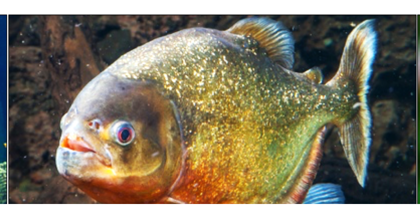
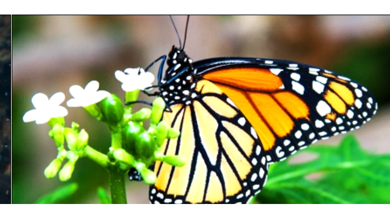
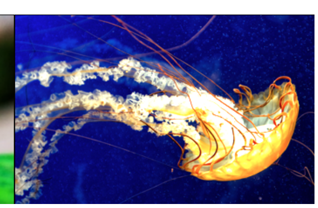
BOOK OF ABSTRACTS

13TH INTERNATIONAL WORKSHOP ON SPALLATION MATERIALS TECHNOLOGY

IWSMT-13

OCTOBER 30TH – NOVEMBER 4TH, 2016

CHATTANOOGA, TENNESSEE
UNITED STATES OF AMERCIA

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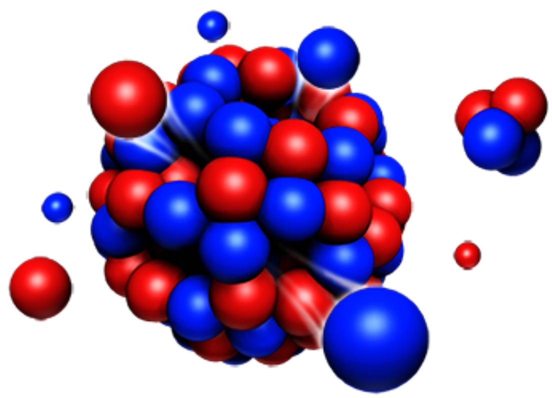

OCTOBER 30TH – NOVEMBER 4TH, 2016

CHATTANOOGA, TENNESSEE
UNITED STATES OF AMERCIA

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Stuart A. Maloy, Los Alamos National Laboratory, USA
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John Haines, European Spallation Source, Sweden
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Reception, Registration, and Banquet Dates

13th International Workshop on Spallation Materials Technology
Chattanooga, TN, USA
October 30th – November 4th, 2016

Sunday, October 30th, 2016

16:00 – 18:00 Registration
18:00 – 20:00 Reception at the Chattanooga Marriot Hotel

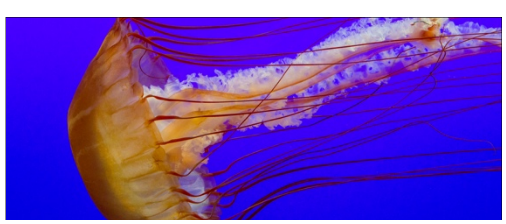
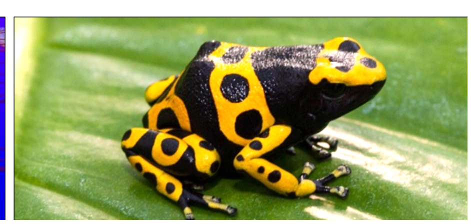
Monday, October 31st, 2016

07:00 Registration

Wednesday, November 2nd, 2016

19:00 – 22:30 Banquet at the Tennessee Aquarium (Ocean Journey building)



Scope

The International Workshop on Spallation Materials Technology (IWSMT) is the premier international forum to present and discuss the latest research and development of materials utilized in spallation sources and accelerator driven systems (ADS). It is also an outstanding forum for exchanging information from operational experience and post irradiation examination (PIE) of irradiated specimens and components.

The thirteenth meeting of IWSMT will again focus on the progress with materials-related studies for accelerator based neutron sources and on issues with operating and future spallation facilities. Our intent is to promote open discussion of the latest material advances and challenges and to encourage new international collaborations. For this meeting organizers would like to invite collaboration with colleagues in the High Energy Physics and Radioactive Ion Beam communities in order to facilitate and promote cooperation between the different international communities that share many common materials-related challenges.

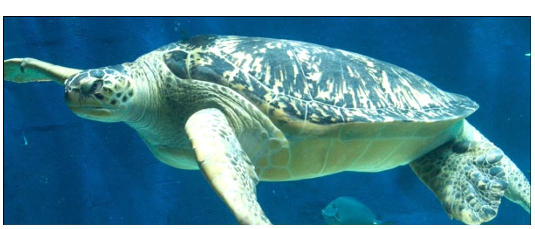
Topics

1. Effects of radiation damage, helium, hydrogen, and other transmutation elements in target and structural materials
2. Operational experience and results from PIE of irradiated components
3. Research in the area of mitigating pressure wave effects in pulsed targets, i.e., cavitation erosion and fatigue
4. Research and development (R&D) of spallation targets for neutron and ADS applications
5. Source technology and engineering: fatigue analysis, welding, weld evaluation, and thermo-mechanical/fluid modeling
6. Production and handling of volatile transmutation elements and activated target components
7. Innovative materials technology and techniques for R&D of applications with extreme irradiation environments
8. Application of materials data to the design and safety study of components in spallation sources and ADS systems

Presentations and Publication

High-quality full papers with sufficient materials science and engineering content relevant to journal themes will be selected by the organizers to be included in a special volume of the Journal of Nuclear Materials.

Information on the scope of the Journal of Nuclear Materials is located at: http://ees.elsevier.com/jnm/




WORKSHOP AGENDA

Monday – October 31, 2016

07:00 – 08:15	**Registration**	
	Session 1: Welcome and Keynote Addresses	
08:20 – 08:30	Opening Address	Bernie Riemer
08:30 – 08:40	Welcome Address	Don Abercrombie
08:40 – 09:00	Plenary Address 1 - The International Workshops on Spallation Materials Technology	Lou Mansur
09:00 – 09:40	Plenary Address 2 - Impact of H and He transmutation products on radiation effects in materials	Steve Zinkle
09:40 – 10:00	**Coffee Break**	
10:00 – 10:25	Status of the European Spallation Neutron Source	Eric Pitcher
10:25 – 10:50	Present status of Chinese Spallation Neutron Source project	Xuejun Jia
10:50 – 11:15	The Proton Power Upgrade project and its Impact on the SNS First Target Station	Bernie Riemer
11:15 – 11:40	Current status of the Chinese ADS project	Zhiguang Wang
11:40 – 12:50	**Lunch**	
	Session 2: Progress of Accelerator Driven System with Emphasis on Materials	
12:50 – 13:15	SINQ - A Status Report	Michael Wohlmuther
13:15 – 13:40	Present status of JSNS mercury target	Takashi Naoe
13:40 – 14:05	Compatibility research of structural materials for China lead-based research reactor	Chunjing Li
14:05 – 14:30	Spallation materials R&D and application for Beam Intercepting Devices (BID) at CERN	François-Xavier Nuiry
14:30 – 14:50	**Coffee Break**	
14:50 – 15:15	Investigation of target material for muon production under high power proton beam irradiation	Shunsuke Makimura
15:15 – 15:40	Pulsed heavy-ion irradiation of tungsten	Jemila Habainy
15:40 – 16:05	Re-examination of ion irradiation as a credible tool to simulate high energy neutron and proton-induced void swelling for accelerator-driven devices	Frank Garner
16:05 – 16:30	The RaDIATE Collaboration – exploring high power target materials response to radiation damage – goals, status, and future plans	Patrick Hurh
16:30 – 18:00	**Poster Session**	

Tuesday – November 1, 2016

	Session 3: Radiation-Induced Effects in Structural Materials	
08:20 – 08:45	Development of advanced ferritic steels for high dose applications	Stuart Maloy
08:45 – 09:10	Tensile testing of steels from the STIP-V irradiation	Tarik Saleh
09:10 – 09:35	Tensile properties characterization of irradiated AISI 316L from high-use target modules at the Spallation Neutron Source using digital image correlation	David McClintock
09:35 – 10:00	Swelling, creep, and embrittlement of D9 stainless steel cladding and duct in two FFTF driver fuel assemblies at high neutron exposures	Frank Garner
10:00 – 10:20	**Coffee Break**	
10:20 – 10:45	The behaviour of $AlMg_3$ after irradiation at high proton and neutron fluences in SINQ targets	Yong Dai
10:45 – 11:10	Post irradiation examination of the MEGAPIE samples at JAEA (2)	Shigeru Saito
11:10 – 11:35	The transmission electron analysis of dislocation loops in T91 steels from MEGAPIE and TWIN-ASTIR irradiation programs	Milan Konstantinovic
11:35 – 12:00	Barrier strength of defects and helium bubbles for hardening of martensitic steels irradiated in STIP	Lei Peng
12:00 – 13:10	**Lunch**	
	Session 4: Research and Development of Target System Materials	
13:10 – 13:35	Mechanical properties and fracture behavior of pure tungsten and tantalum after irradiation in SINQ	Yong Dai
13:35 – 14:00	Annealing effect on the microstructure and hardness of irradiated tungsten	Barbara Horvath
14:00 – 14:25	Formation of oxide layers on tungsten in mildly oxidizing gas	Jemila Habainy
14:25 – 14:50	Low-Z material R&D and application for Beam Intercepting Devices (BID) at CERN	François-Xavier Nuiry
14:50 – 15:10	**Coffee Break**	
15:10 – 15:35	Fatigue properties of tungsten from different processing routes	Jemila Habainy
15:35 – 16:00	Luminescent materials development for beam-on-target imaging at the European Spallation Source	Thomas Shea
16:00 – 16:25	Design and fabrication of a passive irradiation module utilizing the high neutron flux from the 5 MW Spallation Source at ESS	Yong Joong Lee
16:25 – 17:00	**Discussion - 1**	

Wednesday – November 2, 2016

	Session 5: Research and Development of Target System Designs	
08:20 – 08:45	The ESS helium cooled rotating target	Fernando Sordo
08:45 – 09:10	Design modification of ISIS TS2 target in order to improve longevity amid spallation reactions	Arghya Dey
09:10 – 09:35	Simulating performance of tantalum-clad tungsten targets	Dan Wilcox
09:35 – 10:00	Manufacturing of ESS cold moderator – machining, welding, and testing of Al 6061-T6 alloy	Yannick Bessler
10:00 – 10:20	**Coffee Break**	
10:20 – 10:45	Weldability of diffusion bonding between Invar alloy and stainless steel by hot isostatic pressing	Takashi Wakui
10:45 – 11:10	Measured SNS mercury target vessel strain responses to beam pulses and comparison to simulations with variations on mercury material model behavior	Drew Winder
11:10 – 11:35	Failure analysis on the welded part with steep change of thickness employed in JSNS mercury target vessel	Takashi Wakui
11:35 – 12:00	Thermomechanical analysis of ESS spallation material	Fernando Sordo
12:00 – 13:00	**Lunch**	
19:00 – 22:30	*Banquet at the Tennessee Aquarium in the Ocean Journey building*	

Thursday – November 3, 2016

Session 6: Compatibility of Liquid Metals with Structural Materials		
08:20 – 08:45	Cavitation damage in double-walled mercury target vessel	Takashi Naoe
08:45 – 09:10	Effect of oxygen concentration on LME susceptibility of CLAM steel in liquid lead bismuth eutectic	Liu Jing
09:10 – 09:35	Numerical modeling of impurities mass transfer in a wire wrapped fuel assembly under flowing lead bismuth eutectic	Alessandro Marino
09:35 – 10:00	Development of SIMP steel for accelerator driven system in China	Zhiguang Wang
10:00 – 10:20	**Coffee Break**	
10:20 – 10:45	Low cycle fatigue behavior of 15-15Ti steel in static lead-bismuth eutectic with 10-6wt% oxygen concentration at 550°C	Chunjing LI
10:45 – 11:10	Corrosion behavior of Ti_3SiC_2 in flowing lead bismuth eutectic at 1000°C	Zunqi Xiao
11:10 – 11:35	Oxidation behaviors of CLAM steel in stagnant liquid lead-bismuth Eutectic at 500 °C	Shaojian Yan
11:35 – 12:00	**Discussion - 2**	
12:00 – 13:10	**Lunch**	
Session 7: Analysis of Target System Materials and Components		
13:10 – 13:35	Positron lifetime calculation of vacancy clusters in tantalum containing hydrogen and helium	Qiu Xu
13:35 – 14:00	Investigation of samples of F/M and ODS steels irradiated in the spallation source SINQ by positron annihilation	Vladimir Krsjak
14:00 – 14:25	Rate theory analysis of growth process of helium bubble in F82H Irradiated at SINQ	Koichi Sato
14:25 – 14:50	Investigation of SINQ-irradiated samples by single detector Doppler-broadening spectroscopy	Jozef Snopek
14:50 – 15:10	**Coffee Break**	
15:10 – 15:35	Experimental investigation of irradiation effects in beryllium beam window after exposure in the NuMI beamline: preliminary results and plans	Viacheslav Kuksenko
15:35 – 16:00	Plans for the RaDIATE high-energy proton materials irradiation experiment at the Brookhaven Linac Isotope Producer facility	Kavin Ammigan
16:00 – 16:25	HiRadMat at CERN SPS - A dedicated test facility with high intensity beam pulses to material samples	Adrian Fabich
16:25 – 17:00	**Discussion - 3**	

Friday – November 4, 2016

Session 8: Simulations of Radiation Effects on Materials in Spallation Applications		
08:20 – 08:45	Application of rigorous two step methodology for neutron and proton transmutation calculations to spallation targets.	Steven Lilley
08:45 – 09:10	Displacement damage, helium and hydrogen production in different materials irradiated in STIP-VI	Yong Dai
09:10 – 09:35	Simulation of hydrogen thermal desorption characteristics in metals containing large voids	Hayato Yamashita
9:35 – 9:55	**Coffee Break**	
9:55 – 10:20	Material selection of the beam profile monitoring devices at the ESS Target Station	Yong Joong Lee
10:20 – 10:45	The present status and prospective of STIP	Yong Dai
10:45 – 11:15	**Discussion - 4**	
11:15 – 11:45	**Summary and Concluding Remarks**	

IWSMT-13 ORAL PRESENTATION ABSTRACTS

The International Workshops on Spallation Materials Technology

MANSUR, Louis K. (Oak Ridge National Laboratory – retired)

The first International Workshop on Spallation Materials Technology was held in 1996 at Oak Ridge, Tennessee. Important goals were to make preliminary materials selections for new liquid metal target high power spallation neutron sources in conceptual design, and to confront the great challenge of how to obtain needed materials performance information in a reasonable time within available resources. Participants were asked to review and assess the sparse materials experience at existing spallation devices, as well as to evaluate the relevancy of available information from fission, fusion, and lower energy charged particle irradiation research. Plans were formulated to develop the experimental data and calculations on which to base design decisions and target lifetime estimates. Opportunities for collaborations among the relatively few internationally dispersed spallation materials groups were explored; joint work was planned.

In brief introductory remarks on the occasion of the twentieth anniversary of the first meeting we mention the beginnings. We visit the originally stated goals and assess to what extent they have been achieved. The core group of researchers present at the first meeting and their collaborators developed the continuing IWSMT series. Powered by strong international collaborations, the infant has grown to produce the highly successful work that has created the present wealth of knowledge.

The first International Workshop on Spallation Materials Technology was held in 1996 at Oak Ridge, Tennessee. Important goals were to make preliminary materials selections for new liquid metal target high power spallation neutron sources in conceptual design, and to confront the great challenge of how to obtain needed materials performance information in a reasonable time within available resources. Participants were asked to review and assess the sparse materials experience at existing spallation devices, as well as to evaluate the relevancy of available information from fission, fusion, and lower energy charged particle irradiation research. Plans were formulated to develop the experimental data and calculations on which to base design decisions and target lifetime estimates. Opportunities for collaborations among the relatively few internationally dispersed spallation materials groups were explored; joint work was planned.

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Impact of H and He transmutation products on radiation effects in materials

ZINKLE, Steve (University of Tennessee)

YAMAMOTO, Takuya (University of California Santa Barbara)

Due to threshold energies for fast neutron (n,xp) and (n, xα) reactions that are near 1-10 MeV for many isotopes, the typical production rates of H and He transmutation products per unit fast fluence (or displacement damage level) in many materials are at least one order of magnitude higher for spallation and D-T fusion neutron spectra compared to fission spectra. This can lead to enhanced radiation hardening and embrittlement at low irradiation temperatures (below ~0.35 T_M, where T_M is the melting temperature), either enhanced (fusion) or suppressed (spallation) void swelling at intermediate temperatures, and enhanced high temperature helium embrittlement of grain boundaries above ~0.5 T_M. In many cases, pronounced helium effects begin to emerge for He concentrations above ~500 appm. This presentation will review current understanding of the mechanisms and consequences of fission-, fusion-, and spallation-neutron relevant H and He production on irradiation processes in materials. Synergistic effects associated with simultaneous He and H production will be discussed. The role of dose rate (e.g., He/dpa, He/s and dpa/s) effects will be discussed in terms of different temperature-and dose rate-dependent regimes for point defect supersaturation.

Status of the ESS target station

PITCHER, Eric (European Spallation Source ERIC)

The European Spallation Source, now under construction in Lund, Sweden, aims to be the world's most powerful pulsed neutron scattering facility. Driven by a 5-MW proton accelerator, the target station incorporates many features designed to accommodate the high-power proton beam. Among these are a rotating tungsten target cooled by helium gas and the option to run without a proton beam window. As of August 2016, the target station project is 17% complete, and construction has started with some components already manufactured and ready to be installed. Along with the central team in Lund, six European partner institutions are cooperating to deliver the target station. Efforts are now focused on

Present status of Chinese Spallation Neutron Source project

JIA, Xuejun (Institute of Physics, CAS)

The CSNS project is designed to provide multidisciplinary platforms for scientific research and applications by national institutions, universities, and industries. CSNS is an accelerator based multidiscipline user facility. It consists of an H- linear accelerator, a rapid cycling synchrotron accelerating the beam to 1.6 GeV, a solid tungsten target station, and 5 initial instruments for spallation neutron applications. The facility operates at 25 Hz repetition rate with an initial design beam power of 100 kW and is upgradeable to 500 kW. The primary challenge is to build a robust and reliable user's facility with upgrade potential at a fraction of "world standard" cost. A brief introduction of the CSNS project and its current status will be given at the talk.

The Proton Power Upgrade project and its impact on the SNS First Target Station

RIEMER, Bernard (Oak Ridge National Laboratory)

GALAMBOS, John (Oak Ridge National Laboratory)

In conjunction with the future Second Target Station (STS) at Oak Ridge National Laboratory's Spallation Neutron Source (SNS) site, the Proton Power Upgrade (PPU) project will double the power of the existing accelerator that will support both STS and the existing First Target Station (FTS) . The design basis for most of the existing FTS was 2 MW with 1.0 GeV proton pulses delivered at 60 Hz pulse repetition rate. 2 MW or more will be available to FTS with PPU, but PPU increases proton energy to 1.3 GeV. Further, the envisioned operation of STS at 10 Hz implies 50 Hz to FTS. Higher proton energy and the altered repetition rate when STS operates - along with the potential to go beyond 2 MW - require re-evaluation of thermal, structural and shielding margins of FTS systems as well as changes in radiation damage rates. Affected target systems include reflectors, moderators and their cryogenic systems, shielding, utilities and close-in instrument hardware. The original FTS design basis employed simplified bounding assumptions for systems heating which are expected to provide margins when evaluated with more accurate methods with 1.3 GeV protons, but this must be verified. Some system upgrades may be necessary.

The mercury target module was an exception in the original FTS as it was initially rated for 1.0 MW and later increased to 1.4 MW. While records have been set, operation of the first-of-a-kind mercury target technology at the MW level has had difficulties with half of the targets experiencing leaks during neutron production. Design changes have incrementally been incorporated, fabrication oversight improved, and soon gas injection will be initiated at a low flow rate. The jump to 2 MW will require more. Part of PPU includes scope for further developing and deploying high-flow helium gas injection into the mercury target to provide maximum mitigation of beam pulse induced cavitation erosion and vessel fatigue stress. These two phenomena have directly or indirectly led to the target leaks. A new target vessel design will build upon lessons learned from operational experience and post irradiation examination of targets.

This presentation will describe the PPU project with emphasis on the FTS scope.

Current status of the Chinese ADS project

WANG, Zhiguang (Institute of Modern Physics, CAS)

ADS TEAM, (China)

Aimed at the challenging issues that confronted by the long-term sustainable development of advanced nuclear energy, a roadmap of the Chinese ADS (Accelerator Driven System) project was planned and the first stage of this project (scheduled from 2011 to 2016) was launched by the Chinese Academy of Sciences at 2011. During the first stage, great progresses have been achieved on the key technologies on accelerator, target and sub-critical reactor, as well as the related researches. The second stage project, CIADS, was approved last year. It consists of a superconducting proton linac with beam current of 10 mA and energy of about 600 MeV, a granular target with tungsten pellets, and a LBE sub-critical reactor. The total thermal power of reactor and beam is 10 MW. It is a demonstration of high power coupling and transmutation. Recently, a new concept of ADANES (Accelerator Driven Advanced Nuclear Energy System) was proposed and put forward as the future plan.

In this presentation, the current status of the Chinese ADS project will be introduced. More, some issues on target and material are also given.

SINQ – A Status Report

WOHLMUTHER, Michael (Paul Scherrer Institut)

BLAU, Bertrand (Paul Scherrer Institut)

GEISSMANN, Kurt (Paul Scherrer Institut)

HEINRICH, Frank (Paul Scherrer Institut)

WELTE, Jörg (Paul Scherrer Institut)

The spallation neutron source SINQ at the Paul Scherrer Institute (PSI) in Switzerland is in operation since 1996. Since then SINQ has constantly been improved, on both the instruments and the target station side. The improvements of the latter have been achieved mainly by successively optimizing the spallation target design. The Mark I targets were composed of solid zircaloy-II rod bundles. In 2000 they were replaced by rod bundles made from lead filled stainless steel tubes (Mark II). These solid targets with canned lead rods are colloquially called Cannelloni targets. Until the end of 2005 three stainless steel Cannelloni targets were used for neutron production in SINQ; each target was in operation for 2 years. In 2006 a lead bismuth eutectic (LBE) liquid metal target (LMT) – MEGAPIE – was operated successfully and provided neutron yields up to 80% higher than achieved so far with the stainless steel Cannelloni targets. Following MEGAPIE the cladding material of the Cannelloni targets was changed from stainless steel to zircaloy-II (Mark III), again increasing the neutron yield as compared to the stainless steel cladded lead target by about 6%. The huge performance increase experienced during the operation of the LMT (MEGAPIE) triggered further target development resulting in a more compact Cannelloni target surrounded by a lead filled reflector (blanket) with an inverted beam entrance window geometry integrated in the double walled $AlMg_3$ safety shroud, the Mark IV target. The evaluated performance increase of 38% was achieved and SINQ is operating with Mark IV targets since 2009. Currently the SINQ upgrade project aims for further improvements by replacing the neutron guide system, planned for 2018/2019, and a redesigned cold D2 moderator setup.

In this paper we will report details of the SINQ operation, provided statistics about beam time on target, integrated beam currents received, operating hours per year and neutron performance. Furthermore, the SINQ Upgrade Project and challenges as well as difficulties faced during operation will be discussed.

Present status of JSNS mercury target

NAOE, Takashi (J-PARC Center, Japan Atomic Energy Agency)

HAGA, Katsuhiro (Japan Atomic Energy Agency)

KOGAWA, Hiroyuki (Japan Atomic Energy Agency)

WAKUI, Takashi (Japan Atomic Energy Agency)

TAKADA, Hiroshi (Japan Atomic Energy Agency)

The Japan Spallation Neutron Source (JSNS) has been operated since May, 2008 to provide pulsed neutrons by using a liquid mercury target for materials science innovative researches. The goal of operational conditions of the mercury target is 1 MW at a 25 Hz repetition rate with a 1 µs pulse duration. In 2015, the proton beam power was ramped up from 300 kW to 500 kW. However, the JSNS target was faced with troubles twice in the water shroud caused from thermal stress at 500 kW.

The JSNS mercury target vessel was made from stainless steel 316L. It has the triple-walled structure consisting of mercury vessel, inner and outer water shrouds, where the boundary between the mercury vessel and the water shroud is filled with helium gas. For the first trouble occurred in April, we concluded that the water leakage from the welding defect outside of the outer water shroud, where the water shroud bolt to the mercury vessel, through a visual inspection and analytical studies.

For the second trouble occurred in November, water leak was detected in the helium layer. The results of FEM simulation suggested that the leak point was seemed to be fatigue crack at around the portion of monolithic structure connecting the mercury vessel with the water shroud due to the cyclic thermal stress induced by beam trips. The problem is thought to be caused by the structure that left an unwelded portion. The leak point is yet to be identified by inspection.

The JSNS target vessel design was improved to reduce welding lines approximately 30% and bolt connection as much as possible. The fabrication by applying new technique, the wire-EDM, is under progress. In the presentation, present status of JSNS mercury target including outline of the water leak incidents and the new design of the mercury target vessel will be reported.

Compatibility research of structural materials for China lead-based research reactor

LI, Chunjing (Key Laboratory of Neutronics and Radiation Safety, Institute of Nuclear Energy Safety Technology, Chinese Academy of Sciences)

JIANG, Zhizhong (Key Laboratory of Neutronics and Radiation Safety, Institute of Nuclear Energy Safety Technology, Chinese Academy of Sciences)

LUO, Lin (Key Laboratory of Neutronics and Radiation Safety, Institute of Nuclear Energy Safety Technology, Chinese Academy of Sciences)

GAO, Sheng (Key Laboratory of Neutronics and Radiation Safety, Institute of Nuclear Energy Safety Technology, Chinese Academy of Sciences)

FDS, Team (Key Laboratory of Neutronics and Radiation Safety, Institute of Nuclear Energy Safety Technology, Chinese Academy of Sciences)

China Lead-based Research Reactor (CLEAR-I) is the reference reactor for the ADS (Accelerator Driven Subcritical system) program in Chinese Academy of Sciences. Liquid lead-bismuth eutectic (LBE) is chosen as the coolant used in the primary cooling system. Three main candidate structural materials in the reactor are 316L and T91 for the main structural components (main vessel, internals, etc.), and 15-15Ti (1.4970) for the fuel cladding tubes. For LBE cooled reactor, one of the main problems is the compatibility of structural materials with LBE. The paper presents the main progress in the compatibility research of the structural materials for CLEAR-I, including the compatibility facilities, representative results of corrosion and mechanical tests. To study the corrosion behaviors and mechanisms of the structural materials under normal conditions of CLEAR-I, KYLIN series loops and some miniature corrosion devices had been developed. In addition, an ultra-high temperature PbBi loop had been developed for evaluating the corrosion resistance of materials under accident conditions up to 1000°C. A 10,000h corrosion test in LBE is under way at 500°C with the dissolved oxygen of $1\sim3\times10^{-6}$ wt% and the flow velocity of 1 m/s. The test has been successfully operated for 5000 h and the data analysis has been finished. To evaluate mechanical properties of the structural materials in LBE, series mechanical facilities including slow strain rate tensile (SSRT), creep and fatigue facilities has been already constructed in 2014. Up to Now, the effects of pre-exposure, oxygen content in LBE and surface stress concentration on tensile properties of T91 steel in LBE had been conducted systematically. Then, LME susceptibility of T91 and 316L steels in LBE had been assessed qualitatively among the temperature between 200°C and 600°C. Also, the low cycle fatigue behavior of 15-15Ti steel had been compared in LBE and air at 550°C. Keywords: CLEAR-I; Lead-Bismuth Eutectic; Structural Materials.

Spallation materials R&D and application for Beam Intercepting Devices (BID) at CERN

CALVIANI, Marco (CERN)

PERILLO MARCONE, Antonio (CERN)

TORREGROSA, Claudio (CERN)

FORNASIERE, Elvis (CERN)

ESPOSITO, Raffaele (CERN)

NUIRY, François-Xavier (CERN)

ABERLE, Oliver (CERN)

LAMAS GARCIA, Inigo (CERN)

PERONI, Lorenzo (Università di Torino)

SCAPIN, Martina (Università di Torino)

SGOBBA, Stefano (CERN)

PEREZ FONTENLA, Ana Teresa (CERN)

GOMEZ PEREIRA, Laura (CERN)

LEAUX, Floriane (CERN)

In this contribution we will review the spallation materials R&D associated with the design, operation and maintenance of beam intercepting devices in the CERN accelerator complex.

Focus will be given to the redesign of the antiproton production target and the results of the HiRadMat experiment carried out in November 2015 to explore the behaviour of high-Z materials impacted by high intensity beams. Triggered by the extreme conditions encountered by high intensity beams impacting on high density materials, we will discuss the studies carried out to understand the behaviour of W and Ir at high temperature and high strain rates.

In the context of the LHC transfer lines, ring collimators and general beam intercepting devices, results will be presented on tests on graphite, carbon fiber materials and silicon when exposed to the extreme conditions of brightness of the LHC machine beams, including the results of another HiRadMat experiment carried out in early 2016.

R&D and investigation activities associated with Ta-cladding on W and TZM materials will also be shown as well as progresses on Ti6Al4V-contained Pb targets in the context of projects such as the new n_TOF spallation target.

The challenges and issues associated with the proposed high power SPS Beam Dump Facility at CERN will be shown as well.

Investigation of target material for muon production under high power proton beam irradiation

MAKIMURA, Shunsuke (KEK)

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Polycrystalline graphite is a principal target material for muon production under high power proton beam irradiation. Actually, it has been used in the muon targets at J-PARC and Rutherford Appleton Laboratory, in the meson target at the Paul Scherrer Institute, and in neutrino targets at Fermilab, J-PARC, CERN and others. Graphite has extremely high performance for these applications due to its thermal properties, mechanical properties, and chemical stability. However, graphite is easily oxidized at high temperature. In the case where air is unexpectedly introduced into the primary beam line during high power beam operation, the rapid oxidation of graphite target is detrimental to safe operations. Therefore, a robust closed area is required to enclose the consequent contamination. Furthermore, the damaged target must be replaced with a new one before restarting beam operation. The presence of graphite oxidation contaminants complicates these recovery procedures. In addition, since a smaller spatial volume of the source of the secondary particles is beneficial to more efficient transport to downstream experiments, the density of the target material should be higher. So, modifying the graphite or developing a replacement material for the graphite that is denser and more resistant to oxidation is desired. Recently, we started to investigate new target materials with higher performance. Silicon carbide (SiC) is an excellent candidate because it has a good heat resistance and high mechanical strength. The density of SiC is 1.8 times higher than graphite. The oxidation resistance of SiC is much higher than the graphite. But a monolithic SiC cannot be used as a structural material under the pulsed heating cycle, because it is brittle. To take advantage of the good mechanical properties of the graphite and the good oxidation resistance of the SiC, SiC coated graphite is investigated. Under the RaDIATE collaboration, the SiC coated graphite will be irradiated at BNL's BLIP facility and post-irradiation examination will be performed [1]. Simultaneously, NITE-SiC composite material, developed at OASIS group, Muroran Institute of Technology, is also investigated [2]. Brittleness of SiC is improved in this composite material. The SiC composite material will be irradiated and investigated at Research Center for Nuclear Physics, Osaka University [3]. In this presentation, the investigation of the target material for muon production will be described.

[1] http://radiate.fnal.gov/index.html [2] http://oasis.muroran-it.ac.jp/english/index.html [3] http://www.rcnp.osaka-u.ac.jp/index.html

Pulsed heavy-ion irradiation of tungsten

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Irradiation of pure tungsten has been carried out at the M3-beamline of the UNILAC facility at GSI Helmholtz Center for Heavy Ion Research. Tungsten specimens of two thicknesses (26 um and 3 mm) were exposed to 4.8 MeV/u uranium and gold ion beams for fluences up to 4.0e13 ions/cm^2. The repetition rate of the pulsed ion beam was 0.5-2 Hz with a pulse length of 150 us.

The experimental setup allowed for simultaneous measurement of the intensity of the incoming beam, the temperature and the mechanical response of the sample with each pulse. In order to study the beam pulse effect on the excitation and propagation of thermal stress waves, a Laser Doppler Vibrometer (LDV) was used. The LDV recorded the surface velocity of the tungsten foils as a function of time. A dynamic mechanical analysis was done using ANSYS, and a theoretical diagnosis was made for the experimental data.

In order to study the irradiation induced hardening as a function accumulated dose, nano-indentation tests are performed on the damaged surfaces of the samples. The measured data are compared with the calculated damage values, and a correlation between the radiation-induced damage and the observed mechanical property is presented. The thermal diffusivities of the different samples are measured with a Laser Flash Apparatus. The results are converted into thermal conductivity as a function of radiation dose and irradiation temperature.

In addition to pure tungsten, oxidized samples were irradiated for the purpose of investigating the oxide adhesion under a pulsed beam, to understand the mechanical behaviour of slightly oxidized tungsten surface in the presence of oxygen impurities in the helium coolant of the target. The irradiated oxide scale is examined using Auger Electron Spectroscopy (AES) and Scanning Electron Microscopy (SEM). Also, the luminescence of tungsten oxide is investigated, as it showed glaring beam spot during uranium beam irradiation.

Re-examination of ion irradiation as a credible tool to simulate high energy neutron and proton-induced void swelling for accelerator-driven devices

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Unavailability of high-flux test reactors requires that ion irradiation serve as a surrogate for investigating factors that lead to swelling resistance of improved alloys. Most current studies are directed toward lower-swelling ferritic-martensitic alloys and their ODS variants. The credibility of using charged particle simulation requires that the impact of all neutron-atypical features of ion irradiation be identified, understood and minimized. Two different approaches at TAMU and KIPT to accomplish this goal are presented and discussed. In order to demonstrate that ion irradiation is a credible tool, it is required that self-ion irradiation reproduce major aspects of neutron-induced swelling dependencies (compositional, fabricational, flux-spectral) observed in neutron tests. It should also reproduce the swelling behavior (bilinear, steady-state after incubation), but should especially reproduce the well-established post-transient swelling rates of 1%/dpa for fcc iron-base and 0.2%/dpa for bcc iron-base alloys. While studies at TAMU and KIPT show very clearly the bilinear swelling behavior of ferritic-martensitic alloys with a post-transient swelling rate of 0.2%/dpa, most studies on austenitic alloys were conducted in the 1970s-1990s, but did not show the expected 1%/dpa for fcc iron-base alloys. Many earlier studies, especially for simple metals and fcc iron-base alloys, are re-examined here in light of recently attained insights and current calculational practices. The results of this re-examination are very encouraging, attesting to the credible use of ion simulation for void swelling. It is shown that dpa calculational codes (EDEP, BRICE, IONDOSE, early versions of TRIM) used in earlier studies overestimated energy deposition rates by ~20-35%, leading to artificially high dpa levels and an incorrect visualization of swelling vs. depth. When these earlier data sets are re-evaluated using SRIM the 1%/dpa steady-state swelling rate is routinely observed. Additionally, injected-interstitial suppression of void nucleation, not clearly observed in earlier studies on austenitic alloys, is equally strong in both bcc and fcc iron-base alloys.

The RaDIATE collaboration – exploring high power target materials response to radiation damage – goals, status, and future plans

HURH, Patrick (Fermilab)

The RaDIATE collaboration (Radiation Damage In Accelerator Target Environments), founded in 2012, has grown to over 50 participants and 11 institutions globally. The primary objective is to harness existing expertise in nuclear materials and accelerator targets to generate new and useful materials data for application within the accelerator and fission/fusion communities. Current activities include post-irradiation examination of materials taken from existing beamlines (such as the NuMI primary beam window from Fermilab) as well as new irradiations of candidate target materials at low energy and high energy beam facilities. In addition, the program includes thermal shock experiments utilizing high intensity proton beam pulses available at the HiRadMat facility at CERN. Status of current RaDIATE activities as well as future plans will be discussed, including highlights of preliminary results from various ongoing RaDIATE activities and the high level plan to explore the high-power accelerator target relevant thermal shock and radiation damage parameter space.

Development of advanced ferritic steels for high dose applications

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ANDEROGLU, Osman (University of New Mexico)

SALEH, Tarik (LANL)

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The Fuel Cycle R&D program is investigating options to transmute minor actinides. To achieve this goal, new fuels and cladding materials must be developed and tested to high burnup levels (e.g. >20%) requiring cladding to withstand very high doses (greater than 200 dpa) while in contact with the coolant and the fuel.

To develop and qualify materials to a total fluence greater than 200 dpa requires development and testing of advanced alloys and irradiations in fast reactors. Specimens of previously irradiated HT9 specimens are being irradiated in a fast reactor to high doses (>200 dpa). In addition, improvements in the radiation tolerance of HT-9 are being made through minor chpanges in the composition. Advanced radiation tolerant materials with fine oxide dispersion strengthening are also being developed to enable the desired extreme fuel burnup levels. Recent progress in high dose irradiated materials testing and materials development will be presented.

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Tensile testing of steels from the STIP-V irradiation

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ROMERO, Tobias (Los Alamos National Laboratory)

DAI, Yong (Paul Scherrer Institute)

We will present the results from the ambient and elevated temperature tensile testing on samples from the STIPV irradiation at the Paul Scherrer Institute. Samples were of various F/M and ODS steels, and irradiation conditions range from 6-15dpa dose at irradiation temperatures from 135-465°C. Due to the irradiation environments in the target of the SINQ accelerator, helium injection ranged from 245-950 APPM. Tensile tests were performed at the Wing 9 Hot Cells at Los Alamos National Laboratory. Tensile results of MA957, MA956 and CROFERAPU alloys will be presented, and compared to similar alloys from the previous STIP-IV irradiation, and other neutron irradiations in reactors.

Tensile properties characterization of irradiated AISI 316L from high-use target modules at the Spallation Neutron Source using digital image correlation

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GUSSEV, Maxim (Oak Ridge National Laboratory)

CAMPBELL, Cody (BWX Technologies Inc.)

GARNER, Frank (Texas A&M University)

A Post Irradiation Examination (PIE) program is maintained at the Spallation Neutron Source to inspect and test material from SNS target modules after removal from service. Tensile specimens were recently produced and tested from SNS Targets 6, 8, and 9; the maximum damage dose reached was approximately 8 dpa. Tensile tests were performed in conjunction with digital image correlation (DIC) characterization, which permitted detailed analysis of the strain distribution and evolution along the specimen gauge sections. For each specimen, true stress - true strain curves were calculated using the experimental DIC data set. Deformation hardening behavior was analyzed in the terms of the plasticity constitutive equations and deformation hardening parameters were discussed as a function of damage dose and helium/hydrogen content. Local yield stress concept was discussed and evaluated to address the possible effects of damage dose gradients. For many specimens, deformation bands were observed in the small strain area; the band parameters were analyzed in detail. Scattering of the mechanical test results was analyzed using the calculated DIC strain maps. The results show while considerable hardening (~700 MPa increase in yield strength) occurred in the 316L SNS target material, appreciable ductility (18-24% total elongation) remained in the material after irradiation to 7-8 dpa in the mixed neutron/proton environment of the SNS target region.

Swelling, creep, and embrittlement of D9 stainless steel cladding and duct in two FFTF driver fuel assemblies at high neutron exposures

GARNER, Frank (Texas A&M University)

Most data sets on void swelling and irradiation creep of austenitic structural steels were derived in relatively small amounts, making it difficult to extract full parametric dependencies for development of predictive correlations of dimensional change. This paper presents a much larger data base derived from fuel cladding and ducts constructed from D9 steel, an improved titanium-modified variant of AISI 316 stainless steel that has been selected for applications to various candidate reactors, including several accelerator-driven spallation concepts.

This report focuses on the swelling, creep, length change, ovality and embrittlement behavior of 20% cold-worked D9 cladding and duct used in two mixed-oxide driver fuel subassemblies designated D9-2 and D9-4. These 217-pin assemblies were irradiated in the FFTF fast reactor to maximum exposures of 25.3 and 21.4 x 10^{22} n/cm2 (E>0.1 MeV) or 115 and 96 dpa, respectively. The D9-2 subassembly operated at lower temperatures compared to those of D9-4, leading to somewhat different swelling behavior. Since the D9-4 duct operated at lower temperatures than the cladding, the swelling of the duct was relatively low, peaking at 6-7%. The fuel pin cladding reached swelling values of 21-28% in D9-4 and 37-38% in D9-2, with much of the in-core portion of the pins having attained the terminal swelling rate of ~1%/dpa. The well-known "creep disappearance" phenomenon was observed to develop at moderate swelling levels.

Void swelling was found to vary with dpa rate, irradiation temperature and small heat-to-heat differences in composition. The latter involved relatively small differences in phosphorus content, but produced significant differences in swelling. Compared to data sets derived from the smaller EBR-II fast reactor, it is shown that the temperature dependence of void swelling in the much larger FFTF is rather invariant over a large range of temperatures. While no pin failures were observed during in-reactor operation, failure arising from severe void-induced embrittlement occurred in several D9-2 fuel pins and duct during post-irradiation handling.

The behaviour of $AlMg_3$ after irradiation at high proton and neutron fluences in SINQ targets

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BRUN, Roger (PSI)

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The Al-alloy $AlMg_3$ has been used as the material of many important components in the Swiss spallation neutron source (SINQ), such as the target safety container, the moderator tank container, the central tube in which the target inserted. These components expose to either high proton plus neutron flux or neutron flux. Radiation damage and helium and hydrogen etc. may induce a strong degradation in mechanical properties of the material and further limit the lifetime of the components. Understanding the behavior of the $AlMg_3$ material in these components is essential to assess the lifetime of them, which is important for the safe operation of the target. Among these components the target safety container receives the highest proton and neutron fluxes, particularly in the proton beam window area. To evaluate the mechanical properties of $AlMg_3$ in the beam window, tensile specimens were extracted from the containers of SINQ Target 3, 4 and 9, which received proton charge of 6.8 Ah, 10.03 Ah and 13.16 Ah, respectively. The maximum proton fluence received by Target 9 is 8.56E+25 p/m^2 which is the highest one ever received by Al-alloys in spallation targets. Tensile test results show that the embrittlement induced by irradiation increases with irradiation dose. At the highest irradiation dose, the samples are broken immediately after yielding in a cleavage fracture mode. This implies that the upper limit of proton fluence for the container is reached. In this talk the results of tensile test, SEM and TEM observations will be presented. The lifetime of the $AlMg_3$ safety container will be discussed.

Post irradiation examination of the MEGAPIE samples at JAEA (2)

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The world's first megawatt-class lead-bismuth target, MEGAPIE (MEGAwatt Pilot Experiment), was dismantled and post irradiation examination (PIE) samples were prepared at PSI hot-lab. The samples were shipped to each institutions including JAEA. The samples were cut from the beam window (BW, T91) and the flow guide tube (FGT, SS316L). And all samples are prepared without LBE. The irradiation conditions of the specimens irradiated at SINQ target were as follows: proton energy was 580 MeV, irradiation temperatures were ranged from 251 to 341°C, and displacement damage levels were ranged from 0.16 to 1.57 dpa. PIE including SP (small punch) and three point bending tests were performed. SP tests were executed for T91 and SS316L specimens at R.T. in air condition. Specimen size for SP test with 2.4 mm steel-ball is 8 mm x 8 mm x 0.5 mm. T91 specimens were cut from the "Spitze (triangle)" sample and polished to thickness of 0.5 mm. The OM/SP specimens of SS316L were polished to thickness of 0.5 mm. Three-point bending tests were executed for SS316L specimens at R.T. in air condition. The bend bar specimens of SS316L which size is 1.7 mm x 1.7 mm x 16 mm without notch were employed. Results of the SP tests and three-point bending tests on the irradiated specimens will be presented at the workshop. Cross sectional observation on the Spitze sample and microstructural observation by TEM will be also reported.

The transmission electron analysis of dislocation loops in T91 steels from MEGAPIE and TWIN-ASTIR irradiation programs

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The defects formed under neutron and proton irradiation in T91 steel, extracted from the MEGAPIE and TWIN-ASTIR irradiation programs were analyzed by transmission electron microscopy. The focus was given to the evolution of dislocation loops as a function of irradiation dose, temperature, and combined neutron/proton versus only neutron irradiation. Different spatial distributions of dislocation loops were observed as a function of dose in MEGAPIE T91 samples. These results are compared with similar observations in neutron irradiated FeCrC within the GETMAT project. Different mechanisms which could be the origin of homogeneous versus heterogeneous dislocation loop distributions are discussed in the light of modeling results.

Barrier strength of defects and helium bubbles for hardening of martensitic steels irradiated in STIP

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Reduced activation ferritic/martensitic steels are considered as baseline structural materials for fusion DEMO and test reactors in future. Structural materials in fusion reactors will undergo high displacement damage and transmutation helium/hydrogen by intense fluence of high energy neutrons. China low activation martensitic (CLAM) steel, F82H and Optimax-A have been irradiated in the SINQ target irradiation program (STIP). A series of studies have been conducted on the mechanical properties and microstructure of CLAM, F82H and Optimax-A steels irradiated in STIP to investigate the dose, temperature and helium effects on hardening and embrittlement after irradiation.

To investigate the effects of helium on hardening after irradiation, annealing experiments at 600°C were done to recover the irradiation-induced changes in mechanical properties and microstructure. Micro-hardness tests and TEM observations before and after annealing of CLAM, F82H and Optimax-A samples irradiated to 5 - 21 dpa at 56 - 471°C were investigated to analyze the relationship between hardening and quantitative results of defects and bubbles. Defects and helium bubbles were observed in CLAM specimens at all doses. The microstructure and hardness of the same specimens before and after annealing could demonstrate that the hardening in the as-irradiated specimens is attributed to a combined effect of defect clusters and bubbles, while that in the annealed specimens should be essentially contributed by helium bubbles. Barrier strength (α) of defects and helium bubbles for hardening was evaluated from equations $\Delta\tau = \alpha Gb(Nd)1/2$ and $\Delta\tau 2 = \Delta\tau defect2 + \Delta\tau bubble2$. With the good relationship between hardness and strength, the relationship between hardening and quantitative results of cluster, dislocation loop and bubble were analyzed.

Mechanical properties and fracture behavior of pure tungsten and tantalum after irradiation in SINQ

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Tungsten has been used as target material in several spallation neutron sources in the world. Due to its poor water corrosion resistance, tungsten has to be clad with tantalum in solid targets which are with a water cooling system. Although pure tungsten and tantalum have been widely used for many years in spallation targets, the understanding of their mechanical properties and fracture behavior in spallation irradiation environments is still very poor. In the SINQ Target Irradiation Program (STIP) some specimens of pure tungsten and tantalum were irradiated at various dose levels and temperatures. In STIP-V, tungsten specimens were irradiated to doses up to ~29 dpa at temperatures up to ~560°C. In STIP-II tantalum specimens were irradiated to about the same maximum dose, but at temperatures below 320°C. In this work, due to the high activity of the STIP-V tungsten specimens, just those of low doses, 1 - 3.5 dpa, were tested. In order to detect the ductile-to-brittle transition temperature (DBTT) of the specimens, bend testing was conducted at temperatures up to 450°C. The results showed all the specimens broken in a brittle fracture mode, even at such low doses. Tantalum specimens were tensile tested. The results indicated the DBTT of tantalum increased slowly with irradiation dose. The fracture behavior of the tested samples was investigated using scanning electron microscopy (SEM), which exhibited intergranular fracture for specimens of both materials broken in brittle manner.

Annealing effect on the microstructure and hardness of irradiated tungsten

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The European Spallation Source (ESS) will be the world's most powerful neutron source facility built in Lund, Sweden [1] as the demand is continuously growing for neutron scattering and imaging techniques for material characterization under controlled conditions. The neutrons are produced through a spallation process in tungsten which has high neutron production rate due to its high atomic number. Tungsten is environmentally friendly compared to other target materials, and its use in helium environment avoids corrosion issues related to water cooling. Its disadvantages, however, are its low ductility and a high ductile-to-brittle transition temperature. Tungsten is the most critical non-structural material and it must operate reliably and predictably for the planned lifetime of the target. In order to estimate the target life, reliable data is needed on the mechanical properties of tungsten after irradiation. Dedicated experiments were established in the Paul Scherrer Institute to examine the behavior of tungsten under representative operational conditions.

In this study, we evaluate the microstructure and hardness of irradiated tungsten sample of 1.6 dpa after annealing in various elevated temperatures (500°C, 600°C, 700°C, 800°C and 900°C for 1 hour). We compare the structure, defects and dislocations in annealed irradiated and unirradiated samples. Focused ion beam (FIB) was used to prepare transmission electron microscopy (TEM) lamellas which were thinned by flash electrochemical polishing. Several low and high magnification TEM images were obtained in different areas of the sample to obtain quantitative information of the dislocations and defect clusters. The hardness of the tungsten has increased after irradiation (from 436 to 650 $HV_{0.5}$, respectively). There is a significant difference in the tungsten microstructure after irradiating, as observed with TEM. The average dislocation density slightly increases by 18% from 16.4 /μm^2 to 19.4 /μm^2 after irradiation. The defect cluster density is 3.20E+05 piece/μm^3 after irradiation. Annealing the unirradiated and irradiated samples in 500°C for 1 hour resulted only in a minor change in hardness. Initial results of annealing in 500°C show no change in the dislocation and cluster density, nor in cluster size distribution.

Formation of oxide layers on tungsten in mildly oxidizing gas mixtures

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Tungsten has been chosen as the target material for the European Spallation Source (ESS) in Lund. The target is cooled by pure helium gas at a flow rate of 150 m/s and 10 bar pressure. As the target can momentarily reach temperatures above 500°C due to beam pulses, impurities in the helium gas, like oxygen and moisture, can lead to the oxidation of tungsten. In the present work, the oxidation behavior of tungsten has been studied using simultaneous thermal analysis (W-foils) and thermogravimetry (disc specimens). The oxidized samples were characterized by Environmental scanning electron microscopy (ESEM), Energy-dispersive spectroscopy (EDS), Auger Electron Spectroscopy (AES) and X-ray diffraction (XRD).

Using a hot stage in an ESEM, non-isothermal studies on the oxidation of a tungsten foil in an atmosphere of water vapor at low pressure (100 Pa) were carried out in situ. The results show that changes in surface morphology are significant only above 600-700°C.

Tungsten discs (20mm in diameter, 3mm thick) were oxidized isothermally at 400 to 900°C for 2 hours in various gas mixtures - He (pO_2 less than 5 ppm atm), He+Ar+H_2O (pH_2O 0,0078 atm), and He+0.5% O_2 (pO_2 less than 0,005 atm). Oxidation of tungsten was observed in all these gas mixtures, even at 400°C. The results show that the oxidation follows a mixed parabolic and linear rate law. At low temperatures (400 - 500°C), the oxide layer is dark, thin and adhesive, and the growth rate is low. Beyond 600°C, the oxide layer was observed to be greenish, thick and porous. The activation energy for the phase boundary controlled reaction was found to be approximately 64 kJ/mol for the He+0.5% O_2 gas mixture, and 74 kJ/mol for the He+Ar+H_2O gas mixture. For the diffusion controlled reaction below 750°C, corresponding activation energy values were estimated to be 95 and 183 kJ/mol.

The oxides formed on the surface of tungsten were characterized using AES, XRD and EDS. The AES study confirmed the presence of a thin oxide layer on specimens oxidized at 400°C (45nm after 2 hours, pH2O 0.0078 atm.). At 500°C, the oxide layer was 10 times thicker. XRD and EDS methods were used to identify the oxides formed and their compositions. Layers with varying stoichiometry were observed, ranging from WO_3 at the surface to WO_2 at the metal-oxide interface.

Low-Z material R&D and application for Beam Intercepting Devices (BID) at CERN

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In the framework of the Large Hadron Collider (LHC) injectors upgrade (LIU) and the High-Luminosity Project at CERN (European Organization for Nuclear Research, in Geneva Switzerland), absorbers in the (Super Proton Synchrotron) SPS to LHC transfer lines as well as ring collimators in the LHC will undergo important modifications in the forthcoming years, mainly focused during the Long Shutdown 2 foreseen in 2019-2020.

In this contribution we will present the low-Z materials R&D related to the high intensity beam test on carbon based materials for the upgrade of SPS-to-LHC Transfer Line Collimators and LHC Injection Absorbers, carried out during 2016 as well as foreseen in 2017.

Focus will be given to the engineering studies of interesting carbon based materials for CERN collimators and the results of the HiRadMat experiment carried out in April 2016. Triggered by the extreme conditions encountered by high intensity beams impacting on low density materials, we will discuss the studies carried out to understand the behaviour of Isostatic graphite and 3D Carbon Composite (3D CC) at high temperature and high strain rates.

In the context of the LHC transfer lines, ring collimators and general beam intercepting devices, results will be presented on tests on graphite and 3D Carbon Composite (3D CC), including perspectives for future studies.

Fatigue properties of tungsten from different processing routes

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Pure tungsten has been chosen as the target material at the European Spallation Source presently under construction in Lund. In this facility, a high energy (2 GeV) - high power (5 MW) pulsed proton beam will irradiate the spallation volume and the target is subjected to cyclic thermo-mechanical loading caused by beam trips and beam pulses. In order to minimize the probability of fatigue failure, a good understanding of the fatigue behavior of tungsten is important in designing the target station.

In this study, the tensile and fatigue properties of pure tungsten from two different processing routes have been determined at room temperature. The rolled and annealed (RA), as well as sintered and hot isostatically pressed (SH) tungsten test specimens were supplied by Plansee Metall GmbH, Austria. Microstructural examination showed that the rolled material had very large, elongated grains (max. 1000μm×50μm) and the HIPed material had an average grain size of -30μm, with finely distributed micropores. Tensile tests showed that both types of specimens were associated with a low total strain (-0.25%) and negligible plastic strain. However, the rolled specimens had a much higher ultimate tensile strength (-1 GPa) as compared to the HIPed specimens (565 MPa).

Due to the brittle behavior of tungsten at room temperature, the fatigue strength was determined primarily by stress-controlled testing, using the staircase method and a stress ratio approaching zero. The results showed fatigue limits of 350 MPa and 180 MPa for the rolled and HIPed materials, respectively. These values correspond to a runout limit of 2.0E6 load cycles during which no specimen failure occurred. In addition, results from strain-controlled multiple-step testing (strain ratio = 0) indicated a slight relaxation in stress for the HIPed material while the rolled specimens showed the opposite behavior with a marginal increase in stress with increasing cycle count. The results obtained in this study have been compared to previous fatigue test data with rolled and forged tungsten specimens, and a correlation between the degree of grain deformation and direction dependent tensile/fatigue properties is presented. The implication of the fatigue test results on the spallation target design is discussed.

Luminescent materials development for beam-on-target imaging at the European Spallation Source

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Luminescent material deposited onto surfaces of the European Spallation Source (ESS) target wheel, proton beam window and tuning dump will provide images of the high power proton beam. To date, no coating has been identified that possesses the required properties for this challenging application. These properties include: tolerance to the radiation environment of the 5 MW spallation target, a short luminescence lifetime that allows imaging within the 2.86 millisecond pulse, adequate luminescence yield at temperatures exceeding 200 degrees C, and mechanical integrity throughout the life of the target components. Building on work first performed in support of a similar imaging system at the Oak Ridge National Laboratory's Spallation Neutron Source, an international collaboration has launched a campaign to develop materials and coating processes in time for ESS target construction and commissioning. We report here on the first series of experiments for the down-selection of the luminescent coating technology. The studies carried out so far have probed the mechanical integrity of the coatings, the luminescence yield, the luminescence lifetime, the emission spectrum and the temperature dependence. These functional characteristics were measured in response to photon and proton bombardment. In addition, X-ray and neutron scattering techniques have provided information about the structural and chemical characteristics of the samples. To assess changes induced by the anticipated radiation environment, several irradiation experiments are planned, including bombardment by low energy ions, as well as in-situ irradiation in target stations.

Design and fabrication of a passive irradiation module utilizing the high neutron flux from the 5 MW Spallation Source at ESS

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A high level of particle radiation field created by the spallation process limits the lifetime of the beam intersecting and moderating components near to the 5 MW target at ESS, due to radiation induced material degradations. These components are exposed to a high flux of protons and neutrons up to a GeV level of kinetic energy. Understanding material degradation under such a radiation environment is important in setting optimal service lifetimes of affected components and thereby realizing cost efficient operation of the facility. While there are extensive studies on the radiation damage of materials in reactor environments, there is a limited amount of data available for proton and fast neutron irradiations.

A STIP type passive irradiation module is being built in the pre-moderator at ESS, which receives the neutron flux with a kinetic energy from sub-meV up to 1 GeV. This irradiation module is dedicated to the study of the radiation damage of the materials of potential target station system applications, for a longer lifetime and a higher reliability of the facility. In this paper, the design and the fabrication status of the irradiation module are presented.

The irradiation module operates parasitically to the operation of the pre-moderator and is cooled by the water flow that moderates the neutrons from the target. A typical stainless steel alloy sample in the module will receive a maximum radiation damage of 7 dpa per operational year at 5 MW. The calculated maximum helium production rate in steel is 110 appm per operational year, making the helium production to dpa ratio of 15.6 appm/dpa. This irradiation condition is also suitable for fusion steel irradiation. As the fusion steels shows a good fast neutron radiation resistance in view of helium embrittlement and swelling, these could also serve as candidate materials for the target vessel at ESS, which receives direct proton beam. For this reason, this module also contains some material samples of common interest to fusion engineering.

A STIP type passive irradiation module is being built in the pre-moderator at ESS, which receives the neutron flux with a kinetic energy from sub-meV up to 1 GeV. This irradiation module is dedicated to the study of the radiation damage of the materials of potential target station system applications, for a longer lifetime and a higher reliability of the facility. In this paper, the design and the fabrication status of the irradiation module are presented.

The ESS helium cooled rotating target

SORDO, Fernando (Consorcio ESS-Bilbao)

The European Spallation Source is an ambitious European project to build a new generation Spallation Neutron Source in Lund (Sweden). The total budget of the project is €1,800,000,000 shared between 17 European countries. The main characteristic of the source is its 5 MW long pulse proton beam which will impact on a tungsten rotating Target cooled by helium gas.

On November 2014, ESS-Bilbao was chosen as ESS partner for the Target wheel, shaft and drive unit. Along 2 years ESS-Bilbao has developed the engineering design to fulfill the technical specifications of ESS and introducing significant modifications compared with the ESS-TDR proposal.

This paper summarizes the Target works, including protyping activities for the different subcomponents: Spallation material, internal structures, vessel, shaft and drive unit.

Design modification of ISIS TS2 target in order to improve longevity amid spallation reactions

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The working front of a target of ISIS Target Station 2 (TS2) is an assembly of Tantalum clad Tungsten core and a Tantalum water cooling jacket. The aim of the tantalum cladding is to prevent the tungsten coming in contact with the pressurized and fast cooling water. However a gamma spectral analysis on the cooling water after a target has failed shows the evidence of transmuted products of tungsten into the water. The neutronic analysts at RAL have found that the activity of some transmuted products such as isotopes of 175Hf and 172Lu are significantly higher than if they were produced from the tantalum cladding layer. A Monte Carlo simulation suggests that the origin of these transmuted products is the Tungsten core. An investigation on the design and manufacturing process of the target has helped us to identify that the electron beam weld joints on the tantalum cladding as one of the areas of weakness. The penetration of the weld using the existing method was found to be inadequate and the grains under the weld line were found to be significantly enlarged. A failure of the cladding through the weld joint would cause the target life to be significantly shortened. In order to improve the weld penetration, a study using different settings of the weld parameters such as weld current, beam rotation and beam focus has been carried out. The study concludes that maximum weld penetration could be achieved using a combination of high weld power, sub surface focus of the electron beam and low material volume around the weld line. A significant modification in the design of the front of the target has also been undertaken to ensure full penetration at the weld joints. This will ensure a delay for water coming in contact with Tungsten through the welding joint. In addition recent EB welding trials with a revised strategy, using multiple pass electron beam welding at different energies, has been successful in achieving 4500µm weld penetration. The researchers also considering Rotary Friction Welding and Tantalum Cold Spray technique to develop the cladding layer on the tungsten core. An early success has been achieved regarding weldability between Tungsten and Tantalum using Rotary Friction Welding.

Simulating performance of tantalum-clad tungsten targets

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Tantalum-clad tungsten targets are important for many accelerator facilities. Significant work on simulating the performance of such targets has been carried out at the ISIS spallation neutron source, driven by a desire to improve the lifetime of existing Target Station 2 (TS2) targets and to develop an improved target for Target Station 1 (TS1).

Detailed FEA simulations were created for the current TS1, upgrade TS1 and current TS2 targets. Possible causes of target failure were investigated. The main concern was found to be failure of the tantalum cladding leading to corrosion of the tungsten core. Factors limiting the cladding lifetime include manufacturing pre-stress, fatigue due to beam-induced thermal stress, and radiation damage. Manufacturing issues such as weld quality must also be considered.

Simulations predict that the Hot Isostatic Press manufacturing process will introduce a large tensile pre-stress in the cladding, which may adversely affect fatigue life. Preliminary neutron diffraction experiments to measure this pre-stress have been carried out using the Engin-X instrument at ISIS, and the results were compared to FEA simulations. Fatigue loading was calculated for targets with and without pre-stress. There is only limited fatigue data available for tantalum; fatigue tests are planned using ISIS tantalum to provide more data in the region of interest. Likely effects of radiation damage were considered qualitatively, but lack of data has prevented any detailed simulation.

Simulation results for current ISIS targets were used to set design limits for the TS1 upgrade target. The new target will be less highly stressed than the current targets, despite being more highly optimized for neutronics. The potential for pushing future tantalum-clad tungsten targets to higher beam powers was investigated, and the likely limiting factors for target design considered.

Manufacturing of ESS cold moderator – machining, welding and testing of Al 6061-T6 alloy

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The European Spallation Source (ESS) will deliver the world's highest flux of cold and thermal neutrons for basic and applied researches investigating the molecular building blocks of matter. In normal operation, the ESS spallation target will be operating at 5 MW proton beam power. The highly energetic proton beam delivered by the linear accelerator hits the tungsten target where the spallation neutrons are produced. The fast neutrons are moderated to cold and thermal neutrons by a moderator and reflector system located in the close vicinity of the spallation target. The engineering challenge is to achieve an optimal balance between moderation, neutron gain and cooling. The current design is highly integrated which poses challenges in manufacturing.

In this paper the manufacturing and testing of the final ESS cold butterfly moderator is described. The cold moderator with 17 bar design pressure and a flow channel structure to enable a high hydrogen flow to be able to remove the neutronic heat deposition is challenging. The welds need to be placed outside the high stress areas. But at the same time the welds must be accessible and testable. Therefore, the cold moderator is unwind into an upper and a lower half. The cold moderator is high speed milled from two solid Al 6061-T6 work pieces to achieve high precision. After machining, to achieve high quality repeatable welding the work pieces must be carefully cleaned. Fat is removed with ultrasonic cleaning. Oxide layers are removed by pickling. New oxide layer build-up is limited by vacuum packing of the work pieces immediately after drying after the pickling bath. The complete welding seam is electron-beam welded in a vacuum chamber. A filler AlSi12 is used to avoid crack formation. After manufacturing and X-ray analyzing of the welds, burst tests with water and with liquid nitrogen, to consider the low operation temperature, has been successfully performed. Finally, a fracture mechanics analysis has been done to confirm the experimental.

Weldability of diffusion bonding between Invar alloy and stainless steel by hot isostatic pressing

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The mercury target vessel is used at the Japan Spallation Neutron Source (JSNS) in the Japan Proton Accelerator Research Complex (J-PARC). It was made of type 316L stainless steel (SS) with a big size of 1.3 x 1.3 x 2.5 m^3. At the rated operational power of 1 MW, it is replaced every year and all used target vessel is stored in the J-PARC site. Aiming at reducing the amount of radioactive wastes, we have been studying an advanced target structure on which front half of the target could be separated from the reusable rear part with a flange. Since two mercury flow channels of inlet and outlet and one helium gas line are connected in the flange, it is required to have high seal performance with the helium leak rate of less than 10-6 Pa・m3/s to prevent radioactive materials leak from the seal part. The important design condition is that the large thermal expansion of less than 0.008 mm is induced inhomogeneously in the flange during the proton beam injection.

We investigated the availability of Invar alloy (Fe-36%Ni) with the thermal expansion coefficient about one tenth smaller than that of SS for the flange. However there is no data about the corrosion resistance of the Invar alloy against the mercury. Therefore, the Invar alloy block is covered with the SS layer and they are bonded by the hot isostatic pressing (HIP).

In this study, bonding temperature were parametrically changed (973, 1173, 1373 and 1473K) under the constant treatment time and pressure (2 hr and 100 MPa) in order to find out the suitable bonding condition. The contents of Ni and Cr and the hardness in the diffusion layer changed gradually and the thickness diffusion layer increased with an increase of the treatment temperature. The tensile tests were carried out. The fractured at the bonding interface before the necking occurred for bonded at 973 K. The other bonded in the higher than 1173 K failed at the Invar alloy side. The grain size became large and the tensile strength decreased with an increase of the treatment temperature. Fatigue tests for bonded at 1173 K were carried out. The fatigue fracture occurred at the Invar side and fatigue life in the region of the low cycle fatigue. This tendency is same as that for the Invar bulk heat-treated at 1173 K. From these results, it was concluded that the optimum treatment temperature is 1173 K.

Measured SNS mercury target vessel strain responses to beam pulses and comparison to simulations with variations on mercury material model behavior

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The dynamic strain response of the SNS mercury target vessel has been measured from beam pulse induced pressure waves for the first time. Measurements have been made on two targets so far. Data was obtained using commercially available fiber-optic strain sensors. The first set of sensors, installed on the 13th SNS target, was manufactured of regular multi-mode fiber. For the 14th SNS target, these sensors were manufactured out of radiation-resistant multi-mode fiber and we also added prototype strain sensors, under development at SNS, made of highly radiation-resistant single-mode fiber. The strain measurements of the 13th target captured responses from 23 single pulses and 6 bursts of 10 pulses delivered at 60 Hz. Sensors failed at different integrated charge depending on their location on the target. Single pulse intensities ranged from 10.6 to 25.4 micro-C (0.6 and 1.4 MW equivalent at 60 Hz). The radiation-resistant multi-mode sensors on the 14th target lasted between 104 and 105 times longer than the standard sensors, and the single-mode fiber sensors approximately 3 times longer than the radiation-resistant multi-mode fibers. The results of the prototype single-mode strain sensors are in the process of being verified.

The proportionality of peak strain responses to pulse intensity were evaluated and found to be quite linear despite the non-linear cavitating behavior of the mercury. Measured responses were also compared to simulation results. The match is very good at the location closest to the beam axis and entrance point. At most locations away from this point the simulation matches the structural response character well but over-predicts peak strain magnitudes. This simulation deficiency was investigated through trial simulations varying mercury and steel material behavior parameters including bulk viscosity (affects volumetric strain rate response), mercury tensile cutoff pressure (cavitation threshold), mercury bulk modulus, and steel Rayleigh damping. This presentation will present, compare and discuss the measured and simulated strain response data and the lessons learned from the investigations on material behavior changes to improve the match to the measurements.

Failure analysis on the welded part with steep change of thickness employed in JSNS mercury target vessel

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At a high power spallation neutron source of J-PARC (JSNS), in 2015 there was trouble with the water-cooled shroud of a neutron production mercury target during the operating periods with a 500 kW proton beam power. The water-cooled shroud is surrounding a mercury vessel inside. These vessel and shroud are made of type 316L stainless steel and designed to endure pressures of mercury and water, thermal stress and dynamic stress induced by the pulse proton beam injection (Power: 1 MW, repetition rate: 25 Hz, pulse duration: 1μs).

Since the operational goal is 1 MW, redesign of the target is needed based on the cause analysis. However, visual inspection was difficult because water leakd inside of the shroud. We focused on the welding line because a defect may exist in it, especially in the positon where it is difficult to weld, and the defect in the water shroud have a potential to initiate water leakage path due to repeated stress in operation. In case of the JSNS water shroud, there was a welding line on which thickness changed steeply from 3 mm to 8 mm. We investigated this welding line by using a partial test model simulating its geometry and welding process along with numerical simulations. Furthermore, crack propagation analysis from the defect in welding was carried out based on the fracture mechanics.

As a result of a microscopic inspection, we found insufficient penetration of welding and a crack of ca. 1 mm in length with intergranular fracture surface near the position where the thickness changed steeply. A numerical analysis simulating welding process showed that temperature became lower near this position because the heat conduction volume increased with increasing thickness, indicating the cause of the insufficient penetration. The repeated stress range due to the 500 kW proton beam pulse reached 76 MPa. And it was estimated that the initial crack of more than 0.9 mm in length could propagate through the thickness from water channel to the outer surface of water shroud (8.5 mm) by the repetition of beam pulse up to occurrence of water leakage (2.5×10^7 times).

These results indicate the probability that water leakage path in the JSNS water shroud was formed by the crack propagation in beam operation from the initial crack generated in the welding process near welding with steep change of thickness.

Thermomechanical analysis of ESS spallation material

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The European Spallation Source is an ambitious European project to build a new generation Spallation Neutron Source in Lund (Sweden). The total budget of the project is €1,800,000,000 shared between 17 European countries. The main characteristic of the source is its 5 MW long pulse proton beam which will impact on a tungsten rotating Target cooled by helium gas.

On November 2014, ESS-Bilbao was chosen as ESS partner for the Target wheel, shaft and drive unit. Along 2 years ESS-Bilbao has developed the engineering design to fulfill the technical specifications of ESS and introducing significant modifications compared with the ESS-TDR proposal.

The proposed configuration for the Spallation material is based on 10x30x80 mm W brick in a cross flow configuration cooled by 2 mm helium channels. The complex geometry of the helium paths and the transient effects produced by the combination of beam and wheel spinning, force to develop a complex CFD strategy.

This paper summarizes the methodology for CFD and mechanical analysis and the operational conditions for the spallation material (temperature and stress) on steady state and transients.

Cavitation damage in double-walled mercury target vessel

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The Japan Spallation Neutron Source (JSNS) is designed to be operated with injecting 3 GeV, 1 MW proton beam at repetition rate of 25 Hz onto a mercury target at Japan Proton Accelerator Research Complex (J-PARC). A mercury target vessel, made of 316L stainless steel, with 3 mm thick at the beam window front is damaged due to the cavitation caused by the proton beam-induced pressure waves in mercury. Severe cavitation damage reduces the structural integrity of the target vessel, and so currently dominant to shorten service life of the target vessel for 2,500 MWh operation rather than radiation damage (5 dpa for tentative criteria). Aiming at mitigating the cavitation damage and the pressure waves by faster mercury flow in narrow channel, double-walled structure with the narrow channel was applied to the beam window front of target vessel in addition to injecting gas microbubble into flowing mercury.

First double-walled mercury target vessel was operated from October, 2013 to May, 2014, resulting in accumulated beam power of approximately 670 MWh for 1670 hours operation. Before replacing the target vessel, the beam window front was cut out using an annular cutter in order to examine the cavitation damage inside the target vessel. As a visual inspection, band-like damage was recognized on cut specimen facing narrow channel. For more detailed inspection, we employed replica method to measure the damage depth. As a result, there was the cavitation damage with maximum pit depth of approximately 25 μm on the band-like damage. Furthermore, we carried out numerical simulation for pressure wave propagation in order to investigate why the band-like damage was formed. The calculated result showed that the distribution of negative pressure period was correlated with the damage distribution in the narrow channel, and the power dependency for negative pressure period was relatively smaller than that of bulk side.

In the workshop, result of cavitation damage inspection will be presented, and the cavitation damage distribution in narrow channel based on the numerically calculated negative pressure period including power and beam profile dependency will be discussed.

Effect of oxygen concentration on LME susceptibility of CLAM steel in liquid lead bismuth eutectic

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Liquid lead bismuth eutectic (LBE) is one of the reference spallation targets and coolant materials for accelerator driven subcritical systems (ADS) due to its excellent heat transfer performance, lower melting point and good neutronics performance, etc. However, the use of LBE gives rise to the performance degradation of structural materials. And structural materials would likely suffer a risk of liquid metal embrittlement (LME) effect. An understanding of liquid LBE embrittlement on these structure materials is one of essential issues for the development of ADS. The China Low Activation Martensitic (CLAM) steel promises to be selected as the one of the promising candidate materials for the future's ADS due to its good strength and toughness at elevated temperature, as well as its low activation after high flux neutron irradiation. Previous studies revealed that the occurrence of LME was closely related to oxygen concentration at interface of solid/liquid. So the influences of dissolved oxygen on LBE embrittlement of CLAM were conducted at 400°C with a constant strain rate of 10^{-5} s^{-1}. The fracture surface and crack propagation behavior were examined by SEM, EDS and EBSD technologies after tested in LBE with the oxygen concentrations of 10-6 wt%, 10-7 wt% and 10-8 wt%, respectively. The results showed that cleavage crack sources appeared on the outer edges of fracture surface, and the sources number was likely independent on oxygen concentration. Also, the EBSD examination indicated that cracking induced by LME was transgranular and translath, and the plastic deformation around the crack tip reduced seriously when the dissolved oxygen decreased to 10-8 wt%. In brief, low dissolved oxygen in LBE promoted the occurrence of LME on CLAM steel, so more attention should be paid on the LME risk caused by the oxygen depletion in LBE at the solid/liquid interface or in crack tips. The effects of LME on the fatigue and creep properties of CLAM in LBE will be conducted in the future.

Keywords: Lead Bismuth Eutectic; China Low Activation Martensitic Steel; Liquid Metal Embrittlement; Oxygen Concentration

Numerical modeling of impurities mass transfer in a wire wrapped fuel assembly under flowing lead bismuth eutectic

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It is widely recognized that the coolant chemistry control is one of the key technologies for the development of lead-bismuth eutectic (LBE) cooled nuclear systems. In particular, oxygen is a critical chemical element for LBE coolant management, because of its main influence on the corrosion rate of structural materials and core internals and its potential contamination of the coolant by forming solid oxide (PbO). The oxygen concentration in LBE should be therefore continuously maintained within a targeted range between the dissolution limit of a protective oxide layer to avoid fast dissolution of the structural material elements and the solubility limit of oxygen to avoid formation of PbO during operation. The supplied oxygen will be consumed mainly by oxidation of fuel cladding. The amount of oxygen necessary for a continuous oxide scale formation on the fuel cladding depends on temperature, local oxygen concentration, flow velocity and type of steel. Thermodynamic requirements for the formation of a protective oxide layer should be evaluated at the interface between structural steels and LBE. Experimental data on steels' corrosion however often refer to bulk oxygen concentration. This is mainly because it is practically impossible to measure the local oxygen concentrations near to steel surface, especially for complex systems. Therefore, numerical modeling is important and necessary to bridge the gap on information needed to evaluate local oxygen concentration levels and, consequently, corrosion rates.

A numerical model of oxygen mass transfer in a 19-pin fuel assembly will be presented, taking into account the oxidation of fuel cladding materials. This simulation provides a full mapping of the oxygen concentration through a fuel bundle enabling the identification of critical areas in terms of oxygen depletion, and consequently corrosion. Furthermore, it is of primary importance to identify rate-limiting factors and determine representative experimental conditions for the R&D program on corrosion.

A numerical model of corrosion products release from a fuel assembly will be also presented assuming that the dissolution process of steels is limited by mass transfer. Different boundary conditions will be applied at the steels/liquid interface depending on the local oxygen content. This simulation will allow to quantify the effect of the oxygen concentration on mass transfer limited corrosion.

Development of SIMP steel for accelerator driven system in China

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Focused on the materials requirement for the development of ADS in China, a novel material with a whole martensitic phase, tentatively named SIMP steel, has been fabricated under the collaboration between Institute of Modern Physics and Institute of Metal Research, CAS. The SIMP steel will be used in Lead-Bismuth eutectic (LBE) environment and it will support some innovative design of CIADS. In the past several years, optimized design, fabrication, processing and evaluation for the SIMP steel have been done, and a pilot scale of 5,000 kg SIMP ingot has been achieved. The routine performance, high temperature properties as well as resistances to intense ion irradiation and LBE corrosion of the new material are systematically tested. It is found that SIMP steel exhibits good properties under high temperature, high stress, liquid metal corrosion and ion irradiation.

In the present report, the current status of the ADS project in China and the research progress of the SIMP steel will be introduced.

Low cycle fatigue behavior of 15-15Ti steel in static lead-bismuth eutectic with 10-6wt% oxygen concentration at 550°C

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Accelerator Driven subcritical System (ADS) has been considered as an effective technological solution for nuclear waste transmutation. Chinese Academy of Sciences (CAS) had launched an ADS research project since 2011. China LEAd-based Reactor (CLEAR) which was designed by Institute of Nuclear Energy Safety Technology (INEST/FDS Team), CAS, was selected as the reference reactor for ADS. The first stage of the project is to construct a 10MW lead-bismuth cooled research reactor named China Lead-based Research Reactor (CLEAR-I). The mechanical properties of structural materials in Lead-bismuth eutectic (LBE) may exhibit the tendency of degradation, and the extent of degradation is closely related to dissolved oxygen concentration. The candidate material for fuel cladding tubes of CLEAR-I is 15-15Ti steel and will suffer from cyclic stresses and strains. Therefore, it is necessary to study the fatigue behavior of 15-15Ti in LBE under the work condition to ensure the safety of the fuel cladding tubes. In this paper, the low cycle fatigue behavior of 15-15Ti steel in static LBE was studied with a self-designed fatigue test system at 550°C and 10-6 wt% dissolved oxygen. The experiments were carried out with the symmetrical cycling (R=-1) and a frequency of 0.5 Hz and a total elongation ($\Delta\epsilon t/2$) between 0.3% and 2%. After the fatigue tests, surface oxide scales, fatigue cracks, fracture features of the samples were analyzed respectively, with scanning electron microscopy (SEM), energy dispersive spectrometry (EDS) and electron backscatter diffraction (EBSD). The 15-15Ti steel in static LBE did not show significant degradation of fatigue life under the test conditions of low strain amplitude, slow strain rate and 10-6 wt% dissolved oxygen, compared to air. The microscopy analysis indicated that the surface oxide film blocked the propagation of fatigue crack. The paper showed that the fatigue life of 15-15Ti steel in static LBE under above conditions could be expected with its fatigue data in air.

Corrosion behavior of Ti_3SiC_2 in flowing lead bismuth eutectic at 1000ºC

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Lead Bismuth Eutectic (LBE) has been considered as one of the promising candidate coolants and spallation neutron target materials for the applications in lead alloy cooled fast reactors (LFRs) and accelerator driven systems (ADS), due to its favorable thermal-physical, chemical, and neutron properties. Compatibility of cladding materials with the coolant at temperature above 650°C is required for the feasibility of the high-efficiency LFRs. It is expected that Ti_3SiC_2 can be the candidate of the cladding materials in such reactors. So the corrosion behavior of this material in lead alloy above 650°C, especially above 800°C is important in order to improve the efficiency of LFR. To evaluate the corrosion behavior of the candidate structural materials in LBE at elevated temperatures, an ultra-high temperature thermal convection loop was designed and built in Institute of Nuclear Energy Safety Technology, Chinese Academy of Sciences. The designed maximum temperature for the loop is 1000°C and the flow velocity in the test section is about 0.1 m/s. In this study, corrosion behavior of Ti_3SiC_2 in flowing LBE was carried out in the ultra-high temperature loop at 1000°C for 1000 h. The weight and composition of the specimens before and after corrosion were tested and analyzed. The microstructure of the specimens was observed with scanning electron microscope (SEM). The results showed that a double-layer oxide formed on the Ti_3SiC_2 after corrosion. The external layer of the oxide was composed of TiO_2 with inclusions of Ti-Pb-Si-O and the internal layer was composed of TiO_2 with inclusions of SiO_2. And no dissolution attack was observed after corrosion. Longer time experiments will be carried out further in the near future.
Keywords: Lead Bismuth Eutectic; Ti_3SiC_2, Corrosion, Lead Alloy Cooled Fast Reactors

Oxidation behaviors of CLAM steel in stagnant liquid lead-bismuth eutectic at 500°C

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Liquid lead-bismuth eutectic (LBE) is under wide investigation and development for accelerator driven subcritical systems (ADS) and high-power spallation neutron targets, due to its high neutron economy and good thermal-physical properties etc. However, one of the key challenges of design and application of LBE cooled systems in advanced nuclear reactors is the ability of available structural materials to resist corrosion. Reduced activation ferritic/martensitic steels (RAFMs), which replace elements producing long-lived isotopes by reduced activation elements, are considered suitable for nuclear applications, as their sensitivity to swelling under irradiation is lower. In this respect, some investigators have contributed in assessing the corrosion and oxidation behaviors of RAFMs in contract with LBE. China low activation martensitic (CLAM) steel, as one of the main RAFMs, was exposed to static LBE with around 10-6, 10-7, and 10-8 wt.% dissolved oxygen at 500°C up to 2,000 h, in order to investigate the corrosion and oxidation behaviors of CLAM steel. The microstructure, chemical composition, and thickness of the oxide scale of the exposed steels were characterized by means of scanning electron microscopy and energy-dispersive X-ray spectrometry. The experimental data showed that a multi-layered oxide scale, consisting of Fe_3O_4, $Fe(Fe_xCr_{1-x})$ 2O4 and internal oxidation zone (IOZ) was formed, which could isolate the steel matrix from LBE on the surface of CLAM steel. The thickness of oxide scale was a function of dissolved oxygen concentration and exposure time. The results indicated that CLAM steel had potential in the application of LBE cooled advanced nuclear systems. Keywords: Lead Bismuth Eutectic; Corrosion; Oxidation Behaviors; CLAM Steel

Positron lifetime calculation of vacancy clusters in tantalum containing hydrogen and helium

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Tungsten (W) is usually selected as the material for solid target of a spallation neutron source because of its high neutron yields. However, corrosion resistance of W is poor in the cooling water. Tantalum (Ta) claddings can solve the problem of the corrosion, therefore, Ta-cladded W is considered to be the optimized target for spallation neutron source. Investigation of mechanical property changes and microstructural evolution in Ta under high-energy neutron irradiation is important to maintain the safety of target of spallation neutron source. In the present study, the effects of helium and hydrogen, which are produced by nuclear reaction, on lifetime of vacancy clusters were investigated using quantum-mechanical electronic-structure calculations based on the density-functional theory (DFT). Positron lifetime calculations were carried out using the Kohn-Sham approach based on the DFT. In order to investigate the effects of H and He bound with vacancies on the positron lifetime, DFT calculations were performed as a function of the vacancy-clusters size in Ta containing H or He separately and H and He complexly. The lattice constant for Ta is 0.33 nm while the distance between the bulk nearest-neighbours is 0.2858 nm. The cubic cell with periodic boundary contains 2000 atoms. The equilibration process was 5 ps and the calculation of positron lifetimes was made after another 40 ps. The calculation results indicate that the lifetime of perfect Ta is 117.3 ps, and it is 216.0 ps when single vacancies are exists. The lifetime increases with increase in number of vacancy in vacancy cluster. The lifetime of single vacancies decreases to 191.7 ps when the vacancy contains a H atom, while it decreases to 158.6 ps when the vacancy contains a He atom. It is clear that the effect of He on lifetime change is larger than that of H.

Investigation of samples of F/M and ODS steels irradiated in the spallation source SINQ by positron annihilation

KRSJAK, Vladimir (Paul Scherrer Institut)

DAI, Yong (Paul Scherrer Institut)

Extensive positron annihilation experiments on spallation samples of F/M steels were performed at the Paul Scherrer Institute. Evolution of radiation defects and formation of helium bubbles was investigated in the temperature range of 100 – 500°C and up to the maximum displacement damage of 20 dpa. Combining TEM investigation results enabled us to evaluate specific trapping rate of positrons into He-Vacancy clusters. The results show that small (~1nm) He-bubbles with relatively high number density ($\geq 10^{24}$ m^{-3}) are important traps for positrons. They are trapped at the bubbles at a rate similar to the trapping rate at mono-vacancies (1.3×10^{-14} m^3s^{-1}). Helium-to-vacancy ratio has been found to vary between 0.2 - 1.0 and it seems to be inversely proportional to the sizes of the vacancy clusters. In low-dpa samples, the helium is predominantly confined in small vacancy clusters (He/V~1), while in high-dpa samples contain large helium bubbles in which vacancies prevail. The obtained results are discussed with respect to molecular dynamics simulations published in literature. Comparison of more than 70 measurements on various steel samples show saturation of positron trapping for samples irradiated above 300°C. It seems that there is only little (if any) connection between this saturation and displacement damage (dpa). The reasonable interpretation of this is an enhanced coalescence of helium bubbles, decrease of their number density and the related increase of positron mean free path (i.e. decrease of positron trapping rate). Our measurements also show, that presents of oxide dispersions can suppress the saturation of positron lifetime. While the saturation has been observed in the non-ODS Eurofer already at around 250°C, no saturation could be distinguished in the ODS Eurofer until 450°C. This suggests a better resilience against radiation damage and He production during irradiation in spallation targets.

Rate theory analysis of growth process of helium bubble in F82H irradiated at SINQ

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In spallation neutron sources and accelerator driven systems, irradiation damage is significantly more serious than that in fission reactors. The reduced activation ferritic/martensitic steel is a candidate for the structural materials of the facilities, which is more irradiation-resistant than the traditional nuclear reactor materials. One feature of spallation neutron source materials is the formation of a large amount of gas atoms by the nuclear spallation reaction, and the formation of helium bubble has great influence on mechanical properties of structural materials. We investigated the growth process of helium bubbles by positron annihilation lifetime (PAL) measurements. The positron mean and long lifetimes decrease at low doses (<13 dpa) and increase at high doses (>13 dpa) in F82H due to the results of PAL measurements. But transmission electron microscopy shows that the bubble size increases and the concentration decreases with increasing the irradiation dose and temperature. This phenomenon is attributed to the absorption of helium atoms by helium bubbles. In this study, the effect of helium on the growth of helium bubbles was examined by rate theory.

The growth of point defect clusters were calculated using the dynamic rate theory under the irradiation conditions for the second SINQ (Swiss Spallation Neutron Source) target irradiation program (STIP-II). Irradiation temperature was 360-572 K, and irradiation dose was 6.1-20.3 dpa. The following effects were ignored: (1) The effect of hydrogen on the evolution of defect structure, (2) the interaction of helium with interstitials and interstitial clusters, (3) the effect of the large primary knock-on energies, (4) the one dimensional motion of interstitial clusters.

The bubble size was almost constant and the concentration of bubbles increased with increasing the irradiation dose at 373 K. The bubble size and the concentration of bubbles increased with increasing the irradiation dose at 473 and 573 K. At 10 dpa, the bubble size increased and the concentration of bubbles decreased with increasing the irradiation temperature. In previous study, high pressure and low pressure helium bubbles were detected in one sample by PAL measurements. The change in the ratio of helium to vacancy will also be discussed.

Investigation of SINQ-irradiated samples by single detector Doppler-broadening spectroscopy

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A recently developed method for positron annihilation spectroscopy using internal 44Ti/44Sc source was applied for a measurement of Doppler broadening spectroscopy (DBS). This approach enables to perform fast and effective comparative characterization of steel samples irradiated in the spallation source SINQ. Despite a common standard in using 2-detector coincidence setup of DBS (CDBS), previously published work [1, 2] shows that transmutation helium, produced during spallation reaction, provides a characteristic peak in high-count part of spectra. This suggests, that coincidence mode, which effectively suppresses background and noise in the spectra, is not necessary in such investigation. First measurements using new 1-detector DBS and comparison to previous data obtained from CDBS confirm this hypothesis. Our DBS data can be acquired within few hours, instead of few days when using CDBS. Moreover they are free of kapton component (annihilation in the encapsulation of radioisotope source) since only internal positrons were used for the measurement.

Experimental investigation of irradiation effects in beryllium beam window after exposure in the NuMI beamline: preliminary results and plans

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It is now recognized that materials degradation by radiation damage is the one of the most challenging factors of design of the next generation of multi-megawatt high intensity proton accelerator facilities. The existing database on materials response to radiation effects, relevant to proton accelerator environments, is very limited and this complicates materials selection and lifetime predictions. An international research collaboration, Radiation Damage In Accelerator Target Environments (RaDIATE), was recently launched in order to explore radiation damage issues in different candidate materials under the relevant environment. Investigation of Be covers a large part of the RaDIATE collaboration, since it has been selected as a promising candidate for beam windows and target components in new generation of proton accelerator particle sources.

A beryllium beam window (grade PF60) from the "Neutrinos at the Main Injector" (NuMI) beamline has been irradiated by 120 GeV protons over 7 years of beam exposure at 70°C up to an integrated damage level of 0.5 dpa and now is available for analysis. Up to 2000 appm of He and 1800 appm of H have been generated during the irradiation.

The evolution of microstructure of the beam window during irradiation has been investigated at different scales by microscopy and through nanoscale investigation of the evolution of impurity distributions using Atom Probe Tomography. Descriptions of the experimental program and the results of the current activities will be presented.

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Plans for the RaDIATE high-energy proton materials irradiation experiment at the Brookhaven Linac Isotope Producer facility

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The RaDIATE collaboration (Radiation Damage In Accelerator Target Environments) was founded in 2012 to bring together the high-energy accelerator target and nuclear materials communities to address the challenging issue of radiation damage effects in beam-intercepting materials. Success of current and future high intensity accelerator target facilities requires a fundamental understanding of these effects including measurement of materials property data. Toward this goal, the RaDIATE collaboration has organized a materials irradiation run at the Brookhaven Linac Isotope Producer facility in early 2017. The experiment utilizes the 181 MeV proton beam to irradiate several capsules, each containing many material samples, configured for specific post-irradiation examinations. Materials expected to be tested include various grades/alloys of beryllium, graphite, silicon, iridium, titanium, TZM, and aluminum. Attainable peak damage from an 8-week irradiation run ranges from 0.03 DPA (beryllium) to 7 DPA (iridium). Helium production is expected to range from 5 appm/DPA (iridium) to 3,000 appm/DPA (beryllium). The motivation, specimen materials, irradiation parameters, experimental set-up and post-irradiation examination plans of this experiment will be described.

HiRadMat at CERN SPS - A dedicated test facility with high intensity beam pulses to material samples

FABICH, Adrian (CERN)

HiRadMat (High Irradiation to Materials) is a facility at CERN designed to provide high-intensity pulsed beams to an irradiation area where material samples as well as accelerator component assemblies can be tested. The high energy beam parameters (440 GeV protons with a pulse energy of up to 3.4 MJ and a pulse length of 7.2 us) can be tuned to match the specific needs of each experiment. HiRadMat is an area designed to perform single pulse experiments in order to evaluate the effect of high-intensity pulsed beams on materials used in accelerated-driven devices or high power targets in a dedicated environment. The facility is designed for a 10^{16} maximum number of protons per year to be distributed among the experiments. This paper will demonstrate the possibilities for research using this facility and show examples of experiments scheduled in the period 2012-2016. The facility, a unique place for performing state-of-the art beam-to-material experiments, operates under EuCARD2 transnational access and therefore welcomes and financially supports, under certain conditions, experimental teams to perform their experiments.

Application of rigorous two step methodology for neutron and proton transmutation calculations to spallation targets

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The rigorous two step (R2S) method can be used to calculate the decay radiation source term at some time after the end of irradiation. The first of the two steps is to calculate the multigroup energy spectrum as a function of position using a Monte Carlo code such as MCNP, this could be done using cell tallies, an overlaid mesh or a conformal mesh. In the second step the multigroup energy spectrum for each cell or mesh element is then used in a transmutation inventory solver such as FISPACT-II, ALARA, ACAB to calculate the activity and emitted gamma spectra at a given decay time for a given irradiation scenario. The inventory codes use cross section libraries for each interaction, can deal with complex irradiation histories and multistep reactions. The data for each cell or mesh element can be combined to make a 3 dimensional map of the induced activity in a facility. An optional additional step is to use the gamma spectra to generate a source term for use in a Monte Carlo code to calculate shutdown dose around irradiated items. Rigorous two step methodologies have been used in fusion applications for several years however they have not been extensively applied to high energy facilities such as spallation neutron facilities. Typically, in high energy facilities other codes such as Fluka and Cinder have been used which use physics models to calculate the decay. One of the main reasons for this is a lack of available cross section data compatible with the transmutation solvers and the need to include not only neutron activation but also proton activation. The recent TENDL 2016 cross section library includes data up to 1 GeV for protons and neutrons, which provides the possibility to utilise the rigorous two step method as an alternative to physics model based code.

This work describes the initial development of a prototype R2S code for spallation neutronics which uses MCNP for the first step and can use ALARA or FISPACT-II for the second step. The new R2S code uses and expands upon the routines in Pyne, an open source nuclear engineering toolkit. The prototype code is then applied to the ISIS target station 1 and target station 2. The results are compared to results from Fluka. In the future we hope to compare with post irradiation measurements of target components.

Displacement damage, helium and hydrogen production in different materials irradiated in STIP-VI

AULET, David (PSI)

BERGMANN, Ryan (PSI)

WOHLMUTHER, Michael (PSI)

DAI, Yong (Paul Scherrer Institut)

The SINQ Target Irradiation Program (STIP) has been developed at Paul Scherrer Institut (PSI) during the last twenty years. It is aimed to analyze the radiation damage induced by spallation in different structural materials which contributes to a major understanding on the irradiation effects. It is also providing data for developing future spallation sources as well as advanced spallation targets. STIP VI was conducted in SINQ Target 9 during the years 2011 (6,366 mAh) and 2012 (6,994 mAh) receiving a total proton charge of 13.16 Ah. The irradiation parameters such as proton and neutron fluxes, displacement damage (dpa), helium and hydrogen production have been calculated based on a computer simulation using the MCNPX code. The target mainly consists on an inverted $AlMg_3$ semi-sphere as the beam entrance window and the rod-container box which consists on 306 rods fitting 36 rows. Two different proton beam (575 MeV) profiles have been implemented for comparison. Beam 1 has an experimentally measured profile obtained from the gamma mapping of the irradiated $AlMg_3$ beam window. Beam 2 has a two-dimensional truncated Gaussian distribution fit from the experimental data. The proton and neutron fluxes for the experimental beam at the center of the calotte are 8.56x1025 p/m^2 and 4.36x1025 n/m^2. Regarding the target, the maximum proton flux is 8.32x1025 p/m2 at row 0 and the maximum neutron flux is 1.97x1026 n/m^2 at row six. It has been also calculated the radiation damage in the center of the calotte obtaining 8.76 dpa and 2410 appm He. For steels irradiated STIP-VI the maximum radiation damage is 32 dpa with 2,990 appm He and 12,600 appm H, which is the highest values ever reached in spallation targets. Besides steels, the irradiation parameters of Al, zircaloy. Ta, W and SiC were also calculated. It was found that the maximum damage is 72 dpa which was produced in zircaloy, while the maximum helium concentration is 3,540 appm produced in Ta. The detailed neutronic information of STIP-VI will be presented in this talk.

Simulation of hydrogen thermal desorption characteristics in metals containing large voids

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KOMAZAKI, Shin-ichi (Kagoshima University)

It is well known that a large number of hydrogen and helium atoms are formed in the structural materials of the spallation neutron sources. Hydrogen atoms are trapped by a variety of defects in steels such as vacancies, small vacancy clusters, dislocations, grain boundaries, precipitates, voids/bubbles. So, it is important to know about the interaction between hydrogen and such defects. Thermal desorption analysis (TDA) is often utilized to know trapping sites of hydrogen atoms. In the present study, the simulation technique for studying the effect of irradiation-induced defects on hydrogen desorption curves has been developed.

Simulations for pure Fe were performed. The simulation model of hydrogen desorption and adsorption in consideration of hydrogen coverage of sample surface was proposed by Davenport et al [1]. This model was expanded in this study. The dissolved hydrogen were calculated by a kinetics calculation in consideration of trapping and dissociation of hydrogen in the irradiation-induced defects. In the interaction between hydrogen atoms and voids, the effect of dissociation and adsorption, recombination of hydrogen at void surface was also taken into account. In addition, hydrogen diffusion inside of samples was also calculated according to the diffusion equation.

The effect of void surface on the desorption curves was examined. The peak height and temperature of simulational curves increased with the increase of the void radius and the number density. Large voids observed by scanning electron microscopy (SEM) are formed in the creep ruptured pure Fe (10 MPa/973 K). Although the void number density obtained by SEM was used, the simulational curve did not correspond with the experimental one. The experimental curve was influenced by the defects which cannot be observed by SEM. The effect of such defects (including irradiation-induced defects) was also discussed.

[1] J.W. Davenport and G.J. Dienes, Phys. Rev. B25 (1982) 2165.

Material selection of the beam profile monitoring devices at the ESS target station

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In order to monitor properties of the 5 MW proton beam directed onto the ESS target, a number of beam diagnostics will be inserted in a region upstream of the spallation target. The planned diagnostics at the ESS target station are the multi-wire profile monitor, the halo aperture monitor, and beam footprint-imaging system. The beam footprint imaging system consists of luminescent coatings on the target wheel and proton beam window, and optical components to transmit images to cameras located outside of the high radiation area. Since some of these devices intersect the high power proton beam, the beam induced dynamic thermomechanical loads and radiation damage in the monitoring materials limit the service lifetime of the beam diagnostics.

In this paper, we present the working principles and the materials selection of the beam diagnostics at ESS. The radiation damage and the thermomechanical stresses in the candidate beam-intersecting materials under the ESS beam condition are calculated. The simulated results are compared to dedicated experimental results and to known operational records of other major high power accelerator facilities, in order to estimate the material dependent service lifetimes. The impact of the beam-intersecting part on the proton beam transverse emittance and the measurement uncertainties caused by the fast recoil electrons from the beam upstream region are analyzed.

In this paper, we present the working principles and the materials selection of the beam diagnostics at ESS. The radiation damage and the thermomechanical stresses in the candidate beam-intersecting materials under the ESS beam condition are calculated. The simulated results are compared to dedicated experimental results and to known operational records of other major high power accelerator facilities, in order to estimate the material dependent service lifetimes. The impact of the beam-intersecting part on the proton beam transverse emittance and the measurement uncertainties caused by the fast recoil electrons from the beam upstream region are analyzed.

The present status and prospective of STIP

DAI, Yong (Paul Scherrer Institut)

The SINQ Target Irradiation Program (STIP) has undergone 20 years since it was initiated at the First International Workshop on Spallation Materials Technology in 1996 in Oak Ridge. During this 20-year period, seven irradiation experiments, STIP-I to STIP-VII, were conducted. In each experiment more than one thousand test specimens from about forty kinds of materials were included. The maximum irradiation dose of each experiment, except for that of STIP-I, is 20 dpa or more in steels. A large amount of results have been obtained from mechanical testing and microstructural investigation of STIP samples, which have shown a great impact on the R&D of high power spallation targets, and even on fusion community. However, on the other side, there are several unexpected incidents happened in STIP irradiations, e.g. in some cases the end-caps of the specimen rods fell off due to various reasons. Many specimens were lost because of such damages. In this presentation an overview of STIP will be given. The failures of the damaged specimen rods will be shown and discussed. The future plan of the STIP irradiation will be described.

Temperature measurement for in-situ crack monitoring under high-frequency cyclic loading

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XIONG, Zhihong (Sun Yat-sen University)

TSUTSUI, Kihei (Lancemore Co)

FUTAKAWA, Masatoshi (Japan Atomic Energy Agency)

At the Japan Spallation Neutron Source (JSNS), a mercury target made of 316L stainless steel suffers radiation damage under high proton and neutron radiation environment. Along with the radiation damage, the target vessel suffers cyclic impact stress caused by the proton beam-induced pressure waves. Actually, the JSNS target vessel suffers higher than 2×10^8 cyclic loading with a strain rate of approximately 50 s^{-1} at maximum during the service life of 2,500 h at 1 MW proton beam operation. In the previous study, we carried out an ultrasonic fatigue test to investigate the gigacycle fatigue strength of the structural material of target vessel, and found that specimen surface temperature rose abruptly just before resonance frequency changing due to occurrence of fatigue crack. For nuclear facilities such as power plants and neutron sources, it is necessary to develop a non-destructive in-situ monitoring system to detect fatigue crack in early stage. In this study, as a fundamental study, in order to clarify the mechanism of temperature rise due to occurrence of crack, specimen surface temperature during the ultrasonic fatigue test was measured using a 2D thermography. The experimental results showed that surface temperature rose locally only at the tip of fatigue crack and the position of the maximum temperature moved with the crack propagation. Therefore, we carried out numerical simulation to understand the mechanism of temperature rise caused by cyclic loading. The simulation results suggested that the heat due to plastic deformation at the crack tip is dominant for the temperature rise rather than the heat due to the friction between crack surface. The mechanism of temperature rise due to plastic deformation has the potential to detect microcracks at the very early stage. The measurement of temperature distribution is applicable technology as an in-situ monitoring related to structural integrity.

Characterization of W suppliers for ESS target

SORDO, Fernando (Consorcio ESS-Bilbao)

JAVIER, Gil (CEIT)

The European Spallation Source is an ambitious European project to build a new generation Spallation Neutron Source in Lund (Sweden). The total budget of the project is €1,800,000,000 shared between 17 European countries. The main characteristic of the source is its 5 MW long pulse proton beam which will impact on a tungsten rotating Target cooled by helium gas.

On November 2014, ESS-Bilbao was chosen as ESS partner for the Target wheel, shaft and drive unit. Along 2 years ESS-Bilbao has developed the engineering design to fulfill the technical specifications of ESS and introducing significant modifications compared with the ESS-TDR proposal.

The new target proposal is based on commercial hot rolled tungsten bricks (10x30x80 mm) that can be produced by several suppliers all over the world. In order to select the best partner for the production of the spallation material, ESS-Bilbao has received samples from six different suppliers and its mechanical characterization has been completed at CEIT-IK4 facilities.

The paper summarizes the results from the characterization process and the selected techniques.

Structural integrity investigation of JSNS mercury target vessel by nondestructive Inspection

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WAKAI, Eiichi (Japan Atomic Research Agency)

KOGAWA, Hiroyuki (Japan Atomic Energy Agency)

NAOE, Takashi (Japan Atomic Energy Agency)

HAGA, Katsuhiro (Japan Atomic Research Agency)

TAKADA, Hiroshi (Japan Atomic Research Agency)

FUTAKAWA, Masatoshi (Japan Atomic Research Agency)

Mercury target vessel for the Japan Spallation Neutron Source (JSNS) have been fabricated and installed at Materials & Life science experimental Facility (MLF) in Japan Proton Accelerator Research Complex (J-PARC). The mercury target vessel, which is composed of type 316L stainless steel, is designed as multi-walled structure composed of the mercury vessel and double-walled water shroud taking into account of the safety. The double-walled water shroud with cooling flow channels covers the mercury vessel to prevent the mercury leakage to outside when occurring of failure of the mercury vessel. The mercury vessel and water shroud are assembled by the tungsten inert gas (TIG) welding. In the assemble process, the water shroud is fixed on the mercury vessel by bolts and the area around bolt are welded to secure the airtightness. At moment when the pulsed proton beams injecting the mercury target, temperatures of the vessel and mercury increases rapidly because of the abrupt heat deposition, and the mercury target vessel suffers large thermal and dynamic cyclic stress caused by beam trip and pressure waves, respectively. Thus, it is concerned about the crack propagation originated from welding defect. In this study, the nondestructive inspection by the phased array ultrasonic system (GEKKO, M2M) with the total focusing method (TFM) was carried out on the welded 316L steel sample and JSNS mercury target vessel to evaluate the structural integrity of the target vessel. Furthermore, the radiographic tests were also carried out. The size and distribution of the weld defects were investigated by two kinds of inspection methods. The allowable proton beam power was estimated based on the size of weld defects in the mercury vessel. In this workshop, these results will be presented.

Experimental work for gas injection systems in ORNL SNS target

BARBIER, Charlotte (ORNL)

DOMINGUEZ-ONTIVEROS, Elvis (ORNL)

Target reliability at high power (>1.3MW) is a central issue at the Spallation Neutron Source (SNS) at Oak Ridge National Laboratory (ORNL). When the powerful, short pulse, 60Hz proton beam impacts the liquid mercury target, an intense pressure wave propagates into the mercury and the stainless steel vessel which lead to large numbers of stress cycles (>109) and cavitation damage to the target vessel. It has been shown that the presence of small, non-condensable gas bubbles distributed in the liquid metal could considerably mitigate the pressure wave, and potentially the cavitation erosion associated with it. Thus, a gas injection system will be implemented in the SNS mercury target at ORNL starting 2017. Because of the challenges associated with the presence of gas in the mercury loop, two gas injection systems are currently pursued. The first generation gas injection system consists of an inlet orifices bubbler (IOB) and will be limited to low gas flow rate injection (< 1 SLPM), whereas the second generation gas injection system, the swirl bubblers, will allow to reach up to 1% of volume fraction in the target (~15 SLPM). Experiments in water and mercury for both gas injection systems will be presented. Experimental methods used to determine bubble size distribution will be presented in details.

Status of SNS Second Target Station conceptual design

RENNICH, Mark (Oak Ridge National Laboratory)

Selection of key elements of the SNS Second Target Station during over the last 12 months has solidified the basic concept for the target station. Most importantly a solid rotating target was selected because of operating and safety advantages. This choice led to a series of related decisions with major consequences in the configuration of the monolith and building layout. Since the target is mounted vertically rather than on a horizontal cart a large hot cell is not required for operations. Elimination of the shielded cell resulted in the process systems being moved to the proton beam side of the monolith. The rotating target decision also resulted in a moderator/reflector plug configuration designed to shift on rails in order to be removed without disturbing the target. An Architect-Engineer study of the overall second target station site incorporated these changes into a building design which maintained earlier concepts for neutron beam line bunkers and multiple operating floors.

Performance of dual-phase ferritic-martensitic steel EP450 as cladding, wire wrap and duct of high burn-up fuel assemblies after irradiation to 108-163 dpa in BOR-60

GARNER, Frank (Texas A&M University)

Ferritic-martensitic steels are proposed to serve as structural components of various candidate advanced reactor concepts that will experience very high displacement doses. Two important requirements are low void swelling and an ability to allow very high fuel burn-up, without leading to failure by embrittlement. Previously there has been no demonstration that such goals can be reached, but a successful fuel pins series has now shown that such promise can be achieved.

Four vibropacked 37-pin fuel assemblies were irradiated in the BOR-60 fast reactor with cladding, wire wrap and ducts constructed from the EP-450 dual-phase ferritic-martensitic steel. Irradiation proceeded to high exposures with one assembly reaching 30% burn-up and the structural steel receiving more than 160 dpa. No pin failures occurred during irradiation in any of the four assemblies. The upper limit of both burn-up and steel dpa were therefore not determined in this experiment, even though the lower half of the assemblies experienced a reduction in DBTT.

The EP450 steel displayed a remarkable stability to irradiation under all conditions studied. The major microstructural components, tempered martensite and ferrite grains in the ratio of 1:1, were essentially unchanged by irradiation. Secondary phases formed within the tempered martensite and ferrite phases, varying primarily as a function of temperature, but there was no negative consequence on pin integrity. Very low levels of void swelling were found, not exceeding 1%. At the highest exposure the primary contribution to cladding deformation arose from mechanical stresses caused by gas pressure and especially fuel swelling. These results offer assurance that ferritic-martensitic steels may serve well in both neutron or accelerator-driven reactors.

Microstructure and mechanical properties of prototypic electron beam welds used to fabricate AISI 316L target vessels for the Spallation Neutron Source

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MCCLINTOCK, David (Oak Ridge National Laboratory)

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A total of 14 target modules have been installed and operated at the Spallation Neutron Source (SNS) since becoming an operational facility in 2006. The tenth operational target module developed a leak during operation at an electron beam (EB) weld, which was made by welding through the carbon-rich atmosphere of a surface hardening treatment called Kolsterising(R). Though the exact cause of the failure was not established, it was suspected that welding through the Kolsterising(R) treated layer might introduce carbon into the weld and degrade the joint strength. Therefore, a series of weldments were produced to simulate the actual target fabrication procedures, which included EB welding with and without Kolsterising(R) layer, double-pass EB welding, and TIG repairing welding. Tensile and microstructure samples were produced from the different representative weld conditions and characterized using electron probe microanalysis (EPMA) microscopy, electron back-scattered diffraction (EBSD) microscopy, and tensile testing accompanied with digital image correlation. Microstructure characterization revealed that the carbon atmosphere of the hardened Kolsterising(R) layer appears to dissolve into solid solution after welding. Some insignificant elemental segregation of alloy elements and ferrite formation were observed in the weldment and heat affect zone but were not considered detrimental to the weld strength. The tensile tests demonstrated high ductility of the welded material; local strength variations in the welds, in general, were in good agreement with the literature. No weld performance degradation due to welding through the carbon-rich Kolsterising(R) layer was observed for the weld configurations examined.

Hardness of a carburized surface layer on AISI 316L stainless steel from high-use Spallation Neutron Source target vessels after irradiation

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During operation the mercury-facing surfaces of the target vessel at the Spallation Neutron Source (SNS) are damaged by cavitation-induced erosion. A method implemented to increase the resistance of the target surfaces to cavitation-induced damage is a surface-hardening treatment called Kolsterising®. The treatment increases the carbon content at the surface to ~6 wt%, which gradually decreases to the base metal carbon concentration of ~0.2 wt% C at approximately 50 μm. At the time of deployment, the stability of the carbon atmosphere after irradiation was unknown. Specimens were sampled from high-use SNS target vessels after removal from service and characterized to evaluate the stability of the hardened surface layer, which were characterized using Vickers microhardness testing. Results show that the carbon-rich hardened layer was still present on all specimens after irradiation to 6.4 – 7.4 displacements per atom (dpa) and gradual decrease in the hardness profile was preserved. While some slight hardening was observed from irradiation, the hardened Kolsterising® layer appears to remain stable after irradiation in the mixed proton/neutron environment of the SNS target region and will continue to be implemented in future target designs.

Neutron poison burnout and effects on SNS moderator performance

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Two of the four SNS moderators providing thermal and cold neutron beams to the scattering instruments are decoupled from the surrounding reflector by cadmium layers, and poisoned by gadolinium poison plates in the center of the bulk water and liquid hydrogen vessel to produce narrow neutron pulses essential for high-resolution time-of-flight techniques in resolving the neutron energies of the beams incident to and scattered by samples. Due to the high neutron flux levels, the decoupler and poison materials burn out over time such that the moderators need to be replaced. Required poison and decoupler thicknesses were established in the design phase to guarantee a component life of 30,000 MWh. The design calculations assumed as the life-limiting criterion that the transmission of the poison plate not be higher than that of 50 µm fresh gadolinium. Using this criterion, the end-of-life of the moderators will be reached in year 2016.

As simultaneous burnup of poison and decoupler was not considered in the design analyses, and a clear impact on the moderator performance was not investigated, a more rigorous effort was launched to re-calculate the burn-out effects. The poison plate and decoupler were segmented to evaluate the local variation of burn-up with MCNPX-2.7.0 calculations. Isotope transmutation rates were directly calculated in MCNPX using the continuous energy cross sections and fed into a scripted burnup program that iteratively calculated the isotopic changes in the material zones and fed these back in subsequent MCNPX analyses by a predictor-corrector algorithm.

The local variation of burn-up is large; the poison plates burn out faster in the moderator center with peak burnout in zones closer to the target; the decoupler is degraded fastest at the sides facing the coupled moderator exhibiting peak burnup close to the target. The poison and decoupler burnup manifests itself in two effects: a gradual increase of the peak brightness reaching 10-20% at 40,000 MWh for the decoupled water and hydrogen moderators, and an increase of the long-time tail of the pulse shapes, which is more pronounced for wavelengths >2Å. Massive broadening of the pulse shapes is seen only above 40,000 MWh delivered beam energy. Because the moderator pulse width is only one factor of the instruments' energy resolution, the moderator life time can probably be extended from 30,000 MWh to 40,000 MWh with little impact on science. Instrument performance monitoring is clearly recommended approaching the end of life of the moderators.

Examining SNS first target station for the Proton Power Upgrade project

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As a prerequisite for the SNS Second Target Station (STS), the Proton Power Upgrade (PPU) project is aimed to double the current accelerator capability from 1.4 MW to 2.8 MW with a more energetic proton beam of 1.3 GeV. Among the enhanced capability 2.0 MW is to be delivered at the first target station, of which most components have design limits of 1.4 - 2.0 MW. Hence it is imperative to reexamine the physical strain of the first target station for engineering concerns or redesign purpose. In this presentation, the moderator performance, the energy deposition to the whole target monolith and the radiation damage to certain target station components are investigated. Some preliminary results show that by increasing the proton energy to 1.3 GeV the performance of the downstream moderators is moderately improved by ~8% while that of the upstream ones keeps the same. However, the more energetic proton beam shifts the energy deposition burden to the downstream part of the target station as well. If normalized to the same beam power, the energy deposition of the 1.3 GeV beam throughout the majority of the inner reflector plug (IRP) and outer reflector plug (ORP) is generally less than that of the 1.0 GeV beam. But at the region around the transition and back part of the target vessel, it sees a factor of 2-3 increase of the energy deposition load. A detailed study is undergoing to evaluate its engineering impact. The radiation damage study also finds the peak displacement production rate for the ORP occurs at a similar location with more than a factor of 2 increase, which results in ~15 dpa for the designed ORP lifetime.

Neutronics analyses for the Spallation Neutron Source Second Target Station

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Two of the word-class neutron scattering facilities, the Spallation Neutron Source (SNS) and the High Flux Isotope Reactor (HFIR), are located at the Oak Ridge National Laboratory. The SNS and HFIR are funded by the U.S. Department of Energy, Office of Science, Office of Basic Energy Science, and are operated as user facilities, available to researchers from all over the world. Currently there are thirteen neutron scattering instruments in operation at the HFIR and twenty at the SNS First Target Station.

The SNS was designed from the beginning to allow addition of the Second Target Station (STS), and an upgrade of the accelerator power. At this time both advances: the accelerator upgrade and the construction of the STS are in preparation.

Current baseline design for the STS calls for a rotating compact tungsten target. The target station is driven with short (<1 μs) proton pulses at 10 Hz repetition rate and 466 kW proton beam power, and is optimized for high intensity and high resolution long wavelength neutron applications. The possibility to accommodate the STS operation at 20 Hz and 932 kW is also being considered. The proton beam footprint as small as acceptable from the mechanical and heat removal aspects is planned to generate a compact-volume neutron production zone in the target, which is essential for tight coupling of the target and the moderators and for achieving high-intensity peak thermal and cold neutron fluxes. The STS will allow operation of approximately 22 beamlines and will expand and complement the current national neutron scattering capabilities.

This paper will present an update of the status of the neutronics analyses performed for the STS and will discuss the performance of the moderators, heating rates, radiation damage, activation analyses, and shielding calculations which provide input in the engineering design.

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