Docket file

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 11, 1994

Docket No. 50-289

Mr. T. Gary Broughton, Vice President
and Director - TMI-1
GPU Nuclear Corporation
Post Office Box 480
Middletown, Pennsylvania 17057

Dear Mr. Broughton:

SUBJECT: GENERIC LETTER (GL) 92-01, REVISION 1, "REACTOR VESSEL STRUCTURAL INTEGRITY," THREE MILE ISLAND NUCLEAR STATION, UNIT 1 (TAC NO. M83741)

By letters dated July 2, 1992, and December 13, 1993, GPU Nuclear Corporation provided its response to GL 92-01, Revision 1 for Three Mile Island, Unit 1 (TMI-1). The NRC staff has completed its review of your responses. Based on its review, the staff has determined that GPU Nuclear has provided the information requested in GL 92-01.

The GL is part of the staff's program to evaluate reactor vessel integrity for Pressurized Water Reactors (PWRs) and Boiling Water Reactors (BWRs). The information provided in response to GL 92-01, including previously docketed information, is being used to confirm that licensees satisfy the requirements and commitments necessary to ensure reactor vessel integrity for their facilities.

A substantial amount of information was provided in response to GL 92-01, Revision 1. These data have been entered into a computerized data base designated Reactor Vessel Integrity Database (RVID). The RVID contains the following tables: A pressurized thermal shock (PTS) table for PWRs, a pressure-temperature limit table for BWRs, and an upper-shelf energy (USE) table for PWRs and BWRs. Enclosure 1 provides the PTS table, Enclosure 2 provides the USE table for your facility, and Enclosure 3 provides a key for the nomenclature used in the tables. The tables include the data necessary to perform USE and RT_{pts} evaluations. These data were taken from your responses to GL 92-01 and previously docketed information. References to the specific source of the data are provided in the tables.

We request that, within 30 days, you provide confirmation of the plant-specific applicability of the Babcock & Wilcox topical reports BAW-2178P and BAW-2192P and submit a request for approval of the topical reports as the basis for demonstrating compliance with 10 CFR Part 50, Appendix G, Paragraph IV.A.1. To demonstrate that the topical reports are applicable to TMI-1, you must compare the limiting material properties of the TMI-1 reactor vessel to the values reported in the topical reports.

In addition, we have determined that additional data is required to confirm that the USE at end-of-life (EOL) is greater than 50 ft-lb because you have provided a generic mean value for the unirradiated USE for two plates. This value does not consider heat variability. When the unirradiated USE for a particular heat of plate has not been determined, you can determine the lower tolerance limit with 95 percent confidence that at least 95 percent of the population is greater than the tolerance limit. If the lower tolerance limit results in a projected USE at EOL of less than 50 ft-lb, then you must demonstrate, in accordance with Appendix G, 10 CFR Part 50, that lower values of USE will provide margins of safety against fracture equivalent to those required by Appendix G of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code.

We request that you submit within 30 days a schedule for performing these analyses. Further, we request that you verify that the information you have provided for your facility has been accurately entered in the summary data file. If no comments are made in your response to the last request, the staff will use the information in the tables for future NRC assessments of your reactor pressure vessel. Once your response is received and your schedule is determined to be satisfactory, the staff will consider your actions related to GL 92-01, Revision 1, to be complete. When your analyses are submitted, they will be reviewed as a plant-specific licensing action.

The information requested by this letter is within the scope of the overall burden estimated in GL 92-01, Revision 1, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f)." The estimated average number of burden hours is 200 person hours for each addressee's response. This estimate pertains only to the identified response-related matters and does not include the time required to implement actions required by the regulations. This action is covered by the Office of Management and Budget Clearance Number 3150-0011, which expires June 30, 1994.

Sincerely,

Original signed by:

Ronald W. Hernan, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Pressurized Thermal Shock Table
2. Upper-Shelf Energy Table
3. Nomenclature Key

cc w/enclosures:
See next page

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BElliot	DMcDonald				

OFFICE	LA:PDI-4	PM:PDI-4	PD:PDI-4		
NAME	SNorris	RHernan:bp	JStolz		
DATE	4/11/94	4/11/94	4/11/94	/ /	/ /

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In addition, we have determined that additional data is required to confirm that the USE at end-of-life (EOL) is greater than 50 ft-lb because you have provided a generic mean value for the unirradiated USE for two plates. This value does not consider heat variability. When the unirradiated USE for a particular heat of plate has not been determined, you can determine the lower tolerance limit with 95 percent confidence that at least 95 percent of the population is greater than the tolerance limit. If the lower tolerance limit results in a projected USE at EOL of less than 50 ft-lb, then you must demonstrate, in accordance with Appendix G, 10 CFR Part 50, that lower values of USE will provide margins of safety against fracture equivalent to those required by Appendix G of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code.

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Sincerely,

Ronald W. Hernan

Ronald W. Hernan, Sr. Project Manager
Project Directorate 1-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Pressurized Thermal Shock Table
2. Upper-Shelf Energy Table
3. Nomenclature Key

cc w/enclosures:
See next page

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GPU Nuclear Corporation

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Unit No. 1

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Summary File for Pressurized Thermal Shock

Plant Name	Beltline Ident.	Heat No. Ident.	ID Neut. Fluence at EOL/EFPY	IRT_{ndt}	Method of Determin. IRT_{ndt}	Chemistry Factor	Method of Determin. CF	%Cu	%Ni
Three Mile Island 1 EOL: 4/19/2014	Lower Nozzle Belt	ARY 059	0.66E19	10°F	MTEB 5-2	51	Table	0.08	0.72
	Upper Shell	C2789-1	0.75E19	30°F	MTEB 5-2	58	Table	0.09	0.57
	Upper Shell	C2789-2	0.75E19	30°F	MTEB 5-2	25.266	Calculated	0.09	0.57
	Lower Shell	C3307-1	0.72E19	30°F	MTEB 5-2	82	Table	0.12	0.55
	Lower Shell	C3251-1	0.72E19	30°F	MTEB 5-2	73	Table	0.11	0.5
	Nozzle Belt/Upper Shell Circ. Weld WF-70	72105	0.66E19	-26°F	NRC Generic	148.02	Calculated	0.35	0.59
	Upper/ Lower Shell Circ. Weld WF-25	299L44	0.72E19	-5°F	Generic	221.25	Calculated	0.35	0.68
	Lower Shell/ Dutchman Circ. Weld WF-67	72442	0.004E19	-5°F	Generic	173	Table	0.24	0.6
	Upper Shell Axial Welds WF-8	8T1762	0.75E19	-5°F	Generic	152.25	Table	0.2	0.55
	Lower Axial Welds SA-1526	299L44	0.65E19	-5°F	Generic	221.25	Calculated	0.35	0.68
	Upper/ Lower Shell Circ. Weld Atypical	72105	0.66E19	90°F	Plant Specific	123.41	Calculated	0.41	0.1

Reference for TMI-1

Fluence, IRT_{ndt}, and chemical composition data are from BAW-2166.

IRT_{ndt} for WF-70 weld was determined from an approved method in a February 22, 1994 letter from C.Y. Shiraki (USNRC) to D.L. Farrar (Comm. Ed. Co.). In order to use this value of IRT_{ndt}, the licensee will need an exemption from determining the IRT_{ndt} in accordance with 10 CFR 50.61 (b)(2)(i).

Chemistry factors for WF-70, WF-25 and SA-1526, and Atypical welds were determined from the surveillance programs of the following plants:

WF-70; Oconee 2, and Davis-Besse,
WF-25 and SA-1529; Surry 1 and TMI-1,
Atypical weld; Crystal River 3.

Summary File for Upper Shelf Energy

Plant Name	Beltline Ident.	Heat No.	Material Type	1/4T USE at EOL/EFPY	1/4T Neutron Fluence at EOL/EFPY	Unirrad. USE	Method of Determin. Unirrad. USE
Three Mile Island 1 EOL: 4/19/2014	Lower Nozzle Belt	ARY 059	A 508-2	64	0.396E19	75	10°F Data
	Upper Shell	C2789-1	A 302B Code Case 1339	63	0.45E19	75 [7]	Generic
	Upper Shell	C2789-2	A 302B Code Case 1339	82	0.45E19	98	Direct
	Lower Shell	C3307-1	A 302B Code Case 1339	93	0.432E19	112	Direct
	Lower Shell	C3251-1	A 302B Code Case 1339	62	0.432E19	75 [7]	Generic
	Nozzle Belt/Upper Shell Circ. Weld WF-70	72105	Linde 80 SAW	EMA[2]	0.396E19	EMA[2]	---
	Upper/ Lower Shell Circ. Weld WF-25	299L44	Linde 80 SAW	EMA[2]	0.432E19	EMA[2]	---
	Lower Shell/ Dutchman Circ. Weld WF-67	72442	Linde 80 SAW	EMA[2]	0.003E19	EMA[2]	---
	Upper Shell Axial Welds WF-8	8T1762	Linde 80 SAW	EMA[2]	0.45E19	EMA[2]	---
	Lower Axial Welds SA-1526	299L44	Linde 80 SAW	EMA[2]	0.39E19	EMA[2]	---

[7]Additional information required to confirm value.

[2]Licensee must confirm applicability of Topical Reports BAW-2178P and BAW-2192P.

Summary File for Upper Shelf Energy

Plant Name	Beltline Ident.	Heat No.	Material Type	1/4T USE at EOL/EFPY	1/4T Neutron Fluence at EOL/EFPY	Unirrad. USE	Method of Determin. Unirrad. USE
	Upper/ Lower Shell Circ. Weld Atypical	72105	Linde 80 SAW	EMA[2]	0.396E19	EMA[2]	---

References

The UUSE data for plates C2789-2 and C3307-1, and forging ARY 059 and materials types for all plates and forging are from BAW-1820.

Fluence and chemical composition data are from BAW-2166.

The UUSE data for plates C2789-1 and C3251-1 are from BAW 1895.

Nomenclature and Tables

PRESSURIZED THERMAL SHOCK TABLES AND USE TABLES FOR ALL PWR PLANTS

NOMENCLATURE

Pressurized Thermal Shock Table

Column 1: Plant name and date of expiration of license.
Column 2: Beltline material location identification.
Column 3: Beltline material heat number; for some welds that a single-wire or tandem-wire process has been reported, (S) indicates single wire was used in the SAW process, (T) indicates tandem wire was used in the SAW process.
Column 4: End-of-life (EOL) neutron fluence at vessel inner wall; cited directly from inner diameter (ID) value or calculated by using Regulatory Guide (RG) 1.99, Revision 2, neutron fluence attenuation methodology from the quarter thickness (T/4) value reported in the latest submittal (GL 92-01, PTS, or P/T limits submittals).
Column 5: Unirradiated reference temperature.
Column 6: Method of determining unirradiated reference temperature (IRT).

Plant-Specific
This indicates that the IRT was determined from tests on material removed from the same heat of the beltline material.

MTEB 5-2
This indicates that the unirradiated reference temperature was determined from following MTEB 5-2 guidelines for cases where the IRT was not determined using American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, NB-2331, methodology.

Generic
This indicates that the unirradiated reference temperature was determined from the mean value of tests on material of similar types.

Column 7: Chemistry factor for irradiated reference temperature evaluation.
Column 8: Method of determining chemistry factor.

Table
This indicates that the chemistry factor was determined from the chemistry factor tables in RG 1.99, Revision 2.

Calculated
This indicates that the chemistry factor was determined from surveillance data via procedures described in RG 1.99, Revision 2.

Column 9: Copper content; cited directly from licensee value except when more than one value was reported. (Staff used the average value in the latter case.)

No Data
This indicates that no copper data has been reported and the default value in RG 1.99, Revision 2, will be used by the staff.

Column 10: Nickel content; cited directly from licensee value except when more than one value was reported. (Staff used the average value in the latter case.)

No Data

This indicates that no nickel data has been reported and the default value in RG 1.99, Revision 2, will be used by the staff.

Upper Shelf Energy Table

Column 1: Plant name and date of expiration of license.
Column 2: Beltline material location identification.
Column 3: Beltline material heat number; for some welds that a single-wire or tandem-wire process has been reported, (S) indicates single wire was used in the SAW process. (T) indicates tandem wire was used in the SAW process.
Column 4: Material type; plate types include A 533B-1, A 302B, A 302B Mod., and forging A 508-2; weld types include SAW welds using Linde 80, 0091, 124, 1092, ARCOS-B5 flux, Rotterdam welds using Graw Lo, SMIT 89, LW 320, and SAF 89 flux, and SMAW welds using no flux.
Column 5: EOL upper-shelf energy (USE) at T/4; calculated by using the EOL fluence and either the cooper value or the surveillance data. (Both methods are described in RG 1.99, Revision 2.)

EMA
This indicates that the USE issue may be covered by the approved equivalent margins analysis in a topical report.

Column 6: EOL neutron fluence at T/4 from vessel inner wall; cited directly from T/4 value or calculated by using RG 1.99, Revision 2, neutron fluence attenuation methodology from the ID value reported in the latest submittal (GL 92-01, PTS, or P/T limits submittals).

Enclosure 3 (Cont.)

Column 7: Unirradiated USE.

EMA
This indicates that the USE issue may be covered by the approved equivalent margins analysis in a topical report.

Column 8: Method of determining unirradiated USE.

Direct
For plates, this indicates that the unirradiated USE was from a transverse specimen. For welds, this indicates that the unirradiated USE was from test date.

65%
This indicates that the unirradiated USE was 65% of the USE from a longitudinal specimen.

Generic
This indicates that the unirradiated USE was reported by the licensee from other plants with similar materials to the beltline material.

NRC generic
This indicates that the unirradiated USE was derived by the staff from other plants with similar materials to the beltline material.

10, 30, 40, or 50 °F
This indicates that the unirradiated USE was derived from Charpy test conducted at 10, 30, 40, or 50 °F.

Surv. Weld
This indicates that the unirradiated USE was from the surveillance weld having the same weld wire heat number.

Equiv. to Surv. Weld
This indicates that the unirradiated USE was from the surveillance weld having different weld wire heat number.

Sister Plant
This indicates that the unirradiated USE was derived by using the reported value from other plants with the same weld wire heat number.

Blank
Indicates that there is insufficient data to determine the unirradiated USE.