Transient Analysis of Station Blackout for Framatome Nuclear Power Plant

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1. Introduction

Fukushima accident was caused by long hours of Station Black Out (SBO) caused by natural disaster beyond Design Based Accident (DBA) criteria. It led to the reactor core damage.

SBO (Station Blackout) refers to the complete loss of alternating current electric power to the essential and nonessential switchgear buses in a nuclear power plant. Station blackout therefore involves the loss of offsite power concurrent with turbine trip and failure of the onsite emergency ac power system, but not the loss of available ac power to buses fed by station batteries through inverters or the loss of power from alternate ac sources [1].

The purpose of this study is to provide strategies for maintaining core cooling and protecting the reactor core in SBO.

The thermal-hydraulic transient analysis using RELAP5/Mod3.3 code was performed to provide insights for Framatome NPP. It will be helpful for developing AMP (Accident Management Program) to cope with SBO for Framatome NPP.

2. Nodalization Model

The RELAP5/MOD3.3 code has been developed for best-estimate transient simulation of reactor coolant system (RCS) during accident. This code is a tool that allows users to model the coupled behavior of the reactor coolant system and reactor core during accidents. The reactor coolant system behavior is calculated using a two-phase model, which allows unequal temperatures and velocities for the two phase flow [2]. The modeling of the Framatome NPP has been developed using Hanul Unit 1&2 design data. Fig.1 shows the nodalization model of Framatome NPP for the analysis. The nodes of reactor are composed of the down-comer, lower plenum, upper plenum, core, and junction to connect with the hot leg. The pressurizer includes the 8 sub-control volumes and imaginary control volume to maintain the pressure uniformly during the steady state. However, this volume is removed during the transient state. The secondary side of Steam Generator (SG) includes the nodes of the main feed-water system, evaporator, riser, separator, and dome.

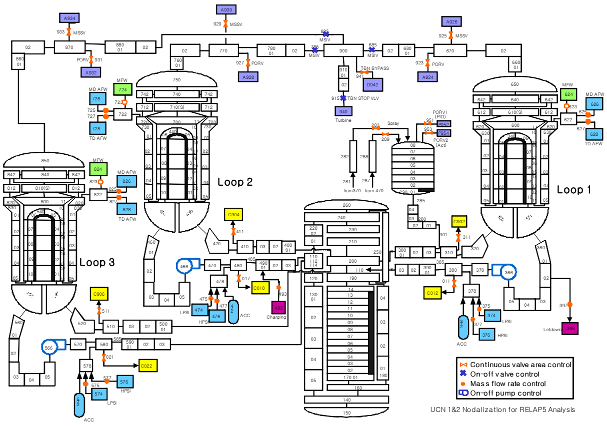

Fig. 1. Hanul Unit 1&2 Nodalization Model

3. Initial Conditions and Assumptions

3.1 Initial Conditions

Initial conditions calculated for 100% FP operation are well agreed with design values as shown in Table 1.

Table 1: Initial Conditions for 100% FP operation

Parameter	Design Value	Calculated Results	Error (%)
Core Thermal Power (MWt)	2,775	2,775	0
Cold leg Temperature (℃)	287.3	287.59	0.1
Hot Temperature (℃)	322	322	0
Pressurizer Pressure (bar)	155.1	155.1	0
Pressurizer Level (%)	63.15	63.73	0.91
Reactor Coolant Flow per Loop (kg/sec)	4,790	4,747	0.89
SG Pressure (bar)	57.7	57.36	0.6
SG Steam Flow (kg/s)	504.4	503.52	0.2
Feedwater Flow (kg/s)	504.4	503.52	0.2

3.2 Assumptions

SBO is assumed to occur during full power operation and in the event of an accident, the following systems and equipment become inoperable.

- Turbine
- Reactor Coolant Pump (RCP)
- Main Feedwater Pump
- Auxiliary Feedwater Pump (Motor Driven)

- Pressurizer Heater
- Charging and Letdown System
- Pressurizer Spray
- Safety Injection Pump

Regardless of SBO, the passive systems such as accumulator, main steam safety valves and pressurizer safety valves can be operated. Control valves of turbine-driven auxiliary water pump and atmospheric steam dump valves, which are powered by batteries, can also be operated.

The maximum flow rate of the turbine-driven auxiliary water supply pump assumed in the accident is 165 ㎥/hr, which is the design flow rate, corresponding to approximately 3.0% of the main water supply flow at full power.

RCP seal leakage rate is assumed to be 25 gpm/RCP and after 8 hour, the mobile generator supplies power to the loads necessary to ensure plant safety.

Assumptions in the analysis are as follows:

- Case 1
  - No operator actions
- Case 2
  - Feedwater supply using turbine-driven auxiliary water pump by operator (1,800 sec)
  - Cooldown the primary system using atmospheric steam dump valves by operator (1,800 sec)

4. Calculation Results

The sequence of events is provided in Table 2. Case 1 shows RCS response when the RCP seal leakage rate is assumed to be 25 gpm/RCP without secondary cooling by the operator action. The results of the accident analysis are presented in Fig.2 to Fig. 6.

Table 2: Event Sequences for SBO

Event	Time [sec]	
	Case 1	Case 2
• SBO occurs • Reactor trip	0.0	0.0
• TDAFP activation • atmospheric steam dump valve manual open	-	1,800
• SG dryout		-
• Pressurizer safety valve open	1,800	-
• Core boiling begins		
• Core uncovered	5,820	-
• Core damage	6,970	-
• Calculation end	7,050	28,800

After SBO, RCS pressure increased rapidly and then the Pressurizer safety valve opened to limit the RCS pressure. Thereafter, RCS pressure decreases due to cooling by SG and leakage of RCP, as shown in Fig. 2. However, cooling of RCS is over due to the SG dry-out as shown in Fig. 3. RCP seal leak continuous but not sufficient for the depressurization. Finally, the RCS pressure reaches the set-point of opening the Pressurizer safety valve. Fig .4 shows flowrate of Pressurizer safety valve. RCS inventory is reduced because the reactor coolant is released through the pressurizer safety valve. Fig. 5 and fig. 6 shows fuel cladding temperature and core level. Eventually, upper core was completely uncovered at 5,820sec, and core damage time was 6,970sec.

Case 2 is identical to Case-1 except that case 1 adds secondary cooling by the operator. RCS pressure is the same as case 1 until operator action. Operator action is performed at 1,800 sec and operator manually opens atmospheric steam dump valve and starts turbine driven auxiliary water pump for cooldown the RCS. Fig .4 shows flowrate of pressurizer safety valve. In Case 2, pressurizer safety valve is not opened due to the cooling by secondary side. RCS inventory is reduced by RCP seal leak, but reduction rate of RCS inventory is slower than Case 1 due to no release to pressurizer safety valve and accumulator injection by RCS depressurization. Finally, core is not uncovered within 8 hour as shown in Fig. 5 and Fig. 6.

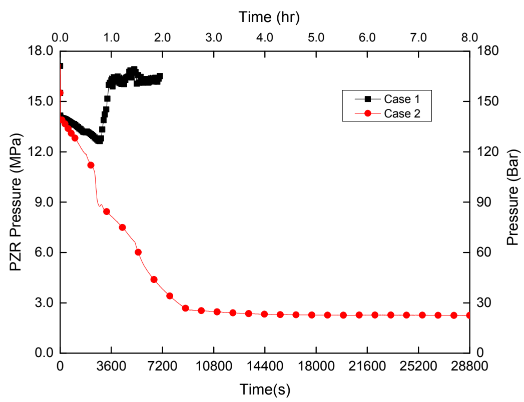

Fig. 2. PZR Pressure

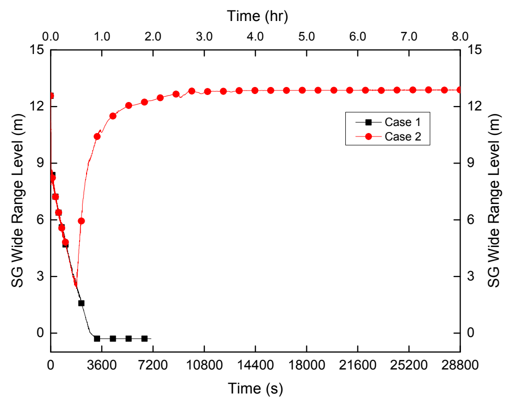

Fig. 3. SG Level

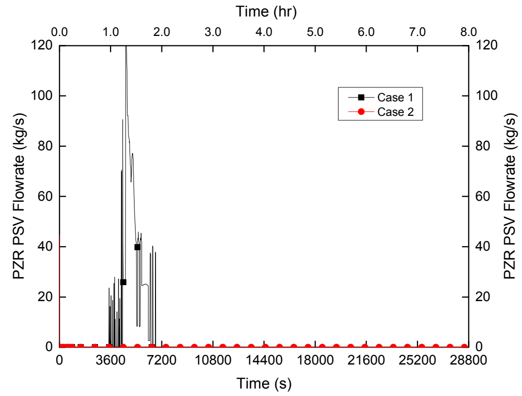

Fig. 4. Flowrate of Pressurizer Safety Valve

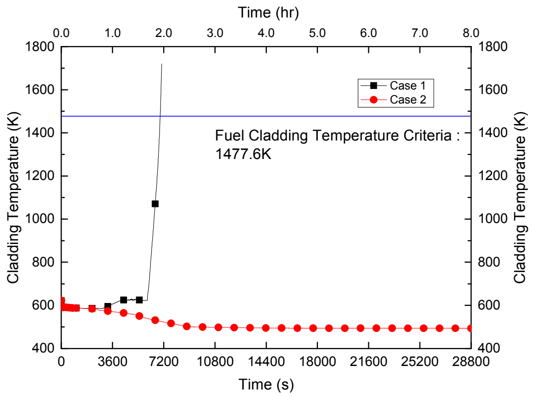

Fig. 5. Fuel Cladding Temperature

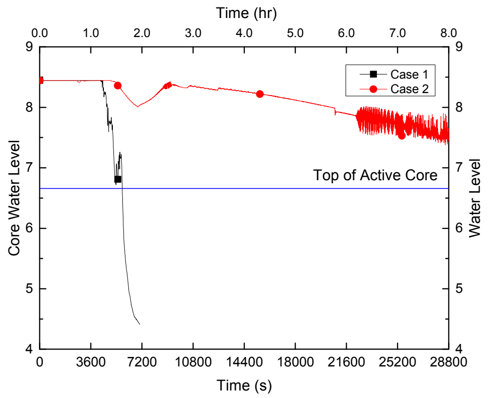

Fig. 6. Core Level

5. Conclusion

The thermal-hydraulic transient analysis was performed to provide insights for SBO.

If there is no operator action after SBO, there is no system for safety shutdown and maintain due to the loss of all AC power, and the pressure increases rapidly after exhausting the steam generator. Finally, the core damage occurs at 6,970 sec due to the loss of RCS inventory. If turbine-driven auxiliary water pump started and steam dump valve is opened by operator manual action within 1,800 sec, then 8 hour is available to avoid core uncover and damage. 8 hour is sufficient time for operator to prepare the mobile generator for supplying AC power.

This calculation results can be directly applied to the development of EOP (Emergency Operating Procedures) and MOG (MACST (Multi-barrier Accident Coping Strategy) Operating Guidelines) for Framatome NPP.

REFERENCES

[1] US NRC Regulatory Guide 1.155, Station Blackout, August 1988
[2] NUREG/CR-6150, "SCDAP/RELAP5/MOD 3.3 CODE MANUAL," Rev.2, Vol.3, Jan, 2001.