AD/A-003 457

SURVEILLANCE PROGRAM FINDINGS FOR ARMY MH-1A REACTOR AT END OF CORE 3

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Naval Research Laboratory
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November 1974

NRL Memorandum Report 2942

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Engineering Materials Division

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SECURITY CLASSIFICATION OF THIS PAGE (When Data Entered)

REPORT DOCUMENTATION PAGE

READ INSTRUCTIONS BEFORE COMPLETING FORM

1. REPORT NUMBER	NRL Memorandum Report 2942
2. GOVT ACCESSION NO.	
3. RECIPIENT'S CATALOG NUMBER	AD/A-003 457
4. TITLE (and Subtitle)	SURVEILLANCE PROGRAM FINDINGS FOR ARMY MH-1A REACTOR AT END OF CORE 3
5. TYPE OF REPORT & PERIOD COVERED	Interim
6. PERFORMING ORG. REPORT NUMBER	
7. AUTHOR(s)	J. R. Hawthorne C. Z. Serpan, Jr. H. E. Watson
8. CONTRACT OR GRANT NUMBER(s)	
9. PERFORMING ORGANIZATION NAME AND ADDRESS	Naval Research Laboratory Washington, D.C. 20375
10. PROGRAM ELEMENT, PROJECT, TASK AREA & WORK UNIT NUMBERS	NRL Prob. 63M01-14 USAENPG MIPR 44017
11. CONTROLLING OFFICE NAME AND ADDRESS	Department of the Army (U.S. Army Eng. Power Group) Ft. Belvoir, Virginia
12. REPORT DATE	November 1974
13. NUMBER OF PAGES	18
14. MONITORING AGENCY NAME & ADDRESS (if different from Controlling Office)	
15. SECURITY CLASS. (of this report)	UNCLASSIFIED
15a. DECLASSIFICATION/DOWNGRADING SCHEDULE	

16. DISTRIBUTION STATEMENT (of this Report)

Approved for public release; distribution unlimited.

17. DISTRIBUTION STATEMENT (of the abstract entered in Block 20, if different from Report)

18. SUPPLEMENTARY NOTES

19. KEY WORDS (Continue on reverse side if necessary and identify by block number)

Embrittlement
MH-1A Reactor
Neutron dosimetry
Notch ductility
Pressure Vessel
Tensile properties
Radiation effects
Reactor surveillance

20. ABSTRACT (Continue on reverse side if necessary and identify by block number)

Charpy-V and tensile test results for five surveillance capsules removed from accelerated exposure, vessel wall, and above-core locations in the ARMY MH-1A reactor are presented. The capsules were originally installed in the reactor prior to its initial startup. Neutron flux and fluence determinations from a vessel wall neutron flux monitor assembly are also presented. A maximum fluence to the vessel of 8.46×10^{19} n/cm^2 >1 MeV is projected for a 20-year full power (45 MWt) operating period. (Abstract continues)

DD FORM 1473, 1 JAN 73 EDITION OF 1 NOV 65 IS OBSOLETE
S/N 0102-014-6601

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CONTENTS

20. (Continued Abstract)

Experimental findings indicate a retention of high fracture resistance by the vessel forging at the end of Core 3 operations.

SURVEILLANCE PROGRAM FINDINGS FOR ARMY MH-1A REACTOR AT END OF CORE 3

Background

Surveillance of the Army MH-1A reactor vessel (AISI Type 316 stainless steel) is being accomplished using specimen capsules and neutron flux monitor assemblies. The specimen capsules are placed in three locations: (a) adjacent to the vessel inner wall, (b) adjacent to the fuel core (accelerated exposure location), and (c) at an above-core location (thermal control). Neutron flux monitor assemblies are located adjacent to the vessel inner wall only. The flux monitor assemblies permit surveillance of the incident neutron flux to the vessel from a point about 3-in. below to a point about 42 in. above the fuel core centerline. The vessel upper circumferential weld is located approximately 8.5 in. above the fuel core centerline and thus resides within the region surveyed by the flux monitor assembly.

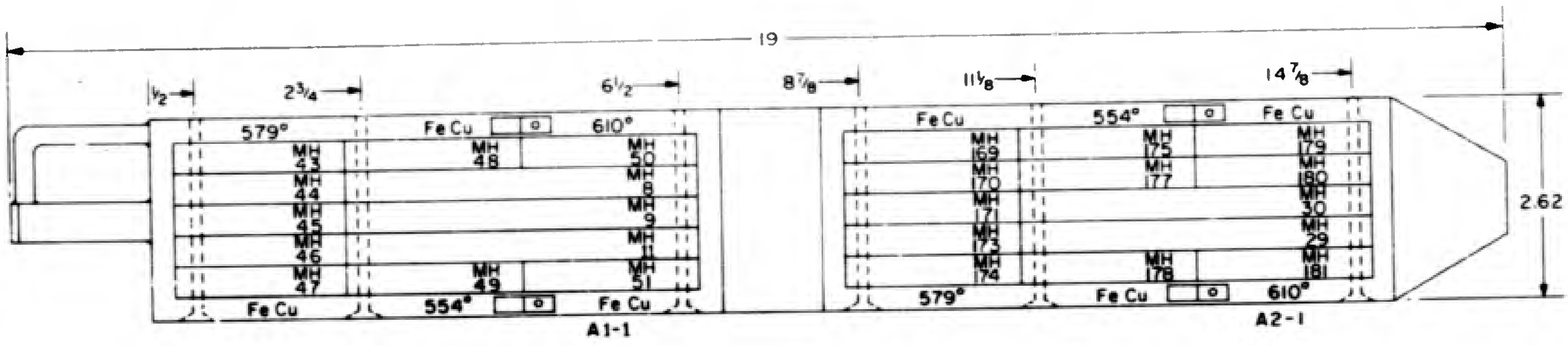
Details of the original MH-1A reactor vessel surveillance program, including specimen capsule locations and contents, are given in Ref. 1. Experimental findings from capsules and flux monitor assemblies removed during the Spring 1969 refueling outage (end of Core 1) and during the Spring 1971 refueling outage (end of Core 2) are documented in Ref. 2 and 3.

Progress

Postirradiation assessments have been completed for a third series of five specimen capsules and one flux monitor assembly discharged from the reactor at the end of Core 3 (Fall 1972 refueling outage). All five capsules were originally installed in the reactor prior to its initial startup in early 1967. In contrast, the flux monitor assembly was a replacement assembly installed during the Spring 1971 refueling outage and was exposed during Core 3 only. Specimen capsule loading diagrams are reproduced in Figs. 1-3 from Ref. 1. All specimens (Charpy-V and tensile) were of vessel forging material. Specimen capsules and the flux monitor assembly each contained iron and copper neutron dosimeter wires; however, neutron flux determinations were based on the analysis of the iron wires only ($^{54}Fe(n,p)^{54}Mn$ reaction). Neutron flux and fluence determinations are summarized in Tables 1 and 2.

Charpy-V (C_V) results for the accelerated exposure

Manuscript submitted October 31, 1974.



NOTES:

1. FLUX MONITORS AND TEMP. MONITORS WELDED IN STAINLESS STEEL TUBES
2. Fe WIRE (NRL'64); Cu WIRE (NRL'65) Co CONTENT ≤ 0.2 PPM
3. 554° HAS TWO LINES ON QUARTZ TUBE; 579° HAS NO LINES, 610° HAS FOUR LINES
4. SPECIMENS SECURED TO FRAMES BY 10-32 SLOT HEAD SCREWS WELDED TO FRAME
5. CHARPY SPECIMENS NOT NOTCHED
6. TENSILE SPECIMENS GAGE DIAM. 0.262 IN
7. MATERIAL-316 STAINLESS STEEL (NOZZLE CUT-OUT)
8. "V" NOTCH MUST BE CUT IN CHARPY BARS PERPENDICULAR TO SCREW HOLE AFTER IRRAD
9. INCONEL SPRINGS SCREWED TO FRAME (TACK WELD SCREW)

Fig. 1 - Specimen loading schemes for accelerated exposure capsules A1-1 and A2-1 (1).

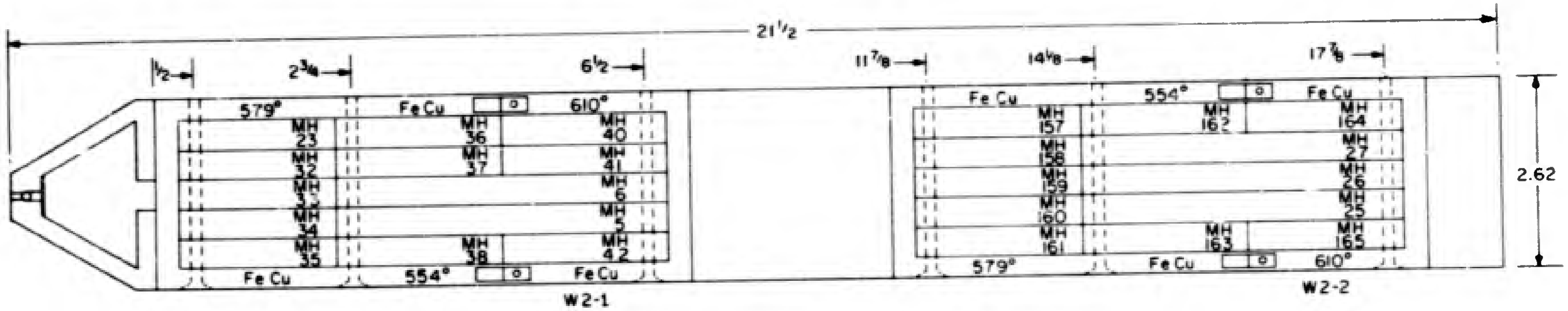


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Fig. 2 - Specimen loading schemes for vessel wall capsules W2-1 and W2-2 (1).

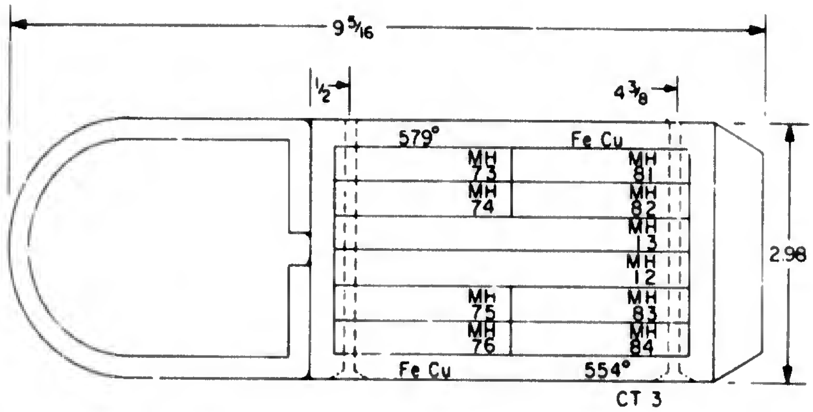

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Fig. 3 - Specimen loading scheme for above-core thermal control capsule CT-3 (1).

<u>Table 1</u>

Neutron Exposure Determinations for Surveillance Specimen Capsules

Capsule Surveillance Location	Capsule Number	Capsule Section*	Neutron Flux $n/cm^2 \cdot sec \times 10^{11}$ (>1 MeV)	Neutron Fluence $n/cm^2 \times 10^{19}$ (>1 MeV)	Charpy-V Specimen Number (Typ.)	Tensile Specimen Number (Typ.
Accelerated	A1-1	T	(a)	-	43	-
		M	(a)	-	48	8
		B	21.0	16.5	50	8
Accelerated	A2-1	T	51.5	40.4	169	-
		M	53.1 (est.peak)	41.7	175	30
		B	(a)	-	179	30
Vessel Wall	W2-1	T	0.87	0.68	23	-
		M	0.93	0.73	36	6
		B	1.19	0.93	40	6
Vessel Wall	W2-2	T	1.40	1.10	157	-
		M	1.57 (peak)	1.23	162	27
		B	1.37	1.08	164	27
Above-Core (Thermal Control)	CT-3	T	1.33	1.04	73	13
		B	2.76	2.17	81	13

*T - Top; M - Middle; B - Bottom.

[a]Neutron dosimeter wire not recovered in capsule disassembly.

Table 2

Neutron Exposure Determinations for Neutron Flux Monitor Assembly

Sample Number	Sample Elevation (in.)*	Neutron Flux $n/cm^2 \cdot sec \times 10^{11}$ (>1 MeV)	Neutron Fluence $n/cm^2 \times 10^{19}$ (>1 MeV)	Neutron Flux $n/cm^2 \cdot sec \times 10^{11}$ (>1 MeV)(Core 1)[2]	Neutron Flux $n/cm^2 \cdot sec \times 10^{11}$ (>1 MeV)(Core 2)[3]
1	1	1.23	0.43		
2	2	1.26	0.44		
3	3 (fuel ϕ)	1.26	0.44	1.32	1.01
4	4	1.29	0.45		
5	5 (peak)	1.34	0.47	1.47	1.02
6	6	1.28	0.45		
7	8	1.29	0.45		
8	10	1.25	0.44		
9	11	1.22	0.42		
-	11.5 (weld ϕ)	1.20	0.42	1.30	0.88
10	12	1.17	0.41		
11	13	1.10	0.38		
12	16	0.96	0.33		

*Elevation in inches from bottom of assembly.

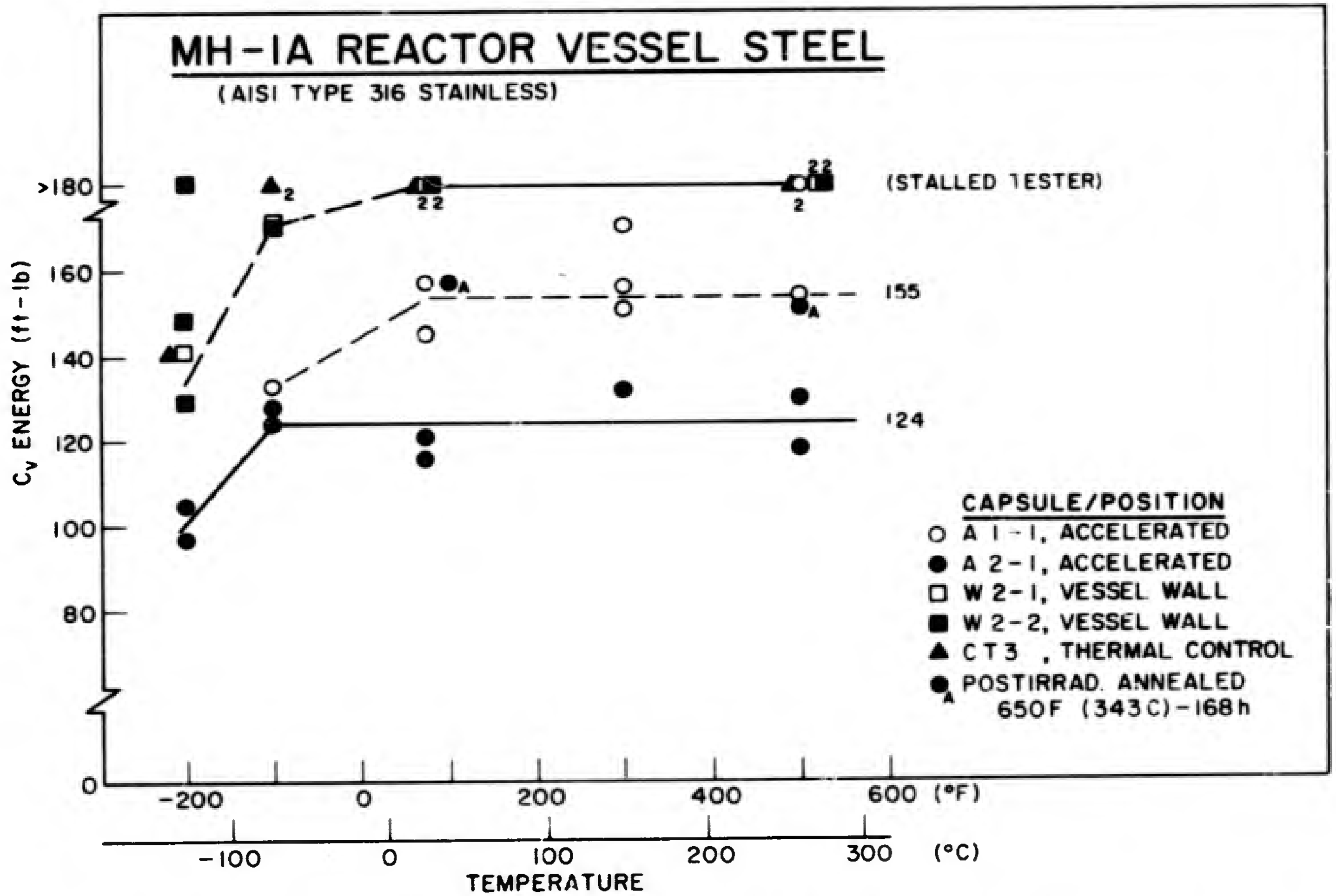

Fig. 4 - Charpy-V notch ductility of MH-1A reactor vessel forging as observed from specimens contained in accelerated exposure, vessel wall, and above-core (thermal control) surveillance capsules. The specimens were removed from the reactor during the Fall 1972 refueling outage.

capsules (A1-1, A2-1), the vessel wall capsules (W2-1, W2-2), and the thermal control capsule (CT3) are given in Fig. 4. Specimens from the latter two capsule locations showed energy absorption capabilities exceeding the capacity of the in-cell test machine at room temperature (75°F, 24°C) and at the reactor operating temperature (500°F, 260°C). Below room temperature, a progressive decrease in specimen energy absorption with decreasing temperature was observed wherein energy values at -200°F (-129°C) were significantly lower than room temperature values. This energy decrease, however, is of academic interest only. An equivalence of results from vessel wall and above-core exposures is observed and is consistent with the measured fluences for those locations (Table 1). The fluence for the above-core (thermal control) location, although low, was somewhat higher than originally expected.

The results for the accelerated exposure capsules, in contrast to the vessel wall capsules, indicate a significant reduction in C_V energy absorption at the higher fluences. Specimens at the peak flux elevation (filled circle points, capsule A2-1) had an average energy absorption of 124 ft-lb in the temperature range 74 to 500°F (24 to 260°C). Specimens at a lower flux elevation (open circle points, capsule A1-1) showed less of an irradiation effect with a minimum energy absorption of ~155 ft-lb. Data scatter for capsule A1-1 is readily traced to individual specimen elevations in this capsule. Overall, energy absorption values for the vessel wall and accelerated exposure conditions are quite high and should preclude any concern for fracture resistance of the forging at this point in service.

A limited evaluation of the effectiveness of 650°F (343°C)-168 hour heat treatment for postirradiation notch ductility recovery was possible with excess specimens from capsule A2-1. The data, included in Fig. 4, indicate that some notch ductility recovery is possible by this method should the need ever arise.

Results of tensile tests are summarized in Table 3. Data trends are illustrated in Figs. 5 and 6. Open symbols refer to 75°F (24°C) tests; Closed symbols refer to 500°F (260°C) determinations. Neutron fluences for the individual locations, in general, decrease from left to right in the figures. As with Charpy-V observations, a significant difference in irradiation effect was not found between vessel wall and above-core exposure locations whereas an appreciable elevation in yield and tensile strength was produced by the accelerated radiation exposure. It is noted that the data trends for 75°F (24°C) and 500°F (260°C) tests are comparable for all three surveillance locations. Of greater significance, the highest fluence exposure (capsule A2-1) did not cause an

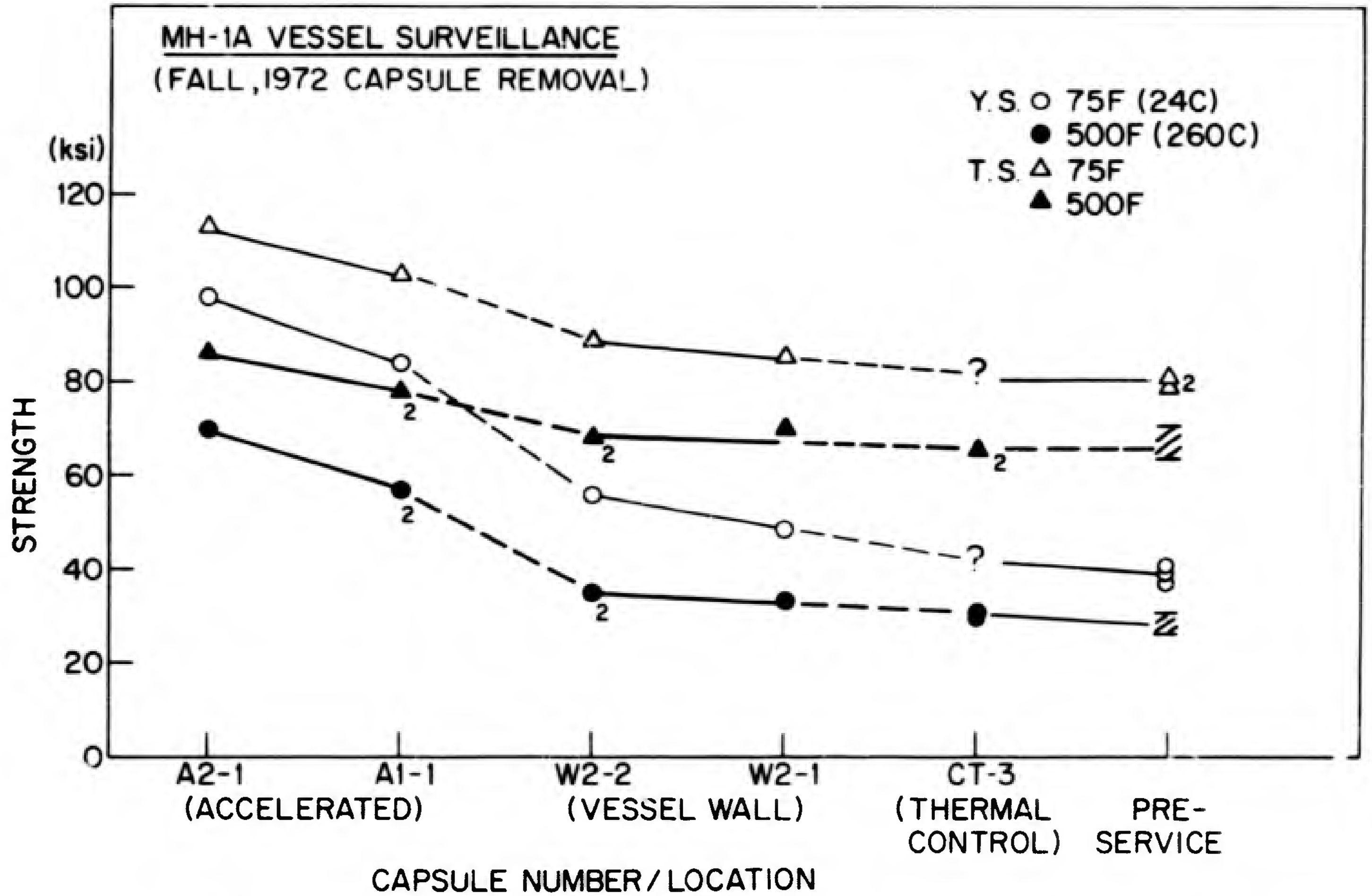

Fig. 5 - Yield and tensile strength of MH-1A reactor vessel forging after irradiation in vessel surveillance capsules (Core 1, 2, and 3 exposure).

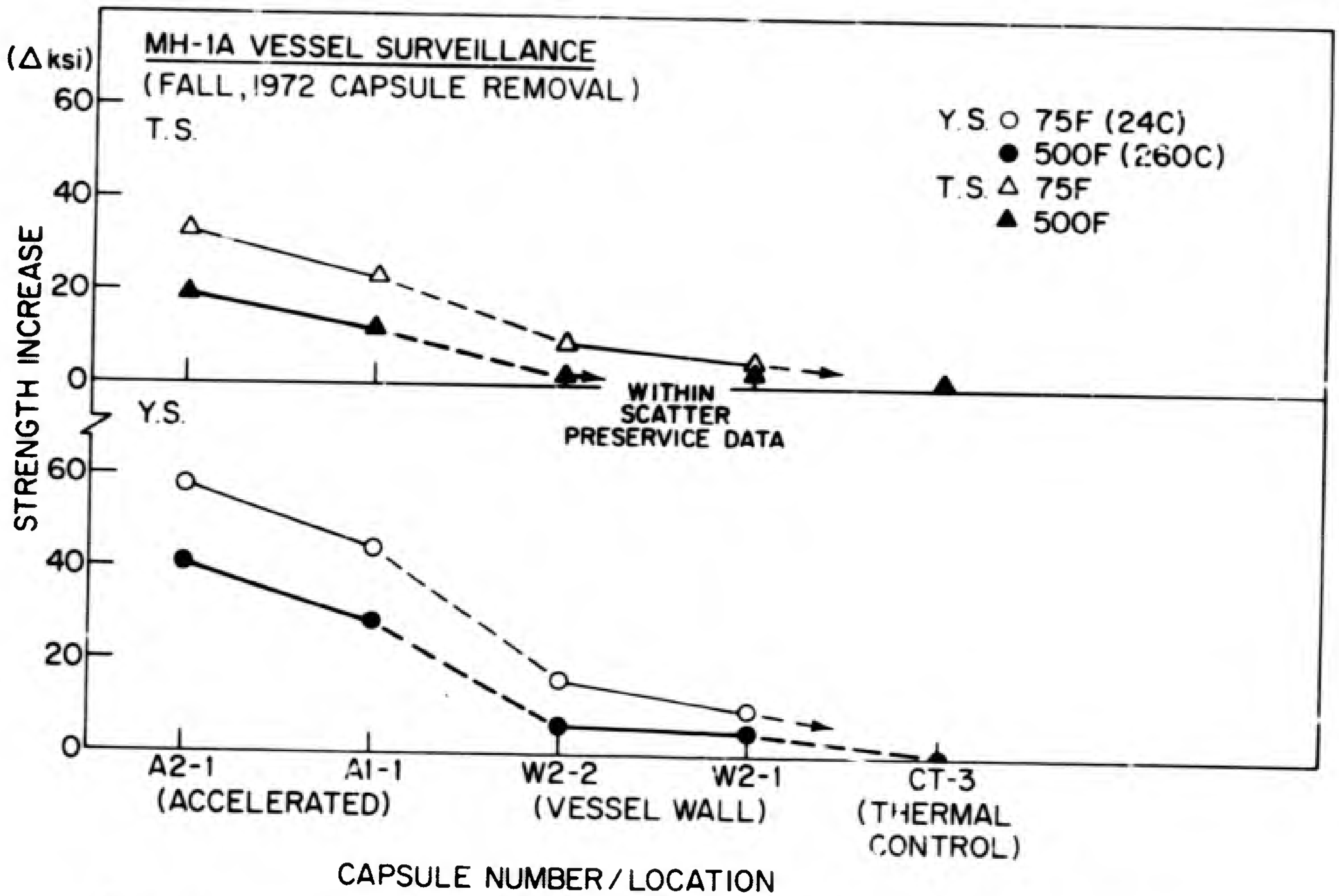

Fig. 6 - Increase in yield and tensile strength of MH-1A reactor vessel forging produced by surveillance capsule irradiations(Core 1, 2, and 3 exposure).

Table 3

Tensile Properties of MH-1A Vessel Forging (Surveillance Capsule Irradiations)

Capsule Number	Surveillance Location/ Fluence	Test Temp. (°F)	Yield Strength (ksi)	Tensile Strength (ksi)	Specimen Number
A2-1	Accelerated (41.0×10^{19})[a]	75[b]	98.3	113.3	30
		500[c]	69.7	86.1	29
A1-1	Accelerated (16.0×10^{19})	75	84.2	103.7	9
		500	55.6	78.6	8
		500	57.5	78.5	11
W2-2	Vessel Wall (1.16×10^{19})	75	56.0	89.6	27
		500	34.3	68.7	25
		500	35.4	68.3	26
W2-1	Vessel Wall (0.83×10^{19})	75	49.2	85.7	6
		500	33.8	70.1	5
CT-3	Above-Core (Thermal Control) (1.60×10^{19})	500	31.2	65.7	12
		500	29.9	64.7	13

[a] Average fluence, n/cm^2 >1 MeV.
[b] 24°C
[c] 260°C

elevation in yield strength above 100 ksi. The conclusion of high vessel fracture resistance based on C_V determinations is thus reinforced by the yield strength observations.

Results from the neutron flux monitor assembly, presented in Table 2, indicate a peak flux on the vessel inner wall of $1.34x10^{11}$ n/cm^2·sec, assuming a fission spectrum neutron energy distribution at this location. A calculated spectrum for the MH-1A vessel is not yet available. The peak flux location is shown to be approximately 2 in. above the fuel centerline elevation. By comparison, the fluxes at the centerline and at the vessel circumferential weld elevation are, respectively, $1.26x10^{11}$ n/cm^2·sec and $1.20x10^{11}$ n/cm^2·sec. For 20 full power years of operation (45 MWt), corresponding fluences for these three locations would be $8.46x10^{19}$, $7.95x10^{19}$, and $7.57x10^{19}$ n/cm^2 >1 MeV. From Core 1 and Core 2 results, estimates of the 20-year fluence at the peak exposure location were, respectively, $9x10^{19}$ and $6.9x10^{19}$ n/cm^2 >1 MeV.

The difference in vessel wall peak flux determinations from monitors in the wall capsule versus the flux monitor assembly is small ($1.57x10^{11}$ versus $1.34x10^{11}$ n/cm^2·sec) and may be due to a small difference in position relative to the wall proper and also to the difference in total exposure history (Core 1 + 2 + 3 versus Core 3 only). Good agreement is also observed for peak flux values determined from flux monitor assemblies exposed consecutively for single core periods; i.e., Core 1 versus Core 2 versus Core 3 (see Table 2). Accordingly, it is considered that the projection of maximum fluence for 20 full power years of operation is well based. In Table 2, flux values for the vessel circumferential weld determined from Core 2 and Core 3 flux monitor assemblies differ somewhat more than those for the peak flux position.

Continuing Program

The continuing MH-1A vessel surveillance program involves further exposures of vessel forging material but, in addition, now includes weld deposit material simulating the vessel circumferential girth weld. This modification was prompted by the similarity in flux between the peak flux location and the vessel weld location and by a recent observation of a potential in weld deposit for low fracture resistance compared to the vessel forging both before and after 500°F (260°C) irradiation. Accordingly, the inclusion of both materials in the surveillance program should permit a full assessment of vessel fracture resistance in service.

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(1) C.Z. Serpan, Jr. and H.E. Watson, "Pressure-Vessel Surveillance Program for the Army MH-1A Floating Nuclear Power Reactor," NRL Report 6604, Naval Research Laboratory, Sep. 22, 1967.

(2) L. E. Steele, et al, "Irradiation Effects on Reactor Structural Materials, QPR, 1 Aug-31 Oct 1970," NRL Memorandum Report 2181, Naval Research Laboratory, Nov. 15, 1970.

(3) L. E. Steele, Editor, "Irradiation Effects on Reactor Structural Materials, QPR, 1 Aug-31 Oct 1972," NRL Memorandum Report 2531, Naval Research Laboratory, Nov. 15, 1972.