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Methodology for Estimating the Number of Failed Fuel Rods in Operating PWRs Using Diffusion and Kinetic Models

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Abstract

A methodology for estimating the number of failed fuel rods based on the primary coolant activity in operating PWRs has been developed. This method deals with both the diffusion and the kinetic models. In case of small or medium cladding failures, the diffusion model which can consider different sizes of failure is used, whereas for large cladding failures the kinetic model is used. From the kinetic model, the release-to-birth rate ratio (R/B) is represented as a linear function of the number of failed fuel rods. This has been done by expressing the escape rate coefficient in terms of the slope of log(R/B) versus log λ. *The present method has been applied to the cases of 26 cycles of several nuclear power plants for which ultrasonic testings were performed. The results show that the present method gives better predictions than the existing computer codes such as IODYNE and CADE.*

1. Introduction

When a reactor fuel rod is failed, a leak path exists so that coolant can enter into the fuel rod gap and fission products can escape into the primary coolant system. Analysis of the primary coolant activity can provide important information on the conditions of the core, particularly such as the number of failed fuel rods, the cladding failure size and the location, and the contribution of tramp uranium to the coolant activity. A lot of computer codes or methods have been developed [1-5] to estimate the number of failed fuel rods based on the primary coolant activity. These existing methods that adopted the kinetic model can be classified into two categories of 'total activity method' and 'nonlinear regression method'.

The total activity method [1,2] calculates the number of failed fuel rods by comparing the measured total primary coolant activity with the predicted activity released from a single failed fuel rod into the primary coolant. But this method is difficult to consider the contribution of tramp uranium to the coolant activity which is of importance in the analysis of the coolant activity. The non-linear regression method [3-5], on the other hand, can consider the contribution of tramp uranium to the coolant activity, but this method has inherent pitfalls of non-linear equation. Therefore, solutions by this method are largely dependent on the regression method and the initial value.

These problems of the kinetic model can be overcome if the release-to-birth rate ratio *(R/B)* is represented as a linear function of the number of failed fuel rods. In the present study, this has been achieved by determining the escape rate coefficient from the slope of log*(R/B)* versus log λ. In addition, based on experimental data obtained from a reactor with a single failed fuel rod, Lewis et al. [6] proposed that diffusion process can be a dominant transport mechanism in the gap in case of small or medium size of failures. Considering transport mechanism in the gap as diffusion process enables to handle the situation when there are different sizes of cladding failure in the reactor core. In the present work, therefore, both the diffusion and the kinetic models are used, depending on the predicted cladding failure size and the number of iodine isotopes of which activity is measured. The cladding failure size is predicted from the slope of linear-relationship between log*(R/B)* and log λ [6]. The present method has been applied to the cases of 26 cycles of several nuclear power plants for which ultrasonic testings were carried out.

2. Analysis

2.1 Diffusion Model

In the case of small or medium cladding failure, it can be assumed that the iodine compounds bound to the internal surfaces after reaction with the wet steam become mobile. Therefore, diffusion is considered to be the dominant mechanism for the transport of iodine in the gap [6]. For the physical situation as shown in Fig. 1, transport of the iodine isotope i in the gap is governed by the following steady-state diffusion equation:

$$D_{ij}\frac{d^2C^g_{ij}}{dz^2} - \lambda_i C^g_{ij} - \lambda_i \alpha_{ij}\left(\frac{S}{V}\right)C^g_{ij} + R^f_i = 0 \tag{1}$$

with two boundary conditions,

$$C^g_{ij} = C^0_{ij} \qquad (z=0)\ , \tag{2}$$

$$\frac{dC^g_{ij}}{dz} = 0 \qquad \left(z = \pm\frac{L_f}{2}\right)\ , \tag{3}$$

where D is the diffusion constant, C is the concentration, α_{ij} is the iodine absorption fraction in the gap, (S/V) is the surface-to-volume ratio of the gap, R is the release rate, L_f is the length of the fuel rod, the subscript j is the failure size parameter, and the superscripts g and f denote the gap and fuel pellet, respectively. When the failure size is small or medium, an analogous expression to Eq. (1) holds for the diffusion in the leak channel. Because there is no source term in the leak channel, Eq. (1) becomes

$$D_{ij}\frac{d^2C^l_{ij}}{dx^2} - \lambda_i C^l_{ij} = 0 \tag{4}$$

with two boundary conditions,

$$C^l_{ij} = C^0_{ij} \qquad (x = 0)\ , \tag{5}$$

$$C^l_{ij} = 0 \qquad (x = t_{eff})\ , \tag{6}$$

where t_{eff} is the effective thickness of the cladding and superscript l denotes the leak channel. In these boundary conditions, C^0_{ij} is still unknown but can be determined by the following condition (Fig. 1):

$$R_{ij} = R^f_{ij} + 2R^g_{ij} \tag{7}$$

where R_{ij} is the total release rate into the leak channel, R^f_{ij} is the direct release rate from the fuel to the leak channel, and R^g_{ij} is the release rate from the gap to the leak channel.

Solving Eqs. (1) and (4) and using Eq. (7), Chun and Kim [7] obtained the following expression:

$$(R/B)^c_{ij} = f_{ij}\,(R/B)^f_i \tag{8}$$

where f_{ij} is the attenuation coefficient which is a function of the failure size, decay constant, and the diffusion coefficient in the gap. Assuming k different sizes of the failures (j=1,2,3,...,k) and using Eq. (8), the release fraction from all failed fuel rods is represented by

$$(R/B)^c_i = \sum_{j=1}^{j=k} X_j (R/B)^c_{ij} + C = (R/B)^f_i \sum_{j=1}^{j=k} X_j f_{ij} + C \tag{9}$$

where X_j is the number of failed fuel rods with failure size j and C is the contribution of tramp uranium to the coolant activity. Again, assuming that the dominant release mechanism of fission products from the fuel pellet to the gap is diffusion process, $(R/B)^f_i$ in Eq. (9) can be obtained from the booth's diffusion model.

$$(R/B)^f_i \approx (R/B)^f_{i,\,Diffusion} = 3\left(\frac{D'}{\lambda_i}\right)^{0.5} H_i \tag{10}$$

where D' is the empirical diffusion coefficient in the UO_2 fuel pellet and H_i is the factor to account for the effect of precursors on the diffusional release of fission products in the fuel pellet.

Therefore, using Eq. (9), X_j can be evaluated from the measured coolant activities if C is determined.

2.2 Kinetic Model

2.2.1 Balance Equations

A schematic diagram of the three-region kinetic model is shown in Fig. 2. The release phenomena of fission products from a failed fuel rod are governed by the following balance equations for a iodine isotope i in the fuel pellet, in the gap, and in the primary coolant, respectively:

$$\frac{dN^f_i}{dt} = \dot{F} y_i - (\nu_i + \lambda_i)\, N^f_i \tag{11}$$

$$\frac{dN^g_i}{dt} = \nu_i\, N^f_i - (\varepsilon_i + \lambda_i)\, N^g_i \tag{12}$$

$$\frac{dN^c_i}{dt} = \varepsilon_i\, N^g_i - (\lambda_i + \beta)\, N^c_i \tag{13}$$

where N is the number of atoms at any instance, $\dot{F}$ is the fission rate per rod, y_i is the fission yield, ν_i is the escape rate coefficient from the fuel pellet to the gap, ε_i is the escape rate coefficient from the gap to the primary coolant, and β is the purification rate. As can be seen in the existing kinetic models [3-5], the following familiar *(R/B)* expression under the steady-state condition can be obtained from the above equations:

$$(R/B)_i = \frac{XG\varepsilon_i}{\lambda_i^{1.5} + \lambda_i^{0.5}\varepsilon_i} \times H_i + C \tag{14}$$

where X is the total number of failed fuel rods and G is the empirical constant.

The left hand side of the above equation can be determined from the measured coolant activity as follows:

$$(R/B)_i = \frac{(\lambda_i+\beta)}{\lambda_i} \times A_i \times \frac{V_c}{Fy_i} \tag{15}$$

where A_i is the measured coolant activity and V_c is the coolant volume. Because it is impossible to directly evaluate X using Eq. (14), the non-linear regression method has been used in the existing codes. However, if the escape rate coefficient can be determined, Eq. (14) is linearized as follows:

$$(R/B)_i = L_i X + C \tag{16}$$

where

$$L_i = \frac{G\varepsilon_i}{\lambda_i^{1.5}+\lambda_i^{0.5}\varepsilon_i} \times H_i \tag{17}$$

Therefore, the number of failed fuel rods and the contribution of tramp uranium to the coolant activity can be obtained by linear regression method using Eqs. (15) and (16) when the measured coolant activities of at least two iodine isotopes are available.

2.2.2. Determination of the Escape Rate Coefficient

The slope of linear-relationship between log*(R/B)* and log λ can be approximately determined from the measured coolant activities of two isotopes of i and j, typically I^{131} and I^{133}, as follows:

$$n = \frac{\log(R/B)_i - \log(R/B)_j}{\log\lambda_i - \log\lambda_j} \tag{18}$$

After substituting Eq. (15) into Eq. (18), Eq. (18) can be rearranged in terms of (A_i/A_j).

$$\frac{A_i}{A_j} = \frac{\lambda_i y_i(\lambda_j+\beta)}{\lambda_j y_j(\lambda_i+\beta)} \times \left(\frac{\lambda_i}{\lambda_j}\right)^n \tag{19}$$

From the steady-state solution of Eqs. (11), (12), and (13), (A_i/A_j) can also be obtained as follows:

$$\frac{A_i}{A_j} = \frac{\lambda_i y_i \nu_i \varepsilon_i(\lambda_j+\nu_j)(\lambda_j+\varepsilon_j)(\lambda_j+\beta)}{\lambda_j y_j \nu_j \varepsilon_j(\lambda_i+\nu_i)(\lambda_i+\varepsilon_i)(\lambda_i+\beta)} \tag{20}$$

Here, the following assumptions are made:

- $\nu_i \ll \lambda_i$ (21)
- Diffusion is the dominant release mechanism of fission products from the fuel pellet to the gap.
- $\frac{\varepsilon_i}{\varepsilon_j} = \left(\frac{\lambda_i}{\lambda_j}\right)^{-m}$ (22)

Equating Eq. (19) to (20) and using the above three assumptions, the escape rate coefficient as a function of n is obtained as follows:

$$\varepsilon_i = \frac{\lambda_j\left(\frac{\lambda_i}{\lambda_j}\right)^{-m} - \lambda_i\left(\frac{\lambda_i}{\lambda_j}\right)^{(n-m+1)}}{\left(\frac{\lambda_i}{\lambda_j}\right)^{(n-m+1)} - 1} \tag{23}$$

3. Results and Discussion

To evaluate the number of failed fuel rods, either diffusion model or kinetic model should be selected in the present method. This is performed based on the number of iodine isotopes of which activity is measured and the predicted cladding failure size which is deduced from the slope of linear-relationship between $\log(R/B)$ and $\log \lambda$. Diffusion model requires at least three measured iodine activities (typically, I^{131}, I^{133} and either I^{134} or I^{135}), but kinetic model can be applied with two measured iodine activities (typically, I^{131} and I^{133}). In addition, diffusion model is selected in case of small or medium cladding failure while kinetic model is used for large cladding failure.

The present method has been applied to the cases of 26 cycles, including 21 cycles of foreign PWRs and 5 cycles of domestic PWRs, for which ultrasonic testings were performed at EOC. As shown in Fig. 3, the results show that the present method gives better predictions than existing computer codes such as IODYNE developed by ABB-CE [2] and CADE developed by Westinghouse [4]. IODYNE seems to underpredict and CADE seems to overpredict for the cases of 26 cycles collected in the present study.

The present model should be extended to the reactor transient condition, such as iodine spiking phenomena in particular, since the present method needs the stable activity data during steady-state reactor operation. It is also necessary to build up a more reliable database to improve the present method.

Acknowledgement

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References

1. Beraha, R et al., "Fuel Survey in the Light Water Reactors Based on the Activity of the Fission Products," *Nucl. Technol.*, **49**, 426 (1980).
2. "PWR Fuel Follow from Coolant Activity Analysis, Training Manual," CE-NPSD-308-P (1991).
3. Beyer, C.E. et al., "Methodology Estimating Number of Failed Fuel Rods and Defect Size," EPRI-NP-6554, (1989).
4. Burman, D.L. and Benzvi, M. S., "CADE-Coolant Activity Data Evaluation Personal Computer Program, User's Manual," Westinghouse (1988).
5. Lewis, B.J., Green, R.J., and Che, C.W.T., "A Prototype Expert System for the Monitoring of Defected Nuclear Fuel Elements in Canada Deuterium Uranium Reactors," *Nucl. Technol.*, **98**, 307 (1992).
6. Lewis, B.J., "Fundamental Aspects of Defective Nuclear Fuel Behaviour and Fission Product Release," *J. Nucl. Mater.*, **160**, 201 (1988).
7. Chun, M.H. and Kim, Y.S., "Attenuation Effect in the Gap and Leak Channel for Fission Products Release," *Trans. Am. Nucl. Soc.*, **68**, 304 (1993).

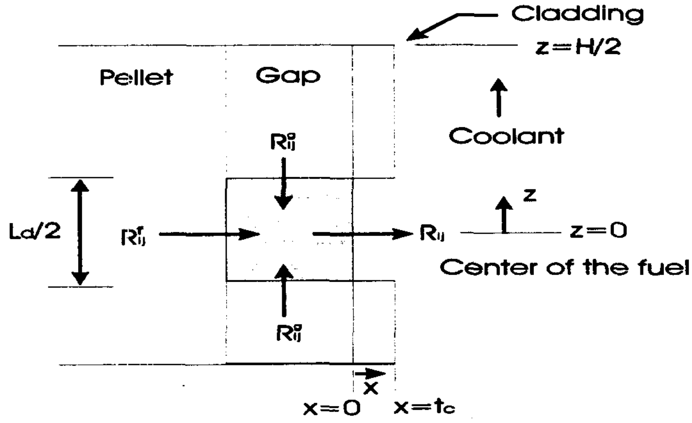

Fig. 1 Physical Model for the Failed Fuel Rods

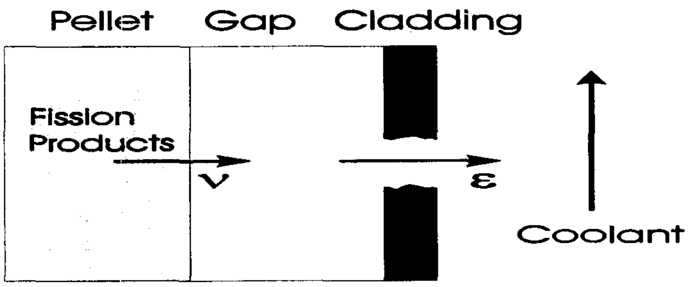

Fig. 2 Leak Path of the Fission Products (Three-Region Kinetic Model)

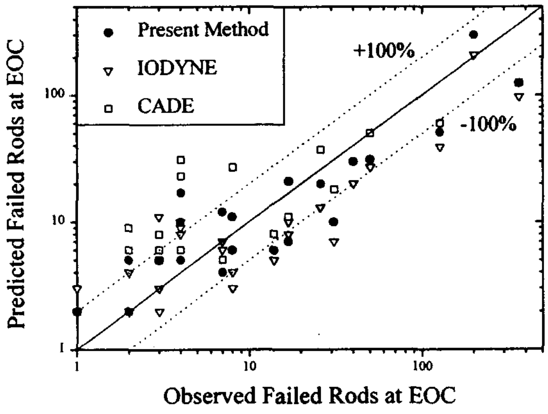

Fig. 3 Predicted Number of Failed Fuel Rods