Wisconsin Electric
POWER COMPANY

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VPNPD-94-051
NRC-94-037

May 2, 1994

Document Control Desk
U.S. NUCLEAR REGULATORY COMMISSION
Mail Station P1-137
Washington, DC 20555

Gentlemen;

DOCKETS 50-266 AND 50-301
GENERIC LETTER 92-01, REVISION 1,
"REACTOR VESSEL STRUCTURAL INTEGRITY"
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

Nuclear Regulatory Commission (NRC) Generic Letter (GL) 92-01, "Reactor Vessel Structural Integrity," dated March 6, 1992, was issued to obtain information from licensees to enable the NRC to assess compliance with regulatory requirements and commitments regarding reactor vessel integrity. Our response to GL 92-01 was provided to the NRC on June 25, 1992, and supplemented on July 30, 1992, and November 1, 1993 (VPNPD-93-186). On March 17, 1994, the NRC requested that Wisconsin Electric (WE) confirm the applicability and accuracy of information previously provided by WE and the Babcox & Wilcox Owners Group. This letter provides our response to your request.

In letters dated May 21, 1993, and November 1, 1993 (VP-NPD-93-185), WE referred to B&W Nuclear Technologies (BWNT) topical reports BAW-2178P and BAW-2192P, respectively, for Point Beach Units 1 and 2. These topical reports were previously submitted to the NRC by BWNT on behalf of the B&W Owners Group. We have reviewed and confirmed the accuracy of the limiting material properties in these reports that applies to the Point Beach units. However, we noted during our review that the cold leg temperature for the Point Beach units was incorrectly listed as 552.5°F in BAW-2178P instead of 542°F. Wisconsin Electric has confirmed with BWNT that Point Beach Units 1 and 2 remain bounded by the fracture mechanics analyses presented in BAW-2178P at this lower operating temperature (Attachment 1). The introduction to the final version of BAW-2178P will include reference to the correct cold leg temperature as shown in Attachment 1. We request that the NRC review and approve the final versions of BAW-2178P and BAW-2192P as the basis for demonstrating compliance with 10 CFR Part 50, Appendix G, Paragraph IV.A.1 for Point Beach Units 1 and 2.

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Wisconsin Electric has also reviewed the information contained in Enclosures 1 and 2 of your March 17, 1994, letter. We have confirmed the accuracy of the material chemistry and property data as presented in your enclosures. However, since the time of the submittal of our response to GL 92-01, we have performed a further analysis of reactor vessel dosimetry as part of our reactor cavity neutron monitoring program. The results of this analysis (Westinghouse WCAP-12794, Rev. 2 and WCAP-12795, Rev. 2) provided updated fluence values that differ slightly from those provided in Enclosures 1 and 2 of your letter. The most current end-of-license fluence projections for Point Beach Units 1 and 2 are presented in Attachment 2.

Please contact us if you have any questions or require additional information regarding this response.

Sincerely,

Bob Link
Vice President
Nuclear Power

JRP/kmc

Attachments:
1) BWNT Letter to Wisconsin Electric dated 4/25/94
2) Fluence Predictions for the Point Beach Units at 32 EFPY

cc: NRC Regional Administrator, Region III
NRC Resident Inspector

BW

B&W NUCLEAR TECHNOLOGIES

3315 Old Forest Road
P.O. Box 10935
Lynchburg, VA 24506-0935
Telephone: 804-385-2000
Telecopy: 804-385-3663

April 25, 1994

NIMS
FILE A3.6

JRP

Mr. J. R. Pfefferle
Wisconsin Electric Power Company
231 W. Michigan St.
P. O. Box 2046
Milwaukee, WI 43201

Dear Jim:

In BAW-2178P report, the cold leg temperatures of Point Beach Units 1 and 2 are listed as 552.5F referring to BAW-1543, Rev.3. According to Rev.4 of BAW-1543, these temperatures should be 542F. This correction is made to the approved version of this report as shown in the attached. This temperature doe not affect the results of this analysis since Point Beach vessels were not used for a lower-bounding analysis.

Attached page shows an addition to the introduction of BAW-2178P. This will be issued as BAW-2178PA as the approved version of the report showing the NRC SER in the report.

Sincerely,

Ken Yoon

Kenneth K. Yoon

Attachment:

APR 2 9 1994

Addition to the end of INTRODUCTION section on page 1-1.

Following receipt of the NRC SER, this report is reissued as BAW-2178PA with a typographical error correction in paragraph 4.1 and the changes in cold leg temperature column of Table 5-1 to reference a later version of BAW-1543 (Revision 4), to be consistent with BAW-2192PA. These changes do not affect the results of this analysis.

Fluence Predictions for Point Beach Unit 1 at 32 EFPY

Beltline Identification	Heat No. Identification	ID Neutron Fluence at EOL (n/cm^2)	1/4T Neutron Fluence at EOL (n/cm^2)
Nozzle Belt Forging	122P237VA1	3.17E18	2.15E18
Int. Shell Plate	A-9811-1	2.78E19	1.88E19
Lower Shell Plate	C-1423-1	2.43E19	1.65E19
Int. Shell Axial Welds SA-812	1P0815	1.78E19	1.21E19
Lower Shell Axial Welds SA-847	61782	1.63E19	1.10E19
Circ. Weld SA-1101	71249	2.43E19	1.65E19
Nozzle Belt to Int. Shell Circ. Weld SA-1426	8T1762	3.17E18	2.15E18

Fluence Predictions for Point Beach Unit 2 at 32 EFPY

Beltline Identification	Heat No. Identification	ID Neutron Fluence at EOL (n/cm^2)	1/4T Neutron Fluence at EOL (n/cm^2)
Nozzle Belt Forging	123V352VA1	3.70E18	2.51E18
Int. Shell Forging	123V500VA1	2.88E19	1.95E19
Lower Shell Forging	122W195VA1	2.62E19	1.77E19
Int./Lower Shell Weld SA-1484	72442	2.52E19	1.71E19
Nozzle Belt to Int. Shell Circ. Weld	CE Weld	3.70E18	2.51E18

Arneson, Kriesel, Link, R Newton,
Ulmer
file

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

March 17, 1994

Associated w/
GL 92-01-01

~~DAW~~
~~IGM~~
~~EAB~~
~~FDP~~
KVA - last

Docket Nos. 50-266
and 50-301

Mr. Robert E. Link, Vice President
Nuclear Power Department
Wisconsin Electric Power Company
231 West Michigan Street, Room P379
Milwaukee, Wisconsin 53201

Dear Mr. Link:

SUBJECT: GENERIC LETTER (GL) 92-01, REVISION 1, "REACTOR VESSEL STRUCTURAL INTEGRITY," POINT BEACH, UNITS 1 AND 2, (TAC NOS. M83737 AND M83738)

By letter dated June 25, 1992, as supplemented July 30, 1992, and November 1, 1993, you provided your response to GL 92-01, Revision 1. The NRC staff has completed its review of your response and determined that: (1) based on the available surveillance data, the Point Beach reactor vessels will be below the pressurized thermal shock (PTS) screening criteria in 10 CFR 50.61 when their operating licenses expire; and (2) based on the analyses in BAW-2192 and BAW-2178P, the Point Beach reactor vessels will be able to satisfy the upper shelf energy (USE) requirements of 10 CFR 50, Appendix G, throughout the term of their operating licenses.

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 and permittees 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 PTS table for PWRs, a pressure-temperature limit table for BWRs and a USE table for PWRs and BWRs. Enclosure 1 provides the PTS and/or pressure temperature 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, pressure-temperature limit, and RT_{pts} evaluations. These data were taken from your response to GL 92-01 and previously docketed information. The information in the RVID for your facilities will be considered accurate at this point in time, and will be used in the staff's assessments related to vessel structural integrity. References to the specific source of the data are provided in the tables.

We request that you confirm within 30 days the plant-specific applicability of the NRC staff approved revisions of the topical reports BAW-2192 and BAW-2178P, and request review and approval in accordance with 10 CFR Part 50, Appendix G. This review will be a plant-specific action. We further request that you verify that the information you have provided for your facilities 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 your confirmation is received, the staff will consider your actions related to GL 92-01, Revision 1, to be complete.

GL 92-01/4

If you desire to use surveillance data from another plant to determine the chemistry factor and margin value for welds in your beltline, you must compare the actual beltline irradiation temperatures (cold leg temperatures) of the two plants to determine whether a temperature correction is required. In addition, the data must: (1) be from a weld fabricated using weld wire with the same heat number and with the same type of flux as the beltline weld; and, (2) meet the credibility criteria of Regulatory Guide 1.99, Revision 2 (the scatter of the data about the best-fit line should normally be less than 28 °F).

The information requested by this letter is within the scope of the overall burden estimated in Generic Letter 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,

Allen G. Hansen, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV/V
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

Mr. Robert E. Link
Wisconsin Electric Power Company

Point Beach Nuclear Plant
Unit Nos. 1 and 2

cc:

Ernest L. Blake, Jr.
Shaw, Pittman, Potts & Trowbridge
2300 N Street, N.W.
Washington, DC 20037

Mr. Gregory J. Maxfield, Manager
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Town Chairman
Town of Two Creeks
Route 3
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Chairman
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Resident Inspector's Office
U.S. Nuclear Regulatory Commission
6612 Nuclear Road
Two Rivers, Wisconsin 54241

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
Point Beach 1 EOL: 10/5/2010	Nozzle Belt Forging	122P237VA1	2.95E18	50°F	Plant Specific	115.2	Table	0.11	0.82
	Int. Shell Plate	A-9811-1	2.68E19	1°F	Generic	92.556	Calculated	0.20	0.056
	Lower Shell Plate	C-1423-1	2.33E19	1°F	Generic	47.466	Calculated	0.12	0.065
	Int. Shell Axial Welds SA-812	1P0815	1.71E19	-5°F	Generic	138.2	Table	0.17	0.52
	Lower Shell Axial Welds SA-847	61782	1.56E19	-5°F	Generic	167.6	Table	0.25	0.54
	Circ. Weld SA-1101	71249	2.33E19	10°F	Plant Specific	180.0	Table	0.26	0.60
	Nozzle Belt to Int. Shell Circ. Weld SA-1426	8T1762	2.95E18	-5°F	Generic	152.25	Table	0.20	0.55

References

Chemical composition, fluence, and IRT_{ndt} data are from June 25, 1992, letter from B. Link (WEPCo) to USNRC Document Control Desk, subject: Response to Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity, 10 CFR 50.54(f)

IRT_{ndt} for plates A-9811-1 and C-1423-1 were determined per pp 3-18 of BAW 10042P. The values are conservative relative to MTEB 5-2.

Percent copper for Nozzle Belt. Forging determined by the NRC staff from data from similar forgings in letter dated November 1, 1993. The value is a tolerance limit with 95 percent confidence that at least 95 percent of the population is less than the tolerance limit (TL).

$$(TL = \overline{X}+K\sigma, \text{ where } \overline{X}=0.06,\ K=3.187,\ \sigma=0.015)$$

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
Point Beach 2 EOL: 3/8/2013	Nozzle Belt Forging	123V352VA1	3.50E18	40°F	Plant Specific	113.25	Table	0.11	0.73
	Int. Shell Forging	123V500VA1	2.92E19	40°F	Plant Specific	49.771	Calculated	0.09	0.70
	Lower Shell Forging	122W195VA1	2.66E19	40°F	Plant Specific	28.63	Calculated	0.05	0.72
	Int./Lower Shell SA-1484	72442	2.56E19	-5°F	Generic	173	Table	0.24	0.60
	Nozzle Belt to Int. Shell Circ. Weld	CE Weld	3.50E18	-56°F	Generic	232.5	Table	0.27	0.90

References

Chemical composition, fluence, and IRT_{ndt} data are from June 25, 1992, letter from B. Link (WEPCo) to USNRC Document Control Desk, subject: Response to Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity, 10 CFR 50.54(f)

Percent copper for Nozzle Belt. Forging determined by the NRC staff from data from similar forgings in letter dated November 1, 1993. The value is a tolerance limit with 95 percent confidence that at least 95 percent of the population is less than the tolerance limit (TL).

$$(TL = \overline{X} + K\sigma,\ \text{where } \overline{X}=0.06,\ K=3.187,\ \sigma=0.015)$$

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
Point Beach 1 EOL: 10/5/2010	Nozzle Belt Forging	122P237VA1	A 508-2	51	2.0E18	59	NRC Generic
	Int. Shell	A-9811-1	A 302B	54	1.81E19	70	65%
	Lower Shell	C-1423-1	A 302B	60	1.58E19	77	65%
	Int. Shell Axial Welds SA-812	1P0815	Linde 80, SAW	EMA[2]	1.16E19	EMA[2]	Generic
	Lower Shell Axial Welds SA-847	61782	Linde 80, SAW	EMA[2]	1.06E19	EMA[2]	Generic
	Circ. Weld SA-1101	71249	Linde 80, SAW	EMA[2]	1.58E19	EMA[2]	Generic
	Nozzle Belt/ Int. Shell Circ. Weld SA-1426	8T1762	Linde 80, SAW	EMA[2]	2.0E18	EMA[2]	Generic

References

% Drop in USE for plate A-9811-1 determined from surveillance data in accordance with RG 1.99, Rev. 2, paragraph 2.2.

Chemical composition and fluence data are from June 25, 1992, letter from B. Link (WEPCo) to USNRC Document Control Desk, subject: Response to Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity, 10 CFR 50.54(f)

UUSE data for weld SA-847 are from June 25, 1992, letter from B. Link (WEPCo) to USNRC Document Control Desk, subject: Response to Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity, 10 CFR 50.54(f)

UUSE for plate C-1423 and A-9811-1 are from WCAP-10736

UUSE data for other welds are from BAW-1803, Revision 1

UUSE for Nozzle Belt. Forging determined by using 65 percent correction factor on data from similar forging reported in a letter dated November 1, 1993 from B. Link (WEPCo) to USNRC. The UUSE value is a tolerance limit with 95 percent confidence that at least 95 percent of the population is greater than the tolerance limit (TL)

($TL = \overline{X}-K\sigma$ where: $\overline{X}=99$, $\sigma = 12.69$, $K = 3.187$)

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[2]Licensee must confirm applicability of Topical Reports BAW 2192 and BAW 2178P.

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
Point Beach 2 EOL: 3/8/2013	Nozzle Belt Forging	123V352	A 508-2	74	2.37E18	89	65%
	Int. Shell	123V500VA1	A 508-2	108	1.98E19	117	65%
	Lower Shell	122W195VA1	A 508-2	85	1.80E19	94	65%
	Circ. Weld SA-1484	72442	Linde 80, SAW	EMA[2]	1.73E19	EMA[2]	Generic
	Nozzle Belt/ Int. Shell Circ. Weld	Not provided	No data available	53	2.37E18	75	NRC Generic

References

Chemical composition and fluence data are from June 25, 1992, letter from B. Link (WEPCo) to USNRC Document Control Desk, subject: Response to Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity, 10 CFR 50.54(f)

UUSE data for weld SA-1484 are from June 25, 1992, letter from B. Link (WEPCo) to USNRC Document Control Desk, subject: Response to Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity, 10 CFR 50.54(f)

UUSE for forging 122W195VA1 and 123V500VA1 is from BAW-2140, which analyzed capsule S

UUSE for forging 123V352 reported in a letter dated November 1, 1993 from B. Link (WEPCo) to USNRC.

UUSE for Nozzle Belt/Int. Shell Circ. Weld is the NRC staff value for Combustion Engineering fabricated welds that was reported in a letter dated December 3, 1993 to T.L. Patterson (OPPD) from S. Bloom (USNRC).

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[2]Licensee must confirm applicability of Topical Reports BAW 2192 and BAW 2178P

PRESSURIZED-THERMAL SHOCK TABLES AND USE TABLES

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 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 ASME 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 were 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, Rev. 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, Rev. 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 copper 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.

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.

Column 8: Method of determining unirradiated USE

<u>Direct</u>

This indicates that the unirradiated USE was from a transverse specimen

<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>Equi. to Surv. Weld</u>

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

This indicates that there is insufficient data to determine the unirradiated USE. These licensees may need to utilize an approved equivalent margins analysis to demonstrate USE compliance to Appendix G, 10 CFR 50.

4/14/94

TO: MIKE BAUMANN

FROM RICH LAUFER

These two sentences more clearly describe what was intended in the first sentence on the second page of our 3/17/94 GL92-01 letter.

We request that, within 30 days, you provide confirmation of the plant-specific applicability of the Babcock & Wilcox topical reports BAW-2178P and BAW-2192P and submit a request for approval of the topical reports as the basis for demonstrating compliance with 10 CFR Part 50, Appendix G, Paragraph IV.A.1. To demonstrate that the topical reports are applicable to [redacted], you must compare the limiting material properties of the [redacted] reactor vessels to the values reported in the topical reports.

POINT BEACH UNITS 1+2