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

May 11, 1994

Docket Nos. 50-390
and 50-391

Mr. Oliver D. Kingsley, Jr.
President, TVA Nuclear and
Chief Nuclear Officer
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

Dear Mr. Kingsley:

SUBJECT: WATTS BAR NUCLEAR PLANT - GENERIC LETTER 92-01, REVISION 1, "REACTOR VESSEL STRUCTURAL INTEGRITY" AND LOW UPPER SHELF ENERGY EVALUATION (TAC NOS. M83525, M83526, and M85712)

By letters dated July 7, 1992 and October 15, 1993, TVA provided its response to Generic Letter (GL) 92-01, Revision 1. The NRC staff has completed its review of TVA's responses. Based on its review, the staff has determined that TVA 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(s), Enclosure 2 provides the USE tables for your facility, 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.

By letter dated October 15, 1993, the staff received an analysis entitled, "Watts Bar Unit 1 Low Upper Shelf Energy Evaluation" for review. This analysis was performed by Westinghouse and follows the approach used in the Westinghouse Owners Group (WOG) report on equivalent margins analyses (WCAP-13587, Rev. 1). The staff has completed its review of this analysis and has determined that TVA used methodology, modeling procedures, and acceptance criteria which fall within the scope of Draft Regulatory Guide DG-1023 and ASME Code Case N-512. On such basis, the staff determined that TVA has

demonstrated margins of safety against fracture for intermediate shell forging 05 that are equivalent to those required by Appendix G of the ASME Code. This completes licensing action M85712. Details of the staff's review will be published in Supplement 14 of the Watts Bar Safety Evaluation Report (NUREG-0847).

The staff requests that, within 30 days, you verify that the information you have provided for your facility has been accurately entered in the summary data 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 vessel. Once your response is received, the staff will consider your actions related to GL 92-01, Revision 1, to be complete.

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
Peter S. Tam, Senior Project Manager
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Pressurized Thermal Shock or Pressure-Temperature Limit Tables
2. Upper-Shelf Energy Tables
3. Nomenclature Key

cc w/enclosures:
See next page

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Document Name: GL9201.WBN

Enclosure 1

# 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 |
|---|---|---|---|---|---|---|---|---|---|
| Watts Bar 1 | Int. shell 05 | 527536 | 3.18E19 | 47°F | Plant Specific | 132 | Table | 0.17 | 0.80 |
| EOL: Not listed | Lower shell 04 | 528522 | 3.18E19 | 5°F | MTEB 5-2 | 51 | Table | 0.08 | 0.83 |
| | Circ. weld | 895075 | 3.18E19 | -43°F | Plant Specific | 68 | Table | 0.05 | 0.70 |

References

Chemical composition, $IRT_{NDT}$, and fluence data are from July 7, 1992, letter from R. M. Shell (TVA) to USNRC Document Control Desk, subject: Browns Ferry Nuclear Plant, Sequoyah Nuclear Plant, and Watts Bar Nuclear Plant--Response to Generic Letter 92-01 (Reactor Vessel Structural Integrity)

| 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 |
|---|---|---|---|---|---|---|---|---|---|
| Watts Bar 2 | Int. shell 05 | 527828 | 3.18E19 | 14°F | Plant Specific | 31 | Table | 0.05 | 0.78 |
| EOL: Not listed | Lower shell 04 | 528658 | 3.18E19 | 5°F | MTEB 5-2 | 31 | Table | 0.05 | 0.81 |
| | Circ. weld | 895075 | 3.18E19 | -50°F | Plant Specific | 68 | Table | 0.05 | 0.70 |

References

Chemical composition, $IRT_{NDT}$, and fluence data are from July 7, 1992, letter from R. M. Shell (TVA) to USNRC Document Control Desk, subject: Browns Ferry Nuclear Plant, Sequoyah Nuclear Plant, and Watts Bar Nuclear Plant--Response to Generic Letter 92-01 (Reactor Vessel Structural Integrity)

# 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 |
|---|---|---|---|---|---|---|---|
| Watts Bar 1<br><br>EOL: Not listed | Int. shell 05 | 527536 | A 508-2 | 43 (EMA) | 1.90E19 | 62 | Direct |
| | Lower shell 04 | 528522 | A 508-2 | 69 | 1.90E19 | 88 | 65% |
| | Circ. weld | 895075 | Rotterdam Grau Lo, SAW | 102 | 1.90E19 | 131 | Direct |

References

Chemical composition, UUSE, and fluence data are from July 7, 1992, letter from R. M. Shell (TVA) to USNRC Document Control Desk, subject: Browns Ferry Nuclear Plant, Sequoyah Nuclear Plant, and Watts Bar Nuclear Plant--Response to Generic Letter 92-01 (Reactor Vessel Structural Integrity)

The applicability of the Equivalent Margins Analysis (EMA) for Forging 05 was addressed in the October 15, 1993 letter from W.J. Museler to USNRC, "Watts Bar Nuclear Plant, Unit 1, Generic Letter 92-01 Response to Request for Additional Information and Withdrawal of Exemption Request." NRC staff review of the equivalent margins analysis is in process.

| 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 |
|---|---|---|---|---|---|---|---|
| Watts Bar 2<br><br>EOL: Not listed | Int. shell 05 | 527828 | A 508-2 | 86 | 1.90E19 | 110 | Direct |
| | Lower shell 04 | 528658 | A 508-2 | 82 | 1.90E19 | 105 | 65% |
| | Circ. weld | 895075 | Rotterdam Grau Lo, SAW | 108 | 1.90E19 | 139 | Direct |

References

Chemical composition, UUSE, and fluence data are from July 7, 1992, letter from R. M. Shell (TVA) to USNRC Document Control Desk, subject: Browns Ferry Nuclear Plant, Sequoyah Nuclear Plant, and Watts Bar Nuclear Plant--Response to Generic Letter 92-01 (Reactor Vessel Structural Integrity)

ENCLOSURE 3

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 I 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.

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