Transactions of the Korean Nuclear Society Autumn Meeting
PyeongChang, Korea, October 30-31, 2008

A Comparative analysis for control rod drop accident in WH and CE type

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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, 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 occur 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. Due to the above reason, the Korea Electric Power Research Institute(KEPRI) had decided to develop the new safety analysis code system for Korea Standard Nuclear Power Plants in Korea. 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 investigation for few candidates, RETRAN-3D has been chosen as a system analysis code. As a part of the feasibility estimation for the methodology and code system, CRD(Control Rod Drop) accident which an event of Non-LOCA accidents for Uljin units 3 & 4 and Yonggwang 1 & 2 was selected to verify the feasibility of the methodology using the RETRAN-3D. And the results were compared with those mentioned in the final safety analysis reports (FSARs) of the plants.

2. ANALYTICAL METHOD AND PROCEDURE

To begin with, the CRD accident is classified as an ANS condition II event. The CRD event initiated by an electrical or mechanical failure those results in one or more control rods from the same group of a given dropping to the bottom of the core. The negative reactivity insertion from the dropped control rod causes a prompt reduction in nuclear power followed by decrease in Reactor Coolant System (RCS) pressure and hot leg temperature. As the amount of dropped rod worth, a direct reactor trip would be occurred following the dropped rod event. If reactor trip would not occur, the initial nuclear power could be restored by reactivity feedback or bank withdrawal.

3. RETRAN Model for Application plants

Yonggwang units 1 and 2 (YGN-1/2) are 900 MWe 3-loop pressurized water reactors (PWRs) designed by WH.

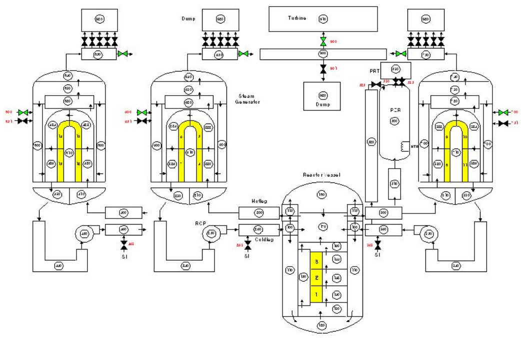

Figure 1. YGN 1/2 nodal diagram

For making nodal of the plants, the operation and design data of those were used, and in some cases the data of Kori units 3 and 4 (KRN-3/4) were selected to fill up the lacks of reference data. In fact, the YGN-1/2 and KRN-3/4 are typically the same plants at the viewpoint of safety analysis. The model is composed of 67 volumes, 104 junction, 3 reactor core heat conductors, 112 trip cards, and 227 control block description cards. Entire loops are modeled separately to assure the capability to analyze the loop asymmetry events. The nodal diagram is shown in Fig.1.

Uljin units 3 and 4 (UCN-3/4) are 1,000 MWe 2-loop PWRs and designed as the first Optimized Power Pants 1000 (OPR1000). On the viewpoint of safety analysis, however, UCN-3/4 could be classified as CE-type plants.

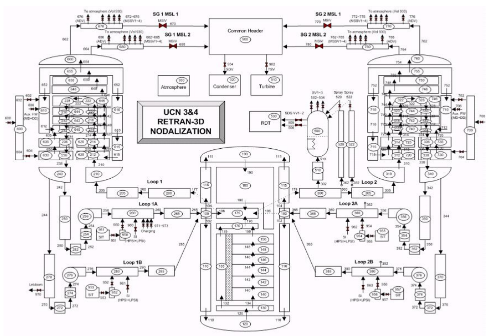

Figure 2. Uljin 3/4 nodal diagram

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 4 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.

4. Results

In response to the control rod drop, the core average power decreases rapidly, leading to a decline in the hot leg temperature. The decrease in hot leg temperature consequently reduces the energy available for removal through the steam generator, resulting in a reduction in SG pressure and temperature and RCS cold leg temperature. The cooling of the RCS produces a decrease in the RCS pressure and subsequent outsurge from the pressurizer.

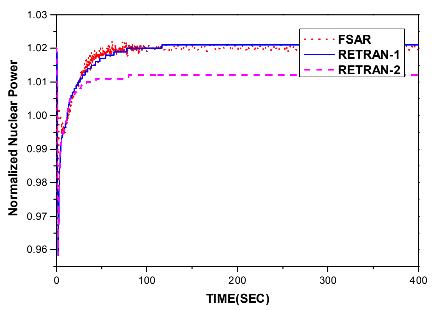

Figure 3. Normalized Power (UCN-3/4)

The indicated nuclear instrumentation power is influenced by the core nuclear power and excore detector tilt. The excore detector tilt modeling in this study cause the indicated nuclear instrumentation power to be significantly less than the actual core power. The mismatch between the indicated nuclear instrumentation power and the turbine power demand, in conjunction with a Tref – Tavg mismatch, causes the RCS to generate a signal to withdraw the control bank. As the control bank is withdrawn, the core average power begins to increase. The combination of reactivity feedback, control bank worth, and dropped control rod worth is sufficient to cause the continued withdrawal of the control bank such that the core power eventually overshoots the initial value. The DNBR response to the event shows that the DNBR increase immediately after the event initiation due to the decrease in core power. As core power begins to recover, the DNBR peaks then begins to decrease. The time of the MDNBR for the event, limiting statepoint condition, occur at approximately the same time as the peak core thermal power. At the time of MDNBR, the pressurizer pressure and core inlet temperature indicates increasing trend but not higher than their initial value.

The results performed in the different condition show that the statepoint conditions of the CRD event become less limiting as the dropped control rod worth is increased. If the dropped control rod worth is great enough, a reactor trip will occur by the low pressurizer pressure signal. This study also indicates that control rod event become more limiting as the worth of the inserted control bank is increased. A great inserted control bank worth allows the rod control system to more easily overcome the power reduction induced by the dropped control rod. The increased control bank worth also means a power overshoot is more likely to occur as a result of the control bank withdrawal.

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