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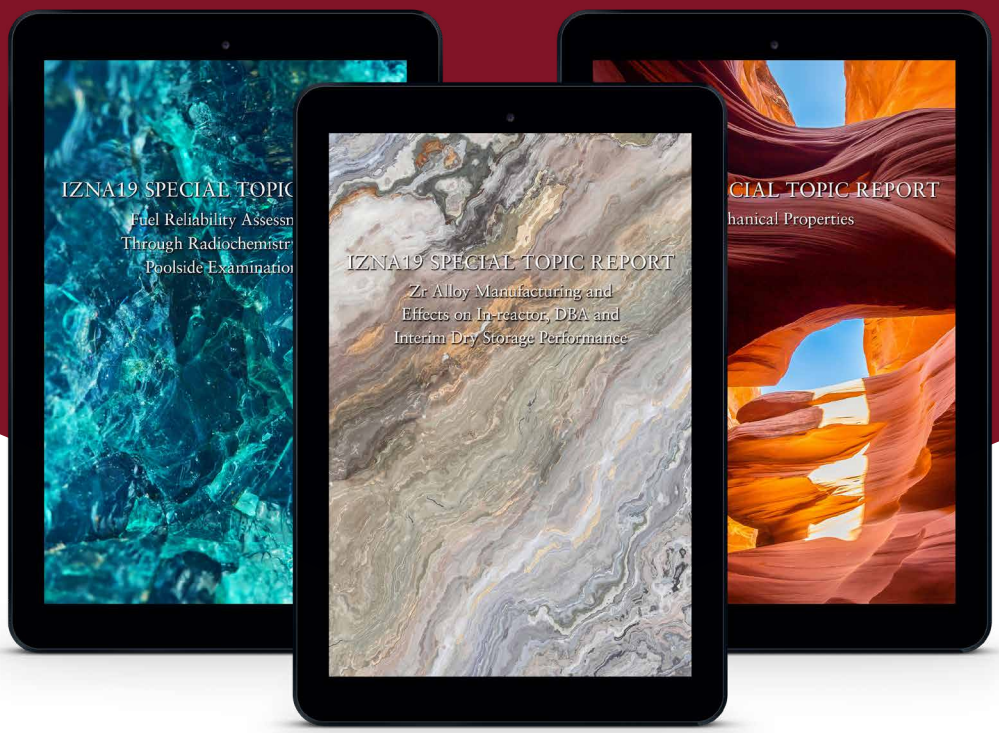
IZNA™

Information on Zirconium Alloy Technology Programme



The Annual IZNA Programme is focused on fuel assembly material issues and open to fuel vendors, regulators and research laboratories.

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Deliverables

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

- Searchable electronic report version with the following contents:
  - » High-resolution pdf files with complete IZNA Special Topic 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 Special Topic 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.
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- Hardcopy, four colour reports will be provided if specifically requested by the customer. The printed Special Topic Reports will be provided as soon as they are printed.
- 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.

"Since 2003 and until now, IZNA products offered by ANT International have been one of the main source of information for the training of our new engineers."

CRISTINA MUÑOZ-REJA RUIZ
Fuel Rod Technology Manager ENUSA, Industrias Avanzadas S.A. Spain

"In June 2008, Peter Rudling and Ron Adamson gave a one-day seminar at GNF's Wilmington, North Carolina site. The seminar was conducted with an open manner with plenty of audience interaction, a feature that was well appreciated by all attendees."

DR. YANG-PI LIN
Lead Engineer Global Nuclear Fuel, USA

Read more at: www.antinternational.com/IZNA

IZNA19 Programme

The overall objective of the IZNA Programme is to:

Interpret published R&D results and nuclear plant experience on Zr alloy materials to help the customer to make informed decisions for the improved reliability, safety and economics of operations.

Point out areas where more R&D is needed to resolve potential issues (indications just seen but no problem yet)

The objective is met through review and evaluation of the 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.

IZNA19 Special Topic Report

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 is discussed in detail below.

Zr Alloy Manufacturing and Effects On In-reactor, DBA and Interim Dry Storage Performance

To ensure safe operation of nuclear fuel, certain performance criteria must be met both during normal operation, anticipated operational occurrences and during postulated accidents. Zr Alloy material properties impact some of the most important performance criteria such as:

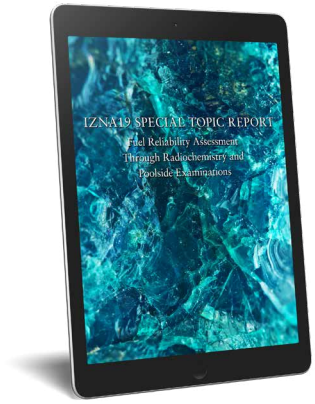
- Pellet Cladding Interaction (PCI) Pellet Cladding Mechanical Interaction (PCMI) corrosion and hydriding properties which in turn, are dependent upon:
  - » Chemical composition.
  - » Crystallographic texture and cladding microstructure.
  - » Cladding strength/ductility/fracture toughness.
- Re-opening of the pellet-cladding gap, lift-off, which is partly related to the fuel cladding creep properties.

- Excessive dimensional changes (resulting in e.g. excessive Fuel Assembly (FA) bowing) of fuel components that are a function of creep (including oxide induced and residual stress relaxation creep), irradiation growth, and hydrogen pickup in the components.
- Loss of Coolant Accident (LOCA) performance that is related to hydrogen pickup both during the base irradiation, before the LOCA event, as well as during the high temperature LOCA oxidation.
- Maintaining fuel integrity and fuel retrievability under normal dry storage and transportation conditions.
- Maintaining sub-criticality and shielding requirements under transportation accidents.

The Zr-alloy properties are a function of the reactor environment (fast neutron flux, temperature, water chemistry, etc.) and the Zr-alloy microstructure. The microstructure is a function of material chemistry and manufacturing process. A better knowledge of the impact of manufacturing changes on the Zr alloy microstructure and in-pile material properties can lead to improved capabilities to respond to current fuel performance issues and to ensure that any changes in the current manufacturing processes of the Zr alloys does not impair fuel's safety performance criteria. The overall objective of this Special Topic Report (STR) is to provide this knowledge. Below is the report content list.

Content List

1. Reactor characteristics,
2. Fuel Design+ material used
3. Basics
   - 3.1 Effects of Irradiation on coolant, oxide and Zr alloy
   - 3.2 Microstructure in Zr alloys (texture, Kearns factors, etc.)
   - 3.3 LWR - Normal operation, AOO, DBA
     - 3.3.1 Corrosion and HPU, Irradiation growth and creep,
     - 3.3.2 PCI, PCMI
     - 3.3.3 Seismic event, LOCA and RIA
   - 3.4 Interim Dry Storage- Main regulatory requirements related to dry storage
   - 3.5 CANDUs pressure tubes operation conditions
4. Zr source material manufacturing (VanArkel, Electrolytic process, Kroll)
   - 4.1 Effects on Zr alloy performance during LOCA
5. Ingot melting process
   - 5.1 Effects on microstructure (chemical composition and impurties from ingot melting) and in-pile performance, DBA
6. Cladding tube (fuel rods, including liner) guide tube manufacturing
   - 6.1 Effects on microstructure and in-pile performance, DBA
7. Effects of microstructure on dry storage
8. Strip and sheet manufacturing
   - 8.1 Effects on microstructure and in-pile performance
9. CANDU pressure tube manufacturing and effects on in-pile performance



Fuel Reliability Assessment Through Radiochemistry and Poolside Examinations

The primary objective of this report is to provide guidance in assessing fuel reliability for decisions regarding core operation, the cause of failure, returning fuel to the core for continued irradiation and of used nuclear fuel in dry storage containers. The first step is to determine during operation if the core is defect free or not through radiochemistry. If defected rods are detected or suspected, it is crucial that the failed rods be identified during poolside examination and the root cause be assessed in order to take actions to ensure that this failure mode does not occur again. To reach these objectives, various nondestructive poolside examinations techniques may be used to identify failed rod(s) and failure cause(s). In many cases root cause analysis also requires hot cell examinations.

If failed fuel is degrading during operation, fuel washout may contaminate the core with tramp uranium, which will increase the plant radiation levels for many years even after the failed and degraded rod(s) have been removed from the core. With good methodologies, whether fuel washout is occurring or not can be assessed during reactor operation. If significant fuel washout occurs during operation, plant management may decide to shut down the plant to remove the failed degraded rod(s) to stop fuel washout.

A secondary objective is to promote clear documentation of the state (intact or damaged) of all spent fuel assemblies or bundles for subsequent operations associated with the back-end of the fuel cycle. The latter includes wet storage at the power plant, interim wet or dry storage, transportation, and eventually disposal in a geologic formation. Primary fuel parameters of interest are fuel design, burnup, and leak tightness.

Content List

Mechanical Properties

Fuel cladding and support components used in water-cooled nuclear reactors are subjected to stress, various chemical reactions and irradiation by neutrons. This report is intended to provide basic understanding on the mechanical properties of zirconium alloys, stainless and ferritic steels and nickel alloys. The information can then be used by customers to evaluate their components.

Adequate mechanical properties are required to ensure safe and satisfactory performance during normal and accident conditions and during post-operation storage. In this report testing techniques and standards for unirradiated and irradiated materials are described. Typical properties are yield strength, ductility and fracture toughness. Properties and effects of relevant variables are discussed with a view on how this information may be used. The consequences of reactor environment are outlined.

The generation of point defects during irradiation leads to an evolution of microstructure with subsequent changes in mechanical properties, usually an increase in strength and loss in ductility. Transmutation may have to be considered. Chemical reactions may lead to stress corrosion cracking and pickup of hydrogen that may produce embrittling hydrides. Mechanisms of deformation such as slip and twinning and processes leading to fracture are discussed. Analysis through tensors and yield criteria provide understanding of the effects of crystallographic texture and anisotropic deformation.

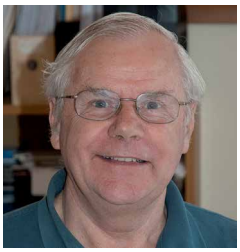
Content List

1. Mechanical Properties of:
   1.1 Zr-Alloys, Ni-Alloys
   1.2 Steels: Austenitic and Ferritic
2. Mechanical Property Testing Techniques of Archive And Irradiated Materials
3. Deformation Mechanisms, Irradiation Effects
4. Specific Effects Of Hydrides

Report Authors

The authors are: **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**, retired from EPRI, **Dr. Clément Lemaignan** formerly at CEA and **Mr. Peter Rudling**, President of A.N.T. International.

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 retired 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.

"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

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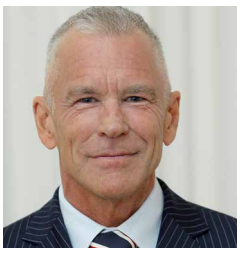
Dr. Malcolm Griffiths obtained his PhD in Physical Metallurgy from the University of Birmingham in 1981. After a three-year post-doctoral term working on radiation damage in Ti-alloys he joined AECL at the Chalk River Laboratories in 1984. He has worked on various aspects of materials performance in 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 IZNA Membership appears in the associated Proposal.

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Insurances of the owner and of others in respect of a nuclear incident shall exclude any right of recourse against the supplier and his sub-suppliers of every tier and kind.

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