# 16th International Workshop on Spallation Materials Technology (IWSMT16)

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# Book of Abstracts

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**Poster session / 3**

## Structural and mechanical responses of (ZrTiNbTa)C4 and ZrC ceramics under energetic He-ions

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High entropy carbide ceramics (HECCs) are potential structural materials for advanced reactor concepts where numerous helium atoms would be produced due to the (n, α) nuclear transmutation reaction. However, the irradiation behaviors of high entropy ceramics are not well understood until now. In present work, the responses of high entropy (ZrTiNbTa)C4 and its binary constituent ZrC ceramics to energetic He-ions irradiation have been studied. X-ray diffraction analysis showed that He-ions irradiation resulted in peak shift and broadening of diffraction peaks in both (ZrTiNbTa)C4 and ZrC, which denotes lattice expansion and structural damages induced by the irradiation. Transmission electron microscope observation revealed that nano-sized spherical helium bubbles distribute uniformly in the grain interiors of (ZrTiNbTa)C4 while string-like bubbles with a preferred orientation are observed in ZrC grains. The accumulation and coarsening of He bubbles at grain boundaries are confirmed for both (ZrTiNbTa)C4 and ZrC, but no grain boundaries tearing was observed. Nanoindentation characterization gives that the irradiation induced hardening of (ZrTiNbTa)C4 is less definite than that of ZrC, which is attributed to irradiation induced bubbles and defect clusters act as obstacles during the deformation upon nanoindentation. Based on the experimental results, it can be concluded that the high entropy (ZrTiNbTa)C4 exhibits a less microstructural damage and reduced hardening than ZrC under identical irradiation conditions, suggesting the HECCs may possess a better irradiation resistance than the binary carbides.

**Facility overview, updates and developments. Operational experience of targets, beam windows, cooling and ancillary systems / 4**

## Operation status of CSNS Target Station

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The China Spallation Neutron Source (CSNS) marked its inaugural operational milestone by generating a neutron beam in August 2017. In August 2018, it passed national acceptance and officially started operation. The CSNS target station is distinguished by its stationary tungsten target and three distinct moderators: Decoupled and Poisoned Hydrogen Moderator (DPHM), Coupled Hydrogen Moderator (CHM), and Decoupled Water Moderator (DWM). The Target-Moderator-Reflector (TMR) are integrated into a compact, coupled configuration, which optimizes the neutron flux yield. At present, the power of the incident proton beam on the target has been increased to 160 kW, which notably exceeds the 100 kW goal set for the CSNS phase-I. The target station's subsystem, encompassing the target assembly, moderator and reflector, water-cooling loops, and cryogenic system, exhibits robust operational integrity and demonstrates the potential for further increase of power ahead of the CSNS phase-II implementation. Remote maintenance has been successfully executed, with three target plug replaced to date. Additionally, remote inspection and analyses of target plug failed seal rings have been performed. The fourth target plug is currently in service. The condition of the moderator-reflector Plug has been maintained satisfactorily, with no discernible deterioration in neutronic performance observed, even with the increased power output and extended operational periods. In pursuit of a comprehensive understanding of the target material's behavior and performance under conditions of intense proton irradiation and high-velocity water flow, the CSNS target station has initiated post-irradiation examination on the target plug. To facilitate these studies, specialized equipment for the precision cutting, sampling, and preparation of target material specimens

has been developed. Furthermore, a dedicated testing room has been established to conduct post-irradiation mechanical property evaluations. In parallel, preparations for the remote replacement of the proton beam window are underway.

**Poster session / 5**

## NUMERICAL MODELING OF HIDDEN DAMAGE ACCUMULATION DUE TO RADIATION EXPOSURE AND HYDROGEN EMBRITTLEMENT

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NUMERICAL MODELING OF HIDDEN DAMAGE ACCUMULATION DUE TO RADIATION EXPOSURE AND HYDROGEN EMBRITTLEMENT
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Hydrogen is generated by nuclear transmutation reactions in fusion and fission reactors, spallation neutron sources, and high-energy charged particle environments. Additionally, hydrogen is generated by environmental sources such as corrosion, radiolytic decomposition and recoil injection, particularly in systems involving water cooling and moderation, such as pressurized water reactors (PWRs) and boiling water reactors (BWRs). In Western-design PWRs and in some BWRs a hydrogen overpressure is also maintained on the coolant to suppress the buildup of corrosive chemical species. Notably, hydrogen has been shown to play a crucial role in the evolution of damage microstructure, which can in turn influence the mechanical properties and cracking behavior of structural materials.

This study examines the computational finite element modeling (FEM) of irreversible deformation and damage accumulation in the material of structural elements subjected to the combined influence of thermal, force, and irradiation fields.
A detailed description is provided of the calculation methodology employed for the determination of the stress-strain state and the long-term strength of austenitic steel. The results obtained with this methodology are also presented. The method is based on a complete mathematical formulation of the boundary-initial value problems of creep accompanied by irradiation effects. The following phenomena are considered: elastic, thermoelastic, plastic, thermal, and irradiation creep, irradiation swelling strains, damage due to thermal and irradiation creep, and hydrogen embrittlement. The constitutive equations for the numerical modeling were constructed using qualitative and quantitative data from experimental studies conducted at the micro, meso, and macro levels. The effect of hydrogen embrittlement is represented by the data from experiments performed at the macro level, which include changes in the modulus of elasticity. The constructed constitutive equations allow the description of the essential non-linear interaction of processes. The numerical solution of the boundary value problems is performed by FEM, and the initial value problems are solved by time integration. To estimate the effects of cyclic deformation and fracture, the procedures of asymptotic methods and averaging over cycle periods are employed.
The obtained results reveal that deformation and damage accumulation caused by irradiation effects in the material, including irradiation creep, irradiation swelling, and embrittlement, can significantly limit the safe operation of nuclear reactor vessel internals when interacting with thermal creep. The analysis suggests that the primary contributor to the accumulation of hidden damage is the synergistic interaction between irradiation swelling and hydrogen embrittlement. Furthermore, our findings indicate that on the surfaces of the elements, where the maximum values of the dose accumulation

rate are present, hydrogen embrittlement processes significantly intensify the damage accumulation rate. Such a behavior can be attributed to the growth of stress components caused by increased strains of irradiation swelling.

**Liquid/solid and particle beam interactions and associated studies: pressure waves, cavitation, erosion, corrosion etc and mitigation techniques / 6**

# Conceptual development of beamstrahlung photon absorber for FCCee

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The FCCee is a proposed 90-km electron-positron collider to be developed at CERN, intended to succeed the LHC.

During its operation, the FCCee will generate high-energy photon beams on both sides of the experimental insertions with an anticipated power of up to 500-600 kW in the 45.6 GeV, ~1.4 A operating phase. Photon energies will average between 2 to 63 MeV, with higher energy tails reaching several hundred MeV at the Z-pole and up to a few GeV at the ttbar threshold (182.5 GeV).

The significant power output presents challenges for photon absorption, particularly due to the absorber's geometric constraints and the specific characteristics of photon interaction. The design requires windows to separate the ultra-high vacuum (UHV) of the machine from the inert atmosphere of the absorber.

This contribution explores initial absorber concepts and system designs, ranging from gas-cooled graphite solutions to pure liquid lead sloped flow systems. Additionally, it outlines the research and development (R&D) and prototyping activities planned to address these challenges.

**Results from Post-Irradiation Examination of target and structural materials, innovative experimental techniques in study of irradiated materials / 7**

# Autopsy of the n_TOF spallation target at CERN

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The neutron time-of-flight (n_TOF) facility at the European Laboratory for Particle Physics (CERN) functions as a pulsed white-spectrum neutron spallation source. The facility's target, composed of pure lead, is impacted by a high-intensity 20 GeV/c pulsed proton beam. The facility enables the study of neutron-nucleus interactions across a wide spectrum of neutron kinetic energies, ranging from a few meV to several GeV. Such studies have significant applications in nuclear astrophysics, nuclear technology, and medical research.
Following the end-of-life of the second-generation target (target #2), the facility underwent substantial upgrades during CERN's Long Shutdown 2 (LS2, 2019–2020), culminating in the installation of a third-generation neutron target (target #3), operational since 2021.
A major enhancement in the third-generation target is the cooling technology, transitioning from water to gaseous nitrogen as the coolant medium to minimise erosion, corrosion, and contamination. Ensuring the structural integrity and performance of these targets over time is crucial for operational efficiency and safety. The creep behaviour of pure lead at high operational temperatures is particularly critical for long-term use. Additionally, the spent target must be packaged to be conform to the Host States requirements for the conditioning of radioactive waste.
To investigate the effect of long-term irradiation on Target #2 and to prepare it for its conditioning, a waste packaging and autopsy project for n_TOF Target #2 was initiated in 2022 and executed in 2024. Given the high radiation dose rate of the spent target after ten years of operation, the autopsy was conducted entirely remotely using a robotic system. This contribution details the methodologies, preparations, and techniques implemented, emphasising non-destructive autopsy methods such as visual inspections, ultrasonic testing, and 3D measurements. These non-destructive testing (NDT) techniques enable thorough internal inspection, identifying defects like cracks, voids, deformations, and material degradation while significantly mitigating the risk of contamination spread.
This contribution also outlines the waste packaging process and the subsequent disposal pathway.

**Facility overview, updates and developments. Operational experience of targets, beam windows, cooling and ancillary systems / 8**

## The upgrade plan and progress of China Spallation Neutron Source Target Station

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The target station converts high energy (1.6GeV,62.5μA) protons into lower-energy (< 1 eV), short-pulsed neutron beams optimized for the neutron scattering instruments. The interaction between the pulsed high-energy proton beam with the light water-cooling tungsten target produces the fast neutrons through the spallation reaction. Then the fast neutron will be moderated into short the cold, thermal and epithermal neutrons by the decoupled and poisoned hydrogen moderator, the coupled hydrogen moderator and the ambient decoupled water moderator. These moderators provide neutron pulses with different energy spectra to satisfy the requirements from various neutron instruments. The CSNS target station uses the flat tungsten target plates and the single layer container to reduce the distance between the tungsten target and the moderators. This compact mode makes the CSNS target station has very high neutron efficiency. When CSNS upgrades to 500KW during its Phase II, the target station also use the fixed solid target and keep the target-moderator compact mode in order to keep the high neutron efficiency of the target station. We will replace the light water to heavy water to cool the target and the beryllium reflector to increase the neutron intensity for the moderators. The key import thing is to ensure of the safety of the center components of our target station. We consider the different handling methods to cope with the different risk levels based on the thermal hydraulic simulation results. We also have develop the Post Irradiation Examination(PIE) since PIE is very important to minimize the risk.

**Spallation neutron/muon source component, systems & materials related technology and innovation / 9**

# Preparation of Zr-4 Alloy Coated W target plates by HIP Diffusion Welding

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Due to its high density, high melting point, and high thermal conductivity, W is used as a solid target material for scattering neutron sources. However, because of the poor corrosion resistance of W, it is necessary to coating a good corrosion-resistant layer to ensure the long-term use of the target material, Ta metal is a good choice and both have good applications in ISIS and CSNS.
CSNS has initiated the power upgrade project, and by the end of 2029, the power of CSNS will reach 500kW. Under high power conditions, decay heat removal of the target under abnormal beam stop conditions will be a challenge, reducing the decay heat of the target is one way. The calculation results show that the decay heat of zirconium alloy is only one tenth of that of Ta, using zirconium alloy replaced Ta as the target coating material can effectively reduce the decay heat of the target.
Zr- 4 alloy is widely used as a cladding material for nuclear reactors due to its low neutron absorption rate, excellent high-temperature performance, and corrosion resistance. In this study, Zr-4 alloy diffusion welding on the surface of W plate was carried out using hot isostatic pressure (HIP) method. The microscopic analysis of the interface shows that after HIP diffusion welding, a clear diffusion layer is formed at the interface of W and Zr-4 alloy, the selected area electron diffraction pattern analysis results show that the intermediate diffusion layer is ZrW2 phase and the nanoindentation hardness test results show that the hardness of the ZrW2 intermediate layer is significantly higher than that of W and Zr-4 alloy matrix. Meanwhile, as the HIP temperature increases, the thickness of the intermediate layer significantly increases from 0.2μm at 930 ℃ to 11.2 μm at 1400 ℃.

**Spallation neutron/muon source component, systems & materials related technology and innovation / 10**

# Alternative designs for the Beam Dump Facility (BDF) production target and in-beam test plans at CERN

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CERN's upcoming Beam Dump Facility (BDF) will host a new high-Z production target. The device will act as a beam dump to safely absorb 356kW of the 400 GeV/c proton beam delivered by the Super Proton Synchrotron (SPS). At the same time, it will generate a large number of high energy particle collisions in an attempt to produce hidden sector particles for the Search for Hidden Particles (SHiP) experiment.

Previously, the baseline target design of tantalum clad TZM and W blocs was tested in-beam via a scaled target prototype which underwent pre- and post-experiment material characterisation as described in contribution to IWSMT16 by Rui Franqueira Ximenes.

Since the baseline design was tested, several alternative designs have recently been considered which aim to improve physics and operational concerns, as well as waste packaging and disposal aspects. These include Niobium-alloy clad W, helium cooled W plates, and W/Cu peripherally cooled plates. Several of these designs will be tested in an existing slow-extraction test-bench in the North-Area

at CERN, aimed at baselining the final configuration by mid-2025 in view of the delivery of the Technical Design Report by end of 2025.

This presentation will describe the thermal and mechanical design of the alternative target technologies being considered, as well as the status and plans for the future in-beam tests.

**Facility overview, updates and developments. Operational experience of targets, beam windows, cooling and ancillary systems / 11**

## Proton beam window replacement in CSNS

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The Proton Beam Window (PBW) of the target station is one of the key equipment in China Spallation Neutron Source (CSNS). The PBW 1 has been running stably since2018. The PBW 2 has been completed and is planned to replace PBW 1during the CSNS summer maintenance in 2024.
This will be the first time to replace the PBW at CSNS. As an extremely important task, it poses great difficulty, tight timelines, and heavy responsibilities. The success of the PWB replacement is crucial for the next running, therefore, our team has conducted extensive preparation work before this replacement, including remote pipeline cutting, remote vacuum line connection, remote equipment lifting, as well as the hoisting, transport and storage of the PBW mockup. The simulation tests of new PBW installation was also done. The relevant equipment has been prepared to deal with possible unexpected situations. Moreover, the radioactive radiation, radioactive gas monitoring and the personnel protection in the maintenance process have been fully considered.
This report provides a detailed description of the PBW replacement process at CSNS.

**Results from Post-Irradiation Examination of target and structural materials, innovative experimental techniques in study of irradiated materials / 12**

## Outcome of the Post-Irradiation Examination of the baseline design prototype target and Nb-cladding R&D for the Beam Dump Facility (BDF) at CERN

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The Beam Dump Facility (BDF) and the SHiP experiment [1] have been selected to investigate hidden sector physics at CERN's ECN3 experimental cavern starting in 2030. To produce feebly interacting particles via a fixed target impacted by 400 GeV protons from CERN's Super Proton Synchrotron (SPS), the design of this target must safely absorb up to 356 kW of beam power [2]. With the initiation of the High-Intensity ECN3 (HI-ECN3) project's Technical Design phase, multiple concepts for the BDF high-power target are being explored, alongside essential material R&D. This new phase builds on the significant insights gained in recent years, particularly during the Comprehensive Design Phase, which included the development of a baseline design, construction and irradiation testing of

a prototype, and subsequent post-irradiation examination (PIE) along with R&D on niobium alloys for cladding.

The baseline design comprises a water-cooled eighteen molybdenum-based alloy TZM and pure W cylinders, cladded with Tantalum-2.5W alloy [3]. A prototype was constructed and irradiated at CERN in 2017 [4-5]. A post irradiation examination (PIE) has then been carried out to evaluate and characterize the resilience of the target core materials and Ta-based cladding [6].

Concerns regarding high decay heat from the cladding, particularly under a Loss-of-Coolant Accident (LOCA) scenario, prompted a feasibility study for niobium-based cladding materials. Pure Nb, Nb1Zr, and Nb10Hf1Ti were downselected, leading to the production of prototype cladded capsules via Hot Isostatic Pressing (HIP) and their subsequent thermo-mechanical characterization [7].

In this contribution, the HI-ECN3 project and Targetry developments are introduced, particularly a new He-cooled W concept for the Target. It details the outcomes and key conclusions from the post-irradiation examination of the baseline prototype target, as well as the findings from the R&D on niobium-based cladding.

[1] https://cds.cern.ch/record/2704147
[2] https://doi.org/10.23731/CYRM-2020-002
[3] https://doi.org/10.1103/PhysRevAccelBeams.22.113001
[4] https://doi.org/10.1103/PhysRevAccelBeams.22.123001
[5] https://doi.org/10.18429/JACoW-IPAC2021-WEPAB365
[6] https://doi.org/10.1002/mdp2.101
[7] https://doi.org/10.18429/JACoW-IPAC2023-THPM017

**Collaborations, opportunities and future plans e.g. for materials irradiations & PIE / 13**

## CSNS remote handling system and PIE plan

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The remote handling system is an important part of the target station and an important guarantee for the safe, stable and efficient operation of the China Spallation Neutron Source (CSNS) facility. Since CSNS formal operation began in 2018, the remote handling system has generally been operating well and a significant amount of remote handling operations have been successfully completed for target station maintenance objects such as targets, shutter inserts and ion exchange resins, etc. This report will first introduce the operation and maintenance experience summary of the remote handling system as well as the problems and challenges encountered. Secondly, it will also introduce the progress of CSNS-II Remote Handling System for neutron source target station upgraded to 500kW and new muon source target station. Finally, it will discuss the R&D on key technology/equipment for remote handling system and PIE plans for supporting the better development of higher power target station in the future.

**Facility overview, updates and developments. Operational experience of targets, beam windows, cooling and ancillary systems / 14**

## 40 years of ISIS targets

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The 16th December 2024, will mark 40 years since neutrons were first produced at the ISIS pulsed neutron and muon facility at STFC Rutherford Appleton Laboratory in Oxfordshire, UK. Since that date, the facility has operated several types of targets and added muon capabilities and the second target station (TS2). This talk will provide a brief history of the targets employed, touch on some of the challenges, design iterations and continuing development work, as well as looking to the future with the preliminary planning and conceptual design for the future facility, currently called ISIS-II.

ISIS'initial operation began with uranium target plates clad in zircaloy, however as the power level from the accelerator was increased, issues with plate swelling began to significantly impact operations and a decision was made to move to the all tantalum targets. At the start of the 21st century, the targets were again changed, this time for tungsten plates clad in tantalum and although the design has recently been updated with the TS1 project, this typed of target is still employed.

In early 1987, muons were first produced from an intermediate graphite target down the extracted proton beam (EPB) for the first target station (TS1). This facility has also grown and developed over the subsequent years and continues to provide a programme of complimentary science to the neutrons from TS1 and TS2.

Since 2009 a solid tungsten core clad in a tantalum sleeve has formed the heart of TS2, a lower repetition rate target station focussed on longer wavelength neutrons. The life times of the TS2 targets has not yet reached our initial design specification and has been the subject to much investigation and development work since first operation.

As ISIS heads towards its milestone later in the year, work is already underway to scope out and conceptually design a brand new neutron and muon facility for the second half of the century. Currently titled ISIS-II, this facility is planned to eventually replace the existing ISIS, while maintaining and enhancing the UK's neutron and muon provision, in a way complementary to the European Spallation Source (ESS), in order to continue to support the UK and global research communities. This facility is likely to feature next generation MW-class target stations.

**Results from Post-Irradiation Examination of target and structural materials, innovative experimental techniques in study of irradiated materials / 15**

# Multimodal characterization of samples from Spallation Neutron Source components exposed to extreme radiation and transmutation-inducing environments

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Simultaneous high-energy proton and neutron irradiation induce microstructural and mechanical responses in structural materials that are unique from irradiation with fission/fusion neutrons or accelerator-based ion beams. The target module and proton beam window (PBW) at the Spallation Neutron Source (SNS) at Oak Ridge National Laboratory (ORNL) are irradiated in these unique environments during operation. In addition to recoil damage up to about 10-15 displacements per atom (dpa) at moderately low component temperatures (100-120℃), transmutation reactions produce large levels of helium (up to ~190 appm/dpa) and hydrogen (up to ~740 appm/dpa). To maintain neutron production and extend component lifetime, accurate understanding of the radiation-induced changes to the microstructure and mechanical properties is essential, with particular focus here on the interactions with transmutation produced hydrogen and helium. Tensile testing with digital

image correlation has shown that the materials in these components have interesting and potentially unique behaviors not found in other radiation environments. Samples of 316L stainless steel target module material deformed via deformation waves with little martensite transformation, and material from a solution-annealed 718 PBW displayed a partial recovery in ductility with increasing radiation dose. To investigate these interesting phenomena further, specimens from these materials were characterized using a multitude of microstructural and microanalytical characterization techniques including scanning transmission electron microscopy (STEM) with energy dispersive x-ray spectroscopy (EDS) and electron energy loss spectroscopy (EELS), thermal desorption mass spectrometry (TDS), differential scanning calorimetry (DSC), and in-situ scanning electron microscopy (SEM) and TEM heating and tensile/compression testing.
In the as-irradiated condition, both the 316L and 718 materials exhibit a high density of nanometer-size cavities. These cavities were assumed to contain both He and H, similar to the cavities found in metals and alloys that have a high He/H concentration after triple-beam ion irradiation or neutron irradiation and transmutation. However, the results show that the cavities did not necessarily contain both gasses after the low temperature irradiation, as only H was observed by STEM-EELS. Post-irradiation annealing above 500 ℃ resulted in the colocation of H and He in the same cavities, though with a core-shell structure of H surrounding He, coinciding with the growth in cavity size and drop in cavity number density. To understand why this is the case and to examine the stability of the transmutation gas/defect complexes and their effect on mechanical properties, the materials were characterized using multiple different methods before and after post-irradiation annealing. TDS and DSC experiments were used to gain insights into the types of defects that are present and how the gasses are stored in the materials. DSC experiments have previously shown that a large amount of stored energy from irradiation is present in defects that are too small to be seen by TEM. Here, the radiation-induced stored energy –difference between energy released during first scan and subsequent annealed scans –were strongly correlated with the radiation dose and H and He release levels and recrystallization of amorphized phases. This correlation may help explain why there is an increase in radiation ductility with increasing dose in the PBW 718 material. In-situ SEM and TEM deformation testing will be presented to highlight the mechanisms for the ductility recovery.
With the DSC and TDS results in mind, post-irradiation annealing microstructural characterization and mechanical testing revealed that 1) the nanocavities retain relatively stability to high temperatures and are highly pressurized –enhancing dislocation emission during straining and thus work hardening and 2) annealing transforms the grain structure and dislocation structure enabling further recovery of ductility while maintaining strength greater than the reference condition. The microstructural process(es) responsible for the increased ductility with radiation dose could provide insights into deformation mechanisms that might be exploited in future materials.

**Collaborations, opportunities and future plans e.g. for materials irradiations & PIE / 16**

# Capacity building and PIE research on CSNS remote handling sample preparation and mechanical property testing

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The CSNS target operates in a composite environment of high-energy protons 1.6GeV and fast neutrons meV~GeV, and the evaluation of its operating status is crucial. In order to study the irradiation damage of the target, especially to provide more data for power upgrade, the CSNS target station PIE team has developed a small sample teleoperation preparation and mechanical properties testing system that can operate in the hot cell. This report will introduce the progress and research plans of CSNS in PIE research on the core components of the target station.

**Liquid/solid and particle beam interactions and associated studies: pressure waves, cavitation, erosion, corrosion etc and mitigation techniques / 17**

# Damage inspection of mercury target vessel operated at 1 MW in J-PARC neutron source

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At the Materials and Life Science Experimental Facility (MLF) in J-PARC, liquid mercury target for the pulsed spallation neutron source is in operation. A target vessel made of SS316L is damaged by pressure wave induced cavitation in mercury in addition to the 3 GeV proton and neutron irradiation. We are improving the target vessel to mitigate the cavitation erosion and gradually ramping-up the proton beam power for the goal of 1 MW 25 Hz, 1 MW stable operation in two month was achieved in May, 2024. To demonstrate the mitigation effect, beam window portion of the vessel was cut and inspect the depth of erosion. Furthermore, correlation between the erosion depth and beam induced acoustic vibration during operation are investigated to reflect the in-situ prediction of damage mitigation effect during operation. In the workshop, recent progress of mercury target in J-PARC will be introduced mainly result of damage inspection of 1MW-operated target.

**Results from Post-Irradiation Examination of target and structural materials, innovative experimental techniques in study of irradiated materials / 18**

# Irradiation Effect of SIMP Steel as Candidate Structural Materials for Accelerator Driven System

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The technological challenge presented by the conceptual future ADANES is the inability of current materials and components to withstand the harsh nuclear environment. Continue long-running program was performed to pursue development of low activation structural materials, with the greatest effort directed at the SIMP steels. Recent progress on the irradiation respondence of candidate structural material SIMP steel will be introduced in this study.
Microstructure evolution of SIMP and T91 steels under energetic Fe ions irradiation at the elevated temperature to the dose of 260 dpa has been investigated by using a TEM with the cross-sectional specimen technique. The swelling due to the formation of the cavities increased significantly and reach a peak value at the irradiation temperature of 440℃. It was noted that an important feature of SIMP steels microstructure under irradiation at high temperature regime was the appearance of cavities swelling replaced by irradiation induced second-phase particles. It is revealed that the appearance of the precipitates may play an important role in inhibiting swelling in the martensite lath. The relationship between radiation induced precipitation and cavity swelling in SIMP steels has been discussed in this wok.

**Application of new materials data and/or safety codes, computational modelling/analysis / 19**

## Investigation of some operational properties of the new-style ISIS TS-1 target

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During 2021–2022, a new style of multi-plate tantalum-clad tungsten spallation-neutron-producing target was installed in Target Station 1 of the ISIS Spallation Neutron and Muon Source. It is therefore of relevance for investigations to be made into the thermal and structural performance of the irradiated target. Some challenges of deducing target performance parameters from measurements that in practice have to be made remotely are described.

**Spallation neutron/muon source component, systems & materials related technology and innovation / 20**

## APT characterization of irradiation-induced evolution of nano-particles in ODS steels

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Oxide dispersed strengthened (ODS) ferritic/martensitic steels have been extensively studied in various fusion and fission materials R&D programs. Due to their high strength at high temperatures, ODS steels are tentative structural materials in ADS. The objective of this study is to determine irradiation-induced evolution of nano-sized clusters in ODS steels. The selected ODS steels are Eurofer-ODS developed by KIT Germany for fusion materials program and 9Cr and 14Cr-ODS steels developed by CEA France for fuel claddings. The samples were irradiated in SINQ Target-7 and -10 to doses up to 20 dpa.
Two kinds of particles were analyzed, including pre-existing oxide particles (ODSP) and irradiation-induced nano-particles (INP). The size, volume density and composition ODSP and INP have been quantitatively characterized using atom probe tomography (APT). The results showed that the INP are composed of Si, Mn, Ti, Sc, Ca and K, where Ti, Sc, Ca, and K are spallation transmutants produced during irradiation. The evolution of ODSP depends on irradiation temperatures, which may be attributed to different mechanisms.

**Poster session / 21**

## Characterization of Helium Bubble Formation in FeCr Alloys Using Positron Annihilation Lifetime Spectroscopy

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Positron annihilation lifetime spectroscopy (PALS) has been used to improve the understanding and prediction of early-stage helium-assisted radiation aging of structural materials. The present study

integrates a unique high-energy helium ion irradiation experiment with spallation target irradiation experiments, offering experimental PALS data that reveal the microstructure and a wide range of radiation-induced cluster sizes. Through these analyses, the study estimates the helium-to-vacancy ratio and enhances the existing theoretical model of positron trapping coefficients to account for cluster size and irradiation temperature. The findings confirm the model's effectiveness in characterizing helium-vacancy clusters and helium bubbles up to 1 nm in diameter, formed at irradiation temperatures ranging from room temperature to 300℃. This validation underscores the feasibility of using helium implantations for experimentally simulating spallation radiation environments.

**Liquid/solid and particle beam interactions and associated studies: pressure waves, cavitation, erosion, corrosion etc and mitigation techniques / 22**

## Comparison of simulated and observed cavitation-induced erosion damage in Spallation Neutron Source target vessels

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Cavitation-induced erosion patterns on the mercury-facing surfaces of Spallation Neutron Source (SNS) mercury target vessels were simulated using a newly developed parameter called the maximum bubble size. Explicit dynamic finite element simulations of the target vessels'structural response to beam pulses were performed. The cavitation bubble growth was calculated using a user defined subroutine based on the Rayleigh-Plesset equation with an initial bubble radius of 10 μm. The contour maps of the maximum bubble size value at the end of the simulation for each finite element on the mercury-facing surfaces were called the maximum bubble size maps. The predicated erosion patterns from the simulations were benchmarked with observations of erosion damage in the SNS target vessels during post-irradiation examination (PIE) and compared to calculations using a previous cavitation potential simulation technique called the saturation time. The gas injection technique has been deployed in the SNS target vessels to mitigate the cavitation-induce erosion damage and the maximum bubble size simulation was modified to account for the effect of gas injection. The patterns in the maximum bubble size maps for targets operated with and without gas injection agreed well with observations of erosion patterns on PIE samples from target vessels after service. The effect of gas injection rates on the damage potential simulation and an alternative parameter incorporating the bubble volume as a possible measure were also investigated.

**Results from Post-Irradiation Examination of target and structural materials, innovative experimental techniques in study of irradiated materials / 23**

## An overview of STIP results – lifetime assessment of components in spallation targets

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Irradiation experiments (STIP) in the targets of Swiss spallation neutron source (SINQ) at the Paul Scherrer Institute were started in 1998. More than nine thousand specimens of various Fe-, Ni-, Al-, Zr-, and W-alloys etc. were irradiated in a wide range of irradiation dose 2-30 dpa (in Fe) and temperature of 80-550 ℃, with 20-100 appm He/dpa and 200-400 ape H/dpa depending on the materials and the local conditions of the irradiation position in a SINQ target.

To date, numerous specimens of austenitic steels (SS316L(N), JPCA), ferritic-martensitic steels (T91, EM10, HT9, F82H, Eurofer etc,), Ni-alloys (Inconel 718 and Inconel 600), Al-alloys (AlMg3 and Al-6061), Ti-alloys (Ti6Al4V and Ti-Zr) have been tested. The materials cover a large part, if not all, of the structural materials used in the present spallation targets worldwide. In addition, the irradiation conditions in terms of dose and temperature range of the STIP specimens also cover those of the spallation targets. Therefore, the post-irradiation examination results of STIP specimens have been widely used, particularly for those spallation targets operating at 1 MW level.
In the presentation, an overview of STIP results will be given. The limiting factors for various materials under different conditions are discussed, which can be very useful to the lifetime evaluation of important components such as target containers or proton beam windows in spallation targets.

**Fundamental studies on the effects of radiation damage in materials. Innovative radiation damage resistant materials technology / 24**

## Hydrogen and tritium in various materials irradiated in STIP

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Among the PIE results of samples irradiated in STIP, the results of hydrogen and its isotopes release measurements are very interesting but less understood. Unlike helium, the actual hydrogen concentration of an irradiated specimen cannot be well predicted without measurement, as it strongly depends on the irradiation temperature and the irradiation environment of the specimens. In most cases, the hydrogen concentration measured by melting specimens is either very low (< 20%) or very high (few times) as compared to the calculated values. Comprehensive TDS analysis was performed on alloys SS 316, F82H, zircaloys-2 and AlMg3 as well as pure metals such as Al, Ti, Fe, Ni, Cu, Nb, Ta, Au and Pd in the temperature range of 40-1250 ℃ at Pacific Northwest National Laboratory (PNNL, USA). Gas species detected included atomic masses 2 to 6, including the hydrogen species H2, HD, HT, DT, D2 and T2, and the helium isotopes 3He and 4He. Some interesting trends were observed. For example, in the case of steel specimens, the main hydrogen release peak occurred a lower temperature and was narrower for lower dose (~10 dpa) specimens as compared to higher dose (~20 dpa) specimens. Significant deuterium and tritium releases were found for all specimens, ranging from ~17% to ~30%, and from ~2% to ~9% of the total hydrogen content, respectively. Recently tritium release experiments were performed at PSI and the results showed a good agreement with those obtained at PNNL. In this presentation, the experimental results achieved so far are summarized and compared with calculated values. The trapping mechanisms of hydrogen in various materials are briefly discussed.

**Liquid/solid and particle beam interactions and associated studies: pressure waves, cavitation, erosion, corrosion etc and mitigation techniques / 25**

## Embrittlement effect of liquid lead-bismuth eutectic on irradiated ferritic/martensitic steels

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Ferritic/martensitic (FM) steels have been selected as structural materials for applications in liquid metal spallation targets, such as the MEGAPIE (megawatt pilot experiment) target, and in the future accelerator driven systems (ADSs). Liquid lead–bismuth eutectic (LBE) is a candidate target and coolant material for such systems. Apart from the serious degradation of the mechanical properties

induced by intensive proton and/or neutron irradiation, liquid metal embrittlement (LME) is another factor of great concern for the components made of FM steels and used in a LBE environment.
In this work, LBE embrittlement effect on ferritic/martensitic (FM) steels T91, F82H, HT9 and EP823 after irradiation in a mixed spectrum of spallation neutrons and high energy protons in a target of Swiss Spallation Neutron Source has been studied. Slow-strain-rate tensile (SSRT) testing at a strain rate of 1 × 10-5 s-1 was conducted on T91 and F82H steels irradiated to doses up to 20 dpa in either Ar or liquid LBE at temperatures between 150 and 500 ℃. Tests in Ar showed significant irradiation-induced hardening and embrittlement effects (loss of ductility) as compared to the unirradiated ones. Tests in LBE revealed an additional embrittlement effect induced by LBE, which increased with irradiation dose. Consequently, the fracture strain of irradiated specimens was reduced from above 5% to 2-3%. 3-point bending (3PB) testing was performed on pre-cracked T91, HT9 and EP823 steel specimens irradiated to 6-14 dpa in either Ar or LBE at 150-250℃ and different strain rates. Similar to the SSRT test results, the 3PB test results show that the irradiated FM steels exhibited both irradiation-induced embrittlement effect and LBE induced embrittlement effect on fracture toughness of the steels. The LME effect is more pronounced at a lower strain rate. Scanning electron microscopy revealed an enhanced cleavage fracture mode for specimens tested in LBE.

**Spallation neutron/muon source component, systems & materials related technology and innovation / 26**

## Suitability of TZM as an ISIS Target Cladding Material

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The ISIS Facility uses a Tantalum clad Tungsten target in order to produce neutrons which facilitates unique and innovative research. The properties required for a material to qualify as a potential target material are strict, resulting in scarce options. Often these materials are very expensive due to their rarity and are also very difficult to work with from a machining and manufacture perspective. Due to this, any material that is deemed to potentially fit the criteria should be investigated. TZM, a Molybdenum alloy is one such material put forward by the Target Design Group.
To ascertain its credibility, an in-depth investigation was carried out, firstly, researching the properties of the material, focusing on those most relevant to the criteria of a target material. Secondly, testing its manufacturability, using the machining techniques available to the Target Manufacture Facility, to compare the optimum parameters found for machining TZM against the known parameters for Tantalum.
This talk will cover an overview of the research and the in-depth practical analysis carried out, with comparisons to our current cladding material, along with the results and conclusions drawn from said analysis.

**Application of new materials data and/or safety codes, computational modelling/analysis / 28**

## Radiation damage assessment of the seventh SINQ target irradiation program based on MCNPX simulation

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The SINQ Target Irradiation Program (STIP) is one of the crucial research projects performed by the Paul Scherrer Institute (PSI). The seventh experiment of the SINQ Target Irradiation Program (STIP-VII) was conducted in SINQ Target 10 during 2013 and 2014 and the total proton charge received by the target is 11.7 Ah. A radiation damage assessment was performed by using the Monte Carlo N-Particle eXtended (MCNPX) code to obtain the key irradiation parameters, i.e., displacement damage (dpa), helium (He) and hydrogen (H) concentrations.
After irradiation, the gamma mapping was performed at the entrance window of proton beam and provided the distribution of the proton fluence for the initial proton source definition. In the MCNPX simulation, the fluences of proton and neutron along the target as well as the energy deposition were calculated. Corresponding to the average current 1.45 mA of proton beam, the maximum proton and neutron fluxes obtained are 2.3e14 p/(cm2·s) and 5.6e14 n/(cm2·s), and the maximum energy deposition in the Zircaloy-2 cladding tubes is about 500 W/cm3. The irradiation parameters of six selected materials (9Cr-1Mo, Zry-2, Al, SiC, Ta and W) in each of 16 specimen rods were evaluated by multiplying the proton and neutron fluences with the corresponding cross section data. The maximum displacement damage was calculated for Zircaloy-2 in Row 2 with a value of 44.8 dpa (displacement per atom). The maximum value of helium concentration was calculated for tantalum in Rod 1 with a value of 2560 appm and the maximum hydrogen concentration was calculated for tungsten in Rod 1 with a value of 13050 appm. This assessment will be used as a reference in the subsequent post-irradiation examination of the STIP-VII specimens.

**Results from Post-Irradiation Examination of target and structural materials, innovative experimental techniques in study of irradiated materials / 29**

# Post-irradiation examinations of solution annealed and cold worked Inconel 718

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Ni-based alloys have been widely utilized as proton beam window (PBW) materials in existing spallation neutron sources and are potential materials for advanced generation IV reactors like molten salt reactors. However, the radiation-induced loss of ductility is still a concern, and the post-irradiation examination between different heat treatment methods is limited. This study aims to investigate the behavior of solution annealed (SA) and cold worked (CW) Inconel 718 after irradiation at the Swiss spallation neutron source (SINQ). The findings of this study will be useful for materials selection and lifetime prediction of PBWs and improve the understanding of radiation damage for Ni-based alloys in nuclear applications.
Inconel 718 alloys were solution annealed at 1065℃ for 0.5 h to form austenitic structure and part of that were cold worked. Specimens were irradiated in the STIP-II program at 7.6-19.5 dpa in a temperature range of 137-395 ℃. Post-irradiation tensile tests were performed at room temperature and irradiation temperatures. The microhardness of the irradiated sample was measured after tensile test. The fracture morphology and irradiation-induced defect structures were observed by scanning electron microscope and transmission electron microscope.
The tensile test results indicate that both SA and CW Inconel 718 alloys retained a significant amount of ductility after irradiation, except for a high dose (18 dpa) and irradiation temperature (395 ℃) condition, where the SA alloys show intergranular fracture surfaces while CW alloys show quasi-cleavage fracture surfaces. In the SA condition, the microhardness increased after irradiation, whereas the hardening was less pronounced at higher irradiation dose and temperature due to the formation of fewer frank loops. In the CW condition, the hardness decreased after irradiation, despite the existence of frank loops and He bubbles could strengthen the alloy. The reduction of high-density dislocations in initial microstructure of the CW condition was mainly cause of the irradiation-induced softening. Moreover, the microstructures in deformed areas were also observed.

**Fundamental studies on the effects of radiation damage in materials. Innovative radiation damage resistant materials technology / 30**

## High-Entropy Alloy R&D for Accelerator Beam Window Applications

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High-Entropy Alloys are a class of novel material that can offer improved resistance to beam-induced radiation damage and thermal shock. Development of these new alloys to serve as beam windows in multi-megawatt accelerator target applications is ongoing at Fermilab. Currently we are investigating AlCoCrMnTiV alloy systems of 4-6 components for service as beam windows; these compositions are predicted by CALPHAD simulation to have a low density and single-phase BCC crystal structure. The microstructures of these systems are being studied by electron microscopy techniques such as energy dispersive X-ray spectroscopy (EDS) to determine elemental homogeneity and composition, electron backscatter diffraction (EBSD) to quantify grain structure and orientation, and transmission electron microscopy (TEM) to observe defect structures and precipitate formation; nanoindentation is used to probe microstructural mechanical properties. Initial bulk property characterization utilizes differential scanning calorimetry to quantify specific heat capacity ($c_p$), dilatometry to determine the coefficient of thermal expansion (CTE), and time-domain thermoreflectance (TDTR) to measure thermal conductivity ($K$). A miniature tensile testing apparatus is also being developed to test tensile properties. Post-irradiation examination is currently ongoing for several of these compositions that have been irradiated by low-energy ions to damage levels and at a temperature relevant to beam window applications at future next-generation accelerator facilities. This talk will briefly describe the alloy design and synthesis before going into more depth covering microstructural pre-characterization, and post-irradiation examination results from low-energy ion irradiated specimens. This will be followed by the plans for alloy down selection and future prototypic proton irradiations.

**Spallation neutron/muon source component, systems & materials related technology and innovation / 31**

## Structural Optimization Design and Manufacturing Technology Development of Moderator Reflector System at CSNS-II

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As part of the CSNS-II Project, a new design of moderator and reflector plug will be installed at CSNS-II target station.
As the proton beam power will be upgraded from 100KW to 500KW, the structure and the manufacturing technology of the reflector will be optimized to accommodate the higher thermal deposition. The material 6061-T6, which has stronger resistance to stress corrosion, will be used as the material for the reflector container. In order to reduce the welding deformation, the electron beam welding process test of 6061-T6 aluminum alloy was carried out, and this new technology will be applied in the manufacture of the new reflector plug.
In addition, in order to ensure the stable operation of the moderator under a harsher working conditions, the manufacturing process of cadmium, a neutron decoupling material of the moderator, has been optimized. We solved the problem of uniformity of flame-sprayed cadmium coating,and we have also studied the preparation process of gadolinium coatings, but due to the low thermal conductivity of gadolinium, this decoupling material can only be used in areas with low thermal deposition.

**Fundamental studies on the effects of radiation damage in materials. Innovative radiation damage resistant materials technology / 33**

# Development of Carbon-Nanotube Reinforced Tungsten Alloy

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Tungsten is used as a proton-accelerator-target material. However, a significant amount of helium is produced through spallation process under high-energy proton irradiation, compared to nuclear fission and fusion materials. The produced helium forms bubbles at grain boundaries with increasing temperature, leading to fatal embrittlement of the material. Recently, it has been reported that adding carbon nanotubes to aluminum alloys suppresses helium bubble formation and enhances mechanical properties. Inspired by these methods, we have started developing tungsten alloys with carbon nanotubes via powder metallurgy. We conducted single helium ion irradiation on the first prototype materials. In this presentation, we will report a status of the manufacturing and the analysis of these materials.

**Results from Post-Irradiation Examination of target and structural materials, innovative experimental techniques in study of irradiated materials / 34**

# Irradiation response of SIMP steel in STIP

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Ferrite/martensite (F/M) steels containing 9-12% chromium are considered as one of the most competitive candidate materials for advanced nuclear energy systems due to its excellent properties such as resistance to irradiation swelling, high thermal conductivity and low thermal expansion coefficient. SIMP steel is a novel F/M steel with about 10.24%Cr and 1.2%Si specially developed for the CiADS (China Initiative Accelerator Driven System) application. Preliminary test results revealed showed good corrosion resistance to lead-bismuth eutectic (LBE) at 550℃. To investigate it irradiation response, specimens of two batches of SIMP steel were irradiated in Target 10 and 13 of the Swiss spallation neutron source (SINQ), namely STIP-VII and -VIII.

Tensile specimens of SIMP steel were irradiated in STIP-VIII at temperatures between about 130 and 270 ℃ to doses between 4 and 10.4 dpa. Tensile tests were performed at both room temperature (RT) and irradiation temperatures. The fracture surfaces of selected tensile tested specimens were characterized by scanning electron microscopy (SEM). Moreover, the microhardness was measured from the grip section of tensile specimens.

Irradiation-induced hardening was shown by the tensile and microhardness results. At RT, total elongation of irradiated specimens reduced from 18% of unirradiated specimens to about 6%. At

higher temperatures between 150 and 300 ℃, the total elongation continuously decreases with the test temperature to about 12% and 4% for unirradiated and irradiated samples, respectively. The fracture morphology of the samples showed that the fracture mode changed from ductile to quasi-cleavage fracture with increasing dose.
The dose of the specimens irradiated STIP-VII reached 23 dpa. They will be tested also at higher temperatures. The results will be included in this presentation.

**Collaborations, opportunities and future plans e.g. for materials irradiations & PIE / 35**

## Conceptual Study of Post Irradiation Examination (PIE) Facility at J-PARC

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JAEA has been developing accelerator-driven systems (ADS) for nuclear transmutation to reduce the volume and hazardousness of high-level radioactive waste generated by nuclear power plants. In order to prepare the material irradiation database necessary for the design of ADS and to study the irradiation effects of candidate structural materials for ADS in liquid lead-bismuth eutectic (LBE) alloys, a proton irradiation facility is under consideration at J-PARC. In this proton irradiation facility, a 400 MeV and 250 kW proton beam will be injected into the LBE spallation target, and irradiation tests under LBE flow will be performed for the candidate structural materials. Furthermore, semiconductor soft-error tests, medical RI production, and proton beam applications will be performed. Among these, post irradiation examination (PIE) of irradiated samples and separation and purification of medical RI, for instance Ac-225, will be carried out in the PIE facility to be constructed near the proton irradiation facility. In this PIE facility, PIE of the equipment and samples irradiated in the other facilities in J-PARC will also be performed. In this presentation, first, the conceptual study of the PIE facility, including the items to be tested and the test flow will be described. And then, the specifications and quantities of the facilities and the test equipment required to perform these test items will be shown. Finally, the layout of the PIE facility will be proposed.

**Fundamental studies on the effects of radiation damage in materials. Innovative radiation damage resistant materials technology / 36**

## Applicability of Ferritic/Martensitic Steels for Spallation Applications: Comparison of Performance in Fission, Fusion to Spallation Systems

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Ferritic/Martensitic (F/M) steels have become the lead material for a number of applications in extreme neutron and ion irradiation environments due to their robust resistance to irradiation effects

and, in some cases, low activation. In the range from ~400 to 650℃, these materials exhibit low or no swelling in fission irradiation environments while maintaining excellent mechanical properties. For applications in more fusion neutron or spallation proton environments, the production or implantation of helium and hydrogen becomes a significant additional challenge.
We have conducted a large study on fission neutron irradiation effects on large numbers of Fe-Cr-base model alloys for their dimensional stability and resilient mechanical properties performance. This work is based on microstructural evolution during irradiation over a range of temperatures from 300℃ to nearly 600℃ to doses up to 10 dpa. The work also includes mechanical properties measurements of these irradiated materials where changes in properties can be associated with irradiation-induced microstructural changes. Unique to our program is the use of a synchrotron x-ray source to monitor in sit deformation processes.
The results of our work can be related to other work on F/M alloys from the spallation irradiation structures and mechanical properties which have been exposed to similar doses over the same range of temperatures. One relevant finding from our work is that the early evolution of defect structures and low dpa levels produces significant hardening or strengthening even at temperatures above 450℃. This behavior is tied directly to the dislocation loop structures and their evolution with dose. At higher doses, the growth of these loop structures diminishes their effect on mechanical properties, but the evolution of irradiation-induced precipitate structures, particularly the G-phase and alpha-prime at higher Cr levels, produce other hardening and strengthening effects. This talk will compare the irradiation performance of our alloy systems to the results for available spallation systems studies.

**Spallation neutron/muon source component, systems & materials related technology and innovation / 37**

# Status of the STS Target Segment Manufacturing R&D

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The target assembly for the Second Target Station (STS) at Oak Ridge National Laboratory is working towards final design approval in late 2025, and R&D continues to focus on the manufacturing of the target segment. The novel design of the target segment and unique combination of materials (tungsten, tantalum, copper, and Inconel) has driven the process development towards solid state bonding techniques (vacuum hot pressing (VHP) and hot isostatic pressing (HIP)). The mismatch in thermal expansion between tungsten and Inconel, combined with the wedge-shaped geometry, create challenges for material interface bonding; however, the resulting "shrink fit"of the Inconel shroud around the tungsten/tantalum spallation package creates a beneficial state of compressive residual stress in the tungsten. The corresponding tensile residual stress state in the Inconel must therefore be accounted for in the fatigue lifetime estimate of the shroud. The HIP process is being developed through a series of small-scale test articles that mimic geometrical features of the target segment, and a smaller number of full-scale test articles to exercise the HIP process on prototypical geometry. VHP coupon samples are used to characterize material interface shear strength and thermal diffusivity as a function of bonding pressure. Corresponding simulations of the HIP process are leveraged to predict the bonding pressure, so that the expected bond quality can inform performance models. Furthermore, a verification is required that the residual stress prediction in the Inconel is conservative, so diffraction-based techniques using X-rays and neutrons have been used to compare with the simulation predictions. The process development status and current results are presented along with the remaining R&D tasks supporting the final design completion.

**Liquid/solid and particle beam interactions and associated studies: pressure waves, cavitation, erosion, corrosion etc and mitigation techniques / 38**

# Experimental Activities in SCK CEN on Corrosion of Structural

## Materials in Static and Dynamic Liquid Pb-Bi Eutectic - Overview

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SCK CEN conducts an extensive experimental program within the MYRRHA and SFR-SMR R&D programs to characterize candidate structural materials for their application in heavy liquid metals. Main challenges lie here in the qualification of candidate materials for a variety of environmental parameters of the liquid metal system, i.e.: temperature; flow velocity; liquid metal chemistry with respect to oxygen; alloying and structural state of structural materials; etc. The main aim of the tests is to produce the reliable quantitative data and characterize the candidate steels under conditions representative for HLM reactors, i.e.: 200-400 ℃, Pb-Bi flow velocity of 2 m/s and concentration of oxygen dissolved in the liquid metal around 2×10-7 mass%.
In this presentation we will summarize results from different experimental campaigns carried out in stagnant corrosion installations and in flowing liquid metal using CRAFT loop. Unique results of heavy liquid metal corrosion after very long exposure times (8000+ hours) of a ferritic/martensitic steel (T91) will be discussed. The results show that continuous addition and tight controlling of dissolved oxygen in the liquid metal is a key to successful corrosion mitigation. During the performed long-term exposure under well controlled conditions regarding temperature, flow velocity (2 m/s) and oxygen content in the liquid metal the steel underwent oxidation which resulted in formation of Fe-Cr based oxide films. Localized dissolution corrosion, as a result of local degradation of initially formed oxide film, are characterized as-well. The main results are compared with tests performed under the similar conditions in CORRIDA loop (KIT). Some results on Alumina-Forming Austenitic steels (AFA) and their application as potential structural materials in heavy-liquid metals (Pb, Pb-Bi) will also be briefly discussed.

**Liquid/solid and particle beam interactions and associated studies: pressure waves, cavitation, erosion, corrosion etc and mitigation techniques / 39**

## Study on Electromigration Effect in High-temperature Liquid Metal Pool

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Liquid lead (Pb) is candidate coolant of accelerator driven systems, fast reactors and fusion reactors. However, the chemical compatibility of liquid Pb with structural materials is one of the important issues. The flow of electrons and ions in the liquid metal components may result in electrical current flow. The electrical current flow in a liquid metal pool can promote a mass transport phenomenon through electromigration. The electromigration can accelerate the corrosion of structural materials. The purpose of this study is to clarify the effect of electromigration on the material compatibility in liquid Pb pool. Two specimens of 304 stainless steel were installed on the electron receiving side and the electron influx side of the crucible filled with liquid Pb. The electromigration test was conducted under the electrical current flow of 10 A at the temperature of 773 K. The results of SEM/EDX analysis performed on the tested specimens indicated that two different mass transport phenomena took place in each specimen. Dimple-like patterns and cavities were formed on the surface of the electron-receiving side, and they indicated that dissolution corrosion was promoted. The metallic impurities were precipitated on the specimen surface on the electron influx side. The precipitation was identified as CrNiAs crystals through XRD and EDX analysis. The collision of the Pb atoms on the specimen surface of the electron-receiving side promoted the dissolution corrosion according

to the electromigration. The dissolved elements and impurities escaped from the electron-receiving side onto the electron influx side according to the electromigration of Pb atoms. The dissolved elements and impurities precipitated on the specimen surface of the electron influx side according to the oversaturation condition which was induced by the electromigration.

**Liquid/solid and particle beam interactions and associated studies: pressure waves, cavitation, erosion, corrosion etc and mitigation techniques / 40**

## Excellent corrosion resistance of self-healing Al-rich oxide layer on FeCrAl alloy in flowing lead bismuth eutectic

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Liquid lead bismuth eutectic (LBE) is a candidate target material for accelerator driven systems. The chemical compatibility of liquid LBE with structural materials is one of the important issues to be addressed. FeCrAl alloys are candidate structural materials since they can form Al-rich oxide layers on their surface which can function as an anti-corrosion barrier in liquid metals. The purpose of the present study is to clarify the corrosion resistance of FeCrAl alloy APMT (Fe-21Cr-5Al-3Mo) in flowing LBE. The corrosion test facility of OLLOCHI (Oxygen-controlled LBE LOop for Corrosion tests in HIgh-temperature) was operated for the current corrosion study, which was installed at JAEA, J-PARC. The corrosion tests were performed with the APMT rectangular specimens having a size of 14 mm x 10mm x 1mm. They were subjected to the pre-oxidation treatment in an air atmosphere at 1373 K for 10 hours before exposure to flowing LBE. The specimens were installed in the test holder and exposed to flowing LBE at 723 K for 2000 hours. The flow velocity of liquid LBE around the specimens was 1 m/s in average. The specimens were taken out from the specimen holder after the test. The surface of some specimens was artificially damaged. They were then continuously exposed to liquid LBE to clarify the in-situ self-healing behavior of the protective oxide layer. The specimens tested in the flowing LBE were cleaned with a mixed solution of acetic acid, ethanol and 30% H2O2 solution (1:1:1). The corrosion of the specimens was metallurgically analyzed with SEM/EDX, XRD and STEM. The test results indicated that the APMT specimens with the pre-oxidation treatment revealed excellent corrosion resistance due to the presence of the protective oxide layer. The in-situ self-healing of the Al-rich oxide layer which was artificially damaged was also indicated.

**Application of new materials data and/or safety codes, computational modelling/analysis / 41**

## Operating Experience of ISIS Targets

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As part of the ISIS TS1 Project, a new design of spallation target was installed on the first target station (TS1), which has now been operating for almost a year. Detailed Finite Element Analysis (FEA) simulations were carried out, and compared to measured operating data. FEA has also been used to investigate unexpected observations on some target plates. Attempts were made to measure

radiation-induced changes in the thermal conductivity of tungsten in-situ on a working spallation target.
ISIS TS2 targets continue to be replaced ~1.5 years into their nominally 5-year design lifetime. The achievable life is limited by increasing activation of the cooling water, thought to be due to tungsten in direct contact with water. Recent irradiated property data combined with detailed FEA now predicts target lifetimes which are consistent with the observed failures. This raises the possibility that such failures could be predicted and avoided in future.

**Application of new materials data and/or safety codes, computational modelling/analysis / 42**

## ISIS-II Target Concept Development

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Conceptual design studies are now underway for ISIS-II, the successor to the UK's pulsed neutron and muon source. Appropriate target technologies must be selected for each of the two proposed neutron target stations, to achieve a balance between neutronic performance and engineering reliability.
This talk will present the status of preliminary designs for both stationary and rotating target concepts for ISIS-II. Safe operating limits must be defined for direct beam on target, as well as residual decay heat in a Loss of Coolant Accident (LOCA) scenario. Irradiated material properties are a key design driver, as are oxidation properties of tantalum and tungsten. Alternative choices for core and cladding materials will also be discussed.

**Results from Post-Irradiation Examination of target and structural materials, innovative experimental techniques in study of irradiated materials / 43**

## Post irradiation examination and lifetime limits of highly irradiated components at the Spallation Neutron Source and High Flux Isotope Reactor

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The microstructure and mechanical properties of components in high-dose environments are altered during operation, which typically result in a loss of ductility and fracture toughness. The decrease in ductility limits the useful component lifetime due to concerns of fracture during operation. The useful lifetimes of components are established by reviewing previous results from characterizations of a particular alloy and selecting dose values that limit the risk of failure; however, test results for materials and irradiation environments under consideration are not always available. The most desirable information for establishing lifetime limits are results from testing material samples from actual components after operation. Post irradiation examination (PIE) of components after removal from service provides the most accurate information on radiation-induced changes in material properties. A PIE program is maintained at the Spallation Neutron Source (SNS) for targets and proton beam windows to ensure the administrative lifetime limits are optimized to provide reliable operation. Material samples form targets and proton beam windows are routinely obtained and characterized to measure the radiation-induced changes to mechanical properties. Another PIE project at Oak Ridge National Laboratory (ORNL) is focused on sampling a section of a 6061-T6 aluminum beam tube that was installed and operated in the High Flux Isotope Reactor (HFIR) for 25 years. During this

presentation the various current PIE projects at ORNL will be described, followed by a discussion on how the results are used to establish administrative dose limits and limit the risk of component failure.

**Application of new materials data and/or safety codes, computational modelling/analysis / 44**

## Feasibility Study of Diffusion Bonding of Zirconium to Tungsten Using Vanadium Interlayer and Its Implication on Decay Heat Driven Power Limit of a Water-Cooled Tungsten Target

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In developing a high-power water-cooled tungsten target for spallation neutron production, one of the important factors that limits the beam power on the target is decay heat driven temperature rise in tungsten in loss of coolant accidents. When tungsten is exposed to water vapor, tungsten oxides formed in steam is known to become volatile at above 800 oC causing radiological hazards with a risk of significantly elevated inhalation dose. A significant fraction of decay heat in the spallation volume is deposited in the cladding volume that protects tungsten from aggressive corrosion and erosion by coolant water in radiation environments. Tantalum has proven operational records as tungsten cladding material at world's leading spallation sources, the TS1 and TS2 targets at ISIS, the LANSCE target at LANL and the CSNS target. However, its high specific decay heat deposition per volume in spallation environment is a limiting factor in further increasing the beam power on the target. To study the feasibility of replacing tantalum with a low decay heat cladding material, we studied the feasibility of diffusion bonding of zirconium to tungsten. Zirconium alloys have extensive operational records in hadron radiation environments, and post irradiation material properties data are available in nuclear and spallation communities. It was used as cladding material for early generation uranium spallation targets at KENS, IPNS and ISIS and has been used as canning material for the MW class lead target at PSI. For this study, a vanadium interlayer was used between W and Zr to avoid the known formation of a brittle ZrW2 intermetallic layer along the W-Zr bond line during high pressure and high temperature diffusion bonding processes. Single step vacuum hot pressing was performed with W, V and Zr plates. The W-V and V-Zr bonding layer did not show a trace of ZrV2 intermetallic subdomain formation, showing the feasibility of using zircalloy alloys for tungsten cladding material. To demonstrate the full feasibility of three-dimensional diffusion bonding, a single step hot isostatic pressing of V and Zr plates on a tungsten brick was performed. Particle transport and thermal simulations showed that a zirconium clad spallation target has a higher decay heat driven beam power limit of 800~kW compared to the 400~kW power limit calculated for a tantalum clad tungsten target.

**Spallation neutron/muon source component, systems & materials related technology and innovation / 45**

## Applying a Pressure System Safety Code to Spallation Target Design

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The Second Target Station (STS) project at Oak Ridge National Laboratory (ORNL) has entered preliminary design review activities with an innovative edge-cooled spallation target. The tungsten/tantalum laminate spallation block is clad in copper and contained inside a thick Inconel shroud. The shroud provides structural support. It also provides thermal management by flowing water through cooling channels. This structure is subsequently bonded to a stainless steel support.
This talk will present challenges and successes in applying a pressure system safety code to the STS target design. Specifically, Section VIII Division 2 Part 5 (Design by Analysis) of the ASME Boiler & Pressure Vessel Code (BPVC) is used to analyze the Inconel shroud and stainless steel support against static and cyclic failure modes.
For static loading, an elastic-plastic analysis is appealing, however, the special geometry and loading of the spallation target presents challenges with the factored load methodology. For cyclic loading, Inconel 718 in the solution annealed form is considered a special material by the Code, and the Hot Isostatic Pressure (HIP) manufacturing operations induce significant residual stresses in the shroud. The plastic deformation and subsequent residual stresses from manufacturing create challenges with determining elastic-plastic material curves, fatigue curves with mean stress effects, and stress margins. Modifications and comparisons will be presented for fatigue models and plastic ratcheting assessment.
As will be shown, residual stresses from manufacturing drive the design performance. The challenges of simulating HIP, machining, and welding steps will be presented. These enhanced simulation tools are then used to further inform design choices for manufacturing.

**Facility overview, updates and developments. Operational experience of targets, beam windows, cooling and ancillary systems / 46**

## Status of the Second Target Station Project with Target Systems Highlight

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The Second Target Station (STS) project is an expansion to the existing Spallation Neutron Source (SNS) in Oak Ridge, TN. The project is at the end of the preliminary design phase. STS seeks to provide world leading peak cold neutron brightness to a suite of new instruments, eight of which are included in the project scope. SNS is currently in the final stages of the Proton Power Upgrade Project (PPU), which will leave the linear accelerator and accumulator ring capable of providing 2.8 MW delivered in micro-second long pulses at 60 hz. The First Target Station (FTS) will accept a maximum of 2 MW, leaving power available for STS. Every 4th pulse from the accumulator ring will be directed to STS via a new beam transport line. Thus, STS will receive 700 kW at 15 hz and the FTS will operate at 2 MW with an irregular 45 pulses per second. STS will achieve world leading peak brightness using the following ingredients: 1) compact, short, high net energy (46.7 KJ) pulses, 2) a compact rotating tungsten target with a low coolant volume fraction, and 3) closely coupled compact hydrogen moderators surrounded by water pre-moderator and beryllium reflector.
This presentation will provide an overall technical description of the project and its status. Current design and major evolutions across all systems will be presented with a focus on Target Systems. Additionally, an index of spallation materials related challenges and achievements will be provided.

**Spallation neutron/muon source component, systems & materials related technology and innovation / 47**

## Status of the STS Target Assembly Design

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The Second Target Station will be a world-leading neutron facility with cold neutron brightness an order of magnitude better than the Oak Ridge National Laboratory (ORNL) Spallation Neutron Source First Target Station. This facility aims to produce world-class brightness neutrons to advance fundamental science.

A rotating 1.2m diameter target disk of 58mm thick Tungsten has been selected for the Second Target Station. The target disk has been divided into twenty segments with separate cooling water passages. Individual target segments can be replaced without removing significant amounts of shielding. The segmented approach was selected in 2021 and has progressed into detailed design.

This talk will provide a general update on the target design, emphasizing the tungsten target, drive mechanism, and shaft arrangement.

The tungsten target design for the STS involves a solution-annealed Inconel 718 outer casing, a compliant copper intermediate layer, and a laminated stack of tungsten and tantalum. The Inconel layer provides structural support and passages for water cooling the target. Details will be presented on the general design approach, fabrication methods, fluid analysis, and remaining design challenges.

The target drive mechanism is a compact drive arrangement that rotates the target system, directs water to individual target segments, and provides for fast target removal in unexpected failure. The design centers around a compact arrangement employing an open-frame motor, high-capacity bearings, and a dry-running gas seal. The presentation will describe the general arrangement, key trade-offs, and expected maintenance activities.

The shaft of the Target Assembly provides stiffness, shielding, and a locating feature for the target segments. The talk will cover the design challenges of the shaft, including deflection requirements, clearances to surrounding shielding, and their influence on the prompt radiation levels.

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# Synergistic effect of ion irradiation and LBE corrosion on martensitic steel

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Accelerator Driven sub-critical System (ADS) has been considered as a powerful technology for transmuting long-lived nuclides and minor actinides (MAs) [1]. In these systems, a spallation target located inside a sub-critical fast reactor core provides the external source neutrons to drive the operation of the ADS. The neutrons are produced via spallation reaction of high power proton beams colliding with the target material.
The lead-bismuth eutectic (LBE) was commonly selected as the target material and the coolant material due to its higher neutron yield and high good thermal-physical properties. As the most critical component in the spallation target, the beam window will be subjected to intensive irradiation of the proton/neutron mixed spectrum and liquid LBE corrosion. The degradation of mechanical properties of the beam window under such combined harsh environment are essential for the safe operation of spallation target. For the safety and reliability of the target, it is necessary to assess the behaviors of the beam window material under both irradiation damage and molten LBE.

In order to systematically investigate the synergistic effect of irradiation and LBE corrosion on structural materials of the spallation target, the Heavy Liquid Metal and Irradiation Facility (HLMIF) using heavy ion as an irradiation source was constructed in the National Laboratory of Heavy ion Accelerator in Lanzhou. Based on the HLMIF, 247MeV Ar ions was used to irradiate SIMP sample simultaneously exposed to flowing LBE, and a synergistic effect was investigated. The irradiation damage on the interface of the SIMP steel/LBE was 1.36dpa, 4.8dpa and 13.6dpa. The results shows that the Ar ions irradiation not only accelerates the corrosion rate, but also remarkably alters the microstructure of surface oxides compared to the only corroded sample. The mechanism of the synergistic effect will be briefly discussed in this study.

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**Application of new materials data and/or safety codes, computational modelling/analysis / 49**

## BOILER AND PRESSURE VESSEL ANALYSIS OF A MINI-CHANNEL BEAM DUMP

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The Facility for Rare Isotope Beams (FRIB) is a high-power heavy ion accelerator facility at Michigan State University completed in 2022. Its driver linac is designed to accelerate all stable ions to energies above 200 MeV/u with beam power of up to 400 kW. Currently, FRIB is operating at 10 kW, delivering multiple primary beam species. The beam dump absorbs approximately 75% of the primary beam power. The existing beam dump head can accommodate up to 10 kW operation, with a planned transition to a mini-channel beam dump design. Presented here is an overview of the mini-channel beam dump head and details the Boiler and Pressure Vessel Code analysis in ANSYS ® simulations.

**Poster session / 50**

## Development of 3D-printed beam windows for COMET Phase alpha and Phase 1

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The COherent Muon to Electron Transition (COMET) project is at operation of phase-α in J-PARC Hadron experimental facility. The project objective is to explore the lepton flavor violation process by searching the neutrino-less conversion of muons into electrons. In the beam line, the muon transport solenoids are composed of superconducting magnets which are kept cool by liquid Helium (LHe). The installed beam windows should be robust enough to withstand against the rapid and high pressure increase over 0.3 [MPa] in emergency of LHe quenching until rupture disks break. In a same time, for high transmission efficiency the material density must be low, and the thickness must be as thin as possible, while minimizing the nuclear heat generation by beam energy loss. Thus, for the phased-α we have developed the 3D-printed windows made by Ti-6Al-4V which has radius curvature on to the beam passing area, instead of a conventional thin and flat shape.
Now we started the R&D of the beam windows for coming phase-1. In the phase, the proton beam power will be increased to 3.2 [kW] and there are other beam windows to be developed. At present we focus on the development of the window for the beam duct. The window needs to satisfy; (1) installed in bore diameter of 260 [mm], (2) bonded with diameter of 250 [mm] Aluminum duct, (3) withstands over 0.3 [MPa] in emergency, (4) sustains atmospheric pressure from opposite face of the window in operation condition. By the R&D we decided to manufacture the window by 3D-printer with the material AlSi10Mg which can weld to the beam duct directly to maximize the beam passing area. In the presentation we present the R&D results and history of Ti64 window for phase-α and the progress of the Aluminum alloy window for phase-1.

**Fundamental studies on the effects of radiation damage in materials. Innovative radiation damage resistant materials technology / 52**

## Status and Plan of Materials and the Radiation Damage Analysis in FRIB

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The Facility for Rare Isotope Beams (FRIB) is a scientific user facility for nuclear science. FRIB's superconducting radio frequency (SRF) linear heavy-ion accelerator can accelerate all the ions up to uranium to energies above 200 MeV/u. The design beam power is 400 kW, which, once achieved, will extend the heavy-ion accelerator power frontier by more than one order of magnitude. FRIB currently operates at a primary beam power of 10 kW.
This presentation will focus on the status and plan of materials and the radiation damage analysis in FRIB for the challenges and discuss the technical several issues. Analysis based on material irradiation effects caused by heavy particle beams and the influence of the environment in which they are used is necessary. In particular, the function of the beam dump is planned to include isotope production in water, so the fabricability including high reliable designing, soundness, and durability of the thin vessel structure of the beam dump are one of very important subject to be evaluated. The radiation damages for the related facilities will be also included and discussed.

**Facility overview, updates and developments. Operational experience of targets, beam windows, cooling and ancillary systems / 53**

## Challenges of FRIB Facility and Current Status

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The FRIB accelerator systems include ion sources, front-end, linear accelerator, targetry, fragment separator, beam delivery, the cryogenic plant, beam dump, and supporting systems. This presentation will focus on the challenges, plan, and current status of FRIB accelerator systems and discuss the technical issues to be achieved. Very recently, ramp-up test at 21.9 kW in average power using 82 Se beam accelerated to 228 MeV/u has been successfully performed at the FRIB facility on July 16, 2024.

**Application of new materials data and/or safety codes, computational modelling/analysis / 54**

## Gas Production in the Proton Beam Window and Target Vessel at SNS

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Due to its profound effects on the microstructure, void swelling and embrittlement of structural materials, the transmutation gas production of hydrogen and helium is an important metric in evaluating radiation damage on structural materials in a spallation facility. A systemic measurement of gas productions at the proton beam window (PBW) and mercury target vessel at SNS is underway. Detailed calculations were carried out in preparation for the measurements for an Al-6061 and Inconel 718 PBW and a SS316L target vessel. To accurately predict gas production at different locations of the PBW and target vessel, incident proton beam profile and direction distributions at the PBW were carefully constructed from the real time measurements of proton beam distributions at a harp located at 7.2 m upstream of the PBW. The resulting proton beam at the PBW is thus a time-averaged beam profile with a distribution of beam directions at each pixel of the profile over the operation time of the PBW or target vessel. New hydrogen and helium production cross sections were developed to include cross sections for each gas isotope, instead of for each gas element as used in the current cross section library. In such a way, the calculated gas productions will be better checked by a mass spectroscopy based experimental method capable of distinguishing each gas isotopes. Recent versions of both CEM and INCL4 intra-nuclear cascade (INC) models were used in generating this new set of hydrogen and helium production cross sections. The differences in these cross sections are discussed, although the main purpose of using different INC models was to compare and benchmark them with the planned measurements in the near future.

**Application of new materials data and/or safety codes, computational modelling/analysis / 55**

## Monte Carlo-Driven Analysis of Key Nuclear Parameters in Target Design for Optimised Neutron Yield: Insights from ISIS-TS1

# and Notional Design Evaluation

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The nuclear and engineering design of the target is critical for optimizing the performance and efficiency of neutron sources. This study presents a Monte Carlo-driven analysis, conducted with the FLUKA code, of the currently operating ISIS-TS1 target and several derived notional designs. The analysis aims to quantify the impact of key nuclear parameters—such as proton range and the neutron absorption cross sections of the coolant and target vessel—on the effective neutron yield. The study also investigates the significance of the coupling between the reflector and target on overall system performance. This analysis provides insights into the relationship between some selected target design parameters and the neutron yield, uncovering important lessons that can help to better address future design efforts. A clear understanding of the role of some nuclear parameters, and how this role is influenced by coupling with the reflector, could be particularly relevant for future programmes focused on developing more efficient neutron source concepts .The findings of this analysis can contribute to a better understanding of the effective impact of some nuclear parameters of the target design to the effective neutron leakage and underscore the importance of integrating advanced MC simulation tools early in the design process.

**Liquid/solid and particle beam interactions and associated studies: pressure waves, cavitation, erosion, corrosion etc and mitigation techniques / 56**

# Long term fuel assembly material and hydraulic behavior in HLM systems

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During a series of fuel assembly (FA) tests at the refurbished MEXICO facility at SCK CEN, extensive research was conducted to evaluate the material behavior and performance of heavy liquid metal (HLM)-cooled wire-wrapped fuel assemblies. The primary objective was to assess long-term material compatibility such as corrosion resistance, and thermal-hydraulic behavior under representative operating conditions, which are crucial for the sustained viability of HLM-cooled systems like the MYRRHA reactor.
The experimental setup was designed to replicate MYRRHA's operational parameters, including maintaining an oxygen concentration of 2 x $10^{-7}$ wt.%, nominal FA inlet flow rates, and thermal-hydraulic conditions to simulate the reactor's maximum cladding temperature. These conditions were maintained over a six-month testing period, effectively simulating two operational cycles. Precise control of oxygen concentration was achieved through advanced cold trap technology, ensuring minimal oxygen variation and stable chemical conditions within the HLM coolant.
Post-experiment material analysis revealed a thin, non-uniform spinel oxide layer on the fuel pin cladding, with an average thickness of approximately 500 nanometers. Minor material degradation was observed, with a maximum damage depth of around 40 microns. These findings suggest that the materials used in the FA exhibit acceptable corrosion resistance under MYRRHA reactor nominal operating conditions.
Key findings from the tests also include the stability of the FA's thermal-hydraulic performance over time, attributed to the controlled oxygen concentration. This resulted in minimal oxide particle formation within the coolant, reducing the risk of oxide deposition/accumulation and its potential impact on FA heat transfer.
Finally, the data collected provide an invaluable dataset for validating numerical simulations related to both chemical and thermal-hydraulic modeling. This validation is crucial for enhancing predictive

capabilities and ensuring the accurate assessment of material performance over the reactor's lifecycle, particularly for systems like MYRRHA that rely on heavy liquid metals as coolants.

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## Understanding the fracture of nuclear grade graphite with radiation damage and thermal heating

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Nuclear-grade graphite is often used as target materials in a particle accelerator; it is also used as moderator and structural materials in nuclear fission reactor cores in the UK as well as in some GenIV reactor designs. This work will provide an overview of the types of graphite materials used in these applications and will detail the differences in their microstructure and thermal-mechanical properties. These nuclear grade graphite materials will often shrink upon irradiation causing dimensional changes and subsequent stresses leading to fracture; but there are also exceptions where low temperature irradiation leads to swelling first at low dose. These will be discussed. When used with CO2 coolant in a nuclear fission reactor, there is radiolytic oxidation reducing the density of the graphite. Details will be given in terms of the irradiation induced damages in these graphite materials and how their behaviours change with thermal heating to temperatures at 1100℃.

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## Resistivity of Frenkel Pair in BCC W from First Principles

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Tungsten is a key material for both nuclear fusion and as a spallation target for neutron sources. In nuclear fusion, W is being considered for heat divertors thanks to its good thermal conductivity, resistance to irradiation damage, and high melting temperature. Furthermore, due to its high density and atomic mass, W is an excellent material as a target for neutron sources. In both settings, W is exposed to high amount of damage measured as displacement per atoms (DPA), which results in the creation of Frenkel pairs, voids, dislocations, and transmutation reactions. Tungsten will see irradiation damage of 10-35 DPA in its working lifetime. This amount of damage will negatively affect the properties of the material. In this work, we propose a method to calculate from First Principles using Density Functional Theory the changes in resistivity due to a Frenkel Pair. Specifically, a vacancy and a self-interstitial (crowdion in the [111]). The calculated resistivity from a vacancy is 8.07 μΩm, while the experimental value is 7 μΩm. For the Frenkel pair, the resistivity is 31.5 μΩm, while the experimental value is 27 μΩm. This method can be expanded in multiple ways. First, to consider the changes in resistivity due to presence of transmutation elements like rhenium, osmium, and tantalum under neutron irradiation in W. Second, with hexagonal closed-packed materials like titanium, which are being considered as containment and window material for the graphite target for the Long Baseline Neutrino Facility. Finally, this method can be further developed to calculate changes in electronic contribution to thermal conductivity. Knowing how the properties of the material change due to irradiation can help in maximizing its service life, finding ways of improving the properties of the material, and enables to test novel materials before manufacturing.

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## Erosion Problem for Granular Tungsten Targets Using Circulating Fluidized Beds- Experimental Investigation of Surface Damage

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For a higher power neutron or muon source, a flowing granular target would offer superior thermal performance and enable a greater particle yield than a monolithic target. However, the flow of tungsten would necessitate solid particle impingement erosion conditions to occur. Previous work at RAL to produce a pilot plant conveying granular tungsten resulted in severe erosion of stainless-steel pipes. Erosion is a well understood phenomenon when considering sand-like materials such as silica, but modelling still requires experimental coefficients for correlations to be correctly applied to each tribosystem. Understanding of the erosion conditions created by high density media is much more limited, and the applicability of wear correlations, developed for low density materials, is untested.

A new test methodology has been developed at the University of Sheffield for recreating accelerator erosion conditions, and initial tests have been conducted for a small selection of metals and conditions. Damage to the eroded surfaces has been analysed and wear features found to be like those created in erosion by silica materials. Roughening was found to be strongly related to the impingement angle of the particles on the surface, and it is suggested that the size and shape of the impinging particle is also significant. Simple material comparisons such as hardness were also shown to be limited in applicability for these high energy particle impacts. Future work in this area will enable confident material selection for granular target containers.

**Collaborations, opportunities and future plans e.g. for materials irradiations & PIE / 60**

## Collaborations and opportunities for materials development, manufacturing and irradiations at University of Birmingham

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In this talk, we will be presenting our capacities at the University of Birmingham for alloy development, manufacturing, testing and irradiation. Examples will be given on additive manufacturing of Cu/Cu alloys and W, in situ mechanical testing of steels, and irradiation tests using the cyclotron.