NUCLEAR POWER PLANT SAFETY AND MECHANICAL INTEGRITY

DESIGN AND OPERABILITY OF MECHANICAL SYSTEMS, EQUIPMENT AND SUPPORTING STRUCTURES

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BIOGRAPHY

Mr Antaki has nearly 40 years of experience in nuclear power and process engineering. He is a Fellow of the American Society of Mechanical Engineers (ASME). He is the Chairman of ASME III Working Group Piping Design and ASME B31.1 Mechanical Design Committee, and a member of the ASME Operation and Maintenance Subgroup. Mr Antaki was an engineer and a group manager at Westinghouse, and is currently Chief Engineer, Becht Nuclear Services. Mr Antaki resides in Aiken, South Carolina.

Mr Gilada has nearly 35 years of experience in nuclear mechanical and civil projects. He is a member of ASME III Working Group Piping Design, the Working Group Flaw Evaluation, and the ASME Operation and Maintenance Subgroup. He is the Principal Consultant Engineer at the Comanche Peak Nuclear Power Plant, Texas, with responsibilities in the field of civil and structural mechanics, and also in piping and suspended systems engineering. Mr Gilada resides in Cleburne, Texas.

PREFACE

This is a book about issues and tasks faced by engineers in nuclear power plants, with a focus on the safe and reliable operation of mechanical systems and equipment, and their supporting structures. The issues and challenges described in this book vary from the most complex subjects, such as replacing steam generators or tackling complex material damage mechanisms, down to the simpler tasks like maintenance, cleaning, and painting that, while simpler, call for logical and technically sound solutions.

This book is a compilation from our personal experiences that span many decades. Our sole purpose is to aid a new generation of engineers navigate through the vast amount of information in this field, and to emphasize the importance, in our business, of approaching every day's challenges with the same degree of rigor and thoroughness to achieve safe and reliable plant operation.

We have tried to describe the thought process followed in defining the issues at hand, and solving them using knowledge from codes, standards, regulations, past experience, and the basic principles of mechanical, materials, and structural engineering. This knowledge converges into a field, best described as Safety and Mechanical Integrity, which we chose for the title. It is the field of engineering in which we face challenges to plant equipment and plant operation (whether human induced, or caused by operational transients, ageing and obsolescence, or natural phenomena hazards) and have to make important operability and run-or-repair decisions.

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The opportunity to work in nuclear power plants and to support their safe operation has provided us many opportunities to contribute and also to constantly learn the art of engineering. We are thankful for the support provided to us by plant management in permitting us to share some of this experience for the benefit of others in the industry.

We are honored to be part of Code committees in the field of nuclear power, and thank our colleagues, our friends of so many years on Code committees, for the constant give-and-take that takes place in translating everyday knowledge into codes and standards, and eventually into books like this one.

At times, with the day-to-day emergencies, we thought we may not make our deadlines, but thanks to the excellent support and insights of Chelsea Johnston and Poulouse Joseph of Elsevier, we persevered. Finally, we would like to acknowledge the support we have received from our families during the two year journey to produce this book.

ACRONYMS AND DESCRIPTION

ACI	American Concrete Institute
ACRS	Advisory Committee on Reactor Safeguards
AEC	Atomic Energy Commission
AFW	Auxiliary feedwater
AG	Code on Nuclear Air and Gas Treatment
AHU	Air Handling Unit
AISC	American Institute of Steel Construction
AISI	American Iron and Steel Institute
ANS	American Nuclear Society
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASME O&M	Operation and Maintenance of Nuclear Power Plant
ASTM	ASTM International
AWS	American Welding Society
AWWA	American Waterworks Association
BDBE	Beyond-Design Basis Event (and BDB Earthquake)
BEZ	Break Exclusion Zone
BL	NRC Bulletin
BTP	Branch technical position
BWR	Boiling water reactor
B&PV	Boiler and Pressure Vessel
CC	Code Case
CEA-CEN	Commissariat a l'Energie Atomique Centre d'Etudes Nucleaires
CFD	Computational fluid dynamics
CFR	Code of Federal Regulations
CGD	Commercial Grade Dedication
CP	Construction permit
CR	Contractor report
CRDM	Control rod drive mechanism
CS	Core support and Containment spray
DMW	Dissimilar metal welds
EPRI	Electric Power Research Institute
EQ	Environmental qualification and Equipment qualification
ERDA	Energy Research and Development Administration
ET	Eddy current testing
FA	Functionality assessment
FAC	Flow-accelerated corrosion
FEA	Finite element analysis
FEMA	Federal Emergency Management Agency
FIV	Flow-induced vibration
FLEX	Diverse and flexible coping strategies

FMEA	Failure mode and effects analysis
FSAR	Final safety analysis report
GDC	General design criteria
GIP	Generic implementation procedure
GL	NRC Generic Letter
GMRS	Ground motion response spectra
GPM	Gallons per minute
HE	High Energy
HVAC	Heating, ventilation and air conditioning
IAEA	International Atomic Energy Agency
ID	Inside diameter
IEEE	Institute of Electrical and Electronics Engineers
IER	Industry Event Report
IN	NRC Information Notice
INPO	Institute of Nuclear Power Operations
IOE	Industry operating experience
ISI	Inservice inspection
IST	Inservice testing, ISTB (pumps), ISTC (valves), ISTD (snubbers)
LBB	Leak before break
LER	Licensee Event Report
LOCA	Loss of coolant accident
MC	Metal containment
ME	Moderate energy
MEB	Mechanical engineering branch
MR	Maintenance rule
MRP	Material Reliability Program (EPRI)
MSIV	Main steam isolation valve
MSS-SP	Manufacturers Standardization Society – Standard Practices
MT	Magnetic testing
NDE	Nondestructive examination
NDT	Nondestructive testing
NEI	Nuclear Energy Institute
NEMA	National Electrical Manufacturers Association
NQA	Nuclear Quality Assurance
NRC	Nuclear Regulatory Commission
NUREG	NRC Regulatory Guidance
OBE	Operating basis earthquake
OD	Operability determination
OD	Outside diameter
OSHA	Occupational Safety and Health Administration
O&M	Operation and Maintenance
PCC	Post-Construction Code
PRA	Probabilistic risk assessment

PRHA	Pipe rupture hazards analysis
PT	Penetrant testing
PTS	Pressurized thermal shock
PVRC	Pressure Vessel Research Council
PWR	Pressurized water reactor (and pipe whip restraint)
PWSCC	Primary water stress corrosion cracking
P&ID	Piping and instrumentation diagram
QA	Quality assurance
QC	Quality control
QME	Qualification of Active Mechanical Equipment
RAHA	Radial Arm and Hoist Assembly
RCS	Reactor coolant system
RG	Regulatory guide
RHR	Residual heat removal
RT	Radiographic testing
SAM	Seismic anchor motion
SAR	Safety analysis report
SCC	Stress corrosion cracking
SG	Steam generator
SMA	Seismic margins assessment
SMACNA	Sheet Metal & Air Conditioning Contractors' National Association
SPRA	Seismic Probabilistic Risk Assessment
SQUG	Seismic Qualification Utilities Group
SRP	Standard review plan
SS	Stainless steel
SSC	Structure, system and component
SSE	Safe shutdown earthquake
TIG	Tungsten inert gas (welding)
TS	Technical specification
UT	Ultrasonic testing
VT	Visual testing
ZOI	Zone of influence

Section III

NCA	ASME Section III Subsection General Requirements for Division 1 (Piping & Components) and Division 2 (Concrete Containments)
NB	ASME III Subsection for Class 1 Components
NC	ASME III Subsection for Class 2 Components
ND	ASME III Subsection for Class 3 Components
NE	ASME III Subsection for Class MC Components
NF	ASME III Subsection for Pipe Supports

Section XI

IWA	ASME XI General Requirements
IWB	ASME XI Requirements for Class 1 Components of Light-Water Cooled Plants
IWC	ASME XI Requirements for Class 2 Components of Light-Water Cooled Plants
IWD	ASME XI Requirements for Class 3 Components of Light-Water Cooled Plants

CONTENTS

CHAPTER 1

Regulations, Codes, and Standards

Chapter 1 covers the regulations that govern the design, construction, and operation of mechanical systems and components in nuclear power plants. Regulations are the building blocks of nuclear power plant engineering; they are defined in each plant's safety analysis report (SAR), and they must be understood and followed at each step of the engineering process. Following the regulations, we address Codes and Standards, with emphasis on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PV Code) because it covers piping systems, vessels, pumps, and valves, which are of particular interest in this book. We close this chapter explaining how structures, systems, and components (SSCs) are classified into safety classes, seismic categories, and essential classes.

1.1 REQUIREMENTS

1.1.1 Regulation

Why start the book with regulations?

Engineering activities at a nuclear power plant are highly regulated. An engineer in a nuclear power plant has to focus on three types of requirements, all of them essential and complementary: (1) following the company procedures, (2) following good engineering practice (codes, standards, guides, etc.), and (3) Following the regulations applicable at the plant.

It is useful to first look at the regulatory aspect of nuclear plant engineering in the United States, before turning our attention to codes and standards, because nuclear power regulations control the application of codes and standards, and provide supplementary requirements that are not contained in codes and standards. Also, regulations control how to document engineering issues and their resolution.

How are engineering activities regulated in the nuclear power industry?

The engineering of nuclear power plants is closely regulated by the US Nuclear Regulatory Commission (NRC). The NRC was formed from the

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Atomic Energy Commission (AEC) through the Energy Reorganization Act of 1974, which replaced the Atomic Energy Act of 1954. It made the NRC a regulatory agency, no longer involved in the development and promotion of nuclear power as was its predecessor the AEC.

The NRC is empowered by the Congress to regulate and provide oversight of the complete life cycle of nuclear power plants, from design and construction, through operation, life extension, and decommissioning.

What are the functions of the NRC?

The NRC has several functions, which include: (1) to develop regulations and guidance, (2) to license applicants to operate the nuclear facility; (3) to oversee compliance with the licensing basis and safety requirements of the plant; (4) to evaluate operational experience and communicate lessons learned, and to impose plant-specific or industry-wide actions if necessary; (5) to conduct research; and (6) to hold hearings to address safety concerns related to nuclear plant operations.

How is the NRC organized?

At the top of the NRC structure are five NRC Commissioners, including the chairperson. Reporting to the Commissioners is the NRC staff and the Advisory Committee on Reactor Safeguards (ACRS). Within the NRC staff, under the Executive Director of Operations, there are several departments, including four geographical regions, the office of Nuclear Reactor Regulation, the office of New Reactors, and the office of Nuclear Regulatory Research.

What is the ACRS?

The ACRS is an advisory group of foremost experts, independent of the NRC staff, who report directly to the Commission, with four primary objectives: (1) to review safety studies, (2) to advise the Commission on safety standards, (3) to review generic safety topics, and (4) to advise on radiation protection.

How are the regulations organized?

At the top of the hierarchy of regulations is the Code of Federal Regulations (CFR), for nuclear power the CFRs of interest are title 10, referred to as 10 CFR, in particular:

- 10 CFR Part 20 "Standards for Protection Against Radiation"
- 10 CFR Part 21 "Reporting of Defects and Noncompliance"

- 10 CFR Part 50 "Domestic Licensing of Production and Utilization Facilities"
- 10 CFR Part 100 "Reactor Site Criteria"

To what extent do the CFRs affect engineering?

The CFRs are the foundations over which the plant's SAR, codes, standards, and procedures are developed. They permeate every aspect of engineering. A good example is 10 CFR section 50.72 "Immediate Notification Requirements for Operating Nuclear Power Reactors." This CFR section has specific requirements regarding reporting of conditions and activities to the NRC Operations Center. Certain conditions must be reported in as little as 4 hours.

What are the upper-level safety requirements?

10 CFR Part 50 contains Appendix A, which spells out upper-level general design criteria (GDC). The following GDCs are of particular interest in the design and qualification of SSCs:

- Criterion 1 "Quality Standards and Records"
- Criterion 2 "Design Bases for Protection Against Natural Phenomena"
- Criterion 3 "Fire Protection"
- Criterion 4 "Environmental and Missile Design Bases"
- Criterion 5 "Sharing of Structures, Systems, and Components"
- Criterion 13 "Instrumentation and Controls"
- Criterion 14 "Reactor Coolant Pressure Boundary"
- Criterion 15 "Reactor Coolant System Design"
- Criterion 16 "Containment Design"
- Criterion 17 "Electric Power Systems"
- Criterion 18 "Inspection and Testing of Electric Power Systems"
- Criterion 19 "Control Room"
- Criterion 21 "Protection System Reliability and Testability"
- Criterion 22 "Protection System Independence"
- Criterion 23 "Protection System Failure Modes"
- Criterion 24 "Separation of Protection and Control Systems"
- Criterion 30 "Quality of Reactor Coolant Pressure Boundary"
- Criterion 34 "Residual Heat Removal"
- Criterion 35 "Emergency Core Cooling"
- Criterion 38 "Containment Heat Removal"
- Criterion 41 "Containment Atmosphere Cleanup"
- Criterion 44 "Cooling Water"

- Criterion 50 "Containment Design Basis"
- Criterion 53 "Provisions for Containment Testing and Inspection"
- Criterion 54 "Piping Systems Penetrating Containment"
- Criterion 55 "Reactor Coolant Penetrating Containment"
- Criterion 56 "Primary Containment Isolation"
- Criterion 57 "Closed System Isolation Valves"
- Criterion 60 "Control of Releases of Radioactive Materials"
- Criterion 61 "Fuel Storage and Handling and Radioactivity Control"

Other 10 CFR 50 appendices of importance to the subject of this book are Appendix B "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"; Appendix J "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors"; and Appendix R "Fire Protection Program for Nuclear Power Facilities."

What is the next level of regulation, below the CFRs?

At the next engineering level is the Standard Review Plan (SRP) which details the engineering methods and criteria to be followed in designing, qualifying, and operating the plant. Each plant commits to meet the SRP and documents its compliance in the SAR. Plants have a Preliminary Safety Analysis Report at the construction permit stage, and a Final Safety Analysis Report (FSAR) at the operating license stage. The commitments made in the plant FSAR will guide the engineering activities throughout the plant lifetime. New plants have a preapproved design basis, captured in a Design Control Document, which also follows the SRP.

The SRP is (US Nuclear Regulatory Commission Regulation) NUREG-0800 "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants." Until 1975 it was published as Regulatory Guide (RG) 1.70 "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants." The SRP describes the methods and criteria that are acceptable for the design, construction, operation, inspection, and repair of nuclear power plants. Regarding SSCs, the SRP is in many ways a technical guide to preparing project-specific engineering design, analysis, and qualification procedures. The SRP is subdivided into 19 chapters, and each chapter addresses a specific engineering discipline. The 19 chapters are:

- Chapter 1: Introduction and Interfaces
- Chapter 2: Site Characteristics and Site Parameters
- Chapter 3: Design of Structures, Components, Equipment, and Systems