Proceedings of the Korean Nuclear Society Spring Meeting
Cheju, Korea, May 2001

Application of A Best-Estimated Kinetics Code to RIA Evaulations

Hideaki Ikeda
Toden Software, Inc.
6-19-15 Shinbashi, Minato-ku
Tokyo 105-0004, Japan

Takafumi Anegawa
Tokyo Electric Power, Co.
1-1-3 Uchisaiwai-cho, Chiyoda-ku
Tokyo 100-0011, Japan

Abstract

A best-estimated kinetics code, SIMULATE-K, was applied to numerical simulations for a control rod drop accident (CRDA) in a BWR core, to ensure the basic fuel behavior under the reactivity-initiated accident (RIA) events, and the sensitivities of the analysis models. The recent revise in terms of the fuel cladding failure thresholds accepted for the safety evaluation in Japan, are taken into account in the present analyses. The results show that the fuel cladding failure can occur in highly irradiated and low-reactive fuels near the dropped control rod. The sensitivity study indicates that the coolant density feedback reactivity plays an important role to suppress the power excursion. It should be noted that a fine thermohydraulic nodalization in the numerical representation for core coolant hydraulics behavior, more than 100 multi-channels for instance, is indispensable for accurate modeling of strongly space-dependent reactivity behavior around the dropped control rod.

1. Introduction

Application of the three-dimensional neutron model for the evaluation of the LWR core dynamic behavior is one of the recent topics for the reactor physics methods in Japan. Such an advanced model coupled with the detailed thermohydraulics (T-H) modules is frequently used in analyses for neutron flux oscillations occurred in BWR cores[1,2], and also for hypothetical RIAs in PWR/BWR cores.[3,4]

The recent experimental data in terms of the simulated RIA show that the safety threshold for the mechanical fuel cladding failure strongly depends on the burnup exposure.[5,6] Furthermore, the RIA occurring in a large LWR core shows strong space-dependency from the viewpoints of neutronic and thermohydraulic behaviors. These situations motivate an application of a three-dimensional code to accurately model the core exposure distribution for the RIA safety evaluations. The present article presents a recent activity on the RIA analyses, particularly the CRDA analyses in BWR cores, using a best-estimated kinetics code, the kinetics version of SIMULATE (SIMULATE-K).[7]

2. Analysis Models

SIMULATE-K solves the three-dimensional two-group neutron diffusion equations with six-group delayed neutron precursor by the advanced nodal method. Fuel heat conduction and transfer to coolant is modeled by the radial one-dimensional thermal diffusion in each neutronic calculation node. Two-phase coolant hydraulic behavior is simulated by the axially one-dimensional five-equation model in the individual fuel assemblies. Subcooled boiling phenomenon observed under the power excursion can be accounted by using the Saha-Zuber correlation and the Lahey mechanistic model. In addition to the original SIMULATE-K, the present code has a capability of numerically representing the core hydraulics by the collapsed multi-channels in order to obtain computational efficiency further.[1]

Dynamic behavior of the pin-wise fuel enthalpy is computed using the local power that is obtained by synthesis between the assembly-homogenized nodal power and the predetermined intra-assembly relative pin power distribution. The relative pin power and pin burnup exposure distributions are specified based on the representative fuel assemblies for each depleted fuel cycle.

The neutronic cross section and reactivity feedback models are fully consistent with the steady-state version of SIMULATE,[8] accounting for the coolant density and Doppler feedbacks, although the application of the coolant density feedback is not accepted for the current safety evaluation of RIA in Japan.

3. Analysis Results

The aforementioned code was applied to the CRDA analyses in a GE BWR5 core at the EOC statepoint under the cold and hot-standby (HSB) conditions. The reactivity worth of the dropped control rod was specified to be 1.3 $\%\Delta k$ by adjusting the macroscopic fission cross sections of fuel assemblies adjacent to the dropped control rod. The radial positions of the dropped control rod are shown in Fig. 1, and Table 1 summarizes the base analysis condition. The applied analysis condition mostly coincides with that accepted for the current safety evaluation in Japan, with the exception of coolant density feedback.

Fig. 2 contains the time-traces of the maximum fuel enthalpy obtained in the base CRDA evaluations with the above analysis condition. We find that increase of the fuel enthalpy is effectively suppressed by the negative coolant density feedback after the power excursion, particularly in the HSB CRDA due to the lower subcooled inlet coolant condition. In the present analysis, the number of the fuel cladding failure was evaluated based on the two mechanisms, the PCMI and the high-temperature rupture. The thresholds used in the Japanese safety evaluation have been newly revised accounting the recent experimental data, which were mostly obtained in the JAERI/NSRR facility.[6] Fig. 3 shows the threshold lines for the failure evaluation and the analysis results in terms of the correlation between fuel exposure and enthalpy. Because the failure thresholds are strongly dependent on the fuel exposure, the analysis predicts that the failure occurs even in the fuels showing low enthalpy increase.

Table 2 shows calculation results for the sensitivity of the analysis conditions for the CRDA evaluation. The calculation results are found to be extremely sensitive to the treatment of the coolant density feedback. This indicates that the conservatism obtained by neglecting the coolant density feedback is quite large.

In addition, we ensured the sensitivity of the T-H numerical nodalization, and these results are shown in Table 3. Here, the number of the fuel-assembly-rings, where the T-H channel/assembly nodalization was applied around the dropped control rod, was increased from 1 to the maximum. The channels of assemblies which were not individually represented, were appropriately collapsed to several groups. It can be found that the T-H nodalizaion has large impact on the RIA evaluation. This means that the CRDA analysis is quite sensitive to the local thermohydraulic performance, and that more than 100 T-H numerical channels is required for accurate modeling of reactivity behavior under the CRDA in a typical equilibrium core.

4. Summaries

The best-estimated kinetics code was applied to the RIA (CRDA in a BWR core) analyses. The results showed that the fuel cladding failure can occur with relatively small increase in fuel enthalpy in the highly irradiated fuels, due to the recent revision of threshold decrease in the higher burnup range. It was found that by considering the coolant density feedback, it effectively suppresses the fuel enthalpy increase, particularly in the HSB CRDA. This also showed that there exists a large conservatism by neglecting the feedback.

In addition, applying the best-estimated code to the RIA evaluations for the typical equilibrium LWR core, we have to pay attention to the T-H nodalization in the numerical model. Because the core consists of the various reloaded fuels having different characteristics in the thermohydraulic feedback reactivity, a coarser T-H nodalization leads to inaccurate calculations. In the CRDA analysis in a typical BWR core, more than 100 T-H channels are required in its numerical nodalization for accurate modeling of reactivity behavior around the dropped control rod.

References

1 Y. Takeuchi et al., *Nucl. Tech*, **105**, 162 (1994).
2 Y. Bessho et al., Nucl. Tech, **127**, 49 (1999).
3 H. Yamada, et al., 'Realistic Evaluation of Reactivity Insertion Accidents for a Typical BWR', *Proc. Int. Mtg. On Best-Estimate Methods in Nuclear Installation Analysis, BE-2000* (CD-ROM) (2000).
4 K. Matsumura et al., 'A Study on the Fuel Loading Effect in Rod Drop Accident Analysis using a Three Dimensional Transient Code', *Proc. 7th Int. Conf. on Nuclear Engineering, ICONE-7* (CD-ROM) (1999).
5 F. Schmitz, et al., 'Investigation of the Behavior of High Burn-up PWR Fuel under RIA Conditions in the CABRI Test Reactor', *22nd Water Reactor Safety Meeting* (1994).
6 T. Fuketa, et al., 'New Results from the NSRR Experiments with High Burnup Fuel', NUREG/CP-0149, Vol. 1, *23rd Water Reactor Safety Meeting* (1995).
7 J. A. Borkowski, et al., *Trans. Am. Nucl. Soc.*, **71**, 456 (1994).
8 SIMULATE-3 Methodology, STUDSVIK/SOA-92/02 (1992).

Table 1 Base Analysis Condition for CRDA evaluations

	Cold CRDA	Hot Standby CRDA
Initial Core Power (%Rated)	10^{-6}	10^{-4}
Inlet Coolant Flow (%Rated)	20	20
Inlet Coolant Subcooling (K)	92	5
Core Pressure (kg/cm^2a)	1.58	72.3
Initial Fuel Enthalpy (cal/g)	2	18

Table 2 Sensitivities of Analysis Models to CRDA evaluation Results

Analysis Conditions	Cold Core				Hot-Standby (HSB) Core			
	Peak Power (xRated)	Max Fuel Enthalpy (cal/g)	No. of Fuel Failures		Peak Power (xRated)	Max Fuel Enthalpy (cal/g)	No. of Fuel Failures	
			PCMI	High Temp.			PCMI	High Temp.
Base Case	4.1	119	25	42	8.9	106	49	39
No Void Feedback	4.1	+14%*	25	91	9.9	+97%*	141	889
Initial Power :Base x100	3.2	-4%	0	42	4.6	-24%	0	0
Coolant Flow :40%Rated	4.1	+4%	29	45	8.9	0%	49	41
Core Pressure :Base +1kg/cm^2	4.1	-4%	25	42	8.9	+2%	52	41
Inlet Subcooling :40(Cold)/0(HSB)K	4.7	-1%	73	49	7.5	-30%	0	0

* Difference from the base case.

Table 3 Sensitivities of T-H Nodalization to CRDA evaluation Results

No. of Fuel Assembly Rings applied 1-mesh/assembly T-H Nodalization	Cold Core					Hot-Standby (HSB) Core				
	Total No. of T-H Channels	Peak Power (xRated)	Max Fuel Enthalpy (cal/g)	No. of Fuel Failures		Total No. of T-H Channels	Peak Power (xRated)	Max Fuel Enthalpy (cal/g)	No. of Fuel Failures	
				PCMI	High Temp.				PCMI	High Temp.
∞*	764	4.0	117	25	42	764	8.8	105	45	37
5**	147	4.1	+2%***	25	42	158	8.9	+1%***	49	39
5	120	4.1	+2%	29	42	129	9.1	+2%	51	41
4	91	4.2	+3%	37	43	91	9.4	+4%	58	45
3	59	4.3	+8%	67	63	59	10.0	+8%	72	65
2	35	5.2	+28%	133	138	35	11.6	+16%	135	139
1	19	9.4	+148%	1106	120	19	16.1	+41%	282	203

* All the fuel assemblies are represented by 1-mesh/assembly T-H nodalization. Base case in the present table.

** Corresponding T-H nodalization used in the calculations shown in Table I.

*** Difference from the base case.

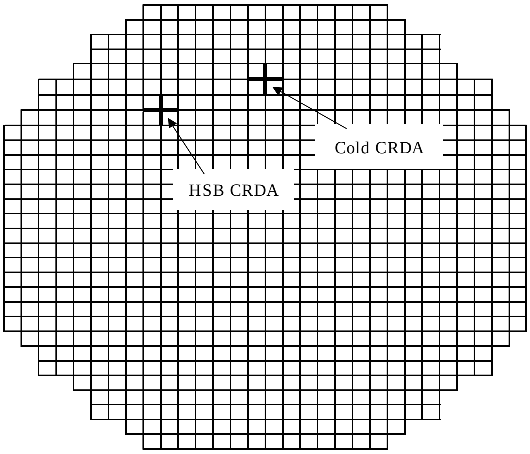

Figure 1 Dropped Control Rod Positions

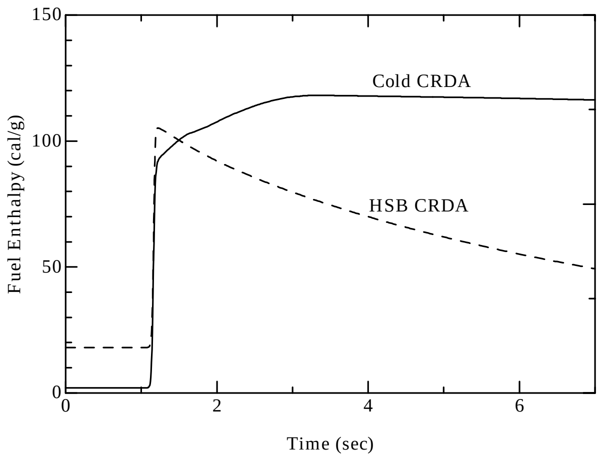

Figure 2 Time-Traces of Calculated Maximum Fuel Enthalpy

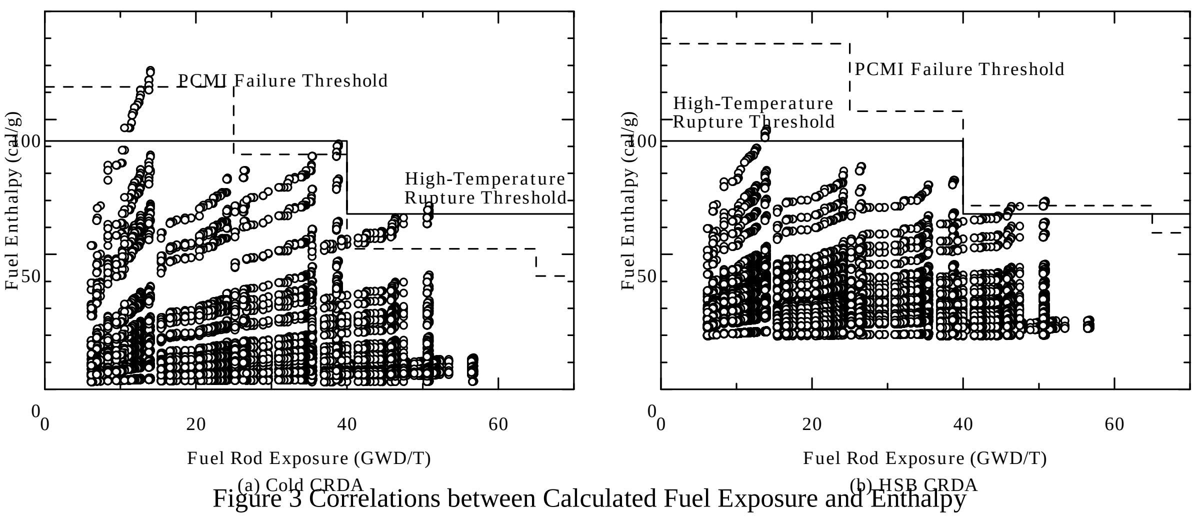

Figure 3 Correlations between Calculated Fuel Exposure and Enthalpy