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ANT International Reports

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AUGUST 2017

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The objective of the handbooks and reports on current issues is to provide guidance for those needing to get an introduction to and an initial understanding of the topics covered or to update and refresh the memory of those with materials/plant chemistry background. This group includes individuals ranging from young engineers, researchers, specialists to upper management. The handbooks and reports are written and explained in such a way that those not familiar with the topic can follow the report, and find and grasp the appropriate information. This means that the reports can be used by the organisation in the training of the internal staff with or without additional assistance from the ANT International staff. The handbooks and reports are offered to nuclear utilities, fuel vendors, research laboratories and regulatory agencies.

For more information and/or an offer welcome to contact
Mikaela Strand at mikaela.strand@antinternational.com

Please also visit our website for the latest updated information
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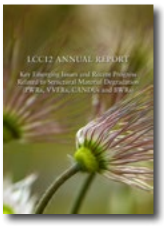
Previously published LCC Reports

Structural Material Degradation



Sample

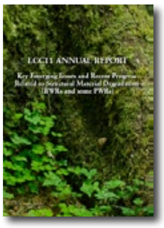
Key Emerging Issues and Recent Progress Related to Structural Material Degradation 2015 (LCC12 AR)

This Report discusses the PWR/VVER/CANDU/BWR highlights from the 17th International Conference on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, that was held in Ottawa, Canada in 2015.

Sample

Key Emerging Issues and Recent Progress Related to Structural Material Degradation in PWRs, VVERs and CANDUs 2014–2015 (LCC11 AR)

This Report reviews the key information presented at the “Fontevraud International conference” which was held in Avignon, France, in September 2014. In this conference, research work, related to field failures and issues of various reactor components, is also presented. This Report reviews the papers relevant to PWRs, some BWRs, VVERs and CANDU reactors.



Sample

Key Emerging Issues and Recent Progress Related to Structural Material Degradation (BWRs and some PWRs) 2014–2015 (LCC11 AR)

This Report reviews the key information presented at the “Fontevraud International conference” which was held in Avignon, France, in September 2014. Research work, related to field failures and issues of various reactor components, is also presented. This Report addresses the issues of BWRs along with the issues common to some BWRs and PWRs. In addition, this Report summarises papers in the Civil Engineering (concrete structural material degradation) track as well.

Sample

Key Emerging Issues and Recent Progress Related to Structural Material Degradation 2013–2014 (LCC10 AR)

This Report discusses the PWR/VVER highlights from the 16th International Conference on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, that was held in Ashville, NC in August 2013.

Sample

Operational Issues, Practices and Remedies (LCC9 AR)

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

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

Sample

Key Emerging Issues and Recent Progress Related to Structural Material Degradation (LCC8 AR)

This Report discusses the highlights from the 15th International Conference on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, which was held in Colorado Springs in August 2011.

Sample

Key Results from Recent Conferences in Structural Material Degradation in Water Cooled Reactors (LCC6 SR)

This Report reviews the key papers published in the following conferences:

1. The 14th International Conference on “Environmental Degradation of Materials in Nuclear Power Systems”, Virginia Beach, USA, August 24–28, 2009.
2. The Fontevraud 7 Conference “Contribution of Materials Investigations to Improve the Safety and Performance of LWRs”, Avignon, France, September 26–30, 2010.
3. Jubilee day seminar on “Stress Corrosion Cracking of Nickel Base Alloys at CEA – Coriou Effect”, Saclay, France, January 26, 2010.

Sample

Annual Report covering results published during 2008–2009 (LCC5 AR)

The Report covers the following topics:

1. PWR/VVER primary side coolant chemistry.
2. BWR coolant chemistry.
3. PWR and VVER secondary side chemistry.
4. Materials degradation.
5. Intergranular stress corrosion cracking and irradiation-assisted stress corrosion cracking of cold worked/irradiated stainless steels in de-oxygenated PWR-type coolants.

Sample

Annual Report covering results published during 2007–2008 (LCC4 AR)

The Report covers the following topics:

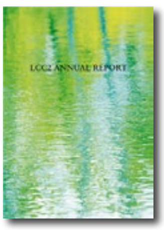
1. Nuclear plant primary water chemistry experience (PWR, VVER and BWR).
2. Water chemistry sampling and monitoring (PWR, VVER and BWR).
3. Corrosion product control and sampling technique (secondary side, PWR, VVER).
4. Material degradation management.
5. Fuel /water chemistry interaction.
6. Behaviour of radiolysis gases in BWRs and PWRs.

Sample

Annual Report covering results published during 2006–2007 (LCC3 AR)

The Report covers the following topics:

1. Secondary side steam generator tube problems.
2. Structural Materials Degradation.
3. Electrochemical membrane methods in LWR cleanup systems.
4. Effect of water chemistry on fuel cladding.

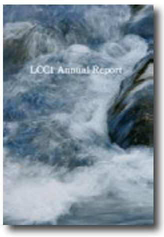


Sample

Annual Report covering results published during 2005–2006 (LCC2 AR)

The Report covers the following topics:

1. Coolant Quality and Control Issues – PWR/VVER Water Chemistry – BWR water chemistry.
2. Structural Materials Degradation.
3. Primary Circuit Corrosion (BWRs and PWRs) – SCC, PWSCC in PWRs – SCC in BWRs and remedies (HWC, NMCA).
4. Dose Rate Buildup and Control (BWRs and PWRs).



Sample

Annual Report covering results published during 2004–2005 (LCC1 AR)

The Report covers the following topics:

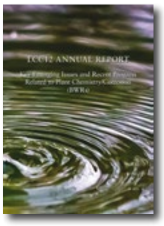
1. Coolant Quality and Control Issues – PWR Water Chemistry – BWR water chemistry.
2. Materials selection for the primary BWR and PWR circuits.
3. Primary Circuit Corrosion (BWRs and PWRs) – SCC, PWSCC in PWRs – SCC in BWRs and remedies (HWC, NMCA).
4. Dose Rate Buildup and Control.
5. Fuel/Water Chemistry Interaction.

Coolant Chemistry and Corrosion

Sample

Key Emerging Issues And Recent Progress Related to Plant Chemistry/Corrosion in PWR/VVER/CANDU, PHWR, and Auxiliary Systems) 2016 (LCC12 AR)

This Report cover the key information in PWR/VVER/CANDU Reactors presented at the 20th Nuclear Power Chemistry Conference, in October 2016. The Report not only summarises but also analyse the results to assess in which specific situation the results are applicable.



Sample

Key Emerging Issues and Recent Progress Related to Plant Chemistry/Corrosion in BWR Reactors, 2016 (LCC12 AR)

This Report cover the key information presented at the 20th Nuclear Power Chemistry Conference, in October 2016. The Report not only summarises but also analyse the results to assess in which specific situation the results are applicable.

Sample

Key Emerging Issues and Recent Progress Related to Plant Chemistry/Corrosion in PWR/VVER/CANDU Reactors 2014–2015 (LCC11 AR)

This Report is covering two technical aspects dealing with:

1. Recycling of waste in primary coolant and,
2. Secondary water chemistry options and Hide out return of impurities from Steam Generators (SG).

Sample

CRUD in PWR/VVER Coolant Volume II – Control of Crud in The PWR/VVER Coolant and Mitigation Tools (LCC11 STR)

The purpose of the Volume II of this Report is to describe the tools and their application to adequately control the coolant crud in order to improve the fuel and out-core radiation performance.

Sample

Key Emerging Issues and Recent Progress Related to Plant Chemistry/Corrosion in BWR Reactors (LCC10 AR)

This Report cover the key information presented at the Nuclear Power Chemistry Conference, in October 2014. The Report not only summarises but also analyse the results to assess in which specific situation the results are applicable.

Sample

CRUD in PWR/VVER Coolant, Volume I – Sources, Transportation in Coolant, Fuel Deposition and Radiation Build-up (LCC10 STR)

The topic is covered in two separate volumes. This document, Volume I, covers the following topics:

1. Introduction.
2. Sources of CRUD.
3. CRUD Transportation in PWR/VVER Coolant.
4. CRUD Deposition in the Core
5. CRUD Release from Core and Radiation Build-up Mechanism.

Sample

Key Emerging Issues and Recent Progress related to Plant Chemistry/Corrosion in PWR/VVER/CANDU Reactors (LCC10 AR)

This Report cover the key information presented at the Nuclear Power Chemistry Conference, in October 2014. The Report not only summarises but also analyse the results to assess in which specific situation the results are applicable.

Sample

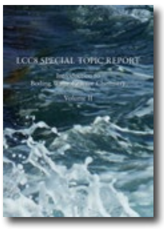
Effects of Coolant Chemistry on Fuel Performance (LCC9 STR)

The intent of this Report is to provide a state-of-the-art knowledge of the mechanisms of the various forms of Zr-alloy corrosion and hydrogen pickup (HPU) and how water chemistry impacts fuel performance, including corrosion and HPU. This knowledge will help implement actions to reduce Zr alloy corrosion and hydrogen pickup.

Sample

Introduction to Boiling Water Reactor Chemistry Volume I (LCC7 STR)

This Report is the first volume out of two focusing on BWR chemistry. The Report includes the basics of water radiolysis, the dynamic process that establishes the concentration of oxygen and hydrogen gas in the reactor water, the consequences of tramp uranium and leaking fuel rods on chemistry and other relevant topics.



Sample

Introduction to Boiling Water Reactor Chemistry Volume II (LCC8 STR)

This Report is the second volume out of two focusing on BWR chemistry. This volume II takes a closer look at corrosion of structural materials covering the following topics:

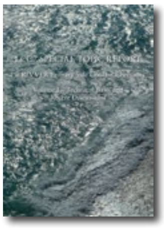
1. Corrosion Considerations.
2. Stress Corrosion Cracking Mitigation.
3. Start-up IGSCC Mitigation.
4. Shutdown Dose Rate Minimization.
5. Reactor Water Purity Transients.
6. Surveillance Programs.

Sample

PWR/VVER Primary Side Coolant Chemistry, Volume II – Water Chemistry Tool to Mitigate the Concerns (LCC8 STR)

This report is the second volume out of two focusing on PWR/VVER chemistry. Volume II covers the following topics:

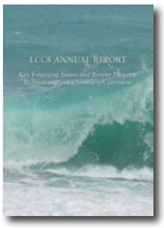
1. Chronology of Coolant Chemistry.
2. Current Coolant Chemistry Programs.
3. Coolant Chemistry Guidelines & Practices.
4. Purification and Filtration of the Coolant Monitoring Concept.



Sample

PWR/VVER Primary Side Coolant Chemistry, Volume I – Technical Basis and Recent Discussions (LCC7 STR)

This report is the first volume out of two focusing on PWR/VVER chemistry. This document is intended to provide a detailed description of the PWR/VVER Primary Side Coolant Chemistry. Furthermore, it should provide a strong support to the utilities for establishing a responsive plant specific chemistry program. It may also help the Manufacturers and Regulators at having a detailed approach of primary water chemistry and corresponding issues.



Sample

Key Emerging Issues and Recent Progress Related to Plant Chemistry/Corrosion (LCC8 AR)

This Report contains plant chemistry, corrosion issues and their recent progress from the following three conferences held in the year 2012:

1. The EPRI Steam Generator Secondary Side Management Conference.
2. BWR issues presented in the EPRI Conference.
3. The Nuclear Power Chemistry Conference.

Sample

Operational Issues and Practices in PWRs (LCC7 STR)

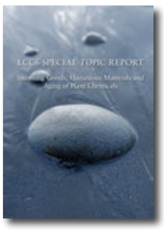
This Report combines topics of potentially important consequences:

1. Total Organic Carbon (TOC) in primary and secondary system water.
2. Best practices for corrosion products removal in reactor coolant systems (RCS) of VVERs and the use of mechanical filters or macroporous resins.
3. Best practices for the use of Ion Exchange Resins for RCS, SG blowdown, and the spent fuel pool.
4. Best practices to mitigate condenser leakages and recommendations for Operation procedures in case of large impurities ingress.
5. The Use of Dispersant to Mitigate Steam Generator Corrosion Product Deposition.

Sample

Effect of Zinc in BWR and PWR/VVER on Activity Build-Up, IGSCC and Fuel Performance (LCC6 STR)

This Report gives a comprehensive understanding of the zinc chemistry mechanism and information on how Zinc Chemistry in BWR and PWR plants was introduced in the plants and explains the results achieved.



Sample

Incoming Goods, Hazardous Materials and Aging of Plant Chemicals (LCC6 STR)

The objective of this Report is to provide a comprehensive understanding of controlling the quality of incoming goods in Nuclear Power Plants.

Sample

Radiochemistry in Nuclear Power Reactors (Light Water Reactors) Volume I (LCC6 STR)

The objective of this Report is to provide a comprehensive understanding of radiochemistry in Nuclear Power Plants which has a large impact on dose rates, operational exposures, maintenance activities, shutdown process, safety issues, environmental constraints and control of proper plant operation.

[Sample](#)

Start-up and Shutdown Practices in BWRs as well as in Primary and Secondary Circuits of PWRs, VVERs and CANDUs (LCC5 STR)

The Report provides a worldwide review of Startup and Shutdown Procedures both in the Primary and in the Secondary Circuit of PWRs, CANDUs and VVERs, and in the reactor coolant and main steam containing systems of BWRs.

[Sample](#)

Consequences of Power Uprating (LCC3 STR)

A relatively high percentage of all operating LWRs in the world have implemented or are considering some kind of increase of the power (power uprate). This Report provides a worldwide reviewof power uprates. Water chemistry and radiology experiences of power uprates are presented separated on BWRs and PWRs. Key issues are identified and discussed and available options to assure maintained or improved conditions at power uprates are presented.

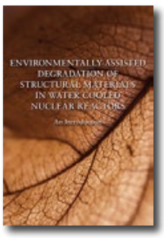
[Sample](#)

CRUD in PWR/VVER and BWR Primary Circuits (LCC2 STR)

The objective of this Report is to provide members with the basic understanding of the mechanisms involved and their relationship to material performance and activity buildup.

Stand Alone Reports LCC

Structural Material Degradation



[Sample](#)

Environmentally-Assisted Degradation of Structural Materials in Water Cooled Nuclear Reactors – An Introduction (SMDR)

The first Report in this series covers all different degradation modes of stainless steels and nickel base alloys.

[Sample](#)

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

The second Report in this series focuses on the degradation of carbon and low alloy steels.

[Sample](#)

Environmentally-Assisted Degradation of Stainless Steels in LWRs (EADS)

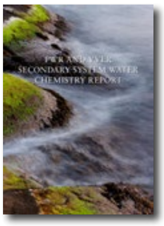
The third Report in this series covers degradation of austenitic stainless steels.1

[Sample](#)

Environmentally-Assisted Degradation of Nickel-Base Alloys in LWRs (EADN)

The fourth Report in this series is related to degradation of nickel-base alloys in LWR primary coolant systems.

Coolant Chemistry and Corrosion



[Sample](#)

PWR and VVER Secondary System Water Chemistry (SSWC)

Adequate secondary side water chemistry is crucial for safe and economical PWR and VVER plant operation. A good practised water chemistry control avoids degradation problems of the key components, such as steam generators (SG), carbon steel components and turbine. This Report gives a complete overview of the various rationale approaches to optimise the water chemistry according to the design and materials as well as the specific situation of each Nuclear Power Plant.

LCC™

LWR Chemistry and Component Integrity Programme

The annual LCC Programme is focused on reactor coolant, secondary chemistry and material issues and open to nuclear utilities, manufacturers and vendors, research and engineering organisations as well as regulatory agencies. In the LCC11 Programme, currently 17 organisations in North America and Europe, representing 21 nuclear plants, are members. The Programme was started in 2004.

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.
- Establish an independent meeting point for experts to enable free and critical discussions and experience exchange.

Find more information at: www.antinternational.com/lcc

Customer Feedback

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

MR. MICHAEL BOLZ
NPP Philippsburg

"A perfect balance between the theoretical knowledge and explanation of the fundamental facts and their practical application"

MR. JUAN DE DIOS SÁNCHEZ
Iberdrola

"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."

MS. JAYASHRI N. IYER
Westinghouse Electric Company

Previously published ZIRAT Reports

Manufacturing of Zr Alloys

[*Sample*]

Impact of Manufacturing Changes on Zr Alloy In-pile Performance (ZIRAT14/IZNA9 STR)

This Report provides an overview of Zr alloy manufacturing process and relationship between fabrication, microstructure and in-pile performance.

Sample

Manufacturing of Zr-Nb Alloys (ZIRAT11/IZNA6 STR)

The Report covers the impact of manufacturing on microstructure and in-reactor performance of Zr-Nb alloys. The Report was co-authored by Russian scientists involved in the development of the Russian Alloys E110, E635 and E125.

Sample

Manufacturing of Zirconium Alloy Material (ZIRAT5 STR)

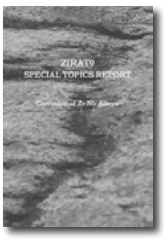
This special topical Report on zirconium alloy manufacturing and impact on in-pile fuel performance has been prepared within the ZIRAT5 Programme. The objective of this Report is to provide members with the basic understanding of the mechanisms involved and their relationship to fuel performance.

Corrosion and hydriding of Zr Alloys

Sample

Corrosion Mechanisms in Zirconium Alloys (ZIRAT12/IZNA7 STR)

The intent of this Report is to discuss the corrosion mechanisms and the various forms of corrosion features (uniform, nodular and shadow corrosion) in BWRs and PWRs/VVERs. This knowledge will give the possibility to implement actions to reduce corrosion.



Sample

Corrosion of Zr-Nb Alloys in PWRs (ZIRAT9/IZNA4 STR)

The Report gives an overview of the development of the Zr-Nb alloys in the nuclear industry, as well as the basic metallurgy and manufacturing of the Zr-Nb alloys. The Report focuses on the out-of-reactor as well as in-reactor corrosion and hydrogen pickup performance.

Sample

The Effects of Zn Injection (PWRs and BWRs) and Noble Metal Chemistry (BWRs) on Fuel Performance (ZIRAT8/IZNA3 STR)

This Report reviews the experience of the impact of Zn-injection (in PWRs and BWRs) and Nobel Metal Chemical Addition (NMCA) in BWRs on fuel integrity. The Report also discusses fundamentals in crud formation related to Zn-injection and NMCA.

Sample

Corrosion of Zirconium Alloys (ZIRAT7/IZNA2 STR)

This Report covers all aspects of Zirconium alloy corrosion and hydrogen pickup out-of-reactor as well as in-reactor.

Sample

Water Chemistry and Crud Influence on Cladding Corrosion (ZIRAT6/IZNA1 STR)

The purpose of the Report is to describe the fundamentals in crud formation, transport, and deposition, in order to provide a solid basis for evaluating and analysing any fuel operational problem due to crud deposits.

Properties of hydrides and impact on fuel in-reactor performance

Sample

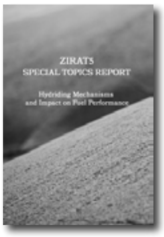
Effect of Hydrogen on Zirconium Alloy Properties – Vol I (ZIRAT13/IZNA8 STR)

The intent of these two Reports is to discuss the basics of hydrogen, hydrides and their effect on the zirconium alloy properties in Volume I and in-pile performance during normal and accident conditions as well as dry storage in Volume II.

Sample

Effect of Hydrogen on Zirconium Alloy Performance (Normal Operation, LOCA/RIA and Dry Storage) – Vol II (ZIRAT13/IZNA8 STR)

The intent of these two Reports is to discuss the basics of hydrogen, hydrides and their effect on the zirconium alloy properties in Volume I and in-pile performance during normal and accident conditions as well as dry storage in Volume II.



Sample

Hydriding Mechanisms and Impact on Fuel Performance (ZIRAT5/IZNA1 STR)

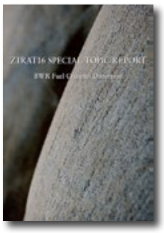
The objective of this Report is to provide the basic understanding of the hydrogen pickup and redistribution mechanisms involved and the impact of hydriding on fuel performance.

Dimensional changes of fuel assemblies/channels and components

Sample

Charged Particle Bombardment of Zirconium Alloys – A Review (ZIRAT19/IZNA14 STR)

This Report reviews the effect of ion irradiation (or ion bombardment, used interchangeably in this Report) on not only the standard dimensional stability topics but also the full range of properties that can be influenced by ion irradiation. The primary questions addressed are: What properties of zirconium alloys can ion irradiated simulate (emulate or substitute for) neutron irradiation? How, or to what extent, can ion bombardment complement neutron irradiation?



Sample

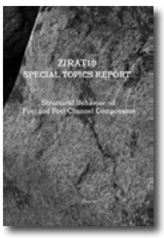
BWR Fuel Channel Distortion (ZIRAT16/IZNA11 STR)

This Report is primarily focused on channel distortion; that is, change of shape during in-reactor service: length, bow, bulge and twist. This Report addresses a broad range of channel features which can contribute to normal and problematic performance. In particular, the so-called "shadow corrosion induced" channel bow phenomena are analysed.

Sample

In-Reactor Creep of Zirconium Alloys (ZIRAT14/IZNA9 STR)

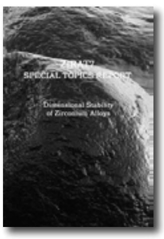
In this Report we have attempted to summarise data and mechanisms by addressing the variables that are known to affect irradiation creep. A number of conclusions are reached which give insight into the uses of creep data, mechanisms and models to assess in-service performance of reactor components.



Sample

Structural Behaviour of Fuel Channel Components (ZIRAT10/IZNA5 STR)

The objective of this Report is to provide a better understanding of the mechanisms behind the dimensional changes of structural materials in LWRs and CANDUs. Such improved understanding may improve both fuel reliability and reactor safety.



Sample

Dimensional Stability of Zirconium Alloys (ZIRAT7/IZNA2 STR)

This Report provides the understanding of the mechanisms of dimensional instability (including irradiation creep and growth) in the aggressive environment of the nuclear core and the potential effects on fuel performance.

Mechanical properties

Sample

Mechanical Testing of Zirconium Alloys – Volume I (ZIRAT18/IZNA13 STR)

This Report covers separate effects testing of Zr alloys.

Sample

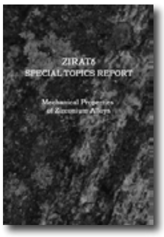
Mechanical Testing of Zirconium Alloys – Volume II (ZIRAT18/IZNA13 STR)

This Report covers integral or specific phenomena testing of Zr alloys

Sample

Pellet–Cladding Interaction (ZIRAT11/IZNA6 STR)

The objective of this Report is to give a better understanding of the PCI mechanisms, pertinent methods to test PCI performance and PCI remedy technology may improve fuel reliability and reactor safety.

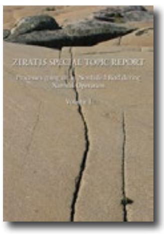


Sample

Mechanical Properties of Zirconium Alloys (ZIRAT6/IZNA1 STR)

The objective of this Report is to provide the basic understanding of the mechanical properties of the fuel components that are crucial for the fuel in-pile performance.

Fuel Pellet (UO2 and MOX) Performance during normal operation and accident conditions



Sample

Processes going on in Nonfailed Rod during Normal Operation (ZIRAT15/IZNA10 STR)

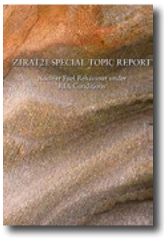
With increased burnup, processes going on inside a nonfailed fuel rod decrease pellet density, lower the fuel melting temperature and thermal conductivity, increase the fission gas release. Also, a high burnup structure is formed at pellet average burnups in excess of about 50 MWd/kgU. This Volume I of the Report describes the processes going on inside a nonfailed fuel rod during normal operation.

Sample

Processes Going on in Nonfailed Rod during Accident Conditions (LOCA and RIA) (ZIRAT15/IZNA10 STR)

This Report describes the fuel performance during accident conditions (RIA and LOCA). The associated Volume I of the Report describes the processes going on inside a nonfailed fuel rod during normal operation.

Fuel Performance during accident conditions (LOCA and RIA)



Sample

Nuclear Fuel Behaviour under RIA Conditions (ZIRAT21/IZNA16 STR)

This Special Topic Report (STR) give insights and understanding of the parameters impacting the fuel RIA performance and reviews the applicability of the data to high burnup fuel cladding.

Sample

Processes Going on in Nonfailed Rod during Accident Conditions (LOCA and RIA) (ZIRAT15/IZNA10 STR)

This Report describes the fuel performance during accident conditions (RIA and LOCA). The associated Volume I of the Report describes the processes going on inside a nonfailed fuel rod during normal operation.

Sample

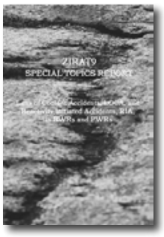
Effect of Hydrogen on Zirconium Alloy Properties – Vol I (ZIRAT13/IZNA8 STR)

The intent of these two Reports is to discuss the basics of hydrogen, hydrides and their effect on the zirconium alloy properties in Volume I and in-pile performance during normal and accident conditions as well as dry storage in Volume II.

Sample

Effect of Hydrogen on Zirconium Alloy Performance (Normal Operation, LOCA/RIA and Dry Storage) – Vol II (ZIRAT13/IZNA8 STR)

The intent of these two Reports is to discuss the basics of hydrogen, hydrides and their effect on the zirconium alloy properties in Volume I and in-pile performance during normal and accident conditions as well as dry storage in Volume II.



Sample

Loss of Coolant Accidents, LOCA, and Reactivity Initiated Accidents, RIA, in BWRs and PWRs (ZIRAT9/IZNA4 STR)

This Report gives insight and understanding of the LOCA and RIA issues and reviews the applicability of the relevant data. Specifically, the background data to the current (non-revised) LOCA criteria are discussed.

Fuel Reliability

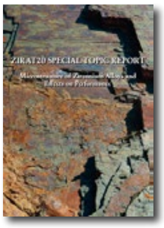
Sample

Fuel Reliability (ZIRAT20/IZNA15 STR)

The Report provides information on how to increase fuel reliability by:

1. Reducing the primary fuel failure frequency and minimize the consequences of fuel failures when they occur and
2. Minimizing operational effects due to factors such as fuel assembly and channel bowing, that can affect thermal margins (LOCA, DNB, Dryout) and core control capabilities (control rod insertion).

Microstructure of Zr alloys and effects on performance



Sample

Microstructure of Zr alloys and effects on performance (ZIRAT20/IZNA15 STR)

This Report provides examples illustrating how microstructures are modified during fabrication and service and how these microstructures affect in-reactor properties.

Other topics

Sample

Hot Cell Post-Irradiation Examination Techniques for Light Water Reactor Fuels, Vol II (ZIRAT21/IZNA16 STR)

This Report is a complement to the previous report on the same topic published within the ZIRAT19/IZNA16 Programme (Vol. I). The Report:

1. gives an overview of post-irradiation examination (PIE) and inspection techniques for CANDU nuclear fuel and other zirconium alloy components used in reactors;
2. discusses PIE techniques along with real world examples of in-reactor microstructural changes and impact on material behaviour;
3. provides information on PIE capabilities of some of the major hot cell facilities.

Sample

Hot Cell Post-Irradiation Examination Techniques for Light Water Reactor Fuels, Vol I (ZIRAT19/IZNA14 STR)

This Report provides an overview about the status of post-irradiation examination (PIE) and inspection techniques for nuclear fuel and their applications for analysis of material degradation during fuel operation in a reactor core. Emphasis is given to advanced non-destructive and destructive PIE techniques applied to LWR fuel rods and bundle hardware. The objective of this Report is to provide this knowledge.

[Sample]

High Burnup Fuel Design Issues and Consequences (ZIRAT17/IZNA12 STR)

This Report reviews the potential consequences of increased burnup on fuel. Recommendations are also given on how to remedy the high burnup issues.

[Sample]

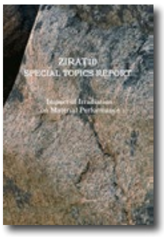
High Burnup Fuel Issues, their Most Recent Status (ZIRAT8/IZNA3 STR)

This Report reviews the potential consequences of increased burnup on fuel.

[Sample]

High Strength Nickel Alloys for Fuel Assemblies (ZIRAT17/IZNA12 STR)

This Report is a comprehensive review and evaluation of the high strength nickel alloys (625, 718, X-750 and A286) used for PWR/VVER and BWR fuel assembly components.



[Sample]

Impact of Irradiation on Material Performance (ZIRAT10/IZNA5 STR)

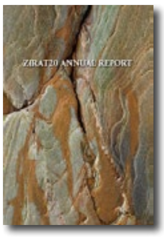
This Report focuses on the behaviour of zirconium alloys used for the main structural components in the fuel bundle and should provide a solid understanding of the role that irradiation plays in component performance. This Report is a complementary companion to the Structural Behaviour of Fuel Components.

Annual Reports presenting key results published during 18 months

[Sample]

ZIRAT21/IZNA16 Annual Report

Covering the Zr related results published during 2015–2016



[Sample]

ZIRAT20/IZNA15 Annual Report

Covering the Zr related results published during 2014–2015

Sample

ZIRAT19/IZNA14 Annual Report

Covering the Zr related results published during 2013–2014

Sample

ZIRAT18/IZNA13 Annual Report

Covering the Zr related results published during 2012–2013

Sample

ZIRAT17/IZNA12 Annual Report

Covering the Zr related results published during 2011–2012

Sample

ZIRAT16/IZNA11 Annual Report

Covering the Zr related results published during 2010–2011

Sample

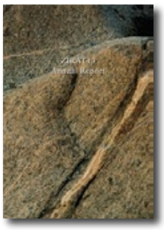
ZIRAT15/IZNA10 Annual Report

Covering the Zr related results published during 2009–2010

Sample

ZIRAT14/IZNA9 Annual Report

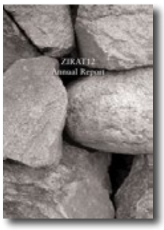
Covering the Zr related results published during 2008–2009



Sample

ZIRAT13/IZNA8 Annual Report

Covering the Zr related results published during 2007–2008



Sample

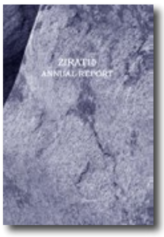
ZIRAT12/IZNA7 Annual Report

Covering the Zr related results published during 2006–2007

Sample

ZIRAT11/IZNA6 Annual Report

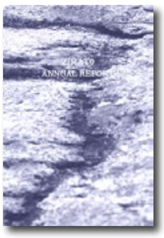
Covering the Zr related results published during 2005–2006



Sample

ZIRAT10/IZNA5 Annual Report

Covering the Zr related results published during 2004–2005



Sample

ZIRAT9/IZNA4 Annual Report

Covering the Zr related results published during 2003–2004

Sample

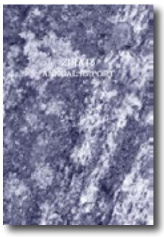
ZIRAT8/IZNA3 Annual Report

Covering the Zr related results published during 2002–2003

Sample

ZIRAT7/IZNA2 Annual Report

Covering the Zr related results published during 2001–2002



Sample

ZIRAT6/IZNA1 Annual Report

Covering the Zr related results publis-hed during 2000–2001

Sample

ZIRAT5 Annual Report

Covering the Zr related results published during 1998–2000

Stand Alone Reports ZIRAT

Sample

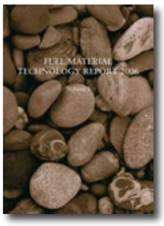
Fuel Design Review Handbook (FDRH)

This handbook intends to provide a guide to the items that have the greatest influence on fuel performance and prioritise the audits that are recommended.

Sample

Fuel Fabrication Process Handbook (FFPH)

The objective of the Fuel Fabrication Process Handbook is to provide guidance for a cost effective audit which uses audit time on areas which are most likely to affect the performance of the PWR/VVER and BWR fuel. The FFPH focuses on a “Process Audit” procedure, the audit of the fabrication process parameters for making high quality fuel.



Sample

Fuel Material Technology Report (FMTR) Vol I

This first volume covers short general outlines of the designs of PWR, BWR, CANDU, VVER and RBMK reactors but with focus on the corresponding fuel assembly and supporting structure designs together with the rational for the selection of the materials used in the different applications. This volume also gives an overview of the in-reactor fuel performance, of fuel performance codes and the manufacturing process of the fuel assembly.

Sample

Fuel Material Technology Report (FMTR) Vol II

In this second volume, the following topics for different water cooled type reactors are covered:

1. Effect of radiation on the various fuel assembly materials as well as the interaction between materials and cooling water chemistry in this radioactive environment.
2. Water chemistries in different reactor types.
3. Fuel design criteria and operating limits
4. Fuel performance during Loss Of Coolant Accidents and Reactivity Initiated Accidents.
5. Fuel performance during intermediate storage.
6. Trends and potential burnup limitations.

Sample

Control Assembly Technology Report (FMTR) Vol III

The Report constitutes Volume III of the series of Fuel Material Technology Reports. It describes the designs, manufacturing, performance and issues related to BWR/PWR/VVER/CANDU Control Assemblies with Ag-In-Cd (AIC), B4C, Hf absorber materials and stainless steel structural materials.

Sample

Fuel Material Technology Report (FMTR) Vol IV

The primary objective of this volume is to provide guidance in improving fuel reliability. To reach this objective, various Poolside and Hot Cell Examinations techniques may be used. A good knowledge of the pros- and cons- of the different techniques can guide the utility or fuel vendor to select the most cost efficient techniques for their specific goals. A second objective of this Report is to document this knowledge in a form which can be updated as new information, and methods become available.

Sample

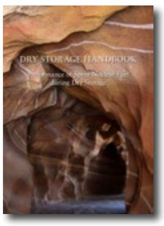
Management of BWR Control Rods (MBCR)

The objective of this Report is to provide the customers with information which will be a useful tool for management of control rods.

Sample

PWR Zr Alloy Cladding Water Side Corrosion (PZAC)

The Report discusses the different parameters impacting PWR fuel corrosion and provides a computer code which allows an equivalent comparison of new alloys irradiated in different reactors at different conditions. The computer code may also assist in identifying the mechanism why the corrosion rate starts to accelerate under certain conditions.



Sample

Dry Storage Handbook – Fuel Performance in Dry Storage

This handbook contain a technical assessment of the expected performance of spent nuclear fuel (SNF) during extended dry-storage time periods and the condition of such fuel at the end of dry storage. The principal focus of the reviews is on SNF and the effects of dry storage rather than on dry-storage containers and the related storage facilities. The objective is to provide background information on the likely behaviour of materials comprising water reactor fuel assemblies and on the performance of integral assemblies under conditions typical of dry storage for extended intervals of time.

ZIRAT™

Zirconium Alloy Technology Programme

The annual ZIRAT Programme is open to nuclear utilities and laboratories world-wide. The overall objective of the ZIRAT Programme is to enable the Members to:

- Identify potential zirconium alloy material advantages and limitations as well as suggest solutions for current issues.
- Gain increased understanding of material behaviour related to successful core operation.

ZIRAT is an annual programme, different from other national or international fuel materials related programmes because it offers to the member on a yearly basis the only in-depth comprehensive, periodic, timely review and analysis of all the new zirconium technology data in the public domain.

Find more information at: www.antinternational.com/zirat

Customer Feedback

"We find ZIRAT a great help and, in a way, it's like having an extra person in the team "

MR. TED DARBY
Rolls-Royce

"A good opportunity to learn issues which are very close to practical application"

MR. FRANK HOLZGREWE
BKW

"The ZIRAT meeting exceeded my expectations. The most valuable component of the meeting was the ability to speak directly with the ANT International experts. The depth of knowledge and understanding of all speakers in the ANT network is terrific. The size of the meeting was perfect."

MS. JAYASHRI N. IYER
Westinghouse Electric Company

IZNA™

Information on Zirconium Alloy Technology Programme

The annual IZNA Programme is open to fuel vendors, regulators and research organisations. The IZNA deliverables are similar to those of the ZIRAT Programme, however no annual Seminar is provided. A tailored seminar on site is provided at an additional fee. IZNA customers are fuel vendors, research laboratories and regulatory agencies world wide.The overall objective of the IZNA Programme is to enable members to:

- Identify potential zirconium alloy material problems and solutions related to nuclear fuel.
- Gain increased understanding of material behaviour related to successful core operation.

The IZNA Programme annually reviews the latest information on zirconium alloy technology. The review is the only complete, annually updated references on non-proprietary zirconium alloy technology in the industry. The information is organised by well established topics such as alloy research, corrosion, mechanical properties, dimensional stability, potential high burnup limitations etc.

Find more information at: www.antinternational.com/zirat

Customer Feedback

"My company joined the IZNA Programme in year 2003 starting with the IZNA1 and IZNA2. Since that year until now, products offered by ANT International have been one of the main source of information for the training of our new engineers."

MS. CRISTINA MUÑOZ-REJA RUIZ
Enusa

"The Seminar was attended by members of GNF's Materials Technology and Fuel Reliability team as well as researchers interested in Zircaloy corrosion at GE's Global Research Center."

DR. YANG-PI LIN
Global Nuclear Fuel

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