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Analysis of Loss of Offsite Power Transient Using RELAP5/MOD1/NSC;

II: KNU1 Design-Base Simulation

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RELAP5/MOD1/NSC를 이용한 원자력 1호기 외부전원상실사고해석

Ⅱ 설계기준사고

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Abstract

The KNU1 (Korea Nuclear Unit 1) loss of offsite power transient as a design-base accident has been simulated using the RELAP5/MOD1/NSC computer code. The analysis is carried out using the best-estimate methodology, but the sequence and its assumptions are based on the evaluation methodology that emphasizes conservatism. Important thermal-hydraulic parameters such as average temperature, steam generator level and pressurizer water volume are compared with the results in the KNU1 Final Safety Analysis Report(FSAR). The present analysis gives much lower RCS average temperature and pressurizer water volume, and much higher S/G water volume at the turn-around point, which may be considered to be additional improved safety margins. This is expected since the present analysis deals with the best-estimate thermal-hydraulic models as well as the initial conditions on a best-estimate basis. These additional safety margins may contribute to further validate the safety of the KNU1 in this type of accidents(Decrease in Heat Removal by the Secondary System).

요 약

원자력 1호기의 설계 기준 사고인 외부 전원 상실 사고를 열. 수력학적 최적 계산용 코드인 RELAP5/MOD1/NSC를 사용하여 모의하였다. 본 분석은 최적 계산모델로 수행 되었으나, 사고 전개 및 가정등 보수성을 갖는 평가 방법에 의거하였다. 해석결과중 노심평균온도, 증기발생기 및 가압기 수위등의 중요한 열·수력학적 변수를 원자력 1호기의 최종 안전성 분석보고서의 결과와 비교하였다. 본 해석결과에서 노심평균온도와 가압기 수위는 보다 낮게, 증기발생기 수위는 보다 높게 나타남으로써 더 향상된 안전한계치를 확인하였다. 이것은 본 해석에서 최적 열·수력 모델을 사용하였을 뿐

만 아니라 초기치로써 최적 값을 택하였기 때문에 얻어지는 결과이며, 또한 이와 같은 유형의 사고 (2차 계통의 열제거 능력 상실 사고)에서 원자력 1호기의 안전성을 더욱더 입증시켜 주는 것이다.

1. Introduction

Continued efforts and research activities toward the safer operation of nuclear power plants have brought significant advances in the technology of nuclear safety analysis. These advances made it possible to identify and resolve many technical uncertainties, and hence to re-examine lower safety margins imposed by the conservative approach. The additional margin, made available through the implementation of the advanced technology, can be utilized to increase the overall economics of nuclear plants.

The implementation of the technology was realized through the development of the advanced system thermal-hydraulic computer codes based on the best-estimate methodology as described in the preceeding paper(Kim, et al.; 1986b). It is indicated that the capability of the code be confirmed within a reasonable range in the application to accident analyses.

Much attention was addressed to the assessment of the advanced, best-estimate T/H code, RELAP5/MOD1/NSC, in the preceeding papers dealing with the plant transient analysis on the loss of normal feedwater(Kim, et al.; 1986a) as well as the loss of offsite power(Kim, et al.; 1986b). These analyses confirmed that the code responds well to the plant transient and has a higher capability to be applied to this type of accidents(Decrease in Heat Removal by the Secondary System).

Therefore, a best-estimate thermal-hydraulic code, the RELAP5/MOD1/NSC, is here applied to the analysis of the KNU1(Korea Nuclear Unit 1) loss of offsite power accident as a design-base accident. The accident sequence and the boundary conditions of the present analysis are almost identical to those of the evaluation methodology specified in KNU1 Final Safety Analysis Report(FSAR) which emphasizes conservatism. However, the initial conditions are based upon the best-estimate methodology. Important thermal-hydraulic parameters such as reactor coolant system(RCS) average temperature, steam generator(S/G) level and pressurizer(PZR) water volume were compared with the results in the KNU1 Final Safety Analysis Report. The main objective of the present analysis is to identify the safety margin which may be additionally deduced by the application of the best-estimate methodology in stead of the evaluation methodology to this type of design-base accident.

2. Sequence Description

KNU1 is a 587 MWe two-loop PWR(Pressurized Water Reactor) consisting of a reactor vessel, two inverted U-tube S/Gs, two water-sealed reactor coolant pumps(RCPs), a pressurizer and various inter-connecting pipings. Each coolant loop is designated loop-A and loop-B, and the pressurizer is attached to loop-A.

A loss of offsite power transient, which may be caused by a multitude of failures, is characterized by a reduction in the secondary heat removal capability. If the reactor were not tripped properly, the core damage could result from the sudden loss of heat sink. Even the proper reactor trip may not ensure the safety of the reactor if the auxiliary feedwater were not supplied to the S/G, in which case the residual heat would heat the primary system to the point where water relief from the pressurizer occurs, thus leading to the core damage. Then the main focus of the loss of offsite power transient analysis, as a design-base, is to identify the capability of the secondary system to remove the residual heat, and hence prevent the water relief from the primary system.

Assumptions used in the present analysis are almost identical to those of the evaluation methodology, specified in the KNU1 FSAR, as follows;

i) Reactor/turbine trip is actuated by the S/G low-low level signal, and at the same time, both reactor coolant pumps begin to coastdown.

ii) Only one auxiliary feedwater pump, delivering a full capacity of 13.5 kg/sec to one S/G, is available one minute after the reactor/turbine trip.

iii) Only the self-actuated safety relief valves (SRVs) are available for secondary system steam relief.

iv) The pressure in the primary system is controlled only by the self-actuated safety relief valves.

The maximum relieving capacity of the primary and the secondary SRVs are 87 kg/sec at 17.24 MPa and 514.7 kg/sec at 7.89 MPa, respectively.

3. Code and Input Model Description

The system simulations were performed using the RELAP5/MOD1/NSC computer code and the detailed description of the code as well as the nodalization scheme for KNU1 is given in the preceeding paper (Kim, et al.; 1986b) of this series. Both SRVs in the secondary system which were not modeled in the preceeding nodalization is now added as shown in Fig. 1.

Steady-state calculation at full power operation was performed and the results are identical to those of the preceeding paper. The uncertainties in the initial conditions, that were incorporated in the FSAR analysis, were not considered in the present simulation since the present analysis deals with the best-estimate initial values, i.e. the most probable values, in the system performance.

However these differences in the initial conditions were found to give no significant effect on

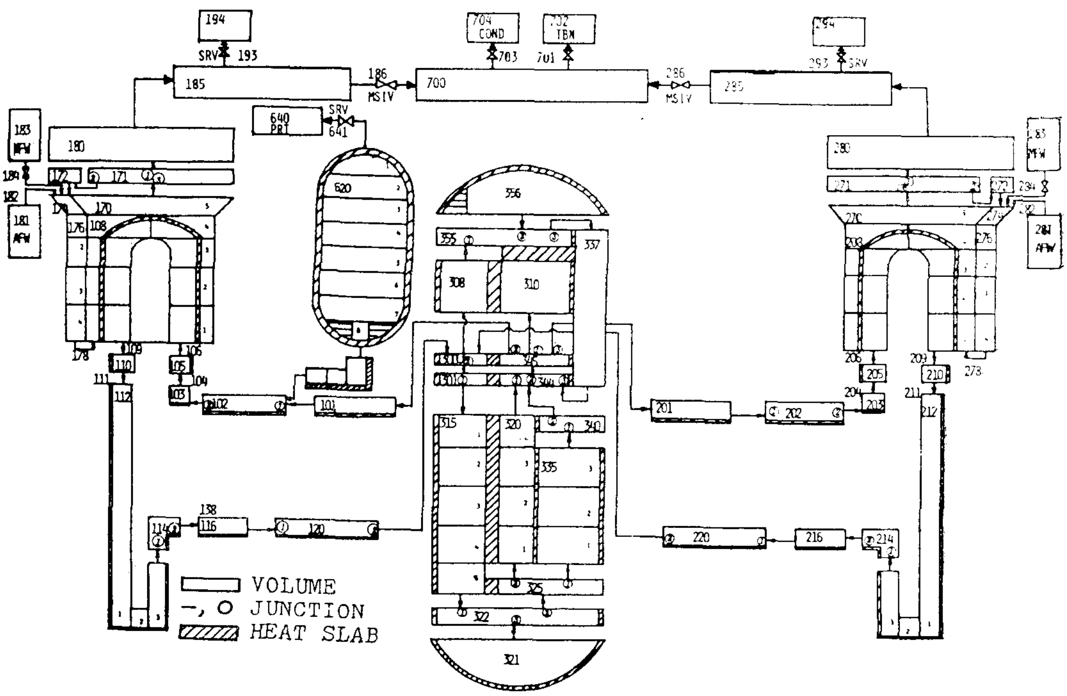


Fig. 1. RELAP5/MOD1/NSC Nodalization for KNU1 Accident Analysis

the results during the progression of the accident.

4. Results and Discussion

Present analysis was performed based upon the initial and the boundary conditions described in sections 2 and 3. The initiating events and the major simulated events during the progression of the transient are summarized in Table 1. The simulated thermal-hydraulic parameters are shown in Figs. 2~10. The key safety parameters in this type of accident, such as the S/G level, RCS average temperature and the PZR level etc., are compared with the results in the FSAR which used the LOFTRAN code(Burnett, et al.; 1972) for the calculations.

The rapid decrease in the S/G level, shown in Fig. 2(the ordinate represents the distance from the bottom of the U-tube sheet), is initiated by the loss of main feedwater. This decrease causes the reactor/turbine to trip by the S/G low-low level signal and both RCPs are also tripped at the same time. Following the turbine trip, the fast closure of the turbine stop valves causes the steam flowrate to drop(Fig. 3) leading to the S/G pressure increase (Fig. 4)

The S/G pressure increases up to the SRV setting value(7.51 MPa) of the lst class SRVs, thus causing them to open. The steam relief through the SRVs(Fig.5) provides the secondary

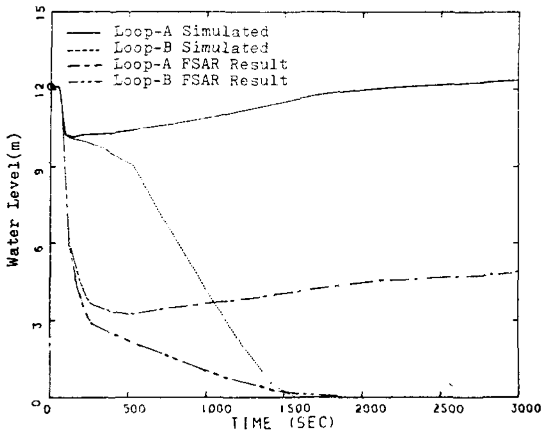


Fig. 2. S/G Water Level vs. Time

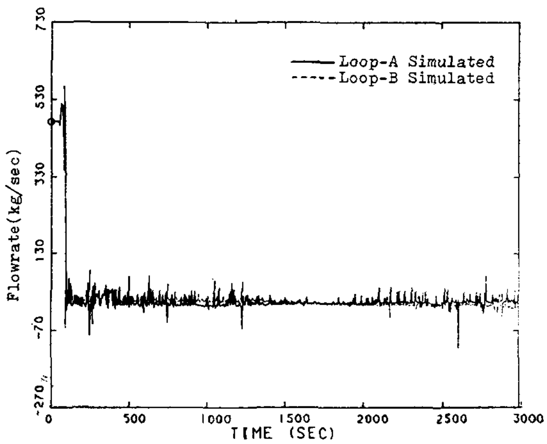


Fig. 3. S/G Steam Flowrate vs, Time

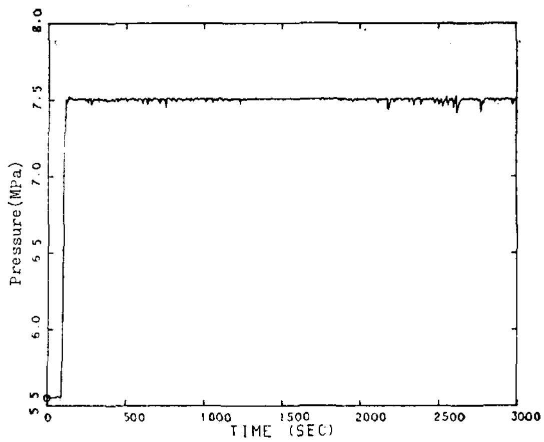


Fig. 4. S/G Steam Pressure vs. Time

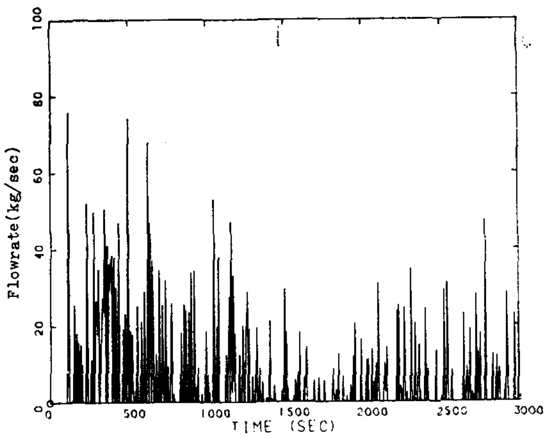


Fig. 5. Steam Flowrate Through S/G Steam Safety Relief Valve vs. Time

Table. 1. Sequence of Events for KNU1

Time(sec)	Simulated Events(FSAR)	Simulated Events(RELAP5)
0.0	—100% Power Operation	—100% Power Operation
50.0	—Low-Low S/G water level —Reactor trip —Reactor coolant pumps begin to coastdown	—Loss of main feedwater
92.75		—Low-Low S/G water level —Reactor Trip, Turbine trip, Reactor coolant pumps begin to coastdown
110.0	—One S/G begin to receive aux. feedwater from one motor-driven pump	
112.75		—One S/G begin to receive aux. feedwater from one motor-driven pump
152.0		—Full rated aux. feed-water is taken
1,600.0		—S/G-B dryout occurs
1,700.0		—PZR SRV open
2,050.0	—S/G-B dryout occurs	
2,350.0	—Peak Tavg occurs —Peak PZR water level occurs	
2,800.0		—Peak Tavg occurs
3,000.0		—Peak PZR water level occurs

system with additional capability enough to remove the residual heat and then the pressure remains at that setting value. This steam relief through the SRVs and the reduction of the S/G void fraction due to the abrupt pressure jump further accelerate the decrease in S/G water level initially caused by the loss of main feedwater discussed above(Fig.2).

Since the auxiliary feed, actuated 60 seconds after the reactor trip as indicated in Table 1, is supplied to S/G-A only, the S/G-A water level slowly rises throughout the whole transient period, whereas the level in S/G-B continues to decrease resulting in the complete uncovery of the S/G U-tube sheet at 1500 second. The uncovery of S/G-B reduces the heat removal capability in contrast to the S/G-A which retains the capability.

Referring to Fig. 2, one can note that, even though the phenomenological trend and the S/G-B dryout time($\sim$1500 sec) are quite similar in both cases(simulated and FSAR), the S/G-A water level in the present simulation with the best-estimate methodology is 6.2m higher than that of the FSAR results based on the evaluation methodology. This indicates that, in this type of accidents(Decrease in Heat Removal by the Secondary System), the higher S/G water level can be considered as providing a larger safety margin. Therefore, in a sense, the best-estimate calculation further validates the safety of KNU1.

The loop coolant flowrate, shown in Fig. 6, experiences a large and abrupt drop following both RCP trips and decreases smoothly until the stable natural circulation is fully established. Because the natural circulation is driven by the hot-cold leg temperature difference, it strongly depends on the heat removal capability of the secondary side. Therefore, approximately at 1500 second when the U-tubes of S/G-B become completely uncovered, the decrease in the heat removal capability of the secondary side raises the loop-B cold leg temperature, thus causing the hot-cold leg temperature difference to become smaller. This leads to the decrease in the loop-B flowrate as shown in Fig.6, while the loop-A

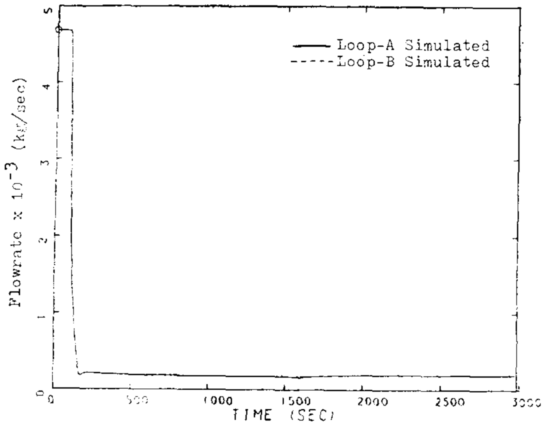


Fig. 6. Loop Coolant Flowratervs. Time

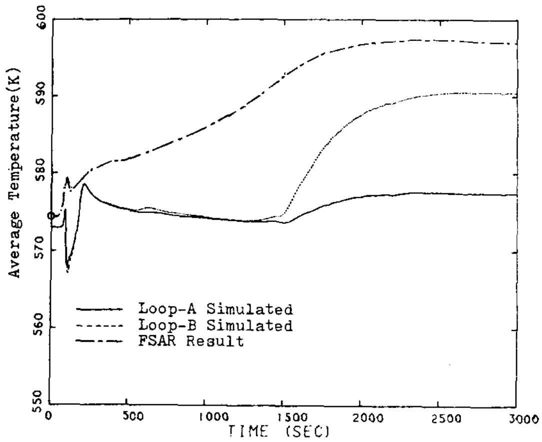


Fig. 7. Loop Coolant Average Temperature vs. Time

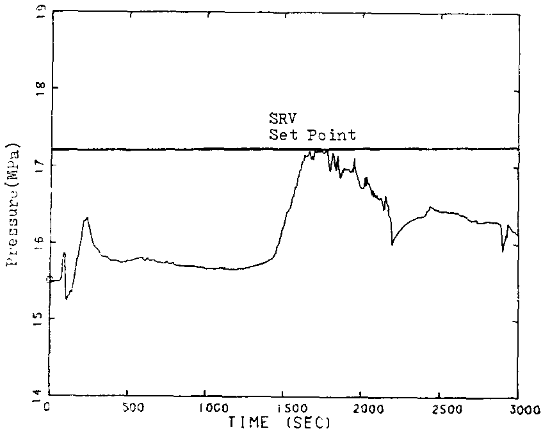
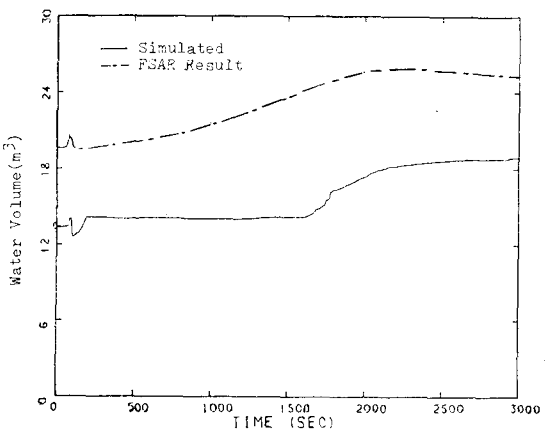


Fig. 8. Pressurizer Pressure vs.



Time Fig. 9. Pressurizer Water Volume vs. Time

flowrate remains constant. Furthermore, the reduction in loop-B flowrate subsequently causes the RCS average temperature to increase(Fig. 7) and also brings about the increase in the PZR pressure and level(Fig. 8 and Fig. 9) due to thermal expansion.

As can be seen in Fig. 7, the RCS average temperature increases at the initial stage of the accident due to the decrease in the secondary heat removal capability brought about by the loss of main feedwater. The rapid drop in the reactor power following the reactor trip leads to the drop in the RCS average temperature. The subsequent RCP coastdown raises the RCS average temperature until the heat removal capability, due to the steam flow through the S/G SRVs together with the auxiliary feedwater, overcomes the residual heat. The RCS average temperature then slowly decreases as the core decay power decreases.

As the U-tubes of S/G-B become completely uncovered at 1500 second, the heat transfer mode in S/G-B changes from pool boiling mode to superheated natural convection mode, resulting in the rapid decrease in heat removal capability and consequently the loop-B cold leg temperature rises. This increase in cold leg temperature as well as the reduction in natural circulation flowrate as discussed in Fig. 6, brings about the increase in loop-B coolant average temperature.

Because of the thermal mixing of coolants of both loops in the core, the rise in loop-B coolant temperature affects loop-A coolant temperature to rise slightly. This re-heating of the reactor coolant causes the pressurizer pressure to increase up to the SRV set point.

The RCS average temperature continues to rise until the heat removal capability, improved by the actuation of PZR SRVs and by the increase in S/G-A water level, becomes balanced with the core decay power at around 2500 second. As can be seen in Fig. 7, at the time when the plant equilibrium state is reached(2500 sec), the best-estimate analysis results in 13K lower RCS average temperature than the FSAR result. One can note that this also validates the safety of KNU1, as well as the higher S/G water level (Fig.2).

Pressurizer Pressure variation also shows similar trend to the RCS average temperature, discussed in Fig. 7, up to the SRV actuation time (Fig. 8). Afterwards, although the RCS pressure is controlled by the actuation of SRV to remain below the SRV set point, the PZR pressure slowly decreases as the sub-cooled water surges from the RCS loop to the pressurizer. The pressurizer volume(water level x area) shown in Fig. 9 also reflects the phenomena as discussed in Fig. 7. As can be observed from Fig. 10, the increase in the pressurizer level following the thermal expansion of the RCS, causes the water/steam inteface to rise. That is, the interface, which was in volume ⑤ initially, rises to volume ④ near 1300 second and, due to the continued rise of the interface, the liquid void fraction of volume ④ also continues to increase until it reaches 0. 62 when the plant condition is in near equilibrium condition. The existence of interface in volume ④ has an important meaning in that the fluid through the SRV is purely steam throughout the whole transient.

Back to Fig. 9, one can observe that the

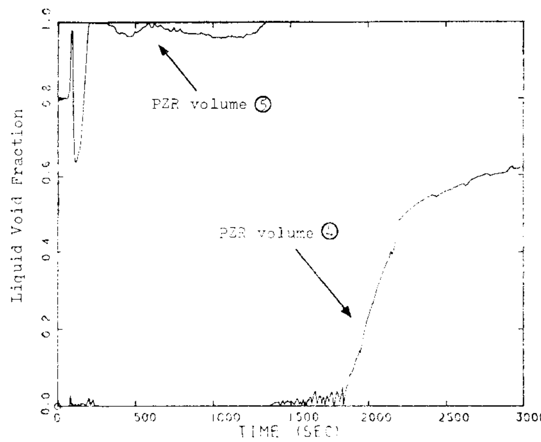


Fig. 10. Liquid Void Fraction at PZR Volume with Liquid-Steam Interface vs. Time

pressurizer water level obtained from the present best-estimate methodology is lower than the FSAR result. It can also be noted that the lower PZR water contributes to further validate the safety of KNU1 as well as the S/G water level (Fig. 2) and the RCS average temperature(Fig. 7).

5. Conclusion

Analysis of KNU1 loss of offsite power accident, as a design base, has been performed based on the sequence and the assumptions deduced from the evaluation methodology which emphasizes conservatism in contrast to the present best-estimate methodology.

The important thermal-hydraulic parameters, such as RCS average temperature, steam generator water level and pressurizer water volume, are compared with the results shown in the KNU1 Final Safety Analysis Report. The present analyses give much lower values in RCS average temperature and PZR water volume, and much higher value in S/G water level at the turn-around point, which may be considered to be additional improved safety margins. This is expected since the present analysis deals with the best-estimate thermal-hydraulic models and initial conditions on a best-estimate basis. The additional safety margin deduced from the present best-estimate analysis may, in a sense, contribute to further validate the safety of the KNU1 in this type of accidents (Decrease in Heat Removal by the Secondary System).

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