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

May 20, 1994

Docket Nos. 50-348
and 50-364

Mr. D. N. Morey, Vice President
Southern Nuclear Operating Co., Inc.
Post Office Box 1295
Birmingham, Alabama 35201-1295

Dear Mr. Morey:

SUBJECT: RESPONSE TO GENERIC LETTER (GL) 92-01, REVISION 1, "REACTOR VESSEL STRUCTURAL INTEGRITY," JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. 83461 AND 83462)

By letters dated July 1, 1992, and November 23, 1993, Southern Nuclear Operating Company (SNC) provided its response to GL 92-01, Revision 1. The NRC staff has completed its review of your response(s). Based on its review, the staff has determined that SNC 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 and Enclosure 2 provides the USE tables for Farley Units 1 and 2. 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(s) 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 Generic Letter (GL) 92-01 review, the staff has identified two open issues for each of your plants:

(1) The amounts of nickel in welds fabricated using weld wire heat numbers 33A277, 6329637 and 90099 in Farley, Unit 1, and weld wire heat numbers 5P5622 and 83640 in Farley, Unit 2, were determined as mean values from the Westinghouse Owners Group (WOG) data base. The Pressurized Thermal Shock (PTS) Rule, 10 CFR 50.61, requires the amounts of copper and nickel to 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 vessel weld. If these values are unavailable, upper limiting values given in the material specifications to which the 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 within 30 days of receipt of this letter you provide the 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.

(2) Surveillance data was used to determine the unirradiated upper shelf, energy values for weld wire heat numbers 6329637 and 90099 in Farley, Unit 1, and weld wire heat numbers 5P5622, 83640 and HODA in Farley, Unit 2. However, the surveillance weld data in these cases were from a different heat than the beltline welds. Additional information is required to justify use of the surveillance data. Since the surveillance data are from a different heat, a statistical analysis addressing heat variability may be appropriate. When the unirradiated USE for a particular heat of 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 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 the SNC 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 of receipt of this letter SNC submit a schedule for performing and submitting the results of this statistical analysis to the staff.

Further, we request that you verify that the information you have provided for your facilities has been accurately entered in the summary data file. If no comments are received within 30 days of receipt of this letter, the staff will use the information in the tables for future NRC assessments of your reactor

May 20, 1994

pressure vessel. Once your response is received, the staff will consider your actions related to GL 92-01, Revision 1, to be complete. Your response to the chemical composition and surveillance weld concerns 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:

Byron L. Siegel, Project Manager
Project Directorate II-1
Division of Reactor Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Pressurized Thermal Shock Tables
2. Upper-Shelf Energy Table(s)
3. Nomenclature Key

cc w/enclosures:
See next page

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DATE	5/19/94	5/20/94	5/20/94		

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Sincerely,

Byron L. Siegel

Byron L. Siegel, Project Manager
Project Directorate II-1
Division of Reactor Projects
Office of Nuclear Reactor Regulation

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1. Pressurized Thermal Shock Tables
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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
Farley 1 EOL: 6/25/2017	Int. Shell B6903-2	C6294-1	3.75E19	0°F	MTEB 5-2	91	Table	0.13	0.60
	Int. Shell B6903-3	C6308-2	3.75E19	10°F	MTEB 5-2	82.2	Table	0.12	0.56
	Lower Shell B6919-1	C6940-1	3.75E19	15°F	MTEB 5-2	88.831	Calculated	0.14	0.55
	Lower Shell B6919-2	C6897-2	3.75E19	5°F	MTEB 5-2	98.2	Table	0.14	0.56
	Int. Shell Axial Welds	33A277	1.24E19	-56°F	Generic	78.689	Calculated	0.25	0.21 [9]
	Circ. Weld	6329637	3.75E19	-56°F	Generic	113	Table	0.23	0.20 [9]
	Lower Shell Axial Welds	90099	1.24E19	-56°F	Generic	92	Table	0.17	0.20 [9]

REFERENCES FOR FARLEY 1:

Fluence, IRT_{ndt} and chemistry values from November 23, 1993, letter from D. Morey (SNOC) to USNRC Document Control Desk, subject: Responses to Requests for Additional Information Regarding GL 92-01.

[9] Chemical composition from mean value of WOG data. Additional information required.

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
Farley 2 EOL: 3/31/2021	Int. Shell B7203-1	C6309-2	3.8E19	15°F	Plant Specific	100	Table	0.14	0.60
	Int. Shell B7212-1	C7466-1	3.8E19	-10°F	Plant Specific	145.0	Calculated	0.20	0.60
	Lower Shell B7210-1	C6888-2	3.8E19	18°F	Plant Specific	89.8	Table	0.13	0.56
	Lower Shell B7210-2	C6293-1	3.8E19	10°F	Plant Specific	98.7	Table	0.14	0.57
	Circ. Weld 11-923	5P5622	3.8E19	-40°F	Plant Specific	76	Table	0.13	0.20 [9]
	Lower Shell Axial Welds 20-923 A/B	83640	1.23E19	-70°F	Plant Specific	49	Table	0.05	0.20 [9]
	Int. Shell Axial Welds 19-923A	HODA SMAW	1.23E19	-56°F	Generic	10.01	Calculated	0.02	0.96
	Int. Shell Axial Welds 19-923B	BOLA SMAW	1.23E19	-60°F	Plant Specific	10.01	Calculated	0.02	0.93

REFERENCES FOR FARLEY 2:

Fluence, IRT_{ndt} and chemistry values from November 23, 1993, letter from D. Morey (SNOC) to USNRC Document Control Desk, subject: Responses to Requests for Additional Information Regarding 92-01.

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[9] Chemical composition from mean value of WOG data. 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/EFPY	1/4T Neutron Fluence at EOL/EFPY	Unirrad. USE	Method of Determin. Unirrad. USE
Farley 1 EOL: 6/25/2017	Int. Shell B6903-2	C6294-1	A 533B-1	73	2.34E19	99	65%
	Int. Shell B6903-3	C6308-2	A 533B-1	65	2.34E19	87	65%
	Lower Shell B6919-1	C6940-1	A 533B-1	66	2.34E19	86	65%
	Lower Shell B6919-2	C6897-2	A 533B-1	62	2.34E19	86	65%
	Int. Shell Axial Welds	33A277	Linde 1092, SAW	115	7.73E18	149	Surv. Weld
	Circ. Weld	6329637	Linde 0091, SAW	115 [10]	2.34E19	149 [10]	Equiv. to Surv. Weld
	Lower Shell Axial Welds	90099	Linde 0091, SAW	115 [10]	7.73E18	149 [10]	Equiv. to Surv. Weld

References for Farley 1

Fluence, heat number and UUSE values from November 23, 1993, letter from D. Morey (SNOC) to USNRC Document Control Desk, subject: Responses to Requests for Additional Information Regarding GL 92-01.

[10]Surveillance weld is from a different heat than beltline welds, additional information required to confirm licensee's 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
Farley 2 EOL: 3/31/2021	Int. Shell B7203-1	C6309-2	A 533B-1	72	2.369E19	100	Direct
	Int. Shell B7212-1	C7466-1	A 533B-1	62	2.369E19	100	Direct
	Lower Shell B7210-1	C6888-2	A 533B-1	76	2.369E19	103	Direct
	Lower Shell B7210-2	C6293-1	A 533B-1	72	2.369E19	99	Direct
	Circ. Weld 11-923	5P5622	Linde 0091, SAW	110 [10]	2.369E19	148 [10]	Equiv. to Surv. Weld
	Lower Shell Axial Welds 20-923A/B	83640	Linde 0091, SAW	131 [10]	7.67E18	148 [10]	Equiv. to Surv. Weld
	Int. Shell Axial Welds 19-923A	HODA	SMAW	131 [10]	7.67E18	148 [10]	Equiv. to Surv. Weld
	Int. Shell Axial Welds 19-923B	BOLA	SMAW	131	7.67E18	148	Surv. Weld

REFERENCES FOR FARLEY 2:

Fluence, heat number and UUSE values from November 23, 1993, letter from D. Morey (SNOC) to USNRC Document Control Desk, subject: Responses to Requests for Additional Information Regarding GL 92-01.

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[10] Surveillance weld is from a different heat than beltline welds, additional information required to confirm licensee's value

Nomenclature and Tables

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.)

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 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 tha 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.