Journal of the Korean Nuclear Society
Volume 27, Number 1, February 1995

Development of TASS Code for Non-LOCA Safety Analysis Licensing Application

Han Young Yoon, Geun Sun Auh, Hee Cheol Kim, Joon Sung Kim, and Jae Don Choi
Korea Atomic Energy Research Institute
(Received July 26, 1994)

Non-LOCA 인허가 해석용 TASS 코드의 개발

윤한영 · 어근선 · 김희철 · 김준성 · 최재돈
한국원자력연구소
(1994. 7. 26 접수)

Abstract

Since the current licensed system codes for Non-LOCA safety analysis are applicable only for a specific type PWR, it is necessary to develope a new system analysis code applicable for all types of PWRs. As a R&D program, KAERI is developing TASS code as an interactive and faster-than-real-time code for the NSSS transient simulation of both CE and Westinghouse plants. It is flexible tool for PWR analysis which gives the user complete control over the simulation through convenient input and output options. In this paper the code applicability to Westinghouse type plants was verified by comparing the TASS prediction to plant data of loss of AC power and loss of load transients, and comparing to the prediction of RELAP5/MOD3 for feedline break, locked rotor, steam generator tube rupture and steam line break accidents.

요 약

현재 사용중인 Non-LOCA 해석용 인허가 코드들은 특정한 형태의 가압경수로에 맞게 짜여진 것들이어서 모든 형태의 가압 경수로에 적용할 수 있는 범용 코드의 개발이 필요한 실정이다. 이를 위하여 한국원자력연구소에서는 웨스팅하우스 및 CE형 발전소에 공히 적용할 수 있는 과도현상 해석 코드인 TASS 코드를 개발하고있다. 이 TASS 코드는 실시간 보다 빠르게 핵증기계통에 대한 모의 계산을 수행하며 대화식의 입출력을 통하여 사용자가 원하는 과도현상을 정확히 모사할 수 있다. 본 논문에서는 웨스팅하우스형 발전소에 대하여 TASS 코드를 적용하여 Non-LOCA 인허가 해석을 하기 위한 검증을 위해, 교류 전원 상실사고와 부하상실사고에 대하여 발전소 실측자료와의 비교계산을 수행하였고 주급수관 파단사고, 펌프축 고착사고, 증기발생기 세관 파열사고 및 주증기관 파단사고들에 대하여 대형코드인 RELAP5/MOD3 코드와의 비교계산을 수행하였다.

1. Introduction

There are two licensed Non-LOCA safety analysis codes available in Korea for PWR design applications, i.e., NLOOP[1] and CESEC[2]. NLOOP is used for NSSS transient simulation of Westinghouse plants and CESEC is used for CE plants. Since the two codes are hardwired programs, it is not possible to apply NLOOP code for CE plants or CESEC for Westinghouse plants. In order to overcome this applicability limit, an unified computer code applicable for both types of PWRs is necessary. As a R&D program, TASS (Transients and Setpoints Simulation) code has been developed to meet this need through generalization of inputs and controller models which provide perfect simulation of all kind PWR controllers. TASS provides a digital simulation of a Nuclear Steam Supply System (NSSS) for a wide range of operating conditions. TASS is a highly flexible analytical tool which models major plant components for both the primary and secondary systems as well as the control and protection systems. It calculates the transient behavior of a PWR for normal and abnormal conditions including accidents. TASS determines the core power and heat transfer through the NSSS. It also computes the thermal and hydraulic behavior of the reactor coolant in the primary and secondary systems. It includes the primary and secondary control systems and the balance-of-plant fluid systems. Extensive testing and verifications were performed for Westinghouse nuclear power plants by comparing the results of TASS with plant data (loss of AC Power and loss of load), and with the predictions of RELAP5/MOD3[3], which is a NRC developed computer code used for accident analysis including LOCA. For the comparisons with RELAP5/MOD3, the following Non-LOCA design basis events were analyzed.

- Seized Rotor Accident
- Feedline Break Accident
- Steam Generator Tube Rupture Accident
- Steamline Break Accident

Each of the above four events was chosen as a most extreme case of flow transients, RCS heatup transients, RCS coolant flow loss transients, and RCS cooldown transients.

2. Mathematical Model

2.1. Reactor Coolant Model

The RCS thermal-hydraulic model is formulated with five one-dimensional conservation equations. The conservation variables are mixture (liquid and steam) mass, liquid mass, mixture energy, steam energy, and mixture momentum. The mass and energy for the mixture are calculated for each node. Mass flowrate is calculated for each flowpath. The conservation equations are :

- Conservation of Liquid Mass

$$\frac{dM_l}{dt} = \sum(1-x)\,W + W_{cond}$$

- Conservation of Mixture Mass

$$\frac{dM}{dt} = \sum W$$

- Conservation of Mixture Energy

$$\frac{dE}{dt} = \sum Wh + Q$$

- Conservation of Steam Energy

$$\frac{dE_{stm}}{dt} = \sum xWh_{stm} + Q_{stm} - W_{Cond}\,h_{stm} + W_{cond,\,boil}\,(h_{stm} - h_g) + W_{cond,\,surf}\,(h_{stm} - h_f) + W_{cond,\,wall}\,(h_{stm} - h_f)$$

The summations in the above four equations are over all momentum and non-momentum paths connected to each given node.

- Conservation of Mixture Momentum

$$\frac{1}{g}\left(\frac{L}{A}\right)\frac{dW}{dt} = (P_u - P_d) - K_f\frac{\phi W|W|}{\rho g A^2} - K_g\frac{W|W|}{\rho g A^2} + \triangle P_{elev} + \triangle_{pump}$$

where,

M = mixture mass, M_l = liquid mass, x = steam quality, W = mixture flow rate, W_l = liquid flow rate, W_{cond} = total condensation rate,

E = mixture energy, E_{stm} = steam energy, Q = heat rate,

h_g = saturated steam enthalpy, h_f = saturated liquid enthalpy,

h_{stm} = steam enthalpy, Q_{stm} = heat rate to steam region,

$W_{cond,\ boil}$ = boiling rate (negative), $W_{cond,\ wall}$ = condensation rate on wall

$W_{cond,\ surf}$ = condensation rate on liquid surface, L = path legnth,

A = flow area, P_u = upsteam pressure, P_d = downstream pressure,

$\triangle P_{elev}$ = pressure change by elevation,

$\triangle P_{pump}$ = pressure change by pump,

K_f = Reynolds number dependent friction k-factor for turbulent flow[4],

K_g = geometric k-factor, g = gravity constant, ρ = density,

ϕ = two-phase friction multiplier (Tom and Martinelli-Nelson)[5, 6].

The above equations are obtained by means of standard integration procedures of the multidimensional conservation of mass, energy and momentum equations[7, 8, 9]. In addition, for convenience during the integration procedure, an additional conservation equation (conservation of steam mass Ms)

$$\frac{dM_s}{dt} = \sum xW - W_{cond}$$

is integrated. The code will use the pairs M_l, M or M_s, M depending on the non-equilibrium condition (pressure, and liquid and steam enthalpy).

2.2. Secondary System Models

The secondary system for a PWR is represented by node-flowpath model shown in Figure 1. Three nodes represent the secondary side of each steam generator-a downcomer (saturated or subcooled liquid and saturated steam), an evaporator /riser/economizer region (saturated or subcooled liquid and saturated steam) and a dome (saturated or superheated steam). One additional node represents the common steamline header. This system representation allows accurate modeling of the recirculation phenomena and the downcomer and evapoartor water levels. The model represent all major components including the secondary safety valves, atmospheric dump and bypass valves, main steam isolation valves, steamline and feedline check valves, and a steam generator blowdown system. The steam generator secondary model conserves mass and energy in each node by

$$\frac{dM}{dt} = \sum W_{in} - \sum W_{out} \ and$$

$$\frac{dU}{dt} = \sum (WH)_{in} - \sum (WH)_{out} + \sum Q_{in} - \sum Q_{out}$$

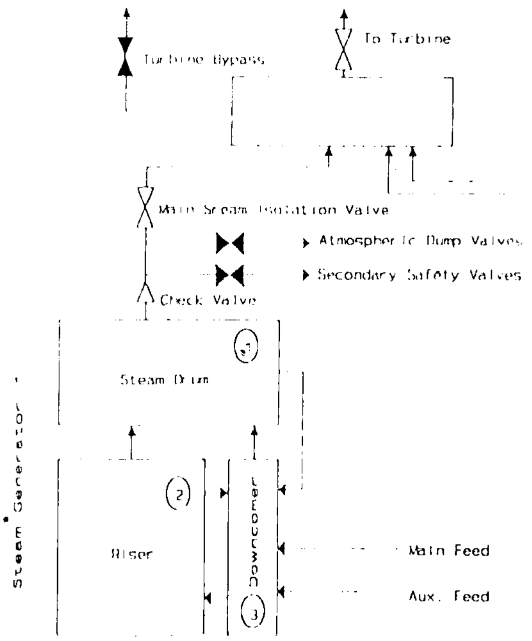

Fig. 1. Steam Generator Secondary System Geometric Model

where,
M = mass, U = internal energy, W = flow rate,
Q = heat transfer rates, H = specific enthalpies.

Once the above two equations are integrated over a computational time step, the new pressure is found by an iterative solution of the equations,

$$M_f V_f(P) + M_g V_g(P) = V,$$
$$M_f + M_g = M,$$
$$M_f U_f(P) + M_g U_g(P) = U,$$

where,
M_f, M_g = liquid and steam masses,
V_f, V_g = liquid and steam specific volumes, function of pressure,
U_f, U_g = liquid and steam specific internal energies, function of pressure, P = pressure, V = total volume, M = total mass, U = total internal energy.

2.3. Control System

TASS provides a very flexible method to handle control systems for the core, primary system, secondary system and selected balance of plant systems. This is done by shell program and a set of generic control system modules. The shell provides interface for the process modules, control modules and the user. The systems provided by the shell are :

- Reactor protective system,
- Pressurizer level control and CVCS,
- Pressurizer pressure control,
- Primary system relief valves,
- Safety injection systems,
- Turbine control, admission, isolation, dump, relief, and bypass valves,
- Feedwater main and auxiliary, and
- Control rod regulating systems.

The actual controllers are defined by means of the generic control system modules. The system modeller selects the level of detail to be provided for each controller and then assembles each controller using the generic modules as building blocks. This provides a very flexible capability to handle plant specific characteristics of control systems. The resulting control system modeling scheme enables the system modeler to define simple or more sophisticated controllers as needed. The basic building block available to the control system modeller is the functional element. It is generic module, designed to perform the following three functions :

- Input : The element receives information from Global Common (e.g., pressure, setpoint, control group outputs)
- Proccesing : The element processes the input data according to its specific design. TASS provides twenty-eight functional elements types, performing simple arithmetic and logical operations, differential and integral transfer functions, block branching, simulation of valve characteristics, and specialized functions.
- Output : The element generates a single numerical result which is available as input to other elements, or as the final output of the control group.

Figure 2 demonstrates how these inter-element communication rules are observed.

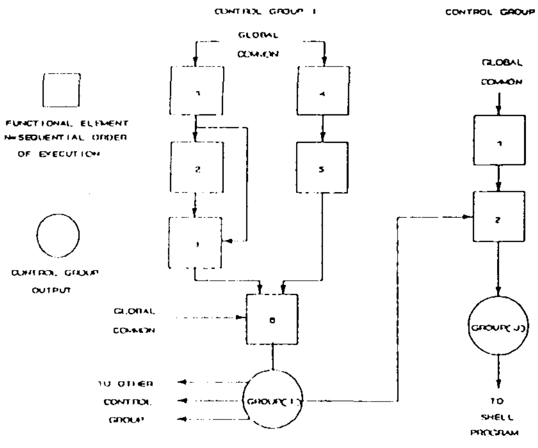

Fig. 2. Schematic Structure of Control Groups

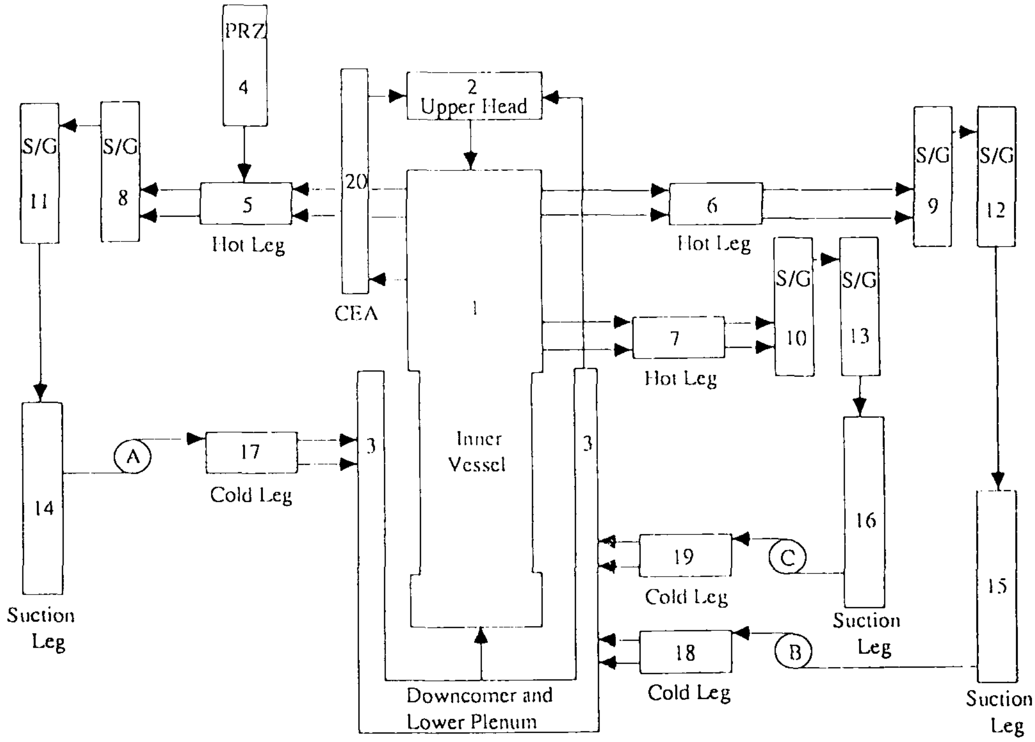

Fig. 3. TASS Code Nodalization

3. Verification for Westinghouse Type PWRs

Verification of TASS included comparison of plant behavior as predicted by TASS both to measured plant data obtained during plant operation and to the results of accident simulations typical of those performed in support of Plant A. Plant A is one of two virtually identical units on the same site. The Nuclear Steam Supply Systems (NSSS) for both units were supplied by Westinghouse. The initial license was to operate each of the facilities at a core thermal power output of 2775MWt. Site parameters, the major systems and components including the engineered safety features and the containment structures were evaluated at a core power level of 2775MWt.

3.1. Comparison to Plant Data

Two plant transients of Plant A were selected for comparison to the predictions of the TASS code, i.e., complete loss of AC power from 100% power, and loss of electrical load test from 100% power. The basis for the choice of the transient is that sufficient data was taken during the transient to perform a meaningful comparison and that the transient exercised major models of the TASS code.

Complete Loss of AC Power from 100% Initial Power

The complete loss of AC power from 100% power transient was initiated by a breakdown in main transformer, which resulted in complete loss of AC power. Simultaneously, DC power was automatically provided to the safety-related equipment. The complete loss of AC power was immediately followed by a turbine trip and a reactor trip. Table 1 show the se-

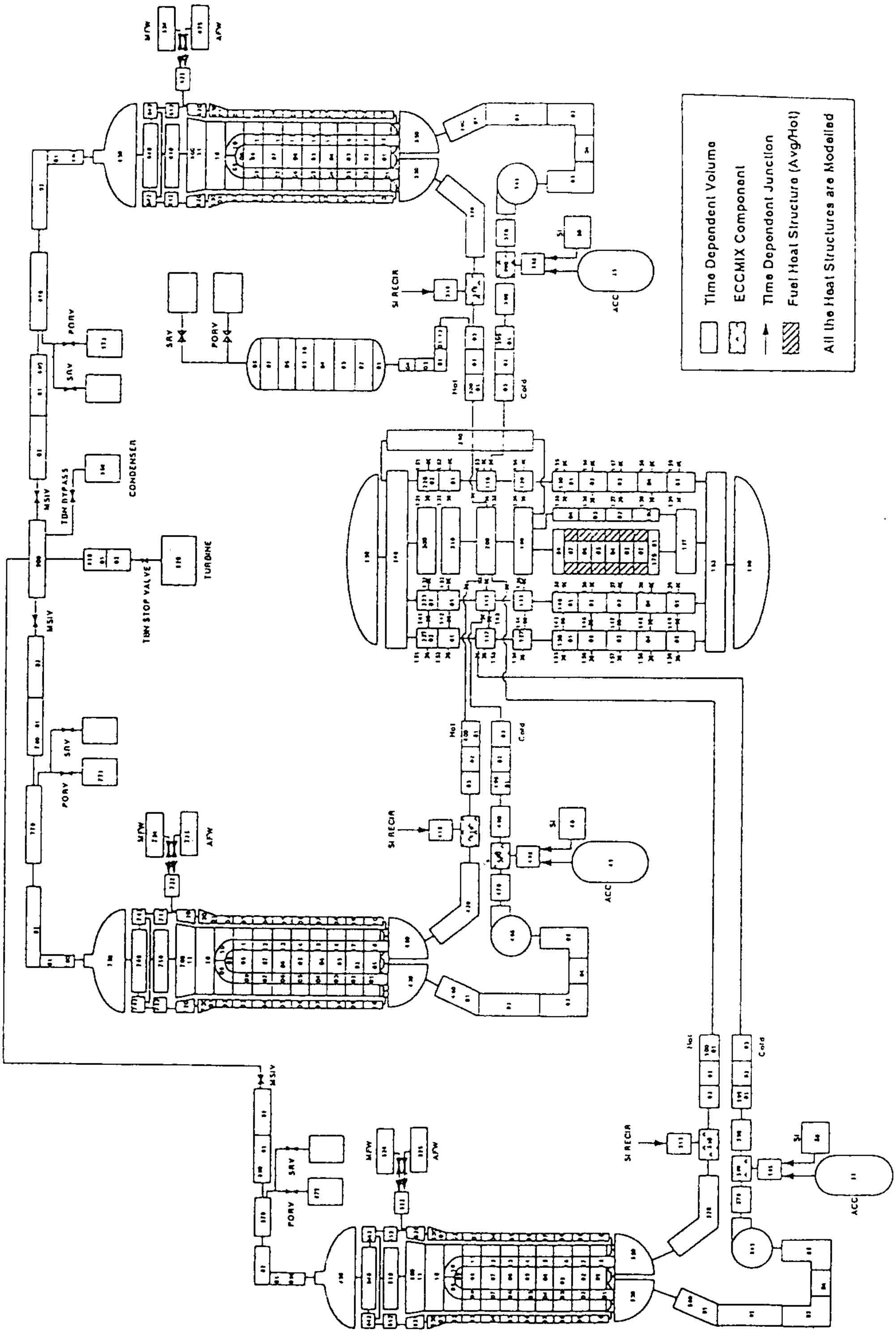

Fig. 4. RELAP5/MOD3 Nodalization

Table 1. Sequence of Events for Complete Loss of AC Power from 100% Initial Power

Time (sec)		Event	Value	
TASS	Plant Data		TASS	Plant Data
0.0	0.0	Main Transformer Damage	–	–
0.2	0.2	Turbine Trip	–	–
0.2	0.3	Reactor Trip	–	–
1.3	1.2	RCP Coastdown	–	–
68.0	68.0	Minimum Pressurizer Pressure, MPa	14.5	14.3

quence of events from both the plant test data and the TASS simulation of the event. Figures 5 and 6 portray comparisons of important parameters. The results show that the differences between the prediction of TASS and the plant data are within acceptable ranges.

Loss of Electrical Load Test from 100% Initial Power

The objectives of this test are to demonstrate the ability of the plant to sustain a 95% load-loss at 100% power without reator trip and turbine trip, and to evalute the system reponse to the transients. However, the plant was tripped by a pressurizer pressure low signal which was caused by an over-response of the pressurizer Power Operated Relief Valve (PORV). Three of the four steam dump bypass valve banks opened within 5 seconds after the Power Circuit Breaker (PCB) opening. The fourth bank opened 8 seconds after PCB opening. The PORV also opened due to the increase of the pressurizer pressure. Table 2 shows the sequence of events from both the plant test data and the TASS simulation of the event. All of the control systems were automatically controlled in the TASS simulation. Figures 7 and 8 portray comparisons of important parameters. TASS predicted the maximum pressurizer pressure at 6 seconds after the PCB opening while the test data shows the maximum pressurizer pressure at 8 seconds after PCB opening(Fig.7). However, overall comparison of the pressurizer pressure shows good agreement.

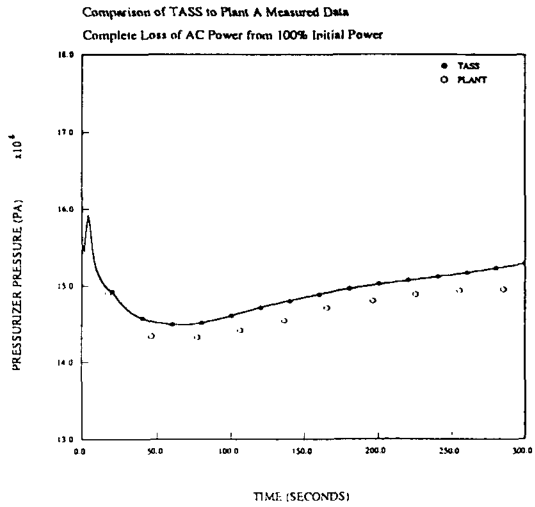

Fig. 5. Pressurizer Pressure

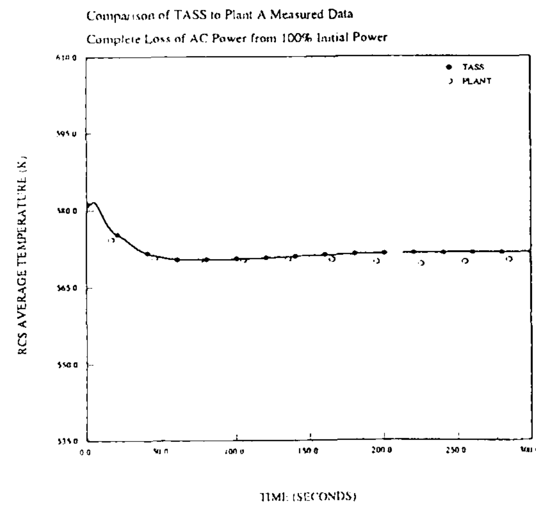

Fig. 6. RCS Average Temperature

Table 2. Sequence of Events for Loss of Electrical Load Test from 100% Initial Power

Time (sec)		Event	Value	
TASS	Test		TASS	Test
0.0	0.0	PCB Open	–	–
6.0	8.0	Maximum Pressurizer Pressure, MPa	16.3	16.4
9.7	10.0	Reactor & Turbine Trip	–	–
12.0	13.0	Maximum Hot Leg Temperature, K	600.4	599.7
16.0	16.0	Maximum Steam Header Pressure, MPa	7.4	7.5
54.0	45.0	Minimum Pressurizer Pressure,MPa	13.5	13.1

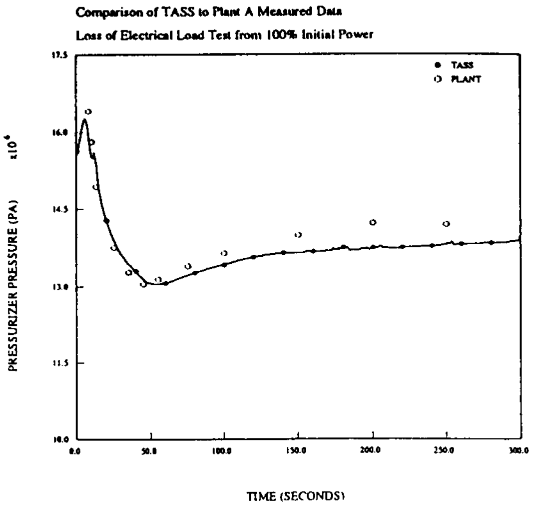

Fig. 7. Pressurizer Pressure

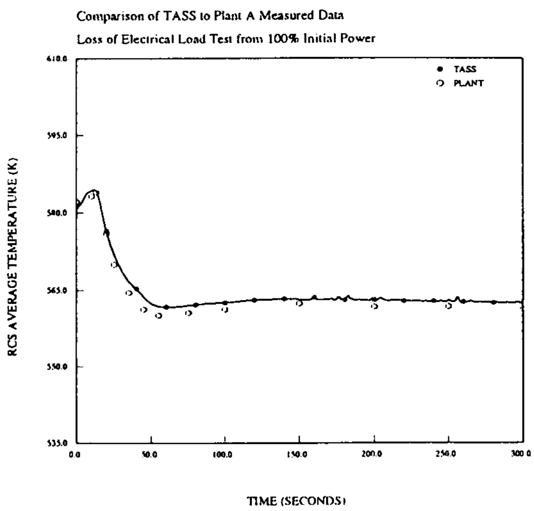

Fig. 8. RCS Average Temperature

3.2. Comparisons to RELAP5/MOD3 Prediction

A set of RELAP5/MOD3 results for the Plant A unit were examined for comparison to the plant reponse as predicted by TASS. The basis for the case selection was to challenge the TASS models in order to identify any significant differences in the results of the two codes. Therefore the most severe design basis events were selected for comparison. Figure 3 and 4 show the nodalization schemes used for the TASS and RELAP5/MOD3 simulations.

Feed Line Break

For Westinghouse type plants, the most limiting feedline rupture is a double-ended rupture of the largest feed line. Thus, a single feed line break case was simulated using the TASS and the RELAP5/MOD3 codes for a double-ended rupture of the largest feed line. The case assumes that a largest break (0.13 M^2) occurs in the feed line to one of the steam generators, downstream of the feedwater check valve. Table 3 shows the sequence of events from both the RELAP5/MOD3 and TASS

Table 3. Sequence of Events for Feed Line Break

Time (sec)		Event	Value	
TASS	RELAP5		TASS	RELAP5
0.0	0.0	Feed line break, M^2	0.13	0.13
5.0	5.0	S/G lo-lo level trip signal is generated, %	17.0	17.0
7.0	8.0	Rods begin to drop	–	–
7.0	8.0	RCP begin to coastdown	–	–
7.0	8.0	Turbine trip and main feedwater is terminated	–	–
65	66	Auxiliary feedwater is delivered, M^3/sec	0.028	0.028
70	80	Low steamline pressure setpoint is reached, MPa	4.135	4.135
85	87	Main steamline isolation	–	–
90	92	HPSI is delivered to each cold leg	–	–
361	341	Pressurizer safety valves open, MPa	17.236	17.236
~2000	~2000	RCS temperature begins to decease	–	–

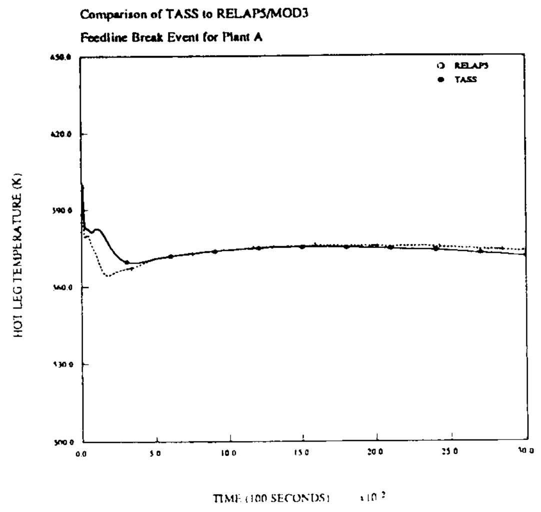

Fig. 9. Hot Leg Temperature-Affected Loop

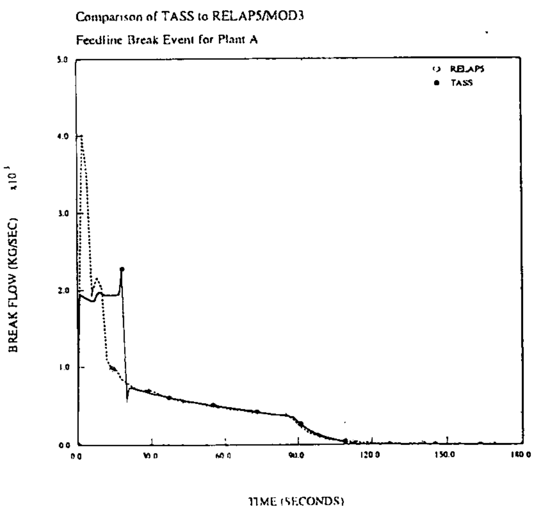

Fig. 10. Break Flow

simulations. Figures 9 and 10 provide comparisons of important parameters as calculated by the TASS and the RELAP5/MOD3 codes. The major concerns of this event are : short term RCS cooldown until the affected steam generator is empty, RCS heatup after the steam generator is empty, MSIV closing time, and the long term cooling capability of the two intact steam generators by auxiliary feedwater flow. During the period of event until steam generator for the broken loop is empty, the RCS temperature calculated by TASS is a little higher than that calculated by RELAP5/MOD3 due to the different liquid mass in the affected steam generator. The rapid decrease in liquid mass in the affected steam generator results in lower heat removal in TASS than RELAP5/MOD3.

Seized Rotor

A single reactor coolant pump rotor seizure can be caused by seizure of the upper or lower thrust-journal bearings. Following seizure of a reactor coolant pump shaft, the core flow rate rapidly decreases to the value which occurs with only two of the reactor coolant pumps in operation. The reduction in core flow with the associated increases in core coolant inlet temperture will reduce the margin to the DNB safety limit and increase the system pressure. For Plant A the event is terminated by the Low Reactor Coolant Flow reactor trip. A single seized rotor event was simulated using the TASS and RELAP5/MOD3 codes. The case assumes that a reactor coolant pump stops instantaneously at the initiation of event. Table 4 shows the sequence of events from both the RELAP5/MOD3 and TASS simulations. Figures 11 and 12 provide comparisons of important parameters as calculated by the TASS and the RELAP5/MOD3 codes. Figure 11 shows a small difference in pressurizer pressure between the two set of the results. The other system parameters show good agreement, especially for the loop mass flow rates which are the key parameters for this transient.

Steam Generator Tube Rupture

The steam generator tube rupture accident is a penetration of the barrier between the reactor coolant system and the main steam system which results from the failure of a steam generator U-tube. Integ-

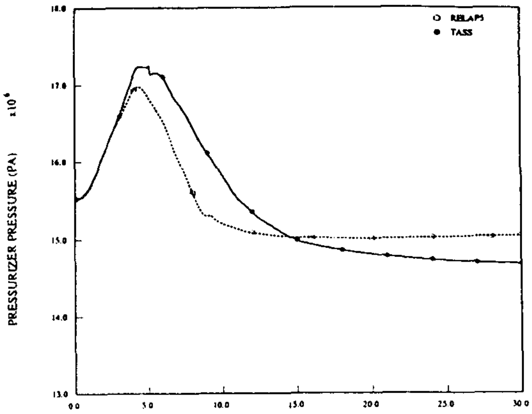

Fig. 11. Pressurizer Pressure

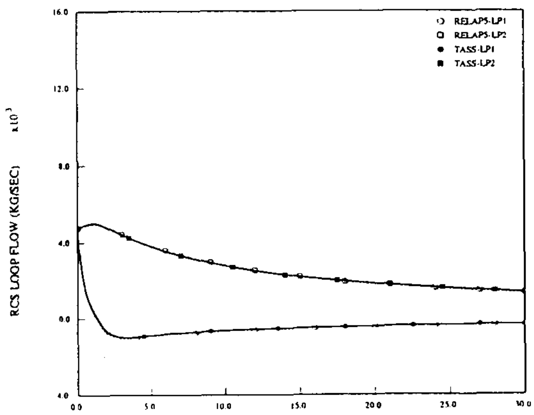

Fig. 12. RCS Loop Flow

Table 4. Sequence of Event for Locked Rotor

Time (sec)		Event	Value	
TASS	RELAP5		TASS	RELAP5
0.0	0.0	Seizure of a single RCP shaft	–	–
0.1	0.1	Low RCS flow Rx. trip signal, %	87.	87.
1.1	1.1	Rods begin to drop	–	–
1.1	1.1	Loss of offfsite power ;	–	–
1.1	1.1	coastdown of remaining RCPs		
5.1	4.3	Turbine trip	–	–
9.5	4.2	Peak pressurizer pressure, MPa	17.25	16.98
		S/G safety valves open, MPa	8.27	8.27

Table 5. Sequence of Events for Steam Generator Tube Rupture

Time (sec)		Event	Value	
TASS	RELAP5		TASS	RELAP5
0.0	0.0	Rupture of one S/G tube, cm^2	1.87	1.87
217	190	PRZ lo-lo pressure trip, MPa	13.513	13.513
219	192	Rods begin to drop	–	–
219	192	Turbine trip	–	–
219	192	Coastdown of RCPs	–	–
229	212	S/G safety valves open, MPa	8.274	8.274
270	263	HPSI actuation	–	–
305	298	Aux. feedwater is delivered	–	–

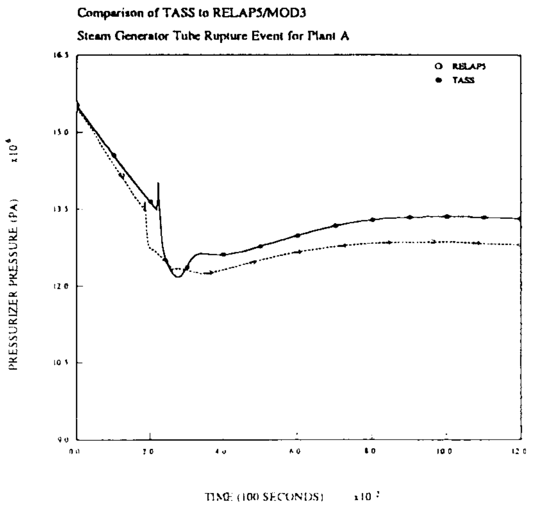

Fig. 13. Pressurizer Pressure

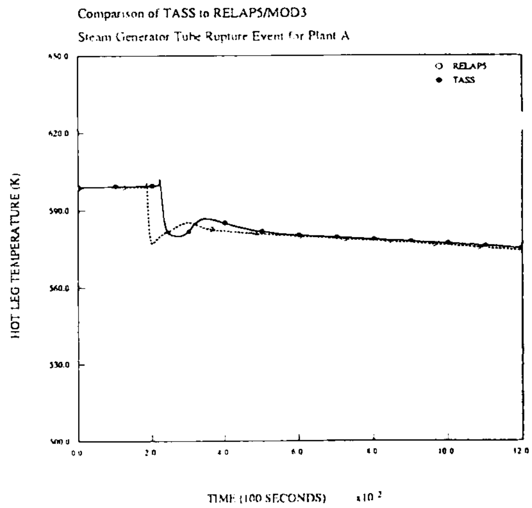

Fig. 14. Hot Leg Temperature-Affected Loop

rity of the barrier between the RCS and main steam system is significant from a radiological release standpoint. A steam generator tube rupture event causes a depressurization of the RCS. A reactor trip is generated by either the over-temperature delta-T trip or the low pressurizer pressure trip. For this analysis, a reactor trip is assumed to occur when the pressurizer pressure reaches the trip setpoint. This is the latest time at which a reactor trip would occur. A single steam generator tube rupture case was simulated using the TASS and RELAP5/MOD3 codes. Table 5 shows the sequence of events from both the RELAP5/MOD3 and TASS simulations.

Figures 13 and 14 provide comparisons of important parameters as calculated by the TASS and the RELAP5/MOD3 codes. Figure 13 traces the pressurizer pressure calculated by TASS and REALP5/MOD3. The rate of depressurization calculated by TASS is slightly slower than that calculated by RELAP5/MOD3 due to the fact that the break flow predicted by RELAP5/MOD3 is slightly higher than that of TASS. For the calculation of critical flow, TASS uses the Henry-Fauske model however RELAP5/MOD3 uses a equation derived from the Bernoulli equation which slightly overpredicts the critical flow in most cases. Since the reactor trip is by the low pressurizer pressure in this case, this overprediction of TASS for the pressurizer pressure resulted in a delay in reactor trip time. Except for the effect of different trip times, the comparisons show good agreement.

Steam Line Break

A single steam line break case was simulated using the RELAP5/MOD3 and the TASS codes. The case assumes that a double ended guillotine break occurs in the main steam line inside the containment building from the zero power initial condition. This case does not assume a loss of AC power so that the reactor coolant pumps continue to operate throughout the event. Table 6 shows the sequence of events from both the RELAP5/MOD3 and TASS simulations. Figures 15 and 18 provide comparisons of important parameters calculated by TASS and the RELAP5/MOD3 codes. As shown in Figures 16 and 17, RELAP5/MOD3 results show that the break flow stops at around 130 seconds even though the liquid mass of the affected steam generator is still present due to the stagnant region in moisture separator. In contrast, TASS results show that the break flow exist until the affected steam generator is totally empty. Due to this difference in the break flow, the cold leg temperature of the affected loop predicted by TASS increases more slowly than that of RELAP5/MOD3 at about 100 seconds at which the neutron power starts to increase. The hot leg temperature of the af-

Table 6. Sequence of Events for Steam Line Break

Time (sec)		Event	Value	
TASS	RELAP5		TASS	RELAP5
0.0	0.0	Main steamline break, M^2	1.13	0.13
10	10	Main steamline isolation	–	–
10	10	Aux. feedwater is delivered	–	–
21.4	23	Low steamline press. sig., MPa	4.14	4.14
33.4	35	HPSI system is actuated	–	–

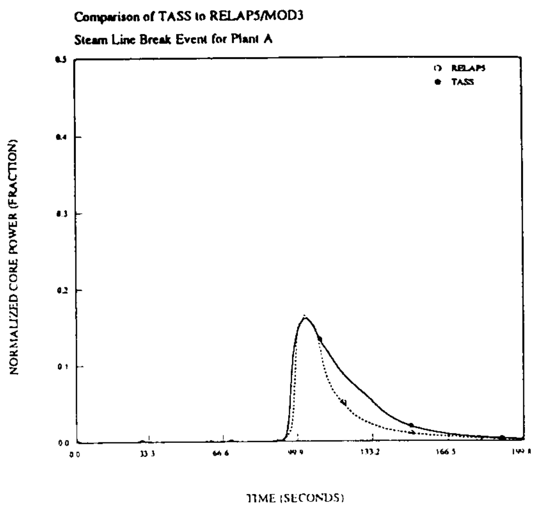

Fig. 15. Normalized Core Power

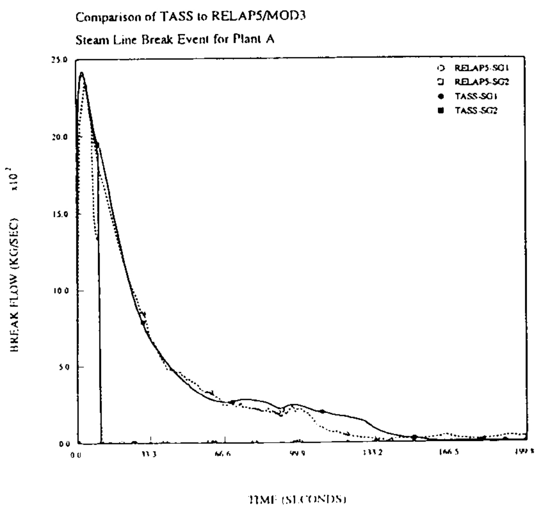

Fig. 16. Steam Generator Steam Flow

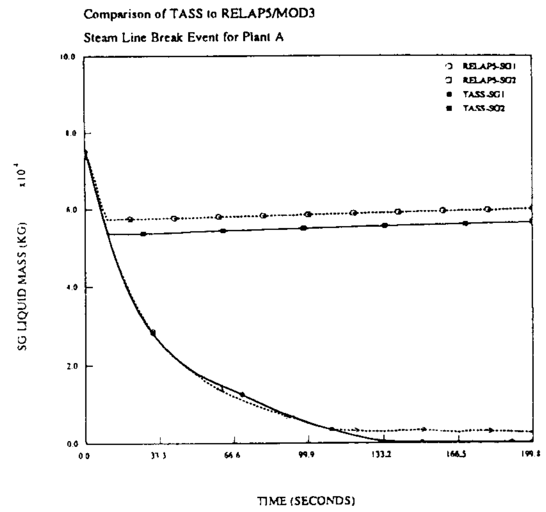

Fig. 17. Steam Generator Liquid Mass

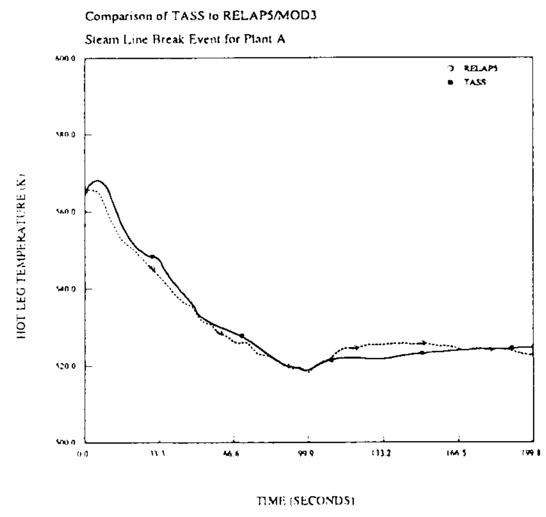

Fig. 18. Hot Leg Temperature-Affected Loop

fected loop predicted by TASS in Figure 18 increases for a while at the beginning of event due to the outsurge flow from the pressurizer(Fig.3). However, as shown in Figure 4, the hot leg temperature calculated by RELAP5/MOD3 does not increase due to the fact that the RELAP5/MOD3 has several control volumes for the hot leg and the compared hot leg temperature was taken from the node connected just after the core outlet nozzle without the pressurizer surge line connection.

4. Conclusions

The cases demonstrate that the TASS models can accurately predict Westinghouse type PWR plant responses to upset conditions. The verification effort supports the following conclusions :

- TASS has a numerically stable solution methodology with a proper conservation of mass, momentum, and energy.
- TASS reproduces measured plant behavior for a range of different events.
- TASS satisfactorily reproduces the plant behavior as predicted by RELAP5/MOD3.
- TASS is basically a best estimate code. Appropriate conservatism of licensing analyses of Non-LOCA design basis events can be introduced primarily through code inputs.

TASS is shown here to be capable of predicting system responses for Westinghouse type PWR Non-LOCA design basis events. Thus, TASS can be effectively used as a predictive tool for licensing analysis of Non-LOCA events.

References

1. Uwe Laudi, "Code Manual on NLOOP," September, 1987.
2. "CESEC Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," Enclosure 1-P to LD-82-001, January 6, 1982.
3. EG&G, "RELAP5/MOD3 Code Manual, Vol. I, II, III, IV, V," NUREG/CR-5535, EGG-2596, June, 1990.
4. Wallis, Gramham B., "One-Dimensional Two-Phase Flow," McGraw Hill, Inc., 1969.
5. Tom, J.R.S., "Prediction of Pressure Drop During Forced Circulation Boiling of Water," International Journal of Heat and Mass Transfer, Volume 7, pp. 709−724, 1964.

6. Martinelli, R.C. and Nelson, D.B., "Prediction of Pressure Drop During Forced Circulation of Boiling Water," Transactions of the ASME, August, 1948.
7. Bird, R.M., Stewart, W.E., and Lightfoot, E.N., "Transport Phenomena," Jhon Wiley and Sons, Inc., 1960.
8. Kocumustafaogullari, G., "Thermo-Fluid Dynamics of Separated Two-Phase Flow," Ph. D Thesis, School of Mechanical Engineering, Georgia Institute of Technology, Atlanta, Georgia, December, 1971.
9. Ishii, M., "Thermal-Fluid Dynamics Theory of Two-Phase Flow," Collection de la Direction des Etudes et Recherches D'Electricite de France, Eyrolles, Paris, 1975.