An Evaluation of Kori-1 Natural Circulation Cooldown Capability

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1. Introduction

According to the US NRC BTP RSB 5-1 requirement [1], nuclear power plants can be brought from the normal operation to cold shutdown under the Natural Circulation Cooldown (NCC) condition using only safety-grade systems with only onsite or offsite power available and assuming a single failure.

In response to these requirements, an analysis has been performed to verify the NCC capability of the Kori Nuclear Power Plant Unit 1 (Kori-1) as part of the feasibility study for the unit's extended operation after its allowed lifetime.

2. 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 (RCPs) 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 Kori-1 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.

SG PORVs are used as a means of the RCS cooldown and stabilization by controlling the SG secondary steam release. PZR PORVs are credited as a means of the RCS depressurization because the main and auxiliary PZR spray systems are not designed as a safety-grade system. Auxiliary feedwater system is also used as a safety-grade means of the SG level control to maintain the normal SG water level for NCC analysis [3]. A charging pump is used to control the PZR level and inject the boric acid through the normal charging bypass line or high pressure safety injection line. Reactor vessel upper head vent path is used to control the PZR level as a safety-grade alternative letdown system [4].

The analysis has been performed using the CENTS computer code [5]. The CENTS code is an interactive computer code for simulation of the NSSS and related systems. It calculates the behavior of a PWR for normal and abnormal conditions including accidents. It is a flexible tool for PWR analysis which gives the user complete control over the simulation through convenient input and output options. The control systems for reactivity, level, pressure and steam flow are simulated as well as the balance of plant systems.

The analysis consists of a series of hot standby, cooldown, and depressurization phases of NCC to RHR entry conditions. The main purpose of this analysis is to maximize the auxiliary feedwater usage in order to confirm that the designed auxiliary feedwater source is enough for use.

3. Simulations and Results

Figure 1 through 5 show the analysis results of NCC event occurred in full power condition.

Immediately following the loss of offsite power, RCPs trip and begin to coast down. The flow through the core decreases and results in reactor trip. Shortly after the reactor trip, SG pressure increases to the SG PORVs setpoint, and then the operator manually controls the SG PORVs to stabilize the NSSS at hot standby conditions.

The plant is maintained at hot standby for 4 hours consistent with the BTP RSB 5-1 requirements. The operator controls SG PORVs to maintain the steam generator pressure (Figure 4) and thus the RCS cold leg temperature (Figure 1).

During the cooldown process, the operator performs a 18 °F/hr cooldown by increasing steam flow through SG PORVs. As the RCS starts to cooldown, the pressurizer level and pressure decrease, and the operator controls charging and letdown flow as necessary to maintain pressurizer level (Figure 3). The charging flow provides inventory control during the cooldown, adds negative reactivity, and increases the subcooling of the RCS.

During depressurization process, the operator stops cooldown to prevent RCS overcooling, opens the PZR PORV to depressurize RCS.

At 17.1 hours after the event initiation, the RCS pressure and temperature reach the RHR entry conditions of 384 psia and 330 °F, respectively.

4. Conclusion

As an NCC analysis result of Kori-1, the amount of auxiliary feedwater used is estimated to be about 153,000 gallons (Figure 5). This demonstrates that the NCC from normal operation to RHR entry conditions, per the BTP RSB 5-1 requirements, can be performed well within the minimum available capacity of 200,000 gallons.

It is concluded that Kori-1 can be cooled and depressurized to RHR entry conditions in conformance with the restrictive assumptions of US NRC BTP RSB 5-1.

ACKNOWLEDGEMENT

This work has been carried out under the Project funded by the Korea Hydro & Nuclear Power Co., Ltd. (KHNP).

REFERENCES

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5. "Technical Description Manual for the CENTS Code," Rev. 0, December 2002.

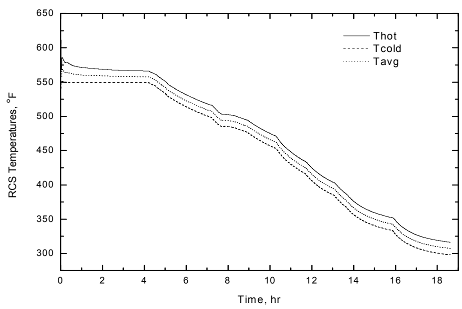

Figure 1. RCS Temperature

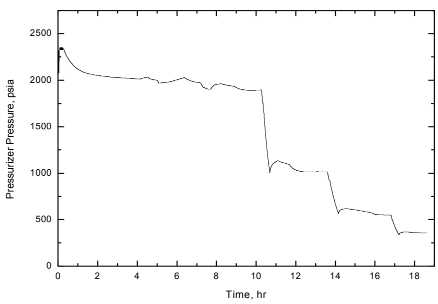

Figure 2. Pressurizer Pressure

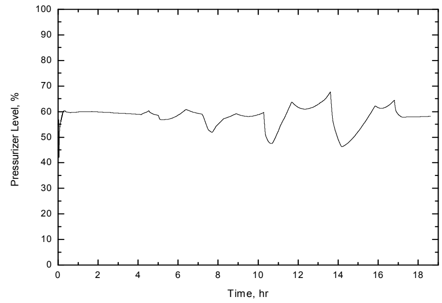

Figure 3. Pressurizer Level

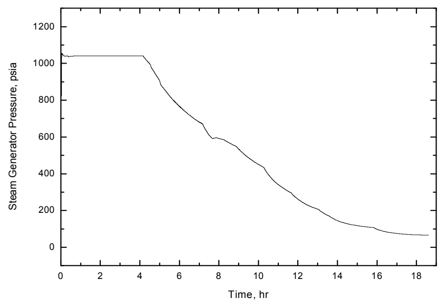

Figure 4. Steam Generator Pressure

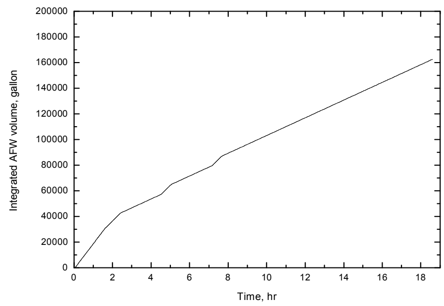

Figure 5. Integrated Auxiliary Feedwater Flow