Development of Audit Calculation Methodology for RIA Safety Analysis

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1. Introduction

Current acceptance criteria of reactivity initiated accidents(RIAs) used for licensing application were established in 1974 to limit the extent of fuel rod fragmentation based on the experimental data of SPERT and PBF tests with low burnup fuel[1]. But in the mid-1990s, it was discovered that the 170 cal/g criterion is not always adequate to protect the fuel rod failure, especially for the high burnup fuel, and failure criteria based on steady-state departure from nucleate boiling(DNB) or critical power ratio(CPR) may not be appropriate for fast transient. Thereafter, extensive experimental works for the establishment of new safety criteria have been conducted in CABRI, NSRR, BIGR[2]. Based on these data, interim acceptance criteria were established by U.S. NRC in 2007[3].

The interim criteria contain more stringent limits than previous ones. For example, pellet-to-cladding mechanical interaction(PCMI) was introduced as a new failure criteria. And both short-term (e.g. fuel-to coolant interaction, rod burst) and long-term(e.g., fuel rod ballooning, flow blockage) phenomena should be addressed for core coolability assurance. For dose calculations, transient-induced fission gas release has to be accounted additionally[3].

Traditionally, the approved RIA analysis methodologies for licensing application are developed based on conservative approach. But newly introduced safety criteria tend to reduce the margins to the criteria. Thereby, licensees are trying to improve the margins by utilizing a less conservative approach[4]. In this situation, to cope with this trend, a new audit calculation methodology needs to be developed. In this paper, the new methodology, which is currently under developing in KINS, was introduced. And preliminary calculation results and required further works for the establishment of the methodology were described as well.

2. Analysis Details

2.1 Utilized methodology and codes

Currently considered audit calculation methodology is based on the realistic approach. Realistic approach is a well-known methodology for LOCA safety analysis. Realistic approach is composed of the best-estimate codes that sufficiently describe the behaviors of power insertion and fuel behaviors during RIAs, and uncertainty analysis. For the uncertainty analysis, uncertainties in the analysis method and inputs must be identified, and uncertainty quantification has to be done with some appropriate statistical treatments.

Fig.1 shows a utilized code system for RIA analysis. For the generation of fuel power during fast transient, SCALE and PARCS code were used. Specifically, SCALE code packages were used for the generation of group constants, and kinetic nodal analysis with a control rod ejection accident(REA) were conducted by PARCS code in 3D core-wide perspective. Initial state of fuel rod and rod behaviors during transient were assessed by FRAPCON and FRAPTRAN code, respectively.

2.2 Information on the calculation

Preliminarily calculation on the REA of advanced power reactor(APR)1400 was carried out. In this calculation following methods and assumptions were made.

- Initial core of APR1400 in a state of hot zero power(HZP) was analyzed.
- Two group constants were utilized for kinetic nodal analysis, and these were generated by changing the fuel temperature, moderator temperature and density, boron concentration, and insertion state of control rod.
- For the 3D analysis, total 241 fuel assemblies in the core were simulated and the active core length of 3.81m was divided into 25 axial planes of equal length.
- Before the REA, all control rods were fully inserted. And ejection of single full-strength control rod located at the core periphery was assumed. Rod ejection was completed at the time of 1s after accident initiation.

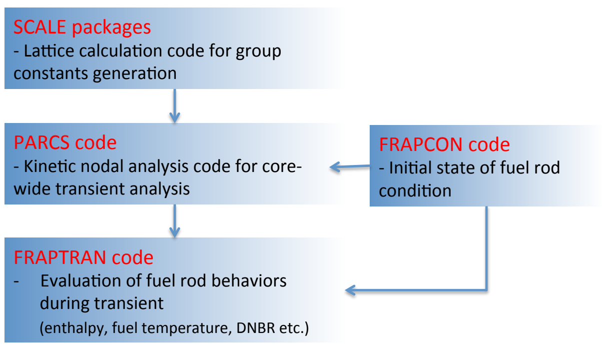

Fig. 1. Utilized computer code system for audit calculation of RIA safety analysis

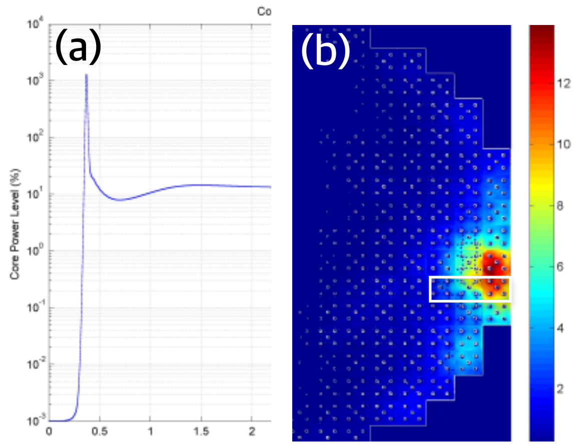

Fig. 2 (a) Evolution of core power and (b) power distribution in the half-core at the time of 0.37s after REA.

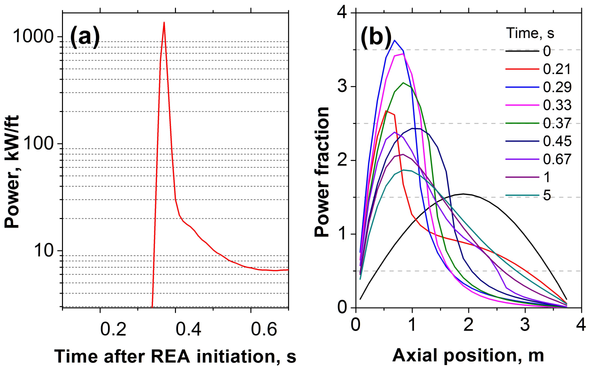

Fig. 3 (a) Evolution of the hottest pin power in the core, and (b) change of axial power shapes with time progression.

- Best-estimate fuel behaviors during transient were estimated based on the nominal conditions of fuel rod and thermal-hydraulic boundary conditions.
- During transient, every single fuel rod has used 25 different axial power shapes. Fuel rod was divided into 20 axial nodes with equal length.
- EPRI-1 critical heat flux correlation(default model in FRAPTRAN) was used for the assessment of DNBR.

3. Results and Discussion

3.1 Power evolution

3.1.1 Core power

Fig. 2(a) shows the core power evolution during REA. Core power was increased rapidly due to the insertion of positive reactivity with the control rod ejection, and it reached maximum at a time of 0.37s after accident initiation. Maximum power is about 127 times larger than the full power condition of the APR1400. After that, negative reactivity was inserted strongly due to the reactivity feedback, and it involved a sudden drop of core power. Power distribution in the half-core at the time of 0.37s is shown in the Fig. 2(b). This shows that,

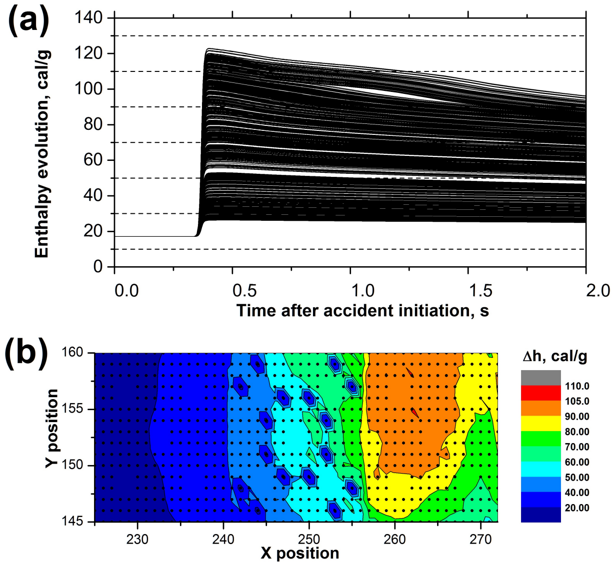

Fig. 4. (a) Evolution of fuel enthalpy and (b) distribution of peak enthalpy increase in the location of the core, indicated in Fig. 2(b).

as expected, the significant fuel power increase is limited at the local area of the core.

3.1.2 Pin power evolution

Fig.3(a) shows the most hottest pin power evolution in the core. As coincide with the core power transient, shown in Fig.2(a), maximum pin power of 1,361kW/ft(rod average) was developed at the time of 0.37s after the accident, and full width half maximum(FWHF) was about 20 ms. Fig.3(b) shows the change of axial power profiles as the time preceded. Before accident initiation, the power shape was like a cosine, but after the accident initiation, strongly bottom skewed peak power profiles were developed. The most highly bottom peaked power was observed at the time of 0.27s, and after that, peaking was rapidly relieved also, but significant bottom peaking was still observed after power transient was stabilized.

3.2 Fuel behaviors

3.2.1 Fuel enthalpy evolution

Fig. 4(a) shows the fuel enthalpy evolution curves at the selected area of the core, indicted in the Fig.2(b). Peak entalphies were observed at the time of 0.4~0.48 ms after the accident, and they were reduced gradually. Observed axial positions of peak enthalpy was mostly 0.95m away from the bottom. Maximum entalphy in the core was 123 cal/g. This is well below the one of the coolability criteria, 230 cal/g enthalpy limit. Fig.4(b) shows the distribution of maximum entalphy rise in the core. Several lower entalphy rises appeared in the mid-position of the figure are due to the presence of burnable absorber. Maximum entalphy rise was 105.9 cal/g, and and it was observed at the position of 262/155(x/y position). This shows a relatively sufficient margin to the interim PCMI failure criteria of fresh fuel, 150 cal/g.

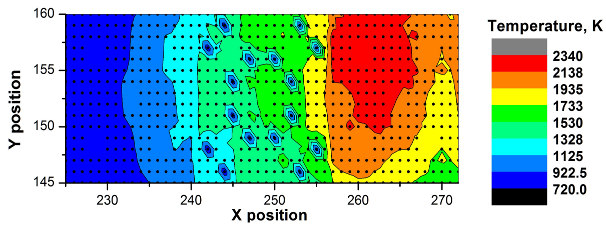

Fig. 5 Distribution of maximum fuel centerline temperature in the selected areas of the core, indicted in

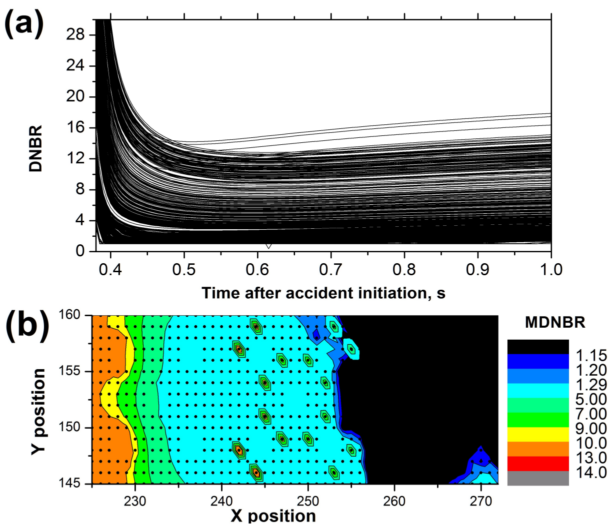

Fig. 6 (a) Evolution of DNBR and (b) distribution of MDNBR in the selected areas of the core, indicted in Fig. 2(b).

3.2.2 Maximum fuel temperature

Fig. 5 shows the distribution of maximum fuel centerline temperature at the selected areas of the core. As expected, distribution of centerline temperature was very similar to the entalphy distribution, shown in Fig. 4(b), and maximum temperature was observed at the position of 262/155 also. Maximum temperature was 2,335K. This also shows a sufficient margin to the one of the coolability criteria, fuel melting.

3.2.3 DNBR evolution

Fig. 6(a) shows the evolution curves of DNBR at the selected areas of the core. DNBR of each fuel rod was greatly reduced about 0.3ms after the accident, and it reached minimum between ~0.4ms and ~1s depending on the injected powers. Minimum DNBR(MDNBR) was ranging from ~14 to 1.00. Observed axial position of MDNBR was below the 1.14m from the bottom position. But, this is slightly above the observed position of peak fuel entalphy. Fig.6(b) shows the distribution of MDNBR at the selected area. It reveals that about 280 fuel rods have lower MDNBR than the current DNBR safety limit. And if we consider the whole areas of the core, much more rod failure can be expected. This suggest the possibility of rod failure at the rated power conditions also.

3.3 Further works

Preliminary calculation on the HZP REA of APR1400 has been done by utilizing the best-estimate code. But for the establishment of realistic evaluation methodology, uncertainty quantification has to be done also. For this, following areas need to be studied further.

- Refinement of accident scenario.
  - For example, completion of rod ejection time.
- Identification of uncertainty parameters
  - Related to the fuel rod, models used in the codes, heat transfer, thermal-hydraulic boundary conditions, power rise & width during transient etc.
- Analyze the effect of fuel burnup and initial power condition before transient.
- Selection of key uncertainty parameters and statistical treatment method.

4. Summary

For the development of audit calculation methodology of RIA safety analysis based on the realistic evaluation approach, preliminary calculation by utilizing the best-estimate code has been done on the initial core of APR1400. Followings are main conclusions.

- With the assumption of single full-strength control rod ejection in HZP condition, rod failure due to PCMI is not predicted.
- And coolability can be assured in view of entalphy and fuel melting.
- But, rod failure due to DNBR is expected, and there is possibility of fuel failure at the rated power conditions also.

And, for the establishment of audit calculation methodology, required further works are listed.

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