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

May 31, 1994

Docket Nos. 50-369
and 50-370

Mr. M. S. Tuckman
Senior Vice President
Nuclear Generation
Duke Power Company
P. O. Box 1006
Charlotte, North Carolina 29201

Dear Mr. Tuckman:

SUBJECT: GENERIC LETTER (GL) 92-01, REVISION 1, "REACTOR VESSEL STRUCTURAL INTEGRITY," MCGUIRE NUCLEAR STATION, UNITS 1 AND 2 (TAC NO. M83480 AND M83481)

By letter dated July 3, 1992, Duke Power Company (DPC) provided its response to GL 92-01, Revision 1. The NRC staff has completed its review of your response. Based on its review, the staff has determined that DPC 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 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 table, Enclosure 2 provides the USE table for your facilities, 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 response to GL 92-01 and previously docketed information. References to the specific source of the data are provided in the tables.

We request that you verify that the information that you have provided for your facilities has been accurately entered in the summary data files. 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 and the staff will use the information in the tables for future NRC assessments of your reactor pressure vessel.

DF01

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:

Victor Nerses, Project Manager
Project Directorate II-3
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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Sincerely,

Victor Nerses

Victor Nerses, Project Manager
Project Directorate II-3
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

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1. Pressurized Thermal Shock Tables
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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
McGuire 1 EOL: 6/12/2021	Int. shell B5012-1	C4387-2	2.02E19	34°F	Plant Specific	54.51	Calculated	0.11	0.61
	Int. shell B5012-2	C4417-3	2.02E19	0°F	Plant Specific	100	Table	0.14	0.61
	Int. shell B5012-3	C4377-2	2.02E19	-13°F	Plant Specific	75	Table	0.11	0.66
	Lower shell B5013-1	C4315-1	2.02E19	0°F	Plant Specific	99.1	Table	0.14	0.58
	Lower shell B5013-2	C4374-2	2.02E19	30°F	Plant Specific	65	Table	0.10	0.51
	Lower shell B5013-3	C4371-2	2.02E19	15°F	Plant Specific	65	Table	0.10	0.55
	Int. shell axial welds M1.22	20291 and 12008	1.46E19	-50°F	Plant Specific	171.06	Calculated	0.20	0.87
	Int. to lower shell circ. weld G1.39	83640	2.02E19	-70°F	Plant Specific	39.8	Table	0.05	0.12
	3-442 ABC Lower shell axial welds M1.32	21935 and 12008	1.46E19	-56°F	Generic	209.6	Table	0.22	0.86

References for McGuire 1

IRT_{ndt} data are from July 3, 1992, letter from H. B. Tucker (DPCo) to USNRC Document Control Desk, subject: McGuire Nuclear Station, Units 1 and 2; Catawba Nuclear Station Units 1 and 2; Oconee Nuclear Station, Units 1, 2, and 3: Response to Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity.

Fluence data are from the March 17, 1994 letter to USNRC subject: Pressurized Thermal Shock

Chemical compositions are from the response to GL 92-01 dated 12/20/93 and from the March 17, 1994 letter on Pressurized Thermal Shock.

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
McGuire 2	Forging 05 Int. shell	526840	2.04E19	-4°F	Plant Specific	87.98	Calculated	0.16	0.85
EOL: 3/3/2023	Forging 04 Lower shell	411337-11	2.04E19	-30°F	Plant Specific	115.8	Table	0.15	0.88
	Int. to Lower shell Welds Shell W05	895075	2.04E19	-68°F	Plant Specific	33.48	Calculated	0.C3	0.70

References for McGuire 2

Chemical composition and IRT_{NDT} data are from July 3, 1992, letter from H. B. Tucker (DPCo) to USNRC Document Control Desk, subject: McGuire Nuclear Station, Units 1 and 2; Catawba Nuclear Station Units 1 and 2; Oconee Nuclear Station, Units 1, 2, and 3: Response to Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity.

Fluence data are from Table 6-22 of WCAP-13516.

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
McGuire 1 EOL: 6/12/2021	Int. shell B5012-1	C4387-2	A 533B-1	71	1.20E19	89	65%
	Int. shell B5012-2	C4417-3	A 533B-1	68	1.20E19	89	65%
	Int. shell B5012-3	C4377-2	A 533B-1	81	1.20E19	100	65%
	Lower shell B5013-1	C4315-1	A 533B-1	64	1.20E19	84	65%
	Lower shell B5013-2	C4374-2	A 533B-1	77	1.20E19	96	65%
	Lower shell B5013-3	C4371-2	A 533B-1	68	1.20E19	85	65%
	Int. shell axial welds M1.22	20291 and 12008	Linde 1092, SAW	70	0.87E19	110	Direct
	Int. to lower shell 9-4442 circ. weld G1.39	83640	Linde 0091, SAW	101	1.20E19	126	Direct
	Lower shell axial welds M1.32	21935 and 12008	Linde 1092, SAW	59	0.87E19	90	Direct

<u>References</u>

Chemical composition and UUSE data are from July 3, 1992, letter from H. B. Tucker (DPCo) to USNRC Document Control Desk, subject: McGuire Nuclear Station, Units 1 and 2; Catawba Nuclear Station Units 1 and 2; Oconee Nuclear Station, Units 1, 2, and 3: Response to Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity.

UUSE data for welds M1.22, G1.39, and M1.32 are from Table A-1 of WCAP-10786.

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
McGuire 2 EOL: 3/3/2023	Int. shell Forging 05	526840	A 508-2	74	1.225E19	100	Direct
	Lower shell Forging 04	411337-11	A 508-2	71	1.225E19	97	65%
	Int. lower shell Welds W05	895075	Grau Lo, SAW	112	1.225E19	140	Direct

References

Chemical composition and UUSE data are from July 3, 1992, letter from H. B. Tucker (DPCo) to USNRC Document Control Desk, subject: McGuire Nuclear Station, Units 1 and 2; Catawba Nuclear Station Units 1 and 2; Oconee Nuclear Station, Units 1, 2, and 3: Response to Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity.

Fluence data are from Table 6-22 of WCAP-13516.

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 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.