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July 6, 2015
L-15-191

ATTN: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT:
Beaver Valley Power Station, Unit No. 1
Docket No. 50-334, License No. DPR-66
Generic Letter 92-01 Response Clarification (TAC No. M83431)

This letter provides information to clarify and correct the July 8, 1992 response to Generic Letter 92-01, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f)," Revision 1, for Beaver Valley Power Station, Unit No. 1 (BVPS-1). The clarification and corrections are described in the attachment and do not affect the conclusions discussed in the July 8, 1992 response letter.

There are no regulatory commitments contained in this submittal. If there are any questions or additional information is required, please contact Mr. Thomas A. Lentz, Manager - Fleet Licensing, at (330) 315-6810.

I declare under penalty of perjury that the foregoing is true and correct. Executed on July 6, 2015.

Sincerely,

E. A. L

Eric A. Larson

Attachment: Generic Letter 92-01 Response Clarification for BVPS-1

cc: NRC Region I Administrator
NRC Resident Inspector
NRC Project Manager
Director BRP/DEP
Site Representative BRP/DEP

ATTACHMENT
L-15-191

Generic Letter 92-01 Response Clarification for BVPS-1

Section 50.60(a) of Title 10 of the Code of Federal Regulations (10 CFR 50.60(a)) requires that all light water nuclear power reactors, licensed by the Nuclear Regulatory Commission (NRC), meet fracture toughness requirements and have a material surveillance program for the reactor pressure boundary. These requirements are set forth in Appendices G and H to 10 CFR Part 50. 10 CFR 50.61 provides fracture toughness requirements for protecting pressurized water reactors against pressurized thermal shock events. Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations," was issued to call attention to Revision 2 of Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials."

Generic Letter (GL) 92-01, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f)," Revision 1, states in part that:

> This generic letter is part of a program to evaluate reactor vessel integrity and take regulatory actions, if needed, to ensure that licensees and permit holders are complying with the requirements of 10 CFR 50.60 and 10 CFR 50.61, and are fulfilling commitments made in response to GL 88-11.

A discrepancy was noted in the July 8, 1992 response to GL 92-01, Revision 1, for BVPS-1 (hereafter referred to as the BVPS-1 response). Following investigation it was found that clarification of the methodology for determining the reactor vessel material properties and other corrections of the BVPS-1 response were needed. The five findings and resulting response changes are described below.

1. Materials Certification Information for
   Table 3, Intermediate Shell B6607-2
   Table 4, Lower Shell B7203-2

The data associated with two sets of three Charpy V-notch tests performed at room temperature (RT) (one set for each material listed above) was not included in tables 3 and 4 of the BVPS-1 response. The "Charpy Impact and Fracture Tests – Transverse," information in Table 3 (page 7), and Table 4 (page 8) of the BVPS-1 response is revised as shown below to include this additional test data. In addition, the lateral expansion value in Table 3 for the final test performed at a temperature of 210 degrees Fahrenheit (°F) was documented incorrectly as 65 mils, but should have been documented as 67 mils based on the original test data sheets. The "Charpy Impact and Fracture Tests – Transverse," information in tables 3 and 4 of the BVPS-1 response is hereby replaced with the information shown below. Side bars are provided to indicate the areas of change.

Table 3, Charpy Impact and Fracture Tests – Transverse

Temp °F	Ft-Lbs	% Shear	Lat. Exp. (mils)
-40	13.5	10	6
-40	15.5	10	8
-40	18	10	10
0	30	20	20
0	19	15	11
0	17	15	12
50	26.5	30	18
50	27	30	22
50	29	30	22
RT	40	40	32
RT	42	40	33
RT	43	40	33
110	49	50	40
110	44	50	38
110	52	50	42
160	71	85	57
160	60	85	52
160	76	90	63
210	81	100	65
210	86	100	70
210	81	100	67

Temp °F – Temperature in degrees Fahrenheit
Ft-Lbs – Foot-Pounds
% Shear – Percent Shear
Lat. Exp. – Lateral Expansion
RT – Room Temperature

Table 4, Charpy Impact and Fracture Tests – Transverse

Temp °F	Ft-Lbs	% Shear	Lat. Exp (mils)
-40	26	10	17
-40	13	10	9
-40	12	10	6
0	27	30	23
0	20	30	16
0	19	30	17
40	34	35	30
40	37	35	32
40	54	40	42
RT	59	55	49
RT	58	55	50
RT	60	55	49
110	79	75	64
110	66.5	70	54
110	72.5	80	58
160	86.5	100	71
160	81.5	100	67
160	86	100	67
210	91	100	72
210	75	100	62
210	81	100	67

Temp °F – Temperature in degrees Fahrenheit
Ft-Lbs – Foot-Pounds
% Shear – Percent Shear
Lat. Exp. – Lateral Expansion
RT – Room Temperature

2. Materials Certification Information for
   Table 6, Intermediate Shell Seams 19-714 A and B
   Table 7, Intermediate Shell to Lower Shell Seam 11-714
   Table 8, Lower Shell Seams 20-714 A and B
   Table 9, Intermediate Shell to Lower Shell Seam 11-714

The materials certification information for the welds listed above are included in tables 6 through 9 of the BVPS-1 response. Intermediate shell to lower shell seam 11-714 appears twice since there were two separate flux lots used to create the weld. In the tables showing the weld material, the reference temperature for nil-ductility transition (RTNDT) used for each material is shown as -56°F with an asterisk. The asterisk states, "Estimated per NRC Regulatory Review Plan [Materials Engineering Branch Technical Position] MTEB 5-2." The -56°F value was used as a generic estimate of the RTNDT for welds in Combustion Engineering vessels, but was not referenced in MTEB 5-2. The source of this -56°F value was identified as 10 CFR 50.61 in the last bullet on page 18 of the BVPS-1 response. Paragraph 10 CFR 50.61(c)(1)(ii) states that:

> For generic values of weld metal, the following generic mean values must be used unless justification for different values is provided: 0 °F for welds made with Linde 80 flux, and -56°F for welds made with Linde 0091, 1092 and 124 and ARCOS B-5 weld fluxes.

Intermediate shell seams 19-714 A and B were fabricated with Linde 1092 weld flux, intermediate shell to lower shell seam 11-714 was fabricated with Linde 0091 weld flux, and lower shell seams 20-714 A and B were fabricated with Linde 1092 weld flux. Therefore, the note at the bottom of tables 6 through 9 in the BVPS-1 response is hereby corrected as follows to state that the -56°F value is a generic mean value in accordance with 10 CFR 50.61(c)(1)(ii).

* Generic mean value in accordance with 10 CFR 50.61(c)(1)(ii)

The NRC's June 9, 1994 letter included a table of information from the Reactor Vessel Integrity Database that was based on the responses to GL 92-01 for BVPS-1 and previously docketed information. The table of information, and thus the Reactor Vessel Integrity Database, state that the RTNDT values for the beltline welds were "Generic" meaning from the generic Combustion Engineering values. Although the note referencing MTEB 5-2 as the source of the -56°F RTNDT value in tables 6 through 9 is incorrect, the source of this value is correctly documented in the Reactor Vessel Integrity Database.

3. Materials Certification Information for
   Table 2, Intermediate Shell B6607-1
   Table 10, Surveillance Plate (Vessel Lower Shell) B6903-1
   Table 11, Surveillance Weldment
   Table 12, Surveillance Weld Heat-Affected Zone Material

The BVPS-1 response includes charpy impact and fracture test data in tables 2, 10, 11, and 12. These tables include column headings that indicate lateral expansion values

are shown in inches. These tables should indicate that the lateral expansion values are shown in mils.

Therefore, column headings in tables 2, 10, 11, and 12 of the BVPS-1 response are hereby changed to indicate that the lateral expansion values are shown in mils.

4. Materials Certification Information for
   Table 5, Lower Shell Plate B6903-1
   Table 10, Surveillance Plate (Vessel Lower Shell) B6903-1

Charpy V-notch data for the limiting plate material, lower shell plate B6903-1, is documented in tables 5 and 10 of the BVPS-1 response. Table 10 of the BVPS-1 response has 12 additional data points for this material as compared to Table 5. The full Charpy V-notch data set summarized in Table 10 of the BVPS-1 response was used to establish the unirradiated RTNDT value for lower shell plate B6903-1. The original test report indicates that the lateral expansion value for the first test performed at a temperature of 210°F is 70 mils. This value was documented incorrectly in Table 10 as 80 mils. This value was correctly reported in Table 5 of the BVPS-1 response.

Based on the foregoing, the lateral expansion value documented in Table 10 of the BVPS-1 response for the first test performed at a temperature of 210°F is hereby changed from 80 mils to 70 mils.

5. Determination of Unirradiated RTNDT for Beltline Region Materials
   Page 18

Subarticle NB-2331 of American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section III requires both drop-weight test data and Charpy V-notch test data from transverse specimens for determination of unirradiated RTNDT values. This methodology is paraphrased in the first bullet on Page 18 of the BVPS-1 response.

The appropriate test data necessary to use these ASME Code requirements for determination of unirradiated RTNDT values was obtained and documented in the BVPS-1 response for beltline plate materials (Intermediate Shell Plates B6607-1 and B6607-2 and Lower Shell Plates B6903-1 and B7203-2). However, even though drop-weight tests were performed and summarized in the BVPS-1 response for each of the four beltline plate materials, the second bullet on Page 18 of the BVPS-1 response implies that the methodology used in establishing the unirradiated RTNDT values was based on the 30 foot-pound (ft-lb) temperatures from each of the full Charpy V-notch curves. When drop-weight test data is not available, this approach is consistent with the methodology outlined in MTEB 5-2 for SA-533 Grade B, Class 1 plate materials.

Enclosure 1 of the NRC's June 9, 1994 letter regarding the response to GL 92-01 for Beaver Valley Power Station, Unit No. 1, indicated that the method of determining unirradiated RTNDT (identified in Enclosure 1 as IRT_{ndt}) values for the reactor vessel beltline plate materials B6607-1, B6607-2, B6903-1 and B7203-2 was the MTEB 5-2

methodology. These references are also included in the NRC's Reactor Vessel Integrity Database (RVID).

The methodology used to determine the unirradiated RTNDT values for BVPS-1 reactor vessel beltline plate materials (Intermediate Shell Plates B6607-1 and B6607-2 and Lower Shell Plates B6903-1 and B7203-2) was consistent with the methodology described in ASME Code Section III, Subarticle NB-2331. Therefore the second paragraph and first two bullets on page 18 of the BVPS-1 response are hereby replaced with the following text.

> The unirradiated RTNDT of the beltline region materials was established from the drop weight NDTT tests and the Charpy V-notch tests, using the guidance provided in Subarticle NB-2300 of the ASME Boiler and Pressure Vessel Code, Section III.
>
> - The NDTT temperature, as determined by drop weight tests (ASTM E-208), is RT_{NDT} if, at 60° F above the NDTT, at least 50 ft-lbs of energy and 35 mils lateral expansion are obtained in Charpy V-notch tests on transverse specimens. Otherwise, the RT_{NDT} is the temperature at which 50 ft-lbs and 35 mils lateral expansion are obtained on transverse Charpy specimens, minus 60°F. These criteria were applied in determining the initial RT_{NDT} values for the surveillance plate and weldment.