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Best Estimate Small Break LOCA Analysis for KNGR SIS Optimization

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Abstract

The KNGR has an advanced ECCS design feature which employs four mechanically-separated SI trains where each train consisting of one HPSI pump and one SIT injects ECC water directly into the reactor vessel downcomer annulus. To demonstrate that the KNGR ECCS design features meet the EPRI ALWR requirements of no core uncovery for a break of up to 6 inch diameter, small break LOCA cases with various break sizes were analyzed using a best-estimate analytical procedure. Two kinds of break locations are considered: cold leg and DVI line breaks. It was observed that the KNGR ECCS design can tolerate a cold leg break of up to 10 inches with no core uncovery. However, since DVI line break with 6 inch diameter undergoes slight core uncovery, further investigation is required for KNGR SIS optimization.

1. Introduction

The Korean Next Generation Reactor (KNGR), which is rated at 3914 MWt, is an advanced pressurized light water reactor. In the KNGR plant, the safety injection system (SIS) employs four mechanically-separated SI trains where each train consisting of one HPSI pump and one safety injection tank (SIT). The emergency core cooling water coming from the in-containment refueling storage tank (IRWST) and SIT is injected directly into the reactor vessel downcomer annulus instead of the cold leg injection adopted for the existing plants. Also low pressure safety injection (LPSI) pumps are removed in the SIS. These characteristics of KNGR ECCS may influence the system response to a postulated LOCA.

The purpose of this analysis is to confirm that the KNGR ECCS can maintain the core covered with two-phase fluid throughout the transient for a break of up to 6 inch diameter. As allowed in the EPRI URD [1], best-estimate analytical procedures and assumptions are employed for this analysis. Several small break LOCA cases are analyzed to verify that the KNGR ECCS design with the advanced design features will satisfy the requirements of no fuel damage specified in the EPRI ALWR requirements document [1].

2. Analysis Procedures

For the analyses presented, best-estimate input decks of CEFLASH-4AS/REM [2] and PARCH/REM [2] were prepared. The CEFLASH-4AS/REM [3] is a best-estimate thermal-hydraulic computer code, which has been designed for realistic analysis of Small Break LOCA (SBLOCA) transients. The code predicts the general thermal-hydraulic transient behaviour of the primary and secondary system in response to small break LOCA. Figure 1 shows the nodalization scheme of CEFLASH-4AS/REM for KNGR SBLOCA analysis. The main aspects of the analysis assumptions are summarized below, which are the conservative "best estimate" assumptions:

1) The plant is assumed to be operating at 100% of rated core power (3914 MWt).
2) The reactor trip is assumed to occur when low pressurizer pressure trip signal is generated at 1825 psia.
3) All reactor coolant pumps (RCPs) are assumed to trip at the same time when the reactor trip occurs.
4) The charging pump flow is not credited in more conservative manner.
5) The safety injection is assumed to start when safety injection actuation signal (SIAS) is generated at 1825 psia. Arithmetic average values of maximum and minimum HPSI pump delivery data were used. In case of cold leg breaks, four HPSI trains and four SITs are credited. In case of DVI line breaks, however, three HPSI trains and three SITs are credited because the ECC water delivered to the broken DVI line is assumed to be directly spilled into the containment.
6) The YGN 3&4 decay heat curve including the actinide decay and uncertainties for the first 4 cycles [4] was used.
7) The Henry-Fauske and Homogeneous Equilibrium Model is used for the break discharge model.

The PARCH/REM [3] is a best-estimate hot rod heatup computer program, which also has been developed for realistic fuel temperature calculations for a small break LOCA transient. The code predicts the heatup behaviour of hot rod using the core conditions (core pressure, core mixture level, saturated liquid mass in the core, core inlet flow, and core inlet enthalpy) generated from CEFLASH-4AS/REM calculation.

3. Analysis Results and Discussions

3.1 Cold Leg Breaks

For cold leg breaks, break sizes ranging from 3 inch (0.05 ft^2) to 10 inch (0.55 ft^2) diameter were investigated. The analyzed cases are: 0.05, 0.1, 0.15, 0.2, 0.25, 0.3, 0.35, 0.4, 0.5, and 0.55

ft^2. Core uncovery is not predicted for all break cases up to 10 inch diameter break as shown in Figure 2. It was observed that, among the break cases analyzed, a 0.2 ft^2 (6 inch diameter) cold leg break attains the lowest minimum core mixture level. The safety injection flow is determined by the primary system depressurization, which depends on the inventory loss through the break. Therefore, for small break LOCA, the core mixture level is determined by the competing effect of inventory loss through the break and coolant makeup by safety injection flow. For cold leg breaks smaller than 0.2 ft^2 break, inventory loss through the break is small enough for HPSI flow to keep the core remain covered. For cold leg breaks larger than 0.2 ft^2 break, the rapid primary system depressurization resulted in large amount of HPSI and SIT water being injected into the RCS.

3.2 DVI Line Breaks

For DVI line break, break sizes ranging from 3 inch (0.05 ft^2) to 8.5 inch (0.4 ft^2) diameter were investigated. The analyzed cases are: 0.05, 0.1, 0.15, 0.2, 0.25, 0.3, 0.35, and 0.4 ft^2. As shown in Figure 2, DVI line breaks smaller than 0.2 ft^2 (6 inch diameter) break size maintains the core covered throught the transient. However, DVI line breaks larger than 0.2 ft^2 break size undergoes core uncovery. The detailed transient behaviour for 0.2 ft^2 DVI line break is described below.

Following the break, the RCS pressure decreases rapidly (Figure 3). When the pressurizer pressure reaches 1825 psia, reactor trip and safety injection actuation signals are actuated. The reactor trip occurs at 12.72 seconds with time delay of 1.15 seconds. Simultaneously with the reactor trip, offsite power is assumed to be lost. Therefore, RCP trip occurs, causing the vessel flow to drop rapidly (Figure 4). At total time delay of 40 seconds after the SIAS, HPSI flow starts at 51.58 seconds (Figure 5).

After reactor trip, the RCS pressure drops to a typical plateau slightly above the secondary system pressure (Figure 3). It stays on the plateau until the break uncovers. As the break is uncovered at 185 seconds, steam is well removed through the break, causing the RCS pressure to fall below the secondary system pressure, which in turn increases the SI flow into the RCS (Figure 5).

After the complete coast-down of RCPs, the SG U-tubes and hot legs begin to drain by reverse flow due to the RCP head loss. During this process, spike drop in the core mixture level occurs (Figure 6). Since the SI flow is still smaller than the break flow, the core mixture level decreases again. The core mixture level reaches the top of core at 405 seconds and is maintained at this level until the SIT flow starts at 441 seconds (Figure 5). However, Since the core mixture level remains near the top of core, fuel heatup is not expected to occur.

For DVI line breaks larger than 0.2 ft^2 break, since the HPSI flow cannot make up for the inventory loss due to the large break flow, core uncovery occurs for a significant period of time. Figure 7 shows the coolant and cladding temperature of the axial node 19 (hottest node) predicted by PARCH/REM for 0.4 ft^2 DVI line break. The first peak before the reactor trip is due to the increase in core power by the positive reactivity insertion from the decreasing moderator density. Since the film boiling occurs on the cladding surface due to excessive power, the heat transfer coefficient drops suddenly as shown in Figure 8. The second peak occurs for a short period time when the hot node uncovery occurs. The third peak occurs during the long period of core uncovery time due to the deficiency of HPSI capacity. The fuel heatup starts at 135 seconds and the maximum cladding temperature is calculated when the SIT water is injected at 205 seconds. Since the predicted peak clad temperature is around 1100 ^{o}F which is well below the limiting temperature of 2200 ^{o}F specified in 10CFR50.46, fuel damage is not expected to occur.

4. Conclusions

The core two-phase mixture level stays above the top of core up to the 10 inch and 6 inch diameter of cold leg breaks and DVI line breaks, respectively. For DVI line breaks larger than 6 inch, fuel heatup occurs due to the core uncovery.

In conclusion, it can be demonstrated that the KNGR ECCS design can tolerate a six inch diameter of Small Break LOCA with no core uncovery for cold leg breaks. However, since a DVI line break of 6 inch diameter undergoes slight core uncovery, further investigation is required for KNGR SIS optimization.

References

1. ALWR Utility Requirement Document, Chapter 5 : Engineered Safeguards Systems, Section 2.3.9, April 1987.
2. CEN-373-P, Vol.3, "Realistic Small Break LOCA Evaluation Model : Computer Program Input and Output Description," Nuclear Fuel Engineering, C.E., April 1988.
3. CEN-373-P, Vol.1, "Realistic Small Break LOCA Evaluation Model : Calculational Models." Nuclear Fuel Engineering, C.E., April 1988.
4. D.E. Uhl, "Decay Heat for YGN 3/4 Feed and Bleed Analysis," Y34-FE-0021, June 1992.

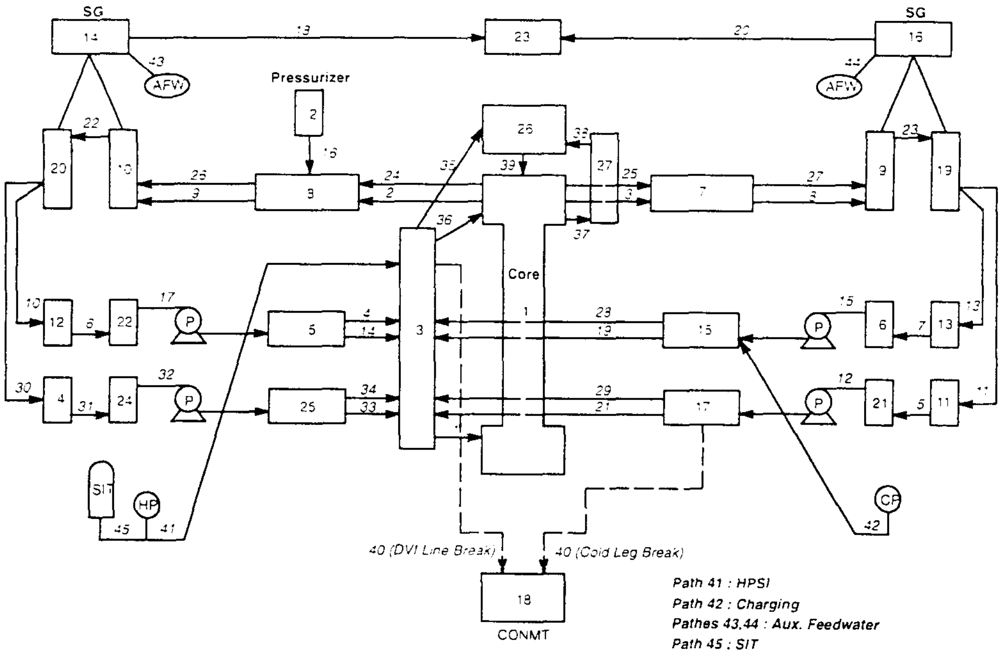

Figure 1. Node Diagram of CEFLASH-4AS/REM for KNGR SBLOCA Analysis

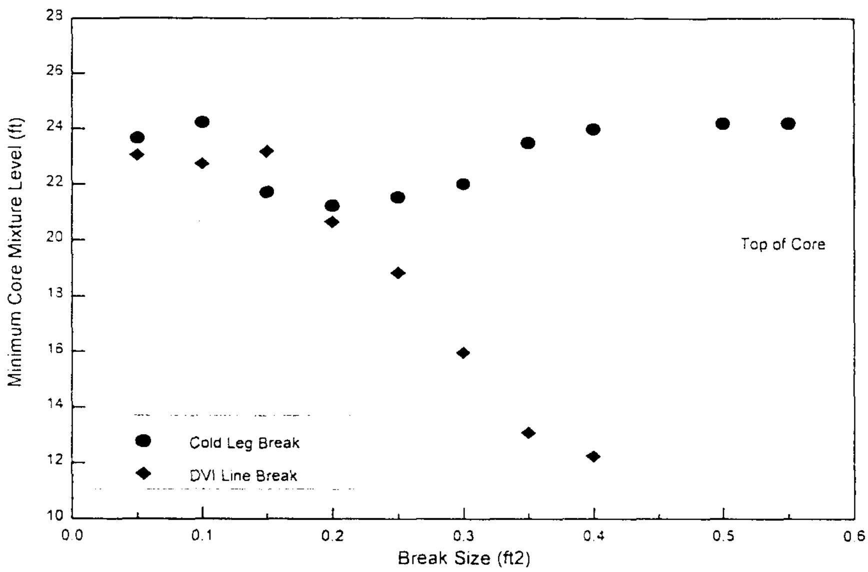

Figure 2. Minimum Core Mixture Level As a Function of Break Sizes Analyzed

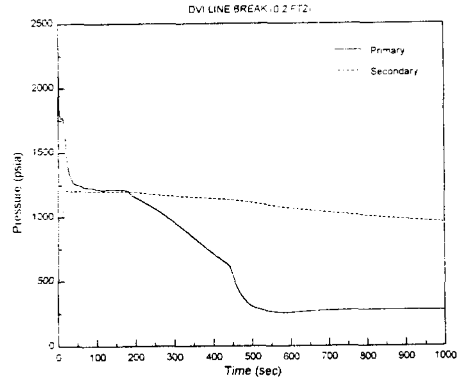

Figure 3. Primary and Secondary System Pressure (Node 1 and 14)

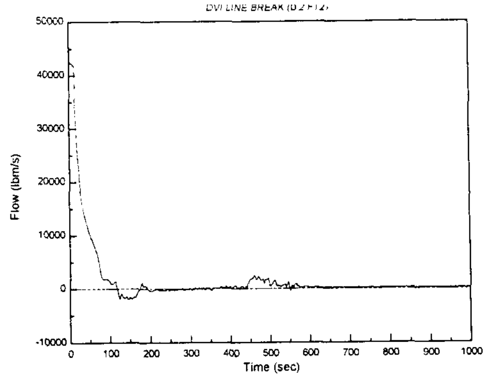

Figure 4. Core Inlet Flow (Path 1)

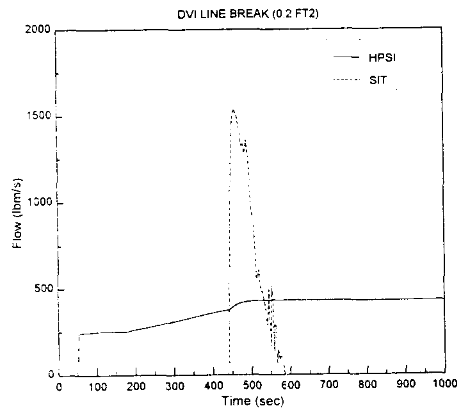

Figure 5. Total HPSI and SIT Flow (Path 41 and 45)

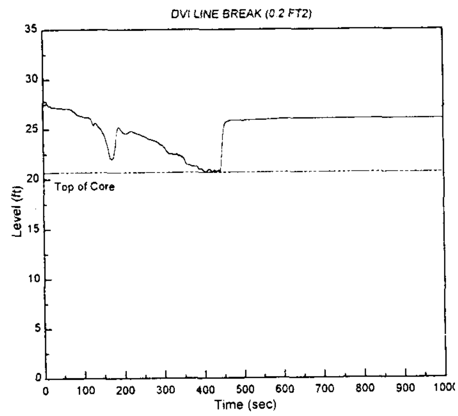

Figure 6. Core Mixture Level (Node 1)

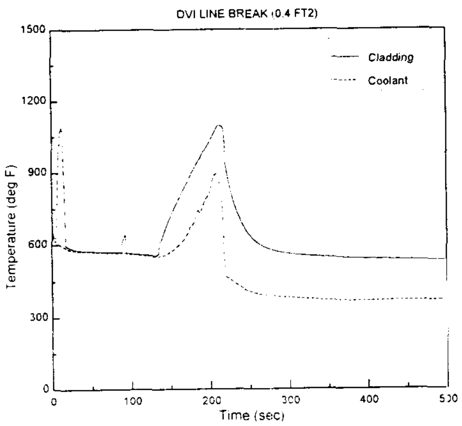

Figure 7. Maximum Cladding Surface and Coolant Temperature (PARCH/REM)

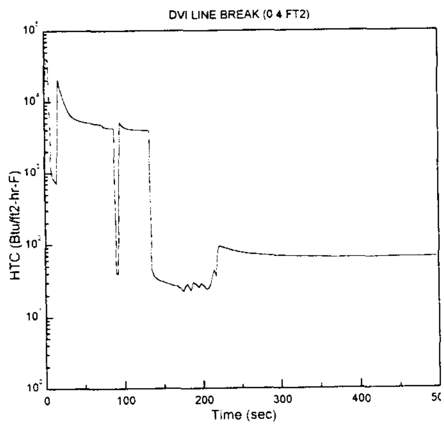

Figure 8. Cladding Surface Heat Transfer Coefficient (PARCH/REM)