GULF STATES UTILITIES COMPANY

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U.S. Nuclear Regulatory Commission
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Washington, D.C. 20555

Gentlemen:

River Bend Station - Unit 1
Docket No. 50-458

This letter provides Gulf States Utilities Company's response to Generic Letter 92-01, Revision 1 "Reactor Vessel Structural Integrity". This generic correspondence was issued to assess licensee compliance with requirements and commitments regarding reactor vessel integrity. Specifically, this letter evaluates compliance with 10 CFR50.60 and 10 CFR50.61 and licensee commitments made in response to Generic Letter 88-11.

The enclosure to this letter furnishes the information requested in the generic letter required to address this issue for River Bend Station. Each requested item is directly followed by the response in a format comparable to that set forth in the generic letter.

If you have any questions or comments, please contact Mr. Leif L. Dietrich of my staff at (504) 381-4866.

Sincerely,

W.H. Odell

W. H. Odell
Manager - Oversight
River Bend Nuclear Group

WAE/LLD/JRH/WJB/EJZ/WJS

Enclosure

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

STATE OF LOUISIANA)

PARISH OF WEST FELICIANA) Docket No. 50-458

In the Matter of)

GULF STATES UTILITIES COMPANY)

(River Bend Station - Unit 1)

AFFIDAVIT

W. H. Odell, being duly sworn, states that he is a Manager-Oversight for Gulf States Utilities Company; that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission the documents attached hereto; and that all such documents are true and correct to the best of his knowledge, information and belief.

W. H. Odell

W. H. Odell

Subscribed and sworn to before me, a Notary Public in and for the State and Parish above named, this 2nd day of July, 1992. My Commission expires with Life.

Claudia F. Hurst

Claudia F. Hurst
Notary Public in and for
West Feliciana Parish, Louisiana

ENCLOSURE

RESPONSE TO GENERIC LETTER 92-01

BACKGROUND

Generic Letter 92-01, Revision 1 has been issued by the NRC "to obtain information needed to assess compliance with requirements and commitments regarding reactor vessel integrity". This request grew out of NRC efforts to resolve neutron embrittlement concerns for the Yankee Nuclear Power Station reactor vessel. Responses to the NRC requests for information covering three general areas are given below.

ITEM 1.

> Certain addressees are requested to provide the following information regarding Appendix H to 10 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 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).

RESPONSE TO ITEM 1

The reactor vessel materials surveillance program for River Bend Station is in compliance with 10 CFR 50, Appendix H. River Bend Station Unit-1 reactor materials surveillance specimens were provided in accordance with ASTM E 185-73 with exception of limiting weld material as described in the River Bend Station Safety Evaluation Report and outlined below. Details regarding the surveillance program are described in the River Bend Station Updated Safety Analysis Report (RBS USAR) section 5.3.1.6 (reference 1). In its assessment of the River Bend Station surveillance program compliance with 10 CFR 50, Appendix H, the NRC staff stated the following (reference 2):

"On the basis of the information provided by the applicant, the staff has concluded that the applicant's surveillance program contains the limiting base metal and HAZ material, but does not contain the limiting weld material. In accordance with ASTM E 185-73, the limiting weld metal is 5P6756/0342. The weld metal contained in the applicant's surveillance capsule is weld metal 492L4871/A421B27AF.

Although the limiting weld metal is not contained in the applicant's surveillance program, the applicant will be required to determine the effect of neutron irradiation damage on the limiting weld metal using RG 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials". The staff has found that the methods of predicting neutron irradiation damage that are documented in RG 1.99 are conservative. Hence, the use of this guide to predict neutron irradiation damage is an acceptable alternative to testing the limiting weld metal as part of the surveillance program.

Based on its evaluation of compliance with Appendices G and H, 10 CFR 50, the staff concludes that the applicant has met all the fracture toughness requirements of these appendices except for Paragraph II.B of Appendix H. However, the staff will require the applicant to perform a conservative evaluation of the neutron irradiation on the fracture toughness of the reactor beltline materials. The staff considers this evaluation an acceptable alternative to meeting the material surveillance requirements of Paragraph II.B of Appendix H 10 CFR 50."

As required by 10 CFR 50, Appendix H, surveillance specimens removed from the River Bend Station reactor vessel will be tested in accordance with the version of ASTM E 185 in effect at the time of testing.

<u>ITEM 2.</u>

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.

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.

(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 tests;

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

(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;

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

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

(6) the heat number of the wire used for determining the weld metal chemical composition if different than Item (3) above.

RESPONSE TO ITEM 2.a.

The Charpy upper shelf energy (USE) values for RBS reactor vessel beltline materials (see RBS USAR Table 5.3-1 included as Attachment 1 to this response) are predicted to be greater than 50 foot-pounds at the end of 32 Effective Full Power Years (EFPY) of reactor operations when using the guidance of Regulatory Guide 1.99, Revision 2, Paragraph C.1.2 (reference 3). This is based on the initial Charpy USE values given in RBS USAR Table 5.3-1 and an end-of-life peak 1/4 thickness depth (1/4T) neutron fluence of 4.8 x 10^{18} neutrons/square centimeter (reference 4). This conclusion is consistent with that given by the Nuclear Reactor Regulation Staff as given in its safety evaluation related to Amendment No.45 to the RBS operating license (65.6 ft. lbs. for the material with the lowest initial upper shelf energy) (reference 5).

RESPONSE TO ITEM 2.b.

The RBS reactor vessel was designed, fabricated, tested, inspected and stamped in accordance with the 1971 edition of the ASME Boiler and Pressure Vessel Code, Section III, Class I requirements up to and including the Summer 1973 Addenda (reference 6). Information regarding the evaluations of material properties pursuant to 10 CFR 50 , Appendix G is given in RBS USAR Table 5.3-1 (Attachment 1 to this report) and section 5.3.1.5 (reference 7).

ITEM 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.

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

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 energy for each beltline material as predicted for December 16, 1991, and for the end of its current license.

RESPONSE TO ITEM 3.a.

Operation with RBS beltline temperature below 525°F was not considered in the Appendix G analysis because the steady state operating temperature of the coolant in the beltline region is slightly greater. Based on the temperature in the recirculation suction piping, which draws water directly from the beltline region, the steady state temperature in the beltline region is greater than 533°F (reference 8).

Only during startup and operation without feedwater heating, which occurs when feedwater heaters are out of service or when the turbine is off-line and reactor steam is routed through the turbine bypass, does the beltline experience coolant less than 525°F when the core is critical. The effective full power time of operation in these transient conditions has been estimated to be less than 1%, and the associated temperatures for most of that time are 515°F or higher. Thus, the cumulative fluence for operations below 525°F is estimated to be less than $1x10^{17}n/cm^2$.

The combination of low fluence and small deviation from the 525°F level is not expected to significantly affect beltline reference temperature for nil-ductility transition (RT-NDT) or upper shelf energy predictions. Furthermore, irradiation temperature effects, if any, will influence surveillance test results since the surveillance specimens are also exposed to this temperature environment. Thus, any influence of temperature will be accounted for automatically when surveillance results are evaluated to 10 CFR 50, Appendix G requirements.

RESPONSE TO ITEM 3.b.

As reported in item 1 above, no surveillance specimens have been removed from River Bend Station-Unit 1 as of this reporting date. The first surveillance capsule is scheduled for removal at 6 EFPY of reactor operation. Removal of the first capsule is expected to occur during the fifth refueling outage in 1994. Therefore, surveillance results have not yet been considered in neutron embrittlement predictions.

RESPONSE TO ITEM 3.c.

This item is not applicable to River Bend Station because, as reported above, no surveillance specimens have been removed from the RBS vessel.

REFERENCES:

1. River Bend Station Updated Safety Analysis Report section 5.3.1.6, page 5.3-6.

2. Safety Evaluation Report Related to the Operation of River Bend Station, NUREG-0989, pages 5-15,16.

3. Gulf States Utilities Company letter to U.S. Nuclear Regulatory Commission dated 11-03-88 (RBG-29292, GSU response to Generic Letter 88-11.)

4. GSU letter to the NRC dated 05/14/90 (RBG-32835), Attachment 1--GE Report SASR 89-20, Rev. 1, page 2-3. (GSU request for change of Tech. Spec. 3/4.4.6; subsequently approved as Amendment 45.)

5. NRC letter to GSU dated 08-01-90, regarding Amendment 45 to RBS operating license, Safety Evaluation, page 3.

6. RBS USAR, section 5.3.3.1.1.1, page 5.3-15.

7. RBS USAR, section 5.3.1.5, pages 5.3-3 through 5.3-5.

8. RBS USAR, Figure 1.1-1 and steam tables.

ATTACHMENT 1

RBS USAR

TABLE 5.3-1

RIVER BEND STATION UNIT 1 REACTOR VESSEL CHARPY TEST RESULTS
VESSEL BELTLINE CHEMICAL COMPOSITION AND EMBRITTLEMENT EFFECTS

1) Vessel Plate (Beltline)

Heat Number	C	Mn	Si	P	S	Ni (Percent)	Cu (Percent)	Mo	V
C3138-2	0.19	1.37	0.25	0.012	0.015	0.63	0.08	0.58	-
C3054-1	0.19	1.30	0.26	0.007	0.020	0.70	0.09	0.57	-
*C3054-2	0.19	1.30	0.26	0.007	0.012	0.70	0.09	0.57	-

2) Vessel Welds (Beltline)

Heat/Lot No.	C	Mn	Si	P	S	Ni (Percent)	Cu (Percent)	Mo	V
492L4871/ A421B27AE	0.07	1.06	0.37	0.018	0.025	0.95	0.04	0.50	0.02
*492L4871/ A421B27AF	0.07	1.17	0.32	0.020	0.020	0.98	0.03	0.51	0.02
5P6756/0342(1)	0.078	1.24	0.53	0.010	0.012	0.92	0.09	0.46	0.006
5P6756/0342(2)	0.063	1.27	0.57	0.010	0.011	0.93	0.09	0.45	0.006

1) Vessel Plate (Beltline)
→4

Heat Number	Start RT_{NDT} °F	R.G. 1.99, Rev. 2 Extrap. RT_{NDT} °F	32 EFPY RT_{NDT} °F	Transverse Charpy Upper Shelf (ft-lb)
C3138-2	+9	75	84	86, 74, 78
C3054-1	-20	80	60	94, 93, 93
*C3054-2	+2	80	82	92, 102, 92

2) Vessel Welds (Beltline)

Heat/Lot No.	Start RT_{NDT} °F	R.G. 1.99, Rev. 2 Extrap. RT_{NDT} °F	32 EFPY RT_{NDT} °F	Transverse Charpy Upper Shelf (ft-lb)
492L4871/ A421B27AE	-60	86	26	151, 160, 161
*492L4871/ A421B27AF	-50	65	15	126, 129, 136
5P6756/0342(1)	-50	153	103	95, 99, 96
5P6756/0342(2)	-60	153	93	89, 94, 91

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*Selected for reactor vessel test specimen.
(1) Tandem wire process
(2) Single wire process