LCC™

LWR Chemistry and Component Integrity Programme

The LCC9 deliverables.
For information on earlier published LCC Special Topic Reports please see *page 18*.

The annual LCC Programme is focused on reactor coolant, secondary chemistry and RCS material issues and open to nuclear utilities, manufacturers and vendors, research and engineering organisations as well as regulatory agencies. In the LCC8 programme, currently 23 organisations in North America and Europe, representing 112 nuclear plants, are members. This Programme was started in 2004 and has gained large interest.

Deliverables

ANT International will provide the LCC Members with the following in January–February time frame:

- Hardbound copy(-ies) in colour of the Reports. The Reports will be provided as soon as they are printed with the aim of before the Seminar in March.
- Searchable CD-ROM(-s) with the following contents:
  - High-resolution pdf files with the Reports in colour.
    - The files can be copied to a company server, with full read access for everybody with access to the server.
    - The contents from the Reports in pdf-format can be printed.
      Also, the contents from the pdf-files can be copied and pasted electronically into other documents, e.g. Word files.

  The CD-ROM(-s) of the Reports will be provided before the Seminar (see below).
- A Seminar will be held in Europe to present the results of the LCC Programme.
  All the LCC presentation material will be provided to the customers prior to the Seminar. This will enable the customers to print out the presentation material, e.g. in colour with high resolution. The number of full time employees per Member that may attend meetings is limited to eight (8) people per organisation. The language of the LCC Programme will be English.
- The Members of the Programme organisation of LCC will be available for consulting throughout the year. A few telephone or e-mail consultations requiring no additional work are provided at no additional cost to Members.

*"The LCC Seminar has been very enjoyable and informative.
It has broadened my understanding of many topics and has given me ideas/research to explore in the near future."*

JOHN MCGRADY
Rolls-Royce sponsored EngD Student, University of Manchester

LCC Programme Content and Description

Nuclear utilities have to reduce costs for operation, maintenance and fuel, keep the highest level of safety and lowest possible level of radiation exposure to employees and the general public and minimize environmental impact of liquid and solid effluents and wastes. Emphasis on safety, longer fuel cycles, higher burnup of fuel, increased fuel duty with more nucleate boiling in Pressurised Water Reactor (PWR's) and plant power uprates as well as more technical issues like Axial Offset Anomaly (AOA also called Crud Induced Power Shift/CIPS), Stress Corrosion Cracking (SCC) all point to the increased importance of high quality water chemistry and control.

It is our goal that the LCC Programme shall assist the LCC Members in meeting all these water chemistry and material related challenges in a most efficient way. This Programme reviews and evaluates the developments and trends in Light Water Reactor (LWR) primary coolant and secondary side chemistry and also structural materials technology (excluding fuel materials). This is accomplished by identification of relevant information and a discussion of its significance for the Programme. The Programme reviews all relevant information through publications and international conferences and, when necessary, comments and background information are added.

Additional benefits for the LCC Members can be seen in the fact, that the Members gain an increased understanding of power plant water chemistry and material integrity to facilitate more efficient plant operation. Furthermore, the LCC Members can be assisted in the training and education of a new generation of chemistry and material experts in their organisations.

The overall objectives of the LCC Programme are to enable the LCC Member to:

- Gain increased understanding of reactor water chemistry related to a successful plant operation and continued integrity of Reactor Coolant System (RCS) materials while keeping radiation exposure low
- Guiding the plant operators to apply adequate PWR secondary side chemistry for safe, economical and environmentally friendly plant operation with high availability and without significant steam generator degradation or fouling problems or carbon steel Flow Accelerated Corrosion
- Assist in the training and education of a new generation of chemistry and materials experts
- Establish an independent meeting point for experts to enable free and critical discussions and experience exchange

These objectives are met through critical review and evaluation of the most recent data related to reactor water and secondary side chemistry, identification of the most important new information, and discussion of its significance in relation to water chemistry now and in the future.

The evaluations are based on the large amount of non-proprietary data presented at technical meetings and published in the literature.

Listen to Juan de Dios Sánchez Zapata,
C.N.Cofrentes Plant Life Management at Iberdrola.

LCC9 Programme

Two separate Reports and will be prepared within the LCC9 Programme. The topics are:

- Annual Report (AR) on Operational Issues, Practices and Remedies similar to that issued in the LCC7 Programme but addressing new issues
- Special Topic Report (STR) on Effect of Coolant Chemistry on Fuel Performance

At the LCC9 Seminar, the above reports will be presented. However, also a presentation will be given at the LCC9 Seminar on the "Structural Materials Degradation Report (SMDR), rev. 1" which will be furnished within the LCC10 Programme.

The presentation gives engineers and scientists an introduction to the topic of environmentally-assisted degradation of structural materials in water-cooled nuclear reactors.

The presentation intends to focus on the following topics:

- Essentials of the physical metallurgy of relevant structural alloys and the physical chemistry of corrosion
- Advances in understanding of PWSCC of Ni-base alloys in PWRs and approaches to mitigation
- Advances in the understanding of PWR secondary side steam generator tube corrosion
- Advances in stress corrosion cracking and irradiation assisted stress corrosion cracking of stainless steels in PWR primary water
- Advances in modeling flow assisted corrosion
- The cracking observed at dissimilar alloy weldments
- Environmental assisted fatigue of reactor materials in LWR coolant environments
- Reactor experience of mitigation actions for cracking of BWR structural alloys and core internals

The two Reports are described more in the following.

Operational Issues, Practices and Remedies

This LCC9 AR on Operational Issues, Practices and Remedies will be authored by Dr. Francis Nordmann, Mr. François Cattant, Dr. Samson Hettiarachchi, Dr. Robert Cowan and, Mr. Dewey Rochester. Operational Issues, Practices and Remedies are important issues for plant personnel and designers. These issues may require optimization or an exchange of knowledge to improve plant operation in a safe, economical, environmentally sustainable way.
In addition to the help for utilities, it is of interest for manufacturers and regulators in understanding some plant issues and remedies.

This report combines the following subjects of limited extent but potentially important consequences:

- Degradation of the primary coolant barrier together with mechanical remedies,
- The potential benefits of Enriched Boric Acid (EBA),
- Primary coolant (Co-58, colloids) inventory,
- Degradation of concrete structures in NPPs,
- Colloids, Zeta Potential and Activity Transport,
- Electrochemical Corrosion Potential (ECP) measurements and,
- Key points, "lessons learned" and "best practices" of several recent conferences.

Mechanical and design mitigation of various types of corrosion encountered in the PWRs' second containment barrier

Over the past thirty years, corrosion has been the prevalent type of degradation of the PWRs' second containment barrier (primary system boundary) leading to extensive and expensive repairs, component replacements (i.e., SGs) and lost power generation. In order to maintain a high level of safety and keep the plants operating as economically as possible, corrosion mitigation techniques have been developed. Although major efforts have been made by utilities and vendors to apply these techniques in the field, some types of corrosion are still active.

This report describes eight types of corrosion of the second containment barrier which are, or have been active (Stress Corrosion Cracking such as primary water SCC, SCC in polluted environments, SCC of sensitized stainless steels, irradiation assisted SCC, general corrosion (cation release, boric acid corrosion), pitting, erosion-corrosion and atmospheric corrosion). For each of these degradation phenomena, a brief description of the mechanism is provided along with the types and properties of the materials that have been impacted. A list of the typical components of concern is also provided and some field failures of these components are presented.

This report lists more than a dozen corrosion mitigation techniques along with field examples. These examples show that most have been successful in solving one or more of the corrosion issues mentioned above.

Enriched Boric Acid (EBA)

Fuel of higher enrichment, longer fuel cycles or use of MOX, requires increased amounts of neutron poison to maintain reactivity control which means higher content of boron-10 at the beginning of fuel cycles and in some safety injection tanks. The inconveniency of a higher concentration of natural boric acid, with only ~19.6% to 20 % of B-10, may be counteracted by using Enriched Boric Acid, with typically ~40% 10B, which means a twice lower total boric acid concentration.

The report will cover the fuel issues, including use of MOX, the various advantages of using EBA, particularly associated with the possibility of achieving the optimum pHT (close to 7.4 at 300°C) with an acceptable lithium concentration, i.e. (i) lower dose rates, (ii) lower risk of corrosion of Alloy 600 and other nickel base alloys, (iii) lower risk of corrosion of 304 stainless steel. The key issue of impact of EBA on Axial Off-set Anomaly (AOA) and on waste amounts will be discussed. The benefit of using tanks with lower boric acid content will be explained, both for new PWR/VVER units and for avoiding heavy modification in tanks of already operating units. The process for switching from natural boric acid to EBA will be also described.

Finally, the interesting use of high enrichment of EBA (>90%) to recover the B-10 depletion from neutron reaction in operating units will be explained, with very low primary water coolant renewal, thus generating much lower liquid wastes.

"The long experience of the LCC Expert Team provides useful information for 'sunny and cloudy days' of a chemist's job!"

MICHAEL BOLZ
Head of Chemistry Department at NPP Philippsburg

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The Behaviour of Co-58 and nickel in a High Duty Core

Some units in the U.S. have experienced high concentrations of Co-58 and nickel in the reactor coolant during the cycle. The increase in concentrations has occurred at approximately one hundred effective full power days (EFPD) into the cycle. Nickel is of concern in the coolant because it can increase the probability of Axial Off-set Anomaly (AOA). Furthermore it can lead to an increase in the production of Co-58 which is a radiation dose concern.

This report evaluates the occurrence of this phenomenon and investigates the possible factors that may cause it. These factors include steam generator materials, core duty, tubing manufacturing processes, changes in fuel assembly steaming rates, power transients, and chemical additives such as zinc.

Sampling techniques, system dose rates and behaviour of other crud species will also be addressed.

Degradation of Concrete Structures in NPPs – The Problem

Over the past three years increasing attention has been paid by a US Utility and US NRC to understand the potential mechanism of concrete degradation that appeared to have occurred in a US NPP. The degradation involves developing fine cracks in the concrete that may take up to 5 to 15 years to develop. The discovery that was made in 2010 during intense inspection in preparation for application for license extension of a US NPP has not affected continued plant operation, but has generated interest in understanding the mechanism of crack generation with the intent of identifying mitigation methodologies.

The problem surrounds what is known as alkali-silica reaction (ASR), a slow chemical reaction that occurs between alkaline cement and reactive silica found in some aggregates used to make concrete. It has been known that when concrete is exposed to moisture over extended periods of time, ASR develops forming a gel that expands causing micro-cracks that affects concrete properties. Efforts are underway to understand how ASR affects mechanical strength of concrete structures used in NPPs.

This section of the report will summarize factors affecting ASR, provide publicly available information on ASR and potential mitigation methods that can be applied to existing and new structures.

Colloids, Zeta Potential and Activity Transport – general for BWRs and PWRs

Insoluble corrosion products such as simple oxides, mixed oxides and oxyhydroxides present as colloids in aqueous suspensions in nuclear reactor heat transport systems develop surface electrical charges and zeta potentials as a result of hydration and dissociation of surface

"ANT International provides excellent material for education, this supports the very important transfer of knowledge in times when alternation of generation becomes a problem in many nuclear power plants."

BERNT BENGTSSON
Senior Engineer & Chemistry Advisor at Ringhals NPP,
Vattenfall AB, Sweden

Read more

hydroxyl groups depending on the pH of the system. These charged corrosion products then deposit on heat transfer and other surfaces such as flow venturis and flow orifices causing fouling. Such fouling increases the pressure drops across flow venturis causing an adverse effect on nuclear power generating systems affecting the plant power output by as much as 1 to 4%.

Furthermore, deposition of activation corrosion products such as Co-60 on out-of-core surfaces can also affect radiation fields in the NPPs.

This section of the report summarizes the basics of colloid formation, surface charge and zeta potential generation, activity transport and particulate deposition on critical surfaces of NPPs, with some emphasis on how to mitigate the negative effects of particulate deposition.

ECP Monitoring in BWRs and PWRs – Why and How

ECP monitoring has been widely used in BWRs since the early 80’s for optimizing hydrogen addition during the implementation of hydrogen water chemistry (HWC). This type of monitoring has continued in BWRs following the introduction of newer technologies such as NMCA in mid 90’s and On-line NMCA in 2005. Therefore, BWR ECP monitoring is extensive covering more than 50 BWRs with over 400 reactor operating years experience. Furthermore, BWR ECP monitoring has been collectively performed at ten different locations in the BWR heat transport circuit.

On the other hand, relatively limited amount of ECP monitoring has been performed in PWRs to control hydrogen in the primary system to maintain corrosion potential away from the Ni/NiO stability line where maximum PWSCC crack growth rate appear to occur. Some ECP monitoring has also been performed in the secondary side of PWRs to control oxygen and detect potential redox transients. PWR ECP monitoring has also been limited to a few primary and secondary side locations including the feedwater piping.

This section of the report will summarize ECP monitoring in BWRs and PWRs providing details of why and how these measurements have been performed.

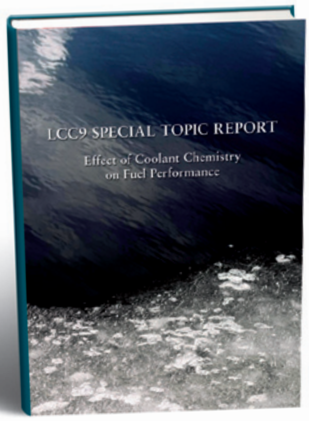
Review of 2013 Radiation Reduction and Source Term Conferences

This summary will address the radiation source term and dose rate “lessons learned” and “best practices” experienced at operating BWRs and PWRs. The emphasis is on understanding individual plant behaviour that can then be adopted by other units. The ANT International experts will evaluate and comment on the applicability of the various findings to other operating BWRs and PWRs. Specifically, the following toipcs will be addressed:

- The technical basis of radiation field buildup and technologies for reducing radiation fields throughout the plant for PWRs and BWRs.
- The characterization of radiation fields including the isotopic contribution to these fields,
- Remote monitoring of radiation fields and advances in dose rate detectors. The use of these technologies to apply them to radiation field reduction will also be discussed.

Summary of the important BWR items from the 16th International Conference on Environmental Degradation of Materials in Nuclear Power Systems

This important conference is held every other year with a focus on the effect of water chemistry, irradiation and material selection on degradation of structural and fuel materials. The emphasis of our reporting will be on the findings leading to better understanding of current issues plus an evaluation of new emerging issues important for new and extended reactor operations.



STR on Effect of Coolant Chemistry on Fuel Performance

The deregulated market means that the nuclear utilities must reduce operating, maintenance and fuel cycle costs to remain competitive. Also reactor safety must be improved while the plant radiation build-up must be limited to cope with the change in political environment towards nuclear power.

To achieve the above mentioned goals the following changes have been and are being introduced:

- Longer cycles. Longer cycles do not necessarily meet any of the goals noted above; they do however reduce licensing needs and save storage space. Two-year cycles start to be marginal in some cases economically.
- Higher discharge burnups
- Modified water chemistries
  - e.g. elevated LiOH in PWRs,
  - increased or decreased hydrogen coolant contents in PWRs,
  - Nobel metal additions in BWRs,
  - Zn-injection in BWRs and PWRs
- Plant power uprates
- More aggressive fuel management methods

Changes in water chemistry may interact with several of the above parameters challenging the corrosion and hydrogen pickup performance of the Zr alloys being used as fuel claddings.

The fuel vendors on their part are developing new fuel designs, including more advanced materials to partly reduce the fuel cladding corrosion and hydrogen pickup rates. However, these alloys may respond differently to the parameters mentioned above. Also, very little corrosion and hydrogen pickup data exist for these new alloys.

The intent of this Report is to provide a state-of-the-art knowledge of the Zr-alloy corrosion mechanisms and the various forms of corrosion features in BWRs and PWRs/VVERs. This knowledge will help the utilities to implement actions to reduce Zr alloy corrosion and hydrogen pickup.

"For new engineers and chemists, this could be a very useful training tool. For experts in a given field, knowledge of experience in other related fields facilitates improvements in their own fields. ANT International plays an important role in fulfilling this need in the nuclear industry through the LCC program."

MS. JAYASHRI N. IYER
Principal Engineer Materials and Fuel Rod Design at Westinghouse Electric Company, USA

Read more

The following aspects are intended to be discussed in this STR:

- Introduction
- Basics of the corrosion and hydrogen pickup process
  - Corrosion as an electrochemical process with the oxide as the electrolyte
  - Mobile species in the oxide and factors controlling them
  - Barrier-layer and non-barrier layer concepts
  - Oxide growth and breakdown processes
  - Hydrogen uptake mechanisms
- Description of different forms of corrosion in BWRs and PWRs/WWERs and how these relate to the corrosion mechanisms
  - Uniform corrosion
  - Accelerated uniform corrosion
  - Nodular corrosion
  - Shadow Corrosion
  - Enhanced Shadow Corrosion
- Key parameter impacting corrosion mechanisms and therefore corrosion rate
  - Temperature
  - Irradiation
  - Water coolant chemistry (the Report focuses on this parameter)
  - Material microstructure (e.g. Second Phase Particles characteristics and their dissolution, solutes, degree of coldwork/recrystallisation)
- Various forms of accelerated corrosion and their correlation to the corrosion mechanisms
  - Chalk River Unidentified Deposits (CRUD) water chemistry induced, including CRUD Induced Localized Corrosion (CILC) failures
  - LiOH concentration in the zirconium oxide
  - Cladding microstructural changes
  - Hydrides in the fuel cladding
- Reporting on PWR/VVER and BWR cases where water chemistry had a major part in the development of fuel failures. Discussions on the reasons for the failures and how they could have been mitigated.
- Discussions, implications and summary

LCC Programme Organisation

The Programme manager of LCC is Peter Rudling, President of ANT International. He will be joined by Dr. Francis Nordmann, formerly at Électricité de France, Dr. Peter Ford, formerly at General Electric, Dr. Wilfried Rühle, formerly at Philippsburg, Germany, Dr. Suat Odar, formerly at AREVA NP GmbH (former Siemens AG, KWU), Dr. Jan Kysela Director at Rez, Dr. Peter Scott, formerly at AREVA, Dr. Pierre Combrade, formerly at AREVA NP, Dr. Robert Cowan, formerly at GE, Dr. Samson Hettiarachchi, formerly at GE Hitachi, Mr. Dewey Rochester, formerly at Duke Energy, Mr. François Cattant, formerly at Électricité de France and Mr. Claude Amzallag, formerly at Électricité de France.

Dr. Francis Nordmann has 40 years of experience in power plant chemistry. He is retired from Électricité de France (the French Utility) in 2007, where he was an international expert in charge of chemistry and corrosion in the corporate offices. He was in charge of managing the engineering studies for the French fleet of 58 PWR units and of several international programmes. His Ph. D degree was obtained at the French Atomic Energy Commission, in connection with the University of Mulhouse in 1973. He also worked for 8 years within the French manufacturer Framatome (now Areva). He has been active for example in the following areas:

- Water Chemistry evolution and studies for the various systems (primary coolant, secondary steam-water system, condenser cooling systems, and intermediate circuits)
- Developing the Chemistry Specifications for the French NPP and some others
- Interface with Manufacturers and Regulatory Body
- Chemistry and corrosion Training
- Steam Generator blowdown and condensate polishers strategy
- Optimisation of secondary water chemistry for various objectives
- Steam Generator experience feedback and relation with chemistry
- International projects with various countries and organisations
- Organising Committee of several International Conferences on Chemistry for Nuclear Reactors

Dr. Peter Ford received his bachelors and doctoral degrees from Cambridge University (UK) and a masters degree from RPI (USA), with an emphasis on metallurgical engineering and corrosion science. Initially he "learned the trade" in the 1960s when he was as an engineering apprentice for a turbine manufacturer and an alloy developer for the Olin Corporation in the USA. Since then he has been associated with the nuclear power industry, with a focus on developing mitigation actions for various modes of environmentally-assisted degradation of structural materials. He worked initially as manager of the corrosion group at the research laboratory of the Central Electricity Generating Board (UK) and then for 23 years with the General Electric Corporate Research and Development Center (GE-CRD) (Now the GE Global Research Center) where he was manager of the Corrosion Mitigation and Coatings Laboratory. This laboratory interacted closely with General Electric Nuclear Energy and their customers in the USA, Asia and Europe resolving issues with Boiling Water Reactors. Seminal contributions were made in the areas of;

- Choice of structural materials for current and future reactors to minimize environmentally-assisted degradation
- Optimized fuel cladding compositions to counteract nodular oxidation

- Radioactivity build-up control via zinc injection
- Life prediction codes for environmentally-assisted cracking of materials both in and out of the reactor core. This mechanistically-based methodology provided the fundamental basis for water chemistry specifications for BWRs used world-wide
- Water chemistry optimization methods for mitigating environmentally-assisted cracking, including "Hydrogen Water Chemistry" and "Noble Metal Technology"
- Underwater repair and cladding techniques

Upon retirement from GE in 2000, he served at the US Nuclear Regulatory Commission for 4 years as a member of the Advisory Committee for Reactor Safeguards addressing issues ranging from materials degradation, new reactor design certification, license renewal, etc. for all light water reactor designs. He became associated with ANT-International in 2005. Dr. Ford is active in various societies and international cooperative groups in the field of nuclear materials degradation, including consultancies with reactor vendors, utilities, universities and national labs, etc. He has authored or co-authored 100 papers, handbooks and patents and is a Fellow and recipient the Willis Rodney Whitney Award from NACE-International for "outstanding contributions to the science of corrosion", and specifically predicting environmentally assisted cracking in BWRs.

Dr. Wilfried Rühle has been working in nuclear power industry for about 35 years. His degree as Ph.D. in chemistry, physical chemistry and radiation biology he has got from the faculty of natural sciences at Heidelberg University, Germany. His thesis on a subarea of nuclear fuel reprocessing he has made in the Institute for Radiochemistry at the Nuclear Research Centre in Karlsruhe, Germany. With a background in chemistry, radio chemistry and radiation biology he joined the German energy supplier EnBW. There he was in charge of the chemistry department for two Nuclear Power Plants (NPP), one BWR and one PWR, working within the power plant organization resulted in widespread practical and operational experience in general plant operation. Experience in water chemistry of both light water reactor types, BWR and PWR, broadened his knowledge in the interaction of metals and water.

Special items of interest and experience are:

- Corrosion under different water conditions or water treatments
- Radioactivity build-up
- Water processing techniques
- Water radiolysis and behaviour of radiolysis gases in water and steam
- Operational chemicals especially diesel fuel and lubricants
- Plant operation with enriched B-10

For many years he was member/ chairman of VGB working groups dealing with BWR and PWR chemistry topics and several times be worked for IAEA as an OSART member.

Dr. Suat Odar has 40 years of experience in power plant chemistry. He is retired from AREVA NP GmbH (Former Siemens and KWU) in February 2008, where he has held since mid of eighties various service and managerial positions for power plant chemistry. In the last seven years he was responsible for the water chemistry of the nuclear power plants in his company. His degree as Ph.D. in Physical Inorganic Chemistry was obtained from the Technical University of Darmstadt, Germany, in 1970. He has been active for example in the following areas:

- Water Radiolysis and Post LOCA Hydrogen Control in PWR Containment
- Commissioning of PWR plants

- Developing Chemistry Control concepts for PWRs
- Water Chemistry Guidelines
- Consulting in Power Plant Operation (Chemistry part)
- Improvement of Steam Generator Performance
- Man-Rem-Reduction
- Plant Life Extension (Chemistry measures)
- Steam Generator Chemical Cleaning
- Plant Chemistry Training Programmes
- Secondary Side System Design & Material Review to improve Steam Generator Performance

Dr. Jan Kysela studied at the Faculty of Inorganic Chemistry at The Institute of Chemical Technology, Prague (ICT), where he graduated from the Department of Nuclear Chemistry with an MSc. in 1967. He received his Doctors diploma (PhD) at the Faculty of Water Technology Institute of Chemical Technology, Prague. He was first employed as a chemist on the LR-0 research reactor at Nuclear Research Institute Rez. He was responsible for heavy water coolant technology – its analysis, cleaning and overall management. At that time, the Czech heavy water nuclear power plant research programme was also under development, and he participated in a number of projects related to heavy water chemistry. After the heavy water programme ended and the light water programme started, he began working on using boric acid to manage VVER reactor reactivity. Dr. Kysela became a head of Irradiation Experiments Department in 1980. He coordinated work and was responsible for start-up of reactor high pressure water loop RVS-3. He carried up extensive research programme including steam generator tube stress corrosion cracking, tests of electromagnetic filters. He was project manager of development programme of test facilities on high pressure loop such as electrically heated model of fuel rods at in-pile and at out-of-pile test section, corrosion equipment for corrosion fatigue, electro-chemical measurement at high pressure and high temperature and continuous neutron measurement of boric acid concentration. At 1986 Dr. Kysela initiated a three years programme and cooperation with NPP Rheinsberg in Germany for primary water chemistry optimisation – comparison of standard, high pH and hydrazine water chemistry. Programme finished by hydrazine water chemistry at of NAP Rheinsberg. More specifically he has managed within years 1990–2010 the following international projects:

- VGB programme on IASCC tests irradiation experiments of structural materials (RPV and internals) both non-irradiated and pre-irradiated
- HITACHI Programme on BWR material irradiation testing
- NUPEC-MHI Zinc injection irradiation programme
- JAPEIC-MHI Programme on RPV Material Specimens Irradiation
- TVEL Corrosion tests of new Zr alloys
- Experimental water chemistry research programme for Czech and Slovakian NPP

He was responsible for reactor operation and experimental and irradiation programmes. Managing of reactor utilisation for different purposes – including reactor loops, irradiation channels and radioisotope production. Management of five experimental departments and laboratories - reactor operation, irradiation projects, reactor physics, mechanical workshop and analytical chemistry. Recently Dr.Kysela made use of his past experience to also prepare facilities and an experimental programme for GIV (generation four) reactors, specifically reactors cooled with supercritical water or helium and experimental programme on fusion technology research in blanket module system for tritium production and energy conversion. He works today as the Director of the Reactor Services Division at the Nuclear Research Institute at Rez in the Czech Republic. He has been with the company for 40 years.

Dr. Peter Scott received his B.Sc. in chemistry from the University of Sheffield in England in 1965 and then his Ph.D. in physical chemistry from the same university in 1968. He spent two years as a Post Doctoral Fellow in the Department of Applied Chemistry of the National Research Council of Canada before starting his career in the nuclear industry in 1971 in the Materials Development Division at the Harwell Laboratory of the UKAEA. During 19 years at Harwell he became a section head and a recognized expert in corrosion fatigue and stress corrosion cracking of materials in thermal and fast reactor systems. He joined the Framatome Group (now AREVA NP) in 1989 where he was an Expert Consultant for 18 years in corrosion and stress corrosion of materials, mainly in PWRs, until his retirement at the end of January 2008. During the 37+ years he has worked in the nuclear power business, his interests and experience in materials selection, analysis of field cracking events, life prediction, preparation of technical reports and presentations to utility clients and Safety Authorities relating to general and localized corrosion, stress corrosion and corrosion fatigue have been in the following areas:

- Low alloy pressure vessel steels, steam generator shells and secondary piping of nuclear power stations and other non-nuclear structures
- Nickel base alloys used for steam generator tubes and various dissimilar metal junctions between low alloy steels and stainless steels, mainly in the primary circuits of PWRs
- Stainless steels of primary piping and highly irradiated internal PWR core support structures
- Assistance in resolving corrosion problems in the structural components of PWR fuel assemblies
- High strength alloys used for springs, bolts and valve stems

He was also a member of the editorial board of the NACE Corrosion Journal for over 8 years. He is the author or co-author of over 100 scientific publications and in 2000 received the F. N. Speller Award from the NACE for outstanding contributions to the practice of corrosion engineering.

Dr. Pierre Combrade received his first degree from the Ecole Nationale Supérieure des Mines de Paris, in 1967 and earned his doctorate degrees with a thesis on solidification of refractory eutectic alloys for aero engine turbine blades in 1972. He has been involved in corrosion problems for the nuclear industry since 1972. For 22 years he worked partly on corrosion problems in light water reactors and partly on stress corrosion and localized corrosion of high grade stainless steels for the chemical process industry. He joined Framatome (now AREVA NP) in 1994 as head of the corrosion department of the Technical (R&D) Centre and subsequently became an "AREVA International Expert". He retired from AREVA in 2007 and is now active as a consultant, mainly in the fields of corrosion of materials for light water reactors and underground storage of radioactive wastes. Dr. Pierre Combrade is also the author or co-author of over 50 technical publications and several reviews as well as book chapters on stress corrosion cracking and crevice corrosion. He is co-author of a book of metallurgy published in 1997 and re-edited in 2002. He has also been involved in teaching activities in the Ecole des Mines de Paris, in the Ecole des Mines de Saint Etienne and the INSTN (National Institute for Nuclear Science and Techniques), as well as directing several thesis students working on SCC and fretting-corrosion problems.

Dr Robert Cowan, retired Chief Technologist of General Electric's Nuclear Energy Division, with over 40 years of Boiling Water Reactor chemistry, corrosion, structural material and fuel material experience. He holds a Ph.D. in Metallurgical Engineering from The Ohio State University in Columbus, Ohio. At GE he headed the development activities which qualified and commercialized various new technologies for boiling water reactor (BWR) application including: IGSCC resistant grades of stainless steel, hydrogen water chemistry, zinc injection (utilizing both natural and isotopically depleted zinc), noble metal technology and in-reactor ECP monitoring.

- Over 70 Technical Publications
- 20 Issued U.S. Patents
- Consultant to both US and European BWR utilities as well as EPRI
- Contributed heavily to the development and subsequent revisions of the EPRI BWR Water Chemistry Guidelines and NMCA Application Guidelines

Dr. Samson Hettiarachchi has 33 years of experience as a college lecturer, researcher, innovator and a technologist. He has held a variety of technical positions at GE Nuclear Energy as Chief Engineer/Physical Sciences, Chief Technologist/Chemistry, Engineering Fellow and Principal Engineer prior to his retirement from GE in February 2011. Prior to joining GE, he held the position of Electrochemist/Senior Electrochemist at SRI International (formerly Stanford Research Institute) and the position of Lecturer/Senior Lecturer in Chemistry at the University of Colombo, Sri Lanka. Two of his innovations at GE Nuclear Energy, NobleChemTM and On-Line NobleChem are widely used in the US, Japan, Spain and Switzerland to extend the life of Boiling Water Nuclear Reactors. Dr. Hettiarachchi's research experiences include Physical Chemistry, Electrochemistry, Surface Chemistry, Catalysis, Corrosion and Mitigation of materials, Battery Technology, Sensor Technology, and In-situ Generation of Nano-particles. He has worked in the nuclear power industry related work for about 25 years. His specific experiences in the nuclear industry include, Development of ECP sensors, ECP monitoring, High Temperature pH Measurements, Zeta Potential Measurements, HWC Benchmark Tests, Water Chemistry Guidelines, Dose Reduction, Fuel Corrosion, SCC Mitigation, NobleChem Applications, On-Line NobleChem Applications, Inspection Relief Criteria Development, and Plant Chemistry/Materials Education and Training. For many years he has participated in EPRI BWRVIP Mitigation Committee Meetings and several IAEA Meetings. He has over 100 publications in International Journals and International Conference Proceedings and holds 27 issued US patents. He has been a peer reviewer for the Corrosion Journal and the Journal of Nuclear Science and Engineering.

Mr. Dewey Rochester retired from Duke Energy Carolinas LLC in June 2010 after working for thirty six years in the field of nuclear power plant chemistry. He began his career in May 1974 at Duke's Oconee Nuclear Station as a Junior Chemist. He was promoted to site Chemistry Manager in February 1978. In September 1984 he transferred to the corporate office to lead the process qualification programme for the steam generator chemical cleaning at Oconee. From 1989 until 2003 he worked on a variety of projects dealing with primary and secondary water chemistry, and steam generator corrosion issues as well as performing assessments of plant chemistry performance. In February 2003 he was promoted to Duke Energy's Corporate Nuclear Chemistry Manager, where he led the group responsible for the development of the site chemistry programmes at Duke's three nuclear sites. During his career he has worked in all

phases of nuclear power plant chemistry including makeup water production, primary and secondary chemistry, radwaste processing and steam generator chemical cleaning and corrosion. He has authored and co-authored several papers on steam generator chemical cleaning, radwaste processing, steam generator corrosion issues, and the use of dispersants to mitigate steam generator deposition. He also made numerous presentations at various conferences and seminars.
Some of his interests include:

- Chemical cleaning processes and corrosion monitoring
- Core design impact on corrosion product releases
- The use of dispersants for steam generator deposition mitigation
- Post accident sampling systems
- Steam generator corrosion issues
- Primary and secondary water chemistry guidelines
- Zinc addition to mitigate plant dose rates

He served twice as Chairman of the Babcock & Wilcox Steam Generators Owner's Group, Chairman of the EPRI Chemistry Subcommittee and a member EPRI Steam Generator Owner's Group Technical Support Subcommittee and Integration Committee.

Mr. François Cattant graduated in chemical engineering in 1974 and joined Electricity of France (EDF) in 1975 as chemist engineer at the chemical department of the corporate laboratories (Plants Operation Division). At that time, he was involved in power plants water and steam conditioning.
Up to 1995 he worked in the following technical fields as an expert:

- Failure root cause analysis of gas-cooled reactors components, including fuel
- Water & steam chemistry, chemical cleaning and NDE for fossil fired stations
- Failure root cause analysis of nuclear power plants irradiated or contaminated parts & components and reactor pressure vessel (RPV) irradiation programs monitoring
- Examination of Dampierre 1 retired steam generator, to the examination of RPV head penetrations, to the study of thermal embrittlement, to the analysis of wear

Between 1995 and 1998 he was loan-in to the Nuclear Maintenance Application Center at EPRI Charlotte (NC, USA). He was involved in various maintenance guides such as those of pumps or diesel generators. He also acted as EPRI expert for the examination of Ringhals 3 retired steam generator.
In 1998 he moved back to France, at the R&D Materials and Mechanics of Components department where he stayed until his retirement in 2009. He served there as scientific advisor and senior engineer. His area of expertise was again chemistry, corrosion, and metallurgy, with special attention to primary water chemistry, source term reduction, primary water corrosion (Alloys 600/182/82, SSs), PWSCC mitigation and repair, fuel cleaning, innovation strategy. He also served as the EDF representative to the EPRI Materials Reliability program.
During this period:

- he was under contract with EPRI, being EPRI technical expert regarding several destructive examinations such as North Anna 2 RPV head penetrations, South Texas Project 1 Bottom Mounted Instrumentation, Braidwood 1 pressurizer heater #52, San Onofre 3 CEDM #64...
- he was the President of the "Materials, Non Destructive Testing and Chemistry" section of the "French Nuclear Energy Society" (from 2004 to 2008);
- he was also the Materials Ageing Institute (MAI) International Partnership Manager.

During his carrier, he made many presentations and papers in international conferences and scientifics journals.

In 2010, he was sponsored by the MAI to write a "Handbook of Destructive Assays", a 1100 pages document putting together extended summaries of hundreds of destructive examinations performed on LWRs' NSSSs, in France, US, Japan and Sweden.

Mr. Claude Amzallag retired from Electricity of France (EDF) in June 2010 after working for 37 years in the field of nuclear energy. He began his career in November 1973 at the Research Center of CREUSOT-LOIRE Steel Company, Firminy. He was in charge of the Fatigue Laboratory from November 1973 to June 1994.
The main achievements in CREUSOT-LOIRE include:

- Development and characterization of materials for nuclear, mechanical and aircraft industries,
- Realization and Management of Research and Development Programs on Fatigue and Rupture Behaviour of Materials,
- Coordination of Materials Handbooks [Fabrication, Materials Behavior (Fatigue, Corrosion, Rupture and Constitutive Laws), Field Experience],
- Standardization of Fatigue and Rupture Tests in FRANCE (AFNOR) and USA (ASTM),
- Organization of International Seminars, Workshops and Conferences – Editor of the Proceedings (ASTM Special Technical Publications, SF2M-FRANCE).
- Management of Projects for Nuclear, Mechanical and Aircraft Industries.

From July 1994 to June 2010, he worked at the Basic Design Department of Electricity of France (EDF - SEPTEN), Villeurbanne, as Materials Expert.
The main achievements in EDF include:

- Management of the EDF Scientific Program on Components made of Nickel Base Alloys,
- Responsible of the EDF Research and Development Program on Fatigue of Austenitic Stainless Steels,
- Collection and Analysis of International Service Experience in Fatigue and Corrosion,
- Coordination of Materials Handbooks,
- Assistance and Technical Support in Materials (fatigue, rupture, corrosion, constitutive laws, disposition and reference curves) for EDF,
- Review and Justification of fabrication problems.

He has authored and co-authored over 100 technical publications on Fatigue and Corrosion and made numerous presentations at various conferences. He has coordinated several international standards on Fatigue and Rupture Tests and edited several books. He has also been involved in teaching activities in the frame of permanent formation of nuclear engineers.

Mr. Peter Rudling is the President of ANT International, managing the ZIRAT/IZNA/LCC programmes as well as providing seminars and Handbooks on various fuel related topics to the nuclear industry. Peter was a senior consulting scientist at Vattenfall, the largest Swedish power company. Earlier he has also been a Specialist of Fuel Materials at ABB Atom (now Westinghouse) and a Project Manager at EPRI.

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LENA JOHANSSON
Axpo Chemistry Department

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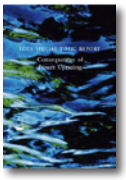

LCC3 STR
Consequences of Power Uprating

LCC4 STR
Environmentally-Assisted Degradation of Carbon and Low-Alloy Steels in Water Cooled Nuclear Reactors

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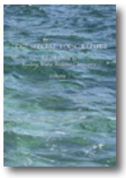

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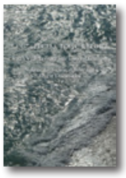

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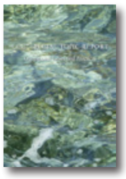

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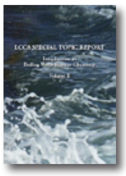

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