Technical Paper

https://doi.org/10.7733/jnfcwt.2021.19.2.215

Development of the Defect Analysis Technology for CANDU Spent Fuel

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(Received December 28, 2020 / Revised January 19, 2021 / Approved March 8, 2021)

The domestic CANDU nuclear power plants have been operated for a long time and various unforeseen spent fuel defects have been discovered. As the spent fuel defects are important factors in the safety of the nuclear power plant, a study on the analysis of the spent fuel defects to prevent their recurrence is necessary. However, in cases where the fuel rods inside the fuel assembly are defected, it is difficult to dismantle the fuel assembly owing to their welded structure and the facility conditions of the plant. Therefore, it is impossible to analyze the spent fuel defect because it is difficult to visually check the shape of the fuel defect. To resolve these problems, an analysis technology that can predict the number of defected fuel rods and defect size was developed. In this study, we developed a methodology for investigating the root cause of spent fuel defects using a database of the earlier fuel defects in the plants. It is anticipated that in the future this analysis technology will be applied when spent fuel defects occur.

Keywords: CANDU, Nuclear power plant, Spent fuel defect, Defect cause, Nuclide

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1. Introduction

Since commercial operation of a CANDU nuclear power plant started in April 1983, four CANDU nuclear power plants have been operated. As these domestic CANDU plants have been operated for a long time, various types of spent fuel defects have occurred in them [1]. The physical properties of the pellet, which is uranium dioxide (UO_2) in the fuel rod, must be free of small holes, cracks, and chipping defects in the process of manufacturing nuclear fuel [2-3].

In addition, since the fuel rod must withstand high heat and radiation in the reactor and maintain its integrity in water of 5 m depth in the spent fuel pool facility for decades, the fuel rods made of Zircaloy material are also inspected by helium leak testing in the manufacturing process. However, during the fuel burning in the reactor, many small defects occur in the welding area of the endcap in the fuel rod. The spent fuel released from the reactor can be stored in a canister designed with a capacity of 60 fuel assemblies if it is determined to be a sound spent fuel without defects by the defect detection system in the power plant. It is then transferred to a dry storage facility [4]. However, in the case of defective spent fuel, it is sealed in a special container (Can) after determining the fuel is defective through sipping inspection system and a visual inspection system [5-6]. Defective fuels are stored separately in the defective spent fuel facility. When it is difficult to check defects through a sipping test and visual inspection, a precise inspection for investigating the root cause of the defect type and the defect location is performed in a hot cell [7].

In order to secure the safety of the CANDU nuclear power plant, the integrity of the spent fuel must be guaranteed. However, complete monitoring of the fuel being burned in the reactor has many restrictions due to high radiation conditions. To date, in order to monitor fuel loaded in the reactor, a method of analyzing the coolant radioactivity of the reactor has been widely used. In Korea, the analysis technology of the coolant radioactivity was previously at a relatively low level where only indication of a fuel defect was identified and it was unable to investigate the root cause of the defect. In particular, in the case of CANDU spent fuel, it is difficult to perform fuel dismantling due to the welded structure of the fuel assembly, as shown in Fig. 1 [8].

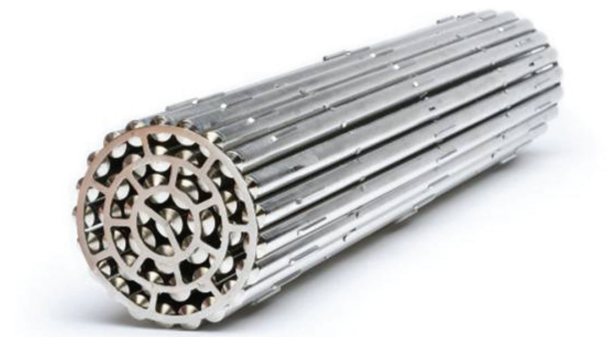

Fig. 1. The structure of the CANDU fuel assembly.

If the fuel rod located inside the fuel assembly is defective, it is impossible to identify the fuel defect. In response, in this study, a methodology for analyzing the number of defective fuel rods and the size of defects and the root cause of defects is described.

2. Development of analysis technology for fuel defect

2.1 Fission product of nuclear fuel

When nuclear fission occurs in the uranium pellet of the fuel rod, fission products are generated. Fission products that are directly generated through fission are called primary fission products, and since these primary fission products are unstable, they can be converted into another nuclide through various decay mechanisms.

The instability of the primary fission product is stabilized by conversion to other nuclides with the release of beta particles (β-particles) due to excessive neutron numbers and is generally accompanied by gamma ray (γ-ray) emission simultaneously.

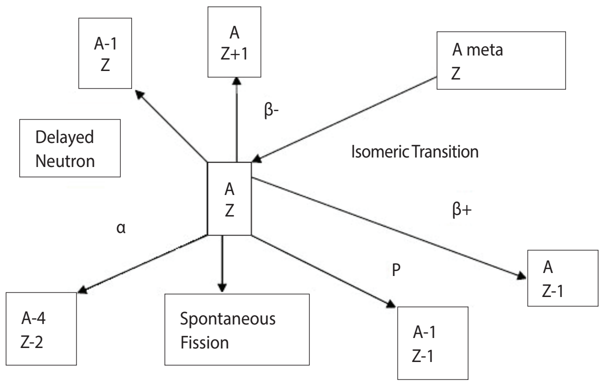

Fig. 2. The concept of decay reaction for fission product (A : Mass number, Z : Atomic number).

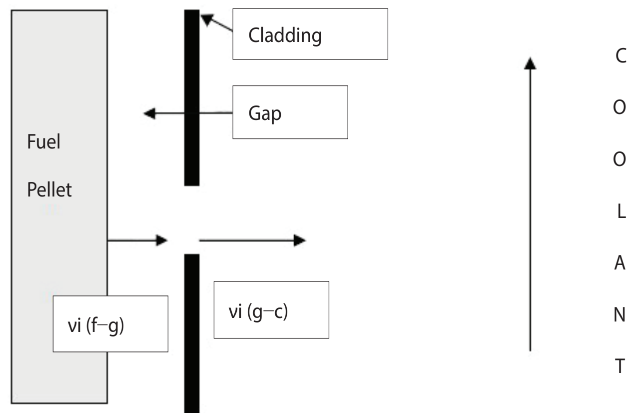

Fig. 3. Schematic diagram of the kinetic model.

Secondary fission products are not generated directly through nuclear fission and come from the neutron capture of the stable fission product. The nuclides generated in this way increase in proportion to the square of the burn-up of the fuel. The method of calculating the amount of nuclides generated in spent fuel is evaluated by the Bateman equation, given as Eq. (1), through various nuclide transformation processes based on the decay concept illustrated in Fig. 2.

$$\frac{dNi}{dt} = + \sum_{j \neq i}[\lambda_{ij}^{d} + \overline{\sigma_{ij}}\,\overline{\text{Ø}}]Nj - \sum_{j \neq i}[\lambda_{ji}^{d} + \overline{\sigma_{ji}}\,\overline{\text{Ø}}]Ni \quad (1)$$

where Ni is the number density of nuclide i, Nj is the number density of the nuclide j that can contribute to the formation of the nuclide i by fission, and λ_{ij}^{d} is the decay constant from nuclide i to nuclide j by beta decay or isomeric transition. $\overline{\text{Ø}}$ is the average neutron flux and $\overline{\sigma_{ij}}$ is the average cross section for conversion from nuclide i to the nuclide j [9].

2.2 Diffusion of the fission product

The fission products generated in the uranium pellets move from the pellets to the gap, which is a space between the pellets and the Zircaloy cladding. In the central region of the pellet, which has temperature above about 1,400°C, the fission products are moved by diffusion. On the other hand, in the outer region of the pellet, where the temperature is below 1,000°C, fission products are moved by recoil and knock-on.

2.2.1 Modeling of the fission product release

In order to evaluate the amount of fission products in the gap of the fuel rod, mass balance equations are used and the mass balance equations can be established in terms of a diffusion model or a kinetic model. If spent fuel is defective, the mass balance equations shown in equation (2) and (3) are used to calculate the transport of fission products from the gap to the coolant according to the kinetic model of Fig. 3.

$$\frac{dN_g(i,t)}{dt} = v_i^{f-g} N_i^f - \lambda_i N_g(i,t) - v_i^{g-c} N_g(i,t) \quad (2)$$

$$\frac{dN_f(i,t)}{dt} = \dot{F} Y_i - (v_i^{f-g} + \lambda_i) N_i^f \quad (3)$$

where $N_g\,(i,t)$ is the number of atoms of the isotopes i in the gap at instant t and $N_f\,(i,t)$ is the number of atoms of the isotopes i in the pellet at instant t.

v_i^{f-g} is the experimental escape rate coefficient from the pellet to the gap, N_i^f is number of atoms of the isotopes i in the pellet, λ_i is decay constant for the isotopes i and v_i^{g-c} is the experimental escape rate coefficient from the gap to the

Table 1. Characteristics of each defective fuel

Defective fuel	A	B	C	D
Defect indication (days after the fuel is loaded)	231	172	197	78
Defect shape	hole shape in the circumference of fuel rod	Hydriding Blister and swelling	Crack on the welded endcap	Separated gap of the endcap
Defect size (mm^2)	98	31	18	52
Burn-up (MWHr/kgU)	158	91	135	49

coolant, $\dot{F}$ is average fission rate in the pellet, Y_i is cumulative fission yield of the isotopes i [10].

2.3 Nuclides for evaluation of the defective fuel

In the event of leakage in the cladding of the fuel rod due to the fuel defect, nuclides such as krypton, xenon, and cesium that leak outside the fuel rod are detected in the coolant system [11].

These gas nuclides such as xenon and krypton exist without reacting with other nuclides in the gap because they are chemically stable. Under normal conditions, nuclides of inert gas are easily released through the defective area. In the case of iodine nuclides, they exist in a gaseous state in the central region of the pellet but when iodine nuclides are released to the gap, which is relatively cooler than the pellet, a small portion of iodine is absorbed in the surface of the pellet in the form of CsI and most of the iodine is absorbed in the surface of the cladding of the Zircaloy material [12]. If the temperature of the pellet surface decreases below the saturated temperature of the water due to reduction of the reactor power, the soluble iodine and cesium nuclides are dissolved by the coolant which flows into the gap of the fuel rod. The soluble iodine and cesium nuclides are leaked to the coolant from the gap of the fuel rod. Because of this phenomenon, the peak of the radioactivity for the iodine and cesium occurs in the coolant by direct release and recoil release.

The presence of a small defect in the cladding of the fuel rod increases the ratio of ^{131}I to ^{133}I. This is because ^{131}I has a longer half-life than ^{133}I.

2.4 Simulation of the fission products behavior

2.4.1 ORIGEN overview

After the fuel assembly is loaded into the reactor core, it is burned in the core for a certain period of time until the fuel is released from the core. At this time, nuclides are generated in the process of the nuclear reaction. ORIGEN code is a program for calculating the decay and generation of these nuclides [13].

Input data are the burn-up, power distribution, decay time, core loading period, etc. The results of the ORIGEN code calculation are the inventory of each nuclide in the fuel rod. If the fuel is defective, the nuclides in the fuel rod are leaked outside the fuel rod and the radioactivity of the leaked nuclide is then measured in the coolant by the detection system. Therefore, if the measured value of leaked nuclides and the inventory of nuclides in the fuel rod calculated by ORIGEN code are known, the defect size can then be predicted even if it is difficult to see the defect, which is located in the internal rod of the fuel assembly.

2.4.2 Characteristics for each defective fuel

During the operation of the nuclear power plant, four fuels that were actually failed were sampled and the defect

Table 2. The verification results for prediction of the number of defective fuel rod

Rod power ($kW \cdot m^{-1}$)	Pred. ^{131}I (uCi/kg)	Meas. ^{131}I (uCi/kg)	Pred. number of defective rod	Actual number of defective rod
48	51.1	49.2	1	1

characteristics of each fuel were analyzed. Table 1 shows the defect characteristics such as the defect indication, defect shape, defect size, and burn-up occurred after loading the fuels in reactor.

2.4.3 Analysis of leakage rate for each defective fuel

The method of measuring the nuclides leaked through the defect hole of the fuel rod was to measure the activity of the nuclei such as iodine, xenon, kryton leaked from the defective fuel using NaI(TI) scintillation detector with circulating coolant including gas after inserting the defective fuel into a sealed canister of a sipping inspection system.

Analysis comparing the inventory of the nuclei in the defective fuel rod calculated by ORIGEN and the nuclei leaked into the coolant through the defect hole of the fuel rod shows that the leakage rate of ^{135}Xe is in the range of 2.21×10^{-04} to 8.56×10^{-02} while ^{133}Xe shows slightly higher leakage rates of 3.61×10^{-04} to 1.68×10^{-01}. This is because ^{133}Xe has a longer half-life than ^{135}Xe. The leakage rate was calculated as shown in equation (4).

$$\text{Leakage rate} = \frac{\text{coolant activity for nuclide i (Ci)}}{\text{inventory for nuclide i in a fuel rod (Ci)}} \tag{4}$$

The leakage rate of ^{88}Kr, a gaseous nuclide, is in a range from 9.20×10^{-05} to 3.40×10^{-02}, which is smaller than the leakage rate of ^{133}Xe and 135Xe. It is also analyzed that as the half-life of ^{88}Kr is much shorter than those of ^{133}Xe and ^{135}Xe, many nuclides of ^{88}Kr disappear in the process of diffusion from fuel rods to coolant. ^{131}I, which is a soluble nuclide, has a leakage rate of 1.86×10^{-03} to 1.49×10^{-02}, which indicates a relatively smaller deviation than that of a

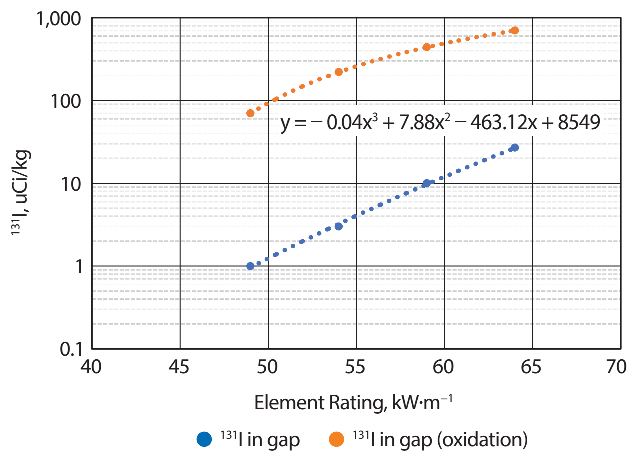

Fig. 4. Activity ^{131}I in gap vs rod linear power.

gaseous nuclide. The ^{131}I nuclide and ^{133}Xe /^{131}I are closely related to the defect size of the fuel rod [14].

Therefore, if the value calculated by ORIGEN and the coolant activity in the event of a fuel defect are known, the defect size can be predicted.

3. Results and discussion

3.1 Prediction for the number of defective fuel rod

It is important to recognize the number of defective fuel rods in the process of the fuel burning in the reactor. As the power of the fuel rod increases, ^{131}I in the fuel rod also increases. Fig. 4 shows the maximum radioactivity of ^{131}I (red line) that can be leaked per one fuel rod according to the oxidation of pellets and the fuel rod power. If the measured activity of ^{131}I (Meas. ^{131}I) in the coolant is within the

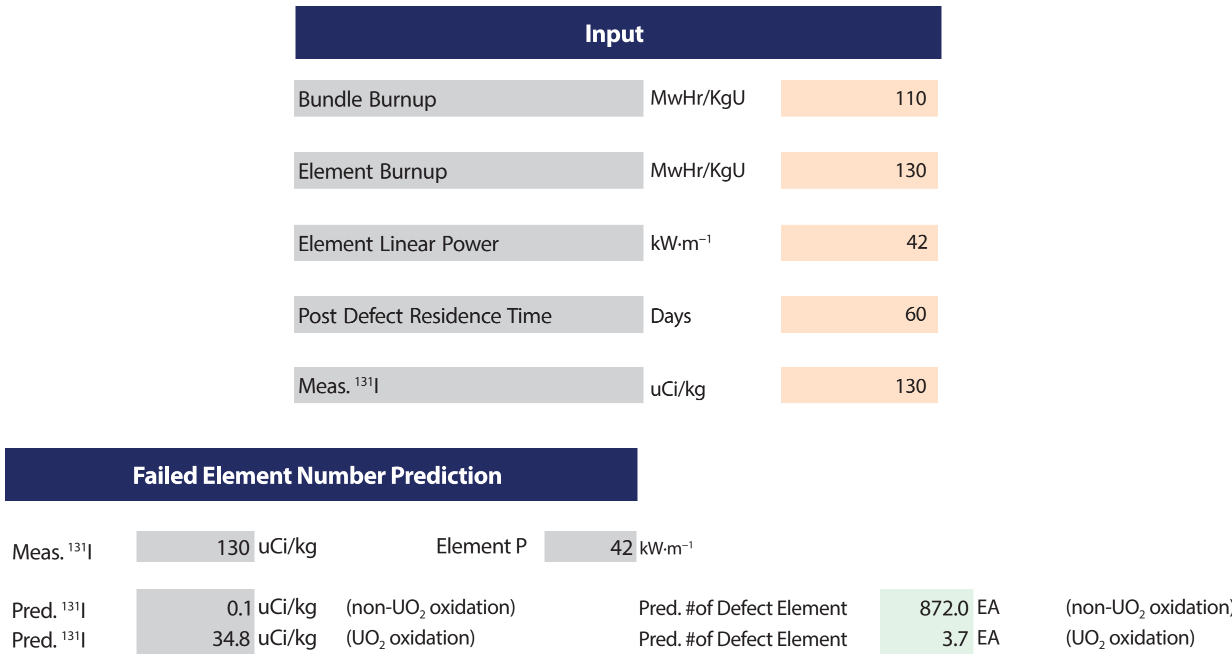

Fig. 5. Program for estimating the number of defective rods.

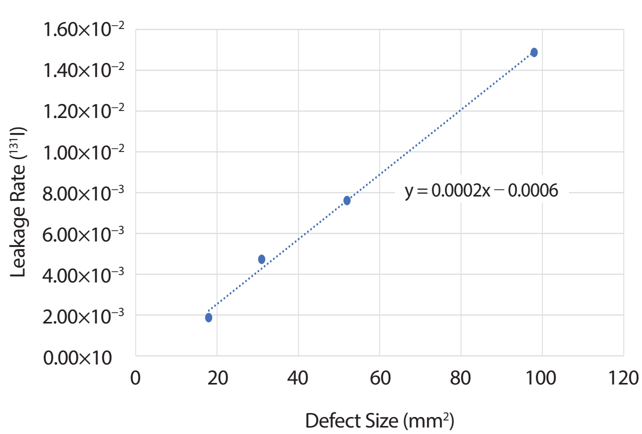

Fig. 6. Defect size vs leakage rate.

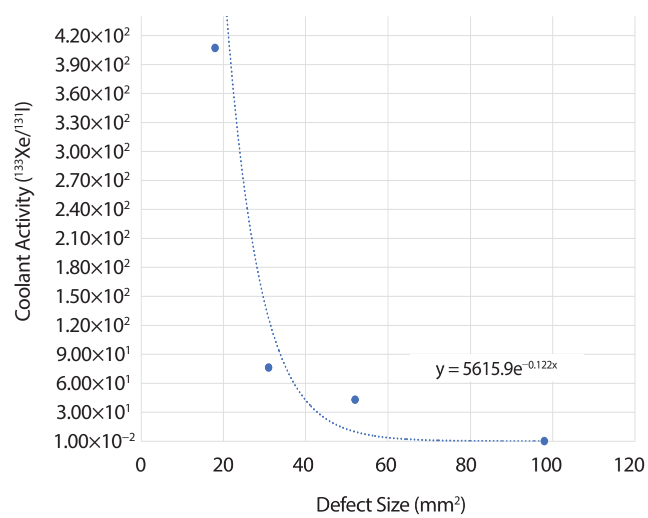

Fig. 7. Defect size vs $^{133}Xe/^{131}I$.

inventory activity of ^{131}I (Pred. ^{131}I) generated in a fuel rod, it is estimated as one fuel rod which is defective and if more activity is measured in the coolant, the number of defective fuel rods is calculated in proportion to measured activity as shown in Fig. 5.

Based on Fig. 4, we developed a program that can automatically estimate the number of defective fuel rods by inputting the element linear power and the coolant activity of ^{131}I (Meas. ^{131}I), as shown in Fig 5.

To check the accuracy for prediction of the number of defective fuel rod, it was applied to the case of the defective fuel. As a result of reviewing to the case that one fuel

Table 3. Analytical method for defect cause (≤ 120 MWHr/kgU)

Defect Shape	Endcap Crack	Hydride Blister	Hole
Root Cause/ w%	Debris /0.5 H_2 Contamination /0.5	Incomplete weld /1	Debris/1

Table 4. Analytical method for defect cause (> 120 MWHr/kgU)

Defect Shape	Endcap Crack	Hydride Blister	Axial crack	Hole
Root Cause /w%	Debris /0.1	Debris /0.2	Power ramp/1	Debris/1
	Endcap Porosity/0.5	H_2 Contamination/0.1		
	Incomplete weld/0.2	Incomplete weld/0.5		
	High burn-up/0.2	High burn-up /0.2		

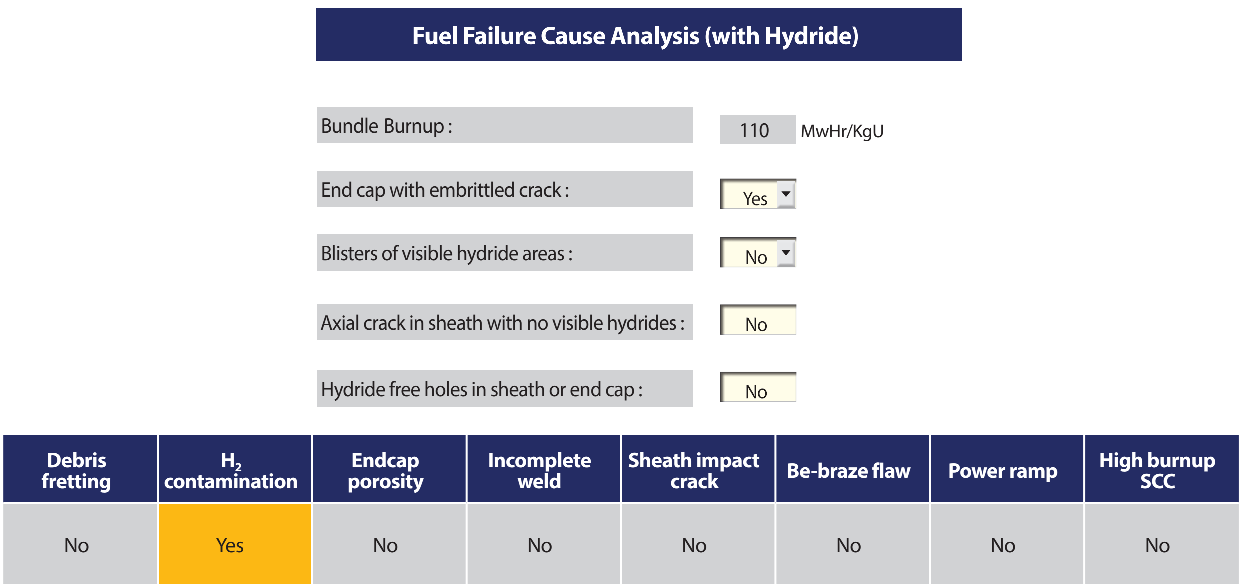

Fig. 8. Program for root cause analysis.

rod was actually defective, it was proved that the predicted value was satisfied as shown in Table 2.

3.2 Prediction for the defect size

The defect size of the fuel rod cannot be verified without seeing the defect. However, the defect size can be predicted without dismantling the fuel using the method for estimating the defect size obtained from Fig. 6 and Fig. 7.

The correlation was analyzed by comparing the leakage rate for each defect size after measuring the actual defect size shown in Table 1. Based on this, even if the defect size cannot be checked visually the leakage rate can be calculated according to Equation 4. the defect size can be estimated in reverse using the leakage rate. Since the data was calculated based on four defective fuels, the accuracy may be somewhat lower but the accuracy is expected to increase when more data is accumulated in the future.

Therefore, it is possible to measure the size of the defect by the leakage rate of ^{131}I or the ratio of ^{133}Xe to ^{131}I. In particular, the ratio ^{133}Xe to ^{131}I is applied to very tiny defects.

3.3 Prediction of defect causes

By analyzing the database of the defective spent fuel, it is found that there are various defect characteristics depending on the burn-up, and typical defect types are classified according to the root cause. Through this data, a method to identify the root cause of the defect can be established according to the defect shape.

The root causes for defective fuel were analyzed by dividing the region of burn-up into high burn-up and low burn-up using the intermediate value. If the fuel burn-up is 120 MWHr/kgU or less, the root causes for each defect shape are as given in Table 3. And if the fuel burn-up exceeds 120 MWHr/kgU, the root causes of each defect type are as shown in Table 4. A program to predict the root cause is shown in Fig. 8.

4. Conclusions

Based on leakage characteristics of fission products and the inventory of nuclides in fuel rods by ORIGEN and analysis of coolant activity, a program was developed for the first time in Korea to predict the number of defective fuel rods and the root cause of spent fuel defects after analyzing the fuel. In addition, a method to estimate the defect size was established using a database for fuel defects.

It is anticipated that this program will be applied in the future to technologies that can predict the degree of defects.

Acknowledgements

This work was carried out under the project "Development of the Root Cause Investigation for CANDU Failed Fuel" in KEPCO Nuclear Fuel Co.

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