Attachment 8

"Corrosion Evaluation of the Limerick Unit 2 N-16D Reactor Vessel Nozzle Modification - Non-Proprietary," Framatome Document No. 51-9271770-002, Non-Proprietary Version framatome

Framatome Inc.

Engineering Information Record

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Corrosion Evaluation of the Limerick Unit 2 N16-D Reactor Vessel Nozzle Modification – Non-Proprietary

framatome

Document No.: 51-9271770-002

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Safety Related? ☒ YES ☐ NO

Does this document establish design or technical requirements? ☐ YES ☒ NO

Does this document contain assumptions requiring verification? ☐ YES ☒ NO

Does this document contain Customer Required Format? ☐ YES ☒ NO

Signature Block

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Record of Revision

Revision No.	Pages/Sections/ Paragraphs Changed	Brief Description / Change Authorization
000	All	Original submittal; Note proprietary version is 51-9271544-000.
001	Sections 1.0 and 6.0	Updated text and Figures 1-1 and 1-2 in Section 1.0, and updated References 1 and 2 to the latest revisions.
002	Multiple	See Proprietary version (51-9271544-003).

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1.0 PURPOSE

The repair of the N16-D reactor vessel instrumentation nozzle in the Limerick Generating Station Unit 2 (LGS-2) reactor vessel changed the penetration configuration in the following ways: 1) the repair exposed the SA-533 Grade B, Class 1 low alloy steel reactor vessel and E8018-NM low alloy steel weld pad to water conditions, 2) included a new Alloy 690 nozzle as part of the pressure boundary, and 3) included a new Alloy 52M weld pad and partial penetration J-groove weld as part of the pressure boundary [1,2]. Also, the reducing insert to nozzle weld is now an Alloy 52M dissimilar metal weld. The original configuration and the final repair configuration are shown in Figure 1-1 and Figure 1-2, respectively. Note that the weld joining the stainless steel reducing insert (i.e., Grade F316) to the stainless steel pipe in the original configuration is assumed to be stainless steel because the joined base metals are both stainless steel.

The following corrosion evaluation considers potential material degradation due to each of these changes. For each degradation mechanism, the evaluation will either justify that the degradation mechanism is not an issue for the remaining life of the plant or the evaluation will conclude that further analysis is required.

[ ]

Figure 1-1: Original Configuration [1,2]

[ ]

Figure 1-2: Final Repair Configuration [1,2]

2.0 ASSUMPTIONS

2.1 Assumptions Requiring Verification

There are no assumptions requiring verification.

2.2 Justified Assumptions

The weld joining the stainless steel reducing insert to the stainless steel pipe in the original configuration is assumed to be stainless steel because the joined base metals are both stainless steel.

3.0 CORROSION OF EXPOSED LOW ALLOY STEEL

The low alloy steel reactor vessel material exposed due to the repair will be in the water space environment given the elevation of the N16-D nozzle. LGS-2 implements the water chemistry control requirements of BWRVIP-190 Revision 1 to mitigate corrosion [3, 4].

3.1 General Corrosion

[ ]

3.2 Galvanic Corrosion

[ ]

[ ]

3.3 Crevice Corrosion

[ ] The environmental conditions in a crevice can become aggressive with time and can cause accelerated local corrosion.

[ ]

The test results are supported by operating experience (and simulated operating experience) in light water reactors. [ ]

3.4 Stress Corrosion Cracking

[ ]

Although it is very unlikely that SCC cracks will initiate and propagate in low alloy steel under normal BWR conditions, it is impossible to completely rule out. Hence, it is prudent to examine the feasibility of performing an allowable flaw size evaluation by applying the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section XI criteria [9]. [.]

[ ]

[ ]

Figure 3-1: [ ]

4.0 CORROSION OF ALLOY 690 AND ALLOY 52M

Stress corrosion cracking failures of Alloy 600 and its associated weld metals (Alloy 82/182) have occurred in domestic and international light water reactors. The BWR industry addressed this issue by replacing or modifying affected materials with a modified version of Alloy 82 [14]. The modified version of Alloy 82 adds carbide stabilizers (Niobium and Titanium) to minimize chromium depletion at the grain boundaries. The PWR industry selected Alloy 690 and Alloy 52/152 as replacement materials [15]. Alloy 690 was also thermally treated to improve the microstructure, but grain boundary chromium depletion of Alloy 690/52/152 was avoided by doubling the chromium content (from ~15% to ~30%) instead of using carbide stabilizers. Alloys 690/52/152 have been in service for decades with no reported failures. Laboratory studies indicate that Alloy 690 and Alloy 52/152 have superior SCC resistance relative to the Alloy 600 and Alloy 82/182.

Although most testing of Alloy 690/52/152 has been under PWR conditions, some studies have been performed in environments more similar to BWRs. Creviced U-bend specimens of Alloy 600 and Alloy 690 were tested at 600°F for 48 weeks with an environment of 6 ppm oxygen [16]. The Alloy 600 readily cracked, whereas Alloy 690 showed no cracking. Also, testing of Alloy 690 in high purity water containing 36 ppm oxygen at 289°C (~550°F) for 47 weeks resulted in no cracking [16].

Extensive testing has been performed on Alloy 52/152 in high temperature deaerated water, which indicate that Alloy 52/152 is much less susceptible to SCC compared to Alloy 82/182 (the Alloy 600 weld metal) [15,17,18]. Test data of Alloy 52/152 in a high temperature oxygenated environment is not readily available, but Alloy 52/152 is expected to have a low susceptibility to SCC under these conditions as well, based on the similarity of Alloy 52/152 to Alloy 690.

The only difference between the Alloy 52M to be used in the repair and Alloy 52/152 are small alloying additions to improve weldability. The corrosion resistance is similar.

5.0 CONCLUSION

The modification of the N16-D reactor vessel nozzle at LGS-2, which exposed the low alloy steel reactor vessel to a water environment and introduced new materials (Alloy 690 and Alloy 52M), is found acceptable for the remaining life of the plant for all degradation mechanisms considered with the exceptions of general corrosion and SCC of the exposed low alloy steel, both of which require disposition by additional analyses. [

] Based on laboratory studies and operating experience, the replacement higher chromium content nickel-based alloys (Alloy 690 and Alloy 52M) have a high resistance to SCC.

6.0 REFERENCES

References identified with an (*) are maintained within Exelon Records System and are not retrievable from Framatome Records Management. These are acceptable references per Framatome Administrative Procedure 0402-01, Attachment 8. See page 2 for Project Manager Approval of customer references.

1. [
]

2. [
]

3. **BWRVIP-190 Revision 1: BWR Vessel and Internals Project, Volume 2: BWR Water Chemistry Guidelines - Technical Basis.* EPRI, Palo Alto, CA: 2014. 3002002623.

4. LGS UFSAR – Appendix A, "Updated Final Safety Analysis Report Supplement," Revision 18.

5. [ ]

6. D.C. Vreeland, et al., "Corrosion of Carbon and Low-Alloy Steels in Out-of-Pile Boiling-Water-Reactor Environment," Corrosion, Vol. 17, No. 6, 1961.

7. [ ]

8. [ ]

9. ASME Boiler and Pressure Vessel Code, Section XI, 2007 Edition, including Addenda through 2008.

10. H. P. Seifert and S. Ritter, "New Observations about the SCC Crack Growth Behavior of Low-Alloy RPV Steels under BWR/NWC Conditions," 11th International Conference on Environmental Degradation of Materials in Nuclear Power Systems, Aug 10-14, 2003, Stevenson, WA, ANS.

11. [ ]

12. License Renewal Application, Limerick Generating Station Units 1 and 2, Facility Operating License Nos. NPF-39 and NPF-85, June 22, 2011, NRC Accession Number ML11179A101.

13. [ ]

14. NUREG/CR-6923, "Expert Panel Report on Proactive Materials Degradation Assessment."

15. *Materials Reliability Program (MRP):Resistance to Primary Water Stress Corrosion Cracking of Alloys 690, 52, and 152 in Pressurized Water Reactors (MRP-111),* EPRI, Palo Alto, CA: 2004. 1009801.

16. Sedriks, A.J., Schultz, J.W., Cordovi, M.A., "Inconel Alloy 690 – A New Corrosion Resistant Material," Corrosion Engineering (Boshoku Gijutsu), vol. 28, pp. 82-95, 1979, Japan Society of Corrosion Engineering.

17. M. J. Psaila-Dombrowski et al., "Evaluation of Weld Metals 82, 152, 52 and Alloy 690 Stress Corrosion Cracking and Corrosion Fatigue Susceptibility," Eighth International Symposium on Environmental Degradation of Materials In Nuclear Power Systems – Water Reactors, Aug 10-14 1997, Amelia Island, FL, ANS.

18. Crum, J.R., Nagashima, T., "Review of Alloy 690 Steam Generator Studies," Eighth International Symposium on Environmental Degradation of Materials In Nuclear Power Systems – Water Reactors, Aug 10-14 1997, Amelia Island, FL, ANS.

The Vice President of Products and Engineering has approved the use of Reference 5.

Carl Fisher (Vice President of Products and Engineering) for Carl Fisher