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AEC RESEARCH AND DEVELOPMENT REPORT

HEAVY WATER MODERATED POWER REACTORS

PROGRESS REPORT
JULY-AUGUST 1963

Technical Division
Wilmington, Delaware

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HEAVY WATER MODERATED POWER REACTORS PROGRESS REPORT JULY-AUGUST 1963

D. F. Babcock, Coordinator
Power Reactor Studies
Wilmington, Delaware

Compiled by R. R. Hood

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E. I. DU PONT DE NEMOURS & COMPANY
EXPLOSIVES DEPARTMENT - ATOMIC ENERGY DIVISION
TECHNICAL DIVISION - WILMINGTON, DELAWARE

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ABSTRACT

The HWCTR was operated intermittently during July and August at power levels as high as 38 MW for irradiation of test fuel assemblies of uranium oxide and uranium metal. Near the end of August, a leak developed in one of the two Zircaloy bayonets in the reactor, and the reactor was shut down for examination and possible removal of the bayonet.

Postirradiation examinations of UO_2 fuel tubes revealed that the outer Zircaloy sheath of four tubes that had been fabricated to relatively low UO_2 density (82 to 87% of theoretical) collapsed during irradiation. Fission gas release in tubes that were irradiated to about 3000 MWD/T ranged from 2% at $\int$kdt values of 15 watts/cm to 60% at an $\int$kdt of 50 watts/cm.

Further evaluations of thorium-U^{233} fuel cycles in D_2O-moderated power reactors indicate that this combination offers a greater extension of U. S. uranium reserves than any converter-breeder combination proposed to date. The Du Pont Engineering Department has undertaken a study of the feasibility of designing and constructing D_2O-moderated pressure tube reactors as large as 8300 MW thermal capacity.

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HEAVY WATER MODERATED POWER REACTORS
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INTRODUCTION

This report discusses the progress of the Du Pont development program on heavy-water-moderated power reactors. The over-all goal of the program is to advance the technology of these reactors so that their use in large nuclear stations would permit generation of electricity at fully competitive costs. The principal phases of the program are: (1) the irradiation of candidate fuels and other reactor components under power reactor conditions in the Heavy Water Components Test Reactor (HWCTR), (2) the development of low-cost fuel tubes for use in large D_2O-moderated reactors, and (3) the technical and economic evaluation of various D_2O-reactor design concepts.

SUMMARY

The HWCTR was operated intermittently during July and August at power levels as high as 38 MW for irradiation of test fuel assemblies of uranium oxide and uranium metal. Near the end of August, a leak developed in one of the two Zircaloy bayonets in the reactor, and the reactor was shut down for examination and possible removal of the bayonet.

Postirradiation examinations were started on sixteen UO_2 fuel tubes from two assemblies that failed during irradiation tests in the HWCTR. During the tests, individual tubes in the two assemblies reached a maximum exposure of about 3000 MWD/T at a maximum $\int kdt$ of 50 watts/cm. The principal findings were:

(1) The outer Zircaloy sheaths of four tubes collapsed during irradiation, apparently from the 1200 psi coolant pressure (see Figure 4). The sheath cracked at the apex of the ridge on two of the collapsed tubes. The sheath collapse occurred only in tubes of relatively low UO_2 density (82 to 87% of theoretical) that operated at an $\int kdt$ of 40 watts/cm or higher.

(2) At $\int kdt$ values below about 15 watts/cm, only 2% of the fission gas was released by the UO_2 cores of the tubes. The gas release increased rapidly with thermal rating. Some 60% of the fission gas was released in tubes that operated at an $\int kdt$ of 50 watts/cm. Nitrogen release, which was a problem in earlier

irradiations, was satisfactorily low in all nine tubes for which gas release data were obtained.

Further evaluations of the operation of D_2O power reactors on a thorium-U^{233} fuel cycle indicate that such reactors offer the following attractive features:

(1) Breeding with thorium fuel in a reactor that can be built with current technology and that can produce economical power.

(2) Greater extension of U. S. uranium reserves than with any converter-breeder combination proposed to date, by virtue of the low inventory of fissionable fuel required and the breeding potential.

For purposes of further study, a conceptual design (Table XII) was outlined for a 1000-MWe thorium-fueled D_2O-moderated power reactor. The neutron balance for the design indicates that a 30-year average breeding ratio of 1.02 could be obtained. The uranium consumption in D_2O reactors operating at this breeding ratio is modest enough that there would be little or no incentive to develop designs capable of breeding ratios as high as 1.05 before the beginning of the next century.

The Du Pont Engineering Department has undertaken a study of the feasibility of designing and constructing D_2O-moderated pressure tube reactors as large as 8300 MW thermal capacity. The study thus far has been concerned only with liquid D_2O coolant; organic coolant will be considered later. A design concept that appears feasible has been developed for a 3500 MW(t) reactor cooled by liquid D_2O. Design problems that have been examined to date include the structural arrangement of the reactor and the coolant piping, design of the pressure tube assemblies, differential thermal expansion in various parts of the reactor structure, deformation of the shield structures, and leak-tightness of the pressurized D_2O system.

One-region bucklings and void effects were measured for D_2O-moderated lattices of nested uranium tubes in critical experiments at the Savannah River Laboratory. Measured bucklings were generally in fair agreement with those calculated by means of the buckling code currently used at Savannah River for calculation of D_2O lattice parameters.

DISCUSSION

I. THE HEAVY WATER COMPONENTS TEST REACTOR (HWCTR)

The HWCTR is a D_2O-cooled-and-moderated test reactor in which candidate fuel assemblies and other reactor components are being evaluated under conditions that are representative of large D_2O-moderated power reactors.[1] Currently, fuel assemblies of uranium metal (coextruded with Zircaloy cladding) and assemblies of uranium oxide (mechanically compacted in Zircaloy sheaths) are being irradiated in this reactor.

A. REACTOR OPERATION

The HWCTR was operated intermittently during this report period at power levels as high as 38 MW for irradiation of test fuel assemblies. Reactor operation is summarized in Tables I and II, and in Figures 1 and 2. Irradiation tests currently in progress are listed in Table III. No fuel failures were experienced during July and August.

Near the end of August, the HWCTR was shut down for investigation of apparent helium leakage into the moderator. This leakage was evidenced by a 50% increase in dissolved helium concentration in the D_2O. Tests made subsequent to reactor shutdown revealed a leak in the Zircaloy bayonet of the boiling D_2O loop. This leak is indicative of some type of failure in the bayonet, and at the end of this report period the bayonet was being examined by means of a borescope in an effort to locate the failure.

B. VIBRATION DAMAGE IN FUEL ASSEMBLIES

It was reported in DP-855* that the stainless steel top fittings on a fuel assembly containing a 10-ft-long UO_2 tube separated from the fuel tube and the housing tube during irradiation of the assembly in the HWCTR; therefore, normal methods for removal of the assembly from the reactor could not be employed. By the use of special gripper tools, this assembly was subsequently removed from the reactor without incident. The fitting failure was tentatively attributed to vibration of assembly components during HWCTR operation, and eight other irradiated UO_2 assemblies with top fittings of the same general design (Figure 3) were inspected. Three of these

* Progress reports in this series are not referenced individually.

contained full-length (i.e., 10-ft-long) fuel tubes; the remaining assemblies comprised columns of 14-in.-long fuel tubes. Sufficient evidence of severe vibration wear was observed to warrant (a) discontinuing further irradiation of certain assemblies, (b) substituting redesigned fittings at the upper ends of the fuel columns in all assemblies that were returned to the HWCTR, and (c) initiating redesign of the coolant flow restrictor upstream of each oxide fuel assembly.

The observed vibration damage is summarized in Table IV. Well-defined wear grooves 1 to 2 ft long and up to 22 mils deep, produced by vibratory contact with the housing tube ribs, were found at the upper ends of the full-length fuel tubes. The corresponding damage at the upper ends of the columns of short fuel tubes was generally less severe, with the wear distributed circumferentially because of rotational movement of the short tubes; the maximum groove depth on the cladding was 10 mils. Rotation of the full-length fuel tubes did not occur, presumably because of their greater weight and length of rib contact.

Although there was vibration wear at the upper ends of the fuel columns in two assemblies that failed in the HWCTR, the fuel failures in each case were caused by buckling of the cladding, as described in Section II of this report.

The observed vibration damage was attributed to a combination of design features favoring vibratory movement of the upper ends of the fuel columns: (a) generation of large-scale eddies by the 2-in.-diameter flow-restricting orifice 10 in. upstream of the fuel gripper fitting, (b) insufficient lateral support (i.e., ribs) for the gripper fitting, and (c) excessive diametral clearance (60-90 mils instead of the normal 30) between the fuel column and the housing tube ribs. The flow restrictor is being redesigned to incorporate small-diameter orifices, which will produce smaller eddies that can be more completely dissipated before impinging on the fuel column. The gripper fittings in the tubular oxide assemblies now under irradiation were modified by adding to each of them six 3-in.-long ribs providing a diametral assembly clearance of 15 mils. Insofar as the availability of suitable components permits, the clearance between fuel and housing tube ribs will be reduced to about 30 mils in test irradiation assemblies to be fabricated in the future.

Three assemblies of short fuel tubes were returned to the HWCTR after the inspection discussed above. The top fuel tube in each of the three assemblies was replaced by a dummy stainless steel tube of which the gripper fitting was made

an integral part. Further irradiation of the full-length oxide tubes was deferred pending consideration of the possibility that the strength of the cladding at the point of maximum groove depth might not be sufficient to withstand the internal gas pressure at shutdown.

C. HEAVY WATER LOSSES

During the latter half of July, D_2O inventory data indicated that the over-all leakage rate of D_2O from the HWCTR was about 50 lb/day, as compared to 30 lb/day during earlier periods of sustained reactor operation. The reactor was shut down on July 24 for investigation of the high leakage rate; five leaks were discovered -- three at leakoff points in valve packings, and two at connections on fuel monitor pins. After these leaks were eliminated, the leakage averaged about 40 lb/day during one week of reactor operation. Attempts were made during this period to ascertain the sources of leakage by means of air samples throughout the building and by detailed scrutiny of D_2O inventories. These steps were helpful in locating additional leakage points during a subsequent shutdown when a total of seven water leaks and eight gas (He + D_2O) leaks were discovered. At the end of the report period, more extensive leakage surveys, repairs of known leaks, retightening of all high-pressure flange bolts and connectors, and installation of leakage detectors were in progress.

The experience cited above prompted detailed re-examination of the program to reduce D_2O losses in the HWCTR. The primary objective of the program is to obtain and analyze data that will be of value in the future design of large D_2O power reactors. The first step in the program is to reduce the leakage through all seals, flanges, connections, and packings to an amount that appears representative of the components when all reasonable maintenance is provided. When this condition is achieved, the next step will be to identify the sources and sizes of outstanding leaks. This information will provide a basis for evaluating the amount of D_2O losses in proposed power reactors, and will enable an estimate to be made of the benefits to be expected in improvements and modifications of particular equipment or component items.

II. REACTOR FUELS AND MATERIALS

Two types of Zircaloy-clad fuel elements are currently under development in the Du Pont program on D_2O power reactors: mechanically compacted tubes of uranium oxide and coextruded tubes of uranium metal. In the development of these two

alternative fuels, primary attention is being given to fuel element designs and fabrication methods that offer promise of low fabrication costs when produced in the volume required for several full-scale reactors.

A. FUEL TUBES OF URANIUM OXIDE

1. Fabrication Development

Nitrogen release from fused UO_2 during reactor operation is objectionable because it is a source of pressure buildup within oxide fuel elements. In the past, postirradiation examination of Zircaloy-clad tubes of crushed, fused UO_2 frequently revealed large quantities of free nitrogen in the cores (DP-645, DP-745, DP-755, DP-765). Consequently, studies have been made of methods of removing the nitrogen prior to fabrication of oxide tubes. It was reported in DP-845 that a high-temperature heat treatment in moist hydrogen is an effective method. More recent experiments show that vacuum outgassing is also effective. In these experiments, the nitrogen content of fused UO_2 particles ranging in size up to -3 mesh was reduced from 300 ppm to less than 1 ppm by vacuum treatment for 4 hours at an estimated average temperature of 1400°C. These findings are supported by the results of postirradiation examination of the SOT-2 oxide tubes, which contained vacuum-outgassed UO_2 (see Section III A3). Comparison of the gas release data in Table VII with those reported in DP-745, Table VII, and DP-765, Table IV shows that the release of nitrogen from fused UO_2 during irradiation was reduced by the vacuum-outgassing treatment.

2. Fabrication of Fuel Tubes for Irradiation Tests

Fabrication was completed on a group of short 2-in.-OD UO_2 tubes in Zircaloy sheaths (designated SOT-7 tubes) that are candidates for irradiation in the HWCTR. Seven of these 14-in.-long tubes are to be irradiated with the objectives of characterizing the behavior of swage-compacted UO_2 tubes at very high thermal ratings (maximum $\int kdt$ of 75 watts/cm) and obtaining additional information on the irradiation behavior of UO_2 that is vacuum outgassed or annealed in moist hydrogen for removal of nitrogen.

3. Postirradiation Examinations

(a) General

Postirradiation examinations were started on short UO_2 fuel tubes from two assemblies that failed during operation in the HWCTR. The results of inspection of sixteen tubes from these two assemblies (designated SOT-2-2 and SOT-2-3) are summarized below. More detailed information is presented in subsequent paragraphs.

(1) The outer sheaths of four tubes collapsed during irradiation to form a ridge running the length of the core (Figure 4).

(2) The sheath cracked at the apex of the ridge on two of the collapsed tubes, thereby releasing gas to the reactor coolant.

(3) Volume decreases from 2.4 to 8.7% occurred in the four elements with collapsed sheaths. The core densities of these elements, all of which were vibratorily compacted to initial densities of 82 to 87% of theoretical, increased to about 88%.

(4) The UO_2 in the collapsed elements migrated into the ridge and, in at least one case, migrated into a gas chamber at the end of the element.

(5) Only small amounts of nitrogen were released by the UO_2 during irradiation.

(6) More than one-half of the fission product gas that formed during irradiation was released in elements operating at an $\int$kdt of 40 to 50 watts/cm. Approximately 2% of the fission gas was released in elements operating at an $\int$kdt of 15 watts/cm or less.

(7) One element contained a large amount of deuterium that must have entered the element as D_2O during irradiation.

(8) Minor sheath collapse occurred at one end of a swage-compacted element; it is believed that the as-fabricated density of the core in the collapsed region was lower than the usual 90 to 92% of theoretical.

(9) Almost all of the Zircaloy studs holding the stainless steel fitting to the top fuel piece in each assembly (see Figure 3) were broken, and the 8-in.-long fittings were either loose or unattached.

(b) Operating Histories

The sixteen fuel tubes in the two assemblies that failed were fabricated from natural UO_2 by vibratory compaction alone or by vibratory compaction plus subsequent swaging. Each tube was sheathed in Zircaloy and was fitted with end plugs containing gas spaces. The purpose of these plugs (DP-775) was to decrease the buildup in gas pressure within the fuel tubes during irradiation. The gas chambers roughly doubled the available gas volume within each fuel tube.

The operating histories of the two assemblies were as follows; more detailed information on the individual fuel tubes is presented in Table V.

Assembly designation	SOT-2-2	SOT-2-3
Total assembly exposure, MWD	54	49
Maximum specific power, MW/ton	39	41
Maximum specific exposure, MWD/ton	3200	2900
Maximum heat flux, pcu/(hr)(ft^2)	335,000	344,000
Maximum $\int$kdt, watts/cm	50	49
Number of scrams from power	7	7
Total time in reactor, days		
at full coolant flow	96	89
at shutdown flow	57	21

(c) Visual Appearance

The outer sheath on four elements collapsed to form a ridge running the length of the core (Figure 4). Tubes 251C and 253C developed small cracks in the Zircaloy sheath at the apex of the ridge (Figure 5). Sheath collapse of the type shown in Figure 4 occurred only in tubes with low initial densities that operated at high thermal ratings. Low-density elements operating at low $\int$kdt ratings (16 watts/cm or below) did not collapse nor did swaged elements when operated to $\int$kdt ratings up to 50 watts/cm. The four collapsed tubes were fabricated by vibratory compaction alone to core densities of 82 to 87% of theoretical; the lower the initial density in these tubes, the more pronounced the ridge.

Incipient sheath collapse occurred at one end of a swaged tube. Collapse is believed to have occurred over this region because of an unusually low oxide density. Low core densities at the ends of swaged tubes are sometimes observed after a portion of the UO_2 is removed mechanically to allow insertion of a gas chamber.

(d) Dimensional Changes

Large decreases in volume and average outer diameter occurred in the collapsed tubes during irradiation. Changes in dimensions for the remaining tubes ranged from +0.002 to -0.002 inch for the mean diameters and +0.1 to -1.3% for the volume. Volume and diametral changes observed in the collapsed tubes are listed in Table VI. In these tubes, the volume decreases ranged from 2.4 to 8.7%, and the mean decrease in OD ranged from 0.014 to 0.054 inch; at the ridge the OD increased as much as 0.105 inch. The final core densities calculated from the volume changes are consistent with the observations on the ridge shapes. After irradiation, the core densities ranged from 87 to 89% of theoretical. Four swaged elements with initial densities of about 90% of theoretical operated wholly or in part in the $\int kdt$ range of 40 to 50 watts/cm without collapse. These results indicate that core densities at least this high are necessary to prevent collapse of oxide tubes operating at these ratings.

(e) Core Cross Section

When two of the collapsed tubes were sectioned, the UO_2 core was observed to fill the sheath ridge and to be in close contact with the sheath, as shown in Figure 6. The extent of the columnar grains which are present in the ridge and elsewhere indicates that the ridge was formed and filled with UO_2 for some appreciable period of time before nuclear shutdown.

(f) Gas Release

The free gas in nine of the tubes was collected, measured, and analyzed; the amounts and compositions are shown in Table VII. The nitrogen content of the gas was satisfactorily low in each case. The fission gas release varied from about 2 to 60% of that formed. The xenon release is plotted in Figure 7 against the maximum $\int kdt$ that was reached during irradiation. These data show that at an $\int kdt$ above about 30 watts/cm, irradiations to high exposures would be complicated by the buildup of internal pressure from fission gas release. The gas from three additional tubes will be collected and analyzed to provide data points at $\int kdt$ values of 23 to 32 watts/cm.

B. FUEL TUBES OF URANIUM METAL

1. Postirradiation Examinations

Two sound assemblies of uranium tubes that were removed from the HWCTR in December 1962 after irradiation to low exposures were examined to complete preliminary inspection of the first group of experimental metal tube assemblies that were irradiated in the HWCTR. Aside from the expected elongation of the twisted-ribbon annular spacers (see DP-845), the two assemblies were essentially undamaged.

A summary of the inspection results and irradiation conditions is presented in Table VIII, which supplements Tables IV and V of DP-845. Both assemblies were taken apart without difficulty. Vibration damage was inconsequential on all surfaces of the fuel tubes and on the twisted-ribbon spacers. With the exception of a few small spots where the oxide coating on the tubes was worn shiny but not further penetrated, the tube surfaces appeared to be in as good condition as when irradiation began. The diameter and length changes of the fuel tubes were very small. The increases in length of the twisted-ribbon spacers, caused by plastic deformation as the result of differential expansion of fuel tubes and ribbons, ranged from 3/8 to 9/16 inch.

III. REACTOR EVALUATION STUDIES

A. THE USE OF THORIUM IN D_2O-MODERATED POWER REACTORS

In March 1963, the Du Pont Company submitted a report[2] to the AEC on the current status of the U. S. development program on heavy-water-moderated power reactors. Included in this report were the results of very preliminary evaluations of self-sustaining thorium-U^{233} breeding cycles. It was concluded from these evaluations that such cycles are an attractive long-range possibility for D_2O reactors, and it was recommended that D_2O-thorium systems be investigated further as a possible means of increasing the utilization of U. S. uranium reserves. Subsequently, the AEC requested Du Pont to continue its evaluations of thorium-fueled, D_2O-moderated reactors. The following paragraphs contain a discussion of the latest results of these evaluations and a review of some of the general considerations relating to the use of thorium in D_2O-moderated reactors.

1. Breeding in Thermal Reactors

(a) General Considerations

In considerations of breeding in thermal reactors with the thorium-U^{233} fuel cycle, a clear distinction must be made as to the significance of the following four factors: breeding ratio, doubling time, conservation of fuel resources, and fuel economics. A high breeding ratio is of value only if it leads to a short doubling time; a short doubling time serves little purpose if it does not conserve fuel resources; and fuel conservation itself is of questionable value unless for some postulated set of economic conditions this conservation leads to lower fuel cycle costs.

The D_2O-moderated thermal breeders discussed in this report are based on consideration of all four of these factors. The reactors do not represent the maximum possible breeding ratio or the minimum doubling time achievable with solid-fueled D_2O reactors. Instead, they represent a judgment of the compromises that will be necessary to achieve excellent conservation of fuel resources while retaining those features of D_2O reactors that are essential to low-cost power generation in the near future. It will be shown below that the breeding gain in any of the D_2O systems considered is too small to be an important factor in fuel cycle costs during the next 50 years or so.

(b) Breeding Ratio

The design factors that affect the breeding ratio are neutron economy, neutron energy spectrum, the neutron flux level in the thorium, and the fuel lifetime in the reactor.

The neutron economy in D_2O reactors is at least as good as that in any other practical reactor. Thermal utilization factors as high as 0.96 can be achieved. However, attempts to obtain maximum neutron economy by increasing the reactor fuel loading may be self-defeating because of the resultant hardening of the neutron spectrum.

The neutron energy spectrum is important because of the energy dependence of the neutron reproduction factor, η, for U^{233}. The value of η for thermal neutron absorptions in U^{233} is 2.29, while that for resonance absorptions is only 2.10. The latter value is little higher than the 2.07 of U^{235}. Also, the resonance integral of U^{233} is about three times that of U^{235}. Resonance absorptions therefore cause a significant reduction in η and in the breeding gain.

Neutron absorptions in Th^{232} lead to Pa^{233}, which decays to U^{233} with a 27-day half-life. The thermal absorption cross section of Pa^{233} is sufficiently large that if the neutron flux in the thorium is high, an important loss of neutrons and of potential U^{233} can occur due to absorptions in Pa^{233}. This undesirable side reaction has led in the past to the consideration of fluid fuel reactors from which the Pa^{233} could be extracted and stored outside the reactor until it decayed to U^{233}. It has also led to reactor designs in which the thorium would be placed in a blanket region while most of the reactor power would be generated in a central fuel region. These designs are disadvantageous in that economical use of reactor core volume requires the generation of power uniformly in all fuel positions, and it is essential for economical fuel cycles that the fuel remain in the reactor until a large fraction of the initial U^{233} is burned (to reduce fuel processing costs and losses). For these reasons, the U^{233} is dispersed in the thorium in the reactors discussed in this report. The neutron losses by absorptions in Pa^{233} are therefore directly proportional to the specific powers at which the Th-U^{233} fuel elements are operated.

The neutron loss to fission product poisons increases with average fuel exposure. For example, the breeding ratio is decreased about 1% when the average fuel exposure is increased from 10,000 to 15,000 MWD/T.

(c) Doubling Time

The important factors that affect the doubling time are (a) the breeding ratio, (b) the specific power, in terms of the total inventory of fissionable isotopes inside and outside of the reactor, and (c) the reprocessing losses, which are dependent not only upon the loss each time the fuel is reprocessed and refabricated, but also upon the fraction of the fissionable material that is burned in the reactor between reprocessing steps. In this regard, D_2O-thorium breeders appear capable of specific powers that are several times those of fast breeders, and the D_2O reactors can burn the equivalent of 60 to 300% of the initial U^{233} in the core between reprocessing steps.

(d) Conservation of Fuel Resources

Two of the important factors in the conservation of uranium resources are: the doubling time and the inventory (related to the specific power) of fissionable material required to start the reactor. In a power generation economy

that grows rapidly compared to the doubling time, the initial inventory of fissionable material is a far more significant factor than is the doubling time. It is the low inventory that gives D_2O reactors such a large advantage over fast breeders in terms of the amount of uranium that must be mined to allow nuclear power generation to keep pace with the growth of electrical requirements over the next 50 years or so.

(e) Fuel Economics

The important factors in fuel cycle economics include: credit for bred fuel, which depends upon the doubling time, fuel inventory charges, fuel life per reprocessing step, and reprocessing and refabrication costs. In the D_2O reactor design presented in Section III A3 of this report, the inventory charges and the credit for bred fuel in the present-day economy each amount to no more than 0.2 mill/kwh, while reprocessing and refabrication costs are 0.7 mill/kwh for a 15,000-MWD/T fuel life. Thus, the characteristically low inventory of D_2O reactors makes it attractive to de-emphasize the breeding ratio and doubling time in favor of longer fuel life and lower fabrication and reprocessing costs.

With natural uranium at \$5/lb of U_3O_8, the burnup charges in a D_2O reactor operating on a uranium-plutonium cycle are about 0.5 mill/kwh and the inventory charges amount to about 0.1 mill/kwh. These costs are so low that there is little incentive now to lower fuel costs by the development of breeder reactors. It probably will be many years before uranium consumption has grown sufficiently to cause significant increases in fuel costs. This fact allows the orderly development of U^{233} breeding in heavy water reactors and places emphasis in the near-term on D_2O reactors that are capable of producing power economically.

(f) Startup of Fuel Cycles

The fissile fuel for starting a prototype or a full-scale D_2O-thorium reactor can be either U^{233}, U^{235}, or Pu^{239}. Studies must be made to determine which of these alternatives is best. For the prototype, U^{233} mixed with thorium in the proportion calculated to exist at steady state has the outstanding advantage that this fuel will give the desired data at the earliest time and with the least uncertainty as to interpretation. The expense of producing U^{233} will be somewhat greater than for producing Pu^{239} or U^{235}. Presumably, one of these latter two nuclides would be used to start full-scale D_2O-thorium reactors.

2. Neutron Economy and Fuel Utilization

(a) Neutron Balance

The neutron balance for thorium-fueled D_2O reactors is presented in Table IX. The values chosen for the reactor discussed in the next section are listed in this table. Also shown are the ranges of values that are within the control of the reactor designer. The first figure in each range corresponds to a low specific power, short fuel exposure, and an efficient blanket; the second figure corresponds to a high specific power, long fuel exposure, and no blanket. The two values of η are for thermal neutrons (2.29) and for neutrons with intermediate energy (2.10). In any actual reactor, η must be somewhere between these limits.

The "net breeding ratio" is the value obtained with thorium-U^{233} fuel without allowance for the buildup of U^{235} and heavier nuclides. The "30-year average breeding ratio" includes the decrease in η due to buildup of these heavy nuclides averaged over 30 years of reactor operation.

In Table X, the neutron balances are compared for several types of thorium-fueled reactors. Although each of the listings is taken out of its original context, the detailed comparison allows an estimate of what could be done with the different reactor types. The data in Table II for the homogeneous reactor apply to isotopically pure U^{233} fuel. Eventually, the buildup of U^{235} would reduce η and the buildup of U^{236} would cause neutron losses; these two effects would combine to reduce the average breeding ratio by about 0.02. The figures for the spectral shift reactor are for a nonbreeder optimized for economical power generation. Edlund[7] has indicated that a conversion ratio of 0.92 would be possible if such a reactor were optimized for thorium utilization. Presumably, this conversion ratio could be achieved by providing a thorium blanket to reduce the neutron leakage, shortening the fuel life to reduce the neutron losses to fission products, using zirconium cladding to eliminate about 2/3 of the neutron loss to the structure and moderator, and shortening the fuel life to reduce the neutron losses to H_2O in the moderator.

The breeding ratio of the gas-graphite reactor could be improved somewhat by operating with a shorter fuel exposure, but as in the case of the homogeneous reactor, the quoted figures are for pure U^{233} fuel. Therefore, the average breeding ratio should be reduced about 0.02 to account for the buildup of U^{235} and U^{236} in the fuel over the life of the reactor.

(b) Fuel Utilization

The amounts of natural uranium that must be mined through the year 2030 to meet the U. S. requirements for nuclear power, as postulated in the 1962 AEC Report to the President[3], are shown in Figure 8. The figure was presented originally in Reference 4. A curve for D_2O reactors operating with thorium at a breeding ratio of 1.00 has been added to the original data. It is concluded from Figure 8 that:

(1) Through the early part of the next century, the use of D_2O-thorium reactors can substantially reduce the requirements for mined uranium, even below the best combinations of D_2O and H_2O converters and fast breeders.

(2) Even after fast breeders provide bred plutonium for their own needs, the use of thorium in D_2O reactors would prevent the D_2O reactors from being a continuing drain on uranium resources.

Figure 9 shows the effect of breeding ratio on the requirements for mined uranium to the year 2030 for a nuclear economy composed entirely of thorium-fueled D_2O reactors. This figure illustrates the following points:

(1) An increase in breeding ratio at the expense of a decrease in specific power may not improve the fuel resource picture because of the larger inventory requirements to start up new reactors. In particular, the uranium requirements for the reactor design described in this report would be essentially unchanged if a more economical reactor design were employed in which the specific power was changed from 35 to 47 MW/T and the breeding ratio reduced from 1.02 to 0.99.

(2) An increase in breeding ratio to 1.05, which corresponds to a doubling time of about 20 years, would reduce the requirements for mined uranium by more than a factor of two and would remove into the distant future the necessity for developing fast breeders. Since D_2O reactors can operate at a specific power per unit of fissionable material of four to five times that in a fast breeder, a breeding ratio of 1.05 in a D_2O reactor is worth a ratio of about 1.20 to 1.25 in a fast breeder in achieving equivalent doubling times.

(3) The uranium consumption in D_2O reactors with a breeding ratio of about 1.00 is modest enough that there will be little or no incentive to develop breeding ratios as high as 1.05 before the beginning of the next century.

(c) Fuel Economics

Presented in Table XI is a "high-spot" estimate of fuel cycle economics for the reactor described in the next section of this report. These data are predicated on arbitrarily assumed values for the breeding ratio, the costs of natural uranium and thorium, and the fuel exposure. The table illustrates the following:

(1) With present uranium costs, the inventory and burnup charges are so low that breeding is a relatively small economic factor in fuel cycle costs. Thorium fuel at 15,000 MWD/T is a standoff with uranium fuel, and the most attractive way to reduce near-term costs is to increase the exposure in the thorium fuel -- even at the expense of reduced conversion ratios. The columns headed "future costs" illustrate the situation if uranium were to cost $100/kg. Here the inventory charges are sufficiently high to make breeding economically attractive. However, longer fuel life still appears to offer additional savings.

(2) Thorium-fueled D_2O reactors can operate with reasonably attractive fuel cycle costs even with uranium priced as high as $100/kg, a price that should not be exceeded for many years if such reactors are used exclusively to meet the U. S. nuclear power demand.

3. Description of a Thorium-D_2O Reactor

For purposes of program formulation and further study, a conceptual design has been outlined for a 1000-MWe thorium-fueled D_2O-moderated power reactor. This "reference" design is basically the same as that of a D_2O reactor that is optimized for economical power generation on a uranium-plutonium fuel cycle. Some of the major design parameters for thorium and uranium-fueled reference reactors are presented in Table XII. A more detailed listing of representative design parameters for uranium-fueled reactors that were optimized for low energy-generation costs may be found in Reference 2. The envisaged reactor is a pressure tube unit that is cooled by liquid D_2O and is operated with an indirect turbine cycle. The fuel assemblies comprise coaxial tubes of metal or oxide clad in Zircaloy; uranium fuel elements of this type are under development now at the Savannah River Laboratory. Coolants other than liquid D_2O are feasible (e.g., liquid H_2O, boiling D_2O, or an organic liquid), and might ultimately prove to be better choices. Liquid D_2O is specified for the present because the technology of this coolant is further advanced than that of any of the other leading candidates.

The following modifications would be necessary to adapt the uranium-fueled design of Table XII to operation on a thorium-U^{233} cycle. All of these modifications are purely mechanical in nature and do not require the development of new technology.

(1) Thorium metal fuel enriched with 1.5% U^{233} is used instead of UO_2 enriched to 1.2% U^{235}. Thorium oxide could be used instead of metal at some sacrifice in breeding ratio.

(2) The number of fuel positions for a 1000-MWe reactor is increased from 517 to 692 to reduce the specific power and the neutron loss to Pa^{233}.

(3) The number of control positions is increased from 37 to 53 to provide the greater control requirements necessitated by the decay of Pa^{233} during a long shutdown. This decay would add about 10% k to the reactor.

(4) A ring of positions, similar in mechanical design to fuel positions, is placed in the radial D_2O reflector and filled with thorium elements containing no U^{233} initially.

(5) Short thorium elements, also containing no U^{233} initially, are placed at the top and bottom of the fuel column in the axial D_2O reflectors.

(6) Control rods sufficient to control about 2% k are made of thorium; cooling and radiation shielding are provided. These are the rods that would be used during steady-state reactor operation.

Rough estimates of construction and operating costs are shown in Table XIII for the 1000-MWe reference reactors described in Table XII. These estimates, which apply to present-day costs and financing charges that are representative of privately owned utilities, were developed with the aid of the cost optimization code described in Reference 8. The estimated cost of power is increased about 0.4 mill/kwh when the reactor is designed for operation with thorium. This increase is due almost entirely to the additional fuel positions that are needed to accommodate the lower specific power and better conversion ratio in the thorium-fueled case.

B. VOID COEFFICIENTS IN BOILING H_2O REACTORS

As part of an exploratory study of the use of boiling H_2O as coolant in a D_2O-moderated reactor, estimates were made of void coefficients of reactivity for this type of reactor. Fuel assembly dimensions were obtained by use of a computer code that was originally written for liquid-water-cooled fuel designs[9]; the code was used in this study to obtain approximate dimensions for boiling-water-cooled designs. The change in reactivity that results from the change in the void content of the coolant was computed with a code that was devised for calculation of the nuclear parameters of heavy water lattices[10].

The fuel assemblies considered in the study consisted of three concentric tubes of uranium oxide clad with Zircaloy-2 and housed in a Zircaloy pressure tube. The void coefficient of reactivity was computed as a function of the fuel mass, lattice pitch, and U^{235} enrichment. The calculated void coefficient is shown in Figure 10 as a function of these variables. The void coefficient is defined as the increase in k_∞ that results from a change in the H_2O coolant density from 0.4 to 0.3 g/cc. Figure 11 shows the value of k_∞ at a coolant density of 0.5 g/cc for the same range of fuel designs as in Figure 10.

These results will be used as a basis for choosing a reference design for economics evaluations. It is apparent from the calculations that the fuel assembly design for a boiling H_2O reactor must be significantly different from that of a boiling D_2O reactor if the positive void coefficient is to be held to a manageable value. The fuel assembly for the boiling D_2O reactor described in Reference 2 had a fuel weight of 25 lb/ft on a 10-in. lattice pitch. It is seen by extrapolation of Figure 10 that this fuel design with H_2O coolant would have a prohibitively high void coefficient. One of the objectives of further evaluations is to determine the magnitude of the economic penalty associated with design modifications required to obtain a near-zero void coefficient with boiling H_2O.

C. ENGINEERING FEASIBILITY OF LARGE D_2O REACTORS

The continuing trend toward installation of larger nuclear power stations in the U. S. and the consideration of very large nuclear reactors for process heat applications intensified interest in the feasibility of building single D_2O-moderated reactors as large as about 10,000 MW thermal capacity. With the objective of exploring the feasibility

question, the Engineering Department of the Du Pont Company has undertaken for the Atomic Energy Commission a study of the feasibility of designing and constructing 3500 MW(t) pressure-tube power reactors, moderated by D_2O and cooled either by liquid D_2O or by an organic liquid. The problems of designing and constructing similar reactors of 8300 MW(t) capacity are also being considered in the study. For both sizes, the study is concerned with over-all plant feasibility, with emphasis being placed on the reactor and its auxiliary systems. "Feasibility" of the plants is interpreted to mean that they can be designed and constructed as conceived and within the expected cost range, and that they will operate as intended.

The study thus far has been concerned only with liquid D_2O coolant. A design concept has been developed for a 3500 MW(t) reactor moderated by cold unpressurized D_2O and cooled by liquid D_2O at 1700 psia. The reactor is a vertical calandria containing 516 Zircaloy pressure tubes. Separate shell-and-tube shield structures are the main supporting members for the calandria and the pressure-tube assemblies. Coolant is supplied to and discharged from the pressure-tube assemblies through individual 3-in. pipes which connect with large ring headers above and below the reactor. Control and safety rod drives are mounted above the reactor, and on-power refueling is accomplished from below the reactor.

The design concept described above appears feasible, although many difficult problems exist. Problems which have been examined include the structural arrangement of the reactor and the arrangement of the large numbers of closely spaced coolant pipes; design of the pressure-tube assemblies and accessibility for their installation, replacement, and refueling; accommodation of differential movements between parts of the structure due to temperature differences; deformation of the shield structures under their great loads; and leak-tightness of the pressurized D_2O system.

No reasons have developed to doubt the feasibility of designing and constructing a 3500 MW(t) plant based on this reactor concept; however, the problems of the portions of the plant outside the reactor have not yet been examined in detail.

The problems of design and construction of an 8300 MW(t) plant are basically the same as those for the 3500 MW(t) plant, but in some cases are aggravated to an extent that they may require quite different solutions. Some of the major problems foreseen include the very large sizes and loadings of reactor components, especially the axial shields, the design and

installation of the individual piping to pressure tubes, the large sizes of pumps and steam generators, the method of achieving adequate containment, and the turbine-generator plant.

IV. EXPERIMENTAL PHYSICS

A. BUCKLINGS AND VOID EFFECTS

One-region bucklings and void effects were measured for D_2O-moderated lattices of nested uranium metal tubes in the Process Development Pile (PDP) of the Savannah River Laboratory. The purposes of these measurements were (a) to provide normalization points for improving buckling calculations on lattices of heavy, tubular fuel assemblies, and (b) to extend a general experimental study of uranium-D_2O systems that has previously included measurements of lattices of 1-in.-diameter metal rod clusters[11] and 0.5-in.-diameter UO_2 rod clusters[12].

1. Description of the Fuel Assemblies

The two types of tubular fuel assemblies used in the present studies are shown in Figure 12. The first, the "ABC" assembly, consisted of a natural uranium rod surrounded by two concentric natural uranium tubes. The assembly was enclosed in a housing tube of 6063 aluminum. The second, the "C" assembly, consisted of a single natural uranium tube in a housing tube of 6063 aluminum. The housing tubes were provided with plugs so that D_2O could be admitted to or excluded from the fuel assembly "coolant" passages. The uranium tubes were protected with a coating of "Kel-F"* that was about 0.007-in. thick. The effective fuel length of the assemblies was 12 ft.

2. Lattices

Both types of fuel assemblies were arranged in D_2O-moderated lattices in the PDP at three triangular lattice pitches: 9.33-in., 12.12-in., and 14.00-in. center-to-center. In each of these six lattices, the radial and vertical bucklings were determined separately with D_2O and with air in the "coolant" passages. In addition, period-reactivity measurements were taken in the air-filled and D_2O-filled "C" tube

* Kel-F 800 resin. Trademark of Minnesota Mining and Manufacturing Co. for a copolymer of chlorotrifluoroethylene and vinylidene fluoride.

lattices at the 9.33-in. pitch. Eighty-five fuel assemblies were used in each lattice. Since this number of assemblies did not completely fill the PDP tank, the effective reactor diameter was reduced by poisoning the outer region of the tank with lithium-aluminum rods. The poison rod arrangements were tailored to each lattice to produce a cylindrical, unreflected pile. The moderator purity in all lattices was 99.59 ±0.02 mol % D_2O. The nominal temperature in each measurement was 22°C.

3. Buckling Measurements

Geometrical bucklings were obtained by a previously validated procedure[11] in which three critical water height measurements (runs) are made with each lattice. In the first run, four vertical and three radial flux profiles were obtained by irradiating gold pins suspended throughout the lattice. The gold pin count-rate data were then fitted by a least-squares method to cosine and J_o functions to obtain the vertical and radial bucklings. In the second run, the critical moderator height was measured with the gold pins removed from the lattice. Finally, in the third run, extra aluminum guide tubes were inserted in the lattice, and the moderator height was again measured to determine the worth of the control and safety rod guide tubes that were always present in the lattice. The differences between critical moderator heights in the three runs were measured with a precision of ±0.01 cm. When combined with the calculated corrections described below, the results from the three runs yielded the vertical and radial bucklings of the "clean" lattice.

The results of the buckling measurements are presented in Table XIV and in Figures 13 and 14. Also included for comparison are calculated material bucklings that were obtained by use of the July 1963 version of the Savannah River BSQ Code[10]. This code, which is routinely used to compute lattice bucklings, includes recent modifications to make allowances for the increase in neutron leakage due to streaming in fuel assemblies containing voided coolant passages. The agreement between calculated and measured bucklings is fair for all lattices except the two 9-in. lattices with D_2O removed from the coolant passages. The material bucklings in Table XIV are the sums of the corresponding vertical and radial bucklings. For the lattices in which the fuel assemblies contain air, anisotropy is appreciable and this sum is dependent upon the reactor geometry.

The calculated corrections to the measured bucklings totaled 11 to 18 μB; these corrections accounted for: (a) uneven distribution of gold pins in the flux mapping run, about 1 μB, (b) different relative locations of the control and safety rod guide tubes, about 0.5 μB, and (c) the poisoning effect of the "Kel-F" coating on the uranium tubes, 10 to 18 μB. A measured correction was applied to normalize all measurements to 22°C.

The uncertainties listed with the results in Table XIV take into account the total range in bucklings encountered in the several traverses of each flux mapping run. The errors listed for the vertical bucklings also include about ±3 μB error due to uncertainties in the moderator temperature and the D_2O purity plus 3 to 8 μB for the uncertainty in the thickness of the "Kel-F" coating on the uranium tubes.

4. Period-Reactivity Measurements

Period-reactivity measurements were made in the 9.33-in. lattices with single-tube fuel assemblies to determine the migration areas for neutrons in these lattices. Positive periods corresponding to changes in vertical buckling were obtained by making small increases in the moderator height of the critical reactor. The neutron flux increase was measured as a function of time on five compensated ion chambers and three BF_3 counters. A calculated period-reactivity relationship was then used to obtain values of Δk_{eff}. Finally, values of $(L^2+\tau)$ were calculated from the values of $(-\Delta k_{eff}/\Delta B_z^2)$ by means of two-group theory. This procedure has been described in detail in previous work.[12]

The period measurements were performed six times in the lattice of air-filled assemblies and eight times in the lattice of D_2O-filled assemblies; various increases in moderator height were investigated. The results of the measurements are shown in Table XV. The errors listed with the values of $(-\Delta k_{eff}/\Delta B_z^2)$ include an uncertainty of about 1 second in measuring the period and an uncertainty of ±0.02 cm in measuring the difference in moderator height between the critical and supercritical conditions.

The results indicate that the experimental value of $(-\Delta k_{eff}/\Delta B_z^2)$ is dependent upon the reactor period at which it is obtained. The effect is shown for the air-filled assemblies in Figure 15. This variation is probably due to errors in the calculated period-reactivity relationship. However, such systematic errors will be essentially canceled in a ratio between the values of $(L^2+\tau)$ observed in two lattices.

TABLE I

Operating Chronology of HWCTR
July-August 1963

July 1-13	Discharged OT-1-3 housing tube Replaced fuel top fitting and top fuel slug of SOT assemblies with dummy section Discharged OT-1-5 and OT-1-2, replaced by new test fuel
July 13	Attained criticality Shut down during temperature ascension because of high O_2 concentration
July 16-17	Attained criticality Shut down during power ascension because of increase in gas activity monitor reading
July 17-24	Attained criticality, raised power to 38 MW
July 24	Shut down to investigate sources of D_2O leakage
July 24-27	Attained criticality, raised power to 38 MW
July 27-31	Level operation at 38 MW
August 1-5	Level operation at 38 MW
August 5	Shut down to investigate sources of D_2O leakage
August 5-8	Made leak repairs
August 8-12	Operated at 18 MW
August 12	Received spurious scram from continuous temperature monitor
August 12-16	Operated at powers up to 38 MW
August 16	Shut down to repair leak in fuel failure instrument
August 16-18	Repaired leak
August 18-25	Operated at powers up to 39 MW
August 25	Shut down for suspected fuel failure
August 25-31	Made preparations for discharge of failed Boiling Loop Bayonet

TABLE II

Operating Summary of HWCTR
July-August 1963

	July		August	
Time reactor critical, %	47.7		63.5	
Maximum power, MW	38		39	
Reactor exposure, MWD	Driver	Test	Driver	Test
Month	393	98	504	129
Total accumulated since startup	4520	1014	5024	1143
D_2O losses (100 mol %), lb	2741[a]		1020	

(a) The loss for July was 2229 lb. An additional 512 lb is included to account for a purity analysis error made in June.

TABLE III

Test Fuel Irradiation Data
July-August 1963[a]

Reactor power	39 MW
Coolant pressure	1200 psig
Coolant inlet temperature	237.5°C
Moderator outlet temperature	250°C

Position	Element Number	Fuel Type	Fuel Assembly Description	Starting Date	Maximum Nominal Conditions: Assembly Power[b], MW	Specific Power[c], MW/ton U	Heat Flux, pcu/(hr)(ft²)	Outlet Temp., °C	Surface Temp., °C	Core-Clad Temp., °C	Core Temp., °C	∫kdt, watts/cm	Maximum Exposure, MWD/ton U[c]: Attained	Planned
37	CANDU	Oxide rod	Five 19-rod bundles of natural UO_2 pellets	10-5-62	0.94	22.4	121,000	265	271	290	-	26.1	3502	9,000
38[d]	OT-3-2	Oxide tube	Single tube of natural UO_2, vibratorily compacted	5-19-63	0.42	36.9	174,000	258	276	328	-	13.6	1568	(d)
39	2ETWO-2	Metal tube	Unalloyed, 2.1% enriched, 2" x OD single tube	5-19-63	1.13	81.5	475,000	253	306[f]	428	497	-	3423	8,000
40	SOT-1-2	Oxide tube	Seven 14" long tubes of 1.5% enriched UO_2[e]	10-5-62	0.55	59.0	292,000	246	293	378	-	23.2	9050	23,000
42	2ETWO-3	Metal tube	Same as position 39	6-11-63	1.14	82.3	477,000	253	306[f]	429	498	-	2793	10,000
55	SOT-5-2	Oxide tube	Seven 14" tubes of natural UO_2, 5 swaged, 2 vibratorily compacted[e]	5-19-63	0.61	38.0	397,000	251	306[f]	433	-	55.6	1477	6,000
56	SOT-1-3	Oxide tube	Same as position 40[e]	1-6-63	0.52	55.7	273,000	246	291	370	-	21.8	7168	10,000
57	SMT-1-2	Metal tube	Ten 11-1/4" alloyed (Fe,Al) natural uranium tubes	5-19-63	0.55	32.4	277,000	251	306[f]	369	426	-	1292	7,500
58	SOT-1-4	Oxide tube	Same as position 40[e]	7-13-63	0.69	74.2	364,000	248	306[f]	411	-	29.0	2014	16,000
59	OT-1-7	Oxide tube	Single 1.5% enriched UO_2 vibratorily compacted swaged tube	7-13-63	0.83	67.5	347,000	251	306[f]	410	-	27.6	1816	23,000
60	SMT-1-3	Metal tube	Five 11-1/4" alloyed (Fe,Al,Si) natural uranium tubes	7-13-63	0.44	34.4	294,000	249	306[f]	373	433	-	978	7,500

(a) Data taken on August 24, 1963; exposures as of August 31, 1963.
(b) "Flow-ΔT" power calculation; does not include gamma heating of moderator.
(c) These values are based on an assembly power of 1.09 times "Flow-ΔT" power to include gamma heating of the moderator.
(d) Liquid loop position. The OT-3-2 fuel element is being used to commission the loop and is not an irradiation test of fuel.
(e) Originally contained eight tubes. Top tube replaced with dummy fuel piece after vibration damage observed on top pieces.
(f) Temperature limited by surface boiling.

TABLE IV

Vibration Damage to Tubular Oxide Assemblies Irradiated in the HWCTR

Assembly No.[a]	Fuel Tube Length	Max. $\int$kdt, watts/cm	Maximum Exposure, MWD/T	Damage to Fuel Tubes	Damage to Gripper Fitting (other than extensions of vibration grooves on fuel)
OT-1-2	9 ft	21	7000	Two grooves 90° apart, extending 12-15 in. from top of tube, 10-14 mils deep. Two other rib marks of negligible depth.	Negligible
OT-1-4	9 ft	22	4000	As in OT-1-2; two grooves 6 in. long, 7-13 mils deep.	Negligible
OT-1-5	9 ft	22	5700	As in OT-1-2; two grooves 26 in. long, 10-22 mils deep.	Negligible
SOT-1-2	14 in.	22	7400	Top slug damaged; worst groove 6 in. long, 10 mils deep.	Negligible
SOT-1-3	14 in.	21	5700	Top slug damaged; 4 mils maximum depth of wear.	Negligible
SOT-2-2[b]	14 in.	49	3200	Significant vibration wear on top slug only; grooves 2 mils deep. Slight vibration wear on mating ends of top slug and gripper fitting.	Three of four bolts broken off; several mils wear due to contact with housing tube.
SOT-2-3[b]	14 in.	49	2900	As in SOT-2-2	All bolts broken off; other wear as in SOT-2-2.
SOT-5-2	14 in.	46	500	Superficial wear marks on top slug.	None

(a) Assemblies OT-1-3 and OT-1-6, which failed during irradiation, have not yet been inspected.
(b) Failed during irradiation due to buckling of the cladding.

TABLE V

Fabrication Information and Irradiation Conditions for Uranium Oxide Tubes

Position in Assembly	Tube No.	Method of Compaction[a]	Collapse	Core Density, % of TD	∫kdt[b], watts/cm	Burnup[b], MWD/Tonne U
Assembly SOT-2-2						
1	257C	V+S	No	89	3	200
2	250B	V	No	83	11	800
3	256A	V+S	No	90	25	1800
4	251A	V	Yes	87	42	2900
5	251C	V	Yes	82	49	3500
6	255B	V+S	No	90	47	3300
7	257B	V+S	Incipient[c]	89	34	2400
8	252C	V	No	82	16	1100
Assembly SOT-2-3						
1	254A	V+S	No	90	3	200
2	251B	V	No	83	10	700
3	256B	V+S	No	90	23	1500
4	252B	V	Yes	83	39	2600
5	254B	V+S	No	89	48	3200
6	253C	V	Yes	85	45	3000
7	257A	V+S	No	89	32	2100
8	253B	V	No	85	15	1000

(a) V signifies vibratory compaction only; V+S signifies vibratory compaction and subsequent swaging.

(b) ∫kdt and burnup estimates are based on HWCTR operating data and were averaged over the length of the element. The ∫kdt ratings correspond to the period of highest power generation.

(c) This element suffered incipient collapse at its "cold" end only; most of the element was unaffected.

TABLE VI

Diametral and Volume Changes in Collapsed Uranium Oxide Tubes

Tube No.	Measured Change OD, inch		Change in Volume, %	Calculated Core Density, % of theoretical	
	Mean[a]	Ridge		Preirradiation	Postirradiation
251A	-0.014	+0.048	-2.4	87	89
251C	-0.054	+0.105	-8.7	82	89
252B	-0.044	+0.080	-4.5	83	87
253C	-0.042	+0.089	-5.4[b]	85	88

(a) The mean change in OD does not include the ridge and adjacent area.
(b) Estimated from OD and ID measurements.

TABLE VII

Postirradiation Gas Content of Uranium Oxide Tubes

Results of measurements on SOT-2 tubes

Tube No.	Gas Release (Major Constituents Only), cc/kg UO_2						Volume of He + A, cc at STP	Initial Free Volume, cc	Approx. Pressure Inside Tube at 20°C, atm.[b]	Xenon Release, % of that produced by fission
	Total	Xe	N_2	D_2	He	A				
Assembly SOT-2-2										
257C	23.5	0.05	0.71	ND[a]	23.3	0.13	86	96	1.0	1.4
251A	69.2	30.8	5.6	ND	27.5	1.3	111	107	2.4	57
251C	Gas lost via sheath cracks									
255B	69.2	38.6	0.24	ND	25.6	0.14	94	92	2.4	62
257B	40.2	14.2	5.0	ND	15.5	0.19	56	96	2.6	33
Assembly SOT-2-3										
254A	30.4	0.06	6.0	ND	24	0.3	86	92	1.2	1.8
251B	30.3	0.28	1.5	ND	27	1.3	104	122	1.1	2.2
252B	76.4	29	11	ND	32	0.2	117	118	2.4	59
254B	302	27	23	218	30	0.2	112	91	10.0	46
253C	Gas lost via sheath cracks									
253B	26.6	0.34	5.9	ND	16	2.5	71	117	1.4	1.8

(a) "ND" signifies none detected.
(b) Calculated from measured composition of released gas, with the assumption that all helium and argon present resulted from pressurization of the tubes to one atmosphere of helium during fabrication.

TABLE VIII

Dimensional Changes in Irradiated Tubes of Uranium Metal

Fuel assembly designation	TWNT-13[c]		TWO-1-2[d]
HWCTR position	60		59
Maximum exposure, MWD/T	370		1105
Exposure to full reactor flow, days[a]	14		58
Exposure to shutdown flow, days[b]	15		39
	Outer Fuel Tube	Inner Fuel Tube	
Fuel tube length decrease, in.	7/64	1/8	3/16
Fuel tube diameter change, average, mils	-1	-1	-4
Twisted-ribbon slack, in.[e]	5/8	3/4	3/4
Twisted-ribbon length increase (calculated), in.	3/8	9/16	3/8

(a) flow = 100 gpm/assembly, 10 ft/sec velocity.
(b) flow = 30 gpm/assembly, 3 ft/sec velocity.
(c) Maximum separation between ribbon and fuel tube, when slack is confined to one pitch length.
(d) Nested assembly of two thin-walled fuel tubes, with twisted-ribbon spacers attached to the outer surface of each.
(e) Assembly containing a thin-walled "outer" fuel tube and an inner housing tube, with twisted-ribbon spacers attached to each.

TABLE IX

Neutron Balance in D_2O-Thorium Reactor

	Range	Reactor Design of Table XII
Specific Power, MW/kg U^{233}	1 - 4	2.60
Exposure, MWD/Tonne of Thorium	5,000 - 30,000	10,000
η	2.29 - 2.10	2.260
$\eta\epsilon$	-	2.278
Neutron losses:		
Leakage	0.01 - 0.05	0.016
Xe and Sm	0.050 - 0.056	0.054
Other fission products	0.03 - 0.06	0.035
Structure and moderator	0.05 - 0.10	0.072
Pa^{233}	0.015 - 0.05	0.030
Total neutron losses	0.16 - 0.29	0.207
Pa^{233} + U^{235} production	1.11 - 0.98	1.071
Pa^{233} lost by neutron absorption	0.015 - 0.05	0.030
Net breeding ratio	1.09 - 0.93	1.041
30-year average breeding ratio in the reactor	1.07 - 0.91	1.021

TABLE X

Neutron Balances for Thorium-Fueled Reactors

	Reference D_2O of This Report	Homogeneous D_2O[5]	Spectral Shift[6]	Gas and Graphite[6]
Specific power, MW/kg U^{233}	2.6	1.0	1.0	1.5
Fuel exposure, MWD/T of thorium	10,000	-	20,000	20,000
η	2.260	2.274	-	-
$\eta\epsilon$	2.278	2.274	2.2215	2.188
Neutron losses:				
Leakage	0.016	0.020	0.0756	0.030
Xe and Sm	0.054	0.052	0.0487	0.040
Other fission products	0.035	-	0.1046	0.093
Structure and moderator	0.072	0.110	0.0503	0.033
H_2O	-	-	0.0860	-
Pa^{233}(a)	0.060	0.020	0.0220	0.036
U^{236}	0.020	-	0.0320	-
Total losses	0.257	0.202	0.4192	0.232
Average breeding ratio	1.021	1.072	0.8023 (0.92)(b)	0.956

(a) Twice actual absorptions in Pa^{233} to account for loss of potential U^{233}.

(b) Edlund and Schutt[7] estimate that a conversion ratio of 0.92 can be achieved in a spectral shift reactor designed for maximum thorium utilization.

TABLE XI

Estimated Fuel Cycle Costs for D_2O Reactors

	Current Costs				Future Costs		
Fuel	UO_2	Th	Th	Th	Th	Th	Th
Equivalent breeding ratio[b]	-	1.00	1.05	1.00	1.00	1.05	1.05
Uranium cost, $/kg U	13	13	13	13	100	100	100
Thorium cost, $/kg U	-	20	20	20	20	20	20
Exposure, 10^3 MWD/Tonne	14	15	15	30	15	15	30
Costs, $/kg of fuel							
Fabrication	22	30	30	30	30	30	30
Shipping	2	5	5	5	5	5	5
Reprocessing	-	35	35	35	10	10	10
Burnup or credit	49	0	-3.6[b]	0	0	-27[b]	-27[b]
Subtotal, $/kg	73	70	66	70	45	18	18
mills/kwh[a]	0.82	0.72	0.68	0.36	0.46	0.18	0.09
Uranium inventory charge, mills/kwh[c][d]	0.05	0.15	0.15	0.15	1.12	1.12	1.12
Thorium inventory charge, mills/kwh[c]	-	0.05	0.05	0.05	0.05	0.05	0.05
Total, mills/kwh	0.87	0.92	0.88	0.56	1.63	1.35	1.26

(a) At net thermal efficiency of 27.2% and load factor of 0.8
(b) Includes allowance for reprocessing and refabrication losses; "equivalent breeding ratio" equals breeding ratio in reactor minus 0.01.
(c) At a charge rate of 10% yr; total inventory assumed 50% greater than reactor inventory.
(d) Assumes that U^{233} is worth only the cost of the natural uranium required to produce it. It is assumed that 3 kg of U^{233} can be obtained from one tonne of mined uranium.

TABLE XII

Description of 1000-MWe Preliminary Designs of D_2O-Moderated Power Reactors

Fuel	Th Metal	UO_2
Initial enrichment, wt %	1.5 (U^{233})	1.2 (U^{235})
Reactor coolant	Liquid D_2O	Liquid D_2O
Lattice pitch, in.	10	10
No. of fuel positions	692	517
No. of control positions	53	37
Core radius, ft	12.7	11.1
Core length, ft	15	15
Coolant inlet temp., °C	264	264
Coolant outlet temp., °C	304	304
Maximum heat flux, pcu/(hr)(ft^2)	350,000	470,000
No. of coolant loops	8	8
Throttle steam pressure (saturated), psia	650	650
Net plant efficiency, %	27	28
Net electrical production, MW	1019	1019
D_2O inventory, tons	910	760
Thorium or uranium inventory, tonnes	105	62
U^{233} or U^{235} inventory, kg	1580	735

TABLE XIII

Estimated Costs of 1000-MWe D_2O Reactors

Basis: (1) reactor designs of Table XII
(2) fuel cycle costs as per Table XI
(3) load factor of 80%
(4) capital charge rate of 14.5%/yr (13.0%/yr for D_2O)

Fuel Cycle	Uranium		Thorium	
	10^6\$	mills/kwh	10^6\$	mills/kwh
Construction costs:				
Reactor and primary system	33.0	0.67	36.3	0.73
Reactor bldg. and auxiliaries	18.5	0.38	25.0	0.51
Turbine plant	51.4	1.04	51.4	1.04
Indirect costs @ 0.48	49.4	1.00	54.0	1.10
Total construction cost	152.3	3.09	166.7	3.38
D_2O @ \$20/lb	30.4	0.56	36.5	0.66
Operating costs		0.44		0.45
Fuel cycle cost (from Table XI)		0.87		0.92
Total generating cost		5.0		5.4

TABLE XIV

Bucklings of Tubular Assemblies of Natural Uranium in D_2O

See Fig. 12 for description of fuel assemblies

Fuel Assembly Designation	Lattice Pitch, inch	Fuel Assembly Contains	Measured Bucklings, μB[(a)]			Calculated Bucklings, μB
			B_z^2	B_r^2	B_m^2	λ_m^2
"ABC"	9.33	Air	212 ±6	378 ±5	590 ±8	543
		D_2O	178 ±8	375 ±3	553 ±9	531
"ABC"	12.12	Air	386 ±6	213 ±3	599 ±7	593
		D_2O	330 ±7	214 ±2	544 ±7	539
"ABC"	14.00	Air	341 ±6	164 ±1	505 ±6	500
		D_2O	282 ±6	166 ±2	448 ±6	437
"C"	9.33	Air	195 ±5	375 ±1	570 ±5	529
		D_2O	286 ±8	385 ±3	671 ±8	655
"C"	12.12	Air	312 ±5	212 ±3	524 ±6	515
		D_2O	341 ±6	217 ±4	558 ±7	542
"C"	14.00	Air	270 ±4	168 ±2	438 ±5	426
		D_2O	269 ±5	171 ±2	440 ±5	420

(a) 1 $\mu B = 10^{-6}$ cm^{-2}

TABLE XV

Experimental and Calculated Values of L^2 Plus τ for Lattices of Single Uranium Tubes in D_2O

Change in Moderator Height, cm	ΔB_z^2, μB[a]	Power Doubling Time, sec	Experimental Results $\Delta k_{eff}/\Delta B_z^2$, cm^2[b]	Experimental Results $L_z^2+\tau_z$, cm^2
Air-Filled Assemblies				
1.87	2.94	52	316 ±5	
1.63	2.56	61	325 ±5	
1.46	2.30	70	324 ±5	
1.32	2.08	80	328 ±6	
1.08	1.70	101	334 ±7	
1.03	1.62	109	330 ±7	
			Avg. 326 ±2.5	357 ±2.5
D_2O-Filled Assemblies				
1.46	4.03	52	230 ±4	
1.44	3.98	52	232 ±4	
1.23	3.40	64	235 ±4	
1.13	3.13	74	230 ±4	
1.12	3.08	74	233 ±4	
0.97	2.69	87	238 ±5	
0.97	2.68	87	238 ±5	
0.78	2.15	113	241 ±6	
			Avg. 235 ±2.3	256 ±2.3

(a) 1 $\mu B = 10^{-6}$ cm^{-2}

(b) Assigned errors are based upon known random errors only and do not include possible systematic errors in the experimental method or the period-reactivity relationship.

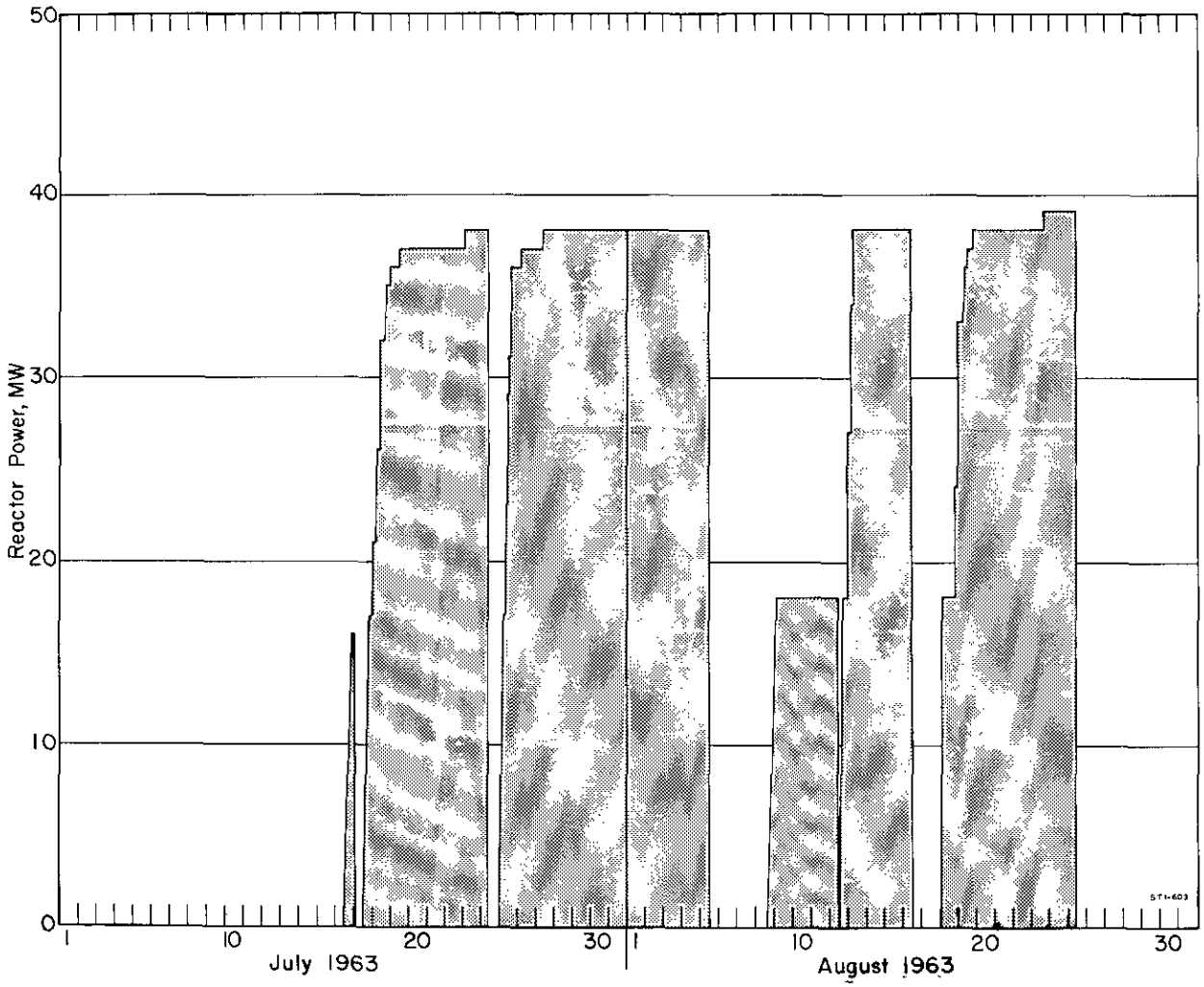


FIG. 1 OPERATING POWER OF HWCTR
July-August 1963

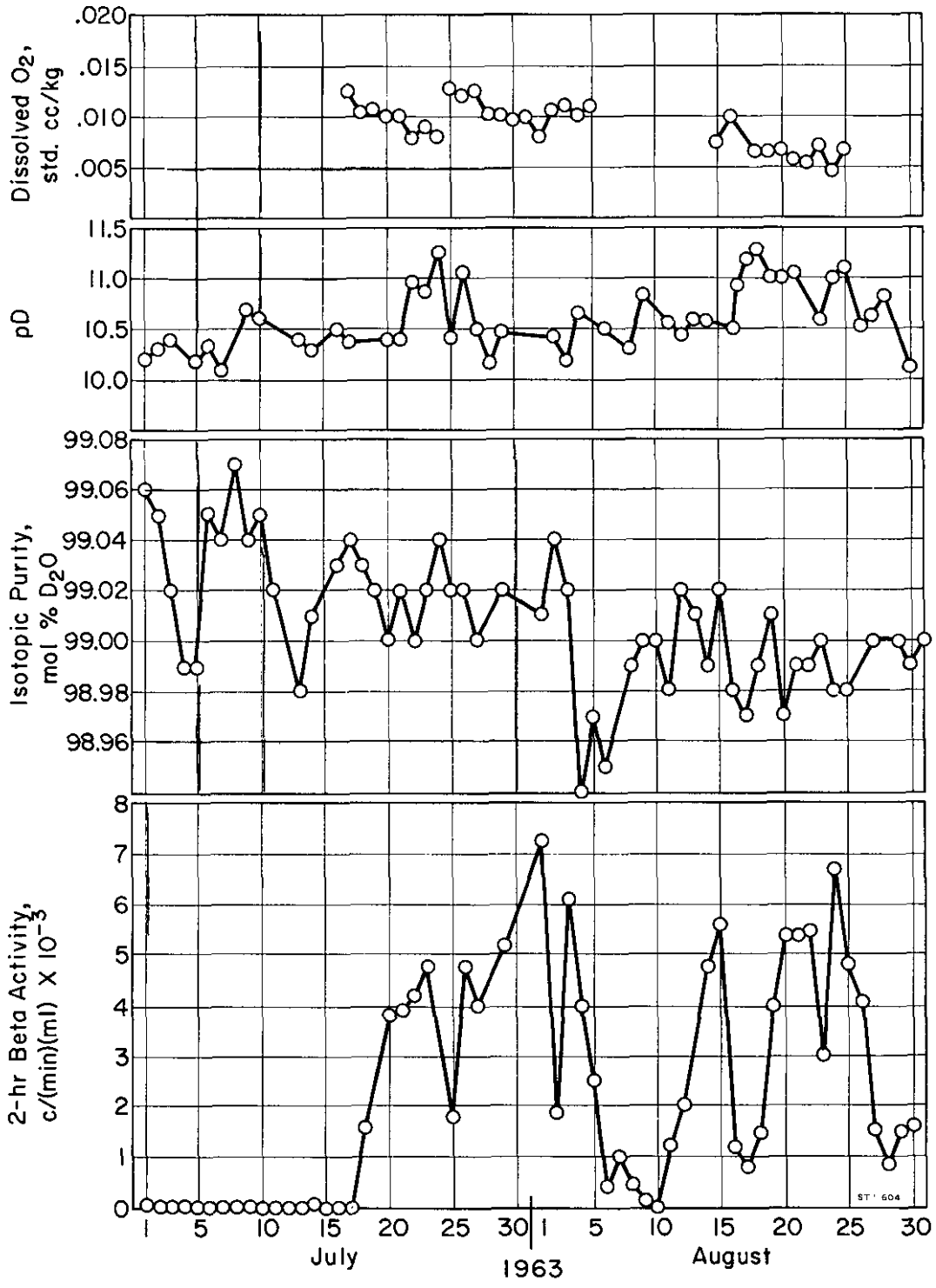


FIG. 2 HEAVY WATER QUALITY IN HWCTR
July - August 1963

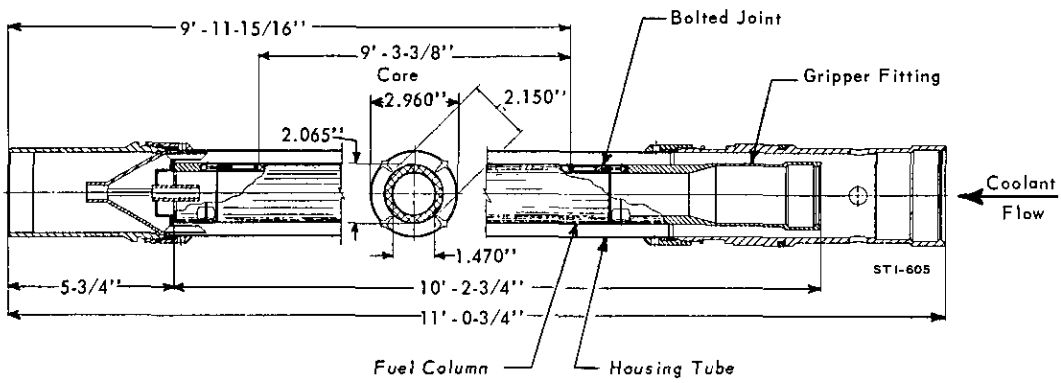
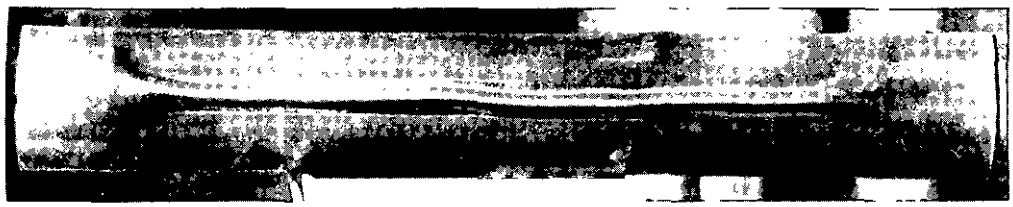
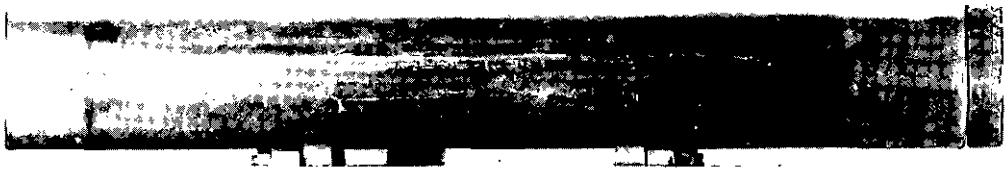


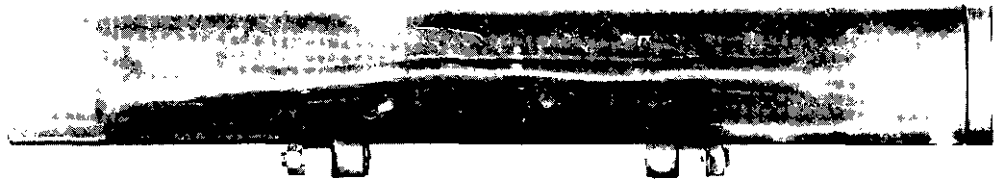
FIG. 3 TYPICAL TUBULAR UO_2 IRRADIATION ASSEMBLY FOR HWCTR TESTS



Neg. 52155 Tube 251C from SOT-2-2



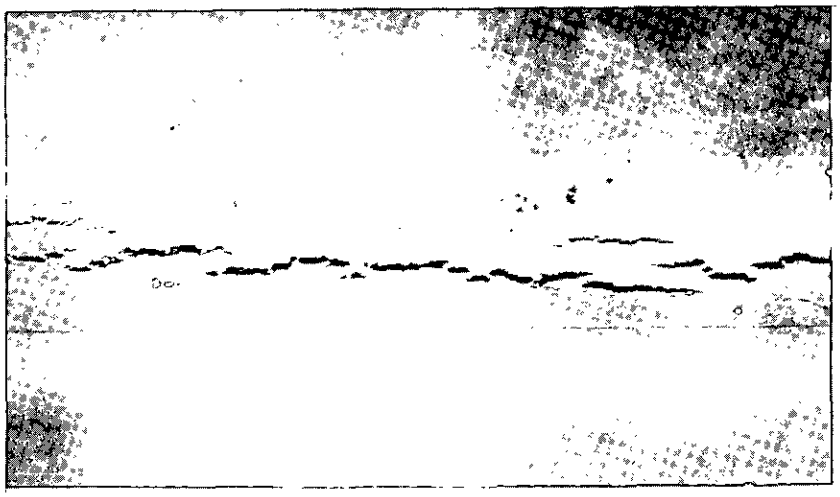
Neg. 52151 Tube 251A from SOT-2-2



Neg. 52070 Tube 252B from SOT-2-3

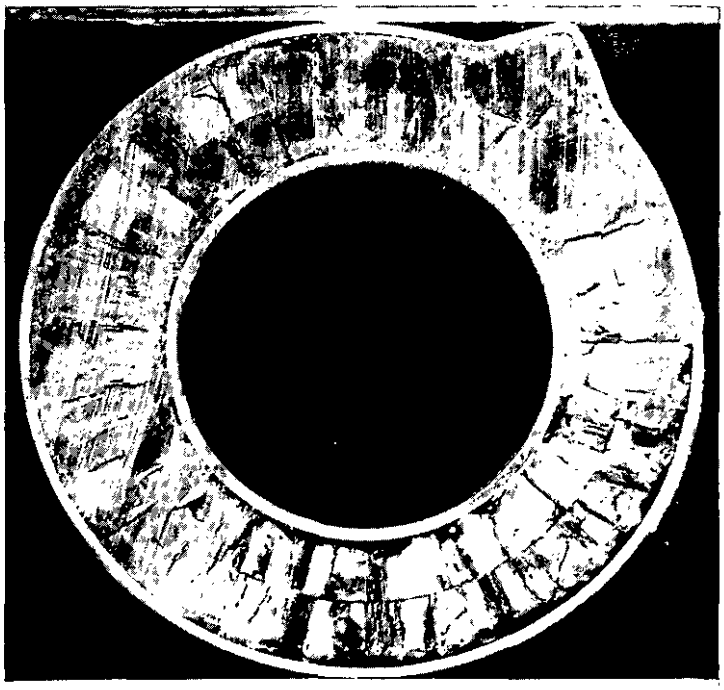
Neg. 52108 Tube 253C from SOT-2-3

FIG. 4 COLLAPSED URANIUM OXIDE TUBES 0.43X
The Zircaloy Sheaths in Tubes 251C and 253C cracked at the apex of the ridge



Neg. 52123

FIG. 5 SHEATH CRACKS IN IRRADIATED URANIUM OXIDE TUBE 20X
Tube 253 C



Neg. 52145 2X

Neg. 52042 4X

FIG. 6 CROSS SECTION OF COLLAPSED URANIUM OXIDE TUBE
Note extent of columnar grain growth and the migration of UO_2 into the ridge.

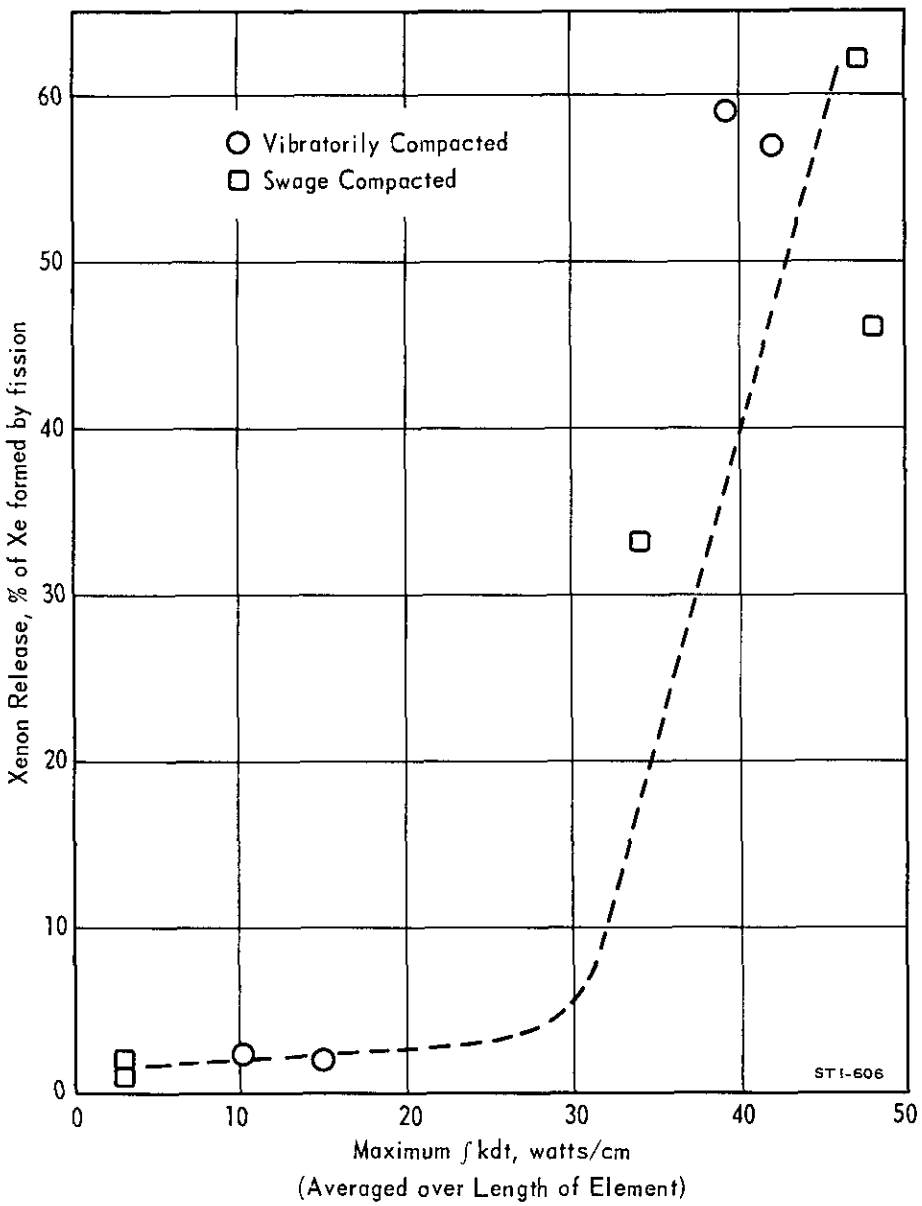


FIG. 7 XENON RELEASE IN URANIUM OXIDE TUBES
(Results of Measurements on SOT-2 Tubes)

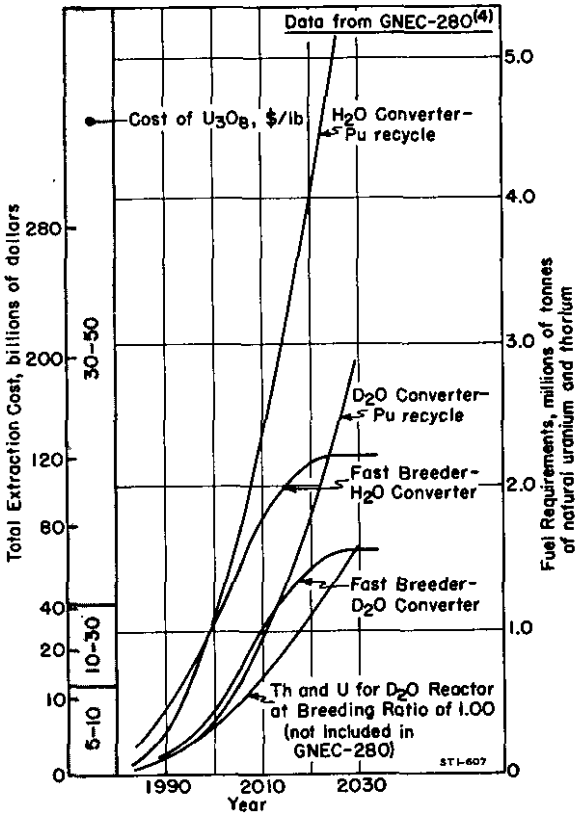


FIG. 8 FUEL REQUIREMENTS FOR PREDICTED U. S. NUCLEAR POWER DEMAND

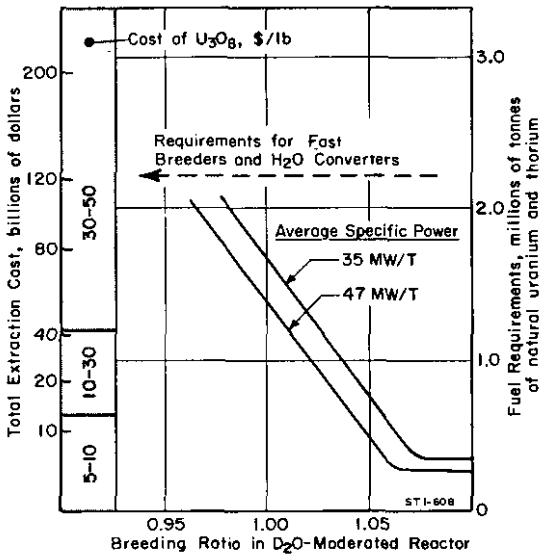


FIG. 9 EFFECT OF BREEDING RATIO ON FUEL REQUIREMENTS TO YEAR 2030

Basis: (1) Average fuel exposure of 10,000 MWD/T
(2) *Reprocessing and refabrication losses of 0.8% per cycle*
(3) Nuclear power growth as per Reference 3
(4) A U^{233} yield of 3 kg per tonne of natural uranium

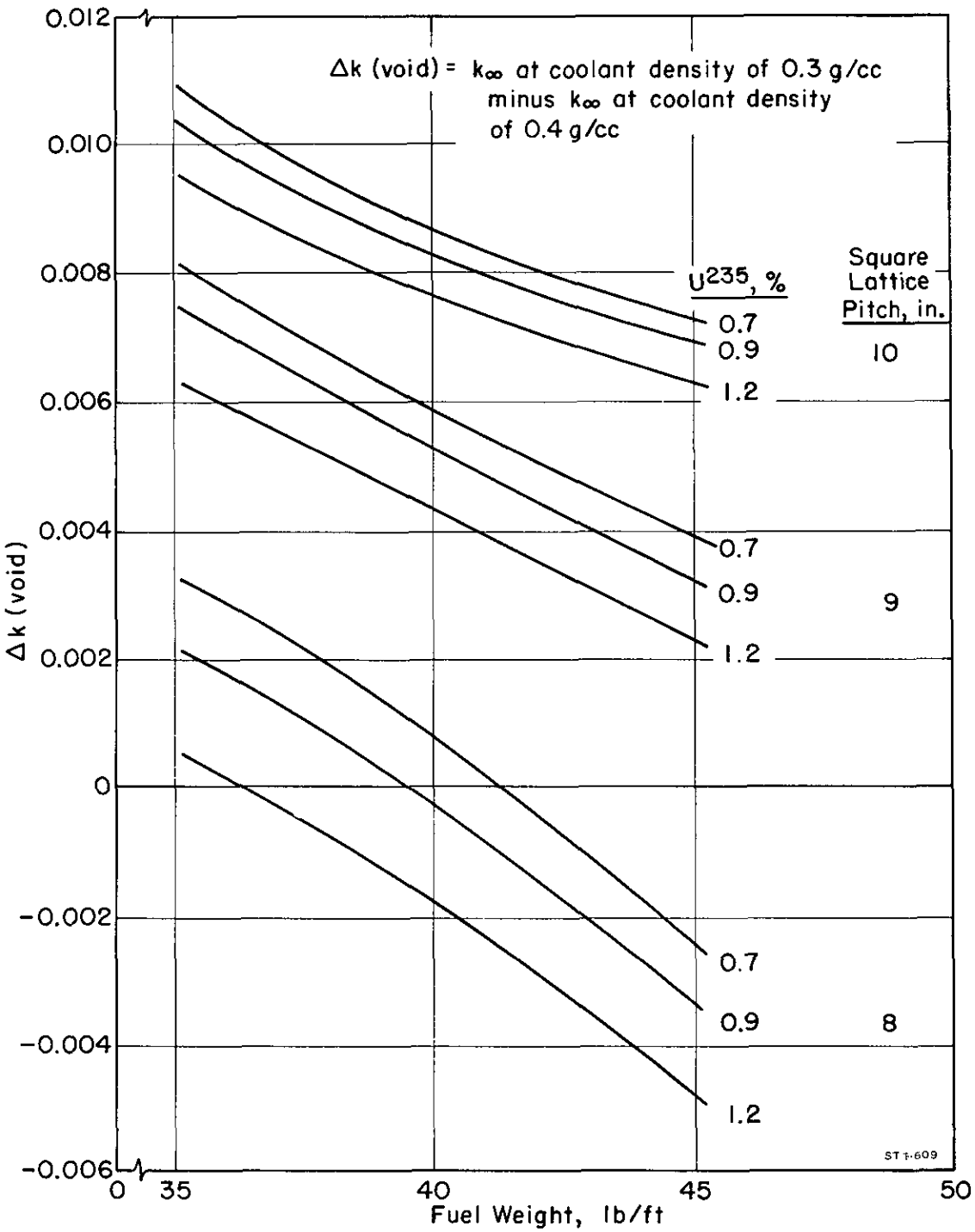


FIG. 10 VOID COEFFICIENTS OF REACTIVITY IN BOILING H_2O REACTORS

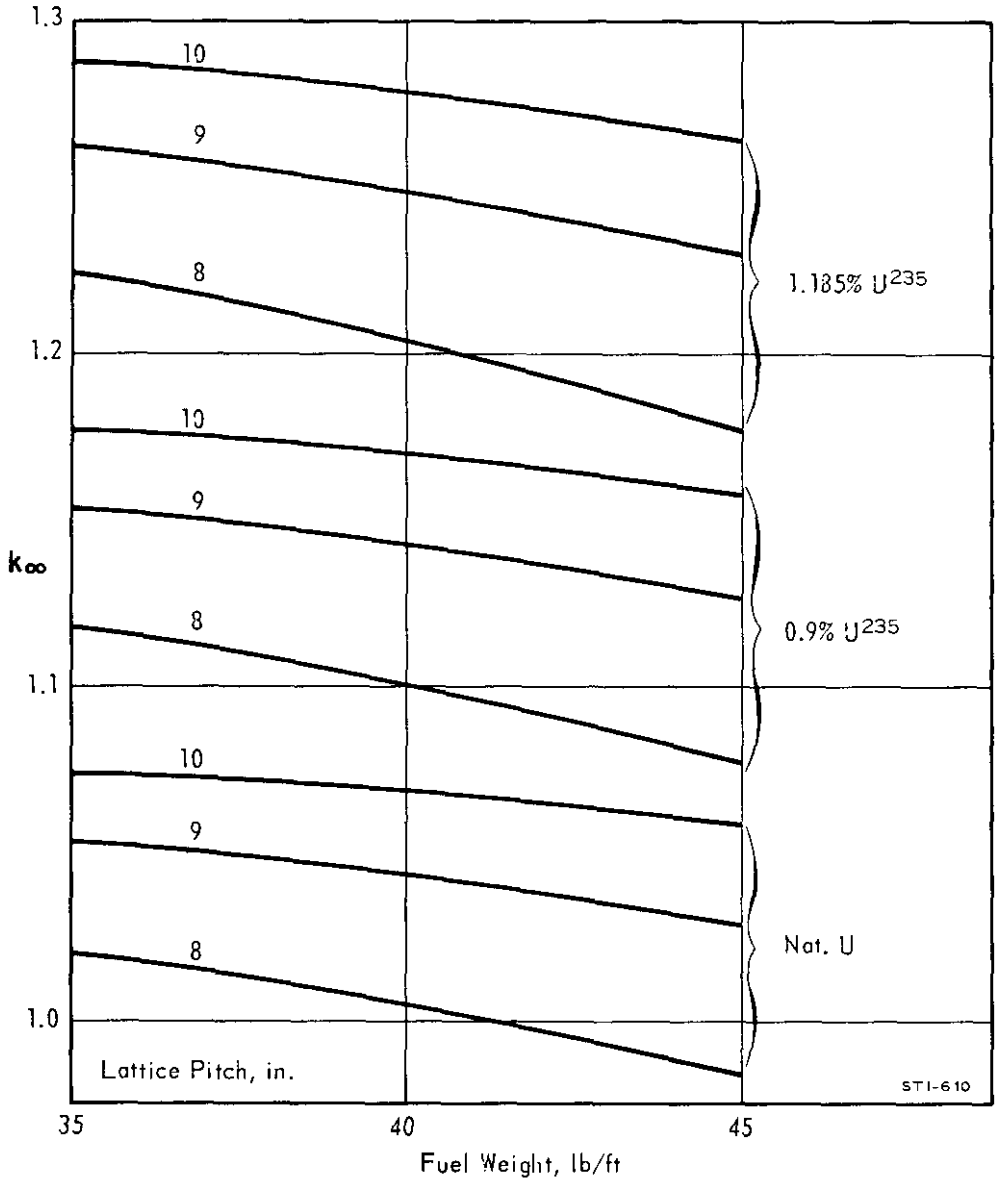
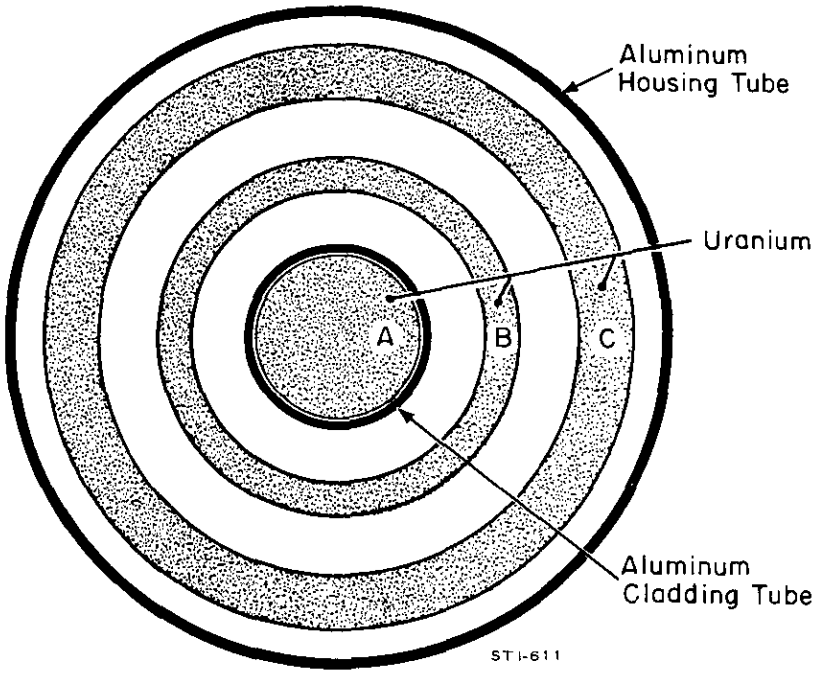


FIG. 11 REACTIVITIES OF FUEL DESIGNS FOR BOILING H_2O REACTORS
H_2O Coolant Density = 0.5 g/cc



	Component	Material	OD, inches	Wall Thickness, inch
"ABC" Assembly	Fuel rod	Uranium	1.000	–
	Cladding tube	1100 Aluminum	1.090	0.032
	Inner fuel tube	Uranium	2.120	0.180
"C" Assembly	Outer fuel tube	Uranium	3.500	0.320
	Housing tube	6063 Aluminum	4.000	0.050

FIG. 12 NATURAL URANIUM FUEL ASSEMBLIES USED IN PDP BUCKLING EXPERIMENTS

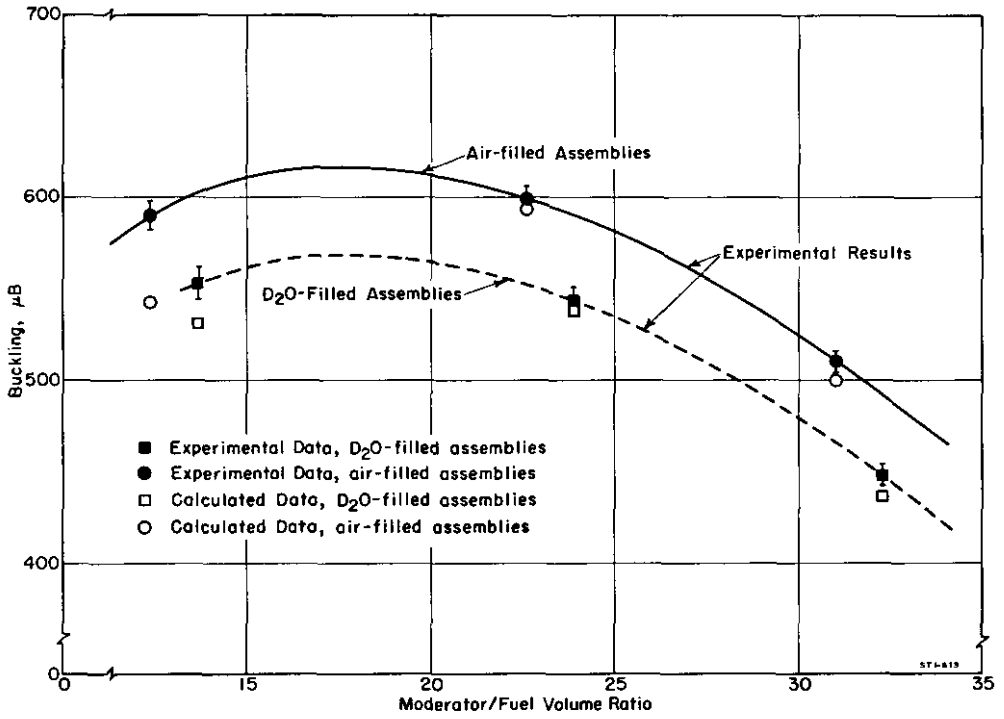


FIG. 13 EFFECT OF MODERATOR/FUEL RATIO ON BUCKLINGS OF NATURAL URANIUM ASSEMBLIES IN D_2O
Type "ABC" Assemblies (see Fig. 12)

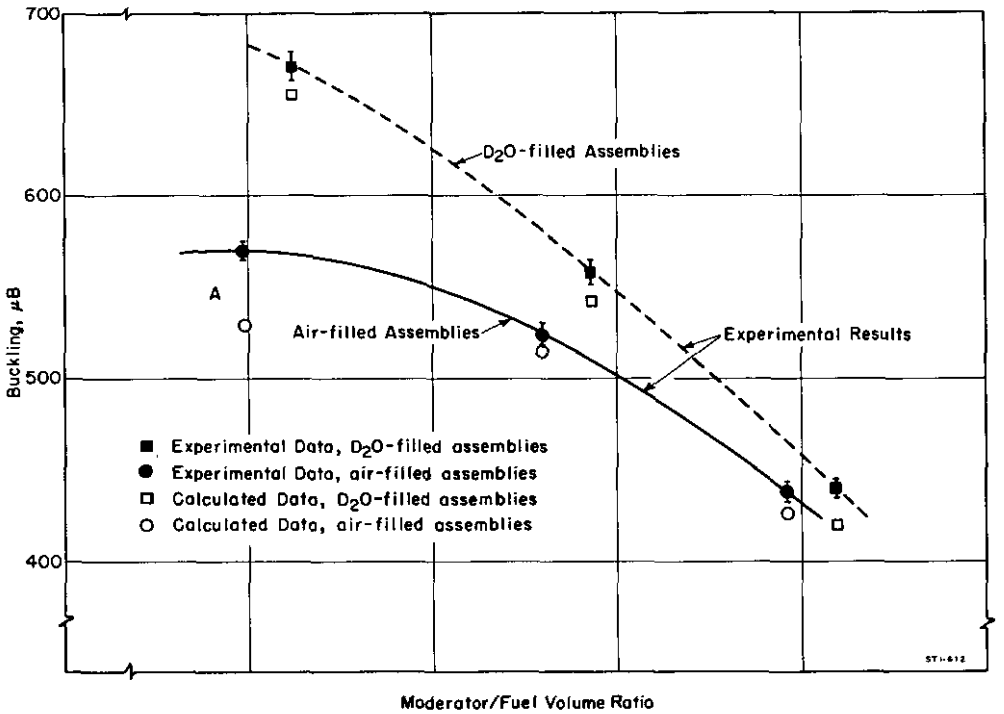


FIG. 14 EFFECT OF MODERATOR / FUEL RATIO ON BUCKLINGS OF NATURAL URANIUM FUEL ASSEMBLIES IN D_2O
Type "C" Assemblies (see Fig. 12)

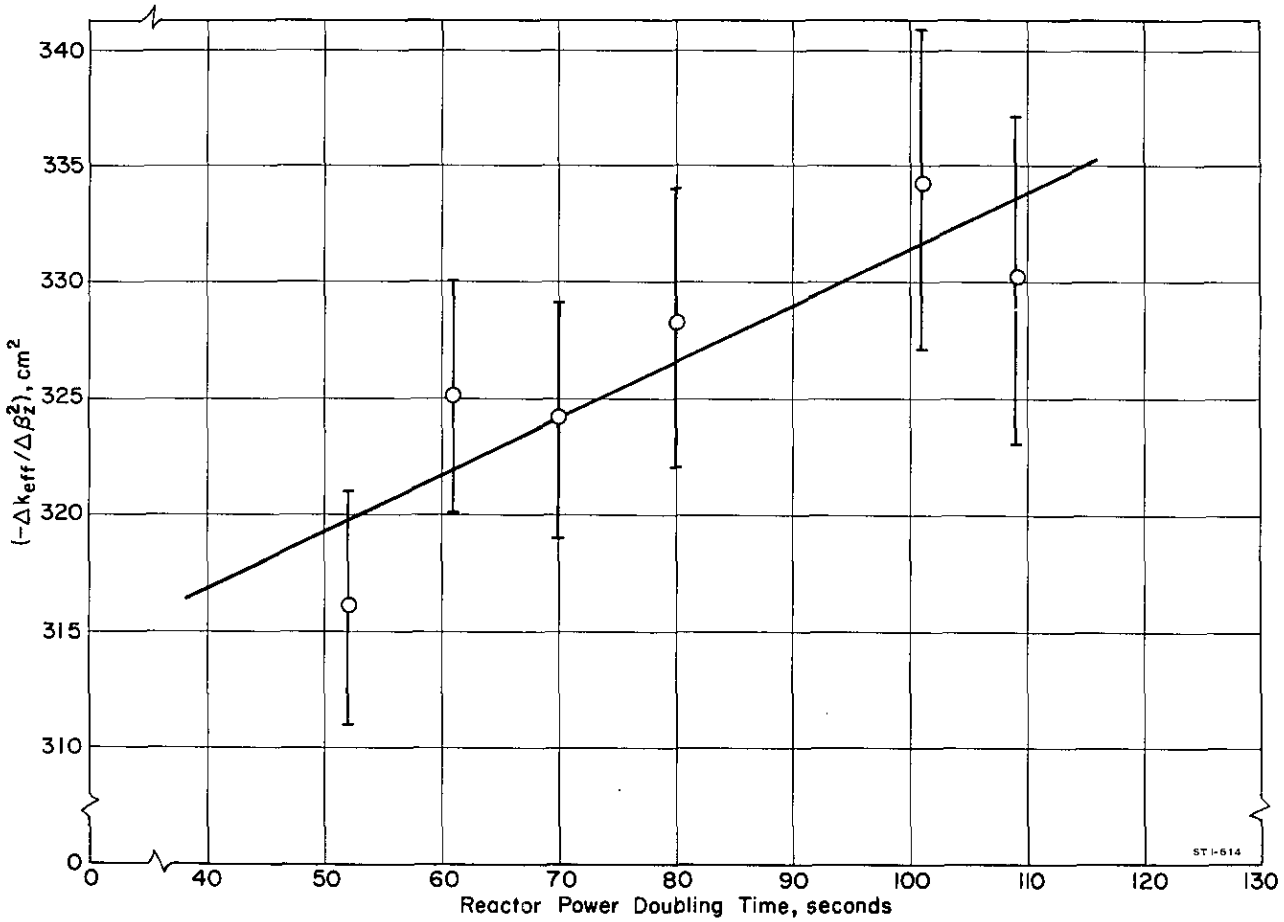


FIG. 15 DEPENDENCE OF MEASURED $(-\Delta k_{eff}/\Delta B_z^2)$ ON REACTOR PERIOD

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13. Previous progress reports in this series are:

DP-232	DP-395	DP-485	DP-575	DP-665	DP-755	DP-845
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Progress for the months of September and October 1963 will be reported in DP-875.