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Transient analysis for total loss of feed water scenario due to postulated feed line breaks in both steam generators

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1. Introduction

The total loss of feed water (TLOFW) scenario is a beyond design basis accident (DBA) for nuclear steam supply system (NSSS) but is considered in the safety analysis report (SAR) for the stress analysis of structures of KEPIC class 1 and the hydrogen generation analysis. The postulated single feed line break (FLB) scenario is DBA and is described in chapter 15.2 of SAR. To evaluate the safety impact, the integrity of plant and the coping measures, a transient of total loss of feed water due to the postulated breaks of both feed line is analyzed for OPR1000 with RELAP5 code [1].

2. Analysis conditions and model

2.1 Analysis condition

In a hypothetical condition of feedline break in both steam generator, the loss of normal feed flow and the loss the SG secondary inventory are followed simultaneously in the both SG feed lines. Because of the loss of integrity of the feedwater pipes, the auxiliary feed water is also unavailable. Although the valve diameter ranges from 6 to 18 in, the effective break area is restricted by the area of the flow distribution holes in economizer which is 1.22ft^2 (0.1133m^2). In the case of small break, that is break flow rate is below 30% of feed flow rate, there is no effect on the plant safety because the feed water pumps have supply capacity up to ~130%.

In this study, a hypothetical scenario of pipe break of both feed line is considered. And an operator action according to the emergency operating procedures (EOP) is considered in the beyond DBA condition. The integrity of nuclear fuel and the effectiveness of the accident mitigation measures are evaluated. In the evaluation, the safety systems and the control systems are available, and the reactor is shutdown with the initiation of accident.

2.2 RELAP model

In the analysis, RELAP5 Mod 3.1K is used [1]. The nodalization of OPR1000 plant is presented in Fig. 1. The reactor is operating with 102% power and the other initial condition is adopted from LBLOCA safety analysis methodology [2].

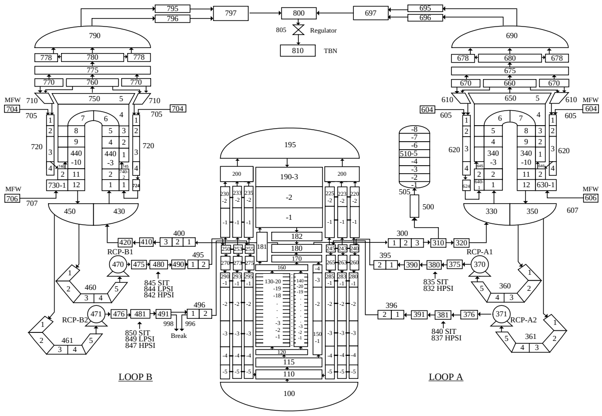

Fig. 1 Nodalization diagram of OPR1000

3. Analysis results

3.1 Transient analysis results

The transient calculation is performed for 3600 seconds. The major sequence of events for the transient is tabulated in Table 1.

Table 1. Major sequence of events

Event	Time [s]
Break start	0.0
Reactor trip	0.0
RCP trip	0.0
Turbine trip	0.26
Steam generator DC level lo	9.3
Pressurizer SV open	283.0
SDS operation	1800.0
HPSI	1948.7
LPSI	1964.2
SIT	2460.3

After the break, the secondary inventory and feed water are lost through the breaks in feedlines as shown in Fig. 2. The secondary inventory is drained within 50 seconds. Because of the imbalance by the decrease of heat removal through the secondary side, the pressure of pressurizer increases up to the safety valve set point as depicted in Fig. 3. Before the operator's action, the pressurizer safety valve is opened and closed repeatedly followed by the loss of primary inventory. After the operation of safety depressurization system (SDS), rapid

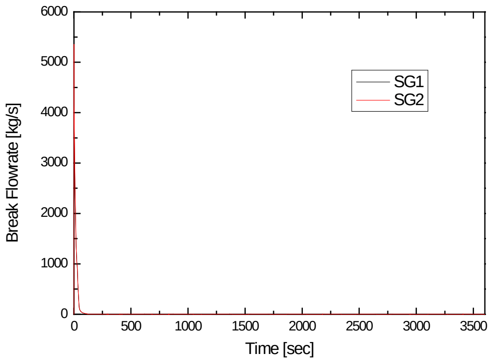

Fig. 2 Mass flow rate of the break

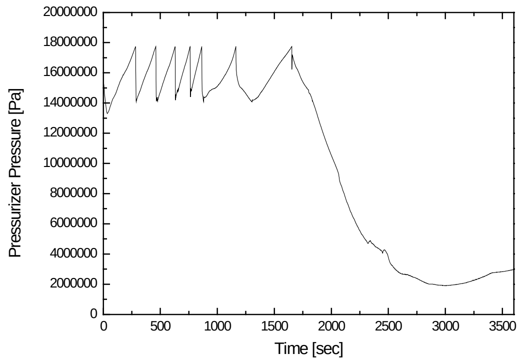

Fig. 3 Pressurizer pressure

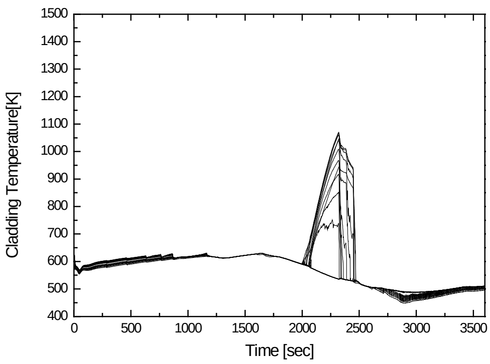

Fig. 4 Cladding temperature

depressurization of primary system occurs as shown in Fig. 3. And the pressure decreases to set point of safety injection. The peak cladding temperature is observed as 1030 [K] in Fig. 4 by the uncovery of the core.

To investigate the sensitivity of the peak cladding temperature on the break size, calculations for eight cases of break size are performed and shown in Fig. 5. The sequence of the initiation and operator action is same with the base case (1.22 ft^2). The peak cladding temperature is generally proportional to the break size as presented in Fig. 6.

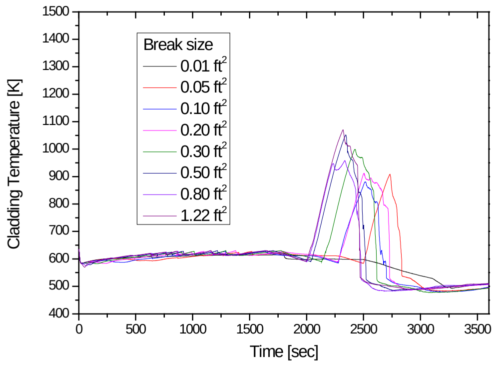

Fig. 5 Peak cladding temperature for various break sizes

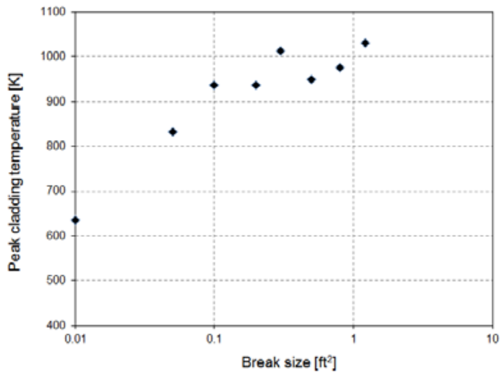

Fig. 6 Sensitivity of PCT on break size

4. Conclusions

To evaluate the safety impact, the integrity of plant and the coping measures, a transient of total loss of feed water due to the postulated breaks of both feed line is analyzed for OPR1000 with RELAP5 code. The calculations show that the operation of safety depressurization system at 1800 seconds is an effective measure to mitigate the core damage due to the uncovery of the core according to the pressurization of primary loop. Through the sensitivity studies, it is presented that the peak cladding temperature is proportional to the break size.

REFERENCES

[1] RELAP5/MOD3 Code Manual, INL, NUREG/CR-5535, 1995

[2] Topical Report for the Realistic Evaluation of Emergency Core Cooling System, KEPRI, TR-KHNP-0002, 2002

[3] Uljin 3,4 Final Safety Analysis Report, KHNP, 2013