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WAPD-MRP-57(Rev.)

Reactors - Power

TECHNICAL PROGRESS REPORT

PRESSURIZED WATER REACTOR PROGRAM

For the period

September 9 to October 20, 1955

Contract AT-11-1-GEN-14

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DATE MAR 16 1957

For The Atomic Energy Commission

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Chief, Declassification Branch

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PART II

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SUMMARY

A. Systems Design

1. The Atomic Energy Commission (the AEC) approval for the Component Cooling Water System was received. (p. 18)
2. The following system descriptions were submitted to the AEC for approval:
 - Decay Heat Removal System (p. 17)
 - Primary Plant Control System (p. 20)
 - Safety Shutdown System (p. 23)
 - Primary Plant Instrumentation System (p. 21)
 - D-C Distribution System (p. 21)
 - Core Removal Cooling System (p. 19)
3. All functional requirements have been completed for the following systems:
 - Decay Heat Removal System (p. 17)
 - Rod Control System (except for rod programming) (p. 21)
4. The majority of the Pittsburgh-Des Moines Steel Company shop drawings for the boiler chambers and the reactor chamber have been approved. (p. 14)
5. A contract has been awarded to Milwaukee Crane & Service Company for furnishing and supervising the erection of the main crane of the Fuel Handling building. (p. 16)
6. The slabs and foundations under the reactor chamber, the two boiler container chambers, the auxiliary container and the Fuel Handling Canal are complete. (p. 24)

B. Component Design and Tests

1. Approval of the plan list and outline drawing for the pressurizer was received from the Atomic Energy Commission (the AEC). (p. 29)
2. The equipment specification for the water storage tank was revised to accommodate changes made as a result of a plant arrangement study. This revised specification was sent to vendors for quotations. (p. 30)
3. The contract for the air vent muffler was awarded to the Burgess-Manning Company. (p. 31)
4. A test of the fixed roller nut feasibility prototype equivalent to one lifetime was run with no scram failures, utilizing alternating periods in hydrogenated and oxygenated water environments, and is continuing. The fixed roller nut prototype mechanism with kicker has been assembled and a test started under simulated operating conditions. (p. 32)
5. A prototype of the rod position indicator has been built and tests are being run on it. (p. 34)

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SUMMARY (Contd.)

6. Equipment specifications have been submitted to the AEC for the PWR nuclear instrumentation and the operational radiation monitoring equipment. AEC approval has been received for the PWR nuclear instrumentation specification, E-565904-A, and formal quotations have been requested from the selected vendors. (p. 35)
7. The bottom hemispherical head of the pressure vessel will be fabricated using the replacement (back-up) plates furnished by the Lukens Steel Co. Examination of the first flange forgings at the Bethlehem Steel Company disclosed an unsatisfactory material condition which has compelled the scrapping of all but one forging which was salvaged by further forming. (p. 46)
8. The photoelastic tests at Brown University have produced preliminary results of the first test, or cap test, which represents a limited portion of the vessel head with penetrations believed most critical. These preliminary results indicate the reference head design appears to be satisfactory from a pressure standpoint. (p. 47)

C. Core Development

1. The single-pass reactor with an expanded seed design will insure the 60 Mw output on 3-loop operation. Calculations showed that the two-pass design would exceed performance requirements, but at the expense of an excessive coolant velocity in the seed and an increase in mechanical design problems. (p. 44)
2. The list and brief mechanical description of PWR reactor instrumentation, with its required plant container penetrations and read-out instruments, was established this month and submitted to the AEC for approval. (p. 42)
3. Drawings of the model for the air flow tests being performed at the Westinghouse Steam Division were approved and released for manufacture. (p. 44)
4. Analyses of the PWR core protection requirements have shown that the 1% flow rate supplied by the decay heat removal system satisfactorily protects the core. (p. 45)
5. Nuclear Metals, Inc. has begun production of Zircaloy-2 tubing for the core manufacture. (p. 48)
6. A new process development at the Bishop & Babcock Company has indicated that their spinning process can produce a tubing with an integral end closure of good quality. Refinements in the process are continuing. (p. 49)
7. The progressive failure test bundle consisting of 9 UO₂ filled rods has been sectioned. Rods which were touching before in-pile exposure are separated 0.001 in. after the test. Second such progressive failure test is still to be sectioned in the lab. (p. 56)

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SUMMARY (Contd.)

8. The production of the initial 16 assemblies of natural UO_2 blanket elements for the Physics UO_2 -Flexible Program was completed October 15. (p.57)

D. Materials Development

1. Defected elements prepared by brazing core rods of U-3.8 w/o Si alloy inside Zircaloy-2 tubes with an Al-6 w/o Cu alloy barrier have satisfactorily withstood 15 days exposure in 680°F water and 14 days in 650°F water. (p.72)
2. Small ternary additions, 0.2 w/o of Mo, Nb or Cu, to arc-melted buttons of U-3.8 w/o Si have reduced the corrosion rate of this alloy by a factor of 2 to 3 as indicated by 42 day corrosion data. These additions have effectively eliminated the presence of undesirable alpha uranium in the microstructure and made the tolerable silicon content less critical. (p.73)
3. One (X 2219) of three experimental Al-base alloys being investigated as future cladding materials has satisfactorily withstood 8 weeks exposure to 680°F water. This alloy displays a linear metal loss versus time relationship amounting to 11.7 mils per year. (p. 75)
4. Zircaloy cladding is effective in restraining irradiation-induced growth of unalloyed or alloyed uranium core alloys but does not prevent irradiation-induced volume expansion. Indications are that the ultimate damage to rod-type fuel elements depends on cladding ductility. A volume expansion of 3 to 4% results in splitting of the Zircaloy cladding. (p.68)
5. Decomposition pressures (hydrogen) measured over corroded samples of gamma phase U-Mo alloys indicate that the hydride responsible for corrosion failure is not simple UH_3 . The corrosion hydride develops higher pressures and has a lower heat of formation than does UH_3 . (p. 60)
6. Irradiation on PWR UO_2 defected specimens was resumed in the WAPD-13 loop at MTR. Fuel element failure instrumentation was installed for the test. A report on the feasibility of a shim rod facility loop utilizing the WAPD-13 external loop equipment was issued. (p. 81)
7. The third UO_2 defect test in the X-1 loop continues in a satisfactory manner. All equipment for the addition of by-pass modification for future chemistry tests has been received. (p. 82)
8. The CR-V central thimble loop with a PWR subassembly proof test began operation in-pile on September 9. (p. 85)
9. Analysis of resin beds from the third UO_2 defect test in loop X-1 shows a uniform release of fission products since the start of the test. (p. 84)
10. Equipment for determining distribution of activity between filterable and non-filterable phases has been received and is being checked out. (p.86)

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SUMMARY (Contd.)

11. Preliminary data have been obtained on the adsorption of Cs¹³⁷, Sr⁹⁰ and Y⁹⁰ on precorroded stainless steel. (p. 88)
12. Sulfamic acid has been used in decontaminating Bettis B-Loop, with an overall decontamination factor of 200 after mechanical removal of loosened scale. (p. 90)

E. Physics

1. The 8 rod down program of control of the PWR has been calculated using the two-dimensional UNIVAC code of Dr. Cuthill. This configuration is considerably better than the blanket type of control in so far as flux distributions are concerned. (p. 92)
2. The fission product activity in the primary coolant has been calculated on the basis of a defected element and on the basis of a disintegrated element. (p. 93)
3. The metal blanket work on the PWR Flexible Critical Experiment is essentially completed. The oxide blanket program was started October 13. (p. 94)
4. The first TRX experiment using oxide rods has been made critical. 1280 rods of 0.600 in. diam, 70% density oxide, were required at a water to equivalent uranium metal ratio of 3-to-1. (p. 94)

F. PWR Parameters

The following is a recapitulation of current PWR parameters:

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

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SUMMARY (Contd.)

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	6.5×10^6 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	1160 psia
Main coolant piping, ID*	15.0 in.
Steam side piping design pressure	1050 psig

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

Primary Loop compartment (contained loop shutdown 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

CORE PARAMETERS

General

Heat output at full power**	790×10^6 Btu/hr
T_h (average outlet temperature)	541.4°F
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

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SUMMARY (Contd.)

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
 <u>UO₂ Region</u>	
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 x 10 ⁶ Btu/hr

Seed Region

Inside diameter	86.6 cm
Outside diameter	121.0 cm
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
Weight of zirconium flux depressors	1.43 tons

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I-A-1

Part I

PWR Engineering

A. Power Plant Analysis & Systems

1. Plant Functional Design and Operation Analysis (55all) J. E. Nolan

This activity includes the determination of plant operating characteristics, the analysis of recent simulator data and the study of plant performance during accident conditions.

Simulator Studies

The PWR Primary Coolant Volume Surge report based on the June simulator runs was submitted. The added effects of pressurizer simulation and coolant compressibility are included in this analysis. The pressure coefficient acts to increase the coolant surge by the introduction of positive reactivity, while the coolant compressibility acts to decrease the surge. The overall effect has been an increase in the coolant volume surge of 15 to 20 per cent. For a power level change of 25 per cent the increase is 2 to 3 cu ft.

The major result of the larger surge and the resulting pressure rise is to increase the possibility of the operation of the pressurizer pilot-operated relief valves. However, a surge induced spray can be used to decrease this pressure rise.

Transient Coolant Flow

A test specification for obtaining special performance data on the PWR primary coolant pump, so that more precise coastdown information can be calculated, was submitted for approval.

Plant Design Parameters and Operating Conditions

A document listing the current plant design parameters and operating conditions was submitted for information.

PWR Hazards Report

Work is proceeding on the draft of the preliminary PWR Hazards Report to be presented on November 15. Calculations have been made of the time to uncover the core after various size breaks. These times range from 19 sec for a double-ended break of the 15 in. main coolant pipe to 26 min for a break of a 2 in. pipe. Detailed calculations to determine the core meltdown after a loss of coolant accident are under way.

A meeting with a subcommittee of the Reactor Safeguard Committee was held at Bettis Plant on October 10. Possible accidents, their probability, the consequences, and corrective action to be taken were discussed.

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2. Reactor Plant Layout and Site Facilities (55a12) R. F. Devine

This project provides for the direction and coordination of the work in the preparation of detailed design and drawings for the site facilities.

Primary Coolant Piping and Components (S.O. 1-AP-8005)

Minor revisions are being made in the plan and elevation drawings of the primary coolant piping in preparation for issuing them to the vendor for shop fabrication of the piping. "Holds" will continue to be shown for the primary components until approved manufacturers' drawings are available.

Recommendations for methods of primary component removal, including pipe and seal weld cutting, and the forces and moments at the component nozzles as determined by a stress analysis, were submitted to the Atomic Energy Commission. This analysis is based upon stresses resulting from maximum operating conditions at 100 Mw gross plant output.

A drawing has been prepared of a revised type of primary pump support in which horizontal movement is accommodated by rollers instead of a sliding plate as previously contemplated. The use of rollers will tend to assure boiler motion more nearly as predicted, which is necessary in order to accurately compute reactions on the boilers from the main steam and boiler feed piping. This drawing has been referred to the Westinghouse Atomic Equipment Division, General Electric, and Allis-Chalmers for their comments as to suitability for use with their pumps.

Pressurizer System (S.O. 1-AP-8006)

Pressurizer nozzle and support locations based upon installation and piping arrangements have been established. Piping reactions on the nozzles have been prepared and transmitted to the pressurizer supplier.

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

Revised issues of the purification cubicle study drawings are being developed. The heat exchanger compartment will be enlarged and the demineralizer vent valve will be located in the auxiliary container cubicle in accordance with the practice of shielding all potentially contaminated vents. A method for shielding flowmeter and dp cell vents is under investigation.

Operational Radiation Monitoring System (S.O. 1-AP-8024)

A study drawing has been prepared of the plant container air-monitoring system, in which air samples are collected at each of the three interconnections to the auxiliary chamber and piped to monitors located in the relatively low radiation field in the auxiliary chamber.

I-A-2 (Contd.)

Internal Shielding (S.O. 1-AP-8032)

A detailed investigation of the shielding requirements for the auxiliary chamber cubicles indicates that the thickness of the concrete shielding walls on the east and west ends of the flash and blow-off and pressurizer compartments may be reduced from 2 ft 6 in. to 2 ft. The concrete wall shielding the 12 ft/diam connecting tube between the reactor sphere and auxiliary chamber will be reduced to 2 ft in thickness and a 2 ft thick wall will be added to each side of the walkway below the pressurizer cubicle. These auxiliary chamber personnel access walkways and cubicles will be shielded to a maximum of 50 mr/hr as established by PWR Radiation Shielding Design Criteria. These shielding thickness requirements for concrete walls are subject to structural investigation by Stone & Webster Engineering Corporation.

A study is being made to investigate the feasibility of replacing concrete with steel shielding at the openings through cubicle walls through which valves will be operated. The intention is to reduce the wall thickness to simplify access to the shielded valve or to reduce stem length in those valves in which the bonnet projects into the wall.

Miscellaneous Utilities (S.O. 1-AP-8033)

A meeting was held between Duquesne Light Company and Bettis Plant to discuss needs for auxiliary piping, the type of service and the general location of interconnections between the DLC turbine area and the reactor plant area. The following services were discussed: Heating steam, heating condensate, sanitary (treated) water, soft water, control air, service air, emergency charging water, contaminated condensate, demineralized water, raw cooling water, and fire protection water. A meeting will be held at a future date to discuss line size, elevation, pressure and temperature requirements.

Reactor Plant Container (S.O. 1-AP-8035)

The majority of the Pittsburgh-Des Moines Steel Company shop drawings for the boiler chambers and the reactor chamber have been approved by Stone & Webster.

The shop drawing submitted by Pittsburgh-Des Moines detailing the air locks for personnel access to the container has been approved.

Pittsburgh-Des Moines recently submitted for approval their welding process specification and welders' qualification records.

Stone & Webster has recently completed and submitted for review their preliminary specification for installing the Gunit concrete lining required in the reactor plant container for missile protection.

I-A-2 (Contd.)

Steam and Boiler Feed Systems (S.O. 1-AP-8040 and 8041)

Isometric detail drawings, complete with bills of material, have been prepared for the main steam and feed lines from the boilers to the agreed upon junction points with the piping in the turbine area at which Duquesne Light Company assumes responsibility.

Neutron Shield Tank and Reactor Vessel Support (S.O. 1-AP-8046)

The Stone & Webster Engineering Corporation has completed revision of the neutron shield tank and vessel support drawing, the principal change from previous studies being the neutron detector tube locations. The present plan is to locate a pair of detectors (one BF₃ and one compensated ion) opposite each of the four corners of the core, oriented as previously recommended to the Atomic Energy Commission (the AEC). The eight tubes will project through and terminate just above the circular closure access walkway.

The design drawings and construction specifications covering the shield tank and support are being prepared for submittal to the AEC for approval.

External Shielding (S.O. 1-AP-8047)

Design descriptions for the PWR reactor plant container shield and the PWR fuel handling shielding were submitted to the Atomic Energy Commission for approval in accordance with the shielding design criteria.

Reactor Plant and Fuel Canal Concrete (S.O. 1-AP-8047)

Stone & Webster is proceeding with the preparation of construction drawings of those walls common to the turbine generator and the reactor plant, including necessary connections and penetrations.

Service and Fuel Handling Building and Related Auxiliary Equipment Structures (S.O. 1-AP-8050)

Bids have been received on a unit price basis for the furnishing, fabrication and erection of the structural steel framing of the reactor plant service building, the fuel handling building and the auxiliary buildings.

The Stone & Webster Engineering Corporation completed the drawings showing floor plans and exterior elevations of the reactor service, fuel handling, and auxiliary buildings. Representatives of the Atomic Energy Commission and Bettis Plant reviewed these drawings during the week of September 26, 1955. These drawings will next be presented to the Bureau of Labor and Industry of the Commonwealth of Pennsylvania for approval.

I-A-2 (Contd.)

Nuclear Plant Cranes (S.O. 1-AP-8051)

A contract has been awarded to Milwaukee Crane & Service Company for furnishing and supervising the erection of the main crane of the Fuel Handling building.

The functional requirements of the other cranes have been prepared and Stone & Webster is preparing the necessary specifications for same. These cranes are as follows:

<u>Location</u>	<u>Capacity</u>
1) Clean Room	15 tons
2) Decontamination Room	20 tons
3) Core Vault	1 ton
4) Sampling Room	3 tons

3. Primary Coolant Fluid Systems (55b11) A. P. Zechella

This project covers the analysis of system requirements, preparation of schematic diagrams and system descriptions, determination of functional requirements for all system components and preparation of test specifications for all primary plant fluid systems.

The below listed PWR Fluid Systems have had Atomic Energy Commission approval subject to the resolution of minor comments:

Reactor Coolant System
Pressurizer and Pressure Relief System
Purification System
Chemical Addition System
Component Cooling Water System

Work is continuing on these systems in reviewing component proposals being received from vendors and preparation of the chapters of the Power Station Manual.

I-A-3 (Contd.)

Coolant Charging System (S.O. 1-AP-8008)

An investigation of the need for the steam blanket on the primary water storage tank was completed. It has been decided to retain the steam blanket as part of the system to maintain similarity with feed water storage systems used by Duquesne Light Company.

A study of the resin charging tanks resulted in the decision to use two resin charging tanks and to eliminate the resin charging tank outlet valves from the system. This decision was based on the fact that the moving of one tank from one boiler container to another would result in a difficult operation. The low cost of these tanks does not warrant the introduction of this additional operational problem.

After an investigation of the hydrostatic test pump, it was decided to use bronze material for the pump and the pump suction and discharge valves. The use of bronze will permit the purchase of commercially available components.

Work is currently in progress to:

Revise Stone & Webster Engineering Corporation flow diagram to conform with present system schematic diagram.

The functional requirements for the system components were revised to concur with the latest system revisions, and equipment specifications are being reviewed for compatibility with system description.

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

Work is currently in progress to:

1. Investigate the possibility of formation of an explosive mixture of hydrogen and air in the blow-off and flash tanks, and the associated design requirements of the tank vent lines to the Waste Disposal System.
2. Develop an emergency procedure for clearing the resin discharge lines in the event resin plugs the lines.

Decay Heat Removal System (S.O. 1-AP-8011)

The system design description and schematic were submitted to the Atomic Energy Commission on September 14, 1955. All functional requirements have been written.

Valve Operating System (S.O. 1-AP-8015)

An alternate scheme, incorporating motor-operated globe valves, has been designed to provide satisfactory system functioning in the event that proof

I-A-3 (Contd.)

tests of the 3-way remote-operated selector valves indicate that these valves will be unsatisfactory for use in the subject system.

Test specifications covering the hydraulic pilot valve and the 3-way motor-operated control valve, both as separate entities, have been prepared. A test specification covering the 18 in. hydraulic stop valves, as a separate entity and together with the above two valves to simulate system operation, has also been prepared.

A series of tests have been performed at the Bettis Test Facility to assure the possibility of operating a pair of 18 in. hydraulic stop valves with the pressurizer isolated from the primary coolant system. Test results indicate that this operation can be performed without causing dangerous pressure surges within the system.

Service Air System (S.O. 1-AP-8016)

A preliminary flow diagram for the Service Air System has been prepared. Design information and functional requirements are being formulated.

Component Cooling Water System (S.O. 1-AP-8017)

The Stone & Webster Engineering Corporation is preparing a revised systems description and schematic piping diagram to include the comments from the Atomic Energy Commission and the revisions as required for the canal water cooling systems design.

Reactor Plant Container Air Cooling System (S.O. 1-AP-8019)

An investigation was made of the comparative merits of using a d-c motor, high pressure air, low pressure air or high pressure hydraulic fluid for the 48 in. butterfly valves operating mechanism. It was concluded that high pressure hydraulic fluid is the most economical and reliable system for present closure time requirements.

A review of the radiation detector system was made and it is planned to recommend a back-up signal detection system. The back-up system will detect high pressure in the Reactor Plant Container due to a rupture of piping or components. The pressure signal will afford a more rapid means of sensing a major rupture in the primary plant.

General Services (S.O. 1-AP-8033)

A preliminary flow diagram and specifications are being prepared for the various miscellaneous piping to be covered by this system. A system description will be prepared for this information and will be forwarded to the Atomic Energy Commission.

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Fuel Element Failure Detection System (S.O. 1-AP-8036)

Preliminary design data have been furnished to the Stone & Webster Engineering Corporation for physical layout studies. Much of the final system design is contingent upon equipment placement. It is expected that the system design will be finalized and submitted to the Atomic Energy Commission by November 30.

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

System design description and schematic will be issued October 31.

The system is designed to contain, treat and dispose of radioactive wastes from the PWR reactor plant and its auxiliaries in such a manner as not to exceed specified tolerance levels. Essentially the system contains ion exchangers, which remove suspended and dissolved radiological impurities, a gas stripper, which removes and concentrates fission gases and decay tanks to hold these fission gases for sufficient time so that their disposal to the atmosphere will not exceed 10 per cent of tolerance.

Interested civil authorities from the State of Pennsylvania and Ohio River Valley Water Sanitary Commission were advised of PWR radioactivity problems and our proposed methods for handling them.

The Waste Disposal Section of the first PWR Reactor Hazards Summary Report is being prepared.

Core Removal Cooling System (S.O. 1-AP-8045)

System design description and schematic were submitted to the Atomic Energy Commission on September 12. Approval has not been obtained to date. All functional requirements have been written.

Safety Injection System (S.O. 1-AP-8053)

A functional requirement has been prepared for this system which will be designed by Stone & Webster to perform the following functions:

Inject water into the primary system as near the core as possible to prevent uncovering the core should a primary system rupture occur.

This system will also not include any installed spare equipment nor will it be powered by a diesel electric generator. It is expected that the system design will be completed by November 15.

Reactor Plant Container Sprinkler System (S.O. 1-AP-8054)

Bettis Plant has issued a functional requirement for the Reactor Plant Container Sprinkler System which will perform the following functions:

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1. Lower the temperature and pressure within the reactor plant container resulting from a primary system rupture.
2. Keep a continuous spray in operation after (1) is accomplished in order to control and limit a possible secondary pressure and temperature rise to the design limits of the container.
3. Keep the container atmosphere saturated with water vapor to reduce the possibility of a H_2-O_2 reaction.

The Stone & Webster Engineering Corporation is proceeding with the design based on no installed spare equipment. In addition, the requirement that the equipment may be supplied power from a diesel electric generator has been eliminated.

Consideration is being given the use of the spray pump as a booster pump to supply the boiler feed pumps for purposes of injecting water into the primary system. It is expected that the system design will be completed by November 15.

Control Air System (S.O. 1-AP-8055)

A preliminary flow diagram for the Control Air System has been prepared. Design information and functional requirements are being formulated. The air requirements have been revised from previous estimates.

Canal Water Cooling System (S.O. 1-AP-8056)

Work is continuing on the Canal Water Cooling System Description and schematic drawing by Stone & Webster to include the component cooling water system heat exchangers as a means of cooling the canal water. The revised systems will be submitted to the Atomic Energy Commission by November 15.

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

This activity includes control and instrumentation systems directly concerned with the parameters of the primary plant systems and their operation.

Primary Plant Control System (S.O. 1-AP-8026)

The Primary Plant Control System Design Description was completed and forwarded for approval. Changes in the functional arrangement of the Reactor Control Console were made, and integrated with the functional arrangement of the main instrument panel. Comments of the Atomic Energy Commission were resolved at a meeting with demonstration of a full scale mock-up of the console on October 13. The functions of all panels of the system have been determined to enable completion of equipment specification preparation by November 15.

Primary Plant Instrumentation System (S.O. 1-AP-8027)

A functional layout of the Main Instrument Panel and a general description of its components was submitted to the Atomic Energy Commission (AEC) for approval.

The integrated design description of this system was forwarded to the AEC for approval on October 4. Portions of the system representing instrumentation for individual primary plant systems were previously forwarded for information and comment. Resolutions of these comments were incorporated in the integrated system design description.

A detailed study of instrumentation accuracy requirements was completed and a meeting held with the AEC to determine the areas of application for magnetic amplifier equipment. Such equipment may be employed in most areas where amplification is required except in the instrumentation for primary loop temperatures (T_h and T_c) and pressurizer pressure. In these areas, the functions of control require greater accuracy than can presently be obtained with magnetic amplifiers, and it is expected that electronic equipment will be employed.

Reactor Plant Electric Power Distribution System (S.O. 1-AP-8028)

The description of the d-c Distribution System was forwarded to the Atomic Energy Commission (the AEC) for comments on September 21.

A study concerning the adequacy of power connections between the station and the Duquesne Light Company system with respect to the reliability necessary for the station auxiliary power system was forwarded to the AEC on October 20. This document will present a recommendation for and description of any additional power supply facilities which might be required.

A number of minor revisions are being made in the 480 v unit substations and motor control centers. These revisions will be covered in the next issue of the system description which is expected to be completed during the next report period.

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

This activity includes control and instrumentation systems directly concerned with the parameters of the core and with the operation of the reactor.

Reactor Rod Control System (S.O. 1-AP-8021)

All functional requirements for the Rod Control System components, with the exception of the rod programming device, have been completed.

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Current information indicates that uniform rod position control will not be used. Group control is expected to be the type of rod programming which will be required. The manner in which the programming will be accomplished is under investigation and is expected to be completed in the next report period.

As reported in the last Technical Progress Report, the system was designed to move two control rods at a time thus requiring an inverter for each pair of control rods. A study has been made to determine the feasibility of moving four rods at one time. If feasible, it would reduce the number of inverters required. A report and recommendation will be submitted to the Atomic Energy Commission by October 21.

Reactor Power and Temperature Control System (S.O. 1-AP-8022)

A number of simulator studies were completed during this period. These studies included (1) an investigation of the pressurizer surge line characteristics, (2) the effect of a controlled steam relief system on plant surges, (3) the addition of a pressure override signal to the temperature control signal, (4) the results of dropping two or four rods from core-center position during power-operation, and (5) the effects of rod jumps on control system stability.

The studies revealed that the surge line is underdamped for fast disturbances. For normal disturbances the damping coefficient is not critical due to the natural frequency of the pipe which is of the order of 0.8 cps. Since disturbances that can occur in the plant due to temperature transients or relief valve operation are of much lower frequencies than 0.8 cps, no additional damping is required for the surge line.

The operation of a controlled steam relief system was simulated. The system is composed of four valves of different sizes which are actuated by large load reductions. The studies indicated that such a system is capable of limiting the surges in the primary plant and preventing relief valve operation.

An override signal was added to the reference control signal which provides rod-in motion when the pressurizer pressure reaches a pre-set point. The results indicate that such a signal is capable of cutting back plant temperatures appreciably, thereby reducing the pressure surge. This signal could be used to minimize relief valve operations if needed.

The studies reveal that the plant may be able to recover from the dropping of two rods from the center of the core if the control rod velocities are such as to produce a reactivity change of the order of 2×10^{-4} per sec. If four rods are dropped the recovery is doubtful. The plant parameter that is most affected during these transients is pressurizer pressure. When four rods were dropped, the pressure dropped to approximately 500 psi before the control system inserted enough reactivity to balance that inserted by the dropped rods if the remaining rods were moved at approximately maximum control speeds.

I-A-5 (Contd.)

If the rods move in jumps instead of smoothly, the control system is not greatly affected if the jumps are of the order of $1.25 (10^{-4}) \text{ Sk}$. If the jumps are as large as $2.5 (10^{-4}) \text{ Sk}$, the transient response of the system is poorer and the possibility of instability is introduced. Therefore, if the jumps are small the difference between smooth motion and jump motion is negligible.

A report of these simulator studies will be issued during the next report period.

Nuclear Instrumentation System (S.O. 1-AP-8023)

Bettis Plant requested approval to delete the functions of auctioneering, averaging, and coincidence of the linear level power range signals. Three continuous linear level signals would be fed to the Safety Shutdown System where any signal operation devices required would be provided. The request was approved, and system component procurement will proceed on this basis.

Reactor Protection System (S.O. 1-AP-8025)

The Safety Shutdown System Design Description was submitted for approval during this period. Studies of sequential loss of flow accidents and start-up accidents are essentially complete and expected to be issued by October 31. Additional accident studies will be conducted as new core design information becomes available.

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

This project covers the simulation of the PWR power plant providing information on the effectiveness of control systems and on the thermal plant conditions as a part of plant analysis work.

The simulator program during September included studies of pressurizer operation, steam relief operation for decreasing power level, and reactor power and temperature control. Pressurizer operation was simulated for isentropic and saturation line compression of steam. The steam relief simulation was made up of four valves providing for controlled steam relief to permit the primary plant to remain in normal operation during large load reductions. Load and reactivity transient runs were made with and without the relay control system. Throughout this study a pressure coefficient of reactivity of $+2 \times 10^{-6}/\text{psi}$ was assumed.

7. Reactor Plant Installation and Tests (59all) R. F. Devine

This activity provides for the necessary coordination required among various contractors and subcontractors for installation.

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The progress of general construction is illustrated in Fig. 1. This photograph shows the type of construction on the reactor portion of the plant which is now in progress under Duquesne Light Company contract.

The slabs and foundations under the reactor chamber, the two boiler vapor container chambers and the auxiliary vapor container chamber are complete. The above slabs along with the walls enclosing the vapor chambers, which are now under construction to a height of 12 to 15 ft, are part of a Duquesne Light Company subcontract. This subcontract calls for approximately 11,000 cu yd of concrete and is 90 per cent complete.

The slabs and foundation under the Fuel Handling Canal are complete. The forms, reinforcing and concrete for the Fuel Handling Canal walls and reactor chamber walls to their full height are now under construction. This portion of the Duquesne Light Company contract calls for approximately 3500 cu yd of concrete and is 40 per cent complete.

A representative of the Stone & Webster Engineering Corporation has been stationed at the site since August 15 and a representative of Bettis Plant since September 1.

These personnel have been performing functions of inspection in an advisory capacity, and coordination of engineering in connection with the construction now in progress under Duquesne Light Company contract.

8. PWR Power Station Manual and Test Program (59cl6) J. E. Nolan

This project includes the coordination of the preparation of instruction manual and plant test specifications needed during the testing and operational phases of the plant development.

Work continued on the resolution of additional Atomic Energy Commission (AEC) comments concerning the administration of the PWR Power Station Manual Program and the submission schedule dates for the individual chapters of the manual. A document outlining the results of this work was submitted during the week of October 17.

The proposal for the PWR Inspection and Test Program was revised to include AEC comments and amplified to include more details concerning specific portions of the program. Work is in progress to submit more detailed information on the remaining portions of the program by November 1 and December 15 respectively.

Fig. 1. - View of Construction of the Reactor Portion
of the PWR Power Plant. October 6, 1955

B. Power Plant Components and Component Materials and Tests

This section covers the engineering development, design, and technical effort required for procurement of the various system components. Included, also, is information relative to non-core materials. The components specifically include valves, pumps, auxiliary pressure vessels, heat exchangers, electric power equipment, and instrumentation.

1. Non-Core Materials (53all) J. R. Hunter

Substitute for Microbrazed

The auric brazing material specimens were not subjected to corrosion testing during this period because it became necessary to review and modify analytical methods and control criteria to insure compliance with test specification requirements. The test will be resumed during the week of October 24.

The Coast Metals Company alloys Nos. 52 and 53 and the Rexweld 64 brazing materials are being machined and will be submitted for corrosion testing within the next reporting period.

Spring Materials Development

The relaxation tests on Haynes 25, Refractaloy 26, and S-816 were continued in oxygenated water at 600°F. The Haynes 25 springs (12%, 42%, and 62% cold reduced wire) tested at 50,000, 100,000, and 120,000 psi stress levels for 161 days exhibited average relaxations of 0.4%, 2.1%, and 2.1%, respectively. The Refractaloy 26 springs, tested at 70,000, 80,000, and 90,000 psi stress levels for 170 days exhibited average relaxations of 3.25%, 2.62%, and 2.56%, respectively. The S-816 springs tested at 70,000, 80,000, and 90,000 psi stress levels for 170 days exhibited average relaxations of 5.0%, 6.5%, and 7.2%. Tests will be continued until a total of 200 days for each specimen has elapsed.

The Haynes 25 springs in test were aged for 4 hr at 1000°F. Mechanical tests on wire cold reduced 40% and aged to 16 hr were attendant with progressive increases in tensile strength from 247,500 + 2500 psi for a 4 hr age to 273,500 + 1500 psi for a 16 hr age. Wire was obtained for tests to determine the optimum aging treatment. Tests to determine the torsional modulus of spring materials, and to evaluate the effect of wire size and chemistry on the torsional modulus, will be started in the next reporting period. Fabrication of the required Inconel X wire was completed but the wire has not yet been received. The required Haynes 25 wire is on hand.

Thirty Cobenium (Elgiloy) springs were heat treated and are scheduled to be placed in test in oxygenated water at 600°F during the next reporting period. Total time of test will be 200 days in autoclave plus periodic shutdowns for observation.

Study of Elastic Properties of Potential Bearing Materials

Bearing alloy specimens for determination of the compression modulus are on order. Ten replicates of each of 3 different heats of 10 alloys are being obtained. Delivery of the machined specimens is expected to start in the next reporting period, at which time testing will begin.

2. Materials for Primary Loop Components (55b16) J. R. Hunter

It was previously reported that it appeared possible to replace the partially bonded stainless clad steel plates for the reactor vessel bottom head with the spare plates then in process of manufacture. These plates, however, were also found to be of poor bond quality and could not be used for replacements. The original plates were salvaged with 1/2-in. diam plug welds on 3-in. centers. Concurrently the steel manufacturer was processing other replacement plates. The choice between the salvaged plates and the replacement plates in process was the subject of a meeting in the offices of Cobustion Engineering, Inc. on October 6. It was agreed that the fabricator would proceed with the manufacture of the lower head using the salvaged plates until it was established that the bonding of the replacement clad plate was satisfactory. The bonding characterizing the replacement plate was proved to be unsatisfactory on October 13. It was then decided to obtain the required replacement material by machining excess carbon steel from a satisfactorily bonded plate approximately 9 1/2 in. thick which was previously produced for the shipping head. The machined plates are scheduled for delivery October 25.

The heavy forgings for the flanges for this vessel were rejected by the manufacturer, Bethlehem Steel Corp., due to internal flaking. New forgings are being processed using a changed procedure which is expected to produce sound forgings.

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

18-in. Prototype Capped Manual Stop Valve

Tests on this valve in accordance with Part I of T-Spec 551926-B were completed at Bettis Plant, Large Site. On October 4, the valve was shipped to the Westinghouse Atomic Equipment Department where it is undergoing operational tests in the 16-in. test loop.

18 x 16 x 18-in. Capped Manual Stop Valves

Outline, assembly, and detail drawings were submitted to the Atomic Energy Commission (the AEC) for approval.

18 x 16 x 18-in. Hydraulically-Operated Main Stop Valves

The assembly drawing and plan list were approved by the AEC and manufacturing release was granted to the vendor for the body and cylinder castings.

18-in. Main Check Valves

An assembly drawing was submitted to the AEC for approval.

Main Relief Valves

A subcontract was agreed upon between Bettis Plant and Crosby Steam Gage and Valve Company covering relief valves meeting E-Spec 565267, 565268, and 565269. The scope of the work includes the design, development, and manufacturing of the pressurizer safety valves, the loop relief valves, and the reactor

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vessel relief valves in quantities to satisfy the requirements of the plant and complete spare units. The subcontract was submitted to the AEC for approval.

Valve Testing

Technical approval was granted by the AEC for the modification to the stainless steel general purpose test loop at Bettis Plant, Large Site. Bids were requested for the required piping subassemblies.

4. Primary Coolant Pumps (55b20) C. M. Miller

Testing of the prototype primary coolant pump is proceeding satisfactorily. All performance tests and the 500 hr proof run were completed, with no difficulties experienced.

The plan list and outline drawing for the 3 complete pumps and one spare motor, as manufactured by the Atomic Equipment Department, were approved by the Atomic Energy Commission. Approval of the general assembly, plan list, and the release for manufacture were granted.

Proposals were received from Allis-Chalmers Mfg. Co. and General Electric Company for the remaining 2 complete pumps and one spare motor on October 6. An order will be placed during the next report period.

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

Vent System Discharge Pumps

At the request of the Atomic Energy Commission (the AEC), the due date for the bids was extended from September 9 to October 5 to allow sufficient time for more vendors to bid.

Charging Pumps

E-Spec approval was received on September 12 and bids were received on October 17. The order will be placed during the next report period.

3000 psi Air Compressor and Accessories

E-Spec approval was granted by the AEC and bids were received on October 17. The order will be placed during the next report period.

6. Primary Coolant Loop (55b26) C. M. Miller

The Pittsburgh Piping & Equipment Co. was chosen as the vendor of the subassemblies of the 18-in. primary coolant piping.

7. Steam Generators (55b32) J. I. Spiegel

To facilitate fabrication and improve the delivery date of the steam generator heat exchanger, the Babcock & Wilcox Co. has revised the fabrication procedure of the heat exchanger elbow, simplifying the assembly and welding

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sequences. Drawings showing the revised heat exchanger elbow were received and distributed for information to all parties concerned. Fabrication of the heat exchanger continued during this design change.

Heat exchanger hand hole diaphragms and gaskets are being made interchangeable between the Foster Wheeler Corp. and Babcock & Wilcox Co. steam generators. Drawings incorporating this feature are being prepared by the Babcock & Wilcox Co. and were previously received from the Foster Wheeler Corp.

All Babcock & Wilcox Co. welding and fabricating procedures were received and approved, except the following:

<u>Item</u>	<u>Status</u>
a. Transition welding procedure	A new welding procedure using 25-20 Cr-Ni electrodes is being prepared. Samples promised for the week of November 7.
b. Stress relieving procedure for the heat exchanger	Received and reviewed by Bettis Plant personnel. Maximum temperature differentials and rates of heating and cooling are being added to the procedure.
c. Tube weld samples	Promised for week of November 7.

Revised drawings of the steam generator outline and assembly, steam drum details and internal parts were received from the Foster Wheeler Corp. and distributed for information to all parties concerned.

8. Reactor Plant Auxiliary Pressure Vessels (55b33) C. E. Sinclair

Pressurizing Tank

Approval of the plan list and outline drawing for the PWR pressurizing tank was received from the Atomic Energy Commission (the AEC). Detailed drawings are now being prepared by the Foster Wheeler Corp.

Bids were received for the pressurizing tank replaceable heaters on October 18, and a recommendation was made to place the order with the low bidder.

The transition sleeve tests to be conducted in Vessel No. 2 were delayed while awaiting delivery of a vessel head closure ring. The examination conducted on the head removed from Vessel No. 1 revealed that a crack was formed in only one of the transition sleeve designs tested. This sleeve was one fabricated with an air gap between the sleeve and the carbon steel portion of the clad vessel wall in order to eliminate contact between the carbon steel and the stainless steel transition sleeve. The crack appeared in the sharp corner formed by the

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sleeve at the point where the sleeve was recessed to form the air gap. This sleeve design will not be considered for future use. No damage, corrosion, or injurious effects were detected in the remaining transition sleeves which consisted simply of a straight sleeve welded into the vessel wall. This design transition sleeve is considered to be acceptable as a means of introducing stainless steel heater wells into the pressurizer wall and will be used as the basis for PWR pressurizer design. The 6-mo test to be conducted in Vessel No. 2 will obtain further information on transition sleeve designs.

Coolant Purification System Demineralizer

The plan list and outline drawing for the PWR Coolant Purification System demineralizer were received from the Alloy Manufacturing Co., Inc. and were forwarded to the AEC for approval.

The outline drawing for this demineralizer specifies the use of a replaceable resin retention element which is designed to utilize the principle of edge-type filtration, although the closure design used on this vessel permits use of other types of retention units. A sample unit will be obtained and tested at the Bettis Plant to establish the acceptability of this type resin-retention element for demineralizer use. The investigation being conducted by Bettis Plant personnel to evaluate all promising replaceable resin retention element designs is being continued.

Chemical Addition System Pressure Vessels

Plan lists and outline drawings for the hydrogen injection vessel and the hydrazine addition vessel were received from the Pittsburgh Piping & Equipment Co. on October 3 and are being reviewed by Bettis Plant personnel before submission to the AEC.

Primary Charging System Resin Storage Tank

Bids on the resin storage tank were received on October 19 and are being evaluated by Bettis Plant personnel. The design of the resin storage tank was changed from a single portable storage tank to individual tanks for each of the 2 demineralizers. This modification was necessitated as a result of a plant arrangement study which revealed that too many difficulties would be encountered in transporting a single portable resin storage tank from one demineralizer to the other.

Primary Charging System Water Storage Tank

As a result of an arrangement study conducted by the Stone & Webster Engineering Corporation, the orientation of the water storage tank on the site was changed. This component will now become an elevated tank instead of a reservoir as originally planned. E-Spec 565434 for this component was revised to incorporate this change as well as comments received from the AEC in their approval letter. The revised equipment specification was sent to vendors for

quotations which are scheduled to be received on November 20.

Valve Operating System Pressure Vessels

Outline and assembly drawings for the Valve Operating System 3000 psig air bottles and 3000 psig water flasks were received and are being reviewed by Bettis Plant personnel. These drawings were submitted to the AEC October 19 for review and approval.

Valve Operating System Air Vent Muffler

The contract for the air vent muffler was awarded to the Burgess-Manning Co.

Filters and Vent Capillary

The bids for the 2-in. 3000 psig Valve Operating System strainer were evaluated and technical recommendations were made for awarding a contract to furnish these strainers.

Coolant Discharge and Vent System Pressure Vessels

The equipment specifications for the flash and blowoff tanks were approved by the AEC with comments and were sent to vendors for bids. Bids on these vessels were received on October 19 and are being reviewed by Bettis Plant personnel.

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

Regenerative and Non-Regenerative Heat Exchangers

Release from the Atomic Energy Commission (the AEC) to purchase the regenerative and non-regenerative heat exchangers from Alco Products, Inc., was received on August 29. Review of their preliminary designs indicated that difficulty would be encountered in building these heat exchangers to Westinghouse process specification. This design would not permit a 2-pass strength weld of tubes to tube sheet having a leakage path length equivalent to 150% of the metal thickness. Before the purchase order was placed on September 22, the vendor redesigned the equipment by using 3/8 in. tubes with a larger ligament to conform with Bettis Plant tube welding requirements.

Request for approval of outline drawings of both sets of exchangers, which were revised by the vendor to include Stone & Webster Corp. comments, was submitted to the AEC on October 3. Approval of the Bills of Material and Plan Lists of both sets of exchangers was requested from the AEC on October 5. A release to procure material was received from the AEC on October 13 and given to Alco Products, Inc., on the same date.

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10. Control Mechanism - Design and Test (56c13) J. J. Gertz

In order to reach a decision on the production mechanism design for PWR Core 1, a parallel development of a fixed roller nut mechanism without a kicker device, and a 2-arm collapsible rotor mechanism similar to that used on STR Core 3A, will be built. These mechanisms were chosen instead of the fixed roller nut design with the kicker developed on project 56c19, primarily because of their basic simplicity. A technical supporting document will be submitted by December 1 in which a final mechanism design will be recommended based on layouts and analyses in conjunction with continued feasibility prototype tests performed under project 56c19.

11. Control Mechanism - Component Development (56c15) T. L. Weybrew

Magnetic Brake

A fourth test was completed in the series aimed at developing a suitable magnetic brake to hold a control rod in position when power to the motor is shut off. The brake shoes for this test had only one pad per shoe compared with 2 pads per shoe for the first 3 tests. The brake shoe material was sintered type 316 stainless steel, the same material used in the first test. The minimum torque available to resist motion of the control rod with the brake shoes against the type 410 stainless steel housing wall was lower than that for any of the previous tests, and only 75% of that found for the first test, indicating that the single-pad design is a less effective brake. A fifth test using sandblasted type 347 stainless steel brake shoes of the double-pad type is in progress.

Roller Nut

A third test was completed in the series to determine the performance of a roller nut and screw of increased size applicable to a PWR control drive mechanism. This test was the same as the first test in this series except that the outer races of the rollers were 17-4 PH stainless steel instead of Stellite 19. The water temperature was 200°F and the screw material was 17-4 PH stainless steel. The results show that the wear of the 17-4 PH stainless steel rollers was approximately 4 times that of the Stellite 19 rollers. A fourth and final test of this series will be run with the same materials and under the same conditions as the third test except the temperature will be 500°F instead of 200°F, in order to duplicate the environment of the second test.

12. Control Mechanism - Prototype Development (56c19) T. L. Weybrew

The second life test of mechanism type XRN-10A (fixed roller nut) was successfully run with no scram failures. This test was run with alternating periods in hydrogenated and oxygenated water environment. This second test is the same as the first life test except that all of the wearing parts were replaced and the test rig was modified to eliminate possible sources of friction. Testing is being continued for a prolonged period to extend the knowledge of the operating characteristics of this mechanism.

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All parts of mechanism type XRN-21A (fixed roller nut with kicker) were received and preliminary tests in air were conducted, after which the mechanism was assembled in an autoclave for tests under simulated operating conditions.

An assembly of mechanism type XRN-20A (expanding roller nut with separate scram magnet) for the simulated operational test is now in process.

A duplication of the life test of the XRN-10A is planned, starting with hydrogenated water.

Detail drawings for mechanism type XRN-19B (2-arm collapsible rotor) were completed.

13. Reactor Plant Electrical Components (56dl2) G. E. Seguin

Pump Speed Changing Switches

These four 2400 v motor-operated disconnect switches are located in the auxiliary power room and provide 2-speed operation of the main coolant pumps. E-Spec 565710 describes these switches and was sent out for bids.

Battery Distribution Switchboard

This distribution board, located in the auxiliary power room, is used to transfer d-c control circuits between the 2 batteries. E-Spec 565718 describing this switchboard was submitted to the Atomic Energy Commission (the AEC) for approval.

Emergency Lighting Transfer Panel

The lighting panel located in the auxiliary power room, and used to transfer emergency lighting circuits from the normal supply to the emergency lighting battery, is described in E-Spec 565831. This equipment specification was submitted to the AEC for approval.

Rod Drive MG Sets

These 3 MG sets are located in the auxiliary power room and provide the d-c supply for the rod drive motor inverters. E-Spec 565909 describing these MG sets was submitted to the AEC for approval.

14. PWR Non-Nuclear Instrumentation (56dl3) C. J. Madsen

As a result of a meeting held October 7 with the Atomic Energy Commission (the AEC), the following specifications are being revised:

E-565581 - General Specification for Primary Plant Instrumentation

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- E-565582 - T_c and T_h Temperature Transmitters
- E-565583 - Reactor Power & Temperature Control System
- E-565584 - Calibrating Thermometer Element & Temperature Interlock
- E-565585 - Multipoint Temperature Recorder
- E-565586 - Pressurizer Water Temperature Recorder
- E-565587 - Pressurizer Steam Temperature Test Element
- E-565599 - Multipoint Temperature Indicator

- E-565588 - Main Circulating Pump D/P Indicator
- E-565589 - Steam Generator D/P Indicator
- E-565590 - Reactor D/P Indicator
- E-565591 - Primary Plant Static Pressure Indicator
- E-565592 - Pressurizer Pressure Indicator
- E-565597 - Flash Tank Wide Range Pressure Indicator
- E-565598 - Blow-Off Tank Wide Range Pressure Indicator
- E-565800 - Flash Tank Pressure Indicator (Low Range)
- E-565801 - Blow-Off Tank Pressure Indicator (Low Range)
- E-565802 - Pressurizer Static Pressure Controller

- E-565593 - Primary Coolant Flow Transmitters & Receivers
- E-565594 - Pressurizer Liquid Level Recorder (Uncompensated)
- E-565595 - Flash Tank Liquid Level Recorder
- E-565596 - Blow-Off Tank Liquid Level Recorder

The following specifications were submitted to the AEC and were approved with comments:

- E-565824-B - Level Indicator - Storage Tank Charging System
- E-565825-B - Level Indicator - Hydrogen Injection Flask - Chemical Addition System
- E-565826-B - Pressure Indicator - Hydrogen Injection Flask - Chemical Addition System
- E-565823-B - Gage Glass - Flash Tank
- E-565827-B - Gage Glass - Blow Off Tank

The following specifications were submitted to the AEC for approval:

- E-565336-B - Pressure Indicator - Storage Tank Charging System
- E-565837-B - Conductivity Meter

Twelve specifications from Stone & Webster Engineering Corp. are being circulated for internal review.

Specifications for a total of 20 remaining items are in process of preparation.

A model of a rod position indicator for PWR was built. This indicator uses multiple transformers, each secondary operating an indicator light directly. Preliminary tests indicate that it operates satisfactorily. Further

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tests will be made on it in conjunction with the PWR prototype rod drive mechanism.

15. PWR Nuclear Instrumentation (56d14) C. J. Madsen

E-Spec 565904-A on the PWR nuclear instrumentation was approved by the Atomic Energy Commission (the AEC) with comments on September 23. These comments were resolved and submitted to the AEC for concurrence. The prospective bidders were furnished with copies of the revised specification for use with their formal quotations which are due November 1.

16. Radiation Monitoring Equipment (56d15) C. J. Madsen

Operational Radiation Monitoring Equipment

E-Spec 565832-A for the operational radiation monitoring system was submitted to the Atomic Energy Commission for approval. The specification was also supplied as advance information to the prospective bidders. When the specification is approved the vendors will be requested to submit formal quotations.

Safety Radiation Monitoring Equipment

A study of the system description is being made to ascertain whether the equipment should be bought as a package, whether individual specifications should be written for each item of equipment and quotations requested on each item, or whether some items should have specifications written while others may be bought from catalogs.

C. Reactor and Auxiliaries

1. Mechanical Design of Reactor Core (53a12) P. A. Halpine

As a result of discussions held between Bettis and AEC representatives October 18, the core design efforts are now being applied to the development of a single pass expanded core design with 0.070 in. fuel plates and channels in the seed and 4 regions of differential orificing in the blanket. This design contains 32 seed clusters with 32 control rods and 113 blanket assemblies, or a total of 145 units. The following are items covered during this period:

Blanket Assembly

A report presenting the design concept of the PWR blanket assembly was submitted to the Atomic Energy Commission (the AEC) for information on September 30. The design, as described therein, is shown in Figs. 2 and 3. Efforts are continuing to simplify the design and to reduce Zircaloy requirements.

The major changes from the original reference design are:

- a. A shell fabricated entirely of Zircaloy, which simplifies both the design and the assembly procedure.
- b. The elimination of the top transition, to permit access to fuel bundles without damaging the shell and to eliminate Zircaloy welding after assembly of the bundles and shell.
- c. Placing the latch spring at the top of the assembly, to assure proper engagement of the fuel element failure detection piping after differential expansion between the core and cage structure.
- d. Incorporation of fuel element failure detection and location system components into the design.

The assembly shell thickness has been tentatively established as 0.148 in. Shell deflections occurring during four-loop operation are now being evaluated from a hydraulic viewpoint.

Seed Clusters

Maximum stresses resulting from the combined action of pressure and thermal loads in the seed subassemblies are being investigated for 60 Mw and 100 Mw power production.

Methods of employing subassemblies fabricated by resistance-welding thin bars to the edges of flat fuel plates are being investigated.

Fig. 2 - Blanket Assembly in Core Structure

Fig. 3 - Blanket Assembly and Details

Core Cage

The cage structure is well established in concept. Work is progressing on the following items in firming the design:

Plate Thickness - To estimate, by analytical methods, the operational deflections and stresses in the support plate and barrel and establish plate thickness.

Fastenings - To evaluate possible methods of joining the support plates to the cage barrel.

FEDAL (Failed Element Detection and Location) Piping - To establish a reference design integrating the system piping with the cage structure.

Control Rods and Shrouds

Work has been initiated on designs employing control rods with about 3 ft tails. Use of such rods would affect the design of both seed clusters and core cage. Two general design approaches are being investigated: rods of about a 3 3/8 in. span, removable through the mechanism port; and rods of about a 4 1/4 in. span, removable through the refueling ports as a part of the cluster.

A control rod of 4 1/4 in. span will require a shroud diameter of about 5 in.

The head penetration for a mechanism is 4 in. An effort is currently being made to establish a design which will provide adequate shrouding for a 4 1/4 in. span rod and which will be consistent with through-the-head refueling requirements.

2. Mechanical Testing of Reactor Components (53a13) P. A. Halpine

The program of mechanical tests has progressed in accordance with the proposed schedule, which was submitted to the Atomic Energy Commission (the AEC) by letter dated September 7.

During this period, manufacturing drawings of the scale model core cage components have been completed and released. Manufacture of components for the mechanical feasibility demonstration assembly was started. Equipment specifications for the vessel required in the full scale component hot loop and full scale component cold loop have been prepared and issued. Work on the layouts of these loops has begun.

3. Reactor Preliminary Design (53a14) L. S. Mims

Core 2 Design Study

Tentative major design objectives for PWR Core 2 were submitted to the Atomic Energy Commission (the AEC) by letter dated October 5. These

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objectives are intended to define the scope of further study prior to establishing final design objectives by April 1, 1956. They are listed below:

- a. Reduction in core cost by use of less expensive cladding, structural materials and improved manufacturing techniques.

The materials and fabrication techniques now used in core construction are quite expensive and result in a fuel cost unacceptable in comparison with a conventional heat source. The use of stainless steel, aluminum and other relatively inexpensive metals would tend to bring nuclear fuel costs to a more acceptable level. Manufacturing and inspection techniques applied to materials now used should be reviewed for possibilities of cost reduction by adopting different processes.

- b. Extended core life at full power operation.

Extending core life at full power operation will significantly reduce the cost of fuel inventory, lengthen the reprocessing cycle and reduce downtime of the plant. It appears desirable to double the Core 1 design objective.

- c. Increased power output per ton of uranium in the core.

Energy production costs could be reduced by using the most economical combination of fuel enrichment. Advances under this objective could be made by the use of either a seed and blanket type core or a slightly enriched core.

- d. Greater power plant maneuverability by providing reactivity for xenon override.

A basic limitation on nuclear power plant performance is flexibility with regard to plant utilization. An effort must be made to provide flexibility for plant maneuvering in response to utility system loads. The degree of maneuverability required to make the plant an effective component of a utility system must be balanced against the increased cost involved in order to remain within the general framework of more economical power.

- e. One hundred (100) Mw gross electrical output with 4 loops operating.

The PWR turbine generator plant is of 100 Mw design and to confirm nuclear power as a reasonable heat source for utilities, it is considered essential to produce a core which can provide sufficient power to operate the plant at full capacity. A 100 Mw core could noticeably reduce the operational costs which are now quite high in comparison with conventional plants.

4. Reactor Specifications (53a15) L. S. MimsMinimum Pressure in Reactor Vessel

Allowable pressure variation studies have been re-examined in the light of the new flux factors. This was done for each of the four reactors now under consideration. These studies were for both 3 and 4 loop operation and assumed the reactor did not suffer a loss-of-flow accident or power surge during the low pressure condition. The minimum pressures for this transient were of the order of 1500 psia.

Chemical Shutdown System

Studies have been completed on the requirements of the chemical shutdown system. These studies have indicated the amount of boron required for cold shutdown if various numbers of rods should stick in their most reactive position. (See following table.) A maximum injection rate of approximately 0.07 lb natural boron/hr - ft³ H₂O has been determined using the assumption that the rods were overriding peak xenon and refused to go back in to follow the xenon burnout. The cleanup requirements were determined considering that any residual poison left in the reactor when full power operation is desired will adversely affect the power ratio.

Emergency Shutdown Poison Requirement

<u>No. of Rods Stuck</u>	<u>Lb of Natural Boron per ft³ of System H₂O</u>
24	0.382
16	0.220
8	0.087
1	0.000

5. Reactor Core Instrumentation (53a16) L. S. MimsFuel Element Failure Detection and Location

The equipment necessary to evaluate the delayed neutron monitoring station has been incorporated into the failure test loop for oxide rods (WAPD-13-6 experiment) installed in the HB-5 facility of the MTR. Also installed on the WAPD-13-6 experiment is the revised gamma monitoring station. The WAPD-13-6 experiment will provide a PWR oxide rod failure signal for direct comparison of the gamma monitoring station with the delayed neutron monitoring station under PWR failure conditions.

The decision was made to discontinue further development work on the gaseous fission product monitor (GFPM), as reported in Chalk River Project Report RDP-51 dated April 16, 1953. The decision was based on the engineering complications involved with incorporating the GFPM into a high temperature and pressure system and the disadvantage, in a recycled system,

of the GFPM sensitivity to long-lived fission products. This sensitivity in a recycled system results in saturation of the monitor (GFPM) activity reading for a considerable time following a failure, even after the failure has been removed from the reactor.

General Reactor Instrumentation

The list of PWR reactor instrumentation, with its required plant container penetrations and read-out instruments, was established this month and submitted to the Atomic Energy Commission (the AEC) for approval. A brief mechanical description and instrument location was attached to the list.

In order to identify the problems associated with development of the small reactor instrument discussed in WAPD-MRP-56 to measure neutron flux, flow, and water temperature, a design study model was prepared. During the preparation of the model, appropriate heat transfer analyses were made to establish the relative sizes of the various components. As a result of this work, consideration is being given to whether the development of this unit should be subcontracted or developed at the Bettis Plant. An attempt is being made to reach a decision by November 1.

BeO Loading in the Reactor

Preliminary calculations are being made of the gamma and neutron heating in the 30 lb of BeO, under consideration as a photoneutron source for the PWR Core 1. These calculations are in progress. The various methods reported in the previous Technical Progress Report, WAPD-MRP-56, for incorporating the 30 lb of BeO will be evaluated against the calculated cooling requirements.

6. Radioactive Materials Handling & Viewing Equipment (53dl1) P. A. Halpine
and
7. Refueling Equipment (53dl2) P. A. Halpine

During this period, the effort in these projects was directed toward initiating procurement of the important long lead time items and making feasibility studies.

A list was made of thirty different equipment items required in these two projects, and shop orders for these items were issued. The list is as follows:

- Subassembly Handling Crane with Extraction Tool
- Transfer Crane
- Core Hoisting Rigging
- Subassembly Cut-off Machine
- Spent Fuel Shipping Containers
- Used-Subassembly Shipping Container
- Head Storage Stand
- Core Assembly Stand (Clean Room)
- Subassembly Test Stand
- Core Support Ring (Deep Pit)

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- Source Handling and Storage Coffin
- Container for Damaged Subassembly
- Control Mechanism Handling and Storage Equipment
- Radioactive Scrap Press
- Radioactive Scrap Shipping Containers
- Used Subassembly Storage Racks
- Closure Seal Weld Cutting Machine
- Head Handling Yoke
- Closure Bolt Handling and Storage Equipment
- Reactor Pit Equipment
 - a. Base Plate
 - b. Subassembly Transfer Fixture
 - c. Shield Deck
- Underwater Optical Equipment
- Underwater Lighting
- Subassembly Handling Truck
- Core Hold-down Barrel Support Stand
- Core Spring Support Stand (Belleville Washer)
- Underwater Television Equipment
- Trellis for Mechanism Services
- General Purpose Manipulator
- Control Rod Exchange Tools
- Control Rods for Storage and Shipping

This list is believed to be complete, although some items may be changed if the core design changes from the present reference design.

FR-0276, PWR Viewing System for Radioactive Materials, was approved internally and by Duquesne Light Company, and has been submitted to the Atomic Energy Commission (the AEC) for approval. A closed circuit television system is available in the Hot Laboratory of the Bettis Plant and will be given a test, simulating the radiation encountered inside the pressure vessel of the PWR reactor soon after shutdown. The vidicon tube of the camera will be exposed to a gamma radiation field of approximately 100,000 r/hr to determine if an acceptable image can be transmitted; lens browning and tube life will not be investigated in this test. The test will be run as soon as it can be worked into the Hot Laboratory schedule.

E-Spec 535741-B, for the subassembly handling crane with extraction tool, was approved by the AEC, subject to revision in accordance with their comments. The necessary revisions clarify the E-Spec and do not materially affect the equipment specified in it. The revisions will be made and issued as E-Spec 535741-C. Proposals have been received from five of the ten suppliers invited to submit proposals on E-Spec 535741-B; the other suppliers did not offer proposals. The proposals were evaluated, and the United Shoe Machinery Company was selected as the vendor. A subcontract is being negotiated on the basis of E-Spec 535741-C.

E-Spec 535423, PWR Closure Weld Cutting Machine, was completed and distributed internally and to Duquesne Light Company for comment.

I-C-6 & 7 (Contd.)

Design of the handling device and installation jig for the complete core will be postponed until firm dimensions are available for the 125-ton crane in the Fuel Handling Building.

Preliminary calculations indicate that cast iron may be superior to lead for shipping casks, both in cost and cooling capacity, when no external cooling system is used. The comparison of materials will be extended to several specific cask designs consistent with the reference design core.

8. Reactor Thermal and Hydraulic Analysis and Tests (55a13) L. S. Mims

Reactor Power

No combination of the design features reviewed in the previous progress report was sufficient to raise the power output of the single-pass reactor to the specified 60 Mw with 3 loop operation. Calculations showed that the two-pass design would exceed performance requirements, but at the expense of an excessive coolant velocity in the seed.

The effect of control rod programming on the power output of the reference single-pass core was determined. Basis of calculations was a set of two-dimensional radial flux plots obtained from UNIVAC. Two cases were studied: one in which eight rods were initially inserted and sixteen rods withdrawn; the other in which twelve rods were inserted and twelve withdrawn. Differences between the two cases were 0.5 Mw, but both showed significant improvements, e.g. 0.4 - 4.5 Mw over the case of uniform position control, although still only 55.5 - 56.0 Mw and, therefore, short of the 60 Mw design goal.

Pressure Vessel Flow Study

The detail drawings of the model to study flow through the pressure vessel and its internal components have been received from the Westinghouse Steam Division and approved by Bettis personnel. Notification of approval was sent to the Steam Division permitting release of the drawings to the model manufacturer in order to begin fabrication. The fabrication of the model is scheduled to be completed by January 10, 1956.

Orifices for Blanket Assemblies

A test program has been set up to determine the correct orifice design for the blanket assemblies. Since the blanket of the PWR core will be orificed differentially, a test program was set up to cover a wide range of interest. Therefore, the flow per assembly will be varied from 4×10^4 to 18×10^4 lb/hr requiring pressure drops across the orifice ranging from 60 to 30 psi. Data will be obtained covering the entire range of expected flows and pressure drops for various combinations of orifice arrangements and these data will guide the selection of orificing. It is expected that this test will be completed by February 29, 1956.

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Control Rod Thermal Distortion Test

Drawings necessary for the fabrication and evaluation of boron steel vs hafnium for use in Core 1 control rods have been completed. The drawings were released to the Fuel Element Division to provide test prototypes and to determine relative manufacturing difficulties of both materials. The test prototypes will be delivered to the Physics Department for a comparative study of both materials in the slab critical assembly. Upon completion of the comparative study, the control rods will be turned over to the PWR Analytical Design Section for thermal distortion tests.

The thermal distortion test will consist of heating the control rod in air by induction heating so that its temperature distribution will duplicate the worst expected temperature distribution during core life. Axial temperature distribution will be obtained by varying the axial spacing of the induction heater coils. Traverse temperature gradients will be obtained by making the heater coil and rod non-concentric. The distortions due to the temperature distribution will be measured by dial indicators.

Decay Heat Removal

The performance of decay heat removal with respect to protection of the core, pressure vessel and control rod mechanisms was studied. The decay heat cooling requirements of four reactor designs were investigated: a single-pass design of the reference and expanded cross-section, and a two-pass design of the reference and expanded cross-section.

It was found that the convective flow rate of 1% full flow supplied by the decay heat removal system permits bulk boiling in the exit of the hot channel for a period that starts 10 to 15 sec after the loss of flow and continues for a period of 300 to 500 sec. However, the reactor power has dropped during this period to the point where the ratio of burnout flux to maximum reactor flux is at the safe value of 7.5 to 1. Therefore, the 1% flow rate supplied by the decay heat removal system satisfactorily protects the core. The pressure vessel and the control rod mechanisms are also protected, since the water exiting from the average channel never reaches the quality region.

Gamma Heating in the Pressure Vessel and Thermal Shields

Gamma heating rates in the pressure vessel and thermal shields are necessary in determining coolant flow rate, contribution of heating in these structures to total heat, and in detail design, especially in the pressure vessel. An engineering "worst-case" study was made of gamma heating rates in the pressure vessel and thermal shields for 100 Mw gross plant output. Maximum and average rates were determined. The maximum values of 10,000 Btu/hr - ft² are well under the design limits of 35,000 Btu/hr - ft² for the pressure vessel.

Safeguard Studies

As part of the evaluation of the safeguard aspects of the PWR core, analyses were made of the core thermal characteristics in the event that the core is completely uncovered by loss of coolant from the system. These calculations were extended to establish the maximum rate of melt-down of the core under such circumstances that no cooling can be provided. These results were incorporated in the power plant evaluation of the vapor container characteristics under such a condition.

Temperature Distributions

Temperature patterns were calculated for reactor metal parts in which thermal stresses are of interest, specifically the core bottom support plate, which is heated by gamma rays; and the seed subassemblies, where a temperature differential is created because of the absence of uranium in the side plates.

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

The Pressure Vessel Review Group, consisting of one representative each from the Atomic Energy Commission (the AEC), Combustion Engineering, Inc., and Bettis Plant, held two meetings during this period. The purpose of this Committee is to coordinate the efforts of the organizations concerned in over-coming engineering and fabrication problems.

Examination of the first flange forgings at Bethlehem Steel Company disclosed an unsatisfactory material condition in the center of the material. This required the Bethlehem Company to discard four of the five forgings; the fifth forging had not progressed beyond the point of no return. This forging was salvaged by modifying the annealing process, and will be delivered about November 15. The ingots were immediately poured for the replacement forgings and delivery of the first replacement is expected early in December. No delay in the vessel delivery is anticipated due to this material problem if the replacement units are acceptable.

Machining of smaller parts for the closure seal development program has been made in the Chicago Plant of Combustion Engineering as the machinist strike at the Chattanooga Plant is still unsettled.

Although the plug welding repair of the cladding bond for the vessel bottom head plates was completed, it was agreed to wait for replacement material from the Lukens Steel Company. One plate of the required material was shipped from Lukens Steel Company; the remaining two plates will be made from heavier plates which have been ultrasonic tested and approved for bond quality. However, 3 in. of the material thickness will have to be removed by machining; these plates are to be machined and ready for shipment on October 26. The use of these replacement plates causes a delay of approximately 8 weeks in the fabrication of the bottom head. It is not presently known what effect this has on the final shipping date of the vessel.

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Strain gages have been attached to the PWR model vessel (1/4 scale) and the assembly is being readied for the first tests.

The photoelastic tests at Brown University have produced preliminary results of the first test, or cap test, which represents a limited portion of the vessel head with penetrations believed most critical. These preliminary results indicate the reference head design appears to be satisfactory from a pressure standpoint.

The head model has been machined on the inside and outside surfaces and penetration holes have been drilled. Preliminary results are expected in approximately 5 weeks.

PART IIPWR DEVELOPMENTA. Fuel Element Development1. Core Raw Materials and Shapes (53c21) B. R. PriceProperties of UO_2

Although it was reported previously in WAPD-MRP-56 that the reclaiming of rejected sintered compacts by crushing and recompacting was not feasible because of the resulting low resintered density, it has been decided to re-examine this problem. The original study was conducted on 100% recycle material of rather coarse size. There is reason to believe that by breaking up the rejected compacts to a very fine size and blending with virgin feed material, high density sintered compacts may be obtained.

Claddinga. Properties of Seamless Tubes

Further investigation of factors affecting the size of rupture in the cladding of water-logged elements has revealed two significant items. It had previously been established that rates of pressure rise in the order of 3000 psi/sec (the maximum which might prevail in a water-logged element in the PWR core) did not increase the size of the rupture beyond that experienced at nearly static rates. It has now been found that at rates of 10,000 psi/sec, the crack area of a ruptured fuel rod is approximately three times that experienced at near static rates. At this high rate of pressure rise, the bursting strength is about 15% greater than under static conditions. Variations in wall thickness of 8% do not cause a measurable difference in the size of rupture in reference fuel rods under static pressure. Seamless tubing is being procured on the basis of 5% maximum wall variation.

Creep tests within an environment of 2000 psi pressure of fuel rods containing different amounts of entrapped water have been initiated. Present tests have been limited to 700°F but an autoclave capable of operation at 1400°F is being procured.

b. Manufacture of Seamless Tubes

Both Bridgeport Brass Company and Wolverine Tube Division have delivered lots of 200 to 300 ft of reference tubing. The preliminary evaluation of this material indicates that the tubing from Wolverine Tube has wide variations in wall thickness and a poor control of the ID. Some of the tubing received from Bridgeport Brass shows cracking with the dimensional check not yet completed. Other tubing from another ingot was not cracked.

Nuclear Metals Inc (NMI) has begun production of Zircaloy-2 tubing for core manufacture. Preliminary reports indicated difficulties in holding dimension, however, the supplier feels that good dimensional control can be maintained by sizing some of the existing tubing. They have now taken steps to make proper inspection of their product. NMI has completed all of their development work including manufacture of 20 ft each of Zircaloy-3A, -B, and -C.

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Material received from Superior Tube Company has been examined metallurgically and compared with the cracked tubing made from the same ingot by Bridgeport Brass. Current microscopic evidence indicates that excessive amounts of cumulative cold work may cause progressive micro damages in areas adjacent to elongated massive precipitate. This precipitate is believed to contribute to cracked tubing. Efforts are now being made to get small extrusions from Wolverine Tube for processing by Superior Tube to final dimensions.

c. Closed Tube Development

A tentative process has been evolved at the Bishop and Babcock Mfg. Company which is expected to provide preliminary evaluation of parts by November 1.

2. Non-Destructive Testing (53c22) J. G. Christ

All major ultrasonic equipment, except the tube traversing equipment, has been received. The tank received from Sperry Products, Inc. for submerged ultrasonic testing is being modified to incorporate a tube driving mechanism.

Work on ultrasonic inspection for end cap weld and tubing flaws and wall thickness continues. The methods look promising but applicability to production testing cannot be demonstrated until the positioning and traversing equipment are in operation.

Eddy-current inspection of tubing appears sensitive to processing variables. To date, the Probalog has given the most consistent indication of flaws and subsequent pressure test failures and will be used for production work.

Zircaloy Tubing

a. Surface Flaws

An immersed, 2-crystal, ultrasonic technique has been successfully used to detect both internal and external longitudinal flaws in Zircaloy tubing. Adaption of the method to production testing will be tried November 15 when the tube conveyor is received.

No correlation has been found between Cyclograph traces run at 20, 100, and 200 kc and local wall thickness variations, grain size, residual cold work, or hardness. A correlation appears to exist, however, between the average level of the Bristol recording at 20 kc and the average wall thickness. In addition, it has not been possible to locate a high percentage of obvious flaws (defects through the wall) or predict pressure test failure with any of the Cyclograph tests.

Checking of tubing under hydrostatic pressure with the Cyclograph and annular coil indicates that readings can be changed approximately 20% at 6000 psi. The changes follow no predictable pattern, but do demonstrate that stress or attendant elastic dimensional changes affect the results of eddy-current tests.

Eddy-current evaluation of tubing from Nuclear Metals, Inc., Bridgeport Brass Company, Wolverine Tube Division, and Superior Tube Company indicates a considerable variation of trace characteristics attributable to

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processing differences. Pip heights corresponding to rejectable flaws for one process or processor do not necessarily indicate rejectable flaws for tubing made by a different process or processor.

b. Tube Wall Thickness

Calibration of the Vidigage for Zircaloy tubing is under way. The instrument is being checked against micrometer and optical comparator readings and new scales will be prepared. This instrument appears most promising for production use in measuring wall thickness. Final evaluation of the equipment for production inspection will be initiated November 15 when the tube conveying equipment is received.

End Closures

a. Porosity and Through Cracks

The process specification for helium leak testing of PWR fuel elements has been submitted to the Atomic Energy Commission for approval. Refinements of the specification, such as decreasing operating pressures and testing times, are being investigated.

b. General Weld Quality

Recent receipt of the ultrasonic scanner and tank will permit a continuation of end closure studies using the shear wave technique. To supplement this work, end welds are being investigated using longitudinal waves normal to the end cap. The methods are to be compared for ultimate selection of a test procedure.

Microradiography has revealed that weld porosity may be correlated with microporosity or other density variations in tubing. Such density differences have been observed in lots of tubing received from different processors.

A specification for the radiographic examination of PWR fuel rod end closures has been drafted. The technique to be used involves two 90° exposures of welds with the welded tube inserted into a drilled Zircaloy block to provide a uniform section for radiography. The technique will show spherical voids as small as 8 mils in diameter.

3. Fuel Rod Process Development (53c23) J. G. Christ

End Closures

Two methods of end closure welding are being developed for production use. These methods are inert chamber arc welding and radiofrequency induction welding. Percussive welding is being developed as a follow-up effort.

a. Arc Fusion Welding of Step Down End Plug

During this period several improvements were made on the welding chamber as an approach to a production facility. A quick-change collet and more powerful instant speed drive motor were installed to provide more rapid

II-A-3 (Cont'd)

handling of the work and truer, smoother rotation during welding. High frequency arc initiation was improved when a high initial welding current was used and then reduced manually to the normal level for the remainder of the weld. An automatic timer and sequencing mechanism incorporating this feature was installed on the welder. This device also tends to eliminate crater cracks.

Thus far the maximum number of welds made in one chamber loading and one pump down was 134 which required 2.5 hr. One hundred and two test specimens have been placed in corrosion tests which will provide data on 204 fusion welds. A preliminary process specification for fusion welding PWR fuel rod end closures has been completed.

Radiographic and metallographic examinations have shown some porosity in the welds. The use of a second pass has been ineffective in eliminating the porosity. The use of lower currents at slower rotational speeds in order to maintain the same energy input is now being investigated. This should also reduce undercutting.

X-rays have shown cracked pellets next to the end caps on fuel rods where reference internal axial clearances of 0.070 to 0.090 in. were used. Studies are under way to determine affect of end clearance on pellet cracking.

b. Radiofrequency Induction Welding

Several fixtures which allow rotation of the work during welding have been completed and tried out with considerable success. The major effort has been concentrated on making a number of reproducible welds on press-fit, plug type end closures with reference (0.156 in.) plug lengths, for mechanical property tests at Bettis Plant. When proper control is maintained on all variables a uniform weld bead approximately 0.050 in. wide and free of harmful oxidation and undercutting is produced without using an elaborate atmosphere chamber. This method of welding has also been successfully used on other types of end closure designs. One of these, the butt welded closure, requires careful control of the upsetting pressure which can be obtained on the rf equipment.

Several of the end closures on fuel filled tubes were deliberately made with more than adequate penetration to evaluate the fuel contamination of the weld. No evidence of weld contamination was found in any of the test samples. Radiographic, autoradiographic, and metallographic methods of investigation were used.

c. Percussive Welding

The testing and rewiring of the electrical system has been completed. The machine has been assembled and some test welds have been made. These welds were made with solid copper stock of 0.418-in. diam. The welds were strong enough when properly made to cause the resulting rod to tear diagonally through the stock along shear planes during a bending test.

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Alignment of the parts on a common axis of motion is of paramount importance in view of the thinness of the tube wall to be welded. Since the amount of burnoff and/or upset is expected to be small, the faces to be welded must be parallel to begin with to prevent arcing from being concentrated along one side of the mating faces. Present work is being concentrated on improvement of mechanical features of the machine.

End Closure Evaluation

Rotating cantilever beam tests have been continued on reference arc fusion welded end closures, rf fusion welded end closures, and others of special design.

End closures welded by radiofrequency have been shown to be comparable to similar samples made by fusion welding. Static tension tests and other correlating data will be obtained from representative samples of tubing used for fatigue tests to improve the accuracy of direct quantitative comparison. Bending fatigue tests on these end closures have demonstrated that internal notches are more damaging than external notches. The internal notches are sharp and caused by inadequate weld penetration whereas the external notches are caused by weld undercutting and are not as severe.

Repeated blow tension impact tests have been discontinued in favor of tension fatigue tests. There is some indication as a result of specimens tested, that this method of stress is more sensitive to structural notches than is the bending fatigue. Tests on stress-relieved samples are in process to verify this observation.

In the previous progress report (WAPD-MRP-56), 500 end closures were reported as having been made by the arc fusion welding method. A limited evaluation is given below as requested by the Naval Reactors Branch. No leaks were found by the helium leak test. A limited number of welds made on oxide filled tubes resulted in an approximately 50% shattered oxide pellets nearest the weld, when reference axial clearances were used. Radiographic inspection by an improved technique shows extensive porosity in 75% of the welds. Two hundred and twenty-four of the end closures were corrosion tested for 14 days in 750°F, 2000 psi steam. Only 2 welds showed a small white spot and none of the welds showed excessive weight gains that would be indicative of a leak. The corrosion tests are continuing. Work is in progress to reduce the extent of porosity.

PWR Seed Development (F. R. Lorenz)

Fuel Plate Fabrication

Sixteen dummy fuel plates were rolled during this period using variations in the cleaning procedures, assembly welding processes, and rolling temperature, and reductions in efforts to eliminate the intermittent bonding which has been obtained between the flange and plate. Metallographic inspection showed that a good metallurgical bond was not produced and further efforts are being made to remedy this situation.

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Initial attempts to fabricate fuel plate flange stock (trapezoidal cross-section) by rolling through a Turk's head having one angular rolling surface have been completed. It appears that flange stock can be fabricated in this manner when rectangular starting stock is used and a cross section reduction of 15% is made. The problem in fabricating flange stock by rolling through a Turk's head is in straightening the drawn stock.

Zircaloy-3 type material for fabricating fuel plates has been received. Different cladding component rolling techniques will be employed with this material.

Only dummy fuel plates have been fabricated for the initial development work. Ten 40-lb Zircaloy-B fuel alloy ingots have been melted and surface conditioned to provide fuel components for future work. Two of these ingots have been forged and rolled out of a salt bath to provide fuel alloy fillers with as-rolled surfaces and machined surfaces for development work.

Subassembly and Cluster Fabrication

Ten fuel plates, sectioned from those rolled during this period, were etched and assembled and welded into a 16-in. long subassembly in order to obtain weld penetration data for the seed design. Over-penetration was obtained, and additional subassemblies are being prepared for similar experiments.

A subassembly was prepared using flat Zircaloy-2 plates with spacing strips along the edges and copper for channel spacers. After welding, the copper was etched out. Measurements showed an average of 0.089 in. with maximum variation of ± 0.001 in. along a 12-in. length. A similar subassembly 4 ft in length will be completed during the next period.

Tests have been made in an attempt to weld sections of short STR subassemblies together to make the longer length subassembly necessary for the PWR seed. The first series of tests has shown that it is possible to get a satisfactory weld with little distortion. To determine the exact amount of distortion from such a process it will be necessary to weld together two full length sections. Further tests are being made to improve the weld and to determine the effect of different methods of machining the subassembly ends upon welding conditions.

Inspection procedures are being reviewed to ascertain what has to be done to adequately inspect the first full length welded seed units and to determine what new procedures and techniques must be developed.

Fuel Rod Process Development (B. R. Price)

Fuel Component

a. Compaction on Stokes Automatic Press

It has definitely been established that sintering time can be shortened from 10 to 5 hr at 1725°C if 125 tsi, rather than 100 tsi, is

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II-A-3 (Cont'd)

used for the compacting pressure. Sound pellets have been made at rates up to 26 per min with this pressure. It is necessary, however, to add 0.2 w/o Sterotex to the powder mix to allow this higher load to be used.

Complete process specifications and product specifications for fuel components for Core 1 blanket fuel rods have been issued. The reference process employs compaction on automatic presses.

b. Vibratory Compaction

Limited experimental work in this period produced a 1-in. UO_2 compact having a sintered density (10 hr at 1725°C) of 89% theoretical. The long-range potential of vibratory compaction appears less promising as a method for producing integral length fuel cores than does the extrusion process; consequently, it has been decided to terminate experimental work on the former. A final report on vibratory compaction will be issued.

c. Extrusion

Centerless ground integral UO_2 fuel cores with sintered densities of 90% theoretical and lengths of 5-in. were produced by extrusion. Ball-milled mixtures of MCW- UO_2 powder, together with 1 w/o Methocel, 0.1 w/o Sterotex and 0.2 w/o TiO_2 were employed as the raw materials. Concentrated effort is being placed on the development of manipulative techniques which will result in nonporous, homogeneous extrusions.

Sintering Studies

Long term (up to 88 hr) sintering of UO_2 at 1725°C indicates that the rate of densification is 0.05% per hr after the first 20 hrs for compacts pressed at 100 tsi.

Seven lots of MCW- UO_2 powder (Lots No. 24 to 30, inclusive) were tested to determine fabricability. Six of the 7 lots fell short of the nominal 92% average density after being compacted with 125 tsi pressure and sintered for 5 hr at 1725°C . It is estimated that an additional 3 to 5 hr would raise the average density to 92% theoretical.

4. Fuel Assembly Process Development (53c24) J. G. Christ

Fuel Rod Bundle Development

One full size all-Zircaloy rod bundle has been completely processed through corrosion testing, a second has been welded and machined, a third is being sectioned for study of weld penetration, and a fourth is being prepared for welding. Space measurements are being made after each step in the process to determine the pattern and extent of processing distortions. The use of strip spacers for control of spacing tolerances is being evaluated.

II-A-4 (Cont'd)

A study to determine the current speed-time parameters needed to achieve full penetration is being made on single rod weld specimens. To date, 50 specimens have been evaluated. Other parameters being studied in this program are the diameter of electrode rotation and clamping of the underside of the tube sheet hole.

Adaptation of the electrode rotating mechanism to the production welding chamber has begun. Detailing of all components for the production chamber is being completed.

Fuel Rod Bundle Evaluation

Results have been obtained on the first test of a program to compare the welded with a mechanically assembled bundle. One of each type was heated to 1350°F and air cooled. These bundles consisted of 24 solid Zircaloy rods surrounding one fuel rod. The fuel rod in the welded bundle buckled more severely, but the spacing distortions were distributed between neighboring rods. The fuel rod in the mechanical bundle showed more end growth and less buckling but the spacing distortions were localized around the distorted rod. A report on this test is being prepared.

Bundles are in process to compare the behavior of the loose and rigid assemblies containing all fuel rods with reference axial fuel clearances in thermal cycling.

Shell Assembly Development

A report on hot bending of Zircaloy-2 was released during this period. The forming of final shapes for the shell assembly must await confirmation of the final design. Operations being developed for the shell assembly are the hot forming of channels, the hot deep drawing of transitions from square to round and the forging of transitions.

The rf resistance welding will be evaluated for the making of structural welds in Zircaloy-2 without an inert atmosphere.

5. Fuel Element Testing and Evaluation (53c25) J. G. Christ

In-Pile Tests

The PWR in-pile test program was designed to test fuel elements and small assemblies of fuel elements under simulated service conditions.

a. WAPD-CR-5

The six month test period on two fuel rod bundles began with the start-up of the Chalk River Reactor on September 9, 1955. The test has been operating satisfactorily since this date except that the average heat flux in the test is 45 percent of the desired value. This is due to a lower effective neutron flux than was originally estimated.

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b. WAPD-MTR-15-1

The progressive failure test bundle of 9 UO₂ filled rods was sectioned in the Hot Laboratory. The rods that were touching before in-pile exposure were separated by about 10 mils after the test. Internal examination of the fuel rods will be made at the same time as is the internal examination of the 15-2 test rods.

c. WAPD-MTR-15-2

This test unit was returned to Bettis Plant on September 13 after 6 weeks exposure. Sectioning of this bundle has been started in the Hot Laboratory but no evaluation results are available as yet. Fission product activity in the coolant system indicated a small fission break near the beginning of the irradiation. However, there was no appreciable increase in the level of activity in the system during the balance of the test period.

Fuel Element Testing and Evaluation (B. R. Price)

Because of the unavailability of suitable autoclaves, essentially no corrosion testing of fuel components in 750°F steam was carried on during this period.

In connection with corrosion studies of cladding material, a comprehensive process specification covering all aspects of corrosion testing for cladding components for both the seed and blanket fuel elements is being prepared.

Metallic Fuel Element Development (F. R. Lorenz)

Corrosion studies have indicated that cast and epsilonized (U₃Si) U-3.8 w/o Si fuel alloy is more resistant to high-temperature water than is the wrought alloy. In order to produce cast material in desired shapes, work has commenced on both direct casting into shaped molds and centrifugal casting methods.

The exact composition required to obtain a structure composed entirely of epsilon (U₃Si) is not known, and heat treated billets of a nominal U-3.8 w/o Si composition have usually had some alpha and/or delta phase present, which is undesirable for corrosion resistance. The analyses of 22 homogenized U-3.8 w/o Si ingots show an average silicon content of 3.806 w/o with a range of individual ingot averages between 3.69 w/o Si and 3.93 w/o Si. This shows that even if the exact composition required to obtain U₃Si were known it would be difficult to attain this with each melt. Ternary alloys are being prepared to try and broaden the composition range for the epsilon phase.

Zircaloy-2 clad billets of U-3.8 w/o Si with a 0.010-in foil of a third metal or alloy between the fuel and clad have been extruded to rods for corrosion mechanism studies. The extrusion of unclad U-3.8 w/o Si ingots in the temperature range of 1450°F to 1550°F has been unsatisfactory; however, limited but promising success has been obtained by extrusion in a copper jacket.

II-A-5 (Cont'd)

Hot hardness tests of the U-3.8 w/o Si alloy have been undertaken in hopes that the results will indicate a suitable fabrication range for the brittle alloy and also yield some information on relative plasticity which will aid in cladding fabrication procedures. Sufficient data have not been collected for analysis.

The brittle nature of the U-3.8 w/o Si alloy makes it extremely difficult to straighten it after extrusion. Hot stretcher straightening is currently being used.

Since corrosion studies have demonstrated rather good corrosion properities for an aluminum-to-U₃Si bond, interest has been expressed in the preparation of aluminum clad rod samples. Work has been initiated on dies suitable for cladding fuel rods with aluminum or aluminum alloys.

Extrusions made to date indicate that the U-3.8 w/o Si fuel alloy can be successfully extruded only in a very narrow temperature range. To determine the variations inherent in various heating and cooling media during processing, a study of dummy extrusion billets is being conducted so the influence of transfer time from salt bath to the extrusion container and time in the container at given temperatures can be evaluated in their effect on the final extrusion quality.

Both Zircaloy-2 and titanium have been back extruded for use as cladding components for fuel rod extrusion without a copper jacket, but with a 1/8 in. copper disc between the material and the stem of the press.

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

A total of \$946,973 was expended for capital equipment on the UO₂ program during fiscal year 1955. Eighty per cent of the equipment has been received and the balance is expected within the next few months. Installation is proceeding on schedule to meet the starting date for fuel component manufacture. Purchase requests for capital equipment, totaling approximately \$350,000, from available fiscal year 1956 funds, have been initiated for the manufacture of blanket bundles and assemblies, seed assemblies, control rods and flux depressors.

Production on the slightly enriched low density, 70% of theoretical, 0.387 in. diam, UO₂ fuel components for the Physics UO₂ - TRX Program is 60% completed. This group is scheduled for completion by January 1, 1956.

Sixteen complete assemblies of fuel elements containing natural UO₂ were delivered for the Physics UO₂ Flexible Program on October 15. This was the first major step in the production of these elements for this program. Manufacturing operations are continuing on a 3-shift basis with many ways being studied for improving the rate of production and increasing the yield of acceptable pellets. The U. S. Tool & Die Company, Pittsburgh, Pennsylvania and the American Lava Corporation, Chattanooga, Tennessee have been used as outside suppliers for end grinding of the UO₂ pellets. Work will be continuing with U. S. Tool & Die, where necessary, to help break production bottlenecks. Other suppliers have been contacted on the basis of doing the entire job and have been making surveys to determine the availability of equipment and the timing of their contribution. Because of the very tight time schedule on this Flexible Program, it is not expected that these outside suppliers will be able to furnish a completed product in time to be of major help.

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Approximately 12.5 tons of natural UO_2 powder were received from Mallinckrodt Chemical Works for the manufacture of core fuel components.

The 3 hydrogen sintering furnaces designed by Bettis personnel are operating satisfactorily at 1725°C . Two semi-automatic Harper sintering furnaces were received. One furnace is completely installed; installation on the other furnace is about 50% complete. Delivery for the second Besly-Welles end grinder is scheduled for the end of October. The alcohol rinse system is completely installed. The carbon dioxide furnace for pretreating fuel components prior to sintering is expected for delivery by December 1.

7. Control Rod Development (56cl2) F. R. Lorenz

Drawings for the first, full length, test hafnium control rod have been received. The rod will be made into 2 short cruciform sections and will be fusion welded together to the proper length. The ingots required have been melted and are being fabricated to slabs. An identical control rod will be simultaneously fabricated from stainless steel containing approximately 1% natural boron. Both are scheduled for completion about November 15, 1955 so that additional fabrication studies can be planned as the need is indicated by inspection. They will then be available for thermal distortion studies.

Consideration is being given to the direct extrusion of hafnium control rods and for this purpose the determination of suitable extrusion temperatures and the extrusion constants was undertaken by extruding 7 small hafnium billets at temperatures from 1300°F to 2000°F . All temperatures in this range were found to be satisfactory and only a slight increase in pressure was noted as the temperature was lowered. The extrusion constant varied between 58,600 psi at 2000°F to 83,100 psi at 1300°F . A cruciform die assembly and guide for a half-scale extrusion have been ordered for the 2150-ton press.

II-B-1

B. Metallurgy of Core Materials

1. Basic Study of Metallic Fuel and Cladding Alloys (53b11) E. F. Losco

Several problems of a basic and unusual nature were revealed in the Alloy Development (53b41) and Irradiation Effects (53b31) programs on metallic fuel and cladding alloys. In order to obtain a clearer understanding of these problems so that future alloys of improved corrosion resistance and irradiation stability could be developed, several investigations were initiated. These include crystal structure and thermal stability determinations on U-Mo alloys of relatively low molybdenum contents, studies on the nature of the hydride phase responsible for failure of gamma phase alloys in high temperature water, and specific heat, resistivity and phase diagram studies directly related to the important phase reversals observed during irradiation of U-Mo and U-Nb alloys. Progress on these investigations are indicated below.

Crystal Structure of Alpha Prime Phase

The crystal structure of the alpha prime phase formed on gamma quenching of U-5 w/o Mo has been identified and classified as belonging to space group Cmm2. The location of the atoms has been determined to be $(\frac{1}{4}, 0, 0)$ and $(0, \frac{1}{4}, \frac{1}{2}) \pm 0.1$ of an atomic spacing. A more precise determination of atomic positions is now in progress.

Effect of Stress on Thermal Stability of U-Mo Alloys

The application of stress appears to accelerate the decomposition of the gamma-quenched U-7 w/o Mo U alloy. X-ray diffraction results indicate the transformation product to be a distorted alpha structure. The thermal stability of this phase is now being studied.

Properties of Uranium Hydride

Work has continued on the calibration of the apparatus for measuring the magnetic properties of UH_3 in corroded U-Mo alloys. Difficulty has been experienced in obtaining a usable polycrystalline sample of bismuth for use as a calibration standard. The high anisotropy of the magnetic susceptibility of bismuth requires that preferred orientation in the sample be eliminated. Provisions are being made to use powdered bismuth to offset the anisotropy problem. A new cryostat has been designed for the apparatus and is being built to obtain better temperature control in the region below -50°C .

Crystal Structure of Uranium Hydride

Further work has been done to investigate the nature of the hydride in single crystals of gamma phase U-12 w/o Mo corroded in 650°F water. Oriented single crystals were corroded for 1, 3, 7 and 14 days. The one-day sample remained in the gamma phase, while all the other samples were completely transformed to the epsilon (U_2Mo) phase. No X-ray evidence was found for alpha uranium, UH_3 or any new hydride phase. Metallography of the 14-day sample revealed the presence of hydride platelets. These, however, were thinner and fewer in number than would be expected for polycrystalline samples similarly exposed. A gamma phase single crystal annealed for 3 days at 650°F without the presence of hydride did not transform. Thus, the internal stress produced by the hydride has a pronounced effect on the transformation kinetics of gamma to epsilon.

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Identification of Hydride Phase Formed During Corrosion of Gamma Phase Alloys

The decomposition hydrogen pressure was measured as a function of temperature over samples of U-12 w/o Mo which were hydrided by corrosion testing for 7, 14 and 21 days in 650°F water. The results were compared with the known values for pure UH_3 to help in identifying the phase formed during corrosion. The data previously reported for the first sample tested, the 14-day sample, do not agree with later information and are now considered in error. The 7- and 21-day samples, containing 270 and 1350 ppm of hydrogen, respectively, produced equilibrium pressures described by the relation:

$$\log P_{\text{mm}} = \frac{-1235}{T} \neq 5.315.$$

This compares with

$$\log P_{\text{mm}} = \frac{-4480}{T} \neq 9.20$$

for the equilibrium between UH_3 , alpha uranium and hydrogen.

The higher pressures obtained over the corroded gamma phase alloy and the lower heat of formation indicate that the hydride phase is not simple UH_3 . Since it was found that molecular hydrogen can be reabsorbed in the corroded samples, a quantitative study of the equilibrium will be made over a wide range of hydrogen concentration for a given sample. Results will be correlated with the appearance of the hydride phase observed metallographically and with magnetic susceptibility measurements.

Specific Heat Measurements

An experiment is in progress to measure specific heats and latent heats of order-disorder of U alloys as a function of temperature. The apparatus was modified to give greater sensitivity. Calibration of the modified apparatus is completed. Specific heat data will be obtained during the next report period.

Electrical Resistivity Measurements at Low Temperatures

A liquid nitrogen cryostat is being constructed for electrical resistivity measurements of U alloys (-196°C to room temperature). Fabrication of the parts is completed. Final assembly is awaiting delivery of the Monel tubing used in the construction.

The U-Nb Phase Diagram

It was previously reported (Section II-B-4 of WAPD-MRP-56) that in-pile irradiation of thermally transformed U-Mo and U-Nb alloys results in reconversion of the low temperature stable phases back to the high-temperature, corrosion-resistant gamma phase. In the case of U-Mo alloys this behavior may be explained on the basis of disordering of the epsilon phase which occurs at 31 at/o Mo. While no such phase has been noted in the U-Nb system, it was felt that such a phase may have been overlooked, possibly, because of its sensitivity to sample preparation. To check this a gamma-quenched sample of U-10 w/o Nb thermally transformed 2 weeks at 550°C was partially dissolved in HCL to remove the alpha phase. X-ray

II-B-1 (Contd.)

analysis of the residue disclosed the presence of the expected BCC gamma phase and an unexpected FCC phase.

Further indication of the existence of a FCC phase was obtained in another sample of U-10 w/o Nb whose surface was specifically prepared for X-ray analysis and then thermally transformed in vacuum. The FCC phase was again observed. Polishing the surface prior to analysis caused the phase to disappear. Acid etching to remove disturbed surface metal caused the phase to reappear, indicating that the FCC phase exists throughout the sample and is sensitive to surface preparation. Solution of the sample in HCl to remove alpha again resulted in detection of the FCC phase in the residue. Tests are now being conducted to determine if this phase is in an ordered or disordered state. Also, the composition of the FCC phase is being determined.

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II-B-2

2. Basic Study of Ceramic Fuels (53b12) Z. M. Shapiro

UO₂-O₂ Equilibrium and Kinetics

As indicated in WAPD-MRP-56, a study of the equilibrium and kinetics in the system UO₂-O₂ has been initiated in an effort to gain more information regarding the structure and stability of UO₂ under various conditions. In this connection, the equilibrium phase diagram for the region UO₂-U₃O₈ is being checked in order to determine the accuracy of the different phase diagrams prepared by Perio and Gronvald. Samples of oxide having O/U ratios of approximately 2.10, 2.20, 2.25, 2.35, 2.45, 2.51, 2.58, and 2.67 have been prepared by controlled oxidation of Mallinckrodt UO₂ (UO₂.03) in air using a thermal balance. The samples were placed in Pyrex or Vycor bulbs and were sealed off in vacuo. These are now being annealed at 360°C and 700°C for periods of 9 and 16 days. Following this treatment the O/U ratios and crystal structures of the samples will be determined. It is expected that this will be completed in approximately 2 months.

A number of samples of various O/U ratios are also being prepared by controlled oxidation in air at 180°C. One group of these samples will be sent directly for crystal structure determinations. Duplicate sets will be annealed in evacuated sealed bulbs at 180°C for periods of 9 and 16 days prior to submitting them for X-ray analysis.

By the above experimental procedure it is hoped to obtain information on the nature of the immediate products of the slow oxidation of UO₂ and the phase changes that occur on annealing.

Kinetic Studies

Work has continued on the rates of oxidation and reduction of UO₂ and U₃O₈ in 650°F water containing oxygen and hydrogen in order to gain a better understanding of the reactivity of UO₂ in water under various conditions.

UO₂ and U₃O₈ powders have been exposed for 45 days in degassed water at 650°F and 35 days to degassed water maintained at a pH of 10.5. Analysis of samples removed at intervals during these periods showed that the powders retained their original oxygen contents. The UO₂ samples analyzed as UO₂.020 ± 0.005; the U₃O₈ samples maintained an O/U ratio of 2.66 ± 0.01. Further experimental work on these samples has been discontinued since they appear to be stable to oxidation and reduction under these conditions.

Samples of U₃O₈ have been removed from water containing 100 cc H₂/kg after exposure periods of 5 to 65 days. Information on O/U ratios and X-ray data on samples removed after exposure of up to 35 days was reported in WAPD-MRP-56. Samples removed after 45, 55, and 65 day exposures are now being analyzed and will be reported on in WAPD-MRP-58.

Samples of UO₂ and U₃O₈ powder have been removed after a 5-day exposure to water containing 2 cc O₂/kg. They are now being analyzed to determine the

II-B-2 (Contd.)

extent of oxidation. These results and the results of 10 and 15 day exposure periods will be reported in WAPD-MRP-58.

Diffusion Studies

Assembly of components and fabrication of apparatus for the experiment to determine the rate of fission release from uranium dioxide during irradiation is progressing. Definite assignment of personnel to the project has been arranged. If equipment deliveries are made on schedule, it is expected that the apparatus can be installed during a Brookhaven pile shutdown in November. Equipment for the determination of the self diffusion of uranium and oxygen in UO_2 as a function of O/U ratio is now under consideration. It will be completed in approximately 2 months. The results of this work will aid especially in understanding the sintering mechanism of UO_2 .

Magnetic Susceptibility Studies

The final report on magnetic susceptibility measurements on uranium oxides has been received from the investigators at the Carnegie Institute of Technology, where the work was performed for the purpose of gaining more information on the structure of non-stoichiometric uranium-oxygen compounds. Conclusions of the report are as follows:

1. The phase boundary of the lower oxygen phase (CaF_2 structure) is at an oxygen-to-uranium ratio of 2.31 ± 0.01 .
2. The phase boundary of the higher oxygen phase (orthorhombic) is at an oxygen-to-uranium ratio of about 2.44. This figure is not as reliable as the foregoing.
3. Dawson and Lister (J. Chem. Soc., 1950, 2177, Ibid., 2181) who performed earlier magnetic susceptibility measurements appear to have an error in their O/U analysis. This means that the X-ray data of Anderson and Alberman (J. Chem. Soc. London, 1949, S 303) are also in error as to composition since the same sample was used in both measurements.
4. When oxygen is added to UO_2 , the anion lattice is undisturbed and oxygen enters in interstitial positions, increasing the exchange interaction between U(IV) ions.
5. With increasing oxygen content it is most likely that U(IV) ions are replaced by U(VI) ions.
6. The difference between U_3O_8 and oxide of slightly lower O/U compositions in the same phase is brought about by creation of oxygen lattice vacancies.
7. The upper limit on the presence of UH_3 in samples studied is of the order of one part in 10^5 .
8. $UO_{2.03}$ has an antiferromagnetic transition at 28°K. This transition is not evident in oxides of greater oxygen concentration; in the latter, anomalies exist in the region of 2 to 28°K.

II-B-3

3. Irradiation Effects on Ceramic Fuels (53b30) Z. M. Shapiro

UO₂ Irradiation Program

As indicated in WAPD-MRP-56, an irradiation program utilizing statistical methods is being designed for insertion into the Hanford reactors. The current NaK capsule design is being changed according to suggestions from Hanford personnel. A Hanford irradiation request is being prepared and will be submitted for approval by November 1.

MTR-WAPD 14-Met 6 Irradiation of Hot Pressed and Steam Oxidized UO₂

This experiment consists of 2 samples of hot pressed steam oxidized UO₂ and 2 samples of hot pressed MCW UO₂ which were irradiated in the Materials Testing Reactor (MTR) to a maximum exposure of approximately 5000 MWD/T. One of each type specimen was punctured and the fission gases released at various temperatures from room temperature up to 250°C were separated. Sections are being removed for metallographic analysis and the remaining portions of the specimens will be re-inserted into the gas apparatus for a measurement of the fission gas release up to temperature of 1000°C and for total gas analyses. It is expected that results will be presented in WAPD-MRP-59.

MTR-WAPD 14-Met 8 Irradiation of Half Length PWR Reference Rods

This experiment was designed to ascertain the simultaneous effect of irradiation and high temperatures on PWR reference rods irradiated under neutron flux conditions comparable to the highest calculated PWR flux. For this purpose, 4 natural UO₂ rods identical to the PWR reference design, except that a 4½ in. length was utilized instead of the reference 10½ in. length, were canned in NaK capsules and irradiated for 3 weeks in the L-45 hole in MTR in a flux of approximately 4×10^{14} thermal neutrons/cm²-sec. These rods were discharged from the reactor on October 3 and will be returned to Bettis Plant by the end of October. For this experiment, the effect of the high temperatures on the UO₂ is of primary interest rather than the effect of exposure. Hence, the samples will be examined and sectioned for metallographic examination as soon as possible after they are returned.

MTR-WAPD 14-Met 9

Irradiation of Half Length PWR Reference Rods to High Burnups

The samples for this experiment are identical to those of WAPD 14-Met 8 and are scheduled for insertion into the L-45 hole in the MTR during the October 24 shutdown. The samples will be irradiated for an indefinite period, the exact irradiation time being dependent on the results obtained from experiment WAPD 14-Met 8.

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MTR-WAPD 19-Met 1 and 2

Irradiation Cycling of Steam Oxidized and MCW UO₂ Rods

Metallographic data on 2 of the 13 cycling samples previously reported in WAPD-MRP-55 and 56 are complete. There is no evidence of a change in the microstructure in either the hot pressed or the cold pressed and sintered steam oxidized samples. Both types of samples show gross cracking. One sample each of hot pressed and cold pressed and sintered MCW UO₂ have been sectioned with a tubing cutter for examination of individual pellets. Samples have been removed for density measurements and the results will be reported in WAPD-MRP-58.

MTR-WAPD 19-Met 3

Irradiation of Waterlogged UO₂ Specimens in MTR Rabbit Facility

Microstructures have been obtained from one of the 6 cold pressed and sintered Mallinckrodt UO₂ specimens which had been cycled 40 times in small, water filled autoclaves into and out of a thermal neutron flux of 5×10^{13} nv. Gross cracking was observed, but the microstructure was apparently unaffected by the combination of cycling and irradiation. In addition to the radial type cracks, it was observed that the specimen contained a circumferential crack which extended completely around the specimen near the outer periphery.

MTR-WAPD 19-Met 5

Irradiation Cycling of Natural and Enriched UO₂ Rods

The 4 enriched UO₂ samples will remain in the low flux A-Piece cycler for a total of 7 MTR cycles instead of 5 as previously reported in WAPD-MRP-56. The burnup of samples discharged on September 12 was previously reported as 16,000 MWD/T in WAPD-MRP-56. This figure is in error; the correct burnup is 1600 MWD/T. The samples are scheduled for discharge from the MTR on November 14. On the basis of recent fission monitor data these samples will receive an estimated exposure of 3500 MWD/T.

The 8 full length natural UO₂ specimens were returned to Bettis Plant on October 6 and preliminary information will be reported in WAPD-MRP-58.

MTR-WAPD 13-Met 4 Irradiation Cycling of UO₂ in MTR HB-5 Loop

The undefected sample of cold pressed and sintered MCW UO₂ which had been cycled 4000 times in the MTR HB-5 loop to an estimated exposure of 600 MWD/T was returned to the Bettis Plant during September. It is expected that preliminary information will be available for presentation in WAPD-MRP-58.

MTR-WAPD-13-Met 6 Irradiation Cycling of Defected UO₂ in MTR-HB-5 Loop

A defected UO₂ specimen cold pressed and sintered to 93% of theoretical density and enriched to 4.7 w/o U-235 has been inserted into the HB-5 loop facility at MTR. It is scheduled to operate at a maximum heat flux of 600,000

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Btu/hr-ft²-°F for 6 weeks. During the first 3 weeks the specimen will remain essentially stationary in the high neutron flux position except for daily checks of the cycling mechanism. During the first 2 weeks of the second 3-week period it will be cycled 5 min into and 5 min out of the high neutron flux region. If no major operating changes have occurred after this period of time the cycle will be changed to 12 hr into and 12 hr out of the high neutron flux region for the remainder of the test. Plans are being made for complete radiochemical analysis of the loop water similar to the analyses performed on the loop irradiations in the Chalk River NRX reactor.

Thermal Cycling

Five Zircaloy-2 clad UO₂ rods and one empty Zircaloy-2 welded tube have been cycled 2000 times in NaK between the temperature limits of 1050°F and 550°F. The holding time at each temperature is 5 min, and the transfer time is 20 to 30 seconds. The experiment was previously outlined in WAPD-MRP-56.

Increase in axial run-out of 0.001 and 0.075 in. have been recorded. Examination of the rods at 2000 cycles indicate that the rods have increased from 0.008 to 0.075 in. in length with the largest increase measured on the empty clad. The largest diametral increase measured was approximately 0.002 in. The majority of the rods showed diametral increases of 0.005 or less. X-ray photographs show distinctly the breaking up of a few pellets, but no evidence of clad damage is visible.

CR-WAPD-Met 4A First UO₂ Defect Test

One undefected UO₂ specimen has been punctured and the fission gases released up to temperatures of 1000°C have been separated. It is expected that preliminary gas analyses will be available by November 15.

CR-WAPD-Met 5 Second UO₂ Defect Test

Density measurements of the UO₂ in the defected specimen and in a control specimen are now being made in an attempt to explain the central void and weight gain in the defected specimen. This was previously discussed in WAPD-MRP-56. It is expected that density information will be presented in WAPD-MRP-58.

CR-WAPD-Met 7 and Met 8 Third and Fourth UO₂ Defect Tests

Both Chalk River X-rod loops are now operating with defected UO₂ specimens which were cold pressed and sintered to 93% of theoretical density and enriched to 4.7 w/o U²³⁵. Previous information on 2 tests was presented in WAPD-MRP-55 and WAPD-MRP-56. The beta-gamma activity of the water from the 2 loops was approximately the same at the time of reactor start-up after the prolonged Chalk River shutdown. However, the water activity in the X-3 loop has increased slightly, and at present the test is operating with an average beta-gamma activity in the water which is 2 to 3 times that of the X-1 test. The defect in each specimen is a 0.005 in. hole located near one end. Further details of loop operation are given under project 53b34 "In-Pile Radiation Exposure Tests of PWR Fuel Materials."

II-B-3 (Contd.)

CR-WAPD-Met 9 Fifth UO₂ Defect Test

Six PWR reference size rods are being prepared for the fifth UO₂ defect test in the X-rod loops at Chalk River. Five of the rods will be placed in the X-1 loop in-pile section replacing the samples from CR-WAPD-Met-7. The sixth rod will be placed in the out-of-pile section as a control sample. The 5 in-pile samples will operate at heat fluxes of 300,000, 400,000, and 500,000 Btu/hr ft²-°F. One defect specimen and one undefected control specimen will be irradiated at each of the 2 lower flux positions. The center sample, operating at the highest heat flux, will also be defecte. It is planned to use 0.005 in. holes for two of the defects and a longitudinal crack starting at the weld for the other. The specimens will be fabricated from enriched UO₂ prepared by the decomposition of uranyl nitrate to UO₃ and subsequent reduction to UO₂. The samples will be compacted with PVA and Sterotex additions and sintered at 1750°C in a hydrogen atmosphere furnace according to PWR reference specifications. The samples are scheduled for delivery to Chalk River in early November for insertion into the pile November 22. The loop will be modified to permit better sampling for radiochemical data from the system operating with defecte. Additional details of the X-1 loop operation and plans are included in Section II-C-2a.

II-B-4

4. Irradiation Effects on Metallic Fuels (53b31) E. F. Losco

Bare Samples of U-Nb

Since the last report period, the seven samples of U-10 w/o Nb irradiated to a maximum exposure of 1820 MWD/T in MTR in various heat treated conditions, have been sectioned into corrosion and metallographic samples. The metallographic samples have been mounted in preparation for structure examination and micro-hardness determinations. Corrosion rates in 650°F water on the gamma-quenched and irradiated samples will be obtained. It is expected that the information obtained from these tests will help to clarify phase reversal indicated by pre-post electrical resistivity measurements.

Clad Samples of U-Nb

Corrosion testing of 10 samples of Zircaloy-2 clad U-10 w/o Nb, fabricated in various manners and irradiated in MTR process water to an exposure of approximately 2000 MWD/T, is continuing. These samples, together with unirradiated control samples, have undergone 14 days of test in 650°F static water without failure.

Twelve specimens of Zircaloy-2 clad natural U-10 w/o Nb, homogenized and aged at various temperatures, have been irradiated 6 weeks in MTR process water to an estimated maximum exposure of 2600 MWD/T. Pre-post dimensional measurements indicated only minor changes due to irradiation. No correlation with degree of transformation, thickness of diffusion zone, or exposure is apparent.

Clad Samples of U-Mo

Seventeen specimens of Zircaloy-2 clad 5 w/o enriched U-9, -10.5, -12 and -13.5 w/o Mo alloys, which were irradiated 13 months in MTR process water to an estimated maximum exposure of 28,000 MWD/T, have been measured for any dimensional changes due to irradiation. The cladding of one specimen (U-10.5 w/o Mo, irradiated in the highest flux position) was split open longitudinally in several places. Close examination indicated that some plastic flow had occurred in the cladding prior to failure. A cross-section of the specimen showed that the crack extended to the middle of the core material and that there was no separation of the clad and core. Pre-post dimensional measurements of all specimens showed an increase in both length and diameter. The length changes were a maximum of 0.4% for the U-9 w/o Mo alloy, and diminished as the molybdenum content increased to a maximum of 0.2% for the U-13.5 w/o Mo alloy. The diameters of all specimens increased; the maximum diametrical change being 2.3%. The volume increases of all specimens appeared to be dependent only on burn-up and amounted to about 4% at the maximum exposure of 28,000 MWD/T. This volume change, required to cause cladding failure, is similar to the volume changes noted for failed rods of zirconium clad natural uranium, described in a later section. It is postulated that the ultimate radiation damage limit is dependent upon the ductility of the cladding material.

II-B-4 (Contd.)

Because of the deemphasis of U-Mo fuel elements and the overload of corrosion tests of irradiated materials, the corrosion testing of 20 Zircaloy-2 clad specimens of as-extruded natural U-9, -10.5, -12 and -13.5 w/o Mo alloys, which were irradiated in MTR process water to a maximum exposure of 2000 MWD/T, has been discontinued.

The 24 samples of 5 w/o enriched U-Mo alloys, which make up experiment MTR-WAPD 11-Met 14, were discharged from the reactor September 12 after being irradiated in process water to a maximum exposure of approximately 12,000 MWD/T. The irradiation of experiment MTR-WAPD 11-Met 15, comprising 24 samples of 5 w/o enriched U-Mo alloys in the as-extruded condition, is continuing.

Loop Tests

The corrosion testing of the irradiation control specimens, which were fabricated from the same extrusions and in the same manner as the Chalk River X-1 loop specimens and were irradiated in MTR process water to a maximum exposure of 8800 MWD/T, has continued. All U-12 w/o Mo specimens, including controls, failed in 35 days or less. Of 12 irradiated U-10.5 w/o Mo samples and 6 non-irradiated controls, 8 irradiated and 3 control samples are continuing in test after 42 days.

The Chalk River X-3 loop section, containing thirteen specimens of Zircaloy-2 clad U-11 and 12 w/o Mo which were irradiated to an exposure of approximately 4000 MWD/T, has been disassembled. A visual examination of all specimens, including one defected specimen, has been made. All specimens were covered with a uniform black crud. No apparent failure or bulging of any specimen occurred. Pre-post diameter measurements indicated no change as a result of irradiation.

Irradiation of Transformed Specimens of U-Mo Alloys

Additional measurements have been made on the transformed and irradiated specimens of U-9, -10.5, -12 and -13.5 w/o Mo alloys described in WAPD-MRP-56. Electrical resistivity measurements at temperatures varying from -196°C to +150°C have been made on the irradiated specimens. These results show that the U-9, -10.5, -12 and -13.5 w/o Mo specimens irradiated in the gamma phase exhibit a negative temperature coefficient after irradiation similar to that prior to irradiation. However, the positive temperature coefficient of electrical resistivity of the transformed specimens has reversed to a negative coefficient after irradiation. Hardness measurements have been completed and these results confirm the observation made in WAPD-MRP-56 that the transformed specimens have decreased in hardness after irradiation to the value of the gamma-quenched condition.

Results of the metallographic examination are puzzling. The structure of the U-10.5, -12 and -13.5 w/o Mo alloy specimens appears to be unchanged with alpha uranium precipitated at the grain boundaries. This contradicts the hardness and electrical resistivity data which indicate that the alpha was taken back into solution. The possibility remains that the phases present in the irradiated samples are gamma plus precipitated alpha. In order to investigate

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this, a sample of U-9 w/o Mo which was transformed to alpha plus epsilon was heat treated at 580°C for 40 min. and water quenched so as to form the alpha plus gamma phases. X-ray examination of this sample after this heat treatment showed alpha and highly disordered epsilon present. The electrical resistivity of this specimen at -196°C showed a 5 percent increase. Since the increase after irradiation was 125% it would appear that the electrical resistivity results can not be explained on this basis. Further attempts will be made to produce the alpha plus gamma phase to verify this observation. In addition corrosion tests will be conducted in 650°F water. Since corrosion resistance is sensitive to the presence of alpha a comparison of thermally transformed specimens with those which were thermally transformed and then irradiated should be enlightening.

Cold Water Irradiation of Prototype Rod Fuel Elements

Prototype fuel rods of unalloyed uranium and U-2 w/o Zr clad by co-extrusion with Zircaloy-1 have been irradiated in the Chalk River NRX reactor to a maximum burnup of 10,000 MWD/T. These elements, containing enrichments which varied from 0.7 to 5.5 w/o U²³⁵, were fabricated by Nuclear Metals, Inc. The unalloyed uranium rods were prepared in the beta-quenched and alpha-annealed conditions. The U-2 w/o Zr rods were prepared in the cold-worked, alpha-annealed and gamma-quenched conditions. At regular intervals during irradiation the rods were removed from the reactor and radiographed so that changes in length could be detected with increasing exposure. The results indicated that the unalloyed uranium samples grew only 0.1 to 0.5% while the U-2 w/o Zr rods grew 1 to 3% in length. However, when dimensional changes were carefully measured in the Bettis Hot Laboratory, it was found that the growth of all samples was very small, varying from 0 to about 1.8% in length. It was concluded, therefore, that the cladding was quite effective in restraining directional deformation irrespective of heat treatment and alloy content.

The percent change in volume determined for the rods ranged from 1 to 3%, increasing with increasing exposure and independent of heat treatment and composition. Visual examination of the rods indicated that 1/4 of them, exposed to 9000 MWD/T or greater, had transverse and longitudinal splits in the cladding near the end cap weld and smaller splits and bulges near the center of the rods. No cladding failures were observed in rods irradiated to lower exposures. Thus, the cladding cannot restrain volume expansion and splits when the expansion amounts to about 3%. Since this occurs at about 9000 MWD/T, this exposure may be considered the upper limit for fuel elements of this type.

II-B-5

5. Alloy Development (53b41) E. F. Losco

U-3.8 w/o Si Clad with Zircaloy-2

The second of two fuel elements prepared by Nuclear Metals, Inc. by coextruding Zircaloy-2 over U-3.8 w/o Si failed in 14-28 days in 650°F water. The first element had previously failed in 0-14 days. Both failures were attributed to bondline attack since the cladding split some distance from the 0.040 in. diam defect. This was confirmed with a bare end sample from the same extrusion which showed bondline attack within 24 hr in 650°F water.

Two extrusions, numbered 26-1 and 27-1, of duplex melted U-3.8 w/o Si alloy clad with Zircaloy-2 prepared at Bettis Plant are being tested as defected fuel elements and bare-end samples in 650°F water. In the as-extruded condition 4 of 8 elements from extrusion 26-1 and 1 of 4 elements from extrusion 27-1 failed in 0-14 days. The others are continuing in test. Examination of failed elements and bare-end samples indicates that failure is due to a poorly corrosion-resistant interface between clad and core. Metallographic examination of extrusion 26-1 showed the presence of residual alpha phase as stringers in the epsilon matrix. This material, extruded at 1650°F, also showed partial unbonding of the cladding. Extrusion 27-1 showed no alpha stringers but excess delta and only a minor and intermittent amount of unbonding. The extrusion temperature in this case was 1600°F.

U-3.8 w/o Si Extrusions with Metallic Barrier Layers

The investigation of U-3.8 w/o Si extrusions with metallic barrier layers is being continued. Two extrusions (Nos. 460 and 462) with niobium barriers and Zircaloy-2 cladding are being studied. Although preliminary examination indicated good bonding for both extrusions, when clad samples from extrusion No. 460 were annealed at high temperatures to form diffusion layers it was found that interdiffusion did not occur between the barrier and the core because of unbonding. The core alloy of extrusion No. 460 contained appreciable quantities of alpha phase. Bare extruded samples split longitudinally within 4 days in 650°F water. Therefore no definite conclusions could be drawn concerning the effectiveness of the niobium barrier for this extrusion. However, clad defected samples from extrusion No. 462 also failed in the first 14-day test period. The core material for this extrusion contained only a small amount of residual delta-phase. The 7-day corrosion rate for bare core samples was quite good, 0.54 mg/cm²/hr. Also, there was no evidence of unbonding between core, barrier and cladding. Metallographic examination of failed rods indicated that failure was due to bondline attack, the core material being in good condition. These results indicate that niobium is not a satisfactory barrier material.

Coextrusions with Mo, U-10 w/o Nb and Cu barriers have been prepared and are now being examined metallographically.

II-B-5 (Contd.)

Clad U-3.8 w/o Si with Cast Aluminum Barrier Layers

It was previously reported that Al-6 w/o Cu alloy appeared promising as a barrier layer between U-3.8 w/o Si and Zircaloy-2, no bondline attack being observed in 14-day tests in 650°F water. Accordingly, several simulated fuel elements, 4 in. long by 0.42 in. OD, were prepared by vacuum casting a 0.035 in. thick layer of Al-6 w/o Cu between bare extruded rods of U-3.8 w/o Si and Zircaloy-2 tubes. Two such elements, containing 0.040 in. diam defects drilled into the core alloy survived 15 days in 680°F water. On the 16th day of test an equipment failure resulted in overheating of the autoclave and the samples were subjected to 750°F steam for approximately 8 hours. One element failed by bulging and splitting of the cladding at one end. The other sample showed no signs of failure and is being continued in test in 680°F water. Four other elements have been in test for 14 days in 650°F water with no signs of failure. The failed sample was sectioned for examination. It was apparent that an excellent bond existed between barrier and core and no bondline attack occurred. Failure was due to lack of bonding between barrier and cladding. While no corrosion of the clad or barrier occurred, the unbonding presented a path for water to reach the end cap area where the core was directly exposed causing corrosion at this location.

Because of the promising results so far obtained with fuel elements prepared with cast Al-6 w/o Cu barriers, an intensive effort is being devoted to investigating this type of element in detail. It is desirable from a physics standpoint to reduce the thickness of the barrier layer to a minimum. The first attempts to prepare 0.010 in. thick barriers indicated that shrinkage or gas voids can occur within the barrier and that bonding of the barrier to the Zircaloy tubing was not entirely satisfactory. It was also indicated that centering of the core rod, which at first was believed to be a difficult problem, offers no difficulty the rod being well centered as the brazing alloy fills the gap between rod and cladding. Studies involving variations in barrier thickness, brazing temperature and time, barrier application techniques, barrier composition and surface preparation are under way.

Tensile Properties of U-3.8 w/o Si

Tensile properties of extruded U-3.8 w/o Si rods are being determined to establish its strength and ductility as a function of temperature. These data will be helpful in indicating whether extruded rods are sufficiently ductile to permit fabrication of fuel elements by a hot drawing operation. The tensile specimens were machined from a rod previously extruded from an epsilonized billet. Complete absence of ductility was indicated at room temperature. An ultimate strength value of 85,000 psi was obtained. Elevated temperature tests require controlled inert atmosphere facilities to prevent oxidation. These are being prepared.

U-3.8 w/o Si Ternary Alloys

Bare arc-melted samples of U-3.8 w/o Si alloys containing ternary additions of 0.2, 0.5 and 1.0 w/o of Mo, Nb or Cu were continued in corrosion

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test in 650°F water. The 1.0 w/o additions resulted in relatively early failures and high corrosion rates. Samples with 0.5 w/o additions have corrosion rates of the order of -1.50 mg/cm²/hr after 42 days of test, some samples having failed by gradual conversion to oxide because of the relatively high corrosion rate. Samples containing 0.2 w/o Mo, Nb or Cu, however, have all survived 42 days of test and continue to display superior performance. The corrosion rates are -0.64, -0.70 and -0.83 mg/cm²/hr which is a factor of 2 to 3 lower than that of the binary U-3.8 w/o Si alloy.

Microstructures of the ternary alloys were obtained. These indicated that the ternary additions were effective in eliminating the residual alpha phase from the epsilonized structure. Increasing amounts of the ternary addition, however, resulted in larger percentages of residual delta and a third phase in the microstructure. It appears, therefore, that an optimum percentage of the ternary addition will greatly improve corrosion resistance and widen the range of tolerable silicon content. To check this, alloys containing 0.1, 0.2 and 0.3 w/o of Mo, Nb or Cu have been prepared. Corrosion tests on these have just started.

Other elements have also been investigated as ternary additions to the U-3.8 w/o Si alloy. Pt was previously shown to affect corrosion properties adversely. Epsilonized arc-melted buttons containing 0.2, 0.5 and 1.0 w/o of Pd, Ru or Ta, which were previously shown to be beneficial in gamma phase U-Mo alloys, were also tested in 650°F water. It was found that these elements, in all concentrations tested, were highly detrimental to the corrosion properties of U-3.8 w/o Si.

Corrosion Mechanism for U-3.0 w/o Si

The U-3.8 w/o Si alloy, unlike gamma phase alloys which fail by disintegration due to internal hydride formation, normally fail by gradual surface oxidation in 650°F water. It was previously shown that the corrosion rate is sensitive to microstructure, being higher when appreciable amounts of alpha or delta are present in the epsilon matrix. Additional observations made on epsilonized and extruded unclad samples have revealed that a thin, discontinuous phase appears between the surface oxide layer and the unattacked alloy during corrosion. This layer is similar in appearance to the excess delta phase and is tentatively identified as delta (U₃Si₂). It is postulated that the uranium in the surface layers of the alloy is depleted by oxidation and that the silicon left behind initiates the formation of U₃Si₂ at the oxide-alloy interface. X-ray identification of this layer is being made.

Samples from one of the extrusions showing the U₃Si₂ corrosion layer (extrusion No. 460 which contained alpha stringers) were annealed 7 days at 800°C. Instead of improving the corrosion resistance, as might be expected, the annealing treatment was harmful. This was confirmed for bare and bare-end samples taken from two extrusions provided by Nuclear Metals Incorporated. Annealing at 800, 850 and 900°C after extrusion greatly increased the corrosion rates in 650°F water. Internal oxidation at grain boundaries of samples taken from extrusion No. 460, after annealing 7 days at 800°C, was observed after 3 days in 650°F water. This type of corrosion was not observed in samples which were not annealed after extrusion. Although the surface attack was heavy

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for this particular extrusion because of relatively large amounts of excess alpha and delta, the internal, intergranular oxidation resulted in an overall expansion of the sample diameter. These results, together with the findings for other extrusions on the adverse effects of heat treating after extrusion, suggest unusual microstructural changes.

The above observations emphasize the need for a detailed study of the corrosion mechanism for bare samples of the U-3.8 w/o Si alloy. This program has been initiated and will be expanded to include effects of ternary alloy additions.

Crushed U-3.8 w/o Si as a Fuel Material

Since the presence of residual alpha uranium in epsilonized U-3.8 w/o Si adversely affects corrosion resistance and is undesirable from the standpoint of dimensional stability under irradiation, the possibility of minimizing its bad effects by employing crushed particles of the alloy was considered. Crushed particles of the alloy, impregnated with a corrosion resistant aluminum base alloy which serves also as a bond to a Zircaloy tube, may result in a fuel element of superior dimensional stability and corrosion resistance. To check this possibility particles of U-3.8 w/o Si of over 40 mesh size, having a bulk density 55% of maximum, were inserted in a Zircaloy-2 tube and impregnated with Al-6 w/o Cu alloy. Excellent bonding was realized between the Al alloy and cladding and the Al alloy and U-3.8 w/o Si alloy particles. However, shrinkage cavities were observed within the Al alloy. During impregnation the U-3.8 w/o Si alloy particles were overheated and transformation of the epsilon phase was evident. Additional work is under way to improve density, reduce the amount of Al alloy used and improve the impregnation technique.

Effect of Defect Size on Corrosion Life of Clad U-12 w/o Mo

The samples of extruded U-12 w/o Mo clad fuel elements being tested to study the effect of defect size on the corrosion life have completed 140 days of test time. The defect diameters are 0.003 in., 0.013 in., 0.020 in., 0.040 in. and 0.080 in. All of the samples with 0.080 in. defects have failed in 70-126 days, and 2 of 4 samples with 0.040 in. defects have failed in 84-98 days. None of the samples with defect diameters smaller than 0.040 in. have failed indicating that the area of exposed fuel is related to corrosion life.

Corrosion of Clad U-2 w/o Zr Alloy

Five simulated fuel elements of U-2 w/o Zr coextruded with Zircaloy-2 cladding which were provided by Nuclear Metals, Inc. have survived 28 days of test in 650°F water. These elements were prepared by drilling 0.007 in. diam defects just through the cladding and then annealing for 2 hr at 1050°C to form a corrosion resistant U-Zr diffusion layer at the base of the defect. The elements had been tested 1235 hr in 600°F water by Nuclear Metals, Inc. before delivery to Bettis Plant. A sixth element was sectioned at the defect as-received. The very small defect was observed to be packed with oxide which apparently protected the underlying core alloy from further oxidation.

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Small defects were also effective in increasing the life of U-12 w/o Mo elements, as indicated above, even when the defects were drilled well into the core alloy. In order to determine if defect size, rather than the diffusion treatment, was responsible for the long life of the Nuclear Metals Inc. elements it was decided to continue the corrosion tests of the five elements after enlarging the original defects. The 0.007 in. diam defect was enlarged to 0.040 in. diam on 4 elements. The defect was drilled 0.030 in. deep (just through the cladding) on 2 elements and 0.040 in. deep (just through the diffusion layer) on 2 other elements. On the 5th element the original 0.007 in. diam defect was enlarged to 0.080 in. diam and drilled 0.030 in. deep (just through the cladding). These samples have been returned to test.

Aluminum-Base Alloys for Cladding Applications

Screening corrosion tests were continued on various experimental aluminum alloys under static, dynamic, and semi-dynamic conditions. Results of a static test showed that one of the alloys (X-2219, Al-6 w/o Cu) has satisfactorily withstood 8 weeks exposure to 680°F water. This is the same alloy being used for brazing U₃Si to Zircaloy tubes. The weight gain per unit area vs. time curve was linear with a corrosion rate equal to 0.13 mg/cm²/day (6.9 mils/yr). Metal loss data were calculated from measurements of film thickness. A straight line was also obtained from the metal loss per unit area vs. time plot and the corrosion rate was calculated to be 0.22 mg/cm²/day (11.7 mils/yr). As was mentioned in WAPD-MRP-56, this same alloy corroded badly after only a 4-week exposure to 680°F steam. This surprising result is being checked. A second alloy (M-388, Al-1 w/o Ni - 0.5 w/o Fe), which survived less than 1 week at 680°F water, has held up in 680°F steam. Weight gains were much greater for this alloy than for the X-2219 alloy, but the data do not follow a definite curve. A third alloy (M-400, Al-1 w/o Ni - 1 w/o Fe) has satisfactorily withstood a 4-week exposure to 680°F steam.

The 3 alloys were subjected to a dynamic loop test for 500 hr at 500°F (pH = 6-8). The weight gains and calculated metal losses were very nearly the same for the 3 alloys but were much higher than those from the static tests. Two of the 3 alloys (M-388 and X-2219) were also subjected to a dynamic loop test at 600°F with the circulating water controlled to a pH of 8-9. After a 2-week exposure, both alloys showed approximately the same weight gains and metal losses. Again, these values were higher than those from the static tests.

In the semi-dynamic test, disc-shaped samples of the 3 alloys were attached to a rotatable shaft in an autoclave. After a run of 6 days at 600°F, during which time the pH of the water changed from 6.9 to 5.2, all of the alloys showed a considerable weight loss. This test is still in progress.

The effect of surface treatment on corrosion behavior was investigated for 2 of the alloys in a static 680°F steam test. Except for those specimens which were chemically coated, all of the surface treated M-388 alloys failed after 3 weeks exposure. On the other hand, the chemically coated X-2219 samples failed after 3 weeks exposure, but the other surface treated samples held up.

A start has been made on the effect of heat treatment on the corrosion behavior of these alloys. Preliminary results for the X-2219 alloy indicated that there is very little effect.

Two autoclaves have been equipped with a pump to circulate the water. Long-time testing of the 3 alloys at pH ranges of 8.0-9.5 and 4.0-6.5 have been started.

II-B-6

6. Ceramic Fuel Materials Development (53b42) Z. M. Shapiro

Preparation of UO_2

Work was continued on the preparation of UO_2 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 UO_3

During this period 4 kilograms of UO_3 were prepared for the irradiation program by hydrogen reduction of UO_3 at 775-820°C. This includes both enriched UO_3 and oxide containing ZrO_2 and SiO_2 additives. In preparing the powders with additives, known amounts of zirconyl nitrate and ethyl orthosilicate were dissolved in the uranyl nitrate solution so that, on pyrolysis and hydrogen reduction, the desired amount of additive would be uniformly dispersed throughout the resulting UO_2 .

Activated UO_2

Work on sulfated UO_2 has continued in an effort to determine the reasons for its superior sinterability. A quantity of UO_3 in which the sulfate ions were added to the uranyl nitrate solution was received from the Mallinckrodt Chemical Works. After reduction with H_2 at 790°C this material was found to contain from 1/2 to 1/3 less sulfate ion and from 2/3 to 1/2 less U(VI) than samples prepared by ball milling and digestion with sulfuric acid. The sinterability of this powder will be determined during the next month. The sulfate additions were found to have no effect on the X-ray pattern of UO_2 .

Since the particle size of these active powders is thought to be smaller than that of the normal material, Mallinckrodt UO_2 was treated with 200 ppm of uranyl sulfate in order to determine the contribution of the sulfate ion itself to the superior sinterability. After analysis, this material will be compacted and sintered in the same manner as were the other activated powders.

ORNL and ANL UO_2

Samples of UO_3 and UO_2 prepared through steam oxidation, spray denitration, precipitation and crystal growth methods have been received from Oak Ridge and Argonne for evaluation in the pressing and sintering program.

The large grain UO_3 hydrates, on preliminary petrographic and X-ray examination, appear to be orthorhombic single crystals. Their dehydration and reduction to UO_2 will be carried out under carefully controlled conditions in an attempt to determine whether single crystals of UO_2 are produced.

Precipitation Studies

A series of ammonium diuranate powders was pyrolyzed at 250, 500, and 800°C and then reduced with hydrogen at 800-900°C. In most cases the final

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products were pyrophoric and had to be stabilized by use of longer reduction periods and/or higher reduction temperatures. These oxides are now being analyzed by chemical and X-ray methods. A study of their pressing and sintering characteristics will be completed in approximately 2 1/2 months.

X-Ray Studies

An investigation was made in order to determine whether the uranium oxides and their parent compounds could be characterized by their X-ray diffraction patterns as to their method of preparation. Samples of ammonium diuranate, U_3O_8 , UO_3 , and UO_2 prepared by various methods were examined. Work to date indicates that most of the compounds studied had normal powder patterns. The powder lines of UO_2 made by high pressure steam oxidation of uranium were broader than those of oxide prepared by other methods, indicating that this process produces a UO_2 with a very small crystallite size. However, U_3O_8 made by ignition of this UO_2 yielded the sharpest X-ray pattern indicating that some sintering occurs during oxidation.

Storage Stability

It has been reported that UO_2 has a tendency to pick up both water and oxygen on standing, the amount being a function of its mode of preparation as well as atmospheric conditions during exposure. Since it is highly undesirable for UO_2 intended for use in the PWR to undergo changes in storage which will greatly alter the nature of the material, studies were made of the storage stability of UO_2 as prepared by different methods. The results indicate that MCW UO_2 is considerably more stable even after further reduction with H_2 or oxidation to U_3O_8 and subsequent reduction with H_2 than powders prepared at the Bettis Plant by precipitation or decomposition techniques.

Carbon Doped UO_2

A series of carbon doped UO_2 powders containing 140, 330, and 860 ppm carbon was prepared by soaking Mallinckrodt oxide in dilute Aquadag suspensions and drying in vacuo. These powders were compacted to approximately 68% of theoretical density by cold pressing in a tungsten carbide die at 175,000 psi. Pellets of each carbon content were sintered for 2 hours at 1450°C in purified argon and hydrogen atmospheres. Density measurements on the sintered pellets indicate that sintering in hydrogen results in a higher density than sintering in argon. Chemical analyses of the sintered pellets is now being carried out to determine the extent of carbon removal.

Microscopic examination of the argon sintered samples failed to reveal whether the carbon was present as a uranium carbide or as graphite.

The argon sintered pellets were corrosion tested in degassed water for 24 hours at 650°F and 2600 psi pressure. Contrary to expectations, no visible change occurred after this time. The samples have been returned to the autoclave for further treatments. The hydrogen sintered pellets are now being corrosion tested.

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Silica Doped UO_2

Powders containing 500, 1000, 2500, and 4000 ppm of silica were prepared as described in WAPD-MRP-56. These powders were pelletized by cold pressing in a tungsten carbide die at 175,000 psi, and the pellets were sintered in purified argon and hydrogen atmospheres for 2 hours at 1450°C. In contrast to the results obtained with the carbon doped specimens, density measurements indicate that an argon atmosphere is superior to hydrogen atmosphere for the production of higher density pellets. The reason for this action is not yet understood.

Polished sections of the argon sintered specimens were examined with a metallographic microscope. Grains of silica were not visible.

Corrosion tests were carried out at 650°F and 2600 psi for 24 hours in degassed water. The argon sintered pellets suffered no visible change. Pellets which were sintered in hydrogen are now being corrosion tested.

Sintering of UO_2

Study of the relative importance of the various factors affecting the sinterability of UO_2 has continued.

Powder Characteristics

A sedimentation fractionation-microscopy technique was tried to determine the size distributions of the various powders being used. Apparently satisfactory initial dispersions were obtained using a hypersonic generator; however, the powder flocculated during sedimentation and the fractionation was incomplete. Fractionation will be attempted in other dispersing agents and mediums as well as in more dilute suspensions.

X-ray line broadening studies of powders being used indicate that, with the exception of high pressure steam oxidized UO_2 , the crystal sizes are too large to be measured by this technique. While MCW powder oxidized to $\text{UO}_{2.1}$ was not significantly different from the starting material, a preparation of MCW powder oxidized to $\text{UO}_{2.2}$ showed splitting of some of the lines indicating the presence of the tetragonal structure.

A comparison made between the X-ray spectrometer traces of the (200) and (111) peaks of as-received MCW powder and a 70% theoretical density cold pressed compact made from this powder showed no significant difference in the breadth of the lines. However, this does not exclude the possibility that some fracturing of the UO_2 crystals does occur during pressing since the initial size of the crystals (1-10,000 Å) is much larger than the upper limit of X-ray line broadening sensitivity.

Pressing Characteristics

The results obtained from a study of the effects of particle size, oxygen-to-uranium ratio and pressure on the as-pressed density indicate that (1) the bulk density of the powder has little relation to the pressed density, (2) the size fraction of MCW powder has no marked effect on the pressed density, and (3) there is a considerable difference in the as-pressed densities of the pellets produced from powders prepared by various methods. Although

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the effect is not marked, it also appears that the as-pressed density decreases somewhat with increase in oxygen-uranium ratio.

Stability of UO_2 in Water

Long term tests on the stability of UO_2 compacts in water have continued. Specimens have now been exposed for 52 days in degassed water at 650°F , 45 days in degassed water maintained at a pH of 10.5 at 650°F , and 58 days in degassed steam at 750°F . In general, the appearance and weights of the compacts have not changed significantly.

Several hydrogen sintered compacts, to which PVA and Sterotex had been added prior to pressing, were exposed to each of the above conditions for 10 days. The compacts appeared unaffected by the exposures. Chemical and spectrographic analyses of duplicate samples indicate that the addition of Sterotex and PVA does not increase the level of impurities in the hydrogen sintered compacts. The following concentrations of impurities were found: 0.1% U(VI), 0.002% carbon, 0.0002% silicon, 0.005% water, and 0.002% fluoride.

The test in degassed water containing 25 cc H_2/kg has been discontinued since no differences have appeared between compacts tested under this condition and compacts tested in degassed water.

Surface Area and Density of UO_2 Powders

Surface area and density measurements have been performed on 3 ammonium diuranate (ADU) powders in an effort to understand the effect of preparative method on the nature of the resultant UO_2 powder.

The ADU powders were prepared from uranyl nitrate by precipitation with either ammonium hydroxide (ADU-13), urea (ADU-14) or ammonium carbonate (ADU-17). The surface area and density values obtained for the 3 powders are $72.4 \text{ m}^2/\text{cc}$ and 5.06 g/cc for ADU-13, $28.2 \text{ m}^2/\text{cc}$ and 5.15 g/cc for ADU 14, and $31.0 \text{ m}^2/\text{cc}$ and 5.66 g/cc for ADU-17.

The ADU prepared by precipitation with ammonia has the largest specific area, the lowest density and, on the basis of X-ray measurement, the smallest crystallite size. This is consistent with the fact that precipitation with ammonia occurs rapidly, very likely forming a finely divided, poorly crystallized powder. Precipitation by the other two methods occurs slowly, yielding a material of larger crystallite and particle size and smaller specific surface.

Although both the ADU-14 and ADU-17 powders were prepared by methods involving slow precipitation, microscopic examination shows the ADU-17 to be composed of uniform spherical particles whereas the ADU-14 particles are somewhat distributed in size and shape. This probably indicates that the ADU-17 is a more carefully crystallized powder than the ADU-14 and contains fewer closed pores and amorphous regions--a conclusion which is borne out

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by the greater density of the ADU-17. The similarity in surface area may indicate that the average particle size of the ADU-17 is smaller than that of the ADU-14.

The above results will be compared with data obtained on the 3 derivative UO_2 powders.

Thermal Conductivity of UO_2

Approval has been obtained from the Atomic Energy Commission for continuation of the work at the Armour Research Foundation on the measurement of the thermal conductivity of UO_2 at high temperature. Work will be resumed at Armour immediately.

The measurement of the center and surface temperatures of UO_2 compacts while under irradiation has been delayed pending receipt of the thermocouples from the supplier. These are expected by next month.

Sintering Additives

It has been found that small additions of U_2O_5 enhances the sinterability of UO_2 in the presence of sulfur and steam. However, cerous nitrate still appears to be the most effective additive for improving UO_2 sinterability.

UO_2 Porcelains and Cermets

The study of the best possible materials for use in connection with the cermets and porcelains program being carried out for Bettis Plant by Battelle Memorial Institute has been completed. Cermets studies will be confined to the addition of silicon, silicon nitride, and beryllium, initially, whereas the porcelain studies will include work on bodies composed of UO_2 and oxides such as BeO , TiO_2 , ZrO_2 , CeO_2 , and SiO_2 .

The subcontract with Corning Glass Works on the determination of the mechanical properties of UO_2 and their relation to thermal fracture has been approved by the Atomic Energy Commission. Work will commence immediately.

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II-C-1

C. Chemistry and In-Pile Loop Engineering

1. In-Pile Screening Tests of PWR Fuel Materials (53b33) A. P. Miller

a. HB-5 (MTR-WAPD-13) Loop

The WAPD-13 loop in the HB-5 beam hole of the Materials Testing Reactor is being used to determine the effect of thermal cycling upon a clad UO_2 specimen.

On October 3, the normally operating loop coolant pump (Westinghouse Model 30-A) ceased to operate properly. After an investigation, the pump was re-started and continues to perform satisfactorily even though current required is slightly higher than normal. If the pump must be replaced, one reactor cycle of operating time probably will be lost.

All shielding construction and further modification of the purification system for fuel element failure instrumentation were completed prior to the October 9 reactor start-up.

The loop is presently operating with the first UO_2 defected specimen in-pile. It is expected that this test will continue for two to three cycles, after which a second defected specimen will be inserted. The first reactor cycle will be static irradiation with power cycling during succeeding reactor cycles.

b. Shim Rod Loop Studies

A study of the feasibility of an L-42 shim rod facility loop utilizing the WAPD-13 external loop equipment has been completed. A report, "Feasibility of a Shim Rod Facility Loop", was transmitted to the Atomic Energy Commission on October 7.

The study indicates that the following items would be required to use the WAPD-13 equipment for a shim rod loop:

- (1) An in-pile reentrant type pressure tube for insertion in the modified shim rod position.
- (2) A hydraulic cycling mechanism capable of moving the specimens in a manner to simulate power changes in a reactor.
- (3) Upgrading of the present loop to allow operation at a maximum of 600°F and 2200 psig and removal of an additional 100,000 Btu/hr of heat.
- (4) A removal and shipping cask to handle larger sized specimens.

The report describes feasible methods to accomplish the necessary modifications.

II-C-2

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

a. X-1 Loop

The present test of two UO_2 specimens (one defected) began on May 12 in the X-1 loop in the NRX Reactor. The major purposes of this test are: (1) to study the structural stability of UO_2 and (2) to study the resistance of such samples to corrosive environments when irradiated with small defects in the cladding.

The loop has been operating in-pile at specified conditions of flow, temperature, pressure and water chemistry since the NRX Reactor resumed operations on September 9 following the July 20 shutdown.

On September 20 a leak to atmosphere in the high pressure water system developed and reached a maximum of 420 g/hr by September 22. As no radiological or operational hazard existed, normal operation continued. On October 2 the leak was located in a purification system tubing socket-weld and repaired.

On September 29 the normally operating coolant pump (Westinghouse Model 150-C) ceased operating due to current overload. Some difficulty was experienced in re-starting the pump even though the overload heaters were replaced. The pump now appears to be operating properly, but its performance is being followed closely by Atomic Energy of Canada, Ltd. (AECL) reactor operating personnel. In the event of any further evidence of malfunctioning of the pump, it will have to be replaced. The duration of the present test depends upon future performance of the pump and AECL acceptance of its performance.

The system pressure drop has been increasing slowly since the removal of KOH after the July 20 reactor shutdown. In order to maintain the specified flow rate, the control valve must be gradually opened. A KOH resin bed has been installed and will be placed in service in the event the pH of the water must be raised to maintain flow.

Future tests in this system will be directed toward obtaining data on high burnup and high heat fluxes for reference PWR fuel elements and more detailed radiochemistry data. Material and drawings have been received at AECL for the prefabrication of the modifications required to permit more convenient and frequent sampling of piping contaminated by solubles and insolubles separately. It is planned that the modification prefabrication be completed at AECL by October 28. The installation and out-of-pile testing will take place following the November 1 reactor shutdown, pending AECL receipt and approval of the Bettis Plant proposal for the fourth UO_2 defect test in X-1. The present test will be removed from the in-pile position during the shutdown and the irradiated test section prepared for shipment to Bettis Plant. The fourth defect test is scheduled to commence following the November 22 shutdown.

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b. X-3 loop

The combined PWR-LSR test commenced on September 9. A reference-length defected PWR blanket specimen is being irradiated.

The loop has continued successful operation during the present report period. However, the specified loop coolant inlet temperature of 557°F to the test section can only be maintained when the reactor is operating at full power. It is necessary to maintain close observation of the surge tank level during periods immediately following a reactor shutdown and during a reactor start-up.

Termination of the test will depend upon the operating level of activity and future irradiation requirements of LSR projects. It is not planned that future PWR irradiations tests will be run in this loop.

3. In-Pile Proof Tests of PWR Fuel Assemblies (53b35) A. P. Miller

a. NRX-CR-V

The primary purpose of the CR-V test in the Central Thimble Facility of the NRX Reactor is to study the performance of PWR fuel material and fuel bundles under plant operating conditions for long periods of exposure.

The irradiation of these specimens commenced when reactor operation was resumed on September 9.

The heat output from the specimens is 150 kw which is approximately 45% of the design value as calculated from the estimated perturbed neutron flux for the central thimble position. The maximum bulk water temperature is 540°F and the maximum specimen surface temperature is 570°F.

The system has been operating at specified conditions of temperature, pressure, flow and water chemistry since the beginning of the test. Operational performance has been satisfactory. Preliminary difficulty in adjusting the Dowtherm cooling system because of the relatively low heat generation by the specimens has been overcome.

The pressure drop across the in-pile test section has increased since the beginning of the test from 31% to 41% of chart reading. A chart reading of 100% is equivalent to a differential pressure of 800 in. of water.

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4. PWR Chemistry Operations (53b36) P. Cohen

This project is responsible for chemistry analyses and interpretation for in-pile loop experiments.

a. Loop X-1, First Defect Test UO₂ and U-Mo

The report, "Radiochemistry of First PWR Fuel Material Test - X-1 Loop NRX Reactor", WAPD-CPM-1, is in reproduction.

b. Loop X-3, First Defect Test, U-Mo Alloy

Analysis of the ion exchange resin that was on stream throughout the tests which ran from March 11 to July 20, is nearly complete. The activities of the isotopes investigated, corrected back to the end of the test, were in the following decreasing order: I^{131} , Sr^{89} , Zr^{95} , Ba^{140} , Ce^{144} . (Analysis for Cs^{137} is not yet completed). The resin bed contained a considerable quantity of crud. Radiochemical analysis of the separated crud showed the activities to be in the same relative order as found on the resin except that Sr^{89} was greater than I^{131} ; fission product activity associated with the crud was about 1/4 of the total fission product activity found on the bed.

c. Loop X-1, Third Defect Test, UO₂

Two more resin beds from this test have been received and analyzed for specific radioisotopes. These resin beds, which were on stream for equal times, had just about the same amount of the various isotopes on them as did the first resin bed from this test.

Following the start-up on September 9, the steady state delayed neutron activity in the loop water has been about 100 c/sec, and the gamma monitor has risen from about 500 - 1000 c/min. Both of these values have shown a steady increase since the test began in May. The gross beta-gamma activity in the loop water preceding the ion exchange bed ranged from 1.9×10^5 to 9.4×10^5 dis/min-ml with an average of 4.2×10^5 dis/min-ml; following the ion exchange bed the values ranged from 1.8×10^4 to 5.7×10^5 dis/min-ml with an average of 1.8×10^5 dis/min-ml. The maximum Kr^{88} and Xe^{133} activities thus far observed in the loop water were 3.5×10^6 and 2.3×10^8 dis/min-ml, respectively.

The KOH ion exchange bed was replaced by an MB-1 bed on September 14 and the loop has since operated at neutral pH.

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d. X-3, Second Defect Test, UO_2 and U-Zr

This test was started with the reactor start-up on September 9. The test section contains one defected UO_2 and one defected Zr-U alloy fuel element. The loop is operating without purification. The steady state delayed neutron activity in the loop water has been 20 - 25 c/sec, and the gamma monitor 200 c/min. The gross beta-gamma specific activity in the loop water ranged from 1.9×10^5 to 2.8×10^6 with an average of 7.2×10^5 dis/min-ml. Delayed neutron peaks at start-up have been very pronounced, to as high as 1×10^5 c/sec. Loop pH is very sensitive to reactor operation, holding steady at about 5.5 during operation, and rising to 6.5 - 7 after shutdown. Nitrate is undetectable, but dissolved CO_2 has been found.

e. Central Thimble (CR-V)

This proof test of PWR type subassemblies, began with reactor start-up on September 9. Activities have been low, about 4×10^4 dis/min-ml before the filter, 4×10^3 dis/min-ml after the filter, and less than 2×10^2 dis/min-ml after the ion exchanger.

5. PWR Radiochemistry (53b37) P. Cohen

This project provides radiochemical assistance for PWR projects.

A simplified procedure was developed for the analysis of fission products in acid pickling solutions. Iron is first removed by extraction, and Cr is then oxidized, to prevent interference with subsequent analyses for Sr, Zr, Ce and Cs. Completion of the new radiochemistry laboratory facilities alleviated the previous overcrowding. A total of 153 determinations were performed for 10 isotopes. Reports covering 110 determinations were received from Nuclear Development Corporation of America under subcontract 73-(14-398).

6. Fission Product Removal Process Development (55b12) W. T. Lindsay, Jr.

a. Examination of Contaminated Facilities

(1) Bettis Loop R-1

Autoradiographs and Parlodion film stripping of the Bettis Loop R-1 piping following disassembly of the loop (WAPD-MRP-56) indicated some preferential activity trapping in blemished areas of the inside pipe surfaces such as deep scratches and circular rings.

Radiochemical analyses of the pickling solutions obtained to determine relative nuclide concentrations as a function of activity levels throughout the loop have not been received as yet.

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(2) Bettis Loop R-3

Bettis Loop R-3 was contaminated to some extent during the R-3 phase VIII test of a cold-pressed UO_2 wafer-type element with multiple clad defects. Loop surveys and autoradiographs of Harlodion films stripped from the removable test section indicated deposition of radioactive particles. Radiochemical analyses of the pickling solution from the test section have not yet been completed.

(3) STR Mark I

Additional decontamination experiments with contaminated hydraulic service system piping from STR Mark I are reported under Project 55b17.

b. Distribution of Activity Between Filterable and Non-filterable Phases

The full stream bed filter to be utilized in conjunction with full-stream filter studies on activity release from ruptured fuel elements in Bettis Loop R-3 (WAPD-MRP-55) has been received from the manufacturer. The pressure vessel housing has been welded in the Bettis R-3 loop facility, which has been modified for accommodation of the filter unit. The bed filter will be subjected to preliminary testing with and without crud additions to the loop in order to determine pressure drops at various flow rates and filtering capacities.

The filter is designated for use in NRX-X-1 during the latter part of November depending upon preliminary test data obtained in the Bettis Loop R-3. Accordingly, testing with ruptured fuel elements will be limited by the available time for scheduling the unit for in-pile service.

The Bettis R-3 facility is presently in operation with a dummy plug insert in order to evaluate leakage across the core seat of the bed filter unit.

c. Leaching of Activity from UO_2

No Bettis R-Loop tests on this topic have been active during the reporting period. The results of autoclave tests are reported under Project 53b30.

d. Erosion Experiments with UO_2

Additional long term erosion studies have been continued in Bettis Loop R-2 (150°F; WAPD-MRP-56, p. 73). Negligible erosion rates were found for UO_2 pellets exposed to a high velocity stream (28.7 fps). The only marked evidence of erosion was noted on the leading edge of the wafer peripheries. Based on the total exposed area, an erosion rate of $2.4 \text{ mg/dm}^2\text{-hr}$ was obtained for one pellet exposed for approximately 1025 hr. Other pellets exposed for shorter periods of time (161 and 291 hr, respectively) were found to yield erosion rates of the same order of magnitude.

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One pellet was fractured intentionally around the periphery and projected into the coolant stream in order to test an extreme case representing a fractured UO_2 fuel element. Based upon the total exposed area, an erosion rate of $4.0 \text{ mg/dm}^2\text{-hr}$ was obtained for a total testing time of 425 hr. It was noted, however, that preferential erosion occurred at the fractured areas of the periphery.

It appears from the above tests that the observed appearance and weight changes of the various wafers do not indicate serious material release from exposed UO_2 fuel elements due to erosion by the coolant stream.

e. Behaviour of Particulate UO_2 in Turbulent Streams

In order to check the reproducibility of deposition rate measurements reported previously (WAPD-MRP-56), a duplicate run using irradiated fuel corrosion product particles has been made. The average deposition rate was 0.26% per pass for the entire Bettis Loop C-1, or 0.13% per pass for each of two drain tees, which comprised the major traps. By measuring the gross activity of the injected solids and comparing this with the activity held on filter discs and the solids discharged from each drain tee during draining, a satisfactory activity balance was obtained. Both drain tees were found to have trapped approximately the same amount of solids. The deposition rate during the above run was nearly four times that found during the previous run. The high value may have been the result of a momentary loop flow interruption which occurred during injection. Additional runs are planned for the near future.

The applicability of results from experiments employing small diameter piping to prediction of effects in the FWR main coolant piping cannot be determined with certainty without further experimentation. A larger diameter, low-pressure loop is being constructed for the purpose of studying scale-up effects. Replaceable loop sections of various diameters will provide the necessary flexibility.

f. Ion Exchange Process Studies

Experiments are being carried out with the object of determining the decontamination factors and capacities obtainable when high purity water containing carrier-free radioisotopes is processed in a mixed-bed resin at an area flow loading of 7.5 gpm/ft^2 . The isotopes Ce^{144} , Pr^{144} , Cs^{137} , Ba^{137} and Ru^{106} have been used so far. The solutions of the isotopes, received as chlorides in hydrochloric acid, were diluted with demineralized distilled water by a factor of 20,000 to produce a solution of approximately 1000 to 3000 c/min-ml. The experimental columns were 10.5 mm ID and had a capacity of 60 ml wet settled resin.

The experiments are not yet complete. Results to date are as follows:

Cs^{137} - Decontamination factors greater than 100 have persisted for 802 column volumes with no evidence of break through.

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~~Ce¹⁴⁴-Pr¹⁴⁴~~ - Decontamination factors over 100 were obtained for 400 column volumes. The removal factor then dropped to 65 and remained at this figure at 678 column volumes with no additional tendency to break through.

~~Ru¹⁰⁶-Rh¹⁰⁶~~ - Incomplete removal observed in the first few column volumes.

Solutions which had been diluted with Pittsburgh tap water showed much earlier break-through of cesium and cerium, as might be expected.

Bettis Loop R-1 is to be rebuilt as an ion exchange and instrumentation test facility. It will consist of a small, high-velocity system with parallel purification units. It will make possible demineralizer evaluation with isotopes in chemical forms which are as similar as possible to those expected in the FWR purification stream. All components required for the reconstruction of this loop are now on order.

7. FWR Fuel Element Rupture Studies (55b13) P. Cohen

Work on this will be resumed in March 1956 when T-Loop will become available.

8. FWR Coolant Development (55b15) W. T. Lindsay, Jr.

a. Adsorption of Activity on Particulate Matter in Water Suspension

Experiments of this nature have been temporarily suspended until adequate data are obtained to show which of the fission products are sufficiently sorbed on stainless steel at FWR conditions to warrant the use of such additives.

b. Contamination Studies

Studies have commenced on the sorption of carrier-free radioisotopes on precorroded stainless steel surfaces. Information concerning the degree and rate of sorption of the various fission products from aqueous solutions at FWR conditions is necessary for a complete evaluation of plant contaminability and purification system effectiveness.

The procedure consists of exposing a known quantity of tracer solution to a precorroded type 316 stainless steel rocking autoclave and sampling the solution periodically. The autoclaves are precorroded for 100 hr with water at 540°F before adding the tracer solution. Samples are taken from the autoclave by means of calibrated stainless steel bombs. The contents of the bombs and two HNO₃ rinses of the bombs are evaporated to dryness in the counting pans. After equilibrium is established at room temperature, the autoclave is heated to a higher temperature and additional samples taken as above.

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A number K can be defined as the ratio of the concentration of a radioactive isotope per unit area of stainless steel surface to the concentration of the isotope per unit volume of solution. Values of the ratio K are tabulated below.

<u>Isotope</u>	<u>Temperature °F</u>	<u>K-value (cm)</u>	<u>Solution Activity at Equilibrium (c/min-ml)</u>
Cs ¹³⁷	Room Temperature	0.13 ± 0.01	2300
Cs ¹³⁷	540	0.04 ± 0.03	2500
Cs ¹³⁷	540	0.02 ± 0.01	4700
Sr ⁹⁰	Room Temperature	0.026 ± 0.008	4400
Sr ⁹⁰	325	0.19 ± 0.01	3800
Y ⁹⁰	Room Temperature	0.28 ± 0.05	3700
Y ⁹⁰	325	14.7 ± 2.2	250

The values for Sr⁹⁰ and Y⁹⁰ are being checked since some difficulty was encountered in obtaining reproducible samples. Similar experiments are in progress with Ce¹⁴⁴ and Ru¹⁰⁶.

c. Pretreatment of PWR System Materials

Experiments of this nature have been temporarily suspended until adequate data are obtained on contamination studies. A knowledge of the sorption of fission products on stainless steel surfaces should aid in selecting pretreatment solutions for the steel surfaces.

9. PWR Decontamination and Waste Disposal (55b17) W. T. Lindsay, Jr.

a. Decontamination Experiments

(1) Simplified Decontamination Procedures

Work on this phase has been temporarily interrupted by reassignment of engineers. Radiochemical analyses of ion exchange resin and section of loop piping removed from Bettis T-Loop following the second low pH, high oxygen test described previously (WAPD-MRP-56, p. 76) showed an unfavorable distribution of fission products between the system surface and the ion exchanger. Additional experiments are planned with simultaneous treatment by complexing agents to prevent contamination by the released fission products.

(2) Bettis Loop R-3

After pickling of the removable test section of the Bettis Loop R-3 facility for activity species analyses following the R-3-VIII test with a multi-defected UO₂ wafer-type element (WAPD-MRP-56, p. 73), the loop was decontaminated with 0.2% periodic acid followed by 2% citric acid. As was noted in previous loop tests in Bettis Loops R-1 and R-2, the citric acid solution specific activity was much larger than the specific activity of the periodic

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pretreatment. In the R-3-VIII run, the specific activity of the citric acid (approximately 1.4×10^6 γ dpm) was 6.3 times that of the periodic. An overall decontamination factor of approximately 2.2 (corresponding to the citric acid run) was indicated by the loop survey readings. It should be noted that the general activity in Bettis Loop R-3 was low in comparison to the level found in Bettis Loop R-1 (Phase R-1-III, WAPD-MRP-51) for which an overall decontamination factor of 4 was obtained with a periodic-citric decontamination.

(3) STR Mark I Hydraulic Service System Piping

The equilibrium exchange of activity between decontamination solution and a pipe wall was examined by allowing an STR HSS pipe coupon to remain in contact with a 0.2% periodic acid solution for 50 hr at 160°F. At this point, the pipe wall activity was no longer decreasing. The contaminated acid was replaced with fresh solution, and the pipe wall activity was found to drop significantly within 15 hr, giving an additional decontamination factor of 2.1. The experiment will be repeated using aged solutions to determine whether or not acid decomposition was partially responsible for the observed effect.

(4) Chalk River Loop X-3

Unlike the results obtained with STR pipe surfaces, prolonged treatment with periodic acid and citric acid gave very little decontamination of piping removed from Chalk River Loop X-3. Although the activity present was largely due to activated iron and cobalt, as in the STR samples, the oxide film, formed at 550°F, is evidently much more resistant to attack than that formed at 450°F. Sulfamic acid, on the other hand, appears to descale the higher temperature oxide surfaces provided temperatures of at least 250°F are maintained during treatment. The minimum contact time is approximately 20 hr.

(5) Bettis B-Loop

The chemical decontamination of this equipment has been completed. The combined decontamination factor for 22 hr treatment with 0.1% periodic acid and 20 hr with 2.0% citric acid was 34 for an average temperature of 160°F. Since descaling was necessary, a subsequent 18 hr treatment with 1.0% sulfamic acid at 270°F was used. The loosened scale was then removed by gentle swabbing, followed by a water spray rinse. The overall decontamination factor for the entire sequence of cleaning operations was about 200. The final surface activity, as determined by smear counting, was less than 200 c/min-ft².

b. Waste Disposal

(1) Disposal of Radioactive Gases

The method of disposal of fission product gases from PWR will involve either retention of the gases in solution during the requisite decay period, or

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steam stripping of the gases followed by storage an/or slow discharge to the air via a stack. In the first case, simple methods to prevent diffusion of Xe^{133} from the liquid into the vapor zone of the holding tanks will be required. Experiments have been planned to determine the rate of xenon diffusion from a bulk liquid surface into the air above, both with and without floating seals present. The simplest floating seal under consideration is a "Microballoon" layer several inches thick. Other types will be investigated if necessary.

The rate of trace gas removal by steam stripping will be compared to the corresponding rate of bulk gas removal in simple batch-type experiments. Information of this type will permit the design of a suitable steam stripping unit for processing of the primary coolant discharge, should this equipment be necessary.

The equipment necessary for the above investigations is nearly completed at present.

(2) Ion Exchange of Decontamination Solutions

A 2% citric acid solution which had been used for decontamination of Bettis Loop R-1 following the R-1 Phase III test was passed through a bed of MB-1 resin. A decontamination factor of more than 10^3 was obtained for 50 column volumes with the citric acid solution having a pH of 2.3 and an initial activity of 10^6 c/min-ml. Diluting the samples with water by factors of 6 and 24 did not increase the decontamination factor.

A solution of 0.2% periodic acid which had been used during the same decontamination cycle was treated in the same way. A decontamination factor of 10^3 was obtained for 24 column volumes of solution having a pH of 1.9 and an initial activity of 10^4 c/min-ml. The factor dropped to 13 after 24 column volumes and remained at approximately this figure for 50 additional column volumes. When diluted by a factor of 12 the decontamination factor has been constant at 2×10^2 for 120 column volumes. To determine the dilution effect, approximately 320 column volumes will have to be processed for comparing data.

These tests will be continued when adequate shielding of the columns has been provided.

D. Reactor Physics

1. PWR Core and Control Physics (52all) S. Krasik

Work has been proceeding using the two group two-dimensional UNIVAC code developed by Dr. Cuthill. The results for the axial flux distributions reported in WAPD-MRP-56 were considered unsatisfactory for thermal design purposes. For this reason various schemes of programming the rods have been studied. The most favorable scheme appears to be a rod program in which eight rods are down completely at hot, clean criticality, while the remaining sixteen rods are almost completely withdrawn. The eight rods down are the four corner rods and the four rods at the center of the flat sides of the annulus. The eight rods down control about 8 to 9% in reactivity. This is supplemented initially by having the remainder of the rod bank in somewhat. The calculated peak-to-average radial flux factor in the seed for eight rods down is 1.37 (exclusive of water holes). An estimated axial factor used in conjunction with this is 2.0 although this has not yet been verified. The blanket fraction of the power is 53% and the blanket peak-to-average flux is 2.26. In this program the four rods at the center of the flat sides are lifted as a group leaving the four corner rods down. With only the corner rods down the radial peak-to-average flux in the seed drops to 1.24 (exclusive of water holes). Finally, when the four corner rods are lifted the seed peak-to-average flux factor drops to 1.13.

Several other programs have been calculated. In one of these alternate rods are presumed to be fully down while the remaining twelve rods are fully withdrawn. The radial peak-to-average power in the seed rises to 1.46. The blanket power fraction was 55% for this case. Several cases have been calculated for a so-called "expanded design". In this design the seed annulus is moved out one row increasing the number of seed clusters to 32. The number of blanket subassemblies is increased to 117 from 93. The flux factors obtained are similar to those obtained for the reference design for comparable cases.

A code (ABNER) for the IBM-650 has been specified, programmed and coded for the solution of few-group diffusion problems in slab, spherical and cylindrical geometry (one-dimension). The internal memory capacity is sufficient to permit 100 points distributed in 10 regions of different group constants. The code is intended to perform the spatial integration using group constants prepared by MUFT-III, and is thus limited to a maximum of eight lethargy groups.

Studies of the resonance escape probability in TRX lattice configurations have continued with results obtained from both Monte Carlo methods (executed by Dr. Richtmyer at NYU) and the approximate energy dependent cell technique reported in WAPD-MRP-55. Although analysis of the Monte Carlo results is not complete, comparison with TRX measurements of resonance escape probability indicates that roughly 1/3 of the capture may occur at energies above 450 ev. It further appears that these higher energy resonances contribute substantially to the surface absorption.

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For neutron energies between 0.5 Mev and 10 Mev, the average cosine of the scattering angle and the average logarithmic energy decrement per collision have been evaluated for carbon and oxygen using the optical model of the nucleus. The calculations are prompted by the large amount of carbon in the PWR Flexible Critical Experiment and by the uncertainties in the detailed angular distribution inferred for oxygen. General expressions for the transformation from center of mass to laboratory system have been obtained by a simpler method than that of Zweifel and Hurwitz (J. Apl. Phys. 25, 1254).

2. PWR Shielding Physics (52a13) S. Krasik

The preliminary analysis of the distribution of the fission product gases in the PWR system has been completed in cooperation with the Chemistry group considering two methods of release. The first method of release considered was the continuous release of fission products into the primary coolant through a fracture in the cladding of a uranium oxide fuel element. The second method was the instantaneous release (disintegration) of one complete element into the coolant water. The specific activities resulting in the primary coolant and reactor plant container were computed for both cases.

The calculated gross fission product activity in the primary coolant resulting from the instantaneous release of one fuel element is as follows: A short time after release the activity is 5×10^5 $\gamma/\text{cm}^3\text{-sec}$ of energy 2.5 Mev and 4.5×10^6 $\gamma/\text{cm}^3\text{-sec}$ of energy 0.8 Mev. After a period of 1 day the specific activity will be approximately 6×10^4 $\gamma/\text{cm}^3\text{-sec}$, assuming no removal of the zirconium by ion-exchange methods.

The first part of the two-part code HYDRA, which is a mathematical formulation of the core corrosion-fission product contamination problem, is near completion. The formulation has been generalized to include three conceivable modes of core corrosion which are characterized by rates of introduction of fission product contaminants into the primary coolant which are: (1) proportional to the quantity of contaminant present in the core, (2) proportional to the fission rate, and (3) instantaneous at some point in time after the reactor has been brought up to power. The complete code will account for isotopes which can be treated as first members of fission product decay chains and also for isotopes which are daughters of first members. The first part of HYDRA deals only with those isotopes which can be regarded as first members of decay chains. Thus far, the results of the code for simplified cases in which deposition and re-solution of contaminants have been ignored are in agreement with parallel hand calculations. Barring unforeseen difficulties, it now appears that the checkout of part I will be completed by the end of October.

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

Work on the highly enriched slab and metal natural uranium blanket has continued. The blanket has been built out to a thickness of 12 in. on each side of the seed. The excess reactivity of the assembly with the 12-in. blankets was estimated to be 12% using the old form of the inhour equation which does not take into account the ages of the delayed neutrons or fast fission

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effects. The adjusted value taking these effects into account is expected to be at least 15% and possibly as high as 20%. The maximum blanket loading in this experiment consisted of six blanket bundles per side, each 2 in. thick. The last row of blanket was worth only 0.25% in reactivity.

A program of rod calibration to determine shutdown characteristics has been completed. It was found that the difference in critical rod bank position for the 5-in. and 3-in. rods was approximately 1 in. The differential rod worth at this point was 1% in reactivity per inch.

Flux plots in both a horizontal and vertical direction along the mid-plane at the different blanket loadings were made. From this, estimated two-dimensional maps through the seed and blanket at the different loadings can be obtained. The reactor was made critical with reduced water height and without control rods in the moderated section. Flux traverses were taken for this condition. From these plots it was possible to determine the vertical reflector savings and thus determine the effective critical height of the reactor. Plots of critical height versus differential water worth have been made. The integrals of these curves cross-check the integrated rod worth curves at the various loadings.

A temperature coefficient determination was performed at the full blanket loading over the range of 28 - 40°C in three-degree steps. The temperature coefficient at 28° is $-1 \times 10^{-4}/^{\circ}\text{C}$; it becomes more negative with increasing temperature.

The oxide seed and blanket slab experiment was loaded and approaches to criticality made starting October 13. The slab was critical with eight seed clusters and a 6-in. layer of blanket on one side. Additional oxide has been added and experimentation is proceeding.

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

A 1.3% enrichment uranium oxide (UO_2) lattice has been built up to criticality. This lattice has a 3-to-1 (actual 3.05-to-1) water-to-uranium volume ratio (calculated as though the uranium in the UO_2 were metal of density 18.9 gm/cm³). The hydrogen-to-uranium atomic ratio is 4.25 and the water-to- UO_2 volume ratio is 1.07. The fuel rods are made up of pellets 0.600 in. diam and about 0.5 in. high which have been compressed to 68.77% of the theoretical density of 10.95 gm/cm³. These pellets are stacked to a height of 48 in. in 0.028 in. wall thickness aluminum tubing in the same manner as the 0.600 in. diam metal fuel.

The number of fuel rods required for criticality is 1280 ± 0.5 . This number represents an effective core radius of 41.43 cm (16.31 in.). The pellets have a hydrogen content of 100 ppm due to the addition of an organic agglomerating agent to make it possible to press the pellets on a production press. The relative high hydrogen content creates some uncertainty in the criticality calculations.

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The temperature coefficient of this lattice has been measured from room temperature to 50°C and found to be in the range of -1.0×10^{-4} to -1.2×10^{-4} per °C. This is considerably more negative than the corresponding water-to-uranium volume ratio lattice of the 0.600 in. diam metal fuel and indicates that the 4-to-1 lattice of the UO_2 fuel will also have a negative temperature coefficient at room temperature.

Work is progressing on measurements of reflector savings, migration area, thermal utilization, fast fission effect and resonance escape probability.

5. Long Term Reactivity Gain Experiment (53b32) S. Krasik

The study of available information concerning long term reactivity changes continues. A trip to the National Reactor Test Station is planned in the immediate future for the purpose of discussing experiments involving irradiations of sample sections of PWR fuel elements in the Materials Testing Reactor, with period reactivity measurements in the Reactivity Measurement Facility. During the visit calibrations of the Reactivity Measurement Facility by Dr. G. Hopkins will be observed, and the possible usefulness of chemical or mass-spectrographic analyses of fuel composition will be discussed with personnel at the Chemical Processing Plant.

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