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International Agreement Report

Assessment of RELAP5/MOD3 Version 5m5 Using Inadvertent Safety Injection Incident Data of Kori Unit 3 Plant

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Office of Nuclear Regulatory Research
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NOTICE

Abstract

An inadvertent safety injection incident occurred at Kori Unit 3 in September 6, 1990 was analyzed using the RELAP5/MOD3 code. The event was initiated by a closure of main feedwater control valve of one of three steam generators. High pressure safety injection system was actuated by the low pressure signal of main steam line.

The actual sequence of plant transient with the proper estimations of operator actions was investigated in the present calculation. The asymmetric loop behaviors of the plant was also considered by nodalizing the loops of the plant into three.

The calculational results are compared with the plant transient data. It is shown that the overall plant transient depends strongly on the auxiliary feedwater flowrate controlled by the operator and that the code gives an acceptable prediction of the plant behavior with the proper assumptions of the operator actions. The results also show that the solidification of pressurizer is not occurred and the liquid-vapor mixture does not flow out through pressurizer PORV. The behavior of primary pressure during pressurizer PORV actuation is poorly predicted because the actual behavior of pressurizer PORV could not be modelled in the present simulation.

Executive Summary

The plant transients following an inadvertent safety injection signal have a relatively high probability of occurrence in the current Westinghouse design. It is well known that the severity of the above event is not so great, however the effect of core over-cooling and pressurizer solidification may become serious in conjunction with the operator misoperation.

There are some experiences of events described above in the Westinghouse 3-loop power plant in Korea. One of these event occurred at Kori Unit 3 in September 6, 1990, which is an Westinghouse 3-loop PWR rated at 900 MW(e) and was commissioned in 1985. The event was initiated by a closure of main feed water control valve of one of the steam generators (SG-B) while the reactor power was maintained to 83 % full power. A SG-B low-low level signal generated by the deduction of feedwater flow tripped the reactor. Although the turbine trip signal was also generated by the reactor trip signal, reactor-turbine inter-trip valve was stuck and consequently actual turbine trip was delayed. During this period, the safety injection system was introduced by the low pressure signal of main steam line and increased the primary water inventory.

The event was simulated using the best estimated computer code, RELAP5/MOD3 Version 5m5, with the proper estimations of operator actions to identify the important thermal-hydraulic phenomena of inadvertent safety injection and to demonstrate the code applicability

to the full scale plant analysis. There are three types of digital records from plant computer in Kori Unit 3 ; the computer daily logging sheet, pre-post trip review record, and sequence record of events. However most plant data were lost due to the computer malfunction during the transient, therefore calculational results were compared with the plant analog strip chart record.

It was found that the code predicted well the plant behaviors, with sufficient accuracy, indicating the code's capability to this type of transients. The analysis shows that the core over-cooling was prevented by terminating the safety injection, and the over-pressurization due to the solidification of pressurizer was prevented with the actuation of pressurizer PORV and injection of auxiliary feedwater. The analysis shows also that the residual heat can be removed by the safety injection and establishment of the natural circulation. However there are some differences between the calculational results and the plant data, particularly in the cold leg temperature. It may be considered that this discrepancy was mainly due to the temperature measurement errors in the low flow rate.

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1. Introduction

The RELAP5 code has been assessed with a great number of data of the separate and integral effect test facilities, but the assessments using the data of full scale power plant were scarce. The resulting data from real power plants can be used to eliminate the scaling problems in small scale test facilities. However the plant data are scarcely available and have great uncertainties compared with the well instrumented test facility data.

The plant transients following an inadvertent safety injection have a relative high probability of occurrence in the current Westinghouse design and the above incidents are considered to be American Nuclear Society (ANS) Condition II events. We have some experiences of these events in the Westinghouse 3-loop nuclear power plants in Korea. One of these events occurred at September 6, 1990 in Kori Unit 3, which is a Westinghouse 3-loop PWR rated at 900 MW(e) and was commissioned in 1985. The event was initiated by a closure of main feed water control valve of one of the steam generators while the reactor power was maintained at 83% full power. A SG-B low-low level signal generated by the deduction of feedwater flow tripped the reactor. Although the turbine trip signal was also generated by the a reactor trip signal, reactor-turbine inter trip valve was stuck and thus actual turbine trip was delayed. During this period the safety injection was actuated by the low pressure of main steam line and increased the primary water inventory.

Such incident result the core over-cooling and pressurizer solidification, thereby the operator should determined if the safety injection signal should be blocked. For a spurious occurrence, the operator would stop the safety injection and maintain the plant in the hot shutdown condition. It is well known that the severity of the above event is not so great, however the effect of core over-cooling and pressurizer solidification may become serious in conjunction with the operator misoperation.

According to the actual sequence of plant transient, the event was simulated using the best estimated computer code, RELAP5/MOD3 [1] Version 5m5, with the proper estimations of operator actions to identify the important thermal-hydraulic phenomena of inadvertent safety injection and check a applicability of the code for the full-scale plant analysis.

Description of the Kori Unit 3 plant and sequence of events are presented in Chapter 2. The plant simulation model and input description is presented in Chapter 3. Results and discussions are presented in Chapter 4. Conclusions are presented in Chapter 6.

2. Descriptions of Plant and Accident Sequence

Kori Unit 3 nuclear power plant in which an inadvertent safety injection event occurred is located on Yangsan, Kyoungnam, Korea. Kori Unit 3 is a 900 MW(e) 3 loops Westinghouse PWR.

Kori unit 3 has been in commercial operation since Sep. 30, 1985. The major design specifications are summarized in Table 1.

Accident happened at 15:35 Sep. 5, 1990. Before the accident occurred, Kori unit 3 was operated in 83% of full power and 800 MW(e) of turbine generator to adjust the time of refueling. The major operating parameters are presented in Table 2.

Sequence of events and major plant data were recorded by the main computer. However a large portion of digital data was lost due to the malfunction of computer. From the available records[2, 3, 4] of the sequence of event, the event can be summarized as Figure 1.

Based on the limited operation records and strip charts of plant parameters, the accident sequence is reconstructed as follows. While operated in 83% of full reactor power, loss of power for 7300 PCB Group 3 Frame 2 happened due to destruction by overheat. This led to loss of control for SG-B main feedwater control valve, which is an fail-close valve. Level of SG-B began to decrease. When the narrow range level reached to 17%, SG lo-lo level signal caused to trip the reactor.

Although turbine trip signal was generated by reactor trip signal, reactor-turbine inter-trip valve was stuck and thus turbine trip was delayed. During this period, steam in SGs was flowing out to turbine. SGs were depressurized until main steam isolation valves (MSIVs) were closed. Depressurization of steam line actuated safety injection signal and main steam isolation signal.

After safety injection signal, the operators tripped turbine manually and opened PCB-7372 and PCB-7300 to protect generator. Because of opening of PCB-7372 and PCB-7300, automatic power transfer to off-site power following generator trip was failed. Thus off-site power was lost for 3 minutes until operator manually transferred the switch to off-site power. At this time the reactor coolant pumps (RCPs) were failed due to a trouble of 480 V Non-1E diesel generator.

Safety injection was supplied for about 12 minutes. Aux. feedwater was also supplied to SGs sufficiently. Without RCPs, heat transfer to SGs maintained by a natural circulation.

Restarting RCP-A at about 38 minutes after accident, the plant was begun to be recovered to hot shutdown condition.

3. Code and Input Description

3.1. Code Description

The RELAP5/MOD3 Version 5m5 is used for the simulation of the event. There are no model change of the code, however a minor correction is done to overcome an input processing failure with the 3 pump controllers for self initialization control. The incorrect coding on the line 139 of subroutine 'rssi.F', 13a(7) = 0, is replaced with 13a(7) = 2.

3.2. Steady-State Calculation

Steady-state calculation is conducted to obtain the major operating parameters which matches to the pre-accident conditions. Since SI signal is actuated by secondary side, the calculation is aimed at the obtaining of the secondary side conditions, especially, steam generator level.

Nodalization of the plant is shown in Figure 2. In nodalization, the whole system was divided into 227 volumes, 274 junctions, and 289 heat structures. The components are described in the following paragraphs.

The plant is nodalized in 3 loops in order to simulate asymmetric effects between loops. 3 loops are connected to a lower plenum component through 3 downcomer components, which interconnected with cross flow junctions. Lower plenum component is connected to upper plenum component through core component and core bypass channel component.

There are two heat structures in the core component to simulate the fuel rods; one is for the average fuel rods, the other for the hottest rod. Heat generation from the two heat structures is 83% of full power. Scram table is used as shown in Table 3. ANSI/ANS-5.1-1979 decay heat model[3] is used in this calculation.

Pressurizer (PZR) component is connected to loop-B. In the steady-state calculation, PZR component is also connected to time-dependent volume of 15.51 MPa in order to obtain the steady-state system pressure. This volume is disconnected in the transient calculation. Pressurizer heaters and spray are not modeled. PZR PORV (Power Operated Relief Valve) component is modelled and described in the following chapter.

Upper plenum component is connected to each hot leg of three loops, and to upper head component through downcomer upper annulus component and guide thimble component.

Pump components are used. The characteristic curves of RCP(Reactor Coolant Pump) are given for four regions. Pump controllers determine speeds of each pump components with the parameters; 336.85 of gain, 1.0 of proportional time constant, 10.0 of integral time constant, and 4582.16 kg/sec of the desired flow rate.

Feedwater of 7.07 MPa and 226.7 C is supplied to steam generators. The feedwater controllers determine each feedwater flow rates. The given control parameters are as the followings; 1.0 of proportional time constant, 5.0 of integral time constant, 0.5 of the desired level, 0.05 of scale factor Sk, and 10.0 of scale factor Sm.

Flowing the riser component from downcomer component to separator component, feedwater is vaporized. In separator component, separated water flows downward to downcomer component and steam flows upward to steam plenum component. Table 4 shows loss coefficients of several junctions in steam generator, which results in the recirculation ratio, 3.95.

Steam within steam plenum component flows to common header component through steam dryer component, dome component, main steam isolation valve component, and steam line component. Common header component connects turbine component and three steam line components.

Because of the characteristic of the feedwater controller, feedwater flow rate has matched with steam flow rate after a considerably long calculation. Moreover, 0.2 sec is given as the maximum time step because a numerical oscillation in separator component was occurred at the lager time step size.

The results of steady-state calculation are summarized and compared with the plant data in Table 5.

3.3. Transient Calculation

At 10 sec of transient calculation, accident is initiated by closing the main feedwater control valve of loop-B. When the SG narrow range level decreases to 17%, reactor trip signal is generated to scram the reactor after 2 sec of signal delay.

Safety injection signal is generated when one of three steam lines is depressurized below 4.13 MPa. In this case, steam line pressures are compensated with lead/lag controller (50/5). Time delays of 2 sec for SI signal and 10 sec for SI pump start are considered.

Although the safety injection flow rate is calculated by the pump characteristic curve, it is adjusted by multiplying 2/3 based on the operator's records. It is injected for 795 sec until the operator terminates.

MFIV and MSIV are closed in 5 sec after signal. Time delays of 5 sec and 10 sec after SI signal are considered for MFIV and MSIV, respectively.

Pressurizer PORV is opened when pressure is increased to 100 psi (689 kPa) above the reference pressure and closed when decreased to 80 psi (552 kPa) above the reference pressure. Pressurizer pressure is compensated by a proportional-integral-differential controller with the parameters; 2249.7 psia (15.5 MPa) of the reference pressure, 5.0 of proportional gain, and 900.0 of reset time constant.

Motor-driven auxiliary feedwater pumps are actuated by one SG low-low level signal. 60 sec of response time and 10 sec of pump delay is considered. Pumps perform their maximum capacity in 10 sec from start. A turbine-driven auxiliary feedwater pump is actuated by two SGs low-low level signal. 60 sec of response time and 2 sec of pump delay are also considered. Pump performs its maximum capacity in

10 sec from start. 7.31 kg/sec of steam is extracted from SG-A and B to simulate more actually the plant.

Auxiliary feedwater system can be controlled by the operator in the accident situation. And there might be several operations to control auxiliary feedwater flow rate. Although those operations are not recorded, those may be deduced based on the operating parameter records as the followings.

(1) For SG-A, auxiliary feedwater flow rate is controlled above 55% of SG level to match with steam flow rate to the turbine-driven auxiliary feedwater flow rate.

(2) For SG-B, auxiliary feedwater flow rate is reduced to 8.85 kg/sec above 30% of SG level.

(3) For SG-C, it is reduced to 5.44 kg/sec above 10% of SG level. The above deductions are reflected in this calculation.

Trip setpoints described in this chapter are summarized in Table 6.

4. Comparisons of Results and Discussions

The calculation is conducted for 3600 sec of problem time. The calculational results are compared with the plant data. However, the plant data have large uncertainties as shown in Tables 7a and 7b[5]. Furthermore, the large time shifts exist in the records since the pens are positioned on the recorders to be able to cross one another. Therefore, the analysis was performed to access a code performance based on the comparison of trends rather than comparison of absolute values.

The major calculated events are summarized and compared with the plant data in Table 8. The timings of reactor trip and safety injection signal are earlier than those of the plant data. It seems that those discrepancies come from the difference between the time delay considered in the calculation and the actual time delay.

After reactor trip, reactor power rapidly decreases to decay power level as shown in Figure 3. Main feedwater and main steam flow rates rapidly decrease by closing the isolation valves and do not flow as shown in Figures 4 and 5. Major parameters along with the accident sequence are discussed in the following sections.

4-1. Steam Generator Level

As mentioned previously, it was inferred that the operator controlled the auxiliary feedwater flow rate according to the SG levels. Estimated auxiliary feedwater flow rate is shown in Figure 6 as a function of time. The calculational results for SG levels are compared with the plant data in Figures 7a, 7b, and 7c. The direct comparison of calculated values with the plant data may be meaningless because there were many unrecorded operator actions for the recovery of SG levels, and thus the resulted plant data was used for the simulation to estimate the operator actions. The major difference between the calculation and the plant data is that SG levels do not drop below '0' in the initial period of the plant data. It is guessed that levels are not correctly recorded because of the dead band of narrow range level gauge.

Although there are a little time shifts and discrepancies after RCP re-running, the general behavior of water level is well agreed with the plant trend. Therefore it is inferred that the operator actions on the auxiliary feedwater control are reasonably considered.

4-2. Steam Generator Pressure and Temperature

The calculational results for SG pressure are shown in Figures 8a, 8b, and 8c. At the beginning of the event, a low-low level signal due to a sudden decrease of main feedwater actuates reactor trip. Normally the reactor trip causes the turbine trip, but in this transient the turbine stop valve does not closed. The SG pressure decreases following the excess outgoing of steam and reduction of void generation due to the reactor trip. Although the actual pressure was not reached to the safety injection set-point (Steam Line Low Pressure; 4.14 Mpa), the pressure signals led by lead-lag controller generate the safety injection signal which close the Main Steam Isolation Valves (MSIVs).

SG pressures rapidly increase as MSIVs close, and then slowly decrease as the supply of the auxiliary feedwater reaches its capacity so that the secondary heat removal begins to overcome the reactor decay power. PORVs (Power Operated Relief Valves) are simulated to open at 7.76 Mpa. The anlysis shows that the supply of the auxiliary feedwater alone can provide the sufficient heat removal capability without the operation of the PORVs.

While safety injection water cools down the primary side and auxiliary feedwater is supplied to cooldown the secondary side, SG pressures gradually decrease. This decrease is continued until safety injection is terminated and then SG pressures are remained constantly. After about 1800 sec, SG pressures increase gradually because of the step-wise reductions of auxiliary feedwater flow rate. After RCP-A re-running at 2376 sec, reactor coolant flows through loop by the forced circulation. The coolant flow leads to the increase of heat transfer in SG. The rate of pressure change increases as shown in the figures. The overall trend in the calculated secondary pressures is in good agreement with the plant data. The discrepancy on the magnitude may be resulted from an estimation of operator action and assumption of the closing characteristics of isolation valves.

The change of temperatures are shown in Figures 9a, 9b, and 9c. It shows that the steam in SGs is under a nearly saturated condition and thus the temperature behavior is almost identical to the pressure behavior.

4-3. Behavior of Primary Side

With the excess cooling of steam generators due to the failure of turbine stop valve closure following the reactor trip, the pressure and temperature in primary side decrease rapidly as shown in Figures 11, 12, 13, and 14. Immediately after the reactor trip, the hot leg and cold leg temperature decrease rapidly. After the steam flow was isolated by SI signal, pressure and temperature turn to increase. The flow coastdown due to RCP trips and the decay heat increase the hot-cold leg difference until the establishment of the natural circulation in the loop. And then RCS temperature gradually decreases because of being cooled down by safety injection and auxiliary feedwater injection. Safety injection flow rate is determined by the characteristic curve of safety injection pump, and thus varies according to RCS pressure as shown in Figure 17.

As reactor coolant contracts due to the decrease of temperature, pressurizer level is decreased. However, as the inventory of reactor coolant increases due to safety injection, pressurizer level stops decreasing and turns to increase according to the amount of safety injection. At this time, pressure also increases to the PORV setpoint according to the increase of the reactor coolant inventory, and PZR

PORV is opened. According to the plant data, it is found that PZR PORV was in the cycling of open/close. PORV was sometimes opened at higher pressure and closed at lower pressure than the setpoint of open/close. It may be resulted from the valve degradation during the process. In the simulation, the above valve degradation was not modeled because of it's complexity.

The calculated RCS cold leg and SG temperatures are compared with the plant data as shown in Figure 15. The calculated RCS cold leg temperatures are almost the same as SG temperatures because the heat is exchanged completely. But, in case of the plant data, RCS cold leg temperatures do not match with SG temperatures, because the temperature measurement in the loop of low flowrate is not reliable.

As safety injection signal is resetted and safety injection is terminated at 861 sec, the increasing rates of RCS hot leg temperatures are blunted and stabilized as shown in Figures 13a, 13b, and 13c. This, in turn, influences to SG pressure to be stabilized.

As the RCS inventory increases no more, PZR level does not increase and is stabilized. As pressure gradually decreases along with heat transfer to the secondary side, PZR pressure decreases below the PORV setpoint and then PORV is closed. There is no more flow through PORV as shown in Figure 16.

As the auxiliary feedwater flow rates are controlled to decrease step by step, RCS temperature increases gradually and reactor coolant swells. Thus, in turn, PZR level increases again. According to the increase of PZR level, PZR pressure increases to open PORV again.

Analysis shows that PZR did not become a solid state and the liquid did not flow out through PORV. Following the restarting RCP at 2331 sec, the operators begin to recover the plant and keep the reactor at the hot shutdown condition by normal charging and letdown. There was no record of operator actions during the recovery phase and thus the simulation was terminated at 3600 sec.

In this study, there are many assumptions or estimations of operator actions and the overall plant trend depends strongly on the operator actions. Moreover the plant digital data were lost during the major events due to the malfunction of data logging computer. However the calculated trends with the proper assumptions of the operator actions are generally agreed with the analog strip chart for the plant trend record, except the loop temperature.

5. Run Statistics

In order to compare the run times with other organization, it should be mentioned that the computer type used is MIPS 32-bit workstation and the operating system is one of UNIX System V, RICSos. The time step sizes and other run statistics are summarized in Table 9. The requested maximum time step size is 0.5 sec in this work. Because of Courant limit, time step sizes become smaller after RCP rerun at 2376.5 sec than before. The total CPU time is shown in Figure 17 as a function of transient time.

The total CPU time required for simulation of the whole transient 3600 sec. is 42092.4 sec. The input processing time is 36.75 sec. and the number of volumes is 227. The number of time steps is 37882. Thus the grind time is 4.89 milli seconds (the grind time = (total CPU time - input processing time) / (number of time step x number of volumes) = (42092.4 - 36.75) / (37882 x 227)).

6. Conclusions

An analysis of an inadvertent safety injection event for Kori Unit 3 was carried out using RELAP5/MOD3 Version 5m5 on MIPS RC-3240 workstation. The results are compared with the available plant data to assess the code applicability. The major conclusions are as the followings.

(1) The calculational trends well agree with the plant data, except the loop temperature.

(2) There are some differences between the calculational result and the plant data. It is due to the uncertainties of the instruments and the estimations of the operator actions.

(3) The asymmetric effects of the plant can be simulated by nodalizing the plant into three loops. The calculation predicts well the asymmetric effects if the predictions of the operator actions are proper.

(4) The characteristics of valve degradation such as pressurizer PORV are recognized to be important in the transient analysis of pressure behavior.

Generally speaking, RELAP5/MOD3 code has the capability of simulating the safety injection incident following the operator actions.

7. Reference

1. "RELAP5/MOD3 Code Manual (Draft)," NUREG/CR-5535, EGG-2596, Aug. 1991.
2. Computed Daily Log Sheet of Kori Unit 3, on 6 September 1990.
3. Computed Post Trip Review Sheet of Kori Unit 3, on 6 September 1990.
4. Computer Sequence of Events Record of Kori Unit 3, on 6 September 1990.
5. "Final Safety Analysis Report, Kori Nuclear Power Plant Unit 3/4," Korea Electric Power Company, 1985.
6. "Emergency Operating Procedure, Kori Nuclear Power Plant Unit 3/4," Korea Electric Power Company, 1990.

Table 1. Design Specifications of Kori Unit 3

Items	Specifications
1. RCS	
- core power	2775.0 MWth
- fuel assembly	17 X 17 standard
- number of FA	157 assemblies
2. Loops	
- number of loops	3 loops
- pump power:	10 MWth / 3 RCP's
3. Pressurizer	
- Pressurizer spray	
- Heaters	Backup Heaters/ Proportional Heaters
- PORV	3 units
- SRV	3 units
4. Steam Generator	Model F Type
- Steam line / SG	1 unit
Flow restrictor/SG	1 unit
- Number of U-tubes	5626
- Swirl vane moisture separator	3 units
- Shevron type dryer	1 unit
- MSIV	1 unit per SG
- MFIV	1 unit per SG
Items	Specifications
5. CVCS	
- Charging Pumps	3 units
- PZR Level Control	Automatic letdown and charging control
6. ECCS	
- Accumulator per Loop	1 unit
- HPSI pumps (Charging pump)	3 units
- LPSI pumps (RHR pumps)	2 units
7. Aux. Feed Water Pump	
- Motor-Driven	2 units
- Turbine-Driven	1 unit

Table 2. Operating Parameters before Occation of the Accident

Parameters	Conditions
Reactor Power	83% of Full Power
Turbine-Generator Power	800 MW(e)
Levels of Steam Generator	50%, 50%, 50%
Boron Concentration	258 ppm
Control Rod Position	D-Bank 172 Steps
RCS Temperature	306.2 C
PCS Pressure	157 kg/cm2

Table 3. Scram Table

Trip Time (sec)	Reactivity ($)
0.0	0.0
0.2	-0.09667
0.4	-0.3664
0.6	-1.2818
0.8	-6.00514
1.0	-7.7259
1.2	-8.5833
1.4	-9.0077
1.6	-9.9426
1.8	-9.3966
2.0	-9.5202
2.2	-9.5703
2.4	-9.667
2.0e20	-9.667

Table 4. Loss Coefficients of Junctions in Steam Generator

Junction #	From Vol.	To Vol.	Loss Coefficient	
			Foreward	Reverse
315	310	320	10.0	10.0
330	330	340	0.45	0.45
330	330	305	0.1	0.1
330	320	330	0.1	0.1

Table 5. Comparisons of Initial Conditions

Parameters	Plant Data	Simulated	Remark*
Reactor Power	83% of Full Power	83% of Full Power	
PZR Pressure	157. kgf/cm2	158.19 kgf/cm2 (2250.1 psia)	800-08
PZR Level	53.0 %	54.95 %	
T(hot) T(cold)	320.7 C 291.7 C	321.5 C 292.1 C	200-01 240-02
S/G Pressure	~70. kgf/cm2	66.58 kgf/cm2 (947.01 psia)	330-01
Feed/Steam Flow	~1.6e6 kg/hr (979.8 lb/sec)	1.54e6 kg/hr (941.13 lb/sec)	
S/G Level	0.50	0.50	

Note: * volume number of calculated value

Table 6. Trip Set Point

Trip	Parameter	Set Point
MFCV-B Close	Event Initiation	10 sec
Rx Trip	1/3 SG lo-lo Level	17%
SI	1/3 Steam Line lo-P	599.7 psia (50/5 lead-lag)
SI Reset	Manual	805 sec after SI
MFIV, MSIV	SI Signal	
Aux Feed		
Motor-Driven	1/3 SG lo-lo Level or SI	17%
Turbine-Driven	2/3 SG lo-lo Level	17%
RCP	Manual	51 sec after Rx Trip
PORV (open)	PZR Pressure	100 psid (PID)
(close)	PZR Pressure	80 psid (PID) (Pref = 2235psig)
Charging & Letdown	Manual	2324 sec after Rx Trip

Table 7a. Uncertainties of Indicators and/or Recorders
(Conditions II, III, and IV)

Indicators/Recorders	Range	Accuracy
T hot (wide range)	-17.8 - 371.1 C	±4%
T cold (wide range)	-17.8 - 371.1 C	±4%
PZR Level	entire distance between taps	+35%
Primary System Pressure	0.0 - 20.7 MPa	8%
Steam Line Pressure	0.0 - 8.96 MPa	14%
Steam Generator Water Level (wide range)	0.0 - 100.0 of Span	±35%
Aux. Feedwater Flow Rate	0.0 - 144.2 m3/hr	±5%

Table 7b. Uncertainties of Indicators and/or Recorders
(Normal Operation)

Indicators/Recorders	Range	Accuracy
T hot (wide range)	-17.8 - 371.1 C	±4%
T cold (wide range)	-17.8 - 371.1 C	±4%
PZR Level (at 15.5 MPa)	entire distance between taps	±3.5%
Primary System Pressure	0.0 - 20.7 MPa	±4%
Steam Line Pressure	0.0 - 8.96 MPa	±4%
Steam Generator Water Level (wide range)	+7 to -5 ft from nominal full load level	±4%
Aux. Feedwater Flow Rate	11.4 - 181.7 m3/hr	±3%

Table 8. Comparisons of Sequence of Events

Event	Plant Data	Simulated
MFCV-B Close	0:10	0:10
Aux Feed (MD) Actuation Signal	0:51	0:42.8
Rx Trip	0:52	0:44.8
SIS	1:03	0:56.2
Aux Feed (TD) Actuation Signal	1:09	1:05.5
RCP Trip	1:43	1:31.8
Aux Feed (MD) (60+10 sec Delay)		1:54.8
Aux Feed (TD) (60+2 sec Delay)		2:07.5
PORV Open		
SI Reset	14:28	14:21.25
Aux Feed-C Closed		15:20.5
Aux Feed-A Partially Closed		22:15.25
Aux Feed-B Partially Closed		28:31.55
Charging & Letdown	39:36	39:29
Aux Feed All Closed	39:39	39:32
RCP Run	39:43	39:36

Table 9. Run Statistics Table

Transient Time(sec)	CPU Time (sec)	Time Step Size (sec)	# of Cycle	Primary Mass Error		Secondary Mass Error	
				(kg)	Fraction	(kg)	Fraction
0.0	36.8	0.02	0	0.0	0.0	0.0	0.0
150.0	1958.9	0.5	1848	236.4	1.36e-3	67.8	4.02e-4
650.0	6673.9	0.25	6097	221.4	1.11e-3	498.1	2.20e-3
1150.0	12033.5	0.25	10728	224.6	1.15e-3	984.0	3.86e-3
1650.0	15783.4	0.25	14009	220.9	1.13e-3	1588.3	5.75e-3
2150.0	19068.7	0.125	16991	230.7	1.18e-3	2972.1	1.05e-2
2200.0	19349.0	0.25	17244	230.4	1.18e-3	3202.1	1.12e-2
2300.0	19996.3	0.5	17834	231.3	1.19e-3	3544.7	1.24e-2
2376.5	20504.5	0.25	18297	229.8	1.18e-3	3853.1	1.34e-2
2876.5	29319.0	0.0625	26306	4.3	2.26e-5	79.8	2.78e-4
3376.5	38149.6	0.0625	34306	9.4	4.99e-5	157.2	5.49e-4
3600.0	42092.4	0.0625	37882	0.9	4.97e-6	176.3	6.15e-4

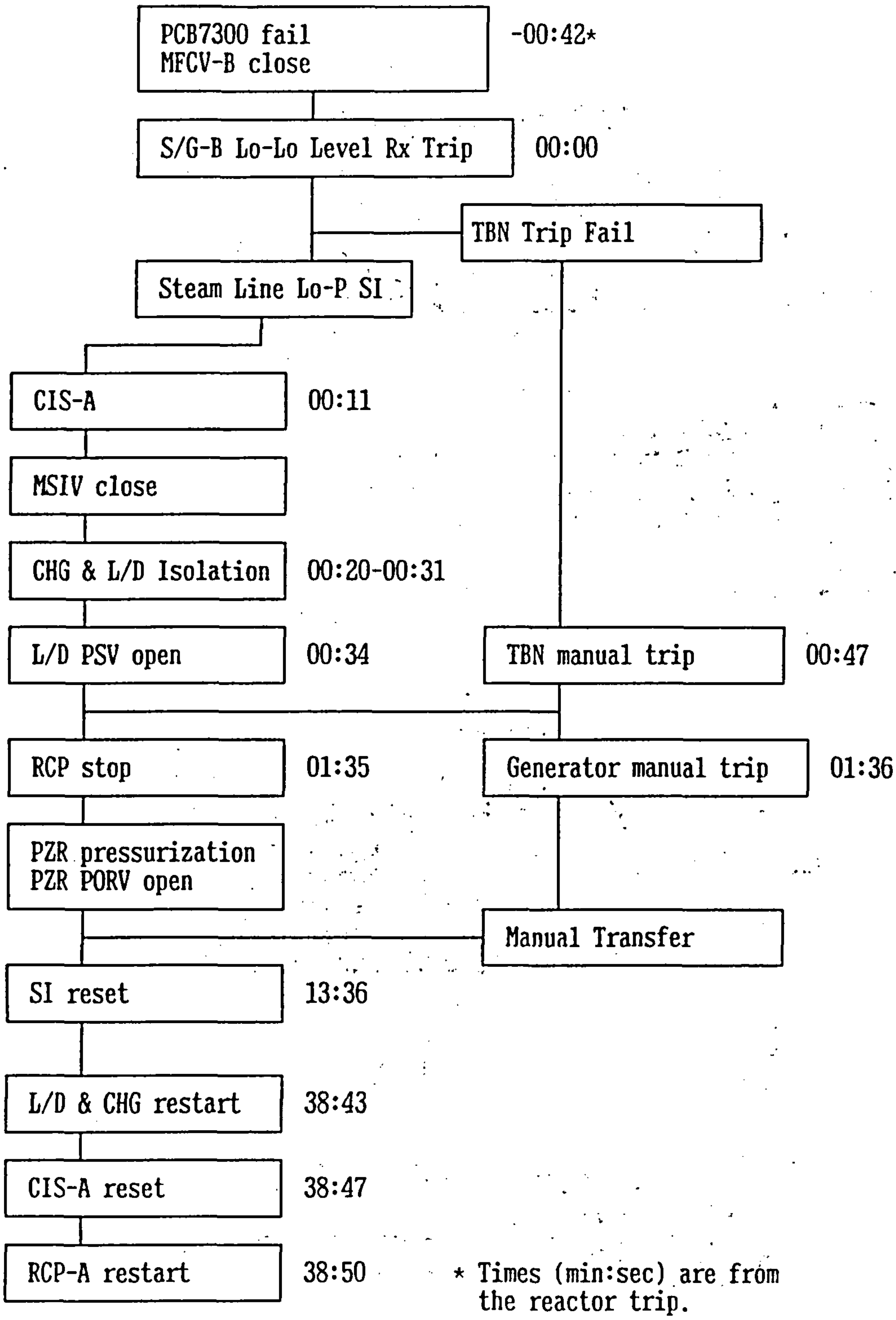

Figure 1. Sequence of Events

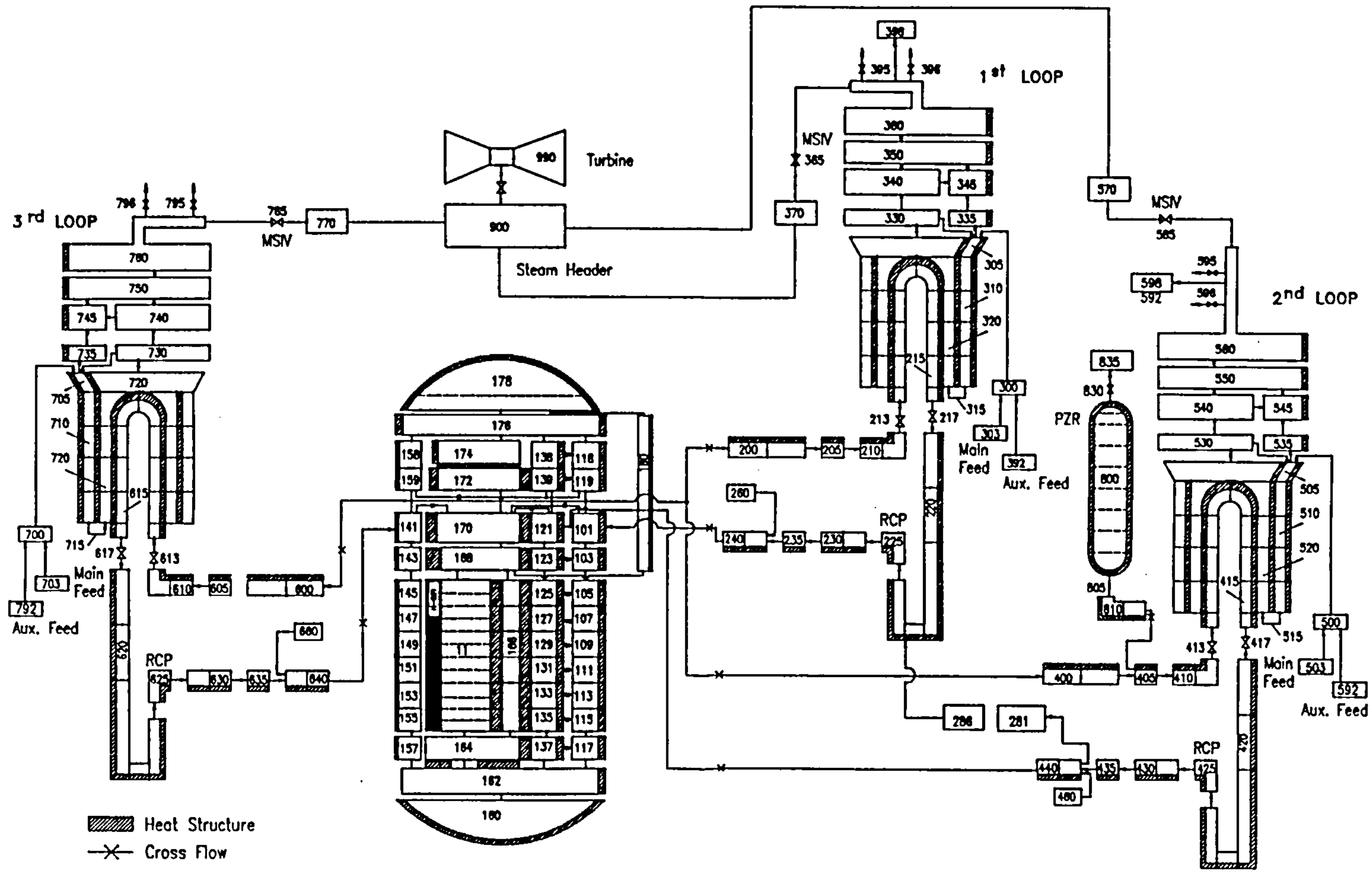

FIG. 2. Kori Unit 3 Nodalization

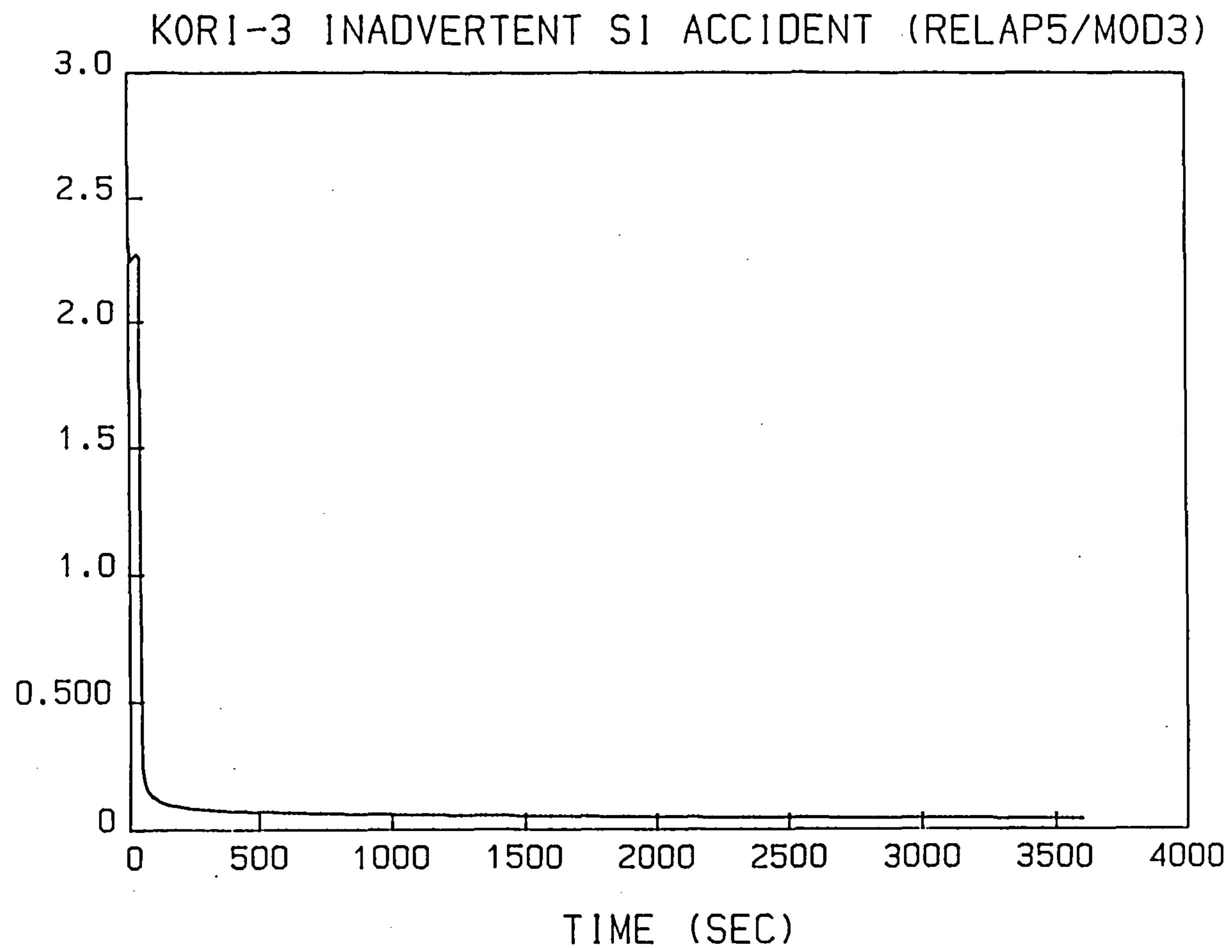

FIG. 3. Reactor Power (Calculated)

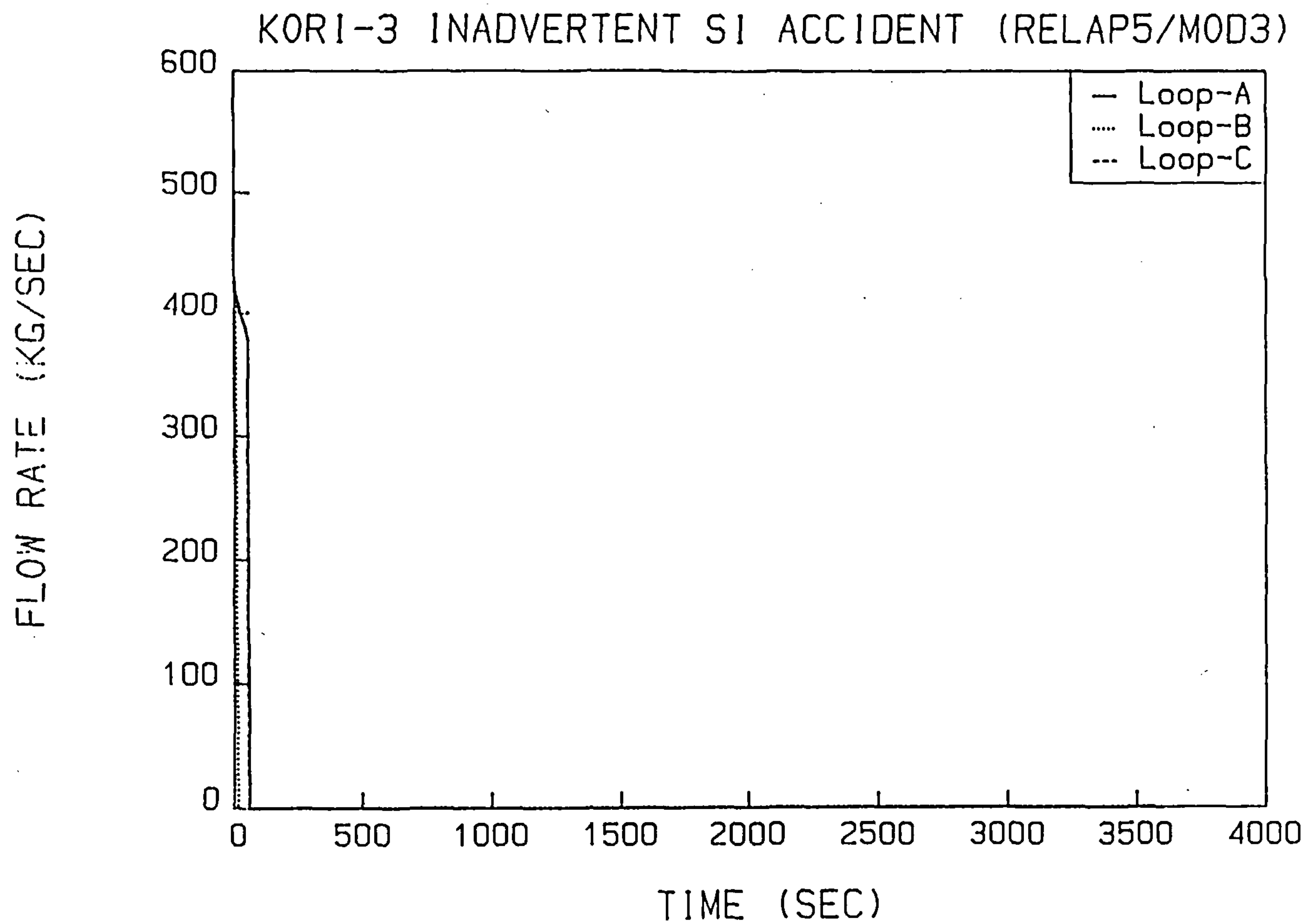

FIG. 4. Main Feed Water Flow Rate (Calculated)

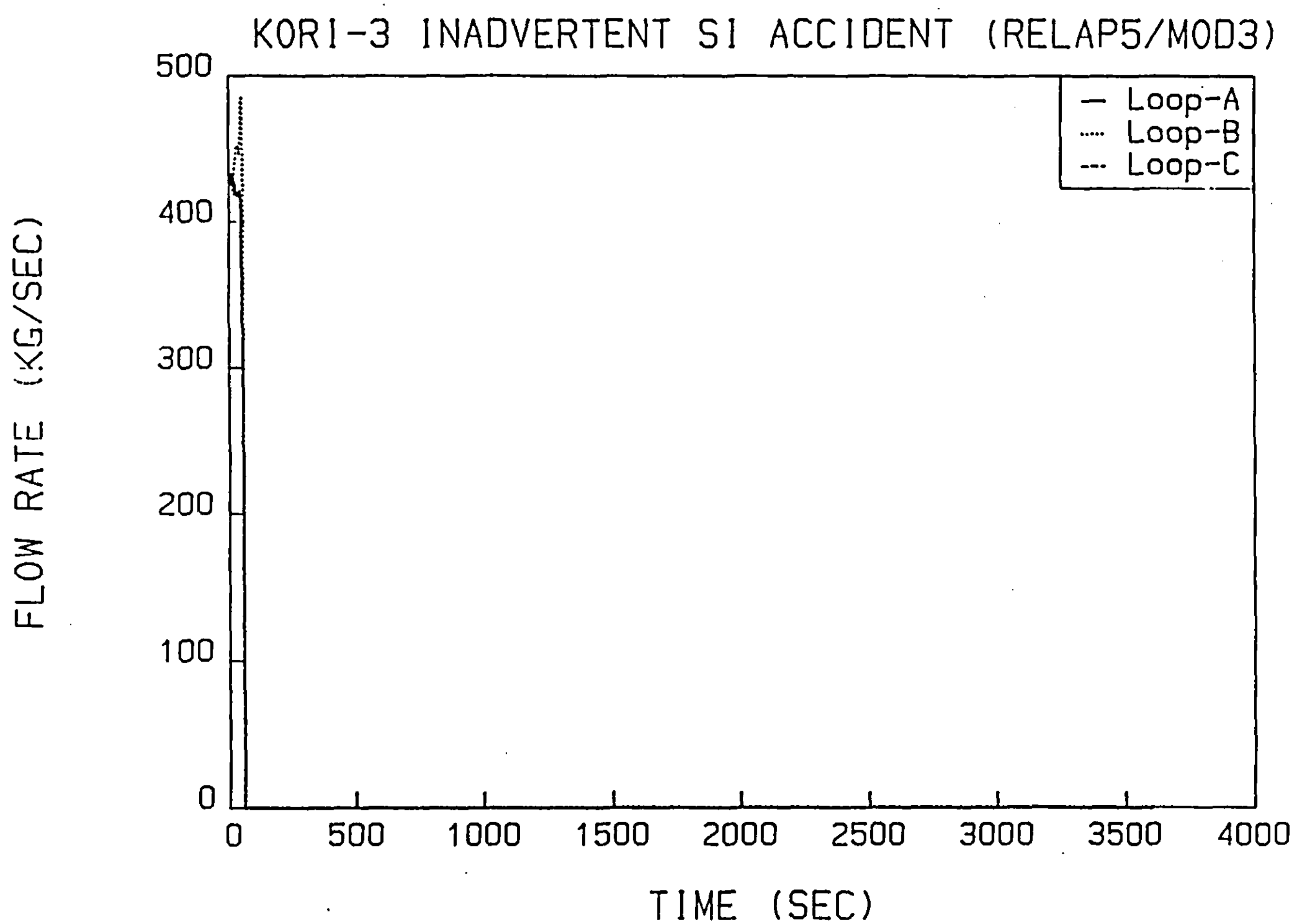

FIG. 5. Main Steam Line Flow Rate (Calculated)

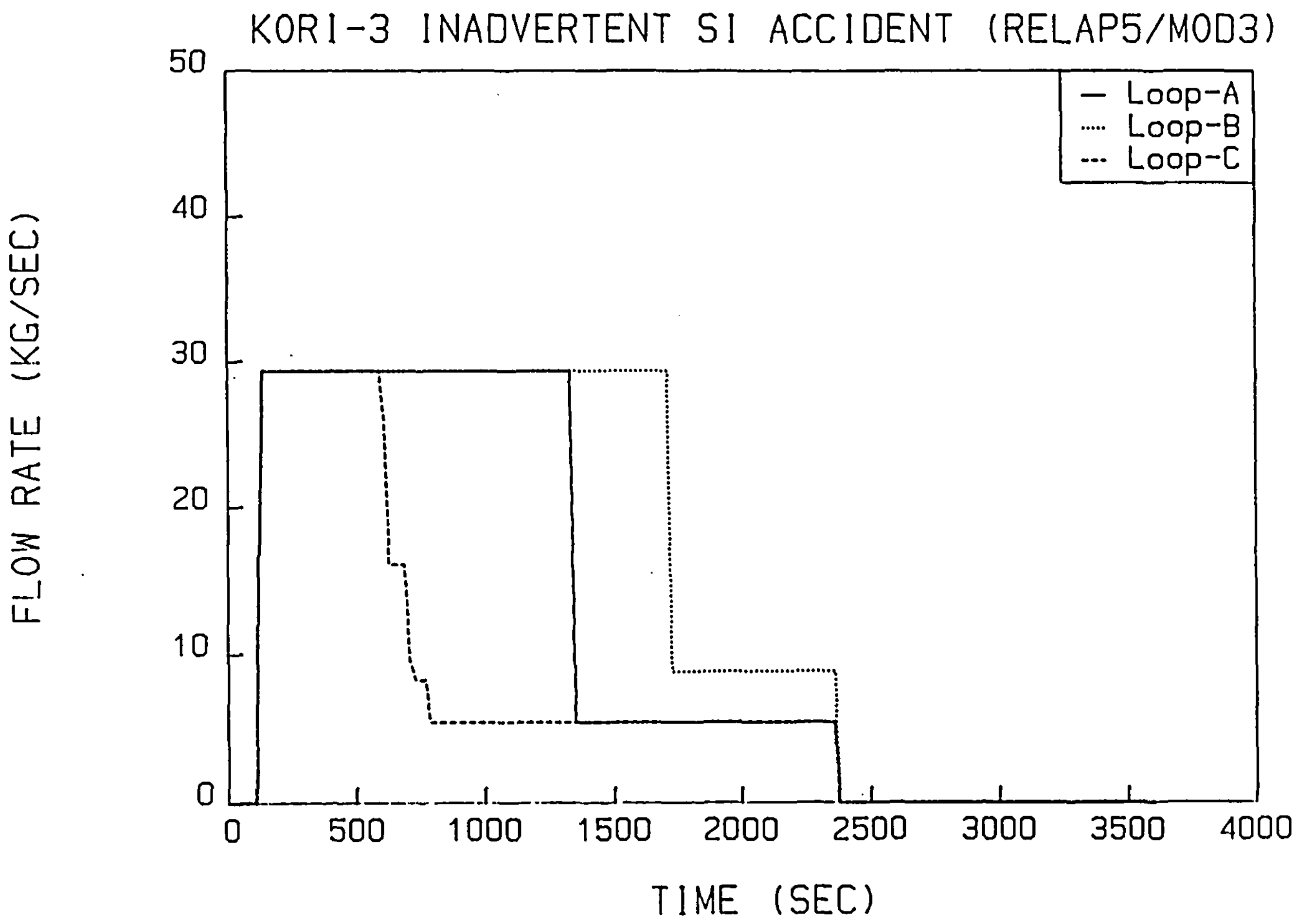

FIG. 6. Aux. Feed Water Flow Rate (Calculated)

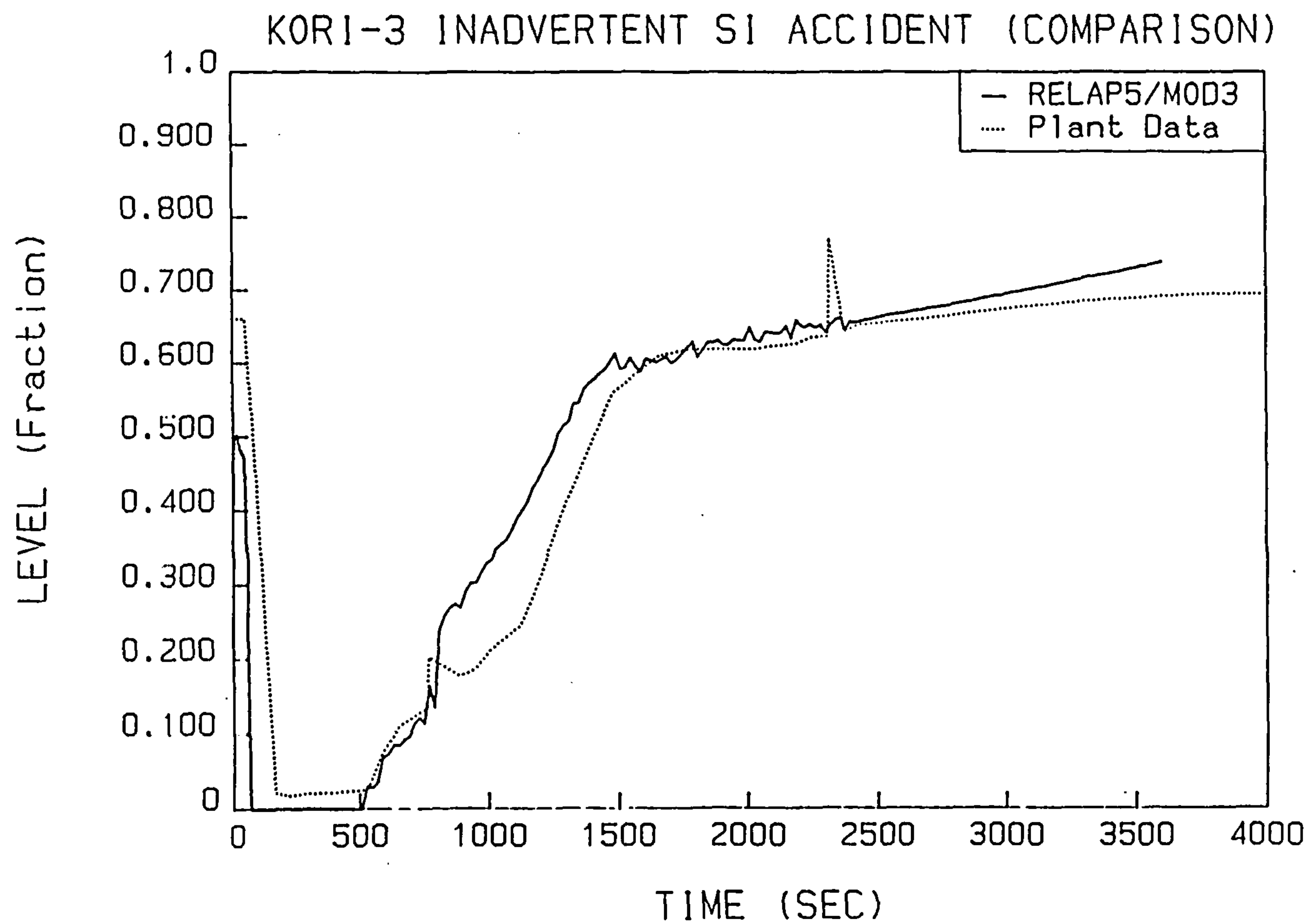

FIG. 7a. Comparison of SG-A Level with Plant Data

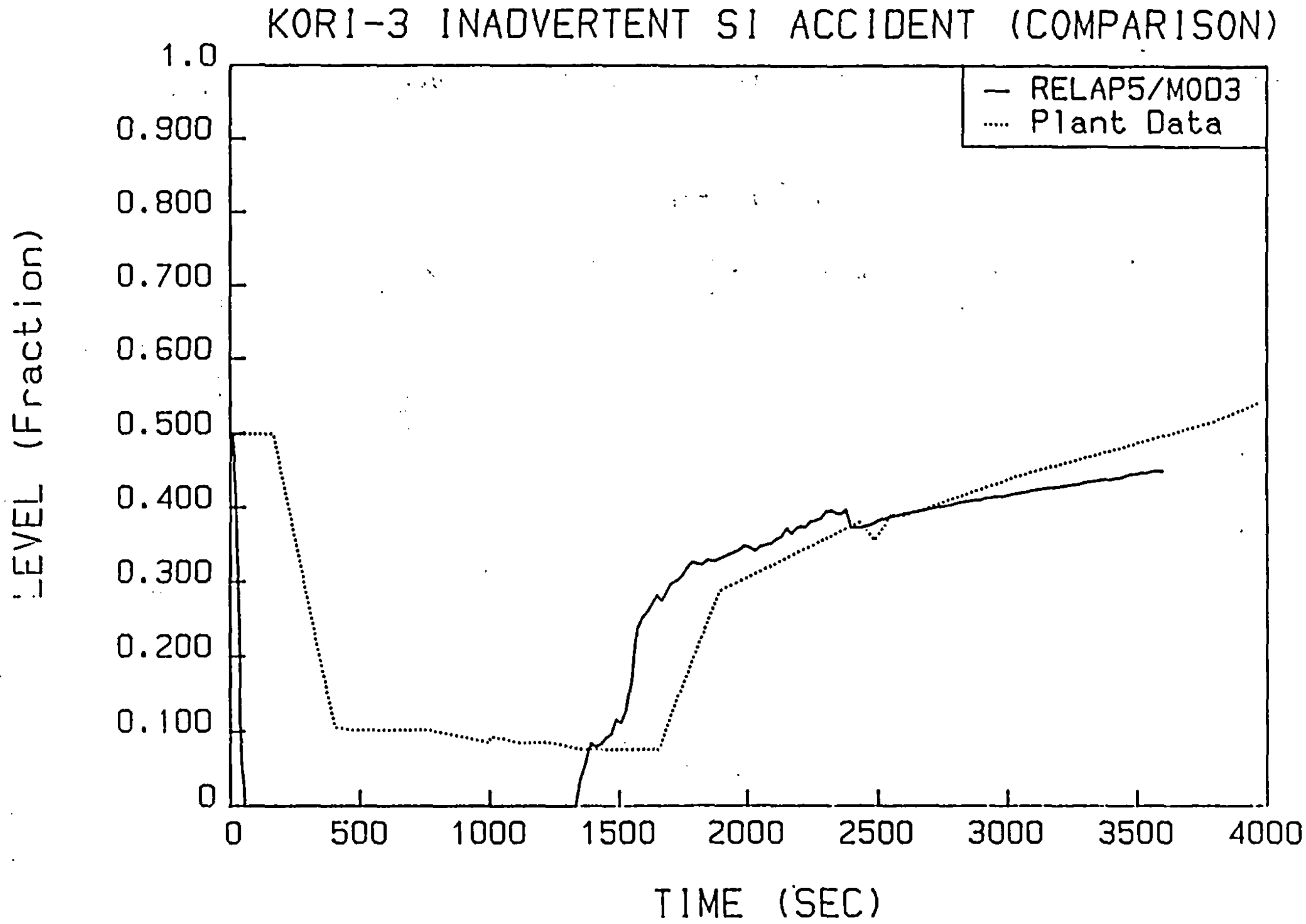

FIG. 7b. Comparison of SG-B Level with Plant Data

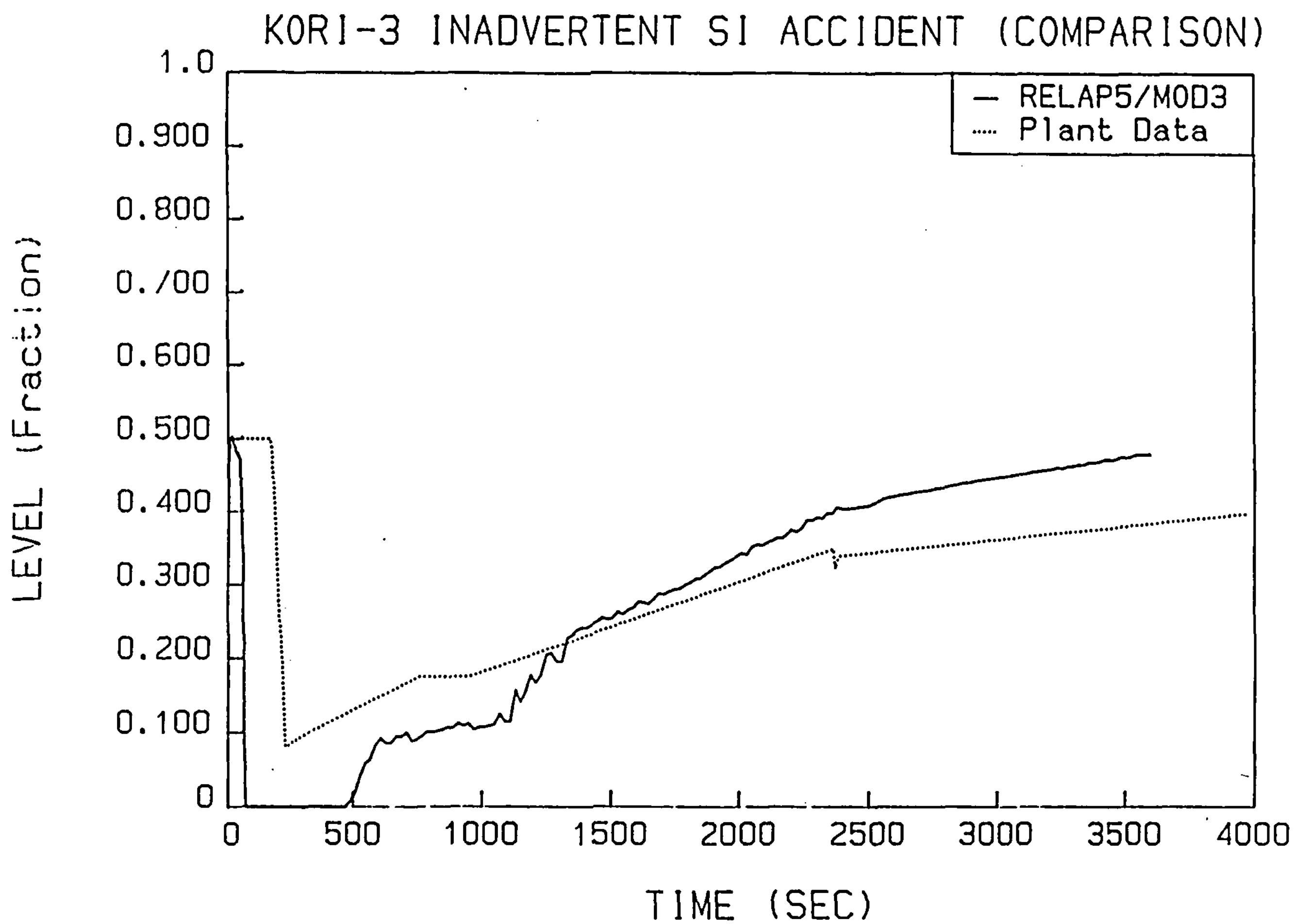

FIG. 7c. Comparison of SG-C Level with Plant Data

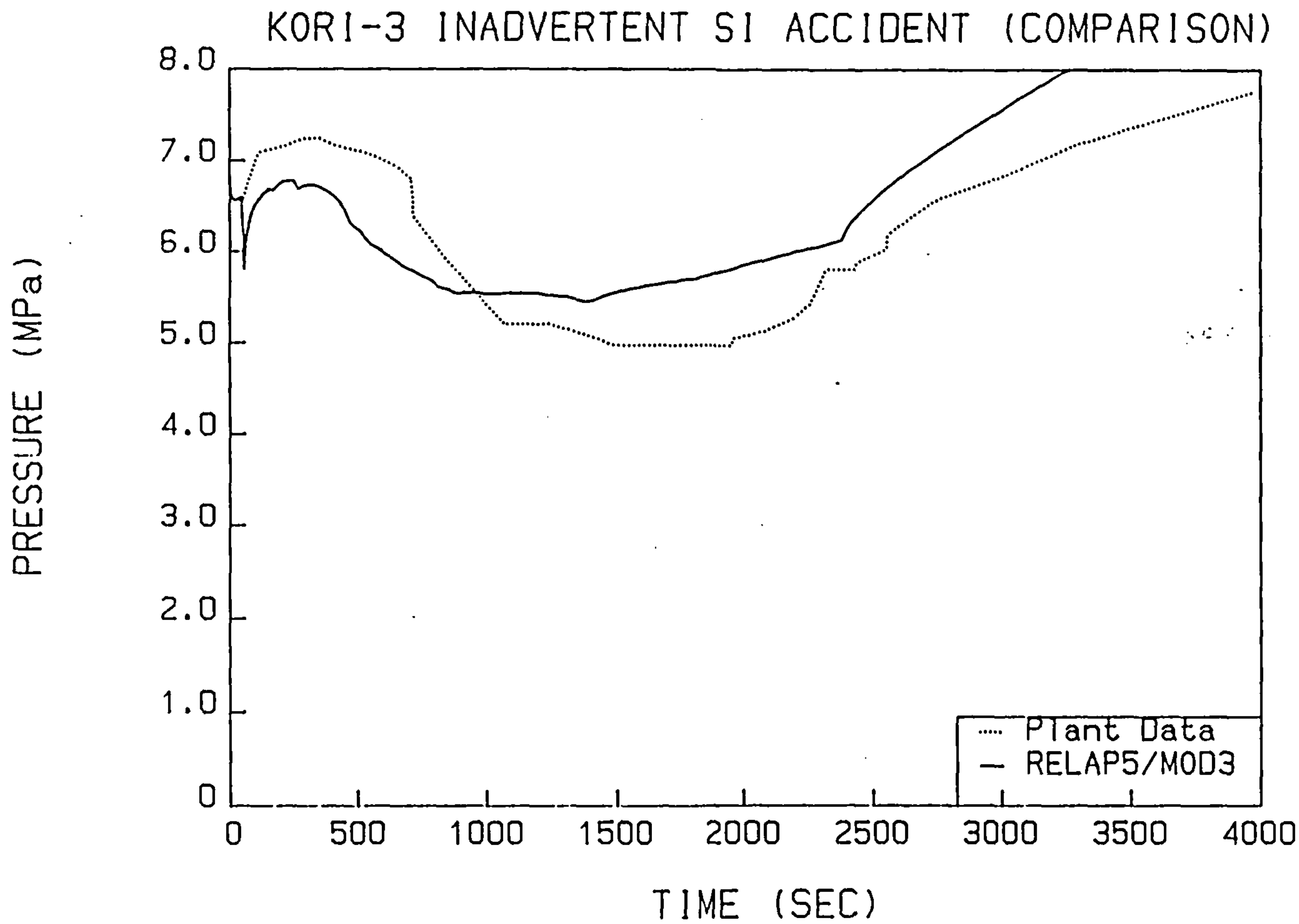

FIG. 8a. Comparison of SG-A Pressure with Plant Data

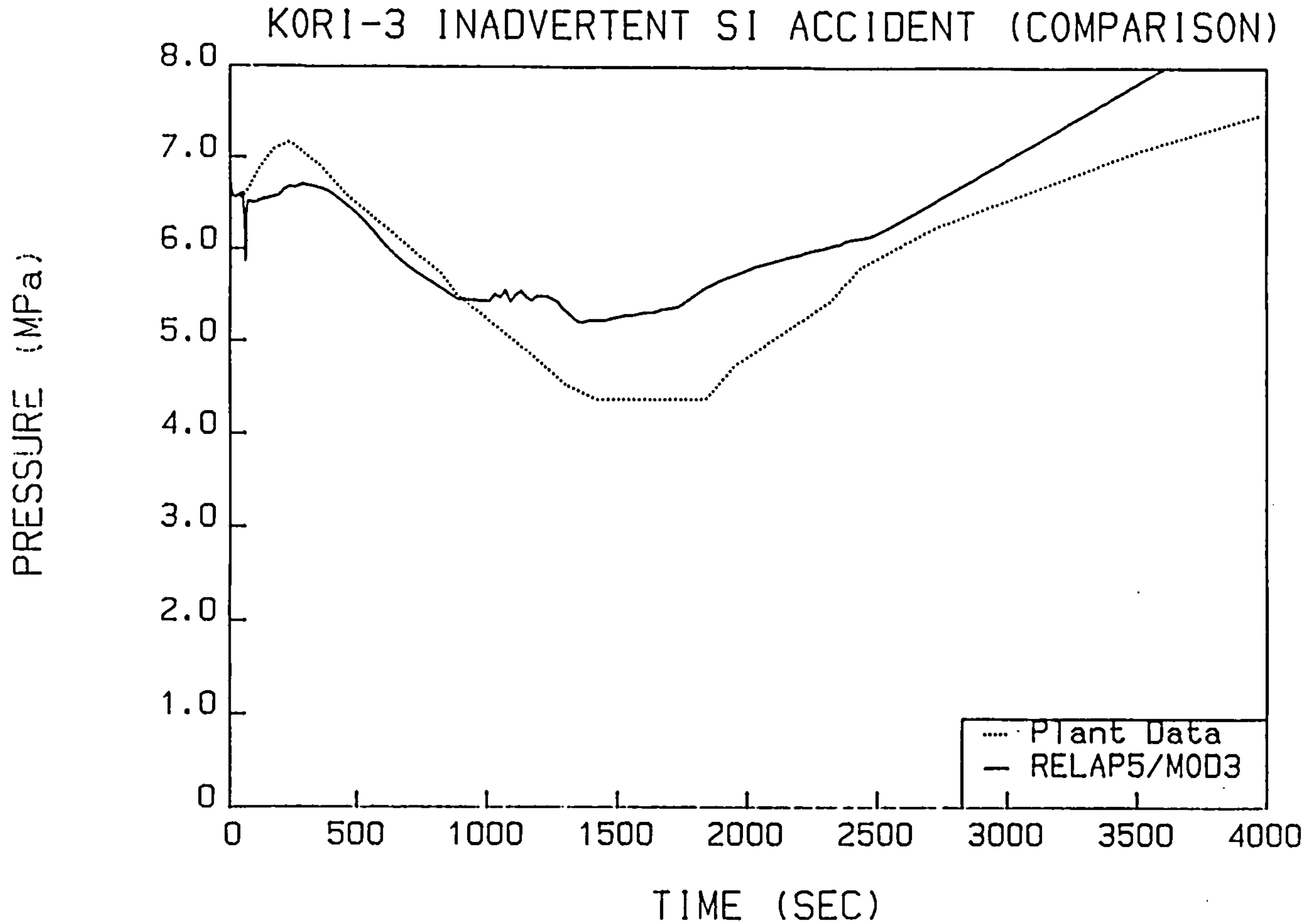

FIG. 8b. Comparison of SG-B Pressure with Plant Data

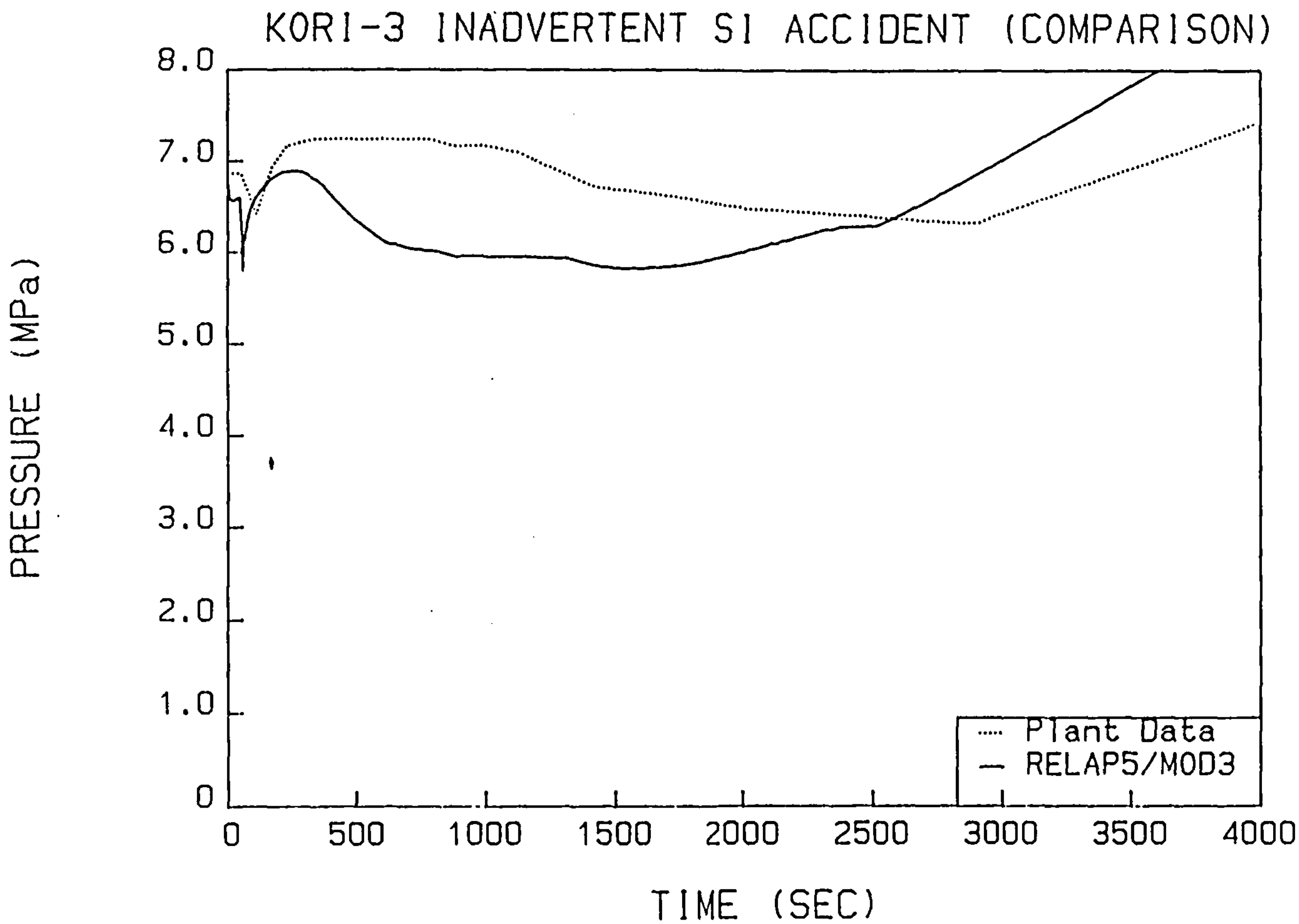

FIG. 8c. Comparison of SG-C Pressure with Plant Data

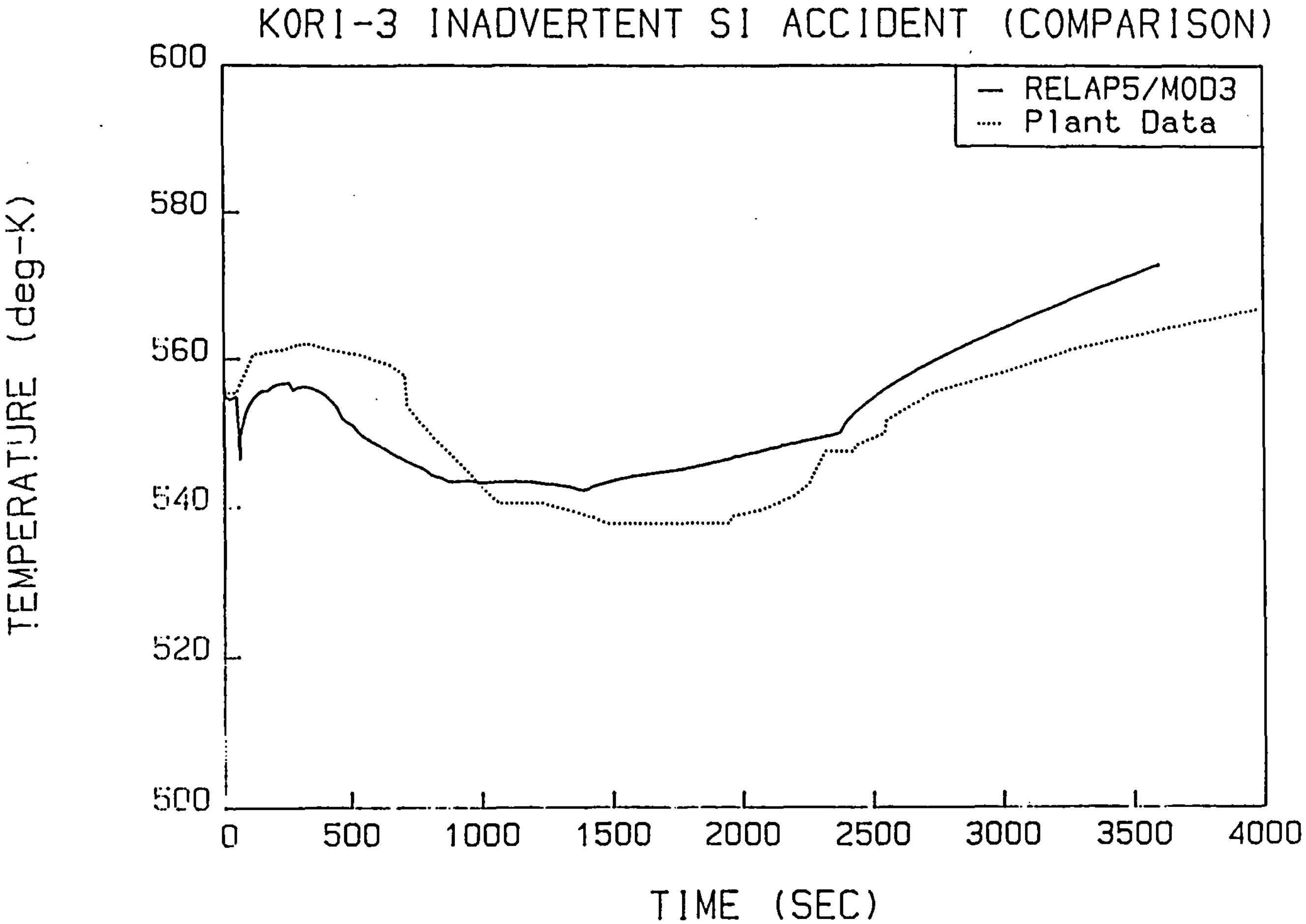

FIG. 9a. Comparison of SG-A Temp. with Data

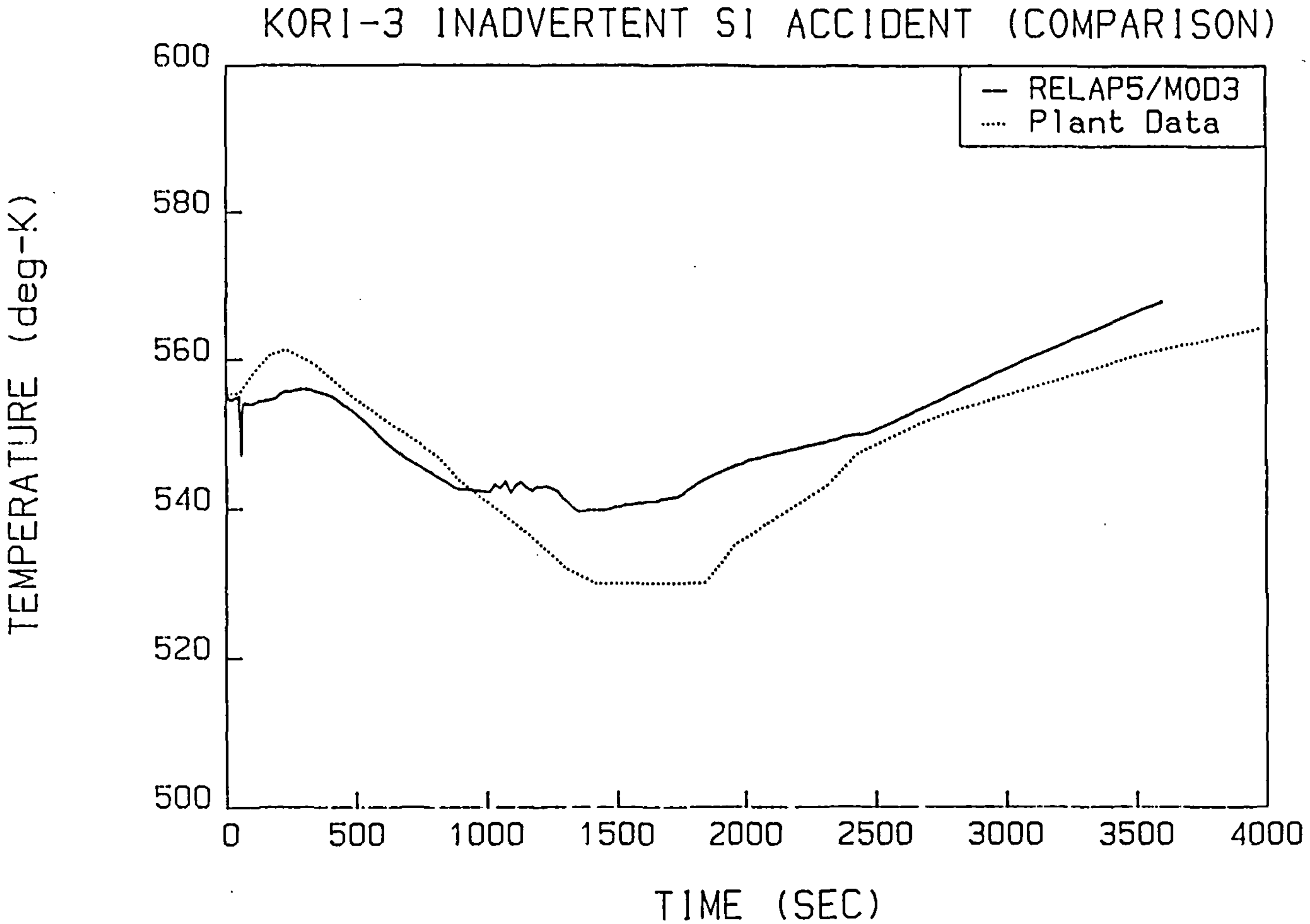

FIG. 9b. Comparison of SG-B Temp. with Data

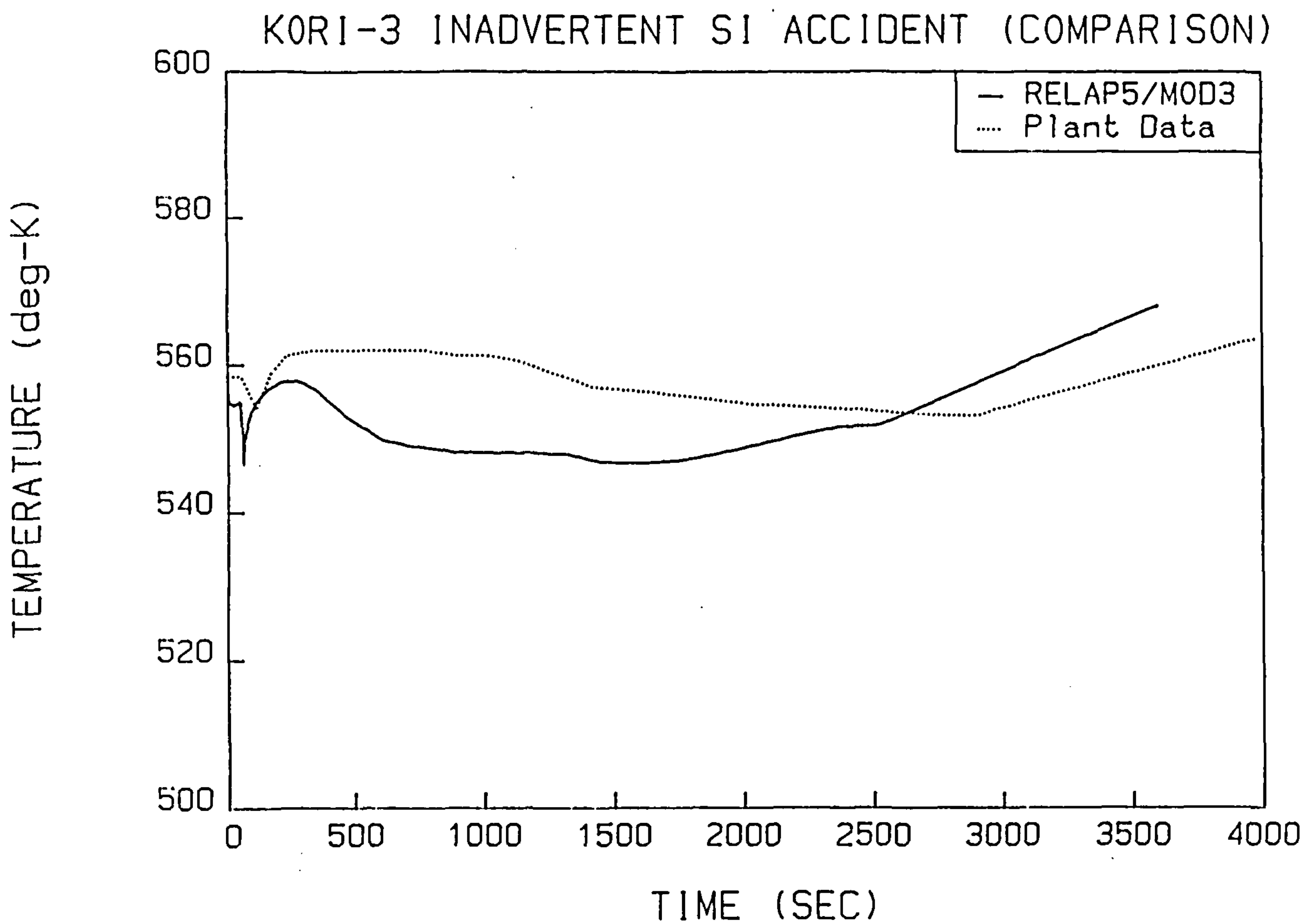

FIG. 9c. Comparison of SG-C Temp. with Data

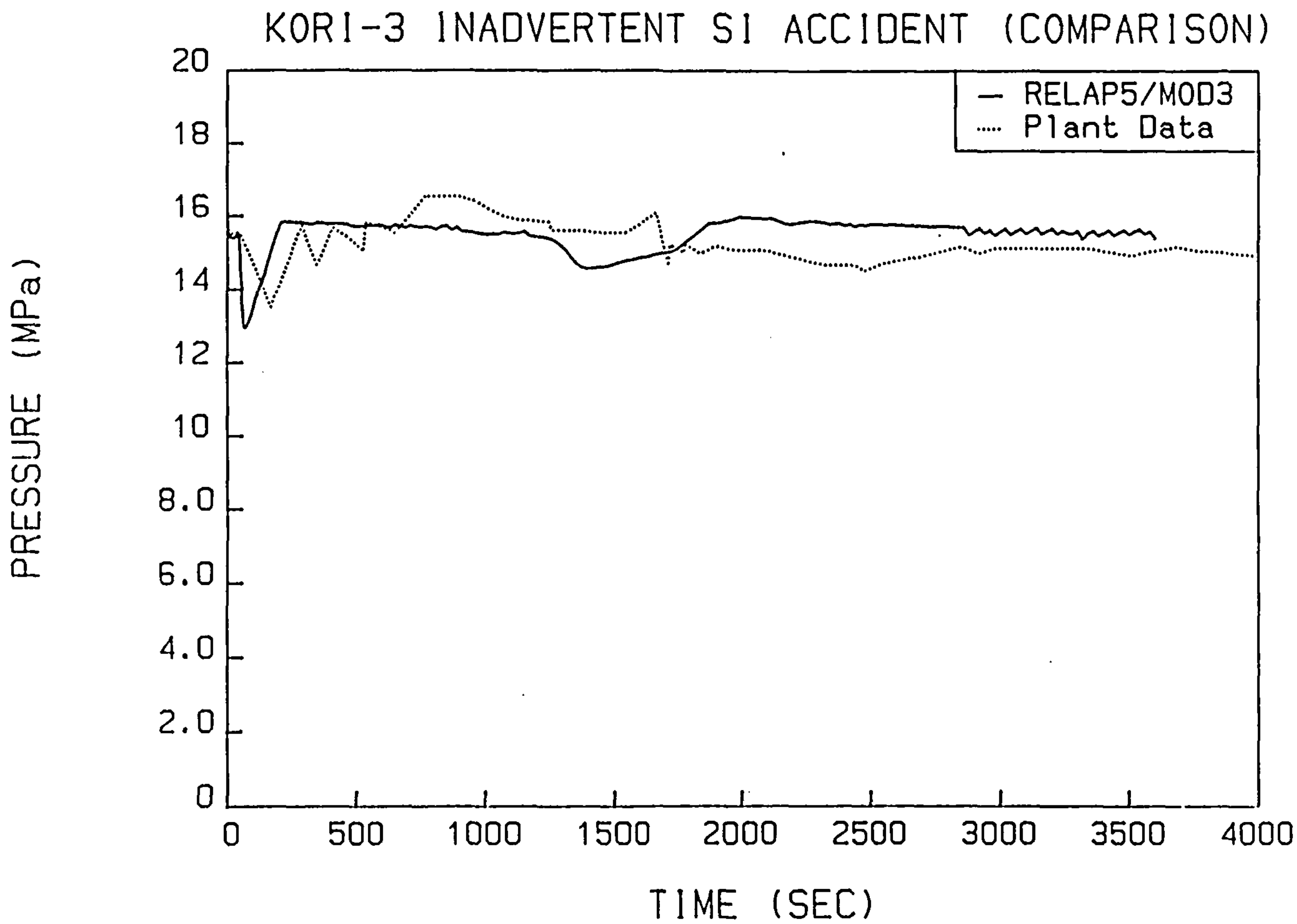

FIG. 10. Comparison of PZR Pressure with Data

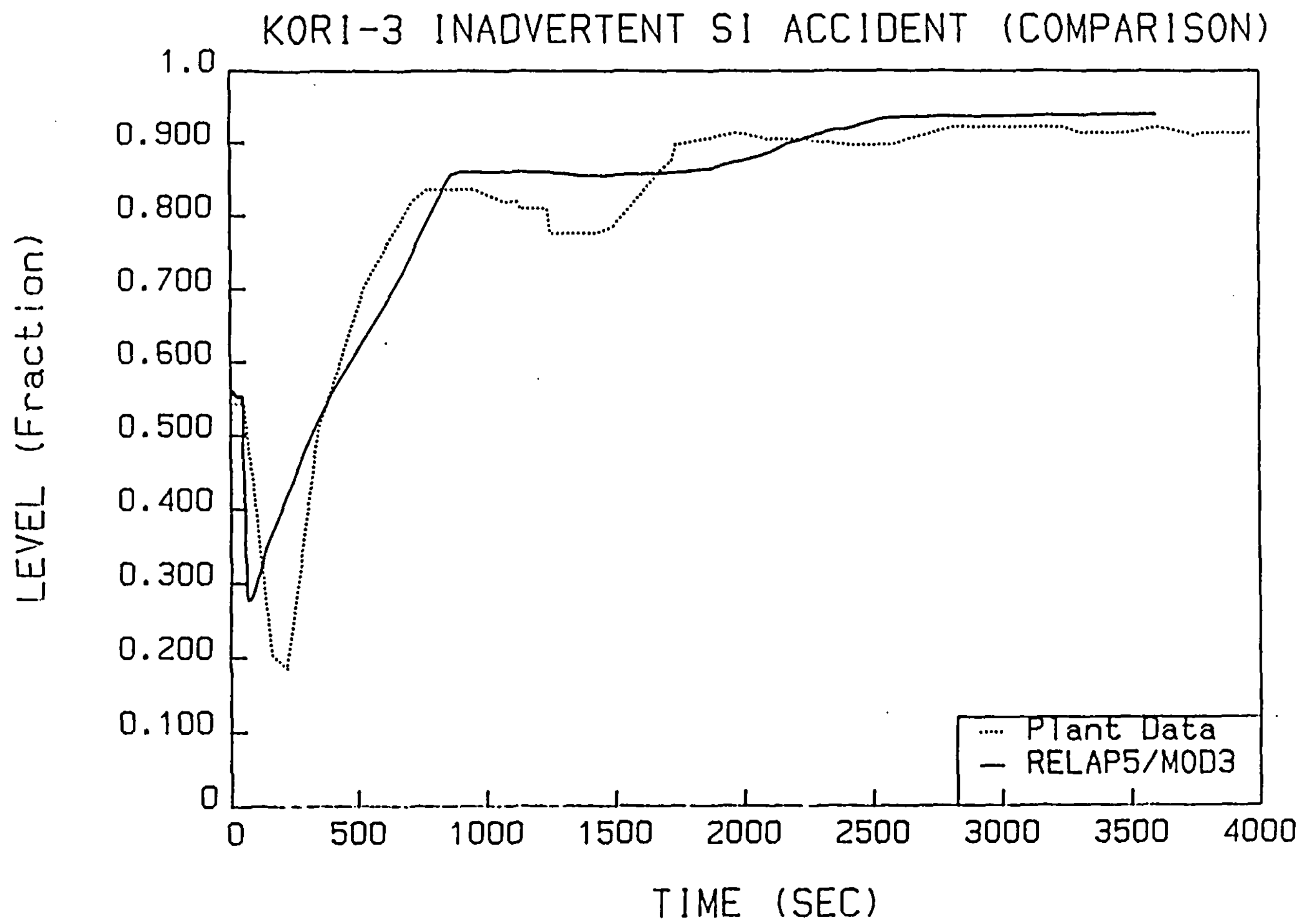

FIG. 11. Comparison of PZR Level with Data

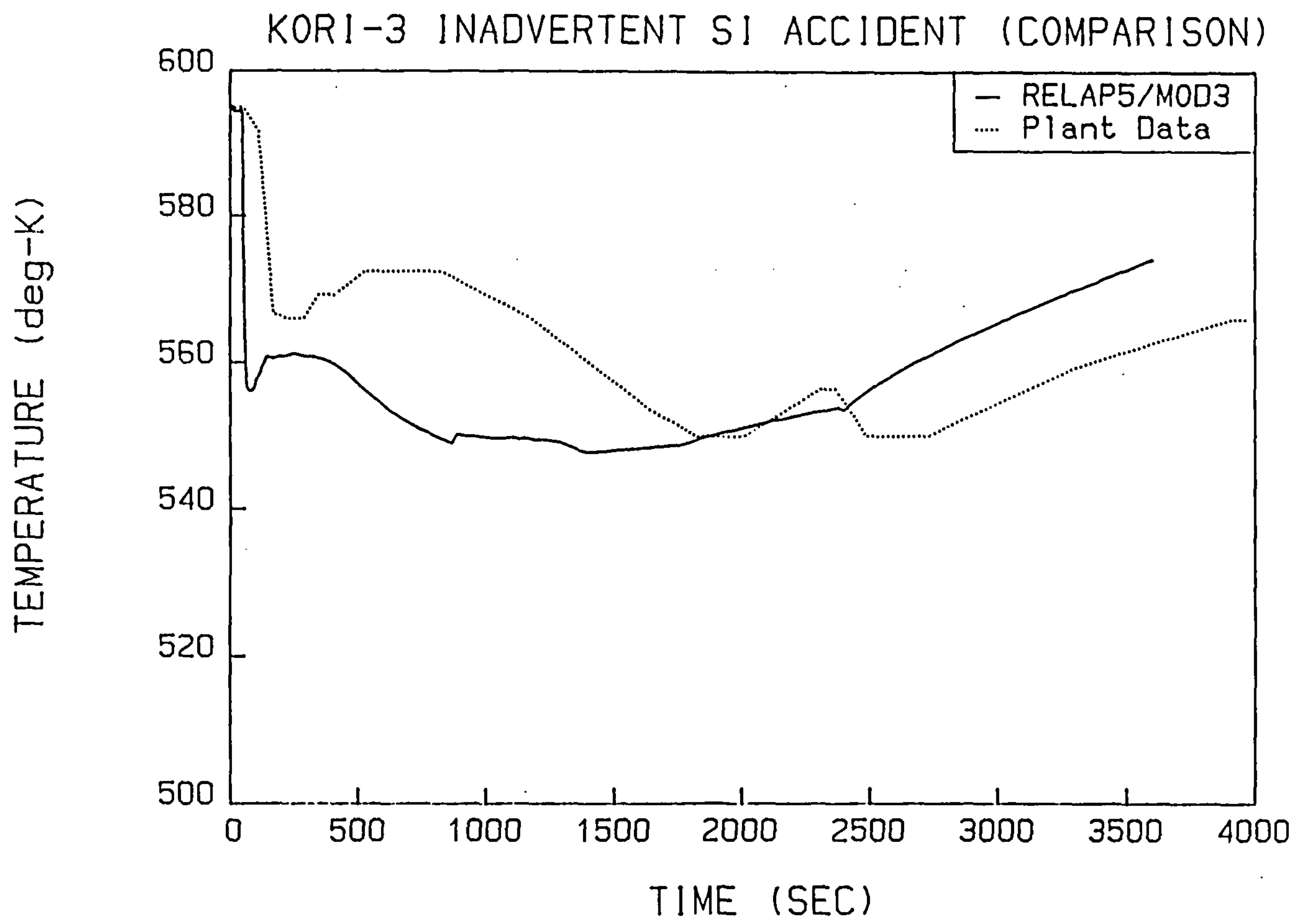

FIG. 12a. Comparison of T Hot (A) with Data

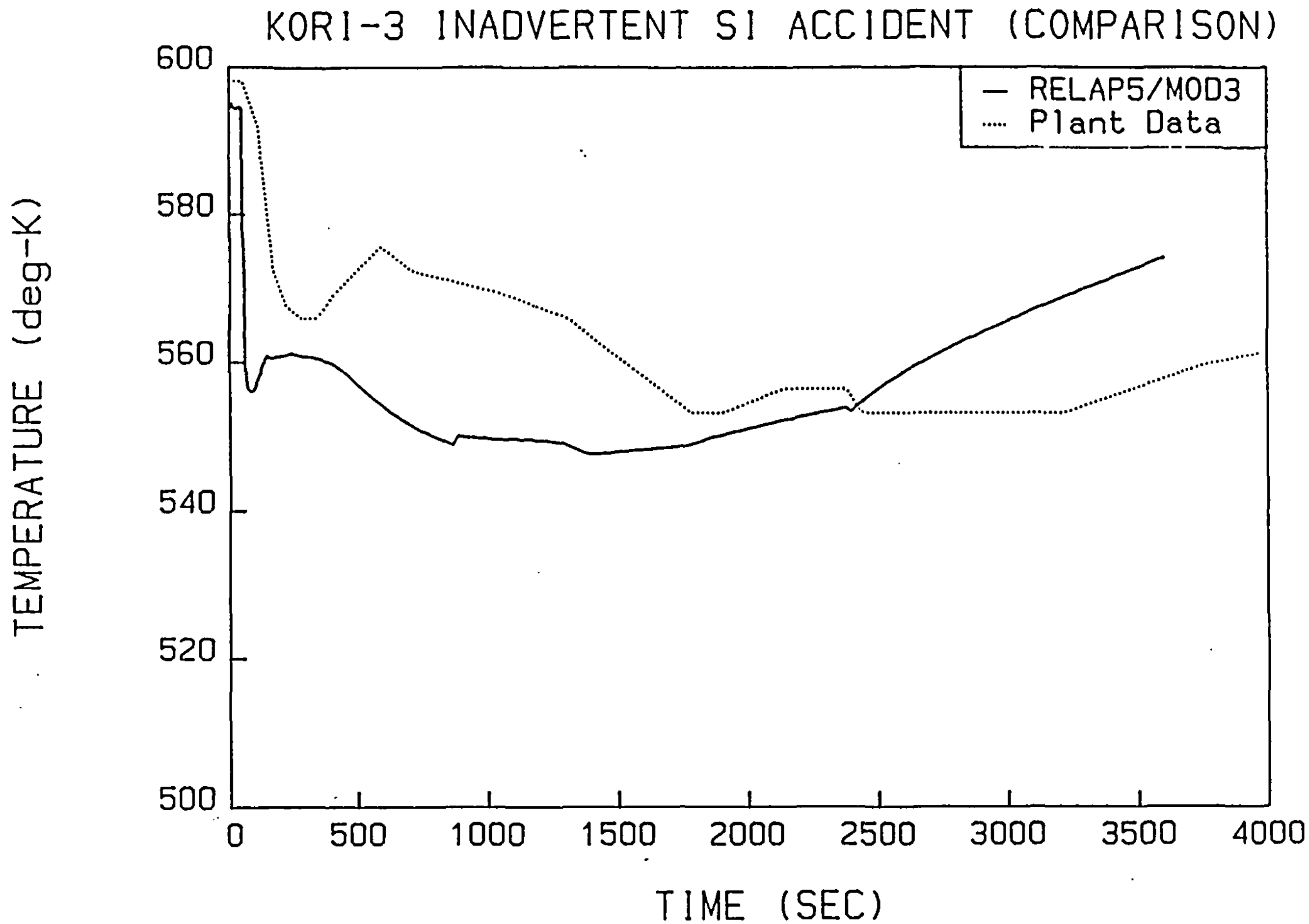

FIG. 12b. Comparison of T Hot (B) with Data

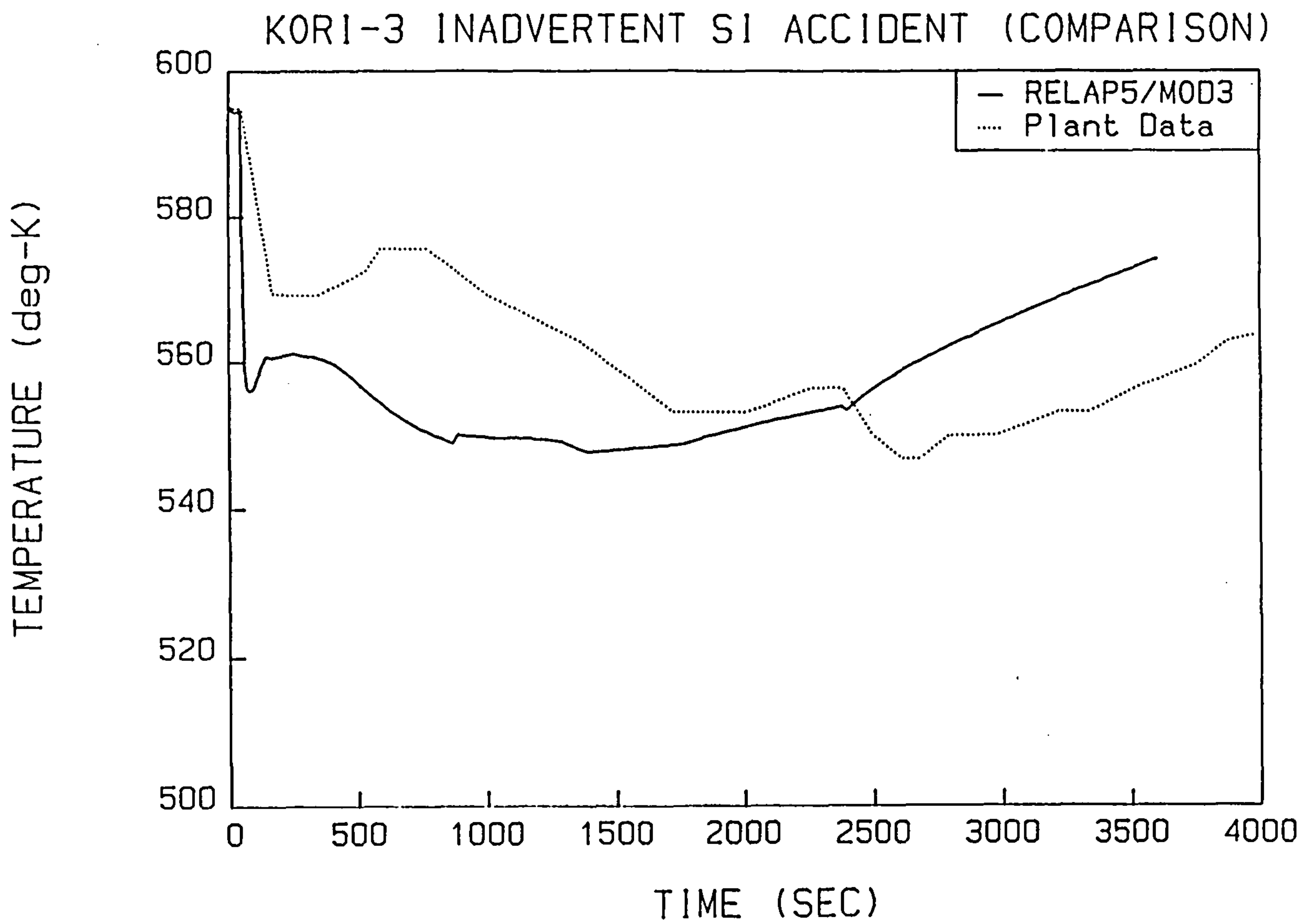

FIG. 12c. Comparison of T Hot (C) with Data

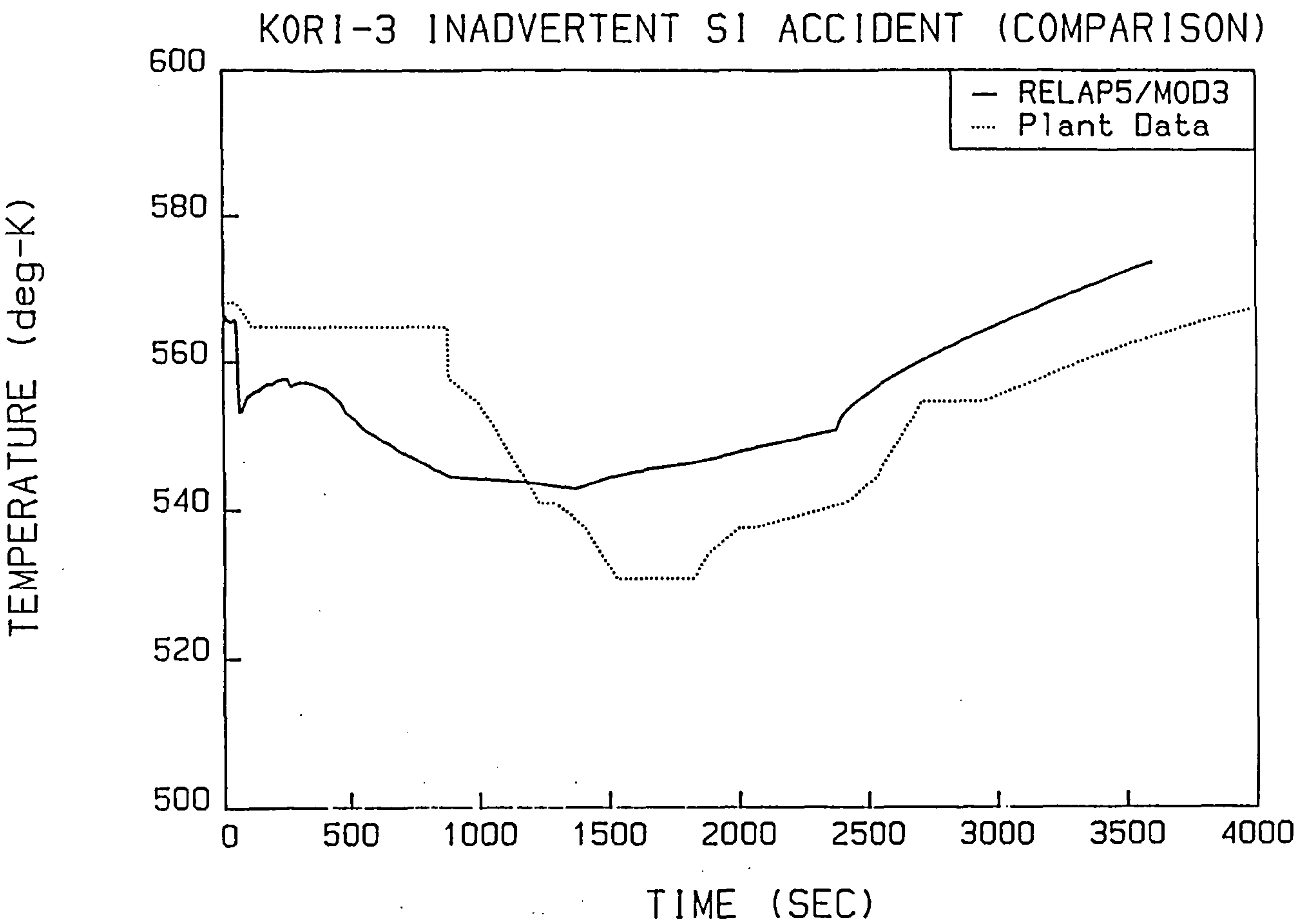

FIG. 13a. Comparison of T Cold (A) with Data

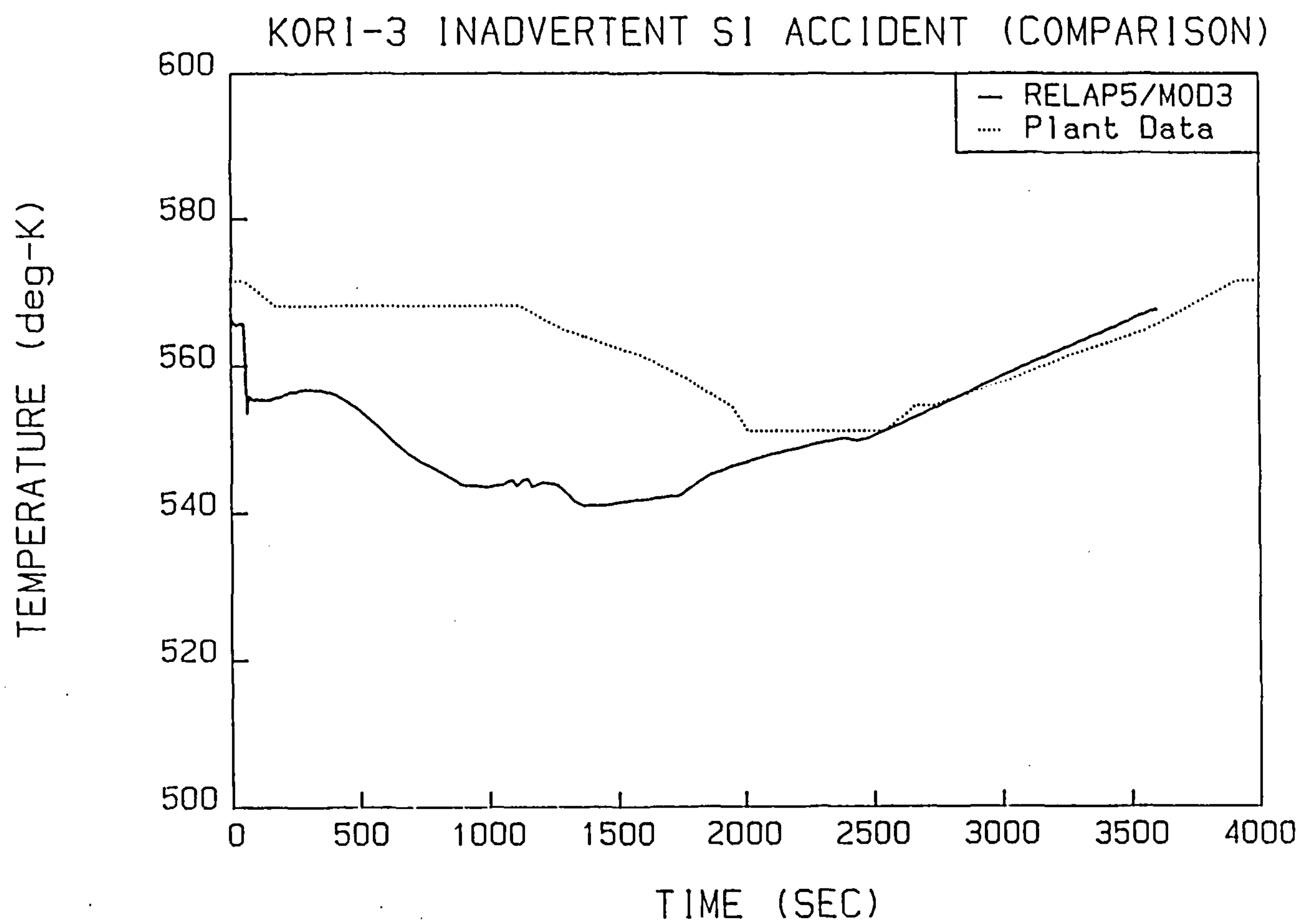

FIG. 13b. Comparison of T Cold (B) with Data

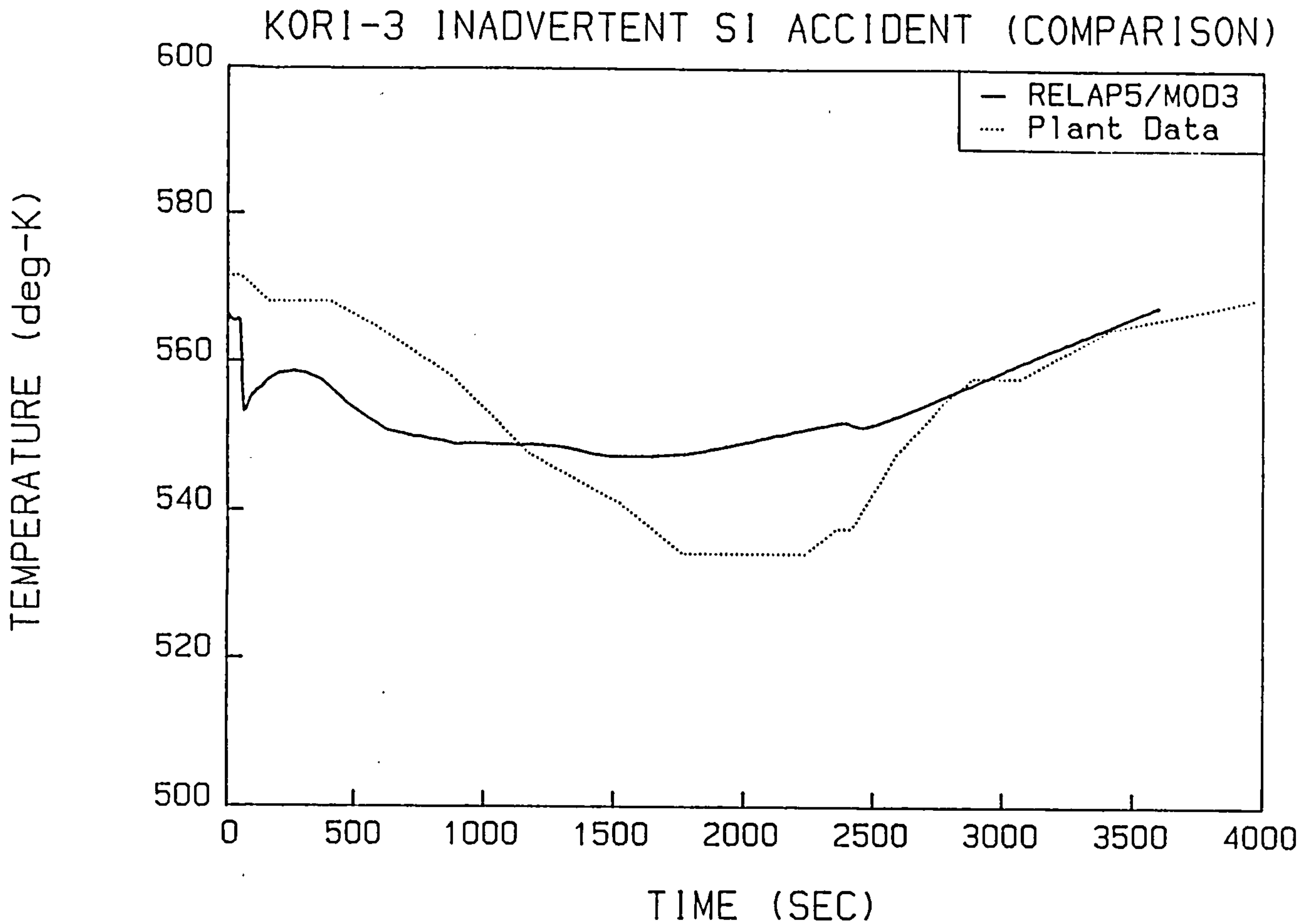

FIG. 13c. Comparison of T Cold (C) with Data

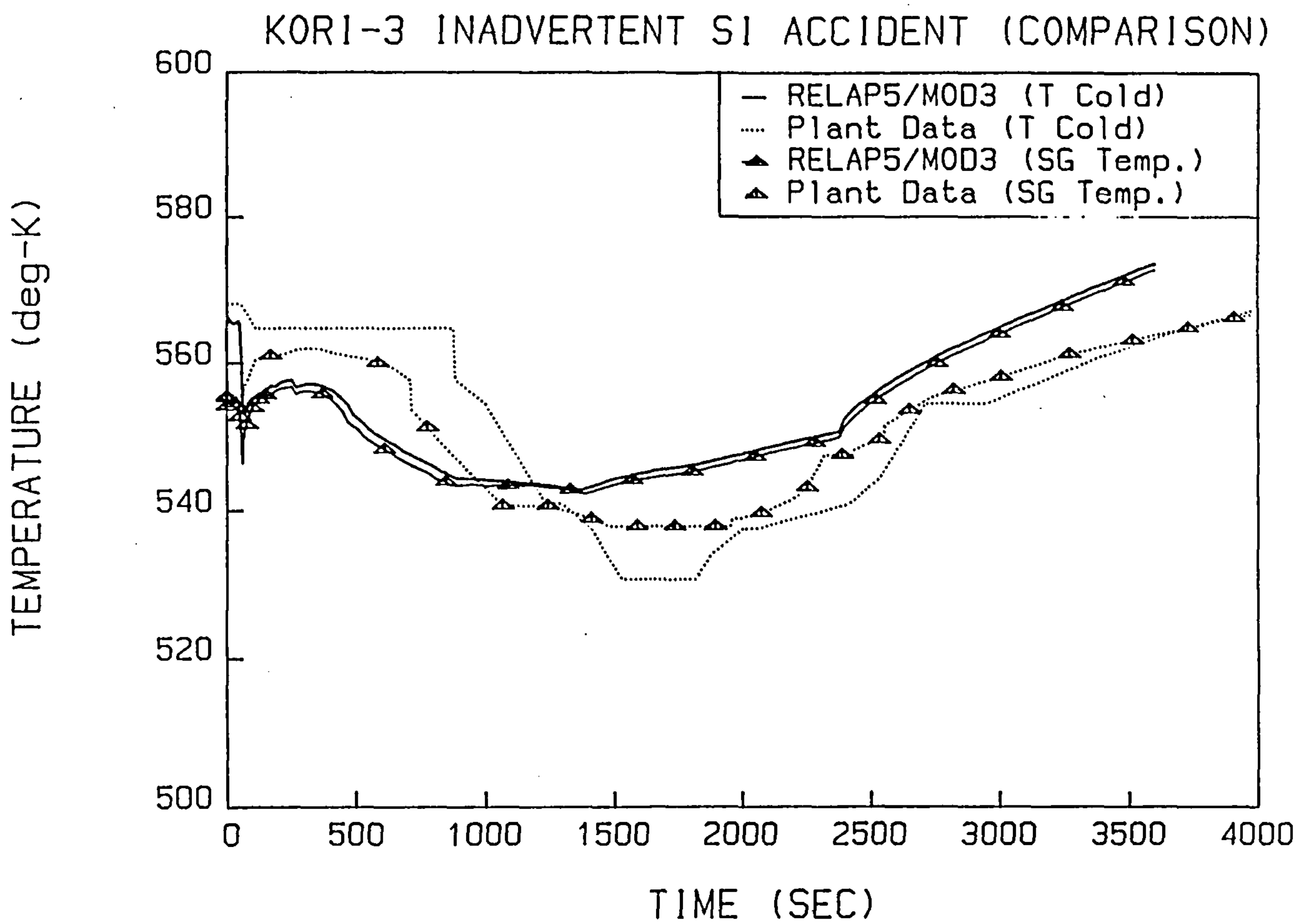

FIG. 14a. Comparison of T Cold (A) and SG-A Temp.

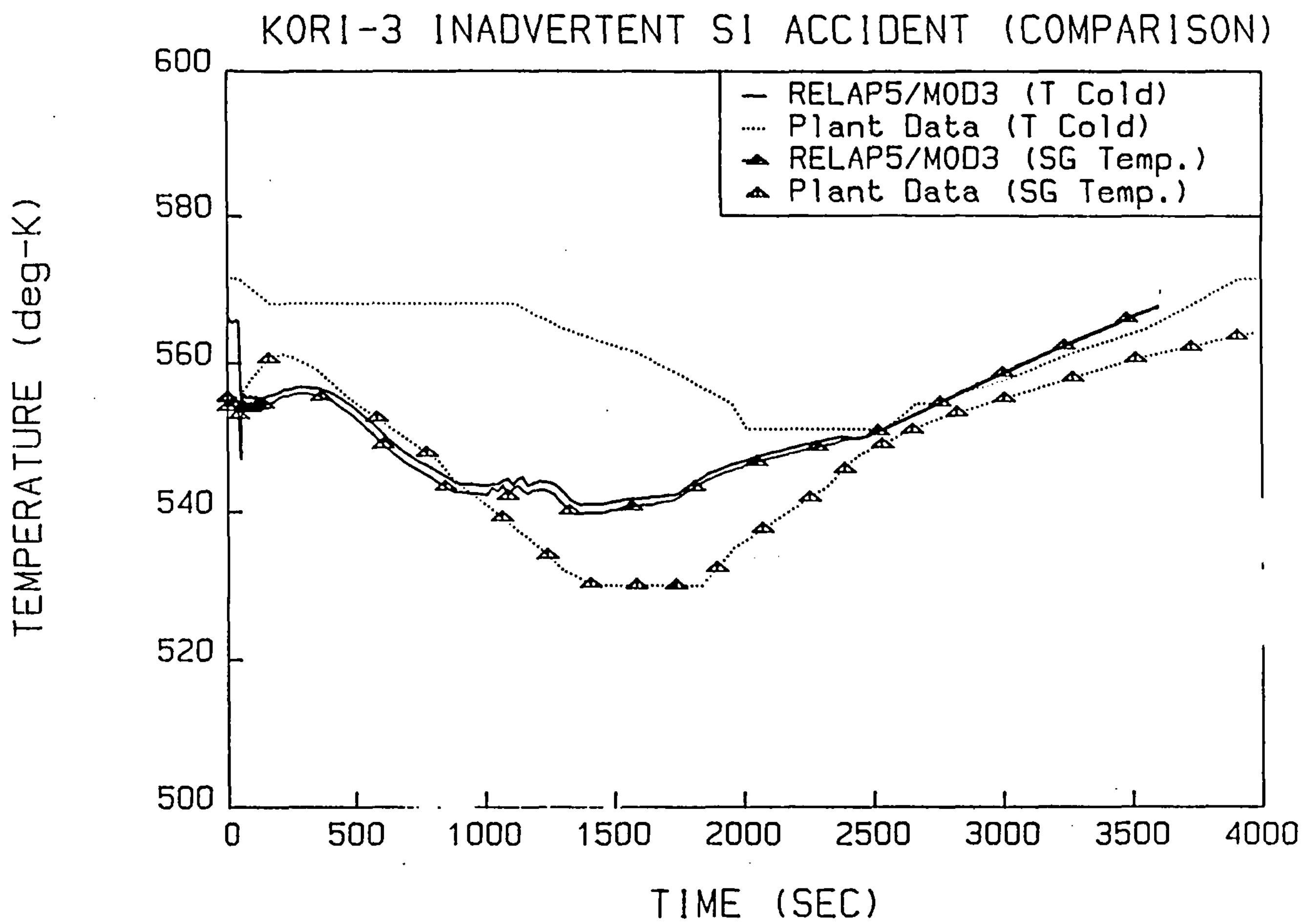

FIG. 14b. Comparison of T Cold (B) and SG-B Temp.

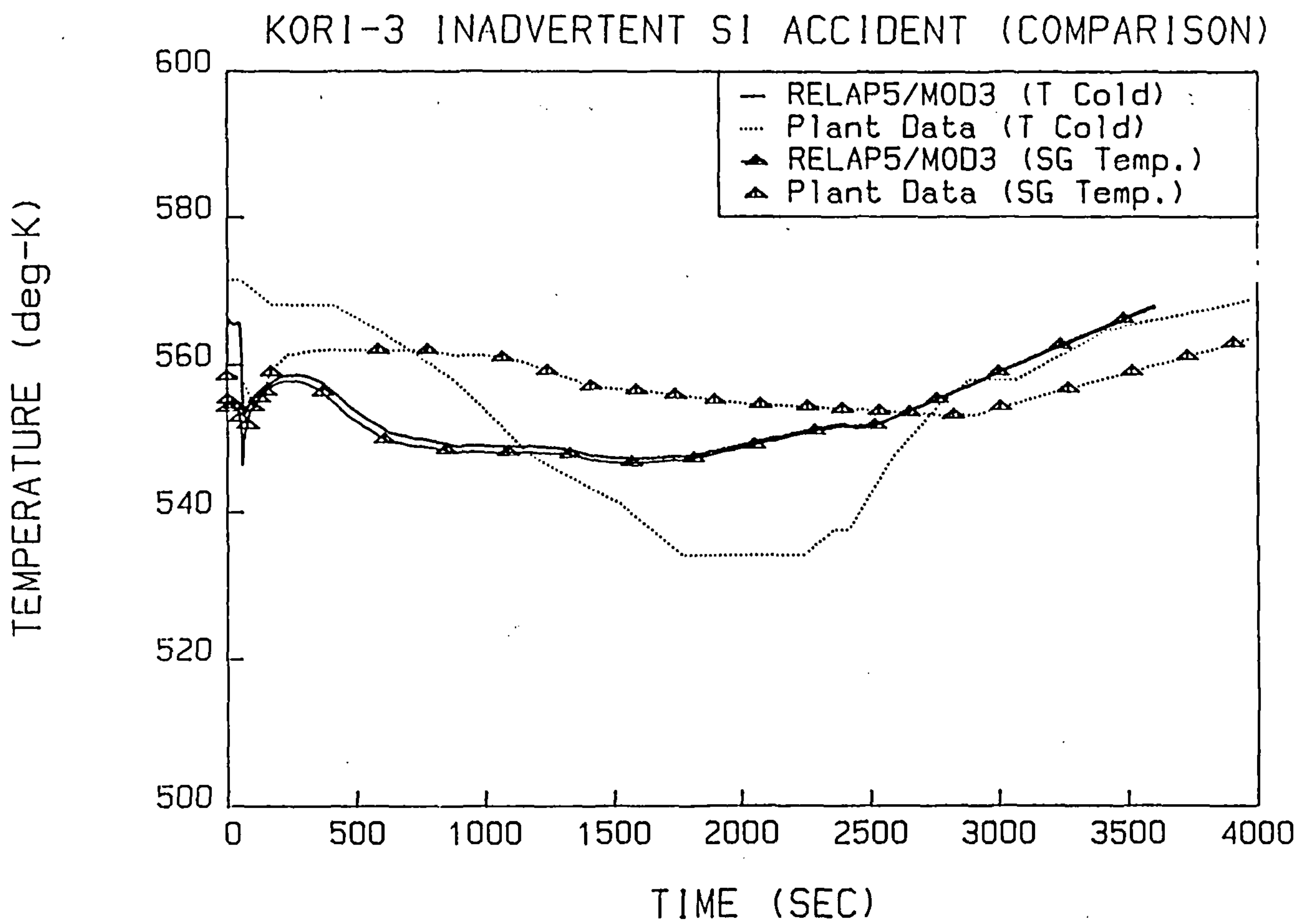

FIG. 14c. Comparison of T Cold (C) and SG-C Temp.

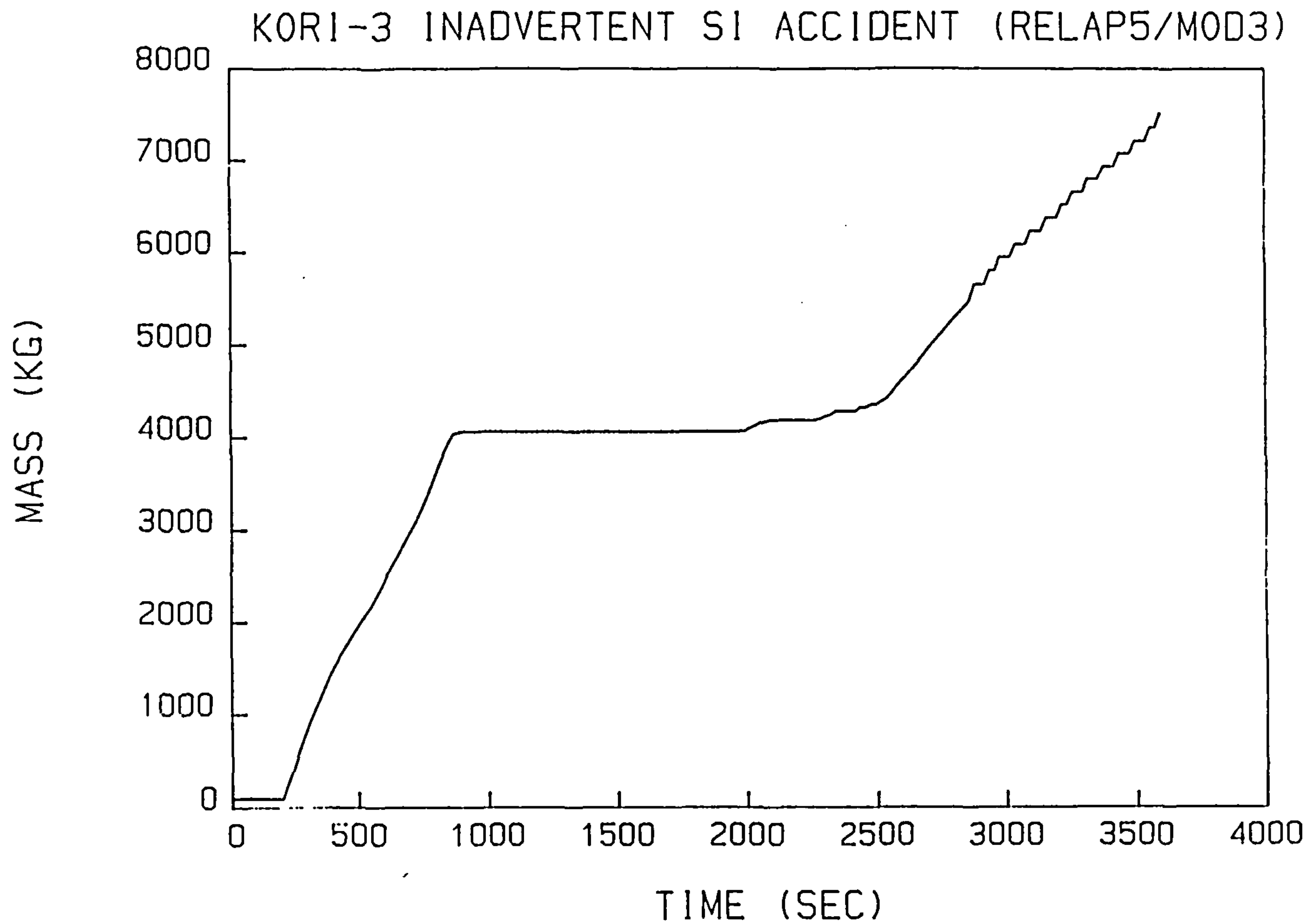

FIG. 15. Integrated Flow of PZR PORV (Calculated)

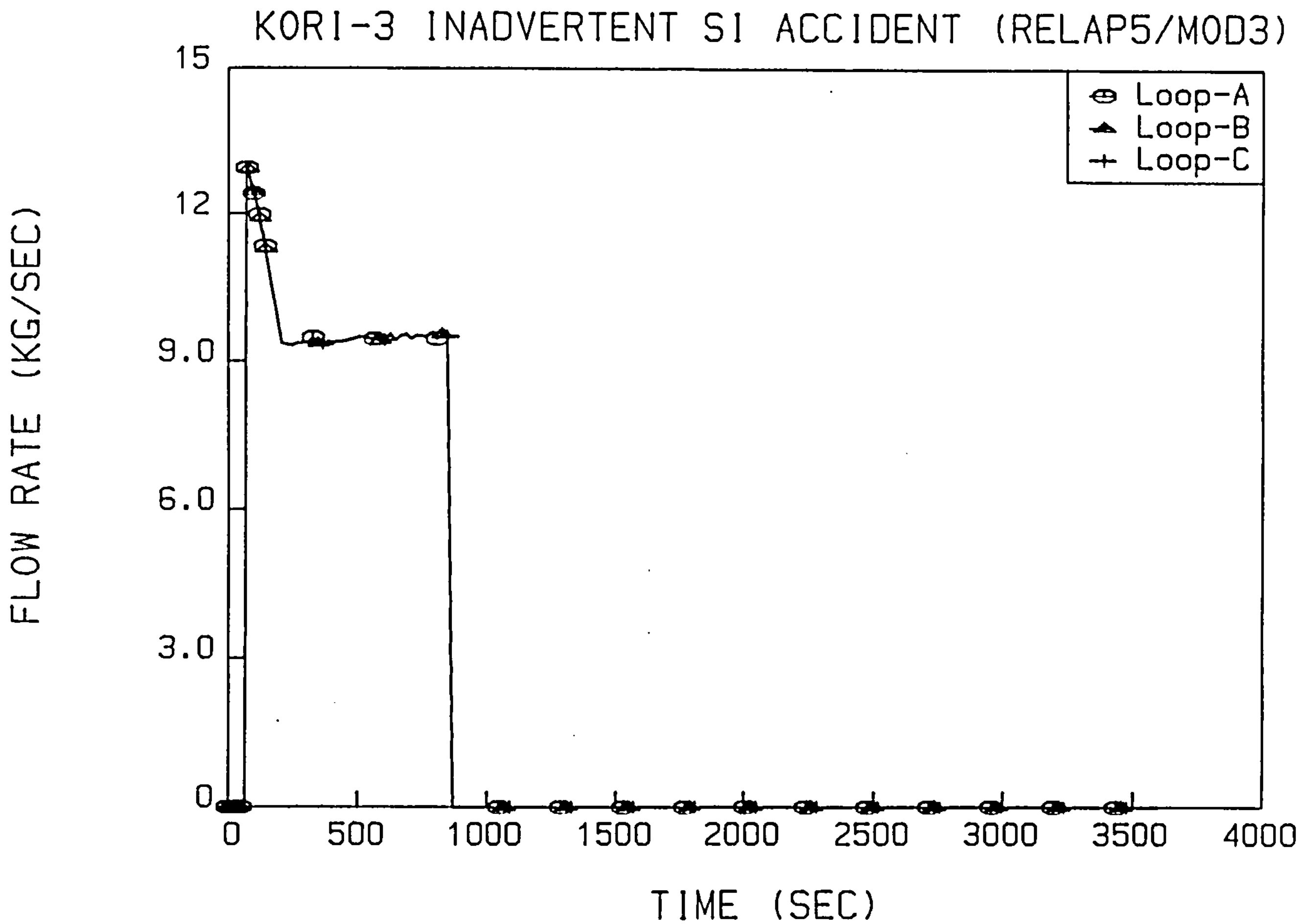

FIG. 16. Safety Injection Flow Rate (Calculated)

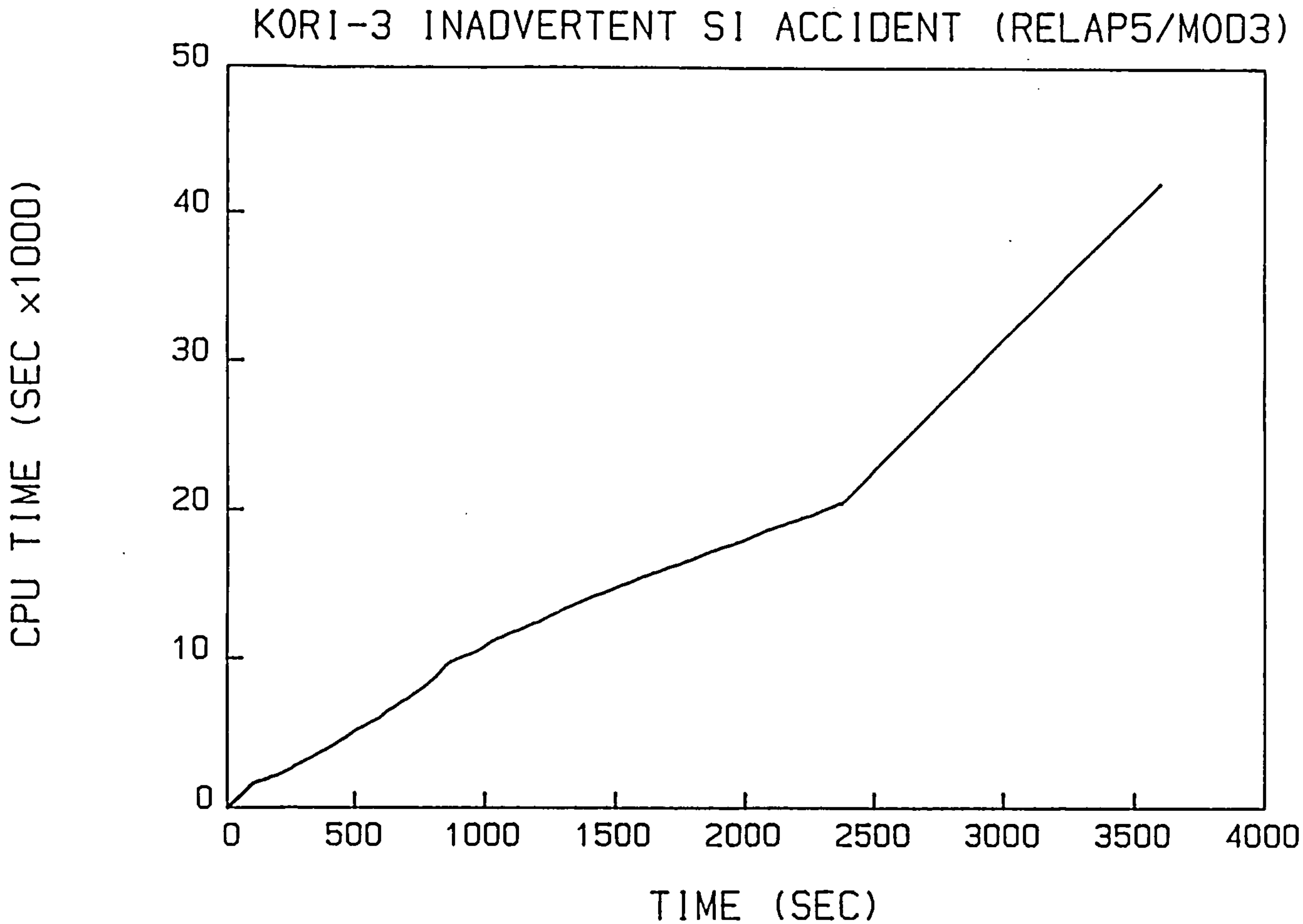

FIG. 17. Total Required CPU Time

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3201, 3202

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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

An inadvertent safety injection incident occurred at Kori Unit 3 in September 6, 1990 was analyzed using the RELAP5/MOD3 code. The event was initiated by a closure of main feedwater control valve of one of three steam generators. High pressure safety injection system was actuated by the low pressure signal of main steam line.

The actual sequence of plant transient with the proper estimations of operator actions was investigated in the present calculation. The asymmetric loop behaviors of the plant was also considered by nodalizing the loops of the plant into three.

The calculational results are compared with the plant transient data. It is shown that the overall plant transient depends strongly on the auxiliary feedwater flowrate controlled by the operator and that the code gives an acceptable prediction of the plant behavior with the proper assumptions of the operator actions. The results also show that the solidification of pressurizer is not occurred and the liquid-vapor mixture does not flow out through pressurizer PORV. The behavior of primary pressure during pressurizer PORV actuation is poorly predicted because the actual behavior of pressurizer PORV could not be modelled in the present simulation.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

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