8. Analysis of Chinshan-1 Power Test Data with the BNL Plant Analyzer, *Ge-Ping Yu, Bau-Shei Pei, Chin Pan (Natl Tsing-Hua Univ–Taiwan)*

The Brookhaven National Laboratory (BNL) plant analyzer, also called the high-speed interactive plant analyzer (HIPA), was developed to reduce the time and cost associated with nuclear power plant transient simulation. The BNL analyzer proved to be quite accurate for the analysis of boiling water reactor (BWR) designs and compared well with the results of TRAC-B01, RAMONA-3B, and RELAP-5 calculations.[1] Modification such as feedwater pump control system, SRV logics, input parameter, and form loss coefficient were made to HIPA's BWR/4 models to accommodate the BWR/4 plant-specific design features of Chinshan-1. This paper presents comparisons of HIPA prediction to Chinshan-1 data for two important power tests.

The plant analyzer (HIPA) of the Taiwan Power Company consists of two AD100 peripheral processors controlled by a VAX 8250 host computer. The analyzer can be operated by a 32-channel analog control panel and a Tektronix oscilloscope or remotely by an IBM-PC with a color monitor, and a Micro VAX work station with graphic display. The purpose of the host computer is to compile the software (written in ADSIM simulation language), assemble the program into a task, execute it, and to provide pretabulated function data to expedite iterative procedures. Each AD100 consists of seven task-specific miniprocessors with a pipeline architecture that allows very high simulation speeds.

Fission power is simulated with a point-kinetic model that accounts for the effects of Doppler, void reactivity, moderator temperature, control rod movement, boron injection, and accidental reactivity insertion. The nonequilibrium, nonhomogeneous two-phase thermal-hydraulics in the reactor vessel is modeled using a four-equation drift flux mixture model, and the steam-line dynamics is modeled with an adiabatic compressible flow model. In addition, the entire balance of plant, the plant protection system, and the pressure suppression pool are modeled, as are the pressure regulator, feedwater flow controller, and recirculation flow controller.[1]

Next, the result of HIPA calculations for an 83% loss-of-feedwater heating transient and 98% feedwater pump trip transient are presented. These results are compared with plant data from the power test conducted at Chinshan-1. Steady-state initialization of the two transients was established by adjusting load set point, core flow rate, and dome pressure from the 100% power initial conditions. The initial conditions used in the HIPA calculation were very close to the test data.

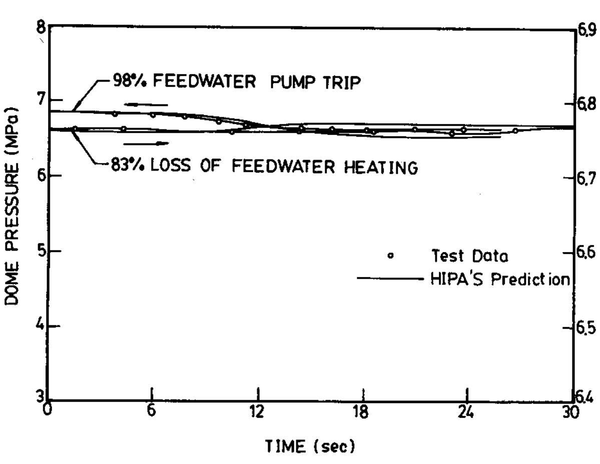

Fig. 1. Comparison of the dome pressure responses from the test data and HIPA's prediction.

LOSS-OF-FEEDWATER HEATING TEST

The purpose of this test is to test whether variation in feedwater temperature can cause normal reactor response. Loss of feedwater heating may be caused by closure of the steam injection line of the feedwater heater, or bypass of the feedwater heater. The steam supply to the heater was initially closed during the test. The bypass valve was opened to avoid feedwater passing the heater by. A heater isolation valve was closed, and the feedwater temperature gradually decreased. A decrease in the inlet temperature of the core flow resulted in a rise of the reactor power. Finally, the system parameters of the core gradually reached another steady-state condition. Test data of the reactor power, dome pressure, core flow rate, and feedwater flow rate were compared with the predictions of the HIPA model. It predicts most of the important plant responses within acceptable ranges. The difference in the reactor power rising rate from prediction to test data is because of the lack of start time of the transient and the drop rate of the feedwater temperature.

FEEDWATER PUMP TRIP TEST

The purpose of this test is to test whether the reactor core can respond normally to avoid a low-water-level reactor trip when a normal feedwater pump trips. Upon manual trip of a feedwater pump, the decrease of the feedwater flow results in a rapid drop in the reactor power and pressure. Finally the major system parameters such as power and core flow rate reach another steady state. Core water level fluctuates because of recirculation pump run back, but still keeps above the low-water-trip setpoint. Variations in feedwater flow rate are modeled by time-dependent boundary conditions that simulate this transient. Test data of the reactor dome pressure, core flow rate, steam line flow rate, and water level are compared with the HIPA model predictions. It predicts similar trends with acceptable results. The timing of major events in the test sequence was simulated well by the prediction. Overprediction of the steam line flow rate also reflects on the core water level. The dome pressure response from the test data and HIPA results can be seen in Fig. 1.

In conclusion, it appears that HIPA is capable of predicting the transient behavior of Chinshan-1. The HIPA Chinshan model will be further improved in the future.

1. W. WULFF, H. S. CHENG, S. V. LEKACH, A. N. MALLEN, "The BWR Plant Analyzer, Final Report," NUREG/CR-3943, BNL-NUREG-51812, Brookhaven National Lab. (Aug. 1984).

9. GVTRAN-PC: A Fast Steam Generator Transient Simulator, *Horacio Nakata (CNEN–Brazil)*

INTRODUCTION

An accurate and inexpensive analysis capability is one of the most desirable qualities sought in a plant analytical tool, because licensing procedures and operations management require a great number of detailed plant performance evaluations under hypothetical accident sequences and initiating conditions. Nevertheless, because of increased analytical complexity, the existing codes are not fast enough to permit inexpensive, thorough examination of the most probable sequences in an accident analysis. This work tries to contribute to the development of a faster and better plant simulator by developing a fast and accurate steam generator simulator.

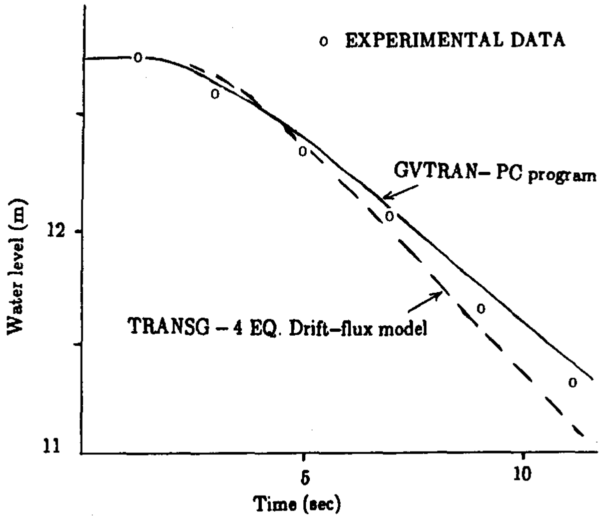

Fig. 1. Feed chamber water level in D.C. Cook Unit 1 transient.

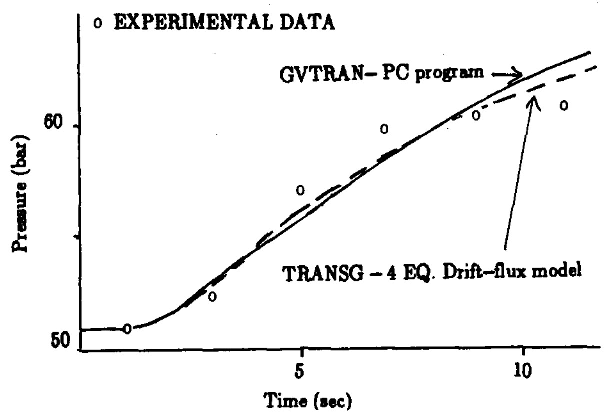

Fig. 2. Steam chamber pressure in D.C. Cook Unit 1 transient.

METHODOLOGY

The methodology developed and implemented in the GVTRAN-PC program (a U-tube steam generator simulator for the microcomputer environment) is summarized here. The primary circuit of the steam generator is simply treated with the mass and energy conservation equations[2] by assuming a constant pressure in the primary tube region. In the secondary circuit, mass and energy conservation are imposed with properly chosen two-phase correlations. The momentum conservation equation was only balanced in the secondary region, and its solution through a quasi-static approach resulted in a significant saving in computation time while retaining good accuracy in the water-level representation.

The simplicity of the GVTRAN-PC program is also warranted by a reduced number of control volumes both in the primary and in the secondary region. The most important effect in the secondary region that must be carefully tracked is the level at which the bulk boiling begins, because the void fraction changes quite abruptly at that point and, consequently, strongly influences the heat transfer rate from the primary to the secondary fluid. The program tracks the boundary of the bulk boiling region with the moving boundary approach by defining only two control volumes in the secondary tube zone and, correspondingly, four control volumes in the primary tube zone.

RESULTS

The GVTRAN program was tested against the actual transient measurements recorded during startup testing at Donald C. Cook Nuclear Station Unit 1 (Ref. 1). The transient consisted of a plant turbine trip from 100% of the rated power. The turbine trip was manually initiated while the plant was operating at full power. After that, the main steam turbine stop valve was automatically closed and, as a consequence, caused the reactor and turbine driven feed pumps to trip.

The feed chamber water level and the steam dome pressure calculated with the GVTRAN-PC program is plotted in Figs. 1 and 2, respectively, together with the experimental data and results calculated with the TRANSG program.[1] The comparison of the results is quite satisfactory in spite of the simplified model employed in the GVTRAN-PC program compared with the TRANSG four-equation drift-flux model. Both water level and pressure deviations are well within the acceptable range considering the margin of uncertainties in the experimental data.

The accuracy of the results was not compromised even with the use of fewer control volumes (only two volumes in the secondary tube region) because the moving boundary approach was adopted at the boundary of the bulk boiling region, thus permitting a fairly good representation of the time variation of the two-phase region height without recourse to a fine-mesh representation.

In conclusion, the methodology employed in the GVTRAN-PC program possesses a combination of features, all of them highly desired for a nuclear plant simulator: coding simplicity for a quick intermachine transfer, computation velocity compatible with on-line monitoring, and accuracy acceptable for a typical transient analysis.

1. J. C. LEE, A. Z. AKCASU, J. J. DUDERSTADT, G. J. Van TUYLE, R. FORTINO, "Simplified Models for Transient Analysis of Nuclear Steam Generators," EPRI NP-1772, Electric Power Research Institute (Apr. 1981).

2. R. B. BIRD, W. E. STEWART, E. N. LIGHTFOOT, *Transport Phenomena*, John Wiley & Sons, New York (1960).