CAROLINA POWER & LIGHT COMPANY

NUCLEAR ENGINEERING DEPARTMENT

SUMMARY REPORT
SR-BNP1-1005-001

BRUNSWICK UNIT 1 RPV SURVEILLANCE PROGRAM

FIRST CAPSULE (300°) REMOVAL AFTER 8 FUEL CYCLES

TEST RESULTS
AND
PROJECTIONS

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CONTENTS

KEY WORDS

Fast Neutron Flux and Fluence

Radiation Surveillance Program

Fracture Toughness

SUMMARY AND CONCLUSION

The test results from the first reactor vessel surveillance capsule to be removed from Brunswick Unit 1 (300° capsule) demonstrate that the current heatup - cooldown curves are conservative; therefore, revision is not required.

INTRODUCTION

The Brunswick (BSEP) 1 Surveillance Program for the reactor pressure vessel materials was designed for the requirements of ASTM E 185-66, except that the number of unirradiated tensile specimens for plate weld and heat-affected-zone was 2 for each rather than the specified 3 and the number of unirradiated Charpies was 3 rather than the specified 15.

Test methods and reporting for the 300 degree capsule, the first to be removed in the program, comply with the ASTM E 185-82 edition.

The results of mechanical properties, chemical analysis, and radiochemical testing of the capsule's contents are provided in Supplement 1 of this report. The tests were performed by General Electric at the Vallecitos Nuclear Center.

Westinghouse Electric used the flux wire activity measurements from Supplement 1 and power and void distributions as varied over time, as supplied by CP&L, for input to neutron transport calculations which produced the fast neutron exposure rates at the capsule and at key maximum locations on the vessel in the core region. The Westinghouse work is included as Supplement 2.

Most of the surveillance program description and core region materials characterization are given in Supplement 3, "Information on Reactor Vessel Surveillance Program." Other information required by ASTM E 185-82 is provided in Supplement 1 or 2. Salient data for core region materials have been collected in Table 1 for ready reference.

TABLE 1
BRUNSWICK 1
EOL: 9/8/2016

BELTLINE IDENT.	HEAT NO. PLATE NO.	%CU	%NI	CHEM. FACTOR (TABLE)	EOL ID FLUENCE (27 EFPY)	EOL 1/4T FLUENCE (27 EFPY)	INITIAL RT_{NDT} (Note 1) (°F)	ΔRT_{NDT} 1/4T
Lower Shell	C4535-2 201	0.12	0.58	82.6	9.4 E17	6.9 E17	10	29
Lower Shell	C4550-1 251	0.11	0.60	74	9.4 E17	6.9 E17	10	26
Lower Int. Shell	C4487-1 301	0.12	0.56	82.2	1.2 E18	8.6 E17	10 (Note 3)	32
Lower Int. Shell	B8496-1 351	0.19	0.58	139.8	1.2 E18	8.6 E17	10	54
Nozzle N16A	Q2Q1VW	0.16	0.82	123	4.6 E17	3.4 E17	0 (Note 4)	29
Nozzle N16B	Q2Q1VW	0.16	0.82	123	4.6 E17	3.4 E17	0 (Note 4)	29
Axial Welds G1 & G2 (Note 2)	S3986	0.05	0.96	68	6.5 E17	4.7 E17	-56 (Note 5)	19
Axial Welds F1 & F2 (Note 2)	S3986	0.05	0.96	68	8.2 E17	5.9 E17	-56 (Note 5)	22
Circ. Welds	1P4218	0.06	0.87	82	9.4 E17	6.9 E17	-56 (Note 5)	28

NOTES:

1. Values based on NDTT, unless otherwise specified.
2. Represented by surveillance weld.
3. CP&L testing of archive plate used to establish RT_{NDT} based on Subsection NB-2331 of ASME Code Section III.
4. Based on additional Lenape Forge drop weight test data for Heat Q2Q1VW (not included in Section 3.6 of NEDO-24161, Revision 1).
5. Generic value.

COMPARISON OF MEASUREMENTS TO PRIOR REPORTS AND PROJECTIONS

Fast Neutron Flux and Fluence

The capsule fast neutron flux was determined to be 1.18×10^9 n/cm^2 - sec. > 1 MeV (see Table 2-1 of Supplement 2). It had been measured at 1.4×10^9 after the first fuel cycle using dosimetry extractable from the capsule in situ. This higher initial flux was to be expected due to the low leakage feature of utilized core loading patterns. At 8.67 EFPY, the fluence for the capsule is 3.2×10^{17} n/cm^2 > 1 MeV.

Whereas the higher flux (1.4×10^9) was used in the development of current heatup - cooldown curves, those curves remain conservative. A small additional conservatism was found in performance of Supplement 2 calculations: the lead factor from the capsule to the maximum 1/4T position was found to be 1.04 compared to the previously assumed lead factor of 1.0. The lead factor from the capsule to the maximum ID azimuth of 45° is 0.80.

Chemistry

Check chemical analyses were performed for the surveillance plate and the surveillance weld (see Table 6-1 of Supplement 1). The heat number and elemental chemistry of the surveillance weld had previously been unknown; however, General Electric recently was able to identify the weld heat as Adcom S3986 and the flux as Linde 124, Lot 3876, Run No. 934. The chemistry and mechanical properties of the reactor vessel beltline fabricated materials are given in Section 3.0 of Supplement 3.

Tensile Tests

Room temperature test results for the surveillance plate and weld, unirradiated (see Section 3.5 of Supplement 3) and irradiated, as taken from Table 5-1 in Supplement 1, were:

	Yield Strength ksi	Ultimate Strength ksi	Total Elongation %
Unirradiated Plate	65	89	25
Irradiated Plate	69	91	20.4
Unirradiated Weld	71.8	86.5	30
Irradiated Weld	76	92	21.6

As demonstrated by the test results, the weld (vs. base metal) has the greater radiation embrittlement despite a lower copper content (.055 versus .099%). Higher phosphorus (.017 versus .009%) or higher nickel (.98 versus .53%) in the weld may account for the difference.

Charpy V Tests

As demonstrated below, the adjusted reference temperatures (ART), as given in Reference 1, are conservative compared to that obtained with the results of this capsule test at 8.7 EFPY.

	Measured ART 300° Capsule 8.67 EFPY (Note 1)	Calculated ART 8.67 EFPY (Note 2)
Plate 301	30.6	63
Weld	-4.4	16
Plate 351 (Note3)		76

NOTES:

1. Reference Table 4-3 of Supplement 1.
2. Per Reference 2.
3. Controlling Material (material specimens not included in surveillance capsule).

The shelf energy after 8.7 EFPY is found in Table 4-3 of Supplement 1 for both surveillance plate and weld to be more than adequate. As converted to the transverse direction, the shelf energy for the plate is 96.6 ft. lb. The shelf energy for the weld is 97.5 ft. lb.

EFFECTS OF LOW TEMPERATURE OPERATION

The changes in mechanical properties with irradiation exposure to the first 8.7 EFPY are sufficiently small to indicate that operation below 525°F during that period is not a reason for concern. Future operation will entail much less low temperature operation, and the issue should be closed with respect to Reference 3.

REMAINING SURVEILLANCE PROGRAM

Brunswick 1 has two capsules remaining in place in the reactor at 30° and 120° azimuths. Removal dates have not yet been selected, because it is desirable to establish an integrated surveillance program for Units 1 and 2 in the future. The first Brunswick 2 capsule is scheduled for removal in 1996. It is important to have the test results from that capsule in order to set the parameters for an integrated program. At that time, the remaining surveillance schedule can be established.

REFERENCES

1. Amendment 172 to Brunswick Facility Operating License, February 15, 1990. (Note: As noted to the USNRC in LER 1-94-005, Amendment 172 was assigned to Brunswick Unit 2 at that time incorrectly.)

2. USNRC Regulatory Guide 1.99, Revision 2, Radiation Embrittlement of Reactor Vessel Materials.

3. Letter, R. B. Starkey, Jr. (CP&L) to USNRC Document Control Desk, NLS-92-180, Response to Generic Letter 92-01, Revision 1, July 6, 1992.

SUPPLEMENT 1

BRUNSWICK UNIT 1

RPV SURVEILLANCE MATERIALS TESTING

(38 PAGES)