THERMALHYDRAULICS
FOR
CANDU REACTORS

AN INTENSIVE SHORT COURSE
FOR NUCLEAR SCIENTISTS AND ENGINEERS

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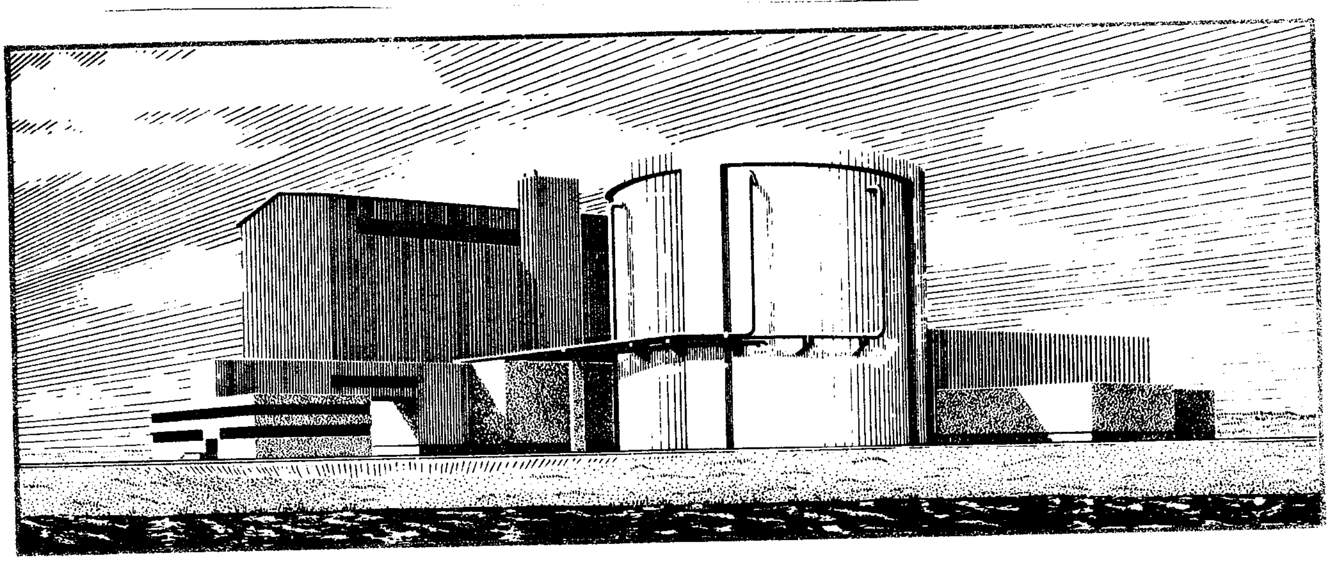
EDITED BY:
RICK BONALUMI - ONTARIO HYDRO
JEN-SHIH CHANG - M.I.E.S.
BILL GARLAND - AECL/EC
JACK KIRKALDY - M.I.E.S.
NORM SPINKS - AECL/EC



LECTURERS AND AUTHORS

CHAPTER

1. Introduction to Design and Analysis (W.J. Garland, AECL/EC)
2. Introduction to Safety Analysis (V. Snell, AECL/EC)
3. Heat and Mass Transfer and Two-Phase Flow I (J.S. Chang, McMaster)
4. Heat and Mass Transfer and Two-Phase Flow II (R.L. Judd, McMaster)
5. Single and Two-Phase Flow Modelling I (W. Brimley, AECL/EC)
6. Numerical Methods I (B. McDonald, AECL/WSNRE)
7. Single and Two-Phase Flow Modelling II (B. McDonald, AECL/WSNRE)
8. Computational Analysis of Multidimensional Flow (M. Carver, AECL/CRNL)
9. Numerical Methods II (P.T. Wan, McMaster)
10. Process Design I (W.J. Garland, AECL/EC)
11. Operational Considerations (R. Bassermann, Ontario Hydro)
12. Process Equipment (W. Brimley, AECL/EC and D. Meranda, Ontario Hydro)
13. Process Control (Q.B. Chou, Ontario Hydro)
14. Chemistry and Materials of Thermalhydraulic Loops (D. Barber, AECL/EC and R. Fleck, Ontario Hydro)
15. Safety Analysis I, Thermohydraulics (N. Spinks, AECL/EC)
16. Safety Analysis II, High Temperature Transient Fuel Channel Behaviour (P.D. Thomson, H.S. Waldman, C.L. Swift-Schulz, D. Caswell, S. Girgis, AECL/EC)
17. Single and Two-Phase Flow Diagnostics and Monitoring (W. Brimley, AECL/EC and J.S. Chang, McMaster)
18. Process Design II (W.J. Garland, AECL/EC)
19. Safety Analysis III, Containment Analysis (D. Pendergast, W.M. Collins, AECL/EC)

WORKSHOP CO-CHAIRMEN

Workshop I - Design (W. Garland, AECL/EC and D. Meranda, Ontario Hydro)

Workshop II - Safety (W. Midvidy, Ontario Hydro and V. Snell, AECL/EC)

THE CANDU POWER REACTOR SYSTEM: A PREFACE

The CANDU nuclear power reactor system has been under development since the early 1950's by Atomic Energy of Canada Limited (AECL) in collaboration with Canadian industry. CANDU stands for CANada Deuterium Uranium indicating that the reactor uses heavy water or deuterium oxide as moderator, and natural uranium as fuel. The success of the system is indicated by the record breaking performance of the Pickering Generating Station of Ontario Hydro on the shores of Lake Ontario close to the metropolitan area of Toronto (Fig. P1).

The four 500 MWe Pickering reactors are known as CANDU-PHW units, to indicate that the coolant is heavy water under pressure. Although the PHW system has received the most attention so far and has been developed to a commercial stage at Douglas Point Pickering and Bruce, other versions are under consideration. For example, a prototype station using light water as coolant was started up in 1971 at Gentilly, Quebec. Another version of CANDU is the OCR for Organic Cooled Reactor. A research reactor using an organic liquid coolant has been operating since 1965. Higher temperatures are a feature which could be of great value when other fuel cycles are being considered or for applications where high temperature steam is required.

The outstanding advantage of the CANDU system is its ability to operate with natural uranium fuel. This permits the purchase of uranium feedstock on the world market without limitations of enrichment services. In addition, the CANDU system extracts more electrical energy per unit of uranium feedstock than any other commercially proven reactor system.

Canada has abundant reserves of uranium but no enrichment plant so that the ability to utilize natural uranium fuel is particularly attractive from a national standpoint, both in terms of balance of payments and freedom from external political and economic pressures. These considerations have been significant factors in the longstanding support given to the development of the CANDU system by the Federal Government.

The major commercial utilization of the CANDU system in Canada has been within the Ontario Hydro system. The large size of this system, the absence in Ontario of significant fossil fuel reserves, and the harnessing of most of the available hydraulic sites, has led Ontario Hydro to commit a series of large multi-unit CANDU stations. Over 35% of electrical demand in Ontario is generated by nuclear power.

Hydro-Quebec's initial entry to the nuclear power field was via the construction and subsequent operation of the Gentilly-1 nuclear power station. This station, owned by AECL, employs the prototype CANDU-BLW reactor. While Quebec still has substantial unharnessed hydraulic reserves (primarily in the James Bay region), Hydro-Quebec has constructed a 600 MWe CANDU station in their system at the Gentilly site.

The potential attractiveness of this size led AECL to adopt it as a standard unit. Substantial success has already been achieved in marketing this unit design. In addition to Gentilly-2 being constructed by Hydro-Quebec, AECL has a contract (in partnership with an Italian company) to supply one unit to Argentina (Cordoba). The New Brunswick Electric Power Commission

WORLD POWER REACTOR LIFETIME PERFORMANCE

1	CANADA	Pickering-2	542 MW	84.5%
2	W. GERMANY	Stade-1	662 MW	83.5%
3	CANADA	Pickering-1	542 MW	83.3%
4	CANADA	Bruce 4	791 MW	78.5%
5	CANADA	Bruce 3	791 MW	78.2%
6	CANADA	Pickering-4	542 MW	77.6%
7	CANADA	Pickering-3	542 MW	77.5%
8	USA	Point Beach 2	524 MW	77.4%
9	USA	Connecticut Yankee	602 MW	75.4%
10	SWEDEN	Barsebaeck 2	600 MW	74.5%

CUMULATIVE LOAD FACTORS FOR REACTORS OVER 500 MW(e) TO END OF SEPTEMBER 1980

Station	Cumulative Load Factor %	Type
Bruce-3	82.0	CANDU
Stade-1	81.2	PWR
Pickering-2	80.9	CANDU
Pickering-1	80.3	CANDU
Point Beach-2	77.4	PWR
Pickering-4	77.3	CANDU
Pickering-3	75.4	CANDU
Prairie Island-2	75.2	PWR
Calvert Cliffs-2	74.7	PWR
Connecticut Yankee	74.6	PWR
Bruce-4	73.5	CANDU
Bruce-1	73.0	CANDU

REF: NUCLEAR ENGINEERING INTERNATIONAL VOL. 25, NO. 307, 1980

Country	Annual load factor%	Number and size of reactors	Cumulative load factor %	Number and size of reactors
Canada	70.11	10 (5818 MWe)	64.90	10 (5818 MWe)
Europe	56.35	53 (38500.3 MWe)	56.48	57 (42284.3 MWe)
USA	56.76	68 (54684 MWe)	54.74	68 (54658 MWe)
Japan	48.41	20 (13852 MWe)	52.40	22 (15117 MWe)
UK	51.77	22 (7949.3 MWe)	53.24	22 (7949.3 MWe)
France	56.63	10 (6429 MWe)	51.87	12 (8343 MWe)
W. Germany	50.49	10 (14299 MWe)	54.95	11 (15199 MWe)

Source: Nuclear Engineering International (March 1980)

FIGURE P1

has built one unit at a site near St. John (Point Lepreau), and another unit is under construction in the Republic of Korea. The four CANDU-600's are at various stages of completeness with Point Lepreau presently approaching full power.

Summaries for performance and penetration of CANDU reactors are given in Figs. P1, P2, and P3. While CANDU carries an initial cost penalty due to the high price of heavy water, this is more than offset during an operating life-time due to cheap natural uranium fuel, efficient burn-up and inherent safety features of the reactor. Indeed the CANDU PHW reactor design with its heavy water moderator, natural uranium fuel and pressure tube concept has certain characteristics that obviate the need for a high strength pressure vessel. Instead, the pressure boundaries are the pressure tubes which are considerably simpler to manufacture to the required quality. Further, experimental evidence and station performance indicates that pressure tubes will leak before they break since their thickness is much less than the critical crack length. Such leaks can be readily detected by monitoring the moisture content and the pressure in the gas annulus between the pressure tube and the calandria tube. This is done on a continuous basis. In addition, ultrasonic scanning devices are mounted on the fuelling machine for periodic in-service inspection of the pressure tubes.

The pressure tube design permits the heat transport system to be subdivided into two separate coolant circuits (loops). In the case of a hypothetical loss of coolant accident, this design feature restricts the consequences of the loss of coolant accident to just one of the loops. This simplifies the design and considerably reduces the burden on the emergency injection and the containment system design.

All reactivity devices are located in guide tubes positioned in the low pressure moderator environment. Thus, there exists no mechanism for rapid ejection of any of these reactivity devices, nor can they drop out of the core. The maximum reactivity rates achievable by driving all control reactivity devices together in the wrong direction is about 0.35 mk per second and well within the design capabilities of the protective systems.

Fuel, coolant and moderator are arranged in a square lattice with a 28.6 cm pitch. This is a near optimum geometry from a reactivity standpoint. Even if all fuel channels were either pushed apart or brought together for whatever reason the net reactivity increase would be at most, 1 milli-k where k is the neutron multiplication constant; and this only for the ideal case of uniform rearrangement. This is, of course, physically impossible. For the case where one, or a few fuel channels are displaced, the net reactivity would at worst not be affected at all or it would decrease, thereby shutting down the reactor. Also, since a lattice of natural **uranium** and light water cannot be made critical in any concentration, there can be no criticality problems in the spent fuel bay of CANDU reactors.

The pressure tube design also makes on-power fuelling a possibility. On-power fuelling results in a reactor with very low reactivity control requirements. Typically, the reactivity decay rate in 600 MW(e) CANDU PHW reactors is about 0.4 mk per day. This is compensated by fuelling about

NAME	LOCATION	TYPE	POWER MWe NET	NUCLEAR DESIGNER	DATE OF FIRST POWER
NPD	ONTARIO	PHW	22	AECL & CGE	1962
DOUGLAS POINT	ONTARIO	PHW	206	AECL	1967
PICKERING A	ONTARIO	PHW	515 x 4	AECL	1971/73
GENTILLY 1	QUEBEC	BLW	266	AECL	1971
KANUPP	PAKISTAN	PHW	125	CGE	1971
RAPP 1	INDIA	PHW	203	AECL	1972
RAPP 2	INDIA	PHW	203	AECL	—
BRUCE A	ONTARIO	PHW	740 x 4	AECL	1976/79
GENTILLY 2	QUEBEC	PHW	640	AECL	1982
POINT LEPREAU	NEW BRUNSWICK	PHW	635	AECL	1982
CORDOBA	ARGENTINA	PHW	600	AECL	1983
PICKERING B	ONTARIO	PHW	516 x 4	AECL	1982
WOLSUNG 1	KOREA	PHW	600	AECL	1983
BRUCE B	ONTARIO	PHW	756 x 4	AECL	1983
DARLINGTON	ONTARIO	PHW	850 x 4	AECL	UNDER CONSTRUCTION
CERNAVODA	ROMANIA	PHW	600	AECL	PROJECTED
		TOTAL	18,208 MWe		

FIGURE P2 CANDU POWER REACTORS, EXISTING OR UNDER CONSTRUCTION

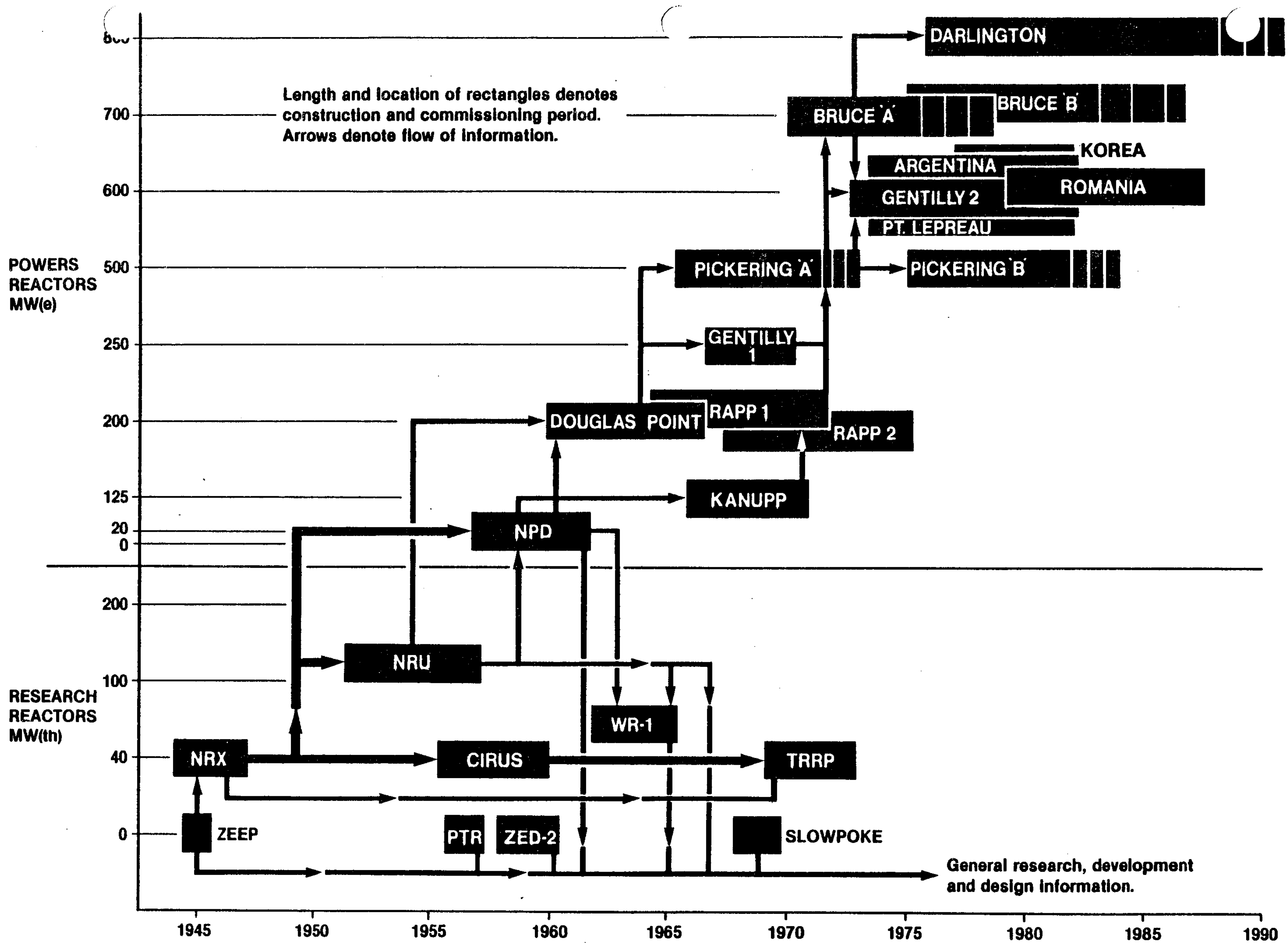


FIGURE P3 GENEALOGY OF CANDU REACTORS

two channels per day. In addition, the pressure tube concept provides an excellent opportunity for locating fuel defects and the on-power fuelling permits the removal of defective fuel as soon as it is detected. This helps to keep the heat transport system essentially free from fission product activity.

Finally, the separation of the moderator from the high pressure heat transport coolant allows the moderator to act under certain circumstances as an additional heat sink for the fuel decay heat, e.g., where one might hypothesize a failure or impairment in the emergency core cooling system following a primary loss of coolant accident.

Thermalhydraulics, which is the central theme of this course, is concerned with safe and effective heat removal from the reactor core for power production. The basic CANDU design, while favourable to both safety and efficiency, must be studied in detail for the development of optimal structures and strategies. The following lecture notes represent the state-of-the-art of this challenging branch of nuclear engineering.

Table of Contents

Chapter | | Page

GLOSSARY OF ABBREVIATIONS AND ACRONYMS

AE	Acoustic Emission
AECB	Atomic Energy Control Board
AESOP	Atomic Energy Simulation of Optimization
ASDV	Atmospheric Steam Discharge Valve
ASSERT	Advanced Solution of Subchannel Equations in Reactor Thermalhydraulics
ASTM	American Society for Testing Materials
BLC	Boiler Level Control
BLW	Boiling Light Water
BOILER	Boiler
BPC	Boiler Pressure Controller
CCP	Critical Channel Power
CHF	Critical Heat Flux
CPR	Critical Power Ratio
CRNL	Chalk River Nuclear Laboratories
CRT	Cathode Ray Tube
CSA	Canadian Standards Association
CSDV	Condenser Steam Discharge Valve
CSNI	Canadian Standards for the Nuclear Industry
DBE	Design Base Earthquake
DCC	Digital Control Computer
DF-ET	Drift Flux-Equal Temperature
DF-UT	Drift Flux-Unequal Temperature
DNB	Departure from Nucleate Boiling
ECC	Emergency Core Cooling
ECI	Emergency Core Injection
EFPH	Effective Full Power Hours
EVET	Equal Velocity Equal Temperature
EVUT	Equal Velocity-Unequal Temperature
EWS	Emergency Water Supply
FBR	Feed, Bleed and Relief
FP	Full Power
HEM	Homogeneous Equilibrium Model
HTS	Heat Transport System
HWP	Heavy Water Plant
HYDNA	Hydraulic Network Analysis
I&C	Instrumentation and Control
IBIF	Intermittent Buoyancy Induced Flow

ICRP	International Commission on Radiological Protection
LOC	Loss of Coolant
LOCA	Loss of Coolant Accident
LOC/LOECC	Loss of Coolant with Coincident Loss of Emergency Core Cooling
LOP	Loss of Pumping
LOR	Loss of Regulation
MCCR	Ministry of Corporate and Consumer Relations
MCS	Maintenance Cooling System
MHD	Magneto hydrodynamics
milli-k	See p. 13-7
NPD	Nuclear Power Demonstration
NPSH	Net Positive Suction Head
NUCIRC	Nuclear Circuits
OECD	Organization for Economic Co-operation and Development
PGSA	Pickering Generating Station A
PHTS	Primary Heat Transport System
PHW	Pressurized Heavy Water
PHWR	Pressurized Heavy Water Reactor
PRESCON2	Pressure Containment
QA	Quality Assurance
RAMA	Reactor Analysis Implicit Algorithm
R&M	Reliability and Maintenance
RB	Reactor Building
rem	röntgen or rad equivalent mammal
RIH	Reactor Inlet Header
ROH	Reactor Outlet Header
RTD	Resistance Temperature Detectors
SDM	Safety Design Matrices
SOPHT	Simulation of Primary Heat Transport
SRV	Safety Relief Valve
TMI	Three Mile Island
TOFFEA	Two Fluid Flow Equation Analysis
UVUT	Unequal Velocity Unequal Temperature
VB	Vacuum Building
VC	Vacuum Chamber
WNRE	Whiteshell Nuclear Research Establishment