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Director
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Waterford 3

W3F192-0094
A4.05
QA

July 6, 1992

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

Subject: Waterford 3 SES
Docket No. 50-382
License No. NPF-38
Generic Letter 92-01, Revision 1, Response

Gentlemen:

In accordance with Generic Letter 92-01, Revision 1, "Reactor Vessel Structural Integrity," attached is the Entergy Operations, Inc. Waterford 3 response.

Please contact me or Robert J. Murillo should there be any questions regarding this response.

Very truly yours,

R.F. Burski

RFB/RJM/ssf
Attachment
cc: R.D. Martin, NRC Region IV
D.L. Wigginton, NRC-NRR
R.B. McGehee
N.S. Reynolds
NRC Resident Inspectors Office

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the matter of	)	
	)	
Entergy Operations, Incorporated	)	Docket No. 50-382
Waterford 3 Steam Electric Station	)	

AFFIDAVIT

R.F. Burski, being duly sworn, hereby deposes and says that he is Director Nuclear Safety - Waterford 3 of Entergy Operations, Incorporated; that he is duly authorized to sign and file with the Nuclear Regulatory Commission the attached response to Revision 1 of Generic Letter 92-01; that he is familiar with the content thereof; and that the matters set forth therein are true and correct to the best of his knowledge, information and belief.

R.F. Burski

R.F. Burski
Director Nuclear Safety - Waterford 3

STATE OF LOUISIANA	)	
	)	ss
PARISH OF ST. CHARLES	)	

Subscribed and sworn to before me, a Notary Public in and for the Parish and State above named this 6th day of July, 1992.

Steve E. Fahy

Notary Public

My Commission expires WITH LIFE.

Response to NRC Generic Letter 92-01 for Waterford 3

Question 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 E185-73, -79, or -82 and who do not have an integrated surveillance program approved by the NRC (see Enclosure 2 of GL 92-01), are requested to describe actions taken or to be taken to ensure compliance with Appendix H to 10CFR 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 Question 1

The Waterford 3 surveillance program was designed to meet the requirements of ASTM E-185-73. Because the reactor vessel was fabricated to Section III of the ASME Boiler and Pressure Vessel Code 1971 Edition through the Summer 1971 Addenda, there were two exceptions:

(1) plate selection for the surveillance program was based on longitudinal data, and
(2) the surveillance capsule assembly is attached to the vessel wall.

The NRC Staff concluded in its Supplemental Safety Evaluation Report (SSER) #1[1] that the surveillance program meets all the requirements of 10 CFR 50 Appendix H, with the exception of Paragraph II.B. The exception is that the limiting weld material, as determined by the Staff, is not the weld metal included in the surveillance program. However, in SSER #1 the Staff also concluded that an exemption from Paragraph II.B was justified, as follows:

> Waterford 3 would calculate the adjusted reference temperature (ART_{NDT}) for the weld metal based on the greater of the measured shift in RT_{NDT} as determined by impact testing of the surveillance weld metal and the predicted shift in RT_{NDT} as determined by Regulatory Guide 1.99 for the Staff-determined limiting weld material.

Subsequent to the Staff's SSER #1, Appendix H to 10 CFR 50 was revised, effective July 26, 1983. As a result of the changes, the Staff concluded in SSER #8[2] that the exemption was no longer required and that the Waterford 3 surveillance program complies

with the revised Appendix H requirements.

In addition to the above, it should be noted that Waterford 3 is committed to the 1982 revision of ASTM E-185, as documented in References (3) and (4).

Question 2.a

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

> 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 Paragraph 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.

Response to Question 2.a

The Charpy upper shelf energy (USE), as predicted in accordance with Regulatory Guide 1.99 Revision 2, does not fall below 50 ft lbs at the end of design life. In their SER related to Generic Letter 88-11 response[5], the NRC documented their conclusion that a conservative prediction of end-of-life Charpy USE is above the 50 ft-lb minimum required by Appendix G. Therefore, no further action is required.

Question 2.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 considerations given to the following material properties in their evaluations performed pursuant to 10CFR 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, and the unirradiated reference temperature for each beltline material, and the method for 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 heat number of wire and flux lot number used to fabricate the surveillance 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 and chemical composition if different than Item (3).

Response to Question 2.b

The Waterford 3 reactor vessel meets the fracture toughness requirements in effect through the Summer 1971 Addenda to the ASME Code. The longitudinal Charpy impact tests were satisfactorily performed, and Charpy V-notch results were delivered to the NRC with Reference (6). Materials to perform additional testing required by the Summer 1972 Addenda and 10 CFR 50 Appendix G were not available.

The available test data were evaluated according to BTP MTEB 5-2, "Fracture Toughness Requirements", as recommended by the NRC Staff in a December 1974 meeting. The results of baseline testing of the surveillance material demonstrated wide margin of conservatism in the evaluation. This information was also included in the Reference (6) submittal to the NRC.

The Staff summarized in its SSER #1, its evaluation of the degree of compliance for Waterford 3 with the fracture toughness requirements of 10 CFR 50 Appendix G. The Staff concluded in relevant part that all requirements of Appendix G were met, with the exception of Paragraphs II.B.1, III.C.1, III.C.2 and IV.A.1, for which sufficient information was supplied to grant exemptions. Additional information was also supplied to the NRC which was sufficient to demonstrate compliance with the requirements of Paragraphs III.B.3, IV.A.3, and IV.B of Appendix G, Part 50[1].

Subsequent to SSER #1, Appendix G was revised, effective July 26, 1983. With that revision, the Staff concluded that, because fracture toughness properties for the reactor coolant pressure boundary materials had been demonstrated equivalent to the requirements of Appendix G, the exemptions were no longer required. This conclusion was documented in their SSER #8[2].

2.b.1 The drop weight test results, unirradiated reference temperature for each beltline material, and the method for determining the unirradiated reference temperature from the Charpy and drop weight test, are noted in Table 1.

2.b.2 The nominal heat treatment process received by test and surveillance program materials is summarized below:

1. 1600°F for 4 hours. Water Quenched. (Austenitizing)
2. 1225°F for 4 hours. (Tempering)
3. 1150°F for 40 hours. Furnace Cooled to 600°F. (Post-weld stress relief)

2.b.3 Beltline material source is as follows:

Plate or Weld No.	Heat No.	Flux Lot No.
M-1003-1	56488-1	N/A
M-1003-2	56512-1	N/A
M-1003-3	56484-1	N/A
M-1004-1	57326-1	N/A
M-1004-2	57286-1	N/A
M-1004-3	57359-1	N/A
101-124 A,B,C	Type 8018 electrode CE Lot No. BOLA, HODA	N/A
101-142 A,B,C	MIL B-4 wire Heat No. 83653	0091 flux Lot No. 3536
101-171	MIL B-4 wire Heat No. 88114	0091 flux Lot No. 0145

2.b.4 Surveillance material source is as follows:

- Base metal is from plate M-1004-2 (lower shell), adjacent to ASME Code Section III test material and at least one-plate-thickness distant from any water-quenched edge.[7]

- Weld metal and HAZ material were produced by welding together sections of the selected base metal and another lower shell plate of the vessel. The HAZ test material was manufactured from a section of the same base metal plate used for base metal test material. The sections used for weld metal and HAZ test material are adjacent to the ASME Code Section III test materials at a distance of at least one plate thickness from any water-quenched edge.[8]

- The welding materials were the same as that used for weld No. 101-171, i.e., MIL B-4 weld wire Heat No. 88114 with Linde 0091 flux Lot No. 0145.

2.b.5 Chemical composition of beltline plates and welds are in Table 1.

2.b.6 Not applicable to Waterford 3.

Question 3.a

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

How the temperature 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.

Response to Question 3.a

The reactor has not been operated with a cold leg temperature (T-cold) less than 525°F for a significant period of time.

The Waterford 3 Technical Specifications (3.2.6) require that, in Mode 1 above 30% of rated thermal power, T-cold be maintained between 544°F and 558°F. Technical Specification 3.1.1.4 requires that the lowest operating loop temperature (T-cold) be greater than or equal to 520°F with the reactor critical. The Action Statement allows 15 minutes to restore T-cold within limits should T-cold fall below 520°F, or be in Mode 3 within the next 15 minutes. Surveillance Requirement 4.1.1.4 requires that T-cold be verified greater than or equal to 520°F at least once per 30 minutes when the reactor is critical and T-cold is less than 530°F.

A review of LER logs showed that no violation of T-cold limits for criticality has ever been reported.

Operating procedures OP-10-001 and OP-903-001 provide controls to ensure that limits on T-cold are maintained. With reactor power greater than 30%, procedures require that T-cold is maintained between 544°F and 558°F. During startup, procedure OP-10-001 requires that the lowest T-cold be verified greater than or equal to 520°F every 15 minutes until criticality is achieved, and at least every 30 minutes with the reactor critical and RCS T-cold less than 530°F. Between critical and 30%, the Technical Specification Surveillance Logs for Modes 1-4 note limits for T-cold; 544°F-558°F in Mode 1 above 30%, and greater than 530°F in Modes 1 and 2.

The plant is thus controlled by the operators such that core-critical operation with T-cold less than 525°F is an abnormal or transient condition, and therefore a temporary condition. The reactor vessel has thus not been operated with a cold leg temperature less than 525°F for a significant period of time.

Conservatism is built into the Adjusted Reference Temperature because the projected fluence of 3.68E19 in the FSAR is based on a high-leakage core, and the plant has operated since the first refueling with a low-leakage core. The first surveillance capsule was withdrawn at the last refueling and is presently being evaluated. The results of the capsule evaluation should provide clarification as to whether there have been any adverse effects from normal plant operating practices. Capsule evaluation results will be submitted to the NRC by November 30, 1992 in accordance with Waterford 3 letter W3F192-0052.

Question 3.b

Utilities are requested to provide information on how their surveillance results have been used in response to GL 88-11 and how the predicted amount of embrittlement has been considered.

Response to Question 3.b

Credible surveillance measurements are not yet available using Regulatory Position 2.1 of Regulatory Guide 1.99 Revision 2. Testing of the first capsule withdrawal has not yet been completed. Waterford 3 requested an extension for submittal of the summary report with Reference (9).

Question 3.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 its current license.

Response to Question 3.c

See response to question 3.b.

References

(1) Supplemental Safety Evaluation Report No. 1, by the NRC Staff, page 5-9, NUREG-0787, dated October 1981.

(2) Supplemental Safety Evaluation Report No. 8, by the NRC Staff, page 5-1 NUREG-0787, dated December 1984.

(3) LP&L letter to NRC, W3P87-1578, dated July 31, 1987.

(4) NRC Staff SER Supporting Change to Bases Section of Technical Specifications, dated 8/20/87.

(5) NRC Staff SER Related to Generic Letter 88-11 Response, dated 11/27/89.

(6) LP&L letter to NRC, LPL8254, dated February 24, 1978.

(7) Waterford 3 FSAR, Section 5.3.1.6.1.1

(8) Waterford 3 FSAR, Section 5.3.1.6.1.2

(9) Entergy letter W3F192-0052, to Mr. Thomas Murley, NRC, dated April 2, 1992.

TABLE 1

WATERFORD UNIT 3
REACTOR VESSEL MATERIALS

Product Form	Material Identification	Drop Weight NDDT (°F)	Initial[d] RT_{NDT} (°F)	Charpy Specimen Orientation	Chemical Content (%) Nickel	Copper	Phosphorous	Sulfur
Plate	M-1003-1	-30	-30	longitudinal	0.71	0.02	0.004	0.010
Plate	M-1003-2	-50	-50	longitudinal	0.67	0.02	0.006	0.007
Plate	M-1003-3	-50	-42	longitudinal	0.70	0.02	0.007	0.009
Plate	M-1004-1	-50	-15	longitudinal	0.62	0.03	0.006	0.008
Plate	M-1004-2	-20	22	longitudinal	0.58	0.03	0.005	0.005
Plate	M-1004-3	-50	-10	longitudinal	0.62	0.03	0.007	0.007
Weld	101-124 A,B,C[a]	-60	-60	N/A	0.96	0.02	0.010	0.016
Weld	101-142 A,B,C[b]	-80	-80	N/A	<0.20	0.03	0.007	0.009
Weld	101-171[c]	-70	-70	N/A	0.16	0.05	0.008	0.008

(a) Intermediate shell course longitudinal seam weld
(b) Lower shell course longitudinal seam weld
(c) Intermediate - lower shell girth weld
(d) Plate RT_{NDT} determined using Branch Technical Position MTEB 5-2;
Weld RT_{NDT} determined in accordance with ASME Code, Section III, NB-2300