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Coupled analysis of Neutronics, Core Thermal-Hydraulics and Fuel Performance to Evaluate Whole Core Fuel Failures in a Sodium-Cooled Fast Reactor

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1. Introduction

A fuel limiting factor analysis in a core thermal design is highly important to assure the safe and reliable operation of a reactor system. 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.

Typical SAFDL employed in a Pressurized Water Reactor (PWR) design is a departure from the nucleate boiling ratio (DNBR). However, the coolant boiling temperature in a sodium-cooled fast reactor (SFR) is around 900℃, which is much higher than that of the water coolant in a PWR. Therefore, instead of DNBR in a PWR, the core thermal design of SFRs requires assuring a proper fuel thermo-mechanical performance, where design limits are highly related to the spatial and temporal variations of thermal power, neutron flux, and temperature under various operating conditions. However, previous SFR analyses separately consider neutronics, core thermal-hydraulics and fuel performance. They neglect the radial peaking effects and conservatively evaluate a single fuel rod behavior.

In this work, the multi-physics analysis of neutronics, core thermal-hydraulics and fuel performance has been developed to predict a thermo-mechanical failure of whole core metallic fuel rods for a sodium-cooled fast reactor. This method reveals the improved computational accuracy and more comprehensive physical information of each code compared to a conventional simple 1-dimensional fuel performance calculation. The developed analysis was applied to evaluate the fuel performance of a candidate PGSFR (Prototype Gen-IV SFR) core by considering the uncertainties of design parameters.

2. Analysis Codes

2.1 Neutronics

The core neutronics performances were calculated using the K-CORE code system. The nuclear predictions start from the generation of the base nuclear cross-section consists of a 2,082-group from ENDF/B-VII.0. Then, the region-wise microscopic cross-section sets were generated by utilizing the effective cross-section generation module of MC^2-3[1]. The cross-section data were collapsed again into a 33-group structure by weighting the group-wise neutron fluxes calculated by TWODANT[2] in an R-Z geometry. The fuel cycle analyses were performed including the neutron flux, burnup and system constraints, and reloading stages. These calculations are carried out with REBUS-3[3]. The neutron flux is calculated based on the nodal transport theory with 33-group cross section in a hexagonal-Z geometry.

2.2 Core Thermal-Hydraulics

For a thermal-hydraulic analysis of a core consisting of subassemblies with a subchannel of a wire-wrapped rod bundle, a subchannel analysis is employed by a MATRA-LMR code[4]. It characterizes the average mass, momentum, and energy balance in every subchannel. It assumes that the axial velocity component is dominant, compared to the components in the transverse direction. A typical triangular subchannel arrangement, a control volume for an axial momentum equation, and control volumes for axial and transverse momentum equations are depicted in Fig. 1. A subchannel is a flow path designated by wire-wraps between fuel rods. There are three types of subchannels such as interior, edge and corner. The flow distribution within the subchannels is calculated from the implemented flow split correlations.

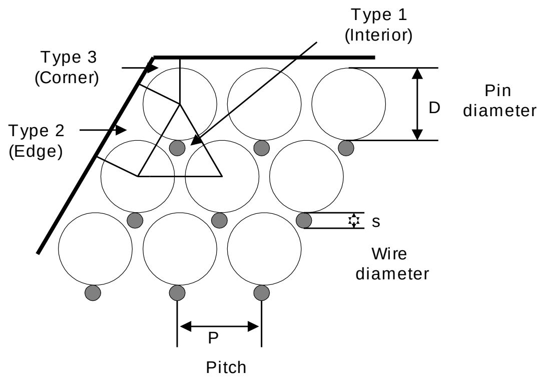


Fig. 1. Core thermal-hydraulic subchannel model.

2.3 Fuel Performance

As shown in Fig. 2, a typical metal fuel pin consists of a solid cylindrical metal fuel slug, sodium bond, gas plenum, and cladding material. The gap between fuel slug and cladding is also filled by sodium. The fuel life time behavior and its failure probability from BOL (Beginning-Of-Life) to EOL (End-Of-Life) is evaluated by the MACSIS code[5], which predicts the thermal performance and dimensional characteristics of metal fuel pins under normal operating conditions. The performance characteristics of the metal fuel rod in liquid metal reactors include fuel swelling due to the accumulation of fission products, fission gas release and the buildup of rod internal gas pressure, constituent redistribution of the fuel alloying elements, fuel cladding chemical interaction, and thermo-mechanical cladding integrity under a fast neutron environment.

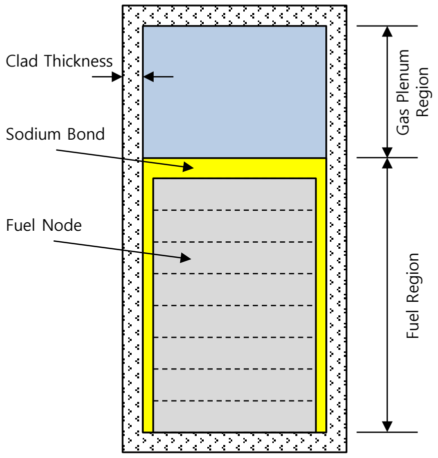


Fig. 2. Fuel pin performance model.

3. Coupling Methods and Results

The PGSFR core employs metallic fuel rods. The SAFDLs are typically cumulative damage fraction (CDF), inelastic strain and hoop stress. To evaluate these SAFDLs for all fuel rods, detailed temporal and spatial data are provided from neutronics and core thermal-hydraulics.

The core is composed of 112 fuel assemblies. They are split into an inner core and an outer region. Even though the fuel assemblies with same enrichment are loaded, a different fuel management strategy is applied by 4/5-batch scheme at the inner/outer core, respectively. Therefore, the coupling analysis for each fuel assembly involves different life cycle history from BOL to EOL. Figure 3 shows an example of the maximum cladding temperature variation in a central assembly (4-batch) during an equilibrium life time. Since a fresh fuel assembly contains the largest fissile material and it

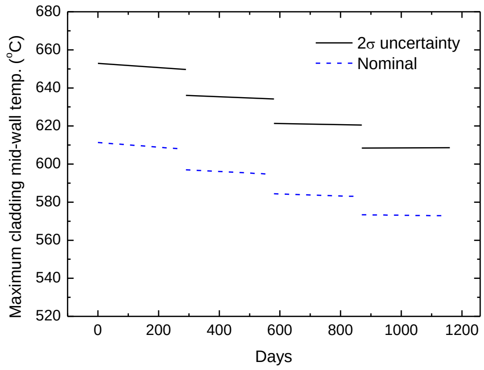
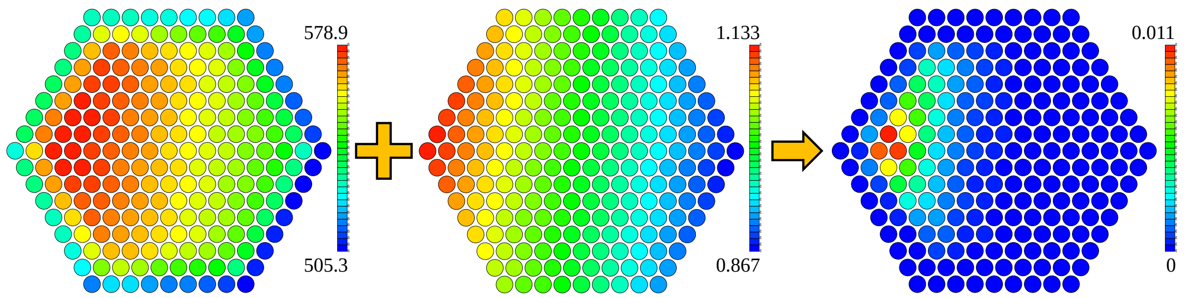


Fig. 3. Temperature variation from BOL to EOL in the core center assembly.

burns out with time, the temperature decreases by cumulative cycle length. In a cycle, temperatures change is determined by a relative position about the primary control assemblies.

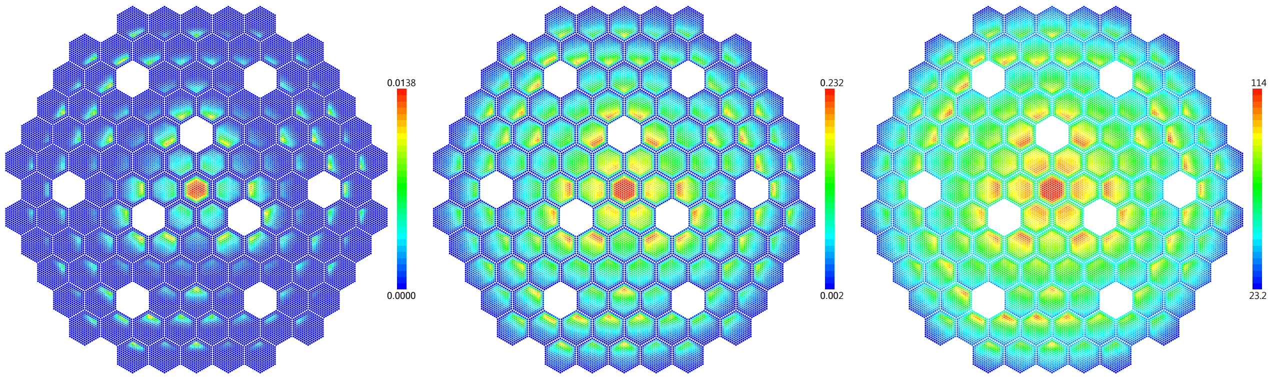
Figure 4 exhibits a spatial coupling example in an outer fuel assembly with 217 fuel rods. Pin thermal power increase as close to the core center. Fuel rods with high thermal power are located near a subassembly wall. However, a subchannel flow rate in the wall region is larger than that in the central region. Thus, the cladding temperature also reduces near the wall. As a result, the maximum temperature and the maximum thermal power are located at different rods in the subassembly. As fuel rods approach the cladding failure limit, the CDF is sensitive to both temperature and power. Therefore, the CDF shows a drastic change as shown in Fig. 4-(c). A typical 1-D analysis employs a conservative calculation using the maximum temperature and power from an assembly. The maximum CDF in the present method is 0.0110, which is less than 0.0144 of the typical 1-D analysis.

The coupled analysis is utilized to evaluate the CDF, inelastic strain and hoop stress for all fuel rods as shown in Fig. 5. To evaluate a life time performance from BOL to EOL, a different cycle pattern is applied for each fuel assembly during the equilibrium core transition. The fuel performance analysis is repeated in a total of 24,304 rods (217x112). The CDF reveals the most rapid spatial change over the whole core. On the other hand, the hoop stress is less sensitive to core location. The maximum value of each parameter satisfies the design



(a) Cladding temperature (b) Thermal power peaking (c) Cumulative damage fraction

Fig. 4. Coupling example of cladding temperature, thermal power and cumulative damage fraction within an outer fuel assembly.



(a) Cumulative damage fraction (b) Inelastic strain (c) Hoop stress

Fig. 5. Fuel performance evaluation over the whole core (112 assemblies/24,304 rods).

criteria as shown in Table I[6]. Although thermal power in the core center region is higher than that of exterior region, the maximum temperature is equalized over the entire core by adjusting flow rates of each fuel assembly. Therefore, the maximum value in each assembly is similar. However, the center assemblies reveal relatively flat distributions, while the exterior regions increase spatial peaking factors. Thus, severe fuel rods are concentered in the core center region.

Table I: Maximum fuel design limits

	Analysis	Limit
CDF	0.0138	<0.05
Inelastic strain (%)	0.232	<1
Hoop stress (MPa)	114	<150

All CDF scattering as a function of cladding temperature is displayed with a histogram in Fig. 6. Since the CDF change abruptly as shown in Fig. 4 and 5, it is represented in the logarithmic scale. The ratio between the maximum and minimum is almost 10^8 in the single core. The CDF and temperature is highly correlated in the entire scattering. The cladding temperature is obviously one of the most important factors in the CDF evaluation.

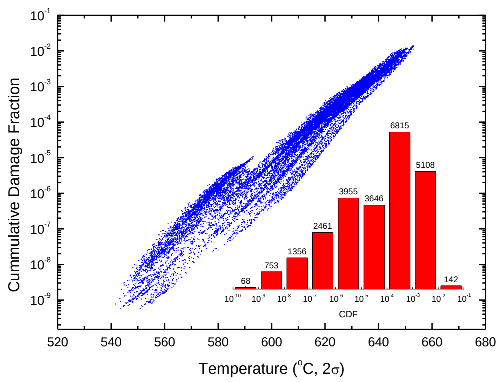


Fig. 6. CDF distribution in the equilibrium core.

4. Conclusions

A new coupled analysis of neutronics, core thermal-hydraulics and fuel performance has been developed to predict a thermo-mechanical failure of whole core metallic fuel rods for a sodium-cooled fast reactor. This analysis reveals the improved computational accuracy over the previous methods. It is evident that the present method is able to increase the plant performance keeping a safety margin compared to the simple conservative 1-dimensional analysis.

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