The Analysis of Uncontrolled CEA Bank withdrawal from low power Accident for APR1400 Using KNAP

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1. Introduction

In Korea, the nuclear industries such as fuel manufacturer, the architect engineer and the utility, have been using the methodologies and codes of vendors, such as Westinghouse(WH), Combustion Engineering(CE), for the safety analyses of nuclear power plants. Consequently the industries have kept up the many organizations to operate the methodologies and to maintain the codes for each vendor. It may cause difficulty to improve the safety analyses efficiency and technology related. So, the necessity another of methodologies and code systems applicable to Non-LOCA, beyond design basis accident and performance analyses for all types of pressurized water reactor (PWR) has been raised. As the first requirement, the best-estimate codes were required for applicable wider application area and realistic behavior prediction of power plants with various and sophisticated functions. After the review on several candidates, RETRAN-3D has been chosen as a system analysis code.

The draft version of the methodology was developed based on the references for the general purpose, and modified to apply it to specific plants in Korea. As a part of the feasibility estimation for the methodology and code system, uncontrolled CEA bank withdrawal from low power accident for the Advanced Power Reactor 1400(APR1400) was selected to verify the feasibility using the RETRAN-3D[1,3]. And the results were compared with the Standard Safety Analysis Reports (SSAR) of APR1400.

2. Application Plant Modeling

The APR1400 is a 1400MW 2-loop plant. Base deck was made with RETRAN code to feasibility study. The core region is divided into 6 heat structures to design nuclear fuel assemblies. The core shroud and control rod motion region is made to realization coolant bypass flow model in reactor vessel. Also, the reactor vessel is divided into 10 control volumes to represent the cold leg nozzles, downcomer, down plenum, core and upper plenum for the analysis of thermal hydraulic behaviors in the reactor vessel and steam generator (SG) region. The U-tubes of SG are divided into 12 control volumes and heat structures to design heat transfer through tubes under the assumption that the bend regions of the tubes do not play an important role in the heat transfer phenomena.

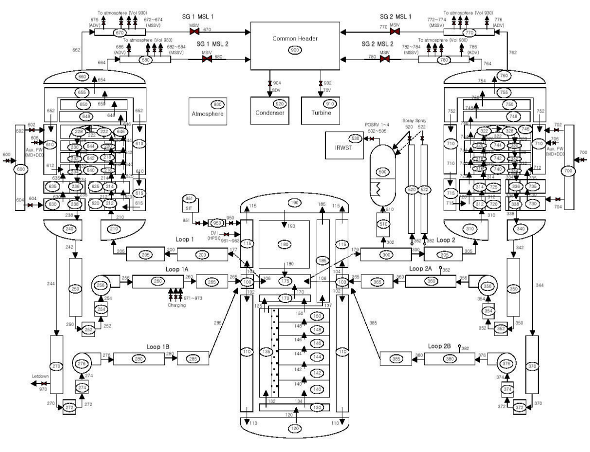

Fig. 1. APR1400 Nodal Diagram

3. Accident Description

An uncontrolled sequential withdrawal of CEAs is assumed to occur as a result of a single failure in the control element drive mechanism (CEDM), control element drive mechanism control system (CEDMCS), or reactor regulating system, or as a result of operator error. The withdrawal of CEAs from low power conditions adds reactivity to the reactor core, causing both the core power level and the core heat flux to increases together with corresponding increase in reactor coolant temperatures and reactor coolant system (RCS) pressure to rise and DNBR and the linear heat rate (LHR) margin to decrease. The pressure increase activates the pressurizer sprays which mitigate the pressure rise. The increase in the coolant and fuel temperature in combination with an expected negative moderator temperature coefficient and Doppler coefficient, which is always negative, causes negative reactivity addition which mitigates the rise in core power and heat flux. The withdrawal of the CEAs also causes the axial power distribution to shift to the top of the core. The associated increase in the axial peak is compensated for by a corresponding decrease in the integrated radial peaking factor. The magnitude of the RETRAN peak change depends primarily on the initial CEA configuration and initial axial power distribution. The withdrawal of CEAs also causes the neutron flux power measured by the ex-core detectors to be recalibrated due to CEA motion.

As the core power and heat flux increase, a reactor trip on the core protection calculator (CPC) or high power trip may occur to terminate the event depending on the initial operating conditions and the rate of reactivity addition[8]. If a trip occurs, the CEA drops into the core and insert negative reactivity which terminates further thermal margin degradation. If no trip occurs and corrective action is not taken by the operator, the CEAs fully withdraw and the nuclear steam supply system (NSSS) reaches to a new steady state equilibrium with higher power, temperature, peak LHR and a lower hot channel DNBR value.

4. Accident analysis

In the case of Uncontrolled CEA Bank withdrawal from low power accident for APR1400, the dynamic behavior of major RCS parameters and core power following the drop of a single CEA are presented. RETRAN and SSAR cases were evaluated under same conditions.

The reactivity insertion rate accompanying the uncontrolled CEA withdrawal is dependent primarily upon the CEA withdrawal rate and the CEA worth, since, at low power conditions, the normal reactor feedback mechanisms do not occur until power generation in the core is large enough to cause changes in the fuel and moderator temperatures.

TABLE I

Initial Conditions for Uncontrolled CEA Bank withdrawal from low power Accident

Parameter	Value
Core power Level, MWt	0.03983
Core Inlet Coolant Temp. ^oF	563.0(295.0)
Core Mass Flowrate, 10^6lbm/hr	153.52
Pressurizer Pressure, psia	2175
Steam Generator Pressure, psia	1161.0
Moderator Temperature Coefficient, $10^{-4}\ \Delta\rho/^oF$	0.5
Doppler Reactivity	least negative
CEA Reactivity Addition Rate, $10^{-4}\ \Delta\rho$/sec	0.890

The initial conditions and NSSS characteristics assumed in this analysis have been determined to be the limiting set of conditions allowed by the limiting conditions for operation (LCOs) in terms of providing the closest approach to the fuel design limits for a CEA withdrawal from low power level. The initial conditions which provide the closest approach to the duel design limits correspond to low power, core inlet temperature of 563 ^oF(295.0 ^oC), core inlet flow of 95% of design flow and minimum RCS pressure of 2175 psia. The initial RCS pressure is chosen to be the lowest allowed pressure within the LCOs since this allows the transient response to the CEA withdrawal to proceed for a longer time by delaying actuation of the high pressurizer pressure trip.

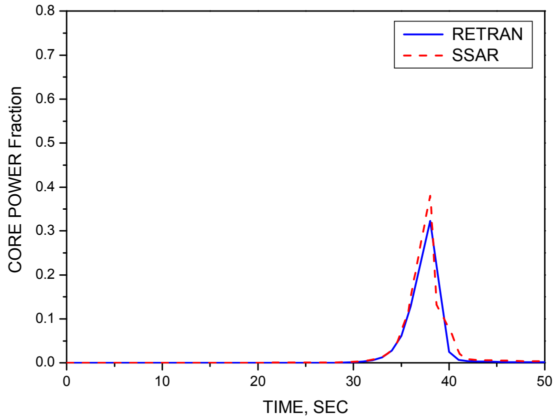

Fig. 2. Core power fraction

5. Conclusion

To develop the Korea Non-LOCA Analysis Package (KNAP) and confirm the feasibility, CRD, Uncontrolled CEA Bank withdrawal from low power, Rod ejection accident of APR1400 plant using the RETRAN code is analyzed. Analysis result of RETRAN code for APR1400 is compared with those of SSAR or using the presented code in SSAR.

Throughout this study, reactivity insertion accidents handled in this paper are presented very similar trend and acceptable results are produced.

REFERENCES

[1] C. K. YANG and Y. H. KIM, "A comparative analysis for control rod drop accident in WH and CE type nuclear power plant using the RETRAN", *Fourth Japan-Korea Symposium on Nuclear Thermal Hydraulics and Safety*, **NTHAS4, Sapporo, JAPAN** (November, 2004).

[2] Y. H. KIM, and C. K. YANG, "Development of Safety Analysis Methodology for Reactivity Insertion Accidents using Modified RETRAN Code", *Journal of NUCLEAR SCIENCE and THCHNOLOGY",* **Vol. 42,** *No. 11, p.1001-1009* (November, 2005).

[3] M. P. Paulsen, et al., "RETRAN-3D : A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems", NP-7450-A, Rev. 5, Vol. 1 & 3, Electric Power Research Institute, (2001).

[4] ABB Combustion Engineering Nuclear Fuel, "CETOP Thermal Margin Model Development", Rev.01, CE-NPSD-150-P, ABB Combustion Engineering, (1991)

[5] C. K. Yang, *et. al.*, "The RETRAN Reactivity Modeling for Westinghouse Nuclear Plant Analysis," *Proc. of KNS 2001 Fall Meeting*, Korea Nuclear Society, Soowon, KOREA, (2001)

[6] Combustion Engineering INC., "CESEC Digital Simulation of a Combustion Engineering Nuclear Steam Supply System", CE PLANT SYSTEM ANALYSIS DECEMBER, 1981