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HEAVY WATER MODERATED POWER REACTORS

PROGRESS REPORT SEPTEMBER 1961

Technical Division

Wilmington, Delaware

E. I. du Pont de Nemours & Co.
Explosives Department - Atomic Energy Division
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HEAVY WATER MODERATED POWER REACTORS

Progress Report
September 1961

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ABSTRACT

At the end of September 1961, construction of the Heavy Water Components Test Reactor was about 90% complete. Thirty-two compacted tubes of crushed, fused uranium oxide in Zircaloy sheaths were fabricated for irradiation tests and destructive evaluation. Irradiation tests of the tubes were started in the Vallecitos Boiling Water Reactor and at Savannah River. The fabrication process for the tubes included steps designed to exclude hydrogenous material from the oxide cores, thereby eliminating the probable cause of sheath failures in previous irradiations. Additional experimental data on heat transfer burnout of tubes in subcooled water at pressures of about 100 to 1000 psi show that the burnout heat flux is not affected significantly by pressure in this range. The data were correlated in terms of water velocity and subcooling.

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INTRODUCTION

This report records the progress of the du Pont development program on heavy-water-moderated power reactors fueled with natural uranium. The goal of the program is to advance the technology of these reactors so that full-scale plants in which they are employed can produce electricity at costs that are competitive with those of fossil-fueled stations. The present effort is divided into two main categories: (1) the development required for the successful construction and operation of the Heavy Water Components Test Reactor (HWCTR), a facility for irradiating fuel elements and other reactor components at power reactor conditions, and (2) the development of the technology required for full-scale power reactor systems.

SUMMARY

At the end of September, construction of the HWCTR was about 90% complete. The installation of reactor internal parts and the rod drive system is in progress. Most of the major equipment for the facility is at the site. Undelivered parts for the rod drives are the most critical item in the completion schedule.

Thirty-two experimental fuel tubes of compacted uranium oxide were fabricated with the objective of evaluating process modifications aimed at eliminating the cause of Zircaloy sheath failures experienced in earlier irradiations of oxide tubes. Moisture or other hydrogenous material in the oxide is believed to have been responsible for Zircaloy sheath failures in previous irradiation tests of oxide tubes. Two of the experimental tubes were placed under irradiation in the Vallecitos Boiling Water Reactor, and six are being irradiated at Savannah River. One-half of the tubes were compacted in Zircaloy-4 sheaths by vibratory compaction and subsequent swaging; and one-half by vibratory compaction alone. Steps incorporated in the fabrication process to reduce moisture in the present tubes were apparently successful. Moisture contents of the finished tubes were an order of magnitude lower than in tubes fabricated previously.

The fabrication of coextruded tubes of unalloyed uranium for irradiation tests in the HWCTR continued without difficulty. Eighteen 1-inch-diameter tubes with Zircaloy-4 cladding have been extruded successfully; postextrusion processing of these tubes is underway. Extrusion of a like number of tubes of 2-inch diameter is scheduled for October. One tube of U - 0.3 wt % Al - 0.5 wt % Si was also extruded successfully.

Additional measurements of heat transfer burnout for subcooled water confirmed earlier indications that variation of system pressure between 150 psia and 1000 psia has little or no effect on the burnout heat flux. The experimental data, comprising 19 measurements with electrically heated tubes of 0.3-inch to 0.9-inch diameter, were correlated by equations containing only coolant velocity and subcooling as independent variables.

DISCUSSION

I. THE HEAVY WATER COMPONENTS TEST REACTOR (HWCTR)

The HWCTR, a D_2O-cooled-and-moderated test reactor, is being built at the Savannah River Plant for the purpose of testing candidate fuel assemblies and other reactor components under irradiation conditions that are typical of large, heavy-water-moderated power reactors. Descriptions of the reactor facility and of the two isolated loops for more specialized irradiations were presented in topical reports[1,2]. Startup of the facility is scheduled for late 1961.

A. CONSTRUCTION STATUS

At the end of September 1961, construction of the HWCTR was about 90% complete. The placement of concrete shield blocks in the annular space around the reactor vessel was completed, and the installation of reactor internal components was started. The rod drive platform is in place, and the reactor top head and head lifting mechanism were attached to the platform. Installation of the rod drive system is in progress. The 27-ft-square construction access opening in the containment shell of the reactor building was welded closed.

Most of the major equipment pieces for the HWCTR have been delivered to the site; the principal exceptions are the fuel transfer coffin, replacement parts of 17-4 PH stainless steel for the rod drive systems (DP-605)*, pumps and associated drives for the isolated coolant loops, and the two Zircaloy bayonets for the loops. The replacement parts for the rod drives are the most critical item at present in the reactor completion schedule.

A recent photograph of the reactor site is presented in Figure 1.

B. NUCLEAR REACTIVITY OF DRIVER ASSEMBLIES

The nuclear reactivity necessary for operation of the HWCTR will be provided by a ring of 24 "driver" fuel assemblies, each consisting of a fuel tube of fully enriched U^{235} alloyed with zirconium, a zirconium housing tube, and a target cruciform of boron stainless steel (see Figure 2). In 1959, the desired boron concentration in the targets was specified tentatively on the basis of critical experiments with a nuclear mockup of a full HWCTR lattice in the Process Development Pile (PDP)[3]. During the present report period, this specification was confirmed by means of a series of PDP experiments in which the reactivities of real driver assemblies were compared with those of the mockup drivers employed in the earlier experiments.

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* All DP reports cited without complete reference are progress reports in this series.

In the more recent tests, critical water heights were measured as two driver mockups were inserted alternately with two actual drivers in central hex positions of a PDP lattice in which the neutron spectrum approximated that expected in the HWCTR. The results are presented in Figure 3 in terms of axial buckling for criticality of the particular PDP lattice as a function of the natural boron content of the targets. On the basis of these results, targets containing 0.34 wt % natural boron, which corresponds to 0.6 gram of boron per foot of target length, were specified for the first HWCTR charge. Drivers containing this amount of boron exhibited the same reactivity as driver mockups containing 0.5 g/ft; the latter provided a satisfactory margin of control in the tests with the full mockup of the HWCTR lattice.

As a separate part of the PDP experiment, a measurement was made of the azimuthal variation of the neutron flux in a driver fuel tube as a consequence of the cruciform target. A series of gold pins at various azimuthal positions on the tube were irradiated in the PDP, and the activation levels of the individual pins were measured. Although some azimuthal variation was observed, there was no correlation with target orientation. The observed ratio of maximum to average flux was 1.03; this ratio will be included in calculations of hot spot factors for the driver assemblies during HWCTR operation.

During the same PDP irradiation, a test was made to determine the feasibility of measuring the axial flux distribution by irradiating wires attached to the target subassembly of the drivers. The PDP irradiation demonstrated that the technique is feasible and that reliable axial distributions can be so obtained. Use of this technique is planned for measuring axial flux shapes in the HWCTR during physics startup tests.

C. MECHANICAL DESIGN OF DRIVER ASSEMBLIES

The mechanical design of one of the end fittings for HWCTR drivers was modified to provide for positive azimuthal positioning of the target cruciform relative to the housing tube on the opposite side of the fuel tube (see Figure 2). The new design precludes positioning a target blade directly opposite a rib on the housing; it thereby increases by about 10% (from 0.8 to 0.9) the factor that is included in computations of burnout heat flux to allow for non-idealities in fuel tube heat transfer. A prototype of the new fitting was incorporated in a driver assembly and was placed under flow test at HWCTR flow and temperature conditions to insure that the fitting does not introduce problems of wear or vibration damage.

II. REACTOR FUELS AND MATERIALS

A. BACKGROUND

One of the most important objectives of the du Pont program on heavy-water-moderated power reactors is to develop fuel elements of natural uranium that can achieve a total fuel cycle cost no greater than 1 mill/kwh in a large power reactor. At present, two types of Zircaloy-clad fuel elements are being developed: coextruded tubes of uranium metal and mechanically compacted tubes of uranium oxide. A few specimens of each type have been irradiated at low temperature in a Savannah River reactor. The present emphasis is on preparation of both metal tubes and oxide tubes for larger scale irradiation tests in the HWCTR. Also scheduled for irradiation in the HWCTR are bundles of swaged uranium oxide rods, which are being procured from a commercial source as a possible alternative to compacted oxide tubes.

B. FUEL ELEMENTS OF URANIUM METAL

1. Fabrication of Irradiation Specimens

Fabrication continued at Nuclear Metals, Inc., on fifteen sets of inner and outer thin-walled tubes of unalloyed uranium clad with Zircaloy-4; these tubes, which are being prepared for irradiation tests in the HWCTR, have the nominal dimensions shown in Figure 4.

By the end of September, a total of eighteen inner tubes (three spares) had been extruded without difficulty and were undergoing postextrusion processing and inspection. Fabrication of the coextrusion billets for the outer tubes was 75% complete, and extrusion of all of the outer tubes was scheduled for October.

2. Cladding Restraint Program

As discussed in DP-625, Nuclear Metals is fabricating a small number of coextruded tubes of unalloyed uranium with various thicknesses of Zircaloy cladding. These tubes will be used in an irradiation program aimed at ascertaining the effectiveness of mechanical restraint in preventing or controlling the swelling of uranium fuel cores during irradiation. The status of the fabrication of the tubes as of the end of this report period is summarized in the table below.

Status of Coextruded Tubes with Various Cladding Thicknesses

Tube Style	Outer Diameter, inches	Outer Cladding Thickness, inch	Inner Cladding Thickness, inch	Fabrication Status
A (tube 1)	2.060	0.060	0.060	Undergoing postextrusion processing.
A (tube 2)	2.060	0.060	0.060	Undergoing postextrusion processing.
B	2.060	0.025	0.040	Complete. Satisfactory.
C	2.060	0.040	0.025	Complete. Satisfactory.
D (tube 1)	1.020	0.060	0.060	Excessive striation. Set aside.
D (tube 2)	1.020	0.060	0.060	Undergoing postextrusion processing.
E	1.020	0.040	0.025	Complete except for cladding thickness verification.

A total of seven tubes have been extruded in an effort to obtain five acceptable irradiation specimens. Two 2-inch-diameter tubes with 0.060-inch cladding were extruded, instead of only one tube, because of the greater fabrication uncertainties associated with this particular type of tube. A second 1-inch-diameter tube with 0.060-inch cladding was made as a replacement for the first such tube, which was set aside because it contained a 0.005-inch-deep striation. Otherwise, the fabrication and postextrusion processing of the tubes have proceeded uneventfully.

3. Fabrication of Uranium Alloy Tubes

Although the emphasis in the du Pont program on uranium metal elements is being placed on unalloyed uranium, a supplementary study is being made of the effect on irradiation behavior of adding small amounts of alloying agents to uranium. The objective is to devise a core composition that is superior to unalloyed uranium in reactor performance and is suitable for a natural uranium reactor. As discussed in previous progress reports, fabrication studies on several promising alloys have been underway at Nuclear Metals. During the present report period, the first attempts were made to fabricate full-size (i.e., 10-ft-long) Zircaloy-clad fuel tubes with cores of these alloys. One coextrusion billet of U - 0.3 wt % Al - 0.5 wt % Si and one of U - 1 wt % Si were fabricated for processing into tubes for HWCTR tests. The sizes of billets were selected to yield tubes having nominal dimensions of 2-inch-diameter thin-walled tubes (see Figure 4). The ternary aluminum-silicon alloy was extruded successfully at 650°C, and is undergoing postextrusion processing. The U - 1 wt % Si billet failed to extrude because an instrument error led to insufficient load being placed on the ram of the extrusion press. This billet will be remade for a later extrusion.

C. Fuel Elements of Uranium Oxide

A total of thirty-two experimental fuel tubes consisting of compacted cores of uranium oxide in Zircaloy-4 sheaths were fabricated at the Savannah River Laboratory, and irradiation tests of some of the tubes were started at the Vallecitos Boiling Water Reactor (VBWR) and at Savannah River. The fabrication of the tubes, the results of non-destructive tests to date, and the irradiation tests are discussed individually below:

1. Fabrication of Compacted Oxide Tubes

With the objective of evaluating improved techniques for fabricating compacted tubes of uranium oxide, thirty-two such tubes were prepared for irradiation tests and for destructive examination. Ten of the tubes contain UO_2 enriched to 5% U^{235}; four of these were shipped to General Electric Co., for irradiation in the VBWR. The remaining twenty-two tubes are of natural UO_2. All of the tubes contain fused UO_2 that was crushed, classified as to particle size, blended, and compacted in Zircaloy-4 sheaths. One-half of the tubes were compacted by vibration and subsequent swaging, and the other half were produced by vibratory compaction alone. Bulk UO_2 densities of the swaged cores averaged 90 to 91% of theoretical; densities of the vibratorily compacted cores averaged 82 to 84% of theoretical. The core lengths for Savannah River and VBWR irradiation specimens were 18 and 36 inches, respectively. Other nominal dimensions of all specimens included an outside diameter of 2.06 inches, an inside diameter of 1.46 inches, and a cladding thickness of 0.030 inch.

Several steps were incorporated in the fabrication process to exclude hydrogenous material from the tubes; the objective was to eliminate the cause of the local hydride attack that caused sheath failures in earlier irradiations of Zircaloy-clad oxide tubes at Savannah River (DP-615). These steps were (1) vacuum outgassing of the oxide at 1300°C prior to loading it into the sheath tubes, (2) loading the oxide in a dry, inert atmosphere, and (3) vacuum drying of the compacted fuel tubes at 200°C before final sealing. To provide a basis for appraising the improved process, some of the oxide tubes are being examined destructively for moisture and gas contents of the UO_2 core and for the hydrogen content of the cladding. Results of analyses of carefully processed samples indicate that the moisture content of the UO_2 cores is less than 10 ppm. By comparison, the moisture in tubes fabricated previously was as much as 100 ppm. The amounts and compositions of the gases extracted from the tubes at 1000 and 1300°C are listed in Table I. The data for enriched oxide reveal that the vacuum outgassing step effected about a tenfold reduction in the gas content of the UO_2. Reliable data are not yet available on the gas content of the original natural UO_2, but preliminary results indicate that the total amount of gas was about the same as in the finished tubes.

The hydrogen content of the Zircaloy-4 sheaths after postfabrication autoclaving averaged between 20 and 30 ppm. Information on the hydrogen content of the as-received tubing is not yet available, but a value of about 20 ppm is expected.

Inspection of the tubes after they were autoclaved revealed that several of the swaged tubes contained a grey longitudinal streak about 1/8 inch wide on the outer sheath surface. A similar streak was also present in samples of the original sheath tubing. Although the streaks are not considered detrimental to the integrity of the elements, tests are being made to identify the cause.

Depressions as deep as 0.017 inch developed in the outer sheath of many of the tubes during the autoclaving operation. These depressions, which were at the extremities of the oxide core, were the result of sheath buckling into areas of the core that had a relatively low density. The low density was attributed to local disruption of the oxide packing when the core ends were counterbored to permit insertion of the end plugs. Improved methods for counterboring the oxide and assembling the end plugs will be developed to eliminate the depressions.

One of the vibratorily compacted tubes was subjected to internal pressure in order to measure the pressure required to collapse the inner sheath. The test was performed with the element at 350°C, the approximate maximum cladding temperature of an operating fuel element in a D_2O-cooled power reactor. The collapse pressure was 600 psi, which is well above the level expected in current irradiation tests.

The results of room-temperature tensile tests on specimens from the outer sheaths of the compacted tubes were in general agreement with results on tubes fabricated previously (DP-625). Vibratory compaction did not alter the Zircaloy properties. Swaging to a sheath reduction of 6% increased yield and tensile strengths, and reduced elongation-to-fracture values more than 50%. The results of the tensile tests are presented in Table II.

2. Irradiation Tests

Eight of the compacted oxide tubes described in the preceding section were placed under irradiation, two in the Vallecitos Boiling Water Reactor and six in a Savannah River reactor. One of the two VBWR tubes was fabricated by vibratory compaction and swaging, and the other was fabricated by vibratory compaction alone. The approximate irradiation conditions for these two tubes are as follows:

Coolant - boiling H_2O
Reactor pressure - 1000 psi
Mean coolant temperature - 285°C
Total element power - 120 kw per tube
Maximum heat flux - 140,000 pcu/(hr)(ft^2)

After irradiation at these conditions for about two weeks, the tubes will be inspected. If inspection results are satisfactory, they will then be irradiated further at a maximum heat flux of about 300,000 pcu/(hr)(ft^2).

Half of the tubes being irradiated at Savannah River are swaged specimens; the other half are vibratorily compacted. The tubes are being irradiated at temperature levels below those projected for a D_2O-cooled power reactor. The irradiation conditions at Savannah River are classified "Secret," and are presented in separate classified reports.

3. Thermal Cycling of Compacted Oxide Tubes

One each of the swaged and vibratorily compacted elements described in Section II C 1 were subjected to a series of thermal cycling tests in order to obtain information on cladding behavior. The cycles consisted of immersion of the tubes in molten lead for five minutes and immersion in water at about 100°C for three minutes. The lead temperature was 450°C in some of the cycles, and 650°C in others. The results are summarized below:

	Incremental Increase in OD, inch	
	Non-Swaged Element	Swaged Element
After 1 cycle, 450°C/100°C	0.001	0.001
After 5 cycles, 450°C/100°C	0.000	0.000
After 20 cycles, 450°C/100°C	0.000	0.000
After 1 cycle, 650°C/100°C	0.002	0.003
After 2 cycles, 650°C/100°C	0.001	0.002
After 5 cycles, 650°C/100°C	0.002	0.001
After 12 cycles, 650°C/100°C	0.003	-
Total growth of element	0.009	0.007

There was no apparent change in inside diameter during the cycling, and the change in length was insignificant. At the completion of the tests there was no apparent indication of weld or clad failure or of abnormal swelling. Although the tests are not representative of any particular conditions that are expected to exist during irradiation of the tubes, they do provide a good general indication of element integrity. On the basis of the results, it is believed that the compacted tubes can withstand considerable power cycling during irradiation.

4. Analysis of Uranium Oxide for Moisture

Two methods were developed at the Savannah River Laboratory and successfully applied for the determination of traces of water in UO_2 for power reactor fuel elements. Comparable results were obtained with the two methods. At moisture concentrations of 10 to 100 ppm, the standard deviation for each method was about 20% with a 5-g sample of UO_2. One method employed pressure-volume-temperature measurements of the water vapor that was liberated when UO_2 was heated in vacuum to 300°C in a glass system. A small correction was applied for the noncondensable inert gases that were also liberated as the oxide was heated. The second method utilized an electrolytic moisture monitor and integrator to measure the water vapor that was evolved when UO_2 was heated in a stream of dry helium. The analytical train was placed in a dry box to minimize sorption of H_2O by UO_2 during transfer of samples from fuel tubes to the furnace.

A study of UO_2 with the first-named method showed that water is held by the oxide by at least two different mechanisms and that a temperature of about 300°C is necessary for complete dehydration. Water that is adsorbed on the surface is apparently released at 25°C under vacuum; absorbed or chemisorbed water is released over the temperature range 100-300°C. An example of the desorption curve in this range is shown in Figure 5. The dotted portion of the curve illustrates the magnitude of the correction for noncondensables.

More thorough studies of the adsorption isotherms of water on UO_2 will be made in support of the development program on oxide fuel elements.

III. HEAT TRANSFER BURNOUT IN SUBCOOLED WATER

Additional measurements of heat transfer burnout were made at the Savannah River Laboratory (SRL) for forced flow of subcooled water at system pressures as high as 1000 psi. The results confirmed and extended the data presented in DP-645 in the following respects:

1. Variation of the system pressure between 150 and 1000 psia has little or no effect on the burnout heat flux.

2. At pressures as high as 1000 psia and at a given subcooling and fluid velocity, the burnout heat fluxes for heated tubes of 0.32- and 0.49-inch ID were the same and were consistent with previous measurements at SRL with different equipment at much lower pressure (~50 psia).

3. The burnout heat flux at 150-1000 psia for heated tubes of 0.87-inch ID was about 20% lower than the corresponding value for 0.32- and 0.49-inch ID tubes and was not consistent with a limited amount of earlier burnout data at 50-80 psia.

The burnout experiments, the results of which are presented in Table III, are being conducted to establish the heat transfer limits for liquid-D_2O-cooled power reactors that operate at pressures of about 1000 psi. The tests were made with electrically heated tubes that were 2 ft long and were cooled by internal downward flow of water. Burnout, or melting of the tube wall, was visually observed through sight glasses in the enclosures around the heated tube.

The data in Table III were correlated by the following empirical equation:

$$(Q/A)_{BO} = K(1 + 0.0365V)(1 + 0.00914T_s) \qquad (1)$$

where

$(Q/A)_{BO}$ = burnout heat flux, pcu/(hr)(ft²)

K = a constant; K is 536,000 for tubes of 0.32- and 0.49-inch ID at system pressures of 80-1000 psia; K is 448,000 for tubes of 0.87-inch ID at system pressures of 150-1000 psia.

V = water velocity at burnout point, ft/sec

T_s = subcooling at burnout point, °C

The correlation fits the data with a maximum deviation of ±16% and a standard deviation of ±8%. The subcooling and velocity terms of Equation (1) are the same as those in the following burnout equation reported in DP-355 for 25-85 psia:[4]

$$(Q/A)_{BO} = 266{,}000\ (1 + 0.0365V)(1 + 0.00914T_s)(1 + 0.0131P) \qquad (2)$$

In Equation (2), P is the coolant pressure at the burnout point, psia.

Drift plots of Equation (1) against subcooling and velocity show that the respective terms in the equation adequately represent the effects of these variables. There is some evidence of a drift of the correlation with subcooling, but the number of data points responsible for this apparent trend are considered too few to warrant adjustment of the subcooling term in the equation at this time. A drift plot against system pressure shows that, for equivalent diameters of 0.32 and 0.49 inch, the correlation adequately describes the experimental data from 1000 psia down to about 80 psia. At lower pressures the burnout flux decreases rapidly with decreasing pressure.

At an equivalent diameter of 0.87 inch, there appears to be no significant pressure effect from 1000 psia down to about 150 psia. Below 150 psia, the burnout flux decreases somewhat with decreasing pressure. The data obtained with the 0.87-inch-ID tube in high-pressure equipment at pressures less than 100 psia are 15-40% lower than four burnout fluxes that were measured for annuli of 0.75- and 1.0-inch equivalent diameter in low-pressure equipment. Pending resolution of this discrepancy by experiments that are now underway, the correlation of the data for 0.87-inch-ID tubes by Equation (1) must be considered tentative.

TABLE I

GAS CONTENT OF UO_2 CORES IN COMPACTED FUEL TUBES
Preirradiation Data

Element No.	Specimen No.	Temp. of Extraction, °C	Total Gas, cc/g	Composition, ppm in UO_2 H_2	O_2	N_2	CO	CO_2
Z185B[a]	C	1300	0.11	0.3	-	61	-	7
Z186B[a]	B	1300	0.10	-	18	81	6	6
Z93B[a]	A	1300	0.11	3	11	56	12	4
Z99B[a]	B	1300	0.07	3	5	27	8	1
Z186B[a]	B	1000	0.04	2	2	5	17	1
Z186B[a]	D	1000	0.05	-	3	12	16	6
Z99B[a]	B	1000	0.05	1	7	26	32	18
Z99B[a]	D	1000	0.04	2	2	7	9	4
ZE194[b]	A	1000	0.05	2	2	16	23	10
ZE194[b]	B	1000	0.05	2	0.8	15	23	13
ZE194[b]	C	1000	0.05	2	0.5	12	19	16
As-Received[b]	-	1000	0.44	18	0.3	170	130	5

(a)natural oxide
(b)5% enriched oxide

TABLE II

RESULTS OF TENSILE TESTS ON ZIRCALOY SHEATHS OF COMPACTED UO_2 TUBES
Preirradiation Data

Tube No.	Tube Condition	Orientation of Tensile Specimen	Yield Strength, psi	Ultimate Strength, psi	Elongation to Fracture, %	Reduction of Area, %
H06-12B	As-received	Circumferential	48,500	84,500	16.5	43.5
H06-12C			45,500	83,000	16.5	41
H06-12D			45,400	82,500	16.5	46.5
H06-12B	As-received	Longitudinal	44,000	72,000	22	33.5
H06-12C			50,500	75,500	25	32
H06-6A	As-received	Circumferential	52,000	76,000	17	44
H06-6B			46,000	82,000	15.5	47.5
H06-6C			50,000	76,000	16	43.5
H06-6C	As-received	Longitudinal	47,000	78,000	15.5	27
H06-6D			45,000	77,500	21	35
H06-6E			43,500	76,400	14	38
Z178A-B	Vibratorily compacted	Circumferential	44,500	73,600	-	43
Z178A-C			40,000	73,000	-	46
Z178A-D			37,200	73,000	14	40
Z178A-6	Vibratorily compacted	Longitudinal	45,000	71,000	24	31.7
Z178A-H			-	71,000	21.5	32.4
Z178A-I			44,500	71,600	-	32.4
Z186A-A	Swaged	Circumferential	54,500	88,400	-	31.5
Z186A-B			55,000	89,300	6.5	26.6
Z186A-C	Swaged	Longitudinal	-	80,000	12.5	25.4
Z186A-F			52,200	80,600	12.5	20.2
Z186A-G			58,800	81,000	10.9	20.4

TABLE III

EXPERIMENTAL BURNOUT HEAT FLUXES IN SUBCOOLED WATER

Pressure, psia	Equivalent Diameter of Tube, inch[a]	Water Vel. (V), ft/sec	Subcooling (T_s), °C	Meas. Burnout Heat Flux $(Q/A)_{BO}$, 10^6 pcu/(hr)(ft^2)	Ratio of Measured to Calculated[b] Burnout Heat Flux
54	0.324	28.7	43.8	1.33	0.86
84	0.321	36.6	41.4	1.80	1.04
87	0.322	38.8	34.8	1.63	0.95
193	0.321	38.6	51.6	1.84	0.97
798	0.322	38.8	35.2	1.67	0.98
994	0.322	35.8	33.1	1.86	1.16
1000	0.320	38.5	76.4	1.94	0.89
1001	0.313	21.3	36.2	1.47	1.16
52	0.487	27.7	41.6	1.16	0.78
93	0.489	24.7	78.2	1.61	0.92
503	0.487	25.3	40.7	1.43	1.01
803	0.495	28.6	33.4	1.41	0.99
1000	0.495	16.4	46.5	1.21	0.99
1002	0.492	33.3	59.0	1.79	0.99
1002	0.491	34.4	57.2	1.80	0.98
26	0.877	19.8	39.4	0.87	0.83
40	0.877	13.3	48.1	0.87	0.91
87	0.880	29.5	29.8	1.01	0.85
193	0.877	10.4	55.7	0.86	0.93
193	0.871	9.8	72.2	1.05	1.04
195	0.869	30.0	30.5	1.10	0.92
501	0.871	19.2	38.6	1.14	1.10
999	0.872	19.7	43.0	1.14	1.06
1000	0.868	29.7	22.5	1.07	0.95

(a) Heated length was 2 ft.

(b) Calculated by the following correlating equations for high-pressure burnout:

$(Q/A)_{BO} = 536{,}000(1 + 0.0365V)(1 + 0.00914T_s)$ for tubes of 0.32- and 0.49-inch ID

$(Q/A)_{BO} = 448{,}000(1 + 0.0365V)(1 + 0.00914T_s)$ for tubes of 0.87-inch ID (tentative)

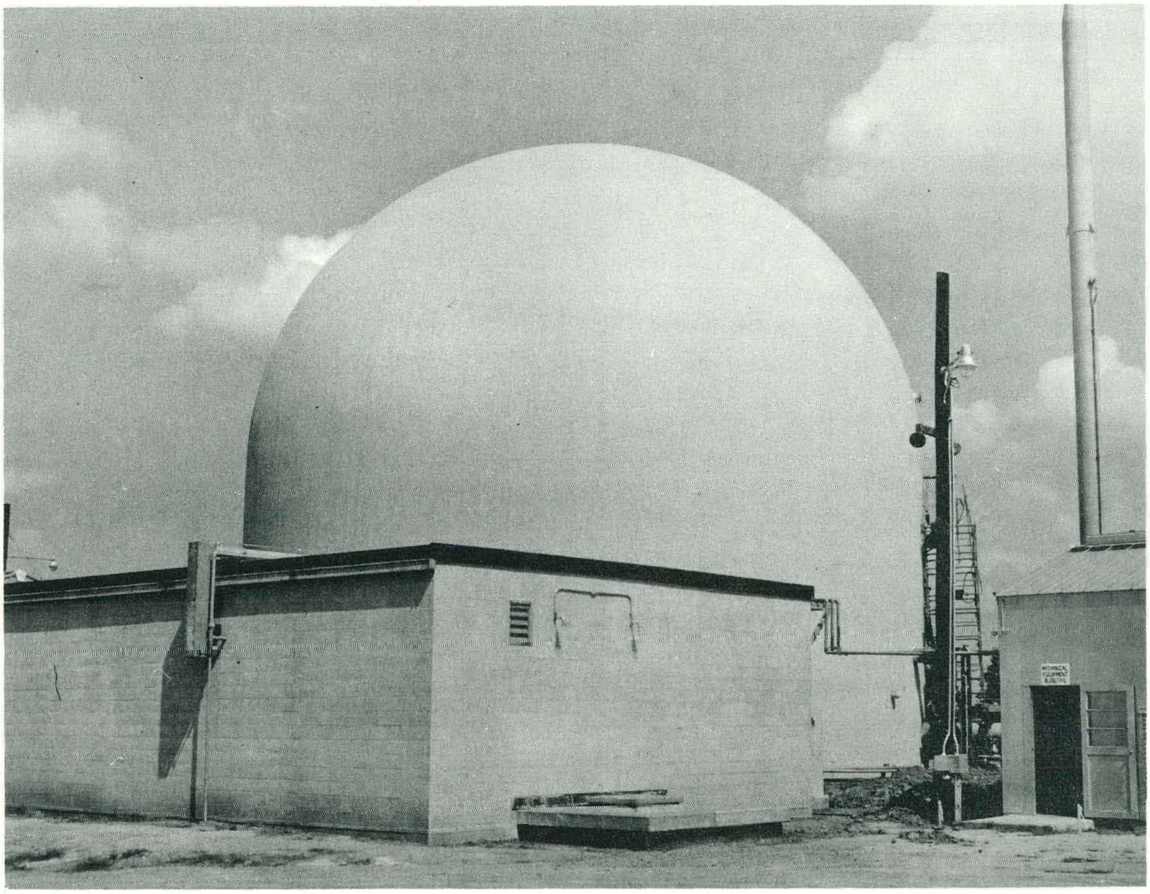


FIG. 1 EXTERIOR OF HWCTR CONTAINMENT SHELL – SEPTEMBER 1961
Concrete block building in foreground is reactor control building.

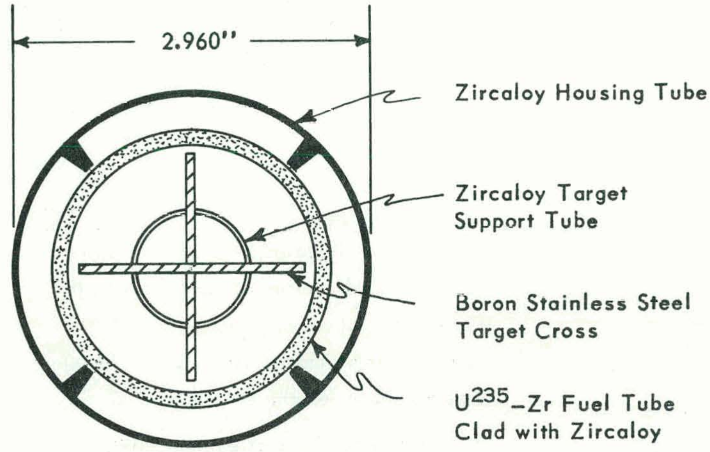


FIG. 2 CROSS SECTION OF HWCTR DRIVER ASSEMBLY

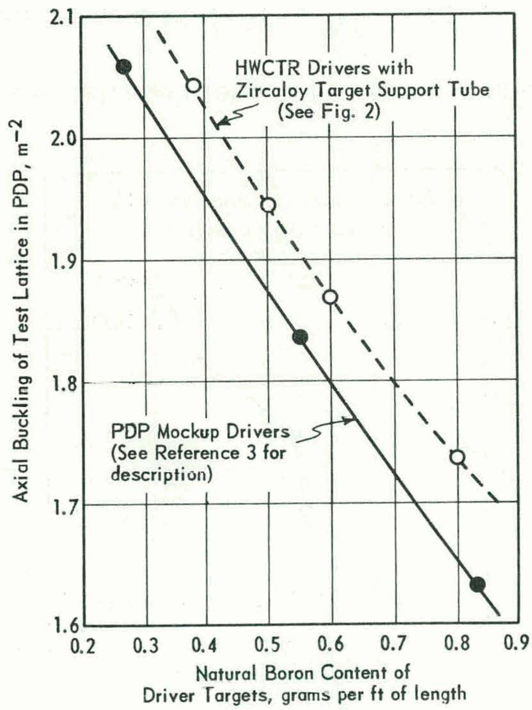
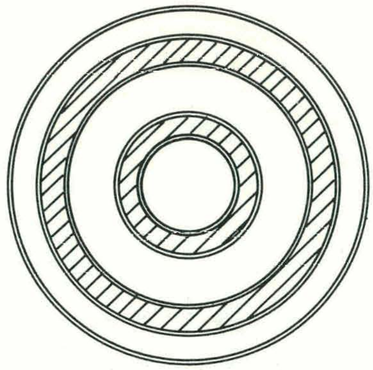


FIG. 3 EFFECT OF BORON CONTENT OF TARGETS ON
REACTIVITY OF HWCTR DRIVER ASSEMBLIES
Results of critical water height measurements
in the Process Developement Pile (PDP)



	Inner Tube	Outer Tube
Clad inner diameter, inches	0.660	1.700
Core inner diameter, inches	0.710	1.750
Core outer diameter, inches	0.970	2.010
Clad outer diameter, inches	1.020	2.060
Approximate length, ft	10	10

Housing tube ID, inches	2.494
Housing tube OD, inches	2.560
Core	Uranium metal
Clad and housing	Zircaloy - 4

Spacers for maintaining tube concentricity are not shown.

FIG. 4 THIN URANIUM METAL TUBES FOR HWCTR TESTS

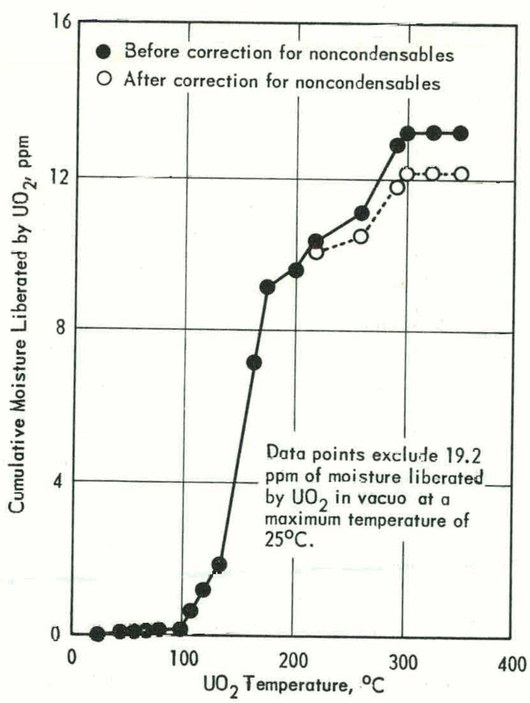


FIG. 5 DESORPTION OF WATER BY FUSED URANIUM OXIDE

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5. Earlier progress reports in this series are:

DP-232	DP-345	DP-425	DP-485	DP-545	DP-605
DP-245	DP-375	DP-435	DP-495	DP-555	DP-615
DP-265	DP-385	DP-445	DP-505	DP-565	DP-625
DP-285	DP-395	DP-455	DP-515	DP-575	DP-635
DP-295	DP-405	DP-465	DP-525	DP-585	DP-645
DP-315	DP-415	DP-475	DP-535	DP-595	DP-655

Progress for the month of October will be reported in DP-675.