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《Original》

Study on the Steam Line Break Accident for Kori Unit-1

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고리 1호기에 대한 증기배관 파열사고 연구

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Abstract

The steam line break accident for Kori Unit 1 is analyzed by a code SYSRAN which calculates nuclear power and heat flux using the point kinetics equation and the lumped-parameter model and calculates system transient using the mass and energy balance equation with the assumption of uniform reactor coolant system pressure.

The 1.4 ft^2 steam line break accident is analyzed at EOL (End of Life), hot shutdown condition in which case the accident would be most severe. The steam discharge rate is assumed to follow the Moody critical flow model. The results reveal the peak heat flux of 38% of nominal full power value at 60 second after the accident initiates, which is higher than the FSAR result of 26%.

Trends for the transient are in good agreement with FSAR results. A sensitivity study shows that this accident is most sensitive to the moderator density coefficient and the lower plenum mixing factor. The DNBR calculation under the assumption of $F_{\Delta H}=3.66$, which is used in the FSAR with all the control and the shutdown assemblies inserted except one B bank assembly and of $F_Z=1.55$ shows that minimum DNBR reaches 1.62 at 60 second, indicating that the fuel failure is not anticipated to occur.

The point kinetics equation, the lumped-parameter model and the system transient model which uses the mass and energy balance equation are verified to be effective to follow the system transient phenomena of the nuclear power plants.

요 약

SYSRAN code를 사용하여 고리 1호기의 증기배관파열사고를 분석하였다. SYSRAN code는 중성자출력과 열선속계산은 각각 점근사 중성자 운동방정식과 집중정수 모형을 이용하고 냉각수 계통 과도현상에 대해서는 전 계통을 균일한 압력으로 취급하여 질량 및 에너지 평형방정식을 이용하여 계산한다.

사고 결과를 심각하게 만드는 노심상태로 부냉각재 온도계수가 커지는 노심말기와 증기발생기의 유체함량이 가장 많은 고온 정지상태를 초기조건으로 하여, 격납용기외부의 가장 큰 배관면적인 1.4ft^2

크기의 증기배관이 파열되었을때 Moody critical flow model에 따라 증기가 방출된다고 가정하여 분석하였다. 그 결과 노심의 최대 열선속은 사고후 60초에 정상상태의 38%로서 FSAR의 26%에 비해 높은 값을 나타냈으나 모든 과도현상의 경향은 FSAR의 결과와 잘 일치하였다. 민감도 조사결과 이사고는 냉각재밀도 계수와 노심 하부공간혼합인자에 가장 민감한 것으로 나타났다.

B bank중 한 개의 RCCA가 완전인출 상태에서 노심에 삽입되지 않았다고 가정했을 경우의 FSAR 분석결과인 $F_{\Delta H}$를 3.66으로 F_Z를 1.55로 하여 DNBR을 계산해 본 결과, 최소 DNBR은 1.62가 되어 핵연료의 손상은 예상되지 않았다.

점근사중성자 운동방정식, 집중 정수모형 및 질량과 에너지평형방정식을 이용한 계통 과도 현상모델은 발전소 전 계통의 과도 현상의 경향을 연구하는데 적합한 것으로 밝혀졌다.

I. Introduction

The steam line break accident corresponds to ANS(American Nuclear Society) condition IV[1] and it must be analyzed as design basis accident.

The steam line break causes the reactor to cool down rapidly and core power would increase in the presence of negative moderator density coefficient. In this case if the most reactive rod cluster control assembly is stuck in its fully withdrawn position after reactor trip, the core has the increased possibility to become critical and return to power. Extremely, fuel damage is also anticipated to occur in circumferences of the stuck assembly due to the increased power peaking factor. The core is ultimately shut down by the boric acid injection delivered by the safety injection system.

When nuclear power plant is designed, it must be demonstrated that the reactor is able to be safely shut down assuming a single failure[2] in the engineered safeguards.

In this paper, the steam line break accident for Kori Unit 1 is analyzed by a code SYSRAN[3] and the results are compared with those of FSAR.[4]

SYSRAN code simulates two reactor coolant loops including two steam generators and the associated systems. It also simulates the reactor kinetics, the reactor control and protection system, safeguards system and other subsystems.

Minimum DNBR is calculated using the single channel thermal hydraulic analysis code, SCAN[5] to predict the probability of the fuel damage.

Sensitivity study is also performed for some parameters which are expected to give comparatively large effects to the transients.

II. Accident description and Protection system

The steam line contains the steam of 1,020 psia, 547°F at zero power and the 805 psia, maximum moisture content 0.25% steam swifts with the flow rate of 7.51×10^6 1b/hr at full power.

The steam release arising from a break of a main steam line would result in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The increased steam flow will remove energy from the reactor coolant system and reduce the coolant temperature. Especially the core has a very large negative moderator temperature coefficient at EOL, hot shutdown condition. In such a condition the cooldown of the reactor core results in a very large positive reactivity insertion into the core. If the most reactive rod cluster control assembly is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will become critical and return to power.

A return to power following a steam line

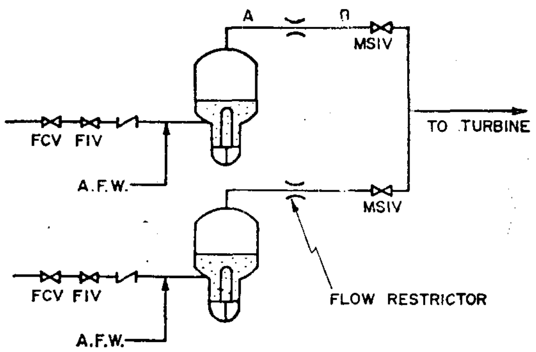

Fig. 1. Schematic of the Main Steam System.

break is a potential problem mainly because of the high power peaking factors which exist assuming the most reactive rod cluster control assembly to be stuck in its fully withdrawn position.

Fig. 1 shows the secondary system of Kori Unit 1. Main steam line discharge area is 4.6 ft^2. Flow restrictor limits the discharge rate to 1.4 ft^2 for the excessive steam discharge. To stop the steam release of intact loop, the isolation valves are established as shown in Fig. 1. If the break occurs at B position in Fig. 1, closing of the isolation valves will quit the steam release from the intact steam generator.

If the break occurs between the isolation valve and the turbine, the cooldown will stop with the isolation valves colsing. Main steam line isolation valves close in 10 seconds by Hi-hi containment pressure,Hi steam line flow rate or Steam line low pressure. Reactor is tripped by Overtemperature ΔT trip, Overpower ΔT trip or Safety injection signal.

Steam line low pressure (525 psia), Pressurizer low pressure(1750 psia), or Containment hi pressure (20 psia) initiates the safety injection system. Safety injection signal gives rise to reactor trip and then 20,000 ppm boron solution comes into the core through the cold legs which insures the shutdown of the reactor.

The core cooldown will be accelerated if the feedwater is supplied continuously.So the safety injection signal closes not only all of the feedwater isolation valves, but also trips the main feedwater pumps. This stops the additive cooldown of the reactor. Auxilliary feedwater is supplied to remove the decay heat after the accident.

These protection systems first make safe, and finally accumulators and residual heat removal system make core cooldown completely.

III. Accident analysis

1. Calculational model

The SYSRAN code can treat 2-loop PWR power plant and contain the routine which simulates core, steam generater, pressurizer, coolant-loop and secondary loop. Fig. 2 shows the nodalized reactor coolant system. Each part is nod alized by a few control volumes which have the same flow areas and the fluid flow and heat transfer is calculated by the mass and the energy balance equation. The heat generation is calculated by the conventional point kinetics equation and the heat transfer to the reactor coolant is calculated using the lumped-paramer model.[2]

A) Reactor plenum mixing model

If the coolant flow conditions are different from each other loop, cross flow is permitted in downcomer for the flow into core to equal. The mixing rate in lower plenum and upper penum is controlled by the mixing factor. Mixing factor of 1 means perfect mixing, and of 0 means no mixing.

B) Reactor core

Reactor core is nodalized into 4 nodes per each loop and axial power shape is assumed cosine. To consider the volume which does not contribute to remove the heat generation in core, core bypass volum is also taken. Axialy averaged water temperature, density and boron concentration are assumed to proportional to

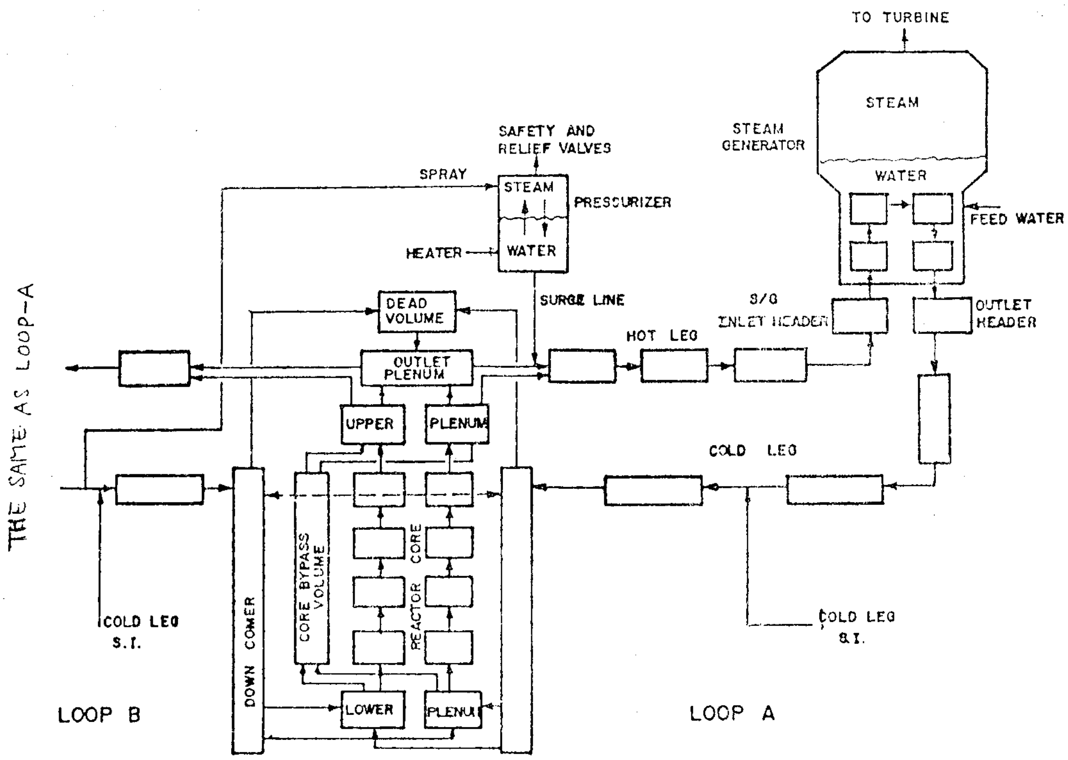

Fig. 2. Reactor Coolant System Flow Model.

cosine square by the perturbation theory.[6] The cross flow from each loop is neglected.

C) Fuel thermal kinetics equation

Fuel is discretized into three regions, two fuel regions and one cladding region. Heat generasion in fuel is assumed to be equal in each fuel region. Calculation of heat transfer to coolant uses Dittus-Boelter correlation[7] for subcooled convection and uses Jens-lottes' correlation[8] for nucleate boiling.

D) Reactor coolant system model

Heat transfer and fluid flow in reactor coolant system are calculated by the energy balance equation and the continuity equation. The pressure change due to the area change from a control volume to next control volume is neglected, and the pressure in reactor coolant system is assumed to be uniform through all the system. The heat transfer to the secondary system through the steam generator is calculated using the logarithmic mean temperature difference.[2] Reactor coolant system pressure is calculated by assuming that pressurizer fluid is saturated and that the specific volume change of pressurizer steam due to the reactor coolant system mass change is to follow the isentropic process.

E) Secondary system model

Secondary side of the steam generator contains fluid part and steam part. Water level is calculated by the bubble-rise model[2] to consider the void generated by the heat from the primary side. In case the water level is lower than U-tube height, the heat transfer from primary to secondary side is assumed to be proportional to the water level under the assumption that the vapor region cannot transfer the heat. Secondary side fluid temperature is calculated from the mass, volume and energy balance equations under the saturated condition. Feedwater enthalpy, flow rate and steam flow must be supplied by the input.

2. **Initial conditions and assumptions**

A) Hot zero power;

It has low mixture level (33% of narrow span), but maximum fluid mass.[9]

B) End of cycle;

The negative moderator coefficient corresponding to the end of life, rodded core with the most reactive rod in the fully withdrawn position gives the largest positive feedback effect causing the severe result. This coefficient is derived from the rodded, 0 ppm curve in Fig. 4.3-30 of FSAR. Doppler coefficient gets the absolutely minimum value of -1.4pcm/°F from Fig. 4.3-26 of FSAR. This condition also increases the positive feedback effect. Boron concentration is assumed to have the smallest value of -10pcm/ppm from Fig. 4.3-31 of FSAR.

C) Double ended break;

1.4ft^2 steam line break occurs between flow restrictor and main steam isolation value.Break flow takes Moody critical flow model.[10]

D) Steam generator discharges saturated steam with quality of 1.0 which causes the maximum heat subtraction.

E) Safety injection system is shown in Fig. 3 and safety injection curve used is shown in Fig. 4 adopted from FSAR. Boron tank contains boric acid of 267ft^3 with concentration of 20,000ppm. After boron tank is exhausted, 2,000ppm boric acid in RWST is injected to the cold leg.

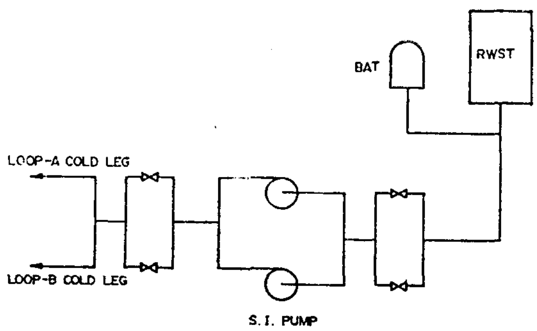

Fig. 3. Safety Injection System.

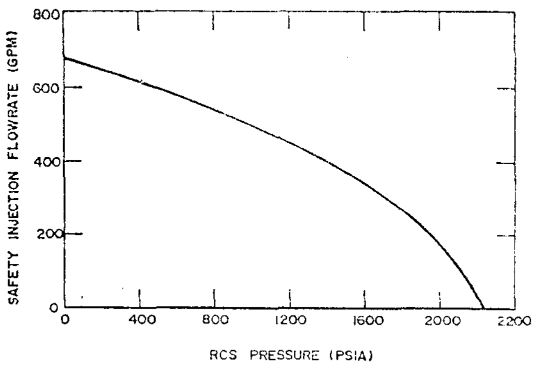

Fig.4. Safety Injection Flowrate vs. Rcs Pressure.

F) Offsite power is available during transient. This gives the larger positive feedback effect than offsite power unavailable.

G) Reactor trip signal occurs at the same time of steam line break accident. The reason why we disregard the delay time is that we focus on the rate of positive feedback effect after insertion of $-1.8\%\ \delta k$, safety shutdown margin at EOL.

H) Safety injection signal starts from steam line low pressure (500psia) with delay time of 10 seconds, and 12 seconds are more needed to arrive at pump full speed. Main steam isolaton and main feedwater isolation value get delay time of 10 seconds. 105% of normal feedwater is supplied to steam generator till main feedwater control valves close and then 5 % of auxiliary feedwater is delivered to steam generator.

I) The condition of reactor core is assumed the same as the broken loop A resulting in a largest positive feedback effect.

J) The mixing condition of upper and lower plenum is also considered, that is, 75% of coolant for one loop remains in the same loop through both plenums.

K) By-pass volume is 30% of reactor core full volume and by-pass flow is 5% of core flow.

L) Pressurizer is located in the intact loop not to mitigate the severences.

IV. Results and Discussion

The analysis of 1.4ft^2 double ended steam line break for Kori Unit 1 has been performed under the previous assumptions. The results are compared with FSAR data in Fig. 5 through Fig. 8.

In Fig. 5, the steam flow rate in loop A is in good correspondance with the value of FSAR with Moody fL/D=0 model.[11] As shown in Fig. 5, the large steam release due to the steam line break results in the excessive heat removal from the reactor coolant system, which causes the abrupt reduction of average coolant temperature~~. At~~ 10 second, however, when the main steam line isolation valves are closed, the decrease in coolant temperature becomes slower.

In Fig.6, the thermal contraction of coolant arising from the reduction of coolant temperature brings down the water level of the pressurizer, and then the pressure of reactor coolant system continues to decrease until the pressurizer becomes empty at 20 secod. At this stage, the pressure of reactor coolant system is maintained by the boiling in the dead volume at the top of the vessel. (In FSAR, the emptification of pressurizer occurs at 14 second. This time lead seems to be caused by the difference between the initial water levels of pressurizer.)

In Fig. 7, the core attains the reactivity of −1.8% at 2 second with all the rod cluster

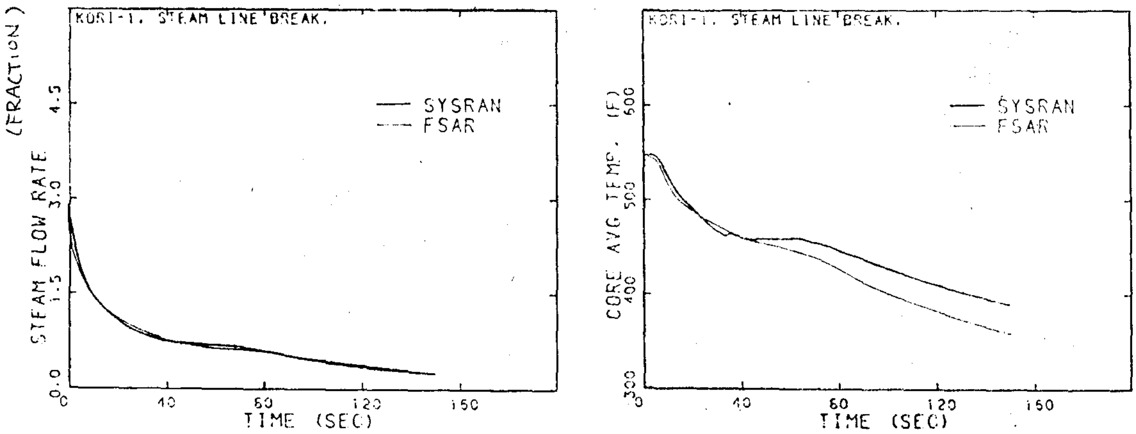

Fig. 5. Steam Flow Rate and Core Avg. Temp. for 1.4ft^2 Break of KORI-1 Steam Line.

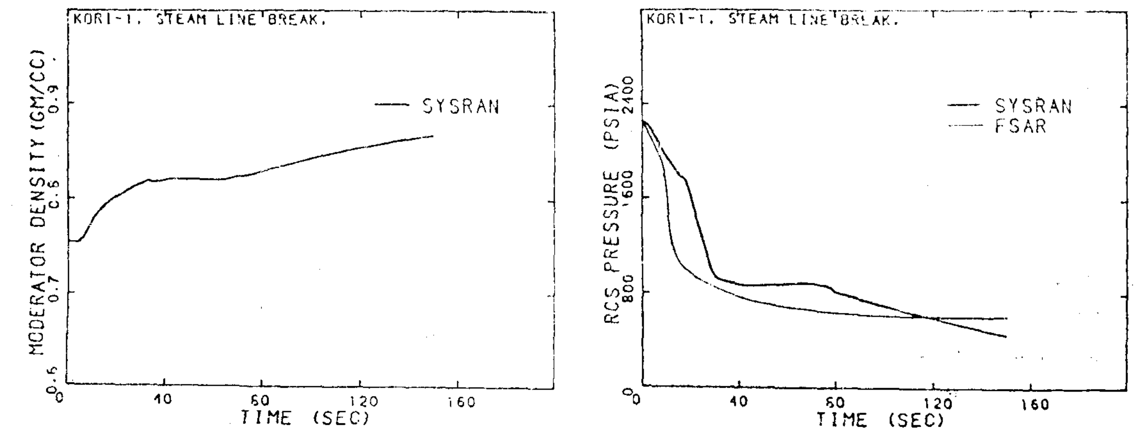

Fig. 6. Moderator Density and RCS Pressure for 1.4ft^2 Break of KORI-1 Steam Line.

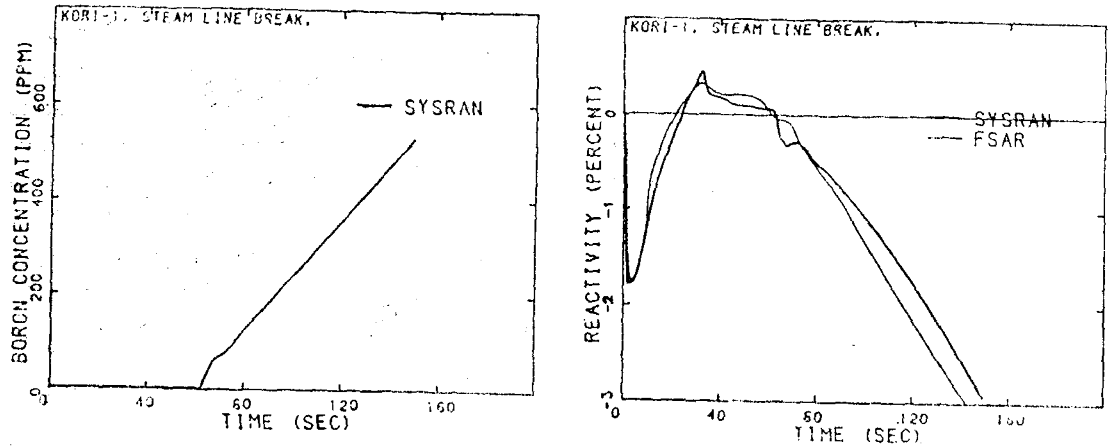

Fig. 7. Boron Concentration and Reactivity for 1.4ft² Break of KORI-1 Steam Line.

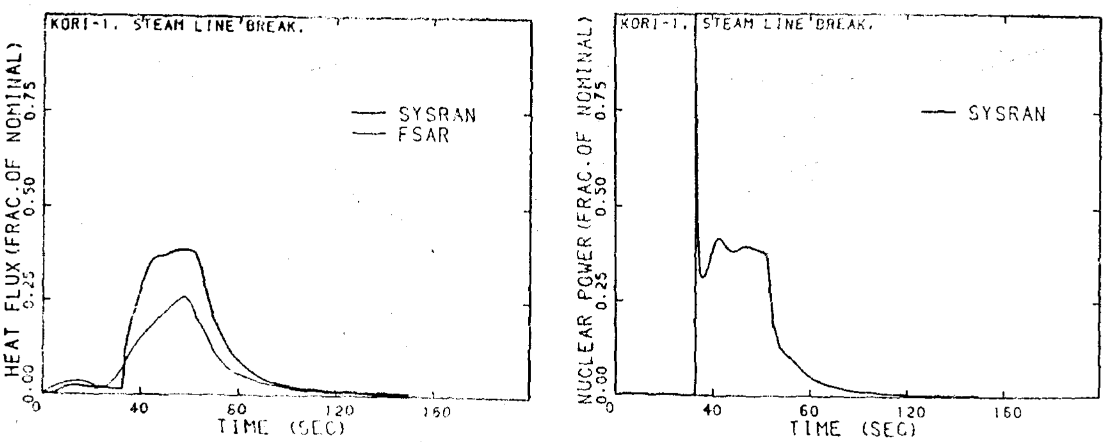

Fig. 8. Heat Flux and Nuclear Power for 1.4ft² Break of KORI-1 Steam Line.

control assemblies inserted except I-7 RCCA(B bank), but at 25 second it returns to the criticality by positive reactivity insertion due to coolant temperature drop, and then at 33 second it goes over the prompt critical of 1.0 \$ (0.0044 $\Delta k/k$).

As shown in Fig. 8, at the superprompt critical state, the neutron flux continues to increase rapidly until the negative reactivity due to Doppler effect of fuel is inserted. And then nuclear power reaches equilibrium at 38% of nominal value. Heat flux also reaches equilibrium with time delay. The higher heat flux makes the coolant temperature higher than the FSAR result as shown in Fig. 5.

At 8 second, the steam pressure in loop A drops below 500 psia, at which the safety injection pumps are initiated. At 18 second, the safety injection pumps start to operate and at 30 second, their speed reaches the maximum value.

As shown in Fig. 7, at 63 second, 20,000ppm boron reaches core and it makes the core subcritical. And then nuclear power and heat flux drop sharply to zero power shutdown. It is shown in Fig. 8.

All the results of this analysis are in good agreement with those of FSAR, expect the occurence of superprompt critical[2] state which seems to be caused by selecting more conservative value of moderator coefficient in this analysis than in FSAR.

The DNBR calculation has been performed using SCAN, a single channel thermal hydraulic analysis code under the assumption of $F_{\Delta H} = 3.66$ which is used in FSAR with all the rod cluster control assemblies inserted except I-7 RCCA (B bank) and of $F_Z = 1.55$. Fig. 9 shows that the minimum DNBR reaches 1.62 at 60 second, indicating that the fuel failure could not be anticipated to occur.

Sensitivity study is also performed for some parameters which are expected to give comparatively large effects to the transients. The results are shown in Fig. 10 through Fig. 13. Sensitivity study on moderator density coefficients is represented in Fig. 10. Maximum heat flux is 43% of nominal value in case moderator coefficient is higher than reference case by 10% and it is 32% in case moderator coefficient is lower by 10%. In both cases the heat -flux is higher than the FSAR result.

Sensitivity study on Doppler temperatur coefficient is represented in Fig. 11. Maximum heat flux, with Doppler temperature coefficient of -2.6pcm/°F which is negatively largest value at EOL, is lower by 6% than the reference case ($\alpha_F = -1.4$pcm/°F). Negatively smallest Doppler

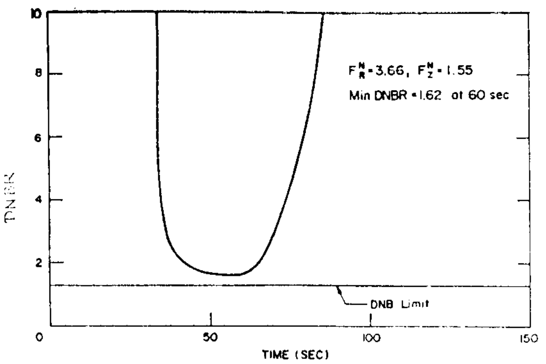

Fig. 9. Min DNBR for 1.4ft² Break of KORI-1 Steam Line.

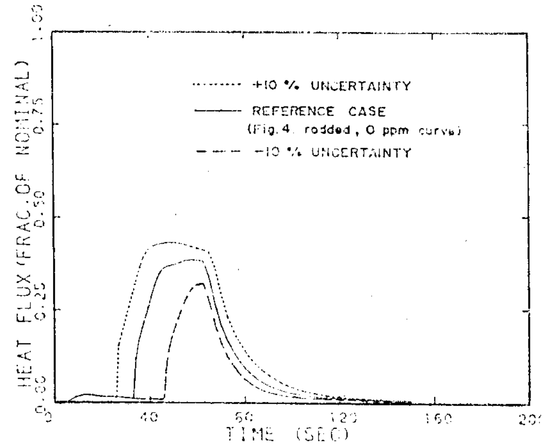

Fig. 10. Sensitivity to Moderator Density Coefficients.

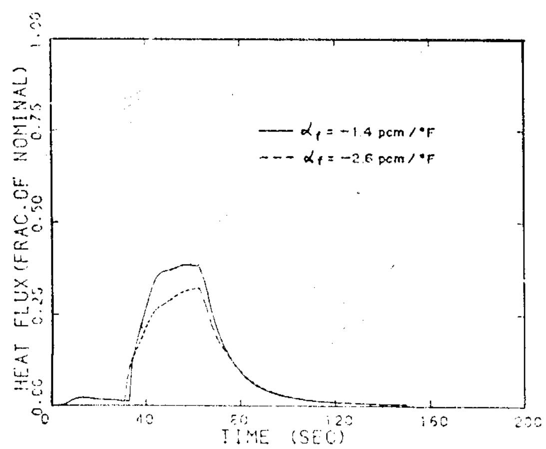

Fig. 11. Sensitivity to Doppler Temperature Coefficient.

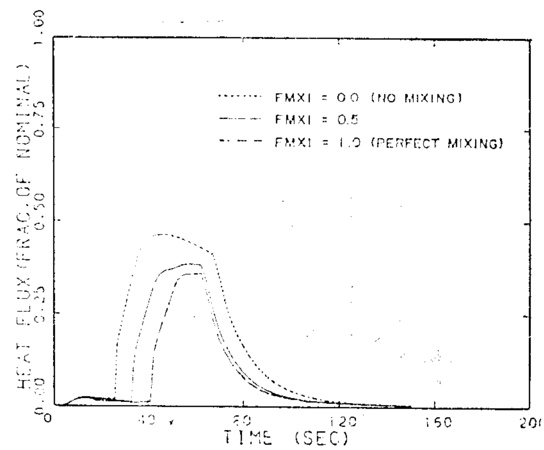

Fig. 12. Sensitivity to Lower Plenum Mixing Factor.

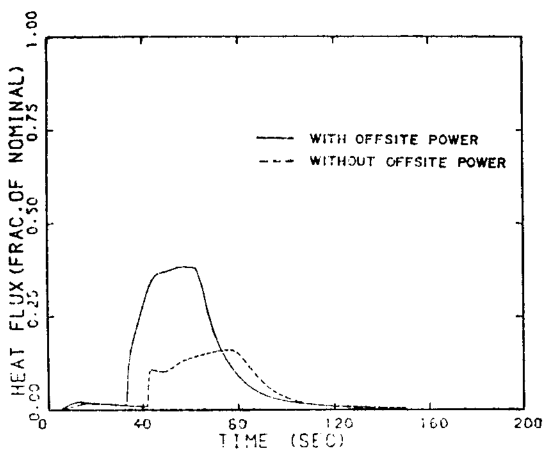

Fig. 13. Sensitivity to Offsite Power Availability.

coefficient brings the smallest feedback effect after the core becomes critical.

Sensitivity study on lower plenum mixing factors is represented in Fig. 12. Maximum heat flux with assumption of no mixing is higher than the reference case by 8% and it is lower by 2% in case of perfect mixing.

In Fig. 12 heat flux is represented in case offsite power is unavailable at the same time the accident occurs. Maximum heat flux is much lower than the reference case (offsite power is available). If offsite power is available, the reactor coolant flow rate will be maintained and the cooldown rate will be faster than offsite power unavailable.

The other parameters does not so large effects to the transients as those parameters mentioned before.

280 cpu seconds is elapsed with computer system CDC-6400 with time steps 0.1 seconds to accident time of 5 second, 0.2 to 50, 0.5 to 100 and 1.0 to 150 second. The transient, with time steps as 4 times detailed as the reference case, does not much change.

V. Conclusion

The analysis of the 1.4ft^2 steam line break for Kori Unit 1 using the point kinetics equation and the system transient model reveals the maximum heat flux of 38% of the nominal value at 60 second, minimum DNBR of 1.62. Those results indicate that the reactor can shutdown without damaging fuel integrity if only one safety injection pump works.

All the results are in good agreement with the FSAR excluding the heat flux which is higher in this analysis.

Sensitivity study also indicates that FSAR assumptions are most conservative. Some parameters including the mixing factor need more accurate data aquired by the experiment.

The point kinetics equations, the lumped-parameter model and the system transient model which uses the mass and balance equation are verified to be effective to follow the system transient phenomena of the nuclear power plants.

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