Georgia Power Company
40 Inverness Center Parkway
Post Office Box 1295
Birmingham, Alabama 35201
Telephone 205 877-7122

C. K. McCoy
Vice President, Nuclear
Vogtle Project

Georgia Power
the southern electric system

June 25, 1992

ELV-03648
001530

Docket Nos. 50-424
50-425

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

Gentlemen:

VOGTLE ELECTRIC GENERATING PLANT
REACTOR VESSEL STRUCTURAL INTEGRITY
GENERIC LETTER 92-01 REVISION 1

In response to Generic Letter 92-01, Revision 1 concerning reactor vessel structural integrity, Georgia Power Company (GPC) is submitting the enclosed information.

Mr. C. K. McCoy states that he is a vice president of Georgia Power Company and is authorized to execute this oath on behalf of Georgia Power Company and that, to the best of his knowledge and belief, the facts set forth in this letter and enclosure are true.

GEORGIA POWER COMPANY

By: C.K. McCoy
C. K. McCoy

Sworn to and subscribed before me this 25th day of June, 1992.

Mary N. Bentley
Notary Public

MY COMMISSION EXPIRES MAY 6, 1995

A001 1/1 Add: D. McDonald 41 Encl 1 1
K. Wickman 1 1
B. Elliot 1 1

300070

Georgia Power

Enclosures: Response to Generic Letter 92-01
Table 5.3.3-2
Table 5.3.3-3
WCAP-11011
WCAP-11381

CKM/PAH/gmb

c(w): Georgia Power Company
Mr. W. B. Shipman
Mr. M. Sheibani
NORMS

U. S. Nuclear Regulatory Commission
Mr. S. D. Ebneter, Regional Administrator
Mr. D. S. Hood, Licensing Project Manager, NRR
Mr. B. R. Bonser, Senior Resident Inspector, Vogtle

700775

ENCLOSURE
VOGTLE ELECTRIC GENERATING PLANT
RESPONSE TO GENERIC LETTER 92-01 REVISION 1

REACTOR VESSEL STRUCTURAL INTEGRITY

Generic Letter 92-01

1. Certain addressees are requested to provide the following information regarding Appendix H to CFR Part 50:

   Addressees who do not have a surveillance program meeting ASTM E 185-73,-79, or -82 and who do not have an integrated surveillance program approved by the NRC (see Enclosure 2), are requested to describe actions taken or to be taken to ensure compliance with Appendix H to 10 CFR Part 50. Addressees who plan to revise the surveillance program to meet Appendix H to 10 CFR Part 50 are requested to indicate when the revised program will be submitted to the NRC staff for review. If the surveillance program is not to be revised to meet Appendix H to 10 CFR Part 50, addressees are requested to indicate when they plan to request an exemption from Appendix H to 10 CFR Part 50 under 10 CFR 50.60(b).

GPC Response

The capsule surveillance program for the Vogtle Electric Generating Plant (VEGP) meets ASTM 185-82. The capsule surveillance program is in compliance with the requirements of Appendix H to 10 CFR Part 50; therefore, a revised program or an exemption is not required.

Generic Letter 92-01

2. Certain addressees are requested to provide the following information regarding Appendix G to 10 CFR Part 50:

   a. Addressees of plants for which the Charpy upper shelf energy is predicted to be less than 50 foot-pounds at the end of their licenses using the guidance in Paragraphs C.1.2 or C.2.2 in Regulatory Guide 1.99, Revision 2, are requested to provide to the NRC the Charpy upper shelf energy predicted for December 16, 1991, and for the end of their current license for the limiting beltline weld and the plate or forging and are requested to describe the actions taken pursuant to Paragraphs IV.A.1 or V.C of Appendix G to 10 CFR Part 50.

GPC Response

Calculations for the Vogtle Electric Generating Plant using the guidance of Regulatory Guide 1.99, Revision 2 were performed by Westinghouse Electric Corporation for both reactor vessels. These calculations indicated that the Charpy upper shelf energy is expected to be above 50 ft-lbs, even after 48 effective full power years (EFPY) of operation. Tables 5.3.3-2 and 5.3.3-3, which are enclosed, were prepared for the next Final Safety Analysis Report (FSAR) revision and show the results of these calculations for each vessel.

REACTOR VESSEL STRUCTURAL INTEGRITY

Generic Letter 92-01

b. Addressees whose reactor vessels were constructed to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition are requested to describe the consideration given to the following material properties in their evaluations performed pursuant to 10 CFR 50.61 and Paragraph III.A of 10 CFR Part 50, Appendix G:

GPC Response:

The reactor vessels at VEGP were constructed to the ASME Summer 1972 addenda of the 1971 edition of the code.

Generic Letter 92-01

(1) the results from all Charpy and drop weight tests for all unirradiated beltline materials, the unirradiated reference temperature for each beltline material, and the method of determining the unirradiated reference temperature from the Charpy and drop weight test;

GPC Response

The results from the Charpy tests for the unirradiated beltline materials for both reactor vessels are found in the FSAR in tables 5.3.2-2, 5.3.2-3, 5.3.2-4, and 5.3.2-5. The unirradiated reference temperature for the materials is also shown on these tables. It should be noted that the RTNDT for plate B8805-3 (in FSAR table 5.3.2-2 sheet 1 of 2) should be +30°F instead of -30°F. This value will be changed in the next FSAR update. The method of determining the unirradiated reference temperature from the Charpy and drop weight tests is located in WCAP-11011, "Georgia Power Company Alvin W. Vogtle Unit No.1 Reactor Vessel Radiation Surveillance Program," and WCAP-11381, "Georgia Power Company Alvin W. Vogtle Unit No. 2 Reactor Vessel Radiation Surveillance Program" in sections 3.1 and 3.3.

The drop weight test data is currently maintained by Westinghouse. The results of the drop weight tests (TNDT values) are shown in tables 5.3.2-2, 5.3.2-3, 5.3.2-4, and 5.3.2-5 of the FSAR for both reactor vessels.

REACTOR VESSEL STRUCTURAL INTEGRITY

Generic Letter 92-01

(2) the heat treatment received by all beltline and surveillance materials;

GPC Response

The heat treatment for VEGP Unit 1 is shown in table A-5 of WCAP-11011; the heat treatment for VEGP Unit 2 is shown in table A-6 of WCAP-11381.

Generic Letter 92-01

(3) the heat number for each beltline plate or forging and the heat number of wire and flux lot number used to fabricate each beltline weld;

GPC Response

The heat number of each beltline plate is on the Combustion Engineering material certification reports maintained by Westinghouse. There are no beltline forging materials. The heat number of the wire and flux lot used to fabricate each beltline weld is found for VEGP Unit 1 in table A-3 of WCAP-11011 and is found for VEGP Unit 2 in tables A-3 and A-4 of WCAP-11381.

Generic Letter 92-01

(4) the heat number for each surveillance plate or forging and the heat number of wire and flux lot number used to fabricate the surveillance weld;

GPC Response

The heat number of each surveillance plate is maintained by Westinghouse. The heat number of wire and flux lot number used to fabricate the surveillance weld is found in table A-3 of WCAP-11011 for VEGP Unit 1 and is found in table A-4 of WCAP-11381 for VEGP Unit 2.

Generic Letter 92-01

(5) the chemical composition, in particular the weight in percent of copper, nickel, phosphorous, and sulfur for each beltline and surveillance material; and

REACTOR VESSEL STRUCTURAL INTEGRITY

GPC Response

The chemical compositions for VEGP Unit 1 are found in tables A-1, A-2, and A-3 of WCAP-11011 and for VEGP Unit 2 are found in tables A-1, A-2, A-3, and A-4 of WCAP-11381. In WCAP-11011, table A-1 compares the results of the Combustion Engineering and Westinghouse chemical analysis of plate B8805-3, and table A-3 compares the results of the chemical analysis of the weld metal used in the core region seam welds. In WCAP-11381, table A-2 compares the results of the Combustion Engineering and Westinghouse chemical analysis on plate B8628-1, and table A-4 compares the results of the chemical analysis on the weld metal used in the intermediate to lower shell closing girth seam weld.

Generric Letter 92-01

(6) the heat number of the wire used for determining the weld metal chemical composition if different from items (3) above:

GPC Response

Not appliable.

Generic Letter 92-01

3. Addressees are requested to provide the following information regarding commitments made to respond to GL 88-11:

   a. How the embrittlement effects of operating at an irradiation temperature (cold leg or recirculation suction temperature) below 525°F were considered. In particular licensees are requested to describe consideration given to determining the effect of lower irradiation temperature on the reference temperature and on the Charpy upper shelf energy.

GPC Response

The Technical Specifications for the Vogtle Electric Generating Plant require that critical operation occurs at a temperature of 551°F or higher. Some physics tests are allowed when the reactor coolant system lowest operating loop temperature (T_{avg}) is greater than or equal to 541°F. Critical operation does not occur at temperatures below 525°F.

REACTOR VESSEL STRUCTURAL INTEGRITY

Generic Letter 92-01

b. How their surveillance results on the predicted amount of embrittlement were considered.

GPC Response

The mean values of copper and nickel were used for the generation of the chemistry factor in the calculation of the change in RTNDT utilizing Regulatory Guide 1.99, Revision 2. For VEGP, the surveillance results indicate the changes in Charpy upper shelf energy and the 30 ft-lb transition temperature shift values are less than those predicted utilizing Regulatory Guide 1.99, Revision 2.

Generic Letter 92-01

c. If a measured increase in reference temperature exceeds the mean-plus-two standard deviations predicted by Regulatory Guide 1.99, Revision 2, or if a measured decrease in Charpy upper shelf energy exceeds the value predicted using the guidance in Paragraph C.1.2 in Regulatory Guide 1.99, Revision 2, the licensee is requested to report the information and describe the effect of the surveillance results on the adjusted reference temperature and Charpy upper shelf energy for each beltline material as predicted for December 16, 1991, and for the end of its current license.

GPC Response

The measured increase in reference temperature does not exceed the mean-plus-two standard deviations predicted by Regulatory Guide 1.99, Revision 2. The measured decrease in Charpy upper shelf energy does not exceed the value predicted using the guidance in Paragraph C.1.2 in Regulatory Guide 1.99, Revision 2. Table 5-6 in WCAP-12256, "Analysis of Capsule U from the Georgia Power Company Vogtle Unit 1 Reactor Vessel Radiation Surveillance Program," and table 5-6 in WCAP-13007, "Analysis of Capsule U from the Georgia Power Company Vogtle Electric Generating Plant Unit 2 Reactor Vessel Radiation Surveillance Program," compare the Charpy upper shelf energy values and the 30 ft-lb transition temperature shift values to those predicted utilizing Regulatory Guide 1.99, Revision 2.

TABLE 5.3.3-2

UNIT 1 REACTOR VESSEL VALUES FOR ANALYSIS OF POTENTIAL PRESSURIZED THERMAL SHOCK EVENTS (a)

Material	Cu wt-%	Ni wt-%	Initial RTNDT (deg. F)	10CFR50.61 Predicted RTPTS (degrees F) Dec. 16, 1991	32 EFPY	48 EFPY	Regulatory Guide 1.99, Rev. 2 RTNDT (deg. F) Dec. 16, 1991	32 EFPY	48 EFPY	Initial USE (ft-lbs)	Regulatory Guide 1.99, R2 Predicted USE (ft-lbs) Dec. 16, 1991	32 EFPY	48 EFPY
Intermed. Shell Plate, B8805-1	0.08	0.59	0	67	100	105	67	100	105	90	78	70	68
Intermed. Shell Plate, B8805-2	0.08	0.59	20	87	120 (b)	125	87	120	125	100	87	78	76
Intermed. Shell Plate, B8805-3	0.06	0.60	30	88	112	116	83	112	116	107	93	83	81
Lower Shell Plate, B8606-1	0.05	0.59	20	74	94	97	74	94	97	116	101	90	88
Lower Shell Plate, B8606-2	0.05	0.58	20	74	94	97	74	94	97	113	98	88	86
Lower Shell Plate, B8606-3	0.06	0.64	10	68	92	96	68	92	96	118	103	92	90
Core Region Longitudinal & Girth Seams (c)	0.04	0.10	-80	-2	20	23	-2	20	23	134	116	105	102

NOTES:

a. RTPTS and RTNDT values are based on the peak fluence at the vessel inner radius of 2.78 E18, 3.16 E19 and 4.75 E19 for Dec. 16, 1991, 32 and 48 EFPY, respectively. The fluence values for 32 and 48 EFPY were developed assuming that uprating from 3411 to 3565 MWt would take place during calendar year 1992, and that calculated design basis neutron flux levels incident on the reactor vessel were applicable over the 32 EFPY design lifetime as well as for 48 EFPY. USE was predicted using the 1/4T fluence values based on the peak fluence at the vessel inner radius. The vessel wall thickness is 8.625 inches at the beltline region. Copper and nickel values for all materials are based on the results of Combustion Engineering chemical analyses. Surveillance capsule material was not used in calculating RTPTS, RTNDT or USE because there has been only one capsule removed from the reactor vessel, hence there is insufficient data at this time.

b. Limiting vessel material.

c. All of the core region welds were fabricated from wire heat 83653. Two Combustion Engineering weld qualifications (CE qualification codes E3.11 and G1.43) were done for welds containing wire heat 83653.

TABLE 5.3.3-3

UNIT 2 REACTOR VESSEL VALUES FOR ANALYSIS OF POTENTIAL PRESSURIZED THERMAL SHOCK EVENTS (a)

Material	Cu wt-%	Ni wt-%	Initial RTNDT (deg. F)	10CFR50.61 Predicted RTPTS (degrees F)			Regulatory Guide 1.99, Rev. 2 RTNDT (deg. F)			Initial USE (ft-lbs)	Regulatory Guide 1.99, R2 Predicted USE (ft-lbs)		
				Dec. 16, 1991	32 EFPY	48 EFPY	Dec. 16, 1991	32 EFPY	48 EFPY		Dec. 16, 1991	32 EFPY	48 EFPY
Intermediate Shell Plate, R4-1	0.06	0.64	10	64	92	96	64	92	96	95	85	74	72
Intermediate Shell Plate, R4-2	0.05	0.62	10	61	84	87	61	84	87	104	93	81	79
Intermediate Shell Plate, R4-3	0.05	0.59	30	81	104	107	81	104	107	84	75	66	64
Lower Shell Plate, B8825-1	0.05	0.59	40	91	114	117	91	114	117	83	74	65	63
Lower Shell Plate, R8-1	0.06	0.62	40	94	122	126	94	122	126	87	77	68	66
Lower Shell Plate, B8628-1	0.05	0.59	50	101	124 (b)	127	101	124	127	85	75	66	65
Core Region Longitudinal Welds (c)	0.07	0.13	-10	71	107	111	71	107	111	152	132	112	109
Intermediate to Lower Shell Girth Weld (c)	0.06	0.12	-30	49	82	86	49	82	86	90	78	67	65

NOTES:

a. RTPTS and RTNDT values are based on the peak fluence at the vessel inner radius of 1.72 E18, 3.17 E19 and 4.76 E19 for Dec. 16, 1991, 32 and 48 EFPY, respectively. The fluence values for 32 and 48 EFPY were developed assuming that uprating from 3411 to 3565 MWt would take place during calendar year 1992, and that calculated design basis neutron flux levels incident on the reactor vessel were applicable over the 32 EFPY design lifetime as well as for 48 EFPY. USE was predicted using the 1/4T fluence values based on the peak fluence at the vessel inner radius. The vessel wall thickness is 8.625 inches at the beltline region. Copper and nickel values for all materials are based on the results of Combustion Engineering chemical analyses. Surveillance capsule material was not used in calculating RTPTS, RTNDT or USE because there has been only one capsule removed from the reactor vessel, hence there is insufficient data at this time.

b. Limiting vessel material.

c. All of the core region welds were fabricated from wire heat 87005. Two Combustion Engineering weld qualifications (CE qualification codes E3.23 and G1.60) were done for welds containing wire heat 87005.