Evaluation of Natural Circulation Cooldown Capability of APR^+

Ju Han Lee *, Je Woo Cho, Myung Jun Song, Chan Eok Park, Jong Joo Sohn
KEPCO E&C Company, Inc., 150 Deokjin-dong, Yuseong-gu, Daejeon, 305-35
**Corresponding author:juhanlee@kepco-enc.com*

1. Introduction

In accordance with the US NRC BTP 5-4 requirement [1], nuclear power plants should have ability to reach the cold shutdown condition after reactor trip through Natural Circulation Cooldown (NCC) only using safety-grade systems powered from onsite or offsite, assuming single failure. In response to the requirements, an analysis has been performed to verify that the NCC capability of Advanced Power Reactor Plus (APR^+), which is a standard plant of 4308 MW thermal power.

2. Methods and Results

2.1 Analysis Methods

For the analysis, the initiating event is assumed to be a loss of offsite power at time zero leading to an assumed loss of power to the reactor coolant pumps thus nearly instantaneous reactor trip. The most limiting single failure is assumed to be a loss of emergency diesel generator.

NCC is governed by decay heat [2], component elevations, primary to secondary heat transfer, loop flow resistance, and void formation. Component elevations in the APR^+ are such that a satisfactory natural circulation flow for decay heat removal is obtained by fluid density differences between the core region and the steam generator tube region [3].

A computer simulation of the NCC analysis of the NSSS was performed with the KISPAC computer code [4]. The KISPAC Code employs a node-and-flowpath approach to model the RCS. For the flow of the Reactor Vessel Upper Head (RVUH) region, the KISPAC code conducts a mechanistic calculation of the flow into and out of the RVUH region under both forced and natural circulation conditions. The NCC analysis was performed from normal full power operating conditions to the conditions that permit the Shutdown Cooling System (SCS) operation. The entire NCC analysis was performed within the requirements of the BTP 5-4.

The flow into and out of the RVUH is very small during forced circulation and even smaller under natural circulation conditions. Following the loss of all forced flow, the RVUH can remain relatively hot and thermally lag behind the remainder of the RCS during a subsequent cooldown. Since the RVUH is at a higher elevation than the RCS, natural circulation induced flow into and out of this region is unlikely. During a cooldown the fluid in the RVUH must either be cooled by heat conduction through the walls or via a feed and bleed operation with the cooler Reactor Coolant System (RCS) fluid. A cooldown procedure utilizing the deliberate draining and re-filling in the upper head region and allowing a RVUH steam bubble formation was selected.

The RVUH is modeled in the KISPAC code as two region non equilibrium thermodynamic control volume. A momentum balance is used to calculate the coolant flowrate into and out of the RVUH and a mass and energy balance is also performed on the RVUH control volume. The state conditions that can exist for the two region model are liquid in both regions (saturated or subcooled); steam (saturated or superheated) over liquid (subcooled, saturated, or two phase); or steam in both regions (saturated or superheated).

The pressurizer is modeled in the KISPAC code as a two region non equilibrium thermodynamic control volume similar to the RVUH. The pressurizer additionally includes a model for the surge line, heat transfer between the pressurizer metal walls and fluid (including heat loss to the containment), main and auxiliary sprays, backup and proportional heaters and pressurizer Pilot Operated Safety Relief Valves (POSRVs). The KISPAC code also includes detailed models of the SIS and the passive auxiliary feedwater system (PAFS) which is used during the NCC event [5].

2.2 Analysis Results

Immediately following the loss of offsite power (and the assumed loss of power to the RCPs), flow through the core decreases as the RCPs coast down. This immediately results in a RCOPS reactor trip on low reactor coolant pump shaft speed. Full natural circulation flow is then established in the RCS in less than 10 minutes.

The plant is maintained at hot standby for four (4) hours consistent with the BTP 5-4 requirements. The operator controls the PAFS start-up valve to maintain the steam generator pressure (Fig.1).

At 4.0 hours after the initiating event (the loss of offsite power), the operator opens pressurizer gas vent valve to depressurize the RCS to the point where the RCS subcooling margin decreases to 15 °C (27 °F). This depressurization is favorable for securing Safety Injection flow to make up the RCS inventory during the subsequent RCS cooldown (Fig.2).

After the depressurization, the operator begins a 27.8 °C/hr (50 °F/hr) cooldown by increasing feedwater flow through PAFS start-up valve (Fig.3).

The RVUH steam void can be detected easily because it is accompanied by an unexpected pressurizer level increase. This steam void could be removed by opening RVUH gas vent valve.

At 10.9 hours after the initiating event, the RCS pressure and temperature reach the shutdown cooling entry conditions of 31.6 kg/cm^{2}A (450 psia) and 176.7 °C (350 °F), respectively.

The amount of safety grade PCCT water used is 1150 m^3 (303,600 gallons). This demonstrates that the NCC to the shutdown cooling entry conditions, according to the BTP 5-4 requirements, can be performed well within the limit of PCCT water capacity (i.e., minimum capacity of 1,666 m^3 (440,000 gallons)) (Fig. 4).

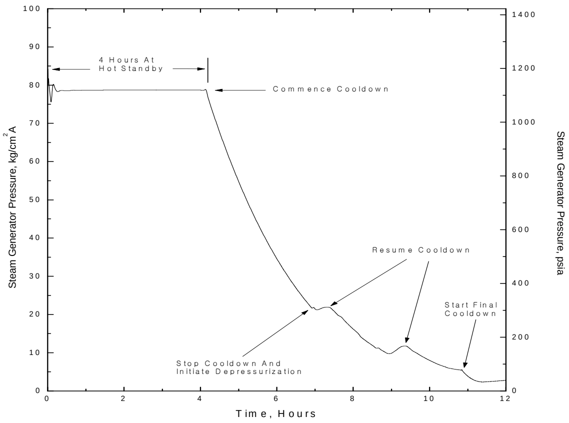

Fig 1. Steam Generator Pressure

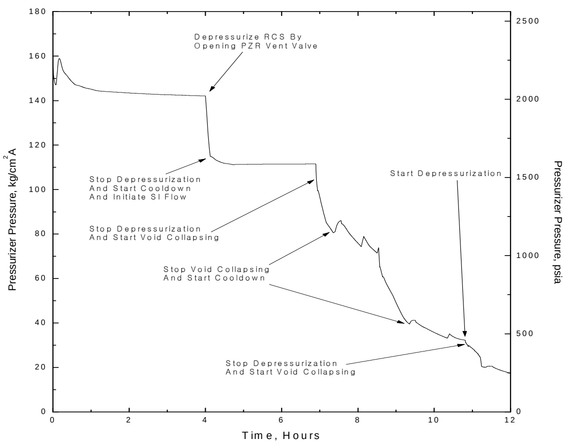

Fig 2. Pressurizer Pressure

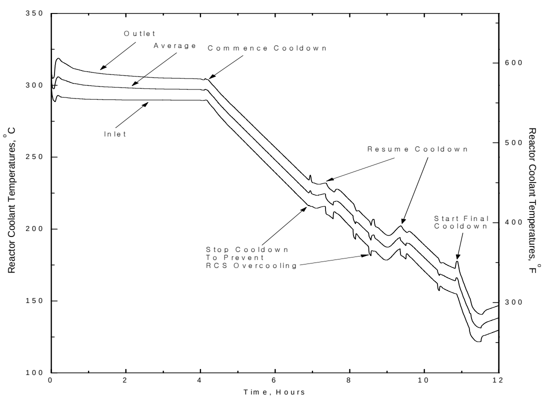

Fig 3. Reactor Coolant Temperatures

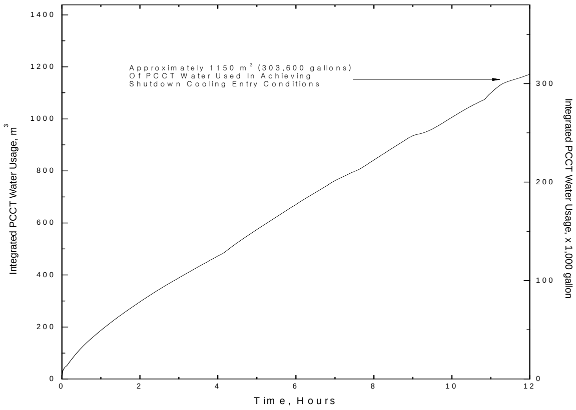

Fig 4. Integrated PCCT Water Usage

4. Conclusions

The NSSS NCC analysis results demonstrate that a cooldown and depressurization to the SCS entry conditions is achievable within the BTP 5-4 requirements. These requirements include the use of only safety grade equipment, the loss of offsite power, a single failure and passive auxiliary feedwater usage within the minimum available capacity. The steam void formed in the RVUH can be easily controlled and eventually collapsed using the reactor coolant gas vent system.

REFERENCES

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[4] Technical Manual for the KOPEC Integrated Systems Performance Analysis Code, Rev. 0, August 31, 1999.

[5] Evaluation Report of Passive Auxiliary Feedwater System Safety, Rev. 0, April 30, 2010, KHNP.