SCE&G
A SCANA Company

South Carolina Electric & Gas Company
P.O. Box 88
Jenkinsville, SC 29065
(803) 345-4344

Gary J. Taylor
Vice President
Nuclear Operations

November 7, 1995
RC-95-0266

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Gentlemen:

Subject: VIRGIL C. SUMMER NUCLEAR STATION (VCSNS)
DOCKET NO. 50/395
OPERATING LICENSE NO. NPF-12
RESPONSE TO GENERIC LETTER 92-01
REVISION 1, SUPPLEMENT 1

Pursuant to 10CFR50.54(f), the Nuclear Regulatory Commission (NRC) issued Generic Letter 92-01, Revision 1, Supplent 1, "Reactor Vessel Structural Integrity", requiring that each plant respond to parts (2), (3) and (4) by November 19, 1995 providing results of actions taken to locate all data relevant to the determination of Reactor Pressure Vessel (RPV) Integrity.

South Carolina Electric & Gas Company (SCE&G) acting for itself and as agent for South Carolina Public Service Authority, hereby submits the attached in response to Generic Letter 92-01, Revision 1, Supplement 1. This submittal addresses the action required by parts (2), (3) and (4).

I declare that these statements and matters set forth herein are true and correct to the best of my knowledge, information, and belief.

Should you have any questions, please call Mr. Jim Turkett at (803) 345-4047, at your convenience.

Very truly yours,

Gary J. Taylor

Gary J. Taylor

JWT/GJT/nkk
Attachment

c: J. L. Skolds
O. W. Dixon
R. R. Mahan (w/o attachment)
R. J. White
S. D. Ebneter
S. Dembek
J. B. Knotts, Jr.
General Managers
NRC Resident Inspector
Central File System
RTS (LTR 920001-1)
File (815.14)

180006

A028
1/1

STATE OF SOUTH CAROLINA :
: TO WIT :
COUNTY OF FAIRFIELD :

I hereby certify that on the 7th day of November 1995, before me, the subscriber, personally appeared Gary J. Taylor, being duly sworn, states that he is Vice President, Nuclear Operations of the South Carolina Electric & Gas Company, a corporation of the State of South Carolina, that he provides the foregoing response for the purposes therein set forth, that the statements made are true and correct to the best of his knowledge, information, and belief, and that he was authorized to provide the response on behalf of said Corporation.

WITNESS my Hand and Notarial Seal

Michael J Zaccone
Notary Public for South Carolina

My Commission Expires

My Commission Expires July 13, 2005
Date

RESPONSE TO GENERIC LETTER 92-01, REVISION 1, SUPPLEMENT 1 PARTS (2), (3) AND (4)

CONTENTS

REFERENCES	2
RESPONSE TO PART (2)	3
RESPONSE TO PART (3)	3
RESPONSE TO PART (4)	4
RESPONSE - 10CFR 50.60 COMPLIANCE	5
RESPONSE - 10CFR 50.61 COMPLIANCE	6
TABLES 1-16	8

References:

(1) SCE&G letter from J. L. Skolds to Document Control Desk, USNRC; dated June 30, 1992; subject: Virgil C. Summer Nuclear Station, Docket No. 50/395, Operating license No. NPF-12, Response to Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity (LTR 920001).

(2) Davidson, J.A., Yanichko, S.E., "South Carolina Electric And Gas Company Virgil C. Summer Nuclear Plant Unit No. 1 Reactor Vessel Radiation Surveillance Program," WCAP-9234, January, 1978

(3) Boggs, R.S., Fero, A.H., Kaiser, W.T., "Analysis of Capsule U from the South Carolina Electric and Gas Company Virgil C. Summer Unit 1 Reactor Vessel Radiation Surveillance Program," WCAP-10814, June 1985.

(4) Colburn, D.J., Lamantia, L.A., Albertin, L., "Analysis of Capsule V from the South Carolina Electric and Gas Company Virgil C. Summer Unit 1 Reactor Vessel Radiation Surveillance Program," WCAP-11726, January 1988.

(5) Chicots, J.M., Lloyd, T.M., Albertin, L., "Analysis of Capsule X from the South Carolina Electric and Gas Company Virgil C. Summer Unit 1 Reactor Vessel Radiation Surveillance Program," WCAP-12867, March 1991.

(6) SCE&G letter from O. S. Bradham to Document Control Desk, USNRC, dated January 12, 1989, subject: Virgil C. Summer Nuclear Station Docket 50/395, Operating License No. NPF-12, Response to Generic Leterr 88-11.

(7) Chicots, J. M., Ramirez, M.A., "Evaluation of Pressurizer Thermal Shock for V. C. Summer," WCAP-13209, March 1992.

Generic Letter Required information (2):

An assessment of any change in best-estimate chemistry based on consideration of all relevant data.

Response:

Reference (1) submitted the best-estimate chemistry for the V. C. Summer Nuclear Station (VCSNS) Reactor Pressure Vessel (RPV) beltline material. This data has been reviewed and verified to be the best-estimate. This data is repeated below with the sources of this information:

DESCRIPTION	HEAT NO.	%CU	%NI	SOURCE
Intermediate Shell Plate	A9154-1	.10	.51	Lukens Steel Company Test Certificate, Dated 3/14/72
Intermediate Shell Plate	A9153-2	.09	.45	Lukens Steel Company Test Certificate, Dated 3/14/72
Lower Shell Plate	A9923-1	.08	.41	Lukens Steel Company Test Certificate, Dated 1/11/72
Lower Shell Plate	A9923-2	.08	.41	Lukens Steel Company Test Certificate, Dated 1/11/72
Beltline Weld	4P4784	.05	.91	Average of data from: Chicago Bridge & Iron Company Certificate of Analysis, Test #PT321A- Tandem Wire, Revised 2/9/73 %CU = .05 %NI = .91 Chicago Bridge & Iron Company Certificate of Analysis, Test #PT321B- Single Wire, Revised 2/9/73 %CU = .06 %NI = .87 Chemical Analysis of Irradiated Charpy Specimen CW-14 Reported in Ref. (3) %CU = .04 %NI = .95

Generic Letter Required information (3):

A determination of the need for use of the ratio procedure in accordance with the established Position 2.1 of the Regulatory Guide 1.99, Revision 2, for those licensees that use surveillance data to provide a basis for the RPV integrity evaluation.

Response:

For the VCSNS Reactor Vessel Radiation Surveillance Program, Chicago Bridge and Iron Company furnished sections of SA533 Grade B Class 1 plate used in the core region of the VCSNS Reactor Pressure Vessel, specifically, from the 7.75 inch intermediate shell plate A9154-1. Also supplied was a weldment made from sections of plate A9154-1 and adjoining lower shell plate C9923-2, using RACO 1 NMM weld wire, heat number 4P4784, linde 124 flux, lot number 3930. This weld wire and flux combination is identical to that used in forming the longitudinal weld seams in the intermediate and lower shell courses and the girth weld joining these two shell courses. Reference (3), Table 4-1, provide a chemical analysis of the RPV surveillance material. This information provides clear evidence that the copper and nickel content of the surveillance material does not differ from that of the vessel material. Therefore, the measured values of ΔRT_{NDT} do not need to be adjusted by multiplying them by the ratio of the chemistry factor for the vessel weld to that for the surveillance weld.

The surveillance data has been fitted to obtain the relationship of ΔRT_{NDT} to fluence. This fit is calculated in table 1. The values of ΔRT_{NDT} used in Table 1 have been revised from that submitted in Reference (1). These revised values are included in Tables 2 and 8 and are underlined. An explanation for these revisions is included below in the response to the Required Information (4) of the Generic Letter.

Generic Letter Required information (4):

A written report providing any newly acquired data as specified above.

Response:

SCE&G has conducted a review of all the data pertinent to the chemical composition and mechanical properties of the material in the beltline region of the VCSNS Reactor Vessel. As a result of this review, several significant changes were made to the data reported in reference (1). There are two primary reasons for these changes as explained by the following:

- A more exact source of data was obtained for the preirradiated Charpy V-Notch Impact Test results for the Reactor Pressure Vessel core region beltline weld metal.

- Previously, the data from the Charpy V-Notch Impact Test results for surveillance capsules were plotted on a graph and then a curve was fitted to the data points by hand. In order to standardize the interpretation of the Charpy V-Notch Impact Test results and to avoid individual interpretation of data, all the results have been replotted using the computer program CVGRAPH, Version 4.0. This program has been developed by the Westinghouse Owner's Group. The CVGRAPH 4.0 Program is a data tool for the fitting of Charpy V- Notch test results.

The program stores the data and graphically displays the test results. The curve fit routine uses a hyperbolic tangent function. The curve calculator provides results in temperature at 30 and 50 ft-lb impact energy.

Tables 2 through 8 are tables that were previously submitted in reference (1). Tables 2, 7, and 8 required revisions due to the reasons noted above. Revised data is indicated by an underline.

Tables 9 and 10 include revisions made to tables of a printout from the database RVID. These revisions are also due to the reasons noted above. Revised data is indicated by a cloud.

Also during this review, existing industry wide databases were reviewed to identify any "Sister" vessels. It was concluded that the intermediate and lower shell axial welds of the Shearon Harris Unit 1 Reactor Vessel were fabricated using the same weld heat material, 4P4784, as that used for welds in the VCSNS vessel. Therefore, the Shearon Harris Reactor Vessel is a "Sister" vessel of the VCSNS vessel. In this regard, discussions between these two units have been instigated and are continuing. The intent of these discussions is for a continuing transfer of data on reactor vessel mechanical properties. The Shearon Harris vessel does not contain any surveillance capsules from the weld material, heat 4P4784. Therefore, no surveillance data will be available for use in the VCSNS surveillance program. However, data from the VCSNS surveillance program will be made available to Shearon Harris.

Additional information required by Generic Letter (4)

(1) the results of any necessary revisions to the evaluation of RPV integrity in accordance with the requirements of 10 CFR 50.60, 10 CFR 50.61, Appendices G and H to 10 CFR Part 50, and any potential impact on the LTOP or P-T limits in the technical specifications.

Response - 10 CFR 50.60 compliance:

This submittal does not make any changes to the Charpy Upper Shelf Energies (USE) that would affect the evaluation reported in reference (1) for the compliance with Paragraphs IV.A.1 of Appendix G to 10 CFR Part 50. This evaluation is repeated as follows:

"A maximum 10 ft-lb, or 7.6%, decrease in USE was determined for plate A9154-1 (longitudinal orientation). This data is plotted on the Regulatory Guide 1.99, Revision 2 curve using the guidance set forth in paragraph C.2.2 of Regulatory Guide 1.99, Revision 2. Based on actual data, the projected decrease in USE at the projected end of life (32 EFPY) fluence of $3.87X10^{19}$, reference 5, is expected to be less than 10%. Using the lowest initial USE value of 75 ft-lb for the intermediate shell plate A9154-1, the resulting end of life USE value is projected to be 67.5 ft-lb. This is above the 50 ft-lb limit and, therefore, is acceptable."

The P-T limits in the Technical Specifications are to assure that plant operations comply with the requirements of 10 CFR 50.60 and Appendix G. The P-T limits are based on an initial unirradiated value of RT_{NDT} which increases as the material is exposed to fast neutron radiation. The initial value of RT_{NDT}, adjusted for radiation exposure, is termed ART. To find the most limiting ART at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the original unirradiated RT_{NDT}. The current P-T curves in the VCSNS Technical Specifications are based on a value of ART, adjusted for radiation exposure, derived from data obtained from three surveillance capsules. The derivation of these curves is summarized in reference (5). The value of the limiting ART, and the corresponding P-T curves, is based on the following:

EFPY = 14

Controlling Material = Lower Shell

Initial RT_{NDT} = 10°F

ART at 1/4 T = 96°F

ART at 3/4 T = 83°F

Revisions to data included in this submittal will revise the data used to derive the values of ART for some of the beltline materials. Tables 11 through 15 provide a revised set of calculations for the derivation of ART (14 EFPY) for all beltline materials. From these Tables, it is concluded that the controlling material remains the lower shell plate. The revised values of ART are as follows (note that they remain unchanged):

Controlling Material (revised) = Lower Shell

Initial RT_{NDT} = 10°F

ART at 1/4 T (revised) = 96°F

ART at 3/4 T (revised) = 83°F

The ASME approach (Part 50 Appendix G sanctioned) for calculating the allowable limit curves for various heatup and cooldown curves specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time.

K_I function (P, T)

$K_I < K_{IR}$

K_{IR} is obtained from the reference fracture toughness curve given by the following equation:

$$K_{IR} = 26.78 + 1.223 \exp[0.0145 (T - ART + 160)]$$

The values of the revised ART at 1/4 T and at 3/4 T are equal to the values used to develop the current P-T curves. Therefore, the current P-T curves remain valid.

In regards to Low Temperature Overpressure Protection (LTOP), the RHR relief valve setpoints have been verified for compliance with Appendix G based upon the pressure and temperature limits of the current P-T curves in the Technical

Specification. Therefore, the current setpoints remain valid for the revised data of this submittal.

It is concluded that the current evaluations contained in reference (5) for compliance with 10 CFR 50.60 (and Appendix G) and for the development of the current P-T curves in the Technical Specifications remain valid for the revised data of this submittal. Therefore, RPV integrity remains in compliance with the requirements of 10 CFR 50.60 and no revisions are necessary for the current P-T curves of the Technical Specifications or the RHR relief valve setpoints for LTOP.

Response - 10 CFR 50.61 compliance:

Using the prescribed PTS rule methodology outlined in 10 CFR 50.61, RT_{PTS} values were generated for all beltline region materials of the VCSNS Reactor Vessel as a function of end-of-life (32 EFPY) fluence values. Table 16 provide a summary of the RT_{PTS} calculated values for all the beltline region materials. The information on this table updates and revises that of references (6) and (7). The fluence data were generated based on the most recent surveillance capsule program results of reference (5). The chemistry factors for the intermediate shell plate, A9154-1, and the welds were calculated using surveillance capsule data and Regulatory Guide 1.99, Revision 2, methodology as shown in Table 1.

Per 10 CFR 50.61, the NRC screening value for RT_{PTS} using the current projected fluence values for the end-of-life (32 EFPY) is 270°F for plates and axial welds and 300°F for circumferential welds. As indicated in Table 16, the maximum projected RT_{PTS} for all plates and axial welds is 113°F for the lower shell plate C9923-1. This is well below the screening criteria of 270°F. The maximum projected RT_{PTS} for the circumferential welds is 22°F. This is also significantly below the 300°F screening criteria. Therefore, the PTS Rule evaluation based on the revised data of this submittal concludes a compliance with the requirements of 10 CFR 50.61.

TABLE 1
CALCULATION OF CHEMISTRY FACTORS USING V. C. SUMMER SURVEILLANCE CAPSULE DATA

Component	Capsule	Fluence	FF	DRTNDT	FF*DRTNDT	$(FF)^2$
Int. Shell, A9154-1	U	0.639	0.874	36	36.464	0.765
(Long.)	V	1.470	1.107	53	58.671	1.225
	X	2.460	1.242	38	47.196	1.543
Int. Shell, A9154-1	U	0.639	0.874	15	13.110	0.765
(Trans.)	V	1.470	1.107	33	36.531	1.225
	X	2.460	1.242	26	32.292	1.543
					219.264	7.065

Chemistry Factor = 219.264 / 7.065 = 31.035

Component	Capsule	Fluence	FF	DRTNDT	FF*DRTNDT	$(FF)^2$
Weld Metal	U	0.639	0.874	23	20.102	0.765
	V	1.470	1.107	47	52.029	1.225
	X	2.460	1.242	23	28.566	1.543
					100.697	3.533

Chemistry Factor = 100.697 / 3.533 = 28.502

TABLE 2
EFFECT OF 550°F IRRADIATION ON NOTCH TOUGHNESS PROPERTIES OF THE
V. C. SUMMER UNIT 1 REACTOR VESSEL SURVEILLANCE CAPSULE MATERIAL

Material	Capsule	Fluence 10^{19} N/CM^2	Average 30 Ft-Lb Temp. (°F)	ΔT_{30}	Average 50 Ft-Lb Temp. (°F)	Average Energy Absorption At Full Shear (Ft-Lb)	USE % Decrease	Reference
Int. Plate A9154-1	unirr.	0	-22	--	7	132	--	2
(Longitudinal)	U	0.639	14	36	52	131	.8	3
	V	1.470	31	53	71	122	7.6	4
	X	2.460	16	38	51	125	5.3	5
Int. Plate A9154-1	unirr.	0	28	--	71	75	--	2
(Transverse)	U	0.639	43	15	94	75	0	3
	V	1.470	61	33	137	76	--	4
	X	2.460	54	26	112	73	2.7	5
Weld Metal	unirr.	0	-53	--	-13	91	--	2
	U	0.639	-30	23	10	87	4.4	3
	V	1.470	-6	47	30	85	6.6	4
	X	2.460	-30	23	5	85	6.6	5
HAZ Metal	unirr.	0	-93	--	-65	130	--	2
	U	0.639	-58	35	-28	121	6.9	3
	V	1.470	-43	50	-13	111	14.6	4
	X	2.460	-37	56	-5	111	14.6	5

TABLE 3

DESCRIPTION:	Reactor Vessel Beltline Plate (Intermediate Shell)
HEAT NO.:	A9154-1
SPECIFICATION NO.:	SA533 Grade B Class 1
SUPPLIER:	Lukens Steel Company
HEAT TREATMENT:	Austenitized at 1625°F for 4 hours Water Quenched Tempered at 1280°F for 4 hours Air Cooled Stress Relieved at 1050°F for 43 hours Furnace Cooled/Air Cooled
COMPOSITION:	.10 %Cu, .51 %Ni, .009 %P, .015 %S
DROP WEIGHT T_{NDT}:	-20°F
RT_{NDT}:	30°F (measured)

PREIRRADIATION CHARPY V-NOTCH IMPACT DATA FOR THE V. C. SUMMER UNIT NO. 1 REACTOR PRESSURE VESSEL INTERMEDIATE SHELL PLATE A9154-1

(LONGITUDINAL ORIENTATION)

TEST TEMPERATURE (°F)	IMPACT ENERGY (FT-LB)
+212	121, 144, 144
+70	80, 107, 112
+40	70, 88, 70
+10	54, 50, 48
-20	17, 10, 15
-100	5, 3, 4

(TRANSVERSE ORIENTATION)

TEST TEMPERATURE (°F)	IMPACT ENERGY (FT-LB)
+212	82.5, 76.5, 83
+120	56, 64, 74
+90	51.5, 60.5, 58
+70	56, 36, 40
+40	36, 41, 38
-20	12.5, 20, 19

TABLE 4

DESCRIPTION:	Reactor Vessel Beltline Plate (Intermediate Shell)
HEAT NO.:	A9153-2
SPECIFICATION NO.:	SA533 Grade B Class 1
SUPPLIER:	Lukens Steel Company
HEAT TREATMENT:	Austenitized at 1625°F for 4 hours Water Quenched Tempered at 1280°F for 4 hours Air Cooled Stress Relieved at 1050°F for 43 hours Furnace Cooled/Air Cooled
COMPOSITION:	.09 %Cu, .45 %Ni, .006 %P, .016 %S
DROP WEIGHT T_{NDT}:	-20°F
RT_{NDT}:	-20° (measured)

PREIRRADIATION CHARPY V-NOTCH IMPACT DATA FOR THE V. C. SUMMER UNIT NO. 1
REACTOR PRESSURE VESSEL INTERMEDIATE SHELL PLATE A9153-2

(LONGITUDINAL ORIENTATION)

TEST TEMPERATURE (°F)	IMPACT ENERGY (FT-LB)
+212	142, 146, 136
+70	108, 107, 105
+40	82, 83, 82
+10	68, 56, 74
-20	54, 58, 54
-100	9, 3, 8

(TRANSVERSE ORIENTATION)

TEST TEMPERATURE (°F)	IMPACT ENERGY (FT-LB)
+212	101, 108, 111.5
+120	93, 90, 94
+70	65, 71, 68
+40	59, 51, 50
+20	43, 42.5, 47
-20	14, 26, 18

TABLE 5

DESCRIPTION:	Reactor Vessel Beltline Plate (Lower Shell)
HEAT NO.:	C9923-2
SPECIFICATION NO.:	SA533 Grade B Class 1
SUPPLIER:	Lukens Steel Company
HEAT TREATMENT:	Austenitized at 1600°F for 4 hours Water Quenched Tempered at 1260°F for 4 hours Air Cooled Stress Relieved at 1075°F for 41 hours Furnace Cooled/Air Cooled
COMPOSITION:	.08 %Cu, .41 %Ni, .005 %P, .015 %S
DROP WEIGHT T_{NDT}:	-10°F
RT_{NDT}:	10° (measured)

PREIRRADIATION CHARPY V-NOTCH IMPACT DATA FOR THE V. C. SUMMER UNIT NO. 1
REACTOR PRESSURE VESSEL INTERMEDIATE SHELL PLATE C9923-2

(LONGITUDINAL ORIENTATION)

TEST TEMPERATURE (°F)	IMPACT ENERGY (FT-LB)
+212	164, 164, 155
+40	86, 76, 85
+10	56, 56, 55
-20	42, 50, 35
-50	8, 6, 12
-100	4, 5, 5

(TRANSVERSE ORIENTATION)

TEST TEMPERATURE (°F)	IMPACT ENERGY (FT-LB)
+212	93, 94, 88
+120	82, 79, 84
+70	55, 50, 51
+50	40, 46, 53
+40	56, 42.5, 37.5
-10	23, 31, 33.5

TABLE 6

DESCRIPTION:	Reactor Vessel Beltline Plate (Lower Shell)
HEAT NO.:	C9923-1
SPECIFICATION NO.:	SA533 Grade B Class 1
SUPPLIER:	Lukens Steel Company
HEAT TREATMENT:	Austenitized at 1600°F for 4 hours Water Quenched Tempered at 1260°F for 4 hours Air Cooled Stress Relieved at 1075°F for 41 hours Furnace Cooled/Air Cooled
COMPOSITION:	.08 %Cu, .41 %Ni, .005 %P, .014 %S
DROP WEIGHT T_{NDT}:	-30°F
RT_{NDT}:	10° (measured)

PREIRRADIATION CHARPY V-NOTCH IMPACT DATA FOR THE V. C. SUMMER UNIT NO. 1 REACTOR PRESSURE VESSEL INTERMEDIATE SHELL PLATE C9923-1

(LONGITUDINAL ORIENTATION)

TEST TEMPERATURE (°F)	IMPACT ENERGY (FT-LB)
+212	148, 147, 150
+40	92, 100, 89
+10	92, 70, 82
-50	52, 60, 44
-75	35, 17, 12
-100	7, 10, 9

(TRANSVERSE ORIENTATION)

TEST TEMPERATURE (°F)	IMPACT ENERGY (FT-LB)
+212	104, 114, 100
+120	74.5, 81, 80
+70	68, 51, 51
+50	47, 44, 43.5
+30	37, 42.5, 35
-30	11, 9, 12

TABLE 7

DESCRIPTION:	Reactor Vessel Core Region Beltline Weld Metal
WIRE HEAT NO.:	4P4784
WIRE TYPE:	RACO 1NMM
FLUX TYPE:	Linde 124
FLUX LOT:	3930
FABRICATOR:	Chicago Bridge and Iron Co.
HEAT TREATMENT:	1150°F for 12 hours
WIRE COMPOSITION:	.05 %Cu, .91 %Ni, .013 %P, .012 %S
DROP WEIGHT T_{NDT}:	-50°F
RT_{NDT}:	-44° (measured)

PREIRRADIATION CHARPY V-NOTCH IMPACT DATA FOR THE V. C. SUMMER UNIT NO. 1 REACTOR PRESSURE VESSEL CORE REGION BELTLINE WELD METAL

TEST TEMPERATURE (°F)	IMPACT ENERGY (FT-LB)
+212	79, 87, 87
+40	64, 70, 66
+10	53, 46, 50
-20	33, 22, 38
-100	8, 7, 8
-120	12, 7, 9

TABLE 8
COMPARISON OF V. C. SUMMER UNIT 1 SURVEILLANCE MATERIAL 30 FT-LB TRANSITION TEMPERATURE SHIFTS AND UPPER SHELF ENERGY DECREASES WITH REGULATORY GUIDE 1.99 REVISION 2 PREDICTIONS

			30 ft-lb Transition Temp. Shift R. G. 1.99 Rev. 2		Upper Shelf Energy Decrease R. G. 1.99 Rev. 2	
Material	Capsule	Fluence 10^{19} n/cm^2	(Predicted) (°F)	Measured (°F)	(Predicted) (%)	Measured (%)
Int. Plate A9154-1	U	0.639	57	36	17	1
(Longitudinal)	V	1.47	72	53	21	8
	X	2.46	81	38	23	5
Int. Plate A9154-1	U	0.639	57	15	17	0
(Transverse)	V	1.47	72	33	21	0
	X	2.46	81	26	23	3
Weld Metal	U	0.639	59	23	18	4
	V	1.47	75	47	22	7
	X	2.46	85	23	25	7
HAZ Metal	U	0.639	--	35	--	7
	V	1.47	--	50	--	15
	X	2.46	--	56	--	15

TABLE 9

REACTOR VESSEL INTEGRITY DATABASE
Summary File for Upper Shelf Energy

Plant Name	Beltline Ident.	Heat No. Ident.	Material Type	USE @ EOL @ 1/4T	1/4T Neut. Flu @ EOL	Unirr USE	Method Determ Unirr USE	% Drop USE @ EOL @ 1/4T	Method Determ % Drop	Cu		
Summer	(Continued)	Docket No.:	50-395									
	LOWER SHELL	C9923-1	A 5338	81	2.431	106	DIRECT	23.4%	Position 1 of RG 1.99, Rev. 2	0.08		
	LOWER SHELL	C9923-2	A 5338	70	2.431	92	DIRECT	23.4%	Position 1 of RG 1.99, Rev. 2	0.08		
	WELDS	4P4784	LINDE 124	78	2.431	84	DIRECT	7.4%	Surveillance data	0.05		
	INTERMEDIATE SHELL	A9153-2	A 5338	82	2.431	107	DIRECT	23.4%	Position 1 of RG 1.99, Rev. 2	0.09		
	INTERMEDIATE SHELL	A9154-1	A 5338	74	2.431	81	DIRECT	8.5%	Surveillance data	0.10		

References for Summer

>>>>>GL 92-01 References<<<<<

Fluence, chemical composition, and IRTndt data are from June 30, 1992, letter from J. L. Skolds (SCE&G) to USNRC Document Control Desk, subject: Response to Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity

UUSE data are from Table A-1 of WCAP-10814

TABLE 10

REACTOR VESSEL INTEGRITY DATABASE
Summary File for PTS

Plant Name	Beltline Ident.	Heat No. Ident.	RTpts @ EOL	ID Neut. Fluence @ EOL	IRTndt	Method Determ IRTndt	ΔRTndt at EOL	Fluence Factor @ EOL	Chemistry Factor	Method of Determin. CF	Margin	Method of Determin. Margin	Cu%	Ni%
Summer	EOL: 08/06/22	Docket No.:	50-395											
	INTERMEDIATE SHELL	A9153-2	92	3.87000	-20	PLANT SPEC	78.2	1.349	58.00	Table	34.00	TABLE	0.090	0.450
	INTERMEDIATE SHELL	A9154-1	89	3.87000	30	PLANT SPEC	41.9	1.349	31.00	Calculated	17.00	TABLE	0.100	0.510
	LOWER SHELL	C9923-1	113	3.87000	10	PLANT SPEC	68.8	1.349	51.00	Table	34.00	TABLE	0.080	0.410
	LOWER SHELL	C9923-2	113	3.87000	10	PLANT SPEC	68.8	1.349	51.00	Table	34.00	TABLE	0.080	0.410
	WELDS	4P4784	22	3.87000	-44	PLANT SPEC	38.5	1.349	28.50	Calculated	28.00	TABLE	0.050	0.910

References for Summer

>>>>>GL 92-01 References<<<<<

Fluence, chemical composition, and IRTndt data are from June 30, 1992, letter from J. L. Skolds (SCE&G) to USNRC Document Control Desk, subject: Response to Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity

UUSE data are from Table A-1 of WCAP-10814

TABLE 11

CALCULATION OF ART FOR V. C. SUMMER REACTOR VESSEL MATERIAL
INTERMEDIATE SHELL PLATE A9154-1

Parameter	14 EFPY 1/4T	14 EFPY 3/4T
Chemistry Factor, CF (°F)	31	31
Fluence, f (10^{19} n/cm^2)(a)	1.08	0.43
Fluence Factor, ff	1.023	0.765
******		
ΔRT_{NDT} = CF x ff (°F)	31.7	23.7
Initial $RT_{NDT,I}$ (°F)	30	30
Margin, M (°F)	17	17
******		
Adjusted Reference Temperature, ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin	79	71
Total ART_{14EFPY} (b)	96	83

****

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E > 1 Mev) = 1.73 at 14EFPY (projections determined from Reference (5), Table 6-13). The V. C. Summer reactor vessel wall thickness is 7.75 inches at the beltline region.

(b) The current V. C. Summer heatup and cooldown pressure/temperature limits are based upon these total ART values for 14EFPY.

TABLE 12

CALCULATION OF ART FOR V. C. SUMMER REACTOR VESSEL MATERIAL
INTERMEDIATE SHELL PLATE A9153-2

Parameter	14 EFPY	
	1/4T	3/4T
Chemistry Factor, CF (°F)	58	58
Fluence, f (10^{19} n/cm^2)(a)	1.08	0.43
Fluence Factor, ff	1.023	0.765

****

ΔRT_{NDT} = CF x ff (°F)	59.3	44.4
Initial $RT_{NDT,I}$ (°F)	-20	-20
Margin, M (°F)	34	34

****

Adjusted Reference Temperature, ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin	73.3	58.4
Total ART_{14EFPY} (b)	96	83

****

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E > 1 Mev) = 1.73 at 14EFPY (projections determined from Reference (5), Table 6-13). The V. C. Summer reactor vessel wall thickness is 7.75 inches at the beltline region.

(b) The current V. C. Summer heatup and cooldown pressure/temperature limits are based upon these total ART values for 14EFPY.

TABLE 13

CALCULATION OF ART FOR V. C. SUMMER REACTOR VESSEL MATERIAL
LOWER SHELL PLATE C9923-1

Parameter	14 EFPY	
	1/4T	3/4T
Chemistry Factor, CF (°F)	51	51
Fluence, f (10^{19} n/cm^2)(a)	1.08	0.43
Fluence Factor, ff	1.023	0.765

****

ΔRT_{NDT} = CF x ff (°F)	52.2	39.0
Initial $RT_{NDT,I}$ (°F)	10	10
Margin, M (°F)	34	34

****

Adjusted Reference Temperature, ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin	96	83
Total ART_{14EFPY} (b)	96	83

****

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E > 1 Mev) = 1.73 at 14EFPY (projections determined from Reference (5), Table 6-13). The V. C. Summer reactor vessel wall thickness is 7.75 inches at the beltline region.

(b) The current V. C. Summer heatup and cooldown pressure/temperature limits are based upon these total ART values for 14EFPY.

TABLE 14

CALCULATION OF ART FOR V. C. SUMMER REACTOR VESSEL MATERIAL
LOWER SHELL PLATE C9923-2

Parameter	14 EFPY	
	1/4T	3/4T
Chemistry Factor, CF (°F)	51	51
Fluence, f (10^{19} n/cm^2)(a)	1.08	0.43
Fluence Factor, ff	1.023	0.765

****

ΔRT_{NDT} = CF x ff (°F)	52.2	39.0
Initial $RT_{NDT,I}$ (°F)	10	10
Margin, M (°F)	34	34

****

Adjusted Reference Temperature, ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin	96	83
Total ART_{14EFPY} (b)	96	83

****

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E > 1 Mev) = 1.73 at 14EFPY (projections determined from Reference (5), Table 6-13). The V. C. Summer reactor vessel wall thickness is 7.75 inches at the beltline region.

(b) The current V. C. Summer heatup and cooldown pressure/temperature limits are based upon these total ART_{14EFPY} values for 14EFPY.

TABLE 15

CALCULATION OF ART FOR V. C. SUMMER REACTOR VESSEL MATERIAL WELD (INTERMEDIATE TO LOWER SHELL)

Parameter	14 EFPY	
	1/4T	3/4T
Chemistry Factor, CF (°F)	28.5	28.5
Fluence, f (10^{19} n/cm^2)(a)	1.08	0.43
Fluence Factor, ff	1.023	0.765

****

ΔRT_{NDT} = CF x ff (°F)	29.2	21.8
Initial $RT_{NDT,I}$ (°F)	-44	-44
Margin, M (°F)	28	28

****

Adjusted Reference Temperature, ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin	13	6
Total ART_{14EFPY} (b)	96	83

****

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E > 1 Mev) = 1.73 at 14EFPY (projections determined from Reference (5), Table 6-13). The V. C. Summer reactor vessel wall thickness is 7.75 inches at the beltline region.

(b) The current V. C. Summer heatup and cooldown pressure/temperature limits are based upon these total ART values for 14EFPY.

TABLE 16

V. C. Summer Reactor Vessel
Beltline RT_{PTS} Values
At End Of Life (32EFPY)

Material	ΔRTNDT(°F) (CF	x FF*)	+	Initial RT_{NDT} (°F)	+	Margin (°F)	=	RT_{PTS} (°F)
Intermediate Shell Plate, A9154-1	(31.0)	1.35		30		17		(89)
Intermediate Shell Plate, A9153-2	58	1.35		-20		34		92
Lower Shell Plate, C9923-1	51	1.35		10		34		113
Lower Shell Plate, C9923-2	51	1.35		10		34		113
Circumferential Weld Seam	(28.5)	1.35		-44		28		(22)
Longitudinal Welds	(28.5)	1.08		-44		28		(15)

( ) Indicates numbers were calculated using surveillance capsule data.

* Fluence factor based upon peak inner surface neutron fluence of 3.87×10^{19} n/cm^2 [Reference (5)], except for longitudinal welds. For longitudinal welds, the fluence factor is based on a neutron fluence of 1.33×10^{19} n/cm^2 [Reference (5)] at the inner surface of the weld.