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WAPD-MRP-54 (Del.)

TECHNICAL PROGRESS REPORT
PRESSURIZED WATER REACTOR PROGRAM

For the period
May 5, 1955 to June 16, 1955

Contract AT-11-1-GEN-14

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WESTINGHOUSE ELECTRIC CORPORATION

ATOMIC POWER DIVISION

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TECHNICAL PROGRESS REPORT

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SUMMARY (Contd.)

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4. As a result of a re-evaluation of the need for strainers in the Primary Coolant Discharge and Primary Coolant Charging Systems, the coolant discharge and vent system strainer and the suction header strainer were cancelled. (Page 20)

C. Core Development

1. A report was written on the fuel element failure instrumentation development to date, particularly for the PWR, but applicable in principle to all pressurized-water type reactors. (Page 23)
2. A review was made of promising core layouts to determine the extent to which the reference design approximates the optimum balance between nuclear, thermal and mechanical requirements. The results indicate that the reference core arrangement presented in WAPD-MRP-53 represents a well-balanced design to meet the desired core objectives. (Page 24)
3. Functional Requirements for the Fuel Handling System were prepared and submitted to the AEC for approval. (Page 30)
4. The Equipment Specification for the subassembly handling crane and extraction tool was written and is being prepared for submission to the AEC for comment. (Page 30)
5. A contract was made with Brown University, Providence, Rhode Island, to conduct the photoelastic study of the reactor vessel closure head with penetrations for 28 control openings, 9 servicing openings and 2 instrumentation openings. (Page 32)
6. A group consisting of a Combustion Engineering, AEC, and PWR representative was organized to conduct weekly meetings for the purpose of expediting the reactor vessel program. (Page 32)

D. Materials Development

1. A second shipment of seamless Zircaloy tubing from Nuclear Metals, Inc. was received which showed considerably better dimensional uniformity and straightness than did the first lot. (Page 37)
2. Preliminary results were obtained which indicate that hydrogen sintering at 1725°C without prior CO₂ treatment effectively removes 1 w/o PVA binder from UO₂ fuel compacts. (Page 45)
3. Evaluation work on fuel rod bundles was narrowed to two designs with the expectation of selecting a reference design by July 1. (Page 49)
4. Samples of clad natural MCW and steam oxidized UO₂, hot pressed and cold pressed and sintered to densities varying from 81 to 92 percent of theoretical, were cycled 1800 times into and out of a thermal neutron flux of 3×10^{14} neutrons/cm²-sec during a six-week irradiation period with no observed dimensional changes. (Page 97)

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SUMMARYA. Systems Design

1. The following system descriptions and schematics were submitted to the AEC:
 - a. Pressurizing and Pressure Relief (Page 5)
 - b. Purification (Page 5)
 - c. Sampling (Page 6)
 - d. Valve Operating (Page 7)
 - e. Steam and Feed Piping (Page 7)
 - f. Nuclear Instrumentation (Page 8)
 - g. Reactor Plant Electrical Power Distribution System (Page 10)
2. Design criteria for the Safety Radiation Monitoring System were submitted to the AEC for approval (Page 1) and functional requirements for the Fuel Handling Water System were issued. (Page 2)
3. A report was submitted to the AEC for information describing methods for prolonging flow coastdown as a part of the study of the loss of flow accident. (Page 2)
4. The Operational Radiation Monitoring System was approved by the AEC. (Page 9)
5. Agreement was reached with AEC and the Duquesne Light Company to proceed with a main control center design consisting of a pictorial control console and a vertical back-up instrument panel. (Page 10)
6. AEC approval of the preliminary design of the Fuel Handling and Service Building and of a weight handling plan was received. (Page 14)
7. Pittsburgh-Des Moines Steel Company was awarded the contract for fabricating and erecting the reactor plant container shell. (Page 14)

B. Component Design and Tests

1. The Equipment Specification for the pressurizing tank was forwarded to vendors for quotations. (Page 19)
2. Bids were received for the three types of 18-in. valves. (Pages 16 and 17)
3. The last of the required drawings on the steam generators was approved by the AEC. (Page 18)

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SUMMARY (Contd.)WAPD-MRP-54

5. Six defected samples of clad natural MCW UO_2 , cold pressed and sintered to 93 percent of theoretical density, were irradiated in small auto-claves under conditions of static high pressure water and reactor cycling which could result in sample failure due to waterlogging. Samples were cycled 40 times, into and out of a thermal neutron flux of 5×10^{13} neutrons/cm²-sec, with no change in dimensions. (Page 100)
6. A defected UO_2 specimen has operated successfully since May 12 under simultaneous exposure to high temperature water and neutron bombardment in the Chalk River X-1 loop. The gross water activity, which is considerably less than that observed on previous tests, reached a level between 10^4 and 10^6 disintegration/min/ml. (Page 101)
7. A mechanism for cycling specimens in the HB-5 loop was built, tested, and shipped to MTR for installation. (Page 102)
8. An error in position of the sample in the HT-1 loop was discovered and steps taken to correct the situation. (Page 104)

E. Physics

1. An intensive review of the method of calculating the reactor physics parameters required for design calculations was completed. (Page 106)
2. A slab reactor approximately the geometry of one side of the seed in the PWR reference design has been constructed in the PWR flexible critical assembly. (Page 108)
3. The calculations for the heating in the thermal shields and wall of the reactor vessel were essentially completed for the reference design and a report is being prepared. (Page 109)
4. The estimates of air-conditioning requirements for the primary loop compartment were completed.

F. PWR Parameters

The following pages contain a recapitulation of current PWR parameters.

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PWR PLANT PARAMETERS

The following PWR plant parameters are based on a net generated electrical output of 60 Mw with 3 coolant loops operating.

General

Reactor Power*	790 x 10 ⁶ Btu/hr
Gross Electrical Output*	68 Mw
Net Electrical Output**	60 Mw
Steam Pressure at Full Load **	600 psia
Normal System Operating Pressure**	2000 psia
System Design Pressure **	2500 psig
Generated Electric Power Range (with reactor in automatic control)	5 Mw to full power
Generated Electric Power Range (automatic control of the throttle by system load dispatching)	20 Mw to full power

Maximum Expected Network Load Swings*

1. Plus 15 Mw or minus 12 Mw at step change rate
2. Plus or minus 15 Mw at 3 Mw per second
3. Plus or minus 20 Mw at 25 Mw per minute

Ambient Conditions within Vapor Container *

1. Maximum temperature, continuous duty	122°F
2. Maximum temperature, emergency conditions	
a. Steam generator compartments	130°F
b. Reactor compartment	200°F
3. Minimum temperature, continuous duty	50°F
4. Minimum plant equipment operating temperature	40°F
5. Normal operating humidity	50% nominal
6. Limits of humidity variation	20% to 70%
7. Normal ambient air pressure	Atmospheric
8. Limits of air pressure variation	To be determined

Ambient Conditions Within Air Conditioned Areas (Control Room, Laboratory, etc.)*

1. Normal design ambient temperature	80°F
2. Limits of ambient temperature variation	70-80°F
3. Normal design humidity	50%
4. Limits of humidity variation	20-50%

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PWR Plant Parameters (Contd.)Ambient Conditions Within Non-Air Conditioned Areas (General Enclosed and Heated Equipment Areas)*

1. Spaces normally attended
 - a. Normal ambient temperature 70-90°F
 - b. Limit of ambient temperature 60-105°F
except 5 hr/yr max will exceed 108°F
 - c. Limits of humidity variation 20-70%
2. Spaces normally not attended
 - a. Normal ambient temperature 60-90°F
 - b. Limits of ambient temperature 50-105°F
except 5 hr/yr max will exceed 108°F
 - c. Limits of humidity variation 20-70%

Conditions for Outside Areas (Open and Non-Heated Enclosed)

1. Actual design conditions should be determined for specific equipment involved in accordance with requirements as imposed by the duty service, location and exposure.
2. General Outdoor Conditions

	<u>Summer</u>	<u>Winter</u>
a. Maximum (Minimum) temperature	Dry Bulb 102°F	-16°F
b. Design temperature	Dry Bulb 95°F	0°F
	Wet Bulb 75°F	
c. Wind velocity (average)	8.9 mph	11 mph
d. Wind Direction	From S to WNW	From S to NW

Pressure Piping Welds*

1. Primary Coolant System
 - a. To 1 in. NPS Socket
 - b. Above 1 in. NPS Butt
2. Steam and Feed System
 - a. To 2 in. NPS Socket
 - b. Above 2 in. NPS Butt

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FWR Plant Parameters (Contd.)Pressure Piping Design*

<u>Item</u>	<u>Design Pressure</u>	<u>Design Temp.</u>	<u>NPS</u>	<u>Sch. No.</u>
Main loop piping	2500 psig	600°F	15.094 ID	18.000 OD
			15.125 ID	18.063 OD
Auxiliary piping	2500 psig	600°F	1/2 to 1	80
Auxiliary piping	2500 psig	600°F	1-1/2 to 8	160
Pressurizer auxiliary piping	2500 psig	675°F	1/2 to 1	80
Pressurizer auxiliary piping	2500 psig	675°F	1-1/2 to 6	160
Auxiliary piping	3000 psig	600°F	1/2	80
*Auxiliary piping	3000 psig	600°F	1-1/2 to 3	160
Auxiliary piping	3000 psig	150°F	1/2 to 1	80
Auxiliary piping	3000 psig	150°F	1-1/2 to 3	160

*Note: Schedule 160 pipe in 2-1/2 in. to 3 in. size will not be satisfactory for 3000 psig and 600°F if wall thickness tolerance of minus 12-1/2% is applied. If this size is required, it will be necessary to order special and specify more rigid tolerances.

Piping penetrations of primary systems - Minimum size, 1 in. to and including first stop valve.

Primary systems piping - Minimum size, 1/2 in. NPS, Sch. 80

Primary system piping surface finish - 125 micro inches
(internal)

Applicable Codes*

1. Reactor Pressure Vessel

a. ASME, Section I, Power Boiler Code

2. Control Rod Mechanisms (Tubular housings only)

a. ASME, Section I, Power Boiler Code

3. Pressurizer

a. ASME, Section I, Power Boiler Code

4. Steam Generators

a. ASME, Section VIII, Unfired Pressure Vessel Code

5. Primary Plant Auxiliary Heat Exchangers and Pressure Vessel Code

a. ASME, Section VIII, Unfired Pressure Vessel Code

b. ASME, Section V, Miniature Boiler Code

(Where due to size limitations Section VIII is not applicable)

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PWR Plant Parameters (Contd.)

6. Main Coolant Pumps

- a. Not a code item; however, design pressure containing parts in accordance with ASME, Section I where applicable.

7. Main and Auxiliary Coolant Piping

- a. Size - ASME, Section I, Power Boiler Code, P. 23
- b. Construction - ASME, Section I, Power Boiler Code, PP. 112, 116, 300
- c. Flexibility - ASA - B-31, Code for Pressure Piping, Section 6, Chapter 3

8. Vapor Container

- a. ASME, Section VIII, Unfired Pressure Vessel Code with exceptions as approved.

9. Valves

a. Safety valves

- (1) ASME, Section I, Power Boiler Code, P. 269 to P. 290 inclusive, where applicable

b. Main coolant valves

- (1) ASME, Section I, Power Boiler Code, P. 299 where applicable
- (2) ASA - B-16.5, 1953, where applicable

c. Other valves

- (1) ASME, Section I, Power Boiler Code, P. 299, P. 318 where applicable
- (2) ASA - B-16.5, 1953, where applicable

Primary Coolant Systems Heating and Cooling Rate*

1. Normal
2. Exceptions

200°F/hr for 400 cycles each

- a. Reactor pressure vessel
- b. System components not subject to normal operating temperature

To be determined
As required

Primary System Hydrostatic Test Conditions*

1. 3750 psig, $T_{\max} = 100^{\circ}\text{F}$
2. 2500 psig, $T_{\max} = 550^{\circ}\text{D}$

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PWR Plant Parameters (Contd.)Primary Coolant Systems Thermal Transients*

1. Normal Operation

	<u>No. of Transients</u>	<u>Temperature Transients*</u>	
		(1)	(2)
a.	200,000	± 4	± 2
b.	100,000	± 7	± 4
c.	25,000	± 10	± 5
d.	7,000	± 13	± 7

* First temperature transient occurs in approximately 50 sec followed by the second transient 100 sec later.

2. Thermal Shock

- a. Discharge System - ambient to 550°F max, instantaneous, 5000 cycles
- b. Others - as required by individual system components

Reactor Temperature Coefficient*

-2×10^{-4} Sk/°F

Water Conditions

- 1. Source of water for Primary Coolant*, Charging Water, Fresh Water, etc.

Boiler feed grade with nominal conditions as follows:
 pH at 25°C, 6.5 min - 8.3 max
 Total insoluble solids - 0.2 ppm
 O₂ content - 0.009 ppm max
 Chlorides - 0.3 ppm max
 Conductivity - 0.5 min - 1.5 max micromhos/cm
 Dissolved N₂ - 0.2 ppm

- 2. Valve Operating System*

Same as 1 except max concentration of 10 ppm O₂ and 33 ppm N₂

- 3. Fresh Water Cooling System

Same as 1 except corrosion inhibitor added equivalent to 500-800 ppm of chromate ion with pH adjusted to 8.3 - 9.5

- 4. Shield Tank

Same as 1 except corrosion inhibitor to be added equivalent to 500-800 ppm of chromate ion in lithium chromate form with pH adjusted to 8.3-8.6

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PWR Plant Parameters

5. Fuel Handling Canal and Storage Area*

Non-deaerated demineralized water having nominal conditions as follows:

pH at 25°C, 6.5 min - 8.3 max
Total insolubles - 0.2 ppm
Chlorides - 0.3 ppm
Conductivity Micromhos/cm
0.5 min; 1.5 max.

6. Primary Water*

pH at 25°C, 6.0 min - 10.5 max
Total suspended solids - 1 ppm
O₂ content 0.14 ppm
H₂ content - 25 to 30 cc (STP)/kg

Decontamination Agent

Compatible with materials of corrosion resistance equivalent to 18-8 type stainless steel; not to cause corrosion rates exceeding 150 mg/dm² mo

Radiation Level Specifications (Based on 4 loop capability of 937×10^6 Btu/hr)

Primary loop compartment (contained loop shut down and decontaminated, but with adjacent loop operating)	200 mr/hr
Purification system compartment (after decontamination of all but demineralizer)	200 mr/hr
Purification system compartment valve access cubicle (after decontamination of all but demineralizer)	50 mr/hr
Auxiliary vapor container (outside of component shields)	50 mr/hr
Turbine room	2 mr/hr
Control room	2 mr/hr
General station area	Less than 5 mr/hr

Heat Source SystemGeneral

Heat output at full power*
T_h (average outlet temperature)

790×10^6 Btu/hr
541.4°F

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PWR Plant Parameters (Contd.)

T_c (average inlet temperature)	508.6°F
T_{ave} (average core coolant temperature)	525°F
Full power coolant flow	20.0×10^6 lb/hr
Reactor temperature coefficient*	-2×10^{-4} Sk/°F
Reactor Fuel Material** (Blanket)	UO ₂
Weight of zirconium	11.44 tons
Weight of natural uranium	11.6 tons
Weight of U ²³⁵	52.0 kg
Mean diameter of core	185.5 cm
Height of core without fuel supports	183.0 cm

UO₂ Region

Full length	183 cm
Active length - cm	164.5
Bundles per subassemblies	7
Number of subassemblies	93
Number of rods	78,771
Gap between bundles	1/16 in.
Active rod length	9-1/4 in.
Rod length	10-1/4 in.
Rod diameter	0.413 in.
Clad thickness	0.028 in.
Pitch	0.468 in.
Rods per bundle	121 (11 x 11)
Flow area	7.02 ft ²
Heat transfer surface	6570 ft ²
Equivalent diameter	0.021 ft
Weight of zirconium - including fuel supports	6.55 tons
Weight of natural uranium	11.6 tons
Power	425×10^6 Btu/hr
Surface coefficient - h	4,890 Btu/hr-ft ² -°F
ΔT average	29°F
ΔT hot channel	66.9°F
T_2 - outlet average	537.6°F
T_{2HC} - hot channel outlet	575.5°F
T_{sm} - max surface temperature (includes peaks at end caps)	621.9°F
Average heat flux - $\bar{q}$	64,700 Btu/hr-ft ²
Max heat flux (includes end cap peaks)	265,000 Btu/hr-ft ²
Power density	34.6 watts/cc average
Peak power density	142 watts/cc
θ - average	13.2°F
θ_{max} HC-peak in hot channel, at end cap	86.5°F
θ_{max} - peak in hot channel, not at end cap	69.3°F

Seed Region

Inside diameter	86.6 cm
Outside diameter	121.0 cm

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PWR Plant Parameters (Cont'd.)

Length	183 cm
Fuel element	Plate
Plate thickness	0.080 in.
Clad thickness	0.015 in.
Channel thickness	0.080 in.
Plate width (active)	2.05 in.
Number of plates	1248
Number of clusters	24
Weight of U ²³⁵	52 kg
Weight of zirconium	3.46 tons
Flow area	1.68 ft ²
Heat transfer surface	2540 ft ²
Equivalent diameter - D _e	0.0129 ft
Power	286 x 10 ⁶ Btu/hr
Surface coefficient - h	9,470 Btu/hr-ft ² -°F
θ - average	11.72°F
θ _{max} HC - max in hot channel	51.6°F
T _{sm} - max surface temperature	581.9°F
ΔT - average	39.5°F
ΔT _{HC}	63.1°F
T ₂ - outlet average	548.1°F
T ₂ HC - hot channel outlet	571.7°F
q̄ average flux	111,000 Btu/hr-ft ²
q̄ _{max} peak flux	316,000 Btu/hr-ft ²
Average power density	90 watts/cc
Maximum power density	257 watts/cc

Main Coolant System

Total number of loops**	4
Steam generators per loop**	1
Pumps per loop*	1
Number of loops operating at full power**	3
Flow per loop at full power	6.5 x 10 ⁶ lb/hr
T _H	541.4°F
T _C	508.6°F
Steam generator design**	
Steam output at full power (total)	861,000 lb/hr
Steam pressure at full power	600 psia
Steam side design pressure	1200 psig
Pump design	
Capacity at full speed,	
Impeller "A" 4 loops*	10,000 gpm at 145 psi TDH at 501°F
3 loops*	12,000 gpm at 136 psi TDH at 506°F

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FWR Plant Parameters (Contd1)

Capacity at full speed,
Impeller "B" 4 loops*

3 loops*

16,200 gpm at 108 psi TDH at
507°F

16,700 gpm at 105 psi TDH at
508°F

Capacity at half speed,
Impeller "A" 4 loops*

3 loops*

5,000 gpm at 36 psi TDH at
501°F

6,000 gpm at 34 psi TDH at
506°F

Capacity at half speed,
Impeller "B" 4 loops

3 loops

8,150 gpm at 27.5 psi TDH at
501°F

8,450 gpm at 26.8 psi TDH at
508°F

Number of speeds**

Impellers*

Volute*

Pump electrical input, full speed

Pump electrical input, half speed

Pump motor operating voltage (60 cycle a-c)**

Pump design temperature*

Main coolant piping ID*

2

Replaceable

Permanently installed

1250 kw

160 kw

2300 v

550°F

15.0 in.

Coolant Pressurizing System

Pressure following maximum expected surges*

Design temperature*

Type of electrical heaters*

Pressurizer auxiliary piping design temperature*

Coolant recirculation flow

Pressurizer internal volume

Volume of water at normal operating level

2185 psia

675°F

Replaceable

675°F

0.5 gpm

260 ft³

100 ft³

Primary Plant Control

Pumps

Pressurizer heaters

Valves (Primary Coolant)

Manual

Manual or automatic

Remote manual

Purification System

Number of units

Maximum coolant flow (per unit)*

Normal coolant flow (per unit)*

2

50 gpm

40 gpm

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PWR Plant Parameters (Contd.)

Demineralizer life	3 months
Fuel element corrosion product release rate*	100 g/day
Loop material corrosion rate*	10 mg/dm ² /mo
Demineralizer maximum operating temperature*	130°F
Demineralizer resin*	Rohm & Hass MB-1 or equivalent
Demineralizer capacity*	12 gram equivalent/cu ft
Demineralizer efficiency for removal of crud and fission products*	100%
Demineralizer cross section flow rate	7.5 gpm/ft ² (minimum with coolant pumps at full speed)

Coolant Charging System

Storage capacity	50,000 gal
Number of fill pumps	1
Capacity of fill pump at 100 psig	200 gpm
Number of charging pumps	2
Capacity of each charging pump at 3000 psig	25 gpm
Maximum fluid velocity in pipe	10 ft/sec
Number of test pumps	1
Test pump discharge head at 2 gpm	3750 psig

Coolant Discharge and Vent System

Discharge System influent flow rates

Hot coolant at 2500 psia and 525°F	40 gpm
Cold coolant at 15 psia and 150°F	75 gpm
Vent gases - maximum	60 cfm

Discharge System effluent conditions

Pressure	5 psig
Temperature (normal)	150°F
Temperature (maximum)	180°F

Materials

a. Between loops and first stop valve	Type 304 stainless steel
b. Between first stop valve and system	Carbon steel

Emergency Cooling System

Number of units	4
Unit maximum heat absorption rate	100 kw
Unit design pressure	1200 psig
Steam side	1500 psig
Condensate side	

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PWR Plant Parameters (Contd.)

Unit operating pressure (steam)	735 psig
Number reservoirs	2
Reservoir capacity	7000 gal
Reservoir water maximum initial temperature	100°F

Corrosion Inhibitor System

Design function**

1. Provision for injection of hydrogen
2. Provision for injection of hydrazine

Piping systems*

Hydrazine addition	Charging System
Hydrogen addition	Sampling System

Hydrazine addition vessel*

Number	1
Capacity	1 gal
Nominal working pressure	15 psia

Hydrogen bubble can*

Number	1
Capacity	1 gal
Nominal working pressure	15 psia

Hydrogen injection vessel*

Number	1
Capacity	500 cu in.
Nominal working pressure	2000 psig

Valve Operating System

Type of system	Charging pump - accumulator
Fluid medium	Uncontaminated demineralized water
Accumulator medium	High pressure air
Type of operation	Remote or local control
Location of control	Outside vapor container
Normal operating pressure	2800-2900 psig
Design pressure*	3000 psig
Design temperature*	
a. Cold sections	150°F
b. Hot sections	600°F

* Tentatively Approved

** Approved

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PWR Plant Parameters (Contd.)

Air storage capacity	40 ft ³
Operating water storage capacity	100 gal
Number of air compressors	1
Capacity of air-compressor	5 cfh at 3000 psig

Fresh Water Cooling System

Design temperature of cooling water*	100°F
Operating pressure (nominal)*	100 psig

Vapor Container Air Cooling and Decontamination System

Type of system	Open
Number of supply trains	2-1 normally operating
Number of exhaust trains	2-1 normally operating
Design air flow capacity for 2 trains	34,200 SCFM
Inlet air cleaning	Electrostatic precipitation and filtration
Outlet air cleaning	Filtration
Heating equipment	Steam air heater
Vapor container isolation	Quick-closing valves

Sampling System

Number of sample points	4
Position of sample points	Inlet and outlet of each Purification System demineralizer
Size of sample piping	1/2 in. NPS, double extra strong
Sample quantity rate	0.16 gpm
Sample discharge	To Waste Disposal System
System design pressure	2500 psig
System design temperature	130°F
Measurements	a. Conductivity b. Quantity of suspended solids c. pH d. Gas analysis e. Water analysis

* Tentatively Approved
** Approved

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PART IPWR ENGINEERINGA. Power Plant Analysis & Systems1. Plant Functional Design and Operation Analysis (55all) O. J. WoodruffSimulator Studies and Primary Coolant Volume Surge Estimates

The EASE Simulator studies being run during June 1955 include simulation of the pressurizer and the compressibility of the primary coolant. The resulting change in reactivity due to the pressure coefficient is then included as well as that due to the temperature coefficient.

An interim report will be issued by the end of June which describes the plant simulation and presents the methods of coolant volume surge computations which have been applied to previous simulator runs. The summary of results of previous runs was presented in WAPD-MRP-53.

Reactor Plant Container Design Criteria

The Reactor Plant Container Design Criteria, as described in WAPD-MRP-53, was forwarded to the AEC on May 16 for approval of the calculated gross and net volume and auxiliary chamber straight side length. Approval was granted by the AEC on June 2.

Safety Radiation Monitoring System Design Criteria

The Safety Radiation Monitoring System Design Criteria have been forwarded to the AEC for approval. This system performs the following functions:

1. Provides indication or alarms for excessive radiation fields in operating and maintenance areas.
2. Provides film badge and dosimeter protection for plant personnel.
3. Provides particulate and gaseous monitoring inside and outside the plant and of all plant effluents.
4. Provides preoperational off-site radiation background information and a continual off-site monitoring program.

The plant monitoring will consist of gamma field monitors, portable air monitors, personnel monitors and survey meters, and liquid and gaseous effluent monitors. The off-site area monitoring will consist of 4 portable off-site monitoring stations and probably a number of inexpensive integrating, rechargeable ion chambers which will also be used for preoperational surveys. Background monitoring of river, steam and well water, river mud soil samples, vegetation and food products will be provided. Trailer mounted off-site stations have been recommended to permit monitoring at various locations and to allow major maintenance of monitoring equipment to be done at the plant site.

I-A-1 (Contd.)

The Stone & Webster study and recommendations for this system have been forwarded to the AEC with the design criteria as supplemental information.

Fuel Handling Water System Functional Requirements

The Fuel Handling Water System Functional Requirements were issued during this period. The System can be defined as the water in the transfer canal and the equipment necessary to perform the following functions:

1. Provide water for the cooling of spent cores which are stored in the canal.
2. Provide for the evacuation, storage and replacing of the water in the canal compartment which is directly over the reactor.

During core removal operations, it is necessary to empty the canal compartment over the reactor, remove the pressure vessel head, and then refill the compartment. It is anticipated that an 110,000 gal tank will be needed to store the water removed from the canal. Refilling the canal compartment from the storage tank will be by gravity.

The system will be flexible enough to perform both functions simultaneously, and will be piped and valved in such a way as to permit any pump to fill or drain water from any canal compartment.

The transfer canal will be filled initially with demineralized water. Maximum canal water temperature will be 90°F.

Reactor Plant Container Sprinkler System

Studies are continuing to determine the requirements of the reactor plant container sprinkler system. This system will reduce container ambient pressure to accommodate a delayed exothermic chemical reaction between the core materials and water, should it occur, after a major rupture in the primary system. Calculations to date indicate that the spray water must be injected within ten minutes time after the rupture to reduce the ambient temperature sufficiently and must be recirculated for continued protection to absorb the heat of reaction and to prevent a hydrogen-oxygen reaction.

A recommendation on this system will be submitted to the AEC by July 15.

Means to Prolong Flow Coastdown

A report was forwarded to the AEC for information on means to prolong flow coastdown. Three feasible methods were investigated: addition of inertia to the main coolant pump, introduction of an axial flow impeller with large inertia into the main primary coolant stream, electrically coupling a synchronous machine to the primary coolant pump motors.

I-A-1 (Contd.)

An analysis of the present reactor flow coastdown requirements indicates that little, if any, increase in flow coastdown is necessary to insure the safety of the reference core. The report indicates that the least expensive method of increasing coastdown time is to increase pump inertia.

PWR Steam Generator Relief Valve Settings

An analytical study to determine the setting and capacity of the steam generator relief valves based on the reference reactor requirements is in process. Recommended settings and capacities will be forwarded to the AEC early in the next report period.

PWR Operation

A program designed to resolve the uncertainties connected with the use of pilot operated power control valves in the main steam system has been developed. The program will consist of the following steps:

1. Based upon present available information, Stone & Webster will survey possible commercial or developmental (if necessary) valve types for limiting plant transients with either complete or partial loss of turbine generator output. This study is scheduled to be completed by July 1.
2. Preliminary studies will be completed by September 1 to establish tentative control and actuation methods for the system.
3. Based upon the above investigations, simulator studies will be made to establish valve capacities and operational characteristics required to limit primary and secondary plant transients during complete loss of turbine generator output and for emergency power changes greater than the established design values. These studies are scheduled for completion by October 15.
4. Considering the results of the simulator studies, final design of the required Steam Dump System will be established by December 1 and made a part of the Safety Shutdown System at this time.

PWR Safety Report

Preliminary material was developed for the first issue of a PWR Safety Report covering the location and topography of the site, reactor plant design, overall site layout, reactor plant and equipment layout, and reactor plant container basic design, penetrations and ventilation system. Material is presently being developed on the consequences of the loss of reactor coolant including preliminary considerations of a core chemical reaction. This report will be issued the last week in July.

PWR Service and Facilities Requirements

During the period a revision of the PWR Service and Facilities requirements was completed and forwarded to the AEC. These requirements are for services and facilities which are supplied to the reactor plant by the Duquesne Light Company.

Architectural additions and changes which have been made since the last issue include an increase in storeroom floor area to 4000 sq ft, sanitary facilities based on 100 man-shifts per day, and a rearrangement in the control panel space requirements.

Additions to the mechanical requirements include information on maintenance of reactor plant equipment and revised steam demand. The waste Disposal System steam requirements have been added at 6000 lb/hr while the plant container steam heating needs were reduced to 4000 lb/hr.

There have been many additions and changes to the reactor plant water requirements. The most significant changes to the water requirements are the elimination of cooling water for the plant container air system, the addition of fire protection water and reactor plant container sprinkler, and the increased volume of the fuel handling canal from 250,000 to 425,000 gal. Lesser changes in flow rates and storage volumes have been made to other systems.

2. Primary Coolant System (55b11) A. P. Zechella

Reactor Coolant System (S.O. 1-AP-8005)

The AEC commented on the revised system description and schematic diagram during the period. The major comments were:

1. System performance data based on the use of various combinations of General Electric Company, Allis-Chalmers Manufacturing Company, and Westinghouse Atomic Equipment Department main pumps must be developed.
2. The loop by-pass lines should be sized for warm-up and cool-down requirements and not for possible decontamination requirements.

Regarding comment one, preliminary work indicates that there would be some degree of unbalance in individual loop flow rates with certain pump combinations. The effect of this unbalance is not yet determined. It is expected that more complete information will be available by July 1.

A revised system description reflecting the AEC comments will be prepared; the date of reissue has not been scheduled as yet.

I-A-2 (Contd.)Pressurizing and Pressure Relief System (S.O. 1-AP-8006)

The revised system description was submitted on May 26. All component functional requirements have been written with the exception of the level indicator-recorder and the spray nozzle. The level indicator-recorder functional requirements depend on pressurizer tank dimensions and must wait selection of pressurizer tank vendor and receipt of his proposed design.

The spray nozzle functional requirement awaits the results of spray effectiveness tests which are expected to be completed in July.

3. Primary Auxiliary and Secondary Systems (55dll) A. P. Zechella

Emergency Cooling System (S.O. 1-AP-8011)

The emergency cooling problem is being studied with a view to simplifying the reference design which utilizes a closed condensing coil in the steam system.

It is evident from the study that if the design conditions previously established for emergency cooling can be relaxed, a simplification and cost reduction can be made. In making this study, Westinghouse Atomic Power Division is evaluating the effect of relaxing these design conditions in the emergency cooling concept in the following manner:

1. Assumption has been made that an operator will be present to isolate any loop in which a primary to secondary leak has occurred.
2. The length of time for no-power operation has been reduced to approximately two hours.

If these simplifications can be realized, they will result in a savings of approximately \$60,000.

By June 24 this study will have been completed and a recommendation will be made as to the type of Emergency Cooling System which will be installed in the PWR plant.

Purification System (S.O. 1-AP-8007)

A revised issue of the Coolant Purification System Description and schematic Diagram was submitted to the AEC for information on June 1. The following revisions were made in the system design:

1. Replacement of the 16 capped manual isolation gate valves with 12 capped manual isolation globe valves. Two of these valves are utilized for manual regulation of the purification flow (1 in each loop).
2. Reduction of the number of bleed-off valves from 8 to 2.
3. Relief valves added to each loop.
4. Design pressure of the shell of the non-regenerative heat exchanger reduced from 2500 to 150 psig.
5. Demineralizer dimensions altered to permit a minimum area flow loading of 7.5 gpm per sq ft (full speed operation of main pumps).

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I-A-3 (Contd.)Coolant Charging System (S.O. 1-AP-8008)

Approval was withheld by the AEC on the Charging System pending a restudy of the requirements of the system and possible simplification of the schematic diagram. The system is currently being revised to incorporate various modifications which were agreed to during conferences with the AEC.

The revised system description and schematic will be submitted for approval prior to June 30.

Coolant Discharge & Vent System (S.O. 1-AP-8009)

The system description and schematic are being revised in accordance with agreements that have been reached with the AEC. The major changes to this system are:

1. Remove the remote reading vent line gas-water indicators.
2. Combine the flash and blow-off tank pump discharge lines to waste disposal.
3. Combine the spray water lines to the flash and blow-off tanks.
4. Add indicating lights on the auxiliary control panel for the flash and discharge tank pumps.
5. Provide no additional facilities in the vent system to vent instruments and capped valves.

The revised system description and schematic drawing was submitted for approval June 10.

PWR Reactor Sampling System (1-AP-8012)

The Sampling System description and schematic piping diagram was revised and submitted to AEC for approval. The Sampling System will provide for four sample penetrations into the Purification System. The penetrations will be located before and after each of the two demineralizers. No provision is being made for obtaining hot samples.

PWR Component Cooling Water System (1-AP-8017)

The name of the PWR Fresh Water Cooling System was revised to the PWR Component Cooling Water System. A draft of the Component Cooling Water System was prepared by Stone & Webster Engineering Corporation for review and comment. The comments will be returned to Stone & Webster and the system description and schematic piping diagram will be revised accordingly. This system will be submitted to the AEC for approval by June 24.

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I-A-3 (Contd.)WAPD-MRP-54Valve Operating System (S.O. 1-AP-8015)

The revised system description and schematic diagram were resubmitted to the AEC for approval. The only remaining area of uncertainty is the thermal shock problem on primary valve operating cylinders and pilot valves. During the report period, thermal shock tests on 3 in. and 7 in. hydraulic valves have been completed. A complete report and analysis of the results of these tests and the thermal shock tests previously completed on a 4-way solenoid operated spool pilot valve will be submitted to the AEC for information by July 1.

Corrosion Inhibitor System (S.O. 1-AP-8014)

Work is continuing on combining all chemical addition requirements of the plant into one system, the Chemical Addition System. This new system will be submitted to the AEC by June 22.

Air Cooling & Air Treatment System (S.O. 1-AP-8019)

A preliminary design of this system has been reviewed by Westinghouse Atomic Power Division, the Duquesne Light Company and the AEC. The system description and schematic diagram is being revised and will be submitted to the AEC for approval by June 23. The system consists essentially of two trains of supply and exhaust fans with filtering media and controls to maintain the ambient temperature in the reactor plant container between 50°F and 122°F. No mechanical refrigeration is utilized for air cooling. Each of the air ducts which penetrates the reactor plant container is provided with two quick closing valves which will assure the pressure integrity of the reactor plant container should a rupture of the primary system occur.

PWR Steam Piping and Feed Water Piping Systems (1-AP-8040 and 8041)

The system descriptions and schematic piping diagrams have been sent to the AEC for approval. The system design pressure for the Steam Piping System and Feed Water Piping System has been revised from 1200 psig to 900 psig. The Blow-down and Steam Sampling Systems are included in the Feed Water Piping System description and schematic piping diagram.

Waste Disposal System (S.O. 1-AP-8043)

This system is being designed by Stone & Webster Engineering Corporation.

A preliminary design study has been received and is being reviewed. Comments will be sent to Stone & Webster Engineering Corporation for their guidance in preparing a system description for approval.

The design study shows that incineration is the most economical method of disposing of combustible waste for PWR, and evaporation, with storage of evaporator bottoms, is the most economical method for disposing of liquid waste.

4. Reactor Control System Design (56a13) N. E. Wilson

This project encompasses that portion of plant control and instrumentation directly related to the reactor as distinct from the primary coolant and related systems. Two of the systems involved have had design approval, one was completed and forwarded to the AEC for approval during the period of this report and the remaining two have yet to be completed.

Rod Control System

Work on this system is proceeding and will be completed for forwarding to the AEC by July 25. The design is based on the use of a fixed roller nut, gravity run-down mechanism. Although an important characteristic of the system design will be the provision for uniform position control of the rods by means of a programming drum which transfers the control signal to pairs of rods in succession, provision is expected to be made for obtaining a non-uniform position control with minimum change in the system.

The system contemplates the use of commutator inverters for direct supply of a maximum of two rods per inverter from a special d-c bus. While continuity of voltage to the d-c bus is a requirement of the system, voltage regulation is not critical. Supply to the bus will be by multiple a-c to d-c power converters (such as motor-generator sets) fed from 480 v station service supply buses, with fast-acting reverse current relays for removing a converter from the d-c bus on failure of a 480 v supply.

The requirements of an inverter and a rod position indicator assembly have been prepared and forwarded to component engineers for procurement of a prototype of these components to be used with a rod drive mechanism to test the functioning of a complete rod drive and position indication unit.

Reactor Power and Temperature Control System

The design of this system has been approved and its general characteristics were reported in the last Technical Progress Report, WAPD-MRP-53. The simulator program to verify further the design and determine the optimum values of gain for the nuclear damping control signal began during the first week in June. The results of this program will be given in the next Technical Progress Report.

The functional requirements of the entire control system and its components have been completed and forwarded to the component engineer for preparation of equipment specifications and selection of vendors.

Nuclear Instrumentation System

The redesign of this system was completed during this report period and forwarded to the AEC for approval. Also completed and forwarded for approval were the general requirements of the system and its components in anticipation of the selection of a list of suitable vendors for the equipment package.

Replies from the twenty-five vendors to whom letters of inquiry were sent have been received and the list of those indicating an interest in supplying This equipment is being reviewed to determine a list of vendors who will be invited to bid on the equipment. Information on nuclear instrument procurement will be reported in the future in section I-B-12 of the Technical Progress Report.

Operational Radiation Monitoring System

Approval of the system design was given during this period, with resolutions of some comments on the design requested. The principal comment on system design related to the need for determining the particular reactor plant container which is the source of contamination in the event of a system rupture which causes radioactive discharge from the stack. It is planned to add air particle detectors to the system with intake air for them from each of the passageways between the two boiler chambers and the auxiliary chamber, and from the passageway between the reactor chamber and the auxiliary chamber. To be resolved is the question of the need for placing these detectors outside of the shield with consequent penetration of the plant container as opposed to placing these detectors in a shielded box in the auxiliary chamber where they could be serviced by plant personnel during operation. This question will be resolved and the system revised to incorporate this and other changes relating to AEC comments by July 15.

Letters of inquiry have been sent to twenty-seven potential vendors to determine an interest in and capability of supplying the radiation monitoring equipment. Replies are being received and evaluated. The evaluation and selection of a list of vendors will be reported in the future in section I-B-12 of this report.

Safety Shutdown System

The program of simulator studies discussed in the last Technical Progress Report, WAPD-MRP-53, is expected to begin on July 1.

Further calculations on flow coastdown and improvements in calculations on nucleate boiling at the hot channel exit indicate that the time available for shutdown may be lengthened. Approximately 500 milliseconds are available for initiation of rod motion following inception of the loss-of-flow accident if zero steam quality is a requirement of the shutdown system. If one second elapses, the steam quality is not expected to exceed 25%. Even if 0% exit quality remains, the criterion, the shutdown circuitry is expected to be improved over that proposed in a preliminary report issued in April. The improvement will be in the elimination of some interlocking of circuit breaker and relay contacts which were incorporated to anticipate loss of flow before this fact could be detected with certainty by the prime source of information - the power to the main coolant pumps. The main difficulty lay in the fact that there are transient conditions on the electrical system, lasting up to 150 milliseconds, during which power to the pumps may be reduced to zero or even reversed, for which no tripping should occur because the condition is transient. With a longer shutdown time now permissible, the system design is expected to

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I-A-4 (Contd.)WAPD-MRP-54

be based primarily on the pump power relays with suitable delays to cover transients. The system design will be completed and forwarded to the AEC by September 15.

5. Non-Nuclear System Design (55el2) N. E. Wilson

This project encompasses the control and instrumentation of the primary plant fluid systems and the auxiliary systems such as Electrical Power Distribution, Remote Viewing, and Plant Communications.

Primary Plant Control

Preliminary reports on this system are being prepared which cover the controls associated with each fluid system as the system design description for the particular fluid system is completed. Reports on the controls for the Main Coolant System, the Coolant Pressurization System, the Coolant Charging System, and the Valve Operating System have been prepared and forwarded for information and comment. These preliminary reports will be combined into a complete Primary Plant Control System design description to be forwarded for approval by August 15.

Agreement was reached by conferences with personnel of the Duquesne Light Company and the AEC to proceed on the basis of a pictorial control console with vertical instrumentation panel in control station design. A layout in plan of the control room, including the Duquesne Light Company portions of the control stations is expected to be completed by July 1 in order that steel for the control building structure may be specified.

Primary Plant Instrumentation

Functional requirements for all of the flow, temperature, pressure and level instrumentation for the Main Coolant System and the Pressurizing System have been completed and forwarded to the component engineer for the preparation of equipment specifications. Component functional requirements for the Coolant Purification System instrumentation were also essentially completed during this period. During the next report period the requirements for all of the fluid systems instrumentation will have been completed and a complete system description prepared. This system description will be forwarded to the AEC by August 5.

The functional layout of the main vertical instrument panel, to be placed behind the main control console and to provide essential plant information to the reactor operator, will be completed by July 15. This panel will contain primary instrumentation for the Main Coolant System and the Pressurizing System. It will not be "pictorial" in presentation except that instruments will be arranged physically to correspond with the pictorial arrangement of the plant on the Main Control Console.

Electrical Power Distribution System

The system design description was completed and forwarded for approval during the report period. It encompasses the 480 v supply and main feeder distribution, 2400 v supply to the main coolant pumps, distribution of battery power to reactor plant controls, and the 120 v control power supplies.

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I-A-5 (Contd.)

The essential features of the system are:

1. 2400 v power to the main coolant pumps will be by four individual feeders, each with its own circuit breaker from the Duquesne Light Company 2400 v station service supply. Each feeder will be connected to a separate 2400 v bus section.
2. 480 v power for the reactor area will be provided by two double-ended, 2400 v to 480 v substations. Each end of a substation will be fed from separate 2400 v bus sections in the Duquesne Light Company 2400 v station service supply. A common 2400 v feeder will be used to supply one end of the two substations.
3. The d-c power for control and emergency lighting will be supplied to the reactor area by two feeders, one from each of the two batteries supplied by the Duquesne Light Company. One battery will be used normally for control, and one will be used normally for emergency lighting.
4. 120 v control power for the reactor area will normally be from a transformer on one 480 v bus section with automatic transfer to a transformer on another bus section.

All of the components (unit substations, control power supply panels, motor starters, battery power panels, etc.) will be located in the auxiliary power room on top of the reactor auxiliary chamber and adjacent to the control building. Control of the 480 v supply bus tie, and main feeder breakers will be from a panel in the auxiliary control area of the main control room.

Remote Viewing System

A preliminary design of the Remote Viewing System has been prepared by the Stone & Webster Engineering Corporation and is being reviewed by Westinghouse and the Duquesne Light Company.

The system incorporates television viewing cameras in each of the four boiler compartments and in the pressurizer compartment, with presentation by closed circuit to a single receiver in the control room. Questions involving the location of cameras, remote control of pan, tilt and focus need to be resolved in order that optimum usefulness may be obtained compatible with minimum cost and complication.

6. Control Systems Tests (56all4) H. E. Thomas

The internal report referred to in WAPD-MRP-53 has been prepared and will be published during the month of June. This report covers EASE simulator data taken in March.

The simulator program for the month of June has been set up using a blanket-seed type core with a negative temperature coefficient and a positive pressure coefficient. Load and xenon transient studies are being made with and without the relay control system.

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7. Plant Layout and Shielding (55all) W. B. LeePrimary Coolant Piping and Components

A description of the primary coolant piping and support system was prepared to accompany the piping plan and elevation drawings submitted to the AEC for approval. Stone & Webster is now revising these drawings to show location of field welds for procurement and fabrication purposes. These drawings will be used to select a vendor to furnish and shop fabricate the piping assemblies.

The three prospective coolant pump vendors have approved the pump support design prepared by Stone & Webster. This support permits the pump to pivot about two trunnions welded to the volute and move longitudinally on base pads, as required by thermal movement of the pipe. Stone & Webster is currently determining the feasibility of designing a pump support which will be adaptable with a minimum of modifications to any of the three makes of pumps.

Until the pumps are selected, the pump positions and connecting piping cannot be finalized since critical volute dimensions of the three pumps differ.

Stone & Webster has submitted revised estimated maximum piping moments and forces in the pressure vessel inlet and outlet nozzles. These reactions, which are considerably less than those previously estimated, have been transmitted to Combustion Engineering.

Vendors' drawings of both the primary coolant hydraulic and motor operated valves have been received and found to be satisfactory from an arrangement and access standpoint.

Pressure Vessel

Stone & Webster has completed a more advanced study drawing of the pressure vessel support structure in which the weight of the vessel is transferred to a cylinder surrounding the vessel through a series of twenty-four radial bearing pins located beneath the closure flange. At the bottom of the vessel the cylinder rests on a conical formed heavy plate extending to the concrete foundation, which supports the vessel, neutron tank, and reactor container.

The studies also cover the region around the top of the vessel to establish clearances for the control rod mechanism electrical cables and cooling water piping, and the ventilation duct for the space in the reactor container dome. The access walkway surrounding the top of the vessel at the top of the neutron tank, which provides access to the closure and neutron detectors, is also included. Fairly heavy shielding is required for both the walkway and parapet to protect against radiation from the exposed coolant piping.

Auxiliary Piping and Equipment1. Steam and Boiler Feed Systems

Stone & Webster transmitted final plan and elevation drawings of the Main Steam and Feed Water Piping Systems within the reactor plant container.

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2. Auxiliary Piping

Stone & Webster has prepared a composite schematic drawing showing the auxiliary piping within the reactor plant area. Detail piping drawings will be issued on an "area" rather than a system basis.

The division of responsibility for design of auxiliary piping between the Duquesne Light Company areas and reactor plant area has been agreed upon by the agencies concerned. In general, Stone & Webster will carry utility piping to a point 5 ft from building walls and process piping into the turbine room; Duquesne Light Company will begin design at those points.

Shielding

Drawings and design descriptions have been prepared for the reactor plant container external housing and plant shield and for the fuel handling canal and pit shield design. These will be submitted to the AEC for approval during June.

The shielding of the radioactive components is designed to provide the following radiation levels:

1. Normally occupied areas during plant operation - 2 mr/hr.
2. Areas requiring frequent unrestricted maintenance access during plant operation - 5 mr/hr.
3. Areas requiring infrequent restricted maintenance access during plant operation - 50 mr/hr.
4. Certain internal areas requiring infrequent short-time maintenance access after shutdown - up to 200 mr/hr.
5. All external areas which are non-restricted during normal operation will be shielded to 2 r/hr following a catastrophe which uniformly distributes the full power equilibrium fission products throughout the reactor plant containers.

These radiation levels are to be attained by enclosing the entire reactor plant by at least 4 ft-9 in. of concrete or its equivalent, and separately shielding certain internal components with concrete, steel or lead. The required structural thicknesses of the reactor plant concrete enclosure equal or exceed the 4 ft-9 in. shielding requirement in all cases except for the wall adjacent to the turbine room and the control room where concrete was added for shielding.

Work has continued during the period on the arrangement of the internal shielding but a final design cannot be reached until the piping and mechanical equipment locations become firm. It is tentatively planned to shield the purification compartments and auxiliary vapor chamber to 50 mr/hr and use them for operation and maintenance cubicles. Preliminary designs were prepared for shielding the area around the reactor vessel head for maintenance and for shielding penetrations through the secondary shield. These items along with the internal shield design will be submitted to the AEC for approval when the equipment arrangements permit.

Upon completion of the internal shield design, an overall shielding drawing will be included in the progress report to illustrate the general configuration of the internal and external shield designs.

8. Reactor Plant Site Facility (55a12) W. B. Lee

The AEC has approved the outline dimensions of the Fuel Handling and Service Buildings. At the same time approval was given to use structural steel and curtain wall type construction with pressed steel ribbed roof decking, insulation and built-up tar and gravel roof.

The AEC approved a recommended plan for weight handling facilities for the Reactor Plant Container and Fuel Handling Building. The approved plan extends the main crane of the Fuel Handling Building to serve the auxiliary chamber and provides a monorail system to serve the main hatches of the boiler chambers of the Reactor Plant Container.

The above approvals allow Stone & Webster to complete the preparation of preliminary design drawings, with preliminary cost estimates for the two buildings. These drawings, consisting of single-line plans, elevations and sections, will be submitted to the AEC for approval not later than June 15. Detail design by Stone & Webster is proceeding on the basis of the above approvals.

A preliminary draft of the functional requirements of the main crane for the Fuel Handling Building is being circulated for internal review. A bridge crane of 125-ton capacity is required in order to handle the reactor vessel head and associated equipment. Stone & Webster is proceeding with the preparation of the equipment specifications, anticipating no major revision in the preliminary draft of functional requirements. Specifications, approved for purchase, should be available not later than July 1 in order to insure delivery and installation of the crane by July 15, 1956, to aid in placing the reactor vessel.

The AEC has approved the outline dimensions of the fuel handling canal.

Reactor Plant Container

Bids for construction of the Reactor Plant Container were opened May 16. The low bidder, Pittsburgh Des-Moines Steel Company was awarded the contract. A technical evaluation of the bids was made by Westinghouse and Stone & Webster.

The personnel air-lock arrangement consisting of six 7 ft diam by 9 ft long openings, two for each boiler compartment and the auxiliary compartment, has been approved by the Commonwealth of Pennsylvania.

Reactor Plant Utilities

Arrangement drawings are proceeding for the following:

1. Auxiliary power room including heating and ventilation.

I-A-8 (Contd.)

2. Air cooling system equipment room and internal reactor container equipment.
 3. Yard service lines.
 4. Canal water handling.
 5. Conduit and wiring arrangements.
9. Instruction Manuals and Test Program (59cl6) O. J. Woodruff

A meeting was held with the AEC on June 14 to review the PWR Station Instruction Manual Program. Based upon the results of this meeting the revised program proposal will be submitted July 15.

The development of the PWR Station Test Program is continuing. Preliminary discussions have been held with representatives of the Duquesne Light Company in an effort to coordinate and define the program responsibilities of the two organizations. It is expected that this program proposal will be submitted to the AEC by July 15.

I-B-1B. Power Plant Components and Component Materials and Tests1. Primary Coolant Loop (55b26) C. M. Miller

Piping E-Spec 565447-B was approved, with comments, by the AEC. Piping loop layouts will be submitted to the AEC for approval by June 16.

2. Primary Coolant Pumps (55b20) C. M. MillerPrototype Primary Coolant Pump

The prototype primary coolant pump was completed during the week of May 9 and placed on test. Testing is now proceeding as per T-Spec 551927.

Orders for the plant primary coolant pumps are being held pending release by the AEC.

3. Pump Component Development (55b22) C. M. Miller

This entire development program was terminated by request of the AEC.

At the conclusion of the terminal test, the nylon and hard rubber terminal insulating sleeves had not failed under the test conditions of 2500 psig and 250°F.

4. Auxiliary Pumps (55b21) C. M. Miller

E-Spec 565380 for the reactor coolant discharge pump, and E-Spec 565382 for the reactor coolant dischargeeductors, were forwarded to the AEC for information. These specifications were also released to vendors to obtain proposals. It is expected that orders will be placed during the week of June 20.

5. Primary Coolant Valves (55b19) F. J. Hartz

The General Specification for Primary Coolant Valves, E-Spec 565200-B, was approved by the AEC.

Seal Development

Crane Co. submitted a cost breakdown for the gasket and packing test program which was technically approved by the AEC. Financial approval by the AEC was requested.

18-in. Capped Manual Stop Valve

Vendors' proposals were received covering a 16-in. valve with 18 in. body connections and to be operated by a portable electric motor unit after the cap is removed. AEC approval was requested to place the order for 8 valves and 1 operator with the Westinghouse Atomic Equipment Department which submitted the low bid.

I-B-5 (Contd.)

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Hydraulically-operated Main Stop Valve

E-Spec 565311-C calling for a 16-in. hydraulically-operated main coolant stop valve with 18-in. body connections was approved by the AEC. AEC approval was requested to place an order for 8 valves with the Westinghouse Atomic Equipment Department which submitted the low bid.

Main Coolant Check Valve

E-Spec 565313-D was revised in accordance with new functional requirements and comments were received from the AEC. The specification was resubmitted to the AEC for information. Bids were received and opened on June 10 and are being evaluated.

18-in. Prototype Main Stop Valve

This valve was completed and is on test at the Crane Co. plant. It is scheduled to be shipped to the Westinghouse Large, Pa. Plant by June 20. Arrangements were made to conduct preliminary tests on the valve in the test loop at the Large Pa. Plant prior to installation in the test loop at the Westinghouse Atomic Equipment Department for engineering tests.

2-in. Hydraulic Pilot Valve

Bids were opened on May 13 for valves of this type covered by E-Spec 565290-B. AEC approval was requested to place the order for 7 valves, including 1 spare with Crane Co. which submitted the low bid.

I-B-66. Steam Generators (55b32) J. I. Spiegel

All drawings on the steam generators were approved with comments by the Atomic Energy Commission (the AEC). Vendors are preparing revised drawings in accordance with AEC and Westinghouse comments.

The following table represents the status of welding procedures, tube-to-tube sheet weld samples, and fabricating procedures: (AEC approval is not required).

<u>Babcock & Wilcox Co.</u>	<u>Received</u>	<u>Approved by Westinghouse</u>
Fabrication Procedure	X	
Manual Welding Procedure (Nozzles)	X	
Cladding Procedure	X	
Submerged Arc Welding Procedure	X	
Tube Welding Procedure		
Transition Welding Procedure		
Tube Weld Samples		
Stress Relieving Procedure		
<u>Foster Wheeler Corp.</u>		
Stainless Steel Welding Procedures		
Manual Metal Arc	X	X
Submerged Arc	X	X
Tube Welding Procedure		
Tube Weld Samples	X (preliminary)	X
Carbon Steel Welding Procedures		
Manual (Pipe)	X	X
Manual (Plate)	X	X
Automatic	X	X
Fabricating Procedure	X	

The Babcock & Wilcox Co. was authorized to fabricate the pattern and die and then to form and machine the heat exchanger return bend.

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I-B-7WAPD-MRP-547. Reactor Plant Pressure Vessels and Degasifier (55b33) C. E. SinclairPressurizer Tank

E-Spec 565362-B for the pressurizing tank was sent to pressure vessel manufacturers for quotations. The bid opening is scheduled for June 28.

The transition sleeve tests being conducted in test vessel No. 1 were continued during this test period. This test will be completed about June 20, at which time this unit will have been subjected to approximately 500 temperature cycles during 2000 hr of operation. At this time, the head will be removed and examined for evidence of damage to the transition sleeves. A new flange, which is currently being manufactured, will be installed and placed in operation. This second head with its transition sleeves will be cycled until approximately March 31, 1956.

The transition sleeve tests in vessel No. 2 have been temporarily discontinued due to a failure in the level indicating instrumentation on this vessel. Tests will be resumed by June 25 after the level instrumentation has been repaired and will continue until 6 months of operating time is accumulated.

Corrosion Inhibitor System Pressure Vessels

E-Spec 565389 for the hydrazine addition vessel was approved by the Atomic Energy Commission (the AEC) on June 9. Competitive bids are scheduled to be opened July 18.

Permission was received orally from the AEC on June 10 to release E-565388, equipment specification for the hydrogen injection vessel for bids. Competitive bids are scheduled to be opened July 11.

Coolant Purification System Demineralizer

E-Spec 565176 was revised to incorporate AEC comments and include the latest modifications in functional requirements. This revised specification was submitted to the AEC for information and was released to prospective manufacturers for quotations on May 9. Bids are scheduled to be opened on June 17.

Coolant Discharge and Vent System Pressure Vessels

E-Specs 565390, Flash Tank, and 565391, Blowoff Tank, were submitted to the AEC for approval on May 23. Approval was requested for these specifications on all items except the method of mounting the circulating pumps and the method to be used to introduce primary coolant discharge into the vessels. Westinghouse is currently contacting prospective vendors to obtain their recommendations on these items. This investigation will be completed by June 20, at which time the results will be incorporated into the specifications for these vessels.

Valve Operating System 3000 psig Air Flasks

E-Spec 565654 for the 3000 psig air flasks was approved by the AEC and was released to vendors for quotations. Final bids are scheduled to be received by July 15.

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I-B-7 (Contd.)

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Valve Operating System 3000 psig Air Bottles

E-Spec 565440 for the 3000 psig air bottles was approved by the AEC and was released to vendors for quotations. Final bids are scheduled to be received by July 15.

Valve Operating System Air Vent Muffler

E-Spec 565442 for the air vent muffler was submitted to the AEC for approval.

Primary Charging System Resin Storage Tank

No further action by the Atomic Energy Commission (the AEC) for approval of the resin storage tank, E-Spec 565433 A, will be taken since functional requirements for this component are now being modified.

Primary Charging System Water Storage Tank

No further action by the AEC for approval of the water storage tank, E-Spec 565434 A will be taken since functional requirements for this component are now being modified.

8. Reactor Plant Auxiliary Heat Exchangers (55b34) J. I. Spiegel

The regenerative and non-regenerative purification heat exchanger specifications as approved by the AEC on May 2 were sent to vendors for bids. Bids are due approximately July 1.

9. Filters and Vent Capillaries (55b35) C. E. Sinclair

The General Equipment Specification for Primary Coolant Strainers, E-Spec 565410, was revised and submitted to the AEC for information. Revision B of this specification deleted the helium leak test requirements and added specific information concerning requirements for cast stainless steel strainer bodies.

The coolant discharge and vent system strainer and the suction header strainer were cancelled. No further action for approval of the specification for the Valve Operating System strainer, E-Spec 565413, will be taken until re-evaluation of system requirements is complete.

10. Emergency Cooling Components (55b45) J. I. Spiegel

Because of restrictions on the system design, a hold was placed on these components. As a consequence, bids previously requested from vendors are being held in abeyance.

11. Reactor Electric Power Components (56d12) G. E. Seguin

Preliminary purchase specifications for the motor-operated speed transfer switches for the main coolant pumps were distributed internally and to the Duquesne Light Company for comments. All comments are to be completed by June 24.

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I-B-11 (Contd.)

WAPD-MRP-54

The Elmira Plant of Westinghouse Electric Corporation has been studying the feasibility of thyratrons for low frequency control rod drive motor supplies. It has been indicated that if the tubes are given a breaking-in run and removed from service after a limited time, their reliability might exceed that of any other available device. The application is being further evaluated.

12. PWR Instrumentation (56d13) C. J. Madsen

The following equipment specifications were issued:

<u>Specifications</u>	<u>Title</u>	<u>Disposition</u>
E-565336 - Charging System Pressure Gage		to AEC for approval
E-565337 - Emergency Cooling System Thermometer		to AEC for approval
E-565339 - Emergency Cooling System Back Pressure Controller		to AEC for approval
E-565340 - Emergency Cooling System Flow Indicator		to AEC for approval

Drawings of a standardized valve position indicator suitable for all valve sizes to be used on PWR were submitted to the Atomic Energy Commission for approval.

An investigation into the use of a simple magnetic switch as a rod position detection device was started.

13. Materials for Primary Loop Components (55b16) J. R. HunterFeasibility of Replacing Forgings with Castings

The work on welded cast type 304 stainless steel pipe was reported for this period in WAPD-MRD-49, and will henceforth be reported in the technical progress reports WAPD-MRD series.

Heliarc Welding of the First Pass for Primary Loop Piping

Sufficient information was obtained on the use of the Electric Boat type consumable insert so that a welding procedure specification will be prepared within the next reporting period for heliarc welding of the first pass by this method. This procedure shall be specified for performing the welding of the first pass in all PWR primary loop piping.

I-B-13 (Contd.)WAPD-MRP-54

Efforts will be continued on welding the first pass without a backup ring to further investigate the effect of joint configuration and base material chemistry. Cast pipe and 2 different heats of heavy walled pipe will be used. Welding will be continued to further determine the effect of using nitrogen gas as the inert gas backup. This work will be completed within the next reporting period.

Steam Generators - Chloride Stress Corrosion

Work on the corrosion resistance of type 347 stainless steel was reported in WAPD-MRD-49.

The Duquesne Light Company, as subcontractor, reports that work has begun on fabrication of 3 loops. Three type 304 stainless steel heat exchangers and condensers were shipped to the Frank R. Phillips Power Station on May 24. Start of testing is anticipated to occur on or before July 1.

All future work on heat exchanger chloride stress corrosion tests run in conjunction with the Duquesne Light Company will be reported in the WAPD-MRD series technical progress reports. Highlights of this work will be reported in future issues of this report.

C. Reactor and Auxiliaries

1. PWR Core Development (53al4) L. S. Mims

Fuel Element Failure Detection and Location

As a result of evaluations reported in WAPD-MRP-53, it has been decided to include an instrumentation test system in the WAPD-15 loop designed for the MTR HT-1 hole. The purpose of this test system is to obtain instrument sensitivity data for use in designing the fuel element failure detection and location instrumentation.

Specifically, the instrumentation test loop will measure the following:

- a. The gamma activity in a sampling line carrying coolant from a subassembly containing a defected fuel element.
- b. The build-up of fission product activity in an anion bed through which the above described coolant sample is passed.
- c. The effectiveness of a cation bed in reducing the loop corrosion product background activity in the coolant samples.

Due to space limitations on the WAPD-15 loop, tests of delayed neutron detection instrumentation have to be performed elsewhere. Consideration is being given to incorporating such tests into the program for the WAPD-13 loop designed for the MTR HB-5 hole. The design of the instrumentation test loop for the WAPD-15 loop has been completed, and arrangements have been made to obtain all of the necessary components. Arrangements have also been made with Phillips Petroleum Company personnel at the MTR to cooperate in effecting this evaluation program. The gamma-sensitive detection instruments will monitor a sample of the MTR (WAPD-15) loop effluent both when the loop contains a subassembly with a bent undefected center rod, and when the loop contains a subassembly with a defected center rod. By comparing the instrument signal under both loop conditions, the instrument "signal difference" due to the occurrence of failure should be obtainable.

A report has been written on the fuel element failure instrumentation development to date, particularly for the PWR, but applicable in principle to all pressurized-water type reactors. This report presents a program and a time schedule for achieving a prototype system for fuel element failure detection and location in pressurized-water type reactors.

Core Temperature and Neutron Flux Distribution Instrumentation

Work on instrumentation to determine distribution of temperature and of neutron flux in the core has been started. Existing methods for making temperature and flux distribution measurements are being reviewed to determine means for instituting such instrumentation in the PWR. Particularly under

investigation are the requirements for the number and distribution in the core of the sensing devices and the means of getting the information from these sensing devices outside of the pressure vessel.

2. Mechanical Design of PWR Core (53a12) P. A. Halpine

A review was made during this period of promising core layouts to determine the extent to which the reference design approximates the optimum balance between nuclear, thermal and mechanical requirements. The results indicated that the reference core arrangement presented in WAPD-MRP-53 represents a well-balanced design to meet the desired core objectives. All design effort is now being applied to this design only.

A program is now in progress to complete an analytical study of temperature and stresses in the reference fuel rod, to determine final dimensions and tolerances, and to establish a schedule of design verification tests. The program is outlined in WAPD-RD-58 which will be submitted to AEC by June 24.

Blanket fuel subassembly design is under investigation. Layouts are being prepared of various end fittings which will support the fuel bundles and provide engagement of the subassembly with the reactor support structure and handling tool. The objective is to simplify subassembly fabrication, assembly and handling.

3. Thermal and Hydraulic Analysis and Tests of PWR Core (55a13) L. S. Mims

Reference Core Characteristics

Core characteristics have been recomputed using the most recent flux plots obtained from the one dimensional simulator. These plots are based on the natural uranium element containing UO_2 , and on a water-to-uranium volume ratio of 3.24. Results are listed below:

Parameters

1. Q - Electrical - 60 Mw; Heat 231 Mw (790×10^6 Btu/hr)
2. $T_{ave} = 525^\circ F$; p = 2,000 psig at outlet
3. Hot Channel Factors

<u>Region</u>	<u>ΔT</u>	<u>θ</u>	<u>q</u>
Seed	1.40	2.03	1.31
UO_2	1.25	1.69	1.06

4. 10% flow allowed for leakage; 10% heat assumed generated outside of fuel elements.

I-C-3 (Contd.)

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General

1. Mean Diameter - cm	185.5
2. Height - without fuel supports - cm	183
3. Tons Natural U	11.6
4. Kg - U ²⁵	52.0
5. Tons of Zirconium	11.44
6. Inlet Temperature - °F	508.6
7. Average Outlet Temperature - °F	541.4
8. Average Core Temperature - °F	525
9. Three-Loop Power - Mw	231
10. Three-Loop Flow - 10 ⁶ lb/hr	20.0
11. ΔP through Pressure Vessel - psi	38.0
12. ΔP through Fuel Elements - psi	20.4

UO₂ Region

13. Full Length - cm	183
14. Active Length - cm	164.5
15. Bundles/Subassembly	7
16. Number of Subassemblies	93
17. Number of Rods	78,771
18. Gap between Bundles - in.	1/16
19. Active Rod Length - in.	9 1/4
20. Rod Length - in.	10 1/4
21. Rod Diameter - in.	0.413
22. Clad Thickness - in.	0.028
23. Pitch - in.	0.468

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24. Rods/Bundle	121 (11 x 11)
25. Flow Area - ft ²	7.02
26. Heat Transfer Surface - ft ²	6,570
27. Equivalent Diameter - ft	0.021
28. Tons Zirconium - including fuel supports	6.55
29. Tons Natural Uranium	11.6
30. Power - 10 ⁶ Btu/hr	425
31. Three-Loop Flow - 10 ⁶ lb/hr	12.05
32. Velocity - fps	9.94
33. Reynolds Number	147,000
34. Friction Factor	0.0209
35. ΔP Blanket - psi (everything in core but orificing)	5.06
36. Heat Transfer Coefficient - h - Btu/hr-ft ² - °F	4,890
37. ΔT Average °F	29
38. ΔT Hot Channel °F	66.9
39. T ₂ - Outlet Average °F	537.6
40. T _{2HC} - Hot Channel Outlet °F	575.5
41. T _{SM} - Max. Surface Temperature (includes peaks at end caps) °F	621.9
42. Average Heat Flux - $\bar{q}$ - Btu/hr-ft ²	64,700
43. Max. Heat Flux - Btu/hr-ft ² (includes end cap peaks)	265,000
44. Power Density - watts/cc average	34.6
45. Peak Power Density - watts/cc	142
46. θ - Average °F	13.2
47. $\theta_{\max HC}$ - Peak in Hot Channel, at end cap	86.5
48. $\theta_{\max}$ - Peak in Hot Channel, not at end cap	69.3

I-C-3 (Contd.)WAPD-MRP-54Seed Region

49. Inside Diameter - cm	86.6
50. Outside Diameter - cm	121.0
51. Length - cm	183
52. Fuel Element	Plate
53. Plate Thickness - in.	0.080
54. Clad Thickness - in.	0.015
55. Channel Thickness - in.	0.080
56. Plate Width (active)	2.05
57. Number of Plates	1,248
58. Number of Clusters	24
59. Kg - U ²³⁵	52
60. Tons - Zirconium	3.46
61. Flow Area - ft ²	1.68
62. Heat Transfer Surface	2,540
63. Equivalent Diameter - De - ft	0.0129
64. ΔP Core - psi	20.4
65. Three-Loop Flow - 10 ⁶ lb/hr	5.96
66. Friction Factor	0.0182
67. Reynolds Number	186,000
68. Velocity - fps	20.5
69. Power - 10 ⁶ Btu/hr	286
70. Heat Transfer Coefficient - h - Btu/hr-ft ² - °F	9,470
71. Θ - Average °F	11.72
72. Θ _{max HC} - Max. in Hot Channel °F	51.6

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73. T_{sm} - Maximum Surface Temperature °F	581.9
74. ΔT - Average °F	39.5
75. ΔT_{HC} - °F	63.1
76. T_2 - Outlet Average °F	548.1
77. T_2 HC - Hot Channel Outlet °F	571.7
78. $\bar{q}$ Average Flux - Btu/hr-ft ²	111,000
79. $\bar{q}_{max}$ Peak Flux - Btu/hr-ft ²	316,000
80. Average Power Density - watts/cc	90
81. Maximum Power Density - watts/cc	257

Thermal Design of End Caps

The temperature at the fuel element interface, between the UO₂ fuel and the Zircaloy end caps, is being determined. Chemical interaction occurs between Zircaloy and UO₂ at temperatures above 1100°F and can endanger the integrity of the element. Preliminary calculations indicate the possibility of temperatures in excess of 1100°F in the present reference design. More detailed calculations now being performed will determine what end cap design changes, if any, are necessary. This work will be submitted in a separate report on about July 15.

Thermal Bowing and Stressing of Rods in Natural Uranium Bundles

Thermal bowing of rods is caused by the variation of average rod operating temperatures throughout a given bundle. Calculations have been made on bowing of UO₂ rods based on the maximum expected temperature gradient in a single bundle. The rods were assumed to be fixed to rigid tube sheets. The following results are for the worst case conditions for 0.413 in. diameter 10 in. long UO₂ fuel rods with no initial axial pellet clearance:

Deflection	-	0.0015 in.
Bending Stress	-	770 psi

These values are considered negligibly small and indicate that rods 10 in. long are satisfactory from the point of view of bowing.

Calculations have been made on the lengthening of UO₂ rods in warming up from room to operating temperature. Where the initial pellet axial clearance is large enough to take up the differential expansion between the UO₂ and the Zircaloy end caps, the rod lengthening is 22 mils. Where the initial clearance is zero, the UO₂ stresses the Zircaloy cladding in the amount of 50,900 psi, and the rod lengthening is 59 mils.

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Thermal stresses in the hottest UO₂ pellet have been investigated. The results given below are for a pellet which is not influenced by the cladding surrounding it. That is, there is an initial radial clearance between the pellet and cladding sufficient to take up differential expansion. These stresses are calculated for maximum center temperature of 2140°F and an interface temperature of 732.5°F. The stresses are based on a Young's Modulus of 21,000,000 psi thermal coefficient of expansion of 6.5×10^{-6} in./in. per °F, and a Poisson ratio of 0.30.

<u>Stress in the Hottest Rod</u> (Reactor producing 80 Mw electrical)	<u>Center, psi</u>	<u>Outside Edge, psi</u>
Radial	81,900 (Compressive)	0
Longitudinal	165,000 (Compressive)	110,000 (tensile)
Circumferential	81,900 (Compressive)	110,000 (tensile)

It is probable that plastic deformation or cracking would reduce the above stresses and would have a secondary effect on the conductivity. Future investigations will be directed toward determining the effect the corresponding strain in the UO₂ has on the 0.028 in. thick cladding enclosing the 0.357 in. OD UO₂ material for various initial clad clearances.

Minimum Pressure in Reactor Vessel

Minimum allowable primary loop pressure during transients versus plant operating power is being determined. Variations in pressure in the reactor vessel are caused by power level changes. These variations in pressure are of short duration, hence calculations have been based on the assumption that local boiling is allowable but that bulk boiling in the core is forbidden. A report of this study is being prepared for issuance in July.

Loss-of-Flow Accident

During this period, the loss-of-flow studies have been concentrated on the use of the UO₂ reference core design and are being extended into the emergency cooling period. These studies are based on the latest calculated main coolant pump inertia value, and the experimentally determined control rod insertion rate. Analyses have been completed on the thermal condition of the seed hot channel, control rod mechanisms, and pressure vessel as a function of time from pump failure to the start of the emergency cooling period. Emergency cooling here refers to the removal of decay heat by natural circulation flow. Analyses of emergency cooling in the core are being made in conjunction with power plant design. Results of the studies on emergency cooling characteristics of the core will be reported about September 1.

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Fig. 1 shows the latest power-flow decay curve with the hot channel exit conditions of zero quality, 40% quality, and burnout superimposed. This curve is based on the latest pump inertia data. It assumes no rod insertion and a temperature coefficient of reactivity of $-2 \times 10^{-4} \text{ } ^\circ\text{K}/^\circ\text{F}$. The curve also illustrates the effect of variation in operating conditions on the time to reach bulk boiling.

PWR Pressure Vessel Flow Study

A quarter scale air flow model of the pressure vessel has been chosen for use in a PWR flow study. The primary purpose of this study is the determining of what flow baffling is necessary in the bottom portion of the pressure vessel. Preliminary specifications for the tests and the model have been prepared; on the basis of these specifications, negotiations with the Westinghouse Steam Division are being completed. The original Steam Division time and cost bid, dated May 16, is presently being modified to factor in revisions made to the test program.

4. Mechanical Tests of PWR Components (53a13) P. A. Halpine

A quarter scale plastic model was made of four PWR subassemblies and an appropriate portion of the supporting structure to explore experimentally the mechanical problems encountered in the latching and unlatching of the subassemblies in the core cage.

5. Core Refueling Equipment (53d12) P. A. Halpine and

6. Materials Handling & Viewing Equipment (53d11) P. A. Halpine

Functional Requirements 1003 for the fuel handling system were written and submitted for AEC approval.

E-Spec. 535741 describing the handling crane and extraction tool was written in accordance with FR-1003 and will be submitted to AEC for comment by June 24.

The layout drawing of the fuel handling canal and the fuel handling equipment was extended to include handling equipment for the complete core and for the pressure vessel head, in addition to the handling crane with extraction tool and the transfer crane.

7. Reactor Vessel Assembly (55b25) P. A. Halpine

The AEC has granted approval to form and machine the reactor vessel cylindrical courses and bottom head in preparation for welding. Four sets of welding procedure specifications have been reviewed and submitted to the AEC. Approval was granted by the AEC for these welding specifications.

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I-C-7 (Contd.)WAPD-MRP-54

A program for the development tests required for the reactor vessel was submitted to the AEC and approval was granted. This program requires the testing of a quarter scale steel model of the vessel having a bolted type closure and a similar model with the alternate design (conical segment closure) and a full diameter test vessel with the bolted closure design. The model vessel (1/4 scale) with the conical segment closure has been completed and is presently being set up for the initial tests.

A contract has been made with Brown University, Providence, Rhode Island, to conduct the photoelastic study of the reactor vessel closure head with penetrations for 28 control openings, 9 servicing openings and 2 instrumentation openings. Two plastic castings are in progress, one 1/10 scale for the conical segment closure head and one 1/15 scale for the bolted closure head.

The change in design from the conical segment closure to the bolted closure required new materials for the large forgings which are of the order of 90,000 lb each. During a meeting between Combustion Engineering and WAPD, a preferred material for these flanges was established. This material is basically the same chemistry as A-302 - Grade "B" with the addition of 0.5% nickel to improve the impact properties. A recommendation of this material was forwarded to the AEC and approval was granted. The forgings were ordered from Bethlehem Steel by Combustion Engineering on June 14.

During a meeting between Combustion Engineering, the AEC and WAPD which was held on June 13, a group was organized consisting of one individual from each of the three organizations. This group has the primary function to review the reactor vessel program and to expedite this program in order to meet the required delivery date of the vessel which is August 1, 1956. This group will meet weekly at Combustion Engineering and the first meeting is scheduled for June 20.

I-C-8WAPD-MRF-548. Control Mechanism Development (56c19) T. L. Weybrew

Following the initial test for 2000 cy in hydrogenated water and the subsequent acceleration tests, mechanism type XRN-10A (fixed roller nut) shown in Fig. 2 was operated for the equivalent of 750 full travel cycles in oxygenated water with 300 scrams attempted. This test was accomplished to obtain some indication of characteristics under adverse water conditions which are not anticipated on PWR, except as an accident condition. Soon after the test began, scram failures occurred and the proportion of failures soon reached 50%. Hydrogen was then added to the water in order to eliminate the oxygen. The mechanism operation promptly improved and all scram failures disappeared. The oxygenated water test was then resumed and the scram failure rate again rose to about 50% for the remainder of this test.

Oxygenated conditions were then terminated and the life test was resumed in hydrogenated water duplicating the original environment. Once again the operation improved greatly, with only 4% scram failures over the next 2000 cy, and none in the first 870 cy. However, the mechanism is now operating with slightly less average net overhauling torque than it demonstrated during the first test.

The type XRN-10A mechanism was built prior to the establishment of firm functional requirements and does not fulfill the current space and performance requirements. Therefore, a redesign of this mechanism, designated type XRN-21A, has been undertaken in an attempt to meet these requirements. This design will be completed by July 1. One major feature of the new design is the allowance of space for inclusion of an additional "kicker" motor as found necessary from the standpoint of scram reliability.

Rebuilding of the original model of the expanding roller nut with separate scram magnet (type XRN-9A) is continuing.

A more detailed investigation into the preliminary design of types XRN-17A, XRN-18A, and XRN-19A (6 in. diam versions for PWR of the expanding roller nut with separate scram magnet, 4-arm collapsible rotor, and 2-arm collapsible rotor, respectively) was conducted in an effort to produce a backup design for the above fixed roller nut mechanism. The 2-arm type XRN-19A mechanism was selected as most suitable, and design studies are proceeding on this type.

II-A-1WAPD-MRP-54PART IIPWR DEVELOPMENTA. Fuel Element Development1. Core Raw Materials and Shapes (53c21)Uranium-Molybdenum Program (F. R. Lorenz)

All development melting of uranium-molybdenum and uranium-niobium alloys was discontinued and all analytical work stopped because of the recent selection of UO_2 fuel material for the PWR blanket. Reports, summarizing the work already completed, are being prepared. All work on extrusion billet components was terminated for the same reason and no further effort was made during this period.

Uranium Oxide Program (B. R. Price)Properties of UO_2 Powders

This work is being undertaken to determine the fabrication characteristics of UO_2 powders. Several experiments were performed which indicate that the incorporation of internal lubricants with the oxide powder reduces the compaction ejection force and improves the compact surface appearance. The addition of 0.2 w/o aluminum stearate to the 1.3% enriched oxide powder for the TRX experiment produced excellent compacts. For CO_2 pretreated compacts, initially containing alumina stearate, residual carbon and hydrogen contents were less than 3200 ppm and 70 ppm respectively.

Due to the urgency of the in-pile test program, work on reclaiming unsatisfactory UO_2 fuel slugs in the as-compacted and sintered condition was inactive during this period.

Claddinga. Seamless Tube Development

Samples of the first lots of drawn seamless Zircaloy-2 tubing manufactured by Nuclear Metals have been hydrostatically tested. As would be expected because of the residual cold work in the drawn tubing, the apparent elastic limits and bursting pressure were higher than for tubes machined from hot-rolled bar stock, reported in WAPD-MRP-52. The properties of drawn seamless tubing of reference size (0.414 in. OD x 0.3585 in. ID, wall thickness 6.76%) tested under internal pressure are given below and in general are 1.5 to 2.5 times greater than the values reported in WAPD-MRP-52. The rate of pressure rise used in these tests was approximately 500 psi/sec.

II-A-1 (Contd.)Environment-Air - One Atmosphere

<u>Temp. °F</u>	<u>No. of Specimens</u>	<u>Range of Apparent Elastic Limit, psi</u>	<u>Range of Bursting Pressure, psi</u>
70	4	10,500-12,500	11,500-13,900
620	2	6,500- 7,000	8,600- 8,900
636	2	6,500	8,800- 9,000
700	2	6,500- 7,000	7,800- 8,300
735	2	6,500- 8,000	7,300- 9,000

Some preliminary results have also been obtained from bursting tests in a pressurized and heated autoclave. In these tests two pressure rates were used. The fast rate, approximately 5000 psi/sec, was used to simulate the rate of pressure rise which might occur in a water-logged element under the fastest reactor start-up conditions. The slow rate corresponded to that employed in the bursting tests reported above.

External Pressure - 2000 psig

<u>Temp. °F</u>	<u>No. of Specimens</u>	<u>Range of Differential Bursting Pressure, psi</u>	<u>Range of Actual Bursting Pressure, psi</u>
636	1 (Slow)	7,900	9,900
	3 (Fast)	13,200-14,600	15,200-16,600
700	1 (Slow)	8,000	10,000
	1 (Fast)	8,400	10,400

Differential bursting pressure equals measured bursting pressure minus external pressure.

The properties shown above are in fairly close agreement with the values predicted from stress calculations and data on the strength of Zircaloy-2 with known amounts of residual cold work. Further work is being done to try to relate fracture size in the tube to energy released by the pressurizing fluid at the time of rupture. The purpose of these measurements is to establish the size of opening in a water-logged and ruptured element through which fuel fragments might escape into the main coolant.

Instruments to measure the magnitude of shock waves attending rupture of a tube in a pressurized autoclave are now being calibrated; knowledge of shock wave intensity being desirable for a complete understanding of the water-logging startup accident.

Approximately 93 billets, both 2-in. and 2-3/4 in. OD, copper-clad, have been extruded into tube shells of 3/4-in. OD x 0.065-in. wall and 1-1/16 in. OD x 0.080-in. wall respectively by Wolverine Tube Division. These shells will be

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processed to finished size in three tube reducing steps with two intermediate anneals.

Processing difficulties have been encountered by Bridgeport Brass in the fabrication of Zircaloy-2 tubes from extruded shells 2-1/2-in. OD x 1/4-in. wall. Short, deep longitudinal cracks starting from the ID became evident during the third tube reducing operation (0.880-in. OD x 0.060-in. wall reduced to 0.625-in. OD x 0.047-in. wall). Although no cracks are evident in the material after the first and second tube reducing operation, it is quite likely that the severe cold working (65% reduction in area) during each of the first two tube reducing steps introduces a condition in the material which leads to severe cracking in the third operation. Evidence for this is the appearance of very minute longitudinal marks on the ID of the tube after the first reduction and noticeably larger marks after the second. Tooling has been ordered to decrease the amount of cold work in each tube reducing step.

Nuclear Metals has completed the first trial order of 331 ft of reference size tubing. Four separate lots were supplied. The dimensional control of both ID and straightness was considerably better on the last lot than on the first. The dimensional variation on the last lot was still a little greater than PWR requirements. A new purchase order has been placed for 700 ft of reference size tubes to be delivered by July 15. This order will permit Nuclear Metals to continue its investigation of straightening tubes and control of the ID to the close dimensional tolerances specified by Bettis. Tubing from the first trial used for the HT-1 in-pile test was proof tested to 10,000 psi internal pressure; 10,000 psi was determined to be approximately 95% of the average apparent elastic limit. Approximately 4 % of the tubing tested took a permanent set of 0.12% on the diameter (0.12% was the minimum detectable change); the remainder of the tubing was not permanently deformed and was judged to be satisfactory for in-pile use.

Samples representing the first lot of tubes from both Nuclear Metals and Wolverine have been in 750°F steam test at 2000 psi pressure for 30 and 14 days respectively. The average weight gains are about 25% higher than normal. It is believed that this situation can be remedied by removing more metal from the ID and OD surfaces. Experiments are in progress to determine the minimum amount of metal that must be removed from the cold drawn surfaces to render them corrosion resistant.

A 7 day, 750°F, 2000 psi steam test was performed on tube samples with all OD surfaces pickled to remove a constant amount on all samples and ID surfaces treated in various ways. The results follow:

<u>ID Condition</u>	<u>Result</u>
As-drawn, washed with Alconox	White corrosion product
Pickled in HNO ₃ -HF solution	Black adherent film
Dipped in nitric acid solution, washed	Spotty white

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From these observations it appears that a HNO_3 -HF pickling treatment to remove .001-in. of metal will be required if maximum corrosion resistance is needed on ID surfaces.

A draft of Purchasing Department Specification 12650 giving the manufacture, process control, and inspection requirements for seamless Zircaloy-2 tubes, was presented to prospective suppliers for their comments. Schedules for completion of development work and the beginning of manufacture for core requirements were reviewed and agreed upon.

b. Closed Tube Development

Use of a Zircaloy tube with an integrally closed end similar to a cartridge case would be a great advantage from the standpoint of assuring cladding integrity by virtue of eliminating one welded end closure. A study has been completed which indicates that manufacture of such a component is feasible. Development is being approached by investigating the possibilities of doing the work at the Bettis Plant and by requesting quotations from commercial fabricators. Eight concerns have been contacted and at least three have indicated interest in undertaking development work.

2. Core Manufacturing Process Development (53c23) F. R. Lorenz

The fuel element development program on the U-Mo fuel rod has been discontinued in accordance with the recent selection of the blanket fuel material. Experimental work already in process has been terminated, with the exception of the corrosion evaluation program, being used to compare the effects of various fuel processing techniques, which will be completed in its entirety over a period of a few months. This group will proceed by taking on the seed development work for fuel plates, fuel assemblies, and control rods. In addition, a program on general fuel alloy and fuel element development is expected to be initiated for future core consideration.

Extrusion Development

During the experimental work for the development of processes for extruding Zircaloy-2 clad fuel rods it was found that chromium plated tools must not contact Zircaloy-2 at extrusion temperatures of 1600°F or greater because of severe adherence between the materials. However, considerable success in extruding Zircaloy-2 was realized by using hard faced dies. The lubricant found best for extruding Zircaloy-2 was a mixture of one part Moly-Kote to four parts Led-Plate thinned with Ucon. The use of this lubricant increased die life approximately 50% and kept striations along the length of the extruded rod to a minimum.

Rod Finishing

Previous work has demonstrated the difficulty in mechanical rod straightening operations, and since it has not been possible to produce rods of sufficient straightness by drawing alone, the program was extended to include hot stretching. The straightening of Zircaloy-2 clad U-12 w/o Mo rods by vertical stretching at temperatures of 25°, 300°, 600°, and 800°C has been investigated. Preliminary results indicate that a stress of 2500 psi at a temperature of 600° C or a stress of 500 psi at a temperature of 800°C is suitable for straightening rods to a maximum curvature of 0.006-in. per foot.

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Tensile data were obtained on the U-12 w/o Mo alloy after a 35 to 1 reduction in area by hot coextrusion in Zircaloy-2. Results from four specimens are:

	<u>Ultimate Stress (psi)</u>	<u>Total Elongation (%)</u>	<u>Reduction Area (%)</u>
Range	128,000 - 135,000	6 - 7.9	39 - 47
Mean	132,000	6.9	43.5

Final reports are being written describing the work accomplished on rod finishing.

Corrosion Testing of Product Rod

A group of 220 rods was placed in corrosion test in 680°F degassed water to determine the life of two different types of end closures. Ten failures have occurred in the first 68 days of testing from the original 110 rods with resistance butt welded end closures. One hundred rods are still in test. No failures have occurred after 54 days of testing of the resistance butt welded rods with a fusion seal weld and all 110 rods are still in test. The seal weld seems to be contributing in a definite manner to an improvement in corrosion life for the butt welded rods.

All of the material for the corrosion program comparing the effects of induction melting, consumable arc-remelting, wrought fuel, and fuel homogenization has been extruded and drawn into Zircaloy-2 clad rods, and will be prepared for the corrosion evaluation.

U-Mo End Closure Studies (J. G. Christ)

The selection of UO₂ as the reference fuel material for PWR Core 1 caused the termination of the alloy fuel rod end closure development program. Evaluation of processes developed to date is being completed. Battelle Memorial Institute is evaluating a process described below. An upset-butt welding process developed at the Bettis Plant was used to prepare about 250 specimens which will be used for evaluation at a later date. This project had been directed toward the development of an end closure process that would produce an absolute seal between the fuel element clad and the end cap and bonding between the fuel alloy and end cap. Most of the experience gained under this project is directly applicable to UO₂ end closures reported under project 53c24.

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Early in this report period the process variables for an upset-butt welded end closure were surveyed in detail. Specimen design was fixed at matching spherical recesses in both the fuel rod and the end cap. The recess was designed to have a maximum depth of 5 mils and to leave a flat, annular projection with a width of 25 mils around the periphery of the joint. The best cleaning procedure developed was to use freshly machined pieces, scrub in acetone, rinse in acetone, and rinse in alcohol. Welding dies were machined to close tolerances to insure uniform contact between the dies and work pieces. The dies were accurately aligned to insure uniform initial contact between mating surfaces of the work and this contact was checked before each weld was made by making an impression on a thin sheet of paper. The fuel rod was clamped in the dies to allow 10 to 15 mils extension beyond the dies and the end cap was allowed to extend about 3/16-inch. The welding current cycle was controlled in two separate phases; the first phase was a high energy, short time impulse to obtain an annular projection weld and the second phase was a longer, lower energy impulse to bond the fuel without excessive fusion. During this period, 110 enriched U-Mo fuel rods were completed by the above process. These rods were intended for the CR-5 proof test assemblies. A series of natural enrichment specimens was prepared for corrosion and destructive testing evaluation.

Work at Battelle Memorial Institute during this period was revised to stop further experimental work, but to finish the evaluation of the upset-butt welding of end closures with the upset restrained by use of an insulated sleeve. Preparation of specimens was completed, and evaluation is continuing on a program prepared jointly by BMI and Bettis.

Work at the Radio Corporation of America on R.F. induction welding of 0.3-in.-diam rods was terminated. The entire effort on this subcontract has been shifted to the UO₂ program.

Sensitizing Studies

Sensitizing studies on U-Mo rods have been discontinued.

Life Corrosion Tests

Life corrosion tests of end closures on a large number of U-Mo rods prepared by early end closure processes are continuing. The results to date on this program are tabulated below:

Days In 680°F H ₂ O	Butt Welded Only		Butt Welded Plus Fusion Seal Welded	
	No. of White Welds	Failed Rods	White Corrosion on Welds	Brown Corrosion on Welds
5	--	2	8	1
12	--	4	14	7
26	18	8	22	11
40	--	8	17	20
54	23	8	37	5
68	36	10	42	7

No. still in test 100 rods 110 rods
 200 welds 220 welds
 (Failed rods are actual penetration to the fuel)

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The above data show that the process selected for early U-Mo in-pile tests was probably satisfactory, but this process would not be satisfactory for long life service. These data also show that a simple screening by corrosion test is not a satisfactory test for end closures since new corrosion failures show up on extended corrosion exposure.

Non-Destructive Testing (J. G. Christ)

To establish the integrity of fuel elements and their components, the development of non-destructive testing techniques has been undertaken at the Bettis Plant. Selection of the UO_2 fuel material has considerably simplified the inspection problem. Whereas previously more effort was concentrated on the inspection of a capped, coextruded rod because of the recognized greater difficulty of such inspection, the effort is now almost completely shifted to the tube and end cap weldment inspection development.

Accordingly, the program outlined in Table III, "Summary of Non-Destructive Testing", WAPD-MRP-52, has been modified considerably and will now follow the tests outlined in Table I of this report.

The basic equipment required to pursue the work outlined is currently on hand or on order. With the exception of the 80-100 kc Magnatest, all necessary equipment is expected before the end of June.

Zircaloy TubingSurface Flaws

A procedure embodying the best technique developed to date has been recommended to the Production Section for the penetrant examination of tubing and finished elements and is reported in WAPD-FE-867, to be released soon. Calibration of the emulsification step in terms of defect depth, and an investigation of the effect of vacuum degassing on the sensitivity of the test, are currently being pursued.

Using collimated 10 mc beam with an immersed circumferential technique, longitudinal flaws 0.004-in. deep x approximately 1-in. long have been detected on the inner surface of 0.357-in. ID tubing. The signal was recordable and can be used to trigger an alarm. To improve reproducibility, more precise positioning and conveying equipment will be required.

The 1 kc Probalog has proved to be extremely sensitive to defects in tubing. Some indications picked up by this equipment could not be correlated with penetrant tests or metallographic examinations, but appeared to bear a relationship to superficial scratches observed with the boroscope. Evaluation and interpretation of the test results will require more work and may never be independent of an accompanying visual examination. Correlation with the Forster type equipment (Magnatest) and Cyclograph will be conducted when this equipment is received.

Subsurface Flaws

With the exception of the use of penetrants the techniques described above are applicable. It is encouraging to state that to date no serious subsurface flaws have been detected by the eddy-current, ultrasonic or X-ray techniques.

II-A-2 (Contd.)WAPD-MRP-54Tube Wall Thickness

Resonant ultrasonic equipment required for thickness inspection of the tubing is expected by mid-June. Tube conveying and crystal positioning equipment is presently being detailed.

General Quality (Proof Strength)

A hydrostatic proof testing method for production testing of tubing is being developed. To make the test practical and economical, a method of coupling with the high pressure line that will not consume Zircaloy tubing is being sought.

End ClosuresWeld Continuity (Porosity and Through Cracks)

Application of penetrant methods to detection of surface discontinuities on the end closure welds is as discussed for the penetrant inspection under Zircaloy tubing.

Fabrication of components for helium leak testing of the first and second end cap welds is proceeding.

Weld Quality and Strength

The Magnatest and Cyclograph, with probe and "C" coils, will be tried for examination of the quality of end closure welds. Also, the hydrostatic test referred to above will be tried.

In addition to the above tests, all end cap welds are being radiographed. X-raying will be required for all elements after both the first and second end closure weldment.

Fuel Element

Assembly voids among the oxide pellets in the tubes are being checked radio-graphically along with the second end closure weldment.

U-Mo Element Inspection

In addition to the above concentrated effort on inspection of components for the UO_2 element, a limited program continues at Battelle Memorial Institute on inspection of the cap-to-clad and core welds for samples welded in the consistency run of the U-Mo end closure reference process developed by BMI.

Fifty-two samples have been run on the 20 kc Multitest with annular coil and on the 100 kc Multitest and 100 kc Cyclographic with "C" coils. The as-welded samples gave spurious results due to roughness of the weld, and machining of the welds will be required if the results are to be interpreted.

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Of five samples tested previously, one showed a bad weld with the 20 kc "C" coil on the Multitest. Examination of the weld indicated extensive unbonding between the cap and core as well as inclusions at the interface.

Radiography of all welds made for the consistency run is nearly complete. Approximately 90% show voids, with some of the voids running as large as 0.070-in.

Examination of U-Mo elements will end when BMI completes the testing of the consistency welding run.

TABLE I

SUMMARY OF NON-DESTRUCTIVE TESTING

<u>Material and Type Defect</u>	<u>Test Method or Technique</u>
<u>Zircaloy Tubing</u>	
<u>Surface Flaws</u>	1. Penetrants 2. Ultrasonics (a) Immersed and/or contact, circumferential technique (single crystal) (b) Immersed, longitudinal technique (one or two crystal) (c) Immersed, diametrical technique (one or two crystal) 3. Eddy-Currents (a) 30 kc Magnatest with annular and probe coils (b) 80-100 kc Magnatest with annular and probe coils (c) 1 kc Probalog (d) Cyclograph - annular and probe coils
Subsurface Flaws	2 & 3 above
Thickness	4. Ultrasonics (a) Contact resonance technique
General Quality (Proof Strength)	5. Hydrostatic Proof Test
<u>End Cap Stock</u>	
<u>Soundness</u>	2-(c), 3-(a) and 3-(d) above
<u>End Closure</u>	
<u>Weld Continuity</u> (Porosity & Through Cracks)	1 above 6. Helium leak test 7. Ultrasonics - immersed, longitudinal technique (single crystal)
Weld Quality and Strength	3-(b), 3-(d) and 5 above 8. Radiography
<u>Fuel Element</u>	
<u>Voids</u>	8 above

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II-A-3WAPD-MRP-543. UO₂ Fuel Element Development (53c24) B. R. PriceCompactionVibratory

The revised experimental vibratory packing machine was received and is being installed.

Cold Pressing (Stokes Automatic Mechanical Press)

Pilot runs to gain greater experience with the reference process were initiated in this period under Process Engineering supervision. The results of these pilot runs will be statistically analyzed to evaluate the need for change or improvements in the reference process.

SinteringFurnaces

Four sintering runs were made in the small molybdenum wound resistance furnace procured from a commercial manufacturer. More than 2000 compacts were sintered to 92% - 95% theoretical density. A cold zone attachment has been designed. This modification will make possible the cooling of a sintering charge in a hydrogen atmosphere at much faster rates than is possible at present.

A check-out run of the large molybdenum wound sintering furnace designed and built at the Bettis Plant indicated the need of rewinding the resistor elements and improving the insulation. Detail drawings for a cold zone attachment to this furnace have been completed.

A de-oxidizing tube furnace unit was installed in the influx side of the CO₂ pre-treatment furnace. The de-oxidizing furnace contains copper turnings heated to 800°C through which the tank CO₂ passes for purification prior to being used in treating compacts for binder removal.

CO₂ Pre-Treatment

A total of 12 pre-treatment runs involving 9000 compacts were made in this report period. These compacts were all used for the various in-pile specimens and proof test assemblies.

It was reported in WAPD-MRP-53 that pre-treating in a CO₂ atmosphere results in densification of the UO₂ compacts. It has since been found that only binder free compacts exhibit this property. Compacts containing a binder, such as PVA, actually become less dense (0.5% - 1.0% decrease in density noted) after the 2 hr - 800°C pre-treatment. The apparent decrease in density is a result of the weight loss due to the removal of the PVA.

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Limited quantitative analytical information indicates that hydrogen sintering at 1725°C without prior CO₂ pre-treatment is quite effective in the removal of the 1 w/o PVA binder. Four sets of analytical results are tabulated below:

	<u>Residual Carbon in %</u>	<u>Residual Hydrogen in %</u>
1 w/o PVA - with pre-treatment	0.001	< 0.005
	0.003	< 0.005
1 w/o PVA - without pre-treatment	0.001	< 0.005
	0.004	< 0.005

Finishing Operations on Sintered UO₂ Compacts

"Glazing" of the silicon carbide wheel during dry grinding of some sintered high density enriched oxides made it necessary to use a coolant. The coolant which was used consisted of one part "Cimcool Transparent Coolant" rust inhibitor plus 100 parts water mixture. This coolant was very effective and eliminated "glazing" of the grinding wheel and chipping of the compacts.

Irradiation Test Samples

Shown in Table II are pertinent facts related to 59 sample elements which were fabricated for in-pile tests during this report period.

Corrosion Studies

A statistical comparison of the corrosion behavior of reference process oxide compacts sintered in hydrogen and tested in 750°F steam and 650°F water (each environment at 2000 psig) is shown in the following table.

Compacts Sintered in Molybdenum Wound Furnace - Hydrogen Atmosphere750°F SteamA. Without CO₂ Pre-Treatment

<u>Condition</u>	<u>20 hr</u>	<u>80 hr</u>	<u>100 hr</u>	<u>7 days</u>	<u>14 days</u>	<u>21 days</u>	<u>28 days</u>
Intact	26	11	12	11	11	6	
Eroded						6	5
Fractured		1		10	1	2	
Powdered				4			
	26	12	12	25	12	14	5

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B. With CO₂ Pre-Treatment

<u>Condition</u>	<u>20 hr</u>	<u>100 hr</u>	<u>7 days</u>	<u>14 days</u>	<u>21 days</u>	<u>28 days</u>	<u>42 days</u>
Intact	15	6	4	7	3	3	2
Eroded							1
Fractured			3		1		1
	15	6	7	7	4	3	3

650°F WaterA. Without CO₂ Pre-Treatment

<u>Condition</u>	<u>20 hr</u>	<u>1 day</u>	<u>6 days</u>	<u>7 days</u>	<u>14 days</u>	<u>21 days</u>
Intact	8	8	18	4	18	6
Fractured	1	2	1			
Powdered	1					
	10	10	19	4	18	6

B. With CO₂ Pre-Treatment

<u>Condition</u>	<u>20 hr</u>	<u>7 days</u>	<u>14 days</u>	<u>21 days</u>
Intact	2	2	4	2
Fractured				
Powdered				
	2	2	4	2

Thermal Shock Experiments

A series of exploratory experiments was completed to determine the reaction of 0.350-in. diam UO₂ cylindrical fuel slugs to controlled thermal shock conditions. Three separate experiments were set up. In the first test, pellets were quenched in water after heating in a hydrogen furnace at 970°C; in the second and third tests, pellets were heated in the flames of gas torches to either 1150°C or 2100°C and quenched in water.

Conclusions:

1. Processing history (argon or H₂ sintering, lapping, and grinding) appeared to have little or no effect on fuel slugs quenched from 970°C to 20°C.
2. UO₂ cylinders whether ground or unground, will remain intact upon quenching from 1150°C or 2100°C if they are heated slowly to temperature. (Cracks which appear during rapid heating are the focal points for fractures which occur on quenching.)
3. All pellets tested showed a small amount of surface fragmentation.

TABLE II

Designated Test	No. of Elements	Tube OD (in.)	Tube ID (in.)	Fuel Diam (in.)	Fuel Length (in.)	Finished Rod Length (in.)	Fuel Material	Av. Fuel Density %
MTR	4	0.4	0.346	0.3445	2.5	3.2	+325 MCW	92 0.010"
Rabbit	2	0.4	0.346	0.3445	2.5	3.2	MCW + 1 w/o PVA	93 Defect
NaK Steady State	4	0.4	0.346	0.3445	4.5	5.2	+325 MCW	93
	2	0.4	0.346	0.3445	4.5	5.2	MCW + 1 w/o PVA	93
Out-of-pile Thermal Cycle	2	0.4	0.346	0.3445	4.5	5.2	+325 MCW	92
	2	0.4	0.346	0.3445	4.5	5.2	MCW + 1 w/o PVA	93
HT-1 (1st Oxide Test)*	10	0.414	0.3585	0.357	9.2	10-15/16	1 w/o PVA 3.5 w/o U ²³⁵	96
HT-1 (2nd Test)**	11	0.414	0.3585	0.357	9-1/16	10-3/4	3.5 w/o U ²³⁵ +1 w/o PVA	96
Central Thimble***	15	0.414	0.3585	0.357	9.25	10-3/4	7.4 w/o U ²³⁵ +1 w/o PVA	96
	7	0.414	0.3585	0.357	9.25	10-3/4	5.4 w/o U ²³⁵ +1 w/o PVA	96

* The first HT-1 test mentioned above consists of a fuel bundle of nine reference size fuel rods, the center rod of which has been bent so as to be forcibly jammed against two adjacent rods. The purpose of this test is to find if a completely blocked channel, such as might occur from a water-logging accident, will induce failure in neighboring sound rods. The 3.5 w/o enrichment is necessary to obtain a heat flux of 311,000 Btu/hr/ft² in the neutron flux available in the in-pile loop. The test was begun at low neutron flux on May 11.

** The second HT-1 test is similar to the first except that the center rod contains about 0.1 cc of water and has not been bent. It is expected that in this test the center rod will rupture during reactor start-up. The subsequent damage, if any, to other rods will be studied.

*** The Central Thimble tests are proof tests of good bundles. In order to simulate the thermal conditions which will arise in the PWR core due to thermal neutron flux gradients, adjacent rows of rods will contain UO₂ of progressively increasing enrichment.

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UO₂ End Closure Studies (J. G. Christ)

Following the selection of UO₂ as the PWR Core 1 fuel material, the total end closure development effort has been shifted to this project. This transition required the machining of new tooling and the reallocation of materials to the various sites participating in the end closure program. This program requires the development of absolute seals for the ends of oxide filled tubes of adequate design and strength to resist thermal distortions and stresses in service.

The flash butt welder at the Bettis Plant has been retooled to evaluate projection-type butt welds and flash butt welds of tubing. These two approaches are presently being compared to determine the major effort for the use of this equipment. The purchase of a percussive welder built in the Fisher Body Division Laboratories has been arranged. This equipment will be installed at Bettis for the development of UO₂ end closures.

An inert-arc fusion welded end closure process was established to provide fuel rods for the immediate in-pile program. This process consisted of a double fusion weld on a fitted joint between the end plug and tube. The end plug was shouldered to a depth of 5/32-in. and an interference fit between the plug and tube was used. A copper chill block was clamped over the fuel and the weld was made as close to the edge of the fuel as possible. The double weld was used to reduce porosity and to insure uniform penetration. Evaluation of variables in the inert-arc fusion welding process is continuing.

The combination fusion welded-hot swaged end closure is being investigated with UO₂ fuel pellets. Variables include the geometry of the ends before swaging, swaging temperature, and reductions. This work has yielded heavier, worked weldments and internal radii at the end closures.

The evaluation of end closure designs by mechanical tests has been started. Comparative results on static tension, impact tension, and rotating beam fatigue tests are being obtained.

The total experimental effort on subcontract work at Battelle Memorial Institute has been shifted to the UO₂ program. Methods being evaluated are flash welding, upset-butt welding in an insulated sleeve, and use of overlapping spot welds between the tube and deep plug. The results of these methods will be compared early in the next report period to select a reference process for detailed evaluation at BMI.

Work on the Uniskan process at RCA has been held up until seamless tubing becomes available. The development of coil designs for R.F. induction welding has been shifted to the oxide rod diameter. The shape of coils developed has increased the allowable tolerances on alignment. Attempts to obtain deeper penetration with coils designed to give a wider weld pool have been unsuccessful.

4. Fuel Element Development (53c25) J. G. ChristBlanket Fuel Assembly Development

The development program involves evaluation of processes and components which will be used to manufacture natural fuel assemblies for the PWR. The work is divided into 2 main categories as follows:

- a. Evaluation of various types of fuel rod bundles and their manufacture.
- b. Development of practices for making the natural assembly shells and accessories which will contain these bundles.

The choice of fuel did not materially affect either of these programs. Although the specific rod size will result in a different level of values obtained in various tests, the generalized comparisons between types of bundles should still be valid.

Rod Bundle Evaluation

Work has been primarily concentrated on the two bundle reference designs. These are the tube sheet (fusion welded) and egg crate (mechanically joined) bundles. Since a final choice on bundle type is to be made by July 1, every effort is being made to have as many data available as possible with this date in view.

Informative discussions have been held with various people interested in this design development and its attendant problems. Full-size Zircaloy-2 bundles are being constructed in order to further evaluate the ease and cost of manufacture and assembly of the two references and to study their general structure.

As a result, it has been shown that the welded structure is more rigid as a unit than its mechanically joined counterpart. Furthermore, the general feeling is that the mechanical type would be the more susceptible to fretting corrosion under service conditions. On the other hand, it would probably be somewhat more easily inspected for general corrosion susceptibility, since individual parts could be corrosion tested and inspected prior to assembly. The welded type must necessarily be tested and inspected subsequent to welding.

Prior to making the final selection, some additional data are being obtained on stress-strain relationships by bend deflection on individual rods as supported within the two types of bundles. Thermal cycling data obtained on an unfixed fuel rod show that there is a longitudinal bowing after 500-1000 cycles between 1050 and 550°F. Consideration must be given as to how the bundle as a whole would react to these conditions.

Other tests being planned from a longer range viewpoint include in-pile proof tests of the two types of rod bundles, thermal distortion and thermal cycling tests on sub-size bundles, fretting corrosion and wear tests on an assembly mock-up containing both types of rod bundles, and a study of rod vibration under dynamic waterflow as influenced by the two reference bundle designs.

Since spacings between fuel rods within a bundle are an important consideration in design, much effort has been devoted to establishing the optimum means of measuring these openings. Currently a small flat probe has been developed which operates mechanically. It is inserted between two rods and a fixed rotary moment is applied to bring it into contact with adjacent rods. This rotary motion is translated into a linear separation measurement. The new probe replaces a similar type where the rotary motion was applied by the operator manually. In this case the rotary moment was established by the "feel" of the operator.

Other possibilities are constantly being reviewed for this job. Presently another mechanical method, a "Dynamic Micrometer", and air gauges are under investigation.

Industrial boroscopes have been proven to be useful in inspecting surfaces in the interior of in-pile test rod bundles. If no less wearisome means of doing this inspection is developed, they will have to be used on production size bundles as well.

A rotating electrode is under development as the initial step in making an automatic operation for welding rod ends in the end plates for the tube sheet bundles. To date the drawings have been completed and construction of the unit which is designed to fit into a laboratory welding box has been started.

Development of Production Techniques for Making Assembly Shells

The rolling of the 12-in. wide Zircaloy-2 strip to be used in the development program is essentially complete. Some special handling in the mill was required since a high temperature, vacuum heat treatment was specified. This feature was a result of the preliminary investigations made on bending properties of Zircaloy-2. The material will be used ultimately for forming into channels and/or other shapes being considered as possibilities for ultimate manufacture of the square (5-1/2-in.) assembly shells.

Six-inch lengths of 0.162-in. thick Zircaloy-2 strip have been formed to a 90° angle with a radius of 0.125-in. by Youngstown Welding and Engineering Company on a development purchase order. Various combinations of applying heat to the work and dies were tried. Generally it seems that heating to the range of 850-1000°F is required for forming to the above conditions.

Another company, Brooks and Perkins Co., has been contacted in reference to doing hot forming operations on Zircaloy-2. They feel they could handle both the bending of channels to form the assembly shells and the deep drawing of transition pieces which will be welded to the ends of the shells. Their experience in hot forming operations is considerable. To date they have formed several small deep-drawn cups from Zircaloy-2 material.

An extensive program for additional investigation of factors affecting the bending properties of Zircaloy-2 has been started. Results of this experimental work should aid in writing future specifications for forming of this material.

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A welding jig to be used to join the components which will make up the square tubes is in the design stage.

Future plans include the completion of the above-mentioned items and the awarding of development purchase orders for trial forming of angles, channels, transition pieces, and other possible shapes which may be used as assembly components.

In-Pile Tests

Following the selection of UO_2 as the PWR Core 1 fuel material, the in-pile testing program was completely revised. The MTR-(HT-1) program was revised to a complete oxide bundle evaluation program. The NRX-CT test was redesigned to allow the testing of two oxide rod bundles. Both bundles will have a power gradient achieved by use of an enrichment gradient along rows of rods within the bundle. One bundle will be a mechanical assembly of rods; the second will be a completely welded bundle.

The fabrication of the components for NRX-CT test section is proceeding to a scheduled completion on June 28. The structural components for the test section have been completed and the fuel rods are in process.

The following bundles have been completed for the MTR-(HT-1) testing facility:

- HT-1A - U-Mo bundle with sensitized and defected center rod (being held)
- HT-1B - Spare for HT-1A (being held)
- HT-1C - UO_2 rod bundle with bent center rod touching 2 neighbors (in test)
- HT-1D - Dummy bundle with one calibration pellet in center rod. Pellet exposed by 3-10 mil holes in tube.
- HT-1E - UO_2 bundle with water-logged center rod

5. Thermal and Hydraulic Properties of Fuel Elements (55a15) J. E. Zerbe

The work on this project for the period from May 20 to July 20 will be covered in WAPD-MRL-6.

6. Pilot Fuel Element Manufacture (53c41) W. J. Hurford

Approximately 64% of the capital equipment scheduled for procurement during fiscal 1955 for the UO_2 program is on hand. An additional 36% is on order. Equipment installation work has been changed from an equal effort on U-Mo and UO_2 fuel element equipment to a maximum effort on UO_2 equipment in line with the selection of UO_2 as the PWR fuel material.

Pilot equipment has been installed for the reference UO_2 fuel pellet process with the exception of hydrogen sintering furnaces and face grinders. Several sintering furnaces are being constructed at the Bettis Plant. The first of these should be in operation by July 15. The face grinder is scheduled for delivery during the latter part of June and should be in operation by July 15. At present, pilot runs are being made on 25 lb batches with the available production equipment. Cold compacted pellets are being produced satisfactorily. Samples from these batches will be sintered in a laboratory furnace to determine their quality.

Slightly enriched (1.3%) UO_2 fuel is being processed into 0.600 in. diam uranium dioxide pellets having a density of 70% theoretical for the two-region critical experiment. Agglomerated and dried UO_2 powder, containing 1.5 w/o polyvinyl alcohol (PVA) did not yield a sound pellet under automatic compacting conditions. The top of the pellet flaked during ejection from the die. Similar processing of natural UO_2 powder results in satisfactory pellets. The different compacting characteristics of the natural and slightly enriched UO_2 are attributed to the greater amount of material finer than 325 mesh in the enriched UO_2 powder. An addition of 0.1 w/o aluminum stearate to the enriched powder provided sufficient lubrication for production of satisfactory pellets. The binders are removed by a CO_2 heat treatment.

Core Manufacture

Request for quotations on Zircaloy-2 tubing, meeting tentative PDS 12650, for the blanket cladding were sent out to commercial suppliers.

B. Metallurgy of Core Materials1. Physical Metallurgy of PWR Core Materials (53b11) Z. M. ShapiroPreparation of UO₂

Work has continued on the preparation of UO₂ by various methods for irradiation and fabrication tests as well as for investigations of the physical and chemical properties of the oxide itself.

Hydrogen Reduced UO₃

During this period, 4 kg of UO₂ were prepared for irradiation tests by hydrogen reduction of UO₃ at 800°C. One kilogram of natural oxide was sent to the Armour Research Foundation for hot-pressing. Two thousand four hundred grams of slightly enriched and 600 g of fully enriched UO₂ were for use in in-pile experiments at both MTR and Chalk River. This powder was prepared so that its properties would be quite close to those of the Mallinckrodt oxide.

A study of the effect of UO₃ particle size on the U(VI) content of UO₂ prepared in a manner similar to that of MCW indicates that the best oxide is obtained by reducing -200 mesh UO₃ with hydrogen at 800°C for 16-20 hr at hydrogen flow rates between 0.1 - 0.2 l/hr/g UO₂. The powder produced in this manner is brown in color and contains less than 1% U(VI) and 0.1% water.

Activated Uranium Dioxide

Battelle has reported that UO₂, prepared from UO₃ treated with dilute H₂SO₄, sinters to almost theoretical density. A quantity of this activated UO₂ was prepared with the object of determining (1) whether the results could be duplicated at Bettis and (2) what the reasons were for the improved sinterability.

A slurry of Mallinckrodt UO₃ in sulfuric acid was tumbled for 4 hr in a Zircaloy-2 ball mill, transferred to polyethylene-lined trays, and dried for approximately 16 hr at 80°C in a forced convection oven. The dry cake was ground to pass through a 100 mesh sieve and then reduced to UO₂ with dry hydrogen at 800°C. The results of the chemical analyses of the UO₂, accurate to within $\pm 0.2\%$ are given below.

Table IActivated Uranium Dioxide

<u>Reduction Time (hr)</u>	<u>% U</u>	<u>% U(VI)</u>
15.5	87.9	2.5
15.5	87.9	2.7
15.5	87.9	2.7
33.5	87.9	1.8
33.5	87.9	2.1

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It will be noted that, for the most part, this material contains more than 2% U(VI) unlike MCW UO_2 which contains between 1 - 1.5% U(VI). Samples of this "activated" oxide are now being analyzed for sulfur. The sinterability is also being checked.

Precipitation Studies

Work has been resumed on the preparation of UO_2 by various methods in order to determine which method produces a material most suitable for application to PWR fuel elements. In this connection, a number of uranium salts were prepared by precipitation with ammonium hydroxide or hydrogen peroxide preparatory to reduction with hydrogen. Variables such as pH, concentration, temperature, rate and manner of precipitation and digestion periods, were carefully controlled.

The various materials are now being characterized by measurement of index of refraction, particle size distribution, crystallite size, structure, etc., prior to conversion to UO_3 or U_3O_8 and UO_2 . Also, equipment is being constructed for performing these conversions under carefully controlled conditions.

In support of the irradiation program, a limited study was made of the solubility of UO_2 in various media in order to determine how best to dissolve irradiated sintered pellets of UO_2 . The results are summarized in the following table. It will be noted that boiling concentrated nitric acid was found to be the best solvent of those tried.

Table IISolubility of 6.5 g Sintered UO_2 Pellets in Various Media

<u>Solvent</u>	<u>Volume (ml)</u>	<u>Temp. (°C)</u>	<u>Time (hr)</u>	<u>Results</u>
concd. H_2SO_4	20	30	24	No change.
concd. H_2SO_4	50	160	24	No change.
concd. H_3PO_4	20	30	24	No change
concd. H_3PO_4	60	150	24	1/4 pellet dissolved
1:1 HNO_3	20	30	24	2/3 pellet dissolved
1:1 HNO_3	60	boiling	2.5	pellet dissolved*
concd. HNO_3	20	30	24	pellet dissolved*
concd. HNO_3	40	boiling	1.5	pellet dissolved*

*uranyl nitrate crystals formed

Carbon Doped UO_2

The carbon doped samples prepared as described in WAPD-MRP-53 in support of the program to determine the sensitivity of sintered UO_2

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compacts to carbon were analyzed and found to contain considerably less carbon than was originally inserted. Since the carbon was probably removed during hydrogen reduction of the UO_3 , attempts are now being made to disperse the carbon homogeneously by soaking the UO_2 in a sugar solution, drying and then decomposing. Samples prepared in this manner are now being analyzed for carbon content and homogeneity.

Oxygen Doped UO_2

The preparation of UO_2 with oxygen-to-uranium ratios greater than two is being continued as indicated in WAPD-MRP-53 in support of the program for determining the effect of composition on the properties of the material. In preparing these samples it has been observed that the rate of reaction with oxygen at a pressure of 1 atmosphere and at temperatures between 150 and 500°C is considerably faster than the rate of diffusion of the gas through the mass of powder. Accordingly, the samples oxidized inhomogeneously unless the powder is thoroughly spread in the reaction tube. On the basis of the difficulty in obtaining duplicate analyses, it appears that when the O/U ratio exceeds 2.3 the oxidation takes place inhomogeneously even if the powder is spread thinly. The results are being checked by X-ray diffraction studies.

Sintering of UO_2

A program has been initiated to study the relative importance of the various factors affecting the sinterability of UO_2 to aid in the determination of the sintering mechanisms as well as to act as a guide in the preparation of suitable material. The factors being investigated include O/U ratio, additives, surface area, crystallite size, particle size and distribution, as-pressed density, sintering atmosphere times and temperature. The effect of these variables on the density, microstructure, O/U ratio, and ratio of connected-to-closed pore volume of the sintered compacts will be observed.

In the preliminary series of runs it was found that a density of 93% could be achieved in 2 hours at 1750°C in a hydrogen atmosphere starting with a 70% dense MCW compact, while a density of 95% could be obtained starting with a 75% dense MCW compact. The O/U ratio of the starting material was approximately 2.03. These results are in conflict with those obtained by the British at Harwell who have indicated that an oxygen-uranium ratio of approximately 2.13 and absence of H_2 from the sintering atmosphere was necessary to achieve these high densities. These results will be checked.

Stability of UO_2 in Water

The long-term tests on the stability of UO_2 compacts in high temperature water, reported on in WAPD-MRP-53, were continued. The specimens which had previously undergone a 4 day test in degassed water

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at 650°F, were exposed for an additional 14 days under the same conditions. No significant changes were observed in the appearance of the compacts at the end of this period. Representative samples of the group were weighed after drying; the weight changes after the full 18 day exposure are summarized in Table III.

Weight losses were observed only in those cases where disintegration had occurred during the initial 4 day test. These losses are attributed to the carry-over of some of the fine UO_2 powder from the test container during operation of the autoclaves. Thus, no quantitative significance can be attached to weight changes obtained with powder samples.

It will be noted that, in general, the weight changes on both the sintered and the green compacts are small, and that the gains observed are due principally to the corrosion film which developed on the Zircaloy-2 containers as evidenced by the gain in weight of the empty container. When corrected for the increase in weight of the Zircaloy-2 container, the largest weight gain (sample 184) was only 5 mg. It will be noted, further, that the weight changes of the green compacts are approximately equal to those of the sintered compacts despite the fact that the surface areas available for reaction in the case of the unsintered compacts are much greater than those available in the case of the sintered specimens. Thus it may be concluded that, at least initially, UO_2 is very resistant to oxidation at 650°F even in water containing no hydrogen. The samples have now been returned to the autoclaves for further test.

A number of samples comparable to those tested in degassed water have been exposed to water containing 25-30 cc of $\text{H}_2/\text{kg H}_2\text{O}$ for 12 days at 650°F. The reaction of the compacts was similar to that of the compacts tested in degassed water, i.e., the compacts sintered in hydrogen in an alundum furnace retained their physical character whereas those sintered in argon in a carbon furnace disintegrated.

Additional UO_2 pellets which have been sintered in hydrogen as well as some which have been hot pressed are now being tested in 650°F degassed water, 650°F water containing 25-30 cc of $\text{H}_2/\text{kg H}_2\text{O}$, and in 750°F degassed steam. An autoclave is also being prepared to test samples at a high pH using LiOH .

Six uncorrosion tested UO_2 compacts, some of which were sintered in H_2 in a alundum furnace and some in argon in a graphite resistor furnace, which were duplicates of numbers 167, 168, 169, 170, 172, 173, in Table III, were analyzed for carbon to determine the degree of pickup on sintering in a graphite resistor furnace. The results are given in Table IV.

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Table III

UO₂ Corrosion Test - 18 Days in Degassed Water at 650°F
(0.04 cc O₂/kg max)

Sample #	UO ₂ Powder	Sample Preparation	Appearance of Compact After Exposure	Wt Change of Compact and Zircaloy Container (gross)
164	MCW 5-8 +325 frac.	cold pressed, unsintered	no change	+0.0064
165	NUR-1	cold pressed, unsintered	no change	+0.0097
166	NOH-18	cold pressed, unsintered	no change	+0.0071
167	MCW 5-8	cold pressed, sintered in argon in carbon furnace	powder	-0.0060
168	NUR-1	cold pressed, sintered in argon in carbon furnace	powder	-0.0055
169	NOH-18	cold pressed, sintered in argon in carbon furnace	chunks and powder	+0.0100
170	MCW 5-8	cold pressed, sintered in hydrogen in alundum furnace	no change	+0.0058
172	NUR-1	cold pressed, sintered in hydrogen in alundum furnace	no change	+0.0086
173	NOH-18	cold pressed, sintered in hydrogen in alundum furnace	no change	+0.0070
180	NOH-14	cold pressed, sintered in argon in carbon furnace	powder	-0.0084
182	MCW	hydrostatically pressed, sintered in hydrogen in alundum furnace	no change	+0.0072
184	MCW	hot pressed at 1750°C - 15 min	no change	+0.0123
185	blank cup			+0.0076
186	blank cup			+0.0063

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Table IV

Carbon Contents of UO_2 Compacts

<u>UO_2 Powder</u>	<u>Sample Preparation</u>	<u>Carbon Content ppm</u>
MCW 5-8	cold pressed, sintered in argon in graphite furnace	50
NUR-1	cold pressed, sintered in argon in graphite furnace	30
NOH-18	cold pressed, sintered in argon in graphite furnace	30
MCW 5-8	cold pressed, sintered in hydrogen in alundum furnace	10
NUR-1	cold pressed, sintered in hydrogen in alundum furnace	10
NOH-18	cold pressed, sintered in hydrogen in alundum furnace	10

It will be noted that the samples fall into two distinct categories with respect to carbon content, yet those sintered in graphite contain only 20-40 ppm more carbon than those sintered in hydrogen. Whether this difference in carbon content is responsible for the marked difference in the resistance of the two types of compacts to desintering is not certain. The possibility that another impurity causes the compacts sintered in the graphite furnace to disintegrate is now being thoroughly checked.

Five sintered UO_2 compacts (obtained from KAPL) to which 1% TiO_2 , Cr_2O_3 , U_2O_3 , CaF_2 , or Al_2O_3 had been added as densifying and strengthening agents, as well as 2 compacts containing 0.25% and 0.5% TiO_2 , were tested in degassed water at 650°F for 5 days. The appearance of the compacts remained unchanged and the weight changes were less than 1 mg in every case. The samples have been reinserted in the autoclaves for further test.

Kinetic Studies

Samples of $UO_{2.004}$ powder, prepared by reducing MCW $UO_{2.03}$ in hydrogen at 800°C, and samples of $U_3O_8(UO_{2.663})$ prepared by air oxidation of MCW UO_2 at 800°C, have been exposed to degassed water, to water containing 100 cc $H_2/kg H_2O$, and to degassed water containing 25 cc $O_2/kg H_2O$, in an effort to determine the rates of oxidation and reduction of uranium oxides in the presence of 650°F water. Several samples have been removed after 5-day exposures. These will be analyzed to determine the changes in oxygen content. Other samples will be removed after they have been in test for 15 days. Additional runs will be made in mixtures of oxygen and hydrogen dissolved in high temperature water and steam. Surface area, density, and X-ray measurements will be made on selected samples in an attempt to relate the rate of chemical reaction to the physical character of the oxides.

Surface Area and Density of UO_2 Powder

Surface area and density measurements have been performed on samples of $UO_{2.004}$ prepared by reduction of MCW $UO_{2.015}$ in hydrogen at $800^\circ C$ for 20 hr, and on $U_3O_8(UO_{2.663})$ prepared by ignition of the same starting material in air at $800^\circ C$ for 6 hours. The purpose of these measurements was to aid in the characterization of the powders used in the study of the kinetics of the reaction of UO_2 with high temperature water containing oxygen and hydrogen. The specific areas obtained for the $UO_{2.004}$ and U_3O_8 samples were 0.61 and $0.31 \text{ m}^2/\text{g}$ respectively after outgassing at a pressure of 10^{-5} mm Hg at $400^\circ C$ for 6 hours. The densities were 10.15 and 7.91 g/cc respectively.

The surface area and density values for the $UO_{2.004}$ sample are very close to the values obtained for MCW $UO_{2.015}$ from which the oxide was prepared. This indicates that little sintering occurs in the powder when it is heated in hydrogen at $800^\circ C$. However, the surface area of the U_3O_8 powder is approximately half that of the starting material in spite of the fact that the oxidation was accompanied by a 22% decrease in density. On the basis of these data it appears that considerable sintering occurs in ignition of UO_2 to U_3O_8 in air at a temperature as low as $800^\circ C$.

Preliminary experiments have been performed on a dye adsorption method for determining the surface area of UO_2 . This work was done in an effort to develop a simple, rapid technique for making routine area measurements.

The degree of adsorption of methylene blue chloride in a propyl alcohol solution by UO_2 powder was determined spectrophotometrically. Samples of MCW and high pressure steam oxidized UO_2 were thoroughly shaken in a $3 \times 10^{-4} \text{ M/l}$ solution of the dye. The weight of dye adsorbed per gram of UO_2 was $6.5 \times 10^{-5} \text{ g}$ in the case of the steam oxidized UO_2 and $8.8 \times 10^{-5} \text{ g}$ in the case of the MCW UO_2 . These results are in opposition to those obtained by the B.E.T. method which indicate that the surface area of steam oxidized powder is twice that of MCW powder. However, the results obtained with the dye may bear some relation to the fact that MCW powder sinters to a higher density than steam oxidized powder in spite of the fact that the initial MCW compact density may be lower than that of the steam oxidized compact. Accordingly, additional experiments will be performed with the dye in an attempt to determine the significance of the results obtained.

An air permeability apparatus for determining surface area of fine powders has been constructed. The surface area values obtained by this method will be compared with the B.E.T. values to determine if the method can be applied to routine measurement of the surface areas of UO_2 powders.

Magnetic Susceptibility Studies

Preparation of the series of uranium oxide samples having O/U ratios between 2.00 and 2.67 has been completed, and the samples have been transmitted to the Magnetism Department of the Carnegie Institute of Technology for measurement. As indicated in WAPD-MRP-53, it is expected that these measurements will not only yield a quantitative indication of the oxygen content but also yield information regarding the oxidation states of the uranium. The measurements are scheduled to be completed by June 30.

Diffusion Studies

Experimental work on the diffusion of inert gases in UO_2 has been delayed pending completion of necessary modifications to the laboratory facilities. It is expected that experimental work will commence again by June 30.

The UO_2 - O_2 Equilibrium

As in the case of the diffusion studies the major portion of the experimental work is being delayed until the alterations to the laboratory can be completed. A thermal balance which will be used in conjunction with this work is now under construction and will be completed by June 30.

The Reaction of Zirconium with UO_2

Work at the Battelle Memorial Institute under subcontract to Bettis Plants on the reaction of UO_2 with zirconium is nearing completion.

The results of the long-term experiments at temperatures of 750, 950, 1150, and 1300°F are now being evaluated. Thus far there has been no indication of reaction at 750°F. The reaction at 1150°F proceeds very slowly while that at 1300°F proceeds more rapidly. It is expected that a detailed report on this work will be completed by BMI by June 30.

Zr- UO_2 Phase Diagram

Work on the determination of the equilibrium diagram of UO_2 and zirconium is being continued at Battelle under subcontract to Bettis Plant.

While the diagram has not been completely explored, sufficient work has been completed to enable the construction of a tentative diagram. A preliminary report will be issued by BMI by June 30.

Thermal Conductivity of UO_2

Work on the measurement of the thermal conductivity of UO_2 at temperatures above 1000°C is being continued at the Armour Research Foundation (ARF).

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The components of the new apparatus have been completed and the apparatus is now being assembled. ARF has indicated that some measurements would be completed before the termination of their contract on June 30.

Sintering Additives

Studies are continuing at ARF under subcontract to Bettis Plant on sintering aids for UO_2 . Considerable improvement in the sinterability of MCW oxide was noted when approximately 1% ZrO_2 was added to the UO_2 in the form of ammonium zirconyl carbonate $[(\text{NH}_4)_2 \text{ZrO}(\text{CO}_3)_2]$. The action of this material was further enhanced by admitting steam to the furnace. In this manner densities of approximately 92% theoretical were achieved on sintering for only 1 hr at 1500°C .

2. Mechanical Metallurgy of Core Materials (53b21) B. Lustman

No work has been done on this project for this period.

3. Corrosion of Core Materials in Static Systems (53b41) E. F. Losco

Since UO_2 was selected as the fuel for the first PWR core, the intensive development work on U-12 w/o Mo and U-10 w/o Nb alloys is being de-emphasized. A new metal alloy development program has been prepared and submitted to AEC for approval. The new program is aimed primarily at developing metallic fuel alloys which have more attractive nuclear properties than the past reference alloy, U-12 w/o Mo or U-10 w/o Nb, and which permit greater flexibility in reactor design and low cost fuel element fabrication. Only those experiments, initiated in the past, which fall in line with the new program and which will contribute to the above aims will be continued. Those which were specifically directed toward evaluation of U-12 w/o Mo or U-10 w/o Nb fuel elements are being rapidly discontinued or brought to a logical stopping point so that summary reports can be prepared. This is indicated for the individual experiments discussed below.

Effect of Water Additions on Corrosion of U-Mo

To understand the corrosion behavior of U-12 when exposed to 343°C water, additions were made to the test water that would act as hydrogen depolarizers and decrease the amount of hydrogen picked up by the samples and hence extend the corrosion life. As reported previously, additions of both Ni and Pt ions greatly retard the hydrogen absorption but only Ni ions offer any pronounced extension of the corrosion life. Some question exists concerning the effect of adding small amounts of $\text{Pb}(\text{AC})_2$ along with the Pt ions. This latter question is currently being studied by testing a new extrusion (samples from the original extrusion have all been used) in a solution of chloroplatinic acid containing lead acetate and in water containing $\text{Pb}(\text{AC})_2$ alone. The effect of adding Hg^+ ions as Hg_2SO_4 is also being studied using the new material. The behavior of samples of this extrusion are also being studied for comparison purposes in degassed water and water containing NiSO_4 .

It is apparent from data taken to date that Hg^+ additions in these amounts have no detrimental effect on the corrosion of U-12 Mo. Hg, as expected, is not effective as a H_2 depolarizer. It is possible that Hg plated out of solution even though fresh additions were made after each corrosion test period. Samples will be tested to failure and the hardness, metallography, and hydrogen content will be observed on samples removed from test prior to failure.

While the U-12 w/o Mo alloy is not of primary interest since UO_2 was selected for the first PWR core fuel, it is representative of gamma phase U-base alloys. The results of this work, therefore, are of general basic interest in studying the mechanism of corrosion of U-base gamma phase alloys. The work will be continued and the data collected for publication in a comprehensive report on effects of water additions.

U-Mo-Pt Ternary Alloys

A series of arc-melted buttons were made to evaluate the effect of Pt additions in the metal, instead of the water, on the corrosion and decomposition of gamma U-Mo alloys. These alloys showed that Pt additions improve the corrosion rates and lives of as-cast samples, gamma-quenched samples, and

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samples annealed at 400 and 500°C. All of the samples that have not yet failed have been discontinued and will be examined for hardness, hydrogen content and metallography. A summary report will be written.

As a follow-up, two duplex melted and co-extruded alloys, U-8 Mo-1 Pt and U-12 Mo-0.5 Pt, were made and tested in various conditions. The results of testing these samples essentially confirm the data on the samples from the arc-melted buttons. These samples will be tested to failure or until considerably more time is accrued in order to bring the experiment to a logical conclusion.

U-Nb Binary Alloys

In order to better evaluate U-Nb binary alloys, bare samples from three duplex melted co-extrusions (U-8, 10, 12 Nb) were corrosion tested in various conditions. Long annealing times (up to 35 days) were studied at 400 and 500°C but all of the samples failed in 0-1 day. These niobium alloys, as previously reported, lack thermal stability at 400 to 500°C. Transformation of the gamma phase quickly destroys their corrosion resistance.

The metallography of the samples annealed for short times at 400 and 500°C has now been completed for the U-8 and 10 Nb alloys. Annealing at 400°C for 7 days results in a high alpha grain boundary precipitate that grows with time, and annealing at 500°C for 2 days results in a transition structure that closely resembles "distorted" alpha which breaks down upon further annealing to a high alpha structure. The samples undergo maximum hardening during the first annealing period and thereafter become progressively softer. The kinetics of the transformation of the 8% alloy are, of course, more rapid than those of the 10% alloy.

The samples now in test will be tested to failure and samples of U-12 Nb are currently being annealed at 400 and 500°C for purposes of post annealing corrosion evaluation. Once this phase of the work is completed, the experiment will be considered complete.

U-Nb-Pt Ternary Alloys

In view of the pronounced improvements offered by Pt additions to U-Mo binary alloys, a duplex melted and co-extruded U-10 Nb-1 Pt alloy was fabricated to evaluate the effect of Pt additions on the corrosion properties and annealing characteristics of U-Nb binary alloys.

It was observed metallographically that annealing at 400°C up to 35 days did not produce heavy grain boundary precipitation but instead a very fine precipitate (not well resolved) throughout the matrix. The hardness was still increasing after 35 days although most of the hardening took place during the first 7 days (from 300 DPH as γ Q to 600 DPH after 7 days at 400°C). Annealing at 500°C for 2 days produced a transition structure similar to that observed in U-Nb binary alloys. Continued annealing resulted in transformation to a high alpha structure. Thus, Pt additions to U-Nb base alloys have a pronounced effect on the metallography of specimens annealed at 400 and 500°C. Whether this effect is one of slowing down the transformation is not readily discernible.

The samples still in test will be continued to failure and a summary report will be written concerning the corrosion resistance and the metallography.

Homogenization of Unclad U-12 w/o Mo Alloys

A program is in progress to determine the annealing time required to obtain optimum corrosion properties of U-12 Mo extrusions when homogenized at temperatures of 600 to 1000°C. The total corrosion time is insufficient for conclusive analysis; however, several significant observations can be ascertained from a study of the 28-day rates as shown in Table I.

Table I

The Effect of Homogenizing Temperature on the Corrosion
Properties of Unclad U-12 w/o Mo Alloys

Treatment	-----28 da Corrosion Rate (mg/cm ² /hr)-----	
	Extrusion 1101-1-13*	Extrusion J959**
As-extruded	Failed 14-21 da	-0.57
600°C/72 hr/W.Q.	Failed 7-14 da	Failed 21-28 da
700°C/72 hr/W.Q.	Failed 14-21 da	Failed 21-28 da
800°C/72 hr/W.Q.	-0.28	-0.25
900°C/72 hr/W.Q.	-0.13	-0.35+
1000°C/72 hr/W.Q.	-0.07	-0.15

* Duplex melted and extrusion clad with Zircaloy-2

** Duplex melted and extruded to fuel billet

+ Sample cracked into 2 pieces between 3 and 7 days

In agreement with past experience, extrusion No. J959 with a reduction of approximately 4 to 1 has much better corrosion resistance as-extruded than extrusion No. 1101-1-13 which had been reduced approximately 25 to 1. This phenomenon occurs because the greater amount of reduction produces a structure of continuous, elongated coring with low corrosion resistance. Previous experience with extrusions having a 4 to 1 reduction had shown high temperature annealing to have essentially no effect upon corrosion properties. In the case of J959, however, annealing at a temperature of 800°C and higher has appreciably improved corrosion resistance. This result is not considered abnormal since extruded billets with different melting and fabricating histories can be expected to have variations in the degree of segregation even at a low reduction of 4 to 1.

Although annealing at 800 to 1000°C has appreciably improved the 28-day rates, annealing for 72 hours at 600 and 700°C has decreased the corrosion resistance of these materials. Metallographic examination of other samples heat treated at the same temperatures show that annealing at 600 and 700°C results in widening of the cored stringers observed in as-extruded samples without effectively eliminating the stringers. On subsequent corrosion testing more general corrosion attack is observed in the widened stringers. On the other hand, only the narrow cored stringers of as-extruded material are attacked during corrosion testing and these stringers are separated by areas of relatively good corrosion resistance. It is felt, therefore, that annealing

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of as-extruded material at 600 and 700°C is to be avoided since partial homogenization may be worse than no homogenization at all. It should be noted from Table I that the minimum corrosion rates were obtained by annealing at the higher temperature.

Data were obtained on the annealing of two additional U-12 w/o Mo extrusions (Mo2cl-16 and 1101-1-13) for various times. Again it was noted that higher annealing temperatures result in lower rates and longer lives. Also, it was noted that increasing the annealing time, both at 900°C and 1000°C, decreases the corrosion rate but has little effect upon the corrosion life of these alloys. The nominal molybdenum analysis of both extrusions was 12 w/o Mo but the actual analysis for extrusion 1101-1-13 is 11.2 w/o Mo. Samples from this latter extrusion have lower corrosion rates and longer lives than samples from extrusion Mo2cl-16. It is not believed that its lower molybdenum content is responsible for better corrosion properties. The microstructure of this extrusion is dirtier than normal, indicating the possibility that impurities may be inhibiting hydrogen absorption. This will be checked.

The information obtained on effects of homogenization while being carried out on U-12 w/o Mo alloy, are generally applicable to U-base alloys. It is considered desirable, therefore, to complete the relatively small amount of work remaining before preparing a summary report.

Low Temperature Annealing of U-12 w/o Mo Alloys

The study of the mechanism by which the corrosion resistance of homogeneous U-12 Mo alloys is improved after annealing at 400°C is being continued. Corrosion results obtained since the past report are shown in Table II.

TABLE IICorrosion of Annealed U-12 Mo in 650°F Water

Condition	Corrosion Rate 35 da (mg/cm ² /hr)	Corrosion Life (Days)
As-extruded	-1.4	35-42
900°C/24 hr/W.Q. (γQ)	-1.0	35-42
900°C/24 hr/W.Q. + 343°C/5 wk/W.Q.	-0.93	35-49
900°C/24 hr/W.Q. + 400°C/5 wk/W.Q.	-0.11	>56
400°C/5 wk/W.Q.	-0.85	35-42

These results are in agreement with the discussion of this subject in WAPD-MRP-53. Although samples annealed at 400°C without a previous gamma quench were seen to corrode more rapidly than the as-extruded samples in the first 21 days, the corrosion life was the same for both conditions. Only the samples annealed at 400°C following the gamma quench showed an appreciable improvement in corrosion rate and life. These samples are still in test after 56 days.

Corrosion samples have been removed periodically from test and analyzed for hydrogen. It was found that 24% of the hydrogen evolved during corrosion

was absorbed by the samples which were gamma quenched and annealed at 400°C. The samples from all other conditions absorbed only 14% of the hydrogen evolved. However, the corrosion rate of the samples gamma quenched and annealed at 400°C is an order of magnitude lower than the rates for the other conditions. The total hydrogen absorbed by the 400°C annealed samples per unit corrosion time, therefore, is much less.

This experiment has indicated that a low temperature anneal of gamma phase U-12 w/o Mo results in appreciable improvement in corrosion properties by affecting the amount of hydrogen absorbed and its distribution. The factors responsible for this type of behavior, whether due to precipitation of impurities or coherent precipitation of a transformation phase, will be considered in the new work program associated with gamma phase alloys which have more attractive nuclear properties. A summary report on the findings for the U-12 w/o Mo alloy will be issued shortly.

Corrosion Life of Extruded Rod Elements

Corrosion testing of material made by extruding directly to the 0.30 in. diam. size is being continued. From this testing information results are being obtained on the effect of low temperature heat treatment on corrosion life. The data indicate that 400°C annealing causes a marked decrease in corrosion life of the extruded material. Since 400°C annealing of bare samples has been shown to be beneficial, there is the possibility that transformation is occurring in core-clad interdiffusion layers which may be depleted in Mo content due to precipitation of MoZr_2 which has been observed in the past. Some of these samples are also being examined metallographically in an effort to obtain an explanation for the apparent discrepancy in corrosion behavior. These results will be included in a summary report.

Corrosion of Clad Fuel Elements

A series of binary and ternary clad fuel elements, with and without barrier layers, are being tested to study the corrosion resistance, susceptibility to heat treatment, and thermal stability of the alloys. The data indicate that the 10 Nb + 1 Pt and the 10.5 Mo with a Nb barrier, which showed good behavior up to the time of the last report, are now failing in thermal stability and in heat treatment after cold drawing. The 12 Nb and the 12 Mo + 0.5 Pt alloys are exhibiting the least variability in corrosion properties when heat treated after cold drawing. The 12 Mo + 0.5 Pt has shown good corrosion resistance in all conditions except the cold drawn, and even these samples have responded to heat treatment. The Nb alloys have shown some measure of thermal stability, though this is contrary to results reported for bare Nb fuel. Of the alloys with barrier layers, the Nb barrier shows the most promise. Fuel element corrosion testing is being continued since the function of barrier layers in promoting corrosion life is of special importance in the future work program.

Samples of U-12 Mo and U-10 Nb fuel elements with a bonded molybdenum layer are being prepared by induction melting in a 1.2 in. internal diameter molybdenum crucible. These will be subsequently extruded, defect and corrosion tested. This experiment is of primary interest in determining the

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role of cladding on the corrosion mechanism. It is desired to study the effect of a non-hydrogen getter such as molybdenum on corrosion behavior of gamma phase alloys. Results will be compared with those obtained when Zircaloy-2, a hydrogen getter, is used as the cladding on the same alloys.

The samples of extruded U-12 Mo clad fuel elements being tested to study the effect of defect size on the corrosion life have 56 days of test time. The defect diameters are 0.003 in., 0.013 in., 0.020 in., 0.040 in., and 0.080 in. No failures have occurred. The test conditions are 343°C and degassed water. This experiment is being continued since results will be applicable to all clad fuel elements having a gamma phase core.

Tests are being run to study the effect of bending stress on the corrosion life of U-12 Mo. Two stressed (95,000 psi) samples of clad U-12 Mo defected fuel elements and two clad control samples (unstressed) have completed 56 days of test time. The test conditions are degassed water at 343°C. No adverse corrosion effects have been observed to date and testing will be continued until signs of failure are observed.

A bare sample, also stressed in bending, was examined after 5 days of corrosion testing. It showed signs of relaxation. On restressing the sample it broke with a brittle fracture at the point of maximum stress. Metallographic examination indicated a preponderance of fine, randomly distributed hydride needles on the compression side and fewer coarser needlless on the tension side. Thus, the state of stress has an appreciable effect on the amount and form of hydride produced. To check the amount of relaxation to be expected as a result of heating alone at 650°F, a sample was prestressed in a bending jig to 95,000 psi and placed in an inert atmosphere furnace. A permanent set of 20% was recorded after 5 hours at temperature. The set increased to 24% after a total of 12 hours at temperature. This test is being continued.

The study being conducted to determine the feasibility of recrystallization and stress-relief heat treatment for cold-drawn U-12 Mo is being continued. Bare and clad defected test samples were made from two induction cast billets homogenized before extrusion for 48 hours at 900°C. Early failure of bare samples annealed at 600 and 700°C was mainly due to cracking, and the cause was traced to gross porosity in the billets. Samples annealed at 900°C for short times show that at least some of the adverse effects of cold drawing can be removed by annealing. Because of the presence of porosity the samples were removed from test. A new study will be made using duplex-melted, pre-extruded, and homogenized alloys which will eliminate the porosity encountered in induction cast alloys. The clad defected fuel elements indicate early failures which may again be caused by the presence of porosity. However, some samples have been in test for over 28 days without failing and are being continued in test. Since drawing is contemplated as a practical method of fabricating metallic fuel elements, the effects of drawing and subsequent heat treatment on corrosion behavior of fuel elements continues to be an important phase of future work.

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Homogenization of U-12 w/o Mo Extrusion Billets

The investigation of the effect of homogenization of U-12 Mo extrusion billets on the corrosion characteristics of the extruded, stripped rods is nearly complete. The latest results indicate that unclad samples annealed at 400°C without a prior homogenization at 900°C fail in 7 to 14 days. Homogenization prior to extrusion results in effectively increasing thermal stability as shown by the longer corrosion life, 14 to 28 days or 28 to 42 days. Since there are no failures in 42 days testing of samples homogenized at 900°C and annealed 400°C after extrusion, it is apparent that more complete homogenization was attained by the gamma quench subsequent to extrusion than was accomplished by the homogenization prior to extrusion. Although annealing at 600°C for 1 hr has reduced the corrosion life of the extruded and extruded drawn rods, the data concur with the other evidence as to the effectiveness of pre-homogenization.

Corrosion tests are being conducted on simulated fuel elements with 0.040 in. diam defect holes through the cladding. From the results to date pre-homogenization increases the thermal stability of the extruded rods when annealed at 500°C for 1 week. As reported in WAPD-MRP-53 these data are in agreement with the results of the testing of bare samples annealed at 500°C. When all samples have been tested to failure, a report covering this study will be issued.

As a follow-up on the above program, 8 duplex melted castings that were originally prepared to study the impurity level obtained by various melting techniques were homogenized at 900 and 1000°C prior to extrusion at 1600 and 1900°F. Bare samples were then tested in 650°F water as-extruded, γ Q, and annealed at 400°C for 5 weeks and 500°C for 1 week both with and without a γ Q after the extrusion operation. The wide range of impurity level makes precise interpretation of the corrosion data difficult. However, the data, with the exception of pre-homogenized 1900°F extrusions, essentially confirm the previous information derived from the above program, that is, homogenizing prior to extrusion improves the corrosion resistance and thermal stability of as-extruded material. The samples will be tested to failure along with the samples annealed at 400°C for five weeks.

An additional investigation is now getting under way to study in greater detail the corrosion properties of bare U-12 w/o Mo samples as affected by homogenization prior to extrusion, extrusion temperature, and post-extrusion heat treatment. While this program was originally designed to investigate factors contributing to observed variability in U-12 w/o Mo alloy, it is being continued since the data will be applicable generally to gamma phase alloys.

Variability in U-12 MoEffect of Melting Practice

In order to clarify existing discrepancies concerning the relative merits of as-cast induction melted and duplex melted materials, eight alloys were tested in the as-induction-cast and as-duplex-cast conditions. In addition, duplex-cast specimens were gamma quenched.

It was observed that no consistent differences exist between the cast materials either after induction melting or after remelting by the consumable arc process. On the whole, cast samples that were duplex melted exhibit the longer lives, but in general this is due to the fact that 10 samples were tested as duplex melted whereas only 1 was tested as induction melted. Also, due to the portion of the castings that were sampled, the specimen integrity of the duplex melted material was higher. This phase of the study is now complete and will be incorporated in another report.

Effect of Extruding and Drawing

Bare samples of duplex melted and co-extruded U-10.5 and 12 Mo alloys were examined as-extruded, extruded plus Q, as-cold-drawn and cold-drawn plus Q. The extrusions were intended to study barrier layer and defect size effects in clad samples. Bare samples were tested to ascertain the integrity of the core.

From the corrosion data to date on these samples, little variability is noted in the corrosion life of as-extruded material. Gamma quenching improves corrosion life considerably as expected, but also increases the spread in corrosion life. Some variation is also seen to exist in cold-drawn and cold-drawn and γ Q specimens. This would tend to indicate that differences between extrusions become more evident and possibly more marked when additional operations such as annealing and cold drawing are subsequently performed.

These samples will be tested to failure and incorporated in another report.

Effect of Fabrication Condition

The program to determine the effects of homogenization upon various properties of induction melted, chill cast alloys is essentially complete and a report incorporating the results of this work will be issued.

Hydrogen Overvoltage Studies

Hydrogen overvoltage measurements have been made on alpha uranium and several uranium alloys in order to determine the feasibility of using these measurements to predict the corrosion characteristics of uranium-base alloys.

Inasmuch as no obvious relationship has been found relating corrosion behavior and overvoltage, these studies have been discontinued and a report is currently being written.

Uranium-Molybdenum Phase Diagram Studies

Two series of samples which had been annealed at 560°C and 610°C have been examined by X-ray and metallographic methods. The results are in agreement with the revisions of the diagram as reported last month. The work on the U-Mo phase diagram determination is now completed and has shown that the gamma phase decomposes eutectoidally to alpha uranium and U₂Mo. A report on the phase diagram is being prepared.

Uranium-Molybdenum-Niobium Phase Diagram Studies

All series of samples which were being annealed have been removed from the furnaces. These samples are now being examined by X-ray and metallographic methods. The work on the U corner of the U-Mo-Nb phase diagram determination is expected to be completed during the next two report periods.

Magnetic Susceptibility Measurements

In accordance with the new alloy development program, equipment is being constructed to measure the magnetic susceptibility of corrosion-tested U-Mo alloys to detect the initial formation of UH₃. The Curie method of measurement will be used. This requires that the product of the magnetic field intensity and the gradient of the field intensity be constant in the region in which the sample is suspended. An attempt has been made to determine the best magnetic pole piece shape to give the desired field configuration over as large a volume as possible. Calculations indicate that a shape of the form

$$y = a(x - x_0)^{-1/2}$$

gives a field such that $H \frac{dH}{dx}$ is relatively constant. During the next report period it is planned to machine the pole pieces and measure the field configuration experimentally.

X-ray Double Crystal Spectrometer for Irradiated Material

Considerable work is planned to investigate the effects of irradiation on the crystal structure of uranium-base alloys. For this purpose an X-ray double crystal spectrometer is being built.

During this report period the X-ray tube holder and the crystal holder have been received. The three remaining major components have been submitted for bids.

Thermal Stability of U-Nb-Zr Gamma Phase Alloys

Past work on U-10 w/o Nb clad with Zircaloy indicated that a U-Nb-Zr diffusion layer, formed during heat treatment, exhibits remarkably good thermal stability and corrosion resistance. An investigation has therefore been initiated to obtain more complete information on the thermal stability of the gamma phase in the U-Nb-Zr system. The primary purpose of this work will be to

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determine the optimum alloy composition corresponding to maximum thermal stability of the gamma phase. In addition, the diffusion coefficients of the components in the alloy will also be determined.

It is planned to make sandwich type diffusion couples in which U-Nb alloys of various compositions will be clad with pure zirconium. The specimens will be annealed at 900°C for 10 and 25 days to permit diffusion and to produce, thereby, a concentration gradient of Zr in the U-Nb alloy. After the diffusion anneal the specimens will be annealed for various periods of time at 500°C to promote decomposition of the gamma phase. The specimens will then be metallographically examined on a plane perpendicular to the diffusion direction to locate the position of the untransformed gamma phase. They will then be machined down to this point and a chemical analysis of the material obtained. Alloy composition as a function of distance along the diffusion direction will also be obtained to determine the diffusion coefficients. The material for this work has been ordered.

Effect of Stress on Transformation

The importance of stress in causing accelerated corrosion and changing the amount and distribution of the hydride phase has been demonstrated for both U-Mo and U-Nb alloys. To obtain a clearer understanding of the role which stress plays, a program will be initiated to study the effect of stress on the decomposition of the gamma phase; it is planned to transform gamma quenched uranium alloys in a creep machine under tensile loading. One of the alloys which will be investigated is U-7 w/o Mo. This material is now being prepared. U-Nb-Zr alloys will also be studied; the particular compositions will be dictated by the results of thermal stability studies mentioned above. Initial tests will be run at 475 and 400°C. The time for the start of transformation will be determined for different stress values. In addition, the decomposition products will be examined for the presence of a metastable phase for which some evidence already exists.

Effect of Multiple Alloy Additions on Thermal Stability of Gamma Uranium

A problem of considerable interest and potential value is the comparison of the effect of multiple additions as compared to a larger amount of single addition in thermal stability of the gamma phase. Such effects are pronounced in ferrous alloys and may lead to stabilization of the gamma phase of uranium with acceptably low alloy content. Therefore, a study is being initiated to determine the effects of Mo, Nb, Zr, Ti, Cr, V, etc., in multiple combinations for comparison with equal atomic percentage additions of single elements. If possible, it is planned to use extremely pure materials for this study. Accordingly, the possibility of refining the uranium, which is considered the principal source of impurities, is now being investigated. Alloy additions will be made to the refined uranium if the refining is successful. If not, presently available uranium will be used. The range of effectiveness of the additions will be determined in the same manner as in the thermal stability experiments on the U-Nb-Zr alloys.

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1. Non-Core Materials (53all) J. R. HunterSubstitute for Microbrazed

Data on brazing clearances of furnace brazed typed 18-8 specimens mentioned in the last report (WAPD-MRP-53) are given below:

FLOW IN 1 IN. LONG BRAZING FLOW SPECIMENS
CONTAINING 1/16 IN. DIAM 10 MIL WALL TUBING

Material	Annular Clearance *				Brazing	
	2 mil	5 mil	8 mil	Useable Clearance	Temp.	Time
24-K Au	100%	100%		2 to 5 mil	1070°C	15 min
80% Au 20% Cu		100%	100%	5 to 8 mil	965°C	15 min
50% Au 50% Cu	100%	100%		2 to 5 mil	1000°C	15 min
Rexweld 64		100%	100%	5 to 8 mil	1040°C	15 min
Microbrazed	100%	100%	100%	2 to 8 mil	1040°C	15 min
Coast Metal #53	100%	100%		2 to 5 mil	980°C	15 min
Coast Metal #52	100%	100%		2 to 5 mil	980°C	15 min

*Clearance on the diameter

These temperatures and clearances provided 100% penetration of the 1 in. diam, 1 in. long brazing flow specimens with a minimum of alloying of the 10 mil wall tubing. This represents satisfactory brazing at the clearances listed in the useable clearance column. If corrosion tests are satisfactory, these materials will warrant serious consideration as substitutes for microbrazed.

Brazing temperatures of the order of 50°C to 100°C, higher than those listed above, are recommended for heavier sections which are seriously affected by alloying of brazing material with the base material.

Corrosion testing of additional brazing materials will be initiated when materials presently on order are received. Present corrosion tests of the auric brazing materials are continuing in order to complete a 200 day test exposure to substantiate favorable short time test results.

High Strength Steel for Reactor Vessel

The creep testing rig design, referred to in WAPD-MRP-53, was submitted to the Brookhaven National Laboratory and to the Knolls Atomic Power Laboratory for comment. It is anticipated that comments will be made at a meeting held at the Brookhaven National Laboratory (BNL) on June 17 at which representatives from the Bettis Plant, Knolls Atomic Power Laboratory, and BNL will be present.

All T-1 and A-302 base metal plate for this program was rolled and will be ultrasonically tested, cut, and shipped to various laboratories by July 1. Testing is anticipated to start about July 15.

In accordance with instructions, orders were given to the vendor to develop as much clad plate as possible from material in work. It is anticipated that the clad T-1 and A-302 plate will be available about August 1.

Stress Corrosion of Alloys

The review of literature on stress corrosion of alloys is being continued on a low level of effort. The emphasis in this study is as follows: (a) collating of existing stress corrosion data for metals of current interest, and (b) developing a test procedure which will evaluate corrosion resistance of metals under varying stresses and at different levels of cold work.

Data were reported in WAPD-MRD-48 on the stress corrosion of copper-nickel alloy after 84 days of exposure. No further work has been done.

Spring Materials Development

Spring materials are being evaluated for relaxation (loss of ability to support a load) after exposure to a corrosive environment (oxygenated water - 600°F) for a period of time. It is estimated that relaxation values $\pm 0.5\%$ fall within the experimental error and are not significant. This is a continuing project and data are made available to design engineers as it is obtained.

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The data on a single heat of Haynes 25 Stellite indicate superiority from a corrosion standpoint to currently used Inconel X and warrant serious consideration as a replacement material if this information can be corroborated on other heats. Presently available data are tabulated below and should be considered as interim values only:

Material (Aged)	Load psi X10 ³	% Cold Reduction	% Relaxation	Weeks in Test	Notes
Haynes 25 Stellite	50	15	0.42	19	(No change in load after 8 wks.)
Haynes 25 Stellite	100	44	2.25	19	
Haynes 25 Stellite	120	62	2.25	19	(No change in load after 8 wks.)
Alleg.-Ludl. S-816	70	AR	7.0	20	Extreme scatter of data
Alleg.-Ludl. S-816	80	AR	7.8	20	Extreme scatter of data
Alleg.-Ludl. S-816	90	AR	8.5	20	Extreme scatter of data
Refractaloy-26	70	AR	2.8	20	Extreme scatter of data
Refractaloy-26	80	AR	2.8	20	Extreme scatter of data
Refractaloy-26	90	AR	2.8	20	No change after 4 weeks
Inconel X	70	15	Less than 0.5%	6	Small pitting on flat ground ends. Black surface.
Inconel X	70	40	Less than 0.5%	6	Small pitting on flat ground ends. Black surface.
Inconel X	70	60	Less than 0.5%	6	Small pitting on flat ground ends. Black surface.

There is on order material to perform corroborative tests where preliminary data indicate a promising material. It is anticipated that this material will go into test during July. Additional screening of promising materials is going forward on Cobenium, Allegheny-Ludlum CM spring wire, and Allegheny-Ludlum high speed steel wire.

During the coming period, this program will be actively pushed to: (a) accumulate additional time on specimens in test, (b) accumulate information on new materials, (c) corroborate existing data on other heats of promising materials, and (d) accumulate data at higher stress levels. Anticipated problems involve availability of test facilities, specifically, autoclaves.

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Statistical analysis of the data on Haynes 25 and Allegheny Ludlum S-816 leads to the following conclusions. After a total of 19 weeks of test, the Haynes 25 springs tested at 50,000 psi corrected torsional stress showed no significant relaxation. The range of relaxation obtained was well within the range attributed to error in measurements. The average relaxation obtained was approximately 0.42%. The Haynes 25 springs being tested at 100,000 psi and 120,000 psi corrected torsional stress reached a constant value of approximately 2.25% relaxation after 8 weeks test.

The Allegheny Ludlum S-816 springs being tested at 70,000 psi, 80,000 psi, and 90,000 psi torsional stress show an increase in relaxation with respect to a continual increase in time. Variations in relaxation between replicates were as large as 4%. The approximate relaxations of each stress level at the end of 20 total weeks test are 7%, 7.8% and 8.5% for the 70,000 psi, 80,000 psi, and 90,000 psi stress levels, respectively.

Sufficient data for statistical analysis have not yet been obtained on the Inconel X springs presently in test.

The more complete investigation of Inconel X and Haynes 25 is being planned. New materials to be investigated are Cobenium, (Elgin National Watch Company), Inconel X (International Nickel Company), and Allegheny Ludlum C-M and high speed steel.

2. Carbon Steel Program (55b14) F. Cohen

The work on this project for this period is covered in WAPD-MRL-6.

D. Chemistry Development1. Fuel Element Testing and Fission Product Removal (55bl2) P. CohenEffect of Coolant Chemistry on Fuel Stability

a. U-Mo Alloy

Work on U-Mo Alloy has been suspended.

b. UO_2

The status of work on the effect of coolant composition on UO_2 fuel is recorded under I-A-2. this report.

Fission Product Removal Studies

a. Out-of-pile tests

(1) U-12 w/o Mo

An estimate was obtained of the settling rates of added solids in loop R-2, Phase IV-D (WAPD-MRP-52), and the results obtained with the synthetic crud solids were extrapolated to the case of UO_2 of the same particle size assuming Stoke's law. The following range in data was obtained for a system with equal volume parallel circuits with average velocities of 6.21 and 1.4 ft/sec respectively.

$-\frac{dc}{dt}$ -c	% Removal of Crud/Effective Pass	Calculated % Removal of UO_2 /Effective Pass
0.016/min.	0.20	0.48
0.005/min.	0.06	0.15
0.019/min.	0.25	0.58
0.0062/min.	0.08	0.18
	Avg. = 0.15	Avg. = 0.35

where c = concentration; t = time

Phase Va testing in loop R-1 (600°F, 2000 psi, 100 cc H_2 /kg, neutral pH) was terminated after approximately 1200 hours (WAPD-MRP-52). Channel specimens were removed and examined with respect to contamination as a function of surface finish. An additional test was initiated and completed at essentially the same conditions in Phase R-1-Vb with various surface finishes (600 hrs, 25 cc H_2 /kg). Subsequent to R-1-Vb testing, additional channels (ranging from 3 to 500 rms) were inserted for R-1-Vc testing, now in progress. Preliminary results obtained from R-1-Va and

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Vb indicate no significant differences in activity contamination among channel specimens with finishes varying from 2 to 125 rms. Comparatively high levels of contamination found with sand blasted (300-350 rms) surface finishes may be attributed to the tremendous surface area presented by said finishes and, as such, represent an extreme case. The table below summarizes preliminary comparisons. "A" represents the mean activity of the channels corresponding to a particular finish while "Ae" corresponds to the mean activity of electro-polished samples present in the same environment.

(Profilometer reading)		<u>Pretreatment</u>	<u>A/Ae</u>
<u>Test</u>	<u>Specimen Surface Finish, rms</u>		
R1-Va	125-140, #304 stainless	None	1.5
R1-Va	65-85, #304 stainless	None	1.6
R1-Va	50-60, #304 stainless	None	1.7
R1-Vb	2, #304 stainless	None	2.8
R1-Vb	side (1) 50-70, #304 stainless	Preoxidized in neutral water	1.8
	side (2) 125, #304 stainless		
R1-Vb	side (1) 50-70, #304 stainless	Preoxidized in LiOH	1.8
	side (2) 125, #304 stainless		
R1-Vb	100, #347 stainless	Preoxidized in neutral water	1.9
R1-Vb	100, #347 stainless	Preoxidized in LiOH	2.5
R1-Vb	300-325, #304 stainless	Sand blasted #40 grit	15.5

Channel specimens from R-1-Va were brushed with Alconox and the percent remaining activity extrapolated to the same initial activity. The results listed below do not indicate any preferential removal as a function of surface finish.

<u>Surface Finish</u>	<u>Percent of Initial Activity Remaining (extrapolated to same initial activity) (Brushed with Alconox)</u>
Electropolished	15.2 (average of 5 channels ranging from 11.2-19.3)
50-60	20.1 (average of 2 channels ranging from 19.2-21.4)
65-85	18.4 (average of 4 channels ranging from 16.2-20.8)
125-140	21.2 (average of 5 channels ranging from 18.9-25.8)

Electrostripping of contaminated channel specimens from the R-2-IV and R-1-Va is in progress.

Five heat treated U-12 w/o Mo specimens to be utilized in conjunction with full stream filter studies in R-3 have been shipped to the MTR for irradiation. Manufacture of the bed filter has commenced but the filter will not be available until July 1, 1955. A limited number of tests will be run with the heat treated specimens to demonstrate the effect of failing corrodible fuel specimens on loop contamination and to determine the distribution of the released activity between filterable and non-filterable phases.

(2) UO_2

The defected clad UO_2 pellet tested in R-3, Phase VI (WAPD-MRP-52) has been dissolved and its specific activity determined to be approximately 0.2 curie.

The R-3 test section as well as some piping removed from the system was pickled with HNO_3 -HF. From the particular piping examined, it is estimated that approximately one mg of UO_2 was lost from the defected pellet assuming that all released activity was associated with UO_2 and not leached from the element. This release corresponds to 0.03% of the total activity in the element containing approximately 3.25 gm of UO_2 . A factor of two might be assumed for the rest of the loop since approximately 1/2 of the loop surface exposed to activity was pickled.

At the termination of R-3-VI, the defected pellet test section was removed and the one loop modified for insertion of a defected (#60 drill hole) cold pressed and sintered UO_2 wafer type element (~ 1 -2 curies). Subsequent operation of R-3-VII has indicated little release of activity to date (~ 500 hr) as indicated by water counts of 10-100 γ cpm/ml.

Loop R-2 has been pickled with HNO_3 -HF in order to reduce activity levels to a background commensurate with further UO_2 testing. An extreme case will be tested in the R-2 facility with a UO_2 cold pressed element containing seven #60 drill holes in the wafer type element. This test (R-2-VIII) was begun on June 13.

(3) C-Loop Tests with UO_2 Suspensions

(a) Construction and Modification

C-1 loop has been modified to permit quantitative solids injection at high velocity via the make-up pump discharge. At the same time, a series of special hot filters were built for installation in the pressure relief line between the main loop and the diaphragm relief valve. The filtration rate can be preset within $\pm 10\%$ by adjusting the make-up pump stroke, and then the flow can be checked by metering the discharge directly on the low pressure side of the relief valve. The bomb injection system was shown to give quantitative transfer at a flow rate of 3.5 fps (110 cc/sec-1/16 in. ID), while quantitative transfer from the loop to the micrometallic filter disc was also verified. It was found that quantitative transfer of UO_2 (0.5-20 μ) was not obtained below 1.5 fps in a complex network of 1/16 in. ID tubing. It was also found that the stroke could be lowered to the point where pressurization was still maintained, but where negligible solids transport to the filter took place. This feature makes it possible to conduct deposition runs over long periods of time without the necessity of correcting the observed "throwout" for removal by the filtration-relief stream.

In order to prevent possible release of airborne activity to the loop room, a tight fitting sheet metal enclosure was built around C-1 loop. Air circulation at ~60 cfm is provided by means of a small blower exhausting through a CWS air filter. Individual panels are easily removable for access to filters and valves.

To date, three Allis-Chalmers Model NP43100 centrifugal pumps have failed in service. APD No. 61 (C-1 loop) failed due to rotor can expansion after approximately 2 weeks at 600°F, 2000 psi. During this time not more than 25 or 30 starts were made. APD No. 59 (C-2 loop) failed after 5 days at 600°F, 2000 psi. In neither case were solids added to the system during operation. APD No. 56 (C-1 loop) failed after 9 days operation at 500°F, 2100 psia. The solids concentration during this period reached 220 ppm on four occasions, although only for several minutes at a time. The time average solids concentration remained below 10 ppm. All three pumps showed extensive rear bearing wear. The vendor is currently repairing the above pumps while attempting to determine the cause of failure. In the meantime, the impeller of a Westinghouse 5A pump is being modified to provide the necessary high-head characteristic required in C-1 loop. If the resultant characteristic meets Atomic Equipment Department design figures, one can expect 30 percent increase in loop velocity when this pump is installed as a substitute in C-1 loop.

(b) Operation

The rate of solids deposition in this high-velocity circulating system has been studied using sludge from the autoclave corrosion of non-irradiated U-Mo alloy slugs. By obtaining data with a simple loop which was relatively free of mechanical constrictions and natural traps, it was hoped that a lower limit to the expected throwout rate for a larger, more complex loop could be established.

After injecting sludge, the loop was operated for a period of either 30 or 60 min at a negligible filtration rate. The filtration rate was then increased to 110 cc/min in order to quantitatively sample the loop contents. The instantaneous deposition rates for ~220 ppm initial solids concentration are given below:

<u>Time After Injection</u>	<u>% Deposited per pass</u>	<u>1st Order Rate Const. min⁻¹</u>
2 min (initial)	0.86	0.50
38 min	0.13	0.077
68 min	0.036	0.021

It can be seen that the loop water solids concentration does not follow a simple exponential decay after injection; this effect might be expected in view of the large variation in settling rates between the various particle sizes present. Following each deposition run, filtration was continued for periods of from 8 to 17 hours. At the end of each so-called "cleaning run", the loop still contained ~57% of the original solids injected

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for that particular run. Although the solids build-up was cumulative, the amount of solids already present in the system before a given run apparently did not affect the deposition rate in subsequent runs. Following the series of four deposition experiments, a long cleanup sequence was made. This consisted of three start-ups, three cooling periods interspersed between 48 hr continuous filtration at 6 system volumes per hour flow. During this period, only 35% of the remaining solids inventory was removed. It can be concluded that on the average (0.65×0.57) or 37% of the solids injected per run remained "irreversibly" trapped within the loop and could not be moved by extended circulation.

Additional solids deposition experiments will be begun in C-1 loop using sludge from the autoclave corrosion of irradiated U-Mo fuel slugs. The activity distribution on test sections of different surface finish and at various points around the loop will be obtained as a function of operating time. Autoradiographs of pipe test sections will also be made.

(c) Pile Wall Survey Instrumentation

In order to permit detailed "in place" surveying of contaminated pipe, particularly in the C and R loops, an NRD SC-45 collimated scintillation probe has been installed on a movable boom attached to a dolly. A rack containing the following components is also mounted on the dolly in order to make the equipment a single working unit:

1. Atomic Model 312 Power Supply
2. Atomic Model 204C Linear Amplifier
3. Atomic Model 570 Pulse Height Analyzer
4. Atomic Model 105 Scaler

This equipment has been used for a point-to-point survey of a contaminated loop (R-1) with satisfactory results. Normalizing to unit specific activity on a horizontal run of 1 in. pipe, the following activities are observed at various locations:

<u>Location</u>	<u>Activity</u>
Horizontal Run	1.0
Vertical Run	0.5
Orifice plate, vertical	1.7
Vertical bend, 90°	3.0
Contraction	8.7
Flange	2.4

Two sections of 1 in. Sch. 80 pipe cut from the upper and lower headers of the Chalk River X-1 loop following the first UO_2 defect test were also studied. Again normalizing to unit activity for the horizontal run of pipe, the activity on a bend where welding back-up rings had been used was found to be 8.1. Identification of Zr^{95} (730 kev) and Ce^{141} (146 kev)

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peaks were made by obtaining the γ -spectrum on both pieces of pipe. The spectra for both pipes were identical, indicating that the method of deposition of activity was probably similar. Further quantitative information will be obtained by decontamination of sections of the piping and radiochemical analysis. Autoradiographs of the horizontal pipe section reveal the typical "hot spots" associated with the deposition of particulate activity.

Contamination-Decontamination Studies

Autoclave coupons (304 stainless, area = 0.306 dm², 20-40 rms) have been precorroded in 3 additional environments and are at present being contaminated with a particle-free mixed fission product solution, at 600°F and 2000 psi. The conditions under which they were precorroded, and weight changes, are presented below. Differences in activity absorption will be reported in the next progress report.

<u>Test Conditions</u>	<u>Accumulated Net Change - mg/dm²</u>		
	<u>100 hr</u>	<u>200 hr</u>	<u>300 hr</u>
all run at 600°F and 2000 psi			
Neutral water with 25 cc/kg H ₂	+5.6	+8.5	+15.7
Versene solution (pH 10.8)	+3.3	+4.0	-
Potassium cyanide solution (10 ppm)	+7.2	+7.8	-

previous test conditions

LiOH without H ₂ (initial pH = 10.5)	+1.8	-0.5	-0.9
LiOH with H ₂ (initial pH = 10.5)	+1.0	+0.5	-1.0

Five tests have been run in a study of activity adsorption on synthetic crud. Test conditions were: neutral, degassed water at 600°F, 2000 psi and a crud concentration of 200 ppm. Difficulties were encountered in establishing the proper sampling and measuring techniques in order to get consistent data. One difficulty which had to be compensated for was the adsorption of activity by the filtering medium which was used to separate the crud from the active solution.

Tests were run in a tipping autoclave which oscillated over a 120° arc. Samples were withdrawn at time intervals which extended from 2-360 min. When the data were plotted, it could be seen that crud-activity equilibrium was evidently reached prior to the first sampling period (2min) and the activity of the crud, filtrate and total activity samples remained essentially constant throughout the run. The approach to the equilibrium distribution of activity between crud and water is apparently very rapid. Below are the data of test No. V.

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Activity Absorption on Synthetic Crud
Test Run 5/27/55

Time (min)	Experimental Data			Calculated Data
	Gross Activity (Ai) (CPM/ml)	Filtrate (F) (CPM/ml)	Crud (C) (CPM/ml)	Crud (Ai-F)
0	4129	4072	163	57
2.8	4740			
4.0		4375	148	365
9.5	4944			
10.7		4745	701	199
20.5	5313			
21.6		4761	155	552
47.0	4844			
48.3		4434	184	410
151.7	5239			
152.5		4810	267	429
359.5	4839			
360.6			127	

One test has been run using Montmorillonite clay instead of synthetic crud. The clay concentration was 200 ppm and the activity removal was approximately the same. Higher concentrations of clay and other activated exchange media will be used in subsequent tests during the next period.

Autoradiographs were taken of autoclave coupons subjected to 100 hr of pre-corrosion followed by 100 hr of exposure to a particle-free mixed fission product water solution (filtered 4 times through Millipore filter paper). In comparison to samples which were contaminated in unfiltered water, activity distribution is uniform over the entire coupon surface and the level of contamination is considerably lower, thus demonstrating that the spotty nature of autoradiographs taken from samples exposed to unfiltered water was correctly interpreted as the result of particulate contamination rather than as the result of a non-uniform distribution of active centers for adsorption.

Adsorption of Specific Radioisotopes on Primary Coolant Container Materials

Adsorption-desorption studies using carrier-free, specific radioisotopes have been started to determine the principal contributors to long-term contamination. Determination of the adsorption isotherms is being done at 600°F and 1500 psig in degassed water using a large surface of 304 stainless steel as the substrate material. Although such variables as surface finish and method of surface pretreatment will be studied, the preliminary work is being done on un-oxidized 304 stainless steel with approximately a 15 micro-inch finish. As shown above, these results should be comparable with other surface finishes.

Currently, the adsorption isotherm for Sr⁹⁰ is being determined with concentration as the variable. At the present time, there are insufficient

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data to draw the isotherm or to report whether or not there is a hysteresis effect in the adsorption-desorption curves.

Progress on this problem will be slow until installation of additional autoclaves which are expected to be available by July 1.

Slug Rupture Studies

Work on this project has been suspended with the selection of UO_2 as the reference core material.

Fouling Effects of Crud Deposits

The final report ("Crudded Plate Heat Transfer Test", ES-1015, Report No. 5426) of results found by the Babcock & Wilcox Company has been received. Constants have been established to predict the heat transfer of two types of crud deposits, which seem to have the same physical and chemical properties except for their color.

$w/h = \frac{\text{mg/dm}^2}{\text{Btu/hr} \times \text{ft}^2 \times ^\circ\text{F}}$	<u>Crud Deposit</u>
$0.144 \times 10^{-5} \pm 0.014 \times 10^{-5}$	Black
$0.278 \times 10^{-5} \pm 0.016 \times 10^{-5}$	Brown

The range of values obtained will permit an estimate of the thermal effects of crud deposits on fuel elements. Since these data apply only to completely sub-cooled conditions, additional work must be done to allow for boiling in the deposit.

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2. PWR Coolant Technology Applications (55b10) P. Cohen

For evaluation of the valve operating system, the overall diffusion rate of oxygen through a static column of water was measured using the thermally insulated surge tank of the D-loop. With the water surface pressurized to 1900-2000 psi the oxygen concentration at the bottom of a 4-ft column increased from 0.7 to 12 ppm in 40 hours. Interpolation of these data to the conditions in the PWR Hydraulic Valve Operating System water flasks indicates that the limiting concentration of 10 ppm would be reached at the bottom of Flask No. 1 (4 ft³ air, 8 ft³ water) in a minimum of 40 hr and at the discharge end of the system in at least a week. This information was transmitted to Fluid Systems Section May 14.

Samples of deposit taken at the time of the inspection of the secondary sides of the port and starboard boilers, Bettis Test Facility, have been analyzed both wet chemically and spectrographically. The chromium and nickel contents were both extremely low, under 0.1%, indicating negligible attack of stainless steel components by secondary boiler water. The deposit consisted primarily of iron oxides which are normal for a boiler; however, the high quantity of ferric oxide indicates either that the boiler operated with low pH and oxygenated water or that rusting occurred during idle periods. Except for a lower total iron content and slightly higher chromium and nickel percentages, the deposit on the tubes was similar to that found loose on the shell surfaces.

Further work was done on the chemistry facilities for the PWR: primarily, the selection of equipment, and establishing the arrangement of the sample preparation room.

Demineralized, untreated water was specified as the reference water for the Fuel Handling Canal. A search has been started for an inhibitor compatible with primary coolant, so far without success.

Consideration was given to the use of aluminum as a material of construction for various PWR Power Station components. Under dry conditions, aluminum may be used alone or in conjunction with steel for air ducts, structural members or enclosures. When moisture is present, consideration must be given to the dissolved impurities and their effect on galvanic and crevice corrosion of cold worked metal. High purity water, demineralized or condensate, such as reactor make-up and canal water, can be safely stored or carried in aluminum tanks and piping. For other waters, individual consideration must be given; alkaline waters, such as contemplated for the emergency cooling tank, are corrosive to aluminum. For cooling towers, aluminum may be preferred to wood but the feasibility will depend on the system, the water, and associated equipment.

E. Irradiation Effects1. Irradiation Effects on Core Materials (53b31) R. H. FillnowUranium Alloy Program

Since UO_2 was selected as the fuel for the first PWR core, interest in U-12 w/o Mo and U-10 w/o Nb alloys has decreased. Irradiation effects on these alloys, therefore, are of interest only when results are generally applicable to a class of U-base alloys and lead to a clearer understanding of behavior of the alloys subjected to neutron bombardment. For this reason experiments which were initially designed to evaluate the effects of irradiation specifically on U-12 w/o Mo or U-10 w/o Nb are being rapidly discontinued or modified in design to obtain more basic information. This is indicated for the individual experiments discussed below.

Cold Water Irradiation of Rod Fuel Elements (CR-WAPD-Met 2A)

The Zircaloy-1 clad U-2 w/o Zr and unalloyed uranium rods have been irradiated to a maximum exposure of about 5500 MWD/T in a Chalk River X-Rod hole. X-ray inspection after this irradiation period revealed a possible rod failure. Therefore, it was requested that these rods be returned to the Hot Laboratory for visual examination and length measurements. While UO_2 was recently selected for the first PWR core, this experiment is of general interest and will be continued.

Irradiation of Clad U-Mo Alloys in X-Rod Loop (CR-WAPD-Met 4A)

The remaining 4 defected, unirradiated, natural and enriched control samples of U-10.5 and U-12 w/o Mo were continued in corrosion test. One of these, U-10.5 w/o Mo, failed in 117 to 124 days. This represents the only 1 of 8 samples, which were extruded at 1750°F, to last over 82 days. The other 3 control samples, all U-12 w/o Mo, were extruded at 1950°F and are continuing in test after 124 days. The higher extrusion temperature appears to have improved corrosion life.

The samples irradiated to an estimated maximum exposure of 1545 MWD/T have been defected and are being corrosion tested in 650°F water.

Irradiation of Clad U-Mo in Chalk River X-Rod Loop (CR-WAPD-Met 6)

The 13 specimens of Zircaloy-2 clad U-11 w/o Mo and U-12 w/o Mo, were inserted in the Chalk River X-3 loop on March 10. The calculated temperature at the point at which fuel is exposed to high temperature water for the one defected specimen is 688°F. After about 2½ months (about 2500 MWD/T) the radioactivity of water samples removed from the loop indicate that the corrosion rate of exposed fuel is extremely low. The water activity has been running continuously at about 2000 counts/ml. Irradiation will be continued for the intended 4-month period unless the facilities are required for UO_2 experiments. This test represents the longest in-pile irradiation of a defected U-12 w/o Mo fuel element and will be of general interest.

MTR Process Water Irradiation of Clad Uranium-Molybdenum Alloys
(MTR-WAPD 11-Met 3, MTR-WAPD 11-Met 4)

Corrosion testing of defected Zircaloy-2 clad specimens of U-9 w/o Mo, U-10.5 w/o Mo, U-12 w/o Mo, and U-13.5 w/o Mo alloys containing 5 w/o uranium-235 has continued. These specimens had been heat treated at 750°C for 0.5 hr prior to irradiation and irradiated in MTR process water to estimated burnups ranging from 6500 to 17,300 MWD/T. The additional corrosion data obtained on irradiated specimens since the issuance of WAPD-MRP-53 indicate that 6 out of 7 specimens for WAPD 11-Met 4 have failed in 0-7 days. The 7th specimen survived 7 days and is still in test. Five specimens for WAPD 11-Met 3 failed in 14-21 days, 2 in 21-28 days, 1 in 26-33 days and 2 are still in test after 28 days.

As previously mentioned in WAPD-MRP-53, two additional groups of control specimens were prepared to determine if the observed decrease in corrosion life was a result of irradiation damage or damage caused by core-clad interdiffusion resulting from heat treatment. Two groups of control specimens are being tested: one group of 8 specimens in the same autoclave as the irradiated specimens and a second group of 16 specimens in a separate autoclave. Half of each group are in the as-extruded condition and half as-heat-treated at 750°C for 0.5 hr. The control specimens being corrosion tested in the same autoclave as the irradiated specimens have undergone 28 days test with only one failure, the 11.9 w/o Mo specimen as-heat-treated failing in the 21-28 day test period. Of the specimens being tested in a separate autoclave, the two U-9 w/o Mo samples which were heat treated failed in the 14-28 day test period, the two U-13.5 w/o Mo samples which were heat treated failed in the 28-42 day test period, and one U-9 w/o Mo sample as-extruded failed in the 28-42 day test period. All others are continuing in test after 42 days of corrosion testing. While it may be too early to draw definite conclusions, the results so far indicate that irradiation or irradiation plus heat-treatment, rather than heat treatment alone, was responsible for the early failures. The corrosion testing will be continued to bring this experiment to a logical conclusion.

MTR Process Water Irradiation of Clad Uranium-Molybdenum Alloys
(MTR-WAPD 11-Met 7)

Twenty defected specimens of U-12 w/o Mo and 3 of U-10.5 w/o Mo, irradiated in MTR process water to maximum exposures of 1200 MWD/T, have been corrosion tested in 650°F static water. Also tested in the same autoclave were 8 control specimens of U-12 w/o Mo and 1 of U-10.5 w/o Mo. All specimens have failed in less than 22 days of test, with the exception of 4 irradiated U-12 w/o Mo specimens. The estimated exposures of the 4 remaining specimens encompass the entire range of exposures for this experiment. Failure of control specimens occurred

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as early as the 0-7 day test period. No correlation of corrosion life and exposure is evident. It is believed that the relatively poor corrosion behavior of this group of specimens is due to poor cladding, which was caused by the overheating of the extrusion billets in the salt bath prior to extrusion. For this reason no further work is contemplated on this experiment.

MTR Process Water Irradiation of Clad Uranium-Molybdenum Alloys
(MTR-WAPD 11-Met 8, MTR-WAPD 11-Met 9)

Samples of as-extruded 3.6 w/o enriched U-10.5 w/o Mo and U-12.0 w/o Mo, which have been irradiated in MTR process water to estimated maximum exposures of 3600 and 6500 MWD/T, have been defected and are currently being corrosion tested in 650°F static water. These samples are from the same extrusions as the samples that were inserted in the X-rod loop at Chalk River as experiment CR-WAPD-Met 4A. The samples from the loop test have been defected and are being corrosion tested in the same autoclave as the MTR irradiated samples along with seven unirradiated control specimens. No failures were noted after 7 days testing.

MTR Process Water Irradiation of Clad Uranium-Molybdenum Alloys
(MTR-WAPD 11-Met 10)

Six as-extruded specimens each of natural U-9.0 w/o Mo, U-10.5 w/o Mo, U-12.0 w/o Mo, and U-13.5 w/o Mo, clad with Zircaloy-2, have been irradiated in MTR process water to an estimated maximum exposure of 2800 MWD/T. The specimens showed no dimensional changes due to irradiation. Five specimens of each composition have been defected and are currently being corrosion tested to failure in 650°F static water. Due to the recent choice of UO₂ as the PWR Core I fuel material, the work on this program will be brought to a logical conclusion and the results reported after the corrosion tests are completed. No failures were recorded after 14 days testing.

MTR Process Water Irradiation of Clad U-Nb Alloys
(MTR-WAPD 11-Met 11)

Ten Zircaloy-2 clad natural U-10 w/o Nb samples, fabricated in various ways as described in WAPD-MRP-52, have been irradiated in MTR process water to an estimated maximum exposure of 2600 MWD/T and returned to Bettis Plant. No dimensional changes due to irradiation were observed. All specimens have been defected and will be corrosion tested in 650°F static water. Autoclave facilities are not now available for these specimens.

MTR Process Water Irradiation of Clad U-Mo Alloys (MTR-WAPD 11-Met 12)

Four specimens each of 5 w/o enriched U-9.0 w/o Mo and U-10.5 w/o Mo, which were heat treated at intermediate temperatures in order to transform the metastable gamma phase to alpha plus epsilon and irradiated to an estimated maximum exposure of 2000 MWD/T, have been measured for any dimensional changes. No gross changes in dimensions were observed. Several specimens will be sectioned and examined for changes in microstructure due to irradiation.

II-E-1 (Contd.)WAPD-MRP-54MTR Process Water Irradiation of Clad U-Mo Alloys (MTR-WAPD 11-Met 13,14,15)

A series of three experiments of Zircaloy-2 clad 5 w/o enriched U-9 w/o Mo, U-10.5 w/o Mo, U-12 w/o Mo, and U-13.5 w/o Mo samples in the as-extruded condition have been inserted into MTR process water. Irradiation is scheduled for 12 weeks, 24 weeks, and 51 weeks to estimated maximum exposures of 7,000, 14,000 and 24,000 MWD/T respectively. Due to the recent choice of UO_2 as the PWR Core I fuel material and the subsequent de-emphasis of U-Mo alloy development, the future work on these programs will be drastically reduced.

MTR Process Water Irradiation of Clad U-Mo Alloys (MTR-WAPD 11-Met 16)

Seven specimens of Zircaloy-2 clad 3.6 w/o enriched U-12 w/o Mo, in the as-extruded condition, were charged March 28 into MTR process water for nine weeks irradiation to an estimated maximum exposure of 5000 MWD/T. These specimens were fabricated from the same extrusions and in the same manner as the specimens for the CR-WAPD-Met 6 loop test. After irradiation, the specimens will be defected and corrosion tested to failure in 650°F static water for comparison with the results obtained from the loop test.

MTR Process Water Irradiation of Clad U-Nb Alloys (MTR-WAPD 11-Met 17)

Twelve specimens of Zircaloy-2 clad natural U-10 w/o Nb, homogenized and aged at temperatures that bracket the range of calculated central temperatures for PWR uranium alloy fuel elements, are being irradiated in MTR process water to an estimated maximum exposure of 2600 MWD/T. After irradiation, the specimens will be measured for dimensional instability, defected, and corrosion tested in 650°F static water to failure. The specimens will be discharged from the reactor on June 20, 1964.

MTR Process Water Irradiation of Clad U-Nb Alloys (MTR-WAPD 11-Met 18,19)

Specimens of Zircaloy-2 clad 5 w/o enriched U-10 w/o Nb, heat treated at various aging temperatures, were scheduled for insertion into the MTR May 30. However, due to the recent choice of UO_2 as the PWR Core I fuel material, no further work is planned on these specimens by the PWR Metallurgy Subdivision.

Irradiation of Clad U-12.0 w/o Mo in the HB-5 Corrosion Loop (MTR-HB-5-Met 1)

The Zircaloy-2 clad 3.6 w/o enriched U-12 w/o Mo sample, defected as described in WAPD-MRP-52, was irradiated 6 weeks in the HB-5 loop in MTR to an estimated exposure of 1200 MWD/T. Loop conditions during test were described in WAPD-MRP-53. The specimen has been returned to Bettis Plant and will be sectioned at the defect and the core examined for corrosion effects when Hot Lab metallographic equipment becomes available after completing higher priority experiments.

II-E-1 (Contd.)WAPD-MRP-54Irradiation of Clad U-10 w/o Nb in the HB-5 Corrosion Loop
(MTR-HB 5-Met 2)

A Zircaloy-2 clad 5 w/o enriched U-10 w/o Nb specimen, defected with a 0.040 in. hole through the cladding, ran for one three-week fuel cycle in the HB-5 loop in the MTR. The calculated temperature at the bottom of the defect was 705°F. Water counts for the first 12 weeks of test were 600 counts/min/ml or less. Data for the last week of operation have not been received to date. This specimen will be returned to Bettis Plant on June 14, sectioned at the defect, and examined metallographically for further evidence of corrosion.

Irradiation of Unclad U-Mo Alloys (MTR-WAPD 12-Met 1)

Seven unclad samples of U-10.5 w/o Mo and seven unclad samples of U-12 w/o Mo were irradiated to a maximum burnup of approximately 1700 MWD/T. Three samples of each composition were corrosion tested in 650°F (343°C) and the results are shown in Table I. The results of this test indicate that irradiation did not accelerate the time to failure for these unclad samples. However, the results of a previous test indicated that irradiation of cast buttons of U-12 Mo to a maximum burnup of about 2500 MWD/T did accelerate the time to failure. Thus, a definite conclusion as to the effect of irradiation up to 2500 MWD/T on corrosion life of bare samples of U-12 Mo cannot be made. The experimental results of this program are now almost complete and will be issued as part of a report on the effect of irradiation on U-Mo alloys.

TABLE I Time to Failure of Unclad U-10.5 w/o Mo
and U-12 w/o Mo Samples

<u>Sample No.</u>	<u>Composition</u>	<u>Exposure MWD/T</u>	<u>Time to Failure Days</u>
11	U-10.5 w/o Mo	1700	22-28
12	U-10.5 w/o Mo	1450	Survived 35 days
13	U-10.5 w/o Mo	750	22-28
24	U-12 w/o Mo	1580	Survived 35 days
26	U-12 w/o Mo	1450	22-28
27	U-12 w/o Mo	750	20-22
5	U-10.5 w/o Mo	Control	21-35
6	U-10.5 w/o Mo	Control	21-35
29	U-12 w/o Mo	Control	21-35
30	U-12 w/o Mo	Control	21-35

Irradiation of Unclad U-Mo Alloys to High Burnups
(MTR-WAPD 12-Met 2, MTR-WAPD 12-Met 3, MTR-WAPD 12-Met 6)

Three samples of gamma-quenched U-12 w/o Mo, three samples of gamma-quenched U-13.5 w/o Mo, and one sample of U-13.5 w/o Mo that was transformed to 60% gamma and 40% epsilon, as determined by X-ray measurements, were irradiated to a maximum burnup of 7350 MWD/T. The irradiation-induced hardness

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II-E-1 (Contd.)WAPD-MRP-54

changes in 1150 of the 61 untransformed samples are shown in Table II. The hardness of the lower exposure samples (3100-6100 MWD/T) increased approximately 80 DPH units and was uniform across the diameter of the specimens. The two high exposure samples (7100 and 7350 MWD/T) showed a variation of approximately 130 DPH units from the edge to the center of the 0.187 in. diam specimens.

A temperature of about 400°C for 12 weeks (the irradiation time for these specimens) is sufficient to cause transformation from the metastable gamma structure to the stable epsilon and alpha uranium microstructure. Thus, the high surface hardness might be accounted for by transformation. However, temperatures of 525 and 548°C existed at the center of the two specimens which showed very low central hardnesses. It would be expected that transformation would certainly occur here and produce high hardness values. The fact that the actual central hardness was less than that before irradiation (see control specimens) indicates the possibility that central porosity may be present.

TABLE II Hardness of Unclad U-12 w/o Mo and U-13.5 w/o Mo Samples

<u>Sample No.</u>	<u>Composition</u>	<u>Exposure MWD/T</u>	<u>Calculated Temp. (°C)</u>		<u>DPH* Hardness</u>
			<u>Surface</u>	<u>Center</u>	
83	U-13.5 w/o Mo	3960	236	297	412 $\pm$ 5 401 Surface** 372 289 283
89	U-13.5 w/o Mo	7100	385	525	257 Center 265 300 402 406 Surface 386 Surface 303
61	U-12 w/o Mo	7350	400	548	289 277 Center 289 367 Surface
63	U-12 w/o Mo	6100	337	452	390 $\pm$ 6
64	U-12 w/o Mo	3140	193	233	365 $\pm$ 7
82	U-13.5 w/o Mo	Control	--	--	322 $\pm$ 4
68	U-12 w/o Mo	Control	--	--	309 $\pm$ 4

* 5 kg load

** Samples 0.187 in. diam; where one number is presented for the hardness value no variation was present across the diameter.

II-E-1 (Contd.)WAPD-MRP-54

These samples also showed decreases in density up to 7.6% as a result of the irradiation and these changes could also account for some of the indicated hardness changes. A clearer insight into the mechanism will be possible as soon as metallographic studies are completed.

A second experiment consisting of the same types of samples is currently being irradiated and is scheduled for discharge from the MTR on June 20 after an irradiation of 36 weeks to a maximum exposure of 18,000 MWD/T. Hardness measurements will also be made on these for confirmation purposes.

Seven samples of gamma quenched U-10.5 w/o Mo were irradiated to a maximum burnup of 12,000 MWD/T. The irradiation induced changes in dimensions and density have been measured and the results are shown in Table III. A maximum decrease in density of 4.1% occurred in the 12,000 MWD/T sample. Previously reported results indicated that U-12 w/o Mo irradiated 7350 MWD/T decreased 6.6% in density and U-13.5 w/o Mo irradiated 7100 MWD/T (transformed to ϵ) decreased 7.6% in density. The difference in behavior of these samples cannot be explained by phase transformations, as complete transformation results in a density increase of only 0.8% in U-10.5 w/o and 0.5% in U-12 w/o Mo. Thus, it is apparent that less density decrease due to irradiation is obtained the lower the Mo content, at least in the range of 10.5 to 13.5 w/o Mo. These samples are currently being mounted for hardness and metallographic examination.

The results of these tests will be issued in a report on the effect of irradiation on U-Mo alloys.

TABLE III Dimensional and Density Changes in Unclad
U-10.5 w/o Mo Samples

Sample No.	Exposure MWD/T	Dimensional Changes		Density Changes
		Diameter	Length	
42	2720	+0.3%	+0.15%	-0.9%
43	6700	+0.3%	+0.5%	-1.6%
45	9000	+1.1%	+0.8%	-2.7%
46	11,600	+1.6%	+1.0%	-3.5%
48	12,000	+1.1%	+1.0%	-4.4%
50	9850	+0.8%	+0.7%	-2.3%
51	5250	+0.5%	+0.4%	-1.4%

Irradiation of Transformed Specimens of U-Mo Alloys (MTR-WAPD 12-Met 4)

Twenty specimens of U-9, 10.5, 12 and 13.5 w/o Mo alloys which were heat treated at 525°C so as to partially transform the gamma phase, are currently being irradiated in the MTR and are scheduled to be discharged on May 30, 1955 after a maximum burnup of about 2500 MWD/T.

Electrical resistivity measurements are continuing on unirradiated control samples in order to follow the transformation of these alloys at 525°C.

II-E-1 (Contd.)

WAPD-MRP-54

Results to date indicate that transformation has not been completed in these alloys after 200 hrs at 525°C.

Additional measurements of the temperature coefficient of electrical resistivity of the U-12 and 13.5% Mo alloy specimens have been made. These results are summarized below:

<u>Hrs at 525°C</u>	<u>U-12 w/o Mo (°C⁻¹)</u>	<u>U-13.5 w/o Mo (°C⁻¹)</u>
0	-.092 x 10 ⁻³	-.051 x 10 ⁻³
8	-.101 x 10 ⁻³	-.041 x 10 ⁻³
25	-.073 x 10 ⁻³	-.021 x 10 ⁻³
48	-.037 x 10 ⁻³	+.021 x 10 ⁻³
76	0	+.120 x 10 ⁻³
312	+.088 x 10 ⁻³	+.179 x 10 ⁻³

The above values were calculated by assuming that the relationship between electrical resistivity and test temperature is linear between -25 and 25°C. The coefficient of resistivity, calculated on the above basis, is seen to change sign after 76 hours of transformation for the U-12 w/o Mo alloy and after about 35 hours for the U-13.5 w/o Mo alloy.

This experiment is designed to determine the effects of irradiation on transformation kinetics of metastable gamma phase U-base alloys. As such it is of special interest for future alloy development work and will be continued and expanded to include other U-base alloys.

Irradiation of Unclad U-3.8 w/o Si Alloy (MTR-WAPD 12-Met 5)

It was indicated in WAPD-MRP-53 that efforts to reveal the microstructure metallographically of the irradiated U-3.8 w/o Si specimen ends had not been successful. Since this time an extensive effort has been made in this respect using various chemical and electrochemical etchants and also cathodic vacuum etching techniques. However, none of the techniques have been successful in revealing the structure of these samples. Other etchant solutions are currently being investigated. Examination of the "as-polished" specimens revealed no cracks or porosity.

Microhardness measurements are currently being made on these specimens. The unirradiated hardness measurements were made in the transverse direction of the extruded samples. However, the irradiated specimen ends were mounted longitudinally for metallographic examination; therefore hardness measurements on these irradiated specimens are being made in the longitudinal direction. It is planned to measure the unirradiated specimens in the longitudinal direction to determine the difference, if any, in directional hardness properties of this material. These results obtained to date are shown below:

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Specimen No.	Approximate Flux* Density In-Pile	Hardness (DPH, 1 kg. pen load)	
		Pre-Irradiation	Post-Irradiation
10	medium (position #3)	227 $\pm$ 6	442 $\pm$ 12
10	high (position #4)		447 $\pm$ 12

- * Specimen #11 was at the approximate centerline of the A-Hole and specimen #10 was directly above this specimen.

It may be seen from the above data that the hardness of this material increased a maximum of 220 DPH units due to irradiation. Control specimens which were heat treated at 200°C, 300°C, 400°C and 500°C for 6 weeks indicated no change in hardness due to the heat treatment.

Because of the increased interest on U-3.8 w/o Si alloy as a promising fuel for future PWR cores additional samples of U-3.8 w/o Si will be irradiated to obtain more complete data on dimensional changes, hardness and corrosion behavior.

MTR Irradiation of Clad U-Nb Alloy Samples (MTR-WAPD 12-Met 7)

Four samples of Zircaloy-2 clad natural U-10 w/o Nb, canned in NaK and irradiated to an estimated maximum exposure of 2000 MWD/T, have been measured for dimensional instability. No change in dimensions were observed. The description of the specimens and the plans for further post-irradiation measurements were given in WAPD-MRP-52.

MTR Irradiation of Unclad U-Nb Alloy (MTR-WAPD 12-Met 8)

Seven specimens of unclad uranium-10 w/o niobium, canned in NaK, are being irradiated for 6 weeks in the MTR to an estimated maximum exposure of 2000 MWD/T. The irradiation consists of 3 specimens homogenized at 900°C for 24 hours and one specimen each homogenized at 900°C for 24 hours followed by the aging treatments shown below.

- 400°C - 2 weeks - W.Q.
- 450°C - 2 weeks - W.Q.
- 500°C - 2 weeks - W.Q.
- 550°C - 2 weeks - W.Q.

The effect of irradiation on the dimensional stability, density, hardness, electrical resistivity, corrosion rate, and microstructure will be determined. Pre-irradiation properties of these samples are shown below.

II-E-1 (Contd.)WAPD-MRP-54

<u>Heat Treatment</u>	<u>Density (gm/cm³)</u>	<u>Hardness (R_a)</u>	<u>Electrical Resistivity (10⁶ ohm cm) R.T. (26°C)</u>	<u>-196°C</u>
900°C-24 hr	16.51	64.5	65.70	71.19
900°C-24 hr / 400°C-2 wk	16.71	76.5	52.15	35.66
900°C-24 hr / 450°C-2 wk	16.76	72.0	44.28	21.80
900°C-24 hr / 500°C-2 wk	16.78	68.5	41.59	18.12
900°C-24 hr / 550°C-2 wk	16.81	66.0	40.21	16.60

Irradiation of Unclad U-10 w/o Nb Samples to High Burnups
(MTR-WAPD 12-Met 9,10)

Two experiments of unclad 5 w/o enriched U-10 w/o Nb, given similar treatments as those shown for MTR-WAPD 12-Met 8, had been scheduled for insertion into the MTR. However, due to the recent choice of UO₂ as the PWR Core I fuel material, these programs will be tabled indefinitely.

Radiation Cycling of Clad U-12 w/o Mo Alloy (MTR-WAPD 19-Met 1, 2)

All of the clad U-12 w/o Mo alloy specimens which were irradiated in the Bettis "A" piece cycling facility have been returned to the Hot Laboratory after 6 weeks irradiation. These rods will be examined visually for gross changes. Further tests of this nature on this material have been discontinued because of the recent de-emphasis on U-12 w/o Mo as a PWR fuel element.

Restrained Uranium (MTR-WAPD 4-Met 4)

Three specimens of unalloyed natural uranium which were rolled in the alpha region so as to produce a high degree of preferred orientation, were assembled into NaK filled Zircaloy-2 capsules and irradiated for a maximum burnup of about 400 MWD/T. The specimens were assembled in the capsules so that they were forced to grow in-pile against a calibrated Inconel X spring. Also, one end of each sample was drilled so that a thermocouple well extended into the middle of the specimen. Stainless steel-clad chromel-alumel thermocouples were inserted into the wells and extended out of the reactor to permit continuous recording of the central temperatures of the uranium specimens. These containers were returned to the Hot Laboratory and disassembled and the following observations were made:

a. The uranium specimens grew in length without buckling or warping so that it was possible to determine the stress exerted on the spring by the increase in length of the uranium specimens.

b. The specimens decreased in diameter as follows: the diametrical decrease in the region of the thermocouple well was approximately equal to the clearance that existed between the specimen and the thermocouple well prior to irradiation. The diametrical decreases in the regions not restrained by the thermocouple well appeared to be greater toward the end of the specimen with the smallest diameter directly in contact with the spring.

II-E-1 (Contd.)WAPD-MRP-54

c. The irradiated springs were recalibrated and it was found that the spring constants had not changed from their unirradiated value.

The specimens were photographed and the negative enlarged to about 8X from which length measurements were obtained. These results are summarized below:

<u>Specimen No.</u>	<u>Estimated Exposure (MWD/T)</u>	<u>Length (in.)</u>			<u>Diameter at Spring Sect. (in.)</u>			<u>Stress Exerted By Uranium (psi)</u>
		<u>Pre</u>	<u>Post</u>	<u>ΔL</u>	<u>Pre</u>	<u>Post</u>	<u>ΔD</u>	
# 6	100	0.752	0.870	0.118	0.251	0.233	0.018	2930
# 3	200	0.752	0.897	0.145	0.251	0.235	0.016	3410
# 2	400	0.753	1.007	0.254	0.251	0.223	0.028	6140

The results of the temperature measurements are not clear since the data have not been forwarded from the MTR. Verbal communication has indicated that the thermocouple exposed to the maximum neutron flux shorted out and the temperature recorded for the next highest neutron flux position was about 250°F. From these results, the following temperatures are estimated at the center of the uranium specimens:

<u>Specimen No.</u>	<u>Estimated Temp. (°C)</u>
# 6	60 to 100°C
# 3	121°C
# 2	250 to 300°C

It is planned to irradiate two additional assemblies similar to this to maximum exposures of about 600 MWD/T and 1200 MWD/T respectively, to determine if a stress level that prevents further growth will be reached.

Irradiation Effects on Core Materials (53b31)UO₂ Irradiation Program R. H. FillnowMTR-WAPD 14-Met 1

Six samples of fused MCW UO₂, hydrostatically pressed and sintered to 95% of theoretical density, were irradiated in the MTR to a maximum burnup of 5600 MWD/T. The details of the experiment and earlier data on four specimens were reported in WAPD-MRP-52 and WAPD-MRP-53. One sample has been sectioned and mounted and is being prepared for metallographic examination. Due to difficulties encountered in the initial determinations of fission gas release from UO₂ samples, it was decided to hold the remaining specimen from this experiment until the method of analysis was fully worked out. It is expected that a measurement of fission gas release from this specimen at temperatures up to 1000°C and a determination of its leaching behavior when exposed to static high temperature water will be completed before July 1.

MTR-WAPD 14-Met 3

Twelve cold pressed UO₂ specimens representing several source materials were irradiated in the MTR to a maximum exposure of 7,000 MWD/T. Details of the experiment, specimens, and irradiation behavior were reported in WAPD-MRP-52 and WAPD-MRP-53.

Metallographic specimens from the sintered MCW UO₂ sample have been prepared but final examination will be delayed until the remotely operable cathodic vacuum etching unit can be put into service at the end of June. The equipment for the measurement of density of small pieces of irradiated UO₂ has been completed. Density measurements on representative samples from the experiment will be made as soon as a suitable method for sectioning the specimens has been worked out.

One of the Mallinckrodt natural UO₂ specimens (7C) was punctured under vacuum, and the fission gases which were liberated at room temperature were isolated by adsorption on activated charcoal at -196°C. An analysis was performed on the radioactive species present. The three nuclides which were identified were 9.4-yr Kr⁸⁵, 12.0-d Xe^{131m}, and 5.3-d Xe¹³³. The disintegration rates of these nuclides were corrected to the time of reactor shutdown to give the initial disintegration rates, D°(obs), listed in column 2 of Table I. Estimates of the total amount of each species present in the specimen at the time of shutdown are shown in column 3. The estimated fraction of the gases liberated at room temperature is given in column 4.

II-E-1. (Contd.)WAPD-MRP-54Table I

<u>Nuclide</u>	<u>D° (obs) dpm</u>	<u>D° (total) dpm</u>	<u>$\frac{D^{\circ}(\text{obs})}{D^{\circ}(\text{total})}$</u>
Kr ⁸⁵	1.77x10 ⁹	3.57x10 ¹⁰	0.05
Xe ^{131m}	2.2x10 ¹¹	1.85x10 ¹²	0.12
Xe ¹³³	4.6x10 ¹³	3.42x10 ¹⁴	0.13

The fraction of Kr⁸⁵ liberated appears to be about a factor of 2 lower than that of the 2 xenon species. However, the estimate of the total Kr⁸⁵ present in the specimen at shutdown is more reliable than those for the xenon isotopes since, with a long half-life, it is much less dependent on fluctuations in pile operating conditions.

Measurements of the gases liberated at higher temperatures could not be made due to failure of the heating apparatus.

WAPD 14-Met 4

Eight cold pressed UO₂ samples were irradiated in the MTR to an estimated maximum burnup of 7000 MWD/T as outlined in WAPD-MRP-53. The examination of these specimens will be delayed until an evaluation has been completed of a number of specimens, recently irradiated, which have been prepared according to the reference fabrication method.

MTR-WAPD 14-Met 7

Four clad samples of MCW natural UO₂, cold pressed and sintered to 93% of theoretical density, were inserted into the MTR in a thermal neutron flux of 3x10¹⁴ nv on May 31. The samples will be irradiated for a period of 72 wk in order to determine the irradiation behavior of UO₂ fuel after long exposures.

MTR WAPD 19-Met 1 (A-Piece Cycling Experiment)

The purpose of this experiment was to determine the effects on UO₂ of radiation cycling under temperature conditions similar to PWR reactor operating conditions. Samples were to be cycled alternately 5 min in and 5 min out of the high neutron flux region. Originally, the specimens were to have been cycled into and out of a flux of 3x10¹⁴ for 6 wk. However, due to a breakdown in the hydraulic system during which no cycling occurred, the specimens received, first, a 3-week period of cycling followed by a 3-week period without cycling during which time only half the specimens were exposed to a high neutron flux. As a result, the irradiation was extended to 9 weeks and the samples were cycled for a second 3-week period. Each sample was cycled approximately 1800 times during the course of the irradiation.

II-E-1 (Contd.)

WAPD-MRP-54

The following specimens were irradiated under this experiment.

Table II

<u>No.</u>	<u>UO₂ Source</u>	<u>Method of Fabrication</u>	<u>Density (% theoretical)</u>	<u>Diam (in.)</u>	<u>Length (in.)</u>
1	MCW	hot pressed	81	0.4	5
1	MCW	cold pressed and sintered	92	0.4	5
2	steam oxidized	hot pressed	89	0.4	5
2	steam oxidized	cold pressed and sintered	90	0.4	5

Of the 6 specimens listed, 3 of the steam oxidized and the hot pressed MCW have been examined and dimensions obtained. No changes in dimensions were observed.

MTR-WAPD 19-Met 2

This experiment is a duplicate of MTR-WAPD 19-Met 1 with the exception that the specimens are being irradiated in a different order to take into account the flux gradient along the specimen train. This experiment was inserted into the MTR on April 18 and is scheduled for removal on June 20.

CR-WAPD-Met 4A First X-1 Rod Loop Test

Four samples of steam oxidized UO₂ containing 6.1 w/o uranium-235, hot pressed to 80% of theoretical density, were irradiated for 6 weeks in the Chalk River X-1 Loop to an exposure of 1540 MWD/T. Two of the samples were irradiated with a 0.005 in. hole drilled through the cladding. Details of the specimens, experiment, and irradiation behavior were reported in WAPD-MRP-52 and WAPD-MRP-53. Sections have been removed from the two undefected specimens and metallographic examination will be completed by the end of June.

A portion of the defected specimen which did not rupture during irradiation was subjected to static leaching conditions in 1500 ml of degassed H₂O at 650°F and 2200 psi. The test ran from May 2 to June 2 at which time the autoclave was shut down and the sample and water removed. Water samples were taken twice weekly during the test period. The rate of leaching of radioactive materials into the water was monitored by separating Cs¹³⁷ from these samples. The activity of the water, which was measured initially with 1 day after the start of the test, remained essentially constant for the duration of the test. The average value of the Cs¹³⁷ content in the unleached UO₂ was estimated to be 2.7×10^{10} dpm. On the basis of this estimate about 6.8% of all the Cs¹³⁷ in the specimen was leached out during this test.

II-E-1 (Contd.)WAPD-MRP-54

The autoclave water was removed on June 2 and is being analyzed for long-lived fission products. The sample was re-inserted on June 3 into the same autoclave in clean degassed water. A test similar to the one discussed above is being repeated to check if any further leaching had been prevented in the first test due to a saturation effect. The residual UO_2 will also be analyzed for the same species in order to determine what fraction of the materials is liberated by the leaching process.

CR-WAPD-Met 5 Second X-1 Loop Test

The defected UO_2 specimen which showed a weight increase of 0.59 g as reported in WAPD-MRP-53, was dried under vacuum at $120^\circ C$ for 24 hr and weight changes were noted. A 0.26 g decrease in weight was observed after the specimen had been dried for 12 hr at $120^\circ C$; exposure of the sample for a second 12 hr period at the same temperature showed no further decrease in weight. During both periods of heating, the exhaust gases were monitored to detect any radioactive fission gases which might be liberated. No gas was detected. The sample is now being sectioned and prepared for metallographic examination.

One of the undefected samples of enriched steam oxidized UO_2 was punctured under vacuum and the fission gas liberated at room temperature was isolated. Successive fractions of gas liberated between $28^\circ C$ and $350^\circ C$ and between $350^\circ C$ and $900^\circ C$ were also removed. The specimen was then held at room temperature under vacuum for approximately 8 hours. The temperature was then raised to about $900^\circ C$ and will be held there for several days to determine if additional gas is removed after prolonged heating. Detailed analyses on the individual gas samples are also being conducted.

It is expected that an experimental determination of the total fission gas content of the specimen will be made by July 15.

CR-WAPD-Met 7 Third X-1 Loop Test

The third Chalk River X-1 Loop experiment was initiated on May 12. Details on the 2 UO_2 specimens, one of which has been defected, are shown below:

Material - UO_2 prepared by decomposition of uranyl nitrate and subsequent reduction in hydrogen

Enrichment - 4.7 w/o uranium-235

Method of Fabrication - cold pressed and sintered in hydrogen atmosphere in molybdenum furnace

Density - 95%

Diam of UO_2 - 0.344 in.

Length of UO_2 core - 2.625 in.

Overall diameter - 0.4 in.

Overall length - 3.5 in.

Defect - 0.005 in. hole

II-E-1 (Contd.)

WAPD-MRP-54

A second specimen similar to the defect specimen was inserted into the loop as a control sample.

The loop has been in operation from May 12 to the present without excessive introduction of fission products in the loop water. The loop water activity has varied between the following limits:

<u>Source of Activity</u>	<u>Activity Level</u>
Kr ⁸⁸	3.4x10 ³ to 2.5x10 ⁵ dpm/ml
Gross <i>β</i>	2x10 ⁴ to 4x10 ⁶ dpm/ml

More complete information on loop operation and water activities is presented under project 53b34.

MTR Waterlogging TestsI Rabbit

The purpose of this test is to subject defected UO₂ fuel elements to static high pressure water under reactor cycling conditions. The tests are carried out in water filled autoclaves pre-pressurized to 2000 psi with helium. These autoclaves are cycled into and out of a thermal neutron flux in the MTR Rabbit facility.

The six specimens listed below were cycled 40 times into a thermal neutron flux of 5x10¹³ nv. The cycle consisted of a 2-minute exposure in the reactor and a 3-minute cooling period outside the region of high flux. The small weight gains which were noted were less than the increase in weight which one would expect as a result of water occupying the small clearance space between the fuel and the clad.

Table III

<u>Specimen</u>	<u>Material</u>	<u>Method of Fabrication</u>	<u>Density</u>	<u>Defect</u>	<u>Dimensional Changes</u>
26S	MCW	CP&S in carbon resistor furnace	94%	5 mil	none
29S	MCW	CP&S in carbon resistor furnace	93%	5 mil	none
30S	MCW	CP&S in carbon resistor furnace	93%	10 mil	none
31S	MCW	CP&S in carbon resistor furnace	93%	10 mil	none
32S	MCW	CP&S in carbon resistor furnace	92%	10 mil	none
35S	MCW	CP&S in carbon resistor furnace	94%	10 mil	none

Examination of the specimens after irradiation indicated that no changes had taken place other than a slight discoloration of the cladding.

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2. In-Pile Radiation Exposure Tests of PWR Fuel Materials (53b34) P. Cohen

Exposure tests of PWR fuel materials in the X-Rod loops at Chalk River proceeded satisfactorily during the report period. Radiochemical information was accumulated as fast as available manpower permitted. Details are presented below.

X-1

After successful out-of-pile operation, the test section containing two oxide samples (one defected) was inserted in the NRX Reactor on May 12. The specimens were cold-pressed and sintered in a hydrogen atmosphere in a molybdenum-wound furnace to 92-94% theoretical density. The pressure drop across the control test section in the loop became excessive and it was replaced with a section of pipe on June 1. The loop otherwise has operated satisfactorily at specified test operation conditions.

Water activities have been extremely low, compared with those found in previous defect tests. Kr^{88} activity began at 1.9×10^5 dpm/ml, fell to 3.4×10^3 dpm/ml on May 18, and subsequently rose to 2.4×10^5 dpm/ml on May 25. Gross beta-gamma activity on evaporated samples counted one hour after sampling has ranged from 2.2×10^4 dpm/ml to 4.4×10^6 dpm/ml before the ion exchanger, and from 2.4×10^4 dpm/ml to 1.3×10^5 dpm/ml after the ion exchanger. These values do not, of course, include rare gas (Kr and Xe) activity, since the samples are evaporated. There is evidence, too, that some iodine activity is lost during evaporation. In addition, aliquots of the same sample, evaporated at different times, but all counted at the same time (1 hr) after sampling, show, in general, increasing activities with increasing elapsed time between sampling and evaporation, indicating a growth of Rb^{88} from Kr^{88} during this time, prior to expulsion of the Kr^{88} . For this reason, gross activities after the ion exchanger are too high, and gross decontamination factors across the ion exchanger have not been calculated. Decay curves indicate that about 90% of the activity at time of sampling has an 18 min half life corresponding to Rb^{88} .

Delayed neutron activity has also been low, ranging from 30 to 40 counts per second at full power. At reactor start-up, the delayed neutron activity peaks sharply to as high as 400 counts per second, dropping to its steady level within several minutes; this is apparently caused by expulsion of water saturated with iodine and bromine delayed neutron emitters as the fuel temperature rises. Gamma monitor readings also show peaks at start-up, and occasionally also show a rise at shutdown.

Coincident with the peak in gamma and delayed neutron readings, iodine analyses also rise at start-up, by as much as a factor of 200 from 1.5×10^4 to 3×10^6 dpm/ml counted approximately 2 hr after sampling. Decay curves indicate the major iodine activity has a half life of about 2 hr, only about 10% of the iodine activity being 8 day I^{131} . No tellurium was found, even during periods of high iodine activity. Apparently 77-hr Te^{132} is not released from the fuel element, though its daughter, 2.4-hr I^{132} is released.

II-E-2 (Contd)

Ion exchange decontamination factors for iodine range from 150 at low iodine levels to 3.4×10^4 at high iodine levels. Mo99, counted 1.2×10^5 dpm/ml during a start-up peak compared with $4.5 - 6.5 \times 10^4$ dpm/ml during steady operation. Ion exchange decontamination factors for molybdenum ranged from 40 to 105.

General activity surrounding the sampling drybox is less than 5 mr/hr. Survey readings were taken inside the shielded loop bay 8 hr after a shutdown following more than 2 weeks of operation. The general radiation level in contact with the piping was 30 mr/hr, the highest reading being 60 mr/hr at the pump inlet. These readings are comparable with those made prior to the start of the test, and indicate that little long-term contamination has been caused to date by this test.

Analyses are still proceeding on the ion exchange resins from the first and second defect tests.

This test is scheduled for 4 months irradiation.

X-3

The X-3 loop containing U-Mo specimens (1 defected and 12 undefected) has continued to operate at design operating conditions without incident since insertion on March 7.

Gross beta-gamma activity on evaporated samples has risen slightly, ranging generally now from 1.8×10^4 to 1.4×10^5 dpm/ml (compared with $4 \times 10^3 - 5 \times 10^4$ dpm/ml for the previous period). There have been several peaks of activity, one to 1.7×10^6 occurring when power failure caused a loss of flow and pile trip, one to 7×10^5 occurring during pile shutdown, and a third evidenced on the gamma monitor when flow was suddenly increased. These peaks are apparently due to crud, since the Kr⁸⁸ activity did not increase following the first peak. Kr activity is up by about a factor of two from the last period to about 10^3 dpm/ml. The delayed neutron monitor reads about 10-15 counts per second.

Future Work

Future tests will be directed toward obtaining more detailed radio-chemistry data. To this end modifications will be made in loop X-3 after scheduled removal of the present test in July, to permit more convenient and frequent sampling of contaminated piping, and a PWR length oxide defected test element will be tested in loop X-1 at the end of the present test in that loop. By the start of these tests it is expected that up to 3 radiochemists will be available for work at Chalk River, as required.

3. In-Pile Screening Tests of PWR Fuel Materials (53b33) P. Cohen

The HB-5 (MTR-WAPD-13) loop is presently being modified for thermal cycling testing. A new in-pile tube and a hydraulic cycling mechanism are being set up at the MTR for out-of-pile testing which will continue until June 20.

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At that time, tests will start on an undefected UO₂ sample.

The U-Nb specimen inserted on April 20 was removed after one 3-week cycle to allow insertion of the burnout test (WAPD-20) on May 9. The burnout sample was removed during a special shutdown on May 20 after a rupture of the fuel element. A more detailed report of the burnout test is presented under II-E-4.

Activities observed during the high pH (9.1 - 9.8) test of the defected U-Nb alloy were remarkably constant, averaging 6000 dpm/ml on evaporated samples, with a gross ion exchange decontamination factor of 10. One sample for iodine showed its activity to be 2000 dpm/ml, and an ion-exchange decontamination factor of greater than 1000.

In preparation for testing with defected UO₂ specimens, a study is being made to determine what loop modifications are necessary for installation of sufficient shielding for protection against the higher activity levels. These modifications are presently scheduled to be made at the completion of the first cycling test, which will run from 9 to 15 weeks.

4. In-Pile Proof Tests of PWR Fuel Assemblies (53b35) P. Cohen

NRX-CR-V

Prefabrication of piping for the WAPD-CR-V loop (Central Thimble - NRX Reactor) and all the assembly work at Bettis Plant has been completed by the subcontractor.

The following frames have been completed and were shipped to Chalk River on the dates indicated.

- a. Frame "A" - Main Loop System - May 13
- b. Frame "B" - Catch Tank - May 19
- c. Frame "C" - Purification System - May 25.

All miscellaneous equipment and material necessary for the completion of the CR-V loop has been shipped to Chalk River with the exception of the following items:

- a. One (1) 150 D Pump
- b. Spare parts for Fisher & Porter Flowrator damaged in transit to Chalk River

It is expected that the above items will be shipped by June 7.

The status of the installation of the major components of the loop at Chalk River as of June 1 is as follows:

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- a. Frame "A" - 50% completed
- b. Frame "B" - 100% completed
- c. Frame "C" - 100% completed
- d. Frame "D" - 100% completed
- e. Frame "E" - 100% completed
- f. Frame "F" - 100% completed
- g. Frame "G" - 100% completed

- (f) The installation of interconnecting piping, service piping, shielding, and electrical systems is proceeding at a satisfactory rate.

From an overall viewpoint, the installation of CR-V loop appears to be on or ahead of schedule, and barring unforeseen difficulties, the loop should be ready for out-of-pile testing starting July 5.

Fabrication of the in-pile tube was started by the sub-contractor. Completion is scheduled for June 17.

MTR-HT-1

Out-of-pile testing on the HT-1 loop (MTR-WAPD-15) was completed and the test section was inserted in-pile on May 9 for the first progressive failure test with a uranium oxide fuel assembly. In this sample the center fuel rod is bent so that it contacts two adjacent fuel rods. Heat balances on the test sample could not be obtained because of an installation error on the thermocouples used for measurement of the temperature rise across the in-pile test section. By the time the instrumentation was corrected, the fuel arrangement in the reactor was changed, decreasing the flux at the HT-1 hole, so that the flux at the HB-5 hole could be raised for the WAPD-20 burnout test. As a result, the heat release rates measured on HT-1 were low, as expected. At the May 31st. shutdown the reactor was reloaded in the standard arrangement. After start-up on June 2, measurements were again made of the heat release and these were still found to be low by a factor of two to two and one-half, the uncertainty being due to the low value of the temperature rise being measured. The perturbed flux could not initially be calculated to better than a factor of two and the observations could be explained by too high an estimated flux and too low enrichment of the test section. However the possibility was considered that the sample was not in juxtaposition with the reactor fuel. This was checked and it was found that the sample was off-set five inches with respect to the fuel loading. Further investigation disclosed that the stop, fixing the position of the sample had been located on the drawings for a north insertion of the in-pile tube, whereas the actual insertion had been made from the south side. Since the reactor fuel is loaded on the north side, the observed offset resulted.

The following steps are being taken to correct this error. First, a new in-pile tube with correct location of the sample is being procured. Second, means are being investigated for trimming the sample for the second test so that

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it can override the stop of the present tube, and rest against a flow transition in the presently inserted tube, in approximately the correct position. Third, another subassembly of approximately double the enrichment of the first sample is being procured. The tentative plan for testing is as follows. At the next shut down, June 20, an attempt will be made to insert the second sample in the correct position by overriding the stop. If this can be achieved, the heat release will be measured during operation, and the necessity for higher enrichment determined. If the sample cannot override the stop, MTR would be requested to rearrange the fuel, to provide an approximately correct juxtaposition of fuel and sample, the heat release measured during operation and the need for higher enrichment determined as above. If either higher enrichment or prolonger lattice rearrangement are required for design test conditions, a new in-pile tube would be inserted at the August 1 shut down, with the second reference enrichment sample, or a third higher enrichment sample, as required.

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F. Reactor Physics1. PWR Core and Control Theoretical Physics (52all) S. Krasik

An intensive review of the method of calculating the reactor physics parameters needed for design calculations has been completed. The methods used for computing the age, resonance escape probability, diffusion constant for the fast and slow group, fast fission factor and thermal utilization were examined in detail and procedures established for each of these constants. These procedures have been used to establish the parameters for the analysis of the PWR reference design which is now being carried out.

Of these parameters the age (τ) and resonance escape probability (p) presented the greatest difficulties. Furthermore these parameters have a most significant effect on the calculation of the performance of the seed and blanket design.

After examination of several alternatives, it was decided to compute the resonance escape probability by the Wigner method using the Dancoff correction. This procedure will be used pending completion of the Monte Carlo calculations of p on the UNIVAC and NORC. For the uranium oxide the surface-to-mass term is treated to account for the reduced uranium density in the UO_2 ; a Dancoff correction factor is applied for the resonance flux depression caused by neighboring rods. When tested by application to the TRX metal critical experiments, the values calculated by this scheme apparently agree within 1% with the values deduced from the experimental results. Another point of interest is the method of treating the blanket subassembly which is surrounded by a substantial annulus of water. This is calculated first determining the resonance escape probability of the interior of the subassembly and then treating the subassembly and water annulus as a cell. This produced a slight increase in p over a homogeneous calculation.

The two group age for the multiple mixture presented some problems which were resolved by using an equivalent one group formula fitted to results of the CPC multigroup calculations. Multigroup results using the MUFT-II code have been completed for a number of binary mixtures.

The two group age, fast removal cross section and fast diffusion constant are written as follows:

$$D_f = \left[3 \sum_i N_i \sigma_{tr}^i \right]^{-1}$$

$$\Sigma_f = \sum_i N_i \xi^i \sigma_s^i$$

$$\tau = \left[3 \sum_i N_i \sigma_{tr}^i \sum_j N_j (\xi^j \sigma_s^j) \right]^{-1}$$

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where the N_i are the respective atomic densities of the i^{th} constituent. The results of the numerical calculations of binary mixtures are fitted with the metal-to-water volume ratio S^* as the variable so that

$$D_f = D_f^w \frac{(1 + S^*)}{(1 + aS^*)}$$

$$\Sigma_f = \Sigma_f^w \frac{(1 + bS^*)}{(1 + S^*)}$$

$$\tau = \tau^w \frac{(1 + S^*)^2}{(1 + aS^*)(1 + bS^*)}$$

where the superscript w refers to the value of the parameter for water. From MUFT-II results for hydrogen, water with and without oxygen degradation, water-uranium mixtures and water-zirconium metal mixtures the following constants are deduced:

	σ_{tr} (barns)	$\xi \sigma_s$ (barns)
Hydrogen	2.67	0.711
Oxygen	3.00	0.060
Uranium	9.10	0.264
Zirconium	4.00	0.000

These effective cross sections, when used in the formula above, fit the calculations from which they were deduced to better than 5% for $0 \leq S^* \leq 1$. The calculations made for UO_2 -water mixtures were not used in the fitting process and thus afford a check on the accuracy of the scheme. This comparison is tabulated below.

	MUFT-II Calculation	One-Velocity Fit
$S^* = 0.5$		
D_f	1.212 cm	1.24 cm
τ	33.5 cm ²	34.4 cm ²
Σ_f	0.0362 cm ⁻¹	0.0361 cm ⁻¹
$S^* = 1.0$		
D_f	1.195 cm	1.23 cm
τ	41.7 cm ²	42.3 cm ²
Σ_f	0.0286 cm ⁻¹	0.0291 cm ⁻¹

A recently obtained result of considerable interest in the seed calculations is the calculated zirconium-water age at $S^* = 1.0$. This is 59.9 cm² for zero buckling and about 60.5 cm² at a buckling of 50×10^{-4} cm⁻² which is the buckling of the clean ZPR-I experiments from which an age of 62.2 was inferred

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experimentally. In this calculation the effect of anisotropic scattering by zirconium (estimated by fitting of Los Alamos data) increases the age at $S^* = 1.0$ by 12 cm² over the value with isotropic scattering.

A similar one-velocity fitting process has been completed for the fast fission factor ϵ . From this fit an effective inelastic scattering cross section can be inferred for oxygen, uranium and for hydrogen. Using the following form for a homogeneous mixture

$$\epsilon - 1 = \frac{(v-1 - \frac{\Sigma_c}{\Sigma_f}) \Sigma_f}{\Sigma_{in} + \Sigma_c - (v-1) \Sigma_f}$$

the results given in the table were obtained. The tabulated cross sections provide a fit to within 1% in $(\epsilon - 1)$ over the range $0.25 \leq S^* \leq 1$. It is assumed that $v = 2.48$ and $\alpha = 0.14$ for uranium.

	σ_f	σ_c	σ_{in}
Hydrogen	0.00	0.00	1.38
Oxygen	0.00	0.024	0.171
Uranium	0.30	0.042	2.32

2. PWR Flexible Critical Experiment (52b11) R. T. Bayard

The work on the first metal critical assembly has been completed and the core disassembled. A report is in preparation.

The second critical assembly (uranium-molybdenum metal critical assembly) has been loaded to 340 bundles and is still subcritical. The k_{∞} is probably of the order of 1.05 as judged by the shape of the inverse counting rate vs geometric buckling curve. The core will be disassembled and reassembled using a mixture of bundles with and without molybdenum in order to achieve criticality with a reasonable number of bundles. This work will proceed only as the seed and blanket critical experiment schedule permits, since the major effort is now on the slab critical.

The third assembly (a highly enriched slab and blanket) has been started in the first tank. This is a slab reactor having the geometry and loading corresponding to one side of the seed in the PWR reference design. The slab is built out to a length equivalent to eight clusters; this is sufficient to make the reactivity close to that of an infinite length slab. A first slab was built with a loading density corresponding to the PWR seed and an annular width of about 5-1/4 in. This slab did not become critical with a water reflector. A second slab with an annular width of about 5-3/4 in. was then built; this slab likewise did not become critical with a water reflector. It was estimated that this slab was about 2-1/2% subcritical. A uranium metal blanket layer was then added to one side of the slab. With a 2 in. blanket layer on one side the reactor was critical. The analysis of these experiments has not yet been completed.

3. Two-Region Critical Experiments (52b12) J. R. Brown

Measurements are just being completed on the 3-to-1 water-to-uranium volume ratio lattice of the 0.387 in. diam fuel rods of 1.3% enrichment. The lattice of 2-to-1 water-to-uranium ratio will be started in the next two weeks. The buckling measurements on this lattice were reported in WAPD-MRP-53.

The results obtained from the temperature coefficient measurement are as follows:

Temperature Range °C	Average Temperature °C	Temperature Coefficient $\Delta\rho/\Delta T$
21.37 - 27.53	24.45	- 0.55 x 10 ⁻⁴
26.83 - 32.48	29.66	- 0.63 x 10 ⁻⁴
32.48 - 36.53	34.50	- 0.71 x 10 ⁻⁴
36.52 - 43.69	40.10	- 0.81 x 10 ⁻⁴
41.29 - 47.23	44.26	- 0.87 x 10 ⁻⁴

The data for determination of M^2 by the water height method has been completed.

Measurements of the thermal utilization (f), the resonance escape probability (p) and the fast fission effect (ϵ) have also been made for this lattice and the data is being analyzed. This analysis will be completed in approximately three weeks.

4. Primary Loop Shielding Physics (52a13) S. Krasik

The proposed shield arrangements for the secondary piping penetrations through the concrete shield have been examined and found to be satisfactory. A more detailed analysis will be made when firm arrangement drawings are received. The present arrangements consist of a lead collar around the pipe and a lead plate in front of the elbow of the pipe. This arrangement shields against any direct radiation streaming through the penetrations.

The calculations for the heating in the thermal shields and walls of the reactor vessel have essentially been completed and a report will be issued as soon as possible. The axial average heating rate in the thermal shields varies from 1 watt/cm³ at the zone opposite the flat side of the core to 2 watts/cm³ opposite the corner of the core. The heating rate in the pressure vessel wall is well below the design limit.

According to present arrangement (Drawing 76-J-478) the heating at the inside surface of the thermal shields will be approximately 2 watts/cm³ at the region where the core is nearest to the thermal shields and will decrease to approximately 1 watt/cm³ for the zone of greatest separation between the core and thermal shields.

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The calculations for the air-conditioning requirements for the primary loop compartment have been completed. These calculations have been based on a very simple model for the transport of fission products throughout the primary system. The results of the calculation indicate that a maximum of about 6 air changes per hour and a minimum of 3 may be necessary in order to keep the air activity below tolerance if a 10 liter per hour leak rate is assumed.

5. Physics In-Pile Irradiation Experiments (53b32) S. Krasik

The work carried out under this project will be reported in the STR progress report, WAPD-MR-50, under project 43b36 under the new defined rules for reporting work administered by other projects.

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