Docket file

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

June 24, 1994

Docket Nos. STN 50-454, STN 50-455
and STN 50-456, STN 50-457

Mr. D. L. Farrar
Manager, Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III, Suite 500
1400 OPUS Place
Downers Grove, Illinois 60515

Dear Mr. Farrar:

SUBJECT: GENERIC LETTER (GL) 92-01, REVISION 1, "REACTOR VESSEL STRUCTURAL INTEGRITY," BRAIDWOOD STATION, UNITS 1 AND 2, AND BYRON STATION, UNITS 1 AND 2 (TAC NO.(s) M83436, M83437, M83443, and M83444)

By letters dated July 2, 1992, and November 19, 1993, Commonwealth Edison Company (CECo) provided its response 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 CECo 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 database 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 tables, 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 responses 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 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.

The information requested by this letter is within the scope of the overall burden estimated in GL 92-01, Revision 1, "Reactor Vessel Structural





DF01 1/1

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

Ramin R. Assa, Acting Project Manager
Project Directorate III-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:
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Mr. D. L. Farrar
Commonwealth Edison Company

Byron/Braidwood Power Stations

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ENCLOSURE 1

# 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 |
|---|---|---|---|---|---|---|---|---|---|
| Braidwood 1 | Lower Nozzle Belt Forging | 5P-7016 | 6.82E18 | 10°F | Plant Specific | 26 | Table | 0.04 | 0.71 |
| EOL: 10/17/ 2026 | Upper Shell Forging | 49C344-1-1 /49D383 | 3.03E19 | -30°F | Plant Specific | 31 | Table | 0.05 | 0.73 |
| | Lower Shell Forging | 49D867-1/ 49C813-1 | 3.03E19 | -20°F | Plant Specific | 20 | Table | 0.03 | 0.73 |
| | WF645 Upper Circ. Weld | H4498 | 6.82E18 | -30°F | Plant Specific | 41 | Table | 0.03 | 0.50 |
| | WF562 Middle Circ. Weld | 442011 | 3.03E19 | 40°F | Plant Specific | 41 | Table | 0.03 | 0.65 |
| | WF653 Lower Circ. Weld | 31401 | 1.00E17 | -40°F | Plant Specific | 150.8 | Table | 0.19 | 0.56 |

References

Chemical compositions and initial $RT_{NDT}$ data for all materials are from the July 2, 1992 letter from M.A. Jackson to T.E. Murley, Subject: Braidwood Station, Units 1 and 2.

Fluence data are from WCAP-12685, "Analysis of Capsule U from the Commonwealth Edison Company Braidwood Unit 1 Reactor Vessel Surveillance Program," August, 1990.

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 |
|---|---|---|---|---|---|---|---|---|---|
| Braidwood 2 | Lower Nozzle Belt Forging | 5P-7056 | 6.82E18 | 30°F | Plant Specific | 26 | Table | 0.04 | 0.90 |
| EOL: 12/18/ 2027 | Upper Shell Forging | 490963-1-1 /495904-1-1 | 3.03E19 | -30°F | Plant Specific | 20 | Table | 0.03 | 0.71 |
| | Lower Shell Forging | 500102-1-1 /50C97-1-1 | 3.03E19 | -30°F | Plant Specific | 37 | Table | 0.06 | 0.75 |
| | Upper Circ. Weld | H4498 | 6.82E18 | -30°F | Plant Specific | 41 | Table | 0.03 | 0.50 |
| | Middle Circ. Weld | 442011 | 3.03E19 | 40°F | Plant Specific | 41 | Table | 0.03 | 0.65 |
| | Lower Circ. Weld WF-696 | 1084-18 | 1.00E17 | -16°F | Plant Specific | 54 | Table | 0.04 | 0.6 |

References

Chemical composition and initial $RT_{NDT}$ data for all materials are from the July 2, 1992 letter from M.A. Jackson to T.E. Murley, Subject: Braidwood Station, Unit 1 an 2...

Fluence data are from WCAP-12845, "Analysis of Capsule U from the Commonwealth Edison Company Braidwood Unit 2 Reactor Vessel Surveillance Program," March, 1991.

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 |
|---|---|---|---|---|---|---|---|---|---|
| Byron 1 | Lower Nozzle Belt Forging | 123J218 | 2.159E19 | 30°F | Plant Specific | 31 | Table | 0.05 | 0.72 |
| EOL: 10/31/ 2024 | Int. Shell Forging | 5P-5933 | 2.159E19 | 40°F | Plant Specific | 31 | Table | 0.05 | 0.73 |
| | Lower Shell Forging | 5P-5951 | 2.159E19 | 10°F | Plant Specific | 26 | Table | 0.04 | 0.64 |
| | WF-501 Upper Circ. Weld | 442011 | 2.159E19 | 10°F | Plant Specific | 41 | Table | 0.03 | 0.63 |
| | WF-336 Middle Circ. Weld | 442002 | 2.159E19 | -30°F | Plant Specific | 41 | Table | 0.03 | 0.46 |
| | WF-472 Lower Circ. Weld | 31401 | 1.00E17 | 10°F | Plant Specific | 164.65 | Table | 0.23 | 0.57 |

<u>References</u>

Chemical composition and initial $RT_{ndt}$ data are from the July 2, 1992 letter from M.A. Jackson to T.E. Murley, Subject: Braidwood Station, Unit 1 and 2...

Fluence data are from WCAP-13880, Analysis of Capsule X from the Commonwealth Edison Company Byron Unit 1 Reactor Vessel Surveillance Program," January, 1994.

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 |
|---|---|---|---|---|---|---|---|---|---|
| Byron 2<br><br>EOL: 11/6/ 2026 | Lower Nozzle Belt Forging | 4P-6107 | 6.82E18 | 10°F | Plant Specific | 31 | Table | 0.05 | 0.74 |
| | Upper Shell Forging | 49D329-1-1 /49C297-1-1 | 3.03E19 | -20°F | Plant Specific | 20 | Table | 0.01 | 0.70 |
| | Lower Shell Forging | 49D330-1-1 /49C298 | 3.03E19 | -20°F | Plant Specific | 31 | Table | 0.05 | 0.73 |
| | WF-562 Upper Circ. Weld | 442011 | 6.82E18 | 40°F | Plant Specific | 41 | Table | 0.03 | 0.65 |
| | WF-447 Middle Circ. Weld | 442002 | 3.03E19 | 10°F | Plant Specific | 68 | Table | 0.05 | 0.62 |
| | WF-614 Lower Circ. Weld | 31401 | 1.00E17 | 40°F | Plant Specific | 144.4 | Table | 0.18 | 0.54 |

References

All chemical composition and initial $RT_{ndt}$ data are from the July 2, 1992 letter from M.A. Jackson to T.E. Murley, Subject: Braidwood Station, Units 1 and 2.

Fluence data are from WCAP-12431, "Analysis of Capsule U from the Commonwealth Edison Company Byron Unit 2 Reactor Vessel Surveillance Program," October 1989. EOL USE values for the RG 1.99, Rev. 2 and the lower limiting value of 1.17. Cu for plates and forgings.

Fluence data for weld WF-614 is from the January 17, 1986 letter from G.L Alexander to H.R. Denton, subject: Zion Station Units 1 and 2; Byron Station Units 1 and 2; Braidwood Station Units 1 and 2; Pressurized Thermal Shock.

# 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 |
|---|---|---|---|---|---|---|---|
| Braidwood 1<br><br>EOL: 10/17/ 2026 | Lower Nozzle Belt Forging | 5P-7016 | A 508-3 | 145 | 4.09E18 | 162 | Direct |
| | Upper Shell Forging | 49C344-1-1 /49D383 | A 508-3 | 93 | 1.66E19 | 118 | Direct |
| | Lower Shell Forging | 49D867-1/ 49C813-1 | A 508-3 | 117 | 1.66E19 | 136 | Direct |
| | WF-645 Upper Circ. Weld | H4498 | Linde 80, SAW | 75 | 4.09E18 | 87 | Direct |
| | WF-562 Middle Circ. Weld | 442011 | Linde 80, SAW | 55 | 1.66E19 | 70 | Direct |
| | WF-653 Lower Circ. Weld | 31401 | Linde 80, SAW | 63 | 6.00E16 | 79 | Direct |

**References**

UUSE data for plate 49DC344-1-1/49D383-1-1 are from the July 2, 1992 letter from H.A. Jackson to T.E. Murley, Subject: Braidwood Station, Unit 1 and 2 ...

Fluence data are from WCAP-12685, "Analysis of Capsule U from Commonwealth Edison Company Braidwood Unit 1 Reactor Vessel Surveillance Program," August 1990.

UUSE data for the remaining materials are from the November 19, 1993 letter from T.W. Simpkin to T.E. Murley, "...Braidwood Station Units 1 and 2, Response to Request for Additional Information Regarding NRC Generic Letter 92-01. EOL USE for weld WF-653 was calculated using RG 1.99, Rev. 2 methodology.

# 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 |
|---|---|---|---|---|---|---|---|
| Braidwood 2 | Lower Nozzle Belt Forging | 5P-7056 | A 508-3 | 115 | 4.09E18 | 128 | Direct |
| EOL: 12/18/2027 | Upper Shell Forging | 49D963-1-1 /495904-1-1 | A 508-3 | 94 | 1.66E19 | 119 | Direct |
| | Lower Shell Forging | 50D102-1-1 /50C97-1-1 | A 508-3 | 124 | 1.66E19 | 150 | Direct |
| | Upper Circ. Weld WF645 | H4498 | Linde 80, SAW | 75 | 4.09E18 | 87 | Direct |
| | Middle Circ. Weld WF-562 | 442011 | Linde 80, SAW | 55 | 1.66E19 | 70 | Direct |
| | Lower Circ. Weld WF-696 | 1084-18 | Linde 80, SAW | 63 | 6.00E16 | 78 | Direct |

<u>References</u>

UUSE data for plate 49D963-1-1/495904-1-1 are from the July 2, 1992 letter from M.A. Jackson to T.E. Murley, Subject: Braidwood Station, Units 1 and 2...

Fluence data are from WCAP-12845, "Analysis of Capsule U from the Commonwealth Edison Company Braidwood Unit 2 Reactor Vessel Surveillance Program,' March 1991.

UUSE data for the remaining materials are from the November 19, 1993 letter from T.W. Simpkin to T.E. Murley, "...Braidwood Station units 1 and 2, Response to Request for Additional Information Regarding NRC Generic Letter 92-01. EOL USE for weld WF-696 was calculated using RG 1.99, Rev. 2 methodology.

## 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 |
|---|---|---|---|---|---|---|---|
| Byron 1 | Lower Nozzle Belt Forging | 123J218 | A 508-2 | 111 | 1.179E19 | 138 | Direct |
| EOL: 10/31/2024 | Int. Shell Forging | 5P-5933 | A 508-2 | 111 | 1.179E19 | 138 | Direct |
| | Lower Shell Forging | 5P-5951 | A 508-2 | 120 | 1.179E19 | 150 | Direct |
| | WF-501 Upper Circ. Weld | 442011 | Linde 80, SAW | 63 | 1.179E19 | 73 | Direct |
| | WF-336 Middle Circ. Weld | 442002 | Linde 80, SAW | 60 | 1.179E19 | 74 | Direct |
| | WF-472 Lower Circ. Weld | 31401 | Linde 80, SAW | 64 | 6.00E16 | 72 | Direct |

<u>References</u>

UUSE data for forging 5P-5933 are from the July 2, 199? letter from M.A. Jackson to T.E. Murley, Subject: Braidwood Station, Units 1 and 2.

Fluence data are from WCAP-13880, Analysis of Capsule X from the Commonwealth Edison Company Byron Unit 1 Reactor Vessel Surveillance Program," January 1994.

UUSE data for the welds and initial $RT_{NDT}$ for weld WF-501 are from the November 19, 1993 letter from T.W. Simpkin to T.E. Murley, "...Braidwood Station Units 1 and 2, Response to Request for Additional Information Regarding NRC Generic Letter 92-01. EOL USE for weld WF-472 was calculated using RG 1.99, Rev. 2 methodology. EOL USE for weld WF-336 was calculated using RG 1.99, Rev. 2 assuming the lower limiting value of 0.5% Cu. for welds.

UUSE data for forgings 123J218 and 5P-5951 are from WCAP-11651, "Analysis of Capsule U from the Commonwealth Edison Co. Byron Unit 1 Reactor Vessel Radiation Surveillance Program," November, 1987.

EOL USE values for the forgings were calculated using RG 1.99, Rev. 2 and the lower limiting value of 0.1% Cu for plates and forgings.

# 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 |
|---|---|---|---|---|---|---|---|
| Byron 2 | Lower Nozzle Belt Forging | 4P-6107 | A 508-2 | 131 | 4.09E18 | 155 | Direct |
| EOL: 11/6/ 2026 | Upper Shell Forging | 49D329-1-1 /49C297-1-1 | A 508-2 | 117 | 1.66E19 | 149 | Direct |
| | Lower Shell Forging | 49D330-1-1 /49C298-1-1 | A 508-2 | 99 | 1.66E19 | 127 | Direct |
| | WF-562 Upper Circ. Weld | 442011 | Linde 80, SAW | 60 | 4.09E18 | 70 | Direct |
| | WF-447 Middle Circ. Weld | 442002 | Linde 80, SAW | 53 | 1.66E19 | 67 | Direct |
| | WF-614 Lower Circ. Weld | 31401 | Linde 80, SAW | 67 | 6.00E16 | 74 | Direct |

References

UUSE data for forgings 49D330-1-1/49C298-1-1 are from the July 2, 1992 letter from M.A. Jackson to T.E. Murley, Subject: Braidwood Station, Units 1 and 2.

Fluence data; and UUSE data for forgings 4P-6107 and 49D329-1-1/49C297-1-1 are from WCAP-12431, "Analysis of Capsule U from the Commonwealth Edison Company Byron Unit 2 Reactor Vessel Surveillance Program," October 1989. EOL USE values for the forgings were calculated using RG 1.99, Rev. 2 and the lower limiting value of 0.1% Cu for plates and forgings.

UUSE data for the welds are from the November 19, 1993 letter from T.W. Simpkin to T.E. Murley, "...Braidwood Station Units 1 and 2. Response to Request for Additional Information Regarding NRC Generic Letter 92-01. EOL USE for weld WF-614 was calculated using RG 1.99, Rev. 2 methodology.

Fluence data for weld WF-614 is from the January 17, 1986 letter from G.L Alexander to H.R. Denton, subject: Zion Station Units 1 and 2; Byron Station Units 1 and 2; Pressurized Thermal Shock.

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 the approved equivalent margins analysis in the B&W Owners Group Topical Reports: BAW-2178P and BAW-2192-P.

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 the approved equivalent margins analysis in the B&W Owners Group Topical Reports: BAW-2178P and BAW-2192P.

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.

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:

Ramin R. Assa, Acting Project Manager
Project Directorate III-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

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