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

May 31, 1994

Docket Nos. 50-338
and 50-339

Mr. J.P. O'Hanlon
Senior Vice President - Nuclear
5000 Dominion Blvd.
Glen Allen, Virginia 23060

Dear Mr. O'Hanlon:

SUBJECT: GENERIC LETTER (GL) 92-01, REVISION 1, "REACTOR VESSEL STRUCTURAL INTEGRITY," VIRGINIA ELECTRIC AND POWER COMPANY (VEPCO) NORTH ANNA POWER STATION, UNITS NO. 1 AND NO. 2 (NA-1&2) (TAC NOS. M83488 AND M83489)

By letters dated May 29, 1992, September 28, 1992, October 22, 1992, December 29, 1992, September 23, 1993, and February 9, 1994, you provided a response to GL 92-01, Revision 1. The NRC staff has completed its review of your responses.

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 the Reactor Vessel Integrity Database (RVID). The RVID contains the following tables: a pressurized thermal shock (PTS) table for PWRs, a pressure-temperature limits table for BWRs and an upper-shelf energy (USE) table for PWRs and BWRs. Enclosure 1 provides the PTS tables. Enclosure 2 provides the USE tables for NA-1&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.

We have determined that additional data is required to confirm that the USE at end-of-life (EOL) for one of your beltline materials, forging 05, for NA-1, is greater than 50 ft-lb because you have provided a generic mean value for the unirradiated USE. These types of values are unacceptable because they do not consider material variability. When the unirradiated USE for a particular material has not been determined, you can set the USE equal to the lower tolerance limit calculated for the group of similar materials. The unirradiated USE should be determined such that there exists 95% confidence

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that at least 95% of the population is greater than the lower 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 Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. We request that, within 30 days receipt of this letter, you submit a schedule for providing this required data.

Additionally, we have determined that additional data is required to confirm the value provided for the nickel content of weld 05B of the NA-2 reactor vessel. The value of 0.10 provided in the GL 92-01 submittal was cited as an "estimated" value. However, the supporting data and methodology for determining the estimated value were not provided. The Pressurized Thermal Shock (PTS) Rule, 10 CFR 50.61, requires that the amounts of copper and nickel be best-estimate values. According to the PTS Rule, a mean value is acceptable for welds fabricated using the same heat number as that which matches the critical reactor vessel weld. If these values are unavailable, upper limiting values given in the material specifications to which the reactor vessel was built may be used. If not available, conservative estimates (mean plus one standard deviation) based on generic data (data from reactor vessels fabricated to the same material specification in the same shop as your vessel and in the same time period) may be used if justification is provided. If none of these alternatives are available, 1.0 percent nickel must be assumed. We request that you provide the Westinghouse Owners Group (WOG) data that was used to determine the amount of nickel and that you determine the best-estimate amount of nickel in accordance with the PTS Rule, 10 CFR 50.61, within 30 days of receipt of this letter.

Further, we request that you verify that the information you have provided for NA-1&2 has been accurately entered in the summary file. If no comments are made in your response to this request, the staff will use the information in the tables for future NRC assessments of your reactor pressure vessels. 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 plant-specific licensing actions.

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)

Leon B. Engle, Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Pressurized Thermal Shock Tables
2. Upper-Shelf Energy Tables
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	IRT_{ndt}	Method of Determin. IRT_{ndt}	Chemistry Factor	Method of Determin. CF	%Cu	%Ni
North Anna 1 EOL: 4/1/2018	Nozzle shell forging 05	990286/ 295213	2.51E18	6°F	MTEB 5-2	121.5	Table	0.16	0.74
	Int. shell forging 04	990311/ 298244	3.95E19	17°F	Plant Specific	86	Table	0.12	0.82
	Lower shell forging 03	990400/ 292332	3.95E19	38°F	Plant Specific	73.503	Calculated	0.16	0.80
	Weld 04	25531	3.95E19	19°F	Plant Specific	93.089	Calculated	0.09	0.11
	Weld 05A	25295	2.78E18	0°F	Generic	138.5	Table	0.30	0.17
	Weld 05B	4278	2.78E18	0°F	Generic	58.5	Table	0.11	0.11

References

The nickel contents for weld 05A and 05B are values from Sequoyah 1&2. (Same weld wire heat numbers).

Chemical composition and IRT_{ndt} data are from BAW-2168, which is attached to the GL 92-01 response.

Fluence data:

WCAP-11777: ID EOL fluence is 3.95E19 n/cm²

Table 2-1 of BAW-2146, which is attached to December 27, 1991, letter from W. L. Stewart (VPCo) to USNRC Document Control Desk, subject: Request to Change Technical Specifications: Pressure/Temperature Limitations, Low Temperature/Overpressure Protection System Setpoints, states that forging 05, and welds 05A and 05B have fluences that differ from forgings 03 and 04, and weld 04.

Note:

Chemical composition values for forging as are averages from the beltline material data and the surveillance data.

A margin of 69°F (σ_I = 20°F $\sigma\Delta$ = 28°F) has to be used for weld 05A and weld 05B for which a generic IRT_{ndt} of 0°F has been derived.

Summary File for Pressurized Thermal Shock

Plant Name	Beltline Ident.	Heat No. Ident.	ID Neut. Fluence at EOL	IRT_{ndt}	Method of Determin. IRT_{ndt}	Chemistry Factor	Method of Determin. CF	%Cu	%Ni
North Anna 2 EOL: 8/21/2020	Upper shell forging 05	990598/ 291396	4.47E19	9°F	MTEB 5-2	51	Table	0.08	0.77
	Int. shell forging 04	990496/ 292424	4.47E19	75°F	Plant Specific	35.112	Calculated	0.10	0.85
	Lower shell forging 03	990533/ 207355	4.47E19	56°F	Plant Specific	96	Table	0.13	0.83
	Weld 04	716126	4.47E19	-48°F	Plant Specific	10.398	Calculated	0.09	0.08
	Weld 05A	4278	4.47E19	0°F	Generic	58.5	Table	0.11	0.11
	Weld 05B	801	4.47E19	0°F	Generic	87.0	Table	0.18	0.10 [7]

References -

The nickel content for weld 05A is from Sequoyah 2 (the same weld wire heat number and the same flux).

BAW-2168, which is attached to June 29, 1992, letter from W. L. Stewart (VPCo) to USNRC Document Control Desk, subject: Response to Generic Letter 92-01, Reactor Vessel Structural Integrity, contains chemical composition and the initial RT_{ndt} (IRT_{ndt}) data for all the beltline materials

Fluence is from Table 6-13 of WCAP-12497

Note:

Chemical composition values for forging 04 are averages from the beltline material data and the surveillance data.

A margin of 69°F (σ_I = 20°F $\sigma\Delta$ = 28°F) has to be used for weld 05A and weld 05B for which a generic IRT_{ndt}, of 0°F has been derived.

[7] Additional information required to confirm value

Summary File for Upper Shelf Energy

Plant Name	Beltline Ident.	Heat No.	Material Type	1/4T USE at EOL	1/4T Neutron Fluence at EOL	Unirrad. USE	Method of Determin. Unirrad. USE
North Anna 1	Nozzle shell forging 05	990286/ 295213	A 508-2	62	1.74E18	75[7]	Generic
EOL: 4/1/2018	Int. shell forging 04	990311/ 298244	A 508-2	68	2.49E19	92	Direct
	Lower shell forging 03	990400/ 292332	A 508-2	58	2.49E19	85	Direct
	Circ. Weld Weld 04	25531	SMIT 89, SAW	71	2.49E18	102	Direct
	Nozzle to Int. Shell Weld 05A	25295	SMIT 89, SAW	78	1.74E18	111	Sister Plant
	Nozzle to Int. Shell Weld 05B	4278	SMIT 89, SAW	---	1.74E18	105	Sister Plant

References

The fluence data for weld 04 is from June 29, 1992 letter to NRC (Response to GL 92-01); fluence data for other materials are from September 23, 1993 letter to NRC (Response to GL 92-01 RAI).

Chemical composition and UUSE data for forging 03 are from BAW-2168, which is attached to the GL 92-01 response.

UUSE data for forging 04 and weld 04 are from BAW-1911, Rev. 1.

Note: Weld 05A is 94% of thickness of the Nozzle to intermediate shell weld and 05B is the remainder. Therefore, it is not necessary to evaluate the EOL USE for weld 05B because it is not at the 1/4T location.

[7] Additional information required to confirm value

Summary File for Upper Shelf Energy

Plant Name	Beltline Ident.	Heat No.	Material Type	1/4T USE at EOL	1/4T Neutron Fluence at EOL	Unirrad. USE	Method of Determin. Unirrad. USE
North Anna 2	Upper shell forging 05	990598/ 291396	A 508-2	64	2.0E18	74	Equiv. to forging 04
EOL: 8/21/2020	Int. shell forging 04	990496/ 292424	A 508-2	51	2.82E19	74	Direct
	Lower shell forging 03	990533/ 207355	A 508-2	58	2.82E19	80	Direct
	Weld 04	716126	LW 320, SAW	69	2.82E19	107	Direct
	Weld 05A	4278	SMIT 89, SAW	86	2.82E19	105	Sister Plant
	Weld 05B	801	SMIT 89, SAW	---	2.82E19	---	---

References

BAW-2168, which is attached to June 29, 1992, letter from W. L. Stewart (VPCo) to USNRC Document Control Desk, subject: Response to Generic Letter 92-01, Reactor Vessel Structural Integrity, contains chemical composition data for all the beltline materials. However, it contains UUSEs for forging 04 and weld 04 only.

Fluence and UUSEs for forgings 05 and 03 are from December 29, 1992 letter to NRC.

Note: Weld 05A is 94% of thickness of the Nozzle to intermediate shell weld and 05B is the remainder. Therefore, it is not necessary to evaluate the EOL USE for weld 05B because it is not at the 1/4T location.

PRESSURIZED THERMAL SHOCK 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.)

<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 either owners group or plant-specific equivalent margins analyses.

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 either owners group or plant-specific equivalent margins analyses.

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.