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Development Status of Accident Tolerant Fuels for Light Water Reactors in Korea

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1. Introduction

Research on accident tolerant fuels (ATFs) is aimed at developing innovative fuels, which can mitigate or prevent the consequences of accidents.

In Korea, innovative concepts are being developed to improve fuel safety and reliability of LWRs during accident events and normal operations. ATF technologies will be developed and commercialized through a sequence of long-lead and extensive activities. The interim milestone for new fuel program is that we would be ready for an irradiation test in commercial reactor by 2021. This presentation deals with the status of ATF development in KOREA and plan to implement new fuel technology successfully in commercial nuclear power plants.

2. Accident Tolerant Fuel Pellets

Desirable attributes for enhanced accident tolerance fuel pellets are increased thermal conductivity and retention capability of corrosive and radioactive fission products (FPs). A high uranium density fuel is also required in particular ATF cladding concepts, to compensate for the anticipated reduction of neutron economy.

In Korea Atomic Energy Research Institute (KAERI), two kinds of new pellet concepts are being evaluated. As for near-term technology, a microcell UO_2 pellet is being studied. Focus is to use existing infrastructure, experience, and expertise to the maximum extent possible, so that this evolutionary concept can be used in the relatively near future. For the long-term perspective, nitride- and/or silicide-based composite pellets that have a high uranium density and high thermal conductivity are being investigated.

Kepco Nuclear Fuel Company (KepcoNF) is developing SiC-TRISO composite fuel technology. The main benefits of TRISO based coated particle fuel are superior FPs retention capability and thermal conductivity.

2.1 Microcell UO_2 [1-3]

In microcell UO_2 concepts, UO_2 grains or granules are covered by thin cell walls. There are two kinds of microcell UO_2 pellets under development in KAERI, classified according to the material type composing the cell wall. The first is a metallic microcell UO_2 pellet and the second is a ceramic microcell UO_2 pellet.

The metallic microcell UO_2 pellet is a high thermal conductive pellet with a continuously connected metallic wall. Recent impact assessments of the thermal conductivity of the fuel in a loss-of-coolant accident (LOCA) progressing in a pressurized water reactor (PWR) showed that an increase in thermal conductivity reduces both the peak cladding temperature and the quench time of the fuel rod.

The main purpose of the ceramic microcell UO_2 pellet is to minimize the FPs release contained in the pellet structure by providing a microcell structure with oxide additives. An improvement in FP retention capability leads to a reduction of the inner surface cladding corrosion caused by FPs as well as the internal pressure of the fuel rod. A soft thin wall facilitates the fast creep deformation of the pellets, thereby reducing the mechanical loading of the cladding under operational transients. A mesh-like rigid wall structure is also expected to prevent the massive fragmentation of pellets during a severe accident.

Fabrication feasibility for microcell UO_2 pellets has been demonstrated by applying conventional sintering process. Thermo-physical property measurements under normal operating and selected accident conditions showed that design concepts are successfully implemented in both metallic and ceramic microcell UO_2 pellets. For a metallic microcell UO_2 concept, the design and composition of the metallic wall is being modified to improve the chemical stability with high-temperature steam. The irradiation test with highly instrumented fuel rods under normal PWR operating conditions in the Halden reactor is also an important ongoing mission.

2.2 High density pellets [1,2,4]

The exploration of UO_2-uranium mononitride (UN) and UO_2-uranium silicide (U_3Si_2) composite fuels is ongoing for a long-term application. UN and U_3Si_2 are known to have many benefits, such as higher uranium density and thermal conductivity. However, there are several drawbacks in the utilization of these materials as a fuel for water reactors. In this study, the feasibility of composite fuel concepts in which the uranium nitride or uranium silicide particles are embedded in a UO_2 matrix is being evaluated. These concepts are based on the assumption that a protective matrix of UO_2 would prohibit water corrosion of the nitride phase or suppress irradiation swelling or chemical interaction of silicide at temperatures relevant to LWR operation.

2.3 SiC-TRISO composite fuel

TRISO particles embedded SiC ceramic fuel is expected to have superior retention capability of FPs and high thermal conductivity.

Although these attractive features makes this concept one of the potential candidates for accident-tolerant fuel, there are several key issues that need to be addressed for the LWR application, such as the fabrication of dense composite pellet by using pressure-less sintering process, increase of TRISO packing density, and using enriched uranium of 20% U^{235} and uranium nitride kernel to keep the today's fuel cycle length.

KepcoNF are developing SiC matrix sintering and joining technology to enhance the pellet fabrication economy. Efforts are focused on development of pressure-less sintering process by using small amount of additives to enable engineering-scale production of composite pellets. Thermo-mechanical properties of fabricated pellets are being evaluated as well.

3. Accident Tolerant Fuel Claddings

Many ATF cladding concepts are being considered to improve on the performance of current Zr-based alloys, especially in terms of oxidation resistance and mechanical strength under accident conditions.

In KAERI, the surface-modified Zr alloy and the SiC ceramic composite concepts are developed as ATF claddings as a way to reduce hydrogen generation and to mitigate ballooning and rupture opening of tube under the accident. The development plan of the surface-modified Zr alloy concept is focused on as a near-term application and that of the SiC composite concept is considered a long-term application.

3.1 Surface-modified Zr alloy cladding [1,2,5,6]

Surface-modified cladding has coated layers on the outer surface of Zr alloy tube. The role of coated layer is an enhancement of the oxidation resistance and mechanical strength of Zr alloy cladding. Since this concept is able to utilize existing infra-structure of current Zr cladding technology without major change, it could be implemented in the industry in relatively near future.

Cr-based alloy systems are developed for outer coating layer to reduce oxidation rate of cladding under the high temperature steam and coolant water. With a coating layer of Cr-based alloy alone, however, the mechanical strength of the Zr alloy tube could not be increased. Partially oxide dispersion strengthened (ODS) layer between the outer coating and Zr alloy has been designed to increase high temperature strength, which would lead to reduced ballooning and rupture. By a combination of two types of surface modification technology of the surface coating and partial ODS treatment, the accident tolerance of Zr alloy cladding can be considerably improved.

Three dimensional (3D) laser coating method has been developed to make dense and adhesive coating layers on the Zr alloy tube. This coating technique can be applied to 4m-long tube without much difficulty and changing the cladding property.

Zr alloy was coated with Cr-based alloy and then subjected to high temperature steam of 1200°C to evaluate oxidation behavior. The test result showed that the oxidation rate was only 1/1000 of that of uncoated Zr alloy, implying that hydrogen generation would be also decreased by the same degree.

For partial ODS treatment, Y_2O_3 particles are applied to the surface of Zr alloy specimen, using 3D laser scanning, and then performed tensile test. While the ODS alloyed layer was only 20% of the specimen thickness, the test showed that strength of the specimen increased about 2 times at 500°C.

3.2 SiC composite cladding [2,7,8]

SiC ceramics, especially in their composite form (SiC_f/SiC composites), have superior high-temperature properties, excellent irradiation resistance, inherent low activation and other superior physical/chemical properties. Compared to the current Zr alloys, the SiC composites also offer a reduced neutron absorption cross section and an outstanding oxidation resistance to high-temperature steam. The combination of these attractive features makes the SiC composites one of the leading candidates for accident-tolerant LWR fuel cladding and core structures. Although the SiC composites are expected to provide outstanding safety features under severe accident conditions, there are several key issues that need to be addressed for the LWR application, such as the fabrication of thin-walled long tubes, the hermetic joining of end-cap seals, and capability of FP retention, corrosion under the normal operating conditions of an LWR.

The corrosion of SiC under PWR-simulating water conditions was significantly reduced with the control of dissolved hydrogen, by retarding the formation of the surface oxide layer. Fiber winding patterns affected the hoop strength of SiC composite tubes, mainly due to the difference in fiber volume fraction. Efforts on optimal tube designs are in progress for an improvement of mechanical reliability of composite tubes.

4. Irradiation Performance Test

An irradiation program for microcell UO_2 pellets and surface modified claddings under bilateral cooperation with the International Consortium is ongoing. The main objectives are to study the in-reactor performance, by irradiating highly instrumented fuel rods under normal PWR operating conditions in the Halden reactor.

In the case of microcell UO_2 pellets, the focus is on the fuel performance parameters such as fuel centerline temperature, fuel densification and swelling, and FGR behavior. After the irradiation test, several post-irradiation tests will be performed to evaluate the fission damage on the wall structure, transport kinetics of volatile FPs through or along the wall to the cladding inside, chemical interactions between the wall and FPs, and redistribution of wall components in a pellet, and so on.

The irradiation performance of surface modification cladding has to be evaluated since the most important factor of ATF cladding materials is the in-pile performance such as the corrosion, creep, growth, and microstructural characteristics at the interface between the coated layer and Zr alloy under normal operation conditions.

REFERENCES

[1] Y.H. Koo, J.H. Yang, J.Y. Park, K.S. Kim, H.G. Kim, D.J. Kim, Y.I. Jung, K.W. Song, KAERI's development of LWR accident tolerant fuel, Nucl. Technol. 186, p. 295-304, 2014.

[2] H.G. Kim, J.H. Yang, W.J. Kim, Y.H. Koo, Development Status of Accident-tolerant Fuel for Light Water Reactors in Korea, Nuclear Engineering and Technology, 48, p.1-15, 2016.

[3] D.J. Kim, Y.W. Rhee, J.H. Kim, K.S. Kim, J.S. Oh, J.H. Yang, Y.H. Koo, K.W. Song, Fabrication of micro-cell UO_2-Mo pellet with enhanced thermal conductivity, J. Nucl. Mater. 462, p.289-295, 2015.

[4] J.H. Yang, D.J. Kim, K.S. Kim, Y.H. Koo, UO_2-UN composites with enhanced uranium density and thermal conductivity, J. Nucl. Mater. 465 p. 509-515, 2015.

[5] H.G. Kim, I.H. Kim, Y.I. Jung, D.J. Park, J.Y. Park, Y.H. Koo, Microstructure and mechanical strength of surface ODS treated Zircaloy-4 sheet using laser beam scanning, Nucl. Eng. Technol. 46 (4) p. 521-528, 2014.

[6] H.G. Kim, I.H. Kim, Y.I. Jung, D.J. Park, J.Y. Park, Y.H. Koo, Adhesion property and high-temperature oxidation behavior of Cr-coated Zircaloy-4 cladding tube prepared by 3D laser coating, J. Nucl. Mater. 465 p.531-539, 2015.

[7] D. Kim, H.-G. Lee, J.Y. Park,W.-J. Kim, Fabrication and measurement of hoop strength of SiC triplex tube for nuclear fuel cladding applications, J. Nucl. Mater. 458 p.29-36, 2015.

[8] D. Kim, H.-G. Lee, J.Y. Park, J.-Y. Park, W.-J. Kim, Effect of dissolved hydrogen on the corrosion behavior of chemically vapor deposited SiC in a simulated pressurized water reactor environment, Corros. Sci. 98 p. 304-309, 2015.