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FINAL SURVEILLANCE OF
EBWR PRESSURE VESSEL FOLLOWING
THE PLUTONIUM RECYCLE PROGRAM

by

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I. RETROSPECT

In December 1964 the decommissioned EBWR primary system was nondestructively inspected to determine its suitability for operation in conjunction with the AEC Plutonium Recycle Program. The inspection included visual examination of the interior, arc-deposited Type 304 stainless steel-clad surfaces of the flat closure head and conical ring forging of the cylindrical, carbon-steel (SA-212-B) reactor pressure vessel. (The ring forging is machined to accommodate 44 head bolt studs.) No visible defects were noted on the surfaces of these components.

Similar examination was made of immediately accessible surfaces of intermittent-welded stainless steel panels (each 32 in. wide, 110 in. long, and 1/10 in. thick), which comprise the cladding below the ring forging. In this instance, numerous surface "decorations" were observed. As a result of these initial observations, the steam duct and shock shield were removed from the vessel for greater accessibility. In addition, a shielded platform was suspended from the bolt studs to facilitate comprehensive destructive investigation of the "decorated" panels and adjoining panels representing the upper half of the vessel cladding. Investigation of the lower half was precluded by the high radiation levels and the thermal shield.

ANL-7117 describes the nature, results, and conclusions of this investigation in detail.¹ Briefly, initial application of a foaming detergent on the panel surfaces and introduction of nitrogen at 800 psig between the panels and the vessel wall revealed many of the "decorations" to be through cracks. Moreover, cracks were found in five of the eight panels comprising the upper course of cladding. Over 75% of the cracks were in the steam zone and were clustered between the weld nuggets adjacent to the arc-deposited welds joining the panels. Progressively less cracking was observed in the water zone. Follow-up dye-penetrant tests disclosed very superficial cracks in two of the "good" panels, and none in the third. Similar tests were made on the cladding of the closure head and of the upper ring forging; no cracks were observed.

Subsequent destructive examinations were made of samplings from the panels to determine whether the cracks propagated into, or were present in, the SA-212-B vessel wall, and to establish, if possible, the mechanism of cracking. These examinations included:

1. Exploratory grinding and intermittent dye-penetrant testing of the cracks. Approximately 40 linear feet of cracks were ground out to the cladding-vessel wall interface. After the vessel surface was carefully cleaned, grinding and dye-penetrant testing was continued to a depth of 0.010-0.015 in. In some instances, the panel cracks appeared to have originated at the interface; in others, the origin was traced to a point remote from the interface, such as the weld nugget. In the latter cases, the cracks also could have resulted from a changing stress pattern in the cladding as the material was removed. In any event, there was no evidence of cracking of the SA-212-B base metal or parting in the resistance welds bonding the cladding to the vessel wall.

2. Metallographic examination, chemical analysis, and mechanical tests on boat samples ($2\frac{1}{2}$ in. long, $1\frac{1}{2}$ in. wide, $1\frac{1}{8}$ in. deep) of the clad vessel, and strip samples (8 in. long, $1\frac{1}{2}$ in. wide, $1\frac{1}{10}$ in. thick) of the cladding. In addition, scrapings were taken from the interfaces to determine the chloride content of the deposit. For comparison, similar samples were cut from a 2-in.-thick cladding test plate and from 4-in.-thick remnants of the base metal used to fabricate the vessel. One remnant had been exposed to the reactor environment for 11,225 hr prior to its removal in 1960. The test plate and other remnants had been stored outdoors since 1959.

Metallographic examination of the vessel samplings showed that all cracks in the strip samples and cladding on the boat samples were intergranular, except for a few hot tears inside the weld nuggets. In all instances, the microstructures and Strauss reagent tests showed that the stainless steel was severely sensitized. None of the internugget cracking propagated across the unbounded interface into the base metal. In the bonded region, the cracks were stopped by the fusion boundary of the kernel. Microcracking of the kernel was contained inside the dendritic structure of the kernel.

Similar examination of the base-metal samples revealed several microcracks up to 0.008 in. deep. Several of the deeper microcracks were filled with Type 304 stainless steel intrusions. Comparisons with microstructures of remnant specimens showed conclusively that the microcracks existed before the cladding was applied and were probably formed during initial fabrication of the plate. Other than the stainless steel intrusions, there was no evidence of propagation of the surface microcracks as a result of vessel fabrication, cladding process, or subsequent reactor operation.

Spectrochemical analysis of the cladding samples showed neither abnormal concentrations of residual elements (Mo, W, Ti, Al, and Cu), nor deleterious amounts of the heavy elements (Pb, As, Sn, Bi, and Sb). Except for a slightly higher sulfur content (0.035 vs 0.030%), chemical analyses showed that the cladding met ASME Boiler and Pressure Vessel Code materials requirements.

Scrapings from the exposed vessel wall and interface side of the cladding were identified as Fe_3O_4 . There was no evidence of local pitting or of gross generation of Fe_2O_3 . Analysis of the Fe_3O_4 scale showed ppm chloride ion concentrations; however there was no evidence that the interface chlorides participated in the clad-cracking process.

Evaluation of the information available, and consideration of the fracture mechanisms, radiation effects, temperature, and corrosion phenomena, led to the following conclusions:

1. Initial cracking within the cladding most probably was caused by thermally induced stress ruptures during vessel fabrication. These unsurfaced cracks propagated by low-cycle fatigue during reactor operation.
2. The mechanism of cladding failure would not be different in the inaccessible lower regions of the vessel.
3. Regardless of the mode of fracture, we believe that cracks in the cladding will not propagate into the vessel wall.
4. Continued operation of the reactor would cause more cracks to surface, and efforts to repair existing cracks or sampled areas would accelerate the cracking mechanism.
5. The cracking of the cladding did not significantly impair the integrity of the vessel.
6. From a corrosion standpoint, the total exposed surface of the base metal was less than 0.5% of the internal vessel area. In 500°F oxygenated water (the proposed operating conditions for the Plutonium Recycle Program). SA-212-B corroded at 400 mg/(dm²)(mo), or about 0.0025 in./yr. Since the thickness of the wall in the exposed areas is 9/16 in. greater than that required by the Vessel Code, prolonged operation with this corrosion rate could be tolerated.

In support of these conclusions, a surveillance program was implemented to provide information on changes incurred in the cladding and the exposed vessel wall during EBWR operation with a plutonium-fueled core. Preparations for this program included:

1. Grinding sharp edges and notches in all sampled areas to reduce stress concentrations.
2. Selection of four reference areas for surveillance; more specifically, as shown in Fig. 1, one in the water zone of two panels (No. 1 and 5) with essentially no surface cracks, and one in the steam zone of two panels (No. 3 and 8) with ground-out cracks and exposed vessel wall.
3. Establishment of reference points in the exposed wall surfaces (to be measured with an optical micrometer) to determine the extent of subsequent corrosive attack.
4. Installation of removable "windows" in the shock shield to facilitate access to these surveillance areas.
5. Provisions for periodic monitoring of coolant-water purity, pH, gross activity, and entrained corrosion products during reactor operation.
6. Fabrication and installation within the vessel of five helium-filled irradiation capsules containing 75 front-chamfered Charpy vee-notch specimens machined from the quarter-plate-thickness regions of a 4-in.-thick SA-212-B steel remnant. The objective was to follow irradiation damage of the steel by removal and examination of specimens after various exposures.

Fig. 1. Orientation of EBWR Reactor-vessel Cladding Panels, Showing Location of Surveillance Areas

Loading of plutonium fuel subassemblies into the EBWR vessel started in September 1965. The loading comprised 36 central subassemblies,

each containing fuel pins fabricated from a homogeneous mixture of 1.5 w/o PuO_2 (8 a/o ^{240}Pu) and depleted UO_2 . Extending radially outward, the balance of the loading consisted of 60 shim subassemblies (two rows) containing fuel pins of UO_2 enriched to 6% in ^{235}U , 49 outer subassemblies containing natural UO_2 fuel pins, and an antimony-beryllium source. Each subassembly had an active height of 4 ft and contained 36 fuel pins (0.372-in. OD) clad in Zircaloy tubing (0.025 in. thick).

The nature and results of the experiments that followed are not reported here. Briefly, they included measurements of capture-to-fission ratios of ^{238}Pu and ^{241}Pu , control-rod worths, boric acid worths, temperature coefficients of reactivity, shutdown margins, and verification of eight- and nine-rod shutdown criteria for the full loading.

During these experiments, the EBWR vessel and cladding accumulated 4,830 hr exposure to the following operating environments:

Power, MWt	44 (avg)
Pressure, psig	600
Temperature, °F	489
Water conditions	
Total recirculation flowrate, lb/hr	7.0×10^6
pH	6.9-7.2
Resistivity, ohm-cm	$2-1 \times 10^6$
Oxygen content, ppm	
Reactor water	0.24
Steam	28.0
Maximum nvt (>1 MeV)	$\sim 3 \times 10^{17}$

Upon termination of the experimental program in July 1967, the irradiation test capsules of SA-212-B specimens were removed to a high-level cave for destructive examination; the plutonium core and the control rods were transferred to the EBWR storage pit; and the water level in the vessel was raised above the core shroud to facilitate inspection of the previously documented surveillance areas. The rest of this report describes the findings and conclusions of the vessel surveillance program.

II. POSTEXPERIMENT SURVEILLANCE

Initial examination and measurements of the vessel surveillance areas were carried out in a high-radiation background (~ 350 mR in the open space of the steam zone) and under cramped conditions; only a partially shielded platform was available for suspension from the vessel head bolts. These conditions were subsequently alleviated somewhat by using a fully shielded platform suspended from the building crane. Nonetheless, radioactivity levels in the vessel above the water fill line were such that personnel were limited to individual exposures of about 15 min.

A. Corrosion

1. Optical Micrometer Measurements

Corrosion-depth readings of exposed vessel-surveillance areas were taken with the optical micrometer shortly after unloading the core in July 1967, and in November and December. In each instance, the micrometer was mounted on a rectilinear bridge capable of travel along both coordinates. Readings were taken by positioning four conical feet of the bridge into four punch marks outlining the area to be traversed by the micrometer. In the water zone (Panels No. 1 and 5), the bridge was held in place by being clamped to the shock shield window; one technician made these measurements, as shown in Fig. 2. Measurements in the steam zone (Panels No. 3 and 8) required two technicians: one to position the bridge against the punch marks, the other to operate the micrometer.

Fig. 2. Clamped to Window in Shock Shield, Optical Micrometer-Rectilinear Bridge Assembly Being Used to Measure Corrosion of Exposed Vessel Wall in Water Zone (Panel No. 5)

Table I lists the values recorded in this manner. Also listed are the pre-experiment values. By comparison, the initial readings (July 1967) in the steam zone (Panel No. 3) indicated an apparent metal loss of 0.025 in. As mentioned earlier, these readings were taken under cramped conditions (partially shielded platform). After installation of the crane-operated platform, similar measurements showed a corrosion-product buildup of about 0.002 in. The latter value is believed to be more

reliable, since no significant measurable changes were incurred after wiping the areas with tissue and scrubbing the locating points lightly with glass wool. The discrepancy between the July readings and the later ones can be attributed to difficulties in making the preliminary measurement; particulate matter lodged in the punch marks; failure to engage the marks; improper seating of the bridge feet in the marks; or some combination of factors arising during the short time available to perform the measurements.

TABLE I. Optical Micrometer Readings^a of EBWR Vessel Wall Surveillance Areas before and after Plutonium Recycle Experiment (in inches)

Panel No.	Bridge Coordinates		Pre-expt July 1966	July 1967		November and December 1967					
	Vert	Horiz		Not Cleaned	Diff	Not Cleaned	Diff	Wiped	Diff	Scrubbed	Diff
Steam Zone											
3	Max	0	0.628	0.654	-0.024	0.627	+0.001	0.627	+0.001	0.629	-0.001
		1	0.650	0.675	-0.025	0.645	+0.005	0.648	+0.002	0.648	+0.002
		2	0.681	0.705	-0.024	0.676	+0.005	0.677	+0.004	0.678	+0.003
		3	0.702	0.728	-0.026	0.701	+0.001	0.701	+0.001	0.702	0.000
		4	0.719	0.743	-0.024	0.717	+0.002	0.717	+0.002	0.717	+0.002
8	Max	0	0.803			0.787	+0.016	N.G.	-		
		9	0.811			0.791	+0.020	0.792	+0.019		
		10	0.819			0.794	+0.025	0.792	+0.027		
		11	0.831			0.803	+0.028	0.792	+0.031		
Water Zone											
1	22	4	0.657			0.683	-0.026	0.723	+0.036		
		5	0.699			0.685	+0.014	0.728	+0.037		
		6	0.719			0.701	+0.018	0.789	-0.004		
		7	0.737			0.732	+0.015	0.703	+0.003		
5	20	13	0.759			0.773	-0.014				
		14	0.765			0.775	-0.010				
		19	0.785			0.795	-0.010				
		20	0.736			0.749	-0.013				

^a+ indicates buildup of corrosion products.

- indicates loss of metal.

Initial readings on Panel No. 8 in the steam zone indicated a corrosion-product buildup of about 0.022 in. After wiping, repeat readings showed little change. The inconsistencies in the steam-zone readings have not been completely explained.

On the other hand, measurements on Panels No. 1 and 5 in the water zone showed substantial buildup of corrosion products. This was expected and was confirmed to some extent by subsequent analyses of the products. The anomalous reading in Panel No. 1 (V-22, H-4) was not investigated further because of the relatively high radiation level and the extremely cramped space in which the panel is located.

2. Analyses of Corrosion Products

Spectrographic analyses were made of scrapings of vermillion-colored deposits observed on the exposed carbon steel vessel surfaces in the water zone, and on brown deposits from exposed areas in the steam zone. Table II lists the results of the analyses.

TABLE II. Results of Spectrochemical Analyses of Corrosion Products on Exposed SA-212-B Surfaces in Steam and Water Zones of EBWR Vessel

Element	Steam Zone	Water Zone	Element	Steam Zone	Water Zone
Al	-	T	Mn	-	S
Cr	-	VS	Ni	W	S
Cu	M	M	Ti	-	S
Fe	VS	VS	Zr	-	VW

Symbol	Meaning	Approx Conc, %
VS	Very strong	100-10
S	Strong	10-1
M	Moderate	1-0.1
W	Weak	0.1-0.01
VW	Very weak	0.01-0.001
T	Trace	0.001-0.0001

As evidenced by the table, the steam-zone deposits contained only iron, nickel, and copper oxides, whereas deposits in the water zone also contained aluminum, manganese, chromium, titanium, and zirconium oxides. Thus, as would be expected, deposits in the steam zone were almost entirely formed "in situ," but those in the water zone include a substantial portion of corrosion products entrained by the coolant from other areas of the primary system. Other than titanium, no unusual elements were detected; its presence was unexpected, and the source was not determined.

B. Cladding Cracking

Visual examination of the accessible cladding surfaces revealed no significant change in the cracking pattern around the weld nuggets from that described in ANL-7117.

Dye-penetrant inspection of the vessel wall surfaces in the surveillance areas (see Fig. 3) showed that no material changes were incurred during the experiment.

In ANL-7117, it was reported that the cladding was more severely cracked in the 2-in.-thick "welding practice" plate, and the cracking mechanism was different from that observed on strip samples of cladding removed from the vessel. The difference has been resolved by a review of mill test reports on the cladding used for the practice plate and the vessel proper.

Fig. 3. Dye-penetrant Appearance of EBWR Vessel Surveillance Areas in Steam and Water Zones after Plutonium Recycle Experiment

As summarized in Table III, the cladding for the practice plate was significantly hardened by cold-rolling at the mill before delivery to the vessel fabricator for (1) welding to the SA-212-B backing plate, and (2) subsequent rolling to a cylindrical sector ($R = 42$ in.). The latter operation involved additional forming and stress-relieving at 1150-1200°F. This additional working resulted in the severe sensitization of the stainless steel. Thus the probable causes of initial cracking of the vessel cladding advanced on p. 6 of this report still hold.

TABLE III. Summary of Mill Test Reports on Cladding
for EBWR Vessel and for Practice Plate

Vendor	Heat No.	Mill Invoice Date	Hardness, R_B	Specification, Grade, Finish
Allegheny-Ludlum ^a	334053	5/19/55	75/77	Allegheny Metal 18-8S; Type 304; No. 1 Finish.
Republic ^b	H-26800	10/23/57	82	SA240, Gr. S; Enduro 18-8-S; Type 304 CR Sheets; No. 4 Finish, one side polished.

^aEBWR vessel cladding.

^bPractice plate cladding.

On p. 37 of ANL-7117 it was observed that incremental bending (in a vise) of samples of the composite 2-in. practice plate containing superficial internugget cladding cracks resulted in enlargement of the cracks without propagation across the interface into the base metal. (Also see Fig. 10 of ANL-7117.) This observation was confirmed by more sophisticated bend tests upon conclusion of the Plutonium Recycle Experiment. These tests were made on six curved beam samples ($12\frac{1}{8}$ in. long, $1\frac{1}{2}$ in. wide, $\frac{3}{8}$ in. thick) from the same plate. Figure 4 shows two typical samples.

Fig. 4. Final Appearance of Bend Specimens from 2-in.-thick Practice Plate of Type 304 Clad SA-212-B Base Metal. Dye penetrant reveals that, in tension, superficial surface cracks (A) are arrested by thumb print weld nuggets, (B) propagate between, or (C) detour around, nuggets to interface (D), but do not extend into the base metal.

In all cases, the cracks propagated, in tension, through the cladding between the resistance-welded "thumbprints." However, none of the cracks extended into the SA-212-B base metal, although the strain was beyond the yield point, i.e., 1% total. All beams were permanently bent.

Calculations based upon uniform beam sections showed the beam strength was variable and intermediate between values for an unclad section and for a clad, full-thickness member. Variable load-carrying capacities of the beams, at the elastic yield strength, were influenced by the presence of previous cladding cracks. Bend-test results are summarized in Table IV.

TABLE IV. Summary of Load-carrying Capacities of Composite Test Beams at 50,040 psi^a

Sample No.	Beam Dimensions, in. ^b		Load Capacity, lb		
	Width	Thickness	Calculated	Measured	% Calc
1	0.522	0.387	216	187	86.7
2	0.554	0.385	228	164	75.2
3	0.495	0.388	206	165	80.1
4	0.520	0.377	204	168	82.4
5	0.520	0.384	212	167	78.8
6	0.552	0.353	190	c	-

^aYield strength of SA-212-B practice plate.

^bIncludes cladding thickness of 0.109 in.

^cAccidentally overbent in fatigue-test machine before bend test.

C. Radiation Effects

Three irradiations designed to provide data pertinent to the EBWR vessel surveillance program were completed between spring 1959 and termination of EBWR operation in 1967. Two irradiations were conducted with uranium-fueled cores, and one with the plutonium core. Analysis of the data indicated that the vessel wall had not been significantly damaged by radiation during its operational life.

1. Wire and Foil Activations

These activations represented the first irradiation performed with the 48-in., 20-MW uranium core in 1959. The objective was to establish the neutron-flux profile outside the pressure vessel and the character of damaging radiation. To this end, bare wires (20 gauge) of Au, Ag, Ni, Cu, Ta, and Pt and bare and cadmium-covered foils of gold, ²³⁸U, and ²³⁵U were taped to a flat aluminum slat. This slat was suspended in a flattened aluminum tube, which was inserted into the reactor cavity between the vessel outer insulation and the cavity steel liner (see Fig. 5). The reactor was then operated for 24 hr at an average power level of 17.4 MW. Upon completion of the exposure, the reactor was shutdown for conversion to 100-MW capability.

Fig. 5. Location of EBWR Vessel Irradiation-surveillance Facilities. Also shown is the core loading for the Plutonium Recycle Experiment.

Subsequent gamma scans along the lengths of the wires indicated an assymetric flux profile incident on the vessel wall. Except for the platinum wire, the scans were made on the principal characteristic gamma line of the element. For the platinum wire, an adjoining gamma peak of

lower intensity was used to preserve counting efficiency. As shown in Fig. 6, the activation profiles for five of the six wires are similar. The apparently greater activation of the platinum wire is attributed to the scanning procedure, which failed to discriminate contributions of a nearby gamma peak (of the active platinum gamma-radiation spectrum) to the total count rate.

Fig. 6. Bare-wire Activations Outside EBWR Vessel Wall

Data from the bare and cadmium-covered gold foils revealed that the cadmium ratio was 1.09 at the 4.9-eV resonance peak, i.e., a very hard neutron spectrum. Fission-fragment analyses of the ^{235}U and ^{238}U foils for ^{99}Mo also showed a predominantly fast fission spectrum. Respective fast-flux values were 3.2×10^8

and 3.1×10^9 . These experimental values are in good agreement with those for the gold foils. Also, the value from the ^{238}U foil agrees well with the calculated value of 8×10^9 reported in ANL-5544² for a 48-in. core without thermal shielding.

The "hard" nature of the incident neutron flux is caused by the shielding installed inside and outside the vessel wall. On the core side, the vessel wall is screened by a 1.2 w/o natural boron-stainless steel thermal shield 1.10 in. thick. On the reactor-cavity side, the biological shielding includes a 1/2-in.-thick layer of ferro-boron concrete containing ~18 w/o natural boron.

2. SA-212-B Izod Impact Specimens

The second irradiation involved round, unstressed, miniature, multinotched Izod impact specimens. All specimens were contained inside Type 304 stainless steel capsules initially pressurized with helium to 500 psig. These capsules were loaded into newly installed bare and shielded Type 304 stainless steel irradiation test tubes located in the peak thermal-flux region of the reflector, approximately 6 in. from the edge of the 60-in.-diameter, uranium-fueled, 100-MW core. The shielded tubes featured a concentric sleeve of 2 w/o natural boron-stainless steel (24 in. long, 1/2-in. wall). All tubes were dimpled at the lower end to retain the capsules, yet permit natural circulation of the primary coolant water around the capsules. The capsules were loaded from the top of the reactor vessel. Similarly encapsulated control specimens were loaded into the steam duct (13.5 ft above the core) to monitor the thermal effects.

The specimens were irradiated to a calculated integrated exposure of 1.34×10^{18} nvt ($E > 1$ MeV) during subsequent reactor operation at an average power of 18.6 MW. Postirradiation tests³ revealed that no significant damage was incurred by the exposure.*

3. SA-212-B Charpy Vee-notch Specimens

The final vessel irradiation surveillance was made on modified (front edges chamfered 0.015/0.020 in.) Charpy vee-notch specimens during the Plutonium Recycle Experiment. Four capsules, each containing 15 specimens and pressurized with helium to 500 psig, were loaded into the irradiation tubes shown in Fig. 5. Control specimens were similarly loaded into Capsule No. 4 (not shown). As for the Izod specimens, this capsule was installed in the steam duct (~13.5 ft above Capsule No. 3). The objective here was to establish the long-term thermal effects at 489°F during the 4,830-hr irradiation test period. Capsule No. 4 was essentially unirradiated by neutrons. The rest of the capsules accumulated an estimated exposure of 8.3×10^{18} nvt ($E > 1$ MeV) during subsequent reactor operations, totaling 4,830 hr at an average power level of 44 MWt.

*Calculated exposure not available at time of publication of Ref. 3.

Differences between the radiation damage suffered by the "bare" and "shielded" specimens were obscured by inaccuracies in the impact-testing procedures. Therefore, all results were combined into a band representation as shown in Fig. 7. Three gross results are apparent:

- a. An upward shift of the NDT ($\sim 125^{\circ}\text{F}$) at the 15-ft-lb level (between the controls and irradiated specimens).
- b. A concurrent depression of 30-35% in the notched-bar impact strength at high temperature.
- c. An inconsistent enhancement of the impact strength of the control specimens.

Fig. 7. Effect of EBWR Plutonium Core Irradiation on Impact Strength of SA-212-B Charpy Vee-notch Specimens. Also included are data from three impact-test-machine calibrations with an unirradiated, 4-in.-thick SA-212-B plate.

Figure 7 includes an additional band of data taken during impact-test-machine calibrations from a 4-in.-thick plate of unirradiated SA-212-B material. The objective here is to resolve the long-term effects of out-leakage of helium gas and inleakage of water (see Table V) on the impact strength of the control specimens. A comparison of both groups of unirradiated specimens shows that the impact strength of the controls increased during the exposure. This increase may be attributed to either long-term

TABLE V. Relative Hardness of Irradiated Charpy Vee-notch Impact Specimens

Capsule No.	Final Capsule He Pressure, psig ^a	Environmental Composition, %	Rockwell B Hardness		
			Avg	Min	Max
1	42	99.9 He	91.0	85.7	92.4
2	494	99.9 He	89.9	88.8	90.8
3	509	99.9 He	92.1	90.2	94.0
4	Steam	N.D. ^b	80.2	78.2	89.2
5	475	99.4 He; 0.4 N ₂ ; 0.1 O ₂	90.6	89.5	92.1

^aInitial helium pressure = 500 psig.

^bNot determined; contained water drops.

annealing of residual machining stresses or decarburization by water vapor. The latter process is suspected because the controls were softer and their surfaces were coated with a uniform, dense-black oxide.

Capsule No. 1 also lost the bulk of its initial helium charge. Based on volumes of recovered gas, calculations indicated a residual capsule pressure of 42 psig; the capsule may have overheated during the irradiation period. However, impact-test results from specimens in this capsule were not markedly different from those irradiated in fully pressurized capsules.

As evidenced by Table V, nominal hardening of the SA-212-B specimens occurred during the irradiation cycle. Except for Capsule No. 4, the average hardness values are based on three readings on each of 30 pieces of specimens from each capsule after impact tests. For Capsule No. 4, the average is based on one reading on 12 pieces from six Charpy test specimens. In this instance, the lower hardness values are attributed to long-term annealing and/or decarburization by water vapor.

From the results of these irradiations, which also establish the belt-line environment, we concluded that the vessel has not been damaged by the accumulated neutron dosage. The significant maximum/minimum limits are:

- a. Integrated vessel dosage (from shielding calculations and foil irradiations at 17.4 MW): $8.3 \times 10^{17} / 3.2 \times 10^{17}$ nvt ($E > 1$ MeV).
- b. Izod impacts dosage (from 100-MW EBWR core irradiation at 18.6 MW): 1.34×10^{18} nvt ($E > 1$ MeV).
- c. Estimated Charpy impacts dosage (from plutonium-core irradiation at 44 MW): 8.3×10^{18} nvt ($E > 1$ MeV). Manganese-54 determinations were made on drill chips from the surfaces and centers of eight broken Charpy vee-notch bars to verify the final integrated dosage. Based on a fast cross section of 86.5 mb for the $^{54}\text{Fe}(n,p)^{54}\text{Mn}$ reaction, the determinations indicated a dose of 8.5×10^{18} nvt ($E > 0.1$ MeV).

Thus the vessel dosage is probably more than one order of magnitude below that at which measurable damage (impairment of impact properties) was detected by the irradiation surveillance program for the EBWR pressure vessel.

III. CONCLUSIONS AND DISCUSSION

Neither the visual observations made nor the data obtained during postexperiment surveillance of the EBWR vessel and cladding give cause to change the conclusions in ANL-7117. In fact, additional information on the

fabrication history of the 2-in.-thick practice plate supports the hypothesis for the most probable cause of cladding cracks advanced in that document.

Visual examination and optical micrometer measurements of the bared SA-212-B areas indicate that no accelerated corrosion of the base metal occurred. Anomalies in the measurements are probably due to imperfect "seating" of the rectilinear bridge feet in the punch indentations or to slight differences of micrometer position on the x-y axis. The surveillance areas traversed were not previously machined or polished; hence elevation differences of several thousandths of an inch could exist within very short distances. Moreover, the rather high radiation level (15-min exposure, max) in the water zone precluded extensive repeat measurements.

A quantitative assessment of irradiation-damage effects to the EBWR vessel belt-line section has been afforded by two accelerated in-vessel irradiations of SA-212-B samples. The first, which consisted of round, multinotched Izod specimens during operation at 100 MW with a uranium-fueled core, revealed no measurable damage by the 1.3×10^{18} nvt ($E > 1$ MeV) exposure. The second, conducted on Charpy impact specimens during the Plutonium Recycle Experiment, revealed two simultaneous irradiation effects at the estimated 8.3×10^{18} nvt ($E > 1$ MeV) exposure: (1) an increase of $\sim 125^\circ\text{F}$ in the transition temperature, and (2) a 30-35% reduction in the high-temperature impact resistance. No deleterious effects were observed in the thermal control specimens exposed in the steam zone of the vessel. From these two irradiations, we concluded that the vulnerable part of the EBWR vessel (belt-line) has not been degraded by the 8.3×10^{17} nvt ($E > 1$ MeV) exposure accumulated during its operational lifetime.

Although results from the EBWR vessel surveillance program supported the conclusion that integrity of stainless steel cladding applied by resistance welding, submerged-arc process, and manual weld overlays is adequate, continued surveillance of all such vessels is adjudged mandatory. Details of any surveillance program must be established on an individual basis.

The EBWR surveillance results probably are not applicable to metallurgically roll-bonded, wrought-clad vessel steels; however, a small amount of such composite material is used in the lower-head nozzle stubs. These stubs are exposed to very low stresses and are located in a region of low neutron flux. Impact tests on the base plate before fabrication established that it was as notch-resistant as the SA-212-B base metal from which the vessel was fabricated.

Viewed in retrospect, the pre- and post-Plutonium Recycle Experiment in-vessel examinations, of necessity, differed in part because of the unexpected high radiation backgrounds. For example, unlike the pre-experiment examination residence time of 4 hr, individual exposures were limited to 15 min. With respect to procedures, hindsight would reflect the following modifications:

1. Reference measuring points should be prepared in the x, y, z directions for location of the conical feet of the rectilinear bridge. The optical micrometer mounted thereon is capable of measuring depths to ± 0.0001 in. To avoid anomalous readings, the locating points should be drilled holes, with a fixed radial chamfer, say $1/16$ in. Preferably, these holes should be less in depth than the cladding thickness and of a diameter that permits positive seating of the cones on the radius. If the cylindrical portion of the cladding is intermittent spot-resistance-welded, as in EBWR, the holes should be drilled in the nuggets to avoid "springing" under the locating load.

2. Reference locations should be provided for cameras and lighting so that photographic comparison of pre- and postoperation surfaces can be ensured. In addition, lighting intensity should be the same, as should the focal length of the camera lenses employed.

3. Additional shielding should be provided to extend personnel access time for the resolution of measurement anomalies.

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REFERENCES

1. N. Balai, C. R. Sutton, E. A. Wimunc, and R. F. Jones, Inspection, Evaluation, and Operation of the EBWR Reactor Vessel, ANL-7117 (Nov 1965).
2. M. Grotenhuis and J. W. Butler, Experimental Boiling Water Reactor (EBWR) Shield Design, ANL-5544 (Aug 1956), Fig. 5.
3. N. Balai, Impact, Magnetic, and Resonance Properties of Irradiated SA-212-B Pressure Vessel Steel, Trans. ANS 7, No. 1 (1964), p. 116.