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Thermo-mechanical behavior of all the metallic fuel rods in a sodium-cooled fast reactor

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ABSTRACT

A detailed analysis of neutronics, core thermal-hydraulics, and fuel performance has been developed to predict the thermo-mechanical failures of all the metallic fuel rods in a sodium-cooled fast reactor. The neutronics analysis provides a reload pattern for fuel assemblies and power/flux spatial distributions for each fuel rod. The core thermal-hydraulics allows the calculation of the cladding temperatures via subchannel analysis. The fuel performance analysis evaluates the thermo-mechanical integrity of all the fuel rods by considering the detailed spatial and temporal variations of the thermal power, neutron flux, and cladding temperature during the in-core lifetime of each fuel assembly. This analysis leads to improved computational accuracy and provides more comprehensive physical information for each code than a conventional one-dimensional fuel performance calculation. It is evident that, compared with a simple conservative one-dimensional analysis, the present method will improve plant performance while maintaining the thermal margin. The developed analysis has been applied to evaluate the fuel performance of a Prototype Gen IV sodium-cooled fast reactor candidate core by considering the uncertainties of the design parameters.

1. Introduction

A fuel limiting factor analysis in core thermal design is highly important to ensure the safe and reliable operation of reactor systems. The Nuclear Regulatory Commission (NRC) provides general design criterion (GDC) 10 for core thermal-hydraulic design (NRC, 2017a), which is equally applied to the liquid metal reactor GDC of ANSI/ANS 54.1, section 3.2.1 (ANS, 1989). This GDC states that the reactor core shall be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs).

The typical SAFDL used in pressurized water reactor (PWR) design is a departure from the nucleate boiling ratio (DNBR) on the fuel cladding surface. However, the sodium thermal conductivity of a sodium-cooled fast reactor (SFR) is approximately hundreds of times larger than the thermal conductivity of water. Moreover, the coolant boiling temperature in an SFR is approximately 900 °C, which is much higher than that of the water coolant in a PWR. Considering typical operating temperatures, an SFR has a thermal margin of approximately 300 °C to its boiling point. Therefore, unlike a DNBR in a PWR, the core thermal design of SFRs requires ensuring an appropriate fuel thermo-mechanical performance, where the design limits are highly related to the spatial and temporal variations of the thermal power, neutron flux and temperature under various operating conditions.

The fuel design criteria in metallic fuel SFRs during normal operation and AOOs, which include the cumulative damage fraction (CDF), inelastic strain and hoop stress, require no eutectic liquefaction and no fuel melting. These criteria are related to the cladding integrity because the fuel cladding provides the primary barrier to prevent the release of radioactive materials. Thus, the cladding strength must have a sufficient margin to assure that the probability of cladding failure is acceptably low. Eutectic liquefaction and fuel melting are temperature limits that can be directly calculated from the core thermal-hydraulic code. However, the other criteria should consider multi-physics phenomena from various design domains such as neutronics, core thermal-hydraulics, and fuel performance.

In previous SFRs, each design domain has been separately constructed. Neutronics is used to attempt flattening the spatial power distribution and reducing the power peaking factors to maximize the thermal margin. Core thermal-hydraulics is used to perform a flow allocation for each subassembly and calculate the core temperature distributions, satisfying the temperature design limits. Then, the fuel performance is used to assess the allowable limiting temperatures based on conservative input parameters over the entire core or fuel assembly; this analysis neglects the radial peaking effects and conservatively

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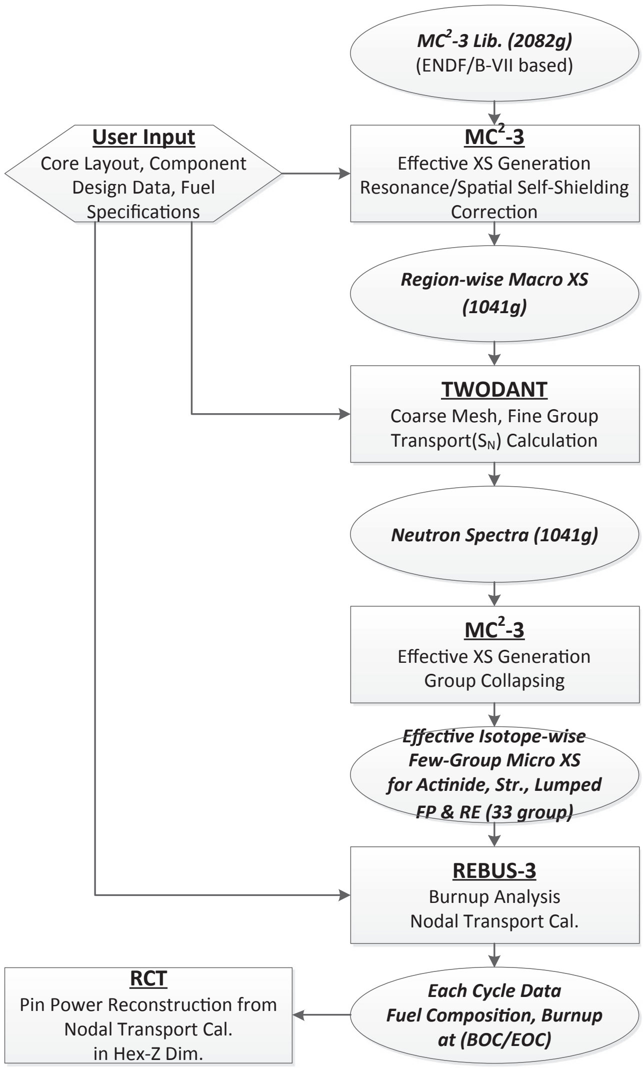

Fig. 1. Calculation procedure for PGSFR core neutronics design.

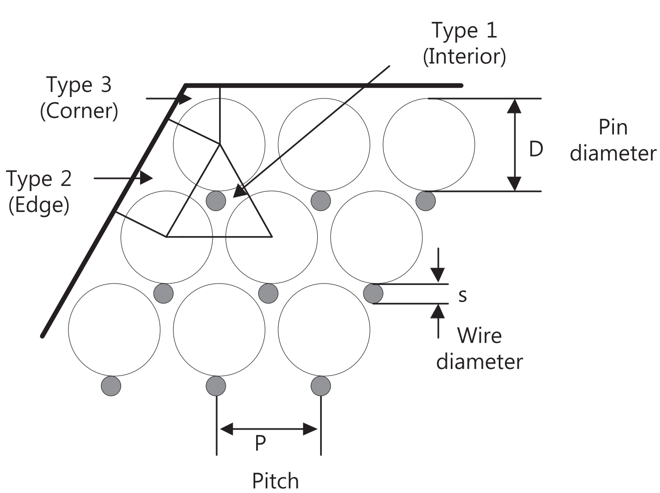

Fig. 2. Core thermal-hydraulic subchannel model.

evaluates the behavior of a single fuel rod.

In this work, a multi-physics analysis of neutronics, core thermal-hydraulics, and fuel performance was developed to predict the thermo-mechanical failures of the entire set of metallic fuel rods in an SFR under normal full power operation. This method considered the detailed spatial and temporal variations of thermal power, neutron flux, and cladding temperature at the fuel rod level during the in-core lifetime of each fuel assembly. Compared to a conventional simple one-dimensional fuel performance calculation, this analysis realized improved computational accuracy and provided more comprehensive physical information for each code. In general, the fuel rod with the maximum thermal power is located near the subassembly wall close to the core center, whereas the fuel rod with the maximum temperature is located around the subassembly center. However, as a typical one-dimensional analysis uses the largest input values over the subassembly, the predicted fuel design limits are greater than the actual values. Unlike normal operation, design basis accidents allow a certain fuel damage fraction, depending on the accident occurrence frequency and the corresponding off-site radiation dose criteria (ANS, 1983). The present method not only showed an intuitive fuel damage potential distribution but also provided initial conditions at the fuel rod level for transient fuel performance analysis during a design basis accident. Each fuel rod experiences an accident with different initial conditions of fuel temperature, cladding wastage, gas plenum pressure, accumulated CDF, etc.

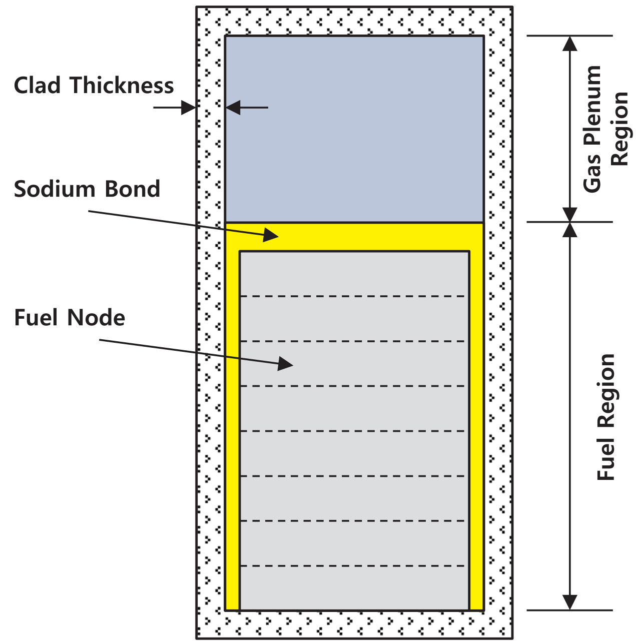

Fig. 3. Fuel pin performance model.

The developed analysis was applied to evaluate the fuel lifetime performance of the Prototype Gen IV SFR (PGSFR) candidate core by considering the uncertainties of the design parameters (Yoo et al., 2016). The Korea Atomic Energy Research Institute (KAERI) has performed SFR design with the final goal of constructing a prototype plant by 2028. The PGSFR project is aimed at developing the conceptual design of an SFR, which is necessary for the efficient utilization of uranium resources and the reduction of high waste volumes and toxicity levels. The PGSFR core is loaded with 112 fuel assemblies of low enriched uranium metallic fuel (U-10%Zr). Each fuel assembly consists of 217 fuel rods arranged in a triangular configuration within a hexagonal duct. This method, which evaluated a total of 24,304 (112 × 217) fuel rods after approaching the equilibrium core, was performed by considering the uncertainty of each design parameter as 2σ according to nuclear plant design criteria and guidelines.