A Study on System Integrity during LOCV with Delayed Scram Event

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1. Introduction

Event categorization of design basis condition (DBC) 2 with delayed scram is introduced as DBC 3 for EUR [1]. Delayed scram means the relevant scram (trip) is assumed on the second actuation limit reached during event sequences. The limiting analysis case with delayed scram is determined on screen analysis procedure, and a loss of condenser vacuum (LOCV) with delayed scram is chosen as a limiting event for the system integrity. To check the system integrity during LOCV with delayed scram, a computer code of RETRAN-3D is used. Screen analysis procedure to determine the relevant event is described herein, and the results are shown to evaluate system integrity.

2. Methods and Results

The screen analysis procedure is established to determine the limiting one among the events with delayed scram. In this section the screen analysis procedure and the thermal hydraulic results are given.

2.1 Screen Analysis Procedure

The screen analysis procedure to determine the limiting event with delayed scram is shown in Fig. 1. At first, the analyst evaluates each event qualitatively based on plant operating condition and boundary conditions, and then selects the limiting event with the delayed scram. Next, the analyst performs the thermal-hydraulic analysis for the limiting event using a computer code such as RETRAN-3D. If additional reactor trip (delayed trip) is required to meet the acceptance criteria for the relevant event, the trip signal shall be added and its setpoint shall be determined.

Events related to pressure increase among DBC 2 events are as follows;

- Loss of normal feedwater
- Malfunction of steam generator main feedwater system
- Loss of non-emergency AC power to the station auxiliaries
- Main steam isolation valves closure
- Loss of external load
- Turbine trip
- Loss of load
- Chemistry and volume control system (CVCS) malfunction
- Loss of condenser vacuum
- Total loss of core coolant flow

With respect to system integrity (system peak pressure), LOCV event is selected as representative event among DBC 2 events via the screen analysis procedure.

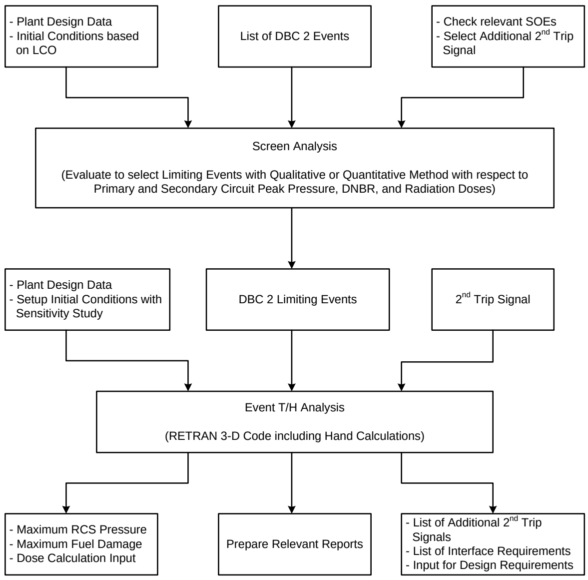

Fig. 1. Screen Analysis Procedure

2.2 Reactor Trip

Primary and secondary reactor trips for an LOCV event are on high pressurizer pressure and low steam generator level, respectively. Therefore, for an LOCV with delayed trip event, the secondary trip is determined as a trip caused by low steam generator level.

2.3 Computer Program

The computer program, the RETRAN-3D [2], is used in the quantitative evaluation of the DBC 2 events with delayed scram. Sections 2.3 through 2.6 provide the relevant information.

RETRAN code is a transient thermal-hydraulic analysis code designed for use in best-estimate evaluation of light water reactor systems, and RETRAN nodalization of the primary and secondary systems and the major components of the EU-APR, which is a modified APR1400 to meet EUR, are used.

RETRAN modeling consists of 137 control volumes including core model (split core) for NSSS major systems and above 190 junctions for connecting each control volume or defining boundary conditions. Trip and control cards are also used to simulate the relevant setpoints and response times of the related reactor protection and control systems.

2.4 Acceptance Criteria

Acceptance criteria on system integrity for LOCV with delayed scram is that the pressure of the item to be protected against overpressure stays lower than 1.1 times the design pressure of the protected item.

2.5 Selection and Preparation of the Input Data

The major initial conditions used are summarized in Table 2.

Table 2. Initial Conditions

Parameters	**Nominal Values**
Core Power Level, MWt	3,983.0
Pressurizer Pressure, MPa	15.51
Core Inlet Temperature, ^{o}C	290.56
Reactor Coolant Pump Flow, kg/sec per pump	5,247.8
Pressurizer Water Level, m^3	34.6
Steam Generator Pressure, MPa	6.895

For LOCV with delayed scram, the failures in safety valves of the protected system are assumed for pilot-operated safety relief valves (POSRVs) for the primary system and MSPOSRVs for the secondary system according to the following rules;

Total number of safety valves	Failed
2 ~ 3	1
4 ~ 8	2
9 ~	One fourth of the number of safety valve

Operator action is not considered in the analysis for 30 minutes after the event initiation. At 30 minutes after the event initiation, the operator, via the appropriate emergency operating procedures, initiates plant cooldown to bring the plant into a safe shutdown condition.

Overpressure analysis for LOCV event is carried out with assumption that all plant systems operate as designed excluding the system failure causing initiating event. The LOCV event is assumed to abruptly and completely terminate both main steam and feedwater. Therefore, feedwater control system (FWCS) is not able to be performed its functions and tasks as planned. All safety control systems and non-safety control systems except reactor power cutback system (RPCS) are not conservatively credited. The loss of offsite power is not assumed.

In order to fulfill the acceptance criteria for pressure during transients, the RPCS is designed as a safety class 3 system. Therefore, the RPCS is credited to perform their functions, and it is assumed to be automatically operated to cutback the reactor power in the early stage of transients and consequently mitigate the RCS overpressure.

2.6 Results

The sudden reduction of steam flow, caused by the LOCV, leads to a reduction of the primary-to-secondary heat transfer. The moderator reactivity is constant prior to reactor trip due to a zero moderator temperature coefficient (MTC), even though the average core temperature increased from the initial conditions. The rapid primary system pressurization and the sudden reduction of steam flow generate RPCS actuation signal through the steam bypass control system (SBCS).

The pressure in primary system increases to high pressurizer pressure trip setpoint (16.39 MPa) as first reactor trip signal which is not credited. The secondary trip signal (45%WR) is actuated later by low steam generator level. Table 3 shows the relevant sequence of event summarizing the major events.

Table 3. Sequence of Event

Time (sec)	Event	Setpoint	Remark
0.00	Loss of Condenser Vacuum	-	
0.17	RPCS Actuation Signal	-	
3.90	MSADV Opens, MPa	7.83	
4.13	High Pressurizer Pressure Trip Setpoint, MPa	16.39	1st reactor trip
6.54	Max RCS Pressure, MPa	17.0	
7.51	Max SG Pressure, MPa	8.04	
19.62	Low SG Level Trip Setpoint, %WR	45.0	2nd reactor trip
20.77	Low SG Trip Signal Generated		
204.65	Aux Feedwater Actuation Signal Generated, %WR	25.0	
1,800	Operator Initiated Plant Cooldown		

Fig. 2 and Fig. 3 show primary pressure for short term and long term, respectively, and the maximum RCS pressure is about 17.0 MPa. Fig. 4 and Fig. 5 show secondary pressure for short term and long term, respectively, and the maximum steam generator pressure is about 8.04 MPa.

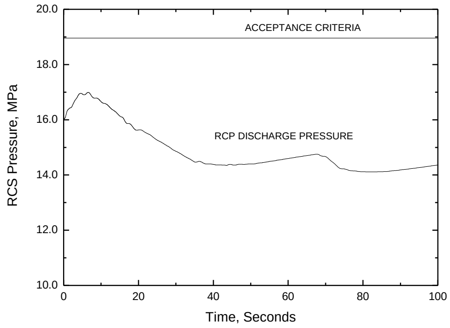

Fig. 2. RCS Pressure vs. Time (short term)

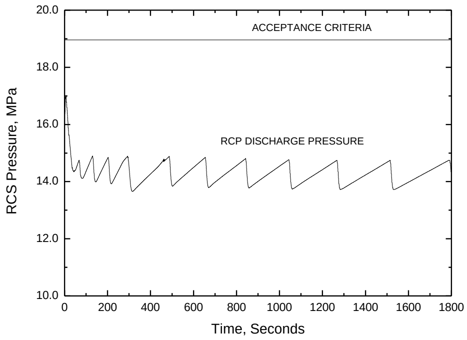

Fig. 3. RCS Pressure vs. Time (long term)

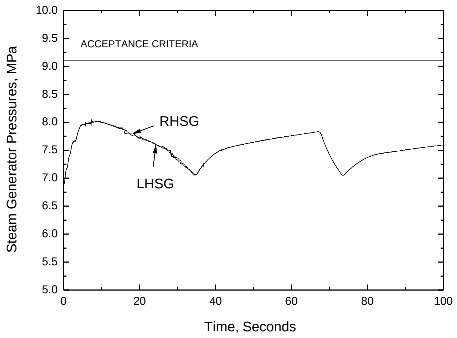

Fig. 4. SG Pressure vs. Time (short term)

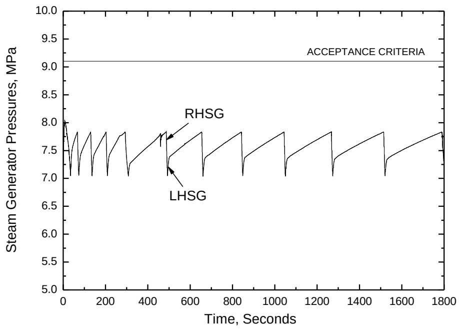

Fig. 5. SG Pressure vs. Time (long term)

3. Conclusions

The analyses performed for the overpressure events indicate that the pressure boundary integrity of the system is ensured by the various mitigating design features such as control systems and safety and relief valves.

For LOCV with delayed scram, the maximum primary and secondary pressures remain below acceptance criteria of 18.97 MPa and 9.07 MPa, respectively, ensuring the primary and secondary system integrities.

REFERENCES

[1] European Utility Requirements for LWR Nuclear Power Plant, Revision D, October 2012.

[2] EPRI NP-7450, "RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," Computer Code Manual, Computer Simulation & Analysis, Inc. and Electric Power Research Institute, December 1997.