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Items to Be Improved in MAAP-ISAAC Code for Best Estimate PSA Level 2

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1. Introduction

The evaluation of phenomena of severe accidents has been the subject of considerable research over the last several decades. By applying modern analysis tools and techniques, the US Nuclear Regulatory Commission (NRC) has developed a body of knowledge regarding the realistic outcomes of the selected severe reactor accident scenarios for the Peach Bottom and Surry Power Stations and as a result, published the related report, NUREG-1935, in 2012 [1].

Currently, the KHNP project (namely, K-SOARCA) [2] for best estimate Level 2 PSA is being performed by applying best modeling practices drawn from the collective wisdom of the severe accident analysis community. This paper describes the items for improvement in MAAP-ISAAC [1] (Modular Accident Analysis Program-Integrated Severe Accident Analysis Code for CANDU plants) [3] that are required for best estimate Level 2 PSA.

2. Improvement items of MAAP-ISAAC for the application of Level 2 PSA

In Korea, most severe accident analyses for CANDU plants have been performed by using the MAAP-ISAAC code. The improvement items of the code described in this paper are based on MAAP-ISAAC 4.03 which is being used in K-SOARCA.

2.1 Reflection of the design characteristics of the CANDU plant

The degasser condenser relief valves installed for the overpressure protection in a CANDU plant operate in the two following modes.

- Power mode : open at the pressure of 11.27 MPa(g) in the reactor outlet header
- Spring mode : start to open at the pressure of 10.06 MPa(g) in the degasser condenser tank and open fully at 10% accumulation pressure

The modeling of these valves applied in current MAAP-ISAAC code is only for the power mode. Because the power mode is impossible in a station blackout accident, the code cannot relieve the pressure of the primary system. Therefore, valve modeling for spring mode operation must be added in code.

Unlike a PWR, the turbine governor valves of a CANDU plant are not closed at reactor shutdown. That is, the open fraction of these valves is adjusted automatically to maintain the secondary side pressure. But, in the current MAAP-ISAAC model, these valves are closed simultaneously with reactor shutdown, so MSSVs (main steam safety valves) open due to the unreasonable increase of the secondary side pressure as shown in Fig. 1. This unreasonable opening of MSSV causes the steam generator (SG) secondary side inventory to empty faster and consequently, the severe accident condition to happen faster. Therefore, the model of the turbine governor valves needs to be modified based on the design characteristics of a CANDU plant.

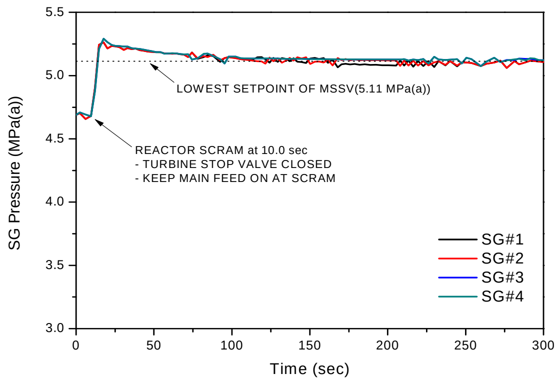

Fig. 1. Pressure behavior of the secondary side of the steam generators during SBLOCA

2.2 Reflection of the decontamination factors in ISLOCA and SGTR

The decontamination effect due to the deposition in a pipe was considered in US SOARCA. The containment vent pipe model is included in the latest version (5.04) of the MAAP code [4], which can estimate the decontamination effect due to the deposition in a pipe. Because this function is not included in the current MAAP-ISAAC code, the amount of released radionuclides is estimated conservatively. To estimate (preliminarily) the decontamination effect due to the

[1] MAAP is an Electric Power Research Institute (EPRI) software program that performs severe accident analysis for nuclear power plants including assessments of core damage and radiological transport. A valid license to MAAP4 and/or MAAP5 from EPRI is required.

deposition in a pipe in ISLOCA, the containment vent pipe model of MAAP 5.04 was used by applying the information estimated by MAAP-ISAAC such as thermal-hydraulic conditions in the reactor building and the amount of radionuclides released into the auxiliary building (the break discharge point of the ISLOCA). The result of the estimation shows that the decontamination factor for fission product groups other than Sb and Te_2 is 2.5 (60% decontamination effect) (refer to Fig. 2). Therefore, for the analysis of accidents such as ISLOCA in which decontamination due to deposition in a pipe is expected, the containment vent pipe model adopted in MAAP 5.04 needs to be included in MAAP-ISAAC code.

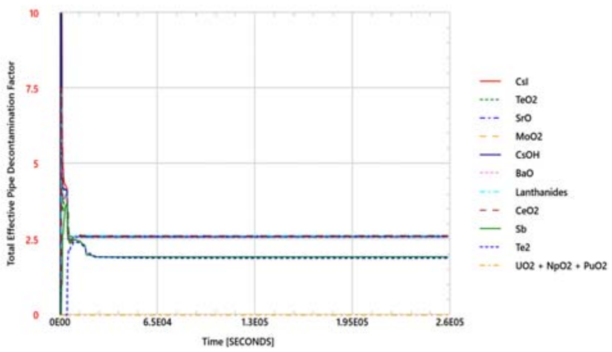

Fig. 2. Decontamination factors of each nuclide group in ISLOCA

One of the strategies to reduce the amount of radionuclides released to the environment in a SGTR accident is to inject the water into the secondary side of the steam generator. In the current MAAP-ISAAC code, the results of the latest research (such as ARTIST experiments [5, 6]) regarding the decontamination effect in a SGTR accident are not applied. FAI suggested the following decontamination factors, which can be conservatively applied to analysis of SGTR accidents of CANDU plants based on the ARTIST.

DF_{tube} within SG Tubes = 2 : without condensation
DF_{SG} for Dry SG = 1 : RCS has depressurized
DF_{SG} for Wet SG = 33 * DF_{pool}
Where, DF_{pool} is the value for bare pool estimated in current MAAP-ISAAC.

Using this recommendation, the best-estimate of the amount of radionuclides released into the environment in a SGTR, the decontamination factor needs to be reflected in MAAP-ISAAC model based on the results of the latest research regarding SGTR accidents.

2.3 Potential items for best-estimate plant modeling

The release of pressure using the Containment Filtered Vent System (CFVS) as one of the mitigation actions is considered in CANDU plants. In the current MAAP-ISAAC code, because this CFVS model is not incorporated, the user needs to implement a model for this function based on the site operation strategy. Also, the user must provide the decontamination factor using the junction model in which only one decontamination factor is applied regardless of the differing characteristics of each radionuclide. Therefore, a model of the pressure release based on the site operation strategy and a model applying the different decontamination factors according to the radionuclide characteristics should be added in the MAAP-ISAAC model code.

One of the SAMG entry conditions for mitigation of the severe accidents is to meet the subcooling margin criteria in the reactor inlet headers (less than 5℃). In the current code (unlike with the latest version of MAAP which provides the information of the thermal-hydraulic conditions of each node), only the average information of each loop is provided. So, the definite subcooling conditions in the reactor inlet headers cannot be determined from code calculations. Therefore, to definitely identify the subcooling margin condition, the thermal-hydraulic conditions of each node should be provided instead of the average ones of each loop. Another alternative is that model could be added to the code to provide the subcooling margin condition.

3. Conclusions

Through the K-SOARCA project, items have been identified as needed improvements in the MAAP-ISAAC code for best estimate PSA Level 2 analysis of CANDU plants. These improvement items are required to expect more realistically the phenomena of severe accidents and the amount of the released radionuclides such as Cs-137. If all of these improvement items are implemented appropriately in the MAAP-ISAAC code, it is expected to be applicable for the performance of best-estimate PSA Level 2 analyses.

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REFERENCES

[1] U.S.NRC, State-Of-the-Art Reactor Consequence Analyses Report, NUREG-1935, 2012.
[2] Kwang-Il Ahn, Seok-Won Hwang, The KHNP SOARCA Project: Development Strategy, KHNP Severe Accident and SOARCA Workshop, Sandia National Laboratories, Albuquerque, NM, April 17, 2017.

[3] KAERI, ISAAC Code User Manual, KAERI/TR-3645, 2008.
[4] EPRI, Modular Accident Analysis Program – MAAP5 v5.04 for Windows, Product Id: 3002007340, 2016.
[5] Lind, T., Dehbi, A., Guntay, S., “Aerosol Retention in the Flooded Steam Generator Bundle during SGTR,” Nuclear Engineering and Design, Vol. 241, pp 357-365, 2011.
[6] Güntay, S., Suckow, D., Dehbi, A., Kapulla, R., “ARTIST: introduction and first results”, Nuclear Engineering and Design, Vol. 231, pp. 109-120, 2004.