CORROSION SCIENCE AND TECHNOLOGY, Vol.20, No.4(2021), pp.189~195
[Research Article]
pISSN 1598-6462 / eISSN 2288-6524
DOI: https://doi.org/10.14773/cst.2021.20.4.189

Failure Evaluation Plan of a Reactor Internal Components of a Decommissioned Plant

Seong Sik Hwang†, Sung Woo Kim, Min Jae Choi, Sung Hwan Cho, and Dong Jin Kim

Korea Atomic Energy Research Institute, 989-111 Daedeok-daero, Yuseong-gu, Daejeon 34057, Korea

(Received July 02, 2021; Revised July 09, 2021; Accepted July 09, 2021)

A technology for designing and licensing a dedicated radiation shielding facility needs to be developed for safe and efficient operation an R&D center. Technology development is important for smooth operation of such facilities. Causes of damage to internal structures (such as baffle former bolt (BFB) of pressurized water reactor) of a nuclear power reactor should be analyzed along with prevention and countermeasures for similar cases of other plants. It is important to develop technologies that can comprehensively analyze various characteristics of internal structures of long term operated reactors. In high-temperature, high-pressure operating environment of nuclear power plants, cases of BFB cracks caused by irradiated assisted stress corrosion cracks (IASCC) have been reported overseas. The integrity of a reactor's internal structure has emerged as an important issue. Identifying the cause of the defect is requested by the Korean regulatory agency. It is also important to secure a foundation for testing technology to demonstrate the operating environment for medium-level irradiated testing materials. The demonstration testing facility can be used for research on material utilization of the plant, which might have highest fluence on the internal structure of a reactor globally.

Keywords: *Nuclear power plant, PWR, Reactor internals, Materials harvesting, IASCC*

1. Introduction

Irradiation deterioration of the stainless steel material inside the nuclear reactor is a part that must be checked when considering the safety of a long-term operating nuclear power plant, and many related studies are being conducted at home and abroad [1-3].

The Korea Atomic Energy Research Institute (KAERI) has a testing facility called Irradiated Material Examinations Facility (IMEF) for irradiated materials. However, this facility has large scale concrete hot cells that handle high level radioactive materials such as spent fuel, and is not suitable for precise analysis of radioactive materials at medium or low levels due to concerns such as secondary contamination and diffusion of contamination.

Currently, some 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.

In the case of mechanical property evaluation technology of irradiated materials, some test specimens with gage specifications of 9 mm in length, 2 mm in width and 0.5 mm in thickness were used for evaluation of the tensile properties of the irradiated materials using a research reactor. However, it is difficult to obtain a test specimen due to the validity of the test results.

In addition, KAERI has micro-size analysis technology and facilities for medium level radioactive materials and continues to develop related technologies for non-irradiated materials to secure nanoscale level analysis technologies that are being utilized by foreign research laboratories.

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 [4].

†Corresponding author: sshwang@kaeri.re.kr
Seong Sik Hwang: Principal Researcher, Sung Woo Kim: Principal Researcher, Min Jae Choi: Senior Researcher, Sung Hwan Cho: Senior Researcher, Dong Jin Kim: Principal Researcher

A Korean research institute has developed a replacement equipment for BFB and the dismantling of barrel former bolts for the repair of reactor internal structures of Westinghouse-type nuclear power plants, and has applied this technology to harvest internal structures from the nuclear reactor.

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 facilities has not been installed in Korea.

A regulatory study on hydrodynamic analysis of the reactor internal structures and evaluation of primary coolant and structure interaction has also been conducted in Korea.

Radioactive inventory evaluation technology and radionuclide analysis technology are considered as core technologies related to dismantling. One of the domestic agencies conducted an assessment of radioactive inventory considering the reactor specific operational history, which is a method differentiated from the existing radioactive inventory assessment technology, and conducted a comparison review with the measured values by radioactive species using dose monitors. The analysis of radionuclides of radioactive materials is conducted only by some domestic research institutes, due to the difficulty of setting up high cost equipment and radiation management zones. There is no data on analysis of radionuclides of high level radioactive materials.

Because the BFB, one of the key research items in this study, is a radioactive waste from decommissioning nuclear power plants, it is expected that technology development of the procedures for bringing in and out of the plant will be necessary.

Thus, with the support of the government, long-term research projects have been launched [5], and this paper discusses the research items of the project and how they will be used in the future

2. Research method

Methods for remotely handling specimens exposed to neutrons and radiation were analyzed, and facilities for evaluating the IASCC characteristics of stainless steel materials embrittled by neutrons were summarized. Specimens of the power plant that showed defects were analyzed and methods for determining the cause of the defects were summarized.

Foreign data were investigated and analyzed on the hotcell construction method for handling radioactive specimens, and the domestic facility construction method was discussed.

The numerical analysis method to identify the cause of the defect is as follows.

The flow evaluation methodology is analyzed and the target location is determined. After optimizing the preliminary flow analysis model, a detailed flow analysis model is developed. Fluid induced load is calculated and a fatigue evaluation methodology that is considered as one of the damage mechanisms is developed. After that, the IASCC initiation evaluation methodology is analyzed and the impact of deterioration is comprehensively evaluated through the IASCC initiation evaluation.

The radiation dose calculation for the transport of radioactive specimens uses RAPTOR-M3G+ACT2, a commercial code. A representative nuclear reactor core is selected for the calculation and the amount of radiation during neutron irradiation of each chemical component is calculated based on the chemical composition of type 316 stainless steel presented in NUREG/CR-3474. This result even performs mapping through the visualization program. In particular, it is the key to perform the calculation of radiation by applying the nuclear fuel loading and operation history of the power plant. Considering the convenience of calculation and core symmetry, the 1/4 core model is used.

3. Result and discussion

3.1 Technology development of facility verification for IASCC test

USNRC (US Nuclear regulatory commission) and EPRI (Electric power research institute) 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 [6]. Spain, as well as the US nuclear power plant, is also carrying out a research program on

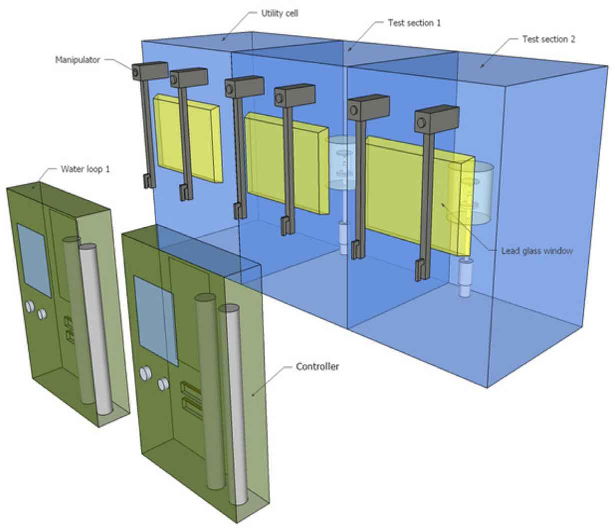

Fig. 1. Schematic of the verification test facility for IASCC

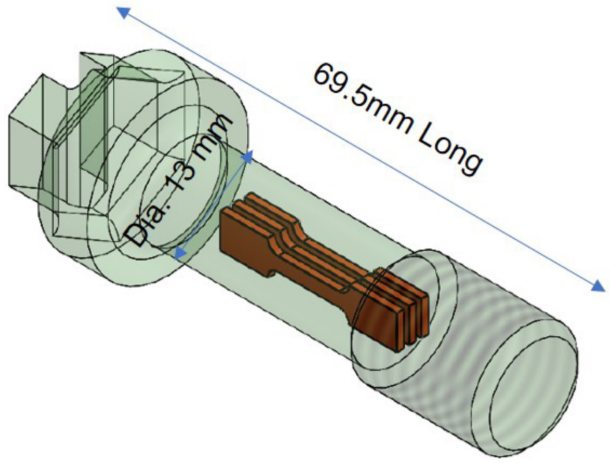

Fig. 2. Schematic of small size tensile test specimen

the decommissioning Zorita nuclear power plant. They are actively participating in the material evaluation program of the Korean nuclear power plant.

The development of the demonstration test technology for IASCC is a study using materials taken from the decommissioned nuclear power plant. Considering foreign cases, it is understood that LIDEC of France and SCK-CEN of Belgium, Studsvik of Sweden have built and operated one stop service-enabled facilities, ranging from precision processing of irradiated materials to precision testing and evaluation and technical support. However, as the neutron irradiated materials transporting cost is more expensive abroad than the evaluation costs itself, and difficulties in utilizing foreign facilities, such as the limitations of the test specimen's fluence, size and volume, we plan to establish the IASCC verification test facilities in Korea as shown in Fig. 1.

Tiny test specimens smaller than the typical ASTM specimen specification are used to assess tensile properties of radioactive materials. In addition, although this small specimen is being used to assess the characteristics of materials that are difficult to secure large test specimen, such as 3D printed materials, it is necessary to conduct an applicability assessment according to the difference between materials, as the results of small test specimen differ greatly depending on the type or microstructure of materials. In this research project, we also intend to develop technology with the aim of producing a tiny tensile specimen of the shape shown in Fig. 2. The detail dimension of the bolt sample is shown in Fig. 3.

The development of the optimal specimen specifications and test conditions for mechanical property evaluation for radioactive specimen will be carried out. We will also select optimal tensile specimen specifications and test conditions by taking into account the specimen fabrication process and limited BFB handling space within the hot cell.

In the field of development of the tensile properties evaluation technique of irradiated materials, the technology by using the Instrumented Indentation Techniques(IIT) will show the change in mechanical properties in the length direction of the BFB.

On the other hand, we plan to show machining accuracy by verifying that the machining tolerances of the test specimen (100 micrometers) given in the ASTM E8 [8] specification are met by using the measurement device for the survey specimen after precision processing.

Foreign agencies have research facilities that allow precise analysis of microstructures in atomic units below the nano size for medium level radioactive materials. The USA continuously complements microstructure analysis facilities installed in Oakridge National Lab.(ORNL), Idaho National Lab.(INL), and France has secured and utilized facilities for nano size level microstructure evaluation of radioactive materials, including high level nuclear fuel since 2010. The United Kingdom also contributes to the study of degradation of nuclear materials by establishing precision assessment facilities including hot cells within the National Nuclear User Facility (NNUF).

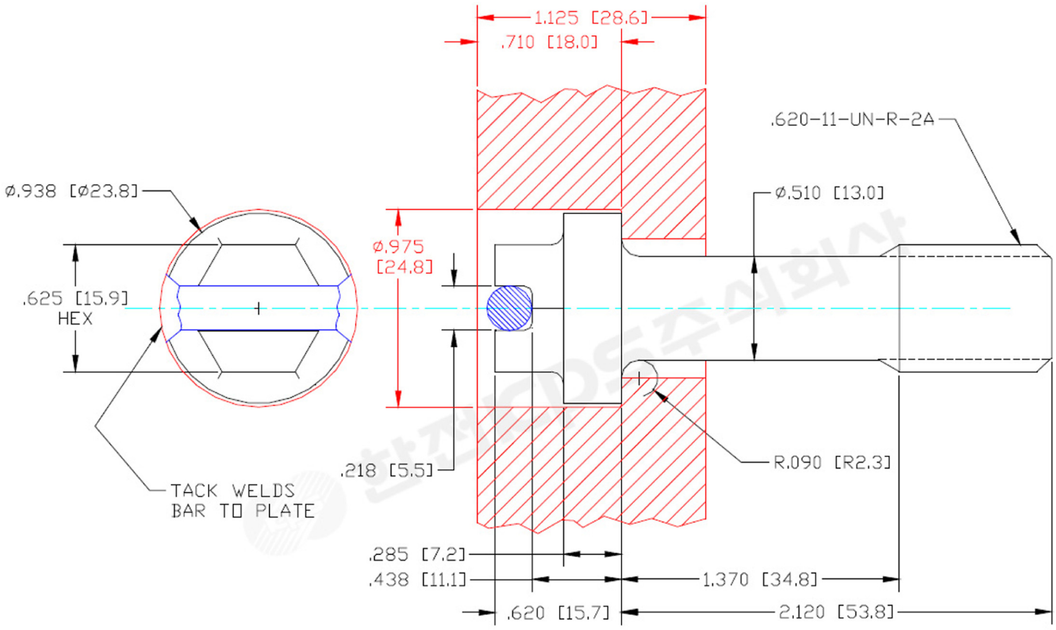

Fig. 3. Dimension of baffle former bolt of the Korean plant [7]

This study area aims to establish a microstructural analysis laboratory and develop core technologies for supporting radioactive nuclear materials. The content includes the development of the preparation process for the electron microscope analysis test specimen, the development of high resolution microstructural analysis technology of the radioactive nuclear material, and the establishment of a test procedure for the analysis of medium and low level radioactive materials.

3.2 Root cause analysis of flaw indication of the Korean plant

Two of the eight faulty bolts in former level A (bottom) of the plant will be selected to conduct an analysis of the shape of the defect and cause of damage by analyzing its fracture surface [9]. 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 Fig. 4 below [10].

The detailed analysis of the causes of damage is performed through the breakdown analysis of mechanical characteristics, fracture surface analysis, microstructure analysis, and detailed stress evaluation methods such as neutron irradiation effects (embrittlement, gamma-heating), flow induced vibration, and IASCC sensitivity assessment. The results of the analysis of domestic and foreign BFB damage experience including analysis of overseas damage cases are also conducted, and the results of the root cause analysis are intended to increase the reliability of the results through consultation from overseas institutions.

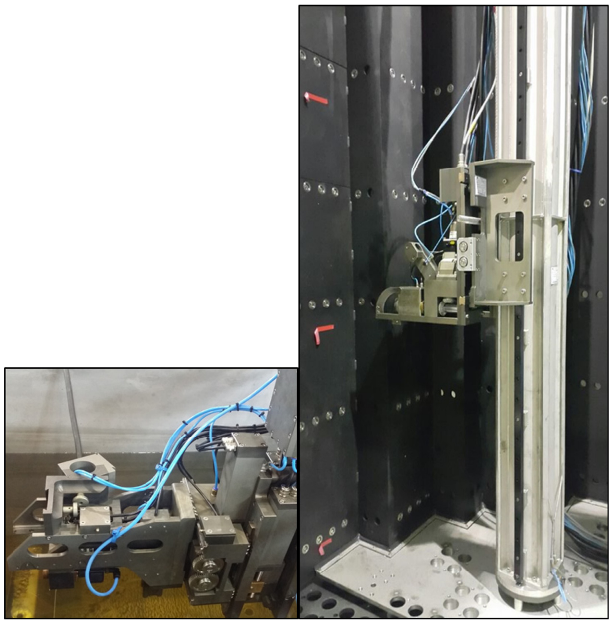

Fig. 4. Replacement facility for BFB from the Korean plant [10]

3.3 Design and development of hot cell facilities

Hot cells are essential for handling and analysis of radioactive specimens. The licensing process of the regulatory body is essential to make use of the implementation. 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, subject to the conditions of working 40 hours a week. Specific research items include analysis of radiation protection licensing requirements, preparation of hot cell design requirements, easy-to-disassemble/assemble and movement hot cell design, hot cell shielding design, calculation of space dose applied to Montecarlo methodologies, human simulation, and assessment of worker exposure.

3.4 Structural integrity analysis by flow distribution near BFB

A numerical analysis technique is used as a method to identify the cause of damage of BFB to be withdrawn. USNRC in the USA presents temperature and fatigue caused by flow induced vibration, IASCC, etc. as the main causes of damage to the reactor internal structures, including BFBs. This study will also assess the integrity of the key components of the reactor internal structure from the perspective of fatigue and IASCC. The main parts of the reactor internal structure that could be damaged will be derived and the effects of the flow induced vibration and fatigue and the IASCC will be assessed comprehensively. Fig. 5 shows a preliminary analysis result on stress distribution on the bolt. The picture of the defective bolt included in this Fig. 5 was obtained from a study case [11] at an overseas power plant, and it was judged that a similar defect would be observed in this study, so it was included in the figure as reference data.

3.5 Evaluation on effect of neutron irradiation

Radiation doses from the vicinity of BFB are essential data for the planning of withdrawal and transfer. In

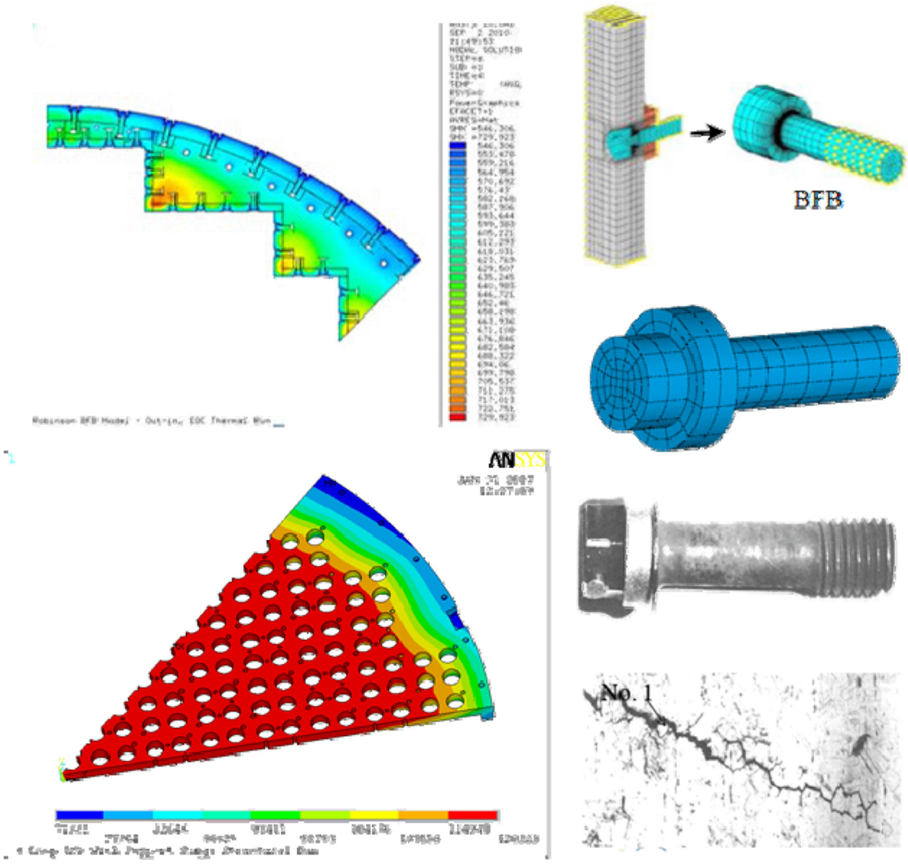

Fig. 5. Expected evaluation result for IASCC susceptibility

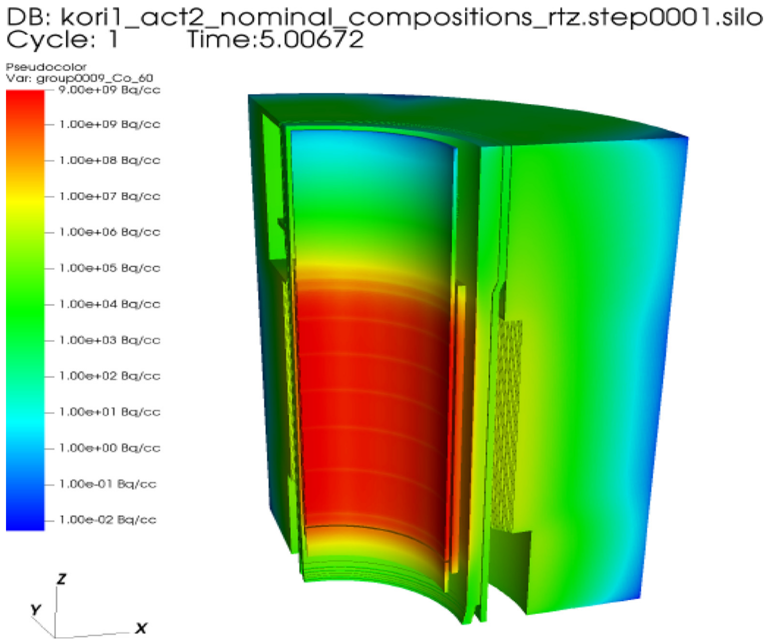

Fig. 6. Radiation level of Co-60 in 5 years after final operation

addition, the location specific survey data will be used to reduce employee exposure and perform efficient decommissioning.

In this task, radiation dose calculations are performed reflecting the entire operation cycle of the Korean plant. The cumulative neutron irradiation by location of the BFB is evaluated by applying the RAPTOR-M3G three dimensional transport calculation code and BUGLE96 neutron response section area library, reflecting the model of the reactor and the internal structure, the fuel loaded by cycle, the uranium enrichment, and combustion diagram.

Fig. 6 shows a preliminary evaluation of the expected distribution of Co-60 radiation levels five years after the permanent shutdown of the Korean plant.

In addition, we will analyze nuclides near BFBs, improve the ability to analyze radionuclides using standard materials, analyze Gama and Beta ray emitting nuclides and analyze the essential nuclides required for disposal of radioactive wastes in accordance with the NSSC's Notice 2017-60.

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 Korean plant will be surveyed.

· It is important to secure a foundation for testing technology for demonstration of high-temperature, high-pressure environment of medium-level irradiated materials.

· The five research areas of the project are 1) Technology development of facility verification for IASCC test, 2) Root cause analysis of flaw indication of Korean plant, 3) Design and development of hot cell facilities, 4) Structural analysis of flow analysis of assembly and major parts, 5) Effect of neutron irradiation.

Acknowledgements

This work was supported by Korea Institute of Energy Technology Evaluation and Planning (KETEP) grant funded by the Korea government (MOTIE) (2019151 0301140, Development of Failure and Degradation Analysis Technologies of Reactor Internal Baffle Former Bolts from Decommissioning NPP).

References

1. Z. Jiao, J. T. Busby, and G. S. Was, Deformation micro-structure of proton-irradiated stainless steels, *Journal of Nuclear Materials*, **361**, 218 (2007). Doi: https://doi.org/

10.1016/j.jnucmat.2006.12.012

2. O. K. Chopra and A. S. Rao, A review of irradiation effects on LWR core internal materials – IASCC susceptibility and crack growth rates of austenitic stainless steels, *Journal of Nuclear Materials*, **409**, 235 (2011). Doi: https://doi.org/10.1016/j.jnucmat.2010.12.001
3. Y. S. Lim, D. J. Kim, S. S.Hwang, M. J. Choi, and S. W. Cho, Effects of Proton Irradiation on the Microstructure and Surface Oxidation Characteristics of Type 316 Stainless Steel, *Corrosion Science and Technology*, **20**, 158 (2021). Doi: https://doi.org/10.14773/cst.2021.20.3.158
4. Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 1-A), EPRI, Palo Alto, CA, USA (2019).
5. S. S. Hwang, *et al.*, *PRIMA-NET workshop,* Development of failure and degradation analysis technologies of reactor internal baffle former bolts from decommissioning NPP, Jan. 16, Gimcheon, Korea (2020).
6. J. Smith, *EPRI – NRC Materials Meeting*, EPRI Irradiated Materials Testing Programs, ML13162A571, USNRC, USA (2013). https://www.nrc.gov/docs/ML1316/ML13162A571.pdf
7. C. G Lee, *Project for facility modification of baffle former bolt retrieval from Kori unit 1*, KEPCO KPS (2020).
8. ASTM E 8-01, Standard Test Methods for Tension Testing of Metallic Materials, ASTM International, West Conshohocken, PA (2001). Doi: https://doi.org/10.1520/E0008-01
9. K. H. Na, *et al.*, *KNS workshop*, Research progress and plan for the utilization of Kori 1 harvesting materials, Oct. 23, Ilsan, Korea (2019).
10. S. Hwang, *et al.*, *Development of Life Prediction and Extension Technologies of Nuclear Reactor Internals,* KAERI/RR-4098/2016, Daejeon, Korea (2016).
11. G. F. Somville, *Proc. 17th International conference on nuclear engineering* (ICONE17), ASME, July 12-16, Brussels, Belgium (2009).