DP - 455

Reactors - Power

AEC Research and Development Report

HEAVY WATER MODERATED POWER REACTORS

PROGRESS REPORT

December 1959

Technical Division

Wilmington, Delaware

January 1960

E. I. du Pont de Nemours & Co.
Explosives Department - Atomic Energy Division
Technical Division - Wilmington, Delaware

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Progress Report
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Power Reactor Studies
Wilmington, Delaware

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ABSTRACT

At the end of 1959, 25% of the construction and 85% of the firm design of the Heavy Water Components Test Reactor (HWCTR) were complete. Further safeguards analyses of the HWCTR, done with the aid of analog and digital computers, reinforced earlier data that indicated that the reactor is highly self-regulating and that the safety system should prevent release of radioactivity outside the containment building. Fabrication tests of metallic uranium fuels and the preparation of irradiation specimens of swaged uranium oxide fuel tubes continued. A tube of Zircaloy-2-clad U - 2 w/o Zr failed during a low temperature, low pressure irradiation test to modest exposure. Preliminary examinations were made of tubular metallurgical joints between Zircaloy and stainless steel and these joints continued to show promise.

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INTRODUCTION

This report is one of a series that records the progress of the du Pont study of heavy-water-moderated, natural-uranium-fueled power reactors. The current effort is divided into two main categories: (1) the research and development required for the successful design, construction, and operation of the Heavy Water Components Test Reactor (HWCTR), a high temperature fuel irradiation facility, and (2) the experimental and theoretical studies required for developing the technology of a full-scale D_2O-moderated power reactor plant. Earlier reports on this study are:

DP-232	DP-315	DP-405
DP-245	DP-345	DP-415
DP-265	DP-375	DP-425
DP-285	DP-385	DP-435
DP-295	DP-395	DP-445

Progress for the month of January 1960 will be reported in DP-465.

SUMMARY

At the end of December 1959, about 25% of the construction and 85% of the firm design of the Heavy Water Components Test Reactor were complete. The progress of construction during December is shown in Figures 1 and 2.

Further safeguards analyses were made of the dynamics and control of the main cooling system of the HWCTR. The results of IBM digital computations of the transients that follow cold water accidents are shown in Figures 4 and 5. Computations performed with an analog simulation of the system showed that the control instrumentation, the reactor equipment, and the auxiliary equipment will perform satisfactorily under normal and transient conditions and that the reactor will be highly self-regulating by virtue of the HWCTR's large, negative temperature coefficient of reactivity.

The fabrication of irradiation specimens of uranium metal continued at Nuclear Metals, Inc. The billet design was chosen for a Zircaloy-clad tube of pure uranium, enriched to 3% U^{235} content, for irradiation in the Vallecitos Boiling Water Reactor. Four extrusions, with natural uranium cores and different end shape designs, provided the data upon which the choice was based. Postextrusion processing of a Zircaloy-clad U - 1.5 w/o Mo tube that was produced for irradiation testing at the Savannah River Plant, showed that the tube was unacceptable for this purpose because the minimum cladding thickness specification was not met.

Additional results were obtained in the study of beta-phase uranium alloys that are expected to be dimensionally more stable under irradiation than the less creep-resistant alpha-phase uranium alloys. Heat treatment and aging experiments indicated that a U - 0.3 w/o Cr - 0.3 w/o Mo alloy, extruded at 620°C and gamma solution heat treated, provides the most stable beta phase of the alloys studied.

Five 2-foot-long specimens of stainless-steel-clad uranium oxide tubes that were fabricated at the Savannah River Laboratory by the cold swaging of crushed, fused oxide were selected for irradiation testing. The tubes, which are approximately 2 inches in outer diameter and 1/4 inch thick, will be irradiated in the low temperature, low pressure coolant of a Savannah River reactor. Several cold-swaged rods of Zircaloy-clad UO_2 are being prepared for SRP irradiation in lead-insulated, stainless steel containers. These tests are designed to provide additional information on the effects of central temperature, exposure, oxygen content, and method of preparation on the in-pile behavior of uranium oxide fuel.

A coextruded tube of U - 2 w/o Zr clad with Zircaloy-2 failed in a Savannah River reactor during irradiation at low temperature and low pressure. This tube was the third Zircaloy-2-clad U - 2 w/o Zr tube to fail in a Savannah River reactor; however, five other such tubes, as well as several zirconium-clad pure uranium tubes, performed satisfactorily during irradiation in the past. The present emphasis in the irradiation program is to test the additional pure uranium tubes and the new alloy tubes that Nuclear Metals, Inc. is fabricating.

Fabrication of tubular metallurgical joints between Zircaloy and stainless steel continued at Nuclear Metals, Inc. with promising results. A small loop for cyclically testing the joints at a maximum pressure of 1000 psig and a maximum temperature of 260°C was completed at the Savannah River Laboratory. Testing of the joints is scheduled to start early in January.

DISCUSSION

I. HEAVY WATER COMPONENTS TEST REACTORS (HWCTR)

The HWCTR is a test reactor in which numerous fuel elements will be irradiated under conditions of temperature, pressure, and neutron flux that are typical of those expected in D_2O-moderated power reactors. A description of the reactor was presented in DP-383[1] and in earlier progress reports. Construction of the facility was authorized by the Atomic Energy Commission in November 1958. The goal for startup is early 1961. The total cost of the test facility, which is designed for a thermal output of 61 MW, is estimated to be $7,700,000 plus an additional $1,000,000 for two isolated coolant loops in which special fuel assemblies will be irradiated. Progress during the month of December on the HWCTR design, construction, and supporting experimental work is summarized in this section.

A. STATUS

1. Construction

At the end of 1959, construction of the HWCTR was approximately 25% complete. The progress of construction during December is recorded in the photographs of Figures 1 and 2. Preparations neared completion for the pouring of the 4-1/2-foot-thick concrete slab at the zero level of the containment building. The forms for the slab were erected; the stub columns for support of the rotating bridge crane were placed; the hold-down bolts for the steel shell of the building were installed; and much of the embedded piping, conduit, and reinforcing steel was completed. Pouring of the concrete is expected early in January.

An analysis of the effects of the recent steel strike on the startup of the HWCTR showed that the facility would probably not be in the possession of the operating group until June or July 1961. The controlling item in establishing the new schedule is the date of delivery of the reactor vessel to the plant site. Delivery of the steel to the vessel fabricator has been delayed six months by the strike, and because the vessel is the limiting piece of equipment in the construction schedule, startup of the reactor has been delayed an equal time. Other important items in the schedule whose delivery has been delayed substantially by the strike are the transfer coffin for the irradiated fuel, the control and safety rod drives, and the steel shell of the containment building.

(1) DP-383, Preliminary Hazards Evaluation of the Heavy Water Components Test Reactor, D. S. St. John, et al., May 1959.

2. Design

Firm design of the HWCTR was approximately 85% complete at the end of 1959.

3. Isolated Coolant Loops

At the end of December 1959, approximately 20% of the firm design of the two isolated coolant loops - one for liquid D_2O coolant, the other for boiling D_2O coolant - was complete.

It is now planned to use Zircaloy-4 rather than Zircaloy-2 for one of the two pressure tubes required for the isolated coolant loops of the HWCTR. The purpose of this plan is to evaluate both alloys as candidate materials for the pressure tubes of a large-scale power reactor. An evaluation of the two alloys as sheathing materials is also to be made in the HWCTR. The reason for the interest in Zircaloy-4 is that the lower nickel content of Zr-4, as compared to Zr-2, may decrease hydrogen pickup and subsequent embrittlement of the Zircaloy during long-term exposure in a reactor.

B. SAFEGUARDS ANALYSES

1. Cold Water Accidents

Analyses were reported in DP-405 and DP-415 of the transients in temperature, pressure, and heat generation that follow various postulated accidents in operating the HWCTR. A detailed analysis of a "cold water" accident was given in DP-415, pp. 13-17. In the cold water accident of DP-415, it was assumed that immediately before the accident (a) the reactor was critical and at low power, (b) the reactor was cooled by two independent loops (see Figure 3), (c) loop A was operating with its steam valve closed, (d) the reactor and loop A were at a temperature of about 200°C, and (e) loop B was at a temperature of 30°C, with its pump running and the isolation valves closed. The accident was initiated by the opening of the isolation valves in loop B and the subsequent injection of cold water from loop B into the reactor. The cold water caused an increase in reactivity which, in turn, led to overheating and melting of the fuel; however, the radioactivity released was contained within the reactor system.

During December, the cold water accident was re-examined to determine the effect of one of the assumptions made in the calculations reported in DP-415. In that report, it was assumed that the rates of heat transfer from the fuel to the coolant were linear functions of fuel temperature, at all power levels reached during the transient. In actuality, at power levels well above the design power of the HWCTR, vapor blanketing of the fuel and subsequent heat transfer burnout limit the rate at which heat can be removed from the fuel surfaces. In the calculations that are reported in the succeeding paragraphs, the heat transfer rate from the fuel to the coolant was assumed to be limited to 100 MW by this film burnout phenomenon. This limitation on the rate of heat transfer during a cold water accident increases the rate of fuel temperature rise and decreases the rate of moderator heating, relative to the rates computed in the earlier study. The slower rate of moderator heating implies that the large, negative moderator temperature coefficient will take longer to affect the power excursion, and the maximum power reached, therefore, will be higher.

The transients following a "standard" cold water accident with the 100-MW limitation on heat transfer to the coolant are shown in Figure 4. The reactor was assumed to be operating at 5 MW on one coolant loop, and the accident was initiated when D_2O flow in the cold, stagnant loop was suddenly started. A prompt temperature coefficient of -2.6×10^{-5} $k/^{o}C$ was assumed. If the safety rods were to fail to function, the power level would reach 100 MW in 8.6 seconds, and would rise rapidly thereafter. The fuel elements would melt and probably vaporize. The rate of power increase that would be obtained if there were no limitation on the rate of heat transfer imposed by film burnout is shown in the same figure for comparison. In the latter case, which was also discussed in DP-415, the uranium metal fuel would melt, but the melting point of zirconium would not be reached and no vaporization would occur.

In Figure 5, the following cold water accidents - all with the maximum rate of heat transfer limited by film burnout of the fuel - are compared: (a) a "standard" accident, as defined above, (b) an accident in which the reactor has an initial power of 5 MW and a prompt temperature coefficient of zero, and (c) an accident in which the reactor has an initial power of 0.1 MW and a prompt temperature coefficient of -2.6×10^{-5} $k/^{o}C$. All of these accidents would lead to vaporization of the fuel in about 12 seconds, unless some corrective action were taken.

The inset in Figure 5 shows the slopes of the power-versus-time curves corresponding to various periods. On each curve, the point P indicates the time at which the period reaches the scram set point of 15 seconds. Allowances of one second for the instrument to respond and of two seconds for the safety rods to fall are made to estimate the scram times, denoted by S. Because all of the scrams occur at powers well below 100 MW, the effect of the film burnout limitation on the rate of heat transfer is to make the consequences of a cold water accident more severe only if the safety circuits fail and the power exceeds 100 MW.

2. Analog Simulation of Reactor Cooling System

The analog computations of the dynamics and control of the main cooling system of the HWCTR (DP-425) were completed during this report period by the du Pont Engineering Department. The objectives of the computations were (1) to investigate the performance of the reactor and the steam generators under normal and transient conditions, (2) to determine the effects of various accidents, and (3) to provide data for design changes, if needed. Preliminary results of this study showed that the control instrumentation, the reactor equipment, and the auxiliary equipment will perform satisfactorily and that the system will be highly self-regulating by virtue of its large, negative temperature coefficient of reactivity. The results of the study are discussed below.

Earlier computations of the HWCTR performance were made on an IBM digital computer at the Savannah River Laboratory and were reported in DP-405 and DP-415. The digital computations provided much information concerning the transient and steady-state behavior of the HWCTR. However, the analog simulation was particularly useful in studying the action and interaction of the control instrumentation for both the primary and secondary coolant systems, by virtue of the greater ease with which a larger number of differential equations describing the system could be handled on the analog, as compared to the digital computer.

The following items were simulated on the analog computer: (a) the reactor kinetics, including the delayed neutrons, the moderator temperature coefficients, and the movements of the control rods, (b) the moderator level in the reactor, which is controlled by a liquid purge, (c) the heat removed from the reactor by the steam generators, (d) the steam pressure control, (e) the H_2O level in the steam generators, which is controlled by boiler feed pumps which in turn are driven by steam turbines, using steam from the generators, (f) the boiler blowdown, and (g) the supply of boiler feedwater from the standpipe. In the computations, the temperature coefficients of reactivity for the fuel and moderator were combined and the boiler blowdown was made proportional to the main flow of steam. Addition of helium to the system to control the reactor pressure was deleted from the analog simulation because analysis showed that the maximum addition rate of 4 scfm would have a negligible effect upon the short duration transients of interest in this study. A flowsheet for the HWCTR system is given in Figure 22 of DP-375.

The conclusions from the analog study are as follows:

(1) The system is stable as designed; there is negligible hunting and no drift in the reactor power level. The steam generator controls for steam pressure, boiler feed rate, and steam flow to the turbine drives of the feedwater pumps perform as desired under such normal operations as startup, shutdown, power change, and moderator temperature adjustment.

(2) Startup of the reactor and the achievement of full power was studied, partly to compare automatic steam pressure control with manual control. As shown in DP-245, in the operation of a hot moderator reactor such as the HWCTR, the position of the control rods determine the moderator (i.e., reactor effluent) temperature, while the steam valve opening determines the power level at that moderator temperature. In the computation of reactor startup, the slow removal of control rods at a constant rate of 0.003 Δk/minute from the initial critical condition

with cold moderator (100°C) was simulated. This rate of rod withdrawal caused the moderator temperature to increase at about 2°C/minute. Steam pressure and reactor pressure also increased progressively. At the start of the control rod withdrawal, the reactor power increased rapidly and in one minute attained a power of about 15 MW and then remained constant; this power satisfied the demands of the feedwater turbine and of the system heat capacity for a rise of 2°C/minute. When the desired steam pressure was reached in the steam generator, the steam valve was opened as required to maintain constant pressure. On the analog, automatic and manual openings of the steam valve were simulated. From this point, with constant steam pressure and with the control rods still moving out at constant rate, the moderator temperature increased less rapidly while the power level increased more rapidly than before. The rod withdrawal was stopped when the desired power was attained. The time required for the startup was about one hour. For this method of startup, the analog computations showed that the desired reactor power was established somewhat more smoothly with automatic control of steam pressure rather than with manual control. However, it is more likely that, for startup, the moderator temperature will be established first at low power by control rod positioning and then the power will be increased by opening the steam valve stepwise. In this latter procedure, the steam pressure is a maximum at zero steam flow and then decreases to the operating level as power is increased. An automatic steam pressure controller would then serve no function in this method of startup.

(3) In shutting down the reactor by insertion of control and/or safety rods, the steam pressure may be controlled at a particular level or the steam valve opening may be left as it was during full power operation. The choice of steam valve opening would depend on the reason for shutting down. Controlling the steam pressure limits the decrease in reactor temperature and pressure. When the reactor is scrammed (-0.1 Δk/minute), if the steam valve opening were to remain at the same position as during full power operation, the most rapid cooling rate would result (i.e., about 60°C/minute initially, or 100°C decrease in 7 minutes). The reactor pressure and the steam pressure also would decrease rapidly. The rate of cooling would not be so rapid as to cause thermal stresses to exceed prescribed limits.

(4) During startup the transition from the gravity flow of boiler feedwater from the standpipe to the turbine-driven pump system was shown to be smooth.

(5) For a reactor heatup rate of 1°C/minute, the liquid level in the reactor initially should be about 18 inches below the level-control point to allow for D_2O expansion with a minimum of D_2O drawoff and a minimum transient increase in the reactor pressure above that desired for steady operation; for a heatup rate of 2°C/minute, the initial level should be 30 inches below the level-control point.

(6) There is essentially no interaction between the steam generators. The distribution of the heat load between the two steam generators depends only upon their individual operating conditions, such as the steam pressure, D_2O flow, and feedwater supply to the particular generator. The reactor power is the sum of the power removed by each generator. Thus, if one steam generator were removed from the system, perhaps as for example, by the failure of the D_2O pump in its loop, the reactor power would be reduced to about half power and only a slight increase in moderator temperature would occur. In another situation, if the turbine-driven feedwater pump for one steam generator were to fail, the reactor power level again would reduce to about half but would oscillate with a period of about seven minutes and with an amplitude of about 3 MW (i.e., a 10% variation). This oscillation occurs because the H_2O side of the steam generator is boiled dry; the steam escapes, lowering the pressure; a small amount of water enters from the standpipe; this water is evaporated; and then the process repeats itself. Each repetitive entry and evaporation of water is accompanied by the small surge in power.

It might appear, as a corollary to the above discussion, that it is not even necessary to scram the reactor power upon the loss of AC power to one of the two D_2O pumps. The reactor should be scrammed, however, to avoid possible "cold water" accidents, such as those discussed in the preceding article and in the next section. Moreover, the analog did not simulate the reactor hydraulics; the analog computations only simulated the complete stoppage of D_2O flow in one of the main loops. In actuality, backflow would occur through the loop with the nonenergized pump; cold D_2O would enter directly into the moderator space of the reactor; and an undesired "cold water" incident would result. The use of check valves in the process piping to prevent backflow is not considered to be a good practice because of the possible water hammer that could occur if the check valve were not to close until full backflow was established.

(7) "Cold water" accidents of any type (i.e., accidents that result from the injection of relatively cool D_2O into the reactor moderator space) cause sudden undesirable increases in reactor power. This is the penalty associated with the negative temperature coefficient of reactivity that in all other respects makes the reactor stable and self-regulating. The "cold water" accidents simulated were (a) a sudden increase in steam flow causes a sudden drop in boiling H_2O temperature, and (b) with the reactor initially at power and operating with only one loop, flow is started through the cold loop.

C. COMPONENTS TESTING

1. Rod Drive Packages

Prototypes of a safety rod drive package and a control rod drive package have been under test at Alco Products, Inc., the consultants on the design and the fabricators of the prototype packages. Testing of the safety rod package was completed in November and showed the mechanism to be satisfactory (DP-445). Testing of the prototype control rod package was completed satisfactorily during this report period.

The test of the control rod package was started early in November and consisted of 3000 cycles of up-and-down drive. No failure of equipment occurred and no significant wear was found except on the drive pinion. The test was intended to simulate ten years or more of reactor service at design temperature and pressure, 285°C and 1500 psig, respectively. After completion of the service test, measurements were made of torque input required to drive the rods, drift in the limit switch settings, and the cooling water flow through the control rod and lower guide tube. All were found to be within the design limits and not significantly different from the initial values.

Tests were also carried out on an improved motor for control and safety rod drives. The motors that were supplied earlier did not meet specifications on variation of speed with load. The new motor was found to be satisfactory.

During testing of both the control and safety rod packages, shaft seal leakage (two different seals) was measured at 1500 psig and 285°C. This leakage, approximately 100% of which is recoverable, ranged from 1.9 to 3.9 lb/hr of outleakage and from 0.2 to 1.3 lb/hr of inleakage. All test work by Alco is now complete and a topical report is in preparation.

2. Prototype Gripper Mechanism for Fuel Transfer Coffin

Testing of the prototype gripper mechanism for the irradiated fuel transfer coffin continued at the Savannah River Plant. New equipment that was installed and tested to satisfaction during December on the mockup included the complete prototype of the fuel element cooling system and a slip clutch to impart some downward driving force to the rack. This clutch is necessary because the weight of the mechanism is not adequate to overcome friction and also supply the desired seating force for components. Up-and-down cycling of the gripper drive was started during December as a part of a durability test, and is now continuing.

3. Seal Leakage

a. Pump Seals

The average rates of liquid water leakage from the inboard and outboard mechanical seals of the centrifugal pump in a hydraulic loop of the Savannah River Laboratory were 31 and 125 gal/yr, respectively, after 32 days of operation at a pump suction pressure of 850 psig. The leakage rates were not significantly different from those reported previously for the first 90 hours of operation. The higher leakage from the outboard seal is not unusual for this pump; approximate measurements during earlier tests of 18 months duration also indicated that the leakage from the outboard seal was much higher than from the inboard seal. In the absence of detailed measurements of the seal components, which will be made at the end of the current tests, there is at present no explanation for the higher leakage from the outboard seal.

The test pump, which has a capacity of 2800 gpm and is driven at 3600 rpm by a 200-hp motor, recirculates filtered and deionized water through the test loop at 260°C. Figure 6 is an assembly drawing of one of the mechanical seals. In normal operation, all of the liquid leaking from the seal drains by gravity through an opening in the lower part of the seal flange; vapor leakage, if any, flows out partly through the liquid drain and partly through the annular opening between the "Teflon" throttling ring and the pump shaft. Although the seal coolant is usually maintained at a temperature below 65°C, local temperatures on the seal faces may be high enough to vaporize part of the liquid leaking through the seal. Liquid may also leak past the throttling ring, but this has not been observed in operation to date.

In the experimental leakage collection system shown in Figure 6, liquid leakage from the seal is collected in a bottle. Vapor leakage from the seal is of considerable interest, because in power reactor operation it may not be readily recoverable. Vapor leakage from the experimental seal is estimated by passing dry nitrogen through the seal chamber at a metered rate and then analyzing the effluent gas for moisture content. The dry gas is injected and the moist gas discharged through special openings in the seal flange. Vaporization of liquid by the small flow of gas through the collection bottle is minimized by means of a condenser. Liquid that leaks into the seal chamber is vaporized somewhat by the flow of gas over it; this makes the estimates of vapor leakage higher than would be expected under normal operating conditions. The maximum vapor leakage experienced thus far, expressed as liquid, has been estimated to be about 1.5 gal/yr.

b. Valve Stem Closures

A test program was started at Savannah River Laboratory to determine experimentally the leakage rate from several typical valve stem closures.

The bonnet-and-stem assemblies of four valves were installed in the cyclic tester that was previously used for testing the closure seals of the HWCTR reactor vessel. The assemblies include the valve stem closures of one 6-inch, one 3-inch, and two 3/4-inch valves. Drawings of these assemblies are shown in Figures 7, 8, and 9, respectively. Leakage rates are to be determined with the valve stem backseated (i.e., open). The leakage out the lantern gland will be measured by a system similar to the one shown in Figure 6. During the proposed tests, the valves will be maintained stationary and conditions cycled 100 times. In a typical three-hour cycle, the deionized water in the valve stem assemblies will be maintained for one hour at a maximum pressure of 1000 psig and a maximum temperature of 260°C; the remainder of the cycle will be used for heating, cooling, and venting to the atmosphere at the end of each cycle. Startup of these tests is expected during the next report period. Later, the equipment will be modified so that the entire valve stem assembly may be enclosed. Nitrogen gas will be passed through the enclosure, and the moisture pickup will be measured. The vapor leakage through the valve stem glands will be thereby determined.

In addition to the stationary tests of valve stems discussed above, it is planned to measure the leakage from a complete 3-inch gate valve that is cyclically opened and closed. In this test, in which deionized water will flow through the valve at 260°C and 1000 psig, the valve will be opened and closed at intervals of 2 to 5 minutes. Modification of existing equipment for testing the valve is about 50% complete, and the test is scheduled to begin early in January 1960.

c. Gasketed Joints

Cyclic testing of the HWCTR gasketed joints at 1500 psig was temporarily postponed because of the more urgent need for information on the leakage from valve stem closures that were discussed above. Although the originally planned 100-cycle test was not completed, the test gaskets were exposed in two different tests to a total of 66 cycles at a temperature of about 260°C and a pressure of 1500 psig. The tests were stopped after 53 cycles because the leakage rates from some of the test joints exceeded the range of the moisture detection system. The range of the instrument was subsequently extended, and the tests were resumed for 13 more cycles with new gaskets in each joint.

Although the data from the tests are not yet fully analyzed, preliminary results on the leakage from the 6-inch flange gasket of a control rod enclosure and on the combined leakage from the 1-inch gasket and the "Swagelok" tubing fitting of a monitor pin closure, show both these leakages to be below design limits. The design maxima for these leakages are 94 in^3 of water/month from the 6-inch flange gasket and 12.2 in^3 of water/month from one monitor pin closure. The leakage from another monitor pin closure, however, exceeded the specified maximum rate. This closure consisted of a "Midlock" tubing fitting and a gasket identical to the one used with the "Swagelok" fitting. Testing of the "Midlock" fitting and the gasket was therefore terminated.

D. IRRADIATION TESTING OF HWCTR DRIVER TUBES

An enriched driver tube of 9.3 w/o oralloy, alloyed with zirconium and clad in 0.015-inch-thick Zircaloy-2, was successfully irradiated in a Savannah River reactor to a modest burnup. Further irradiations of

HWCTR driver fuel elements are planned. Quantitative data for the tests will be given in a forthcoming classified report.

II. TECHNOLOGY OF FULL-SCALE REACTORS

A. PHYSICS - THERMAL DIFFUSION AREA OF REACTORS CONTAINING VOIDS

The effects of lattice voids is of much importance in the calculation of the nuclear reactivity of reactors that are cooled with gas or a boiling liquid. These effects were studied theoretically by Behrens.[1-4] The results of exponential measurements on D_2O-moderated lattices of gas-cooled natural uranium metal plates were presented in DP-285, DP-315, and DP-345. These measurements were intended to test the reliability of the calculated reactivities for the gas-cooled power reactor lattices that were described in the latter report. However, the experiments proved to be difficult to interpret because of neutron streaming effects in the gas coolant channels. To better understand these streaming effects, supplementary exponential measurements were made with nonmultiplying lattices of lead-mercury alloy plates.

1. Description of "Fuel" Assemblies

The Pb-Hg "fuel" assemblies consisted of different numbers of the Pb-Hg plates arranged in a regular array within thin-walled aluminum cans. The geometries of the various assemblies are shown in Figure 10. They are identical to the natural uranium assemblies that were used in the earlier experiments. The plates were 57 inches long, 1.188 inches wide, and 0.18 inch thick. The plates were maintained 0.38 inch apart (i.e., face to face) by thin aluminum spacers that were located at three different elevations in the assembly.

The purpose of the Pb-Hg alloy was to mock up the absorption and scattering cross sections of uranium so that measurements of the thermal neutron diffusion area could be made without the complications of fissions occurring within the plates. The desired composition of Pb-Hg for this purpose was about 2.9 w/o Hg in Pb. However, the actual concentration as determined by chemical and danger coefficient measurements was 3.4 w/o Hg. This change in concentration affects the absolute values of the measured diffusion areas, but does not affect the neutron streaming measurements, which were the measurements of primary interest.

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(1) Behrens, D. J., The Effect of Holes in a Reacting Material on the Passage of Neutrons, with Special Reference to the Critical Dimensions of a Reactor, AERE T/R 103, (1958).
(2) Behrens, D. J., Calculation of the Neutron Migration Length in an Infinite Lattice, AERE T/R 221, (1958).
(3) Behrens, D. J., The Migration Length of Neutrons in an Infinite Lattice, Third Report, AERE T/R 239, (1958).
(4) Behrens, D. J., The Migration Length of Neutrons in a Reactor, Fourth Report, AERE T/R 877, (1952).

2. Theory of Measurements

The exponential measurements of the nonmultiplying, anisotropic lattices were interpreted by the equation

$$L_z^2 = \left(\kappa_z^2 - \frac{L_r^2}{L_z^2} B_r^2\right)^{-1}$$

where κ_z is the relaxation length of the neutron flux in the vertical direction, B_r^2 is the radial buckling of the exponential tank, and L_r^2 and L_z^2 are the radial and vertical diffusion areas, respectively. Values of κ_z and B_r^2 were obtained from the vertical and radial flux distributions in the exponential facility. The ratio L_r^2/L_z^2 was determined by Behrens'[1] theory. Although the work was intended, in part, as a check on Behrens' theory, only minor uncertainties were introduced by the use of this theory to determine L_r^2/L_z^2 because κ_z^2 was in all cases much larger than B_r^2.

3. Experimental Results

The experimental results are shown in Figure 11. An almost linear relationship is obtained between the measured L_z^2's divided by the function $(1 + \Phi)^2$, where Φ is the void fraction in the lattice, against the moderator-to-fuel volume ratio. Within the experimental error, the points also fall on a calculated curve, which was obtained as follows. The lattice cells were divided into a homogenized Pb-Hg-void region, a pure void region, and a D_2O region. This model was then used for the P-3 calculations of flux distributions, which in turn were used to obtain L^2's not corrected for streaming. Then Behrens' theory was applied to calculate the streaming-corrected L_z^2's. The fact that the calculated L_z^2's fall slightly below the experimental points was expected because the model underestimates streaming effects between individual plates. However, the experiments generally can be regarded as an excellent verification of Behrens' theory as applied to streaming in gas-cooled fuel assemblies of the type considered here.

B. ENGINEERING - SRL HIGH PRESSURE HEAT TRANSFER LOOP

Fifty per cent of the construction of the high pressure heat transfer loop, in which flow instability of boiling systems will be investigated, was completed at Savannah River Laboratory by the end of December. All components except the pump, valves, and flow meters were received. Shipment of the remaining parts is expected early in January. A schematic diagram of the loop, which is designed to operate at a pressure of 1700 psi and a temperature of 315°C, is shown in Figure 12.

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(1) Behrens, D. J., op. cit.

The heat exchanger was mounted, and the cooling water piping installed and tested. The quench tank, pressurizer, and most of the primary piping were installed. Detailed design of the test section was completed and construction started. Checkout of the loop (i.e., stresses, weld integrity, hydrostatic test) and operation are scheduled early in January.

The major experimentation in this high pressure loop will be directed toward the problems of flow instability in boiling systems. Some work on heat transfer burnout limits is also planned. Present plans are to force water through electrically heated tubes of 0.25 to 0.75-inch diameter with maximum lengths of 18 feet. Later, tubes will be connected in parallel to common plenums. Power will be pulsed and plenum pressure will be pulsed. Flows, temperatures, and pressures will be observed. This work will be integrated with the other boiling heat transfer experiments that will be conducted at Columbia University (DP-425, pages 14-16).

C. REACTOR FUELS AND MATERIALS

The chief objective of the program on reactor fuels and materials is the development of a low cost natural uranium fuel, of oxide or metal, that can withstand the exposures and temperatures contemplated for D_2O-cooled-and-moderated power reactors. The work on uranium metal tubes is being conducted at Nuclear Metals, Inc., where the immediate emphasis is on the coextrusion of Zircaloy-clad tubes of various core compositions for irradiation tests. The major effort on oxide fuel is at the Savannah River Laboratory, where experimental studies of a cold swaging process for direct mechanical compaction of Zircaloy-clad elements of oxide are underway. The progress of these programs is reviewed in this section.

1. Uranium Metal Tubes for Irradiation Tests

a. Unalloyed Uranium for VBWR Tests

The four Zircaloy-clad natural uranium tubes, extruded as prototypes for the 3% enriched tubes that will be irradiated in the Vallecitos Boiling Water Reactor, were evaluated to permit selection of the preferred end seal design. Preparations for the fabrication of these tubes were described in DP-445. The principal data concerning the end shapes of the four experimental billets are summarized in the following table.

	Specification	No. 64	No. 65	No. 66	No. 67
Over-all Core Length, inches	45.5 max.	40.1	40.6	40.4	39.2
Uniform Core Length, inches[(a)]	25-30	27.2	27.2	29.6	27.1
Core Taper, inches					
Front	6-1/4 max.	7.0	7.0	5.6	5.2
Rear	--	5.9	6.4	5.1	6.9
Minimum Clad Thickness, inch[(b)]					
Outer - Front	0.010	0.015	0.020	0.021	0.023
Middle	0.012	0.027	0.030	0.027	0.027
Rear	0.010	0.018	0.030	0.016	0.027
Inner - Front	0.010	0.022	0.026	0.028	0.027
Middle	0.012	0.012	0.016	0.017	0.018
Rear	0.010	0.030	0.026	0.030	0.028

(a) Defined as the section having a core wall thickness at least 90% of that specified

(b) Measured by autoradiography and adjusted to allow for subsequent etching

These data show that the outer cladding over the uniform core section of each tube varies only slightly from the nominal 0.030-inch thickness, but thins somewhat near the front and rear ends. In contrast, there is essentially no thinning of the inner cladding near the core ends, but some thinning in the mid-section of the tube. This latter effect was not observed in previous extrusions and is not understood at present. All tubes, however, meet the clad thickness specifications. The tube surfaces are of good quality, except for moderate scoring of Tube 64, which resulted from slight pickup on the extrusion tools. All tube samples that were tested for bond strength gave satisfactory results.

None of the four end shape designs may be rejected because of major defects. Therefore, the choice of a single design for the enriched tubes depends on other considerations. The billet designs for Tubes 64 and 65 resulted in front tapers that are slightly longer than specification, and somewhat more thinning of the inner clad at the middle of the tube. The billet design for Tube 67 requires a greater length of uranium core stock, a disadvantage because of the limited present supply of 3% enriched uranium. This design also involves more difficult and complex machining of billet end shapes, thus increasing the risk of machining errors. Accordingly, the billet design for Tube 66 was chosen for the enriched tubes. This design incorporates 6-degree front and rear compensation angles; 10-degree, 1/8-inch corner chamfers; and a single 1/8-inch annular tooth on the front and rear of the core to provide interlocking with the end seals.

The 3% enriched uranium has been cast. Subsequent processing steps refined the grains of the casting. Billet cores for three tubes will be machined from this material.

b. New Alloy Systems

Postextrusion processing has continued on the Zircaloy-clad U - 1.5 w/o Mo alloy tube that was produced for irradiation testing at the Savannah River Plant. In preparation for heat treatment, the tube was pickled to remove the copper extrusion jacket and autoradiographed to determine the clad thickness. These data indicate some thinning of the cladding near the ends of the core, with the most severe thinning occurring on the inside near the rear end taper. In this location the thinnest cladding, after the tube has been etched in the usual manner, will be about 0.0083 inch. In view of the minimum cladding thickness specification of 0.010 inch for this area and the ±0.001-inch uncertainty of the estimate, this tube is considered to be unacceptable for irradiation testing. Accordingly, processing of this tube has been discontinued and means are being considered for reducing the clad thickness fluctuations in future extrusions of this alloy.

c. Beta-Phase Uranium Alloys

Additional results were obtained in the study of beta-phase uranium alloys that are expected to be dimensionally more stable under irradiation than the less creep-resistant alpha-phase uranium alloys. Four alloys that were selected by small-scale screening tests (DP-445) were prepared in larger quantities for further study of fabrication characteristics, structure, and effects of heat treatment.

Extrusions of two of the alloys, the U - 0.3 w/o Cr and the U - 0.3 w/o Cr - 0.3 w/o Mo, showed the rather unusual metallurgical effect that the phase transformation temperature depended on pressure. This dependency in metals is usually an unimportant second-order effect. The results that demonstrated this phenomenon were as follows. Specimens of the two alloys consisted of alpha-phase material after a one-hour heat treatment at extrusion temperature (i.e., 620°C for the Cr alloy and 590°C for the Cr-Mo alloy) followed by quenching in water to room temperature. However, after extrusion of these alpha-phase alloys (at nominally the same temperature as the heat treatment) the material was essentially all in the beta phase. In addition, when the U-Cr-Mo alloy was extruded at a 30°C higher temperature (i.e., 620°C, in the beta phase), the usual difficulty with beta extrusions was encountered; portions of the extruded rod broke into short lengths as it emerged from the die.

An explanation for the unusual behavior cited in the preceding paragraph has been postulated. When the alloy in the alpha phase passed through the extrusion die, the temperature of the alloy increased because of the energy expended in deforming the material.

However, the alloy did not transform to the beta phase at that time; the extruded rod did not break into pieces as does most beta material. The increased pressure on the material during extrusion may have caused the transition temperature to be raised. After the material passed through the die and the pressure was decreased, the transition temperature may then have been lowered to its normal value, thereby permitting the material to go into the beta phase. The combined effects of cooling rate and alloy content then stabilized this phase.

Verification of the stability of the beta phase of the two alloys discussed above and two other alloys, U - 0.4 w/o Al - 0.4 w/o Si and U - 0.3 w/o Cr - 0.4 w/o Si, was obtained through further heat treatment and aging studies. The results now indicate that the U - 0.3 w/o Cr - 0.3 w/o Mo alloy, extruded at 620°C and gamma-solution heat treated, provides the most stable beta phase of the alloys studied. Further study of the fabrication properties of this alloy, however, are required.

2. Fuel Elements of Uranium Oxide

a. Swaged Tubes

Evaluations of the tubular specimens of cold swaged uranium oxide that were described in DP-445 showed that the test pieces are suitable for irradiation testing in a Savannah River reactor. These specimens will be the first swaged tubes of oxide to be irradiated anywhere. The specimens were fabricated by loading crushed, fused oxide into the annulus formed by two coaxial tubes of 0.022-inch-thick Type 304 stainless steel and then cold swaging the assembly over a hardened steel mandrel. Twelve 2-foot-long specimens were cut from six tubes that were swaged in this manner. Stainless steel fittings were welded into each end of the tubes. Seven specimens were examined destructively; irradiation of the remaining five specimens will start next month.

Seven specimen tubes were examined destructively to study packing density, oxygen-to-uranium ratio, oxide penetration into the sheathing, and dimensional variations. The packing densities for these tubes ranged from 88 to 89% of theoretical, which was about 2% less than the densities obtained in previously fabricated tubes. The density variation along the tubes was 0.4% or less. The oxygen-to-uranium ratio for four tubes filled with oxide from the Norton Company averaged 2.013. The O/U ratio was 2.025 for a single tube containing Spencer oxide.

Dimensional control was satisfactory. Outer and inner diameters averaged 2.137 and 1.457 inches, respectively. Eccentricity of one sheath with respect to the other averaged 0.009 inch, with a maximum value of 0.015 inch. Cladding thickness for inner and outer sheaths averaged 0.017 and 0.021 inch, respectively, with minimum values of 0.015 and 0.019 inch. The maximum penetration of oxide into the sheaths, determined metallographically, was 0.005 inch for the inner sheath and 0.004 inch for the outer sheath. Thus, the minimum cladding at any point may be as low as 0.010 and 0.015 inch for the inner and outer sheaths, respectively. To investigate the behavior of the swaged oxide tubes, in the event that irradiation were to cause the release of large amounts of fission gases within the oxide, the annulus of one tube was pressurized with helium in increments of 50 psi. The inner sheath buckled when a pressure of 1200 psi was reached. Deformation measurements indicated that the inner sheath deformed elastically until buckling occurred. Buckling was accompanied by rupture of the inner weld at the gas inlet end of the tube. The 1200-psi buckling pressure was calculated to be equivalent to the pressure that would be caused by the relase of 100% of the fission gases in oxide that was irradiated to 9340 MWD/ton of UO_2, if the gases were uniformly distributed within the void spaces of the lattice at a temperature of 800°C. This pressure is ten times greater than the pressure that is computed to be required to buckle the inner sheath, if the sheath were not restrained by the UO_2 core.

b. Swaged Rods

Zircaloy-2-clad rods of fused UO_2 are also being prepared by swaging for irradiation tests in an SRP reactor. These tests are designed to give additional information on the effects of central temperature, exposure, oxygen content, and method of preparation on the in-pile behavior of fused uranium oxide fuel. The rods will be enclosed in lead-insulated, stainless steel containers. The purposes of the containers are to raise the surface temperature of the rods to the range of interest for power reactor application and to provide secondary containment of fission products in case of failure of the Zircaloy cladding.

Several grades of uranium oxide were selected in order to provide different oxygen-to-uranium ratios in the test specimens. The behavior under irradiation of fused UO_2 from the Norton Company, fused UO_2 from the Spencer Chemical Company, and sintered UO_2 from the Hanford Atomic Products Operation will be compared. The oxygen-to-uranium ratios of the materials to be tested are as follows:

Oxygen-to-Uranium Ratios of the UO_2 Selected for the Swaged Rod Irradiations

UO_2 Type	O/U
Norton fused	2.004
Norton fused	2.036
Norton fused	2.060
Spencer fused	2.008
HAPO sintered	2.007

After the oxide was crushed, the particle size distribution of each type of oxide was adjusted to eliminate particle size as a variable. Swaging of the specimens is scheduled for early next month.

c. Vibratory Compaction

Tests were conducted at the Dayton T. Brown Company to investigate the possibility of using vibratory compaction to increase the packing density of tubular fuel elements filled with uranium oxide. This increased density is desired in order to reduce the sheath wrinkling or cracking that frequently occurs when loosely packed elements are swaged. Six elements, 3 to 4 feet long and filled with fused-and-crushed UO_2, were vibratory compacted from approximately 62% of theoretical density to 70-76% of theoretical density. The most dense packing was obtained when the vibration was at the lowest resonant frequency; for these tubes, this resonant frequency was between 255 and 310 cycles per second. An effort will be made to transport these tubes, without loss of densification due to handling during transit, to the Savannah River Laboratory for swaging studies.

Six additional tubes were prepared for a second series of compaction tests. These tubes are 7 feet long.

3. Stainless Steel - Zircaloy Joints

One of the chief problems that must be faced in the design of pressure tube power reactors is that of joining pressure tubes of Zircaloy to coolant distributors of stainless steel. It is believed that a metallurgical joint is potentially better suited to this application than is a mechanical joint. Not only is less space required for a metallurgical joint, but also it should be more resistant to the detrimental effects of repeated cycles of temperature and pressure. Therefore, a program of fabrication development and testing of metallurgically bonded joints has been started. The status of the development work was reviewed in DP-445; progress during December 1959 is reported below.

a. NMI Fabrication

The manufacture of metallurgically bonded tubular joints of stainless steel and Zircaloy continued at Nuclear Metals, Inc. The several phases of the problem under investigation include scaleup to full-size joints, mechanical properties, and corrosion resistance.

To provide a demonstration joint of the full size required for reactor pressure tubes, three joints were fabricated. These joints were patterned after previous small-scale tests, and destructive examination of one joint revealed that similar results were obtained with both sizes.

Studies continued of the mechanical and corrosion properties of various joints. The favorable results reported last month for a corrosion test of a joint in 250°C water were not sustained in further testing at 300 and 360°C. These tests showed preferential attack in the area of the Zircaloy-to-stainless-steel bond. In view of the good corrosion behavior of joints that had been fabricated by a slightly different process, emphasis is being placed on obtaining satisfactory bonding and mechanical performance of the latter type of joint. An extensive series of small-scale tests is planned; such parameters as fabrication temperature, cooling rate, and type of austenitic stainless steel will be varied.

b. Facility for Testing Zircaloy-to-Stainless-Steel Joints

Construction of a small loop for cyclically testing Zircaloy-to-stainless-steel tubular joints at a maximum temperature of 260°C and a maximum pressure of 1000 psig was completed at the Savannah River Laboratory. The new facility, which was designed to test up to four joints simultaneously, operates in parallel with the existing facility that is used for cyclic tests of gasketed joints. A drawing of a typical test unit and test joint is shown in Figure 13. Testing of the following joints is scheduled to start early during the next report period.

Description of Zircaloy-to-Stainless Steel Tubular Joints

Test Joint No. 1: Fusion-bonded tubular joint between Zircaloy-2 and 400 series stainless steel. The joint is about 3.2 inches OD with a wall thickness of 0.21 inch.

Test Joint No. 2: NMI-fabricated tubular joint between Zircaloy-2 and 347 stainless steel. The joint is about 1.8 inches OD with a wall thickness of 0.13 inch.

Test Joints No. 3 & 4: Diffusion-bonded tubular butt joint between Zircaloy-2 and 304 stainless steel. Each joint is about 2.0 inches OD with a wall thickness of 0.11 inch.

Test Joint No. 5: Similar to Joint No. 2. The OD of the joint is about 3.85 inches and the wall thickness is 0.13 inch.

Test Joint No. 6: Same as Joints No. 3 & 4 but with an OD of about 5.4 inches and a wall thickness of 0.19 inch.

D. IRRADIATION TESTING

1. Metal Fuel Elements

The diffusion heat-treated tube of U - 2 w/o Zr clad in 0.015-inch-thick Zircaloy that was being irradiated in a Savannah River reactor at relatively low temperature and pressure failed after achieving a modest exposure. The first indications of failure were a measured 2% decrease in coolant flow and an accompanying 2.7°C increase in coolant temperature over a period of 25 minutes. Analyses of a water sample taken from the assembly during discharge from the reactor showed the presence of much radioactivity in the water, thereby confirming the failure of the tube.

Thirteen Zr-clad tubes have been irradiated at relatively low temperature and pressure at the Savannah River Plant; exposures in the range of interest to power reactors were achieved in some of the tests. Five of the tubes contained cores of unalloyed uranium, and the remainder were of U - 2 w/o Zr. Most of the unalloyed tubes were clad with zirconium, whereas all of the U - 2 w/o Zr tubes were clad with Zircaloy-2. The operating conditions and exposures during some of these irradiation tests are presented in classified reports that also discuss the test results.[1,2] In general, the irradiation conditions for the U - 2 w/o Zr tubes were more severe; the metal temperature was higher and the cladding thinner, than for the unalloyed tubes. The tube mentioned above was the third U - 2 w/o Zr tube that failed during irradiation.

Although the Savannah River data are useful in comparisons of alternative fuel compositions, they are not representative of power reactor conditions. In at least two respects, the test conditions may have been unfavorable. The restraining force of coolant pressure, which supplements cladding restraint in opposing swelling of the core, was much lower than in a power reactor. The cladding temperature also was lower, with possible detrimental effect on the ductility of the zirconium. In Canadian irradiations of Zircaloy-clad rods of U - 3.9% Si at power reactor temperatures and pressures, failures did not occur at cladding strains that were 2 to 3 times greater than those of the failed elements at Savannah River (~ 3% vs. 1 to 1.5%). The Canadian results are encouraging, but a definitive evaluation of the metallurgical limits on exposure of metallic elements must await much more irradiation data at power reactor conditions.

Currently, the only other natural uranium metal fuel elements undergoing irradiation at the Savannah River Plant are four lead-insulated slugs

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(1) Gleaves, W. H., Irradiation of Natural Uranium Tubes Clad with Zirconium, E. I. du Pont de Nemours & Co., Inc., DP-244, October 1957 (Secret).

(2) Olcott, R. B., et al., Irradiation of Zr-Clad Uranium Metal Tubes for Power Reactors, E. I. du Pont de Nemours & Co., Inc., DP-404, September 1959 (Secret).

(see DP-445). However, preparations are in progress for the testing of other uranium metal alloys as well as other pure uranium fuel elements.

2. Effects of Irradiation on Zircaloy-2

The first six Zircaloy-2 specimens that were exposed in the NRX reactor at Chalk River (DP-445) were discharged and are scheduled to be shipped to Savannah River Laboratory for measurements of stress relaxation. The specimens were strained before irradiation so that the initial stress at the operating temperature would have been from 11,000 to 15,000 psi. Preliminary estimates of the temperature and exposure are in the range of 280°C and 10^{19} fast nvt. More exact data are to be transmitted from AECL to SRL when the specimens are shipped.

An additional six specimens were fabricated, stressed, and delivered to the NRX reactor for exposure in the next two cycles. The initial stress levels are about the same as in the first six specimens.

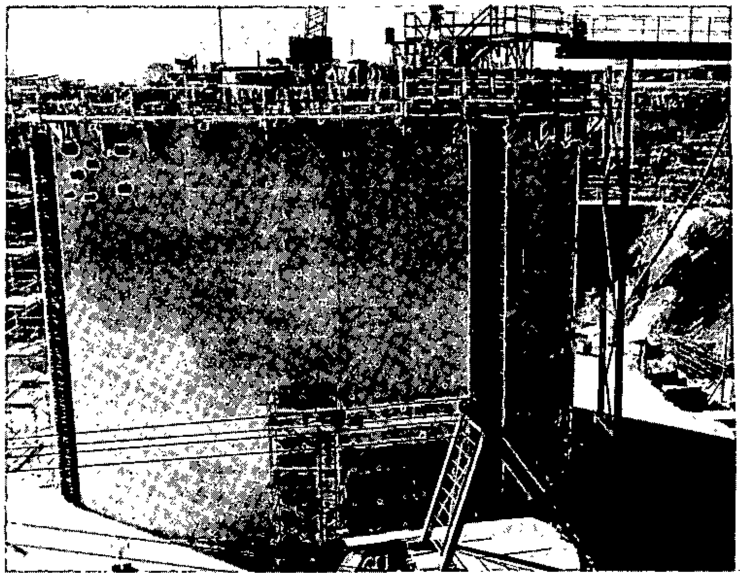


FIG. 1 STATUS OF HWCTR CONSTRUCTION
End of November 1959

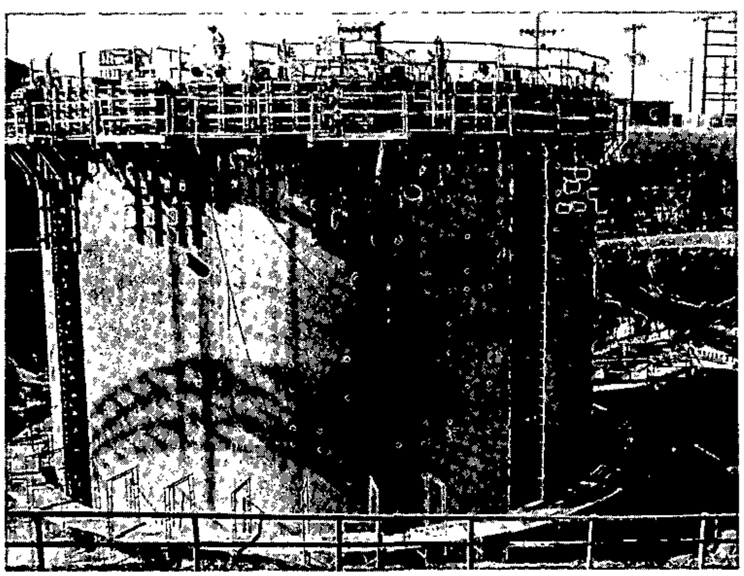


FIG. 2 STATUS OF HWCTR CONSTRUCTION
End of December 1959

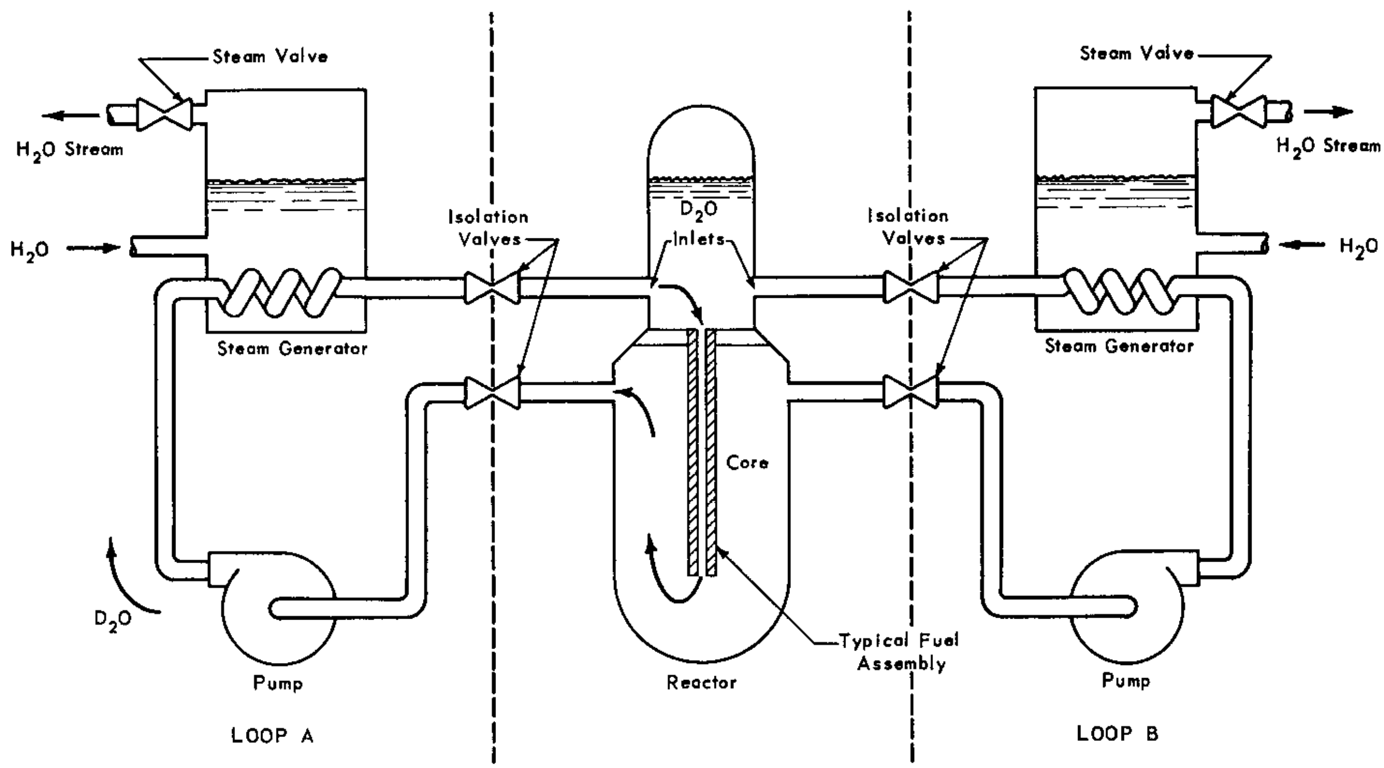


FIG. 3 SCHEMATIC DIAGRAM OF HWCTR FLOW SYSTEM

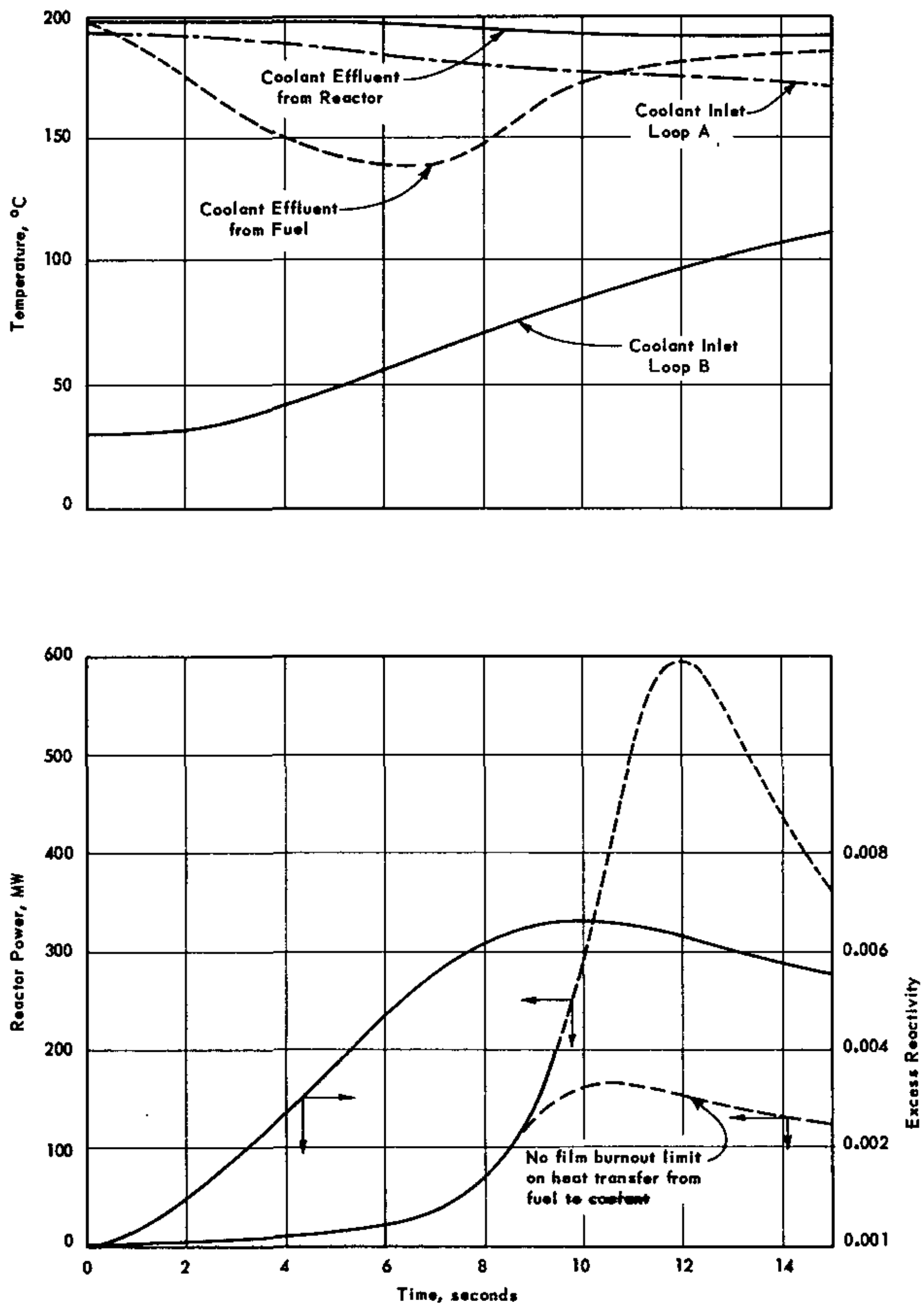


FIG. 4 "STANDARD" COLD WATER ACCIDENT IN THE HWCTR
(Rate of heat transfer from fuel to coolant limited by film burnout of the fuel)

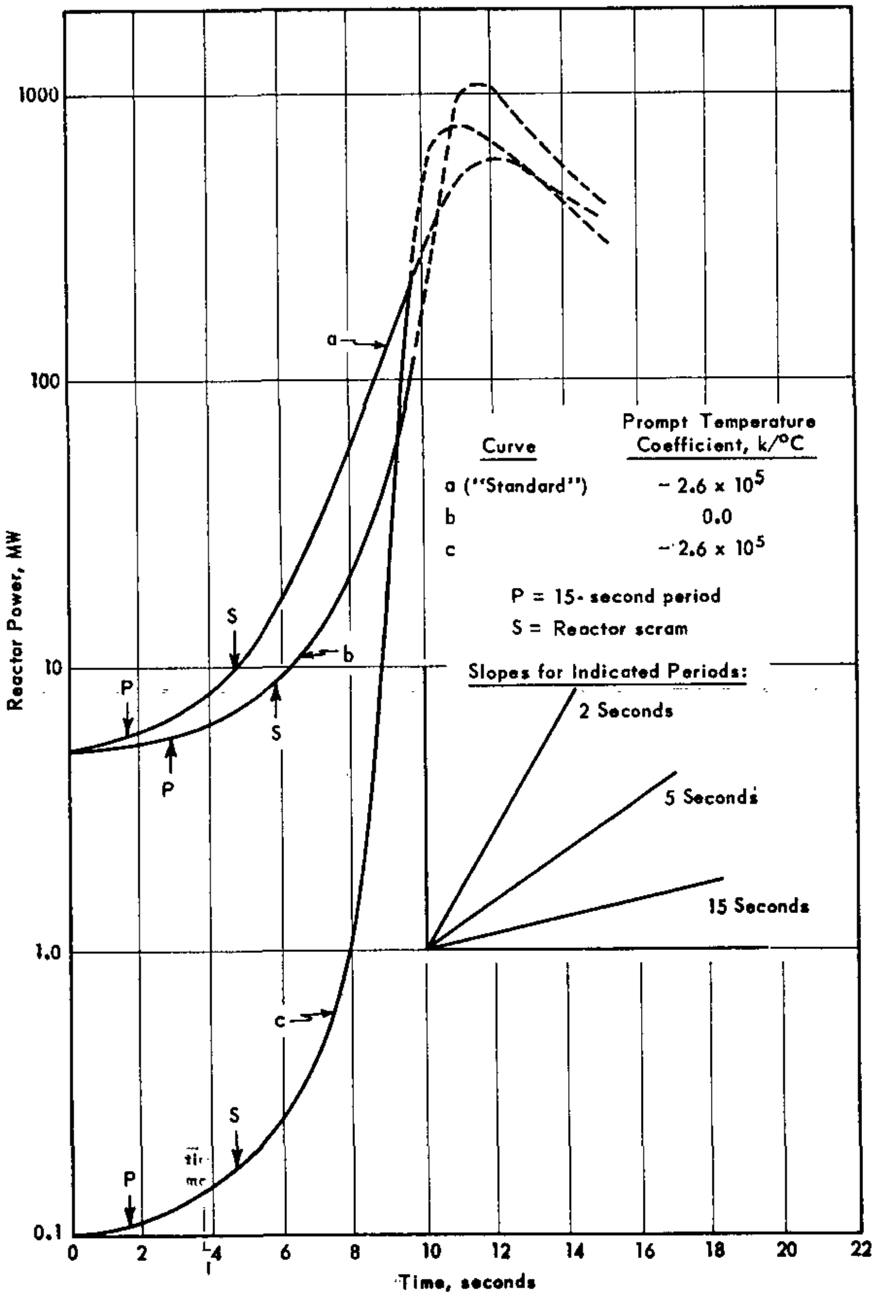


FIG. 5 COLD WATER ACCIDENTS IN THE HWCTR
(Rate of heat transfer from fuel to coolant limited by film burnout of the fuel)

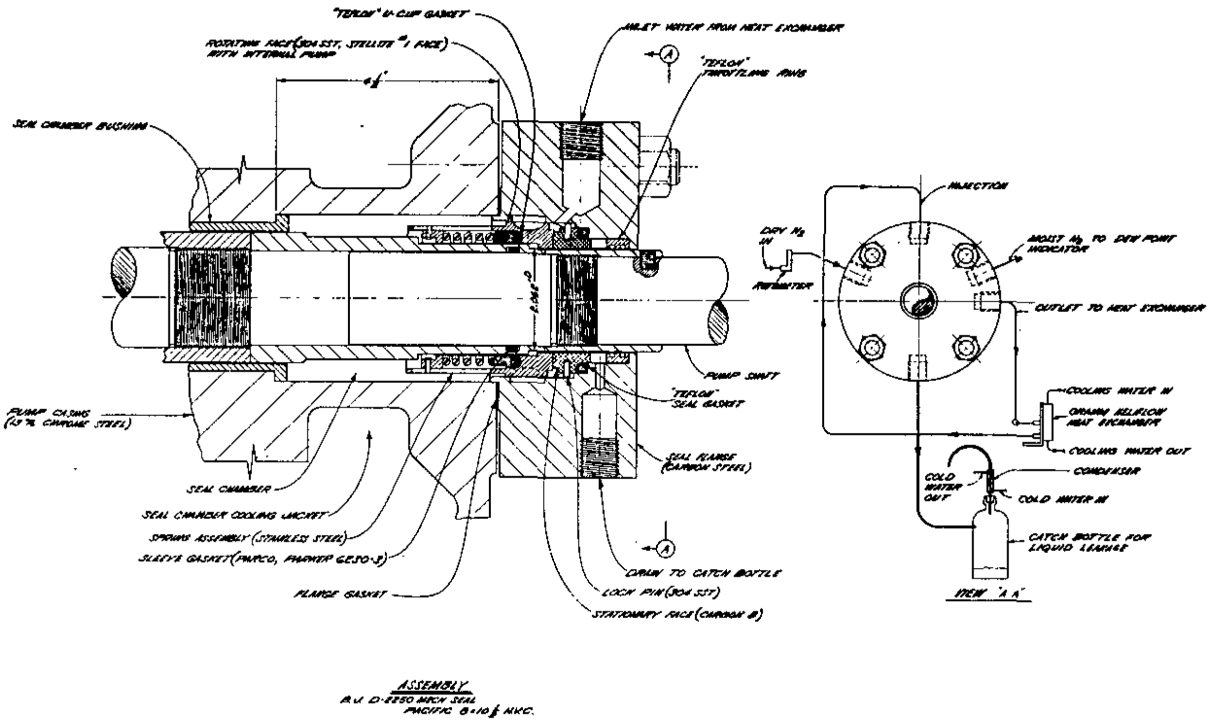


FIGURE 6 – MECHANICAL SEAL FOR PROCESS WATER PUMP AND LEAKAGE COLLECTION SYSTEM

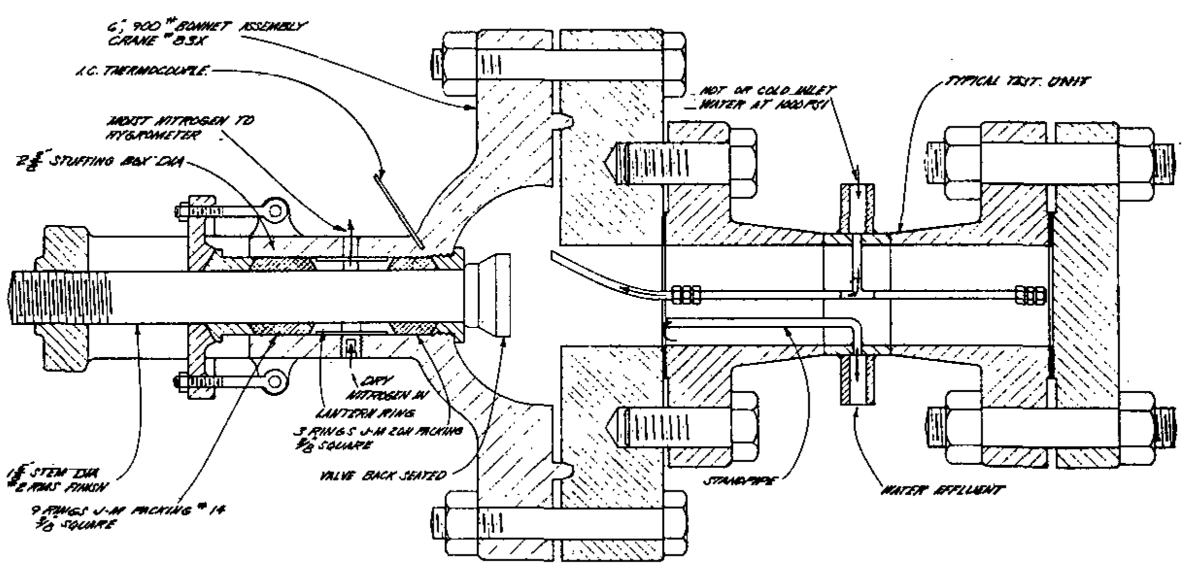


FIG. 7 VALVE STEM LEAKAGE TEST: 6-INCH, 900-LB GATE VALVE

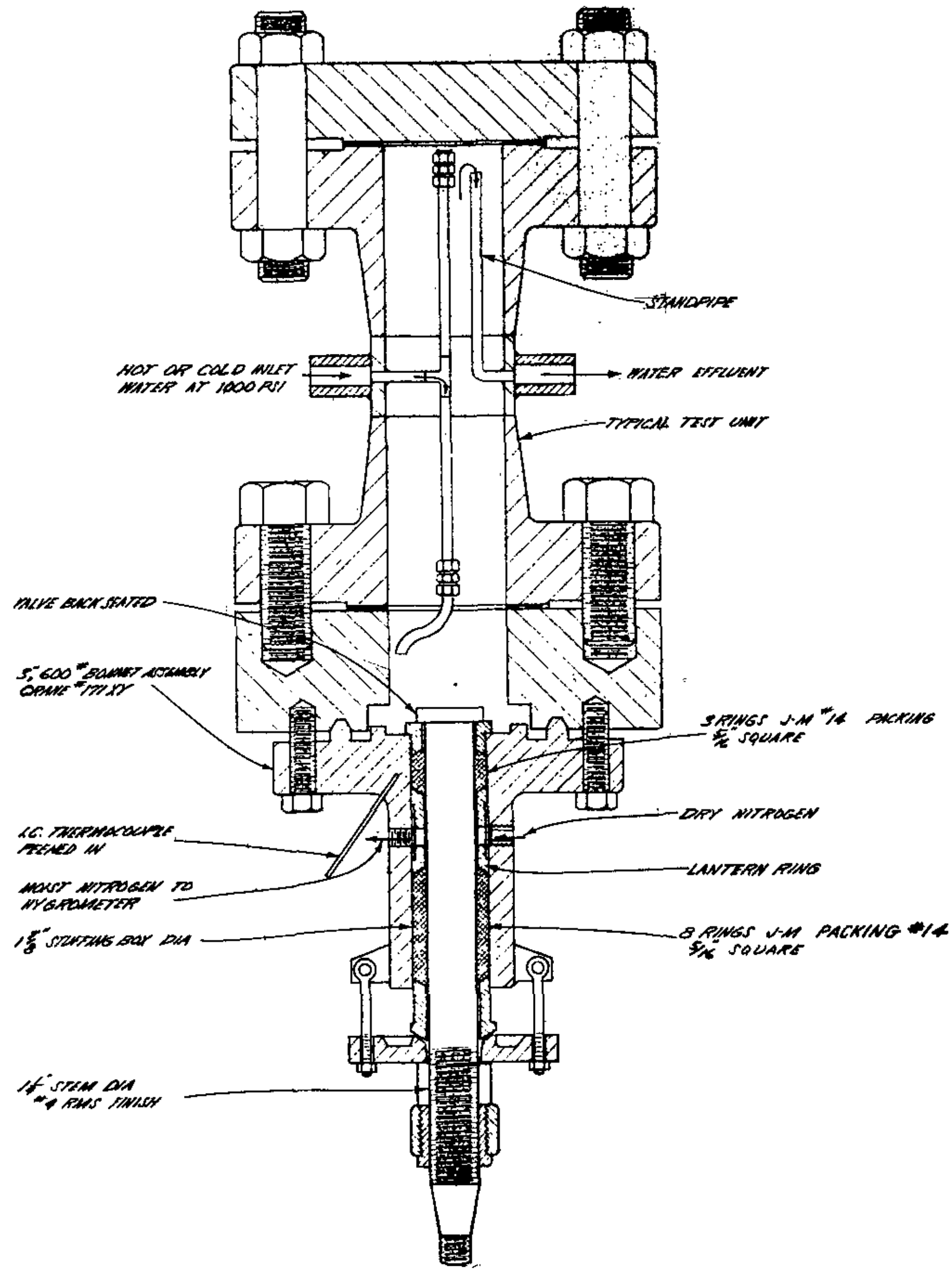


FIG. 8 VALVE STEM LEAKAGE TEST: 3-INCH, 600-LB GLOBE VALVE

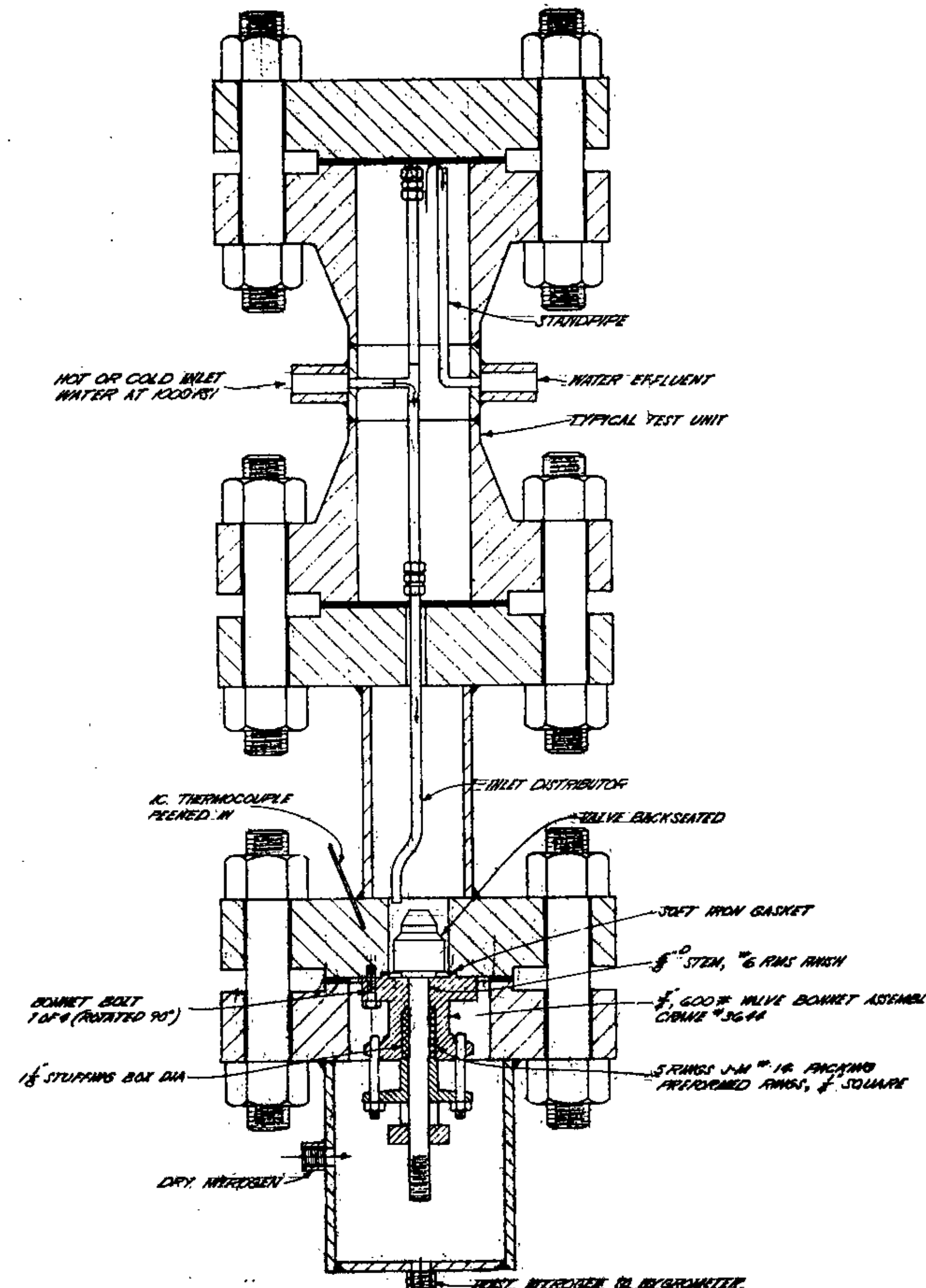
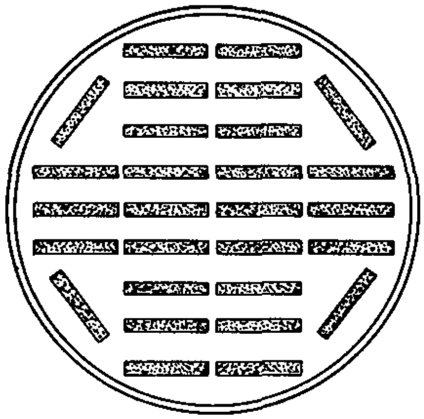
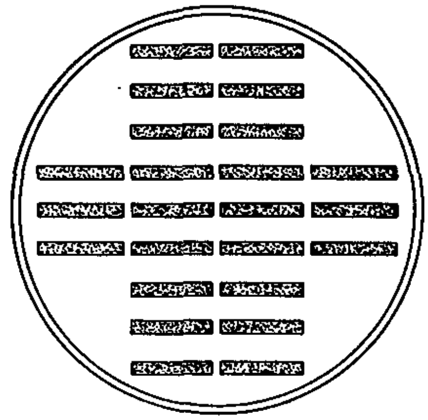


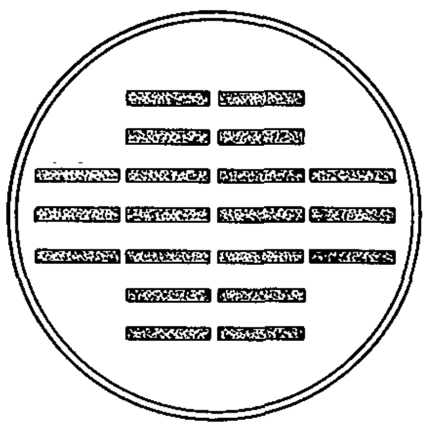
FIG. 9 VALVE STEM LEAKAGE TEST: 3/4-INCH GLOBE VALVE



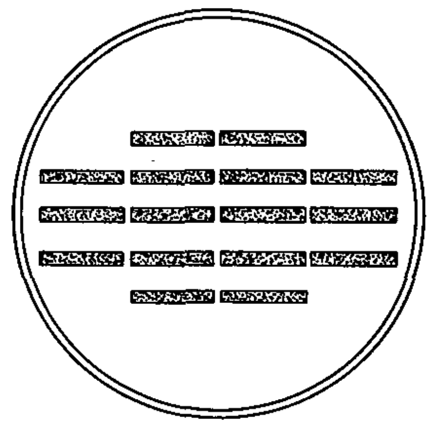
28–Plate Assembly



24–Plate Assembly



20–Plate Assembly



16–Plate Assembly

Plate Dimensions, inches

Width	1.188
Thickness	0.180
Length	57
Spacing	0.38 between plate faces

Alloy Composition: 96.6 w/o lead
3.4 w/o mercury

FIG. 10 EXPERIMENTAL ASSEMBLIES OF GAS-COOLED PLATES

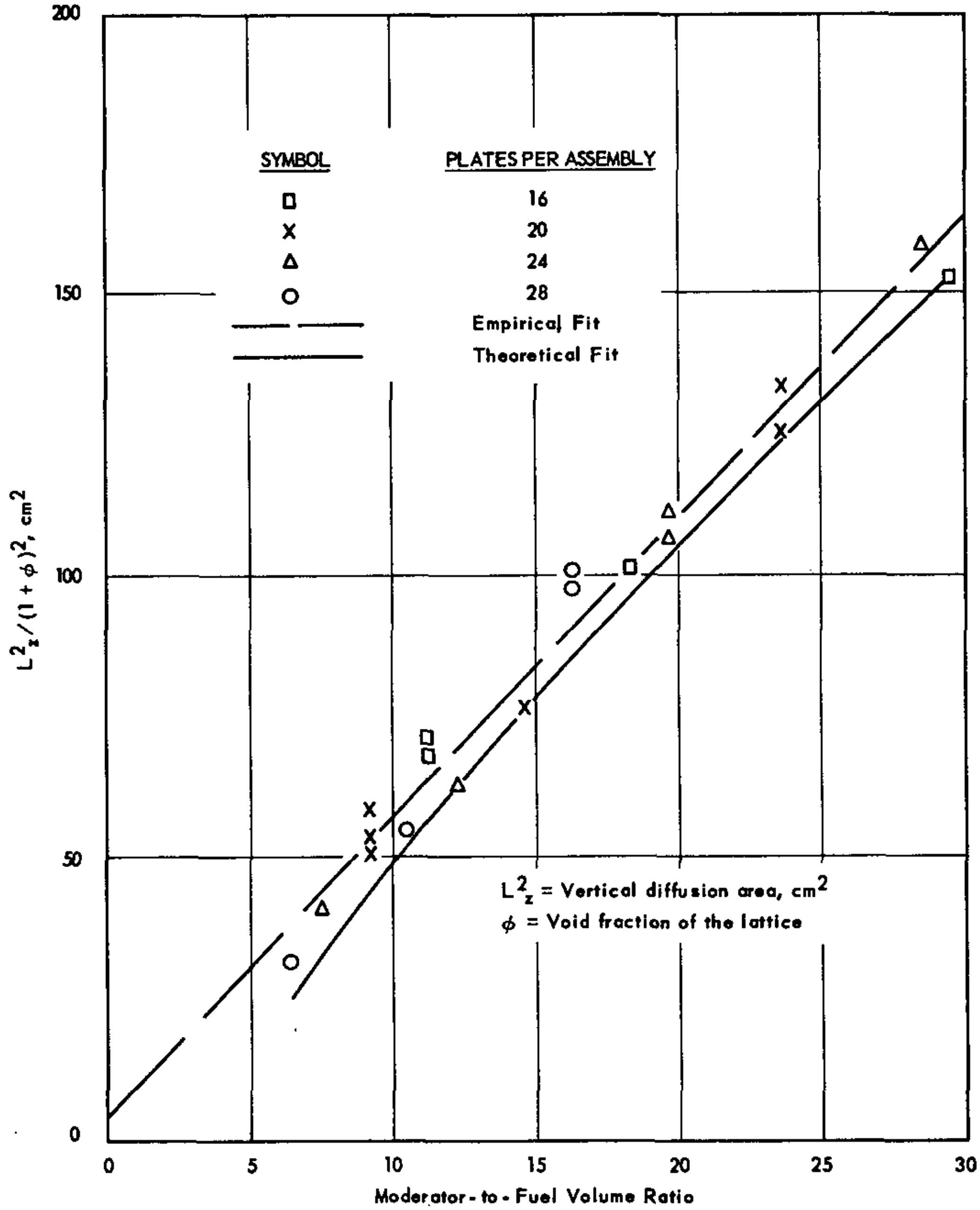


FIG. 11 THERMAL DIFFUSION AREA OF Pb-Hg LATTICES CONTAINING GAS CHANNELS

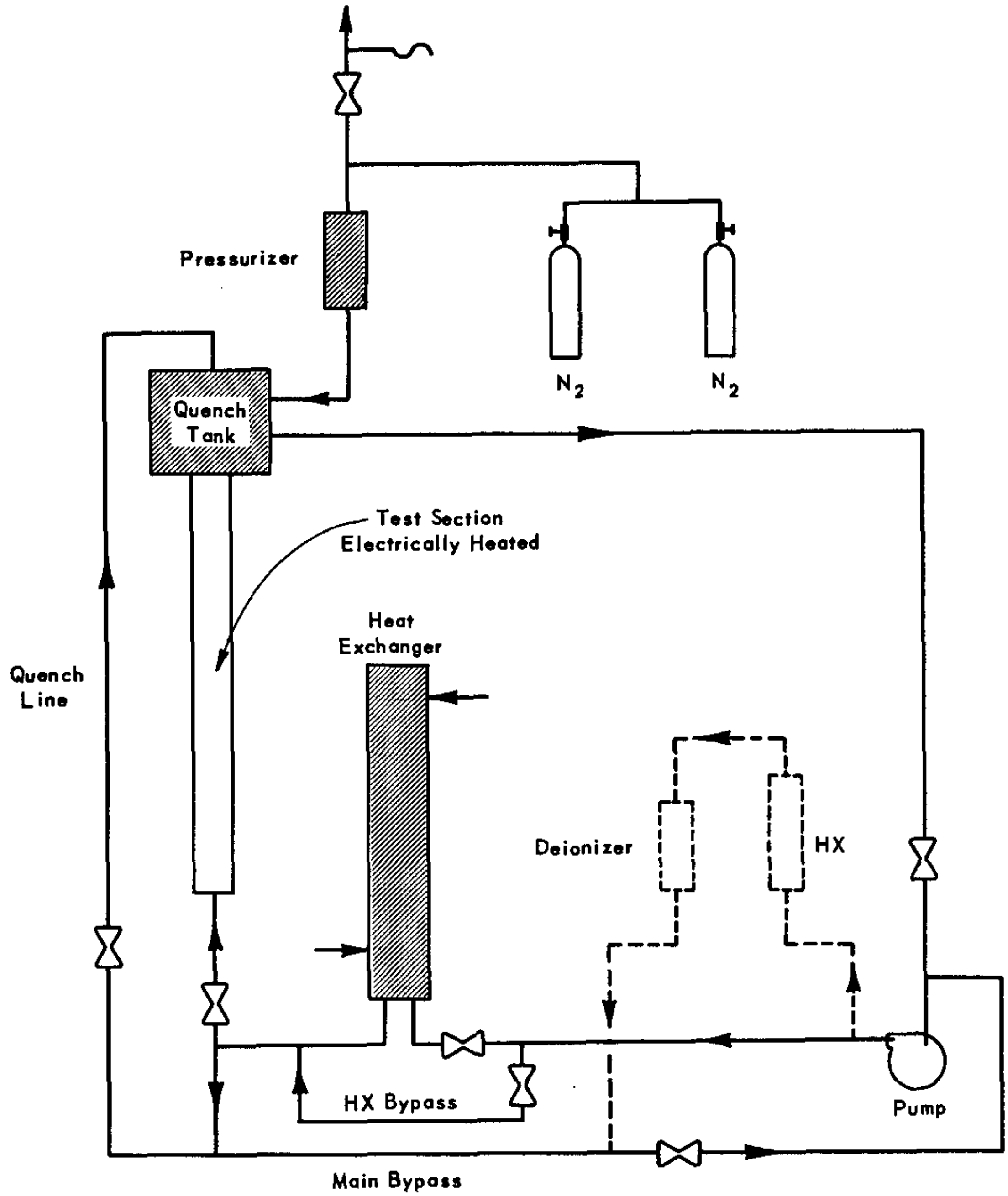
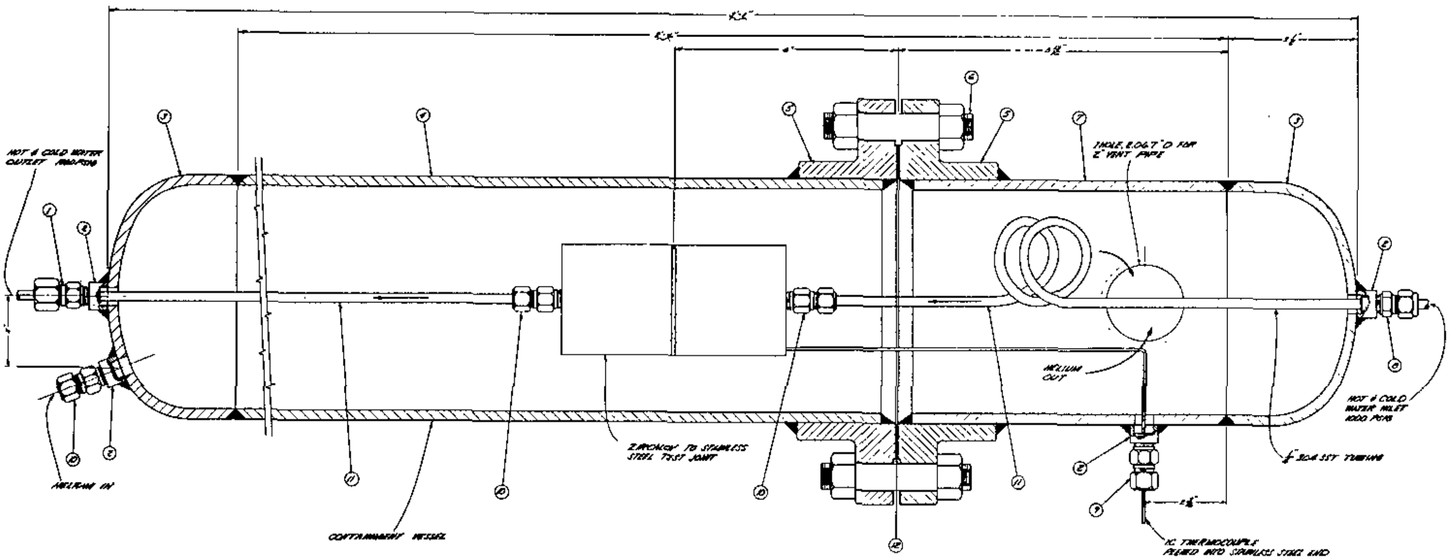


FIG. 12 FLOW LOOP FOR BOILING WATER EXPERIMENTS

Design Pressure 1700 psi
Operating Temperature 315° C



1 303 stainless steel "Conax" packing gland, 1/4" tube to 1/4" pipe, PG-2, "Lava" sealant
2 Carbon steel 1/4" 3000 lb half coupling
3 Carbon steel 6" Sch 40 pipe cap
4 Carbon steel 6" Sch 40 pipe
5 Carbon steel 6" 150 lb slip-on flange
6 3/4" stud, 4½" long with nuts
7 Carbon steel 6" Sch 40 pipe
8 316 stainless steel "Swagelok" fitting 1/4" pipe to 1/4" tubing, 400-1-7, drilled through
9 303 stainless steel "Conax" packing gland 1/4" pipe to 1/8" tubing, PG-2 "Teflon" sealant
10 316 stainless steel "Swagelok" fitting 1/4" pipe to 1/4" tubing 400-1-4
11 305 stainless steel 1/4" OD x 0.035" wall, stainless tubing (length to suit)
12 1/16" asbestos gasket

FIG. 13 FACILITY FOR TESTING ZIRCALOY-TO-STAINLESS-STEEL JOINTS