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Application of Flow Network Models of SINDA/FLUINT^{TM} to a Nuclear Power Plant System Thermal Hydraulic Code

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Abstract

In order to enhance the dynamic and interactive simulation capability of a system thermal hydraulic code for nuclear power plant, applicability of flow network models in SINDA/FLUINT^{TM} has been tested by modeling feedwater system and coupling to DSNP which is one of a system thermal hydraulic simulation code for a pressurized heavy water reactor. The feedwater system is selected since it is one of the most important balance of plant systems with a potential to greatly affect the behavior of nuclear steam supply system. The flow network model of this feedwater system consists of condenser, condensate pumps, low and high pressure heaters, deaerator, feedwater pumps, and control valves. This complicated flow network is modeled and coupled to DSNP and it is tested for several normal and abnormal transient conditions such turbine load maneuvering, turbine trip, and loss of class IV power. The results show reasonable behavior of the coupled code and also gives a good dynamic and interactive simulation capabilities for the several mild transient conditions. It has been found that coupling system thermal hydraulic code with a flow network code is a proper way of upgrading simulation capability of DSNP to mature nuclear plant analyzer (NPA).

1. Introduction

Nuclear power plant consists of many complex flow systems which, if malfunctioning, may put other major plant system such as nuclear steam supply system (NSSS) into anomaly. Many system thermal hydraulic codes such as RELAP5 focus on the detailed and mechanistic simulation of NSSS behavior. However, nuclear plant analyzer (NPA), a computerized tool for safety analysis and engineering, is generally focused not only simulate and evaluate the behavior of NSSS but also other complex engineering systems. The NPA is not limited to nuclear plant analysis, but can be also used to analyze data of any origin using appropriate simulation codes, and, after coupled with system thermal hydraulic codes, an interactive engineering simulator can be produced. Therefore, in order for a simulation system to be an NPA, coupling of system thermal hydraulic code with other engineering simulation code is inevitable for dynamic and interactive capability.

DSNP[1] is also one of the nuclear plant system thermal hydraulic code mainly simulating NSSS . Even though DSNP is originally developed for fast breeder reactor thermal hydraulic simulation, there are other versions of pressurized light water reactor (PLWR) and of pressurized heavy water reactor (PHWR). Especially, this PHWR version of DSNP is supposed to be a base code for future interactive engineering simulator at IAE[2], and in order to obtain this final goal, as mentioned earlier, the simulation scope should be enhanced further to include special engineering systems. In this case, coupling with other simulation codes is a general trend. In this aspect, simulation capability of balance of plant systems is important area that should be provided to the major system thermal hydraulic code, such as DSNP.

The purpose of this study is thus to enhance the premature NPA being developed at IAE by dynamically simulating balance of plant responses in the power plant operational and upset conditions thus providing a flow and pressure boundary condition to the steam generator models in DSNP

A flow network simulation code, SINDA/FLUINT™[3] has been used to model the feedwater train from main condenser to the inlet nozzle of the steam generator of 600 MWe PHWR. The feedwater system is selected since it is one of the most important balance of plant systems with a potential to greatly affect the behavior of NSSS. The main condenser and feedwater system has a feedback effects to the NSSS during normal and transient operation conditions. It includes main and auxiliary condensate extraction pumps, regenerative feedwater heaters and deaerator, main and auxiliary steam generator feedwater pumps, piping and valves for complete system, with rigid and elastic supporters and the necessary structural steel and necessary instrumentation and accessories. Under the appropriate simplifications, model for these system has been developed using SINDA/FLUINT™ coupled to DSNP.

2. Model Description

SINDA/FLUINT™ is a comprehensive finite-difference, lumped parameter (circuit or network analogy) tool for analyzing complex thermal/fluid systems. It is NASA standard analyzer for thermal control system. SINDA simulates transient or steady-state energy storage and flow in a system modeled as a "thermal circuit". And FLUINT simulates transient or steady state mass and energy storage and flow in a system modeled as a "fluid circuit". Although FUINT can be used by itself for purely hydrodynamic analyses such as pump matching, manifolding(flow distribution), or water hammer, it can also be combined with SINDA to simulate combined thermal/hydraulic systems. In this paper SINDA/FLUINT means the FLUINT combined with SINDA.

Generally for the fluid circuit, any components should be modeled using lumps (junctions or tanks) connected by paths (tubes or connectors). Tanks have a control volume which may grow or shrink with wall compliance C_W. Whereas the junctions have no volume and therefore no storage capacity. Paths conserve momentum and they are assumed to be one-dimensional, hence described by a single mass flow rate. Tubes cannot change the flow rate instantaneously. Rather, the fluid in the tubes undergoes acceleration or deceleration. But connectors can change flow rate instantaneously, therefore connectors are used for simulate pumps, valves, etc. The governing differential equations of them are as follows.

For junctions,

$$Mass:\ \sum\vec{W}=0$$

$$Energy:\sum h\cdot\vec{W}+Q=0$$

For tanks,

$$Mass:\ \sum\vec{W}=\frac{dM}{dt}$$

$$Energy:\sum h\cdot\vec{W}+Q-P(\frac{dV}{dt}+\frac{dP}{dt}\cdot V\cdot C_W)=\frac{dU}{dt}$$

For tubes (momentum),

$$\frac{dW}{dt}=\frac{A}{L}\left(P_{up}-P_{down}+HC+FC\cdot W\cdot|W|^{FPOW}+A\cdot W^{2}-\frac{K\cdot W\cdot|W|}{2\rho\cdot A^{2}}\right)$$

For connectors, (time step n to n+1)

$$\Delta W^{n+1}=GK^{n}(\Delta P_{i}^{n+1}-\Delta P_{j}^{n+1})+HK^{n}$$

Where,

W	: Mass flow rate,	A	: Flow area,
L	: Tube length,	FC	: Irrecoverable loss coefficient,
P_{up}	: Pressure of upward lump,	P_{down}	: Pressure of downward lump,
FK	: Additional K-factor loss,	HC	: Recoverable loss coefficient,
FPOW	: 0 (for laminar) to 1(for fully turbulent),		
GK	: Partial derivative of the flow rate w.r.t. pressure drop changes,		
HK	: Flow rate offset,	Q	: Heat source or sink of lump,
C_W	: Tank wall compliance,	V	: Volume of tank,
h	: Enthalpy of lump,	U	: Internal energy of lump

Using the components of SINDA/FLUINT™ described above, the condenser and feedwater system is simplified by 31 fluid lumps and 41 fluid paths(Fig.1, Fig.2). Three LP heaters, deaerator, and 2 HP heaters are modeled by thanks, and 2 main condensate pumps, 3 main feedwater pumps, and several control valves are described by connectors. The long pipes and any kinds of friction losses are modeled by tubes. Condenser gives the boundary condition to the entire system, therefore it is modeled by a plenum which is a type of lump and has infinite volume, so the pressure and temperature can be set as constant values. The heat which is given by the extraction steam from turbine to feedwater heaters, is calculated according to the turbine power using interpolation of turbine heat balance data table of the Wolsong nuclear power plant.[4]

The control logic of Wolsong nuclear power plant is applied to this model. According to this logic the feedwater control valve works by the BLC(Boiler Level Control) program, and it is shown as follows.

$$Valve\ Lift=K_{1}\times LE+\sum K_{2}\times LE+K_{3}\times PB-K_{4}\times\Delta\frac{PB}{\Delta t}+K_{5}\times(W_{s}-W_{f})$$

Where,

LE	: Level Error,	K_1	: Proportional Gain,
ΣK_2	: Integral Gain,	$K_3 \times PB$	: Bias,
W_s	: Steam Flowrate,	W_f	: Feedwater flowrate,

$K_4 \times D\frac{PB}{Dt}$: Swell term

Through the feedwater control valves the feedwater system developed in this work is connected to the NPA simulation system.

3. Results and Discussion

The steady state of this system is adjusted using the data of 100% full power of Wolsong nuclear power plant. The transient analysis has been performed from this steady state which shows 100% normal full power condition. The transient analysis cases are selected according to the recommendation of SOPHT code which is a comprehensive thermal hydraulic analysis package and is developed for simulation of various normal and abnormal operation conditions of a CANDU nuclear power plant.[5] Considering this recommendation and the capability of NPA system in IAE, the transient analysis cases are limited to normal(Level A) and upset or operational transient(Level B) conditions. For the evaluation of feedwater system, turbine trip and loss of class IV power transients, which are closely related to the feedwater system, are selected.

In the case of turbine trip transient, turbine trip occurs at t = 10 sec from the normal 100% full power condition. It means there's no extraction steam from turbine to HP, LP heaters and deaerator, so the pressure and temperature decreases. Fig. 3 shows the trend of deaerator pressure.

Loss of class IV power transient, which means the electrical power supplied to some plant components disappears, is simulated by the sudden trip of the all feedwater pumps. At t = 0, 2 main feedwater pumps and 1 auxiliary pump is all tripped. Fig. 4 shows the feedwater flow rate to steam generator. All feedwater flow goes down to zero only in 2 secs. Fig. 5 shows the level of the deaerator storage tank. The level increases fast to 3810mm, then it stops. Because at the level of 3810, the "very high" level alarm initiated, then the extraction steam line and condensate flow line to deaerator are isolated. Therefore there's no further increase above the level of 3810mm.

The feedwater system is greatly affected by turbine load. Therefore the turbine load variation test for the feedwater system model is very important to know the limit and capability of the model. Fig.6 shows the turbine power trend for the analysis of turbine load variation. At first the turbine power decreases to about 60% and then increases again to 100%. Fig.7 shows the trend of the heat transferred by extraction steam from turbine. Fig.8 and 9 shows the pressure of the deaerator and feedwater flow rate respectively. The overshoot of the feedwater flow is mainly due to the level error of the steam generator. The level error increases the feedwater control valve lift signal as in Fig. 10, and it is the main portion of the valve control signal. The present model of steam generator is based on the homogeneous equilibrium 2 phase flow model, it is not so good an assumption for the fast transient of the 2 phase flow phenomena. The instability of flowrate is due to this assumption, and it should be changed some more detailed one in the further study.

4. Conclusion

The flow network of CANDU nuclear power plant feedwater system using SINDA/FLUINT™ code has been developed to enhance the dynamical and interactive simulation capability of IAE nuclear plant analyzer. The result has shown reasonable response for the most cases of mild operational transient(Level B), such as turbine trip and loss of class IV power. Although some cases show unstable results, it is not due to the coupling of 2 codes, but mostly due to the inherent problem of 2 phase flow model in NPA. Therefore the coupling system thermal hydraulic codes with a flow network code is a proper way of the improvement the simulation capability of NPA.

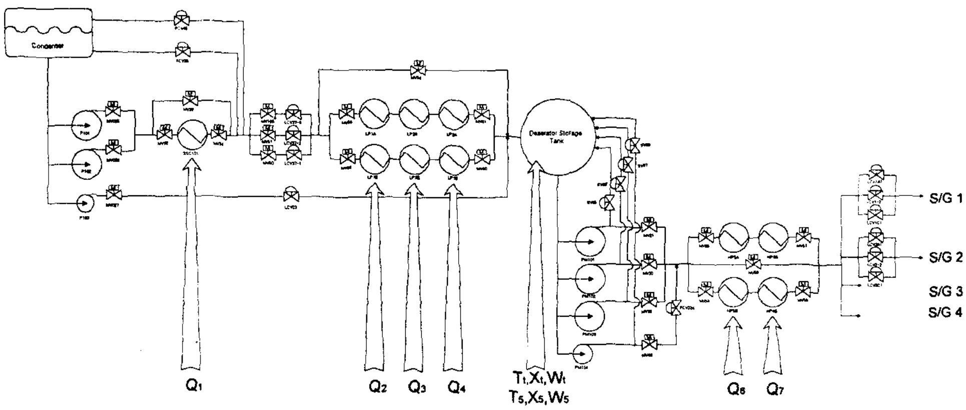

Fig 1. Feedwater System for the CANDU 6 Power Plant

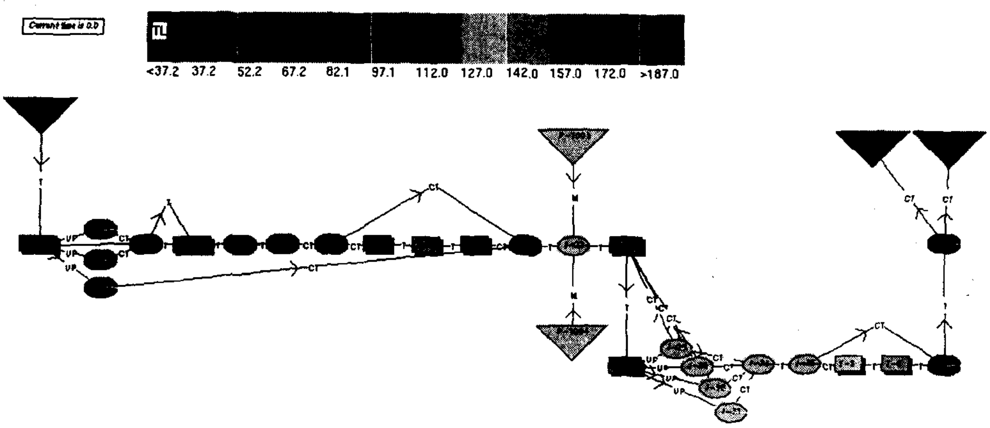

Fig 2. Schematic diagram of feedwater system modeled by SINDA/FLUINT

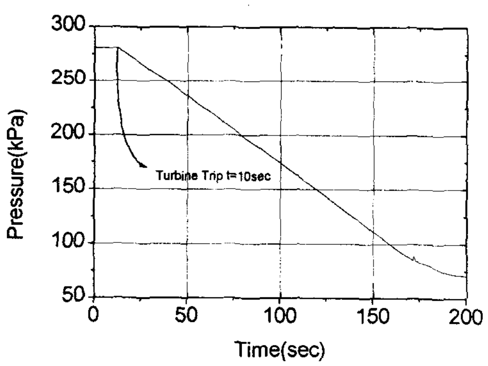

Fig.3 Deaerator pressure(turbine trip)

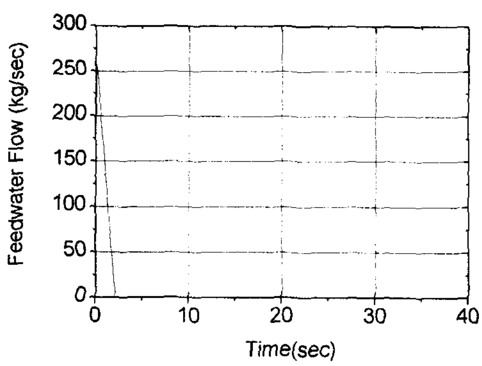

Fig.4 Feedwater flow (Loss of Class IV)

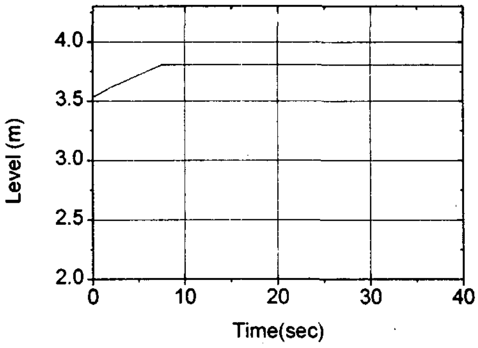

Fig.5 Deaerator tank level (Loss of Class IV)

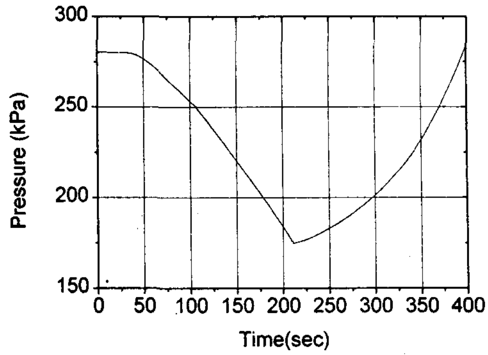

Fig.8 Deaerator tank pressure

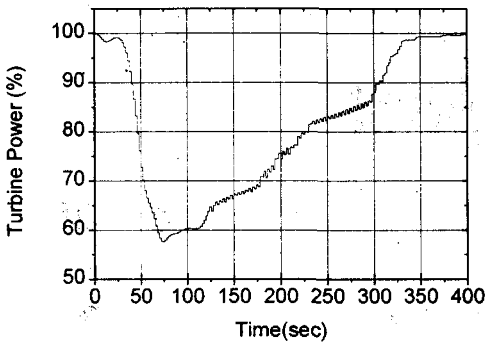

Fig.6 turbine load variation

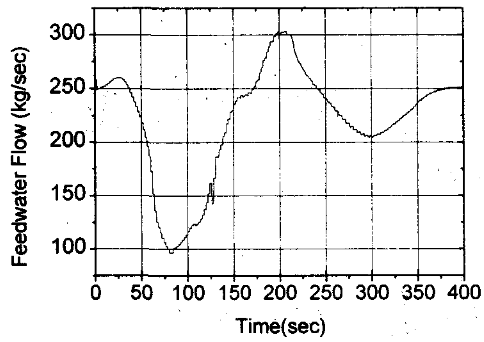

Fig.9 Feedwater flow rate

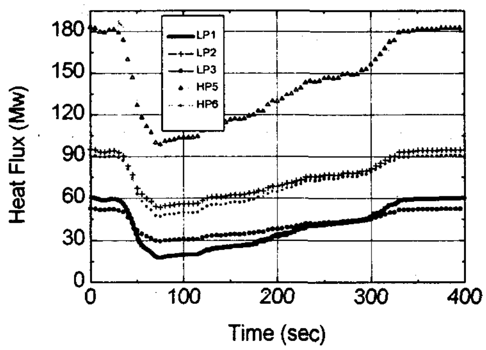

Fig.7 Heat transferred by extraction steam

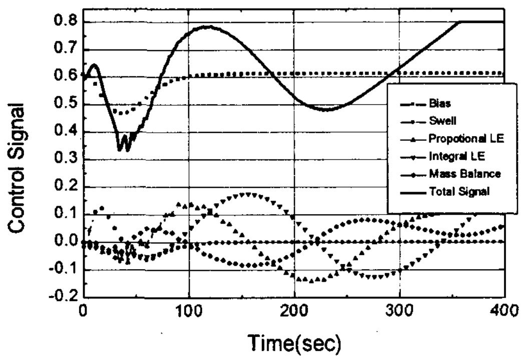

Fig.10 Feedwater control vlave control signal

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[1] J. Zhang and D.A. Meneley, "DSNP: Dynamic Simulator for Nuclear Power Plant - CANDU", 1993

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[3] Cullimore and Ring Technologies, Inc. "User's Manual for SINDA/FLUINT Ver.3.1", 1995

[4] 월성 원자력 2,3,4 호기 최종 안전성 분석 보고서, 제 6 권

[5] Primary Heat Transport System Transient Analysis, Wolsong NPP 2,3,4, 86-33100-AR-001, AECL