Omaha Public Power District
444 South 16th Street Mall
Omaha, Nebraska 68102-2247
402/636-2000

April 5, 1995
LIC-95-0086

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-137
Washington, DC 20555

SUBJECT: Fort Calhoun Station Reactor Vessel Material Data

References:
1. Docket No. 50-285
2. Letter from OPPD (W. G. Gates) to NRC (Document Control Desk) dated September 17, 1993 (LIC-93-0237)
3. Letter from OPPD (W. G. Gates) to NRC (Document Control Desk) dated October 15, 1993 (LIC-93-0258)
4. Letter from OPPD (W. G. Gates) to NRC (Document Control Desk) dated October 15, 1993 (LIC-93-0260)
5. Letter from NRC (W. T. Russell) to OPPD (T. L. Patterson) dated March 14, 1995

This letter provides the Omaha Public Power District (OPPD) response to the Reference 5 letter from the NRC. Reference 5 notified OPPD that the information previously considered proprietary and submitted via References 2 and 3 would be released to the public unless, within 30 days, OPPD requested withdrawal of that information and the request was approved. The information affected also includes corrections submitted by Reference 4.

OPPD has received permission from the owner of the affected information (ABB/Combustion Engineering) to release the information as non-proprietary. Enclosures 1 and 2 of this letter provide the same information, in non-proprietary format, as that previously submitted in enclosures to References 2, 3, and 4. (The corrections submitted by Reference 4 are included in Enclosure 1.) This is the same information used by the NRC as a basis for review and approval of Amendment No. 158 to Fort Calhoun Station Facility Operating License DPR-40.

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OPPD therefore requests that all of the previously submitted enclosures of References 2, 3, and 4 be withdrawn from the docket. Please contact me if you have any questions.

Sincerely,

T. L. Patt

T. L. Patterson
Division Manager
Nuclear Operations

TLP/tcm

Enclosures

c: LeBoeuf, Lamb, Greene & MacRae (w/o attachments)
L. J. Callan, NRC Regional Administrator, Region IV (w/o attachments)
S. D. Bloom, NRC Project Manager
R. P. Mullikin, NRC Senior Resident Inspector (w/o attachments)

LIC-95-0086
ENCLOSURE 1

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION
REACTOR VESSEL STRUCTURAL INTEGRITY
FORT CALHOUN STATION UNIT NO. 1

(Previously submitted in LIC-93-0237 and LIC-93-0260)

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION
REACTOR VESSEL STRUCTURAL INTEGRITY
FORT CALHOUN STATION UNIT NO. 1

NRC Request Concerning Question 2a in Generic Letter 92-01

The response indicates that the initial upper shelf energy (USE) values for all beltline welds, except for the surveillance weld, are not known. Either provide the Charpy USE for each beltline weld with no documented initial USE value or provide the Charpy USE and analysis from welds that were fabricated using the same vendor, fabrication time frame, fabrication process, and material specification to demonstrate that all beltline welds with no documented initial USE values will meet the USE requirements of Appendix G, 10 CFR 50. If this information cannot be provided, then submit an analysis which demonstrates that lower values of USE will provide margins of safety against fracture equivalent to those required by Appendix G of the ASME Code.

Omaha Public Power District (OPPD) Response

This response provides the requested USE data for beltline welds at Fort Calhoun Station Unit No. 1 (FCS). References noted in this discussion are listed on the last page.

All information pertaining to weld identification was obtained from the materials information submitted to the NRC by OPPD (Reference 1). Weld copper content was also taken from the response to Generic Letter (GL) 92-01, Revision 1 with the exception of weldment 3-410 which incorporates modified and averaged chemical compositions based on individual weld wire heat information (Reference 2).

The USE data for the FCS reactor beltline welds were taken from a variety of sources with a hierarchy as specified in Reference 3. The hierarchy is as follows:

1. Certified Material Test Reports (CMTRs) - When CMTRs were available for the same material heat(s) and flux type, but not necessarily the same flux lot, USE values were generated from ASTM standards (Reference 4).

2. ORNL PR-EDB - If CMTRs were not available on the material, a search was performed for surveillance material with the same heat(s) and flux type, but not necessarily the same flux lot, using individual plant responses to GL 92-01 and reported data in the NRC surveillance capsule test results database for vessels fabricated by Combustion Engineering (CE) (Reference 5).

3. Generic Flux Type - When no specific USE data values were available, USE values were obtained from a statistical average of reported USE data for a particular flux type from CE-fabricated vessel welds. The reported USE data were taken from sources such as the ORNL PR-EDB and prior documentation.

Each of the three beltline welds is addressed below:

Weld 2-410

The 2-410 axial (longitudinal) weld was fabricated using a Mil B-4 type wire (heat number 51989) and a Linde 124 flux (lot 3687). The USE value for wire heat 51989 was derived from a statistical average for 19 welds fabricated with Linde 124 flux. USE values come from two sources. The first source comprised 16 USE values from CMTRs, compiled in a calculation (performed by ABB Combustion Engineering for Calvert Cliffs Unit 2) which has been submitted to the NRC by Baltimore Gas & Electric Co. as proprietary information. The second source comprised three additional data points from Wolf Creek, St. Lucie 2 and Callaway 1 (Reference 5). Individual USE values, along with the average value, are summarized in the following Table 1.

TABLE 1: Summary of Initial USE for Linde 124 Flux Welds

Number	Wire Heat/Flux Lot No.	Initial USE (ft-lbs)
1	89408/0751	111
2	3P7246/0951	106
3	3P7317/0951	99
4	ES6906/0662	89
5	91762/0662	88
6	4P7927/0662	114
7	4P7869/1061	97
8	5P8866/1061	107
9	69025/1061	89
10	89833/1061	96
11	90144/1061	91
12	3P7802/0171	109
13	651A708/0871	94
14	4P8632/0281	107
15	3P8013/0281	92
16	LP5P9744/0281	108
17	90146/1061	100
18	83637/0951	115
19	90077/1061	112
Average		101.3

Weld 3-410

The 3-410 axial (longitudinal) weld was fabricated using a Mil B-4 modified type of wire (heat numbers 12008, 13253, and 27204) with a Linde 1092 flux (lot 3774). A tandem arc process, utilizing two weld wires, was used in the fabrication of the FCS reactor vessel. Based on extensive review of CMTRs, no record was found indicating the use of wire heat number 12008 in combination with itself, or wire heat number 13253 in combination with wire heat number 27204 during the welding process. Therefore, these two combinations were excluded from consideration. The four possible combinations evaluated were:

(1) Wire heat number 13253 combined with wire heat number 13253

(2) Wire heat number 27204 combined with wire heat number 27204

(3) Wire heat number 13253 combined with wire heat number 12008

(4) Wire heat number 27204 combined with wire heat number 12008

Each of these evaluations is summarized below.

(1) Wire Heat Number 13253

The USE data for weld wire heat number 13253 are from two separate CMTRs for D.C. Cook 1 and Salem 2. The D. C. Cook value of 110 ft-lbs represents an average of two Charpy tests at 110°F (104 ft-lbs exhibiting 95% shear, 106 ft-lbs exhibiting 99% shear) and two Charpy tests at 160°F (114 ft-lbs exhibiting 100% shear and 117 ft-lbs exhibiting 100% shear). The Salem 2 USE value of 111 ft-lbs represents an average of three Charpy tests with 100% shear at 100°F (105, 108, and 119 ft-lbs). The more conservative value of 110 ft-lbs was used in the FCS USE evaluation (Reference 6).

(2) Wire Heat Number 27204

The USE data for weld wire heat number 27204 are from reported values for Diablo Canyon 1. A 94 ft-lbs value was reported in response to GL 92-01 (Reference 7) and a value of 98 ft-lbs was reported in the NRC database (Reference 5). The more conservative value of 94 ft-lbs was used in the FCS USE evaluation (Reference 6).

(3) Wire Heat Numbers 13253/12008

The USE value, 112 ft-lbs, for the weld wire heat numbers 13253/12008 combination was derived from a statistical average of 13 USE values for Linde 1092 flux welds as reported in Reference 5. Individual USE values, along with the average value, are given in the following Table 2.

TABLE 2: Summary of Initial USE for Linde 1092 Flux Welds		
Number	Wire Heat/Flux Lot No.	Initial USE (ft-lbs)
1	305424/3889	112
2	27204/3714	98
3	27204/3869	121
4	305414/3951	104
5	W5214/3617	112
6	W5214/3600	118
7	W5214/3692	120
8	1P3571/3958	126
9	12008/3854	112
10	1P3571/3958	105
11	3277/3833	118
12	39B196/3617	104
13	13253/3833	111
Average		112

(4) <u>Wire Heat Numbers 27204/12008</u>

USE data for the weld wire heat numbers 27204/12008 combination are from a CMTR for Diablo Canyon 1. A value of 85.5 ft-lbs was determined from the average of two Charpy tests exhibiting 100% shear at 110°F (86 ft-lbs and 85 ft-lbs).

<u>Weld 9-410</u>

The 9-410 circumferential weld was fabricated using a Mil B-4 modified wire (heat number 20291) with a Linde 1092 flux (lot 3833). The USE data for weld heat number 20291 are from a CMTR for McGuire 1 for a tandem arc weld using heat numbers 20291/12008. The average of two Charpy tests at 160°F (111 ft-lbs exhibiting 100% shear and 110 ft-lbs exhibiting 100% shear) and one Charpy test at 110°F (93 ft-lbs exhibiting 95% shear) is 105 ft-lbs. USE values reported to the NRC (References 8 and 9) show a shelf energy of 112 ft-lbs for the same wire heat combination. The more conservative value of 105 ft-lbs was used in the FCS USE evaluation (Reference 6).

Summary

Table 3, which is consistent with the data above, duplicates the USE summary information previously transmitted to the NRC (Reference 6).

TABLE 3: Upper Shelf Energy Summary for Weld Material (August 2013)

Weld No.	Heat No.	Cu Content %	Initial USE (ft-lbs)	RG 1.99 Position 1.2 % USE Decrease	RG 1.99 Position 1.2 Predicted Irradiated USE (ft-lbs)
2-410	51989	0.17	101.3	30.7	70.2
3-410	13253	0.21	110	34.6	71.9
3-410	27204	0.21	94	34.6	61.5
3-410	13253/ 12008	0.22	112	35.4	72.4
3-410	27204/ 12008	0.22	85.5	35.4	55.2
9-410	20291	0.21	105	39.1	63.9

The FCS predicted irradiated USE values for August 2013 were calculated in accordance with the methodology described in Regulatory Guide 1.99, Revision 2, Position 1.2. All of the predicted USE values at the August 2013 end-of-life are greater than the 50 ft-lbs minimum screening criterion contained in 10 CFR 50, Appendix G.

References:

1. Letter from OPPD (W. G. Gates) to NRC (Document Control Desk), dated July 6, 1992 (LIC-92-203R), "Response to NRC Generic Letter (GL) 92-01, Revision 1: Reactor Vessel Structural Integrity."

2. Letter from OPPD (W. G. Gates) to NRC (Document Control Desk), dated July 16, 1993 (LIC-93-0124), "Revision of Response to NRC Generic Letter (GL) 92-01, Revision 1, "Reactor Vessel Structural Integrity."

3. Letter from OPPD (W. G. Gates) to NRC (Document Control Desk), dated June 23, 1993 (LIC-93-0119), "Fort Calhoun Station 5-Year Construction Period Recovery."

4. ASTM Standard E 185-82, "Standard Practice for Conducting Surveillance Tests for Light Water Cooled Nuclear Power Reactor Vessels," Volume 3.01, American Society for Testing and Materials, Philadelphia, PA.

5. F. W. Stallman, F. B. Kam and B. J. Taylor, "PR-EDB, Power Reactor Embrittlement Data Base, Version 1," NUREG/CR-4816 (ORNL/TM-10328), Oak Ridge National Laboratory, Oak Ridge, TN, June 1990.

6. Letter from OPPD (W. G. Gates) to NRC (Document Control Desk), dated August 12, 1993 (LIC-93-201), "Supplemental Information to Support Extension of the Fort Calhoun Station (FCS) Operating License Expiration Date."

7. Letter from Pacific Gas and Electric (G. M. Rueger) to NRC (Document Control Desk), dated June 30, 1992 (DCL-92-150), "Response to Generic Letter (GL) 92-01, Revision 1, Reactor Vessel Structural Integrity."

8. Combustion Engineering, CEOG Task 749, Report No. CE-NPSD-766, Rev. 01, "Evaluation of Low Upper Shelf Energy for Combustion Engineering Nuclear Steam Supply Systems Reactor Pressure Vessels," June 1993.

9. Letter from Duke Power Company (H. B. Tucker) to NRC (Document Control Desk), dated July 3, 1992.

LIC-95-0086
ENCLOSURE 2

UPDATED INFORMATION ON WELD CHEMICAL CONTENT, ESTIMATED FLUENCE, PROJECTED RT_{PTS}, AND PROJECTED USE FOR THE FORT CALHOUN STATION REACTOR VESSEL

(Previously submitted in LIC-93-0258)

UPDATED INFORMATION ON WELD CHEMICAL CONTENT, ESTIMATED FLUENCE, PROJECTED RT_{PTS}, AND PROJECTED USE FOR THE FORT CALHOUN STATION REACTOR VESSEL

WELD CHEMICAL CONTENT

This section lists weld chemical content for each of the three reactor vessel beltline welds at Fort Calhoun Station (FCS). The FCS reactor vessel was fabricated in 1967-1968 by Combustion Engineering, Inc. (CE), using materials and processes typical of that time period. The sources for determining the chemical content of the welds are also identified. References noted are listed on the last page.

The chemical content data for the FCS reactor beltline welds were taken from a variety of sources with a hierarchy as indicated below:

1. **Weld Chemical Analyses - (FCS)**

   When weld chemical analyses were available for the same material heat(s) and flux lot, chemistry factors were generated from 10 CFR 50.61 and Regulatory Guide (RG) 1.99, Rev. 2 requirements.

2. **Weld Chemical Analyses - (Non-FCS)**

   When weld chemical analyses were available for the same material heat(s) and flux type, but not necassarily the same flux lot, chemistry factors were generated from 10 CFR 50.61 and RG 1.99, Rev. 2 requirements.

Weld 2-410

The 2-410 axial (longitudinal) weld was fabricated using a Mil B-4 type wire (heat number 51989) and a Linde 124 flux (Lot 3687). The chemical content of the weld was determined from samples of two FCS vessel closure head longitudinal welds (Reference 1). The chemical analyses of the weld samples for copper and nickel are shown in Table 1.

TABLE 1: SUMMARY OF CHEMICAL CONTENT FOR WELD 2-410			
SAMPLE	WIRE HEAT/FLUX LOT NO.	Cu (w/o)	Ni (w/o)
1	51989/3687	.16	.20
2	51989/3687	.18	.13
AVERAGE		.17	.17

The chemistry factor of 89.45°F was interpolated from RG 1.99, Rev. 2, Table 1 (Reference 2) and 10 CFR 50.61.

Weld 3-410

The 3-410 axial (longitudinal) weld was fabricated using a Mil B-4 modified type of wire (Heat Numbers 12008, 13253, 27204) with a Linde 1092 flux (Lot 3774). A tandem arc process, utilizing two weld wires, was used in the fabrication of this FCS reactor vessel weld. Based on extensive review of weld records, no record was found indicating the use of either wire heat number 12008 in combination with itself, or wire heat number 13253 in combination with wire heat number 27204, during the welding process. Therefore, these two combinations were excluded from consideration. The four possible combinations evaluated were:

(1) Wire heat number 13253 combined with wire heat number 13253
(2) Wire heat number 27204 combined with wire heat number 27204
(3) Wire heat number 13253 combined with wire heat number 12008
(4) Wire heat number 27204 combined with wire heat number 12008.

Each of these evaluations is summarized below.

(1) Wire Heat Number 13253

The chemical content of the weld was determined from the average of two samples of a FCS vessel closure head weld (Reference 1), the D. C. Cook 1 surveillance weld (Reference 5), and five samples of the Salem 2 surveillance weld (Reference 6). The chemical analyses of the weld samples are shown in Table 2.

TABLE 2: SUMMARY OF CHEMICAL CONTENT FOR WELD 3-410 (13253)

SAMPLE	WIRE HEAT/FLUX LOT NO.	Cu (w/o)	Ni (w/o)
1	13253/3774	.14	.73
2	13253/3791	.27	.74
3	13253/3774,3833	.25	.73
AVERAGE		.22	.73

The variation in the copper content is explained in Reference 1. The chemistry factor of 188.45°F was interpolated from RG 1.99, Rev. 2, Table 1 (and 10 CFR 50.61).

(2) Wire Heat Number 27204

The chemical content of the weld was determined from a weighted average of multiple weld deposit records (Reference 7), including the Diablo Canyon 1 surveillance weld. The results are summarized in Table 3. The same information was previously transmitted to the NRC by Consumers Power Company (Palisades) for weld chemistry of the 27204 heat number.

TABLE 3: SUMMARY OF CHEMICAL CONTENT FOR WELD 3-410 (27204)			
SAMPLE	WIRE HEAT/FLUX LOT NO.	Cu (w/o)	Ni (w/o)
1	27204/3714,3724	.21	NR
2	27204/3714,3724	.21	NR
3	.7204/3714,3724	.25	NR
4	27204/3774	.24	1.00
5	27204/3774	.22	1.10
6	27204/3774	.18	.96
7	27204/3714	.22	1.00
8	27204/3791	.19	1.00
9	27204/3791	.20	1.02
10	27204/3714	.20	1.00
WEIGHTED AVERAGE		.21	1.00

NR = Not Reported

Based on the results above, a chemistry factor of 229.00°F was determined from RG 1.99, Rev. 2, Table 1.

(3) <u>Wire Heat Numbers 13253 & 12008</u>

The chemical content of the weld was determined based on a sample of a Maine Yankee Nozzle Dropout by Wyle Labs from CE archived material. The chemical analysis (Reference 13) of the weld sample determined copper as .21 w/o and nickel as .86 w/o for a chemistry factor of 206.60°F as interpolated from RG 1.99, Rev. 2, Table 1.

(4) <u>Wire Heat Numbers 27204 & 12008</u>

The chemical content of the weld was determined to be 0.19 w/o copper and 0.97 w/o nickel from the Mihama 1 surveillance weld (References 8 and 9).

The corresponding chemistry factor of 215.65°F was interpolated from RG 1.99, Rev. 2, Table 1.

<u>Weld 9-410</u>

The 9-410 circumferential weld was fabricated using a Mil B-4 modified wire (heat number 20291) with a Linde 1092 flux (Lot 3833). The chemical content of the weld was determined from surveillance weld test records at Cooper Nuclear Station (Reference 10). The average copper content was .23 w/o with an average nickel content of .75 w/o which yields a chemistry factor of 194.50°F when RG 1.99, Rev. 2, Table 1 is interpolated.

Summary

The chemistry factors for the FCS beltline reactor vessel welds were determined from RG 1.99, Rev. 2, Table 1 using chemical content information available. OPPD is currently participating in the ABB Combustion Engineering Reactor Vessel Group program, which is assembling an extensive reactor vessel materials data base from CE fabrication records. Completion of this data base is expected to provide OPPD with additional plant-specific and sister vessel information for use in supplementing and refining the FCS reactor vessel materials data base.

FLUENCE ESTIMATES

The fluence estimates for the reactor vessel surface (ID) and ¼t location (vessel wall thickness is 7.125 inches, neglecting clad thickness) for August 2013 are shown in Table 4.

TABLE 4: FLUENCE ESTIMATES FOR AUGUST 2013		
PLATE/WELD	SURFACE (ID) FLUENCE ESTIMATE FOR AUGUST 2013	¼t FLUENCE ESTIMATE FOR AUGUST 2013
All Plates	2.4×10^{19} n/cm^2	1.6×10^{19} n/cm^2
Weld 2-410 & 3-410	1.49×10^{19} n/cm^2	0.97×10^{19} n/cm^2
Weld 9-410	2.4×10^{19} n/cm^2	1.6×10^{19} n/cm^2

These vessel surface fluence estimates were conservatively calculated by ABB Combustion Engineering by using the DOT 4 computer code with ENDF/B-IV cross sections after benchmarking to the Pool Critical Assembly (PCA) (References 3 and 4). The vessel ¼t fluence estimates were calculated using RG 1.99, Rev. 2, Equation 3.

PROJECTED RT_{PTS} VALUES

The 10 CFR 50.61 information for RT_{PTS} is contained in Table 5, which updates the corresponding Table 1 in Reference 4.

TABLE 5: RT_{PTS} FOR FORT CALHOUN BELTLINE MATERIALS USING 10 CFR 50.61 AND REG. GUIDE 1.99, REV. 2

WELD SEAM/PLATE	HEAT NO.	Cu w/o	Ni w/o	Chemistry Factor(°F)	RT_{PTS} (°F) 8/9/2013
2-410 (longitudinal)	51989	0.17	0.17	89.45	109.33
3-410 (longitudinal)	27204*	0.21	1.00	229.00	264.29
9-410 (circumferential)	20291	0.23	0.75	194.50	250.39
D-4802 (intermediate shell, SA-533 Gr. B plate)	C 2585-3 A 1768-1 A 1768-2	0.12	0.56	82.20	135.60
D-4812 (lower shell, SA-533 Gr. B plate)	C 3213-2 C 3143-2 C 3143-3	0.12	0.60	83.00	136.58

* Weld composed of wire heat numbers 12008, 13253, and 27204. The most limiting heat number or combination is reported.

Reg. Guide 1.99, Rev. 02 Equation: $RT_{PTS} = I+M+(CF)f^{(0.28-0.10\log f)}$

Where: CF = Chemistry Factor determined from tables in Reg. Guide 1.99, Rev. 2 and 10 CFR 50.61
f = calculated value of neutron fluence at the reactor vessel/clad interface divided by 10^{19}

For Weld Material:
I = generic mean value of initial reference temperature = -56°F for welds made with Linde 1092 and 124 fluxes
M = margin to cover uncertainties in initial RT_{NDT} = 66°F since generic value of I was used

For Plate Material:
I = initial reference temperature of irradiated material as defined in the ASME code = 0°F for reactor vessel beltline plate material (Reference 12)
M = margin to cover uncertainties in initial RT_{NDT} = 34°F since a measured value of I was used

The PTS screening criteria applied to the vessel ID are RT_{PTS} = 270°F for axial (longitudinal) weld seams and plate material, and RT_{PTS} = 300°F for circumferential weld seams.

All weld and plate material RT_{PTS} values remain less than the 10 CFR 50.61 screening criterion at the August 2013 end-of-life date.

PROJECTED UPPER SHELF ENERGY (USE) VALUES

Based on the previously noted revised chemical content values for the welds and ¼t fluence values from Table 4, the projected USE summary table for beltline weld materials included in Reference 11 requires revision. The updated information is shown below in Table 6.

TABLE 6: Upper Shelf Energy Summary for Weld Material (August 2013)

Weld No.	Heat No.	Cu Content %	Initial USE (ft-lbs)	RG 1.99 Position 1.2 % USE Decrease	RG 1.99 Position 1.2 Predicted Irradiated USE (ft-lbs)
2-410	51989	0.17	101.3	30.7	70.2
3-410	13253	0.22	110	35.4	71.0
3-410	27204	0.21	94	34.6	61.5
3-410	13253/ 12008	0.21	112	34.6	73.2
3-410	27204/ 12008	0.19	85.5	33.5	56.8
9-410	20291	0.23	105	41.3	61.6

The FCS predicted irradiated USE values for August 2013 were calculated in accordance with the methodology described in Regulatory Guide 1.99, Revision 2, Position 1.2. All of the predicted USE values at the August 2013 end-of-life are greater than the 50 ft-lbs minimum screening criterion contained in 10 CFR 50, Appendix G.

CONCLUSION

Based on the preceeding updated reactor vessel weld chemistry values for copper and nickel, and on the resulting re-evaluation of both RT_{PTS} and USE, the conclusions stated in References 4 and 11 remain valid. These conclusions are that, for FCS at the August 2013 end-of-life date, the projected RT_{PTS} values of all beltline materials will remain less than the 10 CFR 50.61 PTS screening criteria, and the projected USE values will be greater than the 10 CFR 50, Appendix G screening criterion of 50 ft-lbs.

REFERENCES

1. Letter from OPPD (R. L. Andrews) to NRC (A. C. Thadani) dated January 23, 1986 (LIC-86-024)

2. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials" dated May 1988

3. Letter from OPPD (W. G. Gates) to NRC (Document Control Desk) dated June 23, 1993 (LIC-93-0119)

4. Letter from OPPD (W. G. Gates) to NRC (Document Control Desk) dated August 12, 1993 (LIC-93-0200)

5. Letter (Attachment 6) from AEP, Indiana Michigan Power (E. E. Fitzpatrick) to NRC (T. E. Murley) dated July 13, 1992

6. Letter (Table 7) from Public Service Electric & Gas Company (S. Labruna) to NRC (Document Control Desk) dated August 4, 1993 (NLR-N93125)

7. Letter from ABB Combustion Engineering (S. T. Byrne) to OPPD (K. C. Holthaus) dated January 11, 1993 (O-MECH-93-002) - Palisades 27204 report transmittal

8. Letter from Pacific Gas & Electric (J. D. Shipper) to NRC (S. A. Varga) dated January 17, 1986 (DCL-86-006)

9. Letter from Nebraska Public Power District (G. R. Horn) to NRC (Document Control Desk) dated July 1, 1992 (NSD920629)

10. Letter from Nebraska Public Power District (G. R. Horn) to NRC (Document Control Desk) dated February 25, 1993 (NSD930270/GE-NE-523-159-1292)

11. Letter from OPPD (W. G. Gates) to NRC (Document Control Desk) dated September 17, 1993 (LIC-93-0237)

12. Letter from OPPD (W. G. Gates) to NRC (Document Control Desk) dated July 6, 1992 (LIC-92-0203R)

13. Wyle Laboratories Report #40602-04, "Chemical Analysis of Weld Material Specimens for Baltimore Gas & Electric Company," dated July 27, 1989