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DP-945

AEC RESEARCH AND DEVELOPMENT REPORT

HEAVY WATER MODERATED POWER REACTORS

PROGRESS REPORT
NOVEMBER-DECEMBER 1964

Technical Division
Wilmington, Delaware

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HEAVY WATER MODERATED POWER REACTORS PROGRESS REPORT

November-December 1964

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Wilmington, Delaware

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ABSTRACT

Operation of the Heavy Water Components Test Reactor (HWCTR) was terminated on December 1, as the result of a decision by the USAEC to curtail its development program on D_2O-cooled power reactors. The facility is being placed in standby condition.

Ten-ft-long Zircaloy spacing ribs were attached successfully to the inside surface of Zircaloy housing tubes by electron beam welding.

The buckling that was observed in an unalloyed uranium tube at 6830 MWD per metric ton was attributed to cavitation swelling.

Corrosion coupons of various materials that were exposed to hot D_2O in the HWCTR were examined.

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HEAVY WATER MODERATED POWER REACTORS
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INTRODUCTION

This report reviews the progress of the Du Pont development program on heavy-water-moderated power reactors. The goal of the program is to advance the technology of these reactors so that they could be used in large power stations to generate electricity at fully competitive costs. Currently, the effort is concerned with (1) irradiation tests of candidate fuel assemblies and other reactor components in the Heavy Water Components Test Reactor (HWCTR), and (2) the development of low-cost fuel tubes for use in large water-cooled reactors.

SUMMARY

Operation of the HWCTR was terminated on December 1 as the result of a decision by the U. S. Atomic Energy Commission to curtail its development of D_2O-cooled power reactors. The reactor is being placed in standby condition in anticipation of possible resumption of operation at a later date.

Ten-foot-long Zircaloy spacing ribs were attached successfully to the inside surface of Zircaloy housing tubes by electron beam welding. It appears that, with further development, this technique can be used to attach similar ribs to Zircaloy-clad UO_2 fuel tubes. In a flow test of an alternative spacing device, prohibitive cladding wear was observed in an assembly of two nested tubes that were spaced radially by short stainless steel ribs affixed to washers. One rib penetrated through the Zircaloy cladding of a fuel tube, and three other ribs formed cladding indentations as deep as 0.005 inch.

Postirradiation examination of a Zircaloy-clad tube of unalloyed uranium that buckled during HWCTR irradiation to 6830 MWD/MTU* revealed that the buckling resulted from cavitation swelling of the uranium.

Corrosion coupons that were exposed in the HWCTR to D_2O at a maximum temperature of 250°C were examined for type and extent of corrosion. The maximum corrosion rate of carbon steel specimens that were exposed in the reactor vessel for 679 days was 0.24 mil per year; the maximum pit depth was 7 mils. One sample of 17-4 PH stainless steel that was heat treated at 900°F cracked by stress corrosion within 566 days. Welds in Zircaloy samples exhibited the white corrosion product that is characteristic of "breakaway" corrosion. Stainless steel and "Inconel"** specimens did not corrode significantly.

* Megawatt-days per metric ton of uranium.
**Trademark of International Nickel Co.

DISCUSSION

I. OPERATION OF THE HEAVY WATER COMPONENTS TEST REACTOR (HWCTR)

Operation of the HWCTR was terminated on December 1, as part of the curtailment by the U. S. Atomic Energy Commission of the development of D_2O-moderated reactors that are cooled with liquid D_2O. The reactor is being placed in standby condition in anticipation of possible resumption of operation at a later date. All of the fuel was removed from the reactor and shipped to the Receiving Basin for Off-Site Fuel, where the fuel will be stored or prepared for final disposal. The heavy water was drained from the reactor system and shipped to the Heavy Water Production Plant, where it will be upgraded to 99.75% D_2O prior to storage for possible future use. The reactor, the isolated coolant loops, and the primary coolant system are being dried and will be left slightly pressurized with dry nitrogen. Detailed inspections are being made of reactor components and process systems. Status reports are being prepared to describe the present condition of the facility, to outline the plans for modifications that were contemplated at the time of shutdown, and to recommend other steps that should be considered if the reactor is to be operated again.

Prior to the final shutdown, the HWCTR was in operation 65% of the time during November at a power of about 50 MW. Operating data are summarized in Tables I and II and in Figures 1 and 2. The status of irradiation tests at the end of December is shown in Tables III and IV. Early in November, the boron stainless steel poison tubes in the driver fuel assemblies were replaced with Zircaloy tubes. This change was made to gain nuclear reactivity and thus to prolong the useful life of the driver fuel. The poison tubes were removed readily from all except three of the twenty-four drivers. The three poison tubes that could not be removed easily were left in place.

Concurrent with final shutdown of the HWCTR, all fuel fabrication and related development work for the reactor was terminated except for (1) completion of Zircaloy housing tubes for a new set (M-3) of metal driver assemblies (9.3% U in Zr), (2) completion of the fuel tubes for three UO_2 driver assemblies, and (3) postirradiation examinations of fuel assemblies. Since fuel tubes for the metal drivers have already been fabricated, completion of the housings will make it possible to place the new drivers in the HWCTR on short notice. The three sets of UO_2 driver tubes will be reserved for pilot irradiations whenever the HWCTR is started up again.

II. SPACING DEVICES FOR UO_2 FUEL TUBES

A. WELDED RIBS

With the ultimate objective of developing techniques and equipment for attaching spacing ribs to Zircaloy-clad UO_2 fuel tubes by electron beam welding, Zircaloy ribs were welded to Zircaloy housing tubes (3.20-in. ID x 0.035-in. wall) that are intended for use with the M-3 driver assemblies in the HWCTR. Ten-foot-long ribs were successfully welded to the inside surface of 21 of these housing tubes. The techniques and results are discussed in the following paragraphs.

The quality of the welded tubes was satisfactory for housing tubes; samples of the rib and end fitting welds were black and lustrous after an autoclave treatment in 400°C steam for 48 hours. Metallurgical samples were machined off the ends of ten tubes and examined; a typically centered rib and an example of an off-center rib (0.011 inch) are shown in Figure 3. Minor equipment revisions to improve positioning accuracy would be required to produce welds with fusion over 100% of the rib-tube interface.

The welding facility, as described in DP-905, comprises a commercial electron beam welder with 16-inch-diameter by 16-foot-long extensions attached to each side of the vacuum chamber, and a drive and positioning mechanism to position the ribs relative to the tube and to move the assembly accurately beneath the electron beam during welding. A precision-machined mandrel positions and supports the ribs within the tube during the welding operation. Experience gained in welding the 21 tubes is summarized below:

Straightness of Weld - The two side guide rolls position the mandrel and tube accurately as they travel under the electron beam. At first, however, excessive rotation of the tube was caused by tilting of the rear support truck as it passed over the uneven surface of the support table. Shimming of the worst low spots reduced this deviation to ±0.005 inch from a straight line.

Roundness of Finished Tube - The 3.270-inch OD x 0.035-inch wall Zircaloy tubing was all within ±0.015-inch-diameter limits as received from the tubing vendor. However, in the first short test sections that were welded, the tube became "square" by as much as 0.100-inch diameter greater at the ribs (major axes) than at points 45° from the ribs (minor axes). Addition of a pair of weighted top rolls to force the tube down against the mandrel and hence into a more rounded shape during welding reduced this imperfection to no more than 0.025 inch, which is within the specified tube diameter tolerance of 0.030 inch.

<u>Rib-Circle Diameter</u> - On the first group (19) of tubes welded, the average rib-circle diameter was 0.007 to 0.025 inch greater than the specified nominal rib-circle diameter. This deviation apparently was due in part to the "squareness" outlined above and to oversize tubing but also to a hump approximately 1/4 inch wide by 0.003 to 0.007 inch high produced in the tubing over each rib during welding. On subsequent tubes, rib heights were increased 0.007 inch and the rib circles were within the ±0.007-inch tolerance on average rib height.

<u>Mechanical Equipment</u> - Evacuation of the cabinet to the 0.1-micron absolute pressure required for welding requires only 5 minutes. Since at least this much time is needed for the preweld double traverse of the mandrel to verify its proper alignment, no loss of production time has been incurred by opening the cabinet and manually rotating and locking the mandrel between each rib. Therefore, efforts have been abandoned for the time being to perfect the motor and solenoid brake for rotating and locking the mandrel.

A simple drawbench was constructed which utilizes a hand operated winch with attached cable to load and discharge the expanding mandrel from the tubes. The maximum forces required for the two operations were 600 and 1100 pounds, respectively.

B. <u>SPACING WASHERS</u>

A mockup of the proposed two-tube UO_2 driver assemblies for the HWCTR (DP-915) was subjected to a flow test in water to investigate the feasibility of using short stainless steel ribs for radial spacing of the nested fuel tubes. Inspection of the mockup assembly after 97 days of flow test revealed that one of the ribs on the inner fuel column penetrated through the inner cladding of the outer fuel tube. Other ribs made visible indentations on the cladding. Additional development would be necessary before spacing devices of this type could be considered safe for use in the HWCTR.

The UO_2 drivers proposed for the HWCTR contain a 10-foot-long outer fuel tube and a column of three shorter tubes positioned within the outer tube by some kind of spacing device. Dimensions of the fuel tubes were reported in DP-915. It was planned that short washers of stainless steel with four spacing ribs attached would be positioned at the extremities of the inner fuel column and between the individual fuel tubes in that column. The arrangement of these ribbed washers is shown in Figure 4, and the washer design is illustrated in Figure 5. The mockup assembly consisted of a Zircaloy outer housing tube with integral ribs, a 9-foot-long UO_2 outer fuel tube clad with Zircaloy-2, and a dummy inner fuel column made from three pieces of stainless steel rod to duplicate the weight of the inner fuel column. The mockup dimensions and flow test conditions are compared in Table V with design values for the oxide drivers.

During the flow test, one of the ribs wore a 1/8-inch-wide x 2-inch-long hole through the 0.030-inch-thick cladding of the outer fuel tube. This rib exhibited moderate wear damage. Three other ribs formed indentations as deep as 0.005 inch in the fuel cladding.

Because of the decision to shut down the HWCTR, no further work is planned on the spacer development program. At present, to separate two concentric oxide fuel tubes clad with Zircaloy, only integral ribs assure satisfactory performance. Additional development work would have to be done to establish whether ribbed spacers can be used.

III. POSTIRRADIATION EXAMINATIONS OF UNALLOYED URANIUM TUBES

As reported in DP-895 and DP-935, two thin-walled fuel tubes with unalloyed uranium cores and Zircaloy cladding survived irradiation in the HWCTR to 6830 and 6470 MWD/MTU at time-averaged uranium temperatures of 500 and 515°C, respectively. Visual examinations and dimensional measurements showed that severe buckling of the inside surface occurred during irradiation. The outside diameter of the tubes changed as much as ±0.050 inch. The ridges that formed when the inner surface buckled were 0.1 inch high.

Metallographic examination of sections cut from the regions of severest buckling in one of the tubes revealed bands containing high concentrations of cavities that surrounded a central region of lower cavity concentration (Figure 6). The regions near the surfaces of the tube also contained a few cavities. These cavities are believed to be the result of the process of cavitation swelling that has been observed in other irradiations of unalloyed uranium[1] and in U - 2 wt % Zr alloy (DP-935).

A radial crack was observed in the uranium core in a sample removed from the buckled portion of the tube. As shown in Figure 7, this crack extended completely across the core; it had begun to penetrate both the inner and outer Zircaloy cladding. A number of large zirconium hydride needles were at the tips of the crack. In other regions of the cladding, a few hydride needles were observed in a circumferential orientation. The grain structure of the cladding showed no change from that before irradiation even in the regions of severest buckling or adjacent to the crack.

IV. EVALUATION OF CORROSION SPECIMENS FROM THE HWCTR

A group of corrosion specimens that were exposed to D_2O for various time intervals in the HWCTR were examined for type and extent of corrosion. The specimen materials, their locations, and the exposure periods are listed in Table VI. Data on reactor innage and D_2O composition were presented in previous reports in this series[2]. The nominal D_2O temperature during reactor operation ranged from 180 to 250°C. The results observed for the individual materials are presented below.

A. CARBON STEEL IN THE REACTOR VESSEL

The following ASTM grades of plain carbon steel were immersed in D_2O above the top shield for 399, 566, and 679 days:

- ASTM A-210 - Specified for medium-carbon, seamless-steel boiler and superheater tubes
- ASTM A-106 - Specified for seamless, carbon steel pipe for high-temperature service
- ASTM A-212 - Specified for high-tensile-strength carbon-silicon steel plates for boiler and other pressure vessels
- ASTM A-285 - Specified for low and intermediate tensile-strength carbon steel plates of flange and firebox qualities (plates 2 inches and under in thickness)

Corrosion rates of specimens exposed 566 and 679 days in the reactor vessel were about the same as for specimens exposed for 399 days (0.24 mil per year maximum; see Tables VII and VIII). Maximum pit depth increased from about 3.5 mils in 399 days to about 7 mils in both 566 and 679 days. Corrosion rates were the same in and out of the D_2O moderator and at the liquid-vapor interface, although pitting was somewhat more severe in the immersed specimens (Table IX).

B. STAINLESS STEEL, ZIRCALOY, AND "INCONEL"

Corrosion by D_2O was the same in "in-flux" and "out-of-flux" tests of stainless steel, Zircaloy, and "Inconel" during 566 days of exposure (Table X). Corrosion rates of these materials were negligibly small.

C. TYPE 17-4 PH STAINLESS STEEL

Stressed specimens of 17-4 PH stainless steel, previously heat treated at 900 and 1100°F, were exposed in the moderator out of the flux zone and in the vapor above the moderator for 566 days (Table XI). Failure occurred in one specimen heat treated at 900°F, a heat treatment which promotes susceptibility to stress corrosion cracking. This specimen was also exposed in the light-water dummy run before nuclear startup of the reactor.

D. LIQUID BAYONET LOOP TESTS

Zircaloy-2 and -4 specimens from the forgings and the extruded tubes used in the bayonets of the isolated coolant loops were tested for 415 days in the Zircaloy bayonet of the liquid D_2O loop (Table XII). Some breakaway corrosion occurred adjacent to the welds in both the Zircaloy-2 and -4. As these welds are representative of welds on the bayonet, the latter welds may be approaching failure.

V. OXYGEN REMOVAL FROM HWCTR COOLING WATER

In DP-925, recent severe pitting attack of the carbon steel tubes in the HWCTR steam generators was attributed tentatively to a change in quality of the sodium sulfite that was used to scavenge oxygen from the feedwater. This conclusion has been confirmed by laboratory tests of the efficacy of two commercial brands of catalyzed sulfite. The tests demonstrated that the brand used for about 6 months in 1964 is not consistently effective in oxygen removal, whereas the brand used at all other times in the HWCTR is consistently effective regardless of the water or the other additives in the water. These two products will be identified in this report as Sulfite "A" and Sulfite "B", respectively.

Laboratory measurements were made of the rate of the oxygen scavenging reaction

$$1/2\ O_2\ (\text{aqueous}) + SO_3^{=} \rightarrow SO_4^{=}$$

in oxygenated water containing additives and sulfite in concentrations appropriate to the water supply from the HWCTR wells. After a given time interval subsequent to sulfite addition, the reaction was quenched by addition of excess I_2 solution. After removal of an aliquot for determination of the I_2 excess, the Winkler method for dissolved oxygen was applied to the remaining solution, with production of more I_2. The I_2 titrations were performed potentiometrically with $Na_2S_2O_3$.

Figure 8 illustrates the two types of behavior encountered with the catalyzed sulfites. Either the reaction proceeded almost immediately, or it required several minutes induction time, after which it proceeded rapidly. The effect of variables on the scavenging reaction was evaluated by determining the oxygen residual after the reaction proceeded for 1 minute. This was sufficient time if the rapid reaction were to occur. In Table XIII the scavenging action is rated "positive" if less than 0.3 ppm oxygen remained and "negative" if more than 0.3 ppm remained after the 1-minute reaction time.

Sulfite "B" was consistently effective, but Sulfite "A" was not. Sulfite "A" performance was not improved by lowering the pH with NH_4Cl or H_2SO_4 to make it similar to Sulfite "B". However, addition of 0.3% Co^{++} made Sulfite "A" as satisfactory as Sulfite "B" for the particular samples that were tested.

TABLE I

Operating Chronology of HWCTR

Nov. 1-10	Replaced driver targets with Zircaloy inner housings and inspected test fuel
10	Attained criticality
11	Attained 36 MW
12	Attained 38 MW
13	Attained 39.5 MW
13-15	Operated at 39.5 MW
15	Attained 41 MW
15-17	Operated at 41 MW
17	Attained 47.5 MW
18	Attained 50 MW
18-23	Operated at 50 MW
23	Attained 51 MW
23-25	Operated at 51 MW
25	Shut down - electrical power interruption
25	Attained criticality
26	Shut down - control rod drives inoperable
26	Attained criticality and 30 MW
27	Attained 51 MW
27-29	Operated at 51 MW
29	Attained 52 MW
Nov. 29 - Dec. 1	Operated at 52 MW
Dec. 1	Shut down - HWCTR operation terminated
2	Attained criticality and 10 kw
3	Shut down - end of wire irradiation test at low power
4	Started fuel discharge
7	Completed fuel discharge - started fuel shipments
8	Started discharge of non-fuel reactor components - started draining liquid loop D_2O
9	Replaced corrosion coupon holders in reactor
11	Started inspection of reactor vessel
14	Drained poison injection system
15	Started draining deluge tank
16	Completed fuel shipments - completed reactor inspection - installed reactor head
17	Completed draining of reactor vessel
21	Started vacuum drying of high pressure systems
22	Started draining D_2O low pressure system
29	Completed vacuum drying high pressure systems
30	Completed draining D_2O low pressure system - completed dedeuterizing purification deionizers
31	Completed all shipments of material from spent fuel basin - completed gas purge of low pressure system and established nitrogen blanket

TABLE II

Operating Summary of HWCTR

	November		December	
Time reactor critical, %	65.2		2.1	
Maximum power, MW	52		52	
Reactor exposure, MWD	Drivers	Test	Drivers	Test
For month	684	140	24	5
Accumulated in H-2 cycle	5085	1040	5109	1045
Losses				
D_2O (100 mole %), lb	487		+84[a]	
% of inventory per year	8.3		-	
Deuterium, g	3564		0	
Helium, scf	78,102		47,535	

(a) Gain

TABLE III

Test Fuel Irradiation Data
November-December 1964[a]

Reactor power	52 MW
Coolant pressure	1200 psig
Moderator temperature	200°C
Coolant inlet temperature	181.5°C

			Maximum Nominal Conditions							Maximum Exposure[d], watt-days/g	
Position	Element Number[b]	Assembly Power[c], MW	Specific Power[d], watts/g	Heat Flux, pcu/(hr)(ft^2)	Outlet Temp., °C	Surface Temp., °C	Core-Clad Temp., °C	Core Temp., °C	$\int k d\theta$, watts/cm	Attained	Goal
37	TMT-1-3	1.20	53.5	410,000	212	253	380	483	-	3,489	20,000
38	SOT-6-2	0.61	41.5	234,000	245	267	323	-	25.5	5,067	30,000
39	SOT-8-2	1.03	58.0	284,000	198	233	327	-	27.2	3,987	30,000
40	TMT-1-2	1.21	54.0	415,000	211	253	381	486	-	3,578	20,000
42	SOT-8-3	1.08	61.5	298,000	198	234	332	-	28.4	4,311	30,000
55	3EMT-2	0.80	107.5	535,000	196	253	391	472	-	4,972	10,000
56	SOT-1-4	0.52	51.2	228,000	190	225	291	-	18.2	12,637	30,000
57	SMT-1-3	0.43	34.2	266,000	200	280	340	395	-	5,600	10,000
58	SOT-1-2	0.44	41.0	184,000	188	216	269	-	14.7	17,333	30,000
59	SOT-9-2	0.85	57.0	322,000	221	281	358	-	35.0	4,038	20,000
60	RMT-1-2	0.61	46.0	240,000	222	277	425	459	-	3,319	10,000

(a) Data taken on 11/30/64; exposures as of 12/31/64.
(b) Elements are identified in Table IV.
(c) "Flow-ΔT" power calculation; does not include moderator heating.
(d) These values are based on an assembly power of 1.09 times "Flow-ΔT" power to include moderator heating.

TABLE IV

Fuel Identification Data

Designation	Shape	OD	ID	Unit Length	Units	Description
SOT-1	Tube	2.06"	1.47"	14"	7	1.5% enriched UO_2 vibrated and swaged in Zircaloy
SOT-6	Tube	2.54"	1.83"	14"	7	Natural UO_2 vibrated and swaged in Zircaloy
SOT-8	Tube	3.67"	2.99"	14"	7	1.2% enriched UO_2 vibrated and swaged in Zircaloy
SOT-9	Tube	2.54"	1.83"	14"	7	1.2% enriched UO_2 vibrated and swaged in Zircaloy
TMT-1	Tube	2.55"	1.85"	10'	1	1.4% ^{233}U in thorium metal core in Zircaloy
3EMT-2	Tube	2.06"	1.70"	37"	1	3% enriched uranium metal alloyed with 1.5% Mo in Zircaloy
SMT-1-3	Tube	1.70"	1.24"	11-1/4"	5	Natural uranium metal alloyed with Fe, Al, Si in Zircaloy
RMT-1-2	Tube	2.07"	1.57"	10'	1	Unalloyed natural uranium metal in 60-mil Zircaloy

TABLE V

Comparison of Dimensions and Operating Conditions for Oxide Driver Assemblies and Flow Test Mockup

		Two-Tube UO_2 Driver Assembly	Driver Mockup
Outer housing tube:	OD, inches	3.260	2.960
	ID, "	3.200	2.900
Outer annulus thickness,	"	0.220	0.138
Outer fuel tube:	OD, "	2.760	2.623
	ID, "	2.290	2.012
Intermediate annulus thickness	"	0.360	0.366
Inner fuel tube:	OD, "	1.570[a]	1.280[b]
	ID, "	1.050	-
Total coolant flow, gpm		300	200
Coolant velocities, ft/sec			
Outer annulus		17.3	13.1
Intermediate annulus		22.0	26.0
Inner annulus		17.6	-
Spacer rib dimensions:			
Length, inches		4.0	2.37
Width, inch		0.109	0.078
Assembly clearance:			
Outer housing rib circle to outer fuel OD, inch		0.030	0.036
Inner fuel rib circle to outer fuel ID, inch		0.030	0.027

(a) The inner fuel tube of the UO_2 driver assembly will comprise three pieces ≈38 inches long to facilitate cave examination after irradiation.

(b) Diameter of stainless steel rod to duplicate inner fuel weight.

TABLE VI

Description of Corrosion Specimens in the HWCTR

Material	Specimen Location	Exposure, days
Carbon steel	Reactor vessel	
	D_2O - above shield	566, 679
	D_2O - in neutron flux	566
	Interface, D_2O-vapor	566
17-4 PH stainless steel	Reactor vessel	
	D_2O - above shield	566
	Vapor above D_2O	566
304 SS, 316 SS, Zircaloy-2 and -4, "Inconel"	Reactor vessel	
	D_2O - above shield	566
	D_2O - out of neutron flux	566
Zircaloy-2 and -4	Liquid bayonet loop	415

TABLE VII

Pitting of In-Reactor Corrosion Coupons

	Pit Depth, mils					
	399 Days		566 Days		679 Days	
Material	Mean	Max.	Mean	Max.	Mean	Max.
A-210	1.9	3.1	3.4	6.7	3.0	5.3
A-106	2.6	3.9	3.7	7.3	3.8	7.7
A-212	2.5	4.6	3.9	7.7	3.5	6.6
A-285	2.2	3.4	3.3	6.0	3.4	6.6

TABLE VIII

Corrosion Rate[a] for In-Reactor Corrosion Coupons

Material	Corrosion Rate, mils per year		
	399 Days	566 Days	679 Days
A-285	0.18	0.17	0.20
	0.19	0.17	0.16
A-106	0.23	0.24	0.23
	0.22	0.23	0.24
A-210	0.15	0.20	0.20
	0.16	0.10	0.21
A-212	0.21	0.23	0.19
	0.19	0.19	0.20

(a) Determined by weight loss.

TABLE IX

Carbon Steel Corrosion at D_2O-Vapor Interface

ASTM Type A-106, stressed and exposed for 566 days

Specimen No.	Position	Corrosion Rate, mils per year	Pit Depth, mils		Pits/in²
			Max.	Avg.	
30	Vapor	0.055	4.3	2.1	60
31	Vapor	0.061	1.9	2.1	75
35	Vapor	0.072	7.0	1.8	66
36	Vapor	0.060	2.8	1.3	39
32	Interface	0.060	3.9	1.7	57
33	Immersed	0.088	5.0	2.0	132
34	Immersed	0.092	6.0	2.1	132
37	Immersed	0.081	4.5	2.0	87
38	Immersed	0.086	5.1	2.1	117

TABLE X

Corrosion Rate[a] In- and Out-Of-Flux Zone

Material	Corrosion Rate, mils per year: Out of Flux 399 Days	Out of Flux 566 Days	In Flux 566 Days
A-106 carbon steel	0.26	0.16	No data
A-210 carbon steel	0.16	0.066	No data
Type 304 SS, annealed	-	0.0082	0.015
Type 304 SS, sensitized	nil	0.0010	0.018
Type 304 SS, 1% boron	0.01	0.0013	0.0005
Type 316 SS	nil	0.010	0.012
Zircaloy-2	0.03	nil	0.018
Zircaloy-4	0.02	nil	0.021
"Inconel"	0.01	0.0004	0.0064

(a) Determined by weight loss.

TABLE XI

Corrosion of 17-4 PH Stainless Steel

Specimen No.	Holder No.[a]	Light Water Test	Specimen Condition	Results
54	1-11	No	48% cold work, 8 hr at 1100°F	No cracks
64	1-11	No	8 hr at 1100°F	No cracks
74	1-11	No	48% cold work, 1 hr at 900°F	No cracks
84	1-11	No	1 hr at 900°F	No cracks
43	1-11	Yes	1 hr at 900°F	Cracked
49	1-16	No	48% cold work, 8 hr at 1100°F	No cracks
69	1-16	No	8 hr at 1100°F	No cracks
79	1-16	No	48% cold work, 1 hr at 900°F	No cracks
89	1-16	No	1 hr at 900°F	No cracks

(a) Holder 1-11 - out of flux in moderator.
Holder 1-16 - in vapor phase above moderator.

TABLE XII

Liquid Bayonet Loop Specimens

Holder No.	Specimen No.	Material
L-4	37	Zircaloy-2 - forging
L-4	38	Zircaloy-2 - forging welded to extruded tube
L-4	39	Zircaloy-2 - extruded tube
L-4	40	Zircaloy-4 - forging
L-4	41	Zircaloy-4 - forging welded to extruded tube
L-4	42	Zircaloy-4 - extruded tube

TABLE XIII

Efficacy of Catalyzed Sodium Sulfites for Oxygen Removal from HWCTR Well Water

Key: O = No Na_3PO_4
P = 30 ppm PO_4^{3-}
+ corresponds to <0.3 ppm O_2 within 1 minute
- corresponds to >0.3 ppm O_2 at 1 minute

	Deionized Water (8.5 ppm O_2)		Well Water Samples: No. 58 (6.8 ppm O_2)		No. 60 (3.8-4.4 ppm O_2)		No. 67 (8.5 ppm O_2)	
	O	P	O	P	O	P	O	P
Sulfite "A" (pH 10.5)	-	+	+	±	+	-	-	+
Sulfite "B" (pH 8.4)	+	+	+	+	+	+	+	+
Modified Sulfite "A"								
pH 9.4 with NH_4Cl	-		+	-	-	-		+
pH 8.5 with H_2SO_4			+	+	-	-		
0.3% Co^{++} added	+	+	+	+	+	+	+	+

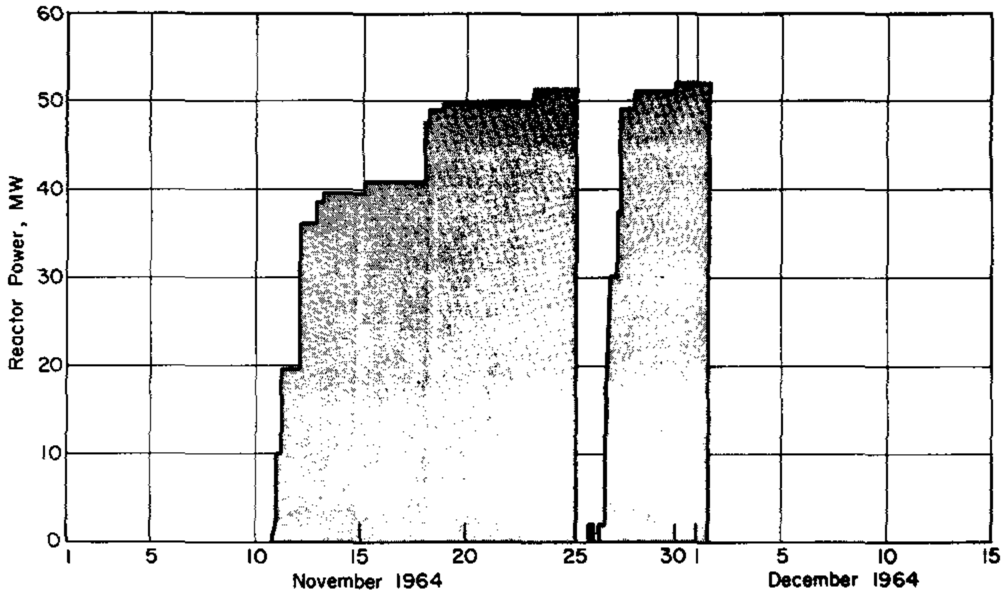


FIG. 1 OPERATING POWER OF HWCTR

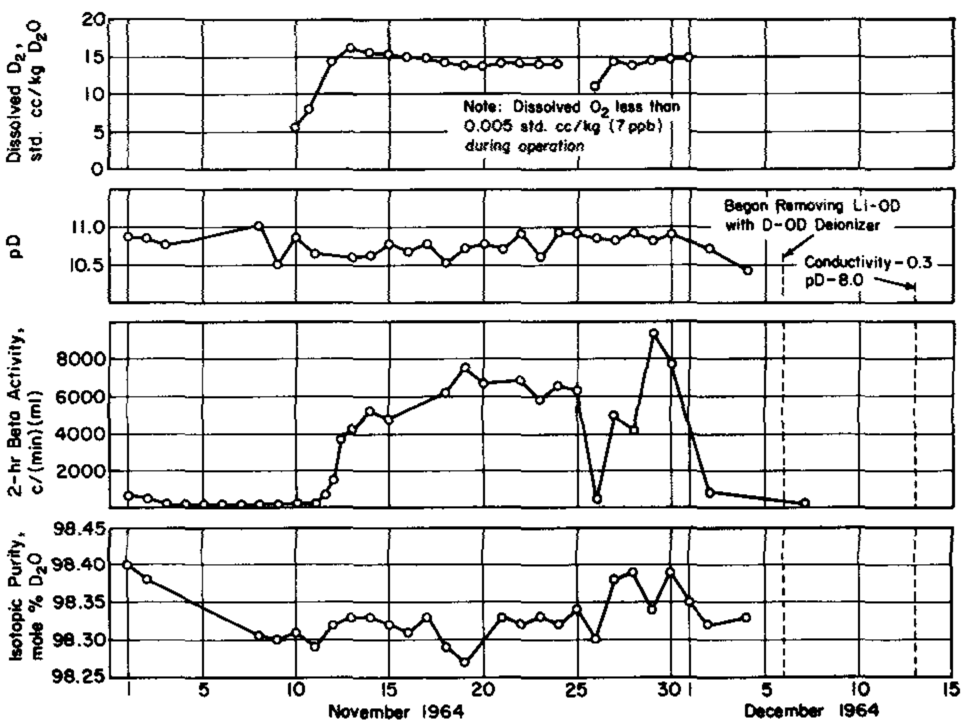
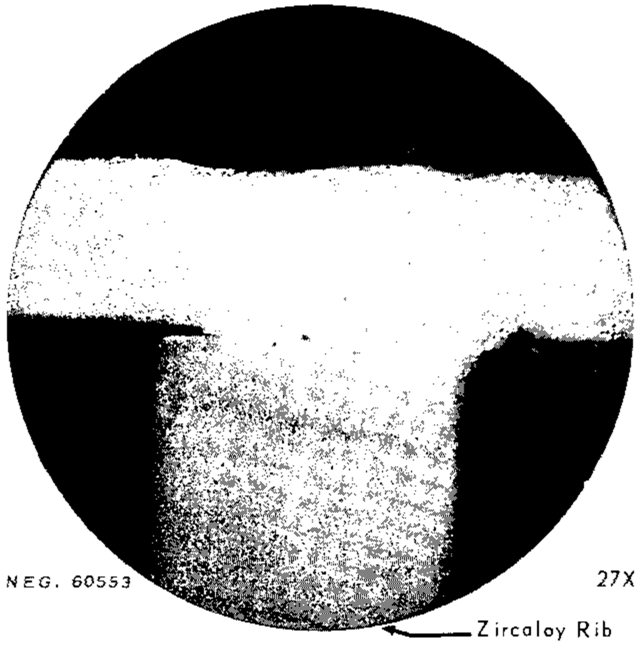
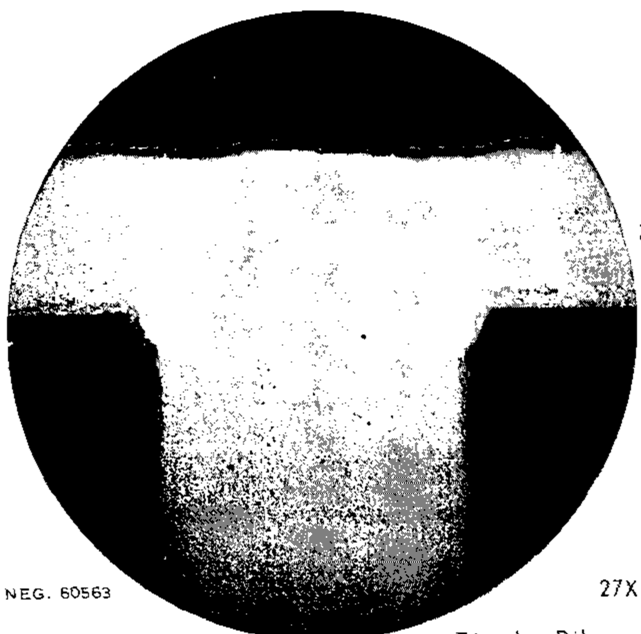


FIG. 2 HEAVY WATER QUALITY IN HWCTR

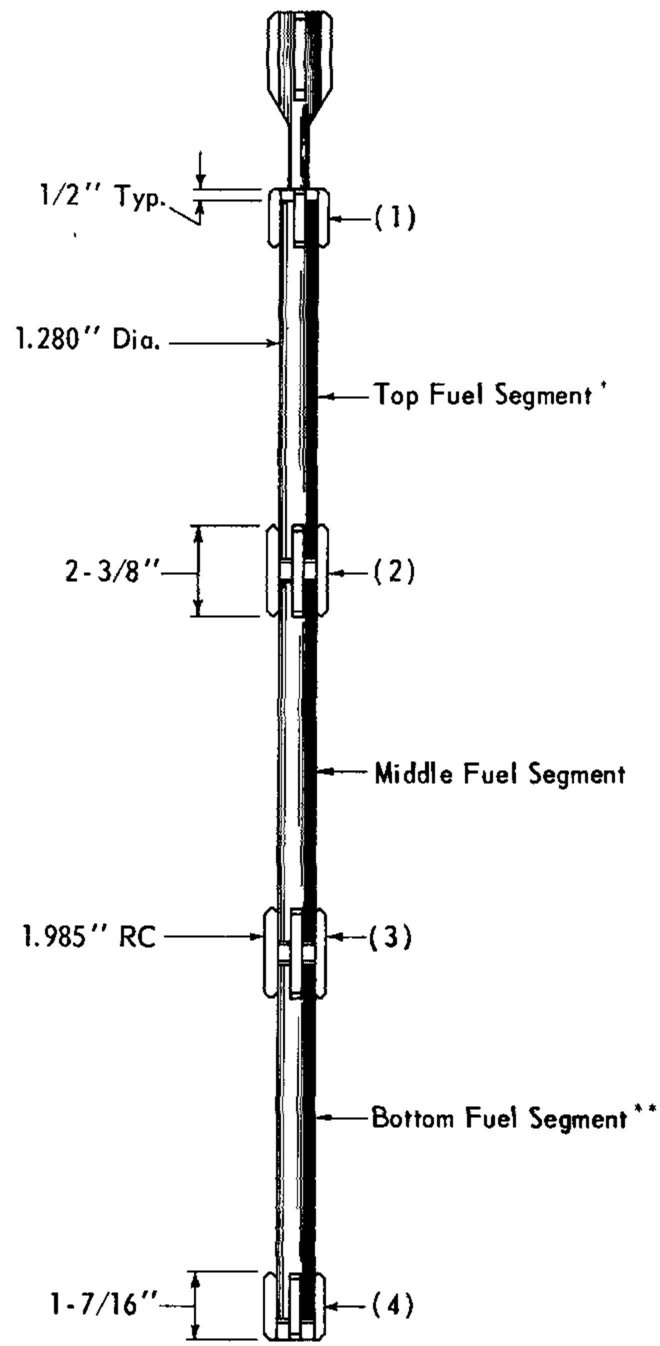


Weld produced with electron beam 0.011 inch away from center of rib



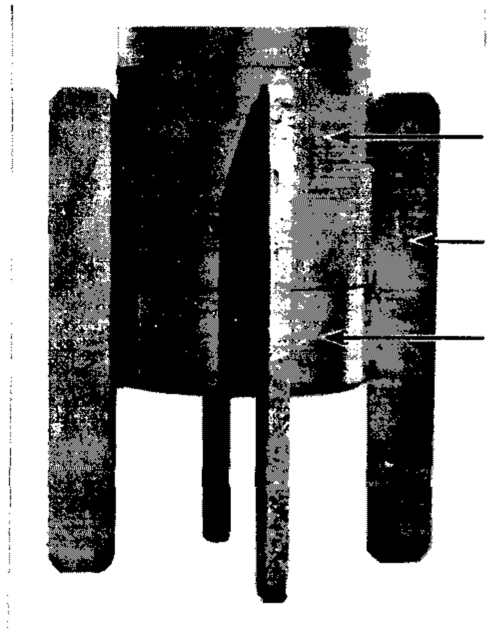
Weld produced with electron beam aligned with center of rib

FIG. 3 SECTIONS OF WELDED RIBS ON ZIRCALOY HOUSING TUBES



* Rib spacers (1) and (2) attached to top segment
** Rib spacers (3) and (4) attached to bottom segment

FIG. 4 DUMMY INNER FUEL COLUMN FOR MOCKUP OF OXIDE DRIVER ASSEMBLIES



DPSTF 1-8558-5 1X

FIG. 5 RIBBED SPACER USED IN FLOW TEST OF OXIDE DRIVER MOCKUP

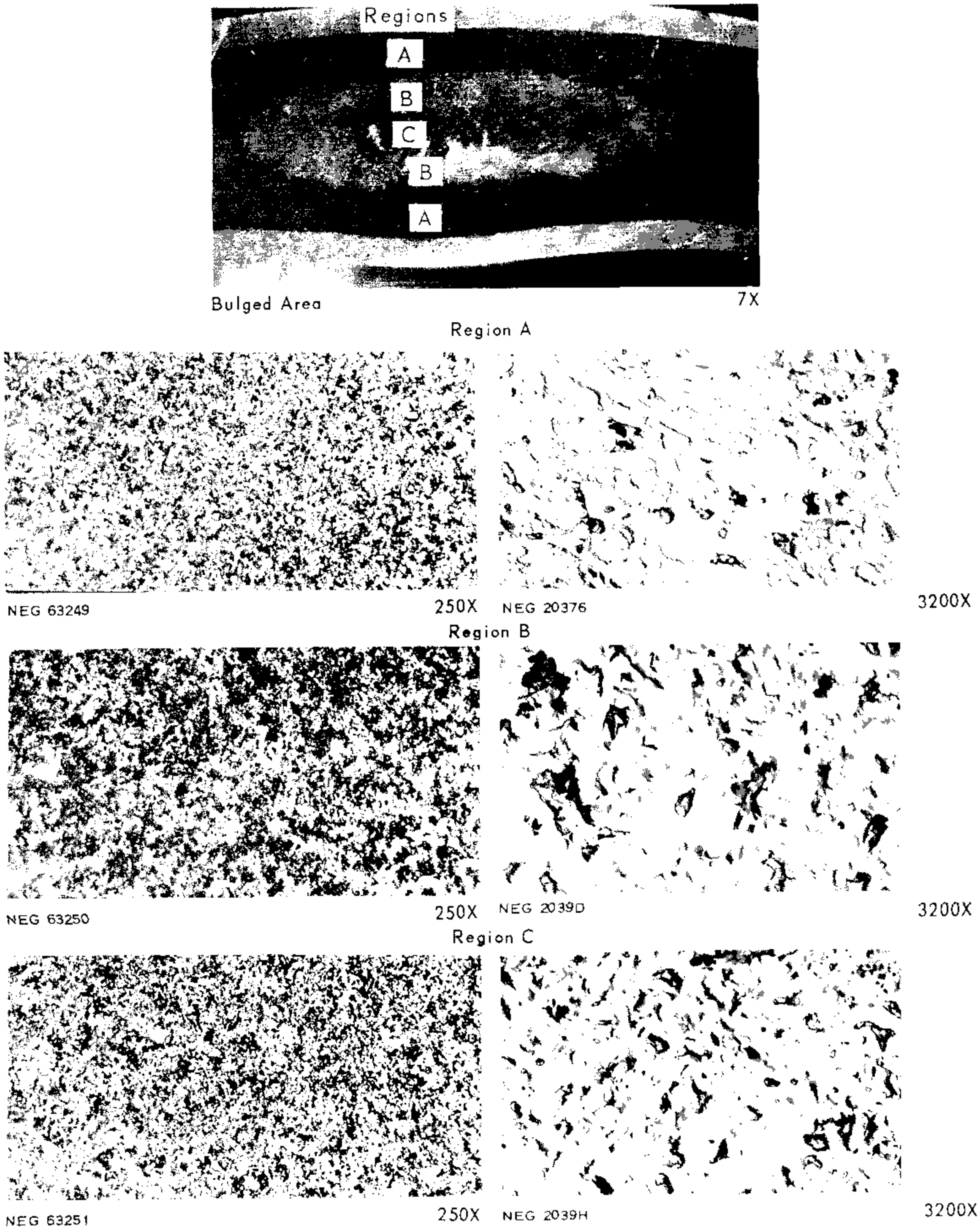
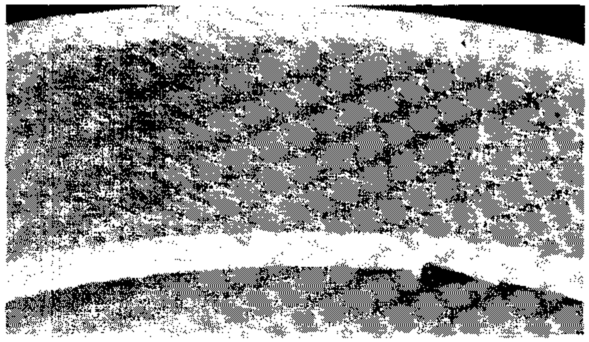
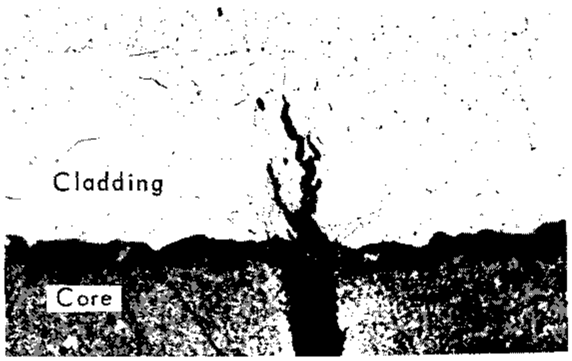


FIG. 6 MICROSTRUCTURE IN REGIONS OF SEVERE SWELLING OF UNALLOYED URANIUM TUBE



7X

a. Radial core crack

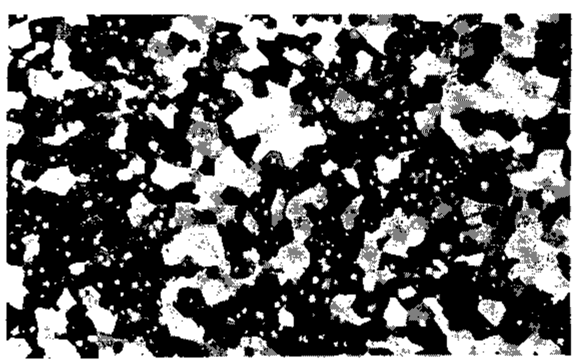


NEG 63235 250X

b. Tip of crack

NEG 63231 250X

c. Typical hydride content of cladding



NEG 63260 250X

d. Typical grain structure of cladding

FIG. 7 MICROSTRUCTURE OF ZIRCALOY-2 CLADDING FROM IRRADIATED TUBE OF UNALLOYED URANIUM

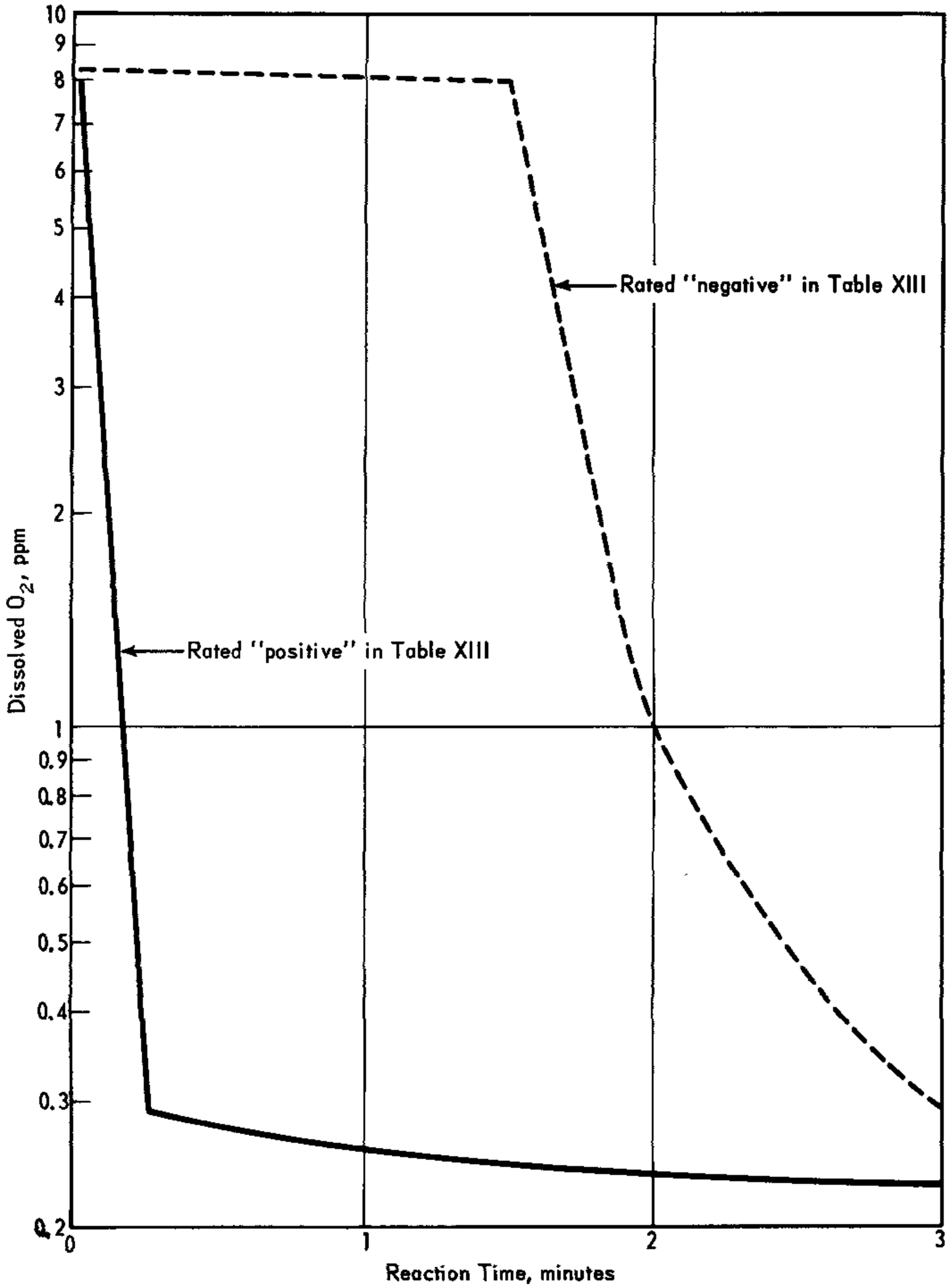


FIG. 8 TYPICAL REACTION RATES BETWEEN CATALYZED SODIUM SULFITE AND DISSOLVED OXYGEN

BIBLIOGRAPHY

1. C. L. Angerman and G. R. Caskey, Jr. "Swelling of Uranium by Mechanical Cavitation." J. Nucl. Mater. (to be published in 1965).

2. Previous progress reports in this series are:

DP-232	DP-395	DP-485	DP-575	DP-665	DP-755	DP-845
DP-245	DP-405	DP-495	DP-585	DP-675	DP-765	DP-855
DP-265	DP-415	DP-505	DP-595	DP-685	DP-775	DP-865
DP-285	DP-425	DP-515	DP-605	DP-695	DP-785	DP-875
DP-295	DP-435	DP-525	DP-615	DP-705	DP-795	DP-885
DP-315	DP-445	DP-535	DP-625	DP-715	DP-805	DP-895
DP-345	DP-455	DP-545	DP-635	DP-725	DP-815	DP-905
DP-375	DP-465	DP-555	DP-645	DP-735	DP-825	DP-915
DP-385	DP-475	DP-565	DP-655	DP-745	DP-835	DP-925
						DP-935