Docket file

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

April 14, 1994

Docket No. 50-336

Mr. John F. Opeka
Executive Vice President, Nuclear
Connecticut Yankee Atomic Power Company
Northeast Nuclear Energy Company
Post Office Box 270
Hartford, Connecticut 06141-0270

Dear Mr. Opeka:

SUBJECT: GENERIC LETTER (GL) 92-01, REVISION 1, "REACTOR VESSEL STRUCTURAL INTEGRITY," MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2 (TAC NO. M83483)

By letters dated July 6, 1992, January 4, 1993, and September 10, 1993, Northeast Nuclear Energy Company (NNECO) provided its responses 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 NNECO 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 Millstone 2, 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.

As a result of our GL 92-01 review, the staff has identified one open issue for Millstone 2. 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, either a mean value from welds fabricated using the same flux type or a value based on your surveillance material. These types of values are unacceptable because they do not consider heat variability of the unirradiated USE. When the unirradiated USE for a particular heat of material has not been determined, you can determine the lower tolerance limit with

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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 variables. The licensee must demonstrate tnat there is a physical (metallurgical) difference in the welds and a statistical difference in the data to utilize a generic unirradiated USE for a particular flux type or other welding variables.

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 Millstone 2 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,

Guy S. Vissing, 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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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
Millstone Unit 2 EOL: 7/31/2015	Int. Shell C-505-1	C-5843-1	2.29E19	20°F	Plant Specific	92	Table	0.13	0.64
	Int. Shell C-505-2	C-5843-2	2.29E19	20°F	Plant Specific	92	Table	0.13	0.64
	Int. Shell C-505-3	C-5843-3	2.52E19	5°F	Plant Specific	92.25	Table	0.13	0.65
	Lower Shell C-506-1	C-5667-1	2.4E19	6°F	Plant Specific	108.77	Calculated	0.14	0.61
	Lower Shell C-506-2	C-5667-2	2.4E19	-14°F [9]	MTEB 5-2	100.25	Table	0.14	0.61
	Lower Shell C-506-3	C-5518-1	2.4E19	37°F [9]	MTEB 5-2	93.5	Table	0.13	0.7
	Int. Shell Axial Welds, 2-203A	A8746	1.95E19	-50°F	Plant Specific	72	Table	0.12	0.20
	Lower Shell Axial Welds, 3-203A	A8746	1.95E19	-50°F	Plant Specific	72	Table	0.12	0.20
	Upper/Int. Shell Circ. Weld 8-203	33A277	0.355E19	-80°F	Plant Specific	144.5	Table	0.30	0.18
	Upper/Int. Shell Circ. Weld 8-203	10137	0.355E19	-60°F	Plant Specific	105.8	Table	0.23	0.06
	Int./Lower Shell Circ. Weld 9-203	10137	2.4E19	-55°F	Plant Specific	76.09	Calculated	0.23	0.06
	Int./Lower Shell Circ. Weld 9-203	90136	2.4E19	-60°F	Plant Specific	135.5	Table	0.30	0.06

Reference for Millstone 2

Fluence, chemical composition and IRT_{NDT} for the welds are from the July 6, 1992 letter from J.F. Opeka to USNRC Document Control Desk, Subject: Haddam Neck Plant; Millstone Power Station, Units 1,2 and 3: Reactor Vessel Structural Integrity, 10CFR50.54(f), (Generic Letter 92-01, Revision 1).

[9]Staff determination of IRT_{NDT} from longitudinal CVN data by converting longitudinal data to transverse data using the 65% correction factor, in accordance with MTEB 5-2.

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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
Millstone 2 EOL: 7/31/2015	Int. Shell C-505-1	C-5843-1	A 533B-1	55	1.21E19	88	Direct
	Int. Shell C-505-2	C-5843-2	A 533B-1	55	1.21E19	89	Direct
	Int. Shell C-505-3	C-5843-3	A 533B-1	58	1.33E19	95	Direct
	Lower Shell C-506-1	C-5667-1	A 533B-1	67	1.27E19	108	Direct
	Lower Shell C-506-2	C-5667-2	A 533B-1	53	1.27E19	86	65%
	Lower Shell C-506-3	C-5518-1	A 533B-1	54	1.27E19	88	65%
	Int. Shell Axial Welds, 2-203A	A8746	Linde 124 SAW	68 [6]	1.03E19	93 [6]	Generic
	Lower Shell Axial Welds, 3-203A	A8746	Linde 124 SAW	68 [6]	1.03E19	93 [6]	Generic
	Upper/Int. Shell Circ. Weld 8-203	33A277	Linde 0091 SAW	110	1.88E18	160	Sister Plant
	Upper/Int. Shell Circ. Weld 8-203	101337	Linde 0091 SAW	90	1.88E18	130	Surv. Weld
	Int./Lower Shell Circ. Weld 9-203/3999	10137	Linde 0091 SAW	71	1.27E19	130	Surv. Weld
	Int./Lower Shell Circ. Weld 9-203/3999	90136	Linde 0091 SAW	71	1.27E19	130	Surv. Weld

[6]Licensee utilized a generic UUSE value; additonal information required to confirm the value.

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

UUSE data for the plates are from the September 10, 1994 letter from J.F. Opeka to USNRC Document Control Desk, Millstone Nuclear Power Station, Unit No. 2. "Response to Request for Additional Information, Generic Letter 92-01, Revision 1.

Confirmation of the methodologies employed for determining the UUSE data were from table 7-6 of BAW 2142, "Analysis of Capsule W-104, Northeast Nuclear Energy Co., Millstone Nuclear Power Station unit 2, November, 1991.

Fluence data reported in a November 27, 1991 letter from J.F. Opeka (NNECO) to USNRC

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).

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.

<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.