ANS Conference

Tritium 2016

April 17-22, 2016
Charleston Marriott
Charleston, South Carolina
United States

11th International Conference on Tritium Science & Technology

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Publication of the Program and Proceedings was provided by a grant from the United States Department of Energy Office of Fusion Science.

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GENERAL MEETING INFORMATION

TRITIUM TECHNICAL SESSIONS BY DAY

TRITIUM TECHNICAL PAPERS

Letter From the Mayor of Charleston

City of Charleston

JOHN J. TECKLENBURG
MAYOR

Welcome to the City of Charleston!

On behalf of the citizens of the City of Charleston, I extend a warm Southern welcome to the participants of the 11th International Conference on Tritium Science and Technology. As time allows, we hope that you will explore the rich history of Charleston.

Charleston South Carolina, founded in 1670, is defined by its cobblestone streets, horse drawn carriages and pastel pre-Civil-War-era houses, particularly in the bustling French Quarter and Battery areas. The Battery promenade and Waterfront Park both overlook Charleston Harbor, while Fort Sumter, a Federal stronghold where the first shots of the Civil War rang out, lies across the water.

The Charleston area is comprised of many historic houses, churches, and public buildings that are still in daily use. We are proud that Charleston has been voted #1 Best U.S. City by Conde' Nast Traveler Readers' Choice Awards for the past five years.

With my very best wishes for an enjoyable and successful conference,

I am Most sincerely yours,

John J. Tecklenburg

Letter From General Chair

TRITIUM 2016
11th International Conference on Tritium Science & Technology
April 17-22, 2016 • Charleston, SC • Charleston Marriott

GENERAL CHAIR	TECHNICAL PROGRAM CHAIR	PUBLICATIONS CHAIR
Dr. Robert P. Addis	Dr. James E. Klein	Dr. Kevin R. O'Kula
(803) 725-3325	(803) 725-4203	(803) 502-9620
addis@tritium2016.org	klein@tritium2016.org	kevin.okula@aecom.com

Dear Colleagues,

On behalf of the National Organizing Committee, the Technical Program Committee, the International Steering Committee, our corporate exhibitors and conference supporters, we welcome you to Charleston South Carolina and the 11th International Conference on Tritium Science and Technology (Tritium 2016).

Tritium 2016 has been organized by the American Nuclear Society and is being hosted by the Savannah River National Laboratory. The Technical Program Committee, whose membership is one third from Asia, one quarter from Europe and the remaining 40% from North America, almost equally from the US and Canada, has provided a strong technical program.

The conference is being supported by a broad spectrum of corporate leaders who have enabled the conference to provide, to registered conference attendees and spouse/guests, an ample supply of meals and breaks. These include a welcome reception Sunday night, a dinner cruise off Charleston Tuesday evening, a banquet Thursday evening, lunches Monday, Tuesday and Thursday, as well as breakfasts and breaks each day. The publication of the Conference Program and the peer reviewed papers in the ANS Journal Fusion Science and Technology is provided by a grant from the U.S. Department of Energy, Office of Fusion Energy Sciences. We are grateful to these corporate and government leaders who are furthering tritium science and technology.

This conference series has a 36-year history starting with the American Nuclear Society national topical meeting Tritium Technology in Fission, Fusion and Isotopic Applications held in 1980 in Dayton, Ohio. Since then it has been held an additional nine times, rotating between North America, Europe and Asia approximately every three years. The objective of the conference is to provide a forum for an exchange of information on science, technology, engineering, and general experience in the safe and environmentally sound handling of tritium for fusion, fission, pharmaceuticals, and other isotopic applications.

We also welcome you to Charleston, founded in South Carolina in 1670; she is one of the grand southern ladies of American cities. Located in the deep-south, Charleston has a long history of Ante-bellum plantations, Revolutionary and Civil War related sites and artifacts. It has a flourishing artistic community, and its cuisine is delightful. Charleston is somewhat similar to Nice, France, where the last conference was held, in that it retains the charms of a bygone era, yet is fresh and ready for business today. We hope you and or your spouse/guest will avail yourself of the charms of this gem of the South during your stay, but do attend the meeting!

We wish you a productive and enjoyable conference.

Regards,
Robert Addis
General Chair, Tritium 2016

Letter From Technical Program Chair

TRITIUM 2016
11th International Conference on Tritium Science & Technology
April 17-22, 2016 • Charleston, SC • Charleston Marriott

ANS

GENERAL CHAIR
Dr. Robert P. Addis
(803) 725-3325
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Dr. James E. Klein
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Dear Colleagues,

On behalf of the Technical Program Committee, it is our pleasure to welcome you to the 11th International Conference on Tritium Science and Technology (Tritium 2016), in Charleston, SC USA, from April 17 – 22, 2016. The International Conference on Tritium Science and Technology brings together the world experts in tritium science and technology to report on technical progress in a number of areas. This conference continues the tradition of a high quality forum for exchange of scientific, technological, and engineering information related to tritium which is only surpassed by the next International Conference on Tritium Science and Technology!

We are departing from tradition of recent conferences by having parallel oral sessions. Feedback from funding organizations has come back loud and clear: likelihood to attend increases greatly if making an oral presentation. The majority of 247 abstracts submitted were from international participants so parallel oral sessions were needed to maximize conference participation. Simultaneous sessions were limited to two and attempts were made to minimize overlap or competing interests of theses sessions. Strategic use of conference space will maximize social interactions between conference participants, exhibitors, and sponsors. We hope this "experiment" with parallel sessions is an acceptable compromise for greater conference participation and the inclusion of more published manuscripts in special editions of Fusion Science and Technology.

The high quality of the technical program is reflected by the leadership and assistance of members of the Technical Program Committee. The committee members, respected leaders in the tritium community, screened all abstracts to make the difficult decisions of selecting the oral presentations for the conference. Committee members balanced technical content with representation between various research organizations located throughout the world. It is believed the best technical program was created thanks to the help and support of these committee members.

In closing, we encourage you to interact and participate with the tritium community attendees during the program presented this week. We hope that when you leave, you will have many great ideas for future research and collaborations to explore before the next conference. Again, welcome to Tritium 2016! We welcome your feedback as you enjoy the conference.

Sincerely,

James Klein

James Klein
Technical Program Chair of Tritium 2016

National Organizing Committee

11th International Conference on Tritium Science and Technology

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Savannah River National Laboratory

ASSISTANT GENERAL CHAIR
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Rosalind H. Blocker
Savannah River National Laboratory

Technical Program Committee

11th International Conference on Tritium Science and Technology

NAME	ORGANIZATION	COUNTRY
Jim Klein (Chair)	SRNL	USA
Paul Cloessner	SRNL	USA
Walter Shmayda	Univ. of Rochester, LLE	USA
David Demange	KIT	Germany
Karine Liger	CEA	France
Christian Gisolia	CEA	France
Alexey Golubev	ISC	Russia
Sam Suppiah	CNL	Canada
Hugh Boniface	CNL	Canada
Armando Antoniazzi	Kinectrics Inc.	Canada
Sei-Hun Yun	NFRI (National Fusion Research Institute)	South Korea
Kyu Min Song	KHNP (Korea Hydro and Nuclear Power)	South Korea
Satoshi Fukada	Kyushu University	Japan
Yasuhisa Oya	Shizuoka University	Japan
Satoshi Konishi	Kyoto University	Japan
Takumi Hayashi	JAEA	Japan
Clark Snow	SNL (Sandia National Laboratory)	USA
Lee Cadwallader	INL	USA
Shuming Peng	CAEP	China
Dario Castiglione	AWE	UK
Kimberly Burns	PNNL	USA

International Steering Committee

11th International Conference on Tritium Science and Technology

2015 MEMBERS	REPRESENTING
Robert Addis	(SRNL)* US
Walter Shmayda	(Rochester) ICF
Christian Grisolia	(CEA)* EU
Manfred Glugla	(ITER) ITER
Beate Bornchein	(KIT) EU
Sam Sappiah	(AECL) Canada
Arkadiy Yakhimchuk	(RFNC) RU
Shuming Peng	(INPC) CN
Prof. Yuji Hatano	(U. Toyama)* JA
Toshihiko Yamanishi	(JAEA)* JA
Scott Willms	(ITER/LANL) Chair

Schedule at a Glance

Monday, April 18

8:00 am-10:00 am	Session 1. Opening and Plenary
10:20 am-12:20 pm	Session 2. Operating and Tritium Facilities
1:40 pm-3:00 pm	Session 3. Tritium Facility Developments
3:20-5:20 pm	Session 4A. Tritium Interaction with Materials: Hydride Formers
3:20-5:20 pm	Session 4B. Tritium Confinement and Safety: Design Considerations
5:20-7:00 pm	Session 5. Poster Session 1
7:00-9:00 pm	International Steering Committee Meeting (Opal Two)

Tuesday, April 19

8:00 am-10:00 am	Session 6A. Tritium Interaction with Materials: Metals—I
8:00 am-10:00 am	Session 6B. Tritium Decontamination and Waste Management
10:20 am-12:20 pm	Session 7A. Tritium Interaction with Materials: Metals—II
10:20 am-12:20 pm	Session 7B. Tritium Processing: Use of Zeolites
1:40 pm-3:00 pm	Session 8A. Gas Phase Detritiation
1:40 pm-3:00 pm	Session 8B. Tritium Storage
3:20-5:00 pm	Session 9A. Tritium Facilities: Operations and Maintenance
3:20-5:00 pm	Session 9B. Tritium Processing, Purification, and Separations
5:00-6:20 pm	Session 10. Poster Session 2

Wednesday, April 20

8:00 am-10:00 am	Session 11A. Liquid Phase Detritiation
8:00 am-10:00 am	Session 11B. Tritium Biological Effects
10:20 am-12:20 pm	Session 12A. Tritium Interaction with Materials: Reactor Materials
10:20 am-12:20 pm	Session 12B. Tritium Breeding and Extraction—I

Thursday, April 21

8:00 am-10:00 am	Session 13A. Gas Phase Tritium Measurements
8:00 am-10:00 am	Session 13B. Tritium Environmental Effects
10:20 am-12:20 pm	Session 14A. Liquid Phase Tritium Measurements
10:20 am-12:20 pm	Session 14B. Tritium Breeding and Extraction—II
1:40 pm-3:00 pm	Session 15A. Future Tritium Facility Planning
1:40 pm-3:00 pm	Session 15B. Tritium Pumping
3:20-5:00 pm	Session 16A. Other Tritium Topics
3:20-5:00 pm	Session 16B. Tritium Confinement and Safety
5:00-6:20 pm	Session 17: Poster Session 3
5:30-7:00 pm	Banquet Reception
7:00-9:00 pm	Conference Banquet

Friday, April 22

8:00 am-9:40 am	Session 18. Near-Term, New Tritium Capabilities
10:00 am-11:00 am	Session 19. Panel Discussion: Progress in Tritium Standards and Regulation
11:00 am-12:00 pm	Session 20. Closing Ceremony

Conference Information

REGISTRATION HOURS

Location: Crystal AB Promenade

Sunday, April 17	3:00pm-6:00 pm
Monday, April 18	7:30 am-1:00 pm 3:00 pm-6:00 pm
Tuesday, April 19	7:30 am-1:00 pm 3:00 pm-6:00 pm
Wednesday, April 20	7:00 am-1:00 pm
Thursday, April 21	7:30 am-1:00 pm 3:00 pm-6:30 pm
Friday, April 22	7:30 am-1:30 pm

SPEAKER INFORMATION

Speakers for each day of the Tritium 2016 should attend the Speaker Breakfast that morning so they can transfer their talk to a computer for presentation later that day. Please bring Power Point files on a USB drive for transfer (bring a PDF version as back-up). Bring short biographical sketch (one paragraph maximum) for introduction by the Session Chairs. Please report to the Session room at least 15 minutes before the start of the session. Please sit near the front of the room near the Session Chairs to facilitate communications. Non-plenary presentations should be no longer than 15 minutes; each paper will have 2-5 minutes for questions at the end.

SPEAKER READY ROOM

Location: Opal Two

Monday, April 18	7:00 am-5:00 pm
Tuesday, April 19	7:00 am-5:00 pm
Wednesday, April 20	7:00 am-12:00 pm
Thursday, April 21	7:00 am-5:00 pm
Friday, April 22	7:00 am-10:00 am

SPEAKER BREAKFAST

Location: Opal Two

Monday, April 18	7:00 am-8:00 am
Tuesday, April 19	7:00 am-8:00 am
Wednesday, April 20	7:00 am-8:00 am
Thursday, April 21	7:00 am-8:00 am
Friday, April 22	7:00 am-8:00 am

EXHIBITS Location: Crystal AB and CD Promenades

Please set up Exhibit booths between 12:00 pm and 4:00 pm on Sunday, April 17. The official opening of the exhibit area is Monday morning, April 18. Booth removal should be planned for Wednesday, April 20, no earlier than 12:00 pm. We thank Tritium 2016 Exhibitors for their support.

POSTERS SET-UP Location: Crystal DEF

Monday April 18

Posters can be put up during lunch (12:30 pm-1:30 pm) or the afternoon break (3:00 pm-3:20 pm) and must be in place by 3:20 pm Monday. Monday posters must be removed Tuesday by the end of the morning break (10:20 am).

Tuesday April 19

Posters can be put up during lunch (12:30 pm-1:30 pm) or the afternoon break (3:00 pm-3:20 pm) and must be in place by 3:20 pm Tuesday. Tuesday posters must be removed Thursday by the end of the morning break (10:20 am).

Thursday April 21

Posters can be put up during lunch (12:30 pm-1:30 pm) or the afternoon break (3:00 pm-3:20 pm) and must be in place by 3:20 pm Thursday. Thursday posters must be removed Friday by the end of the morning break (10:20 am).

Conference Information

ATTENDEE MEALS

Sunday, April 17
7:00 pm-9:00 pm
Opening Reception - Courtyard

Monday, April 18
7:00 am-8:00 am
Continental Breakfast - Crystal CD Promenade

10:00 am-10:20 am
& 3:00 pm - 3:20 pm
Coffee Breaks - Crystal Promenade

12:30 pm-1:30 pm
Lunch - Emerald

Tuesday, April 19
7:00 am-8:00 am
Continental Breakfast - Crystal CD Promenade

Tuesday, April 19
10:00 am-10:20 am & 3:00 pm - 3:20 pm
Coffee Breaks - Crystal CD Promenade

12:30 pm-1:30 pm
Lunch - Crystal CD Promenade

Wednesday, April 20
7:00 am-8:00 am
Continental Breakfast - Crystal CD Promenade

10:00 am-10:20 am
Coffee Breaks - Crystal CD Promenade

Thursday, April 21
7:00 am-8:00 am
Continental Breakfast - Crystal CD Promenade

Thursday, April 21
10:00 am-10:20 am & 3:00 pm - 3:20 pm
Coffee Breaks - Crystal CD Promenade

12:30 pm-1:30 pm
Lunch - Crystal CD Promenade

5:30 pm-7:00 pm
Banquet Reception - Crystal Promenade

7:00 pm-9:00 pm
Conference Banquet - Crystal Ballroom ABC

Friday, April 22
7:00 am-8:00 am
Continental Breakfast - Crystal CD Promenade

9:40 am-10:00 am
Coffee Breaks - Crystal CD Promenade

DINNER CRUISE: TUESDAY, APRIL 19

Dinner Cruise on the SpiritLine Cruise Ship "Lowcountry"

Bus departure from the Marriott Charleston is scheduled for 6:45 pm. Cruise is from 7:30 pm to 9:30 pm.

Join other Tritium 2016 Conference attendees for a relaxing cruise around the Charleston Harbor. Sites that we will see during the cruise are Castle Pinckney, Charleston Battery, the Aircraft Carrier USS Yorktown, the iconic Arthur Ravenel Bridge over the Cooper River, and the historical landmark Fort Sumter (which received the first shots of the U.S. Civil War, April 14, 1861).

Enjoy the sites while sipping a cool drink and enjoying a menu of South Carolina BBQ and other fixins', prepared especially for the event. No cost – we are grateful for conference supporters for fully defraying the cost of this exceptional dinner cruise.

TECHNICAL TOURS: WEDNESDAY, APRIL 20

Savannah River Site

Bus departure from Marriott Charleston at 9:30 am, with the tour scheduled to conclude by 4:30 pm. Return to hotel tentatively by 9:00 pm. The tour fee includes a box lunch, and will stop on the return trip for an on-your-own dinner at 5:00 PM (Miller's Bread Basket, Blackville, SC). Dress in comfortable clothing with closed-toe shoes (required). Limited to 45 attendees, but currently sold out - See Conference desk for any change in this status.

The Savannah River Site (SRS) is a key U.S. Department of Energy industrial complex responsible for environmental stewardship, environmental cleanup, waste management and disposition of nuclear materials. On your tour of this 310-square-mile site, you'll discover the patriotic history of the Site's beginnings and the important role that SRS continues to play in addressing the nation's most challenging issues. You'll experience the vast infrastructure and facilities that continue to meet national security needs, while maintaining the highest possible safety and security standards. And, you'll hear about the expertly engineered processes in place to remediate and protect the environment. Primarily, however, you will learn how SRS contributes to making the world a safer place.

CHARLESTON AREA TOURS

Technical sessions are not scheduled for Wednesday afternoon. Instead, this midway point in the Conference is available as your own free time. We suggest that you consider the recreational and cultural opportunities available in this area.

Individual and group tour possibilities are numerous in the Charleston and surrounding area. See the Marriott Charleston concierge for more information, and the schedule for the free shuttle bus to cultural, Charleston Battery and other districts.

Technical Sessions: Monday, April 18

1 - Opening and Plenary

Location: Crystal Ballroom ABC

8:00 am-10:00 am

1-1 8:00 am
Welcome and Introduction
Cochairs: Robert Addis (SRNL), Steve Xiao (SRNL), James E. Klein (SRNL), John Tecklenburg (Mayor of Charleston), Terry Michalske (SRNL), Scott Wilms (ISC),

1-2 8:40 am
Preparation of and Upgrades to the Active Gas Handling System for the Planned JET TT and DT Campaigns
Robert Smith, invited, John Tecklenburg, Terry Michalske, Scott Wilms, Robert Addis, James Klein

1-3 9:20 am
Tritium Activities at the University of Rochester's Laboratory for Laser Energetics
W. T. Shmayda, M. D. Wittman, J. L. Reid, and R. F. Earley

2 - Operating and Tritium Facilities

10:20 am-12:20 pm

Location: Crystal Ballroom ABC

Cochairs: Masashi Shimada (INL), David Demange (KIT), Tukumi Hayashi (JAEA)

2-1 10:20 am
Fifteen years of Operating in a Tritiated Waste Treatment Nuclear Facility
Christophe Douche, S. Rochefort, J. Avenet, A. Arseguel, O. Leagaie

2-2 10:50 am
Research Plan of Fusion Fuel Cycle and Related Tritium Science in Korea
Ki Jung Jung, Sei-Hun Yun, Hongsuk Chung, Kyu-Min Song

2-3 11:20 am
Review of the TLK Activities Related to Water Detritiation, Isotope Separation Based on Cryogenic Distillation and Development of Barriers Against Tritium Permeation
Ion Cristescu, A. Bükki-Deme, H. Dittrich, N. Gramlich, R. Grösle, F. Krammer, N. Lohr, C. Melzer, F. Rehlinghaus, P. Schäfer, Stefan Welte

2-4 11:50 am
Overview of Fusion-Related Tritium Research in Japanese Universities
Yuji Hatano, Yasuhisa Oya, Satoshi Konishi, Satoshi Fukada, Akio Sagara

3 - Tritium Facility Developments

1:40 pm-3:00 pm

Location: Crystal Ballroom ABC

Cochairs: Walter T. Shmayda (Univ of Rochester), B. Bornschien (CAEP), Ion Cristescu (KIT)

3-1 1:40 pm
Development of Tritium Science and Technology in INPC
Shuming Peng, Xiaosong Zhou, Xiaojun Chen, Zhilin Chen, Heyi Wang

3-2 2:00 pm
Cernavoda Tritium Removal Facility—Evolution in TRF Design
L. Stefan, N. Trantea, A. Roberts, S. Strikwerda, A. Antoniazzi, D. Zaharia

3-3 2:20 pm
The Five Phases—The Way to Standard Tritium Operation of KATRIN
Beate Bornschein, Uwe Besserer, Markus Steidl, Michael Sturm, Kathrin Valerius and Jürgen Wendel

3-4 2:40 pm
Water Detritiation System for ITER—Evaluation of Design Parameters
Hugh Boniface, Nirmal Gnanapragasam, Donald Ryland, Sam Suppiah, and Alex Perevezentsev

4A - Tritium Interacation with Materials: Hydride Formers

3:20 pm-5:20 pm

Location: Crystal Ballroom ABC

Cochairs: Clark Sheldon Snow (SNL), Xiaosong Zhou (Chinese Academy), Satoshi Fukada (Kyushu Univ)

4A-1 3:20 pm
Tritium and ^{3}He in $ErT_{2-x}He_x$
Clark S. Snow, James F. Browning

4A-2 3:40 pm
^{3}He Release, ^{3}He Retention and Structural Evolution in Ti, Er and Zr Tritides: Tritium Content, Phase, Microscopic Structure and Aging Effects
X.S. Zhou, G.J. Chen, W. Ding, W.D. Wang, S.M. Peng, X.G. Long, J.H. Liang

Technical Sessions: Monday, April 18

4A - Tritium Interacation with Materials: Hydride Formers

3:20 pm-5:20 pm

Location: Crystal Ballroom ABC

Cochairs: Clark Sheldon Snow (SNL), Xiaosong Zhou (Chinese Academy), Satoshi Fukada (Kyushu Univ)

4A-3 4:00 pm
Release of Radiogenic Helium-3 from Uranium Tritide
J. Northall and J.P. Knowles

4A-4 4:20 pm
Effects of Grain Morphology on Aging Properties of Erbium Tritide Film
Wang Weidu, Zhou Xiaosong, Ding Wei, Peng Shuming, Long Xinggui, Liang Jianhua, Cheng Guijun, Liu Jinhua, Chen Jin, Liu Qiong

4A-5 4:40 pm
Electron Microscopy of Helium Bubbles in a Palladium Alloy
David B. Robinson, Mark R. Homer, Joshua D. Sugar, E. Lynn Bouknight, Kirk L. Shanahan

4A-6 5:00 pm
Hydrogen Isotope Effect on Getter Rate
Bruce Schmitt, Monte Elmore, Ed Love, Kim Burns

4B - Tritium Confinement and Safety: Design Considerations

3:20 pm-5:20 pm

Location: Crystal Ballroom DEF

Cochairs: Satoshi Konishi (Kyoto Univ), M. S. Lyttle (ORNL), Nicephore Bonnet (Kurion)

4B-1 3:20 pm
Tritium Challenges and Plans for ITER Pellet Fueling and Disruption Mitigation Systems
M.S. Lyttle, L.R. Baylor, R.E. Battle, S.J. Meitner, D.A. Rasmussen, J.M. Shoulders

4B-2 3:40 pm
Preliminary Design of a Tritium Control System for Fluoride-Salt-Cooled High-Temperature Reactors
Xiao Wu, David Arcilesi, Xiaodong Sun, Richard Christensen, Piyush Sabharwall

4B-3 4:00 pm
Tritium Constraints and Strategy for the Fusion Deployment
Satoshi Konishi, Shutaro Takeda, and Ryuta Kasada

4B-4 4:20 pm
Use of TAS in Assessment on Tritium Self-sufficiency, Resource Sustainability and Radioactive Safety for Fusion Reactor
Muyi Ni, Baojie Nie, Jieqiong Jiang, Yican Wu, FDS Team

4B-5 4:40 pm
Estimation of Tritium Release from LLCB TBM & Its Ancillary Systems and Tritium Management in Different Locations of ITER
Priyanka Brahmbhatt, Amit Sircar, Rudreksh Patel, E. Rajendra Kumar, Sadhana Mohan, Kalyan Bhanja

4B-6 5:00 pm
Estimation of Tritium Permeation Rate to Cooling Water in Fusion DEMO Condition
Kazunari Katayama, Youji Someya, Kenji Tobita, Hirofumi Nakamura, Hisashi Tanigawa, Makoto Nakamura, Nobuyuki Asakura, Takumi Chikada, Yuji Hatano, Satoshi Fukada

Technical Sessions: Monday, April 18

5 - Poster Session 1

5:20 pm-7:00 pm

Location: Crystal Ballroom DEF

Cochairs: Anita S. Poore (SRNL), Melissa Golyski (SRNS), Benton Randall (SRNL)

Tritium Confinement and Safety

5-01

Design Considerations for Tritium Handling Systems: An Engineering Approach

Benjamin Huffer

5-02

Model for the Production, Diffusion, and Containment of Tritium in PWRs

N. Bonnet, G. Bonhomme, Y. Iwasaki, D. Carlson, J. Raymont

5-03

Dynamic Tritium Fuel Cycle Analysis for Fusion Reactor Based on System Dynamics Model

Baojie Nie, Muyi Ni, Jieqiong Jiang, Yican Wu, FDS Team

5-04

The Status of the Tritium Transport Libraries in EcosimPro

C. Moreno, A. Rueda, J. Serna, F.R. Urgorri, E. Carella, J. Castellanos

Tritiated Water Processing

5-05

Conception of Multi-Purpose Heavy Water Detritiation Plant

S.D. Bondarenko, I.A. Alekseev, O.A. Fedorchenko, K.A. Konoplev, V. Vasyanina

5-06

Theoretical Analysis for Setting up a Mixed Catalytic Packing that Equips a Catalytic Isotopic Exchange Column

Anisia Bornea, Marius Zamfirache, Ioan Stefanescu

5-07

Detritiation and Tritium Storage Technology Development

Hongsuk Chung, Yeanjin Kim, Kwangjin Jung, Seungwoo Paek, Sei-Hun Yun, Do-Hee Ahn

Tracer Techniques

5-08

Application of Imaging Plate Technique and Beta-Ray-Induced X-ray Spectrometry to Measurements of Tritium Distributions in Divertor Tiles Used in JET ITER-Like Wall Campaigns

Yuji Hatano, Kazuya Yumizuru, Seppo Koivuranta, Jari Likonen, Jussi Ikonen, Masao Matsuyama, JET contributors

5-09

Using Dye Tracer Studies to Characterize Tritium in Groundwater; No News can be Good News

Nadia Glucksberg, Kenneth Alepidis

5-10

Geotritium: Implications of DeP Weak Interaction Fusion in Magmatic Systems

Thomas E. Ward

Tritium Decontamination and Waste Management

5-11

Safe Decommissioning for Tritium Storage U Bed

Gui-Ping Dan, Wen Wei, Zhong-Sheng Ma, Jun-Hui Zeng, Zhao-Yi Tan

5-12

Isotope Exchange of Two $LaNi_{4.25}Al_{0.75}$ Hydride Process Beds

Greg Staack

5-13

^{14}C and ^{3}H Determination in Irradiated Graphite from the Thermal Column of the VVR-S Research Reactor

V Fugaru, C Postolache

5-14

Safety Aspects of an Interim Storage Facility for Tritiated Waste from Nuclear Fusion Facilities

F. Derasse, D. Canas, C. Decanis, J. Pamela

5 - Poster Session 1

5:20 pm-7:00 pm

Location: Crystal Ballroom DEF

Cochairs: Anita S. Poore (SRNL), Melissa Golyski (SRNS), Benton Randall (SRNL)

Tritium Supply, Transport, and Storage

5-15

Romanian Tritium for Nuclear Fusion

Richard J. Pearson, William J Nuttall, Olivia Comsa, Liviu Stefan

5-16

Effect on the Secondary Buffer Vessel from the Upper Supplying Failure in Multi-Feed System

Sei-Hun Yun, Min Ho Chang, Hyun-Goo Kang, Dong You Chung, Seungyon Cho, Hyeon Gon Lee, Ki Jung Jung, Hongsuk Chung, Kyu-Min Song, Euy Soo Lee, In-Beum Lee, Jae-Uk Lee, Byeong Eon Park

Tritium Facilities and Operations

5-17

HYSYS/ASPEN+ Advanced Tritium Transfer Modelling Tools for ITER/DEMO Plant Systems

J. M Nougués, J.A. Feliu, G. Campanya, O. Osychenko, L. Batet, L. Sedano

5-18

Operation Experience on Safety System of Tritium Process Laboratory for 28 Years in Japan Atomic Energy Agency

R. Kurata, M. Yamada, T. Suzuki, H. Nakamura, Y. Iwai, K. Isobe, and T. Hayashi

5-19

CODAC Prototyping for Dynamic Tritum Mass-Balance Control Demostration in Tritium Breeding Systems

D. Marchante, L. Batet, R. Granados J. Abellà, S. Colomines, L. Sedano

5-20

Design of Tritium Handling Facility for Tests of ITER SDS Components

Kyu-Min Song, Byung-Wook Ko, Soon Hwan Son, Sei-Hun Yun, Min Ho Chang, Hyun-Goo Kang, Ki-Jung Jung

Other Tritium Topics

5-21

The structure Design of Producing Tritium Blanket for Fusion-Fission Hybrid Reactor Driven by Z-Pinch

Zeng Herong, Liu Zhiyong, Shaohua Wang

5-22

Finite Element Simulation for Liquid Deuterium-Tritium Inside a Capsule with Fill-Tube

Xin Huang, Xiaosong Zhou, Mingming Yu, Chengwei Wen, Shuming Peng

5-23

Analysis of Tritium Migration and Proposed Capture/Removal Methods in Fluoride Salt-Cooled High-Temperature Reactors (FHRs)

Floren Rubio, Bryan Wallace, Edward D. Blandford

Tritium Biological and Environmental Effects

5-24

Using Atmosphere-Forest Measurements to Examine the Potential for Reduced Downwind Dose

Brian J. Viner, Sydney Goodlove

5-25

Uncertainties in Dynamic Modelling of Environmental Tritium Transfer

Anca Melintescu, Dan Galeriu

5-26

Determination of In Vitro Lung Solubility and Intake-to-Dose Conversion Factor for Tritiated 13X Zeolite

Greg Staack, Yung-Sung Cheng, Yue Zhou, Tom LaBone

5-27

Consequences of Different Anthropogenic Sources on Organically Bound Tritium Concentrations in Watersystems

L. Ducros, F. Eyrolle-Boyer, S. Charmasson, D. Mourier

Technical Sessions: Monday, April 18

5-28
Tritium Levels in Milk in the Vicinity of Chronic Tritium Releases
Pierre Le Goff, Philippe Guetat, Laurent Vichot;
Nadine Leconte; Pierre-Marie Badot, Frédéric Gaucheron,
Michel Fromm

5-29
Exposure to Tritium: Dangerous or Not?
L. Lebaron-Jacobs, Florence Menetrier

5-30
Tritium Concentration in Carrot Plant after Short-term Exposure to Atmospheric HTO
Huifang Shen

Technical Sessions: Tuesday, April 19

6A - Tritium Interacation with Materials: Metals—I

8:00 am-10:00 am

Location: Crystal Ballroom ABC

Cochairs: Paul Korinko (SRNL), Dongxun Zhang (SINAP), G. N. Luo (Inst of Plasma Physics)

6A-1 8:00 am
Revealing the Behavior of Gas Species in Materials with In Situ TEM
Daniel C. Bufford, Clark S. Snow, David Robinson, Khalid Hattar

6A-2 8:20 am
Low Temperature Thermal Water Splitting Using Nanostructured Metal Oxides
George K. Larsen, Simona E. Hunyadi Murph

6A-3 8:40 am
A Framework for Evaluating the Effect of Radiation on the Diffusivity of Hydrogen Isotopes in a Metal Lattice
Bruce Schmitt, Ed Love, Kim Burns

6A-4 9:00 am
The Impact of Hydrophobicity of Stainless-Steel Surfaces on Tritium Inventories
C. Fagan, M. Sharpe, W. T. Shmayda, and W. U. Schröder

6A-5 9:20 am
Characterization of Tritium Permeation Profiles via Helium-3 Analyses
Walter G. Luscher, David J. Senor

6A-6 9:40 am
Overview of Fuel Retention in the JET ITER-Like Wall
A. Widdowson, E. Alves, A. Baron-Wiechec, N. Barradas, N. Catarino , J. P. Coad, K. Heinola, J. Likonen, S. Koivuranta, S. Krat, G. F. Matthews, M. Mayer, P. Petersson and M. Rubel and JET Contributors

6B - Tritium Decontamination and Waste Management

8:00 am-10:00 am

Location: Crystal Ballroom DEF

Cochairs: D. Canas (CEA), V. Fugaru (Horia Hulubei Natl Inst for Physics and Nuclear Eng) C.R. Shmayda (Univ of Rochester)

6B-1 8:00 am
Decontamination of Proton Exchange Membranes After Tritium Exposure
Craig Muirhead, Hongqiang Li, Melissa Byers, Rob Carson, Hugh Boniface, Sam Suppiah

6B-2 8:20 am
Comparison of Solutions to Reduce the Tritium Inventory in Purely Tritiated Metallic Waste
C. Decanis, M. Kresina, D. Canas, J. Pamela

6B-3 8:40 am
Developments in Robust Radioanalytical Techniques for the Determination of ^{3}H in Decommissioning Wastes and Environmental Matrices
Ian Croudace, Phil Warwick, Richard Marsh

6B-4 9:00 am
H-3 Measurement in Radioactive Wastes: Efficiency of the Pyrolysis Method to Extract Tritium from Aqueous Effluent, Oil and Concrete
R. Brennetot, M. Giuliani, S. Guegan, P. Deloffre, P. Fichet, C. Mougel

6B-5 9:20 am
Tritium Recovery from Mixed Waste
C. R. Shmayda, J. Cruz, W. T. Shmayda

6B-6 9:40 am
Leaching Tests of the Tritium Wastes Immobilized in New Cement Mixtures
V Fugaru, C Postolache, Maria Gheorghe, Lidia Radu, Nastasia Saca

Technical Sessions: Tuesday, April 19

7A - Tritium Interacation with Materials: Metals—II

10:20 am-12:20 pm

Location: Crystal Ballroom ABC

Cochairs: Yuji Hatano (Univ of Toyama), Mike Morgan (SRNL), Takuji Oda (Seoul Natl Univ)

7A-1 10:20 am
Influence of Surface Roughness and Gold Plating on the Adsorption and Absorption of Tritium into Stainless Steel (Type 316)
M. Sharpe, C. Fagan, W. T. Shmayda, and W. U. Schröder

7A-2 10:40 am
Electrochemical and Thermal Treatments for Stainless Steel Passivation
P. S. Korinko, R. B. Wyrwas, W. A. Spencer

7A-3 11:00 am
Molecular Dynamics Simulation on Hydrogen Trap Effects by Vacancy Clusters in bcc-Fe
Takuji Oda, Deqiong Zhu

7A-4 11:20 am
Effect of Tritium on Cracking Threshold in Aluminum 7075
Andrew J. Duncan, Michael J. Morgan

7A-5 11:40 am
Deuterium Diffusion and Retention Behaviors in Erbium Oxide Single Layer and Ceramic-Metal Multilayer Coatings
Takumi Chikada, Seira Horikoshi, Jumpei Mochizuki, Cui Hu, Freimut Koch, Takayuki Terai, Yasuhisa Oya

7A-6 12:00 pm
Quantitative Analysis of Deuterium Depth Profile in FeAl-Al_2O_3 Tritium Permeation Barrier Coatings and its Application in Deuterium Permeation Behaviors
Q. Zhana, H. G. Yanga, Y. Hatanob, X. M. Yuana, X. X. Zhua, X. C. Guoa

7B - Tritium Processing: Use of Zeolites

10:20 am-12:20 pm

Location: Crystal Ballroom DEF

Cochairs: Caroline Chambelland (CEA), David James (SRNL), Rodrigo Antunes (KIT)

7B-1 10:20 am
Zeolite Membranes for Tritium Processing: Preliminary Inactive Results and CAPER Upgrade for Separation Experiments with Tritium
Rodrigo Antunes, Olga Borisevich, David Demange, Nando Gramlich, Thanh-Long Le, Alejandro Ovalle

7B-2 10:40 am
Deuterium Depleted Water from TCAP R&D Using H_2/D_2 Feed
Xin Xiao, Henry T. Sessions, Ben Randall, Lucas Angelette, Michael Brown, Dave Babineau

7B-3 11:00 am
Experimental and Simulated Hydrogen Isotopes Adsorption Isotherms Over Zeolite Materials Under Cryogenic Conditions
M. Macaud, C. Gauvin, S. Lectez, J.M. Salazar, J.M. Simon, G. Weber, I. Bezverkhyy, J.P. Bellat

7B-4 11:20 am
Reduction of Glovebox Stripper System Water Loading
Dr. Jason Wilson, Dr. James Klein, Dr. Kirk Shanahan, Dr. Paul Korinko, and Anita Poore

7B-5 11:40 am
Hydrogen Isotopes Adsorption over Zeolite Materials: Modeling of Solid Gas Interactions by Molecular Dynamics Simulation
M. Macaud, S. Lectez, J.M. Salazar, J.M. Simon, G. Weber, I. Bezverkhyy, J.P. Bellat

7B-6 12:00 am
Quantitative Study of the Radiolytic Gas from Tritiated Water Adsorbed in Zeolite 4A
Manuel Grivet, Laëtitia Frances, Didier Ducret, Mikaël Douilly and C. Chambelland

Technical Sessions: Tuesday, April 19

8A - Gas Phase Detritiation

1:40 pm-3:00 pm

Location: Crystal Ballroom ABC

Cochairs: Aurelien Chassery (CEA), Steve Xiao (SRNL), Akira Taguchi (Univ of Toyama)

8A-1 1:40 pm
Catalyst Evaluation for Oxidative Stripper
Xin Xiao, Henry T. Sessions, Anita Poore

8A-2 2:00 pm
Hydrolysis of Tritiated Sodium: Advances in Understanding and Modelling of the Tritium Behaviour
Aurelien Chassery, Helene Lorcet, Joel Godlewski, Karine Liger, Pierre Trabuc, Christian Latge, Xavier Joulia

8A-3 2:20 pm
Preparation of Alveolate "Hydrophobic" Catalyst for Tritium Waste Gas Treatment
Yong Yang , Shuming Peng, Heyi Wang, Yang Du

8A-4 2:40 pm
Ce Based Oxide Loaded Honeycomb Catalyst for Detritiation
Quanwen Wu, Wenhua Luo, Daqiao Meng, Jingwen Ba

8B - Tritium Storage

1:40 pm-3:00 pm

Location: Crystal Ballroom DEF

Cochairs: Sei-Hun Yun (NFRI), Katie J. Heroux (SRNL), Kirk L. Shanahan (SRNL)

8B-1 1:40 pm
Basic Absorption/Desorption Experiment on Depleted Uranium Bed for Tritium Storage
Hyun-goo Kang, Dong-you Chung, Yun Hee Oh, Min Ho Chang, Sei-Hun Yun, Kyu-Min Song, Hongsuk Chung

8B-2 2:00 pm
Measurement of Uranium Hydride Storage Bed Engineering Parameters
P. J. Foster, R. S. Willms, W. K. Hollis, D. Dogruel

8B-3 2:20 pm
Hydriding-Induced Wall Stress Evaluation on a Four-Inch SHort (FISH) Tritium Hydride Bed
Katie J. Heroux, Edwin G. Estochen

8B-4 2:40 pm
Optimal Design and Fuel Inventory of the Multi-Bed Storage System of the SDS Considering Start-up and Shut-down Operation of the Tokamak
Jae-Uk Lee, Min Ho Chang, Sei-Hun Yun, Jin-Kuk Ha, Euy Soo Lee,In-Beum Lee, Kun-Hong Lee

9A - Tritium Facilities: Operations and Maintenance

3:20 pm-5:00 pm

Location: Crystal Ballroom ABC

Cochairs: Lee C. Cadwallader (INL), A. Morono (CIEMAT), Benjamin Hufer (SRS), D. Holunga (LLNC)

9A-1 3:20 pm
Operational Maintenance Philosophy for Tritium Processing Systems
Melissa Golyski

9A-2 3:40 pm
Management of Tritium at the National Ignition Facility
P. Epperson , K. Kasper, T. Kohut, R. Beale, S. Moyle, M. Castro, R. Thacker, D. Nelson, K. Coffee, D. Holunga, S. Brereton

9A-3 4:00 pm
Tritium Plasma Experiment Upgrade for Enhancing Tritium PMI Science
M. Shimada, C.N. Taylor, R.J. Pawelko, L.C. Cadwallader, B.J. Merrill

9A-4 4:20 pm
Recent Upgrades at the Safety and Tritium Applied Research Facility
Lee Cadwallader, Brad Merrill, Dean Stewart, L. Shayne Loftus

9A-5 3:20 pm
Direct Gas-Filling Technology Based on Thermal Gradient for Deuterium-Tritium Cryogenic Target with Micro Fill-Tube
Yu Ming-ming, Chen Shao-hua, Li Hai-rong, Wen Chen-wei, Xia Li-dong, Yin Jian, Wang Wei-wei, Huang Xin, Chen Xiao-hua, Zhou Xiao-song, Peng Shu-ming

9B - Tritium Processing, Purification, and Separations

3:20 pm-5:00 pm

Location: Crystal Ballroom DEF

Cochairs: Gregg A. Morgan (SRNL), Stefen Welte (KIT), Marius Valentin Zamfirache (Natl R&D Inst for Cryogenics & Isotopes Tech)

9B-1 3:20 pm
Direct Decomposition Processing of Tritiated Methane by Helium RF Plasma
Kazunari Katayama, Satoshi Fukada

9B-2 3:40 pm
Short Way Separation of D/T from He with Superpermeable Membranes in the Post-ITER Devices
Alexander Livshits, Arkady A. Yuchimchuk

9B-3 4:00 pm
Evaluation of the Effects of Impurities on SAES® ST198 Hydrogen Gettering
Gregg A. Morgan, David W. James

9B-4 4:20 pm
Experimental Performance Test of Key Components of the KATRIN Outer Tritium Loop
S. Welte, S. Fischer, M. Sturm, N. Tuchscherer, L. T. Le

9B-5 4:40 pm
A Review of Separation of Hydrogen Isotopes by Cryogenic Distillation in CAEP
Xia Xiulong

10 - Poster Session 2

5:00 pm-6:20 pm

Location: Crystal Ballroom DEF

Cochairs: Jason Wilson (SRNL), Mark Wittman (Univ of Rochester), Fanny Derasse (CEA)

Tritium Facilities and Operations

10-01
H/T Isotopic Exchange: Free Water vs Adsorbed Water on Zeolite
A. Lis, C. Chambelland, I. Moysana, L. Frances, F. Legoux, M. Douilly

10-02
Comparison of Gas Species Adsorptions on 4A and 13X Zeolites
C. Chambelland, K. Chesnel, A. Lis, I. Moysan, C. Gauvin, E. Schaer

10-03
Experimental Investigation of ZrCo Getter Beds as Candidate Process for the Tritium Extraction Systems of the European Test Blanket Modules
A. Bükki-Deme, P. Calderoni, D. Demange, E. Fanghänel, T.-L. Le, M. Sirch, I. Ricapito

10-04
Theoretical Considerations for Purification System used in Hydrogen Isotopes Separation Plants
Marius Zamfirache, Anisia Bornea, Ioan Stefanescu

10-05
Performance Characterization of a Pd-Ag Diffuser
Gregg A. Morgan

10-06
Formation of Carbonaceous Products under Radiolysis of Carbon Oxide in the DT-Fuel of Fusion Reactor
Alexey V.Golubev, Valentina N.Golubeva

10-07
Recovery of Low-Concentration Hydrogen Isotopes with Zr_2Fe Alloy
Xingbo Han, Yuan Qian , Wei Liu

10-08
Hydrogen Recovery from Methane and Water using Catalytic Reaction and Pd Membrane
Woo-Chan Jung, Pil-Kap Jung, Joung-Won Kim, Hung-Man Moon, Sei-Hun Yun, Hyeon-Gon Lee

10-9
Design of Cryogenic Distillation for Tritium Separation in China Fusion Engineering Test Reactor
Yuqi Yu, Yong Yao, Jiangfeng Song

10-10
Methods for Reducting and Quantifying Methane and Tritiated Methane Formation in High Pressure Hydrogen Isotope Diffusers
Peter R. Bossard, PhD, Andrew Kaldor, PhD

Tritium Interacation with Materials

10-11
Lessons from Twenty Years of Tritium Exposure to Polymers at SRS
Elise B. Fox

10-12
The Experimental Research of ^{3}He Retention Mechanisms by Aged La-Ni-Al Alloys
Binbin Liang, Weiwei Zhao, Hongguang Yang, TMT Team

10-13
Tritium Aging Effects in Some Pd-Cr, Ni, Co Alloys
Kirk L. Shanahan

10-14
Deuterium Retention in Neutron Irradiated Molybdenum
C.N. Taylor, Y. Yamauchi, M. Shimada, Y. Oya, Y. Hatano

10-15
Thermal Desorption Behavior for Helium in Aged Zirconium Tritide Films
Guijun Cheng, Xiaosong Zhou, Shuming Peng

10-16
Theoretical Calculations of the Interaction Between Hydrogen and Alloying Atom in Nickel
Wenguan Liu, Yuan Qian, Wei Liu

10 - Poster Session 2

5:00 pm-6:20 pm
Location: Crystal Ballroom DEF
Cochairs: Jason Wilson (SRNL), Mark Wittman (Univ of Rochester), Fanny Derasse (CEA)

Tritium Interacation with Materials

10-17
Lab Scaled Facilities Dedicated to the Study of Tritium Retention and Outgassing from JET Plasma Facing Components
X Lefebvre, A Hollingsworth, A Santucci, M Incelli, A Widdowson, P. Coad, R Smith, P Batistoni, N Bekris and JET contributors

10-18
Study on Radiation Effect of Poly (Vinyl Alcohol) Films Irradiated by Tritium Decay
Hairong Li, Shuming Peng, Xiaosong Zhou, Mingming Yu, Lidong Xia, Xiaohua Chen, Chenwei Wen, Shaohua Chen, Weiwei Wang

10-19
The Adsorption of Hydrogen Isotope on Graphite in Molten Salt System
Xiaoling Wu, Yuan Qian, Wei Liu

10-20
In-Situ Investigation of Deuterium Induced Corrosion of Cerium
Xiaoqiu Ye, Changan Chen

10-21
Low-Pressure and High-Temperature Tritium Behavior on Carbon
Stephen T. Lam, Ronald Ballinger, Charles Forsberg, John Stempien

10-22
Surface Treatments to Render Stainless Steel Inert for Tritium
P. S. Korinko, D. Li, W. A. Spencer

10-23
Surface Modification of ZrCo and Zr_2Fe Alloy for Enhancing the Hydriding Properties
Xiaojing Qian, Deli Luo, Chang'an Chen, Guoqiang Huang, Zhiyong Huang

10-24
Deuterium Gas Driven Permeation Behavior in W Coated V-5Cr-5Ti
Yuping Xu, Haishan Zhou, Feng Liu, Xiao-Chun Li, Ningbo Sun, Yingchun Zhang, Jing Wang, Tao Lu, Haodong Liu, Fang Ding, Guang-Nan Luo

10-25
Deuterium Retention in Deposited Tungsten Exposed in EAST
Jing Wu, Peng Wang, Li Qiao, Liang Gao, Guang-nan Luo

10-26
Tritium Aging of $LaNi_{4.15}Al_{0.85}$ (LANA.85)
David W. James, Gregory C. Staack, Simona Hunyadi-Murph

Technical Sessions: Wednesday, April 20

11A - Liquid Phase Detritiation

8:00 am-10:00 am

Location: Crystal Ballroom ABC

Cochairs: Satoshi Fukada (Kyushu Univ), Andrey V. Ovcharov (ITER), Lucas Angelette (SRNL)

11A-1 8:00 am
Preparation and Catalytic Hydrogen Isotope Exchange Activity of Porous Styrene-Divinylbenzene Polymer Supported Pt Catalyst
Akira Taguchi, Yusuke Nagaki, Takahiko Sugiyama, Masahiro Tanaka, Kenji Kotoh

11A-2 8:20 am
More Precise Values of Separation Factors in Water-Hydrogen Isotopic Exchange for Modeling of Combined Electrolysis and Catalytic Exchange Process
Andrey V. Ovcharov

11A-3 8:40 am
Aspects Concerning Manufacutre of Reproducible and Homogeneous Batches of PT/C/PTFE Catalyst for Hydrogen-Water Isotopic Exchange
Gheorghe Ionita , Gh. Titescu, I. Stefanescu, Adriana Marinoiu, Amalia Soare

11A-4 9:00 am
Tritium Water Distillation Assisted with Adsorption and Isotopic Exchange
Satoshi Fukada, Tomohiro Motomura, Satoru Hirano, Jyunji Mizutani, Yoshiaki Miho

11A-5 9:20 am
Recent Progress in the Experimental Study of LPCE Process on "EVIO" Pilot Plant
O.A. Fedorchenko, I.A. Alekseev, S.D. Bondarenko T.V. Vasyanina

11A-6 9:40 am
Rigorous Two-Fluid and Three-Fluid Liquid Phase Catalytic Exchange Models and their Application
Anthony Busigin

11B - Tritium Biological Effects

8:00 am-10:00 am

Location: Crystal Ballroom DEF

Cochairs: C. Grisolia (CEA), Greg Staack (SRNL), Carmen Varlam (ICIT)

11B-1 8:00 am
Tungsten Dust in Fusion Devices: Impact of Morphology and Characteristics of Particles on Tritium Retention/Desorption, Associated Toxicological Studies
C Grisolia

11B-2 8:20 am
Determination of *In Vitro* Lung Solubility and Intake-to-Dose Conversion Factor for Tritiated $LaNi_{4.15}Al_{0.85}$
Greg Staack, Yung-Sung Cheng, Yue Zhou, Tom LaBone

11B-3 8:40 am
HTO and OBT Exposure Induces Differential Effects on Haematopoiesis and Iron Metabolism
Jean-Marc Bertho, Dimitri Kereselidze, Line Manens, Cecile Culeux, Joel Surette, Melinda Blimkie, Lindsey Bertrand Heather Wyatt, Maâmar Souidi, Marc Benderitter, Nick Priest, Jean-René Jourdain

11B-4 9:00 am
Tritium Level Evolution in the Environment at Experimental Pilot for Tritium and Deuterium Separation -ICIT
Carmen Varlam, Ioan Stefanescu, Ionut Faurescu, Irina Vagner, Denisa Faurescu, Diana Bogdan

11B-5 9:20 am
Modeling Tritium Transfer in a Grassland Ecosystem in Response to ^{3}H Releases and Validating with Experimental Data
Séverine Le Dizès, Hugo Renard, Fabien Vermorel, Denis Maro, Céline Aulagnier, Marianne Rozet, Didier Hébert, Luc Solier

11B-6 9:40 am
Effects of Tritiated Thymidine on Neural Stem and Progenitor Cells
Granotier-Beckers C., Soussi I., F.D. Boussin

Technical Sessions: Wednesday, April 20

12A - Tritium Interacation with Materials: Reactor Materials

10:20 am-12:20 pm

Location: Crystal Ballroom ABC

Cochairs: Xavier Lefebvre (CCFE), Michael J. Morgan (SRNL), David J. Sensor (PNNL)

12A-1 10:20 am

Pebble Fabrication and Tritium Release Properties of a $Li_{2+x}TiO_{3+y}$ with $Li_2Z_rO_3$ Solid Solution

Tsuyoshi Hoshino, Saerom Kwon, Masayuki Ohta, Shinichi Urai, Yuki Edao, Kentaro Ochiai, Yoshinori Kawamura

12A-2 10:40 am

The Effect of Tritium Generation on Alloys Corrosion in Molten Li_2BeF_4 Salt

Guiqiu Zheng, David Carpenter, Michael Ames, Yakov Ostrovsky, Gordon Kohse, Lin-wen Hu

12A-3 11:00 am

Application of Tritium Imaging Plate Technique to Measuring Hydrogen Solubility and Diffusivity in $BaInO_{2.5}$

Kenichi Hashizume, Yusaku Oki

12A-4 11:20 pm

Development of H, D, T Simultaneous TDS Measurement System and H, D, T Retention Behavior for DT Gas Exposed Tungsten Installed in LHD Plasma Campaign

Yasuhisa Oya, Cui Hu, Hiroe Fujita, Kenta Yuyama, Shodai Sakurada, Yuki Uemura, Suguru Masuzaki, Masayuki Tokitani, Miyuki Yajima, Yuji Hatano, Takumi Chikada

12A-5 11:40 am

Simulation of the Migration of Tritium in Tungsten Materials in Fusion Devics (ITER and DEMO)

E. A. Hodille, C. Grisolia

12A-6 12:00 am

Tritium Behavior in the Water-Cooled Ceramic Breeder Blanket of CFETR

Guang-Nan Luo, Yuping Xu, Haishan Zhou, Feng Liu, Qiang Qi, Jing Wang

12B - Tritium Breeding and Extraction—I

10:20 am-12:20 pm

Location: Crystal Ballroom DEF

Cochairs: D.J. Senor (PNNL), Paul W. Humrickhouse (INL), Takahiko Sugiyama (Nagoya Univ)

12B-1 10:20 am

Modeling to Support In-Situ Data Interpretation in the TMIST-3 Irradiation Experiment

D.J. Senor, W.G. Luscher, K.K. Clayton

12B-2 10:40 am

Hydration--Carbonation of Gamma Lithium Aluminate

Brad Johnson, Larry Bagassen, Jarrod Crum, Carmen Rodriquez, Jian Liu, Herbert Schaef, Monte Elmore, Nathan Canfield, Walter Luscher, Dave Senor

12B-3 11:00 am

Preliminary System Modeling for the EUROfusion Water Cooled Lithium Lead Blanket

Fernando R. Urgorri, Carlos Moreno, Elisabetta Carella, Jesús Castellanos, Alessandro Del Nevo

12B-4 11:20 am

Overview of Tritium Target Program and Results of Lithium Aluminate

Ingrid Burgeson, Larry Bagaasen Dave Baldwin, Matt Edwards, Monte Elmore

12B-5 11:40 am

Tritium Permeation and Extraction in the Fusion Nuclear Science Facility

Paul W. Humrickhouse, Brad J. Merrill

12B-6 12:00 pm

Tritium Behavior in HCPB Breeder Blanket Unit: Modeling and Experiments

E. Carella, F. R.Urgorri, D. Demange, J. Castellanos, C. Moreno

Technical Sessions: Thursday, April 21

13A - Gas Phase Tritium Measurements

8:00 am-10:00 am

Location: Crystal Ballroom ABC

Cochairs: Sandra Romanelli (Culham Centre for Fusion Energy), Steven James Bell (National Physical Lab), DAvid Robinson (SNL)

13A-1 8:00 am
Compact Determination of Hydrogen Isotopes and Helium
David B. Robinson, Weifang Luo, Trevor Y. Cai, Kenneth D. Stewart

13A-2 8:20 am
Upgraded Analytical Gas Composition Technique in the Tritium Fuel Cycle of JET
S. G. Romanelli, A. Hollingsworth, A. Withycombe, P. Camp, R. Smith

13A-3 8:40 am
Design and Operation of a Monitoring System Which Separates and Measures High and Low Concentrations of Tritium in Air
Robert Goldstein, Dell Williamson

13A-4 9:00 am
An Integrated Bubbler-LSC for On-Line Measurements of Gaseous and Aqueous Tritium
Steven James Bell, Tom Deakin, Ben Russell

13A-5 9:20 am
Implementation of a New Tritium Monitor Calibration Facility
Steve Phillips, Anne Mely, Vincent Werth

13A-6 9:40 am
Tritium Impurities Measurement with Gas Chromatography
F. Bachelet, A. Fabre, D. Salmon, J.L. Brix

13B - Tritium Environmental Effects

8:00 am-10:00 am

Location: Crystal Ballroom DEF

Cochairs: Carl A. Mazzola (CB&I Federal Services), Volodymyr Y. Korolevych (CNL), Satoshi Konishi (Kyoto Univ)

13B-1 8:00 am
Field Testing and Intercomparison of Advanced Tritium Transfer Models
Volodymyr Korolevych, Sang Bog Kim, Nana-Owusua Kwamena, Masakazu Ota, Severine Le-Dizes-Maurel, Denis Maro, Celine Ailagnier, Luc Patryl

13B-2 8:20 am
Oceanic Behavior of Tritium Upon Deep-Sea Release
Shutaro Takeda, Ryuta Kasada, Fumito Okino, Shigeki Sakurai, Satoshi Konishi

13B-3 8:40 am
A Comparison of Tritium Dispersion Methodology for Accident Analysis in U.S. Department of Energy Complex Facilities
Kevin R. O'Kula, David C. Thoman

13B-4 9:00 am
Investigation of the Potential Impact of Storage Place on Tissue Free Water Tritium and Organically Bound Tritium Activity Determination. Feedback of an Interlaboratory Exercise
Nicolas Baglan, Eric Ansoborlo

13B-5 9:20 am
Relevance of Night Production of OBT in Crops
Dan Galeriu, Anca Melintescu

13B-6 9:40 am
Specific Activities of OBT in Soil around Nuclear Power Plant
Lin Du, Qin Zhang, Yu-hua Ma, Ling Wang, Lai-lai Qin, Ke Deng, Zheng-hai Xia, Wei Liu

14A - Liquid Phase Tritium Measurements

10:20 am-12:20 pm

Location: Crystal Ballroom ABC

Cochairs: Laura L. Tovo (SRNL), Sebastian Mirz (KIT), Luis Angel Sedano (Fus_Allanz S/E&C)

14A-1 10:20 am
Radioactive Characterization of Tritiated Heavy Water using ESR Spectrometry
C Postolache, D Negut, V Fugaru

14A-2 10:40 am
Design of a Spectroscopy Experiment for All Hydrogen Isotopologues in the Liquid Phase
Sebastian Mirz, Uwe Besserer, Beate Bornschein, Robin Größle, Bennet Krasch, Stefan Welte

14A-3 11:00 am
Isolating Low Concentration of Tritium in Potable Water using Simple Designed and Fabricated Process
Peter Ozemoyah, John Robinson

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14A - Liquid Phase Tritium Measurements

10:20 am-12:20 pm

Location: Crystal Ballroom ABC

Cochairs: Laura L. Tovo (SRNL), Sebastian Mirz (KIT), Luis Angel Sedano (Fus_Allanz S/E&C)

14A-4 11:20 am
Development of Fast-Response Solved-Tritium Concentration Diagnostics
L. Sedano, F. Medina, J. Abellà, J. Llorca

14A-5 11:40 am
Tritium Measurement with Organic Waste-Less Method
Etsuko Furuta, Akira Taguchi, Takuya Saze

14B - Tritium Breeding and Extraction—II

10:20 am-12:20 pm

Location: Crystal Ballroom ABC

Cochairs: Cochairs: Blair P. Bromley (CNL), Christian Day (KIT), Elisabetta Carella (CIEMAT)

14B-1 10:20 am
Overview of the European R&D on Tritium Technology for DEMO Breeding Blanket
D. Demange, R. Antunes, F. Arbeiter, L. Frances, C. Moreno, A. Morono, D. Rapisarda, A. Santucci, R. Smith, M. Utili

14B-2 10:40 am
Preparation of a Core-shell Adsorbent for Lithium Isotope Separation
Takahiko Sugiyama

14B-3 11:00 am
Lithium Enrichment Needs for Tritium Breeding
Thomas Giegerich, Christian Day

14B-4 11:20 am
Tritium Recovery Efficiency under Array of PbLi Droplets in Vacuum
Fumito Okino, Laetitia Frances, David Demange, Ryuta Kasada, Satoshi Konishi

14B-5 11:40 am
Development and Modelling of a Multi-Nozzle Vacuum Sieve Tray Extraction Facility
L. Frances, D. Demange, S. Konishi, M. Mertens, A. Munoz, F. Okino

14B-6 12:00 pm
First-Principle Study on Structures and Energetics of Intrinsic Vacancy Defects in Li_4SiO_4
Man Jiang, Mingjie Zheng, Jingping Xin, Wenyi Ding, Qunying Huang, FDS Team

15A - Future Tritium Facility Planning

1:40 pm-3:00 pm

Location: Crystal Ballroom ABC

Cochairs: Rachel Elizabeth Lawless (CCFE), Armando Antoniazzi (Kinectrics, Inc.), Ki Jung Jung (NFRI)

15A-1 1:40 pm
Tritium Plant Technology Development for a DEMO Power Plant
Rachel Lawless Barry Butler, Rebecca Shaw, Patrick Camp, Sebastian Russell-Smith, Sophie Bashforth, Christopher Timberlake, Alessia Santucci

15A-2 2:00 pm
Technology Decision Making for a Tritium Removal Facility
Stephen Strikwerda, Armando Antoniazzi

15A-3 2:20 pm
A Large Scale Demonstration Facility for Light Water Detritiation
N. Bonnet, G. Bonhomme, Y. Iwasaki, D. Carlson, J. Raymont

15A-4 2:40 pm
ITER-like Tokamak Exhaust Gases in JET Active Gas Handling System: Process Optioneering
P Camp, R.C. R. Shaw, R. Olney, R. Smith, R. Walker

15B - Tritium Pumping

1:40 pm-3:00 pm

Location: Crystal Ballroom DEF

Cochairs: Gregg A. Morgan (SRNL), Thomas Giegerich (KIT), Hongsuk Chung (KAERI)

15B-1 1:40 pm
Thermo-Mechanical Design Study of a Linear Mercury Diffusion Pump for Tritium Processing
Thomas Giegerich, Christian Day, Xueli Luo, Ralf Müller, Santiago Ochoa, Holger Strobel

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15B-2 2:00 pm
Performance Testing of a Potential Replacement for the Normatex® Pump
Gregg A. Morgan

15B-3 2:20 pm
Positive Displacement Scroll Pump Technology for Tritium Processing Systems
Bryce Shaffer, Louis Boone

15B-4 2:40 pm
Tritium Aspects of Fueling and Exhaust Pumping in Magnetic Fusion Energy
Bryce Shaffer, Louis Boone

16A - Other Tritium Topics

3:20 pm-5:00 pm
Location: Crystal Ballroom ABC

Cochairs: Lutz Bornschein (KIT), Todd Whitehome (CNL), Kanetsugu Isobe (JAERI)

16A-1 3:20 pm
Tritium Control and Capture in Salt-Cooled Fission and Fusion Reactors
Charles W. Forsberg, David M. Carpenter, Dennis G. Whyte, Raluca Scarlat, Liu Wei

16A-2 3:40 pm
Metal Tritides as Power Sources for Tritium Betavoltaics
Brian L. Ellis, N. Philippi, A. Haavisto, S. Thomson, S. Suppiah

16A-3 4:00 pm
Status of the Karlsruhe Tritium Neutrino Mass Experiment KATRIN
Lutz Bornschein, Beate Bornschein, Sylvia Ebenhöch, Moritz Hackenjos, Florian Priester, Marco Röllig and Michael Sturm for the KATRIN collaboration

16A-4 4:20 pm
Tritium Production and Partitioning from the Irradiation of Lithium-Beryllium Fluoride Salt
David Carpenter, Michael Ames, Guiqiu Zheng, Gordon Kohse, and Lin-wen Hu

16A-5 4:40 pm
R&D Progress of Tritium Technology on the Broader Approach
Kanetsugu Isobe, Makoto Oyaidzu, Takumi Suzuki, Hirofumi Nakamura, Yasunori Iwai, Masayuki Yamada, Rie Kurata, Dai Inomiya, Yukihiro Murodate, Takumi Hayashi, Toshihiko Yamanishi

16B - Tritium Confinement and Safety

3:20 pm-5:00 pm
Location: Crystal Ballroom DEF

Cochairs: Ene Daniela (ESS), Kimberly A. Burns (PNNL), Kenneth Kasper (LLNL)

16B-1 3:20 pm
Tritium at the National Ignition Facility—Where Did It Go?
Kenneth Kasper, Rick Thacker

16B-2 3:40 pm
Key Management Tritium Issues at ESS facility
Daniela Ene

16B-3 4:00 pm
Progress of ISO Standards on the Confinement Systems in Fusion Facilities using Tritium
L. Lepetit, P. Cortes, L.Thomas

16B-4 4:20 pm
The Research Progress of Tritium in HTR-10
Yi Xu, Feng Xie, Hong Li, Jianzhu Cao, Jiejuan Tong, Xuegang Liu, Xiaogui Feng

16B-5 4:40 pm
Tritium Production in Secondary Sources
Kim Burns, Ed Love

17 - Poster Session 3

5:00 pm-6:20 pm
Location: Crystal Ballroom DEF

Cochairs: Elise Fox (SRNL), Sam Suppiah (AECL), Joseph Wheeler (SRNL)

Tritium Measurement, Monitoring, and Accountancy

17-01
Tritium Measurement of Waste in Large Volume Drums
A. Godot, K. Galliez, G. Jossens, C. Mathonat, D. Demange

17-02
Large Volume Calorimeter LVC 1380 for the Tritium Measurement in Radioactive Waste Packages
K. Galliez, G. Jossens, F. Bachelet, A. Godot, C. Mathonat

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17 - Poster Session 3

5:00 pm-6:20 pm

Location: Crystal Ballroom DEF

Cochairs: Elise Fox (SRNL), Sam Suppiah (AECL), Joseph Wheeler (SRNL)

Tritium Measurement, Monitoring, and Accountancy

17-03

A Seebeck-type Approach to Hydride Bed Inventory Tracking, PseudoSeebeck Calorimetry

Kirk L. Shanahan

17-04

Speciation of Tritium with Organic Molecules in Nuclear Plant Discharges

C. Merignac, C. Landesman, I. Deniau, A. Bacchetta, H. Schneider, G. Heisbourg, M.B. Mokili, G. Montavon

17-05

Development Trials of an Acid Dissolution Line for Tritium Assay

Damaris Roffey

17-06

Overview of Tritium Monitoring Technology for Fusion Reactor Fueling Systems

Jacky M. Shoulders, Ronald E. Battle

17-07

Obtaining of Thin Layer Radioactive Surface Sources Through Radio-Induced Grafting of Labeled Monomers

C. Postolache, V. Fugaru

17-08

Tritium Counting by Europium Coordination Complex

Masanori Hara, Haruna Sakaguch, Masato Nakayama, Shinsuke Abe, Masao Matsuyama, Takayuki Abe, Tsukasa Aso

17-09

TEACUP: A Tritium Management and Supplemental Core Follow Program

Jennifer Lyons, James Livingston, Edward Love, Kimberly Burns

17-10

A New Device for Tritium Activity Measurement of Tritiated Water

Florian Priester

17-11

First Calibration of an IR Absorption Spectroscopy System for the Measurement of H2, D2 and HD Concentration in the Liquid Phase

Robin Größle, Alexander Kraus, Sebastian Mirz, Sebastian Wozniewski

17-12

Tritium Imaging Techniques for the Study of Tritium Transport in the Graphite Fuel Elements of Fluoride-Salt Cooled High-Temperature Reactors (FHRs)

Huali Wu, Quentin Deslot, Raluca O Scarlat

17-13

Performance of Inorganic Oxides for the Non-Reversible Trapping of Hydrogen: Application to Tritium Analysis

Kévin Galliez, Alain Godot, David Lambertin

17-14

Design Considerations for a Tritium Process Analytical Measurement System

Laura L Tovo, Nancy V. Halverson, Anita S. Poore

Tritium Breeding and Extraction

17-15

Study on Hydrogen Isotopes Behavior in LiPb Forced Convection Flow

Mao Kinjo, Satoshi Fukada, Ryosuke Yoshimura, Taiki Muneoka, Kazunari Katayama

17-16

The Effect of MHD Mixed Convection on Tritium Transport in Fusion-Relevant Configurations

Hongjie Zhang, Alice Ying, Mohamed Abdou

17-17

Tritium Contamination Prevention Using Sacrificial Materials

Paul Korinko, Simona Hunyadi Murph, George Larsen

Tritium Breeding and Extraction

17-18
Study on the Gas Extraction in Molten Salt Reactor for Tritium Control
Hua Li, Ling Wang, Yuan Qian, Wei Liu

17-19
Chemical Kinetics Study of Reactions Between HTO and H2 in Purge Gas Flow in Tritium Breeding Blanket
Michiko Ahn Furudate, Seungyon Cho

17-20
Removal of Low-Concentration Deuterium from Fluidized Li Loop for IFMIF
Yushin Yamasaki, Satoshi Fukada, Kazuma Hiyane, Kazunari Katayama

17-21
The System Design of Tritium Extraction From Ceramic Breeder Material With H_2O Added In Helium Purge Gas
Y. Yao, D. L. Luo, R. J. Xiong, Y. Q. Yu, J. F. Song

17-22
Tritium Extraction from Lithium-Lead EU DEMO Blanket using Permeator Against Vacuum
V. D'Auria, S. Dulla, P. Ravetto, L. Savoldi, M. Utili, R. Zanino

17-23
Investigation of Correlation Between Irradiation Defects and Deuterium Release
Qiang Qi, Jing Wang, Guangnan Luo

17-24
Tritium Production Assessment for the DCLL EUROfusion DEMO
Iole Palermo, D. Rapisarda, I. Fernández, C. Moreno, F.R. Urgorri, A. Ibarra

Technical Sessions: Friday, April 22

18 - Near-Term, New Tritium Capabilities

8:00 am-9:40 am

Location: Crystal Ballroom ABC

Cochairs: : David W. Babineau (SRNL), Satoshi Konishi (Kyoto Univ), Hugh Boniface (CNL)

18A-1 8:00 am
Isotope Separation System at the University of Rochester's Laboratory for Laser Energetics
M. D. Wittman, W. T. Shmayda, J. L. Reid, and R. F. Earley

18A-2 8:20 am
Heat Transfer Experimental Verification of a High Power Tritium-Titanium Rotating Target Prototype
Qianfeng Yu, Gang Wang, Xiang Ji, Wen Wang, FDS Team

18A-3 8:40 am
Preliminary Designing and R&D Progresses of Fuel Cycling System for CFETR
Heyi Wang, Shuming Peng, Xiaolin Wang

18A-4 9:00 am
Tritium Facilities in JAEA and Future Plan
Kanetsugu Isobe, Takumi Hayashi

18A-5 9:20 am
Renewing the Tritium Facilities at Chalk River
Hugh Boniface, Kevin McCrimmon, Don Ryland, Sam Suppiah

19 - Panel Discussion: Progress in Tritium Regulations and Standards

10:00 am-11:00 am

Location: Crystal Ballroom ABC

Cochairs: William W. Weaver (DOE), Donald Ryland (AECL), Antonio Provenzano (AWE Plc)

19-1 10:00 am
Progress in Tritium Regulations and Standards
Panelist: William W. Weaver (DOE),
Antonio (Tony) Provenzano (AWE Plc), Loic Lepetit (ITER)

20 Closing Ceremony
11:00 am-12:00 pm
Robert P. Addis (SRNL), Steve Xiao (SRNL),
James E. Klein (SRNL)

Preparation of and Upgrades to the Active Gas Handling System for the Planned JET TT and DT Campaigns

R Smith[1] plus contributors

[1]UKAEA, Culham Centre for Fusion Energy, Abingdon, UK, OX143DB, robert.smith@ccfe.ac.uk

The planned TT and DT campaigns on JET are the largest Fusion Tritium experiments yet performed and the only large scale campaigns undertaken since 1997. During DTE1 100g of Tritium was fed to the JET tokomak by the AGHS from an inventory of 20g whilst for the TT and DT campaigns foreseen 1000g of Tritium will be supplied to JET from the AGHS from an inventory of 55g. This represents a 10 fold increase in Tritium feed and nearly a 4 fold increase in the number of times the AGHS Tritium inventory is recycled through the plant.

Significant upgrades and modifications to, and revalidation of, the existing AGHS are required to meet the demands of such campaigns and below are some of the subjects to be discussed;

- The upgraded AGHS calorimeter and the receipt and accounting of the Tritium deliveries required for the JET Tritium campaigns.
- The design, construction and operation of a JET Water Detritiation System
- A new AGHS/JET Tritium accountancy system based on a solid state detector system optimized for use in the sub mbar pressure range.
- The use of mechanical cryo-coolers to provide a backup pumping capability for Tritium recovery from the JET machine.
- Upgrades to the Gas Introduction and Delivery system to facility the Tritium feed rates required for the upcoming Tritium campaigns
- A review of the aging effects and problems faced in operating an aging AGHS.
- Review of maintenance and operational issues.
- Upgraded analytical systems.
- An improved Impurity Processing system.

This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the EURATOM research and training programme 2014-2018 under grant agreement No. 633053 and from the RCUK Energy Programme [grant number EP/I501045]. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

Paper 1-3: Monday, April 18

Tritium Activities at the University of Rochester's Laboratory for Laser Energetics

W. T. Shmayda, M. D. Wittman, J. L. Reid, and R. F. Earley

Laboratory for Laser Energetics, University of Rochester, Rochester, NY, wshm@lle.rochester.edu

The Laboratory for Laser Energetics (LLE) routinely fields 860-μm diam thin-walled plastic shells filled with deuterium–tritium (DT) gas or DT ice layered on the inner surface of the shells to study inertial confinement fusion. The multistep filling operation involves desorbing the DT fuel from uranium storage beds, measuring the tritium inventory, and removing decay helium-3 from the DT fuel prior to filling the evacuated shells.

Warm targets are filled with DT gas by gradually compressing the DT to pressures in the range of 10 to 30 bar in the presence of the shells. An aluminum coating on the outside of the shells extends the permeation time constant into the tens-of-hours range. At the end of the pressurizing cycle, the DT is returned to the uranium storage beds. The filled targets are transferred to cold storage until required for experiments.

Cryogenic targets are filled in two steps: first, the gas pressure is compressed from 1 bar to ~120 bar, and subsequently-with a second-stage compressor to pressures in the range of 500 to 700 bar depending on the final ice thickness required. At the end of the pressurization stage, the targets are cooled to 15 K so that the DT gas that has permeated into the shells can condense on the inner surface of the plastic spheres. The gas outside the shells is returned to the storage beds. From this moment on, these targets are maintained at temperatures between 17 and 40 K until they are used to study implosion of DT ice driven by the laser system.

LLE has expanded its infrastructure to reduce emissions, adjust the isotopic ratio of the DT fuel supply, nondestructively measure the gas pressure inside warm targets prior to shot time, and study the properties of DT ice. This paper will outline the key steps used to fill and handle targets with DT fuel up to shot time; summarize the annual emissions from the facility; and highlight the key features of the Isotopic Separation System, the Bremsstrahlung detection system to measure DT gas pressure in plastic targets, and the optical and x-ray diagnostic station used to investigate the properties of DT ice.

This material is based upon work supported by the Department of Energy National Nuclear Security Administration under Award Number DE-NA0001944, the University of Rochester, and the New York State Energy Research and Development Authority. The support of DOE does not constitute an endorsement by DOE of the views expressed in this article.

Fifteen years of Operating in a Tritiated Waste Treatment Nuclear Facility

Christophe Douche, S. Rochefort, J. Avenet, A. Arseguel, O. Leagaie

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Waste contaminated by tritium are produced during operation and decommissioning in tritium facilities. A new nuclear facility was commissioned for the processing of tritiated waste. This paper intends to summarize the experience of CEA in tritiated waste strategy.

Before reprocessing, waste are separated by compounds (metallic, plastic, glass…) and first wall waste must be treated to limit tritium inventory and tritium outgassing.

Various detritiation processes are carried out in glove boxes to limit environmental impact of tritium:

- melting (1600°C) for metal waste to produce ingots,
- steaming (100°C) organic waste,
- tritiated water adsorption on zeolithe…

In addition to reduce releases to the environment, gloves boxes atmospher are decontaminated by a detritiation unit (tritium oxydation and water adsorption on zeolithe).

Waste packaging in metalic drums to take into account tritiated materials are performed before interim storage on site in dedicated buildings to enable the tritium decay (50 years) before final disposal.

Research Plan of Fusion Fuel Cycle and Related Tritium Science in Korea

Ki Jung Jung[1]*, Sei-Hun Yun[1], Hongsuk Chung[2], Kyu-Min Song[3]

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[2] *Korea Atomic Energy Research Institute, Daejeon, Korea*

[3] *Korea Hydro and Nuclear Power CRI, Daejeon, Korea*

Korea has focused on the tritium technology development since joining for the ITER project. WTRF is one nuclear power plant that has the tritium removal facility in Korea, hence every year approximately 500 g of tritium is being stored. Growing up of research in tritium application and of interest on the fusion fuel cycle is to be requested for near future: The process study on tritium recovery from the TRF and the successive separation of He-3 is being considered in the KHNP (Korea Hydro and Nuclear Power) and an applicable facility is under construction. NFRI is also studying on the construction of fuel cycle process research. So, collaboration and cooperative studies between NFRI, KHNP CRI and KAERI are to be taken into account gradually in Korea. This paper deals with the current status of research plan on the fusion fuel cycle and study plan on the related tritium science in Korea.

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Paper 2-3: Monday, April 18

Review of the TLK Activities Related to Water Detritiation, Isotope Separation Based on Cryogenic Distillation and Development of Barriers Against Tritium Permeation

Ion Cristescu, A. Bükki-Deme, H. Dittrich, N. Gramlich, R. Grösle, F. Krammer, N. Lohr, C. Melzer, F. Rehlinghaus, P. Schäfer, Stefan Welte

Karlsruhe Institute for Technology, Institute for Technical Physics, Tritium Laboratory Karlsruhe, Herrmann von-Helmholtz-Platz 1, D-76344 Eggenstein - Leopoldshafen, Germany, ion.cristescu@kit.edu

The design of ITER tritium processing systems shall benefit from experimental data and process validation based on experimental facilities that are ITER scale relevant. Several rigs and experimental facilities have been enhanced and developed at the Tritium Laboratory Karlsruhe (TLK) in order to explore a wide range of envisaged scenarios of tritium plant systems, such as the Water Detritiation System (WDS), Isotope Separation System (ISS) and highly tritiated water processing. In the last years, detailed experimental investigations and process modeling have been conducted in relation to the Combined Electrolysis Catalytic Exchange and Isotope Separation (CECE-ISS) system which were focused on evaluation of the impact of deuterium build-up and accumulation in the CECE system. An enhanced configuration of the ITER WDS has been developed, that allows mitigation of the effects due to deuterium accumulation and reduction of the tritium inventory within the electrolysis system. In addition, the benefits concerning the interface between the WDS and ISS will be presented.

Significant efforts have been made to enhance the simulation tool TRIMO++ that was calibrated against the experimental results collected from the experimental rigs. The new features of the simulation tools will be introduced as well.

The main references of a new method aiming to mitigate the tritium permeation from the tritium processes streams into the non-contaminated streams such as steam generators will be introduced. The reference configuration of first phase of the experimental rigs and the preliminary experimental results will be as well presented

Paper 2-4: Monday, April 18

Overview of Fusion-Related Tritium Research in Japanese Universities

Yuji Hatano[1], Yasuhisa Oya[2], Satoshi Konishi[3], Satoshi Fukada[4], Akio Sagara[5]

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[3]*Kyoto University, Uji 611-0011, Japan*

[4]*Kyushu University, Kasuga 816-8580, Fukuoka, Japan*

[5]*National Institute for Fusion Science, Toki 509-5292, Japan*

The objective of this presentation is to report the current status of research activities in Japanese universities on tritium (T) science and technology in the field of fusion engineering. Four major collaboration programs are currently organized in this field: the Fusion Engineering Research Project in National Institute for Fusion Science (NIFS), Bilateral Collaboration Research between NIFS and Hydrogen Isotope Research Center (HRC), U. Toyama, the Japan-US Joint Research Project PHENIX and the Broader Approach (BA) Activities.
The Fusion Engineering Research Project includes the development of tritium handling and safety technologies, such as those for tritium decontamination and tritium removal. High sensitivity tritium detectors are also under development for environmental safety.

Under the Bilateral Collaboration Research, HRC is open to Japanese and overseas researchers as a joint-use program of NIFS. In each fiscal year, 10–15 research groups visit HRC to perform T experiments. Major research topics are T retention in and release from plasma-facing materials and development of T measurement techniques. Retention of T in tungsten (W) has been examined after surface modifications by He implantation and plasma exposure in fusion devices (the Large Helical Device in NIFS and a spherical tokamak QUEST in Kyushu U.), and the effects of He bubble formation and impurity deposition have been discussed.

The objective of T study in the PHENIX project is to understand the influence of neutron irradiation on retention and permeation of T in W and W alloys. Neutron irradiation will be performed at 500–1200 ˚C to 1–1.5 dpa in the High Flux Isotope Reactor, Oak Ridge National Laboratory. T retention and permeation will be examined in Idaho National Laboratory using a linear plasma machine TPE and a gas permeation device TGAP. As a related activity, transport of T and D in W irradiated with 14 MeV neutrons have been examined in Shizuoka U. Significant trapping effects have been observed despite relatively low damage level (~10-6 dpa).

The BA Activities is the Japan-EU collaboration program, and researches in Japan are organized by Japan Atomic Energy Agency. Japanese universities have contributed to researches on tritium-material interactions and tritium measurements.

Hydrogen isotope transport in solid and liquid T breeders and breeding blanket systems have been examined in U. Tokyo, Shizuoka U., Kyoto U. and Kyushu U. Enhanced deuterium release from a liquid Li-Pb droplet due to oscillation has been found in Kyoto U. The development of permeation barrier coating has been performed in U. Tokyo and Shizuoka U. The performance of a high temperature gas reactor as a T production device has been examined in Kyushu U. The penetration and release of T into/from concrete and soil have also been examined in Kyushu U. for safety assessment of fusion power plants. Social influence of T release from fusion power plants have been studied in Kyoto U.

Paper 3-1: Monday, April 18

Development of Tritium Science and Technology in INPC

Shuming Peng, Xiaosong Zhou, Xiaojun Chen, Zhilin Chen, Heyi Wang

Institute of Nuclear Physics and Chemistry, China Academy of Engineering Physics, Mianyang, 621900, China

Tritium plays an important role in the fusion energy with the ITER coming into construction and China planning to build the CFETR reactor. Tritium is the hydrogen isotope that is the smallest atom. It can enter all kinds of materials because of its special physical properties. Which leads to great challenge to be limited to a certain space for the radioactivity demand. Tritium science and technology research work has been started since last century. Under the financial support of fusion-fission reactor project, INPC has a wide experience in tritium science and technology research work including in pile tritium extraction, a series tritium technology development and tritium safety. Concept design of D-T fuel cycle of CFETR and Z-pich driven fusion-fission reactor has been promoted.

Theoretical and experimental research of the properties of metal tritides and the behavior of helium in metal tritides were performed systemically. The first-principles calculations have been performed to investigate the relative stability of the various phases of zirconium hydrides combined with the special quasirandom structures (SQS) method. The thermodynamic, dynamical, electrical, thermal properties and isotope effect of metal hydrides, such as Ti, Zr, Er, Sc, U et al., have been studied systemically. Effects of tritium content on crystal lattice, ^{3}He retention and structure evolution during aging of tritides were investigated.

Theory based on a 1D model has been established to calculated energy deposition in ionization chambers with size larger than the range of beta rays quantitatively, while Monte-Carlo method was employed in the calculations for chambers with small size. A heat-resistant hydrophobic zeolite molecular sieve has been synthesized on the surface of an alveolate cordierite by employing hydrothermal processing to obtain high flow rate gas detritiation catalyst. Effect of tritium on human peripheral blood lymphocytes has been investigated, and results indicated that comparing with gamma ray of Co-60, tritium might lead to more serious damage.

In this talk, we will review the recent research work and tritium activities under the financial support of the national fusion energy in INPC, which is aim to design and construct the tritium plant of CFETR.

Cernavoda Tritium Removal Facility—Evolution in TRF Design

L. Stefan[1], N. Trantea[2], A. Roberts[3], S. Strikwerda[4], A. Antoniazzi[5], D. Zaharia[6]

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[5]*Armando Antoniazzi, Kinectrics Inc., Toronto, Canada, Armando.Antoniazzi@Kinectrics.com*

[6]*Danut Zaharia, Kinectrics Inc., Cernavoda, Romania, Danut.Zaharia@Kinectrics.com*

ICSI has recently completed the conceptual design of the Cernavoda Tritium Removal Facility (CTRF). CTRF is sized to process heavy water from 2 CANDU reactors, treating 40 kg/h of 10-54 Ci/kg heavy water over 40 years. CTRF removes tritium using liquid phase catalytic exchange (LPCE) paired with cryogenic distillation (CD). The CTRF design has implemented improvements based on design and operational knowledge from DTRF, WTRF, ICSI pilot plant, other tritium laboratories, and industry. Additionally, there are site, client, and regulatory requirements that have imposed differences from other TRF designs. This paper identifies the key improvements and requirements, explains the rationale for the design choice and highlights drawbacks. The key improvements and requirements, grouped under four categories, include:

- *Safety* – a Safe Shutdown State, higher seismic qualifications, restrictions on D2O transfers, extensive use of double containment.
- *Core Systems* - use of a mixed catalyst bed for the LPCE, no catalytic oxidation skid,helium refrigeration system cooling of the cryoadsorbers, better control of CD cascade by using pumps on reverse flows, and the use of a CuO reactor with molecular sieves dryers for cleanup of tritium in glovebox atmospheres
- *Site, client and regulatory requirements* – lower worker dose limits, independent utilities from nuclear Units 1 and 2, different targets for environmental releases and management of external hazards, and the application of the latest reactor grade Regulatory Standards in force in Romania.
- *Auxiliary systems, utilities, and the building* – removal of H_2-O_2 recombiner catalyst from the Air Detritiation System, use of a PEM electrolytic cell for D_2 makeup, and no need for steam in the CTRF facility.

The Five Phases—The Way to Standard Tritium Operation of KATRIN

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The most sensitive way to determine the neutrino mass is to measure the shape of the tritium beta spectrum near its kinematic endpoint. Within an international collaboration the KArlsruhe TRItium Neutrino experiment KATRIN (Fig. 1) is currently being built up on site at KIT thereby making use of the unique expertise of the Tritium Laboratory Karlsruhe (TLK). After 3 years of data taking KATRIN will determine the neutrino mass with a sensitivity of 0.2 eV (90% CL).

KATRIN requires a strong windowless gaseous source of almost pure molecular tritium (95%) and a throughput of 40 g tritium (1.5E16 Bq) per day stabilized to the 0.1% level (Standard Tritium Operation). These requirements can only be met by using a closed tritium loop for stabilized injection, and for withdrawing a small tritium fraction (<5 %) into an outer loop for clean-up. This loop is almost identical with the closed tritium cycle of the TLK infrastructure [2].

Since the last large components (WGTS, CPS) have been delivered in summer 2015, the work is now focused on the commissioning of the whole KATRIN experiment. A particular challenge is the commissioning with tritium, which will mark the point of no return regarding the contamination of our large magnet cryostats, pumps and loops. Technical problems arising after tritium commissioning of those components would imply serious consequences: repair work would either be done under difficult circumstances or would even be impossible. For that reason, we have developed a 5-Phase scenario covering all necessary work to be done to come to the standard tritium operation of KATRIN in a safe and reliable way.

This conference contribution will describe our 5-Phase scenario in detail.

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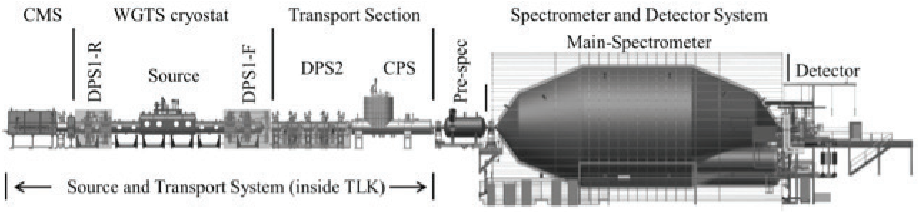

Fig. 1: Sketch of the KATRIN experiment (about 70 m): Electrons from tritium βdecay will leave the WGTS cryostat and will be guided by magnetic fields through the transport section, while the remaining tritium gas will be removed by differential and cryogenic pumping (DPS, CPS). Precise energy analysis and subsequent detection of the electrons will be done in the spectrometer and detector section.

Water Detritiation System for ITER—Evaluation of Design Parameters

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The Water Detritiation System (WDS) designed for ITER is based on the combined electrolysis and catalytic exchange (CECE) process to ensure the emission of tritium to the environment is maintained below very strict limits. The CECE process is one of the processes for tritium removal that CNL (Canadian Nuclear Laboratories, formerly Atomic Energy of Canada Ltd.) has studied, developed and successfully demonstrated. In this work, CNL evaluated ITER's design conditions of the exchange column and the electrolyser—the two key components of the CECE process (and the ITER WDS system)—to assess the effectiveness of tritium removal and investigate options to improve it. The evaluation was done using CNL's CECE process model according to a protocol set out by ITER. Initially, calibration (benchmarking) of CNL's hydrogen-water exchange column model was performed with a standard data set for a specified column to determine modeling parameters that result in a good match to the data. The calibrated model was then applied to the current WDS design. Some optimized conditions for the CECE process that could improve performance of the WDS to meet its design criteria were determined. The details of some of these assessments are presented here with particular attention to the WDS case where the feed water contains high levels of deuterium.

Tritium and ^{3}He in $ErT_{2-x}He_x$

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$ErT_{2-x}He_x$ samples have been studied over a period of three years using a variety of experimental techniques including X-Ray Diffraction (XRD), Transmission Electron Microscopy (TEM), and Nano-Indentation (NI). Complimentary experimental techniques like Neutron Diffraction and gas release measurements combined with atomistic simulations have presented a very complete picture of the evolution of helium in $ErT_{2-x}He_x$ samples. ErT_2 is a cubic crystal (#225) where helium builds up in the crystal from the decay of tritium into ^{3}He. The age of a metal tritide is conveniently expressed as a ratio of the amount of helium to metal in a sample.

Helium bubbles have been observed in $ErT_{2-x}He_x$ to form as plate-like inclusions and to grow along the four {111} planes. The helium bubbles grow in three distinct phases. First, the bubbles nucleate and grow as "Griffith-cracks" until an age of ~0.15 He:M. Second, around 0.15 He:M the diameter stops increasing and instead the bubbles grow in thickness by punching dislocation dipoles. Third, the bubbles grow in size until ~0.3 He:M at which point the bubbles begin to link causing a catastrophic helium release event.

Atomistic calculations using Density Functional Theory (DFT) provide a unique perspective and understanding of tritium and helium movement and site stability in $ErT_{2-x}He_x$. The calculations show that helium migrates via a concerted motion from a filled octahedral to a filled tetrahedral position and then to an unoccupied octahedral site.

Paper 4A-2: Monday, April 18

^{3}He release, ^{3}He Retention and Structural Evolution in Ti, Er and Zr Tritides: Tritium Content, Phase, Microscopic Structure and Aging Effects

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Long-term storage of the hydrogen isotope deuterium (D) and tritium (T) is important for nuclear applications, including next-generation thermo-nuclear energy production and neutron generators. Many transition and rare earth metals are being applied for the long-term solid-state storage of hydrogen isotopes. Titanium(Ti), erbium (Er), and zirconium (Zr) are of practical interest, since they may be used for prolonged and safe storage of hydrogen isotopes containing tritium.

Tritium decays with a 12.32 year half-life into ^{3}He, most of which is retained in the tritides. Only a small quantity of ^{3}He, typically 0.01-0.1%, was released from the tritides. Even though the ^{3}He can be kept in the tritides for several years, most of the generated helium could be released from the tritides when the atomic ratio of ^{3}He to metal in the tritides, i.e., ^{3}He/M, reaches to a critical value. The sudden release of retained ^{3}He could cause disasters due to integrity failure of the pressure vessel, crucial for the life-time of neutron generator. Therefore, this ‘aging’ phenomenon has received significant scientific interests.

Effects of tritium content, phase, microscopic structure changes on ^{3}He release/retention and crystal lattice evolution during aging of Ti, Er and Zr tritide films were investigated using a thermal desorption technique, X-ray diffraction and analysis of 3He release. Results showed that The rapid ^{3}He release values of these three kinds of metal tritides are different, the value from higher to lower are fct zirconium tritide (0.50~0.53 ^{3}He/Zr), fcc erbium tritide (~0.32 ^{3}He/Er), titanium tritide (0.25~0.28 ^{3}He/Ti). Initial tritium content in the Ti tritides showed significant effects on ^{3}He retention due to the existence of α+δ phase boundaries and lattice symmetry in the Ti tritide. The contents of α phase and γ phase in the Er tritide films showed significant different effects on 3He release/retention. The initial tritium stoichiometry or excess tritium atoms accommodated in the octahedral sites and the microstructure (i.e., the texture and Er_2O_3 oxide inclusions) played an important role for the ^{3}He release and the evolution of ^{3}He bubbles in the βphase Er tritide films. Desorption profiles of the tritium and helium varied significantly with the evolution of ^{3}He contents. The amounts of helium desorption in different stages were estimated, and the dissociation energy of helium from different trap states as a function of the aging duration was obtained. An effective helium diffusivities of $0.04\text{-}2.39\times10^{-20}$ m-2/s in fct zirconium tritide, $0.36\text{-}1.51\times10^{-20}$ m-2/s in fcc titanium tritide and $0.88\text{-}31.39\times10^{-19}$ m^{-2}/s in fcc erbium tritide at 300 K have been found by helium release on constant temperature. Variation of crystal lattice structures of α+δ, δ and ε phases in the Ti tritides, ß and ß +γ phases in the Er tritides and ε phases in the Zr tritides has been investigated, and evolution of ^{3}He during aging is mainly governed by the finite defects of self-interstitial atoms, dislocation loops, ^{3}He bubbles, and dislocations created by formation and growth of ^{3}He bubbles. The effect of microscopic structure on helium retention has been revealed. Results showed that small grain size tritides with strong fiber texture would lead to lower the ^{3}He/M value of rapid ^{3}He release and larger relative lattice parameter change.

Release of Radiogenic Helium-3 from Uranium Tritide

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Uranium is a commonly used tritium storage material owing to a low dissociation pressure of UT_3 at ambient temperature but a high dissociation pressure at reasonably low temperatures (e.g. 1 bar at ca 406 °C). Tritium undergoes a rapid decay to ^3He with a half life of *ca* 12 y resulting in tritium storage materials accumulating significant quantities of ^3He. To ensure efficient uptake of tritium, storage beds typically contain an excess of uranium which means in practice the uranium is only ever partially loaded, i.e. UH_{3-x} where x is related to the excess uranium. Understanding the ^3He release behavior from uranium tritide beds of varying loading proportion and isotopic composition as a function of time is of fundamental importance to understanding the through life behavior. This work details the accumulation and release behavior of ^3He from five uranium tritide test beds over a five year period.

Test beds containing 100 g of finely divided depleted uranium were prepared by the concomitant absorption of tritium, deuterium and hydrogen at ambient temperature at initial pressures of 700 ±80 mbar. Following absorption, the head space of the beds was evacuated to a vacuum of $<1x10^{-2}$ mbar and then sealed. Storage and surveillance of the test beds was conducted in a nitrogen glove box at an average temperature of 25 °C. The finely divided depleted uranium was prepared by the absorption (700 mbar at 200 °C) and subsequent desorption ($<1x10^{-2}$ mbar at 360 °C) of deuterium.

The ^3He released from each test bed was determined using the ideal gas law. Periodically the ^3He accumulated in the head space was transferred to a calibrated volume using a gas process line and the pressure and temperature measured. The pressure of ^3He remaining in the head space following each measurement was $<1x10^{-2}$ mbar.

The volume of ^3He released as a function of time from the five test beds did not show a trend with respect to the tritium content. However, normalising the release with respect the total absorbed hydrogen isotope content enabled the data sets to converge to a common trend, which suggests the release mechanism is not affected by loading proportion or isotopic composition. It is concluded that ^3He is accumulated within the uranium that reacted during the loading and the presence of hydrogen and deuterium influences the release.

Effects of Grain Morphology on Aging Properties of Erbium Tritide Films

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Rare earth films are widely used as hydrogen storage material for scientific and industrial applications. To provide for increased ^{3}He retention and lifetimes of metal hydride film for these applications, the erbium hydride film fabrication processes have been developed. The erbium film is evaporated onto the molybdenum substrate in an ultra high vacuum system equipped with an electron beam deposition gun. The erbium film loading tritium in another ultra high vacuum system with the using of temperature programmed process. To form an expected phase structure, the exact temperature based on the pressure-composition-temperature (PCT) curves of erbium tritium (Er-T) system is needed in the loading process. The fabrication processes of ditritide erbium (β phase) with tritium atoms located at the tetrahedral site ($\beta_{Tet)}$ of face-centered cubic (FCC) was specially considered.

A thermal desorption spectra (TDS) techniques are used to determine the thermal stability and the occupation of general metal hydrides will be presented, and the corresponding relations of fine structure and TDS peaks of erbium hydrides were successfully established. To provide for manufacturing quality control of grain morphology, SEM analysis techniques are used to observe grain size and grain boundary. Under the proper conditions of fabrication processes, it mainly is tritium loading temperature and the rough of substrates, the SEM images show that the micro morphology of erbium ditritide films inherited the main structure of erbium films in a great extent, so the micro morphology of erbium ditritide films control will become easy.

Effects of morphology and phase changes on ^{3}He release/retention (RF) and crystal lattice evolution during aging of erbium tritide films were investigated with mass spectrometer and X ray diffraction. It shows that existence of metastable phase (i.e. β phase erbium tritium with tritium atoms octahedral occupancies, β_{Oct}), decrease of grain size and rarefaction of grain boundary will lead to a significant increase in the proportion of 3He release, and the mechanism of grain size effects of erbium tritide on ^{3}He release/retention properties was discussed.

Paper 4A-5: Monday, April 18

Electron Microscopy of Helium Bubbles in a Palladium Alloy

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The formation of nanoscale helium bubbles in metals is a common effect of exposure to radiation and radioactive substances. The rapid radioactive decay of tritium absorbed in metals leads to an extreme example of this, but the safe study of tritium-exposed samples can be challenging. We have developed methods to gently extract most of the last traces of tritium from samples, and to minimize sample volume. These enable the study of helium bubbles in tritium-exposed samples by modern electron microscopy techniques, including 3D imaging by electron tomography, compositional maps near atomic resolution by aberration-corrected microscopy, and microfabricated environmental stages that allow study of bubble properties as a function of temperature. We have recently initiated studies involving these techniques.

The tritium absorption properties of a palladium-nickel alloy sample (5 atom % Ni) were previously published [JS Holder and JR Wermer, J. Alloys Compounds 231 773 (1995)]. After further tritium exposure (about 3.8 years total), we have removed the tritium from this sample to a few tens of microcuries per gram using cycles of exposure to deuterium gas and vacuum without significant heating of the sample. Sections with submicrometer thickness and submillimeter lateral dimensions were prepared using an ultramicrotome. The radioactivity of the resulting samples is low enough that they can be studied with general-purpose instruments without significant safety risks. Transmission electron microscopy reveals 2 nm bubbles in the material at a density on the order of several million bubbles per cubic micrometer. Tomography of the samples is expected to provide more quantitative determinations of the distribution of size and spacing of bubbles, which can provide useful input to theoretical models of bubble formation [DF Cowgill, Fusion Sci. Tech. 48 539 (2005)].

Sandia National Laboratories is a multi-program laboratory managed and operated by Sandia Corporation, a wholly owned subsidiary of Lockheed Martin Corporation, for the U.S. Department of Energy's National Nuclear Security Administration under contract DE-AC04-94AL85000. Work at Savannah River National Laboratory was performed under contract number DE-AC09-08SR22470 with the U.S. Department of Energy (DOE) Office of Environmental Management (EM). SAND2015-8812A

Paper 4A-6: Monday, April 18

Hydrogen Isotope Effect on Getter Rate

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Tritium Producing Burnable Absorber Rods (TPBARs) use a nickel plated, zirconium getter for gettering tritium. During the development phase of the getter design H_2 gas was used to determine the getter rate. The verification phase of getter performance also used only H_2 gas. Thus, potential isotope mass effects on getter rate were not evaluated. A modified getter rate test using both H_2 and D_2 gas was performed to determine potential isotope mass effects. Test procedures were developed to minimize experimental uncertainties with respect to surface cleanliness, minimize variability of test results between different samples, and evaluate the temperature effect on isotope mass by conducting the experiment.

The getter rate test was based on supplying a calibrated molar flow rate of H_2 and D_2 gas to a getter and measuring a pressure response. The getter was positioned in a quartz tube of known volume. The quartz tube was initially evacuated to a low vacuum and placed in a heated oven with temperature control. The oven temperature was controlled to obtain the desired temperature in the test sample. Upon opening the gas supply, the time dependent pressure was recorded until an equilibrium pressure was achieved. The gas supply was then shut off and the time dependent pressure decay recorded. This provided two methods of evaluating getter rate, the equilibrium pressure with a known molar flow rate and the pressure decay with a known volume. A total of 4 temperatures were evaluated using both H_2 and D_2 gas, yielding 8 measurements. A single sample was used for all 8 measurements. A duplicate set of measurements was made using a second sample. A distinct isotope mass effect on getter rate was observed that was consistent between test samples and over the temperature range of the test. A discussion of the test procedures and an evaluation of the impact on getter rate will be presented.

Tritium Challenges and Plans for ITER Pellet Fueling and Disruption Mitigation Systems

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ITER will include a “Pellet Injection System” for supplying deuterium-tritium fuel to the fusion plasma and a “Disruption Mitigation System” for rapidly injecting impurity gases (argon or neon mixed with helium and/or deuterium) into the plasma for the purpose of mitigating the harmful effects of plasma disruptions. Each system has unique challenges for the safe handling and confinement of tritium.

In the case of pellet injection, the deuterium-tritium fuel is cryogenically frozen and then extruded into a continuously flowing ribbon. A pellet cutter is used to rapidly (up to 16 Hz) cut cylinders from the extruded material. A gas valve then opens to propel the cylindrical fueling “pellet” down a series of guide tubes and into the plasma. A sub-system is included to capture the unused extruded material and recycle it within the pellet injection system, minimizing the processing load on the plant exhaust reprocessing system.

In the case of disruption mitigation, cryogenic pellets formed in-situ are held in a “pipe gun” type injector backed by a fast acting propellant gas valve. Should the plasma control system detect the onset of a disruption (loss of plasma magnetic confinement within the vacuum vessel), the pellet is injected by releasing the propellant gas. The pellets are made of argon or neon with a shell or mixture of deuterium. The propellant gas is helium or deuterium. The deuterium used in the disruption mitigation system is reprocessed gas from the plasma exhaust, and as such, has some level of tritium that must be properly contained.

The two systems share similar environmental conditions while under operation. Their locations within the tokamak building expose them to significant background gamma, neutron, and magnetic fields. Both systems use pressurized gases (up to 40 bar) which will contain tritium. Confinement and purge gas systems are planned for both systems in order to provide barriers between the tritium and the outside environment, and to provide an inert atmosphere to eliminate the possibility of potentially explosive atmosphere. Here each system is briefly described and the unique challenges for the safe handling of tritium are detailed. The planned confinement and purge gas systems are described, as are the environmental conditions in which the systems must operate.

Preliminary Design of a Tritium Control System for Fluoride-Salt-Cooled High-Temperature Reactors

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The Fluoride salt-cooled High-temperature Reactor (FHR) is a Generation-IV molten salt reactor. It combines the advantages of the Sodium Fast Reactor (SFR) and the High-Temperature Gas-cooled Reactor (HTGR). However, tritium generation from neutron activation of the primary coolant FLiBe (a mixture of molten LiF and BeF2) is a significant issue in current development of FHRs. As estimated, a 2400 MWth FHR during startup will generate as high as 5000 Ci of tritium per day. For comparison, a 1000 MWe PWR only generates 1.9 Ci of tritium per day. Currently a three-loop FHR is under consideration with an extra loop for tritium removal. As the cost and increased size brought by the additional loop offsets the advantages of FHR, a tritium control system that applied to a two-loop FHR is investigated in this study.

Tritium control and mitigation in FHRs consists of three major tasks. The first one is the amount control or elimination of TF, which is highly corrosive to structural materials. Second is the extraction of tritium from the primary coolant. It would be beneficial if the tritium extracted is collected for further use. Third is the prevention of tritium permeation out of the primary loop through intermediate heat exchanger or structural materials.

A successful tritium removal facility needs to meet three requirements: large mass transfer area to volume ratio, mixing of molten salt to enhance tritium diffusion in the molten salt and large tritium concentration gradient on the mass transfer surface to provide sufficient diffusion driving force.

In this paper, a tritium control and mitigation system is presented. It consists of three major facilities: the redox control facility, the tritium removal facility and the IHX with tritium permeation barrier. In addition, Al2O3 coating as a surface treatment on structural materials is applied as needed to prevent potential tritium leakage. The designs of the facilities are developed and optimized via computational simulation using COMSOL Multiphysics and MATLAB.

Keywords: FHR, Tritium control

Tritium Constraints and Strategy for the Fusion Deployment

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This paper describes the possible constraints for fusion energy to be introduced and deployed to the future energy market, and strategy to resolve them.   Economy, safety, environment, and security will be discussed with simplified and generalized models to identify and consider the problems.

First problem is a tritium supply and economy for the operation of DT reactor fuel systems. It is quite obvious that fusion reactors need at least kilograms to more than 10kg of tritium as an inventory in their fuel cycle to stably operate as energy source. Without having a confidence for initial and stable supply of fuel, none cannot even initiate any project to plan a reactor. The authors have suggested that with adequately designed breeding blanket system, moderately extended period of DD operation of fusion reactor planned in the commissioning phase will produce sufficient amount of tritium by DD reaction that generates both tritium and neutron to be used for breeding. The fuel cycle model suggests the detailed characteristics and dynamics of plant inventory to be described as the combinations of various mechanisms and time constants to hold tritium. Their variations are: a gram and seconds level of plasma to 100 grams and hours in the main fuel loop, 10~1000 grams and hours~days in blankets, smaller amount of tritium in large amount of media from secondary and wastes, and extremely long period and large amount from solid wastes including PFC and blankets. This feature suggests that external initial supply is not practical, if fusion would be a viable energy for the world particularly in its early phase of introduction in the market. Also, inventory control to avoid excess tritium will be inevitable.

While safety of fusion energy is generally controlled by multiple confinement of tritium in the plant both for normal and accidental events to limit personnel dose, acceptance of fusion by public will be strongly affected by environmental behavior of tritium. This paper will report long-term accumulation and spreading of tritium from both airborne and aqueous releases that will not cause significant personnel dose but well above natural background. Particularly environmental pathway of tritium accompanied with waste heat has a major impact. Environmental model suggests that when adequately utilized, global circulation of water could dilute and isolate tritium from human activities with the time constant longer than the half-life of tritium.

Security issue of tritium is currently governed by export control under an internationally agreed regime. Since actual inventory measurement in fusion reactor with short interval and accuracy will remain technically difficult, traffic of significant amount of tritium will need to be limited. Some aspects of fusion energy system from nuclear material control will be discussed.

Use of TAS in Assessment on Tritium Self-Sufficiency, Resource Sustainability and Radioactive Safety for Fusion Reactor

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The International Thermonuclear Experimental Reactor (ITER) will be the first fusion reactor to achieve long pulse Deuterium-Tritium plasma burning, but only has six Test Blanket Module (TBM) and without full tritium cycle. To close the tritium fuel cycle and achieve self-sufficiency, a fusion engineering test reactor (e.g. FNSF, CFETR, MFX…) is necessary to be constructed before the DEMO plant. Since the tritium resource shortage, serious permeability and natural gamma radioactivity, before construction, the first fusion reactor has to find a reliable tritium source for initial startup and validate the environmental compatibility before large inventory DT operation.

For this concern, a Tritium Analysis program for fusion System (TAS) has been developed by Institute of Nuclear Energy Safety Technology, Chinese Academy of Sciences (INEST·CAS), which dedicate to the dynamic simulation of tritium cycle, release source item, environmental dispersion, and public impact for fusion reactor. In the 3.0 version, a system dynamic method was utilized to rebuild the fuel cycle model for fusion reactor, and a new Gaussian puff model was coupled with visual point source to simulate the tritium migration and dynamic re-emission in environment.

By using the TAS, 1) key influence factors of tritium self-sufficiency and initial supply were taken into detailed consideration, and 2) full sequence from released tritium to public absorbed dose was simulated dynamically depending on various operation modes and environment conditions. From the perspective of tritium self-sufficiency, resource sustainability and radioactive safety, the main challenges facing by fusion reactor were addressed. Finally, relevant countermeasures and required R&D towards DEMO reactor were also proposed.

Key words: Fusion reactor; TAS; Tritium cycle; Tritium safety

Estimation of Tritium Release from LLCB TBM and Its Ancillary Systems and Tritium Management in Different Locations of ITER

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The Indian Lead Lithium Ceramic Breeder (LLCB) Test Blanket Module (TBM) is to be installed in one half of equatorial port #2 for testing in ITER machine. Liquid Pb-Li and solid Li_2TiO_3 are the tritium breeder materials in LLCB TBM. Tritium permeates through structural materials in particular at higher temperatures, which is a major operational and safety concern. Therefore, tritium flows in different locations of ITER Tokamak complex have been estimated.

Tritium transport from LLCB TBM and its ancillary systems into process rooms has been studied and analyzed in this work. A steady state diffusion limited permeation model neglecting surface effects has been used for the analysis. Tritium permeation to the vacuum vessel, pipe forest area, port cell, pipe chase area, tokamak cooling water system vault annex and tritium process room in L-2 level has been estimated.

The requirement to be fulfilled in each equatorial port cell is that the Tritium concentration in the port cell during maintenance operations should be below the admissible limit for human access (regulatory maximum allowable value < 1 DAC, Derived Air concentration). The presence of the Detritiation System (DS) in the Port cell has to be taken into account [1].

This admissible limit for human access has to be reached in a sufficiently short time (target = 12 h) after plasma shutdown. Additional release during maintenance and radiological zoning with recommended <10 μSv/h need to be considered. Management of concentration of permeated tritium in different locations considering above requirement has also been discussed in this paper.

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Estimation of Tritium Permeation Rate to Cooling Water in Fusion DEMO Condition

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Estimation of tritium permeation rate to a coolant is very important to design a fusion plant and to assess its safety. The increment of tritium concentration in the coolant raises the potential risk of the fusion reactor. The tritium removal system with the scale applicable to the permeation rate would be necessary in the coolant loop. Tritium loss into the coolant affects the estimation of the required tritium breeding ratio. Practically, the estimation of tritium permeation rate in the plasma facing wall is not easy because mass transfer phenomena such as erosion, deposition, helium implantation, etc. take place complexly. Also, the understanding of the chemical form of the tritium released from the solid breeder is not complete although the permeation rate of the bred tritium in the blanket depends on the HT/HTO ratio strongly. However, the approximate estimation of tritium permeation rate under the acceptable assumption from a safety point of view is surely useful to progress the design activities for a fusion DEMO reactor. In this work, tritium permeation rates in the blanket and the divertor were estimated by the simplified steady state model under the recent DEMO condition in the water-cooled blanket with solid breeder as a first step. The following is one of representative results.

Doyle & Brice model was applied to estimate tritium permeation rate through the tungsten (W) armor in the blanket first wall and the divertor. Hydrogen isotope diffusivity in W by Fauenfelder and recombination coefficient on W by Anderl were used. The permeation through structural material and cooling pipe were ignored. In the blanket first wall, the thickness of W armor was 0.5 mm, the temperature was uniformly 400 ^{o}C, particle flux was 1×10^{20} 1/m^2/s and plasma facing area was 1070 m^2. The estimated permeation rate was 1.8 g/day in the blanket first wall. In the divertor, the thickness of W armor was 5 mm, the temperature was 600, 700 or 800 ^{o}C, and particle flux was 1×10^{20}, 1×10^{22} or 1×10^{24} 1/m^2/s depending on the position. Total plasma facing area was 341 m^2. The estimated permeation rate was 1.6 g/day in the divertor.

Hickman's law was applied to estimate the permeation rate of the bred tritium from the purge gas containing hydrogen to the cooling water through the F82H. Hydrogen isotope diffusivity and solubility by Serra were used. The thickness of F82H was 1 mm, the temperature was uniformly 300 oC. T_2 pressure and H2 pressure in the purge gas were 0.5 Pa and 100 Pa constant, respectively. The total permeation area was 14250 m^2. The estimated permeation rate was 2.3 g/day in the breeding region.

Total tritium permeation rate was estimated to be 5.7 g/day. Tritium production rate in the recent DEMO condition is assumed to be 230 g. Therefore 2.5 % of the bred tritium is permeated into the cooling water.

This work was performed under the Broader Approach Activities.

Design Considerations for Tritium Handling Systems: An Engineering Approach

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Due to the radioactive and elusive nature of tritium (^3H), a comprehensive approach to designing tritium processing systems is essential for understanding, preventing, and mitigating tritium-related hazards. Tritium handling systems must have an exceptionally low probability of leaking. Consequences of tritium leakages can include assimilation into personnel, environmental release, ignition, and violation of operating permits. Important design considerations to counteract such consequences include, but are not limited to, material compatibility, overpressure protection, radiological protection, component selection, active/passive control methods, and confinement concerns.

Effective tritium process design begins with an understanding of the chemical behavior of tritium, which behaves similar to its more common hydrogen isotopes. However, certain mechanisms and reactions change significantly due to the ionizing β-particles emitted by tritium, which greatly influence material selection and the design of primary and secondary confinement systems. Inerted gloveboxes combined with stripper or getter systems are often used to protect workers and to reduce the risk of release of tritium to the environment. Tritium also readily effuses and permeates faster than most gases, which influences mechanical component selection and the design of overpressure protection systems to meet stringent leak requirements of tritium systems. Passive controls are preferred over active controls, but a combination of the two is imperative when designing a process that is inherently safe. Common passive controls include limiting combustible loads and equipment/piping system design (e.g. fully welded vs fitted), while some active controls include intra-process monitoring (e.g. pressure, temperature, flow, gas composition) and the use of control valves for operation and material isolation purposes.

The objective of this work is to identify and detail several key design considerations which can be applied to ensure the safe and effective operation of tritium handling systems. Operational experiences and lessons learned from distinguished tritium facilities which influence the design process are also enveloped within this effort.

Model for the Production, Diffusion, and Containment of Tritium in PWRs

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Following March 2011 nuclear accident in Fukushima, the future of the civilian nuclear power industry is more than ever dependent on its ability to meet increasing demands in public safety at a sustainable cost. So far, the tritium produced in pressurized water reactors (PWRs) has been mainly discharged to the environment in light of the reputed low biohazard of diluted tritium and the non-availability of economical light-water and gas detritiation technologies. However, the acceptable level of these releases could be revisited as a function of evolving water standards, public acceptance, and technological advancements.

In this context and as an extension to EPRI's original work [1], the present contribution proposes a generic model and simulation code for the production and diffusion of tritium in PWR systems, with the aim to help monitor the inventories of tritium and release pathways as a function of water management and treatment strategies within a typical fuel cycle. The model mainly incorporates the following input parameters: reactor size and power; fuel cycle duration; load-following ratio; fuel burnup; boron and lithium enrichment and coupled chemistries; method of boron chemistry adjustment as a function of burnup (dilution/removal); water management during refueling; and importantly recycling and/or detritiation of letdown water. The simulation output includes the time evolution of tritium inventories in the different parts of the PWR system, along with the volumes and activities of tritiated water and gas released to the environment.

Additionally, if detritiation is enabled in the simulation, the model may be coupled with an economic assessment of competitive light-water detritiation technologies [2], including Kurion's newly developed Modular Detritiation System (MDS) [3].

As an example scenario, it is shown that (i) all the tritium produced during a given fuel cycle may be removed from the primary loop by diluting the primary loop water before refueling, thus avoiding tritium diffusion into the refueling canal and evaporative releases, which are thought to be much more biohazardous than liquid releases (ii) if considered, detritiation would be best performed within the first 10 months of the fuel cycle, enabling a reduction of up to 80% of liquid activity releases by detritiating approximately 100,000 gallons of water per core and per fuel cycle.

[1] EPRI Tritium Management Model, EPRI, Palo Alto, CA (2005). Product ID 1009903.
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Dynamic Tritium Fuel Cycle Analysis for Fusion Reactor Based on System Dynamics Model

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As the first fusion reactor for deuterium-tritium plasma burning, the International Thermonuclear Experimental Reactor (ITER) will adopt two pulse operation scenarios. The first scenario is that the pulse length of plasma will be up to around 400 s, and the second scenario is that the pulse length of plasma will be up to around 3000 s. Therefore, long pulse operation scenario is also a possible way to achieve fusion energy.

In order to achieve self-sufficiency of tritium fuel cycle for pulse operation fusion reactor, tritium fuel cycle analysis is an essential research issue. Based on this issue, a system dynamics tritium fuel cycle model for pulse operation fusion reactor was developed by using a commercial software STELLA. In this model, about 12 subsystems were introduced to analyze the stocks and flow of tritium in D-T fusion reactor. Based on this model, tritium self-sufficiency analysis was performed for long pulse operation fusion reactor. Meanwhile, sensitivity analysis of pulse length, burning ratio, tritium cycling time and recovery ratio were carried out. The preliminary results showed the minimum tritium breeding ratio (TBR) increases with pulse length, burning ratio and recovery ratio, while decreases with tritium cycling time. The results can provide tritium self-sufficiency data guidance for pulse operation fusion reactor.

The Status of Tritium Transport Libraries in EcosimPro

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One of the main problems that arises in a fusion reactor is that the consumption of tritium as fuel must not exceed both the amount available and the production capacity. This makes having suitable systems for producing and recovering tritium especially important.

Because of the diffusive properties of tritium, the main priority in these systems is how to control its inventory and permeation to the environment. This is where the ability to simulate the behaviour of tritium takes on particular importance, and models have to be developed to study the transfer phenomena that take place. The results of these simulations can be of great use in developing tritium control systems and designing devices like breeding blankets.

This is what has motivated the creation of models of these phenomena using the simulation tool EcosimPro. Using this program offers considerable advantages because its object-oriented nature facilitates implementation of these processes, the possibility it offers of mixing various disciplines (e.g., transport, control, hydraulics, etc.) and the robustness of its equation-solving algorithms.

EcosimPro and its TRITIUM_LIBS libraries enable the user to build simulation models from a simple permeation membrane to all European Breeding Blanket concepts and auxiliary systems both ITER and DEMO conditions.

The status of the libraries contained in TRITIUM_LIBS is the following:
TRITIUM_BALANCE: basic components to build tritium transport systems.
TRITIUM_TEST: validation experiments and theoretical problems.
TRITIUM_BREEDING BLANKET: HCLL, HCPB, WCLL and HCPB blanket components and ancillary systems.
TRITIUM_FUEL_CYCLE: Fuel cycle steady-state model for a nuclear fusion reactor.
TRITIUM_PLANT: Cryogenic Distillation Column (ISS) and Water Detritiation System (WDS) models.

The libraries helps to study different tritium phenomena using an exhaustive material database and to perform steady state, parametric studies and transient studies of these models thanks to the powerful features of the EcosimPro experiment. The most important results obtained with tritium libraries in EcosimPro are showed and discussed. Some upgrades of the codes are presented as a future work in the tritium transport simulation field.

Conception of Multi-Purpose Heavy Water Detritiation Plant

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Heavy water is used as a neutron moderator and coolant in nuclear power and research thermal reactors. During operation of heavy water reactors, heavy water becomes contaminated with tritium and protium. Protium comes from various sources such as water vapor uptake from the atmosphere. Tritium is produced in heavy water owing to neutron capture by deuterium atoms. Thus, heavy water reactors require facilities to keep deuterium concentration and remove tritium. Schematic diagram of facility has been developed to maintain the characteristics of heavy water in the heavy water reflector. The plant is designed to address the problems related to handling of heavy water at the reactor site.

The installation allows us to solve several problems simultaneously:

- to remove tritium and protium from the reactor water.
- to process heavy water wastes with different content of tritium and protium, formed during reactor operation and operation of the detritiation plant;
- to produce both reactor-grade heavy water and heavy water with low tritium content.

Protium recovery capacity of the plant is not a fixed value. It can vary widely depending on the actual leakage of light water in a heavy water reactor loop.

The plant contains five LPCE columns of different heights and diameters, an electrolyzer and a low power cryogenic distillation column. The evaluation of the basic parameters of the installation has been done using computer models. The initial data for modelling were obtained in the course of long-term operation of EVIO pilot plant [1].

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Theoretical Analysis for Setting Up a Mixed Catalytic Packing that Equips a Catalytic Isotopic Exchange Column

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Water - hydrogen catalytic isotopic exchange process represents the front-end of detritiation technology, chosen and used in the pilot plant at ICSI Rm. Valcea, and for the Tritium Removal Facility that are to be built at the Cernavoda NPP, Romania.

Catalytic mixed packing which equip catalytic isotopic exchange columns is a key element which determines the performance of isotopic exchange process.

Both catalyst and hydrophilic package is necessary to provide a larger surface at fluid interface participating in isotopic exchange. Also their arrangement should facilitate a uniform flow both along it as well as through section. To achieve a high performance packing was necessary to conduct theoretical and experimental research over several years.

Improving catalytic mixed packing is a research project currently taking place both for the structure of materials, catalyst / hydrophilic package ratio, arrangement and not only. The mathematical model developed contributed greatly to theoretical analysis, directing research in the optimal direction.

Both in the pre and post experimental stage, the theoretical analysis was based on mathematical models developed within the Institute for description of water - hydrogen isotopic transfer, in their turn being in continuous improvement.

The paper presents a theoretical analysis developed to set up a high performance mixed catalytic packing, made of hydrophobic catalyst Pt / C / PTFE and hydrophilic stainless steel.

To achieve an effective mixed packing is necessary first to determine by laboratory analysis the isotopic transfer performance separately for each of the two fillings. Efficiency is defined both by mass transfer constant as well as the height of theoretical plates. Thus, it determine the isotopic transfer efficiency between water vapor and gas in case of hydrophobic catalyst Pt / C / PTFE. For the hydrophilic package also determine separately the isotopic transfer efficiency between liquid and vapor.

Based on these characteristics and using the mathematical models developed, we conducted a theoretical analysis on the catalyst / hydrophilic package arrangement in order to achieve high performance isotope transfer. We modified the catalyst / hydrophilic package ratio and arrangement as well, showing the level of contribution to isotopic transfer.

The paper presents the results of those analyzes as well as graphical and numerical mathematical models developed that allowed this analysis.

Detritiation and Tritium Storage Technology Development

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Korea is operating 24 nuclear power plants and a highly advanced neutron application reactor HANARO. And she is designing tritium storage and delivery system (SDS) for ITER. Therefore, we have been developing detritiation and tritium storage technologies since the operation of Wolsong CANDU station in 1983.

The Wolsong Tritium Removal System has been designed to remove tritium generated in moderator and heat transport heavy water. Catalysts transfer tritium from tritiated heavy water to gaseous tritiated deuterium. The hydrogen isotopes, including tritium, are transported to a cryogenic distillation system where the tritium is removed for safe storage. Conventional high-pressure storage tanks can be dangerous for storage of radioactive tritium gas. We have been studying various kinds of metal hydride, such as titanium, zirconium cobalt, and depleted uranium. Titanium was proven to store tritium safely and efficiently for a long time. Zirconium cobalt, however, incorporate tritium safely and compactly, and temporarily hold large quantities of it that can be recovered easily under safe, controlled conditions. But due to the disproportionation characteristics of zirconium cobalt, we are now developing depleted uranium hydride safe handling technologies.

SDS is a part of ITER fuel cycle, installed in the tritium plant. It stores and delivers the hydrogen isotopes and pure gases required for the tokamak. SDS serves as a safe storage facility for the tritium and deuterium in a metal hydride form for the short-term operation and the long-term maintenance. It has various functions: delivery of the hydrogen isotopes and pure gases; storage and delivery of tritium according to the fuel cycle operation scenarios; safe storage of tritium, inventory accountancy, and tritium tracking; operation of metal hydride bed for tritium; long-term storage of tritium, handling and accountancy facility for the imported tritium. Thus, we are going to present our progress since the last tritium conference.

Application of Imaging Plate Technique and Beta-Ray-Induced X-Ray Spectrometry to Measurements of Tritium Distributions in Divertor Tiles used in JET ITER-Like Wall Campaigns

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One of the main problems that arises in a fusion reactor is that the consumption of tritium as fuel must not exceed both the amount available and the production capacity. This makes having suitable systems for producing and recovering tritium especially important.

Because of the diffusive properties of tritium, the main priority in these systems is how to control its inventory and permeation to the environment. This is where the ability to simulate the behaviour of tritium takes on particular importance, and models have to be developed to study the transfer phenomena that take place. The results of these simulations can be of great use in developing tritium control systems and designing devices like breeding blankets.

This is what has motivated the creation of models of these phenomena using the simulation tool EcosimPro. Using this program offers considerable advantages because its object-oriented nature facilitates implementation of these processes, the possibility it offers of mixing various disciplines (e.g., transport, control, hydraulics, etc.) and the robustness of its equation-solving algorithms.

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The libraries helps to study different tritium phenomena using an exhaustive material database and to perform steady state, parametric studies and transient studies of these models thanks to the powerful features of the EcosimPro experiment. The most important results obtained with tritium libraries in EcosimPro are showed and discussed. Some upgrades of the codes are presented as a future work in the tritium transport simulation field.

5-09 Poster Session 1: Monday, April 18

Using Dye Tracer Studies to Characterize Tritium in Groundwater: No News can be Good News

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Boiling water reactors (BWRs) often result in tritium contamination to groundwater. Typically releases are associated with buried piping associated with the Auxiliary Buildings or to Refueling Water Storage Tanks (RWST). Characterizing the subsurface is further complicated by the vast network of underground piping, structures, and foundations power plants. Fluorescent dyes are a useful tool for characterizing groundwater flow, both in the vadose zone and within the shallow aquifer. This paper will discuss three case studies where dyes were successfully used to track the pathway of tritiated water, and demonstrate that any off site discharges were via monitored pathways. These studies were completed with no dyes visibly reaching public water bodies.

The studies include:

- Fluorescein and Rhodamine WT dyes used at the base of tanks to track vadose zone migration to vaults within the protected areas.

- ßEosine dye used in subsurface piping to demonstrate the integrity of the piping and surrounding encasement below the Auxiliary Building.

- ßFluorescein and RWT dyes used to evaluate shallow vadose and deeper groundwater pathways to demonstrate historic releases

Dye tracer results from these three case studies demonstrate that this is a cost effective and valuable tool to document groundwater flow paths. These studies also demonstrated that, at times, the lack of measurable data (dye detections) can be as useful as positive results when developing and validating conceptual site models.

Geotritium: Implications of DeP Weak Interaction Fusion in Magmatic Systems

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Tritium in the environment comes from several sources: anthropogenic nuclear weapons and commercial nuclear reactors, atmospheric cosmic-ray production, and geotritium. Geotritium and its decay product ^{3}He is found in the earth's magmatic system evidenced by tritium emission from volcanoes and ^{3}He anomalies associated with Mid Ocean Ridge Basalts (MORB), Oceanic Island Basalts (OIB) and deep mantle plumes. The ^{3}He can be accounted for assuming a radiogenic ^{3}He component in combination with mantle depletion of U and Th and degassing of the upper/lower mantle. It is proposed that the $d(pe^-, \nu)^3t$ weak interaction fusion (WIF) reaction (Q = 5.474 MeV) is responsible for both the geotritium and anomalous ^{3}He found in nature. The radiogenic lifetime for the WIF reaction, $dp+e^- \rightarrow {}^3t+\nu_e$, in a metal hydride geologic environment, such as FeH/FeD in the core and mantle of the earth, was calculated to be $_{Tdep} = 1.61\mathrm{x}10^9$ yr. The anomalous ^{3}He produced by the WIF reaction can be used as a radiochronometer to date the mantle basalt processes. The $p(pe^-, \nu)d$ WIF reaction (Q=1.442 MeV) with a lifetime $_{Tdep} = 2.54\mathrm{x}10^{12}$ yr. was also examined as a possible radiochronometer for the formation interval of the oceans, the duration of the earth's melting and degassing prior to the formation of the oceans. The formation age of the oceans was determined to be 3.92 billion years ago in good agreement with geologic magmatic activity.

Safe Decommissioning for Tritium Storage U Bed

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In tritium system and laboratory, tritium is usually stored in tritium storage beds, such as uranium beds. The early use of the tritium storage bed is mostly made of glass. Since the glass vessels are easily broken and the releasing activated uranium powder would burn when it encounters air, the tritium and uranium airborne could be formed. For decreasing the radiochemical toxicity resulting from tritium and uranium airborne, nowadays, the glass uranium beds have been abandoned for its bad safety, and the safe decommissioning of the tritium storage uranium beds will be one of the most important parts in tritium laboratory safety.

In safe decommissioning, the key facts are removing tritium and avoiding the uranium airborne to be formed when the activated uranium powder encounters air. Therefore, the physical and chemical properties of tritium and uranium are needed to considered, besides the policies of the radioactive waste management are taken into account. Thus, the method of decommissioning glass uranium bed was studied, and the method of removing tritium through high temperature analysis combined with hydrogen isotope exchange, and stabilizing the active uranium powder by oxidation was set up. Based this management method, the tritium storage uranium bed experimental management device was made. The device could dispose less than 100g uranium powder containing tritium less than 1×10^{12} Bq was developed. The tritium storage glass uranium bed management technology was validated by using our laboratory awaiting decommissioning tritium storage glass uranium bed. The result shows that the technology could dispose the glass uranium beds by meeting the requirements. After being disposed, the waste could be transferred to tritium and U_3O_8, which is a stabilized oxide material. Through the technology, the waste volume decreases 95%, and the impact for the environment is slight.

Isotope Exchange of Two $LaNi_{4.25}Al_{0.75}$ Hydride Process Beds

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The Savannah River Tritium Enterprise has used $LaNi_{4.25}Al_{0.75}$ (commonly abbreviated as LANA.75) metal hydride beds for over two decades for high density storage of tritium and other hydrogen isotopes. Beds exposed to significant quantities of tritium have a limited service life due to radiolytic decay to helium-3 within the metal matrix. Unlike hydrogen, helium-3 “born” in the metal is insoluble and stresses the surrounding crystal lattice. These stresses partially manifest themselves in a reduced bed capacity, due to the formation of a “heel” of hydrogen isotopes that cannot be desorbed under normal process conditions. If kept in service long enough, the bed will begin to weep helium-3. When either the capacity impacts process throughput or the bed can no longer deliver helium-3 free hydrogen isotopes, the bed is removed from service.

In preparation for removal from the process, each bed undergoes a series of isotope exchanges in which a measured amount of protium or deuterium is loaded onto the beds. The beds are then desorbed to a tank, and the composition of the desorbed gas is measured by high resolution mass spectrometry. The process is repeated several times, effectively diluting the tritium concentration in the heel. This operation serves two purposes, to recover as much tritium from the heel as practical, and to reduce the potential for pressure buildup in the retired vessel due to continued decay of tritium to helium-3.

One of the challenges associated with performing isotope exchanges is calculating the residual tritium inventory of the bed. Application of a straightforward mass balance can sometimes lead to the conclusion that more tritium was removed from the bed than was originally present. This obviously incorrect conclusion may be attributed to many possible errors, most of which are unavoidable in an active process environment. Application of a dilution model, in lieu of a mass balance, will ensure that the final tritium inventory of the bed is a positive value.

Two LANA.75 hydride beds recently reached the end of their service lives in the Savannah River Tritium Enterprise processing facility. Both beds have undergone multiple isotope exchanges. Exchanges were performed by loading the beds with nearly pure deuterium gas from one tank and desorbing the bed to another tank. Gas analysis was performed on both tanks to aid in determining the reduction in tritium inventory in the bed being exchanged. Prior to the isotope exchange process, initial tritium inventory was determined using the flow-through calorimetry. Processing of the data and final results are reported.

5-13 Poster Session 1: Monday, April 18

^{14}C and ^{3}H Determination in Irradiated Graphite from the Thermal Column of the VVR-S Research Reactor

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Due to the large volume, the graphite thermal column comprises considerable amounts of low-medium radioactive waste.

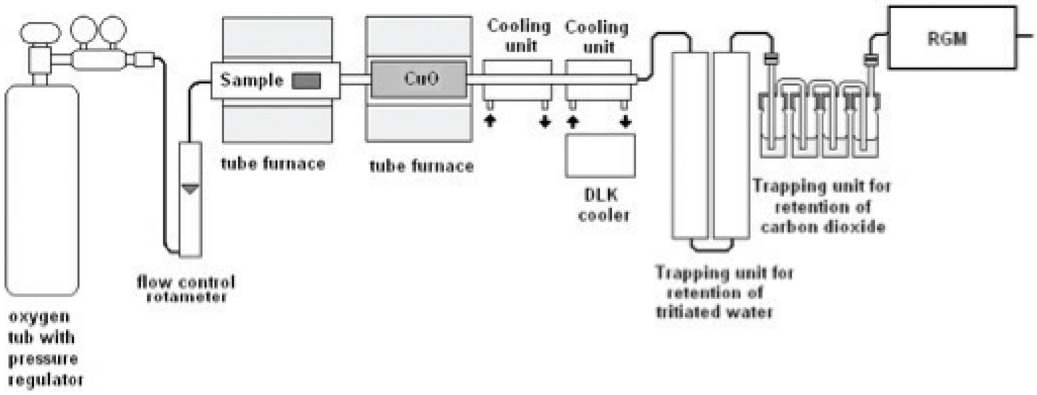
In this study, a new oxidizing combustion method has been developed to determine, separately, the ^{14}C and ^{3}H activities in the samples collected from the irradiated graphite of the VVR-S research reactor.

The experimental methodology concerns the oxidation in two steps of the samples in oxygen atmosphere. HTO and $^{14}CO^{2}$ fractions were collected separately in their cocktails in 10 ml vials and the measurements of ^{3}H and ^{14}C activities were performed using a Liquid Scintillation Counter.

A schematic diagram of the experimental setup is presented in the figure:

- Pressure oxygen tube with pressure regulator and flow control Rota meter
- 2 tube furnace with temperature controller (RT 50-250/11 Nabertherm type), one for graphite oxidation (at 8000 C temperature) and a second one for catalytic oxidation of resulted gases of? water and carbon dioxide. (700-8000 C in
the presence of CuO wires or ~ 4500C in the presence of Pt/Alumina catalysts)
- Cooling units for flue gases with two-stage cooling, first with air and secondary using a recycling cooler DLK 40 Lauda type coupled at a Proline thermostat.
- Trapping unit for retention of tritiated water over sicative CaCl2
- Trapping unit for retention of carbon dioxide using Carbo-sorb E+
- Radioactive gas monitor (RGM) with Ionizing Chamber Overhoff type

The retention yield was calculated and the value obtained was 92-94%.



Safety Aspects of an Interim Storage Facility for Tritiated Waste from Nuclear Fusion Facilities

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The management strategy for tritiated waste produced by licensed nuclear facilities in France is based on the design, safety and operability of interim storage solutions defined for each category of tritiated waste.

A 50-year period for such interim storage was defined considering the available feedback from existing storage facilities for fission-related waste (not especially dedicated to tritiated waste) on the one hand, and the relatively short half-life of tritium with a decay factor of 16 (4 half-life decay periods) on the other hand. Therefore, the duration of 50 years for this building is consistent with the requirements governing interim storage, whereas the lifetime for disposal facilities is designed to last much longer, which is not the case.

This route provides waste producers with new possibilities in the field of tritiated waste since waste with higher levels of tritium becomes acceptable.

A key point for the acceptability of fusion as an energy source is being able to implement the services required to manage radioactive waste generated by the ITER project on a timescale that is compatible with the project's needs. Within this framework, the CEA has to design, build and operate the interim storage facility called INTERMED. This facility will mostly be dedicated to the tritium decay of ITER waste (VLLW and SL-LILW produced during the operational and deactivation phases).

Taking into account the strategy for tritiated waste management and nuclear safety issues, the main specificities of the INTERMED project are:

- providing a "bridging" function between the raw tritiated waste generated by ITER (and other producers) and ANDRA's disposal facilities
- designing a building with an operational lifetime of around 70 years
- taking into account the high mobility of tritium.

The purpose of this paper is to describe the legal framework of INTERMED within this specific context, before detailing its safety functions as submitted to the French regulator (ASN), together with the main safety options. The next steps in the licensing process are also discussed in this paper. We provide a brief overview of the various facilities that handle tritium, thereby showing that tritium management is an international concern.

Romanian Tritium for Nuclear Fusion

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An energy producing fusion reaction has not yet been demonstrated, yet the potential for fusion power to provide future baseload energy generation continues to drive global research. The demand for tritium from fusion research is expected to surge in coming decades, and may significantly reduce the global tritium stock due to a sizeable mismatch between CANDU supply and fusion demand.

Tritium must be manmade and for commercial purposes is solely sourced from CANDU-type heavy water reactors as a by-product of the fission process. Owing to the aforementioned mismatch in supply and demand, no external source of tritium could sustain a commercial fusion power programme. Therefore demonstrating the concept of tritium breeding and consequently achieving "tritium self-sufficiency" will be essential for commercial fusion power. However, in the short-term, an external source of tritium is required to demonstrate tritium breeding (amongst other unproven fusion technologies also requiring tritium).

Currently only two facilities are capable of producing tritium from CANDU reactors worldwide. Romania is set to become the third as it currently plans to build a tritium removal facility to detritiate its two longstanding CANDU reactors at Cernavoda. Until now the rationale for detritiation has been driven solely by targets to improve radiological safety, and maintain efficient reactor performance. However this research considers the implicit yet largely overlooked opportunities associated with the resulting production of tritium, and resulting helium-3 (decay product of tritium).

Romania has a unique opportunity to become a major player in all future tritium ventures, specifically fusion. The introduction of a tritium removal facility could help to ease the strain on tritium stock from future fusion demand by providing a tertiary supply source, and given its price at roughly $30m per kilogram, it also has potential to provide significant positive economic impact for Romania. The experience of handling of tritium could also provide Romania with the opportunity to produce a generation of tritium experts, whom could be highly valuable to global fusion research.

In addition to the commercial value of tritium, opportunities also exist for its decay product helium-3. Helium-3 is also rare and highly valuable. Demand is currently dominated by the security sector, but implications for fusion-use of helium-3 may also see fusion research drive demand in the near future. The opportunities for the supply of helium-3 are thus implicit and also considered.

Romania is in an almost unique position to enter a rare isotope supply business. Although current focus is on tritium removal for sustaining safe and efficient reactor operation, there are clear opportunities given the expected future demand-pull for tritium from fusion research, and similar yet more immediate prospects in the demand for helium-3. Forecasts showing the potential for supply of tritium and helium-3 from a tritium removal facility at Cernavoda allows for better understanding of the opportunities, and may be of importance to three separate parties. On the supply-side Romania could see significant ongoing economic benefit from sales, as well as the potential for future research openings, and those involved on the demand-side a provided with a better insight into the availability of tritium and helium-3; namely in the fusion research community and security sector respectively.

Effect on the Secondary Buffer Vessel from the Upper Supplying Failure in Multi-Feed System

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This is the continuous work on the “Tank efflux study” as a similarity test of multi-hydride bed system:

When one part of supplying feed among the water efflux tanks has a failure such as clogging valves or having a very low delivery handicap in the upper water tanks, then the liquid head of the secondary buffer vessel shows that it is uneasy to keep a stable liquid level control. In realistic application with multi-tank efflux operation the fluctuation of the secondary buffer vessel shows a certain expands of the liquid head change during the supplying failure from the upper level tank efflux. In this study the variation of liquid head in the secondary buffer vessel is to be analyzed in theoretical approach, and then the required supplementary amount from the feeding point or the alternative delivery from the standby water tank will be discussed. Here, the authors propose a similar approach for the delivery amount control in the hydride bed system.

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HYSYS/ASPEN+ Advanced Tritium Transfer Modelling Tools for ITER/DEMO Plant Systems

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Tritium transfer predictive modelling is a historical top level scientific milestone of nuclear fusion technology. Thus, the development and qualification of predictive modelling tools for tritium transfers among systems represents today a technological challenge for final design and licensing of tritium systems in ITER. The use and exploitation have crucial implications on *(1) flexible operational reliability for the demands of experimental facilities; (2) the safe management and control of radioactive inventory for systems having large functional complexity; (3) the economy of expensive and scarce fuel.*

Expecting for coming scaled experimental demonstration, the only proofs of tritium self-sufficiency in tritium breeding systems could come from predictive modelling. Thus, tritium transfer modelling tools are key tools for the conceptual design and certification of tritium fusion systems.

Up to now, there exist no qualified nuclear tools for ITER and only QA guidelines for their development and use.

Under the partial auspices of Spanish Ministry for Industrial R&D (CDTI), advanced tritium transfer modelling tools for ITER/DEMO Plant Systems HYSYS/ASPEN+ are developed based on our large experience of Chemical Plants and Chemical Systems modelling and scientific background and tritium expertise.

Modelling routines for key unitary operations for tritium transfer isotopic processes at Plant systems: (1) burn-up, (2) permeation, (3) absortion/desorption, (4) cryodistilation, (5) catalytic exchanges; (6) electrolytic are developed and proposed.

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Operation Experience on Safety System of Tritium Process Laboratory for 28 Years in Japan Atomic Energy Agency

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The TPL (Tritium Process Laboratory) at the JAEA (Japan Atomic Energy Agency) has been the only facility, which can handle over 1 gram of tritium in Japan. Major objectives of the TPL are to establish the tritium processing and tritium safety technologies for nuclear fusion reactor. The operations of the safety systems with tritium started from March 1988 after the transportation of tritium to the TPL.

The maximum amount of tritium, which is permitted at the TPL at April 2001 is 22.2 PBq (about 63 g). The concept of the safety systems at the TPL is the triple confinements. Each confinement has own detritiation system, which is composed of adequate catalytic reactor and molecular sieve beds.

The effluent tritium removal system (ERS) for primary confinement, which receives exhaust tritiated gases from the experimental apparatuses, the negative pressure preservation in glove boxes and the vent gas from the liquid waste storage tank system. The ERS has treated 5.71×10^4 m^3 of the exhaust gases from 1988, and has exhausted the gas through a stack of the TPL after tritium elimination. The glove box-gas purification system (GPS) for second confinement, which removes tritium permeated or leaked out from experimental apparatuses in glove boxes, has processed circulatory 2.93×10^7 m^3 of the glove box-gas from 1988. The air clean-up system (ACS) for third confinement has treated 1.68×10^6 m^3 of the gas by the air glove box ventilation, and 7.29×10^5 m^3 of the gas by the local ventilation during the large component maintenances, and has exhausted the gas through a stack of the TPL after tritium elimination.

The average tritium concentration in a stream from a stack of the TPL to environment, whose flow rate was 38,800 m3/h, was 1.37×10^{-4} Bq/cm^3 during the last 10 years. The average value is calculated for every three months. This concentration is 0.0275 smaller than the tritium concentration limit for HTO in air. After the Great East Japan Earthquake, the TPL has damaged, and took about one and half years until the safety systems were operated normally. The above results at the TPL have been thus much valuable database for a design and maintenances of safety systems of the tritium handling facilities. This report will summarizes the operation results of the safety systems at the TPL for 28 years. The maintenance and improvement works for the systems are reported with failure data of several main components of the systems.

CODAC Prototyping for Dynamic Tritum Mass-Balance Control Demostration in TBM Systems

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The confidence on the achievement within the present ITER R&D decade of "active" tritium diagnostic solution opens, in parallel with tritium dynamic transfers proven "predictive" modelling skills (today also under development and qualification), a gate to a new continuous inventory and operation tritium control strategy not only for TBM systems and for the overall tritium plant systems.

Discrete mass-balance tritium management strategy provides time-to-time tritium mass-balance checks through integrated inventory measurements in a measuring devoted system (TMS). Drawbacks of static approach relate to a constrained system operation flexibility, on a poor tritium safe control and on an inherent difficulty to provide tritium self-sufficiency demonstration, a key committed TBM Programme mission. In opposition, dynamic mass-balance tritium management approaches are considered to simultaneously provide reliable system operation flexibility, to contribute to a robust tritium management and control safety approaches and finally to support the required kinetic mass balance for tritium self-sufficiency demonstration. Similarly to the dynamic control of any other specimen in a general plant, in

particular isotopes in nuclear facilities, the robustness of a continuous dynamic tritium control and management relies on three pillars:

I) the achievement of continuous tritium atmosphere monitoring and T-concentration diagnostic probes at TBM effluents within the ranges of accuracy for pre-established required tritium mass-balance matching;

II) the availability of performant predictive tritium system models capable to anticipate tritium concentration and in-components/sub-systems inventories within the final (or assumed) range of accuracy of tritium ambiance monitors and tritium forms sensing solutions in effluents;

III) the final development of a proven and qualified TBM CODAC Plant System architecture fitting ITER CODAC tiers, layers and interlocks configuration and global requirements.

This paper explores dynamic "control" solution targeting in-prototype hardware realization
Dynamic. control strategy for tritium needs to be substantiated, developed, implemented, globally qualified (benchmarked) and certified according to ITER nuclear standards and regulations.

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Design of Tritium Handling Facility for Tests of ITER SDS Components

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The tritium handling facility is designed and fabricated at KHNP CRI in 2016. This facility is used for tests of the ITER SDS beds. This system consists of TRS (Tritium Recovery System), PGRS (Pure gas recombiner system), TADS (Tritium Assay and Dispensation System), TSS (Tritium Supply System), two glove boxes and three air-purged-enclosures. TADS and TSS using high level tritium are fabricated in the nitrogen glove boxes. TRS, PGRS and GC using relatively lower level tritium are fabricated in the air-purged-enclosures. Leak rate as one of key design requirements of glove box is less than 0.001% volume of GB per hour. 99% of tritium leaked out from TADS or TSS into GB is recovered for 8 hours by TRS. Detritiation rate of TRS is 95%. Ultimate concentration of tritium in GB becomes less than 1 mCi/m^3 after TRS operation. Total 7.4 PBq of tritium and 1.85 PBq per batch can be used in this facility. TSS is operated with a metering tank, a metal bellows reservoir, a furnace and two uranium beds for temporary storage and clean-up. In this facility, a new method of metering and dispensing tritium is introduced. A bellows reservoir is additionally employed to improve the operability of conventional PVT-c measurement system. This reservoir has a variable volume controlled by a bidirectional plunger, which is moved by a stepping motor with a high precision laser distance sensor. This tritium handling facility is utilized for the functional test and performance assessment of the storage beds developed by NFRI and KAERI.

The Structure Design of Producing Tritium Blanket for Fusion-Fission Hybrid Reactor Driven by Z-Pinch

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Nowadays, energy is the foundation of modern society development and the restriction factor of economy development. For the considerable long term, the China energy consumption mainly depends on coal which will produce a large of fog, powder, PM2.5, NOx and CO2 and so on and bring out considerable entironment destruction. In order to realize the low carbon society aim, the energy structure should be heavily reformed in China. In fact, the research shows fission and fusion energy are security, clear and reliable energy. However, the low uranium resource usage ratio and the complex post-disposal of lack fuel are the key problems which restrict the fission energy development, the material enduring irradiation, high energy gain of commercial producing electricity and large scale usage of tritium resource are the another key problems which restrict the fusion energy development. Therefore, the Peng xianjue academician of China Academy of Engineering Physics puts forward a new type fusion-fission hybrid reactor driven by Z-Pinch. In such system, in order to realize tritium breed and self-sustain, the structure design of producing tritium blanket is necessary to carry out for the future Demo reactor. On the basis of producing tritium blanket physics analysis, the materials which include helium gas, ZIRLO, H2O, Li4SiO4 and SS316Lβ00Cr17Ni14Mo2βstainless steel are chosen for such producing tritium blanket. Thereinto, the helium gas is used as the carrier of tritium, the ZIRLO is used as zirconium cladding, the H2O is used as the moderator, the solid ceramic Li4SiO4 is used as the tritium breed material, the SS316L stainless steel is used as the structure material. Under the support of national inertia confinement fusion energy research project, using Pro/e CAD software, the producing tritium blanket model is established for the next finite element analysis work

Key words: Z-Pinch, fusion-fission hybrid reactor, producing tritium blanket, structure design

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Finite Element Simulation of Liquid Deuterium-Tritium Inside a Capsule with Fill-Tube

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The inertial confinement fusion (ICF) design requires smooth and uniform deuterium-tritium (D-T) ice layers in a spherical shell. The first important step in the growth of high quality ice layers is the nucleation of crystalline seeds. For a perfect capsule without defects on the inner surface, formation of seed crystal is, to a great extent, governed by the thermal profile in the capsule. A finite element model for liquid D-T inside capsule shell suspended at the center of a spherical layer by a fill-tube is presented to analyze the thermal and hydrodynamic characteristic in the cryogenic target. The simulation is conducted within the framework of the steady compressible flow with heat conduction and mass transfer model. The simulation results show that the presence of fill-tube encourages the coldest spot on the inner surface of the shell, which usually means the nucleation position, to migrate from the equatorial plane of the shell to the top. The pressure of rare tamping helium gas, which is related to the thermal conductivity, also has a significant influence on the temperature profile in the target, and therefore the formation of seed crystal. The calculated results provide some theoretical explanations for a variety of experimental observations.

Analysis of Tritium Migration and Proposed Capture/Removal Methods in Fluoride Salt-Cooled High-Temperature Reactors (FHRs)

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The Fluoride salt-cooled High-temperature Reactor (FHR) is a candidate generation IV nuclear reactor class. FHRs utilize a molten fluoride salt, flibe (Li_2BeF_4), as a coolant. Flibe is an attractive coolant with advantages such as a high boiling point and a high resistance to breakdown under irradiation. The ^{6}Li in unenriched flibe has a high cross section to thermal neutrons producing ^{3}H. It has been estimated that the tritium production for an operating FHR is 2400 Ci/day, which is higher than comparable LWR technologies however lower than HWR technologies. FHRs operate at high temperatures in which H and its isotopes readily permeate through structural materials. It is necessary to understand where this tritium migrates and permeates throughout the primary reactor boundary in order to prevent its release. A model has been developed to track tritium production and predicted migration and permeation pathways. Preliminary consensus is major sinks for the tritium generated within an FHR includes graphite reflector pebbles in the core and heat exchangers used for power conversion and decay heat removal. In order to mitigate tritium release, methods of tritium sequestration and capture are being investigated. Current candidate methods under further investigation include tritium capture in nuclear grade carbon, tritium “gettering”, and inert gas sparging. Preliminary assessments have been made on the efficiency of these methods. At UNM, research is ongoing to demonstrate the efficiency of inert gas sparging can be greatly increased by the application of ultrasonic energy. In a preliminary experimental investigation using simulant fluids, a sonomechanically-enhanced sparging approach resulted in a 70% reduction in degassing time. In addition, UNM is researching candidate tritium “getters”, permeation barriers, and sweeping fluids for use in double-wall heat exchangers.

Using Atmosphere-Forest Measurements to Examine the Potential for Reduced Downwind Dose

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A 2-D atmospheric dispersion model was developed to address how airborne plumes interact with the forest at Savannah River Site. Parameters describing turbulence and mixing of the atmosphere within and just above the forest were estimated using measurements of water vapor concentration made at the Aiken AmeriFlux tower for a range of stability and seasonal conditions. Water vapor is an ideal component for this study due to its similarity to tritium oxide in regards to its interactions with vegetation and the environment.

Stability periods when the greatest amount of mixing of an airborne plume into the forest were found for 1) very unstable environments, when atmospheric turbulence is usually at a maximum, and 2) very stable environments, when the plume concentration at the forest top is at a maximum and small amounts of turbulent mixing can move a substantial portion of the plume into the forest. Plume interactions with the forest during stable periods are of particular importance because these conditions are usually considered the worst-case scenario for downwind effects from a plume. The pattern of plume mixing into the forest was similar during the year except during summer when the amount of plume mixed into the forest was nearly negligible for all but stable periods.

If the model results indicating increased deposition into the forest during stable conditions can be confirmed, it would allow for a reduction in the limitations that restrict facility operations while maintaining conservative estimates for downwind effects. Continuing work is planned to confirm these results as well as estimate specific deposition velocity values for use in toolbox models used in regulatory roles.

Uncertainties in Dynamic Modelling of Environmental Tritium Transfer

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Impact assessment of tritium is required for design, licensing, emergency preparedness and management and consequently, the associated dynamic models for tritium transfer into the environment are subject to constraints regarding their robustness, moderate conservatism, and uncertainty. The uncertainty of models predictions including both routine and accidental releases of tritium were studied in international research programmes coordinated by IAEA in the last ten years emphasising that the doses in accidental emissions have unacceptable large uncertainties, especially for rain and night conditions. The case of organically bound tritium (OBT) during the night time is the subject of another contribution at this conference.

In the present study, the steps for an improved model for tritium accidents are discussed together with uncertainty analysis of the model’s structure and parameters. For wet deposition of tritium during the accident, uncertainty is reduced using recent results for pesticide spray on crops. The authors’ model regarding the dynamics of OBT in animals is improved by considering the role of protein turnover in maintenance metabolism and recent results on carbon half-times after ^{12,13}C diet shits. The CROPTRIT model previously developed by the authors for tritium transfer as tritiated water (HTO) and OBT in crops is improved. The source of OBT production in plants is leaf and the canopy exchange velocity is commonly involved in photosynthesis, air to leaf transport, and transpiration flow from roots as coupled processes. Consequently, the model time steps is adjusted considering plant adaptation to the changes in environmental conditions and to the availability of input meteorological conditions (averages between ten minutes and one hour). Beside the isotopic fractionation of tritium in plants, the processes involved in tritium transfer are the same with those used in crop growth models, but using a shorter time step (ten minutes up to one hour). Only the partition of newly formed dry matter and OBT are considered with a time step of a day. The meteorological conditions before, during and after the accident affect the tritium dynamics in soil and root water and consequently, the soil sub-model uses specific space grid, root properties and a fast solution for the coupled equations of water, HTO, and root uptake. The vapour and liquid phases in soil are considered and the importance of thermal gradients in soil is scrutinised.

The model uncertainty coming from model parameters and alternate sub-models for photosynthesis, transpiration, and soil are analysed for winter wheat. The model is favourable tested for winter wheat where the experimental data base for tritium is large. For other crops as maize and pasture, the experimental data base for tritium is sparse and the key model parameters are defined together with their influence on the uncertainty and sensitivity study.

Determination of In Vitro Lung Solubility and Intake-to-Dose Conversion Factor for Tritiated 13X Zeolite

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Special Tritium Compounds (STCs) are defined as: “Any compound, except for H2O, that contains tritium, either intentionally or inadvertently.” The Savannah River Tritium Enterprise (SRTE) relies on many different materials to process tritium. These STCs can retain various amounts of tritium in particulate form and present special radiological protection challenges different than those of elemental tritium or tritiated water. These challenges can be divided into three areas: detection, physical and chemical behavior, and radiation emission, any of which can complicate deconvolution of bioassay results.

Uptakes of elemental or tritiated water are readily distributed throughout and released from the body and can be detected by normal bioassay sampling. Inhalation of STCs may or may not be detected, depending on how rapidly tritium evolves from the material inhaled. Tritium evolution from inhaled particulate occurs through either through dissolution of the host material or by diffusion of the tritium from the intact host material. For simplicity, in this discussion the release of tritium from a tritide by either mechanism is referred to as "dissolution." Materials that readily release tritium are classified as “soluble” and result in lower DCFs than materials that retain tritium. Once released from a soluble material, the tritium is flushed from the body via normal metabolic processes. Materials classified as insoluble allow tritium beta decay energy to deliver overall higher dose to a localized area.

A sample of tritiated 13X zeolite, similar to that used at the Savannah River Tritium Enterprise (SRTE), was analyzed to estimate the particle size distribution and the rate at which tritium evolves from this material in a simulated lung environment. This information was used to calculate the committed effective dose a worker would receive after inhaling tritiated 13X. These doses, calculated as a function of particle size using the same methodology given in the DOE Tritium Handbook, are presented as inhalation intake-to-dose conversion factors (DCF). The DCF for tritiated 13X zeolite particulate with an Activity Mean Aerodynamic Diameter (AMAD) of 5 μm was determined to be 1.11E-11 Sv/Bq for this material, indicating that tritium in a urine bioassay can be interpreted as if it had resulted from an intake of HTO if the worker was potentially exposed to any mixture of HTO and 13X.

Consequences of Different Anthropogenic Sources on Organically Bound Tritium Concentrations in Watersystems

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Naturally occurring in the environment, due to its cosmogenic or telluric origin, tritium was also considerably produced in the past by nuclear weapons atmospheric testing which led to tritium concentrations in rains more than one hundred times above its natural levels during the peaking emissions in 1963 (UNSCEAR, 2000). Besides, tritium is historically one of the radionuclides that are the most released by nuclear power plants in France (Gontier et Siclet, 2011).
Due to these various source terms tritium is currently ubiquitous in the environment, encountered as gaseous forms (HT, CH_3T), liquid forms (free water (HTO), Tissue Free Water Tritium (TFWT)) or associated with organic molecules in a more or less exchangeable way (Exchangeable Organically Bound Tritium (E-OBT), Non Exchangeable Organically Bound Tritium (NE-OBT)). These two last forms are integrated both within the water cycle and the carbon cycle through biological or other microbial processes (Diabaté and Strack, 1997). Thus their behaviour and fate within the environment are partially governed by the organic matter biodegradation cycle.

Several studies indicate a disequilibrium between tritium bound and free forms, especially for various components of aquatic or terrestrial systems (Clark et al., 2010; McCubbin et al., 2001) and more precisely the possible role of the origin and quality of organic matter on this disequilibrium within a same compartment (Eyrolle-Boyer et al., 2014). The delayed transfer of terrigenous organic matter previously contaminated by atmospheric fallout or other anthropogenic sources of tritium, towards aquatic systems would explain this common disequilibrium.

To demonstrate this hypothesis two approaches will be presented. The first one aims at comparing OBT/HTO ratio measured in samples collected in areas located far from the influence of the releases from nuclear facilities and that are only impacted by global fallout. Two kinds of areas will be considered in this case, areas most impacted by global fallout (maximum rainfall) and areas less impacted located in the south of France under dry Mediterranean climate.

The second approach consists in the sampling campaign of various components (litter, soils, aquatic plants,, sediments…) from terrestrial and aquatic systems located close to nuclear fuel reprocessing plant currently in the process of dismantling, in order to compare the surrounding tritium levels for the various forms.

The preliminary results from these studies will be presented together with comparison of tritiated water levels from different compartments of the hydrosphere (atmosphere/rivers/marine waters) in order to highlight the huge range of variability of HTO between atmosphere (rainfall), continental surfaces (river water) and marine areas (sea water).

Tritium Levels in Milk in the Vicinity of Chronic Tritium Releases

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Tritium can be integrated into most biological molecules. Even though its radiotoxicity is weak, the effects of tritium could be increased following concentration in critical compartments of living organisms. For a better understanding of tritium circulation in the environment and to highlight transfer constants between compartments, we studied the tritiation of different agricultural matrices chronically exposed to tritium.

Milk is one of the most frequently monitored foodstuffs in the vicinity of points known for chronic release of radionuclides firstly because dairy products find their way into most homes but also because it integrates deposition over large areas at a local scale. It is a food which contains all the main nutrients, especially proteins, carbohydrates and lipids. We thus studied the tritium levels of milk in chronic exposure conditions by comparing the tritiation of the main hydrogenated components of milk, first, component by component, then, sample by sample. Significant correlations were found between the specific activities of drinking water and free water of milk as well as between the tritium levels of cattle feed dry matter and of the main organic components of milk. Our findings stress the importance of the metabolism on the distribution of tritium in the different compartments.

The present results highlight certain relationships that link milk components with external sources (air, feed, and drinking water) and their effects on the isotopic ratios of the non-exchangeable fraction of lipid, casein and acid whey dry matter.

Exposure to Tritium: Dangerous or Not?

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The French nuclear safety authority (ASN) decided to set up several think-tanks due to the unexpected tritium concentrations measured in the local fauna in Cardiff Bay (UK) resulting from uncontrolled discharges of tritiated molecules, and the forecasts of ever-increasing tritiated discharges from French civilian facilities. These think-tanks comprising experts, nuclear operators, stakeholders and associations were given the task of thoroughly analyzing data on tritium bioaccumulation in the different biological compartments and of assessing the effects of tritium on our health and the environment.

The work of these think-tanks led to a book entitled “Livre blanc du tritium” (Tritium White Book), which revealed shortcomings in the fields of metrology, environmental monitoring and tritium discharge management, but above all in assessing the impact of tritium on human health. Recommendations in terms of safety, monitoring and research have since been made and new avenues of research have also been put forward to the government.

This paper provides an overview of the issues identified for ongoing research and for future regulation.

Tritium Concentration in Carrot Plant after Short-Term Exposure to Atmospheric HTO, Huifang Shen
(Shanxi Agricultural Univ)

Tritium Concentration in Carrot Plant after Short-Term Exposure to Atmospheric HTO

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In order to study HTO uptake and OBT formation in carrots following a short atmospheric HTO release, potted carrot plants were exposed to evaporated HTO for 1h in an exposure box at four different growth stages from the early to late growth stages. The concentration of tissue free water tritium (TFWT) and organically bound tritium (OBT) in leaves and root were measured at the end of exposure and at harvest, respectively.

The results showed that the percentages of the leaf TFWT concentration at the end of exposure to the estimated equilibrium concentrations were 71.81%, 116.3%, 132.84% and 139.11% at four different growth stages. These results suggest that in most cases, it usually takes shorter than 1 h for the TFWT concentrations in the carrot leaves to reach squasi-equilibrium with the HTO concentration in the ambient air moisture. In the leaves of carrot, the exposure at the later growth stage generally ended with a higher TFWT concentration at the end of exposure, whereas such a tendency was just contrary in the roots of carrot. The earlier stage exposure gave rise to lower TFWT concentrations at the harvest of the leaves and roots. During the period between the exposure and harvest, the TFWT concentrations decreased by factors of 86—389 for the leaves and 3—116 for the roots. After exposure at the vigorous growth stage for the leaves and roots, the OBT concentration were greatest at the end of exposure and at harvest of both parts. At the earlier growth stage exposure, the OBT concentration in the leaves was higher than that of in the roots at harvest. But the later stage exposure leaded to higher OBT concentration in the roots at the harvest. At harvest, the relative OBT concentration from the fourth exposure was lower than that from any other exposure by a factor of about three in the radish top. In the root, the relative OBT concentration from the first exposure was two to four times higher than that from any other exposure. The difference of the relative OBT concentration in root between at the end of exposure and at harvest was not so significant. The ratio of between OBT and TFWT for the roots at harvest was from 3 to 21 depending on the exposure times. The OBT-to-TFWT ratio for the leaves was higher than that for the root in the first exposure, whereas the ratios for roots were higher than that for leaves in the other exposures. The radiological importance of OBT for the carrot roots would generally be much greater than that of TFWT.

The selection of parameters and improvement for tritium concentration prediction model in plant should be based on the different growth stages for accidental release of atmospheric HTO so that the estimation model for tritium could be estimated accurately and objectively.

Revealing the Behavior of Gas Species in Materials with In Situ TEM

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Service lifetimes of tritium-containing materials are directly related to microstructural effects caused by tritium decay and corresponding ^{3}He accumulation. Understanding the processes by which ^{3}He alters hydrogen isotope and bubble evolution at the microstructural level is crucial to understanding and controlling the changes manifested at the bulk level. Accelerated testing methodologies that enable rapid screening and avoid hazards associated with tritium are also desirable. Techniques that resolve structural changes at small length scales, such as transmission electron microscopy (TEM), can reveal the microstructural origins of these behaviors.

The *In Situ* Ion Irradiation TEM (I^{3}TEM) facility at Sandia National Laboratories enables accelerated studies of hydrides and gases in materials with nanometer resolution. Two particle accelerators are attached to a TEM, so a sample can be irradiated and observed concurrently. Ions from a 5 kV Colutron stop within the sample; implanted ^{4}He concentrations can rapidly reach levels that would require years of natural tritium aging, while deuterium can be implanted to alter hydrogen isotope concentrations. More energetic ions from a 6 MV Tandem produce greater amounts of displacement damage. Additionally, a variety of sample holders can manipulate the local specimen environment, including heating to 1000 °C, quantitative mechanical loading, and gas/liquid environmental cells. The TEM has sub-nm resolution, and a recently added diffraction-based automated orientation mapping capability quantifies the local crystallographic orientation with resolution of a few nm. The combination of irradiation, local environment manipulation, and TEM observation enables studies of microstructural evolution in metals containing hydrogen isotopes and gas bubbles in complex environments with unprecedented resolution.

This talk presents work with model face- and body-centered-cubic (fcc and bcc) and hexagonal-close-packed (hcp) metal systems under ion implantation, heating, and gas flow conditions. First, bubble growth and coalescence were captured during *in situ* heating after ^{4}He implantation. We also show differing microstructural damage resulting from multiple concurrent beam in situ irradiation in which combinations of ^{4}He and deuterium were implanted, while an energetic Au beam simultaneously produced large amount of displacement damage. Finally, the effects of a changing local gas environment on hydride structure will be shown. Sandia National Laboratories is a multi-program laboratory managed and operated by Sandia Corporation, a wholly owned subsidiary of Lockheed Martin Corporation, for the U.S. Department of Energy's National Nuclear Security Administration under contract DE-AC04-94AL85000.

Paper 6A-2: Tuesday, April 19

Low Temperature Thermal Water Splitting Using Nanostructured Metal Oxides

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Thermochemical water splitting cycles (TCWS) based on the repeatable reduction/oxidation reactions of nanostructured metal oxides (MOs) are investigated as possible replacements for the current process used in the Tritium Extraction Facility (TEF) at Savannah River Site, which relies on single use Mg beds. TCWS is a simple two-step cycle, where in the first step the MO is reduced by heating to an elevated temperature. The second step occurs at a lower temperature, where the reduced MO is re-oxidized by water, producing H2 through water splitting:

(1) $MO_{oxidized}$ + Thermal Energy β$MO_{reduced}$ + (1/2)O_2

(2) $MO_{reduced}$ + H_2O + Thermal Energy β$MO_{oxidized}$ + H_2

The combination of the regeneration and water splitting steps creates one thermochemical cycle, where the only inputs are water vapor and thermal energy, and only outputs are oxygen and hydrogen gas. A distinct advantage of the TCWS cycle is that H_2 and O_2 are evolved during temporally-separated stages, allowing for independent processing of the outputs and avoiding combustible mixtures of H_2 and O_2.

While the simplicity of the process is certainly attractive, there are two challenges that must be addressed before TCWS can be considered as a replacement for Mg beds in the current Z-bed recovery system. The first challenge is that the endothermic reduction of bulk MOs typically occurs at very high temperatures (>1200 °C). The second challenge is that the high temperature of the reduction step causes sintering of MO media, which gradually reduces the efficiency of the TCWS cycles over time. It is important to note that these two challenges are directly related to the large reduction temperatures of normal MO materials. However, new nanostructured MO materials have recently been discovered that avoid these difficulties because they have significantly lower reduction and oxidation temperatures. These new materials make TCWS an excellent candidate for replacing the Mg beds in glovebox stripping systems. This report describes recent investigations into using nanostructured MO materials for TCWS at lower temperatures.

A Framework for Evaluating the Effect of Radiation on the Diffusivity of Hydrogen Isotopes in a Metal Lattice

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Irradiation enhanced permeation of hydrogen isotopes in austenitic steels has been observed in several experiments. The exact mechanisms are not well understood, such as whether the dominate contributor is neutron displacement damage, ionizing radiation in the lattice, alloy segregation, defects, or bypass diffusion of the lattice via grain boundaries and dislocations. Surface reaction rates can also be an important phenomenon for transport rate whether due to surface contamination or surface reaction rate kinetics. However, this is beyond the scope of this report. In order to assess these potential mechanisms, a permeation model is proposed to address the potential for radiation enhanced diffusion in a lattice due to ionizing gamma (γ) radiation. Use of transition state theory (TST) and density functional theory (DFT) is made to define the relationship. The intent is to develop a more advanced model for diffusion in a lattice to allow for investigating and solving the more complex phenomenon associated with hydrogen isotope permeation in real metals under irradiation. While the primary focus is diffusivity, solubility and permeability are also discussed. This would be just a first step for evaluating radiation enhanced diffusion of hydrogen isotopes which must include additional phenomenon, including neutron (n) damage.

Ionizing γ radiation will impact the lattice electron jump frequency and number of free electrons, the hydrogen isotope vibrational energy and jump frequency, the lattice diffusion activation energy, and potentially the enthalpy of solution. The classic Arrhenius relationship for permeability lumps these phenomenon into an activation energy and pre-exponential. The activation energy is comprised of the activation energy of diffusion and the enthalpy of solution. A constant pre-exponential represents the jump distance squared times jump frequency. Using TST and DFT these can be decomposed into more fundamental definitions such as zero point and saddle point energies, jump paths and Fermi energy that provide greater insight into the potential for enhanced diffusion/permeability due to an ionizing radiation. A relationship is presented using these more fundamental definitions and an evaluation of the potential impact on hydrogen isotope diffusion/permeability in a lattice.

The Impact of Hydrophobicity of Stainless-Steel Surfaces on Tritium Inventories

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Previous studies have shown that a significant fraction of the tritium inventory in a metal sample accumulates in the surface water layers.1 All metals support a surface water layer. The tritium concentration in this water layer is significantly higher than that present in the metal bulk. This tritium concentration controls the rate at which tritons enter or leave the metal bulk.2 Hydrophobicity of the surface controls the amount of surface water present on the sample. This study examines the consequences of altering the hydrophobicity of a surface on the tritium inventory present in the surface water layers and on the rate of triton migration into and out of the metal bulk.

A set of 30 stainless-steel (type 316) samples has been prepared for this study. The surfaces have been skimmed to expose bulk metal by removing any surface alterations that may have occurred during fabrication. All samples were initially degreased with acetone. Subsequently, the samples were soaked in varying concentrations of HNO3 to change the hydrophobicity of the stainless-steel surfaces. The samples were tritiated by exposure to one atmosphere, 50/50, DT gas at room temperature for 24 h and then stored under a helium cover gas. The total uptake of tritium was measured using linear thermal desorption. The fraction of tritium present in the surface water layer was measured using a pulsed RF low-temperature plasma.2

The paper outlines the details of the process used to modify the hydrophobicity of the stainless-steel surface, correlates the total quantity of tritium absorbed with hydrophobicity, and discusses the impact of hydrophobicity on the tritium concentrations achievable in the water layers present on the stainless-steel surface.

This material is based upon work supported by the Department of Energy National Nuclear Security Administration under Award Number DE-NA0001944, the University of Rochester, and the New York State Energy Research and Development Authority. The support of DOE does not constitute an endorsement by DOE of the views expressed in this article.

1. R.-D. Penzhorn et al., Fusion Sci. Technol. 64, 45 (2013).
2. M. Sharpe, W. T. Shmayda, and W. U. Schröder, "Tritium Migration to the Surfaces of Stainless-Steel 316, Aluminum 6061, and Oxygen-Free, High-Conductivity Copper," submitted to Fusion Science and Technology

Characterization of Tritium Permeation Profiles via Helium-3 Analyses

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The TMIST-2 experiment evaluated in-reactor tritium permeation through a 316 stainless steel (316SS) sample tube at two different temperatures (i.e., 292° and 330°C) and three different tritium partial pressures (i.e., 0.1, 5, and 50 Pa). TMIST-2 was conducted in an instrumented lead assembly within the Advanced Test Reactor (ATR) at Idaho National Laboratory (INL), which permitted in-situ measurement of tritium permeation. Following irradiation, the test assembly was returned to Pacific Northwest National Laboratory (PNNL) for subsequent post-irradiation examination (PIE). Although numerous examinations were performed, there was significant interest in developing a method for evaluating the tritium permeation profile along the axial length of the sample. It was hoped such a method would help verify that the majority of tritium was permeating from the central region of the sample, held at a uniform temperature by virtue of the irradiation capsule design and active temperature control. This portion of the sample was designated as the “active” permeation region.

Because Helium-3 (^{3}He) is a decay product of tritium, ^{3}He analyses were performed along the axial length of the sample tube in an attempt to infer the tritium permeation profile. In order to restrict permeation to the active region of the tube where the temperature gradient was minimal (< ±5°C), the irradiation capsule design specified that the cooler ends of the sample tube would also be sputter-coated with aluminum. Cooler temperatures and the presence of a coating with greater permeation resistance to tritium were expected to minimize extraneous tritium permeation from the sample ends, which could potentially confound permeation results from the active permeation region of the sample tube. During PIE activities at PNNL, the 316SS tube was sampled axially for 3He analyses between the sputter-coated and active permeation regions of the sample. Samples were collected by machining adjacent rings with a lathe over this transition zone and each ring was then then subjected to the 3He measurement technique.

Results of the ^{3}He measurements will be presented within the context of the in-reactor permeation test, TMIST-2, which was conducted in the ATR at INL. The methodology used to obtain the measurements, isotope-dilution mass spectrometry, during PIE at PNNL will be described in detail.

Overview of Fuel Retention in the JET ITER-Like Wall

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The JET tokamak operates with a metal wall consisting of predominantly beryllium in the main chamber and tungsten in the divertor. The choice of wall materials in JET reflects that of the ITER tokamak and is so called the JET ITER-Like Wall (JET ILW). One motivation for the selection of these materials in tokamaks is to reduce the retention of fuel (i.e., hydrogen isotopes) in machine components compared with carbon (C) based machines. This is largely achieved by eliminating the hydrogen-carbon chemical sputtering mechanism which dominates fuel retention in C machines through co-deposition of fuel constituents with eroded C and long range migration of volatile hydrocarbons. To date the global fuel retention and retention rates following deuterium (D) operations in the metal JET-ILW is assessed at ~ 0.2% of injected D atoms and 4.9×10^{18} D/s respectively – at least an order of magnitude lower than the preceding C configuration. The assessment of fuel retention and retention rates is needed to evaluate the expected tritium (T) inventory of JET following T and DT operations commencing in 2018 for safety and waste liability purposes.

An overview of fuel retention for the JET-ILW is presented. The retention results are drawn from a range of surface analysis techniques of components removed from JET which have been analysed as part of a two year European wide EUROfusion project. The *post-mortem* analyses for individual components provide fuel retention data for plasma facing surfaces, recessed surfaces and also areas remote from the plasma; each of which are dominated by different retention processes. The results presented summarise retention in these different regions and include the latest information on retention in tile gaps. From the area specific results the global retention and retention rates are derived and an evaluation of future T retention provided.

*See the Appendix of F. Romanelli et al., Proceedings of the 25th IAEA Fusion Energy Conference 2014, Saint Petersburg, Russia

Paper 6B-1: Tuesday, April 19

Decontamination of Proton Exchange Membranes After Tritium Exposure

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Canadian Nuclear Laboratories (CNL) has an active collaboration with Tyne Engineering Inc. (Tyne) to develop a Proton Exchange Membrane-based (PEM) electrolyser intended for tritium removal from water using the Combined Electrolysis and Catalytic Exchange (CECE) process. To better understand the tritium compatibility of PEMs, some membrane samples were exposed to various levels of tritium by soaking the membranes in tritiated water for extended periods of time. In this study, commercially available Nafion® N 1110 membranes were exposed to a tritiated water (with a β activity of about 1000 Ci/L) which was made in the Tritium Facility at CNL. Two equivalent batches of Nafion® N 1110 membranes (each with a 4 cm x 4 cm dimension) received β-doses of 67 kGy and 155 kGy, respectively.

The exposed membranes were then decontaminated for characterization and testing. A few different decontamination methods were experimentally studied. These methods can be categorized as water flushing and chemical soaking. It has been found that the measured tritium concentrations in elution water decreased quickly in the first 30 days followed by a slow decay afterwards until reaching a plateau after about 100 days. The received dose did not appear to have significant impact on tritium removal from the membrane during water flushing. Chemical soaking proved to be more effective than the water flushing method. Table 1 shows the liquid scintillation count results of β-doses released from the tritium-exposed membranes. The single step chemical soaking of Nafion® N 1110 can release at least 50 times more tritium than the water flushing method. The results and experience reported in this paper may be relevant to other tritium decontamination activities.

Table 1
Tritium Removed by Water Flushing vs. Chemical Soaking

Membrane Sample	Final Elution of Water Flushing (Bq)	One Step of Chemical Soaking (Bq)
Nafion® N-1110 at a β-Dose of 67 kGy	7 000	371 000
Nafion® N-1110 at a β-Dose of 155 kGy	32 500	255 000

Comparison of Solutions to Reduce the Tritium Inventory in Purely Tritiated Metallic Waste

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During operation and decommissioning of fusion facilities, radioactive waste will be generated. Most of these wastes are expected to be tritiated and will require a specific management strategy taking into account the physical and chemical properties of tritium. The reference solution for tritiated waste that cannot be accepted directly for disposal is a 50-year interim storage in order to allow for tritium decay (The 50-year intermediate storage corresponds to 4 tritium radioactive periods i.e. a tritium reduction by a factor 16). The most tritiated categories may need to be processed using a detritiation technique applied to the radwaste for reducing the tritium content or tritium outgassing as an alternative or as a complement to an interim storage phase.

The detritiation of metallic waste by thermal treatment or melting has been investigated and the main characteristics are described in this paper. Melting provides a better detritiation factor than thermal treatment. A comparative analysis has been also performed taking into account several independent parameters related to the environment, safety and technical feasibility aspects. For purely tritiated metallic waste made from stainless steel, in any case, performing a detritiation appears to be very attractive, allowing a significant decrease of interim storage duration.

Paper 6B-3: Tuesday, April 19

Developments in Robust Radioanalytical Techniques for the Determination of ^{3}H in Decommissioning Wastes and Environmental Matrices

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Tritium is ubiquitous in and around nuclear plants, being formed via neutron capture by ^{2}H, ^{6}Li, ^{10}B and ^{14}N and via ternary fission. The highly mobile nature of ^{3}H species results in widespread distribution of the radionuclide. Predictive modelling of ^{3}H activity concentrations is challenging and direct measurement of ^{3}H activities in materials is the preferred approach to underpin waste and environmental assessments. The UK nuclear industry is currently engaged in a significant programme of site decommissioning resulting in a high demand for the rapid characterisation of ^{3}H in a diverse range of matrices, including concretes, metals, plastics, sludges, resins, soils and biota. To support such assessments, it has been necessary to develop dedicated instrumentation in parallel with robust radioanalytical methodologies; namely a multi-tube furnace and a high-capacity closed oxygen combustion system. Data are presented on the development and validation of these instruments, designed specifically to enable the quantitative extraction of ^{3}H from diverse sample types. Furthermore the furnace system has been employed as a tool to gain insights into the ^{3}H association in decommissioning and environmental matrices exposed to ^{3}H arising from nuclear power plant operations through thermal evolution profiling (figure 1). The impact of the chemical speciation of ^{3}H on analytical strategy is discussed. A major benefit of the multi-sample furnace is its ease of use and applicability to ^{3}H determination in virtually any sample type. The complementary oxygen combustion system has been evaluated for the quantitative oxidation of organic-rich samples (e.g. wood, plastic, oil, biota) and data on its effectiveness will be presented.

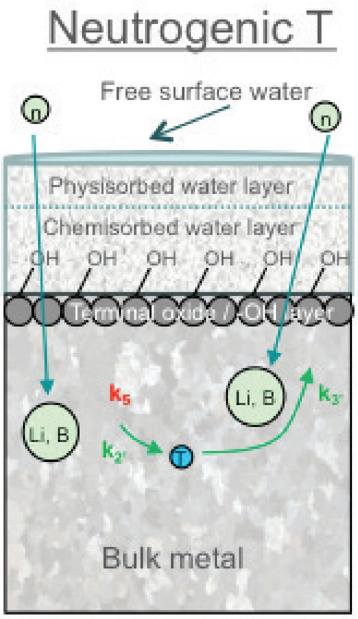

Figure 1 – Schematic indicating one possible illustration of 3H origin and speciation in metals.

Paper 6B-4: Tuesday, April 19

H-3 Measurement in Radioactive Wastes: Efficiency of the Pyrolysis Method to Extract Tritium from Aqueous Effluent, Oil and Concrete

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Due to its labile and chemical behavior (organic or inorganic forms), tritium is one of the most difficult radionuclide to measure. However for D&D projects concerning tritium facilities, stakeholder considers its characterization as a priority.

Liquid Scintillation Counting (LSC) is the reference method for tritium measurement. In the case of solid sample usually found in D&D project, the pyrolysis method is generally used prior to measurement in order to isolate tritium from the matrix and separate it from possible interfering species such as C-14 or S-36. The furnace used possesses two independent heating zones. The first one, containing the sample, can be set up (temperature gradient) whereas the second one is maintained at 950°C. The sample is placed into the cold furnace in porcelain boats after a precise weighing and is swept by a nitrogen flow during the heat-up period. In the second zone, O_2 is added to the stream of N_2 in order to guarantee the complete oxidation of the gases released during the pyrolysis taking place in the first zone. As the temperature rises, extracted gases flow through a bubbler filled with HCl 0.1M to fix H-3. The bubbler is systematically cooled for a better trapping efficiency.

As Certified Reference Materials (CRM) are rather scarce for nuclear waste materials, our analysis are validated through proficiency tests and with homemade materials (concrete, oil). Different matrices were investigated, and for each we determined the tritium recovery rate after pyrolysis. We will present the outcomes of the temperature program optimization to obtain the best tritium extraction from the tested matrices (liquid wastes (oil and aqueous effluent) and solid wastes (concrete)). For optimization, we also used the available CRMs proficiency tests for aqueous samples and intercomparison between laboratories for oil. For concrete, a specific study has been conducted in order to identify the influence of the particle size and of the sample quantity on H-3 recovery after pyrolysis.

Paper 6B-5: Tuesday, April 19

Tritium Recovery from Mixed Waste

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Nuclear Sources and Services (NSSI) in Houston, TX houses a unique infrastructure to recover tritium from a variety of different streams for re-use. NSSI accepts gaseous, aqueous, and tritiated-solid feedstock. For example, titanium-tritide foils from spent neutron generator sources are detritiated at 1000oC. Tritium gas is recovered from a variety of metal getter beds and the emptied beds are returned to service. Tritiated liquid that is desorbed from zeolite drier beds or generated while reconditioning nickel beds is transferred to a Combined Electrolysis Catalytic Exchange (CECE) system to concentrate the tritium fraction before the gas is transferred to a hydrogen isotope separator for enrichment. Mixed-waste liquids, typically generated in the pharmaceutical industry, are converted to tritiated water and inactive gaseous effluents using an EPA approved, high-temperature, two-stage, catalytic-oxidation destruction process. The destruction efficiency of solvents is extremely high, typically in the 99.999% to 99.9999% range.[1] The resultant water containing HTO is also directed to the CECE subsystem. At present, liquid activities ranging from the 10 µCi/L to 20 Ci/L can be processed economically. The CECE subsystem has detritiated 2 metric tonnes of heavy water per year and is designed to detritiated up to 30 metric tones of light water per year. NSSI is licensed to operate within a 2-gram tritium inventory limit.

Recently the Mixed-Waste Oxidation Facility that can process up to 40 liters of liquid per day has been upgraded to reduce the corrosive effects of chlorine-based mixed wastes on the effluent scrubbing system. This paper will outline the key features of this system and discuss the effectiveness of the upgrades and the operational parameter space.

Leaching Tests of the Tritium Wastes Immobilized in New Cement Mixtures

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The study was undertaken to determine the immobilization performance of low-level tritium waste in new cement mixtures. The grouts analyzed are fine mortars, fluid, self-compacting, with flowability, able to fill the smallest gaps in the solid waste container. The grouts curing (28 days minimum) showed compact and structural integrity without pores and cracks, so as to ensure radioactive tritium wastes encapsulation.

Two representative types of solid/liquid tritium radioactive wastes with known tritium activity have been prepared: one type containing hydrophilic tritium compounds and the other hydrophobic tritium compounds. The tritium activity was presumed to have a uniform distribution inside the waste.
The tritium wastes were mixed with three different types of grout, with water/cement ratio 0.4, and poured in a cylindrical mould with a diameter of 2 cm and a length of 10 cm. Superplasticizers were added in the cement mixtures composition.

The leaching of tritium from cement samples of cylindrical shape in an aqueous environment has been studied.

Samples from the liquid surrounding the cements blocks were collected at 1, 3, 5, 7, 14, 30 and monthly thereafter and placed in the liquid scintillation cocktail and measured. The quantity of tritium released was then calculated

The experimental set-up and the results of the experiments data on the leaching rate are given in the paper. The potential of the new cement mixtures to be use as immobilizations matrixes for low and intermediate tritium wastes intermediate storage is discussed.

Influence of Surface Roughness and Gold Plating on the Adsorption and Absorption of Tritium into Stainless Steel (Type 316)

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The Fluoride salt-cooled High-temperature Reactor (FHR) is a Generation-IV molten salt reactor. It combines the advantages of the Sodium Fast Reactor (SFR) and the High-Temperature Gas-cooled Reactor (HTGR). However, tritium generation from neutron activation of the primary coolant FLiBe (a mixture of molten LiF and $BeF2$) is a significant issue in current development of FHRs. As estimated, a 2400 MWth FHR during startup will generate as high as 5000 Ci of tritium per day. For comparison, a 1000 MWe PWR only generates 1.9 Ci of tritium per day. Currently a three-loop FHR is under consideration with an extra loop for tritium removal. As the cost and increased size brought by the additional loop offsets the advantages of FHR, a tritium control system that applied to a two-loop FHR is investigated in this study.

Tritium control and mitigation in FHRs consists of three major tasks. The first one is the amount control or elimination of TF, which is highly corrosive to structural materials. Second is the extraction of tritium from the primary coolant. It would be beneficial if the tritium extracted is collected for further use. Third is the prevention of tritium permeation out of the primary loop through intermediate heat exchanger or structural materials.

A successful tritium removal facility needs to meet three requirements: large mass transfer area to volume ratio, mixing of molten salt to enhance tritium diffusion in the molten salt and large tritium concentration gradient on the mass transfer surface to provide sufficient diffusion driving force.

In this paper, a tritium control and mitigation system is presented. It consists of three major facilities: the redox control facility, the tritium removal facility and the IHX with tritium permeation barrier. In addition, $Al2O3$ coating as a surface treatment on structural materials is applied as needed to prevent potential tritium leakage. The designs of the facilities are developed and optimized via computational simulation using COMSOL Multiphysics and MATLAB.

Keywords: FHR, Tritium control

Paper 7A-2: Tuesday, April 19

Electrochemical and Thermal Treatments for Stainless Steel Passivation

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Tritium reacts many containment vessel surfaces containing water, hydrocarbons, and other protium bound materials. In addition, it can diffuse into the containment vessel and exchange with alloying elements and create undesirable compounds. Subsurface elements may also diffuse out of the vessel and dilute the isotopic composition of the gas. Savannah River Tritium Enterprise (SRTE) uses a proprietary surface treatment that is intended to render the surface inert. Unfortunately, this process is somewhat an art and not a process based on scientific understanding. SRTE has funded a project that will determine the effects of vacuum thermal treatments and electropolishing on surface passivation, in terms of rendering the surface inert, of stainless steel. A series of experiments will be discussed that will optimize the acid concentration, current density, and other electrochemical process variables for stainless steel. The surfaces will be analyzed using Laser Induced Breakdown Spectroscopy (LIBS), Auger Electron Spectroscopy (AES), Atomic Force Microscopy (AFM), and Scanning Electron Microscopy (SEM). Novel techniques to characterize the passive layers will also be developed. Finally, gas sample bottles will be loaded with protium and deuterium mixtures to determine the relative exchange characteristics of the treated vessels. This comparison is generally valid and applicable to tritium service since previous work has demonstrated that if the gas mix does not dissociate in protium and deuterium then the treatment is generally stable for the tritium environment.

Paper 7A-3: Tuesday, April 19

Molecular Dynamics Simulation on Hydrogen Trap Effects by Vacancy Clusters in bcc-Fe

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Tritium retention and leakage behaviors in/through reactor materials such as tungsten and steels are relevant with the safety and economics of fusion reactors. In order to accurately predict the tritium behaviors during reactor operation, understanding of interactions between hydrogen isotopes and irradiation defects is needed. Since vacancies basically act as strong traps of hydrogen in metals, many studies have been devoted to quantify the trap effects of vacancies, especially of mono-vacancies. However, the knowledge on vacancy clusters is still limited. In the present work, therefore, with focusing on the influence of vacancy clustering, effects of hydrogen-vacancy interaction (H-V interaction) on hydrogen behaviors in metals are studied using molecular dynamics (MD) simulation. β-iron is chosen as the model system, because it is relevant with reduced activation ferritic/martensitic (RAFM) steels and a reliable potential model regarding H-V interaction is available.

All MD simulations are conducted using LAMMPS code. Systems of various conditions on hydrogen and vacancy concentrations are studied. For each condition, two sorts of initial atomic configurations are prepared: vacancies are agglomerated as a single vacancy cluster in one configuration, while vacancies are isolated as multiple mono-vacancies in the other. In order to quantify the influence of vacancy clustering on H-V interactions, the effective diffusivities of hydrogen and the H/V ratios in H-V complexes are evaluated. Then, the evaluation results are compared between the system with a single vacancy cluster and the system with multiple mono-vacancies.

The MD simulations clearly indicate strong trap effects of vacancies. The H/V ratio in H-V complex is lowered by vacancy clustering when the concentration of hydrogen in the system is high. It means that vacancy clusters have weaker trapping effects than isolated vacancies, if the concentration of vacancies is fixed. As a general trend, the H/V ratio becomes smaller as the size of vacancy cluster becomes larger. On the other hand, when hydrogen concentration in the system is low, less significant influence of vacancy clustering is observed. We construct a kinetic model based on the rate theory to explain these observations and validate the model in comparison with the MD results. Isotope effects in H-V interactions are also quantified by performing the MD simulations with (i) hydrogen atoms, (2) tritium atoms and (3) a mixture of hydrogen and tritium atoms.

Effect of Tritium on Cracking Threshold in Aluminum 7075

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The effect of long-term exposure to tritium gas on the cracking threshold of Aluminum Alloy 7075 is being investigated. The alloy is the material of construction for a pressure vessel used to contain tritium in an accelerator designed for inelastic scattering experiments on nucleons. In general, the alloy is resistant to embrittlement by hydrogen isotopes except under specific environmental conditions. This is because the passive oxide layer acts as a barrier to hydrogen dissolution by preventing disassociation of the molecule. However, embrittlement associated with hydrogen has been observed in Aluminum 7075 in moist air, because the water molecule dissociates on the aluminum oxide surface. For aluminum in a tritium environment, monatomic tritium is available for diffusion into the alloy because of radioactive decay. Tritium that diffuses into the aluminum could lower the threshold for cracking leading to delayed failure by slow crack growth. Research on tritium effects on high strength aluminum alloys is limited and the data do not include engineering parameters such as crack growth rates and threshold stress intensity values. Measured cracking thresholds may depend on the amount of dissolved tritium, and time of exposure, because of the additional embrittlement from its radioactive decay product, helium-3. Figures or equations can be included in the abstract for the Conference Program Booklet.

The purpose of this program is to determine the likelihood of detrimental aging effects of Aluminum 7075 in tritium atmospheres and recommend design strategies for effective mitigation. Cracking thresholds are being measured by using pre-stressed fracture mechanics specimens described in ASTM E1681. The specimens were cut from the alloy in the form of compact tension specimens that could be bolt-loaded to fixed crack-opening displacement values. Twenty-eight specimens were loaded to a range of stress intensity values less than the fracture toughness value of Aluminum 7075, K_{Ic}, measured in laboratory air. The stress intensity values ranged from 45% to 90% of $K_{Ic.}$ The stressed samples were than place in a pressure vessel and held in 17.2 MPa tritium gas for up to twelve months. The samples will be examined for cracking after four, eight, and twelve months of exposure. Tests on companion specimens in moist hydrogen and air are planned to isolate the effects from hydrogen isotopes from decay helium. Post exposure characterization will be utilized to ascertain the primary mechanisms of embrittlement. The program plan and early results are reported in this study.

Paper 7A-5: Tuesday, April 19

Deuterium Diffusion and Retention Behaviors in Erbium Oxide Single Layer and Ceramic-Metal Multilayer Coatings

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Tritium permeation through structural materials is one of critical issues in fusion systems from the viewpoint of an efficient fuel cycle and radiological safety. A promising solution is to fabricate thin ceramic coatings as a tritium permeation barrier (TPB). Erbium oxide (Er2O3) coatings have been investigated intensively as TPBs and showed high permeation reduction factors of up to 105. Moreover, our previous study proved that Er2O3-metal two-layer coatings suppressed corrosion by liquid lithium-lead with their properties as TPB. In this study, deuterium permeation and retention in the coatings with one, two, and three layers were investigated to elucidate hydrogen isotope migration behavior in single and multi-layer coatings.

Er2O3 single-layer, Er2O3-Fe two-layer, and Er2O3-Fe-Er2O3 three-layer coatings were prepared on reduced activation ferritic/martensitic steel F82H substrates by physical vapor deposition methods. The layer thickness of Er2O3 and Fe was 1.2–2.6 μm and 0.8–1.0 μm, respectively. A smooth layer structure was confirmed by cross-sectional observation. Thereafter, deuterium permeation measurements were performed using a gas-driven permeation system at 500β700 ℃. Finally deuterium was introduced in the coated sample with a deuterium pressure of 8.00 × 104 Pa at 500 β700 ℃, and the sample was immediately cooled down to room temperature to keep deuterium in the sample. Deuterium depth profiles were examined using a tandem accelerator via 3He(d,p)4He nuclear reaction.

Deuterium permeability for Er2O3 and Er2O3-Fe coatings showed similar values, indicating the deuterium permeation was controlled by the Er2O3 layer. On the other hand, the permeability for the Er2O3-Fe-Er2O3 coating decreased by a factor of 104 in comparison with that for an uncoated substrate. The permeability was one order of magnitude lower than that of the Er2O3 single layer coating and one order of magnitude higher than that for an Er2O3 coated sample on both sides of the substrate, indicating two diffusion barriers contributed to higher permeation reduction, and the deuterium recombination process on the back surface was also an important factor. Deuterium depth profiles of the coatings clearly showed the relationship between deuterium concentration and grain density of the Er2O3 coating. In addition, approximately three times higher deuterium concentration was found in the three-layer coating in comparison with those in Er2O3 and Er2O3-Fe coatings, indicating an increase of tritium retention in the multilayer structure.

Paper 7A-6: Tuesday, April 19

Quantitative Analysis of Deuterium Depth Profile in FeAl-Al2O3 Tritium Permeation Barrier Coatings and its Application in Deuterium Permeation Behaviors

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It is very important to predict and control the deuterium and tritium production, permeation retention in fusion materials. The permeation and retention of D&T through the boundary of the reactor should be minimized. Tritium permeation barrier coating technology is one of the most effective methods to solve the problem of permeation and leakage of tritium in different tritium breeder blanket modules technologies. However, it is very difficult to explain the detail permeation behaviors of the FeAl-Al2O3 tritium permeation barrier coatings.

In this paper, the standard Ti-1.3%.at D and Ti-9.2%.at D samples have been prepared by quantitative absorption in pure deuterium gas. And then, a depth profile calibration of deuterium element in solid sample was established by glow discharge optical emission spectroscopy (GDOES) with "standard quantification mode". Based on this, the deuterium depth profiles for CLAM substrate, Fe-Al layer and Fe-Al/Al2O3 composite coatings were measured by GDOES. According to the data, the concentration of deuterium in Al2O3 film area with the thickness about 2μm was decreased sharply. It can be inferred that Al2O3 film was the main barriers for preventing the deuterium permeation in the composite coatings. Moreover, the important parameters for permeation behaviors, such as, surface absorption and desorption constants, diffusion coefficient of different layers were also calculated by measured the surface concentration and depth profiles of the D in Fe-Al/Al2O3 composite coatings.

Zeolite Membranes for Tritium Processing: Preliminary Inactive Results and CAPER Upgrade for Separation Experiments with Tritium

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Zeolites are well known absorbing material used as pebbles in detritiation systems. Using zeolites in the form of membrane could provide continuous operation. Such promising process has been proposed by the Tritium Laboratory Karlsruhe (TLK) to be used in the Tritium Extraction System of the solid Breeding Blanket of the fusion reactors. Experiments are carried out to identify the best zeolite membranes and measure their separation and permeation performances. Tests with inactive mixtures of H2/He and H2O/He show rather high permeances (~0.5 – 0.7 μmol/sm2Pa for H2 at 25 °C for MFI-ZSM5). Although limited separation (maximum 1.7 towards hydrogen in 0.1% - 10% H2/He at 25 °C) is measured the implementation of a membrane cascade will allow to achieve the process requirements [1].

The proof of concept of this technology for tritium process will be further evaluated using the unique features of the CAPER facility that can produce and handle dry and wet tritiated mixtures at several degrees of dilution [2]. The CAPER facility is being upgraded to accommodate the zeolite membranes and instrumentation required for the future membrane separation experiments with tritium. The test of zeolite membranes under tritiated atmosphere (Q2/He, Q2O/He, Q = H, D, T) will allow investigating possible isotopic and ageing (degradation) effects on the separation performances and permeation rates.

This paper aims to present at first a summary of the results obtained with H2/He and H2O/He mixtures in MFI-ZSM5 and NaA type zeolite membranes that are relevant for the future experiments under tritiated environment. The detailed design of the new setup (membrane and membrane module, flow - pressure - temperature controls, tritium monitoring…) to be integrated into the CAPER glovebox together with the expected process parameters and experimental plan are also presented.

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Deuterium Depleted Water from TCAP R&D Using H_2/D_2 Feed

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SRS Tritium Facility relies upon the Thermal Cycling Absorption Process (TCAP) to separate tritium from other hydrogen isotopes. HT-TCAP system performs protium-tritium separation primarily for detritiation of hydrogen gases to make stack releasable raffinate (< 5 ppm T_2). Raffinate purity is an important factor considered during development of advanced TCAP using H_2/D_2 surrogates. Conventional analytical methods for determining protium and deuterium concentration have sensitivity limitations compared to ion-chambers used for tritium measurements. It was known that the TCAP raffinate (H_2) had less than natural abundance D2 (~150 ppm) but quantitative analysis was desired. On the other hand, water isotope analysis by spectroscopic method is very sensitive and accurate for trace level D/H ratio determination. TCAP raffinate samples were collected in a hydride bed, then converted to water by oxidization, and analyzed for deuterium concentration. It confirmed that the advanced TCAP can reduce deuterium concentration from 50% in the feed to 20 ppm in the raffinate. The 25,000 $D2_2$ reduction ratio would make 2.0 ppm D_2 raffinate if the feed had only 5% D_2, or almost no deuterium if a hydrogen gas cylinder (contains ~150 ppm deuterium from natural abundance) were used as feed. The hydride bed setup, hydrogen oxidation and safety considerations will be discussed. Additionally, potential health effects of deuterium depleted water will be discussed from literature survey.

Paper 7B-3: Tuesday, April 19

Experimental and Simulated Hydrogen Isotopes Adsorption Isotherms over Zeolite Materials under Cryogenic Conditions

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Reverse Chromatography Process (RCP) is one of the most suitable processes for decontamination of HDT gases. Separation of HDT mixtures is obtained over zeolite materials under cryogenic conditions.

Design of a new material could be a way to improve the separation process. In order to identify the adsorbent c haracteristics (nature of compensation cations, geometry and opening diameter of pores, silicon/aluminium ratio…) which give the best separation, adsorption of hydrogen or deuterium has been performed by manometric experiments.

The adsorption properties of zeolites, such as faujasite or LTA type were measured in a temperature range between 40 K and 120 K. Experimental datas will be compared to the results obtained by molecular dynamics simulations, and finally correlated to the different zeolite framework. Moreover, it will be shown that isotopic effect is more pronounced when the temperature decrease.

Reduction of Glovebox Stripper System Water Loading

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In facilities containing tritium, all process equipment is contained in inerted gloveboxes operating at slightly negative pressure relative to the process rooms. The gloveboxes have recirculation systems which include a stripper system. The glovebox stripper systems capture tritium from the glovebox atmosphere to minimize facility emissions with the possibility of recovering the tritium.

Hydrogen isotopes released into the gloveboxes are converted to oxide form and removed from the glovebox atmosphere by the glovebox stripper systems – the intended function of these systems. Protiated water (and oxygen) enters the glovebox system in various ways. All water in the gloveboxes is ultimately removed by the stripper system molecular sieve beds which are then processed or disposed of as waste. The water and oxygen enter the glovebox in locations both internal and external to the gloveboxes. The majority of oxygen and water originates external to the gloveboxes in current facility operations.

This study evaluated approaches for water source reduction i.e. reducing the amount of water entering the gloveboxes. The second approach explored options to segregate or prevent the mixing of protiated water in the glovebox with the tritiated water formed as part of the tritium oxidation and capture process used to reduce facility emissions.

Hydrogen Isotopes Adsorption over Zeolite Materials: Modeling of Solid Gas Interactions by Molecular Dynamics Simulation

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Zeolites of the faujasite type are well-known porous material for the separation of hydrogen isotopes under cryogenic conditions. In these materials, heavier hydrogen isotopes are those adsorbed preferentially. This last suggest why reverse chromatography process (RCP) is one of the most suitable processes for tritium decontamination of HDT gases. However, the underlying mechanisms governing the hydrogen isotopes separation on zeolite under cryogenic conditions are not completely elucidated. On porous materials, such as zeolite, the gas separation by an adsorption process could be obtained by one or several of the following mechanisms:

- Preferential adsorption of one isotope over others. This can due to the different gas-solid interaction strength, known as the thermodynamic equilibrium effect,
- Faster adsorption of one isotope rather than other produced by different diffusing rates (kinetic effect),
- Better accessibility to the porosity of the adsorbent for one isotope given by the size exclusion (molecular sieving effect),
- Difference of diffusing rates in narrow micropores, especially for light molecules such as hydrogen isotopes (quantum sieving effect).

We modeled hydrogen isotopes adsorption by a zeolite by using a molecular dynamics (homebrew code) code. Our main concern is to understand solid-gas interactions during the separation process. Our code allowed determining the H2 and D2 adsorption isotherms at low temperatures. We show that the parameterization of the Lennard-Jones (LJ) potential parameters (for the interactions between hydrogen and the zeolite framework atoms) have a strong influence on the modeled adsorption isotherms. Moreover, the implementation of the Feynman-Hibbs correction of the LJ interaction potential allows to obtain a consistent correlation between experimental simulation results.

Paper 7B-6: Tuesday, April 19

Quantitative Study of the Radiolytic Gas from Tritiated Water Adsorbed in Zeolite 4A

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In tritium plants, the atmosphere clean-up systems, based on catalytic oxidation of tritium, produce low-activity radioactive water. An adsorption process in zeolite 4A, belonging to the class LTA, was developed for this water (less than 18.5 TBq.L-1). Self-radiolysis of tritiated water (HTO) adsorbed in zeolites 4A shows large differences compared to free-bulk water radiolysis. We studied the roles played by zeolites on the quantities of released gas. We used an experimental approach which consisted in studying the precise case of self-radiolysis for two tritiated water activities (27.8 and 60.0 TBq.L-1) for about 1000 days. Experimental data from periodic analysis of gas overhead in tanks, confirm the stable behavior of water adsorption over time.

A special care was taken on the influence of Water Loading Ratio (WLR), which is given as a percentage and represents the ratio between the mass of water and the mass of dry zeolite. It ranges between 0 and 20% of their mass. Indeed, the location of water molecules and the interaction between zeolite and water so evolve along water adsorption. As a consequence we chose to focus our experiments on the influence of Water Loading Ratio (WLR).

On one hand, we found that zeolites 4A firstly increase hydrogen release and in the same way oxygen release too. On the other hand, zeolites favor next a recombination between these two radiolytic products, with a strong dependence on their water loading ratio. Such a phenomenon, is revealed for the first time by this experiment, and doesn't occur during the free-bulk water radiolysis.

Finally, the time elapsed between the experimental starting and the beginning of the slow recombination phenomenon appears as a function of the tritiated water activity and of the water loading ratio of the zeolites.

Catalyst Evaluation for Oxidative Stripper

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Catalysts are used in the SRS Tritium glove box stripper to oxidize elemental tritium and tritiated hydrocarbons into tritiated water for subsequent removal via sorption in zeolite beds. Design criteria are set to remove 99% of tritium at 1 ppm in the glovebox. Development and evaluation of the process typically start with nonradioactive protium surrogate. This raises challenge of detecting unreacted protium at 10 ppb level. GC/DID (Discharge Ionization Detector) from GOW-MAC Instrument was identified as the instrument for this task. Catalyst performance was evaluated for H_2 and CH_4 oxidation activity at various temperatures and flow rates. Application of reaction engineering concepts will be discussed in detail such as prevention of channeling and change of reaction mechanism in order to obtain catalyst intrinsic activity. The GC/DID set up, carrier gas purification/management, calibration and maintenance will also be included.

All catalysts were active to oxidize ppm level of H2 and CH4 with initial 100% conversion, but their performance experienced some degrees of deactivation at low to moderate temperatures. The deactivation was reversible by temporarily operation at 425°C. The data supports a deactivation mechanism by moisture loading to the catalyst, which hinders further oxidation reaction on the active sites. The deactivation is reversible and can be re-activated at a higher temperature by driving off moisture. Among the catalysts evaluated, palladium catalyst outperformed the platinum catalysts.

Paper 8A-2: Tuesday, April 19

Hydrolysis of Tritiated Sodium: Advances in Understanding and Modelling of the Tritium Behaviour

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Within the framework of sodium fast reactors (SFR), some studies are carried out in order to propose a better understanding of the tritium behaviour during sodium cleaning processes implemented in handling or dismantling operations. Liquid sodium, used as a coolant in SFR, contains some impurities such as hydrogen (H,T) and oxygen which can be present in a soluble or precipitate form (NaH, NaT, Na_2O). Cleaning processes must lead to the complete disposal of reactive sodium in a safe way. Most of the sodium cleaning processes is based on the Sodium Water Reaction (SWR): this reaction leads to the production of energy and generates a liquid effluent, mainly composed of an aqueous solution of sodium hydroxide, and a gaseous effluent, mainly composed of an inert gas, hydrogen and steam. Tritium is distributed between the liquid and the gaseous effluents, and according to two different chemical forms: tritium hydride HT and tritiated water HTO. HTO being 10,000 times more radiotoxic than HT, a precise knowledge of the mechanisms governing the distribution of tritium is necessary in order to estimate the exhaust gas releases and design the process needed to treat the off-gas before its release into the environment.

Some specific hydrolysis experiments have been carried out at CEA Cadarache. Small quantities of sodium containing tritium have been hydrolyzed by water injected drop by drop. Visualization techniques such as very high-speed visualization and infrared visualization have been used to observe the reaction area more accurately. A chemical characterization of the effluents generated by the process allowed us to study the influence of several parameters on the tritium distribution.

The analysis of these results highlights an important influence of the total activity of the sodium on the tritium distribution according to its chemical form HT and HTO. Tritiated water HTO is generated in the liquid effluent and part of it is vaporized by the heat of reactions distributed in the aqueous solution of soda.
Based on these new experimental results, a phenomenological modelling of the tritiated sodium hydrolysis is proposed in this paper.

Acoustic and Structural Analysis of an Ultrasonic Cleaning System for Fuel Assemblies Containing Tritium Producing Burnable Absorbers, Gianluca Longoni, Robert O. Gates, Kenneth Ian Johnson, Kimberly A. Burns, Edward F. Love (PNNL)

Paper 8A-3: Tuesday, April 19

Preparation of Alveolate “Hydrophobic” Catalyst for Tritium Waste Gas Treatment

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The tritium handling site is often faced the problem of huge volume gas exhaust containing a little amount of tritium. Oxidation of tritium gas into tritium water by catalyst and recycling tritium water is generally used detritiation technics. Catalyst is the key technology of this technics. High air resistance in the catalyst will make air circulation difficult, so that the air resistance of catalyst should be low. The heating of a large catalyst bed is difficult, so that the catalyst should have good catalysis efficiency at room temperature. Because vapor will poison the catalyst at normal temperature, catalyst should be hydrophobic. A small amount of hydrogen is often added in the air flow to elevate the catalyzed tritium oxidation efficiency, but adding hydrogen raises the temperature of the catalyst surface, organic hydrophobic catalysts are fairly combustible, so that hydrophobic catalyst which is able to endure high temperature is needed.

To prepare a catalyst for the detritiation of waste gases at high flow rates, a heat-resistant hydrophobic zeolitic coating was synthesized on the surface of alveolate cordierite by hydrothermal processing. The support was characterized by X-ray diffraction and scanning electron microscopy. The synthetic conditions under which the zeolite layers were grown on the substrate were investigated. An alkaline environment was a key factor for successful zeolite synthesis, as well as sufficient alveolate cordierite for perfect zeolite crystals. Pretreatment of the cordierite with vitriol favorably influenced zeolite formation, and the amount of raw material also had an effect. The surface contact angle of alveolate cordierite support is about 2°, after the support is coated with hydrophobic coat, the the surface contact angle is 12°-16°.

A alveolate “hydrophobic” catalyst is prepared from the support by impregnation of Pt. At room temperature, when the space velocity is 3370h-1, the oxidation efficiency of 1% hydrogen in humid air is higher than 99%, which is much higher than Pt-Al2O3(62.6%). When the space velosity is 16000h-1, the pressure drop on the alevolate “hydrophobic” catalyst is<0.01Mpaβthe pressure drop on Pt-Al2O3 is 0.15Mpa, the conversion of H2 is 98.9% over the alevolate “hydrophobic” catalyst. After the alevolate “hydrophobic” catalyst is calcined at 300β the conversion of 1% hydrogen oxidation in room air at room temperature over the catalyst is still higher than 99% at 3370h-1.

A heat-resistant “hydrophobic” catalyst with low air resistance is prepared by hydrothermal synthesis and impregnation. The catalyst was essentially waterproof and not easily poisoned by water vapor. At room temperature, the conversion of low concentrations of H2 in air over the catalyst was very high at different space velocities and relative humidities.

Paper 8A-4: Tuesday, April 19

Ce Based Oxide Loaded Honeycomb Catalyst for Detritiation

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Because of its permeability and radioactivity, tritium must be rigorously fenced in tritium systems. Detritiation devices are required in tritium system, for example in the inner deuterium/tritium fuel cycle of ITER, GDS (Glove box detritiation system) VDS (Vent detritiation system) and ADS (Atmosphere detritiation system) are equipped. The method of catalytic oxidation and adsorption is widely used for atmosphere detritiation, but traditional particle catalyst has large gas resistance, which limited the space velocity for detritiation. Honeycomb catalyst can enormously increase the gas handling capacity due to its low pressure drop and high dispersity of active ingredients, but has not been used in detritiation so far. For glove box detritiation, metal getter is usually used, but it is easily envenomed by oxygen. Ce based oxide loaded catalysis is a candidate material for glove box detritiation, because Ce based oxide can release O during the catalytic reaction, when the inlet gas is hypoxic.

Ce based oxide CeO2, Ce0.7Zr0.3O2 (CZ), Ce0.65Zr0.25M0.1O2 (CZM M=La Y) are prepared by sol-gel, and the capability of redox is tested by H2-TPR and O2-TPO. Pt/CeO2, Pt/CZ and Pt/CZM are prepared by impregnating, and the catalytic activities are evaluated by H2 conversion rate in gas mixture (with different H2/O2 ratio and various flow rates). The catalyst is deposited on the honeycomb substrate of cordierite using ultrasonic technology. Catalytic activities of honeycomb catalysts are evaluated by H2 conversion rate in gas mixture with different H2 concentration and various flow rates. The results indicated that doping of Zr La and Y in CeO2 improved the redox and oxygen storage capacity of CeO2. The loading of Pt decreases the reduction temperature to about 200β which is important for the application in GDS. The conversions of hydrogen was 100% at room temperature when oxygen is ample, at 200βwhen oxygen is deficient. The gas space velocity is up to 6×104 h-1 with honeycomb catalyst, comparing to the particle catalysts of about 1000 h-1. The results suggest that Ce based oxide loaded honeycomb catalyst can be used both in GDS and ADS with high space velocity and high tritium conversion rate even if the oxygen is deficient.

Basic Absorption/Desorption Experiments on Depleted Uranium Bed for Tritium Storage

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For safe and efficient operation of Storage and Delivery System in tritium plant for fusion fuel cycle, metal hydride bed for recovery and delivery of hydrogen isotope gas is essential. DU (depleted uranium) has no degradation due to tritium aging. And, the decomposition pressure of DU is flat almost all the way through complete decomposition. It has potential for high pressure delivery. Because of these reasons, depleted uranium is one of the strong candidates as tritium storage material for fusion fuel cycle in spite of its radioactivity and limitation of handling.

Because metal hydride materials expand upon hydrogen absorption, determining the process vessel (PV) expansion stresses to ensure adequate wall thicknesses is an important aspect of hydride bed development. This paper details the hydriding-induced wall stress evaluation on a prototype FISH bed as a major step toward qualifying this bed design for tritium service. The results revealed that the design modifications to improve FISH bed capabilities and performance did not result in a measurable PV stress increase. The maximum tensile wall stress measured at high hydrogen loadings (H/M>0.7) was determined to be <50% of the ASME VIII-I allowable limit for 316L stainless steel and minimal hydriding-induced wall stress was observed in the optimal operating range of LANA.85. The strain gages exhibited typical hysteresis upon gas loading and unloading, in which the wall stress accumulated gradually upon absorption but was rapidly relieved upon incremental desorption. The results described herein were in good agreement with previously reported studies on similar hydride storage beds.

Paper 8B-2: Tuesday, April 19

Measurement of Uranium Hydride Storage Bed Engineering Parameters

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Metal hydride getter beds will be used to store, measure and deliver tritium and are therefore a critical part of the fuel cycle of ITER. A metal hydride getter bed called Self-assaying Tritium Accountability and Containment Unit for ITER (STACI) was previously designed and built at Los Alamos National Laboratory as part of the ITER project to study bed static self-assaying performance. Results of those tests were reported previously [1, 2, 3]. The storage bed contains approximately 5 kg of depleted uranium (DU) distributed uniformly in two copper lattices consisting of twenty one vertically oriented compartments each. The simplicity of this design employs bed segments that are small in size and does not require radiation shields. The operating capacity of the bed is 150 grams of tritium, with a total capacity of 200 grams tritium. High sensitivity measurement of the bed temperature is accomplished by twelve thermocouples distributed uniformly about the outer surface of the primary container. Tritium measurement accuracy is estimated at 1% at 150 grams tritium.

There is a need for other engineering data beyond the self-assaying data that were collected, and the STACI bed is well-suited to this task.LANL performed several tests with hydrogen to measure individual hydride bed performance characteristics. Following activation of the bed, tests were performed to measure bed capacity, hydriding/dehydriding rates, pressure drop across the bed (before and after repeated hydriding/dehydriding cycles), cooling rates and to evaluate uranium containment effectiveness. Radiographs were taken before bed activation and after several hydriding/dehydriding cycles. Results of these tests are reported here. These engineering data will be used to support construction of hydride storage beds for the ITER Storage and Delivery System (SDS).

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[2] Nasise, J. E., C. R. Walthers and R. W. Basinger; “A Self-Assaying Tritium Storage Bed For ITER”; Fusion Technology Vol. 28, pp. 1055-1060 (1995).
[3] STACI bed mechanical drawings, 1998, ITER UID NANJB8.

Hydriding-Induced Wall Stress Evaluation on a Four-Inch SHort (FISH) Tritium Hydride Bed

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The Savannah River Site (SRS) Tritium Facilities (TF) is home to the largest metal hydride-based tritium facility in the world. Metal hydrides allow for safe and compact absorption, storage, and desorption of hydrogen isotopes at various points in the tritium processing cycle. The Four-Inch SHort (FISH) tritium hydride bed is the third generation of metal hydride storage beds designed for use in the TF. This hydride bed uses a $LaNi4.15Al0.85$ (LANA.85) alloy in which the aluminum content was varied slightly from that used in previous generation beds to achieve optimal absorption/desorption pressures. The advanced design features of the FISH bed significantly reduce costs by eliminating the need for a Hot and Cold Nitrogen (HCN) system, simplifying bed activation and installation processes, and improving bed performance and efficiency, In-Bed Accountability (IBA) calibration capabilities, end-of-life removal, and potential He-3 recovery.

Because metal hydride materials expand upon hydrogen absorption, determining the process vessel (PV) expansion stresses to ensure adequate wall thicknesses is an important aspect of hydride bed development. This paper details the hydriding-induced wall stress evaluation on a prototype FISH bed as a major step toward qualifying this bed design for tritium service. The results revealed that the design modifications to improve FISH bed capabilities and performance did not result in a measurable PV stress increase. The maximum tensile wall stress measured at high hydrogen loadings ($H/M>0.7$) was determined to be <50% of the ASME VIII-I allowable limit for 316L stainless steel and minimal hydriding-induced wall stress was observed in the optimal operating range of LANA.85. The strain gages exhibited typical hysteresis upon gas loading and unloading, in which the wall stress accumulated gradually upon absorption but was rapidly relieved upon incremental desorption. The results described herein were in good agreement with previously reported studies on similar hydride storage beds.

Optimal Design and Fuel Inventory of the Multi-bed Storage System of the SDS Considering Start-up and Shut-down Operation of the Tokamak

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In our previous work, we developed a mathematical model for optimal design of the multi-bed storage system of the SDS with assumptions that the burn and dwell operation of the Tokamak is repeated endlessly, and the details of Tokamak demand pulses are identical. However, in real world, the Tokamak has a start-up and shut-down period when different demand specification occurs compared to normal fueling operation. The start-up and shut-down operation can affect the number of getter beds of the SDS. In this study, we developed an algorithm to find optimal number of getter beds considering start-up and shut-down operation using our previous design model. Moreover, this algorithm can estimate the minimum inventory of tritium or deuterium in a getter bed to keep supply fueling until shut-down. The inductive operation mode of the Tokamak is presented to illustrate the applicability of the proposed model.

Operational Maintenance Philosophy for Tritium Processing Systems

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The high contamination potential of the release of radioactive tritium facilitates the demand for and development of a stringent and comprehensive approach to operational maintenance of tritium systems. Prompt and efficient maintenance is necessary to ensure the accepted operational safety basis is adhered to and a continued safe state of operation is achieved. This will help to mitigate and avoid potential hazards that result from a tritium release to the public and facility personnel. Because of the hazards associated with a release of tritium contamination the process systems are in large kept within a series of inerted glovebox environments that must be maintained to keep structural integrity. The nature of the tritium release from a glovebox could have significant consequences for the general public as well as for personnel. As such, the maintenance philosophy is developed to help facilitate operations in the adherence to the facility's safety code of conduct.

To effectively facilitate the safe operation goals mentioned a well-defined maintenance philosophy has been developed that encompasses routine and non-routine maintenance activities. Examples of routine activities include preventative maintenance such as line-break inspections, helium leak tests to ensure components are leak tight, weld inspections and overall surveillance testing of essential components and infrastructure. Predictive maintenance also falls into this category. Predictive maintenance activities are developed over time in response to non-routine maintenance work. Non-routine maintenance or corrective maintenance activities are performed in response to a specific failure or to resolve a particular inadequacy in performance of tritium systems. When corrective maintenance is performed trends are often studied and more predictive maintenance can be scheduled to compensate for more routine failures.

This paper will identify key operational maintenance considerations which when applied, will ensure that tritium handling systems are operated safely. Several operational experiences and lessons learned will be discussed that have shaped the operational maintenance philosophy over the years.

Paper 9A-2: Tuesday, April 19

Management of Tritium in NIF

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The National Ignition Facility (NIF) is the world's highest-energy laser and is designed to implode target capsules filled with deuterium and tritium gas mixtures. NIF targets and associated gas lines contain up to 7 TBq of tritium that post-shot are contained within a 10 meter spherical target chamber. In addition to the main target chamber, NIF has numerous diagnostic, positioners and laser optic assemblies that are exposed to tritium. After a target shot, target positioner and diagnostics are withdrawn from the target chamber for workers to manually reconfigure in preparation for the next shot.

This paper discusses our strategy of managing tritium in the target chamber and within target positioner and diagnostic assemblies after laser implosions following commencement of tritium operations in September of 2010. Each diagnostic and positioner assembly can be isolated from the target chamber and has its own vacuum and ventilation systems. The majority of tritium is captured by the Tritium Processing System (TPS) which converts molecular tritium in the vacuum pump exhaust streams to HTO and captures it on a molecular sieve bed. The TPS is capable of scrubbing 14 m^3/min of exhaust using two parallel skids, each skid having lead and lag molecular sieve vessels with a capacity of 27 litres of water per vessel.

Residual tritium within the position and diagnostic vessels is scrubbed by purging with either dry or moist air to levels that allow the unit to be accessed by workers. Simple diagnostic and positioner changes occur in the target bay by workers accessing the vessel through ventilated openings. More significant modifications requiring removal of components are transferred to a work area dedicated to handling contaminated material.

Contaminated components removed from service are processed in the Hazardous Material Management Area (HMMA), a 650 m^2 processing area configured for decontamination, inspection and refurbishment. The HMMA has fume hoods, walk-in enclosures, ultrasonic cleaners and ventilated cabinets for working with items ranging in size from small target assemblies to large 5 meter diagnostic packages.

Tritium Plasma Experiment Upgrade for Enhancing Tritium PMI Science

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The Tritium Plasma Experiment (TPE) is a unique high-flux linear plasma device that can handle beryllium, tritium, and neutron-irradiated plasma facing materials, and is the only existing device dedicated to directly study tritium retention and permeation in neutron-irradiated materials with tritium [1,2]. The plasma-material-interaction (PMI) determines a boundary condition for diffusing tritium into bulk PFCs, and the tritium PMI is crucial for enhancing fundamental sciences that dictate tritium fuel cycles and safety, and are of high importance to a Fusion Nuclear Science Facility (FNSF) and demonstration fusion power plant (DEMO).

Recently the TPE has undergone major upgrades in its electrical and control systems. New DC power supplies and a new control center enable remote plasma operations from outside of the tritium contamination area, minimizing the possible exposure risk with tritium and beryllium. We discuss the electrical upgrade, enhanced operational safety, improved plasma performance, and development of an optical spectrometer system. This upgrade not only improves worker occupational safety, but also enhances TPE plasma performance to better simulate extreme plasma-material conditions expected in ITER, FNSF, and DEMO.

[1] M. Shimada et.al., Rev. Sci. Instru. 82 (2011) 083503
[2] M. Shimada, et.al., Nucl. Fusion 55 (2015) 013008

This work was prepared for the U.S. Department of Energy, Office of Fusion Energy Sciences, under the DOE Idaho Field Office contract number DE-AC07-05ID14517.

Paper 9A-4: Tuesday, April 19

Recent Upgrades at the Safety and Tritium Applied Research Facility

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This paper gives a brief overview of the Safety and Tritium Applied Research (STAR) facility operated by the Fusion Safety Program (FSP) at the Idaho National Laboratory (INL). FSP researchers use the STAR facility to carry out experiments in tritium permeation and retention in various fusion materials, including wall armor tile materials. FSP researchers also perform other experimentation as well to support safety assessment in fusion development. This lab, in its present two-building configuration, has been in operation for over ten years. The main experiments at STAR are briefly described. This paper discusses recent work to enhance personnel safety at the facility. The STAR facility is a Department of Energy less than hazard category 3 facility; the personnel safety approach calls for ventilation and tritium monitoring for radiation protection. The tritium areas of STAR have about 6 air changes per hour, with air flow being once through and then routed to the facility vent stack. Additional radiation monitoring has been installed to read the laboratory room air where experiments with tritium are conducted. These ion chambers and bubblers are used to verify that no tritium is present in these experiment rooms. Standby electrical power has been added to the facility fan blower so that proper ventilation will now continue to operate during commercial power outages.

This work was funded by the US DOE Office of Fusion Energy Sciences under the DOE Idaho Field Office contract DE-AC07-05ID14517 with the Idaho National Laboratory.

Direct Gas-Filling Technology Based on Thermal Gradient for Deuterium-Tritium Cryogenic Target with Micro Fill-Tube

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Inertial confinement fusion (ICF) has been seen as the most hopeful solution for future world's energy demands. Actually, ICF has been come true by fuel target implosion. The majority of current designs of ICF fuel target refer to a thin-walled spherical capsule with a diameter of 1~2mm. Within the sphere capsule, a uniform condensed hydrogen fuel layer is formed at the thick of 100~200μm. Improvement of filling accuracy and formation of uniform condensed hydrogen fuel layer in ICF target has been a major challenge for target fabrication.

The gas-filling technology has been developed based on temperature difference between fuel reservoir and target cell, by controlling the temperature respectively for cryogenic target with fill-tube. The effect of temperature gradient on fuel filling processes with different size target has been studied by theoretical calculation and experiment. These results suggest that the initial fill pressure difference is slight with respect to different final temperature for fuel reservoir. In other words, it is suitable for gas-filling process for target with any diameter under control by temperature gradient.

One can note that the fuel injectivity is more accuracy for the wider range temperature gradient between the reservoir and target cell, with the target diameter expansion. The 3μm/K sensitivity for the 1.6mL reservoir at the operation temperature of 75K is quite adequate for layer thickness control purposes, with regard to the target at the diameter of 2mm. These results could afford to an important foundation for highly accuracy of fuel injectivity of cryogenic target.

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Keywords: temperature controlled independently; temperature gradient; cryogenic target; filling processes; injectiveity under control

Paper 9B-1: Tuesday, April 19

Direct Decomposition Processing of Tritiated Methane by Helium RF Plasma

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In the fuel cycle system of a fusion reactor, it is preferable that tritium is extracted from tritiated hydrocarbon as DT or T_2 because tritium is injected into fusion plasma as DT or T_2. Plasma decomposition reaction of hydrocarbon has been utilized for the production of carbon based materials such as carbon nanotube. It has been observed by the present authors using a flow-type plasma reactor that CH_4 is effectively decomposed in helium plasma [1]. In the decomposition process of CH_4, some amounts of other hydrocarbon and carbon particles and carbon deposits including hydrogen will be generated. In order to adopt a plasma reactor into tritium extraction system, understanding of hydrogen and carbon behavior is important from viewpoints of processing efficiency and tritium safety. In our previous work, a flow-type plasma reactor utilizing capacitively coupled plasma followed by the double tube permeator of Pd-Ag, which is shown in Fig.1, was developed and tritium extraction from tritiated methane in CH_4/He gas was demonstrated [2]. The apparent decomposition rate of tritiated methane was smaller than that of methane. Moreover, the amount of extracted tritium by Pd-Ag was smaller than expected. Because the concentration of tritiated methane in these experiments was very lower than that of CH_4, generated hydrogen, carbon and a small amount of water vapor was considered to be involved in tritium behavior in plasma and Pd-Ag surface. In this study, in order to understand tritium behavior in the plasma reactor, the Pd-Ag permeator was bypassed and component concentrations in the outlet gas of plasma reactor were investigated.

The typical concentration of components in the input helium gas was 1000ppm CH_4, 50ppm H_2, 100Bq/cc CH_3T, 4Bq/cc HT, 4Bq/cc HTO. The flow rate was set to be 116, 47 or 23 cc/min. Total gas pressure in the plasma reactor was adjusted to be 590Pa. Concentrations of CH_4 and H_2 in the outlet gas was measured by a gas chromatograph. Concentrations of tritiated hydrocarbon, HT and HTO were measured by oxidation-water bubbler system. Plasma density was measured to be 5×10^{14} m^{-3} by the Langmuir probe method.

Fig.2 shows comparisons of conversion rate of CH_3T and selectivity of HT with and without the Pd-Ag permeator. CH_3T conversion ratio and HT selectivity without Pd-Ag were 1.7 times greater than that with Pd-Ag. This result suggests that HT and carbon based particles generated in plasma was transferred to the permeator and react on the Pd-Ag and then CH_3T was generated. In the presentation, tritium retention in carbon deposits in the plasma reactor will be also discussed.

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[2] K.Katayama, et al., Fusion Sci. and Technol. 60 (2011) 1379-1382.

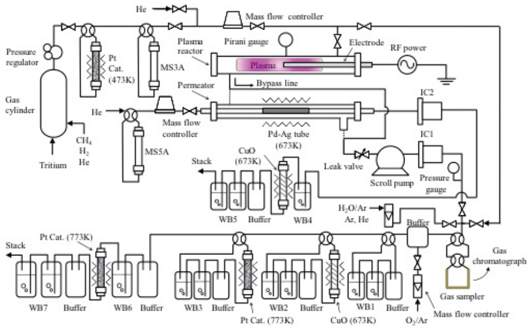

Fig.1

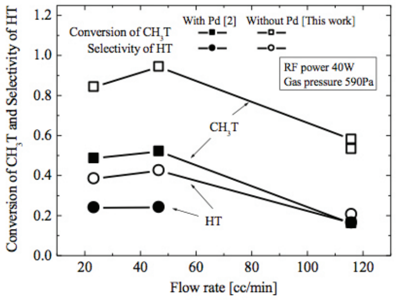

Fig.2 Conversion ratio of CH3T and selectivity of HT in the plasma reactor with or without Pd-Ag permeator.

Paper 9B-2: Tuesday, April 19

Short Way Separation of D/T from He with Superpermeable Membranes in the Post-ITER Devices

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Due to well-known limitations on the content of He in the plasma, the exhaust of fusion reactor will mainly consist of unburned fuel, including tritium. If all this exhaust mixture is directed to the tritium plant for its separation from helium and other impurities (as planned in ITER), this leads to an unacceptable tritium inventory in the post-ITER devices (e.g. DEMO).

To solve the problem it would be necessary to separate the major portion of D/T mixture from He in the immediate vicinity of the divertor output, compress it there and direct it back into the plasma by the shortest way passing the tritium plant.

The membranes superpermeable to suprathermal hydrogen particles can be applied for the realization of such "short way separation" scheme in future fusion machines and first of all in DEMO. Suprathermal hydrogen particles whose energy (kinetic, chemical or internal) exceeds ≈1 eV are able to pass through the superpermeable membrane (SPM) almost like an opening of the same area. SPM separates the permeating hydrogen from any impurities including He as well as automatically compresses it by orders of magnitude.

The "Prometheus" setup (Sarov) was specially designed for large scale SPM studies with tritium. The selective pumping of hydrogen isotopes (H/D/T) with the speed of ≈1 m^3/s followed by automatic compression was demonstrated with SPM made of V using the "Prometeus" setup [1].

The combination of SPM with the plasma generation of suprathermal hydrogen appears to be preferable for post-ITER devices and such scheme was successfully tested in a number of laboratories in technical scale [2, 3]. Taking into account these results, the SPM-plasma experiments in favor of DEMO must be mainly purposed for modeling, testing, optimization and scaling of possible arrangements for the short way refueling (rather than for the basic studies [4]). The "Prometheus" facility is available for such R&D and open for the international cooperation.

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4. T. Giegerich and C. Day, 12th Int. Workshop on Hydrogen Isotopes in Fusion Reactor Materials, June 2 – 4, Toyama, Japan

Paper 9B-3: Tuesday, April 19

Evaluation of the Effects of Impurities on SAES® ST198 Hydrogen Gettering

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Various getter materials are used in the processing of hydrogen isotopes and associated impurities. SAES® ST198 is a zirconium-iron alloy that is typically used for the removal of low levels of hydrogen isotopes from a process gas stream. However, various impurities may be present in process gas streams and some of these impurities may have a deleterious effect on the hydrogen absorption capabilities of the ST198.

A series of experiments were completed to determine the effects of various impurities on the hydrogen gettering ability of the ST198 as a function of the bed operating temperature. Changes in hydrogen getter performance were tracked using the analysis of Residual Gas Analyzer data. Baseline conditions of 0.1% hydrogen within a nitrogen rich stream were evaluated at both 350°C and ambient temperature conditions (24°C). Various concentrations of impurities were also explored to determine the effects on the hydrogen gettering of ST198. It has been determined that one benefit of ST198 is that it shows no appreciable interaction with N_2. However, gas impurities of carbon monoxide, methane, and ammonia were shown in this work to have an effect on the hydrogen gettering abilities of ST198.

This paper presents the findings relating to the evaluation of the effect of carbon monoxide, ammonia, and methane impurities on the hydrogen gettering ability of the ST198. Lower operating temperature conditions made the ST198 getter bed more susceptible to deactivation in the presence of impurities. In the event that the studied impurities exist in the process gas stream, the ST198 material could possibly become deactivated towards hydrogen isotope absorption at lower operating temperatures.

Paper 9B-4: Tuesday, April 19

Experimental Performance Test of Key Components of the KATRIN Outer Tritium Loop

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One of the main tasks of the Tritium Laboratory Karlsruhe (TLK) in the next years is to operate the Windowless Gaseous Tritium Source (WGTS) of the Karlsruhe Tritium Neutrino (KATRIN) experiment. The aim of KATRIN is an absolute measurement of the neutrino mass by precise spectroscopy of the tritium βspectrum close to its maximum energy of 18.6 keV. While the inner loop system of KATRIN has the task to provide a stabilized tritium throughput of 40 g day-1 in the WGTS, the outer loop is required for tritium clean-up, purification, and accountancy before reinjection of the gas into the inner loop. It is essential to maintain an isotopic purity of >95% of tritium in the inner loop. For this reason, the performance and interaction of key components of the outer loop have been investigated in a dedicated test as shown in figure 1.

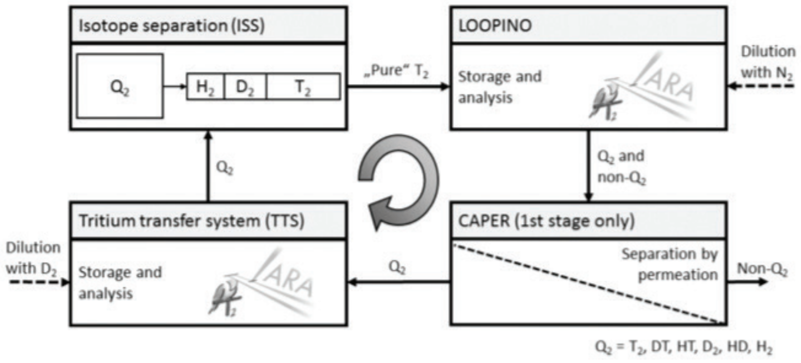

Fig.1 Schematic of the gas processing steps in the performance test.

Three different scenarios of processing tests were performed for tritium supply and exhaust removal. To reach a quantitative conclusion with regard to tritium inventories and stability of gas composition, Laser Raman (LARA) spectroscopy was employed.

The paper will describe the gas processing tests which were done with tritium batches of approximately 1 mol each and 20 mol in total. This amount of tritium was processed on a day to day basis in the TLK tritium loop, while LARA measurements were performed. Results will be given on total tritium turnover rate compared to total loss rate, recoverable loss rate and irrecoverable tritium loss rate. Furthermore the stability of gas composition during operation of the tritium loop systems will be presented.

A Review of Separation of Hydrogen Isotopes by Cryogenic Distillation in CAEP

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Study of separation of hydrogen isotopes by cryogenic distillation started from scratch in 2003 in CAEP. Over the past ten years, research efforts have been focused on both theoretical simulation together with experimental tests.

A simulation package has been developed under MATLAB platform. Based on dynamic stage model, this package can simulate separation performance of all the six hydrogen isotopes. It is especially designed for muti-column separation system and those with side-stream isotope exchange reactions for HD,HT and DT. Combined with front-end process like LPCE (Liquid Phase Catalytic Exchange), a conceptual design is proposed and analysis is completed for heavy water detritiation and upgrade.

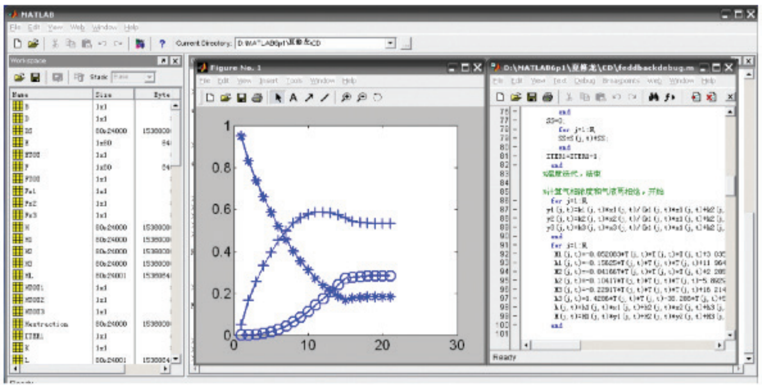

Fig.1 Simulation package for hydrogen isotopes by CD

In a design of D-T fuel cycle of CFETR(China Fusion Experimental Test Reactor),a compact 3-column concept is also recommended together with a He/D_2 tritium extraction carrier gas for TBM.

Experimental separation tests have been carried out since 2006. In 2014, H_2/HT separation is carried out with tritium recovery efficiency greater than 0.98.

H/T Isotopic Exchange: Free Water vs Adsorbed Water on Zeolite

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Self-radiolysis of tritiated water adsorbed on 4A zeolite is different than free bulk water radiolysis. Presence of zeolite favour recombination of radiolytic products (hydrogen and oxygen). To study if this phenomenon is influenced by tritium concentration, we need to synthesize tritiated water adsorbed on zeolite with different concentrations.

Two different methods were performed to obtain tritiated water adsorbed on zeolite.

Tritiated water is synthesized thanks to exposure between free light water and tritium. H/T isotopic exchange between water and gas was followed during three months. At last, the activity obtained for tritiated water was about 1,600 Ci/L. This water was then adsorbed on zeolite.

On the second one, study was carried out by exposing adsorbed water to tritium gas. Activity of adsorbed water on zeolite has reached a level of 50,000Ci/L after 5 months exposure. The adsorbed tritiated water was then obtained in situ.

Kinetic of the H/T isotopic exchange observed in those two adsorbed tritiated water synthesis will be discussed.

Comparison of Gas Species Adsorptions on 4A and 13X Zeolites

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An experimental work was carried out for studying adsorption and desorption of several gas species on 4A and 13X zeolites. The purpose of the study was to optimize the use of these kind of zeolites in our applies. Indeed, these ones could be a source of impurities if molecules desorption occur inside processes.

In literature, gas adsorption on zeolite was widely studied though some mechanisms remain unexplained. Adsorption and desorption depend of the kind of zeolite, suppliers, pressure, temperature and others molecules adsorbed on zeolites.

An instrumented process was designed to enhance understanding of adsorption on 4A (1/8 UOP Honeywell Company) and 13X (CECA Siliporite) zeolites. Experimental results show that argon, nitrogen, oxygen, methane were adsorbed on 4A zeolite at room temperature. It was also demonstrated that nitrogen and oxygen were adsorbed on 13X zeolite at room temperature. Results will be presented through the session.

These results allow a better understanding of gas adsorption and desorption on zeolites previously cited.

Experimental Investigation of ZrCo Getter Beds as Candidate Process for the Tritium Extraction Systems of the European Test Blanket Modules

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ZrCo is a well-known tritium storage material and has been applied in various research and laboratory facilities [1]. The most interesting properties with regards to the thermodynamic of the ZrCo-H system are the very low H_2 partial pressure of $ZrCoH_3$ at room temperature and the ease to reach sufficiently high temperature to completely release the stored H_2. The same properties motivate also the study described in this paper, in which the material is used not for direct storage but rather for the detritiation of inert gases, in particular helium. With such function, ZrCo getter beds are the reference solution adopted in the conceptual design of the tritium extraction system of the European Test Blanket Modules to replace the cryogenic molecular sieve bed previously proposed [2].

Two experimental campaigns were carried out at the Tritium Laboratory Karlsruhe on ZrCo in order to consolidate this choice and especially to confirm the performances and operability of this technology to extract and recover the tritium from the EU-TBM in ITER. A small scale bed containing 6.5 g of ZrCo material was operated in the CAPER facility and tests with tritium (DT) were implemented. In parallel, a technical scale bed containing 450 g of ZrCo material was operated in the HYDE loop where H_2 was used to study the process performances at relatively high He flow rates up to 20 NL/min.

Within both experimental campaigns, we analyzed carefully and parametrically the absorption and desorption procedures, as well as the one necessary for the initial activation (or re-activation) of the getter material. In general the experiments highlighted a very high absorption efficiency and sharp breakthrough when the amount of H_2 (or HT) reached more than 50% of the maximum storage capacity. On the other hand, unexpected and dramatic increase of the pressure drop through the bed during the absorption was recorded. From these first experimental campaigns to characterize ZrCo for recovering traces of tritium from flowing He, we can conclude that the principle of this process is proved but its practicability should be limited to relatively low loading of the getter material.

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Theoretical Considerations for Purification System used in Hydrogen Isotopes Separation Plants

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ICSI Rm. Valcea is the leading research institute involved in the Romanian heavy water detrititiation program. ICSI has built a Tritium Removal Facility which is an experimental pilot plant for deuterium and tritium separation - its main objective being to demonstrate the detritiation technology followed by the implementation to the CANDU nuclear power plant from Cernavoda.

Within isotope separation installations using cryogenic distillation process, the required gas purity must be high to avoid the risk of impurities condensation. A preferred and recommended process for purification consists of solidifying the impurities over a large material area of heat exchanger devices. Such a system is usually regenerative type to ensure continuous operation. Gas Purification is achieved either by means of reversible heat exchangers either in thermal regenerators.

Reversible exchangers and regenerators have a periodic operation, a warm period and a cold period. During the warm period, the heat exchanger or regenerator mass heat up cooling the purified gas, while in the cold period, the cold waste gas heat up the exchanger or the regenerator.

Essentially, the purification process by solidifying impurities is the same for both reversible exchanger and for regenerator, but because of their different realization the process describing will be different and so did the design method. Due to periodic operation of a regenerative system the process is unsteady, his description being highly complex mathematics. For this reason is of particular interest the realization of a very well developed mathematical apparatus for describing non-stationary heat exchange processes, respectively the simultaneous mass and heat exchange processes taking place in regenerative systems.

This paper presents a conceptual scheme of a purification unit consisting in two stages (the first being a drying system followed by an advanced cryogenic purification). For the second stage will be developed a theoretical analysis of the process. Due to cyclic operation (cooling, retention, cleaning) the process is carried-out in non-stationary regime, thus the mathematical description is complex and needed to design such a system.

It also presents a theoretical analysis on the purification of a gas containing various impurities, using the calculation model developed with the proposed regenerative system.

Performance Characterization of a Pd-Ag Diffuser

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The SRS Tritium Facility currently uses Johnson-Matthey (JMI) Pd-Ag hydrogen purifiers for the separation of hydrogen isotopes from inert gas species in the process. The Pd-Ag diffusers in the process are critical for removing the various inert gas species from the tritium-containing gases. One set of purifiers is used to remove low concentrations of tritium from the process waste gases that are to be sent through the exhaust stack. JMI is no longer manufacturing Pd-Ag diffusers; leaving the SRS Tritium Facility without an in-kind replacement for the diffusers in the process. Other commercial entities that manufacture Pd-Ag diffusers or purifiers were requested to manufacture a replacement tritium-compatible hydrogen purifier to meet SRS specifications.

New purifiers must be characterized by evaluating their operating envelope at SRNL (or equivalent testing elsewhere). It is not recommended or preferred to install equipment in the Tritium Facility without previous testing data. SRS has recently procured several Pd-Ag diffusers from a small company. SRNL will evaluate the diffuser operation and characterize the various diffuser designs to determine the optimal performance conditions.

A series of performance characterization experiments were completed using various gas compositions, flow rates, as well as various tube and shell pressures. This paper will summarize the results of the performance testing of the new Pd-Ag diffuser design compared to the existing Johnson Matthey design.

Formation of Carbonaceous Products under Radiolysis of Carbon Oxide in the DT-Fuel of Fusion Reactor

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Hydrocarbons are formed in the DT-fuel of fusion reactor due to the tritium ß-radiolysis of such admixtures as carbon oxides CO and CO_2/1/. The number of hydrocarbons has been observed with general formula C_xQ_y by micro gas chromatography, while the most of them can be characterized as isomers of C3 и C_4. Authors/1/ concluded that «the presence of higher hydrocarbons and possible other unidentified carbon compounds shows the complexity of developing individual impurity process streams for each and every compound. Hydrocarbons have a concerning impact on ITER Fuel Cycle Subsystems and the issue have to be included in the design of these systems…»

The objective of the given paper is to present research results which provides additional information on possible unidentified compounds and their behavior under irradiation. In particular, it was also previously reported that the most of gaseous products of tritium ß-radiolysis of CO and CO_2 has molecular weight of 12-24 a.u., however at the same time there has been observed formation of so called «white dust» /2/. The «white dust» was characterized as a polymer $C_xH_yO_z$ of variable composition.

The authors have found that CO consumption rate under ß-radiolysis in DT-mixture is substantially higher than the formation rate of gaseous products (mainly hydrocarbons). Besides, the total amount of the formed products is substantially below the amount of decomposed carbon monoxide. Relationship of CO concentration on irradiation time at pressure above normal one demonstrates this quite well/3/. These can be interpreted in the way that carbon monoxide is converted under the radiolysis in DT-gas mixture mainly into the primer product - solid polymer $C_xH_yO_z$, which exit the gas phase. In this case formation of βCH_4, H_2O, CO, CO_2 can be due to the polymer decomposition under irradiation. These results bring us to the conclusion that the polymer being deposited inside the fusion reactor fuel cycle subsystems may contaminate DT-fuel with hydrocarbons due to its radiolysis.

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Recovery of Low-Concentration Hydrogen Isotopes with Zr_2Fe Alloy

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Due to the selection of Li salt as coolant, the Thorium-based Molten-Salt Reactor (TMSR) produces significant amount of tritium, which may damage the mechanical properties and the lifetime of the structure materials, and have large impact to the environment. Therefor tritium control is one of the most important issues in TMSR. The recovery of the extracted tritium as much as possible from off-gas stream becomes a key step in tritium control. Zr_2Fe alloy is considered as a suitable tritium trapping materials since it shows rather high trapping efficiency for tritium in inert atmosphere.

In this work, The Zr_2Fe alloy was prepared by arc-melting the mixture of Zr_2Fe. The crystal structure of the alloy was examined by scanning electron microscopy (SEM) and powder X-ray diffraction (XRD), and a single phase C-16 CuAl2-type structure was found. The activation and thermodynamics character (the pressure–composition isotherms and absorption kinetics) of Zr_2Fe alloy were measured at different temperature by the volumetric method. The alloy could be fully activated after vacuum when the treatment temperature higher than 500°C, and the Zr_2Fe alloy presents a very fast absorption kinetic and a maximum absorption capacity of about 1.8 wt.%. The reaction rates were measured at the initial stage of each run in the subdivided sections of the absorption PC isotherms, and rate determining partial reaction step are presented.

A Zr_2Fe particle bed under a once-through operation was investigated for continuous recovery of low-concentration hydrogen and deuterium from argon stream at different temperature. The particle bed could be activated after purging with argon gas at 500°C for 24h.

As a conclusion, the Zr_2Fe particle bed can recover hydrogen isotopes in argon gas until less than 10 ppb at certain temperature, and heating the Zr_2Fe bed higher than 150°C assures the high performance for hydrogen and deuterium removal in the argon stream. The research shows that the hydrogenation of Zr_2Fe alloy fulfills the demand of tritium trapping in the argon stream.

Hydrogen Recovery from Methane and Water using Catalytic Reaction and Pd Membrane

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In D-T fusion reaction, there are hydrogen isotopes Q_2 (Q means H, D or T), tritiated molecules like CQ_4, Q_2O in the exhaust gas. All of the tritium components should be recovered before released to the environment. In order to recover the hydrogen isotopes from CQ_4 and Q_2O, catalytic reaction combined with palladium membrane permeation can be used. The Q component from CQ_4 can be liberated by catalytic reaction with H2O such as SMR (Steam Methane Reforming), WGS (Water Gas Shift) and so on. As a result, CQ_4 is converted to CO, CO_2 and Q_2. Then, using a palladium membrane which permeates only hydrogen isotopes, the produced Q_2 can be separated.

In these experiments, CH_4 and H_2O were used as a feed gas and platinum catalyst was applied. Firstly, catalytic reaction tests were performed to find out the proper reaction conditions for CH_4 conversion. After these tests, circulation system was made to increase H recovery rate by installing Pd membrane and a circulation pump after the catalyst reactor.

In case of H recovery test from H_2O, CO gas was used for WGS reaction to convert H_2O to H_2 and CO_2. After catalytic reaction tests, circulation tests were also conducted to recover produced H_2.

Design of Cryogenic Distillation for Tritium Separation in China Fusion Engineering Test Reactor

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Cryogenic distillation is an effective technology for large scale hydrogen isotopes separation. A brief introduction was given to such key technology as gas purification, refrigeration and vacuum, measure and control, safety and protection. The wide applications of cryogenic distillation in heavy water production, heavy water detritiation and upgrade, fuel recycle in fusion reactor were applied.

In this work, a large scale of cryogenic distillation for hydrogen and deuterium gas mixture was designed with the whole feed flow at ~20 cubic meter per hour and the deuterium concentration at 500~1000 ppm. The cryogenic distillation process at ~20K would be used for further tritium separation purpose in the water detritiated system of China Fusion Engineering Test Reactor (CFETR). The system consists with a Helium cryogenic refrigerator, a distillation column with about 6 meters height, a reboiler at the bottom and a cooling part at the top of the column, a measuring & control subsystem and some gas storage tanks. The feed of hydrogen/deuterium gas mixture will lead into a proper port with the column. Some fundamental calculation work was done on this system.

Methods for Reducing and Quantifying Methane and Tritiated Methane Formation in High Pressure Hydrogen Isotope Diffusers

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Hydrogen diffusers using Palladium alloy membranes are used to separate hydrogen isotopes from a gas stream that generate tritiated methane in both the permeate gas stream as well as the residual gas stream. The rate of tritiated methane produced depends on: the amount of tritium in the gas stream, operating pressure, operating temperature, carbon content in materials used to build the diffuser and the distribution of carbon in the material. Power & Energy has substantial experience with the chemistry in its microchannel Pd-alloy separators and has seen the substantial catalytic chemistry at high temperatures with carbonaceous components in the presence of large abundance of H2 (and Tritium as well) and has developed process technologies that significantly reduce the methane generated and therefore the tritiated methane. Additional methane is generated into both gas streams for each pass of hydrogen through a diffuser. This will affect the efficiency of the overall process, in a closed loop system, unless there is a step in the process that removes methane. The efficient removal of methane from hydrogen and quantifying the methane content is a process we have developed for assuring that the hydrogen used in Fuel Cell vehicles will meet the current specifications for hydrogen purity. These processes could be scaled to meet the much higher gas flow rates required for hydrogen isotope diffusers. This could be significant in the design of the next generation of tritium processing equipment. This presentation will discuss how the current level methane produced in hydrogen diffusers can be reduced by two orders of magnitude and a low cost process that could quantify and monitor methane and potentially tritiated methane at 1 ppb levels in process streams.

Lessons from Twenty Years of Tritium Exposure to Polymers at SRS

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Polymers are an integral material used in sealing, joining, and containment and are used in several different systems within the Tritium Facility. In an ideal tritium system, we would be able to remove all polymer components due the damaged incurred by the resulting beta decay and reduce the required maintenance of the systems and its components. With the loss of capabilities, such as the Normetex pump, it is necessary to identify and develop polymers that can better with stand the bombardment of beta.

Polymers undergo damage from radiation by two methods: chain scission and cross-linking. The primary method of degradation is dependent on the polymer and its chemical structure. They are more susceptible to damage than metals and ceramics due to the long chain structures that form its properties. The most common resulting physical changes are an increase in hardness and a decrease in elongation. The increase in hardness can be detrimental to sealing applications which rely on compression set and plasticity to seal and prevent permeation of both gases and liquids. Compression set is typically set at 90% for most polymer applications. The decrease in elongation is more sensitive to radiation damage then other mechanical properties such as tensile strength, shear strength, impact strength, and modulus.

The SRS polymer program began in earnest over 20 years ago when concerns about BUNA-N, a synthetic acrylonitrile butadiene rubber, arose. Synthetic rubbers experience negligible radiation damage up to an absorbed dose of 10^5 Gy, begin to exhibit some damage up to 10^6 Gy, and exhibit very severe damage at higher absorbed doses. Due to the lower threshold for radiation damage, alternative materials such as EPDM, ethylene-propylene-diene-monomer, we investigated and recommended for service. EPDM is in continuous use, although investigations to increase the radiation stability of both EPDM and alternative polymers continue to this day. This presentation will highlight the efforts over the course of the last twenty years to increase the radiation stability of polymers, identify potential replacement materials, and support process improvements.

The Experimental Research of ^{3}He Retention Mechanisms in Aged La-Ni-Al Alloys

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To investigate the ^{3}He retention mechanism in aged La-Ni-Al alloys, we compared ^{3}He retention properties at room temperature and tritium releasing temperature after more than 3 years, respectively. The effects of age have been evaluated by following ^{3}He retention on different aging times. The change of thermodynamic and structural properties due to ^{3}He retention on aging time has been determined by PCT diagrams, X-ray diffraction (XRD) and Thermal desorption spectroscopy (TDS). Transmission electron microscopy (TEM) was also preformed for observation of La-Ni-Al aging tritides.

The results indicate that La-Ni-Al alloys retain more than 99.5% of ^{3}He for at least 3 years of aging at room temperature and tritium releasing temperature. The isotherms show a loss of the typical plateau structure, and a decreasing capacity. In the analysis of ^{3}He thermal releasing from TDS, it is found that there are 5-6 ^{3}He release peaks with temperature increase and a large amount of ^{3}He release at the temperature of 800K-1000K (see figure 1), and we also get the activation energy of each peak. As helium accumulation, Nano-helium bubbles have been observed by TEM and the helium bubbles begin to link each other in selected areas (see figure 2). La-Ni-Al Crystal grain begins to refine, and the unit cell is obviously swollen according to XRD results (see figure 3). The mechanism of 3He retention in La-Ni-Al alloys is also discussed.

Key words: Thermal desorption spectroscopy, La-Ni-Al, ^{3}He Retention, Tritide

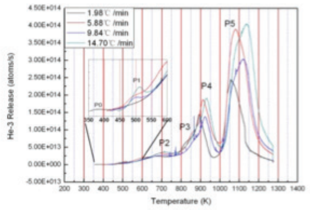


Figure 1.Thermal desorption spectra

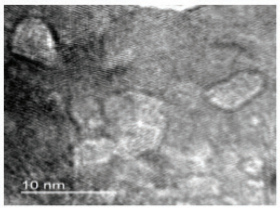
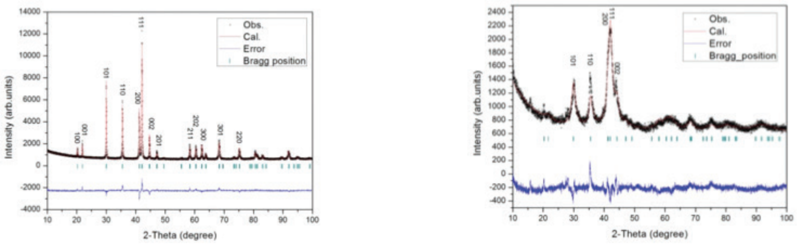


Figure 2. Transmission electron microscope of aged La-Ni-Al



Original sample

aged sample

Figure 3. XRD results of original (left) and aged (right) sample

Tritium Aging Effects in some Pd-Cr, Ni, Co Alloys

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Tritium decays to ^{3}He, and when this decay occurs inside a metal tritide, the ^{3}He is largely retained in the material's bulk. This impacts the subsequent behavior of the hydrogen isotope absorption and desorption, altering the materials thermodynamic characteristics.

Chemical substitution can form alternative miscible hydridable metal alloys over some concentration ranges with modified thermodynamic properties. This allows the 'tuning' of metal hydride characteristics to expand the inventory of available materials for use, potentially allowing a closer match to desired performance characteristics. It is important to quantify these effects in order to predict the long term behavior of metal hydride materials.

Pd alloys of nominal 5 and 10 at% Cr, Ni, and Co were made into foils, loaded with tritium, and stored for several years with occasional isotherm determination to study the tritium induced changes caused by tritium decay while in static storage. Typical effects such as plateau pressure depression and heel formation were noted. Details will be presented and compared to other related work. Also noted was the unusual sensitivity of these materials to the isotherm determination, which tended to partially reverse, or "heal", the tritium decay effects. (One of these samples was removed from its storage unit and further studied with additional techniques, which is the subject of another talk in this symposium.)

Deuterium Retention in Neutron Irradiated Molybdenum

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Understanding and managing deuterium retention in plasma facing components is essential for tritium safety and inventory in fusion reactors. While tungsten will be used in ITER as the divertor material, retention in other refractory metals such as molybdenum are of interest. Molybdenum samples have been exposed to neutron irradiation in the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory, and subsequently tested for deuterium retention in NIMIIX (neutron irradiated materials ion implantation experiment) at Idaho National Laboratory. Deuterium retention in neutron irradiated Mo was compared with deuterium retention in virgin Mo. Results show that overall retention is ~3x higher in virgin Mo than neutron irradiated Mo. however the retention mechanisms, as demonstrated in the desorption spectra in Figure 1, are largely unique. A most prominent difference is in the initial desorption temperature, where desorption from neutron irradiated Mo is delayed 50 K from the virgin samples. These results will be discussed in the context of irradiation temperature, neutron dose, ion flux and fluence as well as surface finish. Possible retention mechanisms and total inventories will be compared and discussed.

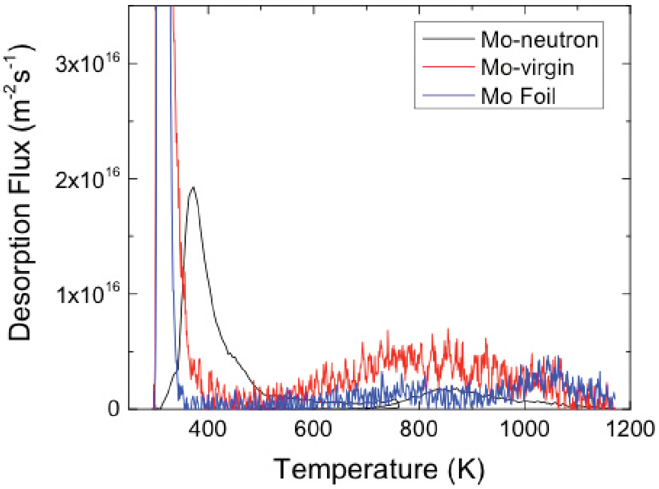

Fig. 1. Deuterium desorption spectra for neutron irradiated Mo, virgin Mo, and a virgin Mo foil sample.

Thermal Desorption Behavior for Helium in Aged Zirconium Tritide Films

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Zirconium are excellent tritium storage materials because of their extremely low hydrogen equilibrium pressure. But the accumulation of helium atom generated by tritium decay could result in a swelling of the lattice, and sometimes this influence is fatal. Therefore, it is quite necessary to understand how these bubbles form, grow, and interact with these two different materials in theory and experiment. Thermal helium desorption spectrometry (THDS), a technique devoted for defect investigation in metals originally, is a particular experimental technique well-suited for studying helium behavior in metals. In this paper, a static thermal desorption method was used to investigate the behavior of helium in aged zirconium tritide films.

Zirconium films were prepared by electron beam evaporation technique in a vacuum chamber at first, and then the deposited films were transported to a tritiding facility and loaded with high-purity tritium gas to produce tritide films with the initial composition of $ZrT_{1.70\sim1.95}$. The samples were heated in vacuum (10-_{6}Pa) up to 1773K at a heating rate of 1K/s, and helium released from samples was measured with a quadrupole mass spectrometer (QMS).

An obvious feature of the THDS is that the helium desorption signal of all samples during aging can be evidently divided into five main zone. The zone I(room temperature~1500K), II(1050K~1800K), III (950K~1600K), IV(600K~950K) and V(400K~700K). All the zones represent helium basic bounding states, which are mobile helium atoms occurring in all aging samples, helium bubbles inside the grain lattice, bubbles in the grain boundaries or dislocations, network of helium bubbles, and helium bubbles near or linked with the film surface for later aging stage. More than 90% helium atoms stay in the form of helium bubbles in the earlier stage, and the concentration of mobile helium atoms in tritide films keeps constant and is less than 0.01% after nucleation. The network of bubbles linked with the grains and grain boundaries are majority formed when He/Zr is about 0.26.

Theoretical Calculations of the Interaction Between Hydrogen and Alloying Atom in Nickel

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Knowledge of the behavior of hydrogen (H) in Ni-based alloy is essential for the prediction of Tritium behavior in Molten Salt Reactor, and can be used to guide the Tritium control and management processes. First-principles calculations were performed to investigate the interaction between H and 3d transition metal (TM) alloying atom in Ni-based alloy. It is easy for first-principles calculations to investigate the system without the influence of other factors.

Judged by the total energy, H prefers the octahedral interstitial site to the tetrahedral interstitial site energetically. The TM-H interaction can be described with the interaction energy. Most of the 3d TM elements (except Zn) attract H. The attraction to H in the Ni-TM-H system can be mainly attributed to the differences in electronegativity as shown in Fig. 1.

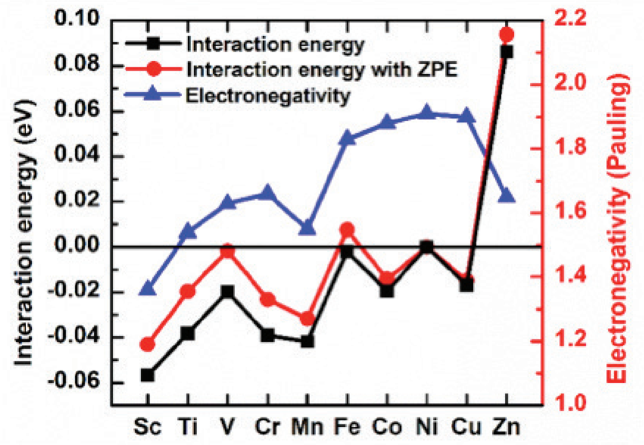

Fig. 1. Interaction energy between TM and H atoms in Ni with and without the zero point energy correction and the electronegativity (in Pauling) of the TM atom.

With the large electronegativity, H and Ni gain electrons from the other TM elements, resulting in the enhanced Ni-H bonds which are the source of the attraction to H in the Ni-TM-H system. Judged by the calculated charge densities, the obviously covalent-like Cr-H and Co-H bindings are also beneficial to the attraction to H. On the other hand, the repulsion to H in the Ni-Zn-H system is due to the stable electronic configuration of Zn. Neither the elastic mechanism nor the “reverse stability” rule can explain the change trend of the interaction between 3d TM and H atoms in Ni.

We mainly utilize the results calculated in 32-atom supercell which corresponds to the case of a relatively high concentration of hydrogen. Our results are in good agreement with the experimental ones. The effects of concentrations of the alloying elements will be addressed in our future works.

Lab Scaled Facilities Dedicated to the Study of Tritium Retention and Outgassing from JET Plasma Facing Components

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Tritium permeation, outgassing and retention are key issues for fusion reactors as a significant amount of tritium can be trapped in in-vessel materials, and contributes to reach progressively the tritium inventory limit outside storage beds; 11g at JET, 700g in ITER.

The WPJET3 project aims to the technical exploitation of the second deuterium-tritium campaign (DTE2) to be held at JET in 2019. This project includes studies of tritium retention, permeation and outgassing from the JET machine and especially from the Plasma Facing Components (PFCs) and divertors plates made of beryllium and tungsten, respectively.
At JET, the tritium permeation can occur because of two different mechanisms:

- During plasma operations, energetic tritium ions hit the PFCs and the divertor and a fraction can be retained in the materials.
- When the pump divertors which trap the ashes and the unburnt fuel are regenerated, the gas is pumped to the JET Active Gas Handling System (AGHS) via the vacuum vessel. A significant amount of tritium (11g, i.e. approximately 4 PBq maximum) can be temporarily released in the torus.

In order to investigate the relative importance of these two different mechanisms, two laboratory scaled facilities have been designed by CCFE and ENEA and are currently being constructed. The Tritium Loading Facility (TLF) will study the short term outgassing and the retention of beryllium under plasma relevant conditions using an ion sputter gun and the Tritium Soaking Facility (TSF) will be used to determine the outgassing and retention of beryllium and tungsten under torus relevant conditions, including torus venting.

This paper describes the design of the facilities, the methodologies, the samples origin and the foreseen experimental plan.

This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the EURATOM research and training programme 2014-2018 under grant agreement No. 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

Study on Radiation Effect of Poly (vinyl alcohol) Films irradiated by Tritium Decay

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The polyvinyl alcohol (PVA) was widely used as a kind of deuterium-tritium (DT) fuel gas-barrier material for multilayer plastic microsphere in inertial confinement fusion (ICF) experiments. The samples of PVA films were exposed to a mixture of DT (1:1) gas at different temperature, pressure and time. The tritium-hydrogen isotopic exchange reaction of the irradiated PVA samples mainly occurred at C-H bond and the FT-IR absorption peak of C-D bond didn't appear. For strong hydrogen bonding interaction, the isotopic exchange reaction don't occurred at O-H bond.

The samples of PVA films were exposed to 0.6MPa DT gas for 5 days at elevated temperature from 30C° to 60C°. The FT-IR absorption peak intensity of C-T bond at 1700cm^{-1} increased linearly with the increase of DT gas temperature, and the amount of C-H bond changed to C-T bond in the irradiated PVA samples increased linearly from 5.4% to 12.2%. The activation energies of tritium-hydrogen isotopic exchange reaction of C-H bond in the irradiated PVA samples were 22.57kJ/mol. The elongation percentage of the irradiated PVA samples decreased linearly from121.0% to 65.4% with the increase of DT gas temperature.

The samples of PVA films were exposed to 0.6MPa DT gas at 25βfor the time range of 10d~30d. The FT-IR absorption peak intensity of C-T bond at 1704cm^{-1} increased linearly with the increase of radiation time, and the amount of C-H bond changed to C-T bond in the irradiated PVA samples increased linearly from 8.1% to 12.5%. The isotopic replacement rate of C-H bond by C-T bond was 0.219%·d^{-1} of the amount of C-H bond in the original PVA films. The elongation percentage of the irradiated PVA samples decreased linearly from 163.8% to 32.7% with the increase of radiation time.

The samples of PVA films were exposed to 0.2MPa~0.9MPa DT gas for 15 days at 25C°. The FT-IR absorption peak intensity of C-T bond at 1704cm^{-1} increased linearly with the increase of DT gas pressure, and the amount of C-H bond changed to C-T bond in the irradiated PVA samples increased linearly from 1.4% to 5.0%. The elongation percentage of the irradiated PVA samples decreased linearly from 245.6% to 5.3% with the increase of DT gas pressure.

The radiation effect of PVA films irradiated by tritium decay was mainly –C-C- main chain bond rupture and degradation. The weight average molecular weight (M_w) of PVA films decreased gradually from 148600 to 21000 with the increase of DT gas pressure. The molecular weight distribution (M_w/M_n) of PVA films increased linearly from 1.35 to 2.05 with the increase of DT gas pressure. The number of low-molecular-weight molecules became more and more with the increase of DT gas pressure.

The permeability of DT gas through the irradiated PVA samples over the temperature range of 20C°~65C° were studied. The permeability coefficient increased exponentially from 1.01×10^{-18} mol·m/m^2·s·Pa to 3.6×10^{-17} mol·m/m2·s·Pa (65C°) with the increase of temperature. The permeation activation energy of the irradiated PVA samples was 74.25kJ/mol.

The Adsorption of Hydrogen Isotope on Graphite in Molten Salt System

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With the development of the nuclear power, the generation IV nuclear reactors are more safe, sustainable and economical. As one of the six candidates of generation IV nuclear reactors, MSR is a popular type of breeder which attracts attentions by many countries.

The molten salt FLiBe is used as coolant in MSR, which releases tritium after neutron irradiation. As an important radioactive element, the concentration of tritium should be controlled in all sites. The tritium adsorption, diffusion and permeation are three main pathways that domain the state of tritium distribution in reactor. While the adsorption of tritium on graphite should be studied in depth as it directly affects the amount of tritium monitored at the entrance of the off-gas system. Some literatures have shown the atomic fraction of tritium retention on graphite at high temperature in Ar/H2 mixture gas phases, while the trap of tritium on graphite in molten salt FLiNaK system is rare studied. In this study, tritium as the hydrogen isotope is replaced by hydrogen whose adsorption quantity will be measured by using volumetric method (GC and QMS). The experiment is taken within the temperature from 737K to 1173K (just like the operating temperature of MSR) under the pressure of 1atm. The concentration of hydrogen (in Ar gas phase) changing from 500ppm to 5000ppm is used. The experiments are taken in both Ar/H2 mixture gas phases and in molten salt FLiNaK phase, respectively. Before the experiment, an annealing process is taken for 4 hours at 673K in vacuum oven in order to remove water and other organic compounds.

The data obey a Langmuir isotherm function which combines DFT simulation in additionally. In order to understand the energy and the crystallization structure of the graphite after hydrogen adsorption, FEED and XPS are used in the research. With a small amount of hydrogen adsorption on graphite, some kinds of hydrocarbons are also found during the experiments in both phases. Through the FEED and XPS, some energy and structure changes are also found. Afterwards, further researches, the adsorption of hydrogen in molten salt system under neutron irradiation, then can be taken into consideration due to these basic researches.

In-Situ Investigation of Deuterium Induced Corrosion of Cerium

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In-situ study of metal and hydrogen reaction, has important scientific significance and engineering significance. It is not only the prototype of the molecule and solid reaction, but also involved in many practical catalytic reactions. To further understand the hydriding process of cerium, and promote the development of other related studies, in-situ investigation of deuterium induced corrosion of cerium was carried out in the present work by pressure-volume-temperature (PVT) method and hot-stage microscope (HSM) technique.

The results show that hydrogen absorption rate of cerium at room temperature is quick that nearly 3.0 D/Ce ratio can be reached in 10 minutes. Cracks formed and increased on the surface of cerium foil during the reaction. Thermal desorption spectroscopy(TDS) shows that cerium deuteride has four desorption peaks, which indicates that deuterium atoms occupied different sites in the lattice of cerium deuteride. Raman spectra provided the vibrational properties of deuterium atoms. Possible mechanisms of deuterium induced corrosion of cerium was discussed.

Low-Pressure and High-Temperature Tritium Behavior on Carbon

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Research characterizing hydrogen behavior on carbon has been primarily focused on collecting data at near-ambient temperatures and pressures for storage or for relatively large quantity applications such as fusion (5 g for Tokamak Test Reactor) [1] [2]. Previous transport models of a small modular 236 MWt fluoride salt-cooled high-temperature reactor (MK-1 PB-FHR [3]) estimate that the production of tritium is relatively low (2410 Ci/EFPD) resulting in sub-atmospheric partial pressures ranging between 0-20 Pa [4]. Operating temperatures in an FHR range from 700-800 $^{\circ}$C. Under these operating conditions, the interaction between hydrogen on carbon is currently un-characterized. Since an FHR contains large quantities of carbon (reflectors, fuel, structures), the tritium behavior in carbon must be considered in order to develop methods to control tritium release rates to environment and to control material corrosion. Additionally, preliminary modeling suggests high performance is achievable in a carbon adsorption tower (reducing release from 2410 to 7.5 Ci/EFPD) [5]. This research aims to significantly reduce uncertainties by attempting to accurately measure hydrogen uptake and kinetics on different types of carbon at prototypic conditions.

In this study, 99.9999% research grade H2 will be used as a tritium surrogate in sorption experiments on different types of carbon including high surface-area activated carbon AX-21 and nuclear grade graphite IG-110U. The temperature range tested will be from 400 to 1100°C and pressure ranges between 0 - 1 atm. Volumetric analysis will be performed using a Quantachrome Autosorb-1 to create absorption and desorption isotherms with a pressure accuracy of +3.3*10^{-4} Pa and a sample capacity up to 5.50 cm^{3}. Diffusion data will be collected and kinetic curves will be produced by gravimetric analysis. This will be performed using an IGA-01 microbalance to detect small weight changes with an accuracy + 1 μɠ in a sample of up to 5 g. In addition, this study may consider the effects BET surface area and pore size distribution of each carbon. If accurate isotherms and kinetic curves can be produced, data gathered from experiments will be used to improve existing models and develop new FHR simulations. Consequently, the feasibility of an out-of-core graphite absorber bed can be re-assessed.

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Surface Treatments to Render Stainless Steel Inert for Tritium

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Tritium is highly reactive with many surfaces and can penetrate the surface and modify the gas composition by interdiffusion as well as reaction with surface elements and contamination. Savannah River Tritium Enterprise use a proprietary surface treatment that is intended to render the surface inert. Unfortunately, this process is not as robust as is needed for containing tritium gas standards. SRTE has funded several projects to determine if there are better ways to prevent the gas contamination and reduce the off-gassing of hydrogen in the stainless steel as well as to reduce the diffusion of tritium into the stainless steel. This presentation will describe test results from treating the surface of gas sample bottles with an aluminide coating, a chromide coating, chromium plating, and a nanodiamond impregnated coating. The testing has been conducted with protium and deuterium, but previous work has demonstrated that if the gas mix does not dissociate in protium and deuterium then the treatment is generally stable in tritium use.

SRNL-STI-2015-00646

Surface Modification of ZrCo and Zr_2Fe Alloy for Enhancing the Hydriding Properties

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Metal hydrides are known to absorb hydrogen isotopes reversibly at moderate conditions. They have been proposed as safe hydrogen isotopes storage and separation materials except for some practical problems. The hydriding properties will be reduced when the metal hydrides are pulverizated and poisoned. The ZrCo and Zr_2Fe alloy particles were modified in order to improve their hydriding properties. Scanning electron microscope (SEM) imaging was applied to determine the surface morphology of the alloy particles after modification. The hydriding kinetics, the resistance of poisoning and pulverization (only ZrCo alloy) of the two alloy particles before and after surface modification were studied. The results reveal that the hydriding properties of modified alloy are excellent. The quantities of hydrogen stored by the treated one are a little greater than the untreated; the hydrogenation rate of modified alloy is faster than that of the original alloy. The results clearly indicate that the resistance of poisoning property with CO, CO_2 and atmosphere is remarkably improved after surface modification. The resistance of pulverization property of ZrCo alloy is also evidently enhanced after surface modification. It is obvious that the hydriding properties of ZrCo and Zr_2Fe alloy particles were both improved after surface modification.

Deuterium Gas Driven Permeation Behavior in W Coated V-5Cr-5Ti

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Alloys based upon the V-Cr-Ti system (e.g. V-4Cr-4Ti, V-5Cr-5Ti) are attractive candidate structural materials for the First Wall (FW) in future fusion reactors because of their low activation properties, high thermal stress factor, good strength at elevated temperatures, and good compatibility with liquid lithium. The interaction between plasma and FW, especially physical sputtering will limit the FW lifetime under normal operation. Therefore tungsten armor is necessary to protect the FW due to its high melting point, low tritium inventory, and low erosion rate. However, owing to its high ductile to brittle transition temperature, difficulties in machining and welding, it is still challenging to use the bulk W to fabricate the FW. Electro-deposition is a promising and attractive method for industrial manufacture of W coating, owing to its simple technics, capability to cover complex surfaces and relatively low cost. A high density (>97%) and thick (>1 mm) W coating has been successfully achieved on vanadium alloys with this technology by researchers from University of Science and Technology Beijing.

The permeation behavior of fuel tritium in the steels is vital for the economy and safety of the reactor. Previous studies showed that the permeability of hydrogen isotopes in vanadium alloys are rather high compared with other structural materials like the reduced activation ferritic/martensitic steels and SiC. However the effect of W coating on the hydrogen isotopes permeation through vanadium alloys is still unknown.

In this paper, a W layer with a thickness of 20 μm was successfully obtained on V-5Cr-5Ti. Scanning electron microscopy (SEM) was used to characterize the surface/cross-section morphologies of the W coatings, X-ray photoelectron spectroscopy (XPS) was used to character the chemical states of the coatings. The deuterium permeation behavior of W coated V-5Cr-5Ti was studied by the gas driven permeation device built in ASIPP. Permeability and diffusion coefficients at temperature ranging from 500-700 oC were obtained and compared with bare V-5Cr-5Ti and pure W. The Tritium Migration Analysis Program (TMAP) was used to analyze the permeation process.

Deuterium Retention in Deposited Tungsten Exposed in EAST

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It is important to evaluate the influence of re-deposited tungsten formed on plasma facing wall on tritium retention in the vessel of a fusion reactor from a viewpoint of safety. In this work, deposited tungsten samples with different thickness and porosity produced by the sputtering method on polycrystalline tungsten substrates were used to simulate the re-deposited tungsten, and retention behavior of hydrogen isotopes were investigated after the exposure of tungsten deposits to plasma in Experimental Advanced Superconductive Tokamak (EAST).

The deposited tungsten samples together with bulk tungsten for comparison were exposed in EAST by Material and Plasma Evaluation System (MAPES) at the mid-plate of H-port for 2005.296 s. Tungsten marker samples with molybdenum interlayers were also exposed to semi-quantitatively study the erosion and eliminate its effect on retention. The Te of the plasma near the surface is about 10 eV by probes, and the maximum temperature rise of the samples is about 347 oC according to 2 thermal couples at the bottom of the samples.

After deuterium plasma exposure, the desorption behavior of hydrogen isotopes from the deposit was observed by the thermal desorption spectroscopy method. It was found that not a small amount of deuterium is retained. The desorption peaks appear around 870 K and 1080 K in this work, which is rather high temperatures for tungsten, indicating deep diffusion of deuterium in both bulk tungsten and deposited tungsten. The rolling direction of the tungsten substrates is perpendicular to the plasma facing surface, which could also bring heavier diffusion of adsorbed fuel on surface into bulk. The location of samples in EAST and even on the head of MAPES may also affect. Furthermore, it could be found that the amount of deuterium retained in the deposits decreases with the porosity. The structure of the deposited films before and after the exposure was investigated by X-ray diffraction and scanning electron microscopy. And ion beam analysis including nuclear reaction analysis and Rutherford backscattering spectrometry are under way to study the depth profile of the deuterium in both deposition layers and bulk tungsten, and the erosion effect on this retention. Simulation work would be done to explain the mechanism of deuterium retention on/in the deposited tungsten.

Tritium Aging of $LaNi_{4.15}Al_{0.85}$ (LANA.85)

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The Savannah River Tritium Enterprise (SRTE) has used hydride beds to store and process hydrogen isotopes for over two decades. New beds are being designed to have a different system for heating and cooling. These beds will involve a hydride material, $LaNi_{4.15}Al_{0.85}$ (LANA.85), that has a lower plateau pressure than the material previously employed. The new beds, like previous beds, are expected to have a limited service life due to radiolytic decay of tritium to He-3 within the metal matrix. The goal of this multi-year project is to examine how the properties of LANA.85 are impacted by tritium exposure before full implementation in the Tritium Facility.

Tritium aging was initiated on two LANA.85 metal hydride samples. The purpose of the first sample was to monitor for He-3 saturation over time and the purpose of the second sample was to look for changes in desorption isotherm performance. Tritium removed during annual gas sampling was replaced. Desorption isotherms were collected at 80, 100, 120, and 160 °C annually. A single absorption isotherm was also collected each year at 120°C. In the first year of aging, it was discovered that isotherm collection at 80°C was not possible because the alpha plus beta to alpha transition had dropped below the analytical capabilities of the sample assay system manifold. As a result, the lowest temperature isotherm was increased from 80°C to 100°C. After testing, each sample was reloaded with tritium for quiescent aging until the following year. Both samples were stored in the beta phase.

Results collected on the virgin material and annually for two years of tritium exposure are presented and discussed. The results have shown no unexpected behavior of the LANA.85 materials over the course of tritium exposure. Before implementation in the Tritium Facility, further annual monitoring and evaluation is recommended to track the effects of tritium exposure on He-3 release and isotherm behavior. Continued evaluation of these samples will reduce the likelihood that unanticipated behaviors will be encountered in full scale production beds.

Paper 11A-1: Wednesday, April 20

Preparation and Catalytic Hydrogen Isotope Exchange Activity of Porous Styrene-Divinylbenzene Polymer Supported Pt Catalyst

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Styrene-based polymers, especially styrene-divinylbenzene (SDBV) polymers, are attractive catalyst support for hydrogen isotope exchange reactions because of their hydrophobic nature. Recently, novel macroporous divinylbenzene polymers were developed via spinodal decomposition using polymethylsiloxane as the phase separator. In this study, we attempted to prepare SDVB polymers with controlled macroporosity. A Pt supported catalyst was also prepared and its activity in catalytic hydrogen isotope exchange reaction was investigated.

Porous SDVB polymers were prepared following the method in literature with slight modification.[1] The starting mixture was as follows; divinylbenzene (mixture of isomers, 6.0 mL), styrene (4.0 mL), trimethylbenzene (14.2 mL), 2,2,6,6-tetramethylpiperidine 1-oxylβ(0.10 g), benzoyl peroxide (0.10 g), acetic anhydrous (0.05 mL) and dimethylsiloxane (1.1 g, DMS-T23). Polymerization reaction was done at 95 ºC for 1.5 h, followed by 125 ºC for 48 h.

A continuous macroporous structure was confirmed by FE-SEM measurements. Mercury porosimetry study revealed the macropore diameter and macropore volume of 2.36 µm and 1.03 cm^3/g, respectively. The surface area was found to be 18 m^2/g by N^2 sorption study. The active sites of Pt for H-T exchange reaction were impregnated by incipient wetness method using $H_2PtCl_6 \cdot 6H_2O$/ethanol, following the thermal decomposition under Ar and reduction under H2 at 200 ºC. The resultant Pt/SDVB catalyst showed a catalytic activity (overall mass transfer coefficient; k [cm/s]) comparable to conventional catalysts.[2]

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More Precise Values of Separation Factors in Water-Hydrogen Isotopic Exchange for Modeling of Combined Electrolysis and Catalytic Exchange Process

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Catalytic gas phase reactions of isotopic exchange between water vapor and hydrogen constitute the principal process leading to isotope separation in the combined electrolysis and catalytic exchange (CECE) process. That makes the precision of equilibrium data for isotopic exchange reactions between water vapor and hydrogen important for the design of current and future hydrogen isotope handling systems including ITER's water detritiation system. Existing codes for the simulation of CECE process make use of separation factors or directly equilibrium constants straightforwardly obtained from the partition function ratios given in [1]. In this article separation factors for ideal gas phase isotopic exchange reactions between water vapor and hydrogen were calculated for deuterium-protium exchange over the full range of deuterium concentration in the presence of trace amounts of tritium using adiabatic correction factors calculated by Bardo and Wolfsberg [2,3]. The results obtained support the conclusions made by Bardo [2] that the application of adiabatic correction factors leads to slightly lower and more precise values of equilibrium constants or separation factors. The difference for protium-trace tritium exchange is relatively low, at 333K it amounts to 2.2%. Nevertheless the simplicity of the adiabatic correction makes its application desirable in the modeling of the CECE process. Comparison with published experimental data [4,5,6] shows that the corrected values better reproduce experiment at least at temperatures below 383 K that are relevant for the CECE process. Results are given in the form of 2D polynomial fits over wide range of deuterium concentration and temperature.

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Aspects Concerning Manufacutre of Reproducible and Homogeneous Batches of PT/C/PTFE Catalyst for Hydrogen-Water Isotopic Exhange

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Tritium removal from nuclear effluents produced in fission or in fusion reactors is one of the most important issues in the field, in order to protect the environment and staff and to recover a very valuable material (tritium). One of the most attractive process for tritium removal it's isotopic exchange between hydrogen and water in liquid phase (LPCE). The key role in LPCE process it's played by the catalyst which have to be very efficient and very stable on time.

The platinum on charcoal and ploytetrafluor-etylene(Pt/C/PTFE) it's one of the most active and stable catalyst used for tritium removal from nuclear liquid effluents by isotopic exchange between hydrogen and water. This type of catalyst can be used alone or mixed with hydrophilic packing in so called- „mixed catalytic packing". An Experimental Tritium Removal Facility(Exp. TRF), based on liquid phase catalytic exchange combined with cryogenic distillation has been commissioned at ICSI Rm -Valcea in order to test tritium removal from tritiated water used as moderator in CANDU reactors from Cernavoda NPP. Many types of mixed catalytic packing, based on Pt/C/PTFE catalysts has been manufactured and tested in LPCE columns from Experimental Tritium Removal Facility from Institute of Cryogenics and Isotopic Technologies.

The paper presents significant aspects concerning the manufacture of reproducible and homogenous batches of catalyst, as follow: impregnation, drying, mixing, shaping and syntherisation. The main characteristics of Pt/C/ PTFE catalyst surface area; pores volume, metallic area, platinum particle size and SEM images aredetermined in order to check the reproducibility and the homogeneity of catalyst batches manufactured by different workers. A comparison between handmade batches and automatic manufactured batches it's presented, too.

The results of measurements on four different batches showed a much closed characteristics which confirms the reproducibility and homogeneity of manufacture method. As result, the mixed catalytic packing, CP-001 type has been selected and proposed to equip the LPCE columns from the future Industrial Tritium Removal Facility from Cernavoda NPP.

Tritium Water Distillation Assisted with Adsorption and Isotopic Exchange

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It is demanded to process tritiated water stored in the Fukushima daiichi nuclear power station after the 3.11 tsunami accident. The conditions of wastewater to be detritiated are the flow rate of 400 m^3/day, the feed tritium concentration of 1-10 kBq/cm^3 and the total decontamination factor of 100. A small amount of other radionuclides or ions are included in the stored water after the ALPS treatment. At present, since it is designed to make detritiation of the wastewater lower than the legally regulated level of 60 Bq/cm^3 without diluted effluent, some new processes were proposed. Judging from viewpoints of previous tritium enrichment technology, LPCE and electrolysis used in the detritiation processes for heavy water reactor might be applied. However, if it is applied as it is, it may lead to usage of large amount of expensive Pt catalyst or high operating cost. Water distillation process needs also a large amount of heat when it is operated in usual facilities. In the present study, a new process to offset evaporation heat against condensation heat is developed under vacuum distillation condition of 60°C (19.7 kPa). Only compressor power is supplied from outside to distillation tower. In addition, materials having properties of water adsorption and isotopic exchange are packed in the tower, and a larger isotope separation factor between HTO and H_2O is expected in a similar way to previous water distillation conditions [1,2]. Experiment of HDO or HTO enrichment is tested using vacuum or atmospheric distillation tower packed with some adsorbents under the total reflux condition. Higher stage separation factor is achieved compared with normal packing materials of Dixon ring. Another Sulzer packing having water adsorption property will be also tested for actual applications to Fukushima wastewater. The details will be presented in the conference.

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Recent Progress in the Experimental Study of LPCE Process on "EVIO" Pilot Plant

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A new LPCE column (LPCE-3) of 2 m packing height and 50 mm inner diameter expands the experimental possibilities of "EVIO" pilot plant [1, 2].

Fresh RCTU-3SM catalyst of somewhat greater average percentage of Platinum (1.3% versus 0.8-1%) and a little bit larger dimensions of SDBC carrier (spherical granules of 0.8-1.2 mm versus 0.5-1.0 mm) has been tested in LPCE-3. Both hydraulic and isotope separation characteristics of LPCE-3 filled with alternating layers of the catalyst and stainless steel spiral-prismatic packing in the volume ratio of 1:4 (the same packing and the same ratio as it used in LPCE-1 and LPCE-2 columns) have been studied. Experimental results are presented and discussed in comparison with ones received on LPCE-1 and LPCE-2 earlier.

LPCE-3 demonstrates very good capabilities. The throughput and performance of the column increased to a considerable extent comparing to the characteristics of elder columns.

Dependence of separation performance on specific hydrogen load has been discovered.

Separation performance expressed by a 3-flow model attribute $\boldsymbol{K_c}$ – mass-transfer coefficient for catalytic exchange (at fixed mass-transfer coefficient for phase exchange) is the same for different isotopes (H, D and T) in contrast to separation performance expressed by the overall mass-transfer coefficient, $\mathbf{K_y a}$ [3].

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Rigorous Two-Fluid and Three-Fluid Liquid Phase Catalytic Exchange Models and their Application

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Liquid Phase Catalytic Exchange (LPCE) is used in water detritiation systems to counter-currently exchange hydrogen isotopes between liquid water and elemental hydrogen gas. Rigorous and accurate simulation of LPCE is required for process design and analysis.

The Two-Fluid model simulates exchange between gas and liquid using an overall mass transfer coefficient model, without separate consideration of liquid/vapor and vapor/gas mass transfer. The Three-Fluid model simulates liquid/vapor and vapor/gas mass transfer explicitly.

This paper presents an improved Two-Fluid model that combines liquid and vapor flow as compared to previous Two-Fluid Models that simply ignored vapor flow. The improved Two-Fluid model is close in accuracy to the more \rigorous Three-Fluid model.

In the general case in which any of the hydrogen isotopes (H, D and T) may be present at non-negligible concentration, no analytical solution is possible. A solution is obtained by discretization of the LPCE differential equations using finite difference approximations for derivatives, resulting in a system of nonlinear equations. The resulting equations have block-tridiagonal structure. The blocks are (2x2) and (3x3) in dimension for the Two-Fluid and Three-Fluid models, respectively. The nonlinear equations are solved iteratively, with a block-tridiagonal linear system of equations being solved rapidly at every iteration.

In the ternary H, D, and T system, the mass transfer coefficients are interdependent and rigorous simulation requires solution of the Maxwell-Stefan equations along the length of the column. Simulations are presented with rigorous solution of Maxwell-Stefan equations versus simplifying approximations for a range of H, D, and T compositions. In most practical cases tritium is present at trace level, so the treatment of mass transfer coefficients can be simplified.

Both Two-Fluid and Three-Fluid models can be fitted to experimental data by adjustment of mass transfer coefficients and then using the “calibrated” model to design an LPCE column to meet specific design objectives. The differences between using Two-Fluid and Three-Fluid models for design purposes are discussed. Columns designed using the Two-Fluid and Three-Fluid models are similar if the improved Two-Fluid model described in this paper is used.

Tungsten Dust in Fusion Devices: Impact of Morphology and Characteristics of Particles on Tritium Retention/Desorption, Associated Toxicological Studies

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Tritium retention in dust particles generated during ITER operation is a major concern for the evaluation of the safety of the machine. These dust of variable size will be activated and tritiated, and could store a great amount of tritium in the vacuum vessel. This has numerous consequences regarding safety issues including the possible release of tritium with such activated aerosols in case of failure of the first barrier of confinement as a loss of vacuum accident (LOVA). Accidental inhalation of such radioactive particles should thus be harmful for fusion workers.

In this paper and in order to prepare the ITER operation, the desorption kinetics of tritium from W powders of different type and characteristics is described. W particles are prepared by planetary milling, by magnetron plasma sputtering and gas condensation, by laser ablation or by filtering particles provided by an industrial supplier. The various dusts are first characterized: size distribution, Specific Surface Area (SSA), concentration of defects and crystallographic structure. Two techniques are used for tritium implantation: gas infusion and microwave plasma interaction. Both processes are compared in term of tritium inventory. Moreover, the impact of the tritiation processes on the particles morphology and characteristics is also addressed, in order to compare relevantly the various experiments. After tritiation implantation, tritium inventory and the dynamic of desorption from the various particles are studied. In all the cases, just after implantation, a rapid release of tritium is observed which ends after 2 to 5 days. Then, the tritium inventory is measured and tritium contents greater than 1GBq/g and up to 40Gbq/g are observed, which is more than 2 orders of magnitude greater than for massive samples. Tritium inventory increases with the dust SSA and with the dust properties, which stresses the importance of studying different type of powder. Almost all the tritium remaining in the sample is recovered at 500°C. However, 1% of the initial tritium inventory remains in the sample at 1000°C corresponding to a maximum of 0.4GBq/g.

In addition to this tritium behavior study, biology experiments are led in order to characterize potential toxicity of these particles being inhaled. In vitro test are undertaken with a cellular model (mucilair®). Tritiated and non tritiated particles are deposited on the apical side of the cells. In the paper, the consequences of such contamination will be presented. The internalization of the particles will be addressed as well as their cytotoxicity.

Paper 11B-2: Wednesday, April 20

Determination of In Vitro Lung Solubility and Intake-to-Dose Conversion Factor for Tritiated $LaNi_{4.15}Al_{0.85}$

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Various metal tritides are encountered at the United States Department of Energy (DOE)'s Savannah River Site (SRS) Tritium Facilities. These metal tritides are essentially tritium in a particulate form, which can present radiological protection challenges that are distinctly different than those of elemental tritium or tritiated water. These challenges can be divided into three areas: detection, physical and chemical behavior, and radiation emission, any of which can complicate deconvolution of bioassay results. Uptakes of elemental or oxidized tritium are readily distributed throughout and released from the body and can be detected by normal bioassay sampling. Inhalation of tritiated particulate may or may not be detected, depending on how rapidly tritium evolves from the material. Tritium evolution from inhaled particulate occurs through either through dissolution of the host material or by diffusion of the tritium from the intact host material. For simplicity, in this discussion the release of tritium from a tritide by either mechanism is referred to as "dissolution." Materials that readily release tritium are classified as "soluble" and result in lower DCFs than materials that retain tritium. Once released from a soluble material, the tritium is flushed from the body via normal metabolic processes. Materials classified as insoluble allow tritium beta decay energy to deliver overall higher dose to a localized area.

A sample of tritiated lanthanum nickel aluminum alloy ($LaNi_{4.15}Al_{0.85}$, commonly abbreviated as "LANA.85"), similar to that scheduled for use at the Savannah River Site Tritium Facilities, was analyzed to estimate the particle size distribution and tritium evolution rate in a simulated lung environment. This information was used to calculate the committed effective dose a worker would receive after inhaling tritiated LANA.85. The results, calculated for several particle sizes using the same methodology given in the DOE Tritium Handbook, are presented as inhalation intake-to-dose conversion factors (DCF). The DCF for inhaled 5.0 μm tritiated LANA.85 particulate was determined to be 1.01×10^{-11} Sv Bq^{-1}, indicating that tritium in a urine bioassay can be interpreted as if it had resulted from an intake of pure HTO if the worker was potentially exposed to any mixture of HTO and LANA.85.

Paper 11B-3: Wednesday, April 20

HTO and OBT Exposure Induces Differential Effects on Haematopoiesis and Iron Metabolism

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Environmental exposure to tritium is mainly due to tritium releases by nuclear reactors. This exposure may increase in future years due to potential increased releases by the ITER reactor. However concerns exist about the reliability of the current ICRP recommendations. As a result, a collaborative project was developed between IRSN and CNL to study various aspects of the toxicity of tritium.

In this project male C57BL/6J mice were exposed to a chronic contamination through drinking water with three different concentrations of tritium (HTO), or tritiated amino acids (OBT) or low dose external gamma radiation (1.3 and 31.2 μGy.h-1), equivalent to tritium exposure, for either 1 or 8 months. Drinking water concentrations of tritium were 10 kBq.L-1, 1 MBq.L^{-1} and 20 MBq.L^{1}, the lowest being relevant to the current regulatory level in Canada set at 7000 Bq.L^{-1}, WHO recommendations at 10 kBq.L^{-1} and French regulatory level set at 100 Bq.L^{-1}. At 1 and 8 months of exposure, blood, spleen, liver, kidney and intestine were harvested and subjected to various analyses including blood cell count and differential, measurement of biochemical parameters in the plasma, and relevant genes expression quantification.

Results of blood cell count analyses indicated slight but significant decrease in RBC at 1 month of exposure with OBT but not with HTO. This effect was not observed at 8 months of exposure, but a significant increase in mean corpuscular volume and decreased mean cell haemoglobin content were observed. These results suggest the appearance of a macrocytic anaemia at 1 month of exposure with OBT, which was compensated by an increased half-life of red blood cells at 8 months of exposure. This macrocytic anaemia was accompanied by a decrease in serum iron concentration at 8 months of exposure in OBT-exposed animals with a dose dependent effect suggesting that an iron deficit was at the origin of this macrocytic anaemia. Such an iron-dependent macrocytic anaemia may have several origins, including a central defect of haematopoiesis, a defect in EPO production by the kidneys, a defect in the capture of red blood cells in the spleen or a defect in iron metabolism in the liver and/or in the intestine. At the moment, results of gene expression analysis and cytokine measurement exclude the three first hypotheses. Gene expression of molecules implicated in iron metabolism showed a slight decreased expression of erythroferrone in the liver. However, the main change was observed in the intestine with a dose-dependent decreased expression of transferrin light chain both at 1 and 8 months of exposure. Work is currently in course to confirm these results through biochemical analyses in these tissues. However, these results already demonstrate a differential effect of HTO and OBT at the same level of exposure, while no such effect was observed in animals irradiated at a gamma dose rate corresponding to the measured concentrations of tritium. These results suggest that deterministic effects may appear during chronic ingestion of tritium and that these effects may be significantly more important with the OBT form of tritium as compared to HTO. This may be explained by the integration of tritiated amino acids in the general metabolism. The implication of these results for radioprotection remains to be defined.

Tritium Level Evolution in the Environment at Experimental Pilot for Tritium and Deuterium Separation—ICIT

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The only CANDU type nuclear power plant (NPP) of Europe is Cernavoda NPP. In view of the long period of operation of one unit of Cernavoda NPP, 19 years, and the raising tritium level in moderator, there was a clear need to develop the technology for tritium and deuterium separation from heavy water.

The Experimental Pilot Plant for Tritium and Deuterium Separation (PESTD) is an experimental installation within the national nuclear energy research program. The PESTD processes were tested using heavy water and tritiated water below exemption level of the Romanian national legislation. The tritium monitoring program started in 2010.

PESTD is part of the Institute for Cryogenics and Isotopic Technologies (ICIT), situated at a distance of about 10 km from the city Rm. Valcea. It is close to several rural settlements, and it is located on the city‘s industrial estate. The Olt River is the only source for industrial water supply and is situated about 1 km from ICIT. The wastewater is discharged outside the Institute in the sewage system of industrial estate.

Tritium measurements were performed in air, surface water, and precipitation on a monthly basis. The method used was liquid scintillation counting. The soil and sediments were measured once per year. The home-grown fruits, vegetable and grains from the gardens located in the vicinity of PESTD were collected monthly during the growing period, each year. Tissue Free Water Tritium was measured using azeotropic distillation and liquid scintillation counting method.

Tritium concentration in precipitation during the monitoring period, 2010-2014, exhibits an average of 10.8 ± 1.7 TU (1 TU= one tritium atom to 10^{18} hydrogen atoms). A minimum value below 4.5 TU in November 2014 and a maximum of 21.9 ± 2.1 TU in July 2014 have been recorded. These are typical values that can be found in precipitation for Vienna, monitoring station of Global Network of Isotopes in Precipitation.

The yearly average tritium concentration for the five locations varied between 11.6 ± 2.52 mBq/m3 in 2013 and 23.2 ± 3.39 mBq/m^3 in 2012. A minimum value of 3.8 ± 3.39 mBq/m^3 has been recorded in December 2013, and a maximum of 97± 3.39 mBq/m3 in February 2012. These values are slightly higher than recorded values for the preoperational monitoring programme.

The values of tritium concentration in surface water, soil, fresh products used in human consumption, didn’t exceed 4 Bq/kg. Nuclear activity of PESTD didn’t have any impact in the environment until now.

Modeling Tritium Transfer in a Grassland Ecosystem in Response to ^{3}H Releases and Validating with Experimental Data

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Tritium (^{3}H) is a major radionuclide released in several forms (HTO, HT) by nuclear facilities under normal operating conditions. In terrestrial ecosystems, tritium can be found under two forms: tritium in tissue free water (TFWT) following absorption of tritiated water by leaves or roots and Organically Bound Tritium (OBT) resulting from TFWT incorporation by the plant organic matter through metabolic processes like photosynthesis at day.

In order to study transfers of tritium from atmospheric releases to terrestrial ecosystem such as grasslands, an in-situ laboratory has been set up by IRSN and EDF on a ryegrass field plot located 2km downwind the AREVA NC La Hague nuclear reprocessing plant (North-West of France). The objectives of this experimental field are: (i) to better understand the OBT formation in plant (by photosynthesis and during nighttime), (ii) to evaluate transfer processes of tritium in several forms (HT, HTO) from the atmosphere (air and rainwater) to grass and soil, (iii) to develop a modeling allowing to reproduce the dynamic response of the ecosystem to tritium atmospheric releases depending of variable environmental conditions. For this purpose, tritium activity will be measured in grass (monthly measurements of HTO, OBT), air, rainwater, soil (daily measurements of HT, HTO) and CO_2, H_2O fluxes between soil and air compartments will be carried out. Then, the TOCATTA model [1] implemented in the SYMBIOSE platform [2] will be refined to simulate tritium transfers to pasture on a hourly time-step basis [3,4]. Based on the modeling previously developed at this time scale, developments will take account for processes specific to tritium. The model is being tested by a comparison between simulated results and field measurements obtained from the experimental plot over one year.

The objectives of this presentation are (1) to present the experimental design of the VATO study (VAlidation of TOCATTA) dedicated to transfers of tritium in a grassland ecosystem, (2) to document the major assumptions, conceptual and mathematical modelling of tritium transfers in soil-plant systems implemented in TOCATTA [1], and (3) to discuss the results of the model-measurements comparison.

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Paper 11B-6: Wednesday, April 20

Effects of Tritiated Thymidine on Neural Stem and Progenitor Cells

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Ionizing radiation during pregnancy impairs brain development leading to microcephaly and mental retardation, which might be due to the depletion of radiosensitive neural stem and progenitor cells (NSPC) that give rise to the main types of neural cells of the brain. Microcephaly and neurobehavioral changes have been also reported after prenatal exposition to high doses of tritiated thymidine (3H-T) in mice. We have therefore investigated the effects of 3H-T on primary cultures of human and murine NSPC. 3H-T is incorporated into DNA during replication leading to chronic ß-irradiation of the contaminated cells and their progeny.

In human NSPC, high doses of 3H-T (≥ 1.85 KBq/ml during 7 days) decreased cell growth due to activation of intra-S and G2/M cell cycle checkpoints, induction of cell death or premature senescence. 3H-T impaired also stem cell functions by triggering premature differentiation and by enhancing cell migration. We developed a cellular dosimetry method by comparing silver grains quantification from autoradiography with liquid scintillation counting after DNA TCA- precipitation. Thereby, tritium dosimetry revealed that those cellular effects were evidenced when nuclear contamination was more than 60 000 fold the level authorized by WHO for tritium in drinking water. Nevertheless, metaphase spreads analysis evidenced that one chromatid break is detected for 39 to 62 tritium disintegrations occurring in cell during S and G2/M phases.

We then compared the chromosome instability after one or several rounds of replication. Interestingly, there was no increase in chromosome aberrations suggesting an adaptive response in contaminated cells. Similar results were obtained in murine NSPC and in mouse embryonic fibroblasts (MEF).

In order to understand the molecular mechanism involved, we investigated the importance of DSB (Double-Strand Break) repair pathways in 3H-T treated NSPC. We have tested the effects of 3H-T on NSPC derived from mice, deficient for Rad54 or Cernunnos, proteins which are involved in HR (Homologous Recombination) and NHEJ (Non-Homologous End Joining) respectively. Whereas HR-deficiency had no significant impact on cell survival, NHEJ-deficiency decreased cell viability, increased chromosome instability and abrogated the adaptive response in response to 3H-T contamination in both, murine NSPC and MEF. These data suggest therefore that DSB induced by 3H-T are predominantly repaired by NHEJ.

Altogether these data confirm that high doses of 3H-T target the NSPC compartment and may impair brain development highlighting that mechanistic studies induced by tritiated water and other tritiated compounds with different intracellular localizations in human NSPC are lacking.

Paper 12A-1: Wednesday, April 20

Pebble Fabrication and Tritium Release Properties of $Li_{2+x}TiO_{3+y}$ with Li_2ZrO_3 Solid Solution

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Solid-solution pebbles of Li2+xTiO3+y with Li2ZrO3 (LTZO) has been developed as a super advanced tritium breeder. Considering the tritium release characteristics of the blanket, the optimum grain size after sintering is <5 μm. Therefore, the pebble fabrication by using emulsion method was carried out to obtain the target value (Fig. 1). LTZO20 (Li2+xTiO3+y with 20 wt% Li2ZrO3 added) was more easily sintered under air compared to Li2+xTiO3+y pebbles, and the average grain sizes of the LTZO20 pebbles were <5 μm. These results indicate that the developed LTZO solid solution has a strong potential for use as a super advanced tritium breeder.

Furthermore, the tritium release properties of tritium breeders are required for DEMO blanket design. In the present study, an evaluation of the tritium release properties of LTZO20 pebbles was performed by using DT neutron irradiation experiments at the fusion neutronics source (FNS) facility in JAEA. To measure in-situ tritium release properties, two ionization chambers (IC) have been adopted as on-line gas monitor. The tritium gas consisted of tritiated hydrogen (HT) and tritiated water (HTO) gases. The HT and HTO were measured in the first IC. The HTO gas was then collected by first water bubbler, and HT was successfully measured in the second IC.

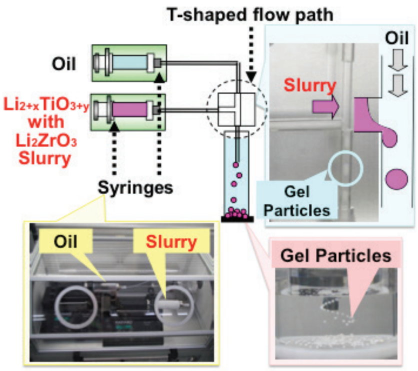

Fig.1 Pebble fabrication by emulsion method.

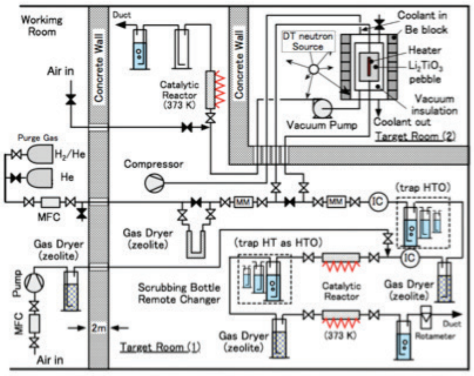

Fig.2 DT neutron irradiation experiment.

Paper 12A-2: Wednesday, April 20

The Effect of Tritium Generation on Alloys Corrosion in Molten Li_2BeF_4 Salt

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In support of materials development for fluoride salt-cooled high-temperature nuclear reactors (FHRs), the high-temperature in-reactor corrosion tests of 316 stainless steel and Hastelloy N® in molten Li2BeF4 (FLiBe) have been successfully accomplished in the MIT research nuclear reactor at 700°C for up to 1000 hours. One objective of these in-reactor corrosion tests is to evaluate the compatibility of structural alloys with the molten FLiBe under neutron irradiation. Unlike out-of-reactor corrosion tests, the corrosion behavior of alloys in molten FLiBe is more complicate due to the tritium generation in molten salt in addition to the radiation-induced structural damage during in-reactor corrosion tests. In the MITR operating at a power of 6MW, with axial-average neutron fluxes of 2.35x1013 n/cm2-s thermal and 1.16x1014 n/cm2-s fast (E>0.1MeV), tritium is produced when neutrons react with the constituents of FLiBe salt:

$$\begin{gathered}
n + {}^{7}_{3}LiF \rightarrow {}^{4}_{2}He + {}^{3}_{1}HF(TF) + n' \\
n + {}^{6}_{3}LiF \rightarrow {}^{4}_{2}He + {}^{3}_{1}HF(TF) \\
n + {}^{9}_{4}BeF_2 \rightarrow {}^{4}_{2}He + {}^{6}_{2}He + 2F \\
n + {}^{19}_{9}F \rightarrow {}^{17}_{8}O + {}^{3}_{1}H(T)
\end{gathered}$$

From these transmutations, it is known that the tritium exists in the molten FLiBe as tritium fluoride (TF), and TF is dissolvable in molten salt as T+ and F- ions. TF chemically behaves like HF in that it is oxidizing to structural alloys: 2 (_1^3)HF(TF)+M βMF_2+ T_2β After this corrosion reaction, TF then converts to T2(g) in molten salt. Moreover, T2 physically and chemically behaves like H2 that is not only reducing to the metallic ions in molten salt but also highly permeable through structural metals. Therefore, the ratio of transmutation products (TF) to oxidation product (T2) in molten salt, both representing as oxidizing and reducing agents, determines the redox potential of molten salt, which significantly controls the corrosion rate of alloys in high-temperature molten FLiBe.

This study focuses on the investigation of tritium generation effect on the corrosion behavior of promising candidate alloys for the FHR, 316 stainless steel and Hastelloy N®, in molten FLiBe salt in a nuclear reactor. It includes the measurement of the tritium retention in irradiated FLiBe salt and alloys, the microstructural characterization of tested alloy samples, as well as the chemistry analysis of molten salt by using electrochemical techniques.

Application of Tritium Imaging Plate Technique to Measuring Hydrogen Solubility and Diffusivity in $BaInO_{2.5}$

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A tritium imaging plate technique (TIP) has been used to measure the hydrogen solubility and diffusivity in Zr-doped $BaInO_{2.5}$ which is one of proton conducting materials.

The $Ba_2In_2O_5$ specimen was prepared with a typical solid state reaction method. The $BaCO_3$, ZrO_2 and In_2O_3 powders were mixed in an alumina mortar with methanol, and dried and calcined in air at 1573 K for 3h. The mixed powder was die-pressed into a disk and then isostatically pressed at 200 MPa. The disc was sintered in air at 1773 K for 5 h. The black surface layers of the specimen were removed with an abrasive paper, then its size was ca. 8 mm in diameter and 2 mm in thickness. Tritium charging into the specimen was carried out with a gas absorption method using tritiated water vapor of 2.5kPa diluted with hydrogen (T/H ~ 10^{-6}) at a temperature ranging from RT to 873 K for a given time. After the charging, the specimens were cut in half, and then the surface and cross section were exposed to IP (TR2025) at RT to obtain a photo-stimulated luminescence (PSL) with an IP reader (BAS 2500).

For the high temperature charging, tritium clearly diffused into the specimen as shown in Fig. 1 which is a PSL image of the cross section of the cut specimen exposed to tritiated water vapor. Based on the PSL distribution profiles and intensities, hydrogen diffusivity and solubility in the specimens were obtained. The hydrogen solubility and diffusivity in Zr-doped $BaInO_{2.5}$ clearly depended on the content of Zr. This characteristic on the Zr content is in agreement with the Zr-content dependency of its proton conductivity.

Paper 12A-4: Wednesday, April 20

Development of H, D, T Simultaneous TDS Measurement System and H, D, T Retention Behavior for DT Gas Exposed Tungsten Installed in LHD Plasma Campaign

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In Large Helical Device (LHD) at National Institute for Fusion Science (NIFS), D plasma discharge experiment will be planned. In the D plasma discharge, energetic tritium (T) will be produced by DD reaction during plasma discharge and implanted into plasma facing materials. To evaluate fuel distribution behavior, it is quite important to estimate all of hydrogen isotope retention. Therefore, in this study, H, D, T simultaneous TDS measurement system has developed and the TDS experiment was demonstrated by DT gas exposed W.

For the development of H, D, T simultaneous TDS measurement system, two different measurement techniques using a quadruple mass spectrometer (QMS) for H, D and proportional counters were used. The sample was placed in a quartz tubing with flowing Ar gas and it was heated with steady heating rate by an infrared gold image furnace. A small amount of desorbed gas was introduced into QMS to measure H, D and rest of the gas was flown into the proportional counters with mixing methane as counting gas.

To evaluate the performance of present TDS system, the mirror polished W samples with size of β10 mm × 5 mm were placed in LHD 2014 plasma campaign and exposed H plasma discharge. ~200 nanometer of carbon dominated mixed-material layer was deposited on the typical PWI area sample. Before DT gas exposure, 1 keV D2+ implantation was performed with the fluence of 5.0 × 1021 D+ m-2. It was found that major D desorption was occurred at the temperature of 600-700 K and 900-1000 K. These desorption temperature was quite high compared to that for pure-W. These facts indicate that the D would be trapped by irradiation damages and carbon to form C-D bond in the carbon dominated mixed-material layer. In this presentation, the result of DT gas exposure experiment will be also presented and desorption behavior of all of hydrogen isotope in plasma exposed W will be discussed.

Paper 12A-5: Wednesday, April 20

Simulation of the Migration of Tritium in Tungsten Materials in Fusion Devics (ITER and DEMO)

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During ITER operation, a large part of the high flux (1022-24 m-2.s-1) of deuterium (D) and tritium (T) interacting with the walls will be implanted in the tungsten ITER Plasma Facing Components (PFC). Then the implanted particles will diffuse into the bulk of the materials or back to the surface. Thus, fuel can potentially be retained inside the wall and even permeate through the actively cooled PFC to the cooling loops. Since tritium is radioactive, its retention/permeation in the machine wall causes serious safety issue. To contribute to this open, macroscopic rate equation (MRE) models are efficient tool to investigate the migration of hydrogen isotopes (HIs) inside W materials at large time and length scale comparable to those of ITER. Once the diffusion and trapping properties of D/T in W are extracted from well-known laboratory experiments, it is relatively straightforward to extrapolate the tritium retention/permeation in W ITER wall.

In this presentation, we will first present the code MHIMS (Migration of Hydrogen Isotopes in MetalS) [1] based on MRE model. Then, the effective trapping parameter of D/T for complex polycrystalline tungsten (PCW) material will be determined by fitting thermal desorption spectrometry (TDS) done after Deuterium implantation in annealed PCW [2, 3]. The simulation shows that 1 to 3 different traps can be distinguished with detrapping energy between 0.85 eV and 1.5 eV with trapping density around 10-3-10-4 at.fr. Of course, our goal is to determine the trapping parameters into PFC placed in a real fusion environment with 14 MeV neutrons irradiation. This is done by simulating a TDS experiment done after Deuterium implantation on self-damaged tungsten which is a good tool to simulate neutron damaging [4]. In this case, the detrapping energies, between 1.6 eV and 1.9 eV, are much higher than in the case of annealed PCW enhancing the retention at high temperature. Thus, we will show that, in order to recover the majority of tritium trapped in such PCW, heating above 1300K for a long period of time e.g. more than 1000s is needed.

Finally, using all the obtained trapping parameters, the retention and permeation of tritium/deuterium through PCW during ITER thermal cycles are estimated taking into account the presence of these high-energy neutrons trapping sites.

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Tritium Behavior in the Water-Cooled Ceramic Breeder Blanket of CFETR

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To realize nuclear fusion as an acceptable commercial energy source, the fuel tritium should be self-sustainable, i.e. the overall tritium breeding ratio (TBR) of fusion reactor blankets must be >1. The TBR is generally decided by tritium production rate, release properties of the breeder and the transport inside structural/first wall materials. This paper presents the efforts to evaluate the tritium behavior in a water-cooled ceramic breeder (WCCB) blanket, which is a candidate concept for the China Fusion Engineering Test Reactor (CFETR) at the Institute of Plasma Physics, Chinese Academy of Sciences (ASIPP).

In the present research, tritium behavior in the first wall, breeder and as well as the structural materials has been taken into account. Using gas-driven permeation and plasma-driven permeation methods, hydrogen isotopes transport parameters have been measured for the plasma-facing material (i.e. tungsten) and as well as structural materials, including some recently developed reduced activation ferritic/martensitic steels. Tritium release from a ceramic breeder Li2TiO3 has been investigated by deuterium ion implantation and subsequent thermal desorption spectroscopy (TDS). Meanwhile, the effects of surface conditions and irradiation defects on hydrogen permeation/retention have been investigated theoretically and experimentally.

Deuterium permeability, diffusivity and recombination coefficients in plasma-facing/structural materials have been successfully measured. For the ceramic breeder, the released deuterium from the surface is mainly in the forms of HDO and D2O. Surface conditions of the blanket walls have been found to affect the hydrogen isotopes permeation flux and inventory. For the first wall using tungsten as the armor material, the surface conditions effects at the plasma-facing side on hydrogen permeation are less sensitive than those of bare RAFM steel walls. These experimental data are utilized as the input parameter for TMAP calculations to predict the tritium behavior under reactor-relevant conditions. Finally, a system-level code is developed to evaluate the overall tritium behavior for the WCCB blanket.

Modeling to Support In-Situ Data Interpretation in the TMIST-3 Irradiation Experiment

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The Pacific Northwest National Laboratory (PNNL) supports the National Nuclear Security Administration's Tritium Sustainment Program in the design, development, demonstration, testing, analysis, and characterization of tritium-producing burnable absorber rods (TPBARs). The TPBARs are irradiated in the Tennessee Valley Authority's Watts Bar Nuclear Unit 1 commercial power plant to produce tritium for the U.S. strategic stockpile. The Program has identified specific data needs to support TPBAR design and performance modeling. In response to these needs, in-reactor experiments are conducted to elucidate fundamental mechanisms governing TPBAR irradiation behavior. The TMIST-3 experiment focuses on in-reactor $LiAlO_2$ pellet performance, and it will be inserted in the Advanced Test Reactor at Idaho National Laboratory in mid-2016.

Tritium is produced in TPBARs from the transmutation of ^6Li in the $LiAlO_2$ pellets. The time, burnup, burnup rate, and temperature dependence of the total tritium release from TPBAR pellets has not been measured previously. However, this release is the basis for understanding tritium transport within the TPBAR. By understanding tritium release kinetics and its dependence on irradiation parameters, better TPBAR performance models can be developed that will predict tritium transport in TPBARs with higher fidelity than currently possible. Evaluating the effects of pellet grain size, porosity, and pore morphology on tritium release will provide further insight into the rate-limiting mechanisms associated with these phenomena and improve the ability to predict TPBAR performance based on variations in pellet characteristics.

The TMIST-3 experiment is an instrumented lead test with two separate test trains consisting of a total of 41 individual capsules containing a variety of pellets with different microstructural features subjected to a variety of irradiation conditions. The TMIST-3A test train will be irradiated first and includes nine flow-through capsules in which the tritium released by the test specimens is swept out of the capsules by flowing He and carried to ion chambers and bubblers for liquid scintillation counting to provide real-time measurement of total tritium release. All of the capsules employ active temperature control during irradiation via adjustable He-Ne gas gap mixtures regulated by thermocouple feedback. The focus of the present paper is on diagnostic modeling that will support interpretation of the in-situ data obtained from the nine flow-through capsules. Comparison of the data with model predictions will provide insight into the important mechanisms that may be governing the observed tritium release rates.

Paper 12B-2: Wednesday, April 20

Hydration-Carbonation of Gamma Lithium Aluminate

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Li-6 enriched $\beta LiAlO_2$ can be used as a target material for the production of tritium through a neutron capture reaction that yields both tritium and helium. Unfortunately, this ceramic material is susceptible to environmental degradation under ambient temperatures and humidities, and the degradation process accelerates rapidly as either of these increase. This process damages the microstructure of the ceramic, compromising its mechanical integrity, as well as chemically altering the phases present in the material. The degradation of lithium aluminate has been studied as a function of temperature and relative humidity using both tubular monoliths and ground specimens. Adsorption thermogravimetric analysis was used to characterize the hydration-carbonation reactions. The reaction appears to require a minimum of 50% relative humidity, and can be fitted with a third degree polynomial expression. Hot-stage x-ray diffraction and differential thermal analysis was used to analyze the decomposition process. The results show that Li_2CO_3, and $(LiAl_2(OH_6)_2)CO_3 \bullet XH_2O$ are the reaction products. The hydrates decompose by approximately 350°C, whereas the final carbonates (e.g. Li_2CO_3) are retained until 750°C. The final decomposition products include $LiAl_5O_8$ and $\beta LiAlO_2$ Scanning electron microscopy was used to evaluate the impact of hydration-carbonation reactions on the microstructure. The degradation process preferentially occurs at free surfaces, along grain boundaries, and at intra-granular interfaces. Potential strategies for minimizing this materials sensitivity to moisture will be discussed.

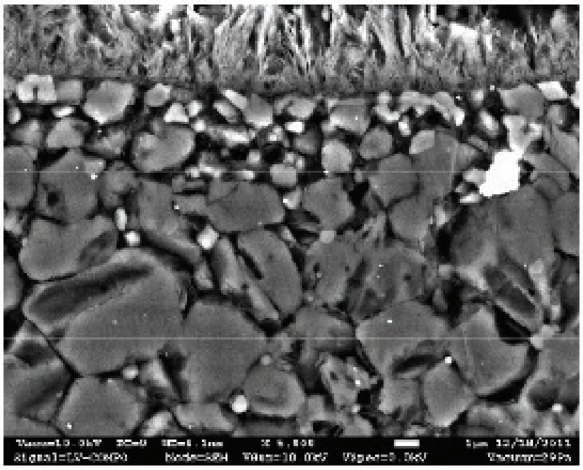

Figure 1. Degraded $LiAlO_2$ microstructure

Paper 12B-3: Wednesday, April 20

Preliminary System Modeling for the EUROfusion Water Cooled Lithium Lead Blanket

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The Water Cooled Lithium Lead (WCLL) blanket is one of the four breeder blanket technologies under consideration within the framework of EUROfusion Consortium activities. The aim of this work is to develop a preliminary model that can track the tritium concentration along each part of the WCLL blanket and their ancillary systems at any time.

Because of tritium's nature, the phenomena of diffusion, dissociation, recombination and solubilisation have been carefully taken into account when describing the tritium behavior inside the lead-lithium channels, the structural materials and the water coolant circuits. The simulations have been performed using the object oriented modeling software EcosimPro.

Results have been obtained for a pulsed generation scenario for DEMO. The tritium inventory in every part of the blanket has been computed. Permeation rates have been calculated as well allowing to know how much tritium ends up in the coolant system and how much remains in the liquid metal. The amount of tritium extracted from the lead-lithium loop has been also obtained. All this information allows to have a global perspective of tritium behavior all over the blanket at any time.

Besides, a parametric study has been performed. The influence of the lead-lithium flow, the influence of the extraction efficiency and the influence of the dissociation and recombination constant of EUROFER have been studied by tuning the corresponding parameters accordingly and analyzing its impact in the inventory, the permeation and the extraction at any time of the pulse. That way it is possible to deduce important information like which is the dominant factor that affects the permeation rate.

The model provides valuable information for the design of the WCLL blanket. More complex upgrades are planned to be implemented based on this model in future stages of the EUROfusion Project.

Overview of Tritium Target Program and Results of Lithium Aluminate

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The USDOE's National Nuclear Security Administration is responsible for maintaining the safety and reliability of the nation's nuclear stockpile. The current method for preparing tritium to replenish this stockpile is to irradiate Tritium Producing Burnable Absorber Rods (TPBARs) in the Tennessee Valley Authority's (TVA's) Watts Bar Nuclear (WBN-1) commercial light water reactor. A number of normal absorber rods are replaced with TPBARs, which are removed at each refueling outage and shipped to the Tritium Extraction Facility to collect the tritium for the stockpile. A TPBAR consists of a column of annular lithium aluminate pellets that is sandwiched between a zirconium-based, inner liner tube and an outer nickel-plated zircaloy getter tube enclosed in stainless steel cladding. This talk will give a historical overview of the Tritium Technology Program followed by discussion of the results of lithium leaching.

A potential issue has been the fate of the lithium aluminate absorber material during a loss of coolant accident (LOCA) at the reactor. During a LOCA it is necessary to understand the potential leaching of lithium from TPBARS. Lithium aluminate pellets were exposed to simulated post LOCA conditions in mini-TPBARs fabricated to mimic a guillotine breach. The mini-TPBARs were fabricated with irradiated and un-irradiated lithium aluminate pellets and leached for up to one year. These mini-TPBAR assemblies were exposed to 200°C in a sealed vessel charged with pH adjusted borated water. The lithium leaching durations varied from short term, (1-30 days), mid-term, (60-100 days) and long term (200-365 days). The lithium leaching progressed in a first order manner. After 365 days the irradiated pellets had leached approximately 30% of the total lithium, whereas the un-irradiated pellets leached approximately 25% of the total lithium.

Paper 12B-5: Wednesday, April 20

Tritium Permeation and Extraction in the Fusion Nuclear Science Facility

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The Fusion Nuclear Science Facility (FNSF), viewed as a critical step in the pathway to fusion energy in the US, is intended to demonstrate long-pulse burning plasma operation and provide for necessary materials and component testing in a fusion nuclear environment prior to pursuit of a larger demonstration power plant. Long-pulse burning plasmas require a sufficient supply of tritium fuel which the FNSF itself must provide, i.e. it must also achieve a tritium breeding ratio (TBR) of ~1. This work describes modeling of tritium transport in the current FNSF design, which employs a dual-coolant lead-lithium (DCLL) blanket, with MELCOR and TMAP. As with any fusion system, tritium losses from the blanket system must be kept acceptably low, with strict limits set by safety and environmental concerns. Such losses can be prevented by a combination of permeation barriers and efficient tritium extraction systems, and using the models we outline the requirements for both in the FNSF. Tritium extraction is accomplished using a vacuum permeator, long envisioned for use in a DCLL-type blanket. Based on the requirements identified by the modeling, we outline some necessary R&D on both permeation barriers and vacuum permeators that must be undertaken prior to operation of the FNSF.

Tritium Behavior in HCPB Breeder Blanket Unit: Modeling and Experiments

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Tritium behavior in a breeding blanket is a key design issue because of its impact on safety and fuel-cycle best performance. Nowadays there is no reference and fully validated tool with predictive capabilities in terms of determining the amount of tritium that will permeate in the cooling plates and in the coolant from the breeding zones, such as of the amount that will remain trapped in the solid blanket. Considering the difficulty in handling tritium, it is intended to prepare a simulation tool for tritium transport in the HCPB breeder zone/breeder unit.

A preliminary model for tritium transport at component level has been developed, focusing on the multi-physics of the release, diffusion, permeation, recombination phenomena in a breeder unit scale. The numerical technique here presented is based on EcoSimPro simulation tool, a program with an object-oriented nature which offers the possibility of mixing various disciplines by robust equation-solving algorithms. The objective of this work is to evaluate tritium inventory inside several components of the breeder unit (pebble bed and purge gas) and its losses into the cooling plates considering a discretized profile in all the components.

The model is based on the integration of fluid flow, concentration and tritium mass transport phenomena together with the isotope interactions between tritium atoms and the atoms of the purge gas flow. We adopted some simplified assumptions and the results obtained with our simulations are compared with others studies obtained by different tools. Based on this, more complex upgrades (including the coolant purification system, the tritium extraction system and the tritium behavior in the beryllium pebble bed) are planned to be implemented in the next future.

Paper 13A-1: Thursday, April 21

Compact Determination of Hydrogen Isotopes and Helium

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Gaseous mixtures of diatomic hydrogen isotopologues and helium are often encountered in the nuclear energy industry and in analytical chemistry. Compositions of stored mixtures can vary due to interactions with storage and handling materials. When tritium is present, it decays to form ions and helium-3, both of which can lead to further compositional variation. Monitoring of composition is typically achieved by mass spectrometry, a method that is bulky and energy-intensive. Mass spectrometers disperse sample material through vacuum pumps, which is especially troublesome if tritium is present. Our ultimate goal is to create a compact, fast, low-power sensor that can determine composition with minimal gas consumption and waste generation, as a complement to mass spectrometry that can be instantiated more widely. We propose calorimetry of metal hydrides as an approach to this, due to the strong isotope effect on gas absorption, and demonstrate the sensitivity of measured heat flow to atomic composition of the gas. Peak shifts are discernible when mole fractions change by at least 1%. A mass flow restriction results in a unique dependence of the measurement on helium concentration. A mathematical model is presented as a first step toward prediction of the peak shapes and positions. The model includes a useful method to compute estimates of phase diagrams for palladium in the presence of arbitrary mixtures of hydrogen isotopologues. We expect that this approach can be used to deduce unknown atomic compositions from measured calorimetric data over a useful range of partial pressures of each component.

Our presentation will discuss our recently published paper in the International Journal of Hydrogen Energy, available online at http://dx.doi.org/10.1016/j.ijhydene.2015.08.033

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Upgraded Analytical Gas Composition Technique in the Tritium Fuel Cycle of JET

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Operation of a magnetic-confinement fusion-reactor requires monitoring of the total gas composition at different stages of the tritium re-cycle process. The main focus of the analyses is assessing a) the concentration of hydrogen isotopologues (H2, HD, HT, T2, DT, T2) and helium 'ash', b) impurities like noble gases, air components or hydrocarbons.

At JET, gas composition analyses are performed via an Analytical Gas Chromatographic Isotope Separation System (AN-GC) installed in a depressurized glovebox within the tritium plant: the Active Gas Handling System [1]. This is the only analytical tool available on site, where the complete analyses of a tritiated gas sample and therefore tritium accountancy can be carried out. The fact that tritium is a radioactive element and the requirements for fusion specific analytical techniques imply that such a system is not commercially available. As the analytical requirements have changed since the first AN-GC system was installed in 1989 and in view of the upcoming JET Deuterium-Tritium Campaign (DTE2) a new AN-GC system has been designed. The new AN-GC system is based on an off-the shelf analytical gas chromatograph unit which will undergo major modifications to fulfil JET requirements and safety criteria to be tritium compatible.

This paper will give details of the existing and the new requirements such as analyses of hydrogen isotopologues, air component, long-chain hydrocarbons, noble gases, ammonia and water and the challenges for the AN-GC system. The new system will be equipped with several bespoke features amongst which i) separation column for the hydrogen isotopologues, ii) new design of small volume radiation detectors for tritium, iii) closed chimney Flame Ionization Detector (FID) to confine and collect the exhaust gas stream into the AGHS primary system, iv) installation of a cryogenic Dewar inside the glovebox. Moreover, in order to be prepared and respond to further analytical requirements (e.g. analyses of components which are not calibrated with the gas chromatograph) a mass spectrometer (1.6 to 1050 AMU) will be coupled to the GC exhaust. The ability of the system of analyzing all requested components with one single sample injection and the installation of the AN-GC coupled with a mass spectrometer in order to identify non-calibrated GC components, are unique and the first in a nuclear fusion laboratory.

This work has been carried out within the framework of the Contract for the Operation of the JET Facilities and has received funding from the European Union's Horizon 2020 research and innovation programme. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

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Design and Operation of a Monitoring System Which Separates and Measures High and Low Concentrations of Tritium in Air

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Nuclear power plant accidents like Fukushima may be a thing of the past with the advent of thorium reactors versus uranium fueled reactors. Thorium is more abundant, less melt-prone, and therefore safer to use. China and the US DOE are partnering to develop a Thorium reactor in China. If it succeeds, nuclear power would become more efficient, cheaper, and safer than today's uranium-based reactors. It would also produce less nuclear waste. However it produces Tritium in copious quantities. The experimental fusion reactor, the ITER project in France is using and recycling Tritium waste as fuel. Despite efforts to utilize Tritium, separating and measurement of high and low concentrations of Tritium in air is a critical environmental consideration.

Overhoff Technology, a subsidiary of US Nuclear Corp, has designed an instrument for the separation of high and low Tritium in air. The primary design principle is a semi-permeable membrane used to isolate Tritium Oxide from all other sample constituents such as other radioisotopes, pollutants, aerosols, and particulates prior to measurement. The measurement system consists of dual low level proportional counters and dual high level ionization chambers. An oxidizer is used to separately measure total Tritium or Tritium Oxide.

For measuring ulta-low levels of Tritium a dual proportional chamber system is used because Gamma compensation is necessary. The simultaneous pulse rise and duration signal processing selects only those pulses corresponding to Tritium Beta decay.

For measuring high levels of Tritium a dual ionization chamber measurement system is used, again because Gamma compensation is necessary. These have a 2 liter nominal volume capacity for each chamber.

An Integrated Bubbler-LSC for On-Line Measurements of Gaseous and Aqueous Tritium

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Sensitive measurements of tritium vapor-in-air require a bubbler to capture and concentrate the tritium for off-line liquid scintillation (LSC) measurement. This process is slow and labor intensive. To address these issues, a collaboration between LabLogic Systems Ltd and the National Physical Laboratory (NPL) has been established to develop an automated, on-line radioactive gas monitor. The automation of sample collection, preparation and measurement will reduce labor requirements. On-line measurement will reduce the delay between sampling and measurement analysis.

The proposed instrument is based on LabLogic's on-line radiation-in-water monitor, WILMA. The WILMA instrument will be upgraded to incorporate a bubbler train, furnace and automatic sample transfer which will extend its capability to monitor for HTO, HT and tritiated organics in both aqueous and gaseous streams. In addition, the instrument will be capable of monitoring for C-14 in air and Sr-90, Y-90, Tc-99, Cs-137 and Am-241 in aqueous streams. As such, the instrument will be a flexible solution for nuclear sites who may be interested in contaminated ground water, liquid effluent or gaseous discharges.

Presented as an oral presentation detailing the specification, design and intended applications of the proposed integrated bubbler-LSC instrument. This work is partly funded by the EMRP through the MetroDECOM project. The EMRP is jointly funded by the EMRP participating countries within EURAMET and the European Union.

Implementation of a New Tritium Monitor Calibration Facility

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Following changes in the regulations governing the use of Tritium monitors in France combined with an extended period of growth in our business coupled with the lack of availability of a reliable test laboratory, Premium Analyse, a world leader in the development and marketing of Tritium monitors, took the decision to develop a tritium test facility in house.

This paper is written to describe the activities and decisions necessary in order to design and implement a new calibration facility which is not only compliant with all the French regulations in force and those that are foreseeable, but also to take into account the requirements of the tritium users and the applicable regulations outside France to create a capability not available elsewhere to our knowledge.

The subjects covered include:

- The reasoning to develop this capability along with the evaluation of the likely trends in the future requirements worldwide.
- The considerations necessary when developing the capability to have a traceable, usable, modular and flexible system.
- A full description of the technical solutions covering all the perceived needs.
- The results obtained with typical tritium monitors available including the ability to supply a monitoring system in real time with changing calibrated concentrations of HT, HTO and mixtures of the two.

This has resulted in a unique capability which allows for quick and accurate calibration of both our monitors and those of other suppliers. This system allows calibration of the systems in the presence of natural levels of Radon progeny, something that was not tested with sealed systems. This capability allows us to offer a full calibration service to the market, but in addition gives great benefits when new products need development.

As a by-product of this development many clients are asking us to provide standalone calibration rigs which can justifiably provide local calibration to the relevant standards at a very cost effective price and avoiding the downtime and transportation costs associated with traditional calibration procedures.

Tritium Impurities Measurement with Gas Chromatography

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Various analytical techniques are available to determine the composition of tritium gas mixtures. In addition to the mass spectrometry, gas chromatography is conventionally used to measure the gas impurities concentration contained in tritium after purification stages. The main difficulty with the chromatography is the implementation and routine work within tritium processes. Two of the main concerns are the work at low pressure and the tritium compatibility of materials.

With the aim of overcome these issues, the CEA in collaboration with INTERSCIENCE Belgium has developed a new apparatus of Compaq Gas Chromatography (CGC) dedicated at working in a tritium glovebox. This paper discusses the characteristics of this new technology and the associated analytical results in terms of measurements of gas impurities (He-3, CH4, N2, O2/Ar) in tritium matrix.

Field Testing and Intercomparison of Advanced Tritium Transfer Models

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Decades of experimental investigation and modeling of tritium transfer through atmosphere-vegetation-soil system resulted in an harmonized approach to key processes encountered in advanced models. Examples include foliar atmospheric HTO uptake, removal of soil HTO via the evapotranspiration route, formation of organically bound tritium through carbon fixation in photosynthesis and parameterization of feedbacks between these processes and external drivers (solar radiation, meteorology and soil moisture availability) responsible for stomata opening. Such conceptual similarity between modern tritium models provides a convenient platform for continuous and realistic evaluation of field-scale transfer of airborne tritium, compatible with sparse and highly variable point observations.

Participants of the Tritium Work Group of IAEA MODARIA (Modelling and Data for Radiological Assessments) conducted simultaneous blind testing of four models against three field scenarios. Participants tested and compared advanced tritium models including TOCATTA-(implemented in the SYMBIOSE platform) of IRSN, France, CTEM-CLASS-TT of CNL, Canada and SOLVEG-II of JAEA, Japan as well as more traditional model CERES of CEA, France. Two field experiments were conducted in Canada: one at the Chalk River site by CNL in 2008 and the other near a tritium light facility in Pembroke by the CNSC in 2012. The third and most elaborately instrumented field experiment began in 2013 near the La Hague processing plant in Cherbourg, France under VATO Programme.

The diversity provided by three field campaigns and the similarity of key modeled processes allowed participants to determine the minimal complexity requirements of advanced tritium models for adequate quantification of environmental transfer of airborne tritium

Oceanic Behavior of Tritium Upon Deep-Sea Release

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Nuclear fusion power plant operation cannot avoid releases of tritium to the environment. One of the major expected emission sources is the cooling water: nuclear fusion power plant with an output power of 1 GW is estimated to release ~80 m3/s of HTO-containing cooling water to the ocean. After release, tritium in the cooling water evaporates to the atmosphere or contaminates the food fish to be caught in the area if concentration is significant.

The authors propose releasing of cooling water to the deep sea as a possible mitigation method for this problem. By releasing the HTO-containing cooling water to the deep sea, the tritium would stay in deep waters for a period of time, preventing the tritium to evaporate or to contaminate the food fish in the area if the behavior of the water is favorable. Given the relatively short half-life of tritium of 12.3 years, this method would keep majority of tritium underwater until its decay, minimizing the amount of tritium migrates to the surface.

In this study, the authors simulated and compared behaviors of tritium released to the ocean for four scenarios: release to the surface level, to 50 m deep, to 200 m deep and to 1,000 m deep from the east coast of Japan. The Princeton Ocean Model was used as the oceanic model in this study, and the particle tracking method through Lagrangian TRANSport model (LTRANS) developed by University of Maryland Center for Environmental Science was used to simulate the behavior of tritium as particles. For salinity, temperature and current data required for this study, published data of OFES (OGCM for the Earth Simulator) by JAMSTEC was adopted. Simulated particle distributions were then converted to particle density and radiation activity of tritium factoring the half-life into the calculation.

The results showed that releasing HTO to 200 m or deeper ocean prevents majority of tritium from entering to food fish habitation area (0-200 m deep) for 45 days, indicating the potential benefit of this method. However, it has to be noted that the effect of this method may vary depending on local conditions, especially the temperature of the cooling water and the local ocean current.

Paper 13B-3: Thursday, April 21

A Comparison of Tritium Dispersion Methodologies for Accident Analysis in U.S. Department of Energy Complex Facilities

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In the U.S. Department of Energy (DOE) Complex, new and existing nonreactor nuclear facilities, including those storing, processing, and handling tritium, must meet the applicable Code of Federal Regulation (i.e., 10 C.F.R. Part 830) requirements for the preparation of documented safety analyses (DSAs). The primary methodology for performing this function, DOE-STD-3009-2014 (Preparation of Nonreactor Nuclear Facility Documented Safety Analysis), was revised in late 2014 and now contains three options for performing the atmospheric dispersion and consequence analysis. This analysis and the associated source term estimates support the estimation of DSA radiological dose estimates. The dose estimates are applied as part of a process for determining whether structure, system, component (SSC) need to be identified to prevent or mitigate radiological exposures from postulated accident events in the analyzed facility.

The three options yield distinctively different dose results when combined with a given source term for the nuclear facility. Each of the DOE-STD-3009-2014 options is characterized by differing sets of assumptions and inputs as well as increasingly more realistic modeling. The options used to evaluate atmospheric dispersion include:

- Option 1: A process based on Nuclear Regulatory Commission (NRC) Regulatory Guide 1.145, Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants;
- Option 2: Use of a DOE-approved toolbox code and application of conservative input parameters; or
- Option 3: Use of site-specific methods and parameters as defined in a site/facility specific DOE-approved modeling protocol.

In this paper, unit sources of tritium oxide and tritium gas are used in combination with the three DOE-STD-3009-2014 options to evaluate the resulting radiological exposures for a U.S. DOE site for the three options. The DOE Central Registry code, MACCS2 (Chanin, D. and Young, M., 1998), is applied for Options 2 and 3, and the results are compared to those obtained with the tritium radiological dispersion code, UFOTRI (Raskob, W., 1993). Key objectives of the analysis are to quantify the level of conservatism among the options and between the two computer codes, and to evaluate the sources for the differences in the dose outcome.

Investigation of the Potential Impact of Storage Place on Tissue Free Water Tritium and Organically Bound Tritium Activity Determination. Feedback of an Interlaboratory Exercise.

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When focusing on tritium environmental monitoring water is the most common matrix. Nevertheless, a part of these monitoring concerns also Free Water Tritium (TFWT) and Organically Bound Tritium (OBT) analysis in environmental samples. Moreover, aiming to improve knowledge on tritium behavior and migration, TFWT and OBT determination are also performed for expertise works at low activity level. TFWT and OBT determinations could also be performed after incidental releases to determine potential impact to the environment and to Man. As hydrogen isotopes are prone to isotopic exchange with atmospheric moisture, the key point when analyzing TFWT and OBT will be to preserve the sample from its sampling to its analysis.

Aiming to answer to the following question; does the storage condition could change activity level within the sample; an interlaboratory exercise was organized within the framework of the French tritium working group to follow both TFWT and OBT activity scrutinizing for any variation on a one year period. For that about 50 kg of grass were sampled dried, homogenized, packaged and frozen prior dispatching to the participants. Six aliquots were sent to each participating lab to determine both TFWT and OBT, starting when all the labs received the sample, continuing two weeks, one month, three months, six months and one year after to check for any variation in the measured activity.

The results were treated for each analysis date, for each participating lab and considering the overall population by calculated the mean and associated standard deviation. As all values are consistent, the main conclusion of this work is that TFWT and OBT activity concentration remain constant when appropriate storage conditions are used. This point is of importance as sample conservation is always a critical parameter for tritium determination in environmental samples and also when considering the fabrication of certified reference materials.

Relevance of Night Production of OBT in Crops

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Tritium has a complex behaviour once released into the environment. In the past 15 years it was noticed that organically bound tritium (OBT) production during the night time in crops cannot be ignored and, unfortunately, the present models do not consider it adequately.

During the day time, the OBT production in crops can be directly related to the actual photosynthesis rate and actual concentration of tritiated water (HTO) in leaves. For night time, the maintenance and growth are based on starch /sucrose reserves in leaves accumulated under light. The starch is gradually hydrolysed during the night time in soluble sugars (sucrose) which is used for night metabolism. HTO can enter in various biochemical reactions during the starch hydrolyse and subsequent processes and OBT can be produced. The following processes are analysed: a) The role of starch and circadian clock in plant metabolism; b) The contribution of the night processes to the overall maintenance and growth of the plant and to the edible plant parts during the development stage; c) The night and day canopy respiration, correlation with canopy transpiration and night growth; d) The role of protein turnover in maintenance respiration; e) The meaning of the night biochemical reactions producing OBT and how these reactions depend on plant type and genotype.

The carbohydrate dynamics in leaves is a highly regulated and complex process and is not well understood (e.g. respiration paradigm, starch and circadian clock). For an operational model many simplifications are needed. The first simplification is to ignore the protein turnover. In this case, tritium transfer can be considered based on a simple water addition process over a skeleton model. The skeleton model considers starch and soluble carbohydrate dynamics in leaves and leaves export (after leaf respiration). The fraction of photosynthesis used for starch production and the usage rate of sucrose are model parameters. The starch hydrolyses rate is considered in four typical situations (linear rate, wheat, maize and soybean data). The analytical or numerical methods are used for the dynamics of leaf starch and sucrose during the development stage of plants. The time step is less than one hour and the plant growth is reproduced. For tritium, a simple balance (stoichiometry) of water addition reactions are used for tritiated sucrose, starch formation, as well as starch hydrolyses rates. The OBT formation highly depends on leaf HTO dynamics and the night canopy conductance and transpiration are of special concern.

The operational model is tested favourably for wheat and rice, where experimental data base is large and the environmental data and crop growth characteristics are documented. The potential application to other plants of interest is discussed. Finally, the protein turnover is considered in a simplified manner. Further experimental work is suggested (interdisciplinary research).

Paper 13B-6: Thursday, April 21

Specific Activities of OBT in Soil around Nuclear Power Plant

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As one of the most important environmental mediums, soil plays a significant role in tritium migration and conversion. In order to explore the distribution and formation mechanism of OBT in soil, the spacial distribution and particle size distribution of OBT in soil around Qinshan Nuclear Power Plant in China, were investigated in this study.

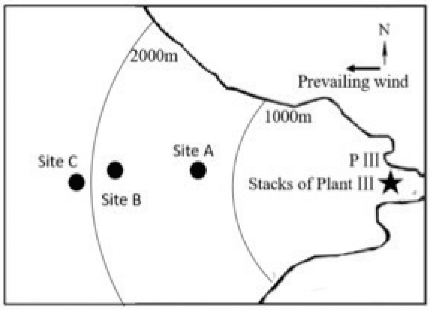

Fig.1 Sites of collection of soil samples from the Qinshan NPP

As shown in Fig.1, 4 soil sampling locations (PIII, A, B, C) were chosen considering wind direction, distance to the stacks of Plant β(two HWRs). Soil cores in subsurface (25cm below) were sampled at A, B, C and surface soil (0-5cm below) at Pβsites were collected. Each soil sample was divided into different groups according to their particle sizes using sieve shaker, including 2000-250μm, 250-53μm, 53-20μm and <20μm. The OBT in soils was converted to HTO using combustion technique and the HTO concentration was determined by LSC (OBT in <20μm soil particles was not measured because of the little sample size).

In terms of the results of OBT in surface soils (0-5cm) at different sites, the OBT concentrations in the soil of Pβ and site A are signicantly higher than that of site B and C, revealing the correlation between OBT activities and the distance to the HWRs. The vertical profile of OBT activity in soil collected at site A, B, C demonstrates that the OBT activities in surface soil are lower than that in deep soil. The different kinds of organical matters, microrganisms and the variation of HTO concentration caused by evapration in surface and deep soil are likely responsible for the variation. The variation of OBT with different soil particle sizes indicates that OBT prefer to be concentrated in medium soil particle sizes (250-53μm) in deep soil and the kinds of microorganisms in different soil sizes may have a correlation with the variation.

Radioactive Characterization of Tritiated Heavy Water using ESR Spectrometry

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Usually, the radioactive concentration of tritiated liquids is determined by Liquid Scintillation Spectrometry (LSC). The measurement of radioactivity for tritiated heavy water (DTO) with high specific activities using LSC implies multiple isotopic dilutions leading to liquid radioactive wastes with medium activities.

In the present study, a novel method for determination of radioactive concentration in samples of tritiated heavy water was proposed. The method relies on the accumulation and detection of free radicals radio-induced by self-radiolysis in frozen samples.

The ESR analysis of the samples stored in liquid nitrogen confirmed the accumulation of DO• radical (triplet with $g = 2.013$ and $a = 0.69$ mT). Because is not stable at the -196° C temperature, D•, the secondary radical generated in self-radiolytical processes of the DTO was not identified.

The linear dependence between ESR signal of the DO• radical and storage time at -196° C temperature suggests a constant accumulation of DO• radical. But a recombination tendency of the formed radicals at high accumulation time has been identified.

The ratio between the ESR amplitude signal and the radioactive concentration is linear on the analyzed range of the radioactive concentrations.

The obtained results sustain the possibility to use the ESR spectrometry in determination of radioactive concentration for high activity DTO samples.

Paper 14A-2: Thursday, April 21

Design of a Spectroscopy Experiment for All Hydrogen Isotopologues in the Liquid Phase

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An integral part of the fuel cycle of future fusion facilities is the isotope separation system (ISS). The Tritium Laboratory Karlsruhe (TLK) is currently operating a combined ISS/WDS (Isotope Separation/Water Detritiation System) as a contribution to this development [1].

A wide range of concentrations of the hydrogen isotopologues Q2 (H2, HD, D2, HT, DT, T2) will occur in the cryogenic distillation columns of the ISS, whereby liquid tritium T2 is accumulated at the bottom. In order to monitor and control the separation process, an inline and real time analysis method is necessary to measure the Q2 concentration in the liquid phase.

Liquid inactive hydrogen isotopologues were already successfully analyzed via infrared (IR) absorption spectroscopy and calibration data for D2 is provided by previous experiments at TLK [2], [3]. Based on this knowledge, the new experiment T2ApIR is designed to be fully tritium compatible to perform a calibration of the IR absorption measurement system with all six hydrogen isotopologues in the liquid phase under conditions similar to the ITER-ISS. Therefore, T2ApIR will provide a combination of a chemical and a coolable ortho/para catalyst, in order to provide isotope mixtures with non-equilibrium chemical and ortho/para compositions. The produced gas mixtures are examined with a spectroscopic Raman analysis system [4], able to simultaneously determine the chemical and ortho/para composition in the gas phase.

This contribution presents the design of the new T2ApIR experiment with the focus on the calibration of the IR absorption measurement system for the concentration of all six hydrogen isotopologues in the liquid phase.

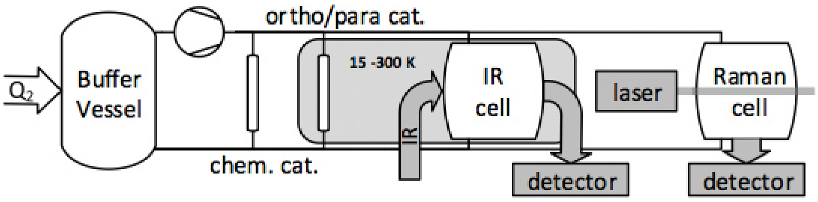

Figure 1: Schematic of the tritium compatible T_2ApIR experiment

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Paper 14A-3: Thursday, April 21

Isolating Low Concentration of Tritium in Potable Water using Simple Designed and Fabricated Process

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Tritium in everyday water (portable water) is frequently of a level that is too low for measurement with conventional instrumentation that is affordable by small laboratories. Scintillation counters that can measure in fraction Becquerel per litre (Bq/L) are usually out of the reach of most laboratories, especially in developing countries. By concentrating the tritium to a predetermined amount, it can reach measurable levels that can be converted back to the original concentration. Affordability of the concentrating process is vital in the overall process. This work is not about inability or non-availability of equipment that can detect low concentration of tritium in water, but about affordability of such equipment in the laboratories of many establishments, particularly in developing countries. The simple and affordable enrichment process will make this possible.

A simple concentrating process based on purification and electrolysis was designed and fabricated. The purification process comprised of an evaporation, condensation, and deionization of the sample water. This was followed with electrolysis using proton exchange membrane (PEM) electrolyser. Though the alkaline electrolyser generates higher enrichment factor than the PEM electrolyser, the PEM electrolyser was chosen in this work since the product of the electrolysis would not require the removal or reduction of hydroxide concentration. The tritium isotope enrichment level, the volumetric reduction and the time frame required for the enrichment were determined using the simple designed and fabricated process, and an easily affordable scintillation counter. Reference readings were obtained using known quantity of tritium in the electrolysis process.

To ensure that no special condition will be required for the process, relationship between temperature and time of the electrolysis, water consumption rate, and tritium content of various portable water were determined. The simple designed and fabricated system effectively concentrated the tritium in the sampled water by several factors

A process simple enough to be fabricated in a small tools shop, and using mostly readily available shelf components was designed, fabricated, and used to enrich tritium in various portable water samples. The enrichment resulted in the output product being measured in non-expensive scintillation counter.

Development of Fast-Response Solved-Tritium Concentration Diagnostics

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Dynamic control of tritium inventory is required for flexible and reliable operation of complex tritium plant experimental systems. In ITER, the extremely demanding tritium inventory management requirements imposed by regulators could drive into static tritium accountancy strategies consisting in the full transfer of tritium to measuring beds for in-situ calorimetry. Such static strategies anticipate serious Plant operational drawbacks. Thus, a final availability of on-line/real-time tritium concentration diagnostics in-plant-effluents linked to new dynamic control strategy will impact strongly operational flexibility of ITER or its safety device. On-line/real-time diagnostic solutions and performances are today unavailable. At present, tritium cannot be reliably monitored real-time/on-line in pressurized Helium (in the required vppm ranges) nor in LLE (Lead-Lithium Eutectic) and such unavailability does not permit tritium inventory dynamic control nor real-time tritium balance matched certifications. In tritium breeding systems, the conventional high diffusivity (βFe, Nb, PdAg,) sub-mm thin diffusion membrane sensors in LLE (in 1-100 Pa partial pressure ranges) show time-lag diffusion responses larger than ITER pulse duration that makes unreliable bred concentration dynamic real-time monitoring.

Electrochemical monitoring principles based on PEM (Proton Exchange Membrane) sensors have been proposed (for pressurized gases and LLE) as sensing principle being developed and qualified in key essential aspects: 1) response calibration ranges; 2) PEM material stability in LLE; 3) discrimination of isotopic effects; 4) rad-hash environment impact on sensor cap and MI cable electronics; 5) LLE and tritium aging lifetime effects on PEM; 6) other.

In this work, a new sensor cap principle is proposed for tritium real-time/on-line measurements as a solute in LLE or as free gas in He-pressurized phase. The new cap principle is based on the use of double-wall/dual-side thin porous membrane: a closed high tortuous porosity structure in contact with LLE and a connected porosity with low tortuosity. Today, fast prototyping additive powder manufacturing techniques can be tailored for porous cap fabrication with convenient material (high diffusivity material βFe or alloy with high material versus LLE solubility ratio). The cap dual-porous topology: (1) permits a very high specific contact area of lead-lithium with the porous structure for solved gas transfer limiting LLE penetration by capillarity; (2) minimize response times by several orders of magnitude through the reduction of effective diffusion path from solid membrane down to inter-pore thickness; (3) minimize low concentration threshold through the reduction of open interpore/capillar desorption dead volume; (4) avoid PEM chemical stability issues in LLE.

Together with anticipated theoretical responses our paper presents general design features of optimized dual-walled cap design topology and further analyses for manufacturing routes and qualification.

Paper 14A-5: Thursday, April 21

Tritium Measurement with Organic Waste-Less Method

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Introduction Tritium measurement is important for fusion science and technology development. Until now, a liquid scintillation counter (LSC) has been used to measure tritium; however, there is an issue of generation of radioactive organic wastes, which are environmental loads. For LSC measurement, we developed a novel plastic scintillator (PS) method to substitute PS for liquid scintillator1,2). Radioactive organic wastes were never generated by using PS; however, there was a limit for available sample volumes with a normal 20 mL vial. Therefore, we developed a special plastic scintillation counter (PSC) for PS-sheets3) and PS-cylinders. We report here the new PSC device and a method for tritiated water measurement using PS-cylinders which was used as an alternative material of a general liquid scintillator.

Experimental For volatile sample measurement, PS-cylinders of EJ-200 (G-teck, Japan), which was a 3 mm cylindrical form in both of diameter and length, was used by filling full in an container for PSC. It was approximately 34 g per each vial. A screw cap for a vial had a cone gum inner (uGV2-500-1: Meridian Biotechnologies Ltd.). Sample of tritiated water (HTO, Moravek) was diluted approximately from 1.3 Bq mL-1 to 500 Bq mL-1 with distilled water and was applied to a container filled with PS-cylinders using a pipette. Then, a cap was immediately screwed to the container. The counting efficiency was measured with diluted series of HTO and a detection limit was calculated by using a background of the PS-cylinders.

Results and discussions When a 5 μL of HTO was applied to the 20mL vial, the counting efficiency was more than 50% after elapsing 2 hours, which was higher than that of liquid scintillator. Four hours were necessary to get stable counts that the sample solution evaporated in interspace of the container. When 2mL of HTO was measured, the counting efficiency was approximately 3.5%, and the detection limit was the lowest when the sample volume was 2mL. Because a detection limit calculated with a background of PSC was 0.3 Bq mL-1 at 10 h measurement, the PSC, which has no shield, needs some shield to decrease the background.

Conclusion The PSC with PS-cylinders showed high measurement efficiency and it was effective because the device did not generate radioactive organic wastes.

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Paper 14B-1: Thursday, April 21

Overview of the European R&D on Tritium Technology for DEMO Breeding Blanket

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Along the pathway towards nuclear reactors using the Deuterium Tritium fusion reaction, the demonstration reactor DEMO shall prove the Tritium self-sufficiency of the machine. The several hundred grams of tritium daily consumed by the machine shall be produced inside the Breeding Blanket (BB) and extracted from it via the Tritium Extraction System (TES). Within the EUROfusion Power Plant Physics and Technology studies, the BB project is working in parallel on four different BB concepts: Helium Cooled Pebble Bed (HCPB), Helium Cooled Lithium Lead (HCLL), Water Cooled Lithium Lead (WCLL), and Dual Coolant Lithium Lead (DCLL).

The TES designs of the four BB have to be tackled in an integrated manner optimizing the synergy with the directly interfacing inner fuel cycle, while minimizing the tritium permeation into the coolant. Considering the very high DEMO requirements in terms of TES throughput and efficiency, it is likely that the techniques currently foreseen for the TES of the ITER Test Blanket Modules are not adequate, so that only advanced and powerful technologies would be suitable for the TES of the DEMO BB. The EU R&D activities on Tritium technology for the BB are organized in four strongly interconnected main areas of work:

(a) Design: conceptual design of the tritium extraction system (TES) of the four BB;
(b) Simulation: tritium transport modelling at both TES and breeder unit levels;
(c) Processes: experimental work to consolidate advanced extraction technologies;
(d) Tritium Control: experimental work on permeation mitigation strategy.

This paper aims to overview the five year work program and objectives, and to summarize the main results achieved so far. Emphasize is given on the experimental work to demonstrate advanced technologies such as vacuum sieve tray or permeator against vacuum for tritium extraction from PbLi, and advanced inorganic membranes and catalytic membrane reactor for tritium extraction from He. Other activities related to the TES conceptual design, simulations on tritium transport, and experimental activities for tritium control are also presented.

Preparation of a Core-Shell Adsorbent for Lithium Isotope Separation

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Lithium-6 is used as a blanket material in DT fueled fusion reactors. A technology to enrich lithium-6 over 90 isotopic percent is, then, necessary to establish the fuel cycle. Though the lithium-mercury amalgam method had developed in practical use, an environmentally preferable method such as ion-exchange method needs to be developed. A good separative performance by displacement chromatography has been reported using a crown ether resin immobilized on porous silica beads as adsorbent. An equilibrium separation factor of the crown ether resin is about 1.04 and is larger than typical cation-exchange resins by an order of magnitude. On the other hand, the height equivalent to a theoretical plate (HETP) of such a crown ether resin was reported as a few centimeters and is relatively larger than typical cation-exchange resins. One of the reasons which cause the large HETP value was pointed out as a large mass transfer resistance in intraparticle diffusion. We, then, propose to use a 'core-shell type' particle for a carrier of the adsorbent. The purpose of the present study is to prepare an adsorbent using a core-shell type of silica beads.

Two types of porous silica beads were used in the present study. One was a core-shell type (Phenomenex, Kinetex 5 μm HILIC) and the other was a fully porous type (Phenomenex, Luna 5 μm HILIC). Benzo-15-crown-5 was selected as an adsorbent and xylene solution of the adsorbent was filled into the pore of both types of beads. Enrichment factors and HETP values for the packed column were experimentally obtained and reported. The overall separative performance of the process needs to be investigated in further research.

Lithium Enrichment Needs for Tritium Breeding

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For tritium breeding in fusion power plants, large amounts of lithium are needed in the breeding blankets. Here, the lithium reacts with the neutrons generated in the plasma and forms tritium that is being extracted from the breeder material.

Natural lithium contains the two isotopes, 6Li and 7Li. Most important for tritium breeding is the isotope 6Li, as it has a significantly higher cross-section for the breeding reaction. Unfortunately, the natural occurrence of 6Li is only 7.4 % whereas 40 – 90 % are needed in the blankets. An enrichment process is thus needed to adjust the isotope composition. In history, lithium enrichment was done excessively in the 1950s and 1960s in the K-25 site Oak Ride, TN, US. The process used there was called COLEX and applied large amounts of mercury. This facility was dismantled in the 1980s. Since then, no large-scale production facility exists anymore and the relatively small demand on global scale is satisfied by the stockpile built up at this time.

For a demonstration fusion power plant (DEMO), currently under preparation in Europe with the goal to produce fusion electricity to the grid in the 2050s [1], some ten tons of 6Li will be needed; for future fusion power plants, even more 6Li will be needed. It is unclear where this large amount will come from, as no production facility exists that could supply this market.

Even worse, detailed information on lithium enrichment processes is most difficult to retrieve as this work has been done more than 50 years ago for military applications. It is therefore required now to re-start activities in this direction to develop a reliable supply path.

This paper discusses the need of lithium enrichment and presents the isotope separation processes developed in the past. Based on this, an assessment is done leading to the proposal of an enrichment process suitable to satisfy in due time the demand for DEMO. Furthermore, an accompanying R&D program is suggested.

[1] F. Romanelli et al., Fusion Electricity – A roadmap to the realization of fusion energy,
EFDA Garching, Germany, November 2012.

Paper 14B-4: Thursday, April 21

Tritium Recovery Efficiency under Array of PbLi Droplets in Vacuum

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The authors discovered the two orders of magnitude enhancement of hydrogen isotope release dissolved in a falling PbLi droplet in vacuum. It is expected to be applied for the efficient tritium recovery from breeding blankets. Single nozzle experiments were performed, whereas multiple droplet nozzles are required to fulfil the mass flow requirements for the scale blanket modules for DEMO. In the multiple nozzle systems degradation of extraction efficiency due to absorption of once released tritium to another droplet, as schematically shown in Fig. 1, is concerned. This study presents the analysis of the net extraction efficiency and contribution of design configurations, such as mutual nozzle pitch, number of nozzles, release chamber size, and tritium exhaust port location, etc.

Analysis is performed under the worst case assumption, i.e., all re-absorbed tritium on another droplet dwells there and not release again during falling. The black body radiation theory is applied to presume the probability of interception by another droplet, as schematically shown in Fig. 2. Release of the molecule under rarefied gas condition is based on the Lamberts' law, same with the radiation from the black body. Results show the most significant effect to the efficiency reduction can be caused by the exhaust port design, port size and numbers. By the optimal design, net efficiency of 60% is expected. Plural number of reflections, of which the released tritium passes through the rain of droplets each time until finally attained the exhaust port, have decisive effect. The nozzle pitch has less significant effect, though not negligible.

The results suggest that the tritium extraction under multiple droplets is viable. Since this study is based on the conservative assumption and a simplified model, the experimental verification is currently in progress to estimate the actual effects in the multiple droplets system, and the optimized experimental setup is intended to be disclosed at the conference.

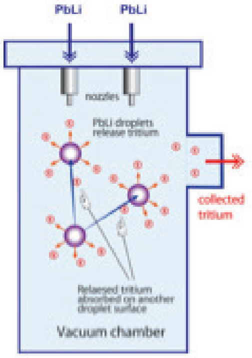

Fig. 1 Schematic of the released tritium behavior

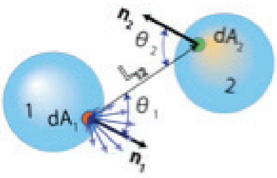

Fig. 2 Analysis model

Paper 14B-5: Thursday, April 21

Development and Modelling of a Multi-Nozzle Vacuum Sieve Tray Extraction Facility

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Tritium extraction from Pb-16Li liquid breeding blanket seems still very challenging with the prospect of DEMO; advanced and very efficient technologies are likely required to ensure a proper tritium management in fusion plants. The Vacuum Sieve Tray (VST) technique consists in letting the liquid metal flow through submillimeter diameter nozzles in a chamber maintained under vacuum where droplets are formed from which tritium can be extracted. This technique was previously rejected, but new experimental results obtained with deuterium and single nozzle VST highlighted enhanced extraction likely due to the high frequency oscillations of the Pb-16Li droplets when falling [1]. Since VST is relatively simple and compact, it is considered now as a very promising method for extracting tritium from Pb-16Li blankets. Nevertheless the study of multi-nozzle flanges is required to scale this technology up to DEMO breeding blankets since the possible reabsorption of tritium extracted from one droplet by another one could be an issue.

Recently TLK developed a detailed model to simulate Pb-16Li flow rate and velocity as well as extraction efficiency. This model takes into account accurate calculations of all pressure drops along the VST, but neglects possible multi-nozzle disturbances. The presentation will focus on our strategy to quantify these, by comparing the experimental results obtained along multi-nozzle extractions of deuterium with the above mentioned single nozzle predictions. This will be fulfilled thanks to the multi-nozzle experiments to be analyzed with an empirical model including different contributions to the decrease of extraction efficiency due to reabsorption.

The multi-nozzle facility we developed comprises mainly two chambers whose geometries were optimized to favor initial dissolution of deuterium in Pb-16Li, to limit parasitic effect like permeation, and to favor the extraction efficiency. The most suitable multi-nozzle configuration consists in a hexagonal flange easy to scale up due to symmetries and equivalences between nozzles. Alternative geometries of the multi-nozzles flange are studied to understand and accurately quantify the disturbances and subsequent extraction efficiency decrease. This multi-nozzle VST experiment is a key step to scale this technique up for DEMO.

[1] F. Okino, K. Noborio, R. Kasada, S. Konishi, Fusion Sci. Technol., 64 (3), 543, 2013

Paper 14B-6: Thursday, April 21

First-Principle Study on Structures and Energetics of Intrinsic Vacancy Defects in Li_4SiO_4

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The tritium breeder is one of critical components in future deuterium-tritium fusion reactor because of the achievement of tritium self-sufficiency. Li4SiO4, possessing comparably high lithium density but maintaining excellent thermo-mechanical properties relative to earlier studied compound of Li2ZrO3 and LiAlO2, has been considered as the primary candidate materials in tritium breeding Test Blanket Module (TBM) designs in the world.

Diffusion of tritium in Li4SiO4 and its subsequent release from surface are affected by the defects induced by nuclear reaction of 6Li(n, β)T, energetic tritium and helium ions, and other processes, but related mechanism is not clear. In this work, ab initio calculations were performed with the Vienna ab initio Simulation Package (VASP) based on density functional theory (DFT). The project augmented wave (PAW) pseudopotential and PW91 exchange-correlation functions were employed to calculate electronic structure, atomic geometry and formation energy of intrinsic vacancy defects in Li4SiO4. Differences between the calculated lattice parameters and the experiment date are less than 1.2%. Minimum formation energies of neutral Li vacancy (VLi) and O vacancy (VO) is 0.7 and 5.7 eV under O-rich condition, 3.8 and 0.4 eV under Li-rich condition respectively. Calculated stable vacancy defects are VLi-, VO2+ and VSi4+ in as-grown Li4SiO4. The lowest energy migration paths of intrinsic vacancy defects also discussed, Energy barriers of most VLi- are less than 0.4 eV, suggesting that migration of VLi- is easy occurred. These results may provide primary date for experimental studies of intrinsic point defects and their influence on tritium diffusion in Li4SiO4.

Paper 15A-1: Thursday, April 21

Tritium Plant Technology Development for a DEMO Power Plant

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An architecture for the DEMO tritium plant, including system blocks and requirements, has been previously defined as part of work undertaken within the Eurofusion PPPT TFV work-package. A review of available technologies has been undertaken for each of the identified system blocks within the tritium plant. These technology surveys cover the majority of functions required in each system block of the tritium plant, including: isotope separation systems, storage, exhaust processing, PEG separation and exhaust detritiation. Mature, well tested technologies were considered along with ideas in the initial stages of development. This resulted in a list of possible technologies able to satisfy the functions of the tritium plant. Optioneering methods based upon careful criteria selection and MADA analysis have been used to select the most promising of the potential technologies for further development. This work resulted in a ranked shortlist of technologies for each system block. To the best of the authors' knowledge, this is the first time that such a comprehensive and thorough approach has been taken to determining the most suitable technologies for the DEMO tritium plant. In some cases decisions on technologies to be employed at DEMO have been made; in other cases a shortlist requiring further research has been drawn up. These shortlists then were compared with technologies to be employed in the ITER tritium plant; an overlap in the two lists suggests that experimental R&D may be unnecessary for some areas. The output of this work has been used to define an R&D roadmap through to 2018 for the conceptual development of a DEMO tritium plant. The WPTFV programme is now focusing upon:

- The selection and development of isotope separation technologies to meet the requirements for protium removal, isotope rebalancing and any other isotope separation systems required
- Further requirements analysis for tritium extraction and processing from breeder blankets, the coolant purification system and plasma enhancement gas species and amounts
- A study on the viability of the continued use of uranium as a storage media
- Tritium accountancy requirements and technology development
- Plasma enhancement gas processing
- Coolant purification system technology selection
- Plant integration

This work represents a crucial step in the development of the DEMO tritium plant, as well as showcasing a new and realistic roadmap towards conceptual design.

This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the EURATOM research and training programme 2014-2018 under grant agreement No. 633053 and from the RCUK Energy Programme [grant number EP/I501045]. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

Technology Decision Making for a Tritium Removal Facility

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A new Tritium Removal Facility (TRF) is being investigated for the Bruce Power Nuclear Power Plant (Ontario, Canada) in order to maintain reduced tritium concentrations in eight CANDU reactors. Five technology options for use in the new TRF were examined:

- Vapour Phase Catalytic Exchange with Cryogenic Distillation (VPCE+CD)
- Liquid Phase Catalytic Exchange with Cryogenic Distillation (LPCE+CD)
- Direct Electrolysis with Cryogenic Distillation (DE+CD)
- Combined Electrolysis and Catalytic Exchange with Cryogenic Distillation (CECE+CD)
- Liquid Phase Catalytic Exchange with Gas Diffusion and Thermal Diffusion (LPCE+GD+TD)

These five technologies were chosen for examination as they have operated as a nuclear station TRF or have clear technology providers. The five technologies were assessed against 15 criteria under the categories of cost, fit to client requirements, design and regulatory considerations, operability and maintainability, construction and commissioning considerations, and safety. A further nine criteria were considered but not used in the assessment, as no significant difference between the five technologies for these criteria was found.

The assessment was performed using HiView3, a decision modelling tool. The technologies were scored against each criterion. Where possible, the scoring was based off of one or more quantitative characteristics of the technological options. Where only qualitative data or relative attractiveness existed, the MACBETH function in HiView was used to develop a score. The criteria and criteria groups were then weighted based on relative importance.

The modelled result was the following order of attractiveness (most to least): LPCE+CD, VPCE+CD, LPCE+GD+TD, followed equally by CECE+CD and DE+CD. A sensitivity analysis on each of the criteria group weighting was performed which showed the LPCE+CD option to be a robust choice.

Paper 15A-3: Thursday, April 21

A Large Scale Demonstration Facility for Light Water Detritiation

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The future of the civilian nuclear power industry depends on its ability to meet increasing demands in public safety at a sustainable cost. Currently, the tritiated water generated at nuclear power reactor sites and spent fuel reprocessing facilities is mainly released to the environment owing to the reputed low biohazard of diluted tritium and the non-availability of economical light-water detritiation technologies. However, the amount of these releases could be reconsidered as a function of evolving water standards, public acceptance, and technological advancements.

This paper aims at describing the ongoing development of an engineering scale light water tritium separation facility by Kurion, Inc. [1]. The separation facility uses the combined electrolysis catalytic exchange (CECE) process, offering the best combination of efficiency, maturity, and cost among alternatives [2]. The process has already been demonstrated and characterized at a laboratory scale corresponding to a water treatment capacity of 0.1 m^3/day. The goal of the current project is to scale the process to a treatment capacity of 1.3 m^3/day and improve its efficiency further.

On the catalytic exchange part, the scaling approach involves designing solutions for the exchange column dimensions and geometry (Figure 1), internal packing, water distribution, heating, and humidification.

On the electrolysis part, the scaling approach implies the development of large proton-exchange membrane (PEM) cell based systems (Figure 1) up to 5 MW, and thus also relates to the emergence of the hydrogen economy.

Finally, the technical feasibility and proximity to industrial deployment is also being evaluated through the application of technical readiness assessment (TRA) criteria to the process.
In summary, the project hereby described involves scaling up, optimizing, and validating a cost efficient technology for light water detritiation at low concentrations and large volumes.

Figure 1: Exchange column (left); electrolyzer (right)

[1] http://www.kurion.com/en/technologies/tritium-removal

[2] N. Bonnet, G. Bonhomme, S. Abeta, Y. Iwasaki, D. Carlson, J. Raymont, Economic Comparison of Light-Water Detritiation Technologies for Large-Scale Applications, to be submitted.

ITER-like Tokamak Exhaust Gases in JET Active Gas Handling System: Process Optioneering

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The AGHS collects Tokamak exhaust gases from the JET machine, recovers the hydrogenic components and recycles the deuterium and tritium for fuelling the plasma. With the increasing fusion power and all-metal first wall and divertor, the forthcoming DTE2 experiment will see a change in the Tokamak exhaust composition compared to the DTE1 first series of experiments. A range of gases additional to those for fuelling the plasma will be added for critical applications such as plasma detachment and disruption mitigation.

One of these candidate gases is neon, which is anticipated to have a negative influence on storing the recovered hydrogen at an early stage of AGHS reprocessing. This paper will outline the trials and plant modifications which are in hand to provide a solution and enable downstream processes to operate as during DTE1. This will comprise a scrubbing circuit to mitigate the blanketing action of the neon in sorbing the hydrogen isotopes onto the existing depleted-uranium storage beds.

A second approach is also under study that has the potential to bring the tritium process circuit into closer alignment to that intended by ITER, thus providing process data for supporting the ITER fuel cycle design and increasing further the value of the JET experiments to the ITER project. JET's torus gases recovered onto AGHS's ITER Prototype Cryosorption Pump (PCP) will, on regeneration, be representative of the gas compositions received into ITER's Tokamak Exhaust Processing (TEP) system front end. The following adaptions of AGHS are being considered to make processing more representative of ITER:

- Helium, neon and hydrogen isotopes from 80K regenerations of the PCP could be routed directly to a Pd-Ag permeator that will replicate the first stage of TEP hydrogen processing.
- Other gases liberated from regenerating the PCP at 130K (so-called "warm" regenerations) could be cycled around a nickel bed and permeator train that will approximate to the operation of a palladium membrane reactor which is a second processing route within the TEP system.

This work has been carried out within the framework of the Contract for the Operation of the JET Facilities and has received funding from the European Union's Horizon 2020 research and innovation programme. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

Paper 15B-1: Thursday, April 21

Thermo-Mechanical Design Study of a Linear Mercury Diffusion Pump for Tritium Processing

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For torus vacuum pumping in DEMO, the KALPUREX process has been proposed by KIT [1] and is now under development. This process uses linear mercury diffusion pumps (LDPs) as primary pumps, backed up by liquid ring vacuum pumps.

A prototype LDP is currently in a detailed design phase. This pump – already fully tritium compatible and in relevant scale – shall be manufactured and tested in the THESEUS facility, a dedicated pump test facility at KIT. The results obtained by these experiments shall be used to benchmark a simulation code for this kind of pump developed in parallel [2].

This paper presents the detailed design of the prototype pump. The prototype LDP comprises three exchangeable pumping stages, an ejector stage leading to a higher foreline resistance and a two-stage inlet baffle system that traps the mercury vapor and that avoids that any mercury can penetrate back into the vacuum system.

The development of the prototype LDP has been done in an iterative process where three operating scenarios have been assumed and the resulting mechanical stresses have been simulated using finite element methods: A normal operation scenario, a bake-out scenario and an internal explosion scenario. In addition to the mechanical simulations, thermo-hydraulic calculations for the two baffle stages have been done to assess temperature distribution and pressure drops. Based on the baffle requirements and on other operational requirements (e.g. cool water- and mercury heating requirements), the specific infrastructure requirements have been derived.

[1] T. Giegerich, C. Day, The KALPUREX-Process, new vacuum pumping process for exhaust gases in fusion power plants, Fus. Eng. Des. 89 1476–1481 (2014).

[2] X. Luo, T. Giegerich, C. Day, Simulation of a Large Linear Jet Mercury Diffusion Pump with the Test Particle Monte Carlo Method, 62nd AVS Conference, Oct. 18 – 23, San Jose, CA, US.

Performance Testing of a Potential Replacement for the Normatex® Pump

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A Normatex® Model 15 scroll pump backed by metal bellows (MB) pump made by Senior Flexonics Inc. has been the preferred technology for use in tritium facilities world-wide. Normatex® pumps used world-wide for tritium service are no longer available. Identification of alternate pumps is required for performance and compatibility with tritium gas. Many of the pumps that could be used to meet the functional performance requirements (e.g. pressure and flow conditions) of the Normatex® pump have features that include the use of polymers or oils and greases that are not directly compatible with tritium service.

One pump that has been identified as a potential replacement is the Edwards nXDSI15iC. A series of pump tests were completed using this pump as a candidate replacement pump. The purpose of the pump testing is two-fold: (1) obtain baseline vacuum pump characteristics for the Normatex® replacement pump intended for use in tritium service; and (2) verify that low pressure hydrogen gas can be transported over various distances by the candidate pump. A series of pump tests were performed at various configurations using hydrogen gas (no tritium) to ensure that this pump can meet the performance. Various configurations of associated lengths of ¾ inch tubing were used in combination with hydrogen and nitrogen flow rates ranging from 25-400 standard cubic centimeters per minute (sccm) to determine whether the proposed pump could be used as a potential replacement for the Normatex® pump.

Though the Edwards pump meets the performance requirements, there are other tradeoffs that must be considered. As for cost considerations, the Edwards scroll pump is ~$8K whereas a Normatex®/Met-Bel system is ~$100K, which would significantly reduce initial start-up costs in a facility. However, the lifetime of the Edwards pump in a nearly pure tritium process stream is speculated to be only 3-6 months due to the polymer tip seals whereas the all-metal Normatex® pump can last for 5 years. This paper summarizes the results of the tests that have been performed using various pump configurations.

Paper 15B-3: Thursday, April 21

Positive Displacement Scroll Pump Technology for Tritium Processing Systems

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For general use in tritium service, positive displacement high vacuum pump options are very limited. For over two decades, a Normetex scroll pump was a mainstay for tritium processing systems; however, with the pump recently discontinued, there is renewed interest in a replacement pump.
This Pump development is very unique, as Tritium operation limits the use of materials commonly found in scroll vacuum pumps. Additionally, the pump must be hermetically sealed to avoid leakage to the environment or contact with other operating components. All components in contact with the working fluid have to be made of metal. No polymer seals, oils, or greases can be used due to degradation when exposed to ionizing radiation.

In 2014, Air Squared and the Savannah River Tritium Enterprise (SRTE) began development on a suitable replacement Tritium pump. In parallel to the development of the Air Squared pump, SRTE did exhaustive research on available pump options. Including testing modifications to off the shelf scroll pumps. SRTE has determined the Air Squared pump offers the best available option to the Normetex pump. In December of that year, Air Squared completed design, fabrication, and preliminary testing of a prototype scroll vacuum pump for Tritium service. SRTE has since continued performance testing of the pump.

Several novel design features are used in the pump. A flexible bellows allows all running hardware to be semi-hermetically separated from the working fluid. Tip seals - a polymer based component that seals between the fixed and orbiting scroll - have been removed from the pump entirely. Alternatively, sealing is achieved by precision machining the fixed and orbiting scrolls and using a lapping process to further reduce running clearances.

The following paper presents the unique design Air Squared has employed and the final test results conducted at SNRL. Without using any polymers or oils, the pump has accomplished all performance metrics required by SRTE for Tritium processing systems, while maintaining the reliable and efficient operation that is expected from a scroll device. With success of the pump, Air Squared has a pathway for a new standard positive displacement scroll pumps for Tritium service.

Tritium Aspects of Fueling and Exhaust Pumping in Magnetic Fusion Energy*

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Magnetically confined fusion plasmas generate energy from deuterium-tritium (DT) fusion reactions that produce energetic 3.5 MeV alpha particles and 14 MeV neutrons. Since the DT fusion reaction rate is a strong function of plasma density, an efficient fueling source is needed to maintain high plasma density in such systems. Energetic ions in fusion plasmas are able to escape the confining magnetic fields at a much higher rate than the fusion reactions occur, thus dictating the fueling rate needed. These lost ions become neutralized and need to be pumped away as gas to be reinjected into the plasma as fuel atoms.

The technology to fuel and pump fusion plasmas has to be inherently compatible with the tritium fuel. Injectors that produce solid cryogenic pellets of DT mixtures have been developed that can handle the tritium self-heating and are nearing a maturity level to implement on the ITER burning plasma experiment. A steady-state reactor will need further development of this technology to produce the reliable fueling needed.

The pumping capability for exhausting tritium containing exhaust gases from a fusion reactor is not as well developed as the fueling capability. Short duty cycle cryopumps are presently being implemented for ITER, but are not suitable for a steady-state reactor. Concepts of continuous cryopumps and other tritium compatible schemes exist, but need significant development.

An ideal solution would be to couple the pumping and fueling such that the pump exhaust is directly fed back into pellet formation with minimal filtering of impurity gases. This would greatly reduce the processing needs for the exhaust gases. Concepts to accomplish this will be discussed along with the fueling and pumping needs for a DT fusion reactor.

* This work supported by the Oak Ridge National Laboratory, managed by UT-Battelle, LLC for the US DOE under DE-AC05-00OR22725.

Paper 16A-1: Thursday, April 21

Tritium Control and Capture in Salt-Cooled Fission and Fusion Reactors

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The Workshop on Tritium Control and Capture in Salt-Cooled Fission and Fusion Reactors: Experiments, Models and Benchmarking was held on October 27-28, 2015 in Salt Lake City. The workshop objectives were to bring together researchers involved in experiments, modeling and benchmarking for tritium control at ~700°C in liquid salts and related systems to (1) exchange information and enable the future exchange of information, (2) initiate an effort for benchmarking of experiments and models, and (3) encourage cooperation between different groups working on the same challenges.

Three advanced power systems use liquid salt coolants that generate tritium and thus face common challenges. The Fluoride-salt-cooled High-temperature Reactor (FHR) uses the same graphite-matrix coated-particle fuel as high-temperature gas-cooled reactors and fluoride salt coolants. Molten salt reactors (MSRs) dissolve the fuel in a fluoride or chloride salt with release of fission product tritium to the salt. In both systems, the base-line salts contain 7Li. Isotopically separated lithium is used to minimize tritium production. The Chinese Academy of Science plans to start operation of a 10-MWt FHR and a 2-MWt MSR by 2020. High-magnetic field fusion machines proposed to use lithium enriched in 6Li to maximize tritium generation—the fuel for a fusion machine. Advances in superconductors that enable higher power densities may require the use of lithium salts as coolants.

The workshop results are summarized including: characteristics of salt-cooled fission and fusion machines; mechanisms and quantities of tritium generation; the environment for tritium capture; tritium generation, transport, migration, and capture in 700°C salt models; alternative strategies for tritium control; and ongoing experimental work.

Metal Tritides as Power Sources for Tritium Betavoltaics

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Tritium betavoltaics are one of the family of nuclear batteries which convert natural radioactive decay from a radioisotope into electricity that can provide continuous power without the requirement for replacement or recharging. These devices can be considered to be primary batteries which are in perpetual discharge and can have very long life times depending on the half-life of the radioisotope used. They are ideal for applications where battery replacement is difficult and are suitable for various low-powered military, aerospace and medical applications. Of the most common beta emitters, tritium is ideally suited to this application due to its high specific activity, low shielding requirements and relatively high availability as it is a by-product of CANDU heavy water nuclear reactors.

Owing to safety and environmental concerns over tritium leakage, metal tritides films have been widely studied as tritium betavoltaic sources. In particular, metal tritide films are used in direct betavoltaic cells, which require the beta radiation source to be in intimate contact with the semiconductor unit due to the poor penetration of beta radiation in solids. A schematic diagram of a direct betavoltaic cell based on a III-V semiconductor with a titanium tritide source is shown in Figure 1. Titanium tritide is a promising source material as it can be formed at low temperatures (<300 °C), has high chemical and thermal stability and titanium is inexpensive.

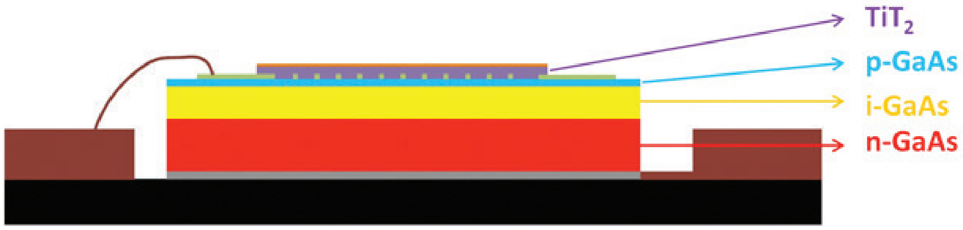

Fig. 1: Schematic diagram of a direct betavoltaic cell with a titanium tritide source.

Here we present a study of titanium hydride/tritide thin films. As with hydrogen storage materials, particle size greatly influences the kinetics of hydride formation. The mechanism of hydride formation will be discussed along with methods to suppress the formation of surface oxides. Betavoltaic devices based on III-V semiconductors have been constructed using titanium tritides films. Results from the fabricated devices show this new type of microbattery is a promising technology for low-power applications.

Paper 16A-3: Thursday, April 21

Status of the Karlsruhe Tritium Neutrino Mass Experiment KATRIN

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The discovery of non-vanishing neutrino masses was the first prove of physics beyond the Standard Model of particle physics. It was rewarded in 2015 with the Nobel Prize for physics for Takaaki Kajita (Super Kamiokande Experiment) and Arthur B. McDonald (SNO Experiment). However, these neutrino oscillation experiments are only measuring differences of neutrino mass squares, not their absolute values. Only a direct neutrino mass measurement with sub-eV sensitivity would allow to determine the neutrino mass scale in a model independent way and to clarify the role of neutrinos in astrophysics and cosmology as well as in particle physics.

The most sensitive way is to measure the shape of a tritium beta spectrum near its kinematic endpoint. Tritium is the nucleus of choice because of its low endpoint energy, super-allowed decay and simple atomic structure. Tritium beta decay experiments have been performed for more than 7 decades [1] yielding in an upper limit of the electron antineutrino mass of 2 eV. Within an international collaboration the KATRIN (KArlsruhe TRItium Neutrino) experiment is currently being built up at site of KIT thereby making use of the unique expertise of the Tritium Laboratory Karlsruhe (TLK). After 3 years of data taking KATRIN will determine the neutrino mass with a sensitivity of 0.2 eV (90 % CL).

KATRIN will use a windowless gaseous tritium source (WGTS) at 30 K, filled with molecular tritium gas of high isotopic purity (> 95 %). The tritium gas will be injected through a capillary at the middle of the source tube with a rate of 40 g per day. It diffuses over a length of 5 m to both ends and is pumped out by a series of differential turbo molecular pumps. The resulting density profile has nearly a triangular shape with a source column density of 5·1017 molecules/cm2. To minimize systematic uncertainties, the column density should have stability of the order of 0.1 %. This requires an appropriate stability of the source parameters as well as an accurate continuous monitoring of activity and composition of the KATRIN tritium source [2]. One of the main techniques employed will be activity monitoring with BIXS (Beta-Induced X-ray Spectrometry) [2, 3].

This contribution will present the status of the KATRIN experiment, thereby focusing on the recently delivered main components and on various techniques in tritium analytics employed at KATRIN.

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Tritium Production and Partitioning from the Irradiation of Lithium-Beryllium Fluoride Salt

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The MIT Nuclear Reactor Laboratory (NRL) has irradiated lithium-beryllium fluoride (flibe) salt as part of an on-going U.S. Department of Energy-funded Integrated Research Project (IRP) involving collaboration between MIT, the University of California, Berkeley, the University of Wisconsin-Madison, and the University of New Mexico. The objective of this joint project is the development of a path forward to a commercially-viable flibe-cooled reactor, named the Fluoride Salt High-Temperature Reactor (FHR).

As part of this project, the NRL has carried out two irradiations of FHR materials in static flibe at 700°C in the MIT Research Reactor. These represent the first irradiation of Li-7 enriched flibe originally produced for the Molten Salt Reactor Experiment at Oak Ridge National Laboratory since the 1960s. These irradiations marked the start of a program evaluating the tritium production and release from the liquid salt system at high temperature; in particular, there is interest in the evolution of tritium from the salt into solid materials and cover gasses. FHR development depends on an understanding of tritium behavior and a capability to control the transport of tritium within the primary system of the reactor. It will be crucial to understand tritium holdup and release from the substantial amount of graphite fuel and structural material within the core, as well as selection and optimization of a tritium capture system and barriers to mitigate tritium release from the primary system to the environment. After the completion of the irradiation, materials from the test were extracted from the salt and examined for tritium uptake and microstructural changes.

This paper describes the NRL irradiations and experience gained from the irradiation of flibe with respect to the detection and handling of tritium. It covers the development of techniques for monitoring the evolution of tritium from the salt during irradiation and the factors particular to the FHR system that influence this process, including the radiolytic production and release of volatile fluorine and fluoride products as a function of temperature. In addition, it discusses the measurement of tritium partitioning between the different materials in the experiment due to the confluence of diffusion, adsorption, and chemical and radiolytic reactions.

R&D Progress of Tritium Technology on the Broader Approach

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In the one of Broader Approach (BA) activities aiming to the development for a prototype fusion reactor, the R&D of tritium technology has been carried from 2007. The period consists of Phase 1 (2007-2010) and Phase 2 dividing into Phase 2-1 (2010-2011), Phase 2-2 (2012-2013) and Phase 2-3 (2014-2016). In the Phase 1, International Fusion Energy Research Center (IFERC) including DEMO R&D building was constructed in Rokkasho BA site of Japan. The R&D building is a facility to handle tritium, other radioisotopes and Beryllium. From the Phase 2-1, R&D of tritium technology has been carried out at the facility. Main R&D activities of tritium technology consist of the following;

Task-1) Maintain of R&D building though the tritium operation
Task-2) Development of tritium accountancy technology
Task-3) Development of basic tritium safety research
Task-4) Tritium durability test

From the start of phase 2 on the BA activity, the collaboration research programs between Japan Atomic Energy Agency (JAEA) and Japanese universities were started in filed of the tritium technology and many studies has been carried out by JAEA as well as Universities.

Presentation gives the progress of R&D activities on tritium technology together with overlook of significant results obtained on above-mentioned tasks in BA activity.

Tritium at the National Ignition Facility—Where Did It Go?

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The National Ignition Facility in Livermore, California is the world's largest and most energetic laser system. It is used for experimental studies involving high energy density, including those associated with inertial confinement fusion. When the NIF first introduced tritium into the Target Chamber, there was substantial uncertainty about how this material would migrate within facility systems, many of which are at high vacuum, and then into the work environment. After 5 years of operations involving tritium, the movement of tritium is now highly predictable and has been generally favorable in terms of hazard management.

During fusion experiments, about 20 curies of tritium are released into the 10-meter diameter NIF Target Chamber, which operates at about 10-4 Pa (10-6 torr) (vacuum between the earth and the moon). Most of the tritium is collected on cryogenically-cooled surfaces of the cryopumps. When the cryopumps are regenerated (warmed), the tritium is directed towards the Tritium Processing System (TPS), which collects 99% of NIF's tritium on molecular sieve material. A small amount of tritium is released to the environment via the stack of NIF's ventilation system. A very small amount remains on the surfaces of the Target Chamber and on target and diagnostic components that are inserted into the Target Chamber during experiments. These components are retracted into vessels, which are brought to atmospheric pressure so that the components can be reconfigured to meet the demands of the next experiment.

Components that are inserted near Target Chamber Center become contaminated from their exposure to tritium-containing targets (up to about 1E6 disintegrations per minute per 100 square centimeters). Components that are exposed to Target Chamber environments but are recessed back beyond the plane of the Target Chamber during experimental shots are generally not contaminated. The 192 laser optic lines attached to the Target Chamber are also not contaminated (with the exception of the final debris shields), which is contrary to NIF's original expectations. Extensive survey data that has been collected over the last five years of tritium use has been used to redefine NIF's hazard management protocols. This shift has resulted in reduced or eliminated contamination controls, a marked decrease in contamination control consumables and associated reduction in radioactive waste.

Paper 16B-2: Thursday, April 21

Key Management Tritium Issues at ESS Facility

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This paper is an overview of the different aspects of the tritium management in European Spallation Source (ESS) facility. The ESS facility is based on a linear driver (linac) directing the proton beam (5 MW of 2 GeV) of 2.8 ms long pulses with a 20 Hz on a tungsten target, where neutrons are produced via spallation reactions. Further the neutrons moderated to thermal and subthermal energies in a couple of moderators placed around the target will be guided through 22 beamlines to the scattering instruments. The project is now in the detailed design and construction phase.
As a by-product of the spallation reactions, tritium will be produced everywhere the primary protons and high-energy secondary particles will interact with ESS machine and its surroundings.

Due to the modifications of the design or increasing of the degree its details updates of the estimates of the tritium production were necessary to be performed using Monte Carlo calculations through over facility. The results of the new estimates are reported here. Based on the outcome of the calculations, derivation of the source terms to the environment as well as the waste characterization was further assessed.

The highest tritium inventory originates in the tungsten target and is assumed to be released completely into the helium-cooling loop and further to the atmosphere. This discharge leads to unacceptable consequences to the public. Three options for tritium removal from the helium loop were further investigated accounting for various types of criteria. It resulted from this analysis that the use of getter SAES (ST198) might be the optimum solution. The ST198 getter detritiation efficiency during transient phases was also demonstrated and found that this option has the least impact on the operation of the spallation source and meets emission guidelines with a reasonable safety margin.

Further environmental impact assessments were carried out both for chronic release as well as for off-normal events.

Paper 16B-3: Thursday, April 21

Progress of ISO Standards on the Confinement Systems in Fusion Facilities using Tritium

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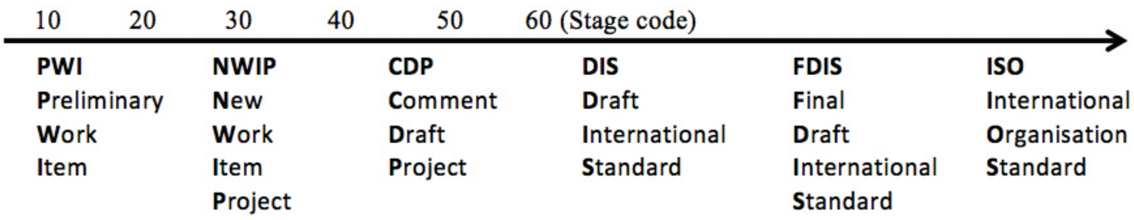
The International Standard Organization technical committee related to nuclear energy (TC 85) has a sub-committee on radiation protection (SC2) comprising in particular one working group dealing with confinement, ventilation and shielding issues (WG23). This Working Group develops standards which specify the applicable requirements related to the design and the operation of containment and ventilation systems of nuclear facilities. A preliminary work item is launched for fusion facilities using tritium (PWI 16646). Such international standard would apply especially to containment and ventilation systems of fusion facilities using tritium as a fuel and dedicated fuel handling facilities in order to protect the workers, the public and the environment from the dissemination of radioactive contamination originating from the operation of these installations.

In association with ISO/TC85/SC2/WG23, this paper presents the progress on this new standard more dedicated to tritium risks inside nuclear facilities. The successive working drafts follow design development of ITER project aiming at selecting the best technical solution. Overview of Standard development following ISO/IEC Directives, part 1 is summarized below:



ISO standard development stage starting from proposal submitted to TC85 with rational on the need of a new standard. NWIP prepare the working draft. After getting consensus at Working Group level, then at Sub Committee, the work draft become DIS. The DIS is distributed by ISO Central Secretariat to all ISO member bodies for voting and comment. FDIS is approved if 2/3 votes are positive and no more than ¼ are negative to be then published as ISO. The full process for having a validated standard takes generally between 3 and 5 years.

The Research Progress of Tritium in HTR-10

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The Very High Temperature Reactor (VHTR) is thought to be one of six proposed Generation IV concepts. The 10MW high temperature gas-cooled reactor (HTR-10) is a helium cooled, graphite-moderated, and thermal neutron spectrum reactor. For Tritium (H-3) has an effect on the environment and public radiation dose, it has been received more and more attentions on the environmental impact assessment of the nuclear power plant. The production mechanism and the transport property of H-3 in HTR-10 have been analyzed. H-3 is mainly generated from heavy nuclei fission reactions and activation reactions of impurities in the reactor core. The calculations show the majority of H-3 is trapped in intact particle fuels and a part can be entered into the primary circuit.

Recently, several experiments in HTR-10 have been carried on including decomposing and analyzing post-irradiation graphite spheres from the core, and sampling H-3 from the primary coolant. Preliminary measurements indicate H-3 is an important nuclide in the primary loop of HTR-10 and can exist both in the graphite and the helium. Further analysis is under way. Based on the experience of HTR-10, several channels have been set for sampling and analyzing H-3 in the primary loop of the high temperature gas-cooled reactor pebbled-bed module (HTR-PM). The chemical form of H-3 will be discussed which may greatly influence its transport and adsorption behavior. The research results can provide useful information about H-3 in HTR-10, and promote the study of H-3 in the high temperature gas-cooled reactors (HTGRs).

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Paper 16B-5: Thursday, April 21

Tritium Production in Secondary Sources

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Currently there are large uncertainties in the attribution of tritium in a Pressurized Water Reactor (PWR) Reactor Coolant System (RCS). The measured amount of tritium in the coolant cannot be separated out empirically into its individual sources. Therefore, all sources of tritium in the RCS of a PWR must be understood theoretically. One potential source of tritium in the RCS is due to tritium production in secondary sources. Neutron sources provide a flux of neutrons that are used to support reactor startup. Primary startup neutron source rods made of 252Cf are inserted into the reactor during the first cycle of a new nuclear reactor. The primary neutron sources are used to produce enough neutrons through spontaneous fission to create a sufficient neutron flux to be seen by the ex-core neutron detectors and facilitate reactor startup. Antimony-Beryllium secondary startup neutron sources are also inserted in the first reactor cycle to provide a neutron source for startups in future cycles. The secondary sources must be irradiated for a cycle to build up enough neutron emissions to support start-up of the next cycle. The Beryllium in the secondary sources is also a source of tritium when irradiated in a neutron flux. This paper will discuss tritium produced within the secondary sources, and released to the coolant.

Tritium Measurement of Waste in Large Volume Drums

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KEP Nuclear proposes two nondestructive methods to measure tritium activities in large volume waste drums: 3He in-growth method and calorimetry.

Firstly, the amount of tritium enclosed inside a waste drum can be determined by the measurement of the leak rate of 3He of this latter. After few months for establishing the equilibrium between the 3He production inside the drum and the 3He drum leak, the apparatus enables the measurement of an activity as low as 1 GBq of tritium in a 200 liters drum containing organic wastes. In practice sampling 3He outside the drum can be realized by the mean of a confining chamber that collect the 3He outflow. To reduce the time, a sampling intrusive system allows having directly a 3He sample in the drum.

Secondly, the calorimetry is the better approved method to quantify tritium in large volume drums. The new LVC1380 is a differential calorimeter with a new design to reduce the dimensions of the device and to regulate more easily its temperature. Contrary to the classical differential calorimeters, LVC1380 doesn't have a symmetrical reference cell but a ghost reference cell concentric at the measurement chamber. The calorimeter has also been co-patented by KEP Nuclear and CEA. This device will allow to measure tritiated waste generated by dismantling operations of a nuclear facility. This device is designed for the management of tritiated wastes in the range of 30 mg to 10 g of 3H (1.08 104 GBq to 3.6 105 GBq) within 100 L to 385 L containers. The accuracy of the measurement is 2% in the range of 1 W [3 g 3H] and the limit of detection currently is 9 mW [28 mg 3H].

Both techniques have their own advantages and limitations but combining both allows covering a very broad range of tritium activities and types of tritiated materials to be characterised.
This paper presents both nondestructive methods used to quantify tritium in large volume drums and their performances after R&D development.

Large Volume Calorimeter LVC 1380 for the Tritium Measurement in Radioactive Waste Packages

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The assessment of the amount of tritium in radioactive waste packages has always been a technical and environmental challenge. Indeed neither a destructive analysis on the waste nor a sampling of radioactive matter inside the package, too dependent of the physical state of tritium (solid, liquid or gaseous), cannot be considered. Moreover a direct nuclear counting of the package cannot be considered because of the low energy of beta particles of tritium stopped by few micrometers of metal (Emax = 18.6 keV, Emoy = 5.7 keV). KEP Nuclear however succeeded in developing a completely nondestructive analysis device to determine the exact mass of tritium in a radioactive waste package. This device is a large volume calorimeter called LVC 1380.

LVC 1380 calorimeter developed by KEP Nuclear

The measurement of 3H mass is based on the heating of package induced by interaction with matter of beta particles emitted. This heat is quantified by the calorimeter. The mass of tritium in the package is then assessed by knowing the specific power of tritium (324 mW/g). This technique eliminates the matrix effects (pressure, chemical composition and heterogeneity). This large volume calorimeter is capable of measuring drums from 100 liters to 385 liters with accuracy lower than 1 %. This new calorimeter can measure powers from 3 mW to 3500 mW corresponding to 1 mg up to 10 g of tritium. The calorimeter has also been co-patented by KEP Nuclear and CEA. This device will allow to measure tritiated waste generated by dismantling operations of a nuclear facility.

The paper presented will include a full description of the calorimeter and the principle of the calorimetric measurement. Different validation assays will also be discussed, as well as the possibility of application such as the integration of the LVC 1380 into a complete line of radioactive waste characterization developed by KEP Nuclear.

A Seebeck-Type Approach to Hydride Bed Inventory Tracking, PseudoSeebeck Calorimetry

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The performance of a prototype LaNi4.15Al0.85 hydride bed was studied. That study used several internal and external thermocouples that were logged via a data acquisition computer along with other process parameters. An additional 32 external surface-mount thermocouples were added and also separately logged. The thermal profile of the bed obtained during the process of acquiring data for a protium 50oC absorption/ desorption isotherm showed significant thermal variations of up to 25 oC between the middle and ends of the bed. A high quality statistical correlation was obtained between summed thermocouple signals (a Seebeck calorimetric procedure) and moles of hydrogen absorbed or desorbed as long as other process parameters such as heater performance remained similar. The use of this type of approach for dynamic bed inventory and, separately, tritium bed content will be discussed. The recommended installation was not a typical Seebeck calorimetric configuration where the thermocouples would be physically connected in series, but instead uses the virtual addition of the thermocouple signals to develop the calibrated signal. For that reason the method is named PseudoSeebeck Calorimetry (PSC).

Speciation of Tritium with Organic Molecules in Nuclear Plant Discharges

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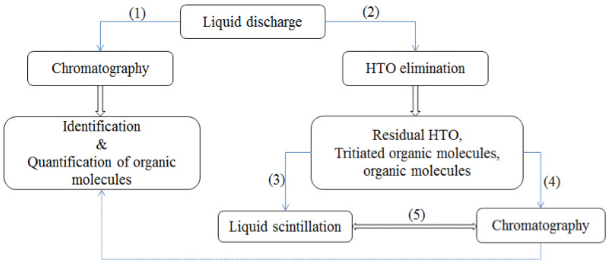
Tritium, produced in the primary coolant by neutron activation of bore and lithium, is the main contributor of the radioactivity released in the discharges of nuclear power plants. It is then present in liquid and gaseous discharges.

The limitation of the dosimetric impact associated to tritium present in the discharges is achieved by discharging tritium under a liquid form (97-98%). This dosimetric impact depends on the speciation of the tritium in the discharges which is mainly (over 99%) under HTO form (tritiated water) but the knowledge of the others forms is less accurate.

For EDF, the development of an analytical method allowing the determination of the speciation of tritium in liquid discharges is thus of main importance.

The aim of this work is then to develop a specific methodology for the quantification of the main organic molecules which are potentially radiolabeled with tritium.

The successive stages of this methodology are described in the following schematic:



The elimination of tritiated water(2) was used to determine (by liquid scintillation(3)) the tritiated organic molecules. Two techniques (freeze drying and distillation under reduced pressure) were tested to eliminate the tritiated water fraction, to concentrate the organic molecules of interest and to determine the concentration rate of these molecules(4). Different chromatography techniques (ion chromatography and reverse phase HPLC) have been used to quantify the different organic molecules present in the liquid discharges (amines: morpholine, hydrazine; carboxylic: acetate, formate; acid: EDTA)(1).

The fraction containing the molecules of interest are then collected and analyzed (off-line) by liquid scintillation in order to determine if the organic molecules are labelled or not with tritium(5). The results obtained by liquid scintillation have showed that the activity of tritium labelled with organic molecules is under the decision threshold.

Development Trials of an Acid Dissolution Line for Tritium Assay

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Tritium is a mobile substance which readily penetrates surrounding materials including metals; quantification of the total tritium content of metals is required at AWE. Trials are underway to design and develop an acid dissolution line to quantify the total tritium content in metal samples through complete acidic decomposition, using Quickfit® glassware and a furnace to oxidise hydrogen isotopes.

A mixture of oxidising and non-oxidising acid is used to dissolve stainless steel, ensuring breakdown of the protective passivation layer and complete decomposition of the metal resulting in liberation of any hydrogen isotopes present. Gaseous product from this reaction is passed over a heated platinised alumina catalyst bed, converting HT to HTO, which is collected in sequential bubbler traps. Non-active trials have proven the suitability of the system for use with 1 gram stainless steel samples using aqua regia (HCl:HNO3, 3:2).
The addition of a known volume of tritiated solution to the reaction flask during initial trials evaluated the transfer efficiency of tritium through the system as calculated by the final measured activity of the bubblers.

This poster describes the stages involved in assessing the recovery of tritium through the system, including issues concerning recovery of the known standards.

Overview of Tritium Monitoring Technology for Fusion Reactor Fueling Systems

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A fusion reactor pellet injection system provides tritium and deuterium fuel in the form of frozen pellets to the fusion plasma. The pellets are propelled into the plasma by a mixture of deuterium and tritium at high-pressure. The fuel pellets and propellant gas will be stored in gas lines contained in an enclosure with a nitrogen cover gas that is monitored for explosive levels of deuterium and tritium and for trace amounts of tritium that could leak into the enclosure. It is important to monitor not only for the presence of deuterium and tritium inside this enclosure to prevent an explosion, but also to quantify the concentrations in a post-accident evaluation. Because the quantity and throughput of deuterium and tritium could leak into the enclosure and result in explosive levels inside the enclosure, the instrumentation monitoring for explosive gas concentration should be classified as safety-related. In a fusion reactor, the pellet injection enclosure will be exposed to significant gamma radiation (nominally 50 KGy total integrated dose), neutron radiation (nominally 1012 n/cm2), and to a significant magnetic field (nominally 50- to 150 mT). Options for implementing safety-related monitors for explosive and trace levels of hydrogen and tritium in the environment of a fusion pellet forming enclosure are examined.

A significant limitation for commercial hydrogen and tritium detectors is that they are constructed with their electronics integrated with their sensors, which is problematic for fusion pellet injection systems in high radiation fields. Commercial instruments will fail at radiation levels much less than those listed above, but the thickness of gamma and neutron shielding required to ensure a detector's survival is not practical for the limited space available for pellet injection enclosures near the fusion plasma. Magnetic fields also affect the operation of some gas concentration sensors enough that magnetic shielding would be required.

Methods considered to monitor for explosive levels include electro-chemical sensors; infrared and other optical sensors; combustion; and gas density. Methods evaluated for monitoring for trace amounts of tritium using beta decay of tritium include liquid scintillation counting; ionization chamber; and crystal diamond detectors. The applicability of optical spectroscopy and mass spectroscopy of measuring gas concentrations are also reviewed. Each of these methods is evaluated for the fusion pellet injection system. Recommendations and options, considering current commercially available technologies, are included in the paper.

Obtaining of Thin Layer Radioactive Surface Sources Through Radio-Induced Grafting of Labeled Monomers

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Tritium (T) and 14C surface contamination can be determined using proportional counter with gas flow detectors. For the detectors calibration, the surface reference sources for each analyzed radionuclide are required.

In this paper was analyzed the obtaining of the reference surface sources with T or 14C by radio-induced grafting of labeled acrylic acid on polyethylene terephthalate (PET) thin foils.

The acrylic fragments radio-insertion on PET structure has been performed through Fourier Transform Infrared spectrometry with Attenuated Total Reflection (FTIR ATR). For grafted PET samples characteristic peaks of acrylic acid (especially a large band at 3000-3800 cm-1 assign to OH carboxylic groups) has been identified.

The grafting yields have been determined using tritiated acrylic acid with 37 kBq/mg specific activity. The mass of grafted monomers on PET foils was determined by: sample oxidation at HTO using a dedicated oxidizer, retention of HTO in ethylene glycol and determination of samples radioactivity using a Liquid Scintillation Counter (LSC) TRICARB TR2800 type.

The experimentally determined grafting yields have been analyzed in relationship with absorbed dose and dose rate. The obtained results suggest a linear dependence between radiochemical yields and dose.

The T and 14-C surface sources have been accomplished using following protocol:

- obtaining of monomers labeled with T and 14C respectively
- depositing of PET thin foils onto stainless steel support plates
- free radical inducing in PET foils using a gamma 60Co source (32 kGy absorbed dose and 9.32 kGy/h dose rate)
- exposing of irradiated PET foils/stainless steel at labeled monomer

The obtained surface sources were analyzed in point of view of total deposited activity and uniformity of the surface activity. The uniformity of the labelled fragments surface distribution was analyzed using Beta TLC Scanner RITA STAR type.

17-08 Poster Session 3: Thursday, April 21

Tritium Counting by Europium Coordination Complex

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The liquid scintillation counting (LSC) has been used for the tritium measurement. The LSC requires an organic liquid scintillator to detect beta particles. The organic liquid scintillator contains some harmful substances and it produces flammable wastes. An inorganic liquid scintillator would be less harmfulness and nonflammable one, although it has not been developed.

An europium coordinate complex is a luminous matter in aqueous solution. This luminous character would be utilized for the tritium counting. In this study, the luminescence of europium coordinate complex induced by beta particles of tritium was observed. The intensity of the luminescence was proportional to the tritium concentration. The potential of the europium coordinate complex can be shown as an inorganic liquid scintillator.

Since dipicolinic acid (2, 6-pydidinedicarboxylic acid or DPA) is a relatively nontoxic and water-soluble, it is a suitable ligand for our purpose. The aqueous solution of DPA adjusted pH 11 was prepared. The Eu3+ solution was mixed with the DPA solution to yield [Eu(DPA)3]3+. The formation of the complex was confirmed by the absorption, emission and excitation spectra. The emission spectrum of the aqueous solution of [Eu(DPA)3]3+ has the sharp peak at 615 nm. The sharp peak was assigned to 5D0 -> 7F2 transition. The emission was resulted from the ligand to metal energy transfer. Consequently, the ligand takes the energy of radiation, the excited ligand transfers the energy to the europium ion and the excited europium ion generates the luminescence.

The tritiated water was added to the [Eu(DPA)3]3+ solution. The luminescence induced by beta particles was detected with using the photomultiplier tube. The intensity of luminescence linearly increased with increasing the amount of tritium. However, the intensity of the luminescence of the [Eu(DPA)3]3+ solution is much weak in comparison with that of the organic liquid scintillator, Ultima Gold. On the other hand, the molecular design of the ligand can improve the efficiency of luminescence of europium coordination complex. The design of ligand is important issues among the efficiency of luminescence, the quantitative tritium counting and the harmful to the living. In the study, the potential of europium coordination complex for tritium counting was confirmed.

This study is supported by KAKENHI 26610065 and KAKENHI 15

TEACUP: A Tritium Management and Supplemental Core Follow Program

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The Tritium Technology Program (TTP) at Pacific Northwest National Laboratory (PNNL) is responsible for designing Tritium-Producing Burnable Absorber Rods (TPBARs). TPBARs are the in-reactor core components that are designed to produce and contain tritium in a commercial nuclear power plant. TPBARs have a passive reactivity control function, designating them as basic safety components which require integrity monitoring. Tritium Effluent Analysis and Core-follow, Up-to-date and Predictive (TEACUP) was developed to assist in the management of the plant's compliance with the regulatory limits for worker and offsite dose, as well as to monitor the integrity of the TPBARs during irradiation.

TEACUP is a tritium management and supplemental core follow program that allows its users to account for reactor coolant system (RCS) tritium sources, generate discharge release estimates, account for downstream river flows and concentrations, and calculate corresponding uncertainties. The program incorporates water balance methodologies, tritium production estimates from secondary startup neutron sources, soluble boron content, reactor coolant system tritium measurements, and seasonal river flow estimates.

TEACUP was designed specifically to facilitate the tracking of TPBAR permeation since measuring in-reactor permeation directly is not feasible. A number of models, calculations, and correlations were developed in order to quantify all of the leading sources and loses of tritium in the RCS. By comparing all of the known contributors and discharges from the RCS tritium inventory to the measured RCS tritium concentration, the unaccounted for balance (within some band of uncertainty) can be attributed to TPBAR permeation. The tritium release estimates to the river generated from TEACUP are validated by comparing them to the measured tritium releases which match well and give confidence that TEACUP is tracking and accounting for tritium appropriately. An additional check on the methodologies within TEACUP is that the cycle-to-cycle trends for tritium permeation per TPBAR are consistent in behavior and the estimated release per TPBAR across each cycle is the same within their uncertainty.

A New Device for Tritium Activity Measurement of Tritiated Water

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Every time a component in contact with tritium also needs a water cooling system, like it is the case with many components in future fusion power plants as well as in current laboratory setups, permeation of tritium into the coolant has to be checked regularly. This is the case e.g. in water cooled breeder blankets of fusion reactors or pumps of their primary vacuum system. On water detritiation systems for example, an efficient, reliable and affordable in-line and near to real time process monitoring is desirable to get the best performance out of your water detritiation system.

A common technique to determine the activity concentration of water is liquid scintillation counting (LSC). On one hand, LSC is a well-established method with good performance with regard to the detection of low levels of contamination. On the other hand, sample taking is necessary which only allows measuring a few samples per day. Each sample needs the addition of a toxic liquid scintillator cocktail. Therefore, it is not possible to treat the sample taken afterwards in usual water detritiation systems but handle it as radioactive waste which increases the overall operational costs. Furthermore, the LSC method cannot be used in-line for monitoring of process parameters.

For this reason, the Tritium Laboratory Karlsruhe (TLK) develops new methods for tritium accountancy. Among others, we develop a small solid-scintillator based accountancy system that can be used stand-alone or in-line as a process-monitoring tool. The work will describe the system briefly and gives an overview of the status of the TRAMPEL (Tritium Activity Measurement with a PhotomultipliEr in Liquids) experiment. The work focuses on the updates to the TRAMPEL experiment with regard to an enhanced readout and signal processing leading to a lowered detection limit with regard to previous work presented. The paper will also address future measures to even further enhance the setup as a whole to fulfill the challenging task for an in-line and near to real time process monitoring tool able to operate under demanding conditions.

First Calibration of an IR Absorption Spectroscopy System for the Measurement of H2, D2 and HD Concentration in the Liquid Phase

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An integral part of the fuel cycle of future fusion facilities is the isotope separation system (ISS). The Tritium Laboratory Karlsruhe (TLK) is currently operating a combined ISS/WDS (Isotope Separation/Water Detritiation System) as a contribution to this development [1]. A wide range of concentrations of the hydrogen isotopologues Q2 (H2, HD, D2, HT, DT, T2) will occur in the cryogenic distillation columns of the ISS, whereby liquid tritium T2 is accumulated at the bottom. In order to monitor and control the separation process, an inline and real time analysis method is necessary to measure the Q2 concentration in the liquid phase.

Liquid inactive hydrogen isotopologues were already successfully analyzed via infrared (IR) absorption spectroscopy and calibration data for D2 is provided by previous experiments at TLK [2], [3]. But these measurements were restricted to mixtures equilibrated at room temperature. By this constraint the HD concentration is fixed for a given atomic ratio of H and D. This equilibrium condition is not valid in cryogenic distillation. Therefore it is mandatory to expand the calibration to non-equilibrated mixtures. Since IR spectra of liquid hydrogen are dominated by nonlinear processes, especially from molecular dimers [5], calibration is not trivial and spectra from equilibrium mixtures differ to non-equilibrium in an unexpected manner. So each absorption line can have a different concentration dependent behavior and therefore line selection for calibration has to be done carefully.

This contribution presents the worldwide first calibration of IR absorption spectroscopy for mixtures of liquid H2, D2 and HD. With accuracy better than 5 % a very good first calibration is available and the technical and physical feasibility is demonstrated. This will be extended to tritiated samples with the T2ApIR experiment, which is under development at TLK [6].

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Tritium Imaging Techniques for the Study of Tritium Transport in the Graphite Fuel Elements of Fluoride-Salt Cooled High-Temperature Reactors (FHRs)

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Fluoride-Salt Cooled High Temperature reactor (FHR) is an advanced reactor design that combines low-pressure fluoride salt coolant LiF-BeF2 (flibe) with solid fuel elements formed of ceramic encapsulated fuel (TRISO particles) dispersed in a graphite matrix material. The flibe coolant operates in the 600 to 700oC temperature range, and the fuel elements operate below 800oC. One important challenge for FHR is tritium production in the coolant due to neutron reaction with 6Li, 7Li, and 9Be. For the Pebble Bed FHR Mark I design, the equilibrium tritium production rate for a 236 MWth reactor is 2.4E5 Ci/yr, which is 3500 times more tritium, per MWe, than a typical 1000MWe AP1000 – 700 Ci/yr. The Molten Salt Research Experiment, which was in operation in 1960s, indicated that graphite blocks in reactor core had large affinity for tritium. Considering the large surface area and the large volume of graphite in the FHR core, it is expected that graphite has the potential to be an efficient tritium sink for FHR; a tritium loading of 0.3 wppm on the graphite that is present in the fuel elements would be sufficient to absorb all tritium produced in the FHR coolant.

At University of Wisconsin (UW), there are ongoing experimental efforts to study hydrogen and tritium transport characteristics into A-3 matrix graphite, which is the main constituent of FHR fuel elements. In one of the planned experiments at UW, real-time tritium absorption in graphite will be measured by a differential pressure gauge. Complementary to this measurement, we are also interested in mapping the concentration distribution of tritium in the graphite samples. Tritium, is a low energy βemitter (average βenergy is 5.6 keV) with half-life of 12.32yr. Because the emitted β particle has very low energy, it is a challenge to map low concentrations of of the tritium absorbed by graphite. Established detection techniques that can be used for low concentrations of tritium include Liquid Scintillation counter (LSC) used for tritium measurement in water, and Ionization chamber for measuring tritium in a mixture gas.

In this paper, several tritium imaging techniques will be discussed as candidates the low-concentration, high-temperature absorption experiments at UW.

Performance of Inorganic Oxides for the Non-Reversible Trapping of Hydrogen: Application to Tritium Analysis

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We studied a classical hydrogen getter based on manganese and silver oxides. Our work consisted of parametric study influencing the trapping kinetics; highlighting the non-reversibility of the trapping; and elucidating the trapping mechanisms. The application field was the transportation and storage of radioactive waste containing radiolysis hydrogen.

Another application field has also been considered: the trapping of tritium under its forms T_2, HT, or DT. Previous studies have evidenced the reactivity of tritrium with MnO_2 getter. This property to trap tritium can be used to immobilize tritium in order to mitigate the radiological risk or to measure its activity. We propose a new way for tritium analysis by a trapping of tritium with MnO_2 getter followed by a proportional or scintillation counting.

In a first time we synthetized and studied various forms of manganese oxides in order to optimize the trapping kinetics. Parameters such as specific surface area, nature of the promotor (silver oxides, silver) and promotor weight percentage in the getter were also studied.

In a second time we studied the chemisorption of dihydrogen on the getter. The chemisorption has been highlighted thanks to magnetometry and infrared spectroscopic measurements. The non-reversibility of the trapping has been verified under anaerobic conditions up to 150°C during several weeks. The good tolerance and applicability of the getter under irradiation has also been confirmed.

We finally managed to explain the trapping mechanisms thanks to XRD and allowed us to highlight the trapping mechanisms at an atomic scale.

New experiments on trapping of tritium are currently undergoing to prove the feasibility of using the getter in tritium activity measurement.

All this results will be displayed and explained for the TRITIUM 2016 conference.

Design Considerations for a Tritium Process Analytical Measurement System

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Tritium, a radioactive isotope of hydrogen, is used in modern nuclear defense, in fusion energy programs, and in proposed medical isotope production processes. For economic, security, safety, and environmental reasons, tritium is generally recycled and purified. The main components of a tritium purification system are a supply and storage system, an impurity removal system, and an isotope separation system. Incorporation of an analytical measurement system is also required for efficient operation and control of the tritium purification system. The design of an analytical system can be challenging due to the presence of radioactive tritium. For example, modification of instruments may be necessary due to space limitations within containment (glovebox), materials used in the design must be compatible with tritium to limit failure and introduction of degradation products into the system, and exposure to personnel and the environment must be controlled during transport of a sample to an instrument. These and other considerations in the design of a tritium process analytical measurement system will be discussed in this presentation.

Study on Hydrogen Isotopes Behavior in LiPb Forced Convection Flow

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It is necessary to estimate tritium behavior in a liquid blanket of a D-T fusion reactor that will be constructed in the future. LiPb is a promising material as the liquid blanket candidate. However, there are few research results to clarify the behavior of tritium permeation under fluidized LiPb condition. To better understand the tritium permeation mechanism, here we use a dual cylindrical tube system. Hydrogen isotope dissolved beforehand in a LiPb forced convection flow is transferred through a permeable tube to an Ar purge gas flow. We found that hydrogen permeation flux can be correlated in terms of diffusivity, solubility and concentration boundary layer. These finding suggests that behavior of hydrogen under fluidized LiPb flow can be quantitatively correlated using the three dimensionless numbers(Sherwood, Reynolds and Schmidt) The results obtained here will give important information to estimate the tritium transfer rate in the future fusion reactor blanket.

The Effect of MHD Mixed Convection on Tritium Transport in Fusion-Relevant Configurations

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In the Dual-Coolant Lead-Lithium (DCLL) blanket, the liquid Pb-Li circulates slowly for tritium breeding and heat removal. Thermal gradients in the fluid, due to non-uniform volumetric heating as well as cumulative bulk heating along the flow direction, cause free convective flow which cannot completely be damped by MHD effects. The resulting mixed-convection MHD flow may exhibit re-circulation and stagnation regions affecting tritium transfer efficiency and the permeation rate. The goal of the present study is to investigate the effect of MHD mixed convection on tritium transport in the DCLL blanket-relevant configurations.

MHD flow, temperature fields and tritium concentration were computed for buoyancy-assisted and buoyancy-opposed flows using our recently developed computational framework, including transport inside the PbLi MHD flow, transfer across the material interface, and permeation through the structural materials and into the helium coolant. Significant differences exist in tritium distribution and permeation between buoyancy assisted and buoyancy opposed flows. Present computations predict a strong shear layer at the center of the channel and buoyancy-induced reverse flow near the front wall for the case of downward flow. Consequently, high tritium concentration is observed in the re-circulation flow region. Meanwhile, for upward flow, we can see a more uniform tritium distribution, and there are no high local tritium concentrations except a peak in the front-top corner due to low velocity in that area. As a result, tritium inventory for the case of upward flow dropped by nearly 80% compared to the case of downward flow. The calculated tritium permeation loss for the case of upward flow is also about 20% lower than the permeation loss for the case of downward flow.

This work represents an attempt at computation of the highly complex subject of tritium transport in a MHD mixed convection flow. The results provide some preliminary evaluation of tritium transport performance affected by MHD mixed convection, which justifies the interest in developing a better understanding of the phenomenon.

Tritium Contamination Prevention using Sacrificial Materials

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Tritium is produced by irradiating Tritium Producing Burnable Absorber Rods (TPBARs) in a Commercial Light Water Reactor at the Tennessee Valley Authority Watts Bar Reactor 1. The TPBARs are manufactured with strict materials specification for contaminant levels within all of the components. Despite meeting these requirements, gamma emitting contamination in the form of 65Zn was detected in a glovebox that was designed to contain only tritium. A forensic examination of the piping revealed that the radioactive zinc was borne from naturally occurring zinc within the materials. Furthermore, this 65Zn deposits at an anomalous distance from the extraction furnace based on vapor pressure considerations alone. A method to capture 65Zn was developed that is intended to prevent its further spread. This method relies on an operating filter media at a specific temperature and location. While this approach is acceptable for limited operations, as the facility undergoes increased utilization, there is a possibility of scheduling conflicts for maintenance and increasing dose to workers. In order to preclude these issues, methods to contain the 65Zn within the furnace module, an area designed for high radiation dose, have been developed. These approaches use bulk materials and nano-materials deposited on various substrates that are compatible with tritium and the extraction process. These materials were tested to ascertain their zinc capturing capability, capacity, and characteristics. The first generation material was optimized and a process lid has been fabricated for testing.

This presentation will describe the 65Zn deposits found in the process piping, methods to prevent further contamination of the piping, and current research directed at containing the contaminants in the high radiation area.

This document was prepared by Savannah River Nuclear Solutions LLC (SRNS), under Contract No. DE-AC09-08SR22470 with the United States Department of Energy (DOE).

SRNL-STI-2015-00596

Study on the Gas Extraction in Molten Salt Reactor for Tritium Control

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Tritium (hydrogen-3) is produced continuously in the fluoride salt reactor, and became a challenge in radioactive protection of the molten salt reactor. Therefore, the tritium removal system is an important issuer in the molten salt reactor. And a similar blanket tritium recovery system is also developed in ITER. In our case, the gas removal system is composed of two key devices, a bubble generator and a bubble separator. The bubble generator is to generate tiny bubbles. Bubbles in the molten salt play a signification role for mass transfer of tritium, while the bubble separator is for bubble separation. In the current study, a venturi-type bubble generator and an axial-flow centrifugal type bubble separator are firstly developed for application. And then a water test loop is designed for investigating the performance of these devices.

Fig. 1 the components of gas extraction system: 1. bubble generator, 2. bubble separator

The systematic experimental studies on the air core formation with a gas–liquid separator and a venturi-type bubble generator were carried out in an air–water system. When the air is charged into the water flow through the small feed-holes, it breaks into small bubbles in the diverging section of the venture-tube. Based on the results of Kress's formulation on turbulence dissipation rate, the bubble size correlation with the Reynolds number and the surface tension coefficient was derived. The effects of Re, gas volume ratio on the initial bubble size are analyzed. The experimental data proved that the volume averaged bubble diameter has a -1 power dependence on the Re, which agrees well with the theoretical derivation. A new formulation which includes the fact that the critical Weber number is dependent on the gas volume ratio was proposed. The air core behavior was recorded with different swirl numbers and Reynolds numbers in a gas–liquid separator. The results indicate that the flow pattern depends on the back pressure and the evolution of air core can be classified as four stages, which are air core with suction, tadpole-shaped core, cloudy core, and rod core, respectively. The back pressure under which the rod core is formed is defined as the critical back pressure, which depends on Re and S. The experimental data indicates that the critical back pressure increases as the Re increases but decreases with the increasing of swirl number. The critical back pressure defines the lower pressure limitation and another critical parameter, i.e., the liquid entrainment ratio, is introduced to define the upper limit of back pressure. To obtain a low cost and efficient separation, the back pressure should be maintained close to the critical value.

Chemical Kinetics Study of Reactions Between HTO and H2 in Purge Gas Flow in Tritium Breeding Blanket

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Tritium recovery process is one of the most important key techniques to realize sustainable tritium breeding to a fusion reactor. In a tritium breeding blanket in a fusion reactor, breeder pebbles is expected to release tritium mostly in the form of HTO into a purge gas flowing around the pebbles. The purge gas usually consists of mostly helium and a small amount of hydrogen. Tritium is expected to recover in the form of HT as a result of chemical reactions between HTO molecules and hydrogen molecules in the purge gas flow. Therefore, for a design of an efficient tritium recovery system, it is important to understand chemical kinetic behaviors in the purge gas flow.

In this paper, reaction pathways of HTO+H2 -> HT+H2O are firstly studied by using a
commercial electronic structure program to find possible elemental reactions and intermediates. Then, the quasi-classical trajectory (QCT) methodology is adopted to investigate chemical kinetics of elemental reactions. In QCT, the trajectory of molecules in the potential energy surface is calculated by classical dynamics, on the other hand the initial and the final internal energy states of the molecules are characterized by the discrete energy levels. For the purpose of simplification, potential energy surface is approximated by the sum of Morse potential energy functions for possible atomic pairs.

The initial energy states of the molecules are randomly given according to the population distribution in the equilibrium states for the given temperature. Then,
the equations of motions are integrated with time to obtain the trajectories of each atom. The final states are determined by the distance between each atom and the angular momentum of the molecules. By counting the number of reactive collisions in the trials, reaction probability is calculated. Finally, the chemical reaction rate coefficients are deduced by a Monte Carlo integration manner.

Removal of Low-Concentration Deuterium from Fluidized Li Loop for IFMIF

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For the effective operation of the International Fusion Material Irradiation Facility (IFMIF), it is important to recover hydrogen isotopes as impurities, which are included in liquid lithium (Li). Impurities limit goals in IFMIF are determined as deuterium (D) < 10 wppm, and tritium (T) < 1 wppm for safety and Li flow stability. The most effective recovery method of hydrogen isotopes is to use yttrium (Y) hot trap because Y has good compatibility with Li and higher solubility of hydrogen isotopes than Li. Therefore, it is necessary to understand the transfer of hydrogen isotopes from Li to Y. In this study, in order to demonstrate the effective operation of a Y hot trap and to obtain actual proof based on experimental data, we evaluated the recovery performance of Y hot trap. Schematic diagram of the experimental apparatus for evaluation the Y hot trap is shown in Fig. 1. Previously, we clarified that Y can work as a hydrogen getter which can absorb D < 10 wppm or T < 1 wppm at 573 K of IFMIF operation temperature at static Li conditions. In addition, we also demonstrated that it is possible to recover D to less than 10 wppm and achieve recovery ratio 99.0 % at fluidized conditions, where only D is included in Li. Recovery ratio of D in Li by Y hot trap at 573 K is shown in Fig. 2. However, other impurities such as N or O may be included in actual IFMIF operation conditions. There is possibility in that N in Li may pollute the surfaces of Y. For this reason, to clarify the influence of N in Li against the recovery performance of a Y hot trap, Experiments are conducted under the Li fluidized conditions at 523 or 573 K, where N about 100 wppm is absorbed to Li.

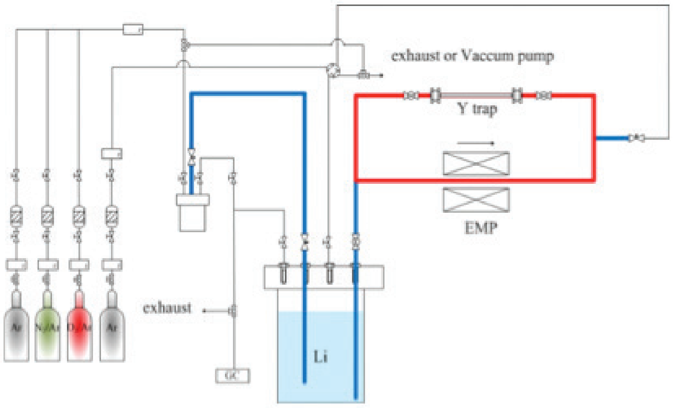

Fig. 1 The experimental apparatus

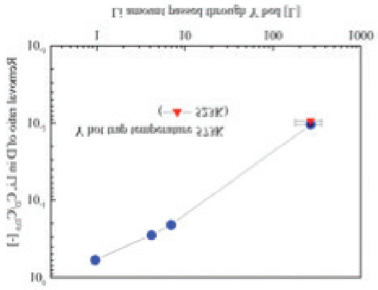

Fig. 2 Recovery ratio of D in Li

The System Design of Tritium Extraction From Ceramic Breeder Material With H_2O Added in Helium Purge Gas

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Hydrogen is always added into purge gas to improve the extraction of tritium from ceramic breeder materials via isotope exchange reactions. However, water vapor is more effective to increase tritium release rate at comparatively lower temperature. Tritium extraction with H2O/He purge gas has decisive advantages: higher tritium extraction efficiency at low temperature, less tritium inventory in breeder blanket, lower tritium permeability to environment. It is significant to establish a cost-effective method to recovery tritium from H2O/He purge gas, where tritium would be in the form of tritiated water.

In the present study, a tritium recovery system from tritiated water in purge gas was designed, three sets of fixed-bed reactors are connected in sequence in this system. The first is a renewable bed packing with modified Fe nano-scale powder, which could decompose more than 90 vol.% water vapor and transfer tritium into hydrogen isotope molecules at 400β The second bed is a non-renewable bed packing with ZrNi5 micron-sized powder, which could convert residual 10 vol.% water vapor to hydrogen isotope molecules below 500β Water vapor concentration at the outlet of the second bed was keeping below 1ppm. The third bed is a hydrogen getter bed packing with ZrCo micron-sized powder to recovery hydrogen isotope from helium with high efficiency. Afterward, a certain amount of water vapor was added in dehydrogenated gas and sent back to breeder box for continuous tritium extraction.

Fe powder in the first bed was synthesized by solvent combustion method with addiction of 10 at.% Zr and 0.5 at.% Rh, Zr was used to inhibit the sintering of powder for avoiding bed performance degradation and Rh was used to enhance the catalytic activity at low temperature. The first bed could be regenerated by hydrogen at 500βwith no production of solid waste. The second bed could be in small size for most of the water vapor was decomposed in the first bed, thus it could service for a long duration with tiny amount of waste production. In order to predict the performance of the system, one-dimensional gas-solid reaction models for the three sets fixed-bed reactors were established based on kinetics parameters from experiments.

Tritium Extraction from Lithium-Lead EU DEMO Blanket using Permeator Against Vacuum

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The history of DEMO breeding blanket designs in the EU is now over 20 years long. Several of the options currently under study, namely the Dual-Coolant Lithium-Lead (DCLL), the Helium-Cooled Lithium-Lead (HCLL) and the Water-Cooled Lithium-Lead (WCLL), use liquid Lithium-Lead (PbLi) as tritium breeder (and multiplier).

As the tritium is steadily produced in the blanket during the reactor operation, and as it is needed to close the fuel cycle in situ, while at the same time it cannot be left to accumulate an inacceptable inventory in the machine, suitably efficient strategies for the tritium extraction system (TES) from the breeder must be developed.

Among the TES options currently under consideration, the permeator against vacuum (PAV) appears to be one of the most promising solutions for the DCLL, HCLL and WCLL BB.

The PAV operating principle is simple: PbLi with concentration of tritium CT,in flows in a channel delimited for a given length L by a membrane permeable to tritium. Vacuum is maintained on the other side of the membrane, so that the difference between the tritium partial pressure on the two sides of the membrane drives the flux of tritium from the PbLi side to the extraction, leaving a lower tritium concentration CT,out at the outlet of the length L. The efficiency of the PAV can then be defined as ββ1 – CT,out/ CT,in.

As opposed to the undesired permeation of tritium through the walls in the rest of the circuit, which may lead to safety issues and reduced Tritium Breeding Ratio, here the relatively low solubility and the resulting high partial pressure of tritium in PbLi, together with a suitably permeable and corrosion resistant material for the membrane, are at the basis of the PAV potential for operation.

In the paper, the adopted semi-analytical model for the prediction of the PAV efficiency is applied to the HCLL and WCLL. The model is then used to assess by a sensitivity study how the main TES design parameters (e.g., material, diameter and length of the permeable membrane; PbLi circulation speed; Sievert's constant and mass transfer coefficient of tritium in PbLi), when varied in a reasonable range and depending on the system considered (HCLL vs. WCLL), affect the PAV system size for a given target efficiency β

Acknowledgements

This work was partially financially supported by EUROfusion.

Investigation of Correlation Between Irradiation Defects and Deuterium Release

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In the future D - T fusion reactor, the self-supply of tritium is one of the critical issues to maintain normal operation of the reactor. Due to the effects of irradiation defects, tritium migration in tritium breeder materials is complex. The correlation between irradiation defects and deuterium release in Li2TiO3 irradiated by 120 keV D+ have been investigated. Thermal desorption spectroscopy (TDS) experiments were performed to elucidate the release behavior of deuterium from trapping sites. Li2TiO3 samples become blue and grey after annealing in vacuum and they become white again after annealing in air. This may be attributed to oxygen defects. From the Ti-2p spectra, the peak position of original sample migrates towards to lower binding energy which denote as Ti3+. This indicates that the color change was due to the valence change from Ti4+ to Ti3+ because of the decrease in the oxygen content. The color of the sample resumed to be white by heating in air, due to the recovery of oxygen defects.

Electron spin resonance (ESR) has been performed to investigate the defects evolution in irradiation samples and annealing samples. The samples annealed in low vacuum have more defects than others, even more defects than irradiated sample with the dose of 2e17/cm2. However, the sample annealed in high vacuum has lower defects. This indicates that if Li2TiO3 works in low vacuum the annealed defects must be considered. The defects increase as the irradiation doses go up. However, when the temperature arrives at 450K the defects concentration begin to increase. This may be due to defects re-combination and release more electron vacancy. From the ESR spectra and the defect concentration, the defects decrease in about the first 5minutes, and then go up as the annealing time increase. This also may be attributed to defects re-combination and release more electron vacancy as the conditions of annealing in different temperature.

Tritium Production Assessment for the DCLL EUROfusion DEMO

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In order to guarantee the viability of a fusion reactor the most important condition to achieve is the tritium self-sufficiency measured through the Tritium Breeding Ratio (TBR) parameter. Due to the various uncertainties and plant-internal losses that occur during DEMO operation that made difficult to predict exactly the produced Tritium, it is required to obtain a TBR target of 1.1 to have a 10% of margin (for a final net TBR ≥ 1.0) to account for the possible losses generated by the unevaluated aspects.

Taking these assumptions as the basis of this work, an assessment of different parameters representing the tritium production, as the TBR, the Tritium Production Rate (TPR) density and their poloidal and radial variations along the breeder structures has been performed for the last DCLL DEMO design developed in the frame of the EUROfusion Programme.

The overall value of 1.104 obtained allows accomplishing the fuel self-sufficiency requirement. This TBR value includes not only the contribution of the Breeding Blanket (BB) modules but also of the Back Supporting Structures (BSS). In fact, the new Manifold design with an high amount of LiPb is highly producing tritium especially in the Inboard (IB) side in which the radial BB thickness is limited by the Central Solenoid presence. The novel BSS design is resulted fundamental to reach the 1.1 criterion.

Lastly, the influence of the integration in the reactor of the heating and current drive (H&CD) systems that will penetrate the breeder volume reducing the amount of material available for tritium production has been evaluated. In fact, different heating systems, such as NBI (Neutral Beam Injection), ECRH (Electron Cyclotron Resonance Heating) and ICRH (ion Cyclotron Resonance Heating) systems, are supposed to be installed inside the Equatorial and Upper Ports of the DEMO tokamak entering the BB modules that could be affected in their functionality. Assuming different configurations for them, the TBR loss has been determined.

All the calculations have entailed the use of an appropriate particle transport code, the MCNP5 Monte Carlo code, and nuclear data library, JEFF 3.1.1. Parallel computations have been carried-out in CIEMAT EULER cluster. Direct simulation results have been normalized to 5.581x1020 neutrons per second source, corresponding to a 1572 MW fusion power.

Isotope Separation System at the University of Rochester's Laboratory for Laser Energetics

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The Laboratory for Laser Energetics (LLE) operates within a 1.5-g tritium inventory limit. The gas is used to fill thin-walled plastic shells with deuterium–tritium (DT) gas or DT ice layered on the inner surface of the shells to study inertial confinement fusion.

Over the past few years, the fuel supply has gradually degraded from a T/D/H ratio of 55/45/0 to 38/59/3. The net effect of this gradual shift in the isotopic ratio has been to make the fuel deuterium rich and to concentrate noticeable amounts of protium inside the target, which interferes with implosion kinetics. Both effects have degraded the fusion neutron yield.

A review of the hydrogen isotope separation technologies suggested that a scaled-down version of the thermal cycling absorption process (TCAP)1 developed at Savannah River National Laboratory would be ideally suited to meet LLE's needs. The system offers several unique advantages. It has no moving parts with the exception of one automated valve. Gas is shuffled by heating or cooling storage beds and columns. The system is compact. Atomic mass sorting of the hydrogen species occurs on a palladium on kieselguhr (Pd/k) column. Molecular hydrogen isotopologues are sorted by mass on a cold molecular sieve (MS) column. Both processes are complementary and reinforce the movement of the heaviest species to the front end of the Pd/k column while the lightest species prefer to accumulate at the exhaust end of the MS column.

This paper outlines the key features of the Isotope Separation System, discusses commissioning results using hydrogen/deuterium mixtures, and summarizes the outcome of purifying LLE's fuel supply.

This material is based upon work supported by the Department of Energy National Nuclear Security Administration under Award Number DE-NA0001944, the University of Rochester, and the New York State Energy Research and Development Authority. The support of DOE does not constitute an endorsement by DOE of the views expressed in this article.

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Heat Transfer Experimental Verification of a High Power Tritium-Titanium Rotating Target Prototype

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High intensity D-T fusion neutron generator (HINEG) is an experimental facility for neutronic physics, which was developed by Institute of Nuclear Energy Safety Technology, Chinese Academy of Sciences. Tritium target is a challenging component of D-T neutron generator, especially for high intensity D-T fusion neutron generator system. According to the theoretical study, the heat flux was about 2.4 kW/cm2 for 1012 n/s rotating target system. As well known, the tritium will be obviously released while the temperature of tritium-titanium target is higher than 200 oC.

In this contribution, a series of experimental researches were carried out. A high power fiber laser (6 kW) was used as the heat source to simulate the thermal effects of deuterium ion beam on the rotating target. The experiments were carried out in air, and it found that the surface of rotating target didn't have any sign of oxidized after heating by laser. The temperature on surface of rotating target were measured by infrared thermography, and the maximum temperature on rotating target was about 105 oC while the actual heat flux was 2.4 kW/cm2. The results showed that the temperature was lower than the safety operation temperature of rotating target (200 oC).

It indicated that the rotating target could operate successfully in the condition of neutron yield of about 1012 n/s.
In addition, a principle verification facility design for 1013 -1014 n/s rotating tritium target system of the D-T fusion neutron generator was also proposed in this study and the preliminary experiment were carried out and the results shown that maximum temperature of the target surface was about 89.4 oC under 60 kW heating condition.

Preliminary Designing and R&D Progresses of Fuel Cycling System for CFETR

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CFETR (China Fusion Engineering Test Reactor) is a tokamak-type reactor, which is being designed by China national integration design group, and one of the objectives of the CFETR is to demonstrate the tritium self-sufficiency. It may be a good complementary with ITER, its demonstration fusion energy(Pf) will be 50 ~ 200MW,and its long pulse or steady-state operation duty cycle time may be 0.3 ~ 0.5.

As CFETR is an engineering test reactor, the D-T fuel cycling system will be quite different from ITER, it must be simpler for system and higher efficient for processing in order that the tritium could be self-sufficiency.A D-T fuel cycling system must be developed which include inner D-T fuel cycling system , outer D-T fuel cycling system and tritium confinement system.The inner D-T fuel cycling system functioned as recovering D-T fuel from the tokomak plasma exhaust gases, purifying the obtained D-T fuel, and further separating the hydrogen isotopes , then stored for feeding back to the plasma vessel. And the outer D-T fuel cycling system functioned as breeding and extracting enough tritium for its self-sufficiency, and also further separating the hydrogen isotopes , then stored for feeding back to the plasma vessel. While, the tritium confinement system functioned as holding the D-T fuelprocessing system and recovering tritium from the tritiated gases and liquid effusions.

The requirements of D-T fuel cycling system for CFETR were analyzed, and the preliminary designed D-T fuel cycling system and the R&D progresses of CFETR will be briefly introduced in the presentation. For the inner and outer D-T fuel cycling system, it has contained storage and delivery system(SDS), pumping system, tritium breeding and extraction system(TES), tokomak exhaust processing system (TEP), isotope separation system (ISS), and tritium analysis system (ANS) , etc. which is almost the same as the fuel cycling system of ITER, but completely different in TEP and ISS. For the tritium confinement system, it has contained vent detritiation system (VDS), atmosphere detritiation system (ADS), water detritiation system (WDS) and tritium detection system, etc. which is also quite similar to the tritium confinement system of ITER, but also completely different in WDS.

Paper 18A-4: Friday, April 22

Tritium Facilities in JAEA and Future Plan

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Tritium Process Laboratory (TPL) at the Nuclear Science Research Institute of Japan Atomic Energy Agency (JAEA) is the first facility which uses tritium over 1 gram in the fusion research of Japan. Tritium handling at TPL started in 1988 and 8.5 PBq of tritium is stored in the TPL. Main objective of the TPL is to study tritium engineering technology for Fusion and its activity has continued until now.

The Broader Approach (BA) activities aiming to the development for a DEMO fusion reactor started from 2007. In BA activities, the R&D of tritium technology was selected as important issue for the DEMO reactor. To carry out the R&D of tritium technology, International Fusion Energy Research Center (IFERC) including DEMO R&D building was constructed in Rokkasho BA site of Japan. In contrast with the TPL, the R&D building is a facility to handle not only tritium, but also other radioisotopes and Beryllium.

Presentation gives the status of two facilities and the progress of tritium engineering technology together with overlook of significant results in both facilities. And future plan of new tritium handling facility of JAEA at Rokkasho BA site will be also introduced.

Renewing the Tritium Facilities at Chalk River

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For at least four decades, AECL and now CNL has maintained a facility at the Chalk River Laboratory site for studying and handling tritium in all its forms. Licensed to store and handle large quantities of tritium, it has provided focused, but diverse capabilities to the CANDU reactor community, as well as commercial and public interests, such as supplying tritium for emergency lighting and R&D expertise to various nuclear fusion research programs. The facility is unique in Canada, so maintaining, upgrading and expanding it is considered essential to providing continued access to Canadian-sourced tritium and the existing tritium-related capabilities for Canadian and international customers. The existing Chalk River tritium facility is housed in an old and insecure building, with many support services that are not up to current standards.

Over the past few years plans have taken shape to build a new tritium facility and move the existing operation into a facility that meets current codes and standards. The new facility will of course be built with up-to-date materials and equipment and allow for expansion of operations, both in scale and diversity. This presentation describes the efforts to design state-of-the-art facilities and replace existing operating equipment with improved and modernized designs. It will also describe the diverse activities within the facility and the further diversification of the R&D program.

Paper 19-1: Friday, April 22

Progress in Tritium Standards and Regulations

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The tritium standards and regulations panel will address regulations, standards and good practices as they relate to tritium processing and tritium environmental impacts. There are few organizations that publish tritium guidance documents, the most notable ones being the International Atomic Energy Agency (IAEA) which produced the Technical Report Series, Number 324, Safe Handling of Tritium, published in 1991, in addition to the French Nuclear Safety Authority (ASN) guidance documents, and those from the Canadian, UK and U.S. (Department of Energy). In addition, there are internal documents associated with ITER that represent excellent resources.

This panel will host discussions and advice on compliance-based issues and provide insight into those areas in which regulatory guidance is currently unclear. The full life cycle of tritium: generation, processing, storage, transportation and eventual disposition to the environmental impacts of routine and accidental tritium releases will be discussed. Good practices associated with operational improvements that have that have the potential to minimize environmental impact and doses to operators will be discussed including one such improvement associated with bulk tritium measurement and accountancy at AWE.

A final topic of interest current research and development activities underway that may allow more realistic assumptions to be considered in safety analysis space. The panel will discuss current and future directions in the Standards & Regulations areas from the perspective of various national and international agencies and organizations as the panel members are comprised of members from the UK, Canada, ITER and the U.S.

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