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# Development the DEC PIRT for APR1400 ATWS

Bum-Soo Youn [a*], Jae-Hwa Koh [a], Dong-Hyuk Lee [a]
[a]*Safety Analysis Group, Nuclear System Safety Lab., KHNP Central Research Institute*
[*]*Corresponding author: bsyoun81@khnp.co.kr*

## 1. Introduction

Since the Fukushima accident, the concern has increased internationally about the disaster and the severe accident. In particular, the importance of severe accidents prevention and mitigation has been highlighted. In June 2015, the KOREA revised the “Nuclear Safety Regulations[1]”, the severe accident management has been included in the existing design basis accident management. Currently, KHNP is pushing for the development of integrated safety analysis codes applicable to multiple failure accident. It is necessary to the extension development of a code for apply the multiple failure accident to the SPACE which is developed for thermal analysis of domestic PWR. In order to apply the SPACE code to multiple failure accident, the PIRT(Phenomena Identification and Ranking Table) has to develop considering the physical phenomena expected in multiple failure accident.

## 2. Methods and Results

This paper deal with PIRT of ATWS(Anticipated Transient Without Scram) in the accident management scope caused by multiple failures.

### *2.1 Definition of Accident*

Anticipated Transient Without Scram(ATWS) are anticipated operational occurrences accompanied by failure of reactor trip when required. ATWS events are of concern since, under certain conditions(e.g., additional component and/or system failures), these could lead to unacceptable consequences up to including core melt and release of radioactivity to the environment. The major concern of the ATWS derives from the consequences of the expected high primary system pressure, which is the characteristic of the transient.

### *2.2 Major Scenarios and Characteristics*

The sequence of event for ATWS shows the table Ⅰ. The phase Ⅰ is the reactor coolant system temperature and pressure increase phase after the steam generator is exhausted by loss of feed water. The phase Ⅱ is the POSRVs are open for depression of the primary system, and system pressure and water level is safety phase. The main thermal-hydraulic trend of the system shows the fig 1~6.

Table Ⅰ: Sequence of Event for ATWS[2]

| Phase | Time (sec) | Event | Set point |
|---|---|---|---|
| Phase Ⅰ | 0.0 | Loss of main feedwater | |
| | 30.7 | Steam generator level reaches low level trip set point | 45% WR |
| | 52.3 | Auxiliary feedwater enters steam generator | 25% WR |
| | 73.6 | Pressurizer pressure reaches high pressure trip set point | 2,374 psia |
| Phase Ⅱ | 74.0 | POSRVs open | 2,470 psia |
| | 81.0 | Main steam isolation valves fully close | |
| | 86.5 | Pressurizer solid begins | |
| | 97.3 | Peak primary pressure occurs | |
| | 115.7 | Hot leg saturation condition reached | |
| | 285.4 | Main steam safety valves open | 1,230 psia |
| | 484.4 | Safety Injection Actuation Signal | 1,810 psia |
| | 620.0 | Safety injection begins | |
| | 1,800.0 | Operators take manual actions to control plant | |

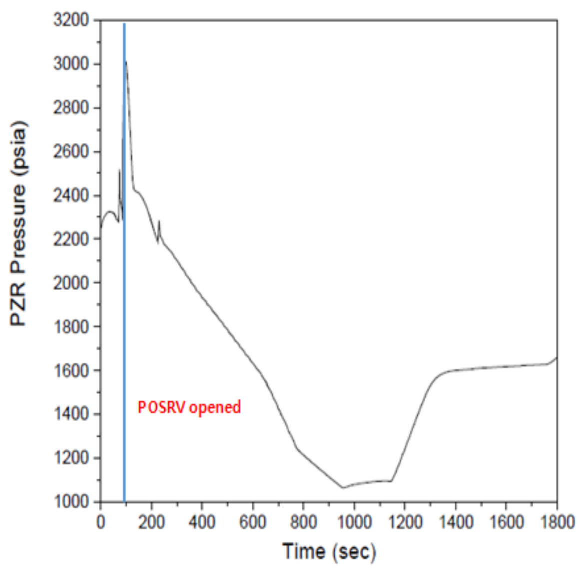


Fig. 1. Pressurizer pressure (SKN 3,4 ATWS).

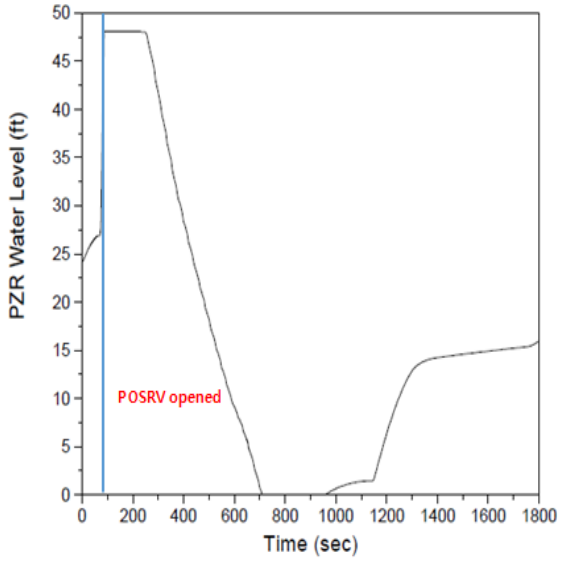


Fig. 2. Pressurizer water level (SKN 3,4 ATWS).

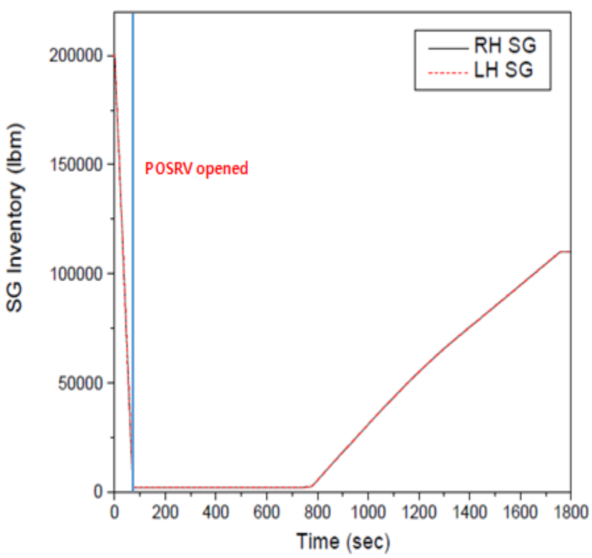


Fig. 4. Steam Generator water level (SKN 3,4 ATWS).

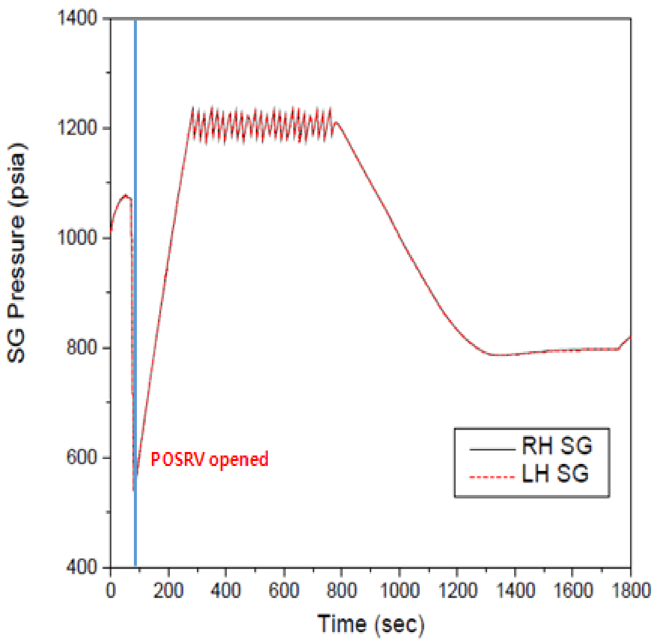


Fig. 3. Steam Generator pressure (SKN 3,4 ATWS).

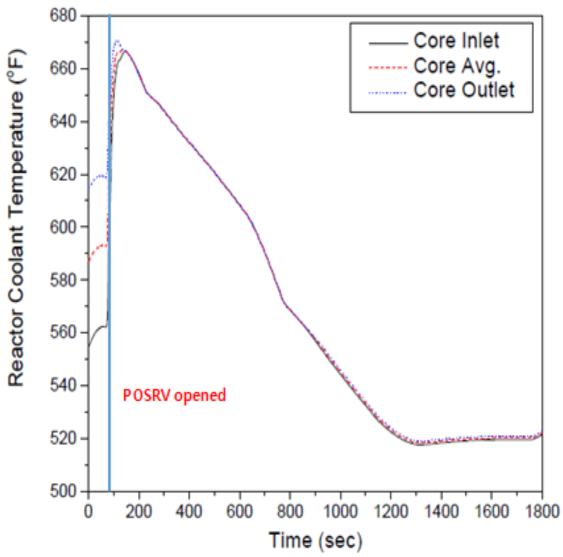


Fig. 5. Reactor coolant temperature (SKN 3,4 ATWS).

*2.3 Safety Assessment Criteria*

The acceptance criteria for ATWS safety analysis are as follows. The cladding temperature don't have to exceed 1,204℃(2,200°F). The reactor pressure boundary maximum pressure doesn't have to exceed 3,200 psig.

*2.4 Major Thermal-Hydraulic Phenomenon*

In addition to the accident scenarios and phase distinctions determined in the previous step, the system/the structure, and the components are categorized to help determined the ranking of major thermal-hydraulic phenomena. The major thermal-hydraulic phenomena in the component in the accident scenarios were identified and determined by the expert consultation.

*2.5 PIRT*

The ATWS PIRT shows the table Ⅱ.

Table Ⅱ: ATWS PIRT

| System/ Structure | Component | Phenomena | PIRT | |
|---|---|---|---|---|
| | | | Ⅰ | Ⅱ |
| RPV | Core (fuel) | Core power | H | H |
| | | Boron transport | - | M |
| | | Reactivity feedback (MTC/FTC) | H | H |
| | | Fuel heat transfer | M | M |
| | Downcomer | Boron mixing | - | L |
| | Lower plenum | Asymmetric flow | - | L |
| Pressurizer | Vessel | Level swell (volume change) | H | H |
| | | Level swell (flashing) | - | L |
| | | Wall condensation | L | - |
| | Surgeline | CCF | - | L |
| | | Flow resistance | L | L |
| | POSRV | Discharge flow | - | H |
| | | Entrainment | - | L |
| | Spray | Bulk condensation | M | - |
| SIS | SIP | SIP discharge flow | - | L |
| Steam Generator | U-tube | Wall heat transfer | L | L |
| | Secondary Side | Wall heat transfer | H | H |
| | | Collapsed water level (downcomer) | H | L |
| | | Pressure build up by MSIV closure | - | H |
| | MSSV | Discharge flow | - | L |

In this table, H means high influence on FOM(figure of merit). M means moderate influence on FOM. L means low influence on FOM.

As a result of developing PIRT, the core power, reactivity feedback, level swell(volume change) in the pressurizer vessel, and wall heat transfer in the steam generator secondary side were confirmed to be an important phenomenon.

## 3. Conclusions

It developed a major thermal-hydraulic phenomenon PIRT for ATWS accidents for expanding the SPACE code to apply to the design extended conditions. The major scenarios and the major thermal-hydraulic phenomenon of the system/structure and the component were derived through the developing the PIRT. PIRT was able to derive the thermal-hydraulic model needed to expand the SPACE code.

**Acknowledgements**

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## REFERENCES

[1] Nuclear Safety and Security Commission Regulation No. 17, "Regulation about the technical standard of reactor facility etc.", partial revision, June 30, 2016.

[2] KOPEC/NED/TR/04-006, Rev.0, "Analysis Report for ATWS Event of SKN 3&4," June 28, 2004.