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Beyond Design Basis Accidents Analysis for iPOWER

Chul Jin Choi*, Jong Cheol Park, Chan Eok Park
Safety Analysis Group, KEPCO E&C 1045 Daeduck-daero, Daejeon 34057
**Corresponding author:cjchoi@kepco-enc.com*

1. Introduction

After the Fukushima accident, the safety became more important to the public acceptance of the nuclear power plant (NPP). An advanced NPP with a passive safety system such as AP1000 has been developed and constructed in many countries. In Korea, an innovative passive optimized worldwide economical reactor (iPOWER) with the enhanced safety and the affordable construction cost has been developing conceptually based on the APR+ design [1]. The iPOWER adopted the passive safety system consisting of a passive emergency core cooling system (PECCS), a passive containment cooling system, and a passive auxiliary feedwater system (PAFS). Two beyond design basis accidents, which are the total loss of secondary heat removal system and the prolonged station blackout, are analyzed to confirm the feasibility of the design. Design changes are identified based on the analysis results.

2. Methods and Results

A realistic analysis method is used to calculate beyond design basis accidents. The realistic RELAP5 computer code is employed. The RELAP5 input such as PECCS and PAFS are also modeled based on the APR+ basedeck.

2.1 Design Description

The PECCS provides cooling water into the reactor coolant system (RCS) to maintain inventory during accidents. The PECCS consists of the hybrid safety injection tank (HSIT), the SIT and the in-containment refueling water storage tank (IRWST). The HSIT is the same with the SIT except the pressure balance piping connecting the HSIT top with the cold leg. The automatic depressurization valves (ADVs) with four stages are installed. The four pilot operating safety relief valves (POSRVs) mounted on the pressurizer top are functioned as the first to third stage ADVs. The last stage ADVs are located at the both hot legs. Since any active components operating with alternating current (AC) power sources are not used in the safety injection system, the failure of the emergency diesel generator system cannot be considered as a single failure. Based on this design features, the PECCS is simplified as the two train system. The direct vessel injection located at the lower vessel than the hot leg level is adopted for effective safety injection as well. The simplified PECCS diagram is shown in Fig. 1. The isolation valves are located at the pressure balance line between the cold leg and the HSIT. The conventional SIT with nitrogen gas is installed as the HSIT.

HSIT injection is activated at low pressurizer pressure or low wide range steam generator (SG) level coincident with high hot leg temperature. The first stage ADV actuation signal is generated at low level in HSIT, the second and third stage ADV are connected to the first stage ADV with time delay. The fourth stage ADVs are actuated at the low-low level in HSIT. The major setpoints are described in Table I.

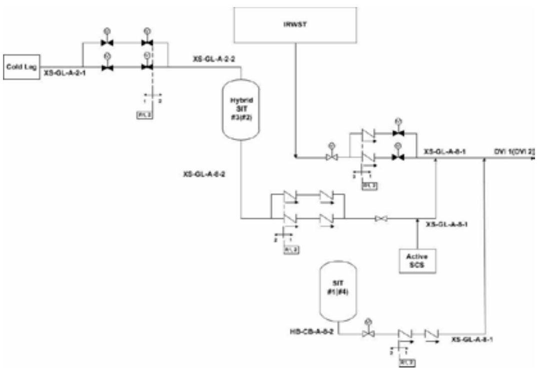


Fig. 1. Simplified iPOWER PECCS Diagram

Table I: Major PECCS Setpoints

Feature	Setpoints
HSIT	RCS Pressure < 10 MPa SG WR Level < 45% & Hot Leg T > 608 K
SIT	RCS Pressure < 4 MPa
IRWST	RCS Pressure < 0.2 MPa
ADV#1	Hybrid SIT Level < 40 %
ADV#2	ADV#1 open with 70s delay
ADV#3	ADV#1 open with 190s delay
ADV#4A	Hybrid SIT Level < 20 % Hot Leg Liquid Level < 3 in
ADV#4B	ADV#4A open with 60 s delay

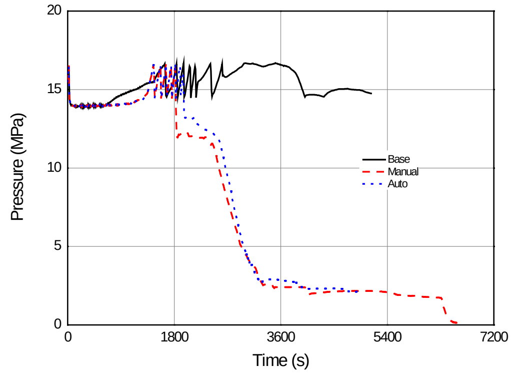
2.2 Total Loss of Secondary Heat Removal System

The total loss of secondary heat removal system (TLOSHRS) event is analyzed to verify the functional diversity of the decay heat removal. All feedwater system and the PAFS are assumed to fail in the accident. Since these coincident failures is very low probable, this accident has been categorized as a beyond design basis accident. However, both this accident and the prolonged station blackout are chosen as accidents to be designed at the similar level to the design basis accidents (DBAs) in accordance with the iPOWER top tier requirements.

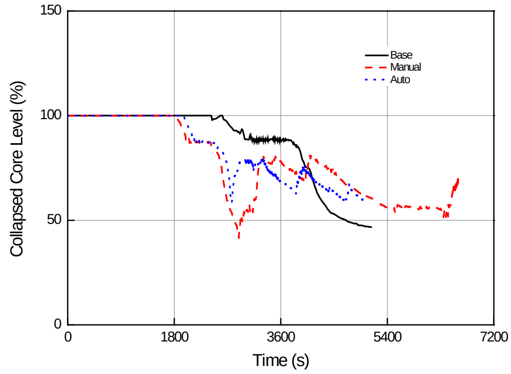
The analysis results are plotted in Fig. 2. Just after the accident initiation, the RCS pressure is decreased, and pressure the SGs is increased. Thereafter, the RCS pressure is also increased until a reactor trip is occurred either due to low SG water level or high pressurizer pressure. The decreased core power, after a reactor trip, is removed by steam effluence through the turbine bypass valves and/or main steam safety valves (MSSVs). The PAFS is automatically initiated at a low SG level for core decay heat removal, however, it is assumed to fail. Since SG water inventory depletion results in RCS pressurization, the POSRVs open just after SG dryout. The core decay heat in combination with POSRV opening maintains the RCS pressure in-between the POSRV opening and closing pressures. The POSRV discharges steam into the IRWST. The HSIT injection is initiated at the low wide range SG level coincident with high hot leg temperature, however, the inventory loss through POSRVs cannot be compensated by this injection flow. Continued depletion of the RCS inventory via the POSRV open/close cycle results in core uncovery, and eventually leads to core damages.

Two cases in addition to the base case are calculated. The manual feed and bleed (F&B) operation is begun by the manual ADV opening at 30 minutes after the reactor trip and the automatic HSIT injection is followed. If the secondary heat removal system is failed, the decay heat can be removed by the safety injection and depressurization. Safety injection provides cooling water to remove decay heat. The HSIT is capable of providing sufficient injection flow. The RCS bleeding is accomplished using the ADVs. The RCS pressure is rapidly decreased to the HSIT injection set point. The earlier HSIT injection than the base case results in SIT injection and eventually IRWST injection although a minor heat-up is occurred before the SIT injection. The manual F&B operation to mitigate a beyond design basis accident is acceptable since the operator action at 30 minutes after the accident initiation is allowed for previous NPPs by the regulatory body.

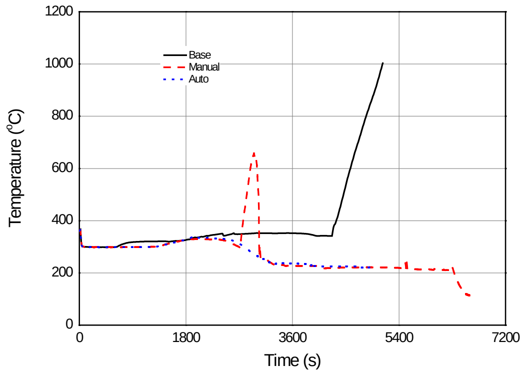
To improve the mitigation system of this accident to the DBA design level, operator actions should be avoided for at least 72 hours after the initiating event. To comply with this requirement, the automatic ADV actuation is assumed at the low wide range SG level coincident with high hot leg temperature. The analysis results as shown in Fig.2 show that the automatic F&B operation is feasible although the ADV opening timing is postponed compared to the manual F&B operation case.



(a) RCS Pressure Transients



(b) Collapsed Core Level Transients



(c) Fuel Cladding Temperature Transients

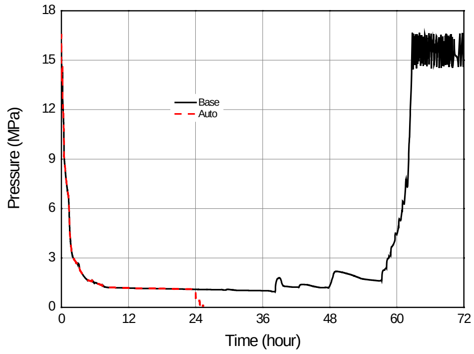
Fig. 2. Thermal-hydraulic Behavior for TLOSHRS

2.3 Prolonged Station Blackout

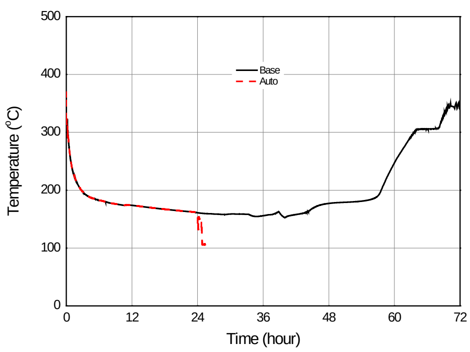
Following the Fukushima accident, concern on the prolonged station blackout (SBO) has been increased. The prolonged SBO scenario involves a loss of offsite power and failures of all diesel generators including alternate AC sources. It is assumed that AC power exists only on the AC buses powered by inverters connected to the station batteries. Loss of all AC power results in unavailability of all normal electrical equipment and most of the safety electrical equipment. The reactor trip and the PECCS are available. This scenario considers the likelihood of loss of offsite power from external events such as earthquakes and flooding.

The coping time for the accident provides the room to implement measures to either restore the failed systems or activate an effective heat removal system. The 72 hours coping capability is adopted for DBAs. The adoption of the same coping capability for the prolonged SBO is reasonable to reflect lessons learned from the Fukushima accident.

Thermal hydraulic behavior is plotted in Fig. 3. Main feedwater pumps and reactor coolant pumps are failed at the same time with the accident because of loss of in-house and offsite power. Core power is rapidly reduced to the decay power level after the reactor trip at the RCS low flow condition. The PAFS is activated at low SG level, and the reduced RCS pressure results in the HSIT injection. The core decay heat is removed by the PAFS. The RCS pressure is stabilized at higher pressure than the IRWST injection actuation pressure, which means the safe shutdown state is not reached. After the PAFS is ineffective, the primary system pressure is increased, eventually, to the POSRV opening pressure. Until at least 72 hours, this condition is maintained and primary inventory is continuously reduced. During this period, the fuel cladding temperature is accordance with the RCS temperature, however, no heat-up is occurred as shown in Fig. 3(b).



(a) RCS Pressure Transients



(b) Fuel Cladding Temperature Transients

Fig. 3. Thermal-hydraulic Behavior for Prolonged SBO

According to the utility requirement [2], the PECCS should be designed so that the safe shutdown state should be achieved within 36 hours and no credit for operator action should be necessary until at least 72 hours following the initiating event. To comply with this utility requirement, an addition of ADV#1 actuation signal on the extended loss of AC power sources, which was used in the AP 1000 design [3], is introduced. The extended duration is determined to be 24 hours corresponding to the battery capability. The loss of AC power signal is based on the detection of an under-voltage condition by the sensors connected to the battery chargers. ADV#1 opening causes ADV#2-4 actuations, followed by the IRWST injection. As shown in Fig.3, the plant can be achieved to the safe shutdown state by the addition of ADV#1 actuation signal.

3. Conclusions

Two beyond design basis accidents were analyzed. Design changes to mitigate automatically the accidents are identified and its feasibility is confirmed by analysis results.

REFERENCES

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[2] Advanced Nuclear Technology: Advanced Light Water Reactor Utility Requirements Document, EPRI, 2014.
[3] AP1000 Design Certification Documents Tier 2, Chapter 7, Revision 19, Westinghouse.