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

April 12, 1994

Docket No. 50-309

Mr. Charles D. Frizzle, President
Maine Yankee Atomic Power Company
83 Edison Drive
Augusta, Maine 04336

Dear Mr. Frizzle:

SUBJECT: GENERIC LETTER (GL) 92-01, REVISION 1, "REACTOR VESSEL STRUCTURAL INTEGRITY," MAINE YANKEE REVIEW STATUS (TAC NO. M83479)

By letter dated July 2, 1992, Maine Yankee provided its response to GL 92-01, Revision 1. In its letter dated October 1, 1993, the NRC staff requested additional information from Maine Yankee, to which you responded in your letter dated December 3, 1993. The staff has completed its review of your responses. Based on its review, the staff has determined that Maine Yankee has provided the information requested in GL 92-01, Revision 1, but additional confirmatory data is required.

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, Revision 1, including previously docketed information, is being used by the staff 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. This data has been entered into a computerized database designated the 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 to this letter provides the PTS table for Maine Yankee, Enclosure 2 provides the USE table for Maine Yankee, and Enclosure 3 provides the key for the nomenclature used in the tables. The tables include the data necessary to perform USE and RT_{pts} evaluations. This data was 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.

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 unirradiated USE value that was either a mean value from welds fabricated using the same flux type, or a value based on your surveillance material. This type of value is unacceptable because it does not consider heat variability of the unirradiated

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USE. When the unirradiated USE for a particular heat of material 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. The tolerance limit should be for all welds fabricated by the reactor vessel vendor, unless it can be demonstrated that the welds are separable by flux type or other welding variable(s). Maine Yankee must demonstrate that there is a physical (metallurgical) difference in the welds and a statistical difference in the data to use a generic unirradiated USE for a particular flux type or other welding variable(s).

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 to 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 plant has been accurately entered in the pressurized thermal shock and upper shelf energy tables for Maine Yankee (Enclosures 1 and 2, respectively). If your response gives no comment to this summary data file verification request, the staff will use the current information in the enclosed 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
E. H. Trottier, Project Manager
Project Directorate I-3
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

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E. H. Trottier, Project Manager
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Pressurized Thermal Shock Table

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
Maine Yankee EOL: 10/21/2008	Int. Shell Plate D-8406-1	B-7955-1	1.80E19	-10°F	Plant Specific	119.53	Calculated	0.15	0.59
	Int. Shell Plate D-8406-2	B-7955-2	1.80E19	0°F	MTEB 5-2	119.53	Calculated	0.17	0.56
	Int. Shell Plate D-8406-3	C-3982-5	1.80E19	0°F	MTEB 5-2	83.3	Table	0.12	0.62
	Lower Shell Plate D-8407-1	B-8330-1	1.80E19	-20°F	MTEB 5-2	176	Table	0.24	0.62
	Lower Shell Plate D-8407-2	B-8330-2	1.80E19	2°F	MTEB 5-2	169	Table	0.23	0.62
	Lower Shell Plate D-8407-3	B-8324-1	1.80E19	0°F	MTEB 5-2	92.25	Table	0.13	0.68
	Circ. Weld 9-203	IP3571	1.80E19	-56°F	Generic	240.96	Calculated	0.31	0.76
	Axial Welds 2-203	51989	1.48E19	-56°F	Generic	89.45	Table	0.17	0.17
	Axial Welds 3-203	13253 and 12008	1.48E19	-56°F	Generic	206.4	Table	0.22	0.84

References for Maine Yankee

IRT_{NDT}, Cu and Ni data (other than Ni value for weld 9-203) and heat numbers from Attachment F of MY letter to USNRC dated December 2, 1988 (MN-88-116) subject: Proposed Change No. 145--Combined Heatup, Cooldown and Pressure-Temperature Limitations. Ni value for weld 9-203 and fluence in MY letter to USNRC dated Oct. 28, 1991 (MN-91-151), Subject: Update of PTS Assessment to Address the Revised PTS Rule.

Upper-Shelf Energy Table

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
Maine Yankee EOL: 10/21/2008	Int. Shell Plate D-8406-1	B-7955-1	A 533B-1	80	1.073E19	115	Direct
	Int. Shell Plate D-8406-2	B-7955-2	A 533B-1	62	1.073E19	84	65%
	Int. Shell Plate D-8406-3	C-3982-5	A 533B-1	71	1.073E19	90	65%
	Lower Shell Plate D-8407-1	B-8330-1	A 533B-1	57	1.073E19	86	65%
	Lower Shell Plate D-8407-2	B-8330-2	A 533B-1	56	1.073E19	82	65%
	Lower Shell Plate D-8407-3	B-8324-1	A 533B-1	64	1.073E19	82	65%
	Circ. Weld 9-203	IP3571	Linde 1092, SAW	56	1.073E19	105	Surv. Weld
	Axial Welds 2-203 A,B,C	51989	Linde 124, SAW	53	8.82E18	75[5]	NRC Generic
	Axial Welds 3-203 A,B,C	13253 and 12008	Linde 1092, SAW	70	8.82E18	107 [2]	Sister Plant

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[5]Generic value for welds fabricated by Combustion Engineering using Linde 1092, 0091, 123 and Arcos B-5 fluxes (Ref. Letter from S. Bloom of USNRC to T.L. Patterson of Omaha Public Power District, dated December 3, 1993.

[2]Additional information required to confirm value

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

References for Maine Yankee

UUSE datum for intermediate shell plate D-8406-1/B-7953-1 is from WCAP-9875

UUSE data for other beltline plates are from J. W. Sheckherd and R. A. Wullaert, "Unirradiated Mechanical Properties of Maine Yankee Nuclear Plant Pressure Vessel Materials," CR 75-269, Effects Technology, February 1, 1975

Plate heat numbers are from Attachment F to December 2, 1988, letter from J. B. Randazza (MYAPCo) to USNRC Document Control Desk, subject: Proposed Change No. 145--Combined Heatup, Cooldown and Pressure-Temperature Limitations

UUSE data for welds 2-203 A,B,C and 3-203 A,B,C are from December 3, 1993 letter from James R. Herbert to NRC, Response to Generic Letter 92-01, Rev. 1.

Fluence based on data reported in MY letter to USNRC dated October 28, 1991 (MN-91-151), Subject: Update of PTS Assessment to Address the Revised PTS Rule.

ENCLOSURE 3

NOMENCLATURE KEY FOR PTS AND USE TABLES

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 as 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. 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 copper 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 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).

Column 7: Unirradiated USE.

<u>EMA</u>
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.

<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 plant(s) with the same weld wire heat number.

<u>Blank</u>
Indicates that there is insufficient data to determine the unirradiated USE.