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Review of Safety Criteria for High Burn-up Fuel and Evaluation of Fuel Performance

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1. Introduction

The problems on the high burn-up fuel are mostly related with the integrity of fuel cladding material. The high burn-up problems which have not experienced yet are important issues not only in Korea but also in other developed nuclear countries. Since the nuclear fuel cladding needs a lot of R&D funds to handle its radioactive material and get the experiment data, most of the nuclear countries are cooperating with the other countries or organizations as a type of international R&D program such as CABRI, SCIP, and HALDEN program. In our standpoint, the international R&D program is a good opportunity to get a lot of the results from that kind of programs, even by the small fund. The nuclear fuel cladding is consisted of Zirconium alloy, and in domestic commercial reactors, the Korea Nuclear Fuel Company has used the ZIRLO alloy as the fuel cladding material from several years ago. The pending problems in integrity evaluation of cladding material can divide two items, and first problem is CRUD deposition during normal operation and the other issue is related with high burn-up. The CRUD deposition can be removed by ultra-sonic cleaning technology, and recently, KINS has approved that technology in its review process. Even though there are no important issues of fuel cladding material during current operation, the integrity evaluation of the high burn-up fuel cladding must be considered as a safety pending issue.

In this study, the safety criteria to evaluating the high burn-up fuel cladding were reviewed and the states of the each safety criteria were analyzed. At the same time, the safety evaluation of commercial cladding was performed using verification code, FRAPCON, and the safety factors and R&D issues of cladding material will be presented for the high burn-up operation.

2. Fuel Safety Criteria

The safety criteria can be divided to several items to evaluate the fuels. It is mainly possible to classify as the Figure 1, according to the three operating conditions. The current safety criteria for light water reactors, which form the large majority of the existing commercial nuclear power plants in the world, were developed during the late 60's and early 70's. During the development of these criteria and methods, high burn-up was thought to occur around 40MWd/kg; data up to this burn-up had been included in data bases for criteria, codes, and regulatory decisions, and it was believed that some extrapolation in burn-up could be made. By the mid 1980s, however, changes in pellet microstructure had been observed from a variety of data at higher burn-up along with increases in the rate of cladding corrosion. Thus, it became clear that something new was happening at high burn-up and/or new operating environments, and that continued extrapolation of transient data from the existing low burn-up (traditional operating environment data base) was not appropriate.

On the other hand, in order to optimize fuel cycle cost, the nuclear industry began work in the mid 80's on new fuel and core designs with the aim of increasing the fuel burn-up, e.g. for extending the cycle length or upgrading the power level. This again lead to a number of basic design changes and new cladding materials. Fuel design should be in concord with the general design criteria that govern the design and operation of power plants. Thus existing fuel safety criteria are examined against design elements as applicable to date. Figure 1 classifies the current fuel safety criteria according to the operation categories. It shows that most of the safety criteria are affected by high burn-up, as one of the new design elements (such as new fuel design, new core design, new cladding material, new manufacturing process, long fuel cycle, up-rated power, high burn-up, MOX, water chemistry change and new operating practices). Because there are close connection between safety criteria and integrity of cladding, it is very important to make a conformation of safety criteria during design basis accident such as LOCA and RIA. The cladding failure by these accidents must be related with the pellet cladding interaction mechanism and high temperature oxidation. Therefore, we will discuss the safety criteria for high burn-up, its failure mechanism, and regulatory position for LOCA and RIA in this study.

3. Fuel Performance Evaluation

3.1. FRAPCON-3 Computer Code

The FRAPCON-3 fuel rod performance computer code has been developed and recently updated by the U.S. NRC and Pacific Northwest National Laboratory (PNNL)[1]. The current 3.3 version calculates steady-state fuel behavior at high burnup (up to 65GWd/MTU, depending on application). The major changes include an improved thermal conductivity model for urania fuel pellet, addition of MOX fuel thermal properties, and the addition of corrosion and hydrogen pickup parameters to advanced cladding types such as ZIRLO™ and M5™.

3.2 Evaluation of the Code Performance

3.2.1 PWR X-Rod

The code performance was evaluated by utilization of commercially used a PWR fuel rod. The X-rod, with an initial U-235 enrichment of 4.5% and a ZIRLO™

cladding, has been irradiated for 3 cycles (1424 days) in the CE type NPP until a rod average burnup reached to 53 GWd/MTU. The PIE of the used X-rod has been done in a hot cell at room temperature in atmospheric pressure. Table 1 shows an oxide thickness, rod OD, axial length, internal pressure and FGR of PIE and code predicted results. Considering the best-estimate nature of the code, it revealed generally acceptable analysis results except for oxide thickness. FRAPCON-3 code estimated a half percent lower oxide thickness in the ZIRLO™ fuel rod than the measured data, and fuel vendor code also predicted non-conservative oxide formation when used 0.75 RevNAC model.

3.2.2 Halden tests (IFA-432)

IFA-432 tests have been performed at Halden heavy boiling water reactor to test the long-term steady-state performance of BWR type fuel rods, changing diametral gap thickness of the fuel rods from 76 to 380 microns [2]. FRAPCON-3.3 code used to reevaluate the fuel centerline temperature and FGR. Fig. 2 shows the changes of fuel centerline temperature depending on the gap size. As the gap size increased, over predicted fuel centerline temperature can be found in accordance with the over-prediction of the FGR.

From the above analysis results, we could know that the FRAPCON-3 to be used as a performance evaluation code for high burn-up fuel rods, and it is necessary to closely monitor the followings.

1) The appropriateness of corrosion model to the advanced cladding alloys especially for high burn-up,
2) FGR model and its influences on the thermal conductivity degradation,
3) and, did not mentioned above analysis, the effect of crud formation on the accelerated corrosion of the fuel rod.

4. Summary

In this study, the fuel safety criteria for high burn-up were reviewed, and especially, the re-establishment of safety criteria related with design basis accident and the regulatory positions were presented. On the other hand, the applicability of performance analysis code was also discussed.

Table 1. Summary of the PIE and computer code analysis results of PWR X-rod

	PIE	FRAPCON-3	Vendor code
Max. Oxide Thickness, μm	73.8	36.4	54.9[1] 73.1[2]
ROD OD (at 3400mm), mm(Δ)[3]	9.68 (-0.02)	9.67 (-0.03)	9.66 (-0.04)
Axial Rod Length, mm, (Δ)	4110.5 (16.8)	4108.4 (14.7)	4110.2 (16.5)
Internal Pressure, MPa (Δ)	4.3 (1.6)	4.72 (2.02)	3.85 (1.15)
FGR, %	1.13	1.82	2.70

[1] and [2] was calculated by 0.75RevNAC and 0.92RevNAC model, respectively.

[3] Δ means the difference between initial and measured/calculated value, initial – measure/calculated.

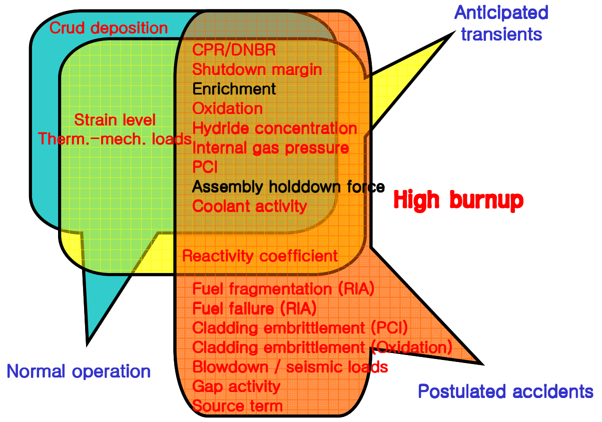

Fig.1 Categories of current safety criteria for nuclear fuel

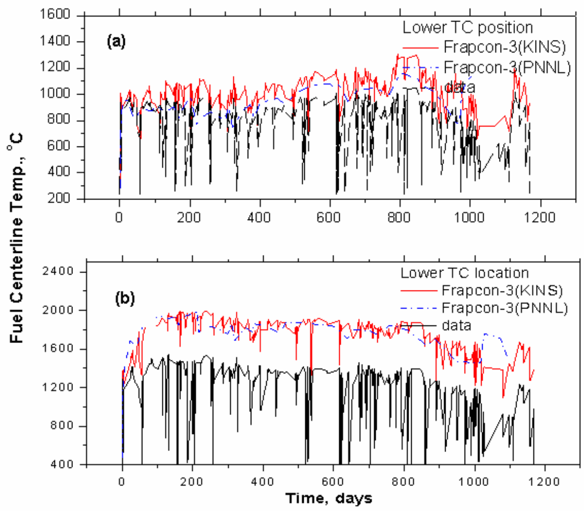

Fig.2 Changes of fuel centerline temperature in IFA-432 test rods, (a) Rod3 (small gap, 76μm) and (b) Rod2 (large gap, 380 μm).

REFERENCES

[1] G.A. Berna, et. al., FRAPCON-3 : A Computer Code for the Calculation of Steady-State, Thermal-Mechanical Behavior of Oxide Fuel Rods for High Burnup, NUREG/CR-6534 Vol.2, 1997.

[2] D.D Lanning et. al., FRAPCON-3: Integral Assessment, NUREG/CR-6534 Vol.3, 1997.