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# Evaluation of Fuel Cladding Failures from the Fission Product Activities in the Reactor Coolant

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# 원자로 냉각수내의 핵분열생성물 방사능에 의한 핵연료피복관 파손 평가

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## Abstract

An efficient procedure of evaluating the fuel cladding failures occurring in the normal operations of typical PWR's has been investigated through the analysis of fission product(FP) activities in the reactor coolant using an analytical model, FIPREL code. Performed by this code is an extensive study on the sensivities of FP activities to such physical parameters as enrichment, burnup, and operation temperature of failed fuel rod as well as the effective failure size quantified in terms of the magnitude of gap release coefficient. The results of study are generally in agreement with those by PROFIP method. In the presence of tramp uranium the portion of activities released from failed rod is separated by an iterative calculation based on the activity ratios of fission nuclides chemically more stable than iodines. Obtained are the linear power density and the number of failed rods, the effective failure size, and the mass of tramp uranium. The operation experiences of 4 cycles of Kori Unit 1 are analyzed and the results show that the model is highly reliable for the survey and evaluation of fuel rod conditions during reactor operations.

## 요 약

FIPREL 전산코드를 사용하여 원자로 냉각수 내의 핵분열 생성물에 의한 방사능을 분석함으로써 PWR의 운전시에 발생하는 핵연료 피복관 파손을 평가할 수 있는 효과적인 절차를 모색하였다. 이 코드를 이용하여 핵연료의 농축도, 연소도, 가동온도 및 갭유출계수의 크기로 정량화되는 실제적 파손 크기등의 물리적 파라미터에 대해서 핵분열 생성물의 방사능이 나타내는 민감도에 대한 방대한 계산을 실시하였으며 그 결과는 PROFIP방법에 의한 것과 대체적으로 일치한다. 노출 우라늄이 존재하는 경우에는 옥소보다도 화학적으로 더 안정된 핵종간의 방사능비에 근거하여 반복계산을 실시함으로써 파손된 핵연료 봉에서 유출된 방사능만을 분리해 낸다. 개발된 전산코드로 파손 핵연료봉의 선형출력 밀도, 갯수, 실제적 파손 크기 및 노출우라늄의 질량등을 계산할 수 있다. 고리 1호기의 4주기에 걸친 운전 경험을 이 모델에 의해 분석한 결과에 의하면 본 모델은 원자력발전소 정상운전시 핵연료봉의 상태를 감시·평가하는데 아주 적합한 것으로 판명되었다.

## 1. Introduction

The improved techniques in fuel rod design and fabrication have significantly reduced the risk of rod failures caused by internal aggressions originating from manufacturing defects, pellet-cladding interactions and/or fatigues of cladding material. But the integrity of cladding can be always perilled by certain external aggressions of already identified effect like the baffle-jetting, or of yet unknown. So it is necessary to survey and evaluate the conditions of fuel rods during reactor operations to assess their integrity as well as to identify the anormalies of system causing rod failures.

The fission products, particularly the volatile species, escaping in large quantity from fuel are accumulated in the gap between fuel and cladding, and also leak to the coolant through defected areas to be dispersed in the system. The evaluation of cladding failures from fission product(FP) activities in the reactor coolant thus requires a good understanding of their behaviors not only in the fuel but also in the coolant system.

The release mechanisms from fuel to gap[1] are defined in rigorous mathematical equations while the gap-to-coolant release is not well understood physically. This is why the design methods of construction companies like Westinghouse[2] rely on the simple assumption that the FP concentration in the gap is in equilibrium with that in the fuel and then calculate the release fraction directly from the production rates multiplied by the escape rate coefficient ($\upsilon_i$) and the defect level. Because $\upsilon_i$ is assumed to be constant over the variations of dominant parameters, the claculated defect level is an effective one with respect to the importance of radiological consequences and does not represent the real conditions of fuel rod.

Another difficulty in the estimation of defect level arises from the presence of tramp uranium whose recoil sources contribute to the observed activities. Recently Aoki[3] has proposed a model based on the graphic display of $\upsilon_i$, defect level, and mass of tramp uranium as the function of iodine ratios(I-131/I-133, I-133/I-135). This model is actually an extension of the iodine correction method of Westinghouse.[2]

An improved model[4] capable of localizing the failed assembly and estimating the operation temperature, the effective failure size, and the number of failed rods has been proposed by CEA in France. This model is based on the results of sensitivity studies, by PROFIP code,[5] of activity ratios to the dominant parameters like burnups, fuel temperature, and effective failure size. Though some remarkable results have been reported, it seems that the precision of the model depends inevitably on the experiences of selecting certain key coefficients.

Domestically the knowledges in this field are not well established and there is no valuable calculational tool. So it has been decided to apply the improved model of CEA to Kori Unit 1 reactor to confirm its adequacy for the evaluation as well as to acquire the experiences needed. To this end a computer code called FIPREL has been developed by adopting the main features of PROFIP and also a new method is incorporated for the evaluation of tramp uranium source. In this method the activity ratios of more chemically stable nuclides than iodines are considered to determine the contribution of tramp uranium.

It is noted that the present study constitutes the first step to the final goals covering the establishment of more realistic operational limits for coolant activity and the quantification of safety margins to limiting radiological consequences of postulated accidents. Accordingly the objectives of this study are set up as follows;

1) Verification of FIPREL code against the experiences of commercial reactors

2) Better understanding of FP behaviors in the reactor system

3) Confirmation of the reliability of the prop osed model

## 2. Description and Verification of FIPREL Computer Code

### 2.1 Source Terms of Fission Products

The production rate of a nuclide by fission-depends on several parameters such as enrichment and burnups of fuel, neutron flux distribution, and geometry of fuel rod. Only the complicated neutronic codes can claculate the precise values of source terms. Accordingly for FIPREL code the results of LEOPARD-K[6] are least-square fitted to obtain quickly the mass of fissile atoms and their fission cross sections at a given step without losing the accuracy.

### 2.2 Modeling of Fission Product Releases and Behaviors in the Coolant System.

Under normal operating conditions the release of FPs from fuel is assumed to be governed by 3 modes, namely, recoil, ejection, and temperature-activated migration (diffusion);

$$\text{Recoil} \quad \cdots F_i^R = \frac{R_i^R}{B_i} = \frac{1}{4}\,\frac{\rho S}{V}$$

$$\text{Ejection} \quad \cdots F_i^E = \frac{R_i^E}{B_i} = \frac{vFN_i}{B_i}$$

$$= \frac{S}{V}\,\frac{vF\rho}{(vF+\lambda_i)}\left(1-e^{-(vF+\lambda_i)t}\right)$$

$$\text{Diffusion} \cdots F_i^D = \frac{R_i^D}{B_i}$$

$$= f_D \frac{\nu_i}{\nu_i+\lambda_i}\left(1-\bar{e}^{(\nu_i+\lambda_i)t}\right)$$

Where $F_i^x$=Release Fraction of Nuclide $i$ by One of the 3 Modes

$R_i^x$=Release Rate of Nuclide $i$ by One of the 3 Modes (#/sec)

$B_i$=Production Rate of Nuclide $i$ (#/sec)

$\rho$=Mean Recoil Length of a Fission Fragment (cm)

$V$=Fuel Volume ($cm^3$)

$v$=Volume of Fuel Ejected by a Fission Fragment ($cm^3$)

$F$=Fission Rate (#/sec)

$N_i$=Concentration of Nuclide $i$ (#/$cm^3$)

$\lambda_i$=Decay Constant of Nuclide $i$

$f_D$=Diffusion Fraction of Nuclide $i$

$\nu_i$=Release Coefficient ($sec^{-1}$)

$=\nu_0 e^{-E/RT}$

$\nu_0$=Proportionality Constant ($sec^{-1}$)

$E$=Activation Energy (cal/mole)

$R$=Gas Constant (cal/mole °K)

The fission nuclides released from fuel decay to other nuclides, and/or are deposited on the inner surface of cladding, and/or leak to the coolant through the defected area. The leakage rate is simply assmed to be proportional to the concentration in the gap;

$$R_i = \nu_g N_i^g$$

where $R_i$=Release Rate (#/$cm^3$ sec)

$\nu_g$=Gap Release Coefficient ($sec^{-1}$)

$N_i^g$=Concentration of Nuclide $i$ in the Gap (#/$cm^3$)

The mass balance equation of nuclide $i$ in the gap is written as follows;

$$\frac{dNi}{dt} = R_i^f - [\nu_g + (\lambda_i + \sigma_i\phi)(1+\alpha_i\frac{S}{V})]N_i^g$$

where $N_i = N_i^g + N_i^d$

$N_i^d$=Concentration of Nuclide $i$ Deposited on the Inner Surface of Cladding (#/$cm^3$)

$=\alpha_i \frac{S}{V} N_i^g$

$\alpha_i$=Deposit Coefficient (cm)

$S$=Surface Area (cm$^2$)

$V$=Gap Volume (cm$^3$)

$\phi$=Neutron Flux in the Gap (#/cm$^2$sec)

$\sigma_i$=Absorption Cross Section of Nuclide $i$ (cm$^2$)

$$R_i^f = R_i^R + R_i^E + R_i^D$$

The radioactive FPs contribute to the activity measured and are purified as the coolant circulates. The purification system and related functions modelled in FIPREL code include demineralizers, partition of gaseous species into gas and liquid phase in the pressuriser and the volume control tank of CVCS, leakage of primary coolant, and boron dilution accompanying makeup water injection. The mass balance equation in the coolant system is expressed as follows;

$$\frac{dN_i^c}{dt} = \nu_g N_i^g - [\lambda_i + P_i + \sigma_i \phi] N_i^c$$

where $N_i^c$=Concentration of Nuclide $i$ in the Coolant (#/cm$^3$)

$$P_i = \frac{L_{\text{eff}}}{M} \beta_i$$

$L_{\text{eff}}$=Effective Letdown Flow Rate Contributing to Purification (g/sec)

$M$=Total Coolant Mass (g)

$\beta_i$=Purification Coefficient of Nuclide $i$

$\phi$=Neutron Flux in the Coolant (#/cm$^2$ sec)

### 2.3 Verification of FIPREL Code

The general capability of the code is verified against the experiences of Fessenheim 1, a PWR of 900 MWe in France, in which during cycle 2 two fuel rods with defected cladding were known to exist. The calculation is carried out with gap release coefficients equal to $10^{-4}$ $s^{-1}$ and $10^{-6}$ $s^{-1}$; these values correspond to the 2 small failure sizes defined in reference 4 and in

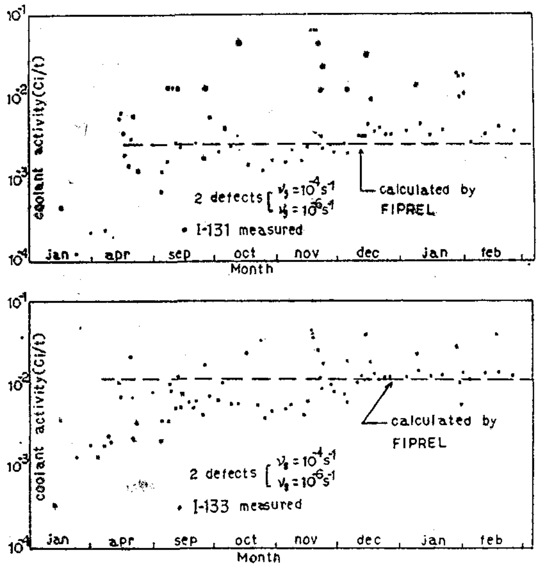


**Fig. 1. Coolant Activity Calculated vs. Measured for Cycle 2 of Fessenheim 1**

section III. 2. Figure 1 shows the results of calculation for $^{131}$I and $^{133}$I activities in the coolant along with the measured values. The good agreement between the calculated and the average of measured values assures the acceptability for subsequent analysis of Kori Unit 1, which in turn will demonstrate further qualification of the code.

## 3. Evaluation of Fuel Cladding Failures

### 3.1 Localization of Failed Rod

The assembly with failed rods is characterized by its proper nuclear design parameters such as the enrichment, the burnups, and the linear power density. The sensivity studies by FIPREL code reveal that the activity ratio of $^{134}$Cs to $^{137}$Cs in the gap has a strong dependence on the burnups of fuel and is affected little by the variation of linear power density. This fact is in correspondance with the result published in reference 4. Figure 2 shows the cesium ratio calculated for varying enrichments and burnups of typical 14×14 fuels.

The cesium isotopes are usually not observed

**Table 1. Burnups of Defected Fuel Assembly Predected versus Observed and Failure States**

| Cycle | 2 | | | | 3 | | 4 | |
|---|---|---|---|---|---|---|---|---|
| Date | 12/08 1980 | 26/08 1980 | 16/12 1980 | 30/01 1981 | 09/06 1981 | 17/04 1982 | 12/02 1983 | 07/04 1983 |
| $\frac{Cs-134}{Cs-137}$ | 0.349 | 0.433 | 0.559 | 0.559 | 0.848 | 1.03 | 0.711 | 0.567 |
| $\left(\frac{MWD}{MTU}\right)^p$ | 4,300 | 5,100 | 6,400 | 6,400 | 9,500 | 11,200 | 8,000 | 6,500 |
| $\left(\frac{MWD}{MTU}\right)^o$ | 4,700 | 5,150 | 6,100 | 6,800 | 10,500 | 10,500 | 7,500 | 6.600 |
| Failure States | Guill. Break | Guill. Break | Severe Flatt. | Severe Flatt. | Grid Break | guill. Break | Guill. Break | PCI |

Note A. $\left(\frac{MWD}{MTU}\right)^p$=Burnups of Failed Assembly Predicted from Cesium Ratio Observed

B. $\left(\frac{MWD}{MTU}\right)^o$=Assembly Average Burnups Observed to be the Nearest to That Predicted at the Given Date

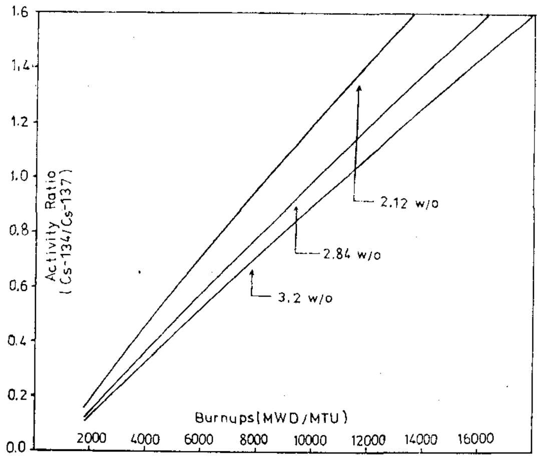


**Fig. 2. Activity Ratio of $^{134}Cs$ to $^{137}Cs$ Calculated for Increasing Burnups in Typical 14×14 Fuels.**

in normal conditions while a large spiking release occurs in transients like reactor trip or rapid power level change. So it is logically thought that the cesium ratio observed at the moment of reactor trip subsequent to steady operations of long period is of the same value that has existed in the gap just before the transient. The localization of failed rod is thus accomplished by comparing the observed cesium ratio with those calculated. This argument is verified against the experiences of Kori Unit 1, which is summarized in Table 1. Taking measurement uncertainties into account the correspondance between the design and the calculated burnups of failed assembly is well acceptable.

### 3.2 Effective Failure Size and Defect Level

Since the FIPREL code does not represent the real physical process occurring in the FP release through the defected area of cladding, the failure size must be expressed effectively with respect to the quantity of FP activity released. As noted in section II. 2 the leakage of FPs from gap to coolant is determined by the magnitude of gap release coefficient ($\nu_g$). It is thus this value that quantifies the effective failure size.

The systematic sensitivity study of activity ratios to this parameter has been carried out using FIPREL. It is found out that the Kr-87 to Xe-135 and the Kr-85m to Kr-87 ratio are most appropriate for the determination of effective failure size because they are strongly dependent on this parameter but little affected by the other ones like burnups or operation temperature of fuel. It is also observed that the effective failure size corresponding to a certain activity ratio belongs to one of the characteristic groups represented by $10^{-6}\ s^{-1}$, $10^{-4}\ s^{-1}$, and $10^{-2}\ s^{-1}$ in growing magnitude. This implies that all the

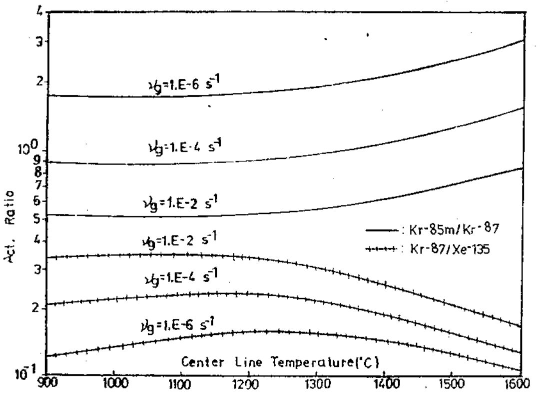


**Fig. 3. Activity Ratios of Krypton for Varying Fuel Temperature and Magnitude of Failure Size in Typical 14×14 Rods.**

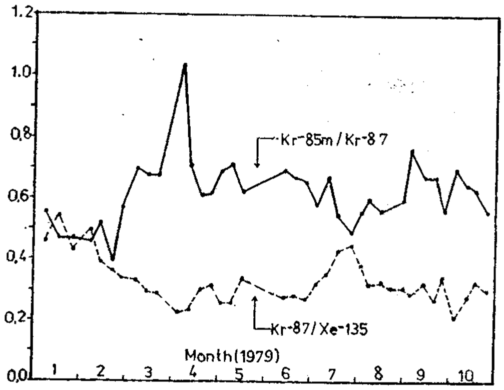


**Fig. 4. Activity Ratios of Krypton Measured for Cycle 1 Kori Unit 1.**

types of failures under steady-state conditions can be divided into 3 characteristic groups, turning out to be in accordance with Reference 4. Figure 3 shows the activity ratios of above-mentioned nuclides for the 3 values of $\nu_g$ and varying fuel temperatures in typical 14×14 fuels. The evaluation of failure size is thus accomplished by comparing the observed ratio with this figure. Shown in Figure 4 is the evolution of activity ratios during cycle 1 at Kori Unit 1. The corresponding failure size is of $10^{-2}$ $s^{-1}$, the largest among the 3 groups. In fact an assembly with 2 guillotine-breaked fuel rods was found by the end-of-cycle fuel examination.

Once the effective failure size is determined in such a way the operation temperature of

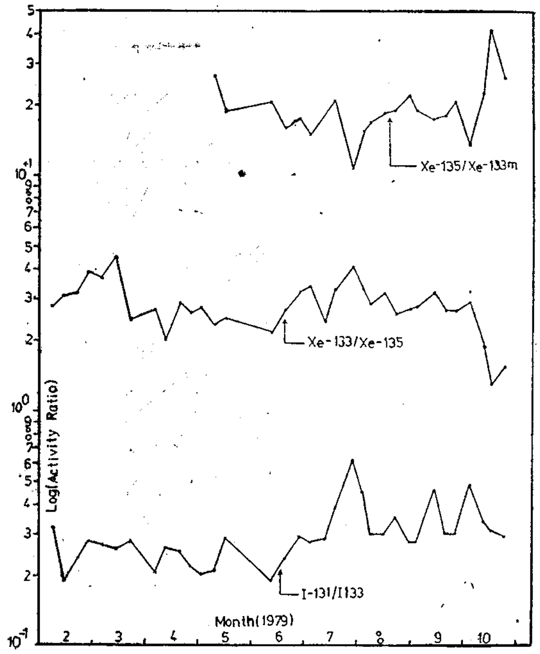


**Fig. 5. Activity Ratios of Xenon and Iodine Measured for Cycle 1 Kori Unit 1.**

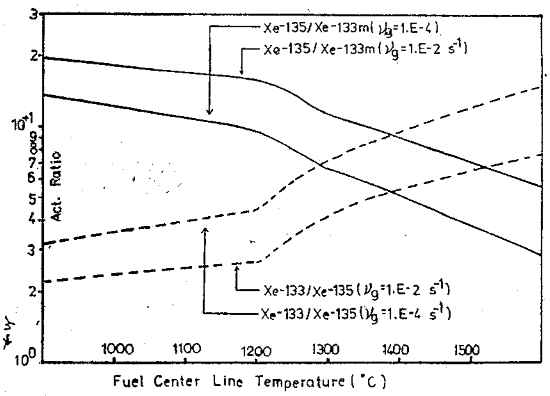


**Fig. 6. Activity Ratios of Xenon Versus Fuel Center Line Temperature for Typical 14×14 Fuels.**

failed rod is discovered by considering the activity ratios of nuclides showing strong dependence on it. Instead of using only one ratio, namely, Xe-133/Xe-135 as in Reference 4, it is decided to introduce the Xe-135/Xe-133m according to the results of sensitivity study by FIPREL. This brings the advantageous effect of lessening the uncertainty caused by the fluctuation of observed activities. Shown in Figure

**Table 2. Evaluations of Failed Fuel Rods for Cycle 1 at Kori Unit 1** (Failure Size $(\nu_g)=10^{-2}s^{-1}$, $Tc=1,150°C$)

Defect Level=0. 03%

| Date | $I^M$-131 | $I^C$-131 | $N$ | $I^M$-133 | $I^C$-133 | $N$ | $Xe^M$-133 | $Xe^C$-133 | $N$ | $Xe^M$-135 | $Xe^C$-135 | $N$ | $Kr^M$-85m | $Kr^C$-85m | $N$ |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 19/03 1979 | 1. 41 $E-2$ | 1. 80 $E-3$ | 7. 8 | 5. 60 $E-2$ | 6. 48 $E-3$ | 8. 6 | 3. 37 $E-1$ | 3. 87 $E-2$ | 8. 7 | 1. 16 $E-1$ | 1. 44 $E-2$ | 8. 1 | 2. 15 $E-2$ | 2. 57 $E-3$ | 8. 4 |
| 26/03 1979 | 1. 27 $E-2$ | 〃 | 7. 1 | 5. 84 $E-2$ | 〃 | 9. 0 | 2. 44 $E-1$ | 〃 | 6. 3 | 9. 36 $E-2$ | 〃 | 6. 5 | 1. 83 $E-2$ | 〃 | 7. 1 |
| 02/04 1979 | 1. 22 $E-2$ | 〃 | 6. 8 | 6. 05 $E-2$ | 〃 | 9. 3 | 3. 94 $E-1$ | 〃 | 10. 2 | 1. 42 $E-1$ | 〃 | 9. 9 | 2. 50 $E-2$ | 〃 | 9. 7 |
| 09/04 1979 | 1. 43 $E-2$ | 〃 | 7. 9 | 6. 71 $E-2$ | 〃 | 10. 4 | 3. 50 $E-1$ | 〃 | 9. 0 | 1. 50 $E-1$ | 〃 | 10. 4 | 2. 79 $E-2$ | 〃 | 10. 9 |
| $\bar{N}\pm\sigma$ | | | 7. 4±0. 5 | | | 9. 3±0. 7 | | | 8. 6±1. 4 | | | 8. 7±1. 5 | | | 9. 0±1. 4 |

Note $A^M$=Measured Activity ($\mu Ci$/cc), $A^C$=Activity Calculated for Failed Rod($\mu Ci$/cc)
$N$=Number of Failed Rods Calculated

5 is the evolution of these ratios during cycle 1 at Kori Unit 1. This is compared with the calculated ones in Figure 6 to find that the center-line temperature of failed rod is less than 1, 200°C. To this temperature correspond a linear power density of about 180w/cm and a relative power level of about 0. 8. This low power density or level can be most probably found in the peripheral fuel assemblies,namely, in zone 3 for cycle 1 at Kori Unit 1. In fact all of the assemblies with failed fuel rods detected by the end-of-cycle test belong to the outer zone.

Finally the number of failed rods is decided by dividing the observed activity by the calculated for one failed rod with pre-determined failure size and operation temperature. The so-called defect level is then calculated based on the number and the linear power density of the failed rod. Presented in Table 2 is the result of calculation for cycle 1 of Kori Unit 1. The number of failed rods is of 9 and the defect level is of 0. 03%, a low level compared with the realistic 0. 12% adopted by ANS 18. 1. [7] The number of failed assembly discharged during reload is of 10 but the exact number of failed rods is not known. Nevertheless the calculated number seems to be a reasonable value taking the number of discharged assemblies into account.

### 3.3 Determination of Tramp Uranium Source

The loss of $UO_2$ pellets through the severely damaged cladding is accompanied by an augmentation of coolant activity. The so-called tramp uranium means the kind of fuel whose fission fragments can reach the coolant directly by their recoil energies, thus making the activity a composite one;

$$A_i^M=A_i^D+A_i^R$$

Where $A_i^M$=Activity Observed of Isotope $i$

$A_i^D$=Activity of Isotope $i$ Released from Failed Fuel Rods

$A_i^R$=Activity of Isotope $i$ Recoiled from Tramp Uranium

To find $A_i^D$ another equation for isotope $j$ of same nuclide is set up and the activity ratios between these 2 isotopes are used to eliminate the unknowns:

$$R^D=A_i^D/A_j^D$$

$$R^R=A_i^R/A_j^R$$

then $$A_j^D=\frac{(A_i^M-A_j^M\cdot R^D)}{(R^R-R^D)}$$

The $R^R$ is easily calculated by FIPREL because

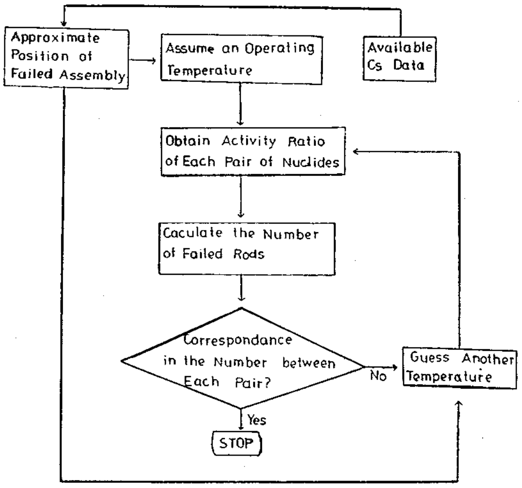


**Fig. 7. Methodology Adopted for the Evaluation of Rod Failures in the Presence of Tramp Uraniums.**

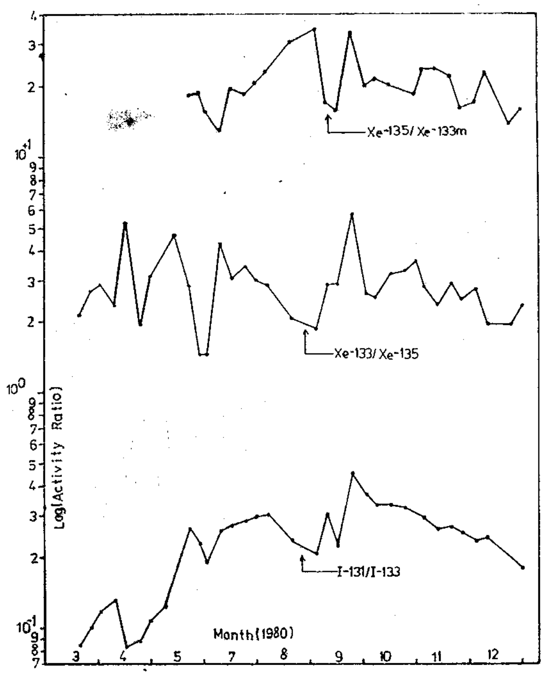


**Fig. 8. Activity Ratios of Xenon and Iodine Measured for Cycle 2 Kori Unit 1.**

it is not affected by the variation of tramp uranium mass. To calculate the $R^D$ the important parameters such as enrichment, burnups, and failure size as well as operation temperature of fuel must be known. But the activity ratios of volatile species are so insensitive to the variation of enrichiment and burnups of failed rod that their effects can be neglected. The failure size can be determined by comparing the observed activity ratio with $R^R$ and $R^D$

**Table 3. Evaluations of Failed Fuel Rods for Cycle 2 at Kori Unit 1.**
(Failure Size $(\nu_g)=10^{-2}\ s-1$, $Tc=1370°C$)

Defect Level=0.03%

| Date | $Xe^M$-133 | $Xe^M$-135 | $Xe^D$-135 | $Xe^C$-135 | $N$ | $Xe^M$-133m | $Xe^D$-133m | $Xe^C$-133m | $N$ |
|---|---|---|---|---|---|---|---|---|---|
| 10/09 1980 | 1.16 E+0 | 4.03 E−1 | 1.57 E−1 | 2.70 E−2 | 5.8 | 2.37 E−2 | 1.72 E−2 | 2.74 E−3 | 6.2 |
| 17/09 1980 | 1.33 E+0 | 4.59 E−1 | 1.80 E−1 | // | 6.7 | 2.91 E−2 | 2.23 E−2 | // | 8.1 |
| 02/10 1980 | 1.26 E+0 | 4.79 E−1 | 1.55 E−1 | // | 5.7 | 2.37 E−2 | 1.42 E−2 | // | 5.2 |
| 15/10 1980 | 1.87 E+0 | 5.81 E−1 | 2.78 E−1 | // | 10.3 | 2.87 E−2 | 1.71 E−2 | // | 6.3 |
| 05/11 1980 | 1.46 E+0 | 5.16 E−1 | 1.94 E−1 | // | 7.2 | 2.18 E−2 | 1.02 E−2 | // | 3.7 |
| 19/11 1980 | 1.73 E+0 | 5.89 E−1 | 2.38 E−1 | // | 8.8 | 2.65 E−2 | 1.38 E−2 | // | 5.0 |
| 05/12 1980 | 1.59 E+0 | 5.75 E−1 | 2.07 E−1 | // | 7.7 | 3.39 E−2 | 2.44 E−2 | // | 8.9 |
| 10/12 1980 | 1.29 E+0 | 6.59 E−1 | 9.67 E−1 | // | 3.6 | 2.90 E−2 | 1.46 E−2 | // | 5.3 |
| $\bar{N}\pm\sigma$ | | | | | 7.0±1.9 | | | | 6.1±1.6 |

Note $A^M$=Measured Activity ($\mu Ci$/cc), $A^C$=Activity Calculated for 1 Failed Rod ($\mu Ci$/cc), $A^D$=Activity Released from Defected Fuel Rod ($\mu Ci$/cc), $N$=Number of Failed Rods

**Table 4. Evaluations of Failed Fuel Rods for Cycle 3 at Kori Unit 1**
(Failure Size $(\nu_g)=10^{-2}s-1$, $Tc=1370°C$)

Defect Level=0.13%

| Date | $I^M$-131 | $I^M$-133 | $I^D$-133 | $I^C$-133 | $N$ | $Xe^M$-135 | $Xe^M$-133m | $Xe^D$-133m | $Xe^C$-133m | $N$ |
|---|---|---|---|---|---|---|---|---|---|---|
| 03/12 1981 | 2.26 $E-1$ | 1.19 $E+0$ | 3.24 $E-1$ | 1.51 $E-2$ | 21.5 | 1.47 $E+0$ | 8.40 $E-2$ | 5.89 $E-2$ | 2.74 $E-3$ | 21.5 |
| 07/01 1982 | 2.67 $E-1$ | 1.26 $E+0$ | 4.27 $E-1$ | 〃 | 28.3 | 1.28 $E+0$ | 9.81 $E-2$ | 8.53 $E-2$ | 〃 | 31.1 |
| 21/01 1982 | 2.50 $E-1$ | 1.27 $E+0$ | 3.73 $E-1$ | 〃 | 24.7 | 1.33 $E+0$ | 9.44 $E-2$ | 7.79 $E-2$ | 〃 | 28.4 |
| 11/02 1982 | 2.81 $E-1$ | 1.34 $E+0$ | 4.46 $E-1$ | 〃 | 29.5 | 1.41 $E+0$ | 9.61 $E-2$ | 7.76 $E-2$ | 〃 | 28.3 |
| 25/02 1982 | 2.73 $E-1$ | 1.37 $E+0$ | 4.12 $E-1$ | 〃 | 27.3 | 1.65 $E+0$ | 1.06 $E-1$ | 8.21 $E-2$ | 〃 | 29.9 |
| 05/03 1982 | 2.72 $E-1$ | 1.24 $E+0$ | 4.49 $E-1$ | 〃 | 29.7 | 1.50 $E+0$ | 1.13 $E-1$ | 9.73 $E-2$ | 〃 | 35.5 |
| 18/03 1982 | 2.62 $E-1$ | 1.31 $E+0$ | 3.97 $E-1$ | 〃 | 26.3 | 1.45 $E+0$ | 9.75 $E-2$ | 7.80 $E-2$ | 〃 | 28.5 |
| 08/04 1982 | 2.50 $E-1$ | 1.35 $E+0$ | 3.49 $E-1$ | 〃 | 23.1 | 1.26 $E+0$ | 7.96 $E-2$ | 6.08 $E-2$ | 〃 | 22.2 |
| $\bar{N}\pm\sigma$ | | | | | 26.3±2.8 | | | | | 28.2±4.3 |

Note $A^M$=Measured Activity ($\mu Ci$/cc), $A^D$=Activity Released from Defected Fuel Rods ($\mu Ci$/cc), $A^C$=Activity Calculated for 1 Defected Fuel Rod ($Ci$/cc), $N$=Number of Failed Rods

**Table 5. Evaluations of Failed Fuel Rods for Cycle 4 at Kori Unit 1**
(Failure Size $(\nu_g)=10^{-2}s^{-1}$, $Tc=1370°C$)

Defect Level=0.03%

| Date | $Xe^M$-133 | $Xe^M$-135 | $Xe^D$-135 | $Xe^C$-135 | $N$ | $Xe^M$-133m | $Xe^D$-133m | $Xe^C$-133m | $N$ |
|---|---|---|---|---|---|---|---|---|---|
| 11/11 1982 | 1.61 $E+0$ | 8.32 $E-1$ | 1.17 $E-1$ | 2.70 $E-2$ | 4.3 | 4.00 $E-2$ | 2.30 $E-2$ | 2.74 $E-3$ | 8.4 |
| 26/11 1982 | 1.53 $E+0$ | 7.46 $E-1$ | 1.28 $E-1$ | 〃 | 4.7 | 3.68 $E-2$ | 2.19 $E-2$ | 〃 | 8.0 |
| 15/12 1982 | 2.26 $E+0$ | 9.69 $E-1$ | 2.38 $E-1$ | 〃 | 8.8 | 3.50 $E-2$ | 1.10 $E-2$ | 〃 | 4.0 |
| 30/12 1982 | 1.76 $E+0$ | 8.37 $E-1$ | 1.55 $E-1$ | 〃 | 5.8 | 3.26 $E-2$ | 1.28 $E-2$ | 〃 | 4.7 |
| 06/01 1982 | 2.47 $E+0$ | 1.08 $E+0$ | 2.50 $E-1$ | 〃 | 9.3 | 3.80 $E-2$ | 1.09 $E-2$ | 〃 | 4.0 |
| 20/01 1983 | 2.09 $E+0$ | 9.97 $E-1$ | 1.83 $E-1$ | 〃 | 6.8 | 3.60 $E-2$ | 1.13 $E-2$ | 〃 | 4.1 |
| 10/02 1983 | 1.26 $E+0$ | 6.66 $E-1$ | 8.62 $E-2$ | 〃 | 3.2 | 2.60 $E-2$ | 1.02 $E-2$ | 〃 | 3.7 |
| 17/03 1983 | 1.66 $E+0$ | 8.02 $E-1$ | 1.41 $E-1$ | 〃 | 5.2 | 3.90 $E-2$ | 2.28 $E-2$ | 〃 | 8.3 |
| $\bar{N}\pm\sigma$ | | | | | 6.0±2.0 | | | | 5.7±2.0 |

Note $A^M$=Measured Activity ($\mu Ci$/cc), $A^D$=Activity Released from Defected Fuel Rods($\mu Ci$/cc), $A^C$=Activity Calculated for 1 Failed Rod ($\mu Ci$/cc), $N$=Number of Failed Rods

calculated for 3 representative failure sizes (see section 3.2).

Since the operation temperature is left still unknown, additional one or two pairs of nuclides are combined to calculate the number of failed rods in an iterative manner as illustrated in figure 7. Usually one or two iterations are sufficient to get the corresponding number between each pair of nuclides if the approximate position of the assembly with failed rods is already known based on available cesium ratio data. Actually 3 pairs of nuclides including Xe-133/Xe-135, Xe-135/Xe-133m, and I-131/I-133 are utilized. This algorithm is more reliable than Aoki's method because it takes into account explicitly the effect of fuel temperature on the activity ratio and also is primarily based on those species chemically more stable than iodines.

The experiences of 4 cycles of Kori Unit 1 are analysed with this method. Shown in figure 8 is the evolution of activity ratios observed during cycle 2. The analysis reveals that the number of failed rods is between 6 and 7 as explained in Table 3 and the mass of tramp uranium is of 100g. The effective failure size is of $10^{-2}$ $s^{-1}$, the largest size among the 3 representatives, and the linear power density corresponding to the estimated fuel temperature is equal to about 220w/cm. The end-of-cycle examination finds 6 assemblies damaged not to be reused and that the one with most severely damaged rods have operated at average linear power density ranging from 218w/cm to 220w/cm. Refer to Tables 4 and 5 for cycles 3 and 4, respectively.

## 4. Conclusion

The general capability of FIPREL, and analytical model developed in this study, has been verified against the experiences of commercial reactors with failed fuel rods whose conditions are well known. An improved method for the separation of tramp uranium source from the measured activity is proposed to calculate the correct defect level. It has been found that the FIPREL code along with the separation method becomes a very efficient tool for the evaluation of fuel rod failures when combined with the following procedures:

1) The cesium ratio observed at reactor trip is a representative parameter of burnup characteristics of failed fuel rod. So it is used to localize the assembly with failed rods.

2) The two activity ratios such as Kr-87/Xe-135 and Kr-85m/Kr-87 are sensitive only to the variation of effective failure size. So they are used to determine the failure size.

3) The xenon ratios such as Xe-133/Xe-135 and Xe-135/Xe-133m are very sensitive to the variation of fuel temperature. So they are used to determine the operation temperature or the linear power density of failed rod.

4) In the presence of tramp uranium the operation temperature of failed fuel rod can not be known by the above method. So an iterative calculation for this variable is done based on multiple pairs of chemically stable nuclides to determine the number of failed rod.

The results produced by these procedures are in good agreement with the operation experiences of Kori Unit 1. So it is concluded that the present model is highly acceptable for the survey and evaluation of fuel rod conditions during normal operations of typical PWRs. But further studies, especially for the transient behavior of FPs, are still required and the results of the present study make the beginning to attain the final goals including the establishment of more persuasive operational limits for coolant activity and the quantification of safety margins to limiting radiological consequences of design-basis accidents.

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