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ABSTRACT

A methodology is described to evaluate the dosimetry and metallurgical data from the two-year ORR-PSF metallurgical irradiation experiment. The first step is to obtain a three-dimensional map of damage exposure parameter values based on neutron transport calculations and dosimetry measurements which are obtained by means of the LSL-M2 adjustment procedure. Metallurgical test data are then combined with damage parameter, temperature, and chemistry information to determine the correlation between radiation and steel embrittlement in reactor pressure vessels including estimates for the uncertainties. Statistical procedures for the evaluation of Charpy data, developed earlier, are used for this investigation. The data obtained in this investigation provide a benchmark against which the predictions of the "PSF Blind Test" can be compared. The results of this investigation and the Blind Test comparison will be discussed.

INTRODUCTION

The two-year Pressure Vessel Simulator (PVS) metallurgical irradiation experiment at the Oak Ridge Research Reactor (ORR) Poolside Facility (PSF) was performed in order to simulate, as closely as possible, the irradiation conditions in commercial reactor pressure vessels and surveillance capsules (Fig. 1). Of primary interest was the question whether results obtained from surveillance capsule evaluations in commercial power reactors can be extrapolated safely. Since there are considerable differences in fluence rate and fluence spectrum between the pressure vessel wall and the surveillance capsules, possible effects of these factors on the irradiation damage need to be investigated. The magnitude of these effects, if any, may also be different for different types of materials, e.g., plate material vs. welds or between materials of different chemical compositions.

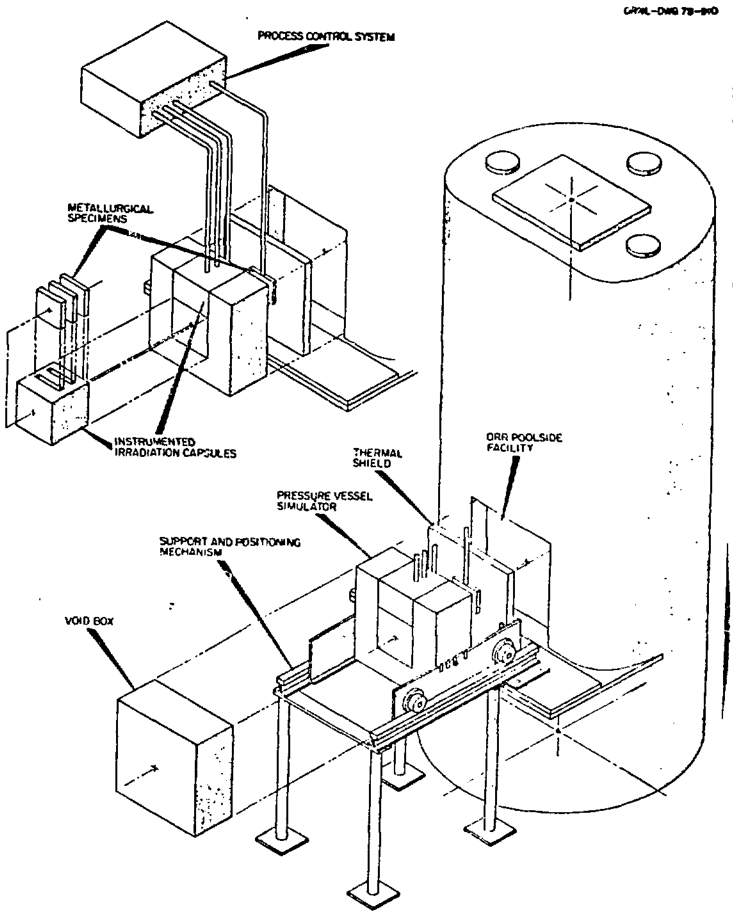

Fig. 1. View of the PSF-PVS Facility.

In order to answer these questions, five capsules were irradiated in the PSF-PVS experiment, each containing metallurgical specimens of the same mix of plate and weld materials (see Fig. 2). Two of the capsules received high-fluence rates, characteristic of surveillance capsules (SSC capsules), and the other three were irradiated to about the same total fluence but at lower fluence rates and over a longer time period (two years vs. one to two months).

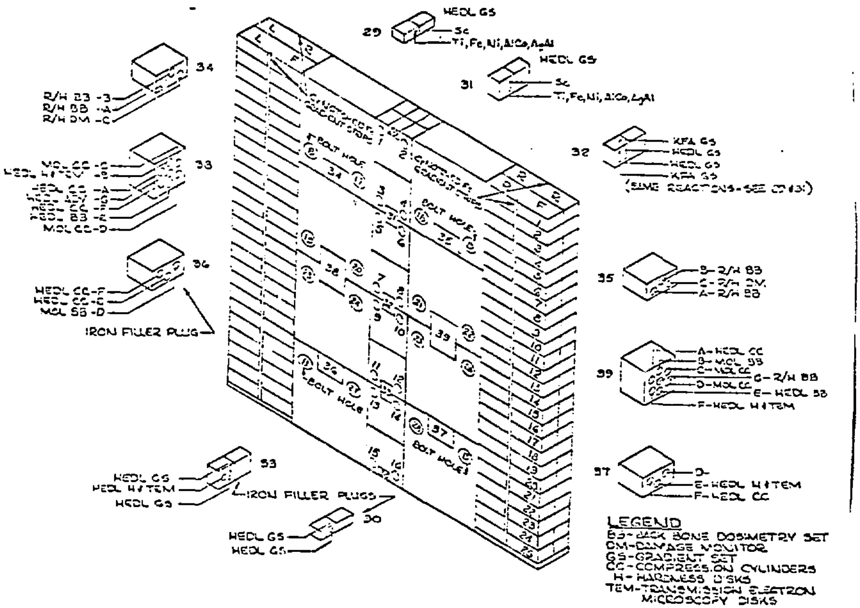

Fig. 2. Illustration of dosimeter and metallurgical specimen location in the irradiation capsules.

In order to serve its intended purpose, namely as a benchmark for testing damage prediction methodologies, both damage fluences and materials damage must be determined with very high accuracy. Reliable estimates for the uncertainties of all data must be provided to ascertain whether differences between predictions and benchmark results are significant or attributable to measuring errors. High accuracies and reliable uncertainties are particularly important in determining possible effects of fluence rate and spectra, since these effects are likely to be small. A careful statistical evaluation of the PSF-PVS experiment is therefore necessary, details of which are reported in the next section.

MATERIALS AND PROCEDURES

The following input data were used for this analysis:

<u>Neutron physics calculation</u>. A detailed description of the calculation is given in Refs. 1,2.

<u>Dosimetry</u>. All dosimetry data used in this analysis were radiometric measurements provided by HEDL (2). Additional dosimetry provided by other U.S. and European Laboratories were not considered at this time.

Metallurgical tests. The metallurgical test results from this experiment are published in Refs. 3,4.

The first step in the analysis was the determination of the damage parameter values, fluence > 1.0 MeV, fluence > 0.1 MeV, and dpa using the LSL-M2 adjustment method (5). The calculations (1) were used as input spectrum together with a one-dimensional ANISN calculation for energies between 0.1 MeV and 0.4 eV. The dosimeters used were $^{63}Cu(n,\alpha)$, $^{45}Ti(n,p)$, $^{54}Fe(n,p)$, $^{58}Ni(n,p)$, $^{237}Np(n,f)$, $^{235}U(n,f)$, $^{59}Co(n,\gamma)$, $^{58}Fe(n,\gamma)$, and $^{45}Sc(n,\gamma)$. The $^{238}U(n,f)$ reaction could not be used because of large perturbations due to Pu burn-in at high fluence. The damage parameter values - including thermal fluence and total fluence - at the centers of the five irradiation capsules with uncertainties are given in Table 1. Values outside the capsule centers can be calculated from a cosine-exponential formula, which was obtained from least squares fits of the ajusted values, namely

$$P(X,Y,Z) = P_0 \cos B_X(X-X_0) \cos B_Z(Z-Z_0) e^{-\lambda(Y-Y_0)} \qquad (1)$$

P(X,Y,Z) is the given damage parameter for the X-Y-Z-coordinate system in Fig. 3. The coefficients of formula (1) are given in Table 2. More details can be found in Ref. 6. .

Six different plate and weld materials were irradiated in each of the five metallurgical capsules. Type and chemical composition for each material is listed in Table 3. The raw Charpy data from Ref. 4 were processed with the CV81 Charpy curve fitting procedure (7) to obtain the values for ΔNDT and upper shelf drop with uncertainties for each irradiation capsule and material. The results are listed in Table 4. They agree within uncertainties with the values in Ref.4 (see Table 5). Details are given in Ref. 7.

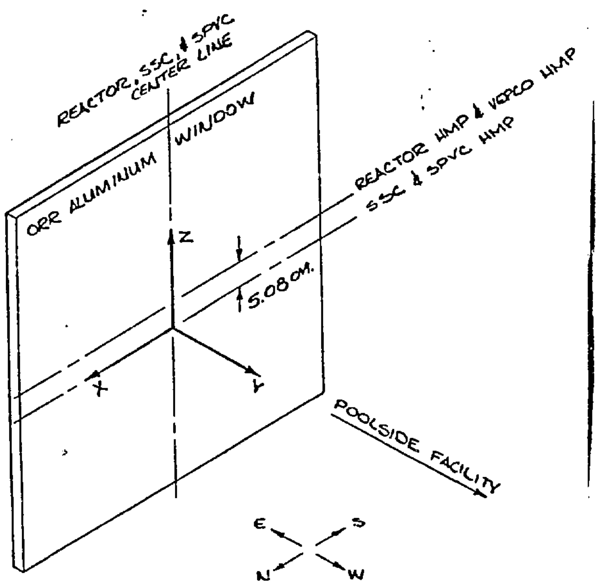

Fig. 3. Coordinate system for the ORR-PSF metallurgical experiment.

Table 1. Fluences and dpa at capsule centers

		ϕ>1.0 MeV	Std. dev. (%)	ϕ>0.1 MeV	Std. dev. (%)	ϕ<0.4 eV	Std. dev. (%)	ϕ_{total}	Std. dev. (%)	dpa	Std. dev. (%)
SSC1	H4	2.56*	5.1	7.74	5.8	1.26	7.4	14.20	5.8	4.07	4.9
SSC2	H9	5.50	5.1	16.84	5.8	2.79	7.4	30.55	5.5	8.80	4.9
0-T	H14	4.10	5.1	12.26	5.8	6.29	7.6	27.66	5.8	6.56	4.9
1/4T	H19	2.21	5.2	8.98	6.0	0.84	7.9	14.75	5.5	4.13	5.2
1/2T	H24	1.05	5.4	5.83	6.0	0.27	8.3	9.17	5.6	2.39	5.4

*Read values for ϕ>1.0 MeV, ϕ>0.1 MeV, ϕ<0.4 eV, and ϕ_{total} as 2.56×10^{19} neutrons/cm^2, etc.

Table 2. Fitting parameters for formula (1)

	P_0*	B_X (cm^{-1})	X_0 (cm)	B_Z (cm^{-1})	Z_0 (cm)	λ (cm^{-1})	Y_0 (cm)
SSC1							
ϕt>1.0 MeV	2.500E+19	0.0499	[illegible]	0.0436	0.97	0.176	13.29
ϕt>0.1 MeV	7.607E+19	0.0507	0.37	0.0464	0.80	0.134	13.29
dpa	3.995E-02	0.0502	0.38	0.0449	0.90	0.156	13.29
SSC2							
ϕt>1.0 MeV	5.341E+19	0.0528	-0.95	0.0457	0.03	0.176	13.29
ϕt>0.1 MeV	1.648E+20	0.0539	-0.88	0.0484	-0.02	0.134	13.29
dpa	8.580E-02	0.0533	-0.91	0.0470	0.02	0.156	13.29
0-T							
ϕt>1.0 MeV	3.924E+19	0.0517	-0.69	0.0395	0.72	0.107	24.05
ϕt>0.1 MeV	1.214E+20	0.0522	-0.64	0.0432	0.71	0.042	24.05
dpa	6.452E-02	0.0516	-0.67	0.0414	0.71	0.079	24.05
1/4T							
ϕt>1.0 MeV	2.143E+19	0.0478	-0.96	0.0378	1.30	0.134	28.56
ϕt>0.1 MeV	8.823E+19	0.0486	-0.86	0.0425	1.14	0.070	28.56
dpa	4.037E-02	0.0481	-0.91	0.0407	1.21	0.097	28.56
1/2T							
ϕt>1.0 MeV	1.016E+19	0.0441	-0.94	0.0349	1.94	0.146	33.70
ϕt>0.1 MeV	5.727E+19	0.0452	-0.79	0.0413	1.48	0.089	33.70
dpa	2.333E-02	0.0450	-0.83	0.0395	1.59	0.107	33.70

*Values for ϕt > 1.0 MeV and ϕt > 0.1 MeV are in neutrons/cm^2.

Table 3. List of materials and chemical compositions (wt-%)

Material	Heat code	Supplier	P	Ni	Cu
A302-B (ASTM reference plate)	F23	NRL	0.011	0.18	0.20
A533-B (HSST plate 03)	3PS, 3PT, 3PU	NRL	0.011	0.56	0.12
22NiMoCr37 forging	K	KFA	0.009	0.96	0.12
A508-3 forging	MO	MOL	0.008	0.75	0.05
Submerged arc weld (single vee type, A533-B base plate)	EC	EPRI	0.007	0.64	0.24
Submerged arc weld (single vee type A533-B base plate)	R	Rolls-Royce & Assoc. Ltd.	0.009	1.58	0.23

Table 4. 41J Charpy shift vs. fluence > 1.0 MeV for different materials in the PSF experiment

	SSC1	SSC2	0-T	1/4T	1/2T
A302-B Plate					
$\phi t > 1.0$ MeV (10^{19} n/cm^2)	2.59	5.41	3.73	2.15	1.05
ΔNDT- 41J (°C)	78	94	77	65	52
A533-B Plate					
$\phi t > 1.0$ MeV (10^{19} n/cm^2)	2.35	4.97	3.47	1.99	0.98
ΔNDT- 41J (°C)	71	84	71	69	52
22NiMoCr37 forging					
$\phi t > 1.0$ MeV (10^{19} n/cm^2)	1.64	3.44	2.80	1.51	0.73
ΔNDT- 41J (°C)	52	109	81	66	66
A508-3 forging					
$\phi t > 1.0$ MeV (10^{19} n/cm^2)	1.79	3.72	2.97	1.61	0.77
ΔNDT- 41J (°C)	15	39	27	23	22
Submerged arc weld (EC)					
$\phi t > 1.0$ MeV (10^{19} n/cm^2)	1.73	3.57	2.90	1.59	0.76
ΔNDT- 41J (°C)	112	123	125	96	94
Submerged arc weld (R)					
$\phi t > 1.0$ MeV (10^{19} n/cm^2)	2.47	5.06	3.59	2.08	1.03
ΔNDT- 41J (°C)	230	309	294	270	242

METALLURGICAL BLIND TEST

The metallurgical "Blind Test" (8) was initiated in order to test current damage prediction methodologies for pressure vessel surveillance against the established test data of the PFS-PVS benchmark experiment. The Blind Test participants were supplied with dosimetry and metallurgical test data from the SSC1 capsule, a detailed description of the experimental configuration, and a complete irradiation-temperature-time history. The participants were asked to determine the damage fluence at locations of the metallurgical specimen and to predict from it the amount of radiation damage in the specimen. A first summary of the Blind Test results is published in Ref. 9. Results were submitted by eight participants. The predicted values of fluence > 1.0 MeV were compared with the ORNL evaluation. All participants came to within ±30% of the ORNL data and more than 60% of the predictions were within ±10%. Neither over nor under prediction of fluences appears to be correlated to over or under prediction of damage by the same laboratory. Table 5 presents a summary of the Blind Test prediction of Charpy shifts compared to results of this evaluation. The predictions are fairly symmetrically scattered around the experimentally based evaluation with bounds which are roughly twice the variance of the statistical evaluation. Substantial and consistent under prediction of damage can be found only in the code R weld material, which appears to be more radiation-sensitive than other materials of similar chemical composition. Thus, the Blind Test results show neither consistent biases or other glaring defects of current damage prediction methodologies. Adequate safety margins must still be provided to avoid non-conservative predictions. The following suggestions for improvement are offered:

- Procedures for determining and reporting uncertainties should be included in the damage prediction methodology. Many Blind Test participants failed to report uncertainties altogether or did so only indirectly (e.g., by referring to contributing uncertainties). Uncertainties are important for meaningful benchmark comparisons and for the establishment of safety margins.

- Predictions based on fluence and chemical composition alone are not always reliable. Any new material should be tested in a variety of fluences to establish material-dependent trend curves with uncertainties which can be used for the prediction methodology.

CONCLUSION

Comparison of current damage prediction methodology with results from the PSF-PVS experiment shows no substantial biases or deficiencies, although improvements are desirable, particularly in

Table 5. Comparison between experimentally determined Charpy shift and Blind Test predictions

	Determined from Charpy curves			Smallest and largest values predicted by Blind Test participants		Difference Blind Test – CV81	
	CV81* (°C)	Std. (°C)	MEA** (°C)	Min. (°C)	Max. (°C)	Min. (°C)	Max. (°C)
A302-B							
SSC1	78	±12	82	71	98	−7	+20
SSC2	94	±11	94	75	112	−19	+18
0-T	77	±10	81	71	96	−6	+19
1/4T	65	±18	67	65	81	0	+16
1/2T	52	±10	50	45	66	−7	+14
A533-B							
SSC1	71	±11	61	45	69	−24	−2
SSC2	84	±10	81	62	99	−22	+15
0-T	71	±13	75	60	87	−11	+16
1/4T	69	± 9	69	54	63	−15	−6
1/2T	52	±10	53	26	52	−26	0
22NiMoCr37							
SSC1	52	±16	61	57	77	−5	+25
SSC2	109	±14	94	65	110	−44	+1
0-T	81	±16	72	63	97	−18	+16
1/4T	66	±18	78	52	76	−14	+10
1/2T	66	±13	56	45	64	−21	−2
A508-3							
SSC1	15	± 7	20	6	43	−9	+18
SSC2	39	± 7	39	11	53	−28	+14
0-T	27	± 7	25	10	49	−17	+22
1/4T	23	± 6	20	8	42	−15	+19
1/2T	22	± 7	14	6	35	−16	+13
Submerged arc weld (EC)							
SSC1	112	±33	108	99	118	−13	+6
SSC2	123	±60	119	130	153	−7	+30
0-T	125	±50	124	121	135	−4	+10
1/4T	96	±18	94	91	115	−5	+19
1/2T	94	±20	89	63	103	−31	+9
Submerged arc weld (R)							
SSC1	230	±12	222	218	227	−12	−3
SSC2	309	±38	289	246	319	−63	+10
0-T	294	±15	286	239	288	−55	−6
1/4T	270	±25	256	180	218	−90	−52
1/2T	242	±44	239	143	189	−99	−53

*ORNL evaluation.
**Evaluation in Ref. 4.

the field of uncertainty analysis. Space does not permit a more thorough discussion of the experimental results, such as effects of fluence rate and spectrum. A detailed discussion is given in Ref. 7.

REFERENCES

1. R.E. Maerker and B.A Worley. "Calculated Spectral Fluences and Dosimeter Activities for the Metallurgical Blind Test Irradiations at the ORR-PSF," to be presented at the 5th ASTM-EURATOM Symposium on Reactor Dosimetry, Geesthacht, FRG, September 24-28, 1984.

2. W.N. McElroy and F.B.K. Kam, eds., PSF Blind Test SSC, SPVC Physics, Dosimetry, and Metallurgy Data packages, distributed to Blind Test participants, February 17, 1984.

3. J.R. Hawthorne, B.H. Menke, and A.L. Hiser, Light Water Reactor Pressure Vessel Surveillance Dosimetry Improvement Program: Notch Ductility and Fracture Toughness Degradation of A 302-B and A 533-B Reference Plates from PSF Simulated Surveillance and Through-Wall Irradiation Capsules, NUREG/CR-3295, MEA-2017, Vol. 1, U.S. Nuclear Regulatory Commission, Washington, DC, 1983.

4. J.R. Hathorne and B.H. Menke, Light Water Reactor Pressure Vessel Surveillance Dosimetry Improvement Program: Notch Ductility and Tensile Strength Determinations for PSF Simulated Surveillance and Through-Wall Irradiation Capsules, NUREG/CR-3295, MEA-2017, Vol. 2, U.S. Nuclear Regulatory Commission, Washington, DC, 1983.

5. F.W. Stallmann, "LSL-M1 and LSL-M2: Two Extensions of the LSL Adjustment Procedure for Including Multiple Spectrum Locations," to be presented at the 5th ASTM-EURATOM Symposium on Reactor Dosimetry, Geesthacht, FRG, September 24-28, 1984.

6. F.W. Stallmann, Determination of Damage Exposure Parameter Values in the PSF Metallurgical Irradiation Experiment, NUREG/CR-3814, ORNL/TM-9166, U.S. Nuclear Regulatory Commission, Washington, DC, 1984.

7. F.W. Stallmann, Statistical Evaluation of the Metallurgical Test Data in the ORR-PSF-PVS Irradiation Experiment, NUREG/CR-3815, ORNL/TM-9207, U.S. Nuclear Regulatory Commission, Washington, DC, 1984.

8. W.N. McElroy and F.B.K. Kam, eds., PSF Blind Test Instructions and Data Packages, distributed to Blind Test participants, March 22, 1983.

9. G.L. Guthrie, E.P. Lippincott, and E.D. McGarry, Light Water Reactor Pressure Vessel Surveillance Dosimetry Improvement Program: PSF Blind Test Workshop Minutes, HEDL, April 9-10, 1984.