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

May 11, 1994

Docket Nos. 50-445
and 50-446

Mr. William J. Cahill, Jr.
Group Vice President, Nuclear
TU Electric Company
400 North Olive Street, L.B. 81
Dallas, Texas 75201

Dear Mr. Cahill:

SUBJECT: GENERIC LETTER (GL) 92-01, REVISION 1, "REACTOR VESSEL STRUCTURAL INTEGRITY," TEXAS UTILITIES ELECTRIC COMPANY, COMANCHE PEAK STEAM ELECTRIC STATION, UNIT NOS. 1 AND 2 (TAC NOS. M83451 AND M83452)

By letter dated July 2, 1992, Texas Utilities Electric Company (TU Electric) 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 TU Electric 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(s), Enclosure 2 provides the USE tables 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 the information you have provided for your facilities have been accurately entered in the summary data file. 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 /1

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

Thomas A. Bergman, Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
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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Honorable Dale McPherson
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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
Comanche Peak 1 EOL: 2/8/2030	Int. shell R1107-1		3.04E19	10°F	Plant specific	37	Table	0.06	0.65
	Int. shell R1107-2		3.04E19	-10°F	Plant specific	37	Table	0.06	0.64
	Int. shell R1107-3		3.04E19	10°F	Plant specific	31	Table	0.05	0.68
	Lower shell R1108-1		3.04E19	0°F	Plant specific	51	Table	0.08	0.64
	Lower shell R1108-2		3.04E19	20°F	Plant specific	31	Table	0.05	0.59
	Lower shell R1108-3		3.04E19	0°F	Plant specific	44	Table	0.07	0.64
	Int. shell axial welds 101-1~4A/C	88112	3.04E19	-70°F	Plant specific	43	Table	0.04	0.19
	Lower shell axial welds 101-142A/C	88112	3.04E19	-70°F	Plant specific	43	Table	0.04	0.19
	Circ. Weld 101-171	88112	3.04E19	-70°F	Plant specific	43	Table	0.04	0.19

<u>References:</u>

The nickel content for all welds is from the average of 0.17 (GL 92-01 response dated July 2, 1992) and 0.2 (WCAP-13422).

Chemical composition (copper and nickel only) for all beltline materials, fluence, and IRT_{NDT} data are from WCAP-13422, which is not in the PR_EDB. Surveillance materials copper, nickel, phosphorus, and sulfur data are from WCAP-13422.

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
Comanche Peak 2 EOL: 2/2/2033	Int. shell A3807-1		3.04E19	-20°F	Plant specific	37	Table	0.06	0.64
	Int. shell A3807-2		3.04E19	10°F	Plant specific	37	Table	0.06	0.64
	Int. shell A3807-3		3.04E19	-20°F	Plant specific	31	Table	0.05	0.60
	Lower shell A3816-1		3.04E19	-30°F	Plant specific	31	Table	0.05	0.59
	Lower shell A3816-2		3.04E19	0°F	Plant specific	20	Table	0.03	0.65
	Lower shell A3816-3		3.04E19	-40°F	Plant specific	26	Table	0.04	0.63
	Int. shell axial welds	89833	3.04E19	-50°F	Plant specific	37.75	Table	0.07	0.05
	Lower shell axial welds	89833	3.04E19	-50°F	Plant specific	37.75	Table	0.07	0.05
	Circ. weld	89833	3.04E19	-60°F	Plant specific	34.05	Table	0.05	0.05

References:

The nickel content for the circ. weld is from the average of 0.03 (GL 92-01 Response dated July 2, 1992) and 0.07 (WCAP-10684).

The chemical composition, IRT_{NDT}, and unirradiated upper shelf energy (UUSE) data are found in WCAP-10684, which is attached to December 16, 1985, letter from W. G. Counsil (TUECo) to V. S. Noonan (USNRC), subject: Fracture Toughness Properties of Unit 2 Reactor Vessel Materials

End of license (EOL) fluence datum is from WCAP-13422, which analyzes surveillance capsule U of Comanche Peak 1

Enclosure 2

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
Comanche Peak 1 EOL: 2/8/2030	Int. shell R1107-1		A 533B-1	73	1.812E19	94	Direct
	Int. shell R1107-2		A 533B-1	81	1.812E19	103	Direct
	Int. shell R1107-3		A 533B-1	69	1.812E19	88	Direct
	Lower shell R1108-1		A 533B-1	66	1.812E19	85	Direct
	Lower shell R1108-2		A 533B-1	61	1.812E19	78	Direct
	Lower shell R1108-3		A 533B-1	77	1.812E19	98	Direct
	Int. shell axial welds 101-124A/C	88112	Linde 0091, SAW	98	1.812E19	125	Surv. Weld
	Lower shell axial welds 101-142A/C	88112	Linde 0091, SAW	98	1.812E19	125	Surv. Weld
	Circ. Weld 101-171	88112	Linde 0091, SAW	98	1.812E19	125	Surv. Weld

References:

Chemical composition (copper [Cu] and nickel [Ni] only) for all beltline materials and fluence data are from WCAP-13422, which is not in the PR_EDB. Surveillance materials chemical composition (Cu, Ni, phosphorus [P], and sulfur [S]) and UUSE data are from WCAP-13422

UUSE data for beltline materials other than surveillance capsule are from FSAR.

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
Comanche Peak 2 EOL: 2/2/2033	Int. shell A3807-1		A 533B-1	84	1.812E19	108	Direct
	Int. shell A3807-2		A 533B-1	79	1.812E19	101	Direct
	Int. shell A3807-3		A 533B-1	82	1.812E19	105	Direct
	Lower shell A3816-1		A 533B-1	84	1.812E19	107	Direct
	Lower shell A3816-2		A 533B-1	83	1.812E19	106	Direct
	Lower shell A3816-3		A 533B-1	84	1.812E19	108	Direct
	Int. shell axial welds	89833	Linde 0091, SAW	131	1.812E19	172	Direct
	Lower shell axial welds	89833	Linde 0091, SAW	131	1.812E19	172	Direct
	Circ. weld	89833	Linde 124, SAW	[illegible]	1.812E19	98	Surv. Weld

References:

The chemical composition and unirradiated up[illegible]f energy (UUSE) data are found in WCAP-10684, which is attached to December 16, 1985, letter from W. G. Counsil (TUECo) to V. S. Noonan (USNRC), subject: Fracture Toughness Properties of Unit 2 Reactor Vessel Materials

End of license (EOL) fluence datum is from WCAP-13422, which analyzes surveillance capsule U of Comanche Peak 1

Enclosure 3

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).

<u>Plant-Specific</u>
This indicates that the IRT was determined from tests on material removed from the same heat of the beltline material.

<u>MTEB 5-2</u>
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.

<u>Generic</u>
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.

<u>Table</u>
This indicates that the chemistry factor was determined from the chemistry factor tables in RG 1.99, Revision 2.

<u>Calculated</u>
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 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 plants with the same weld wire heat number.

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