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CEA Ejection Accident Analysis for APR1000

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1. Introduction

The project to develop basic design of the Advanced Power Reactor 1000 (APR1000) plants has conducted by Korea Electric Power Corp. (KEPCO) since the end of 2009. The APR1000 has been designed to consider the operational experience of Optimized Power Reactor 1000 (OPR1000) plants and emerging safety features. To confirm the feasibility of the design concepts of APR1000, some selected design basis accidents (DBAs) have been analyzed using Korea Non-LOCA Analysis Package (KNAP). In this study, the Control Element Assembly (CEA) ejection accident was analyzed on the view point of the system response to examine the feasibility of the APR1000 design concepts. The results were compared with those values of the OPR1000 Final Safety Analysis Report (FSAR).

2. Plant Modeling

2.1 Reactor Coolant System Modeling

The APR1000 [1] is a typical 1,000MWe pressurized water cooled reactor based on the OPR1000. So, in spite of the improvement in safety features or systems, the APR1000 is treated as the similar plants to the OPR1000 in the nuclear steam supply system (NSSS) modeling for KNAP. The APR1000 NSSS was modeled according to the KNAP topical report [2] and modified to implement the improved design features. The detailed modeling is as the preliminary study [3] and Fig. 1.

2.2 Hot Spot Modeling

The whole core was modeled in two channels, *i.e.*, the average core channel and hot channel, according to the KNAP Hot Spot Model (HSM) [1, 4]. The average channel and hot channel were modeled to represent the whole core and the hottest channel caused by the accident, respectively [3, 4]. For the average channel, the whole core was divided and modeled as the 6 stacked control volume and corresponding heat conductors. The hot channel was modeled to represent the hot single channel with 25 stacked volumes and heat conductors. The detail fuel data, such as gap gas composition, fuel pellet plena pressures, *etc*., was developed based on the outputs of fuel design code, FATE.

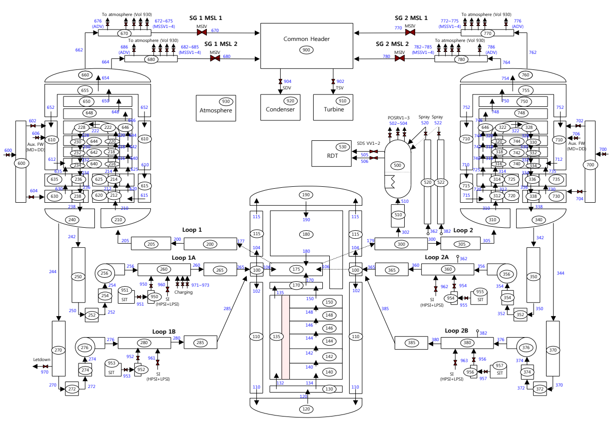

Fig. 1 RETRAN nodal diagram for APR1000

3. Rod Ejection Accident Analysis

To analyze the CEA ejection, the initial conditions and assumptions recommended by Regulatory Guide 1.77 of US Nuclear Regulatory Commission or corresponding guidelines of Korea Institute of Nuclear Safety were considered and implemented (Table 1).

Table 1. Initial Conditions and Assumptions

Parameter	Value
Core power Level, % to Rated Power	102 or 0
Core Inlet Temp. ^{o}F	572
Core Mass Flow, % to Rated Flow	95
Pressurizer Pressure (Pressure Case), psia	2,350
Pressurizer Pressure (Enthalpy Case), psia	2,000
Moderator Temperature Coefficient, $\Delta\rho/^{o}$F	0.0
Ejected CEA Worth, $10^{-2}\ \Delta\rho$	0.1584
Total SCRAM Worth, $10^{-2}\ \Delta\rho$	-6.0
Postulated CEA Ejection Time, sec	0.05
Maximum Radial Peaking factor	2.855

To analyze the system response to the accident, the analysis was carried through two cases, *i.e.*, the pressure case to confirm the maximum pressure and the enthalpy case to confirm the maximum fuel temperature or radially averaged enthalpy. The minimum departure from nucleate boiling ratio (DNBR) was calculated in the enthalpy case condition. For the power level, the hot full power and hot zero power conditions were considered.

The results of this study were compared with those of OPR1000 to examine the applicability and to confirm the effects of the design improvements. As given at table 2, the trends of the transients are similar each other.

Table 2. Sequence of Events

Event	OPR1000		APR1000	
	Time	Value	Time	Value
CEA Ejection	0.0		0.0	
Reactor Trip	0.03		0.03	
Max. Power (HFP), %	0.08	164.2	0.08	157.3
Max. PZR Press, psia	2.44	2500.0	2.23	2500.0
Max. Fuel Temp., °F	3.44	4,875.0	3.53	4,693
Max. SG Press, psia	6.35	1,316	9.55	1,326

The power trends of APR1000 show the similar trends mentioned in FSAR of reference plants (Fig. 2, 3). The difference in power trends was caused by the ejected CEA worth and radial peaking factor. By the adoption of PLUS7 fuel assembly of APR1000, the milder power trends were estimated.

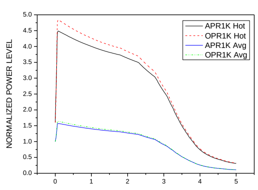

Figure 2. Power (HFP)

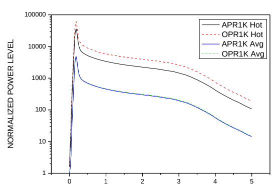

Figure 3. Power (HZP)

The maximum temperature of fuel (Fig. 4) and cladding surface (Fig. 5) showed the similar trends each other. The temperatures of APR1000 were showed lower values, but the difference was insignificant values on the view point of specified acceptance design limits (SAFDL).

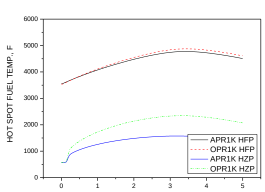

Figure 4. Max. Fuel Temp.

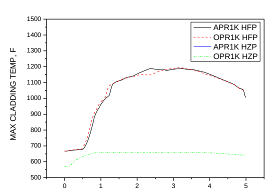

Figure 5. Max. Clad Temp.

Because the fuel enthalpy was estimated by the fuel temperature, the trends of the fuel enthalpy followed those of fuel temperature. In both cases, HFP and HZP, the APR1000 showed lower values than the reference plants (Fig. 6). The difference, however, were insignificant. By the calculation of minimum DNBR (Fig. 7), it was induced that the failed fuel fraction of APR100 would not exceed that of the reference plants due to the slightly higher values of APR1000.

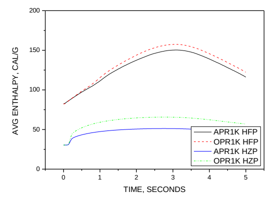

Figure 6. Fuel Enthalpy

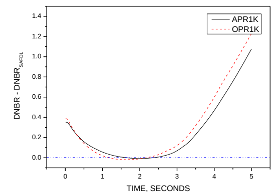

Figure 7. DNBR-$DNBR_{SAFDL}$

The pressure responses were mentioned in the preliminary study [3]. The design review to mitigate the steam generator shell-side pressure has been performed and the setpoints to open or close the main steam safety valves will be adjusted according to the review results.

4. Conclusion

The feasibility study to confirm the APR1000 design concepts was performed using KNAP. The pressure and enthalpy cases were analyzed under the HFP and HZP conditions. The analysis results were compared with those of FSAR of reference plants. Through the study, it was concluded that the APR1000 design concepts could be acceptable on the view point of CEA ejection accident design criteria.

Acknowledgements

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