Enclosure 1

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69 NED 37
CLASS I
July 1969

MECHANICAL PROPERTY SURVEILLANCE OF GENERAL ELECTRIC BWR VESSELS

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ABSTRACT

The General Electric Company furnishes pressure vessel steel surveillance specimens for each BWR to permit the plant owner to account for radiation-induced changes in mechanical properties of the reactor vessel during service. This report presents several methods for determining the magnitude of these property changes. It also describes the specimens, specimen inventory, capsule design, associated equipment, material selection, and instructions for handling the specimens if they are tested.

1. SUMMARY

General Electric (GE) furnishes a special set of specimens with each Boiling Water Reactor (BWR) which may be used to monitor the effect of fast neutron fluence on the mechanical properties of the reactor vessel steel. At the fast neutron fluence levels for the 40-year lifetime of current BWR's these changes are minimal. This report suggests a variety of ways in which these potential radiation-induced changes may be accounted for in the operation of BWR power plants.

Tensile test and Charpy V-notch test bars representing the reactor-pressure-vessel base metal, weld metal and weld-heat-affected-zone (HAZ) metal, and neutron flux dosimeters are installed in the vessel. Sufficient samples are provided to measure changes to vessel mechanical properties throughout the vessel life if they are removed and tested periodically by the owner. These specimens are placed in the reactor for insurance and for economic reasons. They may be used to give *absolute measurement* of any change in mechanical properties, if required, or they may be used to demonstrate that the pressure vessel material is less sensitive to radiation effects than material of this type as defined by a conservative published curve.

Exposure levels in the BWR cause no change in tensile properties for A302B/A533B steel which would affect plant operation. The nil-ductility transition temperature (NDT) of this steel is increased by exposure to neutron fluences greater than 10^{17} nvt. A large body of experimental data has been gathered to document the extent of this change. Calculations for predicting neutron fluence at the vessel wall are being verified by experimental techniques, and it will soon be possible to relate electrical power output of the plant to neutron fluence at the wall with absolute confidence.

When the neutron fluence is known, it can be related to published curves of shift in transition temperature versus fluence to provide a conservative measure of the minimum temperature where the reactor vessel material will remain ductile. The plant owner thus has a variety of choices to account for mechanical property changes. These include measuring or calculating the maximum vessel-wall fast neutron fluence and either testing irradiated specimens or using published data to relate neutron fluence to radiation-induced property change.

2. INTRODUCTION

Pressure vessel steels in nuclear power plants are exposed to an environment of neutron flux and temperature which may produce a measurable effect on the mechanical and physical properties of vessel material. From the viewpoint of the plant operator, changes in the nil-ductility transition temperature of the pressure vessel are of interest. The temperature where ferritic steels may break in a brittle rather than a ductile mode is raised by exposure to a neutron fluence greater than 10^{17} nvt ($>$ 1 MeV). Therefore, some adjustment of the minimum vessel pressurization temperature may be required to accommodate shifts in the NDT. The tensile yield strength is increased and ductility is decreased by exposures above 10^{18} nvt ($>$ 1 MeV). No action on the part of the plant operator is necessary to accommodate changes in the vessel steel tensile properties. General Electric recommends the minimum vessel pressurization temperature be at least 60°F above the NDT to assure ductile behavior of the vessel material.

The predicted maximum vessel wall fluence for the current BWR product line is presented in Table 1. Note that these numbers are calculated conservatively since they assume the plant operates at 100 percent power, 100 percent of the time during a 40-year plant life. Sound practice requires that any radiation-induced changes in the NDT temperature, although small at these fluence levels, be accounted for in plant operation. General Electric furnishes special groups of test specimens for each BWR and provides the owner with several options to determine the extent of pertinent radiation-induced mechanical property changes.

Table 1
CALCULATED MAXIMUM VESSEL WALL FLUENCE LEVELS FOR STANDARD GE PLANTS

Vessel Inner Diameter	Number of Fuel Elements	Thermal Power Rating (MW)	Maximum Wall Neutron Flux (nv> 1 MeV)	40-Year Neutron Fluence[1] (nvt> 1 MeV)
183	368	1590	1.1×10^9	1.4×10^{18}
218	548	2380	5.5×10^8	7.0×10^{17}
218	560	2440	6.8×10^8	8.6×10^{17}
251	764	3440	3.0×10^8	3.8×10^{17}
269	880	4240	1.4×10^8	1.7×10^{17}

Note 1. Calculation assumes 100% power, 100% of the time during plant life.

3. TECHNICAL BASIS

Considerable data are available relating neutron fluence to change in NDT for ferritic materials. For A302B/A533B - Class I steels, these data are usually based on the 30 ft-lb Charpy V-notch energy transition temperature which has been correlated[1] with the dropweight specimen NDT for this steel. Figure 1 shows results from earlier work performed by Carpenter, et al., on the effect of neutron radiation on various heats of A302B/A533B - Class I steel exposed to reactor operating conditions.[2] Note the curves and equations for radiation-sensitive heats (upper and lower curves respectively). Figure 2 presents a comparison of the basic Carpenter[2] reference data shown in Figure 1 against results of several GE-BWR surveillance programs. In addition, Figure 2 includes the curves and equations for the upper limit for the original Carpenter data (upper curve labeled *worst-case*) as well as the upper limit for GE-BWR operating experience. The upper *worst-case* curve represents the most conservative, technically justified estimate of the radiation sensitivity of A302B/A533B - Class I steel. The lower curve reflects the upper limit for GE 550°F BWR operating experience. Note that the GE data include radiation effects on A302B/A533B - Class I weld and weld HAZ material as well as base metal.

Tabulated results of how the minimum pressure test temperature may be adjusted, provided the fluence for a given period of time is known, are given in Table 2. These results are based on *worst-case* performance on material with an initial NDT of 40°F and include the 60°F margin above NDT required by the GE Technical Specifications.

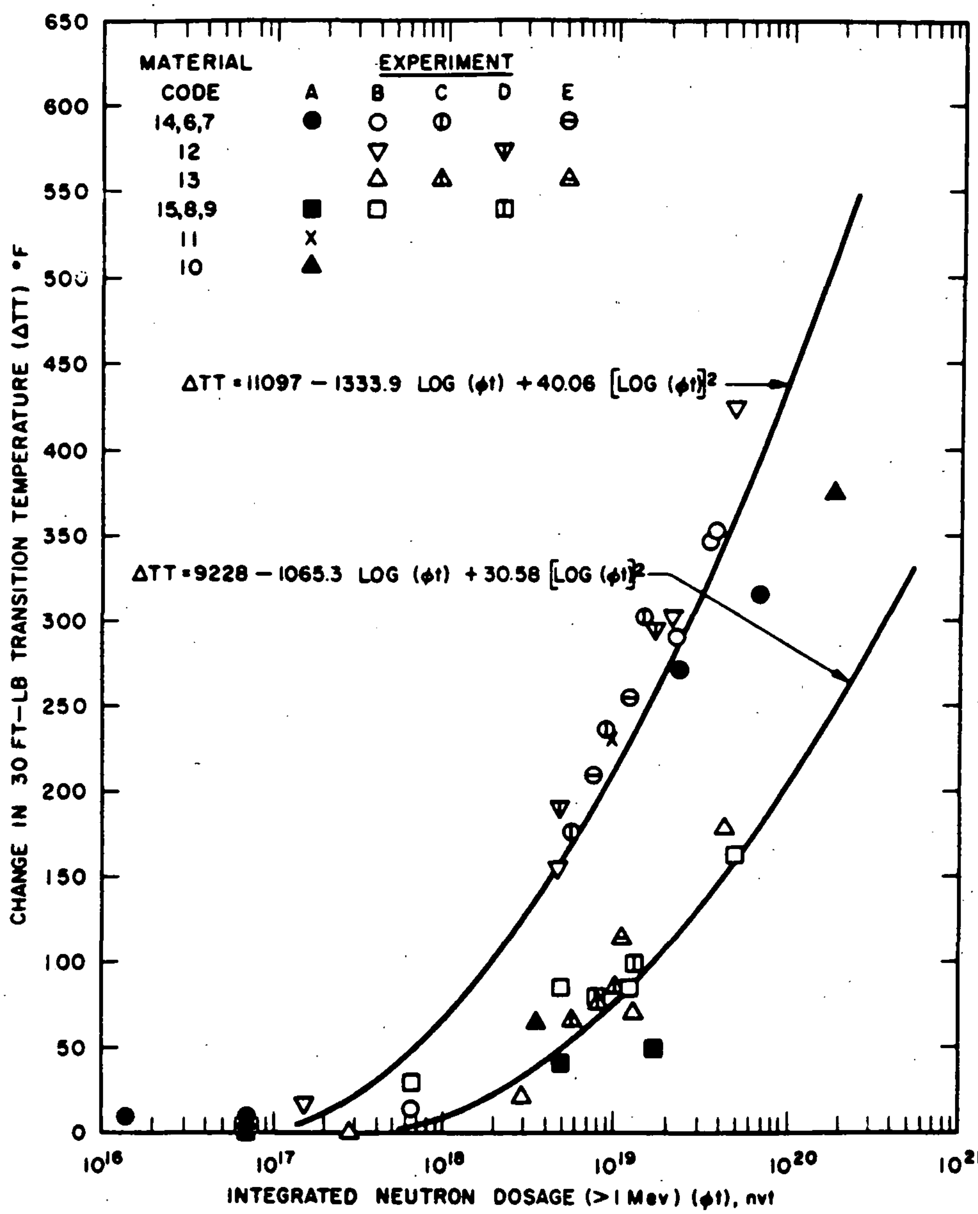


Figure 1. The Effect of Irradiation on Various Heats of A302B Steel "Sensitive" and "Insensitive" Irradiation Behaviors are Demonstrated by the Upper and Lower Curves, Respectively. (By permission of the American Nuclear Society, Sept 8, 1964)

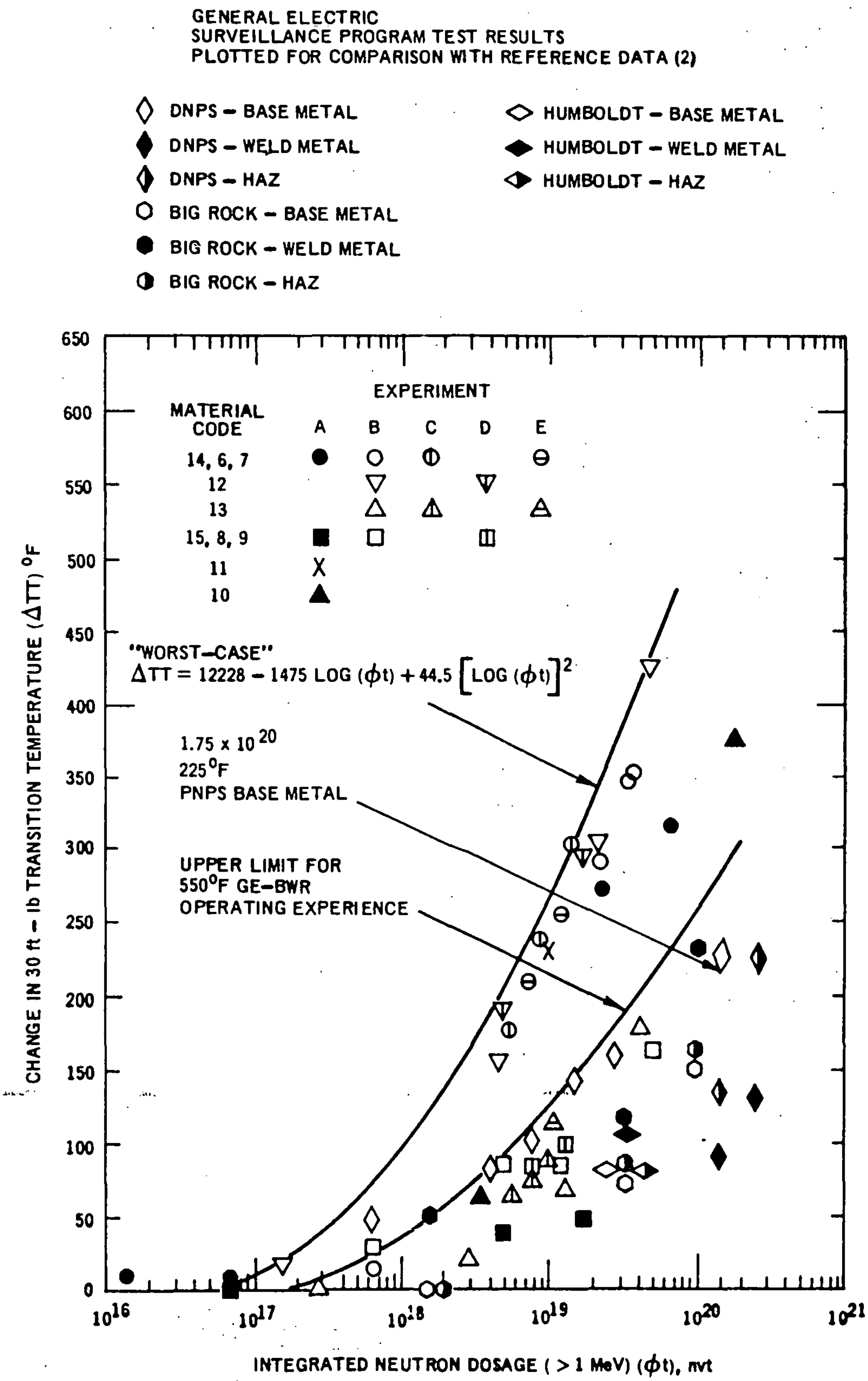


Figure 2. The Effect of Irradiation on Various Heats of A302B/A533B-Class 1.

Table 2
MOST CONSERVATIVE ESTIMATE* OF MINIMUM PRESSURE TEST TEMPERATURE (°F) FOR 40-YEAR LIFETIME

Years of Operation	Projected Neutron Fluence (nvt > 1 MeV)						
	5×10^{16}	1×10^{17}	2×10^{17}	5×10^{17}	1×10^{18}	2×10^{18}	5×10^{18}
0	100	100	100	100	100	100	100
5	100	100	100	108	115	128	165
10	100	100	100	115	128	155	200
15	100	100	106	122	142	174	220
20	100	100	112	130	155	190	242
25	100	103	115	136	165	200	251
30	100	106	118	142	174	210	262
35	100	109	121	148	182	220	277
40	100	112	125	155	190	230	295

* Based on maximum initial 40°F NDT plus required 60°F margin and worst-case radiation sensitivity curve for A302B/A533B - Class I: $(\Delta TT)°F = 1228 - 1475 \log(\phi t) + 44.5 [\log(\phi t)]^2$

4. OPERATING PHILOSOPHY

Surveillance programs for early nuclear power plants were development oriented since, at the time of their construction, there were little data available on the effect of neutron radiation on pressure vessel steels. As results from these early surveillance programs and other General Electric environmental exposure development programs have become available, sufficient data have been generated for A302B/A533B steel to develop empirical equations adequate to predict the effect of various nuclear environments.

This increased information, coupled with a design change in the BWR which has reduced the neutron fluence at the vessel wall, has lessened the necessity for testing surveillance specimens to monitor mechanical property changes. It is now possible to relate the calculated or measured neutron fluence at the vessel wall to mechanical property changes by using published data. For the present generation of BWR plants, GE provides a standard series of surveillance specimens plus a special flux monitor capsule. The owner now has multiple options to determine any radiation-induced changes in mechanical properties. These options are presented in the succeeding sections.

In this regard, special attention should be called to the fact that changes in the NDT need be considered only if the owner, for whatever reason, chooses to pressurize the reactor vessel prior to power operation. The mode of operation of a BWR does not depend on pressurization by external means. The 1000 psi - 550°F operating conditions are dictated solely by the physical characteristics of the BWR coolant. This pressure-temperature relationship results in no appreciable stress on the reactor vessel until the water coolant temperature is in excess of 400°F (well above any temperature where potential shifts in the NDT become a factor for consideration). Further, there is no normal operating mode which would require that a BWR be pressurized at temperatures below where boiling would occur. Under these circumstances, surveillance of radiation-induced mechanical property changes become academic.

NEUTRON FLUX

The fast neutron flux at the vessel wall may be determined by either of the following techniques.

4.1.1 The neutron flux may be determined by analyzing dosimeter wire. This need be done only once, preferably after 1 year of operation or the first scheduled plant shutdown. Since there is a direct linear relationship between thermal power generation and neutron flux, the fluence at the vessel can be accurately calculated at any given time.

4.1.2 An alternate approach would be to compute the flux at the vessel wall based on nuclear considerations. This can be done currently with a slight risk of gross error. This risk will be eliminated within 5 years by experimental work. The neutron fluence in BWR plants is so low that even gross errors will cause no operating risk.

4.2 RADIATION EFFECTS ON A302B/A533B STEEL

The owner may account for radiation effects on A302B/A533B vessel steel by utilizing any of the four methods outlined below.

The upper limit for GE-BWR data relating change in 30 ft-lb transition temperature to fluence as shown in Figure 2 may be used. There is some risk that a particular heat of steel may be more sensitive than the curve indicates, although the data include sufficient points to make this risk very low. The low BWR neutron fluence reduces the risk even more.

4.2.2 A more conservative approach is to use the upper boundary for all data developed for this steel (worst-case) including BWR, PWR, and test reactor experimental data (Figure 2). This method may prove to be costly in terms of extra time and expense for preheating the reactor vessel prior to hydrotest.

4.2.3 A third approach is to test one set of impact specimens irradiated in the reactor to establish the radiation sensitivity of the material. This option should be considered only if the cost of preheating the vessel to the higher temperature dictated by a conservative curve per 4.2.1 or 4.2.2 exceeds the cost of testing specimens. It incurs a secondary risk that the test may demonstrate that the material is as sensitive as the conservative curve.

4.2.4 Finally, the owner may choose to test all the impact specimens over the life of the plant. There is little engineering and no economic incentive to do this since so much data for A302B/A533B steel are already available.

4.3 RADIATION EFFECTS IN WELD AND WELD HAZ MATERIAL

The owner may account for radiation effects in weld and weld HAZ of A302B/A533B material by any of the four following methods.

4.3.1 The upper limit for GE-BWR data (Figure 2) may be used. At present, this involves some uncertainty since limited data are available for these materials. The risk is low because of the low neutron fluence of the BWR.

4.3.2 A more conservative approach is to use the *worst-case* curve for base material. This includes results from a wide variety of heats and radiation temperatures ranging from 450 to 575°F. By analogy, the weld and weld HAZ metal can be expected to behave no worse than the worst of these. Use of this curve may be costly for reasons given in 4.2.2 above.

4.3.3 A third option is to test one set of impact specimens for economic reasons per the rationale developed in 4.2.3.

4.3.4 A final choice is to test all impact specimens over the life of the plant to develop additional data with the objective of developing a more reliable empirical curve.

4.4 COMBINED TESTING

If the owner decides to test irradiated specimens and exercises options 4.2.3, 4.2.4, 4.3.3, or 4.3.4, he should consider combining forces with other nuclear power reactor owners to establish either one-point or three-point curves by irradiating the steels from each reactor vessel in a single experiment, in one reactor and having all post-irradiation tests performed at the same laboratory at one time.

5. PROGRAM

IMPLEMENTATION

The surveillance specimens must accurately represent the pressure vessel to be monitored. Accurate measurement or calculation of the maximum neutron flux at the vessel wall is required. Good equipment design is required to assure accurate specimen positions and protection for long periods, and yet allow inexpensive fabrication and post-irradiation handling.

General Electric provides tensile and Charpy V-notch mechanical test specimens from the reactor vessel base metal, weld HAZ metal, and weld metal from a joint made on the reactor steel which simulates a welded joint in the reactor vessel. Neutron dosimeter wire is included with each group of specimens. Specimens are placed near the reactor vessel wall where the exposure is similar to that of the vessel wall.

5.2 TYPE OF SPECIMENS

Small tensile specimens (see Drawing 921D276*) are provided and may be used to measure the effect of neutron irradiation on the yield strength, and ductility of the pressure vessel materials.

Standard Charpy V-notch specimens (see Drawing 921D277) are provided and may be used to measure the effect of neutron irradiation on the fracture energy transition temperature of the pressure vessel materials.

5.3 MATERIALS

Charpy V-notch and tensile specimens are made from the base metal used to fabricate the vessel (see Appendix B). The specimens are made from flat slabs cut parallel to and one-quarter plate thickness from both of the plate surfaces, and are machined with their longitudinal axis parallel to the plate rolling direction. The notches of the Charpy specimens are cut perpendicular to the plate surface.

A test weld representing a vessel welded joint is fabricated from vessel base material. Charpy and tensile specimens representing the weld HAZ and the weld metal are fabricated from the test weld material.

The weld Charpy specimens are made transverse to the weld direction; thus, only the central section of the specimen is necessarily composed of weld-deposited metal. Specimens are taken throughout the weld section to a distance of 3/4 inch from the weld root. Their long axes are parallel to the plate surface and normal to the weld length. The Charpy notch is made perpendicular to the plate surface.

The gage lengths of the weld tensile specimens are made entirely of weld metal by taking the specimens parallel to the weld length and parallel to the plate surface.

The weld HAZ tensile specimens are made transverse to the weld length and parallel to the plate surface. The center of the specimen is located at the joint between the base metal and weld deposit

The weld HAZ Charpy specimens are made transverse to the weld length and parallel to the plate surface. The axis of the notch is perpendicular to the plate surface, and the notch radius is located at the intersection of the base metal and weld deposit.

All specimens are marked with FAB Code (see Appendix C) on one end.

5.4 BASKET AND CAPSULE DESIGN

The tensile and impact specimens are placed in capsules which are held in capsule baskets. These baskets are in turn fastened to a holder (Drawing 117C3910) and suspended from a bracket on the reactor vessel wall inside diameter. The capsule basket (Drawing 117C3911) is binary coded according to the reactor number and sequence. The identification number system for various GE reactors is contained in Drawing 129B3578.

The basket confines the capsules to a small area, which ensures that the neutron fluence for each capsule will be similar. It reduces the amount of reactor handling at removal time because only one transfer is required. It also reduces the number of vessel brackets required in the reactor structure to hold the specimens. The basket can serve as a container for storage or shipping if the specimens are to be tested in a hot laboratory.

The irradiation capsules are designed to hold the specimens in a dry helium atmosphere in a small space to reduce neutron flux variation among the samples. The tensile capsule is a thin-wall tube. An aluminum filler piece is used to ensure good heat flow between the specimen gage length and the capsule wall (Drawing 117C3912). The impact capsule is a rectangular box (Drawing 117C3913) with thin walls. Reactor water pressure assures good contact between the capsule wall and the specimens. Under these conditions, gamma heat produced in the specimen flows out of the capsule with a minimum gradient, and the specimen temperature is close to the water temperature.

5.5 CAPSULE LOADING

Two tensile specimens are placed in each tensile capsule. These capsules are FAB-coded on one end. Drawing 117C3912 shows a typical tensile specimen capsule loading. Twelve impact specimens, an iron, a nickel, and a copper dosimeter wire are placed in each impact capsule. The impact capsules are binary-coded sequentially as shown in Drawing 117C3913.

5.6 SPECIMEN LOCATIONS

The specimens are positioned at three locations in the reactor vessel adjacent to the inner vessel wall at positions radially adjacent to the core mid-plane where the neutron flux will be highest. Typical locations are shown in Drawing 921D435. These specimens are exposed to a neutron flux with a rate and spectrum similar to that of the vessel wall.

* All drawings are shown in Appendix A.

5.7 CAPSULE DOSIMETERS AND TEMPERATURE MONITOR

Each impact capsule contains iron, nickel, and copper wire dosimeters. These dosimeters can be used to determine the neutron fluence by performing radiochemical analyses for the particular nuclear reaction of interest as outlined in Appendix D. These radiochemical measuring techniques require great care to attain valid results.

Because the boiling water reactor is a constant-temperature device, no special temperature monitoring devices are required.

5.8 VESSEL WALL DOSIMETERS

The specimen capsules and their flux monitors are always a finite distance from the reactor pressure vessel wall. If one is to calculate the neutron fluence at the pressure vessel wall using data from the capsule flux monitors, he must know the exact location of the capsule monitor with respect to the wall and the flux gradient between the capsule monitor and the wall. Since the flux gradient is very steep, small errors in determining the position or the amount of the gradient can cause gross errors in calculating the wall fluence. This error will always be conservative but it can be an economic problem since it may lead to excessive preheating costs. General Electric provides a precise measurement of the distance between the capsules and the wall. In addition, one capsule basket has special iron and copper dosimeter wire contained in a capsule holder (Drawing 158B7636) located as close as practical to the wall. These dosimeters (Drawing 158B7635) can be removed independently of the surveillance samples. General Electric recommends that for plants where the owner decides to measure one of these monitors that it be removed during the first refueling outage since the iron dosimeter wire is most accurate for a period of about 1 year.

5.9 TYPICAL PROGRAM

Capsules are installed in the reactor prior to startup. Capsule removal schedules may vary, depending on the type of information needed, which, in turn, depends on the integrated neutron flux at the vessel wall. A list of specimens included in a typical surveillance program is presented in Table 3.

Details of all specimens with their locations in the capsules, all capsules with their locations in the baskets, and all baskets with their locations in the reactor, are shown on drawings in Appendix A.

5.10 SPECIMEN TESTING AND REPORTS

The specimens must be tested under special conditions, using remote handling and shielding. Such tests are made most conveniently in a radioactive materials laboratory. Careful supervision of all tests is required to assure accurate results and to avoid wastage of the limited number of specimens. Suggestions for capsule handling and testing are given in Appendix E. Reports should be requested for each batch of specimens removed and tested.

Table 3
SPECIMENS FURNISHED FOR TYPICAL SURVEILLANCE PROGRAM

Reactor Group Number	Type Specimen	Number of Specimens Base	Weld	HAZ
	Unirradiated Base-Line Specimens			
-	C(a)	12	12	12
	T(b)	3	3	3
	In-Reactor Specimens			
1	C	12	12	12
	T	2	2	2
2	C	8	8	8
	T	2	2	2
3	C	8	8	8
	T	2	2	2
	Out of Reactor Spares			
-	C	12	12	12
	T	3	3	3
	Total Specimens			
-	C	52	52	52
	T	12	12	12

(a) C = standard Charpy V-notch impact specimen
(b) T = 1/4 inch gage diameter tensile specimen

REFERENCES

1. Pellini, W. S., and Puzak, P. P., *Practical Considerations in Applying Laboratory Fracture Test Criteria to the Fracture-Safe Design of Pressure Vessels,* Journal of Engineering for Power, Trans. ASME, Vol. 86 - Series A - No. 4, 1964, pp. 429-443.
2. Carpenter, G. F., Knopf, N. R., and Byron, E. S., *Anomalous Embrittling Effects Observed During Irradiation Studies on Pressure Vessel Steels,* Nucl. Sci. Eng., 19, 18-38 (1964).
3. *Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors* (E185), Book of ASTM Standards, Part 31 (1964).

APPENDIX A

DRAWINGS

The following drawings are contained in this appendix.

129B3578	Surveillance Program Number Identification of GE Reactors
921D435	Surveillance Program
921D277	Charpy Impact Specimen
921D276	Tensile Test Specimen
117C3910	Specimen Holder Surveillance Program
117C3911	Capsule Basket
117C3912	Tensile Specimen Capsule
117C3913	Impact Specimen Capsule
158B7636	Capsule Holder
158B7635	Neutron Dosimeter

GENERAL ELECTRIC

UNLESS OTHERWISE SPECIFIED USE THE FOLLOWING:— APPLIED PRACTICES | SURFACES √ | TOLERANCES ON MACHINED DIMENSIONS: FRACTIONS, DECIMALS, ANGLES

REV NO. 5

129B3578

CONT ON SHEET 2 SH NO. 1

TITLE
SURVEILLANCE PROGRAM NUMBER IDENTIFICATION OF G.E. REACTORS

FIRST MADE FOR

REACTOR NO.	PLANT NAME	SURVEILLANCE PROGRAM DWG.	BINARY CODE
1	VBWR (G.E.)		
2	DRESDEN UNIT #1 (COMMONWEALTH EDISON)	197E793 SH1&2	
3	GARIGLIANO POWER STA. (SENN)	117B1400	
4	JPDR (JAERI)		
5	BIG ROCK POINT N.S. (CONSUMERS POWER)	104B1343 SH3	
6	HUMBOLDT BAY UNIT 3 (PG&E)	104B1099 SH3	
7	VESR (G.E.)	104R667 SH.3	
8	DODEWAARD POWER STA. (SEP-GKN)		
9	TARAPUR (GOV'T OF INDIA)	129B3807	
10	KRB (RWE-BAYERNWERK)	129B3806	
11	NINE MILE POINT N.S. (NIAGARA MOHAWK)	129B3891	
12	OYSTER CREEK UNIT #1 (JERSEY CENTRAL)	129B3892	
13	DRESDEN UNIT #2 (COMMONWEALTH EDISON)	921D401	
14	SANTA MARIA DE GAROÑA (NUCLENOR)	ZB-1969-13	
15	MILLSTONE (CONN. LIGHT AND POWER, HARTFORD ELECTRIC LIGHT, WESTERN MASS. ELECTRIC CO.)	921D406	
16	PILGRIM (BOSTON EDISON)	921D507	
17	TSURUGA (JAPC)	921D225	
18	DRESDEN UNIT #3 (COMMONWEALTH EDISON)	921D401	
19	MONTICELLO (NORTHERN STATES POWER)	921D435	
20	QUAD CITIES #1 (COMMONWEALTH EDISON)	886D477	
21	QUAD CITIES #2 (COMMONWEALTH EDISON)	886D477	
22	FUKUSHIMA 1 (TEPCO)	921D457	
23	BROWNS FERRY 1 (TVA)	921D456	
24	VERMONT YANKEE (CENTRAL VT. PUBLIC SERV. & GREEN MT. PWR. CORP)	921D514	
25	PEACH BOTTOM 2 (PHIL. ELECTRIC)	921D496	
26	BROWN FERRY 2 (TVA)	921D456	
27	PEACH BOTTOM 3 (PHIL. ELECTRIC)	921D496	
28	J. FITZPATRICK (PWR. AUTHORITY ST. OF N.Y.)	921D465	
29	LILCO (LONG ISLAND LIGHTING CO.)		
30	COOPER STATION (CONSUMERS PUBLIC PWR.)	921D709	
31	BROWNS FERRY 3 (TVA)	921D456	
32	BELL STATION (N.Y. STATE G&E)		

NOTES:

1. THE HOLE PATTERN IS A BINARY NUMBER MADE BY ADDING THE VALUES OF THE HOLES AS INDICATED BELOW.

1 2 4 8 — CODING REF. HOLES

(THIS METHOD USED FOR REACTORS 1 THRU 7)

1 2 4 8 16 — CODING REF. HOLES

(THIS METHOD USED FOR REACTORS 8 THRU 31)

1 2 4 8 16 32 — CODING REF. HOLES

(THIS METHOD USED FOR REACTORS 32 THRU 63)

REVISIONS

			PRINTS TO
4 Feb 3 '69 H.H.T.	ADDED SHEET 2. ADDED DWG. NO. TO REACTORS NO. 16, 22 THRU 27 & 31.	3 Oct 18 '68 H.H.T.	GP2
5 H.H.T.	NE-22438 WCS 11-5-69	REDRAWN WITH CHANGES. ADDED REACTOR NO. 21 TO 32. ADDED DWG. NO. TO REACTOR NO. 13, 15, 17 THRU 21	RV2
			SC2

MADE BY P. LOVIER OCT. 4, 1965 | APPROVALS M.S.H. 10-5-65 | APED SAN JOSE | 129B3578 CONT ON SHEET 2 SH NO. 1

CHK'D BY: W. Swan 10-13-68 WCS 1-27-69

R

GENERAL ELECTRIC

UNLESS OTHERWISE SPECIFIED USE THE FOLLOWING:—

APPLIED PRACTICES	SURFACES	TOLERANCES ON MACHINED DIMENSIONS FRACTIONS	DECIMALS	ANGLES
	√	±	±	±

REV NO. Ø5

129B3578

CONT ON SHEET F SH NO. 2

TITLE
SURVEILLANCE PROGRAM NUMBER IDENTIFICATION OF G.E. REACTORS

FIRST MADE FOR

REACTOR NO.	PLANT NAME	SURVEILLANCE PROGRAM DWG.	BINARY CODE
33	FUKUSHIMA II (TEPCO)	921D471	●○○○○●
34	AKM (BERNISHE KRAFT WERK)	ZB-1969-16	○●○○○●

REVISIONS

FOR REVISIONS SEE SHEET 1.

PRINTS TO
GP 2
RV 2
SC 2

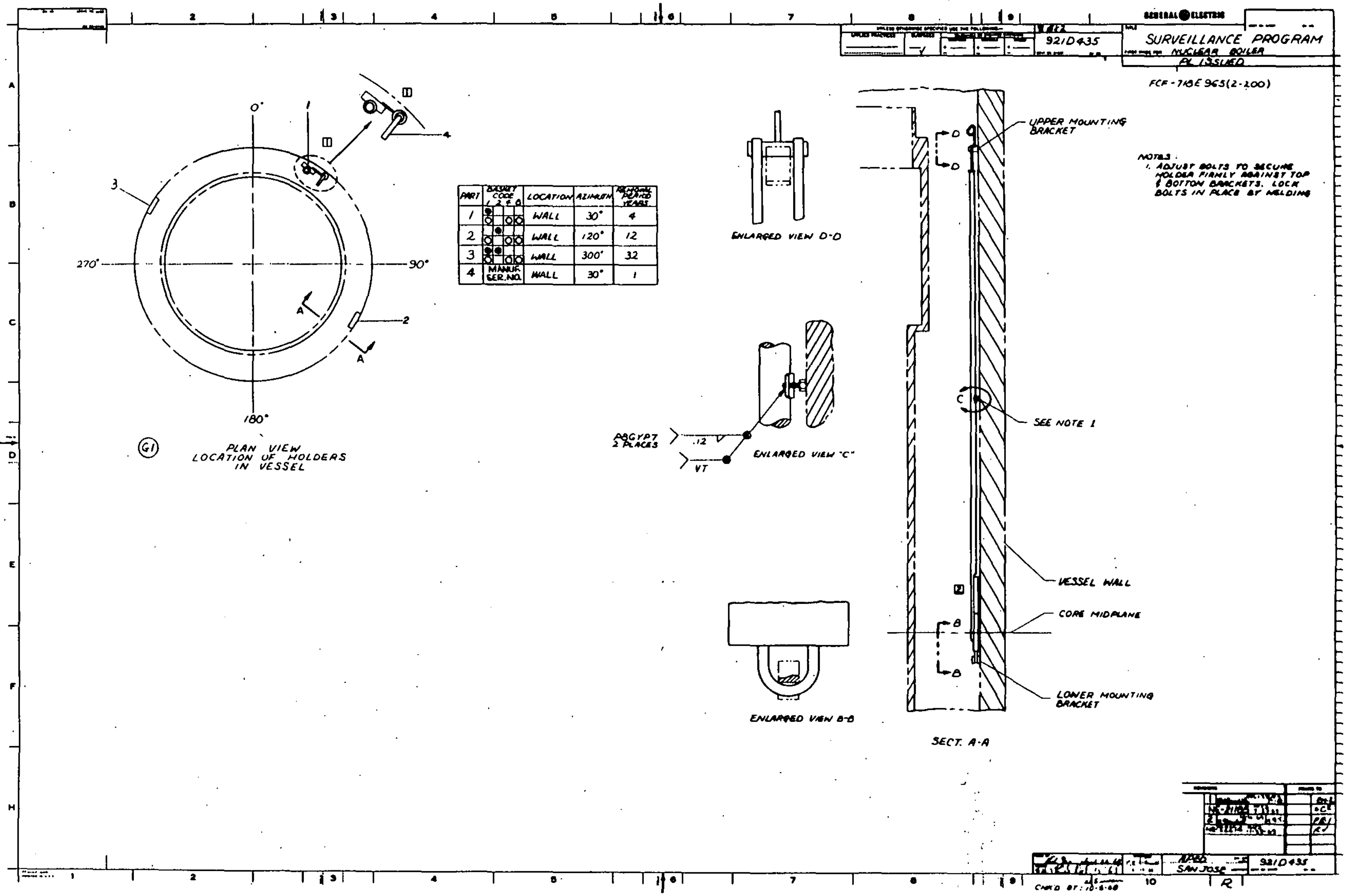
MADE BY Edwar Bu Jan 23 '69 | APPROVALS R J Locke 2-11-69 | APED SAN JOSE | 129B3578 CONT ON SHEET — SH NO 2

ISSUED H. H. Third Feb. 13 '69

CHK'D BY: W Swan 1-27-69

R

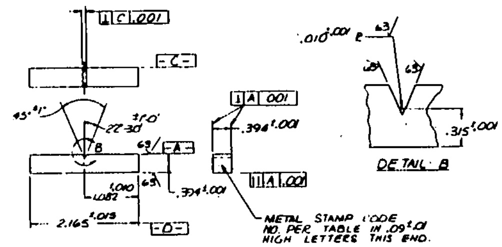
FF-000-F (12-67) PRINTED IN U.S.A.



PART	BASKET CODE 1 2 4 8	LOCATION	AZIMUTH	REMOVAL PERIOD YEARS
1		WALL	30°	4
2		WALL	120°	12
3		WALL	300°	32
4	MANUF. SER. NO.	WALL	30°	1

CHARPY IMPACT SPECIMEN
SURVEILLANCE PROGRAM

921D277



PCF:
P60 TO P62 - 117C4351
P63 TO P65 - 117C3913

PCF P1 TO P3 - 117C3572
P4 TO P13 - 117C3728
P14 TO P23 - 117C3570
P24 TO P26 - 117C3769
P27 TO P29 - 117C3915
P30 TO P32 - 117C3911
P33 TO P38 - 117C4087
P39 TO P47 - 117C4059
P48 TO P50 - 117C4066
P51 TO P53 - 117C4028
P54 TO P56 - 117C4083
P57 TO P59 - 117C4191

NOTE:

1. 63/ SURFACES SHALL BE GROUND OR MACHINED WITHOUT NOTICABLE TOOL MARKS.

PART	MATERIAL TYPE	MATERIAL SPECIFICATION	FAB. CODE NO. PER 158B7275 NO. TO RUN SERIALLY	PART	MATERIAL TYPE	MATERIAL SPECIFICATION	FAB. CODE NO. PER 158B7275 NO. TO RUN SERIALLY	PART	MATERIAL TYPE	MATERIAL SPECIFICATION	FAB. CODE NO. PER 158B7275 NO. TO RUN SERIALLY
1	BASE	21A1106	1I1 TO 1BY	27	BASE	21A1112	DI1 TO D3Y	54	BASE	21A1115	JB1 TO JOY
2	WELD		1C1 TO 1TY	28	WELD		D41 TO D6Y	55	WELD		JE1 TO JKY
3	HEAT AFFECTED ZONE (HAZ)		1U1 TO 27Y	29	HAZ		D71 TO DBY	56	HAZ		JL1 TO JPY
4	BASE	21A1109	5I1 TO 55Y	30	BASE	21A1114	3K1 TO 3PY	57	BASE	21A1100AJ	EP1 TO EUY
5	WELD (SUB-ARC)		6I1 TO 65Y	31	WELD		3Y1 TO 4IY	58	WELD		EY1 TO J2Y
6	WELD (ELECTROSLAG)		561 TO 5EY	32	HAZ		421 TO 45Y	59	HAZ		J31 TO J5Y
7	HAZ (SUB-ARC)		661 TO 6CY	33	BASE	21A1111	736 TO 77Y	60	BASE	21A1100AF	5I1 TO 55Y
8	HAZ (ELECTROSLAG)		5J1 TO 5UY	34	WELD		7A1 TO 7CL	61	WELD		541 TO 56Y
9	BASE	21A1109	KI1 TO K5Y	35	HAZ		7CM TO 7JC	62	HAZ		5H1 TO 5BY
10	WELD (SUB-ARC)		LI1 TO L5Y	36	BASE	21A1111	7LM TO 7YC	63	BASE	21A1112	JCP TO JEE
11	WELD (ELECTROSLAG)		K61 TO KEY	37	WELD		7YD TO AI5	64	WELD		JEJ TO JKA
12	HAZ (SUB-ARC)		L61 TO LCY	38	HAZ		AI6 TO A3Y	65	HAZ		JKB TO JH4
13	HAZ (ELECTROSLAG)		KJ1 TO KUY	39	BASE	21A1111	DJ1 TO DLY				
14	BASE	21A1113	MI1 TO M5Y	40	WELD		DM1 TO DTY				
15	WELD (SUB-ARC)		PI1 TO P5Y	41	HAZ		DU1 TO EIY				
16	WELD (ELECTROSLAG)		M61 TO MEY	42	BASE	21A1111	E51 TO E7Y				
17	HAZ (SUB-ARC)		P61 TO PCY	43	WELD		EA1 TO ECY				
18	HAZ (ELECTROSLAG)		MJ1 TO MUY	44	HAZ		ED1 TO EJY				
19	BASE	21A1113	TI1 TO T5Y	45	BASE	21A1111	6A5 TO 66Y				
20	WELD (SUB-ARC)		UI1 TO U5Y	46	WELD		671 TO 6BL				
21	WELD (ELECTROSLAG)		T61 TO TEY	47	HAZ		6BM TO 6EC				
22	HAZ (SUB-ARC)		U61 TO UCY	48	BASE	21A1100AK	AG6 TO AAY				
23	HAZ (ELECTROSLAG)		TJ1 TO TUY	49	WELD		AB1 TO AOL				
24	BASE	21A1110	BI1 TO BCY	50	HAZ		AOM TO AKC				
25	WELD		BD1 TO BYY	51	BASE	21A1110	YI1 TO Y3Y				
26	HEAT AFFECTED ZONE (HAZ)		CI1 TO CCY	52	WELD		YA1 TO Y6Y				
				53	HAZ		Y71 TO YBY				

921D277

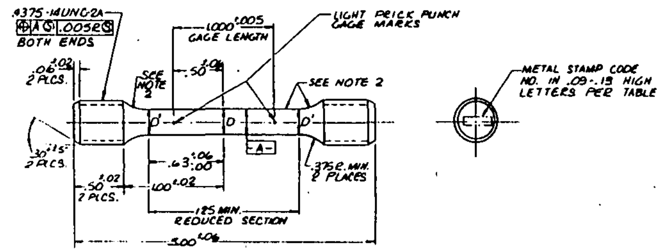
GENERAL ELECTRIC

921D276

TENSILE TEST SPECIMEN

SURVEILLANCE PROGRAM

MSASABL



FCF-P1 to P3 - 117C3572

P4 TO P13 - 117C3726
P14 TO P23 - 117C3731
P24 TO P26 - 117C3839
P27 TO P29 - 117C3342
P30 TO P32 - 117C3370
P33 TO P35 - 117C4046
P36 TO P41 - 117C4057
P42 TO P50 - 117C4065

P51 TO P53 - 117C4078
P54 TO P56 - 117C4082
P57 TO P59 - 117C4192
P60 TO P62 - 117C4850
P63 TO P65 - 117C3912

NOTES:

1. D = .250 ±.001 DIA. AT CENTER OF REDUCED SECTION.
   D' = ACTUAL D DIA. +.002 TO .005 AT ENDS OF REDUCED SECTION TAPERING TO D AT CENTER.
2. GRIND REDUCED SECTION & RADII TO 32√. RADII TO BE TANGENT TO REDUCED SECTION WITH NO CIRCULAR TOOL MARKS AT POINT OF TANGENCY OR WITHIN REDUCED SECTION. POINT OF TANGENCY SHALL NOT LIE WITHIN REDUCED SECTION.

PART	MATERIAL TYPE	MATERIAL SPECIFICATION	FAB. CODE NO. PER 15B07275. NO. TO RUN SERIALLY
1	BASE		2Ä1 TO 2ÄY
2	WELD	21A1106	2Č1 TO 2ÖY
3	HEAT AFFECTED ZONE (HAZ)		2Ë1 TO 2J̈Y
4	BASE		6Ḋ1 TO 6ĖY
5	WELD (SUB-ARC)		6Ṗ1 TO 6İY
6	WELD (ELECTROSLAG)	21A1109	6J̇1 TO 6K̇Y
7	HAZ (SUB-ARC)		6U̇1 TO 6ẎY
8	HAZ (ELECTRO SLAG)		6L̇1 TO 6ṀY
9	BASE		LḊ1 TO LĖY
10	WELD (SUB-ARC)	21A1109	LṖ1 TO LİY
11	WELD (ELECTROSLAG)		LJ̇1 TO LK̇Y
12	HAZ (SUB-ARC)		LU̇1 TO LẎY
13	HAZ (ELECTRO SLAG)		LL̇1 TO LṀY
14	BASE		PḊ1 TO PĖY
15	WELD (SUB ARC)		PṖ1 TO PİY
16	WELD (ELECTROSLAG)	21A1113	PJ̇1 TO PK̇Y
17	HAZ (SUB-ARC)		PU̇1 TO PẎY
18	HAZ (ELECTROSLAG)		PL̇1 TO PṀY
19	BASE		UḊ1 TO UĖY
20	WELD (SUB-ARC)		UṖ1 TO UİY
21	WELD (ELECTROSLAG)	21A1113	UJ̇1 TO UK̇Y
22	HAZ (SUB-ARC)		UU̇1 TO UẎY
23	HAZ (ELECTROSLAG)		UL̇1 TO UṀY
24	BASE		CḊ1 TO CJ̇Y
25	WELD	21A1110	CK̇1 TO CṀY
26	HEAT AFFECTED ZONE (HAZ)		CṖ1 TO CU̇Y

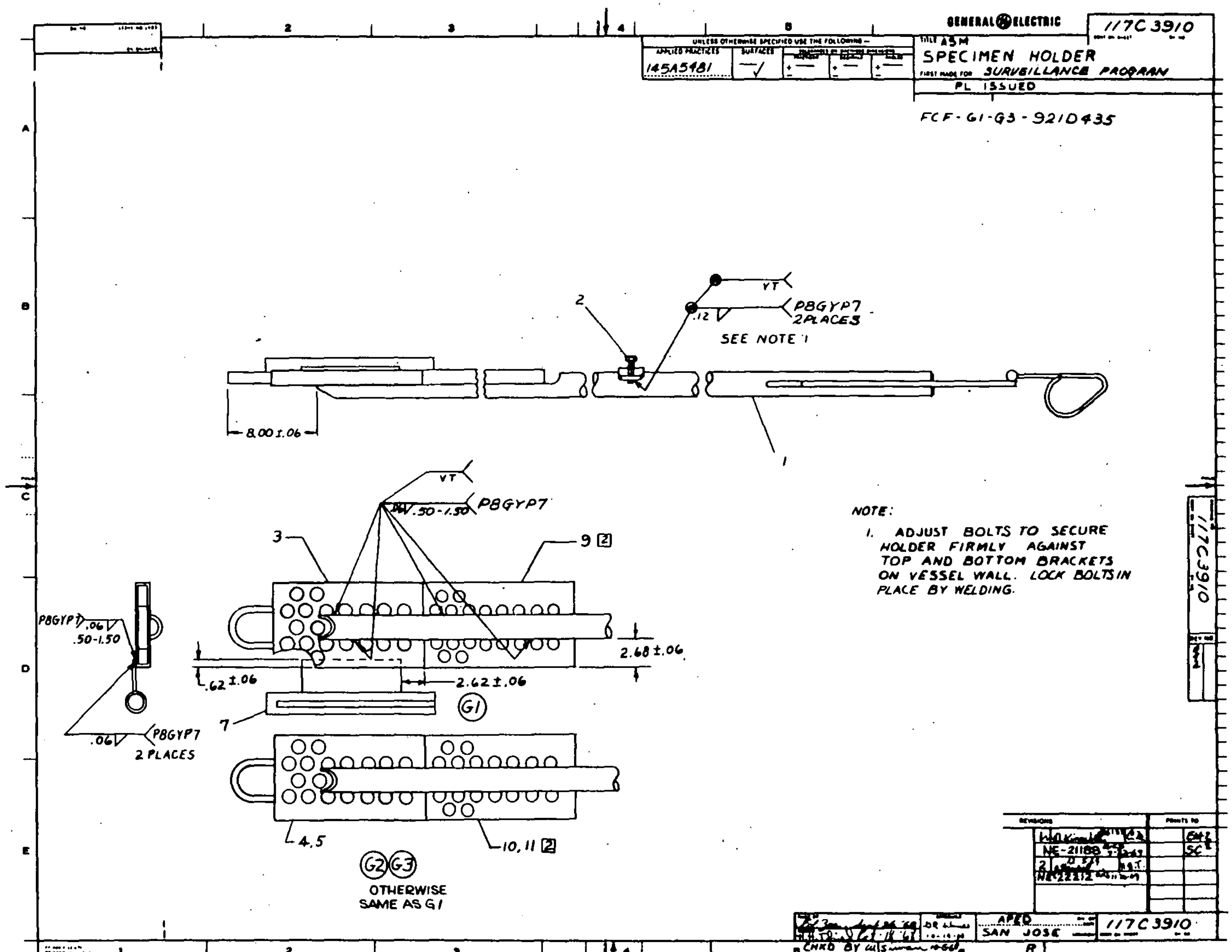
PART	MATERIAL TYPE	MATERIAL SPECIFICATION	FAB. CODE NO. PER 15B07275. NO. TO RUN SERIALLY
27	BASE		DČ1 TO DČY
28	WELD	21A1112	DḊ1 TO DḊY
29	HAZ		DĖ1 TO DĖY
30	BASE		4G̈1 TO 4G̈Y
31	WELD	21A1114	47̈1 TO 47̈Y
32	HAZ		4Ä1 TO 4ÄY
33	BASE		7J̈D TO 7K̈5
34	WELD	21A1111	7K̈6 TO 7K̈Y
35	HAZ		7L̈1 TO 7L̈L
36	BASE		A4̈1 TO A4̈L
37	WELD	21A1111	A4̈M TO A5̈C
38	HAZ		A5̈D TO A6̈5
39	BASE		E2̇1 TO E2̇Y
40	WELD	21A1111	E3̇1 TO E3̇Y
41	HAZ		E4̇1 TO E4̇Y
42	BASE		EK̇1 TO EK̇Y
43	WELD	21A1111	EL̇1 TO EL̇Y
44	HAZ		EṀ1 TO EṀY
45	BASE		6ËD TO 6J̈5
46	WELD	21A1111	6J̈6 TO 6J̈Y
47	HAZ		6K̈1 TO 6K̈L
48	BASE		AK̈D TO AL̈5
49	WELD	21A1100AK	AL̈6 TO AL̈Y
50	HAZ		AM̈1 TO AM̈L
51	BASE		YĊ1 TO YĊY
52	WELD	21A1110	YḊ1 TO YḊY
53	HAZ		YĖ1 TO YĖY

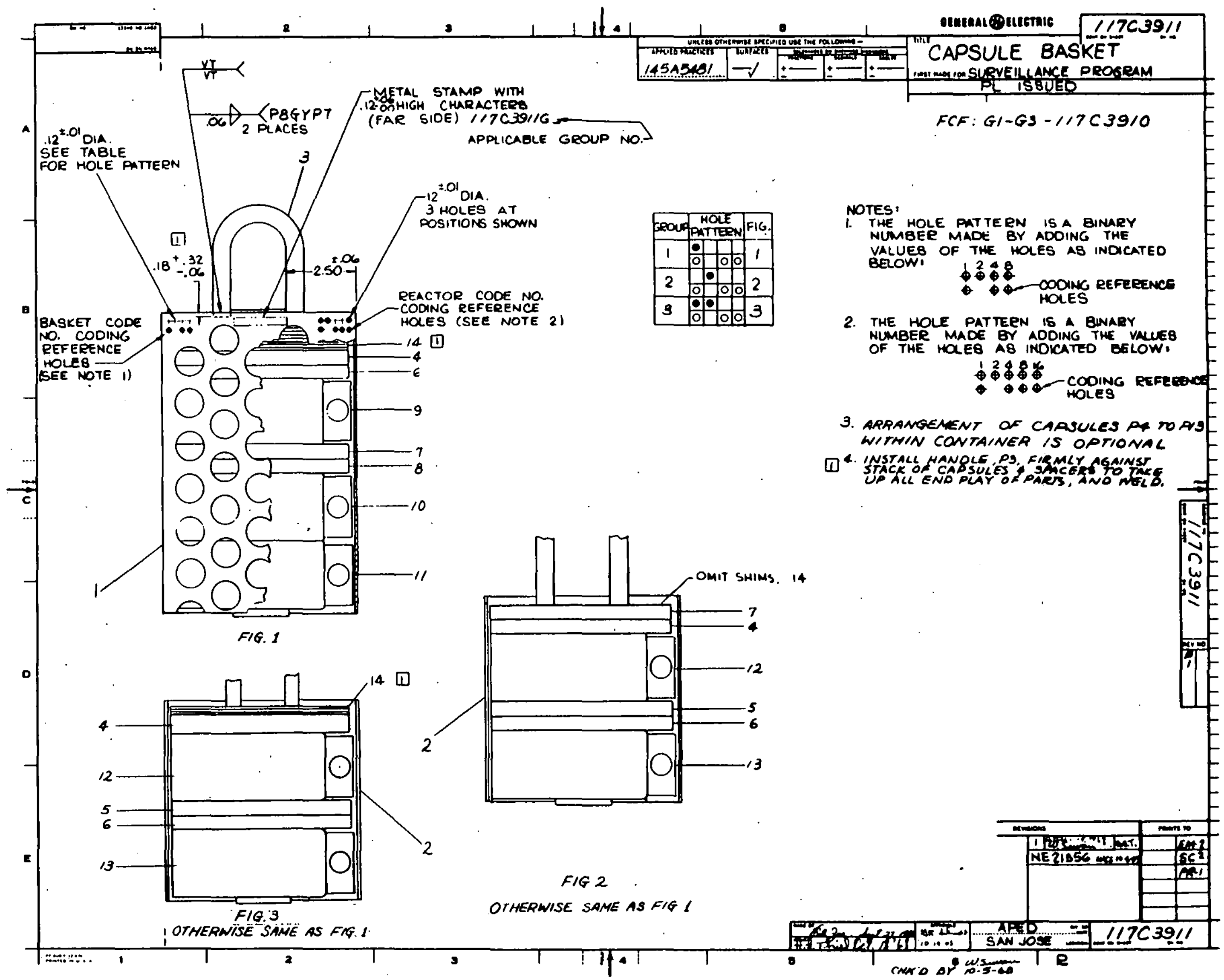
PART	MAT'L TYPE	MATERIAL SPECIFICATION	FAB. CODE NO. PER 15B07275. NO. TO RUN SERIALLY
54	BASE		JṪ1 TO JṪY
55	WELD	21A1115	JU̇1 TO JU̇Y
56	HAZ		JẎ1 TO JẎY
57	BASE		JĠ1 TO JĠY
58	WELD	21A1100AJ	J7̇1 TO J7̇Y
59	HAZ		JȦ1 TO JȦY
60	BASE		5C̈1 TO 5C̈Y
61	WELD	21A1100AF	5D̈1 TO 5D̈Y
62	HAZ		5Ë1 TO 5ËY
63	BASE		JB̈1 TO JB̈E
64	WELD	21A1112	JB̈J TO JC̈4
65	HAZ		JC̈5 TO JC̈M

ADDED PT 30 TO 32
ADDED PT 33 TO 35
ADDED PT 36 TO 38
ADDED PT 27 TO 29

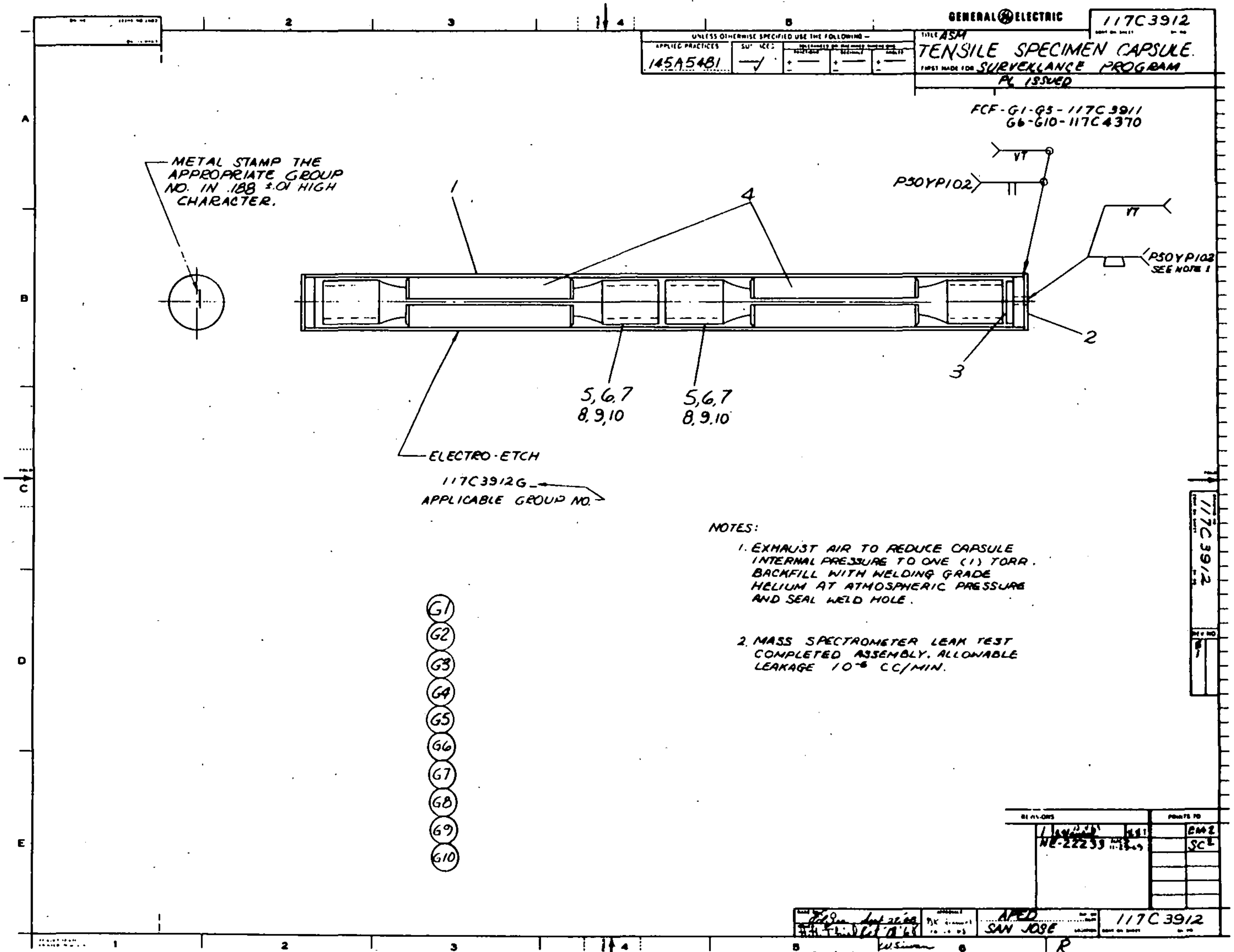
APED SAN JOSE

921D276


GENERAL ELECTRIC
117C3910
ASM
SPECIMEN HOLDER
SURVEILLANCE PROGRAM
PL ISSUED
145A5481
FCF-G1-G3-921D435
2
VT
P8GYP7
2 PLACES
SEE NOTE 1
8.00 ± .06
1
P8GYP7
.50-1.50
3
9
2.68 ± .06
.62 ± .06
2.62 ± .06
7
G1
.06
2 PLACES
4,5
10,11
G2 G3
OTHERWISE
SAME AS G1
NOTE:
1. ADJUST BOLTS TO SECURE HOLDER FIRMLY AGAINST TOP AND BOTTOM BRACKETS ON VESSEL WALL. LOCK BOLTS IN PLACE BY WELDING.
NE-21188
NE-22212
SAN JOSE
117C3910



GENERAL ELECTRIC
117C3911
CAPSULE BASKET
SURVEILLANCE PROGRAM
PL ISSUED
145A5481
FCF: G1-G3-117C3910
METAL STAMP WITH .12 HIGH CHARACTERS (FAR SIDE) 117C3911G
APPLICABLE GROUP NO.
P8GYP7
2 PLACES
.12±.01 DIA. SEE TABLE FOR HOLE PATTERN
.12±.01 DIA. 3 HOLES AT POSITIONS SHOWN
2.50
BASKET CODE NO. CODING REFERENCE HOLES (SEE NOTE 1)
REACTOR CODE NO. CODING REFERENCE HOLES (SEE NOTE 2)
GROUP
HOLE PATTERN
FIG.
NOTES:
1. THE HOLE PATTERN IS A BINARY NUMBER MADE BY ADDING THE VALUES OF THE HOLES AS INDICATED BELOW:
CODING REFERENCE HOLES
2. THE HOLE PATTERN IS A BINARY NUMBER MADE BY ADDING THE VALUES OF THE HOLES AS INDICATED BELOW:
CODING REFERENCE HOLES
3. ARRANGEMENT OF CAPSULES P4 TO P13 WITHIN CONTAINER IS OPTIONAL
4. INSTALL HANDLE, P3, FIRMLY AGAINST STACK OF CAPSULES & SPACERS TO TAKE UP ALL END PLAY OF PARTS, AND WELD.
FIG. 1
FIG. 3
OTHERWISE SAME AS FIG. 1
OMIT SHIMS, 14
FIG 2
OTHERWISE SAME AS FIG 1
NE21856
APED
SAN JOSE
117C3911
CHK'D BY


GENERAL ELECTRIC
117C3912
TITLE ASM
TENSILE SPECIMEN CAPSULE
FIRST MADE FOR SURVEILLANCE PROGRAM
PL ISSUED
145A5481
FCF-G1-G5-117C3911
G6-G10-117C4370
METAL STAMP THE APPROPRIATE GROUP NO. IN .188 ±.01 HIGH CHARACTER.
P50YP102
SEE NOTE 1
5,6,7
8,9,10
ELECTRO-ETCH
117C3912G_
APPLICABLE GROUP NO.
NOTES:
1. EXHAUST AIR TO REDUCE CAPSULE INTERNAL PRESSURE TO ONE (1) TORR. BACKFILL WITH WELDING GRADE HELIUM AT ATMOSPHERIC PRESSURE AND SEAL WELD HOLE.
2. MASS SPECTROMETER LEAK TEST COMPLETED ASSEMBLY. ALLOWABLE LEAKAGE 10^{-6} CC/MIN.
G1
G2
G3
G4
G5
G6
G7
G8
G9
G10
NE-22233
APED
SAN JOSE
CHK'D BY: 10-5-68

GENERAL ELECTRIC

117C3913

UNLESS OTHERWISE SPECIFIED USE THE FOLLOWING —

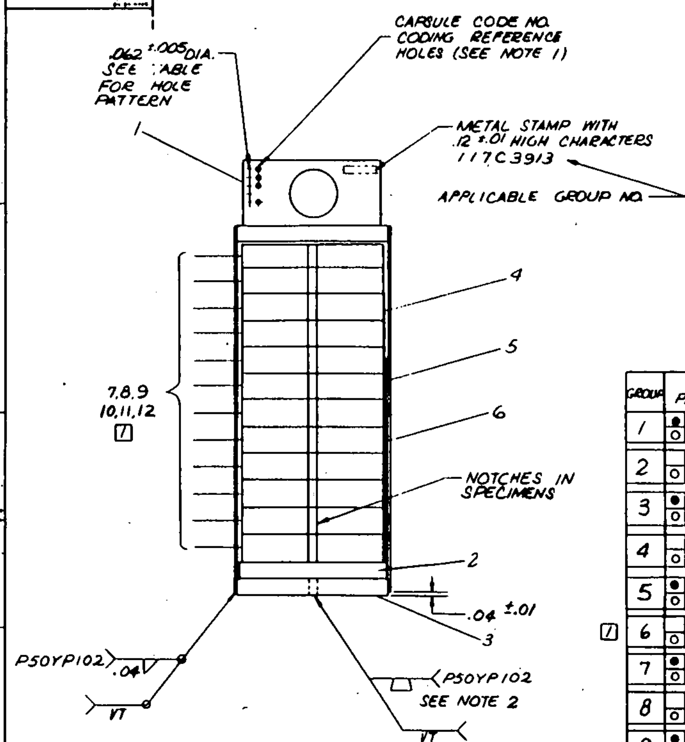
APPLIED PRACTICES	SURFACES
145A5481	√

TITLE: IMPACT SPECIMEN CAPSULE

FIRST MADE FOR: SURVEILLANCE PROGRAM

PL ISSUED

FCF: G1-G5-117C3911
G6-G10-117C4370



	GROUP	HOLE PATTERN				
	1	●				
		○		○	○	○
	2		●			
		○		○	○	○
	3	●	●			
		○		○	○	○
	4			●		
		○		○	○	○
	5	●		●		
		○		○	○	○
1	6		●	●		
		○		○	○	○
	7	●	●	●		
		○		○	○	○
	8				●	
		○		○	○	○
	9	●			●	
		○		○	○	○
1	10		●		●	
		○		○	○	○

NOTES:

1. THE HOLE PATTERN IS A BINARY NUMBER MADE BY ADDING THE VALUES OF THE HOLES AS INDICATED BELOW: 1 2 4 8 16 — CODING REFERENCE HOLES
2. EXHAUST AIR TO REDUCE CAPSULE INTERNAL PRESSURE TO ONE (1) TORR, BACKFILL WITH WELDING GRADE HELIUM AT ATMOSPHERIC PRESSURE AND SEAL WELD HOLE.
3. MASS SPECTROMETER LEAK TEST COMPLETED ASSEMBLY. ALLOWABLE LEAKAGE 1×10^{-6} CC/MIN.

REVISIONS: 1 — NE-22234 — M.V.T.

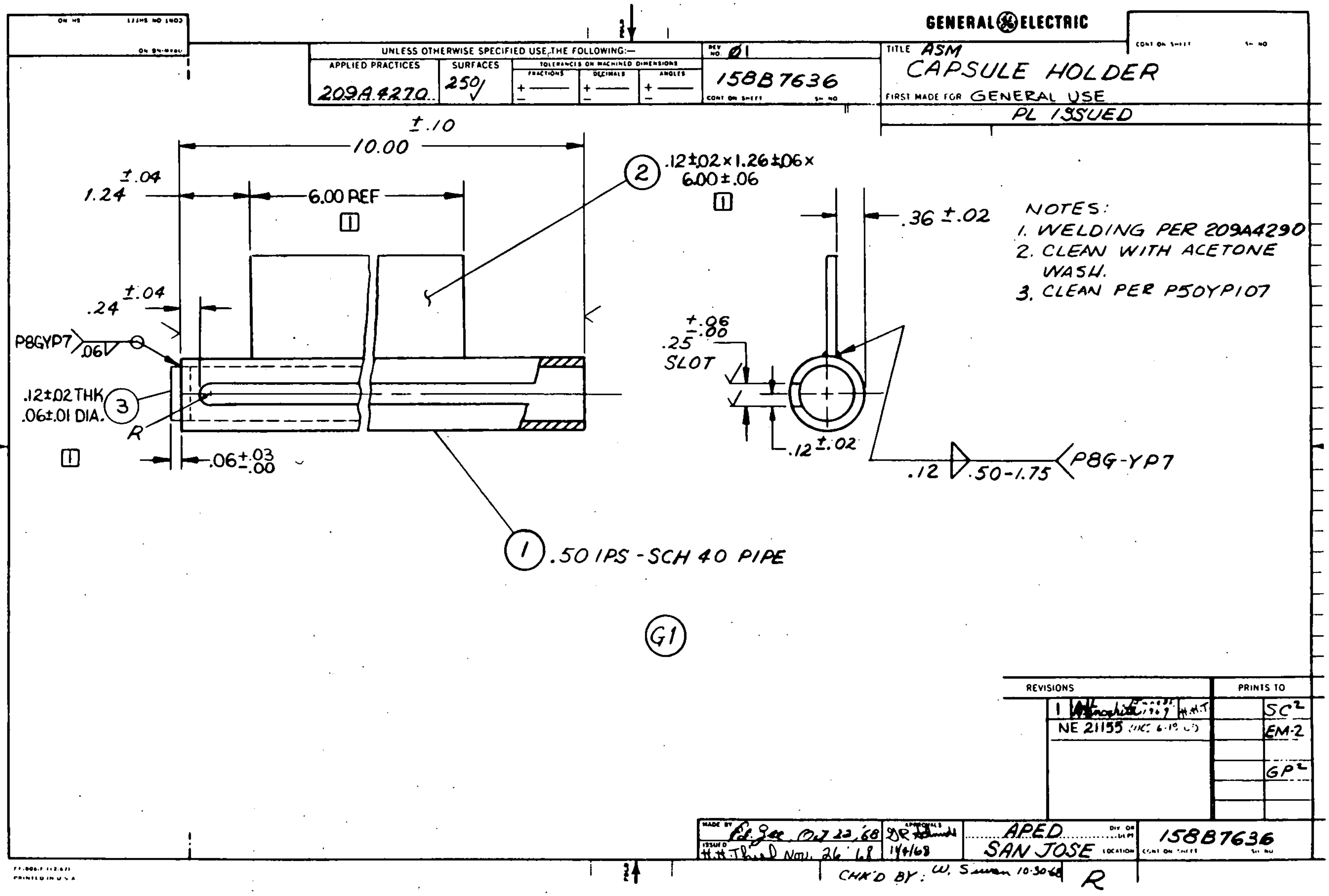
PRINTS TO: EM & SC

CHK'D BY: W. Swan 10-5-68

APED SAN JOSE

117C3913

R


GENERAL ELECTRIC
UNLESS OTHERWISE SPECIFIED USE THE FOLLOWING:—
APPLIED PRACTICES 209A4270
SURFACES 250√
TOLERANCES ON MACHINED DIMENSIONS FRACTIONS DECIMALS ANGLES
REV NO. Ø1
158B7636
TITLE ASM CAPSULE HOLDER
FIRST MADE FOR GENERAL USE
PL ISSUED
±.10
10.00
±.04
1.24
6.00 REF
2
.12±.02×1.26±.06×
6.00±.06
.36 ±.02
NOTES:
1. WELDING PER 209A4290
2. CLEAN WITH ACETONE WASH.
3. CLEAN PER P50YP107
±.04
.24
P8GYP7
.06
+.06
-.00
.25
SLOT
.12±.02 THK.
.06±.01 DIA.
3
R
.06 +.03 -.00
.12 ±.02
.12
.50-1.75
P8G-YP7
1
.50 IPS - SCH 40 PIPE
G1
REVISIONS
PRINTS TO
NE 21155
SC²
EM-2
GP²
MADE BY
ISSUED
Nov. 26 '68
11/4/68
APED
SAN JOSE
158B7636
CHK'D BY:
R

GENERAL ELECTRIC

UNLESS OTHERWISE SPECIFIED USE THE FOLLOWING:—

APPLIED PRACTICES	SURFACES	FRACTIONS	DECIMALS	ANGLES
209A4270	√	±	±	±

REV NO. 0

158B7635

TITLE ASM NEUTRON DOSIMETER

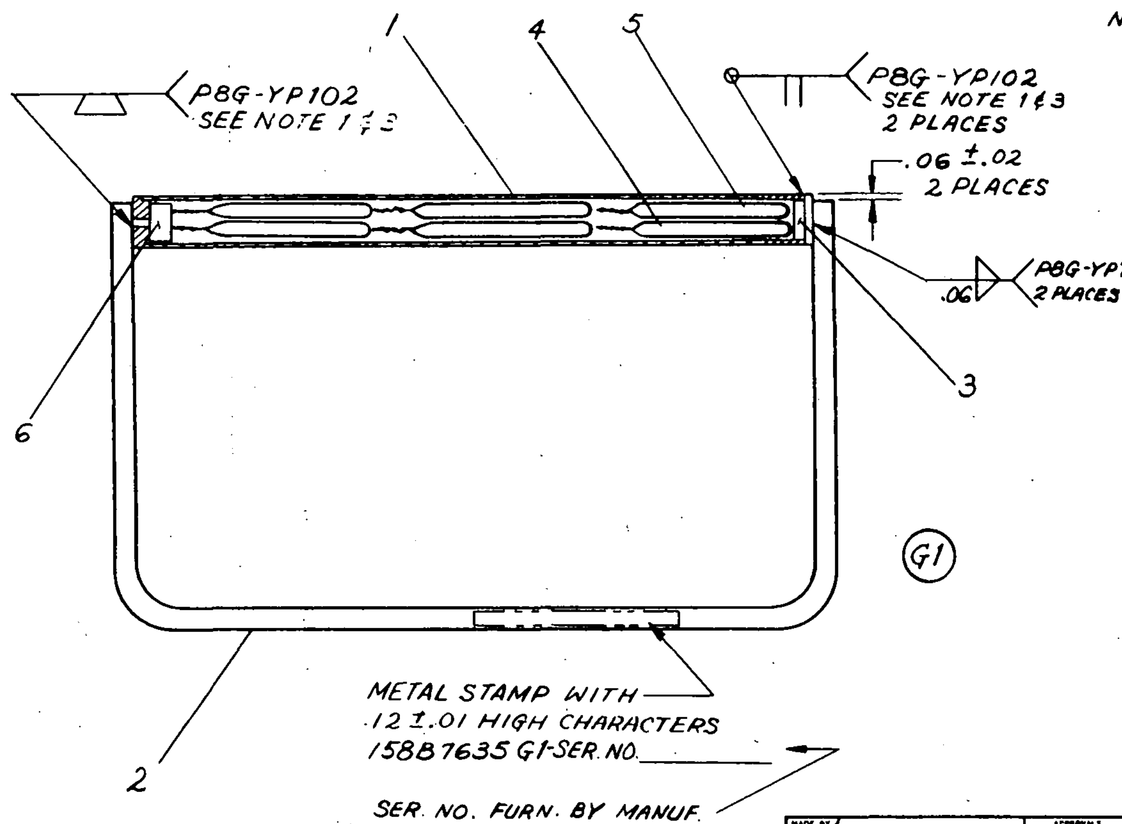
FIRST MADE FOR SERVEILLANCE PROGRAM

PL ISSUED

FCF: G1-886D345

NOTES:

1. BEFORE ASSEMBLING P2 EXHAUST AIR TO REDUCE CAPSULE INTERNAL PRESSURE TO ONE (1) TORR. BACKFILL WITH WELDING GRADE HELIUM AT ATMOSPHERIC PRESSURE AND SEAL WELD HOLE.
2. MASS SPECTROMETER LEAK TEST COMPLETED ASSEMBLY ALLOWABLE LEAKAGE 10^{-6} CC/MIN.
3. WELDING PER 209A4290



REVISIONS	PRINTS TO
	RV2
	EM-2
	PR-1
	GP2

MADE BY Bl Zec Oct 22, 1968 — APPROVALS D.R. [illegible] 1/4/68 — APED SAN JOSE — 158B7635

ISSUED H.H. Flint Nov. 26 '68

CHKD BY: W. Simon 10·30·68 R

APPENDIX B

PREPARATION OF BASE METAL SAMPLE MATERIAL

B.1 STEEL PRODUCTION

The pressure vessel steel normally is melted in an electric furnace and aluminum-treated in the ladle. Each ingot is hot-rolled into one large plate. The plates are air cooled after being rolled, and are then shipped to the fabricator for additional heat treatment and fabrication.

B.2 VESSEL FABRICATION HISTORY

Most BWR pressure vessels are constructed of SA-533 Grade B, Class I steel plate, 6 to 8 inches thick. The vessel shell plates are furnished to the vessel fabricator in the hot-rolled condition. The plates are austenitized, followed by liquid quenching, tempering, and furnace cooling. They are cold-formed into half shells. Stub ends of the plates are removed after this operation and are set aside for the surveillance material. The half shells are welded together to form the vessel shell rings. Several rings are welded together to form the vessel. Postweld heat treatments are performed after each circumferential weld is completed.

B.3 SPECIMEN PREPARATION

Mechanical test specimens are prepared from the vessel plate in accordance with the GE specification for the pressure vessel surveillance program.

APPENDIX C

THREE-DIGIT, TWENTY-SYMBOL MARKING SYSTEM (FAB CODE)

Specimens for a reactor pressure vessel materials surveillance program are being installed in each GE power reactor. Several types of steels and several thousand specimens are to be handled. All the specimens look alike, and a positive means for identification is required. The following items are monitored by the marking system:

1. Reactor

2. Type of Material
   - Interior Base
   - Exterior Base
   - Weld
   - Heat-Affected Zone (HAZ)

3. Location (within limits) on Stock Plate

4. Individual Serial Numbers

These specimens must be handled remotely after irradiation, and the largest possible marking is required. After considering all the factors, a modification of a system developed by the U.S. Steel Corporation Applied Research Laboratory was adopted for surveillance specimen marking. This system (FAB Code) uses three digits and twenty symbols, and can accommodate 8000 specimens. The sequence is illustrated in Table C-1. Note that "look alike" symbols are avoided as much as possible.

Table C-1

FAB CODE - DECEMBER 1963

Arabic Number Series	FAB Code Series	Three-Digit FAB Code Series																			
1	1	*11	*21	*31	*41	*51	*61	*71	*A1	*B1	*C1	*D1	*E1	*J1	*K1	*L1	*M1	*P1	*T1	*U1	*Y1
2	2	*12	*22	.	.	.	.	.	*A2	.	*C2	.	.	.	.	.	*M2	.	.	.	*Y2
3	3	*13	*23	*33	.	*53	.	.	*A3	.	.	.	.	.	.	*L3	.	*P3	.	.	*Y3
4	4	*14	*24	.	*44		.	*74	*A4	.	.	*D4	.	.	*K4	.	.	.	*T4	.	*Y4
5	5	*15	*25	*35	.	.	.	.	*A5	.	.	.	.	*J5	.	.	.	.	.	*U5	*Y5
6	6	*16	*26	.	.	.	*66	.	*A6	.	.	.	*E6	.	.	.	.	.	.	.	*Y6
7	7	*17	*27	*37	.	.	.	.	*A7	.	.	.	.	.	.	.	.	.	.	.	*Y7
8	A	*1A	*2A	.	*4A	.	.	*7A	*AA	.	.	.	.	.	.	.	*MA	.	.	.	*YA
9	B	*1B	*2B	*3B	.	*5B	.	*7B	*AB	.	*CB	.	.	.	.	*LB	.	*PB	.	.	*YB
10	C	*1C	*2C	.	.	.	.	.	*AC	.	.	.	.	.	*KC	.	.	.	*TC	.	*YC
11	D	*1D	*2D	*3D	.	.	.	.	*AD	*BD	.	.	.	*JD	.	.	.	.	.	*UD	*YD
12	E	*1E	*2E	.	.	.	.	.	*AE	.	.	.	*EE	.	.	.	.	.	.	.	*YE
13	J	*1J	*2J	*3J	*4J	.	.	.	*AJ	.	.	.	.	.	.	.	.	.	.	.	*YJ
14	K	*1K	*2K	.	.	.	.	.	*AK	.	.	.	.	.	.	.	.	.	.	.	*YK
15	L	*1L	*2L	*3L	.	*5L	.	.	*AL	.	.	.	.	.	.	.	.	.	.	.	*YL
16	M	*1M	*2M	.	*4M	.	.	.	*AM	*BM	.	.	.	.	.	.	.	.	.	.	*YM
17	P	*1P	*2P	*3P	.	.	*6P	.	*AP	.	.	.	.	.	.	.	.	.	.	.	*YP
18	T	*1T	*2T	*3T	.	.	.	.	*AT	.	.	*DT	.	.	.	.	.	.	.	.	*YT
19	U	*1U	*2U	*3U	.	.	.	*7U	*AU	.	.	.	.	.	.	.	.	.	.	.	*YU
20	Y	*1Y	*2Y	*3Y	*4Y	*5Y	*6Y	*7Y	*AY	*BY	*CY	*DY	*EY	*JY	*KY	*LY	*MY	*PY	*TY	*UY	*YY
Cumulative Total		20	40	60	80	100	120	140	160	180	200	220	240	260	280	300	320	340	360	380	400

NOTES:

1. *denotes first digit of marking code and is furnished in instructions. It will be a FAB Code letter or number such as 2, 4, 7, A, D, J or Y.
2. . denotes number in sequence left out to reduce labor in making table.
3. For ease of marking, it is suggested that a set of stencils be set up in the order shown in column headed "FAB Code Series."

APPENDIX D

DESCRIPTION OF FLUX MONITORS

D.1 NICKEL WIRE

Commercially-pure English nickel containing 0.0729 percent cobalt is used.

The nuclear reactions of interest are:

Ni^{58} (n,p) Co^{58}
Co^{59} (n,γ) Co^{60}

D.2 IRON WIRE

Commercially-pure iron thermocouple wire, 24-gage, is used.

The nuclear reaction of interest is: Fe^{54} (n,p) Mn^{54}

D.3 COPPER WIRE

Very high purity (99.999 percent) copper wire is used.

The nuclear reaction of interest is: Cu^{63} (n,α) Co^{60}

D.4 REFERENCE

Hogg, C. H., and Weber, L. D., *Fast Neutron Dosimetry at the MTR-ETR Site,* ASTM STP No. 341, 133 (1963).

General Analysis of Radio Isotopes, ASTM E181-62 (1968).

Fast-Neutron Flux By Radioactivation of Iron, Measuring, ASTM E263-65T.

Fast-Neutron Flux By Radioactivation of Nickel, Measuring, ASTM E264-65T.

APPENDIX E

CAPSULE HANDLING AND GENERAL SUGGESTIONS FOR CARRYING OUT THE PROGRAM

E.1 GENERAL INSTRUCTIONS

E.1.1 Binary Code

Capsules and baskets in the surveillance program are binary-coded. The use of the binary code is illustrated in GE drawing 129B3578, and in Appendix A of this report.

E.1.2 FAB Code

The FAB Code is described in Appendix C of this report.

E.2 CAPSULE REMOVAL FROM REACTOR

Every effort is made to design the capsules and locate them to assure reasonable ease of removal. The respective drawings of capsule and pressure vessel should be reviewed and a removal plan should be made prior to the operation.

E.3 SHIPPING

If individual capsules are removed from the baskets, they should be placed in a cask liner prior to inserting them in a shipping cask, since it is very difficult to retrieve small objects from the bottom of a cask at the hot laboratory. It may be necessary to get a license to ship byproduct materials prior to shipping the capsules. The receiver may need a license to receive such materials. The receiver should have a set of drawings of all items shipped.

E.4 CAPSULE OPERATIONS AT HOT LABORATORY

E.4.1 Gamma Scan

Each capsule should be gamma-scanned to obtain a measure of the neutron exposure variations within each capsule and among the several capsules.

E.4.2 Impact Specimen Capsule

This capsule has a 1/4-inch-thick stainless steel spacer at the bottom; if possible, the capsule should be parted in this area. The flux monitors are loose on the seam side of the can, and are identified by their length. The specimens should slide out. If they do not and the capsule requires slitting down one side, this should be done on the weld-seam side because the numbered ends of the specimens are opposite to this side.

E.4.3 Tensile Specimen Capsule

This capsule may be opened at either end by cutting the weld. Some skill may be required to get the specimen out because the fillets at the ends of the gage length tend to wedge the aluminum spacers. When the specimens stick as they are being loaded into the capsules, forcing techniques are not successful because forcing increases the wedging action. Back-and-forth, gentle rapping should break the specimens loose, and then they slide easily out or in. There are no flux monitors in these capsules.

E.5 EXAMINATION OF SPECIMENS PRIOR TO TESTING

The specimen numbers in each capsule and their orientation are described in the schedule drawing.

E.6 SUGGESTIONS FOR MECHANICAL TESTING (See ASTM Specification E185)[3]

The following data should be obtained by testing the Charpy specimens:

1. Energy Transition curve at temperatures which completely bracket the brittle ductile transition range, and which define the lower and upper shelf energies.
2. Fracture transition curve determined by examination of the fractured surfaces, using ASTM Specification A370 as a guide.
3. Photograph of each pair of fracture surfaces at a minimum 1X magnification.
4. Hardness data on representative specimens near the notch on the notch surface prior to impact test.

There are sufficient specimens to develop a transition curve if care is taken during testing and if curves previously developed are used as a guide. No duplicate tests should be made before the full extent of the curve is determined. The specimen notch location is referenced to the unmarked end. If an end reference stop is used to align the specimens in this machine, the unmarked end of the specimen must be used with it.

The following data should be obtained by testing the tensile specimens:

1. 0.2 percent yield strength.
2. Ultimate tensile strength.

3. Complete autographic load-elongation curve, preferably obtained with extensometer on specimen gage length.
4. Uniform elongation from curve.
5. Total elongation from curve.
6. Reduction of area from broken specimen.
7. Photograph of each pair of fractured surfaces at a minimum 1X magnification.
8. Photograph of the length of the broken specimen, with a precision ruler adjacent to the gage length to provide a judgment of the type of necking.

Testing parameters should also be reported, such as type and make of machine, loading rate, strain rate, and temperature. If possible, the tensile tests should be made at 550°F. Each tensile specimen should be checked for gage marks prior to testing because some specimens may lack them. Gage marks should be applied to specimens which have none, with a commercial gage marker.

E.7 CONTROL SPECIMENS

A group of specimens from each type of material is available to determine the initial condition of the material prior to temperature cycling or irradiation. These should be tested with the first group of irradiated specimens. The tests should be made on the same machine used for the irradiated specimens.

E.8 METALLOGRAPHY

If funds permit, metallographic specimens should be made of representative samples—especially at the longer time intervals.

E.9 DOSIMETRY (See Appendix D)

Each impact capsule contains an iron, a nickel, and a copper wire. It is recommended that these wires be analyzed by radiochemical means to obtain a measure of the integrated neutron flux exposure of the specimens. It should not be necessary to analyze all the flux wires of all capsules. Wires from selected capsules can be analyzed, and the remainder can be retained as spares, as required.

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TECHNICAL INFORMATION SERIES

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SUMMARY

The General Electric Company furnishes pressure vessel steel surveillance specimens for each BWR to permit the plant owner to account for radiation induced changes in mechanical properties of the reactor vessel during service. This report presents several methods for determining the magnitude of these property changes. It also describes the specimens, specimen inventory, capsule design, associated equipment, material selection, and instructions for handling the specimens if they are tested.

By cutting out this rectangle and folding on the center line, the above information can be fitted into a standard card file.

For list of contents – drawings, photos, etc. and for distribution see next page (FN–610–2)

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