

Omaha Public Power District

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402/636-2000

November 17, 1995
LIC-95-0211

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station PI-137
Washington, DC 20555-0001

- References:
1. Docket No. 50-285
 2. Letter from CPPD (W. G. Gates) to NRC (Document Control Desk), dated August 12, 1993 (LIC-93-0200)
 3. NRC Generic Letter 92-01, Revision 1, Supplement 1, "Reactor Vessel Structural Integrity," dated May 19, 1995
 4. Letter from OPPD (T. L. Patterson) to NRC (Document Control Desk), dated April 21, 1995 (LIC-95-0097)
 5. Letter from OPPD (T. L. Patterson) to NRC (Document Control Desk), dated August 17, 1995 (LIC-95-0156)

SUBJECT: Response to Items 2-4 of Generic Letter 92-01, Revision 1, Supplement 1, "Reactor Vessel Structural Integrity" (TAC No. 92678)

This letter with attachments provides the Omaha Public Power District (OPPD) response to Items 2-4 of Generic Letter (GL) 92-01, Revision 1, Supplement 1 (Reference 3); the response to Item 1 was provided by Reference 5. This response applies to the Fort Calhoun Station (FCS) Unit No. 1 Reactor Pressure Vessel (RPV). Reference 3 required all addressees to identify, collect and report any new data pertinent to analysis of structural integrity of their RPVs and to assess the impact of that data on their RPV analyses relative to appropriate requirements.

Items 2-4 and the OPPD responses are detailed in Attachment 2 of this letter. OPPD has evaluated the best estimate chemistry data for the FCS RPV weld materials and the Reactor Vessel Integrity Database (RVID); this included coordination and agreement with sister vessel utilities to resolve potential discrepancies in the industry database. Any additional information not previously available was also included in Attachment 2. The review of the industry database and the additional information did not result in significant changes to the PTS results from those previously submitted by OPPD. The chemistry factor for the limiting reactor vessel weld did, however, improve from 229°F (Reference 4) to 226.3°F, providing a corresponding improvement in the time projected to reach the 10 CFR 50.61 PTS screening criterion for the limiting (axial) weld. 200149

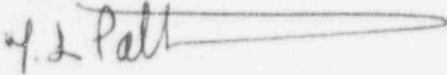
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OPPD has determined that use of the ratio procedure (in accordance with Position 2.1 of Regulatory Guide 1.99, Revision 2) is not needed at this time for FCS, because surveillance data are not used as a basis for the RPV integrity evaluation. However, OPPD continues to evaluate the future use of Position 2.1 as a means of providing additional margin in the PTS evaluation.

Please contact me if you have any questions.

Sincerely,



T. L. Patterson
Division Manager
Nuclear Operations

TLP/tcm

Attachments

c: Winston & Strawn
L. J. Callan, NRC Regional Administrator, Region IV
L. R. Wharton, NRC Project Manager
W. C. Walker, NRC Senior Resident Inspector

LIC-95-0211
Attachment 1

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of)
)
Omaha Public Power District) Docket No. 50-285
(Fort Calhoun Station)
Unit No. 1))

AFFIDAVIT

T. L. Patterson, being duly sworn, hereby deposes and says that he is the Division Manager - Nuclear Operations of the Omaha Public Power District; that as such he is duly authorized to sign and file with the Nuclear Regulatory Commission the attached information concerning the response to requirements of NRC Generic Letter 92-01, Rev. 1, Supplement i; that he is familiar with the content thereof; and that the matters set forth therein are true and correct to the best of his knowledge, information, and belief.

T. L. Patterson
T. L. Patterson
Division Manager
Nuclear Operations

STATE OF NEBRASKA)
) SS
COUNTY OF DOUGLAS)



Subscribed and sworn to before me, a Notary Public in and for the State of Nebraska on this 17 day of November, 1995.

David Kc'Bunski
Notary Public

OPPD Response to Generic Letter 92-01, Revision 1, Supplement 1
"Reactor Vessel Structural Integrity" (TAC No. 92678)
Required Information Items 2-4

References

1. NRC Generic Letter 92-01, Revision 1, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f)," dated February 28, 1992
2. NRC Generic Letter 92-01, Revision 1, Supplement 1, "Reactor Vessel Structural Integrity," dated May 19, 1995
3. Letter from OPPD (W. G. Gates) to NRC (Document Control Desk), dated August 12, 1993 (LIC-93-0200)
4. Letter from OPPD (W. G. Gates) to NRC (Document Control Desk), dated October 15, 1993 (LIC-93-0258)
5. Letter from OPPD (W. G. Gates) to the NRC (Document Control Desk), dated October 27, 1993 (LIC-93-270)
6. Letter from OPPD (W. G. Gates) to the NRC (Document Control Desk), dated June 17, 1994 (LIC-94-0135)
7. Letter from OPPD (T. L. Patterson) to NRC (Document Control Desk), dated April 21, 1995 (LIC-95-0097)
8. Letter from OPPD (T. L. Patterson) to NRC (Document Control Desk), dated August 17, 1995 (LIC-95-0156)
9. CEOG Task 781, CE NPSD-944-P, "Quantitative Evaluation of Chemical Composition Variability in Reactor Vessel Welds." Final Report dated December 1994
10. Combustion Engineering Reactor Vessel Group Report O-PENG-ER-001, Revision 00, Volume I and II, "Records Evaluation Program Phase II Final Report for the Fort Calhoun Reactor Pressure Vessel Plates, Forgings, Welds, and Cladding," dated October 1995
11. NRC Reactor Vessel Integrity Database, Version 1.1, Distribution Package, published July 1995
12. NRC Administrative Letter 95-03, "Availability of Reactor Vessel Integrity Database," dated August 4, 1995

Required Information-Item 2

An assessment of any change in best-estimate chemistry based on consideration of all relevant data

OPPD Response

A thorough review was made of the information contained in the reactor vessel integrity database (RVID) compiled by the NRC. When a discrepancy was identified in the consistency of the information presented for weld material, an evaluation was performed to resolve the apparent inconsistency. OPPD participated in numerous discussions and documentation reviews with sister vessel utility personnel to firmly establish the basis and validity of data provided by the utilities to the NRC for incorporation into the RVID.

The results of the evaluations for the reactor vessel beltline welds at Fort Calhoun Station (FCS) are summarized in the table below:

Weld Seam	Weld Wire Heat No.	Flux Lot No.	Original Values ⁽¹⁾			Revised Values ⁽²⁾			ΔCF
			Cu	Ni	CF	Cu	Ni	CF	
2-410	51989	3687	.17	.17	89.45	.17	.17	89.45	0
3-410	13253	3774	.22	.73	188.45	.22	.73	188.45	0
	27204 ⁽³⁾	3774	.21	1.00	229.00	.20	1.02	226.3	-2.7
	13253 ⁽³⁾ /12008	3774	.21	.86	206.60	.20	.86	202.7	-3.9
	27204/ 12008	3774	.19	.97	215.65	.19	.97	215.65	0
9-410	20291	3833	.23	.75	194.50	.23	.75	194.50	0

- (1) Original values from References 4 and 6
- (2) Revised values from RVID evaluation (if necessary)
- (3) Revised value due to incorporation of Reference 9

The following sections detail the weld chemical content for each of the beltline welds. The sources for determining the chemical content of the welds are also identified.

Weld 2-410

No changes from the original information in Reference 4 and the RVID were identified in the chemical content for wire heat number 51989 (with a flux lot number of 3687) or the source for determining the chemical content of this weld.

Weld 3-410

Wire Heat Number 13253

No change in the chemical content for wire heat number 13253 (with a flux lot number of 3774) was identified. There were additional sources located as shown in the table below. The value for the Pilgrim 1 unit in the RVID is a bare wire, prior to copper coating being applied; thus, it is not utilized in the best estimate calculation.

Specimen	Cu	Ni	Reference
Supplier Test Report (Pilgrim)	NA	0.72	Reference 10
Closure Head Weld (FCS)	.14	0.73	Reference 4
Surveillance Welds (Salem 2)	0.254	0.73	WCAP-8824, -10492, -1154, and -13366
Surveillance Weld (DC Cook 1)	0.27	0.74	WCAP-8047
Best Estimate Average	0.221	0.73	

Wire Heat Number 27204

The chemical content for wire heat number 27204 (with a flux lot number of 3774) changed due to the incorporation of additional data source information as shown in the table below. The values reported in the RVID were evaluated to be correct and no individual changes were required. There was a net improvement in the chemistry factor for this wire heat number of -2.7°F.

Specimen	Cu	Ni	Reference
Surveillance Weld (Diablo Canyon 1)	0.20	1.00	WCAPs-11567 and -13750
FCS Submittal (10 samples)	0.21	1.00	Reference 4
CEOG Task 781 (18 samples)	0.192	1.07	Reference 9
Best Estimate Average	0.20	1.023	

Wire Heat Numbers 13253/12008

The chemical content for wire heat numbers 13253 and 12008 (with a flux lot number of 3774) changed due to the incorporation of additional data source information as shown in the table below. The values reported in the RVID were evaluated to be conservative. Maine Yankee, Fitzpatrick, and Fermi analytically combined the separate chemistry data for heat numbers 13253 and 12008 from welds containing each of the components. The values reported as 13253 and 12008 for DC Cook 1 are conservatively estimated from the surveillance weld material (13253). The values listed in the table below were from actual weld samples of heat number 13253 combined with heat number 12008.

Specimen	Cu	Ni	Reference
Nozzle Dropout	0.20	0.875	Wyle Laboratories Report #40602-04
CEOG Task 781 (18 samples)	0.194	0.844	Reference 9
Best Estimate Average	0.197	0.859	

Wire Heat Numbers 27204/12008

No change from the original information submitted in Reference 4 was identified in the chemical content for wire heat numbers 27204 and 12008 (with a flux lot number of 3774) or the source for determining the chemical content of this weld. The Pilgrim 1 value reported in the Reactor Vessel Integrity Data Base (References 11 and 12) was determined to be based on the Mihama 1 Surveillance weld test report as evaluated by the Kobe Technical Institute in September 1973; however, the Pilgrim 1 value includes added conservatisms. Because the Mihama 1 Surveillance weld test report is the most accurate information, it is the basis for the determination of the FCS value.

Weld 9-410

No change from the original information submitted in Reference 4 was identified in the chemical content for wire heat number 20291 (with a flux number of 3833) or the source for determining the chemical content of this weld. Additional information from the Cooper Nuclear Station surveillance weld was evaluated; however, the best estimate copper and nickel did not change (0.224 and 0.744, respectively) from the values of 0.23 and 0.75 reported previously.

Plate Material

No change from the original information submitted in References 4 and 6 was identified in the chemical content for the plate materials or the source for determining the chemical content.

Required Information -Item 3

A determination of the need for use of the ratio procedure in accordance with the established Position 2.1 of Regulatory Guide 1.99, Revision 2, for those licensees that use surveillance data to provide a basis for the RPV integrity evaluation

OPPD Response

OPPD does not presently use surveillance weld data to provide a basis for RPV integrity; thus, use of the ratio procedure established in Position 2.1 of Regulatory Guide 1.99, Revision 2 is not needed. However, recent discussions with Diablo Canyon staff personnel provide the opportunity to voluntarily utilize the ratio procedure for the FCS vessel. Diablo Canyon is a sister vessel with the limiting FCS weld material (wire heat number 27204) in its surveillance program.

Required Information-Item 4

A written report providing any newly acquired data as specified above and (1) the results of any necessary revisions to the evaluation of RPV integrity in accordance with the requirements of 10 CFR 50.60, 10 CFR 50.61, Appendices G and H to 10 CFR Part 50, and any potential impact on the LTOP or P-T limits in the technical specification or (2) a certification that previously submitted evaluations remain valid. Revised evaluations and certifications should include consideration of Position 2.1 of Regulatory Guide 1.99, Revision 2, as applicable, and any new data.

OPPD Response

As supported by the above responses, OPPD certifies that the previously submitted RPV integrity evaluations remain valid. FCS complies with the requirements of 10 CFR 50.60, 10 CFR 50.61, Appendices G and H to 10 CFR Part 50, and the LTOP or P-T limits in the technical specifications. The incorporation of the additional data in Reference 9 improved the margin to the screening criterion for the limiting weld. The fracture toughness of the plates and welds remain bounded by the equivalent margins analysis previously submitted to the NRC (Reference 5). As previously indicated in Reference 8, OPPD continues to participate in the CEOG Reactor Vessel Working Group task on Weld Properties Evaluation in order to characterize the FCS vessel weld chemistry properties as accurately as possible.