Transactions of the Korean Nuclear Society Spring Meeting
Gwangju, Korea, May 30-31, 2013

Development of Coolant Radioactivity Interpretation Code

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1. Introduction

Nuclear fuel failure evaluation by the coolant radioactivity analysis in reactor operation requires a new improvement of its evaluation, as their operational environments and fuel designs are changed. In Korea, the coolant radioactivity analysis has been performed by using the computer codes of foreign companies such as CADE (Westinghouse), IODYNE and CESIUM (ABB-CE). However, these computer codes are too conservative and have involved considerable errors. Furthermore, since these codes are DOS-based program, their easy operability is not satisfactory. Therefore it is required development of an enhanced analysis algorithm applying an analytical method reflecting the change of operational environments of domestic nuclear power plants and a fuel failure evaluation software considering user' conveniences. We have developed a nuclear fuel failure evaluation code able to estimate the number of failed fuel rods and the burn-up of failed fuels during nuclear power plant operation cycle [1].

A Coolant Radio-activity Interpretation Code (CRIC) for LWR has been developed as the output of the project 'Development of Fuel Reliability Enhanced Technique' organized by Korea Institute of Energy Technology Evanluation and Planning (KETEP). The CRIC is Windows based-software able to evaluate the number of failed fuel rods and the burn-up of failed fuel region by analyzing coolant radioactivity of LWR in operation.

The CRIC is based on the model of fission products release commonly known as "three region model" (pellet region, gap region, and coolant region), and we are verifying the CRIC results based on the cases of domestic fuel failures. CRIC users are able to estimate the number of failed fuel rods, burn-up and regions of failed fuel considered enrichment and power distribution of fuel region by using operational cycle data, coolant activity data, fuel loading pattern, Cs-134/Cs-137 ratio according to burn-up and U-235 enrichment provided in the code.

2. Deleopment of CRIC Code

For developing the inherent failure evaluation code, its algorithm [2] that is the result of research project of phase 1 has been analyzed and major variables classified. And then, data of input/output has been classified and designed.

Database for input/output of the code has been constructed and management of database for use of damage evaluation of nuclear fuels is done by combining with KHNP's Integrated Fuel Reliability Management System (iFREMA). This code was designed for more detailed analyses for specific reactor core and its fuel types by applying unique plant parameters by types of domestic nuclear power plants. And, the comparison of results between real domestic fuel failure damage cases and oversea codes [3, 4, 5].

2.1 Failed Fuel Rods Estimation Model

Failure of fuel cladding leads to the release of fission products to the primary coolant through the defect of cladding and its failure can be identified by fission product isotopes in the primary coolant. Since the released fission products have the characteristics of their half-lives, fission yields and release rates by nuclide, information of fuel damage can be obtained by using these characteristics [6]. The CRIC's failed fuel rod number estimation method is based on 3 region fission products release model commonly known by using correlation of release rate and failed fuel rod number.

Fission products in fuel pellets can be released to the primary coolant by the two releasing processes, such as from pellets to gaps between pellets and claddings, and from gaps to coolant through the defects.

The variation of nuclides per unit time in pellets is the difference between radioactivity generated by fission and radioactivity by their decay and leakage to gaps. The variation of nuclides per unit time in gaps is the difference between radioactivity released from pellets to gaps and radioactivity by their decay and leakage to coolant through the defects. The variation of nuclides in coolant is the difference between radioactivity released from fuels to coolant and radioactivity by their decay and removal by purification system. For very long times at constant operating conditions of coolant purification and the fission rate with failed fuel rods, radioactivity levels of certain nuclides in coolant are as follows:

$$A_r = \frac{n\,F_f\,Y\,\nu\,\varepsilon\,\lambda}{V(\lambda+\nu)(\lambda+\varepsilon)(\lambda+\beta)} \qquad (1)$$

Where,

- n Number of leaking rods
- F_f Fission rate per rod
- Y Yield of the nuclide measured per fission
- ν Probability that atom from the fuel inventory of the nuclide will be released from the fuel per unit time
- ε Probability that an atom in the gap inventory will be released per unit time
- β Purification cleanup rate
- λ Isotope decay constant

V RCS Volume

Mean radioactivity in coolant is proportional to the number of failed fuel rod because radioactivity in coolant is released from fuel rods. However, radioactivity released to coolant is depended on power of failed fuel rods, burn-up and defect size.

It is able to estimate each failed fuel rod number by using I-131, I-133, I-134, and Xe-133 representative isotopes measured in coolant by fuel failure using "the Estimation Method Based on Iodiones and Xenon" in CRIC. It is able to calculate real activities of iodine isotopes (I-131, I-133) except activities generated by tramp uranium in coolant by using measured I-134 activity and estimate failed fuel rod number by assuming linearly proportional to Xe-133 activity in coolant due to direct recoil from pellet to coolant by using empirically derived average release coefficient, core power and fuel rod power.

"The Combined Method" in CRIC is the estimation method for failed fuel rod number considering difference of leakage characteristics of iodines and offgas by temperature (Linear Heat Generation Rate (LHGR) of failed fuel rod has effect on fission products release), and using least square method for solving undefined coefficient ν and ε of 3 Region Model.

2.2 Failed Fuel Burn-up Estimation Model

For the estimation of failed fuel burn-up there are reference data of Cs-134/Cs-137 ratio calculated by SCALE code (ORIGEN-S 6.1) [7]. And the calculated Cs-134/Cs-137 ratios are calibrated by domestic failed fuel Cs-134/Cs-137 ratio data. The CRIC estimates failed fuel burn-up according to reference data in Fig. 1, based on measured Cs-134/Cs-137 ratio and fuel enrichment.

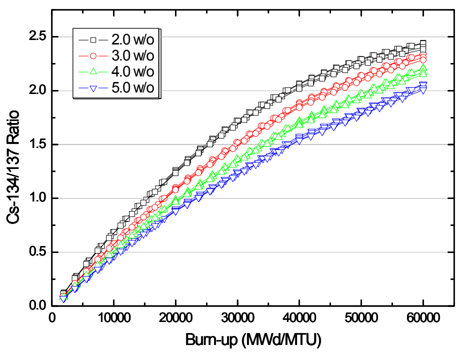

Fig. 1 Reference Data for Measured Cs-134/137 Ratio versus Failed Fuel Burn-up

2.3 Correcting Measured Data by Statistical Technique

When batch analyses for data during a certain period or user selected in measured coolant activity data are performed, the CRIC calculates averages(m) and standard deviation(σ) for each isotope, and simulates code as input adjusted as averages within certain ranges ($m \pm \sigma$ or $m \pm 2\sigma$). A data filtering and adjusting function on the CRIC has also a function for reducing effect on coolant activity by external factor such as rapid rising of I-131 level (iodine spiking) in coolant during reactor power cutback to check the fuel failure.

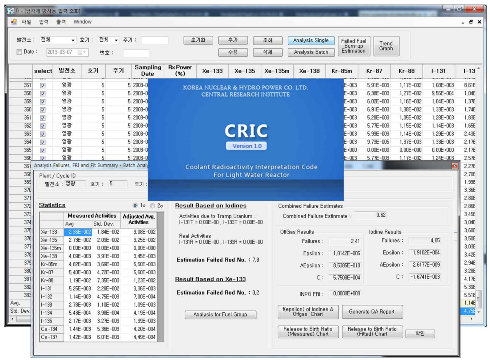

Fig. 2 UI of CRIC (Data Set, Evaluation Result)

3. Conclusions

Due to development of the CRIC, it is secured own unique fuel failure evaluation code. And, it is expected to have the following significant meaning. This is that the code reflecting a proprietary technique for quantitatively monitoring and predicting the effects on the reliability of fuels and core beforehand, in case of a necessity of changing operations in nuclear power plants, hereafter, has been obtained. It is possible to reduce a dependency on foreign technology due to possession of fundamental technology about the unique fuel failure evaluation code, and this product is able to apply to domestic nuclear power plants, or it is expected to be able to provide this code for oversea countries exporting domestic nuclear power plants, hereafter.

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