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A Comparison of SKN 3&4 NCC simulations using SPACE and a current design code

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1. Introduction

The Natural Circulation Cooldown (NCC) analysis of APR1400 was performed using a design code, KISPAC[1] for SKN 3&4 safety analysis. The analysis followed the process and the analysis results met the requirements specified 'US NRC Branch Technical Position RSB 5-1'[2]. In this paper, the NCC analysis results of the design code were compared to those of Korean PWR system safety analysis code, SPACE[3]. This comparison could help the code developers to modify the safety code for use of performance analysis.

2. Definition, Assumptions and Sequence of NCC Analysis

Natural circulation cooldown is an operation process of bring a post-trip nuclear power plant into SCS entry conditions using only safety-grade equipment. The NCC is formed by decay heat in the core, elevation difference of Reactor Vessel (RV) and Steam Generator (SG), primary to secondary heat transfer, Reactor Coolant System (RCS) loop flow resistances and void formation of Reactor Vessel Upper Head (RVUH). A natural circulation flow in the RCS loop in the event of a reactor trip can be achieved by a coolant density difference between RV and SG. As a safety-grade means of RCS cooldown, Atmospheric Dump Valves (ADVs) in the secondary system are used as a heat sink.

NCC analysis is performed under the following assumptions.

- Only safety-grade equipment is used concurrently with a loss of offsite power and a single failure(one diesel generator fail leading to 50% capacity of SIS pumps).
- Plant is initially at full power steady state conditions with normal process parameters.
- The initiating event is the loss of offsite power(LOOP), followed by an assumed loss of power to the Reactor Coolant Pumps (RCPs) that causes the Core Protection Calculator (CPC) reactor trip on low RCP speed, thus resulting in a turbine/generator trip.
- The most negative moderator and Doppler coefficients are used.

These assumptions are applied into SPACE to the maximum extent possible.

The sequence of NCC analysis adopted into both analyses is as follows.

(1) Following the reactor trip, the operator manually controls the ADVs to restore and maintain the secondary pressure to no-load hot standby conditions.
(2) The SG water level is restored, and then maintained at the normal water level by manually controlling the auxiliary feedwater flow rate.
(3) After four hours hot standby period, the RCS is depressurized by using the Pressurizer Gas Vent Valves(PGV) until the RCS subcooling reaches the minimum limit value of 15 ℃ (27 °F).
(4) A cooldown is initiated with 27.8 ℃/hr (50 °F/hr). This rate is slower than the administrative maximum cooldown rate of 41.7 ℃/hr (75 °F/hr).
(5) The operator stops the cooldown and initiates depressurization whenever the RCS subcooling increases to a pre-determined subcooling margin 83.3 ℃ (150 °F).
(6) The operator utilizes the RVUH vent valves to reduce the volume of steam void. This induces colder reactor coolant fill up the RVUH region and thus depressurizes the RCS.
(7) The operator performs cooldown process again until the RCS subcooling margin reaches the maximum value after collapsing steam void by RVUH venting .
(8) Repeat (6)~(7) until SCS entry conditions are achieved with PZR level kept within 30% to 70%.

3. Differences between KISPAC and SPACE codes

Figures 1,2 are the APR1400 nodalizations for KISPAC and SPACE simulations, respectively. The KISPAC has fixed nodalization of CE type plants like OPR1000 and APR1400, while SPACE has the flexibility in building nodes and flow paths. For this reason, SPACE is able to have fine nodalization for typical PWRs such as OPR1000 and APR1400 and this feature can enhance the reliability of simulation results. In NCC analysis, the number of nodes and flow paths used in SPACE is about five times than those in KISPAC.

The KISPAC has only 3 governing equations for liquid field problem solving while SPACE has 10 governing equations for solving two-fluid, three-field thermo-hydraulic problems. Even though KISPAC does not have the ability of treating two phase flow, it does not fail when void is formed in the RVUH region. This is because KISPAC has a special RVUH model which allows void formation. In SPACE, the two phase flow behavior in RVUH may delay code calculation and cause problems if it is not treated well.

4. Simulation Results

Figures 3 through 5 show the comparison of the simulation results of the two codes. The pressure behavior of PZR(figure 3) explains that a stepwise NCC was well simulated with the initial 4 hour of hot standby and two stpes of depressurization and cooldown process and ended with 4 MPa pressure which is near at the SCS entry condition.

The pressure of SG is shown in Figure 4. The pressure was at almost constant value for the hot standby period and then went down sharply during the first period of cooldown. This means that RCS cooldown by ADVs was well simulated in both codes.

The PZR level is shown in Figure 5. During the first 4 hours of hot standby, the level was kept almost constantly. The cooldown of RCS caused the coolant contraction and this brought the level down to 30% level which should be increased above 30% to avoid PZR empty. This could be done with having sufficient SIS mass addition to RCS. more steps of depressurization and cooldown. The level was rapidly increased at 5.3 hour due to void formation in RVUH region.

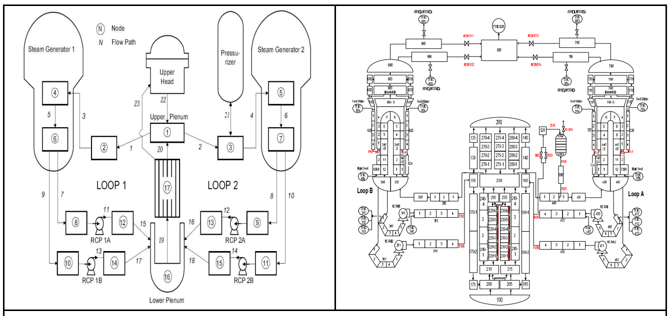

Figure 1. Nodalizations

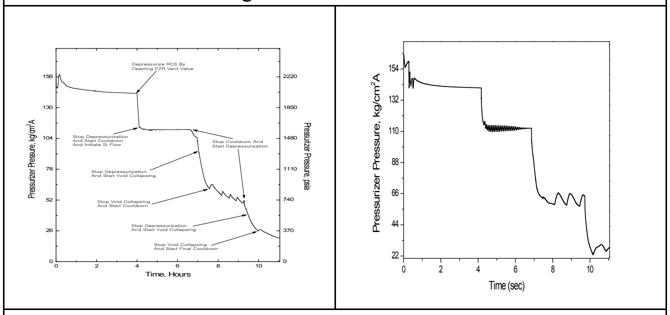

Figure 2. PZR Pressure

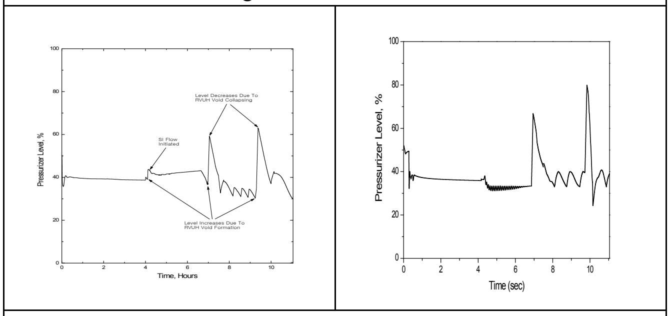

Figure 3. PZR level

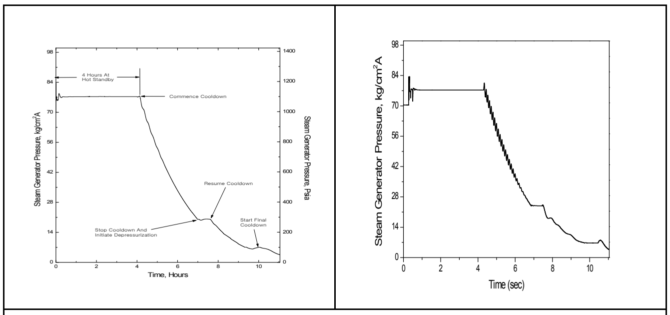

Figure 4. SG Pressure

5. Conclusions

The NCC process for APR1400 was simulated using the KISPAC and SPACE codes. In both codes, major steps of NCC process, such as hot standby, cooldown and depressurization, were achieved fairly well with anticipated properties at each phase. The throttle control of SI flow during the second cooldown was found only in KISPAC simulation, but it seems to be possible in SPACE code by using control functions.

So far, the SPACE code needs much bigger time than the KISPAC code for the simulation of NCC. This is because the SPACE code has more equations to slove and needs more nodes and flow paths for simulation of typical power plants. If appropriate reduction of the equations and approximation into simple nodalization are properly developed, faster calculation with good agreement could be achievable with the SPACE code. 1400.

Acknowledgment

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REFERENCES

[1] KOPEC, “Development of the Plant Analyzer for Korean Standard Nuclear Power Plants” (2000).

[2] US NRC, "Design Requirements of the Residual Heat Removal System," US NRC Branch Technical Position RSB 5-1, Rev. 2, July 1981.

[3] S. J. Ha, C. E. Park, K. D. Kim, and C. H. Ban, “Development of the SPACE Code for Nuclear Power Plants”, Nuclear Technology, Vol. 43, No. 1, 2011.