Journal of the Korean Nuclear Society
Volume 26, Number 1, March 1994

Analyses of SGTR Accident With Mihama Unit Experience

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(Received Dctober 4, 1993)

미하마 원전경험에 대한 SGTR 사고해석

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(1993. 10. 4 접수)

Abstract

A SGTR accident postulated at Kori unit 1 is simulated with Mihama unit experience, which occurred on February 1991, to evaluate the capability of plant to cope with the transient. The system design and plant conditions of Kori Unit 1 are much similar with those of Mihama Unit 2. Therefore, special concern has been given to evaluate the sequence and the resulting consequence of the postulated SGTR accident at the Kori unit 1. An analysis is performed as realistically as possible, with following the EOP of Kori unit 1. The result indicates that the leak through tube break terminates within about forty minutes, and the Kori unit 1 may be sufficient to cope with SGTR accident with same type of sequence. However, the reconsideration may be required for the design of Kori unit 1 which disconnects non-safety AC power from off-site power on SI signal generation. It may be pointed out that the content of EOP for SGTR accident is not enough to require operator's proper judgements. An analysis of SGTR accident tested in the LSTF which simulated the SGTR accident at the Mihama Unit 2 is performed using the RELAP5/MOD3. The results indicates that the code yields in general good agreement with the test, except the break flowrate at the early stage of the event.

요 약

1991년 2월 미하마원전에서 발생한 증기발생기 세관 파열사고에 대한 경험을 바탕으로, 본 사고에 대한 고리 1호기의 대처능력을 평가하기 위하여 해석을 수행하였다. 고리 1호기의 계통설계 및 운전조건은 미하마 2호기와 아주 유사하기 때문에 고리 1호기에서 발생한 가상의 증기발생기 세관 파열사고시의 사고경위 및 전개에 대한 평가가 필요하였다. 해석은 고리 1호기 EOP를 근거로 현실적으로 가능하게 수행되었다. 해석결과, 파열된 세관을 통한 누출은 사고후 약 40분 후에 정지되었으며, 고리 1호기는 유사한 증기발생기 세관 파열사고의 경우 충분한 대처능력이 있음을 보였다. 그러나, SI 신호작동후 소외전원으로 부터의 비안전등급 AC 전원으로 단절되는 설계에 대한 재고가 필요하며, EOP의 운전절차가 운전원의 적절한 판단을 요구하기에는 다소 충분치 못함을

보였다.

또한 미하마 원전의 사고를 실험적으로 모사한 LSTF의 실험결과를 이용 해석코드인 RELAP5/MOD3의 평가능력에 대하여 해석을 수행하였다. 해석결과 코드는 사고 초기의 누설량 예측을 제외하고는 일반적으로 실험결과와 잘 일치하고 있음을 보였다.

1. Introduction

Severe plant transients following a Steam Generator Tube Rupture (SGTR) have a relative high probability of occurrence. With the degradation of the steam generator U-tube integrity due to vibration, corrosion and crack during long operation, the probability of rupture is relatively higher than the reactor coolant system piping. Thus the degradation of steam generator U-tube integrity has been studied as one of the Unresolved Safety Issues (USIs) for a time.

On February 9, 1991, a single tube in steam generator was ruptured in Mihama unit 2 in Japan. The cause of the tube rupture was reported as the incorrect insertion of anti-vibration bars(AVBs), which could not protect the tube of X45-Y14 from the fatigue by fluid elastic vibration. [1] This accident was ranked first in Japan as the event with ECC injection. In consequence of this accident, the amount of the leakage from the primary into the secondary system was estimated as approximately 55 tons, and the amount of the released steam from the main steam relief valve of damaged steam generator was estimated as approximately 1.3 tons. Mihama unit 2 has been operated commercially since 1972, and its nuclear steam supply system(NSSS) design was supplied by Westinghouse with the Model 44 steam generator. The NSSS design of Mihama is much similar with the Kori unit 1 [2] which was started commercial operation at 1978. The major design and operational data are compared between Kori unit 1 and Mihama in table 1. The steam generator of Kori unit 1 is the Model 51 designed by Westinghouse, which is somewhat different from that of Mihama. Even some differences in design from Mihama, it could not be excluded that Kori unit 1 has potential possibility of steam generator tube rupture(SGTR) accident of that sequence. Kori unit 1 has experienced the steam generator tube leak four times, even not challenging plant protection system. In addition, Kori unit 1 has been recently suffering from frequent troubles with causing unexpected reactor trip. On this regard, special concern has been given to evaluate the possibility of Mihama sequence at Kori unit 1 and then the resulting consequences. Thus, the need of reevaluation for Kori unit 1 comes to the fore and the reanalysis of SGTR accident for Kori unit 1 is carried out. As the similarity of NSSS design between Kori unit 1 and Mihama, this analysis is following the emergency operating procedure(EOP) of Kori unit 1 [3] based on the sequence of event of Mihama SGTR accident including component failure and operator's action. Through this analysis, it may be expected that the safety of Kori unit 1 can be re-evaluated under this type of accident condition and the deficiency of the EOP of SGTR for Kori unit 1 can be identified.

The Japan Atomic Energy Research Institute (JAERI) conducted integral simulation experiments on the SGTR incident that occurred at the Mihama Unit 2 power station. The experiment was performed using the Large Scale Test Facility (LSTF) of the ROSA-IV Program. The objective of the experiment was to provide detailed thermal-hydraulic experimental data, that suppliment the plant record, to be used for in-depth evaluation of the incident and for validation of computer-code analyses of the incident. The experimental results

Table 1. Comparison of Major Design Parameters Between Kori Unit 1 & Mihama Unit 2

	Kori unit 1	Mihama Unit 2
Commercial Operation	1978	1972
Reactor type	PWR	PWR
Core Thermal Power	1723.5 MWt	1456 MWt
Generator Power	560 MWe	500 MWe
Number of RCS Loop	2	2
Vendor	Westinghouse	Westinghouse
RCS Operating Pressure	155 bar	155 bar
RCS Loop Flow	4234 kg/s	4028 kg/s
Number of S/G	2	2
S/G Model	Model 51	Model 44
Number of S/G Tubes	3380	3260
Height of S/G	20.6 m	19.3 m
Diameter of S/G tube	22.2 mm	22.2 mm
Number of Tube Support	7	6
Number of HPSI Pumps	2	2
Number of RHR Pumps	2	2
Number of Accumuiator Tanks	2	2
Number of Cont. Spray Pumps	2	2
Number of Emergency D/Gs	2	2
Number of Aux. Feed Pumps	2(motor)/1(turbine)	2(motor)/2(turbine)

show that the sequence of events and the transient changes in system parameters agree well with the Mihama Unit 2 data, and confirm that there is a large margin in the core cooling capability during the incident.

In the present study an analysis of the SGTR accident performed in the LSTF is examined using the RELAP5/MOD3 to improve common understanding of Pressurized Water Reactor (PWR) thermal-hydraulic response during such a transient and to identify areas for desirable model improvements based on the comparision between data and predictions.

2. Analysis of SGTR Accident for Kori Unit 1

2.1. Kori Unit 1 Modelling

Kori unit 1 is a nuclear power plant producing 560 MW electrical power with the core thermal power of 1723.5 MWt. The NSSS of Kori unit 1 consists of one reactor, two steam generators and one pressurizer. The reactor coolant system(RCS) consists of two loops with the two reactor coolant pumps(RCPs). The RCS inventory is maintained constantly by chemical and volume control system(CVCS) through charging and letdown. Power operated relief valve(PORV) and safety relief valve(SRV) are installed in the pressurizer in order to prevent the RCS from overpressurization. Also, pressurizer spray system provides RCS pressure control, which consists of normal and auxiliary spray system. As the purpose of RCS pressure control, the heaters of 1 MW are installed in a pressurizer. The steam from two steam generator is gathered in a steam header and is supplied to turbine. A main steam isolation valve(MSIV) is installed in each steam line, actuated manually by an operator or automatically by steam line low pressure signal. Each steam line has a PORV and five SRVs to prevent the steam generator from overpressurization. The PORV has additional function providing the heat sink for RCS through steam dump. Another component to dump steam for RCS cooling is prepared in steam header. The feedwater supplied to a steam generator is provided from main and auxiliary feedwater system. Main feedwater system is controlled by steam generator level, steam flow and RCS temperatures. Auxiliary feedwater is supplied by turbine driven auxiliary feedwater pump and motor driven auxiliary feedwater pumps as soon as auxiliary feedwater supply signal actuated with delay time. To cope with the loss of RCS inventory events such as loss of coolant accident and SGTR accident, the emergency core cooling system(ECCS) is prepared, which is consisted of high pressure injection pumps, low pressure injection pumps and accumulator tanks. The injection positions of ECCS are both cold leg side and upper plenum of reactor vessel. To simulate SGTR accident for Kori unit 1, RELAP5/MOD3 code [4] is used, which is known as best-estimated code. For the approach to reality, the control logics and controlled components are modelled based on Kori unit 1 Precaution, Limitation and Setpoints(PL&S) [5] and emergency operating procedure for SGTR accident. Major control logics are RCS pressure control, RCS inventory control, feedwater control and steam dump control. RCS pressure is controlled automatically by pressurizer normal spray system

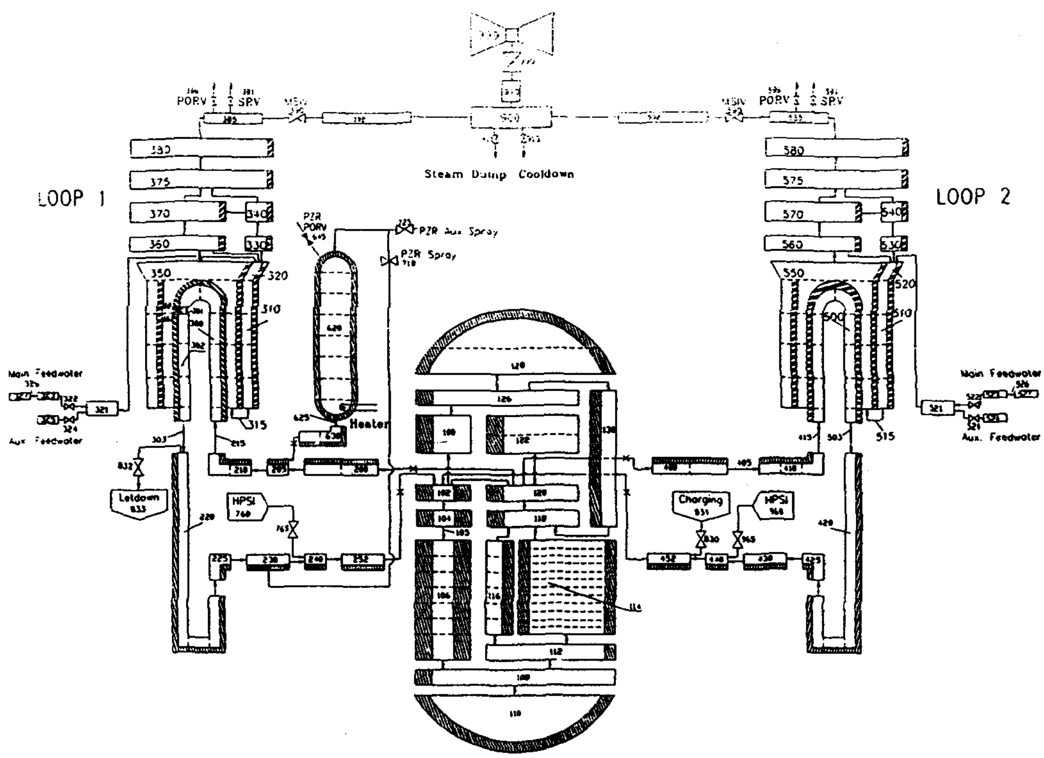

Fig. 1. RELAP5 Nodalization For Kori Unit 1

which connects from cold leg to top of pressurizer and pressurizer heaters installed in the pressurizer bottom. If the pressurizer level drops below 18%, the pressurizer heaters and normal spray are turned off automatically to prevent the overheating of heaters. RCS inventory is maintained by charging and letdown through programmed pressurizer level; Letdown flow is set as letdown orifice flow, otherwise charging flow varies as programmed level. Main feedwater flow is controlled to match steam flow from steam generator and to constantly maintain steam generator level. If turbine trip occurs or turbine load is reduced, the steam dump to condenser makes RCS subcooled margin to be maintained as a certain value. This system is controlled by the relation of RCS reference temperature and auctioneered temperature. The nodalization of Kori unit 1 modelled for SGTR accident simulation is shown as figure 1.

2.2. Evaluation of SGTR Sequence

The Kori unit 1 EOP for SGTR accident is described successively in order to decide an operator actions for mitigating the accident, as briefly shown in figure 2. The analysis of SGTR accident is carried out with the sequence of EOP. However, the failure occurred during Mihama SGTR accident are applied to this analysis, because of the similarity between Kori unit 1 and Mihama unit. The failures in Mihama event was MSIV delayed closure and pressurizer PORV opening failure. MSIV may be closed by two ways; one is fast closing by pneumatic and the other is slow closing by solenoid current. If fast closing is failed, slow closing is actuated by an operator which takes about 5 minutes. In the case of Mihama event, it was expected that slow closing of MSIV was performed after the failure of fast closing. Pressurizer PORV could be used for RCS depressurization. Rapid depressurization of RCS may stop the leakage from primary to secondary, then the accident can be terminated. In spite of the attempt to open the pressurizer PORV by an operator, pressurizer PORV failed to open eventually in Mihama. With above mentioned two exceptions, the sequence of events assumed in this analysis is based on Kori unit 1 EOP as described below. After the reactor trip and safety injection(SI) start, an operator checks proper actuation of various safety functions, and then RCP trip criteria is checked which is RCS subcooled margin of above 10 ℃ and pressurizer level of above 18%. Next, the identification of a faulted steam generator is performed with high radioactive indication in the faulted steam generator blowdown line and/or the behavior difference between both steam generator levels and pressures. An operator closes the MSIV of the faulted steam generator side after the confirmation of faulted steam generator. Through controlling auxiliary feedwater flow, an operator regulates the levels of both steam generators between 8% and 50%. After the reset of SI signal and recovery of the all AC power bus, RCS cooldown is started with maximum rate using steam dump through steam dump valve to condenser or using intact steam generator PORV. RCS cooldown may cease when the core temperature drops to the certain value corresponding to the faulted steam generator pressure. Then an operator checks the availability of pressurizer normal spray to depressurize RCS. If pressurizer normal spray is not available, pressurizer auxiliary spray supplied from CVCS may be used. Until RCS pressure decreases below the faulted steam generator pressure and the pressurizer level recovers above 8%, pressurizer spray may continue with maximum rate. Another criteria for stopping pressurizer spray depends on whether enough RCS subcooled margin and pressurizer level cap be maintained. If an operator judges RCS pressure to remain stable, SI may be stopped. In the present study, the analysis is carried out up to this stage,

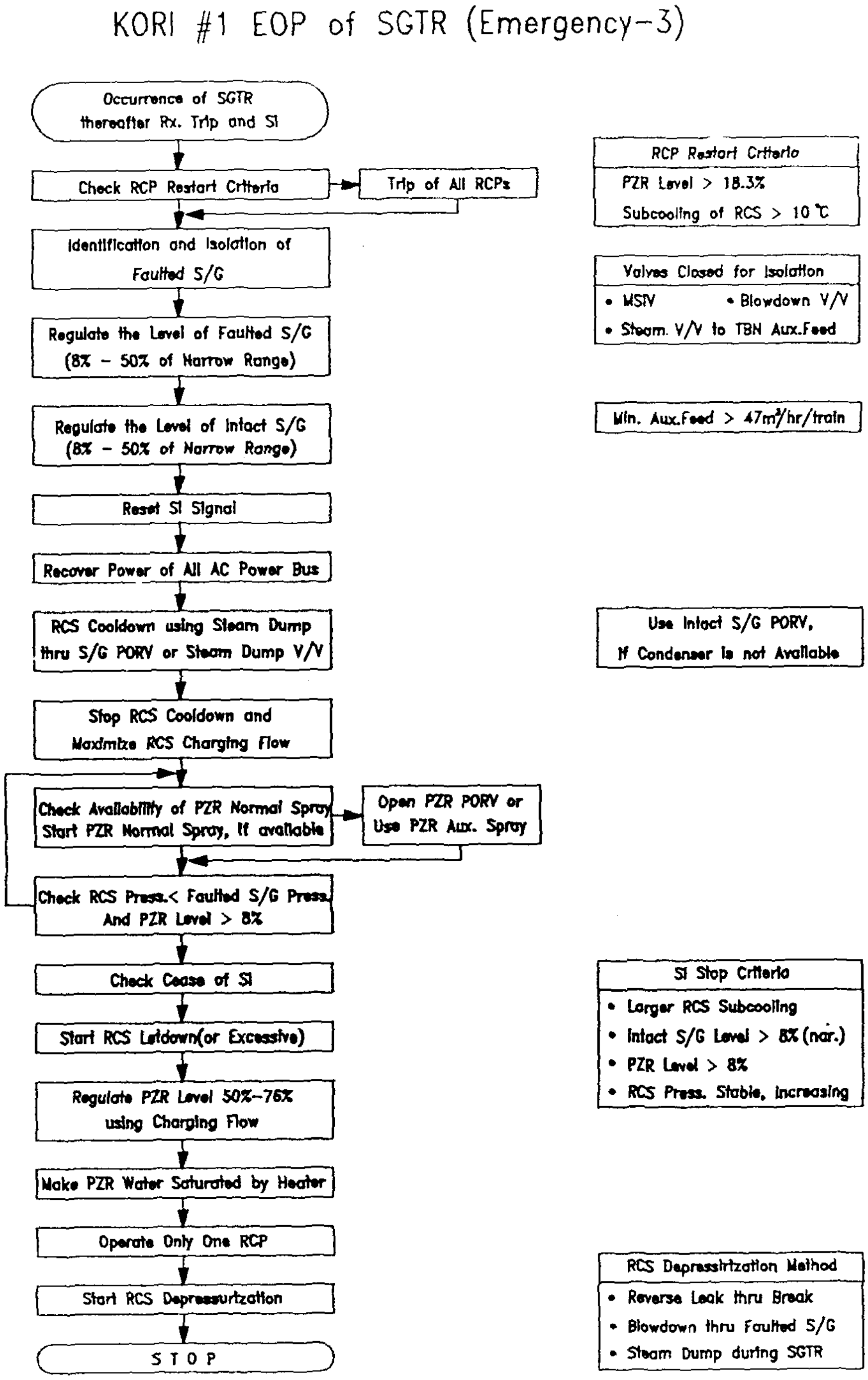

Fig. 2. Kori Unit 1 EOP of SGTR (Emergency-3)

because the stages remaining afterward are only to adjust or control for the termination of accident. After SI ceases, charging and letdown are controlled to maintain RCS inventory. Also, one RCP is restarted or operated to cool the core and to make pressurizer normal spray available. Finally an operator starts RCS depressurization in any way chosen from optional three ways; reverse leak through break, blowdown through faulted steam generator and steam dump.

2.3. Analysis Results

After the break, the RCS pressure decreases slowly in accordance with RCS mass balance, that is, the mass addition from CVCS and the mass reduction by leakage. An operator starts to reduce the reactor power at the rate of 7% per minute when any transient is identified. Corresponding to power reduction, main feedwater flow to a steam generator decreases. For pressurizer pressure control, pressurizer heaters turn on till pressurizer level reaches above 18%. To replenish RCS with coolant, charging flow increases and letdown valves are closed. At 360 seconds after tube break, the pressurizer pressure reaches at the reactor trip setpoint of 122.7 bar, which is the similar reactor trip time in the sequence of Mihama. At this time, the core power is 78% of full power expected as the same power ratio as Mihama. Turbine trip occurs immediately after reactor trip, which causes the increase in steam generator pressures and the decrease in steam generator level due to the level collapsing by the pressure increase. The decrease in steam generator level generate the auxiliary feedwater actuation signal. Turbine driven auxiliary feedwater is supplied immediately, but motor driven auxiliary feedwater is supplied with some time delay for valve arrangement and pump loading. Continuous tube leak reduces pressurizer pressure, and then SI signal is generated by low pressurizer pressure at 8 seconds after reactor trip. The low pressurizer level in SI signal generation logic in Mihama unit is not included in Kori unit 1. SI signal interrupts all non-safety AC electric power bus connection from off-site power, which is a specific design of Kori unit 1. Thus, all RCP starts coastdown and charging flow ceases. With time delay, high pressure core cooling water is injected into both cold legs. Although an operator checks RCP trip criteria and determines whether RCPs restart or not, RCP trip criteria is not satisfied and RCPs remains stopping in this analysis. Auxiliary feedwater fills up both steam generators until the narrow range of steam generator level reaches at 8%. At 630 seconds, the auxiliary feedwater flow to faulted steam generator(S/G-A) stops, which is later than 480 seconds of Mihama sequence. This is expected because the steam generator level of Kori unit 1 is higher than that of Mihama. After the termination of auxiliary feedwater flow to S/G-A, the auxiliary feedwater flow to intact steam generator(S/G-B) increases about 50%. The MSIV of S/G-A is closed at 5 minutes after reactor trip. From this time the pressure behaviors of two steam generators experiences differently. Following Kori unit 1 EOP, operator starts RCS cooldown by steam dump from S/G-B after the narrow range of S/G-B level recovers to 8%. The steam dump for RCS cooldown continues with maximun rate during 445 seconds to reduce the RCS temperature to the value corresponding to S/G-A pressure with sufficient RCS subcooled margin. To refill pressurizer and depressurize RCS, charging pumps are restarted with maximum flowrate at 1890 seconds. An operator judges whether S/G-A pressure is stable or increases and RCS subcooled margin retains above 12 ℃, then pressurizer auxiliary spray is actuated, if normal spray is not available, in order to depressurize RCS and refill pressurizer. At 3350 seconds after tube break, RCS pressure approaches nearly to S/G-A pressure, and the leakage from primary to secondary becomes negligible. Hence, SI is not necessary any more at 3365 seconds. On the whole, the plant behaviors of Kori unit 1 are similar as those of Mihama except steam generator wide range level. RCS pressure shows similar trend as that of Mihama before RCS cooldown using steam dump as shown in figure 3. The cooldown rate for Kori unit 1 is much larger than that of Mihama, and lower RCS pressure of Kori unit 1 is maintained up to SI termination. The steam generator wide range levels and the pressurizer level of both plants are compared as shown in figure 4. For Kori unit 1 early depressurization of RCS causes the refill of pressurizer, however, the initial values of steam generator level of both plants are different significantly. RCS hot leg temperatures of both loops are nearly same in Kori unit 1, whereas for Mihama a little difference between both hot leg temperatures is shown in figure 5. This is expected because the measuring positions of temperatures are different. For Kori unit 1, the cold leg temperature of loop–1 decreases greatly after 2000 seconds, because the loop–1 flow is nearly zero and cold SI coolant is mixed with the coolant in the cold leg.

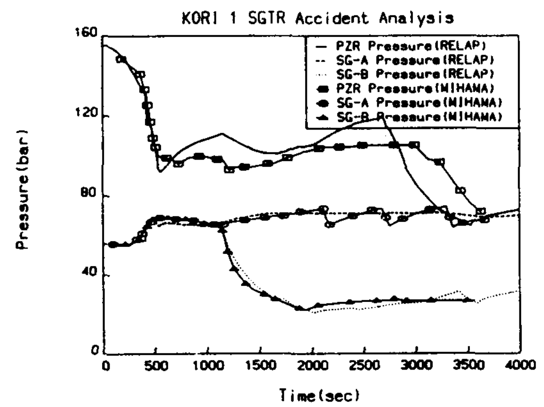

Fig. 3. RCS and SG Pressures

3. LSTF Experimental Facility, Conditions and Procedure

The LSTF is a 1/48 volumetrically scaled model of a Westinghouse type 3423 MWt four loop PWR. It has same major component elevations as the reference PWR to simulate the natural circulation phenomena and large loop pipes to simulate the two-phase regimes and significant phenomena in an actual plant. Figure 6 shows the structure of the LSTF. The facility is designed to be operated at the same high pressures and temperatures as the reference PWR.

The experiment was initiated by opening a

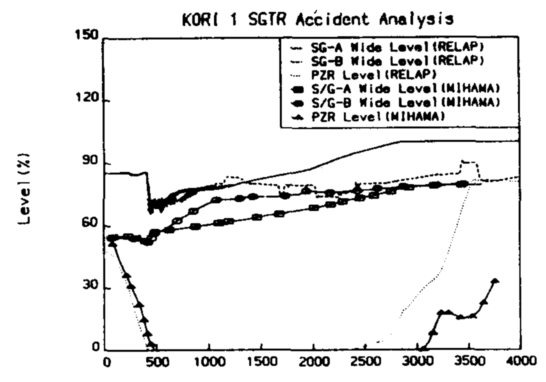

Fig. 4. Pressurizer and SG Water Levels

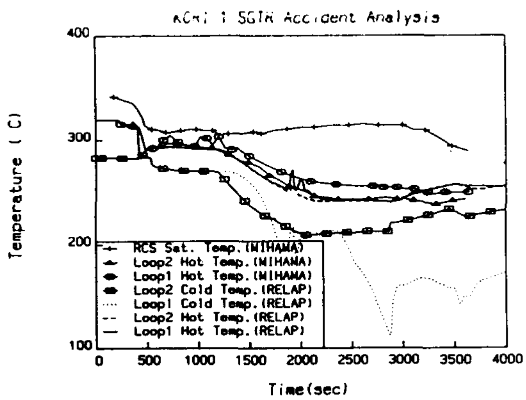

Fig. 5. RCS Temperatures

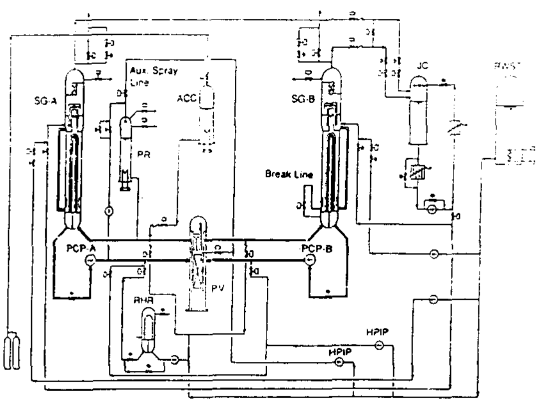

Fig. 6. Structure of ROSA-IV LSTF

break valve nearly at the same RCS pressure and temperatures as at the Mihama Unit 2. The reactor trip signal and safety injection signal were sent automatically at the same RCS setpoint pressures as at the Mihama Unit 2. The damaged steam generator was isolated for 12 min after reactor trip. At the same time, the depressurization of the intact steam generator secondary side was initiated. The depressurization was terminated according to the Mihama Unit 2 Emergency Operating Procedure (EOP). Subsequently, the atmospheric relief valve (ARV) on the damaged steam generator is open and close automatically. The pressurizer auxiliary spray was actuated 44 min after reactor trip to depressurize the RCS. The high pressure injection (HPI) pumps were turned off after the pressurizer water level was recovered. The pressurizer auxiliary spray was turned off after the RCS pressure equilibrated with the damaged steam generator secondary side pressure. Finally, the reactor coolant pump in the intact loop was restarted 65 min after reactor trip. The experiment was ended when the RCS conditions were stabilized.

4. Analysis of LSTF SGTR Experiment

The nodalization used to simulate the LSTF facility of the ROSA-IV program with the

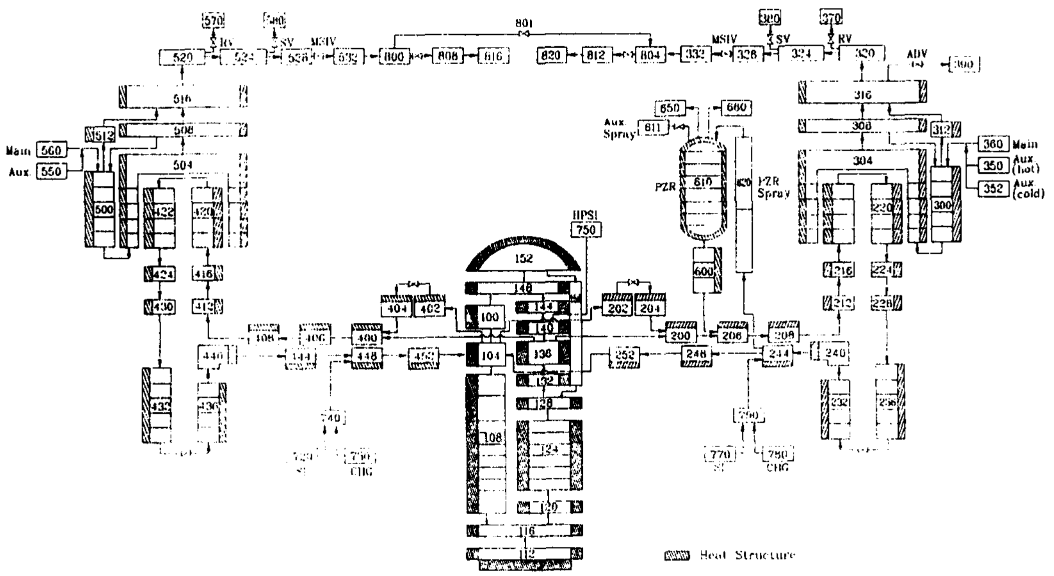

Fig. 7. RELAP5 Nodalization For ROSA IV LSTF

RELAP5/MOD3 code is shown in figure 7. The model is based on 162 volumes connected by 169 junctions and 166 heat structures. The initial conditions, operational set points and conditions are set to be ths same as the experiment. Calculation is terminated by the user when the operator's recovery procedure starts, which is at about 4000 sec after the break initiation.

4.1. Analysis Results

After the initiation of S/G U-tube rupture, the pressurizer pressure and the liquid level decrease monotonically because the amount of break flow to the S/G secondary side is larger than water supplied from the charging pumps. Pressurizer backup heaters are turned on when the pressure reaches 15.4 MPa and off when the liquid level drops to 1 m. When the RCS pressure reaches 13.42 MPa, reactor scrams and the pressure and temperature of RCS and the pressurizer liquid level decrease rapidly. Following the continuous depressurization of the RCS, the RCS pressure reaches the setpoint of the safety injection signal and water is injected to the cold legs and upper plenum of the reactor vessel at 10 sec and 300 sec after the safety injection signal, respectively. Reactor coolant pumps coastdown at 80 sec after the reactor scrams. The pressurizer is emptied completely at about 500 sec. Since the flow between the reactor core and the reactor vessel upper head is quite low, the temperature decrease of the upper head is much less than that of the RCS, while the depressurization rate is almost same as that of RCS. Due to void formation and heat transfer from the reactor vessel to RCS, voiding appears at the upper head of the reactor vessel and the upper head starts to control the RCS pressure instead of the pressurizer.

Voiding appears at about 550 sec. However, its fraction is about 0.15 and approaches to zero later on. After voiding appears in the upper head of the reactor vessel, the RCS pressure increases due to safety injection. The RCS temperature decreases rapidly after the reactor trip. However, after the reactor coolant pumps coastdown, the core flow decreases and the S/G pressure increases. Since the increase in S/G feedwater enthalpy causes less heat transfer between the two systems, the RCS temperature increases. That is, since a large amount of heat per unit mass in the core is transferred and heat loss is reduced, the RCS temperature at the reactor vessel outlet becomes higher. The broken S/G is isolated at 720 sec after reactor scram and simultaneously the intact S/G is depressurized by opening the S/G atmospheric dump valves. The RCS pressure which is increased due to safety injection decreases slowly by the steam dump and the feedwater supply with a decrease of the RCS temperature. The amount of heat stored in the RCS is removed after the opening of S/G steam dump and the RCS temperature decreases at 1200 sec and the RCS subcooling margine increases. Also the voids appeared at the upper head of the reactor vessel are removed at 2000 sec with a decrease in the RCS temperature. The pressure of the broken S/G increases due to RCS inflow with high energy and is controlled by opening the pressure relief valve. The peak pressure of the broken S/G is controlled by the setpoint of the pressure relief valve and the integrity of the S/G is maintained sufficiently.

For the faster depressurization the pressurizer auxiliary spray system is actuated at 2600 sec after the reactor scram instead of the pressurizer relief valves, and the pressurizer liquid level starts to recover after 3100 sec. The high pressure safety injection systems are terminated after the pressurizer liquid level is recovered. When the RCS pressure and the broken S/G pressure become identical, the break flow is stopped. The break flow rate from the RCS to the broken S/G is influenced by the pressure difference between the two systems. Initially the pressure difference is very large ,thus the critical flow rate is formed, and after reactor trip the break flow rate decreases due to less pressure difference. After the safety injection the RCS pressure and break flow rate increase. However, the break flow rate becomes constant later on. After the depressurization using the pressurizer auxiliary spray, the pressure difference between the two systems is nearly identical and the break flow is stopped. The total RCS inventory loss by the total break flow is about 2850 kg, which is equivalent to 60 tons of the Mihama Unit 2.

4.2. Comparison of Results

For the initial depressurization period (up to 500 sec), the calculated RCS pressure drop does not agree well with the measured pressure drop as shown in figure 8. Here, "Pri, Sec-I, Sec-B" represent the conditions at the primary side, intact S/G, broken S/G, respectively. This is mainly due to different prediction of the break mass flow rate from the S/G U-tube to the S/G secondary side. In the experiment the break flow rate decreases rapidly up to 500 sec, however, the break flow rate in figure 9 decreases very gradually and decreases rapidly after the reactor trip. It is considered that this difference results from the modelling of a break nozzle and a related piping. A detailed description for break simulation in the experiment is not available at present, the break is modelled as a simple "trip valve." Since the break mass flow clearly follows the pressure evolution during the transient, this underprediction of the break flow rate may cause the underprediction of the RCS depressurization rate, and consequently the time delay of the reactor trip about 100 to 200 sec.

Figure 8 manifests a level discrepancy at the time the intact S/G steam dump valve is actuated to initiate a cooldown. There exists a large uncertainty at the time of the cooldown initiation and the opening rate of the steam dump valve. The instantaneous opening of a valve shows very large level swell which may be clearly deviated from measured data. Even if the precision in the measured level is not known, the calculated trend diverges from the measured trend most probably due to excessive level swell in the riser region combined with liquid fallback in the separator region. Since the water level measurement in a

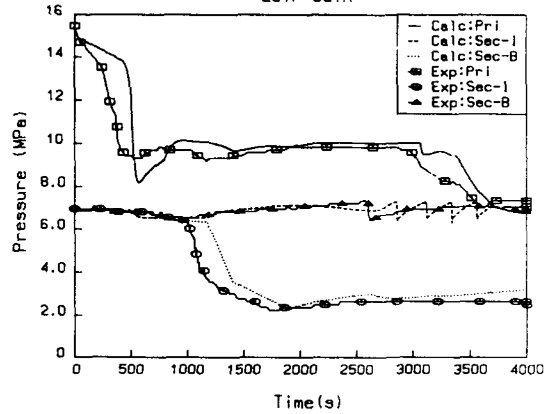

Fig. 8. RCS and SG Pressures

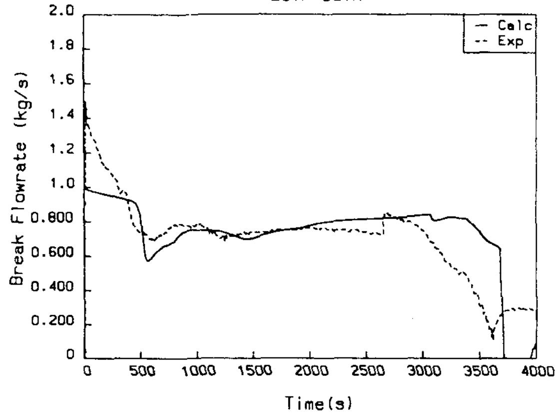

Fig. 9. Break Flow Rate

power plant is used for plant control and also for plant protection, it follows that numerical discrepancies for predicting the S/G water level could have a large impact on the predicted behavior of a power plant, and at the timing of the actuation of emergency features (e.g.auxiliary feedwater).

Figure 8 also shows the behavior of the S/G secondary pressure for both intact and broken sides. The general trend agrees well with the experiment. However, the number of the opening of S/G relief valve is four comparing to one in the experiment. This difference may come from the insufficient modelling of noding, heat capacity, and heat loss in the S/G secondary side.

In the experiment, fluid temperature "bumps" appears in the hot leg to which the pressurizer is connected. This occurs as hot water and steam in the pressurizer penetrated into this hot leg. However, this "bumps" is not observed in the present calculation as shown in figure 10. Here, "HL-I and HL-B" represent the conditions at the intact and broken hot leg, respectively.

Regardless of the differences between the calculation and the experiment described above, other parameters, such as pressurizer level, total break flow rate, total water injection to cold legs

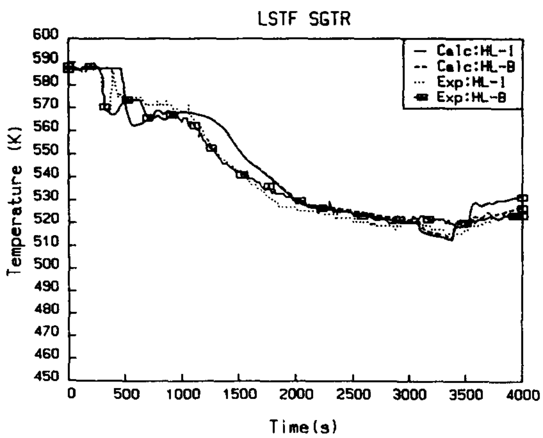

Fig. 10. RCS Hot Leg Temperature

and reactor vessel upper head by HPSI pump, etc. are in good agreement with the experiment.

Finally, the coolant in the core stays as subcooled state throughout the transient. Also the natural circulation flow through the RCS loops is large enough to keep the core coolant subcooled even though the reactor coolant pumps are turned off automatically at 80 sec after the reactor trip.

5. Conclusion

The analysis of SGTR accident for Kori unit 1 is carried out with the EOP of Kori unit 1 based on some sequence of Mihama unit 2. For Kori unit 1, the leak through tube break may be terminated within one hour and no more radioactive materials releases to environment, whereas the eight hour was assumed as the duration of radioactive release in the analysis of Kori unit 1 FSAR[6]. Thus, the accident analysis in the licensing stage is very conservative relatively to the present situation. It may be considered that Kori unit 1 is sufficient to cope with SGTR accident with this type of sequence. However, the Kori unit 1 has some deficiencies in mitigating SGTR accident, that is, the interruption of non-safety AC power bus when SI signal generated. It may need unnecessary operator actions and cause some operator errors. The present EOP of Kori unit 1 is not sufficient to direct an operator to cope with SGTR accident. Insufficient contents of EOP may cause an operator error, because many judgements are needed by an operator. Therefore, it is recommended to revise the current EOP of Kori unit 1 that the safety injection through the upper plenum of reactor vessel by the operator action be described in detail. And a detailed design review on the emergency AC power is necessary to maintain the offsite power supplied to the reactor coolant pump and the condenser after safety injection.

An analysis of SGTR accident is performed using the RELAP5/MOD3 for the code assessment on SGTR accident. The analysis result is compared with the test data of the LSTF which simulated the SGTR occurred at the Mihama Unit 2. The results are in good agreement with the test, except the break flowrate at the early stage of the event. Further analysis with detailed break simulation as in the experiment could be necessary.

References

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