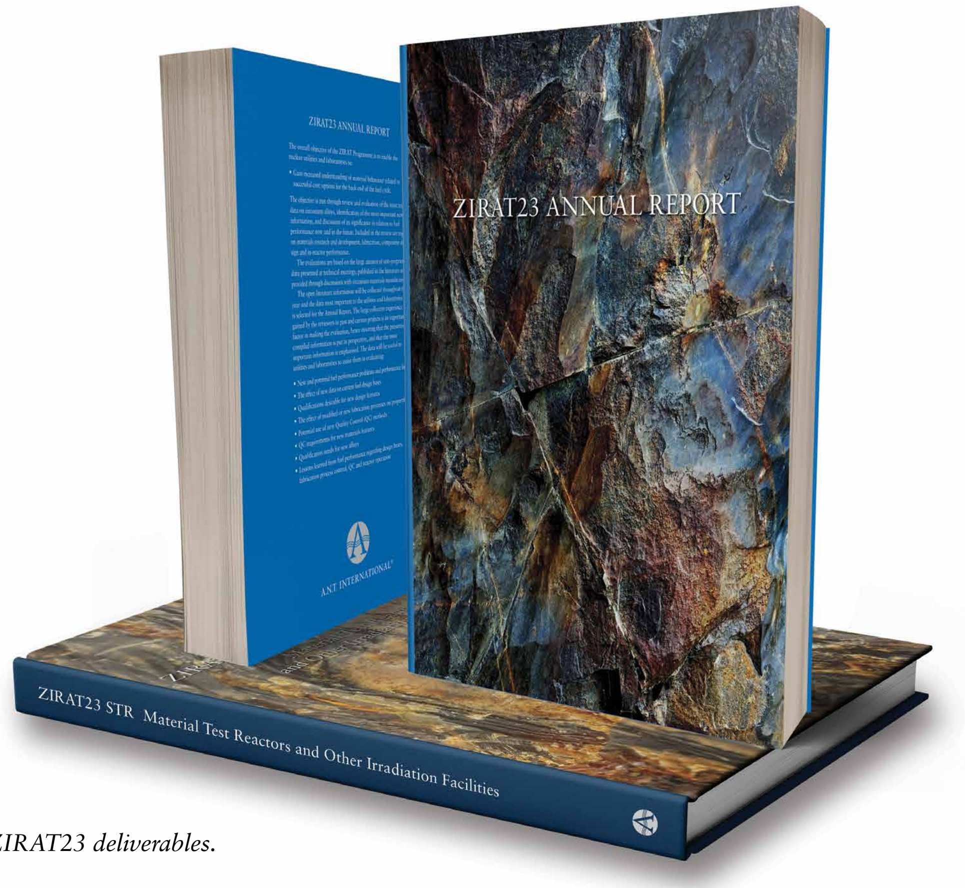
ZIRAT™

Zirconium Alloy Technology Programme



The ZIRAT23 deliverables.

The Annual ZIRAT Programme is focused on fuel assembly material issues and open to nuclear utilities and laboratories. In total, 34 organisations were members in 2017, representing more than 93 nuclear units worldwide. This programme was started in 1996.

A.N.T. INTERNATIONAL®

Deliverables

A.N.T. International will provide the ZIRAT Members with the following:

- Immediate download of high-resolution ZIRAT Annual Report and Special Topic Report PDF files 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 ZIRAT Annual Report and the Special Topic Reports can be printed. Also, the contents from the pdf-files can be copied and pasted electronically into other documents, e.g. Word files.
- The file(s) of the ZIRAT Annual Report and of the Special Topic Report will be provided before the Seminars (see below).
- Hardcopy, four colour reports will be provided if requested. The printed ZIRAT Annual Report and Special Topic Reports will be provided as soon as they are printed with the aim of delivery before the Seminars in February–March.
- Three similar Seminars will be held to present the results of the ZIRAT Programme in USA and Europe in February–March and in China. The number of full time employees per Member that may attend meetings is limited to eight (8) people per organisation. The language of the ZIRAT Programme will be English.
- The authors 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.

Listen to Mr. Andrew Sinclair – Amec Foster Wheeler

Listen to Mr. Michael Brown – Fuelco

"An interesting seminar with very useful information and a close overview of the state of art of nuclear fuel. I will recommend the seminar to my colleagues"

IGNACIO COLLAZO
Iberdrola

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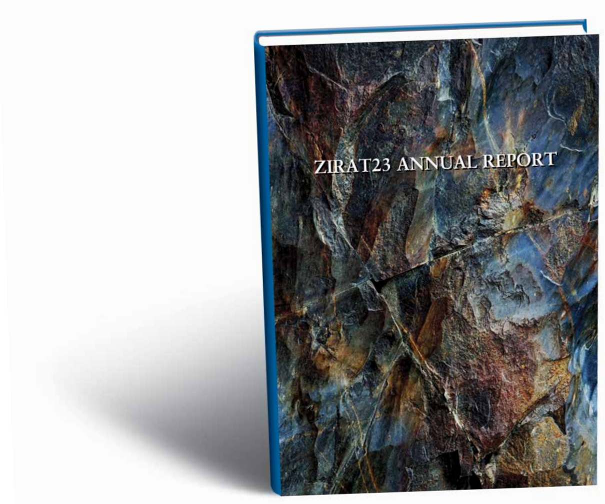
ZIRAT23 Programme



The overall objective of the ZIRAT Programme is to enable the nuclear utilities and laboratories to:

» *Gain increased understanding of material behaviour related to successful core operation and evaluations of options for the back end of the fuel cycle.*

The objective is met through review and evaluation of the most recent data on zirconium alloys, identification of the most important new information, and discussion of its significance in relation to fuel performance now and in the future. Included in the review are topics on materials research and development, fabrication, component design and in-reactor performance.

The evaluations are based on the large amount of non-proprietary data presented at technical meetings, published in the literature and provided through discussions with zirconium materials manufacturers.

The open literature information will be collected throughout the year and the data most important to the utilities and laboratories will be selected for the Annual Report. The large collective experience gained by the reviewers in past and current projects is an important factor in making the evaluation, hence ensuring that the presented compiled information is put in perspective, and that the most important information is emphasised. The data will be useful to utilities and laboratories to assist them in evaluating:

- New and potential fuel performance problems and performance limits.
- The effect of new data on current fuel design bases.
- Qualifications desirable for new design features.
- The effect of modified or new fabrication processes on properties.
- Potential use of new Quality Control (QC) methods.
- QC requirements for new materials features.
- Qualification needs for new alloys.
- Lessons learned from fuel performance regarding design bases, fabrication process control, QC, and reactor operation.
- Issues related to reactor (plant) life extension and refurbishment.

This information will help utility staff to implement actions to maintain or improve fuel reliability. Although the value of recent data endures, the specific technical issues affected by the recent data tend to change with time.

The ZIRAT23 Annual Report will start with a short introduction that will give the background and the current understanding of the topic typically based upon previous ZIRAT reviews. The introductory part will be followed by the review of the relevant data presented since the last ZIRAT review, i.e. ZIRAT22. In addition, each topic will have a final summary sub-section that will provide conclusions and an updated view of the understanding given in the introductory part.

The last section in the ZIRAT23 Annual Report will summarise the current issues related to fuel performance and list the data needed to resolve these issues. The following, currently important, issues are intended to be specifically addressed:

Burnup Achievements and Fuel Performance Issues

- New and potential fuel performance problems and performance limits.
- Trends in fuel operating conditions.
- High burnup fuel performance summary.
- Fuel reliability.
- Fuel performance related utility concerns.
- Fuel related regulatory issues of concerns to utilities and laboratories.

Fabrication

- Changes in zirconium alloy fabrication and QC methods and their potential effect on performance.

In-Reactor Performance of Zr Alloys

- Irradiation effects on microstructure of Zr alloy components such as: Fuel cladding, liner, guide tube, grid/spacer, fuel channel and pressure tube materials.
- The impact of alloying elements, microstructure, and irradiation conditions (temperature, power history, fast flux, fast dose, PWR, BWR and VVER water chemistry) on:
  » Corrosion and hydrogen pickup mechanisms, redistribution, effects on mechanical and corrosionproperties, and dimensional stability.
  » Mechanical properties (e.g. yield and ultimate yield strength, ductility, fracture toughness, fatigue, Delayed Hydride Cracking, Pellet Cladding Interaction.
  » Dimensional stability (irradiation growth, creep, relaxation).
- Recent primary fuel failures (fretting, corrosion, hydriding, Pellet Cladding Interaction (PCI), Pellet Cladding Mechanical Interaction (PCMI)) and secondary degradation, suggested remedies to improve failure resistance, important design and fabrication issues, and impact of plant operation.
- Relationship of fuel rod characteristics to performance in Loss of Coolant Accident (LOCA) and Reactivity Initiated Accident (RIA).
- The direct and indirect impact of water chemistry, CRUD, and chemical additions on the fuel performance.
- Utility and regulator perspectives including burn-up limits based upon the data presented in the ZIRAT23 base Report.

In-Reactor Performance of Accident Tolerant Fuel (ATF)

Prior to the accident at Fukushima, the emphasis of advanced LWR fuel development was on improving nuclear fuel performance in terms of increased burnup for waste minimisation, increased power density for power upgrades, and increased fuel reliability. Fukushima highlighted some undesirable performance characteristics of the standard fuel system during severe accidents, including accelerated hydrogen production under certain circumstances. Thus, fuel system behaviour under design-basis accident and severe-accident conditions became the primary focus for advanced fuels, along with striving for improved performance under normal operating conditions to ensure that proposed new fuels will be economically viable. Fuel vendors, nuclear research laboratories, and universities have embarked on an aggressive schedule for the development of enhanced accident-tolerant fuel (ATF) systems. The programs are in the early phases of R&D and are currently supporting the investigation of a number of candidate technologies that may improve the fuel system. This chapter of the annual report provides a brief review of the various ATF systems that are being developed.

Intermediate storage

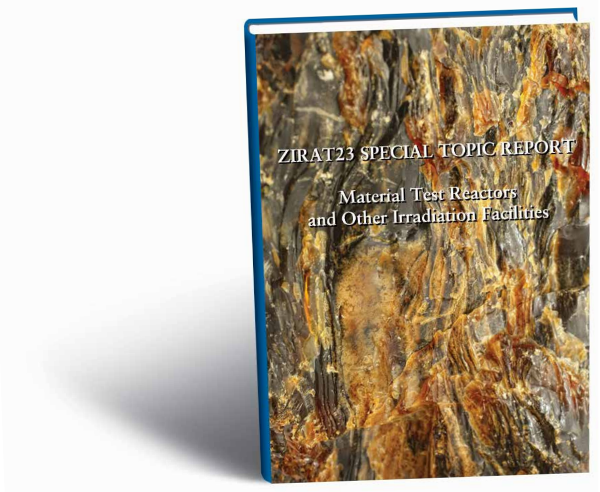
Fuel related issues in dry storage.

ZIRAT23 Special Topic Report

In addition to the ZIRAT23 Annual Report, one Special Topic Report will be prepared on ***Material Test Reactors and Other Irradiation Facilities***.

The Special Topic Report will cover the range from basic information to current knowledge and be written and explained in such a way that engineers and researchers not familiar with the topic can easily follow the STR, find and grasp the appropriate information. This means that the STR could be used by the organisations in the training of their internal staff with or without the additional assistance of A.N.T. International staff. The background and proposed content of the Report are discussed in detail on the next page.

Material Test Reactors and Other Irradiation Facilities



Irradiation damage in materials for nuclear applications primarily results from the production of energetic particles produced in fission, nuclear reactions, and radioactive decay events. The interaction of these energetic particles (fission products, fast neutrons, protons, alphas, and recoil nuclei) with materials results in the production of atomic-scale defects from ballistic collisions and introduction of new chemical elements. In the case of high-energy fission products, the intense ionization along the fission product path can also introduce defects or damage. These irradiation damage processes control the long-term evolution of the materials response to the production and diffusion of defects, fission products and nuclear reaction products.

The response of fuels and materials to radiation is critical to the performance of advanced nuclear systems. Key to understand material performance in a nuclear environment is the analysis of materials irradiated using Material Test Reactors (MTRs) and Ion Beam facilities. Many of the world's non-power-producing nuclear reactors are used for research and training, materials testing, or the production of radioisotopes for medicine and industry. These are basically neutron factories. These reactors are much smaller than power reactors.

Material test reactors, which are the subject of sections 1 and 2 of this STR, are a subset of research reactors. In materials test reactors, materials are subject to intense neutron irradiation to study the induced changes. MTRs have provided essential support for nuclear power programs over the decades. Associated with hot laboratories for the post irradiation examinations, they are structuring research facilities for the fission and fusion domain. These facilities address the development and the qualification of materials and fuels under irradiation with sizes and environmental conditions relevant to nuclear power plants. They provide a means for improving and to demonstrate safe operations of existing and coming power reactors as well as to support future reactors designs. As MTRs, some being high flux research reactors, are able to reproduce material degradation undergone
by materials in power reactors, they provide essential support to the study of ageing of generation II power plants, to the optimisation of generation III plants, and to test fuels and breeder capacities for generation IV.

MTRs are also being used to irradiate new cladding materials and fuels that are being developed to produce Accident Tolerant Fuel (ATF) systems.

According to the IAEA database there are 81 research reactors performing material irradiations distributed over 28 member states [IAEA, 2016]. Section 1 of this STR provides information about some of the most active MTRs around the world. The information on irradiation facilities in these MTRs has been collected from various sources. This information will help in selecting an MTR by the members of a nuclear utility or an engineering laboratory when planning irradiation of their reactor materials to predict the behaviour of the existing material at higher burn ups or generation of irradiation data for verification and licensing of new reactor materials such as new cladding and fuel materials for ATF systems.

One value from testing in MTRs comes from the higher neutron fluxes and radiation damage production rates. However, the dose rate and neutron energy spectra in MTRs are different from those in nuclear power plants. Therefore, the irradiated material data obtained from MTRs cannot be directly applied to power plant conditions without first considering the effects of dose rates and neutron energy spectrum. Section 2 of this STR addresses the opportunities and complexities of using materials test reactors with high neutron fluxes to perform accelerated studies of material aging in power reactors operating at lower neutron fluxes and with different neutron energy spectra. Radiation damage and gas production in different reactors have been compared using the code, SPECTER. This code provides a common standard from which to compare neutron damage data generated by different research groups using a variety of reactors [Griffiths et al, 2017].

"I enjoyed the meeting and the interactions. The annual report always provides not only a concise summary of work worldwide in the Industry over the past year, but also knowledgeable opinions as to its meaning and significance."

BRUCE KAMMENZIND
Bechtel Marine Propulsion Corporation

Read more at: www.antinternational.com/ZIRAT

As mentioned above, traditionally, research to understand radiation-induced changes in materials is conducted via radiation- effects experiments in material test reactors, followed by a comprehensive post-irradiation characterisation plan. This is a very time consuming process because of the low damage rates that even the highest flux reactors exhibit. In addition the high cost of research on irradiated materials in the face of shrinking budgets put additional constraints on this approach.

A promising partial solution to the problem is to use ion irradiation to irradiate mate-rials to very high doses. The advantages of ion irradiation are many. Dose rates (typically 10-3 to 10-4 dpa/s) are much higher than under neutron irradiation (10-7 to 10-8 dpa/s), which means that 100s of dpa can be reached in days or weeks instead of years. Because there is little activation, the samples have little or no radioactivity and often can be handled in a laboratory environment. Control of ion irradiation experiment variables is much better than experiments in nuclear reactors.

Challenges to the implementation of ion irradiation as a surrogate for neutron irradiation include rate effects on microstructures and effective temperature, small irradiation volumes, and accounting for transmutations. The unique effects of ion irradiation and the use of ions to study fast neutron irradiation effects is a critical topic, since development and qualification of new structural materials requires neutron doses that are too high to be obtained in existing nuclear test reactors or spallation neutron sources. The advantages and disadvantages of using ions to simulate nuclear radiation environments, along with available ion irradiation facilities and their capabilities, are reviewed in Section 3 of the STR, with specific examples provided.

Below is an intended content list

Section 1: Available Material Test Reactors (MTRs)

» Material test reactors in
- Americas – ATR, TREAT, HFIR, MITR, ……..
- Europe – Halden, OSIRIS, JHR, ………..
- Asia – HANARO, JMTR
- Russia - SM-3, BOR-60, MIR.M1, RBT-6, RBT-10/2, …….
- Africa
- Australia

» The info for each of the MTRs will include
- Reactor description
- In-core and out-of-core irradiation facilities
- Dimensions
- Environment
- Temp. range and control
- Fast (E>1MeV) neutron flux
- Neutron energy spectrum
- Types of samples
- PIE or packaging and transport facilities

Section 2: Using MTR Irradiation Data for Power Plant Conditions

- Differences between MTR and NPP irradiation environments
- Effects of neutron dose rate
- Effects of neutron energy spectrum
- Differences in gas production
- dpa contribution of gamma radiations
- Methodology for determining MTR and NPP irradiation equivalence
- Example results for various MTR spectra for fuel cladding and structural materials
- New alloy development for fuel materials
- Nickel alloys

Section 3: Ion Irradiation Techniques and Facilities

- Advantages of ion irradiations
- Ion irradiation techniques
- Ion irradiation facilities
- Equivalence of ion and neutron irradiations
- Typical examples of application of ion irradiation data

Ref.

- IAEA RRDB (Research Reactor Data Base), https://nucleus.iaea.org/RRDB/RR/ReactorSearch.aspx, 2016
- Griffiths M, Walters L, Greenwood L. R, Garner F. , Accelerated materials evaluation for nuclear applications, Journal of Nuclear Materials, 488, 2017

Report Authors

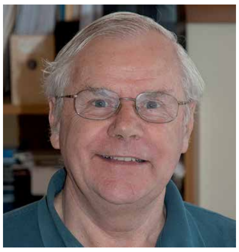
The authors are: Dr. Ron Adamson, formerly at GENE, Vallecitos, Mr. Friedrich Garzarolli, formerly at Framatome ANP, Dr. Charles Patterson, formerly at GNF, Dr. Kit Coleman, earlier at AECL, Dr. Tahir Mahmood formerly at GENE, Vallecitos, Dr. Malcolm Griffiths, earlier at CNL (earlier AECL), Dr. Albert Machiels, recently retired from EPRI, Dr. Clément Lemaignan formerly at CEA and Mr. Peter Rudling, President of A.N.T. International.

Dr. Ron Adamson retired from GE Nuclear Energy in 2000, where he was the manager of Materials Technology. Earlier he graduated from the University of Wisconsin with a B.S. in Mechanical Engineering, an M.S. in Nuclear Engineering and a PhD in Metallurgy. Post-doctoral work on irradiation effects was conducted at AERE, Harwell, England. At the GE Vallecitos Nuclear Center he led research, development and testing programmes for reactor core materials, with special emphasis on zirconium alloys. During his 31 years with GE, Dr. Adamson was actively involved with utilities and the technical community worldwide. He holds 17 patents, has published over 90 technical papers involving nuclear materials technology, and has received several important awards, including the Outstanding Technical Contribution Award from GE Industrial Power Systems, the Mishima Award from the American Nuclear Society, and the Kroll Medal from the ASTM/Kroll Institute. Zirconium alloy areas in which Dr. Adamson has particular interest and experience include: in-reactor dimensional stability; in-reactor corrosion performance and mechanisms; microstructure evolution due to reactor irradiation; mechanical properties of irradiated material; high burnup performance; failure mechanisms and remedies; and fabrication technology. Since retirement he has been actively associated with A.N.T. International, EPRI and others as a consultant in zirconium technology.

Mr. Friedrich Garzarolli retired from Framatome ANP in March 2002, where he has held various managerial and research positions, dealing with fuel rod performance analysis, planning and evaluation of irradiation tests, materials characterisation and evaluation of irradiation effects in materials. His degree as Diplom Ingenieur in metallurgy was obtained from the University of Leoben, Austria, in 1963. He has been active in the following fields:

- Development of new fuel assembly materials, especially cladding for BWRs and PWRs
- Modelling of corrosion for zirconium alloys and stainless steels
- Effect of water chemistry on cladding corrosion
- PCI failures of cladding
- In-reactor dimensional stability
- High-burnup performance
- Failure mechanisms and remedies
- Microstructure evolution due to reactor irradiation

Dr. Charles Patterson retired from Global Nuclear Fuel in 2008 as a Consulting Engineer for Fuel Engineering. During 44 years with GE Nuclear Energy/GNF, he was actively engaged in the development of fuel manufacturing processes, fuel materials, thermal-mechanical and fuel performance models and in the improvement of fuel reliability. This activity involved irradiation and hot cell Programmes in Asia, Europe and the United States to identify in-core material behaviour, validate analytic models and improve fuel reliability. Chuck holds patents in the areas of fuel and cladding materials, fuel assembly design and fuel inspection technology. Dr. Patterson has particular interest and experience in the thermal and mechanical behaviour of fuel, cladding and structural materials, the development of analytic models to describe their behaviour and in the improvement of fuel reliability.



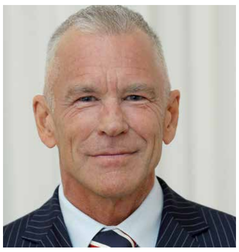
Dr. Kit Coleman has, after receiving a PhD in the UK, spent his working career at the Chalk River Laboratories of AECL. Research interests on zirconium alloys included in-reactor creep, development of improved fuel cladding and pressure tube materials. He retired in 1999 as manager of Material and Mechanics Branch but retains an attachment to CNL (formerly AECL) as a Researcher Emeritus. He has published over 100 papers on zirconium technology and has received the Russ Ogden Award from ASTM and the Kroll Medal from the ASTM/Kroll Institute. He is on the Advisory Editorial Board of the Journal of Nuclear Materials.

Dr. Sheikh Tahir Mahmood *r*etired from Global Nuclear Fuel in 2012 as a Senior Engineer/Technologist for Fuels Engineering at the Vallecitos Nuclear Center. Earlier he received Masters degrees in Physics and Nuclear Technology from abroad and doctorate in Nuclear Engineering from North Carolina State University. His Post-doctoral work on mechanical anisotropy of zirconium alloys and radiation effects on reactor structural materials was done at NCSU and ORNL, respectively. At GE Nuclear Energy/GNF, he was actively engaged in fuel performance and materials technology. This activity involved failure root-cause investigations through hot cell PIE of the failed in-core components, and development and evaluation of material property data bases for new materials developed for in-core use. Tahir has particular interest and experience in mechanical metallurgy, mechanical behaviour of fuel, cladding and structural materials, and in-reactor behaviour of these materials for improved fuel reliability. He has actively participated in various international nuclear industry research programs.

Dr. Malcolm Griffiths obtained his PhD in Physical Metallurgy from the University of Birmingham in 1981. After a three-year post-doctoral term working on radiation damage in Ti-alloys he joined AECL at the Chalk River Laboratories in 1984. He has worked on various aspects of materials performance in nuclear reactor cores during his 32 years with AECL. As manager of the Radiation Damage and Deformation Branch Malcolm was instrumental in developing strategies to support CANDU refurbishment that addressed improving pressure tube performance and the degradation of Inconel X-750 core components. Up until his retirement he was chair and project manager for the Candu Owners Group (COG) R&D program on pressure tube deformation and the COG joint project on Inconel X-750 spacer degradation. From 2003-2013 Malcolm was on the editorial advisory board for the Journal of Nuclear Materials and was an editor from 2013-2016. In 2007 he was recipient of the 2005 Kroll medal from the American Society for Testing and Materials for his pioneering work on microstructure evolution in zirconium alloys during irradiation. He has published nearly 100 papers on material characterisation and reactor core materials performance. He is currently adjunct professor at Queens University, Department of Mechanical and Materials Engineering.

Dr. Albert Machiels retired in June 2017 from the Electric Power Research Institute [EPRI] located in Palo Alto, California, where he was responsible for providing technical expertise on topics related to spent fuel management, fuel cycles, and advanced generation technologies. Dr. Machiels has 50 years of involvement in various fields of nuclear technology R&D, including faculty and program direction positions at several universities and EPRI. In 2012, Dr. Machiels received a Lifetime Achievement Award for his numerous technical contributions to nuclear technology. He holds Chemical and Nuclear Engineering degrees from the Université of Liège, Belgium and a PhD degree in Engineering from the University of California, Berkeley.

Dr. Clément Lemaignan, born in 1949, graduated in material science from school of Mines, St Etienne, got his MS in metallurgy from MIT and his PhD from Polytechnicum of Grenoble. He worked during his entire carrier at the Grenoble Nuclear Center in the field of Nuclear Metallurgy, on fuel behaviour, irradiation damage and RPV embrittlement and mostly focussing on Zr alloy behaviour under irradiation and corrosion. Retired form CEA, he acts currently as scientific consultant and advisor for various nuclear agencies or companies. Emeritus Professor at INSTN (CEA Paris) of nuclear metallurgy, physics of irradiation damage and fracture, Prof. Lemaignan holds 3 patents, has authored 3 books, 6 book chapters, and over 80 peer-reviewed articles. He has been editor for the Journal of Nuclear Materials for more than 15 years. For his research and teaching, Prof. Lemaignan has received many distinctions, among them the Kroll award from ASTM in 2001.



Mr. Peter Rudling is the President of A.N.T. 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.

Price and Terms of Payment

The fixed nominal price for the ZIRAT Membership appears in the associated Proposal.

Terms and Conditions

The term of ZIRAT23 Programme starts from the date of the purchase order and lasts 12 months onwards.

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