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WAPD-MRP-59

REACTORS—POWER

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UNITED STATES ATOMIC ENERGY COMMISSION

PRESSURIZED WATER REACTOR PROGRAM
TECHNICAL PROGRESS REPORT FOR THE
PERIOD DECEMBER 2, 1955 TO JANUARY 12,
1956

Price \$ 0.60

Available from the
Office of Technical Services
Department of Commerce
Washington 25, D. C.

Westinghouse Electric Corporation, Bettis Plant
Pittsburgh, Pennsylvania

Technical Information Service Extension, Oak Ridge, Tenn.

CLASSIFICATION CANCELLED

DATE 2-7-57

For The Atomic Energy Commission

H. F. Canale

Chief, Declassification Branch *md*

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TECHNICAL PROGRESS REPORT

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PART II

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IVSUMMARYA. Systems Design

1. A proposal was submitted recommending a major revision in the Power Station Manual Program. (p.12)
2. Drawings and specifications for the combination neutron shield tank and reactor vessel support were submitted to vendors for bids. Drawings and specifications for the Guniting of the reactor plant container were approved for construction. (p. 3)
3. The system description and system schematic for the following fluid systems were submitted to the Atomic Energy Commission for approval during this report period.
 1. Fuel Element Failure Detection System (1-AP-8036) (p. 7)
 2. Radioactive Waste Disposal System (1-AP-8043) (p. 7)
 3. Safety Injection System (1-AP-8053) (p. 8)
 4. Service Air System (1-AP-8016) (p. 6)
 5. Control Air System (1-AP-8055) (p. 8)
4. The feasibility study of a control system capable of handling the fixed roller nut mechanism, or the collapsible rotor mechanism, or a combination of both, has been approved. (p. 8)
5. The Reactor Protection System Design Description was completed. (p.10)

B. Power Plant Components

1. The 18 in. motor operated prototype valve has been installed in the AED pump test loop and will undergo engineering tests during this period. (p.13)
2. Relief valve test facility is being designed by Stone & Webster to conform with the desires of the American Gas and Electric Service Company. (p.14)
3. The Westinghouse prototype main coolant pump has completed all tests satisfactorily. (p.14)
4. Two piping specifications, one for stainless steel and the other for general piping (carbon steel, cast iron and non-ferrous piping), were prepared by Stone & Webster and submitted by Bettis Plant to the Atomic Energy Commission for approval. (p. 3)
5. Bids were received and the order placed for the Nuclear Instrumentation System. (p.17)

SUMMARY (Contd.)

6. Bids were received on the Operational Radiation Monitoring System and the order will be placed at the beginning of the next report period. (p. 17)

C. Materials, and Rod Control Mechanisms

1. Torch brazed joints of gold and gold-copper alloys show adequate corrosion resistance to hydrogenated water. Testing is continuing in 3 service waters. (p. 18)
2. A report on the results of load-deflection tests on Haynes 25, Refractaloy 26, and S-816 springs is being prepared. Inconel X and Haynes 25 wire for torsional modulus and relaxation tests is on order. A 30-day corrosion and relaxation screening test of Ni-Span-C springs is complete. (p. 18)
3. The final selection of one or more types of control rod drive mechanisms is to be made February 21, after completion of further core design analysis. The manufacturing information was released for one production prototype of a fixed collar nut mechanism and a collapsible rotor mechanism. (p. 19)
4. The life test of the type XRN-21A prototype mechanism is approaching the design goal of 1400 cycles with no scram failures. (p. 19)

D. Reactor Design

1. During this period, the major effort of this Subdivision has been concentrated on preparation of the PWR Reference Design Report. It is now anticipated that the report will be issued by January 15, 1956. Design details of all reactor components, with the exception of blanket assemblies instrumented for power shift, are discussed in this report. (p. 21)
2. Blanket fuel rod tolerances have been established and will be submitted to the AEC for approval about January 15, 1956. A manufacturing release of UO_2 pellets through the steps of agglomeration, compacting and sintering was given on December 16, 1955, as per schedule. (p. 22)
3. The preliminary system description for the fuel handling and fuel servicing system was reviewed by representatives of the other reactor project organizations, and, after revision, the system description was submitted to the AEC for approval. (p. 26)
4. Studies made during this period disclosed that a variety of steady state power characteristics can be obtained without increasing local boiling merely by varying the manner in which the core blanket is initially orificed. (p. 26)
5. The reactor vessel shell welds have been radiographed using the 15,000,000 volt betatron. Preliminary examination of the film has disclosed that there are areas to be repaired. (p. 30)

SUMMARY (Contd.)E. Materials Development & Core Manufacturing

1. The manufacture of 124 fuel bundles for the Physics flexible experiment was completed on December 14, 1955. This included the 113 needed for the experiment plus 11 spares. (p. 46)
2. Agglomeration and compaction of UO_2 powder for the core production has commenced. (p. 45)
3. X-ray spectrometer analysis of a thermally transformed and irradiated U-9 w/o Mo sample has definitely confirmed that irradiation causes transformation of the low temperature alpha plus epsilon phases to the corrosion-resistant high temperature gamma phase. While X-ray evidence of irradiation-induced phase reversal is not available for U-10 w/o Nb samples, changes in properties (density, electrical resistivity and hardness) are similar to those observed for the U-Mo system and indicate that transformation to the high temperature corrosion resistant gamma phase also occurs in this alloy during in-pile irradiation. (p. 59)
4. All three experimental aluminum-base alloys being investigated as possible future cladding materials have survived exposure to 680°F water from 7 to 14 weeks. However, these same alloys failed when exposure to 680°F steam for 6 to 11 weeks. (p. 63)
5. The fourth test of a defected UO_2 specimen in the X-1 loop reactor was terminated because of fusion of a specimen resulting from excess heat generation. (p. 56, 70, 71)
6. Tests of a radial-flow bed filter in the Bettis R-3 Loop have been completed with satisfactory results. The filter has been shipped to Chalk River, Canada, for use in the NRX-X-1 Loop. (p. 73)
7. Additional particle deposition experiments have indicated the need for more detailed examination of the effect of water conditions on particle agglomeration and sedimentation rate. (p. 74)
8. The application of decontamination procedures which do not involve draining of the contents of loop systems has been pursued at an accelerated pace. Cyclic, two-step treatments with very dilute solutions of periodic and citric acids have been shown to be moderately effective in transferring radioactive material from a slightly contaminated system (Bettis Loop R-3) to an ion exchanger. (p. 79)
9. Gas stripping experiments conducted in batch lots by boiling under reduced pressure have shown excellent correlation between the removal from water solution of dissolved oxygen in macro-amounts and dissolved xenon in trace amounts. (p. 79)

VIISUMMARY (Contd.)F. Physics

1. The two-dimensional flux distribution studies have been extended to the 12 and 16 rod down conditions. (p.82)
2. The lifetime studies of the reference design have been carried to 4000 hr in 1000 hr steps. The reactivity change from hot, clean, criticality to the 4000 hr equilibrium xenon condition is 12%. The blanket power fraction increases to 3000 hr and then starts decreasing. (p.82)
3. A comparison in one-dimension of criticality and flux distribution, using 100 points in the lattice instead of 29, shows essentially no change in criticality or power distribution. (p.82)
4. Calculation of radiation heating in the thermal shields and pressure vessel wall for the reference design has been completed. (p. 83)
5. Furfuryl alcohol experiments were completed on two slabs, one bare and one blanketed. Rather small reactivity changes were observed. (p.84)
6. The 3/1 oxide lattice at 69% density has been completed. (p.84)

SUMMARY (Contd.)G. 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

Main Coolant System

Total number of loops**	4
Steam generators per loop**	1
Pumps per loop**	1
Number of loops operating at full power**	3
Flow per loop at full power	6.5 x 10 ⁶ lb/hr
T _h	541.4°F
T _c	508.6°F
Steam generator design**	
Steam output at full power (total)	861,000 lb/hr
Steam pressure at full power	600 psia
Steam side design pressure	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 x 10⁶ 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

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SUMMARY (Contd.)CORE PARAMETERSGeneral

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	13.9 tons
Weight of natural uranium	14.1 tons
Weight of U ²³⁵	69.3 kg
Mean diameter of core	207.5 cm
Height of core without fuel supports	183.0 cm

UO₂ Region

Full length	183 cm
Active length - cm	164.5
Bundles per subassemblies	7
Number of subassemblies	113
Number of rods	95,711
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	8.53 ft ²
Heat transfer surface	7983 ft ²
Equivalent diameter	0.021 ft
Weight of zirconium - including fuel supports	7.93 tons
Weight of natural uranium	14.1 tons
Power	356×10^6 Btu/hr

Seed Region

Length	183 cm
Fuel element	Plate
Plate thickness	0.069 in.
Clad thickness	0.015 in.
Channel thickness	0.069 in.
Plate width (active)	2.05 in.
Number of plates	1920
Number of clusters	32
Weight of U ²³⁵	69.3 kg
Weight of zirconium	4.50 tons
Flow area	2.21 ft ²
Heat transfer surface	3930 ft ²
Equivalent diameter - D _e	0.0129 ft
Power	355×10^6 Btu/hr
Weight of zirconium flux depressors	1.43 tons

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I-A-1PART 1PWR EngineeringA. Power Plant Analysis & Systems1. Plant Functional Design and Operation Analysis (55all) J. E. Nolan

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

Simulator Studies

The PWR Primary Coolant Volume Surge Report based on the June and September simulator runs was issued. The principle results of these runs were reported in WAPD-MRP-57.

Hazards Studies

Investigations have been made of various means of removing decay heat from the core to prevent a meltdown and/or a zirconium-water reaction if a primary coolant system rupture partially or completely uncovers the core. Heat removal by conduction, radiation, and convection were considered, including thermal radiation down the flow channels. Satisfactory cooling appears to occur only if the water level remains above the center of the active portion of the core, when decay heat below the water line may evaporate sufficient steam to dissipate the decay heat above the water line.

Calculations of the pressures in the plant container resulting from a zirconium-water reaction should a primary system rupture cause a core meltdown have been completed. The results indicate the peak pressures in the container under various conditions may be as follows:

Per cent of all zirconium in core assumed to react	<u>20%</u>
Hydrogen burns, no container spray	80 psig
Hydrogen does not burn, no container spray.	52.8 psig
Hydrogen burns, 2400 gpm container spray	52.8 psig

(Hydrogen is a product of the zirconium-water reaction).

A 20 per cent complete reaction is based on recent calculations on zirconium oxidation kinetics by the Metallurgy Department which are to be substantiated by experimental data to be obtained in the next six months.

In order to delay or prevent uncovering the core under certain accident conditions that may cause the reactor pressure vessel relief valves to open

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for an extended period, the removal of these relief valve connections from the reactor inlet pipe to the reactor outlet pipe is being considered. This will eliminate a 6-in. penetration of the inlet piping, reducing the probability of a pipe rupture in this location. It will also eliminate considerable 6-in. pipe and its associated primary coolant volume.

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 construction and installation of the reactor plant.

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

An arrangement study was made to locate a contamination monitoring loop in each of the purification systems. In order to satisfy the requirement that sections of the loop may be cut out during plant operation, the most feasible location appears to be within a small auxiliary shielded cubicle in the purification cubicle.

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

Stone & Webster Engineering Corporation study drawings for internal shielding of the boiler and auxiliary chambers have been received and are being reviewed for shielding of major components. The valve operating cubicle shielding is being developed concurrently with the cubicle arrangement and will be given a final check when the cubicle arrangement drawings are complete.

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

The system design description and schematic drawing covering miscellaneous utilities within the reactor plant container, auxiliary equipment room No. 1, and the auxiliary power room were submitted to the Atomic Energy Commission (AEC) for information.

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

It has been determined that the reactor plant container need not have an overall pneumatic test, provided that each individual chamber is tested separately.

The following sequence of testing will be used:

1. The boiler chambers will be tested separately with the 8 ft diam and 41 in. diam interconnections capped on the boiler chamber end of the interconnections.
2. The auxiliary chamber and the reactor chamber will be tested together, having left the caps on the boiler chamber end of the 8 ft and 41 in. diam interconnections.

This method of testing the reactor plant container was adopted to permit the installation of the Guniting in the boiler chambers while the Pittsburgh-Des Moines Steel Company is erecting the auxiliary and reactor chambers.

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

Drawings covering those portions of the systems within the plant container are being released to Dravo Corporation for procurement and shop fabrication of the piping.

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

Preliminary waste disposal layouts have been examined for shielding requirements, based upon criteria previously approved for the plant, to aid in determining the most economical arrangement for the waste disposal plant.

Neutron Shield Tank & Pressure Vessel Support (S.O. 1-AP-8046)

Drawings and specifications for procurement and shop fabrication of the neutron shield tank and pressure vessel support were submitted to the Atomic Energy Commission (AEC) and concurrently issued to bidders. Three bids were received and are being evaluated.

Piping - General

Two piping specifications, one covering stainless steel and the other general piping (carbon steel, cast iron, and non-ferrous material), were prepared by Stone & Webster Engineering Corporation and submitted to the AEC for approval.

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

Construction drawings have been issued showing the details for the following walls of the concrete enclosure: the north walls of the east and west compartments, the east wall of the east boiler compartment and the west wall of the west boiler compartment. These drawings permit the completion of the above-mentioned walls up to elevation 726. The intent is to have these walls constructed while the Pittsburgh-Des Moines Steel Company is erecting the boiler chambers of the reactor plant container, thereby eliminating the difficulties of forming which would be encountered if these walls were constructed after the completion of the boiler chambers.

Construction drawings have been released for the footings of the Service Building.

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I-A-2 (Contd.)

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

In a meeting at which representatives of the Atomic Energy Commission (AEC) Stone & Webster Engineering Corporation and Bettis Plant were present, it was decided that the ventilation system for the Service Building would include air conditioning for the following areas:

1. Radio Chemistry Laboratory
2. Sample Preparation Area
3. Health Physics Laboratory
4. Dark Room
5. Counting Rooms

Stone & Webster issued a specification covering the following cranes which are required in the Fuel Handling and Service Building:

1. Fifteen ton clean room crane with a half ton auxiliary hoist
2. Twenty ton decontamination room crane
3. One ton core vault crane
4. Three ton sample preparation room crane

Construction drawings have been released for the structural steel framing of the Fuel Handling and Auxiliary Equipment Structures.

3. Primary Coolant Fluid Systems (55b11) C. E. Sinclair

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 following PWR Fluid Systems have been approved by the AEC.

Reactor Coolant System	Decay Heat Removal System
Pressurizer and Pressure Relief System	Coolant Sampling System
Coolant Purification System	Valve Operating System
Chemical Addition System	Core Removal Cooling System
Component Cooling Water System	Reactor Plant Air Cooling System
Coolant Charging System	Steam Piping System
Coolant Discharge and Vent System	Boiler Feed Piping System
	Radioactive Waste Disposal System

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

An analysis of Reactor Coolant System performance based on increased reactor coolant flow rate was completed. The results of this analysis, which was necessitated by modification in core design, is now being discussed with pump manufacturers to determine the ability of existing pump designs to meet the new head-flow requirements.

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Pressurizer and Pressure Relief System (S.O. 1-AP-8006)

The possibility of relocating the reactor vessel relief valve lines from the reactor vessel inlet line to the reactor vessel outlet line was evaluated. It appears that this may be accomplished without serious difficulties while enabling the water volume in the core to absorb much more heat before relief valve discharge uncovers the core.

The internal standpipe in the pressurizer was increased from 100 to 125 in. to accommodate an increase in system surges. These increased surges are the result of incorporating the effect of pressure coefficient in the latest simulator studies.

The use of pressure switches to indicate bellows failure on system relief valves was recommended to the Atomic Energy Commission (AEC).

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

Approval by the AEC to locate a contamination monitoring station in each coolant purification loop was received. In resolving the comments made with this approval, Bettis Plant has incorporated in the design of the station a vertical and horizontal run of test piping. This allows determination of preferential deposition. In addition the monitoring station has been relocated to a position in the inlet piping common to each demineralizer.

The AEC has withheld approval of the palladium hydrogen gauge until receipt of further information confirming the effectiveness of the device. A letter has been submitted furnishing this information.

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

The elevation of the primary water storage tank was increased 10 ft to provide sufficient suction head for the fill and charging pumps during periods of low water level and high temperature in the storage tank.

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

Work is currently in progress to:

1. Develop a method for eliminating the hydrogen-air explosion hazard in the flash and blowoff tanks.
2. Develop a method for restoring the blowoff tank to normal operating conditions within a reasonable time so that it may receive a second relief valve discharge. This scheme must be compatible with the scheme selected for elimination of the explosion hazard in the flash and blowoff tanks.

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

An investigation was made as to the feasibility of using the steam dumped during decay heat removal operations for operating an emergency steam driven

generator. It was concluded that this method of operating the emergency generator was not reliable.

Coolant Sampling System (S.O. 1-AP-8012)

The Sampling System description and schematic piping diagram is in the process of being revised to include the 3 sample trains. Each sample train will include instream monitoring instrumentation and reactor coolant sample removal vessels. The system is scheduled for submission to the Atomic Energy Commission (AEC) on January 16.

Chemical Addition System (S.O. 1-AP-8014)

The maximum working temperature and pressure of the hydrogen injection vessel and the hydrazine addition vessel were increased in order to make these design conditions agree with those for the charging system.

The system description is being revised for submission to the AEC. This revised system description will include all revisions made in the system since initial approval.

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

A means of venting the Valve Operating System piping during filling was approved by the AEC. This scheme utilizes a line run to the auxiliary container access cubicle from the highest point of each of the lines between the heating coils and the main cylinder valves. These lines will slope continually upward to the access cubicle and will terminate with globe valves.

Inclusion of a Safety Injection System in PWR will require some revision to the Valve Operating System. These modifications, particularly the necessity of operating another hydraulic valve, are presently being investigated.

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

The system description, schematic diagram, and all necessary equipment specifications for this system have been submitted to the AEC for approval.

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

Stone & Webster Engineering Corporation is revising the system description to include the latest heat removal requirements for the main coolant pumps. New system requirements will be based on cooling water requirements for the proposed reactor coolant flow increase. The system description and schematic drawing are

also being revised to include two dual service heat exchangers which will serve both the Component Cooling Water System and the Canal Water Cooling System.

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

A study to assure a sufficiently rapid closure of the Air Cooling System butterfly valves by utilizing a pressure surge signal is continuing. The size and height requirements for the exhaust stack are, also, still being investigated.

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

The system description and schematic were submitted to the Atomic Energy Commission (AEC) on December 30. The system design is the same as that described in WAPD-MRP-58. Shop orders have been written for all components in the system.

Feed Water Piping System (S.O. 1-AP-8040)

The Feed Water Piping System has been approved. Bids from vendors of valves have been reviewed and vendors have been selected. The Feed Water Piping System description, as prepared for the operating manual by Duquesne Light Company, has been reviewed.

Main Steam Piping System (S.O. 1-AP-8041)

The Steam Piping System has been approved. Duquesne Light Company and the Pittsburgh Area Office have approved the use of the reverse current valve in the system. Bids from vendors of valves have been reviewed and vendors have been selected. The Duquesne Light Company steam piping layout drawings have been reviewed.

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

The system design description and schematic diagrams were approved by the AEC on December 29.

In order to meet the tight procurement and construction schedule for this system, Bettis Plant has initiated preparation of the equipment specifications for long delivery items such as storage tanks, ion exchangers and heat exchangers. Since system approval has been received procurement of this equipment has been initiated. The remaining equipment will be obtained under a contract covering procurement and installation of the entire Waste Disposal Facilities.

The spray recycle tank cooler has been eliminated since any cooling requirements can be met by establishing a conservative design for the waste liquid cooler.

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

The system description is being revised to indicate the new location for the centrifugal pumps and to show definite pump characteristics. These characteristics

were originally specified on the basis of estimated heat transfer coefficients for the non-regenerative heat exchangers. The vendor of these exchangers has furnished data on which definite characteristics of the pumps will be specified.

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

The system description and schematic diagram for the Safety Injection System were submitted to the Atomic Energy Commission (AEC) for approval on December 22. A new study of this system description is now being made based on the assumption that the Duquesne Light Company boiler feed pumps will not be available for use in injecting cooling water to the reactor vessel.

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

The system description, schematic diagram, and all necessary equipment specifications for this system have been submitted to the AEC for approval.

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

In order to reduce the cost of the fuel storage vault, the canal water storage tank has been relocated from the roof of the vault to a position west of the vault. This has necessitated minor revisions in piping and other associated system auxiliaries.

4. Non-Nuclear Control Systems Design (55el2) W. P. DiPietro

This activity includes control and instrumentation systems directly concerned with the parameters of the Primary Plant Systems and their operation.

The Rod Control System is reported in this section since the major portion of the design work is the integration of a number of electrical components into a system.

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

The feasibility study report of a control system which is flexible enough to control the fixed roller nut mechanism, the collapsible rotor mechanism, or a combination of both, together with a recommended course of action was approved by the AEC.

A system description written on the basis of the study report will be issued during the next report period and submitted for approval. Equipment specifications will be prepared concurrently with the system description.

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

Equipment specifications for the components of the system are being prepared.

Bids on the reactor console have been received, evaluated and forwarded to Purchasing for action.

Bids on the primary plant auxiliary control board are due on February 8.

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

Work is progressing on the functional layout of the auxiliary primary plant instrumentation panel. Work on the Core Instrumentation and the fuel element failure instrumentation will be reported under S.O. 1-AP-8057 in this and all subsequent reports.

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

The revised D-C Distribution System was submitted to the Atomic Energy Commission (AEC) for approval. This system forms a part of the Reactor Plant Electric Power Distribution System and will be combined with it when it is revised and reissued. A reissue of the Reactor Plant Electric Power Distribution System is expected by the week of February 6.

Preparation of equipment specifications for the load centers will be completed and issued during the month of January.

Core Instrumentation

The functional layout of a core instrumentation panel, based on available preliminary functional requirement information, is in progress.

A control scheme for the multiport valve of the Fuel Element Failure Detection System is also in progress.

It is expected that all the control and indication for the core instrumentation and fuel element failure equipment will be located on one panel in the auxiliary area of the main control room.

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

This project covers control system design work for those systems directly related to the performance of the reactor, its protection and monitoring. Radiation Monitoring Systems are also included.

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

At a meeting with the AEC on December 16, the objectives of the Controlled Steam Relief System, a part of the Power and Temperature Control System, were reviewed and agreed upon. It was agreed that the system should have, as its principal objective, the maintenance of pressure surges within the limit set by the pilot-operated relief valve settings for any load reduction that might occur which is larger than the design values. Therefore, a system very similar to the one previously submitted as a design study is expected to be employed. This system employs a signal obtained from a balance circuit in which a fast resetting load signal is compared with a slow resetting load signal. The resulting output is used to operate 1 to 4 pilot-operated relief valves on the main steam header, and to close them in sequence to permit the load on the reactor to be

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reduced by steps within the design limit. Simulator studies made during this period show that standard size relief valves can be selected to provide suitable steps so that the primary plant pressure surge as each valve closes can be kept to approximately 150 psi.

A further study is being made of a system in which a single pilot-operated relief valve with a throttling valve ahead of it would be employed. The throttling valve would be continually positioned by the slow resetting load measuring circuit. This system, while it would be simpler in the relief valve equipment employed, may not be able to cover all cases of load reduction. A final system design is expected to be completed by the fourth week of January.

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

A proposal was forwarded to the Atomic Energy Commission (AEC) to incorporate a "decades per minute" meter in the system as an adjunct to the period metering. The objective is to obtain a trial of this type of meter to learn operator preference. The scale of the meter is linear.

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

At a series of meetings with the AEC, the principles of this system were established and the system design completed and will be forwarded the week of January 16. As completed, the system provides rod insertion signals by normal programming for turbine shutdowns caused by faults in the main generator or its two connected transformers, and for periods on startup below 15 seconds. Both of these are non-lock-in signals, although the first is effectively so, since the circuitry in the generator protection scheme is such that it would not be expected to be reset before the reactor was shut down.

Shutdown by released rod action is provided for (1) loss of coolant flow as indicated by less than 2 loops in operation at any time, (2) nuclear level above approximately 140% of 4-loop full power at any time, (3) loss of two out of three 120-v a-c control voltage circuits to the Nuclear Instrumentation System, after a delay of approximately 1/2 sec, (4) loss of d-c control voltage to the Reactor Protection System after a delay of approximately 1 sec, (5) hot leg temperature above approximately 555°F or core fuel element (seed) temperature above approximately 1050°F, (6) low pressure (approximately 1600 psi) except on cold startup, (7) nuclear level above 120% of 4-loop power capability when 3 loops are in operation, (8) nuclear level above 98% of 4-loop power capability when 2 loops are in operation, and (9) nuclear level above 70% when 2, 3 or 4 loops are in operation with the pumps at one-half speed.

The last 3 shutdown signals provide backup protection at a lower nuclear level for startup accidents when flow is reduced. Protection against inadvertent operation above plant capability caused by excessive turbine throttle opening is provided by utilizing the 120%, 98%, and 70% nuclear level trips in association

with 4, 3 and 2 loops respectively to effect a normal rod insertion until the nuclear level is below the trip point. A fifth nuclear level limit of 35% operates to provide insertion to protect against inadvertent excessive throttle opening when the pumps are operating at one-half speed.

Reactor Plant Monitoring Systems (S.O. 1-AP-8024, 8029, 8030)

The Operational Radiation Monitoring System Design Description (1-AP-8024) was revised and reissued to cover changes agreed upon with the Atomic Energy Commission (AEC), principally the change in Channel 5 from pipe monitors to area monitors. No system design activity took place on the Safety Radiation Monitoring System (1-AP-8029). This system is complete and procurement of equipment is proceeding. The Remote Viewing System design description (1-AP-8030) was revised and reissued to change the monitor from a single receiver to two. One will be a 12-in.-screen monitor, panel mounted in the chief operator's communication panel, with the camera control equipment located in the same place. The other will be a 21-in.-screen monitor on a turntable so that it will be available for visitor viewing. The controls also incorporate provisions for picking up a signal from the fuel handling remote viewing cameras.

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

The simulator program during December included studies of power-flow override, pressurizer operation, and steam dump operation. The effect of rod rate, initial power level, and temperature and pressure coefficients of reactivity were studied to determine power-flow override settings. Pressurizer operation was studied for 3 and 4 loop flow conditions. The steam dump simulation is made up of valves providing for controlled steam dump to permit the primary plant to remain in normal operation during large load reduction. Three steam dump schemes were used. Steam dump studies were conducted under conditions of changing feed water enthalpy, different initial power levels, and with and without control system pressure override.

7. Reactor Plant Installation and Test (59all) C. E. Langlois

This project provides for the necessary coordination and general supervision required among various contractors and subcontractors for the installation of the nuclear portion of the power plant.

The progress of general construction is illustrated in Figs. 1 and 2. These photographs show the general construction of the reactor and turbine generator portions of the power plant and the erection of the east and west boiler chamber steel.

The boiler chamber and auxiliary chamber enclosure walls are rapidly approaching their full height. Of an estimated 22,700 cu yd of concrete in the reactor portion of the power plant, 16,775 cu yd have been poured.

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Fig. 1 - View of Construction of the PWR Power Plant, December 27, 1955

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Fig. 2 - Plant Container Erection, West Boiler Compartment, December 27, 1955

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The shop fabrication of the east and west boiler chambers is very well advanced. The shop fabrication of the reactor chamber and auxiliary chamber is proceeding according to schedule.

The field fabrication of the east and west boiler chambers is proceeding but not as rapidly as was scheduled. Efforts are being made to improve this situation.

The construction of temporary facilities has been started for the use of the Westinghouse Site Organization and the Westinghouse installation subcontractor.

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

This project covers the coordination of the Power Station Manual Program, the program by which test descriptions are prepared for use during the testing and operational phases of station development, and the issuance of a status report on developmental tests of interest to PWR.

A proposal was submitted recommending a major revision in the Power Station Manual Program. It is proposed that, in lieu of Volumes II, III, and IV, written information be given the Duquesne Light Company under the following categories: system descriptions, site description, operating procedures, decontamination procedure, component removal and replacement procedures and core handling and refueling procedures.

B. Power Plant Components and Component Materials and Tests1. Primary Coolant Valves (55b19) A. P. Zechella

This project includes the technical direction required for the development, design, manufacturing follow, tests, and installation of all the primary coolant valves in the PWR Plant.

18 in. Prototype Capped Manual Stop Valve

This valve is being installed in the test loop at the Westinghouse Atomic Equipment Department and will undergo tests in accordance with T-Spec 551926. Testing is expected to begin January 30.

18 in. Motor-Operated Stop Valves

Approval of the assembly and detail drawings was received from the Atomic Energy Commission (AEC). Full manufacturing release was given to Crane Company. T-Spec 550263 for production testing of the valves was submitted to the AEC for approval.

18 in. Hydraulically Operated Stop Valves

Welding procedures and revised detail drawings were approved, and full manufacturing release was given the vendor, Westinghouse Atomic Equipment Department. T-Spec 550260, covering the production testing of the valves, was approved by the AEC.

18 in. Check Valve

Welding procedures were approved and manufacturing release given to the vendor, Edward Valves, Inc.

Pilot-Operated Pressurizer Safety Valve

Technical approval of the design was granted by the AEC. Welding procedures were approved and manufacturing release was given the vendor, Manning, Maxwell & Moore, Inc.

Main Relief Valves

Crosby Steam Gage and Valve Company submitted designs for prototype valves meeting E-Specs 565267, 565268 and 565269. These designs were approved subject to comments regarding the selection of materials and weld configurations. Approval of the designs was granted by the AEC. The vendor was authorized to proceed with detail drawings following incorporation of comments in the revised assembly drawings, to proceed with the procurement of materials for all valves plus 50 per cent excess material for those parts subject to change as a result of tests on the lead valves, to proceed with the machining and assembly of 2

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valves of each type for test, and to proceed with the machining of parts for the balance of the valves except on those parts which may be subject to change after tests of the lead valves are completed. An assembly drawing of each design will be submitted to the Atomic Energy Commission (the AEC) for formal technical approval by January 31.

Formerly it was proposed to construct a full-scale mock-up of the valve for the self-actuated reactor vessel relief valve (E-565269), assemble these parts in an existing valve body and obtain design information by tests. However, this proposal was abandoned for two reasons: first, there is at present no relief valve test facility available to simulate the operating conditions for this valve; secondly, the change in the bellows requirement from a low pressure commercial type to a high pressure developmental item dictates the advisability of constructing the prototype valves of the same materials as those which will be used in the production valves in order to resolve welding problems which might arise in connection with the bellows subassembly.

Relief Valve Test Facility

Stone & Webster Engineering Corporation completed a preliminary design for this facility. A meeting was held with American Gas & Electric Service Corporation (AG & E) personnel at which time many areas of uncertainty were resolved regarding the means of obtaining water and/or steam at the desired test conditions, and the method of disposing of the effluent. Stone & Webster is preparing a new flow diagram and is revising specifications for heat exchanger equipment in line with agreements reached at this meeting. Contractual negotiations are still in progress with AG & E. A meeting is scheduled for January 16 with AG & E to discuss the new flow diagram and heat exchanger specifications.

2. Primary Coolant Pumps (55b20) A. P. Zechella

This project covers the technical direction and coordination necessary for the design, development, manufacturing follow, test, and installation of the primary coolant pumps in the PWR Plant.

Testing of the Westinghouse prototype pump was completed, and the pump disassembled and inspected. No evidences of malfunction or excessive wear were found. Welding procedures for the Westinghouse pump were approved, and manufacturing release given.

The plan list for the Allis-Chalmers Manufacturing Company pump was forwarded to the AEC for approval. The outline drawing as received by Bettis Plant was incomplete, and returned to Allis-Chalmers for revision.

3. Auxiliary Pumps (55b21) A. P. Zechella

Vent System Discharge Pump

A change of the head requirements in 2 of the 4 pumps, and granting of

manufacturing release for all 4 pumps except for the volute which will be constructed of carbon steel, is expected to enable Westinghouse Atomic Equipment Department to meet the scheduled delivery date. This pump will be a standard model 150-D Canned Motor Pump with modifications to the impeller and volute to adapt it to the service requirements of PWR Discharge System.

3000 psi Air Compressor and Accessories

The specification has been revised and approved by the Atomic Energy Commission (AEC) to include 3 high temperature alarm switches connected in parallel to an alarm circuit.

Charging System Pumps

Outline drawings and plan lists for the hydrostatic test pump were received on January 3 and are scheduled to be delivered January 17 and February 1 for the charging pumps and centrifugal fill pump, respectively.

4. Primary Coolant Loop (55b26) R. F. Devine

This project includes the technical direction and coordination required in the development, design, analysis and manufacturing follow of the main coolant loops.

Primary Piping (S.O. 1-AP-8005)

The fabricator for the 18 in. primary piping, Pittsburgh Piping & Equipment Company, has submitted shop drawings of elbows and an offset fitting located at the pump discharge in the Foster Wheeler loop.

The supplier of the pipe, Babcock & Wilcox, has placed mill orders for the billets from which the pipe is produced.

A final stress analysis of the primary piping is being prepared covering all design conditions.

5. Steam Generators (55b32) A. P. Zechella

This project covers the technical direction required for testing, manufacturing follow, and installation of the PWR steam generators.

Minor drawing revisions are being made in accordance with detailed comments from the Stone & Webster Engineering Corporation and the Duquesne Light Company. Babcock & Wilcox shop errors are also being shown on the drawings in order that they represent the unit as built.

6. Reactor Plant Auxiliary Pressure Vessels (55b33) A. P. Zechella

This project covers the technical direction required for design, manufacturing follow, tests and installation of the auxiliary pressure vessels in the PWR Plant.

A contract has been placed with the Robco Manufacturing Company for the terminal connectors for replaceable heater elements. The plan list and outline drawing for replaceable heaters have been submitted to the Atomic Energy Commission (AEC) for approval.

The Flash and Blowoff Tanks

A purchase order for these tanks was placed with Combustion Engineering Co.

Canal Water Filters

The specification for this component was sent to AEC for approval.

7. Reactor Plant Electrical Components (56dl2) W. P. DiPietro

Vendor quotations for the motor-operated speed transfer switches for the coolant pumps were received. They were rejected on the basis that the alternate equipment proposed was not desirable. The specification was resubmitted for bids, and quotes have been received. The bid evaluation was completed and forwarded to Purchasing for action.

The specification for the 480 v a-c unit substations have been sent out for bids and will be sent to the AEC for information on the basis that commercially available equipment will satisfy the specification.

Specifications for the rod drive motor generator sets, the rod drive inverter type power supplies, and the reluctance motor type position indicators have been issued and approved. Procurement on all 3 items has been initiated.

8. PWR Non-Nuclear Instrumentation (56dl3) W. P. DiPietro

Nine additional specifications for instrumentation equipment associated with the control of the primary plant have been approved, subject to comment, and added to the 22 in the process of procurement.

Revisions required to these 9 specifications will be made prior to the placement of the order.

Bid proposals have been received on the 31 specifications and are being evaluated.

Thirteen equipment specifications were prepared using manufacturer's model numbers for identification, so that procurement can proceed on the basis of standard commercial items.

9. PWR Nuclear Instrumentation (56dl4) N. E. Wilson

This project provides the engineering services for the preparation of equipment specifications, procurement and test follow for nuclear instrumentation.

The contract for this equipment was awarded to the Westinghouse Electronics Division on December 15.

10. Radiation Monitoring Equipment (56d15) N. E. Wilson

This project provides the engineering service for the preparation of specifications and procurement of monitoring equipment.

Operational Radiation Monitoring Equipment

Proposals from prospective bidders were opened on December 28. All of the 6 vendors submitted bids. They are being evaluated and the order will be placed early in the next report period.

Safety Radiation Monitoring Equipment

The first trailer-contained mobile monitoring station, being built by Tracerlab, is scheduled for January 13.

Specifications for the Beta Gamma Count Rate Units and Cart Mounted Air Particle Detectors were approved by the Atomic Energy Commission (AEC).

Specifications for the Beta Gamma Detector-Indicator Channel and Counting Room Equipment were submitted for approval.

These 4 equipment specifications are being submitted to prospective bidders for quotations.

Bids have been requested for construction of 3 river effluent sampling stations based on Bettis Plant drawings. Pumps for these stations have been ordered from the Harris Pump and Supply Company. These stations will form part of the preoperational survey equipment.

TV Remote Viewing Equipment

The specification for the PWR Remote Viewing System was submitted to the AEC for approval on December 23. It has also been submitted to prospective bidders for quotations.

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C. Materials, Control Mechanism and Tests1. Non-Core Materials (53all) J. R. HunterSubstitute for Microbrazed

The 3 oxy-hydrogen torch brazed specimens were sectioned longitudinally and examined at 100X and 500X substantiating the examinations made at 10X and 20X. All brazed joints showed adequate corrosion resistance to 600°F hydrogenated water after a 4-mo period with no corrosion attack being observed on the 10-mil wall tubing at either end of the 3 brazing flow specimens.

Maximum pitting of 1 mil in the 24K gold, 1 to 3 mil in the 80% gold - 20% copper, and 2 mil in the 50% gold - 50% copper braze joints was of similar magnitude to the pitting observed in four 80% gold - 20% copper braze joints tested in hydrogenated water at 500°F for 3-1/2 months. Testing will continue under closely controlled conditions in 3 service waters.

One 7-day exposure period in oxygenated water was completed on specimens of Rexweld 64 and Coast Metals Company Alloys Nos. 50, 52, 53, and 57. The final revision of the test specification is complete. All tests in these series are scheduled for a 6-mo exposure.

Spring Material Development

Load-deflection tests of the Haynes 25, Refractaloy 26 and S-816 springs are complete. Data are being evaluated and a report will be written during the next reporting period.

New Inconel X wire will be provided by the supplier for torsional modulus testing. Two of 3 heats of Haynes 25 wire needed for torsional modulus and relaxation tests were ordered. Delivery is promised during January.

Corrosion testing of Cobenium springs is scheduled to start in February.

A 30-day corrosion and relaxation screening test of Ni-Span-C springs is complete and a report is being prepared.

Tests to determine the optimum aging treatment of Haynes 25 wire were delayed temporarily. Testing will be resumed by February 1 following completion of higher priority tests.

Study of Elastic Properties of Potential Bearing Materials

The compression properties of Stellite 3, 6, 12, and 19 will be determined by January 31. Suitable cold-worked Haynes 25 alloy specimens are being procured.

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2. Materials for Primary Loop Components (55b16) J. R. HunterWeldability Tests for Castings and Forgings

Meetings were held during December with representatives of Bettis Plant and vendors of valves and pumps to resolve any difficulties being encountered with interpretation of weldability test requirements.

3. Control Mechanism - Design & Test (56c13) J. J. Gertz

Detail drawings for both the fixed roller nut mechanism, RN-27A, and the 2-arm collapsible mechanism, RN-30A, are complete. Manufacturing information was released for one production prototype of each design. Technical support documents for the fixed nut and the 2-arm mechanisms were submitted to the Atomic Energy Commission on December 8 and December 29. The quantity and type of each mechanism to be located on the reactor vessel will be determined February 21 after receiving a detailed study of core scram requirements on February 1.

4. Control Mechanism - Component Development (56c15) T. L. WeybrewMagnetic Brake

The sixth test is complete in the series to develop a suitable magnetic brake to hold a control rod in position when power to the motor is cut off. The 2-pad brake shoe used in this test was similar to that used in the first 3 and the fifth tests except that the pads were slightly farther apart. The shoes were of sintered type 316 stainless steel. The minimum torque available to resist motion of the control rod with the brake shoes against the type 410 stainless steel housing was 1.3 ft-lb. This torque is comparable to the torque produced by a 2-pad sintered type 316 stainless steel brake shoe of different design tested previously, but only about one-half the value obtained with the sand blasted type 317 stainless steel brake shoe. The test results are being evaluated to determine what additional testing is required.

Roller Nut

The fourth and final test is complete in the series to determine the performance of a roller nut and screw of increased size applicable to a PWR control rod drive mechanism. The 17-1/4 PH stainless steel rollers of this test were operated in 500°F hydrogenated water. An evaluation of the test results for the complete series is in progress and will be summarized in the next report.

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

The life test to obtain knowledge of the scram reliability and other operating characteristics of mechanism type XRN-21A (PWR fixed roller nut developmental prototype mechanism) continued for a total of 1360 cycles and 340 scram operations with alternate periods in hydrogenated and oxygenated water. There were no scram failures during this period of operation. The life test will continue for a total of twice the design goal or 2800 cycles.

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A preliminary test report to be submitted January 23 will describe the life test in detail.

The design of an internal position indicator for mechanism type XRN-21A is complete and the parts are on order. This indicator will be tested after the completion of the life test.

The second life test of SRN-10A (fixed roller nut feasibility mechanism) was terminated after a total of 5600 cycles. A final test report will be prepared and submitted by February 1.

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D. Reactor and Auxiliaries1. Mechanical Design of Reactor Core (53a12) P. A. Halpine

During this period, the major effort has been concentrated on preparation of the PWR Reference Reactor Design Report. Design details of all reactor components, with the exception of blanket assemblies instrumented for power shift, are presented in this report. The component discussion which follows is limited to significant progress made during this report period.

Core Cage

The bottom plate design has been established as a grid-type structure with a relatively thin plate as a top surface to provide for seating and sealing of the fuel units. The possible methods of fabrication are the same as for the top grid, i.e., machining from a solid forging and fabricated by the welding of extruded cruciform shapes. Reference designs have been established for the attachment of fuel element failure headers and pressure taps to the bottom plate.

Current effort is to obtain sufficient design agreement between top grid and bottom plate that either plate can be fabricated from identical raw material, i.e., same size forging or same size extruded cruciforms.

The partial grid structure of welded cruciforms, currently under construction for development evaluation, has not progressed as rapidly as anticipated. Fabrication is now scheduled for completion in late January.

Material procurement is not being delayed until completion of this grid. Orders for both forging and cruciform material are now being prepared.

Blanket Assembly

The reference design for the blanket assemblies is illustrated in Fig. 3. Some of the basic features of this design are:

- a. The fuel bundles are supported on a holding spring located in the bottom of each assembly. The spring supplies the holding force for latching the assembly to the core cage.
- b. The length of the shell has been extended below the fuel region to the bottom of the assembly in order to protect the fuel element failure sampling tube from damage during handling.
- c. A bellows has been added to the bottom of the sampling tube to provide a flexible connection with the bottom plate.
- d. A latch locking device has been added to the latch system to eliminate the need for slotting the support grid.

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SECT. CC

PLAN

SECT. B-B

SECT. A-A

Fig. 3 - Blanket Assembly

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REF ID: A60310

Bundle fuel rod tolerances have been established and will be submitted to the AEC for approval about January 15. A manufacturing release of UO_2 pellets through the steps of agglomeration, compacting and sintering, was given on December 16, 1955 as per schedule.

Seed Clusters

The reference design for the seed clusters is illustrated in Fig. 4. Some of the basic features are:

- a. The design is based on accommodating a $3\frac{3}{8}$ in. span control rod. To accomplish this, the holding spring is located in the bottom of the cluster.
- b. A locking sleeve and spring have been added to key the latch ring in the locked position. This device eliminates the need for slotting the top grid. In addition, should a control rod jam during operation and force the cluster away from the grid, the device insures that the latch is maintained in the locked position.

A cruciform shaped flux depressor has been adopted as the reference design since this shape more easily satisfies design requirements and is more easily fabricated than the box structure. The depressor is attached to the bottom plate and makes a slip fit with the top grid. Cooling flow is supplied from the inlet plenum chamber and passes through channels internal to the depressor.

A 100 curie PoBe source is proposed to be located in each of the 4 corner seed clusters. It will be contained in a circular flow passage in one of the cluster spacers. Source insertion into the cluster will be from the side, at a slight angle to the vertical, and will be accomplished prior to installation of the cluster in the reactor vessel.

Control Rods and Shrouds

Design development is proceeding on the basis of a hafnium control rod of $3\frac{3}{8}$ in. span since it has been tentatively established that such rods will provide adequate shutdown although this will not be affirmed until after the seed loading established February 1. This rod span is the largest which permits the removal of control rod and shroud components through the 4-in. mechanism housing penetration.

The reference design of control rod and shroud is presented in Fig. 5. The design applies more particularly to the run-down mechanism with which it is possible to develop high axial loads on the control rod shafting during scram deceleration, due to the rotational inertia of the rotor. To provide sufficient buckling support for the shaft, bearings are located no further than 3 ft apart. To accomplish this, shafting of nearly constant diameter is used and the buffer occupies a fixed position in the shroud.

Fig. 4 - Seed Cluster

Fig. 5 - Shroud Assembly

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

The mechanical testing of reactor components has continued to progress in accordance with the design and test schedule.

The vessel, loop modifications and stand for the hot loop tests have been ordered with delivery of the vessel promised for June 1. The loop and stand will be completed prior to this date. Prototype components and auxiliary equipment are being detailed in preparation for manufacture.

Erection of the mechanical feasibility demonstration assembly is nearing completion. This assembly has been modified to accept a demonstration model of the core extraction tool.

Data have been obtained on deflection, under load, of core cage component scale models and on deformation of blanket assembly, shell and seed cluster when subjected to internal pressure. The data are being analyzed and will be incorporated in test reports.

Components and facilities for the fretting corrosion tests of blanket assembly and fuel rod bundles are being manufactured.

The preliminary PWR Core Assembly and Shipping Procedure was prepared and transmitted to the AEC for information.

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

Core 2 Design Study

Work on this project was postponed for this period in order to concentrate on the Core 1 Reference Design Report.

4. Reactor Specifications (53a15) L. S. Mims

Chemical Shutdown System

Work on this project was postponed during this period in order to concentrate on the Reference Design Report.

Control Rod Functional Requirements

The requirements of the reactor shutdown system during the most unfavorable sequential loss-of-coolant flow accident are being viewed with reference to the revised control rod functional requirements. It is expected that this work will be completed during the next reporting period.

5. Reactor Core Instrumentation (53a16) L. S. MimsFuel Element Failure Detection Systema. General

A study of the header flow for the Fuel Element Failure Detection System indicated a desirability to provide 11 orifice points along each of the 26 headers at the bottom of the reactor. Depending on the header location, some of these orifice points will control the flow of the effluent samples taken by the sampling rakes and the others will be "dummy" orifices. These dummy orifices will be located at seed cluster positions or blanket positions and their purpose is to provide every header with the same total flow, even though some headers are connected to fewer blanket assemblies than others and each assembly supplies the same amount of sample.

Flow passages machined in the bottom plate adaptors serve as part of the header array for transporting effluent samples from blanket assemblies to the multiport valve. Short "U" tubes connect flow passages between adjacent adaptors to complete the header array on the bottom plate. This header arrangement eliminates much of the tubing that would otherwise be required on the top of the bottom plate and also facilitates assembly of the header array. An orifice insert which fits inside the sampling tube connector provides the proper orificing of flow to each header. Tubing connecting the headers to the multiport valve are contained in two conduits which are attached to the core cage and extend to an instrumentation port in the head.

b. Monitoring Station

The prototype monitoring station will consist of a scintillation counter, the crystal of which will be surrounded by a coil of stainless steel tubing. A sample of the reactor effluent will pass through this coil. The scintillation counter and the coil will be shielded from the ambient room activity by a heavy casing of lead. Fabrication drawings for this prototype reference monitoring station are in progress and will be completed during the next reporting period.

It is expected that this monitoring station will be evaluated in the X-1 loop of the NRX reactor at Chalk River in March. PWR fuel element failure conditions will be simulated by the placement of an artificially defected uranium-oxide rod in the X-1 loop.

c. Sampling Rake

Testing of this device is reported in 55a13.

30d. Header

The equipment for the flow test of the header arrangement was ordered this month, and a general program was set up for testing the arrangement. It was decided to mock up the actual header arrangement proposed for the PWR and test it for cross flow between headers and for the amount of orificing required.

General Reactor Instrumentation (See Fig. 6)

The revised list of reactor instrumentation adapted to the expanded core design was submitted this month to the AEC. This list includes instrumentation to measure seed fuel temperature, seed power distribution and blanket power distribution, both between blanket assemblies and within two individual assemblies. Deleted from the list of instrumentation for Core 1 are pressure vessel and stud temperature measurements and blanket fuel temperatures.

a. Coolant Flow

Flow nozzles integral with the bottom plate adaptor create a differential pressure with low head loss to measure coolant flow into all 32 seed clusters and 22 blanket assemblies. Pressure taps attached to the adaptors are mounted on the upper surface of the bottom plate and rise vertically within three conduits which are attached to the core cage and extend to instrumentation ports in the vessel head.

b. Blanket Exit Water Thermocouple

Thermocouple assemblies supported from fuel ports measure exit water temperatures of the blanket assemblies. Design work is being done on these assemblies so they can be removed from the fuel ports during refueling operations while maintaining electrical continuity between the thermocouples and their terminal connections outside the reactor vessel.

Sourcesa. Primary Source

Four 100 curie PoBe sources mounted at the outside corners of the seed are planned as the primary neutron source.

b. Secondary Source

It is planned to include 20 - 30 pounds of BeO in the core as a secondary source. The BeO would be contained in 339 rods, 3 in each of the 113 blanket assemblies. Investigation of the required specifications on these BeO-containing rods revealed that even an annulus of 3 mils between the BeO pellets and the Zircaloy

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Fig. 6 - Instrumentation Location

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cladding would allow no more than a 600°F temperature of the rods. This temperature is acceptable from a metallurgical standpoint, as no known reactions occur between BeO and Zircaloy at temperatures below 600°F.

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

The preliminary system description for the fuel handling and fuel servicing system was reviewed by representatives of the other reactor project organizations, and after revision the system description was submitted to the AEC for approval.

Subcontract No. 73-(14-426) with United Shoe Machinery Corporation was approved by the AEC. The subcontract as originally approved calls for the design and construction of a fuel assembly handling crane with extraction tool. A proposed Amendment No. 1, which calls for construction of a full-size mock-up of the extraction tool, has been approved by both Westinghouse and United Shoe Machinery Corporation and has been submitted to the AEC for approval. A proposed Amendment No. 2, which calls for construction of a fuel assembly transfer crane with manipulating tool, is being negotiated between Westinghouse and United Shoe Machinery Corporation.

E-Spec 535423-B, which describes the closure seal weld cutting machine, was submitted to the AEC for approval.

The following equipment specifications were prepared and distributed through the Westinghouse and Duquesne Light Company organizations for comment and internal approval:

- E-Spec 566226 - Underwater Optical Equipment
- E-Spec 566227 - Underwater Television Equipment
- E-Spec 566701 - General Purpose Manipulator

8. Reactor Thermal and Hydraulic Analysis and Tests (55al3) L. S. Mims

Reactor Power

More detailed analyses of the reactor steady state performance were made during this period which reflected critical examination of each of the factors involved in the design. As a result of this study, it was found that a variety of steady state power characteristics can be obtained without incurring local boiling by varying the manner in which the core blanket is initially orificed. The advantage of any one orificing scheme over another depends on the extent and nature of power shifts between the seed and blanket, and within the blanket itself, that take place during life because of plutonium build-up. Inasmuch as the drilling of the blanket orificing does not have to be done until December, time is available for further information on the power shift characteristics

within the core. Three cases of interest, nevertheless, have been investigated to indicate the range of performance characteristics available for 3-loop operation with adverse instrument errors of 30 psi and primary system pressure of 5% on inlet coolant temperature. These are as follows:

Case I - In this case, the blanket is orificed to produce maximum reactor power with 630°F hot spot temperature in both blanket and seed early in life. This is the case used to establish the performance characteristics listed in PWR core parameters in the summary. Net power outputs for Case I orificing are listed below:

- 67.0 - without instrument errors and without power shifts
- 62.3 - with instrument errors and without power shifts
- 60.0 - with instrument errors and with 4% shift in blanket power

Case II - In this case, the blanket is orificed to produce just 60 Mw early in life at 630°F hot spot temperatures in the seed and in the low flux blanket regions, Regions 1 and 4 as shown in Fig. 7. With this condition, the hot spot temperature in Regions 3 and 4 are initially at 617°F. A 12% power shift in Regions 2 and 3 would cause the hot spot temperature to rise to 630°F. If, however, the hot spot temperatures in blanket regions 2 and 3, at the time of maximum power shift, were permitted to increase to 636°F, just short of local boiling, then the allowable shift would be raised to 17%.

Case III - In this case, the blanket is orificed so as to produce just 60 Mw early in life with 636°F hot spot temperature in the seed and in the low flux blanket regions, Regions 1 and 4. With this condition, the hot spot temperatures in Regions 3 and 4 are initially at 606°F. For this case, the core could accommodate a 30% power shift in the blanket and still produce 60 Mw.

In computing the above results, advantage was taken of a number of factors not previously assumed in performance calculations and discussed later in this section. The most significant of these are an increase in pumping power believed to be available from prototype tests on the pumps and improved blanket mechanical hot channel factors due to a newly devised statistical approach to hot channel factor computation.

So far as 4-loop operation is concerned, the maximum net power output under Case I is 80.0 Mw without instrument errors and without power shift. With instrument errors but without power shifts, the maximum 4-loop output is 74.4 Mw. In the other cases, where the 3-loop output is given as 60 Mw, the expected 4-loop output would be about 71.5 Mw.

Increased Pumping Power

Tests on the prototype pumps indicated that it may be possible to increase the output of the Westinghouse pumps by as much as 25% and that of the Allis-Chalmers pumps by as much as 10%. This increased pumping power has been considered in the above calculations.

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PWR BLANKET ORIFICE REGIONS

REGION 1 - 21 ASSEMBLIES

2 - 24

3 - 40

4 - 28

SEED (+) 32

Fig. 7

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~~CONFIDENTIAL~~Blanket Hot Channel Factors

A preliminary statistical evaluation of the effect of the separate tolerances on the blanket hot channel factors for pitch and bowing of the individual fuel rods was completed during this period. On the basis of statistical distribution of the flow areas between the fuel rods, as determined by analyses of the effect of rod pitch and bowing tolerances, a considerable reduction in hot channel factors can be effected. Thus, the factor g_{AT} was reduced from 1.30 to 1.05 and the factor g_0 was reduced from 1.40 to 1.10. In this preliminary investigation, certain simplifying assumptions were made regarding the bowing configurations. A more complete investigation not requiring these simplifying assumptions is expected to result in even more greatly improved factors (i.e., g_{AT} about 1.04 and g_0 about 1.06). This investigation is now under way.

A detailed examination of the water flow in the blanket assembly added two hot channel factors and altered a third. A hot channel factor of 1.01 on both g_{AT} and g_0 has been added to account for the loss of flow in heat transfer surface resulting from replacing some fuel rods by failure detection sampling tubes and by BeO rods. A factor of 1.05 has been added to g_{AT} and a factor of 1.04 has been added to g_0 to account for the fact that water flow adjacent to the blanket boxes does not receive the same heating as water flowing inside the bundles. The factor g_{AT} of 1.10 to account for flux peaking in the bundle corner due to water between bundles has been reduced to 1.05 to account for the fact that the flow channel diagonally adjacent to the corner is not fully affected by the flux peak.

Seed Hot Channel Factors

A detailed examination of local conditions in the region of the seed hot spot has led to the recognition of three effects, identified in the calculations as influence factors, which serve to diminish peaking. Since the hot spot occurs in the fuel plate adjacent to an end plate, the effect of the cooler water in the channel on the end plate side is to reduce the over-all factor F_{AT} and F_0 by 0.92 and 0.93 respectively. Furthermore, the hot spot is always found to be located near the edge of the meat. When the conduction along the fuel-less region adjacent to the hot spot is taken into account, a reduction on the over-all factors of approximately 0.95 is found to exist. The effect of the fuel-less metal adjacent to the meat in the seed as a whole is found to be a fin-effect which increases the effective heat transfer surface by a factor of 1.016 or decreases the over-all hot channel factors F_{AT} and F_0 by 0.985.

Steady State Local Boiling

The experimental program at Bettis Plant is proceeding to determine pressure drop and burnout data for PWR temperatures, pressure and velocities. The fluxes are sufficient to give local boiling. It is believed that the results of this program will allow a relaxation of the thermal criteria by permitting steady state local boiling. Some preliminary estimates have been

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Pressure Vessel Flow Study

Orifice to the Blanket Assemblies

Mixing Tests on Blanket Sampling Rake

Control Rod Distortion Study

Thermal Shield Cooling

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Leakage

Calculations were made to determine the water leakage around blanket and seed assembly seals at the bottom support plate. With the 4-loop operation, with a 70 psi pressure differential across the seed, a leakage flow of approximately 500,000 lb/hr, about 2% of the total flow, is expected. This leakage is factored into the over-all cooling flow requirements.

Fuel Rod Tolerances

The maximum Zircaloy-UO₂ interface temperature in the reference fuel rod is 1125°F for the reactor producing 88 Mw net reactor power. This temperature, which is considered satisfactory from the standpoint of low Zircaloy-UO₂ reaction rate, was determined by relaxation analysis based on maximum diametral assembly clearance of 0.008 in.

The effect of flux dipping in the center of the rod due to self-shielding was found to reduce the center temperature significantly. No advantage of this effect has been taken in past calculations.

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

The Pressure Vessel Review Group consisting of one representative each from the AEC, Combustion Engineering, Inc., and Bettis Plant held one meeting during this period.

The PWR reactor vessel bottom head plates have been welded together, Fig. 8, and they are in the process of having the dome section attached. The upper cylindrical vessel course has been machined for the outlet nozzles. The bottom head and cylindrical sections are presently being machined in preparation for welding.

The PWR test head sections have been formed and weld grooves are being machined.

Three new flange forgings have been shipped from the Bethlehem Steel Company and the last flange is scheduled for shipment on January 9.

The 15,000,000 v betatron has been put into operation and the main vessel shell welds have been radiographed. Preliminary examination of the films has disclosed that there are areas to be repaired.

Unclad plate material has been ordered for the final vessel head with penetrations. The protective stainless steel cladding for the inside of the head plate will be applied by weld deposit.

The PWR test vessel shell and bottom head have been finished and have been prepared for joining. The nozzles for this vessel are in place.

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Fig. 8 - "Fit up" of Bottom Head Segments for Welding - PWR Pressure Vessel

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The PWR model vessel with bolted closure (1/4 scale) is presently in the laboratory where strain gages are being applied. Fig. 9 shows the model vessel being welded prior to final machining.

The PWR model vessel with conical segment (1/4 scale) has completed more than 20,000 pressure cycles. About February 15, it will have completed 400,000 cycles after which it will be opened and inspected.

Combustion Engineering has submitted the report on the closure seal weld development. A tubing device is now being made to simulate operating conditions on a segment of the seal, the test program to be finished approximately March 1.

The preferred closure seal is a torus or omega type, designed to be an integral part of the vessel head, and is welded to the head flange by a single weld. The seal may be repaired or replaced after the head has been removed from the vessel. After the vessel head is installed, the seal weld can be made to the vessel flange by means of a 3-pass weld.

A method for attaching control rod mechanism housings and other penetrations to the reactor vessel head has been developed. The housings will be threaded into bosses on the vessel head.

Integral stainless steel sleeves will extend from the housings through the carbon steel vessel head to the stainless cladding on the inside of the vessel head where a seal weld will be made. Thus, the carbon steel and thread portion of the head will be protected from the primary coolant.

A test is presently under way on two designs of threaded penetration attachments. This test will consist of pressure and temperature cycling the attachments, and is expected to be finished by January 30.

The pressure vessel shipment date is now September 24, from Chattanooga.

The photoelastic tests at Brown University have revealed that a 10-in. thickness for the reactor vessel head with penetrations will be satisfactory. The analysis of these photoelastic tests was reported in WAPD-PWR-RD-128. Material for the closure head was ordered and is presently scheduled for delivery the week of March 26.

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Fig. 9 - Welding Vessel Supports - PWR Model Vessel

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PART IIA. Fuel Element Development1. Core Raw Materials and Shapes (53c21) J. GlatterProperties of UO₂

In an effort to reclaim rejected, sintered compacts, controlled quantities of crushed and ball milled sintered material were blended with virgin natural UO₂ powder. The addition of reclaimed powder in amounts ranging from 5 w/o to 30 w/o resulted in a 1.5% reduction in sintered density. For the particular lot studied, the density of virgin material compacted at 100 tsi and sintered for 10 hr at 1725°C was 91%. Addition of the reclaimed powder in the amounts indicated resulted in 89.5% final density under the same compaction and sintering conditions.

The reclamation feasibility of unsintered, rejected 1.3% enriched UO₂ for use in the high density phases of the TRX experiment was established by CO₂ pretreating and ball milling all powder constituents prior to agglomeration. Additives included 0.2 w/o TiO₂ serving as a densifier, 1 w/o PVA serving as a binder and 0.2 w/o Sterotex serving as a lubricant. The conditioned powder was pressed at 80 tsi. A density of 91% was obtained after sintering for 5 hr at 1725°C.

Claddinga. Properties of Seamless Tubes

The creep testing of fuel rods containing different amounts of entrapped water was completed and a report issued. From the data obtained at 700°F and 2000 psi external pressure, it might be postulated that a reference PWR fuel rod containing the maximum amount of entrapped water (6.93 v/o) and held under steady state conditions would fail in approximately 1170 hr and show a diametrical increase of approximately 0.010 in. In general, the time required for burst failure decreases as the amount of entrapped water increases. This relationship would be expected because larger amounts of entrapped water induce higher stresses and, by the fundamentals of creep, higher stresses cause more rapid failure.

The fatigue testing of fuel rods was initiated. Fatigue tests are intended primarily to establish the limit of stresses which might lead to cracks in the cladding as a result of cycling loads in service (testing, fluctuating loads, etc.). Another purpose of these tests is to add information regarding the strength of the reference end closure under the same conditions.

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Fatigue tests are not intended to prove directly the adequacy of the cladding in the reference tubing for PWR Core 1 blanket fuel rods. Reference fuel rods, with metal filler rods simulating the fuel, were fatigue tested at room temperature under internal pressure at 40 to 50 cpm; 5 specimens were tested simultaneously at each pressure. Testing pressures ranged from 4000 to 13,500 psi. No failures were observed at 4000 psi; 3000 cycles were required to cause failure at 7000 psi. After PWR reactor start-up, the normal maximum pressure within a fuel rod, due to fission gases, will not exceed 850 psi. The blanket, under the most adverse conditions, is not expected to be subjected to as many as 3000 pressure cycles (start-up and shutdown). On the basis of combined creep and fatigue information, it may be concluded that a fatigue failure under any condition is impossible.

Conventional burst test failures are confined to the tube wall; all specimens subjected to cyclical stress fail at the welded zone. The failure crack starts at the root of the notch formed by the end plug and the tube wall interface.

b. Manufacture of Seamless Tubes

The processing and evaluation of initial test lots of reference tubing produced by Superior Tube Company was completed. These lots represented extruded shells of 0.750 in. OD x 0.065 in. wall thickness, and 1.062 in. OD x 0.081 in. wall thickness from 3 different ingots. All lots were evaluated at Bettis Plant for dimensions, proof and burst test strength, and were found to meet specifications.

Babcock & Wilcox Co. have completed the piercing of a solid billet and the extrusion of the pierced billet into a tube shell of large size. The extruded shell is being reduced to a size which can be accommodated in Superior Tube Company equipment for conversion to reference tubing. Also, Chase Brass and Copper Company has extruded a hollow billet into a large tube shell for conversion by Superior Tube to reference tubing.

Extruded tube shells of 0.750 in. OD x 0.065 in. wall, and 1.062 in. OD x 0.081 in. wall are being converted to reference tubing by Wolverine Tube Division. Both lots have been processed through the tube reducing operations. Two more draws with intermediate anneals will be required to complete the processing to required size.

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Tooling required for the extrusion of small tube shells is being prepared by Canton Drop Forging & Mfg. Company.

Purchasing Department Specification No. 12650, applicable to zirconium alloy reactor quality seamless tubes, has been revised and submitted for approval so that it may be advanced from the tentative status to standard.

c. Closed-End Tube Development

Evaluation data obtained from 36 closed-end tubes produced by Bishop and Babcock Mfg. Company showed all to be rejectable because of over-size OD dimension and lack of straightness.

The processing of a small lot of closed-end tubes by a revised practice was completed and finished tubes will be evaluated by Bettis personnel. A purchase order was placed with Bishop and Babcock for the conversion of about 40 ft of Zircaloy-2 bar stock into closed-end tubes by revised and improved practices aimed to yield a structurally sound and dimensionally acceptable product.

A draft of a proposed specification covering Zircaloy-2 bar stock was prepared and is being submitted for comment and approval.

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

From the results of non-destructive testing work, two specifications have been drafted; one covers the wall thickness measurement with the Vidigage; the other, the eddy-current inspection of tubing with the Probolog. A development program for the inspection of the PWR seed has been started.

Zircaloy Tubing

a. Surface Flaws

Five techniques for the ultrasonic inspection of tubing have been tried with the following results:

(1) Immersed Circumferential Test: This test has been extensively evaluated and shows promise for detection of longitudinal flaws only, which defects are better detected by the Probolog. Therefore, no further work will be done on the technique at this time.

(2) Contact Angle Beam Test: A spring loaded 2.25 mc crystal has been tried. Difficulties of coupling the crystal to the tubing must be overcome before reliable and reproducible results can be obtained.

(3) Immersed Angle Beam Test: Using a 23° angle and a 5 mc, 1/8 in. collimated beam, this test appears quite sensitive to both longitudinal and transverse flaws. Methods to increase scanning speed and retain sensitivity to defects must be developed to make the test practical for production testing.

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(4) Immersed Through Transmission Angle Beam Test: This test has been tried with generally poor results.

(5) Contact Test From the Tube End: Spurious signals arising from multiple reflections throughout the tube length mask any defect signals. Work on this technique has been discontinued.

A specification for eddy-current inspection of Zircaloy-2 tubing with the 1 kc, Model E Probolog has been submitted to the Atomic Energy Commission for approval. A report covering development of this test for tubing has been prepared and will be issued shortly.

A 100 kc Multitest, test coils, and a traversing mechanism have been received from Battelle Memorial Institute to determine whether this 100 kc eddy-current instrument is better than the 30 kc instrument for detecting flaws in tubing. This instrument lacks sensitivity and is without the phase resolver usually incorporated into Forster eddy-current equipment. Some information is being obtained from this equipment but final evaluation of 100 kc Multitest type instrument will await the receipt of the Magnatest which has the required phase resolver.

Differential annular coils and an improved yoke coil specifically designed to detect laminar flaws have been wound for the 30 kc Magnatest. After balancing, they will be checked for response and sensitivity.

b. Tube Wall Thickness

The specification for checking tubing wall thickness with the Vidigage has been drafted. It will later be modified to include automatic scanning. The tube conveyor for the automatic scanning was not received until December 19, 1955. Attempts are being made to record the Vidigage signal, using a scanning technique and a photoelectric detector.

End Closures

a. General Weld Quality

Of the four ultrasonic tests suggested for inspecting end welds, the immersed end technique and the immersed diametrical technique have been tried. The immersed end technique has given the best results, but the method is not as reliable as the radiographic technique presently specified. The immersed diametrical technique gave poor results because of difficulty in locating weld interfaces and resolving reflected signals. The surface and shear wave techniques are now being investigated.

The specification covering the radiographic inspection of end welds has been approved by the Atomic Energy Commission. Extension of the radiographic technique to include the tubing is being investigated.

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Rod Spacing

Methods to determine the spacing between rods in a bundle are being investigated. At present, a mechanical gauge is being used. This can be mechanized to the extent of automatically scanning of the length of the rods and recording of the measurements. This phase is in process. Optical means of measuring the spacing are also being investigated.

Seed Plates, Subassemblies and Clusters

Techniques used for inspection of other plate type elements have been reviewed for applicability to the PWR seed. In addition, eddy-current and resonance ultrasonic tests to determine unbonding and cladding thickness have been tried with little success. Other applicable tests planned or under way include radiography, autoradiography, conventional ultrasonics, X-ray absorption, penetrants, photoprobes, and mechanical and volumetric measuring methods.

3. Fuel Rod Process Development (53c23) J. GlatterFuel Componenta. Stokes Automatic Press Compaction

The experimental activity utilizing the Stokes Automatic Press consisted of reclamation studies on natural and enriched oxides and cold pressing studies of activated natural UO_2 . The first two studies are presented under 53c21 of this report. The activated powder study is shown below. Development work on the press itself has been completed. To avoid repetition of information, the above heading will be eliminated in future progress reports.

b. Activated Powder - Cold Pressing

Experimental work showed that the addition of TiO_2 in quantities ranging from 0.2 w/o to 0.5 w/o increased the sintered density of as-received natural UO_2 by 3% above that obtained without the additive. A particular lot of oxide which yielded an 89% density without the TiO_2 gave a 92% density under similar pressing and sintering conditions (80 tsi pressure and 5 hr - 1600°C sintering treatment).

c. Extrusion

Laboratory scale production of high density UO_2 fuel rods by hydraulic extrusion was continued during the report period. The main development effort was aimed at establishing a centerless grinding technique for finishing integral length UO_2 rods. The chief obstacle to successful through-feed centerless grinding was the lack of surface uniformity and axial straightness in

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the as-sintered rods. Experiments are now in process to determine the sintering technique necessary to increase the axial straightness of rods. In addition, outside sources have been contacted to assist in solving the existing grinding problem.

d. Slip Casting

Slip cast rods of natural UO_2 having green densities of 57% TD were produced by vacuum treating the fluid slips prior to pouring into plaster molds. The attainment of a high green density was reflected in a sintered density of 93% TD after a 10 hr - 1725°C sintering treatment. Additives included 1.0 w/o Bentonite serving as a shrinkage agent, 0.25 w/o PVA serving as a binder to improve green strength, and 0.2 w/o Daxad 23 serving as a dispersing agent. The addition of 0.2 w/o TiO_2 to the aforementioned constituents gave a 2% increase in density (i.e., from 93% to 95%) for the previously mentioned sintering treatment. The addition of CeO_2 in quantities ranging from 0.1 w/o to 1.0 w/o resulted in density decreases of 2% to 4% when processed and sintered in the same manner.

Sintering Studies

Density measurements were completed on the experiments regarding the determination of sintering characteristics of 10 lots of Mallinckrodt UO_2 utilized in the flexible critical experiment. Evaluation of the data is proceeding.

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

The most serious weld defects have been found to be caused by foreign material in the inside surface of the tubing. This type of defect is reduced by thorough cleaning of the tubing, including the removal of 1 to 2 mils from the surfaces by acid etching. Another cause of weld porosity can be trapped helium. This can be minimized with proper design tolerances. A third, but possibly minor, cause of porosity is the volatilization of impurities in Zircaloy above its melting point. The tube-to-end-plug interface can serve as a nucleation site for volatile impurities.

There has been no correlation between weld porosity and the normal fabrication variables in Zircaloy processing.

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b. Radiofrequency Induction Welding

Acceptance tests on the rf induction welding equipment have been essentially completed at the New Rochelle Tool Corporation. The control system has been modified to cut the weld power when the collapse of the molten annular weld pool is sensed by the equipment. This system compensates for all normal electrical power fluctuations. The equipment is being revised to reduce all possible mechanical variations in operation.

The power equipment has been shipped to Bettis Plant.

c. Percussive Welding

Percussive butt welds have been made between pieces of reference sized fuel tubing. The optimum welds have not yet been obtained. Recent results have indicated that a large increase in capacitance of the machine and modification of the power circuit will be required.

End Closure Evaluation

Tension fatigue tests results have been obtained on fusion welded PWR fuel elements. The endurance limit was found to be 17,500 psi, with all failures occurring in the tube wall at the end of the step plug. Tube fatigue strength was obtained by testing reduced sections. The endurance limits were: 33,500 psi for the 10-15% cold worked tubing, and 32,000 psi for the vacuum annealed tubing. This test program is continuing.

Several hundred rf induction welds were made to evaluate joint design for this application. Step down end plug designs could not be fully penetrated without excessive undercutting of the weld bead. Upsetting of the weld bead was attempted, but excessive corrugation of the tubing over the plug occurred before complete penetration was obtained. The recessed end cap design appears much more promising at this stage. Evaluation of recessed cap designs is continuing.

Internal pressure fatigue tests of tube and end closure welds have been made and are reported under 53c21, part II.A.1.a.

4. Fuel Assembly Process Development (53c24) J. G. ChristFuel Rod Bundle Development

The following innovations were made in bundle welding:

a. Strip spacers were used between the fuel rods during welding in the center as well as the ends of the bundle.

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b. A sequence of welding alternate rods within concentric squares was used in an attempt to balance the heat build-up and heat extractive effects of previous welds. This bundle is being sectioned to determine the weld penetration pattern.

The strip spacers were removed by dissolving in acid after welding. The bundle was vacuum annealed at 1300°F for 1 hr after removal of the spacers. The effect of annealing on spacing was negligible, averaging a 1 mil change. This indicates that the bundle is essentially stress-free after welding. The next bundle will be annealed with the spacers in place to determine whether the rods spacing tolerance can be improved in this way.

Because there has been a problem of the weld metal obscuring the pilot locating holes for drilling of the cooling path, an evaluation of a smaller circumference of electrode rotation is being made. The diameter was reduced from 0.290 in. to 0.200 in. This rotation was still effective in preventing arc wandering, and a somewhat smaller pool was maintained. The evaluation is continuing.

The first production-type fuel rod bundle welding chamber has been received from the manufacturer. A rotating electrode and work indexing mechanism which will be installed in this chamber are still under construction.

A preliminary process specification for the manufacture and inspection of fuel rod bundles was prepared.

Fuel Rod Bundle Evaluation

Two fuel rod bundles have been completed for thermal cycling experiments. One of these bundles was made by the fusion welded tube sheet method, and the second bundle was made by the mechanical assembly of strips and rods. The fuel rods in these bundles were made with 0.070 to 0.090 in. axial clearance on the fuel pellets. A second set of fuel rod bundles is being made with a minimum of axial fuel clearance. The bundles will be thermally cycled to 1300°F.

Blanket Shell Assembly Development

Some difficulty has been encountered in obtaining Zircaloy-2 strip satisfactory for blanket assembly shell development work. Findings of the current investigation indicate that most of the trouble is a result of unsatisfactory cleaning, pickling, and annealing practices resulting in severe surface contamination. Subsequent edge cracking during cold rolling has resulted in a considerable loss of material from the 2 ingots processed. Changes in processing are being investigated. The first trials of forming strip into 7-1/2-ft-long channels by the Brooks and Perkins Company, have been unsuccessful. Contamination of the strip during annealing is probably

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the major cause for breakage on the 3 pieces which have been tried in their press. The remaining material will be blasted (cleaned) and pickled before additional tests are made. Brooks and Perkins is also making some minor improvements on the dies and die heating facilities.

A 5-ft-long internal welding jig has been constructed for welding 2 channels or 4 lengths of strip to form assembly shells. The first tube (2-1/2 ft long) has been welded by the 4-corner method and checked dimensionally. As more material becomes available, both methods can be compared.

A preliminary drawing of the Zircaloy-2 transition piece to be welded on the ends of the assembly shells has been received. Plans are to immediately contact metal fabricators who might be able to make this shape either by an extrusion or die forging process.

Development work has been started on the fuel element failure sampling tube which is to be fastened in 1 corner of the assembly shell. The initial effort is concerned with the design of the support connectors and evaluation of weld techniques to be used in joining the connectors in the shell.

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

Samples of extruded and slip cast sintered natural UO_2 rods were subjected to 636°F water for 7 days and were unaffected in post test appearance. Long-time corrosion tests of these samples are continuing.

Fuel Element Testing and Evaluation (J. G. Christ)

In-File Tests

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

a. WAPD-CR-5

The 6-mo exposure of 2 fuel rod bundles in the central thimble loop of the Chalk River Reactor has continued without incident since the start-up September 9, 1955.

b. WAPD-MTR-15-1

Further examination of this bundle has been deferred due to the limitations of Hot Lab facilities.

c. WAPD-MTR-15-2

External examination of the defected fuel rod showed that the crack in the tubing was about 5 mils wide and 1.5 in. long. Discrete particles of fuel were observed to be still in place under the crack. A report

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has been prepared on the post irradiation examination of the bundle and external characteristics of the fuel rods. Internal examination of the fuel rods and radiochemical analysis have been delayed by the limitations of Hot Lab facilities.

Metallic Fuel Element Development (53c34) F. R. Lorenz

Design work on the adaptation of a vacuum induction melting furnace for centrifugal casting is virtually completed. Several parts of the casting mechanism are being manufactured and engineering work on the furnace is scheduled to begin during the next report period.

Of the three new beryllia crucible washes evaluated during the last report period, only one showed comparable qualities with the wash presently being used. The amount of carbon contamination has been held between 100-400 ppm on several alloys prepared by melting in beryllia washed graphite crucibles. This is not considered excessive contamination. However, the process is not consistent since carbon contamination of 1000 ppm has been found in many alloys. Work is continuing to improve the melting process by different methods of applying the beryllia wash. The silicon analysis of 37 ingots (U-3.8 w/o Si nominal) showed a Si content of 3.786 w/o ± 0.105 .

Investment type molds were used successfully in preparing rods 3/8 in. diam x 5 in. long. Several hundred rods of U-3.8 w/o Si and U-3.8 w/o Si ternary alloys have been prepared in these molds.

Hot hardness studies of epsilonized U-3.8 w/o Si indicate a hardness decrease from about 190 Bhn at 600°F to 20 Bhn at 1600°F. The large hardness range which has been experienced among samples at a particular temperature can probably be attributed to irregularities in the as-cast and epsilonized structure of the specimens.

Nineteen Zircaloy-2 clad U-3.8 w/o Si billets were successfully extruded through a 0.437 in. diam die. A straightening guide was used successfully for this work. An attempt to extrude an epsilonized U-3.8 w/o Si billet in a 0.2 in. wall copper jacket was unsuccessful. One more attempt will be made during the next period to extrude a copper jacketed, epsilonized U-3.8 w/o Si billet of low carbon content (~ 300 ppm). No other extrusion work is planned.

It was found that the erratic readings obtained in the billet heating study being pursued to control extrusion characteristics for the difficult U₃Si alloy, were caused by deterioration of the insulation of the thermocouple wires. Several billets have been prepared using ceramic insulation, and are ready for testing.

Thermocouples welded near the throat of extrusion dies indicated die temperature increases of about 600°F when an extrusion billet heated to 1550°F was pushed through.

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An order for a port hole die to sheath aluminum alloy cladding on U-3.8 w/o Si rods has been placed.

Approximately 1000 ft of Al-20 w/o U235 wire, 0.030 in. diam, was fabricated for use in the PWR flexible flux plot studies.

Master alloys of U-20 w/o Nb, U-30 w/o Nb, U-40 w/o Nb, U-50 w/o Nb, and U-60 w/o Nb are being prepared by special arc melting techniques for ultimate use in making alloys for transformation studies.

Seed Development (53c35) F. R. Lorenz

This activity includes the development of fuel plate fabrication methods and processes, and the assembly of fuel plates into subassemblies and clusters.

Fuel Plate Fabrication

Attempts to process fuel plates for assembly by upsetting have not been successful. It has been calculated that for plates 0.079 in. thick a maximum flange width of 0.145 in. would be possible. This is 0.060 in. less than the requirement for PWR plates. Consequently this method has been eliminated from the major effort, however, flanges will be upset on end plates.

Seventy dummy fuel plates are in process; 44 have been rolled, and 26 are currently being assembled. Cold rolling within the steel jacket has been attempted and has been shown to be effective in decreasing length variations to approximately 1/8 in.

Samples, each 38 in. long, from 6 dummy plates are being corrosion tested in 750°F steam to test the corrosion resistance of hot rolled surfaces. Smaller samples to provide exact weight change measurements are being tested also.

Methods for fabricating flange stock to be used on plates designed with shaped rolls are being developed. Jessop Steel Company has constructed a 4-piece die for the initial development efforts to draw flange stock. Page Steel and Wire Division consented to make initial attempts to roll the flange stock. A fuel plate with flanges that were made from stock produced by drawing through a Turk's head was hot rolled and is being processed.

Thirty-two seed fuel plates containing 5.54 w/o uranium-Zircaloy-2 fuel were rolled. These plates were rolled to compare the merits of both machined and rolled fuel components, and also, 2- and 3-cladding component rolling techniques were used. Preliminary information compiled from data on 16 plates indicates that machined fuel components tend to produce plates with somewhat better fuel thickness tolerances than plates using rolled fuel components. These same plates indicate that recessed cover plates yield better fuel dimensions in the finished plates than does the picture frame technique. A detailed

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analysis will be reported upon the completion of all scheduled work. These first data indicate that it may be possible to improve the seed fuel thickness tolerance so that it is tighter than that which has been used in the core design calculations.

The fabrication of a dummy PWR thermocouple containing fuel plate is nearing completion.

As a result of work accomplished, a preliminary fuel plate design has been determined. Future fuel plates will be of the 2-component recessed cover plate design and the fuel alloy component will have tapered ends. The plates will be hot rolled with flanges roll bonded simultaneously. As-rolled surfaces will be used. Sizing and finishing operations as required and as yet unspecified will be incorporated to complete the fuel plate.

Material

The successful extrusion of large Zircaloy-2 slabs (1-1/2 in. x 7 in. x 15 ft) was accomplished at the Canton Drop Forging & Mfg. Company. These extrusions, made from full size 12-in. diam ingots, were made to provide a new source for the breakdown of the large ingots for the seed plate components.

Seed Subassemblies and Clusters

The first dummy subassembly has been welded together. Thirteen dummy fuel plates with rolled-on flanges and 2 end plates were tacked together with perforated steel spacers. The assembly, welded in the F-3 welding box using 275 amp, is currently being pickled to remove the steel spacers. Initially, the assembly was sent to the Jessop Steel Company for pickling in a hot 10% H_2SO_4 bath; however, the dilute acid was ineffective. Operations in a stronger HNO_3 solution are being completed at Bettis Plant.

Additional subassemblies to complete the first cluster have been included in a revised schedule. Developmental problems, such as pickling of spacers and procurement of Zircaloy-2 strip and machined components, have retarded this part of the development program; however, work on the first cluster will be conducted during the next period.

Welding fixtures for all operations have been received. The welding box carriage drive mechanism was tested with a load simulating a full sized cluster, and a satisfactory response was observed over the welding box working range of 2 to 15 in/min. Ten in/min was determined to be the optimum speed for current subassembly welding conditions.

Several short sections of PWR subassemblies were welded in order to familiarize the personnel with the operation of the welding box and to obtain data regarding welding procedures for full size assemblies. Full penetration was obtained using 275 amp on a short subassembly section of 9 plates with a 0.250 in. wide flange and copper spacers.

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Fixtures for pickling, annealing, and corrosion testing the fuel plates, subassemblies, and clusters have been ordered. These fixtures will be available for development work during the next period.

Flux Depressors

The components for a solid cruciform flux depressor have been received and are being inspected. These components will be welded during the next period. Consideration has been given toward design of box type and hollow cruciform type flux depressors. Active development work will be initiated during the next period on a hollow cruciform type. Although not yet established as a reference design, the hollow flux depressor appears to be the best design.

Control Rods (53c36) F. R. Lorenz

The hafnium control rod, fabricated for test purposes (thermal, mechanical, and worth value), was completed. All work on boron stainless steel alloys for control rod purposes has been discontinued because of poor behavior under irradiation.

A program has been initiated to determine the feasibility of using hafnium crystal bars which have not passed the screening corrosion test (Grade II). Impact, tensile, hardness, corrosion, electrical resistivity, and chemical tests will be conducted on material which has been fabricated utilizing various percentages of Grade I and Grade II crystal bars. Samples will be irradiated in the Materials Test Reactor and the various physical properties will be compared with those of the material prior to irradiation. The samples are expected to be delivered to the MTR site in time for February 16, 1956 charging, if a facility is available.

The development program on substitute materials such as boron and europium dispersions in stainless steel has been examined. It was concluded that the results of the materials process development program would not be available in time to meet PWR Core 1 schedules. A test program has been established to measure the control rod worth of various B₁₀ dispersions in stainless steel in the PWR critical facility.

An operation schedule for the manufacture of control rods, complete with Zircaloy-2 extension and stainless steel adaptor, has been written. Design work has been completed on all welding fixtures. Several of the machining fixtures are being designed, although lack of control rod dimensions has handicapped this work.

The dies for the extrusion of half-scale hafnium cruciforms have been received. Extrusion facilities are expected to become available during January, 1956. An attempt is being made to find an alloy of Cu and Ni that will have the same extrusion characteristics as hafnium so that hafnium itself will not be required for all development work. Alloys of Cu-40 w/o Ni, Cu-45 w/o Ni, and Cu-50 w/o Ni are being obtained.

6. Fuel Element Pre-Manufacturing Operations (53c41) W. J. Hurford

PWR Manufacture

Seed

Conversion of approximately 6500 lb of Zircaloy-2 (7 ingots) to strip

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for manufacturing of fuel plate components and end supports for the seed has been started.

Blanketa. UO₂ Natural Powder

Twelve thousand pounds of specification UO₂ natural powder has been received from Mallinckrodt Chemical Works and is in stock. An additional 36,000 lb will be shipped in January, 1956 and 44,000 lb in March, 1956.

b. UO₂ Fuel Components

Agglomeration and compaction of UO₂ powder for core production have commenced. Operations are proceeding on schedule. Compacts will be pre-treated in a carbon dioxide atmosphere at 800°C for 2 hrs to remove the polyvinyl alcohol binder. Sintering will be done in a hydrogen atmosphere at 1675 - 1700°C for 10 hrs. The sintering process will be used to produce the fuel components for the reactor.

Finished ground fuel components which were fired in the Harper sintering furnace have passed the corrosion test at 750° steam for 20 hr.

c. Zircaloy-2 Reactor Tubing

Six 12 in. diam. weld conditioned ingots (~5800 lb) have been extruded at Canton Drop Forging and Manufacturing Company to 2-1/8 in. bar stock. Thirty-two hundred pounds of this bar has been machined to produce 600-1.85 in. OD x 0.65 in. ID x 5 in. long extrusion billets which are being processed by Nuclear Metals, Incorporated (NMI) into reactor quality tubing. These 600 billets should produce about 10,000 ft of tubing; the remaining bar will be used for extrusion billets for 10,000 ft of additional tubing at NMI, the order for which is currently being negotiated, or for end cap rod stock.

One thousand and forty-nine feet of tubing have been received to date from NMI. The latest 274 ft received was plug drawn prior to annealing, in contrast to moveable mandrel drawing, to the current dimensional specification. Of this, an approximate 85% yield of cut reference lengths to acceptable reference lengths was obtained. Ten per cent were rejected for excessive OD and ID and 5% were rejected for excessive bow (lack of straightness).

Subsequent shipments from NMI are expected to give higher yields as techniques are further developed in the new and less costly plug drawing method.

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Twenty-four billets (2516 lb) machined to 7.74 in. OD x 1.82 in. ID 9-10 in. long, representing 4 ingots (3800 lb), were extruded to 2 in. OD x 3/16 in. wall x ~30 ft long at Chase Brass and Copper Company. These tube shells will be processed by Superior Tube Company to produce about 8000 ft of finished tubing. Delivery of this tubing from Superior Tube Company is expected to start in March, 1956.

d. End Caps

Approximately 1900 lb (2 ingots) of Zircaloy-2 has been processed by ingot extrusion or forging to 3-3/4 in. bars which have been re-extruded, straightened, drawn, annealed, and centerless ground to 0.422 in. rods. C. A. Hukill Machine Company will produce about 50,000 end caps from these rods using Swiss Automatic screw machines.

e. Tube Sheets

Conversion of 1 Zircaloy-2 ingot (590 lb) to hot rolled strip 7 in. wide x 0.310 in. thick has been started and will supply stock for 200 tube sheets.

f. Assembly Shells

One Zircaloy-2 ingot (950 lb) is being converted to hot rolled strip 12-1/2 in. wide x 0.205 in. thick for subsequent processing into 3 assembly shells.

Physics Flexible Experiment

Manufacture of blanket UO_2 fuel bundles for the physics flexible experiment was completed on December 14, 1955. A total of 124 fuel bundles were manufactured. Approximately 27.3 tons of UO_2 powder were processed into 18.2 tons of acceptable UO_2 fuel components, resulting in an overall yield of 66%. A total of 15,154 fuel elements were manufactured.

Physics Two Region Critical Experiment

Manufacture of the slightly enriched, 70% of theoretical density of 0.387 in. diam UO_2 fuel components was resumed. The required completion date of March 1, 1956 will be met.

B. Metallurgy of Core Materials1. Basic Study of Metallic Fuel and Cladding Alloys (53b11) E. F. Losco

Several investigations designed to obtain needed information on basic and unusual problems which developed in Alloy Development (53b41) and Irradiation Effects (53b31) programs were continued. Progress on these investigations is indicated below.

Alpha-Prime Phase in U-Mo Alloys

The alpha-prime phase formed in gamma-quenched U-Mo alloys containing less than 5 w/o of molybdenum is being studied because of the interest in such low alloy-bearing uranium as a reactor fuel. Individual grains of a gamma-quenched U-5 w/o Mo sample have been outlined and photographed. Their crystallographic orientations are being determined preparatory to thermal transformation to the alpha prime phase. The habit plane and orientation relationships in the transformation product will be determined. Other samples of the alpha-prime phase have been prepared and heat treated to determine the thermal stability and kinetics of decomposition of the phase by metallographic and X-ray diffraction techniques.

Effect of Stress upon the Thermal Stability of U-Mo Alloys

X-ray diffraction and metallographic studies on gamma-quenched U-7 w/o Mo samples tested in creep at 400°C and 45,000 psi previously indicated that the applied stress resulted in decomposition of the gamma phase to alpha plus epsilon. Creep-rupture testing was continued on gamma-quenched U-5 and U-7 w/o Mo samples at 350°C. Creep-rupture failures have been obtained at both test temperatures but no correlation between structure and failure time was observed. However, the applied stress was observed to accelerate decomposition of the gamma phase to alpha plus epsilon and occasionally to "strained alpha". Some 350°C tests are still in progress.

A single grain in a gamma-quenched polycrystalline U-7 w/o Mo sample outlined by hardness impressions on the grain boundary failed to transform when heated at 400°C for 300 hr. The remainder of the sample transformed to alpha plus epsilon as expected. The single grain was cut out of the sample and is being analyzed by X-ray diffraction.

Thermal Stability of U-Nb-Zr Alloys

Interdiffusion layers formed in coextruded Zircaloy-2 clad U-10 w/o Nb samples have displayed remarkably good corrosion resistance in 650°F water. The ternary system of U-Nb-Zr, therefore, is being studied by interdiffusion of zirconium in U-Nb alloys containing 2 to 12 w/o niobium. Several couples have been prepared by pressure bonding techniques and are being diffusion annealed to determine diffusion coefficients and to establish the U-Nb-Zr alloy of optimum corrosion resistance and thermal stability. Preliminary

data on couples formed with U-4 and -8 w/o Nb indicate that a 3-week anneal at 1000°C results in a large diffusion zone, about 4 mm thick. About two-thirds of this zone is in the zirconium side of the original interface. Metallographic examination indicates the presence of at least 6 different layer phases in the diffusion zone. Hardnesses of each phase are being determined preparatory to X-ray diffraction and chemical analysis. Tests are also being made to establish a suitable heat treatment to evaluate the relative thermal stability of each layer. One hundred hours at 400°C was found to produce 10% transformation in the gamma phase of the U-8 w/o Nb component of the couple. As the interface was approached the amount of decomposition product decreased. These tests are continuing.

Magnetic Properties of Uranium Hydride

The nature of the hydride phase formed during hot water corrosion of gamma-quenched U-Mo alloys is being studied through magnetic susceptibility measurements. Preliminary measurements at 78°K and a field strength of 3230 gauss on a U-12 w/o Mo sample containing 155 ppm of H₂ indicate that the magnetic moment per gram of platelet material is 7.1. This is assuming that all but 25 ppm (the approximate solubility of hydrogen in the gamma phase) is in the platelets. The magnetic moment for pure UH₃ under the above conditions has been reported as 18.0 emu/g. Thus the magnetic measurements at a fixed field strength and temperature show that the platelets are not UH₃ or (U_{3/4}Mo_{1/4})H₃ as one would expect in a 12 w/o Mo alloy assuming no diffusion of the molybdenum.

The cryostat for the susceptibility apparatus was satisfactorily assembled during this report period. After the completion of the auxiliary plumbing for the cryostat it will be possible to make magnetic measurements from 78°K to about 400°K to determine the Curie temperature of the hydride phase.

Identification of the Hydride Phase Formed during Corrosion of Gamma Phase Alloys

Experiments involving the hydrogen decomposition pressure over corroded samples of U-12 w/o Mo were continued to obtain a clearer understanding of the metallographic and thermodynamic properties of the hydride phase. Metallographic examination of a rod sample containing 1350 ppm of hydrogen showed very coarse hydride platelets after corrosion testing. After heating in an evacuated closed system to 154°C for about 24 hr, without appreciably affecting the hydrogen content, it was found that the platelets were much finer and more uniformly distributed, indicating possible decomposition and local redistribution of the hydrogen. This effect did not appear pronounced in other samples containing lower hydrogen contents (270 and 900 ppm) and similarly treated at higher temperatures (250 to 325°C).

The sample containing 900 ppm of hydrogen was later heated for 24 hr in vacuum at 406°C. This treatment caused a marked reduction in hydrogen content with a gradation from surface to center estimated by metallography

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of zero to 200 ppm, respectively. The platelets remaining in the center were reduced in size and thickness (possibly also in number). The sample surface (to a depth of about 1/32 in.) showed macroscopic intergranular cracks probably caused by severe stresses accompanying the density changes resulting from decomposition of the hydride and evolution of hydrogen. Analysis of hydrogen at the surface and center will be obtained.

Electrical Resistivity Measurements at Low Temperatures

Electrical resistivity measurements will be made down to liquid helium temperatures to study the anomalous negative resistivity coefficient displayed by gamma phase U-Mo, U-Nb, and U-Zr alloys. Samples of alloys of particular interest are being heat treated. Resistivity data will be available during the next report period.

Specific Heat Measurements

Specific heat measurements have been made on 3 additional U-Mo alloys to obtain accurate measurements of transformation temperatures. The transformation temperature recorded for U-11 and -12 w/o Mo samples was observed to be about 20°C lower than predicted from the phase diagram. The presence of minor impurities in the sample may be responsible for this and will be checked on later runs.

The U-Nb Phase Diagram

Additional U-Nb alloys with compositions corresponding to U_3Nb , U_2Nb , UNb , UNb_2 , and UNb_3 have been prepared by a modified arc-melting procedure to minimize alloy segregation. These alloys are being used in an X-ray identification of an unexpected FCC phase which was observed in the U-Nb system and which may be involved in the irradiation-induced phase reversal observed in U-10 w/o Nb samples. The low niobium samples were found to be slightly segregated while the higher-niobium samples showed appreciable segregation on X-ray and metallographic examination. However, heating at 1400°C for 1 hr significantly improved homogeneity so that the samples are considered satisfactory for further work. After additional homogenization, samples will be heat treated to develop desired structures.

Double Crystal Spectrometer

Considerable effort was expended in improving the specimen control mechanism for the double crystal spectrometer being built to study irradiation-induced structural changes in alloy systems. Rotation of the specimen to provide integrated X-ray intensities and horizontal rotation to provide centering control have been achieved. A new wiring diagram was prepared to provide the various motions required. Since lack of suitable space is available in the Hot Laboratory, the instrument will be set up in a specially enclosed area in Room 504, Wing 4 Left.

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2. Basic Study of Ceramic Fuels (53bl2) Z. M. ShapiroUO₂-O₂ Equilibrium and Kinetics

A study of the equilibrium and kinetics of the system UO₂-O₂ has been continued in an effort to gain information regarding the structure and stability of UO₂ under various conditions.

As was reported in WAPD-MRP-58, samples of uranium oxide having O/U ratios from 2.03 to 2.67 were prepared and annealed in vacuo for various times at temperatures of 200, 360, and 700°C and then subjected to analysis by X-ray diffraction. While these analyses are yet incomplete it has been found that samples slowly oxidized at temperatures of 200°C or lower take up oxygen to an apparent O/U ratio of 2.16 without significant change in lattice parameter. As the apparent O/U ratio is increased from 2.16 to 2.30 the lattice becomes tetragonal with a constant c/a ratio of 1.011. This behavior is interpreted to mean that even under conditions of slow oxidation at low temperatures, the oxygen is taken up inhomogeneously with the formation of an entirely new phase, presumably U₃O₇, and that this phase begins to show up on the diffraction patterns only after sufficient conversion has occurred. Beyond this point the cubic pattern disappears and that of the new phase becomes predominant. This appears to be consistent with what Gronvald diagram would predict (a mixture of UO₂ and U₃O₇) in this composition range at temperatures below 150°C.

The X-ray diffraction patterns of samples annealed at 200°C are decidedly different than those of samples in the unannealed condition. Those annealed for 2 weeks show two distinct cubic phases, the α consisting of UO_{2+x} and the β probably consisting of U₄O₉. For powders with O/U ratios between 2.00 and 2.16 the cubic α lines are sharp and the β lines diffuse, while in the range 2.16 to 2.25, the α lines are diffuse and the β lines sharp, indicating that further annealing is required in order to improve the crystallinity of the phases. This was confirmed by the sharper diffraction patterns obtained with samples annealed for 4 weeks. These results appear to confirm the Gronvald phase diagram in the composition range between UO_{2.00} and UO_{2.25} at temperatures of approximately 200°C.

X-ray diffraction patterns of samples of UO_{2.08} and UO_{2.10} annealed at 700°C show only the α -cubic phase. This is also in agreement with Gronvald's phase diagram which shows a region containing only the α -cubic phase UO_{2+x} in this temperature and composition range. On the basis of Perio's diagram, a mixture of the α and β phases would have been expected.

Calculations are now being made to determine whether a lattice contraction occurs in the α -cubic phase as the quantity of oxygen in the lattice is increased. It is expected that the complete results of the X-ray analysis will be reported in WAPD-MRP-60.

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Kinetic Studies in Water

Studies have continued on the rates of oxidation and reduction of UO_2 and U_3O_8 powders in 650°F water containing oxygen and hydrogen in order to better understand the reactivity of UO_2 in water under various conditions.

Samples of U_3O_8 have been removed from water containing 5 cc H_2/kg after exposure periods of 5 to 25 days. Analysis of the samples showed that no changes had occurred in the composition of the samples. Thus, U_3O_8 appears to be stable to reduction under this condition.

Samples of UO_2 and U_3O_8 powder have been removed after 5 and 10 day exposures to water containing 2 cc O_2/kg . As was reported in WAPD-MRP-58, the UO_2 and U_3O_8 samples removed after 5 days had O/U ratios of 2.06 and 2.69, respectively. The UO_2 and U_3O_8 samples removed after 10 days had thin layers of yellow-green powder on the surface indicating that the oxidation did not occur homogeneously. A new group of samples has been prepared for exposure. The powders composing these samples will be thoroughly mixed after each 3 or 4 day exposure period to maintain greater homogeneity throughout each sample.

The fact that UO_2 and U_3O_8 are oxidized to a higher oxide at 650°F in water containing 2 cc O_2/kg indicates that the oxidation of UO_2 does not proceed in the same manner in water as in air. At 650°F , the partial pressure of oxygen in equilibrium with 2 cc/kg of dissolved oxygen is 10 mm Hg. This is, of course, considerably smaller than the partial pressure of oxygen under atmospheric conditions. In air, oxidation of UO_2 proceeds no further than U_3O_8 at any temperature. Thus, the presence of water apparently affects the mechanism of oxidation. This phenomenon will be investigated more fully by studying the oxidation of UO_2 in air containing water vapor.

Diffusion Studies

The apparatus for the measurement of the rate of fission gas emanation from uranium dioxide during irradiation has been installed at the Brookhaven National Laboratory (BNL) on December 23. Actual start-up of the reactor occurred on December 27.

The first specimen is a 3 mm thick flat plate having a density of 92% of theoretical. This shape was chosen in order to maintain essentially isothermal conditions in spite of the fissioning. It is expected that preliminary data on the rate of fission gas release at various temperatures will be available for the next progress report.

The preparation of uranium oxide powders with oxygen-uranium ratios between 2.03 and 2.25 for use in the measurement of the self diffusion of uranium in sintered pellets of these compositions has been completed.

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Oxygen-18 water has now been received and apparatus is being constructed to enable its use in the preparation of uranium oxygen-18 oxides of various oxygen-to-uranium ratios in order to measure the self diffusion of oxygen also. The UO_2 will be prepared by low pressure steam oxidation in a closed system to aid in the conservation of the heavy water.

The analysis of the diffusion couple described in WAPD-MRP-58, in which a layer of enriched UO_2 was pressed onto a layer of natural UO_2 and sintered at 1750°C in hydrogen from 10 hr, has been partially completed. Although the method employed (isotopic analysis by emission spectroscopy) was not entirely satisfactory in detecting small concentration differences in the low concentration range, a 0.020 in. migration of U^{235} was found. From this a diffusion constant of 5×10^{-9} cm²/sec was calculated. However, in view of the fact that considerable shrinkage occurred during the sintering this cannot be considered a valid constant. Moreover, it represents the overall effect of the several processes which occur during sintering. Nevertheless, the number can be used as a relative measure of the migration of uranium during sintering under different atmospheric conditions or with different oxygen-to-uranium ratios.

Diffusion couples have been prepared by evaporating enriched UO_2 on sintered pellets of natural UO_2 and these are now being soaked in vacuum at various temperatures preliminary to the measurement of the diffusion coefficient and the activation energy. In view of the fact that the pellets have already been sintered to high densities for long periods of time, the amount of grain growth during sintering will be negligible, and the experiments should yield true values of the coefficients for bulk diffusion.

3. Irradiation Effects on Ceramic Fuels (53b30) Z. M. ShapiroUO₂ Irradiation ProgramMTR-WAPD-13-Met-6

The defected UO₂ specimen containing pellets cold pressed and sintered to 93% of theoretical density, described in WAPD-MRP-58, has been operating at a maximum calculated heat flux of 600,000 Btu/hr-ft² for 12 weeks in the HB-5 facility at the Materials Testing Reactor (MTR).

After 6 weeks of exposure in the high flux position without cycling, it was cycled into and out of the high flux region for 5-min intervals during the following 3 weeks. Following this, the sample was given 3 to 6 long cycles per day for an additional 3 weeks. Currently this sample is being given a 1-week exposure in the high flux region without cycling following which it will be cycled at 5-min intervals for 1 week and 8-hr intervals for 1 week.

The coolant water activities have been relatively low and no excessive bursts of activity have been noted. Further details of operation are under Section II-C-4.

MTR-WAPD-14-Met-6

Fission gas analysis of one of the specimens containing 90% dense hot-pressed MCW UO₂ pellets and subjected to a burnup of 5000 MWD/T showed a release of 0.82% of the theoretical total gas at room temperature. An additional 0.002% of the total was released between room temperature and 250°C, 0.04% between 250°C and 600°C, and 0.04% between 600 and 1000°C. Thus, a total of 0.90% of the total fission gas present was released.

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

Analyses of the fission monitors which were attached to the 4 half-length PWR reference UO₂ fuel rods irradiated in an estimated neutron flux of 4×10^{14} nv for 3 weeks in the MTR have been completed. These indicate that the average flux values are as follows:

Top of pile	1.4×10^{14} nv
7-1/2 in. from pile top	5.4×10^{14} nv
15 in. from pile top	1×10^{15} nv
22-1/2 in. from pile top	4.3×10^{14} nv

The accuracy of these analyses is estimated to be + 20%. These indicate that the flux in the L-45 position is approximately twice that estimated by the MTR personnel. The maximum central temperature of the rods in the peak flux position is now approximated at 5000°C rather than 2660°C as reported in WAPD-MRP-58. Thus, it is not surprising that melting and burn-through were observed in these rods.

MTR-WAPD-14-Met-9Irradiation of Half Length PWR Reference Rods to High Burnups

This experiment is a repeat of experiment MTR-WAPD-14-Met-8. Although the samples were scheduled for a long irradiation, they were removed from the reactor after only 3 weeks irradiation because of the results with the previous experiment. The capsules have now been returned to the Bettis Plant Hot Laboratory and are being opened for examination.

MTR-WAPD-14-Met-10Irradiation of UO₂ Rods at High Burnup Rates

This experiment consists of 2 half-length PWR reference rods and 2 half-length rods 0.300 in. diam (instead of the reference diameter of 0.413 in.). The smaller diameter samples make it possible to take advantage of the high burnup rates available in the high flux positions of the MTR while still maintaining the proper temperatures. An irradiation for 1 year to approximately 15,000 MWD/T burnup is planned.

MTR-WAPD-14-Met-11Irradiation of Enriched UO₂ Rods to High Burnups

This experiment consists of 4 half-length PWR reference rods containing 5.5 w/o enriched UO₂ instead of natural UO₂. These rods will be inserted in a flux of 3 to 4 x 10¹³ nv during the December 26 shutdown and irradiated for 1 year to a burnup of approximately 20,000 MWD/T.

MTR-WAPD-19-Met-5

The 4 full-length enriched UO₂ specimens in the low flux A-piece cycler at the MTR have now been cycled over 22,500 times and have had an exposure of over 4,500 MWD/T. Because no noticeable change occurred in other cycled specimens (MTR-WAPD-19-Met-4) these specimens have been rescheduled in the cycler for an extended period in order to double the present burnup.

MTR-WAPD-19-Met-8

Four full-length natural UO₂ specimens fabricated from PWR reference type pellets and assembled with diametral clearances of 0.005 in. and 0.010 in. and end cap clearances of 0.070 to 0.090 in. have been charged into the high flux A-piece cycler at MTR. These replace the 8 natural UO₂ specimens (MTR-WAPD-19-Met-7) which were removed on December 26.

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The purpose of this test is to check the effect of the increased PWR design clearances on the stability of the rods under power cycling conditions. The specimens will remain in the cycler for a minimum of 12 weeks.

CR-WAPD-Met-4A

Analysis of the fission gas released from an 80% dense undefected specimen containing hot pressed UO_2 has been completed. The specimen had 1500 MWD/T exposure along with the defected specimen in the first UO_2 test at Chalk River. A total of 10.1% of the total calculated fission gas formed was released. Ten percent was released at room temperature, 0.02% between room temperature and 250°C, 0.02% between 250 and 600°C, and 0.07% between 600 and 1000°C. An autoclave leaching experiment of samples of the UO_2 from this specimen is now undergoing high temperature leaching tests in water. It is expected that considerable additional information will be available for inclusion in WAPD-MRP-60.

CR-WAPD-Met-7

A preliminary examination of the 2 specimens from the third UO_2 defect test in the X-1 loop at Chalk River has been completed. Neither sample showed any change in diameter, but the defected sample grew 0.3% in length. In addition, the defected sample showed a weight increase of 118 mg while the weight of the undefected sample increased only 22 mg. Although the defected specimen was vacuum dried for 24 hr at 120°C, it showed no loss in weight. Most of the weight gain of the undefected sample and at least a portion of the weight gain of the defected sample is attributed to a whitish crud deposited along one side of the specimens. Since this deposit occurs only on one side it is presumed to have accumulated after irradiation during storage or shipment while the specimens were lying horizontally. The defect was examined and appears to be clear.

Data on fission gas release in the undefected specimen will be reported in WAPD-MRP-60.

CR-WAPD-Met-8

The full-length defected UO_2 specimen described in WAPD-MRP-57 has been discharged from the X-3 loop at Chalk River. As reported in WAPD-MRP-58, several bursts of activity occurred late in October following a series of pile start-ups. However, during November the short-lived species of activity apparently disappeared from the coolant water. This would seem to indicate that the escape of short-lived fission products from the UO_2 had ceased, possibly due to plugging of the defect. Results of the preliminary examination of the specimen will be reported in WAPD-MRP-60.

CR-WAPD-Met-9

As reported in WAPD-MRP-58, four 10-1/4 in. long UO_2 rods enriched to 8.35 w/o U^{235} in UO_2 and one 10-1/4 in. UO_2 rod enriched to 20 w/o U^{235} in UO_2 were inserted in the X-1 loop at Chalk River. The 20 w/o rod and 2 of the 8.35 w/o rods were defected. The rods were arranged so that the one having the highest enrichment would be in the highest flux level at the center of the reactor. Three hours after start-up of the reactor an extremely large burst of activity caused the pile to trip although attempts were made to shield the loop so that operation could be continued. It was found necessary to withdraw the test after 20 hr of intermittent operation due to a build-up of an extraordinarily high level of activity. On examination of the specimen it was found that the 20 w/o enriched rod which should have operated at a maximum heat flux of 533,000 Btu/hr-ft², burned out somewhat below its center forming a rather large crater and exposing what was once molten UO_2 . The Zircaloy-2 clad in the immediate vicinity of the crater was apparently heavily oxidized.

In order to determine whether the melting of the UO_2 was confined to the immediate area of the crater or whether it had occurred throughout the length of the rod, the bottom of the specimen was sectioned close to the end cap. Examination revealed that the interior of the rod had melted almost to the end cap and to within approximately 1/32 in. of the edge of the pellet. Three distinct concentric zones were distinguished. The outer zone resembled the original sintered material. The zone immediately adjacent to it was composed of columnar crystals which had recrystallized from the melt while the center had a rather porous crystalline appearance.

Although external examination of the remainder of the rods of the loop revealed no damage, the rod adjacent to the damaged rod was sectioned for examination at the end nearest the burn-out rod. This specimen, which was designed to operate at a heat flux of 425,000 Btu/ft²-hr but may have operated at higher heat fluxes, showed gross cracking but no fusion of the UO_2 .

New reference design specimens containing 6.9 w/o U^{235} are now being fabricated for a new test. It is expected that these will be completed by January 26, 1956, for shipment to Chalk River.

Thermal Cycling

The Zircaloy-2 clad UO_2 rods and the empty controlled specimens which have been cycled in NaK between 550° and 1050°F as described in WAPD-MRP-56 and 58 have been measured and radiographed at 7000 and 10000 cycles.

These measurements indicated a net length increase of 0.16 to 1.6%, a new diameter increase of from -0.02 to +1.0% and a maximum increase in centerline runout at the rod midpoint of 0.048 in. In each case, the largest dimensional change was observed on the empty clad.

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II-B-3 (Contd.)

WAPD-MRP-59

The X-rays at 7000 and 10000 cycles were not noticeably different than those taken at 5000 cycles.

The samples will be replaced in test for an additional 5000 cycles after which one of the rods will be replaced by a fuel rod made according to the new reference specifications, viz. 0.008 in. diametral clearance and 0.060 in. end clearance.

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4. Irradiation Effects on Metallic Fuels (53b31) E. F. LoscoBare Samples of U-Nb

Microhardness measurements on 7 bare U-10 w/o Nb samples, irradiated in NaK capsules in the gamma-quenched and transformed conditions, have been obtained. The samples irradiated in the gamma-quenched condition increased in hardness from 286 DPH to 309-347 DPH, the higher exposure samples (in the range 500 to 820 MWD/T) showing the greatest increase. The samples irradiated in the transformed condition decreased in hardness. These samples had been thermally transformed various amounts in the temperature range of 400 to 500°C. Before irradiation their hardnesses were 558 to 355 DPH, but after irradiation they were 367 to 311 DPH and corresponded fairly closely with the values for samples irradiated in the gamma-quenched condition. These results are in agreement with previously reported density and electrical resistivity changes and indicate an apparent irradiation-induced reversion of the alpha-plus-gamma transformed structure to the high temperature stable gamma structure. No variation in hardness from surface to center of the 3/16 in. diam cross section of samples was noted, indicating that the temperature difference between surface and center during irradiation had little or no effect.

Corrosion tests were conducted in 650°F water on samples cut from each of the irradiated samples. All transformed samples disintegrated within 24 hr. Unirradiated and transformed samples behave similarly so that effects of irradiation cannot be established. Gamma-quenched and irradiated samples showed an increase in corrosion rate (the increase depending on the amount of exposure) when compared with unirradiated control samples.

Clad Samples of U-Mo

Changes in volume resulting from irradiation were reported in WAPD-MRP-58. No further work is contemplated on these samples. All results obtained to date will be included in a comprehensive report on irradiation effects which is now being prepared.

Clad Samples of U-Nb

The 10 samples of gamma-quenched Zircaloy-2 clad U-10 w/o Nb, fabricated in various manners and irradiated in Materials Testing Reactor (MTR) process water to an exposure of approximately 2000 MWD/T, are continuing in corrosion test in 650°F water. To date the specimens have undergone 21 days of test without failure. Control samples being tested in the same autoclave have been in test 12 days with one failure, a gamma-quenched sample of an extrusion straightened by cold swaging which failed in the 5 to 12 day test period.

Loop Tests

The 10 samples of Zircaloy-2 clad U-11 and -12 w/o Mo, from the Chalk River X-3 loop section, and the 7 samples from the same extrusions which were irradiated in MTR process water, have been corrosion tested in 650°F water

for 9 days. Six of the Chalk River samples and 2 of the MTR samples plus 2 unirradiated control samples failed during this test period. No correlation with composition or exposure can be found. The indications are that the corrosion properties and/or the variability inherent in the extrusions are responsible for the early failures.

Volume Expansion of Clad Fuel Elements

As described in WAPD-MRP-58, an experiment involving the irradiation of Zircaloy-2 clad natural uranium rods having hollow centers is planned to evaluate the feasibility of this procedure in lengthening the time for cladding failures due to irradiation-induced volume expansion of the fuel. Specimens are being fabricated. The latter are 1 in. long, approximately 0.244 in. OD and 0.094 in. ID with 0.030 in. thick Zircaloy-2 cladding and Zircaloy-2 end caps. The material will be beta quenched after fabrication to reduce the effects of anisotropic growth. Irradiation will be performed in NaK-filled Zircaloy-2 capsules.

Bare Samples of U₃Si

Seven NaK-filled capsules containing samples of unclad U₃Si machined from a pre-epsilonized and extruded ingot have been shipped to MTR for irradiation. Density, hardness, dimensions, electrical resistivity and corrosion data will be obtained. Due to the lack of suitable facilities at the MTR, the samples could not be charged during the current cycle. It is tentatively planned that these samples will be charged during the week of January 16, 1956, for a 6-week irradiation period.

Irradiation of Transformed Specimens of U-Mo Alloys

It was indicated in WAPD-MRP-58 that although the density, electrical resistivity, and microhardness of the U-Mo alloy specimens which were thermally transformed prior to irradiation showed that these specimens had reverted to the metastable gamma phase, the microstructure of the specimens at low magnification (250X) appeared unchanged. The microstructure of a U-9 w/o Mo alloy specimen which was transformed prior to irradiation has been examined at relatively high magnification (1500X) and the microstructure still was typical of the transformed alpha-plus-epsilon phase. This specimen was then shipped to Hanford Atomic Products Operation and an X-ray diffraction pattern was obtained using their double-crystal spectrometer. This pattern showed that the specimen was in the gamma phase and no alpha uranium was detected. Five diffraction lines were obtained and were analyzed using the Nelson and Riley extrapolation method. It was determined that the lattice parameter a_0 was equal to 3.4203 which corresponds to an 8.8 w/o Mo alloy. Chemical analysis on the ingot from which the specimen was obtained showed a molybdenum content of 8.8 w/o. From these results, there is no question but that irradiation of thermally transformed U-Mo alloys causes a reversion of the structure to the corrosion-resistant gamma phase. While X-ray data are not available for the U-10 w/o Nb alloy, measured physical property changes indicate a similar reversion.

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II-B-4 (Contd.)

WAPD-MRP-59

Details of the in-pile experiment in the Brookhaven reactor wherein the kinetics of the phase reversal reaction will be determined are currently being examined. Tentatively, the kinetics of the following reactions in U-Mo alloys will be investigated:

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

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

Corrosion testing of defected simulated fuel elements prepared with extruded U-3.8 w/o Si rods and an 0.035 in. thick barrier layer of Al-6 w/o Cu alloy is continuing. Two elements tested in 650°F water failed after 49 days. Two other elements are still in test after 77 days. It was found that element failure occurs by attack of the core ends by water infiltration through the poor bond between the Zircaloy-2 clad and the barrier layer. The maximum life obtained in 680°F water was 50 days. Failures at this temperature were also attributed to poor clad-to-barrier bonds.

It was found that a good clad-to-barrier bond can be produced by melting the aluminum barrier alloy inside a closed-end Zircaloy tube in vacuum and holding at 1000°C for a few seconds before allowing the alloy to solidify. Samples prepared in this way with the aluminum alloy exposed at one end and a 0.040 in. diam defect through the clad have withstood 98 days in 650°F water with no bond-line attack. By boring out excess aluminum alloy, leaving a bonded layer on the inside of the Zircaloy tubing, an excellent precoated tube is obtained. This coated tube is suitable for subsequent brazing of U₃Si rods at temperatures low enough (700-800°C) to prevent excessive attack of the core alloy. Simulated fuel elements prepared in this way display excellent clad-to-barrier and barrier-to-core bonds. Several elements are being prepared using various aluminum-base alloys as the barrier layers.

Bonding of Aluminum Barrier Layers to Zircaloy and U₃Si

Other methods of bonding the aluminum barrier to both the Zircaloy-2 clad and U₃Si core are being investigated in an effort to simplify obtaining fuel element samples for further development work. Results to date indicate that vacuum evaporation of aluminum on the inside surface of preheated Zircaloy tubes appears promising, producing coatings which are uniform in thickness. Thicknesses in the order of 5 mils can be developed without difficulty. Replacement deposition of metallic layers of zinc, copper, nickel, and iron from fused salt or acid solutions are also being investigated. Other methods of precoating the Zircaloy, such as dipping in molten aluminum, use of fluxed surfaces and mechanical abrasion of the refractory zirconium oxide during immersion of Zircaloy tubes in molten aluminum have not been entirely satisfactory.

Epsilonization of U-3.8 w/o Si Alloy

It has been definitely established that carbon contamination of U-3.8 w/o Si is responsible for lowering of the peritectoid temperature (930°C) and resultant variability in corrosion behavior of samples epsilonized for various times at temperatures up to 900°C. Samples epsilonized for 7 days at 800°C have 21-day corrosion rates in 650°F water which appear to be directly related to carbon content. Samples with 400 ppm of carbon have

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rates of -0.44 to -0.47 $\text{mg}/\text{cm}^2/\text{hr}$; those with 1000 ppm of carbon have rates of -0.45 to 1.75 $\text{mg}/\text{cm}^2/\text{hr}$, while samples with over 2000 ppm of carbon generally disintegrate in less than 7 days. Metallographic examination of these samples after epsilonization indicated transformation to the desired epsilon (U_3Si) phase for the low carbon material, only partial transformation for material containing 1000 ppm of carbon, and essentially no transformation for the higher carbon samples.

To avoid carbon contamination, modifications in melting practice are being investigated. Various melting crucible materials, such as BeO , ZrO_2 , MgO , BeO -washed ZrO_2 , and improved BeO -washed graphite (previously used) are being evaluated. Preliminary hardness data on U-3.8 w/o Si melted in BeO and ZrO_2 crucibles and cast into silica-coated investment molds indicate that this material is considerably softer than similar material melted in BeO -washed graphite crucibles. Hardnesses in the order of 19 to 23 R_c , compared to 25 to 45 R_c for the carbon-contaminated material, were obtained. Cast U_3Si samples representing a wide range of carbon content are now available. This material will be used to establish the upper limit of acceptable carbon content. The optimum epsilonizing heat treatment will be redetermined for material with acceptable carbon content.

Corrosion of Bare Ternary Alloys

Samples of arc-melted buttons, supplied by Armour Research Foundation, of U-3.8 w/o Si plus 0.2 w/o Mo, Nb, Cu, or Ta are being tested in 650°F degassed water. The 0.2 w/o Nb samples failed in 77 to 84 days, the 77-day rate prior to failure was -0.83 $\text{mg}/\text{cm}^2/\text{hr}$. The 0.2 w/o Cu samples failed in 84 to 91 days with an 84-day rate prior to failure of -0.87 $\text{mg}/\text{cm}^2/\text{hr}$. The 0.2 w/o Mo samples are still in test after 105 days with a rate of -0.68 $\text{mg}/\text{cm}^2/\text{hr}$, and the 0.2 w/o Ta samples have a corrosion rate of -0.87 $\text{mg}/\text{cm}^2/\text{hr}$ after 56 days. It is apparent from these data that the 0.2 w/o Mo samples are the most promising from both a corrosion life and rate point of view.

Cast samples were prepared at Bettis Plant with the same analysis as the Armour buttons to check the above results. The material supplied was melted in BeO -washed graphite crucibles and found to be very high in carbon. Metallographic examination revealed the alloys to be incompletely transformed by the epsilonization treatment at 800°C for 7 days. Corrosion data verified the poor quality of this material. New alloys melted in BeO crucibles have been prepared. Seven-day corrosion rates on these alloys are -0.45 , -0.87 , and -0.78 $\text{mg}/\text{cm}^2/\text{hr}$ for alloys containing 0.2 w/o Mo, Cu, and Nb, respectively. These results are in good agreement with those obtained for the Armour button samples. However, it is too early to tell whether these ternary additions are effective in increasing corrosion resistance since the binary alloy, when low in carbon, has shown rates as low as that for the 0.2 w/o Mo bearing alloy. Aluminum apparently deteriorates corrosion resistance. As little as 0.2 w/o causes the samples to disintegrate within 7 days.

Additional U-3.8 w/o Si alloys with small additions of Ti, V, Zr, Ta, and Ge have been prepared for test and will be corrosion tested during the next report period.

Corrosion Mechanism for Epsilonized U-3.8 w/o Si

The investigation of the mechanism of corrosion of U_3Si in 650°F water was continued. Hardness measurements before and after exposure to hot water for 28 days showed no changes. In the case of gamma-phase alloys, considerable increase in hardness would occur under the same test conditions because of hydrogen absorption during corrosion. Based on these results little or no hydrogen is absorbed during corrosion of U_3Si . Analysis for hydrogen is being made to confirm this.

A small portion of the irregular surface phase, which was previously reported as forming between the oxide and metallic sample surfaces during hot water corrosion of U_3Si , was isolated for X-ray identification. This phase, which was previously believed to be delta (U_3Si_2), was found to correspond more closely to a higher silicon compound USi_2 . Three strong intensity lines, 2 of medium intensity and 5 of weak intensity, obtained from a powder photograph (made by special adaptation of the Weissenberg camera) leaves little doubt of the identity of this phase. Preferential corrosion of the uranium at the surface of the U_3Si apparently causes a build-up of silicon and formation of the USi_2 layer. Silicon then diffuses toward the core to maintain this layer while corrosion proceeds at its exposed surface. The relative corrosion resistance of U_3Si and USi_2 will be determined for specially prepared samples.

The presence of 0.2 w/o Mo, Nb, or Cu results in a rapid build-up of the surface layer corresponding to USi_2 in the binary alloy. After 7 days in 650°F water this layer is about 0.002 in. thick (compared to 0.0005 in. for the binary alloy). Beyond this time the layer thickness remains approximately constant. This layer stains more readily on metallographic etching. X-ray identification is being made.

Stringers in Zircaloy-2

Previous data on stringers found in Zircaloy-2 rods and tubing indicated that they might be gas voids (helium) introduced during melting. Annealing above 1100°C indicated that the stringers could be removed in thin-walled tubes. To obtain more data on the nature of the stringers, solid rod samples are being heat treated in the range 1100 to 1500°C. Quantitative measurements on their size and distribution will be made as a function of heat treatment. Preliminary information indicates that a very short anneal (1 to 10 min) at 1700°C will expand the stringers, suggesting that they are of gaseous origin. This treatment permits easier determination of size and distribution. Changes in size, shape, and distribution will be followed by subsequent annealing at lower temperatures for longer times.

Aluminum-Base Alloys for Cladding Applications

Screening corrosion tests were continued on 3 experimental aluminum alloys under static, dynamic, and semi-dynamic conditions. The alloys tested were X-2219 (Al-6 w/o Cu), M-388 (Al-1 w/o Ni-0.5 w/o Fe) and M-400 (Al-1 w/o Ni-1 w/o Fe). It was reported in WAPD-MRP-58 that the M-388 alloy

had failed after 1 week in 680°F water. An attempt to duplicate this result showed that this alloy in 3 different tempers held up in 680°F H₂O for 49 days. No explanation can be given at this time for the earlier result. The other two alloys have satisfactorily withstood exposures to 680°F H₂O, X-2219 for 98 days, and M-400 for 56 days. The corrosion rates (assumed equal to the limiting slope of a plot of weight gain per unit area versus time) are X-2219, 0.25; M-388, 0.08; and M-400, 0.11 mg/cm²/day. Similar corrosion rates were evidenced from exposure to the steam phase of 680°F H₂O. The addition of hydrogen (300 psi) to static 680°F H₂O had a negligible effect on the corrosion rates of all of the alloys. On the other hand, all 3 alloys including various tempers of each failed after exposure to 680°F steam for times ranging from 42 to 77 days.

Results of circulating water tests for 6 weeks at 680°F under both high and low pH conditions indicated that two of the alloys (X-2219 and M-400) had corrosion rates lower than those from the static neutral water tests. The M-388 alloy, however, blistered badly after only one week under both the high and low pH conditions.

All three alloys have held up from 55 to 81 days in a dynamic loop at 600°F at a controlled pH of 8.5 to 9.0, but the general appearance of the specimens was bad. Pitting occurred in 2 of the alloys (X-2219 and M-388) and all of the alloys formed a flaky scale in addition to the oxide film. Because of this easily removable scale, no significance could be attached to weight change measurements.

Results of the semi-dynamic rotating disc tests for 21 days in 600°F H₂O were equally inconclusive. Though corrosion was not severe, formation of the flaky scale made weight change measurements unreliable.

6. Ceramic Fuel Materials Development (53b42) Z. M. ShapiroProperties and Preparation of UO_2

Work was continued on the preparation of pure UO_2 and UO_2 containing additives for irradiation and fabrication tests as well as for investigations on their physical and chemical properties.

Hydrogen Reduced UO_3

During this period five kilograms of uniformly enriched UO_2 and oxide containing SiO_2 and BeO additives were prepared for the irradiation program by hydrogen reduction of UO_3 at $800^\circ C$.

Activated UO_2

In order to determine the action of ball milling in improving the sinterability of UO_2 , slurries of Mallinckrodt UO_2 (64 ml distilled water/200 g UO_2) were ball milled for 15 hr and then dried in a vacuum oven at $70^\circ C$. When isostatically pressed at 55,000 psi and sintered in hydrogen at $1750^\circ C$, pellets were obtained with densities up to 96% of theoretical. Chemical analysis indicated considerable oxidation had taken place during ball milling and that the O/U ratios were increased from 2.03 to 2.15. Further studies will be made to determine the separate effects of particle size reduction and high O/U ratio.

Oxygen Doped UO_2

Samples of uranium oxide with O/U ratios of approximately 2.2 were prepared by controlled oxidation of Mallinckrodt UO_2 at 200 to $240^\circ C$ and annealed in vacuo for 72 hr. No U_3O_8 could be detected by X-ray analyses. These powders were pressed at 175,000 psi and sintered in flowing argon having a dew point of $-14^\circ C$ at $1550^\circ C$ for 19.5 hr. Chemical and X-ray analyses are in progress to determine whether any change in O/U ratio or composition occurred on pressing and sintering. If the O/U ratios prove satisfactory, corrosion testing will be carried out in degassed water and steam, to determine the effect of oxygen content on the corrosion resistance of uranium oxide.

Storage Stability

Uranium oxide samples containing 2 to 6% silica were found to have a marked susceptibility towards air oxidation at room temperature.

Employing the technique described in WAPD-MRP-58, some correlation was observed between the surface area and storage stability of various UO_2 powders. During the standard 30-day test at $30^\circ C$, powder prepared from precipitated ammonium diuranate with a surface area of $60 \text{ m}^2/\text{cc}$ increased 2 to 4% in U(VI) content; hydrided steam oxidized UO_2 with a surface area of $22.6 \text{ m}^2/\text{cc}$ increased 1.5% in U(VI), and MCW with a surface area of $6.2 \text{ m}^2/\text{cc}$ increased only 0.2% in U(VI) content.

Surface Area and Density of UO_2 Powders

Surface area and density measurements have been performed on UO_2 powders prepared from two types of ammonium diuranate. The ammonium diuranate powders were prepared from uranyl nitrate solutions by precipitation with either ammonium hydroxide (ADU-13) or urea (ADU-14). The surface areas and densities of the diuranates were discussed in WAPD-MRP-57. The diuranates were then pyrolyzed to U_3O_8 and reduced with hydrogen to UO_2 . The specific area and density of the UO_2 derived from ADU-13 were $59.9 \text{ m}^2/\text{cc}$ and 11.11 g/cc respectively. The values for UO_2 prepared from ADU-14 were $60.4 \text{ m}^2/\text{cc}$ and 11.34 g/cc .

These surface area values are the highest obtained on UO_2 powders to date. The largest area measured previously was $22.6 \text{ m}^2/\text{cc}$ for UO_2 prepared by steam oxidation of UH_3 . However, British workers have reported surface area values as high as $100 \text{ m}^2/\text{cc}$.

The data are as yet insufficient to determine whether there is some relationship in surface area between the parent ammonium diuranates and the derivative UO_2 powders. Further studies are in progress to check this point.

The high densities of these powders are probably due to their high O/U ratios (2.08 to 2.14). Densities around 11.2 have been reported in the literature for UO_2 powders with O/U ratios of 2.10.

The surface areas of several UO_2 green compacts, prepared from MCW UO_2 by pressing to 68% of theoretical density, were determined. The value obtained was $1.09 \text{ m}^2/\text{g}$ whereas the surface area of the original powder was $0.63 \text{ m}^2/\text{g}$. Thus it appears that pressing breaks up the powder to some extent, and increases its surface area.

In order to check the validity of the surface area measurements, a TiO_2 powder of known surface area obtained from the National Bureau of Standards was measured. The value of surface area obtained by this laboratory was in good agreement with the known value.

Thermal Analyses

Thermogravimetric and differential thermal analysis curves have been obtained from Mallinckrodt and steam oxidized UO_2 . Preliminary results are in substantial agreement with those obtained at Oak Ridge National Laboratory. In all cases a weight loss, due to the evolution of water, occurred at approximately 100°C . On heating in air, both the DTA and weight change curves indicated two exothermic oxidation reactions, $\text{UO}_2 \rightarrow \text{U}_x\text{O}_y \rightarrow \text{U}_3\text{O}_8$. For the high pressure steam oxidized and Mallinckrodt UO_2 , the intermediate compound was U_3O_7 . However, UO_2 prepared by the low pressure steam oxidation method oxidized to a compound with an O/U ratio of 2.21 (U_2O_5 , U_4O_9). The composition of these intermediate oxides will be checked by X-ray diffraction methods.

The stability range of U_3O_8 was between 400 and 600°C and varied with the type of UO_2 . The U_3O_8 then began to lose oxygen gradually, reaching an O/U ratio of 2.60 at 1300°C. On cooling there was a gradual increase in weight and the oxide reverted to U_3O_8 .

Carbon and Silica Doped UO_2

Studies designed to determine the sensitivity of sintered UO_2 compacts to various impurities were continued.

Previous work has shown that carbon in amounts up to 650 ppm does not affect the corrosion resistance of UO_2 pellets when sintered at 1450°C for 2 hr in argon. It was postulated that more severe sintering conditions are necessary for the interaction of UO_2 and carbon and for migration of the impurities to the grain boundaries. Accordingly, a series of pellets containing carbon was sintered in flowing argon at 1550°C for 10 hr. Although the carbon content of the green pellets was approximately 140, 610, 860, and 6000 ppm, chemical analysis of the sintered pellets revealed that the carbon content was reduced to approximately 10 ppm by the sintering. In view of the fact that the dew point of the purified argon was found to be only -14°C, it is felt that water vapor present in the argon reacted with the carbon to form carbon monoxide and hydrogen and that these gases were subsequently swept away by the argon. Appropriate steps are being taken to reduce the water content of the argon in order to minimize carbon loss during sintering. Moreover, a static atmosphere will be maintained during some of the experiments. Much of the work on carbon addition will be repeated to determine whether the previous results will be reaffirmed.

Since distillation of silica from the furnace during sintering offers the possibility of contamination, a series of silica-doped pellets have been corrosion tested. It has been previously reported that samples containing up to 860 ppm silica which were sintered in either argon or hydrogen at 1450°C for 2 hr did not desinter during corrosion testing. It was felt that more severe sintering conditions were necessary to bring out the adverse effects of silica. Accordingly, compacts made of the silica-doped UO_2 which was used for earlier studies were sintered in argon at 1550°C for 10 hr and in hydrogen at 1750°C for 8 hr. Subsequent corrosion tests of the pellets in 640°F water failed to cause breakup or significant weight change. Chemical analyses of the sintered pellets are now being performed.

A UO_2 powder containing 2000 ppm silica has now been prepared and will be compacted, sintered, and tested for corrosion resistance.

The Sintering of UO_2

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

Compacts which were cold pressed from as-received MCW UO_2 to green densities ranging from 51% to 73% theoretical, and subsequently sintered in

hydrogen at 1750°C for 2 to 64 hr, were examined microscopically. It was observed that many of the particles present in original powder tend to be preserved in the green compacts and form higher density spherical volumes in a matrix of generally lower density. These higher density volumes then sinter to a higher density than the matrix and create voids in their immediate vicinity. This effect tends to become less evident the higher the green density of the compacts. It was also noted that the grain size increases with sintering time and becomes more uniform the higher the initial green density. Thus it appears that the final density and structure of cold pressed and sintered UO_2 is greatly dependent on the character and uniformity of the starting powder.

This work will be extended to compacts made from other size fractions and other types of UO_2 .

Corrosion Failure of Production UO_2 Pellets

It was reported in WAPD-MRP-58 that a number of production run UO_2 pellets had failed during corrosion testing due to non-uniform shrinkage. This difficulty has now been overcome by lengthening the soak period during sintering from 5 to 8 hr.

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 94 days in degassed water at 650°F, 77 days in degassed water maintained at a pH of 10.5 and 650°F, and for 98 days in degassed steam at 750°F. Although, in general, no significant changes were found by observing and weighing the specimens, losses in weight of several milligrams were noted in several cases. Small deposits of powder were also found on the bottoms of the containers holding these samples. This attack does not appear to be chemical in nature, however, and is felt to stem from some erosive action.

II-C-1C. Chemistry and In-Pile Loop Engineering1. In-Pile Screening Tests of PWR Fuel Materials (53b33) A. P. Millera. HB-5 (MTR-WAPD-13) Loop

The WAPD-13 loop in the HB-5 beam hole of the Materials Testing Reactor (MTR) is being used to determine the effect of thermal cycling on a defected UO_2 specimen. The estimated burnup on this specimen is about 2600 MWD/T as of January 12.

The loop has been operating satisfactorily with the specimen being cycled in and out of flux at the frequency indicated below:

<u>MTR Cycle No.</u>	<u>Date</u>	<u>Cycling Conditions</u>
61	Nov. 14 - Dec. 3	Fast - 6 cycles per hr
62	Dec. 11 - Dec. 27	Manual slow cycling - 1 hr in flux and 7 hr out of flux

During reactor cycle 63, December 31 to January 16, the specimen will be fast cycled 6 days, manually slow cycled 6 days and held stationary in flux 2 days.

b. Shim Rod Loop (WAPD-25)

The design effort for the modifications required to the WAPD-13 external loop equipment and for an in-pile tube for the shim rod position have been completed. Procurement of all components is proceeding satisfactorily.

Pending approval of the irradiations program by the Atomic Energy Commission (AEC) and the Materials Testing Reactor (MTR), the loop is scheduled for in-pile operation for at least 3 reactor cycles starting in March 1956.

2. In-Pile Radiation Exposure Tests of PWR Fuel Materials (53b34) A. P. Millera. X-1 Loop

The fourth UO_2 defect test in the X-1 loop of the NRX Reactor, Chalk River, Canada, commenced at 0010 hours on December 15. The purpose of this test was to obtain data on high burnup and high heat fluxes for reference PWR blanket fuel elements and to obtain more detailed radiochemistry data. The test was terminated at 1130 hours on December 16, with only 25.5 MWD accumulated because of intolerable activity in the reactor working areas caused by a rupture in the cladding of the 22% (w/o U-235 in uranium) specimen which was located in the center of the 5 specimen assembly.

The irradiation proceeded normally from start-up at 0010 hours on December 15, to 0315 hours on December 15, when a burst of high activity occurred, causing a reactor trip. The reactor was restarted at 0421 hours but tripped again because of high activity at 0507 hours. Subsequent operation included a number of trips and starts due to other causes, at low power, and additional trips and starts at 34 Mw. During the latter period the activity rose to very high levels. At 1130 hours on December 16, after approximately 11.5 hr of 34 Mw operation, the reactor was deliberately tripped to observe the effects on activity level. The activity monitors on all the loops in the reactor went off scale and the main pile room became a respirator area. The decision was then made to terminate the experiment. The test section was removed from the in-pile position at 2000 hours on December 16.

The test section was cut at Chalk River, and the samples observed in the Atomic Energy of Canada, Limited, (AECL) hot lab. The 20% sample (center) had ruptured, approximately at its center, with little bulging. A piece of cladding about 1-1/4 in. long by about 3/8 in wide had disappeared. UO₂ had been lost from this area, and the residual oxide showed signs of fusion. The other samples appeared to be normal. All of the samples were shipped to Bettis Plant, where they were cut and examined on December 29. Details of specimen examination are given in Section II-B-3.

Analysis of the observed thermal output of the test section indicated that the actual average heat fluxes at 35 Mw pile power were 850,000 and 400,000 Btu/hr-ft², for the central specimen (20%) and the adjacent specimens (8.35%), respectively.

Specifications have been set for another test. Five specimens will be tested of which 3 will have a 5 mil defect. The enrichment will be uniform (6.9%) and the design peak heat flux in the number 2 and 4 specimens will be 400,000 Btu/hr-ft². A proposal and schedule for this test are being prepared.

Following the removal of the test section, the loop was flushed with cold water circulated by the loop pump. Circulation commenced on December 23 and by January 4 the general area activity decreased from 275 mr/hr to 40 mr/hr. Additional decontamination steps will be taken as permitted by the schedule for the next test.

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

a. NRX-CR-V

The PWR fuel material and fuel bundle test at plant operating conditions is proceeding satisfactorily.

The test section pressure drop has exceeded 400 in of water which is the maximum range of the existing differential pressure transmitter. It has been observed in the past that sudden increases in loop flow decrease

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the test section pressure drop due to a "washing off" of the test section. This procedure will be performed during the next reactor shutdown in order to return the pressure drop reading to on scale. The increased pressure drop has not endangered the maintenance of the specified loop flow.

The estimated burnup on these specimens as of January 12 was 1520 MWD/T.

The experiment is scheduled to terminate on March 9, 1956.

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, Fourth Defect Test, UO₂

This test, which began with reactor start-up at 0010 on December 15, was terminated at 1130 on December 16, because of excessive activity and radiation levels. Gamma monitor activity rose from about 10^4 c/min with no absorber between the monitor and the tubing line containing the loop water to greater than 2.5×10^4 c/min with 6 in. of lead between the monitor and the tubing. The reactor tripped several times during the test, and the gamma monitor readings rose sharply after almost every trip. Following a deliberate trip at 1130 on December 16, the general radiation level in the vicinity of a portion of loop piping rose from 7 r/hr to 60 r/hr. Dissolved Cs¹³⁸ activity ranged from 1.7 to 10^5 dis/min-ml to 3.5×10^9 dis/min-ml, the highest value occurring shortly after the loop gamma monitor first tripped the reactor at 0315 on December 15. Kr⁸⁸ activity was 3×10^9 dis/min-ml. The delayed neutron monitor was inoperative because of the high ambient gamma field.

Following removal of the test section and draining of the loop, the radiation level in contact with the lagged 1-in. loop piping decreased from about 900 mr/hr on December 18 to about 300 mr/hr on December 22. Further decay and decontamination resulting from circulation of cold (200°F) water resulted in a further reduction to 15-30 mr/hr by January 4. In contrast to the low levels of permanent pipe contamination recorded, the section of the shielding plug immediately downstream of the test section read 9 r/hr on December 30, and a spot on the strainer body (which does not contain a screen) in the pump suction line read greater than 2.5 r/hr on January 5, indicating deposition of fuel material at these positions.

Analysis of the material collected on the ion exchanger and cold filter, and portions of the system piping, is under way. The hot, full-stream filter had not been installed when the test was terminated.

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b. Central Thimble (CR-V)

This proof test of PWR type subassemblies continued to run without incident, except for the continual increase in pressure across the in-pile test section and shielding plug. Design flow rate can still be maintained, however. Activities remain low, about 4.4×10^4 dis/min-ml before the filter, 4.4×10^4 dis/min-ml before the ion exchanger, and 4.4×10^2 after the ion exchanger.

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

Operation of this test continued satisfactorily until its termination at reactor shutdown on December 12. Operation of the ion exchanger beginning December 1, reduced the gross beta-gamma activity from 5×10^5 c/min-ml to 8×10^3 c/min-ml in 24 hr. The activity remained at about this level throughout the rest of the test. Observations indicated that little short-lived activity was being released until just prior to the end of the test, when the gamma and delayed neutron monitors once again responded to a reactor shutdown.

d. HB-5 (WAPD-13-6) Defected UO_2

This test continued throughout the period. Under slow cycling (3 cy/day), peak activity levels of Cs^{138} and Ba^{139} were approximately ten times lower than under steady irradiation. The same three peaks observed on the delayed neutron monitor with fast cycling are noticeable with slow cycling. The sensitivity of this monitor to gamma radiation or external neutron fields is shown by the fact that, even without flow through the monitor, the first peak is still seen, the second peak is evident but at a delayed time, but the third peak does not appear. Samples taken at various times after insertion of the fuel element indicate that the peak water activity occurs between 10 and 20 min after insertion.

5. PWR Radiochemistry (53b37) P. Cohen

Activity was confined to radiochemical analyses for PWR Chemistry Operation and PWR Fission Product Removal. A total of 138 determinations of Ba, Ce, Cs, Sr, Y, Fr, and I were completed.

6. Fission Product Removal Process Development (55b12) W. T. Lindsay, Jr.a. Examination of Contaminated Facilities(1) Bettis Loops R-1 and R-3

The first of a group of fission product radiochemical analyses in conjunction with these studies has been received. Their significance will

be established pending further analyses for uranium.

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

A second run was completed in Bettis Loop R-3 in establishing further hydraulic and operational data for a radial-flow bed filter (WAPD-MRP-58). During this second run, it was found that the increase in pressure drop upon addition of synthetic crud corresponded closely to the data obtained during the first run for equivalent amounts of crud; e.g., 6.4 psi and 5.2 psi at 600°F, respectively, with the addition of 78 g of crud and a flow rate of 18 gpm. The flow loading was 24 gpm/ft², based on the inside bed area. However, the pressure drop rose rapidly with further crud addition, leveling off during the latter stages of the run. The resulting S-shaped curve might be attributed to plugging followed by redistribution of crud within the bed proper.

The maximum pressure drop found during the second run was 15.9 psi (600°F) at a flow rate of 16.5 gpm, with a total of 122 g of crud charged to the loop and approximately 102 g of crud accounted for in the bed. A bed filter efficiency of approximately 85% was determined. Pressure drops were proportional to the 1.6 - 1.8 power of flow rate for the fouled bed compared to 2.0 for the clean bed.

Operational data obtained with flow, temperature and pressure variation demonstrated satisfactory crud retention within the bed during transient periods of operation.

In general, the results obtained from two continuous crud addition runs in Bettis Loop R-3 demonstrated that the operating characteristics of the bed, including leaching of the filtering media, retention of crud within the bed following filtration, and hydraulic parameters, are compatible with its application of the NRX-X-1 facility. The filter may accumulate approximately 100 g of loop crud with a corresponding pressure drop of only 7.0 psi at 10 gpm (600°F), the operating full stream flow rate of the NRX-X-1 facility.

The bed filter efficiency for radioactive UO₂ particles could not be determined in the available time. A defected U-Mo element (0.05 curies), heat treated to yield catastrophic failure, was inserted in Bettis Loop R-3 subsequent to the second crud addition run. However, no indications of fuel element failure were observed after some 70 hr of operation at 600°F, 2000 psi and an additional 150 hr at high O₂ concentrations (20 cc/kg). Accordingly, some irradiated U-Mo corrosion product was injected into the loop upstream of the bed filter with some 30% of the total charge being accounted for in the bed. Further evidence as to the bed filter effectiveness with respect to filtering of particulate activities will be obtained from HNO₃-HF pickling of the pipe sections before and after the filter in service during the irradiated sludge additions.

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The bed filter housing was removed from Bettis Loop R-3 and shipped to the NRX-X-1 facility to provide information which will assist in evaluation of the behavior of UO_2 fuel specimens under test in that facility. A summary report containing the results of the bed filter test (WAPD-AT-172) has been issued.

c. Leaching of Activity from UO_2

The first of a group of fission product radiochemical analyses from previous R-loop tests (Phases VI, VII, VIII) has been received. Their significance will be determined pending further analyses for uranium.

Testing has commenced in Bettis Loop R-2 with respect to the effect of accidental (or non-operating) water chemistry on UO_2 fuel stability. The first test (R-2-XI) is operating with 1 irradiated cold-pressed and sintered UO_2 defected element (90% theoretical density) and 3 bare non-irradiated UO_2 wafers, at 600°F, 2000 psi, neutral pH, 10 cc O_2 /kg. Activity release from the irradiated element and weight changes on the bare UO_2 wafers will determine the effect of accidental high O_2 (10 cc/kg) at neutral pH on UO_2 fuel material. Future testing will be conducted with the same O_2 concentration (10 cc/kg) at both low and high pH.

d. Erosion Experiments with UO_2

All testing in this project has been completed. A summary report is under preparation.

e. Behavior of Particulate UO_2 in Turbulent Streams

(1) Bettis C-Loop Experiments

A total of thirteen particle deposition runs have been made in Bettis Loop C-1 at 170°F, 300°F and 525°F in demineralized water and in water with 2.5×10^{-3} M LiNO_3 , 5×10^{-3} M LiOH and 5×10^{-3} M HNO_3 . The runs were carried out in the manner previously described (WAPD-MRP-57, 58). The results confirm the previously reported marked effect of temperature on the rate of deposition of the UO_2 particles in downward-directed tees for the runs with demineralized water. The experiments at higher ionic strength showed a smaller temperature effect on the deposition rate (0.06% collection per pass per downward-directed tee at 525°F and 0.012% collection at 170°F with LiOH , 5×10^{-3} M). At constant ionic strength, the results show little effect on changing from neutral to high pH water. At low pH, the results were obscured by partial leaching of fission product activity from the UO_2 particles. The effect of increasing the ionic strength, at 525°F, appeared to decrease the collection efficiency by a factor of about two (0.048% and 0.06% at the high ionic strength vs 0.13%, 0.09% and 0.08% with demineralized water). Further investigation of the effect of ionic strength, pH and other variations of the

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water conditions by sedimentation studies will be carried out at Armour Research Foundation under Subcontract 73-(14-396). The experiments in the Bettis C-loops have been interrupted by reassignment of personnel.

(2) Scale-up Studies

Experiments which were planned to investigate the effect of dimensions of the fluid conduit (to permit the application to PWR of quantitative results from small-scale experiments) have been interrupted by reassignment of personnel. Arrangements are being made to obtain the services of competent consultants in the field of fluid mechanics to assist in the design of future large-scale experiments and in the interpretation of small-scale experiments.

f. Ion Exchange Process Studies

(1) Removal of Fission Products by PWR Purification System

Experiments with demineralized water containing fission product isotopes have continued. The carrier-free isotopes in water solution are processed through columns of MB-1 resin at an area loading of 7.5 gpm/ft². The procedures were outlined in previous progress reports. The following results have been obtained.

- Cs^{137} - Breakthrough had not taken place at the end of 5500 column volumes. The run was discontinued at this point. The highest decontamination factor (D.F.) obtained during the run was 2.3×10^3 . The D.F. rose as the concentration of the isotope in the influent was increased.
- Ce^{144} - Breakthrough has not appeared at the end of 5300 column volumes. The highest D.F. (2.2×10^4) was obtained when the influent concentration was increased to 4.4×10^4 c/min-ml. The D.F. in this test also rose with an increase in influent concentration.
- Ru^{106} - Activity has not broken through at the end of 5000 column volumes. The D.F. rose as the influent concentration was increased. The highest D.F. (5×10^4) was obtained with a solution having a concentration of 5×10^4 c/min-ml.
- Zr^{95} - Breakthrough occurred at about 3000 column volumes. The highest D.F. (4.4×10^3) was obtained with a solution containing 4.4×10^3 c/min-ml. After breakthrough, the D.F. remained at 15. The concentration gradient of Zr^{95} in the bed was disturbed at about 3000 column volumes, and it is possible that breakthrough would not have occurred at this point if the bed had remained undisturbed.

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- Sr^{90} - Breakthrough has not occurred at 2500 column volumes. The D.F. was 2.5×10^3 at this point, with an influent activity of 7.5×10^3 c/min-ml.
- Mixture - The mixture consists of all five of the above isotopes at equal activities with a total of 6000 c/min-ml. Breakthrough has not occurred at 1100 column volumes. The D. F. has averaged 3.1×10^3 .

The results of these tests have shown that the D.F. for all isotopes has increased as the concentration of the influent was increased.

All columns were monitored. The data show that most of the activity was concentrated in the top 3 in. of the resin bed and the bottom of the column was approximately background. The residual capacity of these columns, except for Zr^{95} , remains quite high after 3000-5000 column volumes.

(2) Effect of Demineralization Load on Activity Retention by Resin

Water with a little crud and a resistivity of 500,000 ohm-cm has been put through the column previously loaded with Cs^{137} to determine the effect of this water on the concentration gradient. After 1000 column volumes the concentration gradient, determined by monitoring, was affected slightly. The activity did not move as a concentrated source. Some activity diffused to a point 9 in. from the top. The remainder of the 30-in. column has been unaffected. The resistivity of the water will be decreased to 300,000 ohm-cm, and the test will be continued until breakthrough.

(3) Thermal and Radiation Damage to Resin

Additional experiments have been made to determine the effect of excessive temperature on the physical properties of MB-1 resin. Portions of the resin were heated, underwater, to temperatures of 400°F, 500°F, and 600°F for 12-hr periods. Some resin fines were produced at 400°F, while most of the resin at 600°F had decomposed after the 12-hr period. In no case were gummy or gelatinous masses produced which would interfere with the resin discharge procedure.

(4) Purification and Instrumentation Test Facility

Reconstruction of Bettis Loop R-1P has been completed. The loop is undergoing a series of tests to determine its operational characteristics and will be available for experimental purposes during the next reporting period.

7. PWR Fuel Element Rupture Studies (55b13) W. T. Lindsay, Jr.

No activity on this project during the report period.

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8. PWR Coolant Development (55b15) W. T. Lindsay, Jr.a. Adsorption of Activity on Particulate Matter in Water Suspension

The purpose of this work is to obtain quantitative information with regard to the adsorption of contaminating radioactive species on particulates, either indigenous (crud) or added, in coolant suspension. Previous information (see WAPD-MRP-58) had indicated that such adsorption on magnetite was appreciable for the cases of Ce^{144} and Ru^{106} .

In an experimental attempt to obtain better data, 300 mg of synthetic crud were added to a stainless steel autoclave containing 1500 ml of water with 7700 c/min-ml of carrier-free Ce^{144} . After attainment of equilibrium at 540°F, the crud was filtered out. Only 185 mg of crud was recovered; the remainder adhered to the autoclave walls. Of the initial Ce^{144} , 11.9% was on the crud, 0.5% remained in solution, and 87.6% was on the walls of the autoclave.

Additional experiments are planned to obtain further information on this subject.

b. Contamination Studies

No activity on this project during the report period.

c. Pretreatment of PWR System Materials

The experiment reported under this topic in the preceding report (WAPD-MRP-58) was repeated in an attempt to confirm the results.

Radioautographs were made of a stainless steel autoclave contaminated by Ce^{144} . Prior to contamination the autoclave was precorroded for 200 hr at 540°F with steam and water. One-half was precorroded in steam, one-half in water. The Ce^{144} tracer solution was then added and the autoclave was rocked through a 240° angle. The radioautograph showed the Ce^{144} activity to be concentrated at the location of the steam-water interface during precorrosion. This is contrary to the previous experiment which showed nearly all the activity concentrated on the surfaces precorroded in the water. This phenomenon will be investigated further.

d. Instruments for Continuous Analysis of Coolant(1) Oxygen Analyzer

Arrangements have been made for the Analytical Division of the Westinghouse Research Laboratories to investigate possible ways of continuously analyzing for small quantities (0.1 ppm or less) of dissolved oxygen in the coolant water. The feasibility of employing (1) a gas chromatography-thermal conductivity method, (2) various reducing agents for a differential

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electrical conductivity method, and (3) polarographic methods is to be studied.

Representatives of the A. O. Beckman Company are interested in re-designing their oxygen analyzer for operation at the conditions existing in the PWR sampling stream.

(2) Hydrogen Analyzer

The effect of hydrogen dissolved in water on the electrical resistance of a palladium wire immersed in the water is being investigated. A calibration curve of the concentration of hydrogen in the water vs. the electrical resistance of the palladium has been partially completed. A linear relationship has been obtained for hydrogen concentrations up to 15 cc (STP)/kg H₂O. The temperature of the water was 410°F. This method of analyzing for dissolved hydrogen appears promising; however, the reproducibility and the response rates have yet to be investigated.

A calibration curve of the concentration of hydrogen dissolved in water vs. the amount of hydrogen diffusing into a palladium tube in a 1-hr period has been partially completed. A linear relationship has been obtained for concentrations of hydrogen up to 15 cc (STP)/kg H₂O. For this calibration curve, non-activated palladium was immersed in water at a temperature of 430°F.

A review of the development status of palladium-diffuser devices for hydrogen analysis, as of December, 1955, has been prepared and has been transmitted to the Atomic Energy Commission.

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

a. Decontamination Experiments

(1) Screening Tests of Reagent Formulations

The evaluation at 150°F of alkaline decontaminating solutions containing oxidants and sequestering agents for cleaning up a PWR loop has been carried out. The results indicate that dilute hydrogen peroxide (with acetanilide as a stabilizer), with an EDTA derivative or an organic phosphate or condensed inorganic phosphate as sequestering agent at pH 11, is quite effective in solubilizing uranium dioxide and does not appreciably attack type 304 stainless steel. Such a solution, under the conditions thus far employed, does not dissolve crud to any practical extent.

Experiments with acidic formulations and investigations at elevated temperatures and with reduced concentrations will be carried out during the next period.

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(2) Simplified Decontamination Procedures

The development of non-draining procedures for decontamination, whereby the purification system ion exchangers are used to collect the radioactivity removed from the system and to clean up the reagents as well, has proceeded with the evaluation of successive alternate treatments with periodic acid and citric acid at very low concentrations and at elevated temperatures (250°F and 300°F).

(a) A series of pilot runs of the proposed cyclic treatment in the Bettis R-3 Loop (contaminated by the addition of radioactive corrosion product of U-Mo alloys during the radial-flow bed filter test) has been carried out. The results show that additions of periodic acid, followed by citric acid, at concentrations of 0.005%, 0.01% and 0.02% by weight, with 0.50 hr contact time at 250°F or 300°F before removal by ion exchange, are moderately effective in transferring the contaminating radioactive species to the ion exchanger. The results are not yet to be considered as conclusive since the level of contamination of the loop was quite low.

Some results indicate that periodic acid solutions are effective in eluting activity from an ion exchanger which has previously accumulated activity from the citric acid solutions. Consequently, 2 ion exchangers are recommended for this process: 1 to remove the periodic acid and 1 to clean up the citric acid.

(b) Simultaneously with the above experiments, tests have been made of periodic acid for stability in the presence of corroded stainless steel at 300°F and 400°F. The results show that, at 0.2% and at 0.02% the periodic acid solution decomposed rapidly at 400°F. At 300°F, 21% of the 0.2% solution had decomposed at the end of the 2 hours, while periodate ion was not detected at the end of the first hour in the solution initially 0.02%. Further investigation of the period of stability of the lower concentration solution at 300°F is under way.

(c) An investigation of the corrosiveness of 0.2 g/l paraperiodic acid and 0.2 g/l citric acid with 0.0125 g/l paraperiodic acid upon the various alloys present in loop systems has been carried out. The experimental data show that at temperatures as high as 300°F these solutions generally do not excessively attack stainless steels, Stellites, Inconel, Monel, copper-nickel and copper-beryllium alloys, and even brass and bronze, which the various loop appurtenances might expose to the system.

b. Waste Disposal

(1) Disposal of Radioactive Gases

Experiments measuring the rate of trace gas removal as compared with the rate of bulk gas removal, by stripping, have been completed. The stripping was accomplished by boiling under reduced pressure. Oxygen was

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used as bulk gas. The method employed for measuring the amounts of oxygen before and after stripping was the Non-Referee Method B, ASTM D888-49T. The results of these experiments show an excellent correlation between fractional removal of bulk dissolved O_2 and trace dissolved Xe^{133} . The 8 experiments conducted differed in time and rate of boiling. In each of the experiments, the percentage removal of O_2 and Xe^{133} differed by less than 9%.

Experiments measuring the rate of diffusion of Xe^{133} from liquid surfaces covered with diffusion barriers have been temporarily interrupted as a result of a shortage of xenon. The available supplies of xenon have been used in the stripping experiments.

(2) Removal of Contaminants from Liquid Waste Streams

(a) Preliminary investigation has been started of the effectiveness of adsorbent material in increasing the decontamination factor (D.F.) obtainable with ion exchange beds. An exploratory test was run with montmorillonite clay. A D. F. of 10^2 was obtained by agitating 300 ml of a 4×10^4 c/min-ml solution of Ce^{144} with 2 g of clay for 10 min. After 48 hr the D. F. was 2×10^2 . When Johns-Manville Celite was added to the solution the D. F. rose to 7×10^2 . This work is continuing.

(b) The necessary pretreatment of wastes containing macro-amounts of impurities, in addition to radioactive contaminants, before treatment by the ion exchange units in the FWR Waste Disposal System is satisfactory, continues to be the subject of investigation. Nitric acid, periodic acid and citric acid, among others, may be present in liquid discharged from the equipment decontamination room. A solution containing 2.5% nitric, 30% periodic, and 67.5% citric acid was prepared with an activity of 2.8×10^5 c/min-ml. The solution was boiled for 1 hr and put through a column of resin. A D.F. of only 3 was obtained, showing that this simple pretreatment is not satisfactory. A program for the systematic investigation of methods of pretreatment is in preparation and work will be started during the next report period.

(3) Radiological Hazards

K_2CrO_4 and Li_2CrO_4 have been suggested as corrosion inhibitors in the neutron shield tank. A concentration of 500 ppm of $CrO_4^{=}$ is desired. A study has been made of the possible hazard resulting from use of Li_2CrO_4 or K_2CrO_4 . A respiratory hazard in the vapor container may be created by H^3 (in the form of HTO) by the Li^6 (Th n, α) H^3 reaction. A radiation hazard may be caused by K^{42} formed by the K^{41} (Th n, α) K^{42} reaction and Cr^{51} formed by the Cr^{50} (Th n, α) Cr^{51} reaction.

The calculations indicate that use of Li_2CrO_4 (500 ppm $CrO_4^{=}$) as a corrosion inhibitor might produce the maximum permissible concentration of H^3 , after a period of 1 year, in regions where the air is saturated

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One can also conclude that use of K_2CrO_4 (500 ppm $CrO_4^{=}$) should produce a radiation intensity, outside the shield tank, which is negligible compared to other sources in the vicinity of the neutron shield tank. Potassium chromate has been recommended as the inhibitor on the basis of low cost and low attendant hazard.

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

Analysis of the new reference design with 32 seed clusters is continuing. The two-dimensional cases reported in WAPD-MRP-58 have been extended to rod programs having 12 and 16 rods fully inserted. For the 12 rod down case the peak-to-average flux in the seed region is 1.40. The blanket peak-to-average flux is 1.93.

The one-dimensional, two-group lifetime studies of the reference design are continuing. Using the GODDESS analog computer, the lifetime study has been carried to 4000 hr in 1000 hr steps. At 4000 hr the reactivity was found to be 12% below the hot, clean critical condition. The results also show that the blanket fraction of the power increases continuously with lifetime. The blanket fraction of the total power is estimated to be 55% in the hot, clean core and increases to about 59% at 3000 hr. At 4000 hr the blanket fraction is about 60%.

The lifetime calculation reported in WAPD-MRP-58 as computed on the UNIVAC was found to be in error due to coding difficulties. These have been corrected and studies are now continuing using 300 hr time steps. Results up to 1600 hr of full power operation have been completed. They are found to agree quite well with the GODDESS results described above.

Temperature and pressure coefficients of reactivity have been calculated for the present reference design at 525°F, 2000 psi, for the condition of fully loaded core plus equilibrium xenon. The results obtained are $-2.1 \times 10^{-4} \Delta k/^\circ F$, and $+2.4 \times 10^{-6} \Delta k/\text{psi}$ for the two coefficients.

Use of the IBM-650 code (ABNER) is continuing for analysis of the experimental slab results. Results obtained with the four-group formulation continue to give substantially better agreement than the two-group formulation.

In several cases of interest to PWR design, a comparison of the power sharing has been made of the AML-17 (UNIVAC) code using 29 points with ABNER code using 100 points. The ratio of seed-to-blanket power is the same to within 2% for both codes and the eigenvalues agree to within 0.3%. The fluxes from the LIL ABNER code satisfy balance check equations (within 0.3%) for these cases, which is not true for the corresponding AML-17 results. Apparently the results obtained with the UNIVAC code are trustworthy despite the lack of neutron balance.

An IBM-650 code (DONATE) mentioned in a previous progress report has been in operation for several months. This code calculates the thermal neutron energy distribution and from this the effective thermal group constants. The DONATE code has been used to compute the diffusion constant and absorption cross section averaged over the thermal flux for unit density water at 20°C. The thermal group is defined as lying in the energy range $0 \leq E \leq 0.625 \text{ ev}$. For two values of buckling the following values are obtained:

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B^2 (cm^{-2})	$\bar{\sigma}_a$ (barns)	$\bar{\Sigma}_a$ (cm^{-1})	$\bar{D}$ (cm)	$L^2 = \bar{D}/\bar{\Sigma}_a$ (cm^2)
0	0.2861	0.01911	0.1524	7.973
.01	0.2856	0.01908	0.1530	8.021

The code failed in running the negative buckling ($B^2 = -0.131 \text{ cm}^{-2}$) case which was to be compared with the experiment of DeJuren and Rosenwasser.^{*} A linear extrapolation of the available computations yields a diffusion length of 2.817 cm, which is 2% higher than the measured value^{*} of $2.763 \pm 0.015 \text{ cm}$.

A scheme has been evolved to obtain the thermal neutron spectrum in a water moderated heterogeneous cell. The two regions (water and fuel material) are decoupled by a method similar to that used in the resonance escape probability calculations, neglecting slowing down in the fuel element. This permits treatment of the fuel region as a boundary condition on the moderator. The flux spectrum may then be obtained by a machine solution of the Wigner-Wilkins equation, which in turn determines the thermal neutron spectrum in the fuel region at every point. Preliminary investigations are being made to examine the effect on successive generation calculations of inhomogeneities and scattering in the fuel element.

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

The two-dimensional calculation of the radiative heating mentioned in WAPD-MRP-58 has been completed. This is the heating which may be expected to exist radially through the thermal shield and reactor vessel. This analysis was required for the new reference design because of the substantially reduced spacing between the core and the thermal shields.

The computation consisted of evaluating the radiative heating along three radial rays, P_a , P_b , P_c , extending from the center of the core through the reactor vessel. The first ray, P_a , was taken perpendicular to the flat face of the core and the other two at $\theta = 28.5^\circ$ and $\theta = 45^\circ$ from the first. The table below is a compilation of the total heating as a function of distance r in centimeters from the reflector, core band interface for each ray.

r cm	P_a	P_b	P_c
	watt/cm ³		
0	1.5	2.5	1.6
4.5	.65	1.1	.77
9.0	.33	.60	.40
20.5	.062	.15	.10

A computation was also made of the radiative heating in the control rods. It was found that the average total heating was approximately 30 w/cm^3 of rod material.

^{*} DeJuren, J. and Rosenwasser, H., Diffusion Length of Thermal Neutrons in Water, Jr. Natl. Bur. Stds., Vol. 51, No. 4, October 1953.

REF ID: A57100

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

The enriched seed to be used in the full-scale PWR mock-up has been re-built to make the microscopic fuel location correspond more accurately with that in the actual PWR core. The rearranged seed was found to have approximately 0.5% less reactivity than the original configuration. The loading of the rearranged seed was actually 5% lower than the original configuration which accounts for the reduction in reactivity.

Furfuryl alcohol was substituted for water moderator in a moderator reflected enriched slab. The furfuryl alcohol moderated slab was found to have approximately the same excess reactivity as the water moderated slab. Vertical flux traverses made through the cores indicated that the vertical reflector savings in the two cases were essentially equal.

Substitution of furfuryl alcohol for water in a core composed of the seed flanked by 12 in. of uranium-oxide blanket material on either side decreased the excess reactivity from 6.5% for the water moderated core to approximately 5%.

Two methods of obtaining rod calibrations, the rod drop and rising period methods, have been compared. Calibrations obtained by the two methods agree within 5% up to a total rod worth of 2.5% in reactivity. Further work will be done on this problem.

Flux traverses were taken through the oxide reflected core containing 7/16 in. wide zirconium crosses in the water channels. The purpose of the crosses is to suppress the flux peaks in the water channels. Indications are that the flux peaks are suppressed but a loss in reactivity is also experienced.

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

Measurements on the first lattice of UO₂ fuel rods have been completed. This is the lattice of 0.600 in. diam fuel, 1.3% enrichment, compacted to 68.77% of theoretical density arranged to give a 3-to-1 W/U volume ratio.

The method previously described for obtaining radial and axial fluxes has been used both by exposing foils of fuel material in the fuel rods (by counting fission gammas above 350 kev) and by counting fission gammas above 350 kev directly from the fuel rods themselves limiting the counting region by means of a lead collimator. The reflector savings obtained from these measurements are listed in the table below.

A. Radial Flux: (by collimator counting of fuel rods)

<u>Run No.</u>	<u>Azimuth Angle</u>	<u>Reflector Savings</u>
1	0°	8.30 ± 0.10
2	+60°	8.77 ± 0.14
3	0°	8.60 ± 0.18
4	+30°	8.65 ± 0.15
5	-30°	8.80 ± 0.10
Average:		8.61 ± 0.20

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B. Axial Flux

<u>Run No.</u>	<u>Method</u>	<u>Reflector Savings (total axial)</u>
1	Foils	$18.26 \pm .34$
2	Rods	$18.20 \pm .28$
3	Rods	$17.84 \pm .32$
4	Rods	$17.78 \pm .34$

Average: $18.03 \pm .30$

The critical buckling using these values is calculated to be $28.23 \times 10^{-4} \text{ cm}^{-1}$.

A lattice of 4-to-1 W/U volume ratio of the UO₂ fuel is being set up and all the measurements performed for the 3-to-1 lattice will be performed for this new lattice.

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

Study and evaluation of the Reactivity Measurement Facility (RMF) at the Materials Testing Reactor (MTR) are continuing. Preliminary results indicate the advisability of using period measurements to determine reactivity. Equipment for this purpose has been ordered. Some of this has already been obtained and is being checked out.

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