ZIRAT™

Zirconium Alloy Technology Programme

The ZIRAT21 deliverables.
For information on earlier published ZIRAT Special Topic Reports please see *page 13*.

The Annual ZIRAT Programme is focused on fuel assembly material issues and open to nuclear utilities and laboratories. In total, 33 organisations were members in 2015, representing more than 90 nuclear units worldwide. This programme was started in 1996.

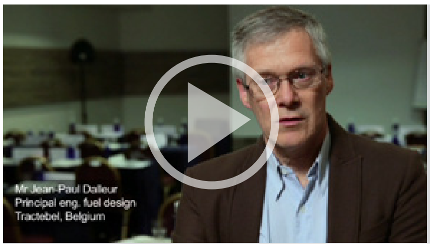
Deliverables

ANT International will provide the ZIRAT Members with the following:

- Searchable CD-ROM(-s) with the following contents:
  - High-resolution pdf files with the complete ZIRAT Annual Report and the 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 ZIRAT Annual Report and 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.

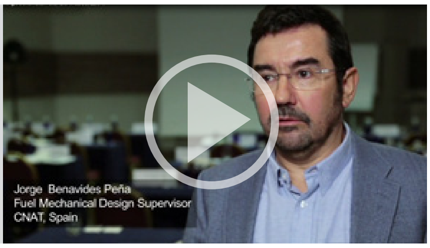
  The CD-ROM(-s) of the ZIRAT AR and of the STRs will be provided before the Seminars (see below).
- Optional reports printed in four-colour. The printed ZIRAT Annual Report and Special Topic Reports will be provided as soon as they are printed with the aim of delivery before the Seminars in February–March.
- Two similar Seminars will be held to present the results of the ZIRAT Programme in USA and in Europe in February-March. 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 Frank Holzgreve, BKW.

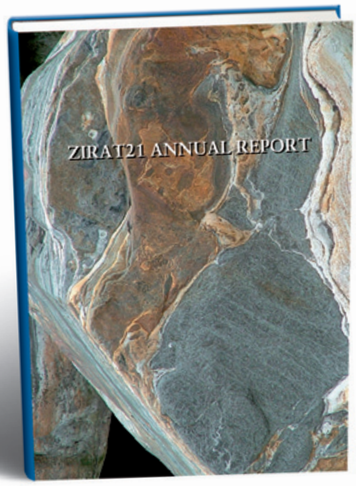


Listen to Jean-Paul Dalleur, Tractebel.

Listen to Jorge Benavides Peña, CNAT.



ZIRAT21 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 emphasized. 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.

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 ZIRAT21 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. ZIRAT20. 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 ZIRAT21 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

- 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 corrosion properties, 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 ZIRAT21 base Report.

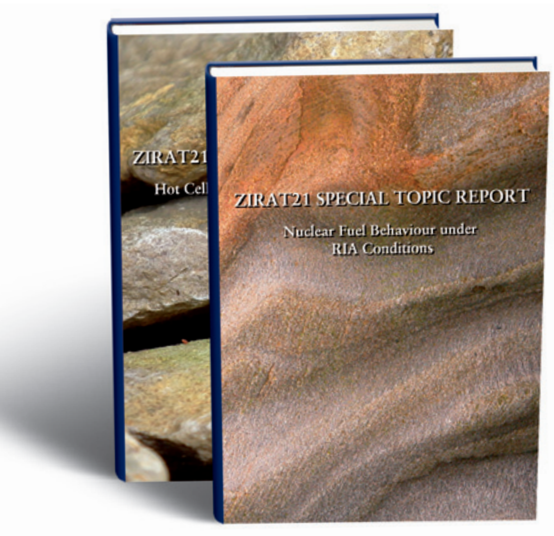
In-Reactor Performance of Accident Tolerant Fuel (ATF)

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 minimization, 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.

ZIRAT21 Special Topic Reports



In addition to the ZIRAT21 Annual Report, two Special Topic Reports will be prepared on Nuclear Fuel Behaviour under RIA Conditions and Hot Cell Post-Irradiation Examination Techniques – Vol. II. The Vol. I, entitled Hot Cell Post-Irradiation Examination Techniques for Light Water Reactor Fuels was issued within the ZIRAT19 Programme.

The Special Topic Reports 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 STRs, find and grasp the appropriate information. This means that the STRs could be used by the organisations in the training of their internal staff with or without the additional assistance of ANT International staff. The background for selecting topics and proposed contents of the Reports are discussed in detail below.

Nuclear Fuel Behaviour under RIA Conditions

The design basis RIA in a PWR is the Control Rod Ejection (CRE), while in a BWR, it is the Control Rod Drop Accident (CRDA). The CRE is based on the assumption of a mechanical failure of the control rod drive mechanism located on the reactor vessel top, followed by the ejection of the mechanism and the control rod by the internal reactor pressure. The resulting, significant power surge is limited partly by Doppler feedback and finally terminated by the reactor trip. The BWR CRDA is assumed to occur if a control rod is detached from its drive mechanism in the core bottom, stays stuck while inserted in the core, then if loosened, drops out of the core by gravity, without involvement of a change in reactor pressure as in the CRE. Partly as a result of these differences, the BWR power pulses are slower than for a PWR. The pulse widths for PWRs are in the range of 10–30 ms and for BWRs in the range of 20–60 ms.

The reactivity transient during an RIA results in a rapid increase in fuel rod power leading to a nearly adiabatic heating of the fuel pellets.

The RIA-simulation experiments conducted in the 1960's and 1970's using zero or low burnup test rods showed that cladding failure occurred primarily by either:

Post-Departure from Nucleate Boiling (DNB). Brittle fracture of the clad material occurring during the re-wetting phase of the overheated heavily oxidised (and thereby embrittled) clad due to the abrupt quenching resulting in large thermal clad stresses. This failure mode is imminent if the cladding is severely oxidized due to the RIA fuel clad temperature excursion.

Cladding contact with molten fuel. Contrary to low burnup rods, the dominant failure mechanism for high burnup rods is Pellet Cladding Mechanical Interaction (PCMI). The change in failure mechanism is due to the decrease in pellet-cladding gap and the embrittlement of the cladding (due to corrosion induced hydriding) with increased burnup. The survival of a high burnup fuel rod in an RIA is dependent on the ability of the cladding to resist PCMI, which in turn depends largely on the cladding ductility. The condition of the cladding has a significant effect on the ductility, specifically the alloy composition, microstructure and texture. In addition, the cladding hydrogen content – most importantly the hydrogen distribution – has a significant impact on the PCMI response. The ductility of the cladding is also highly temperature sensitive and the cladding temperature at beginning of the RIA as well as the power pulse shape are important to whether PCMI-induced failures will occur.

At a later stage of the transient, heat transferred from the pellets may bring the clad outer surface to such a high temperature that dry-out or DNB will occur. If so, the clad material could remain at a temperature above 1000–1200 K for up to 10 s, until rewetting takes place. The elevated temperature may lead to clad outward ballooning and creep burst, in cases where the rod internal gas pressure exceeds the coolant pressure, which may be the case for some high burnup rods.

Provided that the cladding fails, fragmented fuel may disperse into the coolant. This expulsion of hot fuel material into water has potential to cause rapid steam generation and pressure pulses, which could damage nearby fuel assemblies and possibly also the reactor pressure vessel and internal components. Hence, the potential consequences of fuel dispersal are of primary concern with respect to core and plant safety.

This Special Topic Report (STR) will give insight and understanding of the parameters impacting the fuel RIA performance and reviews the applicability of the data to high burnup fuel cladding. The STR also provides the latest RIA regulatory acceptance criteria.

Below is an intended content list

1. Introduction
   - 1.1 Historical background to reactivity initiated accidents
   - 1.2 Consequences of reactivity initiated accidents
   - 1.3 Scope and outline of the report
2. Overview of RIA scenarios
   - 2.1 Reactivity insertion events in major types of reactors
     - 2.1.1 Control system failures
     - 2.1.2 Control rod ejections
     - 2.1.3 Coolant/moderator effects
   - 2.2 Expected power pulse characteristics for various accident scenarios
     - 2.2.1 Pulse width and pulse shape
     - 2.2.2 Pulse amplitude
3. Overview of damage phenomena
   - 3.1 Types of damage to fuel and cladding
   - 3.2 Phenomena with influence on core coolability

"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

Hot Cell Post-Irradiation Examination Techniques – Vol. II

Maintaining and improving reliability of fuel and structural components requires an understanding of their behaviour in reactor and the mechanisms that have been observed to cause failures. A key factor in improving reliability is the identification of the cause or causes of failure. Such information, in turn, requires the examination and analysis of irradiated fuel (including bundle hardware) and structural components at reactor sites (poolside examinations), in hot cells and in related laboratories. Thus, to make progress toward ultra-high reliability fuel and to reduce the potential for fuel failure, it is imperative to examine both failed and non-failed (reference) fuel. Post-irradiation examinations (PIE) provide fuel vendors and nuclear utilities with data on how newly developed or established materials withstand normal operating conditions in new environments. Post-irradiation examinations are largely carried out at a Hot Cell Laboratory where irradiated fuel rods and other hardware can be received, handled, examined, and tested. The investigation results provide information for fuel and component improvement and, thereby, can potentially enhance operating efficiency and reliability. The pool-side and hot cell post-irradiation examination techniques used for LWR fuel and bundle hardware have already been presented in the ZIRAT19 STR titled "Hot Cell Post-Irradiation Examination Techniques for Light Water Reactor Fuels".

Section 1 of this STR will provide an overview about the status of post-irradiation examination (PIE) and inspection techniques for nuclear fuel and other zirconium alloy components used in CANDU reactors and their applications for analysis of materials behaviour in a CANDU reactor core. Emphasis will be given to advanced non-destructive and destructive PIE techniques applied to fuel rods and pressure tubes with examples in the form of case studies.

Microstructure plays a central role for the efficient use of zirconium alloys during service in nuclear reactors. The in-reactor performance of components made from zirconium alloys depends on the environment – neutron flux, high temperature and water chemistry – and the properties of the metal – corrosion resistance, mechanical strength, and ductility. These properties are controlled by the microstructure of the alloy. Microstructure describes the crystal phases present, their spatial and orientation distributions and their defect structure. The initial microstructure is determined by the alloy composition and the various working processes and heat-treatments required to fabricate the component and meet various specifications. During service the microstructure is modified by interactions with neutrons and water. It is, therefore, important to investigate the microstructure of irradiated components. Various microstructural examination techniques from light optical microscopy (LOM) to scanning transmission electron microscopy (STEM) and more advanced and specialized techniques are used for this purpose. Section 2 of this STR discusses these techniques along with real world examples of in-reactor microstructural changes and impact on material behaviour.

Post-irradiation examinations (PIE) are largely carried out at a Hot Cell Laboratory where irradiated fuel rods and other bundle hardware can be received, handled, examined,

> *"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

and tested. A number of hot cell facilities exist in various countries around the world. Each of these facilities has its specific strengths and limitations. Section 3 of this STR provides information on PIE capabilities of some of the major hot cell facilities. This information will be useful for utility engineers when they need to have PIE performed on failed nuclear fuel or other components.

Below is an intended content list

1. Hot Cell PIE of CANDU Fuel and other Zirconium Alloy Components
    - 1.1 Introduction
    - 1.2 CANDU reactor and fuel design
    - 1.3 Zirconium alloy components in a CANDU reactor
    - 1.4 Non-destructive examination techniques
    - 1.5 Destructive methods
    - 1.6 Chemical and microstructural analyses
    - 1.7 Mechanical testing
2. Microstructural Examination Techniques for Nuclear Fuel and Bundle Hardware
    - 2.1 Introduction
    - 2.2 Light Optical Microscopy (LOM)
    - 2.3 Scanning Electron Microscopy (SEM)
    - 2.4 Transmission Electron Microscopy (TEM)
    - 2.5 Advanced Techniques
3. Hot Cell Post-Irradiation Examination Facilities around the World
    - 3.1 Introduction
    - 3.2 Summary of PIE capabilities of the facilities
    - 3.3 PIE tasks and facilities

Appendix A: As-provided information on various hot cell facilities

Report authors

The authors are: Mr. Peter Rudling, President of ANT International, Dr. Ron Adamson, formerly at GENE, Vallecitos, Mr. Friedrich Garzarolli, formerly at Framatome ANP, Dr. Charles Patterson, formerly at GNF and Dr. Kit Coleman, earlier at AECL and Dr. Lars O. Jernkvist.

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 ANT 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 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 behavior of fuel, cladding and structural materials, and in-reactor behavior of these materials for improved fuel reliability. He has actively participated in various international nuclear industry research programs.

Dr. Lars O. Jernkvist is a co-founder and CEO of Quantum Technologies AB, a Swedish company offering science-based consulting services in analysis, modelling and simulation of mechanical-material systems. After earning a MS in Engineering Physics and a PhD in Mechanical Engineering, Lars has gained more than 25 years experience in analyses, modelling and simulation of most aspects of LWR nuclear fuel behaviour. He has particular acquaintance with analyses and simulations of the complex mechanisms responsible for fuel failure under normal reactor operation and storage conditions, such as PCI and hydride embrittlement. Over the last decade, his work has turned to fuel behaviour under off-normal and accident conditions, such as LOCA and RIA, and he is currently engaged in research on methods and computational tools for safety analyses of LWRs and spent fuel facilities.

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.

Price and Terms of Payment

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

Terms and Conditions

The term of ZIRAT21 Programme starts from the date of the purchase order and lasts 12 months onwards.

ANT International shall exercise its best efforts to meet the objectives in this assignment and shall apply to the work professional personnel having the required skills, experience and competence. If the assignment is found to be significantly deficient by the customer within 6 months of its completion, ANT International shall modify the work done within this assignment in such a way that it will become satisfactory to the customer. This modification shall be done without incurring any additional costs to the customer. The total amount of such additional costs due to the modification shall be limited to be less or equal to the amount originally paid to ANT International for this assignment.

It is understood that ANT International is not responsible for any damage, incurred to the customer, their employees, or their plants or to a third party due to the use of the information or the recommendations given within this assignment.

The compiled information and the conclusions, as a result of this work, may be used by the purchasing party for its own use for any purpose provided that the source is given. ANT International retains the rights to the compiled information and the conclusions for other uses.

Nuclear Liability

ANT International and its sub-suppliers, including also suppliers of information and services, of every tier and kind, and everyone engaged by any of them, shall have no liability whatsoever (irrespective of negligence or gross negligence) for any damage or loss whatsoever (including also consequential and indirect loss) resulting from a nuclear incident (as such term is defined in the Paris Convention on third party liability in the field of nuclear energy, as amended from time to time). This shall apply for damage or loss suffered by third parties or the owner and for damage and loss to the nuclear installation, on site property and any other property of any kind, and until the nuclear installation has been definitely decommissioned and irrespective of any termination or cancellation of the proposed work.

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.

Contact

For more information and/or an offer
welcome to contact us at sales@antinternational.com

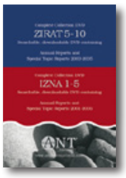
Please also visit our website for the latest updated information, www.antinternational.com

Earlier published ZIRAT Special Topic Reports

Available in hard copy

Available on CD/DVD

Manufacturing of Zr Alloys

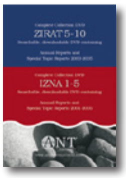


ZIRAT5 STR
Manufacturing of Zirconium Alloy Material

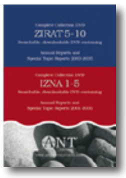
ZIRAT14 STR
Impact of Manufacturing Changes on Zr Alloy In-Pile Performance

ZIRAT11 STR
Manufacturing of Zr-Nb Alloys

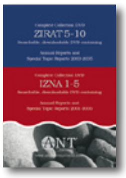
Corrosion and Hydriding of Zr Alloys



ZIRAT6 STR
Water Chemistry and Crud Influence on Cladding Corrosion

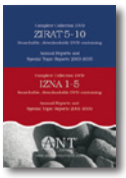


ZIRAT9 STR
Corrosion of Zr-Nb Alloys in PWRs



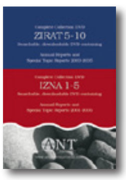
ZIRAT7 STR
Corrosion of Zirconium Alloys

ZIRAT12 STR
Corrosion Mechanisms in Zirconium Alloys



ZIRAT8 STR
The Effects of Zn Injection (PWRs and BWRs) and Noble Metal Chemistry (BWRs) on Fuel Performance

Properties of hydrides and impact on fuel in-reactor performance

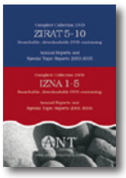


ZIRAT5 STR
Hydriding Mechanisms and Impact on Fuel Performance

ZIRAT13 STR
Effect of Hydrogen on Zirconium Alloy Performance (normal operation, LOCA/RIA and dry storage)

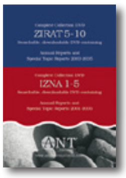
ZIRAT13 STR
Effect of Hydrogen on Zirconium Alloy Properties

Dimensional changes of fuel assemblies/channels and components (fuel assembly/ channel bowing, irradiation growth, creep, relaxation, effect of hydrogen)



ZIRAT7 STR
Dimensional Stability of Zirconium Alloys

ZIRAT16 STR
BWR Fuel Channel Distortion

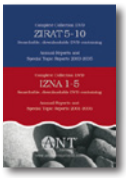


ZIRAT10 STR
Structural Behaviour of Fuel Components

ZIRAT19 STR
Charged Particle Bombardment of Zirconium Alloys – A Review

ZIRAT14 STR
In-Reactor Creep of Zirconium Alloys

Mechanical properties of Zr Alloys



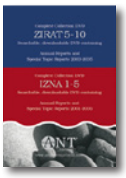
ZIRAT6 STR
Mechanical Properties of Zirconium Alloys

ZIRAT11 STR
Pellet–Cladding Interaction

ZIRAT18 STR
Mechanical Testing of Zirconium Alloys
Volume I

ZIRAT18 STR
Mechanical Testing of Zirconium Alloys
Volume II

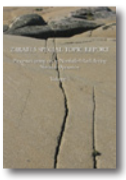
LOCA and RIA fuel performance



ZIRAT9 STR
Loss of Coolant Accidents, LOCA, and Reactivity Initiated Accidents, RIA, in BWRs and PWRs

ZIRAT15 STR
Processes going on in Nonfailed Rod during Accident Conditions (LOCA and RIA)

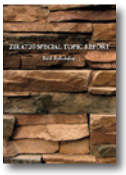
Fuel (UO_2 and MOX) performance



ZIRAT15 STR
Processes going on in Nonfailed Rod during Normal Operation

ZIRAT15 STR
Processes going on in Nonfailed Rod during Accident Conditions (LOCA and RIA)

Fuel reliability

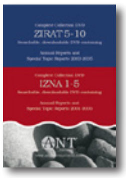


ZIRAT20 STR
Fuel Reliability

Microstructure of Zr alloys and effects on performance

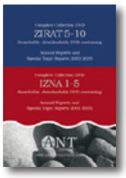
ZIRAT20 STR
Microstructure of Zirconium Alloys and Effects on Performance

ZIRAT Special Topic Reports on other issues:



ZIRAT8 STR
High Burnup Fuel Issues, their Most Recent Status

ZIRAT12 STR
Welding of Zirconium Alloys



ZIRAT10 STR
Impact of Irradiation on Material Performance

ZIRAT16 STR
Performance Evaluation of New Advanced Zr Alloys for PWRs/VVERs

ZIRAT17 STR
High Strength Nickel Alloys for Fuel Assemblies

ZIRAT17 STR
High Burnup Fuel Design Issues

ZIRAT19 STR
Hot Cell Post-Irradiation Examination Techniques for Light Water Reactor Fuels

A.N.T. INTERNATIONAL®
www.antinternational.com

Office Address: Advanced Nuclear Technology International, Analysvägen 5, SE-435 33 Mölnlycke, Sweden.
Phone: +46 (0)31-88 16 00. Fax: +46 (0)31-88 16 01. info@antinternational.com www.antinternational.com