Nuclear Engineering and Design 330 (2018) 166–186

Contents lists available at ScienceDirect

Nuclear Engineering and Design

journal homepage: www.elsevier.com/locate/nucengdes

Thermal-hydraulic phenomena for water cooled nuclear reactors

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ARTICLE INFO

Keywords:
Validation
Scaling
Uncertainty analysis
Accident Analysis
Core design
System design
Computational tools

ABSTRACT

Nuclear Reactor Safety (NRS), Deterministic Safety Assessment (DSA) and Accident Analysis (AA) constitute the general framework for the topic of the present paper. The class of Water-Cooled Nuclear Reactors (WCNRs) is concerned. This includes most of the nuclear reactors in operation, under construction or in advanced design stage. The required licensing process for those reactors, are further necessary elements to establish the context of the performed activity.

Best Estimate (BE) system thermal-hydraulic codes are adopted to demonstrate the safety of WCNR based on AA, namely focusing on the class of Design Basis Accidents (DBAs). On the one hand, the validation of BE codes is a necessary step to prove their applicability to calculate accident scenarios. On the other hand the knowledge of accident scenarios is a requirement for the design and the operation of WCNR. The validation of BE codes and the knowledge of accident scenarios needs the identification and the characterization of Thermal-hydraulic Phenomena (T-HP).

A list of 116 T-HP is derived in the present paper, based on the documents issued in the last three decades by the Committee on the Safety of Nuclear Installations of Nuclear Energy Agency of The Organization for Economic Cooperation and Development (OECD/NEA/CSNI) and by the International Atomic Energy Agency (IAEA). The T-HP list includes the consideration of Separate Effect Tests (SET) and Integral Effect Tests (IET) relevant in Reactor Coolant System (RCS) and Containment of WCNRs. A dozen WCNR types are considered and include Pressurized Water Reactors (PWRs), Boiling Water Reactors (BWRs), Russian design reactor types (e.g. VVER-440, VVER-1000 and RBMK), pressure tube heavy water reactor designs in Canada (CANDU) and in India (PHWR) and so-called 'advanced' reactors (in the text of this paper, they are sometimes assigned as "New Reactors"), which are also equipped with passive systems (for instance, AP-1000 and APR-1400 designed in US and Korea, respectively).

Each T-HP can be characterized by one or more parameters or variables which are part of numerical models and constitute calculational results from system codes. A cross link process can be established between T-HP, parameters and DBA scenarios. The basis for the process and selected cross-link examples are provided and discussed.

A variety of applications for the T-HP list is envisaged in nuclear thermal-hydraulics. Insights are given in the paper in relation to the use of phenomena: a) to address the scaling issue; b) to distinguish between constitutive equations part of the balance equations and 'special models' in BE system codes; c) to prioritize research in nuclear reactor thermal-hydraulics.

1. Introduction

Nuclear Power Plant (NPP) technology is largely based on Nuclear Reactor Safety (NRS). Namely, NRS issues are well known to affect the design and the operation of NPP units. The relation between NRS and design/operation of NPP was established since the first reactor design in the 50's of last century. However, first, the advent of suitable computers and the almost simultaneous issuing of the Interim Acceptance Criteria (IAC) for the design of Emergency Core Cooling Systems (ECCS) by the US Atomic Energy Commission (AEC) in the early 70's, USAEC, 1974, and second, the well known Three Mile Island event at the end of the same decade, triggered massive research projects in NRS and in nuclear thermal-hydraulics. Noticeable inheritances from those massive projects, which ended toward the end of the last century, were:

- database of experiments; and

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https://doi.org/10.1016/j.nucengdes.2018.01.035
Received 28 August 2017; Received in revised form 6 January 2018; Accepted 20 January 2018

Nomenclature

AA	Accident Analysis
ABWR	Advanced BWR
ACC	Accumulator
AEC	Atomic Energy Commission
AM	Accident Management
APEX	Key ITF in US for the identification and characterization of T-HP
AP-1000	Advanced PWR
APR-1400	Advanced Pressurized Reactor
AS	Accident Scenario
ATLAS	Key ITF in Korea for the identification and characterization of T-HP
ATWS	Anticipated Transient Without Scram
BDBA	Beyond Design Basis Accident
BE	Best Estimate
BEPU	Best Estimate Plus Uncertainty
BETHSY	Key ITF in France for the identification and characterization of T-HP
BoP	Balance of Plant
BWR	Boiling Water Reactor
CANDU	Canadian Deuterium Uranium
CCF	Counter Current Flow
CCFL	Counter Current Flow Limitation
CCTF	Key SETF in Japan for the identification and characterization of T-HP
CCVM	Computer (or CSNI) Code Validation Matrix
CFD	Computational Fluid Dynamics
CHF	Critical Heat Flux
CL	Cold Leg
CMT	Core Make-up Tank
CONT	Containment (related)
CRE	Control Rod Ejection
CRGT	Control Rod Guide Tube
CSAU	Code Scaling Applicability and Uncertainty
CSNI	Committee on the Safety of Nuclear Installations
CV	Control Volume
DB	Data Base
DBA	Design Basis Accident
DNB	Departure from Nucleate Boiling
DSA	Deterministic Safety Assessment
ECC	Emergency Core Cooling
ECCI	ECC Injection
ECCS	ECC System
EPR	European Pressurized Reactor
ESBWR	Economic Simplified BWR
ESF	Engineered Safety Feature
FA	Fuel Assembly
FCB	Fuel Channel Blockage
FCO	Forced Convection (HT regime)
FW	Feed-Water
FWLB	FW Line Break
GP	Generalized (thermal-hydraulic) Parameter
HEBR	Header Break
HL	Hot Leg
HOSG	Horizontal Tube SG
HT	Heat Transfer
IAEA	International Atomic Energy Agency
IC	Isolation Condenser
IET	Integral Effect Test
IRWST	In-containment Reactor Water Storage Tank
ITF	Integral Test Facility
I/O	Input/Output
J	Junction
LBLOCA	Large Break LOCA
LCC/SW	Loss of Component Cooling/Service Water
LOBI	Key ITF in Italy (EU ownership)for the identification and characterization of T-HP
LOCA	Loss of Coolant Accident
LOFA	Loss of Flow Accident
LOFT	Key ITF in US for the identification and characterization of T-HP
LOFW	Loss of Feed-Water
LOOSP	Loss of On-site and Off-Site Power
LP	Lower Plenum
LSTF	Key ITF in Japan for the identification and characterization of T-HP
LWR	Light Water Reactor
MBLOCA	Medium Break LOCA
MCP	Main Coolant Pump
MPTR	Multiple Pressure Tube Rupture
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break
NC	Natural Circulation
NCO	Natural Convection (HT regime)
NEA	Nuclear Energy Agency
NPP	Nuclear Power Plant
NRS	Nuclear Reactor Safety
OECD	Organization for Economic Cooperation and Development
OTSG	Once Through SG
PANDA	Key ITF in Switzerland for the identification and characterization of T-HP
PCEI	Parallel Channel Effect and Instabilities
PHW	Phenomenological Window
PHWR	Pressurized Heavy Water Reactor
PIE	Postulated Initiating Event
PIRT	Phenomena (T-HP) Identification and Ranking Table
PIUS	Passive Intrinsic Ultimate Safety (reactor type)
PKL	Key ITF in Germany for the identification and characterization of T-HP
PRISE	Primary to Secondary Leakage
PRZ	Pressurizer
PS	Primary System
PSA	Probabilistic Safety Assessment
PSB	Key ITF in Russia for the identification and characterization of T-HP
PSP	Pressure Suppression Pool
PTS	Pressurized Thermal Shock
PUMA	Key ITF in US for the identification and characterization of T-HP
PWR	Pressurized Water Reactor
PWR-O	PWR equipped with OTSG
PWR-V	PWR equipped with HOSG
QF	Quench Front
RBMK	Boiling Water Graphite Moderated Reactor
RCS	Reactor Coolant System
RD	Reference Document (for the characterization of AS)
RIA	Reactivity Initiated Accident
RNB	Return to Nucleate Boiling
SANB	Saturated Nucleate Boiling (HT regime)
SBLOCA	Small Break LOCA
SBO	Station Blackout
SET	Separate Effect Test
SEMI-SCALE	Key ITF in US for the identification and characterization of T-HP
SETF	Separate Effect Test Facility
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SH-D	Shutdown Transient