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Preliminary Performance Analysis Program Development for Safety System with Safeguard Vessel

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1. Introduction

SMART is an advanced modular integral type pressurized water reactor for a seawater desalination and an electricity production [1]. Major components of the reactor coolant system such as the pressurizer, Reactor Coolant Pump (RCP), and steam generators are located inside the reactor vessel. The SMART can fundamentally eliminate the possibility of large break loss of coolant accidents (LBLOCAs), improve the natural circulation capability, and better accommodate and thus enhance a resistance to a wide range of transients and accidents. The safety goals of the SMART are enhanced through highly reliable safety systems such as the passive residual heat removal system (PRHRS) and the safeguard vessel coupled with the passive safety injection feature.

Figure 1 shows a schematic drawing of the SMART safety systems with a safeguard vessel. The safeguard vessel is a steel-made, leak-tight pressure vessel housing the RPV, SIT, and the associated valves and pipelines. A primary function of the safeguard vessel is to confine any radioactive release from the primary circuit within the vessel under DBAs related to loss of the integrity of the primary system.

A preliminary performance analysis program for a safety system using the safeguard vessel is developed in this study. The developed program is composed of several subroutines for the reactor coolant system, passive safety injection system, safeguard vessel including the pressure suppression pool, and PRHRS. A small break loss of coolant accident at the upper part of a reactor is analyzed and the results are discussed.

2. Preliminary Performance Analysis Program

2.1 Theoretical Model

The developed program is composed of a reactor coolant system including a discharge flow model through the broken pipe, passive safety injection system, safeguard vessel including the pressure suppression pool, and PRHRS.

The reactor coolant system is simplified to be only composed of a saturated steam region and a subcooled water region for the pre-accident period. After a pipe break occurs, the reactor coolant system quickly reaches a saturated state. From this time, the whole reactor coolant system is considered as one volume with a saturated condition and a void fraction is calculated based on the mass and energy balances. Mass and energy conservation equations for the reactor coolant system are as follows:

$$\frac{dm_{RCS}}{dt} = \dot{m}_{SIT} + \dot{m}s_{ump} - \dot{m}_{dis} \qquad (1)$$

$$\frac{dm_{RCS}h_{RCS}}{dt} = Q_{DEC} + Q_{steel} + h_{SIT}\dot{m}_{SIT} + h_{sump}\dot{m}s_{ump} \qquad (2)$$

$$- h_{dis}\dot{m}_{dis} - Q_{PRHRS}$$

The discharge flow through the broken pipe is calculated by using the critical flow model used in RETRAN code. Every time step, incoming flow rates from the SIT and sump are calculated. As the reactor coolant system pressure is lower than the SIT pressure, the water is passively injected from the SIT into the reactor coolant system. The flow rate from the pressure suppression pool and safeguard vessel wall by a condensation into the sump are considered. Safeguard vessel consists of an environment of a saturated state and a pressure suppression pool filled with nitrogen gas and subcooled water. The safeguard vessel environment is considered as one volume and a void fraction is calculated based on a thermo-hydraulic balance. A heat transfer through the wall with outside air is calculated. The PRHRS model simulates steam generators, the steam line, condensing heat exchanger, and feedwater line. The operating flow rate of the PRHRS is calculated at the point where the buoyant force by a density difference is equal to the system pressure drop. SKBK and TSCON correlations are applied into the steam generator and the condensing heat exchanger [2, 3]. Nusselt correlation is utilized for the primary side heat transfer correlation of the steam generator at the condensing mode.

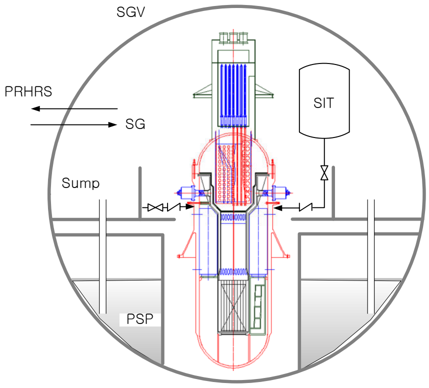

Figure.1 A schematic drawing of the SMART safety systems with a safeguard vessel

2.2 Program Structure

The developed program is composed of SMSIS and SMPRHRS subroutines. The SMSIS subroutine includes the models for the reactor coolant system, passive safety injection system, and safeguard vessel. The SMPRHRS subroutine is for the PRHRS. Figure 2 shows the structure of the developed program.

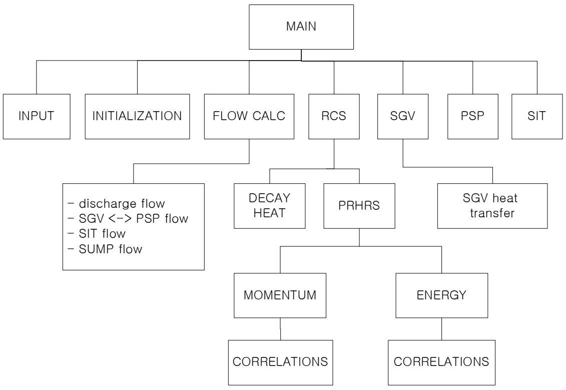

Figure 2. Program structure

3. Results

Using the developed preliminary performance analysis program, 50 mm pipe break scenario located at the upper part of reactor vessel is analyzed. The pre-accident reactor power level is assumed to be the rated value and its value is reduced to the decay heat level after ten seconds. Water is initially discharged through the break line with a large flow rate. As the water level becomes lower than the break point, steam begins to be discharged and the flow rate is reduced. Figure 3 shows the pressure of reactor coolant system and safeguard vessel. As the accident progresses, the pressure of the reactor coolant system decreases, but that of the safeguard vessel increases. After one and half hour, an equilibrium pressure of 1 MPa is achieved. At this time the discharge flow rate has about a zero value and even a negative value. The pressures of the reactor coolant system and safeguard vessel decrease due to the operation of the passive heat removal system and the condensation heat transfer through the safeguard vessel wall. The normalized level of the reactor coolant system shown in Figure 4 drastically decreases for an initial short period. As water is passively injected from the SIT, the normalized level increases. Water is supplied from the SIT for the first stage and from the sump for the second stage.

4. Conclusions

A preliminary performance analysis program for a safety system using a safeguard vessel is developed in this study. The result shows that the pressure equilibrium between the reactor coolant system and the safeguard vessel lets the discharge flow rate through the break point be nearly zero and even a negative value. The present analysis program can be used for the sensitivity study of a safety system using a safeguard vessel.

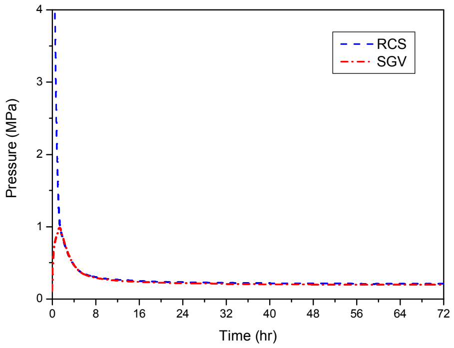

Figure 3. Pressures of reactor coolant system and safeguard vessel

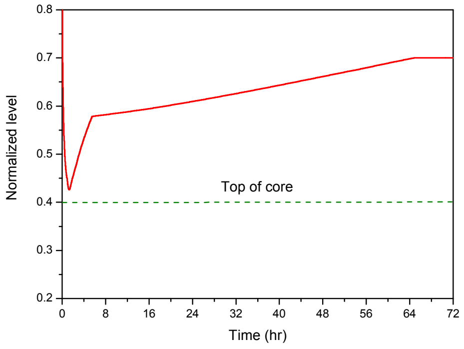

Figure 4. Normalized level of reactor coolant system

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