REACTOR VESSEL FLUENCE REDUCTION APPROACH THROUGH LOW-LEAKAGE FUEL MANAGEMENT AT KORI UNIT 1

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ABSTRACT

An analytical approach to evaluate the relative degree of reactor vessel fluence was performed to support the Kori Unit 1 plant lifetime extension program which has been carried out by Korea Electric Power Corporation (KEPCO) since 1993. In order to reduce reactor vessel fluence, neutron economy requirements regarding lower-leakage of neutron in the core has been considered in determining a loading pattern. The radial peaking factor limit affecting neutron leakage in the core is appropriately relaxed with extra thermal margin ensured from both new steam generators to be installed in 1998 and an advanced thermal design procedure. Using the relaxed peaking factor limit established appropriately, several lower-leakage loading patterns are modeled and analyzed to see a relative effect of each loading pattern on the reactor vessel fluence. It is found that the lower-leakage fuel management, with the relaxed peaking factor limit, can not only reduce the reactor vessel fluence enough to extend the Kori Unit 1 reactor vessel lifetime to 32 EFPY , but also increase the plant availability by improving fuel efficiency. Therefore, it is concluded that the lower-leakage fuel management strategy in Kori Unit 1 is the most economical way to extend its reactor vessel lifetime.

INTRODUCTION

The reactor vessel lifetime is primarily determined by both the upper shelf energy (USE) and the pressurized thermal shock (PTS). Many early generations of nuclear power plants are now faced with reactor vessel integrity problems during PTS. In the PTS events a vessel becomes prone to embrittlement due to fast neutron irradiation and the existence of flaws on the inside surface of the pressure vessel. The PTS rule made by the U.S. NRC establishes pressure vessel screening criteria in terms of a reference temperature (RT) for nil-ductility transition (NDT), or RT_{PTS}, to ensure that the risk from PTS events is acceptable (NRC, 1994). The RT_{PTS} is defined in terms of the vessel neutron (with energy > 1 MeV) fluence at the clad/base metal interface and specifies limits of 130 ^{o}C (270 ^{o}F) for plates and axial welds, and 150 ^{o}C (300 ^{o}F) for circumferential welds in the core beltline. The reduction strategy of the fast fluence with lower leakage loading pattern or fuel assembly replacement, or both, have been employed in many plants (Yang, 1994).

Kori Unit 1, a two-loop Westinghouse type pressurized water reactor, began commercial operation in 1977 and used the out-in high leakage fuel management strategy until Cycle 4. However, a low-leakage fuel management scheme has been adopted from Cycle 5 to the current Cycle 16. In order to assess the feasibility of the Kori Unit 1 design lifetime extension, an evaluation of the reactor vessel life time is now being carried out by KEPCO. As a part of this study, the accumulated vessel fluence of Kori Unit 1 through Cycle 13 was determined using the calculated results obtained with the discrete ordinate transport theory code, DOT, and combining the results with the analyzed consequence of the irradiated surveillance capsule obtained with assuming no fuel management change. Subsequently, the fluence was projected through the extended end of life 32 effective full power years (EFPY) (Hong, et al, 1994). This report determined that RT_{PTS} of the weld material met the screening criteria during the current design lifetime, however, a vessel fluence reduction program, such as lowering leakage loading patterns below the current loading pattern, is required to achieve the goal for the vessel lifetime extension.

The thermal design margin of Kori Unit 1 was investigated to assess the possibility of the radial peaking factor limit relaxation. From the thermal margin point of view, both steam generator replacement and adoption of an advanced thermal design procedure were considered. Assuming the relaxed radial peaking factor limits which were determined by evaluation of the design margin, several loading patterns were generated to evaluate the effect of each pattern on the power level of the core edge facing the circumferential weld and the fluence accumulation of the pressure vessel. The fluence at the vessel is calculated using MCNP code (Breismeister, 1993). Based on the RT_{PTS} assessment, a desired fuel management scheme was derived.

INVESTIGATION OF THERMAL MARGIN

KEPCO, Korea Nuclear Fuel Company (KNFC) and Westinghouse have jointly performed a Reload Transition Safety Analysis (RTSA) for Kori Unit 1 to load an Optimized Fuel Assembly (OFA) on Cycle 16 which is the first transition cycle from Joint Design Fuel Assembly (JDFA). The OFA enables the LOCA F_Q limit to be relaxed by 11. 5% compared to the JDFA, while the $F^N{}_{DH}$ limit of OFA is the same as the JDFA.

The design of a lower-leakage loading pattern in the reload design is mainly dependent on the $F^N{}_{DH}$ limit. The higher the radial peaking factor limit is, the more the fresh fuel can be located in the core interior region, not the peripheral region. As a result, radial neutron leakage can be reduced with a lower power level at the peripheral region. Considering an appropriate level of thermal margin is required to ensure the operational safety, it is thus necessary to optimize the radial peaking factor limit to achieve a desirable lower-leakage fuel management for Kori Unit 1, whose reactor vessel lifetime can be extended.

From the thermal margin point of view, Kori Unit 1 has some potential advantages, since the old steam generator is going to be replaced with a brand new one to improve the deteriorated performance due to increased tube plugging rate. It is clear that no, or a very low, tube plugging rate in the new steam generators increases Reactor Coolant System (RCS) flow rate and the thermal margin is accordingly increased. It is generally known that a 5% reduction in the tube plugging rate gives about 2.5% additional departure from the nucleate boiling ratio (DNBR) margin. Furthermore, an advanced thermal design procedure such as the revised thermal design procedure (RTDP) is able to provide an additional thermal design margin compared to the improved thermal design procedure (ITDP) that is now being applied to Kori Unit 1 thermal design. It is generally known that the RTDP also gives about 8% additional margin compared with the ITDP. Both the new steam generator to be installed in 1998, and the advanced thermal design procedure such as RTDP, are thus enough to yield additional thermal margin which can be directly used to relax the radial peaking factor limit. The preliminary study confirms that the $F^N{}_{DH}$ limit can be considerably relaxed if the advanced thermal design procedure, such as the RTDP, is employed for Kori Unit 1 core design with an appropriate steam generator tube plugging rate and the sufficient current LOCA F_Q margin.

FUEL MANAGEMENT OPTIONS

Several fuel management options have been studied to reduce the neutron leakage from the core peripheral assembly. For each fuel management option, the fuel cycle management study was performed to achieve an ideal equilibrium cycle with three batches (40 feed fuel assembly) whose enrichment is 3.8 w/o of U^{235}. Current fuel inventory is also 3.8 w/o of U^{235}. WABA rods are inserted in some of the fresh fuel assemblies to control the maximum radial peaking and the moderator temperature coefficient (MTC). The core is modeled using ALPHA/PHOENIX-P (Nguyen, et al, 1988) and ANC (Liu, et al, 1985) to simulate the three-dimensional quarter core. The MCNP code (Breismeister, 1993) was used to calculate flux at various vessel locations of interest.

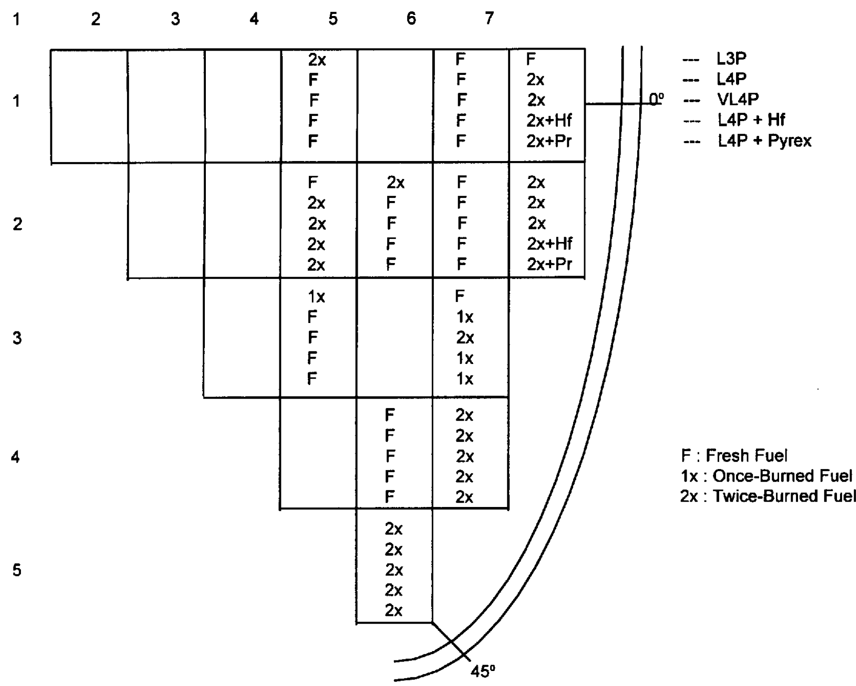

Figure 1
Octant Core Loading Pattern for Each Case

Based on relaxation of the current F^N_{DH} limit mentioned above and the most feasible and realistic fuel management consideration for Kori Unit 1, the following five fuel management schemes have been chosen to evaluate the effects of each fuel management strategy on the reactor vessel fluence as well as fuel cycle economics.

1. low-leakage loading pattern (L3P)
2. low-low-leakage loading pattern (L4P)
3. very low-low-leakage loading pattern (VL4P)
4. L4P plus Hafnium absorber (L4P + Hf)
5. L4P plus Pyrex absorber (L4P + Prx)

Figure 1 shows each fuel management conditions. L3P has fresh fuel, at the core periphery position, while L4P has no fresh fuel, but all twice-burned fuels at the peripheral positions, except the locations (3,6) and (6,3), which were filled with once-burned fuels as shown in Figure 1. L3P is the loading pattern of Kori Unit 1 currently being used in the reload core design. In VL4P, shown in the figure, all the core peripheral locations are filled with twice-burned fuels. L4P+Hf and L4P+Prx has Hafnium annular absorber and Pyrex absorber, respectively, with L4P to reduce the fluence on the circumferential weld part located 14.75 inch below the axial midplane of the active core. Sixteen rods 1.83m (72 inch) long per assembly are inserted to be axially symmetric on the flat assembly at 0 degrees as shown in Figure1.

RESULTS

For each loading pattern, the core peripheral assembly power and the required maximum F^N_{DH} limit are evaluated. L4P requires an increase of about 5.1% F^N_{DH} compared with the current limit being used in L3P. However, the average peripheral assembly power decreases by about 20% and the assembly power of the flat at 0 degrees decreased by about 47%. In order to achieve the VL4P fuel management, it was determined that a relaxation of about 8% in the F^N_{DH} limit is required compared with L3P. The average peripheral assembly power decreases by about 29% and the assembly power of the flat at 0 degrees by about 48%. The energy gain due to the lower leakage loading patterns are 5 and 7 effective full power days (EFPD) for L4P and VL4P, respectively. These correspond to a fuel cycle cost decrease of about 1.4 and 2% per cycle, respectively. The sensitivity calculation of fluence at various vessel locations for L3P, L4P and VL4P indicate that the critical weld location is at 0 degrees and the fluence reduction in this location is about 34% for L4P and 39% for VL4P, respectively, as compared with the L3P.

Table 1 Comparison of Fluence Reduction Fuel Management Schemes

Item / Case	EFPD	Max F^N_{DH}	No. of BA	No. of Hf, Pyrex	Average Peripheral Power (Decrease, %)	Flat Assy Power (Decrease, %)	Fluence Decrease in 0 degree (%)	Life Time Increase (EFPY)
L3P	351	1.435*	208	-	0.549 (-)	0.819 (-)	-	-
L4P	356	1.508	208	-	0.439 (20)	0.431 (47)	34	7.0
VL4P	358	1.550	192	-	0.390 (29)	0.423 (48)	39	8.6
L4P+Hf	350	1.522	176	192 Hf	0.404 (26)	0.326 (60)	64	23.6
L4P+Prx	353	1.512	192	192 Prx	0.420 (23)	0.373 (54)	44	10.6

*** Current design limit**

The assessment of RT_{PTS} also indicates that the L4P and the VL4P can yield an additional 7 and 8.6 EFPY, respectively, as compared with the current L3P before reaching the screening criteria. Table 1 summarizes the calculation results and shows the comparison of each of the fuel management characteristics with the impact on vessel fluence. It appears that the feed fuel assembly on the flat location at 0 degrees mainly contributes to an increase in the average peripheral power.

The use of flux suppression rods such as Hafnium and Pyrex is useful to reduce the flux in the flat assembly, but impacts on the control of the pin power of the core interior regions and the cycle reactivity. As a result, it needs a higher value of F^N_{DH} limit than the L4P assembly. It appears that a relaxation of about 7% in the F^N_{DH} limit is needed to control the pin power for L4P with Hafnium or Pyrex. Since the use of flux suppression rods is unfavorable to the energy gain, due to the negative reactivity insertion, the cycle length of L4P with Hafnium becomes 1 EFPD shorter than the L3P. It is known that the 1.83m (72 inch) long flux suppression rods arranged symmetrically can be replaced with asymmetric 0.91m (36 inch) long rods if there is no impact on the calibration of the NIS reactor protection system (Frank, et al, 1991). By adopting this approach rather than the symmetric 1.83m (72 inch) long rods, the cycle length of L4P with Hafnium conversely becomes 1 EFPD longer than L3P. These kinds of similar trends also appear for L4P with Pyrex. In particular, the insertion of Hafnium or Pyrex rods to suppress the flux at the weld location enables the vessel lifetime to extend by more than 23.6 or 10.6 EFPY, respectively, compared with the current L3P. Figure 2 shows the lifetime extension effect in terms of RT_{PTS} for each loading pattern compared with L3P. As shown in the figure, it is apparent that the vessel lifetime extension is heavily dependent on the fuel management scheme.

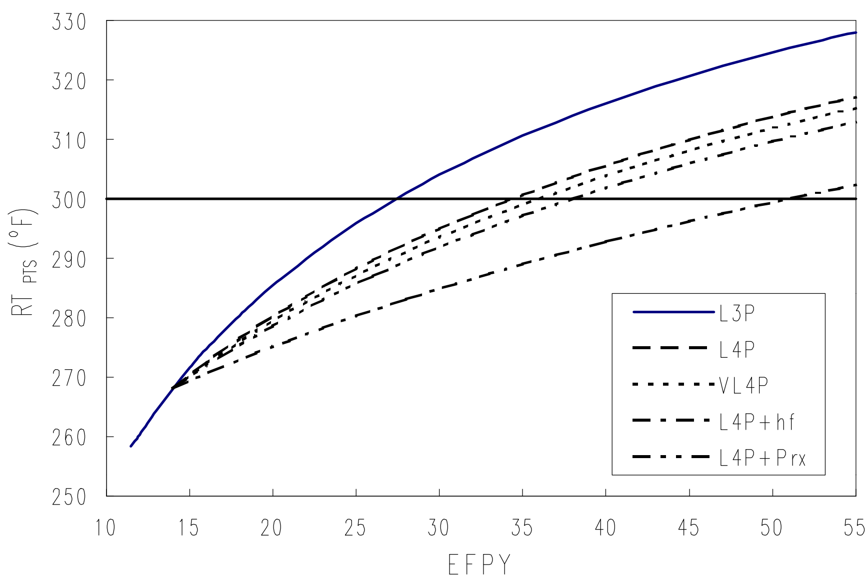

Figure 2 Pressure Vessel Lifetime Extension as Fuel Management

When we relax the F^N_{DH} limit to a higher value using the thermal design margin identified above, compared to the current limit, it is feasible that lower leakage fuel management could reduce the fast flux level and could increase fuel cycle economics. Figure 3 shows the influence of peripheral assembly power on radial neutron leakage for two loop PWRs (Fecteau, 1991) with the relation to the maximum F^N_{DH} (dotted line) obtained from each loading pattern in this study. The higher value of the F^N_{DH} limit is generally required to achieve lower leakage loading patterns in this figure. The dotted F^N_{DH} curve obtained in this study was expected to be linear, but is not, because of a specific loading pattern dependence. It is possible to reduce the vessel fluence if we adopt another approach, such as a reduction of the number of feed fuel assemblies to avoid the placement of a feed fuel assembly in the peripheral location. However, relaxation of the radial peaking factor limit is expected for the reduced feed size core, since the power sharing of the fresh fuel located inside of the core has to be increased to compensate for the reduced peripheral power. Fuel management strategy with 36 feed fuel assemblies enables twice-burned fuel to be located on all the peripheral locations except the (3,6) and (6,3) locations. Under this loading pattern, the average peripheral assembly power is 0.517 and the F^N_{DH} limit increases by about 2.5% to meet the vessel lifetime extension requirement of Kori Unit 1. Consequently, from the requirement of the F^N_{DH} limit relaxation point of view, the reduced batch size of 36 feed fuel assemblies is favorable when compared to the 40 feed fuel management. However, the reduced batch size makes the cycle length decrease by about 28 EFPD compared to the L4P.

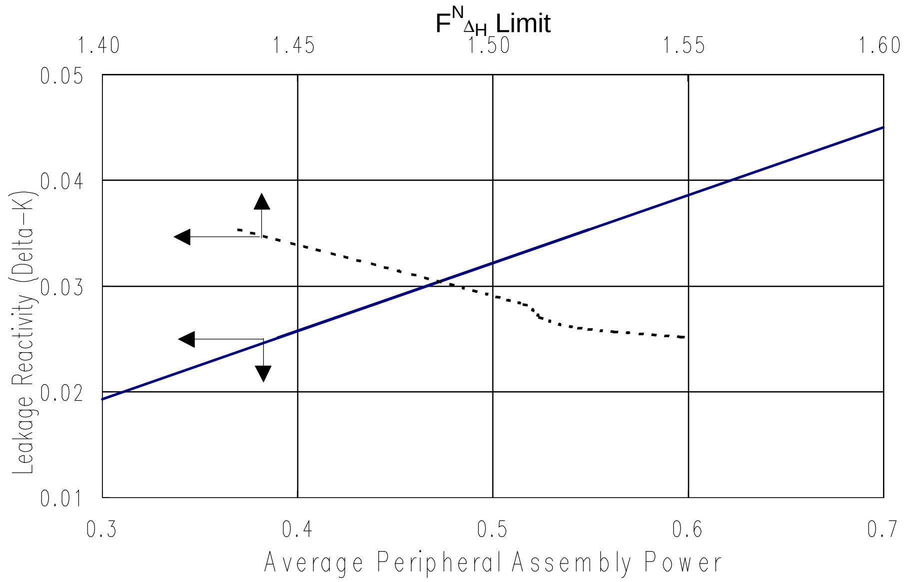

Figure 3 Radial Leakage and F^N_{DH} limit Compared with Average Peripheral Assembly Power

CONCLUSION

Using the new steam generator tube plugging rate assumption, the RTDP and the margin of LOCA F_Q limit, it is confirmed that the radial peaking factor limit F^N_{DH} at Kori Unit 1 can be relaxed to support the L4P or the VL4P fuel management scheme, with three batches (40 feed fuel) whose enrichment is all 3.8 w/o of U^{235}. It is also confirmed that the L4P or the VL4P fuel management scheme is needed to reduce the reactor vessel fluence enough to meet the desired 32 EFPY target life. Furthermore, the L4P fuel management strategy with the flux suppression rods such as Hafnium and Pyrex, can minimize the vessel fluence and thus further increase the vessel lifetime as compared with the L4P option. In addition, the L4P and the VL4P fuel management strategies can improve fuel cycle cost economy by about 1.4% and 2% per cycle, respectively, compared with the L3P being used in the current core design of Kori Unit 1. Therefore, it is concluded that the lower-leakage loading pattern, rather than the current L3P fuel management strategy having relaxed F^N_{DH} limit. is the most economical way to extend the Kori Unit 1 reactor vessel lifetime by the desired level and to improve fuel cycle cost economy.

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