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Investigation on the Allowable Transient Power Levels to Maintain the Mechanical Integrity of the 17x17 KOFA Fuel Rod During the ANS Conditions I and II

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ANS 과도조건 I 및 II에서 17x17 KOFA 핵연료봉의 기계적 건전성이 유지되는 과도상태 허용 출력준위에 관한 연구

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Abstract

Transient power level of the fuel rod is one of the key parameters for the transient fuel behavior. Through the analysis of the fuel performance data bases and sensitivity analyses of such parameters as rod power history, fast neutron flux, fuel enrichment and cycle length, which can affect the transient fuel behavior, a methodology generally applicable to find the allowable transient power level during the ANS Conditions I and II below which the mechanical integrity of the fuel rod is maintained was derived, and allowable transient power levels for the 17x17 KOFA fuel rod have been determined as a function of the burnup. With the introduction of this methodology, design analysis of the transient fuel behavior currently being calculated every cycle can be replaced by the simple check of the peak transient power level achievable during the cycle, and an operational flexibility of the reactor can be obtained by allowing higher transient power level up to 689.5 w/cm at low burnup range than current maximum allowable transient power level, 591 w/cm for the 17x17 KOFA fuel.

요 약

핵연료봉의 과도상태 출력준위는 핵연료봉의 과도상태 거동에서 가장 중요한 변수중의 하나이다. 핵연료 성능 데이타베이스의 분석과 핵연료의 과도상태 거동에 영향을 줄 수 있는 핵연료봉 출력이력, 속중성자속, 농축도 및 주기길이 등의 인자들의 민감도 분석을 통해서, ANS 과도조건 I 및 II에서 핵연료봉의 기계적 건전성이 유지되는 허용가능 과도상태 출력을 구하기 위해 일반적으로 적용이 가능한 방법론이 유도되었으며, 이를 통해 17x17 KOFA 핵연료봉의 허용가능 과도상태 출력이 연소도의 함수로써 결정되었다. 이 방법론을 도입함으로써, 현재와 같이 매 주기마다 핵연료봉 과도상태 설계분석을 수행할 필요가 없이 단지 해당주기에서의 과도상태 최대 출력준위 평가로써 핵연료봉의 과도상태 설계를

대체할 수 있으며, 17x17 KOFA 핵연료에 대해 낮은 연소도영역에서 기존의 최대 허용 과도상태 출력 준위인 591 w/cm보다 큰 최대 689.5 w/cm까지 허용함으로써 원자로 운전에 유연성을 줄 수 있다.

1. Introduction

The objective of the mechanical design analysis of the fuel rod is an assurance of the fuel rod mechanical integrity during the ANS Conditions I and II. The design analysis of the fuel rod is, therefore, performed every cycle to evaluate the performance of the fuel rods loaded in the reactor. The key parameters to evaluate the fuel performance can be largely divided into two categories. One is related to the steady state fuel performance such as the cladding corrosion, the creep strain of cladding due to the pellet swelling and a buildup of the fuel rod internal pressure due to the fission gas release into the fuel-cladding gap, which mainly depend upon the accumulated burnup and the temperature of the fuel during the lifetime, and the other is related to the transient fuel performance such as the fuel temperature increase and the transient cladding strain which can be caused by the overpower transients during the cyle. This study is on the transient fuel performance. The fuel design criteria to verify the fuel mechanical integrity during the transients are that centerline melting of the fuel rod is prevented since it can result in a sudden volume increase of the pellet and subsequent increase of the cladding strain leading to the fuel failure, and total tangential strain of the cladding is less than one percent. The limitation of the transient cladding strain is based upon the results of the ductility measurements of the irradiated cladding.

In this study, through the overall sensitivity analyses of such parameters that can affect fuel performance, a methodology generally applicable to find the allowable transient power level will be derived and the allowable transient power level below which the mechanical integrity of the fuel rod is maintained during the ANS Conditions I and II will be determined for the 17x17 KOFA fuel. For the fuel performance calculation, fuel manufacturing data, reactor thermal hydraulic data and fuel performance model constants are conservatively selected. Sensitivity analyses of such parameters as rod power history, fast neutron flux, fuel enrichment and cycle length will be performed to investigate those effect upon the transient fuel behavior, and the conservative values of those parameters will be used in the determination of generally applicable allowable transient power levels to maintain the mechanical integrity of the 17x17 KOFA fuel during the ANS Conditions I and II.

2. Calculation and Results

2.1. Parameters Affecting the Fuel Performance

Fuel mechanical design code, CARO-D[1] is used for the design analysis of the transient behavior of the KOFA fuel. Fuel performance model constants, fuel rod manufacturing data and reactor thermal hydraulic data are conservatively selected for the calculation of fuel centerline temperature and cladding strain, respectively [2, 3]. Tables 1 and 2 show fuel manufacturing data and reactor thermal hydraulic data of the 17x17 KOFA fuel.

For the parameters which may change cycle by cycle, the conservative values have been selected after a review of nuclear design data bases and subsequent evaluation of those effect upon the fuel performance. For the rod power history, the extremely high and extremely low power variations as shown in Figure 1 are considered in the calculation. Rod power histories represented as a radial power factor (F_{xy}) in Figure 1 can cover the whole spectrum of the rod power histories in the core. For the fast neutron flux, the conservative values are selected from the nuclear design data bases as shown in Table 3

Table 1. Fuel Manufacturing Data of the 17x17 KOFA Fuel

Parameter	Data
Pellet density (g/cm^3)	10.40±0.15
U−235 enrichment (w/o)	3.5−4.2
Pellet diameter (mm)	8.05±0.01
Pellet height (mm)	10±1
Cladding inside diameter (mm)	8.22±0.04
Cladding outside diameter (mm)	9.50±0.05
Active fuel length (mm)	3658±6

Table 2. Reactor Thermal Hydraulic Data of the 17x17 KOFA Fuel

Parameter	Nominal Condition	Thermal Design Flow
Inlet temperature (°C)	292.2	291.2
Outlet temperature (°C)	326.2	329.5
Coolant flow rate (kg/sec.rod)	0.3254	0.3021
Pressure (bar)	157.2	155.1
Core average heat rate(w/cm)	183.1	188.6

Table 3. Fast Neutron Flux of the 17x17 KOFA Fuel

Cycle	ϕ* (Fxy=1.0, E_n > 0.821 Mev) x10^{13} n/cm^2. sec	
	minimum	maximum
1	7.32	10.56
2	8.49	10.95
3	9.07	12.68

(*) ϕ (Fxy) = ϕ (Fxy=1.0) x Fxy

[4, 5]. The cycle length from 300 EFPD(Effective Full Power Day) to 500 EFPD and U-235 enrichment from 3.5 w/o to 4.2 w/o are considered in the calculation.

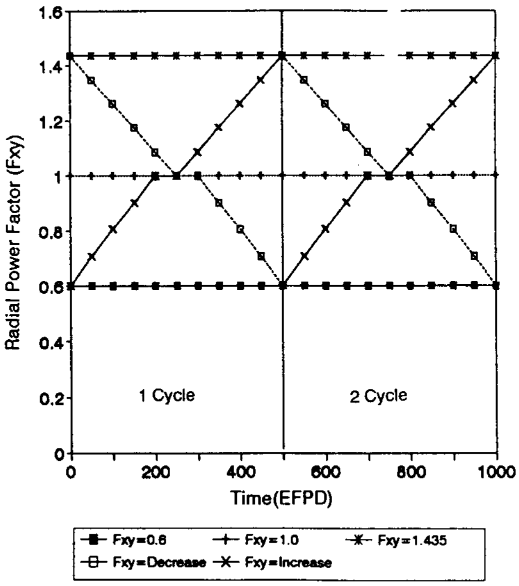

Fig. 1. Selected Rod Power Histories

2.2. Fuel Centerline Temperature

The design criterion of fuel temperature during the ANS Conditions I and II is that fuel centerline temperature be maintained below its melting temperature which tends to decrease linearly with fuel burnup [6].

Figure 2 shows the fuel centerline temperature changes as a function of burnup at the transient power level of 591 w/cm for the various power histories of the 17x17 KOFA fuel. The calculation procedure is such that the fuel rod is burned under the normal condition at the power level shown in Figure 1 and at a certain fixed burnup the power level is increased to the transient power level to simulate the transient condition.

The generic trend of fuel centerline temperature as a function of burnup can be described as follows. The main controlling parameter for fuel centerline temperature is the gap conductance which decreases with increase of both fuel-cladding gap width and fission gas release. As the burnup of the fuel rod

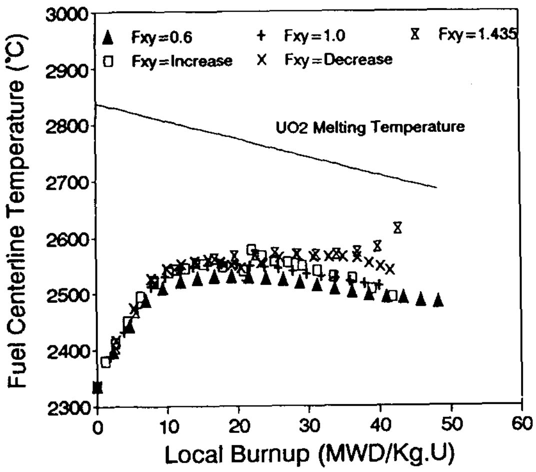

Fig. 2. Fuel Centerline Temperature at the Transient Power Level of 591 w/cm for Different Rod Power Histories

increases, the amount of the fission gas released into the gap increases, and the gap width increases initially due to the pellet densification and decreases later due to the pellet swelling and cladding creepdown. Therefore, the centerline temperature increases initially with the rod burnup as the gap conductance decreases, After it reaches its maximum at the burnup around 10 MWD/kgU, it tends to decrease or stay near constant depending on the power history which determines the variation of gap conductance with the burnup.

The sudden increasing trend for the power history (Fxy=1.435) after the burnup of about 40 MWD/kgU results from fuel temperature increase due to the localized acceleration of cladding oxidation leading to the violation of corrosion design limit(100μm) at the burnup of 41 MWD/kgU. Therefore, this hypothetical situation is prevented by the corrosion design limit which is one of the design criteria of the steady state fuel performance, and there is actually no possibility that the fuel rod maintains that high power level(Fxy=1.435)until the burnup of 40 MWD/kgU.

2.3. Total Tangential Strain of the Cladding

The design criterion of the cladding strain during ths ANS Conditions I and II is that the transient total tangential strain of the cladding be less than one percent[6]. The deformation of the cladding during the transients is caused by the thermal expansion of the fuel pellet at high transient power level.

Figure 3 shows the changes of the total tangential strain of the cladding as a function of burnup at the power level of 591 w/cm for the various rod power histories of the 17x17 KOFA fuel. The calculation procedure is the same as that of fuel centerline temperature explained in section 2.2.

The generic trend of total tangential strain of the cladding as a function of burnup can be described as follows. After initial decrease for short period due to the pellet densification, the cladding strain during the transient increases with the burnup due to the cladding creepdown until the gap closure. After the gap closure, the increase rate of the cladding strain with the burnup slows down due to the absence of the cladding creepdown. The reason for slightly increasing trend with burnup after gap closure is the increase of fission gas release which lowers the gap conductance and subsequently increases the thermal expansion of the fuel pellet and the strain of the cladding. After the gap closure under the normal condition before the transient, the transient cladding strain becomes proportional directly to the difference between the power level before the transient and that after the transient. That is, the fuel rod with lower normal power level before the transient gives the higher transient cladding strain as shown in Figure 3.

2.4. Effect of Fast Neutron Flux, the Rod Power History before the Transient, Fuel Enrichment and Cycle Length upon the Transient Fuel Behavior

The effect of fast neutron flux, the rod power history, before the transient, fuel enrichment and cycle

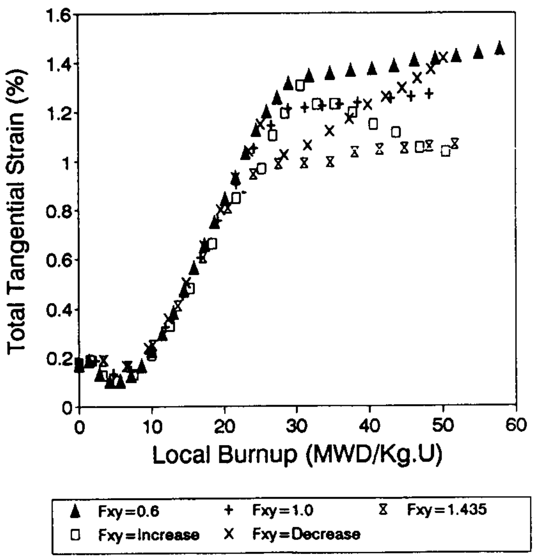

Fig. 3. Total Tangential Strain of the Cladding at the Transient Power Level of 591 w/cm for Different Rod Power Histories

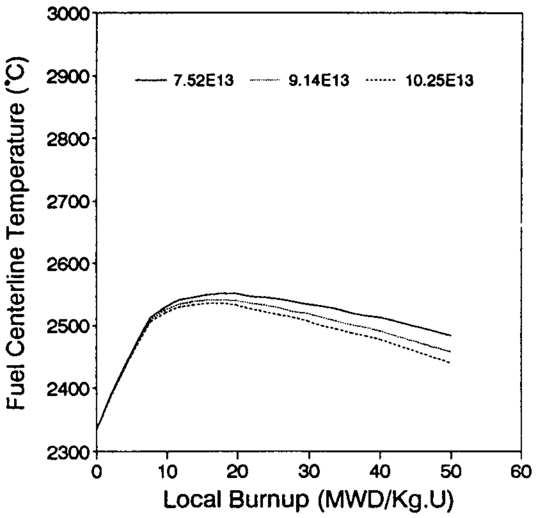

Fig. 4. Fuel Centerline Temperature at the Transient Power Level of 591 w/cm for Different Fast Neutron Fluxes(En ⟩ 0.821 Mev, n /cm^2.sec)

length upon the transient fuel behavior were analyzed. Figures 4 and 5 show the effect of fast neutron flux (E_n ⟩ 0.821 Mev) upon the fuel centerline temperature and total tangential strain of the cladding during the transients, respectively. As the fast neutron flux increases, the cladding creepdown increases and therefore, the fuel centerline temperature decreases and cladding strain increases. However, as shown in Figure 5, after the gap closure, there is no effect of fast neutron flux on the cladding strain since there exists no more creepdown of the cladding after gap contact. In summary, it can be seen from Figures 4 and 5 that fast neutron flux has non-negligible effect on the fuel transient behavior.

To see the influence of different rod power histories before the transient, the power change rates of ± 0.5% per EFPD in normal reactor operation condition have been selected as the limiting conditions after reviewing the nuclear design data bases[3, 4]. The results show that the relative difference of fuel centerline temperature between two limiting power change rates is less than 0.6% and the relative difference of total tangential strain of the cladding between them is less than 1.3%, Therefore, the effect of

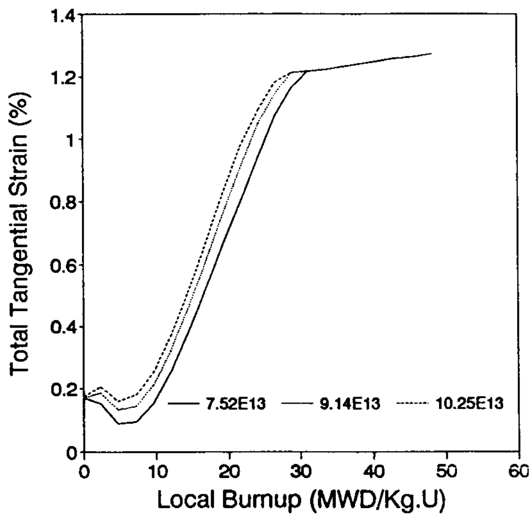

Fig. 5. Total Tangential Strain of the Cladding at the Transient Power Level of 591 w/cm for Different Fast Neutron Fluxes (En ⟩ 0.821 Mev, n /cm^2.sec)

the rod power history before the transient upon the fuel transient behavior is shown insignificant.

The influence of both cycle length(300 EFPD and 500 EFPD) and fuel enrichment (3.5 w/o and 4.2 w/o) on the fuel transient behavior has been analyzed and the results showed that both cycle length and fuel enrichment have a negligible effect upon the fuel centerline temperature and total tangential strain of the cladding during the transient, that is, relative difference of less than 1% both between short and long cycle lengths and between low and high enrichments, respectively.

3. Determination of Allowable Transient Power Levels

Based on the results given in section 2, a methodology generally applicable to determine allowable transient power levels below which the mechanical intergrity of the 17x17 KOFA fuel rod is maintained can be derived as follows.

It can be seen from Figures 2 and 3 that at low burnup less than 10 MWD/kgU, fuel centerline temperature is limiting and at high burnup total tangential strain of the cladding becomes limiting in the determination of allowable transient power level. Conservative reactor thermal hydraulic data, that is, thermal design flow data, fuel manufacturing data and fuel performance model constants are used in the calculation. For the fast neutron flux, conservatively high value for cladding strain calculation and low value for fuel temperature calculation have been selected from the nuclear design data bases as shown in Table 3[4, 5]. To cover the possible variations in the rod power history before the transient, fuel enrichment and reactor cycle length, safety margin of more than 5% which is sufficient enough to cover the effect of those variations upon the transient fuel behavior as discussed in section 2.4 was given to both the fuel centerline temperature and the total tangential strain of the cladding in the determination of allowable transient power level.

Figure 6 shows the allowable transient power level versus burnup for the different power levels under normal operation condition before the transient for the 17x17 KOFA fuel using highly cold-worked and partially recrystalized Zircaloy-4 as the cladding. As explained in section 2.3, the fuel rod with lower power level under normal operation condition before the transient gives lower allowable transient power level. Currently, the maximum allowable power level during the ANS Conditions I and II is limited by 591 w/cm for the 17x17 FOFA fuel. However, it can be seen from Figure 6 that the power level higher than 591 w/cm can be allowed until the burnup of 19 MWD/kgU in the area of fuel rod mechanical behavior. This can give an operational flexibility to the reactor using the 17x17 KOFA fuel.

4. Conclusion

Through the analysis of the fuel performance data bases and sensitivity analyses of the parameters which can affect the transient fuel behavior, a methodology generally applicable to find the allowable

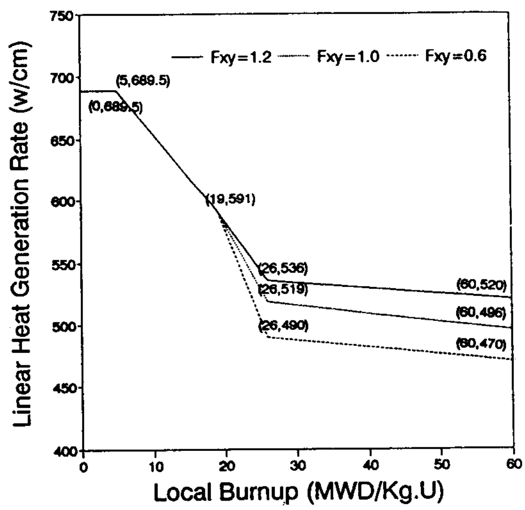

Fig. 6. Allowable Transient Power Levels of the 17x17 KOFA Fuel for Different Power Levels under Normal Condition before the Transient

transient power level during the ANS Conditions I and II below which the mechanical integrity of the fuel rod is maintained was derived, and allowable transient power levels for the 17x17 KOFA fuel rod have been determined. With the introduction of this methodology, design analysis of transient fuel behavior currently being calculated every cycle can be replaced by the simple check of the peak transient power level achievable during the cycle, and an operational flexibility of the reactor can be obtained by allowing higher transient power level at low burnup range than current maximum allowable transient power level.

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