Limitation of Over-Pressurrization due to Unexpected Decrease in RCS Heat Removal

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1. Introduction

For PWRs, the primary and secondary pressures should be maintained within the certain limits required for each plant event category. This over-pressure protection requirement is generally accomplished by pressure regulation systems and pressure relief devices of primary and secondary circuit. According to the U.S. NRC requirement [1], the peak pressure of primary and secondary system should be controlled within 110% of design pressure during anticipated operational occurrences (AOOs). On the other hand, those pressures should not exceed the system design pressure without actuation of the safety valves even in the AOOs under the European Utility Requirements (EUR) [2]. The purpose of this paper is to modify the pressure control system for AOOs to comply with the European acceptance criteria and to demonstrate its performance in terms of the overpressure protection.

2. Development of Limitation Systems

2.1 Requirements

The AOOs are conditions which may occur once or more in the life of the plant ($f > 10^{-2}$). These conditions do not propagate to cause a more serious fault or accident, however, at worst, result in a reactor trip with the plant being capable of returning to operation. Pressurization events are the conditions which make the primary and the secondary circuit pressure increases. As the consequences of those events, the integrity of primary and the secondary circuit from the view point of the system pressure may be challenged by over-pressurization. In order to meet the EUR, certain limitation systems, which are the automatic pressure control systems for AOOs as well as normal operations, are necessary to maintain the system pressure within the safety valve operating setpoints.

2.2 Phenomenological Sequences and Scope

The over-pressurization among AOOs may occur for the following reasons:

- Increase in reactor power,
- Decrease in heat removal from RCS, and
- Increase in RCS inventory.

Unexpected reactor power increase can be caused by certain failures in the reactivity control such as uncontrolled control element assembly withdrawal. Unexpected decreases in the RCS heat removal can be caused by various disturbances including reduction or elimination of main steam flow, reduction or termination of feedwater flow, reduction of reactor coolant flow, or changes in feedwater temperature. As an increase of the RCS inventory, Chemical and Volume Control system (CVCS) malfunction may increase the RCS pressure by the Pressurizer Level Control System (PLCS) malfunction with a loss of off-site power resulting in the maximum charging flow with the minimum letdown flow.

In this paper, we only concentrate on the unexpected decrease in the RCS heat removal. After the event initiation, the primary circuit is over-pressurized by expansion of the reactor coolant as its temperature increases or by increase of the RCS inventory. When the primary pressure approaches a prescribed value as a setpoint, the reactor is tripped automatically. Even after the reactor trip, the primary and secondary circuit pressures continuously increase due to decay heat from the reactor core.

Decrease in RCS heat removal related to pressure increase among the AOOs is as follows:

- Partial loss of core coolant flow,
- Inadvertent closure of a Main Steam Isolation Valve (MSIV),
- Total loss of load and/or turbine trip,
- Loss of load and switch-over to house load operation,
- Loss of main feedwater flow to steam generators,
- Total loss of off-site power (< 2 hours), assuming house load operation failure, and
- Loss of Condenser Vacuum (LOCV).

2.3 Selection of the limiting event

The results of the partial loss of core coolant flow event are no more limiting with respect to RCS pressurization than those of the LOCV event because feedwater and steam flow are instantaneously terminated following the LOCV event. Therefore, the decrease in heat transfer from primary to secondary circuit of the LOCV event is much larger than that of the partial loss of core coolant flow event.

The results of the inadvertent closure of a MSIV event are no more limiting with respect to RCS pressurization than those of the LOCV event because steam flow are not completely terminated following the inadvertent closure of the MSIV event.

The results of the total loss of load and/or turbine trip event are no more limiting with respect to RCS pressurization than those of the LOCV event because feedwater flow instantaneously terminates following LOCV whereas it ramps down following the total loss of load and/or turbine trip event.

The results of the loss of load and switch-over to house load operation event are no more limiting than those of the total loss of load and/or turbine trip event, therefore no more limiting than those of the LOCV event due to above reasons.

The results of the loss of main feedwater flow to steam generators event are no more limiting with respect to RCS pressurization than those of the LOCV event because steam and feedwater flow simultaneously terminate following LOCV whereas only feedwater flow terminates following the loss of main feedwater flow to steam generators event.

The results of the total loss of off-site power are no more limiting with respect to RCS pressurization than those of the LOCV event because the plant experiences a complete loss of forced reactor coolant flow at the initiation of the total loss of off-site power event, and the loss of forced reactor coolant flow results in an earlier reactor trip compared to the reactor trip for the LOCV event.

From above reasons, the LOCV event is determined as the limiting event with respect to pressure increase for the AOOs, except the CVCS malfunction event.

2.4 Design Modifications for Limitation Systems

If condenser vacuum is lost, the turbine is assumed to trip immediately on low condenser vacuum and Turbine Bypass Valves (TBVs) connected to condenser are unavailable. The loss of steam flow due to closure of the turbine stop valves results in a rapid increase in the steam generator pressure. A sharp reduction in primary-to-secondary heat transfer follows rapid heatup of the primary coolant and rapid increase in the RCS pressure.

The increase rate of the pressure in primary and secondary circuit for the LOCV is slow due to a normal operation of the various NSSS control. The RPCS would receive a signal from the SBCS to reduce reactor power by simultaneous drop of one or more selected full strength CEA groups into the reactor core. Normal operation of PPCS and PLCS would reduce the increasing rate of the pressure in primary circuit. The MSADVs open to limit secondary circuit pressure.

In order to fulfill the acceptance criteria for the primary pressure during the LOCV, followings are initially considered and assessed to mitigate the over-pressurization.

- Pressurizer Pressure Control System (PPCS)
- Pressurizer Level Control System (PLCS)
- Reactor Regulating System (RRS)
- Reactor Power Cutback System (RPCS)

The Steam Bypass Control System (SBCS) is not considered as a countermeasure against the over-pressurization because this requires a main condenser to be designed as a safety system and it is not feasible. The Feedwater Control System (FWCS) has no direct effect on the pressure control. While the PPCS including the pressurizer spray, the PLCS, and the RRS have little effect on the over-pressure protection, the RPCS is effective to control the peak primary pressure in the event of LOCV hence it needs to be designed as a safety-class 3. It is automatically operated to cutback the reactor power in the early stage of transients and to mitigate the primary system overpressure. The RPCS initiation logic is shown in Fig 1.

In the secondary system, Main Steam Atmospheric Dump Valves (MSADVs) which have earlier opening setpoint than Main Steam Safety Valves (MSSVs) are capable of discharging steam or water directly to the atmosphere when the SG pressure exceeds the pre-determined setpoint.

Two automatic operated MSADVs are installed on each of the four main steam lines upstream of the MSSVs outside the containment to allow cooldown of the steam generators. Each MSADV is capable of holding the plant at hot standby, dissipating core decay heat and reactor coolant pump heat, and allowing controlled cooldown to the Shutdown Cooling System (SCS) initiation conditions when the Main Steam Isolation Valves (MSIVs) are closed, or when the main condenser is not available as a heat sink.

The capacity, setpoint of the RPCS and the MSADV are summarized in Table I.

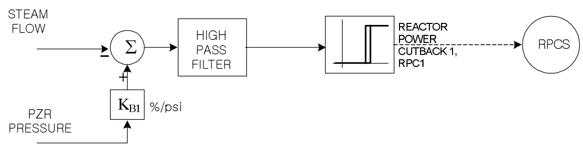

Fig. 1. RPCS Initiation Logic

Table I: Design Data for Safety Analysis

System	No.	Capacity	Delay Time (sec.)
RPCS	-	Bank 5 + 4 (Lead + Second Regulating Bank) 1.01 %$\Delta\rho$	Total : 4.67 Delay Time : 0.5 Full Insertion Drop Time : 4.17
MSADV	4 per SG	Opening Setpoint : 7.832 MPa Closing Setpoint : 7.049 MPa Flow Rate / valve : 204.12 kg/s at 7.832 MPa	Opening Stroke Time : 1.0 Closing Stroke Time : 1.0

3. Analysis and Results

3.1 Initial Conditions and Assumptions

The initial conditions of overpressure protection analysis for the LOCV are assumed to be nominal plant operating values without any additional conservativeness. The initial conditions are summarized in Table. The plant is assumed to be operating at rated power (3,983 MW_{th}) without inaccuracy in power adjustment. The sticking of the most reactive control rod in the reactor scram system is conservatively assumed. The Moderator Temperature Coefficient (MTC) and Doppler reactivity are assumed the value of 0 (10^{-4} $\Delta\rho$/°C) and least negative, the least favorable for LOCV, respectively. The scram rod worth is -8.0 % $\Delta\rho$.

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, FWCS is not credited to perform its functions and tasks as planned. All safety control systems and non-safety control systems except RPCS are not conservatively credited. The loss of offsite power is not assumed.

Table II contains significant conditions and assumptions used for the LOCV event analysis with respect to the primary and secondary circuit overpressure.

Table II: Design Data for Safety Analysis

Parameters	Nominal Values
Reactor Core Power	3,983 MW_{th}
Pressurizer Pressure	15.51 MPa
Core Inlet Temperature	290.56 °C
Reactor Coolant Pump Flow	5,247.8 kg/sec (per pump)
Pressurizer Liquid Inventory	34.6 m^3
Steam Generator Pressure	6.895 MPa

3.2 Codes and Nodalization

The RETRAN is used in the quantitative evaluation of the overpressure protection analyses. The RETRAN code, which is a transient thermal-hydraulic analysis code designed for use in best-estimate evaluation of light water reactor systems, calculates NSSS thermal-hydraulic responses to the initiating events for a wide range of operating condition. The ANS-5.1-1979 decay heat model with +20% uncertainty is used.

Fig. 2 shows the RETRAN nodalization of the primary and secondary systems and the major components of the plant.

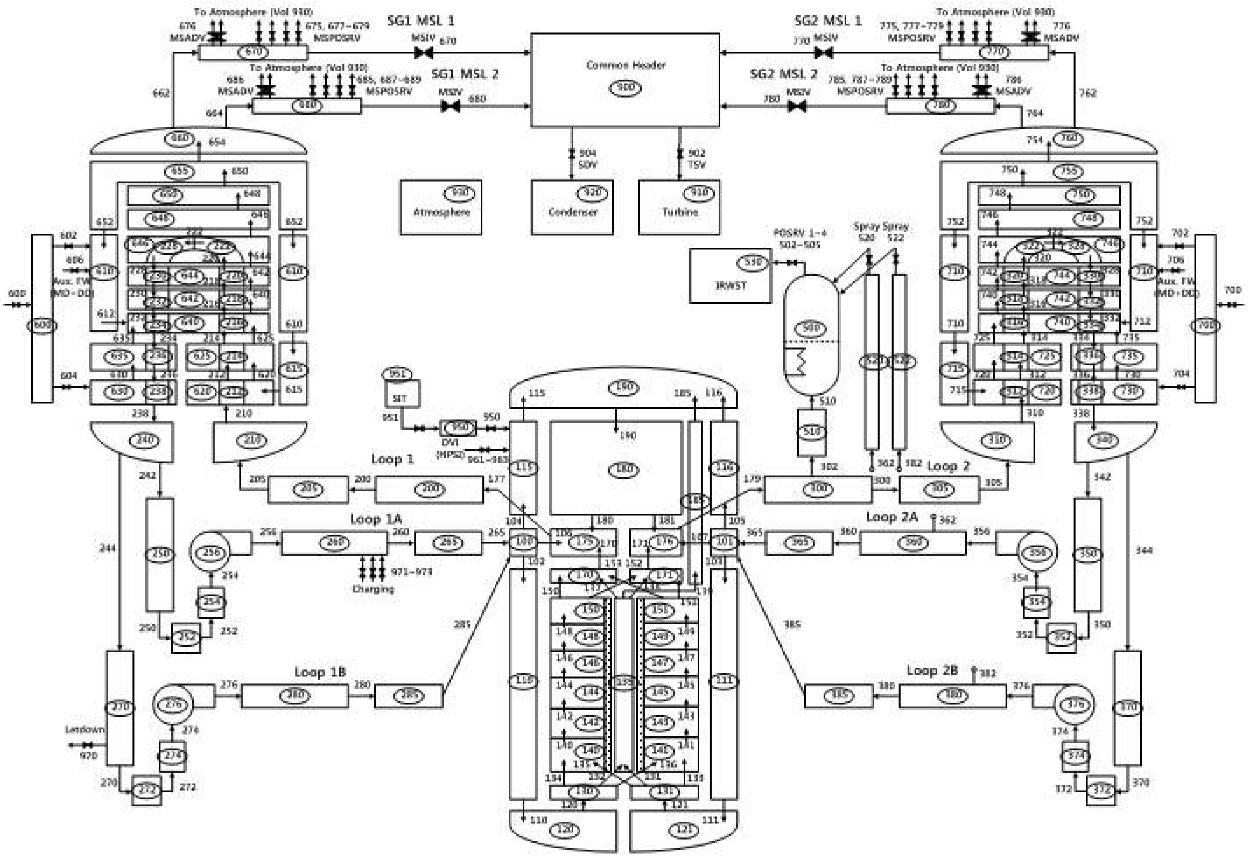

Fig. 2. RETRAN Nodalization for the LOCV Analysis

3.3 Results

The major events occurring during the LOCV are summarized in Table III. The dynamic behaviors in terms of NSSS pressure following the LOCV are presented in Figures 3 and 4.

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 MTC, even though the average core temperature increased from the initial conditions. The rapid primary circuit pressurization and the sudden reduction of steam flow generate RPCS actuation signal through the SBCS at 0.17 seconds. The secondary circuit pressure increases to MSADVs opening pressure at 3.81 seconds. The pressure in primary circuit increases to high pressurizer pressure trip setpoint at 4.13 seconds despite the mitigating design features mentioned above. The reactor trip breakers open at 4.98 seconds, limiting the core power. The maximum RCS pressure of 17.0 MPa is reached at 6.54 seconds and the maximum steam generator pressure of 8.01 MPa is reached at 7.38 seconds. The RCS pressure decreases rapidly due to the combined effects of reactor trip and opening of MSADVs.

Auxiliary feedwater flow initiates automatically at 343.1 seconds. After 30 minutes from the event initiation, the operator initiates a controlled cooldown to shutdown cooling according to the emergency operating procedure.

Table III: Sequence of Event for the LOCV

Time (sec.)	Event	Setpoint or Value
0.00	Loss of Condenser Vacuum	-
0.17	RPCS Actuation Signal by SBCS Generated	-
3.81	Main Steam Atmospheric Dump Valves Open (MPa)	7.83
4.13	High Pressurizer Pressure Trip Setpoint Reached (MPa)	16.39
4.88	High Pressurizer Pressure Trip Signal Generated	-
4.98	Trip Breakers Open	-
6.54	Maximum RCS Pressure (MPa)	17.0
7.38	Maximum Steam Generator Pressure (MPa)	8.01
281.65	Auxiliary Feedwater Actuation Signal Generated (%WR)	25.0
343.10	Auxiliary Feedwater Flow Initiated	-
1,800.00	Operator Initiates Plant Cooldown	-

As shown in Fig. 3, the maximum RCS pressure at the reactor coolant pump discharge is 17.0 MPa which is below acceptance criteria of 17.24 MPa as the design pressure, thus ensuring primary system integrity. As shown in Fig. 4, the maximum steam generator pressure is 8.01 MPa which is below acceptance criteria of 8.27 MPa as the design pressure, thus ensuring secondary system integrity. Consequently, all safety valves in primary and secondary circuit are not actuated.

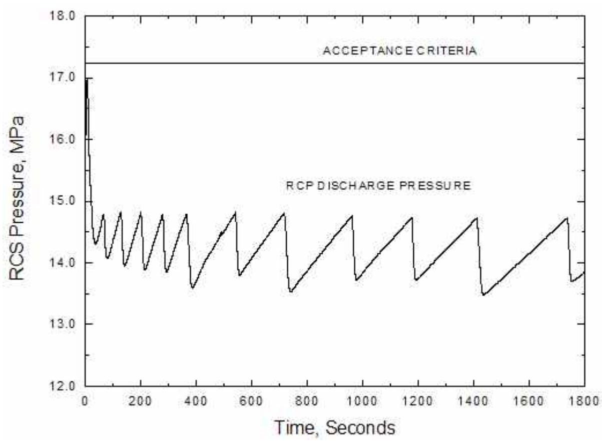

Fig. 3. RCS Pressure vs. Time

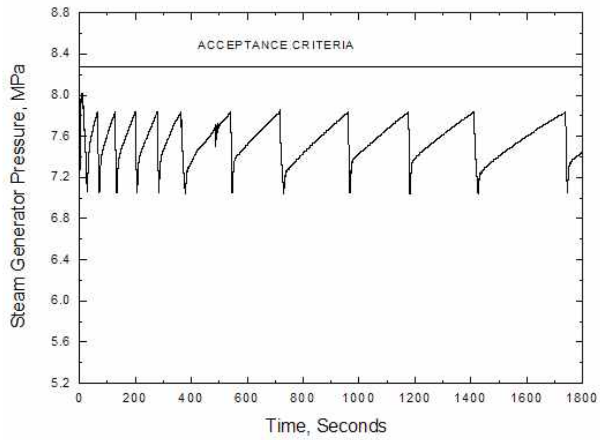

Fig. 4. Steam Generator Pressure vs. Time

4. Conclusions

We have developed the limitation system with the modification of the RPCS for the overpressure protection during AOOs. The limitation system is automatically operated to cutback the reactor power in the early stage of transients and to mitigate the primary system overpressure. The maximum RCS pressure and the maximum steam generator pressure are below acceptance criteria, respectively. It is demonstrated that the primary and secondary circuit pressures can be reduced by means of limitation systems during the over-pressurrization due to unexpected decrease in RCS heat removal.

As a further research, the CVCS malfunction will be assessed from the viewpoint of the over-pressurization reflecting the design changes to comply with the European requirements.

REFERENCES

[1] "U.S. NRC, Standard Review Plan," Section 5.2.2: Overpressure Protection, Rev. 3, NUREG-0800, March 2007

[2] "European Utility Requirement document," Vol. 2 Chapter 2.8 Section 1.1.2.3: Pressure control of core coolant, Rev. D, October 2012.