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THE EFFECT OF AXIAL FUEL ROD POWER PROFILE ON FUEL TEMPERATURE AND CLADDING STRAIN

by

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The most limiting design criteria for nuclear reactor normal operating conditions (ANS Condition I) are known to be rod internal pressure and cladding oxidation, while those for nuclear reactor transient operating conditions (ANS Conditon II) to be fuel centerline temperature and transient cladding total tensile strain. However, the design margins against fuel temperature and transient cladding tensile strain become smaller since power uprating is being or will be utilized for the most of nuclear power reactors to enhance the economics of nuclear power. In order to secure sufficient design margins against fuel temperature and cladding total tensile strain even for power uprating, the current axial rod power profiles used in the reactor transient analysis were optimized to reduce over-conservatism, considering that 118% overpower of a steady-state peak rod average power was not exceeded during the reactor transients. The comparison of the current axial rod power profiles and the optimized ones indicates that the latter reduces the fuel centerline temperature and cladding total tensile strain by 26 °C and 0.02%, respectively.

Key words: nuclear fuel rod, design criteria, fuel centerline temperature, cladding total tensile strain, axial rod power profile

INTRODUCTION

Fuel design requirements are defined in the NRC Standard Review Plan (SRP) to satisfy 10CFR50 (Appendix A "General Design Criteria for Nuclear Power Plants"). In detail, the reactor core and associated coolant, control, and protection system shall be designed with the appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. The NRC SRP4.2 describes fuel system design criteria covering fuel system damage, fuel rod failure and fuel coolability. The fuel system damage and fuel rod failure criteria employed by the most of fuel vendors are steady-state cladding stress and strain, cladding fatigue, cladding fretting wear, cladding oxidation and hydriding, fuel rod growth, fuel rod internal pressure, fuel rod internal hydriding, cladding flattening, fuel centerline temperature, transient cladding stress and cladding total tensile strain, and departure from nucleate boiling ratio (DNBR). The fuel rod integrity is to be evaluated during the reactor normal operation (ANS Condition I) and the reactor anticipated operational occurrences (ANS Condition II) [1]. It is noteworthy that the ANS Condition II related design criteria include fuel centerline temperature, transient cladding stress and total tensile strain, and DNBR. It has been found that the most limiting design criteria for nuclear reactor normal operating conditions may be rod internal pressure and cladding oxidation, while those for nuclear reactor transient operating conditions may be fuel centerline temperature and cladding total tensile strain. Recently, the fuel reliability has been steadily increasing with the help of various robust designs against fuel failure [2-4]. However, fuel vendors are striving to employ reactor power uprating up to about 20% and to increase the batch average fuel burnup from 45.000 MWd/MtU to 55.000 MWd/MtU [5, 6]. With the introduction of power uprating into power reactors, it is obvious that design margins against fuel centerline temperature and cladding total tensile strain will become smaller. It is reported that the Korea Hydro & Nuclear Power Company (KHNP) plans to adopt power uprating up to 5% in some PWR in Korea in the near future. Therefore, it is necessary to have sufficient design margins against fuel centerline

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temperature and cladding total tensile strain even for anticipated power uprating since the most limiting design criteria for nuclear reactor transient operating conditions are fuel centerline temperature and cladding total tensile strain, as mentioned above.

For a 14 14 fuel assembly integrity evaluation under the reactor anticipated operational occurrences (ANS Condition II) in Korea, a box-shaped transient axial rod power profile simulating all control rod-out condition is employed below 380 W/cm, while a bottom-skewed transient one simulating some control rods' insertion is used above 380 W/cm without changing its axial profile with the increase of transient power levels. However, it is found that the current bottom-skewed axial profile generates somewhat a rod-averaged power higher than 118% overpower of a steady-state peak rod-averaged power during the reactor transients. It should be noted that 118% overpower of a steady-state peak rod-averaged power is the maximum rod-averaged power allowed during the reactor transients. In this study, therefore, optimized bottom-skewed transient axial rod power profiles above 380 W/cm are proposed as a function of transient peak power in order to reduce over-conservatism included in the current bottom-skewed axial profile and provide more design margins against fuel centerline temperature and cladding total tensile strain.

PREPARATION OF OPTIMIZED BOTTOM-SKEWED AXIAL PROFILES

The currently used box-shaped and bottom-skewed axial rod power profiles are shown in figs. 1 and 2, respectively. The box-shaped axial profile is used when a rod power increases from a steady-state power to the transient power of 380 W/cm (see step 1 in fig. 3), while the bottom skewed axial profiles are used when a rod power increases from a steady-state power to the

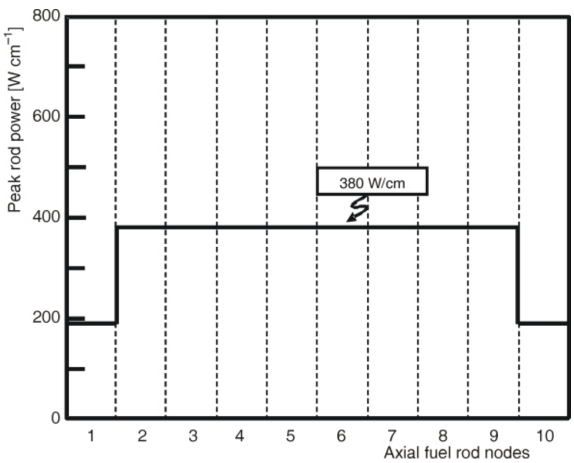

Figure 1. Current box-shaped axial power profile at a peak power level of 380 W/cm

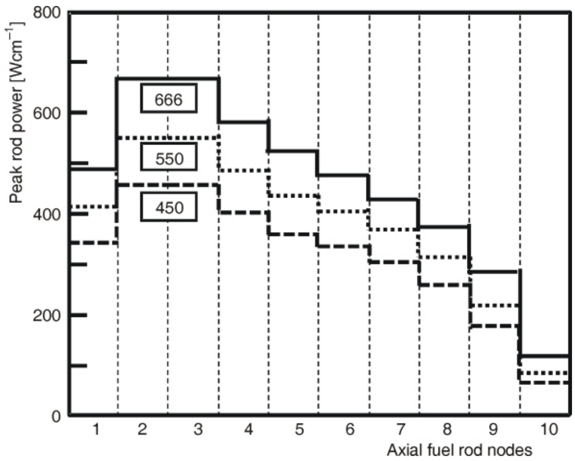

Figure 2. Current bottom-skewed axial power profiles at peak power levels greater than 380 W/cm

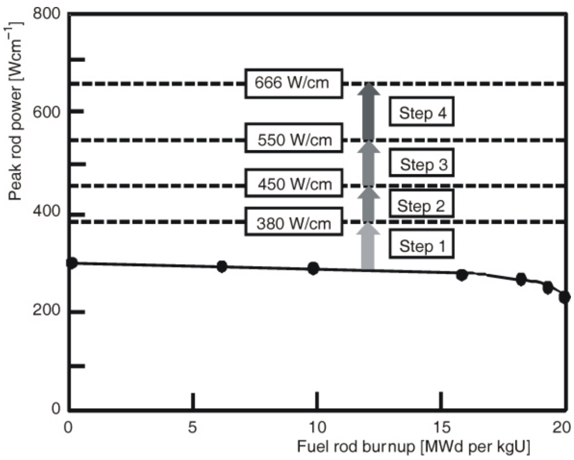

Figure 3. Fuel rod power history and power transient simulation

transient powers above 380 W/cm such as 450, 550, and 666 W/cm (see steps 2, 3, and 4 in fig. 3). The current bottom-skewed axial profiles represent rod averaged powers much higher than the 118% overpower percentage allowed during the ANS Condition II, which generates over-conservative transient average powers above 380 W/cm. In order to reduce over-conservatism included in the current bottom-skewed axial profiles, optimized bottom-skewed axial profiles are proposed at the transient power levels of 450, 550, and 666 W/cm, as shown in fig. 4. It is noteworthy that the areas under the bottom-skewed axial profiles represent transient rod averaged powers. The optimized bottom-skewed axial profiles generate more or less the same rod averaged power as the 118% overpower percentage of the maximum steady-state rod averaged power. The transient rod averaged power used for the optimized bottom-skewed axial profiles may be given as follows

$$q_t \quad [\text{overpower percentage}] F_{\Delta H}\, q_{avg} \qquad (1)$$

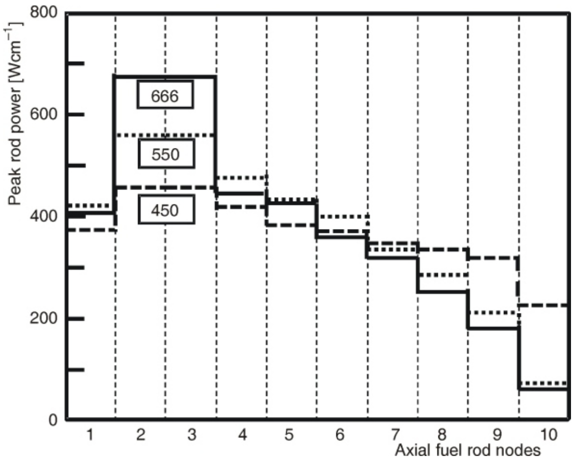

Figure 4. Optimized bottom-skewed axial power profiles at peak power levels greater than 380 W/cm

where q_t is the transient rod averaged power over-power percentage = 118%, F_H – the maximum steady-state radial power factor during the reactor normal operation, and q_{avg} – the core averaged linear power.

For the 14 14 fuel assembly considered in this study, F_H and q_{avg} are given as 1.49 and 212.3 W/cm.

CALCULATION PROCEDURES

With the help of a fuel rod performance analysis code [7], fuel centerline temperature and cladding total tensile strain during the ANS Condition II are calculated for the 14 14 fuel assembly considered in this study. The key input data for the performance analysis code are composed of the reactor core and fuel assembly design parameters (see tab. 1) as well as of the fuel rod dimensions and performance model constants (see tab. 2). For the fuel centerline temperature calculations, the maximum pellet-to-cladding gap and fission gas release are simulated, while for the cladding total tensile strain calculations, the minimum pellet-to-cladding gap and fission gas release are simulated. In detail, for fuel centerline temperature calculations, the minimum values of pellet diameter, dishing and plenum volume, radial relocation model constant, swelling model constant, and cladding creep model constant are utilized, whereas the maximum values of pellet porosity, cladding outer and inner diameters, initial rod pressure, fission gas release model constant and maximum densification model constant. The input data for cladding total tensile strain calculations are just the opposite to those of fuel centerline temperature calculations. The transient peak rod power allowed during the ANS Condition II is 666 W/cm, which falls down from a rod burnup of about 20 MWd/kgU due to the decrease in reactivity with the increase of burnup, as shown in fig. 5. In this study, therefore, the rod burnup of up to 20 MWd/kgU is

Table 1. Reactor core and fuel assembly design parameters for 14 14 PWR in Korea

Design parameters	Values
Core thermal power	1723.5 MW_t
Core average linear power	212.3 W/cm
Local peak linear power achievable for ANS Condition II	666 W/cm
Inlet coolant temperature	282.3 °C
Reactor system pressure	155 bar*
Number of fuel assemblies in the core	121
Number of fuel rods per fuel assembly	179
Fuel rod pitch	14.12 mm

*1 bar = 100 kPa

Table 2. Fuel rod dimensions and performance model constants

Input variable	Input data		
	Minimum	Average	Maximum
Pellet porosity [%]	3.73	5.10	6.50
Pellet diameter [mm]	8.04	8.05	8.06
Clad outer diameter [mm]	9.45	9.50	9.55
Clad inner diameter [mm]	8.18	8.22	8.26
Dishing volume [mm^3]	8.00	11.00	14.00
Plenum volume [mm^3]	5.70	6.40	7.00
Initial rod inner pressure [bar]	21.50	22.50	23.50
Fission gas release model constant	8.00		38.00
Radial relocation model constant	0.48		0.83
Swelling/densification model constant	0.34/3.48		0.46/4.10
Clad creep model constant	0.77E-20*		1.10E-20

*read as 0.77 10^{-20}

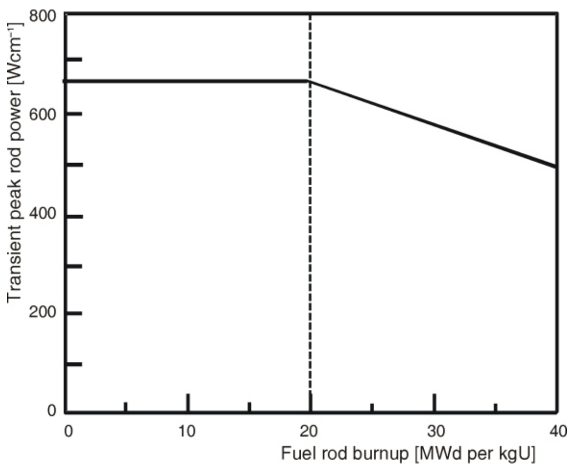

Figure 5. Transient peak rod power *vs.* fuel rod burnup

considered because the transient power decrease below 666 W/cm after 20 MWd/kgU will reduce drastically the fuel centerline temperature and the cladding total tensile strain. The peak steady-state rod power used in this study is given in fig. 3 and the power ramps from the steady-state power to various transient power levels of steps 1 through 4 initiate at the burnups of 0, 6.5, 9.7, 16.2, 18.3, and 20.0 MWd/kgU.

RESULTS AND DISCUSSION

The fuel centerline temperatures and the total tensile strains at 666 W/cm *vs*. the fuel rod burnup are plotted in figs. 6 and 7, respectively. From fig. 6, it can be seen that the fuel centerline temperatures increase with the increase of the fuel rod burnup and the fuel temperatures for the optimized bottom-skewed axial profiles are lower than those for the current bottom-skewed axial profiles, indicating that the over-conservatism included in the latter is reduced. The fuel centerline temperature at the rod burnup of 20 MWd/kgU reaches 2730 °C for the current bottom-skewed axial profiles, while it reaches 2704 °C for the optimized bottom-skewed axial profiles, indicating that the fuel centerline temperature is reduced by 26 °C, comparing with that for the current bottom-skewed axial profiles. It should be noted that the current bottom-skewed axial profiles exceed the design criterion for fuel centerline temperature at the rod burnup of 20 MWd/kgU, while the optimized ones meet the design criterion. The reason that the optimized bottom-skewed axial profiles reduce fuel centerline temperature is that the total energy contained in the fuel rod at the power level of 666 W/cm is lower for the optimized axial profile, comparing the areas under the bottom-skewed axial profiles at 666 W/cm (see figs. 2 and 4), and consequently the amount of fission gas release is lower for the optimized axial profile, as shown in fig. 8. Naturally, the highest fractional fission gas release at 20 MWd/kgU generates the lowest pellet-to-cladding gap conductance even though the pellet-to-cladding gap is the largest at around 8 MWd/kgU (see fig. 8) and subsequently produces the highest fuel temperature at 20 MWd/kgU. On the other hand, the optimized axial bottom-skewed axial profiles might also exceed the fuel centerline melting temperature when a reactor is up-rated only by a few percent. Therefore, it is recommended that the current pellet microstructure be improved against fission gas release. Some fuel vendors have already developed Cr_2O_3-doped pellets or Al_2O_3-doped pellets to increase pellet grain size and then reduce fission gas release [8-10].

From fig. 7, it can be seen that the cladding total tensile strain decreases up to the fuel rod burnup of 7 MWd/kgU and then increases all the way after this burnup. Also, the total tensile strains for the optimized

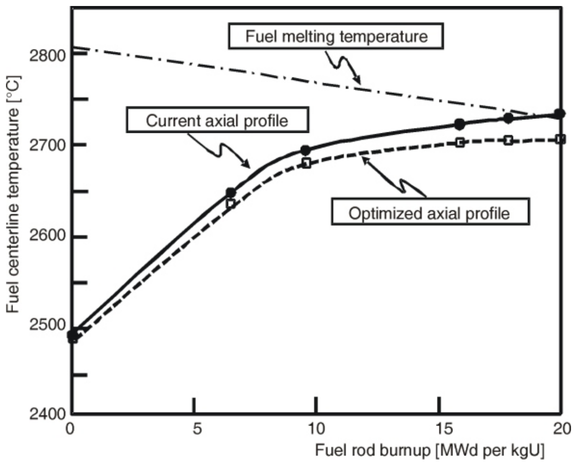

Figure 6. Fuel centerline temperatures at 666 W/cm *vs*. fuel rod burnup

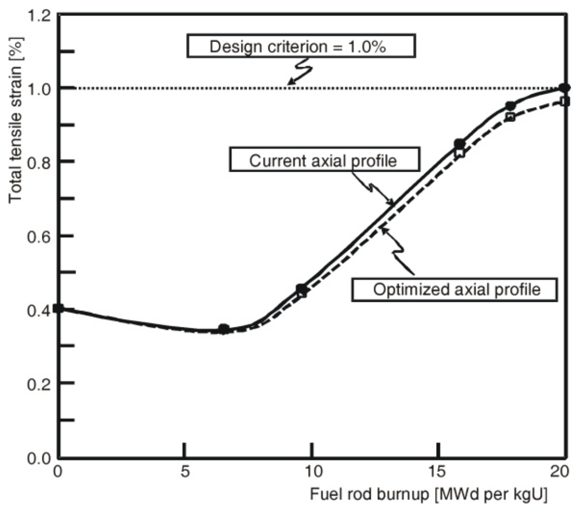

Figure 7. Total tangential strains at 666 W/cm *vs*. fuel rod burnup

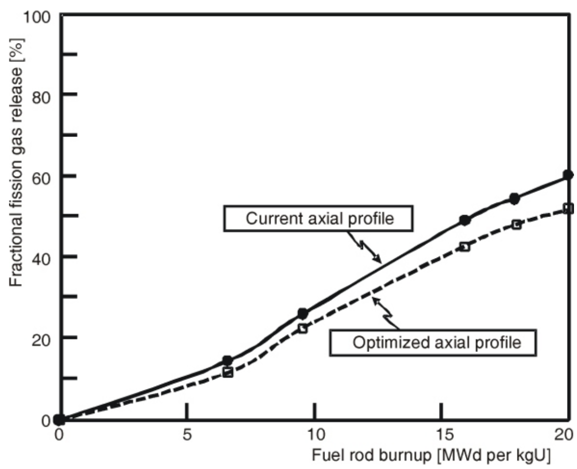

Figure 8. Fractioal fission gas release *vs*. fuel rod burnup

bottom-skewed axial profiles are lower than those for the current bottom-skewed axial profiles, indicating that the over-conservatism included in the latter is reduced. The total tensile strain at the rod burnup of 20 MWd/kgU reaches 1.0% for the current bottom-skewed axial profiles, while that reaches 0.98% for the optimized bottom-skewed axial profiles, indicating that the total tensile strain is reduced by 0.02%, comparing with that for the current bottom-skewed axial profiles. It should be noted that the current bottom-skewed axial profiles barely meet the total tensile strain design criterion of 1.0% at the rod burnup of 20 MWd/kgU, indicating that a small amount of power up-rating will exceed the total tensile strain design criterion with the use of the current bottom-skewed axial profiles. The reason that the lowest total tensile strain occurred at about 7 MWd/kgU can be explained in the following. The cladding total tensile strain is controlled by pellet-to-cladding gap sizes before the power transients and power transient-induced fuel temperatures. The pellet-to-cladding gap sizes before the power transients are shown in fig. 9. The pellet-to-cladding gap size is a function of cladding creepdown and fuel densification/swelling. The cladding diameter is ever decreasing with burnup until the cladding-to-pellet contact. The pellet diameter decreases up to about 8 MWd/kgU due to a dominant effect of fuel densification in a low burnup range but increases due to a dominant effect of swelling after 8 MWd/kgU [11-13]. However, the pellet diameter contraction due to densification is much faster than cladding creepdown. Therefore, there exists a maximum pellet-to-cladding gap creepdown at around 8 MWd/kgU, as shown in fig. 9. Moreover, the amount of fission gas release increases with the increase of fuel rod burnup and the higher fission gas release generates produces the higher fuel temperature. It is noted that the higher fuel temperature generates the larger cladding total tensile strain if the pellet-to-cladding gap is the same before the power transients. The combination of the pellet-to-cladding gap size variation and ever-increasing fuel temperature with burnup may explain why the lowest total tensile strain occurred at about 7 MWd/kgU. On the other hand, the reason that the optimized bottom-skewed axial profiles reduce total tensile strain is that the total energy contained in the fuel rod at the power level of 666 W/cm is lower for the optimized axial profile and consequently the amount of fission gas release is lower for the optimized axial profile, resulting in the lower fuel temperature, as explained above. Since the lower fuel temperature generates the less cladding total tensile strain if the pellet-to-cladding gap is the same before the power transients, the optimized bottom-skewed axial profiles generated lower total tensile strain than the current bottom-skewed ones, as shown in fig. 7.

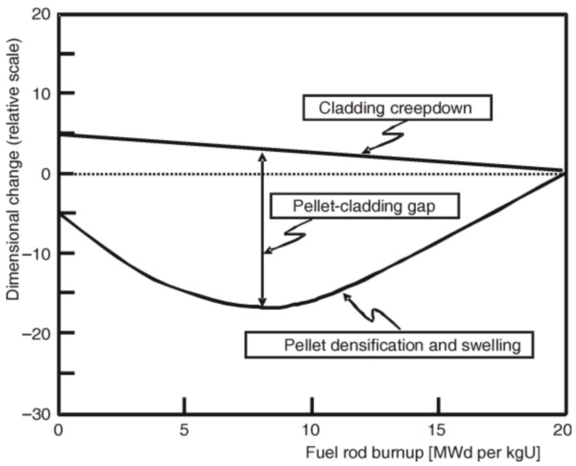

Figure 9. Schematic dimensional change *vs.* fuel rod burnup before power transients

CONCLUSIONS

With the use of the reactor operating condition that 118% overpower of a steady-state peak rod-averaged power will be exceeded during the ANS Condition II, the optimized bottom-skewed axial profiles are proposed to reduce over-conservatism contained in the current bottom-skewed axial profiles. The optimized bottom-skewed axial profiles reduced fuel centerline temperature and cladding total tensile strain by 26 °C and 0.02%, comparing with the current ones. The lower fuel temperatures and total tensile strains for the optimized axial profiles may be explained by lower fission gas release. In addition, the lowest total tensile strain at the fuel rod burnup of 7 MWd/MtU may be supported by the combination of the pellet-to-cladding gap variation and ever-increasing fuel temperature with the increase of the burnup. It is recommended that the current pellet microstructure be improved to reduce fission gas release, which will provides more margins for fuel centerline temperature and cladding total tensile strain.

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Кју-Tae КИМ

ЕФЕКАТ АКСИЈАЛНЕ РАСПОДЕЛЕ СНАГЕ У ГОРИВНОЈ ШИПКИ НА ТЕМПЕРАТУРУ ГОРИВА И НАПРЕЗАЊЕ КОШУЉИЦЕ

Унутрашњи притисак горивне шипке и оксидација кошуљице су најрестриктивнији пројектни критеријуми за нуклеарне електране у нормалним погонским условима, док су за нуклеарне реакторе у нестационарним условима то температура у средишњој оси горива и нестационарно укупно напрезање кошуљице на истезање. Међутим, пројектне маргине за температуру горива и нестационарно напрезање кошуљице на истезање постају још уже, будући да се подиже снага већине нуклеарних реактора, или се то планира ради побољшања економије нуклеарне енергије. У циљу да се обезбеде довољне пројектне маргине за температуру горива и укупно напрезање кошуљице на истезање и у случају пораста снаге, ради редукције претерано конзервативног решења оптимизован је постојећи аксијални профил снаге у горивној шипки који је коришћен у анализама транзијената, будући да током реакторских прелазних стања није долазило до прекорачења већег од 118% стационарног средњег пика снаге у горивној шипки. Поређење постојећег аксијалног профила снаге у горивној шипки и оног оптимизованог показује да се овим другим редукује температура горива у средишњој оси за 26 °C, а укупно напрезање кошуљице на истезање за 0,02%.

Кључне речи: нуклеарна горивна шипка, пројектни критеријуми, температура горива у средишњој оси, укупно напрезање кошуљице на истезање, аксијална расподела снаге у горивној шипки