THERMO-HYDRAULICS OF NUCLEAR REACTORS

This book provides a concise and up-to-date summary of the essential thermo-hydraulic analyses and design principles of nuclear reactors for electricity generation. Beginning with the basic nuclear physics, it leads through technical and quantitative analyses to descriptions of both the normal operation of the various modern nuclear reactor designs and the analyses of the possible departures from normal operation. It then describes both the postulated accident scenarios and summaries of the causes for the three major nuclear power generation accidents, Three Mile Island, Chernobyl, and Fukushima, as well as the major improvements to reactor safety that grew out of those analyses and accidents.

Professor Christopher Earls Brennen was a member of the Mechanical Engineering Faculty at Caltech for over 40 years and retired as the Richard L. and Dorothy M. Hayman Professor of Mechanical Engineering in 2005. As a teacher he was the recipient of a number of teaching awards including the prestigious Richard Feynman Prize.

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Preface

This book presents an overview of the thermo-hydraulics of the nuclear reactors designed to produce power using nuclear fission. The book began many years ago as a series of notes prepared for a graduate student course at the California Institute of Technology. When, following the Three Mile Island accident in 1979, nuclear power became politically unpopular, demand and desire for such a course waned, and I set the book aside in favor of other projects. However, as the various oil crises began to accentuate the need to explore alternative energy sources, the course and the preparation of this book were briefly revived. Then came the terrible Chernobyl accident in 1986, and the course and the book got shelved once more. However, the pendulum swung back again as the problems of carbon emissions and global warming rose in our consciousness and I began again to add to the manuscript. Even when the prospects for nuclear energy took another downturn in the aftermath of the Fukushima accident (in 2011), I decided that I should finish the book whatever the future might be for the nuclear power industry. I happen to believe, despite the accidents – or perhaps because of them – that nuclear power will be an essential component of electricity generation in the years ahead.

The book is an introduction to a graduate-level (or advanced undergraduate-level) course in the thermo-hydraulics of nuclear power generation. Because neutronics and thermo-hydraulics are closely linked, a complete understanding of thermo-hydraulics and the associated safety issues also requires knowledge of the neutronics of nuclear power generation and, in particular, of the interplay between the neutronics and the thermo-hydraulics that determine the design of the reactor core. This material necessarily leads into the critical issues associated with nuclear reactor safety, and this, in turn, would be incomplete without brief descriptions of the three major accidents (Three Mile Island, Chernobyl, and Fukushima) that have influenced the development of nuclear power.

Some sections in Chapter 6 of this book were adapted from two of my other books, *Cavitation and Bubble Dynamics* and *Fundamentals of Multiphase Flow*, and I am grateful to the publisher of those books, Cambridge University Press, for permission to reproduce those sections and their figures in the present text. Other figures and photographs reproduced in this book are acknowledged in their respective captions. I would also like to express my gratitude to the senior colleagues at the California Institute of Technology who introduced me to the topic of nuclear power generation, in particular, Noel Corngold and Milton Plesset. Milton did much to advance the cause of nuclear power generation in the United States, and I am much indebted to him for his guidance. I also appreciate the interactions I had with colleagues at other institutions, including Ivan Catton, the late Ain Sonin, George Maise, and the staff at the Nuclear Regulatory Commission.

This book is dedicated to James MacAteer, from whom I first heard the word *neutron*, and to the Rainey Endowed School in Magherafelt, where the physics Johnny Mac taught me stayed with me throughout my life.

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Mathematical Nomenclature

Roman letters

a	Amplitude of wave-like disturbance
A	Cross-sectional area
A	Atomic weight
b	Thickness
B_g^2	Geometric buckling
B_m^2	Material buckling
c	Speed of sound
c_p	Specific heat of the coolant
C, C_1, C_2, C_R	Constants
C^*, C^{**}	Constants
C_f	Friction coefficient
C_i	Concentration of precursor i
d	Diameter
D	Neutron diffusion coefficient
D_h	Hydraulic diameter of coolant channel
E	Neutron kinetic energy
E'	Neutron energy prior to scattering
f	Frequency
g	Acceleration due to gravity
h, h^*	Heat transfer coefficients
H	Height
H_E	Extrapolated height
Hm	Haberman-Morton number, normally $g\mu^4/\rho S^3$
j	Total volumetric flux
j_N	Volumetric flux of component N
J_j	Angle-integrated angular neutron current density vector
J_j^*	Angular neutron current density vector
k	Multiplication factor
k_∞	Multiplication factor in the absence of leakage

k	Thermal conductivity
$\mathcal{K}$	Frictional constants
l	Typical dimension of a reactor
ℓ	Typical dimension
ℓ	Mean free path
ℓ_a	Mean free path for absorption
ℓ_f	Mean free path for fission
ℓ_s	Mean free path for scattering
L	Neutron diffusion length, $(D/\Sigma_a)^{\frac{1}{2}}$
$\mathcal{L}$	Latent heat of vaporization
$\dot{m}$	Mass flow rate
m	Index denoting a core material
M	Number of different core materials denoted by $m = 1$ to M
Ma	Square root of the Martinelli parameter
n	Integer
$n(E)dE$	Number of neutrons with energies between E and $E + dE$
N	Number of neutrons or nuclei per unit volume
N_f	Number of fuel rods
$\mathcal{N}$	Number of atoms per unit volume
N^*	Site density, number per unit area
Nu	Nusselt number, hD_h/k_L
p	Pressure
p^T	Total pressure
P	Power
$\mathcal{P}$	Perimeter
$(1 - P_F)$	Fraction of fast neutrons that are absorbed in ^{238}U
$(1 - P_T)$	Fraction of thermal neutrons that are absorbed in ^{238}U
Pr	Prandtl number
$\dot{q}$	Heat flux per unit surface area
$\mathcal{Q}$	Rate of heat production per unit length of fuel rod
r	Radial coordinate
r, θ, z	Cylindrical coordinates
R	Radius of reactor or bubble
R_E	Extrapolated radius
R_R	Reflector outer radius
R_{RE}	Extrapolated reflector radius
R_P	Fuel pellet radius
R_O	Outer radius
Re	Reynolds number
s	Coordinate measured in the direction of flow
$S(x_i, t, E)$	Rate of production of neutrons of energy, E, per unit volume
$\mathcal{S}$	Surface tension
t	Time

T	Temperature
u, U	Velocity
$\bar{u}$	Neutron velocity
u_i	Fluid velocity vector
u_N	Fluid velocity of component N
V	Volume
$\dot{V}$	Volume flow rate
x, y, z	Cartesian coordinates
x_i	Position vector
x_N	Mass fraction of component N
$\mathcal{X}$	Mass quality
z	Elevation

Greek letters

α	Volume fraction
α_L	Thermal diffusivity of liquid
α_{mf}	Ratio of moderator volume to fuel volume
β	Fractional insertion
β	Volume quality
β	Fraction of delayed neutrons
ϵ	Fast fission factor of ^{238}U
δ	Boundary layer thickness
η	Efficiency
η	Thermal fission factor of ^{238}U
θ	Angular coordinate
κ	Bulk modulus of the liquid
κ	Wave number
κ_L, κ_G	Shape constants
λ	Wavelength
λ_i	Decay constant of precursor i
$(1-\Lambda_F)$	Fraction of fast neutrons that leak out of the reactor
$(1-\Lambda_T)$	Fraction of thermal neutrons that leak out of the reactor
ξ	Time constant
ξ_1, ξ_2	Constants
μ, ν	Dynamic and kinematic viscosity
ρ	Density
ρ	Reactivity, $(k-1)/k$
σ	Cross section
$\sigma_a, \sigma_f, \sigma_s$	Cross sections for absorption, fission, and scattering
Σ	Macroscopic cross section, $\mathcal{N}\sigma$
Σ_{tr}	Macroscopic transport cross section, $1/3D$

τ	Half-life
τ_w	Wall shear stress
ϕ	Angle-integrated neutron flux
$\phi_L^2, \phi_G^2, \phi_{L0}^2$	Martinelli pressure gradient ratios
φ	Angular neutron flux
ω	Radian frequency
ω_a	Acoustic mode radian frequency
ω_m	Manometer radian frequency
Ω_j	Unit direction vector

Subscripts

On any variable, Q:

Q_o	Initial value, upstream value, or reservoir value
Q_1, Q_2	Values at inlet and discharge
Q_a	Pertaining to absorption
Q_b	Bulk value
Q_c	Critical values and values at the critical point
Q_d	Detachment value
Q_e	Effective value or exit value
Q_e	Equilibrium value or value on the saturated liquid-vapor line
Q_i	Components of vector Q
Q_f	Pertaining to fission or a fuel pellet
Q_s	Pertaining to scattering
Q_w	Value at the wall
Q_A, Q_B	Pertaining to general phases or components, A and B
Q_B	Pertaining to the bubble
Q_C	Pertaining to the continuous phase or component, C
Q_C	Critical value
Q_C	Pertaining to the coolant or cladding
Q_{CI}	Pertaining to the inlet coolant
Q_{CS}	Pertaining to the inner cladding surface
Q_D	Pertaining to the disperse phase or component, D
Q_E	Equilibrium value
Q_F	Pertaining to fast neutrons
Q_{FS}	Pertaining to the fuel pellet surface
Q_G	Pertaining to the gas phase or component
Q_L	Pertaining to the liquid phase or component
Q_M	Mean or maximum value
Q_N	Nominal conditions or pertaining to nuclei
Q_N	Pertaining to a general phase or component, N
Q_R	Pertaining to the reflector

Q_S	Pertaining to the surface
Q_T	Pertaining to thermal neutrons
Q_V	Pertaining to the vapor
Q_∞	Pertaining to conditions far away

Superscripts and other qualifiers

On any variable, Q:

$\bar{Q}$	Mean value of Q
$\dot{Q}$	Time derivative of Q
δQ	Small change in Q
ΔQ	Difference in Q values
Q^m	Pertaining to the material component, m