Determination of Initial Conditions for the Safety Analysis by Random Sampling of Operating Parameters

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1. Introduction

Conservative evaluation methodologies are usually applied for the quantification of safety margin incorporated in the design of nuclear power plants (NPPs). A reasonable conservatism is achieved by the adoption of appropriate code models and by the selection of proper initial and boundary conditions, which constitutes an important part of evaluation methodology.

In most existing evaluation methodologies, which follow a conservative approach, the most conservative initial conditions are searched for each transient scenario through tremendous assessment for wide operating windows or limiting conditions for operation (LCO) allowed by the operating guidelines. In this procedure, a user effect could be involved and a remarkable time and human resources are consumed.

In the present study, we investigated a more effective statistical method for the selection of the most conservative initial condition by the use of random sampling of operating parameters affecting the initial conditions.

2. Determination of Initial Conditions

The initial conditions are obtained through the steady state calculation of the MARS-KS code by changing the operating parameters automatically using the MOSAIQUE program [1], which has been developed by the Korea Atomic Energy Research Institute (KAERI). The selected reference plant is OPR1000. In transient analysis of the OPR1000, it is required to assess the effect of the reactor coolant flow rate, core inlet temperature, pressurizer pressure, pressurizer level, and steam generator (SG) level.

First, the effects of design parameters and setpoint parameters on initial operating parameters are analyzed, and a single variable which affect predominantly on each operating parameter is determined. Then, the distribution function and the range of the selected variables are adjusted to obtain the band of initial conditions required for the transient analysis of the OPR1000 provided in the previous study [2].

2.1 Reactor Coolant Flow Rate

The investigated design parameters affecting reactor coolant flow rate are the velocity, torque, and rated flow of reactor coolant pump. Among these, it is possible to have a reasonably wide range of reactor coolant flow rate by changing the rated flow of pump. For the random sampling of the rated flow, the Gamma distribution with α=2 and β=3 is assumed. And the range of the rated flow is from 4.39 to 18.07. The reactor coolant flow rate determined through 40 times random sampling is from 94.6 to 112.6% of the rated reactor coolant flow rate as shown in Fig.1. It is found that the influence of the pump rated flow on other initial conditions, such as pressurizer pressure, pressurizer level, and SG level, is minimized as described in Fig.2.

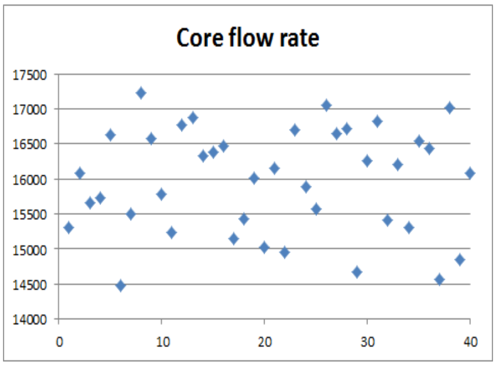

Fig. 1. Range of reactor coolant flow rate obtained from 40 sampling of pump rated flow.

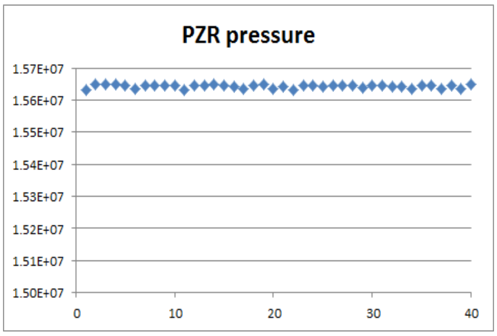

Fig. 2. The influence of pump rated flow on pressurizer pressure.

2.2 Pressurizer Level

The pressurizer level is assessed to be influenced by the charging flow rate, letdown flow rate, and the setpoint of pressurizer level control system (PLCS). The PLCS setpoint predominantly affects the initial pressurizer level as shown in Fig.3. The coefficient of PLCS setpoint is randomly changed between -812 and -785 to obtain the pressurizer level between 41 and 54%

of the nominal value. The coefficient is assumed to have a uniform distribution function.

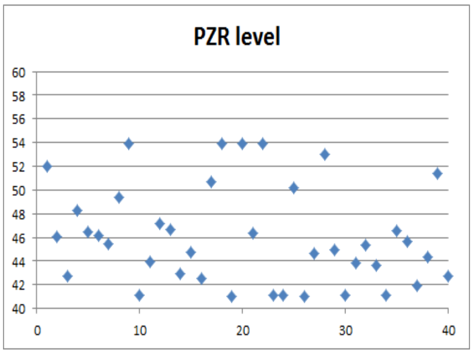

Fig. 3. Range of pressurizer level obtained from 40 sampling of PLCS setpoint.

2.3 Pressurizer Pressure

The pressurizer pressure is expected to be changed by pressurizer heater power, spray flow rate, and the setpoint of the proportional heater in pressurizer pressure control system (PPCS). Among these it is found that the PPCS setpoint on proportional heater most affects the initial pressurizer pressure as shown in Fig.4. It is allowed that the coefficient of PPCS setpoint can be changed from -1.21 to -1.13, which results in the pressurizer initial pressure between 13.79 and 16.03 MPa. This range of pressure is equivalent to 2000~2325 psia in English unit. The uniform distribution function is assumed for the coefficient.

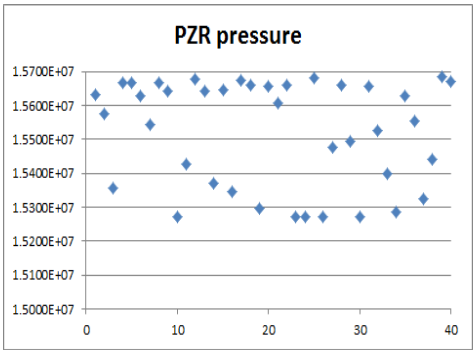

Fig. 4. Range of pressurizer pressure obtained from 40 sampling of PPCS setpoint.

2.4 Steam Generator Level

The steam generator inventory or level is balanced by the feedwater flow rate and steam flow rate. It is found that a wide enough range of SG inventory is obtained by adjusting the coefficient on narrow range fraction in FWCS control logic. The coefficient is assumed to have a uniform distribution, and it is changed between 2.4 and 4.7 to obtain the SG level between 46 and 86 %WR. In Fig. 5, the SG level obtained from 40 random sampling is provided. It is also checked whether the other initial conditions are influenced by the change of the FWCS coefficient as shown in Fig.6. This result suggests the appropriateness of the selected parameter.

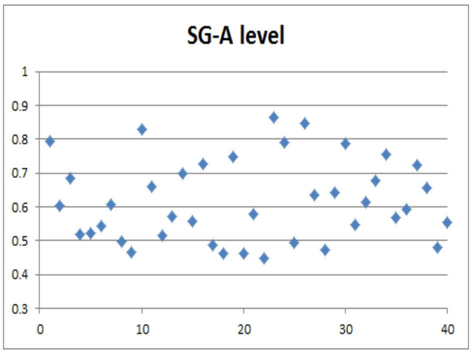

Fig. 5. Range of SG level obtained from 40 sampling of FWCS setpoint.

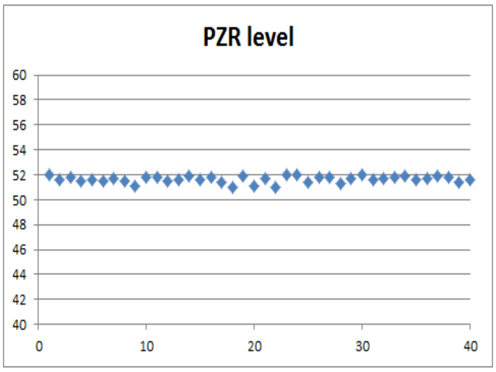

Fig. 6. The influence of FWCS coefficient on pressurizer level.

3. Summary

A method for the determination of initial conditions based on random sampling of plant design parameters is proposed. This method is expected to be applied for the selection of the most conservative initial plant conditions in the safety analysis using a conservative evaluation methodology. In the method, it is suggested that the initial conditions of reactor coolant flow rate, pressurizer level, pressurizer pressure, and SG level are adjusted by controlling the pump rated flow, setpoints of PLCS, PPCS, and FWCS, respectively. The proposed technique is expected to contribute to eliminate the human factors introduced in the conventional safety analysis procedure and also to reduce the human resources invested in the safety evaluation of nuclear power plants. A more thorough methodology would be developed based on the proposed approach in the future.

REFERENCES

[1] KAERI, MOSAIQUE Users Guide, Korea Atomic Energy Research Institute, 2014.
[3] KHNP, Korea Non-LOCA Analysis Package, TR-KHNP-0009, 2007.(in Korean)