Materials harvesting of reactor internals from a decommissioned plant

Seong Sik Hwang [a*], Sung Woo Kim [a], Min Jae Choi [a], Sung Hwan Cho [a], Jong Yeon Lee [a], Hong Pyo Kim [a], Yun Soo Lim [a], Dong Jin Kim [a]
[a] *Korea Atomic Energy Research Institute, 989-111 Daedeok-daero, Yuseong-gu, Daejeon 34057, Korea*
[*] *Corresponding author: sshwang@kaeri.re.kr*

1. Introduction

There is a national project to establish a R&D center for dismantling nuclear power plants [1]. Development of a technology for design and licensing of a dedicated radiation shielding facility for safe and efficient operation of this R&D center is necessary. The causes of damage to the internal structures of the nuclear power reactor, such as the baffle former bolts (BFB) of the Kori unit 1 should be analyzed. To this end, it is important to develop technologies that can comprehensively analyze various characteristics of internal structures of long term operated reactors. The Korean regulatory agency (KINS) requests to identify the cause of the defect.

Thus, with the support of the government, long-term research projects have been launched, and this paper discusses the research items of the project and how they will be used in the future.

2. Background

A lot research institutes have high temperature, high pressure stress corrosion testing facilities and technologies in general experimental zones for non-radioactive components. However, no technology has been developed and no related facilities have been associated with stress corrosion test for medium or low level materials irradiated by neutron.

The importance of degradation management of the reactor internal structure is brought up as the operation time of the operating plants increases, and the materials aging management program (AMP) has been developed and been applied based on EPRI 227 report [2].

The test for identifying the cause of the failure in the internal structure of the reactor shall be conducted in the same high temperature high pressure environment as in the operating plant. Hot cells equipped with autoclave and slow strain rate tester (SSRT) are required for this research, but the device is not installed in Korea.

3. Research detail

3.1 Technology development of facility verification for IASCC test

USNRC and EPRI are working on a government led project to assess material degradation characteristics using survey materials from decommissioned nuclear power plants to secure technical evidence for decommissioning old plants and safe operation of operating plants [3].

The development of the demonstration test technology for IASCC is a study using materials taken from the decommissioned nuclear power plant. Because the neutron irradiated materials transporting cost is more expensive abroad than the evaluation costs itself, and difficulties in utilizing foreign facilities, we plan to establish the IASCC verification test facilities in Korea as shown in Figure 1.

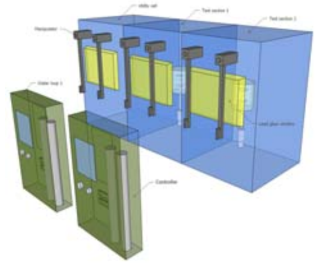

Fig. 1. Schematic of the verification test facility for IASCC.

In this research project, we also intend to develop technology with the aim of producing a tiny tensile specimen of the shape shown in Figure 2.

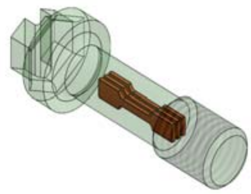

Fig. 2. Schematic of small size tensile test specimen.

3.2 Root cause analysis of flaw indication of Kori-1 BFB

Two of the eight faulty bolts in former level A (bottom) of the Kori unit 1 will be selected to conduct an analysis of the shape of the defect and cause of damage by analyzing its fracture surface [4]. Two sound bolts adjacent to the defect bolts will be also collected and compared to the damaged bolts to identify the cause of the defect.

Defect bolt withdrawal will be carried out by utilizing the equipment developed during a government project in the past as shown in Figure 3 below [5].

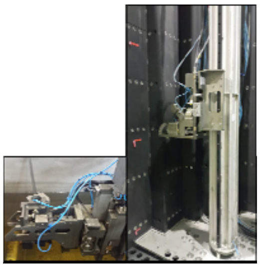

Fig. 3. Replacement facility for BFB from Kori-1.

3.3 Design and development of hot cell facilities

Based on the shielding design criteria (20 mSv/year) given in Notice No.2019-10 of the Nuclear Safety and Security Commission (NSSC), the space dose after the shielding of the test facility will be less than 10 uSv/hour. Specific research items include analysis of radiation protection licensing requirements, preparation of hot cell design requirements, easy-to-disassemble/assembly and movement hot cell design, hot cell shielding design, calculation of space dose.

3.4 Structural analysis of flow analysis of assembly and major parts.

It will be assessed the integrity of the key components of the reactor internal structure from the perspective of fatigue and IASCC. Figure 4 shows a preliminary analysis result on stress distribution on the bolt.

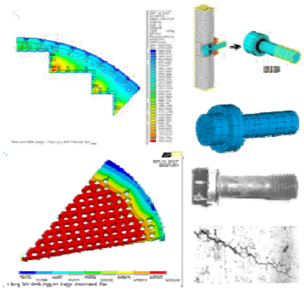

Fig. 4. Expected evaluation result for IASCC susceptibility.

3.5 Effect of neutron irradiation

In this task, radiation dose calculations are performed reflecting the entire operation cycle of the Kori-1 NPP. Figure 5 shows a preliminary evaluation of the expected distribution of Co-60 radiation levels five years after the permanent shutdown of Kori-1 reactor.

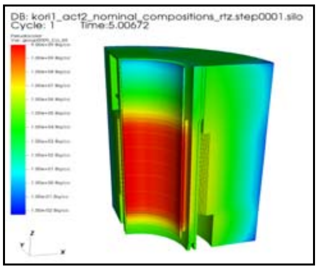

Fig. 5. Radiation level of Co-60 in 5 years after final operation.

4. Summary

- A national project plan to establish a R&D center for dismantling nuclear power plants is described.
- The cause of damage to the BFB of the Kori-1 will be surveyed.
- The project will perform five research areas like 1)Technology development of facility verification for IASCC test, 2)Root cause analysis of flaw indication of Kori-1 BFB, 3)Design and development of hot cell facilities, 4)Structural analysis of flow analysis of assembly and major parts, 5) Effect of neutron irradiation.

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REFERENCES

[1] Seong Sik Hwang, et al., "Development of failure and degradation analysis technologies of reactor internal baffle former bolts from decommissioning NPP" PRIMA-NET workshop, Jan. 16, 2020, Gimcheon, Korea.

[2] MRP: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 1-A), EPRI, Palo Alto, CA, USA, 2019.

[3] Jean Smith, et al., "EPRI Irradiated Materials Testing Programs", ML13162A571, USNRC, USA.

[4] Kyoung Hwan Na, et al., "Research progress and plan for the utilization of Kori 1 harvesting materials" KNS workshop, Oct. 23, 2019, Ilsan, Korea.

[5] KAERI/RR-4098/2016, "Development of Life Prediction and Extension Technologies of Nuclear Reactor Internals" KAERI, Daejeon, Korea, 2016.