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Optimization study of Ultra-long Cycle Fast Reactor core concept

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ABSTRACT

An optimization of an Ultra-long Cycle Fast Reactor (UCFR) design with a power rating of 1000 MW (electric), UCFR-1000, has been performed. Firstly, geometric optimization has been performed in the aspects of core size and core shape in terms of thermal–hydraulic (TH) feedback. Secondly, fuel composition optimization has been performed by adopting pressurized water reactor (PWR) spent fuel (SF) as a blanket material as well as natural uranium (NU). Thirdly, thorium has been loaded in the inner core for the optimization of radial power distribution. Lastly, a small-size UCFR with a power rate of 100 MWe has been developed with optimization of maximum neutron flux and fast neutron fluence limit for a short term deployable nuclear reactor.

The equivalent diameter and the height of the optimized UCFR-1000 core are 5.9 and 2.4 m, respectively, while the equivalent diameter and the height of the optimized UCFR-100 core are 4.3 and 1.0 m, respectively. The size of the optimized UCFR-1000 has been enlarged in the radial direction and shortened in the axial direction from those of the initial UCFR design (Tak et al., 2013a) and this modification makes the burning speed of active core movement slower. It has been confirmed for both designs that a full-power operation of 60 years without refueling is feasible with respect to isotopics and criticality by a breed-and-burn strategy. The core performance characteristics of both designs have been evaluated in terms of axial/radial power shapes, neutron flux and nuclide distributions, breeding ratio, reactivity feedback coefficients, control rod worth, etc. By the design optimization study in this paper, the reductions of maximum neutron flux, fast neutron fluence, and axial/radial power peaking have been achieved, which are favorable for the safety of the UCFR.

1. Introduction

Sodium-cooled Fast Reactor (SFR) is a representative reactor concept among the six reactor systems for Generation IV Reactors (Gen-IV) and many countries have especially focused on SFR development (ANL, 2014; ANS, 2005; Bouchard and Bennett, 2008; U.S.DOE and GIF, 2002; Tak et al., 2013a; NEA, 2012; WNA, 2014). Many SFR concepts have been proposed continuously for reinforcing the advantage and counteracting the disadvantage of sodium-cooled reactors. Among the SFR concepts, various strategies have been developed for the construction of a long cycle reactor. Constant Axial shape of Neutron flux, nuclide densities and power shape During Life of Energy production (CANDLE), proposed by Hiroshi Sekimoto, is the long life reactor concept that adopts a breed-and-burn strategy in the axial direction so the core life can be extended by increasing the axial length. In addition, CANDLE operates such a long cycle for itself without any reactivity control by an operator (Sekimoto et al., 2001). Terra Power Corporation is a nuclear reactor design company and one of the primary investors is Bill Gates. This company has developed Traveling Wave Reactor (TWR) which is also a breeding reactor with a long life core but the breeding direction is radial and travels from the inside out. They have developed a 600 MW (electric) prototype reactor intended to start up around 2022, which is the foundation for full commercialization by the late 2020s (Hejzlar et al., 2013). 4S is another representative SFR reactor concept in Japan with a power rate of 10 and 50 MWe. It applied advanced safety concepts and technologies from the two small size fast reactors, MONJU and JOYO. 4S has a movable reflector surrounding the core for compensating the burn-up reactivity loss over the thirty-year lifetime (Ueda et al., 2005). A once-for-life, uniform composition, blanket-free and fuel-shuffling-free reference core was designed for the Encapsulated Nuclear Heat Source (ENHS) by Korea Atomic Energy Research Institute (KAERI) and the University of California. Its

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design goal is a nearly zero burnup reactivity swing throughout ~20 years of full-power operation up to the peak discharge burnup of more than 100 GWd/t HM. Its nitride fuel core, relative to the reference metallic fuel core, offers up to ~25% higher discharge burnup and longer life, up to ~38% more energy per core, a significantly more negative Doppler reactivity coefficient, and less positive coolant expansion and coolant void reactivity coefficient but a somewhat smaller negative fuel expansion reactivity coefficient (Hong et al., 2005). Argonne national laboratory has presented two long life fast reactors, Ultra-long Life Fast Reactor (ULFR-3000) and Advanced sodium-cooled Fast Reactor (AFR-100) that have lifetimes of more than 40 and 30 years, respectively. AFR-100 aims to be sized for small local grids, transportable from pre-licensed factories to the remote plant site. To achieve such a long life, ULFR-3000 utilizes radial breeding with a radial blanket and an inner blanket while AFR-100 utilizes both axial and radial breeding by uranium enrichment zoning (Kim and Taiwo, 2010; Kim et al., 2012). There is a small size fast reactor concept in Korea, which is a compact sodium-cooled breed-and-burn reactor (B&BR) with CANDLE configuration with a power rating of 250 MWth, and it has been proposed to find the acceptable compact sodium-cooled TWR traveling in the axial direction. It has a very high fuel volume fraction of over 60% and the design has been developed continuously from the neutronics point of view (Hartanto and Kim, 2012). The Ultra-long Cycle Fast Reactor (UCFR) was developed for the purpose of 60-year operation and it has a power rating of 2600 MW (thermal). UCFR utilizes the breed-and-burn strategy by using low enrichment uranium (LEU) as an igniter and natural uranium (NU) as a blanket material. The feasibility of the core has only been reported from the neutronics point of view, therefore, optimization of UCFR was expected to perform thermal–hydraulic (TH) feedback analysis and mitigate the power peaking issue. Also, further study on using spent fuel (SF) for the blanket material was expected, which contributes to waste management issues in Korea by providing interim storage as well as uranium utilization at a high rate (Tak et al., 2013a).

In this paper, the optimization of UCFR-1000 has been performed. Firstly, the overall core shape and size have been changed to satisfy the safety limits from the thermal–hydraulic (TH) feedback of the previous UCFR-1000 (Seo et al., 2012). Secondly, pressurized water reactor (PWR) SF has been tried as the blanket material as well as NU for the fuel composition optimization, which could be one way of solving the SF storage issue in Korea (Braun and Forrest, 2013). Core performance and depletion behavior have been analyzed and core dynamic characteristics have been studied through reactivity feedback parameters and control rod worth. Thirdly, for the optimization of radial power distribution, thorium loaded UCFR-1000 has been designed. The power flattening of the UCFR core has been performed by fuel zoning with an active core region. Lastly, a small-size UCFR with a power rating of 100 MWe (UCFR-100) has also been developed as an alternative model regarding the neutron flux and the fast neutron fluence. It was analyzed with both NU and PWR SF as had been performed for UCFR-1000.

2. Design optimization of UCFR-1000

In this section, UCFR-1000 core design optimization is presented. The core design parameters have been changed from those of the old model (Tak et al., 2013a) in the process of optimizing the core size and core shape to satisfy the safety limit from the TH feedback. Thermal–hydraulic safety analysis has been performed using the Multi-dimensional Analysis for Reactor Safety-Liquid Metal Reactor (MATRA-LMR) code developed by Korea Atomic Energy Research Institute (KAERI). This code performs multichannel analysis of transient and steady-state in rod array for liquid metal reactor (Kim et al., 2002) and Steady-state LMR core Thermal–Hydraulic analysis code based on ENergy model (SLTHEN) which predicts the steady-state temperature field in an LMR core (Yang, 1997). For UCFR, MATRA-LMR performs a single assembly analysis and seeks a peak temperature of center fuel pin and SLTHEN performs whole assembly analysis with the radial peaking factors. The calculation results of the two codes are compared with the thermal safety margin and temperature limit of Design Basis Events (DBE) for sodium cooled fast reactors that are the maximum fuel rod temperature of 955 °C, the maximum cladding temperature of 650 °C, the coolant exit temperature of 560 °C, and the limit of coolant temperature of 1055 °C (Jeong et al., 2010). The TH simulation of the old model has 1478, 876, 867, and 876 °C for each; therefore, the purpose of geometric optimization is to decrease the temperatures by 1/3 so that they are lower than each limit. PWR SF loaded UCFR-1000 has also been analyzed concurrently.

2.1. Core design optimization

2.1.1. Core design requirements and key parameters

As a geometric optimization way of decreasing the temperatures in TH simulation feedback by 1/3, the axial peak factor has decreased by 1/3 through shortening the axial pin length, which leads to a decrease of the coolant flow and maximum neutron flux. The axial pin length could be shortened due to the fact that the core gets larger in the radial dimension and the axial movement of the active core gets slower, which makes it possible to operate a cycle length with a shorter fuel pin length.

The smear density includes not only the radial but also the axial swelling effect by modeling a uniform radial growth of 25% and axial growth of 8%. The axial swelling effect was not considered in designing the old model so the amount of heavy metal decreased while the design power and cycle length remain unchanged. Table 1 shows the design parameters of the optimized UCFR-1000 loading NU model and SF model. The design objective is to analyze a core which operates 60 years once through without refueling with a power rating of 1000 MWe and maintains criticality with non-enriched fuel such as natural uranium or PWR spent fuel. The geometry and other parameters have been determined to satisfy these design criteria.

The equivalent core diameter is 5.9 m and it is 2.4 m tall. The core shape has been shortened axially but enlarged radially compared to the UCFR-1000 core presented by Tak et al. (2013a) whose diameter is 4.8 m and height is 3.6 m. Average power density and linear power have decreased as the core volume increased. From the simulation result of the geometrically optimized UCFR-1000, it has a maximum fuel rod temperature of 767 °C, a maximum cladding temperature of 566 °C, and a maximum coolant exit temperature of 541 °C without the peaking factor so the optimization succeeded in satisfying the TH safety limit (Seo, 2013).

PWR SF was loaded for the blanket material in the UCFR instead of NU. The fuel form of SF-7Zr was decided from the previous SF loading trial of UCFR-1000 (Tak and Lee, 2012a) and it was analyzed in comparison to the NU loaded UCFR (Tak et al., 2013b). To transform the oxide spent fuel into metallic fuel, it only needs the reduction reaction process for metallization and injection molding process with zirconium, which has a nonproliferation level of Direct Use of PWR spent fuel In Candu (DUPIC) process. The composition of the PWR SF is presented in Table 2. It was calculated with the ORIGEN2 code with several assumptions: (a) the discharge burnup is 50 GWD/MTU, (b) 10 year cooling time, (c) the fission gases and fission products with an evaporation point of lower than 1000 °C are removed, (d) no other process to change the actinide composition was performed. ^{238}U is the main nuclide