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

March 29, 1994

Docket No. 50-293

Mr. E. Thomas Boulette, Ph.D
Senior Vice President - Nuclear
Boston Edison Company
Pilgrim Nuclear Power Station
RFD #1 Rocky Hill Road
Plymouth, Massachusetts 02360

Dear Mr. Boulette:

SUBJECT: GENERIC LETTER (GL) 92-01, REVISION 1, "REACTOR VESSEL STRUCTURAL INTEGRITY," PILGRIM NUCLEAR POWER STATION (TAC NO. M83498)

By letters dated July 1, 1992, and August 30, 1993, the Pilgrim Nuclear Power Station (PNPS) 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 PNPS 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 database 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 pressure-temperature table for your facility, Enclosure 2 provides the USE tables 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.

We request that you verify the information you have provided for your facility has been accurately entered in the database. No response is necessary unless an inconsistency is identified. If no comments are received within 30 days from the date of this letter, the staff will consider your actions related to GL 92-01, Revision 1, to be complete.

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 B. Eaton, Project Manager
Division of Reactor Projects - I/II
Project Directorate I-3
Office of Nuclear Reactor Regulation

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

cc w/enclosures:
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Ronald B. Eaton

Ronald B. Eaton, Project Manager
Division of Reactor Projects - I/II
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Enclosure 1

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
Pilgrim	Lower Shell G3109-1	C-2957-1	1.39E18	-10°F	Plant specific	65	Table	0.10	0.48
EOL: 6/8/2012	Lower Shell G3109-2	C-2957-2	1.39E18	-10°F	Plant specific	65	Table	0.10	0.47
	Lower Shell G3109-3	C-2973-1	1.39E18	-10°F	Plant specific	74.45	Table	0.11	0.63
	Lower Int. Shell G3108-1	C-2921-2	1.39E18	-10°F	Plant specific	100	Table	0.14	0.60
	Lower Int. Shell G3108-2	C-2945-1	1.39E18	-10°F	Plant specific	65.5	Table	0.10	0.65
	Lower Int. Shell G3108-3	C-2945-2	1.39E18	-10°F	Plant specific	65.6	Table	0.10	0.66
	Lower Int. Axial Welds 1-338A/C	27204 & 12008	1.31E18	0°F	Plant specific	173.2	Table	0.13	1.06
	Lower Int. to Upper Shell Circ. Weld 3-339B	13253	1.31E18	0°F	Plant specific	95	Table	0.07	0.72
	Lower Int. to Lower Shell Circ. Weld 1-344	21935	1.31E18	0°F	Plant specific	151.65	Table	0.13	0.71
	Lower Shell Axial Welds 2-338A/C	27204 & 12008	1.31E18	0°F	Plant specific	173.2	Table	0.13	1.06

<u>Reference for Pilgrim</u>

IRT data are from August 30, 1993 to NRC (Response to GL 92-01 RAI).

Plate Cu data are from Appendix 1 to TR-6052B-1 added on to the report at a later date.

Except for plate Cu data, Ni, Cu, P data are from "Pilgrim Nuclear Power Station Reactor Pressure Vessel Pressure Temperature Limits," Technical Report TR-6052B-1, Rev. 1, June 26, 1986

S data are from July 1, 1992, letter from R. A. Anderson (BE) to USNRC Document Control Desk, subject: Response to Generic Letter 92-01 (There are no S data for the welds)

Fluence data are from April 16, 1991, letter from J. C. Tascoyeannes (BE) to USNRC Document Control Desk, subject: Pilgrim RPV Pressure Temperature Limits

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
Pilgrim EOL: 6/8/2012	Lower Shell G3109-1	C-2957-1	A 533B-1	67	9.9E17	76	65%
	Lower Shell G3109-2	C-2957-2	A 533B-1	70	9.9E17	79	65%
	Lower Shell G3109-3	C-2973-1	A 533B-1	64	9.9E17	72	65%
	Lower Int. Shell G3108-1	C-2921-2	A 533B-1	70	9.9E17	81	65%
	Lower Int. Shell G3108-2	C-2945-1	A 533B-1	72	9.9E17	80	65%
	Lower Int. Shell G3108-3	C-2945-2	A 533B-1	72	9.9E17	81	65%
	Lower Int. Axial Welds 1-338A/C	27204 & 12008	Linde 1092, SAW	63	9.3E17	75[3]	NRC Generic
	Lower Int. to Upper Shell Circ. Weld 3-339B	13253	Linde 1092, SAW	99	9.3E17	113	Surv. Weld
	Lower Int. to Lower Shell Circ. Weld 1-344	21935	Linde 1092, SAW	63	9.3E17	75[3]	NRC Generic
	Lower Shell Axial Welds 2-338A/C	27204 & 12008	Linde 1092, SAW	63	9.3E17	75[3]	NRC Generic

[3]Generic value for welds fabricated by Combustion Engineering using Linde 1092, 0091 and 124 and Arcos B-5 fluxes (Ref: Letter from S. Bloom, NRR, to T.L. Patterson, OPPD, dated December 3, 1993)

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

Reference for Pilgrim

The UUSE data for the surv. weld is from August 30, 1993 letter to NRC (Response to GL 92-01 RAI).

UUSE, Ni, Cu, P data are from "Pilgrim Nuclear Power Station Reactor Pressure Vessel Pressure Temperature Limits," Technical Report TR-6052B-1, Rev. 1, June 26, 1986

Fluence data are from April 16, 1991, letter from J. C. Tsacoyeanes (BE) to USNRC Document Control Desk, subject: Pilgrim RPV Pressure Temperature Limits

ENCLOSURE 3

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.)

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 the BWR Owners Group Topical Report: NEDO-32205, Revision 1.

Column 5: 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).

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.