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A Fast Neutron Fluence Reduction Strategy for Pressure Vessel Lifetime Extension

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The feasibility of nuclear power plant lifetime extension was examined by the evaluation of fast neutron (>1 MeV) fluence at the reactor pressure vessel (RPV). A fluence reduction option, additional shields installation in outer core structures, was applied to the Kori Unit 1 reactor, and its fluence reduction effect was carefully analyzed. Full-scope explicit modeling of Monte Carlo simulation with MCNP4A code was employed in the fast neutron fluence calculations. An optimized design option was found from various choices in geometry and material for shield structure. It is expected that the magnitude of fast neutron fluence would be reduced by 39% at the circumferential weld of the RPV. On the basis of the criterion of pressurized thermal shock requirement of the RPV, it is predicted to obtain the additional life of 4.4 effective full power years. It was also investigated that the nuclear characteristics and thermal hydraulic factors at the internal core were negligibly influenced by the installation of additional shield structure.

KEYWORDS : Reactor Pressure Vessel, Vessel Lifetime Integrity, Fast Neutron Fluence, Monte Carlo Method

I. Introduction

Currently, many commercial nuclear power plants (NPP) all over the world are approaching the end of their design lifetime. As the operating years of plants grow and the public concern over nuclear safety increases, plant aging and maintenance problems become a more significant matter of concern in the NPP industry.

The lifetime of a NPP is strongly related to the retention of the structural integrity of the reactor pressure vessel (RPV). The aging mechanism of primary concern for RPVs is irradiation-induced embrittlement of material at the vessel beltline. Most of all, the neutrons with energies greater than approximately 1 MeV are the primary cause of RPV embrittlement. Therefore, if the fast neutron fluence at the RPV is effectively reduced by appropriate treatments, vessel integrity will then be ensured beyond the lifetime, and extended lifetime can be produced.

There are several fluence reduction options such as use of low leakage loading pattern, insertion of dummy assemblies at the core periphery, installation of additional shields at vessel internal structures, and power level derating.[(1)] Among the several options, additional shield installation is expected to be more easily applicable to the reactors. Furthermore, it would not cause remarkable changes in core characteristics and also would not cause loss of power generation capacity.

In this study, additional shield insertions were assumed to be applied to the outer core structures, and fluence reduction effect was examined in detail by using the Monte Carlo simulation. The changes of nuclear characteristics and thermal hydraulic factors due to additional shield insertion were analyzed, and the effectiveness of lifetime extension was evaluated quantitatively.

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II. Methodology

The Discrete Ordinates Method is generally used for the calculation of RPV fluence. This S_N transport calculation contains uncertainties associated with the multigroup cross-section libraries, multi-dimensionality, geometric approximations, and angular discretization.[(2,3)] Recently, it has been recognized that an alternative method, the Monte Carlo, is available for accurate results.

Monte Carlo analysis of the neutronic behavior of a LWR in combination with continuous cross-section data is an attractive tool which provides a detailed description of a static LWR core. Furthermore, details of the original nuclear data evaluation can be retained in the cross-section library, and self-shielding in the resolved resonance range is explicitly taken into account. Therefore, for an accurate estimate of the neutron fluence at the RPV and a reasonable description of outer fuel assemblies which is the critical source for RPV fluence, the Monte Carlo simulation is well suited for this task and was used

The ratios of the calculated using Monte Carlo method to experimental surveillance capsule fluence (C/E) for Kori Unit 1 Cycle 1 core are close to 1.0. The C/E values represent the accuracy of neutron fluence calculation.[(4)]

1. MCNP Model of the Kori Unit 1

The Kori Unit 1 is a NPP constructed by Korea Electric Power Company (KEPCO), which utilizes a Westinghouse

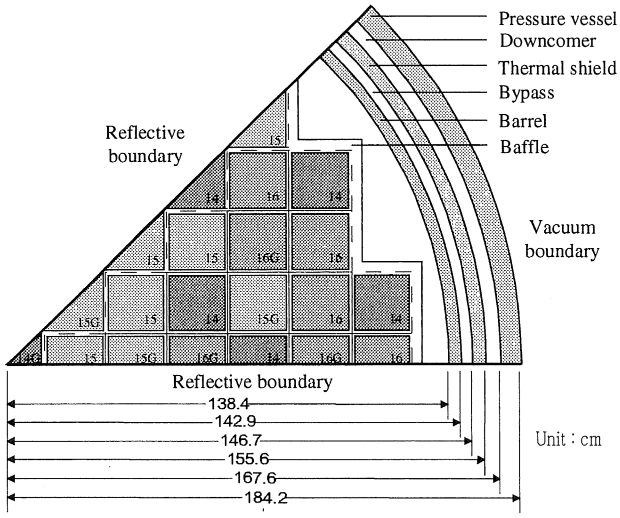

Fig. 1 Cross-Sectional View of the Kori Unit 1 Core

NDR MCNP *Diff.(%)				0.615 0.635 3.2		
			0.976 1.002 2.7	1.135 1.249 10.1	0.413 0.415 0.6	
		1.224 1.240 1.3	1.140 1.161 1.8	1.274 1.365 7.2	0.971 1.035 6.6	
	1.208 1.121 -7.2	1.194 1.174 -1.7	0.911 0.845 -7.3	1.106 1.013 -8.4	1.196 1.211 1.2	0.379 0.344 -9.4
0.943 0.796 -15.6	1.238 1.224 -1.1	1.174 1.104 -5.9	1.259 1.296 2.9	0.950 0.888 -6.5	1.180 1.191 0.9	0.710 0.691 -2.7

*Difference(%) = (MCNP-NDR)/NDR×100

Fig. 2 Relative Assembly Power Distribution of the Reference Reactor Model

nuclear steam supply system. The Kori Unit 1 reactor was chosen in this study because it is the first commercial NPP built in Korea, and thus, life extension of the plant is being considered by KEPCO.

Figure 1 shows the cross-sectional view of one-eighth of the Kori Unit 1 reactor in-vessel components with reflective angular boundaries at 0 and 45 degrees, as modeled by MCNP4A[5]. This model explicitly represents the rectangular and cylindrical domains in three dimensions, and baffle, barrel, thermal shield, and pressure vessel were described clearly. The state of beginning of cycle life in cycle 14 was simulated with HFP and ARO conditions.

To model the cycle 14 core configurations, it is necessary to have uranium, plutonium, and fission product concentrations for representation of the depleted core. For this information, associated depletion calculations were performed by CASMO-3 code[6]. For the purpose of considering the Doppler broadening effect, a new cross-section library was generated at the core temperature by NJOY[7] and ENDF/B-VI.

2. Description of RPV Fluence Calculation

A couple of calculations were carried out to estimate the fast neutron fluence at the RPV. One is a criticality calculation, and the other is a fluence calculation. The criticality calculation employs the KCODE option[5] to obtain k_{eff} eigenvalue of the system and relative power distributions. By confirming that k_{eff} was converged to unity and relative power distribution was consistent with that of the nuclear design report (NDR)[8], the validity of the model was examined.

To estimate the axial power distributions in the fuel rod, each assembly was divided into 4 segments at the axial direction. In four segments of each fuel rod, fission reaction densities were calculated. These results were used as a neutron source information for RPV fluence calculations. After 150 cycle calculations (KCODE), the reference system converged to a k_{eff} value of 0.99690 ± 0.00074. While the criticality calculation was performed, the relative pin power calculations in all rods of the four segments were carried out. Thereafter, the averaged relative assembly power was calculated from the integrated pin powers. **Figure 2** shows the relative assembly power distribution together with the values of the NDR data of the Kori Unit 1 cycle 14. The relative pin power distributions in all rods of the four segments from the previous criticality calculation were converted to the neutron source probabilities in the RPV fluence calculation. By the fixed source problem (SDEF)[5] of MCNP, the probability distribution function of the source was established such that the probability of a neutron starting in a segment of a rod was proportional to the normalized power fraction of that segment.

The pressure vessel of the Kori Unit 1 was fabricated with SA508 class 2 forging and Linde 80 flux circumferential welds. The welds undergo more severe irradiation embrittlement because of their high-copper and -nickel contents. Among the welds of the vessel, the weld situated at 37.5 cm from the centerline of RPV inner wall was highly damaged by fast neutron bombardments. Tally region was set around this high-flux weld line, and that was subdivided into 18 sub-cells by 2.5 degrees along the azimuth. Since the weld flux shows a peak at 0 degree, due to core geometry and the fresh fuel loading position, fluence reduction in this part was focused on in this study.

III. Additional Shield Installation for RPV Fluence Reduction

To intensively mitigate the high flux of the weld at the RPV inner wall, additional shields were installed in the outer core structures. This strategy is not accompanied by the modification of fuel assemblies, and thus, it is predicted that the core characteristics such as power distributions and peaking factors are influenced little by the alteration of outer core structures.

The shields were replaced in the coolant regions because the outer core structural components could not be conveniently

Table 1 Various Configurations of the Shields

Location		Geometry	Material	Specific Design Parameters
Baffle-Barrel	Outer Baffle Surface	Simple Pad	SS-304 Tantalum Graphite	Thickness : 1.5~3.0cm
	Between Baffle and Barrel	Solid Rod	SS-304 Titanium Tantalum Zircaloy-4 Graphite Borated SS	Rod Diameter : 0.9cm, Rod Pitch : 1.4cm
		Hollow Rod		Same Diameter and Pitch of Fuel Rod Rod Thickness : 1.0~4.0mm
	Inner Barrel Surface	Simple Pad		Thickness : 1.0~2.5cm
		Repeated Pad		Thickness : 0.2cm, 0.4cm 3~5 Repeated Pads with Water
Bypass	Outer Barrel Surface	Simple Pad	Tantalum	Thickness : 1.9cm
	Inner Thermal Shield Surface	Simple Pad		Thickness : 1.9cm
Downcomer	Outer Thermal Shield Surface	Simple Pad		Thickness : 1.8cm
	Between Thermal Shield and Vessel	Simple Pad	SS-304 Titanium Tantalum Zircaloy-4 Litium-6 Borated SS	Thickness : 2.0cm
		Repeated Pad		Thickness : 0.2cm, 0.4cm 3~5 Repeated Pads with Water
	Inner Vessel Surface	Simple Pad		Thickness : 1.0~5.0cm
		Repeated Pad		Thickness : 0.2cm, 0.4cm 3~5 Repeated Pads with Water

removed or changed. The space of additional shields possibly inserted are between baffle and barrel, barrel and thermal shield (bypass), and thermal shield and vessel (downcomer). Shield type was divided into two classes: rod type and pad type. The rod type shields needed relatively more volume to be installed, and thus, was only applied to the space between baffle and barrel. However, the pad type shields could be installed at all spaces of the outer core owing to its simple geometry and small size.

The shield materials were chiefly used with stainless steel type 304 (SS-304) and tantalum (Ta). It is well recognized that SS-304 is a typical reactor material and an effective neutron scatterer so that a significant number of fast neutrons are reflected back into the core. Tantalum is also an effective neutron remover widely used to fabricate nuclear reactors. In addition, tantalum is said to have desirable properties such as a high melting point, high strength, and good ductility. Moreover, many materials that are known as effective neutron absorbers and scatterers such as polyethylene, borated stainless, lithium-6, graphite, and titanium were tested for fast neutron shielding. **Table 1** lists the locations, geometries, and materials of the shields tested in this work.

The rod type shield reduced the RPV fluence somewhat for all azimuthal angles, while for the high-flux region (at 0 degree) the effect of reduction was relatively poor. The pad type shields were placed along the outer core structures such as baffle, barrel, and thermal shield. To reduce the maximum fluence around 0 degree of azimuthal angle, partial pad was installed only between 0 and 15 degrees of core azimuthal angles. Repeated pads that are repeated with the coolant by certain spacing were also tested.

Through the various shield configurations, the several cases for effective neutron removing were found as follows:

Case 1. Hollow Ta rods type shield in baffle-barrel region - 2mm and 4mm rod thickness.

Case 2. SS-304 and Ta pads on outer baffle surface - 2.9cm thickness.

Case 3. Three repeated pads of SS-304 and Ta on inner barrel surface - 0.4cm thickness per pad.

Case 4. Two Ta partial pads on outer baffle and barrel surfaces - 1.4cm and 1.9cm thickness, respectively.

Case 5. Three Ta partial pads on outer baffle, barrel and thermal shield surfaces - 1.4cm, 1.9cm and 1.8cm thickness, respectively.

Other materials except SS-304 and tantalum had little effect on the reduction of fast neutrons. It is regarded that these materials have a specific energy range for effective fast neutron removal, but in the reactor core, there are many neutrons with continuous energy spectrum.

Among the five effective shields, the Case 5 with the three Ta pads (**Fig. 3**) most efficiently reduced the high-flux of the weld, and thus, this case was selected as an optimum shield in this study. **Figure 4** shows the angular neutron fluence distribution of the Case 5, compared with normal operation and the Case 4. The maximum fluence and the total fluence were reduced by 39% and 19%, respectively.

IV. Evaluation of Vessel Integrity

1. Nuclear Characteristics Analysis

Installation of the Ta pads in outer core structures can change the core boundary conditions resulting in changes of power distribution and cycle length. However, there will be no effects on safety parameters such as FTC and MTC.

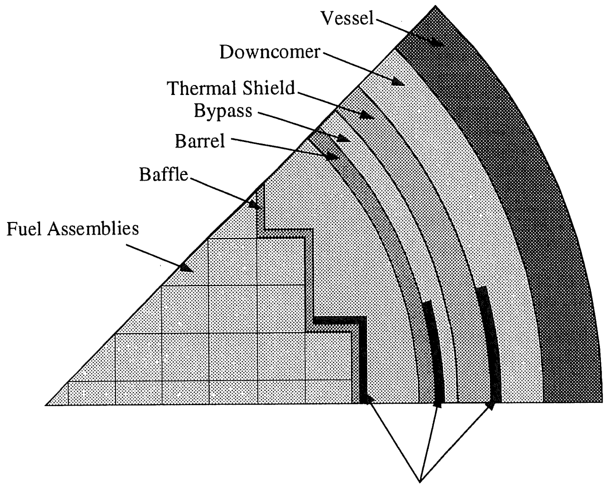

Fig. 3 Cross-Sectional View of the Three Ta Partial Pads Installation

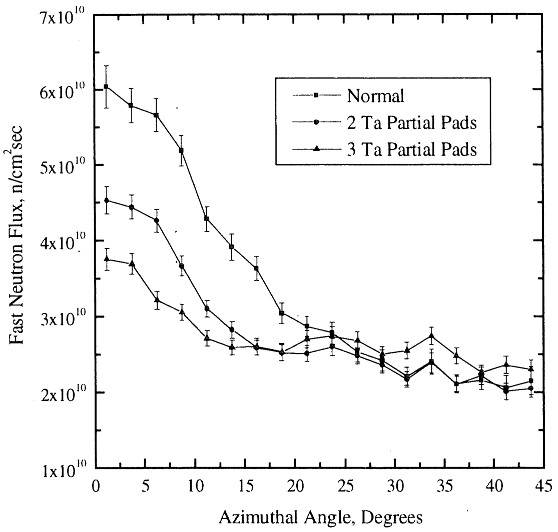

Fig. 4 Angular Fluence Distribution of Ta Partial Pad Cases

Table 2 RT_{PTS} Changes Due to the Ta Pads Installation

EFPY	Operating Year & Calendar Year	ID Fluence 10^{19} n/cm^2	RT_{PTS}
11.5 (11.5)	14.3, 1994.7 (14.3, 1994.7)	1.35 (1.35)	258.4 (258.4)
14.6 (14.6)	18.3, 1998.7 (18.3, 1998.7)	1.70 (1.70)	270.6 (270.6)
24.0 (21.8)	30.0, 2010 (27.2, 2007)	2.81 (3.07)	295.9 (300.0)
26.2 (24.0)	32.7, 2013 (30.0, 2010)	3.07 (3.49)	300.0 (305.9)
32.0 (32.0)	40.0, 2020 (40.0, 2020)	3.76 (5.01)	309.2 (320.9)

(The values in parenthesis are those at the normal case.)

Quite a few changes are expected in rod worths and shutdown margins due to the radial power distribution changes. In this study, only the effects on power peaking were, therefore, assessed for the core with Ta pads.

In order to check only the effect of the Ta partial pads insertion, the first cycle core of the Kori Unit 1 was tested instead of the reload cycle cores. The root mean square (RMS) error, which shows the extent of the radial power distribution change due due to the Ta pad insertion, was 2.12%. By virtue of the thinness of the Ta pad, radial power distribution was not changed greatly even though fast neutron fluence level was greatly reduced.

2. Thermal Hydraulic Effect of Ta Pads Installation

Since the installation of Ta pads at the bypass flow channels, i.e., the spaces between the baffle and the thermal shield, can change the thermal hydraulic parameters, the thermal hydraulic effect was investigated. Assuming the space between the barrel and baffle was fully filled with a metallic shielding pad, 0.7% increase of the core effective flow was estimated. Increment of the effective core mass flow rate can increase DNBR. One percent increase of the core effective flow increased about 0.7% DNBR, conservatively. Therefore, the installation of metallic shielding pads could increase DNBR, which was the positive effect of Ta shielding pads installation.

3. Analysis of Low Upper Shelf Toughness and Pressure Thermal Shock

Due to irradiation embrittlement, the RPV beltline material changes in the fracture toughness characteristics.[9] This can cause a failure of RPV during transients like LOCA. To evaluate the RPV integrity, U.S. NRC provided regulations about the two following issues:

- Low Upper Shelf Toughness (LUST)[10-12]
- Pressurized Thermal Shock (PTS)[13]

The above two issues were studied to evaluate the RPV integrity of the Kori Unit 1 for the extended operation period. As a result, it is verified that the Kori unit 1 has enough safety margins for fracture toughness requirements for all service loading conditions. The PTS screening criterion is the calculated RT_{PTS} to be lower than 300°F for circumferential weld of the Kori Unit 1 RPV.[14]

The reference temperature RT_{PTS} is defined as follow:

$RT_{PTS} = initial\, RT_{NDT} + \Delta RT_{NDT} + Margin$

Detailed procedures to calculate *initial* RT_{NDT}, ΔRT_{NDT}, and *Margin* are given at 10 CFR 50.61 and U.S. NRC Regulatory Guide 1.99 Rev. 2.

V. Predicted Lifetime Extension

As a reference case, *initial* RT_{NDT} and chemistry factor were given as -10°F and 196.0°F, respectively, from the surveillance data of the Kori Unit 1[15]. *Margin* was given as 56.0°F by assuming that $initial RT_{NDT}$ was accurately determined. RT_{PTS} is a function of neutron fluence. For the screening criterion of RT_{PTS}, the cumulative neutron fluence at the RPV should be less than 3.07×10^{19} n/cm^2 at the end of design lifetime.

In the case of no Ta pads installation, RT_{PTS} will exceed the screening criterion in the year 2007, which is within the designed lifetime. In that case, detailed PTS analysis will be needed to ensure the RPV integrity.

When the proposed Ta pads are installed, fast neutron fluence is reduced considerably, and RT_{PTS}, will exceed the screening criterion in the year 2013 as shown in **Table 2**. If the proposed Ta pads are installed, screening criterion will be satisfied within the extended lifetime. Table 2 shows that Ta pads are very effective in lowering the RT_{PTS}. In order to extend the reactor lifetime to the year 2020, however, a more effective fluence reduction program should be prepared in addition to Ta pads installation.

VI. Conclusions

The computational model for RPV dosimetry was applied to the Kori Unit 1 reactor to quantitatively investigate the feasibility of lifetime extension. The fuel loading of cycle 14 was explicitly described by pin-by-pin with MCNP4A Monte Carlo code, and the validity of modeling was confirmed by criticality calculation. Around 0 degree of core azimuthal angle, the fluence showed a peak value due to the core geometry and the fresh fuel loading position, therefore the study focused on the fluence reduction at this high-flux region.

The strategy of additional shields insertion in outer core structures was applied to reduce the RPV fluence. After

various geometries and materials were considered for the additional shields, an optimum shield which consists of three partial pads fabricated with tantalum was finally determined. The maximum fluence at the weld was reduced by 39%. It was investigated that the core nuclear characteristics such as power distribution and peaking factors were little affected by the Ta pads insertion. The bypass flow reduction due to insertion of the pad type shield increased effective core mass flow rate, and thus, optimistically increased DNBR.

On the basis of fracture toughness requirements, so called Charpy upper shelf energy (USE) and PTS criteria, 4.4 EFPYs of additional life was expected. It is expected that the results of this work will be easily applicable for analyzing the lifetime extension of other old reactors.

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