FONTEVRAUD 11

Contribution of Materials Investigations and Operating Experience to LWRs' Safety, Performance and Reliability

PROGRAM

September 14 to 16, 2026

Palais des Papes, Avignon

Organizer

Sfen

Partner

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EUROPEAN NUCLEAR SOCIETY

As the tradition goes, the International Symposium Fontevraud 11 will be dedicated to the **« Contribution of Materials Investigations and Operating Experience to LWRs' Safety, Performance and Reliability »**.

This conference is devoted to promote exchanges regarding the feedback provided by materials analyses and investigations for the improvement on the operation and safety of light water nuclear reactors. The topics usually involved in this conference include all experiences gained by operators and research entities in the analysis of field-experienced failures, wear or degradation, detection of defects, components control and any materials research investigation connected to field failures.

In this perspective, the Fontevraud Conference is a strong opportunity for operators and researchers to share detailed gained experience provided in the framework of their respective failures investigations programs.
This conference mainly addresses LWRs operating experience from the materials point of view in fields as diverse as surveillance programs, corrosion or fatigue investigations and studies, wear, corrosion and aging management.

Presentations will be organized around key technical issues or systems and components :
Pressure Vessel Components | Pressure Vessel Internals | Stainless Steel & Nickel-based Alloys Areas | Piping, Pumps, Valves | Steam Generators | Steam-Water Systems | Turbine, Alternator | Fuel, Control Rod Assembly | Civil Engineering | Non-metallic Materials and Coatings | FAC Flow Accelerated Corrosion

For the 11th edition of Fontevraud conference, the program committee will particularly welcome papers incorporating **Artificial Intelligence tools in the symposium's theme.**

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Faiza SEFTA, OECD-NEA

TOPICS:
T1. Pressure Vessel Components
T2. Pressure Vessel Internal
T3. Stainless Steel & Nickel-based Alloys Areas
T4. Piping, Pumps, Valves
T5. Steam Generator
T6. Steam Water Systems
T7. Turbine, Alternator
T8. Fuel, Control Rod Assembly
T9. Civil Engineering
TB. Flow Accelerated Corrosion

POSTERS:
P3. Stainless Steel & Nickel-based Alloys Areas
P5. Steam Generator
PB. Flow Accelerated Corrosion

PROGRAM OVERVIEW

MONDAY

- Convention center opening
  Check-in
- Welcome address
  Opening keynote speeches
- Poster pitches session
- Welcome reception
- Keynotes
- Technical sessions
  T01.1 | T03.1+T04.1
- Technical sessions
  T02.1 | T03.2

TUESDAY

- Convention center opening
  Check-in
- Technical sessions
  T01.2 | T03.3 | T09
- Technical sessions
  T02.2 | T03.4 | T06
- Buffet lunch
- Technical sessions
  T01.3 | T03.5 | T04.2
- Technical sessions
  T02.3 | T03.6 | T04.3
- Conference party

WEDNESDAY

- Convention center opening
  Check-in
- Technical sessions
  T08 | TB
- Panel discussion
- Buffet lunch
- Technical sessions
  T01.4 | T05.1
- Technical sessions
  T02.4 | T05.2

Opening Speeches
Keynotes & Panel

+120 Presentations
Oral & Posters

Networking
and Social Events

Attendees from
all over the World

Technical sessions

T01.1 Pressure Vessel Components - Irradiation ageing effects
T01.2 Pressure Vessel Components - Manufacturing and environment effect issues
T01.3 Pressure Vessel Components - Fracture mechanics
T01.4 Pressure Vessel Components - Monitoring of irradiation effects
T02.1 Pressure Vessel Internals - Component management and wear
T02.2 Pressure Vessel Internals - Mechanical analysis and fracture toughness testing
T02.3 Pressure Vessel Internals - Evolution of microstructural and mechanical properties under irradiation
T02.4 Pressure Vessel Internals - Effects of environment and IASCC
T03.1 Stainless Steel & Nickel-Based Alloys Area - Stress corrosion cracking of stainless steel piping
T03.2 Stainless Steel & Nickel-Based Alloys Area - Stainless steels 1/3
T03.3 Stainless Steel & Nickel-Based Alloys Area - Stainless steels 2/3
T03.4 Stainless Steel & Nickel-Based Alloys Area - Stainless steels 3/3
T03.5 Stainless Steel & Nickel-Based Alloys Area - Nickel alloys 1/2
T03.6 Stainless Steel & Nickel-Based Alloys Area - Nickel alloys 2/2
T04.1 Piping, Pump, Valves - Stress corrosion cracking of stainless steel piping
T04.2 Piping, Pump, Valves - Experience feedback
T04.3 Piping, Pump, Valves - Miscellanous material investigations
T05.1 Steam Generator - Inconel 690 tube bundle and plugs
T05.2 Steam Generator - SG secondary side
T06 Steam Water Systems
T08 Fuel, Control, Rod assembly
T09 Civil Engineering
TB Flow Accelerated Corrosion

Avignon

Monday 14 September

08:30 - 09:30
Congress center opening - Check-in

09:30 - 10:15
Welcome Adress & Opening Speech

10:15 - 11:00
Keynote

11:00 - 11:30
Poster pitches session

List of posters at the end of this daily program

11:45 - 13:15
Welcome reception

13:15 - 13:40
Keynote

074 - A Review of Major Topics of Corrosion and Environmentally Assisted Cracking of Steels for Mechanical Components of Light Water Reactors from the Past 40 Years
A. Roth (Consultant, Germany)

13:40 - 14:05
Keynote

121 - EDF SCC operating experience on SIS and RHR stainless steels piping: from understanding to mitigation
F. Foct (EDF, France)

14:05 - 14:20
Transition for room changes

14:20 - 16:00
T01.1 - PRESSURE VESSEL COMPONENTS - Irradiation ageing effects

041 - Microstructural characterization of low-dose neutron irradiated RPV material harvested from a decommissioned Japanese PWR plant
Y. Maeda (Institute of Nuclear Safety System, Incorporated., Japan)

033 - Update of Microstructure Characterization of RPV Material Harvested from a Japanese PWR Plant
K. Fujii (Institute of Nuclear Safety System, Inc., Japan)

132 - Evolution of Irradiation-Induced Microstructure in Reactor Pres-sure Vessel Materials for Long-Term Operation
T. Esclapez (EDF Lab Renardières, France)

150 - High Resolution Microstructural Characterization of Nanoscale "Features" in High Ni-Mn Welds
M.G. Burke (Idaho National Laboratory, United States of America)

147 - Restoration of Irradiation Damage in Reactor Vessel Steels: Insights from Atomic-Level Characterization
C. Hossepied (Cea Paris-Saclay - Door Nord, France)

14:20 - 16:00

T03.1+T04.1 - STAINLESS STEEL & NICKEL-BASED ALLOYS AREAS + PIPING, PUMPS, VALVES - Stress corrosion cracking of stainless steel piping

087 - Root cause analysis and lifetime prediction methodology for real 316LN tube in nuclear power plant
E.H. Han (Institute of Corrosion Science and Technology, China)

130 - Surge line welds SCC susceptibility assessment
F. Foct (EDF, France)

107 - Residual Stress and Strain in Austenitic Stainless-Steel Welds: Inte- grated Approach for PWSCC Risk Assessment
T.H. Pham (Electricité de France, France)

122 - Weld root grinding benefits regarding SCC risk – operating experi-ence and mock-ups characterization
F. Foct (EDF, France)

105 - SEM-FIB and TEM investigations of the evolution of the oxide and microstructure in base metal at weld roots from EDF auxiliary lines
P. Cuvillier (EDF, France)

16:00 - 16:30

Coffee break

16:30 - 18:10

T02.1 - PRESSURE VESSEL INTERNALS - Component management and wear

005 - An Update for Aging Management of PWR Internals in U.S. Plants
K. Amberge (Electric Power Research Institute, United States of America)

049 - Laboratory test evaluation of the wear kinetics of thermal sleeves
M. Mori (Mitsubishi Heavy Industries, Kobe Shipyard, Japan)

111 - Towards Reliable Laboratory Simulation of Wear in Nuclear Reactor Thermal Sleeves: Validation through Post-Service Examination
R. Bonzom (EDF R&D, France)

110 - Thermal Sleeve Flange wear: analysis of FROG member operational experience
R. El Hourany (Framatome, France)

129 - Tihange 3 – Guide Tubes Wear mitigations
S. De Vroey (Engie Laborelec, Belgium)

16:30 - 18:50

T03.2 - STAINLESS STEEL & NICKEL-BASED ALLOYS AREAS - Stainless steels 1/3

035 - Local microstructural characterization of small-strain hardening in large austenitic stainless-steel components
E. Plancher (Framatome DTIM - Mechanical Engineering, France)

042 - Structural impacts of viscoplastic flow during hyperquenching of large components in austenitic stainless steel
R. Lacroix (Framatome, France)

077 - Study on the Effect of Thermal Aging on Stress Corrosion Cracking of 308L Stainless Steel Weld Metal in High Temperature Water
Y. Han (Suzhou Nuclear Power Research Institute, China)

059 - Updated Evaluation of the Effect of Sulfur on SCC Growth of Stainless Steel in Hydrogenated Water
T. Moss (Naval Nuclear Laboratory, United States of America)

071 - Update of initiation and crack growth models for a better prediction of SCC of cold worked stainless steels
T. Couvant (EDF, France)

053 - Comparison of Fatigue Crack Mechanisms of 304L Austenitic Stainless Steel in Air and in PWR Primary Environment
N. Huin (Canadian Nuclear Laboratories, France)

100 - Welded tubular specimen to study intergranular stress corrosion cracking initiation on cold worked austenitic stainless steel in PWR hydrogenated primary environment
R. Verlet (EDF R&D, France)

18:50
End of day 1

Tuesday 15 September

07:30 - 08:00
Congress center opening - Check-in

08:00 - 09:20
T01.2 - PRESSURE VESSEL COMPONENTS - Manufacturing and environment effect issues

052 - Fracture behavior of low-alloy RPV steels in the ductile–brittle transi-tion region: influence of processing and test conditions
I. Uytdenhouwen (SCK CEN, Belgium)

088 - Effect of the intercritical heat treatment on the microstructure and impact properties of an MnNiMo bainitic steel
F. TIOGUEM (FRAMATOME Le Creusot, France)

093 - Contribution of local segregations to the scatter of Charpy impact test results in PWR Pressure Vessel Steels, and its analysis with a numerical model of microstructure generated by AI
P. Joly (AREVA NP, France)

019 - Environmentally-assisted cracking of a thermally-aged and irradi-ated RPV steel in oxygenated high-temperature water
H.P. Seifert (Paul Scherrer Institut, Switzerland)

08:00 - 10:00
T03.3 - STAINLESS STEEL & NICKEL-BASED ALLOYS AREAS - Stainless steels 2/3

144 - Oxidation and SCC of a 316L steel in PWR water: effect of tem-perature and loading conditions
T. De Paula (Framatome, France)

025 - Oxygen Effects on Stress Corrosion Cracking Initiation of Stain-less Steel in PWR Primary Water
Z. Zhai (Pacific Northwest National Laboratory, United States of America)

099 - Ongoing research program on environmental effects on Stress Corrosion Cracking susceptibility of Cold Worked Austenitic Stainless Steels in simulated PWR primary water
C. Rainasse (EDF, France)

128 - Mitigating Stress Corrosion Cracking Initiation in Cold-Worked 316L Stainless Steel under PWR Conditions via Zinc Injection
T. Wang (Paul Scherrer Institute, Switzerland)

047 - Development of a generic intergranular oxidation model of Fe-Cr-Ni alloys in PWR primary water
L. Fasquel (EDF, France)

117 - Effect of surface finish and heat-treatment on the microstructural features and oxidation behavior of austenitic stainless steel in PWR primary water
Y. Vidalenc (Framatome, France)

08:00 - 10:00

T09 - CIVIL ENGINEERING

067 - Development of a Quantitative Evaluation Method for Carbonation Depth in Concrete Structures of Nuclear Power Plants Using an Electrochemical Method
M. Maeda (Central Research Institute of Electric Power Industry, Japan)

065 - Investigation of MIC Mechanisms in Gabion Structures and the Role of Cathodic Protection
C. Loutfi (EDF, France)

119 - Comparison between ND 3D X-ray CT and γ-CT visualisation techniques for irradiated aggregates related to CBS
K. Sobek (Research Centre Řež, Czech Republic)

154 - Thermo-hydro-mechanical Study of Concrete Creep at 150 °C and 90% HR: Separating Basic and Drying Creep for Improved Parameter Calibration
S. CHENG (CEA Paris Saclay, France)

142 - Study of the aging of an epoxy coating in the reactor building of a nuclear power plant under accident conditions
A. Dumand (EDF Direction Technique TEGG, France)

091 - Instrumentation and investigation of decommissioned nuclear re-actor containments in Sweden
P. Lundqvist (Vattenfall AB, Sweden)

10:00 - 10:30

Coffee break

10:30 - 12:10

T02.2 - PRESSURE VESSEL INTERNALS - Mechanical analysis and fracture toughness testing

139 - Irradiation Embrittlement of Reactor Pressure Vessel Internals – Testing of Materials Harvested from Decommissioned Swedish Reactors
J. Stjärnsäter (Studsvik Nuclear AB, Sweden)

080 - Fracture Toughness Measurement of a highly irradiated stainless steel fuel alignment pin
B. Hall (Westinghouse, United States of America)

124 - Room temperature fracture properties of a SA304 austenitic stainless steel irradiated in LWR conditions
J. Hure (CEA, France)

031 - Lessons learned from numerical interpretations of toughness tests performed on highly degraded irradiated stainless steels
S. Chapuliot (EDF Lab Renardières, France)

104 - Flaw evaluation of structural integrity for PWR core barrel in Japan
K. Tahara (Mitsubishi Heavy Industries, Kobe Shipyard, Japan)

10:30 - 12:10

T03.4 - STAINLESS STEEL & NICKEL-BASED ALLOYS AREAS - Stainless steels 3/3

036 - Tensile behaviour of Z10C13 martensitic stainless steel under dynamic loading at intermediate strain rates and high temperature
E. Plancher (Framatome DTIM - Mechanical Engineering, France)

127 - Collaborative Study on the SCC Susceptibility of Additively Manufactured Stainless Steels in Light Water Reactor Environments: First Findings of the CAISAM Project
T. Babinský (PSI Center for Nuclear Engineering and Sciences, Switzerland)

152 - Stress corrosion cracking of additive manufactured 316L stainless steel in pressurized water reactor primary water
C. Guerre (CEA, France)

118 - Mitigation of SCC in CANDU Reactor Calandria Relief Ducts
J. Smith (EPRI, Canada)

075 - International activity on comparison of stress corrosion cracking model predictions with operating experience, Part 1: Description of Methodology
K. Heckmann (GRS, Germany)

10:30 - 12:10

T06 - STEAM WATER SYSTEMS

081 - Management of Erosive Degradation Mechanisms in Nuclear Power Plants
R. Wolfe (EPRI, United States of America)

125 - Cavitation Awareness in Operation and Design
M. Weiss (Framatome GmbH, Germany)

126 - Degradations on the APG001RF heat exchangers of 4 NPP units - Impact analysis of the chemistry in operating cycle and layup practices
S. Lamoudi (EDF, France)

114 - Degradations affecting metal bellows expansion joints of low pressure heat exchanger steam feed lines
J.F. Coste (ELECTRICITE DE FRANCE, France)

092 - Impact of the Nb and V additions on the hardness in the HAZ of a microalloyed carbon-manganese steel and its fracture toughness
G. Ben Salem (Framatome, France)

12:30 - 14:00

Buffet lunch

14:00 - 15:20

T01.3 - PRESSURE VESSEL COMPONENTS - Fracture mechanics

086 - Barsebäck Reactor Pressure Vessel Toughness Variation Through Wall Thickness
N. Hytönen (VTT Technical Research Centre of Finland, Finland)

148 - Application of miniature-C(T) for determination of the master curve of a highly irradiated 16MND5 RPV steel
B. Tanguy (CEA, France)

149 - Local approach to fracture-based assessment of the Master Curve reference temperature T0 from mini-C(T) geometry of a highly irradiated RPV steel
P. Francois (CEA, France)

007 - Investigation of the constraint effect on irradiated RPV steels by testing shallow crack SE(B) specimens
J. May (Framatome GmbH, Germany)

14:00 - 16:00

T03.5 - STAINLESS STEEL & NICKEL-BASED ALLOYS AREAS - Nickel alloys 1/2

027 - Degradation of Forsmark 3 bottom nozzle attachment welds
F. Gustavsson (Forsmark Nuclear Power Plant, Sweden)

078 - International activity on comparison of stress corrosion model predictions with operating experience, part 2: Crack growth in Ni-based welds
S. Faust (GRS, Germany)

064 - Evaluating the efficacy of cavitation peening for SCC resistance in Alloy 182 under simulated light water reactor conditions
A. Das (Paul Scherrer Institut, Switzerland)

043 - Effect of Operation and Thermal Aging on Microstructure, Mechanical Behavior and Cracking of A52 Dissimilar Metal Welds
Z. Que (VTT Technical Research Centre of Finland, Finland)

034 - Stress Corrosion Cracking Growth Rates in Additively Manufactured Laser-Powder Bed Fusion Alloy 625
A. Pinkowitz (Knolls Atomic Power Laboratory, United States of America)

108 - Preliminary FOI's evaluation of primary pump studs replacing A286 stainless steel by Alloy 718
Q. Tence (EDF DISC DT, France)

14:00 - 15:20

T04.2 - PIPING, PUMPS, VALVES - Experience feedback

089 - Fracture toughness of a SA335 P1 steel from the original Feed water Pipe Line part of a NPP
I. Uytdenhouwen (SCK CEN, Belgium)

131 - Thermal fatigue - Characterization of sections of Reactor Heat Re-moval System from service in 900 and 1300 MW plants: review of failure investigations since 2011
F. Renaud (EDF - Technical Direction, France)

006 - Thermal Cycling in Normally Stagnant RCS Branch Lines connected to the Residual Heat Removal Systems of the Doel 4 and Tihange 3 Nuclear Power Plants
M. De Smet (TRACTEBEL-ENGIE, Belgium)

113 - Failure Analysis of Krško SI-53 Piping
S. El Shawish (Jožef Stefan Institute, Slovenia)

16:00 - 16:30

Coffee break

16:30 - 18:10

T02.3 - PRESSURE VESSEL INTERNALS - Evolution of microstructural and mechanical properties under irradiation

146 - Microstructural Evolution of Irradiated 403 Stainless Steel CANDU End Fittings
R. Matthews (Canadian Nuclear Laborat, Canada)

133 - The Effect of Irradiation on the NRU G-16 Fast Neutron Cup Ex-posed to CANDU-Relevant Irradiation Conditions
M. Mitchell (Canadian Nuclear Laboratories , Canada)

135 - Sensitivity analysis of swelling in irradiated austenitic stainless steels: a comparison between experimental data and cluster dynamics modeling
G. Adjanor (EDF R&D, France)

022 - Micro-tensile testing of grain boundaries in Fe and He ion-irradiated type 316 stainless steel and model alloys
T. Miura (Institute of Nuclear Safety System, Inc., Japan)

066 - Stress Relaxation Through In-Core Real-Time Mechanical Testing of Structural Materials
R. Song (Idaho National Laboratory, United States of America)

16:30 - 18:50

T03.6 - STAINLESS STEEL & NICKEL-BASED ALLOYS AREAS - Nickel alloys 2/2

021 - Improving SCC prediction of Bottom Mounted Instrumental nozzles
C. Perez (EDF Lab Renardières, France)

062 - Improvement of Alloy 600 SCC initiation and crack growth models to better evaluate to susceptibility to SCC of components
T. Couvant (EDF, France)

094 - Understanding Long-Term Crack Initiation Behavior in Cold Worked Alloy 690
Z. Zhai (Pacific Northwest National Laboratory, United States of America)

106 - Evaluation of the PWSCC susceptibility of nickel base alloys HAZ using calibrated welding defects
J. Caballero-Hinostroza (Framatome, France)

112 - Assessment of manufacturing relevant cold work levels on the SCC susceptibility of thin walled components made of Alloy 600 by EBSD measurement and accelerated corrosion testing
M. Grimm (Framatome GmbH, Germany)

096 - Influence of cold work on mechanical behavior of alloy 690
J. Blaizot (Framatome, France)

090 - Predicting the SCC susceptibility of thin-walled alloy 600 cable sheaths in accelerated lab testing at 350 °C
M. Weiser (Framatome GmbH, Germany)

16:30 - 17:30

T04.3 - PIPING, PUMPS, VALVES - Miscellanous material investigations

134 - Buried Piping Non-Metallic Rehabilitation Technologies for the Refurbishment of Nuclear Stations
C. Cooper (Kinectrics Inc, Canada)

029 - Evaluation of Gas Metal Arc-Direct Energy Deposition Type 316LSi Stainless Steel Valves
F. Gift (Electric Power Research Institute (EPRI), United States of America)

085 - Mechanical testing of thermally aged stainless steel – Base and Welding materials
R. Magnusson (Ringhals - Vattenfall, Sweden)

18:50

End of day 2

20:00 - 22:30

Conference party

Wednesday 16 September

08:00

Congress center opening - Check-in

08:30 - 10:10

T08 - FUEL, CONTROL ROD ASSEMBLY

083 - Mechanistic study of the oxidation behavior of Zr-1Sn-1Nb-0.3Fe alloy in simulated PWR primary water
Z. Li (Institute of Corrosion Science and Technology, China)

050 - Localized oxidation of Zr-2.5Nb alloy – role of material and environmental parameters
S.K. Nouduru (Bhabha Atomic Research Centre, India)

051 - Release of fission products from a failed rod and influence of the oxidation state of its pellets
C. Petit (Framatome, France)

069 - Fretting wear behaviour of Zr alloy cladding tube in simulated primary water of PWR
H. Ming (Institute of Metal Research, Chinese Academy of Sciences, China)

011 - Microstructural evolution of Zr-2.5Nb and its effect on oxidation kinetics during steam oxidation under simulated accident conditions
V.S.V. Anantha Krishna (Bhabha Atomic Research Centre, India)

08:30 - 10:30

TB - FLOW ACCELERATED CORROSION

057 - Examination of an expander affected by FAC on the secondary circuit, located downstream of a weld and after a regulating valve
P. Mestre-Rinn (EDF, France)

008 - Identification of local flow-accelerated corrosion phenomena us-ing data analysis methods
V. Garric (EDF, Direction Technique, France)

101 - "Entrance Effect": local enhancement of FAC rate in the presence of a difference of chromium content
C. Rainasse (EDF, France)

044 - CFD simulations of flow accelerated corrosion cou-pled with electrochemistry
S. Mimouni (EDF Lab Chatou, France)

045 - Computational Fluid Dynamics at the Entrance of Stagnant Lines for Flow-Accelerated Corrosion
J. Varnam (EPRI, United States of America)

032 - Predicting Flow-Accelerated Corrosion and Erosion in Nuclear Power Plants the Homogeneous and Slip Models for Two-Phase Steam Flow
R. Wolfe (EPRI, United States of America)

10:25 - 10:40

Transition for room changes

10:40 - 11:40

Panel discussion

11:40 - 12:40

Buffet lunch

12:40 - 14:00

T01.4 - PRESSURE VESSEL COMPONENTS - Monitoring of irradiation effects

070 - Supervised and non-supervised ML approaches for embrittlement prediction
Y. Hashimoto (Central Researcy Institute of Electric Power Industry, Japan)

145 - Characterization of High-Fluence RPV Surveillance Capsule from Palisades NPP
M. Sokolov (ORNL, United States of America)

048 - Relationship between fracture toughness shift and Charpy shift in Japanese PWR surveillance data
A. Watanabe (Nuclear Regulation Authority, Japan)

030 - Irradiation Surveillance Program - Mechanical and Metallurgical Analyses
K. Jeuland (EDF CNPE Chinon, Technical division, France)

12:40 - 14:40

T05.1 - STEAM GENERATOR - Inconel 690 tube bundle and plugs

102 - High-temperature aqueous corrosion studies of Alloy 690 steam generator tubes
M. Zimina (University of Bristol, School of Physics, HH Wills Physics Laboratory, United Kingdom)

143 - Alloy 690TT Production Method SCC Resistance Testing
B. Capell (EPRI, United States of America)

137 - SMILE Project: Characterization on Harvested Ringhals 2 Alloy 690 TT Steam Generator Tubes
M. Wang (Studsvik Nuclear AB, Sweden)

028 - Sherlock Project – Investigate to Mitigate
G. Klotz (EDF CNPE Chinon, France)

058 - Examination of an Inconel 690TT mechanical steam generator plug after 20 years of operation
P. Mestre-Rinn (EDF, France)

153 - Eddy Current Simulation for Steam Generator Tube Inspections
J. Benson (EPRI, United States of America)

14:20 - 14:30

Transition

14:30 - 16:10

T02.4 - PRESSURE VESSEL INTERNALS - Effects of environment and IASCC

103 - TEM investigations of oxides and base metal at the surface of the confined zone of an uncracked baffle-former bolt irradiated up to 40 dpa coming from a French PWR
P. Cuvillier (EDF, France)

123 - Quantification of grain-scale localized plasticity by micro-DIC for developing physical models of IASCC
W. Karlsen (VTT Technical Research Centre of Finland, Finland)

004 - Hot Cell Testing of Type 304 Stainless Steel Core Barrel Samples Removed from U.S. PWR
F. Gift (Electric Power Research Institute (EPRI), United States of America)

076 - Effects of Chloride and Sulfate on the Crack Growth Rate of Irradiated Stainless Steel in BWR Environments
A. Jenssen (Studsvik Nuclear AB, Sweden)

040 - Formulation of the SCC Crack Growth Rates Equation for neutron irradiated XM-19 under BWR environments
M. Koshiishi (Central Research Institute of Electric Power Industry, Japan)

14:30 - 16:10

T05.2 - STEAM GENERATOR - SG secondary side

063 - Assessment of Steam Generator Tube Support Plate Clogging at Doel 4 and Tihange 3
W. Van Eesbeeck (ENGIE Laborelec, Belgium)

009 - Understanding the spatial distribution of clogging inside steam generators by data analysis
V. Garric (EDF, Direction Technique, France)

084 - Development of simulation method for boiling two-phase flow in the secondary side of steam generators
J. Suzuki (Mitsubishi Heavy Industries, Kobe Shipyard, Japan)

098 - The concept of SCC precursors, experimental work, and its appli-cation to Canadian SG tube extracted from Periodic Inspection Programs
N. Huin (Canadian Nuclear Laboratories, France)

079 - Decrease of hydrogen content in low alloyed steels for steam generator parts
A. Guenot (Framatome Le Creusot, France)

16:10 - 16:10

End of the conference

Posters

P3 - STAINLESS STEEL & NICKEL-BASED ALLOYS AREAS

037 - Fracture Toughness Testing of a Work-Hardened Austenetic Stainless Steel Pre-Cracked by Intergranular Stress Corrosion Cracking
W. Vincent (EDF Lab Renardières, France)

046 - Development of the Ultra-micro Tensile test system for Crack Tip (UTCT)
K. Fujii (Institute of Nuclear Safety System, Inc., Japan)

116 - Impact of Cleanliness of Austenitic Stainless Steel 304 on Corro-sion and Cracking Susceptibility After Long-Term Service in Nu-clear Environments
J. Ben Mohamed (Framatome, France)

138 - Effect of Zn injection on high-temperature water oxidation behaviour in LWR steam generator materials
B. Carley-Macauly (University of Bristol, United Kingdom)

115 - Physics-informed and data-driven Reliable Estimations for Corrosion In Sustainable nuclear Environments (PRECISE): Project Overview
I. Otic (Karlsruhe Institute of Technology Campus North, Germany)

P05 - STEAM GENERATOR

068 - The single effect of microstructure, residual strain and geometric structure on the stress corrosion cracking behavior of scratched alloy 690TT
B. Wu (Institute of metal research, China)

PB - FLOW ACCELERATED CORROSION

039 - Examination of steam generator swirl vanes affected by FAC
S. Le Ber (EDF, France)