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PWSCC of Alloy 600 CRDM in PWRs-History, Repair, Examination

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1. Introduction

Primary water stress corrosion cracking(PWSCC) of alloy 600 in a PWR has been reported in the control rod drive mechanism (CRDM)[1,2], In original PWRs, the SCC was not considered appropriately. Beginning in the mid seventies the world's PWR plants suffered from a sequence of SCC events mostly confined to S/G tubes, initially ODSCC then PWSCC. In thick wall alloy 600 materials, PWSCC was first reported in the Bugey 3 vessel head penetration in September 1991. The first report of a PWSCC in a SG drain nozzle was at the Shearon Harris plant in 1988.1 Two cases of boric acid precipitation were reported on the bottom head surface in two units of a SG in Korea.[3] In addition that, Alloy 600 CRDM nozzle showed crack indications in Korean plant in 2012.

The objective of the present work is to review PWSCC in CRDM nozzle in PWRs and repair and examination technologies for the cracking.

2. PWSCC history of Alloy 600 CRDM

In September, 1991, leakage occurred from the Bugey 3 T54 vessel head penetration in France. After 10 years of monitoring using the acoustic emission method as part of the thermal-hydraulic test, the leak was detected and it was estimated to be approximately 1 l/h. Through metallographic analysis, it was concluded that the crack was PWSCC. Due to vessel head and RPV homogeneity, EDF decided to replace all of the vessel heads with alloy 600 penetrations (54 VHs of 58 are alloy 600, the remaining 4 are alloy 690 penetrations). In France, pressure vessel heads have been replaced since 1993 at Bugey Unit 5.

In 2004, after over 100,000 operating hours, leakage of the 47th CRDM head penetration was detected in Ohi Unit 3 in Japan through visual inspection. Ohi Unit 3 is a power plant where RV head reactor coolant system (RCS) temperature was revised from 289 ^{o}C to 310 ^{o}C in 1997. At the present, 14 plants have replaced RVHs and 7 additional plants will replace RVHs with the alloy 690TT in near future.

The Davis-Besse NPP began commercial operation in August 1978. The plant had accumulated 15.8 effective full power years (EFPY) of operation when the plant shut down for its thirteenth refueling outage on February 16, 2002. On the date, the Davis-Besse began a refueling outage with the intent to perform work that included remotely inspecting the VHP nozzles from underneath the head focusing on the CRDM.[2]

The utility had thought that the boron accumulated on the RPV head was due to leaking CRDM flanges above the RPV head and that such accumulation would not cause extensive corrosion due to water evaporation at the elevated temperatures at that location. Accordingly, the boric acid leaking through the nozzle crack was allowed to corrode the carbon steel head creating a cavity.

The USNRC found that three CRDM nozzles had indications of through-wall axial cracking. The utility replaced the head with a new one, the material of which was the same as original one (Alloy 600) in 2004. Cracks were found again after 6 years of operation (2004~2010) on 24 nozzles.[4]

In Japan, replacement of the pressure vessel heads started at Takahama-1 in 1996, in Spain at Almaraz-1 in 1996, in Sweden at Ringhals-2 in 1996, in Belgium at Tihange-1 in 1999, in the USA at North Anna-2 and in China at Guangdong-2 in 2003. As of 2005, 93 power plants over the world had replaced the pressure vessel heads with alloy 600 penetrations.

3. Repair technologies for CRDM

Since alloy 600 and its welds, alloys 132/182/82, have shown PWSCC in reactor vessel head CRDM nozzles as well as nozzles in pressurizers and SGs, modification or replacement of the degraded area can be necessary. A half nozzle repair for CRDM nozzles with SCC affecting the weld region could be considered as one possible repair option. [5,6] However, a question remains concerning the corrosion rate of the low alloy steel inside the crevice (see Fig. 1) in the high oxygen, air saturated environment during refuelling outages. It seems likely that the protective oxide film built up at high temperature offers protection against crevice corrosion for some considerable time at low temperature during refuelling, but that remains to be demonstrated experimentally.

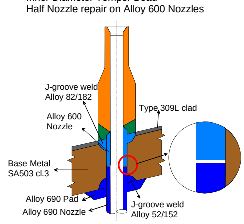

Fig. 1 Concept of half nozzle repair on CRDM

Since the cracks continue to be exposed to the PWR primary water environment and can lead to continued PWSCC flaw growth after initiation. The other form of repair is to embed the flaw under a PWSCC resistant material. Fig. 2 shows an embedment approach to repair PWSCC cracks or leaks in top-head nozzles and welds.[7] The PWSCC susceptible material, shown as the cross-hatched region in the figure, is assumed to be entirely cracked (or just about to crack). PWSCC resistant material, typically alloy 52 weld metal, is deposited over the susceptible material. It should be noted that the resistant material in this repair must overlap the susceptible material by enough length in all directions to preclude new cracks growing around the repair and causing the original crack to be re exposed to the PWR environment. Although this repair approach has been used successfully in several plants, there have been many incidents in which nozzles repaired by this approach during one refueling outage have been found to be leaking at the subsequent outage. These occurrences were attributed to lack of sufficient overlap of the repair, because it is sometimes difficult to distinguish the exact point at which the susceptible material ends.

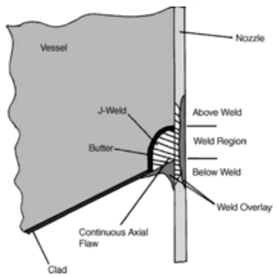

Fig. 2 Schematic of Flaw embedment repair on CRDM [7]

Water Jet Peening is another option for compressive stress improvement at the surface in weld metal which is susceptible to PWSCC. This technique makes use of the energy of the bursting cavitation generated by a high pressure jet water. This technique can improve the stress condition at both the inner and outer surface including weld metals, such as the BMI nozzles (inner surface, outer surface and J-weld) and inlet/outlet nozzles of a reactor vessel. The Laser Stress Improvement Process (L-SIP) introduces compressive stress on the inner surface of the pipe by irradiating the outer surface with a laser rotating around the pipe.

4. Examination of CRDM weld region

ASME Section XI Code case CC-N 729-4, titled Alternative Examination Requirements for PWR Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds Section XI, Division 1' gives us examination requirement on the J weld region of CRDM. It contains in service examination methods and frequency determined using the following some parameters to characterize the susceptibility to crack initiation and the potential for crack propagation. The code case defines an examination volume for the nozzle base metal and examination area for weld and nozzle base metal as shown in Fig. 3

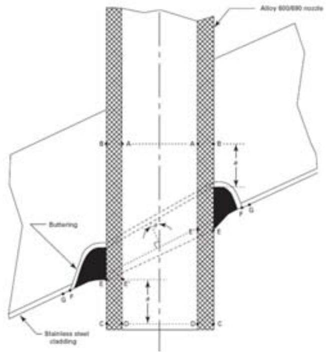

Fig. 3 an examination volume for the nozzle base metal and examination area for weld and nozzle base metal describes in Code Case N-729-4

5. Summary

(1) PWSCC events on Alloy 600 CRDM nozzles were reviewed.
(2) Half nozzle repair, flaw embedment, peening technologies as mitigation and repair tools for the CRDM were introduced.
(3) Examination requirement for the CRDM weld region (CC-N 729-4) was also reviewed.

6. REFERENCES

[1]http://pbadupws.nrc.gov/docs/ML0824/ML082490129.pdf
[2]EPRI-1013419, Materials Reliability Program: Characterization of Cracks in a Control Rod Drive Mechanism (CRDM) Nozzle Base Metal and Weldment from Davis-Besse Reactor (MRP-193) EPRI, Palo Alto, CA, USA, 2006.
[3] KAERI/CR-329/2009, Failure Analysis on Steam Generator Drain Nozzle in Yonggwang Unit 4-Final Report, 2009
[4]IAEA, Nuclear energy series no. NP-T-3.13, Stress corrosion cracking in light water reactors: Good practices and Lessons, IAEA, Vienna, Austria, 2011.
[5] PAYNE R., "Reactor pressure vessel modification approaches," PVP 2005, ASME Pressure Vessel and Piping (Proc. Conf. Denver, Colorado, USA, 2005).
[6] EPRI, Boric acid corrosion guidebook, revision 1, EPRI TR-1000975, EPRI, Palo Alto, CA (2001)
[7] K.R. Rao, Companion guide to ASME boiler & Pressure vessel code, ASME press, Three park Ave. Newyork, USA, Ch.44, pp.63-87, 2009