Unprotected Transient Overpower Accident in a Sodium-Cooled Fast Reactor Using a Three-dimensional Core Thermal-hydraulic Calculation

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1. Introduction

Korea Atomic Energy Research Institute (KAERI) and Argonne National Laboratory (ANL) have developed a detailed three dimensional thermal hydraulics reactor core model. The new model was implemented in ANL and KAERI computer codes for a system-wide safety analysis. The objective of the model was to provide a high-accuracy capability to predict fuel, cladding, coolant, and structural temperatures in reactor fuel subassemblies, and thereby reduce the uncertainties associated with lower fidelity models previously used for safety and design analyses. This approach is especially important for new reactors that have not yet been built and for existing reactors in conditions outside the range in which temperature and reactivity measurements have been made.

2. Three-dimensional Fuel Subassembly Thermal-hydraulic Model

The new model uses a coolant sub-channel treatment similar to that used in COBRA. A formulation of governing equations with variable descriptions is not explained in this paper. The detailed explanation is available in reference [1]. There is an option to use one channel for each coolant subchannel or to combine subchannels where one channel represents a row of subchannels for one sector. The advantages of the later representation are that it uses significantly fewer channels, and requires considerably less computing time and less storage.

The subchannel treatment includes an axial coolant flow parallel to the pins and a cross flow between sub-channels driven by pressure differences and a wire wrap sweeping. Both forced convection and buoyancy-driven natural circulation are handled. There is an option to conserve computer time and reduce computer memory requirements by treating some subassemblies with a detailed sub-channel treatment while other subassemblies are treated in less detail.

The model was specifically designed for integration into the ANL SASSYS-1 [2] code as well as the KAERI SSC-K [3] code. Although there are some differences between the SASSYS-1 and SSC-K features, the thermal hydraulic model was developed to satisfy an interface with both codes. The main frame of the whole core thermal hydraulic model developed by ANL was linked to the SSC-K primary loop model for steady state and transient calculations [4].

The basic conservations equations used include continuity, momentum and energy equations for the coolant. An energy equation is used for the fuel pin. In addition, equations were determined for calculating the pressure driven and the wire-wrap sweeping cross-flow between adjacent coolant subchannels and for calculating the heat flow between adjacent subchannels due to a turbulent mixing and a thermal conduction. Cylindrical geometry with radial heat conduction is used for calculating the fuel pin temperatures.

One of the critical parameters for determining the thermal-hydraulic behavior of a coolant in the subchannels is the heat source from a turbulent mixing and conduction from adjacent subchannels. The CFX code calculations have been performed by KAERI and correlations were suggested to account for a turbulent mixing factor and local heat flux effects in the gap between two subchannels.

A model verification was performed by SASSYS-1 at ANL with an execution of a simulation of the Shutdown Heat Removal Test 17 (SHRT-17) performed in the EBR-II reactor. SHRT-17 was a simulation of a protected loss-of-flow accident. Transient peak coolant temperatures in the instrumented subassembly are shown in Fig. 1. The calculated and measured coolant temperatures are almost identical during the pump coast-down phase of the transient. Later in the transient the calculated and measured temperatures still agree well, with each other.

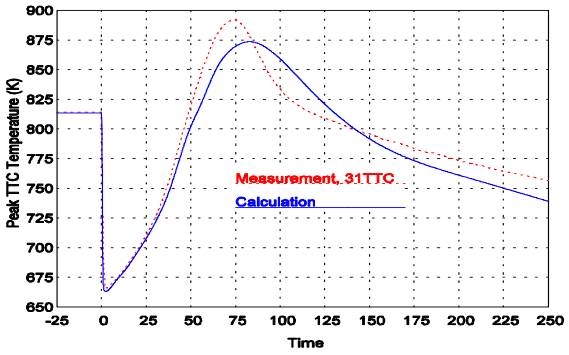

Fig. 1 Transient Peak Coolant Temperatures in the Instrumented Subassembly of EBR II

3. UTOP Analysis

(a) Analysis Method

For the KALIMER-150 UTOP analysis with this model, the whole core was represented by its 1/6 segment, as shown in Fig. 2, based on a symmetric

SA No.	Channel No.	No
1	1 ~ 42	6
2	43 ~ 102	6
3	103 ~ 162	6
4	163 ~ 204	12
5	205 ~ 264	6
6	265 ~ 324	12
7	325 ~ 366	6
8	367 ~ 426	12
9	437 ~ 486	12
10	487 ~ 528	6
11	529~ 570	12
12	571 ~ 612	12
13	613 ~ 654	6
14	655 ~ 696	12

Total core channel number = 6264

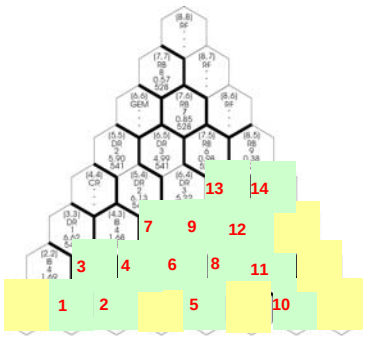

Fig. 2 KALIMER-150 Core Modeling

configuration. Subassemblies with a similar initial power, flow, and irradiation exposure, have been grouped into a total of 14 subassembly types. There are 6 subassembly types, each with 60 channels, for the driver fuel. There are 3 subassembly types, each with 42 channels, for the internal blanket fuel. And, there are 5 subassembly types, each with 42 channels, for the radial blanket fuel. Consequently, there are a total of 696 channels in the whole core subchannel model for the 126 subassemblies in the KALIMER-150 reactor. For the 271 pin driver fuel subassembly model, 60 coolant channels were used to allocate a unique coolant channel to each row of pins in each 60° sector of the subassembly. 39 axial segments including 20 for the active core region were employed in the channel models.

For the UTOP sequence, it is assumed that a control system failure results in an inadvertent withdrawal of a control rod, and the plant scram systems also fail. All other systems, including the coolant pumps and balance-of-plant, are assumed to continue a normal operation. The control rod withdrawal is assumed to insert 0.3 β of reactivity linearly in 15 seconds of time, where β is the effective delayed neutron fraction.

(b) Analysis Results

The reactor power transients calculated by SSC-K and SASSYS-1 are shown in Fig. 3. The observed discrepancy in the power is mainly the result of different rates of reactivity change during the early transient. The reactor power peaks at about 150% and then slowly decreases to seek an equilibrium with the available heat sink provided by the coolant system heat capacity and the heat rejection by the steam generators.

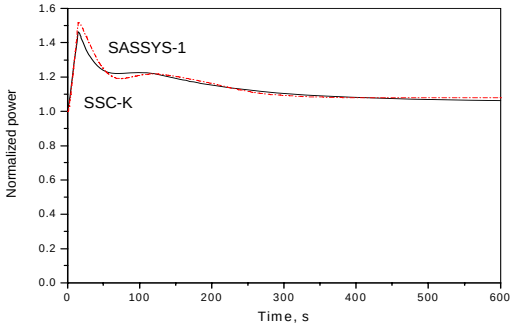

Fig. 3 Power Comparison

The power rise is driven by the assumed control rod withdrawal, and the reactor responds with negative reactivity feedbacks triggered by the elevated fuel and coolant temperatures. The reduced negative feedbacks ultimately lower the reactor power to an eventual equilibrium with the heat removal.

Figure 4 depicts the reactivity components. A close examination of the component reactivities indicates that the difference between the two codes is due to a difference in the core radial expansion reactivity and the CRDL reactivity during the entire transient. The reactivity insertion leads to a power increase, which raises the fuel, coolant, and structural temperatures. These temperature increases cause reactivity feedbacks due to the fuel Doppler effect, fuel and cladding axial thermal expansions, a coolant density decrease, a radial core dilation by a structural thermal expansion at the above-core load pad plane, and a thermal expansion of the CRDL. The net reactivity, which is the sum of the assumed reactivity insertion and the feedbacks, rises initially with the inserted reactivity, but soon peaks and falls as the negative feedbacks counter the only positive feedback from the coolant density reduction. The net reactivity eventually decreases to near zero, and in the long term, begins a slow, low-amplitude, negative-to-positive oscillation as the reactor adjusts to the heat rejection provided by the steam generators.

4. Conclusion

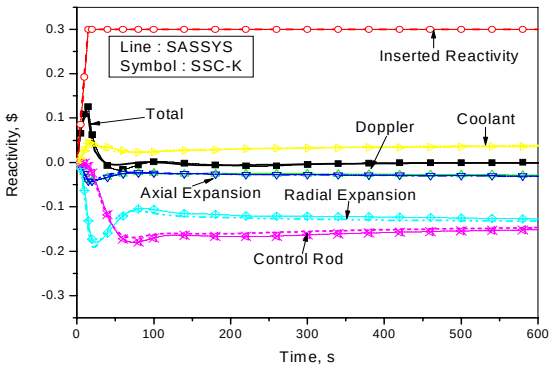

Fig. 4 Reactivity Components

In summary, it has been found that an inclusion of a detailed whole core sub-channel model in the system-wide safety analysis code, SSC-K, increases the accuracy and decreases the uncertainties in the predictions of the reactor safety margins in accident situations.

REFERENCES

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