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NRL Report 6604

Pressure Vessel Surveillance Program for the Army MH-1A Floating Nuclear Power Reactor

September 22, 1967

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NRL Report 6604

Pressure Vessel Surveillance Program for the Army MH-1A Floating Nuclear Power Reactor

Charles Z. Serpan, Jr., and Henry E. Watson

Reactor Materials Branch
Metallurgy Division

September 22, 1967

NAVAL RESEARCH LABORATORY
Washington, D.C.

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ABSTRACT

The pressure vessel surveillance program for the Army MH-1A, barge-mounted reactor was designed by NRL. The NRL-fabricated hardware for the surveillance program included two capsules for vessel wall locations in the reactor, two capsules for accelerated exposure locations, and six capsules for above-core, thermal control locations. Schematic drawings indicate the relative location of the test specimens, neutron flux monitors, and temperature monitors in each capsule section. The manufacturer of the vessel saved the reactor nozzle cutouts from the upper shell-course ring forging for test purposes. The Charpy V-notch and tensile specimens contained in the capsules were machined from one of these reactor nozzle cutouts. Schematic drawings show the location of each test specimen in the parent material. Selected Charpy-V and tensile specimens have been tested to develop preirradiation properties of the AISI 316 stainless-steel reactor vessel for future reference.

PROBLEM STATUS

This completes one phase of the work on the MH-1A surveillance program; work on other phases will continue.

AUTHORIZATION

NRL Problem M01-14
Projects RR 007-01-46-5409,
AT (49-5)-2110, and ERG-3-67

Manuscript submitted June 20, 1967.

PRESSURE-VESSEL SURVEILLANCE PROGRAM FOR THE ARMY MH-1A FLOATING NUCLEAR POWER REACTOR

INTRODUCTION

The MH-1A reactor is a mobile, high-power, pressurized water plant capable of generating 10 Mw(e) at 45 Mw(t) and is the first model of its type. The reactor system has been placed in the converted Liberty Ship *Charles H. Cugle*. Remodeling of the ship required the removal of some equipment but primarily entailed the addition of a large section in the central portion of the ship to house the reactor components. The reconverted vessel, renamed *Sturgis*, is capable of being towed to deep water ports anywhere in the world where there is a need for substantial electrical power in a short period of time.

The prime contract for the MH-1A reactor was awarded by the Army Corps of Engineers to the Nuclear Division of the Martin-Marietta Corporation. The subcontract for pressure vessel fabrication was awarded to the P. F. Avery Corporation of Billerica, Massachusetts. The reactor vessel is made of two 3-1/8-in.-thick AISI Type 316 stainless-steel, ring forgings welded circumferentially in the core region (Appendix A), a welded, hemispherical bottom closure (Appendix B), and a welded head and flange (Appendix C). As discussed later, one of the rings contains welded nozzles, and the cutouts for the nozzles provided the surveillance material.

The reactor design provides for two thermal shields close to the pressure-vessel wall as well as two additional shields adjacent to the core. Despite the four thermal shields, separated by water layers, the relatively compact size of the reactor still allows a significant neutron flux to impinge upon the vessel wall. In expectation of the vessel's exposure to high-energy neutrons as well as to the possible effects of thermal aging, the Army Nuclear Power Field Office deemed it necessary to initiate a program of pressure-vessel material surveillance. The purpose of this program is to monitor the progressive changes in the vessel's material properties. At several locations of engineering interest within the reactor, it was decided that metallurgical test specimens should be exposed to irradiation. This report will document and provide the details of the surveillance program designed by NRL at the request of the Army.

PROGRAM FOR PRESSURE VESSEL SURVEILLANCE

The purpose of surveillance testing is to assess the continuing serviceability of critical components in an operating reactor.* A surveillance program for a stainless-steel, reactor pressure vessel should provide for the representative irradiation of metallurgical test specimens which will depict the loss in notch-impact energy absorption and which will yield tensile data pertinent to the operational history of that reactor. The neutron flux at the pressure-vessel wall should also be established to ascertain the characteristic response of the vessel material to neutron irradiation.

*Recommended Practice for Surveillance Tests on Structural Material in Nuclear Reactors, ASTM E 185-62.

Surveillance Monitoring Locations

Irradiation assemblies containing metallurgical test specimens have been fabricated by NRL. These assemblies are located at the vessel wall, adjacent to the fuel, and above the core out of the high-flux region where specimens are subject to the same temperature history. Assemblies adjacent to the fuel will be referred to as accelerated exposure assemblies, and above the core assemblies will be referred to as above-core thermal control assemblies. In addition, neutron flux monitor wires have been placed against the inside surface of the vessel at the same radial locations as the vessel-wall assemblies. All of the surveillance assemblies and monitor wires were inserted in the reactor prior to the initial startup during the latter part of 1966. The relative locations of these items are shown in Fig. 1.

The placement of neutron flux monitor wires against the vessel provides for a direct measurement of the neutron flux in the critical region at the vessel wall; that is, from the centerline of the active fuel core to the lower edge of the inlet/outlet nozzle region. The flux monitor wires and specimen assemblies at the vessel wall will provide data on the effect of high-energy neutrons and of operational temperature upon the vessel wall material. Data provided by the accelerated exposure assemblies adjacent to the fuel will show in advance the lifetime radiation damage to be expected. The above-core thermal control assemblies will establish the effect of long-term exposure to the operational temperature in the absence of radiation. When one set of measurements has been made on each of the surveillance components, a clear picture of the pattern of radiation damage to be expected for the MH-1A pressure vessel should appear.

Since the vessel-wall specimens are located in a pocket machined from the outside diameter of the outermost thermal shield, the irradiation incident upon the specimens will be slightly greater than that upon the vessel wall. These specimens will also be shielded less than the neutron flux wires which are on the vessel wall. However, the differences in the neutron spectrum at the three locations will probably be so slight as to be essentially negligible. Since the flux monitor wires are along the inside diameter of the pressure-vessel wall, the flux measured at the vessel wall by the monitor wires will be slightly higher than the wall itself. Thus, the higher flux measurement is quite acceptable since the resulting analysis will be slightly conservative.

Test Specimens and Assembly Loading Arrangements

The type of metallurgical specimens selected for the MH-1A surveillance program are ASTM Charpy impact test specimens and modified 0.252-in.-diameter tension test specimens. Since all the specimens were to be exposed directly to the reactor coolant water during irradiation, the Charpy specimens have been left unnotched to prevent any undue corrosion of the V-notch root during exposure. This exposure mode, which is recommended by the ASTM, was selected as being most representative of the exposure of the pressure vessel. Tensile specimens have square, gripping ends, 0.394 by 0.394 in., and a modified gage diameter. For test purposes, the gage diameter has been left 0.010-in. oversize (0.262 in.). After irradiation, the gage diameter will be machined to the proper diameter. For control purposes, however, one tensile specimen machined to the exact dimensions of the gage diameter, 0.2520 in., has been included in one of the above-core thermal control assemblies. Drawings of both type of specimens, machined for irradiation in the MH-1A, are shown in Fig. 2.

The specimen loading arrangements for the two vessel-wall assemblies are shown in Fig. 3. The assemblies are essentially identical and are located in pockets machined into the outer surface of the outer thermal shields. Assembly W1-1/W1-2 has been located in the "forward" pocket, and assembly W2-1/W2-2 has been located in the "aft" pocket of the reactor vessel. The lower sections of the assemblies (W2-2 and W1-2) are

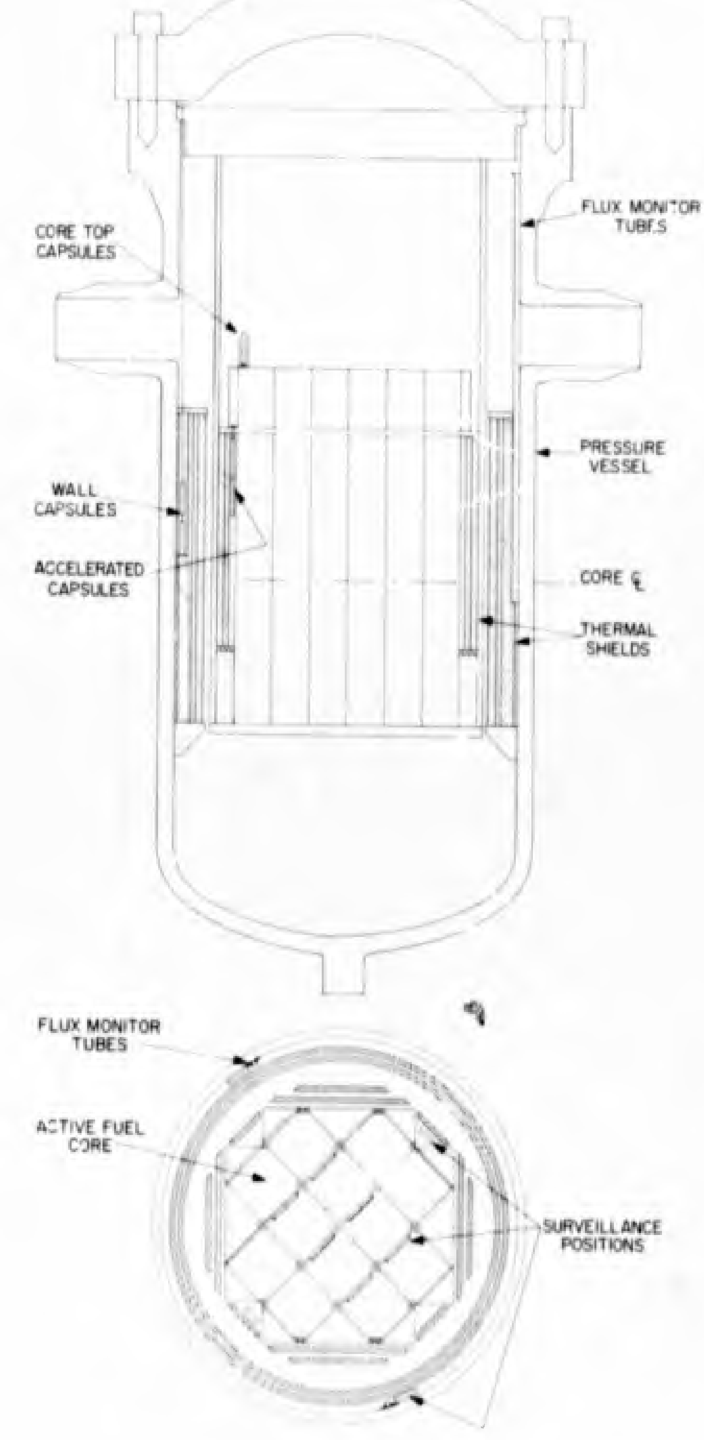
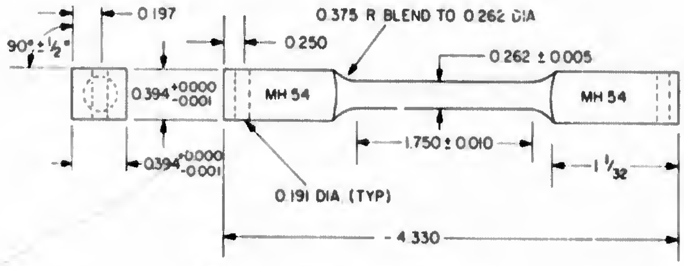
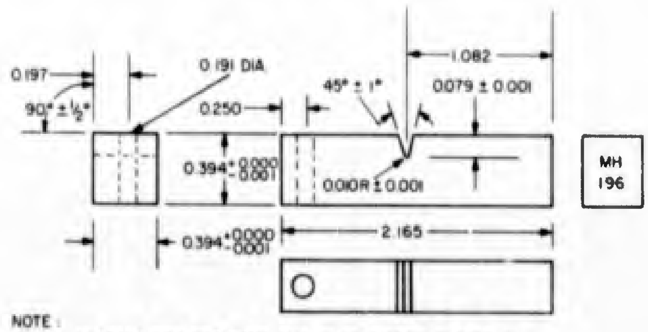


Fig. 1 - The Army MH-1A reactor pressure vessel showing the location of the pressure vessel surveillance assemblies in relation to major reactor components



NOTE:

1. GAGE DIAMETER OVERSIZE FOR FINAL MACHINING IN HOT CELL AFTER IRRADIATION.
2. TOLERANCES ± 0.002 UNLESS INDICATED



NOTE:

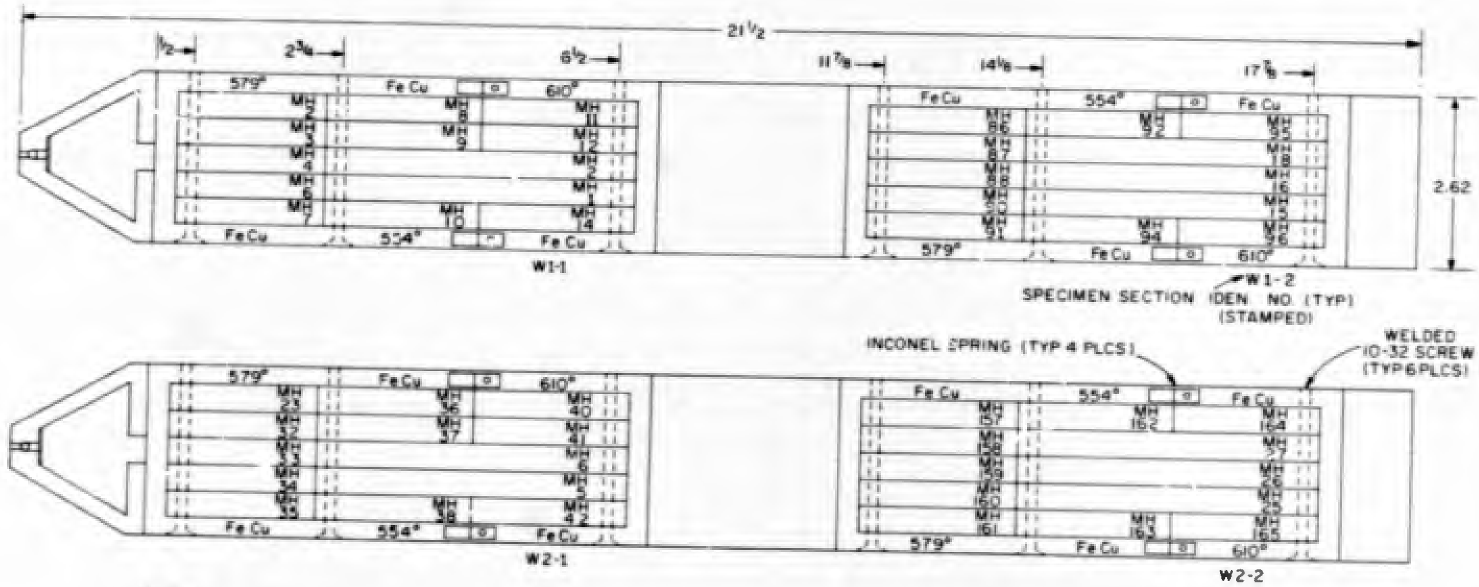
1. V-NOTCH HAD NOT BEEN CUT IN SPECIMENS PLACED IN SURVEILLANCE ASSEMBLIES.
2. WITH THE IDENTIFICATION LETTERING IN AN UPRIGHT POSITION, THE TOP SURFACE SHALL BE THE NOTCHED SURFACE.
3. TOLERANCES ± 0.002 UNLESS INDICATED

Fig. 2 - The tensile and Charpy impact specimens used in the Army MH-1A reactor pressure vessel surveillance program. The gage diameter of the tensile specimen is 0.010 in. oversize, and the V-notch of the Charpy specimen has not been cut in since both specimens will be exposed to the reactor coolant water during irradiation.

positioned so as to be centered about 3 in. below the active core centerline. (At startup the peak flux plane should be about 6 in. below the active core centerline.) The two lower sections contain three tensile specimens and nine Charpy specimens apiece. The two upper sections (W1-1 and W2-1) each contain two tensile and eleven Charpy specimens. These sections are centered on the circumferential weld of the pressure vessel which is 8.86 in. above the active core centerline.

The specimen loading arrangement for the two accelerated exposure assemblies is shown in Fig. 4. These assemblies are essentially identical to one another and are located in slots on either side of the core. No specific vertical locations were required for these assemblies since their only function is to obtain information concerning the response of the material to high energy neutrons in an accelerated mode. The lower section of each assembly is approximately centered on the circumferential-weld plane of the vessel. Both the lower sections (A2-1 and A2-2) and one upper section (A1-2) contain two tensile and eleven Charpy specimens. Section A1-1 has three tensile and nine Charpy specimens.

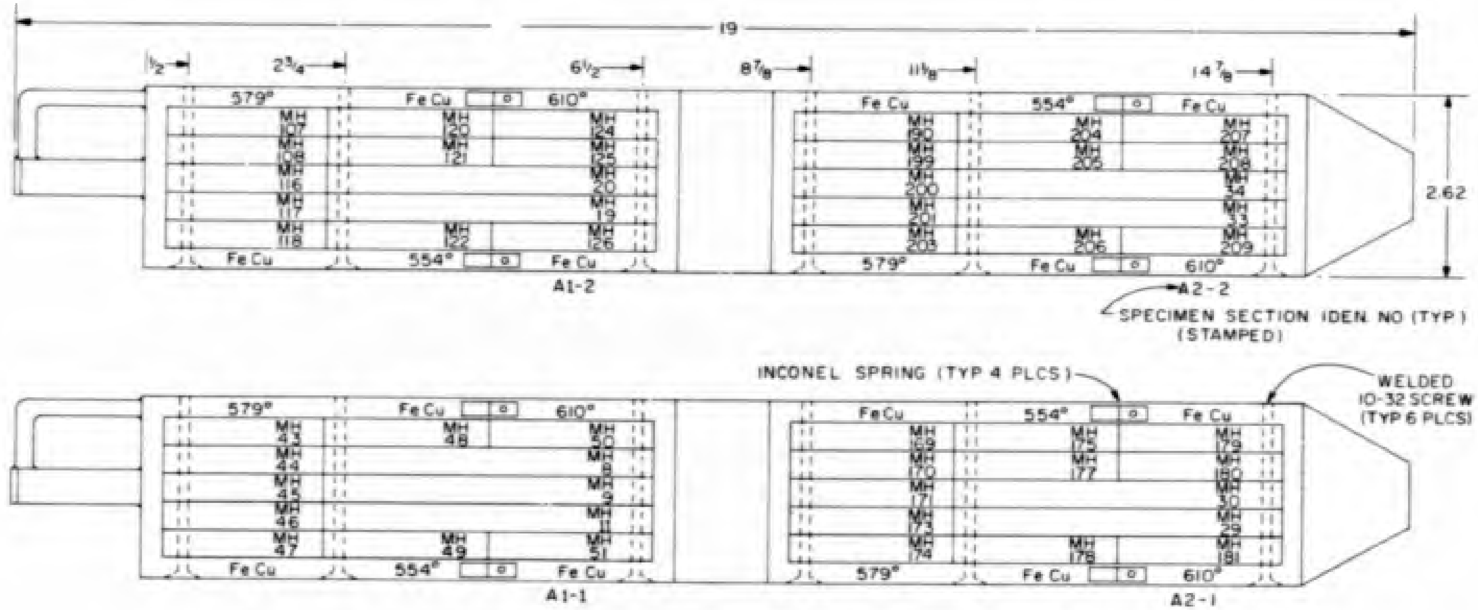
The specimen loading arrangement for the six above-core, thermal control assemblies is shown in Fig. 5. These assemblies are essentially identical and are located in



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
9. INCONEL SPRINGS SCREWED TO FRAME (TACK WELD SCREW)

Fig. 3 - The two assemblies located at the vessel wall surveillance locations of the Army MH-1A reactor pressure-vessel. "MH" and number indicates the specimen blank number and will be further identified in later figures.



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9. INCONEL SPRINGS SCREWED TO FRAME (TACK WELD SCREW)

Fig. 4 - The two assemblies located at the accelerated surveillance locations of the Army MH-1A reactor pressure-vessel

pairs within pockets in the hold-down structure above the core. Each assembly contains two tensile and eight Charpy specimens. Tensile specimen MH-48 located in assembly CT-4 has been machined to exact diameter, 0.252 in. All other tensile specimens in the above-core thermal control assemblies have, as noted in Fig. 2, a gage diameter of 0.262 in.

The frameworks securing the specimens in all of the three types of assemblies were machined in one piece from Type 304 stainless-steel plate. The bottom and middle spacer pieces as well as the top handling fixture were also machined from Type 304 stainless-steel plate. Furthermore, the assemblies have been fabricated with continuous, full-penetration welds. The thickness of all frames and assemblies is 0.394 in. Numerous pockets machined into the inside frame surfaces adjacent to the specimens accommodate temperature monitors (noted on the figures as 554°, 579°, and 610°) and neutron flux monitors (noted on the figures as "Fe, Cu").

Since the frameworks were left open to the reactor coolant water, it was necessary to secure the individual specimens to the frames by means of large-diameter screws running from side to side, through the specimens. The screws were threaded into the frame side and tack welded. The specimens themselves were not threaded but merely contain clearance holes for the screws. The dimensions noted for the approximate center of each screw are provided for future assistance in locating a hot-cell milling cutter for unsecuring the screws prior to removal of the specimens. Charpy specimens are to be notched perpendicular to the screw alignment hole, with the notching surface being up when the identification numbers stamped on each end are upright. Lathe centers are provided on the tensile specimens for final machining.

Inconel springs were affixed to the wall and accelerated assemblies to help prevent undue fluctuations caused by coolant water turbulence. However, during insertion, it became necessary to remove the two bottom springs from each wall assembly to allow proper alignment and fit.

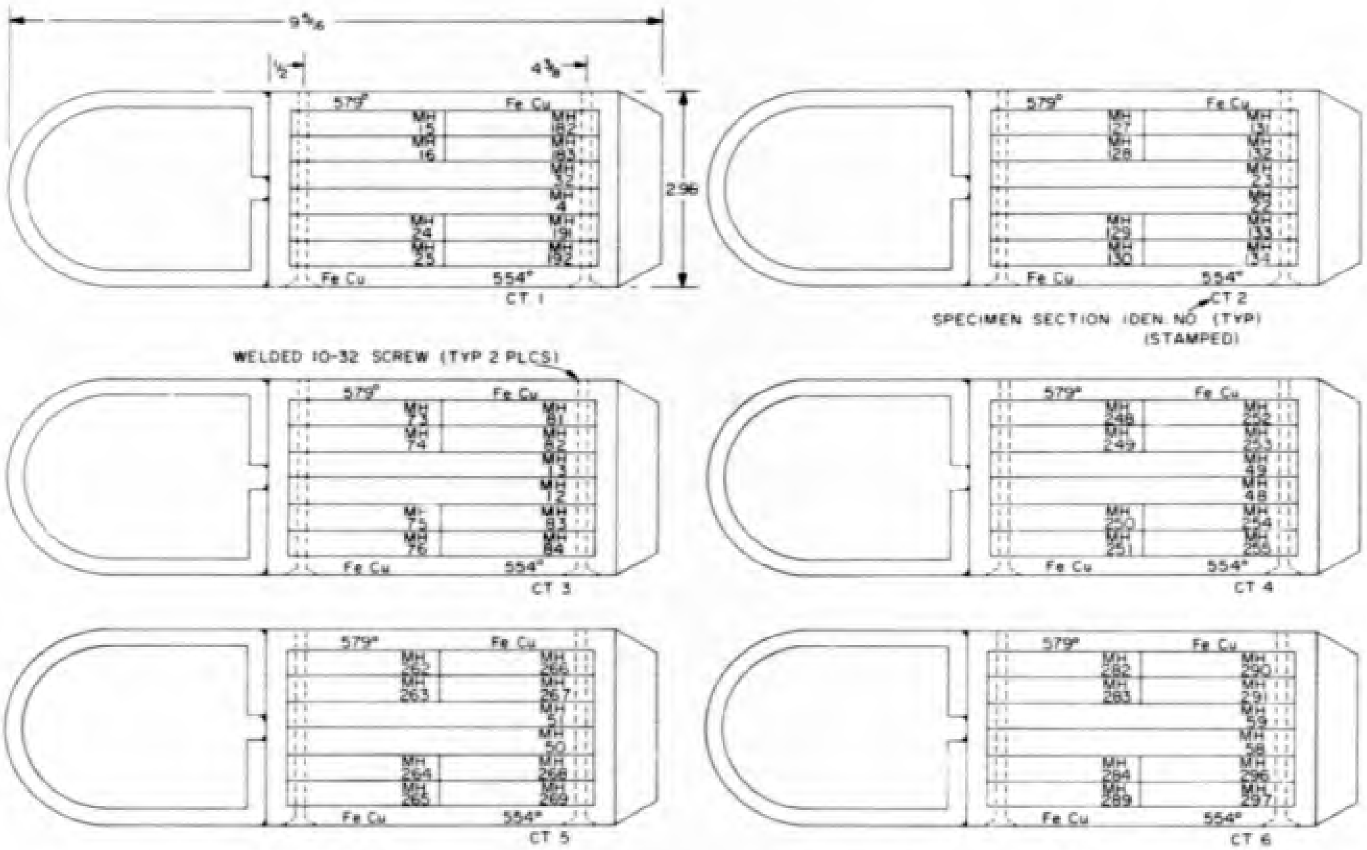
Details of the two neutron flux monitor wire assemblies are shown in Fig. 6. Two identical assemblies have been inserted in access tubes welded onto the vessel wall diametrically opposite each other and adjacent to the vessel-wall assembly positions.

Both sections A and B contain pure iron and copper wires. Both sets of wires in section A are crimped at 2-in. intervals (2 through 22) for rapid sectioning of the wires for counting analysis after irradiation. Likewise, both sets of wires in section B are crimped at 2-in. intervals (26 through 44). The active fuel core and circumferential-weld locations along the length of the wires are indicated for reference in Fig. 6. Because the two sections in each monitor assembly are rather long, a 1-in. section of 3/16-in. rod has been placed between the two sections as well as between section B and the top extension. The assemblies may be cut at these places without any possibility of water entering the wire sections.

MATERIALS

Pressure Vessel

Stock material for all the metallurgical test specimens in the surveillance assemblies for the MH-1A program were taken from one of the two 18-in.-diameter circles removed from the upper pressure-vessel ring forging for inlet and outlet nozzle openings (Appendix A). Diagrams showing the location of all blanks taken for specimen preparation from the parent forging sections are shown in Figs. 7 and 8. The chemical analysis obtained at NRL for the MH-1A pressure-vessel steel is given in Table 1.



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8. "V" NOTCH MUST BE CUT IN CHARPY BARS PERPENDICULAR TO SCREW HOLE

Fig. 5 - The six assemblies located at the above-core thermal control surveillance locations of the Army MH-1A reactor pressure-vessel

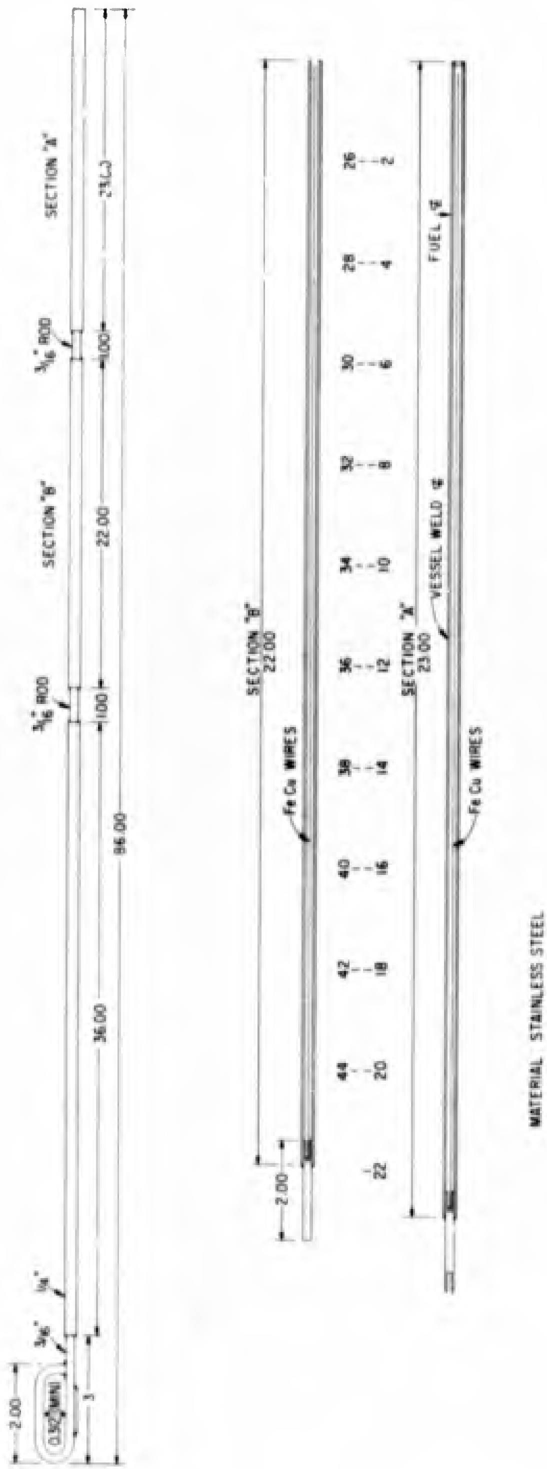

Fig. 6 - One of the two neutron flux monitor wires located along the vessel wall of the Army MH-1A reactor

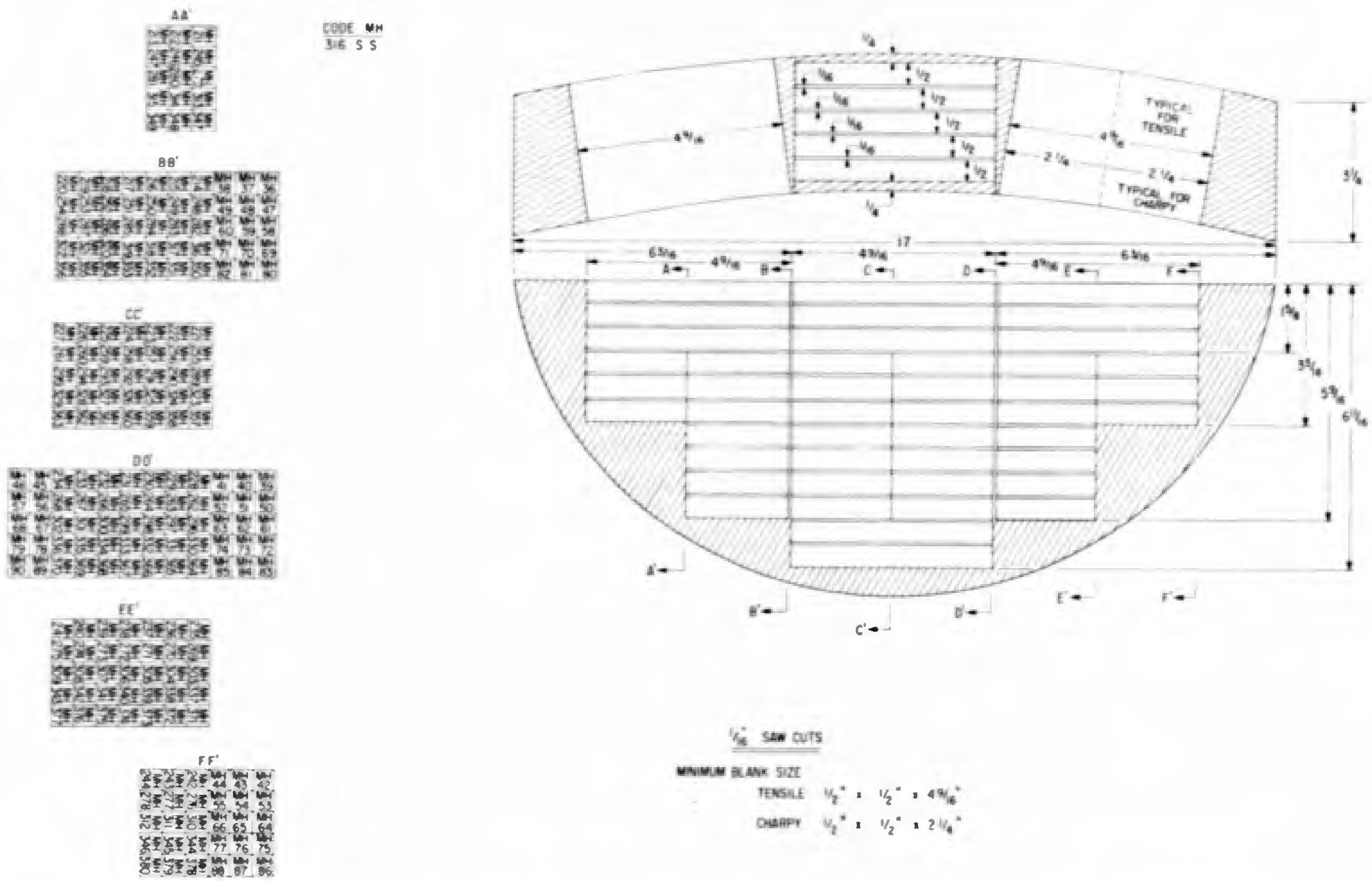

Fig. 7 - A portion of a nozzle cutout from the upper shell course ring forging of the Army MH-1A reactor pressure vessel showing the location of Charpy and tensile specimen blanks. (The other half of the cutout is shown in Fig. 8.)

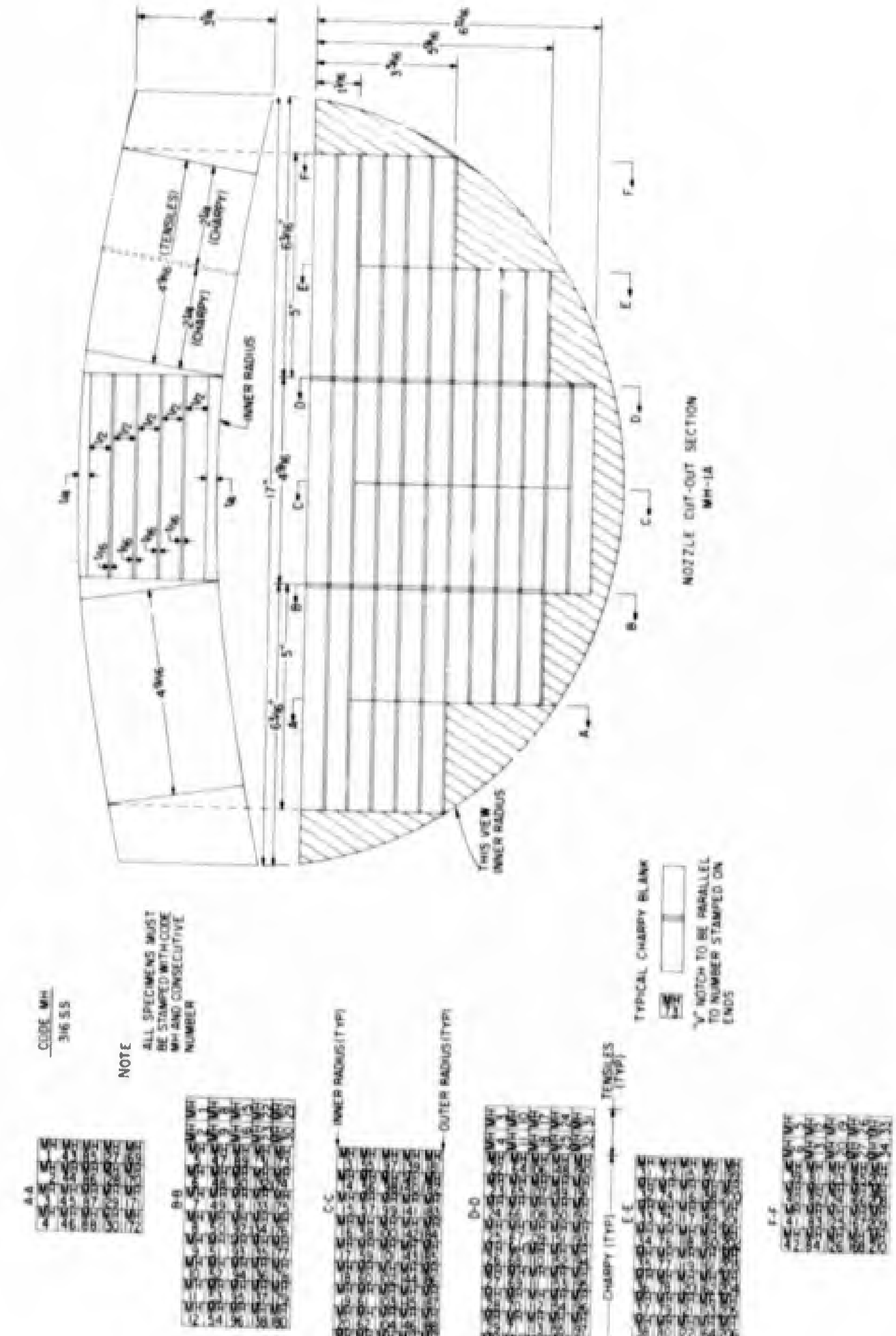

Fig. 8 - A portion of a nozzle cutout from the upper shell-course ring forging of the Army MH-1A reactor pressure vessel showing the location of Charpy and tensile specimen blanks. (The other half of the cutout is shown in Fig. 7.)

Table 1
Chemical Composition of MH-1A Reactor Pressure Vessel Surveillance Steel, Standard Steel Division, Baldwin-Lima-Hamilton Corp. Heat No. E-5367, (ASTM A-182 F316 S/S)

Sample	Chemistry (wt-%)										
	C	Mn	P	S	Si	Ni	Cr	Mo	Co	Ta	Fe
Ladle (BLH)	0.058	1.70	0.018	0.007	0.63	10.90	17.83	2.41	0.02	0.09	–
Check (BLH)	0.063	1.74	0.017	0.009	0.58	10.87	17.83	2.46	0.02	0.09	–
NRL	0.060	1.51	0.014	0.010	0.57	10.80	17.60	3.00	0.01	–	66.2

Test specimens were taken from five layers through the thickness of the 3-1/8-in. forging. These layers have been identified as inner radius, mid-inner radius, middle, mid-outer radius, and outer radius. The Charpy-V specimens representative of the five forging locations were tested over the temperature range -250° to 500°F. In every case, the specimens were bent into a U-shape and absorbed the full impact energy of the pendulum in excess of 200 ft-lb; only slight ductile tearing in the vicinity of the V-notch was observed. In the region of maximum strain around the V-notch, the specimen surfaces developed an "orange peel" appearance which is not unusual for forged stainless steels.

The tensile properties at ambient temperature and at 500°F, as depicted by specimens representing four of the five layers in the 3-1/8-in. forging, are presented in Table 2. Ambient and 500°F, nominal stress vs reduction of area curves are presented in Fig. 9. Again, specimens tested at both temperatures displayed the characteristic "orange peel" surface appearance indicative of material working capability. Both the 500°F and ambient-temperature data show a trend of slight degradation in properties toward the middle of the forging, but these differences are so slight as to be of little consequence. A noteworthy property of the material is the high reduction of area values.

Table 2
Tensile Properties of MH-1A Reactor Pressure Vessel Surveillance Steel, Standard Steel Division, Baldwin-Lima-Hamilton Corp. Heat No. E-5367. (ASTM A-182 F316 S/S)

Forging, Radius Location	Test Temperature (°F)	Yield Strength (0.2% Offset) (ksi)	Tensile Strength (ksi)	Reduction of Area (%)
Mid-inner	Ambient	40.05	81.2	83.0
Mid-inner	500	28.0	64.4	66.0
Mid-inner	500	30.4	66.0	71.7
Middle	500	28.4	70.6	63.6
Middle	500	28.5	68.6	66.9
Mid-outer	Ambient	37.8	79.5	84.6
Mid-outer	500	27.4	64.2	73.0
Mid-outer	500	29.0	65.4	68.7
Outer	Ambient	41.0	81.0	78.8
Outer	500	28.9	64.8	67.9
Outer	500	29.1	70.0	70.4

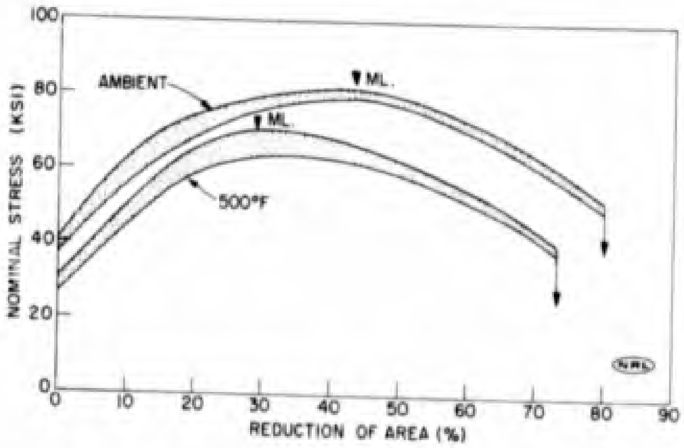

Fig. 9 - Nominal stress vs reduction of area for 0.252-in.-gage-diameter tensile specimens, representing four thickness layers of the 3-1/8-in. AISI 316 ring forging nozzle cutouts of the Army MH-1A pressure vessel. The specimens were tested at ambient and 500°F temperatures.

Tensile and Charpy specimens selected for each assembly section have been taken, with one exception, from a single or individual layer in the stock material. Even though there may be slight differences in the unirradiated properties between layers, such differences will not enter into the analysis of results within any one specimen assembly section after irradiation. It is expected that the total neutron exposure of a single section in any one assembly will be sufficiently uniform to permit the testing of all the specimens in a given section as a unit. The absence of variations in the unirradiated properties of the individual specimens of each section will further assist in this analysis. The layer identification for each assembly specimen section is given in Table 3.

Table 3
Parent Material Thickness Layers for Assembly Specimen Sections

Assembly Section	Thickness Layer	Assembly Section	Thickness Layer
W1-1	Inner	A2-1	Outer
W1-2	Middle	CT-1	Inner and outer
W2-1	Inner	CT-2	Mid-outer
W2-2	Mid-outer	CT-3	Mid-inner
A1-2	Middle	CT-4	Mid-inner
A2-2	Outer	CT-5	Mid-inner
A1-1	Mid-inner	CT-6	Middle

Reference or correlation-monitor material specimens were not included in the initial complement of specimens for irradiation due to the very limited space available for representative exposures.

Neutron Flux Monitor Wires

Pure iron wire, drawn from billets at NRL, has been employed as the primary neutron flux monitor for the MH-1A surveillance program. Each flux monitor assembly contains wires of this material. Within each assembly, there are small welded stainless-steel tubes containing a 1/2-in. section of this iron wire as well as a 1/2-in. section of pure copper wire. Every position on each specimen loading arrangement which is noted "Fe, Cu" is a pocket machined into the frame in which these flux monitor wires within their protective tubes are contained.

The pure copper wire, drawn from billets at NRL, was also used as a neutron flux monitor. Chemical analysis of the copper wire showed that the cobalt content was equal to or less than 0.2 ppm. Copper has been included in the surveillance program in the hope that it will prove useful for very long integrated exposure periods. The reaction of interest for this monitor is $Cu^{63}(n,\alpha)Co^{60}$. Since Co^{60} can also be easily induced from Co^{59} as a result of thermal neutron irradiation, accurate predetermination of the cobalt impurity level in the copper is necessary.

Temperature Monitors

Low-melting-temperature eutectic alloys of three compositions have been included in the specimen assemblies to help determine the highest temperature to which each group of specimens has been exposed. These temperature monitors have been sealed in quartz tubes and then in stainless-steel tubes similar to the neutron flux monitors. The temperature monitors have also been placed in pockets machined into the frame. The specific temperatures at which these monitors will melt are 554°, 579°, and 610°F. Visual inspection of the alloys after irradiation will reveal either a smooth ball or an oblong mass, indicating a temperature equal to or exceeding the nominal melting temperature, or a thin strip of alloy with relatively sharp edges and ends, indicating that the nominal temperature was not exceeded.

Assembly Withdrawal Schedule

No specific time limits have been set for withdrawal* of the individual assemblies of the MH-1A surveillance program. It is expected that one of the accelerated assemblies along with one of the neutron flux monitor assemblies will be removed after approximately 1 year of full power operation. Additional units for reinsertion of these two types of assemblies as well as the above-core thermal control assemblies can be made. The vessel wall assemblies are located in such a manner that replacement will be most difficult if not impossible. Therefore, the initial withdrawal of a vessel-wall assembly will probably not occur until after a minimum of three or four years of operation. Above-core thermal control assemblies may also be removed after this three- or four-year period.

SUMMARY

A pressure vessel material surveillance program has been designed for the Army MH-1A reactor by the Naval Research Laboratory. This program is designed to monitor the progressive changes in the vessel's material properties and to indicate in advance the effect on the material of high energy neutrons and of thermal aging. Using material from nozzle cutouts of the upper ring forging of the AISI 316 stainless steel vessel, NRL fabricated the necessary specimens. Frameworks for all the surveillance assemblies were machined from a Type 304 stainless-steel plate. Drawings of the individual assemblies are presented as well as drawings depicting the location of all test specimens in the parent material. Mechanical properties representative of the unirradiated condition of the surveillance material as well as the chemical analysis are reported for future reference.

*The various tools to be used for the assembly removal operations were designed by the Martin Company. Details of the tools and their operation are contained in the MH-1A Reactor Operating Manual.

Appendix A

MANUFACTURER'S CERTIFICATION AND DOCUMENTATION RELATING TO THE TWO SHELL COURSE RING FORGINGS AND INLET NOZZLES OF THE ARMY MH-1A REACTOR PRESSURE VESSEL

BLH

STANDARD STEEL

A DIVISION OF BALDWIN·LIMA·HAMILTON CORPORATION

BURNHAM, PA. 17009 AREA CODE 717 248-7871

January 20, 1966

Mr. Charles Z. Serpan, Jr.
Reactor Materials Branch
Metallurgy Division
U.S. Naval Research Laboratory
Washington, D. C. 20390

Dear Sir:

SUBJECT: P. F. Avery Corporation Order #J-300-2 and J-300-7
SSD Orders 542-1351 and 662-1134

We have for reply your letter dated January 12, 1966, Ref. #6390-9:CZS:gml, requesting additional information concerning the heat treatment used in producing 316 stainless steel forgings for the P. F. Avery Corporation.

The four forgings produced on subject orders were heat treated (solution treated) as follows:

> Initially, they were heated to 1800F in 6 hours and then held at 1800F for 5 hours. They were then heated to 1960F in 1 hour and held at 1960F for 2½ hours, and then water quenched.

We trust this is the information that you desire, but should you need additional information or have any questions, please feel free to contact us.

Very truly yours,

T. J. CONNARE
Manager - Quality Control

js

STANDARDIZE ON STANDARD FOR
WELDLESS RINGS ROLLED FLANGES FORGINGS ROLLED WHEELS

Fig. A-1 - Heat treatment record for the two vessel shell course ring forgings and two nozzles of the Army MH-1A reactor pressure vessel

STANDARD STEEL DIVISION
Baldwin Lima Hamilton Corp.
Burnham, Pa.

Date: 2-17-64

SUBMISSION SHEET AND TEST REPORT

Customer: P.F. Avery Corporation — Cust. Order: J 300-2

Material: Forged Stainless Steel Cylinders — SSD Order: 542 1352

Destination: Billerica, Mass. — Items 1 & 2

Inspection: Coast Guard — Spec'n: ASTM A 182 F316 S/S
See Order.

SERIAL NO.	HEAT NO.	
MHA-810	E-5367	Item #1

Total Pcs. 1 Total Wgt. 9200#

MHA-811	E-5400	Item #2

Total Pcs. 1 Total Wgt. 11700#

Heat Analysis:

	Heat No.	C	Si	P	Mn	S	Ni	Cr	Mo	Cu	Ti
Ladle	E-5367	[illegible]	.63	.028	1.70	[illegible]	10.90	17.83	2.[illegible]	.03	.05
Check	E-5367	.052	[illegible]	.027	1.[illegible]	[illegible]	10.84	17.[illegible]	2.36	.02	.06
Check	E-5367	[illegible]	.60	.026	1.71	.010	10.62	17.80	2.55	.02	.07
Ladle	E-5400	[illegible]	[illegible]	.017	1.60	.005	10.72	17.76	2.46	.017	.11
Check	E-5400	[illegible]	.52	.018	1.65	.010	10.[illegible]	17.43	2.[illegible]	.02	.10
Check	E-5400	[illegible]	.54	.018	1.67	.010	10.72	17.26	2.[illegible]	.02	.10

	Tensile	Yield	Elong.	Red.
E-5367	7[illegible]500	42500	64.5	78.5
E-5400	77000	37000	68.0	79.0

Corrosion Test: O.K. 180° Bend O.K.

Dwg. SK-1048, Rev. "A" - 1/22/64.

Ultrasonic Inspected per MIL-STD-271B and found to be satisfactory.

Inspected by: Coast Guard.

This is to certify that the chemical analysis (and/or test results) shown in this report are correct as contained in the records of the Company.
No halogen bearing material was used for cleaning forgings.
Date Shipped: 2/18/64 Car No.: Truck

Fig. A-2 - Chemical analysis and test report for the two ring forgings of the Army MH-1A reactor pressure vessel

RSD 679 5M 8-64

BALDWIN-LIMA-HAMILTON CORPORATION
STANDARD STEEL DIVISION
METALLURGICAL DEPARTMENT

Burnham, Pa. 4-2-64 3-28-64 - Date Shipped

Test Report of 2 Forged Stainless Steel Inlet Nozzles For P.F. Avery Corporation

Our Order 662-1134 Customer's Order J-300-7

Specification ASTM A-182-61T, Grade F316, See Order. Car No. Truck

HEAT	MATERIAL	Tensile Strength	Yield Strength	Elong. Per Cent in 2 Inches	Reduction of Area Per Cent	CHEMICAL ANALYSIS Carb.	Sil.	Phos.	Mang.	Sul.	Ni.	Chr.	Mo.	Va.
Item #1														
E-5569	Ser. 4B1453	79000	42000	61.5	76.5									
Item #2														
E-5569	Ser. 4B1454	78500	42000	65.7	78.7									

Heat Analysis:

	Heat No.	C	Si	P	Mn	S	Ni	Cr	Mo	Co	Ta
Ladle	E-5565	.06	.45	.020	1.40	.008	10.74	17.76	2.37	.02	.08
Check	E-5569	.05	.39	.017	1.48	.009	10.58	17.29	2.39	.02	.08
Check	E-5569	.045	.41	.020	1.50	.009	10.63	17.45	2.44	.02	.08

Dwg. SK-1050 Rev. C dated 1/23/64.

No halogen bearing material was used for cleaning forgings.

Ultrasonic tested per ASTM A-388 and found to be satisfactory.

Liquid Penetrant inspected per A165-50T - No Bleeding.

Inspected by Coast Guard - H.E. Gafford.

This is to certify that the chemical analysis (and/or test results) shown in this report are correct as contained in the records of the Company.

L. J. Cannavale Manager, Process Metallurgy

Fig. A-3 - Chemical analysis and test report for the two inlet nozzles of the Army MH-1A reactor pressure vessel

Appendix B

MANUFACTURER'S CERTIFICATION AND DOCUMENTATION RELATING TO THE BOTTOM ELLIPTICAL HEAD OF THE ARMY MH-1A REACTOR PRESSURE VESSEL

Telephone: 215-384-2800
Cable: Carlson, Thorndale, Pa.

Telex: 083-4915
Teletype: 215-383-1693

G.O.CARLSON Inc.
Producers of Stainless Steel, Nickel Alloys and Titanium
THORNDALE, PA 19372

February 18, 1966

U. S. Naval Research Laboratory
Washington, D. C. 20390

Attention Mr. Charles Z. Serpan, Jr.
Reactor Materials Branch
Metallurgy Division

Gentlemen:

Subject: Your Reference 6390-10: CZS: gml
Our GOC 44411A; P. F. Avery Corp.

This refers to your request for actual recorded histories of the various heat treating steps followed during the processing of the following items.

Elliptical head, Type 316 HRAP ASTM A240-61T stainless steel for P. F. Avery Corp., C. O. No. J-300-4, GOC 44411-A, Heat No. 43574-2.

A copy of the record of solution annealing heat treatment performed this material is enclosed.

In reviewing our records we note that similar data were submitted to the U. S. Coast Guard Inspector at the time of inspection.

Cordially,

H. E. Tomlinson, Jr.
Manager - Technical Service

HET:anh
encl.

Fig. B-1 - Confirmation of processing steps for the bottom head of the Army MH-1A reactor pressure vessel

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132
BOS-20-0914

G.O.CARLSON Inc.
Producers of Stainless Steel
THORNDALE · PENNSYLVANIA

TEST REPORT
3/3/64
DATE

SOLD TO P. F. Averv Corporation
67 High Street
Billerica, Massachusetts

DOC 44411-A

CUST NO J-300-4

MARK J-300-4

CHEMICALS AND OR PHYSICALS

DATE OF SHPT 2/28

NO COPIES 4

ATTN Pur. Dept.

TYPED BY JJH

INVOICE

ROUTING (ACTUAL) BRANCH -- PREPAID

4 NOTARIZED TEST REPORTS (DOC REF ONLY)
CHEMS. & MECHANICAL PROP., ULTRASONIC RESULTS & CHECK ANALYSIS

SHIP TO P.F. Avery Corporation
67 High Street
Billerica, Massachusetts

B/N TO CUST 2

TYPE 316 HRAP ASTM A-240-61T
STAINLESS STEEL

w/o 1-27-64 or sooner

C

		Item	Qt Shpd	Weight	Price	Amount
1	ELLIPTICAL HEAD -- 66 ID x 3-1/8 min., 3-1/2 nom., 4 Approx. S.F., 20-1/2 INSIDE OVERALL HEIGHT MACHINED PER DWG. C-SK-1047* INCLUDING 4 TO 1 TAPER ON OD TO HOLD TOLERANCE OF + 1/8, - .000 ANNEALED & PICKLED AFTER FORMING. CONTRACT DA-36-109 ENG-7454 SUBJECT TO P.F. AVERY, MARTIN-MARIETTA CORP. & U.S. COAST GUARD INSPECTION PER CG SPEC. 115	1	1	7031		

RECEIVED
MAR 5 1964
P. F. AVERY CORP.
BILLERICA

CHEMICAL ANALYSIS

Item	Qt	Heat No.											Co.
1	1	43574-2	.048	1.83	.018	.012	.57	17.58	10.62	-	.01	2.28	.18
			CHECK ANALYSIS										
			.050	1.75	.021	.010	.58	17.64	10.70	-	*.01		*.19
			*MODIFIED										

MECHANICAL TESTS

GA	HEAT NO	YIELD STR PSI	TENSILE	% ELONG IN 2"	% RED OF AREA	BRINELL HARDNESS	BEND TEST
3-1/2	43574-2	37,000	82,500	60.0	73.0	137	OK
		40,500	83,500	60.0	71.0	140	OK

Ult asonic Report Attached.

SWORN TO AND SUBSCRIBED BEFORE ME THIS 3 DAY OF March/64

HEREBY CERTIFY THESE TESTS ARE CORRECT TO THE BEST OF MY KNOWLEDGE AND BELIEF

G O CARLSON, Inc

Fig. B-2 - Chemical analysis and test report for the bottom head of the Army MH-1A reactor pressure vessel

FORM 1848 3/63

HEAT TREATING DEPARTMENT

HEAT OBSERVERS REPORT - GREEN ANNEAL BUILDING

ORDER NO. ______ SINGLE LAYER (✓) FCE. NO. 5

GRADE [illegible] IF STACK, HOW MANY PILES ______ CHG. NO. 32

RIDER PLATE USED (✓) DATE 2/6

ON LOAD - COUPLES # [illegible] TIME CHGD. 7:[illegible]

IN LOAD - COUPLES # ______ O.K'd. UNIFORMLY AT TEMP. BY ______ AT ______

TIME	MAX. TEMP.	MIN. TEMP.	OBSERVER & COMMENTS	TIME	MAX. TEMP.	MIN. TEMP.	OBSERVER & COMMENTS
[illegible]	[illegible]	[illegible]	[illegible]				
[illegible]	1960	[illegible]					
[illegible]	2000	1980					
[illegible]	2000	1990					
[illegible]	2010	1990					

3½

1950-2050

3½ HR.

W.Q.

MULTI-CHARGE REHEAT		TIME OUT	AIR COOL				WATER COOL						
IDENTITY OR DESCRIPTION OF PIECE	TIME BACK UP TO TEMP.		(✓) UNLOAD IMMEDIATELY	TIME: NOT UNLOADED AS OF	TIME: UNLOADED ON OR BEFORE	(✓) FAN USED	TIME (Sec.): DOOR OPEN TO WATER	TIME (Sec.): IN WATER	DIP	SPRAY	GOOSENECK	ROTATED	TEMP AFTER WATER
	X	2:00 P					2 MIN.	[illegible]	[illegible]				[illegible]

Attach to back of GREEN copy of "FURNACE PRODUCTION FORM" 5

Fig. B-3 - Heat treatment data sheet and record for the bottom head of the Army MH-1A reactor pressure vessel

Appendix C

MANUFACTURER'S CERTIFICATION AND DOCUMENTATION RELATING TO THE UPPER HEAD AND FLANGE FORGINGS OF THE ARMY MH-1A REACTOR PRESSURE VESSEL

United States Steel Corporation

Homestead District Works

Homestead, Pa.

March 29, 1966

U. S. Naval Research Laboratory
Reactor Materials Branch
Metallurgy Division
Washington, D.C. 20390

Attention: Mr. Charles Z. Serpan, Jr.

Dear Sir:

In answer to your request of January 12, 1966, we are submitting the heat treating cycles used in the processing of the following items:

316 Stainless Steel Flange and Head for P. F. Avery Corporation, Customer Order No. J-300-3, U. S. Steel Order No. HV-40404 dated 11/8/63. Heat and Serial Numbers X25647, 2616-1 and X25646, 2616-2.

We regret the delay in forwarding this information to you.

Very truly yours,

J. L. Giove
Chief Metallurgist

LLJames/jlo
Attachment

Fig. C-1 - Confirmation of processing steps for the upper head and flange forgings of the Army MH-1A reactor pressure vessel

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TREATMENT

SERIAL 2616-1

Charge into a 1000°C furnace - heat to 1065°C - equalize and then maintain 12 hours - water quench cold.

SERIAL 2616-2

Charge into a 1000°C furnace - heat to 1065°C - equalize and then maintain 12 hours - water quench cold.

J. L. Giove, Chief Metallurgist/jlo

Fig. C-2 - Heat treatment of the upper head and flange forgings of the Army MH-1A reactor pressure vessel

UNITED STATES STEEL CORPORATION

STEEL OPERATING DIVISION–METALLURGICAL
HOMESTEAD WORKS

FILE NO.

Report of CHEMICAL and PHYSICAL Tests of Electric Furnace Forging Flange — February 13, 1964

Charged to P. F. Avery Corporation — Shipping Notice

Shipped to Billerica, Massachusetts — Car Number

Customer's Order No. and Date	Mill Order No.	Heat No.	Serial No.	C	Mn	P	S	Si	Cu	Ni	Test Piece Cut From	.2% Y.S.	Tensile Strength lbs. per sq. in.	Elong. in 2" %	Red. of Area %	Bend Test
				LADLE ANALYSIS							Cr Mo Co Ta					
J-300-3	HV-40404	X25647	2616-1	.055	1.57	.026	.014	.54	.06	10.90	17.52 2.31 .08 .01					
11-8-63		X25646	2616-2	.049	1.43	.026	.010	.45	.08	11.15	17.14 2.28 .10 .01					
				CHECK ANALYSIS												
			2616-1	.06	1.56	.026	.013	.48		10.90	17.56 2.40 .07 .01					
			2616-2	.06	1.48	.030	.013	.47		11.34	17.48 2.20 .08 .01					
			2616-1	Tangential								41370	81550	57.5	75.2	
			2616-2	Tangential								39970	80980	57.0	77.6	

ASTM A-182 Grade F 316 with Co .20 max., Ta .10 max.,
Cr 17.00/18.00, Ni 10.50/11.50 Annealed.

gk 3/6/64

RECEIVED FEB 1964 P. F. AVERY CORP. BILLERICA

We hereby certify that the above figures are correct as contained in the records of the company J.T. Giove, Chief Metallurgist

Fig. C-3 - Chemical analysis and test report for upper head and flange forgings of the Army MH-1A reactor pressure vessel

UNITED STATES STEEL CORP.
HOMESTEAD DISTRICT WORKS

ULTRASONIC & MAGNETIC PARTICLE TEST REPORT

TYPE FORGING FLANGE SERIAL 2616-1

HEAT # X2[illegible]7 ANALYSIS T-316 STAINLESS

CUSTOMER P. F. AVERY ORDER # J-300-3

ULTRASONIC TESTED IN ACCORDANCE WITH SPECIFICATION OR PROCEDURE AS DETAILED BELOW:

TYPE OF REFLECTOSCOPE SPERRY - TYPE "UR" MODIFIED

TYPE OF CRYSTAL 1 MC - CERAMIC - RIGHT ANGLE MOUNT

DIRECTION OF SCANNING RADIALLY AND LONGITUDINALLY - BACK REFLECTION SET TO 3" PEAK TO PEAK IN AN INDICATION-FREE AREA.

NO INDICATIONS

Ultrasonic Test Witnessed by
U. S. Coast Guard

[illegible]

OPERATOR

FOREMAN J. [illegible] [illegible]

MAGNETIC PARTICLE TESTED IN ACCORDANCE WITH SPECIFICATION OR PROCEDURE AS DETAILED BELOW:

OPERATOR

FOREMAN

Fig. C-4 - Ultrasonic and magnetic particle inspection report for the flange of the Army MH-1A reactor pressure vessel

UNITED STATES STEEL CORP.
HOMESTEAD DISTRICT WORKS

ULTRASONIC & MAGNETIC PARTICLE TEST REPORT

TYPE FORGING [illegible] HEAD SERIAL 2616-2

HEAT # X25646 ANALYSIS T-316 STAINLESS

CUSTOMER P. F. AVERY ORDER # J-300-3

ULTRASONIC TESTED IN ACCORDANCE WITH SPECIFICATION OR PROCEDURE AS DETAILED BELOW:

TYPE OF REFLECTOSCOPE SPERRY - TYPE "UR" MODIFIED

TYPE OF CRYSTAL 1 MC - CERAMIC - RIGHT ANGLE MOUNT

DIRECTION OF SCANNING THROUGH THICKNESS - BACK REFLECTION SET TO 3" PEAK TO PEAK IN AN INDICATION-FREE AREA.

	%	LOSS	DISTANCE FROM CRYSTAL	DISTANCE FROM EDGE
#1	10	40	[illegible]-1/4"	29-1/2"
#2	20	30	10-1/2"	[illegible]"
#3	20	0	[illegible]-1/4"	30"
#4	10	0	11-1/2"	37"
#5	10	0	9"	32"
#6	50	40	9"	26"
#7	15	30	10-1/4"	1-3/4"
#8	15	25	9"	29-1/2"
#9	15	25	15-1/2"	23"

Ultrasonic Test witnessed by
U. S. Coast Guard

OPERATOR

FOREMAN J. [illegible] SHEA [illegible]

MAGNETIC PARTICLE TESTED IN ACCORDANCE WITH SPECIFICATION OR PROCEDURE AS DETAILED BELOW:

ABOVE [illegible] RETESTED AFTER CONTOURING AND PRIOR TO SHIPMENT. NO INDICATIONS NOTED. THIS INFORMATION [illegible] TEST WAS NOT [illegible] BY THE COAST GUARD REPRESENTATIVES.

DYE PENETRANT TEST ON CONCAVE SECTION DISCLOSED 1 [illegible] LONG. THIS INDICATION WAS REMOVED BY GRINDING APPROXIMATELY [illegible]" DEEP UPON ADVICE FROM THE CUSTOMER.

DYE PENETRANT TEST ON CONCAVE SIDE WITNESSED BY U. S. COAST GUARD.

OPERATOR

FOREMAN J. W. SHEA 2-11-[illegible]

Fig. C-5 - Ultrasonic and magnetic particle inspection report for the upper head of the Army MH-1A reactor pressure vessel

Security Classification

DOCUMENT CONTROL DATA - R & D		
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	2b. GROUP	
3. REPORT TITLE PRESSURE-VESSEL SURVEILLANCE PROGRAM FOR THE ARMY MH-1A FLOATING NUCLEAR POWER REACTOR		
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13. ABSTRACT

The pressure vessel surveillance program for the Army MH-1A, barge-mounted reactor was designed by NRL. The NRL-fabricated hardware for the surveillance program included two capsules for vessel wall locations in the reactor, two capsules for accelerated exposure locations, and six capsules for above-core, thermal control locations. Schematic drawings indicate the relative location of the test specimens, neutron flux monitors, and temperature monitors in each capsule section. The manufacturer of the vessel saved the reactor nozzle cutouts from the upper shell-course ring forging for test purposes. The Charpy V-notch and tensile specimens contained in the capsules were machined from one of these reactor nozzle cutouts. Schematic drawings show the location of each test specimen in the parent material. Selected Charpy-V and tensile specimens have been tested to develop preirradiation properties of the AISI 316 stainless-steel reactor vessel for future reference.

DD FORM 1 NOV 65 1473 (PAGE 1)

S/N 0101-807-6801

Security Classification

14 KEY WORDS	LINK A		LINK B		LINK C	
	ROLE	WT	ROLE	WT	ROLE	WT
Nuclear reactors Pressure vessels Radiation damage Radiation damage surveillance Stainless steel reactors Tensile strength Ductility						