50-298

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

January 9, 1995

Mr. Guy R. Horn
Vice-President, Nuclear
Nebraska Public Power District
Post Office Box 98
Brownville, Nebraska 68321

SUBJECT: GENERIC LETTER (GL) 92-01, REVISION 1, REACTOR VESSEL STRUCTURAL INTEGRITY, COOPER NUCLEAR STATION (TAC NO. M83455)

Dear Mr. Horn:

By letters dated July 1, 1992, July 2, 1993, and September 16, 1993, Nebraska Public Power District (NPPD) provided its responses to GL 92-01, Revision 1. The NRC staff has completed its review of your responses. Based on its review, the staff has determined that NPPD 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. The data has 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 pressure-temperature table(s), Enclosure 2 provides the USE table 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. The data was 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 GL 92-01 review, the NRC staff has identified one open issue for your plant. The initial $RT_{NDT}$ values determined by General Electric's (GE) initial methodology have not been validated and the BWR Owners Group report, GE-NE-523-109-0893, entitled, "Basis for GE $RT_{NDT}$ Estimation Method," did not resolve the issue. GE is in the process of validating its methodology for resolving the initial $RT_{NDT}$ determination issue and will document the results in a topical report. The BWR Owners Group is obtaining approval from its members to provide the GE topical report to the NRC staff for its review and approval. We request that you submit within 30 days a commitment to the BWR Owners Group effort or a schedule for a plant-specific

analysis to resolve this issue. Further, we request that you provide confirmation of the plant-specific applicability of the topical report, NEDO-32205, Revision 1, (as specified in Appendix B of that report) and submit a request for approval of the topical report as the basis for demonstrating compliance with 10 CFR Part 50, Appendix G, Paragraph IV.A.1. We further request that you verify that the information you have provided for your facility has been accurately entered in the data base. If no comments are made in your response to the last request, the staff will use the information in the tables for future NRC assessments of your reactor pressure vessel.

Once you have (1) confirmed the applicability of the topical report, NEDO-32205, Revision 1, to your plant, (2) submitted the request for approval, and (3) provided your commitment to the BWR Owners Group effort or a satisfactory schedule for providing a plant-specific analysis, the staff will consider your actions related to GL 92-01, Revision 1, to be complete. Plant-specific licensing action(s) will be initiated to resolve these issues.

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 July 31, 1997.

Sincerely,
ORIGINAL SIGNED BY:
James R. Hall, Project Manager
Project Directorate IV-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-298

Enclosures: 1. Pressure-Temperature Limit Table
2. Upper-Shelf Energy Table
3. Nomenclature Key

cc w/encls: See next page

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analysis to resolve this issue. Further, we request that you provide confirmation of the plant-specific applicability of the topical report, NEDO-32205, Revision 1, (as specified in Appendix B of that report) and submit a request for approval of the topical report as the basis for demonstrating compliance with 10 CFR Part 50, Appendix G, Paragraph IV.A.1. We further request that you verify that the information you have provided for your facility has been accurately entered in the data base. If no comments are made in your response to the last request, the staff will use the information in the tables for future NRC assessments of your reactor pressure vessel.

Once you have (1) confirmed the applicability of the topical report, NEDO-32205, Revision 1, to your plant, (2) submitted the request for approval, and (3) provided your commitment to the BWR Owners Group effort or a satisfactory schedule for providing a plant-specific analysis, the staff will consider your actions related to GL 92-01, Revision 1, to be complete. Plant-specific licensing action(s) will be initiated to resolve these issues.

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2. Upper-Shelf Energy Table
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cc w/encls: See next page

cc:

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# Summary File for Pressure-Temperature Limits

| 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 |
|---|---|---|---|---|---|---|---|---|---|
| Cooper | Lower int. shell G-2801-7 | C2407-1 | 1.5E18 | -10°F | Plant specific | 92.25 | Table | 0.13 | 0.65 |
| EOL: 12/18/ 2014 | Lower int. shell G-2802-1 | C2331-2 | 1.5E18 | 10°F | Plant specific | 125.3 | Table | 0.17 | 0.58 |
| | Lower int. shell G-2802-2 | C2307-2 | 1.5E18 | -20°F | Plant specific | 162.8 | Table | 0.21 | 0.73 |
| | Lower shell G-2803-1 | C2274-1 | 1.5E18 | 14°F [1] | Plant specific | 153 | Table | 0.20 | 0.68 |
| | Lower shell G-2803-2 | C2307-1 | 1.5E18 | 0°F [1] | Plant specific | 162.8 | Table | 0.21 | 0.73 |
| | Lower shell G-2803-3 | C2274-2 | 1.5E18 | -8°F [1] | Plant specific | 153 | Table | 0.20 | 0.68 |
| | Lower int. shell axial welds 1-233A/C (T) | 27204 & 12008 | 1.5E18 | -50°F | Plant specific | 215.65 | Table | 0.19 | 0.97 |
| | Lower to lower-int. shell circ. weld 1-240 | 21935 | 1.5E18 | -50°F | Plant specific | 175.3 | Table | 0.20 | 0.69 |
| | Lower shell axial welds 2-233A/C | 12420 | 1.5E18 | -50°F | Plant specific | 234.5 | Table | 0.22 | 1.02 |

References:

Chemical composition (copper [Cu] and nickel [Ni]) for weld 2-233A/C, Ni for weld 1-240 and 1-233A/C are from February 25, 1993, letter from G. R. Horn (NPPD) to USNRC Document Control Desk, subject: Submittal of Reactor Vessel Surveillance Test Results.

$IRT_{ndt}$ data for weld 1-233A/C is from G. R. Horne letter of February 25, 1993, cited above.

Other chemical composition, and other $IRT_{ndt}$ data are from July 1, 1992, letter from G. R. Horn (NPPD) to USNRC Document Control Desk, subject: Response to Generic Letter 92-01, Revision 1.

[1] 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 |
|---|---|---|---|---|---|---|---|
| Cooper<br><br>EOL: 12/18/ 2014 | Lower int. shell G-2801-7 | C2407-1 | A 533B-1 | 61 | 1.08E18 | 84 | 65% |
| | Lower int. shell G-2802-1 | C2331-2 | A 533B-1 | 61 | 1.08E18 | 72 | 65% |
| | Lower int. shell G-2802-2 | C2307-2 | A 533B-1 | 67 | 1.08E18 | 84 | 65% |
| | Lower shell G-2803-1 | C2274-1 | A 533B-1 | 60 | 1.08E18 | 73 | 65% |
| | Lower shell G-2803-2 | C2307-1 | A 533B-1 | 61 | 1.08E18 | 75 | 65% |
| | Lower shell G-2803-3 | C2274-2 | A 533B-1 | 60 | 1.08E18 | 72 | 65% |
| | Lower int. shell axial welds 1-233A/C | 27204 & 12008 | Linde 1092, SAW | 85 | 1.08E18 | 112 | Surv. Weld |
| | Lower to lower-int. shell circ. weld 1-240 | 21935 | Linde 1092, SAW | EMA[2] | 1.08E18 | EMA[2] | - - |
| | Lower shell axial welds 2-233A/C | 12420 | Linde 1092, SAW | EMA[2] | 1.08E18 | EMA[2] | - - |

References:

Chemical composition (copper [Cu] and nickel [Ni]) for weld 2-233A/C, Ni for weld 1-240 and 1-233A/C are from February 23, 1993, letter from G. R. Horn (NPPD) to USNRC Document Control Desk, subject: Submittal of Reactor Vessel Surveillance Test Results.

Fluence, the UUSE of the surv. weld, and other chemical composition data are from July 1, 1992, letter from G. R. Horn (NPPD) to USNRC Document Control Desk, subject: Response to Generic Letter 92-01, Revision 1.

Plate UUSE data are from Table 7-1 of MDE-103-0986, which evaluated surveillance capsule 1.

[2]Licensee must confirm applicability of Topical Report NEDO-32205, Rev. 1

PRESSURE-TEMPERATURE LIMIT TABLES AND USE TABLES FOR ALL BWR PLANTS

## NOMENCLATURE

Pressure-Temperature Limits 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 7: Unirradiated USE.

EMA
This indicates that the USE issue may be covered by the approved equivalent margins analysis in the BWR Owners Group Topical Report: NEDO-32205, Revision 1.

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. These licensees will utilize Topical Report NEDO-32205, Revision 1 to demonstrate USE compliance to Appendix G, 10 CFR Part 50.

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 the approved equivalent margins analysis in the BWR Owners Group Topical Report: NEDO-32205, Revision 1.

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 reorted in the latest submittal (GL 92-01, PTS, or P/T limits submittals).