DP - 485

Reactors - Power

AEC Research and Development Report

HEAVY WATER MODERATED POWER REACTORS

PROGRESS REPORT

March 1960

Technical Division

Wilmington, Delaware

April 1960

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

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Progress Report
March 1960

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Power Reactor Studies
Wilmington, Delaware

Compiled by R. R. Hood

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ABSTRACT

Safeguards analyses of the boiling D_2O loop of the Heavy Water Components Test Reactor (HWCTR) show that neither a power failure nor a loss of cooling water will cause a serious accident if the reactor is scrammed. Construction was started on a full-scale mockup of the bayonet for this loop. The mockup will be used for studies of possible vibration phenomena associated with the flow of steam-water mixtures. Emphasis in the development of uranium metal fuel for power reactor use has been shifted to coextruded tubes that have a higher surface-to-volume ratio than those fabricated heretofore. The modified tube design will permit higher powers to be achieved without exceeding thermal limitations, and offers the potential advantage of improved metallurgical behavior during irradiation. A swaged tube of uranium oxide with stainless steel cladding apparently failed during irradiation.

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HEAVY WATER MODERATED POWER REACTORS

Progress Report
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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 power reactors that are fueled with natural uranium. The present effort is divided into two main categories: (1) the 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 development of the technology of full-scale D_2O-moderated power reactor plants. Earlier reports in the series are:

DP-232	DP-345	DP-425
DP-245	DP-375	DP-435
DP-265	DP-385	DP-445
DP-285	DP-395	DP-455
DP-295	DP-405	DP-465
DP-315	DP-415	DP-475

Progress during April 1960 will be reported in DP-495.

SUMMARY

Design of the HWCTR is 90% complete, and purchase orders have been placed for all items of limiting equipment. Construction of the underground part of the reactor building shell is substantially complete, and back-filling is in progress. The last pieces required for fabrication of the reactor vessel are scheduled to be completed by mid-April.

Construction was started on a full-scale mockup of the bayonet that is to be installed in the boiling D_2O loop of the HWCTR. This mockup will be used in experiments to determine whether the flow of steam-water mixtures will result in damaging vibration of the bayonet. A purchase order has been placed for both of the bayonets that will be required for the isolated loops of the HWCTR.

Initial results of the safeguards analysis of the boiling D_2O loop indicate that neither a reduction in D_2O flow through the loop after a power failure nor a loss of cooling water for the loop will cause a serious accident if the reactor is scrammed.

Emphasis in the program to develop suitable fuel elements of uranium metal for power reactor use has been shifted to coextruded tubes that have a higher surface-to-volume ratio than those fabricated heretofore. The higher ratio is achieved by halving the wall thickness of the tubes (1/8 inch vs. 1/4 inch). The thinner tubes are favored because they

permit higher specific powers to be achieved without exceeding limitations on heat flux (in boiling reactors) and maximum temperature (in both boiling reactors and liquid-D_2O-cooled reactors). High specific powers are essential in D_2O-moderated power reactors if competitive power costs are to be realized. The thinner tubes also offer the potential advantage of alleviating factors that cause fuel growth and cladding failures during irradiation (see Table III).

A swaged tube of uranium oxide with stainless steel cladding apparently failed during irradiation to modest exposure in a Savannah River reactor and was removed for examination. The cause of the failure, which was evidenced by the release of gaseous fission products, has not yet been established. A second fuel element of this type was irradiated without incident, and irradiation of a third assembly is in progress. Five additional irradiation specimens were swaged and evaluated.

A mechanical design was developed for a fuel assembly in which rod bundles of swaged uranium oxide can be irradiated in the HWCTR. Efforts will be made to procure swaged rods from a vendor for use as an alternative to swaged tubes if the latter cannot be developed in time for startup of the HWCTR. A special design was developed for an assembly in which insulated swaged rods can be irradiated in a Savannah River reactor, and one such assembly was prepared for irradiation.

DISCUSSION

I. HEAVY WATER COMPONENTS TEST REACTOR (HWCTR)

The HWCTR is a test reactor that is being built primarily for irradiation tests of fuel elements at conditions that typify those expected in D_2O-moderated power reactors. The reactor is scheduled to be placed in operation in mid-1961. A description of the reactor was presented in DP-383 and in earlier progress reports. Progress during March on design, construction, and supporting experimental work is reported in this section.

A. REACTOR STATUS

1. Construction

The application of sprayed concrete to cover the post-tensioning cables around the concrete of the reactor containment building was completed, and back-filling of the excavation around the building was begun. Application of asphalt damp-proofing compound is being completed during back-filling. At the end of March, back-filling had been completed to approximately mid-height of the underground part of the building (see Figure 1). Installation of structural steel and steel grating for the floors in this part of the building was completed.

The bottom head and the remaining two formed plates for the neck of the reactor vessel were shipped during March to the Pacific Coast Engineering Company for vessel fabrication. The top head and one of the two plates for the lower part of the reactor vessel were rejected because they were made from steel plate that did not meet specifications. These pieces are now scheduled for completion at Lukens Steel by mid-April.

2. Design

An order was placed for fabrication of the transfer coffin for irradiated fuel. Purchase orders have now been placed for all limiting items of equipment. Dimensions of the upper cavity of the transfer coffin were increased slightly as a result of a review of problems that might be encountered in removing a damaged bayonet from one of the isolated coolant loops of the reactor.

All Process, Power, and Architectural and Civil drawings scheduled for the HWCTR project have now been issued.

Because of the unattractive price quoted by the fabricator of control and safety rod drives for rework of the two prototype rod drive units for reactor use, the necessary new parts will be purchased by the du Pont Engineering Department, and rework and testing will be carried out at the Savannah River Plant. The action will provide opportunity for early training of maintenance and operating personnel with actual rod drive units.

At the end of March, HWCTR design was approximately 90% complete.

3. Development and Testing

Testing of the prototype gripper mechanism for the irradiated fuel coffin has been completed except for checkout tests after final modifications for installation in the coffin.

4. Isolated Coolant Loops

Construction was started on a full-scale mockup of the bayonet that is to be installed in the boiling D_2O loop of the HWCTR. The purpose of the test facility is to determine whether damaging vibration is likely to occur in the HWCTR bayonet. Steam and water will be passed through the mockup to simulate operation of a fuel assembly that generates boiling D_2O. In the mockup, water at about 260°C passes down an outer coolant channel and then up an inner coolant channel. Saturated steam at 800 psig will be injected through a perforated dummy fuel assembly into the inner coolant channel to simulate the two-phase flow in the HWCTR bayonet. The test facility will also be used to determine the effectiveness of the steam quencher in the bayonet loop and to determine the pressure drop due to two-phase flow in the piping between the bayonet and the quencher. Construction of the test facility is scheduled to be completed in May 1960.

A purchase order has been placed for detailed design and fabrication of the Zircaloy and stainless steel bayonets to be installed within the HWCTR vessel as parts of the two isolated coolant loops. Two special pipe joints that are candidates for use in connecting the inlet and outlet D_2O piping to the loop bayonets have been purchased for testing at the Savannah River Laboratory. These candidates are the Marman "Conoseal" Joint and the Graylock Joint. Test joints will be subjected to temperature and pressure cycling with bending forces applied to simulate the operating loads on the joints. Water leakage will be measured and general performance will be observed.

Procurement has been initiated for all major items of equipment for the isolated coolant loops. Isometric sketches of loop piping have been reviewed, and incorporation of the piping in the field model of the HWCTR will now proceed.

B. SAFEGUARDS ANALYSIS OF BOILING LOOP OF THE HWCTR

Calculations were started to define the transient response of the boiling D_2O loop of the HWCTR to flow disturbances and to reactor scrams. Results now available indicate that neither a reduction in loop D_2O flow after an AC power failure nor a loss of H_2O flow through the loop heat exchanger will cause a serious accident if the reactor is scrammed.

Figure 2 shows a schematic diagram of the boiling D_2O loop. Liquid D_2O is pumped through three paths in parallel: (1) a fuel assembly in the reactor, (2) a heat exchanger that is cooled by H_2O, and (3) a bypass around the heat exchanger. The three streams are recombined in a steam quencher, in which the D_2O steam produced in the fuel assembly is condensed, and are returned to a surge tank. A helium blanket in the surge tank maintains the loop at reactor pressure. The calculations were made for a fuel assembly that consists of a single zirconium-clad tube of uranium metal (2-inch OD x 1.5-inch ID x 10-feet long) at the following reactor conditions (see Table I for details):

Reactor power	- 30 MW
Reactor pressure	- 795 psia
Power in boiling loop	- 1.43 MW
Flow bypassing loop heat exchanger	- None

The calculation procedure is discussed immediately following the presentation of results.

1. Transients Following a Scram

Loop transients following a reactor scram are shown in Figure 3. Steam production in the fuel assembly stops within one second. The loop D_2O cools more rapidly than the D_2O in the reactor, as shown in the following table.

Average Cooling Rates of D_2O During First 35 Seconds of Scram

	Inlet to Fuel	Outlet from Fuel
Loop	35°C/min	50°C/min
Reactor	11°C/min	23°C/min

These cooling rates could be more nearly equalized by (a) opening the steam valves on the HWCTR boilers wider than the setting appropriate to an initial power level of 30 MW (see DP-475 for discussion of effect of valve setting on HWCTR conditions), or (b) bypassing some of the loop D_2O around the heat exchanger. Even without such action, however, this difference in cooling rate between reactor and test loop is not expected to cause any thermal stress damage.

2. Transients Following Loss of H_2O Flow

Loop transients following a sudden complete loss of H_2O flow through the loop heat exchanger are shown in Figure 4. If the reactor is not scrammed, all H_2O boils out of the heat exchanger within 4 seconds; within 19 seconds the steam quencher outlet temperature reaches the saturation temperature of D_2O, and thereafter the steam from the test assembly is only partially condensed. Steam would then flow into the gas space in the reactor vessel, and the liquid level would begin to drop in the loop surge tank. It has not been established yet whether the increased pressure in the reactor as a consequence of the steam flow would scram the reactor before the loop flow stops as a result of inadequate suction head on the circulating pump.

It is more likely that a sudden stoppage of H_2O flow will scram the reactor; this case is shown in Figure 5. Again, all H_2O boils out of the heat exchanger within 4 seconds, although now steam production in the fuel assembly stops within 1 second. The temperature at the pump suction remains low enough that loss of loop flow because of pump cavitation is not a threat.

3. Transients Following AC Power Failure

It is assumed that, following a failure of the AC power supply to the D_2O-circulating pump in the test loop, flow will be reduced by flywheel action until emergency DC power takes over. The D_2O flow is assumed to decrease linearly to one-third of its initial value in 30 seconds, after which it remains constant.

Figure 6 shows loop transients that follow such a reduction in D_2O flow while the reactor continues to operate at a constant power level of 30 MW. In the absence of a reactor scram, steam quality rises rapidly and, within about 11 seconds, the steam generated in the test assembly no longer is completely condensed in the steam quencher. Subsequent events are similar to those described above for the "loss of H_2O" incident.

An AC power failure should, however, scram the reactor; this case is shown in Figure 7. Steam generation ceases within 1 second, and the loop cools steadily. No serious consequences are expected as a result of this incident.

4. Calculation Procedures

In computations of the temperature transients, the boiling loop was considered to be divided into a number of discrete blocks. Each block was assumed to be at a uniform temperature. The division of the system for this purpose is indicated in Figure 2. The temperatures were derived from equations that express the heat balances for the individual blocks. These equations were coded for solution on the IBM 650 computer.

The temperature transients in the loop are obtained by solving eight differential equations:

$$H_1 \frac{dT_1}{dt} = \gamma_1 \psi - \gamma_2^* \left[T_1 - \frac{T_2 + T_3^*}{2} \right] \qquad (1)$$

$$H_2 \frac{dT_2}{dt} = F(T_6 - T_2) \tag{2}$$

$$H_3 \frac{dT_3}{dt} = f_1F(T_2 - T_3) + \gamma_2\left[T_1 - \frac{T_2 + T_3}{2}\right] \tag{3}$$

$$H_5 \frac{dT_5}{dt} = F(T_4 - T_5) \tag{4}$$

$$H_6 \frac{dT_6}{dt} = F(T_5 - T_6) \tag{5}$$

$$H_7 \frac{dT_7}{dt} = f_3F(T_2 - T_7) - \frac{UA}{2}(T_2 + T_7 - T_9 - T_w) \tag{6}$$

$$H_9 \frac{dT_9}{dt} = G(T_w - T_9) + \frac{UA}{2}(T_2 + T_7 - T_9 - T_w) \tag{7}$$

$$\frac{dN}{dt} = 0 \quad \text{(dummy variable)} \tag{8}$$

The nomenclature is as follows; subscripts are identified in Figure 2.

H = heat capacity of the block, MWsec/^{o}C

T = temperature, ^{o}C

t = time, sec

γ_1 = proportionality constant that relates reactor power to the power generated in the fuel assembly in the boiling loop, dimensionless

ψ = reactor power, MW

γ_2^* = the product of (1) the over-all coefficient of heat transfer from the fuel to the coolant and (2) the fuel heat transfer surface, MW/^{o}C

T_3^* = the saturation temperature of the D_2O at the outlet of the fuel assembly, ^{o}C

F = a measure of the total D_2O flow through the loop, MW/^{o}C

f_1 = fraction of total D_2O flow that passes through the fuel assembly, dimensionless

f_2 = fraction of total D_2O flow that bypasses the loop heat exchanger, dimensionless

f_3 = fraction of total D_2O flow that passes through the loop heat exchanger, dimensionless

UA = product of the heat transfer coefficient and area in the heat exchanger, MW/°C

G = a measure of the H_2O flow through the heat exchanger, MW/°C

T_w = the temperature of the H_2O entering the heat exchanger, °C

The boiling point of the D_2O, T_3^*, is expressed as a function of pressure by an empirical equation of the form

$$T_3^* + 273.2 = a_o + \sum_{i=1}^{7} a_i (\log P')^i \tag{9}$$

In Equation (9), the pressure P' is in atmospheres, and a_o and a_i are constants. Since the gas space of the loop surge tank is connected to the gas space in the reactor vessel, P' is the reactor pressure; like ψ, it may be constant or it may be a known function of time, as in the case of a reactor scram.

The over-all coefficient of heat transfer between the fuel and the boiling D_2O is:

$$\frac{\gamma_2^*}{A} = \left(\frac{hk}{hl + k} \right) \tag{10}$$

In this equation, h is the coefficient of heat transfer from cladding to coolant, A is the fuel heat transfer surface, and k and l are the thermal conductivity and thickness of the cladding.

$$\gamma_2^* = \frac{4.58\ hk}{hl + K} \tag{11}$$

In Equation (11), h is in pcu/(hr)(ft^2)(°C), k is in pcu/(hr)(ft)(°C), and l is in ft. The film coefficient at the cladding surface, h, is expressed as follows:

$$h = 0.45\ e^{0.00444P}\ (T_1 - T_3^*)^3 \tag{12}$$

where P is in psia, i.e., P = 14.7P'

The remaining D_2O temperatures are computed from

$$T_8 = \frac{f_2 T_2 + f_3 T_7}{f_2 + f_3} \text{ and} \tag{13}$$

$$T_4 = \frac{xf_1L}{c} + f_1T_3^* + (f_2 + f_3)T_8 \qquad (14)$$

where c is the heat capacity of D_2O in cal/(gm)(^oC), and L is the heat of vaporization of D_2O in cal/gm.

The steam fraction x (steam quality = 100x) is computed from

$$x = \frac{c\gamma_2^*}{Lf_1F}\left[T_1 - \frac{T_2 + T_3^*}{2}\right] - \frac{c}{L}\left[T_3^* - T_2\right] \qquad (15)$$

also,

$$f_1 + f_2 + f_3 = 1 \qquad (16)$$

When no D_2O steam is being generated, the temperature of the coolant exit the fuel assembly (T_3) replaces the saturation temperature T_3^*, and Equation (14) reduces to the following expression:

$$T_4 = f_1T_3 + (f_2 + f_3)T_8 \qquad (17)$$

In addition, γ_2^* is replaced by the quantity γ_2 to express the heat transfer to subcooled water.

$$\gamma_2 = 0.00986\left[1 + 0.0098\left(\frac{T_2 + T_3}{2}\right)\right] \qquad (18)$$

Equation (18) is derived from conventional equations for heat transfer to water in tubes and annuli.

The numerical values that were used in the calculations are listed in Table I, together with the steady-state conditions at the start of the transients.

C. ELECTRONIC INSTRUMENTS FOR THE HWCTR

The successful operation of the HWCTR requires the application of a number of electronic instruments in unusual environments. This section contains a discussion of some of the new mechanical arrangements and modified circuits that have been developed and tested to meet these requirements.

Pulse transformers will be used between the detectors and the main electronic circuits in the startup BF_3 and fission counter systems. The use of the pulse transformers permits the preamplifiers to be located in the control room rather than at the relatively inaccessible detector location inside the containment building. This arrangement reduces both maintenance and space requirements at the counter location.

Another application of pulse transformers is the Low Energy Gamma Monitor (LEGM) which will be provided to monitor the D_2O streams for possible fuel ruptures. The LEGM will consist of a scanning gamma analyzer (commercially available) that is set to monitor a water sample for the presence of a 100-kev peak produced by Np^{239}. The elevated temperature of the HWCTR required the development of a scintillation detector probe that overcomes the temperature limits of photomultiplier tubes. Laboratory tests indicate that the new design will operate successfully.

Leakage of heavy water into the light water cooling streams will be detected by monitoring a continuous sample of light water for radioactive nuclides normally found only in the irradiated heavy water. Two monitor installations will be made, one to detect leaks in either of the two steam generators and the other to monitor the remainder of the system. The light water sample for these monitors will be at a temperature as high as 100°C. Cooling the sample would introduce a time lag which would reduce the monitor sensitivity, since the telltale radionuclides have short half lives. A water-cooled scintillation probe has been developed that is capable of monitoring the hot water sample directly.

Operation of sensitive electronic instruments in close physical proximity to a heavy duty electrical system can result in electrical interactions that cause disturbances in the sensitive detector circuits. To minimize these problems, all signal cables for the HWCTR will be installed in rigid metallic conduit that is completely separated from power and lighting circuits. In addition, doubly shielded coaxial cables will be used for the signal leads, and, where feasible, sources of electrical transients, such as relays, will be equipped with interference filters.

D. SEAL LEAKAGE TESTS

The average leakage rates of water from several gasketed joints of the type to be used in the HWCTR were below design limits during 25 cycles of testing with deionized water at peak conditions of 1500 psi and 260°C. A 6-inch flanged joint for the central control rod enclosure, a 1-inch joint for monitor pin assemblies, and four tubing fittings are being tested. Descriptions of these joints and of the test procedures were presented in DP-395 and in subsequent progress reports. A standard "Flexitallic" gasket is being used in the 6-inch flanged connection, and an asbestos gasket with a stainless steel sheath is being used in the monitor pin joint. The following types of stainless steel tubing fittings are being tested:

1. Crawford "Swagelok", compression type, 1/4-inch pipe to 3/8-inch tubing

2. Conax "Midlock", compression type, 1/4-inch pipe to 1/4-inch tubing

3. Conax, packing type with lava ring insert, 1/4-inch pipe to 1/4-inch tubing

4. Parker, flared type, 1/4-inch pipe to 3/8-inch tubing

In previous tests at 1500 psig, as reported in DP-425, the complete monitor pin assembly, including the primary 1-inch gasket as well as a tubing fitting, was tested for leakage as a unit. Consequently, only the combined leakage from the gasket and tubing fitting was measured. In the current tests, the leakages from the two components are being measured separately. During each typical 3-hour cycle, the joints are maintained at the maximum temperature and pressure for one hour, and the balance of the time is used for heating, cooling, and venting to atmospheric pressure. A summary of the leakage rates from each of the test joints during the first 25 cycles of a 100-cycle test is shown in Table II.

II. TECHNOLOGY OF FULL-SCALE REACTORS

A. REACTOR FUELS AND MATERIALS

One of the most important objectives of the du Pont program on D_2O-moderated power reactors is to develop a fuel element with which a total fuel cost of 1 mill/kwh or less can be realized in a full-scale reactor. Two possible routes to low cost fuel are being investigated:

1. Efforts are being made to develop acceptable fuel elements of uranium metal. This material is potentially cheap to fabricate, but its ability to withstand the desired burnup under power reactor conditions has not yet been demonstrated. Zircaloy-clad tubes of uranium metal for irradiation tests are being fabricated at Nuclear Metals, Inc. via a coextrusion process.

2. A swaging process for production of Zircaloy-clad tubes of uranium oxide is being developed at the Savannah River Laboratory with the objective of decreasing the fabrication cost of suitable oxide elements. One of the principal questions with respect to oxide tubes is whether the inner cladding will collapse from the internal pressure. To obtain early information on this question, irradiation tests of swaged oxide tubes with stainless steel cladding are under way at Savannah River. The stainless steel is being used as a temporary

substitute for Zircaloy sheath stock, which is not yet available in quantity.

1. Fuel Elements of Uranium Metal

Heretofore, the emphasis in the development of fuel elements of uranium metal has been placed on relatively massive, thick-walled tubes (2.0-inch OD x 1.5-inch ID), which are of interest because of their indicated low fabrication cost. These tubes were intended primarily for use in liquid-D_2O-cooled reactors of relatively low capacity, viz., 100 eMW. The emphasis is now being shifted to thinner tubes, the reasons being:

a. Evaluation studies have shown that of all the D_2O-moderated power reactors considered thus far, reactors that are cooled by boiling D_2O offer the greatest promise of eventually producing competitive power. Boiling reactors require fuel elements of relatively large surface area because the limiting heat fluxes are low as compared with liquid-cooled reactors.

b. The evaluation studies have also shown that high specific powers in the fuel are a requisite to competitive power in D_2O-moderated reactors, either boiling or liquid cooled. High specific powers cannot be achieved without encountering fuel temperature limitations unless the subdivision of the fuel is increased.

c. Thin tubes hold promise for attaining higher burnup than thick tubes because at a given specific power: (1) the cladding strain is less per unit volume expansion of the core, (2) uranium temperatures and thermal stresses are lower, and (3) thermal cycles are less severe. These factors are illustrated by Table III, which compares expected service conditions for typical thin tubes with those of typical thick tubes. Prospects are regarded as good for achieving a twofold increase in the fuel exposure (6000 MWD/T vs. 3000 MWD/T) by adopting the thin tubes. If the exposure can be doubled, the cost of power from a large-scale reactor would be reduced by about 1 mill/kwh even if the fabrication cost of the thin tubes were twice that of the thick tubes on a unit weight basis.

In view of the above considerations, the tube fabrication program at Nuclear Metals is being focused on thin tubes, and preparations are being made for irradiation tests of these tubes in a Savannah River reactor, in the VBWR, and in the HWCTR. The cross section of a test assembly of thin tubes for irradiation tests in a liquid-D_2O-cooled position of the HWCTR is shown in Figure 8. Representative operating conditions for this assembly in the HWCTR are included in Table III.

2. Fuel Elements of Uranium Oxide

a. Preparation of Swaged Tubes for Irradiation Tests

Fifteen tubular oxide fuel elements, each 4 feet long, were swaged at the Savannah River Laboratory (SRL) as candidates for irradiation tests. The method of preparation was similar to that used for previous specimens; crushed, fused oxide was loaded into the annulus between concentric stainless steel sheaths, and the oxide was swaged to final density.

The tubes were swaged in two groups. The first group of ten tubes included six that were loaded with "as-received" fused UO_2 from the Spencer Chemical Company with a maximum particle size of -20 mesh. For comparison of swaging characteristics, four tubes were loaded with Norton Company fused UO_2 which was crushed to -20 mesh at SRL. The tubes were loaded to an initial UO_2 density of 64.5% of theoretical. The tubes of Norton UO_2 finished with excellent surfaces, but all of those of Spencer UO_2 had long folds in the outer sheath after swaging.

Five additional tubes were loaded with the Spencer oxide to a density of 66% of theoretical. This higher density was achieved by tapping the tube wall while the oxide was being loaded. When these tubes were swaged to the final diameter, the surfaces were excellent. It was concluded that because of the greater friability of the Spencer UO_2, a higher loading density was required to prevent the outer sheath from folding during the early swaging passes. The greater friability was also apparent from the large increase in the amount of fines, i.e., particles smaller than 200 mesh, observed after swaging. The reason for this behavior is unknown, but must be a result of differences in the fusion processes used by the two vendors. Such differences may also account for the O/U ratio of less than 2.0 for the Spencer oxide, as noted below.

The swaged tubes containing Spencer UO_2 were cut into sections 2 feet long, fitted with end plugs, and welded. The tubes were nondestructively evaluated and five were selected as candidates for irradiation in a Savannah River reactor. The evaluation data for these tubes are shown in the following table.

Evaluation Data for
Five Spencer UO_2 Tubes Selected for Irradiation

Tube Number	OD, inches (Average & Range)	Maximum Eccentricity, inch	Maximum Bow, inch	UO_2 Density, % of Theoretical Average	Minimum	Average O/U Ratio
S-44-A	2.073 +0.002 -0.003	0.014	0.010	88.8	88.5	1.968
S-54-A	2.075 ±0.005	0.014	0.005	89.5	89.4	1.968
S-55-A	2.075 +0.001 -0.002	0.004	0.010	88.5	88.1	1.968
S-56-A	2.077 +0.004 -0.008	0.006	0.012	89.2	88.8	1.968
S-57-A	2.076 +0.003 -0.004	0.005	0.009	88.4	88.3	1.973

The O/U ratios for the Spencer oxide, as determined by ignition of UO_2 to U_3O_8, were all less than 2.0. Since there are no data that indicate the in-pile behavior of swaged oxide with an O/U ratio of slightly less than 2.0, the results of the forthcoming irradiation of this oxide will be of particular interest to the oxide fuel program.

Each of the tubes was inspected for weld integrity by a kerosene bubble test, and the cladding was determined to be free of cracks by "Zyglo" dye penetrant inspection. Representative samples of the inner and outer cladding were also inspected for oxide penetrations. The minimum cladding observed was 0.013 inch.

b. Irradiation Tests of Swaged Oxide Tubes

A swaged tube of uranium oxide clad with stainless steel apparently failed during irradiation in a Savannah River reactor and was removed from the reactor for examination. The failure was evidenced by the release of gaseous fission products into the coolant stream. No significant changes in coolant flow or temperature were noted by the coolant monitors. The cause and nature of the failure are not yet known. One possibility is that the inner sheath collapsed under internal pressure. A more likely cause of failure is that one of the welds developed a leak. A second assembly was discharged from the reactor for examination after achieving modest exposure without incident, and a third assembly was charged into the reactor. Irradiation conditions for these assemblies are classified information and will be reported separately.

c. Mechanical Design of Fuel Assembly Containing Swaged Oxide Rods

The immediate objective of the du Pont investigations of swaged elements of uranium oxide is the fabrication of a load of zirconium-clad swaged tubes in time for startup of the HWCTR. It is not certain that the necessary fabrication development and advance irradiation tests can be completed in time to obtain this objective. Therefore, a parallel effort is being made to procure oxide rods (preferably swaged) that can be used as an alternative to swaged tubes, if necessary. Specifications for the rods are being formulated preparatory to soliciting bids from vendors for their fabrication.

The mechanical design of a fuel assembly for irradiation of the rods in the HWCTR is shown in Figures 9 and 10. The complete fuel assembly, except for the housing tube, is described in Figure 9, and the design of the individual rods is shown in Figure 10. The fuel bundle consists of 19 rods of 0.500-inch core diameter clad with 0.025 inch of Zircaloy-4. In order to fit the bundle into a circular housing of 3.200-inch ID, the outermost rods are arranged on a circle. This arrangement provides better flow distribution through the assembly than if the circular housing contained a bundle with the outermost rods in hexagonal array. Spacing within the bundle is maintained by helically wrapping 12 of the rods with 0.083-inch Zircaloy-4 wire on a 10-inch pitch. The bundle is then held together by a helical wrap of the same wire around the cluster of rods as a whole.

Two prototypes of a single rod were made in order to investigate methods of wire wrapping. No problems were encountered and a tight wrap of wire was obtained.

Material was ordered and fabrication started for a full-size mockup of the bundle. The mockup will be placed in a flow loop for study of pressure drop, flow characteristics, and possible vibration and erosion problems, during extended flow testing under HWCTR conditions.

d. Design of Lead-Insulated Assembly for Irradiation of Swaged Oxide Rods

In advance of the irradiation of swaged uranium oxide under power reactor conditions in the HWCTR, a few swaged rod specimens will be irradiated in a Savannah River reactor to define more clearly the performance characteristics of swaged UO_2 in Zircaloy cladding. These experimental irradiations will permit a study of the effects of heat generation, exposure, compact density, oxygen content of the UO_2, and method of UO_2 preparation on the in-pile behavior of swaged, fused UO_2 fuel. Postirradiation measurements will include dimensional measurements, the quantity of fission gas released to the free volume inside the sheath, and the extent to which structural or other physical changes occur in the fuel. These measurements, plus sampling for burnup analyses, will be carried out in the High Level Caves at the Savannah River Laboratory.

The first assembly, which contains eight rods of UO_2, has been assembled for irradiation; the design is shown in Figures 11 through 13. The fuel rods are made of UO_2 that has been fused, ground, and mechanically compacted to about 90% of the theoretical density in Zircaloy-2 sheaths. Each rod is placed within an insulating layer of lead and an outer shell of stainless steel, as shown in Figure 12. This arrangement serves two purposes during irradiation: (1) the surface of the UO_2 rod is raised to a temperature level that is representative of power reactor operation, i.e., about 350°C; and (2) the steel shell provides strong secondary containment in the event of a failure of the Zircaloy sheath on the UO_2 rod. Each assembly can accommodate as many as sixteen insulated capsules, four per channel, in an aluminum housing tube (see Figure 13). Stainless steel tubes above and below the capsules in each channel provide the necessary absorbing length to minimize flux peaking in the adjacent fuel elements.

e. Zircaloy Sheath Stock for Swaged Oxide Tubes

Forty feet each of outer and inner Zircaloy-2 sheathing in 8-foot lengths were received from Harvey Aluminum Company for use in UO_2 tube swaging experiments. No cracks or other defects were detected by either visual inspection or a "Zyglo" dye penetrant test; and there were no fabrication marks or scratches on the surfaces. Outside diameter measurements were made at the ends and at 2-foot intervals along the length of the tubing. Inside diameter and wall thickness measurements were made only at the ends of the sheaths. The dimensions are summarized in the following table.

Dimensions of Zircaloy-2 Sheath Tubing
Fabricated by Harvey Aluminum Company

	Outer Sheathing, inches		Inner Sheathing, inches	
Dimension	Average	Range	Average	Range
OD	2.507	2.487 - 2.530	1.529	1.510 - 1.542
ID	2.444	2.417 - 2.472	1.466	1.459 - 1.475
Wall Thickness	0.034	0.031 - 0.037	0.032	0.029 - 0.035
Bow	0.059	0.039 - 0.069	0.054	0.021 - 0.082

f. Welding of End Plugs in Swaged Oxide Tubes

After preliminary tests had shown that an electron beam welder can make satisfactory welds in stainless steel, Zircaloy, and aluminum specimens, an attempt to seal weld a swaged tube of uranium oxide with stainless steel cladding was unsuccessful. Although the cladding and end plug were melted by the electron beam, the fused pieces did not join together; instead, the melted part of the cladding pulled away from the end plug. This behavior may have been caused by outgassing of the oxide or by an unsatisfactory weld geometry. Exploratory tests on the effects of weld geometries and gage thicknesses will be made with the electron beam welder in an effort to obtain satisfactory welds.

B. PROPERTIES OF IRRADIATED CLADDING AND STRUCTURAL MATERIALS

1. Zirconium and Zircaloy

Fifteen tensile specimens of pure zirconium fabricated from extruded rod supplied by Nuclear Metals were assembled into perforated aluminum thimbles for irradiation in a Savannah River reactor at a temperature of less than 100°C. The tensile properties of the specimens after irradiation to about 10^{20} nvt ($>$ 1 Mev) will be compared with those of unirradiated specimens. The results of similar experiments with Zircaloy-2 were reported in DP-405, and measurements with Zircaloy-4 are planned.

A group of six specimens that will be used to measure the effect of irradiation at a temperature of 300°C on the relaxation of stresses in Zircaloy-2 has been inserted in the NRX reactor. These specimens will be discharged from the reactor late in April 1960 after they have been exposed for two cycles. Maximum exposure to neutrons with energies greater than 1 Mev will be approximately 2×10^{19} nvt for specimens in this group.

2. Stainless Steel

An experimental program is in progress at the Savannah River Laboratory to ascertain the effects of irradiation on stress relaxation and creep in stainless steel. Initial results (DP-395) demonstrated that irradiation causes stresses to relax at a temperature (less than 100°C) that is much lower than is otherwise required to induce stress relaxation. To apply these results, it is necessary to distinguish between relaxation of shear stresses and relaxation of normal stresses, since the two types of stress exist in various proportions in reactor structures. Therefore, preparations are being made for a second experiment, differing from the first in that the stainless steel specimens will be irradiated while loaded in torsion rather than tension. In the first experiment, the maximum tensile stress was twice the maximum shear stress, but in the second experiment these stresses will be equal. The combined results of the two experiments should indicate which of the two types of stress governs the relaxation.

The design of the torsion specimens is shown in Figure 14. Each specimen is a seamless tube of 0.200-inch OD and 0.125-inch ID. The specimen is inserted in a stainless steel holder, is deformed a predetermined amount in torsion by a loading device, and is held in the loaded position within the holder by means of set screws that bear on a flat at each end of the specimen. After irradiation, the specimen is disassembled, and the angular offset of the two flats is measured. The difference between this offset and the offset before irradiation will be a measure of the relaxation that occurs during irradiation.

Fifty-four specimens were loaded in torsion to stresses of 5,000 to 25,000 psi and were assembled into perforated aluminum thimbles for irradiation in a Savannah River reactor at a temperature of less than 100°C. After exposure to about 10^{20} nvt ($>$1 Mev), all the specimens will be measured for stress relaxation. Half of the specimens will then be twisted to failure to determine the mechanical properties in torsion of irradiated stainless steel. The remaining specimens, together with 18 unloaded specimens that will accompany the loaded ones during irradiation, will be reserved for a possible second irradiation.

TABLE I

VALUES USED IN SAFEGUARDS ANALYSIS OF BOILING LOOP OF HWCTR

Numerical Values of the Parameters

γ_1 = 0.0477	H_1 = 0.007 MWsec/°C
F = 0.0411 MW/°C (154 gpm)	H_2 = 0.440 MWsec/°C
G = 0.0474 MW/°C (180 gpm)	H_3 = 0.033 MWsec/°C
UA = 0.00807 MW/°C (15,300 pcu/(hr)(°C))	H_5 = 0.233 MWsec/°C
T_w = 22°C	H_6 = 0.824 MWsec/°C
P = 795 psia	H_7 = 0.694 MWsec/°C
c = 1.167 cal/(gm)(°C)	H_9 = 0.112 MWsec/°C
	L = 348 cal/gm

Initial Steady-State Conditions

ψ = 30 MW	T_4 = 257.8°C
f_1 = 0.60	T_7 = 170.8°C
f_2 = 0	T_8 = 170.8°C
f_3 = 0.40	T_9 = 52.2°C
T_1 = 298.1°C	γ_2^* = 0.0409 MW/°C
T_3^* = 268.5°C	

Note: See text for nomenclature

TABLE II

SUMMARY OF LEAKAGE RATES FROM HWCTR GASKETED JOINTS AND TUBING FITTINGS

Leakage Rate, in.^{3}/mo Liquid Water (At 25°C)

Seal Description	Average Rate During First 25 Cycles	Allowable Rate (for Design Purposes)	Average Rate: Heating Phase (at 1000 psi)	Average Rate: At 1500 psi and 260°C	Average Rate: Cooling Phase (at 1000 psi)
"Swagelok" tubing fitting	< 0.01	1.2	< 0.01	< 0.01	< 0.01
"Midlock" tubing fitting	0.09	1.2	0.14	0.10	0.04
Conax packing type tubing fitting	0.29	1.2	0.34	0.38	0.16
Parker flared type tubing fitting[a]	--	1.2	--	--	--
6-inch "Flexitallic" gasket	0.7	94	0.70	0.59	0.81
1-inch jacketed asbestos gasket	0.05	11	0.05	0.06	0.05

(a) Tests results not yet completely analyzed. Preliminary examination of data indicates that the leakage did not exceed the allowable rate.

TABLE III

TYPICAL SERVICE CONDITIONS FOR URANIUM METAL TUBES IN D_2O-MODERATED POWER REACTORS

	Thick Tubes	Thin Tubes		
Coolant state (D_2O)	Liquid	Boiling	Liquid	Liquid
Fuel core thickness, inch[a]	0.265	0.130	0.130	0.130
Specific power, MW/T	36	36	36	72
Heat flux, pcu/(hr)(ft^2)	445,000	218,000	218,000	436,000
Temperature, °C:				
Coolant	230	270	230	230
Clad surface	310	285	269	308
Clad-core interface	374	317	301[b]	371
Core, average	463	338	324[b]	413
Core, maximum	507	349	335[b]	434
Core thermal stress ratio	1.00	0.26	0.26	0.47
Thermal cycle ratio (change in max. core temperature with power)	1.00	0.29	0.38	0.74
Cladding strain ratio	1.00	0.49	0.49	0.49

(a) All the fuel elements in this table have pure uranium cores and are clad inside and out with 0.015 inch of Zircaloy-4. For the initial test elements for the HWCTR (Figure 8), a cladding thickness of 0.025 inch is preferred.

(b) Typical values for full-scale reactor and for HWCTR test assemblies. For test assemblies with 0.025-inch cladding thickness, uranium temperatures will be 20°C higher than the tabulated values.

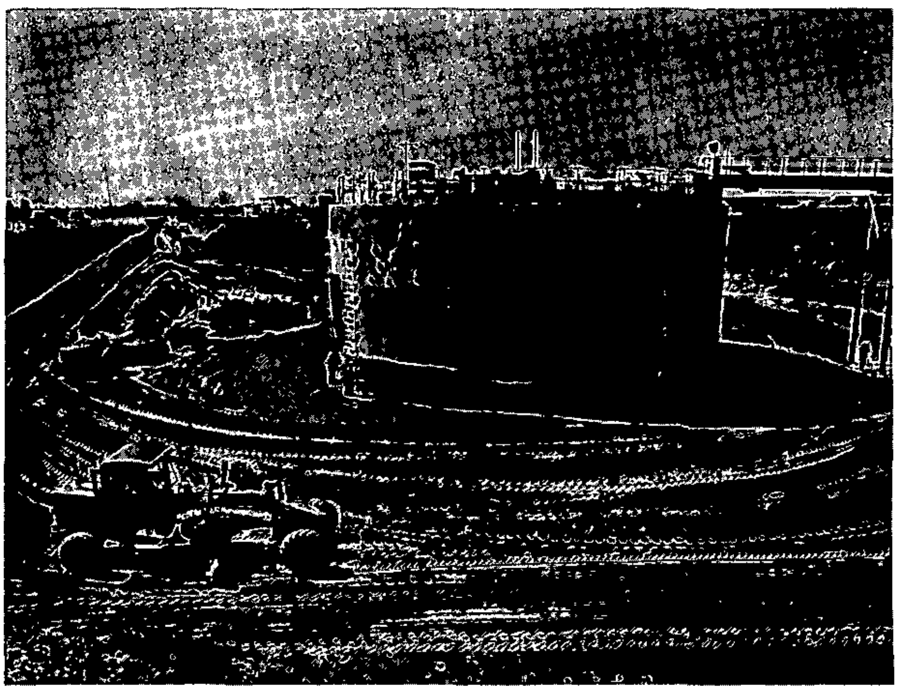


FIG. 1 STATUS OF HWCTR CONSTRUCTION
End of March 1960

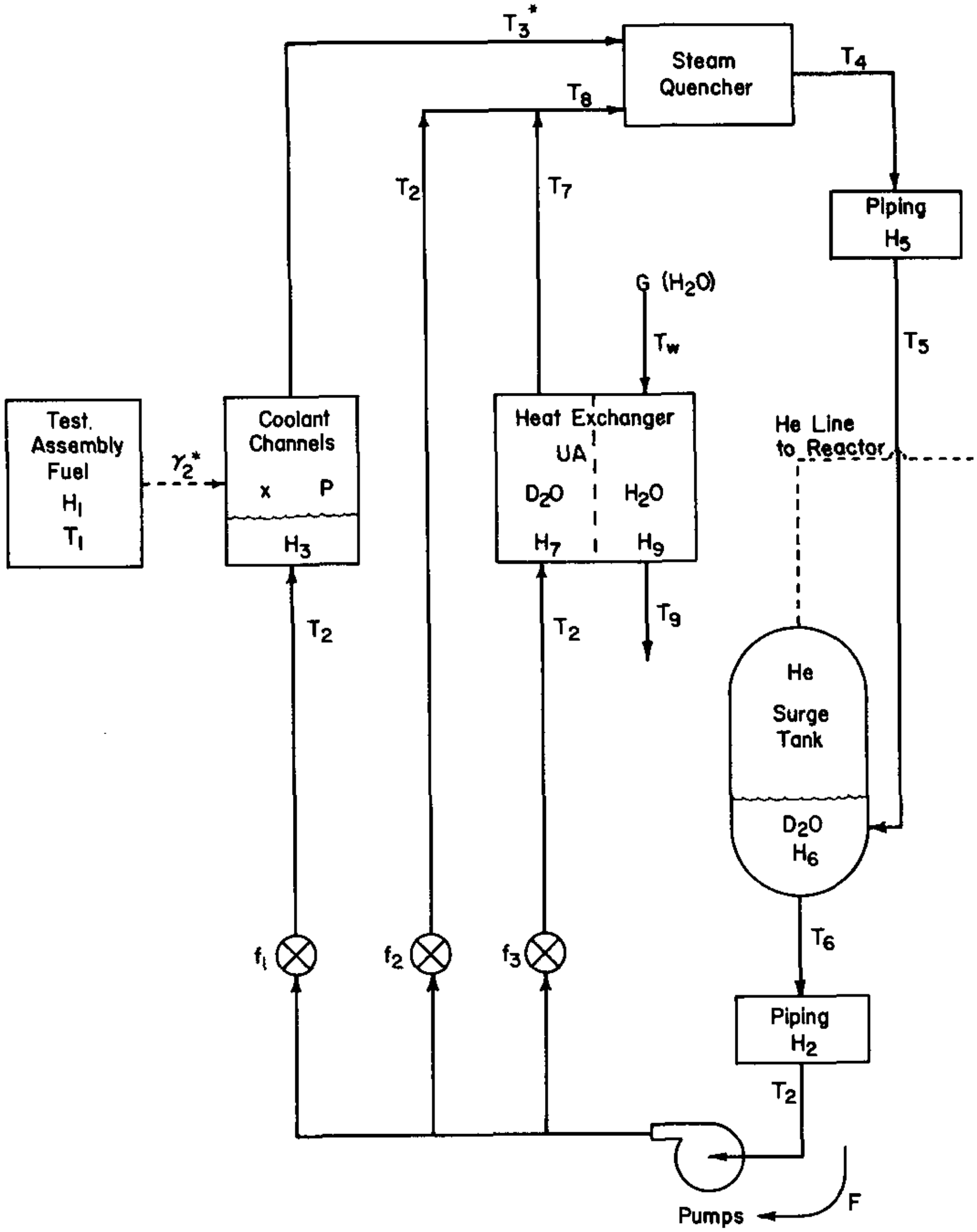


FIG. 2 DIAGRAM OF BOILING LOOP OF HWCTR
See text for nomenclature.

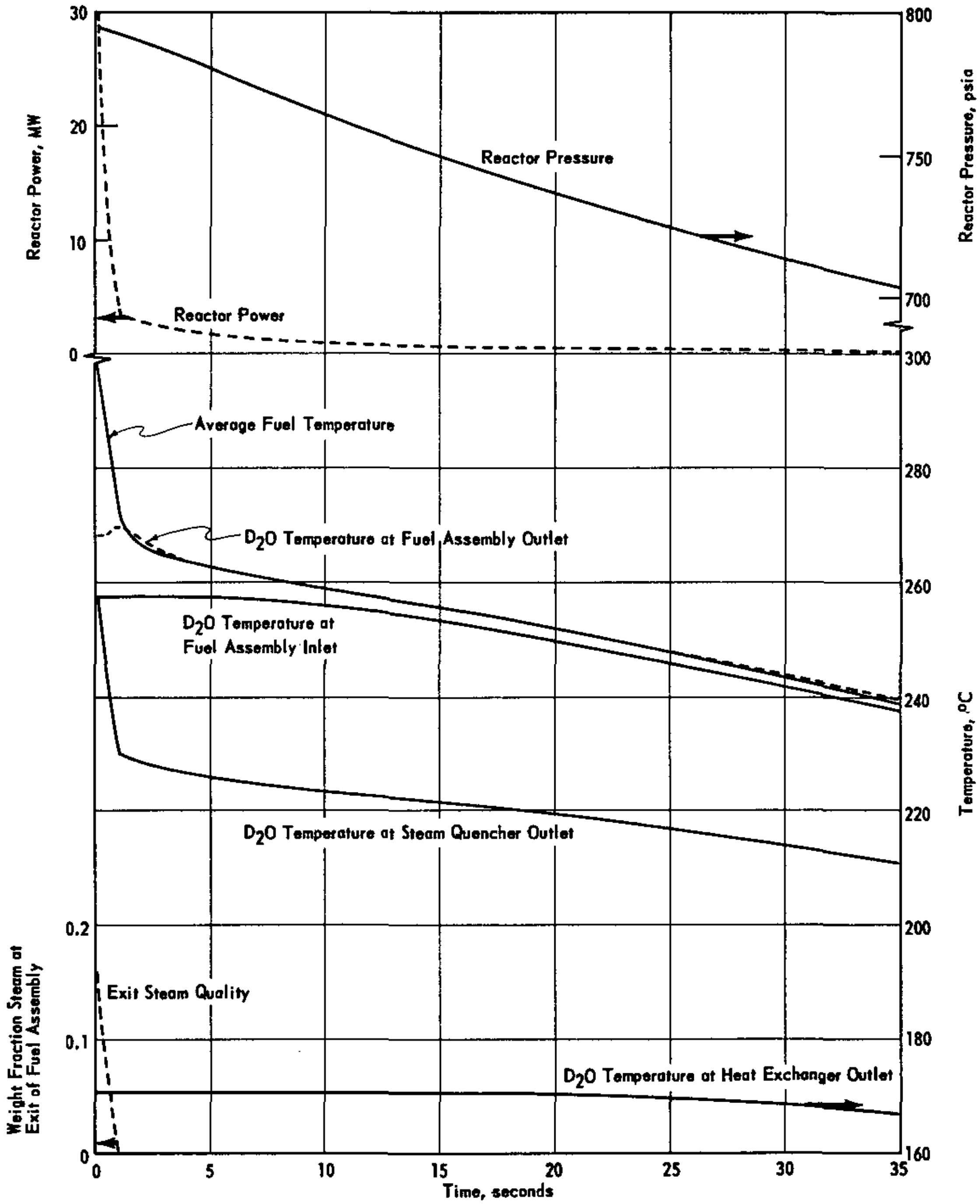


FIG. 3 TRANSIENTS IN BOILING LOOP OF HWCTR FOLLOWING REACTOR SCRAM
Constant Boiler Valve Setting

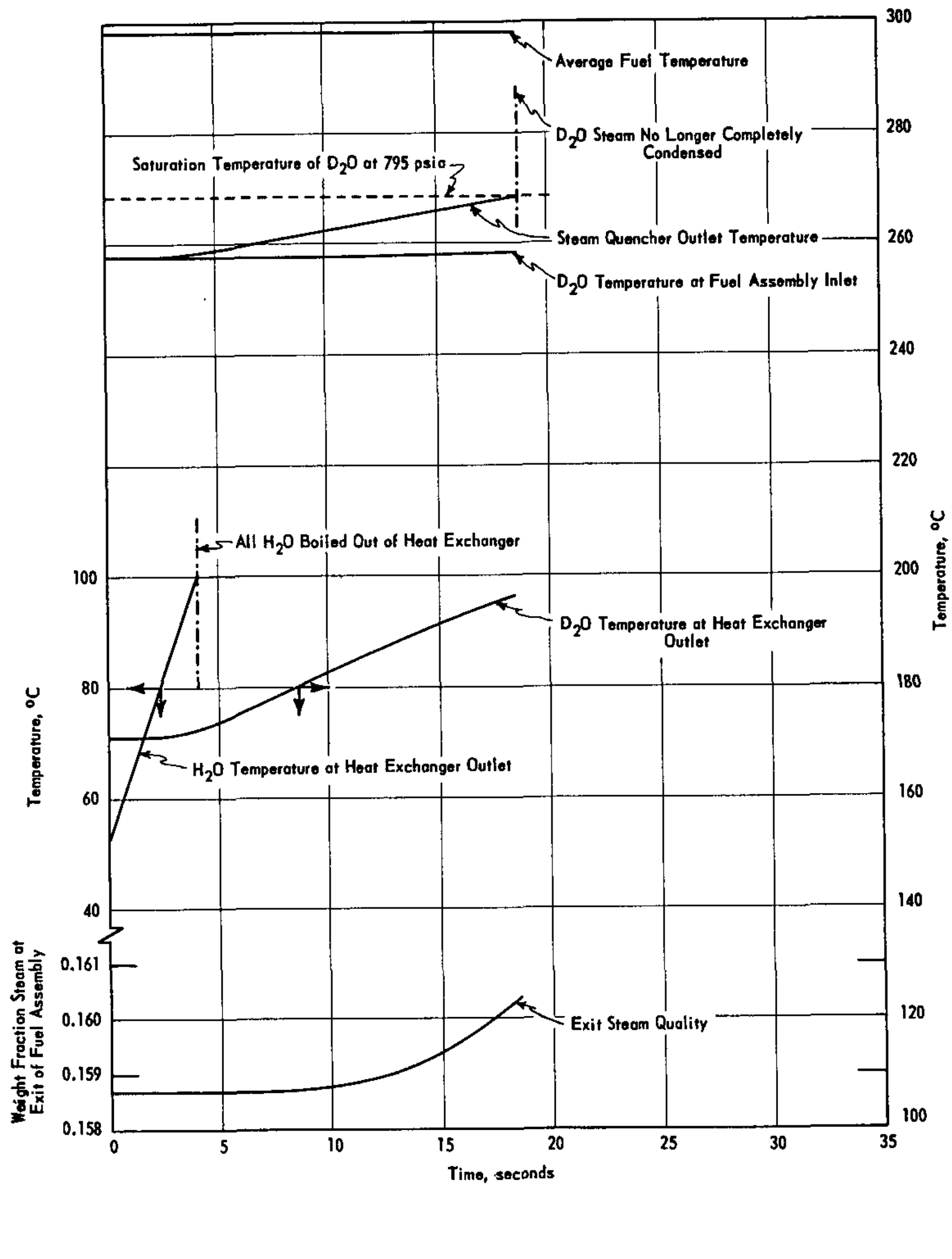


FIG. 4 TRANSIENTS IN BOILING LOOP OF HWCTR FOLLOWING LOSS OF H_2O FLOW – NO SCRAM

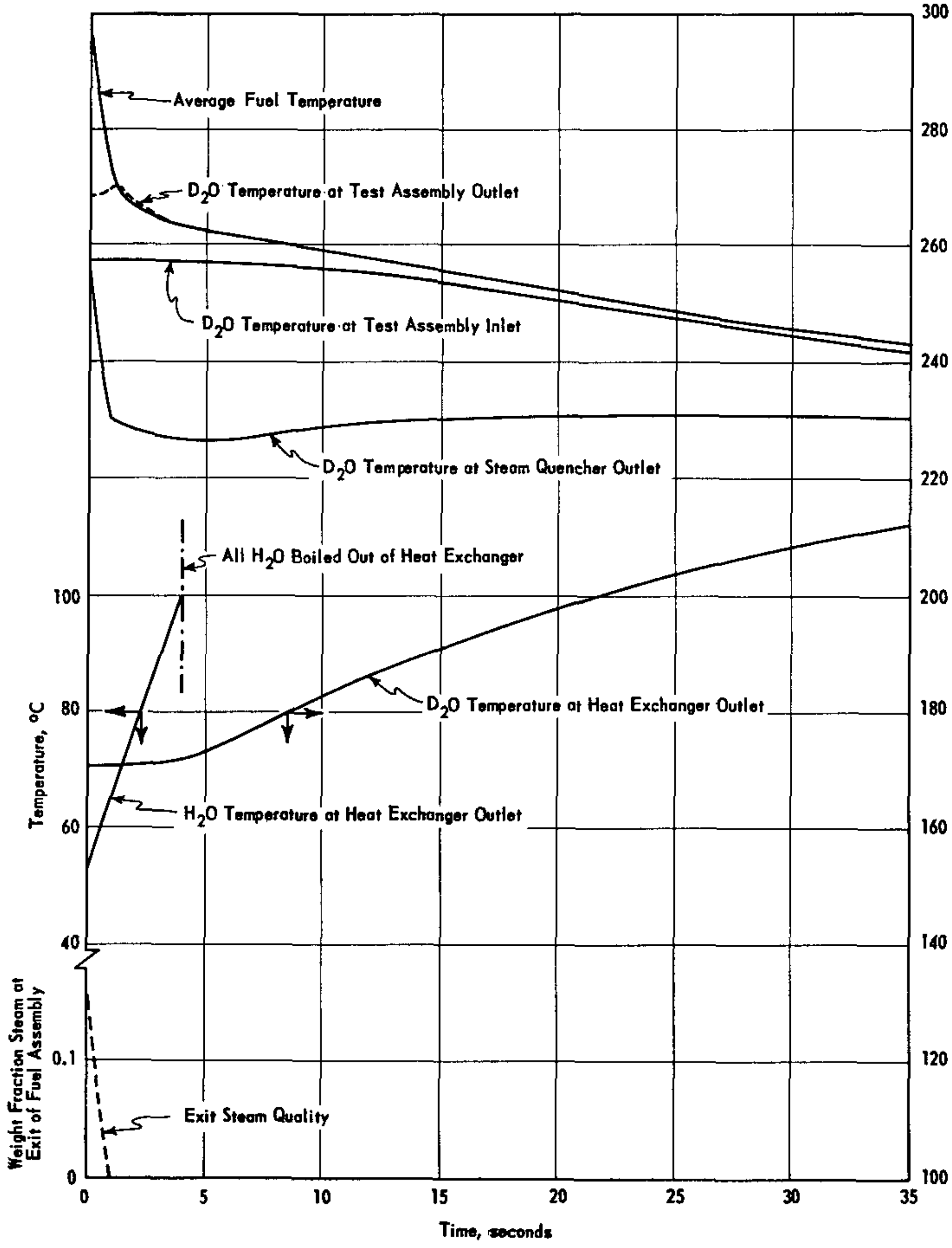


FIG. 5 TRANSIENTS IN BOILING LOOP OF HWCTR FOLLOWING LOSS OF H_2O AND REACTOR SCRAM
Constant Boiler Valve Setting

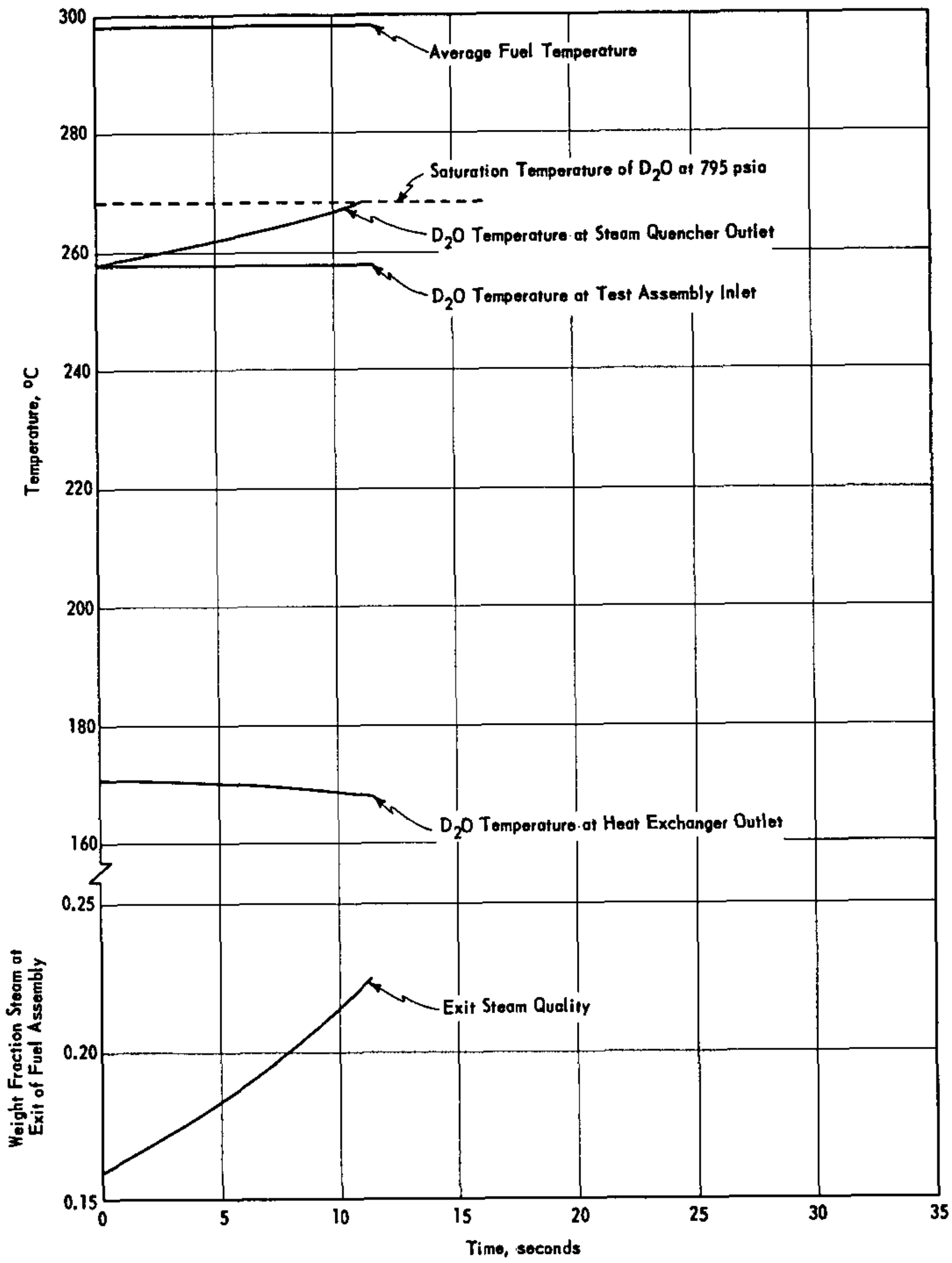


FIG. 6 TRANSIENTS IN BOILING LOOP OF HWCTR FOLLOWING LOSS OF PRIMARY POWER – NO SCRAM
Basis: D_2O Flow Decreases to 1/3 of Its Initial Value in 30 Seconds

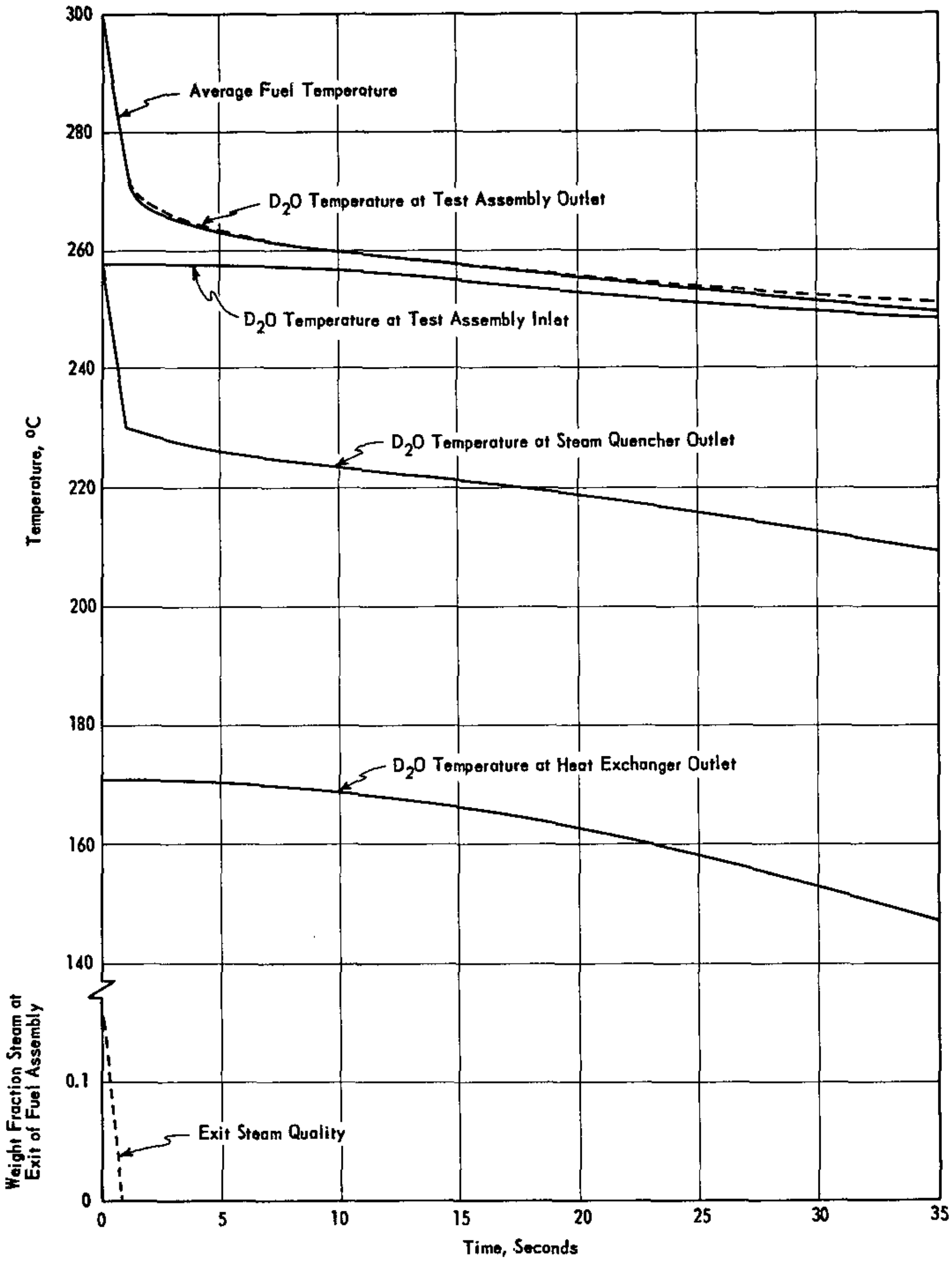
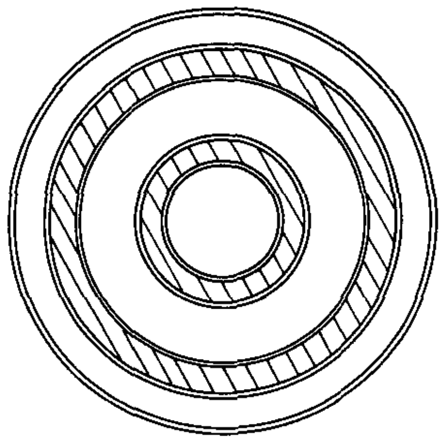


FIG. 7 TRANSIENTS IN BOILING LOOP OF HWCTR FOLLOWING LOSS OF PRIMARY POWER AND REACTOR SCRAM

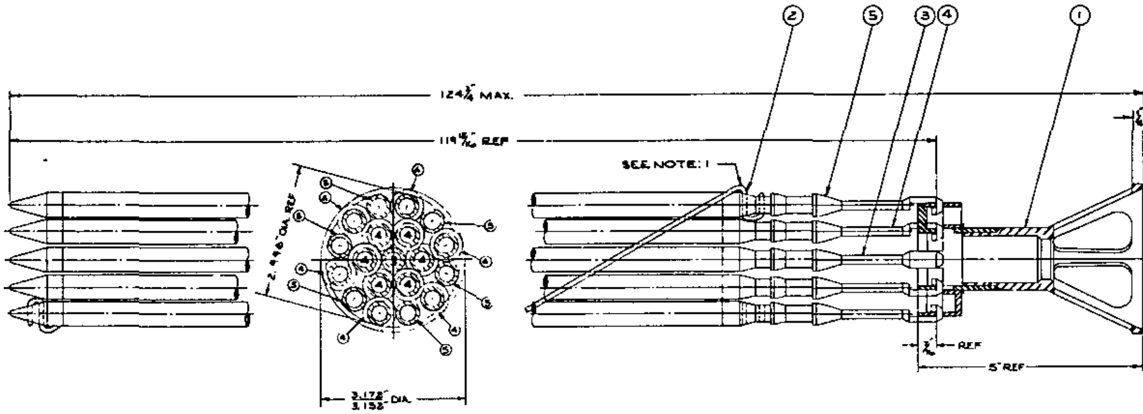
Basis: D_2O Flow Decreases to 1/3 of Its Initial Value in 30 Seconds; Constant Boiler Valve Setting



	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, feet	10	10

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

FIG. 8 THIN URANIUM METAL TUBES FOR HWCTR TESTS



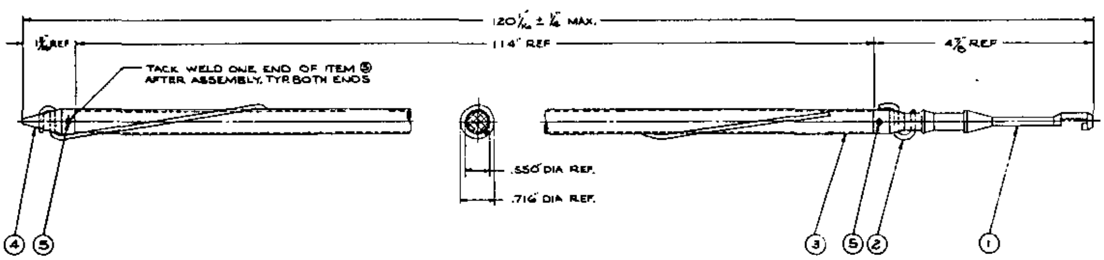
Components

1. Adapter assembly
2. Zircaloy-4 wire (0.083" dia.)
3. Zr-clad oxide rod (center position)
4. Zr-clad oxide rod with wire wrap (See Fig. 10)
5. Zr-clad oxide rod with no wire wrap

Notes

1. Item 2 to be wrapped around bundle on approximately 20" left hand pitch. Start wrap through holes of hook on Item 5 and end on rod tip, Item 5. Tension while wrapping wire not to exceed 15 lbs. Bundle wrap not to cross over any individual rod wrap.
2. "REF" dimensions are nominal.

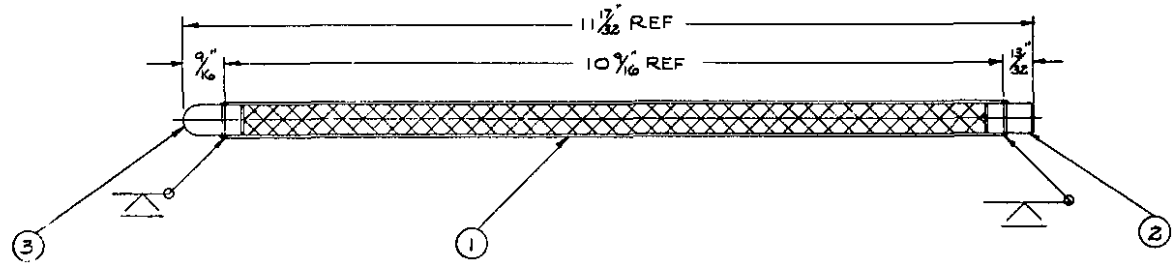
FIG. 9 ASSEMBLY OF URANIUM OXIDE RODS FOR HWCTR IRRADIATION TESTS



Components

1. Rod hook
2. Annealed Zircaloy-4 wire (0.083" dia.)
3. UO_2 rod clad with 0.025" of Zircaloy-4
4. Rod tip
5. "Groov-pin", 1/8" dia. x 1/2" long

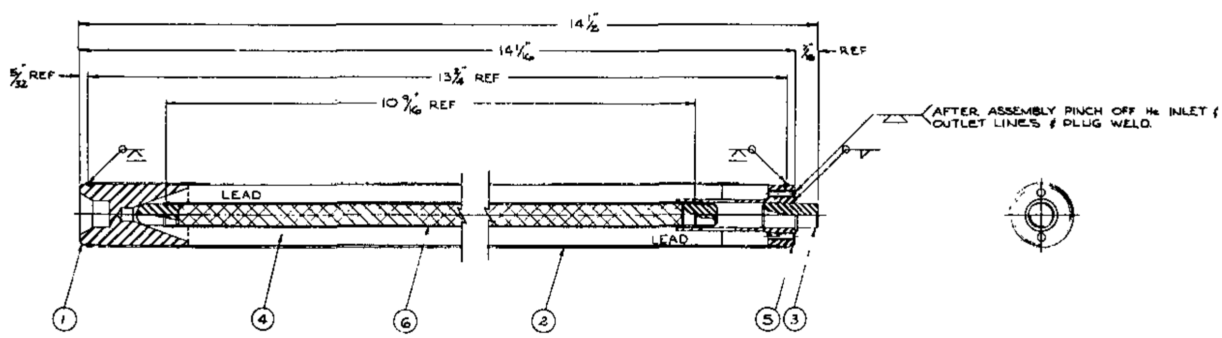
FIG. 10 URANIUM OXIDE ROD FOR IRRADIATION TESTS



Components

1. Swaged UO_2 core with Zircaloy - 2 cladding
2. Top fitting
3. Bottom fitting

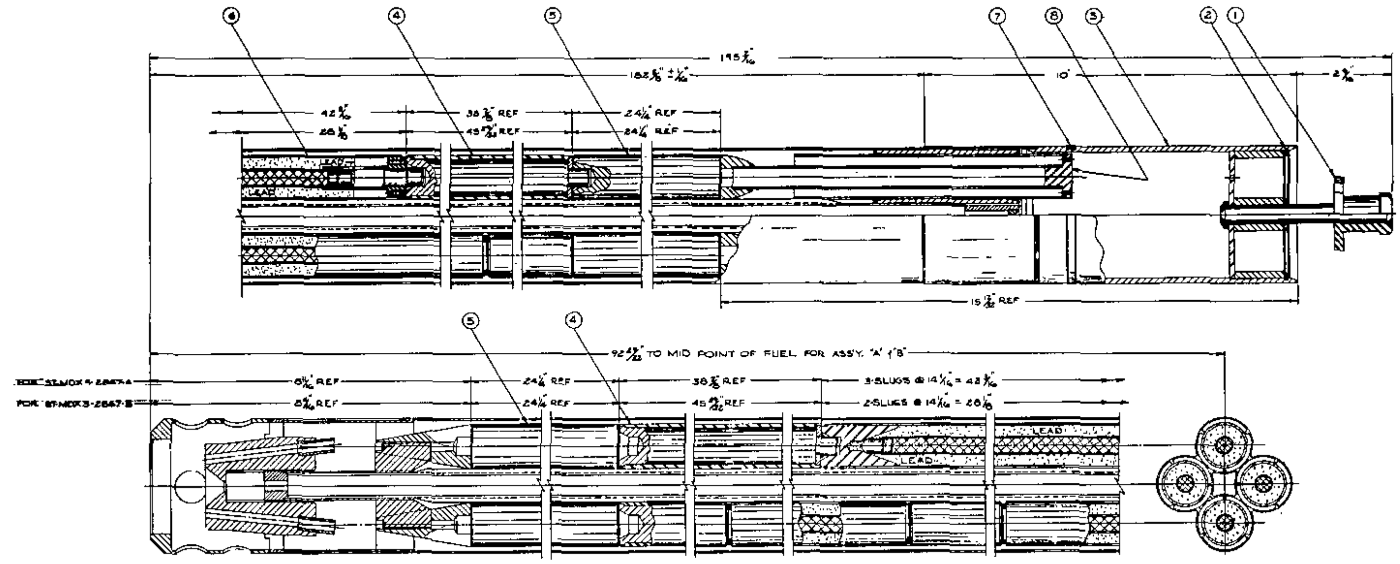
FIG. 11 IRRADIATION SPECIMEN OF SWAGED URANIUM OXIDE



Components

1. Bottom fitting for rod container
2. Lead - insulated container
3. Rod aligning pin
4. Lead ingot
5. Top fitting for rod container
6. Zr - clad rod of uranium oxide

FIG. 12 INSULATED IRRADIATION SPECIMEN OF SWAGED URANIUM OXIDE



Components

1. Stem assembly (for charge - discharge)
2. Retaining ring
3. *Aluminum housing (four contiguous tubes)*
4. Neutron absorber (stainless steel tube)
5. Spacer
6. Insulated specimen of uranium oxide (See Fig. 11 and Fig. 12)
7. Piston ring
8. Coolant orifice assembly

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FIG. 13 ASSEMBLY OF INSULATED URANIUM OXIDE SPECIMENS

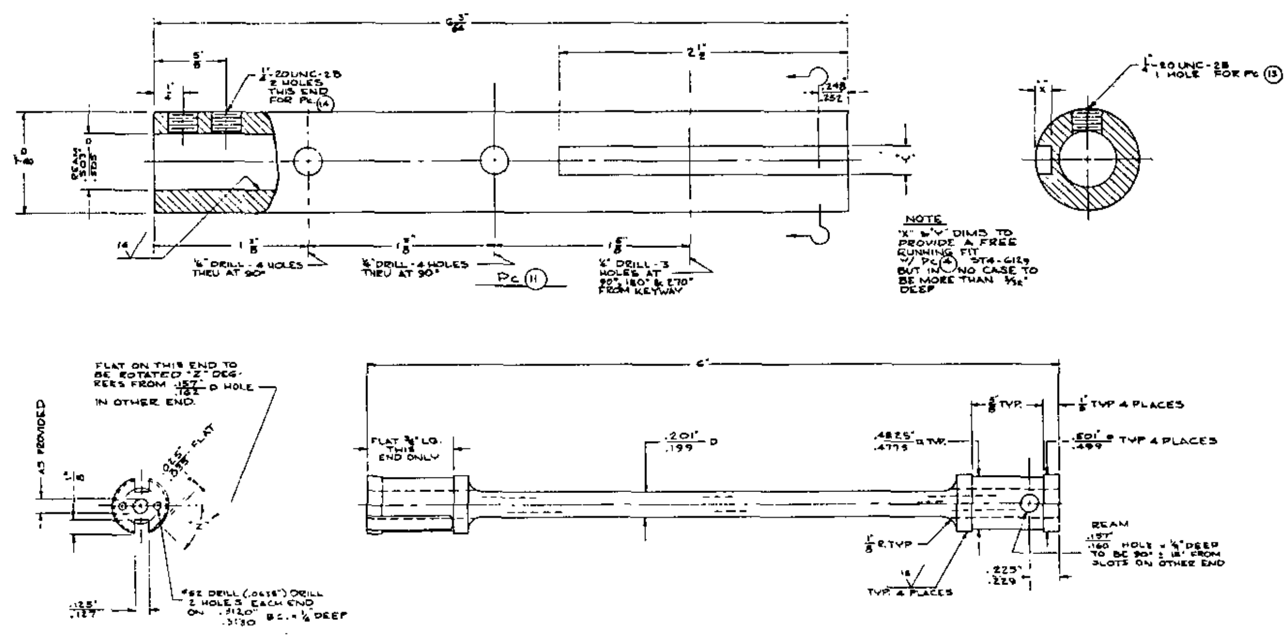

FIG. 14 TORSION STRESS RELAXATION ASSEMBLY