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AEC Research and Development Report

TECHNICAL ACTIVITIES
QUARTERLY REPORT

AEC REACTOR DEVELOPMENT
AND TECHNOLOGY PROGRAMS

APRIL, MAY, JUNE 1971

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AND TECHNOLOGY PROGRAMS
APRIL, MAY, JUNE 1971

By

Staff of Battelle-Northwest

Program Coordinator:

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October 1971

BATTELLE
PACIFIC NORTHWEST LABORATORIES
RICHLAND, WASHINGTON 99352

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FOREWORD

This report covers all programs except the Waste Studies Program conducted at the Pacific Northwest Laboratory for the U.S. Atomic Energy Commission's Reactor Development and Technology Division. The objective of the Technical Activities Quarterly Report is to inform the scientific community in a timely manner of the technical progress made on the programs. The report contains brief technical discussions of accomplishments in all areas where significant progress has been made during the reporting period. The results presented should be considered preliminary and do not constitute final publication of the work. A list of publications and papers is given in the report. Anyone wishing to obtain additional information on the work presented is encouraged to contact the author directly.

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1. SUMMARY

DEFECT TESTING

Postirradiation examination results show that comparable amounts of fuel melting occurred in defected and nondefected mixed-oxide fuel rods operating at a peak linear heat rating of about 23 kW/ft.

REACTIVITY MEASUREMENTS IN A MOCKUP MSBR LATTICE IN THE HTLTR

The experimental measurements on the MSBR mockup lattice have been completed in the HTLTR. Reactivity measurements on the various samples were made at 20, 300, 627, and 1000°C.

MEASUREMENT OF NEUTRONIC PARAMETERS FOR A $\text{ThO}_2\text{-PuO}_2$ HTGR LATTICE IN THE HTLTR

Experiments with a $\text{ThO}_2\text{-PuO}_2$ HTGR lattice have begun in the HTLTR. Criticality was achieved by the addition of HTLTR driver fuel in the graphite stack surrounding the test lattice. Several foil irradiations have been made to determine the amount of mismatch in the neutron spectrum and to determine relative reaction rates for all lattice constituents at room temperature. The infinite-medium neutron multiplication factor for the $\text{ThO}_2\text{-PuO}_2$ HTGR lattice will be measured from room temperature up to 1000°C.

STEADY STATE AND TRANSIENT SUBCHANNEL CODE DEVELOPMENT AND DATA ANALYSIS

An interim report on COBRA-III was issued and a complete card deck and sample problem were sent to 20 selected AEC laboratories and contractors. COBRA-III has the following features:

- It contains all the analysis capability of the COBRA-II program.
- It considers transients up to, but not including, sonic velocity propagation effects.
- The numerical scheme performs a boundary value solution to more properly include the effects of crossflow resistance.
- The numerical solution is stable for all time steps.

Additional effort is being directed toward developing a forced cross-flow model for wire-wrapped nuclear fuel bundles.

Estimates of turbulence intensity and scale in rod bundle subchannels are being obtained by autocorrelation of the turbulent velocity signals. A two-component laser-Doppler velocimeter is being used. The experimental apparatus is complete and in operation. Experimental measurements have been completed for two pairs of interconnected subchannels.

THE EFFECT OF SOLUBLE NUCLEAR POISONS ON CRITICALITY

Analysis and correlation of data from criticality experiments on $\text{Pu}(\text{NO}_3)_4$ solutions containing gadolinium were completed. Agreement between theory and experiment was reasonably good. Further calculations were made to determine the effect of gadolinium on k_∞ of plutonium nitrate solution over the 100 to 300 g Pu/liter concentration range.

NUCLEAR GRAPHITE

Two compressive creep capsules operating at 550 and 800°C, respectively, and a tensile creep capsule operating at 800°C were shipped to Idaho Falls in June for charging into the ETR reactor. The first two are scheduled for a three-cycle irradiation and the latter for a 1-1/2 cycle irradiation.

The radiolysis of helium-diluted mixtures of $\text{CO} + \text{H}_2\text{O}$ has been kinetically monitored for equimolar concentrations at 940, 65, and 10 VPM. The results suggest a large reduction in activation energy, even to negative values with increasing dilution. Possible complication is noted due to the presence of approximately 2 VPM O_2 in the gas stream.

Samples of various sizes recently discharged from the KE-reactor continue to show about the same effect due to sample size as observed and reported earlier. That is, the largest samples have contracted about 20% more than the smallest samples.

Length-change rates of cold-seeded samples are, in some case, different than for unseeded samples by as much as a factor of two. For other

irradiation conditions, there is little or no effect. Significant differences were observed under irradiation conditions of interest to MSR's and HTGR's.

Although inconclusive, it appears that the observed "size-effect" may be largely accounted for by a combined dependence upon sample size per se and cold seeding.

Measurements have been completed on the high-temperature irradiation samples; and the samples will begin the next irradiation cycle at the end of August 1971.

2. DEFECT TESTING

M. D. Freshley

The postirradiation examination of four rods irradiated in the ETR for the purpose of investigating the performance of intentionally defected and nondefected Zircaloy-2 clad mixed-oxide fuels was completed. These rods, which contained UO_2 -4 wt% PuO_2 cold-press-sinter pellet fuel, were irradiated as part of a 12-rod cluster assembly in the ETR M-3 Loop facility. The operating conditions in the loop and calculated temperatures that existed in the different fuel types are summarized in Table 2.1. Specific information related to the four rods examined is summarized in Table 2.2.

TABLE 2.1. Operating Conditions for the Defect Test Assembly in the ETR M-3 Loop

Peak Linear Rod Power Generation	~21.6 kW/ft
Average Linear Rod Power Generation	~17 kW/ft
Peak Surface Heat Flux	513,000 Btu/hr-ft ²
Average Surface Heat Flux	396,000 Btu/hr-ft ²
Calculated Total Power Generation	410 kW
Heat Transfer Film Coefficient	5169 Btu/hr-ft ² -°F
Loop Flow Rate	150 gal/min
Loop Pressure	2000 psi
Coolant pH	9.8 to 10.2
Coolant Oxygen Concentration	<0.05 ppm
Saturation Temperature of Loop	635°F (335°C)
Coolant Velocity Over the Fuel	11.85 ft/sec
Inlet Water Temperature	480°F (244°C)
Outlet Water Temperature	502°F (271°C)
Maximum Cladding Surface Temperature	317°C
Maximum Cladding Temperature	410°C
Maximum Fuel Surface Temperature	630°C
Maximum Fuel Temperature:	
Solid Pellet, 0.006 in. gap	>melting
Solid Pellet, 0.010 in. gap	>melting
Cored Pellet, 0.006 in. gap	~2400°C

TABLE 2.2. Power History for the Four Mixed-Oxide Rods Examined That Were Irradiated in the Defect Experiment in ETR

Rod No.	Type	Diametral Gap-in.	Estimated Peak burnup, Mwd/MTM	Estimated Max. Peak Power, kW/ft			Comments
				Cycle I 35 EFPD*	Cycle II 17.3 EFPD	Cycle III 14.1 EFPD	
F-3	Solid cold-press-sinter pellet	0.010	3630**	21.6	20.3	12.2	Ceramographically confirmed fuel melting
F-10	Solid cold-press-sinter pellet	0.006	1320	13.4	--	--	Probably no fuel melting
F-11	Solid cold-press-sinter pellet	0.006	1970	20	--	--	Fuel melting probably occurred
F-12***	Solid cold-press-sinter pellet	0.010	3380	20	12.5	20.2	Ceramographically confirmed fuel melting

*EFPD = Effective full power days

**¹⁴⁸Nd burnup value

***Operated defected with a 0.020-in. hole during Cycles II and III.

As reported previously, ^(1,2) rod F-12 was intentionally defected by drilling a 0.020-inch diameter hole through the cladding in the peak power region of the rod after irradiation during Cycle I at a peak linear heat rating of 20 kW/ft to a peak burnup of 1200 Mwd/MTM. The defected rod was subsequently irradiated for 17.3 EFPD (Cycle II) at a peak linear heat rating of about 12.5 kW/ft followed by irradiation for 14.1 EFPD (Cycle III) at a peak linear heat rating of about 20.2 kW/ft.

During irradiation of the defected rod at the lower heat rating, gross gamma activity levels monitored in the loop piping system increased in proportion with power with no activity bursts, but remained well within acceptable limits. At the higher heat rating, activity bursts occurred that were associated with changes in reactor power level, some of which were severe.

Axial gross gamma scans show a smooth fission product activity distribution for rod F-10 in which no fuel melting occurred. However, localized variations in fission product activity occurred in rods F-11 and F-12 which were irradiated under conditions that produced fuel melting. These axial discontinuities in fission product activity are caused by axial fuel movement associated with melting. Rod F-3 was not gamma scanned.

Although there was no evidence of clad ridging at pellet interfaces in any of the rods examined, pre- and postirradiation diameter profile measurements show that the cladding plastically deformed during irradiation to cause a maximum reduction in the average rod diameter of about 0.002 inch and an increase in maximum rod ovality of as much as 0.007 inch (Table 2.3). The different as-assembled diametral gaps of 0.006 and 0.010 inch did not have an apparent effect on ovality. However, rod F-10, which did not undergo fuel melting, had slightly less ovality than the other rods which operated with moderate fuel melting.

TABLE 2.3. Dimensional Data Obtained from Four Mixed-Oxide Rods Irradiated in the Defect Experiment in ETR

Rod No.*	Pre-irradiation			Post-irradiation		
	Max. Diameter, in.	Min. Diameter, in.	Max. Ovality, in.	Max. Diameter, in.	Min. Diameter, in.	Max. Ovality, in.
F-3	0.5637	0.5626	0.0013	0.5633	0.5624	0.006
F-10	0.5628	0.5618	0.002	0.5618	0.5603	0.005
F-11	0.5638	0.5628	0.002	0.5623	0.5610	0.010
F-12	0.5643	0.5632	0.0015	0.5630	0.5618	0.007

*Each rod contains UO_2 -4 wt% PuO_2 pellet fuel with an active fuel length of 26.8-inches; 0.40-in. long depleted UO_2 pellets on each end of the fuel column; a 3.0-in. long plenum on the top; and 0.030-in. thick Zircaloy-2 cladding.

The cladding surface in the immediate region of the defect had a white discoloration that trailed off in the downstream direction (Figure 2.1). There were no measurable localized changes in the dimensions of the rod in the defect region.

Examination of the UO_2 - PuO_2 fuel at the defect position in rod F-12 shows that an insignificant amount of fuel washout occurred during irradiation with fuel melting extending to about 23% of the radius at the time of shutdown (Figure 2.2). This amount of melting is consistent with the peak linear heat rating of 20.2 to 21.8 kW/ft. A center void extends to about 12% of the radius at the plane of the defect. An unusual crack pattern that developed in the fuel was probably caused by the leaching action of the steam present in the region of the defect.

F-7952

FIGURE 2.1.

Postirradiation Appearance of the
0.020-inch Diameter Intentional
Defect in $\text{UO}_2\text{-PuO}_2$ Rod F-12

—
2000 μm

Examination of the $\text{UO}_2\text{-PuO}_2$ fuel approximately six inches above the defect in rod F-12 shows that the fuel has oxidized to form U_4O_9 Widmanstätten type precipitates in the microstructure (Figure 2.3). A higher U_4O_9 concentration occurred near the thermal center of the specimen as evidenced by the formation of a continuous U_4O_9 network.

Examination of the cladding in the defected rod shows that a thick, nonuniform ZrO_2 layer formed on the surface of the drilled defect hole (Figure 2.4). A corrugated ZrO_2 layer about 10 μm thick formed on the inner cladding surface in the defect region (Figure 2.5). There was also intermittent reaction between the oxidized $\text{UO}_2\text{-PuO}_2$ fuel and the ZrO_2 layer on the inner cladding surface.

Although there were no indications of massive hydride concentrations in the cladding of the defected rod, higher-than-normal hydride concentrations

FIGURE 2.2. Transverse Section Through the Defected Rod (F-12) Which Contains UO_2 -4 wt% PuO_2 , Cold-Press-Sinter Pellet Fuel. Fuel melting extends to 23% of the radius.

20 μm

(c)

F-8154

(b)

F-8156

(a)

F-8155

FIGURE 2.3. The UO_2 -4 wt% PuO_2 Pellet Fuel Oxidized in the Defected Rod (F-12) to Form U_4O_9 Widmanstatton-type Precipitates in the Equiaxed (a) and Columnar (b) Grain Growth Regions of the Fuel. A continuous U_4O_9 network (c) formed near the thermal center of the rod.

F-8141

FIGURE 2.4. A Thick ZrO_2 Layer Formed on the Surface of the Defect and the Inner Cladding Surface in the Defected Rod (F-12)

F-8148

50 μm

FIGURE 2.5. A Thick Corrugated ZrO_2 Layer Formed on the Inner Cladding Surface in the Defected Rod (F-12). Intermittent reaction between the $\text{UO}_2\text{-PuO}_2$ fuel and the ZrO_2 layer occurred.

which migrated down the thermal gradient to the outer cladding surface did occur (Figure 2.6). Concentration variations around the circumference of the specimen were noted.

Examination of the companion nondefected $\text{UO}_2\text{-PuO}_2$ rod F-3 irradiated during the same experiment showed that fuel melting occurred to about 30% of the radius (Figure 2.7) which is comparable to 23% of the radius at the plane of the defect in the defected rod. In contrast with the defected rod, examination of the nondefected rod showed, (1) that a typical crack pattern developed in the fuel, (2) no evidence of U_4O_9 precipitates, (3) a very thin ZrO_2 layer on the inner cladding surface, and (4) no reaction between the $\text{UO}_2\text{-PuO}_2$ fuel and the ZrO_2 layer on the inner cladding surface. However, a hydrogen concentration and distribution similar to that which occurred in the cladding on the defected rod also occurred in the nondefected rod. The reason for the higher-than-normal hydrogen concentration is not apparent.

F-8186

FIGURE 2.6. Increased Zirconium-Hydride Concentration on the Outer Cladding Surface in the Region of the Defect (F-12)

FIGURE 2.7. Transverse Section Through Nondefected Rod F-3 at the Same Plane of the Defect in Rod F-12. Fuel melting extends to about 30% of the radius.

Essentially the same amount of fuel melting occurred in both the defected (F-12, Figure 2.2) and the companion nondefected (F-3, Figure 2.7) rods. Because both rods were irradiated at comparable peak power conditions (Table 2.2), this observation suggests that the oxidation of the fuel in the defected rod had an insignificant effect on the melting heat rating. Based upon data reported in the literature,⁽³⁾ this is somewhat unexpected.

Also, fuel melting probably occurred in the nondefected rod F-3 while undergoing irradiation during Cycles I and II at a peak power of about 22 kW/ft (Table 2.2). Apparently peak fuel temperatures (i.e., <1700°C) during subsequent irradiation of the nondefected rod at a peak linear heat rating of about 15 kW/ft during Cycle III were not sufficient to allow time-temperature-dependent solid-state diffusion phenomena to eradicate the once-molten fuel structure that formed during prior irradiation.

Reference

1. Plutonium Utilization Program Technical Activities Quarterly Report - March, April, May 1970, BNWL-1442, Battelle-Northwest, June 1970.
2. Technical Activities Quarterly Report - AEC Reactor Development and Technology Programs - July, August, September 1970, BNWL-1522-1, Battelle-Northwest, October 1970.
3. J. A. Christensen, Stoichiometry Effects in Oxide Nuclear Fuels - 1. Power Rating Required for Melting and Oxygen Redistribution in Molten Center UO_{2+x} Fuels, BNWL-536, Battelle-Northwest, December 1967.

3. HIGH TEMPERATURE REACTOR PHYSICS

REACTIVITY MEASUREMENTS IN A MOCKUP MSBR LATTICE IN THE HTLTR

E. P. Lippincott

Experimental measurements on the MSBR mockup lattice have been completed in the HTLTR. The design of this lattice and room temperature measurements have previously been reported.^(1,2) The results are currently being analyzed for comparison with theoretical calculations.

The reactivity measurements on the various central cells, copper, and hastelloy, are summarized in Table 3.1. The first line gives the correction for the reactivity of the N_2 in the void when the central cell is removed. The reactivity worth of each 7-1/2 inch $\times$ 7-1/2 inch $\times$ 24-inch block has been corrected by this amount. No correction for end caps on the central cell is included in Table 3.1. The reactivity worth of copper and graphite are given in ϵ /gram. The copper worth was measured by inserting two, 0.010-inch copper foils on the graphite carrier. The worth of the hastelloy block (graphite) was measured empty and with one hastelloy piece at each temperature, and with additional hastelloy pieces at room temperature and 627°C.

Table 3.2 gives the material worth measurements made by inserting samples in a tube in the center of the cruciform block. Each reactivity measurement is relative to the cruciform block with solid graphite in the center. The worth of these materials corrected for graphite and N_2 is given in Table 3.3. The first column gives the weight of the material in the sample. The graphite worth includes the N_2 in the pores of the graphite. The $^{233}\text{UO}_2\text{-ThO}_2$ refers to the coated particles with one part $^{233}\text{UO}_2$ to three parts ThO_2 . The ^{235}U and $^{239}\text{PuO}_2\text{-ZrO}_2$ are also contained in coated particles. The fuel mix refers to a sample of the $^{233}\text{UO}_2$, ThO_2 , and C mixture loaded into the fuel channels in the central cell.

The material worths in ϵ /kg derived from Table 3.3 are given in Table 3.4. The estimated errors in the final numbers are also presented. By using measured or estimated worths for other elements present, values

TABLE 3.1. Reactivity Measurements (Corrected for N₂ in void)

	20°C	300°C	627°C	1000°C
N ₂ Correction	-0.98¢	-0.49¢	-0.31¢	-0.21¢
Standard Central Cell	3.28¢	1.91¢	1.39¢	1.24¢
Copper	0.0173¢/g	0.0174¢/g	0.0170¢/g	0.0169¢/g
Graphite Carrier	0.39×10 ⁻³ ¢/g	0.47×10 ⁻³ ¢/g	0.50×10 ⁻³ ¢/g	0.56×10 ⁻³ ¢/g
V ₂ O ₃ Cell	22.05¢	21.79¢	22.12¢	22.68¢
Increased Density Cell	4.75¢	3.02¢	2.17¢	1.92¢
4-hole Cell (1-4/16 diameter)	6.84¢	5.68¢	5.32¢	5.38¢
4-hole Cell (2-37/64 diameter)	0.62¢	-3.94¢	-7.48¢	-9.68¢
Single Hole Cell (4 inches)	8.04¢	6.06¢	4.90¢	4.38¢
Single Hole Cell (6 inches)	-2.50¢	-7.83¢	-11.90¢	-14.60¢
Cruciform Block	4.07¢	2.33¢	1.28¢	0.93¢
Graphite Block	12.35¢	14.46¢	16.74¢	18.54¢
Empty Hastelloy Block	11.61¢	13.73¢	15.66¢	17.32¢
Block + One Hastelloy Piece	-24.63¢	-22.74¢	-21.00¢	-19.13¢
Block + Two Hastelloy Pieces				
Adjacent Positions	-53.06¢		-50.31¢	
Opposite Positions	-54.65¢			
Block + Three Hastelloy Pieces	-76.8¢			
Block + Four Hastelloy Pieces	-95.8¢		-99.2¢	

TABLE 3.2. Material Worth Measurements (cents) (uncorrected)

	20°C	300°C	627°C	1000°C
Solid Graphite (Standard)	0	0	0	0
Graphite Powder	-0.29			
Void	-0.75	-0.72	-0.82	-0.91
Copper Foil	-3.53	-3.38	-3.27	-3.25
Beryllium Rod	+0.96	+1.20	+1.35	
Teflon Rod (CF ₂)	-0.59	-0.54		
⁷ LiF	-2.04	-1.91	-1.92	
ThO ₂	-3.80	-3.87	-3.89	-3.89
²³³ UO ₂ - ThO ₂	+4.72	+4.35	+4.07	+3.84
²³⁵ UC ₂	+4.36	+3.98	+3.73	+3.30
²³⁹ PuO ₂ - ZrO ₂	+4.04	+4.34	+5.05	+5.40
Hastelloy Foil	-3.83	-3.78	-3.60	-3.57
Fuel Mix	-1.75	-2.54	-3.30	-3.88

TABLE 3.3. Corrected Material Worth Measurements
(cents) (relative to void)

	<u>Weight, grams</u>	<u>20°C</u>	<u>300°C</u>	<u>627°C</u>	<u>1000°C</u>
Graphite (N ₂ atmosphere)	1844.6	+ .70	+ .69	+ .80	+ .90
Copper	197.1	-3.52	-3.37	-3.26	-3.24
Beryllium	2011.6	+1.69	+1.93	+2.19	- - -
Teflon (CF ₂)	2117.6	+0.14	+0.18	- - -	- - -
⁷ LiF	1311.4	-1.72	-1.59	-1.55	- - -
ThO ₂	250.0	-3.51	-3.59	-3.57	-3.53
²³³ UO ₂ - ThO ₂	33.75	+4.98	+4.60	+4.35	+4.15
²³⁵ U (93%)	11.36	+4.63	+4.24	+4.02	+3.62
²³⁹ Pu - ZrO ₂	75.68	+4.28	+4.57	+5.32	+5.70
Hastelloy	186.4	-3.81	-3.76	-3.58	-3.53
Fuel Mix	2898	-0.96	-1.79	-2.44	-2.91

TABLE 3.4. Material Worths (¢/kg)

	<u>20°C</u>	<u>300°C</u>	<u>627°C</u>	<u>1000°C</u>
Graphite (N ₂ atmosphere)	0.38±0.01	0.37±0.02	0.43±0.02	0.49±0.04
¹² C	0.39±0.01	0.38±0.02	0.44±0.02	0.49±0.04
F	- .03±0.02	-0.01±0.04	- - -	- - -
⁷ Li (78 ppm ⁶ Li)	-4.75±0.05	-4.47±0.11	-4.35±0.11	- - -
⁷ Li (estimated)	-1.84±0.18	-1.58±0.19	-1.55±0.19	- - -
Be	+0.84±0.01	+0.96±0.02	+1.09±.02	- - -
Cu	-17.9±0.1	-17.1±0.2	-16.5±0.2	-16.4±0.3
Hastelloy	-20.4±0.1	-20.2±0.2	-19.2±0.2	-18.9±0.4
ThO ₂	-14.0±0.1	-14.4±0.2	-14.3±0.2	-14.1±0.3
Th	-15.9±0.1	-16.3±0.2	-16.2±0.2	-16.0±0.3
²³³ UO ₂ - ThO ₂ (particles)	147.6±1.0	136.3±1.6	128.9±1.5	123.0±2.9
²³³ U	711±10	661±12	628±11	600±17
²³⁵ U	438±4	401±5	381±5	343±9
²³⁹ PuO ₂ - ZrO ₂	56.6±0.5	60.4±0.7	70.3±0.9	75.3±1.8
Pu (98% ²³⁹ Pu)	919±13	980±18	1140±20	1215±32
Fuel Mix	-0.33±0.01	-0.62±0.02	-0.84±0.02	-1.00±.03

for worths of individual elements are derived. The worth of the actual lithium used is presented together with an estimated worth for pure ^7Li . The worths derived for ^{233}U and Pu (8% ^{239}Pu) and the ^{235}U worth are presented with no correction for flux depression in the particles or in the fuel tube. Such corrections will be necessary before these numbers could be used to estimate worths of small amounts of these materials in a homogeneous geometry.

The MSBR experiment was the first to utilize the new sample changer. This device allows samples to be preheated before insertion in the reactor and thus enables temperature transients during sample changes to be minimized. This results in a time saving during reactivity measurements and an improvement in accuracy. Errors have not been included in Table 3.1, but typical errors might be $\pm 0.02\%$ at room temperature, $\pm 0.04\%$ at 300°C and 627°C , and $\pm 0.07\%$ at 1000°C . Since samples could not be preheated to 1000°C , some temperature transients remained at this temperature and the accuracy is therefore lower. These errors are estimated from the consistency of the data and represent approximately one standard deviation. A second advantage of the sample changer is the elimination of the extension effect which created an uncertainty in the earlier experiments.

References

1. E. C. Davis and E. P. Lippincott, "Design of MSBR Experiment in the HTLTR," Technical Activities Quarterly Report, AEC Reactor Development and Technology Programs, October 1970-March 1971, BNWL-1522-2, Battelle-Northwest, June 1971.
2. E. P. Lippincott, "Room Temperature Measurements on a MSBR Lattice in the HTLTR," Technical Activities Quarterly Report, AEC Reactor Development and Technology Programs, October 1970-March 1971, BNWL-1522-2, Battelle-Northwest, June 1971.

MEASUREMENT OF NEUTRONIC PARAMETERS FOR A $\text{ThO}_2\text{-PuO}_2$ HTGR LATTICE IN THE HTLTR

D. F. Newman

A High Temperature Gas-cooled Reactor (HTGR) lattice which has a $\text{ThO}_2\text{-PuO}_2$ fuel blend was inserted in the central test region of the High Temperature Lattice Test Reactor (HTLTR). Criticality was achieved by the addition of HTLTR driver fuel in the graphite stack surround the test

lattice. Several irradiations have been made at room temperature, with different foil materials loaded in the ThO_2 - PuO_2 HTGR lattice, to determine the amount of mismatch in the neutron spectrum and to determine relative reaction rates for all lattice constituents.

The HTGR fuel blocks are the same graphite blocks used for the previous three HTGR lattice experiments in the HTLTR. In this experiment, the fuel channels in the blocks were filled with a mixture of graphite particles, ThO_2 particles, and PuO_2 -pyrocarbon coated microspheres.⁽¹⁾ The kernels of the microspheres are 209 micron (average) diameter PuO_2 (11.2 g/cm³ density) with 73% ^{239}Pu , 23% ^{240}Pu , 3% ^{241}Pu , and 1% ^{242}Pu . Each PuO_2 kernel is clad with a 91 micron (average) thick pyrolytic carbon coating (1.8 g/cm³ density). The coated microspheres are free of surface contamination. A 37 gram sample of PuO_2 -pyrocarbon coated microspheres has been continuously monitored for alpha contamination, while maintained at 1000°C in an oven for more than 2 months; no contamination was detected.

The completed blocks have a C/Th atom ratio of 250 and C/Pu ratio of 7500. Fuel specifications for the loaded HTGR blocks are listed in Table 3.5. Experiments will be carried out at the following nominal equilibrium temperatures (similar to those on the previous three HTGR lattices): 25, 150, 300, 500, 750, and 1000°C. Measurements will include reactivity of central cell, copper poison, and special sample blocks. A block with V_2O_3 in place of ThO_2 will be used to determine the thorium Doppler coefficient. The infinite medium neutron multiplication factor, k_∞ , for the ThO_2 - PuO_2 HTGR lattice will be determined as a function of temperature, from room temperature up to 1000°C. This will provide valuable benchmark data to which calculational predictions can be compared.

The prediction of neutronic characteristics of plutonium-fueled HTGR lattices requires more sophisticated calculational methods than are needed to handle HTGR lattices fueled with either ^{235}U or ^{233}U . The source of additional difficulties is mainly due to the low energy resonances of ^{239}Pu and ^{240}Pu . The net reactivity effect of granular PuO_2 particle self shielding is a function of both the ^{240}Pu content in the plutonium and the particle diameter.

TABLE 3.5. PuO_2 - ThO_2 HTGR Fuel Specifications

PuO_2 Particle Diameter:	209 μ (average) (± 20 μ Std. Deviation)
C/Pu Atom Ratio:	7500
C/Th Atom Ratio:	250
Pu Isotopic Ratio 239: 240: 241: 242 =	72.8: 22.9: 3.3: 1.0
Radius of Fuel Rod =	0.5969 cm
Effective Cell Radius =	1.07478 cm
Volume Fuel in 25 Rod Block =	1634.76 cm ³
Volume PuO_2 in 25 Rod Block =	2.015 cm ³
Volume Pyrocarbon Coated PuO_2 Microspheres in 25 Rod Block =	14.023 cm ³
Volume ThO_2 in 25 Rod Block =	67.044 cm ³
Mass PuO_2 in 25 Rod Block =	22.51 g
Mass Pyrocarbon Coated PuO_2 Microspheres in 25 Rod Block =	44.14 g
Mass Pu in 25 Rod Block =	19.85 g
Mass ThO_2 in 25 Rod Block =	657.03 g
Mass Th in 25 Rod Block =	577.4 g
Mass Graphite Particles in 25 Rod Block =	1535.1 g
Mass Pyrocarbon in 25 Rod Block =	21.6 g
Mass Graphite Block (excluding end caps) =	5915.5 g
Total Mass of Carbon in Loaded Block (excluding end caps) =	7472.2 g
Fuel Block Dimensions (excluding end caps) =	9.525 $\times$ 9.525 $\times$ 58.42 cm
Mass Graphite End Caps for 25 Rod Block =	372.8 g
Fuel Block Length (including end caps) =	60.96 cm

The present calculational method at BNW uses EGGNIT⁽²⁾ (0.683 eV $\rightarrow$ 10 MeV) to compute neutron capture in the ^{240}Pu resonance considering only neutrons slowing down. At high temperatures, neutrons can scatter up into the 1.06 eV ^{240}Pu resonance as well as slow down into it. Calculations performed by Gulf General Atomic⁽³⁾ using the GTF code (up to 2.38 eV) show that nearly 15% of the neutrons absorbed in the 1.06 eV resonance will have been scattered up from lower energies, when the HTGR moderator temperature is about 1050°K (777°C). A modified version of GRANIT⁽⁴⁾ that is being produced will use the 35 group energy mesh from the LASER version of THERMOS (0 $\rightarrow$ 1.855 eV). Doppler-broadened cross sections for ^{240}Pu will be used by the modified version of GRANIT. The 1.855 eV upper energy for the modified version of GRANIT is identical to the lower energy for group 62 in the EGGNIT epithermal group structure. No modifications will be required for EGGNIT.

References

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2. C. R. Richey, EGGNIT: A Multigroup Cross Section Code, BNWL-1203, Battelle-Northwest, November 1969.
3. J. R. Brown, Private Communication, Gulf General Atomic, Inc., San Diego, California, July 14, 1969.
4. C. L. Bennett, "GRANIT: A Code for Calculating Position Dependent Thermal Neutron Spectra in Doubly Heterogeneous Systems by the Integral Transport Method," Technical Activities Quarterly Report, AEC Reactor Development and Technology Programs, July, August, September 1970, BNWL-1522-1, p. 2.48, Battelle-Northwest, October 1970.

4. STEADY STATE AND TRANSIENT SUBCHANNEL CODE DEVELOPMENT AND DATA ANALYSIS

D. S. Rowe and B. M. Johnson

The objective of this program is to develop improved methods for analyzing heat transfer and fluid flow in rod bundle nuclear elements during both steady state and transient conditions. The program includes both analytical studies for the continued development of computer programs and experimental studies for the verification and implementation of these computer programs.

COBRA-III COMPUTER PROGRAM

The objective of this portion of the program is to develop a mathematical model and computer program for predicting the steady-state and transient thermal-hydraulic performance of rod bundle nuclear fuel elements.

An interim report (BNWL-B-82) was issued to describe progress on the COBRA-III computer program development. A complete card deck and sample problem was also distributed to 20 selected AEC laboratories and contractors, including the Argonne Code Center.

The interim version of the COBRA-III program uses an approach that is very similar to COBRA-II. The bundle flow cross section is divided into flow subchannels that are assumed to contain one-dimensional flow and are coupled to each other by turbulent and diversion crossflow mixing. Both types of crossflow mixing carry mass, energy, and momentum between the subchannels. Boiling and nonboiling conditions are considered by using an equation of state that gives density as a function of enthalpy, flow, heat flux, pressure, position, and time. The model neglects sonic velocity propagation; therefore, it is limited to transients with time duration greater than the time for a sonic wave to pass through the channel. A simplified transverse momentum equation that neglects temporal and spatial acceleration is used to simplify the numerical solution. This gives instantaneous response of the diversion crossflow to changes in subchannel pressure gradient. The equations of the mathematical model are solved by

using a semiexplicit finite difference scheme that is stable for all time steps. This scheme also gives a boundary value flow solution for both steady state and transients where the boundary conditions are the inlet enthalpy, inlet flow rate, and exit pressure. Several successful computer solutions have been run using the above mathematical model and numerical solution. This has included rapid changes in heat flux, inlet flow rate, inlet enthalpy and system pressure. The ability of the boundary value solution to include high crossflow resistance has also been successfully demonstrated. High crossflow resistance has the effect of causing a downstream flow disturbance to be felt upstream. This effect cannot be considered by COBRA-II and other similar computer programs that use an initial value flow solution.

The features of COBRA-III can be summarized as follows:

- It contains all the analysis capability of the COBRA-II program.
- It can consider transients up to, but not including, sonic velocity propagation effects.
- The numerical scheme performs a boundary value solution to more properly include the effects of crossflow resistance.
- The numerical solution is stable for all time steps.

Development of COBRA-III is continuing to improve its capability. A more complete transverse momentum equation has been formulated and incorporated into the COBRA-III program. This equation now includes the temporal and axial component of transverse acceleration. These additional terms show that the diversion crossflow is a convective term and is transported at the coolant velocity. This means that diversion crossflows tend to persist in a given direction unless changed by frictional losses or pressure forces. This is an important difference from the earlier transverse momentum equation where crossflow was governed by pressure and frictional forces only. The use of this new equation also introduces a new parameter that must be determined experimentally. This parameter is the effective ratio of gap spacing to a transverse length for momentum. The new equation also gives more importance to defining the crossflow resistance with greater accuracy since this determines the rate of decay of diversion crossflow. In COBRA-II,

the actual value of crossflow resistance had little effect on the sub-channel flow and enthalpy solutions. Both of the above parameters are presently being evaluated to define their relative importance to the solutions. Solutions using the new momentum equation are performed numerically. A finite difference scheme has been developed and successfully tested on trial problems. The additional momentum terms add considerable stability to the solution.

Additional areas of development are continuing to improve the capability of the COBRA program. Major emphasis is presently being directed toward developing a forced crossflow mixing model for wire wrapped nuclear fuel bundles. A previously developed, but unpublished, wire wrap mixing model for crossflow flow sweeping in COBRA-II is being incorporated into the modified COBRA-III with the improved transverse momentum equation. This model has been successfully applied to forced flow mixing in a 7-pin bundle. Bundles of larger size will be considered to evaluate ability of the mixing model and numerical solution in COBRA-III to describe mixing in large wire wrapped bundles.

Work is also being started to develop reverse flow capability for COBRA-III. This is needed for transient problems where coolant may be expelled from both ends of the bundle. This capability is also required for developing a pressure drop boundary condition.

EXPERIMENTAL STUDY OF FLOW STRUCTURE IN ROD BUNDLES

The objective of this experimental study is to obtain a more complete description of the turbulent flow structure of single-phase flow in rod bundles so that better mixing correlations can be developed for use in subchannel computer programs. Present mixing correlations do not adequately account for the effect of subchannel geometry.

This experimental study is investigating the turbulent flow structure in rod bundle geometry to define the effect of subchannel shape, gap spacing and subchannel interconnections. The axial and transverse fluctuating velocity components in the gap and within the subchannels are the primary measurements being made in these experiments. Estimates of turbulence

intensity and scale in rod bundle subchannels are being obtained by autocorrelation of the turbulent velocity signals. A two-component laser-Doppler velocimeter is being used. This device can measure fluid velocities at a point in a flow stream without inserting a probe into the flow. It operates from the principle that light scattered from very small particles moving with the fluid experiences a frequency shift that is related to the fluid velocity. By using a system of optics and electronics, this Doppler shift can be accurately measured to obtain the instantaneous fluid velocity at the measurement point. This measurement technique is enabling measurements of velocities that have never been obtained in rod bundle geometries.

The experimental apparatus is complete and in operation. A common flow channel is being used to accommodate a variety of subchannel sizes and shapes. This is accomplished by using a flow channel that consists of two common side plates containing glass windows for entrance and exit of the laser beam. Inserts of various thicknesses are placed between the side plates to create rectangular flow channels. Rod bundle subchannels are formed within this channel by inserting rods, or sectors, to form the desired subchannel shapes. Rods with glass windows are being used where appropriate to enable passage of the laser beam. A flow straightening and calming section is included to control the level of upstream turbulence.

Initially, the experiments are concerned with selected pairs of interconnected subchannels to determine how various subchannel shapes affect the transverse velocity fluctuations that are responsible for mixing. Similar measurements will then be performed in a 9-rod bundle to determine how the turbulent flow structure in multiply connected subchannels differs from the structure in simply connected subchannels. This comparison is expected to help determine the minimum size required for rod bundle turbulent diffusion experiments.

Experimental measurements have been completed for two pairs of interconnected subchannels. These are "triangular-wall" and "square-square" at a pitch to diameter ratio of 1.25. Two flows and two temperatures were considered giving Reynolds numbers of 50,000, 100,000 and 200,000. These

measurements show that the distribution of the axial (u'/u_o) and transverse (v'/u_o) components of turbulence intensity across the gap are nearly identical for these two subchannel pair. The axial component of turbulence intensity is about 50% larger than the transverse component and has an average of about 0.06. Both the distribution and magnitude of turbulence intensity is similar to that for fully developed pipe flow. For increasing Reynolds number the turbulence intensity decreases slightly. This result is consistent with the weak Reynolds number dependence observed in rod bundle mixing experiments.

5. THE EFFECT OF SOLUBLE NUCLEAR POISONS ON CRITICALITY

R. C. Lloyd and L. E. Hansen

Experimental criticality data are needed on systems containing soluble poisons in order to accurately determine how effective these poisons would be in possible criticality control use in the storage and processing of solutions bearing fissile material. Accordingly, criticality experiments were performed utilizing gadolinium as a soluble poison in plutonium nitrate solutions. The gadolinium was chosen because of its high neutron cross section, high solubility, and compatibility in the separation process. The experiments were oriented toward providing (1) data on the effectiveness of gadolinium as a soluble absorber for nuclear criticality prevention, and (2) experimental benchmarks for checking calculational techniques and cross-section sets.

Two series of criticality experiments with plutonium nitrate solution homogeneously poisoned with gadolinium were performed. Plutonium concentrations of 116 and 363 g Pu/liter with acid molarities of 1.85 and 4.1, respectively, were used. The ^{240}Pu content was 8.3 wt%. The experimental vessel used in these experiments was a 61.03 cm diameter, stainless steel cylinder, having a wall thickness of 0.079 cm. A second vessel enclosed the criticality vessel to provide for a water reflector.

A preliminary analysis of the data from these experiments was reported previously.⁽¹⁾ The analysis has now been completed. For completeness, the critical values and calculated multiplication factors for all of the experimental assemblies are given in Table 5.1. The 18-group cross-section data used in these calculations were obtained using the GAMTEC-II code. Gadolinium cross-section data were obtained from the ENDF/B-II library. The KENO Monte Carlo code was used to calculate the multiplication factors for these critical systems. From the table, it can be seen that the k_{eff} values fall within a 2% range and are on the nonconservative side. This variation should be considered in applications of similarly calculated extensions of the experimental data.

TABLE 5.1. Criticality of Gd_2O_3 -Poisoned Plutonium Solutions^(a)

Concentration		Critical Values			GAMTEC-KENO Computed k_{eff}
Plutonium, g Pu/l	Gadolinium, g Gd/l	Height, cm	Volume, l	Mass, kg Pu	
116	0.00	15.44	45.17	5.240	0.993
116	0.48	20.16	58.98	6.842	0.980
116	0.96	25.98	75.99	8.815	0.980
116	1.42	34.48	100.85	11.698	0.978
116	1.92	48.28	141.24	16.383	0.976
116	2.38	80.37	235.10	27.272	0.983
363	4.40	26.52	77.56	28.192	0.977
363	5.28	28.07	82.08	29.838	0.986
363	6.28	29.62	86.65	31.497	0.997
363	8.21	32.84	96.10	34.932	0.984
363	9.88	36.04	105.40	38.311	0.992
363	12.85	41.02	120.00	43.608	0.986
363	15.55	46.76	136.76	49.714	0.985
363	18.4	53.70	157.05	57.089	0.977
363	20.25	59.36	173.63	63.113	0.985

a. 24-in. diameter, water reflected, cylindrical vessel

b. Chemical analysis - 116 g Pu/l, 1.85M acid, 238.9 g NO_3 /l total,
1.2552 sp. gr.

- 363 g Pu/l, 4.1M acid, 634.1 g NO_3 /l total,
1.7010 sp. gr.

Isotopic analysis, wt% - $^{238}Pu = 0.044$, $^{239}Pu = 90.677$, $^{240}Pu = 8.379$
 $^{241}Pu = 0.851$, $^{242}Pu = 0.049$

Calculations using the GAMTEC-II code were made for k_{∞} to determine the amount of gadolinium required to reduce k_{∞} to 1 for 100, 200 and 300 g Pu/liter solution containing 100% ^{239}Pu and at no excess acid. The results are plotted in Figure 5.1 and show that 5, 32 and 100 g Gd/liter, respectively, would be required to reduce K_{∞} to unity. The reduction in the effectiveness of gadolinium as a neutron absorber at higher concentration as the spectrum hardens is to be noted.

Reference

1. Technical Activities Quarterly Report, AEC Reactor Development and Technology Programs, October, November, December 1970, January, February, March 1971, BNWL-1522-2, Battelle-Northwest, April 1971.

FIGURE 5.1. Gadolinium Requirement to Safe Infinite Systems of $\text{Pu}(\text{NO}_3)_4$

6. NUCLEAR GRAPHITE

G. L. Tingey

IN-REACTOR CREEP

W. J. Gray

Constant-stress irradiation creep experiments are being conducted to high fluences (up to 2.5×10^{22} n/cm², $E > 0.18$ MeV) to determine whether any change in creep behavior occurs following the onset of rapid expansion at high fluences. Two capsules containing graphite samples under compressive stresses are being irradiated at 550 and 800°C, respectively. A third capsule containing samples under tensile stresses has been constructed and will operate at 800°C. A fourth capsule is planned which will contain samples under compressive stresses and will operate at about 1100°C. This capsule is scheduled to be charged into the reactor about February 1972.

The two compressive capsules operating at 550 and 800°C and the tensile creep capsule were shipped to Idaho Falls in June for charging into the ETR reactor. The two compressive creep capsules contain some samples being irradiated for the third time while some other samples are being irradiated for the first time. A three-cycle irradiation is planned which will raise the total fluence for some of the samples to about 1.1×10^{22} n/cm² ($E > 0.18$ MeV). These two capsules will be discharged about February 1972.

The tensile creep capsule contains samples being irradiated for the first time. The 1-1/2 cycles of irradiation planned for these samples will be equal to a fluence of about 2.3×10^{21} n/cm² ($E > 0.18$ MeV). This capsule is scheduled to be discharged in October 1971.

RADIOLYTIC REACTIONS IN GRAPHITE MODERATED REACTORS

R. P. Turcotte and G. L. Tingey

The radiolytic reaction of carbon monoxide and water vapor has been investigated at equimolar concentrations of 940, 65, and 10 VPM in one atmosphere helium, conditions of relevance to the HTGR system. The results

have been fit to equations of the form G (molecules/100 eV) = $aT^{1/2} e^{-E/RT}$ to yield the constants given in Table 6.1. The $\ln (G/T^{1/2})$ versus $1/RT$ plots from which activation energies were determined are shown in Figures 6.1 and 6.2 for CO_2 and H_2 , respectively. Figure 6.3 shows product yields at 50°C as a function of concentration.

TABLE 6.1.

<u>VPM</u>		<u>a</u>	<u>E(Kcal)</u>
940	H_2	0.061	0
	CO_2	0.150	0
65	H_2	0.004	-1.6
	CO_2	0.014	-1.1
10	H_2	0.004	-1.6
	CO_2	0.005	-1.6

While the results seem reasonably self-consistent, the apparent progression of activation energies from +2.5 Kcal for $[\text{H}_2\text{O}] = [\text{CO}] = 0.5$ atm obtained earlier⁽¹⁾ to -1.6 Kcal for the 10 VPM case is surprising and, if correct, would indicate a lessening of importance of the process with higher dilution and temperature.

The results obtained cannot be considered definitive, however, due to changes in oxygen impurity levels from <0.5 to ~ 2 VPM during the course of the runs, with the higher levels at higher temperatures. We have determined, under somewhat different conditions, (~ 80 VPM reactants and longer reaction times) a linear relation between product yields and oxygen concentration of significant magnitude. As might be anticipated, CO_2 yields increase and H_2 yields decrease; by $\sim 50\%$ for H_2 (~ 1 VPM) and $\sim 40\%$ for CO_2 (4.6-6.6 VPM) for a 1 VPM increase in oxygen concentration. The fact that the activation energies for H_2 and CO_2 formation (Table 6.1) are very close suggests that no such correlation exists for the other data obtained and corrections have not been applied.

FIGURE 6.1. Rate of Formation of Carbon Dioxide from the Radiolysis of a Mixture of Carbon Monoxide, Water Vapor, and Helium

Figure 6.2. Rate of Formation of Hydrogen from the Radiolytic Reaction of a Mixture of Carbon Monoxide, Water Vapor, and Helium

FIGURE 6.3. Radiolytic Yield of Products from the Radiolysis of a Mixture of Carbon Monoxide, Water Vapor, and Helium

It might be pointed out that negative activation energies, while unreasonable for a thermal system, may not be considered so in this radiolytic reaction since the process is complex and not well defined mechanistically. A substantial reduction in activation energy for the $\text{H}_2 + \text{CO}_2$ reaction with increasing dilution has been previously reported.⁽²⁾ The plot given in Figure 6.3 shows good extrapolation to zero yield with zero reactant concentration especially for CO_2 , lending further support to the validity of the technique employed.

The investigation will be continued to a number of other reactant concentrations and to lower oxygen impurity levels. It will be necessary to more quantitatively define the role of O_2 impurity for a full understanding of the system. In this respect, it is interesting to note that the stoichiometric ratio of products (1/1) seems to be approached only at the most dilute concentrations [$\text{CO} + \text{H}_2\text{O} = \text{CO}_2 + \text{H}_2$] where more complex secondary reactions appear minimized.

EFFECT OF SAMPLE SIZE ON DIMENSIONAL STABILITY OF GRAPHITE

W. J. Gray

There is ample evidence which shows that the irradiation behavior of large graphite bars as used in reactor construction is significantly different from that of small samples normally tested in irradiation programs.⁽³⁾ Initially, the investigation of this phenomenon emphasized differences associated only with sample size. However, controlled irradiation of different sizes of samples failed to show the large contraction differences observed.⁽⁴⁾ The program, therefore, was expanded to include an investigation of a phenomenon called "cold-seeding" wherein graphite samples are irradiated briefly at low temperatures prior to the normal high temperature irradiation. Several samples which have been previously "cold-seeded" are currently being irradiated to high fluences at high temperatures in the ETR. Recent results from both the size effect and cold-seeding experiments are discussed below.

The "size-effect" samples which were discharged from the KE-reactor in February 1971 continued to show about the same effect of sample size as

reported previously.⁽⁴⁾ That is, the largest samples contracted about 20% more than the smallest samples after a total fluence of about 3×10^{21} n/cm² ($E > 0.18$ MeV). Irradiation of these samples will be discontinued because of the reactor shutdown.

The cold-seeding capsule was discharged from the ETR in March 1971 after the second period of high-temperature irradiation. Results are shown in Table 6.2. Several things should be noted about these data:

1. Cold-seeding has apparently had a significant effect on length-change rates in many cases, even for fluences as low as 10^{18} n/cm² ($E > 0.18$ MeV).
2. In general, the effect of cold seeding has been less pronounced in the parallel orientation.
3. Samples irradiated at 1000°C or above have not, in general, been affected by cold-seeding (differences of less than about 10% are probably not significant). Poco graphite seeded at 200°C and then irradiated at 1000°C appears to be an exception to this. The other possible exception, TSX graphite seeded at 500°C and then irradiated at 1300°C, is somewhat in doubt because of the large difference between the perpendicular and parallel orientation.
4. The effect of cold seeding, if any, on TSX and TSGBF graphite has been to cause an increase in the contraction rate whereas the effect on Poco graphite has been a reduction of the expansion rate.
5. Because of this difference between Poco and the other two graphites, the percentage difference between seeded and control samples shown in the last column of Table 6.2 has a somewhat different significance. It should be noted, therefore, that the differences of 59% and 47% shown for Poco graphite actually reflect a reduction in expansion rate by a factor of about two.

The TSGFB graphite was irradiated under conditions thought to approximate those applying to samples where the "size-effect" was observed. Whether the difference observed here between seeded and control samples is significant is somewhat questionable. Certainly the difference is not

**TABLE 6.2. Irradiation Conditions and Length Changes
of Seeded and Controlled Samples**

Orientation	Temperature (°C)		TSX GRAPHITE			Length Change Rate ⁽¹⁾		$\frac{\alpha_s - \alpha_c}{\alpha_c} \cdot 100$ ⁽²⁾
	Low	High	Fluence (E > 0.18 MeV)		(%/10 ²¹)			
			Low T	High T		Seeded	Control	
				1st	2nd			
⊥ 	200	750	10 ¹⁸	0.78x10 ²¹	2.24x10 ²¹	-0.193 -0.432	-0.132 -0.338	46 28
⊥ 	200	750	5x10 ¹⁸	0.78	2.24	-0.215 -0.440	-0.132 -0.338	63 30
⊥ 	500	750	10 ¹⁹	0.78	2.24	-0.174 -0.309	-0.132 -0.338	32 - 9
⊥ 	200	1000		0.95	2.85	-0.160 -0.789	-0.153 -0.759	5 4
⊥ 	200	750	5x10 ¹⁹	0.78	2.24	-0.291 -0.531	-0.132 -0.338	120 57
⊥ 	300	500		0.55	1.65	-0.211 -0.321	-0.166 -0.330	27 - 3
⊥ 	500	1000		0.95	2.85	-0.154 -0.775	-0.153 -0.759	0 2
⊥ 	50	750	10 ²⁰	0.78	2.24	-0.224 -0.538	-0.132 -0.338	70 59
⊥ 	200	500		0.55	1.65	-0.254 -0.477	-0.166 -0.330	53 45
⊥ 	200	750		0.78	2.24	-0.276 -0.583	-0.132 -0.338	109 72

TABLE 8.2. (contd)

Orientation	Temperature (°C)		Fluence (E > 0.18 MeV)			Length Change Rate ⁽¹⁾ (%/10 ²¹)		% Increase ⁽²⁾ $\frac{\alpha_s - \alpha_c}{\alpha_c} \cdot 100$
	Low	High	Low T	High T		Seeded	Control	
				1st	2nd			
TSX GRAPHITE (contd)								
⊥	200	1000		0.95	2.85	-0.194	-0.153	27
						-0.774	-0.759	2
⊥	500	750		0.78	2.24	-0.121	-0.132	- 8
						-0.297	-0.338	-12
⊥	500	1000		0.95	2.85	-0.146	-0.153	- 5
						-0.837	-0.759	10
⊥	500	1300		1.00	3.00	-0.222	-0.101	120
						-0.591	-0.754	-22
TSGBF GRAPHITE								
⊥	300	650	10 ²⁰	0.65x10 ²¹	1.95x10 ²¹	-0.275	-0.242	14
						-0.412	-0.330	25
POCO (AXF-8QBG1) GRAPHITE								
Isotropic	500	1000	10 ¹⁹	2.95x10 ²¹	2.85x10 ²¹	+0.491	+0.503	- 2
Isotropic	200	750	10 ²¹	0.78	2.24	+0.108	+0.262	-59
Isotropic	200	1000		0.95	2.85	+0.266	+0.503	-47
Isotropic	500	1000		0.95	2.85	+0.481	+0.503	- 4
Isotropic	500	1300		1.00	3.00	-0.039	-0.047	-17

(1) Length-change rate computer between 1st and 2nd high-temperature exposure only (numbers represent the average length-change rates for 3 to 6 samples in each case).

(2) α_s = length change rate of seeded samples.

α_c = length change rate of control samples.

enough to explain the observed "size-effect". However, conditions not far different (that is, TSX graphite seeded at 200°C and then irradiated at 500 or 750°C) did exhibit a rather marked effect due to cold-seeding. I believe the difference between TSX and TSGBF graphites is not significant here. In view of the uncertainties involved, particularly regarding the conditions applying to the "size-effect" samples, I believe that at least a large part of the observed "size-effect" can probably be attributed to the combined effects of sample size alone plus cold seeding. Again because of the uncertainties, this rather indefinite conclusion is probably the best that can ever be done. It is unfortunate that the mechanism of any effects due to sample size per se are still unknown.

The results of the cold-seeding study are of interest not only in regards to the "size-effect" observation but also as they may apply to the operation of MSR's and HTGR's. Graphite temperatures during initial startup of these reactors may be low enough to produce cold-seeding effects similar to that shown here. If so, contraction rates, total contraction, and stresses generated by contraction differences will be different than predicted on the basis of constant-temperature tests.

It remains to be seen whether the effect of cold-seeding will persist to higher fluences. If it does, the effect on expansion rates at high fluences will be of particular interest.

Although cold-seeding does not appear to produce an effect under all conditions shown in Table 8.2, I believe one more irradiation period for all samples will be worthwhile. Therefore, all samples will be irradiated for a third time beginning about October 1971. Thereafter, irradiation of samples not showing an effect of cold-seeding will be discontinued.

HIGH TEMPERATURE IRRADIATIONS

W. J. Gray and W. C. Morgan

A report⁽⁵⁾ on the high-temperature graphite irradiation data has been prepared and issued; this report covers all sample data from irradiations prior to capsule GEH-13-421.

Measurements have been completed on the samples from capsule GEH-13-421, which was discharged at the end of ETR Cycle 110B. In general, the samples of conventional nuclear graphites irradiated at 1200 to 1300°C have now reached exposures 2 to 3 times as high as those where "turnaround" occurs for samples irradiated at 950 to 1100°C; there is as yet no indication of an incipient onset of the rapid expansion which is characteristic of "turn-around" at the lower temperature range. Figure 6.4, which shows the dimensional changes of NC-8 graphite, illustrates the behavior observed for conventional nuclear graphites in these temperature ranges. Figure 6.5, which is a similar plot for Poco grade AXF-8Q1, illustrates the behavior observed for the fine-grained isotropic graphites. Like the conventional graphites, the samples of Poco graphites are exhibiting a more rapid expansion at 1000 to 1100°C than at 1200 to 1300°C.

Almost all of these samples have been placed in the next irradiation capsule, which is scheduled for about three cycles of irradiation in the ETR beginning with Cycle 113 at the end of August 1971.

MECHANISM OF IRRADIATION DAMAGE

W. C. Morgan and W. J. Gray

As reported in the last Technical Activities Quarterly,⁽⁶⁾ Riley's method⁽⁷⁾ was used to obtain theoretical equations for the graphite crystal lattice expansions and coefficients of thermal expansion as a function of temperature. The final report⁽⁸⁾ on this phase of the work has been submitted for publication in Carbon and served as the basis for an oral presentation at the Tenth Biennial Conference on Carbon (Lehigh University, 28 June - 2 July 1971).

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FIGURE 6.4. Irradiation Induced Length Changes for NC-8 Graphite

FIGURE 6.5. Irradiation Induced Length Changes for POCO Grade AXF-8Q1 Graphite

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