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

April 6, 1994

Docket No. 50-354

Mr. Steven E. Miltenberger
Vice President and Chief Nuclear
Officer
Public Service Electric an as
Company
Post Office Box 236
Hancocks Bridge, New Jersey 08038

Dear Mr. Miltenberger:

SUBJECT: GENERIC LETTER (GL) 92-01, REVISION 1, "REACTOR VESSEL STRUCTURAL INTEGRITY," PUBLIC SERVICE ELECTRIC AND GAS COMPANY, HOPE CREEK GENERATING STATION (TAC NO. M83471)

By letters dated June 30, 1992, September 3, 1992, and July 20, 1993, Public Service Electric and Gas Company (PSE&G), provided its response to GL 92-01, Revision 1. The NRC staff has completed its review of your responses. Based on its review, the staff has determined that PSE&G 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 BWR's. Enclosure 1 provides the PTS and/or pressure temperature 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, pressure-temperature limit, and RT_{pts} evaluations. These data were taken from your responses to GL 92-01 and previously docketed information. The information in the RVID for your facility will be considered accurate at this point in time and will be used in the staff's assessments related to vessel structural integrity. References to the specific source of the data are provided in the tables.

As a result of our GL 92-01 review, the NRC staff has identified one open issue for your plant. The initial RT_{NDT} values determined by General Electric's (GE) initial methodology have not been validated and the BWR Owners Group report, GE-NE-523-109-0893, entitled, "Basis for GE RT_{NDT} Estimation Method," did not resolve the issue. GE is in the process of validating its

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methodology for resolving the initial RT_{NDT} determination issue and will document the results in a topical report. The BWR Owners Group is obtaining approval from its members to provide the GE topical report to the NRC staff for its review and approval. We request that you submit, within 30 days of receipt of this letter, a commitment to the BWR Owners Group effort or a schedule for a plant-specific analysis to resolve this issue. Further, we request that you provide confirmation of the plant-specific applicability of the topical report, NEDO-32205, Revision 1, (as specified in Appendix B of that report) and submit a request for approval of the topical report as the basis for demonstrating compliance with 10 CFR Part 50, Appendix G, Paragraph IV.A.1 We further request that you verify that the information you have provided for your facility has been accurately entered in the data base. 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 you have (1) confirmed the applicability of the topical report, NEDO-32205, Revision 1, to your plant, (2) submitted the request for approval, and (3) provided your commitment to the BWR Owners Group effort or a satisfactory schedule for providing a plant-specific analysis, the staff will consider your actions related to GL 92-01, Revision 1, to be complete. Plant-specific licensing actions will be initiated to resolve these issues.

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,
/S/
James C. Stone, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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

cc w/enclosures:
See next page

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James C. Stone

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cc w/enclosures:
See next page

Mr. Steven E. Miltenberger
Public Service Electric & Gas
Company

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Summary File for Pressure-Temperature Limits

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
Hope Creek EOL: 4/11/2026	Int. Shell	5K3025-1	3.4E17	19°F	Plant Specific	112.75	Table	0.15	0.71
	Int. Shell	5K2608-1	3.4E17		Plant Specific	58	Table	0.09	0.58
	Int. Shell	5K2698-1	3.4E17	19°F	Plant Specific	65	Table	0.10	0.58
	Lower Int. Shell	5K2963-1-2	1.59E18	-10°F	Plant Specific	44	Table	0.07	0.58
	Lower Int. Shell	5K2530-1-2	1.59E18	19°F [1]	Plant Specific	51	Table	0.08	0.56
	Lower Int. Shell	5K3238-1-2	1.59E18	7°F [1]	Plant Specific	58	Table	0.09	0.63
	Lower Shell	5K3230-1-2	1.59E18	-10°F	Plant Specific	44	Table	0.07	0.56
	Lower Shell	6C35-1-2	1.59E18	-11°F [1]	Plant Specific	58	Table	0.09	0.54
	Lower Shell	6C45-1-2	1.59E18	1°F [1]	Plant Specific	51	Table	0.08	0.57
	Axial Welds for Lower Shell	D53040	1.59E18	-30°F	Plant Specific	106.3	Table	0.08	0.63
	Circ. Weld between Lower-Int. & Lower Shells	D53040	1.59E18	-30°F	Plant Specific	106.3	Table	0.08	0.63
	Circ. Weld between Int. and Lower-Int. Shells	D55733	3.4E17	-40°F	Plant Specific	126.4	Table	0.10	0.68
	Axial Welds for Int. Shell	D53040	3.4E17	-49°F	Plant Specific	126.4	Table	0.10	0.68
	LPCI Nozzle Welds	001-01205	2.3E17	-40°F	Plant specific	27	Table	0.02	0.51
		504-01205	2.3E17	-31°F	Plant Specific	20	Table	0.01	0.51
		519-01205	2.3E17	-49°F	Plant Specific	20	Table	0.01	0.53

<u>Reference for Hope Creek</u>

Fluence, chemical composition, and IRT data are from June 30, 1992, letter from S. Miltenberger (PSEG) to USNRC Document Control Desk, subject: Response to Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity, 10 CFR 50.54(f)

NOTE: LPCI nozzles have eight SMAW welds that were made of wires of three heat numbers.

[1]Additional information required to confirm value.

10 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
Hope Creek EOL: 4/11/2026	Int. Shell	5K3025-1	A 533B-1	EMA[2]	2.4E17	EMA[2]	
	Int. Shell	5K2608-1	A 533B-1	EMA[2]	2.4E17	EMA[2]	
	Int. Shell	5K2698-1	A 533B-1	EMA[2]	2.4E17	EMA[2]	
	Lower Int. Shell	5K2963-1-2	A 533B-1	67	1.1E18	75	Direct
	Lower Int. Shell	5K2530-1-2	A 533B-1	67	1.1E18	75	Direct
	Lower Int. Shell	5K3238-1-2	A 533B-1	EMA[2]	1.1E18	EMA[2]	
	Lower Shell	5K3230-1-2	A 533B-1	EMA[2]	1.1E18	EMA[2]	
	Lower Shell	6C35-1-2	A 533B-1	EMA[2]	1.1E18	EMA[2]	
	Lower Shell	6C45-1-2	A 533B-1	67	1.1E18	75	Direct
	Axial Welds for Lower Shell	D53040	Flux type unknown, SAW	117	1.1E18	135	Surv. Weld
	Circ. Weld between Lower-Int. & Lower Shells	D53040	Flux type unknown, SAW	117	1.1E18	135	Surv. Weld
	Circ. Weld between Int. and Lower-Int. Shells	D55733	Flux type unknown, SAW	61	2.4E17	68	10°F data
	Axial Welds for Int. Shell	D53040	Flux type unknown, SAW	122	2.4E17	135	Surv. Weld
	LPCI Nozzle Welds	504-01205 001-01205 579-01205	SMAW	EMA[2]	1.6E17	EMA[2]	

<u>Reference for Hope Creek</u>

Fluence, chemical composition, and IRT data are from June 30, 1992, letter from S. Miltenberger (PSEG) to USNRC Document Control Desk, subject: Response to Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity, 10 CFR 50.54(f)

NOTE: LPCI nozzles have eight SMAW welds that were fabricated using weld wires of three heat numbers.

[2]Licensee must confirm applicability of Topical Report NEDO-32205, Rev. 1

PRESSURE-TEMPERATURE LIMIT TABLES AND USE TABLES FOR ALL BWR PLANTS

NOMENCLATURE

Pressure-Temperature Limits 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.)

<u>No Data</u>
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.)

<u>No Data</u>

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.)

<u>EMA</u>
This indicates that the USE issue may be covered by the approved equivalent margins analysis in the BWR Owners Group Topical Report: NEDO-32205, Revision 1.

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 reorted in the latest submittal (GL 92-01, PTS, or P/T limits submittals).

Column 7: Unirradiated USE.

<u>EMA</u>
This indicates that the USE issue may be covered by the approved equivalent margins analysis in the BWR Owners Group Topical Report: NEDO-32205, Revision 1.

Column 8: Method of determining unirradiated USE

<u>Direct</u>
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.

<u>65%</u>
This indicates that the unirradiated USE was 65% of the USE from a longitudinal specimen.

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

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

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

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

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

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

<u>Blank</u>
indicates that there is insufficient data to determine the unirradiated USE. These licensees will utilize Topical Report NEDO-32205, Revision 1 to demonstrate USE compliance to Appendix G, 10 CFR Part 50.