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Assessment of RELAP5/MOD2 Computer Code Against the Net Load Trip Test Data From Yong–Gwang, Unit 2

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Office of Nuclear Regulatory Research
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Abstract

The results of the RELAP5/MOD2 computer code simulation for the 100 % Net Load Trip Test in Yong-Gwang Unit 2 are analyzed here and compared with the plant operation data. The control systems for the control rod, feedwater, steam generator level, steam dump, pressurizer level and pressure are modeled to be functioned automatically until the power level decreases below 30 % nuclear power. A sensitivity study on control rod worth was carried out and it was found that variable rod worth should be used to achieve good prediction of neutron power. The results obtained from RELAP5/MOD2 simulation agree well with the plant operating data and it can be concluded that this code has the capability in analyzing the transient of this type in a best estimate means.

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1.0 Introduction

The objective of this work was to perform the best estimate transient analysis using the RELAP5/Mod2/36.04 computer code to demonstrate that the Net Load Trip Test, hereinafter named NLTT, of Yong-Gwang Unit 2, performed during its start up test period, can be simulated by this best estimate computer code.

The purpose of 100 % NLTT is to prove the ability of the plant to sustain a net load, which is required as a minimum house load when the transmission line fault occurs, without reactor and turbine trip and to evaluate the interaction between the control system and the system response to the transient [7].

With the plant at 100 steady state conditions and all control systems in automatic mode, NLTT was initiated by manually placing the main transformer high side breaker in the tripped position. During the transient, the major plant operating data were collected by using the on-line computer data logging system and seven strip charts.

The plant geometry and the plant transient are described in chapter 2 and 3. In chapter 4, the plant simulation model and nodalization are presented, while illustrating the initialization methodology. The base calculation results are analyzed in chapter 5 and the analysis of sensitivity study and a run statics are included in chapter 7 and 8 respectively. Conclusions are added in chapter 9 .

2.0 Plant description

This section describes the plant components and control systems which are considered important to understand the Yong-Gwang Unit 2 NLTT simulation. Yong-Gwang Unit 2 ,which is located on the southwestern coast of the republic of Korea ,is a Westinghouse 3 loop PWR rated at 996.8 Mwe and Becthel was the A/E. A schematic representation of the Yong-Gwang Unit 2 is presented in Figure 2.1 [3].

2.1 Primary system

The primary system consists of a pressurized water reactor, reactor coolant system and the associated auxiliary systems. The reactor coolant system is arranged as three reactor coolant loops connected in parallel to the reactor vessel, each containing a reactor coolant pump and a steam generator. As shown in Figure 2.2, an electrically heated pressurizer is connected to the hot leg of one reactor coolant loop. The core is of the three region type and consists of 157 fuel assemblies with 264 fuel rods per assembly. The nominal core power is 2775.5 Mwt. RCCA(Rod Cluster Control Assemblies) are used for reactor rod control and consist of clusters of cylindrical absorber rods. The reactor coolant pumps are Westinghouse vertical, single stage and centrifugal pumps of the shaft seal type. The steam generators are Westinghouse vertical u-tube units which contain inconel tube [5].

The pressurizer pressure control system are making a key role in the primary control system together with the control rod control system. The pressurizer pressure control system whose functional block diagram and setpoints are shown in Figure 2.3 and table 1 maintains pressure at the setpoint value with the help of spray valves (158.95 kgf/cm2), relief valves (164.23 kgf/cm2),safety valves (175 kgf/cm2), proportional heaters and backup heaters [1]. The heaters, spray valve and relief valves maintain the pressure at the setpoint value and prevent reactor trip as a result of pressure variations caused by plant transients.

As shown in Figure 2.4, the automatic control rod control system is designed to maintain a programmed average temperature in the reactor coolant by regulating the reactivity within the core. This control system is designed to control the reactor power automatically in the power range between 15 % and 100 % of rated power for the design transients. Table 2 shows the setpoints of the control rod control system.

2.2 Secondary system

The secondary system conveys steam from the steam generators to the turbine generator system. The system consists of main steam piping, power operated relief valves, safety valves, main steam isolation valves, atmosphere and condenser dump valves. The steam generator level control system and the steam dump control system ,which are shown in Figure 2.5 and 2.7, respectively, are the major control logics in the secondary system.

The setpoints for these systems are shown in table 3 and 4. Each steam generator is equipped with a three element feedwater controller (feedwater flow, steam flow and water level) which maintains a programmed water level (50 % on the secondary side of the steam generator during normal plant operation. This controller continuously compares the measured feedwater flow with the steam flow and a compensated steam generator downcomer water level signal with a water level setpoint to regulate the main feedwater valve opening. The steam dump control system whose schematic representation is shown in Figure 2.6, comprises 16 valves which can bypass steam to the condenser and to the atmosphere. Tavg and steam pressure (Tref) activate the dump system, which is interlocked with plant output to enhance overall control response. If the difference between the required temperature set point of the reactore coolant system and the actual temperature exceeds a predetermined amount (4 c) which is called dead band, a signal will actuate the steam dump to maintain the reactor coolant system temperature within control range until a equilibrium condition is reached. The required number of steam dump valves stroke fully open or modulate, depending upon the magnitude of the temperature error signal resulting from load reduction. The dump valves can be modulated closed after they are full cpen by the reactor coolant average temperature mismatch signal. The steam dump system is comprised of four banks which can bypass steam to condenser and atmosphere , i.e. one single bypass valve with setpoint of 6.48 F , one double bypass valve with 11.44 F, one single dump valve 14.24 F, and one double dump valve with 19.2 F. These valves have a total capacity of 64 % of full load turbine steam flow [1].

2.3 Plant data acquisition

There are three type of digital recording from plant computer, the computer daily logging sheet, pre-post trip review record and sequence record of event. Besides of these digital recordings, there are three analog strip chart recordings of which speed is 2 mm/min. There, of course, exists the uncertainty of the collected data which are due to the reading and instrument error. According to the Ref.[8], the reading and instrument error is 1.5 %. The plant operating data are collected every two seconds, and the simulation data is calculated every one second.

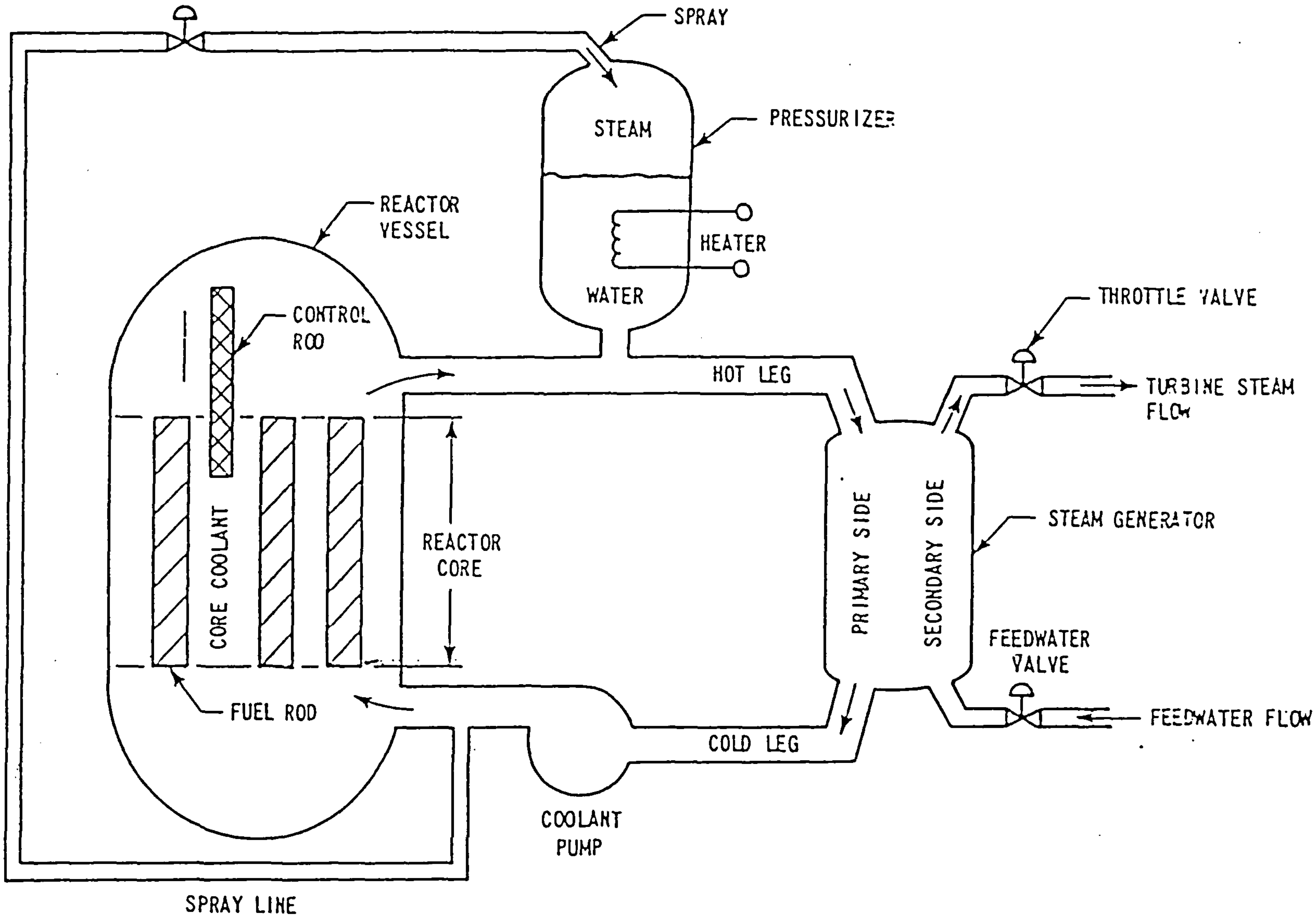


Figure 2-1. Pressurized Water Reactor Schematic Diagram

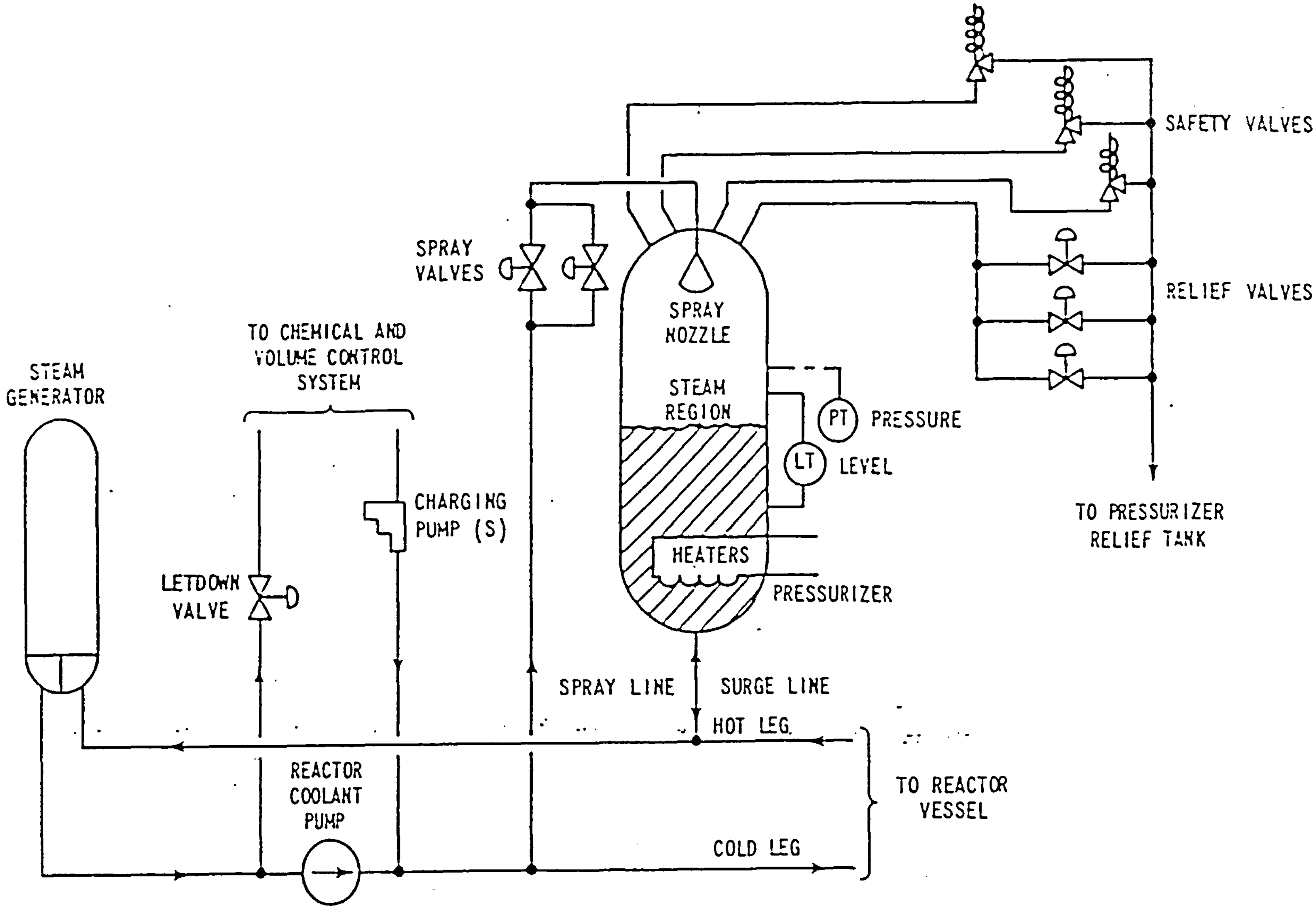


Figure 2-2. Pressurizer Pressure and Level Associated Equipment Arrangement

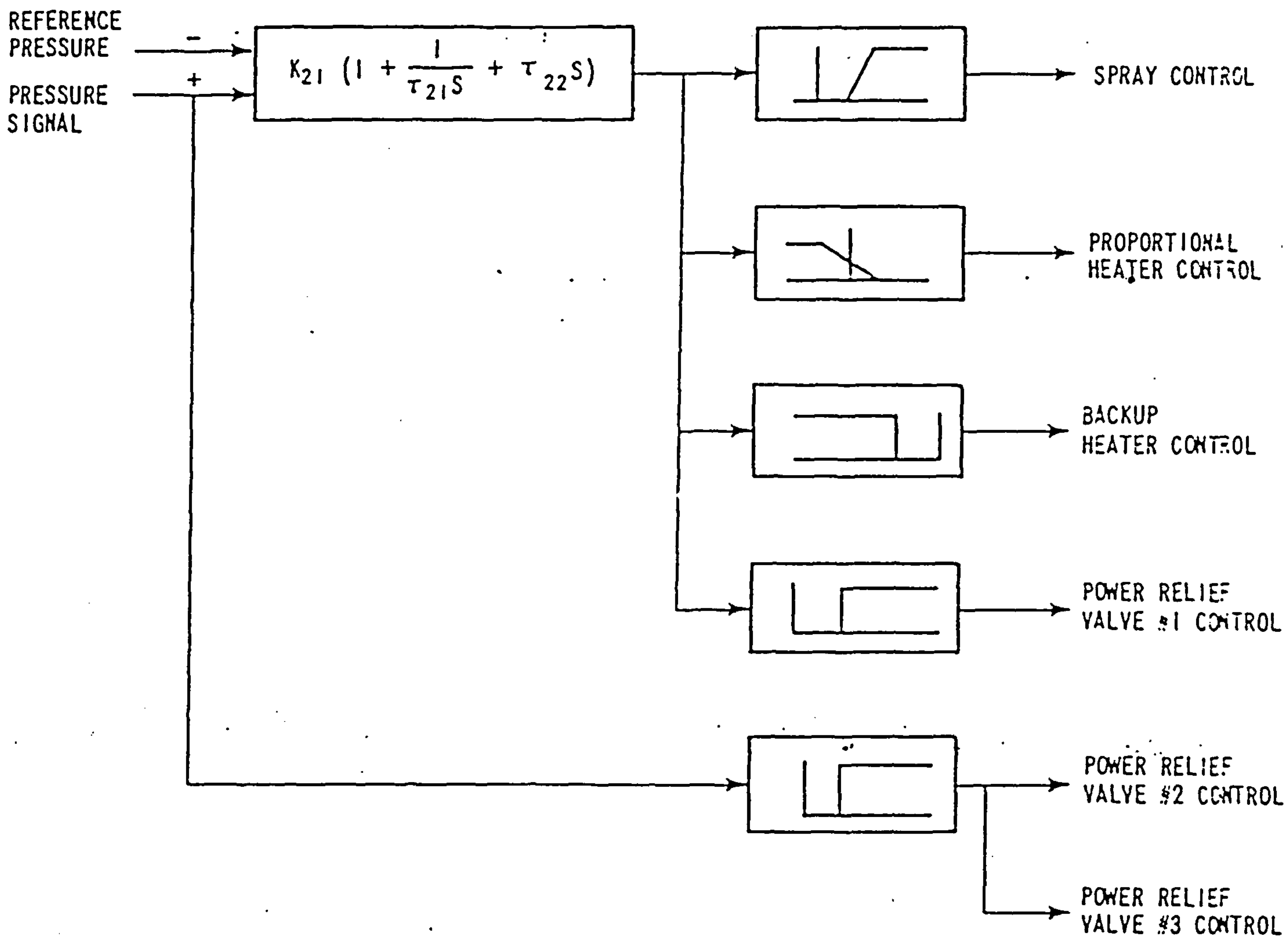


Figure 2-3 . Functional Block Diagram of the Pressurizer Pressure Control System

Table 1. PRESSURIZER PRESSURE CONTROL SYSTEM SETPOINTS

PID Controller	
K_{21}	1 psi/psi
τ_{21}	900 sec
τ_{22}	0 sec
P_{ref}	2,235 psig
Spray controller	
Spray initiation pressure	2,260 psig[a]
Proportional gain	2%/psi
Proportional heater controller	
Proportional gain	-3.33%/psi
Setpoint for full-on proportional heaters	2,220 psig[a]
Backup heater controller	
Backup heaters turned on	2,210 psig[a]
Power relief valve no. 1	2,335 psig[a]
Power relief valve no. 2 and no. 3	2,335 psig

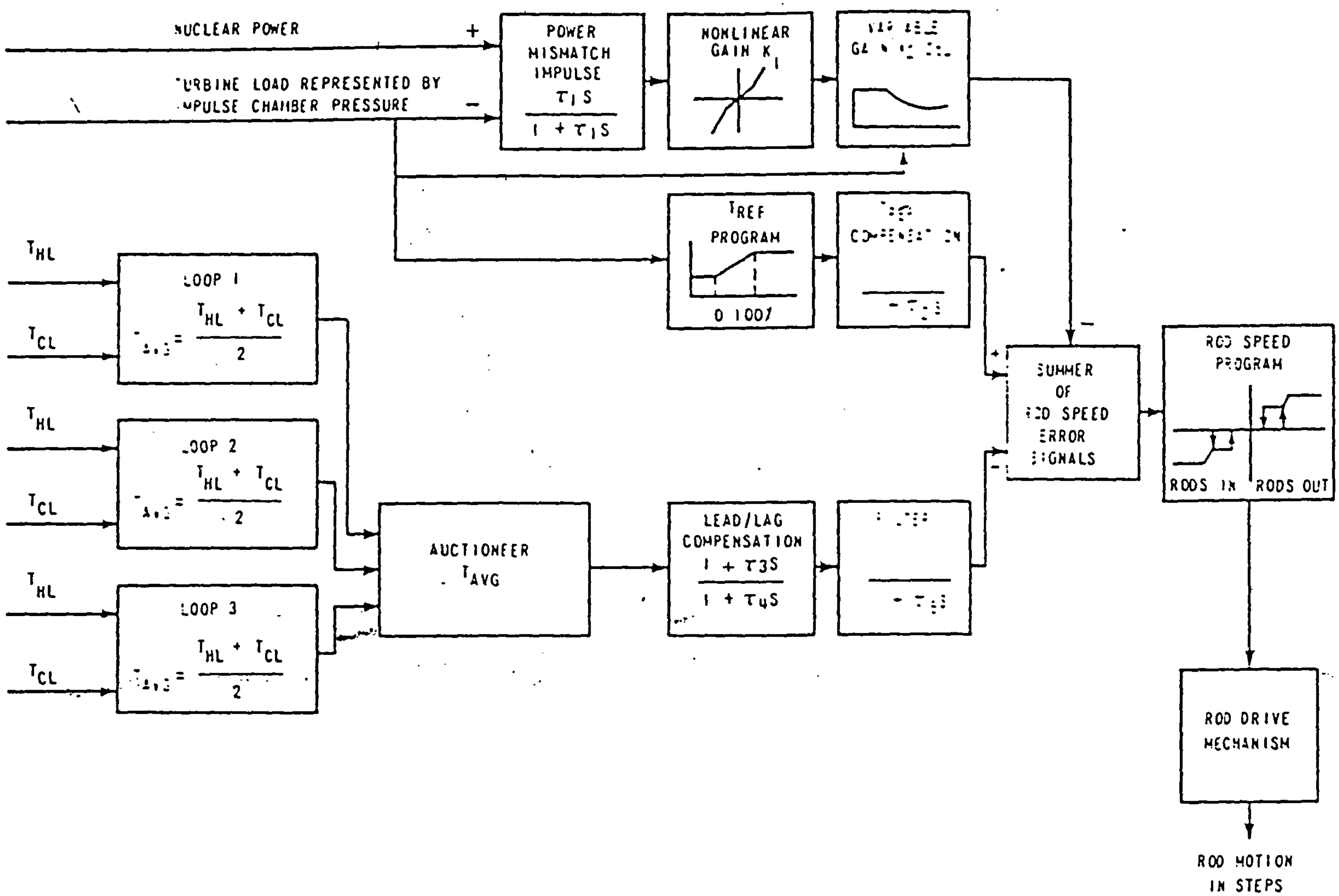


Figure 2.4 Functional Block Diagram of the Rod Control System

Table 2. ROD CONTROL SYSTEM SETPOINTS

Ref gain ($\partial T_{ref}/\partial Q_T$)	For a full load T_{avg} of 588.5	0.315°F/%
τ_3		80 sec
τ_4		10 sec
τ_5		5 sec
No load temperature		557°F
Power mismatch controller		
τ_1		40 sec
Nonlinear gain (K_1):		
error signal at breakpoint of nonlinear gain		± 1%
Low gain		0.3°F/%
High gain		1.5°F/%
Variable gain (K_2) AT 100%		1.0
Down to 50%		Inversely proportional to turbine power
50 to 0%		2.0
Control bank overlap		113 steps
Start B with A		115 steps
Stop A		228 steps
Start C with B		230 steps
Stop B		343 steps
Start D with C		345 steps
Stop C		458 steps

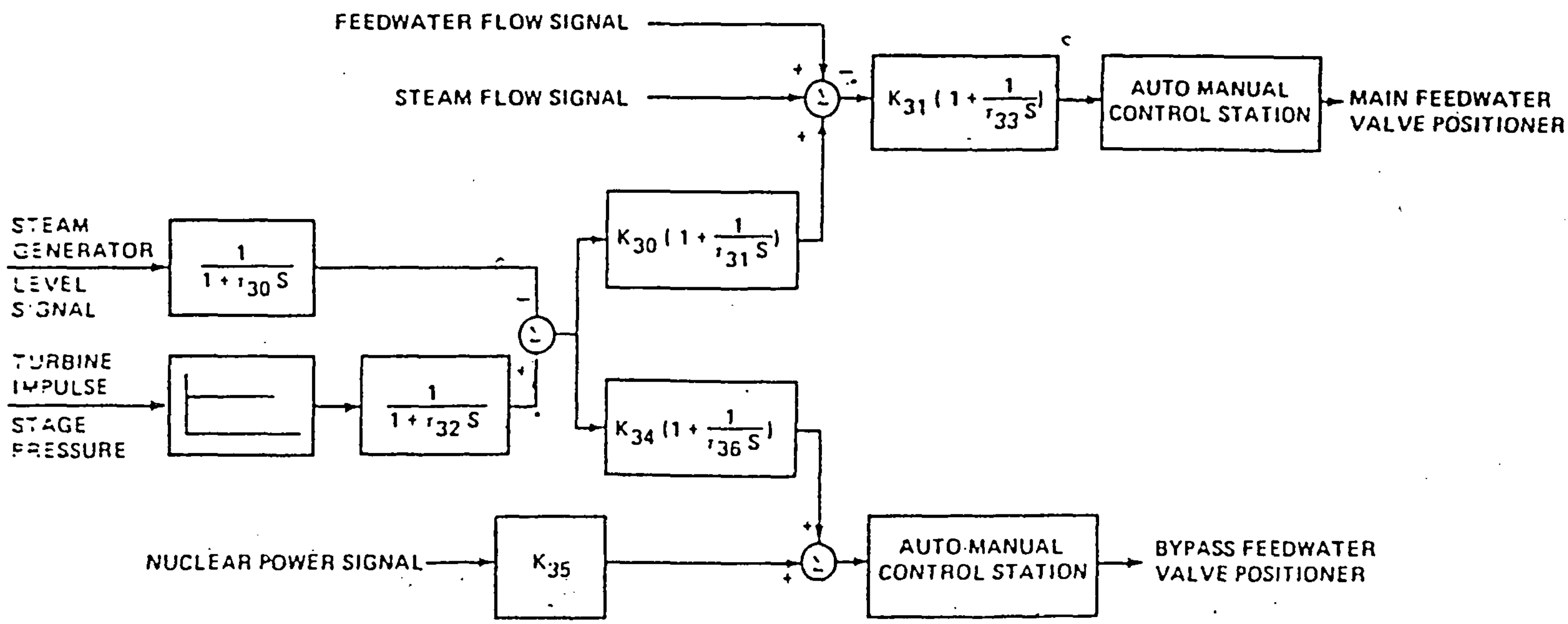


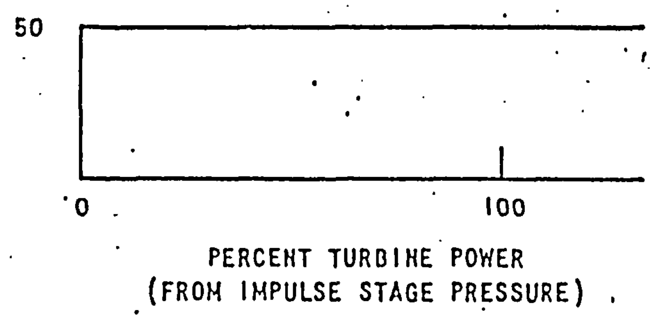
Figure 2-5 Functional Block Diagram of the Steam Generator Level Control System

Table 3. STEAM GENERATOR LEVEL CONTROL SYSTEM SETPOINTS

Level Program

Programmer — 100% Span = 128 inches

Percent of narrow range level span



Level Controller

τ_{30}	5 sec
τ_{31}	200 sec
τ_{32}	5 sec
τ_{33}	200 sec
τ_{36}	750 sec
K_{30}	3.3% full flow/% level span
K_{34}	1% valve lift/% full flow 2% valve lift/% full flow
K_{35}	4% valve lift/% nuclear power

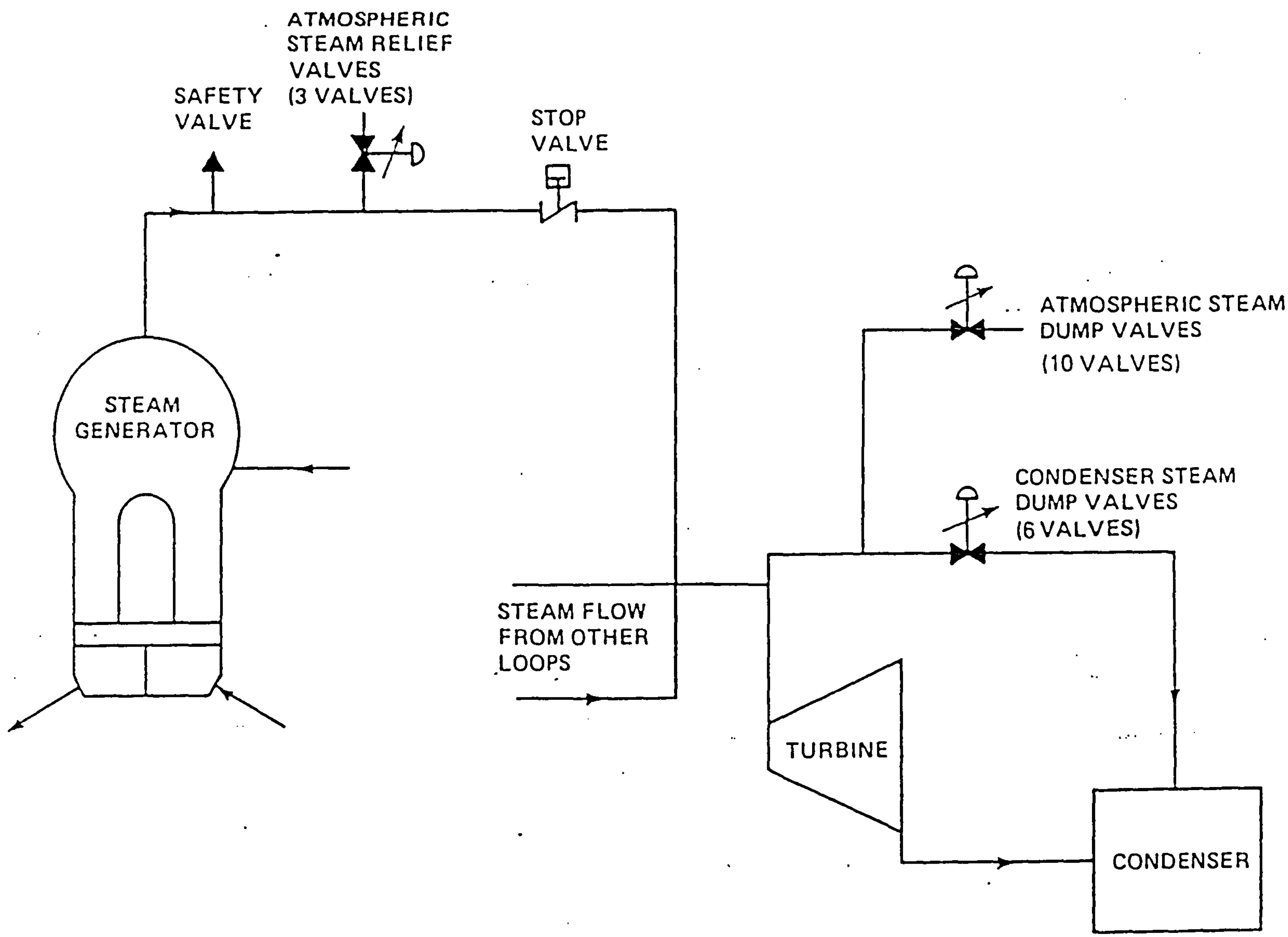


Figure 2-6 Schematic Representation of the Steam Dump Control System

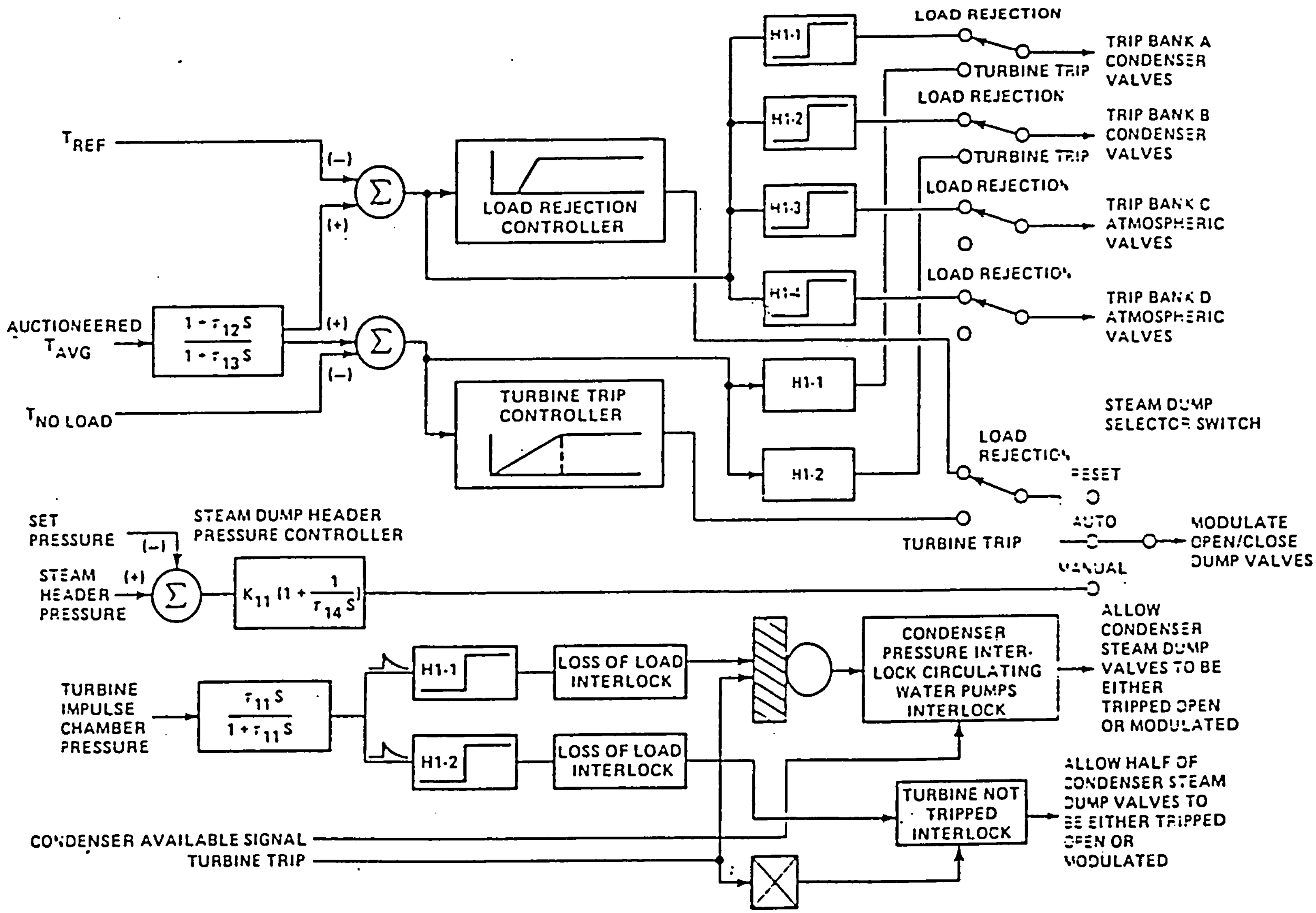


Figure 2.7 Functional Block Diagram of the Steam Dump Control System

Table 4. STEAM DUMP CONTROL SYSTEM SETPOINTS

Load rejection controller	
τ_{11}	120 sec
τ_{12}	10 sec
τ_{13}	5 sec
Deadband	5°F
Proportional gain in percent of total dump capacity per °F[a]	3.8%/°F
Trip open temperatures	9.8°F, 19.6°F, 24.6°F, 31.3°F
Sudden load-loss setpoint (C-7A)	10% of full load
Load-loss greater than 10% (C-7B)	50% of full load
Turbine trip controller	
τ_{12}	10 sec
τ_{13}	5 sec
Dead Band	0°F
Proportional gain in percent of total dump capacity per °F[a]	1.79%/°F
Trip open temperatures	10.4°F, 31.3°F
Header pressure controller	
Set pressure	1,092 psig
Proportional gain in percent of total dump capacity per psi,[a] K_{11}	1%/psi
Reset time constant, τ_{14}	180 sec

3. Plant transient description

With the plant at steady state 100 power conditions and all control systems in automatic mode, NLTT is initiated by manually placing the main transformer high side breaker in the tripped position. Right after the initiation of the transient, the decrease in the S/G steam flow resulting from the large load reduction limits the amount of energy which can be removed from the primary system. The secondary side steam flow rate decrease is not matched by an equivalent drop in the core power level. This results in increasing the primary system pressure and temperature (Tavg). The automatic control rod control system ,whose diagram is shown in Figure 2.4 ,is designed to control the reactor in the power range between 15 % of full power and 100 % of full power, while maintaining a programmed average temperature in the reactor coolant system by regulating the reactivity within the core [2]. The control rod is working according to the two temperature error signals automatically. The error signals are composed of the difference between Tavg and Tref which means the difference between newtron power and turbine power.

As shown in Figure 2.3, the pressurizer pressure and level control system functions to maintain the proper water inventory in the reactor coolant system. Due to the coolant density change during the transient, the pressurizer level and pressure fluctuates as well and the pressurizer functions to maintain the proper water inventory by the help of spray and heaters. On a sudden turbine load reduction, the steam generator level encounters initially a shrink effect due to a collapse of void in the steam generator. Therefore the level error deviation controller calls for an increase in feedwater flow to recover the steam generator level. At the same time, the decrease in the steam flow results in unequal flow at the feed/steam flow deviation controller and this controller calls for a decrease in feedwater flow. Since the level error produced by the shrink is a transitory and short term effect, the output of the level deviation controller is delayed by sensing it through a filtered unit which has 5 second lag time. This allows the flow deviation controller to take the load in controlling the feedwater flow and will drop flow to match the decrease in steam flow.

The excess steam resulting from the load decrease in the turbine is bypassed to the condenser and to the atmosphere via the steam dump system. The steam dump control system ,shown in Figure 2.7, reduces the magnitude of nuclear power plant system response to the transient, following a large turbine load reduction by dumping steam directly to the main condenser and atmosphere. This system has a total capacity of approximately 64 percent of the full load turbine steam flow at full load steam pressure.

This test was conducted as one of the tests which should be done during commissioning in Yong-Gwang Unit 2. The transient initiated by opening the high side breaker , PCB 7400 and PCB 7471 and after 180 seconds the reactor power reached to 50 full power when the plant control was changed to manual operation. After 240 seconds, the reactor power and turbine power reached to 11.5 of full power and 27 Mwe, respectively and after 10 min. synchronized to the grid again and returned to the 100% of power level [7]. This test was performed with satisfying the acceptance criteria specified in Table 5. The result of this test turned out acceptable.

Table 5. Acceptance Criteria for NLTT

- TBN does not reach overspeed trip setpoint
- Safety Injection is not initiated
- RX and TBN must not trip
- No manual intervention should be necessary for control rod, feedwater, steam dump, pressurizer pressure and level until power level decreases below 30 % nuclear power.
- S/G and PZR safety valves should not actuate.

4. Nodalization and Initialization

The used code version was RELAP5/MOD2/CY 36.04 and was processed on a CDC machine cyber 170-875. The input deck was prepared by KEPCO. The nodalization philosophy is based on the guideline of RELAP5 code manual and the detailed data for specific volumes and junctions are based on the design and drawing values for Yong-Gwang Unit 2.

The nodalization can be found in Figure 4.1, which was previously developed for the LOCA analysis [6]. The whole system is devided into 118 volumes, 122 Junctions and 79 heat slabs. The reactor core is described with 6 volumes and core heat slab is modeled by using the RELAP5/MOD2 point kinetics model. The reactor vessel consists of 4 volumes, ie, the downcomer,the lower plenum and core inlet, the upper plenum , the upper head and control rod guide tube. Steam generator is devided into the primary and the secondary system with the boundary of U-tube and the primary system is described with 10 volumes and the secondary with 9 volumes. Two steam generators are lumped in order to match with the RCS loop. Separator is depicted by using the RELAP5 SEPARATOR model. The RCP and the accumulator are also modeled in this simulation. RELAP5 ANS79 model is applied for computing the decay heat and the negative reactivity of temperature and doppler coefficient is considered.

The plant consists of 3 loop coolant system but for the convenience of the nodalization, the reactor coolant system is modeled with two loops, one is the loop which lumps the two loops in which the pressurizer is not connected and the other is the single loop with pressurizer. Each loop is depicted with the RELAP5 pipe component and consider the fluid-metal interaction by using the heat structure. The safety and relief valve nodes of steam generator and pressurizer are not included because it was confirmed by the simple calculation that they did not open during this transient.

The outlets of both steam generators are connected to a single volume, steam header, which is connected to the turbine and the condenser volume. One assumption is needed in order to model this steam dump control system. The atmosphere volume is assumed to have the same pressure as the condenser volume,thus the condenser volume in this nodalization contains the atmosphere volume to which the steam is designed to bypass through the steam dump valve. The steam dump system comprised of 4 banks. So the condenser volume is devided into 4 volumes and connected to the steam header with the survo valve junction. These survo valves behaves realistically in that the valve opening area varies as a function of normalized valve area calculated by steam dump system. For the area change option, the abrupt area change option is selected for valve hydrodynamics in order to calculate the exact steam dump flow rate. Turbine and condenser are modelled as a time dependent volume.

For the steady state initialization, the RELAP5/MOD2 computer code requires the initial conditions which result in zero time deviation for each of balance equation [4]. The initial condition for the RELAP5 simulation should agree with the power plant steady state condition at 100 % full power. The simulated steady state initial conditions along with the plant steady state data for 100 % full power are presented in Table 6, which shows an excellent agreement with the plant data. The steady state condition is obtained in the condition that all primary and secondary control systems are in auto mode.

In the RELAP5/Mod2 simulation, it is very important to get the accurate Tref value because Tref value affects to actuate the control rod and the steam dump system. In order to get the exact Tref value, the steam flow rate at the governor valve is used as the boundary condition because steam flow rate can be considered to be proportional to the turbine power and Tref can be obtained by using turbine power. It is also very important to compute the steam flow at the governor valve exactly in order to calculate the accurate turbine power. Since the curve of valve stem position vs. time or the normalized valve flow area vs. time of the governor valve from the test results is not available, one assumption is made in this analysis. This assumption deals with the steam flow rate calculation at the governor valve. Without knowing the precise behavior of the governor valve, the steam flow rate can not be accurately calculated. Therefore the steam flow rate is intended to be imposed as a boundary condition by applying the time dependent junction at the location of governor valve and can be obtained from Tref value in the plant operating data. The boundary condition is shown in Table 7.

The steam generator level control system is modelled to function automatically with the three element method, ie, feedwater flow, steam flow, and steam generator level. The feedwater valve is working by controlling the feedwater pump speed in the power plant. It is practically difficult to model the feedwater pump speed control and in order to overcome the difficulty encountered, one assumption is made. The feedwater flow rate is determined as a boundary condition by applying the time dependent junction at the feedwater valve and this boundary condition is determined by the normalized feedwater valve area calculated by the feedwater control system. In order to get the exact boundary condition, an approximate method is applied in which the history of valve area is adjusted so that the desired feedwater flow is obtained. This can be done by normalizing the feedwatwer flow rate and the valve flow area response will follow the same curve. At full power, the normalized valve area was 0.88 [5].

In the NLTT of the nuclear power plant, many plants have experience to fail the NLTT due to the OT t trip resulted from the temperature change in the primary system. In the RELAP5 simulation, this OT T is modeled with the limitation in the calculation of the nuclear calibration term for T trips. Since the RELAP5/MOD2 code can not compute the nuclear calibration term, the present modeling contains the OT T trip model without the nuclear calibration term.

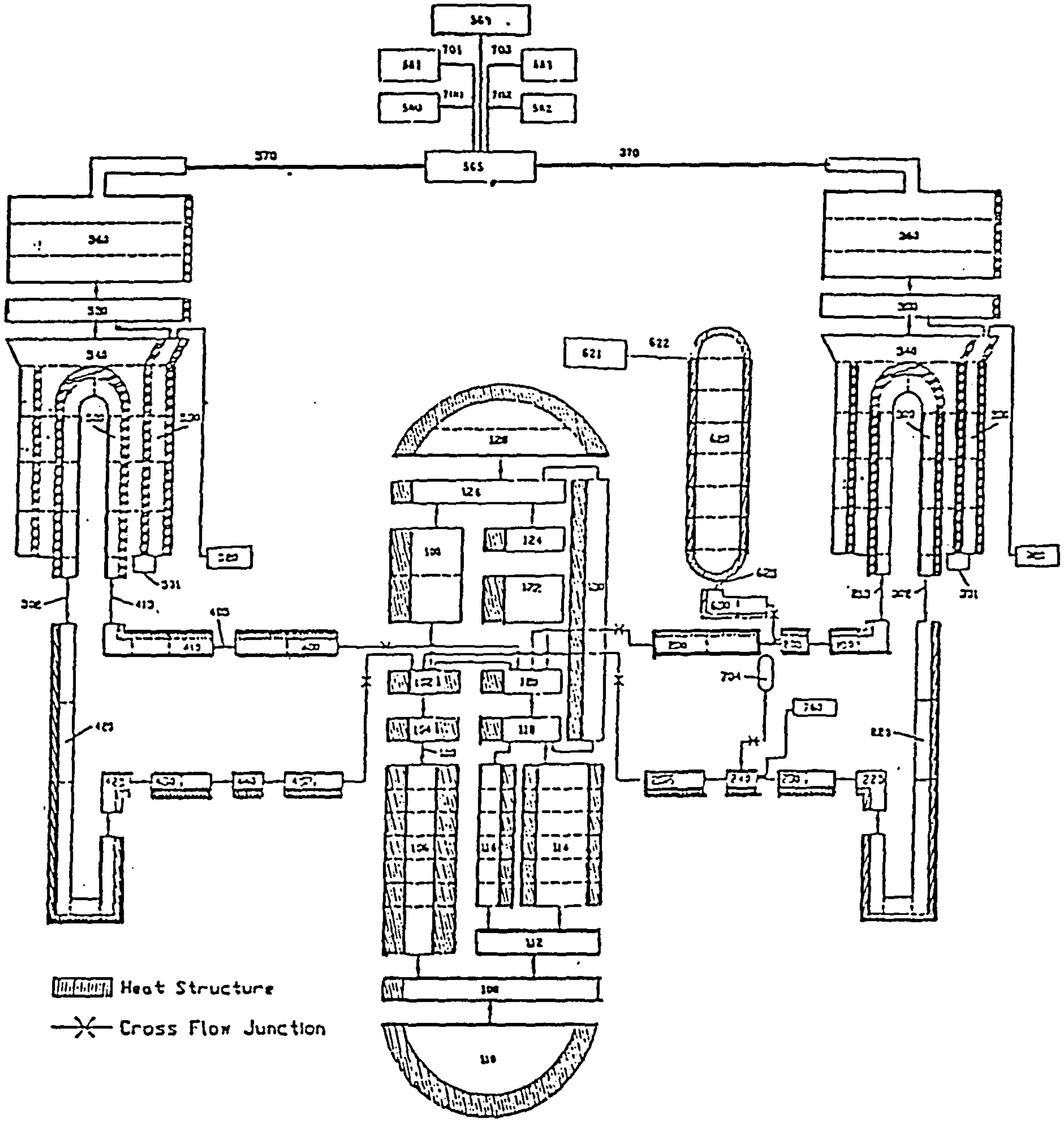


Fig 4.0 Nodalization

Table 2. Plant Initialization Data

	Plant Data	RELAP5/MOD2 Data
1.0 Core		
Inlet Temperature	291.7 (c)	292.62 (c)
Outlet Temperature	326.6 (c)	327.08 (c)
Average Temperature	309.2 (c)	309.85 (c)
Core Power	2775.5 Mwt	2775.5 Mwt
2. Pressurizer		
Level	0.58	0.58
Pressure	157 (kgf/cm2)	157.4 (kgf/cm2)
3.0 Steam Generator		
Pressure	66.8(kgf/cm2)	67.2(kgf/cm2)
Water Level	0.5	0.5
Steam Flow	511.75(kg/sec)	511.55 (kg/sec)
Feed Water Flow	513.88(kg/sec)	512.26 (kg/sec)
Feed Water Temperature	226.5 (c)	226.7 (c)

Table 3. Boundary Condition for Steam Flow at Governor Valve

Time (sec)	Steam Flow (lb/sec)
0.0	3381.0
1.0	2028.6
2.0	1196.7
3.0	983.87
4.0	770.87
5.0	561.24
10.0	135.24
200.0	135.24

5.0 Calculation results and discussions

5.1 Primary system

The core power is controlled by the control rod with the two error signals. Right after the initiation of transient, the control rod begin inserting into the core in order to decrease Tavg and the neutron power. In the RELAP5/MOD2 simulation, the greatest effect of the control rod control system is shown between 20 seconds and 80 seconds. The control rod decreases the reactor power with the rate of 0.33 % of full power per second(Fig 5.1). The reactor power begins to increase again up to 79% of full power for 20 seconds due to the effect of the negative moderator temperature coefficient which appears near 80 seconds. After 100 seconds, the core power continues decreasing toward 30% of full power at which the power plant control mode was changed to the manual control mode. The actual power plant responds differently. It continues to decrease down to 50 % of full power with a reduction rate of 0.28 % of full power per second. The effect of MTC(moderator temperature coefficient) can be found around 100 seconds for 30 seconds. This later effect of MTC in the power plant operating data causes a little lower value after 85 seconds than the simulation data. The main reason of this later effect of MTC may result from the lack of the accurate reactivity information in the core during the test and more details will be added in the sensitivity study. However it can be concluded that the core power calculated by the RELAP5/Mod2 computer code agrees well with the plant operating data.

Following the transient, Tavg increases because of the reduced heat removal capacity of the secondary system. With the help of the sliding Tavg program, reactor power is adjusted to maintain a programmed decreasing Tavg as the turbine load is decreased. Decreasing the turbine load makes Tavg and steam pressure increase. The control rod control system inserted the control rod into the core in order to decrease Tavg equal to the programmed Tavg. The simulated Tavg value in Fig 5.2 shows two peaks of temperature around 10 second and 40 second, respectively. These two peaks can also be found in the pressurizer liquid level, the pressurizer pressure and the spray flow which are shown in Figure 5.4, 5.5, and 5.6, respectively even though these peaks are very small. The timing of these two peak values approximately agreed well with the pressurizer liquid level and pressure but a little latter around 7 seconds than the timing of the spray flow rate. As shown in Fig. 5.2, The simulated Tavg value follows the plant operating data quite accurately up to about 80 second but after that time, it indicates a little higher value which shows the similar trend to core power behavior. The difference between the outlet and inlet temperature in the RELAP5/MOD2 simulation (Figure 5.3) reduecd from 34.46 c to 22.9 c for 180 seconds, in which overall agreement was satisfactory. Right after the initiation of the transient, The cold leg temperature increases very fast due to the sudden reduction of heat removal of the secondary system ,meanwhile the hot leg temperature is constant and this makes the temperature difference between outlet and inlet temperature drop very fast, this reduction continues until 6 second. After 6 second, the hot leg temperature increases due to the constant core power generation until 10 second. After that, it continues to decrease.

The pressurizer level control system functions to maintain the proper water inventory in the reactor coolant system. This level is maintained by controlling the balance between water leaving the system, via the letdown flow path to the CVCS, and water entering the system from charging pump. Since letdown is a fixed amount, the balance is maintained by varing the changing flow. Like this kind of simulation, it is judged that the changing charging flow does not affect much to control the pressurizer level ,so this is modelled as a constant flow. The simulated pressurizer level shows a little lower trend than the plant operating data until 80 second and a little higher after that time. Even though the plant operating data shows an oscillation , the trend of the simulated pressurizer level is similar to that of the plant operating data. The pressurizer pressure is controlled by using the heaters and spray. The decrease of the secondary heat removal induces the Tavg increase and RCS volume increases with yielding the large volume water enters into the pressurizer. During this transient, the pressurizer is insurged.

After insurging, vapor bubble in the pressurizer is compressed and pressure initially increased. This makes the pressurizer spray actuated to reduce pressurizer pressure by condensing steam from the spray. The initial peak pressure in the simulation is shown around 5 seconds which is similar to the plant operating data. The inherent pressure control due to the subcooled water makes the PZR pressure decreases, after that another peak pressure can be obserbed due to the reduction of heat removal of secondary system at 40 second. After peak, it continues to decrease. Generally, the pressurizer pressure trend in the RELAP/Mod2 simulation agrees well with that of plant operating data except a little higher value after 80 second.

5.2 Secondary system

The secondary system cosists of two loops like the reactor coolant system. The calculation results for both loops shows the exactly same trend for the level and for the flow rates, the lumped loop shows double flow rate. For the comparison with the plant operating data, single loop data was used.

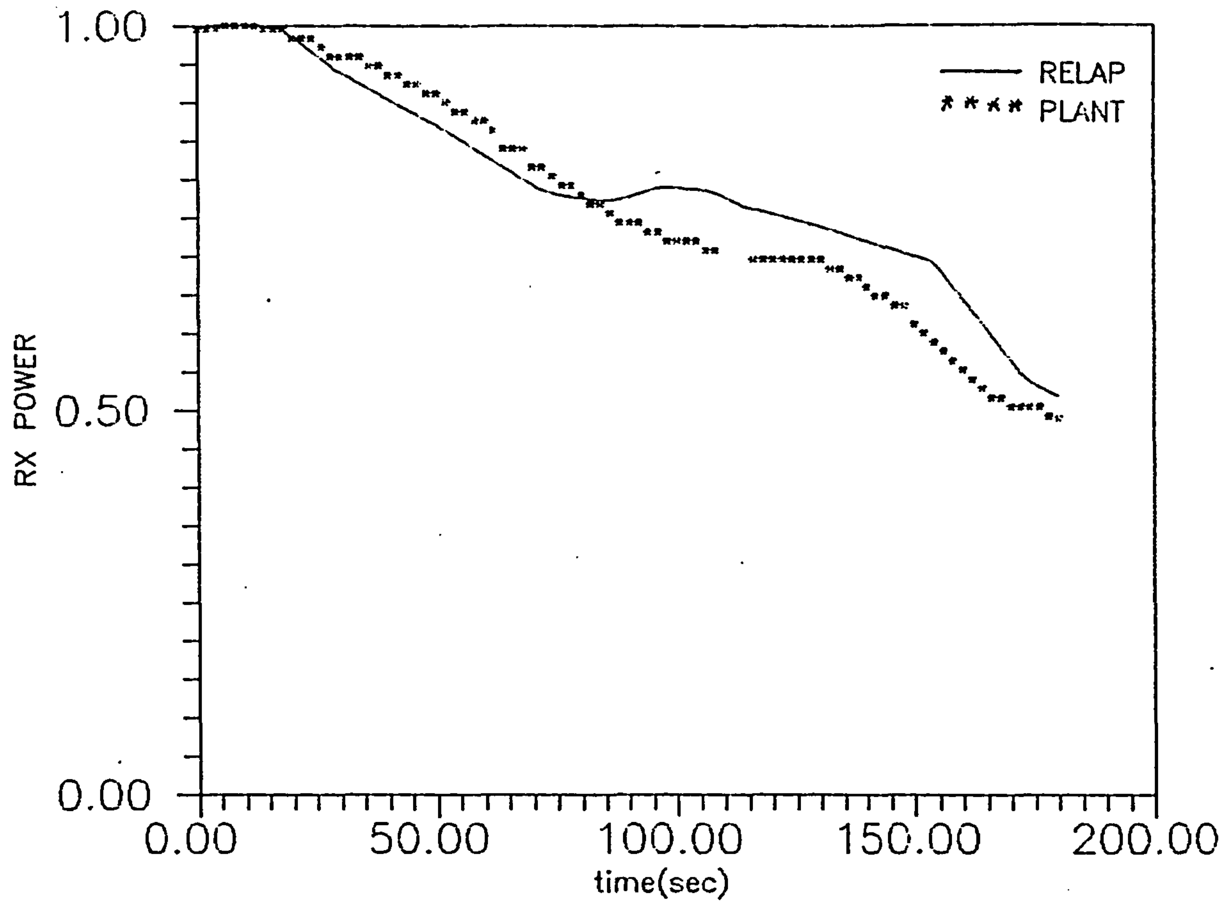
Each steam generator is equipped with three element feedwater controllers (feedwater flow, steam flow and steam generator water level). This controller continuously compares the measured feedwater flow with the steam flow and also compares a compensated signal of water level in the steam generator downcommer with a setpoint of water level to regulate the main feedwater valve opening. The RELAP5/MOD2 simulation indicates that the steam generator level drops to 46% of span level at 4 second (Figure 5.8). This first level drop caused from the fact that on a sudden turbine load reduction, the steam generator level encounters initially a shrink effect due to a collapse of void in the steam generator but because the response of the feed steam flow deviation controller functions much faster than the response from the actual level/programmed level, the level deviation controller calls for an increase in feedwater flow. So the steam generator level begins to increase in order to be equal to the programmed level of 50 % of span level. It recovers up to 49 % level of span about 10 second, and continues the constant level until 110 second, then it begins to decrease. The plant operating data shows a different trend. It drops down to 32% level of span, recovers around 10 second, continue to keep a constant level until 70 second. When it agrees well with the simulation data after 70 second it begins to show oscillation. The general steam generator level trend by RELAP5/MOD2 simulation shows a good agreement with that by the plant operating data except some oscillation after 70 second.

The steam generator pressure increases due to the turbine valve closing and reaches to the first peak pressure at 3 second, then it begins to decrease due to the steam dump valve opening (Figure 5.7). Around 8 second, It begins to increase again because the effect of turbine valve closing is higher than that of steam dump valve opening, and it increase gradually in order to reach the second peak pressure about 30 second. The plant operating data reveals a little higher response (2kgf/cm2) until 80 second. With showing the similar general trend to the simulation pressure, except the oscillation over the range of 6 kgf/cm2 with the trend of gradual increase. But it shows similar two peak pressures which can be explained the same way as above.

The three element valve controller functions to maintain the necessary main feedwater valve position so that sufficient feedwater flows into the steam generator to maintain the level at the programmed value. The three signals that determines the valve position are the level error signal , the steam flow rate signal, and the feedwater flow rate signal. The result of RELAP5/Mod2 simulation (Figure 5.9) shows that the feedwater drops to 450 kg/sec as soon as the transient is initiated (3 sec) due to the effect of the sudden steam flow reduction. The plant operating data shows that the feedwater flow drops to 370 kg/sec at 2 second. This can be explained as follows. On the load reduction, the steam generator level encounters initially a shrink effect due to a collapse of void in the steam generator and the level deviation calls for an increase in feedwater flow. At the same time, the decrease in steam flow developes unequal flow at the feed/steam flow deviation controller and this controller calls for a decrease in feedwater flow. Since the level error produced by the shrink is a transitory , short term effect, the output of level deviation is delayed by sensing it through a filter unit with the 5 second lag time. This allows the flow deviation controller to take the load in controlling the feedwater flow and will drop flow to match the decrease in steam flow. But soon the effect of steam generator level deviation induces the feedwater flow to increase. The simulated feedwater flow shows the peak flow rate of 510 kg/sec but the plant operating data shows 520 kg/sec at 10 sec and 12 sec, respectively. Even though the plant operating data shows a big oscillation with the decrease trend but the simulated feedwater flow shows the continuous decrease after peak flow.

The simulated steam flow (Figure 5.10) shows a sharp reduction down to 340 kg/sec due to a TBN valve closing, meanwhile the plant operating data indicates the same sharp reduction with the value of 250 kg/sec. Following this sharp reduction, it begins to increase with the help of steam dump valve opening. After the peak flow rate, the simulated steam flow decrease gradually with the less sharp slope than the plant operating data. Sharp change of feedwater flow is not seen when compared with steam flow due to the fact that the feedwater controller senses both the error and the accumulation of error. This integration factor is the main reason why the feedwater is less sharp than the steam flow. The plant operating data for the steam generator level, steam flow and feedwater flow show the oscillation but the simulated data reveals that it has some limitation to simulate in details enough to describe the oscillation. In spite of this limitation, it can be concluded that the result of simulation agrees well with the plant operating data.

On beginning the transient, the steam dump valves trip open immediately and remain full open for approximately 35 seconds in the simulation data. After that ,it begins to decrease continuously. The plant operating data is not available, so it is impossible to compare these data with the plant operating data. The Figure 5.11 shows the steam dump flow rate calculated by the RELAP5/MOD2 code. the results of calculation in the primary and secondary system illustrates that the dump system works very accurately because the steam dump flow rate determines the thermal-hydraulic parameters of primary and secondary system.



Fiqure 5.1 REACTOR POWER

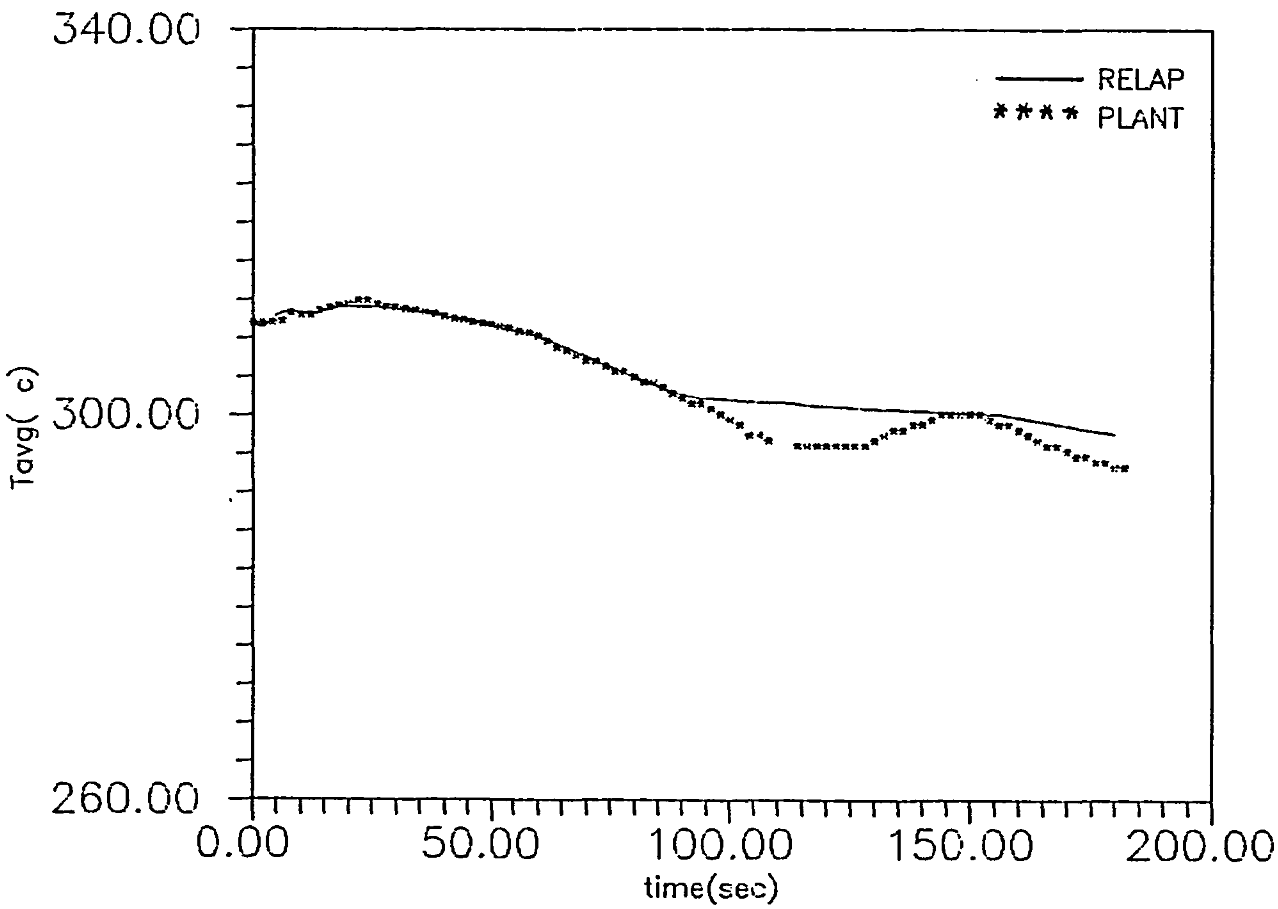


Figure 5.2 Tavg

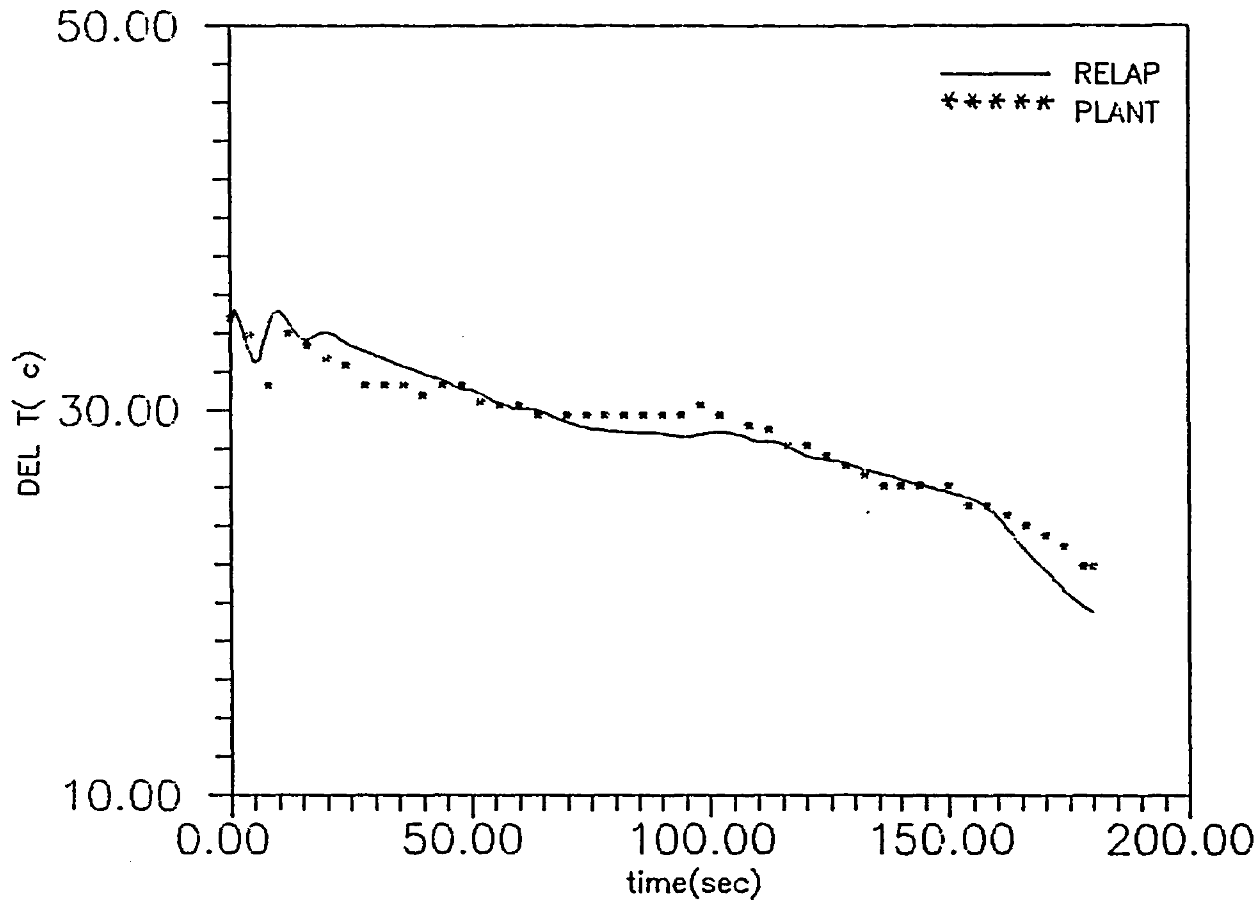


Figure 5.3 DEL T

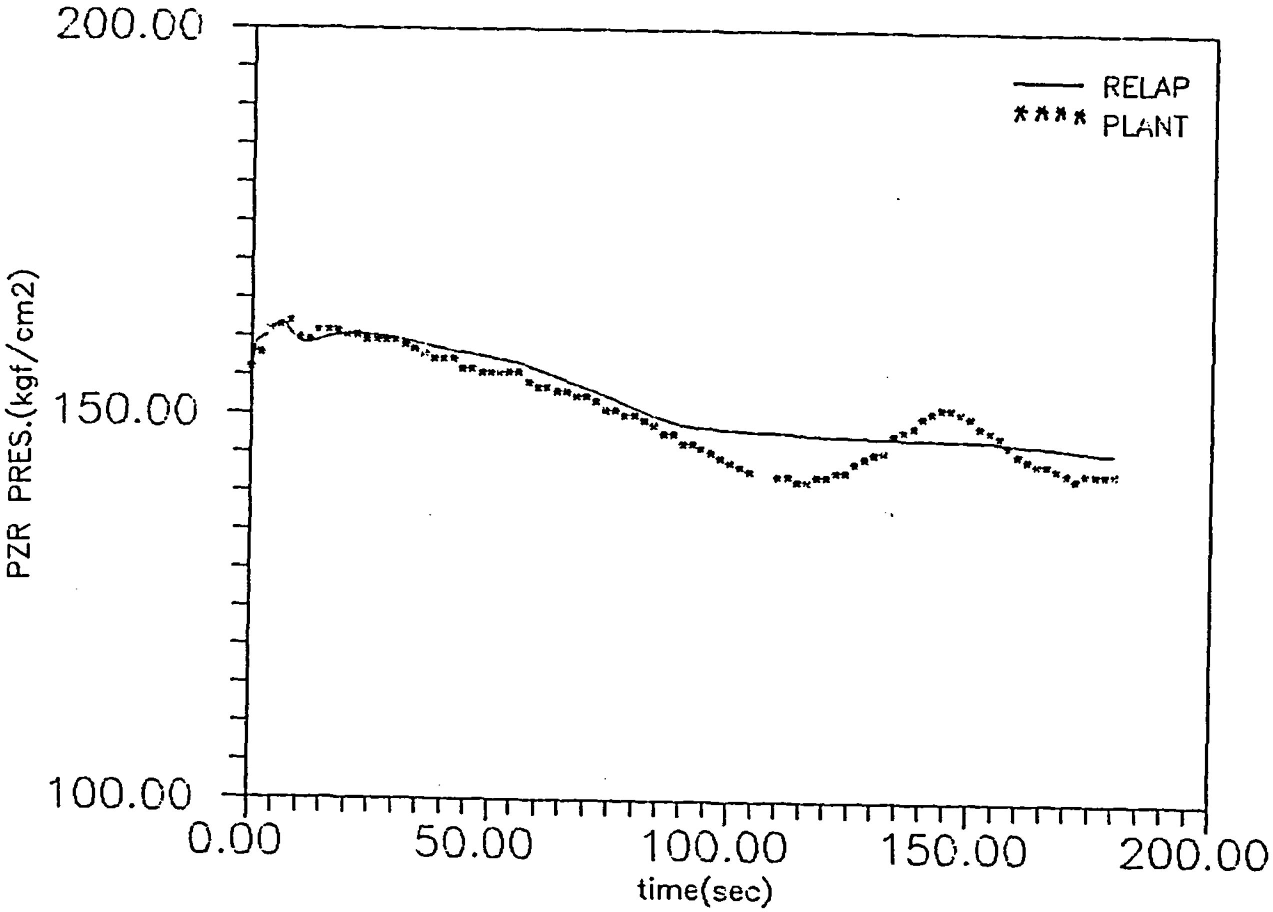


Figure 5.4 PRESSURIZER PRESSURE

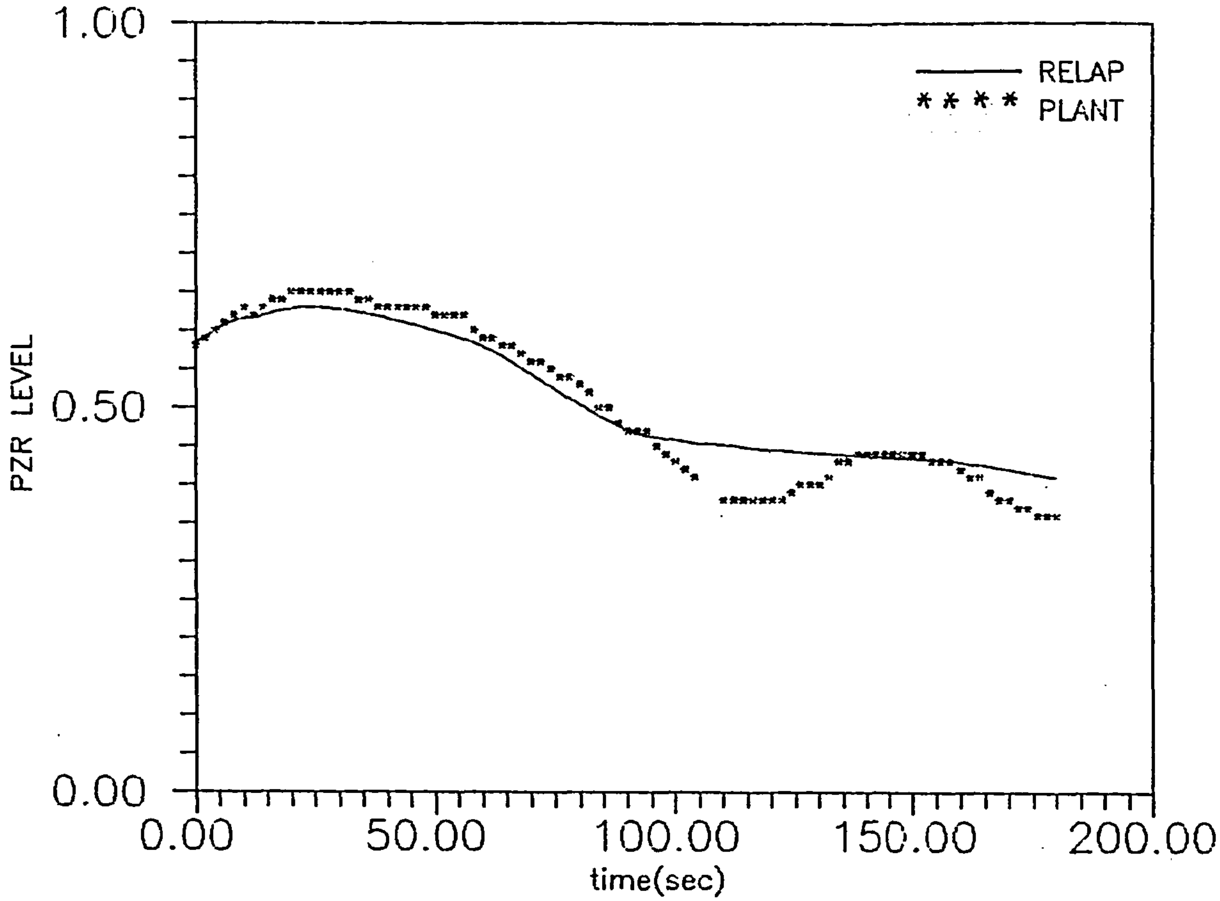


Figure 5.5 PZR LEVEL

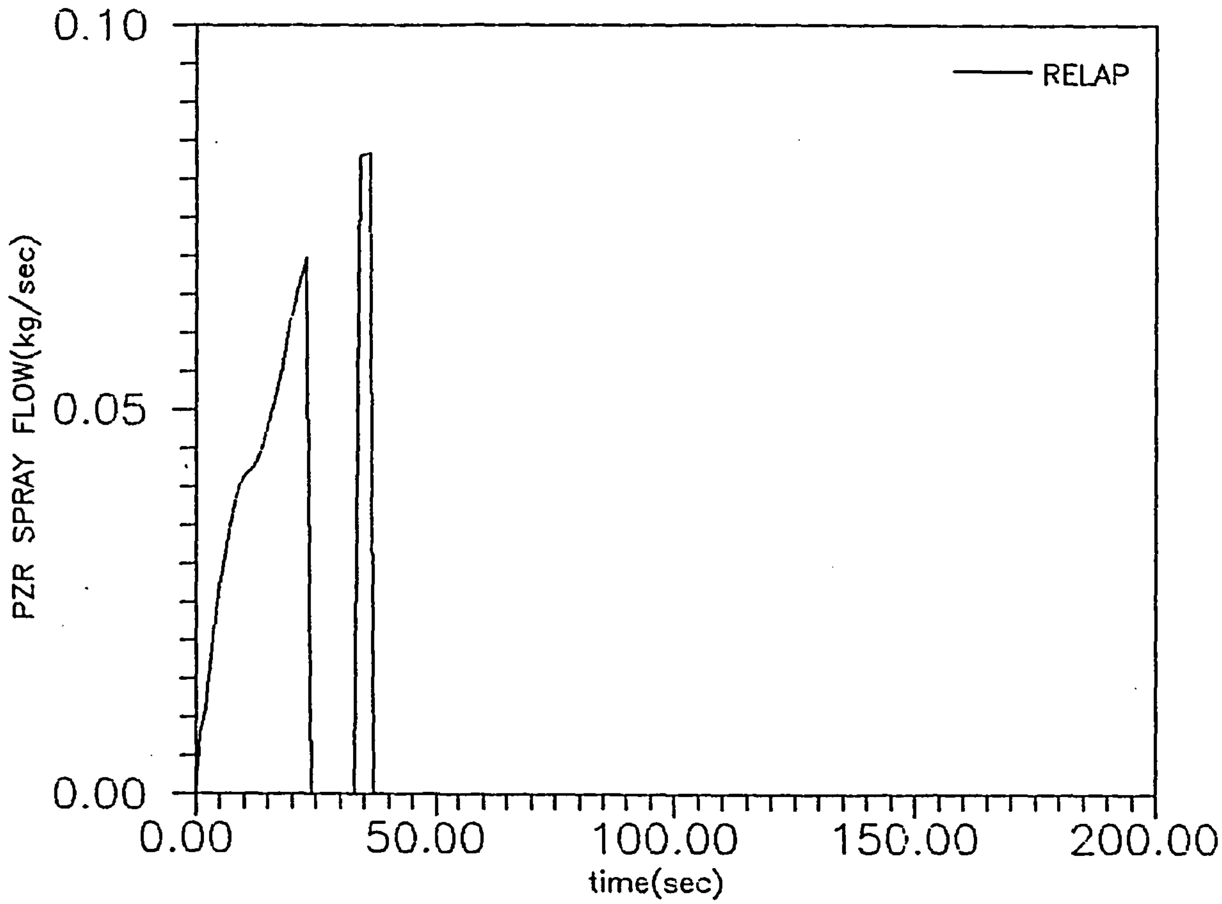


Figure 5.6 PRESSURIZER SPRAY FLOW (kg/sec)

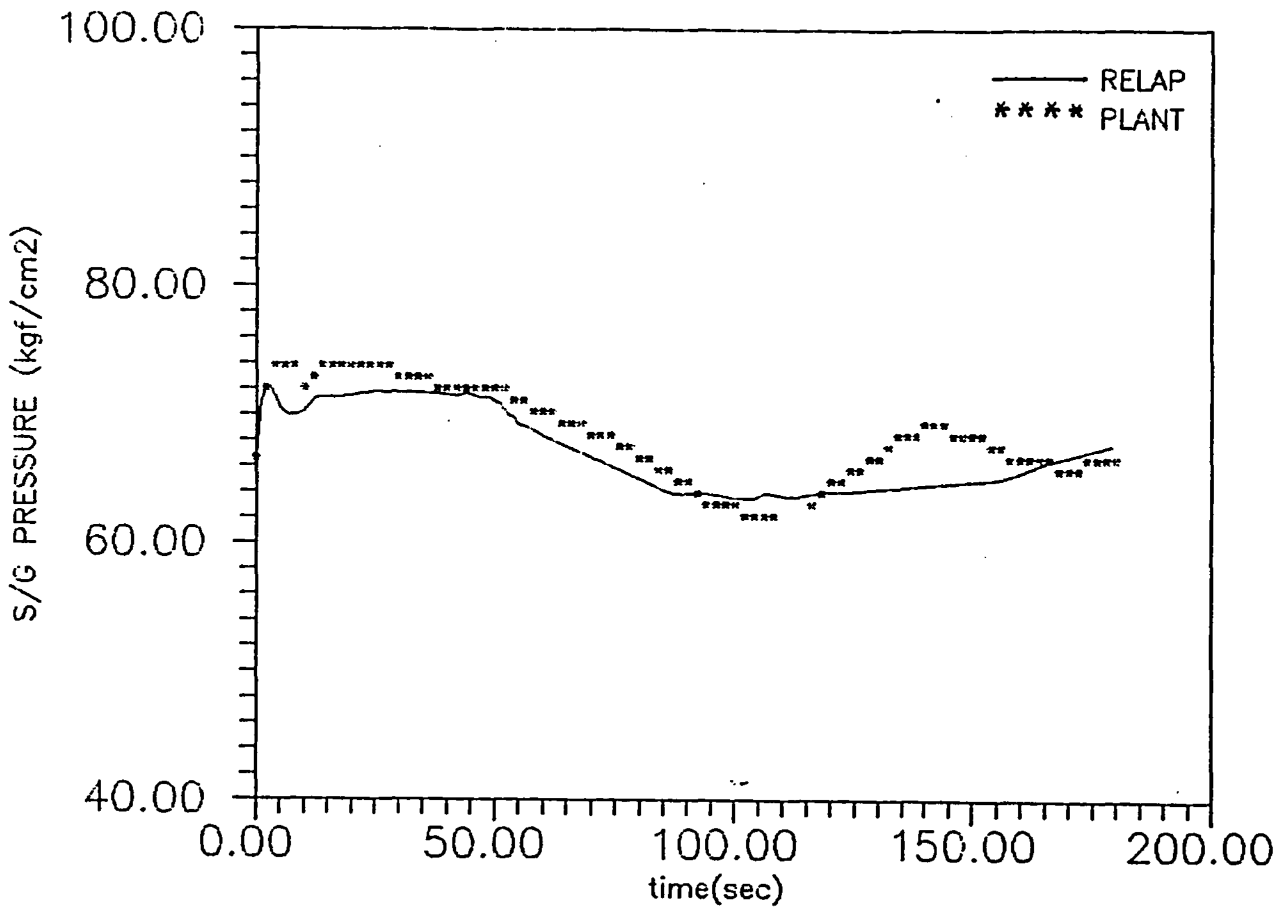


Figure 5.7 STEAM GENERATOR PRESSURE

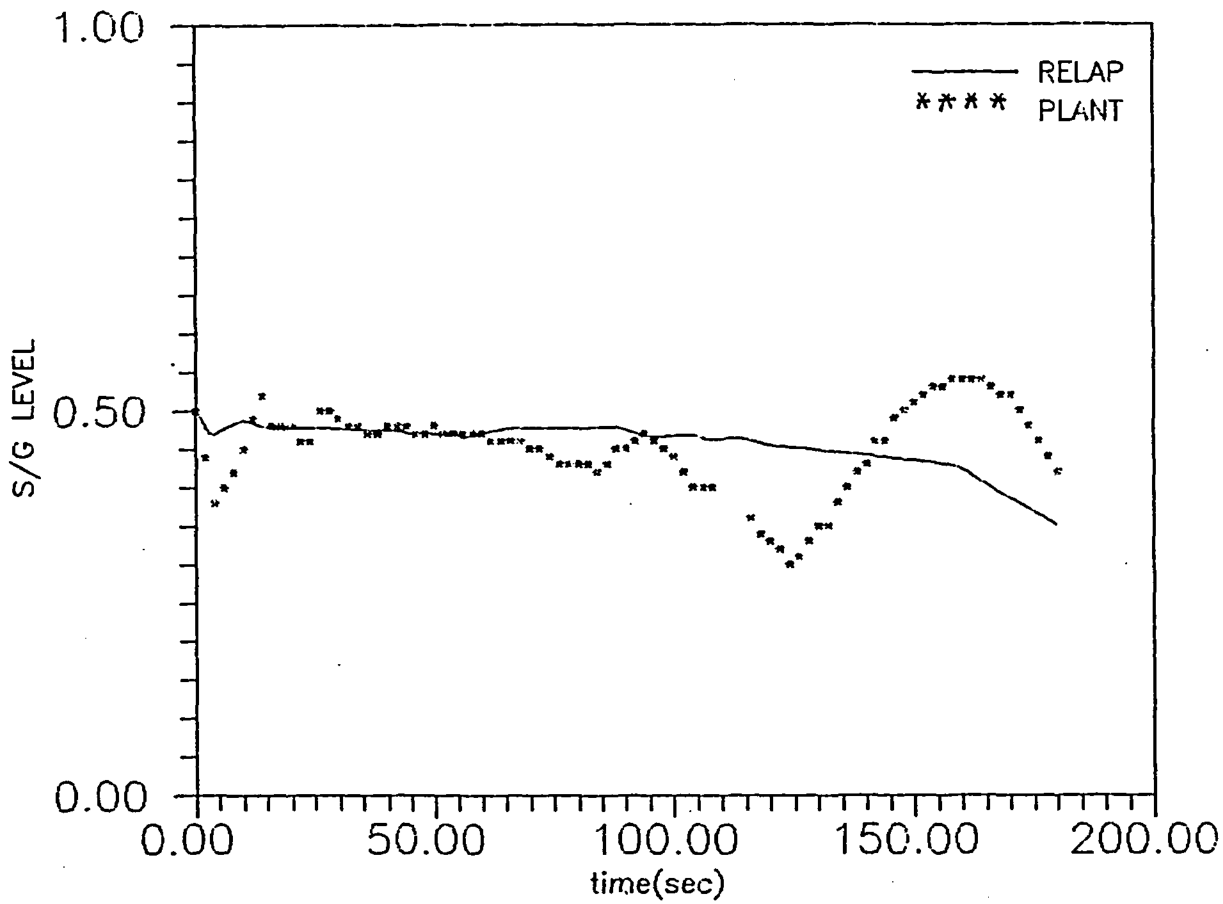


Figure 5.8 STEAM GENERATOR LEVEL

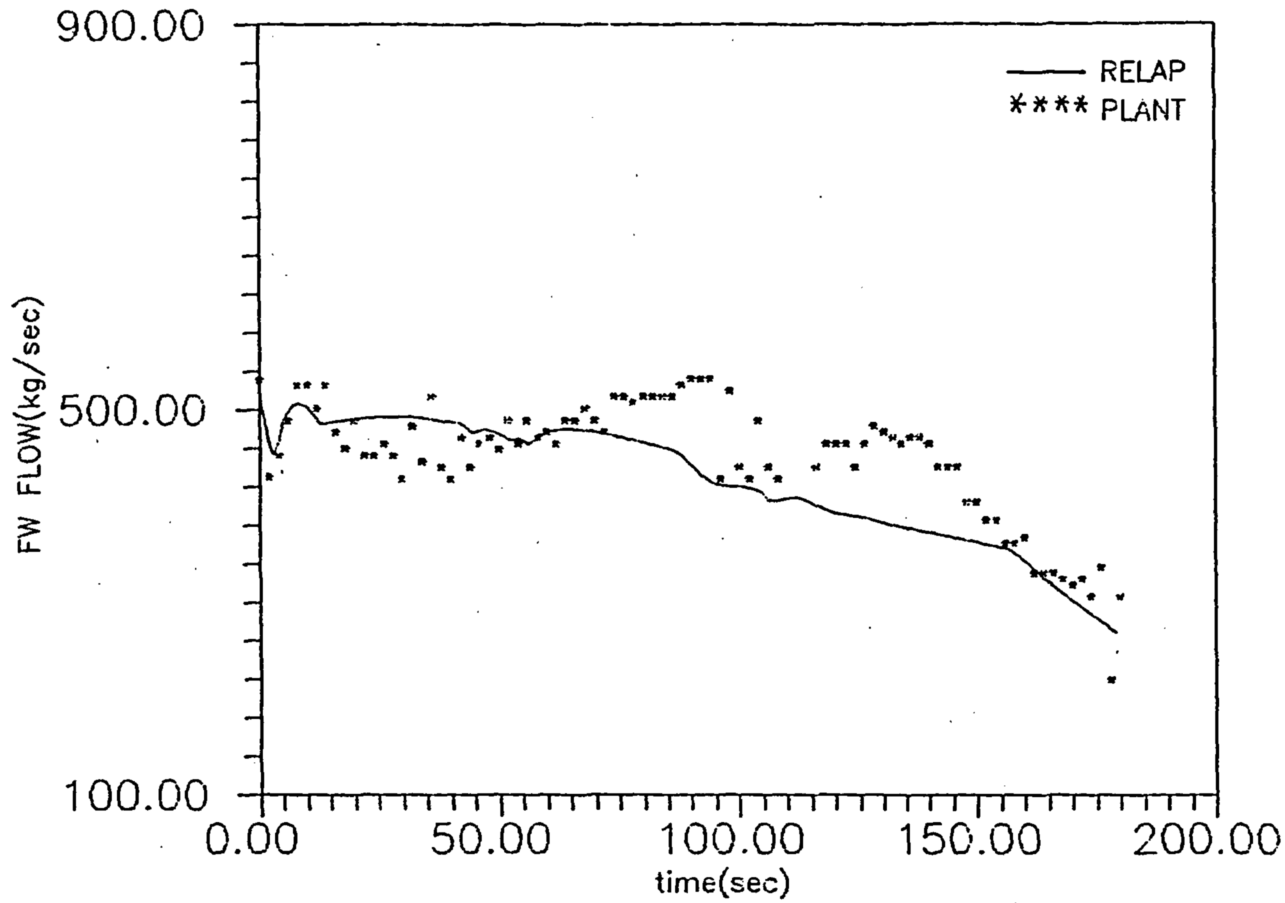


Figure 5.9 FEED WATER FLOW(kg/sec)

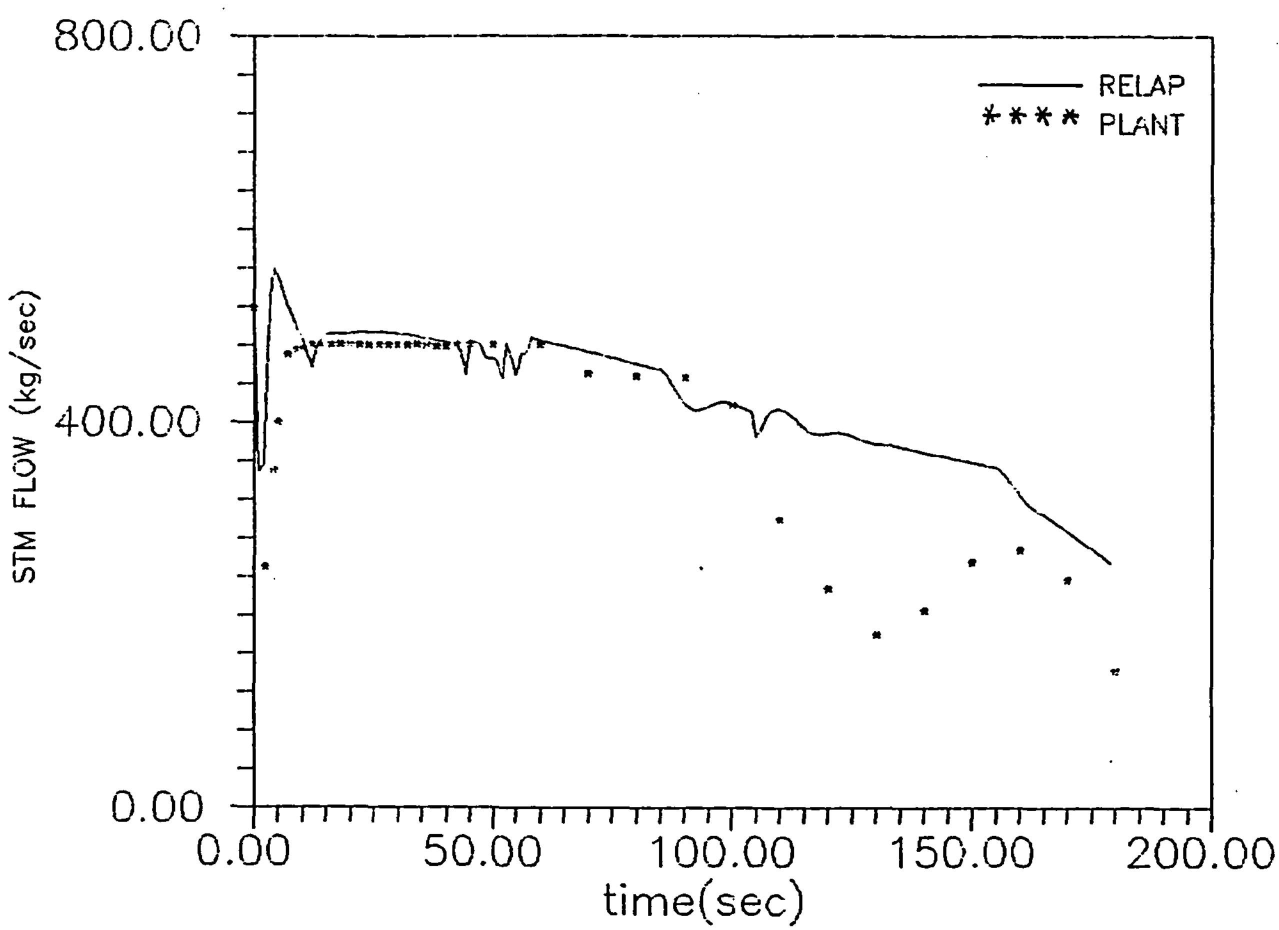


Figure 5.10 STEAM FLOW (kg/sec)

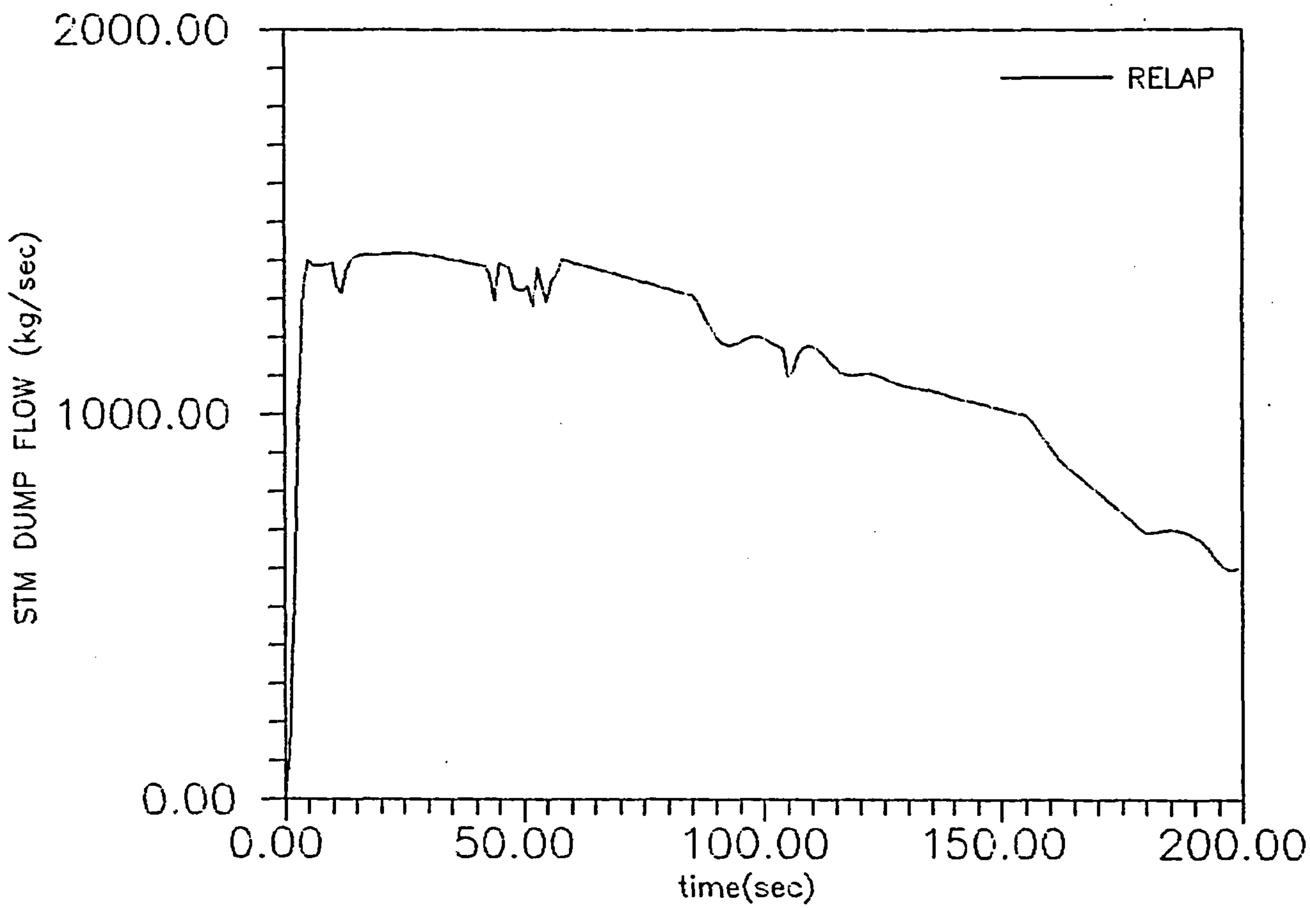


Figure 5.11 STEAM DUMP FLOW

6.0 sensitivity study

The sensitivity study has been performed for the control rod control system. The variable reactivity worth for the control rod was used to achieve good prediction of neutron power and also to check whether the control rod control system is working properly. Figure 6.1 shows the result of simulation which indicates the reactor power with the different control rod reactivity worth. DATA2 has a double reactivity worth compared to the DATA1, which has the reactivity worth with minor modification in the reactivity worth obtained from the nuclear design report. One thing should be pointed out here that it is very difficult to get the accurate reactivity worth in the core during the test. The various reactivity worth was used to achieve the good prediction of neutron power. Because the main purpose of this sensitivity study is to know the effect of the control rod reactivity worth, the other parameter except the reactivity worth was not changed at all. DATA2, which has the double reactivity worth, shows the earlier feedback effect of the moderator temperature coefficient. It reveals this effect around 50 second with the peak power rise up to 80 % full power but DATA1, having the original reactivity worth, indicates the effect of moderator temperature coefficient at 80 second with a less peak power up to 70%full power. After this feedback effect, DATA2 shows the sharper reduction of the reactor power than DATA1. Around 180 second, DATA2 shows 10higher reactor power than

DATA1. The effect on the other thermal-hydraulic parameters of primary and secondary system, i.e. coolant temperature, pressure and steam generator pressure, was shown the same as reactor power.

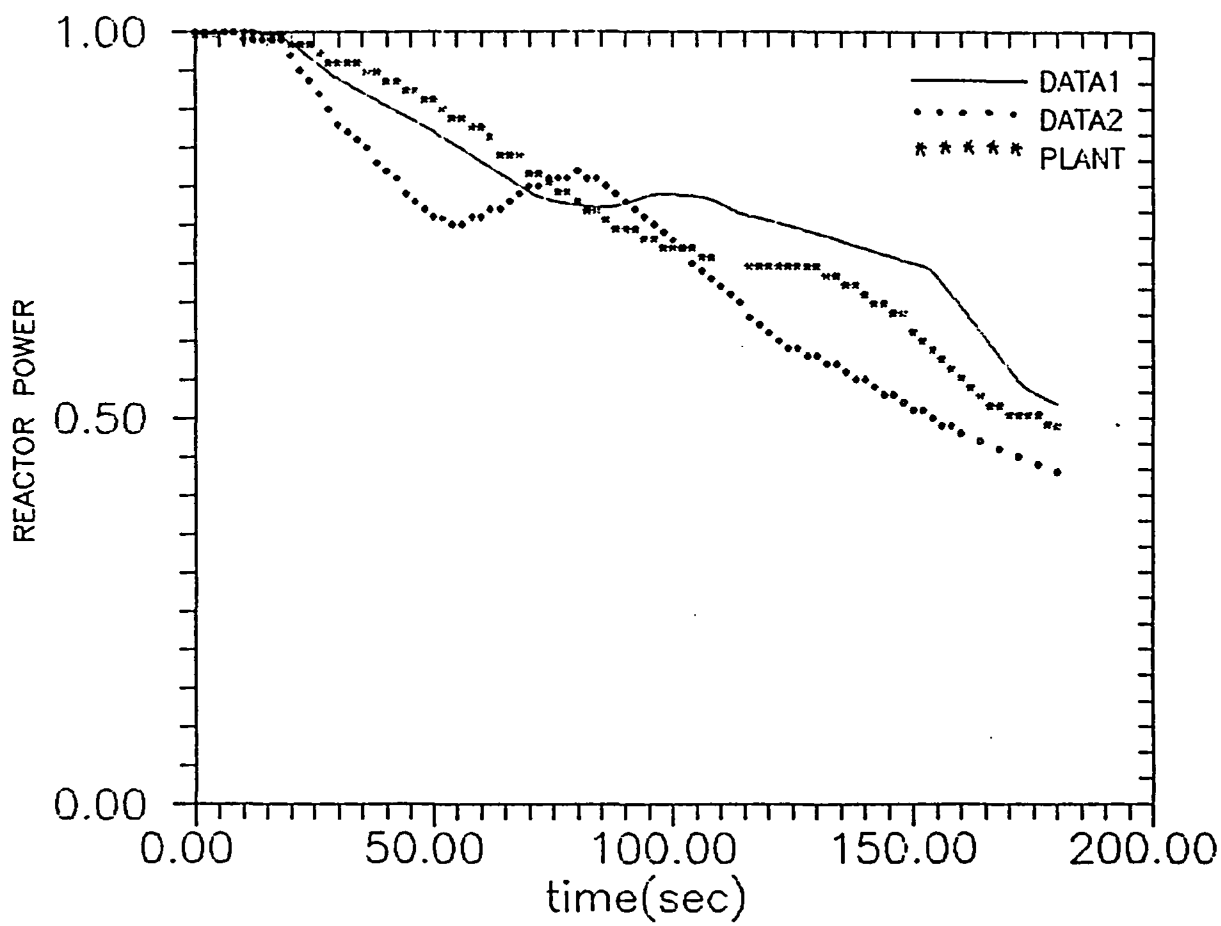


Fig. 6.1 SENSITIVITY STUDY (RX POWER)

Table 4. Reactivity Worth of Control Rod

Rod Position (step)	Reactivity Worth ($)	
	DATA1	DATA2
0.0	0.0	0.0
6.0	-0.0001	-0.0002
12.0	-0.0003	-0.0006
18.0	-0.0035	-0.007
32.0	-0.055	-0.11
58.0	-0.114	-0.228
82.0	-0.181	-0.362
102.0	-0.203	-0.406
130.0	-0.242	-0.484
136.0	-0.292	-0.584
146.0	-0.375	-0.750
154.0	-0.445	-0.890
166.0	-0.482	-0.964
192.0	-0.508	-1.016
204.0	-0.522	-1.044
216.0	-0.566	-1.132
228.0	-0.609	-1.218

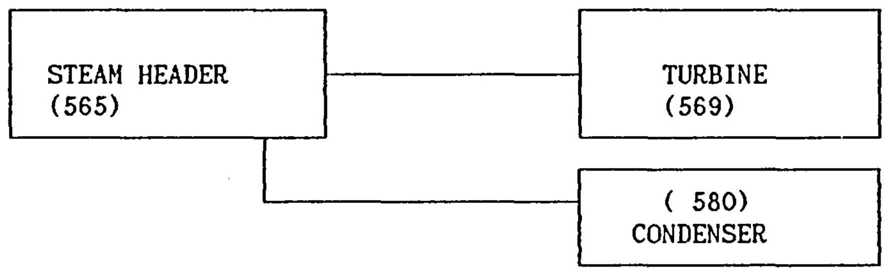
7.0 Nodalization Study

The steam dump system is selected for the nodalization study because the dump steam flow rate via steam dump valves has an important effect on deciding the behaviors of the thermalhydraulic parameters in the primary and secondary system. It is of great interest to find out the effect of nodalization in the steam dump system on thermalhydraulic parameters. As shown in Figure 7.1, two cases of nodalization are selected for this nodalization study. Case 1 illustrates the boundary condition model which uses two volumes(turbine and condenser) and one time dependent junction (steam dump valve) where the steam dump flow rate can be treated as the boundary condition in the steam dump valves. The steam dump flow rate of this boundary condition is determined by the value specified in the P,L,S (precaution,limitation and setpoints) which is a function of temperature error (Tavg-Tref) [1].

CASE 2 describes the steam dump system in detail. The steam dump system is composed of 4 banks with 16 steam dump valves. In order to describe this system, the dump system modeling in the RELAP5/MOD2 simulation consists of 4 volumes for condenser and atmosphere and 4 junctions for valves where the survo valve model was used in RELAP5 and the valve is working according to the normalized flow area calculated by the control system.

The simulations for these two different cases (case 1 and case 2) were performed to evaluate the effect of the different nodalization in the steam dump system . The steam dump flow rates for both cases are shown in Figure 7.2. The steam dump flow rate in Case 1 is around 120 kg/sec lower than that in Case 2 for the first 70 seconds and then drops faster than Case 2. Figure 7.3 and Figure 7.4 show the comparisons of pressurizer pressure and steam generator pressure between two cases. As shown in Fig. 7.2, Case 2 computed the steam dump flow rate higher than Case 1 after 70 seconds and this induced that the pressurizer and steam generator pressure, shown in Figure 7.3 and Figure 7.4, in Case 1 reveals a little higher trend than that in case 2. The results of the comparison indicates that valve junction model is more effective to describe the steam dump system than the boundary condition model .

1.0 BOUNDARY CONDITION NODALIZATION



2.0 VALVE JUNCTION NODALIZATION

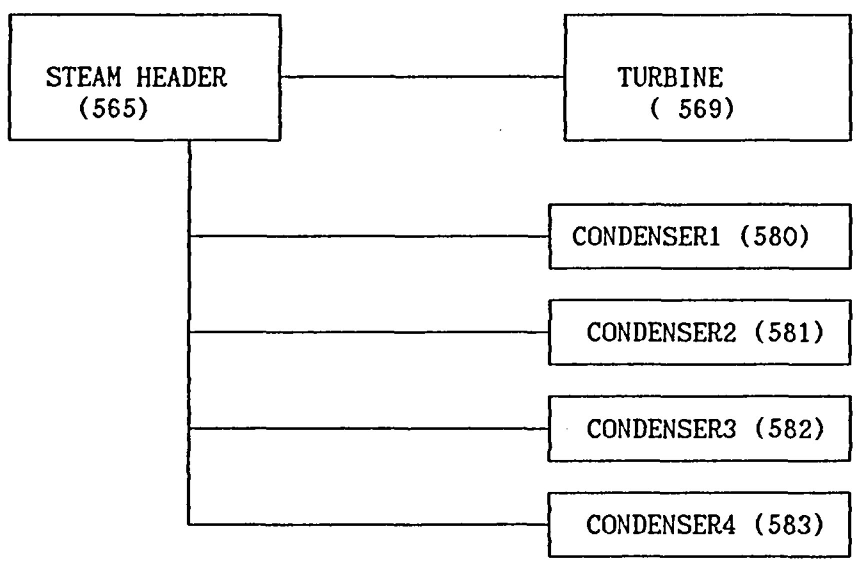


Fig 7.1 Nodalization study (two cases)

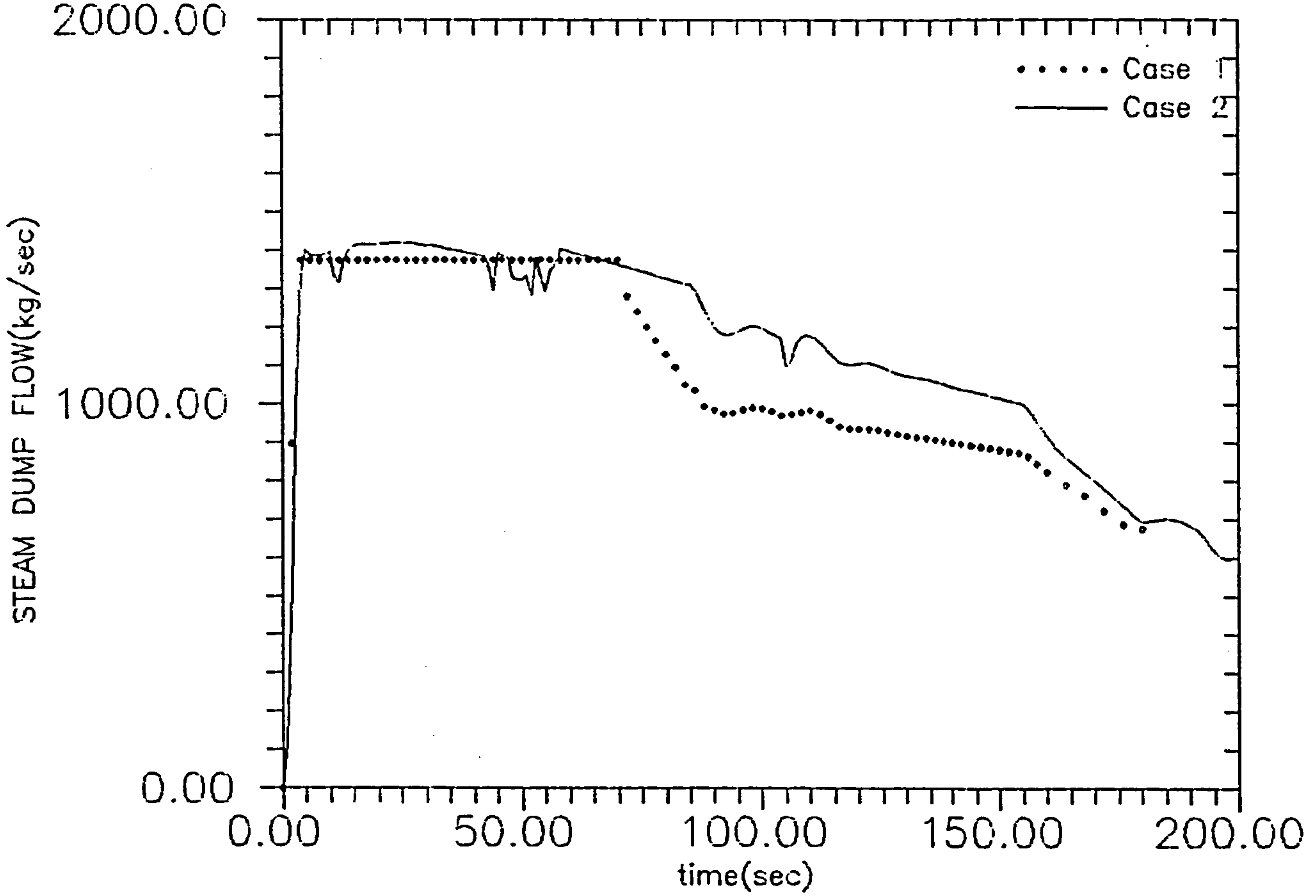


Fig. 7.2 Nodalization study (steam dump flow)

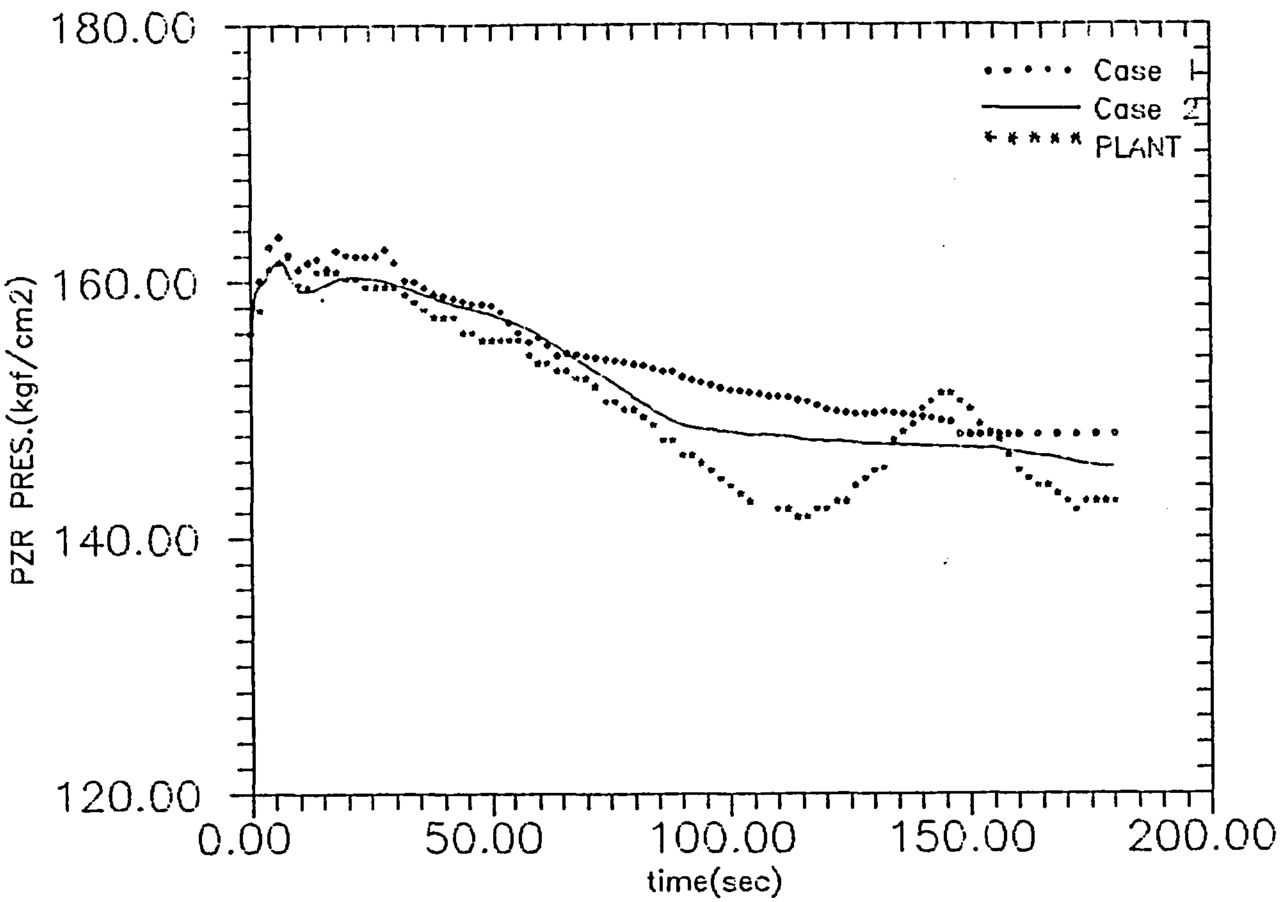


Fig. 7.3 Nodalization study (pressurizer pres.)

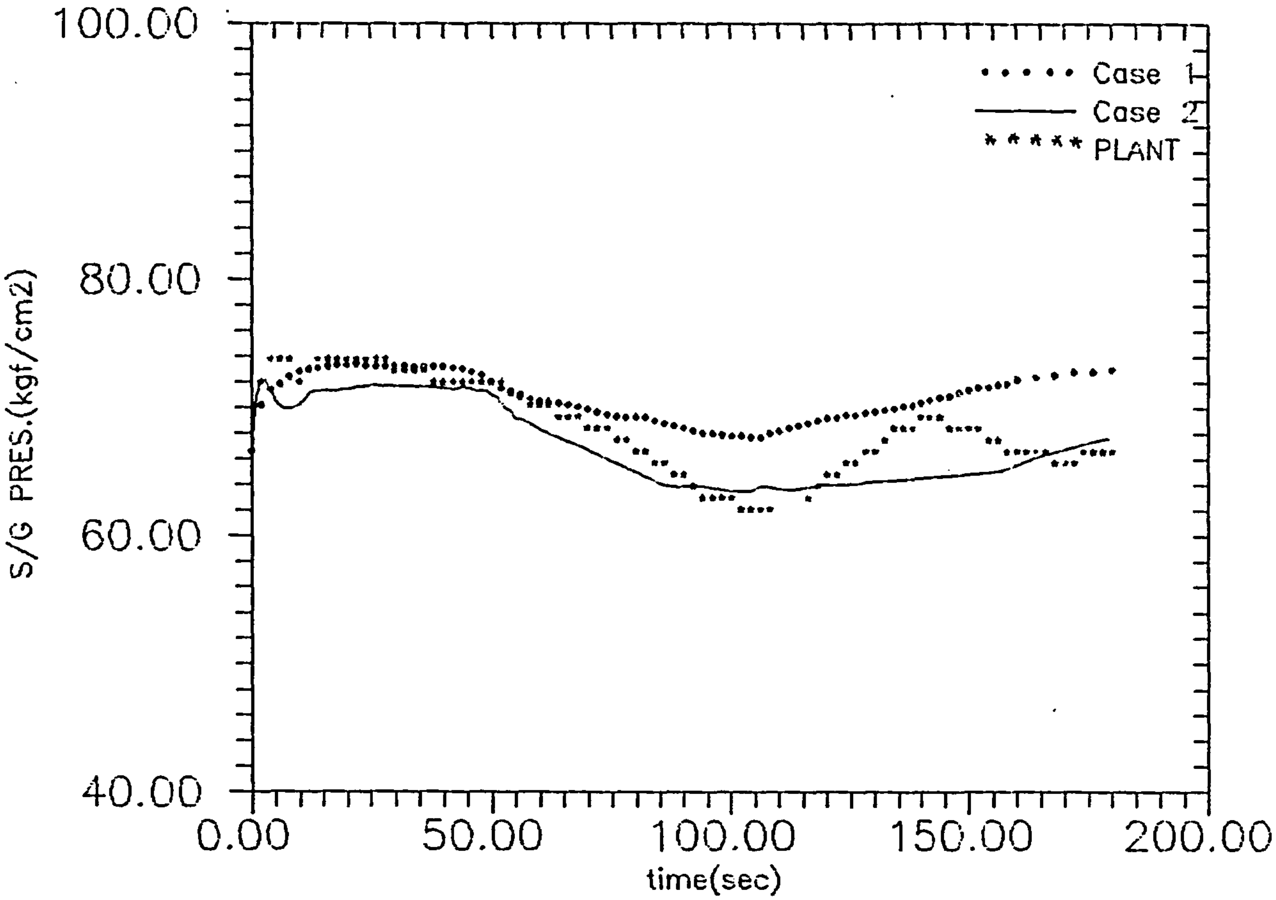


Fig. 7.4 Nodalization study (S/G pres.)

8.0 Run statistics

The computer type used in this analysis is CDC CYBER 170-875 which has one unified extended memory unit and the operating system is the NOS 2.6.1 level 700. The Figure 8.1 and 8.2 show a plot of CPU vs RT and a plot of DT vs RT. The total CPU time fot the simulation of the whole transient 180 seconds is 705.44 second. The calculated grind time is around 1.66 second and the requested maximum time step size is 0.1 second. Figure 8.1 indicates that the CPU time is proportional to the real transient time with the slope of 3.9. The time step size,shown in figure 8.2, shows a constant value(0.05) . This result reveals that this transient is not so severe that the advancement of calculation was succesful with the given maximum time step.

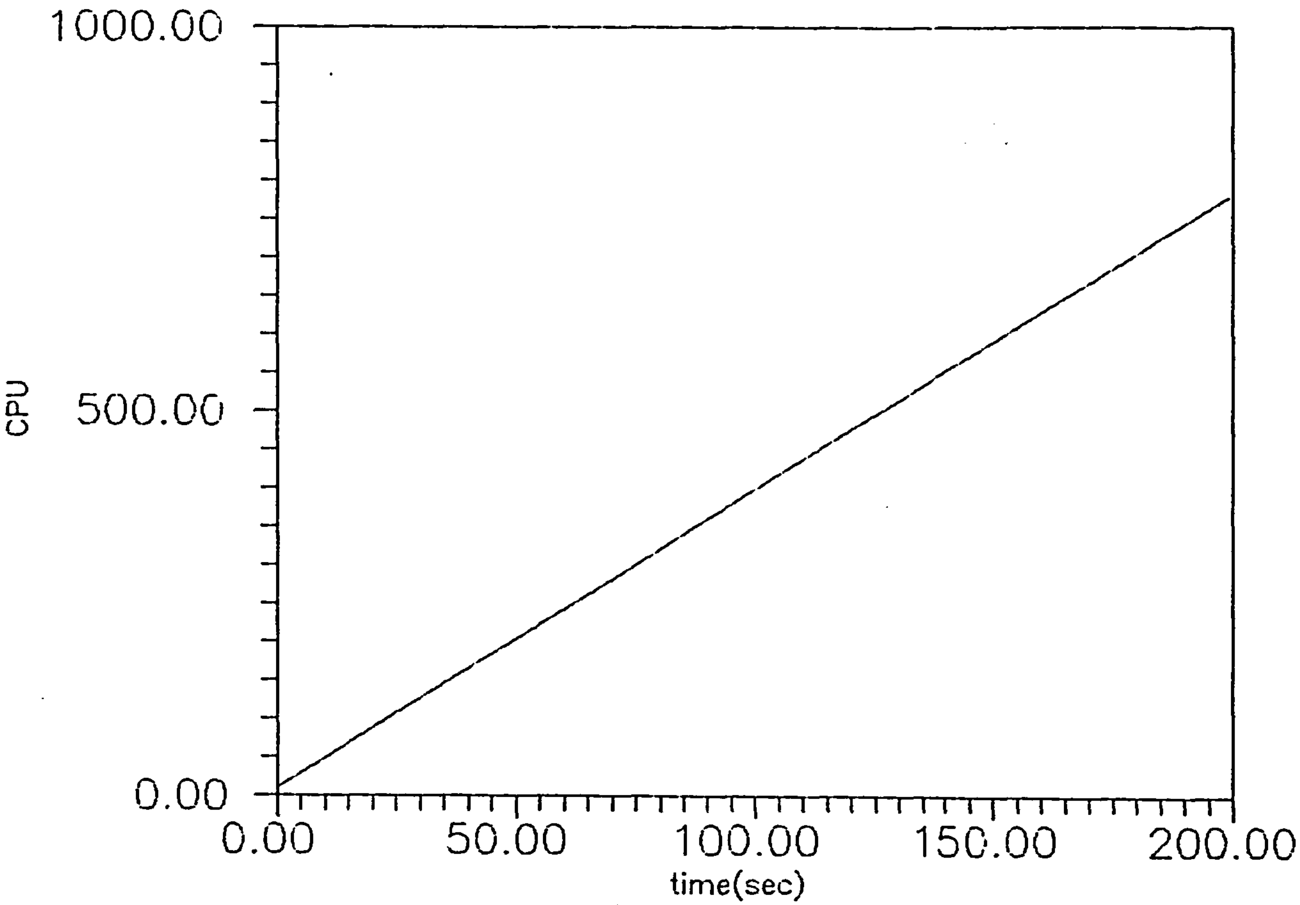


Figure 8.1 CPU

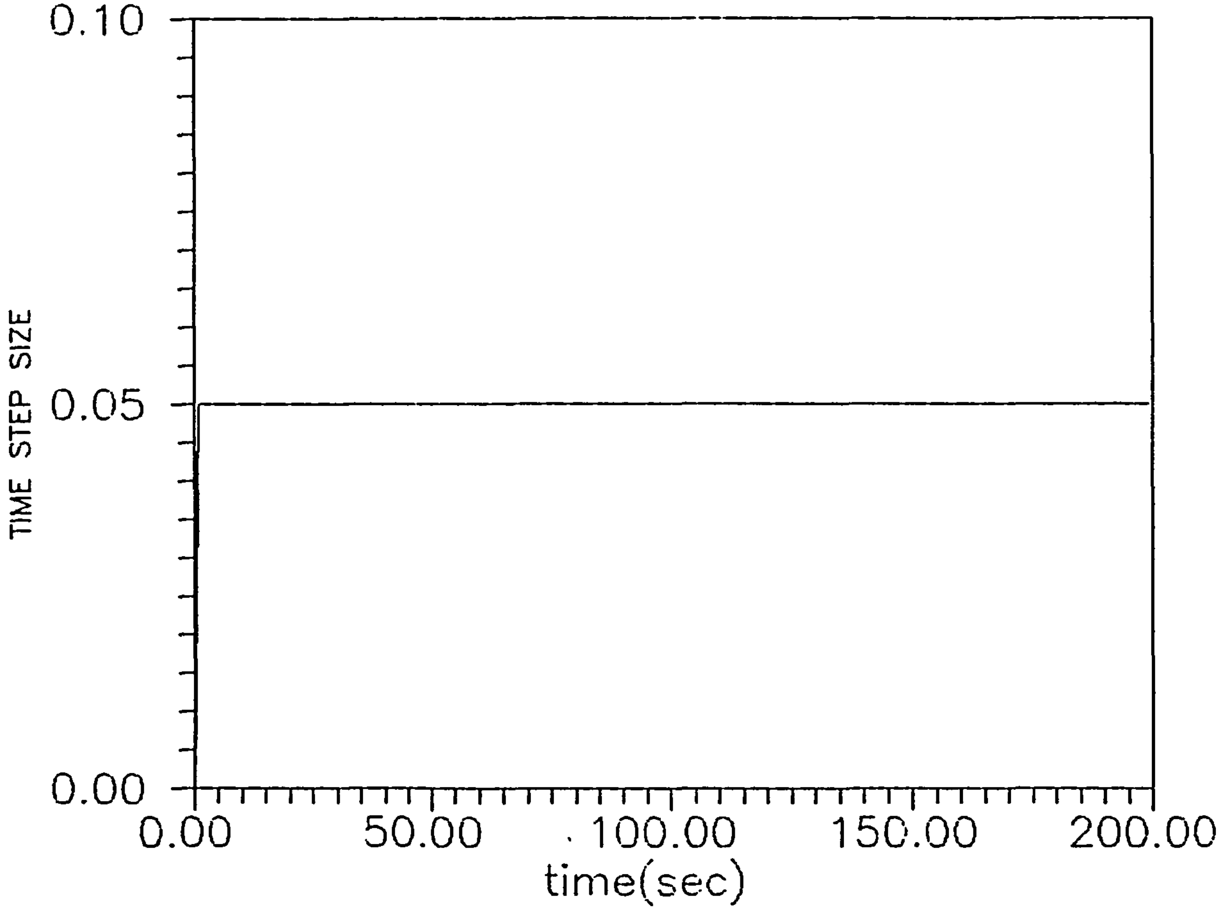


Figure 8.2 TIME STEP SIZE

9.0 conclusions

The best estimate transient analysis for Yong-Gwang unit 2 NLTT was performed by using the RELAP5/MOD2 computer code. Based on the comparisons between the calculated and measured results, the following conclusions can be drawn.

For the NLTT, comparisons of the plant major primary system T/H parameters, such as reactor power, Tavg, PZR level and pressure, indicates that RELAP5 could give the accurate results to the plant operating data. However, the calculated primary thermal-hydraulic parameters show slightly higher than the plant operating data for the latter part of transient(after 70 second). This is due to model uncertainties in RTD sensing line and lower steam generator steam pressure. It is also observed that sucessful prediction of the PZR level and pressure is largely dependent upon the accurate prediction of the transient behavior of the RCS Tavg.

The calculated behaviors of secondary side parameters, such as steam generator water level, steam pressure, feedwater flow and steam flow rates are also agreed well with the plant operating data. If the steam header pressure is prescribed as the boundary condition, the calculated behaviors of the secondary side parameters may be agreed better, but for the auto control of steam dump system, it is not adapted.

The sensitivity study on the control rod worth is carried out for the 100 % NLTT. It was found that various rod worth should be used to achieve good prediction of neutron power. Hence , additional sensitivity studies regarding control rod modeling are required in the future.

The nodalization study for the two cases, boundary condition model and valve junction model, was performed and comparison was made. It is observed that the valve junction model is more effective to describe the behavior of this NLTT transient than the boudary condition model.

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11. ABSTRACT (200 words or less)

The results of the RELAP5/MOD2 computer code simulation for the 100% Net Load Trip Test in Yong-Gwang Unit 2 are analyzed here and compared with the plant operation data. The control systems for the control rod, feedwater, steam generator level, steam dump, pressurizer level and pressure are modeled to be functioned automatically until the power level decreases below 30% nuclear power. A sensitivity study on control rod worth was carried out and it was found that variable rod worth should be used to achieve good prediction of neutron power. The results obtained from RELAP5/MOD2 simulation agree well with the plant operating data and it can be concluded that this code has the capability in analyzing the transient of this type in a best estimate means.

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