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The Analysis of Control Rod Drop 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, Drop Rod 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.

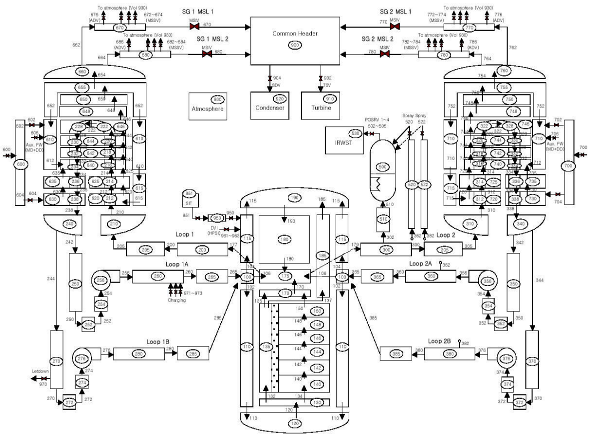

Fig. 1. APR1400 Nodal Diagram

3. Analytical Method and Procedure

In the Korea Non-LOCA Analysis Package (KNAP) methodology, if CEA is dropped into the core, the positive reactivity addition from moderator and Doppler reactivity feedback effects is eventually sufficient to compensate for the negative added by the dropped rod[2]. Therefore, recovery of reduced parameters which power, temperature, pressure, and etc are not dominated by the motion of control rod. It is just depended on reactivity feedback effect[5]. Under considering for above scenario, CRD event for APR1400 is modeled for RETRAN. The results of APR1400 are compared with those calculated by the vendor system code, CESEC-III[4,6].

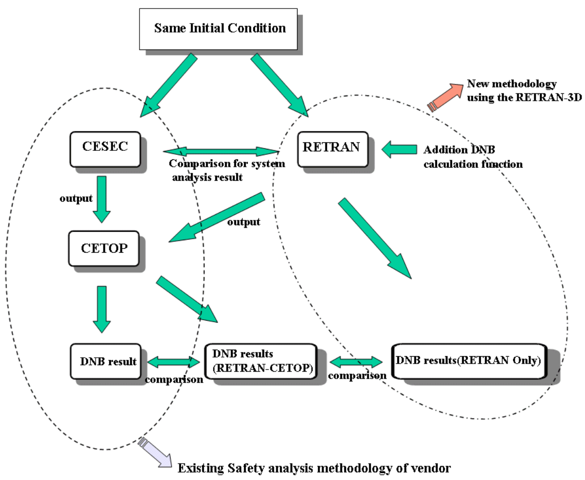

Fig. 2. CRD Analysis flow chart of APR1400

4. Assumption & Calculation

To begin with, all conditions which compared with results of RETRAN with SSAR's results were analyzed at same condition. The reactor regulating system (RRS) is assumed to be in the automatic mode. The nuclear steam supply system response to the single CEA drop was simulated using the CESEC-III. According to the above procedure, RETRAN code results of APR1400 are compared with those mentioned SSAR results.

TABLE I

Initial Conditions for CRD Accident

Parameter	Value
Core power Level, MWt	4062.66
Core Inlet Coolant Temp. °F	563(295.0)
Core Mass Flowrate, 10^6lbm/hr	153.52
Pressurizer Pressure, psia	2175
Steam Generator Pressure, psia	1079
Axial Shape Index	-0.3
Dropped CEA Reactivity Worth, 10^{-2} $\Delta\rho$	-0.10
Moderator Temperature Coefficient, $\Delta\rho$/ °F	$-4.0*10^{-4}$
Doppler Reactivity	Most negative

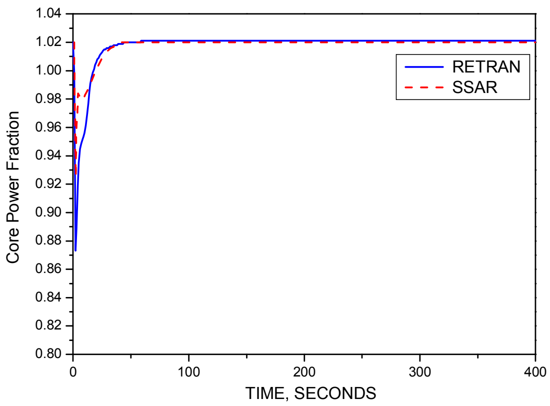

Fig. 3. Core power fraction (CRD event)

5. Conclusion

To develop the Korea Non-LOCA Analysis Package (KNAP) and confirm the feasibility, CRD 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. Afterwards, the system variables produced in this study for CRD accident are used to input of the code calculating DNBR. Also, the code system will be used to wider application area and modified to reflect the problems occurred by the application.

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