

**NUCLEAR REACTOR PROCESS SYSTEMS:  
THERMAL HYDRAULIC ANALYSIS**

**prepared by: Wm. J. Garland, Professor  
Department of Engineering Physics  
McMaster University  
Hamilton, Ontario  
Canada  
February 1998**

for the Thailand Initiative

# TABLE OF CONTENTS:

FOREWORD .....	v	
GLOSSARY OF ABBREVIATIONS AND ACRONYMS .....	vii	
Chapter 1	Course Overview .....	1-1
1.1	Introduction .....	1-1
1.2	Learning Outcomes .....	1-2
1.3	The Course Layout .....	1-3
Chapter 2	Basic Equations for Thermalhydraulic Systems Analysis .....	2-1
2.1	Introduction .....	2-1
2.2	Conservation .....	2-4
2.3	Conservation of Mass .....	2-6
2.4	Conservation of Momentum .....	2-8
2.5	Conservation of Energy .....	2-10
2.6	The Equation of State .....	2-15
2.7	Empirical Correlations .....	2-16
2.8	Solution Overview .....	2-16
2.9	Exercises .....	2-18
Chapter 3	Nodalization .....	3-1
3.1	Introduction .....	3-1
3.2	The Node-Link Concept .....	3-2
3.3	Nodal Diffusion .....	3-3
3.4	Examples .....	3-4
3.5	Matrix Notation .....	3-5
3.6	Exercises .....	3-7
Chapter 4	Equation of State .....	4-1
4.1	Introduction .....	4-1
4.2	Thermodynamic Properties .....	4-2
4.3	The Iterative Method .....	4-3
4.4	The Rate Method .....	4-5
4.5	H <sub>2</sub> O Property Fits .....	4-9
4.6	Exercises .....	4-11
Chapter 5	The Rate Form of the Equation of State .....	5-1
5.1	Introduction .....	5-1
5.2	The Rate Form .....	5-2
5.3	Numerical Investigations: a Simple Case .....	5-3
5.4	Numerical Investigations: a Practical Case .....	5-8
5.5	Discussion And Conclusion .....	5-11
5.6	Exercises .....	5-12
Chapter 6	Thermalhydraulic Network Simulation .....	6-1

6.1	Introduction	6-1
6.2	Porsching's Method	6-2
6.3	Derivation of FIBS	6-2
6.4	Special Cases	6-8
6.5	Programming Notes	6-9
6.6	Conclusion	6-10
6.7	Exercises	6-11
Chapter 7	Empirical Correlations	7-1
7.1	Introduction	7-1
7.2	Empirical Correlations	7-1
7.3	Two Phase Flow Void Correlations	7-2
7.4	Friction	7-3
7.5	Heat Transfer Coefficients	7-4
7.6	Thermodynamic Properties	7-4
7.7	Flow Regime Maps	7-5
7.8	Special Component Data	7-5
7.9	Exercises	7-8
Chapter 8	On Design Tools	8-1
8.1	Introduction	8-1
8.2	The Model's Tenuous Link to Reality	8-1
8.3	Documentation, Verification and Validation	8-2
8.4	Design Tools	8-2
8.5	Notes on Steam Generator Modelling	8-10
References		R-1

## LISTS:

<b>Table 4.1</b>	Summary of the F functions for the rate form of the equation of state	4-12
<b>Table 4.2</b>	Summary of the G functions for the rate form of the equation of state	4-13
<b>Table 5.1</b>	Figure of Merit Comparisons of the Normal and Rate Forms of the Equation of State for Various Convergence Criteria (Simple Case).	5-13
<b>Figure 1.1</b>	Concept map for the course	1-4
<b>Figure 1.2</b>	The cognitive domain.	1-5
<b>Figure 2.1</b>	Derivation path.	2-19
<b>Figure 2.2</b>	The four cornerstone single phase flow equations and the flow of information between them.	2-20
<b>Figure 2.3</b>	The four cornerstone equations for the full two-fluid model.	2-21
<b>Figure 2.4</b>	The four cornerstone equations for the two-fluid model with equal pressure of the two phases.	2-21
<b>Figure 2.5</b>	Simple pool-type research reactor.	2-22
<b>Figure 3.1</b>	A general and connecting links.	3-8
<b>Figure 3.2</b>	Two connected nodes.	3-8
<b>Figure 3.3</b>	Node-link setup for a simple pipe.	3-9

<b>Figure 3.4</b> Node-link setup for an area change in a pipe. . . . .	3-10
<b>Figure 3.5</b> Illustration of convection and diffusion. . . . .	3-10
<b>Figure 3.6</b> Transmission of a step change using the plug flow model and a feeder model with skewing due to differences in transit times. . . . .	3-11
<b>Figure 3.7</b> Transmission of a step change using the Plug Flow model and the Mixing Tank model (1 to 50 tanks). . . . .	3-11
<b>Figure 3.8</b> Simple Tee junction. . . . .	3-12
<b>Figure 3.9</b> Simple Y junction. . . . .	3-12
<b>Figure 3.10</b> Sample node-link connections for a header. . . . .	3-12
<b>Figure 3.11</b> Node-link diagram: 1/4 circuit Darlington G.S. . . . .	3-13
<b>Figure 3.12</b> Node-link diagram: Full circuit Darlington G.S. . . . .	3-14
<b>Figure 3.13</b> 4 node - 5 link diagram. . . . .	3-15
<b>Figure 4.1</b> P-v-T surface for water. . . . .	4-13
<b>Figure 4.2</b> Numerical search for P given $\rho$ and h for a two-phase mixture. . . . .	4-13
<b>Figure 4.3</b> Error correction scheme for pressure in two-phase. . . . .	4-14
<b>Figure 4.4</b> Density vs. pressure at various temperatures in subcooled water. . . . .	4-14
<b>Figure 4.5</b> Basis for curve fitting in the subcooled region. . . . .	4-15
<b>Figure 5.1</b> Simple 2-node, 1-link system. . . . .	5-13
<b>Figure 5.2</b> Program flow diagram for the normal method. . . . .	5-14
<b>Figure 5.3</b> Program flow diagram for the rate method. . . . .	5-15
<b>Figure 5.4</b> Number of iterations per pressure routine call for the normal method with a time step of 0.01 seconds and a pressure error tolerance of 0.001 of full scale (10 mPa). . . . .	5-16
<b>Figure 5.5</b> Integrated flow error for the rate method and the normal method for various fixed time steps, convergence tolerances and adjustment factors. . . . .	5-16
<b>Figure 5.6</b> Schematic of control volumes in the pressurizer. . . . .	5-17
<b>Figure 5.7</b> Flow vs. time for the implicit forms of the normal and rate method. . . . .	5-17
<b>Figure 5.8</b> Pressurizer's pressure transient for the normal method with error tolerance of 0.2%. . . . .	5-18
<b>Figure 5.9</b> Pressurizer's pressure transient for the rate method. . . . .	5-18
<b>Figure 5.10</b> Averaged number of iterations per pressure routine call for the normal method in simulating pressurizer problem. . . . .	5-19
<b>Figure 6.1</b> The simple 4 node - 5 link example. . . . .	6-12
<b>Figure 6.2</b> The four cornertone equations for thermalhydraulic system simulation and the flow of information between them. . . . .	6-12
<b>Figure 7.1</b> Void fraction versus quality for mixtures of saturated liquid and vapour water. . . . .	7-9
<b>Figure 7.2</b> $\alpha$ versus x and $\partial\alpha/\partial x$ . . . . .	7-10
<b>Figure 7.3</b> Flow regimes in horizontal pipes. . . . .	7-11
<b>Figure 7.4</b> Typical power distributions. . . . .	7-12
<b>Figure 7.5</b> complete pump characteristics, double-suction pump, speed = 1800 rpm. . . . .	7-13
<b>Figure 7.6</b> Head characteristics for a typical CANDU pump. . . . .	7-14
<b>Figure 7.7</b> Torque characteristics for a typical CANDU pump. . . . .	7-15
<b>Figure 7.8</b> Choked flow characteristics for a valve. . . . .	7-16
<b>Figure 7.9</b> Control valve characteristics. . . . .	7-16
<b>Figure 7.10</b> Critical heat flux. . . . .	7-17
<b>Figure 7.11</b> Possible thermalhydraulic regimes in a coolant channel. . . . .	7-18
<b>Figure 7.12</b> Power and quality versus length along a fuel channel. . . . .	7-19
<b>Figure 7.13</b> Power versus quality. . . . .	7-19

**Figure 7.14** Critical Power Ratio determination. .... 7-19

## FOREWORD

This is a course about the simulation of nuclear reactor process systems for analysis purposes. Simulation is neither experimentation in the traditional sense of the word, nor theoretical. But clearly, our science and engineering as we now know it would not exist without simulation. Can you conceive of sending a man to the moon without simulation? Or building a nuclear power plant without simulation?

I propose (not originally of course) that there has emerged since the 60's, a new aspect of the scientific method: Simulation, which is orthogonal to experimentation and to theory. This new element alters the manner in which we go about our business. Prior to the advent of simulation tools, theories were posed and experiments were performed, often with severe limitations. Theoretical studies are limited by analytical constraints and experiments are limited by the bounds of cost, hazard, and measurement techniques. With simulation, however, analytical work is extended by numerical calculations and experiments are augmented by simulations. Often a simulation is superior to experiments. Some parameters are now more accurately simulated than they can be measured. Full scale simulations are feasible whereas full scale experiments are usually too risky or too costly to do. Not only is the nature of the scientific method changed, but the extent and scope of the method is vastly enhanced.

The nuclear industry is a typical industry that involves a great deal of fluid processes. It is atypical, however, because one of the process systems, the Heat Transport System (HTS), is of critical importance to the safety of the nuclear station. Sustained loss of cooling of the fuel is a catastrophic event. It has to be shown, a priori, that such events are of negligible probability and that the design is adequate to handle all probable events. Adequate design margin must be demonstrated. To compound the difficulty of the task, there is often insufficient evidence (thankfully) to base arguments on statistics. Consider also that current designs are pushed to their safe limits in order to extract the maximum power at the minimum cost. A nuclear station can typically cost  $\$10^{10}$  (US). A 1% increase in output power can save  $\$2 \times 10^8$  (US) over the life of the station. The key task of design and analysis of the HTS is, then, is to demonstrate safety, performance, reliability and maintainability prior to the actual construction of the facility. Without simulation, this clearly would not be possible.

Typically, the simulation support involves the setting up of a large code such as RELAP and RETRAN (or their Canadian equivalents CATHENA and SOPHT). Large data sets are required as input and copious tables of numbers are the result of the many runs that are required. It can take months to acquire the primary data for such codes in the environment of an engineering design office, although the use of project-wide data bases and CAD/CAE systems have reduced the cycle time somewhat. Manual analysis of the numerical output from a single run can often take days. Clearly, the actual computation time for the computer runs is small compared to the elapsed time of the total engineering task at hand. The bottleneck is not usually the computer; it is the engineer/scientist. It is stark testimony to the achievements of the last 20 years that a very wide scope of problems can be routinely handled by industrial codes. A new era of simulation is upon us! There is a distinct qualitative difference in such simulation tools over the calculations of the past.

For all the bravado of faster and more detailed plant renderings, we would be well advised to step back and look at simulation as an element in a larger project. Much is usually made of the enhancement of a simulator by the discovery of a faster algorithm. Obtaining a speedup of a factor of 2 is a notable event worthy of praise. But is it needed? Where is the bottleneck in your project? For the nuclear industry, the elapsed time for project completion, from project concept to in-service, is not significantly affected by simulation run time. Rather, the engineering phase is governed by concept generation, data preparation, model definition, coding, debugging, code verification, analysis, and design. A slow running code that is easy to use, modify or

develop, even though it is not the last word in accuracy or speed, is a clear winner over the exotic, temperamental, accurate and speedy A-stable, implicit, all singing-all dancing code.

But, alas, the real world demands compromises, a balance must be sought. Some enhancements over a naive explicit number cruncher are essential for stiff systems (for instance) and well worth the price in coding. The key thing to note, however, is that the parameter to optimize is not speed of computation, or stability or robustness per se. We need to optimize the overall project, not the code. In this regard, the optimum code is one that gets the job done with the minimum of fuss and muss. Keep in mind, however, that some careful planning in code design can lead to big payoffs down the line. For instance, effort spent in modularizing a code or generalizing it so that the code serves more than one project is often well spent. The art of simulation is knowing when to stop modularizing or generalizing and when to get down to work.

### ACKNOWLEDGEMENTS

This work was funded by McMaster University and the Natural Sciences and Engineering Research Council of Canada. The author is indebted to Atomic Energy of Canada Ltd. and to Ontario Hydro since this work is inextricably linked to past involvements with these companies. Special thanks go to my students, John Hoskins and Brian Hand, for water property development, and to Raymond Sollychin for his central role in the development of the rate method for the equation of state. A general thanks to all my students over the years; their freshness is a constant delight.

## GLOSSARY OF ABBREVIATIONS AND ACRONYMS

<b>AECB</b>	Atomic Energy Control Board (Canadian nuclear regulatory agency)
<b>AECL</b>	Atomic Energy of Canada Limited (CANDU design company)
<b>AESOP</b>	Atomic Energy Simulation of Optimization (computer code)
<b>ASDV</b>	Atmospheric Steam Discharge Valve
<b>ASSERT</b>	Advanced Solution of Subchannel Equations in Reactor Thermal hydraulics (computer code)
<b>ASTM</b>	American Society for Testing Materials
<b>BLC</b>	Boiler Level Control
<b>BLW</b>	Boiling Light Water
<b>BOILER</b>	Computer code for boiler (steam generator) design
<b>BOSS</b>	BOiler Secondary Side (computer code)
<b>BPC</b>	Boiler Pressure Controller
<b>CAD/CAE</b>	Computer Aided Drafting / Computer Aided Engineering
<b>COBRA</b>	?? (thermalhydraulic computer code)
<b>CANDU</b>	CANadian Deuterium Uranium (reactor type)
<b>CATHENA</b>	Canadian Thermalhydraulic ??? (computer code)
<b>CCP</b>	Critical Channel Power
<b>CHF</b>	Critical Heat Flux
<b>CPR</b>	Critical Power Ratio
<b>CRL</b>	Chalk River Laboratories (part of AECL)
<b>CSA</b>	Canadian Standards Association
<b>CSDV</b>	Condenser Steam Discharge Valve
<b>CSNI</b>	Canadian Standards for the Nuclear Industry
<b>DCC</b>	Digital Control Computer
<b>DF-ET</b>	Drift Flux-Equal Temperature (thermalhydraulic model)
<b>DF-UT</b>	Drift Flux-Unequal Temperature (thermalhydraulic model)
<b>DNB</b>	Departure from Nucleate Boiling
<b>DRIP</b>	Distributed Resistance in Porous Media (computer code)
<b>ECC</b>	Emergency Core Cooling
<b>ECI</b>	Emergency Core Injection
<b>EVET</b>	Equal Velocity Equal Temperature (thermalhydraulic model)
<b>EVUT</b>	Equal Velocity-Unequal Temperature (thermalhydraulic model)
<b>EWS</b>	Emergency Water Supply
<b>FIREBIRD</b>	??(computer code)
<b>FLASH</b>	??(computer code)
<b>FBR</b>	Feed, Bleed and Relief
<b>FP</b>	Full Power
<b>HEM</b>	Homogeneous Equilibrium Model
<b>HTS</b>	Heat Transport System
<b>HUT</b>	Hold-Up Tank
<b>HX</b>	Heat eXchanger
<b>HYDNA</b>	Hydraulic Network Analysis (computer code)
<b>I&amp;C</b>	Instrumentation and Control
<b>IBIF</b>	Intermittent Buoyancy Induced Flow
<b>LOCA</b>	Loss of Coolant Accident
<b>LOC/LOECC</b>	Loss of Coolant with Coincident Loss of Emergency Core Cooling
<b>LOP</b>	Loss of Pumping



<b>LOR</b>	Loss of Regulation
<b>milli-k</b>	Unit of reactivity for reactor physics
<b>NPSH</b>	Net Positive Suction Head
<b>NUCIRC</b>	Nuclear Circuits (computer code)
<b>OH</b>	Ontario Hydro (electrical utility company, Ontario, Canada)
<b>PGSA</b>	Pickering Generating Station A
<b>PHTS</b>	Primary Heat Transport System
<b>PHW</b>	Pressurized Heavy Water
<b>PHWR</b>	Pressurized Heavy Water Reactor
<b>POWDERPUFFS-V</b>	(reactor physics computer code)
<b>PRESCON2</b>	Pressure Containment (computer code)
<b>QA</b>	Quality Assurance
<b>RAMA</b>	Reactor Analysis Implicit Algorithm (computer code)
<b>R&amp;M</b>	Reliability and Maintainability
<b>RELAP</b>	(thermohydraulic computer code)
<b>RETRAN</b>	(thermohydraulic computer code)
<b>RB</b>	Reactor Building
<b>rem</b>	röntgen or rad equivalent mammal or man??
<b>RIH</b>	Reactor Inlet Header
<b>ROH</b>	Reactor Outlet Header
<b>RTD</b>	Resistance Temperature Detectors
<b>SDM</b>	Safety Design Matrices
<b>SOPHT</b>	Simulation of Primary Heat Transport (computer code)
<b>SRV</b>	Safety Relief Valve
<b>THIRST</b>	Thermal-Hydraulics in Recirculating Steam Generators (computer code)
<b>TMI</b>	Three Mile Island
<b>TOFFEA</b>	Two Fluid Flow Equation Analysis (computer code)
<b>TUEC</b>	Total Unit Energy Cost
<b>UVUEUP</b>	Unequal Velocity, Unequal Energy, Unequal Pressure (thermohydraulic model)
<b>UVUT</b>	Unequal Velocity Unequal Temperature (thermohydraulic model)
<b>VB</b>	Vacuum Building
<b>VC</b>	Vacuum Chamber
<b>WRE</b>	Whiteshell Research Establishment (part of AECL)

## NOMENCLATURE

A	area
<b>A</b>	arbitrary vector
C	concentration
$C_p$	heat capacity at constant pressure
$C_v$	heat capacity at constant volume
e	specific internal energy
E	internal heat source or sink
f	friction factor
<b>f</b>	long range or body force
$g_c$	gravitational constant
<b>g</b>	acceleration due to gravity
h	specific enthalpy
$h_N$	heat transfer coefficient
H	total enthalpy in volume, V
<b>I</b>	unity tensor
k	head loss coefficient
L	length
M	mass in volume, V
<b>M</b>	momentum interchange vector
<b>n</b>	unit vector normal to the surface
P	pressure
q	heat flux
Q	lumped heat source or sink
s	surface bounding volume, V
S	surface sink or source
t	time
T	temperature
U	total internal energy in volume, V
V	arbitrary fluid volume
<b>v</b>	velocity vector
W	mass flow
x	quality (weight fraction)

### Greek

$\alpha$	void fraction
$\gamma$	phase volume fraction
$\Gamma$	local sink or source
$\psi$	field variable
$\rho$	density
$\sigma$	stress tensor
$\theta$	angle with respect to horizontal
$\tau$	shear stress tensor

### Operators

$\frac{\partial}{\partial t}$	partial time derivative
$\frac{d}{dt}$	total time derivative
$\frac{D}{Dt}$	substantial time derivative
$\nabla$	Del operator
$\iiint ( ) dV$	volume integral
$\iint^V ( ) ds$	surface integral
$\langle \xi \rangle = \frac{1}{A} \int_S ( ) ds$	cross sectional average

### Subscripts

f	liquid (fluid) phase
g	vapour (gaseous) phase
i	summation index for nodes
j	summation index for links
k	1, 2 (1 = liquid, 2 = vapour)
S	surface
SAT	saturated
IN	ingoing
OUT	outgoing