*Transactions of the Korean Nuclear Society Autumn Meeting*
*Gyeongju, Korea, October 25-26, 2012*

# Coupled Analysis of Core Thermal-Hydraulics and Fuel Performance to Evaluate a Thermally-Induced Fuel Failure in an SFR subassembly

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## 1. Introduction

A limiting factor analysis in a core thermal design is highly important to assure the safe and reliable operation of a reactor system. In a sodium-cooled fast reactor (SFR), the coolant thermal conductivity is about hundreds of times larger than the thermal conductivity of water. Moreover, the coolant boiling temperature in an SFR is around 900°C, which is much higher than that of the water coolant in a PWR. Considering typical operating temperatures, an SFR has about a 300 °C thermal margin to its boiling point. Therefore, instead of DNBR (Departure from Nucleate Boiling Ratio) in a PWR, the core thermal design of SFRs requires assuring proper fuel performance and safety, where the design limits are highly related to the temperature distribution and material behavior under various operating conditions. Typical limiting factors in SFRs are the thermal component of the plastic hoop strain, radial primary hoop stress, and cumulative damage factor during normal operation. However, the previous fuel performance codes only evaluate a single fuel pin performance, which neglects the radial peaking factors and reveals too conservative results. In this work, the multi-physics analysis is performed using both thermal-hydraulic and fuel performance codes.

## 2. Analysis Codes

### *2.1 Thermal-Hydraulic Analysis*

MATRA-LMR (Multichannel Analyzer for Transient and steady-state in Rod Array-Liquid Metal Reactor) was developed specifically for an LMR analysis based on the MATRA code, which was originally developed for a water-cooled reactor based on a subchannel analysis method. For a thermal-hydraulic analysis of a core consisting of subassemblies with a subchannel of a wire-wrapped rod bundle, a subchannel analysis is widely used. 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. Thus, a simplified model can be applied to the transverse momentum equations. 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. Since the pressure drop between the inlet and outlet ports along a subassembly must be constant, subchannels with a small area represent a high flow velocity. Each flow area across a subchannel is also dependent on the axial position of the wire-wraps.

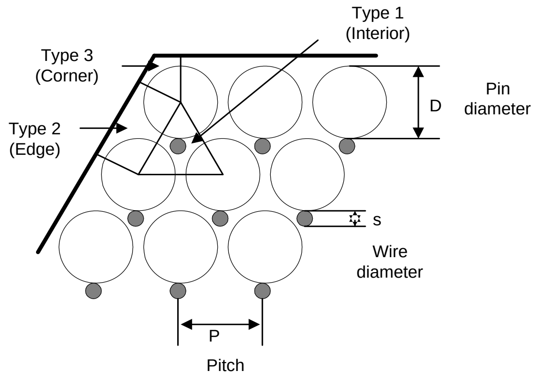


Fig. 1. Thermal-hydraulic subchannel model.

### *2.2 Fuel Performance Analysis*

MACSIS is a computer code for the thermal performance and dimensional characteristics of metal fuel pins under normal operating conditions of a Liquid Metal Cooled Fast Reactor. It is necessary to have a method to accurately assess the thermal performance of a metal fuel pin in a fast neutron environment. The MACSIS computer program was developed as a design tool for a metallic fuel rod.

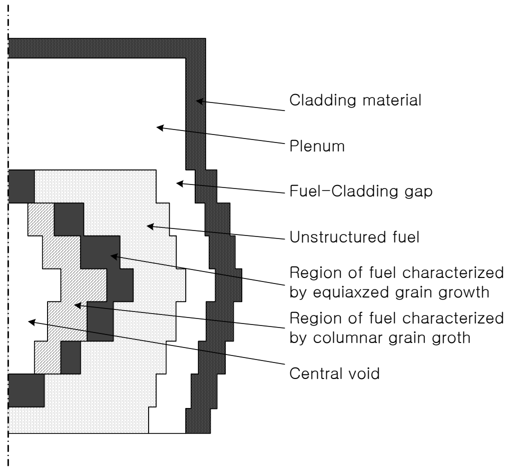


Fig. 2. Fuel pin performance model.

Table I: Comparison of the MATRA-LMR and MACSIS codes

| | MATRA-LMR code | | MACSIS code |
|---|---|---|---|
| Coolant temperature calculation | - 3-dimentional momentum and heat transport | > | - Simple 1-dimentional heat conduction |
| Fuel/cladding temperature calculation | - Single thermal property given by a input list | < | - Temperature/composition dependent thermal property based on experimental correlations |
| Power /neutron distribution | - Radial peaking within a subassembly | > | - Conservative single value |
| Material behavior | - No model included | < | - Deformation and failure model included |

As shown in Fig. 2, the fuel pin is modeled. It computes the one-dimensional temperature distribution and the thermo-mechanical characteristics of a fuel rod under the steady state operation condition, including the swelling and rod deformation. The amounts of fission gas retention and the release during the irradiation of the fuel are also computed. The thermal expansion and gas pressure inside the fuel element are then used to compute the stresses and strains in the cladding. The CDF (Cumulative damage factor) is computed.

MACSIS is constructed from a series of modules with a single set of dimensional units used throughout to provide flexibility in model usage and ease of upgrading as the models developed from future tests are finalized. A radial steady state heat transfer can be computed for 21 axial segments. The code computes all major quantities which affect in-reactor performances of the fuel rod, such as the temperature profile, fission gas generation and retention, fission gas release, swelling, and deformation.

### *2.3Results and Discussions*

Table I summaries a comparison of the MATRA-LMR and MACSIS codes. The MATRA-LMR code reveals a superior performance in the coolant temperature calculation owing to the 3-dimensional momentum transport consideration. It also has an advantage to reflect the radial peaking effects of the power and neutron distributions. However, the thermal properties of the cladding and fuel are only given by an input list. On the other hand, the MACSIS code employs various experimental correlations for the thermal properties, which are highly dependent on the material composition and temperature. Moreover, the MACSIS is capable of evaluating the fuel performance such as fission gas behavior, fuel/cladding displacement, and a cladding rupture analysis under various operating conditions. Accordingly, the present works utilized the MATRA-LMR code for the coolant temperature and radial peaking effects and the MACSIS code for cladding/fuel temperature and material behavior.

The coupled thermal performance analysis was conducted for the demonstration SFR [3]. The hottest 271 pin subassembly located in the central core region was employed, and the results of which are exhibited in Fig. 3 where the subchannel and pin colors correspond to the coolant temperature and pin thermal strain, respectively.

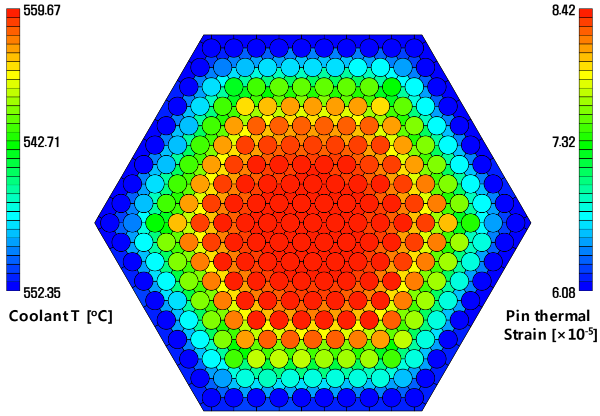


Fig. 3. Coolant temperature and pin thermal strain distribution in a subassembly.

Since metal ductility generally increases with temperature, the pin thermal strain shows a similar behavior compared to the coolant temperature. It is noted that the possibility of a thermally-induced pin failure is apparently dependent on the fuel pin position within a subassembly.

## 3. Conclusions

A new coupled analysis of the core thermal-hydraulics and fuel performance has been developed to predict a thermally-induced failure of metallic fuels for a sodium-cooled fast reactor. It adopts the advantages of each code, such as the whole pin analysis in a subassembly, radial peaking reflection, and various experimental correlations for thermal properties. It is evident that the present method is able to increase the plant performance keeping the same safety margin compared to a simple conservative analysis.

## REFERENCES

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[3] Y. I. Kim, et. Al., Conceptual Design Report of SFR Demonstration Reactor of 600MWe Capacity, KAERI/TR-4598, 2012.