Transactions of the Korean Nuclear Society Autumn Meeting
Gyeongju, Korea, October 25-26, 2012

PWSCC Issues and Material Aging Management for Nuclear Power Plants in Korea

Seong Sik Hwang, Yun Soo Lim, Dong Jin Kim, Sung Woo Kim, Hong Pyo Kim
Nuclear Materials Research Division, Korea Atomic Energy Research Institute,
1045 Daedeog-daero, Yuseong-gu, Daejeon, 305-353,
sshwang@kaeri.re.kr

1. Introduction

The primary water stress corrosion cracking (PWSCC) of alloy 600 in a PWR has been reported in the control rod drive mechanism (CRDM)[1]. Beginning in the mid-seventies, the pressurized water reactor (PWR) plants suffered from a sequence of SCC events mostly confined to S/G tubes, initially ODSCC, and then PWSCC. PWSCC was first reported in Bugey 3 vessel head penetration made of forged alloy 600 materials in September 1991. Other PWRs experienced cracking attributed to the PWSCC of the major primary side weld area made from alloy 182 at the end of the year 2000. Examples of dissimilar metal butt welds between the main austenitic stainless steel primary circuit piping and the outlet pressure vessel nozzles are the cracking of Ringhals 4, V. C. Summer and some J-groove welds of the CRDM of the RVH at Oconee 1 [2].
In addition to the Reactor Vessel Head (RVH), the PWSCC of alloy 182/82 has been reported at bottom mounted instrumentation (BMI) nozzle J-welds, steam generator(SG) J-weld drain nozzle, and SG tube sheet cladding [2].
Two cases of boric acid precipitation were reported at the bottom head surface of a SG in Korea [3,4]. Cracking was found in the cold leg drain nozzles made of alloy 600 in two units, hot side nozzles were fabricated with alloy 690 from the beginning. The cracking of steam generator tubings made of alloy 600 is another concern in Korea, because some plants still have alloy 600 HTMA tubings. The flow accelerated corrosion of secondary pipings is another type of corrosion problems, though it has not been treated as a severe problem in Korea.
To properly manage the corrosion issues and seek out research items for maintaining the integrity of nuclear plants, the PRIMA-Net (Proactive Research and Innovative Material Aging Network) was organized in 2007. The research and development expert group consists of a National research laboratory (KAERI), regulatory body (KINS), utility (KHNP), engineering and design company (KEPCO EC), manufacturer (Doosan Heavy Industries) and universities (SNU, KAIST, UNIST), etc.
The objective of the present work is to review the corrosion issues in Korea and introduce a strategy of Material Aging Management for Nuclear Power Plants in Korea.

2. PWSCC of alloy 600 components

2.1.PWSCC in Drain nozzle of SG

Cracking was found in the cold leg drain nozzles made of Alloy 600 in two units in the Younggwang site in 2007 and 2008, respectively. Figure 1 is an example of a destructive analysis of the cracked nozzle. It was found that the cracking was induced by PWSCC. The utility have replaced the cracked area with Alloy 690, which is more SCC resistant than Alloy 600.

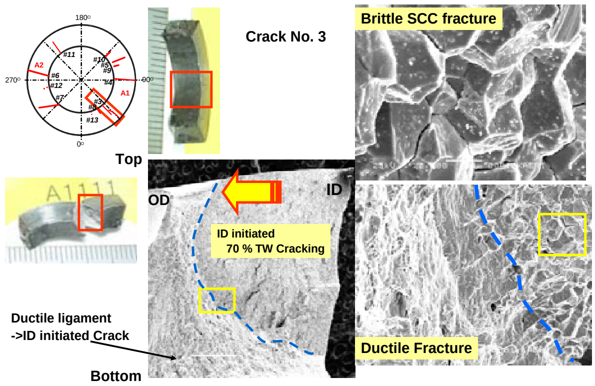

Figure. 1 Feature of cracks developed on the nozzle

2.2 PWSCC in RVH Vent Pipe

Boric acid precipitates were found at the top of the reactor vessel near the pressure vent pipe of YG unit 3 in 2010. The material was also alloy 600 and its weld alloys (alloy 82/182). The crack was initiated on the inner diameter (ID) and propagated to the outer diameter (OD) as shown in Figure 2. The crack propagated along the grain boundaries, and therefore the cracking morphology was purely intergranular

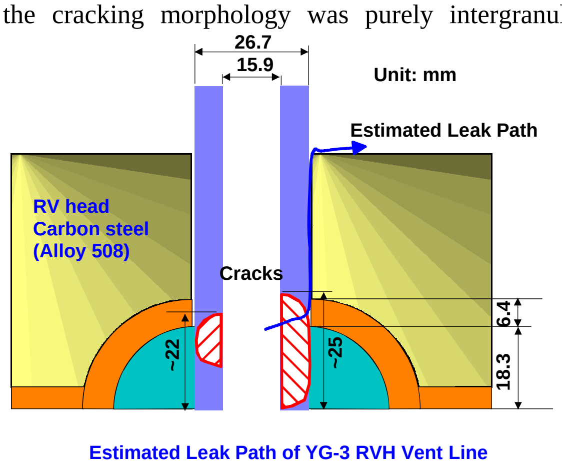

Figure 2. PWSCC of reactor vessel head vent pipe made of alloy 600

SCC. All these facts imply that the cracking was occurred by PWSCC.

3. **Research and Strategy of Material Aging Management**

To properly manage the corrosion issues and seek research items for maintaining the integrity of the nuclear plants, the PRIMA-Net (Proactive Research and Innovative Material Aging Network) was organized in 2007. The research and development expert group consists of a national research laboratory (KAERI), regulatory body (KINS), utility (KHNP), engineering and design company (KEPCO EC), manufacturer (Doosan Heavy Industries), universities (SNU, KAIST, UNIST), etc. Figure 3 shows the role of an expert group from various organizations.

4. Conclusions

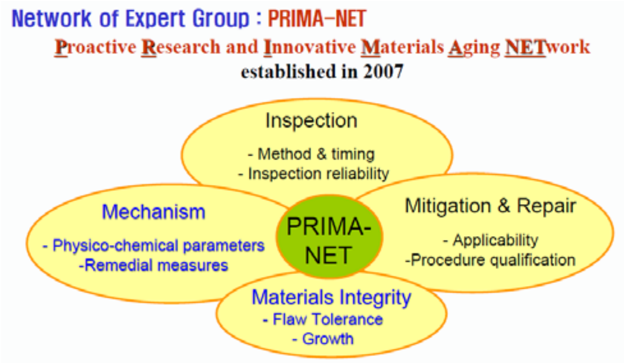

Figure 3. Aging management activities for Nuclear Power Plants in Korea

-Three cases of alloy 600 welds components such as a steam generator drain nozzle and reactor vessel head vent pipe reported in Korea were caused by PWSCC.
-An aging management research group was created in Korea in 2007, the role of which is to seek high priority research items for maintaining the integrity of nuclear plants.

References

[1] IAEA TEC DOC 981, 'Assessment and management of ageing of major nuclear power plant components important to safety: Steam generators 1997.
[2] EPRI TR-103696, "PWSCC of Alloy 600 Materials in PWR Primary System Penetrations", July 1994.
[3] KAERI/CR-277/2007, Microstructure Analysis of a Steam Generator Drain Nozzle in Younggwang Unit 3 - Final Report
[4] KAERI/CR-329/2009, Failure Analysis on Steam Generator Drain Nozzle in Yonggwang Unit 4- Final Report Research Program on Corrosion of Secondary Pipes in PWRs