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Regulatory Assessment Technologies for Aging of Reactor Vessel Internals

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1. Introduction

Since Reactor Vessel Internals (RVIs) are installed within the reactor pressure vessel and surround the fuel assemblies, some of them are exposed to the environment such as high neutron irradiation, high temperature and reactor coolant flow. Those environmental factors can cause damage to RVIs including cracks, loss of material, fatigue, loss of fracture toughness and change of dimension as the operation time of nuclear power plants (NPPs) increases. For long-term operation more than 40 years, aging management of RVIs is important.

The final objectives of this study are to establish the audit calculation system for RVIs and to develop regulatory requirements for aging assessment and management of RVIs considering their operating conditions, materials, and possible aging mechanisms.

In order to develop the audit calculation system, it is required to develop crack evaluation, seismic analysis and thermal-hydraulic analysis techniques for RVIs so that integrity of RVIs under the aging environment can be evaluated and be assured. In addition, regulatory requirements including safety review and inspection guides should be developed in order to assure the quality and uniformity of safety reviews and inspections regarding aging assessment and management of RVIs[1].

2. Categories, Aging Mechanisms and Operating Experiences

Regulatory standards and industrial codes & standards applicable to RVIs are investigated and analyzed. Overseas industrial codes and standards for RVIs include ASME B&PV Code Section III NG, France RCC-M and Germany KTA 3204 for design, and ASME B&PV Code Section XI, Japan JEAC-4205 and Germany KTA 3204 for inservice inspection. Domestic regulatory requirements regarding RVIs can be categorized as the ones applicable to construction and operational stage of NPPs. For the construction stage, regulatory requirements for design, construction and preservice inspection are provided. For the operational stage, regulatory requirements concerning inservice inspection, PSR, continued operation are provided.

Typical design characteristics of the Westinghouse (WH)-type and Combustion Engineering (CE)-type RVIs are investigated and provided. In addition, design characteristics of the domestic RVIs, which include WH, CE and Framatome types, are also analyzed. In this study, design characteristics of RVIs include detailed components of RVIs and their functions, materials, design loads, and design criteria.

Operating conditions such as neutron dose, water chemistry, and operating temperature for RVIs are analyzed. Considering those, 8 possible aging mechanisms(stress corrosion cracking, wear, fatigue, thermal aging embrittlement, irradiation assisted stress corrosion cracking, irradiation embrittlement, void swelling, stress relaxation/creep) and their thresholds or screening criteria are investigated. For aging mechanisms related to the irradiation, more data should be obtained under the pressurized water reactor condition. In addition, degradation and repair/replacement experiences of RVIs are provided.

The draft PSR safety review guide for RVIs is developed based on the analysis results of design characteristics, aging mechanisms, and operating experiences of RVIs. The draft PSR safety review guide is divided into six major chapters and one appendix. Six chapters are (1) Areas of Review, (2) Acceptance Criteria, (3) Review Procedures, (4) Evaluation Findings, (5) Implementations, (6) References.

3. Crack Initiation Prediction

The crack initiation prediction technique of the reactor internals considering fatigue and IASCC was developed, and then applied to BFA(baffle Former Assembly), LCP(Lower Core Plate), and UCP(Upper Core Plate) alignment pin.

As a result of the application, the calculated CUF, IASCC(Irradiation Assisted Stress Corrosion Cracking) initiation damage, and combined damage don't exceed 1 during design lifetime. Therefore, it is found that structural integrity of the BFA, LCP, and UCP alignment pin will be maintained during the design lifetime in the viewpoint of fatigue and IASCC.

4. Seismic Analysis

The Seismic or dynamic analysis for statically determinate beams subjected to in-phase, multi-support motions is performed theoretically and numerically.

The theoretical analysis shows the following facts: The large mass model of the statically determinate beams is approximately equivalent to the effective force model when an appropriate large mass ratio is employed for large mass model. This fact holds true for both of differential equation of motion and the finite element equation of motion.

The finite element analysis based on the large mass model shows the following facts: The accuracy of solutions of LM model is very accurate when the large mass ratio is chosen within an appropriate range. The range of the large mass ratio is 10^3 ~ 10^{11}. However, 10^7 ~ 10^8 is recommended because the range would be slightly varied depending on commercial programs and machines.

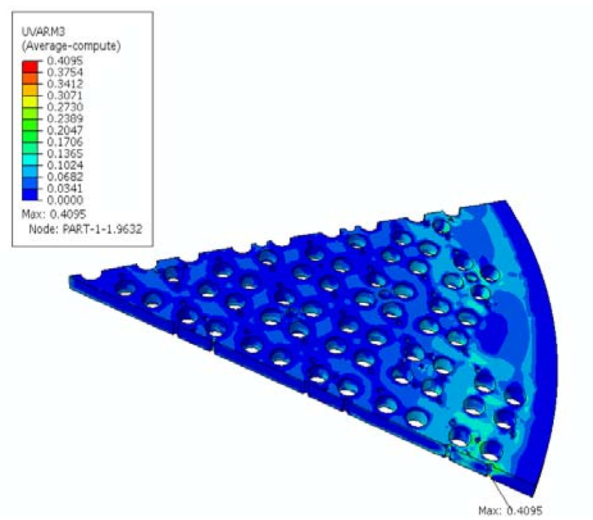

Figure 1. IASCC initiation damage of lower core plate at 40 EFPY

5. Thermal-Hydraulic Analysis

Design documents for one of domestic RVIs are analyzed. As a result, analysis models are developed for six main structures of RVIs (Core Support Barrel, Upper Guide Structure, Core Shroud Assembly, Lower Support Structure, Guide Structure Support System, Control Element Assembly Shroud) to perform thermo-hydraulic and structural analysis. In addition, virtual reality models for RVIs are also developed to understand the geometrical characteristics with the aid of the visualization technique.

The detailed analysis models are optimized for the cost-effective simulation and used in the thermo-hydraulic and structural analyses.

Thermo-hydraulic and structural analyses for RVIs are performed considering periodic pump pulsation pressure and irregular loadings due to turbulent, respectively. As a result, the maximum stress value is found at the location of cold leg where the constraint condition was applied. Analysis results considering turbulent condition showed higher stress values than the ones considering periodic pump pulsation, while maximum stress values are found at the same location for both of the cases.

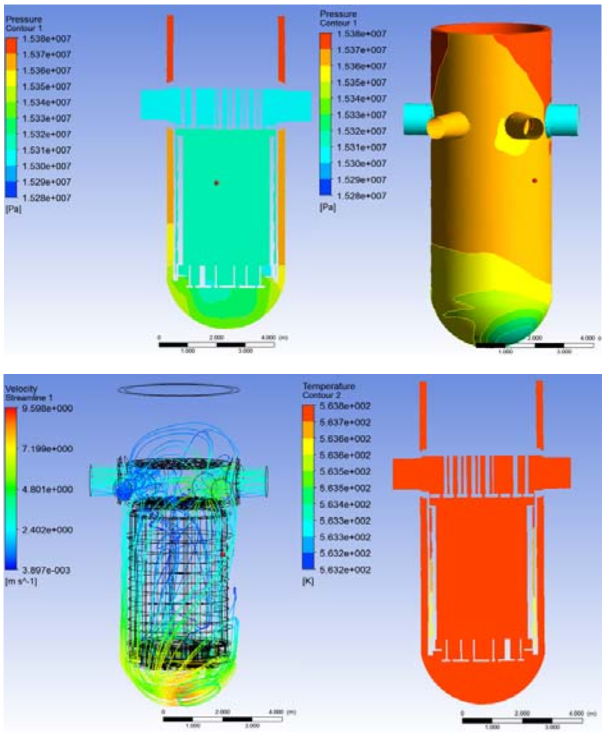

Figure 2. Pressure, velocity and temperature distributions due to random turbulence

6. Summaries

The regulatory technologies developed in this study will be utilized to establish the audit calculation system for RVIs and to develop regulatory requirements for aging assessment and management of RVIs.

The PSR safety review guide for RVIs developed in this study can provide guidance for KINS staffs to assure the quality and uniformity of the safety review regarding RVIs and aging management of them. The analysis results of the regulatory documents, design characteristics, aging mechanisms and operating experiences can also be used to develop regulatory guides including the safety review guide for the continued operation and safety inspection guide for RVIs.

Crack initiation prediction, seismic analysis and thermo-hydraulic analysis methodologies suggested in this study will be studied further and be utilized in the development process of audit calculation system for RVIs. Analysis results provided in this study using these evaluation methodologies will also be considered in the development and revision of the safety review guide for the continued operation and inspection for RVIs.

REFERENCE

[1] Jhung, M.J., *et al*., 2013, Development of Regulatory Assessment and Management Technologies for Aging of Reactor Vessel Internal, KINS/GR-526, Korea Institute of Nuclear Safety.