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DP-975

AEC RESEARCH AND DEVELOPMENT REPORT

HEAVY WATER MODERATED POWER REACTORS

PROGRESS REPORT
MARCH THROUGH JUNE 1965

Technical Division
Wilmington, Delaware

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HEAVY WATER MODERATED POWER REACTORS
PROGRESS REPORT
March through June 1965

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Power Reactor Studies
Wilmington, Delaware

Compiled by R. R. Hood

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ABSTRACT

This report is the last in a series of periodic reports that summarize the progress since 1956 of the Du Pont development program on heavy-water-moderated power reactors.

Postirradiation examination was completed on 14 Zircaloy-clad tubes of mechanically compacted UO_2 that operated at maximum thermal ratings of 25 to 40 watts/cm to exposures as high as 17,000 MWD/MTU. The results indicated that the tubes would have been capable of irradiation to even higher exposures.

Postirradiation examination of Zircaloy-clad tubes of uranium metal alloyed with small amounts of Fe, Si, and Al revealed that these alloys have excellent promise of operating satisfactorily in a D_2O-moderated power reactor to exposures of about 10,000 MWD/MTU.

A multigroup, multiregion code, EPITHET, was developed for calculating neutron flux spectra and neutron absorptions in a cylindrical lattice cell at epithermal energies.

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INTRODUCTION

Since 1956, the Du Pont Company has conducted a development program on heavy-water-moderated power reactors for the U.S. Atomic Energy Commission. The progress of the program, which has emphasized reactors that are cooled with liquid D_2O, has been summarized in a series of periodic reports[a]. The Atomic Energy Commission decided late in 1964 to curtail its activities on water-cooled D_2O reactors, and to concentrate development efforts on organic cooling. Consequently, the Du Pont power reactor program has been terminated; this report is the last progress report on the program.

SUMMARY

Postirradiation examinations were completed on 14 Zircaloy-clad tubes of mechanically compacted uranium oxide that were irradiated in the Heavy Water Components Test Reactor. The tubes were irradiated at maximum thermal ratings ($\int kd\theta$) of 25 to 40 watts/cm to exposures as high as 17,000 MWD/MTU*. They operated in a satisfactory manner, and nothing was observed in the postirradiation examination to indicate that they would not continue to operate satisfactorily to exposures of 20,000 to 30,000 MWD/MTU.

Collapse tests on Zircaloy-clad UO_2 tubes at temperatures as high as 350°C indicated that the UO_2 core prevents collapse of the inner sheath at stress levels below the proportional limit of the sheath material. The hoop stresses at collapse were 4 to 9 times those at which elastic collapse should occur.

Zircaloy-clad tubes of uranium metal alloyed with small amounts of Fe, Si, and Al increased in volume a maximum of 3% when irradiated to exposures as high as 5600 MWD/MTU at a maximum uranium temperature of 425°C. The swelling was much less than that of unalloyed uranium. The U-Fe-Si-Al alloys show excellent promise of operating satisfactorily to exposures of the order of 10,000 MWD/MTU in D_2O-moderated power reactors; however, confirmatory irradiation tests will be necessary.

A multigroup, multiregion method was developed for calculating neutron flux spectra and neutron absorptions in a cylindrical lattice cell at epithermal energies. The method is incorporated in a computer code, EPITHET. Another code, HAMMER, was developed that automates some

* Megawatt-days per metric ton of uranium.

of the calculations for frequently used lattice codes and bases the codes on a common mathematical model. Both EPITHET and HAMMER are available in FORTRAN language.

The temperature coefficient of reactivity of a D_2O-moderated lattice of natural uranium metal (1.00-inch-diameter rods) was measured in an exponential facility. The measured coefficient at 20 to 80°C was -0.75 ±0.16 μB/°C.

DISCUSSION

I. THE HEAVY WATER COMPONENTS TEST REACTOR (HWCTR)

A. REACTOR STATUS

The HWCTR is being maintained in the standby condition described in DP-965. Weekly inspections are made to determine that the ventilating and heating equipment is operating, that the supply of nitrogen for purging the process system is adequate, and that all other conditions are normal. Approximately 50 scf of nitrogen per day are required to maintain a pressure of 5 inches of water in the process system.

The steel surface of the containment vessel underneath the insulation was inspected in June 1965. The condition of the surface was not noticeably different from that of the last previous inspection (in November 1964). The insulation was very wet as a result of heavy showers preceding the inspection. The rate of growth of the pits continues to decrease, as shown in Table I. These observations further confirm the conclusion that the strength of the building is not expected to be seriously impaired in less than 3 to 5 years.

B. REPORT STATUS

Drafts of the following Research and Development reports have been prepared and are in the process of publication.

Report No.	Title
DP-964	Corrosion and Wear of Equipment at the HWCTR
DP-967	Low Power Physics Tests at the HWCTR
DP-968	Containment Building Leak Tests at the HWCTR
DP-971	Performance of Rod Drives at HWCTR
DP-988	Control of Dissolved Gases in the HWCTR
DP-991	Operational Summary of the HWCTR, 1962-1964
DP-600 Suppl. A	Final Hazards Evaluation of the HWCTR

II. REACTOR FUELS

A. SUMMARY OF TEST IRRADIATIONS IN THE HWCTR

Table II summarizes the fuel irradiation tests that were performed in the HWCTR from its startup in October 1962 until its shutdown in December 1964. The major results of the irradiations are reported elsewhere in this report (Sections IIB and IID, below) and in previous reports in this series. Topical reports on test irradiations of uranium oxide tubes and uranium metal tubes are being prepared.

B. POSTIRRADIATION EXAMINATION OF URANIUM OXIDE TUBES

Examination was completed on 14 Zircaloy-clad tubes selected from the 4 assemblies of compacted UO_2 tubes that were under irradiation in the HWCTR when operation was terminated. These tubes operated at moderate ratings and exposures in a satisfactory manner; nothing was observed during the examination to indicate that they would not have continued to operate satisfactorily to goal exposures. A brief description of the objectives and conditions of the irradiation tests is presented below.

Assemblies of UO_2 Fuel Tubes
Selected for Postirradiation Examination

Assembly Designation	Purpose of Test	Maximum $\int kd\theta$, watts/cm	Maximum Exposure, MWD/MTU	Goal Exposure, MWD/MTU
SOT-1-2	Demonstrate behavior of 2.1-inch-diameter UO_2 tubes to high exposures at moderate thermal ratings	25	17,300	30,000
SOT-6-2 SOT-9-2	Demonstrate behavior of intermediate-size (2.5-inch-diameter) UO_2 tubes of reference fuel assembly described in DP-885	29 40	5,100 4,000	30,000 20,000
SOT-8-3	Demonstrate behavior of largest UO_2 tubes (3.7-inch-diameter) of reference fuel assembly described in DP-885	32	4,300	30,000

Representative tubes from each assembly were examined in detail in High Level Caves. Visual and dimensional inspections revealed that no significant changes occurred during irradiation. The free gas within each element was collected and analyzed (Table III). Metallographic examination of cross sections of the tubes was made to correlate changes in core structure during irradiation with exposures and thermal ratings. Burnup analyses are being performed on samples from the two high-exposure elements in the SOT-1-2 assembly.

1. Comparative Behavior of Swage-Compacted and Vibratory-Compacted Tubes

Examination of the swage-compacted and vibratory-compacted UO_2 tubes in assembly SOT-1-2 indicated that UO_2 cores of vibratory-compacted tubes attain central temperatures somewhat above those of swaged tubes irradiated under the same conditions. These results extend previous data obtained on similar elements irradiated to 9400 MWD/MTU (DP-905). One swaged tube (222-E) irradiated to an average exposure of 17,000 MWD/MTU at a peak thermal rating of 25 watts/cm exhibited no change in core structure and released only 5% of the gaseous fission products (Table III). Tube 232-D, compacted by vibration alone, released 22% of the gaseous fission products when irradiated under conditions that were only slightly less severe than those for the swaged tube. This vibrated tube also formed a circumferential crack in the annular UO_2 core; this type of crack generally characterizes central temperatures at or above the temperature of columnar grain growth.

Comparison of the present results for swage-compacted tubes irradiated to 17,000 MWD/MTU with results on similar elements irradiated to peak exposures of 9400 MWD/MTU indicates that the effect of the exposure on the maximum central temperature is slight in the range of conditions tested. However, exposure apparently has an appreciable effect on the temperature of vibratory-compacted tubes. For Tube 232-D (Assembly SOT-1-2, 16,500 MWD/MTU), both the release of fission gas and the structure of the core indicate that the peak central temperature was higher than in Tube 228-G (Assembly SOT-1-3, 9000 MWD/MTU). Both tubes were in equivalent positions in the fuel column and operated at maximum thermal ratings of 24 watts/cm. Possible reasons for the higher central temperature in Tube 232-D are:

(a) Flux peaking of a transient nature.

(b) A decrease in thermal conductivity of the gas phase due to the accumulation of fission gases.

or (c) Lower thermal conductivity in the oxide core due to the slight difference in packing density (84.3% in 232-D vs. 85.5% in 228-G).

2. Intermediate- and Large-Diameter Elements

In the tubes of intermediate size (Assemblies SOT-6-2 and SOT-9-2), the difference in thermal ratings is reflected by both fractional release of fission gas (Table III) and by structure of the oxide core. Extensive grain growth occurred in the cores of SOT-9-2 elements, which operated at a peak $\int k d\theta$ of 40 watts/cm. The cores in SOT-6-2, which operated at a peak $\int k d\theta$ of 29 watts/cm, sintered but did not reach temperatures sufficient for grain growth.

The large-diameter elements (Assembly SOT-8-3) operated at about the same thermal ratings as the SOT-6-2 elements and showed the same irradiation behavior. Both fission gas release and core structure after irradiation were essentially the same in the two assemblies.

3. Cladding Structure

Zirconium hydride concentrated at the coolant side of the Zircaloy-2 cladding, and a thin oxide layer was observed on the core side of the cladding. However, neither the presence of hydride nor the formation of oxide appears likely to impair the serviceability of the cladding. The regions of concentrated hydride penetrated only about 0.007 inch into the cladding. In the outer sheaths, the hydride platelets were oriented preferentially in the rolling plane; in the inner sheaths, the platelets were oriented randomly or in a radial direction. The heaviest concentrations of hydride were observed in the Zircaloy-2 sheaths of the large- and intermediate-size tubes, SOT-8 and SOT-6 tests, irradiated for 88 and 160 days, respectively.

Sheaths fabricated from low-nickel Zircaloy-2 exhibited very little zirconium hydride after irradiation. Test pieces from SOT-1-2 irradiated for 416 days exhibited trace concentrations of the fine hydride platelets similar to hydride concentrations reported earlier on like tubes after an irradiation of much shorter duration (DP-695, DP-705). Hydride distributions between the two tests were different, however. The SOT-1-2 tubes were irradiated under a coolant pressure of 1200 psi, and the pressure within the tubes was always less than the coolant pressure. The resulting sheath stresses led to circumferentially oriented platelets in the outer sheaths and radially or sometimes randomly oriented platelets in the inner sheaths. The earlier irradiation of short duration was performed in a reactor with a coolant pressure below 200 psi. As a result, the internal pressure exceeded the coolant pressure, and the resultant stresses in the sheaths led to random or radially oriented platelets in the outer sheaths and predominantly circumferentially oriented platelets in the inner sheaths (DP-695, DP-705).

C. SHEATH COLLAPSE TESTS

An analysis of collapse tests on four different sizes of UO_2 tubes of interest for power reactors indicates that the UO_2 core prevents collapse of the inner sheath at stress levels below the proportional limit of the sheath material. It is believed, therefore, that the minimum pressure to cause collapse of the inner sheath for tubes with UO_2 densities in the range of 80 to 92% of theoretical can be calculated on the basis of sheath stress without regard to geometry factors such as length or end plug configuration.

Collapse tests reported earlier (DP-765, DP-775, DP-855) demonstrated that appreciable support is afforded the inner sheath of UO_2 tubes by the compacted core. In these earlier tests on Zircaloy-clad tubes, sheath diameters and thicknesses were limited to 1.2 to 1.5 inches and 0.030 to 0.038 inch, respectively. More recent collapse tests have been performed to extend the range of diameters up to 3.0 inches and the range of sheath thicknesses down to 0.022 inch. These tests were performed at three temperatures of interest:

(1) Room temperature.

(2) 100°C - corresponds to maximum sheath temperature during reactor shutdown when coolant pressure is removed.

(3) 350°C - corresponds to maximum sheath temperature during reactor operation.

The recent tests showed that the inner sheaths are significantly strengthened by the UO_2 core (Table IV). The hoop stresses at collapse were 4 to 9 times higher than those calculated for elastic collapse and were sufficiently high to indicate that collapse was initiated by inelastic behavior of the sheath at hoop stresses above the proportional limit of the (cold-swaged) Zircaloy.

D. POSTIRRADIATION EXAMINATION OF URANIUM METAL TUBES

As described in previous reports in this series, tubes of unalloyed uranium and dilute uranium alloys have been irradiated in the HWCTR to exposures as high as 7000 MWD/MTU at temperatures up to 500°C to establish limitations on their performance under power reactor conditions. Results of postirradiation examinations of unalloyed uranium tubes with 0.025-inch-thick Zircaloy cladding were reported in DP-875 and DP-945. Postirradiation examinations have now been completed for the remainder of the tubes, which are listed below and described in Table V:

Unalloyed U with thick (0.060-inch) cladding

U - 330 ppm Fe - 900 ppm Al (0.022-inch cladding)

U - 350 ppm Fe - 300 ppm Si - 800 ppm Al (0.022-inch cladding)

U - 1.5 wt % Mo (0.025-inch cladding)

All of the tubes listed above exhibited less swelling during irradiation than did the unalloyed uranium tubes inspected earlier. It is concluded that alloy additions and external restraints can produce major improvements in the irradiation behavior of uranium metal under power reactor conditions. The best specimens (U-Fe-Al and U-Fe-Si-Al) appeared capable of withstanding higher exposures than were actually attained (5600 MWD/MTU maximum). Prospects are good for achieving exposures of the order of 10,000 MWD/MTU with these alloys. However, irradiation tests at higher temperatures (~500°C) will be necessary to confirm the performance of the alloys.

1. Irradiation Conditions

Irradiation conditions, including those for the unalloyed uranium tubes discussed in DP-945, are summarized in Table VI. The tubes were irradiated in at least two exposure increments to exposures as high as 6800 MWD/MTU at time-averaged central metal temperatures up to 500°C. The maximum temperature usually varied about 75°C from one exposure increment to the next.

2. Dimensional Changes Due to Irradiation

The data on dimensional changes due to irradiation are presented in Table VI and Figure 1. Since length changes were negligible, the uranium volume changes were assumed equal to changes in cross-sectional area; the latter was calculated from measured diameters. The calculated volume changes were negative at low exposures for the RMT and SMT-1-3 assemblies because of a systematic error of 1% in the measurements; consequently, all data for these tubes were adjusted to extrapolate to zero swelling at zero exposure. Accurate swelling data could not be obtained for the ETWO-2 assembly because the fuel tube could not be separated from its inner housing. Also, buckling of the inner cladding of Tube ETWO-3 precluded accurate swelling measurements on that tube. The estimated swelling of 10% for ETWO-2 was inferred from measurements of outer diameter only.

(a) Thick-Clad Tube of Unalloyed Uranium

The irradiation behavior of unalloyed uranium was improved by increasing the cladding thickness from 0.025 to 0.060 inch. With the thicker cladding, the swelling at 3000 MWD/MTU was only 2.5% (Figure 1a), as compared with an estimated 10% for 0.025-inch-thick cladding. The thicker cladding may not have been the only factor responsible for the reduction in swelling; a lower fission rate (38 MW/MTU vs. 85 MW/MTU) may have been a contributing factor. At exposures up to 1500 MWD/MTU, the measured growth was fully accountable by the swelling due to the accumulation of solid fission products. At higher exposures, the formation of cavities in the uranium added to the volume increase. Metallographic examination of the tube revealed that cavities formed at temperatures as low as 250°C.

(b) U-Fe-Si-Al Alloys

At exposures below 5000 MWD/MTU, the swelling of U-Fe-Al and U-Fe-Si-Al tubes did not deviate appreciably from that caused by solid fission products. The volume increase in both alloys was about 2.5% at this exposure. Again, part of the reduction in swelling relative to unalloyed uranium may have been caused by a lower fission rate. Cavity formation may have contributed to the swelling at exposures beyond about 4000 MWD/MTU. A few small cavities were detected in both alloys by metallographic examination.

(c) U - 1.5 wt % Mo

The swelling behavior of U - 1.5 wt % Mo was worse than that of the U-Fe-Si-Al alloys but better than that of unalloyed uranium. Deviations from the line representing solid fission product accumulation were appreciable at exposures above 2500 MWD/MTU, and at 5000 MWD/MTU the volume increase was 5 to 6%. The ID of the tube decreased more than the OD increased (0.017 inch vs. 0.007 inch), which indicates a tendency for the tube to swell by buckling of the inside surface. Relatively large cavities were present in the U-Mo after irradiation.

3. Fuel and Cladding Integrity

During the metallographic examination of the irradiated specimens, occasional cracks were observed that extended radially through the core, but did not penetrate the cladding. Platelets of zirconium hydride were concentrated in the cladding near the tips of the cracks. This circumstance might eventually lead to a cladding failure, but no failures occurred in these tests.

E. POSTIRRADIATION EXAMINATION OF HWCTR DRIVER TUBES

The HWCTR was fueled with large-diameter driver tubes of Zr - 9.3 wt % U (93% ^{235}U) alloy in Zircaloy-2 cladding. Because little data were available on the irradiation behavior of this alloy at HWCTR conditions, the driver tubes were monitored periodically for dimensional changes during their operation in the HWCTR. The last inspection was performed upon completion of HWCTR operation. The results, including those of earlier interim examinations, are summarized in Table VII and are discussed in the following paragraphs.

Irradiation of the driver tubes to 1.83 atom % burnup at time-averaged central metal temperatures as high as 540°C resulted in a maximum volume increase of 5.1%. This swelling occurred as an increase in wall thickness and was caused by the formation of solid fission products and the precipitation of fission gases in small bubbles.

The volume increases were measured by two alternative methods: (1) inside and outside diameters were measured at several points along the 10-ft length of each tube, and (2) outside diameters and wall thicknesses were measured on ring sections that were cut from the tubes. Generally higher volume changes were indicated by the latter method than by the former. The discrepancy was attributed to a systematic error in the measurements on the full-length tubes; therefore, these measurements were normalized to zero volume change at zero burnup. The normalized data are shown in parentheses in Table VII.

The swelling of the driver tubes was about one-half as great as the maximum expected from earlier irradiation tests of small samples of U-Zr alloys containing 6 to 22 wt % uranium. The reason for the improved behavior is not known, but may be the restraining effects of the coolant pressure (1200 psi) and the outer, colder layers of the core inasmuch as these factors were absent in tests of the small samples at comparable temperatures.

Fission gas bubbles up to 0.4 micron in diameter were present in the hottest parts of the driver tubes, but not in the colder regions. Volume increases due to these bubbles, calculated from measurements of bubble diameters and concentrations, ranged as high as 2.1%; the bubble volume plus the swelling due to solid fission products (2.5% per atom % burnup) accounted for the observed volume increases. Sometimes the bubbles were associated with the interface between the ϵ phase and the α zirconium matrix, as well as randomly throughout the α zirconium. No bubbles could be positively identified within the ϵ phase in spite of the much higher uranium concentration there as compared to the matrix.

The irradiation had no apparent effect on the Zircaloy cladding of the driver tubes. Corrosion was negligible. The circumferential tensile strain in the outer cladding due to core swelling was less than 0.3%.

III. REACTOR PHYSICS

A. CALCULATION OF NEUTRON FLUX AND ABSORPTION AT EPITHERMAL ENERGIES

A multigroup, multiregion method has been developed for calculating flux spectra and neutron absorptions in a cylindrical lattice cell at epithermal energies. Initial work on this method is described in Reference 1. The method is now embodied in a computer code, EPITHET, that is available in FORTRAN for the IBM 704 computer.

Within each energy group, the number of collisions in each region is calculated from the slowing-down source of neutrons from all higher groups and from the neutron currents across the boundaries of the region. The neutron currents are calculated from conservation equations involving probabilities of escape from, or collisions in, the region. The group widths and region thicknesses can be varied at will so that the neutron flux can be obtained as a function of energy and radius in any desired detail. The effects of scattering interference and the interaction of resonances either of the same or of different materials can be calculated in detail. The effects of interior surfaces in the fuel assembly can also be taken directly into account.

The EPITHET code is a useful adjunct to complete lattice reactivity codes such as HAMMER or BSQ since it enables a more detailed determination of neutron events in the epithermal energy range.

B. DEVELOPMENT OF THE HAMMER CODE

Since the appearance of large-scale computing machinery in the 1950's, numerous computer programs have been devised for nuclear reactor calculations. Moreover, a "standard" group of reactor codes have come into widespread use at U.S. laboratories for both reactor design problems and the analysis of lattice experiments. Recently, Honeck and Crandall[2] applied a selected set of these codes, namely THERMOS[3], MUFT[4], ZUT and TUZ[5], and FOG[6] to the analysis of some one-hundred critical experiments on clean, D_2O-uranium lattices. Good agreement between theory and experiment was obtained; the average calculated k_{eff} was 0.996. The same battery of codes has also been applied to a series of light water lattices with equally good results[7]. Furthermore, each of these codes has been proven individually through extensive use.

The experience gained in the above-mentioned series of calculations showed a definite need for the automation of these standard codes to eliminate the burdensome tasks of preparing redundant sets of input for the various lattice codes. At the same time, it was thought desirable to base all of the lattice codes on a consistent mathematical model, namely multigroup, multiregion integral transport theory for infinite lattice calculations, followed by homogenized-cell, Fourier transform leakage calculations. Finally, it was desired to produce a sufficiently general system to be of use at most U.S. laboratories in order to facilitate inter-laboratory comparisons of reactor calculations. The result of this project is the HAMMER system (an acronym for Heterogeneous Analysis of Multigroup, Multiregion Exponentials and Reactors), an IBM 704/7090 FORTRAN II code developed jointly at Brookhaven National Laboratory and at the Savannah River Laboratory. The code is now in limited use at Savannah River, where the final stages of code checkout are being performed.

From a single set of data cards, the HAMMER code will sequentially perform any or all of the following operations:

(1) Program CAPN - interpret infinite lattice cell input data and perform extensive error checking; establish path of control among the programs mentioned below.

(2) Program THERMOS - perform a multigroup calculation of thermal flux distribution ($E < 0.625$ ev) from integral transport theory, followed by a multigroup Fourier transform leakage spectrum correction based on homogenized-cell parameters (multigroup flux-volume weighting). Cross section averages and diffusion parameters are provided in the output, as well as fluxes and reaction rates.

(3) Program HAMLET - same as (2), above, except that $0.625 \text{ ev} \leqslant E \leqslant 10$ Mev. This is a major block of new programming, since no suitable codes were available for epithermal integral transport. Collision probabilities are computed under the assumption of cosine currents crossing region boundaries, and the energy spectrum during moderation is computed as in MUFT. This approach automatically produces a heterogeneous calculation of ϵ and p; for the latter, ZUT/TUZ is employed as a subroutine. In addition to the output described in (2), above, a set of four-group (10 Mev - 1.05 Mev - 9.12 kev - 0.625 ev) cross sections is produced, from which the infinite lattice asymptotic spectrum and buckling are determined.

(4) Program FLOG - utilizes the four-group constants produced in (2) and (3), above, to perform a one-dimensional, many-region reactor calculation. This program is a modification of FOG. In addition to the standard criticality searches, FLOG will perform least-squares fitting

of reactor fluxes to obtain "experimental" values of the buckling, and will automatically calculate a series of core size loadings for exponential assemblies.

(5) Program DIED - The fluxes and cross sections obtained in (2) through (4), above, are combined to produce neutron balance sheets in this final edit program. For an infinite lattice cell edit (results of (2) and (3) only), the output is either (a) absorptions vs. fissions by isotope and group, split into "smooth" and "resonance" event contributions, or (b) total absorptions vs. total fissions by isotope and group. For a composite reactor edit (results from (4)), neutron balance sheets are produced for total absorptions vs. total fission neutron production by isotope and region. These latter tables can be produced for all regions, for all fissioning regions, or for selected regions. The flux weighting employed in the tables can be either that appropriate to the region average, the asymptotic core, the interfaces, or all of these.

C. NUCLEAR REACTIVITY OF HWCTR DRIVER ASSEMBLIES

New driver fuel assemblies were ordered for the HWCTR prior to the decision to put the reactor in a standby condition. These new (M-3) drivers are stored now for use if the reactor is put back in operation.

Reactivity measurements were made in the Process Development Pile (PDP) to see how accurately these new M-3 drivers matched the M-1 drivers, the first set used in the HWCTR. The M-3 lattice was slightly less reactive than the M-1 lattice. If the M-3 target tubes used in the PDP test contained approximately the nominal boron content of 0.175 weight percent, the M-3 lattice in the HWCTR would be less reactive than the M-1 lattice by 1 to 2% k. However, little doubt exists as to the lattice reactivity being acceptable for nuclear operation; full-power operation could be attained with no difficulty. The reactivity lifetime of the M-3 fuel is independent of the initial boron content of the target tubes, since the target tubes can later be replaced by Zircaloy inner housings.

D. TEMPERATURE COEFFICIENTS OF REACTIVITY

The temperature coefficient of reactivity of a D_2O-moderated lattice of natural uranium fuel was measured in a 5-ft-diameter exponential facility. The fuel assembies were solid rods of natural uranium, 1.00 inch in diameter, in aluminum tubes having an OD of 1.090 inches and an ID of 1.026 inches. The gap between the uranium and the aluminum contained air. Sixty-one such rods were loaded in a triangular lattice pattern with a 7.00-inch pitch. Measurements of the relaxation length K were made as a function of moderator temperature by means of a traveling flux monitor. The change in K^2 with temperature is shown in the following table; the quoted uncertainties are those attached to the individual K^2 measurements. The value of K^2 at 22°C was subtracted from all measured values of K^2.

Temperature, °C	Change in K^2 from that at room temperature (22°C), μB
22	0 ± 4
30	5 ± 5
49	15 ± 6
61	27 ± 4
72	28 ± 4
83	47 ± 7

If constant radial buckling is assumed, a straight-line fit to the above data, by use of weighting functions inversely proportional to the square of the uncertainty levels, yields a temperature coefficient of -0.65 ±0.15 μB/°C. A calculated change in radial buckling yields an additional contribution of -0.10 ±0.02 μB/°C, resulting in a total coefficient of -0.75 ±0.16 μB/°C. The quoted uncertainty represents a confidence level of 90%. These results are in reasonable agreement with those of Andrews and Freibergs[a].

TABLE I

Summary of Inspections of HWCTR Containment Vessel

Date	Maximum Pit Depth Under Adhesive, mils	Maximum Pit Growth, mils/month
August 1963	15	2.2
May 1964	35	1.2
November 1964	42	0.5
June 1965	45	

TABLE II

Summary of Fuel Irradiation Tests in the HWCTR

All tests performed with liquid D_2O cooling at 1200 psi and ~200°C

Element Name	Maximum Specific Exposure	Time-Averaged Axial Peak Specific Power	Time-Averaged Axial Peak $\int k d\theta$ or T	Peak $\int k d\theta$ or T Achieved	Date Irradiation Started	Date Irradiation Ended	Remarks
A. Uranium Oxide Tubes							
	MWD/MTU	MW/MTU	watts/cm	watts/cm			
OT-1-2	12,390	53	21.5	26	10-5-62	8-7-64	
OT-1-3	5,200	56	22.5	25	1-6-63	6-20-63	Failed
OT-1-4	9,200	54	21.5	26	1-6-63	10-31-64	
OT-1-5	6,030	59	21	23	1-6-63	6-20-63	
OT-1-6	6,035	64	26	29	1-6-63	6-7-63	Failed
OT-1-7	10,300	58	23.5	33	7-13-63	10-31-64	
OT-3-2	1,710	38	13.5	14	5-19-63	8-25-63	
SOT-1-2	17,330	51	19	25	10-5-62	12-1-64	
SOT-1-3	9,380	56	21.5	25	1-6-63	11-29-63	
SOT-1-4	12,640	61	23	31	7-13-63	12-1-64	
SOT-2-2	3,460	37	44.5	50	1-6-63	5-31-63	Failed
SOT-2-3	3,170	36	42.5	49	12-29-62	4-3-63	Failed
SOT-5-2	1,695	35	51	56	5-19-63	9-28-63	Failed
SOT-6-2	5,065	35	23.5	28	12-29-63	12-1-64	
SOT-6-3	2,735	40	27	30	5-11-64	10-31-64	
SOT-7-2	1,110	46	66	68	10-3-63	10-29-63	Failed
SOT-8-2	3,985	54	27	30	8-25-64	12-1-64	
SOT-8-3	4,310	59	28.5	32	8-25-64	12-1-64	
SOT-9-2	4,040	54	36	40	5-11-64	12-1-64	
B. Uranium Metal Tubes							
	MWD/MTU	MW/MTU	°C	°C			
TWNT-5	960	31	360	365	10-5-62	12-26-62	
TWNT-7	700	34	371	375	10-5-62	11-27-62	Failed
TWNT-9	1,005	32	365	368	10-5-62	12-26-62	
TWNT-11	890	34	370	375	10-5-62	12-9-62	
TWNT-12	1,085	35	375	386	10-5-62	12-26-62	
TWNT-13	385	37	380	384	12-2-62	12-26-62	
TWNT-14	1,010	32	365	369	10-5-62	12-26-62	Failed
TWO-1-2	1,195	38	385	390	10-5-62	12-26-62	
ETWO-2	6,830	85	480	495	5-19-63	11-29-63	
ETWO-3	6,470	90	490	515	6-11-63	11-29-63	
SMT-1-2	4,325	29	390	430	5-19-63	8-7-64	
SMT-1-3	5,600	37	425	460	7-13-63	12-1-64	
EMT-2	4,970	87	440	485	5-11-64	12-1-64	
RMT-1-2	3,320	38	445	485	12-29-63	12-1-64	
C. Thorium Metal Tubes							
	MWD/MT	MW/MT	°C	°C			
TMT-1-2	3,560	49	470	505	8-25-64	12-1-64	
TMT-1-3	3,470	48	465	495	8-25-64	12-1-64	

TABLE III

Free Gas in Irradiated UO_2 Tubes

Assembly	Element No.	Peak $\int k d\theta$, watts/cm	Average Exposure, MWD/MTU	Total Free Gas, cc/kg	Xe, cc/kg	Kr, cc/kg	% Release of Xe + Kr	N_2, cc/kg	He, cc/kg	Ar, cc/kg	$O_2 + CO_2$, cc/kg	Notes
SOT-1-2	222-E	25	17,000	68.1	19.7	3.3	5	21.6	19.8	3.4	0.35	
	232-D	24	16,500	224.0	84.0	13.3	22	97.4	21.7	7.1	0.42	(a)(b)
	225-C	23	13,000	73.1	25.3	2.1	8	29.3	15.7	0.5	0.10	
	229-F	16	6,500	23.0	3.7	0.3	2	0.2	18.4	0.3	0.04	(b)
SOT-8-3	297-B	33	4,300	39.5	12.0	1.0	11	8.2	17.4	1.1	0.06	
	290-B	32	4,000	40.0	9.6	0.8	10	13.4	15.8	0.4	0.04	
	289-D	27	3,000	34.2	3.4	0.3	5	13.5	15.7	0.9	0.23	
SOT-9-2	282-A	37	2,800	78.2	11.7	1.1	17	31.6	32.8	0.9	0.2	
	281-B	38	3,800	68.0	19.7	3.2	22	22.5	21.3	0.5	0.1	
	280-C	40	4,000	90.0	40.0	3.4	40	25.0	20.6	0.1	0.5	
SOT-6-2	273-A	28	3,100	37.0	4.2	0.4	5	12.0	19.3	1.0	0.1	
	278-C	27	4,900	48.5	11.6	0.9	9	14.1	21.1	0.8	0.1	
	272-C	27	4,900	35.4	11.6	0.9	9	5.3	16.4	1.0	0.1	
	273-C	23	3,600	40.5	5.0	0.4	6	14.2	19.4	1.4	0.1	

(a) Oxide in SOT-1-2 assembly was vacuum outgassed 1 hour at 1000°C; oxide in other assemblies was outgassed 4 hours at 1500°C.
(b) Vibratorily compacted. All other elements were swage compacted.
(c) Gas volumes are reported for standard conditions of temperature and pressure.

TABLE IV

Results of Short-Time Collapse Tests on Zircaloy-Clad UO_2 Tubes

Tubes were compacted by cold swaging to a UO_2 density of 90 to 92% of theoretical.

	Tube Dimensions, inches	
	3.66 OD x 2.99 ID x 14 long	2.54 OD x 1.83 ID x 13 long
Inner sheath dimensions, in.		
OD	3.05	1.88
thickness	0.033	0.022
Measured collapse pressure, psi		
at room temp.	520	600
at 100°C	435; 475	570; 575
at 350°C	400	490
Hoop stress at collapse, psi		
at room temp.	24,000	25,600
at 100°C	20,100; 22,000	24,300; 24,500
at 350°C	18,500	20,900
Calculated stress for elastic collapse of empty sheath at room temperature, psi[a]	6,000	2,900

(a) Assumptions:
(1) Young's Modulus = 15×10^6 psi
(2) Poisson's Ratio = 0.30
(3) Simple end support
(4) Same dimensions and end geometry as test specimens

TABLE V

Description of Uranium Metal Tubes Irradiated in the HWCTR

Test Series	Clad Dimensions of Tube, inches: OD	ID	Length	Clad Thickness, inch	Nominal Composition,(a) ppm: Fe	Si	Al	Mo, %	Heat Treatment(b)
ETWO-2	2.06	1.70	113	0.025	-	-	-	-	Beta oil-quench
ETWO-3	2.06	1.70	113	0.025	-	-	-	-	Beta oil-quench
RMT-1-2	2.07	1.57	120	0.060	-	-	-	-	Beta air-cool
SMT-1-2	1.70	1.24	10.8	0.022	330	-	900	-	Beta oil-quench, alpha anneal
SMT-1-3	1.70	1.24	10.8	0.022	350	300	800	-	Beta oil-quench, alpha anneal
EMT-2	2.06	1.70	42.5	0.025	-	-	-	1.51	Gamma slow-cool

(a) Except where indicated, the base impurity content was 400 ppm C, 100 ppm Fe, 50 ppm Si, and 10 ppm Al.

(b) β oil-quench - heated in molten salt at 735°C for 10 minutes and quenched in room-temperature oil.
β air-cool - heated in molten salt at 735°C for 10 minutes, air cooled 6 minutes and quenched in room-temperature water.
β oil-quench, α anneal - heated in molten salt for 15 minutes and quenched in room-temperature oil, then heated at 500°C for 60 minutes and air-cooled.
γ slow-cool - heated in an evacuated steel fixture to 775°C for 15 minutes, furnace-cooled to 535°C, held one hour, and air-cooled.

TABLE VI

Irradiation Behavior of Zircaloy-Clad Uranium Tubes During HWCTR Irradiation

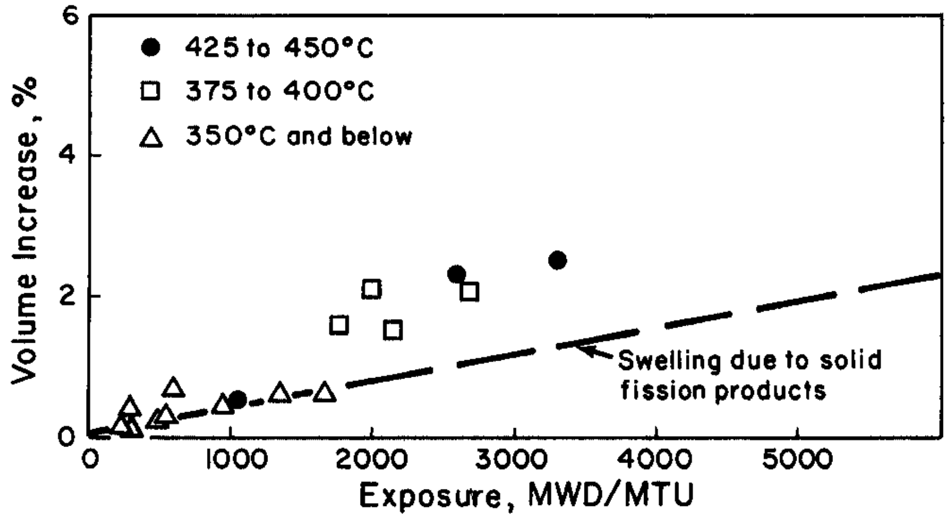
Test Series	Nominal Composition	Time-Avg. Central Temp., °C	Max. Exposure, MWD/MTU	Avg. Power, MW/MTU	Dimensional Changes, inch	Max. Volume Changes, %
ETWO-2	Unalloyed U	495	3400	-	OD +.020	10(est.)
ETWO-2	Unalloyed U	480	6800	85	OD +.040 to -.050 ID -.1	10(est.)
ETWO-3	Unalloyed U	490	6500	90	OD +.040 to -.035	-
RMT	Unalloyed U	445	3300	38	OD +.007 ID -.002	2.6
SMT-1-2	U - 330 ppm Fe - 900 ppm Al	390	4300	29	OD +.004 ID -.010	2.6
SMT-1-3	U - 350 ppm Fe - 300 ppm Si - 800 ppm Al	425	5600	37	OD +.006 ID -.010	3.3
EMT	U - 1.5 wt % Mo	440	5000	87	OD +.009 ID -.017	5.8

TABLE VII

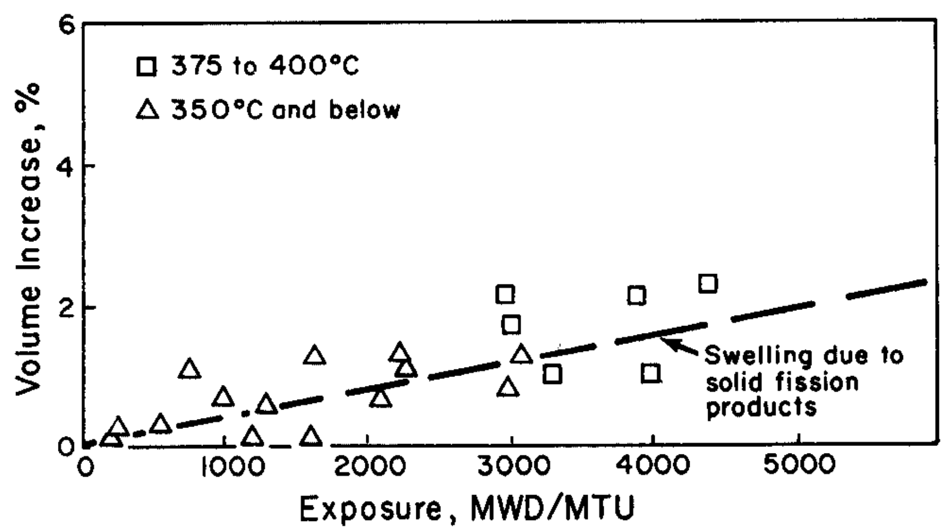
Irradiation Conditions and Dimensional Changes in HWCTR Driver Tubes

	Irradiation Conditions			Dimensional Changes				
Tube No.	Burnup, atom %	TACMT, °C[a]	Max. CMT, °C[b]	ΔOD, mils	Cladding Strain, %	ΔID, mils	ΔV/V, %	R,[c] %/atom %
48	1.47	486	536	4	0.175	-10	3.6 (5.1)[d]	3.4
	1.80	489	532	2	0.10	-11	3.2 (4.7)	2.6
22	1.18	501	531	3	0.13	- 6	2.5 (4.0)	3.4
	1.57	502	530	5	0.22	- 8	3.2 (4.7)	3.0
	1.83	508	528	5	0.17	- 8	3.4 (4.9)	2.7
1	0.32	533	539	-1	0	- 2	0.22	0.7
	0.76	534	569	2	0.094	- 7	2.75 (2.3)	3.3
	1.33	522	565	-1	0	- 6	1.16 (3.2)	2.4
	1.50	530	536	1 (5)	0.024 (0.172)	- 4 (-14)	0.8 (4.3)	2.9
18	0.32	523	539	0	0.147	- 4	1.09	3.4
	0.79	543	585	0	0.276	- 7	2.04 (1.5)	1.9
	1.38	534	572	1	0.084	- 5	1.96 (4.3)	3.1
	1.54	538	509	1 (7)	0.028 (0.269)	- 2 (- 9)	0.56 (4.1)	2.7

(a) Time-averaged central metal temperature.
(b) Maximum central metal temperature during irradiation.
(c) Swelling rate - percent increase in volume per atom percent burnup.
(d) Data based on measurements of ring sections. Volume increases adjusted to extrapolate to zero at zero burnup.

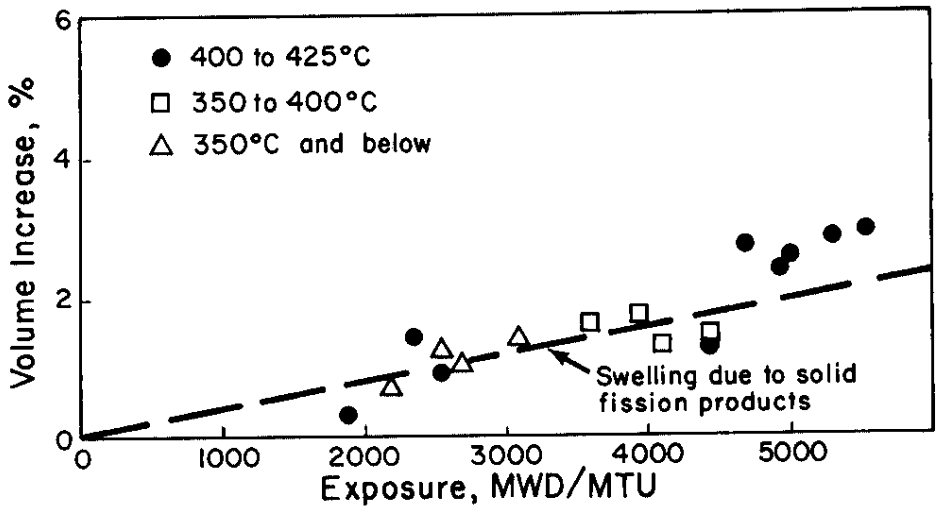


a. Unalloyed Uranium with Thick Cladding

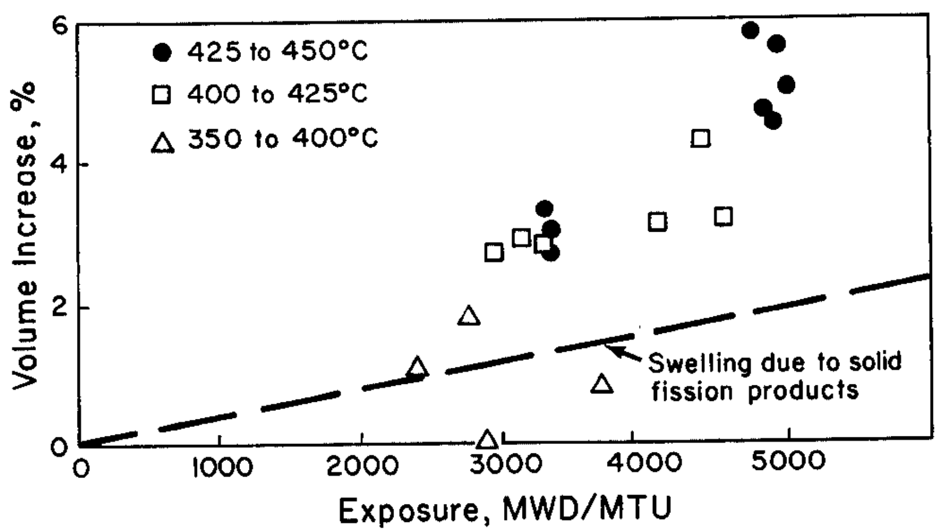


b. U - 330 ppm Fe - 900 ppm Al

FIG. 1 SWELLING OF URANIUM METAL TUBES DURING HWCTR IRRADIATION TESTS



c. U - 350 ppm Fe - 300 ppm Si - 800 ppm Al



d. U - 1.5 wt % Mo

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