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WAPD-MRP-47 (Rev.)

REACTORS—POWER

TECHNICAL PROGRESS REPORT

PRESSURIZED WATER REACTOR PROGRAM

For the period

July 15, 1954 to August 26, 1954

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For The Atomic Energy Commission

H. F. Canale

Chief, Declassification Branch

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ATOMIC POWER DIVISION

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TECHNICAL PROGRESS REPORT

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PART II

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SUMMARY

A report on "Operation Recommendations for PWR" has been completed. The recommendations in this report give additional basis for future plant design.

Progress has been made in establishing the functional requirements for the vapor container system, the coolant purification system and the emergency cooling system.

The primary system design has been approved with comment. Work is proceeding to resolve these comments. Primary system component functional requirements are expected to be complete by September 6.

Detail design of the emergency cooling, purification, chemical control and valve operating systems is proceeding.

The reactor rod control system design will be completed about September 1 and work is proceeding on several other nuclear control system descriptions.

Coordination with the Duquesne Light Co. is proceeding on problems relating to control station locations and functions, and auxiliary power supply system.

Stone & Webster has completed preliminary design of three basically different vapor container schemes. As soon as agreement is reached on which scheme is to be used, detail design will proceed. Coolant piping layouts have been made for each vapor container design.

Detail design of the prototype 18 in. valve, the prototype main coolant pump and the pump test loop are proceeding.

Steam generator detail drawings are about 60 per cent complete. The design will be completed after sufficient information is developed on loop layout to determine steam generator support requirements.

Preliminary design work is being accomplished on other primary plant components such as miscellaneous pressure vessels, auxiliary heat exchangers and instruments.

Examination has been completed of a corrosion tested brazed joint between pieces of type 347 stainless steel plate. From the corrosion standpoint, gold-copper alloy is the most promising material tested.

A test has been completed on a simulated PWR steam generator tube joint. Results indicate no serious difficulties although no final conclusions can be drawn until further testing is completed.

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SUMMARY (Contd.)

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Preliminary reactor designs have been developed for several seed core designs. Attempts are being made to devise a core design which will increase the percentage of power generated in the blanket over that developed in the seed.

Requirements for fuel handling facilities at the site have been developed and forwarded to the AEC.

Combustion Engineering Co. is proceeding with the detail design as well as the manufacturing schedule of the reactor vessel.

Work on melting and preparation of uranium-silicon alloys has been increased. Work is being done on induction melting, arc melting and powder metallurgy methods.

Fuel element fabrication development of Zircaloy clad uranium-molybdenum rods is continuing with work in progress on the clad bonding and diffusion problem, and on rolling, swaging, and drawing of rods. Experiments are also being made using pre-shaped fuel components and extrusion billet nose pieces.

The butt welding process of making fuel element end closures still appears the most promising method. Work is continuing to improve this process.

A program of fabricating fuel rods containing UO_2 has been initiated.

Experimental work is in progress on the forming of fuel rods using uranium-silicon fuel alloy by both rolling and extrusion. Moderate success has been achieved.

The experimental program to develop the thermal and hydraulic data needed for PWR core design is continuing. Experiments have been run to determine pressure losses through parallel rod subassembly joints. Using the air flow loop the circumferential velocity variations of a rod in a parallel flow bundle have been determined. All major components for the heat cycling loop and the high temperature pressure-drop loop have been placed on order. Subcooled burnout testing on the tubular specimen is continuing in the burnout heat transfer loop.

Equipment installation has started in the PWR fuel fabrication building.

A number of studies are in progress to determine the physical metallurgy of core materials. A series of uranium-niobium alloys were prepared and tested to study the "strained alpha" structure. A series of tests have been run to study the phase transformation on uranium-molybdenum alloys. Time required for the start of phase transformation is given for various conditions. The action of the zirconium clad as a getter of

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hydrogen is being studied for zirconium clad uranium-molybdenum fuel elements. A series of uranium-silicon alloys have been annealed various lengths of times at various temperatures to determine the rate of formation of the ϵ compound. Tests have shown that the corrosion resistance of uranium silicon is not improved by the addition of titanium or nickel in minor amounts.

A number of tests are in progress to determine the mechanical metallurgy of core materials. Tensile tests on several uranium-molybdenum specimens have indicated the lack of ductility at room temperature, but show that ductility increases with temperature. Some hardening due to strain aging effects has been noted on uranium-molybdenum alloys. Tensile tests on uranium-silicon alloys have indicated that this material has little ductility and ductility is not increased with moderate increases in temperature. The thermal cycling loop which is being built at Bettis will be placed in operation August 30. Thermal cycling tests will be made on uranium-silicon alloys, uranium-molybdenum alloys and uranium oxide.

Static corrosion tests on extruded fuel elements have shown that the end cap type of failure does not occur when the end caps are properly welded and when corrosion does not proceed along the core/clad interface. Corrosion tests are continuing on a large number of binary and ternary fuel alloy samples. A number of tests are being made to find ways of eliminating the diffusion of clad and fuel alloys thus producing an alloy of poor corrosion resistance at the interface of these two materials. The use of tantalum and molybdenum as barrier layers has shown that these materials do not prevent diffusion with resulting lower corrosion resistance at the bond line. From recent tests on unclad uranium-molybdenum alloys it appears likely that the time for discontinuous failure is closely related to the corrosion rate in this class of alloys. A series of uranium-molybdenum alloys has been given dual heat treatments in an effort to improve corrosion resistance without favorable results. The study of the effect of phase transformation on corrosion resistance is continuing with emphasis on temperatures below 400°C. A series of tests on diffusion couples using zirconium and various uranium alloys has shown that the diffusion thickness is proportional to the square root of time and that rate of growth of the layer is essentially independent of the composition of the materials in the couple. Work has been started to determine the effect on zirconium corrosion resistance of the amount and distribution of precipitated phase in zirconium binary alloys with iron, nickel or chromium. The role of hydrogen in the corrosion mechanism is also being investigated.

A program is in progress to evaluate the suitability of type 304 stainless steel with small additions of boron as a control rod material.

Preliminary operation was begun on carbon steel loop #A-1 and construction was essentially completed on a second carbon steel loop #A-2. Initial tests of approximately 500 hours duration have been completed on two carbon steel loops being operated by Babcock & Wilcox Company. A series of tests on corrosion inhibitors for carbon steel and low alloy steel have been completed and additional tests are continuing.

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The third fission product removal test in loop R-2 is continuing. Eight stainless steel coupons have been removed from the semi-static autoclave in the loop and decontamination tests using a series of decontamination solutions were completed. The greatest success in decontamination was obtained by the use of periodic acid. Additional facilities are being allocated to fission product removal tests. The radiochemical analysis of the activities of five fission products has been completed for; (1) the total water content of the loop, (2) the crud collected, (3) the ion exchanger resin, and (4) 304 stainless steel coupons.

Thermal stability tests of chemical control poison solutions have confirmed the previously reported satisfactory stability of boric acid solutions. Tests to determine the effect on heat transfer properties of poison solutions have indicated no detectable difference between the heat transfer coefficients for water and for 0.15 M boric acid solutions. Corrosion testing of structural materials in boric acid solutions is continuing. Results from four tests are reported.

A number of fuel element samples will be charged into the Chalk River reactor in September. Four of the eight uranium-molybdenum clad samples which were irradiated in MTR and then defected have survived a 44 day corrosion test in 650° water. The remaining four failed during the first 32 days of test, but in no case could the failure be attributed to the presence of the drilled hole in the cladding. It has been tentatively concluded that the variability in the sample corrosion behavior was sufficient to overshadow any harmful effects that irradiation might have added. Results of tests to show the deleterious effects of heat treatment on the corrosion resistance of irradiated uranium-molybdenum alloys are summarized. Radiation appears to be especially injurious to uranium-molybdenum contaminated by diffused zirconium. No changes in dimensions were observed for the fourteen samples of uranium-molybdenum alloys which were irradiated to above 3000 MWD/T.

The first X-rod loop at Chalk River has been placed in operation. The second loop is essentially complete and is being prepared for out-of-pile testing. It is estimated that erection and hydrostatic tests of the in-pile screening test loop will be completed at Bettis by September 15. This loop is to be installed in MTR. The design of the high pressure loop to be installed in the HT-1 hole at MTR will be completed by September 30. Design of the large loop to be installed in the central thimble at Chalk River is continuing.

More than 20 types of seed and blanket core nuclear studies have been completed using the one dimensional spatial simulator. The conclusions drawn from these studies are presented in detail. An analogue computer study of the time required to burnout xenon has been completed.

The assembly of the flexible critical experiment is continuing. Some difficulty is being experienced in obtaining uranium strips with satisfactory surface conditions. Some difficulty has also been experienced in obtaining molybdenum strips which are machined accurately.

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SUMMARY (Contd.)

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A series of measurements of epsilon and thermal utilization have been completed on the two-region critical.

A study is being made of the effects of core corrosion on activity build-up in the PWR system. Calculations of the heat effects of gamma radiation on the inside of the pressure vessel have been completed.

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I-A-1

PART I

PWR ENGINEERING

A. Power Plant Analysis and Systems

1. Plant Functional Design and Operation Analysis (55all) O. J. Woodruff

PWR Preliminary Design Report

This report has been issued in draft form and is now being reproduced for submission to the AEC. The report summarizes the design features of PWR. Included are presently proposed fluid and control system designs and the design and operation parameters for the PWR power plant.

Operation Recommendations

A study has been completed on "Operation Recommendations for PWR". Considering the inherent capabilities of pressurized water reactors and the requirements of commercial power systems the following recommendations were made:

1. That the plant be capable of changing power level at the rate of 3% per second for a range of 25% of full power anywhere in the power range.
2. That the plant be capable of an average rate of power change of 3% per minute over the full power range.
3. That "design life" be defined as that portion of life during which peak xenon may be overridden.
4. That a minimum value of negative temperature coefficient of 2×10^{-4} per °F be provided.

Pressurizing System Functional Requirements

A preliminary analysis of the major primary coolant pressure surges was made to determine certain design parameters of the pressurizing system. This analysis was forwarded to the AEC for information. The method of calculation is presented in detail for the primary fluid volume changes resulting from reactor safety shutdown, from decrease in steam demand while the reactor is under temperature coefficient control only, and from slow changes in steam demand. The method of analysis can be used to give more exact values when more detailed reactor response information is available. Plans are being made for a simulator analysis of the problem to provide improved data.

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Coolant Purification System Functional Requirements

A study is in progress to determine the requirements of the purification system with core corrosion products in the primary coolant. Test results indicate deposition of radioactive fission products on the pipe walls. Therefore, a by-pass purification system may be ineffective in reducing radiation levels in the vicinity of primary coolant piping. Under these conditions the sizing of the purification system is determined by the coolant purity which will prevent deposition causing significant fouling of the heat transfer surfaces. This would result in a small system.

Preliminary indications, from the standpoint of an economic balance between the cost of regenerative heat exchangers for the purification system and the value of the heat lost by returning the purified stream at lower temperatures, are that the stream may be about 75-100°F below average primary coolant temperature.

Emergency Cooling System Functional Requirements

The requirements for a powerless emergency cooling system are being evaluated. Although the high pressure drop cores which fall within the design limitations of the plant will probably require forced convection with an emergency power sources, a more precise evaluation of a powerless emergency cooling system is being undertaken for the first reference core. Since the transient starting conditions are of utmost importance, the loop coastdown time is being evaluated. Simulator studies of convection cooling are to follow.

Vapor Contained System Functional Requirements

Studies are being made to determine the problems of containing radioactive water-vapor escaping from a major break in the Primary Coolant System. Since zirconium and uranium-water exothermic reaction could occur after such a break, this is also being studied. A third problem in such an accident might be the combustion of the hydrogen evolved from the chemical reactions. Preliminary calculations indicate a minimum of 6.5 minutes is available in which to prevent the zirconium-water reaction. Furthermore at least another 6 minutes is required for this reaction to consume the Zircaloy cladding. Efforts are being made to obtain information on uranium-water reaction rates and temperatures in order to calculate the time available in which to take preventive action.

Information is being collected from the electrical and instrument groups in regard to normal ambient conditions which the vapor container air-conditioning equipment will be required to maintain to prevent damage to instruments and controls.

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I-A-1 (Contd.)

The following functional requirements for the system are being studied:

1. Provide internal pressure protection utilizing the 5 compartment vapor container to permit maintenance of the down loop with others at power.
2. Design pressure-tight container to absorb full expansion of primary coolant but do not design container for zirconium-water or uranium-water reaction.
3. Design container according to ASME UPV code except that sound engineering differences in regard to welding checks, materials and test requirements may be necessary.
4. Utilize partial underground construction with earth embankments in order to save concrete enclosure costs.

2. Primary Coolant System (55b11) V. H. Hayden

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

The schematic piping diagram and the system description for the reactor coolant system have been approved by the AEC, subject to comments. The description and piping diagram are being modified in accordance with the comments; they will be re-submitted for information.

Shop orders are being prepared to cover procurement of all components in the system. It is expected that all functional requirements for reactor coolant system equipment will be issued September 6.

Conferences are being held with the Duquesne Light Company concerning their comments on the system design. These comments include questions on the possibility of water hammer in the primary loop, the mode to be used in pipe design and the method of warming up a primary loop by admitting steam to the boilers.

3. Primary Auxiliary and Secondary Systems (55d11) V. H. Hayden

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

AEC has commented on the PWR powerless emergency cooling system, which was submitted during the last report period. These comments are being considered and an answer will be submitted to the AEC by September 17.

In addition to the AEC comments, the following items are also under consideration:

1. Installation of the emergency cooling condenser inside the steam drums (in the form of a pipe coil) to save equipment costs, as suggested by Duquesne Light Company.

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I-A-3 (Contd.)

2. Allowing the cooling water for the emergency cooling condenser to boil in the condenser tubes, as suggested by Duquesne Light Company, to reduce cooling water requirements.
3. Duquesne Light Company has been requested to examine the history of power plant failures to determine the probability of a complete power failure in the FWR plant and the probable duration of such a failure. The results of this study will be the basis for the definition of the emergency period.
4. Stone & Webster has been requested to evaluate the increase in cost versus boiler elevation above the reactor vessel. This study will determine the optimum boiler elevation.
5. A steam driven emergency cooling system is being studied. This scheme utilizes the reactor decay power to generate steam which drives a turbine generator, the output of which drives the TG set condenser circulating water pump and a condensate-boiler feed pump. Primary system flow is by natural circulation.
6. The various emergency cooling schemes are being evaluated with respect to the extremes of reactor designs for FWR (high flow, low pressure drop and low flow, high pressure drop).
7. Consideration is being given to maintaining the plant at full temperature during the emergency instead of gradually cooling it down. This approach eliminates the shrinkage problem and eliminates the expensive air-driven emergency injection flasks.

Results of the above studies will be forwarded to the AEC as they become available.

Information concerning plant constants which affect emergency cooling is being collected so that simulator study of emergency cooling transients can be made. Work on the simulator study will commence in the next report period.

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

The economic study to determine the temperature to which the purification system effluent should be heated (outlined in WAPD-MRF-46) has shown that this effluent should be reheated to 460°F if T average is 525°F. This figure is theoretical; difficulties in regenerative heat exchanger design may lower this value to approximately 425°F.

Chemical Control System (S.O. 1-AP-8013)

Two documents are being prepared for submission to the AEC for information and comment. These documents are:

REF: 1-AP-8007

I-A-3 (Contd.)

1. A system description and schematic diagram for a bleed and feed type chemical control system.
2. An economic comparison of several types of chemical control systems, including bleed and feed, high pressure evaporation, low pressure evaporation and removal of chemical by demineralization.

These documents will be submitted in the next report period.

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

Preliminary studies of valve operating schemes are being made. In the most promising scheme, the valves are operated in one direction by the differential between the valve operating system pressure (3000 psi approximately) and primary system pressure and in the other direction by the differential between primary system pressure and atmospheric pressure. By utilizing this method of operation, it may be possible to eliminate all springs and latches inside the cylinders of the main stop valves. In addition, this scheme uses the least number of auxiliary valves.

A preliminary diagram of the valve operating system will be sent to the AEC for information during the next report period.

4. Reactor Control System Design (56a13) W. H. Hamilton

Reactor Plant Over-all Control System

During the period of this report, a study of the operational requirements of the FWR plant has been made and a report issued to AEC. With the establishment of specific power change requirements for the plant, it will be possible to proceed with over-all and individual system descriptions. The Reactor Plant Over-all Control System Description, first issued March 10, 1954, will be revised during the period of the next report, and reissued on or about October 1.

Reactor Rod Control System

The Reactor Rod Control System Description and its schematic diagram are in preparation and will be issued for approval on September 1. It is based on the use of six safety shutdown rods having a minimum worth of 5% reactivity. During start-up, the rods will be withdrawn two at a time, resulting in an average negative reactivity change rate of approximately 1×10^{-4} per second. Rods fully withdrawn will be on a hold bus available for the application of a run-in frequency sufficient to produce negative reactivity insertion of 1% per second for safety shutdown. A rate of insertion faster than 1% per second will produce no significant change in the reactor power decay curve on loss of flow. It is not intended that rods will be used for attaining criticality.

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Reactor Power and Temperature Control

Additional calculations on the primary plant have been made to determine the best means of compensating a simple temperature-error control system to provide stability. A study of the system on the plant simulator will begin in the month of September. Until this study is made the parameters of the control system will not be firm.

Nuclear Instrumentation System

A study for nuclear instrumentation has been made. The study includes three approaches to the problem. They are:

1. Modification of the Mark II equipment where necessary to make it suitable for FWR.
2. A new design incorporating the latest advancement in circuit techniques and manufacturing processes.
3. An approach similar to (2) but incorporating the more stringent performance requirements and tests of military specifications for shore based equipment.

Approach (1) would provide the lowest immediate investment cost, but would not provide appreciable reduction in cost for subsequent orders. Approach (2) will provide for improved reliability through advancement in design, and have the possibility of lowest reorder cost. Approach (3) will be the most expensive of the three.

Evaluation of the study and a recommendation on the approach to follow will be made during the period of the next report.

Radiation Monitoring System

The functional requirements of this system will be established by health physics requirements, and the requirements for primary system leak detection. They will be established by December 1. The activity to date has consisted of information conferences with equipment suppliers to acquaint them with the general plant arrangement, and problems likely to be encountered.

Emergency Shutdown System

Since the rod speeds and circuit arrangements of the rod control system, the reliability of nuclear instrumentation, and provisions for interlocking of primary system controls involving coolant flow are dictated by the emergency shutdown system. The first system description and diagram of the Emergency Shutdown System is expected to be issued for approval on October 1 and a report on progress on the system will be made in WAPD-MRP-48.

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5. Non-Nuclear Control Systems Design (55e12) W. H. Hamilton

Primary Plant Control System and Primary Plant Instrumentation System

A meeting was held with the Duquesne Light Company on August 5 to discuss mutual problems relative to coordinating the control stations for the primary and secondary plants. It is planned to have the operating panel for primary plant control adjacent to the steam plant control panel. The steam plant control panel will include boiler level control, circulating water control, feedwater heaters, etc., but not turbine control. Turbine control for start-up may be from a panel on the turbine floor separate from the primary plant panel. Automatic load limit control on the turbine to prevent excessive reactor load if less than four pumps are operating is feasible and may be required.

The operating panel for primary plant control will contain only those instruments and controls needed for operating the plant to change load, to control main loop valves and pumps, and to monitor essential plant operating conditions such as flow, temperatures, pressure drop across the reactor, and nuclear level. Part of the panel, or an adjacent panel, will contain the instrumentation and control equipment necessary for bringing the reactor critical and into power operation. A preliminary design of the main primary plant control panel, including the reactor start-up and control function, will be completed on September 1.

At a meeting with the Duquesne Light Company it was decided that the 2300 volt design for the main coolant pumps would be retained. Therefore a distribution system one-line diagram can now be selected to meet the requirements of the reactor plant.

6. Control System Tests (56a14) H. E. Thomas

Preliminary calculations have been completed for the FWR primary plant simulation. The initial setup has been made on the EASE computer, and steady state and transient characteristics have been checked for full power and full flow conditions. System studies will be initiated about September 1.

7. Plant Layout and Shielding (55a14) D. C. Spencer

Stone & Webster Engineer Corporation has completed study drawings and an evaluation including cost estimates of the concrete structures (excavation, backfill, forms, and reinforced concrete), the vapor containers, air conditioning, and the high pressure coolant piping for the following vapor container arrangements.

1. A single vapor container housing the reactor and four boilers with shielding between these units but without pressure segregation located both above and partially below flood level.

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I-A-7 (Contd.)

2. Five cylinders for housing the reactor and the four boilers with pressure segregation and shielding located both above and partially below flood level.
3. A vertical cylinder for housing the reactor and four spheres for housing the boilers with interconnections between the cylinder and the spheres arranged for pressure segregation of each boiler, together with adequate shielding.

Not included in the above evaluations are fuel handling, auxiliary structures, and internal missile protection. Fuel handling facilities and auxiliary structures as planned to date were not included as they do not appear to alter the relative costs of the above vapor container arrangements. In order to evaluate internal missile protection, steps have been taken to establish contacts with various government departments. To date information pertaining to missile protection is very limited and for this reason no attempt was made to include the costs on the above evaluations.

For each of the above arrangements a detail drawing of the cooling piping was made in order to evaluate the effect of system volume on the size of the vapor containers. Each of the piping layouts is based on the principle of hanging the boilers so as to greatly reduce the length of piping required. In this layout the piping and the boiler equipment will be supported on long pin connected members with constant support spring hangers. This piping arrangement will be the most economical. In addition a preliminary piping layout with the boilers anchored is in progress. This alternate layout with the boiler fixed in position is being prepared in the event unforeseen objections are raised by equipment vendors.

Consideration has been given to other forms of vapor containers including:

1. Single sphere
2. Single spheroid
3. Interconnected spheres or cylinders
4. Installing a portion of various containers above ground for emergency use only.

The conclusion reached is that the single cylinder or the five cylinder arrangements are the most promising. During September a vapor container arrangement will be recommended for approval and detail design will proceed, after approval is obtained.

Air conditioning studies are in progress for the various arrangements and include the following:

1. Closed cycle air conditioning
2. Forced air circulation
3. River water cooling of circulated air

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I-A-7 (Contd.)

Internal missile protection is being investigated to determine the best materials and the practical extent of design. Contact is being established through the AEC with various Department of Defense agencies associated with missile protection.

Problems associated with the penetrations of the vapor container, for pipes and cables and auxiliary equipment arrangement are being investigated.

Shielding studies are being continued, based on current arrangements. The studies consist of concrete tests being conducted at MTR and the investigation of special concrete aggregates such as barytes. Detail shielding will not be specified until a final vapor container arrangement is selected.

8. Reactor Plant Site Facility (55a12) D. C. Spencer

Over-all site layouts are being prepared by Duquesne Light company. Final site details cannot be developed until the vapor container arrangement is selected. Present indications are that structural costs can be decreased if the bottom mat of the reactor plant is located above flood level.

A fuel handling system incorporating an open trough type transfer is being studied. This arrangement is based on handling the spent core as a unit. Study drawings are being prepared. This system appears equally adaptable to most vapor container structures being considered. A fuel handling system of the above type appears to be the most flexible when considering present and future core designs. Auxiliary facilities associated with fuel handling such as hot cells, clean room, laboratories, etc. are being studied.

Work is continuing on the facilities for storage of control chemicals, primary water, poisoned water, and decontamination chemicals.

Development of the waste disposal system will continue when more information is available.

9. Instruction Manual and Test Program (59c16) D. C. Spencer

Limited effort may be expended from time to time to supervise preparations of models and visual aids as required in Fiscal Year 1955.

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B. Power Plant Components and Component Materials and Tests

1. Primary Coolant Loop (55b26) P. A. Halpine

Although several materials and methods of manufacture were considered, the present preferred material for the primary coolant loop is type 304 stainless steel seamless pierced and drawn tubing. This material best meets the combined functional, structural, code, and cost requirements.

Carbon steel, although most favorable from an economic standpoint, was rejected because the problems arising from the use of this material will not be resolved in time to meet the schedule for ordering material. Nickel or chrome plated carbon steel was rejected for the same reason.

Piping made from rolled and welded clad plate was rejected due to added fabrication problems and costs such as welding and stress relief which nullified the slight cost advantage over rolled and welded stainless plate. This type of pipe also, was not economically compatible with centrifugally cast stainless steel pipe.

Rolled and welded pipe from stainless steel plate was rejected because, although on the surface it was slightly more favorable from an economic standpoint than pierced and drawn tubing, the added costs due to quality control of plates and longitudinal welds would nullify the economic advantage. This type also, was not economically compatible with centrifugally cast stainless steel.

Centrifugally cast stainless steel pipe is still under investigation. It has an economic advantage over pierced and drawn tubing but several problems such as quality control, over-all structural suitability, and welding have not yet been resolved. See project 55b16 for progress of this type of material.

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

The pump and volute layout drawing was approved and a request for manufacturing release was made. A recommendation that the pump be constructed with a single winding with suitable connections for two speed operation was made.

Approximately 65% of the design engineering and 35% of the drafting work is completed on the prototype pump.

3. Pump Component Development (55b22) P. A. Halpine

Bearing Development

Tests have been run at the Franklin Institute on Graphitar-14 against Stellite Star-J to determine the load carrying ability of this combination. The test was discontinued when the elements seized at 75 psi loading.

Some dry friction tests were also run. In these tests, the runner was operated for a period of 5 minutes each at speeds of 144 rpm, 900 rpm and 1800 rpm.

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Tests of longer duration than this could not be run because of temperature increase. There was no structural damage to the bearing elements in this test.

Malcomized thrust runners are now available at the Franklin Institute and will be used largely in future tests.

Slot Wedge Tests

Three test fixtures were built, each capable of taking varying pressures up to 20,000 psi and temperatures up to 500°F. These fixtures were designed with inserts so that various configurations of wedges as well as various materials might be tested.

To date, three different shape wedges have been tested. Two of the shapes were trapezoidal; one being approximately 3/8 in. thick and the other approximately 1/2 in. thick. The third shape was a modified keystone approximately 1/2 in. thick.

From both a mechanical stress and an assembly standpoint, the modified keystone shape appears to have considerable advantage.

The following wedge materials have been tested at various conditions of temperature and pressure:

1. Mycalex (Glass bonded Mica #410) Mycalex Corporation of America.
2. Micarta (Glass Cloth Silicone Bond #20202-3) Westinghouse Electric Corp.
3. Alsimag (Alumina Ceramic #491) American Lava Company.
4. Alsimag (Alumina Ceramic #576) American Lava Company.
5. Laminated Metal (Bonded with a variety of high temperature bonds).

All of the above materials are scheduled for additional tests, except the Micarta which exhibited excessive creepage during test.

In addition, samples of the following materials have been ordered for testing.

6. Alundum (Alumina Ceramic) - Norton Company.
7. Norbide (Boron Carbide Ceramic) - Norton Company
8. Steatite (Clinoenslatite Crystal) - Stupakoff Ceramic & Mfg. Co.
9. Ceramic Coated Metal Wedge - Solar Aircraft Co.
10. Solid Metal Wedges (Inconel)
11. Glass (Boron Silicate) - Thatcher Glass Mfg. Company.

Tests are also being conducted on slot liner materials. These liners are to be used as a cushion between the rough lamination and the wedges for notch sensitive material such as the ceramics. The liner would also be used as an insulation for the metal wedges.

End Turn Cooling Studies

A heat transfer test fixture has been designed and is in production to determine actual heat transfer coefficients for use in motor end-turn cooling studies. Its design is such that small mock-up coils with various insulation

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schemes may be tested. The heat is generated in this mock-up coil by circulating full load current at low voltage.

Tests are scheduled to establish heat transfer constants of various coil insulation schemes, surface preparation of the heat sink and of various back-up materials that might be used as media between the heat source and the heat sink.

Casting Study

A comprehensive survey has been completed to determine the feasibility of producing a heavy wall stainless steel casting for the prototype pump casing. All possible suppliers have been contacted regarding the design of this part to obtain a reliable pressure-tight structure.

Representative test domes of type 309 stainless steel have been obtained from each of the four most promising casting companies. The test domes have been successfully tested at 10,000 psi hydrostatic and 1000 psi helium test pressures with zero leakage resulting.

The domes were completely radiographed and in general, the negatives disclose good homogeneous material with very slight evidence of sponge or shrinkage present in certain areas. These defects will be further diagnosed as additional metallurgical studies are begun. It is planned to dissect three of the four domes for complete metallurgical investigations. These investigations will include macro and micro etches of the as-cast material as well as after heat treatment. Also, the as-cast material will be subjected to thermal shock equivalent to the requirements of the PWR system. This program is presently in the machining stage and metallurgy studies have begun.

4. Primary Pump Test Loop (55b23) P. A. Halpine

An order for the design of the pump test loop is being negotiated with the Catalytic Construction Co. It is estimated that approximately six weeks will be required to complete the contract.

5. Auxiliary Pumps (55b21) P. A. Halpine

Bettis Site is studying the system to determine requirements for auxiliary pumps. Functional requirements are scheduled to be available in November.

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

The sub-contract with Crane Company for the 18-in. prototype valve was approved by the AEC.

Technical approval of the design was received from the AEC on an outline sectional drawing submitted on June 10. Crane Company is presently working on the manufacturing drawings and will also submit an outline drawing with a bill of material. This drawing will be forwarded to the AEC for approval during the week of August 30.

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Development of the other PWR valves is being held pending the establishment of firm functional requirements.

7. Steam Generators (55b32) K. W. Schwanekamp

Detail drawings for the PWR steam generators being supplied by the Babcock & Wilcox Company and the Foster Wheeler Corp. will not be completed until three weeks after final PWR primary system layout drawings are available. Preliminary layout drawings, from which piping reactions, mounting arrangement and other details required by the steam generator vendors may be determined, are expected to be available by October 15.

The dissimilar metals corrosion test, designed to investigate the problem of using a carbon steel stainless steel combination in the Babcock & Wilcox PWR steam generators has now operated 1000 hours. This unit will be removed from test for final examination as to the effects of galvanic action on September 15. A decision will be reached on the desirability of the Babcock & Wilcox Company PWR design within 30 days after this date.

Plans are underway to run a chloride corrosion test on model heat exchangers operating under PWR conditions. Two units will be tested; one will simulate the Foster Wheeler Corp. design, and the other will simulate the design of the Babcock & Wilcox Company. The model unit simulating the Babcock & Wilcox Company design will be installed in the test loop now being used for the dissimilar metals heat exchanger test. Both tests will be in operation before January 1, 1955.

8. Reactor Plant Pressure Vessels and Degasifier (55b33) K. W. Schwanekamp

Preliminary estimates of size and heater configuration for the PWR pressurizing tank have been made on the basis of using replaceable heater units. Preliminary equipment specifications for this unit and its associated equipment are also being prepared. Final specifications will be prepared when functional requirements are established.

9. Reactor Plant Auxiliary Heat Exchangers (55b34) K. W. Schwanekamp

Preliminary designs for several PWR auxiliary heat exchangers have been completed as an aid to final primary systems evaluation. No final designs or final equipment specifications have been completed to date because final functional requirements have not been established.

10. Filters and Vent Capillaries (55b35) P. A. Halpine

Bettis Site is studying the system to determine the requirements of filters and/or strainers.

11. Emergency Cooling Components (55b45) K. W. Schwanekamp

No work has been done on this project since functional requirements have not yet been firmly established.

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12. Reactor Electric Power Components (55dl2) G. E. Seguin

Component engineering efforts have been limited to long range development items, particularly the power supply for the control rod drive motors. The higher frequency required for driven scram with non-latching type mechanisms presents a problem which requires extensive study and comparison of various types of power supplies. Efforts are confined to general studies in the absence of specific mechanism power requirements.

13. PWR Instrumentation (56dl3) W. T. E. Elmendorf

Liaison work with the Bettis system and control personnel has continued. Informal discussions with instrument vendors have been held to find the availability of instruments meeting primary system requirements on the commercial market.

A proposed differential pressure transmitter specification has been prepared and will be submitted by Sept. 15. A general receiver specification is being prepared on a strictly commercial basis and will be submitted during October.

General development requirements will be determined by the final requirements of the receiver specification and any special control functional requirements which cannot be met with commercial equipment meeting other specifications.

14. Materials for Primary Loop Components (55bl6) J. R. Hunter

Feasibility of Replacing Forgings with Castings

A material specification for type 304 austenitic stainless steel castings was submitted for approval on August 23. This specification will be used for procurement of prototype valves.

A program outlining the work to be done under subcontract for evaluating the acceptability of castings for primary loop components was submitted for approval on August 23.

A 9-ft. length of 18 in. schedule 140 centrifugally cast pipe was received on July 21. Spot radiographic examination was made on each end of the pipe as received. While the exographs portrayed a general condition which, if interpreted as defects, would constitute a basis for rejection on the basis of comparison with acceptance standards, the uniformity of the conditions revealed indicates that radially oriented columnar grains may be responsible. Radiographic indications will be correlated with the results of metallographic examination and mechanical tests which are in progress.

Alternate Brazing Materials for Stainless Steels

Future work on alternate brazing materials will be carried on and reported under project 53all, Non-Core Materials. Results, to date, of corrosion testing performed under project 55bl6 are summarized in this report.

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Transverse sections removed for metallographic examination of the previously exposed edge of hydrogen furnace brazed capillary joints between type 347 stainless steel plate on place specimens static autoclave tested in 600°F oxygenated (1 to 3 cc O₂/liter as charged) and hydrogenated water (100 cc H₂/liter as charged) revealed the following:

- a. Joints made with preplaced copper and silver-manganese (85-15) alloy were slightly attacked by oxygenated water and unattacked by hydrogenated water in 70 day tests.
- b. Joints made with preplaced silver-copper-cadmium-zinc alloy were attacked slightly by oxygenated water and very slightly by hydrogenated water in 70 day tests.
- c. Gold plated joints made with the foregoing alloys were unattacked by either oxygenated or hydrogenated water in 63 day tests.
- d. Joints fed by capillary attraction using gold-copper (80-20) alloy were unattacked in 70 days by both oxygenated and hydrogenated water.
- e. A joint similar to (d) made with gold-manganese alloy (90-10) was unattacked by hydrogenated water in 70 day test. Evaluation of a like joint tested in oxygenated water was not possible because sectioning revealed that the brazing alloy had not penetrated the joint uniformly.

Before gold-manganese alloy can be evaluated, it will be necessary to determine that it will produce mechanically sound joints. If sound joints can be obtained, further tests will be required to determine their resistance to corrosion in oxygenated water. It is considered that from a corrosion standpoint, gold-copper alloy is the most promising of the materials tested.

Steam Generators

A 28-day static autoclave test of an assembly of three tube-in-tube mock-ups made up with type 347 s.s. tubes in carbon steel tube sheets clad on one side with type 347 s.s. weld metal was completed. The mock-ups contained tubes rolled 0.003 in. past contact, with and without grooves measuring 0.002 in. deep by 0.008 in. wide, and a slip-fit tube (0.002 in. to 0.003 in. diametral clearance) containing a groove of the aforementioned dimensions. The test was conducted in boiler water maintained at 450°F, with a temperature of 550°F maintained on the inside of the tubes through the use of resistance heaters. The boiler water, the pH of which was adjusted to 10.5 by the addition of sodium hydroxide, contained 50 ppm phosphate and 0.1 ppm chloride. At the conclusion of the test, the mock-ups were sectioned and both macroscopically and microscopically examined with the following results:

- a. Oxidation of the expanded tube OD and the tube sheet hole was slightly greater near the unclad surface of the tube sheet.
- b. Oxidation of the slip-fit tube OD and the tube sheet hole was uniform over the length corresponding to the thickness of the carbon steel.
- c. Slightly greater oxidation occurred on the OD of the tube and the ID of the carbon steel at the juncture of carbon steel and stainless steel cladding.

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- d. Metallographic examination of both longitudinal and transverse sections from all tubes did not reveal any evidence of pitting, galvanic corrosion or intergranular corrosion of the stainless steel tubing.

Test joints made up with slip fits and grooved tubes both welded and unwelded to double clad tube sheets will be removed on August 28, after completion of 12 weeks testing.

The foregoing tests were conducted using type 347 s.s. tubes because type 304, as proposed by Babcock & Wilcox, was not immediately available. Additional test joint assemblies have been prepared using type 304 s.s. tubing which will be placed in test at higher temperatures corresponding more closely with those expected in PWR i.e., $600^{\circ}\text{F} + 10^{\circ}$ in the inside of the tube and $486^{\circ}\text{F} + 10^{\circ}$ on the outside. Tests will be conducted in boiler waters containing both 0.1 ppm and 15 to 20 ppm chloride.

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C. Reactor and Auxiliaries

1. PWR Core Development (53a14) T. Fahrner

During this period preliminary neutron flux plots and power density plots of various seeded cores were analyzed in an effort to determine the more profitable avenues of investigation.

Power produced in the seed costs several times as much as power produced in the natural uranium blanket; hence, it is desirable to produce as much of the total power in the blanket as the nuclear and heat transfer properties of the materials will allow. Both natural uranium-molybdenum and natural UO_2 have been considered as blanket fuels. Blankets using UO_2 were found to have the more favorable power distribution of the two, being capable of producing up to about 70% of the total power as compared to about 60% for the uranium-molybdenum blanket. It was also found that this distribution is fairly sensitive to blanket thickness, becoming more favorable as the blanket thickness is increased.

For the reactors studied thus far, the size of the enriched seed, if located at the center of the core, is so small that it is excessively difficult to provide adequate cooling. For example, in a reactor containing about 9 tons of uranium in the form of uranium-molybdenum, the volume of the seed was 4% of the total volume and the seed power was 40% of the total power. This problem is now being approached on the basis of a modification in seed design. As presently envisaged the seed is in the form of an annulus surrounding a cylindrical natural uranium-water center, and being in turn surrounded by a natural uranium-water annulus. This arrangement simplifies the cooling problem in two ways. By moving the enriched material away from the center of the core, more of it must be provided for criticality; thus, for the same proportion of total power produced in the seed material, the average seed power density will be reduced. Furthermore, the thickness of the annular seed region will be considerably less than the radius of the cylindrical central seeds originally studied. It is expected that the ratio of peak-to-average power in the seed material will be close to unity. It is hoped that the peak power density will be reduced to the point where normal cooling methods can be used. Calculations based on estimated flux plots have indicated that this is the case, but these conclusions must be verified by flux plots obtained from the analog computer.

Other conclusions indicated by analyses made during this period concern desirable relationships between the seed and the blanket. From the standpoint of reactor transients, it is desirable to have the same hot spot temperature and the same hot channel exit temperature in the seed and in the blanket. This means, in the annular seed design for example, that the flow velocities in the seed will be two to four times higher than in the blanket, whereas the total volume flow will be perhaps one third of that in the blanket. It appears that fuel elements with inherently high friction factors, such as cross flow rods,

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may be suitable for the blanket. Rods in cross flow are presently being investigated in addition to the parallel flow arrangement. For the natural UO_2 - H_2O blankets, rods 0.600 in. in diameter with 36 mil clad thickness and a water-to-uranium volume ratio of 2 1/2 constitute the most promising cross flow design so far investigated. Rods in parallel flow 0.400 in. in diameter with 24 mil clad thickness and a water-to-uranium volume ratio of 2 presently appear to be the most promising parallel flow design.

Since the power density in the seed region will be several times that in the blanket, the fuel structure of the seed must be more finely divided than is that of the blanket. Shapes such as ribbons, wires, or plates are indicated for the seed elements.

Thermal analyses of temperature distribution and gas pressure build-up in rods containing UO_2 are being made, but are limited by lack of information on the conductivity of UO_2 at the higher temperatures.

2. Mechanical Design of PWR Core (53a12) T. Fahrner

Design effort has been directed toward development of the details of a parallel flow rod fuel subassembly and the support structure for a core based upon this subassembly. Several preliminary subassembly designs based upon welded bundles of 10-in. length rods, supported at their ends in a square array, have been completed. Eight bundles, each containing about 80 rods, are stacked end-to-end within a square Zircaloy tube to form the subassembly. Two hundred of these subassemblies are required for a 12-ton core.

A horizontal cross section of a seeded core based upon the annular seed arrangement has been completed.

3. Thermal and Hydraulic Analysis and Tests of PWR Core (55a13) T. Fahrner

Curves showing required emergency coolant flow as a function of power have been prepared for the reference 12-ton uniformly enriched core at a system pressure of 2000 psi. Two situations were considered. In the first case it was assumed that the maximum fuel surface temperature must not exceed the design value of 630°F. In the second case, bulk boiling was permitted at the outlet of the hot channel and the fuel surface temperature was permitted to rise to slightly above T_{sat} (636°F). The first case is considered to be a conservative basis for design of the emergency cooling system. The second case is representative of conditions under which choking of flow in the hot channel and damage to the fuel elements might occur. For a power of 20 Mw (corresponds to the decay power about 10 sec after shutdown from full power) and an average coolant temperature of 525°F, a flow of 0.96×10^6 lb/hr is required to maintain 630°F maximum fuel surface temperature. Boiling at the hot channel outlet will occur at a flow of 0.82×10^6 lb/hr.

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The study of seed power, coolant flow, seed size relationships was completed for plate and rod type fuel elements. The results can be summarized as follows: for cylindrical seeds having diameter equal to height, the relationship between required seed diameter and ratio of seed power to total power based upon cooling requirements is

<u>Seed Power</u> Total Power	<u>Required Seed</u> Diameter
1/4	3 ft
1/2	3 1/2 ft

For rod fuel elements the seeds must be even larger.

Two group analog flux plots have been received for systems having seeds of about 1.3 ft and 2.8 ft equivalent cylindrical diameter. The best power ratio, seed/total $\approx$ 0.3, was obtained for the 1.3 ft seed, which has about 1/10 of the volume required for heat transfer. The "dispersed" or annular seed arrangement is being explored as a possible solution to the cooling problem (See Part I-C-1).

4. Mechanical Tests of PWR Core Components (53a13) T. Fahrner

Fuel Element Vibration Tests

Work during this period has been directed toward improving the experimental setup for measurement of amplitude and frequency of vibration of rod type fuel elements in parallel flow. An optical method consisting of a scribed and painted line on the rod surface, a microscope, and high speed motion picture camera, has been developed to the point where it is believed worthwhile to attempt a series of measurements on rods of various lengths. Attempts to measure frequency by stroboscopic means have been unsuccessful.

5. Core Refueling Equipment (53d12) T. Fahrner

Preliminary requirements for core refueling have been given to the Plant Arrangements Section. It is proposed to provide for handling of entire cores or single subassemblies in a system of water-filled service areas.

6. Radioactive Materials Handling & Viewing (53d11) T. Fahrner

Preliminary estimates of space required for assembly and disassembly of cores, storage of irradiated fuel, and for the hot cell, have been given to the Plant Arrangements Section. A preliminary outline of operations to be performed and equipment required has been prepared in response to a request from the AEC.

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7. Control Mechanism Development (56c19) E. J. Kreh

Design work is progressing on the test rig to be constructed for determination of rundown characteristics of fixed roller nuts with large lead angles.

Misalignment in the operating rig used for wear tests on roller nuts at Large, Pa. resulted in more than normal expected wear at 2000 cycles of operation. Although misalignment was apparently excessive, operation of the nut required less than 3 ft-lb throughout the test. This test is being set up to be repeated with alignment of the test fixture corrected.

Simulation test components are being designed to prove feasibility and establish design constants on a magnetic jack type, and a direct lift magnet type mechanism. This investigation is to determine feasibility of more simple, more reliable mechanism types.

Tests on a collapsing rotor type mechanism employing the expanding roller nut principle are being accomplished on another project and will serve as back-up information should need for a thrown scram arise.

8. Reactor Vessel Assembly (55b25) T. Fahrner

The Bethlehem Steel Company closure study report has been forwarded to Combustion Engineering Company for use in their design work.

A schedule has been set up with Combustion to select, by October 1, 1954, a reference closure joint for development.

Combustion Engineering Company is now working on a design and fabrication schedule for the vessel, based upon delivery by August 1, 1956. The schedule will be prepared by September 15.

Calculations of allowable thermal stresses in the vessel wall, to provide a basis for thermal shield design, are in progress.

Calculations of heating of the vessel wall as a function of core size, power, and thermal shield thickness-composition, are in progress.

It now appears that a reasonable basis for thermal shield design is to provide shielding of such nature that the thermal stress due to gamma heating does not exceed the pressure stress. On this basis, three inches of iron are required inside the pressure wall.

Four reactor vessel layouts have been completed during this period and are being analyzed. A simple cylindrical vessel having the closure opening equal to the inside diameter of the main body (109") and with outlet and inlet nozzles located at top and bottom respectively is preferred at this time. Combustion Engineering has been given a set of reference dimensions for a vessel of this type, for use in preparing the design and fabrication schedule mentioned above.

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PART II

PWR DEVELOPMENT

A. Fuel Element Development

1. Core Raw Materials and Shapes (53c21) B. R. Price

Zircaloy-2 Cladding Cups

A report entitled "Evaluation of Impact Extruded Cups at Alcoa" has been issued. It was found that the surfaces of the extrusion were so good the dimensions may be reduced to more closely approach the size of the finish machined cups. The temperatures of working were not critical—good results were obtained from 1000° to 1400°F. It was concluded that impact extrusion of copper clad Zircaloy-2 billets is a practical method of producing cups.

The Zircaloy-2 cups which were impact-extruded at Wyman-Gordon Company are being examined at Bettis Site for dimensional variations and surface condition.

Zircaloy-2 Tubes

The tubes extruded from copper-clad billets at Allegheny Ludlum (WAPD-MRP-46) have not yet been received at Bettis Site for evaluation.

The tubes extruded by Babcock & Wilcox Company using glass as a lubricant (WAPD-MRP-46) have had sections removed for evaluation at Bettis Site; the balance of the tubes have been returned to Babcock & Wilcox for conditioning of the surfaces and for working down to 1.5 in. OD x 0.150 in. wall by the tube reducing method. After being reduced to the aforementioned size, the tubes will be forwarded to Superior Tube Company for further processing into 0.425 in. OD x 0.042 in. wall and 0.310 in. OD x 0.031 in. wall tubes.

The tube shells scheduled for extrusion at Chase Brass and Copper Company (WAPD-MRP-46) have been further delayed due to the overhauling of their extrusion press. The work has been rescheduled for the latter part of August.

Permission has been granted to negotiate a subcontract for the development of a commercial process for producing small diameter (3/4 in. OD to 1/4 in. OD) Zircaloy-2 tubing. Various suppliers are preparing proposals and inspection of their facilities is being undertaken by Bettis personnel.

Zircaloy-2 Rod

Universal-Cyclops Steel Corp. has shipped the 1000 lb of Zircaloy-2 rod (WAPD-MRP-46). The rod will be evaluated as soon as it arrives at Bettis Site.

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Melting of Uranium-Molybdenum Alloys

Experimental melting studies and melting of material for fuel rod development work were temporarily discontinued in order to concentrate effort on the 3.8 w/o silicon alloy. The only melting undertaken on uranium-molybdenum alloys was to make fuel for the clad fuel rods being prepared for corrosion and radiation studies in the Chalk River loop. Both a 10.5 w/o Mo and a 12.0 w/o Mo alloy were prepared by induction melting in magnesia crucibles. For each alloy duplicate melts were made, one containing 3.6 w/o U²³⁵ and the other only natural uranium.

In all 29.8 lb of the enriched alloys and 65.0 lb of the alloys containing only natural uranium were melted. Duplex melting was not used in preparing any of these alloys.

Melting Uranium Alloys (Miscellaneous)

Eight uranium alloy castings weighing 142.5 lb were made by duplex melting (induction followed by consumable arc) for the alloy development program. Five 200-gram arc melted castings were also made for the same purpose. The 39 remelted buttons discussed in WAPD-MRP-46 have been examined, found to be homogeneous, and are being used for phase diagram studies.

Uranium-Silicon Alloy (3.8 w/o Silicon)

a. Induction Melting

A total of 362.5 lb of alloy were melted using BeO crucibles and ThO₂ stopper rods. This refractory combination has been found to work better than the others which have been tried. Thirty-four pounds of the alloys produced by induction melting have been remelted by the consumable arc process to improve soundness and decrease segregation. All of this material will be heat treated to the epsilon phase prior to machining into extrusion billets.

It has been found that the high purity silicon powder, which is the source of silicon, can be melted quite satisfactorily in a vacuum induction furnace using a SiO₂ crucible. After freezing in the crucible, the ingot of pure silicon can easily be fragmented into small pieces for use as alloy feed stock. Attempts to cold compact the powder into small pellets were completely unsuccessful.

Argonne National Laboratory has been requested to furnish sample castings made on the centrifugal casting machine described in ANL-5123. The castings will be compared with castings made at Bettis Site by the duplex process. Soundness, response to epsilonizing treatments, chemical homogeneity, and corrosion behavior will be the basis of comparison.

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b. Arc Melting

An experiment has been started to study the feasibility of producing sound homogeneous 3.8 w/o silicon alloys by arc melting. The experiment will include arc casting by nonconsumable and consumable methods. The silicon alloy additions will be made as elemental silicon or as uranium + 50 w/o silicon master alloy. To date, 22 lb of the 3.8 w/o Si alloy have been arc cast either from chopped U and Si feed stock or U + 50/50 master alloy feed stock. The initial ingots have been good so far as melting is concerned; however, chemical analyses will have to be made to determine the homogeneity of the alloy.

Homogenization of Uranium Silicon Alloy to the Epsilon Phase

A vacuum annealing furnace is being built for annealing uranium-silicon ingots in sizes up to 3.5 in. diam. The ingots will be heat treated prior to machining into extrusion slugs.

An experiment has been started to study the time-temperature requirements for homogenization of uranium silicon alloys to the epsilon phase. This work will apply both to cast materials and alloys made by powder metallurgy. It is expected that the annealing cycles required for complete homogenization will be somewhat longer for alloys made by powder metallurgy than for those made by melting.

Preparation of Uranium-Silicon Alloys by Powder Metallurgy

As yet no material has been made by Brush Beryllium Company on the purchase order discussed in WAPD-MRP-46, but the equipment modifications needed to make the extrusion slugs are nearly complete. It is expected that some samples will be produced early in September. Hot pressing, rather than cold pressing and sintering, will be used in this work.

Negotiations are being conducted with Sylvania Electric Products, Inc. for a development subcontract to establish fundamental information pertaining to the powder metallurgy of uranium-silicon alloys.

Uranium-Silicon Literature Survey

A preliminary literature survey on uranium-silicon alloys has been made and a report will be issued by the end of August. This report contains information on constitution and structure, density, hardness, thermal conductivity, tension and compression testing, powder metallurgy, and thermal cycling.

2. Core Manufacturing Process Development (53c23) L. B. Prus

Bonding Studies

Previous studies of the Zircaloy-2 to uranium - 12 w/o molybdenum diffusion bond indicated the diffusion of molybdenum from the fuel alloy to the Zircaloy-2. To inhibit this diffusion static bonding

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attempts were made using assemblies containing a molybdenum barrier between the Zircaloy-2 and fuel alloy. Specimens resulting were extremely brittle. Future strength evaluations will be carried out using extruded Zircaloy-2 clad uranium - 12 w/o molybdenum rod samples.

Additional tests to determine the strength of the bond between the core and clad of extruded Zircaloy-2 clad uranium - 12 w/o molybdenum rods have been conducted using a redesigned test specimen. For comparative purposes, samples of solid Zircaloy-2, solid uranium - 12 w/o molybdenum, and rod stock containing 0.002 in. and 0.0005 in. thick molybdenum barrier layers were tested. Data are as shown below:

<u>Material</u>	<u>Barrier</u>	<u>Condition</u>	<u>Bond Strength (psi)</u>
Zircaloy-2 clad U-Mo	None	As-extruded	59,900
Zircaloy-2 clad U-Mo	None	" "	46,500
Zircaloy-2 clad U-Mo	Mo (2 mils)	" "	37,200
Zircaloy-2 clad U-Mo	Mo (1/2 mil)	" "	47,600
Zircaloy-2 clad U-Mo	Ta (2 mils)	" "	30,300
Zircaloy-2 clad U-Mo	Zr (sponge)	" "	52,800
Solid Zircaloy-2	None	" "	44,900
Solid U-Mo	None	" "	85,100*

* Sample cracked at this load.

Extruded Rods

Zircaloy-2 clad uranium - 12 w/o molybdenum rods of 4 in. diam. have been extruded at 1.5/1 and 3.9/1 reduction ratios. Heating prior to extruding was done in a chloride salt bath operating at 1950°F. Resulting extruded stock is being evaluated for core microstructure and dimensions, clad uniformity, and diffusion zone at the core:clad interface. Billet assemblies for similar studies using billets of 2 in. and 3 in. diam. are being completed.

Several Zircaloy-2 billets 1.58 in. diam. containing pre-shaped nose pieces of copper and Ti-Namel, pre-shaped fuel alloy components, and a combination of pre-shaped nose pieces and fuel alloy components, have been extruded at 1950°F through a 0.314 in. diam. hardened steel die. Radiographic examinations of the rods are being completed. One rod produced from a billet containing a pre-shaped, separate Ti-Namel nose piece and a square end fuel alloy component exhibited reduced bulbing at the leading end of the rod. Additional information will be available in the next report period.

The hard faced, Carboloy insert and hardened steel insert dies have been received and tried on a limited basis for extruding Zircaloy-2 clad uranium - 12 w/o molybdenum billets heated at 1950°F. The hard faced die has indicated slightly lower extrusion pressures are required as compared to those required by previously used hardened steel dies; good retention of smooth surfaces on the die; resistance to pick-up of Zircaloy-2, Ti-Namel, and copper; good rod surface. The Carboloy die showed similar traits but not as pronounced.

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II-A-2 (Contd.)

The hardened steel insert die picked up considerable Zircaloy-2 on the first extrusion. A water-cooled die is being prepared for initial extrusion trials.

Zircaloy-2 clad uranium - 12 w/o molybdenum billets, 1.58 in. diam, containing diffusion barrier layers of molybdenum, tantalum, and sponge zirconium, have been extruded to 0.320 in. diam. rods. The barrier layer between the clad and fuel alloy was obtained by wrapping the fuel alloy components in 0.010 in. and 0.0025 in. thick foil prior to billet assembly. Preliminary microscopic examinations of cross-sectional samples taken from the rods show uniform diffusion barriers were obtained. Similar examinations will be made on rods subsequently hot and cold worked by stretching and drawing. Zircaloy-2 clad uranium - 12 w/o molybdenum billets enclosed in copper jackets are being prepared for extrusions at temperatures below 1950°F. Extrusions will be examined for diffusion bonds to determine temperature effect on width of these bonds.

Zircaloy-2 clad billets, 1.58 in. diam, containing enriched (3.6%) uranium - 12 w/o molybdenum and enriched (3.6%) uranium - 10.5 w/o molybdenum have been extruded to rod stock for CR-WAPD-Met 4A irradiation tests. Temperatures of 1950°F and 1810°F respectively and 0.322 in. diam dies were used.

Rolling Rods

Several lengths of 0.900 in. and 0.420 in. diam. compartmented rods consisting of Zircaloy-2 tubes, which contained uranium - 12 w/o molybdenum fuel alloy cores separated by a Zircaloy-2 plug, were rolled at 1020°F and 1600°F. Cracks formed in the clad surrounding the plug separator early in the rolling cycle causing eventual rod fracture in subsequent roll passes. Additional rolling using assemblies with the separating plug welded in place by external weld passes on the tubing are planned.

Swaging Rods

Swaging of compartmented rods consisting of 0.420 in. diam tubes, which contained two uranium - 12 w/o molybdenum fuel alloy fillers separated by a Zircaloy-2 plug, was attempted at 1020°F and 1600°F. Initial trials indicated the Zircaloy-2 tube elongated more than did the fuel alloy material, resulting in separation of the end and separating Zircaloy-2 plugs from the fuel alloy filler. The circumferential clad:core bond and rod dimensions are currently being evaluated.

Drawing Rods

A preliminary attempt was made to hot draw at 1800°F one compartmented rod consisting of a 28 in. long Zircaloy-2 tube (0.437 in. OD, 0.343 in. ID) filled with three 8 in. long uranium - 12 w/o molybdenum fuel alloy components separated by Zircaloy-2 plugs. The assembly fractured after approximately 6 in. of material passed through the 0.359 in. diam. die used. Additional attempts will be made using lower reduction ratios.

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II-A-2 (Contd.)

Straightening Rod Stock

An as-extruded 0.322 in. diam. x 4 ft. long Zircaloy-2 clad uranium - 10.5 w/o molybdenum rod with a maximum curvature of 5/8 in. in 4 ft was horizontally straightened to a curvature of 0.008 in. to 0.013 in. in 3 ft when resistance heated (675 amp - 10 volts) for 15 seconds to 1650°F and elongated 1/4 in. by pulling in a hydraulic drawbench. The rod was not appreciably reduced in cross section. Clad:core uniformity was good and the out-of-roundness was 0.0008 in.

An as-extruded 0.314 in. diam. x 4 ft long Zircaloy-2 clad uranium - 12 w/o molybdenum rod having a maximum curvature of 3/4 in. in 4 ft was vertically straightened to a curvature of 0.025 in. to 0.040 in. in 3 ft when resistance heated (600 amp - 10 volts) for 15 seconds to 1650°F with a 37 lb weight attached. The rod was not elongated or reduced in cross section.

End Closures

Development work on the resistance butt welding method of end-closing Zircaloy-2 clad uranium - 12 w/o molybdenum rods is continuing at Battelle Memorial Institute. Evaluations of additional end-closed rods received from Battelle indicated that (1) in tensile tests, failures occur in the Zircaloy-2 end plug showing the strength of the rod: end plug joint to be satisfactory; (2) upsetting of the fuel can occur during welding. Additional development work is being aimed at determining the cause of this upsetting.

Results of previously reported (WAPD-MRP-46) corrosion tests of eight rods end-closed by resistance butt welding show the following:

1. Specimens tested at 680°F in water with intentional defects can be expected to rupture at the weld. This is a consequence of having corrosion resistant diffusion layers along the length of the rod.
2. Specimens tested in 750°F steam with intentional defects can be expected to fail as in (1) but in less time.
3. Specimens tested in 750°F steam without defects did not fail at the end closure.
4. Wide diffusion bonds are undesirable between Zircaloy-2 and uranium 12 w/o molybdenum fuel alloy.

Approximately 75 Zircaloy-2 clad uranium - molybdenum fuel alloy rods have been end-closed by resistance butt welding and are being processed for experiment CR-WAPD-Met 4A in-pile tests.

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The design for a modified flash butt welder, to be used for end-capping of rods, has been completed by the Taylor-Winfield Company and approved. The machine will be ready for shipment to Bettis Site during the latter part of August.

Initial ultrasonic examination data received from Aeroprojects Incorporated indicate a low strength bond was formed between 0.010 in. and 0.020 in. thick Zircaloy-2 members. Examination of fractured samples showed the bond zone to be contaminated, presumably with oxygen and nitrogen as a protective atmosphere was not used. Additional attempts are being made using an inert atmosphere. In these trials the use of thicker Zircaloy-2 and Zircaloy-2 to uranium - 12 w/o molybdenum bonds will be evaluated.

The subcontract with Nuclear Metals, Inc. (formerly Metallurgical Project, MIT) discussed in WAPD-MRP-45, for development of compartmented long rod with U-12 w/o Mo fuel alloy and integral end-closures is still being negotiated.

Non-Destructive Testing

Adaptors with wedge-shaped faces and a narrow rib to contact the rod surfaces have been used in the through-type sound wave transmission ultrasonic testing setup. It has been noted that improved wave patterns on the oscilloscope screen are obtained. Additional trials will be made on defected standard samples with this arrangement.

Equipment to better evaluate the ultrasonic test setups is being designed. Three designs utilizing different drive mechanisms are being considered. Basically, what is desired is an arrangement whereby rods will be rotated and simultaneously moved between the two transducers of the testing apparatus.

The subcontract on investigation of PWR fuel element specimens for internal defects is continuing at Battelle Memorial Institute. Ultrasonic testing using conventional equipment has proved unsatisfactory. A special 30 megacycle unit has been completed; however, some difficulties must be overcome before this work can proceed. One transducer has been made; however, it was broken during the initial tests. Work is under way to develop suitable transducers. Magnetic tests are being studied using the electron microscope as an electron beam source. In this test, a change in the magnetic field about the specimen through which the current is induced is indicative of an internal flow. Tape recorder heads have been used to pick up distortion in the magnetic field. Contact and brush noises have presented difficulties; however, 1/64 in. diam. defects have been found using this method. Radiography continues to be the most informative method of examination.

A special, fast, X-ray film having small grain size has been obtained from Eastman Kodak for further development work.

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II-A-2 (Contd.)

Rod Assemblies

The aluminum inert atmosphere welding chamber, complete with vacuum equipment, has been placed in operation on the projection welder.

Several attempts have been made to projection weld fluted rod type assemblies in the inert atmosphere chamber, but fixturing has been inadequate. Changes in fixtures have been requested and work will resume before September 1. Zircaloy-2 parts for evaluation of other methods of welding rod type assemblies are on hand and will be welded into assemblies. Because of the attractiveness of the projection welding methods, and in order to make possible a more intensified effort, another 75 KVA welder similar to that on hand has been ordered. This machine will be delivered and installed during September.

3. Advanced Fuel Element Development (53c24) J. S. Theilacker

Pellet Type Fuel Elements

The I-T-E Circuit Breaker Company has delivered 3600 of the 4000 hemispherical cups. These cups were drawn from Zircaloy-2 strip of 0.020 in. and 0.030 in. thickness.

A die for cold pressing UO_2 powder into the drawn hemispherical cups has been placed in operation. It is presently being used in an experiment to determine the effect of particle size variation on UO_2 powder compaction characteristics.

Rod Type Fuel Elements Containing UO_2 Fuel

A rather extensive program concerned with fabricating fuel rods containing UO_2 fuel has been initiated. Various methods of producing such an element will be investigated. Thus far, work has been initiated on determining feasibility of fabricating UO_2 rod type elements by (1) extrusion; (2) tamp filling UO_2 powder into Zircaloy tubing followed by reduction of this composite by either rolling, drawing or swaging to densify the UO_2 ; (3) steam oxidation of uranium metal to UO_2 within Zircaloy-2 tubing; and (4) by compaction of UO_2 powder into cylinders which are then loaded into Zircaloy tubing.

Six Zircaloy-2 clad UO_2 rods are now being fabricated for thermal cycling tests. Slugs $3/8$ diam x 0.3 in. long are being cold compacted and will be inserted into 5.5 in. long Zircaloy-2 tubes. Two tubes each of 70, 80 and 90% bulk density UO_2 will be made. Dies with tungsten carbide sleeve liners are being used for producing the slugs and appear to be much more resistant to the abrasion of the UO_2 than do the hardened tool steel dies formerly used.

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4. Fuel Element Development (53c25) W. J. Hurford

Extruded Rods

An attempt was made to hot extrude two unclad uranium - 3.8 w/o silicon billets, 1.58 in. diam., at 1700°F through a 0.437 in. diam. die. One billet was tried as-cast; the second, as-heat treated (epsilonized) at 800°C for sixteen hours. The as-cast billet failed to extrude. The heat treated billet extruded but ignited and burned. Additional work will be done using copper clad epsilonized billets.

Nine Zircaloy-2 clad billets containing epsilonized uranium - 3.8 w/o silicon fuel alloy components were prepared and extrusion attempts were made at temperatures of 1600°F to 1700°F. Extrusions were obtained with only three of these billets, one each at 1630°F, 1650°F and 1700°F using a 0.437 in. diam. die. Preliminary dimension and metallographic data indicate the following: the maximum variation in clad thickness was 0.007 in.; the maximum variation in core diameter, 0.007 in.; the clad:fuel bond was not continuous throughout the length of any of the three rods. In the 1630°F extrusion, approximately 25% of the clad:fuel interface was unbonded. The 1700°F extrusion was somewhat better with an approximate 12% of the clad:fuel interface being unbonded. Future work, involving the use of copper as an aid to lubrication and various combinations of extrusion temperatures and reduction ratios, is being scheduled.

Rolling Rod

Preliminary investigations of hot drawing at 1800°F as-extruded Zircaloy-2 clad uranium - 3.8 w/o silicon fuel alloy stock were made using a 0.437 in. diam. rod. This rod was extruded at 1700°F from a billet containing an as-cast fuel alloy component. Prior to the drawing attempts, a satisfactory starting end was swaged on the rod. Upon drawing, rod failures occurred at the entrance to the 0.390 in. diam. draw die. Additional investigations using rod extruded from billets containing epsilonized fuel alloy components are in process.

Preliminary attempts were made to hot draw a compartmented rod consisting of a 0.437 in. OD x 0.343 in. ID Zircaloy-2 tube 28 in. long containing three 8 in. long uranium - 3.8 w/o silicon fuel alloy fillers separated by Zircaloy-2 plugs. This assembly was heated inductively to 1800°F and pulled through a 0.359 in. diam. die. Approximately 7 in. of the assembly passed through the die before failure occurred at the filler-plug interface. Similar work using epsilonized fuel alloy fillers will be done.

Plutonium Fuel Rods

All of the fuel components for the plutonium fuel rods being manufactured at Los Alamos Scientific Laboratory (discussed in Section II-B-3, WAFD-MRP-46) have been machined. Welding of end closures on the fuel rods will start during the latter part of August. The rods are scheduled to be completed during September.

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II-A-5

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

The following items were reported in WAPD-MRP-46 under Thermal and Hydraulic Tests of PWR Core Components (55a15), and Basic Core Engineering (55a16). The progress on some of the items covered below is also discussed under progress reports for other projects in cases where the work is of interest to more than the one project.

Parallel Rod Pressure Drop

Experiments to determine pressure losses through joints between parallel rod subassemblies are continuing. Water is used in the tests at Reynolds numbers ranging from about 28,000 to 58,000. End plates having circular holes equivalent to 90%, 100%, and 110% of the tube section flow area are being tested. Models having oblong holes in the end plates have been fabricated and are ready for tests.

These tests measure the total of joint losses and pure friction losses. The pure friction losses are calculated from the Moody curves and subtracted from total pressure losses to determine joint losses. The joint coefficient is believed to be a function chiefly of the Reynolds number based on equivalent diameter within the subassembly, the area ratio, and the ratio of equivalent diameters. Test results are presently being evaluated. An attempt will be made to correlate the above variables after all basic data have been processed.

Velocity Distribution for Parallel Flow

Using an enlarged model in an air loop, the circumferential velocity variations of a rod in a parallel flow rod bundle have been determined to be 7.5%, 6% and 3% at Reynolds numbers of 66,000, 110,000, and 135,000 respectively. These velocity variations were measured at a distance away from the test rod equivalent to one-half the distance between the rods at their nearest point. The prototype lattice of interest consisted of 0.3 in. diam. rods with a 0.375 in. square pitch. The model scale was 8 1/3 times the prototype scale.

Water Heat Transfer Tests for Parallel Flow Rods

A review of subcontract proposals submitted by outside organizations for conducting heat transfer studies in parallel rod flow has been completed. A proposal for an investigation of heat transfer in packed spheres was also evaluated in connection with another project. Since it is anticipated that economy can be effected by placing both investigations with a single vendor, a combined proposal will be solicited from those vendors which were selected in the current review as having the greatest potential for conducting the research involved.

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The minimum conditions required in the conduct of the heat investigations for parallel flow rods :

- a. Reynolds number, up to 500,000 based on hydraulic diameter
- b. Prandtl number 1.0
- c. Triangular and square configurations in the rod bundle with different center-to-center spacings of the rods.

Air Heat Transfer Tests for Parallel Flow Rods

The design of an air flow test section for the measurement of heat transfer numbers will be inferred from mass transfer data. The test section will consist of a naphthalene-surfaced rod surrounded by metal rods in an air flow loop. Depth of surface removal, as measured by a comparator, will be used to calculate mass transfer coefficients.

Average values of heat transfer will also be measured by the transient technique. A hollow metal rod is to be heated uniformly by an internally located Calrod. A section of the tube, well downstream, will be insulated from the rest. The temperature history of this section will be recorded while the rod is being cooled by airflow. Heat transfer coefficients will be calculated from the rates of exponential temperature decay.

The Reynolds number range for these two sets of tests will be from about 60,000 to about 140,000.

Contribution of Eccentric Fueled Region to Hot Channel Factors

In the manufacture of fuel rods it is expected that some variation in clad thickness around the rod must be tolerated. For purposes of computing hot channel factors, an analysis has been made that predicts the ratio of maximum heat flux to average heat flux, for any given eccentricity of the fueled region with respect to the rod as a whole. The result of this work is expressed to a close approximation by the following formula:

$$\frac{\phi_{\max}}{\phi_{\text{ave}}} = 1 + \frac{\frac{e}{b}}{2 \frac{-a}{b} + \frac{k}{bh}} + \frac{\frac{(e/b)^2}{2 \left(2 \frac{-a}{b} + \frac{k}{bh} \right) \left(3 \frac{-2a}{b} + \frac{2k}{bh} \right)}} + \dots$$

Here: a = core radius

b = outside radius of rod

e = eccentricity or distance between centers of rod and core

h = heat transfer coefficient

k = thermal conductivity

ϕ = heat flux through outside surface.

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If we take $b = 0.15$ in., $a/b = 0.8$, $e = \frac{1}{6}(b - a)$, $k = 8$ Btu/hr °F Ft and $h = 9000$ Btu/hr ft²°F, then it results that $\frac{\phi_{\max}}{\phi_{\text{ave}}} = 1.026$.

Bettis Cyclic Loop

All major loop components are on order. The loop detail piping drawings have been completed and released for manufacture with a due date of October 1. Advance material lists were issued to insure that the required material will be on hand when needed. Detail drawings are being prepared for the cross-flow rod calibration test section. The electrical drawings are also being prepared.

High Pressure Pressure-Drop Loop

All fittings and major components for this loop have been placed on order. The expected date to complete delivery of all parts is about September 1. Loop pumps (150-C) are presently undergoing routine tests and are expected to be delivered by early September. Welding on all fittings and piping is being done as they are received. Assembly of the major components and larger pipes is expected to start by early September.

The control panel stand is complete and has been erected at the loop site. Approximately 75% of the wiring is complete. The delivery of all electrical equipment and parts is complete.

Burnout Loop

Subcooled burnout testing on the tubular specimen is continuing. Preliminary calculations have been made and some initial results are tabulated below. The bulk of the data shown below falls between that reported by Purdue and UCLA. Some of the data show about 30% higher burnout points than other data taken under the same conditions. These high points, the last obtained, were taken after two changes were made in the loop. The high flow sensing element was replaced by the low flow range element, and the Tektronix amplifier used in the burnout detector was replaced temporarily for repairs by another amplifier of the same make and model. A systematic check is being made to determine the cause of this discrepancy.

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TABLE I

TABLE OF SOME OF THE TENTATIVE BURNOUT DATA AT 2000 PSIA

Burnout Flux Btu/hr - ft ²	Mass Flow Rate lb/hr - ft ²	Nominal Inlet Velocity-fps	Exit Subcooling OF
4.09	6.06	30	161
3.36	5.68	30	111
2.71	5.42	30	61
2.24	5.14	30	36
1.64	4.81	30	5
3.75	4.98	25	161
3.11	4.84	25	111
2.52	4.59	25	61
2.12	4.26	25	36
1.56	4.04	25	5
4.68	4.24	20	161
3.16	4.08	20	161
2.73	3.76	20	111
2.28	3.7	20	61
1.89	3.57	20	36
1.33	3.33	20	3
4.02	3.21	15	161
3.24	3.11	15	121
3.01	3.01	15	110
3.16	2.18	10	161
2.59	2.1	10	111
1.69	1.08	5	108
1.46	0.699	3	121
1.46	0.7	3	93
1.02	0.45	2	58

Welding experiments on samples of the nickel rectangular test section are being made in order to determine the best technique for welding the actual test section.

6. PWR Pilot Fuel Element Manufacture (53ch1) H. C. Amtsberg

The installation of the 50 kw 3000 cycle generator for the vacuum induction melting furnaces and the small autoclaves for sample testing in the G-Building has been temporarily delayed by higher priority work in the CX-Building. Numerous machine tools, 4 lathes, 1 tool and cutter grinder, and 1 horizontal milling machine, obtained from AEC surplus stocks, have been put into operating condition and readied for installation.

A trip was made to the Watson-Stillman Company, Rosselle, New Jersey, to discuss layout plans for the 1250-ton extrusion press with auxiliary equipment, and to establish design features of the press with respect to billet loader, shear, tooling, and controls.

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The monitor tanks have been installed. Delivery of additional equipment for the monitor tank room, such as spray dryer and vapor distillation units, is expected around October 1.

Pickling and ventilation facilities are being prepared for installation.

II-B-1

B. Metallurgy of Core Materials

1. Physical Metallurgy of FWR Core Materials (53b11) D. E. Thomas

Uranium-Molybdenum Phase Diagram Studies

Points on the phase diagram in the region of 11 to 13 w/o molybdenum at 550°C had previously (Fig. 16, WAPD-MM-413) been reported to be three phases after a three week anneal indicating that the samples had not reached thermal equilibrium. Samples which have been annealed in salt pots for 93 days at 550°C have been examined by X-ray diffraction. These new results indicate that the $\epsilon + \alpha/\epsilon$ phase boundary should be shifted toward higher Mo concentration, and that the $\gamma + \epsilon/\epsilon$ boundary should be lower in temperature as the 12 w/o sample contains the gamma phase.

The specific heat as a function of the temperature has been determined for an 8 w/o uranium-molybdenum alloy. The transformation from $\alpha + \epsilon$ to $\alpha + \gamma$ occurs at $577 \pm 3^\circ\text{C}$. The specific heat begins to increase at about 400°C and continues to increase up to the transformation temperature, thus possibly indicating a latent heat of ordering. Samples of higher Mo content are also being measured.

The intensities of superlattice lines of the ϵ phase are being measured as a function of time. The presence of these lines in a 15 w/o alloy can be detected after 4 hours at 500°C.

"Strained Alpha" Structure Studies

A series of binary alloys of uranium containing 1-6 w/o niobium in steps of 1% were homogenized at 1075°C for a week, then water quenched from 900°C. X-ray diffraction spectrometer traces in the range of 32° - 42° of 2θ show a gradual transition from a well resolved α pattern in the 1% alloy to the ill-defined "distorted alpha" structure in the 6% alloy, the latter having a very broad peak with a maximum at 36.35° and a breadth at half maximum of about 1.5° .

In the 1% alloy the four α lines, - 110, 021, 002, and 111 - are resolved while in the 2% alloy the 021 and 002 peaks are merged into one at 35.75° . As the Nb concentration is increased, this peak becomes broader and shifts toward higher values of 2θ . Also the 111 peak decreases in intensity. In the 6% alloy both the 111 and the 110 peaks are absorbed into the broad 36.35° "distorted alpha" peak.

Hardness measurements show that the hardness passes through a maximum at about 2 w/o Nb.

<u>Alloy Content, w/o Nb</u>	<u>Hardness, Vhn</u>
Derby U	237
1	338
2	380
3	327
4	298
5	237
6	164

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A coarse needle-like structure exists in the 1 w/o Nb alloy, while 3, 4, and 6 w/o Nb alloys exhibit a fine needle-like structure.

X-ray diffraction data on a similar series of alloys containing zirconium also indicate a gradual change to "strained alpha" with increasing alloy content. In this case the transition does not appear to occur as rapidly with increasing alloy content as is the case in the uranium-niobium alloys. Thus the pattern of the 5 w/o Zr alloy is similar to that of the 3 w/o Nb alloy. The gradual evolution of "strained alpha" is indicated also by X-ray diffraction data for a series of dilute uranium-vanadium alloys, in which case vanadium is about as effective as niobium when considered on an atomic percentage basis.

Transformation in Uranium-Molybdenum Alloys

The T-T-T studies have been completed for uranium-molybdenum alloys initially gamma treated at 900°C. The start of transformation for alloys containing 7 to 12 w/o molybdenum as indicated by metallographic examination and X-ray diffraction data is shown in the table below. After 1500 hours at 300°C, 350°, and 400°, there is no definite indication of the presence of any phase other than α , γ , and ϵ in the unstrained material. The nose of T-T-T curve decreases in temperature and increases in time with increasing alloy content.

TABLE II
Time for Start of Transformation - Hours

<u>Reaction</u> <u>Temp. °C</u>	<u>7 w/o Mo-U</u>	<u>8 w/o Mo-U</u>	<u>9 w/o Mo-U</u>	<u>10 w/o Mo-U</u>	<u>12 w/o Mo-U</u>
550°C	0.34	1.4	4.6	760	--
500°C	0.26	1.2	2.5	13	22
450°C	1.0	4	5.2	7.5	35
400°C	19	19	30	18	180
350°C	140	140	800	1,500	1,500
300°C	1,500	1,500	1,500	1,500	1,500

A large grain in a 7 w/o Mo-U was outlined by Tukon hardness impressions which also served to strain the material within the grain. Back reflection Laue patterns have been obtained immediately after straining and after annealing the material at 400°C for 100, 200, and 600 hours. After 100 hours the single crystal gamma spots become darker and better defined and some of the original gamma spots become very weak. The pattern appears to be that of one or two crystals with a symmetry lower than cubic.

Samples of 7 and 8 w/o Mo-U have been gamma quenched and cold rolled and are now being annealed in order to produce the above transformation. An attempt will then be made to determine the structure of the transformation

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II-B-1 (Contd.)

product. Samples of 6, 9, 10 and 12% w/o molybdenum cold-reduced 75% in area showed only an acceleration of alpha precipitation (about 5-10 times) at 300, 350, 400 and 500°C.

The powder pattern for strained alpha in a 4 w/o Mo-U alloy has been obtained and found to agree with that reported by Los Alamos Scientific Laboratory. Preliminary work indicates that the structure is either orthorhombic or monoclinic. The a_0 and c_0 values for the structure have been determined and an attempt is now being made to determine b_0 .

Hydride Formation in Uranium-Molybdenum

An experiment was cited in WAPD-MM-429 in which hydride in the U-Mo alloy portion of a couple with Zircaloy-2 was found to be absent near the plane of the join with Zircaloy-2. This indication of the gettering function of the zirconium cladding is also revealed in two additional experiments.

A specimen of Zircaloy-2 clad U-10 w/o Mo alloy rod was prepared by machining away the cladding along its length except for a short piece at one end. The specimen was cathodically hydrided, and slices were taken at various distances from the end on which the cladding remained. These slices were analyzed for hydrogen content with the results shown in Table III. A lower hydrogen content near the Zircaloy-2 cladding is evident.

TABLE III

<u>Distance from Cladding, in.</u>	<u>Hydrogen Content, ppm</u>
0 - 0.044	166
0.045 - 0.083	158
0.270 - 0.316	250
0.313 - 0.357	226
0.354 - 0.387	180

The core and cladding of a Zircaloy clad U-12 w/o Mo sample containing a simulated cladding defect (a hole 40 mil in diameter) were analyzed in the immediate vicinity of the cladding defect and at distances of 1/2 in. and 1 in. on either side of this defect after 150 hours in corrosion test at 650°F. Analyses on the core material average about 2 ppm of hydrogen. Analyses of the Zircaloy indicate a concentration of hydrogen of 82 ppm in the vicinity of the defect, while that at 1/2 in. and 1 in. distance is 43 ppm. The magnetic test for hydride failed to show the presence of hydride in this rod after corrosion testing. This was confirmed metallographically. It is evident that too little corrosion has occurred in this specimen to result in high hydrogen concentration in the cladding and in the core. Similar samples are being corroded for longer times to show more pronouncedly the distribution of hydrogen.

The alloy hydride formed as a result of cathodic hydriding a U-12 w/o Mo alloy has been examined with the electron microscope to determine whether the platelets consist of one or more phases. The micrographs obtained showed fine

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II-B-1 (Contd.)

structure within the platelets and uneven interfaces with the matrix metal. Regular interfaces would be expected if formed by a shear mechanism. The platelets seem to be a single phase made up of a series of overlapping circular constituents resulting in a transverse fine structure. Further observations are being made to insure that this appearance is not an etching artifact.

The correlation of hydrogen content with microstructure and changes in lattice parameter and density of a 10 w/o Mo-U alloy is still being studied. Initial results indicate that the lattice parameter of the matrix increases with increasing hydrogen content until a limiting value is reached which corresponds metallographically to the appearance of the hydride needles. The density of the alloy also decreases with increasing hydrogen content both before and after the appearance of the hydride phase. From the present results the hydride phase appears to have the composition $\text{UH}_{1.4}$ with a density of 11.1 gm/cc. The samples used in this investigation do not appear to have been hydrided uniformly although they had been ground and sectioned so as to present as nearly uniform volumes as possible for analysis.

The experiment is now being repeated using a molybdenum-clad cylinder which will be cathodically hydrided from an end face. Changes in lattice parameter and microstructure will then be determined as a function of distance from the end of the cylinder. The chips will be used for density determinations and hydrogen analysis.

Uranium-Silicon Alloys

Uranium-3.8 w/o silicon alloys have been annealed various lengths of time at 700 and 900°C to determine the rate of formation of the ϵ compound (U_3Si). The samples were in the form of extruded rod and the initial hardness was 335 ± 5 . At a hardness of 268 the samples appear to be all ϵ metallographically. Corrosion results for these samples show that the as-extruded material disintegrates in less than 24 hours in 650°F water while materials annealed to 268 DPH or less approach a base rate of 4.0 mg/cm²/hr. Further annealing decreases the hardness and increases grain size, but results are not yet available as to the effect on the corrosion rate.

Preliminary results shown in Table V indicate that the addition of 0.5 w/o vanadium or aluminum increases the corrosion resistance of U-3.8 w/o Si appreciably for short times, but has only a minor effect during longer test periods.

II-B-1 (Contd.)

TABLE IV

Average Corrosion Rate in 650°F Water, mg/cm²/hr

Composition, w/o	24 hr	96 hr	168 hr
U-3.8 Si	- 4.5	- 4.0	- 4.6
U-3.8 Si, 0.5 V	- 2.2	- 1.8	- 4.3
U-3.8 Si, 0.5 Al	- 3.0	- 2.1	- 3.9
U-3.6 Si	+ 0.20		Disint.
U-4.0 Si	- 6.0		Disint.

Addition of 0.5 w/o of titanium or nickel to the U-3.8 w/o Si base resulted in poor corrosion resistance. The microstructure of the above mentioned ternary alloys after the epsilonizing treatment consists of epsilon plus a small amount of an unidentified phase. The data in Table IV also confirm that 3.8 w/o silicon alloys have much better corrosion resistance than alloys containing 0.2 w/o more or less silicon.

An epsilon phase has also been found at 25 a/o germanium. A new phase is observed to form by a peritectoid reaction between alpha (or gamma) and U-Ge compound. This material has only minor corrosion resistance to 650°F water.

2. Mechanical Metallurgy of Core Materials (53b21) L. S. Castleman

Fretting of FWR Materials (MM 13)

Six tests (Zircaloy-2 vs. Zircaloy-2, 6,000 psi pressure; 0.003 in. slip; and 2.5×10^6 cycles) have been conducted to check the reproducibility of weight loss measurements as a means of evaluating fretting corrosion damage. Removal of the fretting corrosion product by plastic strain and mechanical brushing appears satisfactory for the above set of conditions. Weight loss measurements for four of the six tests were in agreement and are tabulated below:

Test No.	Debris Collected During Test (mg)	Weight Loss of Plate Specimen (mg)
17	33.0	11.2
18	7.8	1.9
19	6.9	2.3
20	6.9	2.3
21	7.9	1.7
22	14.3	5.5

Two tests showed much higher weight losses, but it was evident that the method of fretting corrosion product removal was not the cause of the disagreement;

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II-B-2 (Contd)

the fretting action was clearly more severe in these two tests. It is probable that uncontrolled atmospheric conditions (temperature and humidity) influenced the results. The fretting apparatus is therefore being equipped to control the atmosphere and temperature, and further tests on reproducibility will be conducted.

Fracture Mechanism of Zirconium and Its Alloys (MM 14)

The tensile properties of arc-melted crystal bar zirconium have been investigated at two testing temperatures - room temperature and -196°C. Samples in the hot rolled, beta quenched, and beta annealed conditions were given a hydrogen solution heat treatment and were subsequently quenched to retain the hydrogen in solution, or slow cooled to precipitate hydride platelets. A pronounced increase in reduction of area at -196°C appears to be associated with the absence of hydride. The fracture is of the typical cup-cone type, whereas the fracture surface of specimens containing precipitated hydride is rough, porous, and granular in appearance.

Compression of Zirconium and Its Alloys (MM 15)

In order to determine the effects of alloying elements on the ductility of zirconium, compression tests on zirconium and zirconium alloys have been performed. Compression testing at 500°F and the testing of additional alloys have supplemented the results reported in WAPD-MRP-46. The 500°F results show that alloys which contain 1 w/o or more of elements such as Fe, Sn, Ni and Cr, or 0.35 w/o O₂ fail in the compression tests, whereas alloys containing smaller percentages do not fail.

Tensile Properties of Uranium Alloys (MM 16)

U-12 w/o Mo Alloy

Preliminary tensile test results have been obtained on as-extruded and on as-extruded and gamma quenched specimens of U-12 w/o Mo alloy tested at temperatures ranging from room temperature to 600°F. The results are summarized in Table V.

In general, the alloy exhibits little or no ductility at room temperature but does show some ductility at higher testing temperatures; this indicates the presence of a brittle transition temperature for the material -- a characteristic of body centered cubic metals.

The load-extension curves indicate a number of interesting and unusual features. Upper and lower yield points appear in the 500°F tests, the maximum yield point elongation being about 2%. Also, in the 500°F and 600°F tests on as-extruded specimens, serrations appear in the load-extension curve; similar serrations appear in the 600°F curve for the gamma quenched material. Apparently, the metastable gamma phase is transforming during the plastic straining, or some other phase is precipitating out of the matrix. Preliminary metallographic and X-ray diffraction studies have failed to reveal any changes in the microstructure of the material following tensile testing. In addition,

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II-B-2 (Contd)

the strain hardening behavior is peculiar. For example, specimen No. E-4 (gamma quenched and tested at room temperature) yields at a true stress of 180,000 psi, whereas the true stress at fracture is calculated to be 170,000 psi; however, specimen No. E-6 (gamma quenched and tested at 500°F) yields at a true stress of 103,000 psi and fractures at a true stress of 144,000 psi. Since work hardening during the room temperature test is either absent or occurs at a low rate, and since work hardening normally decreases with increasing test temperature, it is evident that the hardening observed during the 500°F test must be due to strain aging effects. Although marked serrations are absent in the load-extension curve for 500°F, the curve is not smooth and exhibits definite discontinuities, which indicate that phase changes are taking place and causing the strain aging. Another indication of strain aging was obtained in the tensile test of the as-extruded specimen No. Y-2 at 600°F. During the test, the rate of extension was lowered from 5×10^{-2} in./min to 5×10^{-3} in./min after sufficient straining had occurred to produce serrations in the load-extension curve. The average load immediately increased by 2 percent, and the period of the serrations also increased.

A few substandard Izod impact tests were run on notched and unnotched specimens at room temperature and at 500°F. As expected, the notched specimens were brittle at both testing temperatures; the unnotched specimen was also brittle in the room temperature test, but exhibited moderate ductility at the 500°F test temperature.

U-4.3 w/o Si Alloy

Preliminary tensile test results have been obtained on U-4.3 w/o Si alloy samples epsilonized after extrusion for 64 hours at 800°C; testing temperatures ranged from room temperature to 600°F. The results are summarized in Table V.

It is apparent that material of this composition has little ductility, and that its tensile properties are not influenced by temperature in the range of testing temperatures employed. The three load-extension curves have the same general shape - the load rises steeply during plastic extension, and fracture occurs with little or no necking.

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TABLE V

Tensile Properties of U-12 w/o Mo Alloy

Specimen Number	Heat Treatment	Test Temp. (°F)	0.5% Yield Strength (psi)	Tensile Strength (psi)	Percent Elongation		Percent Reduction In Area
					Uniform	Total	
*E-1	As extruded	R.T.	---	161,000	0	0	0
E-2	" "	250	---	142,000	0	5.9 in 1.5"	40
E-3	" "	500	109,000	116,000	6.9	10.0 in 1.5"	34
*Y-1	" "	500	114,000	120,000	6.1	6.1 in 1.0"	15
Y-2	" "	600	109,000	117,000	6.9	6.9 in 1.0"	22
Y-3	" "	600	108,000	118,000	5.8	6.8 in 1.0"	16
E-4	800°C for 1 hr	R.T.	---	173,000	0	2.1 in 1.5"	12
E-5	" " " "	500	110,000	112,000	2.7	5.2 in 1.5"	32
E-6	" " " "	500	105,000	109,000	4.2	7.0 in 1.5"	30
Y-4	900°C for 24 hr	600	100,000	103,000	1.3	3.8 in 1.0"	33

* Letters refer to ingot number

Tensile Properties of U-4.3 w/o Si Alloy

Specimen Number	Heat Treatment	Test Temp. (°F)	0.5% Yield Strength (psi)	Tensile Strength (psi)	Percent Elongation		Percent Reduction In Area
					Uniform	Total	
L-1	800°C for 64 hr	R.T.	---	55,000	0.4	0.4	0
L-2	" " " "	500	50,000	58,600	1.1	1.1	7.8
L-3	" " " "	600	50,000	51,900	0.6	0.6	1.5

II-B-2 (Contd)

II-B-2 (Contd.)

Damping Characteristics of Zirconium-Gas Alloys (MM 17)

An apparatus has been completed for an investigation of the damping characteristics of zirconium and zirconium alloys between room temperature and 450°C. Torsional damping will be investigated on 0.060 in. diam and 9 in. long specimens subject to very small strain amplitudes (of the order of 10^{-5} in./in.). Zirconium-oxygen alloys will be investigated first. The location and size of the oxygen peak on the internal friction vs. temperature curve will be determined as a function of frequency, and the activation energy for the movement of oxygen interstitials in zirconium will be calculated. The information so obtained should prove of value in determining the effects of oxygen upon the mechanical properties of zirconium and may serve as a convenient analytical tool.

Thermal Cycling of PWR Fuel Elements (MM 18)

Since it is probable that the PWR fuel element will consist of a Zircaloy-clad and bonded or unbonded fuel alloy, the effects of the inherent differential thermal expansion during thermal cycling on the stability of the fuel elements should be ascertained. The thermal cycling apparatus, utilizing sodium-potassium alloy as the heat transfer medium, is being made ready to perform the tests necessary to yield the required information. It is expected that the apparatus will be ready for operation by August 30 and that preliminary results will be available by September 7. The following cycling conditions will be used for all specimens unless the results of early tests indicate a need for changes:

Hot Tank Temperature - 1050°F (est. max core temp. in reactor with metallic fuels)
Cold Tank Temperature - 550°F (est. water inlet temp. to reactor)
Transfer Time - 5 seconds (both directions)
Holding Time - 2 minutes (both tanks)

Samples of the following materials are to be tested:

Unclad and clad (bonded and unbonded) U-3.8 w/o Si alloy
Unclad and clad (bonded and unbonded) U-12 w/o Mo alloy
Clad UO_2 samples with UO_2 cold pressed to various densities
Clad UO_2 samples with UO_2 fused to various densities

3. Corrosion of PWR Core Materials in Static Systems (53b41) D. E. Thomas

Clad Alloys

Testing of extrusion clad alloys has continued. The end cap type of failure appears to be absent when the end caps are properly welded and when corrosion does not proceed preferentially along the clad-core interface. It may be noted that corrosion proceeds between core and cladding and penetrates to the end cap where extensive corrosion and cracking of the specimen ends and splitting of the end caps takes place.

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Clad binary alloys containing 9, 12 and 13.5 w/o Mo are still in test after corrosion testing in 650°F water for 77 days, while a clad rod with a 10.5 w/o Mo core failed in 49 days. At the higher testing temperature of 750°F (steam) some samples of clad 9 and 10.5 w/o Mo have survived 54-68 days while others have failed earlier.

Clad binary alloys containing Nb have also been corrosion tested. Clad rods containing U-6 w/o Nb failed in 7 to 22 days in 650°F water, while rods containing U-10 w/o Nb are still in test after 73 days in 650°F water, and 56 days in 680°F water.

Clad rods containing the ternary alloys 6 w/o Mo - 3 w/o Nb, 7 w/o Mo - 2 w/o Nb, 9 w/o Mo - 0.5 w/o Nb, and 10.5 w/o Mo - 1.5 w/o Nb have survived 42 days test in 650°F water. In 680°F water, clad rods containing the ternary alloys 7 w/o Mo - 2 w/o Nb, 8.5 w/o Mo - 0.5 w/o Nb, 9 w/o Mo - 0.5 w/o Nb, 10.5 w/o Mo - 1.5 w/o Nb, and 12 w/o Mo - 0.5 w/o Nb have survived 28 days, while one containing the ternary alloy 6 w/o Mo - 3 w/o Nb failed within 14 days. A summary list of clad sample corrosion tests is shown in Table VI.

Clad rods containing U₃Si suitable for conducting a meaningful corrosion test have not yet been obtained. In order to evaluate the corrosion behavior of defected clad rods it is necessary to minimize the diffusion layer between clad and core such as is obtained when the epsilonizing treatment follows the extrusion step. Attempts are being made to epsilonize before extrusion, then extrude at temperature below 930°C, the temperature at which epsilon decomposes. Several such attempts have failed to produce a bond. In those cases where apparently successful extrusions resulted, corrosion data are not promising.

Simulated defected fuel elements of extruded rods with molybdenum or tantalum barrier layers 0.001-0.002 in. thick were prepared for corrosion testing in the usual manner. These were then tested in 1500 psi of steam at 400°C for a period of 7 days. Swelling and rupture of the cladding occurred for elements with a tantalum barrier layer, while swelling, especially near the end plugs, and some splitting of the cladding occurred in the molybdenum barrier layer elements. Instead of prolonging the corrosion life of extruded elements, introduction of a barrier layer has apparently accelerated failure.

A cross-sectional photomicrograph of a tantalum barrier layer element after corrosion test shows that the uranium alloy immediately adjacent to the barrier layer has corroded. The poor results for the Mo barrier layer samples is especially puzzling, since elements made by melting and solidifying uranium alloy in a molybdenum crucible and corrosion tested at an exposed face show excellent corrosion resistance and no indication of corrosion attack at the interface, even in samples in which considerable diffusion has occurred. As would be expected from the phase diagram, no separate phase layers occur between the molybdenum and the uranium-molybdenum and no interface attack occurs on hot water testing. The photomicrographs are quite instructive since they indicate that extensive cracking of the tantalum had occurred as a result of the cladding expansion on corrosion. The ductility of tantalum should be sufficient to permit it to deform without cracking to the extent (about 1%)

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II-B-3 (Contd.)

noted in the photomicrograph. It is inferred, therefore, that some pickup of hydrogen by the tantalum had occurred, sufficient to embrittle but not enough to prevent uranium corrosion. The molybdenum barrier layer did not crack under the same conditions, apparently because molybdenum is not a hydride-former.

The following suggestion is made to explain these observations. Corrosion at the base of a defect produces atomic hydrogen and uranium oxide. In the presence of zirconium metallurgically bonded to the uranium alloy, the hydrogen diffuses into the zirconium and is not deleterious to the uranium alloy. However, if a poorly corrosion-resistant layer is interposed between the zirconium and the uranium so that the diffusion path for hydrogen is interrupted, or if a barrier is interposed between core and clad which prevents hydrogen diffusion at a suitably rapid rate, the hydrogen is absorbed into the core where it accelerates corrosion. As a result of corrosion, the core will be split mechanically from the cladding and entry of water to the end closures may occur. Corrosion of an enclosed sample should be even more severe than that of a bare sample since in the latter case much of the hydrogen can escape into the environment. The remedy for this condition would appear to be limitation of core: clad interdiffusion by fabricating at lowest possible temperatures or use of barrier layers of such thicknesses that no obstacle to hydrogen diffusion is presented.

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II-B-3 (Contd.)

TABLE VI

Corrosion Life of Zircaloy Clad Uranium Alloys
in High Temperature Water

Alloy Content (wt %)	Condition	Test Temp. (°C)	Corrosion Life (Days)	Remarks
9 Mo	As-extruded	343	77	In test
"	"	360	40-54	General splitting of clad
"	"	400 (steam)	59-66	General swelling and rupture of clad
10.5 Mo	As-extruded	343	113	In test
"	"	360	82-96	Burst cladding
"	"	400 (steam)	58-65	Pimpling at holes prior to general rupture
12 Mo	As-extruded	343	77	In test
"	"	360	68	In test
"	"	400 (steam)	14-21	Cladding ruptured
13.5 Mo	As-extruded	343	77	In test
"	"	360	68	In test
"	"	400 (steam)	14	General swelling near holes
6 Nb	As-extruded	343	22	Failed
10 Nb	As-extruded	343	59	In test
"	"	360	42	In test
"	"	400 (steam)	14	In test
3.8 Si	As-extruded	343	7	General swelling and rupture of clad
"	800°C/4 hr	343	7	General swelling and rupture of clad
"	800°C/16 hr	343	7	General swelling and splits in clad
"	800°C/20 hr	343	7	General swelling and splits in clad
12 Mo + 0.5 Nb	As-extruded	400 (steam)	28	In test
7.5 Mo + 1.5 Nb	"	400 (steam)	14	In test
9 Mo + 1.4 Nb	"	"	14	In test
8 Mo + 0.3 Ti	"	"	14	In test
6 Mo + 3 Nb	"	343	42	In test
6 Mo + 3 Nb	"	400 (steam)	14	Failed
7 Mo + 2 Nb	"	343	42	In test
7 Mo + 2 Nb	"	400 (steam)	28	In test
9 Mo + 0.5 Nb	"	343	42	In test
9 Mo + 0.5 Nb	"	400 (steam)	28	In test
10.5 Mo + 1.5 Nb	"	343	42	In test
10.5 Mo + 1.5 Nb	"	400 (steam)	28	In test

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II-B-3 (Contd.)

Unclad Alloys

The effect of temperature upon the time to reach discontinuous failure of the U-12 w/o Mo alloy has been determined over the range 316° to 400°C. Tests at 260°C have not revealed discontinuous failure in the times thus far accumulated (70 days). The slope of the curve for log of time for discontinuous failure vs. reciprocal of absolute temperature gives an activation energy of 23,000 cal/mol while the slope of Fig. 3 of WAPD-MM-271 shows an energy of 28,600 cal/mol for the curve of corrosion rate vs. temperature. It thus appears likely that the time for discontinuous failure is closely related to the corrosion rate in this class of alloys.

Double heat treatments of uranium-molybdenum alloys containing 9 to 15 w/o Mo have been completed, and metallography, X-ray, hardness, and corrosion data have been taken. The heat treatments were designed to control the mode of transformation in the hope that a distribution of alpha phase might be obtained which is not detrimental to the corrosion resistance of the material. Table VII. illustrates the corrosion results for the heat treatments thus far completed.

TABLE VII

Failure Period (Days) in 650°F Water for U-Mo Alloys

<u>Heat Treatment*</u>	<u>9 Mo</u>	<u>9.8 Mo</u>	<u>10.7 Mo</u>	<u>12 Mo</u>	<u>13.5 Mo</u>
550°C/3 wk	0-1	0-1	0-1	14	7
400°C/1 wk	1-7	14-28	14-28	28-42	7-28
400°C/3 wk	1-7	7-14	28-42	56	7-14.
400°C/5 wk	0-2	1-7	7-14	56	14-28
500°C/1 wk	0-1	0-1	0-1	7-14	14-56
500°C/3 wk	0-1	0-1	0-1	1-7	14-28
550°C/5 wk + 400°C/1 wk	0-1	0-1	0-1	1-7	1-7
550°C/5 wk + 400°C/3 wk	0-1	0-1	0-1	0-1	1-7
400°C/5 wk + 500°C/1 wk	0-1	0-1	0-1	0-1	28
400°C/5 wk + 500°C/3 wk	0-1	0-1	0-1	0-1	1-7

* All samples were previously held at 900°C for 1 day and water quenched.

These dual treatments do not improve corrosion resistance, and it further appears that the times at the two temperatures are, to a degree at least, additive. The data indicate also that epsilon phase decomposes more readily into alpha than does gamma phase. This is reflected by the corrosion behavior of the 12 w/o Mo alloy. When annealed at 400°C for three weeks after having been gamma quenched, the 12 w/o Mo alloy used in this investigation remained in test in 343°C water for more than 56 days. When identical material was annealed at 550°C for five weeks to produce a mixture of gamma and epsilon and then annealed at 400°C for three weeks, the corrosion life in 343°C water was less than one day. This is substantiated by X-ray data which revealed more alpha in samples having the latter treatment. Additional double heat treatments involving other combinations of the temperatures 400°, 500°, and 550°C are being investigated.

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The study of the effect of transformation upon corrosion resistance is continuing with emphasis on temperatures below 400°C. Annealing gamma quenched alloys at 343°C has, in the case of 6 and 7 w/o Mo alloys, produced marked increases in corrosion resistance. As may be seen below, these alloys in the gamma quenched condition have a corrosion life of less than one day in 343°C water while the same alloys after additional annealing at 343°C for 5 weeks have a corrosion life of 7 to 14 days. The increase in corrosion resistance is accompanied by the appearance of a "strained alpha" structure and a marked hardness increase.

TABLE VIII
Effects of Low Temperature Aging

Treatment	6 w/o Mo					
	X-ray Structure	Hardness DPH	Corrosion Life in 343°C Water	Structure	Hardness DPH	Corrosion Life in 343°C Water
900°C, 24 hr, WQ	γ	180	< 2 hr	γ	260	0-1 day
900°C, 24 hr, WQ; 343°C, 1 week	γ + "Distorted α"	509	> 7 days	γ	268	0-1 day
900°C, 24 hr, WQ; 343°C, 3 weeks	"Distorted α"	549	> 1 day	"Distorted α"	513	> 1 day
900°C, 24 hr, WQ; 343°C, 5 weeks	"Distorted α"	554	7-14 days	"Distorted α"	530	7-14 days

Annealing of gamma quenched alloys containing 9 to 15 w/o Mo at 343°C for 5 weeks has no apparent effect upon corrosion resistance. It thus appears that at least for these times, "strained alpha" is not a low temperature transformation product in alloys containing 9 or more percent molybdenum.

In view of the beneficial effect of low temperature aging upon the corrosion resistance of 6 and 7 w/o Mo alloys, additional samples have been treated for 5 weeks at 343°C followed by transformation at 400° and 500°C to determine the stability of the low temperature transformation product. Data are not yet available.

A large number of ternary alloys are now in test, but no effects other than those previously reported concerning niobium and platinum in slowing down the decomposition of gamma U-Mo have been found. Table IX is a continuation of the data reported in an earlier report showing additional testing times.

TABLE IX

Corrosion Data for Ternary Alloys

<u>Time for Discontinuous Failure in 343°C Water</u>				
<u>Composition w/o</u>	<u>900°C/24 hr/WQ</u>	<u>400°C/1 wk</u>	<u>400°C 3 wk</u>	<u>400°C/5 wk</u>
7.5 Mo 1.5 Nb	98-112	98-112	42-56	0-1
9 Mo	14-28	1-7	1-7	0-1
12 Mo 1.5 Nb	14-28	56-70	56-70	42-56
13.5 Mo	14-28	14-28	28-42	7-14
15 Mo 4.8 Pt	> 238	> 210	> 210	70-84
6 Mo 1 Pt	0-1	7-19	0-1	0-1
8 Mo 1 Pt	42-56	> 70	7-14	1-7
12 Mo 0.5 Pt	56-70	56-70	> 56	> 56
9 Mo 1 Pt	28-42	14-28	7-14	-

Mechanism of Corrosion

Because of the poor corrosion resistance of the diffusion layer materials in Zircaloy clad uranium alloy fuel elements, a study has been made to determine the growth rate of these layers at various temperatures. The following diffusion couples were used:

unalloyed U	vs	crystal bar zirconium
U - 12 w/o Mo	vs	Zircaloy-2
U - 10 w/o Nb	vs	Zircaloy-2
U - 12 w/o Mo	vs	Zircaloy-2, with tantalum barrier layer
U - 12 w/o Mo	vs	Zircaloy-2, with molybdenum barrier layer
U - 3.8 w/o Si	vs	Zircaloy-2

These couples were in the form of extrusion clad rods, with the exception of the unalloyed uranium-crystal bar zirconium couple which was prepared by vacuum induction melting of uranium in a crystal bar crucible. The initial layer thickness was 0.0007 in. $\pm$ 0.0003 in. except in the case of those couples having barrier layers.

The total diffusion layer thickness has been plotted as a function of time for the couples Zr vs. U, Zircaloy-2 vs. U-12 w/o Mo, and Zircaloy-2 vs. U-10 w/o Nb after diffusion at 800° and 900°C. The diffusion layer thickness is proportional to the square root of time. The rate of growth of the diffusion layer is essentially independent of the composition of the partners of the couple and the microstructures of the diffusion layers are similar. The diffusion layer thickness for the Zircaloy-2 vs. U-3.8 w/o Si couple diffused 72 hours at 800°C is one-third that for the couples consisting of Zircaloy-2 vs. gamma phase alloys.

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The diffusion couples discussed above are being corrosion tested in 650°F water. While the exposure times are too short to lead to failure, it is apparent that the corrosion rate of the Zircaloy-2 vs. U-10 w/o Nb samples is independent of the thickness of the diffusion layer, while that of the Zircaloy-2 vs. U-12 w/o Mo samples increases with increasing diffusion layer thickness. Longer exposure times are required to verify this observation.

Zirconium Alloy Development

1. Zircaloy-3 Development Program. The alloys of the program set up by the Zirconium Alloy Corrosion Committee and detailed in WAPD-MM-259 have been in corrosion test for periods up to 142 days. The additional testing time has confirmed the trends noted in WAPD-MRP-46.

2. Zirconium Alloy Corrosion Committee Meeting No. 6. A report on the meeting of the committee held at Bettis Site on May 17, 18, has been written (WAPD-MM-448).

3. Additional testing time has confirmed the trends noted in previous reports for the following alloys:

- a. Zircaloy-2 containing beryllium
- b. Zircaloy-2 prepared from crystal bar zirconium
- c. Zirconium-cobalt binary alloys
- d. Zirconium iron, nickel or chromium binary alloys
- e. Zircaloy-2 containing manganese

Mechanism of Zirconium Corrosion

Work has been started on the investigation to determine the effect on corrosion resistance of the amount and distribution of precipitated phase in zirconium binary alloys with iron, nickel, or chromium. Specimens which have been annealed in the alpha plus compound field for varying periods of time to obtain different concentrations of the insoluble secondary phases ($ZrCr_2$, Zr_2Fe_3 or Zr_2Ni) are in corrosion test at 680°F and 750°F. The amount of precipitated second phase is being determined metallographically.

Specimens from a zirconium-10 w/o molybdenum alloy are being heat treated to determine the effect of structure upon corrosion resistance. The corrosion behavior will be determined for retained beta and for various stages of the transformation of this phase to the equilibrium phases, alpha and $ZrMo_2$.

Samples of Zircaloy-2, cathodically hydrided at 100°C in a buffered phosphate solution (pH 7) with a current density of 0.3 amp/cm², are being corrosion tested in 680°F water and 750°F steam to determine the role of surface hydride upon the corrosion rates and breakaway time. After 14 days exposure, no effect was noted on the specimens in 750°F steam; however, weight losses ranging from -0.12 to -0.69 mg/dm² were observed on the specimens tested in 680°F water. The appearance of the sample after corrosion test was dark brown. This test is continuing to determine if the effects noted are real.

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II-B-3 (Contd.)

The amount of hydrogen picked up by Zircaloy-2 during corrosion test increases with increasing exposure time; however, the ratio of observed hydrogen content to theoretical hydrogen content decreases with increasing exposure time. Specimens of dehydrided Zircaloy-2 are being prepared to determine the role of initial hydrogen content on the hydrogen pickup.

The effect of pre-oxidation of crystal bar and Zircaloy-2 in oxygen at 400°C upon subsequent corrosion in steam at 400°C is planned. The materials will be alpha annealed in vacuum prior to oxidation for various lengths of time at 400°C. If hydrogen pickup is the cause of breakaway, pre-oxidation should delay breakaway until a larger total weight gain is achieved.

A study of the effect of the surface-to-volume ratio on breakaway time is underway, with the idea that the concentration of hydrogen picked up at any given time during corrosion should be lowest in the sample having the smallest surface-to-volume ratio; and, if hydrogen concentration is a factor in breakaway time, should fail after longer exposure time than samples having larger surface-to-volume ratios. Corrosion tests in 680°F water and 750°F steam do not indicate any differences after 44 and 42 days between specimens 30 mils, 1/16 in., 1/8 in., 1/4 in., and 1/2 in. thick. Additional testing time is required to definitely establish this result. A lack of dependence of breakaway time upon specimen thickness would indicate either that hydrogen is not associated with the breakaway phenomenon or that hydride forms in the reaction zone without extensive diffusion into the interior of the sample.

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1. Non-Core Materials (53a11) J. G. ChristDevelopment of Type 304 Stainless Steels Containing Boron

A program is in progress to evaluate the fabricability and properties of type 304 stainless steels with 0.5 to 2.0% boron for use as control rod material. Experimental heats have been made with the boron added as ferroboration and also as 85% pure boron powder.

Five pound heats (ferroboration addition) with 0.42 to 1.70% boron were forged from 1-1/2 in. square bars to 3/4 in. square bars at temperatures of 2050°F to 1950°F. Tensile specimens from these bars, tested in the as-forged condition, fractured at the gage marks, indicating high notch sensitivity. With 0.42% boron the reduction in area was 17.2% and with 1.70% boron it was 6.0%.

Substitutes for Microbrazing

In order to obtain a brazing alloy better than microbrazing for the brazing of types 304 and 317 stainless steel materials as stated in the last progress report (WAFD-MRP-46), test assemblies have been made to determine brazing clearances, heat gradients, and different methods of brazing. Manual torch, automatic torch, and furnace brazing have been used as a means of heating. The fluxing has been done by controlled atmosphere during furnace brazing and high temperature fluxes during torch brazing. Successful brazes have been made with gold; however, other alloys with less or no gold are being investigated. The testing, as planned, will include corrosion and stress corrosion work of brazed joints, operating temperature tensiles of joints both in tensile and shear, and in impact tests. A program covering the entire work is being prepared and will be submitted in the next progress period.

Pressure Vessel Material

A meeting was held with Combustion Engineering, Inc., to discuss materials for the reactor vessel. A program is being prepared to determine the physical properties, including thermal fatigue of both A212 and A302 carbon steels.

2. Carbon Steel Program (55b14) P. CohenBettis Site Activities

Preliminary operation was begun on one carbon steel loop (A-1) and construction was essentially completed on a second carbon steel loop (A-2). Conditions for initial operation of A-1 were: temperature 600°F, pressure 2000 psig, 10 ppm NH₃, 100 cc H₂ per kilo, 100 cc N₂ per kilo. No samples were exposed and a fine screen was inserted to trap welding scale and other debris. Three short cycles of operation were completed. Several grams of weld scale were found after the first cycle; none after the last two. Samples of SA-212 carbon steel and low manganese, titanium bearing experimental carbon steels have been loaded into A-1 for long time tests, under the conditions given above.

Both loops, A-1 and A-2, are equipped with piston pressurizers. It is hoped that this will permit measurement of loop corrosion by hydrogen evolution despite the high initial hydrogen charge. Conditions of test for loop A-2 have not yet been established. At the Carbon Steel Committee meeting at Knolls Atomic Power Laboratory, August 11, the decision was reached to operate loop A-2 without any inhibitor, unless by the time it was ready for test, an inhibitor was discovered that was superior to any so far examined and could appropriately be tested.

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II-C-2 (Contd.)

Investigations are being continued by Jones & Laughlin Steel Corporation on the physical properties of carbon steels in which titanium has been substituted for the greater part of the manganese. Such steels with approximately 0.15 per cent manganese have been found to meet the tensile limits for SA-212 steel. From the standpoint of steelmaking, rolling characteristics, and limited corrosion data, it may be concluded that low manganese steels may be a preferred form of carbon steel for reactor structures. At equal corrosion rates these steels would contribute from 1/5 to 1/10 the manganese to the water that would come from SA-212 steel. Samples of this type of material have been distributed to interested laboratories for corrosion testing, and weld test data are being obtained.

Babcock & Wilcox Subcontract

Under subcontract to the Westinghouse Atomic Power Division, Babcock & Wilcox Co. is operating two carbon steel loops for corrosion tests in the carbon steel program. Both loops are operating at high pH with lithium hydroxide and degassed feedwater, one with 500 cc per kilo of hydrogen, the other with only the hydrogen resulting from the corrosion reactions. Initial tests of approximately 500 hours have been completed in both loops (Babcock & Wilcox Report No. 5081) and the results showed very little difference in the two loops.

Alternate Structural Materials - Corrosion Inhibitors

The evaluation tests of corrosion inhibitors for A 212B and 5% Cr-1/2% Mo steel are continuing. Corrosion specimens are 4 in. x 1/8 in. discs which are rotated at a peripheral velocity of 2 fps in the test solution. Baffles provided inside the autoclave tend to minimize "solution drag" as the specimens are rotated. All specimens are annealed and have ± 63 microinch machined finishes. Specimens are examined and weighed at the end of each 7 day period. Specimens will be descaled with sodium hydride, and weighed upon the termination of each test (28 days).

Inhibitor concentration is maintained throughout the test period. All tests are performed with 0.05 ppm O_2 , at 600°F, unless specified otherwise. Tests completed and in progress are reported below:

TABLE X
Autoclave Tests of Alternate
Structural Materials with Corrosion Inhibitors
600°F - 80 Days

<u>Test</u>	<u>Inhibitor</u>	<u>Concentration</u>	<u>Status</u>
R-3	Na_2HPO_4	5,000 ppm	Completed
R-4	H_2SiO_3	500* ppm	Completed
R-5	$NH_3-H_3BO_3$	0.1 M each	Completed
R-6	H_2	500 m./l	Completed

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II-C-2 (Contd.)

TABLE X (Contd.)

<u>Test</u>	<u>Inhibitor</u>	<u>Concentration</u>	<u>Status</u>
R-7	KTcO ₄	10 ppm	Initiated**
R-8	Li ₂ WO ₄	3,000 ppm	Completed
R-9	Li ₂ HPO ₄	10,000 ppm	Completed
R-10	Li ₂ HPO ₄	~ 2,000 ppm	Completed
R-11	degassed, distilled	H ₂ O	Completed
R-12	Li ₂ HAsO ₄	5,000 ppm	Third 7 day period
R-13	Li ₂ HPO ₄ (~2,000 ppm)	O ₂ (1 ml/l)	Second 7 day period

* Quantity of precipitated silicic acid added to water; concentration in solution less than this figure.

** Test interrupted at the end of 48 hours by mechanical failure of the rotating mechanism.

More detailed information concerning these tests will be given in an interim report to be issued in September.

Results of these tests are summarized briefly below:

R-3 Samples grey, apparently no pitting has occurred. Weight gains of 150-200 mg/dm² probably due in part to phosphate deposition.

R-4 Weight losses 65-115 mb/dm². No pitting, grey-brown mottled appearance.

R-5 Weight gains small (< 20 mg dm²); some pitting noted. This indicates incomplete inhibition and might possibly be prevented by using more concentrated solution or varying the NH₃/H₃BO₃ ratio.

R-6 Weight gains ~10 mg/dm²; slight pitting on one specimen.

R-7 After 48 hours the Tc⁹⁹ had decreased from ~8,000 counts per minute per ml of solution to ~40 counts per ml. Samples were rusty. High activity was found on the sample surfaces. Test is being continued to demonstrate whether or not the deposited layer will act as an inhibitor.

R-8 Weight gains ~100 mg/dm² - no evidence of pitting. Weight gain may be due to inhibitor deposition.

R-9 High weight gains (as high as 500 mg/dm²) largely due to inhibitor decomposition and deposition. Inhibitor has apparently leached Cr from stainless steel autoclave and/or test specimens.

R-10 Weight gains from 40-80 mg/dm² again probably due to inhibitor deposition. No pitting observed.

R-11 Blank - degassed water - samples are grey-black, blistered, and spotted.

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II-C-2 (Contd.)

The test with Li_2HPO_4 is being repeated with the addition of ~ 1 ml $\text{O}_2/\text{kg H}_2\text{O}$ since the presence of oxygen is believed necessary for most effective inhibition. Oxygen apparently participates in protective film repair and reduces phosphate consumption.

Pickled specimens of A212b and 5% Cr-1/2% Mo steels have been descaled by the sodium hydride technique in an effort to determine the magnitude of the blanks which may be expected for this method of descaling. The results are as follows:

0.9% Sodium Hydride Bath

<u>Time of Immersion</u>	<u>A212b</u>	<u>5% Cr-1/2% Mo</u>
5 min	2 mg/dm ²	1.7 mg/dm ²
10 min	0.6	1.8
15 min	0.6	0.6

1.8% Sodium Hydride Bath

5 min	1.0 mg/dm ²	0.2 mg/dm ²
10 min	1.0 mg/dm ²	0.2 mg/dm ²
15 min	0.5 mg/dm ²	0.4 mg/dm ²

The blanks are less for the higher sodium hydride concentration and indicate that descaling in a .2% bath should be quite satisfactory. The reason for the lower weight losses for longer descaling time is not known.

Rotogenerator Studies

The room temperature results for KTcO_4 with A212b are given below:
 KTcO_4 - 20 ppm - peak-to-peak voltage - 3.0 mv
100 ppm - peak-to-peak voltage - 0.4 mv
distilled water - peak-to-peak voltage - 35.0 mv

The 0.4 mv is comparable to the results obtained with the best inhibitors tested thus far. However, a deposit was found in the rotogenerator cell which contains Tc^{99} activity.

Synthetic Crud Investigation

No activity during this period.

D. Chemistry Development

1. Fuel Element Testing and Fission Product Removal (55b12) P. Cohen

The third fission product removal test in loop R-2 begun on July 14 (as reported in WAPD-MRP-46) is continuing. Gross β activity in the loop water fluctuated between 2 and 21 counts per minute per milliliter (c/m.ml) averaging 11 c/m.ml, while the LiOH ion exchanger varied from 0.5 to 12 c/m.ml averaging

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II-D-1 (Contd.)

4.3 c/m.ml, for an average ion exchanger decontamination factor of 2.6. In general, fluctuations in loop activity paralleled those in ion exchanger effluent activity. Samples were counted at 1.9 per cent geometry using a halogen tube having one-half the efficiency of a standard Geiger tube. The micrometallic filter in the loop is in a hot bypass line, separate from the ion exchanger circuit.

Eight stainless steel coupons have been removed from the semi-static autoclave in the loop for decontamination testing, using various decontaminating solutions. Results are tabulated below:

Date Removed	Material	Gross β counts, c/m(1)		Decontami- nation Factor	Decontaminant (2)
		Before	After		
July 23	304 stainless	7139	2157	3.3	CrO ₃ (1 gm/l)
	"	7852	2115	3.7	"
	"	7430	2902	2.6	K ₂ CrO ₄ (1 gm/l)
	"	7219	2864	2.5	"
	347 stainless	12286	2177	5.6	CrO ₃ (1 gm/l)
	"	5822	2015	2.9	"
	"	5710	2294	2.5	K ₂ CrO ₄ (1 gm/l)
	"	8767	3407	2.6	"
July 30	304 stainless	7285	3465	2.1	Na Citrate (20 gm/l) plus NaOH (0.04 gm/l)
	"	3543	1631	2.2	"
	"	5134	945	5.4	HIO ₄ . 2 H ₂ O (20 gm/l)
	"	3921	779	5.0	"
	347 stainless	4931	825	6.0	HIO ₄ . 2 H ₂ O (2 gm/l)
	"	4761	836	5.7	"
	"	3768	1740	2.2	Citric Acid (5 gm/l)
	"	3255	1638	2.0	"

(1) Counted with 8.4 per cent geometry, using a halogen tube having one-half the efficiency of a standard geiger tube. Sample area 0.0625 dm². Both sides of each coupon were counted.

(2) Each decontaminant was circulated past the coupon for two hours at 190-195°F.

It is apparent that periodic acid gives the highest decontamination factors, 1.5 to 3 times greater than those for the other solutions. Concentration of the periodic acid appears to have little effect, the more dilute solution even giving a slightly higher decontamination factor.

Plans have been formulated to greatly accelerate the work on this project by employing additional facilities released by completion of tests on other projects. Erection of T-loop for slug rupture studies will be started early in September.

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II-D-1 (Contd.)

Fission Product Activity - Dynamic Loop Tests

The radiochemical analysis of the activities of five fission products has been completed for (1) the total water content of the loop, (2) the crud (0.1067 g) collected on the sintered metal filter, (3) the ion exchange resin and (4) a type 304 stainless steel coupon. The results are tabulated in Table XI. The data have been corrected for chemical yield and normalized to 100% geometry as of June 28.

One structural material coupon, type A212 carbon steel, was etched to remove the adsorbed fission products and chemically analyzed for five fission products. The samples are being counted to determine their activity and will be reported during the next period.

Table XI

Fission Product Activities* c/m

<u>Nuclide</u>	<u>Water</u>	<u>Crud</u>	<u>Resin</u>	<u>Type 304 Stainless</u>
40h La ^{140***}	1.94x10 ⁵	4.48x10 ⁴	1.09x10 ²	1.28x10 ⁴
12.8d Ba ¹⁴⁰	6.93x10 ²	2.72x10 ³	ø	3.60x10 ²
37d Nb ⁹⁵	2.19x10 ⁴	2.51x10 ⁵	6.38x10 ⁴	9.86x10 ³
65d Zr ⁹⁵	5.16x10 ⁴	2.17x10 ⁵	2.79x10 ⁴	3.25x10 ³
275d Ci ^{144#}	1.71x10 ⁵	7.14x10 ⁵	ø	1.16x10 ⁴

* All activities are corrected for decay to June 28 and, where required, are normalized to the same elapsed time from its parent.

** The total surface area of the coupon is 30 cm²

*** La fraction includes all rare earths except Ce

ø Nuclide could not be chemically separated

Ce fraction includes 275d Ce¹⁴⁴ and 32d Ce¹⁴¹

Phase I-B - Large Defects

The analysis of the crud (0.2968 g) and the solution from the decontamination using periodic acid has been completed. The results are found in Table XII.

The chemical separation of the five fission products has been completed for (1) the total water content of the loop, (2) the ion exchange resin and (3) one structural material coupon type 304 stainless steel. A positive determination of the activities awaits the elapse of two or more half-lives.

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II-D-1 (Contd.)

TABLE II

Fission Product Activities** c/m

<u>Nuclide</u>	<u>Crud</u>	<u>Periodic Acid Solution</u>
40h La ¹⁴⁰	1.94x10 ⁶	8.12x10 ⁶
12.8d Ba ¹⁴⁰	1.77x10 ³	3.81x10 ⁴
37d Nb ⁹⁵	1.26x10 ⁵	1.40x10 ⁶
65d Zr ⁹⁵	1.98x10 ⁶	2.13x10 ⁵
275d Ce ¹⁴⁴	7.22x10 ⁶	1.14x10 ⁵

*See footnotes for Table XI.

**These samples were counted with a hologen-filled Geiger tube which is 50% as efficient as an ordinary Geiger tube.

3. Chemical Control (56a12) Z. M. Shapiro

Thermal Stability Tests of Poison Solutions

The following tests were in progress during the period covered by this report.

<u>Run No.</u>	<u>Solution Composition</u>	<u>Temp (°F)</u>	<u>Static or Rocking</u>	<u>Status</u>
11	NH ₄ HB ₄ O ₇ at 0.16M H ₃ BO ₃ +NH ₄ OH to pH 9.2	600	Static	Continuing
12	NH ₄ B ₅ O ₈ at 0.10M H ₃ BO ₃	600	Static	Continuing
S-6	H ₃ BO ₃ at 0.15M, 0.25M, 0.40M	600	Rocking	Discontinued
S-8	NH ₄ HB ₄ O ₇ at 0.16M H ₃ BO ₃ + NH ₄ OH to pH 9.2	600	Rocking	Continuing

II-D-3 (Contd.)

Results from Run No. 11 showed no significant changes in concentration in the first 1000 hours. Results from Run No. 12 were not yet available at the time of writing of this report. Run No. S-6 operated from May 27 to August 12. This test was split into three periods, different concentrations being tested during each period. The results confirm the previously reported satisfactory stability of boric acid solutions. Run No. S-8 was interrupted by experimental difficulties after approximately 240 hours. No significant changes in concentration were observed during this initial period. The test has been restored to operation and is continuing. Further results were not available at the time of writing of this report.

Physical Properties of Poison Solutions

a. Vapor Pressure Lowering Apparatus

The operation of this experimental facility has been delayed by the installation of new solenoid stirrers for the autoclaves and by difficulties encountered in distilling properly degassed water into the apparatus. It is anticipated that the experiment will be in operation with 1.0M boric acid solution by the end of the period covered by this report. Data will be collected for 1.0M and progressively more dilute solutions during the next period. A considerable body of data must be accumulated before the calculation and interpretation of results can proceed.

b. Distribution Coefficient of Boric Acid between Liquid and Vapor

Boric acid is appreciably soluble in steam as well as in liquid water. Knowledge of the distribution coefficient, defined as the ratio of mole fraction boric acid in the vapor to mole fraction boric acid in the liquid, is necessary for interpretation of the vapor pressure lowering experiment and for estimation of the effect of bubble formation in the reactor core. The magnitude of the coefficient will also determine the feasibility of evaporation processes for recovery of boric acid and water from the chemical control system effluent. A rocking autoclave is being used in an experimental determination in the coefficient at saturation temperatures and pressures. Preliminary results indicate that the coefficient rises from about 0.005 at atmospheric pressure to 0.67 at 2000 psi, approaching unity at the critical point. The experiment is continuing.

Heat Transfer Properties

Over-all and film coefficients for heat transfer to poison solutions are being determined in a test on the Babcock & Wilcox Co. HT-R loop. Ammonium borate solutions, equivalent to 0.15M H_3BO_3 adjusted to pH 9 to 10 with NH_4OH , have been substituted for lithium tetraborate solutions in the test schedule. Results to date indicate no detectable difference between the heat transfer coefficients for water and for 0.15M boric acid solutions. A complete report on the test will be contained in a memorandum to be issued during the next period.

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II-D-3 (Contd.)

Waste Disposal

Brookhaven National Laboratory has conducted preliminary experiments using montmorillonite clays to remove mixed fission products in the presence of boric acid. Results show promise worthy of further investigation, but it is doubtful that the required decontamination factor (approximately 10^6) can be obtained by clay alone. Some resin "polishing" stage appears necessary at least for removal of ruthenium. For a single experiment using a clay column followed by an anion resin, a decontamination factor of 5×10^4 was obtained.

Brookhaven National Laboratory Test Loop

The out-of-pile section of the stainless steel irradiation facility at Brookhaven has been completely overhauled: the valves were modified, the pressurizer packing was changed, and a new water preparation system was added. In addition, the gas analysis apparatus was altered to enable the determination of nitrogen. It is expected that these changes will eliminate the difficulties previously encountered with this system. The unit is now being checked out with water. Results with borate solution will be reported during the next period.

The general design of the in-pile test section of the second irradiation facility has been changed to improve the solution flow pattern and to facilitate construction of a noble-metal lined system. Consequently, the schedule of loop construction and installation has been changed.

The second stainless steel loop (PC-2) is being tested prior to shipment to BNL. Parts for the carbon steel loop (PC-3) are being fabricated. Fabrication of the noble-metal lined loop (PC-4) will be completed as soon as necessary parts are obtained from the vendor.

MTR Test Loop

Immediate plans for a test loop operating in the L-42 position in the MTR have been abandoned. No design which provides the necessary uniform temperature has been devised and it is doubtful whether one could be developed soon enough to meet the current schedule.

The spatial distribution of gamma heating varies with shim rod position and is not known well enough to permit a direct design of a constant temperature system. Moreover, the magnitude of the gamma heating is so large in some areas that sufficient electrical heat cannot be provided at cold spots where the gamma flux may be low, as at the ends of the test section. The test section cannot be shortened without making the test time to reach steady state an appreciable fraction of the operating time between refueling shutdowns.

The smoothing effect of massive amounts of aluminum on the temperature distribution has been computed and found to be insufficient to provide a temperature range of less than 180°F .

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II-D-3 (Contd.)

Temperature ranges in excess of 50°F are undesirable, because of the temperature sensitivity of some radiation effects, and because the temperature rise of the coolant through the PWR core is less than 50°F.

A formal report on the above work is now in preparation and will be issued during the next report period.

Corrosion Testing in Chemical Control Solutions

Corrosion testing of structural materials in boric acid solutions is continuing. Testing with lithium hydroxide-boric acid solutions has been discontinued because of the precipitation of lithium metaborate at test conditions. Tests with ammonium borate have been substituted for the lithium borate tests. These tests will employ 0.1M H_3BO_3 /1 H_2O (at test temperature) and will be adjusted to pH 9.2-9.5 by addition of NH_3 (at room temperature). These tests will be performed in circulating systems at a temperature of 600°F.

Results from the following tests were obtained since the last report:

<u>Test No.</u>	<u>Solution</u>	<u>Gas</u>	<u>Temperature</u>	<u>Time</u>
21	0.1M H_3BO_3	degassed	600°F	60 days
22	0.1M H_3BO_3	100ml H_2 /kg H_2O	600°F	60 days
27	0.1M H_3BO_3	degassed	650°F	30 days
28	0.1M H_3BO_3	100ml H_2 /kg H_2O	650°F	30 days

The thirty day results for tests 21 and 22 were reported in WAPD-MRP-46. All tests were performed in the circulating systems. Results for the various materials are tabulated below.

A212b

Weight losses were not significantly greater in 60 days (10-30 mg/dm²) than in 30 days (tests 21 and 22); however, pitting was evident at the end of 60 days. Pitting occurred in tests 27 and 28.

Low Cr-Mo Alloy Steel

No pitting in tests 21 and 22; weight losses 10-30 mg/dm². Weight gains (approximately 20 mg/dm²mo) were registered in test 27. Weight losses (approximately 20-30 mg/dm²mo) occurred in test 28. Pitting was evident in test 28.

17-4 PH

Good appearance (temper film) in all tests. Weight changes low (<10 mg/dm²) and of mixed sign.

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II-D-3 (Contd.)

Monel

Weight changes low ($< 10 \text{ mg/dm}^2$) and of mixed sign. Surface temper film appeared in nearly all cases.

#304 Annealed, #304 Sensitized

Excellent appearance (temper film) in all tests except test 28 which gave some pitting. Weight gains low (approximately $\sim 10 \text{ mg/dm}^2$) and about the same for the annealed and sensitized material.

#347 Annealed, #347 Sensitized

Same as #304.

#410 Annealed, #410 Aged

Appearance not as good as #304 and #347; most specimens grey. No significant difference between the annealed and aged materials.

Inconel-X

Appearance good (temper film) weight changes between 6 and 20 mg/dm^2 and of mixed sign except for a specimen in test 21 which incurred a weight gain of 40 mg/dm^2 .

Zircaloy-2

Excellent; glossy black. Weight gains were low (3 to 7 mg/dm^2) for pretested samples and were as high as 20 mg/dm^2 for new samples.

Chrome Plate on #347

Appearances good (temper films) but weight losses were $100\text{--}200 \text{ mg/dm}^2$ in all cases.

Stellite 3, Stellite 6

Excellent - temper film - weight changes less than 10 mg/dm^2 .

Titanium

Grey appearance; weight gains of from 3 to 17 mg/dm^2 occurred in all tests.

73-30 Cu-Ni

Appearance good - temper films. High weight losses (350 to 600 mg/dm^2) in degassed tests; weight losses less by factors of 10 to 20 in hydrogenated tests.

II-D-3 (Contd.)

It is recognized that solution concentration gradients are likely to occur in any complex system; e.g., in valves, in weld crevices, or in any location which may temporarily isolate the solution from the main stream. In order to determine the effect concentration cells formed in this manner might have upon corrosion, room temperature concentration cells of 0.1M H_2BO_3 vs 0.9M H_2BO_3 were prepared using types 304, 347 and 410 stainless, and 2-1/4 Cr-1/2 Mo steel coupons as electrodes. The cells were operated degassed for 30 days. Weight changes of the steel electrodes were determined and are as follows:

1. A212b and 2-1/4 Cr-1/2 Mo suffered weight losses as high as 385 mg/dm²mo.
2. The stainless steels suffered weight losses as high as 100 mg/dm²mo.

These results are inconclusive and further work is indicated. It is believed that in practice the concentration gradient will not be as great as that employed in the test. Tests will be performed at a higher temperature and lower concentration gradients.

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II-E-1

E. Irradiation Effects

1. Irradiation Effects on Core Materials (53b31) L. S. Castleman

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

Following approval of the test proposal by the Chalk River Pile Branch, the specimens were scheduled for charging into the reactor in early September.

Hot Water Irradiation of PWR Fuel Materials (CR-WAPD-Met 4)

The decision was made to discontinue the irradiation test as proposed under the above experiment number, since the out-of-pile corrosion resistance of the samples had been impaired by heat treatment and consequent formation of excessive diffusion layer at the core-clad interface. A new irradiation test, designated Experiment No. CR-WAPD Met 4A, will be run on more suitable samples.

Hot Water Irradiation of PWR Fuel Materials (CR-WAPD-Met 4A)

Fourteen samples of U-10.5 w/o Mo and U-12 w/o Mo alloys about 4 in. long and containing 3.6 w/o U²³⁵ and extrusion bonded to Zircaloy-2 cladding are to be irradiated for four months in the X-1 hot water loop in the Chalk River Reactor. Half of the samples are to contain end caps bonded to the core by resistance butt welding, and the other half will have unbonded end caps. Seven of the U-Mo alloy samples will contain defects in the form of one 0.040 in. diameter hole per sample drilled through the cladding into the core (hole is to be located midway between the end caps); the remaining samples will be irradiated without defects. Control defected samples will be given a prior 30 day corrosion test in 650°F water at saturated steam pressure in the as-extruded condition to evaluate the corrosion resistance of the presumably identical in-pile samples. If the corrosion resistance of the controls is satisfactory, the samples to be used in-pile will be given a short corrosion test in the undefected condition; then, defects will be drilled into the samples, and they will be corrosion tested in-pile. The irradiation will start on October 4.

It is also planned to include four rod samples of UO₂ cold pressed and sintered to a minimum of 80% of theoretical density and clad with Zircaloy-2 in the in-pile test, subject to approval of the Canadians. Two UO₂ samples are to contain 0.005 in. diam holes (one per sample) drilled through the cladding to the core; the other two will be in the undefected condition.

Hot Water Irradiation of Enriched Uranium Dioxide (CR-WAPD-Met 5)

The future of this experiment, which was originally scheduled to follow Experiment No. CR-WAPD-Met 4, has now become contingent upon the results obtained in Experiment No. CR-WAPD-Met 4A.

II-E-1 (Contd.)

Constrained Uranium (MTR-WAPD 4-Met 4)

Three specimen container units have been assembled. Insertion of sodium-potassium alloy into the containers has been delayed due to the diversion of effort to Experiment No. MTR-WAPD 12-Met 5.

Constrained Uranium (MTR-WAPD 4-Met 5)

Fourteen samples of natural uranium clad with (but unbonded to) base annealed zirconium machined to 0.020 in. and 0.040 in. wall thickness over one in. gage length were irradiated for six weeks in the MTR at a maximum thermal neutron flux of 3×10^{14} neutrons/cm²/sec. The cladding material was slotted longitudinally in the gage length region to permit circulation of the sodium-potassium alloy heat transfer medium around the uranium specimen. The uranium had been hot rolled at 300°C so as to give it maximum in-pile dimensional instability. Examination of the specimens showed that the cladding had ruptured in every case as a result of uranium growth. Calculations are being made to determine the stress necessary to cause cladding failure. They will be complete by September 7.

Irradiation of U-12 w/o Alloy (MTR-WAPD 7-Met 1)

The simultaneous corrosion and irradiation tests of U-Mo alloy samples in the Van de Graaff generator (see WAPD-MRP-45) have been completed and the samples are being evaluated. Evaluation will be complete by September 1.

MTR Process Water Irradiation of Clad U-Mo Alloys (MTR-WAPD 11-Met 1)

Information obtained to date is summarized in Tables XIII and XIV.

Four of the eight irradiated and defected samples have successfully survived a 44 day corrosion test in 650°F water; the remaining four failed at some time during the first 32 days of testing, but in no case could the failure be attributed to the presence of the drilled hole in the cladding. Unfortunately, during the welding of one of the end caps, the molten Zircaloy-2 at the weld had become diluted with uranium and molybdenum from the core; as a result, the corrosion resistance of the weld metal had become impaired. This condition was not determined prior to irradiation because the samples had only been given a two day corrosion test at 600°F. This is not a sufficiently severe test to show the condition. Apparently, irradiation greatly increased the corrosion rate of the contaminated weld metal.

It is encouraging to find that some of the irradiated specimens have survived longer than the controls. Two of the five as-extruded control samples failed in the 33-44 day test period, and in their case failure is definitely attributable to the presence of the defect. The control samples were fabricated after the weld metal contamination problem had been recognized. Special precautions were taken to prevent weld metal dilution during welding. It may be tentatively concluded that the variability in the sample corrosion behavior was sufficient to overshadow any harmful effects that irradiation might have had.

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TABLE XIII

Irradiated U-10.5 w/o Mo Alloy Corrosion Test Results

Sample Number	Heat Treatment	Exposure (Mwd/t)	Days to Failure	Remarks
B2	None after	3800	23-32	{ Failure caused by corrosion of contaminated weld metal
B3	extrusion	3800	15-22	
B4	"	3600	15-22	
B7	"	3400	O.K. after 44 days	Weld metal corroding
B1	"	3300	O.K. after 44 days	Weld metal corroding
B9	"	2900	O.K. after 44 days	Weld metal corroding
B8	"	2500	8-14	Failure caused by weld metal corrosion
B10	"	2500	O.K. after 44 days	Weld metal corroding
<hr/>				
BC1	700°C for	None	O.K. after 44 days	
BC2	40 minutes	"	O.K. after 44 days	
BC3	"	"	O.K. after 44 days	
BC4	"	"	O.K. after 44 days	
BC5	"	"	23-32	Failed by bulging at defect
<hr/>				
BC6	None after	"	33-44	
BC7	extrusion	"	O.K. after 44 days	
BC8	"	"	33-44	
BC9	"	"	O.K. after 44 days	
BC10	"	"	O.K. after 44 days	

II-F-1 (Contd.)

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TABLE XIV

U-12 and 13.5 w/o Mo Alloy Results

Sample Number	Composition	Heat Treatment	Exposure (Mwd/t)	Days to Failure	Remarks
C1	12 w/o Mo	800°C for	3800	15-22	Swelling at base of end plugs usually observed prior to failure
B6	"	4 hours	3800	5-7	
C2	"	"	3600	5-7	
C3	"	"	3400	2-4	
B5	"	"	3300	2-4	
C4	"	"	1000	0-1	
C5	"	"	1000	5-7	
CC1	12 w/o Mo	800°C for	None	23-32	
CC2	"	4 hours		8-14	
CC3	"	"		8-14	
CC4	"	"		15-22	
CC5	"	"		15-22	
CC6	"	750°C for		O.K. after 44 days	Slight swelling perceptible.
CC7	"	1/2 hour		8-14	Swelling at base of end plugs
CC8	"	"		5-7	usually observed prior to
CC9	"	"		15-22	failure.
CC10	"	"		23-32	
D1	13.5 w/o Mo	800°C for	3800	2-4	
D4	"	4 hours	3800	23-32	
D5	"	"	3400	23-32	Cladding split near defect

II-E-1 (Contd.)

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II-E-1 (Contd.)

The results listed in Table XIV indicate the deleterious effects of heat treatment upon the corrosion resistance of the clad U-12 and 13.5 w/o Mo alloy samples. Five of the ten unirradiated control samples were heat treated for 4 hours at 800°C subsequent to extrusion; one failed during the 23-32 day test period, and the other four failed in 22 days or less. The remaining five unirradiated control samples were heat treated for 1/2 hour at 750°F; one remained intact after 44 days of testing, one failed in the 23-32 day testing period, and the other three in 22 days or less. All but one of the irradiated samples, on the other hand, failed in less than 8 days and the exception failed during the 15-23 day testing period. One sample of the 13.5 w/o alloy failed within 8 days and the other two within 23 days. Although weld metal corrosion of varying degrees was observed on all specimens, except No. C-4, which failed in less than one day, the corrosion was not sufficiently advanced to attribute the cause of failure to its occurrence. Most of the samples showed signs of preferential corrosion along the core-clad interface and bulging at the bases of the end caps. Thus, it appears that irradiation aggravated the inherently poor corrosion resistance of the core-clad diffusion layer present in the samples as a result of heat treatment. Irradiation, therefore, is especially injurious to uranium-molybdenum contaminated - or interdiffused - Zircaloy.

MTR Process Water Irradiation of Clad U-Mo Alloys (MTR-WAPD 11-Met 2, MTR-WAPD 11-Met 3)

Visual examination of the irradiated Zircaloy-2 clad samples of U-9 w/o Mo, U-10.5 w/o Mo, U-12 w/o Mo, and U-13 w/o Mo alloy has shown no sign of dimensional instability.

In view of the post-irradiation corrosion results obtained from experiment MTR-WAPD 11-Met 1 on U-Mo alloy specimens with an excessively thick diffusion layer, the samples irradiated under the above two experiments will not be corrosion tested unless the thickness of the diffusion layer is not found to be excessive. Cross sections of several specimens will be examined metallographically to determine the extent of the diffusion layer. Also, samples taken from specimens subjected to the highest burnup (about 6000 Mwd/t) will be checked metallographically for irradiation-induced phase transformation.

MTR Process Water Irradiation of Clad U-Mo Alloys (MTR-WAPD-11-Met 4)

Zircaloy-2 clad samples of U-9 w/o Mo, U-10.5 w/o Mo, U-12 w/o Mo, and U-13.5 w/o Mo alloys, containing 5 w/o U²³⁵ were discharged from the MTR on August 16, after an irradiation of sixteen weeks (maximum exposure estimated to be about 9,500 Mwd/t). Upon their return, the samples are to be treated similarly to those of Experiments No. MTR-WAPD 11-Met 2 and Met 3.

MTR Process Water Irradiation of Clad U-Mo Alloys (MTR-WAPD 11-Met7)

Twenty-one natural U-12 w/o Mo alloy samples and three natural U-10.5 w/o Mo alloy samples, extrusion clad with Zircaloy-2, were charged into the MTR on August 18. The samples are being irradiated in the as-extruded condition so that the diffusion layer should be of minimum thickness. Measurements

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II-E-1 (Contd.)

of dimensional stability will be made after irradiation, and the samples will be corrosion tested in static water at 650°F after defects are drilled into the cladding.

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

No changes in dimensions were observed for the seven samples of gamma quenched U-10.5 w/o Mo alloy and the seven samples of U-12 w/o Mo alloy which had been irradiated to an estimated maximum exposure of 3000 Mwd/t. Post-irradiation measurements of electrical resistivity are now being made.

Irradiation of Unclad U-Mo Alloys (MTR-WAPD 12-Met. 2, MTR-WAPD 12-Met. 3, MTR-WAPD 12-Met. 4)

Samples of U-10.5 w/o Mo, U-12 w/o Mo, and U-13.5 w/o Mo alloy containing 5 w/o U^{235} , were canned in sodium-potassium alloy filled capsules and charged into the MTR on August 18. Samples are being irradiated for periods of twelve and twenty-four weeks in a thermal neutron flux of 5×10^{15} neutrons/cm²/sec. The effect of irradiation upon the physical, mechanical, and corrosion properties of U-Mo alloys in various heat treated conditions will be determined after the samples are removed from the MTR.

UO₂ Irradiation Program (MTR-WAPD 14-Met. 1)

The apparatus for measuring the fission gas released by the samples has been assembled and gas pressure determinations are now being made. It is expected that results will be available by September 1. Following this, one sample containing a defect is to be corrosion tested in 650°F water; also, the others will be cross sectioned and the state of aggregation of the UO₂ will be determined.*

Irradiation of Natural UO₂ Pellets (MTR-WAPD 14-Met. 2)

The capsules containing the pellets of UO₂ clad with Zircaloy-2 were delivered to Bettis Site on August 13. The capsules will be opened, and the specimens examined for dimensional stability as soon as the equipment now being used for Experiment No. MTR-WAPD 14-Met. 1 becomes available. Subsequently, the pellets will be punctured, fission gas release will be measured if possible and the state of aggregation of the UO₂ will be determined.

Irradiation of Unclad U-3.8% Si Alloy (MTR-WAPD 12-Met. 5)

Seven specimens of U-3.8 w/o Si alloy were inserted into the MTR on July 28. They will be discharged on September 8. The U-3.8% Si alloy was induction melted and extrusion bonded to Zircaloy-2 cladding at 1700°F to produce a rod of 0.304 in. O.D. The cladding was stripped from the rod and specimens 3/16 in. diam x 2 in. long were machined. The specimens were heat treated at 800°C for 91 hours to form the epsilon phase. A summary of the pre-irradiation test data is shown below:

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II-E-1 (Contd.)

<u>Property</u>	<u>As-Extruded</u>	<u>Epsilonized</u>
Density (g/cm ³)	15.34 ± 0.01 (4)*	15.51 ± 0.01 (4)
Electrical Resistivity (microhm-cm)		
at 25°C	57.1 ± 0.5 (5)	55.5 (1)
at -196°C	31.1 ± 0.8 (5)	23.5 ± 0.5 (5)
Temperature Coefficient of Electrical Resistivity Between -100°C and 50°C (°C ⁻¹)	————	0.0016 (1)
Hardness (Vhn)	332 (1)	213 ± 10 (8)

* Numbers in parentheses refer to the number of specimens which were measured.

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

In view of the failure of the samples for the first proposed test in the X-1 loop at the Chalk River reactor, a revised interim program was submitted and has been approved by the Canadians. This program calls for a chemical poison test in loop X-1, a hydrogen inhibitor test in loop X-2, and an ammonia inhibitor test in loop X-3. The status of the loops and test preparations are presented below.

Loop X-1

All necessary work for the completion and final trial operation is now in progress for a September 7 installation. The final installation of the delayed neutron monitor will depend upon the performance of four commercial tubes now under test. Out-of-pile operation began on August 24 using the in-pile test section. Conditions have been adjusted to establish various hydrogen concentrations and degasifier operations. Boric acid will be added on August 31 to establish out-of-pile conditions. The wiring, shielding and adaptor installations have been completed.

Loop X-2

The system is now operating out-of-pile. The in-pile section will be connected during the shutdown period following the removal of X-1. Shielding of the loop will be done during the out-of-pile test. The earliest date for in-pile operations is October 18.

Loop X-3

Complete installation of this system has been delayed due to the drain on manpower required for the other two loops. The main loop is 90% complete. The main loop frame is ready to go into the swimming pool. The sixth and final pump will be shipped from Bettis Site on September 1.

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II-E-2 (Contd.)

Test Section Removal

The equipment for removal of the test section from the reactor has been fabricated and received at Bettis Site. The shielded shipping flasks have been checked for cracks and voids in the lead in the Physics Hot Lab at Bettis Site. A defect observed in the outer shipping flask was repaired, the flasks painted, and shipped to AECL on September 1. The hydraulic shearing device will be tested at Bettis Site prior to shipment to Chalk River.

Defect Test

The proposal for the first defect test is currently being written and will be issued shortly.

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

The piping of the main and supply systems and the control panel wiring have been completed for the screening loop to be installed in the HB-5 hole of the Materials Testing Reactor. Wiring from the panel to the system frames and erection of the system at Bettis Site has been started. It is estimated that erection and hydrostatic test at Bettis Site will be completed by September 15. After an operational test the system will be shipped to the MTR. It is hoped to set a start-up date close to December 1.

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

MTR

All pertinent calculations on the loop system for progressive failure tests in the HT-1 hole, Materials Testing Reactor, have been completed by the Blaw-Knox Co. and submitted to Bettis Site. Check calculations have been made and the results will be submitted to MTR for test section stress analysis and radioactivity hazards evaluation. Approval drawings are currently being checked. It is now estimated that the design will be completed by September 30.

Chalk River

Preliminary flow sheets and electrical designs of the test facility for proof testing in the Chalk River reactor have been prepared by the Blaw-Knox Co. and are being reviewed for conformance to requirements of the Atomic Energy of Canada, Ltd. Drawings of the Chalk River reactor have been requested for design of the experiment layout although the final location of equipment will not be known until September. Work is proceeding, meanwhile, on the basis of the smaller of two possible locations.

The earliest date for the start of irradiations, coinciding with the completion of current Canadian testing, appears to be July - September 1955.

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F. Reactor Physics1. PWR Core and Control Theoretical Physics (52all) A. F. Henry

More than twenty cases of seed and blanket core studies on the one-dimensional spatial simulator have been completed. These studies all have been carried out for a reactor core of spherical shape, in an attempt to determine the effect on the power distribution of various seed and blanket arrangements. Results appear in the following table.

Radius Seed	Thickness Blanket	Seed Loading	Seed Metal-to-Water Vol. Ratio	Age (cm ²)	Control of Seed	Power Ratio Blanket to Seed -- Incl. Fast Effect (12%)	Blanket Fuel	Blanket Water-to Uranium Vol. Ratio
(cm)	(cm)	(kg)						
43	30	70	1.35	$\tau = 110$	Homogeneous	.739	U-Mo	2
43	50	70	1.35	$\tau = 110$	Homogeneous	1.02	U-Mo	2
43	50	50	1.35	$\tau = 110$	Homogeneous	1.04	U-Mo	2
43	50	70	2.70	$\tau = 159$	Homogeneous	1.33	U-Mo	2
43	50	70	1.35	$\tau = 110$	Lumped (10 cm radius)	1.07	U-Mo	2
43	50	70	1.35	$\tau = 110$	Lumped (15 cm radius)	1.11	U-Mo	2
30	50	23.8	1.35	$\tau = 110$	Homogeneous	1.52	U-Mo	2
25	50	13.8	1.35	$\tau = 110$	Homogeneous	1.75	U-Mo	2
25	50	13.8	(τ increased 20 per cent)		Homogeneous	2.02	U-Mo	2
25	50	13.8	1.35	$\tau = 110$	Homogeneous	2.21	UO ₂	2
25	60	13.8	1.35	$\tau = 110$	Homogeneous	2.36	UO ₂	2
43	50	70	1.35	$\tau = 110$	Homogeneous	1.24	UO ₂	2
43	60	70	1.35	$\tau = 110$	Homogeneous	1.49	UO ₂	2
43	50	70	1.35	$\tau = 110$	Homogeneous	1.31	U ₃ -Si	2
43	50	70	1.35	$\tau = 110$	Homogeneous	1.18	U-Mo	3
43	50	70	1.35	$\tau = 110$	Homogeneous	1.22	U-Mo	4
43	50	70	1.35	$\tau = 110$	Homogeneous	1.66	UO ₂	3
43	60	70	1.35	$\tau = 110$	Homogeneous	1.89	UO ₂	3

The following general conclusions may be drawn from these results.

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(1) The power distribution between blanket and seed improves with a reduction in seed diameter. However, since the power density in the seed increases as the seed diameter is decreased, and this power density is a problem even with the largest seeds studied, a smaller seed diameter is probably not feasible.

(2) There appears to be little advantage in increasing the metal-to-water ratio in the seed for the U-Mo cases; an increase of this value by a factor of two results in no significant improvement in the power distribution.

(3) If UO_2 replaces U-Mo as the blanket fuel material, the power distribution is improved. The fact that the neutron age in UO_2 is greater than in U-Mo requires that for fullest utilization of the available neutrons the UO_2 blanket be larger than one composed of U-Mo. Increasing water-to-uranium volume ratio improves the power distribution significantly in the UO_2 case; this is probably because the values of k_{∞} for the UO_2 are nearer unity.

(4) The loading of the seed does not significantly affect the power distribution. Therefore, the seed loading and control can be considered as primarily affecting only the lifetime of the seed.

$$\text{Tables of the integral } I(\beta, x) = \frac{1}{\beta\sqrt{\pi}} \int_{-\infty}^{\infty} \frac{1}{1+y^2} e^{-\left[\frac{x-y}{\beta}\right]^2} dy,$$

representing the Doppler correction factor to the Breit-Wigner single level resonance, have been tabulated for the following range of

$$\beta = 2\sqrt{t} = \frac{4\sqrt{E_k T \frac{m}{M}}}{\Gamma}$$

and $x = 2 \frac{E - E_0}{\Gamma}$; (E_0 being the resonance energy, Γ the total resonance width, T the temperature of the scatterer, and $\frac{m}{M}$ the neutron-to-scatterer mass ratio):

For $t = 0$ to 0.1 , $\Delta t = 0.005$	For $t = 0.25$ to 1.0 , $\Delta t = 0.025$
$x = 0$ to 2 , $\Delta x = 0.05$	$x = 0$ to 4 , $\Delta x = 0.10$
$x = 2$ to 4 , $\Delta x = 0.10$	$x = 4$ to 300 , $\Delta x = 0.50$
$x = 4$ to 300 , $\Delta x = 0.50$	
For $t = 0.1$ to 0.25 , $\Delta t = 0.010$	For $t = 1.0$ to 2.0 , $\Delta t = 0.1$
$x = 0$ to 2 , $\Delta x = 0.05$	$x = 0$ to 300 , $\Delta x = 0.5$
$x = 2$ to 4 , $\Delta x = 0.10$	For $t = 2$ to 25 , $\Delta t = 1.0$
$x = 4$ to 300 , $\Delta x = 0.50$	$x = 0$ to 300 , $\Delta x = 0.5$
	For $t = 25$ to 65 , $\Delta t = 5.0$
	$x = 0$ to 300 , $\Delta x = 0.5$

For $\beta = 16, 17, 18, 19, 20, 25, 30, 35, 40, 45, 50, 60, 70, 80, 90, 100$

$x = 0, 0.5, 1.0, 1.5, 2, 3, 4, 5, 6, 8, 10, 12, 14, 16, 18, 20, 25, 30, 35, 40, 45, 50, 60, 70, 80, 90, 100, 110, 120, 130, 140, 160, 180, 200, 220, 240, 260, 280, 300$

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II-F-1 (Contd.)

These computations were done by the Atomic Energy Commission Computing Facility at New York University, and by the Mathematical Computation Laboratory at the David Taylor Model Basin, Washington, D. C. A report containing the results is being prepared at Bettis.

An analog computer study of the time required to burn out xenon has been completed. In the transients studied, criticality was maintained by coolant temperature variation only. The maximum boiler power, a quantity determined by steam loop limitations, controlled the increase to full reactor power during xenon override. The time required to reach full power and the maximum temperature excursions have been determined as functions of the lower flux value and the temperature coefficient of reactivity (Figs. 1, 2, and 3).

2. PWR Flexible Critical Experiment (52bl1) J. R. Feldmeier

The installation of the reflector tanks has been delayed due to faulty fabrication. One of the tanks has been repaired and is now installed, and the second one should be installed shortly. The general plumbing and electrical wiring are progressing satisfactorily. The mechanical assembly and the electronic controls and instrumentation should be completed by the early part of September.

Most of the zirconium strip has been cleaned and deburred. The difficulties associated with the machining of the natural uranium have been resolved and the total amount of the material will be fabricated and shipped by the middle of September. However, upon cleaning the first two shipments "total 1000 bars" a rough surface condition has been noted. At the present time only 15 per cent of the received materials are sufficiently good to be used without further working. A program is now underway to investigate the effect of the surface imperfections on the flexible core and to develop methods of alleviating difficulties caused by this condition.

The molybdenum fabrication has been stopped pending review by the fabricator to determine the reason for the failure to meet mechanical dimensional specifications. A sorting technique has been worked out which will allow the utilization of materials otherwise out of specification; however, many details still remain to be worked out.

Initial trials of the extrusion cladding machine have been successful and its installation at Bettis is anticipated shortly.

Assuming that no additional problems arise, it is expected that criticality of a test core will be realized in November.

3. Two-Region Critical Experiments (52bl2) J. R. Feldmeier

Measurements of epsilon and thermal utilization for the 2 to 1 water-to-uranium lattice (1.3 per cent enrichment, 0.600 in. diam) have been repeated as a check on the reproducibility of the method. These measurements agreed with previous measurements to be better than 1 per cent.

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MAXIMUM XENON BURNOUT IN PWR

REACTIVITY CONTROL : TEMPERATURE ONLY

INITIAL FLUX $\phi_0 = 4.56 \times 10^{13} \text{ n/v}$

INITIAL DISTURBANCE : STEP FROM 100% TO 50% ϕ_0

POWER INCREASE : THROTTLE FULLY OPENED AT PEAK

XENON WITH POWER LIMITED BY STEAM LOOP

TEMPERATURE COEFFICIENT = -1.5×10^{-4} PER $^{\circ}\text{F}$; ENRICHMENT = 2.5%

ANALYSIS AND COMPUTATION SECTION
ANALOG COMPUTER DATA

Fig. 1

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II-F-3 (Contd.)

MAXIMUM XENON BURNOUT IN PWR

REACTIVITY CONTROL: TEMPERATURE ONLY

INITIAL FLUX ϕ_0 $4.56 (10)^{13}$ nv

INITIAL DISTURBANCE: STEP CHANGE IN FLUX FROM ϕ_0 TO LOWER LEVEL INDICATED.

POWER INCREASE: THROTTLE OPENED FULLY AT PEAK XENON WITH POWER LIMITED BY STEAM LOOP

TEMPERATURE COEFFICIENT: AS INDICATED; ENRICHMENT 25%

ANALYSIS AND COMPUTATION SECTION
ANALOG COMPUTER DATA

Fig. 2

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II-F-3 (Contd.)

MAXIMUM XENON BURNOUT IN PWR

REACTIVITY CONTROL: TEMPERATURE ONLY

INITIAL FLUX $\phi_0 = 4.56 (10)^{13}$ nV

INITIAL DISTURBANCE: STEP CHANGE IN FLUX FROM ϕ_0 TO LOWER LEVEL INDICATED.

POWER INCREASE: THROTTLE OPENED FULLY AT PEAK XENON WITH POWER LIMITED BY STEAM LOOP

TEMPERATURE COEFFICIENT: AS INDICATED; ENRICHMENT 25%

ANALYSIS AND COMPUTATION SECTION
ANALOG COMPUTER DATA

Fig. 3

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II-F-3 (Contd.)

Also repeated was the correction to be applied to the epsilon measurements (due to different decay products of U^{235} and U^{238}) as obtained from the special fission counter exposed in the reflector of the TRX.

A report on all the measurements made on the 1.3 per cent enrichment, 0.600 in. diam fuel rods is being written.

An experiment is being prepared to measure the temperature coefficient of a lattice in which the fuel is spaced in what would ordinarily be a 4 to 1 water-to-uranium lattice, but has some water displaced by inserting 3/8 in. diam aluminum rods at the center of each triangle formed by three fuel rods. This changes the water-to-metal ratio to 1.6 to 1 and the water-to-uranium ratio to 3.25 to 1. The results of the experiment will indicate whether or not a negative temperature coefficient can be obtained with a lattice which normally has a positive coefficient by decreasing the water-to-metal ratio.

4. Primary Loop Shielding Physics (52a13) J. J. Taylor

A study is being made of the effects of core corrosion on activity build-up in the PWR primary system. A theoretical description of the activity build-up due to a given isotope has been formulated. A careful examination of the fission product spectrum is being made to ascertain which of the many species produced contribute significantly to the gross loop activity level. Special attention is being given to those heavy elements which are α -emitters, since discharged water containing them must be kept out of any sources of drinking water and since they may cause serious air contamination problems.

As data are received from the corrosion test loop facility (WAPD-MRF-47, II-D-1), a measure of the accuracy of the theoretical description will be gained, and in turn it is hoped that the postulated build-up mechanism will give further insight into the general experimental problems encountered during operation of the test loop.

The calculations of the heating effect of gammas on the inside of the PWR pressure vessel have been completed. The calculations included the heating due to direct core gammas (prompt and delayed) and the iron capture gammas. The computations were done as a function of core size, reflector thickness and thermal shield thickness required in order to maintain a constant heating of 50,000 BTU/ft³-hr.

Calculations have been made in order to determine the radiation hazard created when an accident occurs which releases the primary coolant water and various fractions of the fission products into a sphere which is not shielded. The dose rate as a function of distance from the surface of the sphere has been obtained. Also the hazard to the surrounding regions created by slow emission of the fission products into the air and their subsequent diffusion has been evaluated. Both of these sets of data are needed for the layout and setting of specifications for the PWR vapor containers.

II-F-5.

5. Physics In-Pile Irradiation Experiments (53b32) J. R. Feldmeier

Reactivity Measurements of Irradiated Fuel

Specimens of Zircaloy-A and Zircaloy-B fuel material have been fabricated to conduct preliminary calibration tests in the Reactivity Measuring Facility (RMF) at the Materials Testing Reactor. Shipment of the specimens to Idaho is awaiting completion of the construction of the RMF. Calculations are continuing on the reactivity changes to be expected. Preliminary results indicate the desirability of calibrating the RMF with known thermal and resonance absorbers such as boron, cadmium, gold, and indium. Such calibrations may make it possible to determine the proper values for the weighting functions to be used in conjunction with the reactivity measurements of irradiated fuels.

Calculations have been started to determine what types of molybdenum specimens are desirable for tests in the RMF. These tests are designed to measure reactivity changes as a function of the surface-to-volume ratio for cylindrical specimens.

The thin hafnium specimens (WAPD-5-3) have been inserted in the active lattice of the Materials Testing Reactor attached to the side of a modified fuel assembly. One cycle of irradiation has been completed and the first specimen has been successfully removed from the fuel assembly for shipment to the Westinghouse Hot Laboratory. The remaining two specimens have been re-inserted into the reactor for further irradiation. Results of the mass spectrometer measurements on the previous experiment, WAPD-5-2, have been received from Argonne National Laboratory and the data are being analyzed. Upon completion of this analysis, a report will be issued.

Water-Logged PWR Fuel Element Experiment

An attempt is being made to circumvent some of the experimental difficulties of laboratory tests (See WAPD-MRP-46, Section CR-WAPD-Met5) on water-logged UO_2 fuel elements. It is proposed to perform this test in a manner similar to the "Short Time Autoclave Experiments" as previously described in WAPD-MRP-46 (53b32-P). These tests would then more closely duplicate reactor conditions.

The water-logged fuel element is to be inserted into the Materials Testing Reactor while housed in an autoclave. As the temperature of the fuel rises, egress of the water through the cladding defect can be monitored remotely by measuring the autoclave pressure. If the pressure within the fuel element rises sufficiently to burst the cladding, the abrupt pressure increase within the autoclave should indicate the incident.

Several simulated fuel elements (using sintered stainless steel cores instead of UO_2) with a 0.005 in. diam defect in the cladding have been water-logged by submerging them in boiling water. These elements absorbed water to fill approximately one-half of the volume available in the simulated fuel material.

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II-F-5 (Contd.)

If it is assumed that the above condition will exist in UO_2 cores and that no water leakage will occur through the cladding defect as the fuel element heats under irradiation, the internal bursting pressure of about 5,000 psi will be obtained when the internal temperature rises to about 500°F. However, if the quantity of water absorbed through a fabrication defect is much smaller than one-half of the available volume, it may be possible for the core temperature to rise to that at reactor operating conditions and still not exceed the pressure required to burst the cladding. The proposed experiments will assist in determining the seriousness of the water-logged fuel incident.

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APPENDIX I

Cross Index of FY 1955 Project Numbers Used in this Report with FY 1954
Project Numbers Used to Report the FY 1954 Program

PART I

PWR ENGINEERING

	<u>FY 1955</u>	<u>FY 1954</u>
<u>A. Power Plant Analysis and Systems</u>	<u>Proj.No.</u>	<u>Proj.No.</u>
1. Plant Functional Design and Operation Analysis	55a11	55a11
2. Primary Coolant System	55b11	55b11
3. Primary Auxiliary and Secondary Systems	55d11	Not Active
4. Reactor Control System Design	56a13	56a13
5. Non-Nuclear Control Systems Design	55e12	Not Active
6. Control System Tests	56a14	Not Active
7. Plant Layout and Shielding	55a14	55a14
8. Reactor Plant Site Facility	55a12	55a12
9. Instruction Manuals and Test Program	59c16	Not Active
<u>B. Power Plant Components and Component Materials and Tests</u>		
1. Primary Coolant Loop	55b26	55b26
2. Primary Coolant Pumps	55b20	55b20
3. Pump Component Development	55b22	55b22
4. Primary Pump Test Loop	55b23	Not Active
5. Auxiliary Pumps	55b21	Not Active
6. Primary Coolant Valves	55b19	55b19
7. Steam Generators	55b32	55b32
8. Reactor Plant Pressure Vessels and Degasifier	55b33	Not Active
9. Reactor Plant Auxiliary Heat Exchangers	55b34	Not Active
10. Filters and Vent Capillaries	55b35	Not Active
11. Emergency Cooling Components	55b45	Not Active
12. Reactor Electric Power Components	56d12	Not Active
13. PWR Instrumentation	56d13	56d13
14. Materials for Primary Loop Components	55b16	55b16
<u>C. Reactor and Auxiliaries</u>		
1. PWR Core Development	53a14	53a14
2. Mechanical Design of PWR Core	53a12	Not Active
3. Thermal and Hydraulic Analysis and Tests of PWR Core	55a13	Not Active
4. Mechanical Tests of PWR Core Components	53a13	Not Active
5. Core Refueling Equipment	53d12	53d12
6. Radioactive Materials Handling and Viewing	53d11	Not Active
7. Control Mechanism - Development	56c19	56c19
8. Reactor Vessel Assembly	55b25	55b25

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APPENDIX I (Contd.)

PART II

PWR DEVELOPMENT

	<u>FY 1955</u>	<u>FY 1954</u>
	<u>Proj.No.</u>	<u>Proj.No.</u>
<u>A. Fuel Element Development</u>		
1. Core Raw Materials and Shapes	53c21	53c21
2. Core Manufacturing Process Development	53c23	53c23
3. Advanced Fuel Element Development	53c24	Not Active
4. Fuel Element Development	53c25	Not Active
5. Thermal and Hydraulic Properties of Fuel Elements	55a15	Not Active
6. Pilot Fuel Element Manufacture	53c41	Not Active
<u>B. Metallurgy of Core Materials</u>		
1. Physical Metallurgy of Core Materials	53b11	53b11
2. Mechanical Metallurgy of Core Materials	53b21	53b21
3. Corrosion of Core Materials in Static Systems	53b41	53b41
<u>C. Materials Application Development</u>		
1. Non-Core Materials	53a11	53a11
2. Carbon Steel Program	55b14	53b14
<u>D. Chemistry Development</u>		
1. Fuel Element Testing and Fission Product Removal	55b12	55b12
2. PWR Coolant Technology Applications	55b10	Not Active
3. Chemical Control	56a12	56a12
<u>E. Irradiation Effects</u>		
1. Irradiation Effects on Core Materials	53b31	53b31
2. In-Pile Radiation Exposure Tests of PWR Fuel Materials	53b34	53b34
3. In-Pile Screening Tests of PWR Fuel Materials	53b33	53b33
4. In-Pile Proof Tests of PWR Fuel Assemblies	53b35	53b35
<u>F. Reactor Physics</u>		
1. PWR Core and Control Theoretical Physics	52a11	52a11
2. PWR Flexible Critical Experiment	52b11	52b11
3. Two-Region Critical Experiment	52b12	52b12
4. Primary Loop Shielding Physics	52a13	52a13
5. Physics In-Pile Irradiation Experiments	53b32	53b32

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