Companion Guide to the ASME Boiler & Pressure Vessel Code

Criteria and Commentary on Select Aspects of the Boiler & Pressure Vessel and Piping Codes
Third Edition

VOLUME 1

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ASME PRESS

ISBN: 978-0-7918-0269-4

The Library of Congress has cataloged the previous edition as follows:

Companion guide to the ASME boiler & pressure vessel code/editor,
K. R. Rao. — 2nd ed
p. cm.
Includes bibliographical references and index
ISBN 0-7918-0218-3 (v. 1) — ISBN 0-7918-0219-1 (v. 2) — ISBN 0-7918-0220-5 (v. 3)
1. Steam-boilers—Standards. 2. Pressure vessels—Standards. I. Rao, K. R., 1933–

TJ289.R36 2006
621.1 83021873–dc22
2005032521

Cover photos:
Volumes 1 & 2: Cover designer: Paul Moran
Paul Nehrenz, photographer; Courtesy of Entergy Corporation:
Volume 1: Ninemile Point; Volume 2: Waterford Nuclear Plant
Volume 3: Cover designer: Rorshach Design.
Background image: Kashiwazaki-Kariwa Nuclear Power Station, Courtesy of Tokyo Electric Power Company, Japan. Foreground image: UK Slugcatcher Vessel Complex, Courtesy of Whessoe Oil and Gas Ltd, Darlington, UK, Top left: Pt. Lepreau plant Provided by NB Power Nuclear, Canada. Bottom left: Column Pressure Vessel (10 bar), Courtesy of Richard Ducros Corporation, France.

DEDICATION TO THE FIRST EDITION

THIS MONUMENTAL EFFORT IS DEDICATED TO THE ASME PRESSURE VESSELS AND PIPING DIVISION AND TO TWO SIGNIFICANT CONTRIBUTORS TO THE DEVELOPMENT OF THE DESIGN-BY-ANALYSIS CON-STRUCTION RULES IN THE MODERN ASME CODE.

This two-volume compendium dedication is not the first recognition of the achievements of Bernard F. Langer and William E. Cooper. The Bernard F. Langer Nuclear Codes and Standards Award, established in 1977, provides a posthumous and lasting tribute to one of these contributors, an intellectual giant who was instrumental in providing the leadership and statesmanship that was essential to the creation of construction rules for nuclear vessels and related equipment. William E. Cooper, the first recipient of the Bernard F. Langer Nuclear Codes and Standards Award, is another intellectual giant instrumental in the creation of the modern ASME Code. In addition, Dr. Cooper acted in a number of ASME Codes and Standards leadership positions. It was my pleasure to join many of my colleagues in April 2001 for the presentation to Dr. Cooper of the ASME President's Award from the 120th President of ASME International, William A. Weiblen. That most prestigious award recognized a lifetime of achievement in ASME and, in particular, in ASME Code activities.

Bernie Langer and Bill Cooper were essential in both the development of the modern ASME Code and in the creation of the forums for technical information exchange that support the Code rules. The publication of these two volumes by ASME International is a legacy of that duality. These volumes continue a long and productive relationship between the development of the modern ASME Code and the technical exchanges on pressure vessel and piping technology sponsored by the ASME Pressure Vessels and Piping Technical Division. This process of technical information exchange, through conference paper and panel presentations, and through refereed paper publication, is an essential step in the reduction to standard practice, standard practice that is eventually embodied in the rules of the ASME Code. Information exchange at technical conferences and in technical publications goes hand in hand with the deliberations of ASME Code bodies.

This relationship goes back to the pivotal events leading up to the development of the modern ASME Code — the appointment of the Special Committee to Review Code Stress Basis in the late 1950s. The principles formulated by that group became the basis for Section III and Section VIII, Division 2 (design by analysis) of the Code. These basic principles were published by ASME in 1968 under the title "Criteria of the ASME Boiler and Pressure Vessel Code for Design by Analysis in Sections III and VIII, Division 2." At the same time that the work of the Special Committee to Review Code Stress Basis was nearing fruition, leaders in the field of pressure vessel design, including Bernie Langer and Bill Cooper, recognized that an improved forum for fundamental technical information exchange was needed. The need eventually led to the formation of an ASME technical division, the Pressure Vessel and Piping (PVP) Division, in 1966.

Many of us who became involved in the PVP Division in the early years were drafted by the leaders in the field to help prepare a compendium of the technical information on pressure vessel and piping technology. The Decade of Progress volumes, as they were known then, were published by ASME in the early 1970s, covering the most significant contributions to pressure vessel and piping design and analysis; materials and fabrication; and operations, applications, and components. The Decade of Progress volumes should be considered the antecedents of these two volumes. Both sets of volumes should be considered as integral parts of the technical literature supporting the Code and the Criteria document.

The PVP Division has acted with great vigor over the years to continue to provide the technical forums needed to support improvements in the modern ASME Code. This year marks the Division's 35th anniversary. When I first became involved in PVP Division activities, the second year had just been completed, with Vito Salerno as the second Chair of the Division Executive Committee. Dana Young had been the first Chair, during 1966–1967, and Gunther Eschenbrenner was ready to become the third Chair, for the 1968–1969 year. Planning was well underway for the first International Conference on Pressure Vessel Technology (ICPVT), scheduled for Delft, the Netherlands, in the following year. The plan was to hold such an international conference every four years, with the Secretariat rotating between Europe (1969), the United States (San Antonio, 1973), and Asia (Tokyo, 1977). Nine of these international conferences have now been held, the most recent in Sydney, Australia, in April 2000.

At the same time, initial planning for the First U.S. National Congress on Pressure Vessels and Piping, to be held every four years in the United States, was also underway. It was my privilege to be the Technical Program Chair for the Second U.S. National Congress on PVP in 1975 in San Francisco, and the Conference Chair for the Third U.S. National Congress on PVP in 1979, also in San Francisco. In addition, the activity within the PVP Division was such that we cosponsored ASME technical conferences with the Materials Division, the Nuclear Engineering Division, and the Petroleum Division in alternate years. This has since led to the annual PVP Conference, the most recent being PVP 2001 in Atlanta, Georgia, in July 2001.

The paper flow from the technical conferences and the network of contributors for the Decade of Progress volumes eventually led to the creation of the *ASME Transactions Journal of Pressure Vessel Technology* in late 1973, only seven years after formation of the Pressure Vessel and Piping Technical Division. Dr. Irwin Berman was its first Senior Technical Editor, with two Technical Editors representing the PVP Division and the Petroleum Division. Once again, I consider it a privilege to have been selected as the Technical Editor for the PVP Division, later becoming the Senior Technical Editor in 1978. The Journal and the technical conferences have provided robust mechanisms for the needed technical information exchange.

But ASME Code rules and the associated technical information exchange is not enough. In one of the very early issues (November 1974) of the *Journal of Pressure Vessel Technology*, two articles were published on the duty and responsibility of engineers and their engineering societies to address public concerns about the safety and reliability of power plants. One, by Bernie Langer, was titled "The Role of the Engineering Societies in Obtaining Public Acceptance of Power Plants." The other, by Bill Cooper, was titled "Nuclear — Pressure Vessels and Piping — Materials: Where to Next." Both articles clearly identified the additional commitment that we all share to bring sound information to the attention of the general public and to policymakers in federal, state, and local jurisdictions. In the almost three decades since the publication of those two articles, this commitment has been extended, as the reach of ASME International, the ASME Boiler and Pressure Vessel Code, and the PVP Division covers the entire world. We owe a debt of gratitude to these two giants, and these two volumes represent a "down payment" on that debt.

Robert E. Nickell, Ph.D.
1999–2000 President

William E. Cooper, Ph.D, P.E.

Acknowledgements to the First Edition

The editor is indebted to several individuals and organizations in the preparation of this two-volume book. Some of them are identified for their assistance in completion of this effort. My thanks are to all of the thirty-nine contributors whose dedicated efforts made this possible by their singular attention to detail, even while they succinctly conveyed the voluminous information.

I wish to thank Dr. Jack Ware, Pressure Vessels and Piping Division who suggested this effort. My thanks are in particular to Martin D. Bernstein who had from the start of this project been my inspiration to rally around during several ups and downs. I also thank Dr. Robert E. Nickell for his encouragement to see the end of the tunnel.

This effort would not have been possible but for the encouragement and support provided by my employer, Entergy Operations Inc., and in particular by Frederick W. Titus, William R. Campbell, John R. Hamilton, Willis F. Mashburn, Raymond S. Lewis, Jaishanker S. Brihmadesam, Brian C. Gray, and Paul H. Nehrenz.

My special thanks to Professor Dr. Robert T. Norman, University of Pittsburgh, for the untiring pains he had taken in training me to undertake efforts such as these — from their very initiation to their logical conclusion.

This unique two-volume publication, which Dr. Frederick Moody aptly called a "monumental effort," would have never taken off had it not been for the vision and sustained support provided by the staff of ASME Technical Publishing. My thanks to them for their support.

Finally, all of this saga-type effort, spread over three years, would have never been possible had it not been for the constant encouragement and untiring support provided by my wife, Dr. Indira Rao, that included all of the sundry chores associated with this project. In addition, I wish to thank other members of my family, Uma and Sunder Sashti, and Dr. Ishu V. Rao, for their zealous support.

Acknowledgements to the Second Edition

This second edition following the success of the first edition has an enlarged scope including the addition of a third volume. This warranted the addition of several contributors who are all experts in their respective specialties. The editor appreciates their contributions, as well as the continued support of the contributors from the first edition.

Editor intends to once again thank Entergy Operations for their continued support. Thanks are especially due to Dr. Indira Rao whose support in several capacities made this voluminous effort possible. My thanks are to the staff of ASME publishing for their continued zeal and support.

ACKNOWLEDGEMENTS TO THE THIRD EDITION

This third edition follows the unprecedented success of the previous two editions.

As mentioned in the first edition, this effort was initiated with the 'end user' in mind. Several individuals and a few organizations had provided support ever since this effort started.

In the second edition the success of the first edition was enlarged in scope with the addition of a third volume, with experts in their respective specialties to contribute chapters they authored.

In response to the changing priorities of Boiler and Pressure Vessel (B&PV) industry and global use of ASME B&PV Codes and Standards the scope and extent of this edition has increased. The result of the current effort is in a 2,550 page book spread in three volumes.

The editor pays homage to the authors Yasuhide Asada, Martin D. Bernstein, Toshiki Karasawa, Douglas B. Nickerson and Robert F. Sammataro who passed away and whose expertise enriched the chapters they authored in the previous editions.

This comprehensive Companion Guide with multiple editions spanning over several years has several authors contributing to this effort. The editor thanks authors who had contributed to the previous editions but did not participate in the current edition and they are Tom Ahl, Domenic A. Canonico, Arthur E. Deardorff, Guy H. Deboo, Jeffrey A. Gorman, Harold C. Graber, John Hechmer, Stephen Hunt, Yoshinori Kajimura, Pao-Tsin Kuo, M. A. Malek, Robert J. Masterson, Urey R. Miller, Kamran Mokhtarian, Dennis Rahoi, Frederick A. Simonen, John D. Stevenson, Stephen V. Voorhees, John I. Woodworth and Lloyd W. Yoder.

The editor appreciates the effort of the continuing contributors from the previous editions, who had a remarkable influence on shaping this mammoth effort, few of them from the very beginning to this stage. The editor gratefully acknowledges the following authors Kenneth Balkey, Warren Bamford, Uma Bandyopadhyay, Jon E. Batey, Charles Becht IV (Chuck), Sidney A. Bernsen, Alain Bonnefoy, Marcus N. Bressler, Marvin L. Carpenter, Edmund W. K. Chang, Kenneth C. Chang, Peter Conlisk, Joel G. Feldstein, Richard E. Gimple, Jean-Marie Grandemange, Timothy J. Greisbach, Ronald S. Hafner, Geoffrey M. Halley, Peter J. Hanmore, Owen F. Hedden, Greg L. Hollinger, Robert I. Jetter, Guido G. Karcher, William J. Koves, John T. Land, Donald F. Landers, Hardayal S. Mehta, Richard A. Moen, Frederick J. Moody, Alan Murray, David N. Nash, W. J. O'Donnell, David E. Olson, Frances Osweiller, Thomas P. Pastor, Gerard Perraudin, Bernard Pitrou, Mahendra D. Rana, Douglas K. Rodgers, Sampath Ranganath, Roger F. Reedy, Wolf Reinhardt, Peter C. Riccardella, Everett C. Rodabaugh, Robert J. Sims Jr., James E. Staffiera, Stanley Staniszewski, Richard W. Swayne (Rick), Anibal L.Taboas, Elmar Upitis and Nicholas C. Van Den Brekel.

Similarly the editor thanks the contribution of authors who joined this effort in this third edition. Sincerity and dedication of the authors who joined in this effort is evident from two instances — *in one case, a contributor hastened to complete his manuscript before going for his appointment for heart surgery! In another case, when I missed repeatedly a correction made by a contributor, he never failed to draw my attention to the corrections that I missed!*

Thus, the editor wishes to appreciate efforts of authors who joined in this edition and worked zealously to contribute their best for the completion of this 'saga'. The authors are Joseph F. Artuso, Hansraj G.Ashar, Peter Pal Babics, Paul Brinkhurst, Neil Broom, Robert G. Brown, Milan Brumovsky, Anne Chaudouet, Shin Chang, Yi-Bin Chen, Ting Chow, Howard H. Chung, Russell C. Cipolla, Carlos Cueto-Felgueroso, K. B. Dixit, Malcolm Europa, John Fletcher, Luc H. Geraets, Stephen Gosselin (Steve), Donald S. Griffin, Kunio Hasegawa, Philip A. Henry, Ralph S. Hill III, Kaihwa Robert Hsu, D. P. Jones, Toshio Isomura, Jong Chull Jo, Masahiko Kaneda, Dieter Kreckel, Victor V. Kostarev, H. S. Kushwaha, Donald Wayne Lewis, John R. Mac Kay, Rafael G. Mora, Dana Keith Morton, Edwin A. Nordstrom, Dave A. Osage, Daniel Pappone, Marty Parece, Michael A. Porter, Clay D. Rodery, Wesley C. Rowley, Barry Scott, Kaisa Simola, K. P. Singh (Kris), Alexander V Sudakov, Peter Trampus, K. K. Vaze, Reino Virolainen, Raymond (Ray) A. West, Glenn A. White, Tony Williams.

The editor thanks Steve Brown of Entergy Operations for his help in the search for expert contributors for this edition.

This edition was initiated by me in August 2006 and has taken over 3000 hours of computer connection time. My thanks are especially to my wife, Dr. Indira Rao whose sustained support for this effort and participation in several chores related to editing. In addition, I appreciate her tolerating my working on it during a 4-month overseas vacation.

The editor thanks the staff of ASME Technical Publications for their unstinted zeal and support in aiming at this publication's target of 'zero tolerance' for 'errors and omissions'.

Finally, the editor thanks all of you, readers and users of this 'Companion Guide' and hopes it serves the purpose of this publication.

CONTRIBUTOR BIOGRAPHIES

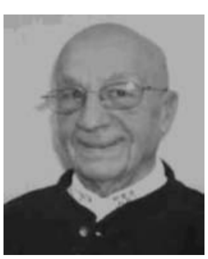
AHL, THOMAS J.

Thomas J. Ahl earned a B.S.C.E. in 1960 and M.S.C.E. in 1961 from University of Wisconsin. He is a Registered Structural and Professional Engineer in Illinois. He held the position of Principal Engineer in Nuclear & Pressure Vessel Design Department, Chicago Bridge & Iron Co., Plainfield, IL, (1961–1998), and was engaged in design and analysis of nuclear related vessels and structural components. Ahl was a Member of ANSI Working Group ANS-56.8 that prepared the ANSI/ANS-56.8-1981—Containment System Leakage Testing Requirements standard.

Ahl is a Member of ASCE, Member of ASCE Hydropower Development Committee, and Conventional Hydropower Subcommittee. He served as Co-Chair of the Task Committee preparing the publication "Manual of Practice for Steel Penstocks ASCE Manual No. 79," Vice-Chair-ASCE Committee preparing the "Guidelines for Evaluating Aging Penstocks," and member of ASCE Hydropower Committee preparing "Civil Engineering Guidelines for Planning and Design of Hydroelectric Developments."

Two of these publications received the ASCE Rickey Award Medal in 1990 and 1994. Thomas Ahl is a member of the Peer Review Group to Sandia National Laboratories and the U.S. Nuclear Regulatory Structural Engineering Branch for the Safety Margins for Containment's Research Program, 1980–2001.

ARTUSO, JOSEPH F.



Joseph F. Artuso is the CEO of Construction Engineering Consultants, Inc. He has over 40 years experience in developing and managing quality control inspection and testing programs for construction materials. He is also actively involved in the Code and Standards writing bodies of ACI and ASME. Mr. Artuso earned a B.S. in Civil Engineering at Carnegie Institute of Technology in 1948 and became a Level III Inspection Engineer at the National Council of Engineering Examiners in 1975. He is a registered Professional Engineer in the states of Pennsylvania, Ohio, New York, Florida, Maryland and West Virginia, as well as being registered as Quality Control Engineer in state of California. His memberships in national committees include A.S.C.E. (Task committee on Inspection Agencies), A.C.I (Committees 214, 304 and 311), A.N.S.I (N-45-3.5 Structural Concrete and Steel), A.S.M.E. (Committee 359 (ASME Sec. III, Div. 2) Construction Materials and Exam.), ACI-ASME (Committee on Concrete Pressure Components for Nuclear Service), ASTM, and NRMCA. He was a contributing editor of McGraw-Hill "Concrete Construction Handbook". Mr. Artuso was the Director of Site Quality Control for the Duquesne Light Company, Beaver Valley, Unit 2. He also supervised construction quality control activities on many nuclear power plants during the period of high construction activity from the 1970's to 1980's.

ASADA, YASUHIDE

Late Dr. Yasuhide Asada was Professor Emeritus of The The University of Tokyo. He has been an internationally renowned scientist in the area of Elevated Temperature Design technology as well as plasticity, creep and creep and fatigue. He was an author of over 150 scientific/technical publications with respect to the technical area of his specialty. His contributions have been awarded by JSME, ASME, ASTM and other engineering organizations.

After six years of field experience at Mitsubishi Heavy Industries Ltd., he was invited to be a faculty member of School of Engineering, The University of Tokyo in 1969. He has been a Professor of Applied Mechanics in the Mechanical Engineering Department since 1980 and retired from the university in 1999 due to the university retirement age of 60.

He was a leader of structural integrity administration at METI on Japanese Nuclear Power plants and components and a leader of Japanese Codes and Standards activity for nuclear and non-nuclear facilities and Japanese representative for ISO/TC11 Boilers and Pressure Vessels where he proposed a new concept of IPEC for international standards.

He has been contributing in international codes and standards activity and was a member of SC. III of ASME B&PV Code Committee, Board on Nuclear Codes and standards of ASME and was a member of ASME Council on Codes and Standards as well as to the Board on Codes and Standards Technology Institute. He has contributed to JSME Power generation Code Committee was a member since 1998 and was chair for first four years.

In scientific activity, he chaired International Council on Pressure Vessel Technology (ICPVT) for 1996 to 2000 and chaired Asian and Oceanic regional Committee (AORC) of ICPVT since 1996. He was a Japanese representative member of International Creep Conference and chaired 7th International creep Conference 2000. Dr. Asada was the recipient of Bernard F. Langer Award and the ASME Dedication Service Award. Professor Emeritus Yasuhide Asada passed away on Nov. 23rd, 2005.

ASHAR, HANSRAJ, G.

Mr. Ashar has a Master of Science degree in Civil Engineering from the University of Michigan. He has been working with the Nuclear Regulatory Commission for the last 35 years as a Sr. Structural Engineer. Prior to that Mr. Ashar has worked with a number of consultants in the U.S. and Germany designing Bridges and Buildings. Mr. Ashar has authored 30 papers related to structures in nuclear power plants.

Mr. Ashar's participation in National and International Standards Organization includes Membership of the NSO and INSO Committees such as American Institute of Steel Construction (AISC), Chairman of Nuclear Specification Committe (January 1996 to March 2008), (AISC/ANSI N690); Member of Building Specification Committee, and Corresponding of Seismic Provisions Committee.

Mr. Ashar's professional activities with The American Concrete Institute (ACI) 349 Committees include Member of the Main committee, Subcommittee 1 on General Requirements, Materials and QA, and Subcommittee 2 on Design. His professional activities also include American Society of Mechanical Engineers (ASME), Corresponding Member, Working Group on lnservice Inspection of Concrete and Steel Containments (Subsections IWE and IWL of ASME Section XI Code), Member, ASME/ACI Joint Committee on Design, Construction, Testing and Inspection of Concrete Containments and Pressure Vessels; Member, RILEM Task Committee 160-MLN: Methodology for Life Prediction of Concrete Structures in Nuclear Power Plants; Member, Federation Internationale du Beton (FIB) Task Group 1.3: Containment Structures, and Consultant to IAEA on Concrete Containment Database (2001 to 2005).

Mr Ashar is a Professional Engineer in the State of Ohio and State of Maryland; Fellow, American Concrete Institute; Fellow, American Society of Civil Engineers; Professional Meer – Posttensioning Institute. Mr. Ashar is a Peer Reviewer of the Papers to be published in ASCE Material Journal, Nuclear Engineering and Design (NED) Periodicals and ACI Material Journal.

BABICS, PETER PAL

Peter Pal Babics graduated as Mechanical Engineer from Bánki Donát Polytechnic Institute (BDPI) of Budapest, Hungary in 1975 and obtained an MSc equivalent degree in 1980 as ME at the Technical University of Miskolc. He post graduated as Welding Engineer at the Technical University of Budapest, and European Welding Engineer (EWE) at the Austrian Institute of Material Science and Welding Technology (SZA), Wien. After graduating he worked as pressure vessel designer and technologist of welding material production (1975–80). From 1980 he directed pressure vessel and pipelines construction in the oil refinery and gas industry. Since 1990 he has been working as member of the Hungarian Atomic Energy Authority (HAEA). His main activity is licensing ISI programs, welding procedures, repairs and replacements of safety related equipment of NPP of the Nuclear Safety Directorate (NSD). In 1997 he took part in a six-month training at the US NRC to study the ASME B&PV CODE Section XI regulatory application and practice. Since 2003 he has been responsible for the ISI System Qualification of the NSD. Since March 2007 he works as leader of Section of Component Supervision in Licensing Department of NSD.

Since 1996 he has been member of the Hungarian Association of Welding and Non-destructive Testing Organisation and his duty is the education, training and examination of metal welders. He has taken part and given presentations in more than 30 conferences and workshop. He is the author of several Hungarian Regulatory Guides.

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Kenneth R. Balkey is currently a Consulting Engineer in Nuclear Services, with Westinghouse Electric Company in Pittsburgh, PA with over 36 years of service in the nuclear power industry. Mr. Balkey provides consultation and advises technology developments related to Codes and Standards and critical asset protection initiatives. He performed and directed reliability and risk evaluations for nuclear and non-nuclear structures, systems and components over his lengthy career. He has produced more than 100 publications and documents relating to risk evaluations of the integrity of piping, vessels and structures, and the performance of components using state-of-the-art probabilistic assessment techniques.

Mr. Balkey is vice chair, ASME Codes and Standards Board of Directors (June 2008 – June 2011), a member of the ASME Board on Nuclear Codes and Standards, past vice president, Nuclear Codes and Standards (June 2005 – June 2008), and past chair, ASME Board on Nuclear Codes and Standards (June 2005 – June 2008). Mr. Balkey also served as a senior technical advisor to the ASME Innovative Technologies Institute LLC, providing consultation on the development of guidance for Risk Analysis and Management for Critical Asset Protection (RAMCAP™) and working with the U.S. Department of Homeland Security. His honors include ASME's Dedicated Service Award (1991), the Bernard F. Langer Nuclear Codes and Standards Award (2002), the Melvin R. Green Codes and Standards Medal (2008), and several other awards from ASME, Westinghouse, and other institutions. Mr. Balkey earned B.S. and M.S. degrees in Mechanical Engineering at the University of Pittsburgh. Mr. Kenneth R. Balkey is a Registered Professional Engineer.

BAMFORD, WARREN

Warren Bamford has been a member of Section XI since 1974, and now serves as Chairman of the Subgroup on Evaluation Standards, whose charter is to develop and maintain flaw evaluation procedures and acceptance criteria. He is a member of the Executive Committee of Section XI, and was also a charter member of the ASME Post Construction Committee, whose goal is to develop inspection, evaluation and repair criteria for non-nuclear plants. He has taught a course on the Background and Technical Basis of the ASME Code, Section III and Section XI.

Warren has been educated at Virginia Tech, Carnegie Mellon University, and the University of Pittsburgh.

Warren's research interests include environmental fatigue crack growth and stress corrosion cracking of pressure boundary materials, and he has been the lead investigator for two major programs in this area. He was a charter member of the International Cooperative Group for Environmentally Assisted Cracking, which has been functioning since 1977.

Warren Bamford has been employed by Westinghouse Electric since 1972, and now serves as a consulting Engineer. He specializes in applications of fracture mechanics to operating power plants, with special interest in probabilistic applications. Over 80 technical papers have been published in journals and conference proceedings.

BANDYOPADHYAY, UMA S.

Bandyopadhyay received his BSME from Jadavpur University (1970), Calcutta, India, MSME from the Polytechnic Institute of Brooklyn (1974). He is a registered Professional Engineer in the states of New York, New Jersey, Connecticut, Massachusetts, Virginia, Wyoming and District of Columbia. He has 28 years of extensive experience in design, engineering and manufacturing of pipe supports and pipe support products for Water Treatment and Waste Water Treatment Facilities, Oil Refineries, Co-generation, Fossil and Nuclear Power Plants. Bandyopadhyay is currently employed by Carpenter and Paterson, Inc. as Chief Engineer and works as a consultant and Registered Professional Engineer for affiliate Bergen-Power Pipe Supports, Inc. Prior to his current employment, he held the positions of Design Engineer (1977–1980), Project Engineer (1980–1986) and Chief Engineer (1986–1992) with Bergen-Paterson Pipesupport Corp. Bandyopadhyay is a member, Working Group on Supports (Subsection NF), since 1993; was an alternate member, Subsection NF (1986–1993). He is also an alternate member, Manufacturer's Standardization Society (MSS), Committee 403-Pipe hangers (MSS-SP-58, 69, 89, 90 and 127) since 1992.

BATEY, JON E.

Jon Batey has been a member of ASME Subcommittee V since 1995 and has served as Chairman since 2002. Jon has served on various sub-tier committees of Subcommittee V since 1990 and currently is a member of the Subgroup on Volumetric Examination Methods, the Subgroup on General Requirements, Personnel Qualifications and Interpretations, the Working Group on Radiography, and the Working Group on Acoustic Emission. He is also a member of the ASME Boiler and Pressure Vessel Standards Committee plus its Honors and Awards Committee, and the ASME Post Construction Standards Committee and its Subcommittee on Inspection Planning. Jon was also a member of the ASME B-16 Standards Committee from 1979 to 1993.

Jon is the Global Inspection Leader for The Dow Chemical Company in Freeport, TX. In his current role, Jon is responsible for inspection performed by Dow or third-party inspectors at supplier fabrication shops. He received a B.S. in Physics from Texas State University. His certifications include ASNT Level III Certificates in Radiography, Ultrasound, Liquid Penetrant, Magnetic Particle, Visual and Leak Test Methods.

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Dr. Becht is a recognized authority in pressure vessels, piping, expansion joints, and elevated temperature design. He is President of Becht Engineering Co. Inc, a consulting engineering company providing services to the process and power industries (www.becht.com, www.bechtns.com for the nuclear services division, and www.tech-training.info for technical training); President of Becht Engineering Canada Ltd.; President of Helidex, LLC (www.helidex.com); and Director of Sonomatic Ltd. (also dba Becht Sonomatic, www.vsonomatic.com) a NDE company that provides advanced ultrasonic imaging. Chuck was previously with Energy Systems Group, Rockwell International and Exxon Research and Engineering where he was a pressure equipment specialist. He received a PhD from Memorial University in Mechanical Engineering (dissertation: Behavior of Bellows), a MS from Stanford University in Structural Engineering and BSCE from Union College, New York. Chuck is a licensed professional engineer in 16 states and provinces, an ASME Fellow since 1996, recipient of the ASME Dedicated Service Award in 2001, and has more than 60 publications including the book, *Process Piping: The Complete Guide to ASME B31.3*, and five patents.

Dr. Becht is Chair of the ASME B31.3, Process Piping Committee; Chair (founding) of the Post Construction Subcommittee on Repair and Testing (PCC), and member of other ASME Committees including the Post Construction Standards Committee (past Chair); Post Construction Executive Committee (past Chair); B&PV Code Subcommittee on Transport Tanks; B&PV Code Subgroup on Elevated Temperature Design (past Chair); B31 Code for Pressure Piping Standards Committee; B31 Mechanical Design Committee; B31 Executive Committee; and is a past member of the Board on Pressure Technology Codes and Standards; the B&PV Code Subcommittee on Design; and the B&PV Code TG on Class 1 Expansion Joints for liquid metal service. He is a member of ASTM Committee F-17, Plastic Piping Systems Main Committee; and the ASME PVP Division, Design and Analysis Committee.

BERNSEN, SIDNEY A.

Dr. Bernsen, earned his B.S.M.E in 1950, M.S.M.E in 1951, and Ph.D. in 1953, from Purdue University. He has been involved in nuclear power activities for over 50 years, initially at Argonne National Laboratory and subsequently with Bechtel Corporation. At Bechtel he held a variety of positions including Chief Nuclear Engineer and Manager of Quality Assurance for Bechtel Power Corporation. Since retirement from Bechtel, after more than 30 years, he has provided consulting services and has remained active in ASME Nuclear Codes and Standards. Dr. Bernsen served as the initial Chair of the ASME Committee on Nuclear Quality Assurance (NQA) and is now an honorary member of the NQA committee.

Dr. Bernsen, was a founding member of the ASME Board on Nuclear Codes and Standards (BNCS) and has continuously served on BNCS since its inception. He was the initial Chair of the ASME Committee on Nuclear Risk Management and completed his second term in 2004. During his tenure, the committee completed and published the first issue of this Standard in April 2002 and the first addendum was published in December 2003. Through his long career, including extensive involvement in licensing and safety, as well as his work in coordinating the development and approval of the Nuclear Risk Management Standard, he has obtained valuable insight into nuclear risk related activities and how they are being and may be applied in the future to ASME Codes and Standards. Dr. Bernsen, an ASME Fellow, has been involved in Codes and Standards for over 35 years.

BERNSTEIN, MARTIN D.

Mr. Bernstein was involved in the design and analysis of steam power equipment since joining Foster Wheeler Energy Corporation in 1960. Retired in 1996, he continued to serve as a consultant to Foster Wheeler and as their representative on the ASME Boiler and Pressure Vessel Committee, on which he had served for more than 25 years. He was Vice Chair, Subcommittee on Power Boilers, Chair, Subcommittee on Safety Valve Requirements, a member of the Main Committee (Standards Committee) and past Chair of Subgroup General Requirements and the Subgroup Design of the Subcommittee on Power Boilers. Since 1986 he and Lloyd Yoder taught a two-day course on Power Boilers for the ASME Professional Development Department. In 1998, ASME Press published *Power Boilers—A Guide to Section I of the ASME Boiler and Pressure Vessel Code* that Bernstein and Yoder developed from their course notes.

Mr. Bernstein was active for many years in ASME's PVP Division. He was also author and editor of numerous ASME publications, including journal articles on ASME design criteria, ASME rules for safety valves, flow-induced vibration in safety valve nozzles, and tubesheet design. Mr. Bernstein obtained a B.S. and M.S. in civil engineering from the Columbia School of Engineering and Applied Science. He was elected an ASME Fellow in 1992, received the ASME Dedicated Service Award in 1994, and was awarded the ASME J. Hall Taylor Medal in 1998. He was a Registered Professional Engineer in New York State. Mr. Bernstein passed away in 2002.

BONNEFOY, ALAIN

Alain Bonnefoy graduated from the INSA in Lyon-France. He began his career at CETIM (French Technical Center of Mechanical Industries) in R & D in the field of Pressure Vessel, Piping and Boilers. In 1976 he held the position of Department manager for the design and analysis of structure and components. Mr. Bonnefoy has published more than 20 papers in structural design particularly seismic analysis, mounded vessels and buried piping areas. Since 2001 he is the Technical Manager of the SNCT (French Pressure Equipment Manufacturer's Association) that publishes the French Codes of Construction such as CODAP (Vessels), CODETI (Piping), COVAP (Boilers). Alain Bonnefoy is also in charge of the French version of the ASME Section VIII Division 1, currently under preparation.

Mr. Bonnefoy is a Member of the different working groups preparing these codes and is also active in European standardization in the field of piping.

BRESSLER, MARCUS N.

Mr. Bressler is President of M. N. BRESSLER, PE, INC., an engineering consulting firm founded in 1977, specializing in codes and standards, quality assurance, design, fabrication, inspection and failure analysis for the piping, power, petroleum and chemical industries. He has over 54 years of experience. He joined TVA in 1971 as Principal Engineer and was promoted in 1979 to Senior Engineering Specialist, Codes Standards and Materials. He took early retirement in 1988 to open up a private consulting practice. His previous experience was with the US Army (1952) where he served as an Industrial Hygiene Engineer; the Babcock & Wilcox Company(1955), where he held the positions of Engineering Draftsman, Stress Analyst, and Boiler Division Materials Engineer; Gulf & Western Lenape Forge Division (1966) where he became Senior Design Engineer, and Taylor Forge Division (1970) as Product Development Manager. At Lenape Forge he developed a design for a quick-opening manway for pressure vessels and piping that was granted a patent in 1971.

Mr. Bressler began his activities in Codes, Standards and Materials in 1960. He has been a member of the ASME B&PV Standards Committee since 1979 to 2009, and is now a member of the Technical Oversight Management Commitee. He is a member and past Vice Chair of the Committee on Nuclear Certification. He is a member of the Standards Committees on Materials and on Nuclear Power, the subgroup on Design (SCIII), the special working group on Editing and Review (SC III), the Boards on Nuclear Codes and Standards and on Conformity Assessment. He is the Chair of the Honors and Awards Committee (BNCS). Mr. Bressler is a member of the ASTM Committees A-01 and B-02 and many of their subcommittees.

Mr. Bressler holds a BME degree from Cornell University (1952) and an MSME degree from Case Institute of Technology (1960). In 1989 he received a Certificate of Achievement from Cornell University for having pursued a course that, under today's requirements, would have resulted in a Master of Engineering degree. He was awarded the ASME Century Medallion (1980), and became a Fellow of ASME in 1983. He is now a Life Fellow. He received the 1992 ASME Bernard F. Langer Nuclear Codes and Standards Award. and is the 1996 recipient of the ASME J. Hall Taylor Medal. He received the 2001 ASME Dedicated Service Award. He is a Registered Professional Engineer in the State of Tennessee (Retired).

BRINKHURST, PAUL

Mr. Brinkhurst graduated from the University of the South Bank, London in 1971 with a BSc (Hons) in Chemical Technology. After spending a number of years in the Chemical and Mining industries he has worked predominately in the nuclear power generation industry since 1983. Mr. Brinkhurst has been employed by the South African electricity utility Eskom since 1993, working mainly in the areas of inservice inspection, integrity and licensing.

Mr. Brinkhurst's specific activities included application of Sections III and XI of the ASME B&PV Code at Koeberg Nuclear Power Station. His current position is Chief Engineer in the Eskom Nuclear Safety Assurance department.

BROOM, NEIL

Neil Broom has been involved with heat exchanger and pressure vessel design and manufacture for the power generation industry for over 30 years. He is currently employed at PBMR with responsibility for Code related activities.

He serves as a member of Subgroup Strategy and Management Divisions 1 and 2 (SC III), Member of Special Working Group High Temperature Gas Cooled Reactors (SC XI).

BROWN, ROBERT G.

Mr. Brown is a Principal Engineer and Director of Consulting for the Equity Engineering Group in Shaker Heights, Ohio. He has experience as both an owner-user and consultant providing engineering support to refineries and chemical plants worldwide. Mr. Brown uses advanced skills in Finite Element Analysis to provide practical and cost effective solutions to solve design and operational issues related to fixed equipment.

Mr. Brown assisted with the development of API 579 *Fitness-For-Service* and has been a consultant for the PVRC effort to develop the new *ASME, Section VIII, Division 2, Boiler and Pressure Vessel Code,* taking into consideration the latest developments in materials, design, fabrication, and inspection technologies.

Mr. Brown is an active member of the Battelle International Joint Industry Project on the Structural Stress Method for Fatigue Assessment of Welded Structures and performs fatigue assessments/reviews of equipment in cyclic service. Mr. Brown also serves on the ASME Subgroup on Design Analysis and performs code compliance calculations and interpretations for pressure vessels. Mr. Brown is a registered Professional Engineer in the States of Ohio and Pennsylvania.

BRUMOVSKY, MILAN

Dr. Milan Brumovsky finished his M.S. in nuclear physics and RNDr. in experimental physics in Charles University in Prague, Czech Republic, and his PhD. in experimental physics at the Moscow Engineering Physics Institute, Moscow, Russia Federation. Then he joined Research and Development Centre of SKODA Nuclear Machinery in Pilsen, Czech Republic (head of Reactor Component Integrity and Safety) – manufacturer of WWER type reactors for Central Europe. After 35 years he moved to the Nuclear Research Institute Rez in Rez, Czech Republic as project manager. Thus, last year he celebrated 50 years in the nuclear power area.

During his career he was involved in many projects connected with the material and integrity research projects, mainly in the field of material qualification for WWER reactors pressure vessels, radiation damage in structural materials, material testing by standard and large scale test specimens and components, fracture mechanics study and application to components integrity. He worked also in preparation of many regulatory documents within the Interatomenergo organization as well as in the Czech nuclear codes. He was also co-ordinator of several IAEA Co-ordinated research projects in the field of radiation damage and fracture mechanics. He is also an active member in the ASTM E-10 Committee as well as in ASME PVP O&C Committee. Finally, he served as a co-ordinator of the European project for preparation of VERLIFE- "Unified Procedure for Lifetime Assessment of Components and Piping in WWER NPPs during Operation".

CANONICO, DOMENIC A.

Dr. Canonico received his B. S. from Michigan Technological University, M.S. and Ph.D. from Lehigh University. He has over 40 years experience in pressure parts manufacturing. Dr. Canonico is currently employed by ALSTOM POWER facilities in Chattanooga, Tennessee. He is Past Chair of the ASME Boiler Pressure Vessel (BPV) Code Main Committee and a member of the ASME Council on C. & S. and Vice President-elect Pressure Technology, C&S. He is a Fellow in ASME, the American Welding Society (AWS) and the American Society for Metals (ASM). In 1999 Dr. Canonico received the ASME Melvin R. Green C&S Medal. He was the 1994 recipient of the ASME J. Hall Taylor Medal, in 1996 and 1999 respectively he was awarded the Dedicated Service Award., and the ASME Region XI Industry Executive Award. In 1978, 1979, and 1985 respectively AWS awarded him the Dr. Rene Wasserman Award, the James F. Lincoln Gold Medal, and the William H. Hobart Memorial Medal; he was the 1983 Adams Lecturer. He is a member of the State of Tennessee Boiler Rules Board.

He has written over 100 technical papers and given technical talks in U.S., Canada, Mexico, Europe and Asia. He is named in Who's Who in Engineering and Men and Women of Science. Dr. Canonico is an Adjunct Professor at the University of Tennessee, Knoxville and on the Advisory Committee of the School of Engineering, University of Tennessee, Chattanooga.

CARPENTER, MARVIN L.

Marvin L. Carpenter graduated with honors from Michigan Technological University (MTU) with a B.S. in Metallurgical Engineering. He continued at MTU and received his Masters in Metallurgical Engineering in 1974. Since graduating, his career has been focused on welding fabrication and testing in accordance with the ASME Boiler and Pressure Vessel Code. ASME Code Committees first caught his attention in the late seventies and he has remained active in the Code ever since. He serves on the Subcommittee on Welding (IX), Chaired the Subgroup on Brazing (IX) and currently Chairs the Subgroup on Materials (IX).

Mr. Carpenter gained expertise in production welding, brazing, failure analysis, coatings, and material testing while working for major corporations including Westinghouse Electric Corporation, The Trane Company, and Bechtel. His experience ranges from supervising a Welding Engineering Develop group to setting up and operating a materials testing laboratory that performed chemical analysis, mechanical testing, metallography, and welding qualification.

In addition to his extensive materials and welding background, he was granted a patent in 1995 for a GTAW-HW circular welding system. His current position is as a Principal Engineer with a major U.S. company that provides power plant equipment. Mr. Carpenter resides in Pittsburgh, PA with his wife, Denise, and two children, Scott and Michelle.

CHANG, EDMUND W. K.

Edmund W.K. Chang, P.E., received his BSME from the University of Hawaii (UHM), 1969. Mr. Chang is currently employed as the Boiler & Welding Maintenance Engineer with Hawaiian Electric Company, Inc., Power Supply Engineering Department, Honolulu, Hawaii. Mr. Chang's responsibilities include being in-charge of all company boiler condition assessments, and National Board (NB) "R" and "VR" Symbol Stamp repair programs. Mr. Chang is also a NB commissioned O/U Inspector, in charge of in-service and acceptance inspections. He is a AWS Certified Welding Inspector (CWI), in charge of welding program, and the company's NDT Level III in PT and MT in charge of the NDT program.

Mr. Chang's professional affiliations include ASME Membership since 1971; association with ASME Hawaii Section as Chairman 2008–2009, Honors & Awards Committee Chair, Webmaster, Newsletter Editor, and Section Chair 1993–1994; ASNT Hawaii Section Director and Webmaster; AWS Hawaii Section Webmaster; and Chair 1996 of Hawaii Council of Engineering Societies. Mr. Chang is a member of the Department of Mechanical Engineering, UHM, Industry Advisory Board.

Mr. Chang's professional publications include as a lead author of "T91 Secondary Superheater Tube Failures Investigation," 1997, ASME PVP Conference, Orlando, Florida; and "Tangential-Fired Boiler Tube Failures, A Case Study," 2007, EPRI International Conference on Boiler & HRSG Tube Failures, Calgary, Alberta, Canada.

CHANG, KENNETH C.

Dr. Chang is a registered professional engineer and received his Ph.D. in applied mechanics from the Department of Mechanical Engineering, University of California, Berkeley. Through his technical and management career at Westinghouse, he has been involved in the design, analysis, and construction of nuclear power plant systems, structures and components, and ASME Section Code development for more than 34 years. Dr. Chang is a renowned professional in the field of fatigue design, structural dynamics, and aging managing for ASME Class 1 components, and authored over twenty-five technical papers and professional presentations. He is a key participant in the development of the new review and audit process for aging management reviews (AMRs) and aging management programs (AMPs) for license renewal applications (LRAs) at USNRC and conducted several training on the subject. Dd. Chang retired as a branch chief of License Renewal Division responsible for performing on-site audits and reviews for the new LRAs. In that capacity he was also a staff member planning for the guidance documents update, involving industry participation, for more efficient preparation and review of LRAs.

CHANG, SHIN

Dr. Chang received her B.S. in 1983 from the National Tsing Hua University in Taiwan. She continued her study in the USA and earned her MS and Ph.D. in Nuclear Engineering from the University of Illinois at Urbana-Champaign, Illinois. After graduation in 1991, she was employed by the Atomic Energy Council at Taiwan (TAEC). Since then, she has been working at the Department of Nuclear Regulation of TAEC for more than 15 years.

During Dr. Chang's career at TAEC, she has been involved in various regulatory safety reviews and inspection works related to nuclear power plants. She has been section chief of the License Review Section and section chief of the Kuosheng Regulatory Task Force of the Nuclear Regulation Department of TAEC. She is currently the section chief of Chinshan Regulatory Task Force of the Nuclear Regulation Department of TAEC. In this role, she is now responsible for all the regulatory activities, which include resident inspections, periodic inspections, maintenance quality inspections, safety review of measurement recapture power uprate application, safety review of the aging management program that includes Chinshan NPP operational safety.

CHAUDOUET, ANNE

Ms Chaudouet earned a Master of Pure Maths at Paris XIII University in 1974 and then obtained a Mechanical Engineering Degree from ENSMP (Mines) in Paris, France in 1976. The same year, she started her career at Cetim (French Technical Center of Mechanical Industries) in R&D in the field of solid mechanics analysed by the Boundary Element Method (BEM).

Soon after, she became in charge of the team responsible for the development of all software developed at Cetim in the domain of 2D and 3D heat transfer and solid mechanics. In that role she had the direct responsibility for the analyses of components by BEM and of fracture mechanics. In 1984, she became head of the Long Term Research Service involved in more theoretical studies and development of design rules for pressure vessels. In the same year she initiated Cetim's participation in PVRC (Pressure Vessel Research Council).

Since 2003, Ms Chaudouet has been actively involved in ASME Boiler and Pressure Vessel Code organization where she became a member of the Subcommittee on Materials, of SC II/International Material Specifications (currently, Chair) and of SC D/Bolted Flanged Joints. She is also an active member of the ASME/API Joint Committee on Fitness for Service. Ms Chaudouet has published over 30 papers in French and in English in the domain of Boundary Elements, Fracture Mechanics and more recently on Fitness-For-Service. Most of these were presented at International Conferences.

Ms Chaudouet has developed professional courses on these topics. In the domain of pressure equipment she has also given courses on the PED (European Directive).

CHEN YI-BIN

Dr. Chen received his B.S. in Nuclear Engineering from National Tsing-Hua University (Taiwan), M.S. and his Ph.D. in Nuclear Engineering from Massachusetts Institute of Technology, Cambridge, MA.

Dr. Chen has spent his entire career in the field of nuclear energy, beginning in research and development of thermal hydraulics and safety for light water reactors, and then with regulatory body conducting safety inspection, audit and review of design, construction and operation of nuclear power plants in Taiwan.

Dr. Chen has held a number of senior management positions including Deputy Director of the Institute of Nuclear Energy Research (INER) and Department Directors of Planning, Nuclear Technology, Radiation Protection and Nuclear Regulation at Atomic Energy Council (AEC). He has also taught graduate courses in the Nuclear Engineering Department of National Tsing-Hua University for more than 10 years.

CHOW, TING

Mr. Ting Chow, specialized in earthquake engineering application to nuclear power facilities, and has been acting head of Seismic Test/Research Laboratory of Institute of Nuclear Energy Research (INER) of Atomic Energy Council of Taiwan since the Lab.'s establishment in 1995.

Mr. Ting Chow has been working on varieties of seismic safety related topics, such as: (1) Seismic Probabilistic Risk assessment at Kuosheng Nuclear Power Plant, (2) Study on Necessity of Installing Earthquake Auto-Scram System for NPPs in Taiwan, (3) Principal review of several NPP's seismic/structural related topics and issues including USI A-46 issue, Chapter 3 of Final Safety Analysis Report of Maanshan PWR NPP, design earthquake review of Lungmen ABWR NPP FSAR, (4) Seismic shake table system for seismic qualification of safety related component (5) INER's seismic shake table set up, (6) Commercial Grade Item's Seismic Dedication, and (7) Soil structure interaction analysis for the Independent Spent Fuel Storage Installation.

Mr. Chow had also been adjunct associate professor in Chun-Yuan Christian University, Taiwan from 1997–2003. Mr. Chow, born in 1955, joined INER right after his MS degree in Civil-Structural Engineering from National Taiwan University in 1979. He also holds MS degree from Massachusetts Institute of .Technology (1987) and a Ph.D candidate from Rensselaer Polytechnic Institute (1990).

CHUNG HOWARD H.

Dr. Howard H. Chung has over 35 years of diversified technical and managerial experience in the nuclear, pressure vessel, and aerospace industries in the areas of analytical and experimental structural mechanics, fluid transport phenomena, flow-induced vibrations, shock and vibration isolation, seismic engineering, hazardous radioactive wastes transportation technologies, pressure vessels and piping design, computer codes development, and nuclear facility configuration management. Currently, he is the President of Structural System Integrity in Naperville, Illinois. He previously worked as a Research Engineer and Project Manager on nuclear reactor and nuclear fuel processing facility R&D programs at Argonne National Laboratory, Argonne, Illinois for twenty-five years (1994–1999). Prior to joining Argonne, he was a technical research staff at MIT Lincoln Laboratory, Lexington, Massachusetts working on U.S. Air Force satellite R&D program for two years (1992–1994).

Howard Chung received BS in Naval Architecture and Marine Engineering from Seoul National University, Seoul, Korea in 1966, MS and Ph.D. in Mechanical Engineering from Tufts University, Medford, Massachusetts in 1971 and 1974, respectively. In addition, he received his MBA degree from University of Chicago in 1983. Dr. Chung has published over 60 technical papers and reports in his fields of expertise and served as an Associate Editor of the ASME Journal of Engineering for Gas Turbines. Dr. Chung served as a member of the ASME Board on Nuclear Codes and Standards (BNCS) for ten years (1987–1997) and as a member of the ASME B&PV Section III Subgroup on Containment Systems for Nuclear Spent Fuel and High Level Waste Transportation Packaging (SG-NUPACK) for more than ten years. In addition, he has been serving as a member of the Committee on Design and Fabrication of Nuclear Structures (N690) of the American Institute of Steel Construction (AISC) since 1991.

Dr. Chung has been also active on serving various professional organizations including the Chair (2002–2003) of the ASME Pressure Vessels and Piping, the Vice-Chair (1995–1996) of the ASME Nuclear Engineering Division and the Vice-Chair (2003–2005) of the Anti-Seismic Systems International Society (ASSISi). He also chaired many international conferences, including the 2002 ASME Pressure Vessels and Piping Conference in Vancouver, Canada. Dr. Chung is an ASME Fellow and received the ASME Dedicated Service Award in 1999.

CIPOLLA, RUSSELL C.

Mr. Russell Cipolla is Vice President, Nuclear Power Generation, and Principal Engineer for APTECH Engineering Services, Inc., Sunnyvale, California (USA). Mr. Cipolla received his B.S. degree in Mechanical Engineering from Northeastern University in 1970, and his M.S. in Mechanical Engineering from Massachusetts Institute of Technology Mechanics in 1972. He has been active in the Nuclear Power Industry since the early 1970s having worked at the nuclear divisions of Babcock & Wilcox and General Electric in the area of ASME Section III design associated with both naval and commercial power plants systems.

Russ has specialized in stress analysis and fatigue and fracture mechanics evaluations of power plant components in operating plants. He has applied his skills to many service problems to include stress corrosion cracking (SCC) of J-groove attachments welds in reactor vessel head penetrations and pressurizer heater sleeves, mechanical and thermal fatigue in piping, SCC in low pressure steam turbine rotors and blades, and fitness-for-service of components supports. Russ was also involved in resolving the NRC Generic Safety Issues A-11 and A-12 regarding fracture toughness and bolted joint integrity. He is well versed in the integrity of threaded fasteners for both structural joints and pressure boundary closures.

In recent years, Russ has been active in both deterministic and probabilistic methods and acceptance criteria for nuclear steam generators (SG) regarding pressure boundary integrity in compliance with NEI 97-06 requirements. In support of industry group efforts, he has made significant contributions to the industry guidelines for the assessment of tube integrity and leakage performance for various degradation mechanisms affecting Alloy 600 and 690 tubing materials. He has development methods for predicting tube burst and leak rates under various service conditions, which have become part of the industry standards.

Russ has been very active in ASME Section XI since joining the Working Group on Flaw Evaluation in 1975, for which he is currently Chairman. Russ is also a member of the Subgroup on Evaluation Standards and Subcommittee Section XI, and has participated in many ad hoc committees on such topics as environmental fatigue, SCC of austenitic materials, and fracture toughness reference curves for pressure vessels and piping, and SG tube examination. Russ has authored/coauthored over 80 technical papers on various subjects and assessments from his past work.

CONLISK, PETER J.

Dr. Conlisk's has a B.S. in Mechanical Engineering and M.S. in Engineering Science from the University of Notre Dame and Ph.D. in Engineering Mechanics from the University of Michigan. He has forty years experience applying engineering principles, com-puters, experimental techniques, and Codes and Standards to solving design of processing equipment and vessels in the chemical industry. From 1960 until 1968, he worked in the Aerospace industry and from 1968 until his early retirement in 1993, Dr. Conlisk worked for the Monsanto Corporation, the last 19 years in the Engineering Department. He was a key member in a team at Monsanto that developed acoustic emission examination for fiberglass and metal tanks and vessels. His services are now available through Conlisk Engineering Mechanics, Inc., a consulting firm he formed in 1994. He has concentrated on design of tanks and pressure vessel, especially fiberglass composite (FRP) vessels. Dr. Conlisk is a nationally recognized authority in FRP equipment design and analysis. He is a member of the ASME committee that developed the ASME/ANSI Standard: "Reinforced Thermosetting Plastic Corrosion Resistant Equipment, RTP-1."

Dr. Conlisk is past chairman and current vice-chairman of the ASME B&PV Code subcommittee, Section X, governing FRP pressure vessels. He is also a past member of the main committee of the ASME B&PV Code. Dr. Conlisk is a registered professional engineer in Missouri.

CUETO-FELGUEROSO, CARLOS

Carlos Cueto-Felgueroso obtained a Naval Engineer (M.Sc.) degree in the Escuela Técnica Superior de Ingenieros Navales of the Polytechnic University of Madrid in 1977. He worked for four years in the Structural Mechanics Section of AESA, the major shipbuilding company in Spain. Carlos moved to IBM Spain, and in 1985 joined the Integrity of Components Group (now Materials and Life Management Unit) of Tecnatom S.A. He specializes in the stress and fracture mechanics analyses of components and piping, in both nuclear and fossil plants. Main activities to date have been in the analytical evaluation of flaws of major components (reactor pressure vessels, turbines, etc.) and piping. He developed Flaw Evaluation Handbooks for streamlining the evaluation of NDE flaw indications. Carlos was involved in the development of Steam Generator tubes plugging criteria for several types of degradation (PWSCC, IGA/ODSCC, etc.). He has expertise in the development of acceptance criteria for PWSCC defects in the CRDM penetrations of vessel heads, and in the Bottom Mounted Instrumentation penetrations.

Carlos has experience in the evaluation of piping failure probabilities for the development of Risk-Informed ISI programs. He participated in the Working Group on Codes and Standards (WGCS) organized by the European Commission and he is member of the ENIQ Task Group on Risk (TGR) dedicated to RI-ISI activities in Europe.

In recent years Carlos participated in several European projects such as the benchmarking of Structural Reliability Models for RI-ISI applications (NURBIM project) and the comparison of structural evaluation methodologies for Thermal Fatigue in piping (THERFAT project) both in the 5th Framework Program of the European Commission. Carlos is a member of the ASME B&PV Section XI Working Groups on Implementation on Risk Based Examination and of the Working Group and on Inspection on Systems and Components.

DEARDORFF, ARTHUR F.

Arthur F. Deardorff has a Mechanical Engineering B.S, from Oregon State University (1964) and MS, University of Arizona (1966). He is a Registered Mechanical Engineer, State of California. He is a Vice President, Structural Integrity Associates, San Jose, California. His professional experience includes 1987 to present with Structural Integrity Associates, San Jose, CA, 1976–1987 with NUTECH, San Jose, CA, 1970–1976 with General Atomic Company, San Diego, CA and 1966–1970 with The Boeing Company, Seattle, WA. His professional associations include American Society of Mechanical Engineers and American Nuclear Society. He is a Past Member of the ASME Code Section XI Subgroup Water Cooled Systems, Working Group on Implementation of Risk-Based Inspection, Task Group on Erosion-Corrosion Acceptance Criteria, Task Group on Fatigue in Operating Plants, and Task Group on Operating Plant Fatigue Assessment, and the ASME Code Post Construction Committee, Subgroup on Crack-Like Flaws.

Mr. Deardorff has expertise in fracture mechanics, stress analysis and reactor systems evaluation, with a strong academic background in thermal-hydraulics and fluid system. His expertise includes PWR and BWR systems and fossil-fired power plants. Art is known internationally for providing ASME Code training in Section III design and analysis and Section XI flaw evaluation.

DEBOO, GUY H.

Guy DeBoo has a B.S., Mechanical Engineering from Northwestern University, 1976 and M.S. Mechanical Engineering from University of Illinois, 1986. His professional experience from 1995 to present is as Senior Staff Engineer, Commonwealth Edison, Senior Staff Engineer with Exelon Nuclear and with Sargent & Lundy Engineers 1976 to 1995. During his 24 years in nuclear power generation, DeBoo has worked on major nuclear design projects including design, inspection and testing phases.

Mr. DeBoo's recent experience includes fatigue, crack growth, flaw stability analyses and operability for power plant components. He supervised functionality and operability evaluations of systems and components to address unanticipated operating events or conditions, which do not meet inspection or test requirements. Mr. DeBoo provides engineering direction for design and operability evaluations of pressure vessels, piping and components and provides technical responses to NRC.

He is a Member ASME B&PV Code Section XI, Secretary, WG Flaw Evaluation, and WG Pipe Flaw Evaluations. Guy is a PE (Illinois), Member of National Society of Professional Engineers.

His industry participation includes PVRC Technical Committee on Piping Systems; publications include "Position Paper on Nuclear Plant Pipe Supports," WRC Bulletin 353, May 1990, and a Tutorial on "An Integrated Approach to Address Engineering of Operating Nuclear Power Plants Functionality and Operability Criteria," 1994, ASME PV&P Conference.

DIXIT, K. B.

K.B. Dixit graduated from the Indian Institute of Technology, Bombay (IITB) with a B. Tech. in Mechanical Engineering in 1972. After completion from the 16th Batch of Training School of Bhabha Atomic Research Centre in 1973, he joined Nuclear Power Corporation of India Limited (NPCIL), Mumbai. His initial field of work was in the area of Structural Analysis and Design of Reactor Components of Indian Pressurized Heavy Water Reactors, using Finite Element Method and ASME Boiler & Pressure Vessel Code, Section III, Nuclear Vessels.

Mr. Dixit was involved in design of nuclear components of Narora Atomic power station, which has India's first indigenous Pressurized Heavy Water Reactors. He has made significant contributions in evolving technology, for design and manufacture, of nuclear components. He was also involved in Design of coolant channel components of PHWRs and development of shutdown systems of Indian PHWRs. He has also made important contributions in resolution of problems associated with core components where remotely operated toolings had to be developed indigenously.

In addition to design and analysis, Mr. Dixit has also gained expertise in Ageing Management, Equipment Qualification and Seismic Revaluation. He has also participated in regulatory reviews for operating reactors as well as those under various stages of design/construction. Publications by him include papers presented at Conferences for Structural Mechanics in Reactor Technology, International conference on Pressure Vessels technology etc. Presently he is working as Executive Director for engineering group of NPCIL and is in charge of all design activities for all systems and components of Indian PHWR plants.

EUROPA MALCOLM

Malcolm Europa is a Senior Engineer with the South African National Nuclear Regulator where he carries responsibility for coordination of safety and technical assessment of nuclear facilities and nuclear equipment under South African nuclear regulations. He has participated in regulatory reviews of the Koeberg nuclear reactors as well as design reviews of pressurized components of the PBMR plant. His background includes nuclear power industry operations experience which includes planning, engineering design, safety reviews and mechanical integrity assessment of pressurised equipment for both nuclear and conventional service.

He currently represents the South African National Nuclear Regulator at the Multinational Design Evaluation Programme (MDEP) on both the Codes and Standards and Vendor Inspection Cooperation working groups.

Malcolm is a Mechanical Engineering graduate from the Peninsula Technikon (1986) and the University of Southern California (1993). He is a Registered Professional Engineer in the Republic of South Africa.

FELDSTEIN, JOEL G.

Joel Feldstein has a Metallurgical Engineering B.S. (1967) and M.S. (1969) from Brooklyn Polytechnic Institute. He has more than 30 years experience in the welding field, ranging from welding research for a filler metal manufacturer to welding engineering in the aerospace and power generation industries. He began his career in power generation with Babcock & Wilcox in 1972 at their R&D Division working on manufacturing-related projects and moved into plant manufacturing in 1984 as the Manager of Welding. There he became familiar with the construction of components for both nuclear and fossil applications. His first assignment on coming to Foster Wheeler in 1993 was in the Technical Center as Manager of Metallurgical Services later taking on the additional responsibility of the Welding Laboratory. In 1998 Joel Feldstein assumed the responsibility of Chief Welding Engineer.

Joel Feldstein, who is currently Chairman of the ASME B&PV Code Standards Committee and a member of the Board on Pressure Technology Codes & Standards began his ASME Code involvement with the Subcommittee on Welding (the responsible subcommittee for Section IX) in 1986. In 1992 he became Chairman of the Subcommittee on Welding and became a member of the B&PVC Standards Committee. He is a recipient of the J Hall Taylor Medal from ASME for the advancement of standards for welding in pressure vessel and piping construction. He has also been an active member of the Subcommittee on Boilers (Section I).

Joel Feldstein is also active in other professional societies including AWS and the Welding Research Council where he served as Chairman of the Stainless Steel Subcommittee, the High Alloys Committee and a member of their Board of Directors.

FLETCHER, JOHN

John Fletcher earned a masters degree in Plasma Physics and spent the early part of his career as a researcher on the South African Tokomak, Tokoloshe. He spent 13 years in a Research & Development environment and during this period authored and co authored five papers in international research journals and thirteen contributions to international conferences. He then spent 10 years as project manager in the South African power generation industry, managing projects on fossil plants and the Koeberg nuclear power plant.

John Fletcher is currently employed at PBMR, with the responsibility for the development and implementation of an ISI program for PBMR. John Fletcher serves as the Chair of the Special Working Group HTGRs established in February 2004.

The SWG has the charter to rewrite SC XI Division 2 Rules for Inspection and Testing of Gas-cooled plants. This SWG has produced a first complete draft of a rewritten SC XI Div 2 for Gas Reactors. The draft introduces the concept of Reliability and Integrity Management (RIM) program that provides the rules and requirements for the creation of the RIM Program for the Modular High Temperature Gas-Cooled Reactor (MHR) type.

GERAETS, LUC H.

Dr. Geraets has an M.S. Degree and a Ph. D. in Mechanical Engineering from the University of Louvain in Belgium. He also holds an MBA from the Mons Polytechnical Institute (Mons, Belgium). He is an expert in the design of mechanical equipment and piping, seismic engineering, and the ASME Section III pressure component code. His fields of expertise include thermal transient, fatigue, water hammer, vibration, pipe rupture, dynamic analysis, finite element stress analysis, inelastic analysis, code compliance, inelastic response of piping, and fitness-for-service criteria. Dr. Geraets' extensive background in engineering mechanics in the areas of analysis, design, criteria development, and management of projects rest on 35 years of engineering experience with the various metamorphisms (Tractionel, Tractebel, SUEZ) of GDF SUEZ, including 20 years in the analysis and design of power plant buildings, mechanical equipment, and supports. He has participated in all phases of power projects including conceptual studies, licensing, design, construction, as-built, modifications, and resolution of operating problems. Based in Brussels (Belgium), Dr. Geraets is now in charge of Strategy, Business Development and Research for the Nuclear Activities Division of GDF SUEZ.

Dr. Geraets joined the ASME Pressure Vessel and Piping Division in 1983. He has published several Conference papers. A founding member of the Seismic Engineering Technical Committee, of which he has occupied various Offices, including Chairmanship (1995–1999), he has been the first International Coordinator of PVP between 2001 and 2004. In 2004, Dr. Geraets became a member of the Executive Committee of the PVP Division, in charge of Honors and Awards from 2004 to 2008, and Vice Chair and Secretary for 2008–2009. He served as Technical Program Chair of the 2008 Pressure Vessels and Piping Conference in Chicago, and will be the Conference Chair for the 2009 PVP Conference in Prague, Czech Republic.

Dr. Geraets is a Fellow of ASME. He has been extensively involved with ASME Code activities, strongly promoting the development of Code knowledge in Belgium, through various means including participation to Section III Code Committees; he has been a member of both the Working Group on Piping and the Working Group on Components Supports between 1984 and 1994. Dr. Geraets has been awarded the Calvin W. Rice Lecture Award in 2008

GIMPLE, RICHARD E.

Richard Gimple has a BSME from Kansas State University (1974) and is a Registered Professional Engineer. Since 1982 he has been employed by the Wolf Creek Nuclear Operating Corporation. Previous employment was with Sauder Custom Fabrication (1979–1982) and Fluor Engineers and Constructors (1974–1979).

As a nuclear utility employee, he has primarily been involved in implementation of ASME's Boiler & Pressure Vessel Code Section III and Section XI during construction and operation activities. Previous non-nuclear experience involved Section VIII pressure vessel and heat exchanger design and construction. At present, as a Principal Engineer, Mr. Gimple provides company wide assistance in the use of ASME Codes, with emphasis on Section III and Section XI.

Mr. Gimple has been active in the Codes and Standards development process since 1984. Mr. Gimple was the 2005 recipient of the ASME Bernard F. Langer Nuclear Codes and Standards Award. He is currently a member of the B&PV Standards Committee (since 2000), the Subcommittee on Inservice Inspection of Nuclear Power Plant Components (since 1994, serving 5 years as Chairman of Subcommittee XI during 2000–2004), the Section XI Executive Committee (since 1992), and the Subgroup on Repair/Replacement Activities (since 1987, serving as Chairman for 7 of those years). Past Codes and Standards participation included 6 years on the Board on Nuclear Codes and Standards and memberships on the Subcommittee on Nuclear Accreditation, Subgroup on Design (in Section III), and three Section XI Working Groups.

GORMAN, JEFFREY A.

Jeff Gorman has been working on materials issues related to nuclear power since 1959, when he was assigned to Naval Reactors. He studied civil engineering at Cornell before working for Naval Reactors. After leaving the Navy, he did graduate work in engineering science, with emphasis on materials science, at CalTech. Since 1968 he has worked as a consulting engineer in the civilian nuclear power program, with most of his work involving materials, corrosion, stress analysis and fracture mechanics.

In 1980, Dr. Gorman was a co-founder of Dominion Engineering, Inc., and is still actively working for the company. A significant part of his consulting work has been for EPRI. His work for EPRI has included preparation of many workshop proceedings involving PWR steam generator technology, preparation of topical reports on materials and corrosion issues, and assisting in revision of water chemistry guidelines. He has also worked extensively for utilities and other industrial organizations on materials and corrosion issues, such as evaluation of the causes of failures of pressure vessels and piping, and developing predictions of the probable rate of failure of PWR steam generator tubes. Dr. Gorman is a registered professional engineer and is a member of AMS, NACE and ANS.

GOSSELIN, STEPHEN R. (STEVE)

Steve Gosselin is a Senior Principal Consultant at Scandpower Risk Management (SRM) with over 30 years nuclear power industry experience. Prior to joining the SRM team in 2008, Mr. Gosselin was Chief Engineer in the Pacific Northwest National Laboratory (PNNL) Materials and Engineering Mechanics Group (1998–2008) and a Project Manager at Electric Power Research Institute (EPRI) from 1993–1998. His work has focused primarily on fitness-for-service, structural integrity, safety, and reliability of pressure vessels and piping components. He has made significant contributions in the areas of fatigue analyses and flaw tolerance methodologies for nuclear pressure vessel and piping components, environmental fatigue computational methods, fatigue crack flaw detection probability, on-line fatigue monitoring, and the development of risk-informed inservice inspection and reliability integrity management programs for nuclear power plant vessel and piping components.

Mr. Gosselin's computational expertise is complemented by over 13 years experience in system engineering and mechanical design analyses at Westinghouse and Combustion Engineering PWR commercial nuclear power plants and 8 years operating experience on U.S. Navy SIC, S5W, and S3G submarine nuclear power plant designs.

Mr. Gosselin is an ASME Fellow (2009) and is a member of the ASME Section XI Working Group on Operating Plant Criteria, ASME Section XI Special Working Group on High Temperature Gas Cooled Reactors, and the ASME Committee on Nuclear Risk Management (CNRM) Subcommittee on Applications. His work has resulted in improved Code rules for operating nuclear power plant piping and vessel component fitness-for-service (ASME Section XI Non-mandatory Appendices E and L) and risk-informed inservice inspection (Code Case N-578).

Mr. Gosselin has a B.S. degree in Mechanical Engineering from the California State Polytechnic University (1980) and a M.S. degree in Mechanical Engineering from the University of North Carolina at Charlotte (1998). He is a registered professional engineer in California. Mr. Gosselin has published 45 papers, articles, and reports in the open literature and is a consulting expert to the International Atomic Energy Agency (IAEA) in the areas of plant life extension, design reconciliation and risk-informed inservice inspection.

GRABER, HAROLD C.

Harold Graber works as an Independent Consultant. Previously he was with the Babcock Wilcox Company in the Nuclear Equipment Division for 34 years. He was Manager of NDT Operations and Manager of Quality Assurance Engineering. Harold Graber is a Member of ASME for 15 years.

He is an active participant on the B&PV Code, Subcommittee V on Nondestructive Examination. He was Vice Chair Subcommittee V; Chair, Subgroup on Surface Examination. He was Member of Subcommittee V on Nondestructive Examination, Subgroup of

Volumetric Examination, Subgroup on Personnel Qualification and Inquiries.

Harold Graber is a Member, American Society for Testing Materials (ASTM) for 26 years. He was Chairman, Subcommittee E7.01 on Radiology. His Committee memberships include Committee E-7 on Nondestructive Examination, Subcommittee E7.02—Reference Radiological Images, Subcommittee E7.06–Ultrasonic Method. He is a Member, American Society for Nondestructive Testing (ASNT). He is a Past Chair, Cleveland, Ohio Section—1971.

Harold Graber is the recipient of ASTM Merit Fellow Award (1992); ASTM Committee E-7—C.W Briggs Award (1989); ASNT Fellow Award (1978). His Certifications include ASNT; Level III certificates in Radiography, Ultrasonic, Liquid Penetrant and Magnetic Particle Methods.

GRANDMANGE, JEAN-MARIE

After graduating in 1972 from the Applied Physics Department of the Institut National Des Science Appliquées in Lyon (France), Jean-Marie Grandemange was a research engineer at the Ecole des Mines research laboratory in Paris, where for three years he worked in the field of fractures mechanics.

He joined the Framatome Group in 1976, working initially in the Safety Dept. on the safety of mechanical components (design assumptions and criteria). He then moved on to the Primary Components Division, working in the Materials and Technology Dept., where he was in charge of the "Design" section from 1981 to 1997, and later Assistant to the head of department. He was named Senior Consultant in 1996.

Since 1978 he has led the Editorial Group in charge of writing the RCC-M design rules. He became a member of the RCC-M Sub-Committee in 1984 and was appointed Chairman of the committee in 1989. Since 1989 he has been responsible for Framatome's contribution to the preparatory work for the construction joint rules for use in the EPR project.

He led the Framatome Structural Analysis Group during the period 1989–1995, representing the company in the Cetim Boilerwork Commission, the RSE-M sub-committee responsible for in-service component inspection rules, and the Working Group on Codes and Standards (WGCS) organized by the European Commission.

Since 1986, he has been the manufacturer's designated expert both to the CCAP (French Central Commission for Pressure-Retaining Equipment) and its Permanent Nuclear Section (SPN) in charge of regulatory text discussion and application. He has lectured in several courses organized by Framatome, EDF, various French Institutes and run seminars in South Korea, Taiwan and China.

GRIESBACH, TIMOTHY J.

Timothy J. Griesbach earned a B.S. in 1972 and M.S. in 1974 in Metallurgy and Materials Science from Case Western Reserve University in Cleveland, Ohio. He is currently an Associate with Structural Integrity Associates (SI) in San Jose, California. Before joining Structural Integrity Associates, Mr. Griesbach was the Director of Technical Services for ATI Consulting. Mr. Griesbach was a Project Manager with the Electric Power Research Institute (EPRI) from 1982 to 1993 where he managed programs on reactor vessel integrity, research on neutron irradiation embrittlement, vessel material toughness properties, fracture mechanics methods, and management of reactor vessel integrity issues including pressurized thermal shock. From 1977 to 1982 he was a Principal Engineer at Combustion Engineering responsible for evaluating the response of nuclear systems and components to severe loading conditions using advanced finite element techniques. From 1974 to 1977 Mr. Griesbach was a Materials Engineer with Pratt & Whitney Aircraft where he was a member of a select research team developing a unique process to produce diffusion bonded jet turbine blades.

Mr. Griesbach is a member of ASME and the American Nuclear Society, and he has been a member of Section XI since 1989. He is chairman of the Working Group on Operating Criteria whose charter is to develop and maintain the Code criteria for operating pressure vs. temperature limits, operating plant fatigue assessment, and related operating plant issues. He is also a member of the Working Group on Flaw Evaluation and Subgroup on Evaluation Standards.

Mr. Griesbach specializes in evaluation of aging degradation mechanisms for nuclear components, including developing databases and modeling predictions on irradiated materials behavior. He has taught courses on managing the integrity of reactor pressure vessels for extended vessel life, and he has written numerous technical papers on these topics.

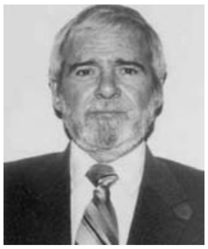
GRIFFIN, DONALD S.

Dr. Don Griffin has 30 years of experience in the structural design, development, and evaluation of nuclear reactor systems. At Westinghouse he developed computer-oriented methods of analysis, structural design criteria, and design procedures for naval, PWR, and fast breeder reactors. He has directed structural analysis of static and dynamic systems including effects of fatigue, fracture, thermal shock, seismic, fluid-solid interactions, and non linear and creep response of materials. He has personal expertise in buckling and instability, inelastic analysis, and elevated-temperature structural design. Current ASME Boiler and Pressure Vessel Code activities include development of design limits for high-temperature and creep buckling, and participation in the Subgroup for Elevated-Temperature Design. Responsible for presentation and resolution of elevated-temperature structural design issues raised during NRC licensing review of CRBRP.

Key relevant experience of Dr. Griffin includes Support of integrity evaluation of the CRBRP Containment Vessel during NRC licensing review; Responsibility for evaluation of Hanford N Reactor pressure tube integrity and pressure tube rupture propagation; Provided structures, seismic, and materials input to the Westinghouse Independent Safety Review of Savannah River Production Reactors; and In-depth review of the Loss of Coolant Accident Design Basis for the Savannah River Production Reactors – Leak Before Break Report.

Dr. Griffin is a Fellow of ASME, past Associate Editor of the Journal of Applied Mechanics, past Chairman of the ASME Pressure Vessel and Piping Division, a recipient of the ASME PVP Medal, and author of 26 publications in applied mechanics, computer methods, and elevated-temperature design. He has been an active contributor to the ASME Boiler and Pressure Vessel Code, Section III, Subsection NH for elevated-temperature nuclear applications. Griffin earned his BME, Cornell University (1952), MS (1953) and PhD (1959), at Stanford University.

HAFNER, RONALD S.



Ronald S. Hafner has more than 40 years experience in a variety of disciplines ranging from radar systems and nuclear instrumentation, to non-destructive testing using gamma-ray sources and electronic devices, to nuclear reactor- and tritium facility-operations, to regulatory issues associated with Department of Energy facilities and the transportation, storage, and disposal of radioactive materials. After a seven-and-a-half year enlistment in the U.S. Air Force, he went to work at Sandia National Laboratories, in Livermore, CA, in 1974, where he specialized in tritium operations and tritium health physics. While at Sandia, he went back to school and received his Bachelor of Science Degree from California State University, Hayward, CA, in 1983, with a major in Physical Sciences and a minor in Physics. In 1987, he moved to Lawrence Livermore National Laboratory where, for the first four years, he worked in tritium operations and tritium facility management. In 1991, he moved to the Mechanical Engineering Division, where he has been part of an engineering consulting organization.

His ASME activities started in 1993, with the Operations, Applications, and Components Committee of the PVP Division. Since that time, he has been involved with the development of more than 90 PVPD Conference sessions on the Transportation, Storage, and Disposal of radioactive materials. He is currently a member of the Executive Committee of ASME's Pressure Vessels & Piping Division.

HALLEY, GEOFFREY M.

Geoffrey M Halley, P.E. holds degrees in Electrical Engineering, Mechanical Engineering, and Engineering Administration (Masters). He is a Registered Professional Engineer in Illinois. From 1993 to the present he is the President of Sji Consultants, Inc., a technical consulting company, providing services to the boiler industry in the areas of product design, development, trouble shooting and forensic investigation/expert witness work. He has 40 years of boiler industry experience, ranging from research/product development, design and applications/installation., primarily in the institutional and industrial segments of the marketplace. He held various positions at Kewanee Boiler Corporation from 1968 to 1986, initially as Supervisor of Research and Development, and as Vice President – Technical Director from 1979 onwards. From 1986 through 1992 he was president of Halcam Associates a Mechanical Contracting Company specializing in commercial, institutional and industrial design/build/service and repair of boiler and HVAC systems. From 1959 through 1968 he was employed in the Aerospace and the Nuclear Engineering industries.

Geoffrey Halley was Chair of ABMA Joint Technical Committee (1981–1986), and has been a member of several boiler industry advisory groups to the USEPA and USDOE. He currently is ABMA Director of Technical Affairs, and was Editor of ABMA Packaged Boiler Engineering Manual. He has been an Instructor at boiler industry technician training schools offered by ABMA/NBBI, and boiler manufacturers. He has authored a number of papers on boiler related topics, published in The National Board Bulletin, Boiler Systems Engineering, and Maintenance Management.

Geoffrey Halley currently is a member of the ASME CSD-1 Committee, and the National Board Inspection Code Sub-committee on Installation.

HANMORE, PETER

Peter has worked within the engineering inspection industry since 1972. After joining Commercial Union Assurance Company as a metallurgist in the engineering laboratory he moved into the New Construction Department in 1984 and has been continuously associated with inspection during the manufacture of many types of work equipment. Although qualified as a metallurgist he has been involved in many related aspects of work equipment. He is an Authorized Inspector Supervisor for the provision of ASME Code services and maintains close links with that organization. Peter is currently a member at Large of the Board of Directors of Codes and Standards. His career within the inspection industry is extremely varied and includes experience such as; Health and Safety Manager, Quality Manager as well Project Manager for the obtaining notification and subsequent implementation of services associated with the Machinery, Lift and Pressure Equipment Directives.

Peter Hanmore has been associated with the Pressure Equipment Directive for many years even before its publication and represents the UK Inspection Bodies on many national and European Committees. Peter represents Bureau Veritas (Notified Body) at the European and UK Conformity Assessment Body Forums for both machinery and pressure equipment. He is a Past-Chairman of the European Conformity Assessment Body Forum (CABF), for pressure equipment and currently Chairman of the UK Notified Body Forum for machinery. During his period as Chairman of the CABF he represented the notified bodies at member states working group meetings; Working Group Pressure, Working Party Guidelines and Working Party Materials, and is still an active member of the latter.

Peter has provided numerous training courses on European Directives and related topics throughout the world for various organisations, e.g. ASME, IMechE, BSI, JSME, European Commission as well as many workshops tailored to individual manufacturers/users requirements.

HASEGAWA, KUNIO

Dr. Kunio Hasegawa graduated from Tohoku University with a Doctor of Engineering degree in 1973. He joined Hitachi Research Laboratory, Hitachi Ltd., over 30 years back. During his term at Hitachi, he was also visiting professors of Yokohama National University and Kanazawa University for several years. Since 2006, Dr. Hasegawa serves as a principal staff in Japan Nuclear Energy Safety Organization (JNES).

Dr. Kunio Hasegawa is a member of Japan Society of Mechanical Engineers (JSME), and is a past member of the JSME Fitness-for-Service Committee for nuclear facilities. He is also a member of ASME and is involved in ASME Boiler and Pressure Vessel Code Section XI Working Group, Subgroup and Subcommittee activities. He has been active for three years as a Technical Program Representative of Codes and Standards Technical Committee in ASME PVP Division.

He has been involved with structural integrity for nuclear power components, particularly, leak-before-break, fracture and fatigue strengths for pipes with cracks and wall thinning, and flaw characterizations for fitness-for-service procedures. Dr, Hasegawa has published for over 100 technical papers in journals and conference proceedings.

HECHMER, JOHN

Mr. John Hechmer has a degree in I Mechanical Engineering from the University I of Notre Dame (1957). He joined the I Babcock & Wilcox Co. (now owned by I McDermott, Inc.) for design and analysis I work for pressure vessels. His work was I primarily for the power generation and I defense industries. His experience included project and engineering management, technology development, and management. His Power Generation products were for both BWR and PWR nuclear electric plants. Defense Industries work addressed Class 1 pressure vessels for the nuclear navy program, primarily nuclear reactors and steam generators for aircraft carriers and submarines. Research products included Breeder Reactor Program, Sodium-steam Generator, Molten Salt Steam Generator. Technology Development was spent in developing tools and procedures for design-analysis interfacing with the Research Center and Engineering Fabrication of Babcock & Wilcox Co. This was enhanced by many years of participation in ASME B&PV Committees, PVRC, and PV&P Conferences. These engineering efforts occurred for 40 years.

Mr. John Hechmer has more than 25 publications, addressing primary and secondary stress evaluation, stress intensity factors, finite element methods and its applications, brittle fracture, welding capability for fatigue, and material's characteristic, examples of this are PVRC Bulletins #429 (3D Stress Criteria Guidelines For Application) and #432 (Fatigue-Strength-Reduction Factors for Welds Based on NDE).

HEDDEN, OWEN F.

Owen F. Hedden retired from ABB Combustion Engineering in 1994 after over 25 years of ASME B&PV Committee activities with company support. His responsibilities included reactor vessel specifications, safety codes and standards, and interpretation of the B&PV Code and other industry standards. He Continued working part-time for that organization into 2002. Subsequently, he has been a part-time consultant to the ITER project and several other organizations. Prior to joining ABB he was with Foster Wheeler Corporation (1956–1967), Naval Nuclear program. Since 1968 Mr. Hedden has been active in the Section XI Code Committee, Secretary (1976–1978), Chair (1991–2000). In addition to Section XI, Owen has been a member of the ASME C&S Board on Nuclear Codes and Standards, the Boiler and Pressure Vessel Committee, and B&PV Subcommittees on Power Boilers, Design, and Nondestructive Examination. He is active in ASME's PVP Division. Mr Hedden was the first Chair of the NDE Engineering Division 1982–1984. He has presented ASME Code short courses in the US and overseas. He was educated at Antioch College and Massachusetts Institute of Technology.

His publications are in the ASME Journal of Pressure Vessel Technology, WRC Bulletins and in the Proceedings of ASME PVP, ICONE, IIW, ASM, and SPIE. He is an ASME Fellow (1985), received the Dedicated Service Award (1991), and the ASME Bernard F. Langer Nuclear Codes and Standards Award in 1994.

HENRY, PHILIP A.

Mr. Henry, Principal Engineer for the Equity Engineering Group in Shaker Heights, Ohio, is a specialist in the design, installation, sizing and selection of pressure relief devices and relieving systems. He is currently chairman of the API Pressure Relieving System Subcommittee's Task Force on RP 520 related to the design and installation of pressure relieving systems. He conducts audits of pressure relieving systems to ensure compliance with OSHA PSM legislation and ASME, API and DIERs standards, codes and publications. He also teaches the official *API Pressure Relieving Systems* course.

Mr. Henry is actively involved in the development of technology for the API Risk-Based Inspection (RBI) methodology. He is co-author of the re-write of API 581, *Risked-Based Inspection Base Resource Document* and is responsible for the development and implementation of Risk-Based Inspection programs for pressure relief valves and heat exchanger bundles at refining and petrochemical plants. He also teaches the official *API 580/581 Risk-Based Inspection* course.

Mr. Henry provides technical support and engineering consulting to all levels of refinery capital projects. He has been responsible for the preparation of purchase specifications, bid tabulations, design reviews and the development and validation of approved vendors lists. He conducts project safety reviews for construction and pre-startup phases of major capital projects. His responsibilities include developing and maintaining engineering specifications in the pressure relief and heat transfer areas and providing overall coordination.

Mr. Henry is a registered Professional Engineer in the States of Ohio and Texas.

HILL III, RALPH S.

Ralph S. Hill III is a Consulting Engineer with Westinghouse Electric Company in Pittsburgh, PA. He has over 30 years of technical and management experience including more than eighteen years in planning, engineering design, construction, and modification for the nuclear power industry and fourteen years providing strategic planning, system engineering, risk management, process evaluation, and project management consulting services to the U.S. Department of Energy in spent nuclear fuel, radioactive waste management, and nuclear materials disposition-related projects.

Mr. Hill is a Member of the ASME Board on Nuclear Codes and Standards where he serves as Chairman of the Risk Management Task Group. Mr. Hill is actively involved in bringing risk-informed probabilistic design methods into the ASME Code and initiatives to support both advanced and next-generation nuclear reactors.

HSU, KAIHWA ROBERT

Kaihwa Robert Hsu earned a B.S. in Civil Engineering from Chung Yuan Christian College, and an M.S. from University of South Carolina. He has thirty years experience applying engineering principles, developing computer codes of corrosion erosion monitoring system, fatigue cycle monitoring system, fatigue crack growth, and fracture mechanics evaluation for nuclear industry. From 1981 until 2003, he worked in Westinghouse and has been involved in the areas of stress analysis, fatigue, fracture mechanics, leak before break, residual stress, primary water stress corrosion crack, and ASME Code related analyses pertaining to PWR. Mr. Hsu is currently a senior engineer with U.S. Nuclear Regulatory Commission (NRC). He is a key member in the development of the review and audit process improvement for aging management reviews. He is an audit team leader for license renewal application, responsible for metal fatigue time-limited aging analyses (TLAA) and aging management programs (AMPs) audit and review.

Publications of Kaihwa Robert Hsu are in the Proceedings of ASME Pressure Vessels and Piping Conference, the Proceedings of 10th Environmental Degradation Conference, and the Proceedings of 8th International Conference of Nuclear Engineering.

HOLLINGER, GREG L.

Greg L. Hollinger is a Senior Principal Engineer for BWX Technologies, Inc. in Barberton, Ohio. He has responsibility for Mechanical/Structural Technology Applications and Design Analysis of Navy Nuclear Pressure Vessel Components and use of the ASME Boiler & Pressure Vessel Code. He chairs the Engineering Department's Technical Support Team responsible for developing technology procedures. He is involved with both nuclear and non-nuclear ASME Certificates of Authorization for BWXT's Nuclear Equipment Division.

Greg is a Fellow Member of ASME, and was the 2004 recipient of the ASME Pressure Vessels and Piping Medal. He is the Chairman of the Subgroup on Design Analysis of the Subcommittee on Design of the ASME Boiler and Pressure Vessel Code. Greg is a member of the Pressure Vessel Research Council (PVRC) and the International Council on Pressure Vessel Technology (ICPVT). He has served on several Boards within the ASME Council on Codes and Standards, and he served as Chair of the ASME Pressure Vessels and Piping Division in 1995.

Greg is an Registered Professional Engineer (Ohio) with 30 years of engineering practice in power-related industries.

HUNT, STEPHEN

Since receiving his BSME from Purdue University in 1995, Steve Hunt has been involved in equipment design, stress analysis and root cause failure analysis of mechanical equipment primarily for the commercial nuclear power and offshore oil industries. This work has included commercial and research nuclear power plants, fossil power plants, floating and fixed offshore oil/gas production facilities, deep diving submersibles, large optical telescopes, tower cranes, paper mills, and chemical plants.

In 1980, Steve Hunt was a co-founder of Dominion Engineering, Inc., and he is currently a Principal Officer. A significant part of Mr. Hunt's recent consulting work has been for the Electric Power Research Institute (EPRI). Major areas of effort have included primary water stress corrosion cracking (PWSCC) of Alloy 600 material, boric acid corrosion, leakage reduction technology, and life cycle management. Publications for EPRI have included many documents related to Alloy 600 PWSCC, the *Boric Acid Corrosion Guidebook*, and most of the Sealing Technology and Plant Leakage Reduction Series reports. Mr. Hunt also works extensively for electric utilities in the areas of Alloy 600 PWSCC failure analysis and strategic planning, life cycle management, and root cause failure analysis. Mr. Hunt also provides technical consulting in the areas of offshore oil production facilities, high pressure sealing technology, large diameter bearings, and pressure vessel failure analysis. Mr. Hunt has authored several hundred reports for a wide range of clients and holds several patents. Mr. Hunt is a registered professional engineer and is a member of ASME and IEEE.

ISOMURA, TOSHIO

Toshio Isomura is a mechanical engineer with over 30 years of experience in all of the aspects of pressure vessels for petro and petrochemical plants. After graduating from Mechanical Engineering of Osaka University in 1972, he joined Chiyoda Chemical Engineering and Construction Co. Ltd, and was engaged in the design and development works of pressure vessels.

He started his new career in the High Pressure Gas Safety Institute of Japan at their Inspection and Certification Department in 2000, and is a manager of technical assessment division from 2006 succeeding Mr. Kajimura. At present, his main work is technical assessments of the technologies of non-standard pressure vessels for the special appraisal under the High Pressure Gas Safety Laws and the standardization tasks for technologies of pressure vessel designs, including Fitness-for-Service code.

He is also a member of the JIS (Japanese Industrial Standards) Pressure Vessels Technical Committee and contributes to the maintenance of existing JIS codes and development of new JIS codes.

He has also been involved with ISO/TC11 activities and Japanese committees and is the secretary of ISO/TC11/WG10 since 2006

JETTER, ROBERT I.

Mr. Jetter has over 40 years experience in the design and structural evaluation of nuclear components and systems for elevated temperature service where the effects of creep are significant. He was a contributor to the original ASME Code Cases eventually leading to Subsection NH. For over 20 years he was Chair of the Subgroup on Elevated Temperature Design responsible for the design criteria for elevated temperature nuclear components. He was Chair of the Subgroup on Elevated Temperature Construction, Vice Chairman of the Subcommittee on Design and a member of the Subcommittee on Nuclear Power. He currently again chairs the SG-ETD. Mr. Jetter has participated in domestic and international symposia on the elevated temperature design criteria. He was a member of a Department of Energy (DOE) steering committee responsible for the design criteria, and was a consultant and reviewer on various DOE projects. As a long time employee of Rockwell International/Atomics International, he was associated from the early sodium cooled reactors and space power plants through all the US LMFBR programs.

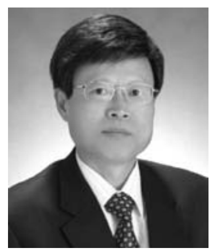
Recently he was an International Fellow for the Power Reactor and Nuclear Fuel Development Corporation at the Monju Fast Breeder Reactor site in Japan. He is a graduate in Mechanical Engineering from Cal Tech (BS) and Stanford (MS) and has a degree from UCLA in Executive Management. He is a fellow of the ASME.

JONES, DAVID P.

Dr. Jones has 40 years experience in structural design analysis and is lead consultant and developer on structural design procedures for SDB-63 (Structural Design Basis, Bureau of Ships, Navy Dept., Washington, D.C.). Dr. Jones is an expert on brittle fracture, fatigue crack growth, fatigue crack initiation, elastic and elastic-plastic finite element methods, elastic and elastic-plastic perforated plate methods, limit load technology, linear and non-linear computational methods and computer applications for structural mechanics. Dr. Jones's key contributions have been developing computer programs that allow use of complex three-dimensional finite element stress and strain results for the evaluation of ASME structural design stress limits. He introduced numerical methods to compute fatigue usage factors, fatigue crack growth, brittle fracture design margins and the like that have now become standards for use in naval nuclear design. He is currently working on using finite element elastic-plastic analysis tools for evaluation of limit load, fatigue, shakedown, and ratchet failure modes.

Dr. Jones has been an active contributor to the ASME Boiler and Pressure Vessel Code Committees; secretary and member of Subgroup on Fatigue strength, Member and chairman of the Subgroup on Design Analysis, Chairman of the Subcommittee on Design, and Chairman of the Task Force on Elastic-Plastic FEA. Dr. Jones was Chairman of Metal Properties Council Task Force on Fatigue Crack Growth Technology. He has also served as Associate Editor of the ASME *Journal of Pressure Vessels and Piping*. He has published over thirty papers on the topics of fatigue, fatigue crack growth, fracture mechanics, perforated plate technology, computational structural mechanics methods, non-linear structural analysis methods, finite element code development for fracture mechanics applications, finite element applications for perforated plate analysis (elastic and elastic-plastic), post-processing finite element results for ASME Boiler and Pressure Vessel Code Section III assessment, limit load technology, and elastic-plastic fracture mechanics. He has been awarded ASME PVP Literature Award – Outstanding Survey Paper of 1992 in ASME *Journal of Pressure Vessels and Piping* and ASME PVPD Conference Award – Outstanding Technical Paper form Codes & Standards – July 26, 2000. Dr. Jones received his BS and MS degrees from the University of Toledo in 1967 and 1968 and his PhD from Carnegie Mellon University in 1972. Dr. Jones is a member of ASME and has worked at the Bettis Atomic Power Laboratory in West Mifflin, Pennsylvania since 1968 where he currently holds the position of Consultant Engineer.

JO, JONG CHULL



Dr. Jong Chull Jo is a mechanical engineer who graduated from Hanyang University, Seoul, Korea in 1979, and obtained his M.S. and Ph. D. degrees from the same university in 1981 and 1985, respectively. Currently, he is a technical consultant of the Organisation for Economic Cooperation and Development (OECD), Nuclear Energy Agency in the area of Nuclear Safety and Regulation and concurrently is affiliated as a principal researcher with the Korea Institute of Nuclear Safety (KINS) Daejon, Korea for which he has been working since 1986. Before that, he worked as a full-time lecturer and subsequently an Assistant Professor of Mechanical Engineering Department at Induk College, Seoul for 5 years.

Dr. Jo's job for over the past two decades relates to the safety regulation of nuclear reactors including safety inspection and licensing review, preparing regulatory requirements and guides, and developing nuclear regulatory technology.

Dr. Jo was Head of Safety Issue Research Department at KINS and concurrently Project Manager of the Regulatory Framework Development for an Integral–Type Pressurized Water Reactor Licensing. He served as a member of the Korean National Technology Road Map Committee and a member of the National R&D Projects Evaluation Committee. He lectured extensively on the technologies for evaluation and resolution of nuclear reactor safety issues at National Research Institutions, Academic Conferences, Engineering Companies and Universities in Korea, and served as a lecturer from 2003–2005 at the Graduate School of Jeonju University, Korea.

Dr. Jo has been a member of the ASME Pressure Vessels and Piping Division (PVPD) since 1999 and has been serving as Chair of the PVPD Fluid-Structure Interaction Technical Committee since July of 2008. He has also been serving as Chair of the Fluid-Structure Interaction Division of the Korean Society of Pressure Vessels and Piping since 2004. He has been a member of the Korean Society of Mechanical Engineers since 1981, a member of the Korean Nuclear Society since 1986, and a member of the Korea Foundation of Science and Technology since 2003. He has published about 50 technical journal papers and over 100 conference proceeding papers. He has also co-authored or co-edited many monographs and books. He has been invited as a peer reviewer of contributing papers for several archival journals such as ASME Journal of Pressure Vessel Technology, Journal of Numerical Heat Transfer, Journal of Numerical Heat and Mass Transfer, Journal of Nuclear Engineering and Design.

Dr. Jo received 'Korean Prime Ministerial Citation' for recognizing contribution to the promotion of science and technology in 1994 and 'Korean Presidential Citation' for contribution to development of science and technology in 2004.

KAJIMURA, YOSHINORI

Yoshinori Kajimura has a Mechanical Engineering B.S. (1966) from Hiroshima University. He has more than 30 years of experience in the design of all kinds of pressure vessels including multi-layered pressure vessels for oil refinery, petrochemical industries and so on for Mitsubishi Heavy Industries, Hiroshima Works as a manager of design.

He began his career in The High Pressure Gas Safety Institutes of Japan (called KHK) at their Inspection and Certification Department in 1995. He responsible as the manager of technical assessment and special appraisal for the pressure vessels under the High Pressure Gas Safety Law and their regulations.

He also has been an active member of the committee of JIS (Japanese Industrials Standards) for pressure vessels since 1990 and also contributes to the development and restructuring of the standards for pressure vessels in Japan.

He has also been involved in ISO/TC11 activities and Japanese committee since 1997 at the restart of ISO/TC11 activities and he began to be the secretariat of ISO/TCI11/WG10 since 2003.

KANEDA, MASAHIKO

Masahiko Kaneda is Senior Vice President of Mitsubishi Nuclear Energy Systems, Inc. He has more than thirty years of experience in development and management of nuclear power plant design in Japan. Mr. Kaneda received a B.S. in Mechanical Engineering from Seikei Univ. in Tokyo, Japan in 1978. From September 2006 to March 2008, he was employed by MHI in Tokyo, Japan as the General Manager of Advanced Pressurized Water Reactor Promoting Department, Nuclear Energy Systems Headquarters. He directed all aspects of the APWR Promoting Department's operations to control activities such as Design Certification of US-APWR, Luminant Project and Potential Customer Engineering. From October 2005 to August 2006, he was employed by MHI in Hyogo, Japan as the General Manager of the Water Reactor Engineering Department, Nuclear Energy Systems Engineering Center, Nuclear Energy Systems Headquarters.

Under the direction of Mr. Kaneda, The Water Reactor Engineering Department got involved in the conceptual and basic design of the entire nuclear power plant facility, and consisted of various sections specialized in the system design, layout design, structural and seismic design, electrical design, instrumentation and control design, turbine system design, and water reactor engineering. In this position, Mr. Kaneda directed the entire operations of the Water Reactor Engineering Department and he established the department's annual operational goals, and planned the budget and resources needed for the achievement of annual operation goals.

Thus, Mr. Masahiko Kaneda has nearly three decades of experience related to plant design for nuclear power plants such as Japanese prototype FBR, Monju and many commercial PWRs in Japan. He is not only a specialist for plant layout design but seismic design as well. Currently, as the General Manager of APWR Promoting Department, Nuclear Energy Systems Headquarters in Mitsubishi Heavy Industries, LTD Japan, he is responsible for the promotion of US-APWR.

KARASAWA, TOSHIKI

The late Toshiki Karasawa graduated with honors form Yamanashi University with a B.E in Mechanical Engineering in 1973. Since graduating, his career had focused on nuclear power technology in Tokyo Electric Power Company (TEPCO). He had more than 30 years of broad experience in Design, Manufacturing, Inspection, Quality Assurance (QA) and Nuclear Fuel. He was the general manager of QA of Nuclear Division at the time of his passing away in March 2008.

During 1980's, Mr. Karasawa demonstrated excellent leadership to introduce ASME Boiler and Pressure Code Sec. III to METI (Ministry on Economy, trade and Industry) Notification No.501, which regulated the detail design and manufacturing of equipment for nuclear power plant in Japan. Following that, he had been contributing to develop Codes and Standards (C&S) in Japan and ASME.

Mr. Karasawa was a member of ASME, a Member-at-large of Board on International Standards (BIS) of CCS, since 1997. His report entitled "ASME Success Story in Japan" was favorably received at BIS meeting in June 2003.

Mr. Karasawa was a member of Japan Society of Mechanical Engineers (JSME) and serves as a Secretary of C&S Main Committee since 2001. He was a member of Nuclear C&S Main Committee of Japan Electric Association (JEA) and he served as Vice Chair of QA committee and Chair QA Sub-committee. He was a member of Structural Design Sub-committee in Thermal and Nuclear Power Engineering Society (TENPES). He was a member of The Japan Welding Engineering Society (JWES) and served as Secretary of Codes and Accreditation Committee since 2001. He was an Executive member of Atomic Energy Society of Japan (AESJ) since 1999 up untill his passing away.

Mr. Karasawa resided in Yokohama City Kanagawa Prefecture, with his wife, Chiharu, and two daughters, Mayumi and Hanae.

KARCHER, GUIDO G.

Guido G. Karcher, P.E. is a consulting engineer with over 48 years of experience in the mechanical engineering aspects of pressure containing equipment. He retired from the Exxon Research and Engineering Co. after serving 30 years as an internationally recognized engineering advisor on pressure vessel, heat exchangers, piping and tankage design, construction and maintenance. On retire from Exxon Research & Engineering Co. in 1994; he became a Consulting Engineer on fixed equipment for the petrochemical industry and related industry codes and standards. Guido has also functioned as the Technical Director of the Pressure Vessel Manufactures Association, for 15 years, in the areas of mass produced pressure vessel construction and inspection requirements.

Guido's code activities include over 35 years of participation in ASME, PVRC and API Codes and Standards activities serving on numerous committees and technical development task groups. He was elected to the position of Chairman of the ASME Boiler & Pressure Vessel Standards Committee for two terms of office (2001–2007) and was elected to the office of Vice President Pressure Technology Codes and Standards (2005–2008). Guido also served as Chairman of the Pressure Vessel Research Council and the American Petroleum Institute Subcommittee on Pressure Vessels and Tanks. He has written numerous technical papers on subjects related to pressure containing equipment.

Guido is an ASME Life Fellow and a recipient of the J. Hall Taylor Medal for outstanding contributions in the development of ASME Pressure Technology Codes and Standards. Guido was also recently awarded the 2007 Melvin R. Green Codes and Standards Medal for outstanding contributions to the development and promulgation of ASME Codes and Standards within the USA and Internationally. Other awards include the API Resolution of Appreciation and Honorary Emeritus Membership of Pressure Vessel Research Council. He earned a B.S.M.E. from Pratt Institute and M.S.M.E. from Rensselaer Polytechnic Institute and is a registered Professional Engineer in the States of New York and New Jersey.

KOSTAREV, VICTOR V.

Victor V. Kostarev is a Mechanical Engineering (Gas and Steam Turbines) graduate of Saint-Petersburg Polytechnic University, Russia. He earned his Ph.D. degree in 1979 for investigation of self-excited vibration of high speed rotors of supercritical steam turbines.

His professional career includes over 35 years experience in analysis and qualification of structures, systems, piping and components for vibration, operational, seismic loads and design basis accidental loads of nuclear power plants and other facilities in different industries. Dr. Kostarev is a founder of the State Laboratory for seismic and external events protection of SSC in CKTI Institute (1977). Then he established in 1992 a private Structural – Mechanical Engineering Company located in Saint-Petersburg, Russia (www.cvs.spb.su) where he is the president up to date. He is an author of more than 50 papers and 10 inventions.

V.Kostarev is a consultant for International Atomic Energy Agency. He is a member of ASME BPVC Nuclear Section III Subcommittee on Nuclear Power and Section III ASME BPVC Working Group on Piping. Victor Kostarev has been the volunteer ASME corresponding author/representative in Russia.

KOVES, WILLIAM J.

William Koves, Ph.D., P.E., ASME Fellow, is a Senior Engineering Fellow at UOP, a high technology company that develops and licenses process and related equipment technology in the petrochemical, process and related industries.

Dr. Koves has 40 years of experience in the design, analysis and troubleshooting of equipment and structures including aircraft, nuclear reactors, and petrochemical equipment. His specialties include stress analysis, fracture, elevated temperature design, heat transfer, stability, vibration, fatigue, fluid mechanics, and mechanics of granular solids.

Dr. Koves is author of numerous publications in the field and holder of 24 US and 3 European patents. He has been very involved with numerous ASME and PVRC committees including, Past Chair of ASME B31.3 Process Piping Committee, Chair of ASME B31 Mechanical Design Committee, Member of the B31 Standards Committee, Member of ASME Boiler and Pressure Vessel Subcommittee on Design Analysis, Elevated Temperature Design, Special Working Group on Design of Bolted Flange Joints and member of the Post Construction Standards Committee and Subcommittee on Repair.

Dr. Koves was Vice-Chair of the Pressure Vessel Research Council (PVRC), Member of PVRC Committee on Piping and Nozzles, Chair of PVRC Committee on Elevated Temperature Design, Chair of PVRC Subcommittee on Shell Intersections, and Past the Chair of the Post Construction Flaw Evaluation Committee and Member of the Main and Executive Committees.

KRECKEL, DIETER

Dieter Kreckel graduated in 1968 from the FH Bingen, Germany with a Dipl.-Ing. (FH) specializing in mechanical/chemical engineering. Dieter Kreckel started his active profession in 1968 within the Department of Engineering and Lay out, piping and components of the UHDE GmbH Company. The nuclear activities that he started in 1971 continue to this date. He is with AREVA NP GmbH (ex. Framatome ANP GmbH, ex. Siemens NP, and ex. KWU).

Dieter Kreckel's work experience includes Engineering components of BWR and PWR, Co-ordination of equipment specifications, QM- Engineering (ENACE Argentina 1981 to 1985), QM-Engineering in various Projects, International co-operation on Design Codes e.g. for EPR Development (GERMAN/ FRENCH), EU DG TREN, WGCS (Working Group "Codes and Standards").

Dieter Kreckel has immense experience in the field of European Nuclear Code activities that include Collaboration within the French REP 2000 Programme, Comparison of German and French Nuclear Codes and their application as a basis for the joint proposals to the **E**uropean Pressure Water Reactor **T**echnical **C**ode for **M**echanical Equipment (**ETC-M**), Elaboration of ETC-M Class 1 to 3 together with partner Framatome ANP SAS, Review of ETC-M Class 1 to 3 proposal together with German

Utilities and EDF, NPP upgrading of RUSSIAN NPP and compliance of the Russian Code analysis for applying to European Codes and Standards.

Dieter Kreckel organized various Seminars for the implementation of the PED and Harmonized EN Standards to Nuclear Codes in Europe. Since 2003 Dieter Kreckel is assigned and acts as Quality Manager for the Olkiluto 3 Project in Finland.

KUO, PAO-TSIN

Dr. Kuo earned an engineering diploma from Taipei Institute of Technology, a MS from North Dakota State University and a PhD from Rice University. He is a Registered Professional Engineer in the State of Maryland. He has been employed by the U.S. Nuclear Regulatory commission (NRC) since 1975. He held various positions in the NRC during this period.

He is currently the Program Director for the License Renewal and Environmental Impacts Program in the Office of Nuclear Reactor Regulation, responsible for guidance development and licensing activities of the license renewal programme as well as environmental reviews of application for license renewal, licensing amendments and early site permits.

Dr. Kuo is a member of the ASME Section XI Special Working Group on Plant Life Extension and former member of the ASME Section III Working Group of Piping Design as well as Task Group on Dynamic Stress Limits. Currently, he is the chairman of Working Group I, General Long Term Operation Framework, IAEA Extra Budgetary Program on Safety Aspects of Long Term Operation of Water Moderated Reactors.

KUSHWAHA H. S.

Mr. H.S. Kushwaha, M.Tech. (Mechanical Engineering), is Director, Health Safety and Environment Group at Bhabha Atomic Research Centre, Mumbai, India. He joined Reactor Engineering Division, Bhabha Atomic Research Centre (BARC) in 1971. Since then, he has been engaged in R&D activities for Structural Design and Safety Analysis of Indian Heavy Water Reactor program. He contributed significantly in the area of computational Mechanics, Pressure Vessel and Piping Design and Analysis and Leak-Before-Break (LBB) study of high energy piping system. Mr. Kushwaha has been associated with Seismic Design, Analysis and Testing of major components of 540 MW(e) Pressurized Heavy Water Reactor built at Tarapur, Maharashtra.

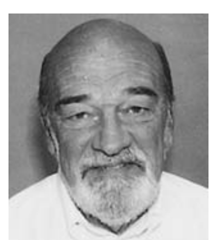
He is currently steering the research activities in the field of structural reliability, radiological risk assessment and uncertainty analysis. Mr. Kushwaha is Chairman of BARC Safety Council and President of Indian Association for Radiation Protection. Mr. Kushwaha is a member of Safety Review Committee for Operating Plants (SARCOP) of Atomic Energy Regulatory Board (AERB). He has published more than 600 technical papers. He is recipient of prestigious Indian Nuclear Society award. He is a Fellow of the National Academy of Engineering.

LAND, JOHN T.

John T. Land, P.E., has been involved in the design, analyses and manufacturing of Westinghouse's PWR nuclear primary equip-ment products for almost thirty years. His product design experience includes reactor internals, steam generators, presurizers, valves, and heat exchangers. Mr. Land also contributed to the design and development of the AP600 and AP1000 MWe Advanced Power Plants, the Westinghouse/Mitsubishi APWR 4500 MWt Reactor Internals, and many of the currently operating Westinghouse PWR domestic and international reactor internals components. In addition, he has directed and reviewed the design and analysis efforts of engineers from Italy (FIAT and ANSALDO), Spain (ENSA), Czech Republic, and Japan (MHI) on several collaborative Westinghouse international efforts. His experience included five years with Westinghouse as a stress analyst on nuclear valves in support of the Navy's Nuclear Reactor Program. Prior to working for Westinghouse, Mr. Land spent eleven years with the General Electric Company on the design and development of Cruise Fan and XV-5A Vertical Take-Off and Landing aircraft propulsion systems. He also holds eleven patents from General Electric, and Westinghouse. Mr. Land received his BS in Mechanical Engineering from Drexel University and his MS in Applied Mechanics from the University of Cincinnati.

Over the past thirty years, John has been active in ASME B&PV Code work. Mr. Land is currently member of the Working Group Core Support Structures and participates in the rule making and maintenance of Sub-Section NG. John is also a member of Sub-Group Design that oversees Section III and Section VIII Design Rules.

LANDERS, DONALD F.



Donald F. Landers, P.E., is currently Chief Engineer of Landers and Associates. He was General Manager and President of Teledyne Engineering Services where he was employed from 1961 to 1999. Mr. Landers, an ASME Fellow, has been involved in ASME Code activities since 1965 serving as a Member of B31.7 and Chairman of their Task Group on Design, Section III Working Group on Piping Design and Subgroup on Design. He continues as a member of these Section III groups as well as Subcommittee III and also served as a member of section XI and the BPVC Standards Committee.

Mr. Landers also served as a member of the Board on Nuclear Codes and Standards and as Vice Chairman. He has served on PVRC committees and was heavily involved in the PVRC research that led to the new seismic design rules in Section III.

He is an internationally recognized expert in piping design and analysis and application of ASME Code and regulatory requirements. Mr. Landers has authored over 20 technical papers related to design and analysis of pressure components.

He is currently involved in providing consulting services to the utility industry in the areas of Life Extension, Code compliance, and Operability issues. Don continues to provide training and seminars on Code Criteria and application internationally. He is recipient of the Bernard F. Langer Award, J. Hall Taylor Award, and ASME Dedicated Service Award.

LEWIS, D. WAYNE

Mr. Donald Wayne Lewis is a Project Engineer for Shaw Stone & Webster Nuclear with over 27 years of experience in commercial nuclear power and Department of Energy (DOE) nuclear related projects. He has worked on a variety of Mechanical/Structural engineering applications including nuclear power system design and construction, MOX fuel assembly design, spent fuel management and related NRC licensing. He has spent 17 years in his primary area of expertise which is related to dry spent nuclear fuel storage and is currently Project Engineer for several Independent Spent Fuel Storage Installation (ISFSI) projects. He has also served as a design reviewer for the DOE Yucca Mountain Project concerning spent fuel processing and disposal.

Mr. Lewis is a Member of the ASME Subgroup on Containment Systems for Spent Fuel and High-Level Waste Transport Packagings. He is the author of two publications related to spent fuel storage which are in the 2003 and 2005 proceedings of the International Conference on Environmental Remediation and Radioactive Waste Management (ICEM) sponsored by ASME.

Mr. Lewis received a B.S. in Civil Engineering from Montana State University in 1980. He is a Registered Professional Engineer in New York, Maine, Iowa, Utah and Colorado.

MAC KAY, JOHN R.

Mr. John Mackay has over 50 years experience as a mechanical engineering specialist in boilers, pressure vessels, steam accumulators, ASME Code construction, Nondestructive examination, heat transfer systems, combustion and municipal incinerator design and construction. John has a Bachelor of Engineering (Mech.), 1951 from McGill University, Montreal and followed it by numerous courses over the years in Management, Management Techniques, and Post-graduate engineering and management courses at Concordia University.

Mr. John Mackay was an employee of Dominion Bridge Company Limited in Montreal from 1951 to 1984 and has since continued to work as a private consultant in his field. His major accomplishments of the hundreds of projects he has been involved include the Primary System Feeder Pipes for the CANDU nuclear reactors, boilers for waste/refuse mass burn disposal systems and design and maintenance of API Storage Tanks. John has extensive experience in the design and construction of heat recovery boilers for the metallurgical industry. John is recognized as one of the leading practitioners of his specialties in Canada.

Mr. John Mackay has been a member of ASME for over 40 years, during which he has served on a variety of committees engaged in updating existing Codes, introduction of new Codes, and the investigation and resolution of questions referred to these committees. He has been a member of Section I Power Boiler Subcommittee since 1968 to present time, Chaired it 1989–2004; Member Standards Committee, 1971–present; Subgroup Electric Boilers (SCI) and chaired it in 1978–84; Member & Chairman Adhoc Task group on Acceptance Criteria. John was a Member and Chair of the Section V Subcommittee on Nondestructive Examination; Joint Task group B31.1/SCI. John is a member of Subgroup on General requirements & Surface Examination (SCV); and is a member of Subgroup on Materials (SCI). John was a member of Honors & Awards Committee (B&PV) from 1989–2006, and chaired in 1995–2006. He was a Member Executive Committee (B&PV Main Committee) from 1992–2004. In addition to ASME John is affiliated with several professional organizations including Engineering Institute of Canada and Quebec Order of Engineers.

John Mackay has several publications and has given lectures on engineering topics both in Canada and USA. John was a participant of several PVP conferences and ASHRAE. He has several hobbies that include Contract Bridge and John is happily married with adult children.

MALEK, M. A.

M. A. Malek is a Professional Engineer (P.E.) registered in the state of Maine, P.Eng. Canada registered in the Province of I New Brunswick and Prince Edward Island. Mohammad is a Certified Plant Engineer, CPE, U.S.A., and has more than 27 years experience in boiler and pressure vessel technology. Presently he is the Chief Boiler Inspector for the state of Florida. Prior to his present position, he was Chief Boiler, Elevator and Tramway Inspector for the state of Maine, Deputy Chief Inspector of state of Louisiana and Chief Boiler Inspector, Bangladesh.

Mr. Malek has demonstrated leadership in B&PV boiler and pressure vessel industry. His achievements include developing and designing a special husk-fired, fire-tube boiler of capacity 500 lbs/hr at 50 psi for developing countries. He has vast knowledge and experience in writing, and enforcing boiler and pressure vessel laws, rules, and regulations. He has written numerous articles and published in several technical journals. Malek obtained his BSME degree from Bangladesh Engineering and Technology, Dhaka (1972) and MBA from Institute of Business Administration, University of Dhaka (1979).

Malek has been a member of ASME since 1980 and Fellow of Institution of Engineers, Bangladesh. He is an instructor of ASME Professional Development courses, and serves on three ASME Committees including CSD-1 Committee, QFO-1 Committee, and Conference Committee of the ASME B&PV Committee. Malek has been a member of the National Board of Boiler and Pressure Vessel Inspectors since 1997.

MASTERSON, ROBERT J.

Masterson has a BSME from University of Rhode Island (1969) and course work for MSME, University of Rhode Island (1973). He is a Registered Professional Engineer in states of RI, MA, IL, NE, MI and AK, and is currently self-employed at RJM Associates in Fall River, MA. Masterson is a retired Captain, U.S. Army Corp of Engineers (1986). His professional experience included New England Electric System (1969–1970), ITT Grinnell Corporation, Pipe Hanger Division, Providence, RI (1972–1979). With ITT Grinnell he was a Manager of Piping and Structural Analysis for the Pipe Hanger Division (1974) and developed stress analysis, and testing for ASME Section III Subsection NF and provided training in Subsection NF for ITT Grinnell, several Utilities, AEs and support for NRC Audit. In 1978 he became Manager Research, Development and Engineering. He was Manager of Engineering (1979) at Engineering Analysis Services, Inc. East Greenwich RI later in 1990 called EAS Energy Services. He was Vice President of Operations (1984) and tasks included NRC audit support, turnkey projects and valve qualification.

Masterson was an alternate member, Working Group on Component Supports (Subsection NF), 1973–1979; Member Subsection NF 1979 to the present. Chaired Task Groups for Subsection NF jurisdiction; Chair of Working Group on Supports (SG-D) (SC III) since May, 2000 and Member of Committee for the First Symposium on Inservice Testing of Pumps and Valves, 1989, Washington, DC, NUREG/CP-0111.

MEHTA, HARDAYAL S.

Dr. Mehta received his B.S. in Mechanical Engineering from Jodhpur University (India), M.S. and Ph.D. from University of California, Berkeley. He was elected an ASME Fellow in 1999 and is a Registered Professional Engineer in the State of California.

Dr. Mehta has been with GE Nuclear Division (now, GE-Hitachi Nuclear Energy) since 1978 and currently holds the position of Chief Consulting Engineer. He has over 35 years of experience in the areas of stress analysis, linear-elastic and elastic-plastic fracture mechanics, residual stress evaluation, and ASME Code related analyses pertaining to BWR components. He has also participated as principal investigator or project-manager for several BWRVIP, BWROG and EPRI sponsored programs at GE, including the Large Diameter Piping Crack Assessment, IHSI, Carbon Steel Environmental Fatigue Rules, RPV Upper Shelf margin Assessment and Shroud Integrity Assessment. He is the author/coauthor of over 40 ASME Journal/Volume papers. Prior to joining GE, he was with Impell Corporation where he directed various piping and structural analyses.

For more than 25 years, Dr. Mehta has been an active member of the Section XI Subgroup on Evaluation Standards and associated working an task groups. He also has been active for many years in ASME's PVP Division as a member of the Material & Fabrication Committee and as conference volume editor and session developer. His professional participation also included several committees of the PVRC, specially the Steering Committee on Cyclic Life and Environmental Effects in Nuclear Applications. He had a key role in the development of environmental fatigue initiation rules that are currently under consideration for adoption by various ASME Code Groups.

MILLER, UREY R.

Mr. Miller is an ASME Fellow and has more than 30 years of experience in the pressure vessel industry. He has participated in ASME Pressure Vessel Code Committee activity for well more than 20 years. He is a Registered Professional Engineer in Indiana and Texas. He is currently a member of the following ASME Boiler and Pressure Vessel Committees:
Boiler and Pressure Vessel Standards Committee
Subcommittee Pressure Vessels—Section VIII
Subgroup Design—Section VIII (Chairman)
Special Working Group for Heat Transfer Equipment (past Chairman)
Special Committee on Interpretations—Section VIII
Subcommittee Design.

Mr. Miller has been the Chief Engineer with the Kellogg Brown & Root Company (KBR), a major international engineering and construction company for the petrochemical industry, since 1992. In this position, he consults on a wide array of subjects including pressure vessesl, heat exchanger, and piping design issues, including application and interpretation of all ASME Code requirements. He has had extensive experience with international projects. He has provided significant engineering support and advice to KBR projects throughout the world. In the role as Chief Engineer, he has traveled extensively providing engineering support for projects in Brazil, Malaysia, Egypt, Algeria, Nigeria, Philippine Islands, South Africa, United Kingdom, Mexico, etc. in addition to a variety of projects in United States. He has experience in refinery, petrochemical, liquefied natural gas, ammonia, phenol, and other types of projects. Previously, he held responsible positions related to process pressure equipment at Union Carbide Corporation and Foster Wheeler Energy Corporation. In addition, he has had over eight years experience in designing pressure vessels for nuclear power generation applications with the Babcock and Wilcox Co. Mr. Miller has a Bachelor's Degree in Mechanical Engineering (cum laud) from the University of Evansville (Indiana).

MOEN, RICHARD A.

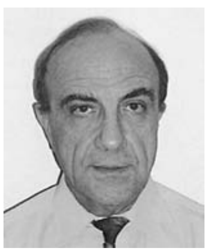
Richard (Dick) Moen has been a member of numerous Boiler and Pressure Vessel Code committees since 1969. Richard (Dick) Moen was an active member of various Boiler and Pressure Vessel Code committees from 1969, until his retirement in 2005. During that time span, he served on the Standards Committee, the Subcommittee on Materials, the Subcommittee on Nuclear

Power, and additional Subgroups and Task Groups serving in those areas. He is a life member of ASM International.

Richard Moen earned a BS degree in Metallurgical Engineering from South Dakota School of Mines and Technology in 1962, with additional graduate studies through the University of Idaho and the University of Washington. He has spent his entire professional career in the field of nuclear energy, beginning in research and development, and then with commercial power plant construction, operation support, and maintenance. He now consults and teaches through Meon Technical Services.

Richard Moen's primary area of expertise is in materials behavior and applications. He has authored numerous papers and has been involved in several national materials handbook programs. And with his long-time involvement in the ASME Boiler and Pressure Vessel Code, he has authored a popular book entitled "Guidebook to ASME Section II, B31.1, and B31.3—Materials Index". His classes are built around that book.

MOKHTARIAN, KAMRAN



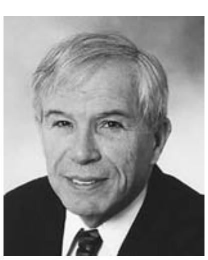
Kam Mokhtarian graduated from the Northwestern University with a Master of Science degree, in 1964. He worked for Chicago Bridge and Iron Company from 1964 through 2000, in a variety of assignments. He was responsible for design and analysis of nuclear vessels and pressure vessels for a number of years. He also provided technical consulting to the engineering staff.

Mr. Mokhtarian has been involved with the ASME B&PV Code Committee, since 1980. He has served as member and chairman of several committees. He was Chairman of Subgroup Design of Subcommittee VIII and the Vice-chairman of Subgroup Fabrication and Inspection. He is presently the Vice-chairman of Subcommittee VIII.

Mr. Mokhtarian is also a member of the Post Construction Standards Committee and the Vice-chairman of the Subcommittee on Flaw Evaluation. He has also served as an associate editor of the ASME's Journal of Pressure Vessel Technology for several years.

Mr. Mokhtarian has been an active member of the Pressure Vessel Research Council (PVRC) since 1980 and has served as Chairman of several committees. He is presently the Chairman of the PVRC. He has authored several WRC Bulletins, including Bulletin 297 that has become a major resource for pressure vessel designers. He has also been teaching a number of pressure vessel related ASME courses.

MOODY, FREDERICK J.



Ph.D., M.S., B.S., Mechanical Engineering (Stanford, Stanford, U. of Colorado) Consulting Engineer, Thermal-Hydraulics, GE Nuclear Energy, 41 years with emphasis on fluid mechanics, thermodynamics, heat transfer, and coupled fluid-structure interaction, pertaining to reactor and containment technology. Adjunct Professor, Thermosciences, San Jose State University, 28 years, Instructor, GE Advanced Engineering Programs. Instructor for ASME Continuing Education courses. Invited courses, lectures in U. S. universities and national labs, Rome, Israel, Holland, Japan, India, Germany, Spain, and Taiwan on unsteady thermofluid behavior. National Academy of Engineering, 2001, Induction into Silicon Valley Engineers Hall of Fame, 2000, ASME PV&PD Award (1999), ASME Fellow (1981); George Westinghouse Gold Medal Award (1980), and Alfred Noble Award for technical paper (1967). Has been Committee chair and co-chair, ASME Fluids Engineering Division, PV&P Division and Associate Editor of ASME Journals.

Participated on NRC-appointed peer review groups, and ongoing consulting assignments with several NRC studies and panels. Publications include *Introduction to Unsteady Thermofluid Mechanics*, Wiley, and *The Thermal-Hydraulics of a Boiling Water Nuclear Reactor*, ANS (co-author), and more than 50 publications in technical journals, and symposium volumes.

MORA, RAFAEL G.

Mr. Rafael Mora is a graduate in Civil Engineering from the University of La Gran Colombia, and holds a Master of Business Administration, UNET-UFPS, Venezuela-Colombia. He is a registered professional engineer in Canada.

Mr. Mora has been working for the pipeline industry for over seventeen years that include pipeline operating; integrity consulting and in-line inspection service companies. He recently joined the National Energy Board as a Technical Leader, Engineering within the Compliance Planning and Analysis Team.

Mr. Mora is currently also a professor in the Pipeline Engineering Master Degree program at the University of Calgary. He has taught courses in pipeline integrity extensively within North and South America and has authored a number of technical papers on this subject.

MORTON, D. KEITH

Mr. D. Keith Morton is a Consulting Engineer at the Department of Energy's (DOE) Idaho National Laboratory, operated by Battelle Energy Alliance. He has worked in the Applied Mechanics Department for nearly 33 years. Mr. Morton has gained a wide variety of structural engineering experience in many areas, including performing nuclear piping and power piping stress analyses, completing plant walkdowns, consulting with the Nuclear Regulatory Commission, developing life extension strategies for the Advanced Test Reactor, performing full-scale seismic and impact testing, and helping to develop the DOE standardized spent nuclear fuel canister. His most recent work activities include performing full-scale drop tests of DOE spent nuclear fuel canisters and developing a test methodology that allows for the quantification of true stress-strain curves that reflect strain rate effects.

Mr. Morton is a Member of the ASME Working Group on the Design of Division 3 Containments, is the Secretary for the ASME Subgroup on Containment Systems for Spent Fuel and High-Level Waste Transport Packagings, and is a Member of the ASME Section III Subcommittee. He has co-authored over twenty-five conference papers, one journal article, and recently co-authored an article on DOE spent nuclear fuel canisters for *Radwaste Solutions*.

Mr. Morton received a B.S. in Mechanical Engineering from California Polytechnic State University in 1975 and a Masters of Engineering in Mechanical Engineering from the University of Idaho in 1979. He is a Registered Professional Engineer in the state of Idaho.

MURRAY, ALAN

Dr. Alan Murray is the Professional Leader Engineering at the National Energy Board in Calgary and an Adjunct Professor in the Chemical Engineering Department of the University of Calgary.

He is a graduate of The Queen's University of Belfast, Northern Ireland in Civil Engineering and Mechanical Engineering and has spent most of his career in Design and Development activities mostly in heavy engineering. He has held a number of senior management positions with a major North American pipeline company and was founding chair of the ASME Pipeline Systems Division. He is the co—author of the ASME book *Pipeline Design and Construction: A Practical Approach*, and has published over 50 papers on a variety of engineering topics.

NASH, DAVID

Dr. Nash is the Vice-Dean of Engineering and a Reader in Mechanical Engineering at the University of Strathclyde in Glasgow, Scotland. After spending several years with a vessel fabricator, Dr Nash joined the Department as a researcher where he gained an MSc and PhD working on local load and saddle support contact problems. His research interests lie broadly in the area of pressure equipment design procedures, and most recently in the area of bolted joints and sealing technology. He has written over 90 papers and authored and contributed to several books. He has co-written and organised a suite of pressure equipment design courses for industry and these have been run on an annual basis since 1986.

Dr Nash is a Fellow of the Institution of Mechanical Engineering and a Chartered Engineer and has been an ASME member since 1987. He was made an ASME Fellow in 2006. He is the present Vice-Chairman of the Pressure Systems Group of the Institution of Mechanical Engineers, is a member of the British Standards Committee for Design Methods (BSi PVE-1–15) and is the current UK national representative to EPERC, the European Equipment Research Council.

NICKERSON, DOUGLAS B.

Douglas B. Nickerson graduated from CalTech with a BSME. He was a registered Engineer in the State of California and is a Fellow of ASME. He worked in the Aerospace Industry until 1965 when he founded his consulting business, Stress Analysis Associates. During his tenure in the Aerospace Industry he developed the Hi-V/L ® pump for aerocraft booster pump application. He was active in dynamic analyses of pumps and valves as a consultant to most of the commercial pump manufacturers including those manufacturing nuclear pumps.

As a corollary to the dynamic analysis of pumps and valves Mr. Nickerson developed a number of computer programs to carry out these analyses. Some of these programs were successfully marketed. Not only active in Engineering he helped organize the Fluid Machinery Section of the Local ASME Section. In recognition of his activities he was made "Engineer of the Month" of Southern California for August 1973.

Mr. Nickerson was on the SURF Board of CalTech and was formerly its Chairman.

Douglas Nickerson had served on a number of ASME Section III Committees and was Chairman of QR Subcommittee of QME. Mr. Douglas B. Nickerson passed away since the completion of the first edition.

NORDSTROM, EDWIN A.

On the personal side, Ed is a native of Kansas who was educated at the University of Kansas as an undergraduate and the Massachusetts Institute of Technology where he earned graduate degrees in both Chemistry and Management – the latter from the Sloan School. He served in administrative positions for 16 years on school boards and 40 years in the Episcopal Church.

Without an engineering degree, Ed rose to be Manager of Process Engineering for a chemical company and then to VP Engineering for A O Smith Water Products Division. In the latter post, he became active in ASME where he has served on Section IV for 25 years. This activity continued across job changes to Amtrol [Manager, Hot Water Maker Sales]; Viessmann Manufacturing [COO for US operations]; Gas Appliance Manufacturers Association; and Heat Transfer Products.

O'DONNELL, WILLIAM J.

Bill O'Donnell has Engineering Degrees from Carnegie Mellon University and the University of Pittsburgh. He began his career at Westinghouse Research and Bettis where he became an Advisory Engineer. In 1970 Bill founded O'Donnell and Associates, an engineering consulting firm specializing in design and analysis of structures and components. The firm has done extensive work in the evaluation of structural integrity, including corrosion fatigue, flaw sensitivity, crack propagation, creep rupture and brittle fracture. Dr. O'Donnell has published 96 papers in engineering mechanics, elastic-plastic fracture mechanics, strain limits and damage evaluation methods. He is Chairman of the Subgroup on Fatigue Strength and a Member of the Subcommittee on Design of the ASME Code. He has patents on mechanical processes and devices used in plants worldwide. He is recognized expert in Failure Causation Analyses. Dr. O'Donnell has given invited lectures at many R&D laboratories, design firms and universities. He is a registered Professional Engineer. He received the National Pi Tau Sigma Gold Medal Award "For Outstanding Achievement in Mechanical Engineering" and the ASME Award for "Best Conference Technical Paper" in 1973 and 1988. The Pittsburgh Section of ASME named Bill "Engineer of the Year." (1988) He was awarded the ASME PVP Medal (1994) and received the University of Pittsburgh ME Department's Distinguished Alumni Award (1996) and Carnegie Mellon University's 2004 Distinguished Achievement Award for distinguished service and accomplishments in any field of human endeavor. He is a Fellow of the ASME and is listed in the Engineers Joint Council "Engineers of Distinction," Marquis "Who's Who in Science and Engineering" and "Who's Who in the World."

OLSON, DAVID E.

David Olson, as part of his career at Sargent & Lundy LLC, has been involved in solving piping and rotating equipment vibration problems at over 50 nuclear and fossil power plants. He has managed the design and successful implementation of preoperational and initial startup piping test programs at BWR and PWR plants. Throughout his career has been responsible for diagnosing and solving field problems with piping systems at both nuclear and fossil power plants. Mr. Olson has also been responsible for initial design, backfits and modifications of both nuclear and fossil power plants. As part of this experience he has developed expertise in vibration analysis, testing and field problem resolution. Mr. Olson is the current and long standing Chairman of ASME Operation and Maintenance Subgroup on Piping Systems, the group responsible for writing the OM-3 standard on piping vibration. Mr. Olson has published numerous technical papers on vibration and piping dynamics, testing and design and has also given numerous training seminars.

Mr. Olson has also managed various industry initiatives including the development of improved guidance for piping design and analysis, piping operability criteria, the development of risk informed methods to reduce seismic loads, and the patented design of radiation shielding pipe insulation. University of Illinois – B.S. Engineering, University of Chicago - MBA, Registered Professional Engineer.

OSAGE, DAVID A.

Mr. Osage, President and CEO of the Equity Engineering Group in Shaker Heights, Ohio, is internationally recognized as an industry expert and leader in the development and use of FFS technology. As the architect and principal author of API 579 *Fitness-For-Service*, he developed many of the assessment methodologies and supporting technical information. As the chairperson for the API/ASME Joint Committee on Fitness-For-Service, he was instrumental in completing the update to API 579 entitled API *579-1/ASME FFS-1 Fitness-For-Service*. Mr. Osage provides instruction on Fitness-For-Service technology to the international community under the API University Program.

Mr. Osage is also a recognized expert in the design of new equipment. As the lead investigator and principal author of the new *ASME, Section VIII, Division 2, Boiler and Pressure Vessel Code*, he developed a new organization and writing style for this code and was responsible for introducing the latest developments in materials, design, fabrication and inspection technologies. These technologies include a new brittle fracture evaluation method, new design-by-analysis procedures including the introduction of elastic-plastic analysis methods, and a new fatigue method for welded joints. Mr. Osage has delivered lectures on the new pressure vessel code in Europe and Japan and will be offering a training course highlighting advantages of the new code for use with refinery and petrochemical equipment.

Mr. Osage was a lead investigator in revamping the API Risk-Based Inspection (RBI) technology and software. The main focus of this effort was a clean sheet re-write of API 581 *Risk-Based Inspection* and the development of a new version of the API RBI software. He is currently working on the next generation of RBI technology where Fitness-For-Service assessment procedures will be used to compute the Probability of Failure for Risk-Based Inspection.

As an Adjunct Visiting Assistant Professor at Stevens Institute of Technology, Mr. Osage has taught graduate level courses in strength of materials and elasticity, structural analysis and finite element methods, and structural optimization.

OSWEILLER, FRANCIS

Francis Osweiller got international recognition for his expertise in French, European and ASME Pressure Vessel Codes & Standards. He has been the head of the French delegation to CEN/TC 54 (European Technical Committee for Unfired Pressure Vessels) for several years and has chaired several committees such as Simple Pressure Vessels, Testing & Inspection, Tubesheets and Bellows. Mr. Osweiller has been actively involved in Europe with the development of the Pressure Equipment Directive and the new CEN Standard for Unfired Pressure Vessels. He gave several courses on these issues in France UK and USA. As member of the Main Committee of CODAP, he developed several design rules for the French Pressure Vessel Code (CODAP). His main contribution was the development of Tubesheet Heat-exchanger rules to replace the existing (TEMA) rules.

Francis Osweiller obtained a Mechanical Engineering degree in Paris, France. He started his career at CETIM-France with FEM analysis applied to pressure vessels. He has published more than 40 papers in France, UK, Germany and US on European Codes, ASME Code and Pressure Equipment Directive and gave lectures at AFIAP, ICPVT (International Conference of Pressure Vessel Technology) and ASME-PVP (Pressure Vessel & Piping Conference). He has been the representative for France at ICPVT andISO/TC11.

Since 1985 Osweiller has been actively involved in ASME Boiler and Pressure Vessel Code organization where he is member of SCII/International Material Specifications, SCSVIII/SWG on Heat Transfer Equipment, Post Construction Main Committee, Board on Pressure Vessel Technology and Council on Codes and Standards. His principal accomplishment is his role for the publication of common rules in ASME Code, European Code and French Code for the design of tube-sheets and expansion bellows. Osweiller is the recipient of several awards and certificates from ASME and PVP and was elevated to the grade of Fellow by ASME in 2001 and is listed in the Who's Who in the World.

PAPPONE, DAN

Mr. Daniel Pappone is Chief Consulting Engineer in Plant Performance at GE-Hitachi Nuclear Energy. He joined GE in 1978. Mr. Pappone has extensive experience in safety evaluations for BWR accident conditions with a primary focus on the vessel and containment response to Loss-of-Coolant Accidents. He is involved in the ongoing development of the generic extended power uprate programs and has held lead technical positions in several stretch and extended power uprate projects.

Currently, Mr. Pappone has been leading research into understanding the fatigue loading acting on BWR steam dryers. His past responsibilities have included degraded core cooling studies, Emergency Procedure Guideline development, and the design, plant application and installation of safety parameter display and plant monitoring computer systems. He brings an overall integrated perspective to the projects, including analysis, system design, operations, and regulatory aspects. Mr. Pappone holds a BS degree in Nuclear Engineering from the University of California, Los Angeles.

PARECE, MARTIN

Martin Parece is Chief Engineer and Vice President, Technology for AREVA NP, Inc. He is responsible for technical oversight and configuration control of pressurized water reactor and high temperature gas reactor designs planned for deployment in North America.

Mr. Parece has B.S. and M.S. degrees in Nuclear Engineering from the University of Illinois and is a member of the American Nuclear Society. During the last 26 years with Babcock & Wilcox, Framatome and AREVA NP, he has gained extensive experience in safety analysis, core reload analysis, plant performance analysis, plant simulation, computer code development, accident mitigation, operator guidance, thermal-hydraulics, plant auxiliary and control systems, Class 1 component design, and licensing. Mr. Parece is the author of numerous papers and topical reports, he also holds a patent on a method and system for emergency core cooling. Mr. Parece is a highly regarded speaker on reactor power uprates, nuclear power plant safety and new reactors.

PASTOR, THOMAS P.

Mr. Pastor has over thirty one years experience working in the areas of stress analysis and pressure vessel design. He holds a Bachelors and Masters degree in Civil Engineering from the University of Connecticut, with emphasis on structural design and analysis.

Mr. Pastor began his career with Combustion Engineering in 1977, where he was a member of the structural analysis group, responsible for performing load analyses of nuclear reactor inernals subject to seismic and LOCA events. Over nine years he developed significant expertise in performing finite element analyses and scientific programming.

In 1986 Mr. Pastor joined the Hartford Steam Boiler Inspection and Insurance Co. (HSB) working in the Codes and Standards Group in Hartford, Ct. During his 22 year tenure at HSB, Mr. Pastor rose from staff engineer, to Manager Codes & Standards, Director, and presently Vice-President Code Services. He has managed the Codes & Standards (C&S) Group for over 17 years, and led the development of several knowledge based databases which are used today to provide Code technical support to over 3000 ASME Certificate Holders and Inspectors worldwide. Mr. Pastor's ASME code expertise is in pressure vessels, and he has taught basic to advanced seminars on Section VIII, Division 1 over 100 times to audiences around the world. He has authored numerous technical papers on the subject of stress analysis and ASME Code developments,

Mr. Pastor is a licensed Professional Engineer in the states of Connecticut and Indiana. He is currently serves on several ASME Committees such as Codes & Standards Board of Directors, Board on Hearings and Appeals, Continuous Improvement Committee, Board on Pressure Vessel Technology, BPV Technical Oversight Management Committee (Vice-Chairman), Standards Committee on Pressure Vessels – Section VIII (Chairman), Subgroup Design – Section VIII, and Special Committee on Interpretations – Section VIII.

PERRAUDIN, GERARD

Gerard Perraudin is a recognized authority on materials in pressure vessel technology. Initially he worked on a variety of assignments for the French Technical Center of Mechanical industries from 1970 through 1980. There on he has been the supervisor of inspectors on a refinery of Exxon Chemical (1980–1983) and later was employed by TECHNIP, a major international engineering and construction company. He has been actively involved in various petroleum and chemical industries over the world (Exxon, BP, Elf). Mr. Perraudin is the Chairman of CODAP Committee responsible for French Pressure Vessel Code. Based on his expertise of Codes he is actively involved in several French and European Code working committees.

PITROU, BERNARD

Bernard Pitrou has more than 40 years of experience in the piping industry. He held the position of manager in Design and Calculations Department, ENTREPOSE (currently called ENDEL). He was engaged in design and analysis of power and process piping as well as nuclear and transportation piping. He is a member of the Pressure Vessel and Piping Committee of the Technical Center of Mechanical Industries and was responsible for several new theoretical developments in the field of piping such as flanges and large openings. Mr. Pitrou served on the first French Piping Committee (1970) created by the SNCT (French Pressure

Equipment Manufacturer's Association) and is now the Chairman of CODETI Committee responsible for French Piping Code. He has been active in the European Standardization and is currently Chairman of Working Group 1 (General) and 3 (Design) of the European Technical Committee 267 in charge of Industrial Piping.

PORTER, MICHAEL (MIKE) A.

Michael (Mike) A. Porter is the Principal Engineer of Porter McGuffie, Inc. In the 40 years since he obtained a Mechanical Engineering degree from the University of Illinois at Champaign/Urbana, he has worked in the natural gas industry, managed a construction firm and served as a consultant to numerous industries in the fields of vibration, thermal and stress analysis. He has published more than 30 ASME conference and journal publications, most of which have been for the Piping and Pressure Vessel Division. His most recent PVP Journal paper "A Suggested Shell/Plate Finite Element Nozzle Model Evaluation Procedure" provides guidance for the application of Finite Element (FE) analysis as it applies to the Boiler and Pressure Vessel Code. He has also authored papers for the Acoustical Society of America and published two books on the application of the FE method of analysis.

Mike has extensive experience in the diagnosis and solution of stress and vibration problems. Much of this experience stems from his work as a technical services engineer at Panhandle Eastern Pipeline Company and as a consultant with Bolt Beranek and Newman. Building on this base, Mr. Porter has established an integrated computational facility for the analysis of mechanical systems and their interaction with fluids. These capabilities include the codes for linear and non-linear stress analysis, computational fluid dynamics and gas/liquid pulsation FE analyses. For the past 15 years, Mike has served as a member of the Design and Analysis Committee of the Piping and Pressure Vessel Division of ASME. He has served as the Technical Program Representative for this committee as well as developing and chairing numerous conference sessions on the subjects of vibration, water hammer, pulsation and stress.

The projects overseen by Mr. Porter cover a broad range of industries and topics. Representative examples include the petrochemical industry (analyses of numerous pressure vessels and components); the aerospace industry (analyses for the International Space Station and FEA training for NASA personnel); and building dynamics (design review and analysis of multistoried structures for the Environmental Protection Agency and the National Ocean Service, among others). These projects have included linear and non-linear stress analyses as well as computational fluid dynamics, structural dynamics and thermal analyses.

RAHOI, DENNIS

D. W (Dennis) Rahoi is an authority on materials used in the pharmaceutical-/biotechnology, chemical process, fossil fuel, and nuclear power industries. The author of more than 50 papers on materials, corrosion and oxidation, he received the Prime Movers Award in Thermal Electric Generating Equipment and Practice by Edison Electric Institute for work published on solving problems in high pressure feedwater heaters. He currently consults in material selections, failure analysis and does other forensic metallurgical work. Mr. Rahoi is also the current editor of Alloy Digest (an ASM International, Inc. publication) and is an active consultant to the Nickel Institute. Mr. Rahoi was the first chairman of NACE's Power Committee and is active on many stainless steel ASTM and ASME (including B31) materials committees. He is the current chairman of the ASME Sub-Group Non-Ferrous Materials for Section II and holds a master's degree in metallurgical engineering from Michigan Technological University.

Mr. Rahoi's work on writing many new ASTM specifications, his active sponsoring of 10 pipe and tube specifications and his active involvement in Welding Research Council and EPRI research proposals on welding and repair keep him in constant touch with the needs of industry. This, combined with his other experiences and consulting, allow him to contribute to the current chapter in this book with authority.

RANA, MAHENDRA D.

Mahendra, an ASME Fellow has a bachelor's degree in mechanical engineering from M.S. University in Baroda, India, and a master's degree in mechanical engineering from the Illinois Institute of Technology, Chicago, Illinois. He is a registered professional engineer in New York State. He is an Engineering Fellow working in the Global Supply System Engineering Department of Praxair, Inc. for the last 34 years. He is involved in the areas of fracture mechanics, pressure vessel design, pressure vessel development, and materials testing. He is also involved in the structural integrity assessment, and fracture control programs of pressure vessels and the member of Board on Pressure Technology, Codes and Standards. Mahendra became the Chairman of the Subgroup on Design and Materials of Subcommittee XII when it was formed in 1996. He is a member of several other ASME Boiler and Pressure Vessel Code committees: member of Subcommittee VIII, member of joint API/ASME Fitness- for Service Committee and the member of the Main Committee of the Boiler and Pressure Vessel Code and the member of Board on Pressure Technology, Codes and Standards. Mahendra is also a member of several ISO, ASTM and CGA (Compress Gas Association) standards committees. He is a Chairman of the Codes and Standards Technical Committee of Pressure Vessel and Piping Division of ASME. He has received several awards from the Pressure Vessel and Piping Division for his contribution in organizing Codes and Standards sessions in Pressure Vessel and Piping Conferences. He has given several lectures in the pressure vessel technology topics in the USA and abroad. He has taught a course on ASME Section VIII, Division 1 to ASME section of Buffalo New York. He is the co-recipient of two patents and the co-author of 25 technical papers. He also has written several technical reports for his company.

RANGANATH, SAM

Dr. Sam Ranganath is the Founder and Pricipal at XGEN engineering, Sam Jose, CA. XGEN, founded in 2003, provides consulting services in fracture mechanics, materials, ASME Code applications and structural analysis to the power plant industry. Before that he held various leadership positions at General Electric for 28 years. Dr. Ranganath is a Fellow of the ASME and has been active in the development of Section III and Section XI, ASME Code rules for the evaluation and inspection of nuclear pressure vessel components. Sam has a Ph.D. in Engineering from Brown University, Providence, RI and an MBA from Santa Clara University, Santa Clara, CA. He has also taught Graduate Courses in Mechanical Engineering at Santa Clara University and Cal State University, San Jose for over 15 years.

RAO, K. R.

KR Rao retired as a Senior Staff Engineer with Entergy Operations Inc. and was previously with Westinghouse Electric Corporation at Pittsburgh, PA and Pullman Swindell Inc., Pittsburgh, PA. KR got his Bachelors in Engineering from Banaras University, India with a Masters Diploma in Planning from School of Planning & Architecture, New Delhi, India. He completed Post Graduate Engineering courses in Seismic Engineering, Finite Element and Stress Analysis, and other engineering subjects at Carnegie Mellon University, Pittsburgh, PA. He earned his Ph.D., from University of Pittsburgh, PA. He is a Registered Professional Engineer in Pennsylvania and Texas. He is past Member of Operations Research Society of America, (ORSA).

KR was Vice President, Southeastern Region, ASME International. He is a Fellow of ASME, active in National, Regional, Section and Technical Divisions of ASME. He has been the Chair, Director and Founder of ASME EXPO(s) at Mississippi Section. He was a member of General Awards Committee of ASME International. He was Chair of Codes & Standards Technical Committee, ASME PV&PD. He developed an ASME Tutorial for PVP Division covering select aspects of Code. KR is a Member, Special Working Group on Editing and Review (ASME B&PV Code Section XI) for September 2007 – June 2012 term.

Dr. Rao is a recipient of several Cash, Recognition and Service Awards from Entergy Operations, Inc., and Westinghouse Electric Corporation. He is also the recipient of several awards, Certificates and Plaques from ASME PV&P Division including Outstanding Service Award (2001) and Certificate for “Vision and Leadership” in Mississippi and Dick Duncan Award, Southeastern Region, ASME. Dr. Rao is the recipient of the prestigious ASME Society Level Dedicated Service Award.

Dr. Rao is a Fellow of American Society of Mechanical Engineers, Fellow of Institution of Engineers, India and a Chartered Engineer, India. Dr. Rao was recognized as a ‘Life Time Member’ for inclusion in the Cambridge “Who’s Who” registry of executives and professionals. Dr. Rao was listed in the Marquis 25th Silver Anniversary Edition of “Who’s Who in the World” as ‘one of the leading achievers from around the globe’.

REEDY, ROGER F.

Roger F. Reedy has a B.S. Civil Engineering from Illinois Institute of Technology (1953). His professional career includes the US Navy Civil Engineering Corps, Chicago Bridge and Iron Company (1956–1976). Then he established himself as a consultant and is an acknowledged expert in design of pressure vessels and nuclear components meeting the requirements of the ASME B&PV Code. His experience includes design, analysis, fabrication, and erection of pressure vessels and piping components for nuclear reactors and containment vessels. He has expertise in components for fossil fuel power plants, and pressure vessels and storage tanks for petroleum, chemical, and other energy industries. Mr. Reedy has been involved in licensing, engineering reviews, welding evaluations, quality programs, project coordination and ASME Code training of personnel. He testified as an expert witness in litigations and before regulatory groups.

Mr. Reedy has written a summary of all changes made to the ASME B&PV Code in each Addenda published since 1950 which is maintained in a computer database, RA-search.

Mr. Reedy served on ASME BP&V Code Committees for more than 40 years being Chair of several of them, including Section III for 15 years. Mr. Reedy was one of the founding members of the ASME PV&P Division. Mr. Reedy is registered Engineer in seven states. He is a recipient of the ASME Bernard F. Langer Award and the ASME Centennial Medal and is a Life Fellow of ASME.

REINHARDT, WOLF D.

Wolf D. Reinhardt earned a Dipl. Ing. Degree in Mechanical Engineering from the Technical University in Braunschweig, Germany, and a Ph.D. from the University of Waterloo, Canada. He is a registered Professional Engineer in Ontario.

His current position as Senior Section Head, Computational Mechanics Development, at Atomic Energy of Canada encompasses the application of numerical analysis to problems in the design, analysis and fitness-for-service evaluation of reactor components. He is also engaged in performing applied research programs for the Canadian nuclear industry.

Previously, he worked as a Lead Engineer in Nuclear Engineering at Babcock & Wilcox Canada on the design and analysis of nuclear components, principally steam generators and heat exchangers, and in the in-service assessment of steam generator tubes.

Wolf is holding an appointment as adjunct professor at Memorial University in Newfoundland. He has published over 50 technical papers in the fields of nonlinear vibration, metal plasticity, computational methods for the nonlinear analysis of pressure vessels, elastic-plastic pressure vessel design and fracture mechanics. He received various Best Paper Awards at ASME PVP conferences and at the ASTM National Symposium on Fatigue and Fracture Mechanics. His current research interests include plastic shakedown analysis as well as structural performance criteria and in-service assessment of piping and reactor components.

Wolf Reinhardt is a member of the ASME B&PV Code Subgroup Design Analysis and participates in the Task Group Elastic-Plastic FEA. He is also contributing to the PVPD Computer Technology Technical Committee and acted as Technical Program Representative at PVP Conferences for this committee. He taught courses on elastic-plastic design using Section III and Section VIII rules, on methods for fitness-for-service assessment, and on the design, analysis and fabrication rules of Section III.

RICCARDELLA, PETER C.

Pete Riccardella received his Ph.D. from Carnegie Mellon University in 1973 and is an expert in the area of structural integrity of nuclear power plant components. He co-founded Structural Integrity Associates in 1983, and has contributed to the diagnosis and correction of several critical industry problems, including:

- Feedwater nozzle cracking in boiling water reactors
- Stress corrosion cracking in boiling water reactor piping & internals
- Irradiation embrittlement of nuclear reactor vessels
- Primary water stress corrosion cracking in pressurized water reactors
- Turbine-generator cracking and failures.

Dr. Riccardella has been principal investigator for a number of EPRI projects that led to advancements and cost savings for the industry. These include the **FatiguePro** fatigue monitoring system, the **RRingLife** software for turbine-generator retaining ring evaluation, **Risk-Informed Inservice Inspection methodology** for nuclear power plants, and several **Probabilistic Fracture Mechanics** applications to plant cracking issues. He has led major failure analysis efforts on electric utility equipment ranging from transmission towers to turbine-generator components and has testified as an expert witness in litigation related to such failures.

He has also been a prime mover on the ASME Nuclear Inservice Inspection Code in the development of evaluation procedures and acceptance standards for flaws detected during inspections. In 2002 he became an honorary member of the ASME Section XI Subcommittee on Inservice Inspection, after serving for over twenty years as a member of that committee. In 2003, Dr. Riccardella was elected a Fellow of ASME International.

RODABAUGH, EVERETT C.

Mr. Rodabaugh has B.S. from Iowa State College, Ames, Iowa (1939) and M.S. from the University of Louisville, Kentucky (1959). He is a Registered Professional Engineer in the State of Ohio.

He has extensive experience in power plant operations and the design of piping and pressure vessels. His previous experience was with Joseph E. Seagram Co., E.I. duPont, U.S. Maritime Service (1943–1946). He was with Tube Turns in Louisville, Kentucky (1946–1961). Mr. Rodabaugh was with Bat-telle-Columbus Laboratories, Columbus, Ohio (1961–1981).

In 1981 he started his own consulting firm and since 1991 he has continued his work on piping and pressure vessels as a consultant.

Mr. Rodabaugh was a member of the original ASME Design Group that prepared ANSI B31.7. He was a member of several ASME Code committees including Section III Committees and Subgroup on Design and Working Group on Piping. Everret Rodabaugh was Chairman of ANSI B16 and its Subcommittees.

Mr. Rodabaugh is also a member of the Pressure Vessel Research Council, Design Division and its Subcommittee on Piping, Nozzles and Vessels. Mr. Rodabaugh published over 60 articles and has written over 100 reports. Everret Rodabaugh is a Fellow in the ASME and received the ASME Bernard F. Langer award in 1998.

RODERY, CLAY D.

Clay Rodery is Technical Authority/ Fixed Equipment for BP North American Products. He has over 27 years of experience consulting in the areas of pressure vessels and piping to Amoco and BP refining, chemicals, and upstream facilities and projects worldwide. After receiving his BSCE from Purdue University in 1981, he joined Amoco Oil Company's Texas City Refinery, where he was involved in project, maintenance, and inspection engineering. In 1990, he moved to Amoco Oil's Refining & Transportation Engineering Department as pressure vessel specialist. In 1995, he became the principal vessel specialist within Amoco Corporation's Worldwide Engineering & Construction Department. In 1999, he moved to BP Chemicals' Technology & Engineering Department as pressure vessel and piping specialist. He became BP Chemicals' Pressure Vessel and Piping Advisor in 2004, until moving to his current role in 2006.

Clay began participating in ASME Boiler and Pressure Vessel Code activity in 1993. He joined the Subgroup on Fabrication & Inspection (Section VIII) in 1997, and the Subgroup on Design in 1999. In May 2000, he was appointed Chairman of the Subgroup on Fabrication & Inspection and member of the Subcommittee on Pressure Vessels. Clay is a member of the ASME Post Construction Standards Committee, and Vice Chair of the Subcommittee on Repair and Testing. He is also a member of the Special Working Group on Flange Joint Assembly.

As a member of the Design & Analysis Technical Committee of the ASME Pressure Vessels and Piping Division, Clay has served as an Author, Session Developer/Chair, Editor, Technical Program Representative, and Tutorial Presenter. Clay is a member of the API Subcommittee on Inspection and the Task Group on Inspection Codes. He is former Team Leader of the Process Industry Practices (PIP) Vessel Function Team.

RODGERS, DOUGLAS K.

Doug Rodgers earned a B.A.Sc (1982) in Engineering Science, with a specialization in Material Science from the University of Toronto and an M.A.Sc (1992) in Mechanical Engineering from the University of Ottawa. Doug has been a member of the ASM International (formerly the American Society for Metals) since 1982 and is currently a Chapter Sustaining Member of the Ottawa Valley Chapter. Doug is a registered professional engineer in the Canadian provinces of Ontario and New Brunswick, and has been a member of ASME since 1999.

Doug has worked for Atomic Energy of Canada Limited for more than 20 years, specializing in performance characteristics of CANDU power reactor materials. Initially with the engineering design group, Doug transferred to the Metallurgical Engineering Branch of the Reactor Materials Division where he was responsible for failure analysis and material surveillance testing of CANDU fuel channel components. With a well-established interest in fracture phenomena, Doug spent several years studying delayed hydride cracking in Zr-2.5Nb pressure tube materials, later becoming the manager of the Materials and Mechanics Branch, responsible for research and development programs relevant to metal fracture. Doug is currently Director of the Fuel Channels Division, incorporating material expertise, varying from computational mechanics, metallurgy, surface science, corrosion, deformation, and fracture, as it is applied to design, research & development, and services for CANDU nuclear reactor systems.

ROWLEY, C. WESLEY

C. Wesley Rowley is Vice President, Engineering & Technical Services, with The Wesley Corporation in Tucson, AZ. He has been with TWC since 1985. Mr. Rowley manages engineering and non-metallic structural repair activities for nuclear power plants. He has published numerous reports and technical papers for EPRI, ASME, ICONE Conferences, Pump & Valve Symposiums, and other nuclear industry events. He is a recognized expert on risk-informed Inservice Testing, as well as non-metallic materials and non-metallic structural repairs.

Mr. Rowley has been a member of the ASME Board on Nuclear Codes and Standards for over fifteen years. He is also a member of the ASME Post Construction Committee, the Subcommittee on Repair & Testing, and the Chairman of the Non-metallic Repair Project Team. Additionally he has been the Chairman of the ASME BPV/Subcommittee II, Materials/Special Working Group, Nonmetallic Material since 2002. He is the past Chairman of the ASME BPV Joint Subcommittee III/XI Project Team for Plastic Pipe. ASME past Vice President, Nuclear Codes & Standards and past Chairman, Board on Nuclear Codes & Standards. He is currently a member of the ASME BPV/Subcommittee III/Special Working Group on Polyethylene Pipe. ASME, Member, Operations & Maintenance Committee (and Sub-group ISTE, Risk-Informed Inservice Testing).

Mr. Rowley is a retired Submarine Captain in the U. S. Naval Reserve. He has a M.A. degree in International Relations and Strategic Studies from the Naval War College (1986). He also has a B.S. in General Engineering (1965) and M.S. in Nuclear Engineering from the University of Illinois (1967). Mr. Rowley is a Registered Professional Engineer.

SAMMATARO, ROBERT F.

The late Mr. Sammataro was Proto-Power's Program Manager — ISI/IST Projects. He was responsible for Proto-Power's Inservice Inspection (ISI) and Inservice Testing (IST) programs. These programs included development and implementation of programs involving ISI, IST, design integrity, design reconciliation, 10CFR50, Appendix J, integrated leakage rate testing, and in-plant and out-plant training and consulting services.

Mr. Sammataro was also responsible for Proto-Power's ISI and IST Training Programs has developed Proto-Power's three-day *Workshop on Containment Inservice Inspection, Repair, Testing, and Aging Management*. He was recognized as an expert in containment inservice inspection and testing.

Mr. Sammataro was the past Chair of the ASME PV&P Division (1999–2000), General Chair of PVP Conference (1999) and was the Technical Program Chair (1998).

He was a member and chair of an ASME Section XI Subgroup and a member of an ASME Section XI Subgroup Subcommittee. He was a past member of the ASME BP&V Code Main Committee (1989-1994). Mr. Sammataro was an ASME Fellow. Mr. Sammataro earned BSCE and MSCE from Rensselaer Polytechnic Institute.

SCOTT, BARRY

Barry Scott is currently Director of Quality Assurance Department (Power) with responsibility to provide QA/QC support for the engineering, procurement and construction phases of Power projects. Barry has experience in the development, implementation and auditing of Quality Programs. He has considerable knowledge of industry Quality Standards, including ISO 9000, 10CFR50 Appendix B, NQA 1 and Government (DOE, DOD) requirements. Barry has extensive experience with projects and project engineering management with special expertise in the structural design of Nuclear Power Plant structures including design of reinforced concrete Containment structures. Barry has been a Member of various ASME Section III committees including Subgroup on General Requirements, Subcommittee on Nuclear Power and Joint ASME-ACI Committee on Concrete Components for Nuclear Service for more than 30 years.

Barry has a Master of Science in Civil Engineering from Drexel University and is a licensed PE (Civil Engineering) in the states of Pennsylvania, California and Washington. He is a certified Lead Auditor in accordance with the requirements of ASME NQA-1 and previously held certification as an ACI Level III Concrete Inspector as required by the ASME Section III Division 2 Code.

SIMOLA, KAISA

Dr. Kaisa Simola is a senior research scientist with 20 years research experience in risk and reliability analysis, analyses of nuclear power plant operating experience, ageing analyses, and risk-informed decision making. Presently her main area of interest is risk-informed in-service inspections at nuclear power plants. She has worked for the Technical Research Centre of Finland, VTT, since 1987. In 2004–2006 she was a Visiting Scientist at the Joint Research Centre of the European Commission in Petten, the Netherlands. She is the chairperson of the Task Group on Risk of the European Network for Inspection and Qualification (ENIQ). She is also a member of the board of directors of the European Safety, Reliability and Data Association (ESReDA).

SIMONEN, FREDRIC A.

Fredric A. Simonen earned B.SM.E in 1963 from Michigan Technology University and a Ph.D. in Engineering Mechanics form Stanford University in 1966. Since joining Pacific Northwest Laboratory in 1976, and before that at the Battelle Columbus Division, Dr. Simonen has worked in the areas of fracture mechanics and structural integrity. His research has addressed the safety and reliability of nuclear pressure vessels and piping as well as other industrial and aerospace structural components.

Since the early 1980's he has been the lead for several studies for the U.S. Nuclear Regulatory Commission (NRC) of the effects of pressurized thermal shock on the failure probability of reactor pressure vessels. This work has advanced the technology of probabilistic fracture mechanics and has developed methods for estimating the number and sizes of flaws in vessel piping welds. During the 1990's Dr. Simonen was a leader on the behalf of NRC and the American Society of Mechanical Engineers in developing the technology and furthering the implementation of risk-informed methods for the inspection of nuclear piping systems.

Dr. Simonen is a member of the Section XI Working Groups on Implementation of Risk-Based Inspection, Flaw Evaluation, and Operating Plant Criteria. He is also a member of the ASME Committee on Nuclear Risk Management and the ASME Research Committee on Risk-Based Technology. He has published over 200 papers, articles and reports in the open literature.

SIMS, J. ROBERT, J.

Mr. Sims is a recognized authority in the field of pressure equipment, with over years experience in risk based technologies for optimizing inspection and maintenance decisions, high pressure equipment, and mechanical integrity evaluation of existing equipment. He has been with Becht Engineering since 1998. Prior to that, he worked for more than thirty years with Exxon as a pressure equipment specialist, developed risk based decision-making tools, led a multi-disciplinary team in development of the flaw evaluation guide that was used as the basis for the API-579 Standard on Fitness for Service, and designed a 30,000 psi reactor vessel. Other positions within Exxon included design and operation of high pressure equipment used in the production of low density polyethylene at facilities worldwide.

Bob is the current Senior Vice President of Codes and Standards of ASME. He is a member of several ASME Committees, such as the Council on Codes and Standards, the B&PV Code Subcommittee VIII for Pressure Vessels, and he chairs the Special Working Group on High Pressure Vessels.

Bob is also the past Chair of the ASME Post Construction Committee, and chair of the Pressure Vessel Research Council Committee on Continued Operation of Equipment. He was previously a member of ASME B31.3 Process Piping Code Committee and Chair of the B31.3 Task Group on High Pressure Piping. He is an ASME Fellow and has more than 20 publications and two patents.

SINGH K. P. (KRIS)

Dr. K.P. (Kris) Singh is the President and Chief Executive Officer of Holtec International, an energy technology company that he established in 1986. Dr. Singh received his Ph.D. in Mechanical Engineering from the University of Pennsylvania in 1972, a Masters in Engineering Mechanics, also from Penn in 1969, and a B.S. in Mechanical Engineering from the Ranchi University in India in 1967.

Since the mid-1980s, Dr. Singh has endeavored to develop innovative design concepts and inventions that have been translated by the able technology team of Holtec International into equipment and systems that improve the safety and reliability of nuclear and fossil power plants. Dr. Singh holds numerous patents on storage and transport technologies for used nuclear fuel, and on heat exchangers/pressure vessels used in nuclear and fossil power plants. Active for over thirty years in the academic aspects of the technologies underlying the power generation industry, Dr. Singh has published over 60 technical papers in the permanent literature in various disciplines of mechanical engineering and applied mechanics. He has edited, authored, or co-authored numerous monographs and books, including the widely used text "Mechanical Design of Heat Exchangers and Pressure Vessel Components", published in 1984. In 1987, he was elected a Fellow of the American Society of Mechanical Engineers. He is a Registered Professional Engineer in Pennsylvania and Michigan, and has been a member of the American Nuclear Society since 1979, and a member of the American Society of Mechanical Engineers since 1974.

Over the decades, Dr. Singh has participated in technology development roles in a number of national organizations, including the Tubular Exchange Manufacturers Association, the Heat Exchange Institute, and the American Society of Mechanical Engineers. Dr. Singh has lectured extensively on nuclear technology issues in the U.S. and abroad, providing continuing education courses to practicing engineers, and served as an Adjunct Professor at the University of Pennsylvania (1986–92).

Dr. Singh serves on several corporate boards including the Nuclear Energy Institute and the Board of Overseers, School of Engineering and Applied Science (University of Pennsylvania), Holtec International, and several other industrial companies.

STAFFIERA, JIM E.

Jim E. Staffiera earned a BS in Mechanical Engineering from Drexel University in 1971 and a Masters in Business from Old Dominion University in 1975. He has been involved with nuclear power plant containment vessel and steel structure design, fabrication, construction, and operation since 1971. Originally employed by Newport News Industrial Corporation (a subsidiary of Newport News Shipbuilding), he assisted with development of commercial nuclear fabrication programs for ASME Code N-type Certificate authorization. This progressed into nuclear component fabrication and construction activities, resulting in his current employment with FirstEnergy Corporation at the Perry Nuclear Power Plant, where he works in the Structural Mechanics Unit and is frequently involved with ASME Code Section XI-related issues.

Jim has been a member of ASME since 1972 and is involved in numerous ASME Boiler and Pressure Vessel Code Committee activities, including holding positions as Chair, Secretary, and Member of various Section XI committees on inservice requirements for operating nuclear power plants. He currently chairs the Working Group on Containment and is also a member of the Section XI Subcommittee, the Subgroup on Water-Cooled Systems, and the Special Working Group on Editing and Review.

Jim is an active member of the ASME Pressure Vessels and Piping Division, having chaired the Codes and Standards (C&S) Technical Committee and been C&S Technical Program Representative for the annual ASME Pressure Vessels and Piping Conference. He has also been a member of the American Society for Quality (ASQ) since 1975.

Jim has been involved in several nuclear industry initiatives, the most recent of which was as a member of the Expert Panel for the EPRI Containment Integrated Leak-Rate Test (ILRT) Interval Extension Project.

STANISZEWSKI, STANLEY (STAN)

Stanley Staniszewski is a senior Mechanical Engineer with the U.S. Department of Transportation, Pipelines and Hazardous Materials Safety Administration. He is a '76 Alumni of the Fenn College of Engineering, from Cleveland State University of Ohio and has completed graduate level course work in Business Administration at Johns Hopkins University and advanced engineering degree work at the University of Virginia. Mr. Staniszewski has been a member of the American Society of Mechanical Engineers, since joining as a student. He currently serves on the ASME Section XII SubCommittee on Transport Tanks, Vice Chairs the Sub Group on General Requirements, and is a member of the ASME Hydrogen Steering Committee, and various taskgroups. Mr. Staniszewski is also a member of the National Board Inspection Code, Main Committee, Subgroups RB, and Nonmandatory Appendices. He has experience in the international standards arena through membership and participation as a governmental technical expert to the United Nations and International Standards Organizations on various Technical Committees, Sub-Committees and Work-Groups on gas cylinders, cryogenic vessels and Hydrogen technologies.

He has 10 years of varied experience in the private sector spanning tool & die, manufacturing, research and product development, design, construction and inspection. Within the federal government he has spent 20 years in the areas of mechanical/electrical/chemical project engineering, management, inspection and enforcement issues that affect hazardous materials/dangerous goods in national and international commerce.

SUDAKOV, ALEXANDER V.

Alexander V. Sudakov was born in 1944 in Rybinsk, Russia. In 1962 after high school graduation he entered Saint Petersburg Polytechnic University, Division of Nuclear Power Stations and graduated in 1968 as a NPP engineer. The same year he started his professional engineering activity in the Central Boiler and Turbine Institute. He held positions from scientific researcher (1968) up to Deputy Director General of the Institute (current position).

Dr. Sudakov obtained a Doctor of Science in 1994 and subsequently held a position as Professor of Saint Petersburg Polytechnic University. Dr Sudakov has authored 10 books and published over 100 papers and manuscripts on thermodynamics, strength analysis and life extension of NPPs components and piping. Dr Sudakov is a Member of a number of Russian scientific committees and nuclear power associations. He was honored with the Russian Federation Government Prize in 1995.

STEVENSON, JOHN D.

Dr. John D. Stevenson is a Senior Consultant for J.D. Stevenson, Consulting Engineer Co. He has extensive experience worldwide in the nuclear power field where he served as a consultant to the IAEA and several non U.S. utilities and consulting firms. He holds a Ph.D. in Civil Engineering from Case Western Reserve University. He has provided structural-mechanical consulting services to the nuclear power industry in the U.S. and abroad for the past 35 years and has been a member of various committees of ASME and B&PVC Section III for the past 35 years. He is currently also a member of several of American Society of Civil Engineers, American Nuclear Society, and American Concrete Institute committees and consultant to government agencies dealing with the structural-mechanical safety of nuclear facilities.

SWAYNE, RICHARD W.

Mr. Swayne has worked as a metallurgist, welding engineer, quality assurance manager, and consultant, in the pressure vessel and piping industry, since 1975. He has experience in design, fabrication, and operation of various power and refinery plant components, including valve design and application, welding and materials engineering, and quality assurance program management for construction and operation. He is an expert and well-known instructor in inservice inspection, inservice testing, and repair/replacement programs in operating power plants. He has assisted many organizations in preparation for new and renewal ASME Certificates of Accreditation and has participated in many ASME National Board Accreditation Surveys. Mr. Swayne has been an active participant since 1977 as a member of ASME and ASTM Codes and Standards Committees. He is a member of the ASME Board on Nuclear Codes and Standards and the ASME Boiler and Pressure Vessel Standards Committee and is the Vice Chair of the ASME Subcommittee on Nuclear Inservice Inspection. Mr. Swayne is also a past member of the Subcommittee on Materials and several working groups under the Subcommittee on Nuclear Power.

He has served as a consultant to utilities, architect/engineers, manufacturers, and material manufacturers and suppliers. He is a Qualified Lead Auditor, and was a Qualified Level II Examiner in several nondestructive examination methods. He has been involved in engineering reviews, material selection and application, and quality assurance auditing.

TABOAS, ANÍBAL L.

Aníbal L. Taboas is an executive generalist who consults on environment, governance, and strategic risk management. His background ranges from nuclear CONOPS, to line management of national laboratories and programs, and conflict resolution. He led various regulatory and legislative initiatives, including changing the disposal limits for transuranic waste. Accomplishments in the federal Senior Executive Service have been recognized by the Vice President's Hammer Award, Secretary of Energy Gold Medal, the University of Chicago Medal for Distinguished Performance, and several Exceptional and Distinguished Service Awards. Taboas has a solid reputation for innovative resolution of regulatory and legislative issues, project management, diversity, and independent peer review. Aníbal actively participates in pro-bono activities, such as the Board of Directors of the Center of Excellence for Hazardous Materials Management, and of the Institute for Regulatory Science, editorial boards, and peer review (e.g., National Science Foundation and International Atomic Energy Agency). Dr. Taboas has a BS in Physics/Theology (Univ. of Dayton), MS in Physics (Indiana State Univ.), MS in Mechanical & Nuclear Engineering (Northwestern Univ.), a PhD *honoris causa* in Environmental Policy (UPAEP), and numerous peer-reviewed publications. Aníbal is Fellow of the American Society of Mechanical Engineers, edited The Decommissioning Handbook, and has served multiple times as Chair of the International Conference on Environmental Management. Aníbal L. Taboas can be reached via electronic mail at: TaboasA2@ASME.org.

TRAMPUS, PETER

Peter Trampus earned his MSc in 1972 in Mechanical Engineering from the Technical University of Budapest, Hungary. He obtained his second degree as Specialized Engineer on Plastic Deformation from the same university in 1979. He earned his PhD in Materials Science in 1985 from the Technical University of Dresden, former East Germany. After graduating, Peter Trampus joined the Csepel Iron and Steel Works, where he worked as research engineer and, then, head of laboratory. In 1982, Peter Trampus moved to the Paks NPP, where he was in charge of the Material Testing and Evaluation Section (1982 to 92) being responsible for in-service inspection and RPV surveillance, basically all structural integrity related issues of the pressurized components, and later (1992 to 95) he was the Head of the Advisory Board to the General Director.

From 1996 to 2003, he worked for the International Atomic Energy Agency (IAEA), a member of the United Nations family, Vienna. He headed projects on managerial and engineering aspects of nuclear power program development, implementation and performance improvement. From 2003 to 2004 he was a visiting scientist at the Institute for Energy, Petten, The Netherlands, one of the seven institutes of the European Commission's Joint Research Center. Here, he was dealing with nuclear safety issues in Central and Eastern European countries. In 2003, Peter Trampus established his own consultant company and now works as principal consultant. Currently, the major focus of his activity is on nuclear power plant life management.

Peter Trampus is a Founding Member of the Hungarian Academy of Engineering (1990), recipient of the IAEA's Distinguished Service Award (2002), the Hungarian delegate of IIW Commission V "Quality control and quality assurance of welded products" (since 2006), President of the Hungarian Association for Nondestructive Testing (since 2005). He is the author of more than 100 papers in technical journals and conference proceedings, several of which are in English.

UPITIS, ELMAR

Elmar Upitis received a B.S. degree in Civil Engineering from University of Illinois in 1955 and did postgraduate studies at the Illinois Institute of Technology. He served in the US Army and was employed by Chicago Bridge & Iron Company from 1955 to 1995 in various capacities, including Chief Design Engineer, Manager of Metals Engineering, and Senior Principal Engineer–Materials. He was also responsible for oversight of CBI engineering in South America, Europe and Africa and Middle East. Mr. Upitis provides engineering consulting services in the areas of codes and standards (ASME, API, ASTM, etc.), design of plate structures, fitness-for-service evaluation, and materials related issues. He is a licensed professional and structural engineer in the State of Illinois, ASME Fellow and a member of various technical committees in the ASME B&P Vessel Code, ASTM Fellow and a member of several ASTM technical committees, former Chair of Pressure Vessel Research Council (PVRC) and an active participant in the PVRC, and a member of AWS and WRC. He is involved in the development of the new B&PV Code to replace the present Section VIII, Division 2 and several other projects related to the ASME B & PV Code.

Mr. Upitis is a co-author of WRC Bulletin 435 on design margins in ASME Section VIII, Divisions 1 and 2, WRC Bulletin 447 on evaluation of operating margins for in-service pressure equipment, WRC Bulletin 453 on minimum weld spacing requirements for API Standard 653, PVRC report on the European Pressure Equipment Directive, and several other published papers on Cr-Mo steel pressure vessels.

VAN DEN BREKEL, NICHOLAS C.

Nicholas C. van den Brekel is a recognized authority on Periodic Inspection for CANDU Nuclear Power Plants (the CANDU equivalent to ASME XI In-Service Inspection requirements). Over the last 16 years, Nick has been a major contributor to the Canadian Standards Association (CSA) N285B Technical Committee on Periodic Inspeciton of CANDU NPPs. Nick has served as an Executive Member and Secretary of this committee for the last 5 years (1999 onwards).

Nick has 23 years of experience in the Inspection and Maintenance of CANDU Nuclear Power Plants. Much of Nick's experience has been in dealing with the unique inspection and fitness for service evaluation challenges posed by the zirconium alloy fuel channels and other reactor internals at the heart of the CANDU reactor design. Many of these components are subjected to irradiation damage, damage that can result in physical changes to the material and material properties, conditions that must be monitored in accordance with Canadian nuclear standards. Nick has been involved in development of new non-destructive evaluation techniques to assess the material condition of zirconium alloy pressure tubes. Nick's experience extends to inspection and maintenance of all CANDU reactor components, including nuclear fuel, fuel channel feeder pipes, steam generator tubes, nuclear piping and vessels, as well as conventional side heat exchangers and steam turbines.

Nick is currently employed as the Technical Advisor to Inspection Services Division of Ontario Power Generation, which provides specialized inspection services to all CANDU reactor units, including 16 operating units in Canada. Nick's experience includes consultation to the international CANDU community on inspection and maintenance related issues.

VAZE, K.K.

K.K. Vaze graduated from Indian Institute of Technology, Bombay (IITB) with a B. Tech. in Mechanical Engineering in 1973. After completion of the 17th Batch of Training School of Bhabha Atomic Research Centre in 1974, he joined the Nuclear Systems Division of Indira Gandhi Centre for Atomic Research, Kalpakkam. He worked in the area of Structural Analysis and Design of Fast Reactor Components using Finite Element Method and ASME Boiler & Pressure Vessel Code, Section III, Nuclear Vessels.

In 1989, he joined the Reactor Safety Division of Bhabha Atomic Research Centre, Mumbai. Mr. Vaze was involved in the Structural analysis and design of Pressurized Heavy Water Reactor (PHWR) Components. The scope of work included Fatigue and Fracture Analysis, Experimental Stress Analysis, Fracture Mechanics, Seismic analysis, Fitness-for-Purpose Evaluation, Residual Life Estimation and Life Extension. He piloted a project on "Leak before Break evaluation of Primary Heat Transport piping of PHWR".

In addition to design and analysis, Mr. Vaze has expertise in Ageing Management, Equipment Qualification and Seismic Revaluation. He is a member of many committees formed by Atomic Energy Regulatory Board to look into the safety aspects of operating reactors as well as those under various stages of design/construction. He has 24 publications in Journals and over 60 papers in International Conferences. His current position is Head, Reactor Structures Section, in Reactor Safety Division of Bhabha Atomic Research Centre, Mumbai. Mr. Vaze resides in Mumbai with his wife, Ashlesha and two daughters, Anuja and Manasi.

VIROLAINEN, REINO

Mr. Virolainen graduated from Helsinki University of Technology (Material Engineering) in 1972. In 1973–1982, he worked at VTT, Electrical engineering laboratory, as a research scientist. Since 1982 he has been working for STUK (Radiation and Nuclear Safety Authority) as inspector, section leader and head of office of risk assessment. His main topics at VTT and STUK have been method development for level 1 PRA, CCFs, reviews of PRA applications for the Finnish NPPs and development of Risk Informed Regulation procedures including Risk-Informed Inservice Inspection (RI-ISI). Mr. Virolainen has been a long term member of Working Group RISK at OECD/NEA/CSNI, Vice Chairman in 1991–1992 and Chairman 1992–1996. He is a member of IAPSAM Board since 2006.

Mr. Virolainen is a special lecturer on systems reliability and risk assessment at Lappeenranta University of Technology. He has several technical publications in U.S., European and International Journals covering PRA, Risk-Informed Regulation and Nuclear Engineering and Design.

VOORHEES, STEPHEN V.

Employed in the Authorized Inspection Agency sector since 1976 with Factory Mutual, Commercial Union Insurance Company, Hartford Steam Boiler I and I, and OneBeacon America Insurance Company. Duties have included inspection of all types of boilers, pressure vessels, heat exchangers, nuclear components as well as supervision of these activities and finally management of same.

Currently serves on Section IV Heating Boilers as Vice Chair, Section XII, Transport Tanks as a member and Chair of Sub-Group Fabrication and Inspection, and serve as member of the Standards Committee.

From 1970 to 1974 served in the US Navy in the Western Pacific on destroyers as a boiler technician.

Married to Louise for 25 years with two sons. Reside in Allentown, PA. Hobbies include hunting, shooting and golf.

WEST, RAYMOND (RAY) A.

Mr. West began his nuclear career in the US Navy in 1971 and then proceeded into its Nuclear Power Program in 1976. He has been a welder, a Level III in several Nondestructive Examination (NDE) methods, and has developed Inservice Inspection (ISI) programs for both Pressured Water Reactors and Boiling Water Reactors. In 1979 he began work in the commercial nuclear industry and he continues to do so today.

His major accomplishments include ASME Engineer of the Year in 1997 in the State of Connecticut for Northeast Utilities, and several other ASME awards including one for the development of the Section XI, Nonmandatory Appendix R, "Risk-Informed Inspection Requirements For Piping," where he served as the ASME Technical Project Manager Responsible for Development and Approval of this Appendix (a 14 year effort), Approved for Publication in the 2005 Addenda of Section XI, October 2004. He has also authored or co-authored many technical papers centering on ISI and Risk-Informed Inservice Inspection (RI-ISI) and the latest was for the ASME 16th International Conference on Nuclear Engineering (ICONE16) in May 2008 that was related to the U.S. Nuclear Regulatory Commission's Rulemaking Process and its effects on the Endorsement of ASME Nuclear Codes and Standards in the USA.

Ray is currently the Vice Chair of the ASME Board on Nuclear Codes and Standards (BNCS), the Co-Chair of the BNCS Task Group on Regulatory Endorsement (TG-RE), a member of the ASME Boiler and Pressure Vessel (BPV) Code Subcommittee XI on Nuclear Inservice Inspection ISI, a member of the Section XI Executive Committee, and several of its lower level BPV Code writing groups. He has been involved with nuclear power for over 30 years. His experience has been focused on welding, NDE, and ISI and he is currently a Technical Consultant and the senior ASME representative for his company Dominion Resources, Inc. at the Millstone Power Station in Waterford, Connecticut.

WHITE, GLENN A.

Glenn White is a principal engineer and principal officer at Dominion Engineering, Inc. in Reston, Virginia. Mr. White manages consulting and analysis projects primarily for the nuclear power industry and often related to aging degradation of materials, boric acid corrosion, or thermal performance.

Mr. White was the principal author of the Electric Power Research Institute (EPRI) safety assessment report for primary water stress corrosion cracking (PWSCC) of U.S. PWR Alloy 600 reactor vessel closure head penetrations. In 2007, he was the principal investigator for EPRI's crack growth and leak-before-break evaluation of PWSCC of PWR pressurizer nozzle dissimilar metal welds in response to indications of circumferentially oriented PWSCC at one plant. Mr. White's projects to evaluate materials degradation include nuclear safety and economic risk assessments and apply analytical tools such as probabilistic Monte Carlo simulation, net present value analysis, Weibull statistical modeling, and stress and fracture analyses.

In the area of thermal performance degradation of nuclear steam generators, Mr. White investigates the sources of steam pressure loss, the fouling deposition process, and the effects of tube deposits on boiling heat transfer and corrosion.

Before joining Dominion Engineering, Inc. in 1993, Mr. White received BS (summa cum laude) and MS degrees in mechanical engineering from the University of Maryland at College Park. Mr. White is a registered professional engineer and is a member of NACE.

WILLIAMS, TONY

Tony Williams is head of the nuclear fuel department of the Nordostschweizerische Kraftwerke AG (NOK), the company responsible for the general management and fueling of the two Beznau PWR units and the Leibstadt BWR in Switzerland. Both plants are renowned for their progressive fuel burnup strategies as well as extensive use of MOX and Reprocessed Uranium fuels.

In addition to fuel procurement, his responsibilities include in-house fuel assembly and core design, administration of reprocessing contracts, planning of interim off-site storage, flask procurement and transport as well as some aspects of final disposal. He is a member of the Swiss nuclear fuel commission and a board member of ZWILAG, the facility responsible for interim dry fuel storage and waste conditioning in Switzerland. In previous positions he was manager of a research program investigating fuel and core issues related to Pebble Bed Modular Reactors as well as working as a reactor physicist for the British commercial nuclear industry.

Dr. Williams holds a diploma in Business Management, has an honors degree in Physics from Durham University (1981) as well as an M.Sc. and Doctors degree in neutron physics from Birmingham University (1984).

WOODWORTH, JOHN I.

John I. Woodworth has BSME from Univ. of Buffalo, 1948. He is engaged in consulting on Steam and Hot Water (hydronic) heating systems and Codes and Standards. He provides information for legal proceedings of hydronic heating systems and equipment. He was previously with Fedders Corp. (1948–1959), as Technical Director of Hydronics Institute (predecessor Institute of Boiler and Radiator Manufacturers.), 1959–1990. Woodworth's professional activities 1990 to date are supported by Hydronics Institute Division, GAMA.

He is a member of ASME, and a member of several ASME Code Committees such as Section IV, (1967–date), Cast-Iron Subgroup; Chair, ASME Section VI; Vice-Chair Controls and Safety Devices for Automatically-Fired Boilers Standards Committee (1973–2000). He was a consultant with the National Institute of Science and Technology (formerly the National Bureau of Standards). Woodworth is a Life Member of ASHRAE, Member of several of its Technical Committees, Secretary, Vice Chair and Chair of SPC. He has written numerous technical articles for trade magazines.

John received ASME Distinguished Service Award (1991), Dedicated Service Award (2000) and ASHRAE Standards Achievement Award (1996). He was a Member, National Fuel Gas Code Committee, VP, Uniform Boiler and Pressure Vessel Laws Society and Liaison to Building Energy Codes & Standards Committee. He was a Member of technical advisory committees for Brook-haven National Laboratories.

YODER, LLOYD W.

Mr. Yoder is a Mechanical Engineering graduate (BSME) of the University of Pittsburgh (1952). He joined ASME as a student member and continued membership until now as an honorary life member. Upon graduation from college, he joined Babcock and Wilcox Company as a graduate student that provided intense training in the operations of all divisions of the Company. Upon graduation from this program, he joined the Company's research center as a test and research engineer. During six years at the research center, he worked on both fossil and nuclear projects and was awarded several patents for fossil boiler inventions.

Mr. Yoder later transferred to the Company's main office, initially as a functional performance contact engineer and later as a design engineer responsible for developing utility boiler Company standards. It was during this time in 1971, that Mr. Yoder became a committee member of the Subcommittee on Power Boilers of the ASME Boiler and Pressure Vessel Code. He continued this membership and is now an honorary member of that Subcommittee. Mr. Yoder later became an engineering manager in Babcock & Wilcox's marketing department and with the Company's international business growing he subsequently joined the International Division as Technical Operations Manager of Licensee and Joint Venture Companies. After retirement in 1996, he became a consultant for several engineering companies on various problems and served as an expert witness in a number of litigations. For fourteen consecutive years, he and the late Martin D. Bernstein taught a continuing education course for the ASME on Section I of the ASME Boiler and Pressure Code. In 1999 the ASME published a book, *Power Boilers: A Guide to Section I of the ASME Boiler and Pressure Vessel Code*, which was co-authored by Lloyd W. Yoder and the late Martin D. Bernstein.

CONTENTS

PART 3: SECTION III — RULES FOR CONTRUCTION OF NUCLEAR POWER PLANT COMPONENTS

PREFACE TO THE FIRST EDITION

This book provides "The Criteria and Commentary on Select Aspects of ASME Boiler and Pressure Vessel and Piping Codes" in two volumes. The intent of this book is to serve as a "Primer" to help the user weave through varied aspects of the ASME Codes and B31.1 and B31.3 Piping Codes and present a summary of specific aspects of interest to users. In essence, this Primer will enable users to understand the basic rationale of the Codes as deliberated and disseminated by the ASME Code Committees. This book is different from the Code Cases or Interpretations of the Code, issued periodically by these ASME Code Committees, although these are referred in the book. It is meant for a varied spectrum of users of Boiler and Pressure Vessel (B&PV) and B31.1 and B31.3 Piping Codes in United States and elsewhere in the world. This book should be considered as a comprehensive guide for ASME B&PV Code Sections I through XI, B31.1 and B31.3 Piping Codes. The contents of these two volumes can be considered as a companion book—a criteria document—for the latest editions of the Code, written by thirty-six professionals with expertise in its preparation and use.

ASME and the industry volunteers have invested immense resources in developing Codes and Standards for the Power and Petrochemical Industry, including nuclear, non-nuclear, fossil, and related. The industry has been relying on these documents, collectively referred to as the ASME Code, on a day-today basis, and regulators consult them for enforcing the rules. Research and development, in both the material science and analytical areas, find their results in the revisions and updates of the Codes. Over a period of time, these B&PV and Piping Codes, encompassing several disciplines and topics, have become voluminous Standards that belie the intent and expectations of the authors of the Codes. In a word, the B&PV Codes can become a "labyrinth" for an occasional user not conversant with the information contained in the Code. Thus, given the wealth of information contained in the Code, these cannot be easily discerned. For example, the B&PV Code, even though it is literally an encyclopedia of rules and standards to be followed by engineers in the nuclear or fossil or related industries, is not easy to comprehend and conform to. Alphanumeric text and graphics are loaded with information, arrived at by a consensus process from the deliberations of practicing engineers, professionals, academia, and regulators meeting several times a year. A lack of understanding of the Code, therefore, can cause not only professional errors but also misplaced confidence and reliance on the engineer's interpretation that could lead to serious public safety hazards. Spread over several volumes and thousands of pages of text, tables, and graphics, it is not easy to decipher the criteria and the basis of these Codes.

Thus, given the importance of these ASME Codes related to the industry and the attendant technological advances, it becomes a professional expediency to assimilate and appropriately apply the wealth of information contained in the Codes. The first step, then, is to ask, "Where is what?" The Code is spread over eleven Sections; attending the tutorials is one way to understand first-hand the various Sections of the Code. However, this is not within the reach of all of the engineers in the industry. The next best solution is to have expert authors, versatile in the individual Sections and Subsections, to make the subject matter understandable to the practicing engineers in a book format such as "A PRIMER."

In this book, all of the Sections I through XI of the B&PV and B31.1 and B31.3 Piping Codes are summarily addressed with examples, explanatory text, tables, graphics, references, and annotated bibliographical notes. This permits engineers to more easily refer to the material requirements and the acceptance criteria whether they are in the design basis or in an operability situation of a nuclear plant or process piping. In addition, certain special topics of interest to engineers are explicitly addressed. These include Rules for Accreditation and Certification; Perspective on Cyclic, Impact, and Dynamic Loads; Functionality and Operability Criteria; Fluids; Pipe Vibration; Stress Intensification Factors, Stress Indices, and Flexibility Factors; Code Design and Evaluation for Cyclic Loading; and Bolted-Flange Joints and Connections. Important is the inclusion of unique Sections such as Sections I, II, IV through VII, IX, and X that enriches the value of the book as a comprehensive companion guide for B&PV and Piping Codes. Of considerable value is the inclusion of an in-depth treatment of Sections III, VIII, and XI. A unique aspect of the book chapters related to the Codes is the treatment of the origins and the historical background unraveling the original intent of the writers of the Criteria of the Codes and Standards. Thus, the current users of these Codes and Standards can apply their engineering knowledge and judgment intelligently in their use of these Codes and Standards.

Although these two volumes cannot be considered to be a perfect symphony, the subject matter orchestrates around a central theme, that is, "The Use of B&PV and Piping Codes and Standards." Special effort is made by the contributors, who are experts in their respective fields, to cross-reference other Sections; this facilitates identifying the interconnection between various B&PV Code Sections, as well as the B31.1 and B31.3 Piping

Codes. The Table of Contents, indexing, and annotated notes for individual Chapters are provided to identify the connection between varied topics. It is worth mentioning that despite the chapters not being of equal length, comprehensive coverage is ensured. The coverage of some sections is intentionally increased to provide in-depth discussion, with examples to elucidate the points citing the Code Subsections and Articles.

K. R. Rao, Ph.D., P. E.
Editor

Robert E. Nickell, Ph.D.
1999–2000 President
ASME International

Preface to the Second Edition

This edition continues to address the purpose of the first edition to serve as a "Primer" to help the user weave through varied aspects of the ASME Codes and B31.1 and B31.3 Piping Codes and present a summary of specific aspects of interest to users. In providing the "end user" all of these aspects, the first edition has been revised appropriately to be consistent with the current 2004 Codes.

Contributors of the first and second volumes had taken immense pains to carefully update their write-ups to include as much of the details that they could provide. Significant changes can be seen in Sections II, III, VIII and XI with repercussions on Sections I, IV, V, VII, IX and X. Thus, these consequences had been picked up by the contributors to bring their write-up up-to-date. Similarly changes of Power Piping (B31.1 Code) and B31.3 (Process Piping) have also been updated.

Included in this edition is a third volume that addresses the critical issues faced by the BWR and PWR Nuclear facilities such as BWR Internals, PWR Reactor Integrity, and Alloy 600 related issues. With the aging of the Nuclear Plants, the regulators perspective can be meaningful, and this has been addressed by experts in this area. In today's industrial spectrum the role of Probabilistic Risk Analysis has taken an important role and this volume has a chapter contributed by recognized authorities. With the increased use of computer–related analytical tools and with ASME Codes explicitly addressing them, a chapter has been devoted to the Applications of Elastic Plastic Fracture Mechanics in ASME Section XI Code.

ASME Codes are literally used around the world. More importantly the European Community, Canada, Japan and UK have been increasingly sensitive to the relevance of ASME Codes. In this second edition, experts conversant with these country Codes had been invited to detail the specifics of their Codes and cross-reference these to the ASME Codes.

Public Safety, more so than ever before, has become extremely relevant in today's power generation. Experts hade been invited to provide a perspective of the regulations as they emerged as well as discuss the salient points of their current use. These include the transportation of radioactive materials and the new ASME Section XII Code, Pipe Line Integrity and pertinent topics involved in decommissioning of nuclear facilities.

K. R. Rao, Ph.D., P.E.
Editor

Preface to the Third Edition

This edition continues to address the purpose of the previous editions to serve as a "Primer" to help the user weave through varied aspects of the ASME Codes and B31.1 and B31.3 Piping Codes, in addition to a discussion of "The Criteria and Commentary on Select Aspects of ASME Boiler and Pressure Vessel and Piping Codes" of interest to "end users". This publication has been revised in providing all of the aspects of the previous editions, while updating to the current 2007 Codes, unless otherwise mentioned. This book in three volumes strives to be a comprehensive 'Companion Guide to the ASME Boiler and Pressure Vessel Code'.

Since the first edition, a total of 140 authors have contributed to this publication, and in this edition there are 107 contributors of which 51 are new authors. Several of the new contributors are from countries around the world that use ASME B&PV Codes, with knowledge of ASME Codes, in addition to expertise of their own countries' B&PV Codes. All of these authors who contributed to this third edition considerably updated, revised or added to the content matter covered in the second edition to address the current and futuristic trend as well as dramatic changes in the industry.

The first two volumes covering Code Sections I through XI address organizational changes of B&PV Code Committees and Special topics relating to the application of the Code. Considering significant organizational changes are taking place in ASME that reflect the industry's demands both in USA and internationally, the salient points of these have been captured in this publication by experts who have first hand information about these.

Volume 1 covers ASME Code Sections I through VII, B31.1 and B31.3 Piping Codes. Continuing authors have considerably updated the text, tables, and figures of the previous edition to be in line with the 2007 Code, bringing the insight knowledge of these experts in updating this Volume. Fresh look has been provided by new authors, who in replacing previous contributors of few chapters, have provided an added perspectives rendered in the earlier editions. In one case, the chapter had been entirely rewritten by new experts, with a new title but addressing the same subject matter while updating the information to the 2007 ASME Code Edition.

ASME Code Committees have spent time and considerable resources to update Section VIII Division 2 that was completely rewritten in the 2007 Code Edition, and this effort has been captured in Volume 2 by several experts conversant with this effort. Volume 2 has chapters addressing Code Sections VIII through XI, refurbished with additional code material consistent with the current 2007 Code edition. Notable updates included in this Volume relate to maintenance rule; accreditation and certification; perspectives on cyclic, impact and dynamic loads; functionality and operability criteria; fluids; pipe vibration testing and analysis; stress intensification factors, stress indices and flexibility factors; Code design and evaluation for cyclic loading; and bolted-flange joints, connections, code design and evaluation for cyclic loading for Code Sections III, VIII and a new chapter that discusses Safety of Personnel using Quick-actuating Closures on Pressure Vessels and associated litigation issues. While few chapters have been addressed by new authors who added fresh perspective, the efforts of continuing authors have provided their insights with additional equations, figures and tables in addition to extensive textual matter.

The third volume of this edition is considerably enlarged to expand the items addressing changing priorities of Codes and Standards. Continuing authors who addressed these topics in the previous edition have discussed these with respect to the ASME 2007 Code Edition. The discussions include chapters on BWR and PWR Reactor Internals; License Renewal and Aging Management; Alloy 600 Issues; PRA and Risk-Informed Analysis; Elastic-Plastic Fracture Mechanics; and ASME Code Rules of Section XII Transport Tank Code. Chapters covering 'U.S. Transportation Regulations for Radioactive Materials'; 'Pipeline Integrity and Security', and 'Decommissioning of Nuclear Facilities' have been considerably revised.

In Volume 3 experts around the world capture 'Issues Critical for the Next Generation of Nuclear Facilities' such as Management of Spent Nuclear Fuel, Generation III1 PWRs, New Generation of BWRs and VERY High Temperature Generation IV Reactors.

The impact of globalization and inter-dependency of ASME B&PV Codes had been examined in the previous edition in European Community, Canada, France, Japan and United Kingdom. Contributors who authored these country chapters revisited their write-up and updated to capture the current scenario.

Significant contribution in the third volume is the inclusion of additional countries with changing priorities of their Nuclear Facilities. In-depth discussions cover the international experts of these countries which own and operate nuclear reactors or have nuclear steam supply vendors and fabricators that use ASME B&PV Code Sections I through XII. This information is meant to benefit international users of ASME Codes in Finland, Belgium, Germany, Spain, Czech and Slovakia, Russia, South Africa, India, Korea and Taiwan that have been added in this third edition.

A unique feature of this publication is once again, as in the previous editions, the inclusion of all author biographies and an introduction that synthesizes every chapter, along with an alphabetical listing of indexed terms

K. R. Rao, Ph.D., P.E.
Editor

INTRODUCTION

This third edition is in three volumes composed of 19 Parts, with Parts 1–5 in Volume 1, Parts 6–11 in Volume 2 and Parts 12–19 in Volume 3. Common to all three volumes is the front matter, including the Organization of the Code. Organization and Operation of the ASME Boiler and Pressure Vessel (B&PV) Committee has been initially authored by Martin D. Bernstein for the first edition but considerably updated in the previous second edition by Guido G. Karcher. However, the current dramatic changes in the ASME B&PV organization these have been captured by Guido Karcher in this current third edition. Included are detailed discussions pertaining to the "Research Projects for the Maintenance and Development of Codes and Standards" and "Realignment Activities of the ASME B&PV Code Committee Structures". An index is provided at the end of each volume as a quick reference to topics occurring in different Code Sections of that volume. In addition to indexing several topics covered in this publication, it is also meant to assist in reviewing the *interconnection* of the ASME Boilers & Pressure Vessel Code Sections/Subsections/Paragraphs occurring in the particular volume. In each chapter, all discussions generally pertain to the latest 2007 Code Edition unless noted otherwise by the chapter author(s). The ASME Code is generally accepted in the United States (and in many foreign countries) as the recognized minimum safety standard for the construction of pressure vessels and piping. Toward that end, the first two volumes can be considered "a primer." Although this primer is authored by several Code Committee members who are considered experts in their respective fields, the comments and interpretations of the rules contained in this book are strictly the opinions of the individual authors; they are not to be considered official ASME Code Committee positions.

Since the first edition, a total of 140 authors contributed to this publication and in this edition there are 107 contributors of which 51 are new authors. Several of the new contributors are from countries around the world that use ASME Boiler & Pressure Codes, with knowledge of ASME Codes in addition to expertise of their own country Boiler & Pressure Vessel Codes. All of these authors who contributed to this third edition considerably updated, revised or added to the content matter covered in the second edition.

Volume 1 has five Parts, each addressing a unique aspect of the Code. Part 1 covers *Power Boilers* (Code Sections I and VII); Part 2 covers *Materials and Specifications* (Code Section II); Part 3 provides an in-depth commentary on *Rules for Construction of Nuclear Power Plant Components* (Code Section III, Divisions 1, 2, and 3); Part 4 covers *Power Piping* (B31.1 Code) and *Process Piping* (B31.3 Code); and Part 5 covers *Heating Boilers* (Code Sections IV and VI).

Volume 2 covers Parts 6–11, with Part 6 covering *Nondestructive Examination* (NDE) (Code Section V); Part 7 providing in-depth criteria and commentary of Code Section VIII, including Divisions 1, 2, and 3; Part 8 covering welding and brazing qualifications of Code Section IX; Part 9 covering Code Section X and pertaining to fiber-reinforced plastic pressure vessels; Part 10 providing in-depth discussions of Code Section XI; and Part 11 covering special topics of interest to ASME Boiler and Pressure Vessel (B&PV) Code Users and Practicing Engineers.

The scope of Volume 3 that contains Parts 12 to 19 has considerably expanded from the previous edition. This volume has in addition to aspects with critical bearing on ASME Boiler & Pressure Vessels addresses countries that have become increasingly important with ASME Codes being applicable to them. Part 12 addresses Current BWR Reactor Internals & Other BWR Issues in chapters that cover License Renewal and Aging Management (NRC), PWR Reactor Vessel Integrity, PWR Reactor Vessel Alloy 600 Related Issues, PRA & Risk Based Analysis, and Applications of Elastic Plastic Fracture Mechanics in ASME Section XI Code Applications. In Part 13 International Codes & Standards Related to ASME B&PV Code are addressed, which include Pressure Equipment Directive used by the European Community, Canadian B&PV Codes & Standards, French Pressure Equipment Codes, Recent Development of Boiler and Pressure Vessel Codes in Japan, and British Codes & Standards. With the recent trends to pay particular attention to Public Safety, more so than previously, Part 14 focuses on Other Ongoing Issues of Public Safety with chapters covering 40-Year Retrospective on the Transportation Regulations for Radioactive Materials, Description of Rules of Section XII Transport Tank Code, Pipe Line Integrity & Security, and Decommissioning of Nuclear Facilities. In Part 15 issues critical for the next generation of nuclear facilities is addressed. In this Part, topics deal with Management of Spent Nuclear Fuel, Generation III+ PWRs, New Generation of BWRs and very High Temperature Generation IV Reactors. Global Pressure Vessel and Piping Issues of several countries of Western and Eastern Europe, Africa and Asia are dealt with in Parts 16 to 19. In each of these Chapters authors with expertise in their Country Codes and conversant with ASME Pressure Vessel Codes provided the write-up. In Part 16 PV&P Codes of West European Countries covered are Finland, Belgium, Germany and Spain. In Part 17 the PV&P Codes of East European Countries included are Czech and Slovakian Codes, Hungary and Russia. Codes and standards used in the nuclear industry in the Republic of South Africa are covered in Part 18. Pressure & Vessel Issues of Asian countries such as India, Korea and Taiwan are included in Part 19.

VOLUME 1

Chapter 1 of the 1st edition was authored by the late Martin D. Bernstein. It discussed *Power Boilers*, Section I of the ASME Code. His objective was to provide an overview of the intent, application and enforcement of Section I rules for the construction of power boilers. This chapter is an abbreviated version of the book *Power boilers, A Guide to Section I of the ASME Boiler and Pressure Vessel Code*, used as the textbook for a two day ASME professional developement course on Section I developed and taught for many years by Martin D. Bernstein and Lloyd W. Yoder. Mr. Yoder has reviewed and updated the 1st edition Chapter 1 for this 2nd edition to commemorate his close friend and associate. In doing so, he found that only minor changes and updating were required because the 1st edition Chapter 1 was so well crafted by Mr. Bernstein, like all of the many things he was known to have written.

Chapter 1 was reviewed and updated by John R. Mackay, long-time member and past chairman of Subcommittee I. The current edition of this Chapter covers revisions to Section I, Power Boilers through the 2007 Edition. Significant additions are included in this update that pertain to Code changes regarding Cold Forming of Austenitic Materials, Hydrostatic Test, increased allowable stresses of many materials, and a new Part PHRSG, Requirements for Heat Recovery Steam Generators.

Chapter 1 covers some of the more important aspects of Section I construction, including the history and philosophy of Section I: how the ASME Code works; the organization and scope of Section I; the distinction between boiler proper piping and boiler external piping; how and where Section I is enforced; and the fundamentals of Section I construction. These fundamentals include permitted materials; design; fabrication; welding and postweld heat treatment; NDE; hydrostatic testing; third-party inspection; and certification by stamping and the use of data reports. A number of design examples also have been included in this chapter.

The design and construction of power boiler involves the use of other sections of the ASME Code besides Section I, such as Section II, *Materials*; Section V, *Nondestructive Examination; and* Section IX, *Welding and Brazing Qualifications*. In a rather unusual arrangement, the construction rules for boiler piping are found partly in Section I and partly in the B31.1 Power Piping Code. This arrangement has led to considerable misunderstanding and confusion, as explained in Chapter 1, Section 1.5, where the distinction between boiler proper piping and boiler external piping is discussed.

In the 1st edition Mr. Bernstein stated *"The ASME B&PV Code changes very slowly but continuously. Thus, although this chapter provides a substantial body of information and explanation of the rules as they now exist, it can never provide the last word. Nevertheless, the chapter should provide the User with a very useful introduction and guide to Section I and its application.*" His words are still true for the reason that Chapter 1, as updated, retains the philosophy and intent of the original author, Martin D. Bernstein.

Chapter 2, authored by Edmund W. K. Chang and Geoffrey M. Halley, covers ASME Boiler & Pressure Vessel Code Section VII, *Recommended Guidelines for the Care of Power Boilers*. This Section is very useful for operators of power boilers, as stated in the Preamble of Section VII, "The purpose of these recommended guidelines is to promote safety in the use of power boilers. These guidelines are intended for use by those directly responsible for operating, maintaining, and inspecting power boilers." In line with the other Code Sections, the nine Subsections, C1–C9, are addressed by the authors, including Fundamentals such as Boilers Types, Combustion, and Boiler Efficiency; Boiler Operation; Boiler Auxiliaries; Appurtenances; Instrumentation, Controls, and Interlocks; Inspection; Repairs, Alterations, and Maintenance; Control of Internal Chemical Conditions; Preventing Boiler Failures; and Guidelines for Safe and Reliable Operation of the Power Boilers.

The authors present the commentary in Chapter 2 from the perspective of Owner–Operator personnel with experience in operating, maintaining, and inspecting industrial and utility power boilers. In some instances, although certain paragraphs are reiterations of Section VII, they, combined with additional information, stress the importance of the aspects covered. It is suggested that the reader review existing literature, such as manufacturer's instructions or existing company procedures, for additional details. Section VII is a Nonmandatory Standard, and it, along with Section VI (Chapter 19 of Volume 1) provides recommended practices and serves as a guideline. However, Section VII touches on many activities that the Owner–Operator personnel must be aware of before a power boiler is commissioned. New personnel who are not familiar with boiler operation, maintenance, and inspection can use Section VII as an introduction to these activities. Experienced personnel will find Chapter 2 to be a good review of the essentials of operation, maintenance, and inspection, with useful figures and references. In the "Summary of Changes" of the 2007 Edition, it was written that "No revisions are contained in Section VII of this Edition." However, there were a few minor corrections made that were not listed. It is the authors' opinion that more effort should be made by the committee to update and expand Section VII as recommended in the Chapter 2 commentary. Like the other Sections, Section VII should also be a living document providing the latest information in this ever-changing world. If the committee does not put in any effort in revising Section VII, they should at least alert prospective purchasers that no changes were made so that holders of the previous Edition do not have to buy it.

Chapter 3 has multiple authors, and in Chapter 3.1, *History of Materials in the ASME Boiler and Pressure Vessel Code*, Domenic Canonico traces the chronological evolution of materials and associated technologies, from the need for materials to accommodate riveted construction to the acceptance of fusion welding as a fabrication process. Included in this discussion are the application of advanced materials, the revisions to the basis for setting allowable stress values, and the acceptance of Material Specifications other than those approved by ASTM. Also covered is the evolution of materials, from their humble beginning as a 35-page inclusion in the 1914 Edition of the Boiler Code to the 3994-page, four-Part 2001 Edition of Section II of the ASME B&PV Code. Chapter 3.1 provides some insight not only into the materials needed for the design and fabrication of power boilers but also into the determination of the Maximum Allowable Working Pressure. With the aid of tables, Domenic discusses the Material Specifications from the 1914 through the present Code Editions.

Chapter 3.2, authored by Richard Moen and Elmar Upitis, discusses Code Section II, Part A—*Ferrous Material Specifications*, adopted by ASME for the construction of boiler, pressure vessel, and nuclear power plant components. They note that all materials accepted by the various Code Sections and used for construction within the scope of the Code Sections' rules must be furnished in accordance with the Material Specifications contained in Section II, Parts A, B, or C, or referenced in Appendix A of Part A—except where otherwise provided in the ASME Code Cases or in the applicable Code Section. Discussions in Chapter 3.2 include The Organization of Section II, Part A, Guideline on the Approval of New Materials, Appendices, and Interpretations.

In Chapter 3.3, Dennis W. Rahoi provides the basis of and commentary on Section II, Part B—*Nonferrous Material Specifications*, adopted by ASME for the construction of boiler, pressure vessel, and nuclear power plant components. He notes that all materials allowed by the various Code Sections and used for construction within the scope of the Code Sections' rules must be furnished in accordance with the Material Specifications contained in Section II, Part B or referenced in Appendix A of Part B-except where otherwise provided in the ASME Code Cases or in the applicable Code Section. Dennis discusses alloy definitions; the organization of Section II, Part B Appendices; the acceptable ASTM Editions; Nonmandatory guidelines; the guideline on the Approval of New Materials; the allowable stresses for alloys; and the basis for material acceptance for Code Construction. Dennis also provides cross-references to weldability; ASME Code Sections I, III, IV, VIII, and IX; and Piping Codes B31.1 and B31.3.

Chapter 3.4, authored by Marvin Carpenter, discusses Section II, Part C—*Specification for Welding Rods, Electrodes, and Filler Metals*. Welding plays a major role in the fabrication of pressure vessels and related components to the requirements of the ASME B&PV Code. Marvin provides the basis for the Specifications and Standards enveloped by Section II, Part C and their relations to the ANSI/AWS specifications. Marvin indicates that Section II, Part C does not include all the welding and brazing materials available to the industry—only those Specifications applicable to ASME Code Construction. Discussions also include Code Cases pertinent to this chapter. Chapter 3.4 highlights the major features of the Welding Material Specifications contained in Section II, Part C and the relationship of these Specifications to other Sections of the Code, including Section IX. Included are the electrode classification system, material descriptions, welding material applications, welding material procurement, and filler-metal certification. Chapter 3.4 should prove useful for one to gain a basic understanding of ASME/AWS Welding Material Classification and Specification.

Chapter 3.5, authored by Richard Moen and Elmar Upitis, covers Section II, Part D—*Properties*. The coverage includes properties of ferrous and nonferrous materials adopted by the Code for design of B&PV and nuclear power plant components. This coverage includes tables of maximum allowable stresses and design-stress intensities for the materials adopted by the various Code Sections, as well as a discussion of yield strength and tensile strength at various temperatures, external-pressure charts, and other properties for the design of items covered by the various Code Sections. With the aid of several tables, they provide in-depth information about "where is what" in Section II, Part D, and in addition, they note that although much of the information in the various Subparts and Appendices of Section II, Part D was compiled in several places in earlier Code Section Editions, in current editions it is compiled entirely in Section II, Part D to reduce the length of, avoid the duplication of, and facilitate the use of the Code Sections. Thus their commentary can be a useful "road map" even for Users of earlier Code Sections, because it encapsulates—all in one place-information crucial to Designers and Practicing Engineers.

In Chapter 4, Roger Reedy provides commentary for the understanding and application of the principles of the ASME B&PV Code. Roger traces the history of the Code, from its initial charter and the voluntary effort of engineers for establishing a Code with a safety record to the current Code developed by a consensus process. Roger suggests that Code Users apply common sense when using the Code and for understanding Code requirements. He emphasizes that "the Code is not a handbook and cannot substitute for the use of engineering judgment." Also, Roger emphasizes the need for a better understanding of the basic principles of the Code Interpretations in the application of safety factors for the Section III Nuclear Code, the Section VIII Pressure Vessels, and the Section I Boiler Code. Roger states that the term *safety factor* is both incorrect and misleading, because a reduction in the factor seems to indicate a reduction in safety. In fact, when the Code Committee considers a reduction in design factor, it allows the reduction only after it determines that other changes in Code requirements have compensated for the reduction.

Chapter 5, authored by Richard W. Swayne, describes the general requirements of Section III applicable to all Construction Classes, including concrete structures and steel vessels, piping, pumps, and valves. It identifies how to classify components and describes how the jurisdictional boundaries of Section III define what is within and what is outside the scope of the Code. This chapter includes coverage of Subsection NCA, which pertains to general requirements for Divisions 1, 2, and 3 of Section III. Division 1 includes steel items such as vessels, storage tanks, piping systems, pumps, valves, supports, and core support structures for use at commercial nuclear power plants; Division 2 includes concrete reactor vessels and concrete containment vessels; and Division 3 includes requirements for the construction of containment vessels for transportation of spent nuclear fuel. The scope of Division 3 now also includes recently-published requirements for construction of storage canisters for spent nuclear fuel and spent-fuel transportation-containment vessels.

Chapter 5 also explains the use of Code Editions, Addenda, and Code Cases. The requirements for design basis, design and construction specifications, and design reports are described, and the responsibilities and quality assurance program requirements of the different entities involved in nuclear power plant construction—from the Material Manufacturer to the Owner—are addressed. Requirements for ASME accreditation, application of the ASME Code Symbol Stamp, and use of Code Data Reports are described. With in-depth information, Mr. Swayne outlines the basis for exemptions, component classification, load combinations, responsibilities, Certificate of Authorization Holders and Quality System Certificate Holders. Also, Mr. Swayne provides cross-referencing to other Code Sections and Subsections, such as Sections III and XI, as well as to pertinent Regulatory Guides, such as the U.S. Code of Federal Regulations (CFR).

Chapter 6 originally authored by John Hechmer for the first edition and updated by Greg Hollinger for the second edition has been largely revised by David Jones for the current third edition.

Authors cover Subsection NB, *Class 1 Components*. In presenting the rules and requirements for Section III, Subsection NB, the authors discuss the theories, on which the rules and requirements are based, the appropriate application for applying the rules and requirements, and the interfaces for design, analysis, and construction. The chapter emphasizes the analytical rules and requirements, and makes reference to the *Criteria of the ASME Boiler and Pressure Vessel Code for Design by Analysis in Sections III and VIII, Division 2, 1968* that is considered the basis document for Sections III and VIII. John provided the design theory and ramifications of the key considerations, with cross-references to other Code Sections discussing the subtle differences between the Section III design criteria and the Section I and Section VIII, Division 1 design criteria.

In addition, commentary is provided on the Code requirements of Class 1 for design by analysis "because of the prominent role played by stress analysis in designing vessels by the rules of Section III . . . and because of the necessity to integrate the design and analysis efforts."—The authors emphasize that the design by analysis theme of NB is to provide high assurance that the failure modes of burst, plastic collapse, excessive plastic deformation, fatigue, ratcheting, brittle fracture, elastic instability (buckling), stress corrosion, and corrosion fatigue. The intent of the rules of NB is to provide assurance that high quality is reached; therefore, stress analysis is added to the "NB rules for all of the disciplines and their interaction" in an effort to reach high quality. Chapter 6 has been updated by Greg Hollinger and David Jones to the 2007 version of the Code including discussions of the differences between Section VIII Division 2 and Section III NB. Discussions have been added on the Section VIII Division 2 rules dealing with Limit Analysis, Finite Element Analysis and Environmental Fatigue, and new methods for fatigue of weldments.

Chapter 7, authored by Thomas J. Ahl for the first Edition, and co-authored by Marcus N. Bressler for the second Edition, provides commentary on Section III, Subsections NC and ND. This commentary addresses pressure atmospheric tanks, and 0–15 psig tanks as presented in the ASME B&PV Code, Section III, Division 1, Subsection NC, Class 2 Components and Subsection ND, Class 3 Components. This chapter does not address piping, pumps, and valves; these are addressed in Chapter 8 for Class 2 and Class 3 Piping, and in Chapter 13 for Nuclear Pumps and Valves. This chapter discusses, in order, each of the eight major Code Articles: Introduction; Materials; Design, Fabrication and Installation; Examination; Testing; Overpressure Protection and Name Plates; and Stamping and Reports. In the 1971 Edition, Subsection NB was fully developed in the evolution of the Nuclear Codes; all other were written by using the outline established for NB. Consequently, many of the basic paragraphs contained in Subsection NB and other reference documents were included verbatim in both Subsections NC and ND, when the subsections were published as separate volumes in the 1974 Edition.

Subsections NC and ND are a combination of rules and requirements taken from Section III, Subsection NB and Section VIII. In Chapter 7, Thomas has referenced all of these Codes and meticulously identified both obvious and subtle differences between Subsection NB, the parent Code, and Subsections NC and ND. Thus, because Thomas addresses the Articles of Subsections NC and ND in this part of the commentary, he presents comparisons, the most probable source of origin of the Code requirements, certain insights as well as contradictions that seem to exist, and the specific source document and some of the underlying theory. He provides cross-references to other Code Sections/Subsections/Paragraphs where applicable. Marcus has taken this work and simplified it where possible, and updated it to the 2007 Edition.

Chapter 8, authored by Donald Landers, discusses Section III, Division 1 (*Piping*). Chapter 7 indicates that the requirements of Section III, Division 1 provide for three classes of components. Chapter 8 indicates that each Class can be considered a quality level, with Class 1 the highest and Class 3 the lowest. These quality levels exist because of the various requirements for each Class in Section III related to materials, fabrication, installation, examination, and design. Design was placed last on the list because sufficient evidence exists to indicate that the other considerations listed are more important than (or, at best, equal to) the design requirements.

In Chapter 8, Donald mentions the foregoing list in his discussions leading to the Code requirements and his commentary regarding the criteria and basis for requirements of Subsections NB, NC and ND Piping. He provides the stress requirements for Nuclear Classes 1, 2, and 3 Piping and the corresponding design processes and Design Specifications, with pertinent references, tables, and figures. His commentary provides insight into load classifications and the responsibility of Owners. The Code rules ensure that violation of the pressure boundary will not occur if the Design Specification satisfactorily addresses all issues necessary for Code compliance. In his commentary, Donald shows the subtle differences between the piping rules and design by analysis, and he explains what items the analyst should be concerned with in satisfying Code requirements. He provides cross-references to B31.7 Code techniques and discusses the current controversy regarding seismic requirements in the piping rules in Section III, Division 1, along with the anticipated revisions that will resolve the controversy.

Chapter 9, has been authored by Kamran Mokhtarian for the previous two editions and now updated considerably by Roger F. Reedy continues the discussion of Subsection NE, *Class MC Components*. This chapter summarizes some of the more significant requirements of Section III, Subsection NE and provides a commentary on such requirements. Kamran's comments and interpretations of the rules are based on his several years of experience in design, analysis, and construction of containment vessels, as well as his participation in various ASME Code Committees. Some comparisons of the rules of Section VIII are included for information. The analysis procedures are not dealt with in any great detail, for they are similar to those of Subsection NB and Section VIII, Division 2; more emphasis is placed on the unique features of Subsection NE. A number of Code Cases and references regarding the rules of Subsection NE are cited, with cross-references to other Code Sections and Subsections. This chapter is based on the 2007 Edition of the Code.

The items covered in Chapter 9 include Scope of Subsection NE; Boundaries of Jurisdiction of Subsection NE; General Material Requirements; Certified Material Test Reports; Material Toughness Requirements; General Design Requirements; Qualifications of Professional Engineers; Owner's Design Specifications; Certified Design Report; Design by Analysis; Appendix F; Fatigue Analysis; Buckling; Reinforcement of Cone-to-Cylinder Junctions; Plastic Analysis; Design by Formula; Openings; Bolted Flange Connections; Welded Connections; General Fabrication Requirements; Tolerances; Requirements for Weld Joints; Welding Qualifications; Rules for Making, Examining, and Repairing Welds; Heat Treatment; Examination; Qualification and Certification of NDE Personnel; Testing; Overpressure Protection; and Nameplates, Stamping, and Reports.

Chapter 10 was authored for the first edition by Robert J. Masterson, who covered Subsection NF (*Supports*). The second and third editions had been updated by Uma S. Bandyopadhyay with the current third edition addressing the changes of the 2007 Code Edition. Robert traced the historical background of this Subsection, which provides a single source of rules for the design, construction, fabrication, and examination of supports for the nuclear industry. Section III, Division 1, Subsection NF was developed to provide rules for the estimated 10,000 piping and component supports existing in a typical nuclear power plant. The criteria and commentary of Chapter 10 provides information on the origin and evolution of design rules and is intended to allow designers, engineers, and fabricators to make better use of Subsection NF. Topics of greatest interest are discussed from both a technical and a historical viewpoint. However, it is not the intent to address every detail associated with the use of Subsection NF.

Subsection NF rules have evolved dramatically over the past 25 years so that today's support rules seldom resemble the original rules of 1973. In Chapter 10, commentary is provided to explain how the criteria are used, the source and technical basis for equations and rationale, and the reasons for change. Robert covers the scope and classification of the types of supports and attachments. Subsection NF contains rules for the material, design, fabrication, examination, testing, and stamping of supports for Classes 1, 2, 3, and MC construction. Robert provides cross-referencing to Subsections NB, NC, ND, NE, and NG, as well as to the B31.1 and B31.3 Codes, and he also addresses Code Cases and Interpretations. Discussions include Subsection NF Appendices and with the help of figures, tables, and references, it is anticipated that the reader will develop a better understanding of Subsection NF and appreciate its complexities and usefulness.

Chapter 11, authored by John T. Land, deals with Subsection NG (*Core-Support Structures*). This chapter provides commentary and practical examples on the materials, design, fabrication, installation, and examination requirements for core-support structures in Section III, Division 1, Subsection NG. In addition, commentary on Section XI as it applies to core-support structure repair, replacement, examination, and inspection requirements is presented. In the first edition, the 1998 Edition of the Code was used to provide examples and discussion points. In this edition, the 2001 Edition of the Code up to and including July 2003 Addenda is used to provide examples and discussion points. The objective of the Subsection NG rules is to provide a Code for the design and manufacture of structures that support the core in pressurized water reactors (PWRs) and boiling water reactors (BWRs). John indicates the subtle differences and overlaps between this Subsection and other Code Subsections. With the aid of figures, tables, and examples, John discusses important considerations in the design of core-support structures, the Owner's Design Specification, and the jurisdictional boundaries between core-support structures and reactor pressure vessels (RPVs). John explains the differences between core-support structures and internal structures, threaded structural fasteners, and temporary attachments. Discussions also include unique conditions of service; construction materials; special materials; fabrication and installation rules; examination and repair; general design rules; design by analysis; testing and overpressure protection; and examples of load combinations for core-support structures.

The third edition of this chapter has been updated to the 2007 Edition of the ASME B&PV Code with new or additional commentary covering: Background on Subsection NG Development; Discussion of Typical Materials Used in CSS, IS, and TSFs; Owner's Design Specification and Design Reports; Environmental Effects; CSS Code Cases; Improvements in Subsection NG; Material Degradation Issues; Compatibility of Subsection NG with Other International Codes; Trends Towards Realistic Design Loads in Reactor Internals; and Summary of Changes to 2007 Edition of the ASME Code for CSS.

Chapter 12, authored by Robert I. Jetter, discusses Subsection NH, 2007 Edition, (*Class 1 Components in Elevated Temperature Service*). The purpose of this chapter is to provide background information on the development and application of the rules for construction of elevated temperature components for nuclear service. Also discussed are the rules for Class 2 and 3 components and core-support structures that are contained in a series of Code Cases. Robert covers all aspects of construction: materials, design, fabrication, inspection, overpressure protection, testing, and marking for Class 1 components in elevated temperature service. In Section III, *elevated temperature* is defined as 700°F for ferritic steels and 800°F for austenitic stainless steels and nickel-base alloys. Elevated temperature behavior and associated failure modes are discussed to provide background for the unique features of the Subsection NH rules. Robert presumes that readers have a basic familiarity with the rules for construction of Classes 1, 2, and 3 components and core-support structures contained in Subsections NB, NC, ND, and NG, respectively, that are discussed in other chapters of this book. Thus Robert provides cross-referencing to these Code Subsections. Based on 40-plus years in the development and implementation of elevated temperature design and construction rules, Robert, with the aid of figures, tables, and references, provides a historical perspective to establish the criteria for the rules contained in Subsection NH. Also discussed are current and future needs.

Chapter 13 was authored for the first edition by the late Douglas B. Nickerson, who held several memberships on Code Committees spread over several decades. He was associated with the design and qualification of pumps and valves, a topic that he covers in this chapter. Marcus Bressler agreed to undertake the updating of this chapter for the Second Edition. Douglas discusses those items that are the driving and controlling forces in hydraulic systems for nuclear power plants. The pump in each system drives the flow through the piping to provide the transfer of energy from one component to another. The valves control the flow through these fluid systems and thus the operation of the systems. Fluid systems have varying degrees of criticality, depending on their function. This commentary explains the relevancy of the ASME Code requirements for safety-related nuclear pumps and valves using the latest issue of the Code. The Code is limited to pressure-boundary requirements. Douglas states that because of this limitation of the scope of the Code, most conditions necessary to the satisfactory design of a nuclear pump or valve are not subjected to Code rules. The Design Specification specifies operational requirements and thus is the most important element in their function and approval. This commentary not only defines the applicable Code but also explains how these components function in their applications.

Chapter 13 also discusses the role of system design and component design engineers, as well as the integrity of the Manufacturer. Douglas provides a historical perspective for the Code rules, cross-referencing other Subsections of the Code. He notes that *Owner's Responsibilities* for system design plays an important part in establishing the rules applicable to the Design Specification for each safety-related pump and valve. Drawing upon considerable practical experience, Douglas covers operational and qualification requirements for the procurement of these items from the Manufacturer. He discusses these items for different service conditions with the aid of schematics and references. Marcus, a member of the subgroup on Design since 1974, and Chairman of the working group on Valves from 1974 to 1977, provides the background to the development of the design rules for valves, and updates the Chapter to the 2007 Edition of the Code.

Chapter 14 describes the bases and provisions of the Code for Concrete Reactor Vessels and Containments updating to 2007 Code Edition. After a short description of the provisions for Concrete Reactor Vessels, the Chapter describes the concrete containment general environment, types of existing containments, future containment configurations, and background development including the regulatory bases of concrete containment construction code requirements. The description covers sequentially the following topics: Introduction, Concrete Reactor Vessels, Concrete Reactor

Containments, Types of Containments, Future Containments, Regulatory Bases for the Code Development, Background Development of the Code, Reinforced Concrete Containment Behavior, Containment Design Analysis and Related Testing, Code Design Requirements, Fabrication and Construction, Construction Testing and Examination, Containment Structural Integrity Testing, Containment Overpressure Protection, Stamping and Reports, Containment Structure and Aircraft Impact, Containment and Severe Accident Considerations, Other Relevant Information, Summary and Conclusion.

The previous editions of this Chapter were developed by John D. Stevenson, and it has been expanded by the current authors, utilizing the expertise of their respective fields. The basic format of this chapter is kept the same as in the previous editions. The updates and additional information relating to the regulatory bases for the code requirements, future containments and considerations for future revisions of the Code included in this update are based on contributions from Hansraj Ashar, Barry Scott, and Joseph Artuso.

In Chapter 15, authored by D. Keith Morton and D. Wayne Lewis, a commentary is provided regarding the containments used for the transportation and storage packaging of spent fuel and high-level radioactive material and waste. John D. Stevenson was the author of this chapter for the earlier two editions of this publication. However, this is a complete rewrite of the Chapter, including a slightly different Chapter title.

In 1997, ASME issued the initial version of Division 3 of Section III. Before the publication of Division 3, Section III, the Section applicable to the construction of nuclear pressure-retaining components and supports had only two divisions: Division 1, for metal construction, and Division 2, for concrete construction. Division 3 was added to cover the containments of packaging for nuclear materials. Currently, the scope for Division 3 is limited to transportation and storage containments for only the most hazardous radioactive materials—namely, spent fuel and other highly radioactive materials, such as high-level waste. Division 3 contains three published subsections: Subsection WA providing general requirements, Subsection WB addressing rules for transportation containments, and Subsection WC addressing storage containment rules. Under active development is Subsection WD, which will provide the construction rules applicable to internal support structures (baskets) for the transportation and storage containments covered by Subsections WB and WC.

Consistent with current Code practice, the primary concern of Division 3 is the integrity of these containments under design, operating conditions (including normal, off-normal, and accident), and test conditions. In particular, the structural and leak-integrity of these containments is the focus of the ASME B&PV Code rules. Division 3 is also concerned with certain aspects of containment-closure functionality because of the potential for leakage, which is a key consideration in the containment function. Division 3 covers all construction aspects of the containment, including administrative requirements, material selection, material qualification, design, fabrication, examination, inspection, testing, quality assurance, and documentation.

Chapter 16, authored by Charles Becht, IV, covers *Power Piping*, the ASME B31.1 Code. This chapter is based on the 2007 edition of ASME B31.1, Power Piping Code. This Code was written specifically for power piping; it is intended to cover fuel-gas and fuel-oil systems in power plants (downstream of the meters), central- and district-heating systems; and water and steam systems in power plants. Charles provides exhaustive coverage of the overlapping and interfacing Codes and Standards that Users of the B31.1 Power Piping should be aware of, and he also discusses the applicability of this Code to various applications and systems. His coverage includes cross-references to ASME B&PV Code Sections and to other B31 Codes, as well as to API, AWWA, ASTM, and other pertinent Standards and publications. Also included is the balance of plant piping beyond the block valve(s) that defines the boundary of the boiler, the rules of which fall entirely within the scope of the B 31.1 Power Piping Code. Charles distinguishes this Code from *Process Piping*, the ASME B31.3 Code, as well as from other B31 Codes.

Chapter 16 provides a commentary, discussing the historical perspective of, information about, and sources of the B31.1 Code. With the aid of equations, schematics, figures, tables, and appendices, Charles elaborates on the basics of the B31.1 Code. Topics include design conditions and criteria, including thermal expansion; design for pressure; flexibility analysis; supports and restraints; an overview of materials; components and joints; requirements for specific piping systems; fabrication, assembly, and erection; pressure testing; and nonmetallic piping systems. References are included for each of these topics.

Chapter 17, also authored by Charles Becht, IV, covers *Process Piping*, the ASME B31.3 Code. This Code has the broadest scope of application of any B31 Code for pressure piping. This chapter covers essentially the entire B31.3 Code, including design, materials, fabrication, assembly, erection, examination, and testing, and includes special topics, such as nonmetallic piping and piping for Category M and high-pressure fluid services. This chapter is based on the 2006 edition of ASME B31.3, Process Piping Code. Changes—some very significant—are made to this Code every year, for which reason the reader should refer to it for any specific requirements. Charles cross-references Sections I, II, III, V, VIII, and IX of the ASME B&PV Code, and he also cross-references API, AWWA, ASTM, and other pertinent Standards and publications.

Charles provides the history of the B31.3 Code and the over-laps of and differences between this Code and other B31 Codes. The B31.3 Code was written specifically for process piping; Chapter 17 provides examples of the typical facilities for which the Code is intended to cover. The exclusions for the applications with the B31.3 Code are discussed, and with the aid of examples, figures, tables, appendices, and references for each topic, a detailed commentary is provided for the following topics: design conditions and criteria; design for pressure; flexibility analysis; supports and restraints, limitations on components and joints; requirements for materials; fabrication, assembly, and erection; examinations; pressure testing; nonmetallic piping systems; Category M piping; high-pressure piping; and the organization of the B31.3 Code.

Chapter 18, was authored by M. A. Malek and John I. Woodworth for the first edition, and co-authored by Geoffrey M. Halley for the Second edition. The current third edition has been revised by Edwin A. Nordstrom. In the first edition, the chapter covered Section IV, *Rules for Construction of Heating Boilers*, using the 1998 Edition, 1999 Addenda, and Interpretations and has now been updated to the 2007 edition. To assist the reader in understanding and using the Code, this chapter is presented in a simplified manner, with the understanding that it is not a Code book and is not written to replace the Code book published by ASME. A historical perspective of Section IV is provided to trace the criteria covered by the Code. The authors define the boilers that fall within the jurisdiction of this Section and provide a detailed discussion of the minimum requirements for the safe design, construction, installation, and inspection of low-pressure-steam boilers and hot-water boilers, which are directly fired with oil, gas, electricity, or other solid or liquid fuels. However, the authors do not cover the operation, repair, alteration, rerating, and maintenance of such boilers, but they do cover potable-water heaters and water-storage tanks for operation at pressures not exceeding 160 psi and water temperatures not exceeding 210°F.

In the first edition, Chapter 18 addressed the Code Interpretations, the Addenda, and the Code Inquiry procedure as they relate to Section IV. The authors mentioned that the format used for this chapter is compatible with the format used in Section IV (1998 Edition, 1999 Addenda, and Interpretations). For the current edition using the 2007 Code, this is still valid. For easy identification, the exact numbers of paragraphs, figures, and tables from the Code book have been used in the running text. The appendices include Method of Checking Safety Valve and Safety Relief Valve Capacity; Examples of Methods of Calculating a Welded Ring Reinforced Furnace; Examples of Methods of Computation of Openings in Boiler Shells; Glossary; and two examples of Manufacturer's Data Report Forms.

Chapter 19 provides criteria and commentary for ASME Section VI, *Recommended Rules for the Care and Operation of Heating Boilers*. This chapter that had been initially authored by M. A. Malek was updated for the second edition by Geoffrey M. Halley with Edwin A. Nordstrom as the author of the current edition. While heating boilers are designed and constructed safely under Section IV, the rules of this Section are nonmandatory guidelines for the safe and efficient operation of steam-heating boilers, hot-water-supply boilers, and hot-water-heating boilers after installation. These rules, however, are not applicable to potable-water heaters. This chapter is divided into nine parts, along with the necessary figures and tables for each part: General, covering the scope, use of illustrations, manufacturer's information, references to Section IV, and glossary of terms; Types of Boilers; Accessories and Installation; Fuels; Fuel-Burning Equipment and Fuel-Burning Controls; Boiler-Room Facilities; Operation, Maintenance, and Repair of Steam Boilers and Hot-Water Boilers; and Water Treatment. The authors have several years of professional field experience in overseeing Code implementation and are conversant with regulatory practice; as such, they discuss the juris-dictional responsibilities and role of licensing agencies.

The authors note that the format used for this chapter is compatible with the format used in Section VI 2007 Code Edition. For easy identification, the exact numbers of paragraphs, figures, and tables from the Code book have been used in the running text. The Exhibits include the maintenance, testing, and inspection log for steam-heating boilers and the maintenance, testing, and inspection log for hot-water-heating boilers and tests. Bibliographical references and notes are also provided.

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The first edition of Chapter 20, was authored by Harold C. Graber, and the subsequent second edition as well as the current third edition have been revised by Jon Batey. The authors discuss Section V, *Nondestructive Examination* (NDE). The purpose of this chapter is to provide Users of Section V insight into the significant requirements, the NDE methods, the NDE methodology, the relationship of Section V with other Code Sections, and the use of ASTM Standards. The information provided is based on the 2007 Edition of Section V. The charter and scope of this Section is to develop and maintain Code rules for NDE methodology and equipment involved with the performance of surface and volumetric testing methods. These test methods are used for the detection and sizing of defects, dis-continuities, and flaws in materials and weldments during the manufacture, fabrication and construction of parts, components, and vessels in accordance with the ASME B&PV Code and other ASME Codes, such as B31.1 (*Power Piping*)

Harold and Jon provide commentary on the contents of Section V, including Subsection A, which contains Articles and both Mandatory and Nonmandatory Appendices that address general requirements, test methods, and specific Code requirements and acceptance criteria; and Subsection B, which contains the ASTM Standards adopted by the ASME B&PV Code. This chapter addresses an audience that includes manufacturers (including equipment manufacturers), insurance companies, architect-engineers, research organizations, utilities, consultants, and the National Board. The authors address additions, revisions, inquires, interpretations, and Code Cases relevant to Section V. An important aspect of this chapter is its coverage of the inter-connection of Section V with other Code Sections and Subsections. This coverage provides insight into how the rela-tionships of the Code Sections are integrated.

Chapter 21 initially authored by Urey R. Miller for the previous first and second editions is revised by Thomas P. Pastor for the current third edition. This chapter covers Section VIII—Division 1, Rules for Construction of Pressure Vessels. The author discusses the historical background of this Section in relation to the construction and safe operation of boilers and pressure vessels. Section VIII Division 1 is written to cover a wide range of industrial and commercial pressure vessel applications. This Section is applicable to small compressed air receivers as well as to very large pressure vessels needed by the petrochemical and refining industry. Section VIII Division 1 is intended for the construction of new pressure vessels. Miller discusses the applicability of Code and Code jurisdictions, as well as situations of the inapplicability and exemptions from this Section.

This chapter provides an overview to each of the parts of Section VIII Division 1 Code. The commentary includes Subsection A—General Requirements for All Methods of Construction and Materials; Subsection B—Requirements Pertaining to Methods of Fabrication of Pressure Vessels; Subsection C—Requirements Pertaining to Classes of Material; Mandatory Appendices; Non-Mandatory Appendices; and Bibliography. The intent of the author is to provide a broad per-spective for the reader to have better understanding of the Code's intent, and to point out, by example, some of the sub-tleties that may not be evident. It is not the objective of this Chapter to provide the reader with a detailed "how to" hand-book. The user of the equipment must define the requirements that are needed for a specific application. With the help of equations, tables, figures and examples Miller provides detailed commentary of Section VIII, Division 1. He comments about several pertinent Code Interpretations and Code Cases pertaining to this Section.

There have been a number of significant changes to Section VIII Division 1 since the First Edition of this Guidebook. The most significant is that the previously non-mandatory rules for tubesheets (Appendix AA) and flanged and flued expansion joints (Appendix CC) are now mandatory and are in Part UHX and Appendix 5 respectively. Also, a new mandatory appendix (Appendix 32) has been added to the Code to allow consideration of local thin spots in shells and heads, and Appendix 33 has been added to define the standard units to be used in Code equations.

The 2nd Edition of the Guidebook was updated to cover the ongoing Code revisions (that affect shell-to-tubesheet joints, Appendix 26 expansion joints, and Appendix M.

The Third Edition of the Guidebook covers revisions to Section VIII, Division 1 from the 2004 Edition through the 2007 Edition. Included are detailed descriptions of several new Nonmandatory Appendices, including Appendix FF: "Guide for the Design and Operation of Quick-Actuating (Quick Opening) Closures, and Appendix GG: "Guidance for the Use of U. S. Customary and SI Units in the ASME Boiler and Pressure Vessel Code". This Chapter also includes extensive updating of referenced figures and tables from the 2007 Edition of Section VIII, Division 1.

Chapter 22, initially authored by Kamran Mokhtarian for the previous two editions has been revised in its entirety by David A. Osage, Clay D. Roldery, Guido G. Karcher, Thomas P. Pastor, Robert G. Brown and Philip A. Henry. This revision covers the 2007 Edition of Section VIII, Division 2. The 2007 Edition is a complete rewrite of the standard, a project that began in 1998 and took nine years to complete. The decision to completely rewrite VIII-2 was made so that the standard could be modernized with regard to the latest technical advances and pressure vessel construction, and also be structured in a way to make it more user-friendly for both users and the committees that maintain it. The 2007 Edition of Section VIII, Division 2 is the result of an extensive collaborative effort involving ASME Volunteers and Staff, the Pressure Vessel Research Council and The Equity Engineering Group.

Mr. David A. Osage was the lead author for the new standard, and he made significant contributions to the design by rule and design by analysis chapters (Parts 4 and 5). He also had responsibility for the assembly of all material that ultimately made up the 9 parts of the new standard: 1. General Requirements, 2. Responsibility and Duties, 3. Materials Requirements, 4. Design by Rule Requirements, 5. Design by Analysis Requirements, 6. Fabrication Requirements 7. Inspection and Examination Requirements, 8. Pressure Testing Requirements, 9. Pressure Vessel Overpressure Protection.

This chapter provides an overview of the development of the new standard, its organization, and a detailed description of each of the nine parts. Emphasis is given to those areas of the standard where new technology was introduced.

Chapter 23, authored by J. Robert Sims, Jr., discusses Section VIII, Division 3 (*Alternative Rules for the Construction of High-Pressure Vessels*). It is intended to be used as a companion to the Code by Manufacturers and Users of high-pressure vessels and also provides guidance to Inspectors, materials suppliers, and others. The chapter's text is generally presented in the same order in which it appears in the Code. Comments are not given about each Paragraph, but Paragraph numbers are referenced as appro priate. The comments apply to the 2007 Edition. The ASME Subgroup on High-Pressure Vessels (SG-HPV) of Subcommittee VIII developed the Code. The comments herein are Bob's opinions; they should not be considered Code Interpretations or the opinions of the Subgroup on High-Pressure Vessels or any other ASME Committee.

This chapter provides commentary that is intended to aid individuals involved in the construction of high-pressure vessels, but it cannot substitute for experience and judgment. Bob covers general, material, and design requirements; supplementary requirements for bolting; special design requirements for layered vessels; design requirements for attachments, supports, and heating and cooling jackets; fracture mechanics evaluation; design using autofrettage; special design requirements for wirewound vessels and frames; design requirements for openings, closures, heads, bolting, and seals; scope, jurisdiction and organization of Division 3; fatigue evaluation; pressure-relief devices; examination, fabrication, and testing requirements; marking, stamping, reports, and records; and Mandatory and Nonmandatory appendices.

Appendix to Part 7 has been authored by Roger Reedy and is a new chapter included in this volume 2 of the third edition. Part 7 covers ASME Section VIII—Rules for Construction of Pressure Vessels with chapters dealing with Section VIII Division 1- Rules for Construction of Pressure Vessels, Section VIII: Division 2-Alternative Rules, and Section VIII, Division 3—Alternative Rules for Construction of High-Pressure Vessels. This Chapter written by Roger F. Reedy deals with items pertaining to Part 7 Safety of Personnel Using Quick-Actuating Closures on Pressure Vessels and Associated Litigation Issues.

The Appendix to Part 7 is new and was written because of the number of lawsuits against manufacturers of quick-actuating closures on pressure vessels. Often manufacturers are sued even though the closures had been operating with no accidents for 20 or 30 years. Because of Worker's Compensation rules, the owner of the equipment often cannot be sued, so the lawyers search for "deep pockets" to compensate their clients and themselves. In order to bring forth litigation, these lawyers would skillfully take words in the Code completely out of context. The Appendix is based on Roger's personal experience in a number of litigations involving quick-actuating closures during the last 25 years. He identifies each of the changes made to the Code rules in Section VIII, Division 1, from 1952 to the 2007 Edition of the ASME Code. In every case where Roger has testified as an expert witness, the manufacturer of the quick-actuating closure was not at fault, and the ASME Code rules had been properly followed. However, the attorneys for the injured party often misinterpret the Code rules to accuse the manufacturer of not having complied with the Code when the closure was made. Based on experience, Roger warns the writers of the ASME Code to assure that the rules are clear, concise and understandable to the common man. The most important point however, is for everyone to understand that in order to avoid severe accidents, users of quick-actuating closures must maintain the equipment and ensure that inferior components are not used as replacement parts, and that the design is not modified or changed. The other key element for safety is that owners of pressure vessels that have quick-actuating closures are responsible for training all employees regarding the proper care and use of the equipment. This training has been neglected by the employer in most accidents.

Chapter 24, authored by Joel G. Feldstein, discusses Section IX, *Welding and Brazing Qualifications*. As the title indicates, this chapter deals with the qualification of welding and brazing proce-dures as well as the qualification of individuals performing those procedures as required by the Construction Codes of the ASME B&PV and Piping Codes. Joel discusses the two-Part organization of the 2007 Edition of Section IX: Part QW, covering welding, and Part QB, covering brazing. Each Part is divided into four Articles. The coverage for Part QW includes general requirements for both welding procedure and welder performance qualification and the variables applicable to welding procedure and welder per-formance qualification. Part QB has a similar format: general requirements for brazing procedure and brazer performance qualification and the variables applicable to brazing procedure and brazer performance qualification. Commentary is provided on all of the Articles with aid of figures and tables, and Code Interpretations are used to provide the Code User with some insight into the requirements of Section IX.

Joel provides a description of the more common welding processes used in Code construction, reviews the qualification rules, provides commentary on those requirements, and covers the historical background leading to the increased use of welding in manufacturing operations. Where comments are provided, they represent Joel's opinions and should not be regarded as the positions of the ASME Code or its Subcommittee on Welding.

In Chapter 25, Peter J. Conlisk covers Section X, *Fiber-Reinforced Plastic Pressure Vessels*, and ASME RTP-1, *Reinforced Thermoset Plastic Corrosion-Resistant Equipment*. Peter mentions that this chapter is tailored for engineers and designers whose experience with vessels is primarily with metal equipment, although he adds that those with experience using fiberglass equipment but not using Section X or RTP-1 will also find this chapter useful, especially its discussions on fiber-rein-forced plastic (FRP) technology. Section X has been enacted into law in 37 jurisdictions in the United States and Canada, whereas RTP-1, although usable as a Code, has not been enacted into law anywhere; therefore, at present, it is a voluntary Standard. Both Standards govern vessels constructed of thermosetting resin rein-forced with glass fibers, but Section X addresses vessels rein-forced with carbon or aramid fibers as well. The pressure scope of Section X is 15 psig internal pressure and greater, the upper limit depending on the size and construction of the vessel. RTP-1 cov-ers tanks and vessels with design pressures of 0 to 15 psig. Both Standards have provisions for vessels with external pressures of 0 to 15 psig.

Neither RTP-1 nor Section X is meant to be a handbook or textbook on FRP vessel design. Chapter 25 is intended to be a manual on the use of these documents. An engineer who specifies an FRP vessel does not need to understand FRP to the same extent that a vessel designer does; however, in specifying the vessel, an engineer necessarily makes many design choices. Peter discusses the basics of FRP technology; the fabrication methods and stress analysis of FRP vessels; the scope of Section X and RTP-1; the design qualification of Section X, Class I, Class II, and RTP-1 vessels; the design qualification overview; Section X example of a Design Specification and its calculations; RTP-1 design examples; and quality assurance of Section X and RTP-1. He provides equations, tables, and figures as well as annotated bibliographical notes indicating the relevance of the cited references.

In Chapter 26, Owen F. Hedden provides an overview of the stipulations of Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*. A chronological overview of the development of Section XI is presented, from its inception in 1968 up to the 2004 Edition including 2006 Addenda. The chapter traces the development, Edition-by-Edition, of important elements of the Code, including the philosophy behind many of the revisions. Emphasis is placed on the 1989 through 2004 Editions, for they apply to the majority of plants in the United States and elsewhere. Through an extensive tabulation of Code Interpretations, this chapter also attempts to give the Code User some insight into clarification of many Section XI requirements.

In the current revisions of Section XI, feedback from operating plants has resulted in new requirements to address stress corrosion cracking mechanisms, weld overlay piping repair techniques, and a program for risk-informed piping inspections.

Owen notes that subsequent chapters of this book address the major areas of Section XI: inservice inspection examination and test programs, repairs and replacements, acceptance and evaluation criteria, containment programs, and fatigue crack growth. Nondestructive examination (NDE) is addressed in this chapter, as its requirements evolve. Owen mentions that Section XI initially had only 24 pages in 1970 but that it now has over 700 pages. Although originally it covered only light-water reactor Class 1 components and piping, now it includes Class 2 and Class 3 sys-tems, metal and concrete containment, and liquid metalcooled reactor plants. With his association with Section XI Code Committee activities since their beginning, Owen is in a good posi-tion to comment on important areas that should not be overlooked as well as unimportant areas that should not distract attention.

In Chapter 27, Richard E. Gimple addresses the repair/replacement (R/R) activities for nuclear power plant items. Article IWA-4000 of Section XI contains the requirements for performing R/R activities on nuclear power plant items. Richard examines the background of these R/R activities and the changes in R/R activity requirements since the original 1970 Edition, and he reviews in detail the requirements in IWA-4000 in the 2007 Edition of Section XI. This information is beneficial to personnel performing R/R activities (e.g., designing plant modifications, obtaining replacement items, and performing welding, brazing, defect removal, installation, examination, and pressure-testing activi-ties). Although the 2007 Edition is used to discuss IWA-4000 requirements, discussions involving earlier editions and addenda of Section XI have been retained from previous editions of the Companion Guide. The thorough discussion of changes from earlier editions and addenda will be very beneficial to personnel using other editions and addenda, especially those updating their Repair/Replacement Programs.

In Chapter 27, Richard uses his unique professional expertise to discuss R/R activity requirements and provides the basis and pertinent explanations for the requirements. The discussion of the scope and applicability of Section XI R/R activities is informative to both new and longtime Users. Richard notes that Section XI is used in many countries, that it is often recognized as an international Standard, and has written Chapter 27 such that it applies regardless of the country where the Section is used. To benefit the reader, numerous Code Interpretations and Code Cases are included in this chapter to help clarify and implement R/R activities. Commentary is provided regarding Interpretations that might be of great benefit in understanding the Code. With over 20 years of association with Code Committee activities, Richard provides clarity and in-depth understanding of Section XI.

Chapter 28, authored by Richard W. Swayne, discusses the Section XI rules for inservice inspection and testing of nuclear power plant components. This chapter covers the general requirements of Section XI applicable to all Classes of components, including concrete structures and steel vessels, piping, pumps, and valves. It identifies the limits of applicability of Section XI, that is, what is within and outside the scope of the Code. Interfaces with applicable regulatory requirements are addressed, and the use of Code Editions, Addenda, and Cases is explained. Mr. Swayne comments on the periodic NDE and pressure testing to ensure the integrity of components, other than containment vessels, within the scope of jurisdiction of this Code. These requirements include NDE, from personnel qualification to conduct of the NDE. They also include the type and frequency of NDE required, including sample expansion and increased frequency required because of defect detection.

Mr. Swayne also addresses periodic pressure testing and pressure testing following R/R activities. Responsibilities and quality assurance program requirements of the different entities involved in examination and testing of a nuclear power plant are discussed.

This chapter addresses many controversial issues and topics of current concern, including the applicability of recent U.S. Nuclear Regulatory Commission (NRC) Generic Letters and Information Notices, and describes ways in which readers can use recent revisions of Section XI to their advantage. References to ASME

Interpretations are included to explain how the Code requirements can be applied to common problems. This Edition contains new information from Mr. Swayne on risk-informed inservice inspection and reliability integrity management programs for high-temperature gas-cooled reactors.

In Chapter 29, which was originally written by Arthur F. Deardorff, and updated and expanded by Russell C. Cipolla, the flaw acceptance criteria and evaluation methods specified in the 2007 Edition of ASME Section XI Code are discussed. Coverage includes the evaluation of flaws in nuclear power plant components and piping using ASME Section XI procedures. The authors discuss flaw acceptance criteria based on the use of predefined acceptance standards and of detailed fracture-mechanics evaluations of flaws. Commentary is provided on flaw characterization and acceptance standards, Class 1 vessel flaw evaluation, piping flaw evaluation (for austenitic and ferritic materials), and evaluation of piping thinned by flow-assisted corrosion. The authors discuss the background and philosophy of the Section XI approach for evaluating inservice degradation, including the rules for inservice inspection of nuclear power plant components and piping as they relate to the criteria, to determine if flaws are acceptable for continued operation without the need for repair.

Drawing upon their participation in Code Committees and professional experience with both domestic and international nuclear plants, the authors discuss step-by-step procedures for the evaluation of flaws in austenitic and ferritic components and piping. The underlying philosophy of Section XI evaluation of degraded components is to provide a structural margin consistent with that which existed in the original design and construction code. Russ has expanded the chapter to describe the updated flaw evaluation procedures for piping, which were added to Section XI in 2002. Also discussed are revised flaw acceptance criteria for Class 1 ferritic vessels in IWB-3610, updated structural factors for austenitic and ferritic piping in Appendix C, and revised fatigue crack growth reference curves, along with the technical basis for these changes.

Russ has also added the historical development and technical basis for Appendices E, G and K, which deal with evaluations for fracture prevention during operating plant events/conditions in the fracture-toughness transition temperature region, and at upper shelf. Further, recent Code Cases N-513 and N-705 to Section XI are described, which cover the requirements and procedures for temporary acceptance of service induced degradation in piping and vessels in moderate energy Class 2 and 3 systems. The degradation can be associated with various mechanisms (cracking, pitting, general wall thinning, etc.) and can include through-wall degradation where leakage can be adequately managed via monitoring. These Cases provide the basis for continued operation until repair can be implemented at a later time.

Wherever possible, the authors cite references to published documents and papers to aid the reader in understanding the technical bases of the specified Code flaw evaluation methods and acceptance criteria. The authors also cite related Section XI requirements that are discussed in Chapters 26, 27, 28, 31, 32, 35, and 39 of this book.

Chapter 30 originally authored by the late Robert F. Sammataro (a well-known and respected colleague well-versed in ASME Codes and Standards) and now updated by Jim E. Staffiera, addresses Subsections IWE and IWL for nuclear containment vessels. Subsection IWE, *Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Plants*, specifies requirements for preservice and inservice examination/inspection, repair/replacement activities, and testing of Class MC (metal containment) pressure-retaining components and their integral attachments and repair/replacement activities and testing of Class CC (concrete containment) pressure-retaining components and their integral attachments for BWRs and PWRs. Similarly, Subsection IWL, *Requirements for Class CC Concrete Components of Light-Water Cooled Plants*, specifies requirements for preservice and inservice examination/inspection, repair/replacement activities, and testing of the reinforced concrete and the post-tensioning systems of Class CC (concrete containment) components for BWRs and PWRs. Together with Subsection IWA, *General Requirements*, a comprehensive basis is provided for ensuring the continued structural and leak-tight integrity of containments in nuclear power facilities.

Subsections IWE and IWL also provide requirements to ensure that critical areas of primary containment structures/components are inspected to detect degradation that could compromise structural integrity. These two Subsections have received significant attention in recent years since the Nuclear Regulatory Commission (NRC) mandated nuclear-industry compliance with these two Subsections of the Code through publication of revised Paragraph 55(a) of Title 10, Part 50, of the Code of Federal Regulations [10 CFR 50.55(a)] in September, 1996. In incorporat-ing these two Subsections into the Regulations, the NRC identi-fied its concern with the increasing extent and rate of occurrence of containment corrosion and degradation. Since that time, numerous additional changes have taken place in all aspects of nuclear power plant inservice inspection requirements, not the least of which have been those for nuclear containment vessels. With increasing emphasis in the nuclear industry on plant life-extension, these changes have resulted in several initiatives cur-rently moving through the ASME Code ‘consensus-committee’ process, including action items addressing the need for more appropriate and effective examinations/inspections and the expanded use of risk-informed inservice inspection activities.

This updated Chapter 30 introduces the latest Commentaries for Subsections IWE and IWL, important documents for users of the Code because of the background information and technical justification provided regarding the reasons for changes made to these two subsections over the years. As noted in the Introduction to this book, the user is cautioned that these documents are the opinions of individuals only. These documents are not products of the ASME Code Committee consensus process, and thus do not represent ASME Code Committee positions.

In Chapter 31, Warren H. Bamford discusses the Code evaluation of fatigue crack growth, consistent with the evaluation methods of Section XI. Fatigue has often been described as the most common cause of failure in engineering structures, and designers of pressure vessels and piping have incorporated fatigue considerations since the first Edition of Section III in 1963. The development of this technology and its application in Section III is discussed in Chapter 39; its application in Section XI is discussed in Chapter 31. With the advancement of the state of the art has come the capability for allowing the presence of a crack, for predicting crack growth, and for calculating the crack size that could lead to failure. This capability has been a key aspect of the Section XI flaw evaluation procedures since the 1974 edition of Section XI; it is discussed thoroughly in Chapter 31.

Warren discusses the background of the criteria for fatigue crack growth analyses and crack growth evaluation methods. Drawing upon his considerable experience in formulating these criteria and his professional expertise in these analyses and evaluations, Warren provides commentary on the calculation of crack shape changes; calculation of elastic–plastic crack growth with the aid of crack growth rate reference curves for ferritic and austenitic steels in air environments; and crack growth rate curves for ferritic and austenitic steels in water environments. He also discusses operating plant fatigue assessment with the aid of Appendix L of Section XI. Also included are discussions pertaining to Appendix A, fatigue evaluation, and flaw toler-ance evaluation. He provides extensive bibliographical notes and references.

PART 11: SPECIAL TOPICS RELATED TO ASME B&PV AND PIPING CODES

Part 11, unlike Parts 1–10, discusses topics that are not covered exclusively by specific Code Sections/Subsections/Paragraphs. Even when the topics are covered, their usage is so overwhelming that a detailed discussion about them is warranted. Thus the chapters of Part 11, written by recognized authorities in their respective fields, not only clarify subtle points of professional interest to the Practicing Engineers but also elaborate on the basic information of the criteria of the subjects discussed.

In Chapter 32, John D. Stevenson covered in first and second edition the Maintenance Rule. In this edition this has been updated by C. Wesley Rowley. July 10, 1991, a Maintenance Rule titled *Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants* was published by the NRC in the Federal Register (56 Fed. Reg. 31324) as 10 CFR 50.65. The rule was developed for the NRC to have an established regulatory framework to provide the means for evaluating the administrative effectiveness of nuclear power plant licensees' maintenance programs. The NRC's overall objective is that structures, systems, and components important to nuclear power plant safety be maintained properly so that plant equipment perform its intended safety functions reliably when required. The Maintenance Rule is performance-based, providing focus on results rather than on programmatic prescriptive requirements.

With his experience in domestic and international nuclear power plants, Wes updates the discussions pertaining to the rudiments of the Maintenance Rule and the criteria on which the Rule is based. Wes also updates historical background, purpose and scope, and expectations of the license holder to abide by the stipulations dictated in the Maintenance Rule. His commentary includes requirements; the methodology to select plant structures, systems, and components (SSCs); the use of existing Standards and programs; establishing risk and performance criteria, goal setting, and monitoring; SSCs subject to effective program maintenance programs; the evaluation of systems to be removed from service; periodic maintenance effectiveness assessments; documentation; and references.

In Chapter 33, Marcus N. Bressler discusses the rules for accreditation and certification and similar issues. This chapter has been revised to address the current 2007 Code revisions and to up-date several accompanying graphics. This chapter is intended to provide an overview of the history of the ASME B&PV Code from its inception through the incorporation of nuclear components to the present. Accordingly the title of the Chapter has also been revised to reflect the scope of the discussions contained in the chapter. From earlier coverage of boilers and pressure vessels, the nuclear initiative required coverage for piping, pumps, valves, storage tanks, vessel internals, and compo-nent and piping supports. Rules for repairs and replacements of nuclear components and the use of newer Codes are discussed, including the need for Code reconciliation and commercial grade dedication. The development of certification and accreditation is covered, with emphasis on the new requirements for organizations seeking ASME accreditation. The globalization of the ASME cer-tificates and stamps is described thoroughly.

Marcus, based on his experience working with many Code Committees, is justly the right person to provide in-depth coverage of the various aspects of the Code and connecting these with the topic of this chapter. A discussion is provided regarding the role of regulators, and the commentary includes accreditation for nuclear construction, Code stamping, the role of the Registered Professional Engineer and Authorized Nuclear Inspector (ANI), and the related Mandatory and Nonmandatory Appendices in the Code. Marcus details two examples of the use of Code reconciliation: repair of steam-generator feedwater-nozzle cracks at a 1970s nuclear plant and the support material requirements of an Example Nuclear Plant (ENP).

Marcus Bressler has provided in this revised script his thoughts on Future Developments in the ASME Boiler & Pressure Vessel Code, as well as ASME's emphasis on Globalization of its Codes and Standards. In addition, applicable Code references, description of a typical nuclear survey, and an annotated bibliography are also provided.

John D. Stevenson initially authored Chapter 34 for the first and second editions of this publication. This update of this chapter is covered by Michael A. Porter.

In the previous editions John Stevenson dealt with perspectives on cyclic, impact, and impulse dynamic loads. John notes that dynamic loads applied to the design of mechanical systems and components are of three basic types: cyclic, impulse, and impact. In addition, there is a fourth potential cyclic-type load in the vibratory motion category. Although vibratory motion is not usually considered in the original design basis, it may be observed during steady-state or transient operations to cause premature fatigue or ratchet failure of metal components. The original Section III definition of a plant's operating life includes design-basis normal, abnormal, emergency, and faulted plant- or system-operating conditions, as defined in the Design Specification. John notes that these operating conditions should not be confused with Service Levels A, B, C, and D currently defined in the Code for design purposes. It is possible to have different Service Level design conditions for the same operating condition, depending on the required response of a component.

John covered in-depth Nonmandatory Appendix N (*Dynamic Analysis Methods*) of Section III. For completeness, he covers other types of dynamic loads not addressed explicitly by Appendix N that the designer of pressure-retaining nuclear components must consider. He also discusses the ASCE Standard used for defining earthquake motions to a building foundation and for supporting the mechanical system or component; these two references deal primarily with earthquake cyclic–type dynamic loads. In addition, commentary is provided on the guidelines used for dynamic impulse and impact loadings provided in the ANS, ASME Appendix B, and B31.1 Code Standards. A discussion is provided about the ASME Operation and Maintenance (O&M) Standard, used for determining the effect of operational vibratory motion independent of the cause of vibration.

In past editions of the handbook, this chapter has primarily addressed issues as they pertained to Section III of the ASME B&PV Code. In particular, it has addressed issues concerning the seismic response of nuclear facilities. Currently, many other facilities covered by the ASME B&PV Code have had to address these same issues, often with little guidance from the appropriate Code section. Modern LNG terminals, for example, have had to undergo extensive seismic reviews. These facilities contain equipment covered by Section VIII and B31.3 of the Code. Neither of these Code section give any guidance to the designer concerning seismic analysis, other than to require that seismic loads be addressed. In this revision of the chapter by Michael A. Porter, some of these issues will be addressed. In addition, a new section discussing the use of computer software for analysis has been included. This section will address some of the issues associated with different computer codes used for different parts of a plant. In addition, the Code references have been revised to reflect the current (2007) Code provisions.

Chapter 35, the functionality was initially authored by Guy Deboo for the previous two editions. This third edition has been updated by Stephen R. Gosselin who revised the discussions pertaining to and operability criteria, which address evaluations for operating plant systems, structures, or components (SSCs) found to be degraded, nonconforming, or subjected to unanalyzed conditions during nuclear plant operation. This revision discusses the methodologies and acceptance criteria applicable to these evaluations. Gosselin introduces typical SSCs that may require operability assessments and functionality evaluations and discuss methods and assessments, failure modes, functionality and operability, and as-built conditions divergent from design. He covers, with the aid of figures, tables, and references, Code requirements as well as short-term and long-term operability acceptance criteria for valves, pumps, snubbers, piping, reactor vessels, tanks, heat exchangers and supports (including component standard and linear supports as well as spring hangers), structural bolts, concrete expansion anchors, and integral welded attachments.

The current practice involves a process of consensus among the regulator viewpoints; plant-specific Technical Specification (TS) requirements; and applicable Codes, Standards, rules, and other licensing-basis compliance requirements. Guy discusses the role of related agencies and committees, such as the U.S. NRC, the ASME Code Committees, and the ASME O&M Code Committees. This chapter includes basic concepts, definitions, evaluation methods, and acceptance criteria from these docu-ments. In this chapter, the scope of SSCs is limited to mechanical systems and their components and supporting structures. Authors discuss the role of the CFR-facility TSs as they relate to the topics of this chapter. Authors provide examples of specified safety func-tions, operating conditions, and events to be considered for some SSCs and piping. The discussions elucidate the often complex, and sometimes nonuniform nature of operability concepts and criteria.

Chapter 36, authored by Frederick J. Moody, covers fluids. Frederick is a recognized authority in the field of fluids; he explains briefly the force predictions from fluid phenomena and behavior, which are significant in the formulations and criteria employed in the ASME B&PV Code. He focuses on the nature of fluid forces imposed on vessels and piping systems from the standpoint of designers, who require guidelines for both the design and operational recommendations of fluid-transport systems. Fluid forces are the result of pressure and shear phenomena and are caused by the energy transfer at pumps or turbines or by disturbances arising from sources such as valve operation, pipe rupture, vapor-void collapse, and the motion of the frame to which the system in question is anchored. A fluid disturbance generally occurs over a pre-dictable time period, and the fluid may respond simultaneously (bulk-flow response) or in a propagation sense (waterhammer response), depending on the system geometry. The nature of bulk-flow and waterhammer responses is significantly different, making it essential to identify the specific fluid response before calculating the resulting forces.

With appropriate equations and references, Frederick covers the basic formulations resulting in fluid forces. He describes the nature of such fluid forces as hydrostatic forces and pressure, as well as shear forces from fluid motion. Predictions made of fluid forces of concern in boiler and piping design are from pressure and shear within internal-flow systems. However, fluid forces are of significant concern in external-flow systems as well, where structures are sub-merged in fluids. Frederick discusses disturbance sources, including motor- or manual-operated valves, safety-relief valves, check-valve closures, pipe ruptures, liquid-column impact at area contractions, liquid-column separation, condensation-induced waterhammer, centrifugal pumps, pipe movements, positive-displacement pumps, gas cushions, and vortex shedding. Frederick also addresses bulk-flow and propagative-flow modeling in pipes, the estimation of fluid-flow forces in pipes, and fluid forces, such as acceleration and standard drag forces, on submerged structures.

Chapter 37 has been revised for this edition by David E. Olson who authored this chapter for the previous two editions. David E. Olson discusses pipe vibration testing and qualification. The discussions center on how piping vibration is typically monitored, quantified and qualified in the power industry. The methods presented for addressing piping vibration are in compliance with the relevant industry codes, standards and regulations, including the 2007 editions of the ASME B&PV Code, the B31.1 Power Piping Code and relevant NRC NUREG's and Regulatory Guides. The chapter addresses the cause and effects of both steady state (e.g. flow induced vibration) and dynamic transient (e.g. water hammer). The development of testing acceptance criteria are discussed along with data acquisition and reduction techniques. Also discussed are vibration prevention and control techniques along with problem resolution examples. The methods presented in this chapter comply with the requirements of ASME O&M-3 "Vibration Testing of Piping Systems".

In Chapter 38, Everett C. Rodabaugh discusses stress intensification factors, stress indices, and flexibility factors with the aid of equations, references, and tables. Everett, a well-recognized authority on the aforementioned topics, notes that piping systems tend to be rather complex structures that include straight pipe and a variety of complex components, such as elbows and tees. A typical piping system might include about 50 components along with many intervening lengths of straight pipe. Each component is subjected to a complex set of loadings. The evaluation of any one component by the detailed analysis methods prescribed in Subsection NB-3200 is an onerous task. The complexity of analyses of piping components and the "standard" aspect of piping components has led to use of stress intensification factors (also called *i*-factors), stress indices, and flexibility factors for evaluations of piping systems. In this chapter, the general concepts behind the development of *i*-factors, stress indices, and flexibility factors are discussed briefly, with references to details of developments.

Everett discusses strain control and stress control as design considerations. Chapter 38 is replete with scholastic discussions and references to substantiate the use of indices. The discussions are provided for the nominal design margin and tests to support these. Stress intensification factors are discussed for girth butt welds; *C* and *K* stress indices are discussed for internal-pressure loading, elbows, and butt-welding tees; moment loading; and thermal-gradient loadings (including branch connections). Fatigue evaluations, ASME Class 2 or 3 piping, and Class 1 Codes for straight pipe, elbows, and seismic analysis are also addressed. Everett provides examples to cover piping systems; moments; Code equations; girth butt welds; elbows; branch connections for Class 2 or 3 piping with branches; best estimates for Class 1 piping; and ASME B31.1 and B31.3 Codes. These examples illustrate how *i*-factors and stress indices are used to check Code compliance and, for a branch connection, to illustrate the quantitative significance of flexibility factors.

In Chapter 39, William J. O'Donnell, whose consulting services have covered all aspects of fatigue evaluations, covers Code design and evaluation for cyclic loading in Sections III and VIII. The author notes that fatigue "is recognized as one of the most frequent causes of failure in pressure vessels and piping components ... for fatigue strength is sensitive to design details, such as stress raisers, and to a myriad of material and fabrication factors, including welding imperfections. Fatigue is also sensitive to such unforeseen operating conditions as flow-induced vibrations, high-cycle thermal mixing, thermal striations, and environmental effects. What is somewhat surprising is the number of fatigue failures that are directly related to poorly chosen design and fabrication details." The ASME B&PV Code was one of the first Codes and Standards to treat the design for fatigue explicitly.

Dr. O'Donnell covers the historical background of fatigue failures. He notes that Section III was the first to include fatigue in its 1963 Edition; Section VIII, Division 2 (*Alternate Rules for Pressure Vessels*). Section VIII, Division 1 (*Rules for Construction of Pressure Vessels)* still does not include explicit fatigue design life-evaluation methods. Fatigue in pressure vessels and piping is of considerable importance. They frequently operate in the low-cycle regime where local stresses are far in excess of yield. This chapter covers the use of strain-controlled fatigue data; stress-strain concentration effects; the effects of mean stress; fatigue failure data; the procedure for fatigue evaluation; cumulative damage; exemption from fatigue analyses; experimental verification of design fatigue curves; and fatigue data for pressure vessel alloys.

More than half of this updated Chapter is devoted to Current and Future Code fatigue design evaluation developments. High temperature water environmental effects are shown to be very important, and the available data is presented in numerous plots. New fatigue design curves proposed by the ASME Code Subgroup on Fatigue are included. Environmental fatigue is expected to be a major Code issue for decades.

Fig. 39.18 presents a new fatigue design curve for austenitic stainless steels in air, revised from the 2006 Second Edition of the *Companion Guide to the ASME Boiler and Pressure Vessel Code*. Figure 39.40 is the corresponding fatigue design curves for austenitic stainless steels in reactor water, also revised from the Second Edition of the *Companion Guide.*

The environmental fatigue design curves in the Second Edition of the *Companion Guide* are independent of temperature. The temperature dependence of reactor water environmental effects on fatigue degradation have recently been determined to be quite significant for carbon, low alloy and stainless steels. Their effects decrease with decreasing temperature below 350ºC (662ºF). Methods of taking credit for such temperature effects have been developed and are included in this new Third Edition of the Guidelines. Section 39.15 describes these new temperature corrections and Figs. 39.41 and 39.42 show their comparison with data for austenitic stainless steels and A333-Gr. 6 carbon steels, respectively. The Chapter includes a comprehensive bibliography.

In Chapter 40, William J. Koves addresses the design of bolted-flange joints and connections, perhaps one of the crucial safety aspects of the power and petrochemical industries because entire piping systems and components are ultimately held together by connections and welded joints. Addressed in this chapter are flanged joints, which are essential and complex components in nearly all pressurized systems. Many factors determine the successful design and operation of a flange joint in service. William notes that the bolted-flange joint involves the interaction between the bolting, flange, and gasket, with important nonlinear variables such as friction and gasket properties considered. The Code design rules provide a method for sizing the flange and bolts to be structurally adequate for the specified design conditions; however, these rules do not address assembly or special requirements, nor do they guarantee a leak-tight joint for all transient-operating conditions.

The purpose of Chapter 40 is to provide the background and basis for the bolted-flange joint design rules contained in the ASME Codes in addition to a discussion of how the rules are applied. Considering this topic's interaction with several ASME Code Sections and Piping and related Codes, William, who has several years of experience serving in ASME Code Committees, translates his professional and Code experience in the writing of this chapter by providing cross-references to the various Codes and Standards. He discusses Codes that address flange-joint design, including design requirements and applicability, and he discusses flange standards as well as flange design for pressure vessels and piping. He provides an in-depth discussion of flange-stress design methods, including the ASME design methodology; the historical background and technical basis; the scope and design philosophy; and the flange types. He comments on flange designs not addressed by ASME: design for external loads, leak-tightness-based design, and flange joint assembly, including ASME Appendix S, bolting and gasket considerations.

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Chapter 41, authored by Hardayal Mehta, presents a review of the applications of many and sometimes unique ways in which the provisions of Section III and Section XI of the ASME Boiler & Pressure Vessel Code have been used in addressing the service induced degradation in the BWR vessels, internals and pressure boundary piping. The vessel internals addressed included steam dryer, shroud and jet pumps. The vessel components considered were feedwater nozzle, stub tube welds, attachment and shroud support welds. Review of pressure boundary piping flaw evaluation methods also included consideration of weld overlay repairs. The service related degradation mechanisms considered were environmental fatigue crack initiation and growth, and stress corrosion cracking. The most form of service-induced cracking in the stainless steel and Ni-Cr-Fe components in the BWR pressure boundary is typically due to intergranular stress corrosion cracking (IGSCC).

Dr. Mehta has included extensive bibliographical references from his own publications, research journals, related EPRI, ASME, and other international publications. Accompanying tables, several figures and examples are used for supporting the detailed discussions regarding several topics he touched upon including BWR Internals, Pressure Vessel, Reactor Pressure Boundary Piping and Crack Initiation & Growth relationships and Plant Monitoring. In the discussions of BWR Internals the inspection, evaluation and repair methods are covered. Dr. Mehta has included in his discussions about BWR Pressure Vessel topics such as the application of probabilistic fracture mechanics for inspection exemption, and low upper shelf energy evaluations. Discussing about Reactor Pressure Boundary Piping, Dr. Mehta expounds the causes of cracking, and remedial/mitigation/repair measures. Fatigue initiation, Relationships for Fatigue and Stress Corrosion Cracking and Crack Growth Monitoring are covered in the last part of this chapter. The updated chapter includes consideration of developments pertaining to the 2007 ASME Code, the NRC actions and EPRI reports.

Chapter 42, initially authored by Drs. Kenneth Chang and Pao-Tsin Kuo for the second edition is currently updated by Robert Kaiwaha Hsu. This provides a detailed description of the NRC's license renewal process and its guidance documents developed for ensuring a uniform format and technical content of a license renewal application and a consistent review of the application by the NRC staff. The authors also provide a clear synopsis of the NRC's technical requirements for license renewal of current operating licenses. In addition, this chapter provides a summary of the emerging issues at the present time and the NRC's interim staff guidance process to capture early any lessons learned from operating experience and/or past license renewal application reviews.

Authors address the License Renewal and Aging Management. The U.S. Nuclear Regulatory Commission (NRC) license renewal process establishes the technical and administrative requirements for renewal of operating nuclear power plant licenses. Reactor operating licenses were originally issued for 40 years and are allowed to be renewed for up to an additional 20 years. The review process for license renewal applications (LRAs) provides continued assurance that the level of safety provided by an applicant's current licensing basis is maintained for the period of extended operation. The license renewal review focuses on passive, long-lived structures and components of the plant that are subject to the effects of aging. The applicant must demonstrate that programs are in place to manage those aging effects. The review also verifies that analyses that are based on the current operating term have been evaluated and shown to be valid for the period of extended operation. As of July 1, 2005, the NRC has renewed the licenses for 33 reactors licenses. Applications to renew the licenses of 15 additional reactors are under review. If the applications currently under review are approved, approximately 50 percent of the licensed operating reactors will have extended their life span by up to 20 years. As license renewal is voluntary, the decision to seek license renewal and the timing of the application is made by the licensee. However, the NRC expects that, over time, essentially all U.S. operating reactors will request license renewal.

Authors recognize the growing interest in License Renewal and mention that to meet this demand, the NRC has established a streamlined process for reviewing applications in a consistent and timely manner. Likewise, the Nuclear Energy Institute (NEI) has developed guidance for the industry on how to prepare an application for renewal. This chapter describes a number of license renewal guidance documents that have been developed to describe interrelated aspects of preparing and reviewing license renewal applications: Standard Review Plan for License Renewal (SRP-LR), Generic Aging Lessons Learned (GALL) Report, Regulatory Guide for License Renewal (RG-LR) which endorses NEI's industry guideline for implementing the requirements of 10 CFR Part 54—The License Renewal Rule (NEI 95-10). The objective of the chapter is to provide back-ground information on the development of these documents and to briefly explain the intended use of the guidance documents singularly and in combination—to facilitate the renewal process from application development to NRC staff review including its audits and inspections of on-site supporting technical documentation. This chapter also provides a brief description of the NEI's environmental review process for license renewal.

Chapter 43, authored by Timothy J. Griesbach, covers *PWR Reactor Vessel Integrity* and the ASME Boiler and Pressure Vessel Code. The authors's objective is to provide an overview of the codes and regulations for prevention of brittle fracture of reactor pressure vessels. The background and bases for the original Section III, Appendix G Code requirements are discussed along with a description of the recent improvements that have been implemented in the Code in Section XI, Appendix G using more up-to-date technology. The changes and improvements are detailed such as the method for determining stress intensity factors, structural factors to account for uncertainties in the analytical methods, and material reference toughness curves. While the Code has incorporated these technical changes, the philosophy of protecting the vessel against brittle fracture has remained the same. The chapter also discusses ongoing efforts to incorporate the Master Curve approach for vessel toughness into the ASME Code, it considers areas for future improvements in the Code method for brittle fracture prevention of PWR reactor vessels, and it summarizes the aging management of PWR reactor vessel internals.

Jeffrey Gorman, Steve Hunt, Pete Riccardella authored Chapter 44 for the previous edition that has been updated by Pete Riccardella and Glenn White for this edition. They have considerable expertise and experience in handling PWR Reactor Vessel Alloy 600 and related issues confronted by the industry. Considering the extreme importance of this topic the authors have covered concerns pertinent to several ramifications of the problem.

Primary water stress corrosion cracking (PWSCC) of Alloy 600 nickel-chromium-iron base metal and related Alloy 82/132/182 weld metal has become an increasing concern to commercial pressurized water nuclear power plants. Cracks and leaks have been discovered in Alloy 600/82/182 materials at a number of locations in PWR reactor vessels and other reactor coolant loop components worldwide. These locations include control rod drive mechanism (CRDM) nozzles, bottom head instrument nozzles, reactor vessel nozzle butt welds, and pressurizer nozzle welds. The consequences of PWSCC have been significant including numerous leaks, many cracked nozzles and welds, expensive inspections, more than 60 reactor vessel heads replaced, and extensive repair and mitigation activities on reactor coolant loop butt welds. A number of plants experienced months-long outage extensions to repair leaks, and one plant was down for over two years as a result of regulatory action following the detection of extensive corrosion to the vessel head resulting from a leaking CRDM nozzle. This chapter addresses Alloy 600/82/182 material locations in reactor vessels, operating experience, causes of PWSCC, inspection methods and findings, safety considerations, degradation predictions, repair methods, remedial measures, and strategic planning to address PWSCC at the lowest possible net present value cost. Recent industry and ASME Code activities to address these concerns are also discussed.

Chapter 45, authored by Sidney A. Bernsen, Fredric A. Simonen, Kenneth R. Balkey, Raymond A. West and Ralph S. Hill III, traces the development of nuclear power plant probabilistic risk assessment (PRA) from its initial evolution as a means for evaluation of public safety through the recognition of its use to identify important safety concerns. This chapter ultimately addresses the PRA's use in

Codes and Standards through the 2007 Edition of the Boiler and Pressure Vessel (BPV) Code and its associated Nuclear Code Cases in helping to determine risk importance and the appropriate allocation of resources, and inservice activities under Section XI of the BPV Code.

It discusses the current status of related Codes and Standards that provide rules and guidance for the development of the PRA and the risk analysis needed to support nuclear power plant applications. The chapter also addresses several specific activities in place or underway to risk-inform the Operation and Maintenance (OM) Code requirements for inservice testing and Code design rules being developed under Section III of the BPV Code. The authors were all active participants in the development and implementation of risk-informed methods for ASME Codes and Standards.

Chapter 46, authored by Hardayal Mehta and Sampath Ranganath, recognized authorities on the Elastic-Plastic Fracture Mechanics (EPFM), are providing in this chapter a review of EPFM applications in ASME Section XI Code. The early ASME Section XI flaw evaluation procedures have been typically based on LEFM. Early progress in the development of EPFM methodology is first reviewed. A key element in the application of EPFM to flaw evaluation is the estimation of the fracture parameter J-Integral. Therefore, the applied J-Integral estimation methods developed by EPRI/GE are first reviewed. Basics of the J-T stability evaluation are then discussed. The first application of EPFM methodology to flaw evaluation of austenitic piping welds is discussed. The extension of EPFM techniques to flaw evaluations in ferritic piping is then covered. Technical background and evolution of Section XI Code Cases (N-463, N-494) and non-mandatory Appendices (C and H) related to pipe flaw evaluation is then provided. Another EPFM based pipe flaw evaluation procedure using the so-called DPFAD approach is also covered.

Drs. Mehta and Ranganath then describe the application of EPFM methods to the flaw evaluations of reactor pressure vessel. An early application has been the evaluation of RPVs with projected upper shelf energy less than that required by 10CFR50. The technical background of Section XI Code Case N-512 and non-mandatory Appendix K is provided. Finally, a proposed Code Case currently under consideration by appropriate Section XI Working Groups, is discussed in detail that would permit the use of EPFM methodology for RPV flaw evaluations per IWB-3610. The updated chapter considers the developments up to 2007 ASME Code as they relate to EPFM flaw evaluation methods discussed.

The authors have included extensive bibliographical references from their own publications, research publications, international journals and related EPRI and ASME publications.

Chapter 47 has been updated with Anne Chaudouet as the lead for this revision with other authors Peter Hanmore and Guido Karcher from the previous editions continuing to be co-authors for this edition, as well. The authors, Francis Osweiller, Peter Hanmore and Guido Karcher all have considerable experience of the US pressure equipment market as well as that in Europe. They have provided a background to the methodology and objectives of the pressure equipment directive and CE marking in general before attempting to portray the detail. The directive is a document of only 55 pages, yet it is applicable to all equipment that can operate at a pressure greater than 0.5 bar and is a mandatory requirement for all pressure equipment to be put into service in the European Union. In common with other European directives the pressure equipment directive specifies general safety objectives which the manufacturer must meet and this leaves considerable scope for interpretation.

Having accumulated considerable experience in the implementation of the directive since its application in November 1999, the authors have provided details of how to design and build pressure equipment to meet the European requirements and thus permit its CE marking and its free movement throughout Europe. After describing the system used to categorize pressure equipment and the conformity assessment requirements that are linked to them the authors go on to explain the routes that can be followed to meet the directive including an explanation of “harmonized stan-dards” and “Notified Bodies”. Each of the essential safety requirements relating to design, manufacture and testing is dis-cussed and guidance provided to assist manufacturers to comply, thus providing the potential exporter to Europe with a wealth of valuable information.

A special attention has been given in this revision to material aspects and to the use of ASME Codes with PED. The areas of compliance of the new Section VIII Division 2 with PED are also highlighted. The authors close with a brief look into a crystal ball and consider the possibility of these “performance based” requirements becoming the basis of global trade.

Chapter 48, authored by Wolf Reinhardt, Nick van den Brekel and Douglas Rodgers, introduces the Canadian pressure vessel standards and explains their relationship to the ASME Code. Distinct features of the Canadian approach to standard development are discussed. The chapter provides the reader with an outline of the Canadian standards and an appreciation of some of their important characteristics. Generally, the Canadian pressure vessel standards adopt the applicable ASME Code sections as the base documents, and then supplement these rules for specific applications as needed. Beyond the ASME Code, the Canadian non-nuclear pressure vessels and piping standard addresses mostly registration and classification of components, and the require-ments for specialized systems such as liquefied natural and petroleum gas systems. The Canadian nuclear construction and in-service inspection standards are based on the corresponding Sections of the ASME Code, but many unique features of the Canadian CANDU (heavy-water) reactor design and licensing basis necessitate additional or modified requirements. In this edition authors added valuable text and graphics to address the ‘Future CANDU Developments’ in the Advanced CANDU Reactor technology as an evolution of the CANDU® 6 reactor with improvements to deliver enhanced safety margins, lower capital and operating costs, improved maintenance and high operating performance. They provide a detailed discussion about the distinctive Canadian nuclear seismic standards. A typical CANDU reactor design is illustrated to allow readers to appreci-ate the background of these rules. Generation III and Generation IV CANDU designs are outlined. A brief outlook to expected future developments in Canadian standards in the international context concludes the chapter.

Chapter 49 was authored by Francis Osweiller, Alain Bonnefoy, Jean-Marie Grandemange, Gerard Perraudin and Bernard Pitrou in the second edition to address “French Codes Dealing with Pressure Equipment”. In the current edition authors have updated the Chapter to be consistent with the current ASME Codes in USA as well as the French Boiler and Pressure Vessel Codes. In France two important organizations SNCT (Pressure Vessel and Piping Manufacturer’s Association) and AFCEN (French Association for Design, Construction and In-Service Inspection) are responsible for the implementation of pressure equipment Codes. SNCT develops Codes such as CODAP for Unfired Pressure Vessels, CODETI for Industrial Piping and COVAP for Steam Boilers and Superheated Water Boilers, with Thermal Fluid Boilers to be included in 2005. AFCEN develops Codes for die nuclear sector namely

RCCM and RCCMR. Chapter 49 deals with pressure equipment covered by the above Codes that deal particularly with the case of a boiler falling within the scope of the PED in which case it shall be considered as an "assembly" i.e. "several pieces of pressure equipment assembled by a Manufacturer to constitute an integrated and functional whole". In Chapter 49 Francis Osweiller assembles contributions of four experts conversant with these Codes, explains the outlines of the organizations and development of these Codes. CODAP is covered by Gerrarand.Perraudin, Barnard.Pitrou covers CODETI, Alain. Bonnefoy discusses COVAP and RCCM is addressed by Jean-.Marie Grandemange.

Originally published in 1943 and updated and republished eight times, the CODAP has been fully revised in 2000 by the French organization of Pressure Vessel and Piping Manufacturers in order to comply with the new European regulation (Pressure Equipment Directive 97/23 EC). The Code is composed of the following Sections: Generals, Materials, Design, Fabrication and Testing and Inspection. The last Section, Testing and Inspection, covers also the task concerning Assessment of conformity to the PED when applicable. The different rules of the 2000 edition are related to the concept of Construction Category which appeared in the 1980 edition. This concept enables the construction quality of a vessel to be adapted and consistent within its future working condition. In this chapter CODAP scope is first detailed both for application in compliance with the PED and for application in accordance with other regulations. Main requirements relating to Materials, Design, Fabrication, Testing and Inspection are presented and significant differences widi ASME VIII Division 1 or 2 are outlined.

CODETI that applies to Industrial Piping i.e., piping intended for industrial plant and covers the same scope as ASME B31-1 and B31.3. CODAP and CODETI are based on the concept of "construction category", which enables the construction quality of a piping to be adapted and consistent with its future working conditions. Originally published in 1974 and updated and republished four times ('79, '82, '91, '95), CODETI has been entirely revised in 2001 by SNCT (French Pressure Equipment Manufacturer's Association) in order to comply with the new European regulation (Pressure Equipment Directive 97/23 EC). Originally CODETI had two sections, the first covering low and medium pressures (P 25 bar; T 350°C), the second covering high pressures. This structure based on early European developments in the field of piping was replaced by the concept of Construction Category which enables the construction quality of a piping to be adapted and consistent with its future working conditions. This concept has been maintained for the 2001 edition. Division 1 applies to industrial metallic piping (i.e. intended for an industrial plant) above ground, ducted or buried. Division 2 and 3 will cover pipeline transportation and water transportation and steel penstock respectively. Scope of Division 1 is detailed both for application in compliance with the PED and for application in accordance with other regulations. Main requirements relating to Materials, Design, Fabrication and Installation, Testing and Inspection are presented. Relevant significant differences with ASME B31.1 and B31.3 are provided.

COVAP applies to steam boilers, super-heated water boilers and thermal fluid boilers and covers the scope as ASME Section I. This code covers all the pressure equipment, which can be assembled by a manufacturer to constitute an integrated and functional whole. The rules of this Code have been established first in order to cover equipment, which shall meet the requirements of the PED, but also to be used where other regulations shall be applied. This Code published by the French organization of Pressure Vessel and Piping Manufacturers is based on the French Standard Serie NF E 32-100 which was withdrawn when the new European regulation (Pressure Equipment Directive 97/23 EC) came in force. Then main requirements relating to Materials, Design, Fabrication and Installation, Testing and Inspection as well as those for Water Quality are presented and significant differences with ASME Section I and Section VIII provided where relevant.

RCCM addresses Inspection rules for Nuclear Island Components and safety related pressure equipment. As indicated in 49.3, the RCC-M was initially based on the ASME III design rules and the French industrial experience. Procurement, manufacturing, and examination practices have since evolved according to the evolution of European and international standards. Design rules applicable to class 1 and 2 components have been updated to comply with applicable regulations and in order to take account of service experience. Less specific work was dedicated to class 3 components, and it is anticipated that more and more reference will be done to applicable non-nuclear industrial standards, and more particularly European harmonized standards, in the near future, as agreed in particular for application to the Finland project. For this reason, the discussions are more particularly dedicated to class 1 and 2 pressure components, with additional information being provided for specific components, such as reactor pressure vessel internals, supports and storage tanks. Additional comments are given in 49.7.10 on construction rules applicable to fast breeder reactor components and in-service surveillance of pressurized water reactor equipment.

Chapter 49 provides the basic philosophy of the Codes and discusses with the help of several tables and graphics General rules, Materials, Design (including flexibility analysis), Fabrication and Installation rules, Testing and Inspection. The authors also discuss their link with the Pressure Equipment Directive or other regulations in addition to a comparison with the relevant ASME Codes. The authors conclude with futuristic ideas and the chapter is replete with pertinent references.

Chapter 50, authored by Kunio Hasegawa, Toshio Isomura, Yoshinori Kajimura, the late Yasuhide Asada and the late Toshiki Karasawa, deals with recent developments of Japanese Codes and Standards for boiler and pressure vessels. This Chapter is revised based on the Second Edition (2006).

Authors provide a brief review of historical background that includes a discussion of the Japan Industrial Standards (JIS) and endorsement of Japan Society of Mechanical Engineers (JSME) Boiler and Pressure Vessel (B&PV) codes. Authors describe the current situation of B&PV Code, including the developments of codes and standards by private sectors such as JSME and its relation to regulation. As a sample of governmental regulation, they cover the situation of High Pressure Gas Safety Law (HPGSL) and Japan Industrial Standard JIS. The authors provide a summary of HPGSL, Designated Equipment Inspection Regulation (DEIR) and Detailed Technical Criteria. In addition, new fitness-for-service rules for pressure vessels published by High Pressure Institute (HPI) are explained briefly.

As part of new trends and deregulation, codes and standards activities of the JSME are discussed covering code and standards for thermal power, nuclear power and fusion power plant components including materials, design and construction, inspection, welding, wall thinning managements, and fitness-for-service. The Fundamental Policy and Organization of the Codes are addressed with the help of several tables and figures. Furthermore, recent trends such as technical developments, upgrading of codes and standards, modification of seismic design standards for nuclear piping, and wall thinning pipes under seismic events are described.

Chapter 51 has been authored by David H. Nash and deals with UK Rules for Unfired Pressure Vessels. The author provides a brief introduction to the UK scene tracing the background and origins of the current Code PD5500 published in 2006 with 2008 updates and its relationship to the European Standard EN 13445 for Unfired Pressure Vessels and the Pressure Equipment Directive (PED) addressed in Chapter 47 of this publication. Dr. Nash points out several issues of PED that have a bear-ing on PD5500. The author thereafter discusses in detail each of the salient design items contained within PD5500. The role of materi-als and design strength for low and high temperature application is included and with the aid of several graphics he discusses the design aspects for shells under internal and external pressure, and buckling related issues. In addition, other vessel component items such as nozzle reinforcements, bolted flanged joints, flat plates and covers, jacketed vessels are also presented. The author also addresses briefly rules for welded joints and inspection and testing. The procedures for design of vessel supports, attachments and other local loading problems are covered and the author discusses the ramifications of these approach especially where failure by fatigue is a possibility. The UK approach for fatigue design is also included and comparisons with the 2007 edition of the ASME codes are made. Design by Analysis is mentioned by the author again cross referencing to the efforts of ASME Codes. Finally a brief overview of the new European code EN13445 is given, this in the light of the PED; Dr. Nash, with the help graphs and tables discusses the various parts and Sections that comprise this Code. Dr. Nash provides exhaustive list of references cited in this chapter.

Chapter 52, authored by Ronald S. Hafner, covers the historical *Development of U.S. Regulations for the Transportation of Radioactive Materials*. The discussion in this Chapter is a highly condensed version of the information presented previously in Chapter 52 of the Second Edition of the *Companion Guide to the ASME Boiler & Pressure Vessel Code*. This Chapter offers a high-level overview of the information presented previously, including all of the appropriate references. Primarily based on the requirements for Type B quantities of radioactive material, the information present-ed in this chapter includes a number of citations that describe the detailed interactions that have taken place between a variety of U.S. governmental agencies, commissions, and departments, such as, the Department of Transportation and its predecessor, the Interstate Commerce Commission, the Department of Energy and its predecessors, the Energy Resource and Development Agency and the Atomic Energy Commission, and the Nuclear Regulatory Commission, etc. The information presented also includes numer-ous citations from the interactions that have also taken place between these governmental agencies and the International Atomic Energy Agency.

From a regulatory perspective, the information presented covers the time period from 1965 through 2004, or about 40 years. Starting in 1978, however, the information also begins to look into the interactions that have taken place between the *regulatory requirements* for transportation packages specified in 10 CFR 71 and the *regulatory guidance* for transportation packages provided by the NRC in the form of Reg. Guides, NUREGs, and NUREG/CRs. As is shown throughout the chapter, the regulatory guidance provided by the NRC specifically notes that the regulatory requirements of 10 CFR 71 can be met using the additional requirements defined in specific sections of the ASME's Boiler & Vessel Pressure Code as a metric for the design, fabrication, assembly, testing, use, and maintenance of packagings used for the transport of Type B quantities of radioactive materials.

Presented in a chronological format, the information provided in this chapter clearly shows how the current system was developed.

Chapter 53, initially authored by Mahendra D. Rana, Stanley Staniszewski and Stephen V. Voorhees provide a Description of Rules of ASME Section XII covering Transport Tank Code of the 2004 edition. This chapter was revised by Mahendra D. Rana and Stanley Staniszewski to incorporate the latest Code changes in the 2007 edition. The first edition of ASME Section XII Transport Tank Code was published in July 2004. This newly developed Code provides rules for construction and continued service of pressure vessels used in transportation of dangerous goods via highway, rail, air or water. The authors provide an overview of Section XII while covering specific topics such as the scope and general requirements, materials and design, fabrication, inspection and testing requirements. The need for a pressure vessel code dealing with the whole spectrum of tanks to transport dangerous goods was a result of the review of USDOT (U.S. Department of Transportation) regulations. The regulations had become cumbersome to use, and in a global market without compromising safety the need to make the rules for transport tanks acceptable internationally became urgent. Hence the inaugural edition of ASME's Section XII focus was Portable Tanks. The subcommittee prepared the Code to be transparent with existing ASME Code requirements such as Section VIII, Div.1, while including the existing DOT requirements that impacted the scope of the charter to prepare the Section XII Code.

This chapter had been coordinated by Mahendra Rana with the help of experts covering topics in their respective fields. Stan Staniszewski dealt with the scope and general requirements of the Code including rules on pressure relief devices, stamping, marking certification, reports and records. The scope of the new Code applies to pressure vessels 450L and above, including additional components and criteria addressed in Modal Appendices that are to be used along with applicable regulations and laws. Steve Voorhees initially handled the sections on fabrication, inspection and testing requirements of Code Section XII. These sections have been further revised by Mahendra Rana to incorporate the 2007 Code changes. From the perspective of fabrication and inspection, Section XII is a mixture of familiar and new concepts to the Boiler and Pressure Vessel Code. Mahendra Rana covered the sections on materials and design rules. The coverage included Design Conditions and Allowable Stresses, Design Temperatures, Design and Allowable Working Pressures, Loadings, Design of Formed Heads, Torispherical Heads, External Pressure Design, Flat Heads and Covers, Openings and Reinforcements, Design of Welded Joints, and Articles covering Portable Cryogenic Tanks including Materials and Design. The rules for fatigue design are also given in the article covering Portable Cryogenic Tanks.

Chapter 54 on Pipeline Integrity and Security Pipelines had initially been authored by Alan Murray for the second edition, and coauthored by Alan Murray and Rafael Mora in this current third edition. The authors note that pipelines are an economic and comparatively safe means of transporting hydrocarbons and many other fluids over great distances. Incidents, while relatively rare, can have serious consequences, so protecting the public, and the environment, is paramount. This is achieved through a combination of regulatory oversight, prudent pipeline integrity management and the use of appropriate technology. Regulatory requirements vary greatly throughout the world and are reflective of an underlying safety philosophy, ranging from the prescriptive approach adopted in the United States to the performance-based methods favored in the United Kingdom. These various approaches are compared before addressing how they, and accompanying codes and standards, are used to formulate the essential elements needed in a sound integrity management plan. The finite resources available for maintaining a pipeline means that, a risk-based approach must be followed, so as to prudently allocate expenditures. Accordingly both qualitative and quantitative methods of assigning risk are described. The four main methods of undertaking integrity assessments are outlined and a detailed approach to assessing defects is provided including several worked examples.

A number of relatively simple means of protecting a pipeline asset are available, ranging from coatings and cathodic protection to preventing third-party damage. These are described, before addressing a relatively new threat stemming from willful targeting a pipeline operation either through physical or cyber attack.

Chapter 55, authored by Anibal L. Taboas for the second edition has been completely revised for the current third edition. In this edition, author presents an editorial view of Decommissioning Technology Development within a context that energy, environment, education, and economy are inextricably linked. The author highlights related approaches by the US Department of Energy and the UK Nuclear Decommissioning Authority. The causes and contributing factors to significantly increased cost estimates are discussed, as well as the cost of delay (missed regulatory opportunities, reluctance to cut back non-productive expenditures, a bias against innovation, and lack of incentives for transition to other significant missions after the completion of environmental remediation).

A compendium of technology challenges and needs is presented, along with an overall program performance rating. The author observes that (1) funding for basic, long, and medium term components, has dwindled as to support primarily paper [or desk] studies, (2) fiduciary responsibility requires demonstrating whether the resources consumed by industry and government in the safe and compliant environmental remediation of radiological and nuclear facilities remains commensurate with the risks averted, and that (3) reverting the funding trend in D&D technology development requires integrated planning, strategic action, and effective communication.

The chapter concludes with a call for a plan to (1) develop the business case for investment; (2) invest in high visibility projects of demonstrably high return for investment; (3) meet customer-identified expectations; (4) track leading indicators and contractual incentives; (5) resolve key policy issues; and (6) transparent independent peer review.

Drs. K.P. (Kris) Singh and Tony Williams collaborate in Chapter 56 to present a comprehensive assay of the backend of the commercial nuclear power cycle. The management of the spent nuclear fuel removed from the reactor after a period of power generation in the reactor core by nuclear fission has been described as the *Achilles heel* of the commercial nuclear industry and the source of much of the disparate political opposition to its use, despite its evidently sterling credentials as a non-polluting and commercially viable alternative to fossil power.

The perceived undesirability of the spent nuclear fuel derives from the transmutation of uranium into an array of isotopes (known as actinides and fission products) that produce copious quantities of radiation for thousands of years after the fuel has been removed from the reactor. Although the rate of dose accretion gradually attenuates with the passage of time, a spent nuclear fuel assembly remains a highly radioactive material for millennia. The technologies developed to manage this unavoidable byproduct of commercial nuclear power generation are discussed in this chapter with a critical assessment of their strengths and weaknesses. For this purpose, spent fuel management technologies are divided into (1) reprocessing, which consists of reclaiming the fissionable portion of the spent fuel for reuse as an energy source and (2) passive storage in either deep pools or in an inert gas environment.

The essential characteristics of the reprocessing technology, namely, the PUREX process, are described in the context of its historical origins and its dependence on chemical separation techniques since the very beginning in the 1940s. The authors explain how the continuance of reprocessing in Europe and its abandonment in the U.S. because of proliferation concerns over thirty years ago led to profound differences in fuel management paths taken by the U.S. and oversea nuclear operators.

Williams and Singh provide a concise description of the wet storage technology that advanced in the U.S. in the wake of the ban on reprocessing in the U.S, but has remained a largely untapped option in those countries that rely on reprocessing and/or dry storage. The evolution of dry storage technologies in the U.S. (ventilated systems) and overseas (metal casks) is also discussed with respect to their technical attributes, safety, reliability, and maintainability. In particular, the role of the ASME Codes in providing a sound platform for the mechanical design and stress analysis of the systems, components, and structures used in wet and dry storage technologies is explained.

The special demands on the used fuel transport packages imposed by the regulations of the USNRC and the guidelines of IAEA to ensure safety in fuel transportation are explained along with the latest developments in the field. Finally, the authors also provide a succinct summary of the methodology to analyze the effect of a postulated aircraft crash on a storage or transport cask to deal with what is an unmistakably unique design consideration in the twenty-first century. (The authors wish to recognize the valuable contribution of Dr. David McGinnes in the preparation of Chapter 56.)

In Chapter 57 Generation III+ PWRS has been addressed in three distinct parts: in Part A: AP1000 by John T. Land, in Part B: EPR by Marty Parece and in Part C: U.S. APWR by Masahiko Kaneda. From the commissioning of the first commercial nuclear reactor more than 50 years ago, the nuclear power industry has been developing and improving reactor technology with particular emphasis on reliability and safety. There are several generations of reactors that have been developed or are being developed. These reactors are generally categorized as Generation I, II, III, III+, and IV reactors. The authors illustrate with the help of schematics the development and the technology distilled from 50 years of successful nuclear operating experience that has led to the Generation III+ pressurized water reactors (PWRs).

Generation I reactors were developed in the 1950s and 1960s, Generation II reactors were developed in the 1970s through the 1990s, and Generation III reactors were developed in the 1990s and 2000s. Generation III reactors are considered to be evolutionary reactors such as the System 80+, and advanced pressurized water reactor (APWR). Generation III+ reactors are based on the nomenclature from the Department of Energy, that is, Generation III reactors with improved economics and safety. Generation IV reactors are new technologies that are being developed for future reactors.

The Generation III+ PWR reactors discussed in this chapter have design features with more robust design improvements, higher availability and longer operating life, extended fuel life, and improved and innovative safety features over the currently operating reactors. The Generation III+ Boiling Water Reactors (BWRs) are discussed in Chapter 58.

The scope of this commentary is to describe in some detail the Generation III+ PWR plant design features, technology, safety and reliability features, and the elimination or mitigation of degradation issues associated with Generation I, and II PWR designs. This chapter provides commentary on the following Generation III+ PWRs that have received Design Certification approval or are in the process of receiving certification approval from the U.S. Nuclear Regulatory Commission (NRC): AP1000, EPR and U.S. APWR. Authors have with the help of illustrations, graphs, charts and figures provided in addition to the historical background and futuristic scope covered the current practices covering each of the three topics of the Generation III+ PWRs.

The coverage for each of these three parts are: In Section 57A the coverage of AP1000 included AP1000 Plant Design, AP1000 Operational Technology, Safety Features, Containment Design, Modularization and Construction, Operation and Maintenance, ASME Code Aspects, Future Direction, References and Nomenclature; In Section 57B discussions regarding EPR covered EPR Development, EPR Plant Design, EPR Safety, Containment Design, Construction, Operation and Maintenance, Code Aspects, Building Now and References were addressed; In Section 57C coverage of U.S. APWR included U.S. APWR Plant Design, Operational Technology, Safety Features, Containment Design, Modularization and Construction, operation and Maintenance, ASME Code Aspects, Future Direction on the U.S. APWR, References and Nomenclature.

Chapter 58 authored by Hardayal Mehta and Daniel Pappone, provides details of the development of boiling water reactor (BWR) based nuclear steam supply system (NSSS) and the role of ASME Code in its design, material selection, fabrication and in-service inspections. A general background of the development of the BWR product line is first provided including the current offerings (the Advanced Boiler Water Reactor, ABWR, and the Economic Simplified Boiling Water Reactor, ESBWR). This includes the description of the reactor and reactor system design, safety system design and the containment design. The authors next describe the key features of the ESBWR including the natural circulation design, operating domain and passive safety features. The ESBWR core and containment cooling systems represent a radical departure from those of the earlier BWR product lines in that the cooling systems are passive and do not rely on electrically driven pumps. The ASME Code aspects covered include the ASME Code versions used in the construction, treatment of environmental fatigue issues, material selection, and others. Future directions in terms of fabrications, modularization, and others are lastly discussed.

Chapter 59, authored by William J. O'Donnell and Donald S. Griffin, describes the structural integrity issues in Section II, VIII, III, and Subsection NH (Class 1 Components in Elevated Temperature Service), and Code Cases that must be covered to support the licensing of High Temperature Generation IV Reactors. It also describes how the Code addresses these issues, and the need for additional criteria to cover unresolved structural issues for very high temperature reactors.

Since the 1980s, the ASME Code has made numerous improvements in elevated temperature structural integrity design criteria. These advances have been incorporated into Subsection NH of Section III of the Code. The current need for designs for very high temperature and for GEN IV systems requires the extension of operating temperature from about 1400ºF (760ºC) to about 1742ºF (950ºF) where creep effects limit structural integrity, safe allowable operating conditions, and design life.

Materials with more creep and corrosion resistance are needed for these higher operating temperatures. Material models are required for cyclic design analyses. Allowable strains, creep fatigue and creep rupture interaction evaluation methods are needed to provide assurance of structural integrity for very high temperature applications. These criteria intended to prevent through-wall cracking and leaking.

The detailed material properties needed for cyclic finite element creep design analyses are generally not provided in the Code. Chapter 59 describes the material models, design criteria and analysis methods which NRC has indicated are remaining needs in the ASME Code to cover Regulatory Issues for Very High Temperature Service:

1. Material cyclic creep behavior, creep-rupture, creep-fatigue interaction and environmental effects.
2. The structural integrity of welds.
3. The development of extended simplified analysis methods (to avoid dependence on "black box" finite element analyses (FEA) for cyclic creep).
4. Test verification of 1, 2 and 3.

Chapter 60 authored by Reino Virolainen and Kaisa Simola cover Risk-Informed Licensing, Regulation and Safety Management of NPPS in Finland. The authors discuss the four operating nuclear power plant units in Finland. The TVO power company has two 840 MWe BWR units supplied by Asea-Atom at the Olkiluoto site. The FORTUM corporation (formerly IVO) has two 500 MWe VVER 440/213 units at the Loviisa site. All the units were commissioned between 1977 and 1982. In addition a 1600 MWe European Pressurized Water Reactor (EPR) supplied by the Framatome ANP—Siemens Consortium is under construction at the Olkiluoto site. Current international safety requirements and especially French and German operating experience have been used in the design. Finnish requirements and operating experience have also been used, especially regarding site-specific features. Severe accident management and protection against collision of a large passenger airplane are implemented in the plant design.

In Finland, risk-informed applications are formally integrated in the regulatory process of NPPs that are already in the early design phase, run through the construction and operation phases through the entire plant service-time-living PRA models have been developed for both the Olkiluoto 1/2 and Loviisa 1/2 NPPs. The PRA studies include level 1 and level 2 models. Level 1 comprises the calculation of severe core damage frequency (probability per year) and level 2 the determination of the size and frequency of the release of radioactive substances to the environment. At the moment, level 1 studies for full power operation cover internal events, area events (fires, floods), and external events such as harsh weather conditions, and seismic events. The shutdown and low power states of level 1 PRA cover internal events, floods, fires, harsh weather conditions and seismic events. Special attention is devoted to the use of various risk informed PRA applications in the licensing of Olkiluoto 3 project, such as RI-ISI, RI-TechSpecs, RI-IST and safety classification of SSCs. In this context this chapter makes several references to the ASME standards on RI-ISI and the European Network for Inspection and Qualification, ENIQ and its RI-ISI related activities. The chapter is has several tables and schematics in addition to references and acronyms to explain the terminology used in this chapter.

The scope of the coverage includes Risk- Informed Regulatory Frame with discussions about PRA in Nuclear Safety Legislation and Risk Informed Regulation Policy sequences. The authors also dwell on PRA IN THE REGULATORY PROCESS and address Development of PRA Requirements for Nuclear Power Plants, and PRA in the Licensing Process of New Designs with tables to support their presentation. Risk-Informed Applications for a Construction License, for Operating License, Risk- Informed Applications during operation control and Review Process of PRA also discussed with the help of schematics, examples and experiences of Olkiluoto 3 NPP Risk Informed Licensing along with EPR safety features. The authors dwell on the European orientation to risk and cover European network for inspection and qualification, ENIQ task group risk, ENIQ documents supporting the RI-ISI framework document, and ENIQ recommended practices. As a part of the extension of risk-informed activities authors dwell on analysis of oil spills, and probabilistic fire simulation, risk-informed regulatory inspections. The authors close the chapter with summary and conclusions.

Chapter 61, authored by Dr. Luc H. Geraets, introduces Belgium as an important actor in the applications of nuclear energy. The author provides a short historical summary of the development of nuclear power in Belgium. He explains the choice made by Belgium to follow the USNRC rules for the construction of its nuclear units, and details how the design and safety analysis of these units have been done by applying the US rules and all the associated documentation (regulatory guides, standard review plans, ASME Code, IEEE standards, ANSI, ANS, etc.). The practical transposition of the ASME Code to the Belgian environment is then presented; in particular, the use of Section XI for repairs and replacements is analyzed in full detail.

This system has proven its workability and efficiency. However, in 2003, Belgium voted a nuclear phase-out law, which provides for abandoning the use of fissile nuclear energy for industrial electricity production; until new legislation happens, there will be no reason to question the rules that would be imposed for design and construction. If it happens, and new plants are built in Belgium, it is likely that the same philosophy as for the currently operating plants would be applied, with the selection of a “fresher” version of ASME Code Section III (and the other Sections called upon by ASME III) and Section VIII Div. 1. Transpositions would be revised, but the general framework and the actors would remain the same.

In Chapter 62, authored by Dieter Kreckel an overview of the Codes and Standards for Pressure Equipment to be applied in Germany is presented. Dieter Kreckel provides an overview of the historical perspective of Pressure Equipment Directive (PED). The Pressure Equipment Directive (PED) (97/23/EC) was adopted by the European Parliament and the European Council in May 1997. Initially it came into force on November 29, 1999, and from May 29, 2002 the Pressure Equipment Directive was obligatory throughout the European Union. Germany as a member of the European Union had to respect the Directives issued by the EU and to transfer these to national law, so the PED came into effect.

Dieter Kreckel discusses several aspects of the pressure equipment directive including series of technical harmonization directives for machinery, electrical equipment, medical devices, simple pressure vessels, gas appliances, and so on. The Directive concerns manufacturers of items such as vessels pressurized storage containers, heat exchangers, steam generators, boilers, industrial piping, safety devices, and pressure accessories. Such pressure equipment is widely used in the process industries (oil and gas, chemical, pharmaceutical, plastics and rubber, and the food and beverage industry), high temperature process industry (glass, paper, and board), energy production utilities, heating, air-conditioning, and gas storage and transportation.

Dieter Kreckel also discusses exclusions from the scope of the PED such as Article 1 that has *Items specifically designed for nuclear use, failure of which may cause an emission of radioactivity.* For Nuclear Power Plant Licensing in Europe the application limits of the PED and the Nuclear Codes have to be agreed with the National Nuclear Licensing Authorities, there is no harmonization process agreed in the European Union. The German Nuclear Power Plants in operation are licensed by the German Codes and Standards for Pressure Vessels effective from the construction time as defined in the operation license. These identify the potentials for the application of the Codes and Standard, as the state of the art. The effort spent in Germany for the development of the new Generation 3 of NPP, for example, the European Pressurized Water Reactor (EPR) or the SWR1000 with respect to the code and standard evolvement is introduced in the AREVA NP engineering process.

Carlos Cueto-Felgueroso discusses, in Chapter 63, pressure equipment regulations, codes, and standards in Spain, in the nonnuclear industry as well as in the nuclear field. In both cases, emphasis is placed on periodic inspections and testing. The basic Spanish regulation on pressurized equipment in the nonnuclear industry may be found in the Regulation on Pressurized Apparatus, published by the Ministry of Industry and Energy in 1979. The regulation consists of a set of general standards and leaves the specifics to a set of Complementary Technical Instructions. After Spain joined the European Community in 1986, a process of modification of the Pressure Equipment Regulation began, with a view to bring it in line with those of the other Member States to facilitate the trade of goods and services within the European Union.

Carlos Cueto-Felgueroso discusses the implications in the process of European harmonization and the Pressure Equipment Directives issued by the European Parliament and Council that became obligatory, regarding the design, manufacture, testing, and conformity assessment of pressure equipment and assemblies of pressure equipment. In the nuclear field, in the absence of a national regulation, the codes and standards of the countries of origin of the design of each reactor are applied. The Spanish nuclear fleet is currently made up of seven pressurized water reactors (PWR) and boiling water reactors (BWR) of U.S. design and one German designed PWR.

Carlos Cueto-Felgueroso’s association with ASME Code Committees is reflected in his discussions pertaining to the several Sections of the ASME B&PV Code. He mentions the application of Section III of the ASME Code in the design and construction of the Spanish nuclear power plants, except in the case of the German designed PWR, for which the KTA rules were used. He indicates that on the other hand, the rules of Section XI of the ASME Code are applied to all the plants for In-Service Inspection (ISI).

The author discusses Spanish Regulation in the Nonnuclear Industry that pertains to Design and Construction, including design, fabrication, and conformity assessment of pressure equipment that is currently regulated in Spain according to PED. The main provisions of the PED are summarized with reference to Spain. He discusses the basic requirements regarding the inspection and testing, including regulation on Pressurised Apparatus and its Complementary Technical Instructions, with particular reference to boilers, economizers, water preheaters, steam reheaters and piping for fluids. The author shows his expertise relating to oil refineries and petrochemical plants. He discusses regulations covering cryogenic tanks and thermal power generation plants using solid, liquid or gaseous fossil fuels. The presentation includes inspection and testing requirements for fossil fuel power generation plants, and oil refineries and petrochemical plants.

Carlos Cueto-Felgueroso's familiarity with ASME Codes and Standards, Section XI is evident from a detailed discussion about Nuclear Industry as it relates to Qualification of NDT for ISI and accounts the U.S. developments regarding Risk-Informed ISI. The author mentions the role of the Nuclear Regulator (CSN) and UNESA (Spanish Utilities Group) and an increasing interest in possible optimization of the ISI programs. There is also a discussion of the international Programme for the Inspection of Steel Components (PISC). In addition, a description of the Spanish NDE Qualification Methodology is described with a mention of the objectives, scope, principles of qualification, functions, and responsibilities of the parties.

The chapter has several tables, graphics and references, including ASME Code Cases, US NUREGs, NRC and PED publications used in the chapter.

In Chapter 64 Dr. Milan Brumovsky discusses the Czech and Slovakian Codes with respect to the Nuclear Power Plants (NPPs) Jaslovske Bohunice (440 MW) in Slovakia, Dukovany (440 MW) and Temelín (1000 MW) in the Czech Republic (both in former Czechoslovakia). Dr. Brumovsky mentions the agreement between the former Czechoslovakia and Soviet Union in context of mutual cooperation in building NPPs. The author traces the Government decisions regarding an extended project of the technical standard documentation of NPPs organized by the International Economic Association "Interatomenergo" in Moscow. The association was set up to cooperate in the field of nuclear power between individual member states of the Council of Mutual Economical Co-operation (CMEA).. The entire complex of technical standard documentation ended in 1990, when GAEN finished the whole project at international level and consequently also in the Soviet Union.

Dr. Brumovsky mentions that the fundamental problem of the project was a question of legal obligation to CMEA standards. Elaboration of obligatory position of state regulatory bodies among the members of the CMEA was done. This facilitated in determining the documentation of technical standards in the form of a legal-agreement. From the point of international relations, the procedure could be considered as sufficient; but from the standpoint of Czech NPPs, the effectiveness of utilizing these standards was at zero point, since effective steps were not organized to bring them into action. The CMEA rules resulted in merely upgrading of the Soviet rules and standards incorporated into new set of Soviet rules and standards issued around 1989. These rules and standards existed for service lifetime assessment of reactor components and were limited only to design and manufacturing; in very special cases these rules were for operation also but not from the lifetime evaluation point of view. Thus, assessment of defects, found during in-service inspection, has to be based on acceptance levels valid for manufacturing and on special procedures, prepared by the Nuclear Research Institute (NRI) Rez and manufacturers of components; for case by case application, these had to be accepted by the Czech State Office for Nuclear Safety (SONS).

SONS requirements for Lifetime Evaluation and mentions that in 1993, the SONS initiated a project "Requirements for Lifetime Evaluation of VVER Main Components" (VVER: Water—Water Energetical Reactor is of pressurized water reactor type but designed and manufactured in accordance with former Soviet codes and rules). Within the scope of this project, preparation of regulatory requirements for lifetime evaluation of reactor components, including all aspects of integrity and degrading processes of these components, was performed. Responsibility of this project was given to the NRI Rez, which focused on reactor pressure vessel (RPV) and reactor internals and issued as a SONS document with recommendations that included Operational Safety Reports. In this document, no practical procedure for lifetime evaluation was given; only general and some detailed technical requirements for evaluation of these two components were described.

Dr. Brumovsky discusses the NTD ASI Code for VVER Reactor Components. He mentions that approximately during the same time, a second activity was initiated by the Czech Association of Mechanical Engineers (ASI), which decided that a set of codes for reactor components, namely, Normative Technical Documentation (NTD) was needed for Czech nuclear industry. A plan for preparation of such codes was discussed, accepted, and put into action, details of which are presented in the chapter.

Next is a discussion of the VERLIFE PROCEDURE which is a proposal for the European Union 5th Framework Programmes that was prepared and accepted with the aim to use proposals of the Section IV as the first document to be discussed, changed, upgraded, enlarged, and finally accepted. The main goal of the project was in the preparation, evaluation, and mutual agreement of a "Unified Procedure for Lifetime Assessment of Components and Piping in VVER Type Nuclear Power Plants." The author thereon discusses the COVERS CONTINUATION. In 2005, a new project within the EU 6th Framework Programmes was opened: COVERS—VVER Safety Research that is also coordinated by the NRI. In this project, WP 4 deals with the upgrading and updating of the VERLIFE procedure to assure that the experience obtained as well as new developments will be appropriately included in the new version. Experts from nine countries are taking part in this project, in addition to VVER-operating countries such as Czech Republic, Slovak Republic, Hungary, Finland, Spain, The Netherlands, Germany, Russia, and Ukraine, as well as from EU-JRC IE (Joint Research Center—Institute of Energy in Petten, The Netherlands) and ISTC (Institute for Scientific and Technical Cooperation).

Dr. Brumovsky concludes that The VERLIFE procedure is now fully accepted as a main regulatory document for lifetime assessment of VVER components in the Czech Republic and Slovakia and partially in Hungary and Finland. Negotiations are now in progress for its use in Ukraine and China.

The chapter has information about several manufacturing companies in the Czech Republic, Slovakia that obtained ASME Certification for manufacturing reactor (and also nonreactor components in accordance with ASME Section VIII) components for export to other countries where ASME Codes are required. The author provides References with annotated bibiliography and author's publications pertinent to this chapter.

Dr. Brumovsky provides detailed information about the Structure of NTD ASI. The final version of the VERLIFE procedure in Czech was accepted as a new version of the Section IV of the NTD ASI. Czech SONS accepted NTD ASI Sections I, II, III, and IV in 2005 and recommended them for their use in the chosen safety important components in NPPs. Similarly in the Slovak Republic, Sections I and II, prepared by the Welding Institute of Slovakia in cooperation with the Welding Institute of the Czech Republic were accepted by Slovak Office for Nuclear Regulation. Structure of the Sections I, II, and III is similar to the appropriate Sections of the ASME Code Sections I, II, and III, where as the structure of Sections IV and V is fully new. The author provides detailed comparison of each of the Czech Codes with ASME B&PV Code Sections I, II, III, IV and V.

Chapter 65, co-authored by Peter Trampus and Peter Pal Babics, provides an overview on the recent activity in Hungary concerning comprehensive adaptation of the ASME Code. The owner of Paks NPP, Hungary's sole nuclear generating facility, is aiming at adjusting the ISI and IST program to meet ASME Code requirements. The objective is to achieve an internationally acceptable level in structural and functional integrity assessment of long-lived and passive as well as of active components, and to create the basis for a proper aging management program in the operations period beyond the design life of the units. Apart from this, it will extend the current four-year inspection interval for Class 1 components up to an eight-year one, which will contribute to a more cost-efficient operation and maintenance. The Hungarian nuclear regulatory regime gives an opportunity for this because the nuclear safety rules do not determine explicitly the applicable codes neither for the design nor for the ISI / IST.

The Chapter briefly describes the background of the Paks life extension project and its regulatory aspects. The basic regulatory principles related to ASME adaptation are summarized. The authors focus on aspects of maintaining the current licensing basis as well as on the necessity to demonstrate the compliance with Section III requirements. The substantial part of the work is the design review of Class 1 and 2 components and, if needed, a re-design of selected components to comply with the Section III requirements. As an example for the ongoing design review process, the comparison of Fatigue Strength Reduction Factors for welds in pressure vessels and piping is presented.

Chapter 66 deals with some aspects of Russian Regulation and Codes in nuclear power and is authored by Victor V. Kostarev and Alexander V. Sudakov. Authors with vast domestic and international experience discuss the Code perspective covering Russia, with appropriate comparisons of Codes of the USA, Canada, Japan, China, India and several European countires. Dr. Kostarev's interaction with ASME is in evidence in detailed discussions presented in this chapter with tables and graphics about Code allowables, Seismic regulations and on-going applications of seismic excitation studies.

The authors present a brief history of regulatory activity and Boiler Codes in Russia and continue with a write-up about System and a list of Standards relevant to the State Safety Regulation in nuclear power. Development of Nuclear Codes for design and analysis of NPPS equipment and piping have been presented along with a comparison of Russian nuclear standard PNAE (Rules and Standards in Atomic Energy Industry of Russia), with ASME BPVC (Boiler Pressure Vessel Code) in application to seismic analysis of a primary loop of PWR (VVER) reactor. Documentation of Guidelines for the Seismic Analysis of NPP (Nuclear Power Plant) Systems, requirements for seismic analysis and capacity, Equipment Classification of the PCLS according to different codes including PNAE, ASME, JEAG, PCLS (Primary Coolant Loop System) have been presented.

The authors through several tables, schematics and graphics have compared the Russian Codes with Codes of Japan and Europe. These include definition of stresses and array of materials, allowable of stresses, formulas for Piping Stress Analaysis, definition of Seismic Loads, Seismic and Dynamic Analysis, and results of Comparative Analysis of PLCS by PNAE and ASME BPVC. Authors also used the Finite Element Analysis results to reinforce their presentation.

European high viscous dampers approach in protecting NPP primary and secondary systems from seismic loads and operational vibration are discussed in this chapter. Using bibliographical notations, schmatics and analytical information, the authors discuss Viscoelastic Piping Dampers, also known as High Viscous Dampers (HVD) extensively used since the mid-1980s in the seismic upgrading of nuclear power stations in Europe and currently used in new nuclear power plant installations in China and India. General operational characteristics of HVDs as a dynamic restraint are discussed in this chapter. The authors discuss their expertise with HVD, as a device that works in a softer manner than snubbers do, providing to the system essential additional damping rather than stiffness. High damping in the device is a result of deformation of a special extremely high viscous liquid that is located in the space between damper's piston and housing.

The chapter includes a glossary of several terms used in this chapter and has 61 references from scholastic publications including Dr. Kostarev's own contributions to international conferences.

Chapter 67, co-authored by Malcolm Europa, Paul Brinkhurst, Neil Broom, and John Fletcher, provides an overview of the codes and standards for pressurized equipment as used in the South African nuclear industry. The applicable legislation, regulatory requirements, and the roles of the respective regulatory bodies governing the use of pressurized equipment are discussed. A historical perspective of the construction, licensing, and operational phases of two pressurized water reactor (PWR) units of Westinghouse design but constructed under license from Framatome is given, with emphasis on the design and quality rules used and risk considerations allowed by the licensing framework.

Furthermore a description is given of the pebble bed modular reactor (PBMR) to be constructed for Eskom, the owner and licensee, by PBMR (Pty) Ltd, a largely state-owned and funded nuclear design company. The PBMR is a high-temperature gas-cooled reactor (HTGR) and is one of the next- generation nuclear power plants (NGNP). The PBMR is designed according to the ASME Section III Codes, utilizing light water reactor (LWR) service conditions and materials. This has imposed certain constraints on the design and required innovative design features which are discussed.

In conclusion, the paper discusses the stated intentions of the South African Government in terms of the expansion of the nuclear industry and the implications thereof. It also reviews the changes being made to the regulatory frameworks, the need for change, and the implications with respect to code and standards usage in the industry.

At present, the Nuclear Power Program in India (Chapter 68, by H.S. Kushwaha, K.K. Vaze, and K.B. Dixit) is based mainly on a series of Pressurized Heavy Water Reactors (PHWRs). This chapter first provides a general overview of the Indian PHWR design and its evolution. The design approach, material selection, and fabrication practices are described for major components such as calandria, headers, steam generators, and piping.

In Indian PHWRs, the design, fabrication, testing, and inspection of all mechanical components basically follow the requirements of appropriate sections of the ASME Boiler & Pressure Vessel Code (ASME B&PV Code). In a few cases, where it was not possible to meet the code criteria, it is the intent of the code that is met.

Other international codes used are (1) Canadian Code CAN/CSA N285.4-05 and IAEA Safety Guide 50-SG-02 for ISI and (2) French Code RCC-G for containment design.

Details are provided of the development and the use of leak-before-break (LBB) criterion to eliminate the need for installation of pipe whip restraints. Results of experiments conducted to determine load-carrying capacity of cracked pipes and the results of fatigue crack growth rate tests in support of LBB criteria are discussed. As a further example of the research and development work conducted in India related to nuclear power plant applications, the development of a modified B2 stress index (used in NB-3600-type stress analyses) for pipe elbows and curved pipes and quantification of additional safety factors to account cyclic tearing in LBB assessment are discussed.

Chapter 69, authored by Jong Chull Jo and Howard H. Chung, provides an overview of the Korean nuclear safety regulatory system and codes for design, manufacturing, operation, and maintenance of nuclear boiler and pressure vessels.

Since the 1970s, Korea has been promoting the nuclear energy industry to produce electricity needed for rapidly expanding industry and for enhancing the quality of human life. In the early stages of the introduction of reactors into Korea, due to lack of a well-established domestic regulatory framework for the safety regulation of operating reactors, the technical safety requirements and safety standards of the countries that supplied the reactors were applied. However, after a while, the Korean regulatory authority streamlined the regulatory framework and applied these rules and regulations to domestic nuclear installations. Furthermore, for strengthening the level of safety of nuclear installations Korea has been making every effort to improve the nuclear safety regulatory system and to continuously update the codes and standards, based on the up-to-date knowledge and experience.

This chapter describes Korean nuclear regulatory organizations, aspects of the regulatory authority including licensing, status of nuclear installations, nuclear reactor regulatory framework and regulations referring to domestic and/or international industrial codes and standards in the area of pressure vessels and piping. The chapter also describes the status of Korean Electric Power Industry Codes (KEPIC) that covers standards for design, manufacturing, operation, maintenance, and testing and inspection of nuclear and non-nuclear mechanical components, including pressure vessels and piping. A comparative assessment of U.S. and Korean codes is also addressed. Besides, a comparison between the KEPIC code and its reference to codes of other foreign countries is also provided.

Chapter 70, co-authored by Y. B. Chen, S. Chang, and T. Chow, provides an overview of the nuclear energy application and development in Taiwan, which includes historical background of the development, role of the regulatory authority and current status of nuclear facilities. This chapter also addresses important issues such as seismic design features of the nuclear power plants (NPPs), PWR/BWR pressure boundary integrity, power uprate and license renewal, and radioactive waste management of NPPs in Taiwan.

Taiwan is located at a complex juncture between the Eurasian plate and Philippine Sea plate, where earthquakes occur frequently. Hence, seismic design/qualification of structures, systems and components (SSC) of NPPs in Taiwan is an important issue. Although Taiwan is prone to earthquake strikes, up to now Taiwan's NPPs have never experienced any earthquakes that challenged the seismic design of the plants. However, the disastrous Chi-Chi earthquake prompted Taiwan's nuclear regulatory authority Atomic Energy Council (AEC) to request the largest and only nuclear utility, Taiwan Power Company (TPC), to install the Automatic Seismic Trip System in all six operating nuclear units to further enhance the plant safety.

N-service inspections (ISI) followed the ASME Section XI for the operating units are conducted during each scheduled outage to ensure the integrity of the SSCs. Recently, more attention has been focused on the inter-granular stress corrosion cracking (IGSCC) in BWRs and primary water stress corrosion cracking (PWSCC) in PWRs especially at the dissimilar metal (DM) welds connecting vessel nozzle to austenitic stainless steel piping according to both foreign or domestic operating experiences. More details of the inspection results are discussed in this Chapter.

To improve the performance of the NPPs, a power uprate project has been launched for the NPPs in operation. The power uprate considered is the Measurement Uncertainty Recapture (MUR) type and up to 1.7% rated thermal power increase for license application. Submission of operating license renewal applications for all the operating units is under planning and preparation. Preliminary feasibility study of stretch power uprate (~5%) is also ongoing. Besides the aforementioned activities, radioactive waste management is also an issue receiving more attention. Right now, the application submitted by TPC for construction permit of independent spent fuel storage installations (ISFSI) at Chinshan site is still under review. Before any ISFSI is allowed to be constructed, the on-site spent fuel storage pools are the only available facilities for the spent fuel management of the nuclear power plants.