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A three-dimensional pin-wise analysis for CEA ejection accident

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1. Introduction

The ejection of a control element assembly (CEA) with high reactivity worth causes the sudden insertion of reactivity into the core. Immediately after the CEA ejection, the nuclear power of the reactor dramatically increases in an exponential behavior until the doppler effect becomes important and turns the reactivity balance and power down to lower levels. Although this happens in a very short period of time, only few seconds, the energy generated can result in significant fuel failures.

The 3-D CEA ejection analysis methodology has been developed using the multi-dimensional code coupling system, CHASER [1], which couples three-dimensional core neutron kinetics code ASTRA, sub-channel analysis code THALES, and fuel performance analysis code FROST using message passing interface (MPI).

This paper presents the pin-by-pin level analysis result with the 3-D CEA ejection analysis methodology using the CHASER. The pin-by-pin level analysis consists of DNBR, enthalpy and Pellet/Clad Mechanical Interaction (PCMI) analysis. All the evaluations are simulated for APR1400 plant loaded with PLUS7 fuel.

2. Methods and Results

The 3-D CEA ejection analysis methodology can be separated into two stages. The purpose of the first stage is to calculate the behavior of core average transient and to generate the power history for all the rods. In the second stage, the pin-by-pin level analysis using the power history is performed to calculate the enthalpy and DNBR values.

2.1 Core Average Transient Analysis Scheme

The Core average transient analysis is accomplished by CHASER using 1/4 assembly node structure. The coupling scheme of CHASER between the kinetics and thermal hydraulic parameters is presented in Fig. 1. The nuclear power calculated by ASTRA, is transferred to FROST via CHASER. FROST calculates the fuel rod temperature and the heat flux using the coolant temperature and heat transfer coefficient transferred from THALES. Thermal-hydraulic data, i.e. effective fuel average temperature, reactor coolant temperature and density related to the effect of reactivity feedback in a core, are passed to ASTRA, and then ASTRA calculates nuclear power considering doppler and moderator feedbacks. The data transfer between codes is performed repeatedly until the heat flux is converged within a criterion.

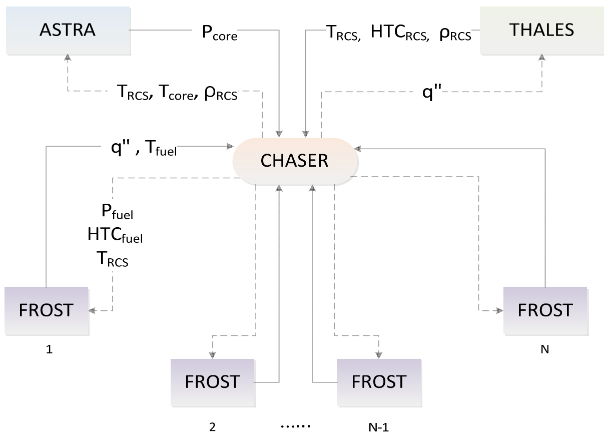

Fig. 1. Coupling scheme of CHASER

The core average transient analysis is already validated by simulating the reference reactivity initiated accident benchmarks, NEACRP 3-D PWR core transient problem [1]. And the sensitivity studies with the kinetics parameters and the other major parameters also have been conducted [2, 3]. As per the core average transient analysis result, the power history for all rods is obtained.

2.2 Pin-by-Pin Analysis Scheme

The pin-by-pin analysis can be conducted using the core power history which is generated from the core average transient analysis.

Fig.2 and Fig.3 show the rods and sub-channel modeling for pin-by-pin level analysis. As shown in Fig.2, the detailed pin-wise modeling is applied for the target node (center) and lumped 1/4 assembly nodes are used in the neighboring nodes. Fig.3 shows that the other assemblies around the detailed nodes are modeled with one node per assembly. In conclusion, the rods and sub-channel modeling simulates the full core as 340 nodes and 327 sub-channels.

The pin-by-pin level analysis can be performed only for the user specified target node. Thus, the pin-by-pin level analysis should be repeated by changing the location of the target node to consider all the interested regions in the core.

2.3 DNBR and Enthalpy Analysis Result

In order to determine the limiting conditions with respect to enthalpy and DNBR, sensitivity studies using a core average transient analysis scheme are performed for various input parameters such as core power, ejected CEA worth, ejected CEA position, kinetics parameters, axial power shape, MTC, FTC, core inlet temperature, core pressure and core inlet mass flowrate. For the limiting cases selected with respect to enthalpy and DNBR, the pin-by-pin analysis scheme is used to calculate pin enthalpies and DNBRs.

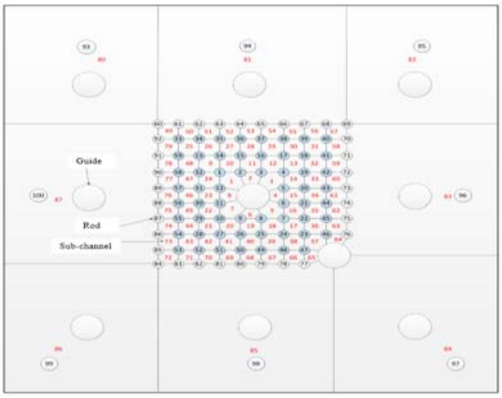

Fig. 2. Rods and sub-channel modeling for pin-by-pin analysis (Detailed node)

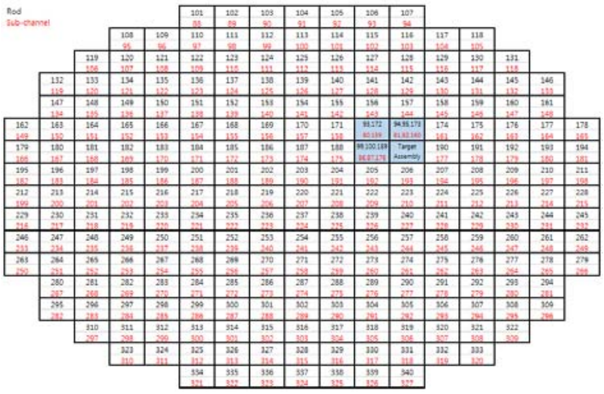

Fig. 3. Rods and sub-channel modeling for pin-by-pin analysis (Lumped part)

In HFP case, pin-by-pin analysis is conducted for all the nodes in the quarter core. Fig. 4 shows the enthalpy result of HFP case. For part power cases (65%, 50%, 20% power levels), most severe 15 assemblies are selected for pin-by-pin analysis. And for HZP case, 20 assemblies are selected and calculated for pin-by-pin analysis. Fig. 5 shows the maximum enthalpy result for HZP case. Fig.4 and Fig.5 show that the maximum enthalpy rise value which represents the most severe fuel pin result is under 110 cal/g and the criteria is satisfied.

DNBR case is also conducted same method of enthalpy calculations. In part powers and HZP, lowest DNBR value is over the DNBR. And in HFP case, some of pins violate the DNBR limit but the number of violating pins are less than 2% of total pins.

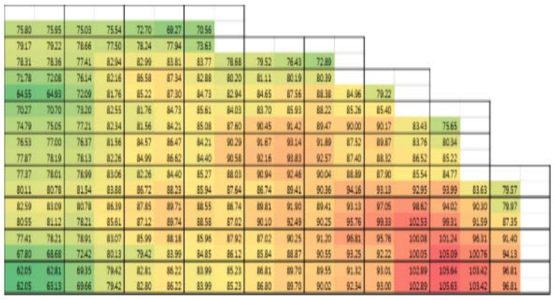

Fig. 4. Maximum enthalpy for HFP case (cal/g)

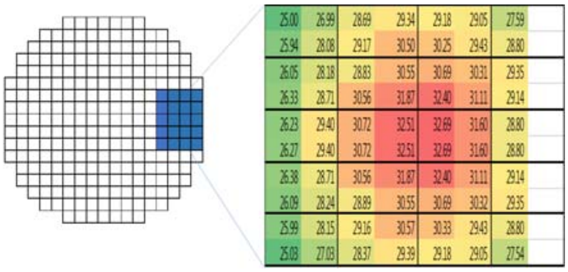

Fig. 5. Maximum enthalpy for HZP case (cal/g)

2.4 PCMI Analysis Result

When the reactivity increases instantaneously, the pellet may expand momently due to the rapidly stored energy and PCMI can occurs. NRC published the fuel clad failure criteria due to the PCMI [4] as the pellet radially averaged enthalpy rise versus the ratio of oxide layer thickness. And KEPCO NF has recently developed the PCMI criteria as a function of burnup instead of the ratio of oxide layer thickness.

From a standpoint of PCMI criteria, HZP shows the most severe power transient, because the core power increased much higher in a short time than the other power levels. Thus only HZP case is conducted for PCMI analysis.

To calculate the maximum enthalpy rise, power histories of all the rods in the core is used and the adiabatic process is assumed for the conservatism.

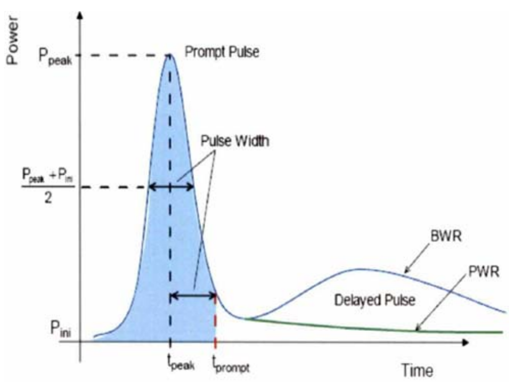

Fig. 6. Definition of prompt pulse width time

The enthalpy rise is defined as the accumulated enthalpy in a rod until the end of the prompt pulse width time. The prompt pulse width time is defined as Fig. 6. The prompt pulse width time for HZP case is calculated as 0.3 seconds, but 2.0 seconds is used as prompt pulse width time for the conservatism in PCMI analysis.

Fig. 7 shows the enthalpy rise result with three fuel types (once-burned, twice-burned, three times-burned) and enthalpy rise criteria versus fuel burn-up (red line). It is observed that the maximum enthalpy rise value is under the enthalpy rise limit. And maximum enthalpy rise appeared at the once-burned fuel.

Fig. 8 shows the distribution of enthalpy rise for all of rods graphically. The black-boxed assembly is ejected CEA position and the assemblies near the ejected CEA position have more increased enthalpy rise values than the opposite assemblies with green color. And the maximum enthalpy rise appeared at the once-burned fuel.

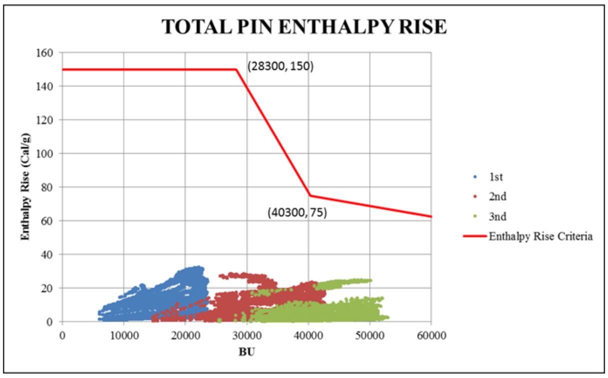

Fig. 7. Enthalpy rise criteria and enthalpy rise

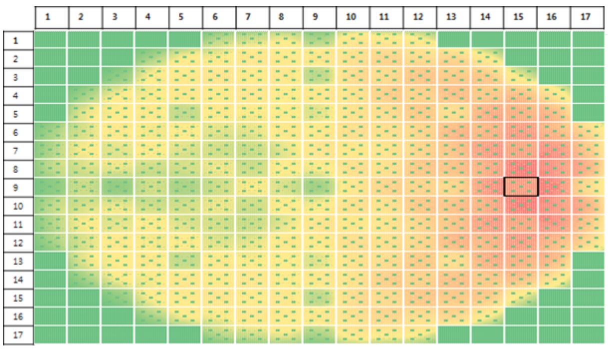

Fig. 8. Enthalpy rise for all of rods (cal/g)
[Green:0~3, Yellow:3~11, Orange:11~19, Red:19~33]

3. Conclusions

In this paper, the pin-by-pin analysis using the multi-dimensional core transient code, CHASER, is presented with respect to enthalpy, DNBR and PCMI for APR1400 plant loaded with PLUS7 fuel.

For the pin-by-pin enthalpy and DNBR analysis, the quarter core for HFP case or 15 ~ 20 assemblies around the most severe assembly for part powers or HZP cases are selected. And PCMI calculation is performed for all the rods in the whole core during a conservative time period.

The pin-by-pin analysis results show that the regulatory guidelines of CEA ejection accident are satisfied.

REFERENCES

[1] J. W. Park et al., Code Coupling for Multi-Dimensional Core Transient Analysis, Transaction of the Korean Nuclear Society Spring Meeting, 2015.

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[4] NRC, Standard Review Plan Rev. 3, NUREG-0800, 2007.