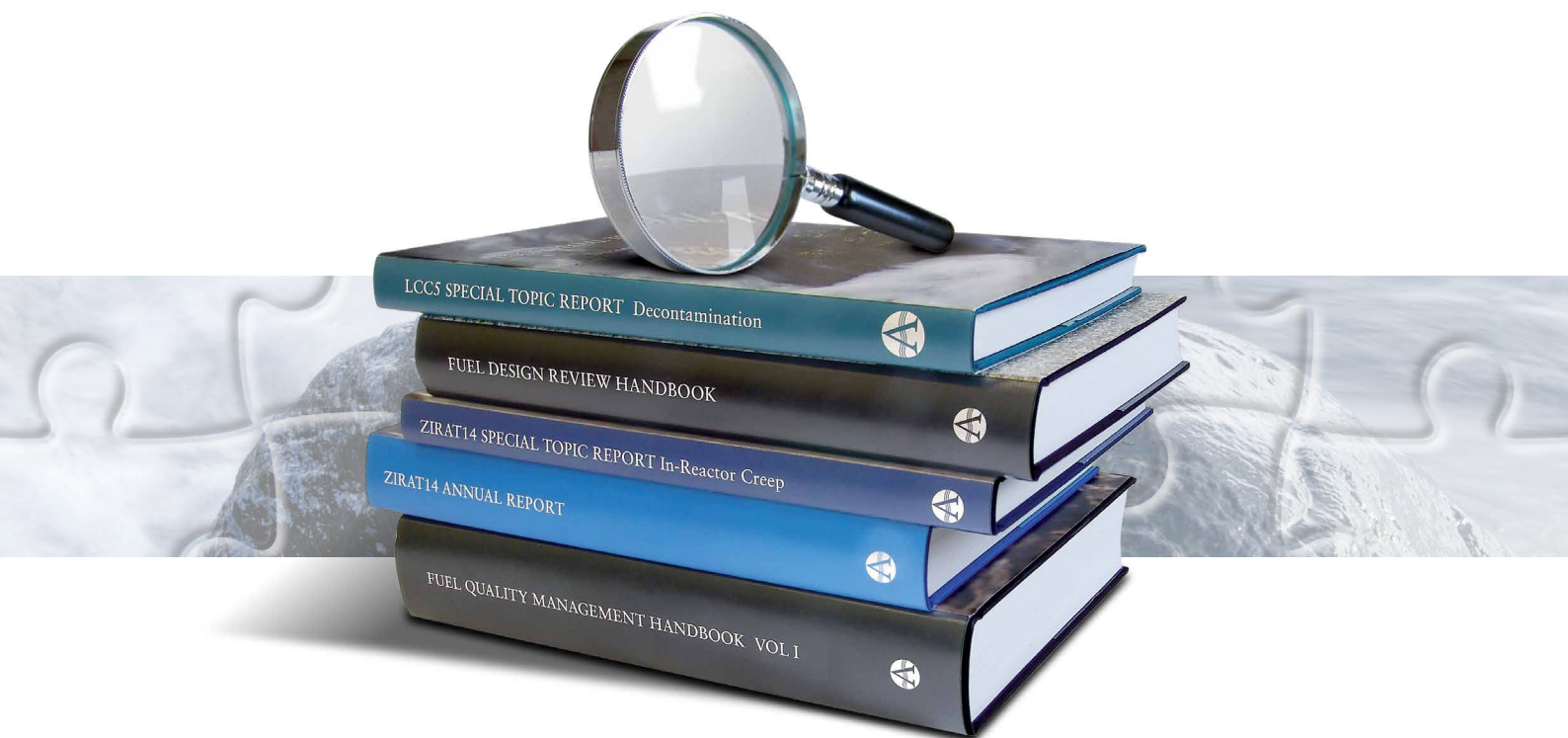




# ANT International Nuclear WIKI (AWIKI)


---



Within the ZIRAT, IZNA and LCC Programmes more than 55 Reports including more than 1 300 articles have been published up to now. To quickly find the information you are looking for, ANT International has developed a Nuclear Wikipedia, AWIKI, based upon the ZIRAT, IZNA and LCC information.

The AWIKI has similar structure as the Wikipedia with a powerful search tool to quickly retrieve the information you are interested in. The AWIKI will also direct you (via links) to information/reports with much more details. For each topic, all the relevant references are listed as well as suggestions on additional information contained in the ZIRAT/IZNA/LCC Reports. The additional information is provided as pdf-files at the bottom of the topic to facilitate fast and easy access to the relevant information.

The AWIKI is continuously updated with the most recent information.



# Boosting your Excellence through Knowledge and Training

[Page](#)
[Discussion](#)
[View Source](#)
[History](#)
[Watch](#)

[my preferences](#)
[my watchlist](#)
[my contributions](#)
[log out](#)

**NAVIGATION**  
[Main page](#)  
[Content Directory](#)  
[Recent changes](#)  
[Random page](#)  
[Help](#)  
**SEARCH**

**TOOLBOX**  
[What links here](#)  
[Related changes](#)  
[Upload file](#)  
[Special pages](#)  
[Printable version](#)  
[Permanent link](#)

## Category:Content Directory

### Structural Material, Fuel Technology and Coolant Chemistry

(previous 200) (next 200)

#### Pages in category "Content Directory"

The following 200 pages are in this category, out of 1,362 total.

**A**

- "KnaIgas" explosion experience
- 800 NiQ
- AFA 3G
- AREVA NP
- AREVA NP Fuel Assembly Design
- AREVA NP fuel reliability data
- ASTM Standard B363 for Zirconium and its Alloys
- ATREIM
- AVI IS
- Absorber
- Absorption Reaction
- Accretion of PWR corrosion
- Accreditation
- Accuracy and Precision
- Actinide
- Actinium
- Activated corrosion products in BWRs
- Activation Energy - General
- Activity
- Activity build-up
- Activity, specific activity and activity concentration
- Advanced Boiling Water Reactors (ABWR)
- Age Hardening - General
- Aging of plant chemicals during storage
- Alkali Metal
- Alkaline Earth Metal
- Alloy
- Alloy 600 MA (Inconel 600 MA)
- Alloy 600 TT
- Alloying Element
- Alpha Decay
- Alpha Particle
- Alpha Stabilizer (of Zirconium)
- Alpha radiation measurement
- Alternative anodes - PWR and VVER secondary side
- Aluminium

**A cont.**

- Axial Power Shaping Rod
- Axial offset anomalies/Crud induced power shift

**B**

- BU achievements and key fuel performance issues
- BU limitations
- BWR CLO and CRUD related fuel rod corrosion failures
- BWR Channel Assembly Fabrication
- BWR Control Rod Design criteria
- BWR Control Rods release of inventory
- BWR Control rod cladding material
- BWR control rod designs
- BWR control rod management
- BWR coolant chemistry
- BWR description and coolant chemistry
- BWR fuel channel box - material modifications to remedy shadow induced bow
- BWR fuel channel design and manufacturing
- BWR fuel channel design criteria
- BWR fuel channel lifetime limits
- BWR fuel channel performance
- BWR fuel performance with zinc injection
- BWR primary water chemistry experience in German plants
- BWR primary water experience in Scandinavian plants
- BWR sampling systems and equipments
- Baffle
- Ball Mill
- Bar
- Bar Stock Production
- Bath
- Batter Fuel Cladding
- Basal Plane
- Basic deformation concepts
- Basic impact of irradiation on fuel materials
- Basic metallurgy of reactor structural materials
- Basics of radiolysis and its effect on corrosion potential in presence and absence of added hydrogen of stainless steels
- Basketweave
- Batch
- Behaviour of radiolysis gases in BWRs and PWRs - Explosion of gases - KnaIgas
- Bending
- Bending Moment
- Bending Stress


**B cont.**

- Boiling Water Reactor
- Boiling Water Reactor (BWR) designs
- Boiling Water Reactor (long)
- Bohmann Equation
- Blending Fuel-Clad
- Boric Acid Treatment (BAT) -secondary side PWR and VVER
- Boron
- Boron Carbide
- Brazers Lattice
- Brazing
- Breeder Reactor
- Breeding
- Bottle Material
- Brookhaven Report
- Buckling
- Bulk Temperature
- Bulk boiling
- Buoyancy
- Burgers Vector
- Burnable Absorber
- Burnable Absorbers (2nd article)
- Burnable Absorbers (3rd article)
- Burnable Absorbers (4th article)
- Burnable Poison
- Burnup
- Burst Strength
- Burst Test
- Burst testing

**C**

- CAN-ULHLM
- CANDU Fuel Assembly Design
- CANDU Pressure Tubes Performance Issues
- CANDU Reactors (long)
- CANDU fuel reliability data
- CANDU reactor start up and shut down experience
- CIPS
- CITROX
- CORD
- CRUD
- CRUD Induced Power Shift

After accessing the AWIKI, the different topics of information are listed.



# Boosting your Excellence through Knowledge and Training

[Page](#)
[Discussion](#)
[View Source](#)
[History](#)
[Watch](#)

[my preferences](#)
[my watchlist](#)
[my contributions](#)
[log out](#)

**NAVIGATION**  
[Main page](#)  
[Content Directory](#)  
[Recent changes](#)  
[Random page](#)  
[Help](#)  
**SEARCH**

**TOOLBOX**  
[What links here](#)  
[Related changes](#)  
[Upload file](#)  
[Special pages](#)  
[Printable version](#)  
[Permanent link](#)

## Post irradiation creep of Zr alloys - dry storage

Contents [hide]

- 1 General
- 2 Saturated primary creep strain
- 3 Effect of the hydrogen content
- 4 Effect of irradiation
- 5 Effect of recovery of irradiation damage
- 6 Allowable creep strain
- 7 Considerations regarding creep
- 8 See also
- 9 Further Reading
- 10 Notes & References

**General**

Long term interim storage is currently considered as a major way for managing SNF. Because of the envisaged long storage duration and the cladding temperature and stress conditions, long term creep is a relevant deformation mechanism, which could potentially lead to a rupture of the cladding.

During dry storage the fuel rod cladding may experience a significant hoop stress loading due to the rod's internal gas pressure from the fission gases released during irradiation, the initial helium charge and the decreasing void volume. This gas pressure depends not only on the FGR but also on the average gas temperature. During vacuum drying after loading of the fuel into the cask, temperatures of 360-500 °C may exist for hours depending on the drying procedure. During long time storage the temperature will decrease from an initial value of 300-400 °C to significantly lower values. The cladding hoop stress will also decrease as consequence of the decreasing temperature of the internal gas during this period. The initial value, which is different for different fuel rods, may be in a range up to a 90-120 MPa.

The state of knowledge on creep of Zr alloys has been reviewed previously by [Johnson & Rudling, 2001/2002<sup>[1]</sup>]. The factors, which may affect creep under dry storage condition are:

- Thermal creep
- Effect of hydrogen on thermal creep
- Effect of irradiation hardening on thermal creep
- Irradiation damage recovery during long-term dry storage.

Estimation of cladding creep strain under long-term dry storage condition for tens of years is in most cases performed by using short-term creep data. From the available experimental data different creep prediction equations have been developed.

The creep behaviour of unirradiated Zr alloy cladding tubes (out-reactor creep) were investigated in much detail in numerous programs. Figure 9.8a and Figure 9.8b show the results from creep tests on Zircaloy-4 internally pressurized cladding tube samples, as reported by [Spilaker et al, 1997<sup>[2]</sup>].

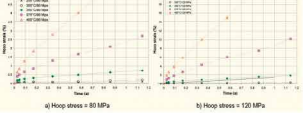




Figure 9.8: Hoop stress of internally pressurized dry-Zr cladding tubes, Material 1 [Spilaker et al, 1997<sup>[2]</sup>].

In general different phenomenological creep models have been applied to describe the creep behaviour for Zircaloy cladding under long time storage. These models express the creep strain as an algebraic equation comprised of individual terms each of which describes the contribution of a single independent variable, namely, stress, temperature, fast neutron fluence and time.

The time (t) dependency of total creep strain (ε) at stresses below yield stress is usually described by a primary strain (ε<sub>p</sub>) and a steady state creep rate (dε/dt<sub>ss</sub>):


**Eq. 9.2**

$$\epsilon = \epsilon_p + \frac{d\epsilon}{dt}_{ss} \cdot t$$

or considering the low strain, usually less than 1%, by a power law:

This gives an example of information about Post irradiation creep of Zr alloys – dry storage.

2



# Boosting your Excellence through Knowledge and Training

[Home](#)
[Log out](#)
[my preferences](#)
[my watchlist](#)
[my contributions](#)
[log out](#)

[Main page](#)
[Content Directory](#)
[Recent changes](#)
[Random page](#)
[Help](#)

[What links here](#)
[Related changes](#)
[Upload file](#)
[Special pages](#)
[Printable version](#)
[Permanent link](#)

## PWR reactor coolant chemistry

[Contents \[hide\]](#)

- 1 Background information
- 2 Recent discussions to improve coolant chemistry
  - 2.1 Control of material degradation
    - 2.1.1 Dissolved hydrogen
      - 2.1.1.1 PWSCC (Crack Growth Rate) (CGR)
      - 2.1.1.2 PWSCC crack initiation
      - 2.1.1.3 Discussions and recommendations
        - 2.1.1.3.1 Discussions
        - 2.1.1.3.2 Recommendations
    - 2.1.2 Lithium and coolant pH value
      - 2.1.2.1 PWSCC crack initiation
      - 2.1.2.2 PWSCC CGR
      - 2.1.2.3 ASCC in stainless steels
      - 2.1.2.4 Discussions and recommendations
        - 2.1.2.4.1 Discussions
        - 2.1.2.4.2 Recommendations
    - 2.1.3 Zinc
      - 2.1.3.1 PWSCC crack initiation
      - 2.1.3.2 PWSCC CGR
      - 2.1.3.3 Discussions and recommendations
        - 2.1.3.3.1 Discussions
        - 2.1.3.3.2 Recommendations
  - 2.2 Radiation field control
    - 2.2.1 Dissolved hydrogen
      - 2.2.1.1 Field experience
        - 2.2.1.1.1 Tongue-2 investigation
        - 2.2.1.1.2 Information from Sata PWR plant
        - 2.2.1.1.3 Bzau investigation
      - 2.2.1.2 Discussions and recommendations
        - 2.2.1.2.1 Discussions
        - 2.2.1.2.2 Recommendations
    - 2.2.2 Lithium an coolant pH value
      - 2.2.2.1 Field experience with different lithium/HT control strategies
      - 2.2.2.2 Discussions and recommendations
        - 2.2.2.2.1 Discussions
        - 2.2.2.2.2 Recommendations
    - 2.2.3 Zinc addition
      - 2.2.3.1 Discussions based on recent field observations
      - 2.2.3.2 Recommendations
  - 2.3 Conclusive summary
  - 2.4 Recommendations
    - 2.4.1 PWR plants with nickel base alloys sensitive to PWSCC in RCS
    - 2.4.2 PWR plants with iron base alloys or nickel base alloys non-sensitive to PWSCC in RCS


[3 See also](#)
[4 Further Reading](#)
[5 Notes & References](#)

### Background Information

Demineralized water is used as reactor coolant in the PWR plants to moderate the fast neutrons to thermal neutrons, which are needed for the nuclear fission reaction to produce energy in the core. Another function of the reactor coolant is to transport the heat of nuclear fission energy produced in the core to SGs. In addition, boric acid is added to the reactor coolant as chemical shim to control the core reactivity. Based on field experience gained during the early PWR operation in the 1950s at several research and power plants, an operation with reactor coolant without further chemical treatment may result in several serious problems with respect to safe plant operation, radiation field control and compatibility of the structural materials. This is due to the fact that coolant is exposed to radiation field in the core, where it decomposes by radiolysis. As a result of this, radiolysis products are generated, some of which are extremely strong oxidants and can jeopardize the compatibility of the structural materials by corrosion. Oxidizing conditions can cause enhanced fuel cladding corrosion or PWSCC in Alloy 600MA that is used as SG tubing material or in numerous reactor penetrations in many PWR plants designed and constructed by US vendors or their licensees worldwide. In addition oxidizing conditions enhance Corrosion Product (CP) (so called "crud") deposition on fuel rods, which may cause crud induced fuel clad corrosion and

Authored by: Susat Odeh  
Date: December, 2011  
Reviewed/Edited by:  
Date:

Here is an example of the information related to PWR reactor coolant chemistry.



# Boosting your Excellence through Knowledge and Training

[Home](#)
[Log out](#)
[my preferences](#)
[my watchlist](#)
[my contributions](#)
[log out](#)

[Main page](#)
[Content Directory](#)
[Recent changes](#)
[Random page](#)
[Help](#)

[What links here](#)
[Related changes](#)
[Upload file](#)
[Special pages](#)
[Printable version](#)
[Permanent link](#)

## Intergranular stress corrosion cracking (IGSCC) and irradiation-assisted stress corrosion cracking (IASCC) of cold worked/irradiated SSs in de oxygenated PWR-type coolants

[Contents \[hide\]](#)

- 1 Introduction
- 2 Austenitic SSs – effects of cold work
  - 2.1 Operating experience
  - 2.2 Laboratory tests
- 3 Austenitic SSs – irradiation effects on stress corrosion cracking susceptibility
  - 3.1 Irradiation damage
    - 3.2 Operating experience of irradiation-assisted stress corrosion cracking (IASCC) in PWRs
    - 3.3 Test reactor and hot laboratory IASCC testing
      - 3.3.1 Characterization of grain boundaries in irradiated austenitic SSs and IASCC susceptibility
      - 3.3.2 Irradiation-induced hardening and IASCC susceptibility
      - 3.3.3 Parametric studies by SIRT of IASCC in highly irradiated austenitic SSs
      - 3.3.4 Parametric studies at constant stress of IASCC initiation in highly irradiated austenitic SSs
      - 3.3.5 Parametric studies of IASCC growth rates in highly irradiated austenitic SSs
    - 3.4 Helium embrittlement
    - 3.5 Hydrogen embrittlement
  - 3.2 Irradiation-induced hardening and IASCC susceptibility
  - 3.3 Parametric studies by SIRT of IASCC in highly irradiated austenitic SSs
  - 3.4 Parametric studies at constant stress of IASCC initiation in highly irradiated austenitic SSs
  - 3.5 Parametric studies of IASCC growth rates in highly irradiated austenitic SSs
- 4 See also
- 5 Further Reading
- 6 Notes & References

### Introduction

Type 304L and 316L SSs (Table 6.1) are the main materials used for primary coolant piping and other components exposed to primary coolant in PWRs (and BWRs). Cast SSs with similar compositions (designated CF-3, CF-3M and CF-8) have also been widely used for large diameter primary piping, elbows and nozzles in primary circuits. Operating experience with respect to environmental degradation of all these low strength materials in PWRs since the mid-1960s has generally been excellent. The only major concerns that have been raised are with thermal aging and embrittlement of some of the cast austenitic grades and with the effects of irradiation on core support structures, as described in Section 6.3.2 in LCSS Annual Report @.

Where higher strength is necessary (for bolts, springs, valve stems etc.), precipitation hardened martensitic SSs, or martensitic SSs, or precipitation hardened martensitic SSs have been used. Environmentally induced cracking of these materials has generally been attributed to excessive hardness and strength on entering service, or in the case of precipitation hardened martensitic SSs, to thermal ageing and hardening in service. Operating experience and associated laboratory studies of these materials are reviewed in the LCSS STR.

### Austenitic SSs – effects of cold work

#### Operating experience

Since the widespread problems that occurred with IGSCC in thermally sensitized weld heat affected zones of austenitic SSs in BWRs in the 1970s [Ford 2006]@, the normal fabrication practice has been to use low carbon grades of Types 304 and 316 SSs, i.e. Types 304L and 316L, in both PWRs and BWRs. An alternative adopted in some countries has been to use niobium or titanium stabilized grades such as, respectively, Type 347 or Type 321 that are very resistant to sensitization because the carbon is trapped as stable niobium or titanium carbides. Nevertheless, there is little doubt that in many older PWRs sensitized Type 304 and 316 SSs exist in considerable quantities but practical experience shows that de-oxygenated, hydrogenated PWR primary water does not cause stress corrosion cracking, in contrast to BWR experience with oxygenated NWC coolant. The reason is clearly related to the presence of dissolved hydrogen in PWR primary circuits, which ensures that any radiolytic decomposition products of water are efficiently scavenged so that overall there is no net decomposition of water, unlike in BWRs. Thus, corrosion potentials of all structural materials in PWR primary circuits are close to the hydrogen/water equilibrium potential and well below the desired protection potential identified for thermally sensitized SSs in BWRs. The exceptions in PWRs primarily concern certain dead legs where air bubbles may be trapped during refuelling, see Section 6.2 in LCSS Annual Report @.

More recently, however, concerns have emerged concerning possible stress corrosion susceptibility of cold worked SSs, even at low corrosion potentials in flowing, normal quality PWR primary water. [Elevare et al 2007/2008]@ One of the most recent incidents to fuel these concerns was a leak of primary water that was clearly due to IGSCC of a Type 316 pressurizer heater tube at the Bradwood Unit 1 PWR, [Chynoweth & Hyres 2007]@. Destructive examination showed that through-wall circumferential cracking had developed from the ID in the heat-affected zone of a socket weld, which appeared to be sensitized. In addition, the internal surface was heavily cold worked. Very recently, a single shallow circumferential intergranular crack was discovered in the heat affected zone of a Type 316 safe end of a dissimilar metal weld between a SG and the primary water inlet piping at Mihama 2 ducting intervention whose primary purpose was to apply a remedial surface treatment to the nickel-based Alloy 152 weld metal. [JAWTI 2008]@. On the other hand, no cracking was observed on the primary water coolant side of the Type 308 SS weld overlay cladding of a PWR pressure vessel despite dynamic straining caused by bulging of the cladding following loss of the low alloy steel support due to boric acid corrosion. [Du et al 2005].

An analysis of PWR operating experience of stress corrosion cracking in austenitic SSs has shown that 85% of 146 recorded cracking events in austenitic SS were due to localised perturbations to primary water chemistry in occluded volumes. [Elevare et al 2008]@. However, the remaining 15% appeared to show that intergranular or mixed IGSCC/IGSCC had occurred without any obvious departure from PWR primary water specifications. These 15% of events concerned mainly pressurizer heater sleeves or cladding and heat exchanger tubing of the Chemistry and Volume Control System (CVCS). There also appeared to be a clear association between the incidence of cracking and hardness >300HV but thermal sensitization was clearly not a risk factor.

Despite a growing body of experimental data reviewed in Section 5.3.1.2 in LCSS Annual Report @ that clearly shows that austenitic SSs can crack in normal quality PWR primary water if they are sufficiently cold worked, in the opinion of this author, the above analysis of operating experience omits some key information that points to another more likely culprit.

Firstly, deliberately strain hardened SSs, mainly Type 316 and to a lesser extent niobium stabilized Type 347, have been used successfully for many decades in PWRs for bolting and other purposes where moderate strength is required, and without apparent problems except when highly irradiated (see Section 6.3.2.2 in LCSS Annual Report @). However, a limit is imposed on the maximum yield strength of such deliberately strain hardened components of 90 ksi (or 625 MPa). [US NRC Reg Guide 1.84, 2007]@. In practical terms for unstabilized austenitic steels with medium carbon content, this corresponds to an upper limit on cold work of ~20% or a hardness of ~300HV. Stress corrosion of other hardenable, high strength, SSs only becomes apparent in PWR primary water service at hardness values greater than ~360HV.

© 2008 The author(s). All rights reserved. This article is intended solely for the personal use of the individual user and is not to be disseminated broadly. No part of this publication may be reproduced, stored in a retrieval system, or transmitted, in any form or by any means, electronic, mechanical, photocopying, recording, or by any information storage or retrieval system, without permission in writing from the American Nuclear Society.

This is an example of the information related to Intergranular stress corrosion cracking (IGSCC) and irradiation-assisted stress corrosion cracking (IASCC) of cold worked/irradiated SSs in de oxygenated PWR-type coolants.

---

## Contact

For more information and/or an offer welcome to contact us at [sales@antinternational.com](mailto:sales@antinternational.com)

Please also visit our website for the latest updated information, [www.antinternational.com](http://www.antinternational.com)



A.N.T. INTERNATIONAL®  
[www.antinternational.com](http://www.antinternational.com)

Advanced Nuclear Technology International  
✉ [info@antinternational.com](mailto:info@antinternational.com)    [www.antinternational.com](http://www.antinternational.com)