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Proceedings of the Research Conference on Post-accident Waste Management Safety (RCWM2016) and the Technical Seminar on Safety Research for Radioactive Waste Storage

November 7th and 8th, 2016, LATOV, Iwaki, Fukushima, Japan

(Eds.) Takafumi MOTOOKA and Isao YAMAGHISHI

Waste Management Division
Collaborative Laboratories for Advanced Decommissioning Science
Sector of Fukushima Research and Development

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(Received January 18, 2017)

Collaborative Laboratories for Advanced Decommissioning Science (CLADS) is responsible to promote international cooperation in the R&D activities on the decommissioning of Fukushima Daiichi Nuclear Power Station and to develop the necessary human resources. CLADS held the Research Conference on Post-accident Waste Management Safety (RCWM2016) on 7th November, 2016 and the Technical Seminar on Safety Research for Radioactive Waste Storage on 8th November, 2016. This report compiles the abstracts and the presentation materials in the above conference and seminar.

Keywords: Post-accident, Waste Management, Safety, Radioactive Waste, Storage, Fukushima Daiichi Nuclear Power Station, CLADS

「事故廃棄物の安全管理に関する研究カンファレンス（RCWM2016）」と「放射性廃棄物保管の安全性研究に関する技術セミナー」講演集

2016 年 11 月 7 日〜8 日、LATOV（福島県いわき市）

日本原子力研究開発機構　福島研究開発部門　廃炉国際共同研究センター
廃棄物処理処分ディビジョン

（編）本岡　隆文、山岸　功

（2017 年 1 月 18 日受理）

廃炉国際共同研究センター（CLADS）は、福島第一原子力発電所の廃止措置の加速化や人材育成に資するため、国内外の研究協力を進めている。CLADS は、「事故廃棄物の安全管理に関する研究カンファレンス（RCWM2016）」を 2016 年 11 月 7 日に、「放射性廃棄物保管の安全性研究に関する技術セミナー」を 11 月 8 日に開催した。本報告書は上記研究カンファレンスと技術セミナーの講演要旨と発表資料を収録したものである。

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1. Preface

To accomplish the decommissioning of the Fukushima Daiichi Nuclear Power Station (NPS), we need to gather the knowledge of domestic and foreign experts, and share understanding of the present needs and future issues. It is also necessary to strengthen R&D activities on not only technologies applicable to the decommissioning work but also fundamental and scientific studies.

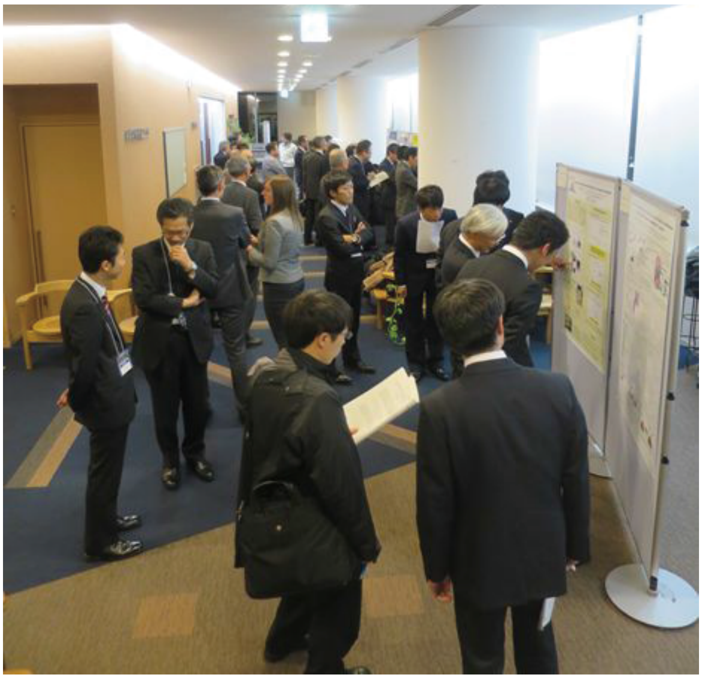
Collaborative Laboratories for Advanced Decommissioning Science (CLADS) established in Japan Atomic Energy Agency (JAEA) is responsible to promote international cooperation in the R&D activities on the decommissioning of Fukushima Daiichi NPS and to develop the necessary human resources.

CLADS held the Research Conference on Post-accident Waste Management Safety (RCWM2016) in Iwaki city on 7th November, 2016. The purpose of the conference was to provide the place and occasion to share the experiences in safe handling and management of radioactive waste in order to promote the research activities in Japan. More than sixty people including nine overseas experts participated in the conference. Young researchers such as Fukushima University and Fukushima, College students joined in the poster session. Participants shared the information on the present state of Fukushima Daiichi wastes and the wider experiences in the waste management. The technical seminar on safety research for radioactive waste storage followed the conference.

Through the discussion in the conference and seminar, participants recognized the importance of the further cooperation and development of the human resources to utilize the experiences of the overseas waste management for Fukushima Daiichi NPS. Presentation materials of lecturers were issued as this review report by mutual agreement.

Participants in RCWM2016

Scenery of RCWM2016



Scenery of Poster Presentation

Scenery of Technical Seminar

2. Research Conference on Post-accident Waste Management Safety (RCWM2016)

2.1 Management of Solid Wastes Arising from Fukushima Daiichi Decontamination and Decommissioning

Tatsuro KOBAYASHI
Tokyo Electric Power Company Holdings, Inc.; kobayashi.tatsuro@tepco.co.jp

1. Waste Arising from 1F Decontamination and Decommissioning

Not only large amount but also many kinds of radioactive wastes (gaseous, liquid and solid wastes) are generated by decontamination and decommissioning works in Fukushima Daiichi (1F), different from decommissioning at ordinary nuclear power stations. Most radioactivity has been concentrated in the solid wastes to be safely stored in 1F site and prevent release and dissemination of radioactivity to the environment.

2. Waste Flow from Waste Generation to Storage

These solid wastes have been temporarily stored in the tentative storage areas. Then, they are transferred to the storage buildings step by step and stored there until waste processing and disposal phase. However, because of the limitation of the capacity of the storage buildings, the wastes should be volume-reduced as much as possible and more stabilized to be able to put them into the storage buildings.

3. Current Status of Tentative Storage

At present, approximately 300,000m^3 of rubble, etc. (rubble, felled tree, and used radioactive protective equipment) stored in 1F site are categorized as follows: (1) Rubble (contaminated metal, concrete, and miscellaneous combustible materials, etc.) stored in tentative storage areas, such as outdoor, sheet covered, container, or soil covered storage, etc. depending on their surface dose rate, (2) Felled trees (trunks, roots, leaves and branches) stored in outdoor accumulation or underground storage pools to reduce fire risks, and (3) Used radioactive protective equipment (coveralls, masks, shoes, and gloves, etc.) stored in outdoor container prior to incineration. In terms of the secondary wastes arising from contaminated water treatment, (4) Used absorbent towers stored in racks or box culverts depending on required shield, and (5) Sludge and concentrated liquid stored in the existing buildings or tanks.

4. Future Challenge

The volume of rubble, etc. is estimated to reach 750,000m^3 in 2028, which will exceed tentative storage capacity. Therefore, reuse and recycling process of low contaminated concrete and metals should be established to reduce the volume. In addition, the estimation of the waste volume should be precise as much as possible applying for knowledge and experiences obtained since the accident. Then volume reduction facilities and additional storage buildings will be designed and installed in a planned manner according to the volume estimation. Finally, tentative storage areas will be released step by step after newly installed storage buildings begin operation.

In terms of the secondary wastes generated in the process of contaminated water treatment system, the large size waste storage building will be constructed to accommodate the wastes from present racks and box culverts. However, because potential risk of leakage still remains, application of dehydration technology is under development using mock-up equipment by the International Research Institute for Nuclear Decommissioning (IRID) in order to stabilize the slurry wastes arising from multi nuclide removal facility (ALPS).

In addition, characterizing the wastes is indispensable for waste processing and disposal in future, so the new research laboratories will be constructed adjacent to 1F site by the JAEA.

Management of Solid Wastes Arising from Fukushima Daiichi Decontamination and Decommissioning

November 7th, 2016

Fukushima Daiichi Decontamination and Decommissioning Engineering Company,
Tokyo Electric Power Company Holdings, Inc.

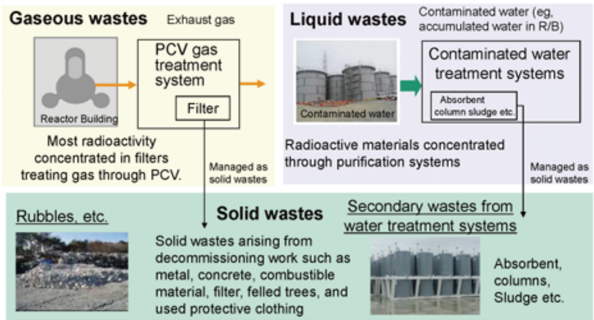
TEPCO

Contents

TEPCO 1

Wastes arising from 1F decontamination and decommissioning

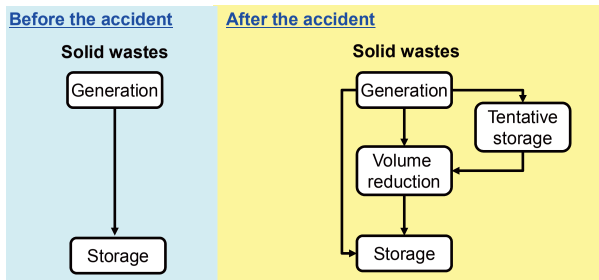
Gaseous, liquid and solid wastes are generated by decommissioning works.
Most radioactivity concentrated in the solid wastes to be safely stored.



TEPCO 2

Wastes flow from generation to storage

Before the accident: Solid wastes stored in storage buildings with containment and shielding
After the accident: Vast amount of wastes generated by recovering works from emergency situation in short time. Due to the limited capacity in waste storage buildings, large part of wastes have been stored outside temporarily as an emergency response. The wastes stored in tentative storage will be volume-reduced as much as possible leading to storage in the solid waste buildings



TEPCO 3

Classification of rubbles, etc. and storage options

Rubbles, etc.

- **Rubbles**: Contaminated concrete, metal, combustible materials, soil, etc.
- **Felled trees**: Trees cut to construct accumulated water treatment tanks, etc. → Stored in buildings after incineration
- **Used protective clothing**: Coveralls, musks gloves and undergarments, etc. → Stored in buildings after incineration

Classified by the surface dose rate of wastes

Outdoor storage	Outdoor Sheet covered storage	Outdoor container storage	Tentative storage tent	Soil covered tentative storage facility	Solid waste storage buildings
0.1mSv/h or less	1mSv/h or less	10mSv/h or less, with β contamination	30mSv/h or less	30mSv/h or less	More than 30mSv/h

TEPCO

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Storage volume of rubbles, etc.

Rubbles As of 2016/9/30

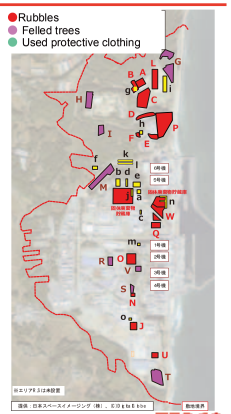
Surface dose (mSv/h)	Tentative Storage Area	Storage Volume (m³)	Storage Capacity (m³)
≦0.1	B, C, F1, J, N, O, P1, U.	130,400	181,200
≦1	D, E1, P2, W	37,100	57,300
1～30	A1, A2, E2, F2, L, Q	20.300	27,700
＞30	No.3 ～No.8 storage building	7,500	12,000
Total	—	195,400	278,200

Felled trees

Category	Tentative Storage Area	Storage Volume (m³)	Storage Capacity (m³)
Outdoor	H, I, M, V	70,200	81,500
Pit	G, T	19,600	24,900
Total	—	89,800	106,400

Used protective clothing

Tentative Storage Area	Storage Volume (m³)	Storage Capacity (m³)
a, b, c, d, e, f, g, h, i, j, k, l, m, n, o	68,300	71,200



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Overview of tentative storage areas for rubbles

Outdoor storage (0.1mSv/h or less)

Outdoor container storage (10mSv/h or less)

Soil covered tentative storage facility (30mSv/h or less)

Inside of solid waste storage building (More than 30mSv/h)

TEPCO

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Storage methods for combustible wastes

- Combustible wastes, such as felled trees, used protective clothing, rubbles (combustible and resistance to flame), and waste oil, have been stored by following methods in site
- Fire protection measures specifically for combustible wastes are taken because they have to be stored temporarily until incineration

Wastes	Storage methods
Felled trees (trunks, roots, branches and leaves)	Outdoor storage Tentative pit
Used protective clothing, etc.	Storehouse Container storage
Rubbles, etc. (combustible and resistance to flame)	Outdoor storage Container storage
Waste oil	Hazardous material storage

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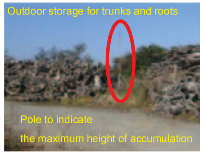
Fire protection measures for trunks and roots (outdoor storage)

◆ Fire protection
- ✓ Limit the accumulating height less than 5m to keep ventilation
- ✓ Separate from hazardous material and ignition device to spark
- ✓ Patrol to check the storage condition once a week
- ✓ Monitor inside temperature of felled tree accumulation (except for trunks) once a week. Increase monitoring frequency 3 times a week during summer season

◆ Early fire detection and extinguishing
- ✓ Daily patrol
- ✓ Deploy fire extinguisher within the storage areas and water tank near the areas
- ✓ Plan to install automatic fire watching camera with fire detection and alarm function

◆ Actions on temperature increase
- ✓ Spray water to high temperature part if its temperature exceeds 60℃
- ✓ Reshuffle the position of felled trees if temperature exceed 60℃ and continue to increase



Monitor inside temperature by thermometer

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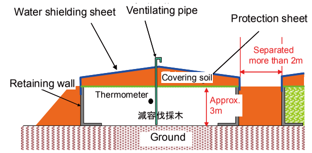
Fire protection measures for branches and leaves (tentative pit)

◆ Fire protection
- ✓ Limit the accumulating height less than 3m
- ✓ Cover upper part by soil to break oxygen
- ✓ Suspend accepting during summer season
- ✓ Separate from hazardous material and ignition device to spark
- ✓ Patrol storage condition once a week
- ✓ Monitor inside temperature and combustible gas once a week. Increase monitoring frequency to 3 times a week during summer season

◆ Early fire detection and extinguishing
- ✓ Daily patrol
- ✓ Deploy fire extinguisher within the storage areas and water tank near the areas

◆ Fire spread protection
- ✓ Separate pit more than 2m from adjacent pits

◆ Actions on temperature increase
- ✓ In case of more than 60℃, prepare actions.
- ✓ In case of more than 70℃, intense monitoring by inserting thermometer in gas release pipe
- ✓ In case of more than 80℃, inject nitrogen gas for cooling



Schematic cross section of tentative pit

Ventilating pipes (center and 4 corners)
- ➢ Area of one pit: within 200m2
- ➢ Capacity of one pit: within 600m3
- ➢ Gas release pipes of one pit: center and 4 corners

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Fire protection for used protective clothing, etc. (container storage)

◆ Fire protection
- ✓Put the wastes into metal container to protect and spread fire
- ✓Separate from hazardous material and ignition device to spark
- ✓Patrol storage condition once a week

◆ Early fire detection and extinguishing
- ✓Daily patrol
- ✓Deploy fire extinguisher in the storage areas and water tank near the areas

◆ Others
- ✓Incinerate coveralls in the miscellaneous incinerator with first priority
- ✓Put ash into drums and store them in the existing storage buildings

TEPCO 10

Fire protection for rubbles, etc. (container storage)

◆ Fire protection
- ✓ Protect fire by putting the wastes into metal container
- ✓ Patrol storage condition once a week

◆ Early fire detection and extinguishing
- ✓ Daily patrol
- ✓ Deploy fire extinguisher in the storage areas and water tank near the areas
- ✓ Plan to install automatic fire watching camera with fire detection and alarm function

◆ Others
- ✓ Evaluate fire risk is low by fire specialist taking account of inside corrosion conditions
- ✓ Put hose and lumber into metal container after crushing into small pieces
- ✓ Put other existing combustible as well as resistance to flame wastes into large size metal container without classification with first priority
- ✓ Classify newly arising combustible as well as resistance to flame wastes by classifying the sorts of material and put them into metal container separately

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Classification of secondary wastes from water treatment systems and storage options

Secondary wastes from water treatment systems

- Used vessels — Stored in tentative storage facilities (outdoor)
- Sludge — Stored in sludge storage facility (indoor)
- Concentrated liquid waste — Stored in concentrated liquid waste storage tanks (outdoor)

There is no plan to increase these volume because equipment which generate these wastes stopped operation.

Classified by characteristics of wastes

Rack	Box culvert	HIC-supporting type box culvert
Wastes which do not need additional shield	Wastes which need additional shield	Wastes which include much water contents
e.g. 2nd Cesium absorption apparatus (SARRY)	e.g. Cesium absorption apparatus (KURION)	e.g. High Integrity Containers (HICs) from multiple nuclide removal equipment

1st facility, 4th facility (Rack, Box culvert)
2nd facility, 3rd facility (HIC-supporting type box culvert)

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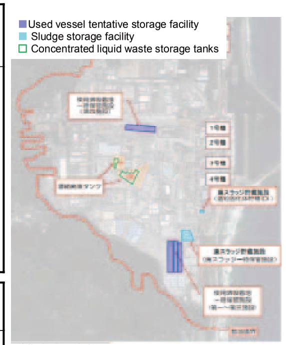
Storage number of secondary wastes from water treatment systems

As of 2016/10/20

Used vessels	Storage number	Storage number / Capacity (%)
Cesium absorption apparatus	754	3,361 / 6,239 (54%)
2nd Cesium absorption apparatus	176	
HICs from multiple radio-nuclides removal system	1,240	
HICs from improved multiple radio-nuclides removal system	923	
Used vessels from high-performance multiple radio-nuclides removal system	73	
Used column from multiple radio-nuclides removal system	9	
Used vessels and filters from mobile-type strontium removal system	186	

Wastes	Storage volume		Storage volume /capacity (%)
Sludge	597	m3	597／700 (85%)
Concentrated liquid waste	9,167	m3	9,167／10,700 (86%)



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Overview of tentative storage areas for used vessels

1st facility	Rack, Box culvert
2nd facility	HIC-supporting type box culvert
3rd facility	HIC-supporting type box culvert
4th facility	Rack, Box culvert

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Forecast of waste generation (before new and additional facilities)

Without additional volume reduction, rubbles, etc. storage volume will reach about 750,000m^3 in 2028.

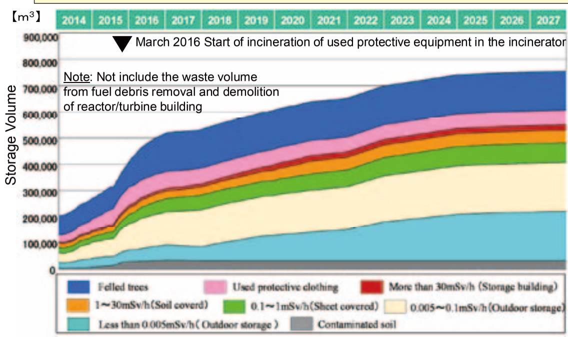

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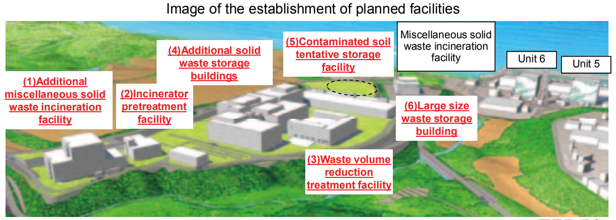
Image of planned facilities to improve waste storage manner

In order to improve our waste storage manner, we have a plan to construct several facilities as follows;
(1)Additional miscellaneous solid waste incineration facility and (2)Incinerator pretreatment facility
(3)Waste volume reduction treatment facility
(4)Additional solid waste storage buildings (storage capacity of approx. 140,000 m^3 (No.10 through No.13 building))
(5)Contaminated soil tentative storage facility
(6)Large size waste storage building (for used vessels of secondary wastes from water treatment)

Image of the establishment of planned facilities



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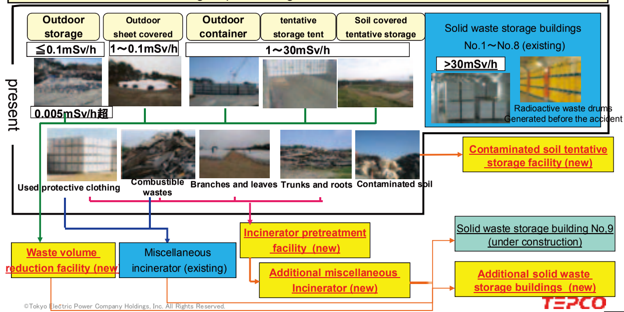
Purpose of new and additional facilities【rubbles, etc.】

- In order to eliminate tentative storage, additional miscellaneous incinerator and its pretreatment facility, waste volume reduction treatment facility, and additional solid waste storage buildings are necessary
- Contaminated soil tentative storage facility is necessary because the method of storage will be decided after considering its processing method



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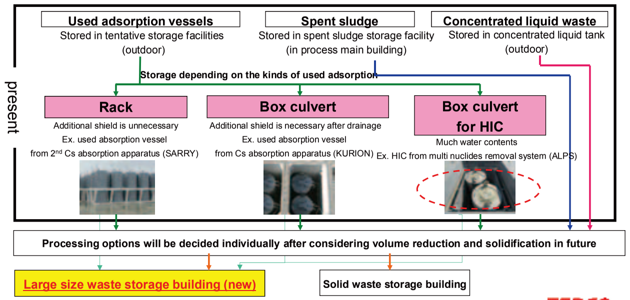
Purpose of new and additional facilities【secondary wastes from contaminated water treatment】

The secondary wastes from contaminated water treatment will be stored in storage buildings to eliminate outdoor tentative storage facilities. Therefore, large size waste storage facility is necessary until volume reduction as well as stabilization of the wastes are decided and carried out



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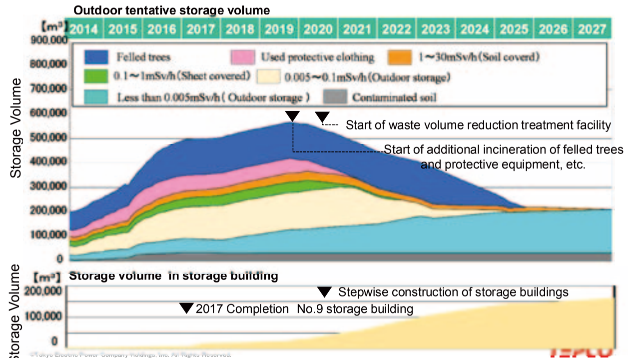
Forecast of waste generation (after new and additional facilities)

By 2028, outdoor tentative storage areas will be eliminated except those for low contaminated rubbles and contaminated soil



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Purpose of new and additional facilities【present layout】

Currently, solid wastes, such as rubbles less than 30mSv/h and secondary wastes from contaminated water treatment have been stored in outdoor tentative storage areas located in site with special permission because the volume of these wastes much exceeds the existing storage building capacity

Current layout of tentative waste storage areas for rubbles and secondary wastes in 1F site

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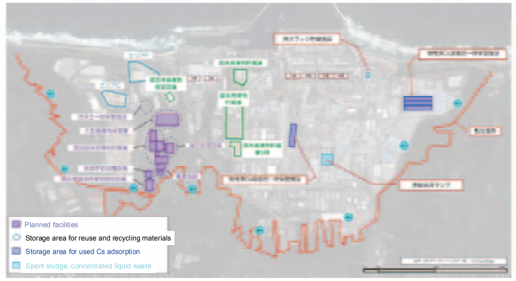
Purpose of new and additional facilities【future image】

In order to reduce risks during storage, the volume of rubbles will be reduced as much as possible, then the wastes will be transferred to inside storage buildings. Finally, outdoor tentative storage will be eliminated by around 2028.

Future layout image of waste storage areas for rubbles and secondary wastes in 1F



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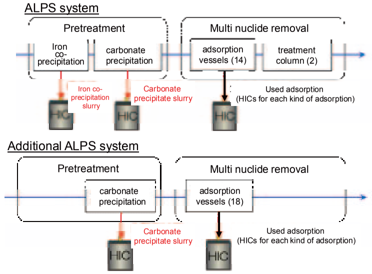
Outline of multi nuclide removal equipment (ALPS)

- 62 kinds of radioactive nuclides are removed by processing contaminated water through pretreatment and multi nuclide removal equipment
  - Slurry wastes are generated in precipitation process in pretreatment equipment
  - Slurry wastes are discharged to HICs
  - HISs are stored in tentative storage facility for used Cs adsorption



Number of HICs as of 2016/10/6
ALPS : 1,229
incl. iron co-precipitation slurry : 319
carbonate precipitation slurry : 820
Additional ALPS : 919
incl. carbonate precipitation slurry : 836

Iron co-precipitation slurry

Carbonate precipitate slurry

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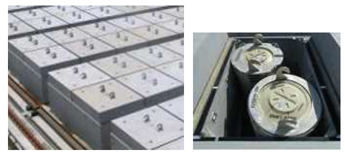
Storage conditions of the slurry wastes

- HICs have been stored in spent Cs adsorption tentative storage facilities
  - location : approx. 35m above sea level
  - capacity : 4,192 HICs
- Concrete culvert with radiation shield as well as watertight to store HICs containing much water contents
- Continue to reduce water level in HICs, extract top liquid, and inspect due to the puddles found on the upper lid of HICs on April 2015
  - Found no puddles since November 2015

Outside appearance of steel reinforcing material

Inside of steel reinforcing material

HIC



Outside appearance of box culvert

Before the concrete lid is on

Spent Cs adsorption vessel tentative storage facility (facility No.2)

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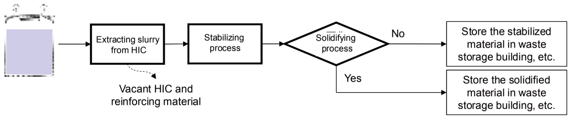
Stabilization and solidification of slurry

- Dehydration and stabilization of the slurry wastes have been under development by the IRID to reduce potential risk of leakage because the slurry include much water contents
- TEPCO is also cooperating on the development to apply the real wastes in future



Process flow of slurry wastes

- R&Ds related to stabilize the slurry wastes are as follows:
  - Evaluate results of applicability to stabilize slurry
  - Extract slurry from HIC, and rinse vacant HIC
  - Restrict dust to scatter from stabilizing equipment, and reduce radiation exposure to workers
  - Consider storage method of stabilized materials
  - Consider applicability and necessity of solidification technology

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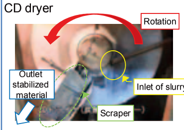
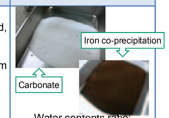
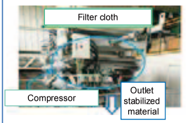
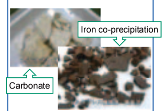
Present status of development to stabilize the slurry wastes

- Dryer, filter press, and centrifugal technologies are selected to carry out dehydration test using mock up cold slurry, considering characteristics of the slurry, remote control, and process speed
- Dryer and filter press technology are perspective to be feasible for stabilizing the slurry (centrifugal separator is insufficient for dehydration and difficult to apply)

Selected technology	Process equipment	Principle, feature	Stabilized material
Disc heating dryer	CD dryer 	<Principle> ○Slurry is applied on heated disc, rotated, then powder is discharged by scraping the surface of disc ○Separated water is discharged as steam <Feature> ・Applicable dependent of the particle diameter ・Necessary for tuning the viscosity to apply on the disc	 Water contents ratio: approx. less than 5%
Filter press machine	Filter press machine 	<Principle> ○Slurry is filtrated and pressed, then dehydrated solid board material is discharged form the bottom of the machine ○Separated water is collected with filter cloth rinsing water <Feature> ・Many examples in sludge treatment ・Possible for high-throughput	 Water contents ratio: approx. 50%

Source: R&D results obtained by the IRID

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Future tasks for stabilization of the slurry wastes

- Extraction of slurry from HICs, and rinsing the vacant HICs
  - Plan to carry out extracting slurry from HICs and rinsing the vacant HICs to confirm feasibility using mock up slurry because viscosity of the slurry is high
    (Vacant HICs after rinsed are assumed to be incinerated and steel reinforcing materials are broken into small pieces in a newly established volume reduction facility)
  - Confirm the feasibility to extract the slurry as well as the residue at the bottom of HICs by the test

- Prevention of dust to scatter and reduce radiation exposure to workers
  - Consider to prevent measures against scattered dust and to reduce radiation exposure to workers using radioactive density obtained by analysis. Specific measures will be evaluated in general design process

- Storage method of stabilized material
  - Consider arising hydrogen gas, transition of moisture, decay heat in storage container for dehydrate material. Specific container will be evaluated in general design process

Source: R&D results obtained by the IRID

TEPCO 26

Present status of development to solidify the slurry wastes

- Applicability and necessity of solidification technology
  - It is confirmed to solidify the mock up slurry in laboratory tests with 2 kinds of cement, and geo-polymer

Solidification material	Ordinary Portland cement	Blast furnace slag cement	Geo-polymer
Iron co-precipitation slurry			
Carbonate precipitation slurry			

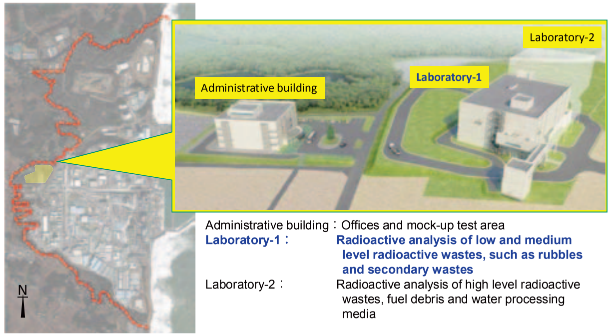
Samples of mixing solidification of dehydrated slurry (filling factor 40%)

- Consider to apply solidification taking account of long term safety and volume reduction as well as effects on disposal

Source: R&D results obtained by the IRID

TEPCO 27

JAEA analysis and research laboratory near 1F site



Administrative building : Offices and mock-up test area
Laboratory-1 : Radioactive analysis of low and medium level radioactive wastes, such as rubbles and secondary wastes
Laboratory-2 : Radioactive analysis of high level radioactive wastes, fuel debris and water processing media

Source: Japan space imaging, (C)DigitalGlobe

JAEA TEPCO 28

Thank you for your attention.

TEPCO 29

2.2 Activities of CLADS Waste Management Division

Takashi ASHIDA
Japan Atomic Energy Agency; ashida.takashi21@jaea.go.jp

1. Introduction of CLADS

Collaborative Laboratories for Advanced Decommissioning Science (CLADS) of JAEA was established in 2015 to promote research and development towards the decommissioning of the Fukushima Daiichi Nuclear Power Station (1F). To achieve this mission, CLADS consists of four divisions.

(1) Waste Management Division
(2) Fuel Debris Handling and Analysis Division
(3) Severe Accident Propagation Behavior Evaluation Division
(4) Remote System and Sensing Technology Division

Main efforts of the CLADS are as follows;

(1) Establishment of a platform to rally the wisdom of experts from around the world
(2) Enhancement of in and outside Japan research on decommissioning
(3) Enhancement of human resource development in mid-and-long term
(4) Establishment of an information dissemination function

Additionally, as an international research and development base, “International Collaborative Research Building” has being constructing near the 1F as a core of the CLADS.

2. Research and Development of Waste Management

The accident at the 1F and restoration works have produced significant volume of various waste. The characteristics of the 1F accident waste are very different from usual waste generated from ordinal nuclear power stations. The research and development works such as characterization of the waste, investigation on safe storage, processing and disposal of the waste are carried out based on the “Mid-and-Long-Term Roadmap” [1]. The contents of R&D activities of CLADS Waste Management Division are as follows.

(1) Characterization
- Analysis of various wastes
- Evaluation of the amount and composition of radioactivity
- Collection of physical/chemical data
- Development of analysis method

(2) Storage
- Soundness of stainless vessel used for cesium adsorption apparatus
- Safety technology for storing nuclear accident-derived wastes

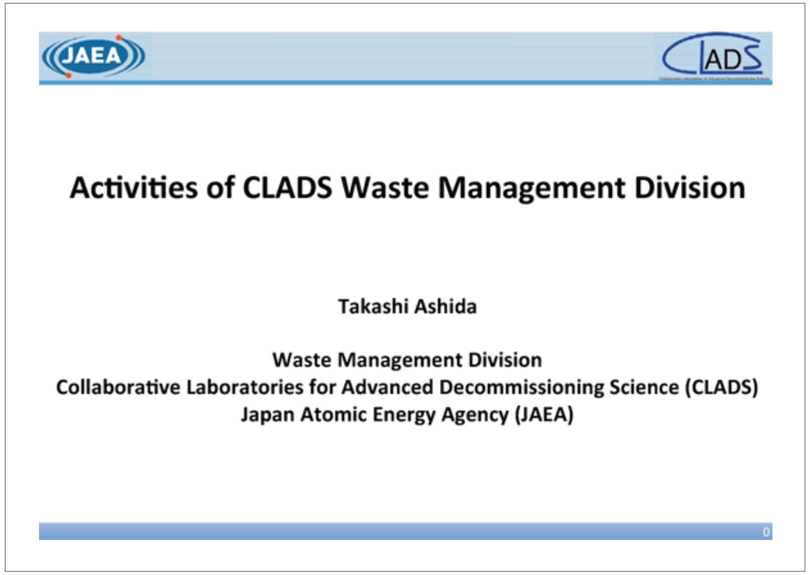
(3) Processing
- Technology survey
- Fundamental test of solidification
- Technology selection

(4) Disposal
- Understanding of the existing disposal concepts
- Examination of disposal concept of accident-derived wastes
- Examination of new disposal concept

These results are integrated to the basic concept of processing and disposal for solid radioactive waste shown in the Roadmap.

Reference

[1] Inter-Ministerial Council for Contaminated Water and Decommissioning Issues, Mid-and-Long-Term Roadmap towards the Decommissioning of TEPCO's Fukushima Daiichi Nuclear Power Station, June 12, 2015.


JAEA
CLADS
Activities of CLADS Waste Management Division
Takashi Ashida
Waste Management Division
Collaborative Laboratories for Advanced Decommissioning Science (CLADS)
Japan Atomic Energy Agency (JAEA)
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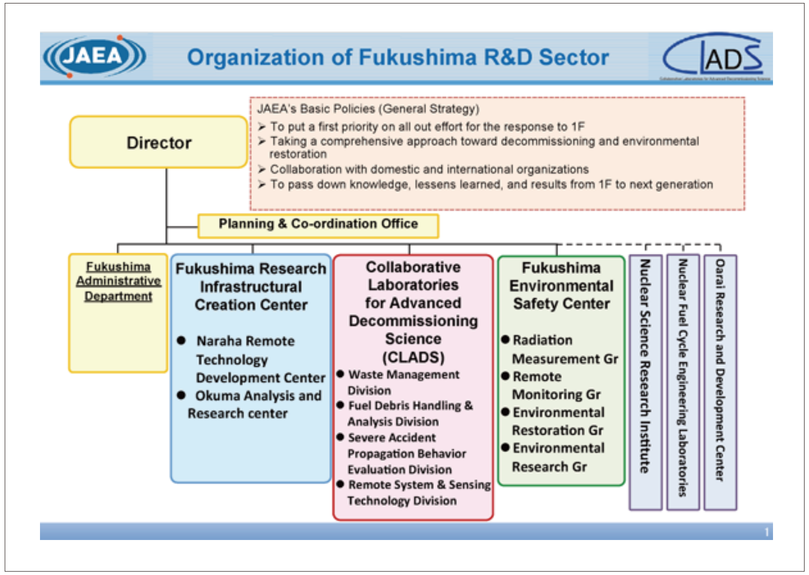
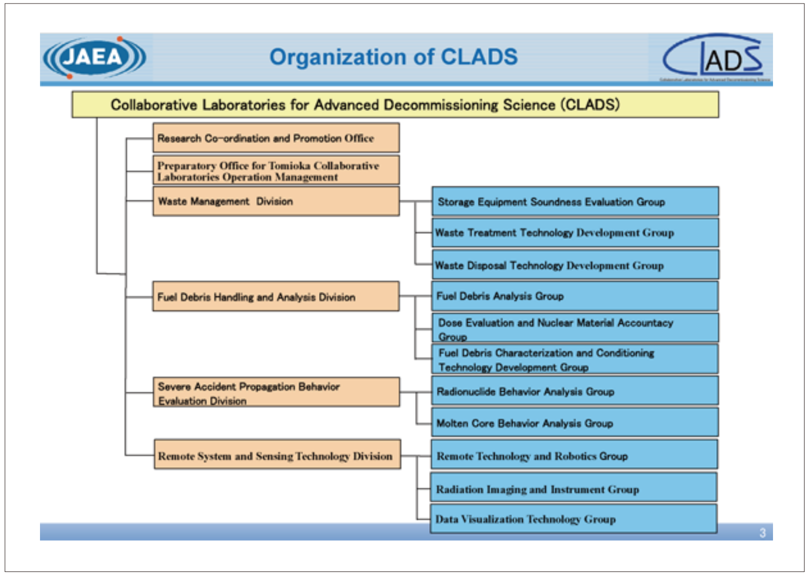
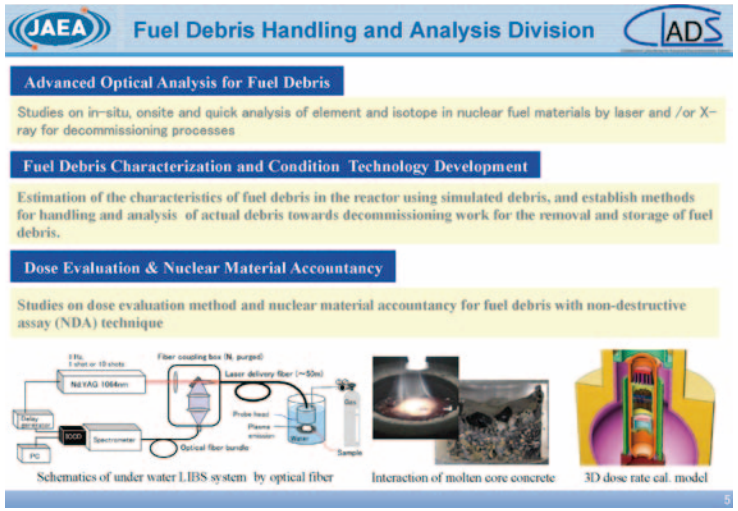
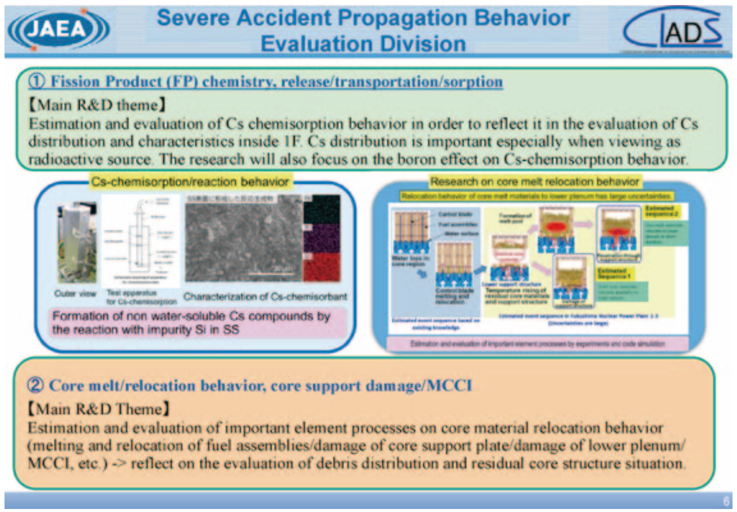

Organization of Fukushima R&D Sector
Director
JAEA's Basic Policies (General Strategy)
➤ To put a first priority on all out effort for the response to 1F
➤ Taking a comprehensive approach toward decommissioning and environmental restoration
➤ Collaboration with domestic and international organizations
➤ To pass down knowledge, lessens learned, and results from 1F to next generation
Planning & Co-ordination Office
Fukushima Administrative Department
Fukushima Research Infrastructural Creation Center
● Naraha Remote Technology Development Center
● Okuma Analysis and Research center
Collaborative Laboratories for Advanced Decommissioning Science (CLADS)
● Waste Management Division
● Fuel Debris Handling & Analysis Division
● Severe Accident Propagation Behavior Evaluation Division
● Remote System & Sensing Technology Division
Fukushima Environmental Safety Center
● Radiation Measurement Gr
● Remote Monitoring Gr
● Environmental Restoration Gr
● Environmental Research Gr
Nuclear Science Research Institute
Nuclear Fuel Cycle Engineering Laboratories
Oarai Research and Development Center
1

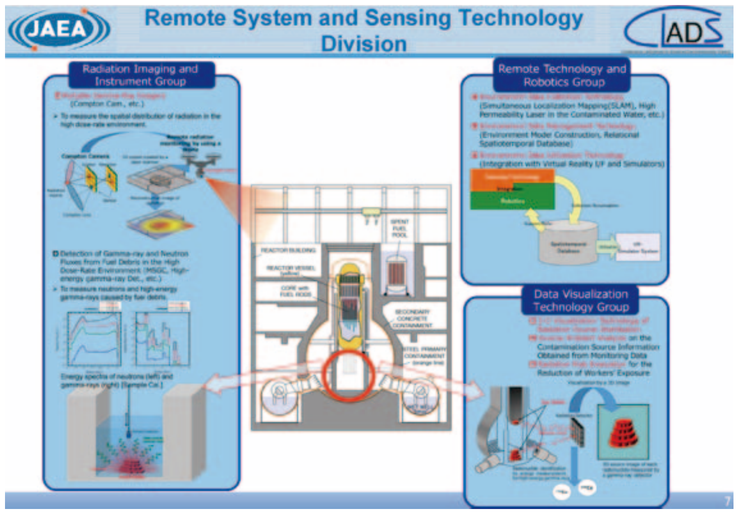

CLADS as international R&D base
Naraha Remote Technology Development Center
Naraha Town, Fukushima Prefecture
Okuma Analysis and Research Center
application
application
International Collaborative Research Building
(near 1F, Tomioka Town, Fukushima Prefecture, under constructing,: 2016~)
Utilizing JAEA's special facilities for handling nuclear fuels and radioactive materials, and irradiation facilities at Tokai and Oarai in Ibaraki Prefecture.
Utilization of results
Fukushima Environmental Safety Center
Research and development on environmental dynamics, radiation monitoring, etc.
『From off-site to on-site』
Cooperation with academia, industry and government.
2


Organization of CLADS
Collaborative Laboratories for Advanced Decommissioning Science (CLADS)
Research Co-ordination and Promotion Office
Preparatory Office for Tomioka Collaborative Laboratories Operation Management
Waste Management Division
Storage Equipment Soundness Evaluation Group
Waste Treatment Technology Development Group
Waste Disposal Technology Development Group
Fuel Debris Handling and Analysis Division
Fuel Debris Analysis Group
Dose Evaluation and Nuclear Material Accountacy Group
Fuel Debris Characterization and Conditioning Technology Development Group
Severe Accident Propagation Behavior Evaluation Division
Radionuclide Behavior Analysis Group
Molten Core Behavior Analysis Group
Remote System and Sensing Technology Division
Remote Technology and Robotics Group
Radiation Imaging and Instrument Group
Data Visualization Technology Group
3


JAEA
Main efforts of CLADS
ADS
(I)Establishment of a platform to rally the wisdom of experts from around the world
●Organization established in April 2015 at Tokai Village, Ibaraki Prefecture
Sector of Fukushima Research and Development
Collaborative Laboratories for Advanced Decommissioning Science
Research Co-ordination and Promotion Office
Preparatory Office for Tomioka Collaborative Laboratories Operation Management
Waste Management Division
Fuel Debris Handling and Analysis Division
Severe Accident Propagation Behavior Evaluation Division
Remote System and Sensing Technology Division
(II)Enhancement of in and outside Japan research on decommissioning
●Inviting foreign researchers
●Collaborative research with oversea research institutes
●Forming a working group including external researchers and experts in the fields necessary to decommissioning
Workshop on decommissioning of 1F
(III)Enhancement of human resource development in mid-and-long term
Integrating analysis technologies from various fields and developing human resources by opening cooperative course with institutions adopted for the "Decommissioning Basic Research: Human Resource Development Program" hosted by the Ministry of education, Culture, Sports.
In order to gather a variety of talented people, the cross-appointment system" are utilized
(IV)Establishment of an information dissemination function
In cooperating with the National Diet Library , JAEA arranges information released by the government and Tokyo Electric Power Company on the basis of IAEA's nuclear accident information categories and transmit the information as the "JAEA Archive".
Transmitting information on documents including JAEA' original research results
4


JAEA
Fuel Debris Handling and Analysis Division
ADS
Advanced Optical Analysis for Fuel Debris
Studies on in-situ, onsite and quick analysis of element and isotope in nuclear fuel materials by laser and /or X-ray for decommissioning processes
Fuel Debris Characterization and Condition Technology Development
Estimation of the characteristics of fuel debris in the reactor using simulated debris, and establish methods for handling and analysis of actual debris towards decommissioning work for the removal and storage of fuel debris.
Dose Evaluation & Nuclear Material Accountancy
Studies on dose evaluation method and nuclear material accountancy for fuel debris with non-destructive assay (NDA) technique
Nd:YAG 1064nm
Spectrometer
Fiber coupling box (N2 purged)
Laser delivery fiber (~50m)
Optical fiber bundle
Schematics of under water LIBS system by optical fiber
Interaction of molten core concrete
3D dose rate cal. model
5


JAEA
Severe Accident Propagation Behavior Evaluation Division
ADS
① Fission Product (FP) chemistry, release/transportation/sorption
【Main R&D theme】
Estimation and evaluation of Cs chemisorption behavior in order to reflect it in the evaluation of Cs distribution and characteristics inside 1F. Cs distribution is important especially when viewing as radioactive source. The research will also focus on the boron effect on Cs-chemisorption behavior.
Cs-chemisorption/reaction behavior
Outer view
Test apparatus for Cs-chemisorption
Characterization of Cs-chemisorbant
Formation of non water-soluble Cs compounds by the reaction with impurity Si in SS
Research on core melt relocation behavior
② Core melt/relocation behavior, core support damage/MCCI
【Main R&D Theme】
Estimation and evaluation of important element processes on core material relocation behavior (melting and relocation of fuel assemblies/damage of core support plate/damage of lower plenum/ MCCI, etc.) -> reflect on the evaluation of debris distribution and residual core structure situation.
6


JAEA
Remote System and Sensing Technology Division
ADS
Radiation Imaging and Instrument Group
Remote Technology and Robotics Group
Data Visualization Technology Group
REACTOR BUILDING
REACTOR VESSEL
SPENT FUEL POOL
7

Waste Management Division (1/3)

Mid-and-Long-Term Roadmap（revised June, 2015）

4－5. Waste management

（2）Processing and disposal

During FY 2017, the "basic concept of processing and disposal for solid radioactive wastes" should be complied, while conducting such measures as characterization of solid waste, studying applicability of processing/disposal technologies widely selected, and developing an analysis method of difficult-to-measure nuclides and an inventory evaluation technology, as well as referring to comments of the Nuclear Regulation Authority.

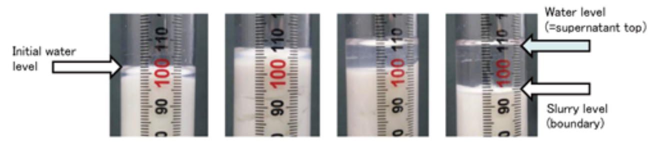
Based on these efforts, actively using radioactive-material analysis/research facilities now under design and accelerating R&D through the characterization of solid radioactive waste, prospects of a processing/disposal method and a technology related to its safety should be made clear by around FY 2021.

8

Waste Management Division (2/3)

Activity on storage of ALPS carbonate slurry

Supernatant of the slurry was observed outside the container (HIC)



Irradiation test of simulated slurry (95g/L)

- Irradiation (0 - 6h): Increase of water level, Formation of bubble and supernatant
- Irradiation (22 - 28h): Growth of supernatant, Decrease of water and slurry levels
- 4 months settling without irradiation: only 2 mm of supernatant were formed. Water level was unchanged.
- Scheme : (1)Gas was generated by water radiolysis (2)Bubbles were retained in the slurry and supernatant was formed (3)A part of bubbles was release from slurry.

R. Nagaishi, T. Motooka, I. Yamagishi, Irradiation experiments of simulated carbonate slurry, EFCOG Nuclear and Facility Safety Workshop, ANL, Illinoi, USA, 9 August 2016

9

Waste Management Division (3/3)

R&D activities : Overview

Obtaining and understanding the waste characteristics

- Sampling waste for radioactivity analysis.
- Estimation of inventory considering with uncertainty.
- Estimation of physical, chemical properties of wastes.

Investigation and development of processing and storage

- Creating candidate technology catalogues on solidification and conditioning, etc.
- Basic experiments of solidification and property measurement.
- Investigation of safety and stability on the storage (support of operation on site, H_2 generation, vessel corrosion estimation, etc.)

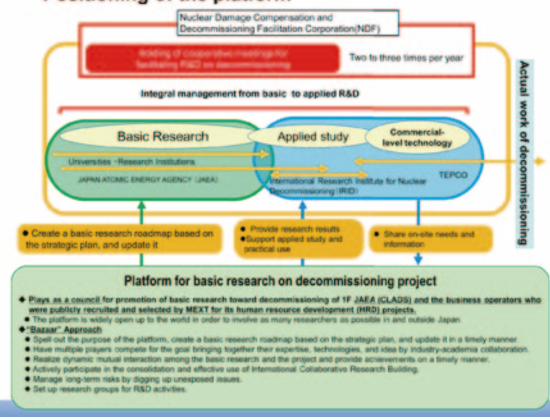
Consideration and investigation of safe disposal systems

- Evaluation for the current established/considered disposal concepts in Japan
- Consideration of new and modified disposal concepts with consideration of foreign concepts and/or those functions

10

Platform for basic research on decommissioning project

Positioning of the platform



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2.3 Session 1　Experiences of Waste Management in Countries

2.3.1 Legacy Waste Management in the United States

Roger SEITZ
Savannah River National Laboratory; roger.seitz@srnl.doe.gov

1. Introduction

Remediation efforts result in large amounts of waste, which has resulted in the need for a comprehensive waste management strategy that includes multiple treatments, storage and disposal options. The options involve considerations for waste acceptance criteria (WAC) and appropriate characterization and a variety of waste forms, containers, transportation options, as well as regulatory approaches that allow for public input. Dedicated on-site disposal cells have been developed at multiple United States Department of Energy (USDOE) facilities and new disposal capacity is being developed at commercial low-level waste (LLW) disposal facilities (see Fig. 1). Facilities designed for disposal of hazardous wastes are also being used for disposal of low activity radioactive waste resulting from remediation efforts. A number of USDOE and commercial facilities are designed consistent with requirements for disposal of hazardous wastes, and thus, are acceptable for disposal of mixed wastes comprising radioactive and non-radioactive hazards. These facilities have a range of different disposal capacities up to more than 15 million tons of waste.

Fig. 1: Facilities with liners and leachate collection systems used for disposal of remediation wastes at the USDOE Hanford (left) and Idaho (right) sites, respectively. The Idaho facility has been used for disposal of reactor vessels from decommissioned test facilities on site as well as a variety of other remediation wastes. (Courtesy: USDOE)

2. Regulatory Approach

Decision-making for remediation and disposal of mixed waste is conducted through a United States Environmental Protection Agency (USEPA) regulatory process involving a rigorous approach for evaluation of alternatives. The approach is conducted with requirements for public involvement, including external regulatory oversight. The generally applied approach involves nine criteria in a structured framework for decision-making (see Table 1). This process has proven to be very effective in evaluating options and communicating the benefits of the selected alternatives, including on-site disposal cells at the major sites. The USEPA has a number of prescriptive requirements (e.g., disposal facility design, waste treatment) that can have advantages for consistency and public acceptance, but can also pose challenges for unique site-specific wastes. The prescriptive design for disposal facilities has been very successful for public acceptance. However, the prescriptive approach for waste treatment can lead to situations where the standard treatment may not be optimal for a specific waste and it is necessary to justify a different approach. When it is necessary to model the performance of some waste forms for a performance assessment, there is a need to develop parameters to support the modeling effort. This has been a challenge, because the detailed data for a performance assessment are not typically collected to support disposal of hazardous wastes.

Table 1: Based on nine criteria used for decision-making.

Regulatory
1. Protection of Human Health
2. Compliance with Federal and State Regulations
Optimization
3. Long-term effectiveness (e.g., WAC and safety assessment)
4. Reduction of toxicity, mobility or volume of contaminants (e.g., treatment, containers)
5. Short-term effectiveness (e.g., worker protection, transportation)
6. Implementability at the site (e.g., WAC)
7. Cost effectiveness
Stakeholders
8. Regulatory acceptance
9. Community acceptance

3. Addressing Challenging Wastes

Remediation has resulted in the generation of a wide variety of wastes that have required a strategic approach to manage. The vast majority of the wastes are routine and relatively straightforward, although disposal capacity is needed (e.g., soils, lightly contaminated debris). However, a variety of specific, more challenging wastes are also produced (e.g., reactor components, ion exchange resins, etc.). These more unique wastes lead to challenges for identification of appropriate treatment and disposal options and also may test the standard regulatory approaches. Solid secondary wastes like ion exchange resins illustrate a variety of strategic challenges associated with remediation wastes in the United States. The presentation includes some practical handling and treatment challenges as well as challenges to identify inputs and assumptions required to model the waste forms.

For efficient identification of research priorities, it is critical to integrate the performance assessment activities with the research teams that conduct research to support the analyses. Initial performance assessment modeling is used to identify assumptions and parameters that are important for the conclusions of the analysis. The researchers and performance assessment modelers work together to identify key areas of need. This provides a benefit for the efficiency of the process (i.e., research is focused on the significant needs) and it also helps the researchers provide justification for the testing and characterization work that they will be conducting.

4. Summary

When considering alternatives for treatment, storage and disposal, there are a number of factors that are addressed in the national strategy. It is a combination of regulatory and technical challenges. The regulatory challenges include: balancing prescriptive and performance based approaches, establishing WAC for treatment and disposal facilities, implementing decision approaches capable of considering multiple criteria, and allowing for a transparent approach with public involvement. Technical challenges will typically focus on the most challenging, unique wastes. The technical efforts must consider WAC for treatment, storage and disposal facilities; transportation requirements; requirements for waste form and containers. The regulatory and technical considerations should be linked to the overall plan for remediation to help identify critical assumptions that may impact the options for final disposal of waste. Given the large amounts of input and assumptions for modeling, it is important to integrate data collection and modeling efforts to help prioritize efforts on the data and assumptions that are most significant for the decision to be made.

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Legacy Waste Management in the United States

Roger Seitz
Environmental Restoration Technologies

Research Conference on Post-accident Waste Management Safety
Hosted by JAEA/CLADS
Iwaki, JAPAN

November 7, 2016

SRNL-MS-2016-00213

Savannah River Nuclear Solutions

Introduction

The United States Department of Energy – Office of Environmental Management (USDOE-EM) is responsible for the largest cleanup program in the world

- **Cleanup activities involve generation of large quantities of waste containing radionuclides and non-radiological hazards**
- **Robust multi-criteria decision-making process involving external regulation and input from stakeholders**

107 USDOE-EM sites - As of September 2012, cleanup has been completed at 90 of those sites (DOE Graphic)

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Contents

- **Waste Management Strategic Considerations**
- **D&D and Remediation Waste Disposal Examples**
- **Regulatory Approach**
- **Waste Management Strategy - Examples for Solid Secondary Waste**

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Waste Management Considerations

National Strategy for technical and regulatory, a number of considerations, for example:

- **Waste Hierarchy**
  - Prevent → Reduce → Reuse → Recycle → Dispose
- **Waste Acceptance Criteria (store, transport, treat, dispose)?**
- **Volume of waste (including remediation wastes)?**
- **Characterization, segregation, treatment decisions (keys for cost estimating, disposal capacity, policies)?**
  - Treatment can increase volume & decrease concentration or vice versa, Potential for improper characterization?
- **Packaging and Transportation requirements ?**
- **Plans for Spent Fuel (on-site, off-site)**
- **Plans for reactor vessels and reactor components**
  - Potential Intermediate-level waste/Greater than Class C in USA ? Disposal and/or storage needed?

Photos Courtesy USDOE

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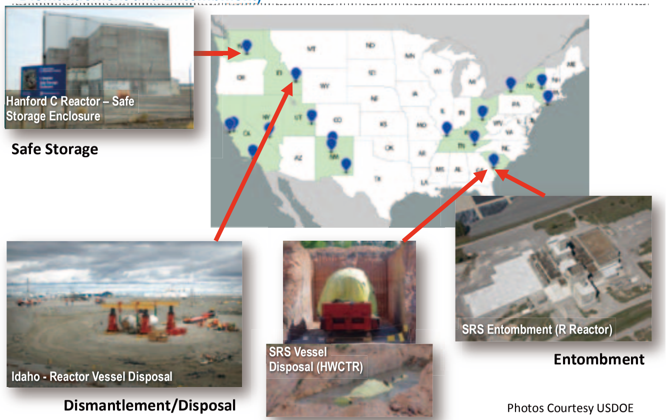
All Three D&D Options Applied in USDOE-EM (Safe Storage, Immediate Dismantlement and Entombment)



Photos Courtesy USDOE

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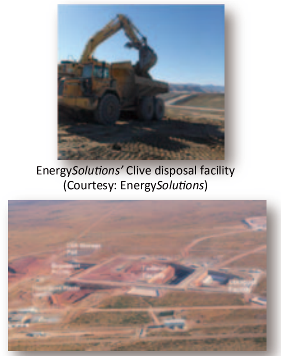
Disposal Options for Remediation and Decommissioning Wastes

- USDOE-EM has the option of developing on-site disposal cells, disposal at the Nevada National Security Site (NNSS) or using commercial disposal facilities
- On-site disposal has been selected as the preferred alternative for much of the waste (can be combined with off-site disposal of some waste)
- Transuranic (Intermediate-level) waste is disposed at the Waste Isolation Pilot Plant
- Spent-Fuel and High-Level Waste currently being stored



Energy*Solutions'* Clive disposal facility (Courtesy: Energy*Solutions*)

Waste Control Specialists Texas disposal facility (Courtesy: Waste Control Specialists)

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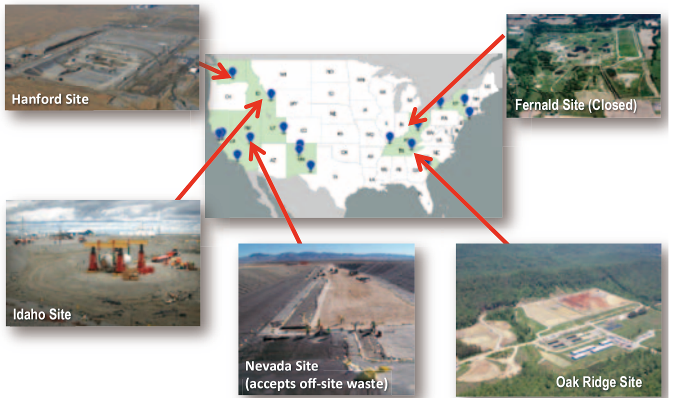
Examples of On-Site Disposal of Cleanup Waste (USDOE)



Photos Courtesy USDOE

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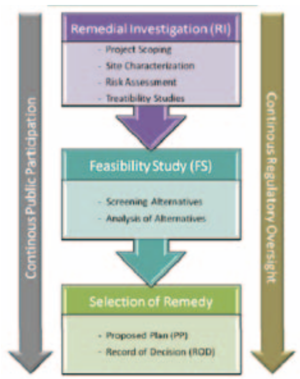
Regulatory Context - Decision-Making Framework

- Most on-site disposal facilities for cleanup waste are being developed under a US Environmental Protection Agency (USEPA) Regulatory Process
- Decision-making through formal process with continuous involvement of USDOE, USEPA, and State regulators and the public



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Key Elements of USEPA Process Applied to Remediation Waste

- Risk goals rather than constraints
- Design standards for disposal facilities and treatment standards for hazardous waste forms (industrial waste) - "Prescriptive" regulation
- Modeling and characterization efforts to support decision-making
- Must meet USEPA requirements and USDOE disposal requirements (USDOE and external regulator review processes are often conducted independently)
- Considers cleanup alternatives
- Nine criteria - Quantitative and qualitative assessment of potential impacts of different alternatives
- Following action, routine reviews (~5 year) are conducted to assess effectiveness of solution

Nine Criteria
REGULATORY
Protect health and environment
Comply with Federal and State regulations
OPTIMIZATION
Long-term effectiveness (WAC)
Short-term effectiveness (workers, transportation)
Implementability (WAC, siting)
Cost-effectiveness
INTERESTED PARTIES
Regulatory acceptance
Community acceptance

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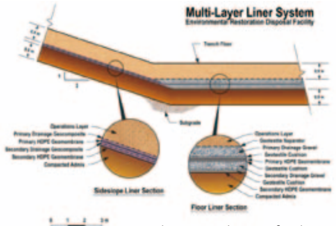
Prescriptive Regulation – Advantages and Disadvantages

- Disposal facilities for cleanup waste are designed to meet US EPA standards for hazardous waste disposal to address the non-radioactive hazards ("Design-based standard")
- Use of standardized and accepted design helps to build public confidence, but introduces challenges if it is necessary to conduct long-term safety assessment
- USEPA prescribes specific treatment approaches for hazardous wastes
- Standard approaches are not always the best option for special cases, need flexibility to consider optimal solution (e.g., grouting of ion exchange resins)



Liner design at the Hanford Site

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Solid Secondary Waste - Background

- Examples of solid secondary wastes and contaminants
  - HEPA Filters (e.g., Tc-99, Cr, I-129)
  - Ion Exchange Resins (e.g., Cs-137, I-129, Tc-99, Cr)
  - Activated Carbon Beds (e.g., I-129, Hg)
  - Silver Mordenite (e.g., I-129, Silver)
  - Miscellaneous Debris

Encapsulation

Solidification

- Prescriptive treatment approaches in regulations for hazardous waste ("debris" and "non-debris") – e.g., encapsulation and solidification/ stabilization using cementitious materials
- Safety assessment models for radioactive waste require numerous inputs (e.g., hydraulic conductivity, moisture characteristics, diffusion coefficients), but limited data are available
- Identification of priorities is important to focus research efforts

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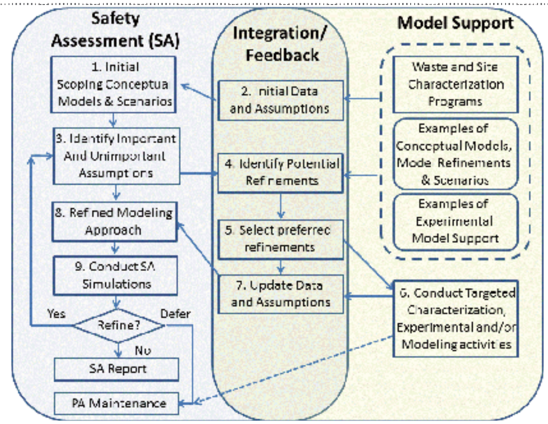
Link Research to Needs of the Safety Assessment (SA)



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Linking initial SA results with prioritization of data collection

Waste Form	Initial PA Assumptions	Potential Data Needs	Vulnerabilities
Filter Media *I-129 release is a key contributor in first iteration of PA*	Minimal credit for K_d and diffusion from filter media Encapsulated in oxidized material with paste properties	Properties of actual encapsulation material, including redox Diffusion coefficients and K_d (Iodine, ?) for encapsulation material	Properties not based on actual formulations Oxidizing conditions increase K_d for I-129
Ion Exchange Resin *Releases from resin expected to be limited, but need data to confirm*	Oxidized resin with no retention of contaminants Stabilized, waste form has properties of oxidized mortar K_d of waste form based on weighted average of K_d in waste & grout	Confirm material properties of final waste form Redox of final waste form K_d of resin and diffusion coefficients in waste form	Properties not based on actual formulations Properties potentially optimistic for unstable resin Oxidizing conditions increase K_d for I-129
Ag-mordenite *Release rates are expected to be low, even with pessimistic K_d*	K_d at low end of range based on testing with water representative of cementitious material Stabilized waste form has properties of oxidized, mortar K_d of waste form based on weighted average of K_d in waste and grout	Confirm material properties of final waste form Redox of final waste form K_d of waste	Defensibility of K_d

Note: This table focuses on vulnerabilities for defensibility, the conservative-bias that is built-in by some of the assumptions is not addressed, but could be addressed as needed in testing.

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Organic Ion Exchange Resins – Considerations and Options

- Prescriptive treatment standard for "hazardous" resins (solidification/ stabilization in cementitious materials, thermal)
- Resins dewatered and sent for treatment in hydrogen form – oxidizing conditions (favorable for I-129, not favorable for Tc-99)
- Organic resins expand posing challenge for cementitious materials (hydraulic properties of solidified matrix may not be maintained)
- Grout formulation/waste form pretreatment needs to address potential expansion and provide sufficient long-term performance (Key question: Safety assessment can help determine what is sufficient?)
- Example Options (Place in container – no pretreatment, Pretreatment to swell resin (Na or Ca form) and solidify, thermal treatment, etc.)

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Strategic Considerations Tied to Road Map for Cleanup

- How well are the wastes characterized (what is the range of uncertainty)? How could uncertainty impact transportation, treatment, storage, disposal options?
- Is treatment increasing or decreasing concentration and volume of contaminants? Will there be limitations of effluents from treatment facility?
  - Disposal facility capacity and WAC considerations (will a more robust facility be needed for smaller volume of waste or is it more effective to dispose of larger volumes?), Will it be more difficult to store and transport a higher concentration waste?
- Would prescriptive regulations be helpful or limit availability of good options? Could solidification of the waste reduce performance (e.g., activated carbon or resin stabilization in cementitious material)
- If Tc-99 and I-129 are both present, would oxidizing or reducing conditions be more favorable? Need input from safety assessment.
- How to address "evolution" of a waste form over time (improve or degradation of properties) and how to address in safety assessment? (e.g., fracture behavior in unsaturated conditions)

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Conclusions

- Variety of different approaches for D&D and remediation waste management in the USDOE-EM Complex
- Decision-making is based on multi-criteria approach with engagement of external regulators and public involvement
- Prescriptive regulations can be helpful (public perception, consistency), but can also limit optimization for special cases
- Safety Assessment helps to identify priority areas for research activities (process helps to provide sound basis for need for research)
- Many competing factors when identifying optimal waste management strategy, research inputs can help to identify and assess critical assumptions that could result in changes in the strategy

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2.3.2 Waste Management for Decommissioning in the UK

Robin ORR
UK National Nuclear Laboratory; robin.orr@nnl.co.uk

The United Kingdom was one of the first countries to adopt nuclear energy as a source of civil electricity production. Indeed, the UK Calder Hall nuclear reactors were, in 1956, the world's first to generate electricity on an industrial scale. Ultimately 26 Magnox reactor units (now closed), 14 Advanced Gas Reactor (AGR) units and one Pressurized Water Reactor (PWR) were built in the UK. To support the development and operation of the UK nuclear power station fleet, research/prototype reactors and research facilities were built as well as plants for enrichment of uranium, fuel fabrication, spent fuel storage and spent fuel reprocessing.

Several research/prototype reactors and other plants have already been successfully decommissioned in the UK but many nuclear facilities remain. The Sellafield site is the largest nuclear site in the UK and represents its largest decommissioning challenge. It is expected that decommissioning and site restoration will continue beyond the year 2100 [1]. Decommissioning the facilities will generate a range of wastes, the majority of which will be classed as Low Level Waste (LLW), destined for near-surface disposal [2]. However, particular challenges during decommissioning arise from the retrieval and management of Intermediate Level Wastes (ILW) and nuclear materials currently stored in legacy facilities.

In the early 1980s the UK initiated a programme to develop treatment processes for solid and slurry ILW from spent fuel reprocessing. A phased development approach was established, from laboratory scale experiments to commissioned plant operation. From this work, four waste cementation plants were built at Sellafield to process operational arising from reprocessing. These plants have now been operating for over 25 years and produced more than 50,000 waste packages. Considerable experience has been established in the UK on the cementation of wastes, development of criteria for disposability [3] and the demonstration of wasteform performance. This experience has been built on to manage decommissioning wastes.

The highest priority tasks at Sellafield are decommissioning the high-hazard legacy plants from historic fuel reprocessing operations that currently store higher activity wastes and materials [1]. Unlike feeds to current waste plants at Sellafield, the ILW from the legacy plants will be a heterogeneous mix of materials, exhibiting a range of hazards, and will vary over the waste retrieval period. In addition, there are significant practical constraints on the engineering that can be implemented to mitigate the hazards from these wastes. These challenges require a more flexible and pragmatic approach than in previous waste treatment processes.

This presentation will provide a brief overview of the UK framework for the management of radioactive wastes. Experience gained from the development of waste treatment and cement immobilization of reprocessing wastes will be summarized before discussing the different challenges posed by ILW retrieved from UK legacy facilities during decommissioning. Examples of strategies for wastes at the UK Sellafield site and associated research will be discussed.

References

[1] Nuclear Decommissioning Authority, Strategy Effective from April 2016.

[2] Nuclear Decommissioning Authority, Radioactive Wastes in the UK: A summary of the 2013 Inventory.
[3] Geological Disposal Generic Waste Package Specification, Radioactive Waste Management Limited.


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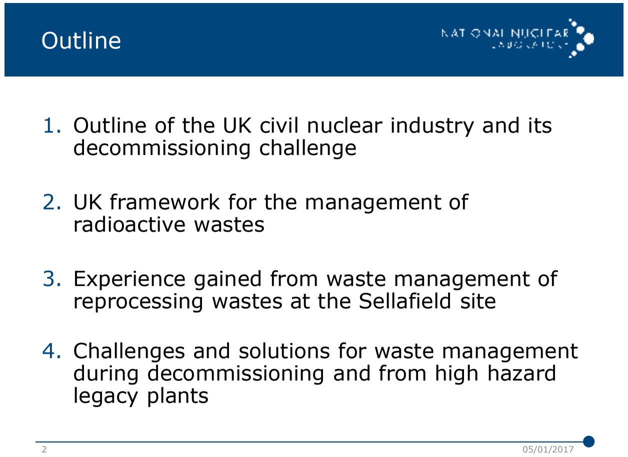
Waste management for decommissioning in the UK
Dr Robin Orr
JAEA/CLADS Research Conference on Post-accident Waste Management Safety
November 7th 2016, Iwaki, Japan


Outline
1. Outline of the UK civil nuclear industry and its decommissioning challenge
2. UK framework for the management of radioactive wastes
3. Experience gained from waste management of reprocessing wastes at the Sellafield site
4. Challenges and solutions for waste management during decommissioning and from high hazard legacy plants
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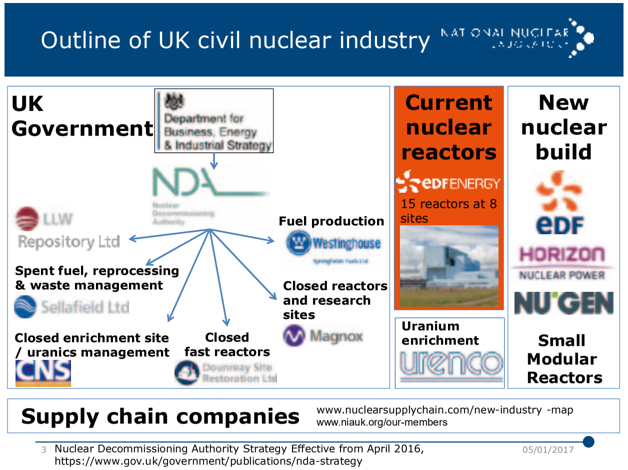
Outline of UK civil nuclear industry
UK Government
Department for Business, Energy & Industrial Strategy
NDA
LLW Repository Ltd
Fuel production
Westinghouse
Spent fuel, reprocessing & waste management
Sellafield Ltd
Closed reactors and research sites
Magnox
Closed enrichment site / uranics management
CNS
Closed fast reactors
Dounreay Site Restoration Ltd
Current nuclear reactors
EDF ENERGY
15 reactors at 8 sites
Uranium enrichment
urenco
New nuclear build
EDF
HORIZON NUCLEAR POWER
NUGEN
Small Modular Reactors
Supply chain companies
www.nuclearsupplychain.com/new-industry-map
www.niauk.org/our-members
3 Nuclear Decommissioning Authority Strategy Effective from April 2016, https://www.gov.uk/government/publications/nda-strategy
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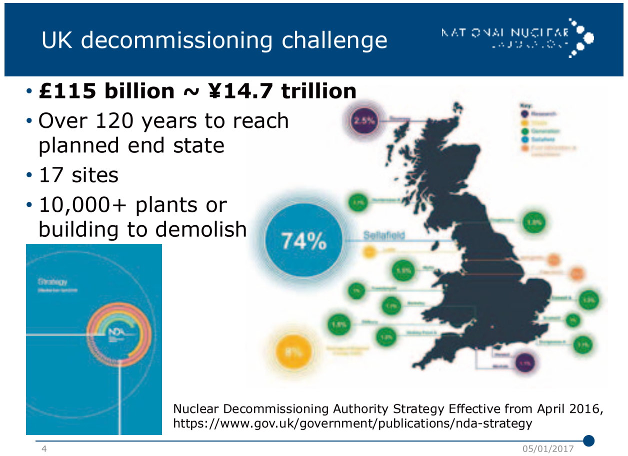
UK decommissioning challenge
• £115 billion ~ ¥14.7 trillion
• Over 120 years to reach planned end state
• 17 sites
• 10,000+ plants or building to demolish
74%
Sellafield
Nuclear Decommissioning Authority Strategy Effective from April 2016, https://www.gov.uk/government/publications/nda-strategy
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Overview of the Sellafield site

- 7 reactors (closed)
- Fuel storage ponds
- 3 reprocessing plants (1 closed)
- Pu finishing lines & PuO_2 stores
- MOX fuel fabrication plants (closed)
- 4 operational waste cementation plants
- Highly Active Liquor (HAL) storage
- HAL vitrification plant
- Multiple waste stores (95% of UK radioactive waste)
- 3 Effluent treatment plants
- Metal decontamination plant
- Research and analytical labs

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UK radioactive waste management framework

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UK framework for management of radioactive wastes

Key principles:
1. Radioactive wastes must not be unnecessarily created
2. Waste must be safely and appropriately managed and treated (passive storage)
3. Waste must be safely disposed of at appropriate times and in appropriate ways

Waste owners are responsible for:
- Developing and implementing waste management
- Costs of management, disposal and regulation

UK regulation of radioactive waste management

Environment Agency — **Environmental protection**

ONR Office for Nuclear Regulation — **Safety and security**

7 Basic principles of radioactive waste management, www.onr.org.uk/wastemanagement.htm 05/01/2017

UK and Japanese classification of radioactive wastes

United Kingdom	Examples	Japan		
High Level Waste (HLW) (<1vol%) Heat generating	Vitrified Highly Active Liquor	**High Level Waste**		
Intermediate Level Waste (ILW) (6vol%)	Pu-contaminated materials (PCM), fuel cladding, effluent sludge	**Transuranic wastes** (TRU)		**Low Level Waste**
	Reactor components, irradiated graphite	Relatively higher radioactive waste	**Reactor wastes**	**Low Level Waste**
Low Level Waste (LLW) (94vol%) <4 GBq/te α <12 GBq/te β/γ	Operation wastes: paper, plastic, tissue, clothing, wood.	Relatively lower radioactive waste	**Reactor wastes**	**Low Level Waste**
Very Low Level Waste (VLLW) <400 kBq per 0.1 m³ (low volume) 4 MBq/te (high volume)	Decommissioning waste: rubble, soil, metal plant equipment	Very low level waste	**Reactor wastes**	**Low Level Waste**
	Used equipment, sludge	Uranium waste		**Low Level Waste**

UK disposal of radioactive wastes: Low Level Waste

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LLW

UK policy to rigorously apply waste hierarchy and maximise alternative disposal routes to the LLW repository through appropriate characterisation, segregation and treatment

VLLW – High volume LLW disposal at specified landfill sites

UK Strategy for the Management of Solid Low Level Waste from the Nuclear Industry, February 2016, www.gov.uk/government/consultations/consultation-on-an-update-of-the-uk-strategy-for-the-management-of-solid-low-level-radioactive-waste-from-the-nuclear-industry

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UK disposal of radioactive wastes: ILW and HLW

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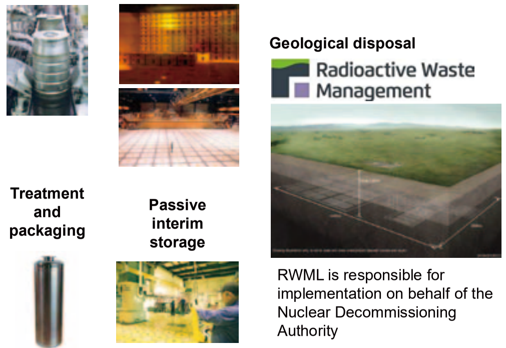
ILW

HLW



Treatment and packaging

Passive interim storage

Geological disposal

Radioactive Waste Management

RWML is responsible for implementation on behalf of the Nuclear Decommissioning Authority

10 Department of Energy and Climate Change, Implementing Geological Disposal White Paper, 2014. 05/01/2017
Department of Energy and Climate Change is now part of Department for Business, Industry and Energy Strategy

UK waste package specification

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- RWML have developed detailed guidance on ILW, HLW and uranium wastes packaging:
  - ★ Provides a definition of expected requirements ahead of site construction
  - ★ Provides a basis for assessment of waste management and allows early packaging
  - ★ Promotes standardisation
  - ★ Will evolve into Waste Acceptance Criteria
  - ★ Permits scrutiny of plans to implement disposal

11 Packing of radioactive waste: specification and guidance, www.gov.uk/guidance/generic-waste-package-specification 05/01/2017

UK waste package specification guidance

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- Waste package identification systems
- Data recording, safeguards and quality management
- Waste container designs:

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UK ILW waste package specification: Waste package requirements

- Waste package
  wasteform + container
  - Activity content
  - Maximum gross mass
  - External dose rate
  - Heat output
  - Surface contamination
  - Criticality safety
  - Gas generation
  - Impact accident response
  - Fire accident response
- Waste container:
  - Dimensions & features
  - Handling features
  - Stackability
  - Identification
- Wasteform
  waste + encapsulant
  - Immobilisation of radionuclides and particulates
  - Exclusion of free liquids
  - Gas generation with limits
    - Radiolysis
    - Corrosion
    - Microbial
    - Chemical degradation
  - Matrix permeable to gas generated
  - Minimise radionuclide mobility
  - Homogeneity/heterogeneity
  - Thermal conductivity
  - Chemical containment
  - Mechanical strength
  - Voids minimised
  - Wasteform evolution, e.g.:
    - Corrosion
    - Encapsulant degradation
    - Radiation damage

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A systematic phased development approach & UK experience of reprocessing wastes at Sellafield

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UK phased approach waste immobilisation development

- UK has a well developed phased approach to waste immobilisation process development
  - Phase 1: Waste characterisation, simulant specification, summary of immobilisation options
  - Phase 2: Selection of preferred immobilisation option
  - Phase 3: Detailed assessment of waste package properties
  - Phase 4: Assessment of waste/process variability
- Phases align with regulatory process and Waste Package Specification requirements

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Phase 1: Waste characterisation, simulant specification & immobilisation options

- Waste Characterisation
  - Waste history
  - Volume arising
  - Physical and chemical properties
  - Variability of the waste properties
- Define waste simulants
  - More than one simulant may be required
- Define immobilisation options
  - Conventional inorganic cements
  - "Alternative" inorganic cements
  - Polymer Modified cements
  - Polymers
  - Bitumen
  - Glass (vitrification)
  - Low melting point metal
  - Ceramics
- Assess immobilisation options ("desktop assessment") and select immobilisation options for further work

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Phase 2: Selection of preferred immobilisation option

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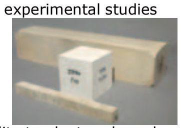
- Demonstrate compatibility of encapsulant with the waste stream
- Typically small-scale (0.1 L to 30 L) inactive/active experimental studies
  - ★ Chemical interactions between waste and encapsulant
  - ★ Physical/mechanical stability
  - ★ Thermal stability
  - ★ Radiation stability
  - ★ Radionuclide leaching



- Process engineering assessment to identify scalability to plant scale and plant requirements
- Assessment of appropriate container type
- Selection of immobilisation process against broad criteria, e.g.:
  - Technical readiness
  - Waste package evolution
  - Impact/Fire performance
  - Total number of packages
  - Radionuclide immobilisation
  - Plant throughput
  - Plant operability
  - Plant footprint
  - Safety of operation
  - Process complexity
  - Secondary wastes
  - Security of materials supply
  - Overall cost

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Phase 3: Detailed assessment of waste package properties

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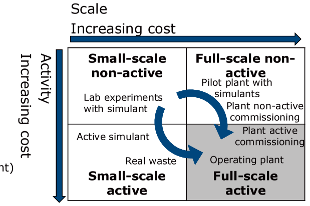
- Detailed practical studies to demonstrate waste package performance of selected process and formulation
  - ★ **Physical properties**
    - ❖ Homogeneity
    - ❖ Density/voidage
    - ❖ Permeability
  - ★ **Physical/chemical stability**
    - ❖ Strength
    - ❖ Dimensional stability
    - ❖ Impact strength
    - ❖ Corrosion of metals (gas generation /dimensional stability)
  - ★ **Thermal properties**
    - ❖ Freeze-thaw
    - ❖ High temperature stability (Fire accident)
    - ❖ Thermal conductivity
  - ★ **Radiation stability**
    - ❖ α and β/γ stability
    - ❖ Gas generation G(H_2)
  - ★ **Define processing parameters**



- Timescales of some studies extend to 360 days curing and beyond
- Experiments and modelling used to demonstrate performance against the Waste Package Specification requirements

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Phase 4: Assessment of waste/process variability

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- Establish waste/encapsulant "formulation envelope"
  - ★ Defines the composition range that achieves acceptable waste package performance
  - ★ Small/Full-scale studies that assess:
    - ❖ Range of waste composition
    - ❖ Range of encapsulant composition
    - ❖ Waste and encapsulant temperatures

- Define the plant operating envelope:
  - ★ Trials using full-scale test rig and (non-active) and commissioned plant to optimise process conditions
  - ★ Defines the range of all process parameters that achieve acceptable product, taking into account process tolerances (uncertainties)
  - ★ Important to ensure waste package quality

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Reprocessing ILWs at Sellafield: Fuel cladding/assembly components

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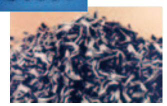
Magnox fuel cladding ("swarf")

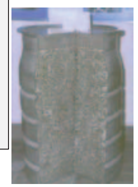
Oxide fuel cladding (stainless steel/ zircaloy)



Vibro-grouting

Waste infilled with fluid slag/Portland cement grout

Vibration used to promote infilling and minimise voids

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Reprocessing ILWs at Sellafield: Slurry wastes

THORP dissolver residues (insoluble fission products/fuel)

THORP dissolver off gas ($BaCO_3$)

Ferric hydroxide floc (effluent actinide removal)

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Reprocessing ILW immobilisation development experience

- Conventional inorganic cements selected:
  - ★High pH give good retention of many radionuclides
  - ★High surface area for adsorption of species
  - ★Low heat of hydration
  - ★High fluidity and long working time
  - ★Bleed free and low water content when hardened
  - ★Known physical, mechanical and thermal properties
  - ★Generally good radiation tolerance/stability
  - ★Often good chemical/physical compatibility with wastes
  - ★Good fire accident performance
  - ★Good impact performance
  - ★Low cost
- Cements now widely used in UK ILW management, but other materials (e.g. polymers) have also been developed and implemented

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Reprocessing ILW immobilisation experience

- Well defined, homogenous waste streams with high levels of characterisation
- High level of waste composition monitoring/control at feeder plant
- Waste-encapsulant interactions relatively easily managed, for example:
  - ★**Magnox cladding**
    - ❖Corrosion of Magnox and uranium
      - ➢Wasteform expansion
      - ➢$H_2(g)$
  - ★**Ferric hydroxide floc**
    - ❖Reactions between ferric floc and cement
    - ❖Minor metal ion species in sludge influence mixing & cement hydration
    - ❖α-radiation stability initially a concern
- 4 cement encapsulation plants **>100 years operation**
- **>50,000** 500 L drum waste packages produced
- Additional waste feeds identified to extent usage of plants

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Decommissioning and Sellafield high-hazard facilities

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Sellafield high-hazard facilities: Magnox Swarf Storage Silos (MSSS)

- Originally built in the 1960s
- 22 silo compartments containing waste stored under water
- Wide range of waste types with little segregation, majority is from Magnox cladding
- Not designed with preparation for waste retrieval and decommissioning

Example waste materials

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Challenges for management of wastes retrieved from MSSS

- Heterogeneous waste
  - ★Varies within compartments
  - ★Total inventory reasonably well known from historic records
  - ★Limited knowledge of distribution
  - ★Sampling has been undertaken but not representative
- Limited characterisation possible at the time of retrieval
- Specific waste hazards to be addressed
  - ★**H_2 production**
    - ❖Corrosion of wastes (Magnox, uranium, aluminium, steels)
    - ❖Radiolysis
    - ❖Gas hold-up in sludge and release
  - ★**Dimensional stability**
    - ❖Corrosion of wastes
  - ★**Thermal stability**
  - ★**Reactive uranium corrosion products**

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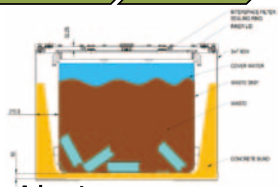
Strategy for MSSS ILW

Retrieval → Interim passive storage ~50+ years → Treatment / Conditioning → Geological Disposal

- **3 m³ standard ILW box**
  - ★Double skinned stainless steel
  - ★Integrated filters in lid
  - ★Concrete bund
- **Waste interim stored as-retrieved**
- **Conditioning**
  - ★Assume liquor removal and entombed in cement
  - ★Future technologies not foreclosed



- **Advantages**
  - ★Simpler process reduces risks
  - ★Retrievals decoupled from immediate waste conditioning
  - ★Only waste retrieval is rate limiting
  - ★Accelerated hazard reduction
  - ★Reduced costs

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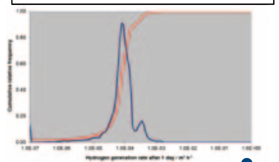
Underpinning retrieval, interim storage and conditioning

- Uncertainty predicting waste package behaviour dominated by inventory
- Statistical approach adopted to predict likely box inventory and distribution of box properties (H_2, expansion)
- Ensures proportionate management of low frequency events

Key research areas

1. Modelling water loss
2. H_2 management
   - ❖ Corrosion rates
   - ❖ Radiolysis
   - ❖ Diffusion from box
   - ❖ Store H_2 levels
3. Waste expansion
   - ❖ Corrosion
   - ❖ Gas hold up
4. Thermal stability
5. Container & filter integrity



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Features of decommissioning and high hazard facility ILW management

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- Increased focus on overall hazard reduction
- Fit for purpose solutions
  - ★Excessive over engineering is counter productive to overall hazard reduction
- Greater emphasis on waste package performance requirements during interim storage and disposal
  - ★Encapsulation may not be required for some wastes
  - ★**Packages need to meet the appropriate requirements at the appropriate time**

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Summary

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1. UK has 120+ year, £115 billion nuclear decommissioning programme
2. UK ILW and HLW will be consigned to geological disposal
3. Detailed guidance established on disposability of ILW and HLW but evolving to optimise resources as decommissioning requirements emerge
4. Flexible approaches to waste characterisation and management are being implemented in the UK for decommissioning
5. Decommissioning requires fit for purpose solutions and emphasis on overall waste package performance

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Acknowledgements

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Sellafield Ltd

Thank you for your attention

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2.3.3 Plan and Implementation of Nuclear Waste Management in Finland

Kari RASILAINEN
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1. Introduction

Finnish nuclear waste management programme is today globally the most advanced one vis a vis low and intermediate level waste and spent nuclear fuel management. The plans and schedules developed over 30 years ago are implemented today with only minor modifications. The features of Finnish success story cover e.g. coupling scientific research, decision making, regulatory needs, and implementor needs in the programme. The well-tested practices in siting a nuclear waste repository, and related site investigations, may be of interest in other countries as well, although one cannot directly transfer methods or approaches from country to country. The characteristic Finnish approach has been to step wise go deeper and deeper into site investigation and disposal concept development using subsequent safety assessments as measure of understanding.

2. Low and Intermediate Level Waste Management

Currently there are two operating licensed low and intermediate level waste repositories in Finland, located at the operating NPP sites Loviisa and Olkiluoto. They are geologic repositories at the depth of around 100 m in crystalline bedrock. There are well-developed plans to extend the repositories in order to dispose of future decommissioning wastes of the respective NPPs.

3. High Level Spent Nuclear Fuel Management

Finland will probably be the first country to license a deep geological repository for spent nuclear fuel. The repository will most likely be based on the KBS-3V concept, with V standing for vertical, meaning that individual disposal canisters containing the spent fuel bundles will be disposed of in individual vertical deposition holes, one canister per hole. The systematic programme for spent fuel disposal has so far lasted over 30 years, see Fig.1.

4. Costs of Nuclear Waste Management

The total costs of all nuclear waste management arising from reactors currently running or under construction was estimated to be 6,5 B€ in 2012, with spent fuel management covering for 3,5 B€. This estimate covers the management of operating waste, spent fuel, and decommissioning waste. Cost estimate for the future management of currently existing (end of 2015) nuclear waste is around 2,45 B€ (no discounting). The money for nuclear waste management is collected from nuclear waste producers into the State Nuclear Waste Management Fund with the idea that at all times there is enough money to manage the currently existing nuclear waste even if the nuclear waste producers would go bankrupt.

Reference

[1] Posiva, 2013. Safety Case for the Disposal of Spent Nuclear Fuel at Olkiluoto - Models and Data for the Repository System 2012, Report Posiva 2013-01, p. 49.

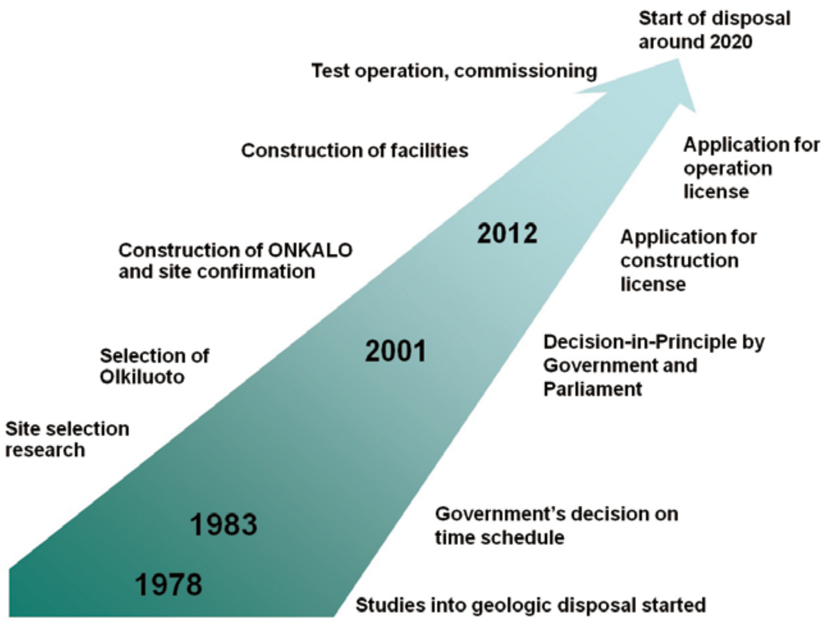


Fig.1. Timeline of spent fuel disposal in Finland [1].

PLAN AND IMPLEMENTATION OF NUCLEAR WASTE MANAGEMENT IN FINLAND

Dr. Kari Rasilainen, Principal Scientist, Research Team Leader
VTT Technical Research Centre of Finland Ltd

Nuclear Waste Management – Pioneering Solutions from Finland 1

PRESENTATION OUTLINE

- Background: Nuclear energy in Finland
- Nuclear waste management policy
- Nuclear Waste Management (NWM) Experiences
  - Low and Intermediate Level Waste (LILW)
  - High Level Waste - Deep Geological Repository (DGR)
- NWM cost estimate in Finland
- Summary

Nuclear Waste Management – Pioneering Solutions from Finland 2

NUCLEAR POWER PLANTS IN FINLAND

Four operational reactors (2 BWRs, 2 VVERs)

- Commissioned 1977-80
- Producing 70TWh (2013)
- Operating at 92% energy capacity factor (world-leader efficiency)

Fifth reactor Olkiluoto 3 (1600 MW, EPR) under construction (2005-)

Sixth reactor Hanhikivi 1 entering construction stage (CLA submitted June 2015)

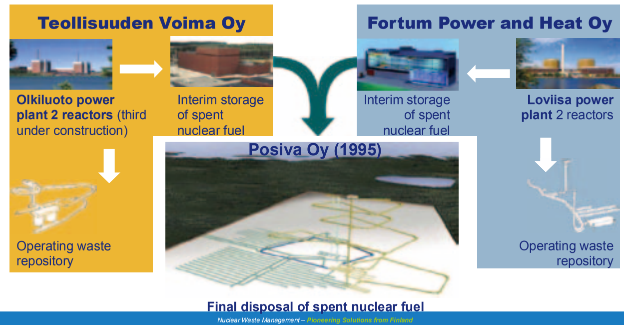
Nuclear Waste Management – Pioneering Solutions from Finland 3

EVOLUTION OF NUCLEAR WASTE MANAGEMENT POLICY

- Government decision in 1978
  - Basic principles
    - Geological disposal of all nuclear waste
- Government decision in 1983
  - Primary objective: irrecoverable transfer of spent fuel abroad
  - Secondary objective: preparedness for final disposal in Finland
  - Schedules of the research, investigation and planning
    - Schedules hold by and large still today
- Amendment of Nuclear Energy Act of 1994
  - Permanent disposal in Finland, no export or import of spent fuel
  - Disposal option is less expensive and complex than reprocessing-recycling
  - Inadequate reliance on foreign or international waste management solutions

Nuclear Waste Management – Pioneering Solutions from Finland 4

FINNISH NWM ORGANIZATION



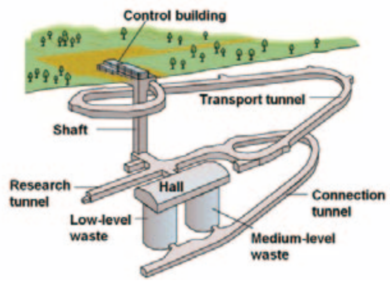
LOW AND INTERMEDIATE LEVEL WASTE

OPERATING WASTE REPOSITORY: OLKILUOTO

Site investigations
1980 - 1990

OPERATING WASTE REPOSITORY: OLKILUOTO

- Operating since 1992
- Licensed routine activity
- Safety cases updated every 15 years
- Depth level 70 – 100 meters
- Two silos: diameter 24 meters and height 34 meters
- LILW conditioning with bituminisation
- Extensions planned for decommissioning waste
- Small-user RW disposal

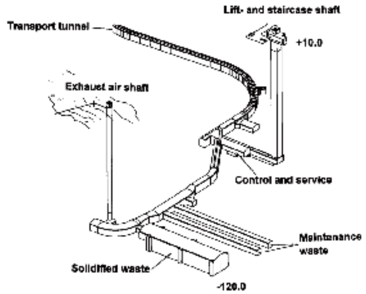


OPERATING WASTE REPOSITORY: LOVIISA

Nuclear Waste Management – Pioneering Solutions from Finland 9

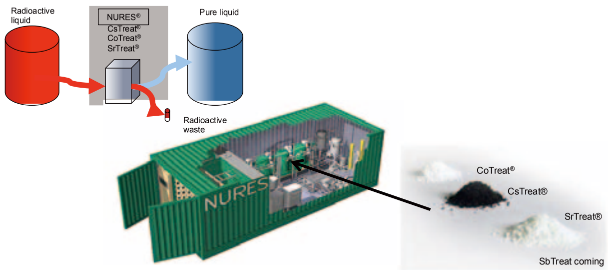
OPERATING WASTE REPOSITORY: LOVIISA

- Operating since 1997
- Licensed routine activity
- Safety cases updated every 15 years
- Depth level 110 – 120 meters
- Two storage tunnels, length 106 meters and a storage cavern length 84 meters
- LILW conditioning with cementation
- Nuclide removal system NURES®
- Extensions planned for decommissioning waste



Nuclear Waste Management – Pioneering Solutions from Finland 10

NURES®: HIGHLY SELECTIVE ION EXCHANGE MATERIALS



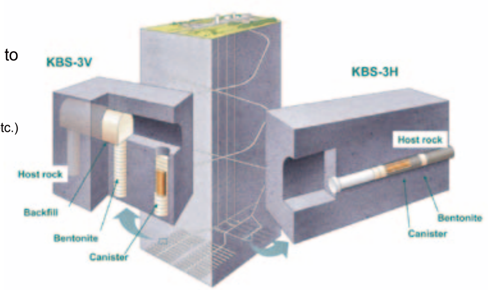
Nuclear Waste Management – Pioneering Solutions from Finland 11

HIGH LEVEL WASTE - DEEP GEOLOGICAL REPOSITORY

Nuclear Waste Management – Pioneering Solutions from Finland 12

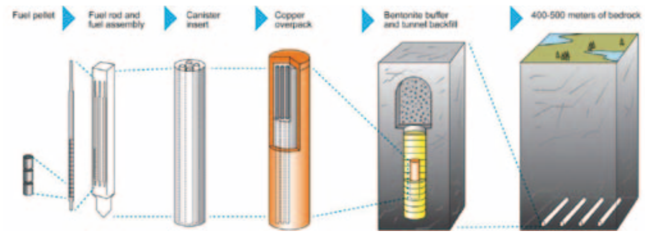
DISPOSAL CONCEPT KBS-3: VERTICAL OR HORIZONTAL DEPOSITION HOLES

Design always need to be tailored to local conditions
(geology, seismology, groundwater, biosphere, etc.)



Nuclear Waste Management – Pioneering Solutions from Finland 13

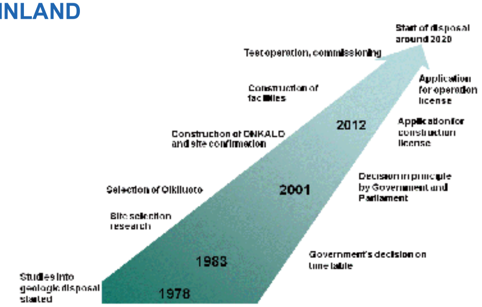
KBS-3 ENGINEERING BARRIER SYSTEM



Multi-barrier requirements: Compressibility+Hydraulic+Chemical+Mechanical

Nuclear Waste Management – Pioneering Solutions from Finland 14

TIMELINE OF SPENT FUEL DISPOSAL IN FINLAND



Source: Posiva Oy

Nuclear Waste Management – Pioneering Solutions from Finland 15

SPENT FUEL SITING APPROACH

N.B. Municipality has right of veto!

- Studies into geologic disposal started in 1978
  - 1978 -1982: General survey
  - 1983 -1985: Screening study of the entire area of Finland
  - 1986 -1992: Preliminary site investigations
  - 1993 - 2000: Detailed site investigations and an environmental impact assessment procedure was carried out for four sites: Romuvaara in Kuhmo, Kivetty in Äänekoski, Olkiluoto in Eurajoki, and Hästholmen in Loviisa.
- A site good enough was sought
  - Goodness was assessed in successive repository safety analyses
- December 2000, the Government made Decision-in-Principle, in May 2001, the Parliament ratified the DiP by 159 votes to 3.
  - Olkiluoto was selected

Nuclear Waste Management – Pioneering Solutions from Finland 16

SPENT FUEL LICENSING APPROACH

- 3-step process for major NWM facilities
  - Decision-in-Principle 2001
    - Olkiluoto was selected as site
  - Posiva submitted Construction License Application for Final Repository to the Government on 28 December 2012
    - Government granted the license 12.11.2015
  - Posiva plans to submit Operation License Application for Final Repository in 2020's
    - perhaps the first licensed spent fuel repository in the world?

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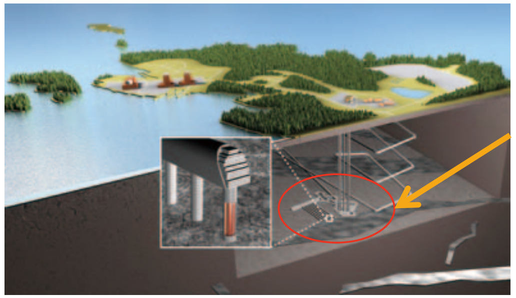
ONKALO UNDERGROUND RESEARCH LABORATORY

Access ramp

Nuclear Waste Management – Pioneering Solutions from Finland 18

OLKILUOTO DEEP GEOLOGICAL REPOSITORY



ONKALO – underground research laboratory

Deposition panels at the beginning

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NWM COST ESTIMATE IN FINLAND

Nuclear Waste Management – Pioneering Solutions from Finland 20

LATEST COST ESTIMATE FROM 2012 COVERING THE WHOLE NWM IN FINLAND

- All wastes
  - Operating waste
  - Spent nuclear fuel
  - Decommissioning waste
- Reactors running or under construction
  - Loviisa 1-2
  - Olkiluoto 1-3
- Estimated total cost 6,5 B€
  - Spent nuclear fuel 3,5 B€
- Cost estimate for the future management of currently existing (end of 2015) nuclear waste is 2,45 B€ (no discounting)

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SUMMARY

Nuclear Waste Management – Pioneering Solutions from Finland 22

FINNISH EXPERIENCE

- Finnish experience in NWM can be considered pioneering
  - NWM programme has progressed for 30 years
    - LILW and spent fuel
  - Coupling scientific research, decision making, regulatory needs, and implementor needs
  - Well-tested practices in siting
    - Site that is good enough has been sought
- Finnish approach has been to step wise go deeper and deeper into site investigation and disposal concept development using subsequent safety assessments as measure of understanding
  - Parallel iterative approach
    - Safety assessments 1985, 1992, 1996, 2000
- Our experience is at your disposal if you so wish

Nuclear Waste Management – Pioneering Solutions from Finland 23

2.4 Session 2 Research activities of Universities

2.4.1 Recovery and Volume-reduction Solidification of Radioactive Elements from Solid Waste

Kenji TAKESHITA
Advanced Nuclear Fuel Cycle Unit, Institute of Innovative Research, Tokyo Institute of Technology; takeshita.k.ab@m.titech.ac.jp

Large amounts of radioactive nuclides were released to the environment by the hydrogen explosion of the buildings of 1F NPP during March 11 to 15, 2011 and forest and farmland were contaminated by the fallout of radioactive elements. Then, large amounts of solid wastes such as soil, tree and rubble were generated. Their generation amounts in the 1F NPP site were about 82500 m^3 for cut tree and 155100 m^3 for soil and rubble. Since these wastes cause the exposure of workers to radiation, the radioactive elements has to be recovered from these wastes, concentrated and immobilized in stable materials. The decontaminated solid wastes are re-used for pavement materials, construction materials etc. As a result, the working environment in 1F NPP site will be improved and the nuclear decommissioning of 1F NPP will be promoted greatly. A part of the contaminated soil is stored with rubble, but the recovery of soil does not progress enough. The results of radiation measurement carried out by JAEA in March, 2015 suggest that high ^{137}Cs radiation of ten thousands to millions Bq/kg was observed in the soil near the reactor buildings. Those of ^{90}Sr in the sampled soils were low values in the range of 300 to 1500 Bq/kg [1]. According to the progress of the decommissioning of 1F NPP, treatment techniques of contaminated soil will be required. Especially, Cs and Sr are adsorbed strongly into 2:1type layered clay minerals, such as vermiculate and montmorillonite. The development of high decontamination technology is indispensable for the desorption of radioactive elements and the reuse of decontaminated soil.

Based on recognition of such current status, an integrated system for the recovery, concentration and immobilization of radioactive elements from environmental waste materials such as soil and tree is developed in this study. However, we have no experience on the treatment of large amounts of environmental wastes. In the current stage, most of radioactive elements (Cs and Sr) in contaminated soil is transferred to 2:1 type layered clay minerals such as vermiculate and montmorillonite and adsorbed strongly into them. Especially, the adsorption of Cs is very strong. Under these circumstances, we develop a new integrated system for the recovery, concentration and immobilization of radioactive elements in the contaminated soil. For the cut tree, the same system will be developed in parallel with the fundamental research on the chemical characteristics of radioactive elements in ash obtained by the combustion of cut tree and the adsorption of them into the ash. This is because the chemical conditions of radioactive elements in the ash are unknown and the fundamental knowledge is needed for the development of suitable system. Then, a new technique to remove Cs from the contaminated soil and highly volume-reduced waste is required for the economical and safe disposal. In this study, we introduce a new recovery and volume-reduction solidification process of these solid wastes.

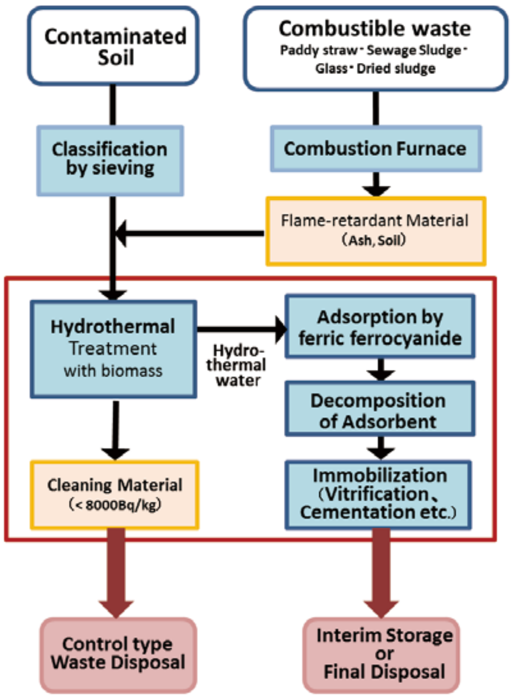

Fig.1 Removal process of Cs from contaminants of Fukushima Daiichi nuclear accident

Firstly we studied the removal process of Cs from contaminated soil, which consists of sieving of contaminated soil, hydrothermal decomposition of clay minerals, adsorption of Cs in ferric ferrocyanide, thermal decomposition of ferric ferrocyanide adsorbing Cs, leaching of Cs from the residue with water and vitrification of Cs, shown in Fig.1, Clay minerals that adsorbs strongly Cs was separated by the sieving of contaminated soil. Secondary Cs in clay minerals was recovered to water by a hydrothermal decomposition method with plant biomass under the conditions of 180-250ºC and 1.0-4.0 MPa [2]. Cellulose and hemicellulose in the plant biomass were decomposed to some organic acids and these organic acids in the subcritical water dissolved clay minerals, especially vermiculite which adsorbs Cs very strongly. More than 95% of Cs in vermiculite was released in hydrothermal water by a repetition of hydrothermal operation and concentrated by a typical Cs adsorbent, ferric ferrocyanide (Prussian Blue), which was decomposed in air at 300ºC [3]. Cs in the residue was recovered easily by pure water. Finally Cs was solidified and further concentrated in borosilicate glass. Mass and radioactive balances of the proposed process were estimated from the results of fundamental experiments. The volume of vitrified waste is reduced to about 1/20000 of that of contaminated soil [4]. These results suggest that both the removal of Cs from contaminated soil and the high volume-reduction of Cs waste can be attained by the introduction of the proposed process.

References

[1] TEPCO Homepage, www.tepco.co.jp/nu/fukushima-np/.../t120730_02-j.pdf

[2] I.Takatsuka, H.Takahashi, T.Fukuda, M.Harigai, Y.Inaba and K.Takeshita, "Fundamental Study on Cs Elution Technique from Vermiculite by Hydrothermal Treatment Using Cellulose-based Biomass (in Japanese)", *Journal of the Society for Remediation of Radioactive Contamination in the Environment*, **4(3)**, 231 – 238 (2016)

[3] J. Munezawa, N.Ito, R.Ogata, S.Inoue, Y.Inabaand M.Harigai, "Development of Cesium Recovery System for High Efficiency and Volume Reduction- Behavior of Cesium salt in Combustion Treatment of Prussian Blue – (in Japanese)", *Journal of the Society for Remediation of Radioactive Contamination in the Environment*, **4(1)**, 35-44 (2016)

[4] K. Takeshita, "Removal Process of Radioactive Cesium from Soil contaminated by Fukushima Daiichi Nuclear Accident", *Atalante 2016*, Keynote Lecture (Paper No.169), Le Corum, Montpellier, France (2016)

2.4.2 Rapid Analysis of Sr-90

Yoshitaka TAKAGAI [1], Makoto FURUKAWA [2], Yutaka KAMEO [3] and Katz SUZUKI [4]
[1] *Fukushima University; takagai@sss.fukushima-u.ac.jp*
[2] *PerkinElmer Japan Co., Ltd.; Makoto.Furukawa@PERKINELMER.COM*
[3] *Japan Atomic Energy Agency; kameo.yutaka@jaea.go.jp*
[4] *Japan Agency for Marine-Earth Science and Technology; katz@jamstec.go.jp*

Radioactive strontium-90 scattered by nuclear power plant accident was specifically quantified by conventional inductively coupled plasma quadrupole mass-spectrometry (ICP-QMS) preceded by on-line chelate column separation (based on lab-on-valve) and oxygen reaction (designated the cascade step). The proposed system overcomes the isobaric interference of ^{90}Zr, whose soil concentration exceeds that of ^{90}Sr by more than six orders of magnitude. In addition, the system requires no ultimate mass spectrometry or radioactive ^{90}Sr standards. The radioactive ^{90}Sr standard was replaced with the stable isotope ^{88}Sr as a pseudo-standard. The modified ICP-QMS system yielded a precise, reproducible sharp ^{90}Sr peak in the ICP-MS profile. The elution time of ^{90}Sr was highly reproducible (RSD = 0.5 %). After implementing the cascade-step, the detection limit (DL) was 0.28 Bq/L (0.06 ppq) in ^{90}Sr analysis and 2.9% of repeatability (10 ppq ^{90}Sr (n = 10)) within approximately 20 min. Solid sample was able to be applied after the treatment of microwave digestion system. The ^{90}Sr from environmental contaminated soil samples collected from areas at a distance of 10 and 20 km from the Fukushima daiichi nuclear power plant ranged from 52 Bq/kg to 73 Bq/kg, with no statistical difference between the proposed and general methods at 95% confidence level. The analytical technique has overcome the analytical problems in radiometry and conventional ICP-MS by using chemical system. The proposed method offers an attractive alternative use for ICP-other ionization mass spectrometry as an enrichment or purification step, thereby expanding the scope of ICP-mass-spectrometric analysis.

References

[1] Y. Takagai, M. Furukawa, Y. Kameo, and K. Suzuki, *Anal. Methods*, **6**, 355 (2014).
[2] M. Furukawa and Y. Takagai, *Anal. Chem.*, 88, 9397 (2016).

2.4.3 Activity of National Institute of Technology, Fukushima College

Shigekazu SUZUKI [1], Shiro JITSUKAWA [1], Seichi SATO [1] and Katsuhiro AOYAGI [1]
[1] *National Institute of Technology, Fukushima College; ssuzuki@fukushima-nct.ac.jp*

1. Introduction

The decommissioning of Fukushima Daiichi nuclear power station is a national challenge. Mid-level engineer training from a medium- to long-term perspective on decommissioning procedures is a pressing issue. The practical training of engineers is the theme of technical college as institution of higher education. The National Institute of Technology, Fukushima College (NIT, Fukushima College), is the closest national institution to the Fukushima Daiichi nuclear power station. Therefore, NIT, Fukushima College, holds a leading position among national colleges across the country in the field of decommissioning education.
Students of NIT Fukushima College have an interest in decommissioning technology. Some students choose to work in nuclear power-related companies. In order to meet the medium- and long-term theme of the region, the decommissioning basic research should be incorporated in graduate research and special studies carried out for human resource development.

2. The Establishment of Decommissioning Human Resource Development Consortium

NIT, Fukushima College had been submitted as a business representative of the public offering of Ministry of Education, Culture, Sports, Science and Technology. As a result, our application has been adopted for one year as a feasibility study. The application was entitled "Human resources training on the decommissioning of nuclear power plant, based on study for graduation --- interdisciplinary challenge from Fukushima".
In the wake of this adoption, "Decommissioning human resource development consortium" was established in 2015 March 17th by NIT, Fukushima College. In this council, the nationwide technical colleges, universities and research institutions pursue together human resource training in the medium- to long-term perspective. During the Council of the Secretariat in the Fukushima College, Takayuki Nakamura, president of Fukushima College, was appointed first chairman, and Tetsuji Choji, president of Kagoshima college, was appointed vice chairman. In April 2016, 34 technical colleges, 9 universities, 3 local government relationship and 6 private companies had joined the consortium.

3. Education Program on Decommissioning

This educational program is designed to correspond with the "human resource development and cooperation between higher education and research institutes in the medium-and-long term viewpoint" which is related to a governmental medium-and-long term roadmap for TEPCO's Fukushima Daiichi Nuclear Power Plant. This education program is constituted by three pillars; research and development, education and the facilities available (Japan Atomic Energy Agency Naraha Remote Technology Development Center). In 2015, classes on decommissioning have been offered in two new subjects. Classes carried out in the intensive course format are "decommissioning and Society" for the third year student and "decommissioning engineering" for the fourth year student.

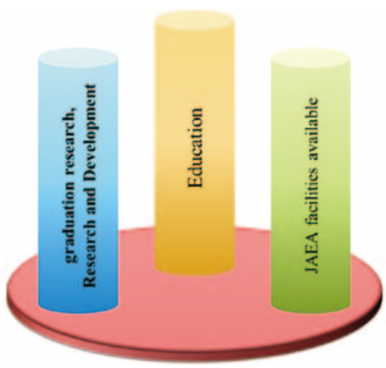

Fig. 1 Three pillars of education program on decommissioning

In 2016, internship programs for decommissioning and the two new lectures have been started. At the same time, the Technical College has promoted the cooperation in graduate research and special research. In 2017, by starting classes on decommissioning in first grade and fifth grade, education program on decommissioning will be completed. Internship is carried out at two locations of companies and city office. In the company learn the decommissioning technology. In the city office understand the duties as a government on decommissioning.

4. Overview of Creative Robot Contest for Decommissioning

In order to get the young generation interested in the decommissioning, education through the creations of robots is believed to be effective. In addition, a variety of practices, such as PBL education and active learning lesson are offered in each technical college. This Robocon is also carried out in PBL and active learning. If the main purpose is to interest students in decommissioning through the manufacturing of robots, it also aims at cultivating the students and contributing to their "creativity education", "problem-solving ability" and "ability to identify challenges".
Overview of the competition challenges are as follows.

1) Field
To select the playing field in each team from the two fields of the following assumes the reactor building.
a) Mock-up stairs
b) Standard step field

2) Field environment
a) Mock-up stairs and b) standard test field both have the following environment features:
• Complete darkness
• Impossibility to see the robot body directly while operating it remotely
• Radio wave does not reach because there is a thick wall of concrete

3) Challenges of robot
a) Mock-up stairs
• Carry a 5 kg luggage up to the second floor from the first floor and return to the original location after dropping off the luggage.
• Determine the objects located on the second floor. It should be noted that the location is undefined.
b) Standard step field
• Examine surrounding obstacles and area (such as area and irregularities).
• Determine objects located in the field. It should be noted that their locations are undefined.

There were submitted from all over the country in the national and public technical colleges of the 13 colleges 15 teams in this contest. Results of document screening, was observed participation of 15 teams.

RCWM2016
7 November, 2016
Iwaki

Activity of National Institute of Technology, Fukushima College

Human resources training on the decommissioning of nuclear power plant, based on study for graduation
- Interdisciplinary challenge from Fukushima -

National Institute of Technology, Fukushima College
O Shigekazu SUZUKI
Seichi SATO
Shiro JITSUKAWA
Katsuhiro AOYAGI
Takayuki NAKAMURA

Contents

- The Nuclear Decommissioning Network of the National Institute of Technology
- Education program on decommissioning
- KOSEN International Summer School
- Creative Robot Contest for Decommissioning

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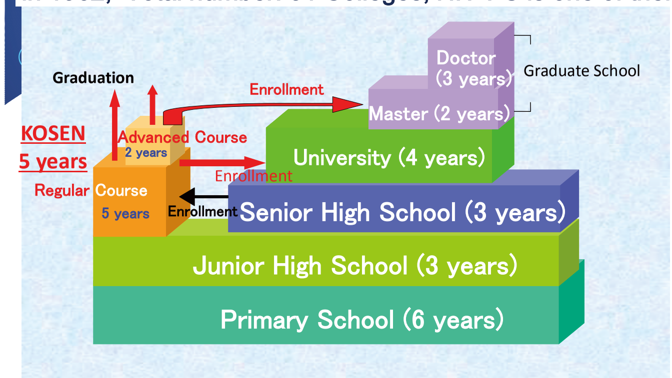
National Institute of Technology (KOSEN) was established in 1962, Total number: 51 Colleges, NIT-FC is one of them



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Outline of NIT Fukushima KOSEN

Department (Number of Students)
Mechanical Engineering (200) --- 40/each year
Electrical Engineering (200)
Chemistry and Biochemistry (200)
Civil Engineering (200)
Business and Communication (200)

Advanced Course (Number of students)
System Engineering for Industrial Technology (32)
Business Communication (8)
Special Course for Regional Rehabilitation (10)

Number of Staffs (As of April 1, 2016)

President	1	Research associate	7
Professors	30	Special appointed professors	4
Associate professors	36	Subtotal of professors	81
Assistant professors	3	Administrative staffs	44
		Total	125

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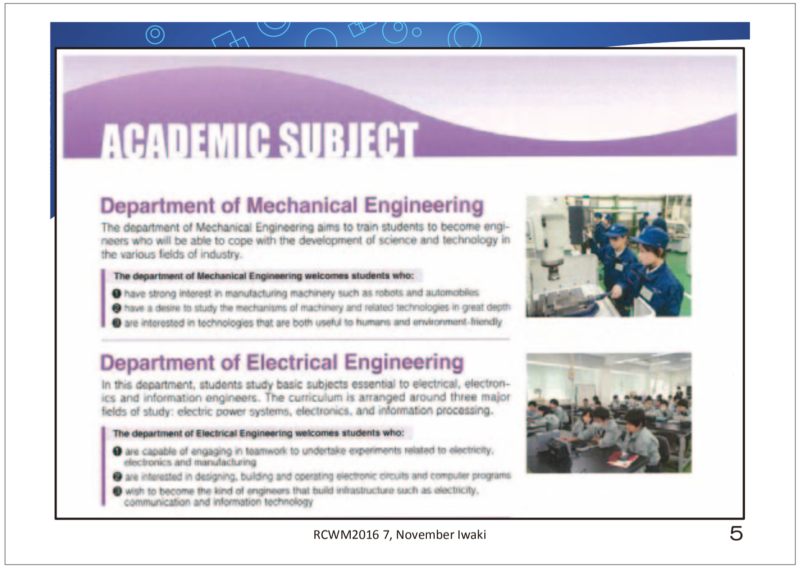
ACADEMIC SUBJECT
Department of Mechanical Engineering
The department of Mechanical Engineering aims to train students to become engineers who will be able to cope with the development of science and technology in the various fields of industry.
The department of Mechanical Engineering welcomes students who:
have strong interest in manufacturing machinery such as robots and automobiles
have a desire to study the mechanisms of machinery and related technologies in great depth
are interested in technologies that are both useful to humans and environment-friendly
Department of Electrical Engineering
In this department, students study basic subjects essential to electrical, electronics and information engineers. The curriculum is arranged around three major fields of study: electric power systems, electronics, and information processing.
The department of Electrical Engineering welcomes students who:
are capable of engaging in teamwork to undertake experiments related to electricity, electronics and manufacturing
are interested in designing, building and operating electronic circuits and computer programs
wish to become the kind of engineers that build infrastructure such as electricity, communication and information technology
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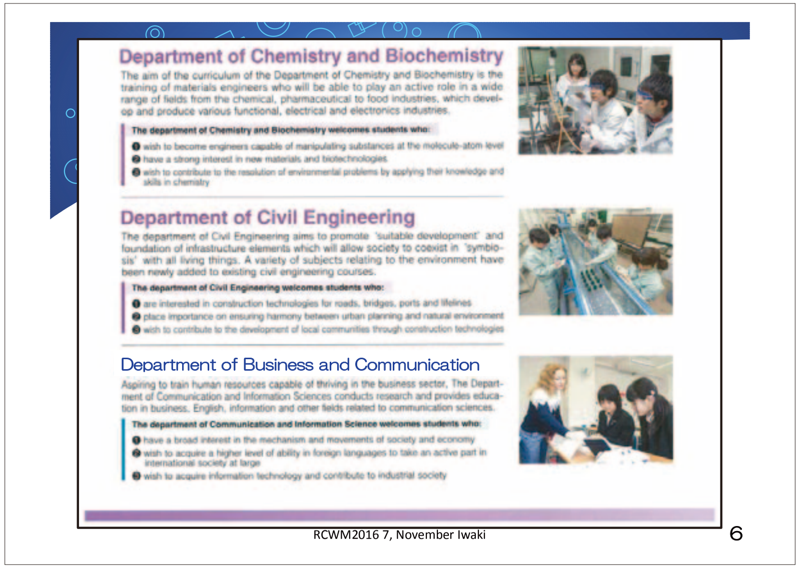
Department of Chemistry and Biochemistry
The aim of the curriculum of the Department of Chemistry and Biochemistry is the training of materials engineers who will be able to play an active role in a wide range of fields from the chemical, pharmaceutical to food industries, which develop and produce various functional, electrical and electronics industries.
The department of Chemistry and Biochemistry welcomes students who:
wish to become engineers capable of manipulating substances at the molecule-atom level
have a strong interest in new materials and biotechnologies
wish to contribute to the resolution of environmental problems by applying their knowledge and skills in chemistry
Department of Civil Engineering
The department of Civil Engineering aims to promote 'suitable development' and foundation of infrastructure elements which will allow society to coexist in 'symbiosis' with all living things. A variety of subjects relating to the environment have been newly added to existing civil engineering courses.
The department of Civil Engineering welcomes students who:
are interested in construction technologies for roads, bridges, ports and lifelines
place importance on ensuring harmony between urban planning and natural environment
wish to contribute to the development of local communities through construction technologies
Department of Business and Communication
Aspiring to train human resources capable of thriving in the business sector, The Department of Communication and Information Sciences conducts research and provides education in business, English, information and other fields related to communication sciences.
The department of Communication and Information Science welcomes students who:
have a broad interest in the mechanism and movements of society and economy
wish to acquire a higher level of ability in foreign languages to take an active part in international society at large
wish to acquire information technology and contribute to industrial society
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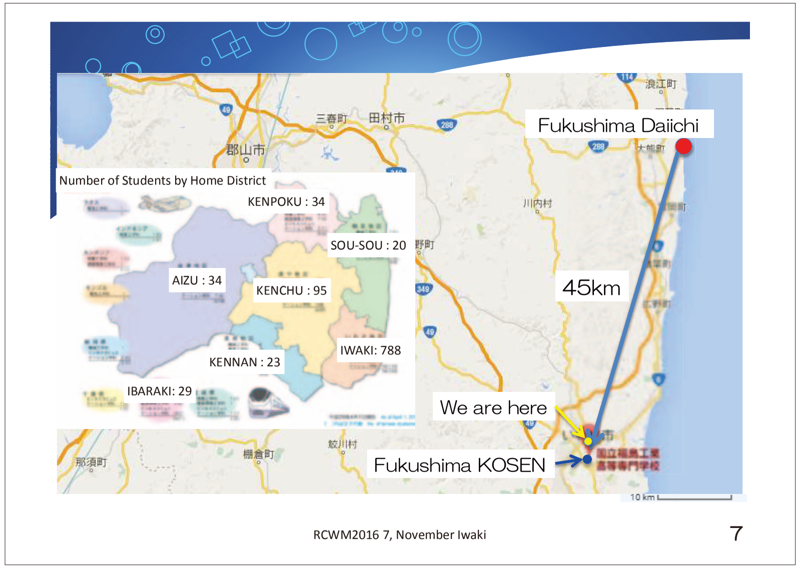
Fukushima Daiichi
45km
We are here
Fukushima KOSEN
Number of Students by Home District
KENPOKU : 34
SOU-SOU : 20
AIZU : 34
KENCHU : 95
IWAKI: 788
KENNAN : 23
IBARAKI: 29
郡山市
三春町
田村市
川内村
浪江町
大熊町
棚倉町
那須町
鮫川村
10 km
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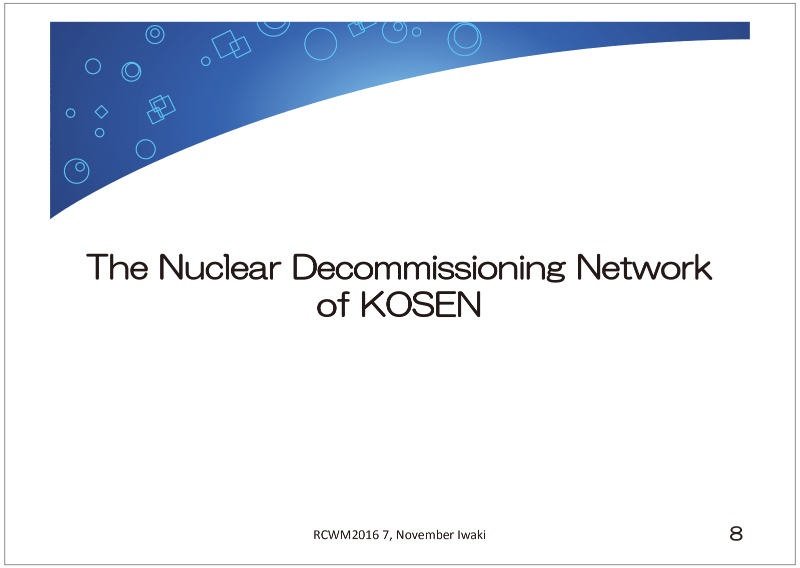
7


The Nuclear Decommissioning Network
of KOSEN
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The Nuclear Decommissioning Network of KOSEN

- Project：Basis research and human resource development, such as for the nuclear power plant decommissioning
- Member：President and professor of the KOSEN, professor of University and academic experts of an industry related to nuclear power, which was agreed for the purpose of the consortium
- Chairman：Nakamura president of Fukushima KOSEN
- Vice-president：Chouji president of Kagoshima KOSEN
- Secretariat：Fukushima KOSEN

	Number of organizations	Number of enrollees	Number of president
KOSEN	31	99	22
University	8	12	
Company	9	17	
Municipality	3	3	
Total	51	131	22

As of October 2016

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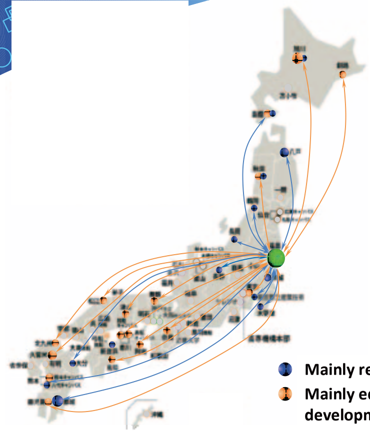
Cooperation KOSEN List



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Education program on decommissioning

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Course on "Decommissioning Technology"

elective subject

- 5th grade: Nuclear accidents, Graduation researches
- 4th grade : Reactor decommissioning, Internship
- 3rd grade: Decommissioning and the Society, Robotics for decommissioning
- 2nd grade: Radiation and radioactivity
- 1st grade: Nuclear power plants

Robotics at Fukushima-Daiichi

Basics on radiation and radioactivity

Basics for Nuclear power plants

Mechanical Engineering
Electrical Engineering
Chemistry and Biochemistry
Civil/Environmental Engineering
Communication/Information Science

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2015
- 3rd grade: (Decommissioning and the Society) 60 students
- 4th grade: (Reactor decommissioning) 10 students

2016
- 2nd grade: (Radiation and radioactivity) 100 students
- 3rd grade: (Robotics for decommissioning) 63 students
- 3rd grade: (Decommissioning and the Society) Students wanted
- 4th grade: (Reactor decommissioning) Students wanted

Visit Naraha Remote Technique Development Center of JAEA

Visit Center for Environmental Creation of Fukushima prefecture

Temporary storage area tour, Radiation measurement practice, Lecture

Important to see the site directly

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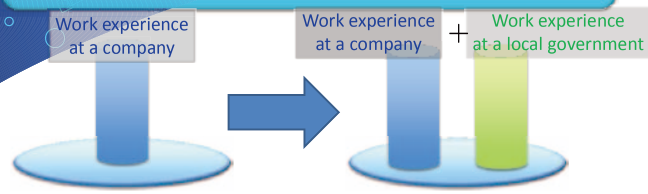
Complex type internship

- Complex type cooperation experience learning with KOSEN, companies and local governments -



Usual internship
- Improvement of professional knowledge
- Understanding of company activities

Complex type internship
- Improvement of professional knowledge
- Understanding of company activities
+
- Understanding of local government activities
- Communication with local citizens

Engineer of training which has a compound eye and the public communication skills

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Graduation researches

(1) Development of remote control equipment in a nuclear reactor towards such sampling techniques for the situational awareness in the nuclear reactor
(2) Development of remote control equipment related to the analytical technique of radioactive material
(3) Waste treatment and disposal research for the safe storage technologies for heat-generating waste.

Basis regions of these challenges
(4) Property analysis and development of remote control infrastructure technology.
(5) Estimation of product material in nuclear reactor.

Social-scientific approach
(6) Analysis of the relationship between work safety and motivation.

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Decommissioning internship

2015
- **Tokyo Power Technology (Radiochemical analysis, 2 students)**
- **KURION (Contaminated water treatment, Robot, 4 students)**
  →9/14 Lecture, 9/20～27 Technical training (USA, Richland, Denver)
- **JAEA (4 students)**
- **The University of TOKYO Uppsala summer school (Sweden, 2 students)**
- **The University of TOKYO Tokai summer school (2 students)**

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Decommissioning internship

2016
- The University of TOKYO International summer school
  (USA, Richland-PNNL/KURION, Huston-NASA, 2 students)
- The University of TOKYO Naraha summer school (2 students)
- Decommissioning technology Study at KURION
  (Tritium removal, 1 students)
- Fukushima Daiichi Nuclear Power Station (5 students)
  Fukushima KOSEN (2 students)
  Tsuruoka KOSEN (Yamagata, 2 students)
  Niihama KOSEN (Ehime, 1 student)
- Iwaki city Nuclear Power Division (2 students,)
- KOSEN International Summer School
  (USA, Richland/Hanford B reactor, Silicon Valley)
  Fukushima KOSEN (1 student)
  Ibaraki KOSEN (Ibaraki, 2 students)
  Kagawa KOSEN (Kagawa, 2 students)
  Kitakyushu KOSEN (Fukuoka, 1 student)
- JAEA (4 students)

Complex type internship

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KOSEN International Summer School

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KOSEN International Summer School

- Competition in cooperation KOSEN
- 8 students applicant
  → 6 students selected by reports evaluation

Schedule
September 18 (Sunday)
- Visit to America
- Study on the history of Hanford (The Reach museum)

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KOSEN International Summer School

Schedule
September 19 (Monday)
- Hanford site B Reactor tour
  → Interviewed by the DOE PR division
- Study on the PNNL
  Public involvement , Research and development of decontamination and environmental recovery
- Decommissioning technology Study at KURION
  Glassification, Remote control, Tritium removal

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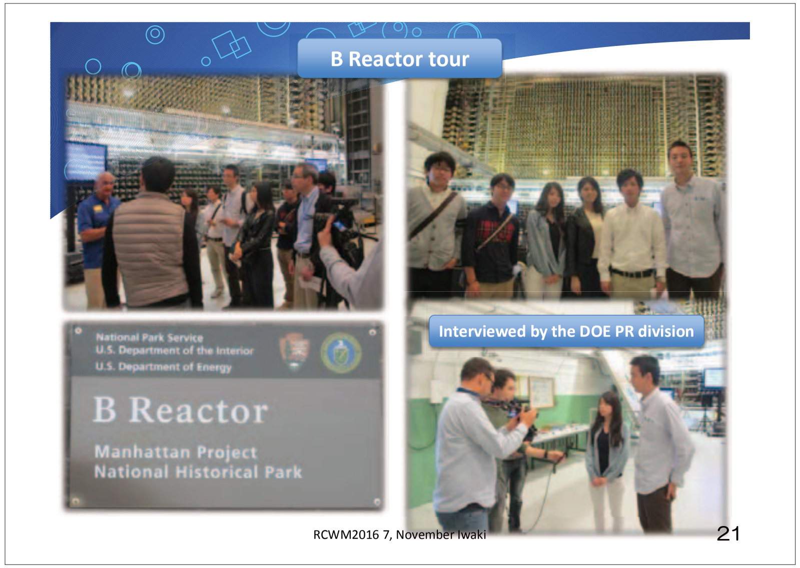

KOSEN International Summer School

September 20 (Tuesday)
- Move from Richland to Silicon Valley
- Create a presentation
- Interaction with entrepreneurs

September 21 (Wednesday)
- Stanford University tour
- Create a presentation
- Presentation (Interviewed by Nikkei)

September 22 (Thursday)
- Move from Silicon Valley to Narita

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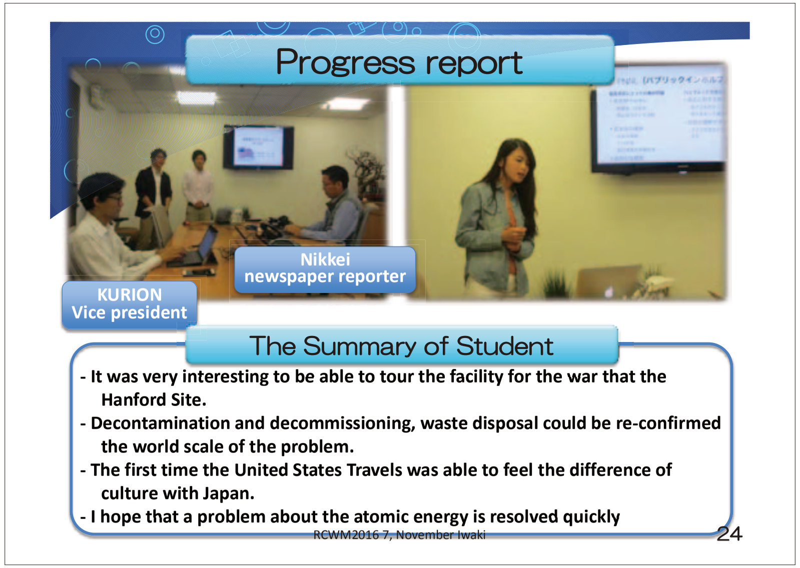

Creative Robot Contest for Decommissioning

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The outline of the contest implementation

Name: The 1st Creative Robot Contest for Decommissioning
Date: Saturday December 3rd, 2016
Place: Naraha Remote Technique Development Center of JAEA
Organizers:
Ministry of Education, Culture, Sports, Science and Technology
The Nuclear Decommissioning Network of KOSEN
Auspices: METI, NDF, IRID, Japan Science and Technology Agency,
FUKUSHIMA pref., IWAKI city, HIRONO town,
NARAHA town, JAEA
Supporters: IHI, ATOX, Joban Engineering
Secretariat: National Institute of Technology, Fukushima College
Administrator:
The executive committee of Creative Robot Contest for Decommissioning

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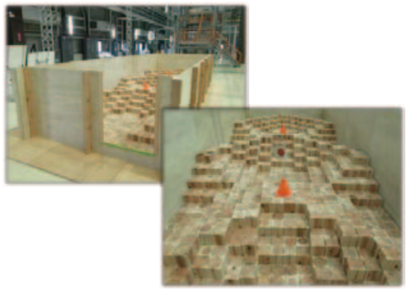
Environment of the Field

Each team select a field from the two options below with the assumption that it is the Fukushima Daiichi site.
1. Mock up stairs 2. Step Field



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Environment of the Field

- Dark without lights
- Robots can't be seen directly due to remote control
- Radio waves can't be reached directly due to the thick concrete walls
- There is a limit for the control of cameras and semi-conduct instruments due to the influence of strong radiation

Tasks for the Robots

(Mock-up Stairs)
- Carry 5kg object from 1st floor to 2nd floor and return to the original position on 1st floor
- Detect an unknown object on 2nd floor
- Others related to the tasks for decommissioning

(Step Field)
- Detect the shape, the area and the up-and-down of step field
- Detect an unknown object which is set in the field
- Others related to the tasks for decommissioning

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Number of the Participants

・15 teams
-11 colleges of National Institute of Technology
Hakodate (Hokkaido, 2 teams), Asahikawa (Hokkaido)
Sendai (Miyagi, 2 teams), Fukushima (Fukushima)
Tokyo (Tokyo), Toyama (Toyama)
Maizuru (Kyoto), Nara (Nara)
Kouchi (Kouchi), Kitakyushu (Fukuoka)
Kumamoto (Kumamoto)
- 1 Tokyo Metropolitan College of Industrial Technology (Metropolitan KOSEN)
- 1 Osaka Prefecture University College of Technology (Osaka Prefecture KOSEN)

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Naraha Summer School

・September 1st : Visit Fukushima Daiichi
・September 2nd : Study of decommissioning at Naraha Remote Technique Development Center

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Summary

- In the mission to Fukushima reconstruction, by the people and technology to grow through technological challenge, safe decommissioning work of Fukushima Daiichi continues.

- Specialty and Ability to overlook the whole
- Flexibility to think about multi scenarios
- Ability to cooperate across the field
- Challenge mind and the executive ability
I would like to educate the student who learned the ability mentioned above continuously

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Thank you for your attention

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2.5 Poster Presentation

2.5.1 *In situ* X-ray Absorption Spectroscopy Study on Hydrogen Recombination Reaction of Precious Metal Nanoparticles Catalysts

Yusaku KIMURA [1], Daiju MATSUMURA [2], Yasuo NISHIHATA [2], Masashi TANIGUCHI [3], Juichiro MIZUKI [1] and Hirohisa TANAKA[1]
[1] *Kwansei Gakuin University; asanisimasa@kwansei.ac.jp*
[2] *Japan Atomic Energy Agency*
[3] *Daihatsu Motor Co., Ltd.*

Abstract
In case of severe accidents in nuclear power plants, and during long-term storage of radioactive waste, hydrogen is generated. The research on hydrogen recombination into safe water by precious metal nanoparticle catalyst has attracted attention. In this study, the mechanism of catalyst poisoning of carbon monoxide has been investigated by in-situ time-resolved analysis on the precious metal using Dispersive X-ray Absorption Fine Structure (DXAFS) at SPring-8. It is found that formation of surface oxide plays an important role in the hydrogen recombination, and CO inhibits the absorption and causes a decrease in activity. Furthermore, the use of ceria-composite-oxides, having oxygen storage capacity, as the support was found to reduce CO poisoning. It was possible to obtain useful guidance in catalyst design.

2.5.2 Application of Automotive Catalyst to Hydrogen Recombiner

Keisuke TAKENAKA[1], Soma KISHIMOTO[1], Masashi TANIGUCHI [2], Yu KAMIJI [3], Yasuo NISHIHATA[3] and Hirohisa TANAKA [1]
[1] *Kwansei Gakuin University; eoy10812@kwansei.ac.jp*
[2] *Daihatsu Motor Co., Ltd.*
[3] *Japan Atomic Energy Agency*

Abstract
At the severe accident of the Fukushima Daiichi nuclear power plant, seawater has been injected for cooling the reactor. When the nuclear waste should be kept isolated, water is decomposed to generate hydrogen and oxygen by radiation in the container. So that highly active recombiner catalyst is required. The intelligent automotive catalyst was found to have excellent activity to recombine hydrogen from a room temperature. In our laboratory, robust catalyst materials have been developed for water vapor and the catalyst poison such as carbon monoxide inhibits the hydrogen oxidation reaction. We are aiming at the realization of highly active recombiner.

2.5.3 Results of Preliminary Analyses on Natural Convection Behavior of Hydrogen Gas Generated in a Radioactive Long-Term Waste Storage Container

Yusei TANAKA [1], Yota SUZUKI [2] and Kazuyuki TAKASE [3]
[1] *Nagaoka University of Technology; s165048@stn.nagaokaut.ac.jp*
[2] *Nagaoka University of Technology; s165047@stn.nagaokaut.ac.jp*
[3] *Nagaoka University of Technology; takase@vos.nagaokaut.ac.jp*

Abstract
In the radioactive waste storage container, because the concentration of hydrogen gas generated by the decomposition of water by radiation increases with time, it is important to accurately grasp the behavior of the hydrogen gas in the sealed container for safety improvement of the storage container. Since the hydrogen gas is heated by the radioactive material accumulated in the storage container, it is conceivable that natural convection is formed. Therefore, in order to clarify quantitatively the natural convection behavior of hydrogen gas due to the decay heat of radioactive materials in the long-term waste storage container, preliminary analyses were performed on the system of a small-scale experimental apparatus in which specification of the long-term radioactive waste storage container is simply simulated. From the present results, the perspective which can predict natural convection phenomena in the long-term waste storage container numerically was obtained.

2.5.4 A Plan of Basic Experiments for Performance Evaluation of a Long-Term Radioactive Waste Storage Container for Reducing Hydrogen Gas Concentration

Yota SUZUKI [1], Yusei TANAKA [2], Kazuyuki TAKASE [3] and Ryutaro HINO [4]
[1] *Nagaoka University of Technology; s165047@stn.nagaokaut.ac.jp*
[2] *Nagaoka University of Technology; s165048@stn.nagaokaut.ac.jp*
[3] *Nagaoka University of Technology; takase@vos.nagaokaut.ac.jp*
[4] *Japan Atomic Energy Agency; hino.ryutaro@jaea.go.jp*

Abstract
In the decommissioning of nuclear power plants, the long-term management of radioactive waste of fuel debris, etc. are necessary. In the process, hydrogen which is the flammable gas is generated by the decomposition of water by radiation. Therefore, it is important to ensure the safety of the waste storage container to reduce the concentration of hydrogen gas, and to keep below the explosion limit (4%). Consequently, a basic experiment to investigate the effectiveness of the waste storage container with the flammable gas concentration reduction mechanism using the passive autocatalytic recombiner (PAR) has been planned. The present study describes the research plan to use a small modeled experimental apparatus.

2.5.5 Potential Applicability of Bioconsolidation for Retardation of Diffusional and Convective Transfer of Effluent Radioactive Species
- The technological Basis and Prospect for Actual Application -

Shigekazu SUZUKI [1] and Kenichi KURUMADA [1]
[1] *National Institute of Technology, Fukushima College; kurumada@fukushima-nct.ac.jp*

Abstract
Bioconsolidation by anaerobic metabolic activities of in-ground microbes was intended to densifying the soil for the purpose of mitigating the effluent aqueous radioactive species accompanied by the transfer of in-ground environmental water. Similar biological process has been found in the spontaneous formation of hard and rigid calcium carbonate scale on the inner wall of drainage pipes in landfill sites, which is sufficiently densified to prevent water from permeating out. The artificial conditioning for the in-ground formation of calcite is expected to contribute to reducing the effluent of radioactive substances from the decommission site. The authors showed that feeding calcium ion and nutritious organic substances with water into the ground can lead to the generation of microbe-derived calcite solid phase which forms dense interconnections amongst the constituent soil particles. To enhance the crystal nucleation of the calcite sub-micron particles, the nucleation-promotive effect of adding carbonyl-containing resin into the soil is to be further investigated. The appropriate conditions for the continuous and short-term growth of the soil-bridging calcite phase need to be experimentally clarified in the future investigation. In parallel with carrying out these fundamental laboratory experiments, the authors seek for collaborative opportunities to implement more actual on-site examinations with the help of advices by those who have superior and broader expertise on what are the more actual technical requirements for the mitigation technologies of the radioactive effluents.

2.5.6 A Study of the Economic Effects by the Nuclear Industry and Development of Human Resources in Regional Economy from the Point of the Younger Generation Consciousness Concerning Nuclear Energy

Aika FUJIKI [1] and Kazunori AKUTAGAWA
[1]*National Institute of Technology, Fukushima College; 10532@fsnct.com*

Abstract
The restoration work from the nuclear accident in Fukushima would continue about 40 years. Development of human resources engaged in the work is very important to conduct the long-term restoration. We surveyed opinions of the Japanese and French young generations' attitude toward nuclear energy after the accident. Opinions of about 100 students of National Institute of Technology in Japan were compared with those of French students.

2.5.7 Enhanced Desorption of Cs from Cs Collapsed Interlayer Region of Vermiculitized Biotite by Hydrothermal Treatment with Eluting Cations

Xiangbiao YIN [1], Hideharu TAKAHASHI, Toshihiko OHNUKI and Kenji TAKESHITA
[1] *Laboratory for Advanced Nuclear Energy, Tokyo Institute of Technology; yin.x.aa@m.titech.ac.jp*

Abstract

Natural soil has ability of immobilizing radiocesium (Cs) within phyllosilicate fraction. The objectives of present study are to clarify how Cs fixation on vermiculite is influenced by structure change caused by Cs sorption at different loading levels and how Cs desorption is affected by various replacing cations induced at different treating temperature. As a result, more than 80% of Cs was readily desorbed from vermiculite with loading amount of 2% saturated Cs (5.49×10^{-6} mol/g) after four cycles of treatment of 0.01M Mg^{2+}/Ca^{2+} at room temperature, but less than 20% of Cs was desorbed from saturated vermiculite. These distinct desorption patterns were attributed to inhibition of Cs desorption by interlayer collapse of vermiculite, especially at high Cs loadings. In contrast, elevated temperature significantly facilitated divalent cations to efficiently desorb Cs from collapsed regions. After five cycles of treatment at 250℃ with 0.01M Mg^{2+}, ~100% removal of saturated Cs was achieved. X-ray diffraction analysis results suggested that Cs desorption was completed through enhanced diffusion of Mg^{2+} cations into collapsed interlayer space under hydrothermal condition resulting in subsequent interlayer decollapse and readily release of Cs^{+}. The results were expected to provide new insights to explore available decontamination technology for Fukushima Cs-contaminated soils.

3. Technical Seminar on Safety Research for Radioactive Waste Storage

3.1 Session 3 Radiolytic Hydrogen and Corrosion

3.1.1 Gamma-Radiolysis Induced Corrosion

Jungsook Clara WREN
Department of Chemistry, Western University, London, Ontario, Canada; jcwren@uwo.ca

Predicting the efficacy, service lifetime and the probability of catastrophic failure of nuclear reactor and used fuel container materials is very important for public and environmental safety. Most of these materials are made of transition metals and their alloys. Corrosion is a key degradation mechanism for these metals and alloys when exposed to humid or wet environments. Corrosion is an electrochemical process involving metal oxidation coupled with reduction of solution species, followed by interfacial transfer of charged species (metal cations, oxygen anions, electrons). In general, the more oxidizing the solution environment is the faster the metal oxidation that leads to metal dissolution occurs.

In nuclear environments ionizing radiation is present. When exposed to γ-radiation water decomposes into a number of different redox active species:

$$H_2O \xrightarrow{\gamma} \{H_2O^+, e^-, H_2O^*\} \rightarrow \bullet e_{aq}^-, H^+, \bullet H, H_2, H_2O_2, \bullet OH \rightarrow\rightarrow H_2, H_2O_2, O_2,$$

Under a continuous flux of γ-radiation the solution environment quickly becomes oxidizing due to accumulation of H_2O_2 and O_2 in water [1,2]. Thus, γ-radiation can accelerate corrosion significantly.

Metal oxidation leads to the formation and growth of solid metal oxide on the surface as well as dissolution of metal cations into the solution phase. The formation and growth of oxide depends on the rate of metal oxidation. However, an oxide layer, preformed or formed during corrosion, is a potential barrier for the interfacial transfer of metal cations [3,4]. Hence, the presence of an oxide layer can decrease the overall corrosion rate. This effect will depend on the physical and electrochemical properties of the oxide. Thus, γ-radiation by increasing the oxidizing strength of water may initially accelerate metal oxidation, but it can also promote earlier formation of a passive oxide layer and thereby, quickly suppress further corrosion. The underlying processes that determine the properties of the oxide formed, and the evolution of these properties as the oxide grows and may convert to other oxides as corrosion progresses are not well understood.

To develop a fundamental understanding of oxide formation and growth on a corroding alloy and the effect of the corrosion environment on the oxide evolution a comprehensive program of study has been put in place. The metals and alloys studied to date include copper, nickel, carbon steel, stainless steels, nickel alloys and a cobalt alloy. The corrosion kinetics is being studied by using a combination of electrochemical techniques and coupon-exposure tests. These measurements are augmented by post-test surface imaging and spectroscopic analyses, and dissolved metal analysis. The experimental parameters are the solution pH, ionic strength, temperature, water volume to surface area, and the absence or presence of γ-radiation.

Gamma-radiation affects corrosion by producing radiolysis products, particularly H_2O_2. This increases the

E_{corr} of the corroding system, and increases the rate of metal oxidation initially. The oxidation of the metal leads to both metal dissolution and oxide/hydroxide formation [3-8].

The driving force for each metal oxidation reaction is the difference between E_{CORR} and the equilibrium potential (E_{eq}) (or overpotential, η_{ox}) of the oxidation. Because it is a potential barrier for metal cation and electron transfer, the presence of an oxide layer reduces the effective overpotential for the oxidation [3,4]:

$$\eta_{ox} \approx E_{CORR} - E_{eq}^{ox} - \Delta V_{oxide}$$

One of the consequences of oxide growth is thus to increase E_{CORR}. That is, both a more oxidizing environment and oxide growth can increase E_{CORR}. We have shown that because the formation of a solidus oxide and its conversion to another solid oxide takes time, the sequence of metal oxidation reactions leading to the formation of different oxides follows the order of the E_{eq} values of the metal oxidation reactions. Thus, E_{CORR} controls not only the rates of individual oxidation reactions but also limits how far along the oxidation sequence occurs.

Because more than one oxide can be formed at a given E_{CORR} more than one corrosion pathway is available. Different solution redox and solution properties, such as oxidant concentration, pH, ionic strength and temperature, affect the kinetics of individual reactions and transport processes involved in corrosion differently. For example, the solution pH does not affect the driving forces for individual oxidation reactions *per se*, but it has a strong impact on the metal dissolution rate. At a pH where the solubility of metal cations is high, the metal oxidation leads to mostly metal dissolution and oxide growth is very slow whereas at a pH where the metal has a low solubility the oxidation leads to faster oxide formation [3-8].

The different effects of solution redox potential and pH on corrosion can be clearly seen by comparing the corrosion of carbon steel with and without radiation present, as illustrated in **Fig. 1**. Gamma-radiation at 150 °C increases iron dissolution at neutral pH while it promotes the formation and growth of a passive and thicker oxide layer at pH 10.6 and with γ-radiation present [8].

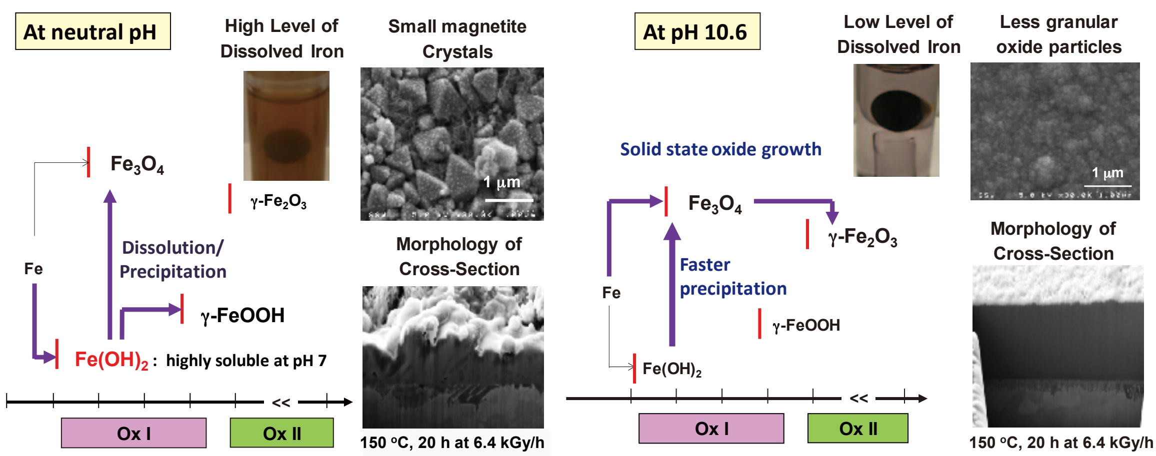

Fig. 1: Illustration of iron oxidation pathways at different pHs in the presence of γ-radiation, leading to different effects on oxide formation and dissolution.

References

[1] J.M. Joseph, B-S Choi, P. Yakabuskie, J.C. Wren. *Rad. Phys. Chem*., 77 (2008) 1009-1020.
[2] P.A. Yakabuskie, J.M. Joseph, J.C. Wren. *Rad. Phys. Chem*., 79 (2010) 777-785.
[3] M. Momeni, J.C. Wren. *Faraday Discuss.,* 180 **(**2015) 113-135.
[4] M. Momeni, M. Behazin, J.C. Wren. *J. Electrochem. Soc.*, 163 (3) C1-C12 (2016).
[5] A.Y. Musa, J.C. Wren. *Corros. Sci.,* 109 (2016) 1-12.
[6] A.Y. Musa, M. Behazin, J.C. Wren. *Electrochim. Acta,* 162 (2015) *185-197*.
[7] Q.W. Knapp, J.C. Wren. *Electrochim. Acta,* 80 (2012) 90-99.
[8] K. Daub, X. Zhang, J.J. Noël, J.C. Wren. *Corros. Sci.,* 53 (2011) 11-16.

Gamma-Radiolysis Induced Corrosion

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Department of Chemistry
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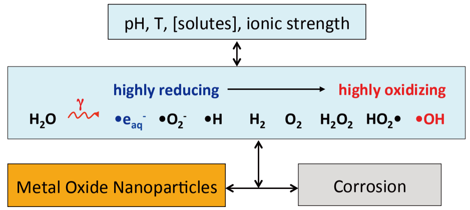
London, Ontario, Canada

Research Conference on Post-accident Waste Management Safety (RCWM2016), Iwaki, Japan, 2016 Nov 8

Radiation Assisted Materials Performance Science Group

2

- RAMPS research focuses on interfacial charge and mass transfer processes occurring in energetic environments such as induced by radiolysis.



RAMPS Corrosion & Related Projects

3

- Research Goal
  - To develop high-fidelity models for corrosion under aggressive environments induced by high T/high P, aggressive chemicals and γ-radiation.
- Nuclear Industry Problems Addressed
  - Corrosion of in-reactor materials and water chemistry controlunder reactor operating conditions
  - Candidate fuel cladding materials for Gen IV supercritical water reactors
  - Reactor vault structure materials
  - Used fuel container materials under deep geological repository conditions
- Materials
  - Cu, Ni, Steels (various ratios of Fe/Cr/Ni), Ni alloys, Co alloys.
- Water phases
  - Humid gases (Ar, O_2/Ar, air), water droplets, liquid to supercritical water.

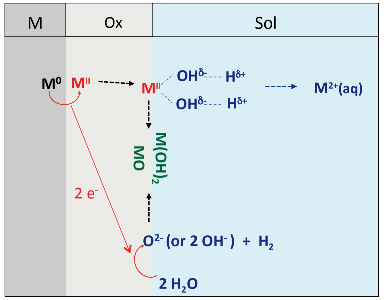
4 γ-Radiation Induced Corrosion

1. Experimental Approach
2. Key Observations
3. Corrosion Mechanism
4. Mass & Charge Balance (MCB) Model
5. Overall Conclusions

Aqueous Corrosion

5

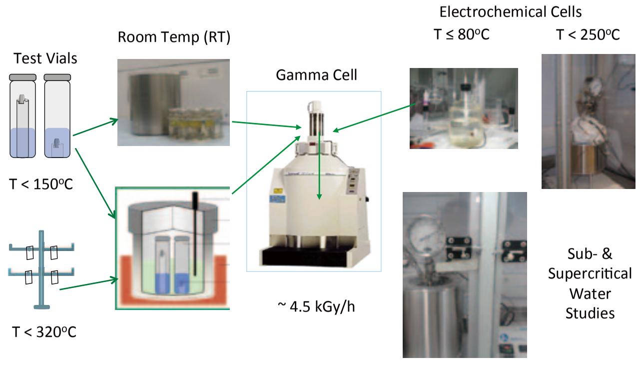
- Corrosion is an interfacial charge and mass transfer process.



- Metallurgical properties
  - Alloy composition
  - Microstructure
- Solution redox chemistry & solvation properties
  - Chemical addition (H_2O_2)
  - Radiolysis
  - pH, T, etc.
- Oxide film properties
  - Affect charge transfer (ions, electrons, holes)
  - Change over long times
  - Oxide growth may be irreversible

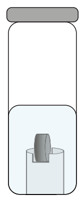
Electrochemical and Coupon Studies

6



Post-Test Analyses

7



- H_2, O_2 by GC-MS

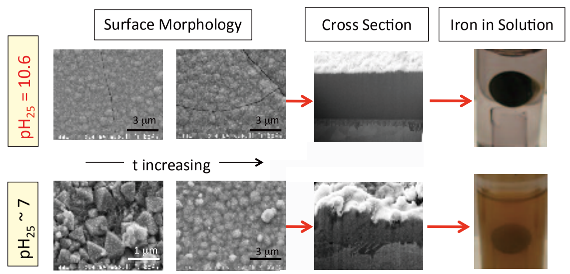
- Dissolved metal by ICP-MS
- pH

- Morphology: SCM, SEM, TEM
- Elemental/Chemical: EDX, XPS, Raman, FTIR, XRD
- Depth Profile: SCM, FIB, AES with Ar^+ sputtering

Effect of γ-Radiation on CS Corrosion

8

150°C, up to 66 h at 6.4 kGy/h, Ar purged



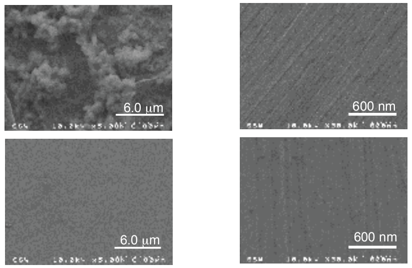
- Radiation influences CS corrosion differently, depending on pH.

K. Daub, X. Zhang, J.J. Noël, J.C. Wren, *Corrosion Science*, 53 (2011), 11-16

Effect of γ-radiation on Corrosion of CS vs SS

9

(pH 8.4, 80 °C)



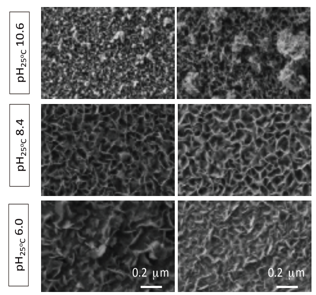
- CS becomes as corrosion resistant as SS in mildly basic solutions under radiation.

L. Wang, K. Daub, Q. Knapp, Z. Qin, J.J. Noel, J.C. Wren*. *J. Electrochem. Soc.* **159** (2012) C503-C512.
K. Daub, X. Zhang, L. Wang, Z. Qin, J.J. Noel, J.C. Wren, *Electrochimica Acta*, *56* (2011), 6661-6672

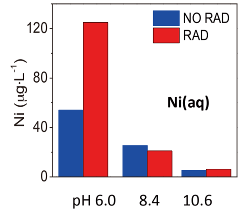
Effect of pH & γ-Radiation on Ni-Cr-Fe Alloys

10

at 150 °C



Dissolved Metal

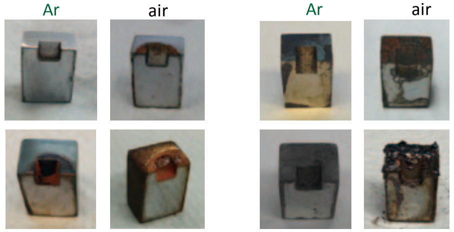


- γ-radiation increases Ni dissolution at pH 6.0, but has a negligible effect at pH 8.4 & 10.6.

A.Y. Musa, J.C. Wren*. *Corros. Sci.* 109 (2016) 1-12.

Crevice Corrosion of CS in Humid Atmosphere

11



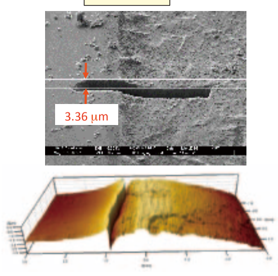
- The crevice surfaces remain clean while the bold surfaces are highly corroded.
- Radiation appears to induce 'inverse' crevice corrosion.

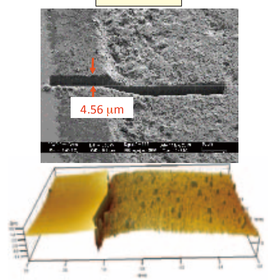
L. Wu, D. Guo, *A. Van Belois*, J.J. Noël, P.G. Keech, J.C. Wren. Manuscript in preparation

γ-Radiation on Galvanic Corrosion of CS-SS

12

pH 6.0, 50 °C, 24 h Immersion

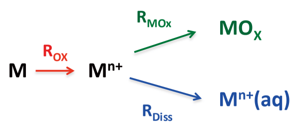


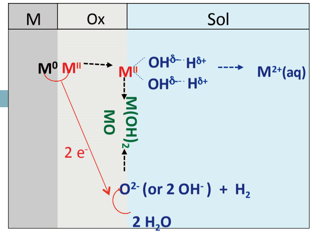


- Radiation accelerates CS corrosion at pH 6.0 without inducing an enhanced galvanic effect.

Corrosion Mechanism (1)

13





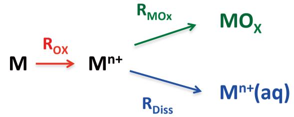
- Metal oxidation leads to solid oxide formation and dissolution:

$$\boxed{R_{OX} = R_{MOx} + R_{Diss}}$$

- These two pathways compete, and their relative rates determine the overall rate of oxide growth.
- The oxide growth impedes metal oxidation. This decreases both dissolution and oxide growth.

Corrosion Mechanism (2)

14

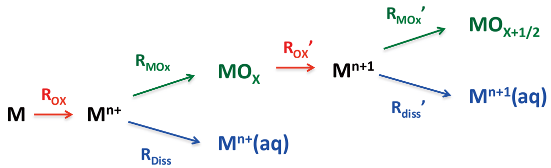


$$\boxed{R_{OX} = R_{MOx} + R_{Diss}}$$

- A chemical oxidant such as H_2O_2 increases R_{OX}
- γ-radiation produces powerful oxidants and increases R_{OX}
- T and pH affect oxide growth by changing the rates of the two competing reactions, R_{MOx} and R_{Diss}.

Corrosion Mechanism (3)

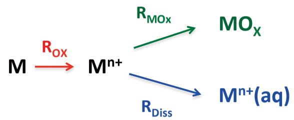
15



- Corrosion may involve more than one metal oxidation reaction.
- These reactions may occur in parallel and/or in series.
- Each oxidation leads to different dissolution and oxide growth.
- Depending on the competition kinetics at each oxidation step corrosion can take a completely different path.

Effect of pH

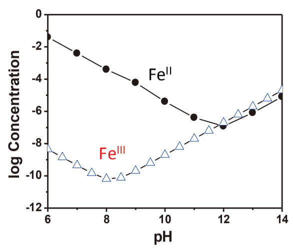
16



$$\boxed{R_{OX} = R_{MOx} + R_{Diss}}$$

- pH does not affect the driving force for an electrochemical reaction, i.e., no direct effect on R_{OX}.
- Metal cation solubility varies widely depending on pH.
- pH affects R_{Diss} and hence R_{MOx} and R_{OX}

Solubility of Fe^{II} and Fe^{III} at 25°C



Effect of Temperature

17

$M \xrightarrow{R_{OX}} M^{n+}$; $M^{n+} \xrightarrow{R_{MOx}} MO_X$; $M^{n+} \xrightarrow{R_{Diss}} M^{n+}(aq)$

$\boxed{R_{OX} = R_{MOx} + R_{Diss}}$

- T has a very small effect on the driving force for an electrochemical reaction.
- T strongly affects the rate of a reaction with a high ΔE_A (the Arrhenius dependence).
- T also affects diffusion and hence, R_{Diss}.
- R_{MOx} increases faster than R_{Diss} with increasing T.

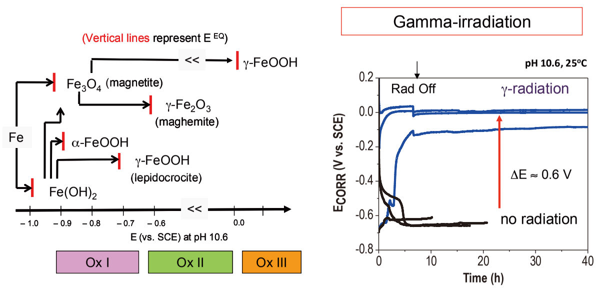
Varying Effect of γ-Radiation Depending on pH, T

18

- γ-Radiation produces strong oxidants such as H_2O_2. This initially accelerates metal oxidation.
- However, the consequence of this increase in metal oxidation depends on pH.
- At an acidic pH where metal solubility is high the increase in metal oxidation leads to mostly an increase in metal dissolution.
- At a mildly basic pH (typically 8 - 11) where the metal solubility is low,
  - The initial increase in metal oxidation leads to faster and earlier formation of metal hydroxide/oxide, which may further oxidize to another oxide.
  - The oxide that forms and grows suppresses the rate of metal oxidation. The overall corrosion is quickly suppressed.
  - The oxide is a potential barrier, and E_{corr} increases as the oxide grows.

E_{CORR} During corrosion of CS under γ-radiation

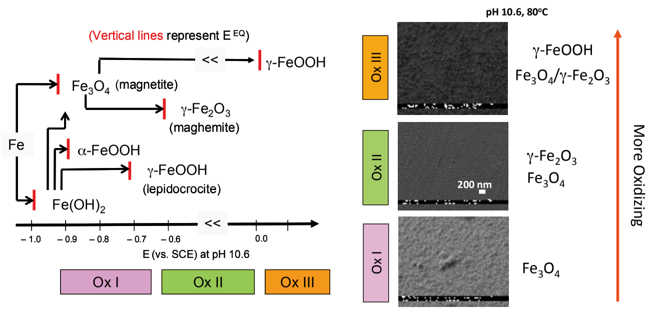
19



K. Daub, X. Zhang, L. Wang, Z. Qin, J.J. Noel, J.C. Wren, *Electrochimica Acta, 56* (2011), 6661-6672
K. Daub, X. Zhang, J.J. Noël, J.C. Wren, *Corrosion Science*, 53 (2011), 11-16

Chemical and phase composition of oxide observed as a function of potential

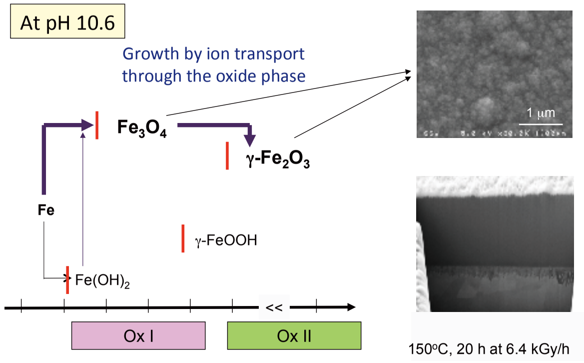
20



K. Daub, X. Zhang, L. Wang, Z. Qin, J.J. Noel, J.C. Wren, *Electrochimica Acta, 56* (2011), 6661-6672
K. Daub, X. Zhang, J.J. Noël, J.C. Wren, *Corrosion Science*, 53 (2011), 11-16

Main oxidation path at pH 10.6

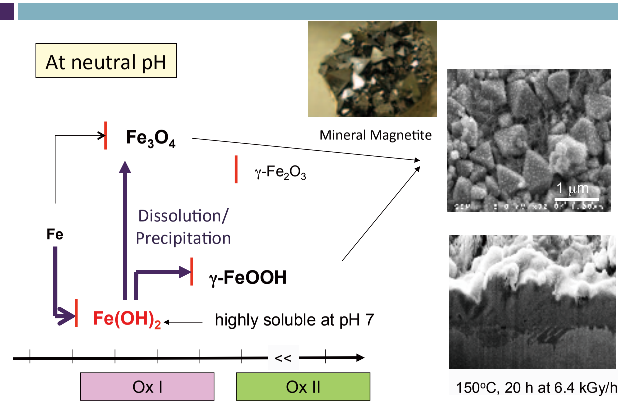
21



21

Main oxidation path at neutral pH

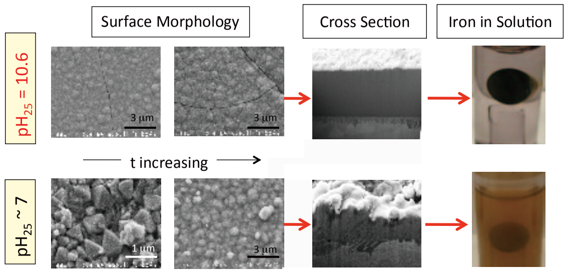
22



Effect of γ-Radiation on CS Corrosion

23

150°C, up to 66 h at 6.4 kGy/h, Ar purged



- Radiation increases the rate of metal oxidation.
- pH and T affect competition kinetics of oxide growth and dissolution.
- The effect of γ-radiation on corrosion varies depending on pH, T

Mass & Charge Balance (MCB) model

24

- The MCB model is a framework that prescribes a method of calculating both oxide growth and dissolution simultaneously during corrosion.
  - The elementary processes that determine the overall corrosion kinetics are electrochemical redox reactions, mass and charge transport across oxide film, oxide formation and metal ion dissolution.
  - The kinetics of the individual elementary processes follow classical chemical reaction rate and mass and charge flux equations.
- The model is capable of calculating time-evolution of oxide growth and dissolution of different alloy elements and as a function of corrosion exposure conditions.
- It has successfully simulated the corrosion tests and electrochemical experiments performed on a Co-Cr alloy, and various Fe-Ni-Cr alloys as a function of potential, pH and temperature .

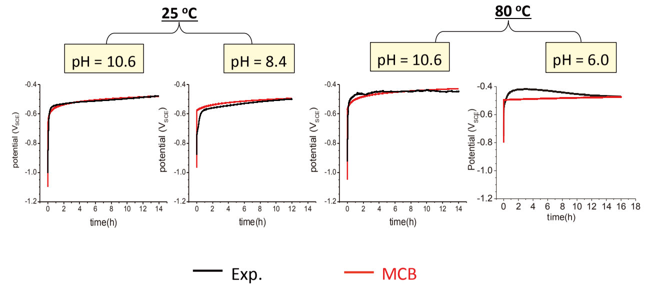
M. Momeni, M. Behazin, J.C. Wren. *J. Electrochem. Soc.*, **163** (3) C1-C12 (2016).
M. Momeni, J.C. Wren*. *Faraday Discuss.* **180** (2015) 113 - 135.

Corrosion Potential

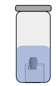
$|\sum i_c| = \sum i_a$ at E_{CORR}

25

- The solution reduction fluxes were formulated the same way as the oxidation fluxes.



3-d Coupon Corrosion Tests



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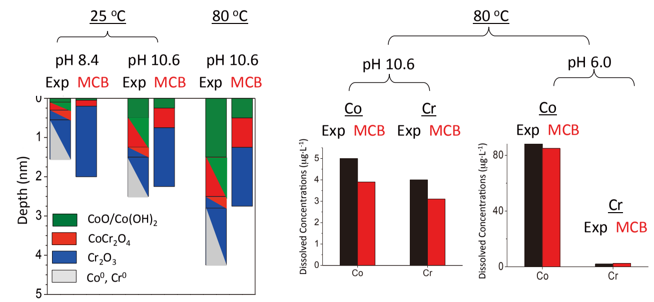


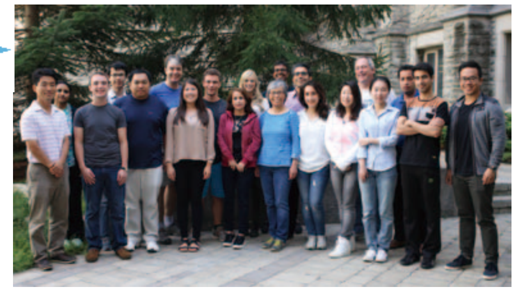
Conclusions

27

- γ-radiation increases metal oxidation rate.
- However, the consequence of this increase varies with pH, T, ionic strength, V_{aq}/A_{metal} because they influence competition kinetics between oxide growth and dissolution.
- Corrosion of an alloy in a given environment may involve more than one metal oxidation, each leading to both oxide growth and dissolution.
- Depending on the competition kinetics at each oxidation step, corrosion can take a completely different path.
- By decoupling the elementary processes involved in corrosion, we were able to develop a corrosion kinetic model that can predict the evolution of corrosion as a function of exposure parameters (pH, T, [oxidant], rad).

Thank You

28



RAMPS

UNENE

ONTARIO POWER GENERATION

NSERC CRSNG

nwmo

3.1.2 Corrosion Risk Management in Fukushima Dai-ichi NPS

Shunichi SUZUKI
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The earthquake occurred on March 11, 2011 and the following tsunami caused the loss of the total AC power sources at Unit 1 to 4. Seawater was temporarily injected into the reactors and the SFPs as an emergency action for fuel cooling and corrosion mitigation actions were urgently required. Various corrosion mitigation measures have been applied to the plants. So far, no major corrosion problems have been identified and continued efforts to make the plant more stable are required because the decommissioning work in the radiation environment is a difficult and long-term mission.
In this presentation, measures for corrosion of the spent fuel pool (SFP), PCV/RPV, canisters for fuel debris & containment vessels for absorbed Cs are introduced.

1. Risk Management for Corrosion of SFP, PCV/RPV

(1) Spent Fuel Pool pH neutralization

In the SFP of Unit3, water pH 11 was found at the first sampling of the water after the accident. The pH increase was estimated as Ca^{2+} dissolution from the concrete/ mortar dropped into the SFP at the hydrogen explosion of the Reactor Building. Boric acid was added to neutralize the pH to mitigate the Al rack corrosion on June 27th and pH9 has been kept so far [1].

(2) Spent Fuel Pool hydrazine(N_2H_4) addition

In Unit 2, 3 & 4 SFP Cl^- concentration was 1500-2000ppm due to sea water injection. Hence, crevice corrosion sensitivity of type 304 SFP liner was expected. As measures, deoxidization process such as N_2 bubbling at make up tank and periodical N_2H_4 addition were applied. N_2H_4 addition was also expected as a method of sterilization in order to reduce the risk of microbiologically influenced corrosion (MIC) [1].

(3) PCV/RPV environments

PCV/RPV nitrogen gas injection started to extract residual H_2 and to keep inertness in PCV to avoid explosion. Same as SPF, deoxygenated water with N_2 gas, N_2H_4 are injected into PCV/RPV [1].

(4) Development of technologies for assessing structural integrity of PCV/RPV.

The structural integrity of various facilities is required until the completion of decommissioning. In order to verify or to make the strategy to recover and keep the PCV integrity for tens of years, seismic assessment based on prediction and mitigation of corrosion have been developed by IRID (International Research Institute of Nuclear Decommissioning). In addition, corrosion mitigation methods available in the high radiation fields such as the sodium pentaborate injection are also being developed [2].

2. Risk Management for Corrosion of Fuel Debris Canister

Upon considering the fuel debris retrieval method, IRID defines the concerns with high priorities in the method from the perspective of “retrieving” the fuel debris and reactor internals, including the design of the storage canister. Safety items for the storage canister are sub-criticality, hydrogen generation, structural strength and material aging such as corrosion as well as heat removal, shielding & confinement [2]. If fuel debris be retrieved and put in the canister in water, corrosion resistant materials, corrosion inhibitors or water quality control must be considered. For the design of a canister, one of the important key parameters is the characteristics of fuel debris, which is supposed to be a complex material composed from various kinds of metals, ceramics and concrete. During the accident progression, seawater was injected to cool the core, which might affect the components of the fuel debris. Takano et al. investigated the possible impacts

of seawater injection on the chemical and physical state of the corium debris. They found that when the oxygen partial pressure is low, calcium is likely to dissolve into the cubic sim-debris phase to form solid solution $(Ca,U,Zr)O_{2+x}$, and also that the crystalline MgO remains affixed on the surface as the main residue of salt components, while a part of it can also dissolve into the sim-debris [3].
Because NaCl vaporizes above 800°C, this result suggests that fuel debris might not contain much chloride but contain cation products which make the fuel debris harder.

3. Risk Management for Corrosion of Cs Removal Vessel

With respect to the long-term storage of the zeolite-containing spent Cs adsorption vessels, the corrosion of the vessel is also one of the most important issues. This is because the chloride ions (Cl^-) from seawater present in the contaminated water exposed to radiation from the adsorbed radioactive Cs, can accelerate the corrosion of vessels made from type 316L stainless steel. Kato et al. investigated the localized corrosion behavior of this material and found that the coexistence of zeolite can suppress the increase in the potential under irradiation, because it causes the decomposition of H_2O_2 generated by the radiolysis of water [4].

4. Conclusion

Because decommissioning of Fukushima Dai-ichi NPS will take a long time, corrosion which risk increases fundamentally with the lapse of time, must be taken into consideration carefully.
At the accident, seawater injected to cool the core & spent fuel pool, but their chloride concentrations have been decreasing dramatically (now below 1 ppm) by the chloride removal apparatus in the water treatment system. On the other hand, at retrieving fuel debris, oxygen may dissolve in the reactor water from the opening in addition to H_2O_2 generated by water radiolysis in core. Hence, corrosion mitigation methods will be implemented in any case as long as submersion methods at fuel debris retrieval are applied.

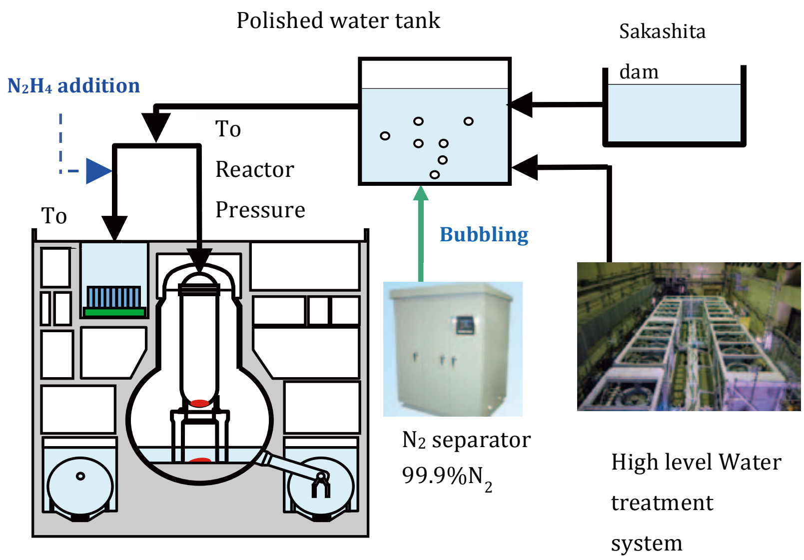

Figure　Outline of corrosion mitigation [1]

References

[1] Y.Fukaya *et al.*, Proceedings of 17th International Conference on Environmental Degradation of Materials in Nuclear Power Systems, August 2015, Ottawa (Canada).

[2] IRID Annual Research Report 2015.

[3] M.Takano *et al.*, Journal of Nuclear Materials 443 (2013), 32–39.
[4] C.Kato *et al.*, Proceedings of 17th International Conference on Environmental Degradation of Materials in Nuclear Power Systems , August 2015, Ottawa (Canada).

Research Conference on Post-accident Waste Management Safety (RCWM2016)

Corrosion risk management in Fukushima Dai-ich NPS

Nov. 8, 2016

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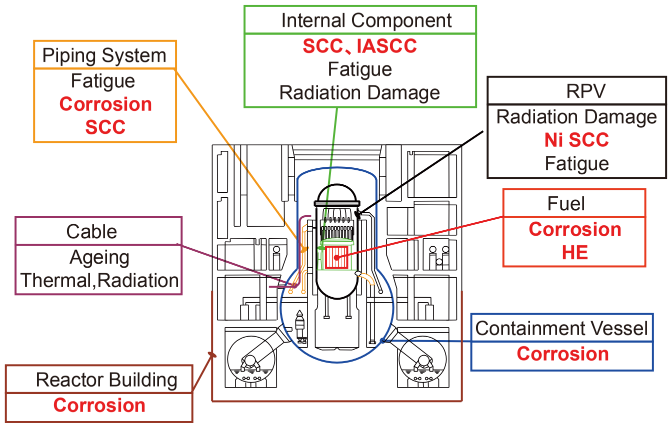
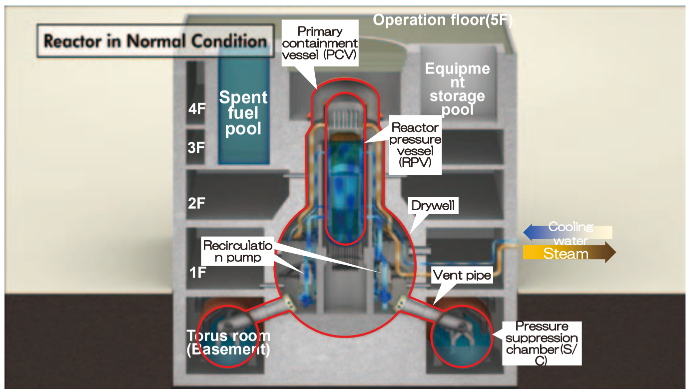
Ageing of operated Nuclear Power Plants



S.Suzuki NACE Speller Award 2013

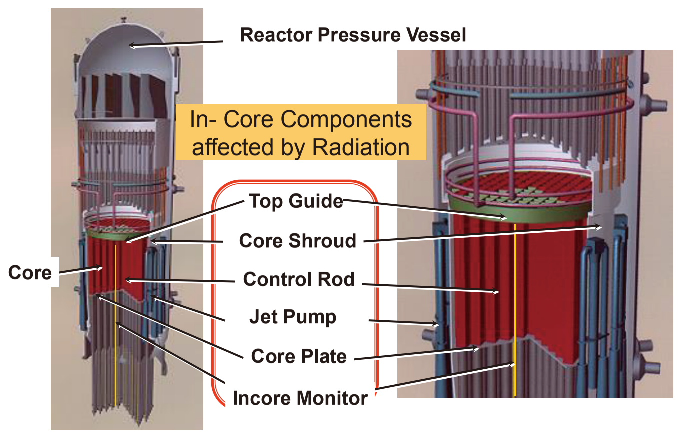
What has happened in the Fukushima Daiichi NPS (estimated)



http://irid.or.jp/en/reports/20150421/

K.Oikawa 'Robot Technologies Developed for the Decommissioning of the Fukushima Daiichi Nuclear Power Station' at the German-Japanese Symposium on Technological and Educational Resources for the Decommissioning of

Boiling Water Reactor (BWR)



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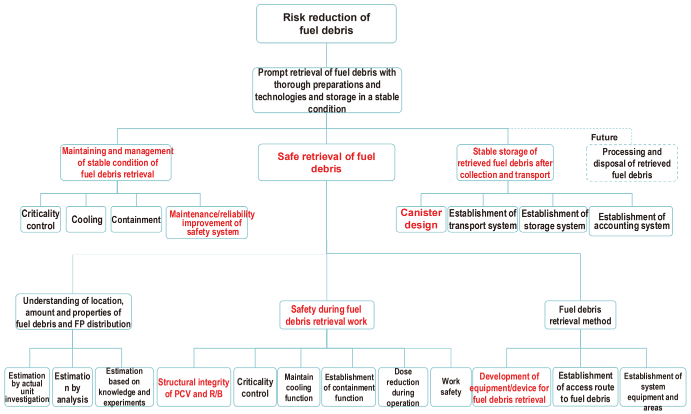
Risk Reduction of fuel debris



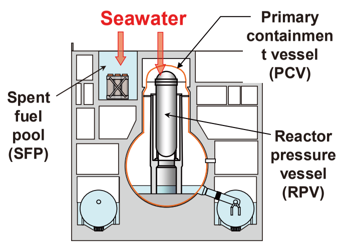
Technical Strategic Plan 2015 for Decommissioning of the Fukushima Daiichi Nuclear Power Station of Tokyo Electric Power Company Towards Amendment of the Mid-and-Long-Term Roadmap in 2015 April 30, 2015 , Nuclear Damage Compensation and Decommissioning Facilitation Corporation

Fukushima Dai-ichi NPP

Corrosion Protection

Shortly after the accident

Fukushima-Daiichi accident

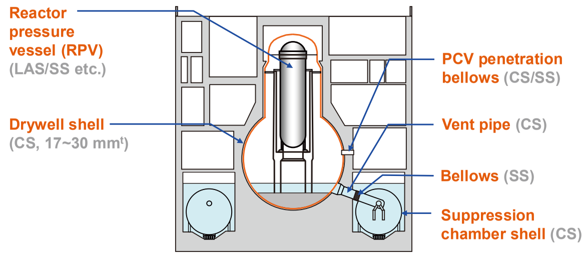


- The earthquake occurred on March 11, 2011 and the following tsunami caused the loss of the total AC power sources at Unit 1 to 4.
- Seawater was temporarily injected into the reactors and the SFPs as an emergency action for fuel cooling.
- Corrosion mitigation actions were urgently required.

Y. Fukaya,et al.,"An Overview of Corrosion Mitigation Activities after Fukushima-Daiichi Accident," Proc. of 17th International Conference on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, August, 2015, Ottawa.

Shortly after the accident

Reactor structure and materials



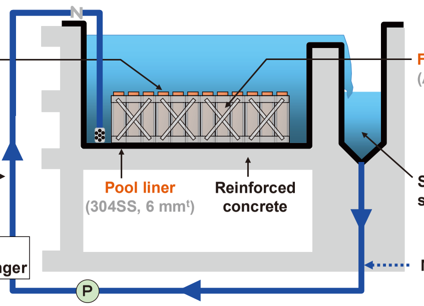
- Great concern: to stabilize the fuel cooling to maintain the **function of storing water**

Material: CS: Carbon steel, SS: Stainless steel, LAS: Low alloy steel

Y. Fukaya,et al.,"An Overview of Corrosion Mitigation Activities after Fukushima-Daiichi Accident," Proc. of 17th International Conference on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, August, 2015, Ottawa.

Shortly after the accident

SFP structure and materials



■Great concern: to stabilize the fuel cooling to maintain the function of storing water

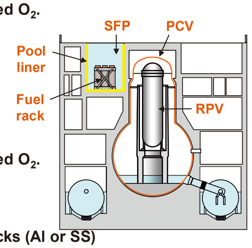
Material: CS: Carbon steel, SS: Stainless steel, LAS: Low alloy steel, Zry: Zircaloy, Al: Aluminum alloy

Y. Fukaya,et al.,"An Overview of Corrosion Mitigation Activities after Fukushima-Daiichi Accident," Proc. of 17th International Conference on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, August, 2015, Ottawa.

Shortly after the accident

Corrosion risk assessment

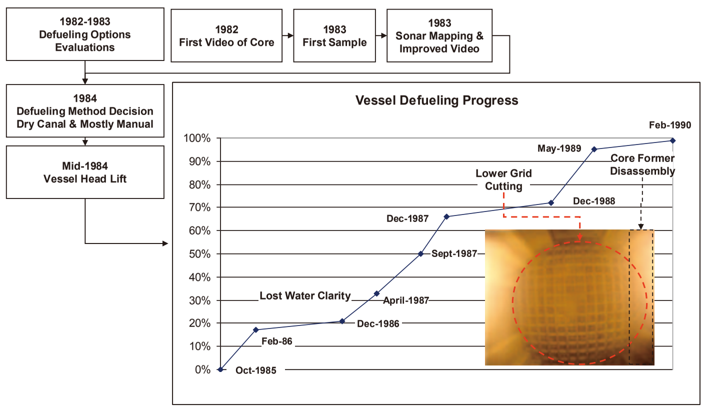
- ■ **Carbon steel (CS) / Low alloy steel (LAS)** e.g. RPV/PCV, Pipe
  - ◆ Uniform corrosion: 0.1 ~ 0.6 mm/y (< 100°C)
  - ◆ The dominant factor for corrosion rate is Dissolved O_2.
    - $Fe \rightarrow Fe^{2+} + 2e$
    - $O_2 + 2H_2O + 4e \rightarrow 4OH^-$
  - ◆ Galvanic corrosion, Flow influenced corrosion
- ■ **Stainless steel (SS)** e.g. Fuel pool liner, Pipe
  - ◆ Passivated (Corrosion rate: Negligible small)
  - ◆ Cl^- can cause localized corrosion* (~ 100 mm/y)
  - ◆ The dominant factor for corrosion rate is Dissolved O_2.
- ■ **Zircaloy (Zry)** e.g. Fuel assembly
  - ◆ Higher corrosion resistance than stainless steel
  - ◆ Cathodically protected by contacting with fuel racks (Al or SS)
- ■ **Aluminum alloy (Al)** e.g. Fuel rack
  - ◆ Uniform corrosion rate: ~ 0.03 mm/y (R.T.)
  - ◆ Galvanic corrosion, Alkaline corrosion (~ 2 mm/y at pH 11.2)
- ■ **Common**
  - ◆ Microbiologically influenced corrosion (MIC)



* Localized corrosion
- Pitting corrosion
- Crevice corrosion
- Stress corrosion cracking

Y. Fukaya,et al.,"An Overview of Corrosion Mitigation Activities after Fukushima-Daiichi Accident," Proc. of 17th International Conference on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, August, 2015, Ottawa.

Defueling Progress and Key Impacts



Chuck Negin,"Three Mile Island Unit 2 Recovery and Cleanup"

Shortly after the accident

Corrosion mitigation method

Mitigation method	Examples of actual measures
Temperature reduction	Cooling
Reduction of dissolved oxygen concentration	Gas bubbling (N_2, Ar)
	N_2H_4, Na_2SO_3
	Organic oxygen scavengers
Cathodic protection	Sacrificial anode (Al, Mg, Zn)
	Impressed current system
Reduction of salt concentration	Dilution by fresh water
	Water treatment
Corrosion inhibitor	NO_2^-, NO_3^-, PO_4^{3-}, MoO_4^{2-}, WO_4^{2-}, organic inhibitors
pH control	NaH_2PO_4, Na_3PO_4, NH_3, Na_2CO_3, NaOH (,H_3BO_3)
Sterilization	Oxidizing biocides (NaClO, H_2O_2)
	Non-oxidizing biocides (N_2H_4)

Selection criteria

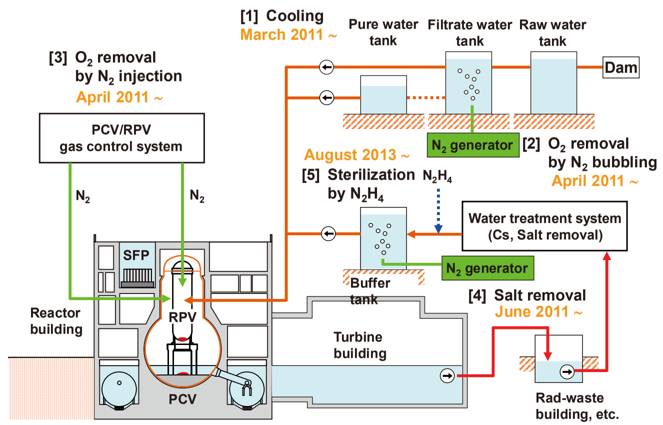
The measures in red have been applied to the reactors and/or the SFPs.

- ■can be applied from the outside of the R/Bs
- ■not affect corrosion of co-existing materials
- ■no side effect on facilities (deposition, precipitation, etc.)
- ■successful service experiences under radiation environment

Y. Fukaya,et al.,"An Overview of Corrosion Mitigation Activities after Fukushima-Daiichi Accident," Proc. of 17th International Conference on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, August, 2015, Ottawa.

Shortly after the accident

Corrosion mitigation of reactor



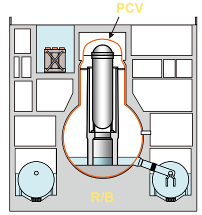
Y. Fukaya,et al.,"An Overview of Corrosion Mitigation Activities after Fukushima-Daiichi Accident," Proc. of 17th International Conference on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, August, 2015, Ottawa.

Shortly after the accident

Status in PCVs after mitigation

Accumulated water in PCVs[1]

Unit	Date of Samples	pH	Conductivity (μS/cm)	[Cl$^-$] (ppm)
1	October 12, 2012	7.2	88	19
2	August 7, 2013	7.4	25	2.9
3	October 22, 2015	6.8	14	< 1



O$_2$ gas concentrations in PCVs[2]

Unit	Date of samples	O$_2$ gas (vol%)	cf. Calculated DO** in accumulated water in PCV (ppm, 50°C)
1	May 28, 2013	0.30* ~ 0.35*	0.08 ~ 0.09
2	April 3, 2013	0.38* ~ 0.62	0.10 ~ 0.16

1) TEPCO web site. 2) Nuclear Regulation Authority web site.

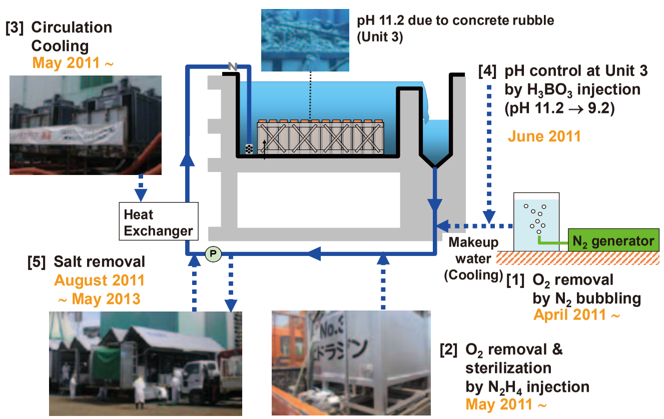
* Detection limit level

** Values from equilibrium calculation

Y. Fukaya,et al.,"An Overview of Corrosion Mitigation Activities after Fukushima-Daiichi Accident," Proc. of 17th International Conference on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, August, 2015, Ottawa.

Shortly after the accident

Corrosion mitigation of SFP



Y. Fukaya,et al.,"An Overview of Corrosion Mitigation Activities after Fukushima-Daiichi Accident," Proc. of 17th International Conference on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, August, 2015, Ottawa.

Shortly after the accident

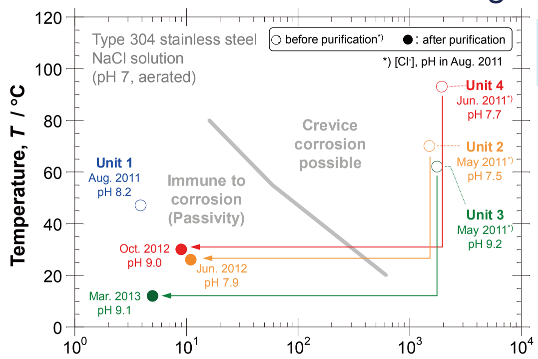
Quality of SFP water

Unit	Date of samples	pH	[Cl$^-$] (ppm)	Cs-134 (Bq/cm^3)	Cs-137 (Bq/cm^3)	Fuel rack material
1	August 19, 2011	8.2	3.9	1.8 x 10^4	2.3 x 10^4	Al
2	August 19, 2011	7.5	1508	1.1 x 10^5	1.1 x 10^5	Al
3	May 8, 2011	11.2	-	-	-	Al
3	August 19, 2011	9.2	1769	7.4 x 10^4	7.4 x 10^4	Al
4	August 20, 2011	7.7	1944	4.4 x 10^1	6.1 x 10^1	SS

Y. Fukaya,et al.,"An Overview of Corrosion Mitigation Activities after Fukushima-Daiichi Accident," Proc. of 17th International Conference on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, August, 2015, Ottawa.

Shortly after the accident

Status in SFPs after mitigation



Additional measures;
- O_2 removal
- Sterilization

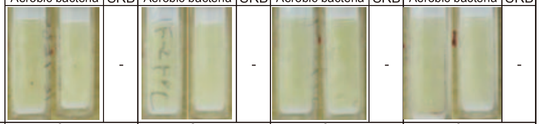
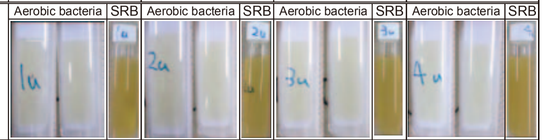
Comparison of water quality of spent fuel pool with the crevice corrosion diagram[1)] for Type 304 stainless steel in aerated condition.

1) M. Akashi, G. Nakayama, T. Fukuda: CORROSION/98 Conf.,NACE International, Paper No. 158 (1998).

Y. Fukaya,et al.,"An Overview of Corrosion Mitigation Activities after Fukushima-Daiichi Accident," Proc. of 17th International Conference on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, August, 2015, Ottawa.

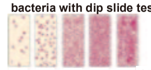
Shortly after the accident

Sterilization effect in SFP

	SFP at Unit 1		SFP at Unit 2		SFP at Unit 3		SFP at Unit 4	
	Aerobic bacteria	SRB	Aerobic bacteria	SRB	Aerobic bacteria	SRB	Aerobic bacteria	SRB
August 2011		-		-		-		-
Result	$< 10^3$ CFU/ml		$< 10^3$ CFU/ml		$< 10^3$ CFU/ml		$< 10^3$ CFU/ml	
	Aerobic bacteria	SRB	Aerobic bacteria	SRB	Aerobic bacteria	SRB	Aerobic bacteria	SRB
July 2012								
Result	$< 10^3$ CFU/ml	non	$< 10^3$ CFU/ml	non	$< 10^3$ CFU/ml	non	$< 10^3$ CFU/ml	non

- The number of the aerobic bacteria was less than 10^3 CFU/ml and no sulfate-reducing bacteria was detected.

Assessment of contamination for aerobic bacteria with dip slide test (Easicult TTC)



1) T. Hirasaki, S. Namatame, Y. Fukaya, and K. Takamori, Proc. 59th Japan Conf. Materials and Environments, Japan society of corrosion engineering, p. 119 (2012).

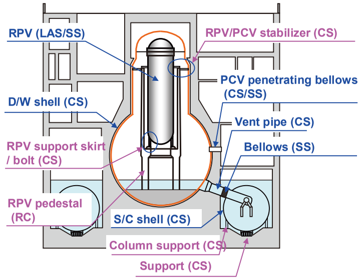
Y. Fukaya,et al.,"An Overview of Corrosion Mitigation Activities after Fukushima-Daiichi Accident," Proc. of 17th International Conference on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, August, 2015, Ottawa.

Mid- and long-term issues

Assessment of RPV/PCV Integrity

- Development of technologies for assessing structural integrity of RPV/PCV
  - Long-term corrosion assessment of PCV and cooling water pipes
  - Development of corrosion mitigation method available under high radiation environment
  - Corrosion of reinforced concrete that composes a RPV pedestal

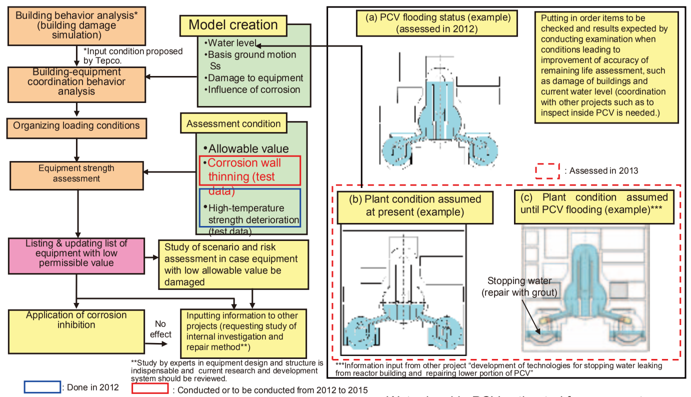
※CS: Carbon steel, LAS: Low-alloy steel, SS: Stainless steels etc., RC: Reinforced concrete



Importance: Function of storing water Structural integrity

Y. Fukaya,et al.,"An Overview of Corrosion Mitigation Activities after Fukushima-Daiichi Accident," Proc. of 17th International Conference on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, August, 2015, Ottawa.

Overall structural integrity assessment flow



Outline flow of remaining life assessment (example)

Water level in PCV estimated from present situation and until PCV is flooded (example)

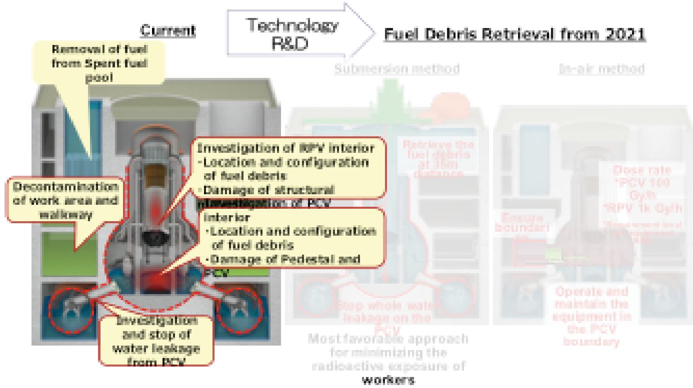
http://irid.or.jp/en/reports/20150409/

T.Sato, 'R&D Activities Related to the Fuel Debris Retrieval from the Fukushima Daiichi NPS' to the 2nd Japan-U.S. Decommissioning and Remediation Fukushima Recovery Forum, April 9, 2015

Decommissioning of Fukushima Dai-ichi NPP

- Fuel debris retrieval, transfer & storage -

Fuel debris retrieval procedure



K.Oikawa 'Robot Technologies Developed for the Decommissioning of the Fukushima Daiichi Nuclear Power Station' at the German-Japanese Symposium on Technological and Educational Resources for the Decommissioning of Nuclear Facilities ,April 27,2015

Comparison : Fuel debris in Fukushima Daiichi and TMI 2

·Burnup and enrichment are higher in 1F
⇒ *Radiation, decay heat and reactivity are higher*
·Fuel debris in 1F contain molten core concrete interaction product
⇒ *Concern of hydrogen generation due to water radiolysis in concrete*
·Seawater injection to the reactor, melting along with the instrumentation cables
⇒ *Effect of salt and kinds of impurities contaminated in the fuel debris*

		1 F	TMI-2
Burnup (Reactor core average)		About 25.8GWd/t	approx. 3.2GWd/t
Enrichment (bundle average (max))		3.7 wt%	2.96 wt%
Cooling period (minimum)		About 9 years (as of June 2020)	About 6 years
Fuel debris location		Inside RPV and PCV (supposition) ⇒Molten interaction with concrete and instrumentation cables as well as fuel constructional material and internal structures are supposed to exist.	Inside RPV ⇒Molten interaction with fuel constructional material and internal structures.
Quantity	Debris	–	134.4t
	(Fuel Assembly)	About more than 460t in total of 3 units	About 122t
	(Uranium) (unirradiated)	About more than 260t in total of 3 units	About 82t
Others		Seawater was injected to the reactor	–

http://irid.or.jp/en/reports/20150419/

T.Sato, 'R&D Activities Related to the Fuel Debris Retrieval from the Fukushima Daiichi NPS' to the 2nd Japan-U.S. Decommissioning and Remediation Fukushima Recovery Forum, April 9, 2015

High temperature reaction between sea salt deposit and U-Zr-O simulated fuel debris

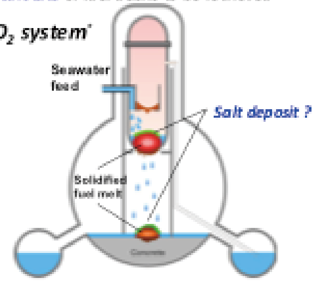
Severe accident at Fukushima Daiichi Nuclear Power Plant, Mar. 2011

- Seawater was injected into cores for urgent cooling
- Salt deposit may have formed on the high temperature debris (crust, corium)
- Various temperature history and oxygen partial pressure depending on position
- Clarify the possible impacts on chemical and physical state of fuel debris to be retrieved

*Preliminary experiments on sea salt–UO_2 system**

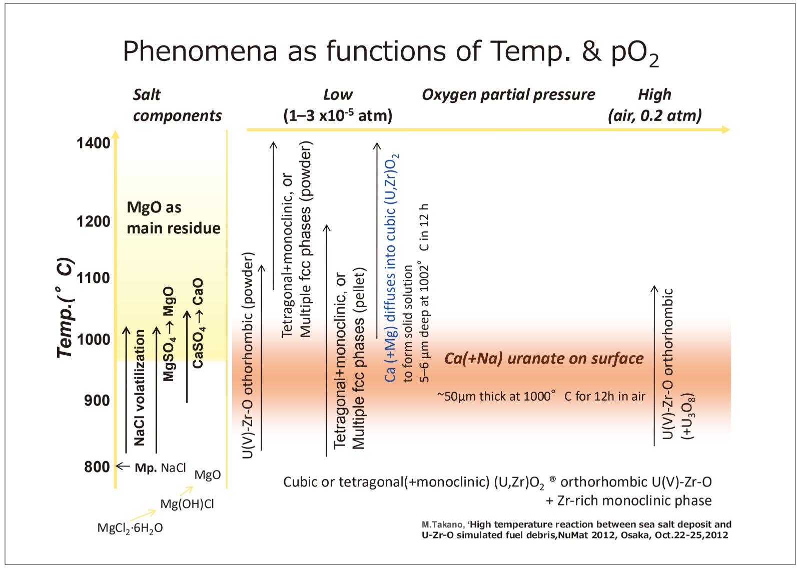
- NaCl vaporization > 800° C
- Corrosive gas evolution (HCl, SO_x)
- MgO as main residue at high temp.
- Ca and a part of Mg soluble into UO_2
  - * *much smaller lattice parameter than UO_2*

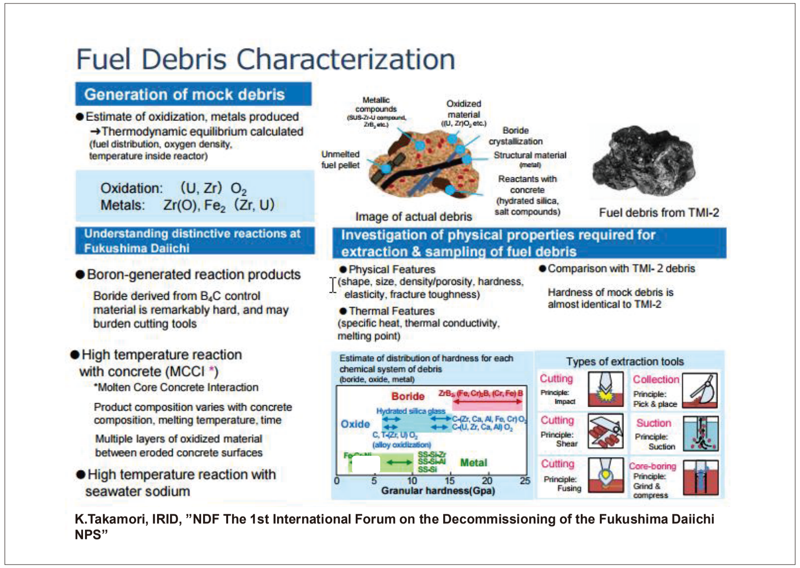
Experiments on sea salt deposit and $(U,Zr)O_2$ simulated fuel debris (sim-debris), as the main constituent of actual debris

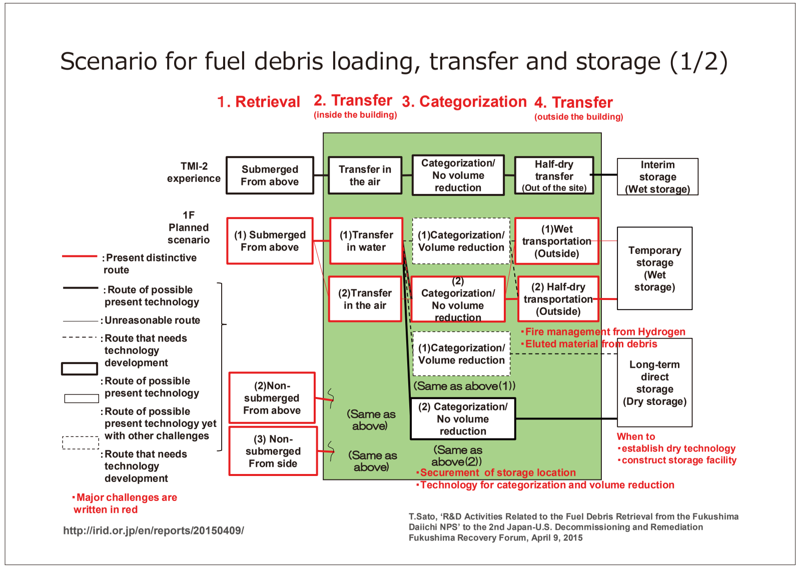


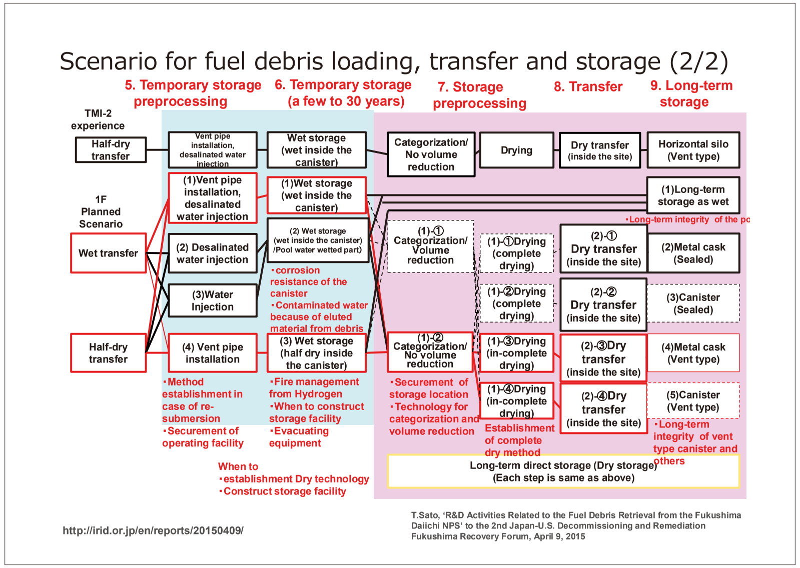
Imaginary drawing of inside PCV

*M. Takano et al., "High temperature reaction between UO_2 and sea salt deposit",ERMSAR 2012, Köln, Germany, Mar. 21-23, 2012, paper 7.2.


Phenomena as functions of Temp. & pO2
Salt components
Low (1–3 x10-5 atm)
Oxygen partial pressure
High (air, 0.2 atm)
Temp.(° C)
1400
1200
1100
1000
900
800
MgO as main residue
NaCl volatilization
MgSO4 → MgO
CaSO4 → CaO
Mp. NaCl
MgO
Mg(OH)Cl
MgCl2·6H2O
U(V)-Zr-O orthorhombic (powder)
Tetragonal+monoclinic, or Multiple fcc phases (powder)
Tetragonal+monoclinic, or Multiple fcc phases (pellet)
Ca (+Mg) diffuses into cubic (U,Zr)O2 to form solid solution 5–6 μm deep at 1002° C in 12 h
Ca(+Na) uranate on surface
~50μm thick at 1000° C for 12h in air
U(V)-Zr-O orthorhombic (+U3O8)
Cubic or tetragonal(+monoclinic) (U,Zr)O2 ® orthorhombic U(V)-Zr-O + Zr-rich monoclinic phase
M.Takano, 'High temperature reaction between sea salt deposit and U-Zr-O simulated fuel debris,NuMat 2012, Osaka, Oct.22-25,2012


Fuel Debris Characterization
Generation of mock debris
Estimate of oxidization, metals produced
→Thermodynamic equilibrium calculated (fuel distribution, oxygen density, temperature inside reactor)
Oxidation: (U, Zr) O2
Metals: Zr(O), Fe2 (Zr, U)
Unmelted fuel pellet
Metallic compounds
Oxidized material
Boride crystallization
Structural material (metal)
Reactants with concrete (hydrated silica, salt compounds)
Image of actual debris
Fuel debris from TMI-2
Understanding distinctive reactions at Fukushima Daiichi
Boron-generated reaction products
Boride derived from B4C control material is remarkably hard, and may burden cutting tools
High temperature reaction with concrete (MCCI *)
*Molten Core Concrete Interaction
Product composition varies with concrete composition, melting temperature, time
Multiple layers of oxidized material between eroded concrete surfaces
High temperature reaction with seawater sodium
Investigation of physical properties required for extraction & sampling of fuel debris
Physical Features (shape, size, density/porosity, hardness, elasticity, fracture toughness)
Thermal Features (specific heat, thermal conductivity, melting point)
Comparison with TMI- 2 debris
Hardness of mock debris is almost identical to TMI-2
Estimate of distribution of hardness for each chemical system of debris (boride, oxide, metal)
Boride
Oxide
Metal
Granular hardness(Gpa)
Types of extraction tools
Cutting Principle: Impact
Collection Principle: Pick & place
Cutting Principle: Shear
Suction Principle: Suction
Cutting Principle: Fusing
Core-boring Principle: Grind & compress
K.Takamori, IRID, "NDF The 1st International Forum on the Decommissioning of the Fukushima Daiichi NPS"

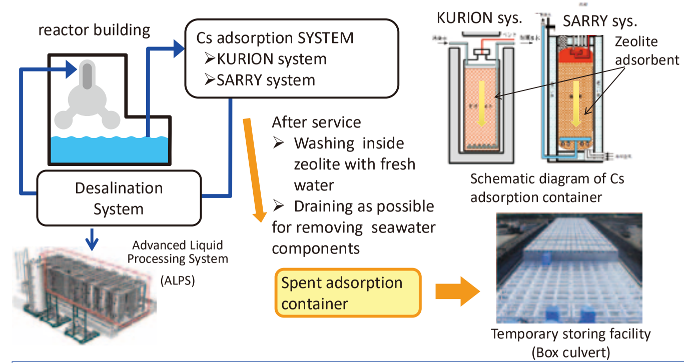

Scenario for fuel debris loading, transfer and storage (1/2)
1. Retrieval
2. Transfer (inside the building)
3. Categorization
4. Transfer (outside the building)
TMI-2 experience
Submerged From above
Transfer in the air
Categorization/ No volume reduction
Half-dry transfer (Out of the site)
Interim storage (Wet storage)
1F Planned scenario
(1) Submerged From above
(1)Transfer in water
(1)Categorization/ Volume reduction
(1)Wet transportation (Outside)
Temporary storage (Wet storage)
(2)Transfer in the air
(2) Categorization/ No volume reduction
(2) Half-dry transportation (Outside)
(1)Categorization/ Volume reduction
(Same as above(1))
Long-term direct storage (Dry storage)
(2)Non-submerged From above
(Same as above)
(2) Categorization/ No volume reduction
(3) Non-submerged From side
(Same as above)
(Same as above(2))
:Present distinctive route
:Route of possible present technology
:Unreasonable route
:Route that needs technology development
:Route of possible present technology
:Route of possible present technology yet with other challenges
:Route that needs technology development
・Fire management from Hydrogen
・Eluted material from debris
・Securement of storage location
・Technology for categorization and volume reduction
When to
・establish dry technology
・construct storage facility
・Major challenges are written in red
http://irid.or.jp/en/reports/20150409/
T.Sato, 'R&D Activities Related to the Fuel Debris Retrieval from the Fukushima Daiichi NPS' to the 2nd Japan-U.S. Decommissioning and Remediation Fukushima Recovery Forum, April 9, 2015


Scenario for fuel debris loading, transfer and storage (2/2)
5. Temporary storage preprocessing
6. Temporary storage (a few to 30 years)
7. Storage preprocessing
8. Transfer
9. Long-term storage
TMI-2 experience
Half-dry transfer
Vent pipe installation, desalinated water injection
Wet storage (wet inside the canister)
Categorization/ No volume reduction
Drying
Dry transfer (inside the site)
Horizontal silo (Vent type)
1F Planned Scenario
Wet transfer
Half-dry transfer
(1)Vent pipe installation, desalinated water injection
(2) Desalinated water injection
(3)Water Injection
(4) Vent pipe installation
(1)Wet storage (wet inside the canister)
(2) Wet storage (wet inside the canister) /Pool water wetted part)
(3) Wet storage (half dry inside the canister)
(1)-① Categorization/ Volume reduction
(1)-② Categorization/ No volume reduction
(1)-①Drying (complete drying)
(1)-②Drying (complete drying)
(1)-③Drying (in-complete drying)
(1)-④Drying (in-complete drying)
(2)-① Dry transfer (inside the site)
(2)-② Dry transfer (inside the site)
(2)-③Dry transfer (inside the site)
(2)-④Dry transfer (inside the site)
(1)Long-term storage as wet
(2)Metal cask (Sealed)
(3)Canister (Sealed)
(4)Metal cask (Vent type)
(5)Canister (Vent type)
・Long-term integrity of the pc
・corrosion resistance of the canister
・Contaminated water because of eluted material from debris
・Method establishment in case of re-submersion
・Securement of operating facility
・Fire management from Hydrogen
・When to construct storage facility
・Evacuating equipment
・Securement of storage location
・Technology for categorization and volume reduction
Establishment of complete dry method
・Long-term integrity of vent type canister and others
Long-term direct storage (Dry storage) (Each step is same as above)
When to
・establishment Dry technology
・Construct storage facility
http://irid.or.jp/en/reports/20150409/
T.Sato, 'R&D Activities Related to the Fuel Debris Retrieval from the Fukushima Daiichi NPS' to the 2nd Japan-U.S. Decommissioning and Remediation Fukushima Recovery Forum, April 9, 2015

Waste Management of Fukushima Dai-ichi NPP

- Storage of Contaminated water -

Coolant treatment system for the contaminated water



The spent containers are stored in temporary storage facilities.
The containers may store for a long period until their final disposal (~2030 in schedule)

C. Kato,et al.,"Effects of Gamma-Ray Irradiation on Spontaneous Potential of Stainless Steel in Zeolite-Containing Diluted Artificial Seawater," Proc. of 17th International Conference on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, August, 2015, Ottawa

Corrosion issue of the spent adsorption vessels

✓The contaminated water had contained seawater components.

✓Type 316L stainless steel has been used for the vessel material.

✓In the aspect of the long-term storage, corrosion may is an important issues.

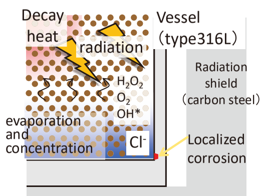
➢Cl^- in remaining water can be concentrated by the decay heat.

➢The radiation can accelerate the corrosion due to water radiolysis oxidants (H_2O_2, O_2) and radicals (OH*).

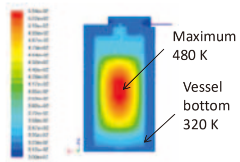
Table 1 thermal analyses and the dose rate at bottom in reference spent KURION vessel*

Decay heat (W / vessel)	Temperature in vessel bottom(K)	Dose rate in vessel bottom (Gy/h)
504 W	320 K	750 Gy/h

*I. Yamagishi, Journal of Nuclear Science and Technology, 51, 1044-1053 (2014).



Schematic diagrams of the vessel bottom



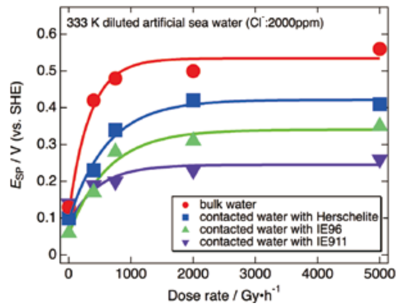
Thermal analyses in the reference spent KURION vessel*

C. Kato,et al.,"Effects of Gamma-Ray Irradiation on Spontaneous Potential of Stainless Steel in Zeolite-Containing Diluted Artificial Seawater," Proc. of 17th International Conference on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, August, 2015, Ottawa

Effect of dose rate in E_{SP}

Order in E_{SP} under irradiation :

bulk water > Herschelite > IE96 > IE911



The relationship between dose rate and the spontaneous potential, E_{SP}, in the bulk water and in contacted with the zeolites

➢ In non irradiation, E_{SP} were almost equal.

➢ Under irradiation, E_{SP} increased with increasing dose rate and the increase in E_{SP} was saturated at a high dose rate.

➢ E_{SP} in contact with zeolites was less noble than in bulk water under the same irradiation conditions.

⇒The contact with the zeolite can suppress the increase in E_{SP} under irradiation

C. Kato,et al.,"Effects of Gamma-Ray Irradiation on Spontaneous Potential of Stainless Steel in Zeolite-Containing Diluted Artificial Seawater," Proc. of 17th International Conference on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, August, 2015, Ottawa

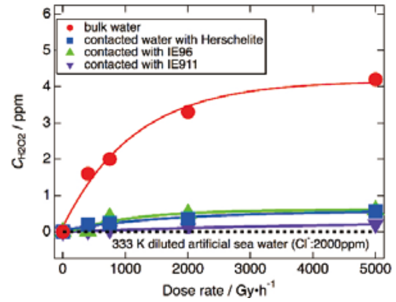
Water radiolysis reactions

	phenomena	Time scale
Initial reaction	H_2O → excitation / ionization	
	excitation → H_2O^*; ionization → H_2O^+ + e^-	10^{-16}s
	H_2O^* → H_2O, H + OH; H_2O^+ + H_2O → H_3O^+ + OH; e^- → e_{aq}^-	10^{-14}s, 10^{-13}s
	reaction → e_{aq}^-, H, OH, H_2, H_2O_2, H_3O^+	10^{-7}s
Secondary reaction (About 40 reactions)	$e_{aq}^- + O_2 \rightarrow O_2^-$ $e_{aq}^- + H_2O_2 \rightarrow OH + OH^-$ $OH + H_2 \rightarrow H + H_2O$ $H + H_2O_2 \rightarrow H_2O + OH$ $H + O_2 \rightarrow HO_2$ $HO_2 + HO_2 \rightarrow H_2O_2 + O_2$	O_2 and H_2O_2 is most important species for evaluation

S.Suzuki, Seminar, Institute of fluid science ot Tohoku Univ.

H_2O_2 concentration after irradiation

The order of H_2O_2 concentrations under irradiation :
bulk water > Herschelite ≒ IE96 > IE911
This order agreed with the order in yielded from E_{SP} values



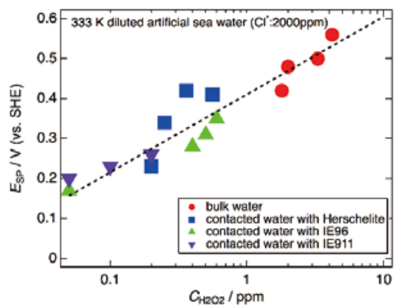
the relationship between dose rate and hydrogen peroxide concentration, C_{H2O2}, in the bulk water and the contacted waters with the zeolite after gamma-rays irradiation.

- H_2O_2 concentration in bulk water increased exponentially with increasing dose rate
- In contact water with zeolites, H_2O_2 concentration was much lower than in bulk water.

⇒The contact with zeolites causes suppression of forming H_2O_2 under irradiation.

C. Kato,et al.,"Effects of Gamma-Ray Irradiation on Spontaneous Potential of Stainless Steel in Zeolite-Containing Diluted Artificial Seawater," Proc. of 17th International Conference on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, August, 2015, Ottawa

Relationship between E_{SP} and C_{H2O2}



The relationship between the spontaneous potential, E_{SP}, and concentration of hydrogen peroxide, C_{H2O2} in bulk water and in the contact water with the zeolites.

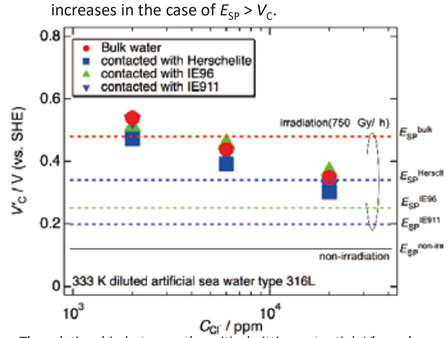
- E_{SP} changed almost linearly with the logarithm of H_2O_2 concentration.
- There is a clear relationship between E_{SP} and H_2O_2 concentration.

⇒The increase in E_{SP} under irradiation was primarily dependent on the H_2O_2 concentration

C. Kato,et al.,"Effects of Gamma-Ray Irradiation on Spontaneous Potential of Stainless Steel in Zeolite-Containing Diluted Artificial Seawater," Proc. of 17th International Conference on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, August, 2015, Ottawa

Effects of the contact with zeolites on localized corrosion initiation

The possibility of occurrence in localized corrosion increases in the case of $E_{SP} > V_C$.



The relationship between the critical pitting potential, V'_C, and concentration of chloride ion, C_{Cl^-}, in the bulk water and in the contact with zeolites. Dashed lines are the E_{SP} at 750 Gy/h and a solid line is E_{SP} at non-irradiation.

- V_C was almost the same in bulk water in the contact with zeolites.

⇒The contact with zeolites don't affect localized corrosion.

- Order in E_{SP} under irradiation is
  bulk water > Herschelite > IE96 > IE911 > (non-irradiation)

⇒The contact with the zeolites can reduce the probability of occurrence in localized corrosion under irradiation.

C. Kato,et al.,"Effects of Gamma-Ray Irradiation on Spontaneous Potential of Stainless Steel in Zeolite-Containing Diluted Artificial Seawater," Proc. of 17th International Conference on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, August, 2015, Ottawa

What is “Risk” ?

$$\text{Risk} = \text{Hazard} \times \frac{\text{Fragility}}{\text{Resilience}}$$

- ✓ Solution of gap between Technology and Social acceptability for nuclear decommissioning
- ✓ Stakeholders’ Involvement

Thinking the way of predicting the future

- ➢ In order to accomplish decommissioning of the Fukushima dai-ichi NPP, challenges which no one has ever experienced will be essential.
- ➢ The key factors to improve the normal construction and operation/maintenance of NPP is how to solve the current constant problems based on the past experiences and new technology. On the other hand, decommissioning of the accident plants requires finding solutions for unpredictable & unexpected problems which changes with time.
- ➢ The other concepts such as the application of Hypothesis driven Thinking may give the hint to lead to success and it is expected that the acquired knowledge, experiences and lessons learned from decommissioning will contribute on solving the difficult missions in the other fields.

– Conclusion –

- In order to measure Corrosion Behavior, which is the stochastic process in nature, risk analyses are essential for its prediction, so that Bayesian analyses have been introduced for the assessment of corrodible structures
- On the other hand, difficult problems still remain for the analyses such as how to predict the unexpected events in future ?
- However, we may prepare for important uncertainties with prediction by washing out the serious events and by proving the hypothesis of the prediction by monitoring them
- Evaluation of material ageing may require such kinds of approaches.

3.1.3 Radiolytic Corrosion of Carbon Steel and Irradiation Facilities
- The effect of pH on the radiation corrosion of iron in dilute chloride solutions **-**

Hiroyuki INOUE [1], Ryuichiro IDEHARA [2], Takao KOJIMA [3] and Akihiro IWASE [4]
[1] *Osaka prefecture university; hiro@mtr.osakafu-u.ac.jp*
[2] *Osaka prefecture university (present affiliation: Daikin Industries, Ltd.)*
[3] *Radiation research center, Osaka prefecture university,* [4] *Osaka prefecture university*

Objective

The effect of the solution pH on the radiation corrosion of iron in a dilute chloride solution was investigated to obtain fundamental experimental data for evaluating the radiolytic corrosion of carbon steel, which may contribute to ensure the long-term stability of the facilities in the damaged Fukushima Dai-Ichi nuclear power plant.

Immersion test under gamma-ray irradiation

The immersion test of an iron sample was carried out for 3 days in Ar-gas atmosphere with irradiating gamma rays, whose absorbed dose rate was 1.8, 12 and 40 kGy/h. A 1mM NaCl solution with a variety of pH was used for the solution; a certain amount of HCl, H_3BO_3, Na_2SO_4, NaOH and $Ca(OH)_2$ was added to a 1mM NaCl solution to modify its pH from 3 to 13. The iron plate (purity : >4N), whose dimensions were 20 x 10 x 1 mm was used for the sample. The volume of the solution was 80mL, so the volume-to-specimen's surface area ratio was roughly 20 mL/cm^2. The sample was supported by the fine glass-rod whose end worked to a hook-shape, which was hooked on to a hole that was drilled at the top of the sample. The corrosion rate was measured with the weight-loss of the immersed sample. The immersion test without irradiating gamma rays was also carried out for comparison.

Fig.1 shows the obtained results. The corrosion of the sample was retarded when the solution pH was within a near-neutral range, regardless of the intensities of the irradiated gamma rays. However, the corrosion in a weak-alkaline solution, whose pH was around 10 was clearly accelerated with an increase in the intensity of irradiated gamma rays, especially of which absorbed dose rate was 12 kGy/h or higher. For an alkaline solution whose pH was larger than 10, the corrosion rate of the sample was decreased with an increase in the solution pH. The surface morphology of the samples after finishing the measurements was examined with a loupe. The surfaces of the sample that had been immersed in the solutions of which pH was 3 to 4 or around 10 were generally attacked. However, no corrosion traces were admitted on the samples that had been immersed in the solution of which pH was either a near-neutral level or 12 and over, even under the intense gamma ray irradiation of which

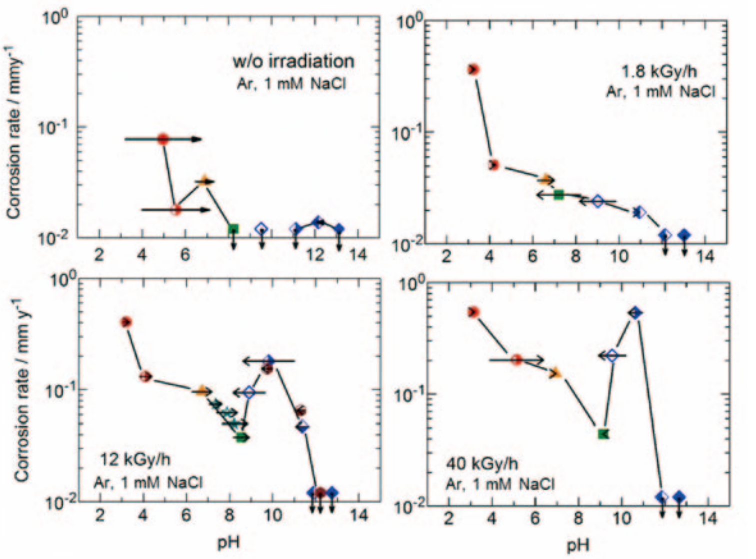


Fig.1 Change in corrosion rate of the iron in a 1 mM NaCl solution that was added a certain amount of acid and base species as a function of the pH under the gamma-ray irradiation of 1.8, 12 and 40 kGy/h as well as without irradiation.

absorbed dose rate was as high as 40 kGy/h.

According to the results above shown, it may be recommendable that the pH of the circulating water should be controlled within a near-neutral range, especially in the place where the irradiation of higher absorbed dose rate is expected. The corrosion damages on reinforcing steel may be concerned if the cover-concreate on the steel had been heavily damaged and the pH of pore water there was decreased as low as 10 by the neutralization of the concreate.

Concentration of dissolved-oxygen and hydrogen peroxide with pH

Fig.2 shows the relationships between the corrosion rate and the concentrations of the oxidizing agents, dissolved-oxygen and hydrogen peroxide as a function of the solution pH. The corrosion rate of the sample was also evaluated by the loss of weight. The concentrations of the oxidizing agents were measured just after finishing the immersion test under the gamma ray irradiation of 8.6 kGy/h for 3 days. The dissolved-oxygen and the hydrogen peroxide concentrations were measured with a florescent-type dissolved-oxygen analyzer and a hydrogen peroxide analyzer using the coulometric titration method as a principle of operation, respectively.

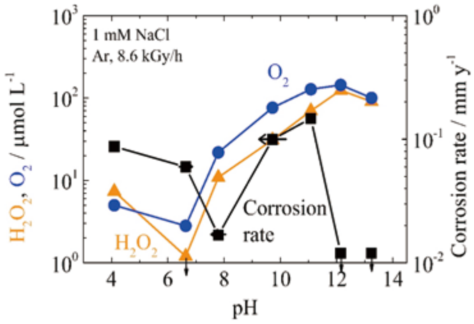


Fig.2 Change in hydrogen peroxide and dissolved oxygen as well as the corrosion rate of iron as a function of the solution pH under gamma-ray irradiation of 8.6 kGy/h

The profile of a change in the concentration of the dissolved-oxygen and the hydrogen peroxide was properly fitted to that in the corrosion rate of the simultaneously immersed sample, within the pH range from near-neutral to about 10.

Electrochemical impedance spectroscopy (EIS) study on the radiation corrosion of iron

The charge-transfer resistance of iron electrode under irradiation in the solution of different pH conditions was measured by the EIS technique. The measurement was carried out under the gamma rays irradiation of 31 kGy/h just after 3 days immersion tests, which had been carried out under the gamma rays irradiation of the same intensity. The obtained impedance spectrums were shown in Fig.3. The corrosion rates were estimated from the charge transfer resistances that were obtained from the impedance spectrums shown in Fig.3.

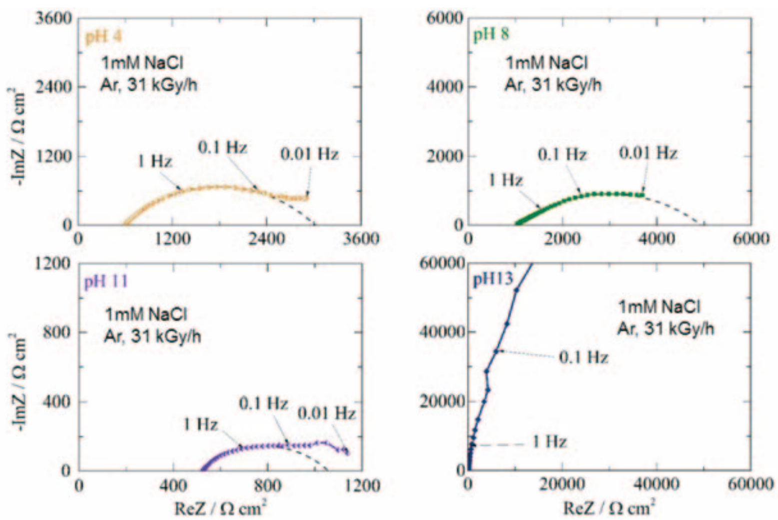


Fig. 3 Impedance spectrum of the iron electrode in a 1 mM NaCl solution of which pH was modified 3, 8, 11 and 13 under the gamma-ray irradiation of 31 kGy/h; the measurements were carried out just after the immersion test for 3 days under the irradiation at the same intensity.

In general, the reducing reaction of an iron electrode for a solution containing certain concentrations of dissolved oxygen and hydrogen peroxide proceeds with either a 4-electron or a 2-electron reaction. It was suggested that the corrosion reaction for the investigated system would proceed with the 4-electron reaction; the reduction of the dissolved-oxygen as well as the hydrogen peroxide evenly acted as the oxidizers for corrosion reaction. The details will be shown in the presentation.

According to the above results the concentration of dissolved-oxygen as well as that of hydrogen peroxide should be focused if it is attempted to monitor the corrosion rate of the structural member of steel structure by the monitoring of the oxidizing species that are induced by the radiolysis of water.

Radiolytic Corrosion of Carbon Steel and Irradiation Facilities

- The effect of pH on the radiation corrosion of iron in dilute chloride solutions -

Hiroyuki Inoue, Ryuichiro Idehara*, Takao Kojima**, Akihiro Iwase

Osaka Prefecture University

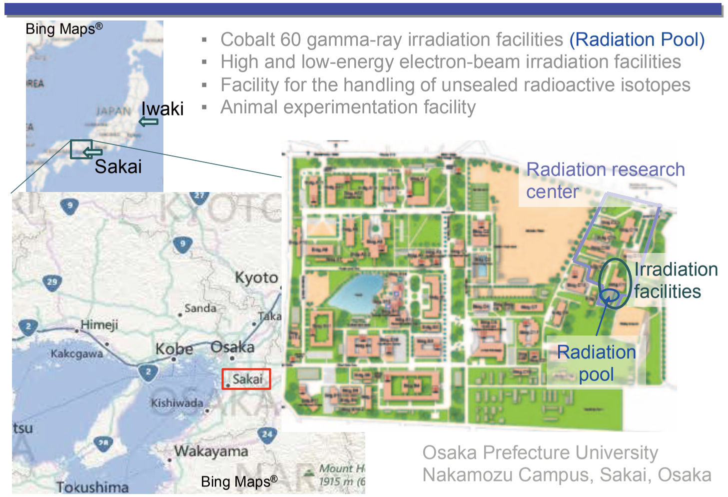
* Present Affiliation: Daikin Industries, Ltd.
** Radiation Research Center, OPU

Topics

- Introduction of the irradiation facility
  (Radiation Research Center , Osaka Prefecture University)
- The effect of pH on the radiation corrosion of iron in dilute chloride solutions
  - Immersion test under gamma-ray irradiation.
  - Concentration of dissolved-oxygen and hydrogen peroxide with pH.
  - Electrochemical impedance spectroscopy (EIS) study on the radiation corrosion of iron.

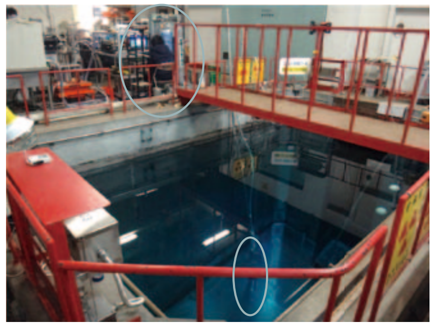
Location of Osaka Prefecture University and Radiation Research Center

- Cobalt 60 gamma-ray irradiation facilities (Radiation Pool)
- High and low-energy electron-beam irradiation facilities
- Facility for the handling of unsealed radioactive isotopes
- Animal experimentation facility



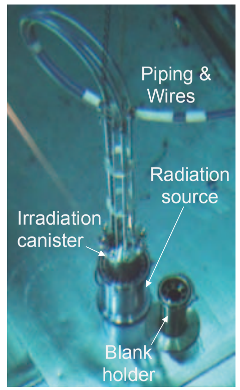
Osaka Prefecture University
Nakamozu Campus, Sakai, Osaka

Irradiation pool, radiation source and irradiation canister



Irradiation pool (5m depth)

3 ^{60}Co radiation sources are placed
(Absorbed dose rate: 26, 7.4, 3.1 kGy/h)
January, 2016



Canister for electrochemical measurement inserted in ^{60}Co radiation source

Apparatuses equipped at the facility

Potentiostat with FRA

Electrochemical impedance spectroscopy study

Fluorescent dissolved-oxygen analyzer

Hydrogen peroxide analyzer

Principle of operation; Coulometric titration of iodine

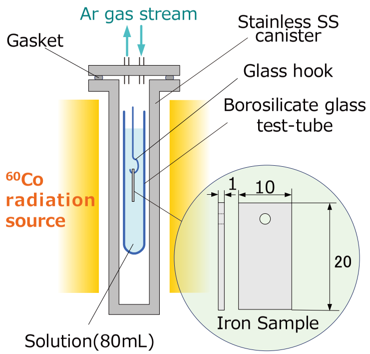
pH meter

Ion-exchange chromatograph

Objective

The effect of the solution pH on the radiation corrosion of iron in a dilute chloride solution was investigated to obtain fundamental experimental data for evaluating the radiolytic corrosion of carbon steel, which may contribute to ensure the long-term stability of the facilities in the damaged Fukusima Dai-Ichi nuclear power plant.

Test methods



Volume-to-specimen surface area ratio; roughly 20 mL/cm^2

Sample : Iron (purity:>4N)

Solution : 1 mM NaCl, 80mL
(pH : 3 to 13)

Absorbed dose rate : 1.8, 12 and 40 kGy/h w/o irradiation

Immersion period : 3 days

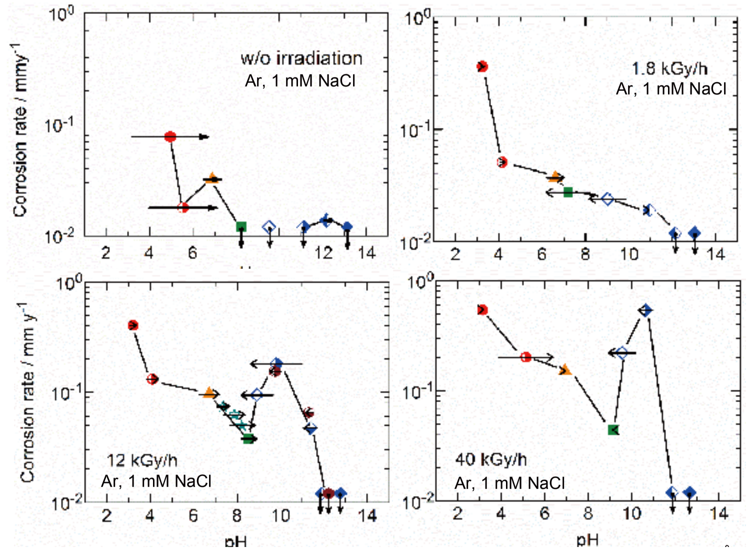
Temperature : Ambient temp.

Corrosion rate measurement : Weight loss of the sample for the immersion period

Acid and base species added to the solution to modify its pH

1mM NaCl solution	w/o irrad.	1.8	12	40
■ : No additive	○	○	○	○
● : + 1.0 mM HCl	○	○	○	○
◐ : + 0.1mM HCl	○	○	○	○
▲ : + 10 mM H_3BO_3	○	○	○	○
★ : + 0.11 mM Na_2SO_4			○	
☆ : + 1.1 mM Na_2SO_4			○	
☆ : + 11 mM Na_2SO_4			○	
◇ : + 0.1 mM NaOH	○	○	○	○
◆ : + 1.0 mM NaOH	○	○	○	○
◆ : + 5.0 mM NaOH			○	
◆ : + 10 mM NaOH	○	○	○	○
◆ : + 100 mM NaOH	○	○	○	○

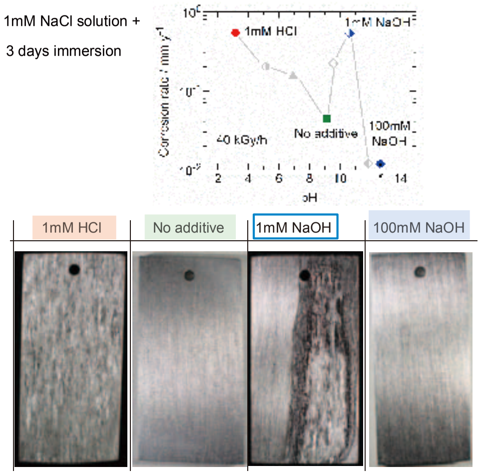
Absorbed dose rate / kGyh^{-1}



The surface morphology of the samples after finishing the tests

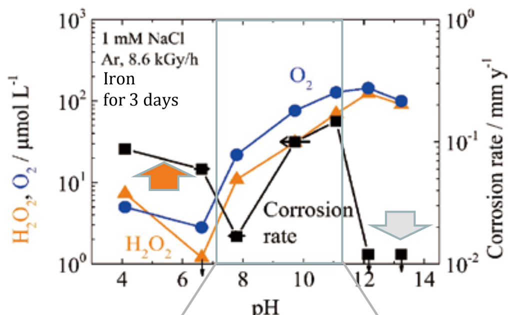
1mM NaCl solution +

3 days immersion



pH 3 to 4 and around 10;
generally attacked

Concentration of H_2O_2 and O_2 as well as corrosion rate of iron with pH



Fenton's reaction

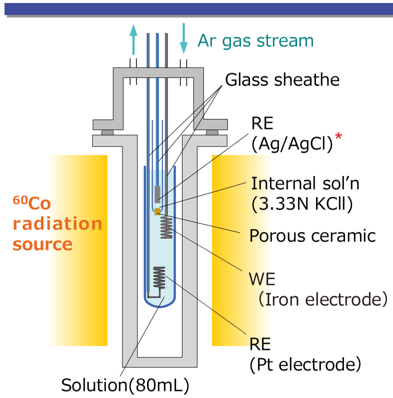
$Fe^{2+} - e^- \rightarrow \boxed{Fe^{3+}}$

$[O_2]$, $[H_2O_2]$

($[O_2] > [H_2O_2]$)

Passivation

Set up for the electrochemical impedance spectroscopy (EIS) study



Sample : Iron wire (purity:>3N)

Solution : 1 mM NaCl, 80mL
(pH : 4, 8 11 and 13)

Immersion period : 3 days

Temperature : Ambient temp.

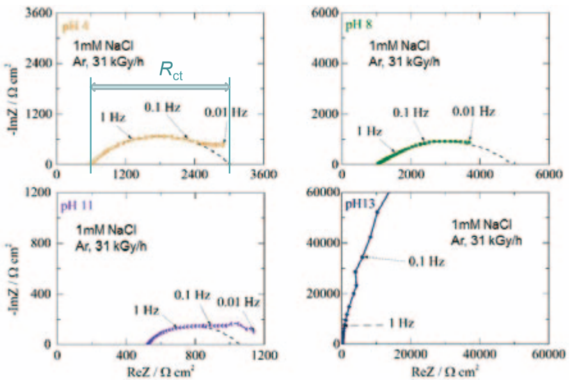
Absorbed dose rate : 31 kGy/h

Electrochemical Impedance Spectroscopy :
10 mHz to 20 kHz, ΔE = ±10 mV

Corrosion rate measurement :
Weight loss of the sample

* Danielson, M. J.
Corrosion **51**(6): 450-455 (1995)

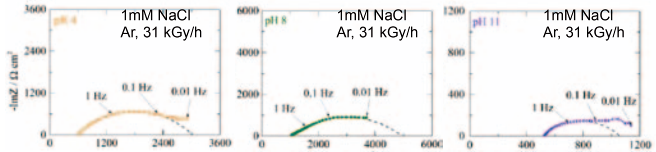
Impedance spectrum of the iron electrode in a 1 mM NaCl solution



R_{ct} (pH 11) < R_{ct} (pH 4) < R_{ct} (pH 8) << $\boxed{R_{ct}\text{ (pH 13)}}$ Passivated

(Higher corrosion rate) (lower corrosion rate)

Corrosion current estimation using Stern-Geary equation



$i_{corr}(R_{ct}) = B / R_{ct}$,

Mass transfer control

$B = \beta_a / 2.3$

β_a : neutral sol'n; 120mV,

acid or base sol'n; 40mV

John O'M.Bockris, Shahed U.M.Khan, "Surface Electrochemistry, A Molecure Level Approach", 765-765 (1993) Plenum press, NY

pH	$i_{corr}(R_{ct})$ (μA cm^{-2})	$i_{corr}(\triangle w)$ (μA cm^{-2})
4	7.2	12
8	4.5	6.9
11	33	60

Corrosion rate estimated from the R_{ct} showed a good agreement with that calculated from the weight-loss, $\triangle w$

4-electrons and 2-electrons reaction on reduction of O_2 and H_2O_2

n = 4 : $O_2 + 4e^- + 2H_2O \rightarrow 4OH^-$; $H_2O_2 + 2e^- \rightarrow 2OH^-$

n = 2 : $O_2 + 2e^- + 2H_2O \rightarrow H_2O_2 + 2OH^-$

n = 4 : $i_{corr} = \{ 4\, D_{O2}\, C_{O2} + 2\, D_{H2O2}\, C_{H2O2} \}(F/\delta)$

n = 2 : $i_{corr} = \{ 2\, n_{O2}\, D_{O2}\, C_{O2} \}(F/\delta)$

δ = 0.05 cm

$D_{O2} = 2.12\times10^{-5}$ cm^2/s, $D_{H2O2} = 1.71\times10^{-5}$ cm^2/s

4-electron and 2-electron reaction on reduction of O_2 and H_2O_2

n = 4 : $O_2 + 4e^- + 2H_2O \rightarrow 4OH^-$; $H_2O_2 + 2e^- \rightarrow 2OH^-$

n = 2 : $O_2 + 2e^- + 2H_2O \rightarrow H_2O_2 + 2OH^-$

pH	$i_{corr}(R_{ct})$ (μA cm^{-2})	$i_{corr}(\triangle w)$ (μA cm^{-2})	i_{corr}(n=4) (μA cm^{-2})	i_{corr}(n=2) (μA cm^{-2})
4	7.2	12	1.4	0.29
8	4.5	6.9	12	4.6
11	33	60	36	14

Oxidizing agents for the radiation corrosion of Iron

(1) NN to weak-alkaline solution: dissolved-oxygen & hydrogen peroxide (4 electrons reaction)

(2) Acid solution: Ferric iron-ion

Conclusion (1)

It may be recommendable that the pH of the circulating water to be controlled within a near-neutral range, especially in the place where the irradiation of higher absorbed dose rate is expected.

The corrosion damages on reinforcing steel may be concerned if the cover-concreate on the steel was heavily damaged and the pH of pore water there was decreased as low as 10 by the neutralization of the concreate.

Conclusion (2)

Within a near-neutral pH range, the mechanism of the radiation corrosion of iron, would be a 4-electrons reaction that the hydrogen peroxide as well as the dissolved-oxygen were reduced to hydroxyl ions.

For acid solutions, however, the dominant oxidizing agent might by the ferric iron-ion yielded by Fenton's reaction.

3.1.4 Applications and Insights from Radiolysis Modeling at Pacific Northwest National Laboratory

Richard S. WITTMAN [1], Edgar C. BUCK and Brady D. HANSON
[1] *Pacific Northwest National Laboratory, P.O. Box 999, Richland, WA 99352, richard.wittman@pnnl.gov*

Past and present applications and insights from radiolysis modeling at PNNL are reviewed. The first is an application to address the possibility that radiolysis of water and water vapor remaining in a spent nuclear fuel storage canister could result in corrosion or in the generation of hydrogen leading to flammability conditions[1]. In addition, concerns of radiolysis of water and water vapor exterior to a canister, as air flows through the overpack, have previously been raised. To analyze such conditions we develop and apply a radiolysis model to predict the radiolytically induced generation of hydrogen and corrosive products such as H_2O_2, O_2, NO_2, NO, and HNO_3 within and around a canister. The model solves the coupled kinetics equations for 111 reactions for 40 gas species to account for radiolytic-induced chemistry, which includes water recombination and reactions with air. Model results give possible bounds on radiolysis product concentrations. For inside the canister fill-gas, the main finding is that significant radiolysis of water vapor requires the presence of residual air to disable recombination. This implies that G-value based estimates would greatly overestimate hydrogen generation. Assuming 1% residual air, calculations indicate that a large volume of residual water (20 L) would be required to reach the 4% H_2 flammability limit in 16 years, and between 3 and 4 L of water would be required to reach the 4% H_2 flammability limit in 300 years. For outside the canister, the main radiolytic products formed in moist air are HNO_3, N_2O, NO_2, CO, and small amounts of O_3. Assuming bounding residence times for air, the highest concentrations are less than 50 ppm and are less than 1 ppm for more typical flow conditions. Dry air gives similar concentrations as moist air with the exception of increased NO_2 and the near absence of HNO_3. Additionally, formation of hydrogen peroxide could be generated in water layers on outside surfaces at $\approx 100\ \mu M$ levels.

And secondly, while the application of vapor phase radiolysis models can guide conditions for storage canister experiments, they can also predict fundamental behavior of mass action chemical kinetics equations. We explore the radiolytic production of H_2O_2 under various externally fixed conditions [2] of molecular H_2 and O_2 that have been regarded as problematic in the literature[3] – specifically, “jumps” in predicted concentrations, and inconsistencies between predictions and experiments have been reported for alpha radiolysis. We computationally map-out a critical concentration behavior for alpha radiolysis kinetics using a comprehensive set of reactions. We then show that all features of interest are accurately reproduced with 15 reactions.

References

[1] R.S. Wittman and B.D. Hanson, *Radiolysis Model Analysis for a Used Fuel Storage Canister ,* IHLRWM-2014 and PNNL report at: http://www.pnnl.gov/main/publications/external/technical_reports/PNNL-22773.pdf
[2] R.S. Wittman, E.C. Buck, E.J. Mausolf, B.K. McNamara, F.N. Smith, and C.Z. Soderquist, *Conditions for Critical Effects in the Mass Action Kinetics Equations for Water Radiolysis*, J. Phys. Chem. A, 118(51), (2014) 12105–12110.
[3] B. Pastina and J.A. LaVerne, *Effect of Molecular Hydrogen on Hydrogen Peroxide in Water Radiolysis*. J. Phys. Chem. A 2001, 105, 9316–9322.

Applications and Insights from Radiolysis Modeling at PNNL

Rick Wittman, Edgar Buck & Brady Hanson
Pacific Northwest National Laboratory

JAEA/CLADS Conference on Post-accident Waste Management Safety

November 8, 2016

Pacific Northwest NATIONAL LABORATORY

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Need for Radiolysis Model Analysis

Cross-cutting gap analysis has identified:

- Possibility of radiolysis of residual water (possibly up to 1L) inside a used fuel canister could result in hydrogen build-up leading to flammability concerns
- Radiolytic generation of hydrogen peroxide and other corrosive products could result in corrosion of fuel cladding and structural components
- Reported canister gas sample data (Bare & Torgerson) show H_2 concentrations greater than the 0.01% detection limit.

  (Some regulators, such as in France, presently require monitoring of canisters for hydrogen)
- Radiolysis of moist air surrounding canister outside walls could possibly generate corrosive products on the wall surface

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Law of Mass Action for Chemical Kinetics 101

Given the Forward and Reverse reactions

$$2A + B \to C \; ; k_F$$
$$C \to 2A + B \; ; k_R$$

the rate of change of species C is

$$\frac{d[C]}{dt} = k_F[A]^2[B] - k_R[C]$$

with an equilibrium condition determined at steady-state

$$\frac{[C]}{[A]^2[B]} = \frac{k_F}{k_R}$$

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Radiolysis Model Description:

- **Model is a Coupled Reaction Point Kinetics Model for 39 species and 101 reactions for calculating time dependent Gas Phase Radiolytic generation of H_2 and oxidizing products such as H_2O_2, HNO_3, O_2 & other species that could potentially accelerate corrosion.**

Mass Action & CSTR Coupled Kinetics Equation:

Generation of 39 Air Species A_i out to 300 years t

***G*-values for Air Species**

$$\frac{d[A_i]}{dt} - \frac{R}{V}([A_i] - [A_i]_0) = \dot{d}\sum_{g=1}^{N_g} G_i^{(g)} w_g [A_g] - \sum_{r=1}^{N_r} k_{ir} \prod_{j_r=1}^{n_r} [A_{j_r}]^{O_{j_r}}$$

Dose Rate

Reaction Rate Constants

External Air Conditions

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Radiolysis Model Description:

- Model is a Coupled Reaction Point Kinetics Model for 39 species and 101 reactions for calculating time dependent Gas Phase Radiolytic generation of oxidizing products such as H_2O_2, HNO_3, O_2 & other species that could potentially accelerate corrosion.

γ Field *G*-values for Air

	G (particles/100 eV)			
Species	H_2O	N_2	O_2	CO_2
H_2O	-7.350	0.000	0.000	0.000
·H	7.400	0.000	0.000	0.000
·OH	6.300	0.000	0.000	0.000
H_2	0.500	0.000	0.000	0.000
O_2	0.000	0.000	-3.300	0.000
O	1.050	0.000	[illegible]	[illegible]
O_2^+	0.000	0.000	2.070	0.000
O^+	[illegible]	[illegible]	[illegible]	[illegible]
e	0.000	2.960	3.300	2.960
N_2	0.000	-4.140	0.000	0.000
·N	0.000	0.295	0.000	0.000
$N(^2D)$	0.000	0.885	0.000	0.000
$N(^4S)$	0.000	1.870	0.000	0.000
N_2^+	0.000	2.270	0.000	0.000
N^+	0.000	0.690	0.000	0.000
CO_2	0.000	0.000	0.000	7.170
CO	0.000	0.000	0.000	[illegible]
CO^+	0.000	0.000	0.000	0.510
CO_2^+	0.000	0.000	0.000	[illegible]

39 Air Species Modeled

No.	Species	No.	Species
1	H_2O	[illegible]	NO_2
2	[illegible]	22	NO_3
3	·H	23	N_2O
[illegible]	·OH	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	H_2	[illegible]	[illegible]
[illegible]	O_2	[illegible]	[illegible]
8	[illegible]	[illegible]	[illegible]
9	O_3	[illegible]	[illegible]
10	[illegible]	[illegible]	[illegible]
11	[illegible]	[illegible]	[illegible]
12	O^-	32	NO
13	H_3O^+	33	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	e	[illegible]	[illegible]
[illegible]	N_2	[illegible]	[illegible]
17	·N	37	CO
[illegible]	$N(^2D)$	[illegible]	CO^+
[illegible]	$N(^4S)$	[illegible]	CO_2^+
20	NO	40	[illegible]

Example External Concentrations:
100% RH Air

Species	Conc. (ppm)
N_2	[illegible]
O_2	[illegible]
H_2O	[illegible]
CO_2	[illegible]
H_2	[illegible]
N_2O	[illegible]
CO	[illegible]
O_3	[illegible]
NO_2	[illegible]

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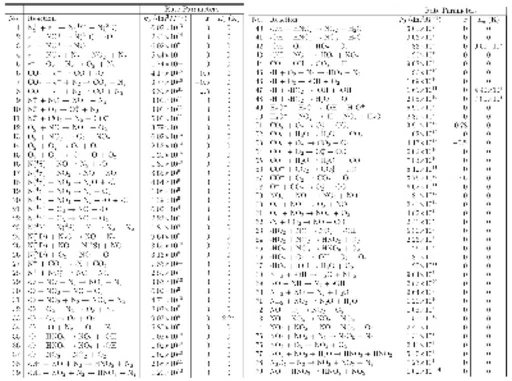
Radiolysis Model Description:

Excitation Transfer, Radical Disappearance & Molecular Species Reactions



R. Atkinson, *et al*., *Evaluated kinetic & photochemical data for atmospheric chemistry: Vol I – gas phase reactions of Ox, HOx, NOx & SOx species*, Atmos.Chem.Phys., **vol 4**, 2004, 1461-1738.

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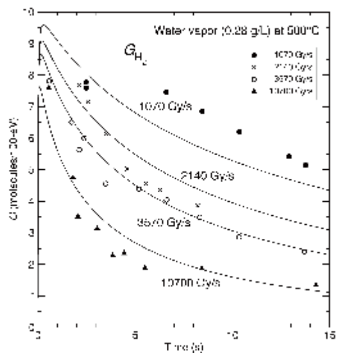
Radiolysis Model Validation:

All results of Arkhipov (2007) were reproduced.



O.P. ARKHIPOV, A.O. VERKHOVSKAYA, S.A. KABAKCHI and A.N. ERMAKOV, "Development and verification of a mathematical model of the radiolysis of water vapor," Atomic Energy 103(5):870 (2007)

Data from:
B.G. DZANTIEV, A.N. ERMAKOV, B.M. ZHITOMIRSKY and V.N. POPOV, "Thermoradiation processes of energy-carrier production," Int. J. Hydrogen Energy, 9(8): 723-728 (1984).

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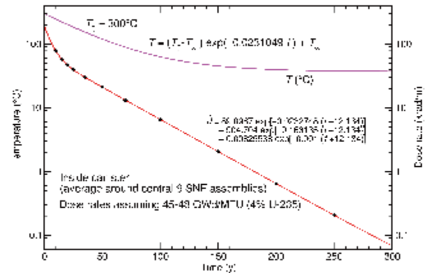
Radiolysis Model Description:

- Time dependence of dose rates inside (near fuel pins) canister [Fits to Bechtel SAIC work (2002 & 2004)]. The temperature change from T_∞ was assumed to decay with a 30 year half-life.



Neutron dose rates are roughly 3 to 4 orders of magnitude less than γ dose rates

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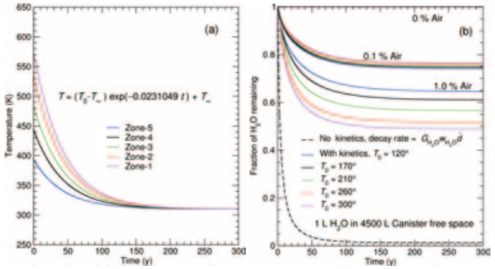
Radiolysis model results for:

Inside the canister fill gas



Dose rates assume a fuel assembly burnup of 45 GWd/MTU and initial enrichment of 4 wt % U-235

Effect of reaction kinetics compared with simple G-value estimate:

[illegible]

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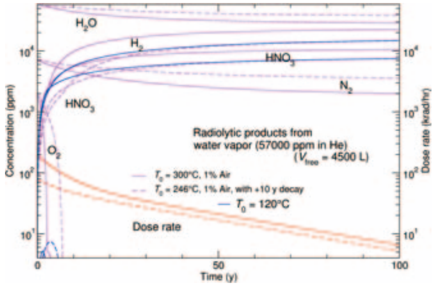
Radiolysis model results for:

Inside the canister fill gas



Dose rates assume a fuel assembly burnup of 45 GWd/MTU and initial enrichment of 3.7 wt % U-235

Fill Gas Composition over 100 Years Assuming Residual 1 L of Water and 1% Air

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Radiolysis model results for:

Inside the canister fill gas (summary)

Fill Gas Composition (%) Assuming 1 L Water at 16 Years

	1% Air				0.1% Air			
T_0(°C)	H_2	O_2	N_2	HNO_3	H_2	O_2	N_2	HNO_3
120	[illegible]	0.000	0.791	0.133	0.654	0.251	0.03[illegible]	0.077
300	1.315	0.000	0.388	0.653	0.350	0.199	0.04[illegible]	0.076

Fill Gas Composition (%) Assuming 1 L Water at 300 Years

	1% Air				0.1% Air			
T_0(°C)	H_2	O_2	N_2	HNO_3	H_2	O_2	N_2	HNO_3
120	1.605	0.000	0.330	0.759	1.382	0.582	0.013	0.119
300	2.347	0.000	0.182	[illegible]	1.251	0.498	0.013	0.119

Composition Correlation Matrix at 300 Years

Comp.	H_2O	H_2	O_2	N_2	HNO_3
H_2O	1.000	0.801	0.747	-0.206	0.362
H_2	0.801	1.000	0.354	0.139	[illegible]
O_2	0.747	0.354	1.000	0.330	0.045
N_2	0.206	0.139	0.250	1.000	0.429
HNO_3	0.362	0.826	-0.245	0.129	1.000

Dose rates assume a fuel assembly burnup of 45 GWd/MTU and initial enrichment of 4 wt % U-235

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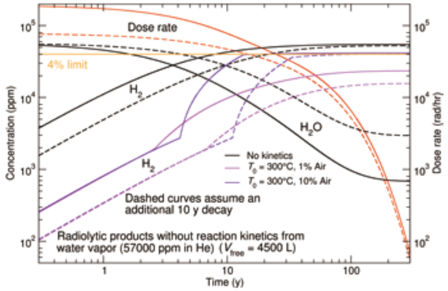
Radiolysis model results for:

Hydrogen inside the canister fill gas (summary)



Dose rates assume a fuel assembly burnup of 45 GWd/MTU and initial enrichment of 3.7 wt % U-235

H_2 Concentration as a Function of Time Comparing the Effect of Reaction Kinetics (magenta & violet curves) with the no Kinetics result (black curves).

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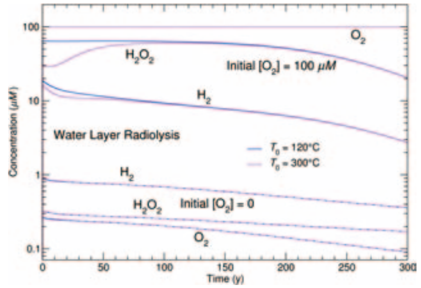
Radiolysis model results for:

Water Layer Radiolysis inside the canister

Dose rates assume a fuel assembly burnup of 45 GWd/MTU and initial enrichment of 4 wt % U-235



Concentrations of Water Layer Radiolysis Products as a Function of Time

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Summary of results for:

Inside the canister fill gas

- Significant radiolysis of water vapor requires the presence of residual air to disable recombination
- Reactions between residual water (1 L) and air in 4500 L of free space result in percent levels of H_2, O_2 and HNO_3 at 300 years, and about one-half the maximum values reached in the first 16 years
- Calculations indicate that a much greater volume of residual water (20 L) would be required to reach the 4% H_2 flammability limit in 16 years, and between 3 and 4 L of water would be required to reach the 4% H_2 flammability limit in 300 years
- Increased residual air results in greater H_2 and HNO_3 concentrations, but also in the depletion of O_2 because it is more effectively removed by a radiolytically induced reaction with N_2
- For lower (0.1 L) residual water and 1% air, all radiolytic products are less than 1%
- H_2O_2 is produced at only very low concentrations (< 10 ppm) in the fill gas, but is the major radiolysis product in liquid water, and could reach 10-100 μM levels in liquid water layers on surfaces

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Further Considerations:

It is likely that alpha dose should be considered for applicability to UO_2 fuel surface for canisters containing fuel debris

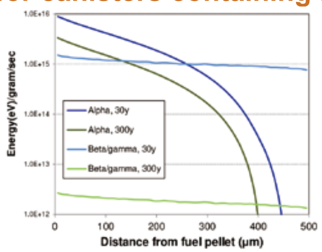

Figure 1.1 The dose to the surface of UNF exposed to a mixed water-air environment (10% water) is dominated by long-lived alpha radiation even at relatively short times out of reactor [Adapted from Radulescu 2011].

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Analytical α Radiolysis Model Description:

- **Subset of reactions sufficient to represent full *alpha* radiolysis model predictions with reaction rate constants ($M^n s^{-1}$)**

	Reaction	k
3	$H_2O_2 \rightarrow H^+ + \cdot HO_2^-$	1.1×10^{-1}
10	$H^+ + \cdot HO_2^- \rightarrow H_2O_2$	5.0×10^{10}
13	$e^- + H_2O \rightarrow \cdot H + OH^-$	1.9×10^{1}
14	$\cdot H + OH^- \rightarrow e^- + H_2O$	2.2×10^{7}
21	$\cdot HO_2 \rightarrow O_2^- + H^+$	1.3×10^{6}
22	$O_2^- + H^+ \rightarrow \cdot HO_2$	5.0×10^{10}
26	$e^- + H_2O_2 \rightarrow \cdot OH + OH^-$	1.1×10^{10}
29	$e^- + O_2 \rightarrow O_2^-$	1.9×10^{10}
36	$\cdot H + H_2O \rightarrow H_2 + \cdot OH$	1.1×10^{1}
42	$\cdot H + H_2O_2 \rightarrow \cdot OH + H_2O$	9.0×10^{7}
43	$\cdot H + O_2 \rightarrow \cdot HO_2$	2.1×10^{10}
50	$\cdot OH + H_2 \rightarrow \cdot H + H_2O$	4.3×10^{7}
51	$\cdot OH + H_2O_2 \rightarrow \cdot HO_2 + H_2O$	2.7×10^{7}
57	$\cdot HO_2 + O_2^- \rightarrow \cdot HO_2^- + O_2$	8.0×10^{7}
80	$H_2O_2 \rightarrow \cdot OH + \cdot OH$	2.5×10^{-7}

Reactions 13, 14, 29 & 36 have negligible effect on the solution which allows a direct solution of [e⁻] reducing the number of equations to three

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SS *alpha* Radiolysis Model Analysis:

- **Three equations for H_2O_2, ·OH and ·H are:**

$$-k_{43}[O_2][\cdot H] + (k_{51}[H_2O_2] + k_{50}[H_2])[\cdot OH] - \tilde{G}_{H_2O_2} + \frac{1}{2}\left(\tilde{G}_{OH} + \tilde{G}_{HO_2} - \tilde{G}_{H} - \tilde{G}_{e}\right)$$

$$(k_{41}[O_2] + k_{42}[H_2O_2])[\cdot H] - k_{50}[H_2][\cdot OH] - \tilde{G}_{H}$$

$$2k_{51}[H_2O_2][\cdot OH] - k_{50}[H_2][\cdot OH] + 2k_{80}[H_2O_2] + \tilde{G}_{H_2O_2} + \frac{1}{2}\left(3\tilde{G}_{OH} + \tilde{G}_{HO_2} + \tilde{G}_{H} - \tilde{G}_{e^-}\right) = 0$$

- **Eliminating ·OH and ·H give a cubic equation for H_2O_2**

$$A_0 + A_1[H_2O_2] - A_2[H_2O_2]^2 - A_3[H_2O_2]^3 = 0$$

We can now determine simple steady-state relations for H_2O_2 (e.g.)

$$\lim_{d \to 0} \frac{[H_2O_2]^2}{d} = \frac{k_{43}k_{50}(2G \ldots G \ldots + G_{H} \ldots G_{HO_2} \ldots G_{e^-})[H_2][O_2]}{4k_{80}(k_{41}k_{51}[O_2] \ldots k_{42}k_{50}[H_2])}$$

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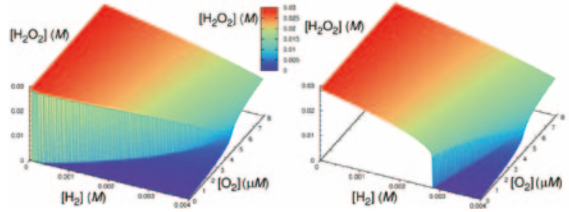
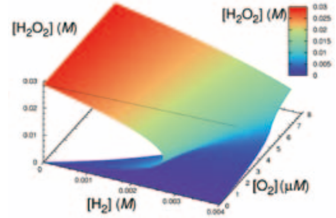
Model Results with Reduced Reaction Set:





Full $[H_2O_2]$ surface at 25 rad/s (5 MeV alpha) for the analytical radiolysis model showing region with three solutions

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Further Considerations:

Our current models could be modified to consider reactions in two coupled volume regions

Free Volume away from UO_2 fuel surfaces receiving only a γ dose rate

$$\frac{d[A_{i,\gamma}]}{dt} = \frac{R}{V_\gamma}\left([A_i]_\alpha - [A_{i,\gamma}]\right) - \dot{d}_\gamma \sum_{g=1}^{N_g} G_{ig}^{(\gamma)} w_g [A_g]_\gamma + \sum_{r=1}^{N_r} k_{ir} \prod_{j_r=1}^{n_r} [A_{j_r,\gamma}]^{O_{jr}} - SH_i^{\ldots}[A_i]_\gamma$$

Active Volume within 700 μm to 3 cm of UO_2 fuel debris surface receiving both an α and γ dose rate

$$\frac{d[A_{i,\alpha}]}{dt} + \frac{R}{V_\alpha}\left([A_i]_\alpha - [A_{i,\gamma}]\right) = \dot{d}_\alpha \sum_{g}^{N_g} G_{ig}^{(\alpha)} w_g [A_g]_\alpha$$
$$-\dot{d}_\gamma \sum_{g=1} G_{ig}^{(\gamma)} w_g [A_g]_\alpha + \sum_{r=1}^{N_r} k_{ir} \prod_{j_r=1}^{n_r} [A_{j_r}]_\alpha^{O_{jr}}$$

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Future Work:

Additional work for Radiolysis model applicability to UO_2 fuel debris canister

- Add H_2 depletion terms consistent with expected getter performance
- Add possible recombination reactions induced by the presence of catalyst surfaces
- Add alpha dose *G*-values for species generated in interstitial water at the exposed UO_2 surface area
- Add possible coupling of the reaction and transport kinetics of interstitial liquid and gas phase species
- Evaluate H_2 generation sensitivity to model parameters
- Guide experiments to model determine unknown model parameters

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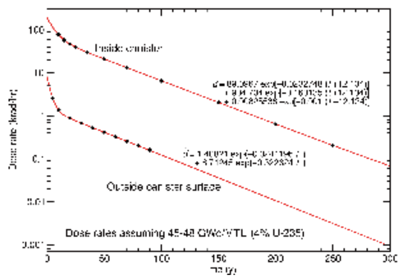
Radiolysis Model (air outside canister):

- Time dependence of dose rates inside (near fuel pins) canister and outside the canister surface [Fits to Bechtel SAIC work (2002 & 2004)].



Neutron dose rates are roughly 3 to 4 orders of magnitude less than γ dose rates

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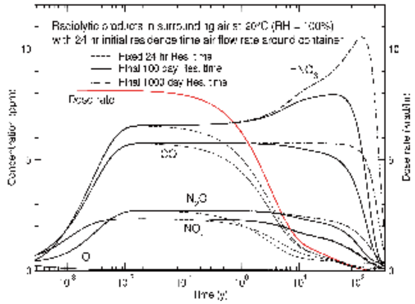
Radiolysis model results for:

Outside canister (inside overpack)



Air (at 20°C) Radiolysis Product Concentration as a Function of Time

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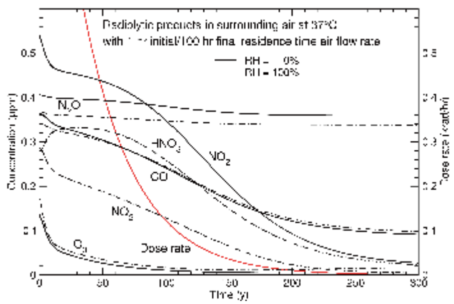
Radiolysis model results for:

Outside canister (nominal case)



Air (at 37°C) Radiolysis Product Concentration as a Function of Time

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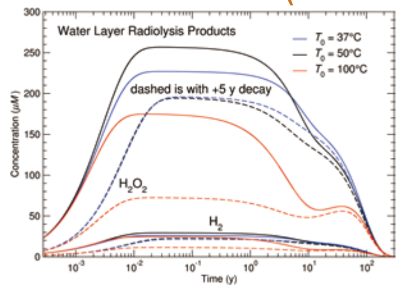
Radiolysis model results for:

On outside canister surface (inside overpack)



Temperature Dependence of Liquid Water Layer Radiolysis Products as a Function of Time

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Summary of results for:

Outside canister (inside overpack)

- The main radiolytic products formed in moist air are HNO_3, N_2O, NO_2, CO, and small amounts of O_3
- Even for extremely long residence times, the highest concentrations are less than 50 ppm and are less than 1 ppm for more typical flow conditions
- Dry air gives similar concentrations as moist air with the exception of increased NO_2 and the near absence of HNO_3
- As in the case of the fill gas, only trace levels of H_2O_2 are produced in the outside air, but could reach 0.05-0.2 mM concentrations in liquid water on the canister surfaces.

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Future Work:

Improve physical accuracy of radiolysis model

- Compare model with (& adjust to) a greater set of fill gas data and external gas sample data around existing fuel canister storage sites
- Couple the reaction and transport kinetics of liquid and gas phase species
- Develop a comprehensive model (as suggested by T. Ahn) that couples gas and liquid phase radiolysis with the electrochemical oxidation of cladding and exposed UO_2

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3.2 Session 4 Hydrogen Safety Research

3.2.1 Activities of the U.S. Department of Energy Facility Contractors Group (EFCOG) to Promote Hydrogen Safety

Kevin R. O'KULA [1,] and Roger D. LANNING [2]
[1] *AECOM Nuclear and Environment Technical Services; kevin.okula@aecom.com*
[2] *Hanford Tank Waste Treatment and Immobilization Plant; rdlannin@bechtel.com*

1. Introduction

Since 2009, the U.S. Department of Energy Facility Contractors Group (EFCOG) has provided hydrogen safety expertise to member contractors and federal organizations through its Hydrogen Safety Interest Group (H2SIG). The H2SIG is aligned under the Accident Analysis Task Group as part of the EFCOG Nuclear & Facility Safety Subgroup, and conducts activities in the following areas:

- **Issue Identification**: Through its communications, the H2SIG identifies common cross-cutting hydrogen safety operations and management issues from throughout the Department of Energy (DOE) Complex, as well as worldwide;
- **Lessons Learned and Experience Sharing**: The H2SIG promotes, coordinates, and facilitates the active exchange of lessons learned and related experiences on programs, practices, procedures, and other pertinent information of common interest on hydrogen safety analyses, including those that have been effective in the context of managing hydrogen hazards in nuclear facilities;
- **Technical Forum Exchange**: Technical, in-person forums and meetings are periodically conducted through EFCOG and technical societies, such as the American Nuclear Society, that include safety and nuclear waste representatives from multiple government, laboratory, facility contractors and consulting companies discussing common hydrogen safety challenges and potential solutions;
- **Database and Methodology Dissemination**: The H2SIG continually attempts to improve the dissemination of available information from U.S. and international hydrogen safety sources by assisting H2SIG members on analytical approaches, methodology, data and information exchange; and,
- **Operational/Accident Event Communications**: As precursor and actual combustible gas events occur, the H2SIG attempts to evaluate the significance for nuclear facilities and processes in general, and especially for those that have similar function(s).

The presentation will highlight activities in each of these five areas.

2. Focus of Work and H2SIG's Approach

The H2SIG is an entirely voluntary technical group of approximately 20 DOE program and field office representatives, contractors, consulting service providers and subject matter experts (SMEs). Seventeen major DOE sites and national laboratories are represented with Sellafield Sites in the United Kingdom comprising an eighteenth major site (Fig. 1). The primary objective of most activities in H2SIG is to help inform the required hazard and accident analyses required under the Nuclear Safety Management rule to support the safety basis for the facility or process, and documented in the facility's safety analysis report (i.e., Documented Safety Analysis, (DSA) for U.S. DOE facilities) [1, 2]. A secondary focus is to improve the tools (data and methods) for performing risk and reliability studies relevant to hydrogen storage and processing.

Of special interest to H2SIG is the resolution of emergent hydrogen and combustible gas issues, application of quantitative risk and other probabilistic approaches, reconciling differences between engineering and safety analysis, and evaluating the impacts of new technical guidance.

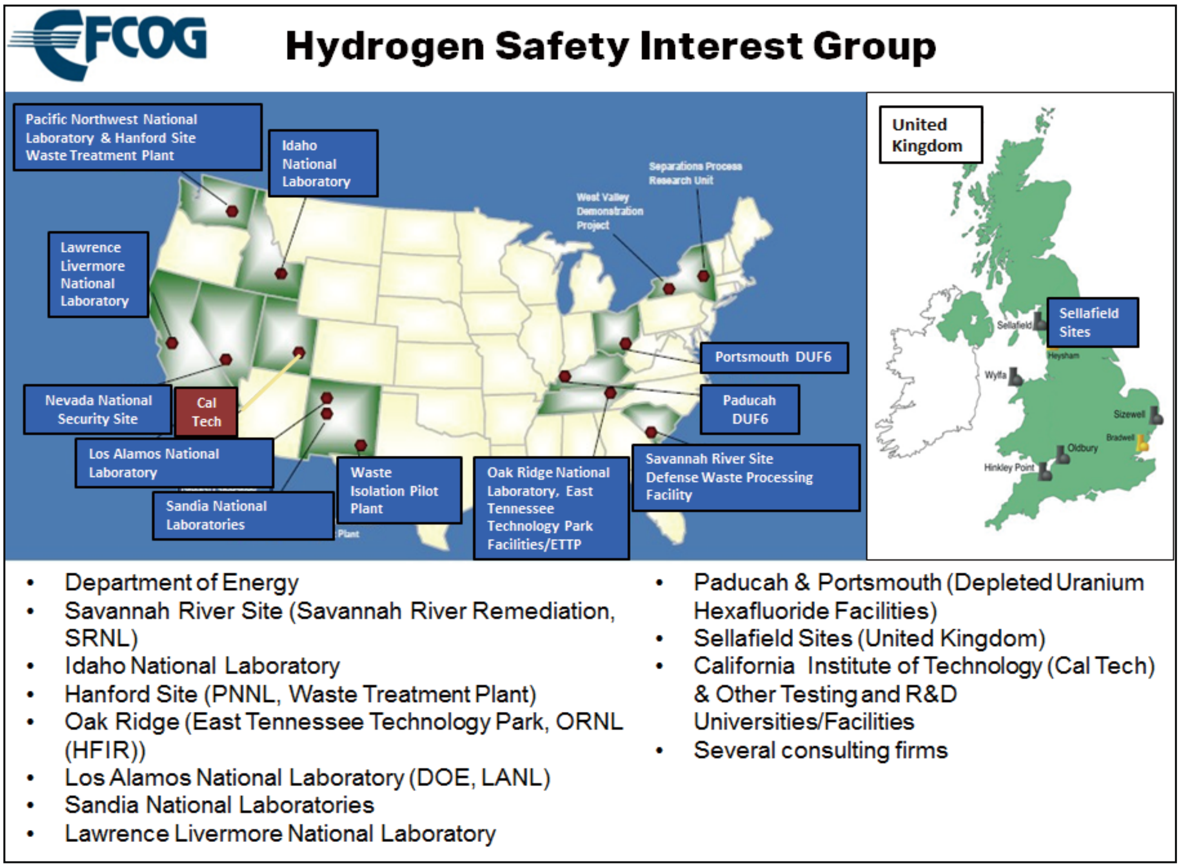

Fig.1 The EFCOG H2SIG has membership from DOE sites in the U.S. and includes the Sellafield Site (UK).

Technical areas addressed by H2SIG include the following: (1) Hazard and Accident Analysis; (2) Risk Management (e.g., waste container and processing issues); (3) Operational Safety and Best Practices; and (4) Tritium Accident Analysis. While the H2SIG carries no regulatory authority, its recommendations are made in a technical advisory capacity. Most communication is informational and topics are scheduled for initial consideration during a monthly conference call with subsequent discussion and other single-issue matters handled on an as-needed basis. Noteworthy examples of recent topics include: (1) Discussion of a new tritium handling and safe storage standard (DOE-STD-1129-2015); (2) Failure rate data for passive components to support design reliability studies in nuclear waste processing facilities; (3) Use of mists and caustic environments to mitigate hydrogen energetic events; (4) Evaluation of the potential for energetic events in waste storage containers; and (5) Comparison of dispersion methodology protocols for a postulated atmospheric release of tritium.

3. Near-Term Focus and Challenges

Despite ongoing budget constraints on the part of member organizations, the EFCOG H2SIG remains a strategic resource for DOE and nuclear facility contractors for informing safety analysis and reliability studies. The H2SIG will continue to work in the topical areas summarized in the previous section as well as pursue initiative in these areas: (1) Determine how to best engage with the Japan Atomic Energy Agency as an EFCOG technical partner in hydrogen and waste management areas; (2) Complete a white paper on tritium airborne release fractions under different accident conditions; and (3) Develop a framework for performing an integrated risk assessment for a large multiple nuclear facility site.

References

[1] 10 CFR 830, Nuclear Safety Management.
[2] U.S. DEPARTMENT OF ENEGY. DOE-STD-3009-2014, Preparation of Nonreactor Nuclear Facility Documented Safety Analysis, November 2014.

Activities of the Department of Energy Facility Contractors Group (EFCOG) to Promote Hydrogen Safety

Kevin O'Kula
AECOM Nuclear and Environment Technical Services

Technical Seminar on Safety Research for Radioactive Waste Storage
Iwaki, Fukushima Prefecture, Japan

8 November 2016

Technical Seminar on Safety Research for Radioactive Waste Storage – November 8, 2016

Outline of Presentation

1. Historical Context
2. Overview of Issue for DOE Contractors
3. One Solution: U.S. Department of Energy Facility Contractors Group EFCOG & Hydrogen Safety Interest Group
4. Hydrogen Safety Interest Group: General Features
5. Characteristics of H2SIG
6. Primary Activity Areas for H2SIG
   a. Issue Identification
   b. Lessons Learned and Experience Sharing
   c. Technical Forum Exchange
   d. Database and Methodology Dissemination
   e. Operational/Accident Event Communications
7. Future Work & Collaborative Opportunities

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1. Hydrogen Hazards in Nuclear Facilities: Historical Context & Selected Events

- Experimental/test reactors – December 1952 Partial meltdown at NRX (Chalk River, Canada)
  - First known severe accident in a reactor of a Western nation
  - Overheating of fuel with cladding rupture and subsequent fuel – coolant interaction
  - Ensuing H_2 explosion
- HLW Waste Tanks – 1980s at Hanford Site, ~LFL concentrations in vent space for Tank SY-101
  - Double-shell tank, ~4 x 10^6 liters of HLW
  - Neutrally buoyant sludge layer over supernatant layer with release of large amounts of H_2
  - Intermittent H_2 gas "burps" over several months
- Combustible residues + plutonium in waste drums at Idaho National Laboratory
  - Waste drum fire in August 2003
  - Waste drum deflagration in November 2005

Chalk River, Canada

Image from inside SY-101 (1989)

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2. Overview of Issue for DOE Contractors

- U.S. Department of Energy (DOE) processes in its facilities, laboratories not immune to hydrogen and other combustible gas hazards from waste processing and storage
- Two inventory types
  - Unwanted: TRU waste drum, waste tank/vessels, and vitrification facilities
  - Planned: Process and operational support
- Many of these issues are common to chemical processing and commercial nuclear facilities
- Numerous and varied approaches taken throughout these facilities for managing hydrogen inventories, limiting the deflagration/detonation, time to reach LFL, identification of Engineering Safety Features (ESFs), among other needs and issues
- As a result, hydrogen hazards can be poorly characterized or not recognized
  - Resulting engineering safety features (i.e., control set) may missing or if improperly characterized, not be suitable

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3. One Solution: Hydrogen Safety Interest Group

- Saves resources through sharing of information to improve consistency among facilities/processes with similar hydrogen issues
- Provides focal point for collecting and disseminating information, data, and lessons learned
- Encourages exchange with technical and contractor groups that have similar or related concerns, from domestic and International colleagues
  - United Kingdom's Hydrogen Working Party (HWP)
  - U.S. Nuclear Regulatory Commission
  - DOE Hydrogen Program/Safety Panel
  - [Potentially, Japan Atomic Energy Agency]
- Supports development of standards and guidance on safety policy, principles and safety analysis methodology
- Provides a technical forum to examine issues, obtain consensus
- Includes consideration of tritium (especially, HTO, T_2O)
- Important: H2SIG - Should not be the only approach

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4. Hydrogen Safety Interest Group: General Features

- Formed in August 2009
- Realigned with Accident Analysis Task Group (TG) in 2015
- Conduct telecon meetings with the Accident Analysis TG
- Meet on annual basis with other EFCOG Nuclear and Facility Safety Groups during Safety Basis Workshops (Albuquerque, New Mexico)
- Support paper presentations and training at biennial Nuclear & Facility Workshop (2016 at Argonne National Laboratory; 2018 (tentatively) in Richland, Washington)
- Includes DOE headquarters and field office technical staff
- Represented by technical subject matter experts from major Laboratories/Facilities/Sites

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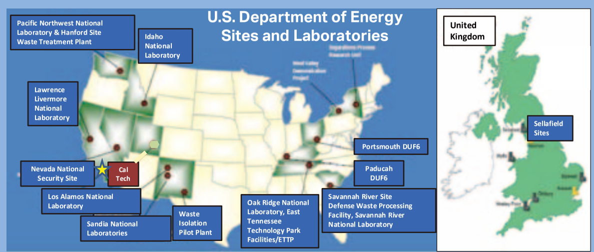
Hydrogen Safety Interest Group Member Sites and Organizations



- Department of Energy
- Savannah River Site (Savannah River Remediation & Savannah River National Laboratory)
- Idaho National Laboratory
- Hanford Site (PNNL, Waste Treatment Plant)
- Oak Ridge (East Tennessee Technology Park, Oak Ridge National Laboratory (HFIR))
- Los Alamos National Laboratory
- Sandia National Laboratories
- Lawrence Livermore National Laboratory
- Paducah & Portsmouth (Depleted Uranium Hexafluoride Facilities)
- Sellafield Sites (United Kingdom)
- California Institute of Technology (Cal Tech) & Other Testing and R&D Universities/Facilities
- Consulting firms (AECOM Technical Services, CBI, etc.)

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H2 Safety Interest Group is Aligned with the Accident Analysis Task Group

Nuclear & Facility Safety (NFS) Subgroup
Jeff Buczek, AREVA – Chair
Nathan Cathey, PNNL – Vice Chair

Accident Analysis Task Group
Roger Lanning, WTP – Chair
John McAllister, AECOM – Vice Chair

Hydrogen Safety Interest Group
Kevin O'Kula, AECOM
Subject Matter Expert

Safety Basis Task Group
Charles Carathers, Chair, ORNL
Ron Beaulieu, NNSS – Vice Chair

Criticality Safety Task Group
Andrew Prichard, PNNL – Chair
James Kuropatwinski, LANL – Vice Chair

Unreviewed Safety Question (USQ) Task Group
Phil Montgomery, Y-12 – Chair
Ron Selvage, LANL – Vice Chair

Nuclear Safety R&D (NSR&D) Task Group
Mukesh Gupta, AECOM – Chair
John Miller, Sandia National Laboratories

Steering Committee
WG Officers
Subgroup Chairs
DOE Sponsors
Mark Mitchell (Past Chair) – LLNL
Mark Joseph – Argonne National Laboratory
Andrew Vincent – SRNS
Kevin Kimball – CNS

NTC Training Liaison
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Jim O'Neil, NNSA/LAFO
Jose Munoz, NNSA/SFO
Greg Jones, DOE-Los Alamos Operations

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5. Characteristics of H2SIG

- Primarily provides subject matter expert (SME) guidance support mostly hazard and accident analysis issue resolution to members
- Voluntary, contractor-initiated
- Members are working level engineers, other technical & regulatory staff
- Apply "graded approach" to situations or analysis
  - Emphasize appropriate analytical solutions relative to hazard
  - Experimental programs are occasionally reviewed
- Three main technical areas & one supplemental area
  1. Hazard and Accident Analysis
     Deterministic applications
  2. Probabilistic safety and reliability applications
  3. Tritium safety analysis

  Supplemental area: Operational safe practices

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6. Primary Activity Areas for H2SIG

1. Issue Identification
2. Lessons Learned and Experience Sharing
3. Technical Forum Exchange of Information
4. Database and Methodology Dissemination
5. Operational/Accident Event Communications

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6.1 Issue Identification

H2SIG identifies and discusses common cross-cutting hydrogen safety operations and management issues from throughout the Department of Energy (DOE) Complex, as well as worldwide

- Waste container situations are closely tracked after Waste Isolation Pilot Plant (WIPP) event in 2014
- Recent observations of waste drum and container issues at Oak Ridge and Los Alamos
- Oak Ridge: Low-level waste Transuranic waste/Pu drums
- Los Alamos: Tritium Waste Containers
- Difference compared to major facilities but should be no less vigilant with respect to worker and environmental protection

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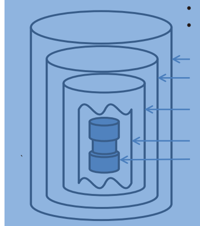
Container Deflagration/Detonation Analysis

Recap/Background

- Transuranic/plutonium oxide waste with layered packaging
- Container inside two nested cans, inside a 2R container, inside a 6M drum, and nested in a 55 gallon drum
- Deflagration/detonation concern on hazards associated with opening these containers for repackaging in WIPP compliant containers
  - Partitioned drum population
  - Some previously repackaged without effective environmental controls
  - Calculation effort analyzed those drums that had reflected deflagration-to-detonation transition (R-DDT) potential & Controls identified



[1] ID of container provided for determining H_2 concentration within the container.
[2] OD of container provided for determining ullage in surrounding container(s).
[3] Internal volume of container provided for determining H2 concentration within the container.
[4] External volume of container provided for determining ullage in surrounding container(s).

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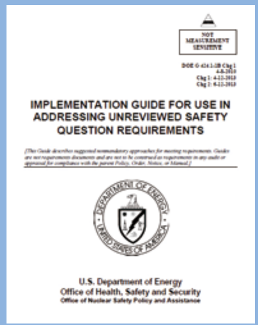
Los Alamos Tritium Waste Containers



- Current situation of interest concerns several Flanged Tritium Waste Containers at Los Alamos National Laboratory
- Containers potentially pressurized with an explosive mixture of hydrogen isotopes and oxygen
- Gas buildup suspected from the radiolysis of tritiated water sorbed to zeolite beds loaded above recommended limit
  - ➢ Some containers also have integrity concerns
- Treating as Unreviewed Safety Question per DOE Guide 424,11-1B, Chg. 2
- Access to the containers prohibited while Evaluation of the Safety of the Situation developed

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6.2 Lessons Learned and Experience Sharing

H2SIG promotes, coordinates, and facilitates the active exchange of lessons learned and related experiences on programs, practices, procedures, and other pertinent information of common interest on hydrogen safety analyses, including those that have been effective in the context of managing hydrogen hazards in nuclear facilities

- Industry risk expert review of flammable gas piping at depleted uranium hexafluoride (DUF6) facilities (Portsmouth and Paducah)
- Consequence analysis of a postulated hydrogen explosion scenarios associated with a proposed steam methane reformer (SMR) unit.
- Past participation from Oak Ridge National Laboratory High Flux Isotope Reactor(HFIR)
  - ➢ Use of low-temperature hydrogen to achieve cold neutron source for materials testing and research

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Experience Sharing Example Collaboration Among Participants

- UK Hydrogen Working Party (HWP) noted work by Sellafield's University Partner, London South Bank University, into explosion mitigation; specifically about the effects of sodium hydroxide and water fogs
- Mitigative Potential
  - ➢ Work suggested that using a combination of the three measures (use of very fine water fog, nitrogen dilution and sodium hydroxide additives in the mitigation of hydrogen deflagration) together produces the optimal mitigation performance and
  - ➢ Can be extremely effective in: inhibiting the burning velocity, reducing the rate of explosion overpressure rise and narrowing the flammability limits of hydrogen-oxygen- nitrogen mixtures
- Other experiences have been shared

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6.3 Technical Forum Exchange

Technical, in-person forums and meetings are periodically conducted through EFCOG and technical societies, such as the American Nuclear Society, that include safety and nuclear waste representatives from multiple government, laboratory, facility contractors and consulting companies discussing common hydrogen safety challenges and potential solutions

Past, Recent and Future Conference and Technical Sessions:

- 2007 Embedded Topical Meeting: **Safety and Technology of Nuclear Hydrogen Production, Control, and Management** (ST-NH2), Boston, Massachusetts, June 2007
- 2010 Embedded Topical: **Safety and Technology of Nuclear Hydrogen Production, Control and Management,** San Diego, California, June 2010
- EFCOG Safety Basis Workshop – Panel Session on **Risk Assessment Approaches to Improve Nuclear Safety in Facilities Containing Hydrogen Hazards**, Nov. 3, 2010
- Panel Session at 2014 ANS Winter Conference on **Hydrogen and Combustible Gas Issues in Nuclear Safety** (Tuesday, November 11, 2014)
- Eleventh International Conference on Tritium Science and Technology (**TRITIUM 2016**), Charleston, South Carolina
- 16th International Topical Meeting on **Probabilistic Safety Assessment and Analysis, PSA 2019,** April 28 – May 3, 2019, Charleston, South Carolina

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Technical Exchange: 2016 JAEA-EFCOG Panel Session

2016 EFCOG Nuclear and Facility Safety Workshop
Argonne National Laboratory, August 2016

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AECOM 17

JAEA – Fukushima Daiichi Research Activities Update

Fukushima Daiichi Site Remediation and Progress on Technical Issues Through the JAEA Collaborative Laboratories for Advanced Decommissioning Science (CLADS)

Title	Authors/Organization
Japanese Activities for Fukushima-Daiichi Decommissioning Research	Koji OKAMOTO, Senior Advisor, CLADS, JAEA and T. Okawa, CLADS, JAEA
High-Temperature Structure Behavior During Severe Accident Conditions	Koji OKAMOTO, B. Jo, and S. Ueda, CLADS, JAEA
Irradiation Experiments of Simulated Wastes of Carbonate Slurry	Ryuji NAGAISHI, Principal Scientist, Storage Equipment Soundness Evaluation Group, Waste Management Division, T. Motooka, and I. Yamagishi, CLADS, JAEA
Seawater Effects on the Soundness of Spent Fuel Cladding Tube	Takafumi MOTOOKA, Principal Scientist, Storage Equipment Soundness Evaluation Group, Waste Management Division F. Ueno, and M. Yamamoto, CLADS, JAEA

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AECOM

6.4 Database and Methodology Dissemination (Innovative Knowledge Management)

Attempt to improve the dissemination of available methodology and supporting information from U.S. and international hydrogen safety sources by assisting H2SIG members on analytical approaches, methodology, data and information exchange

- Highlight availability & improvements in methodology, approaches, and data sources
  - Discussed DOE STD-1129-2015 on Tritium Handling and Safe Storage
  - Summarize use of quantitative risk analysis (QRA) to augment final design of vitrification facility process piping (next five slides)

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AECOM 19

Hanford Tank Waste Treatment & Immobilization Plant (WTP) Quantitative Risk Assessment (QRA)

- Two BWR piping incidents
  - Hamaoka-1 High-Pressure Core Spray system pipe rupture
  - Brunsbuttel BWR core spray system area explosion
- U.S. Nuclear Regulatory Commission Information Notice 2002-15, “Hydrogen Combustion Events in Foreign BWR Piping”
  - Described catastrophic failures of high-pressure BWR process piping attributed to the accumulation of hydrogen gas
- WTP waste produces hydrogen and oxidants due to radiolytic decomposition, and radiolytic and thermolytic decomposition of organic components
- Hydrogen and oxidants can accumulate in stagnant regions of piping when waste containing systems stop flowing
  - Normal process evolutions
  - Faulted condition or accident events (e.g., loss of power, seismic event, etc.)
- Concern on hydrogen accumulation in sufficient quantities in piping systems could causing damage if ignition occurred
- After deterministic approach resulted in an inoperable plant, QRA suggested by expert committee in late 2008 to guide late stages of piping design

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AECOM 20

Black Cell Piping -
No Entry during the life of the plant

Hot Cell Piping
Remote entry possible during the life of the plant

Pre Treatment Hot Cell Looking East

Commercial Nuclear Power Plant PRA Compared to QRA Applied to WTP (2009-2015)

Characteristic	Risk Study: NPP PRA Study	Risk Study: WTP QRA of Hydrogen Events
Nuclear Facility/Mode	Nuclear Power Plant/ Steady-State, Continuous Operation	Waste Processing, Nuclear, Nonreactor/ Batch Operation
Facility Design	Standardized PWR and BWR designs; (multiple containment types)	Unique design; confinement system
PRA type/System	Radiological Risk/ Entire Plant	QRA of hydrogen events/ Individual Piping route
PRA timing "snapshot" PRA performed & Year of Operation/Startup	Operating plants; Published in 1990: 1. Grand Gulf 1 (BWR-6) - 1985 2. Peach Bottom 2 (BWR-4) – 1974 3. Sequoyah 1 (4-loop PWR) – 1981 4. Surry 1 (3-loop PWR) – 1972 5. Zion 1 (4-loop PWR) - 1973	• Pre-operational • Design of the plant to be complete by 2013 • All facilities and systems to be fully operational in 2019
Operational modes	Severe accidents; internal & external initiating events	Design basis events; Normal operation Internal & external initiating events,
Risk metrics	Acute fatality and latent cancer fatality risks; others; Safety Goal Compliance	Frequency-Severity curves for different hydrogen event types
Risk output format	Complementary cumulative distribution functions (CCDFs);Safety goal risks @ mean level	Complementary cumulative distribution functions (CCDFs) of key results

QRA Methodology – *Event Frequency-Severity Analysis*

Structure for hydrogen event frequency-severity analysis
- **Pre-processing** of plant / operational data and definition of required user inputs / assumptions
- **Probabilistic frequency analysis and event progression** with Monte Carlo (MC) simulation
  - Operational Frequency Analysis (OFA): Quantifies frequency of conditions that can support gas pocket formation
  - Gas Pocket Formation (GPF) Model: Quantifies realistic gas pocket parameters (e.g., size, composition, initial pressure)
  - Event Progression Logic (EPL) Model: Evaluates event progression logic to quantify hydrogen event type and severity
- **Post-processing** of MC simulation results and risk aggregation

6.5 Incidents and Accident Event Communications

As precursor and actual combustible gas events are recognized, the H2SIG attempts to evaluate the significance for nuclear facilities and processes in general, and especially for those that have similar function(s)

- DOE Hydrogen Safety - https://h2tools.org/
  - Incidents database - http://h2tools.org/lessons/
- International Association for Hydrogen Safety
  - http://www.hysafe.info/
- Hydrogen Safety and Research (HySAFER) Center at the University of Ulster Northern Ireland (http://hysafer.ulster.ac.uk/)
  - References on Hydrogen Safety Engineering

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7. Future Work & Collaborative Opportunities

- H2SIG – A strategic resource for DOE and other facility contractors worldwide for informing safety analysis and reliability studies on hydrogen hazards in nuclear facilities
- Future Work and Collaboration:

1. Determine how to best engage with the Japan Atomic Energy Agency as an technical partner in hydrogen safety and waste management areas.
2. Develop a framework for performing an integrated risk assessment for a multiple nuclear-facility site considering nuclear & hydrogen hazards.
3. Complete a white paper on tritium airborne release fractions under important accident conditions.

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Thank you for your attention!

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Questions?

2016 EFCOG Nuclear and Facility Safety Workshop Participants
Argonne National Laboratory, August 2016

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3.2.2 Hydrogen Safety Technology: Application of the Automotive Catalyst in a Hydrogen Recombiner

Hirohisa TANAKA [1], Masashi TANIGUCHI [2] and Yasuo NISHIHATA [3]
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1. Background

March 2011, by the Tohoku-Pacific Ocean Earthquake and subsequent Tsunami, severe accident with hydrogen explosions has occurred in the lost cooling function Fukushima Daiichi Nuclear Power Plant. Further, during the long term sequestration storage of the radioactive waste containing water, the generation of water-splitting hydrogen and oxygen by radiation, and the gas filling in those containers should be problematic. In both of the above two cases, highly active catalysts for recombining hydrogen with oxygen into safe water have been strongly desired.

2. Objective of the Project

The object of the project is the development of the recombination catalyst for hydrogen safety, by applying the automotive monolith catalyst technology having high activity, small size, light weight, long durability and mass productivity.

3. A Self-healing "Intelligent Catalyst" for Automotive Emissions Control

A self-healing, named "intelligent catalyst" has been developed and put into practical use for automotive emissions control that can be maintained a high activity for a long period of time [1-3]. The catalyst was attached to an automobile in excess of 6 million units. The intelligent catalyst was also found to have excellent ability to recombine hydrogen even at a room temperature.

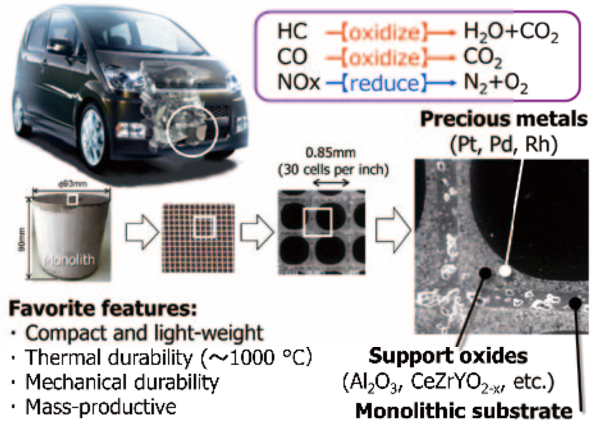


Fig.1. Automotive catalyst.

4. Passive Autocatalytic Recombiner (PAR)

For preventing hydrogen explosion in case of the severe accident at a nuclear power plant, Passive Autocatalytic Recombiner (PAR) has attracted attention as a safety device that does not need any power supply [4-5]. Characteristics of the wide range are required for the PAR:

(A) High activity to recombine with oxygen from the low hydrogen concentration.

(B) On the contrary, for high concentration of hydrogen, the reaction suppression for not becoming a source of ignition.

(C) Fast reactivity to high-speed gas flow.

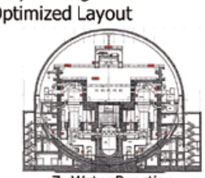
TYPE	RAD WASTE STORAGE	NUCLEAR POWER PLANT
Facility	Backend Facility Decommissioning	Containment Vessel & Building of Nuclear Power Plant
Concept	Environmental Compatible Long Term Durability  High Level Liquid Waste Tanks at Fukushima	Lightweight and Compact Easy Setting Optimized Layout  Zr-Water Reaction Radiolysis of Water

Fig. 2. Passive Autocatalytic Recombiner for hydrogen safety.

(D) Tolerance to catalyst poisoning of CO generated by Molten Core Concrete Interaction (MCCI).

5. Radioactive Waste Storage

The properties required as a hydrogen recombination catalyst for the long-term storage of radioactive waste are also numerous:

(A) High catalytic activity to recombine hydrogen from a room temperature by utilizing the natural convection due to generating water in the closed container, even without assistance mechanism from the outside such as heating or agitation.

(B) Robustness to water vapor, carbon monoxide and nitrogen oxides generated in the vessel.

(C) Furthermore, the activity must be maintained even in the long-term radiation exposure.

6. Achievement of Research

1) Immersion in salt water: The recombination reaction immediately began at room temperature. When immersed in salt water catalytic activity is reduced, but activity emerges from a room temperature after activated.

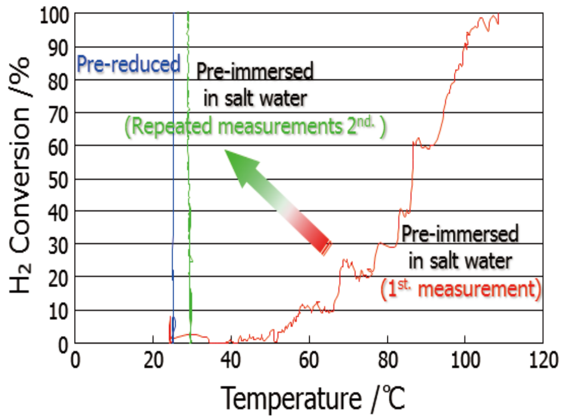


Fig.3. Temperature dependency of hydrogen recombination.

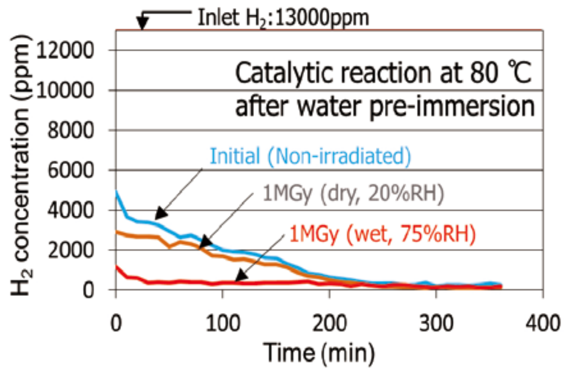


Fig.4. Impact of γ-ray irradiation.

2) γ-ray radiation: It was promoted by a γ-ray radiation of ^{60}Co up to 1.0 MGy, almost corresponding to the cumulative dose for 1 year after loss of coolant accident (LOCA) at boiling water reactor (BWR). Interestingly, the activity of the irradiated catalyst was further improved.

3) CO poisoning: CO tolerance was improved by ceria-based oxygen storage capacity support.

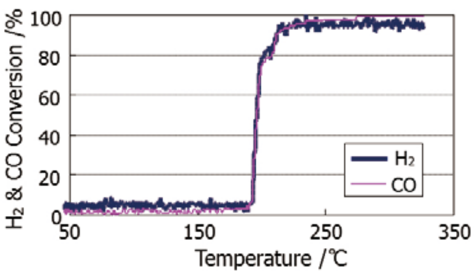


Fig.5. CO poisoning on Pt/Al_2O_3.

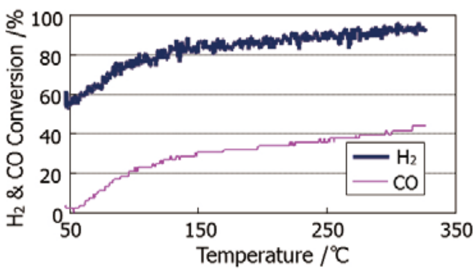


Fig.6. CO poisoning on improved Pt/CZY.

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Remarks

We would like to express our sincere condolences to the victims of everyone in the Tohoku-Pacific Ocean Earthquake and subsequent Tsunami. We sincerely wish that our research and development could contribute to the safety of everyday life. This project on Advancing hydrogen safety was supported by Ministry of Economy, Trade and Industry (METI) from 2013 to 2016. We sincerely thank the group members of Dr. S. Hirata of Kawasaki Heavy Industries and Dr. E.-A. Reinecke of Forschungszentrum Jülich (FZJ) for their kind help on a PAR device experimental.

3.2.3 Hydrogen Safety Research in Europe

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1. Introduction

During the course of a severe accident (SA) in a light water nuclear reactor, large amounts of hydrogen could be generated and released into the containment during reactor core degradation. Additional burnable gases (H_2 and CO) may be released into the containment in case of molten corium/concrete interaction (MCCI). This could subsequently raise a combustion hazard. As observed during the Fukushima accidents, hydrogen combustion could cause high pressure peaks that could challenge the reactor containments and lead to the failure of surrounding buildings. A hydrogen explosion may also be a safety concern in spent fuel storage areas, where flammable conditions may be reached if adequate ventilation is not provided. In this case, the hydrogen explosion may lead to radioactive products dispersion into the environment. To prevent the hydrogen explosion hazard, most of the mitigation strategies adopted in European countries are based on the implementation of Passive Autocatalytic Recombiners (PARs). Nevertheless, studies of representative accident sequences indicate that, despite the installation of PARs, it is difficult to prevent, for all times and locations, the formation of a combustible mixture potentially leading to local flame acceleration. To better understand the phenomena associated with the combustion hazard and to address the issues highlighted after the Fukushima Dai-ichi events, such as the explosion hazard inside the venting systems or the potential flammable mixture migration into spaces beyond the primary containment, complementary R&D projects were launched recently. The expected results will be used to improve the methodology for hydrogen risk assessment and the severe accident management guidelines. The present paper aims to present an overview of the on-going and the planned R&D programs related to hydrogen mitigation and combustion.

2. Hydrogen Mitigation

Passive auto-catalytic recombiners are the key feature of the hydrogen mitigation strategy in many European nuclear power plants. The operational behavior of PARs has been studied in both national and international projects in the past, starting with early test programs conducted primarily by the manufacturers (Siemens, AECL, etc.) [1, 2]. More recently and in the frame of the OECD/NEA THAI and THAI-2 programs at Becker Technologies [3, 4] and the REKO program at JÜLICH [5], significant progress in knowledge has been achieved on the behavior (onset of recombination, recombination rate and ignition potential) of different PAR designs under SA typical conditions. In addition, studies of parallel hydrogen and carbon monoxide recombination are on-going in the frame work of the REKO program [6] and first indicating experiments on interaction of cable fire products with PARs have been conducted [7].
The available experiments have supported the development of advanced mechanistic PAR models such as REKO-DIREKT (JÜLICH) [8] and SPARK (IRSN) [9], which provide reliable PAR performance modeling beyond classical parameter correlations. Recent experiments performed in the frame of the

OECD/NEA THAI2 project have confirmed that the predictive capabilities of conventional parameter correlations are limited under challenging boundary conditions such as oxygen starvation. Thus, for PAR operation under 'academic' or 'clean' conditions (i.e. initially quiescent atmosphere, well defined mixture of hydrogen, air and steam), a large database for model development and validation exists based on the mentioned test programs OECD/NEATHAI and REKO. The simulation of the PAR behavior under these conditions has reached a good level of maturity. However, the boundary conditions during a severe accident include more challenges: first, the gas mixture at the late phase will include not only hydrogen and steam, but also carbon monoxide from MCCI at low-oxygen conditions. Second, the overall flow conditions inside the containment in the course of a severe accident may be quite different from a quiescent atmosphere. In general, the corresponding flow patterns will be strongly influenced by the scenario of the accident, the release location and the containment design. Especially close to the containment walls and structures, a downward-directed flow due to high condensation rates is expected. As many of the PARs installed inside a LWR are installed close to the containment walls, these PARs will have to establish the internal upward chimney flow against the downward local containment flow. Another relevant issue, never investigated before in detail, is the influence of the PAR location (with regard to the source) on the efficiency of the flammable gases conversion. These open safety issues are to be addressed in the upcoming SAMHYCO project, submitted to the European Commission CALL HORIZON 2020.

3. Hydrogen Combustion

Hydrogen is characterized by its combustion properties such as a very wide flammability domain and very low ignition energy. These two characteristics are responsible for the explosion hazard when a release occurs in a confined environment. In the case where hydrogen/air diluted mixtures can be obtained within the flammability limits, a local ignition of the combustible mixture may give initially birth to a slow flame. However, this slow flame can soon strongly accelerate due to all the obstacles that would obstruct its path. This flame acceleration is responsible of high pressure loads that challenge the integrity of the reactor containment building and the safety components operability. One of the major issues/challenges is to predict the level of flame acceleration due to particular geometrical configurations and scales for a given inhomogeneous mixture. Indeed the presence of obstacles will enhance first the combustion leading to the increase in the flame velocity. Then, depending on the geometry and the premixed combustible mixtures composition, it can accelerate the transition to detonation or be quenched after a certain distance. Since 1980's, several experimental programs have been launched with the objective of characterizing the flame acceleration and transition to detonation in closed tubes containing obstacles, in which critical conditions are in principle well-understood [10, 11, 12]. More recently, flame acceleration in uniform and stratified mixtures was investigated in the framework of the ENACCEF program [13]. In addition, the effect of diluent on flame propagation was also studied [14]. The investigation of the effect that spray may have on flame propagation was initiated in the framework of ENACCEF and OECD/NEA-THAI2 projects [13, 15]. The available experiments have supported the development and validation of combustion models. Nevertheless, deficiencies still exist that the scientific community has to address in order to reduce the uncertainty margins within the evaluation of the potential hazard in given scenarios. These deficiencies can be attributed to remaining uncertainties in the determination of the critical conditions, including critical values of mixture expansion ratio, in the detonation cell size data, the laminar burning velocity, the laminar flame thickness and the turbulent flame velocity. To overcome these limitations, recent programs were launched in ICARE Lab in collaboration with IRSN with the objective of characterizing turbulent flame velocity for hydrogen-air-steam mixtures based on experiments performed in a spherical bomb [16]. Complementary investigations are on-going in the framework of the French MITHYGENE project [17].

The extension to hydrogen and carbon monoxide flames is foreseen in framework of the SAMHYCO project.

4. Conclusion

After the Fukushima Dai-ichi accidents, several R&D projects were launched in Europe with the objective of enhancing the knowledge regarding the combustion hazard and the associated mitigation systems. This paper summarized the state of the art as well as on-going and planned R&D related to hydrogen combustion and mitigation in Europe.

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Hydrogen safety research in Europe

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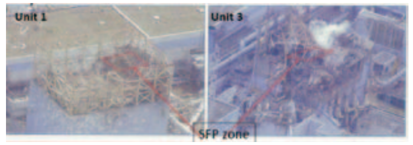
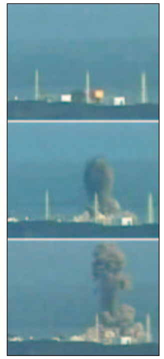
Severe accident and hydrogen explosion in NPP

Three Mile Island Nuclear Power Plant





Hydrogen explosion during Fukushima accident

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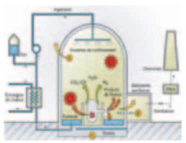
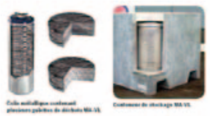
Hydrogen risk in nuclear industry





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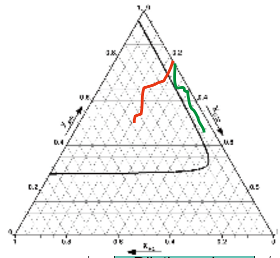
Hydrogen risk mitigation strategies

Limiting oxygen c

Limiting hydrogen



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s and LWRs

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most European NPPs

Dilution using
large volume

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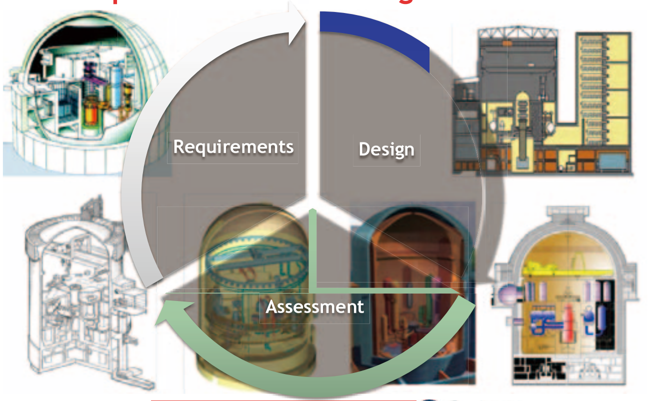
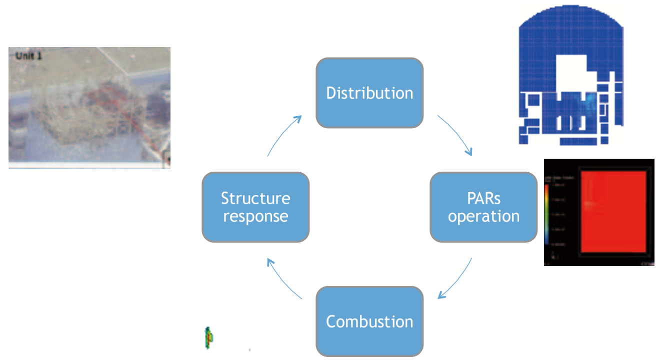
Implementation of mitigation means



Implementation of mitigation means

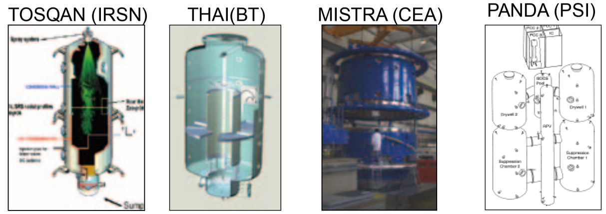


Distribution

Objective: to predict the **composition of the gaseous mixture** inside the containment at each location and at each time

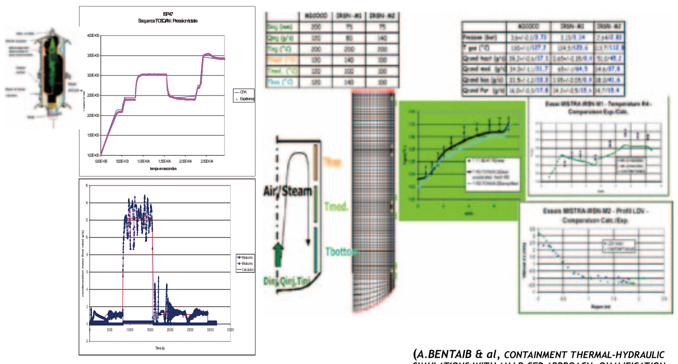
- Validation of LP and CFD tools on SETs and IETs
  - ➡ *Condensation*
  - ➡ *Stratification build-up*
  - ➡ *Erosion of stratification*
  - ➡ *Scaling effect*
- Application to reactor scale

Recent Experimental programs in support of tools validation



TOSQAN & MISTRA programs	THAI (2007-2010)	SETH (2003-2006)	SETH2 (2007-2010)	ERCOSAM (2010-2013)
-SETs condensation	one-phase Stratification/Erosion	SETs one-phase Plumes jets Stratification build-up	one-phase Stratification/ Erosion (PAR, Cooler, Spray)	Scaling effect: accident scenarios scaled from real plants to different facilities
ISP 47				

Topic 1: Condensation



(*A.BENTAIB & al, CONTAINMENT THERMAL-HYDRAULIC SIMULATIONS WITH AN LP-CFD APPROACH: QUALIFICATION MATRIX OF THE TONUS CODE, ICONE-14, MIAMI, USA 2006*)

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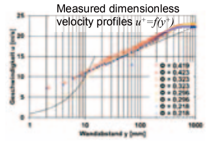
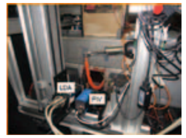
Topic 1: Condensation

- Separate Effect Test for COndensation Modeling (SETCOM @JÜLICH)
  - 6m inclinable test section
  - Optical measurement inside condensing boundary layer under mixed convection heat and mass transfer
  - Development of scalable wall functions for coarse mesh application

(*J.LEHMKUHL & al, Improvement of wall condensation modeling with suction wall functions for containment application, Nucl. Eng. Des. 299, pp. 105-111*)



Application to TOSQAN ISP47



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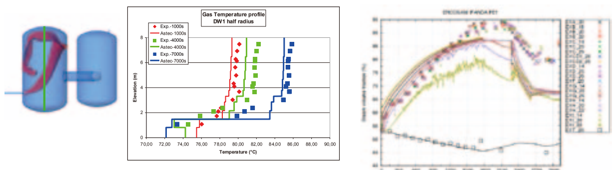
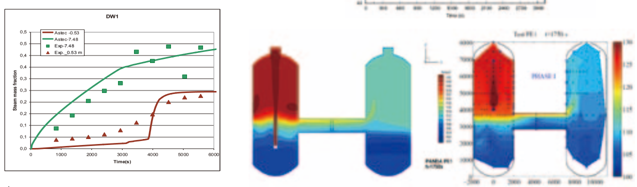
Topic 2: Stratification build-up





(*A.BENTAIB & al, ASTEC validation on PANDA tests, Nureth13, Kanazawa, Japan 2009*)

Fluent temperature distribution at t=1750s

Experimental temperature distribution at t=1750s

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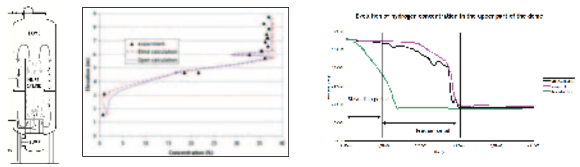
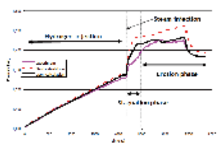
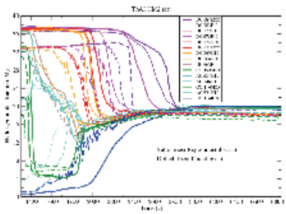
Topic 3: Stratification erosion







(*A.BENTAIB & al, ASTEC validation on OECD/THAI HM-2, ICAPP, Nice, France 2011*)

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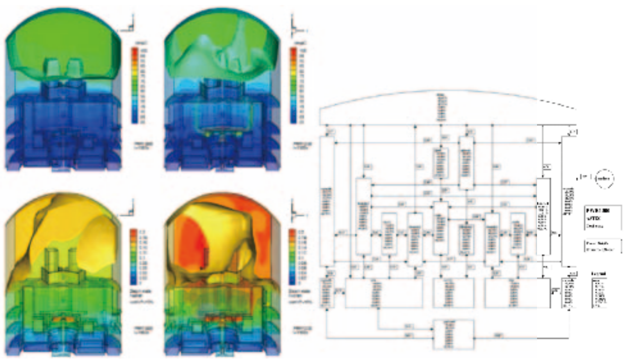
Topic 4: Scaling effect



Mitigation-Recombiners

Objective: to predict the PARs behavior in all situations

- Validation of LP and CFD tools on SETs and IETs
  - Oxygen starvation
  - PARs ignition phenomena
  - PAR interaction with carbon monoxide, fire products, aerosols
  - PAR start-up
- Application to reactor scale

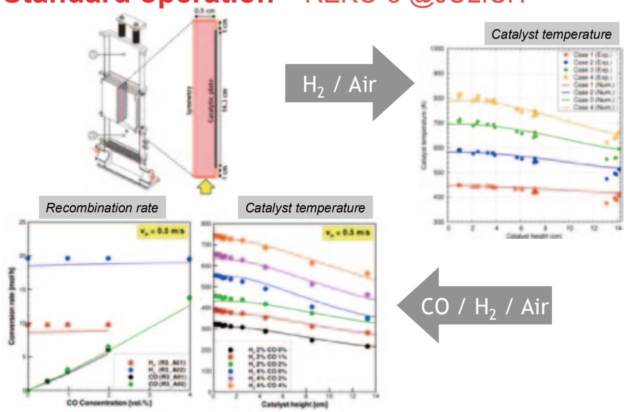
Current experimental facilities (JÜLICH)

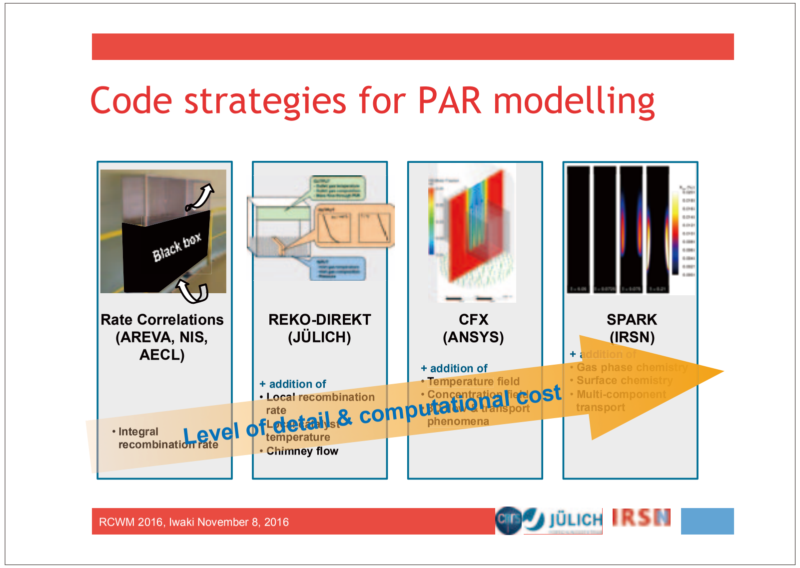
- Catalytic coupons
- PAR hydrogen ignition
- Up to high H_2 concentrations

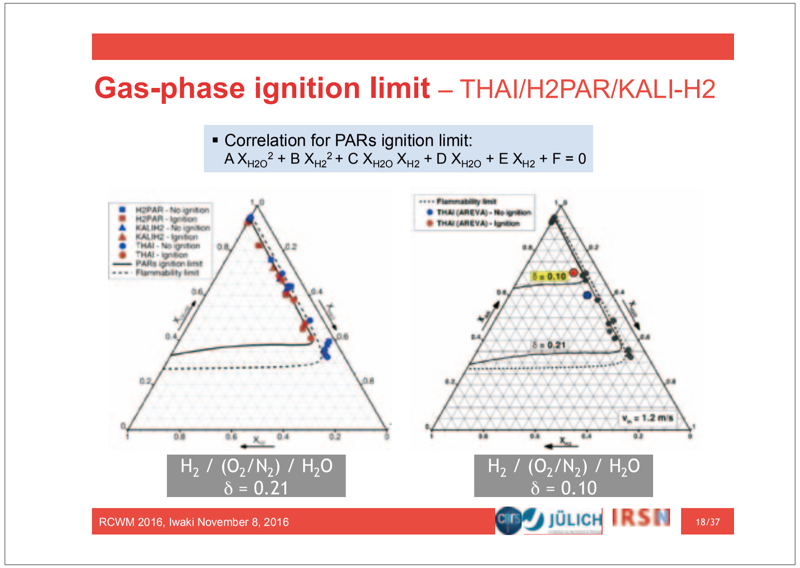
- Catalytic coupons
- PAR hydrogen ignition
- PAR interaction with CO
- Onset of recombination

- FZJ recombiners
- Impact of PAR location
- PAR hydrogen ignition
- Onset of recombination

Standard operation – REKO-3 @JÜLICH




Code strategies for PAR modelling
Black box
Rate Correlations (AREVA, NIS, AECL)
• Integral recombination rate
REKO-DIREKT (JÜLICH)
+ addition of
• Local recombination rate
• Local catalyst temperature
• Chimney flow
CFX (ANSYS)
+ addition of
• Temperature field
• Concentration field
• Multi-species transport phenomena
SPARK (IRSN)
+ addition of
• Gas phase chemistry
• Surface chemistry
• Multi-component transport
Level of detail & computational cost
RCWM 2016, Iwaki November 8, 2016


Gas-phase ignition limit – THAI/H2PAR/KALI-H2
▪ Correlation for PARs ignition limit:
$A X_{H2O}^2 + B X_{H2}^2 + C X_{H2O} X_{H2} + D X_{H2O} + E X_{H2} + F = 0$
H2PAR - No ignition
H2PAR - Ignition
KALIH2 - No ignition
KALIH2 - Ignition
THAI - No ignition
THAI - Ignition
PARs ignition limit
Flammability limit
Flammability limit
THAI (AREVA) - No ignition
THAI (AREVA) - Ignition
δ = 0.10
δ = 0.21
H_2 / (O_2/N_2) / H_2O
δ = 0.21
H_2 / (O_2/N_2) / H_2O
δ = 0.10
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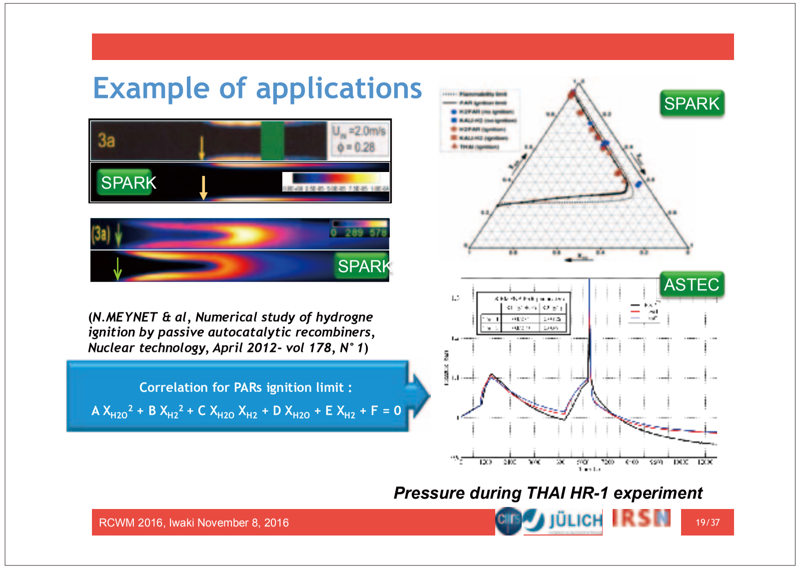
Example of applications
3a
SPARK
(3a)
SPARK
SPARK
ASTEC
(N.MEYNET & al, Numerical study of hydrogne ignition by passive autocatalytic recombiners, Nuclear technology, April 2012- vol 178, N° 1)
Correlation for PARs ignition limit :
$A X_{H2O}^2 + B X_{H2}^2 + C X_{H2O} X_{H2} + D X_{H2O} + E X_{H2} + F = 0$
Pressure during THAI HR-1 experiment
RCWM 2016, Iwaki November 8, 2016
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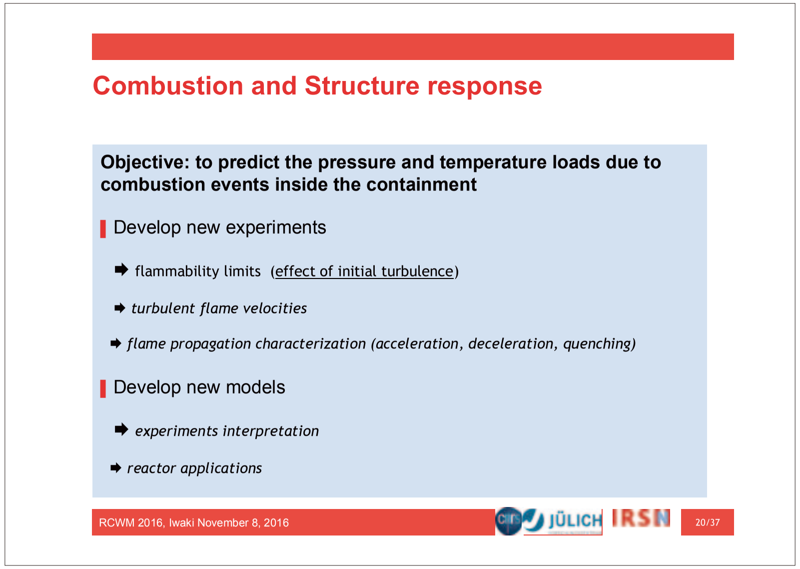
Combustion and Structure response
Objective: to predict the pressure and temperature loads due to combustion events inside the containment
Develop new experiments
flammability limits (effect of initial turbulence)
turbulent flame velocities
flame propagation characterization (acceleration, deceleration, quenching)
Develop new models
experiments interpretation
reactor applications
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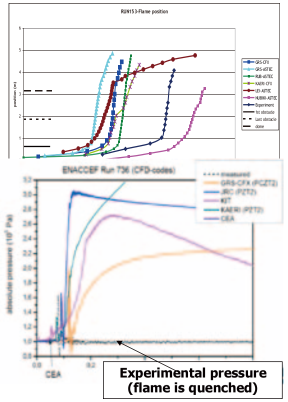
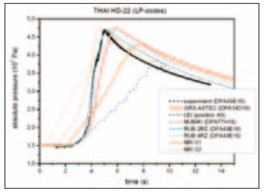
Combustion modeling status

(ISP49 et benchmark SARNET outcomes)





- Combustion models need to be improved to cover the entire range of flame propagation regimes (acceleration, deceleration)
- To date there is no « approved » flame quenching model, neither global, nor local (can be important for the correct prediction of combustion completeness).

Experimental pressure (flame is quenched)

Need of further investigation

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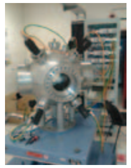
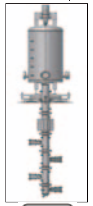
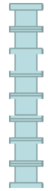
Combustion ongoing programs

SF3 (ICARE) | ENACCEF (ICARE) | ENACCEF2 (ICARE) | THAI(BT)







0,07 m^3 — 1 m^3 — 2 m^3 — 60 m^3

OECD Project : THAI, THAI 2, ISP49
European project : SARNET
National project : MITHYGENE

- Determination of flame acceleration criteria
- Improvement of the existing combustion models
- Investigation of scale effect
- Definition of users best guidelines

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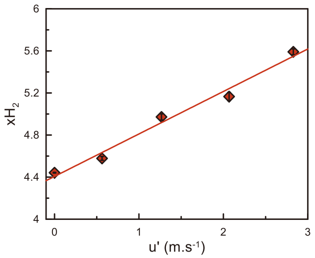
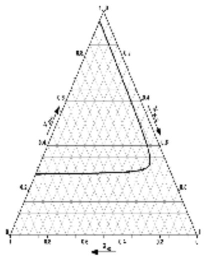
FLAMMABILITY LIMIT: initial turbulence effect





LFL increases linearly with the speed fluctuation

Jules Goulier PhD (University of Orléans, November 2015)

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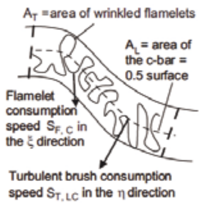
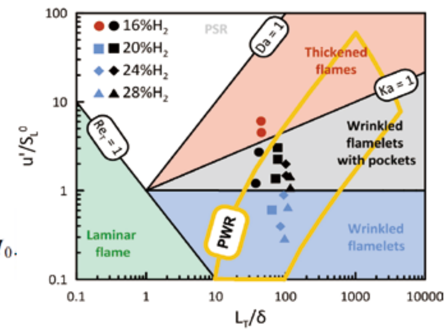
Combustion modelling



$$\frac{S_T}{S_{L0}} = \frac{A_T}{A_L}\frac{\bar{S}_{F,C}}{S_{L0}} = \frac{A_T}{A_L} I_0.$$



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NEW SPHERICAL VESSEL

SPHERICAL VESSEL

- Internal diameter 563 mm
- Volume 95 L
- P_{max} = 200 bar
- T_{max} = 300°C (593K)
- 4 windows of 200 mm optical diameter
- 2 piezo-electric pressure transducers
- Central ignition with Molybdenum electrodes

TURBULENCE

- 8 fans of 130 mm diameter
- 0 to 12 000 rpm

VISUALIZATION

- Z shape Schlieren
- Laser tomography
- Particle Image Velocimetry

Movie
Spherical
flame

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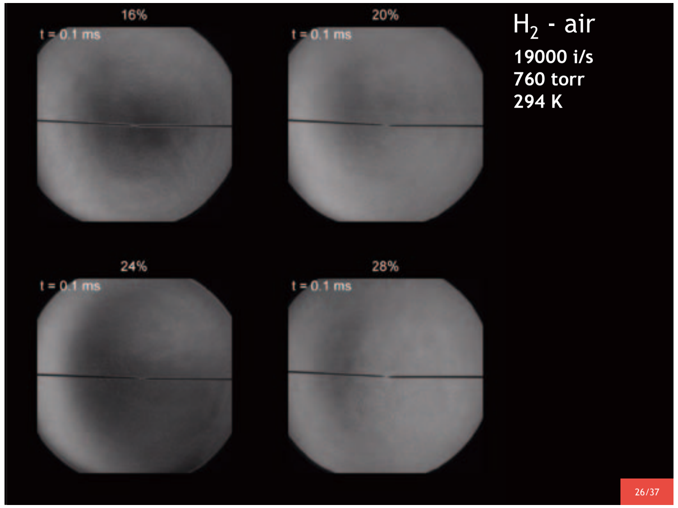

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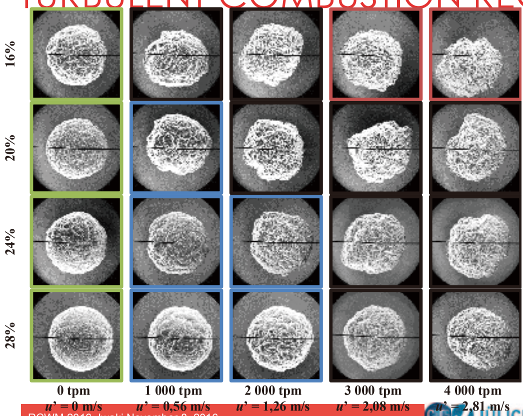
TURBULENT COMBUSTION REGIME



Flame diameter
ø = 120 mm

Green : laminar flames

Blue : wrinkled flamelets

Black : wrinkled flames with pockets

Red : thickened wrinkled flames

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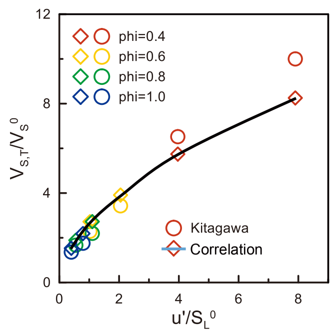
NEW CORRELATION



Only lean hydrogen turbulent flame to our knowledge in the literature

Good agreement of our correlation on Kitagawa experiments

Kitagawa, T. *et al.* Turbulent burning velocity of hydrogen–air premixed propagating flames at elevated pressures. *International Journal of Hydrogen Energy* **33,** 5842–5849 (2008).

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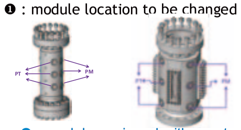
ENACCEFII: New Flame acceleration facility

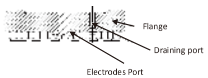
MITHYGENE Project

- Modular
- Obstacles (different shape and blockage ratios)
- Windows for optical diagnostics

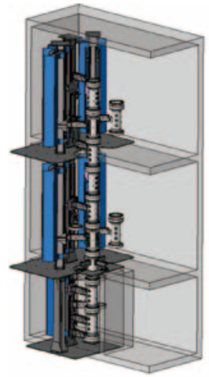
Type ❶ : module location to be changed with type ❷



Type ❷ : module equipped with quartz windows



Type ❸ : Ignition



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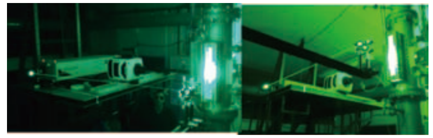
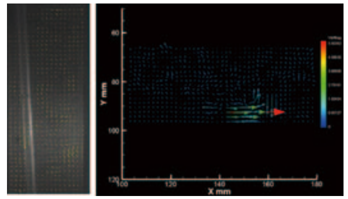
ENACCEFII: New Flame acceleration facility

MITHYGENE Project





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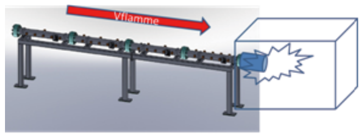
SSEXHY: Structure response to pressure loads

MITHYGENE Project

Facility built at CEA, tests under specification



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On going developments

MITHYGENE Project

Experimental

- Turbulent flame characterization within complementary u' and H2 values
- Flame propagation in dry air
- Flame propagation in wet air
- Effect of spray on flame propagation

Modeling

- Implementation of new correlation in IRSN codes
- Analysis of the experimental data
- Reactor application

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Remaining Open safety issues

Hydrogen stratification build-up and dissolution by different means (spray, venting, cooler, ..)

PAR optimal location

H2/CO recombination

PARs deactivation (fire products, CO,..)

H2/CO migration to auxiliary building

Flame acceleration, deceleration, quenching

Effect of Spray on flame propagation

Structure and safety components response

Management of H2/ CO inside containment

Scaling effect towards reactor application (distribution, combustion, structure response)

In-Vessel Phase | Ex-Vessel Phase | Long term

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SAMHYCO PROPOSAL submitted to EC Call H2020

Towards an improvement of safety management procedures for severe accident late phase including hydrogen and carbon monoxide mitigation and explosion risk assessment models

Participant No	Short Name	Participant organisation name	Country
1	IRSN	Institut de Radioprotection et de Sûreté Nucléaire	France
2	CEA	Commissariat à l'Energie Atomique et aux Energies Alternatives	France
3	CNRS	Centre National de la Recherche Scientifique	France
4	PSI	Paul Scherrer Institut	Switzerland
5	NRG	Nuclear Research and Consultancy Group	Netherlands
6	KIT	Karlsruher Institut fuer Technologie	Germany
7	JÜLICH	Forschungszentrum Jülich GmbH	Germany
8	IJS	Institut Jožef Stefan	Slovenia
9	JAEA	Japan Atomic Energy Agency	Japan
10	CIEMAT	Centro de Investigaciones Energeticas Medioambientales y Tecnologicas	Spain
11	EDF	Electricité de France - EDF	France
12	AREVA	AREVA GMBH	Germany
13	LEI	Lithuanian Energy Institute	Lithuania
14	INCT	Institute of Nuclear Chemistry and Technology	Poland
15	SSTC NRS	The State Scientific Technical Centre on Nuclear and Radiation Safety	Ukraine
16	UCSN	University College of Southeast Norway	Norway
17	KAERI	Korea Atomic Energy Research Institute	Korea
18	BT	Becker Technologies	Germany
19	KINS	Korea Institute of Nuclear Safety	Korea
20	NUT	Nagaoka University of Technology	Japan

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Consortium & Budget

Total budget : 6.5 €
EC requested contribution : 3 M€

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SAMHYCO Main objectives

- ❑ Address specifically ex-vessel conditions (long term phases)
- ❑ Improve the level of knowledge regarding :
  - PARs behavior for MCCI conditions (low oxygen concentrations, carbon monoxide, ignition, start-up, poisoning, ….)
  - Hydrogen and carbon monoxide combustion (flammability limits, flame acceleration, effect of oxygen starvation , flame quenching…)
  - Structure response to combustion loads (concrete and metallic structure)
- ❑ Deduce recommendation to enhance
  - SAMG regarding venting actuation criteria
  - PARs optimal location inside primary containment and inside the connected building (annulus, fuel building, ..)

RCWM 2016, Iwaki November 8, 2016 — 36/37


Thank you for your attention!
For further information, please contact:
IRSN
ahmed.bentaib@irsn.fr

3.3 Session 5 Research and Facility Needs for Collaboration

3.3.1 Steady Irradiation Equipment and its Application to WM Researches

Ryuji NAGAISHI
Japan Atomic Energy Agency; nagaishi.ryuji@jaea.go.jp

Steady irradiation equipment of ionizing radiations of (1) γ-ray, (2) electron beam (EB) and (3) X-ray is now planned by Storage Equipment Soundness Evaluation (SESE) Group to be set up in the coming "secondary" CLADS facility next to the primary one at Tomioka, Fukushima. Their main purpose is not only to do advanced researches on radioactive waste management (WM) for Fukushima Daiichi NPS, but also for engineers (nuclear & other industries) and students (universities & high schools) to learn "radiation effects" through experiments.

In the researches, (a) water radiolysis (seawater & aqueous salt solutions), (b) solid material degradation (metals, alloys & oxides) and (c) radiation-resistance of sensors (elements & units) will be studied in the present plan. To attain them effectively, practical conditions of irradiation fields and/or environments around the wastes should be well reproduced in the experiments.

Specification of Irradiation Equipment

Radioactive nuclides to be removed mainly in Fukushima Daiichi NPS, as well as TMI-2, are Cs-137 and Cs-134 as γ-emitters and Sr-90 and Y-90 as ß$^-$-emitters as shown in Table 1.

Gamma cell-type irradiation equipment (1) with Co-60 source (E_{av} = 1.25 MeV, $t_{1/2}$ = 5.27 y) is planned for the γ-emitters. The dose rate just after supplied will be *ca*. 10 kGy/h, comparable to that in spent adsorption vessel (decay heat: 300-1,000 W) after decontamination of radioactive water. EB accelerator (2) is planned for the ß$^-$-emitters. The maximum energy of incident electron, equal to acceleration voltage, will be 800 or 950 keV. The dose rate, mainly dependent on acceleration current, will be more than 1 kGy/s, comparable to that in normal LWR (mainly neutron & γ-ray). X-ray irradiation equipment (3) is planned for bremsstrahlung from the g- and ß$^-$-rays interacting with metals. The maximum energy of X-ray, equal to tube voltage, will be 150 to 300 keV. The equipment of EB and X-ray is not that restricted in Japanese law concerning prevention from radiation hazards due to radioisotopes, etc.

Table 1 g- and b$^-$-rays emitted from radioactive nuclides.

Nuclide		Cs-137		Cs-134		Sr-90	Y-90	Co-60
Radiation		γ-ray	ß-ray	γ-ray	ß-ray	ß-ray	ß-ray	γ-ray
Energy (keV)	Maximum	662	512 [*1]	698	658 [*1]	546	2,280	1,250
	Average		188		157	196	934	
Average energy of recoiled electron (keV) [*2]		253	-	272	-	-	-	688

*1 primary ß$^-$-ray. *2 obtained from average energy of Compton-scattered photon.

Further Consideration of "Complex Systems" for Radiation Effects

Radiation effects in combinations of water, seawater salts and/or solids as complex systems have been studied at ambient conditions since the Fukushima accident. However, the systems and conditions more specific for radioactive wastes should be further considered for the effects.

For example, in the radiolysis studies (a) on H_2 generation in spent vessel with concentrated radioactive materials, the adsorbent bed of insulated oxides mixed with water after its long-term storage would become anhydrous condition, at which water radiolysis is hard to take place. In the degradation studies (b) of solid oxides, they are contacting with metals in many practical cases, and then the metals would interact with primary radiations to emit secondary lower-energy photons and electrons on the surface of metals, mainly dependent on their thickness.

On the other hand, when the effects are limited to only radiolysis and radiation damage in the narrow sense, they would become initial and partial processes in a series of complicated phenomena in the complex systems. In other words, the phenomena could not be explained only in terms of the effects.

Technical Seminar on Safety Research for Radioactive Waste Storage (closed)
LATOV(6F Seminar room A+B), Iwaki, Fukushima, 8 November 2016

Session 5 Research and facility needs for collaboration

Steady Irradiation Equipment and its Application to WM Researches

Japan Atomic Energy Agency (JAEA)
Collaborative Laboratories for Advanced Decommissioning Science (CLADS)
Ryuji Nagaishi

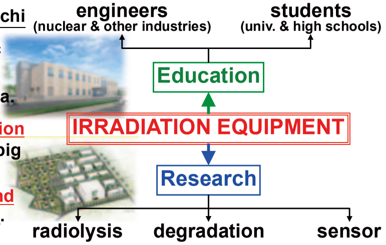
1. Irradiation equipment planned in CLADS
2. Specification of the planned equipment
3. Consideration of systems for radiation effects
4. Introduction to radiolysis researches

Plan of Irradiation Equipment in CLADS 2/12

CLADS building at Tomioka-machi

- The 1st building (next March): basic analysis and research apparatus.
  ⇒ "start-up" of CLADS in Fukushima.

★The 2nd building (planned): irradiation equipment (mainly γ-ray & EB) and big machine for testing fuel assembly.
⇒ indispensable for international and domestic collaboration in CLADS.



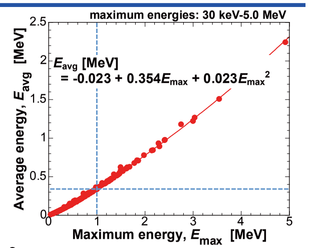
research theme	medium & condition	radiation effect	relation with radiation dose
water radiolysis	water, dissolving salts and coexisting solids	radiolysis directly	quantitative, extensive
degradation of solid materials	metals, alloys and oxides contacting with aq. solns.	radiolysis indirectly (partially damage)	qualitative, intensive
radiation-resistance of sensors	sensing elements and units (devices)	radiation damage (hitting events)	stochastic

Radioactive Nuclides important for 1F 3/12

Wastes emitted from 1F accident

- α-emitters (TRU): exist mainly as "debris" and "spent fuels" to be treated.
  ⇒ *better* INTERNAL irradiation with sources.

★γ- and β-emitters (FP): exist mainly as "radioactive water" and "secondary wastes".
⇒ primary nuclides of Cs-137, Cs-134, Sr-90 & Y-90 in the long-term storage of wastes.
⇒ *more convenient* EXTERNAL irradiation.



β^--ray with continuous energies

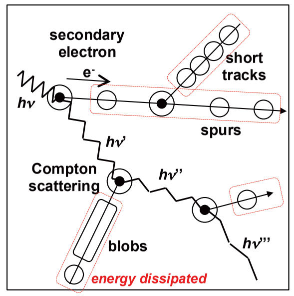
Radioactive nuclide		Cs-137		Cs-134		Sr-90	Y-90	Co-60
Radiation (quality)		γ-ray	β⁻-ray	γ-ray	β⁻-ray	β⁻-ray	β⁻-ray	γ-ray
Energy (keV)	Maximum	662	512*1	698	658*1	546	2,280	1,250
	Average		188		157	196	934	
Recoiled electron (keV)	Maximum*2	478	-	511	-	-	-	1,038
	Average*3	253		272				562

*1 primary β⁻-ray. *2 obtained from minimum energy of scattered photon (Compton edge).
*3 obtained from average energy of Compton-scattered photon.

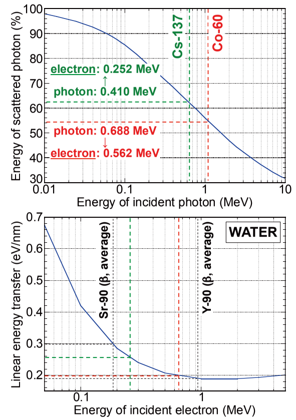
Energy Dissipation of Gamma-rays to Water 4/12

Interaction of photon with matter

- All interactions (photoelectric effect, Compton scattering, electron-pair creation) emit electrons to dissipate energy.
  ⇒ Main interaction for Cs-137 & Co-60 is Compton scattering.



Energy dissipation process of photons



Average energy of scattered photon [U] and energy dependence of LET (e) [B]

5/12

Irradiation Equipment for Radiation Effects

	Source	Radiation: Quality*1	Radiation: Mixture*2	Energy continuity*2	Irradiation beam*2	Remarks
EXTERNAL	① Radioisotopes (sealed)	α, β±, γ, n	Single/Mix.	Mono./Cont.	Steady	^{60}Co, ^{137}Cs, ^{90}Sr-Y..
EXTERNAL	③ X-ray irradiat. equip.	X	Single	Continuous	Steady	efficiency: EZ/(1,600mc^2)
EXTERNAL	Accelerator (LINAC)	e±, p, nucld.	Single	Monochro.	Pulse/Steady	HF e-field, multiple acc..
EXTERNAL	② Accelerator (others)	e±, p, nucld.	Single	Monochro.	Steady	electrostatic, ring..
EXTERNAL	Reactor	n (, γ)	Single/Mix.	Monochro.	Steady	water/C moderated
IN	Radioisotopes (unsealed)	α, β±, γ, n	Mixture	Mono./Cont.	Steady	RI, nucl. materials

*1 Meson, SR, bremsstrahlung, etc. are not included.
*2 Possible to change to single, monochromatic or pulsed beam.

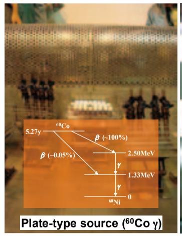
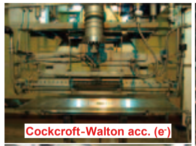
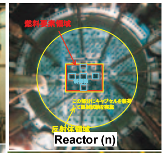
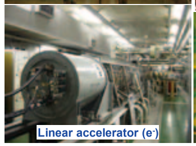


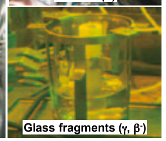
Plate-type source (^{60}Co γ)


Gammacell (^{60}Co γ)


Cockcroft-Walton acc. (e-)


Reactor (n)


Linear accelerator (e-)


Glass fragments (γ, β-)

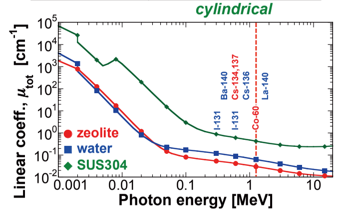
6/12

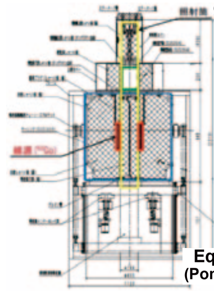
Gamma-ray Irradiat. Equipment, *planned*

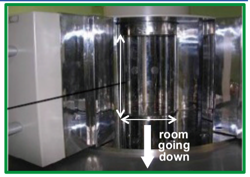
Gammacell-type equipment (Co-60)

- γ-ray: ones emitted from Cs-137 and Cs-134, of which main interaction is Compton scattering.
- researches: water radiolysis, degradation of solid materials and radiation-resistance of sensors.

energy	av.1.25 MeV (1.17, 1.33) *single*
dose rate	max. 10.0 kGy/h (initial)
rate change	radial and axial
irradiat. room	15 cmϕ × 20 cmh *cylindrical*







(a) irradiation room (down to sources)


(b) source vessel

Equipment design (Pony Industry Co., Ltd.)

7/12

2014 Spring Meeting of Atomic Energy Society of Japan (AESJ), G18, Tokyo, JPN, 9 March 2014.

Gamma-ray Irradiation: equipment & holder

Irradiation Facilities (Co-60 source)

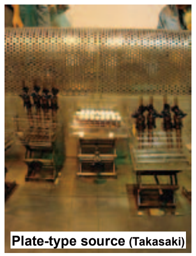
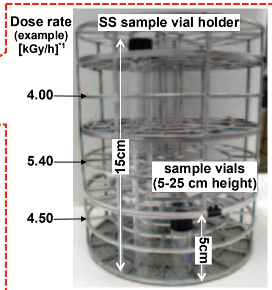


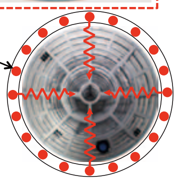
Plate-type source (Takasaki)

Gammacell 220 (Tokai)



	Takasaki (QST)	Tokai (JAEA)
energy	av.1.25 MeV (1.17, 1.33)	
dose rate [rate control]	15-0.5 kGy/h [distance 0-200 cm]	5-0.5 kGy/h [Pb plate covering]
irradiat. room	200cm×200cm×2	15cmΦ×20cmH
irradiation	scheduled (hr)	occupied (min)

pencil-type Co-60 source



Sample setting in Gammacell [upper] and irradiation to samples [bottom]

*1 Dose rate are mainly dependent on numbers of sample vials and wire-meshed partitions.

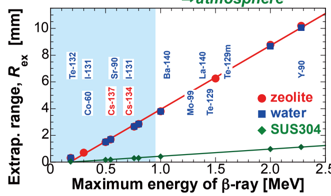
8/12

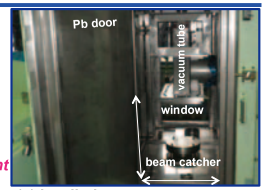
Electron Irradiation Equipment, *planned*

Electrostatic-type EB accelerator

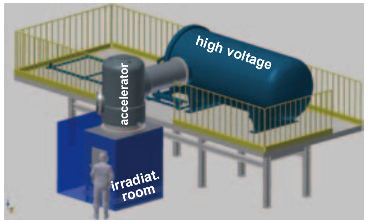
- EB: β--rays emitted from Sr-90 and Y-90, of which energies are average and/or maximum.
- researches: water radiolysis and degradation of solid materials (accelerated tests).

energy (voltage)	max. 0.80 to 0.95 MeV *single at incident*
dose rate	max. 1.0 kGy/s
rate change	current distance from window
irradiat. room	40cmw×50cmd×30cmh *⇒atmosphere*





(a) irradiation room (Pb shielding)



(b) general view (HV set above SUS frame)

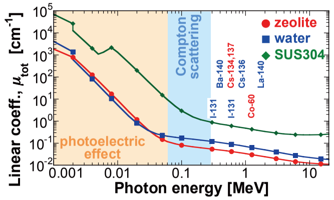
Equipment design (NHV Corporation)

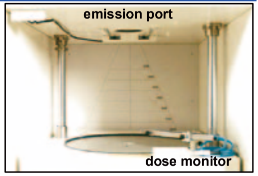
X-ray Irradiation Equipment, *planned* 9/12

X-ray irradiation equipment

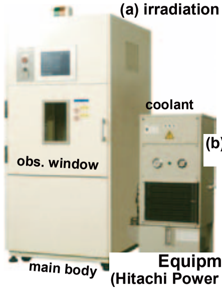
- X-ray: bremsstrahlung emitted from γ- and β⁻-rays interacting mainly with metals.
- researches: water radiolysis, degradation of solid materials and radiation-resistance of sensors.

energy (voltage)	max. 0.15 to 0.30 MeV *continuous*
dose rate	max. 3.0 kGy/h
rate change	current and filters distance from port
irradiat. room	50 cm cubic





(a) irradiation room (Pb shielding)



(b) operation display (preset entry)

Equipment design (Hitachi Power Solutions Co., Ltd.)

Complex Systems for Radiation Effects 10/12

Reaction systems and conditions more specific for the radioactive waste management in Fukushima Daiichi NPS (1F) should be further considered in the radiation effects.

1. Normal complex systems of radiation effects: mixtures of water, solutes and solids (metal, alloy and oxide)
   1) Single reaction systems: radiolysis in aqueous solutions containing solutes at high concentrations (e.g., seawater)
   2) Mixtures of different reaction systems: radiolysis in the submerged zeolite bed (spaces among particles and pores inside particle)
2. Events (phenomena) originated from a series of processes (chemical or physical) containing initial radiation effects
   1) Multi-layer evolution of radiolysis or radiation damage: effects of liquid depth and flow on H2 generation
   2) Multi-step evolution of processes containing the initial process of radiolysis or radiation damages: overflow of water from HIC

↓

Reproduction of condition & necessity of irradiat. experiment

Radiolysis Researches for Collaboration 11/12

PROPOSER: Ryuji NAGAISHI, Ph.D., Principal Scientist, Collaborative Laboratories for Advanced Decommissioning Science (CLADS), Japan Atomic Energy Agency (JAEA)

1. Irradiation Experiments: data acquisition in radiolysis of water necessary for nuclear engineering after Fukushima
   1) Alpha-radiolysis (high LET): damp materials and/or solution systems
   2) Development of irradiation facilities and their applications (CLADS)
   3) Pulse radiolysis: high-concentrated salt systems (seawater, nitrate)
2. Radiolysis Analysis (*urgent*): radiolysis in multilayers of complex systems to reproduce various engineering conditions
   1) Homogeneous effects such as size (liquid depth) and liquid flow
   2) Inhomogeneous effects such as dose rate ("inter-spar")
3. Development of Radiolysis Models (*advanced*)
   1) Solution systems with suspended materials: heterogeneous irradiation and reaction (interaction) fields
   2) Modifications of diffusion kinetics model: salt effects

R. Nagaishi, a CLADS' project on "*Development of Comprehensive Analysis Technique of Water Radiolysis Behavior*" 12/12

Collaboration Plan for Alpha-Radiolysis

1. Background & Purpose

・Overestimation of gases generated from waste forms of α-emitters based on conventional data, *except for those from solutions*

⇒Rational estimation based on latest experimental data

2-1. Candidate nuclides of α-emitters (mass, m_α)

nuclide (Z)	oxidation states[*1]	half lifetime $T_{1/2}$ [year]	av. α-energy E_α [MeV]	specific emitting rate q_α [J g^{-1}s^{-1}]	E_α/Q_{rec}[*2]
Pu-238 (94)	0, +3∼+7	87.7	5.487	0.557	0.981
Am-241 (95)	0, (+2)+3∼+7	432.6	5.480	0.111	0.978
Cm-244 (96)	0, +3, +4	18.10	5.795	2.781	0.982

2-2. Samples of materials (*M*) containing α-emitters

number	1 (Liquid)	2 (Solid)	3 (S)	4 (S)	5 (S)	6 (S)
sample	aqueous solution system	cement	burned ash	soil	debris of oxide-alloy mixture	geopolymer
bulk density [g/mL]	1∼2	1.5	1∼2	1∼2	4∼8	1∼3

comparison

*1 A.J. Bard, R. Parsons, J. Jordan, *"Standard Potentials in Aqueous Solution"*, IUPAC, Marcel Dekker, Inc., NY (1985).
*2 Recoverable average energy, Q_{rec}: J. Katakura and H. Yanagisawa, *JAERI-Data/Code 2002-021*, JAERI (1994).

3.3.2 Research Maps - Tools for Aligning Fundamental and Applied Research with Operations

Kevin M. KOSTELNIK [1, 2] and Stephen M. KING [2]
[1] *Savannah River National Laboratory; kevin.kostelnik@srs.gov*
[2] *Savannah River Nuclear Solutions, LLC; stephen.king@srs.gov*

Roadmaps are graphical, long-range, strategic plans that have been used by government, industry, and academia in a variety of technology-intensive fields. Science and Technology (S&T) Roadmaps are tools that can illustrate the integration of technological advancements into operational or business systems. Research Maps are a type of S&T Roadmap that focus on defining and enhancing the underlying scientific knowledge and required advancements which are necessary to propel technological development to the point that improvements can be made to existing (baseline) operations.

A variety of techniques have been successfully used to create Research Maps because the process is both flexible and applicable to a wide range of fields. While flexible, the process also provides a structured framework for evaluating and communicating the underlying research needs to complex systems over time, such as long-term remediation efforts. A recent US Department of Energy (USDOE) workshop that documents the *Basic Science Needs for Environmental Management* serves as an example approach to defining basic science needs and their alignment with remedial operations.

Important considerations during the initial Planning Phase of the "Mapping" effort should include defining the intended audience of the Map, the use and purpose of the Map, the timeframe being considered, and the desired format of the final product. Multi-organizational development of Maps, although complex, provides diverse input and if developed transparently, builds consensus among key stakeholders by showing direct ties to their interests and issues. Establishment of a cross-functional core team to lead the Mapping effort is essential for project success.

Developing, maintaining and implementing a Research Map can be even more important than the Map itself. The process requires continually strong communication, consensus building, prioritization, decision-making and long-term commitment (social, political and financial).

Research Mapping examples from relevant complex environmental systems within the USDOE will be presented. These examples include environmental remediation of contaminated groundwater and waste processing of high level radioactive liquid waste.

The remediation efforts associated with the Fukushima Daiichi NPP present significant long-term challenges. The current *Mid-and-Long-Term Roadmap* provides a strong, unifying foundation which guides this complex remediation effort. Expanding upon and integrating applicable, fundamental, science considerations into these strategic planning efforts could provide valuable support to this national/international effort.

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Research Maps
– Tools for Aligning Fundamental and Applied Research with Operations

Kevin Kostelnik, Stephen King
Savannah River National Laboratory

November 8, 2016

SRNL-MS-2016-00204

Savannah River Nuclear Solutions

Presentation Outline

- **Introduction**
  - Roadmaps & Research Mapping
  - Tools & Process
- **Examples**
  - Environmental Remediation
  - Waste Treatment
- **Value Proposition**

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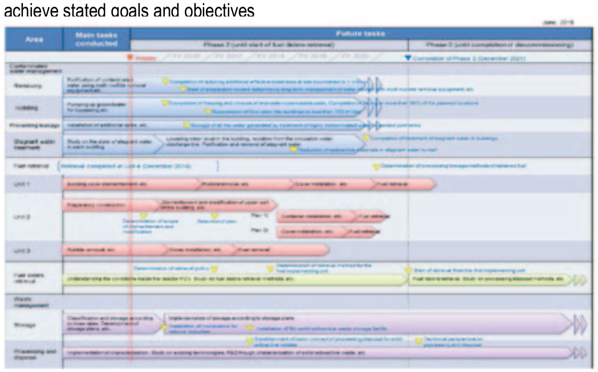
Definitions / Context

- **Roadmaps**
  - Graphical, long-range strategic plans that identify activities and schedules necessary to achieve stated goals and objectives



Source: Mid-and-Long-Term Roadmap towards the Decommissioning of TEPCO's Fukushima Daiichi Nuclear Power Station, June 12, 2015, Inter-Ministerial Council for Contaminated Water and Decommissioning Issues

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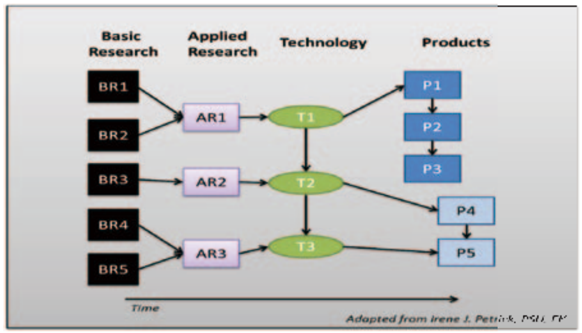
Definitions / Context

- **Research Map**
  - A type of Science & Technology Roadmap, focused on linking underlying scientific knowledge and technological advancements for desired improvements to existing (baseline) operations



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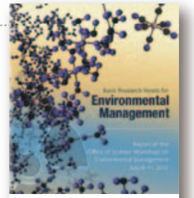
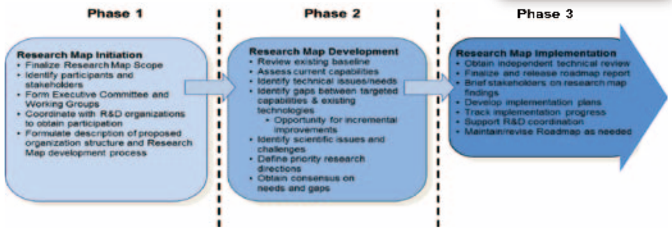
Definitions / Context

- **Mapping (Process)**
  - The flexible process by which a roadmap / map is created, implemented, monitored and updated.
    - *Structured framework*
    - *Various approaches / techniques / outputs*



http://science.energy.gov/~/media/bes/pdf/reports/2016/BRNEM_rpt.pdf



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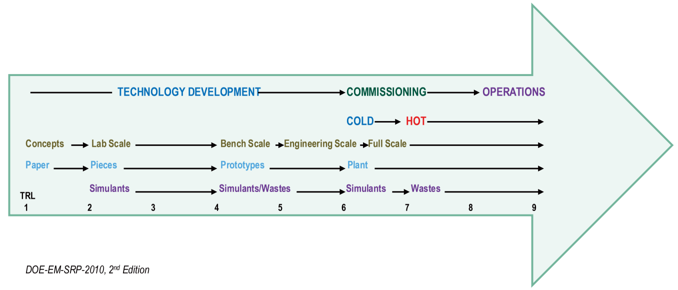
Technology Readiness

- **Technology Readiness Assessments**
  - An assessment of technologies and their readiness for insertion into the project design and execution schedule
- **Technology Readiness Levels (TRL)**
  - An indication of the maturity of a given technology



DOE-EM-SRP-2010, 2nd Edition

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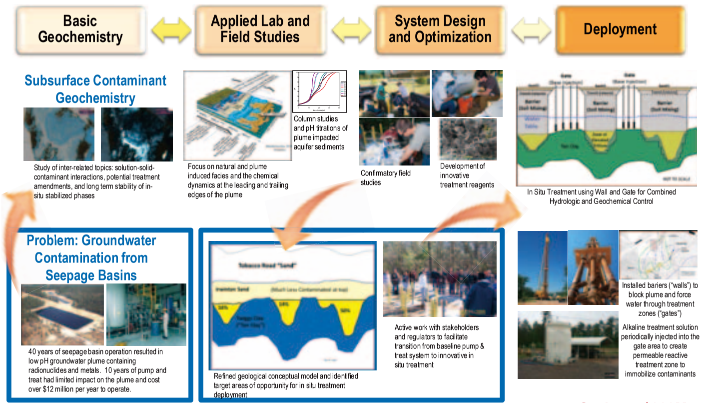
In Situ Groundwater Treatment System – Savannah River Site



Savings: $300M

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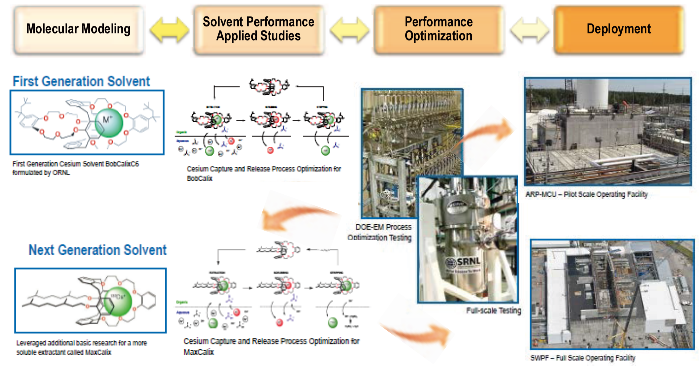
Solvents for Cesium Removal – Savannah River Site



Savings: $600M to $1.8B

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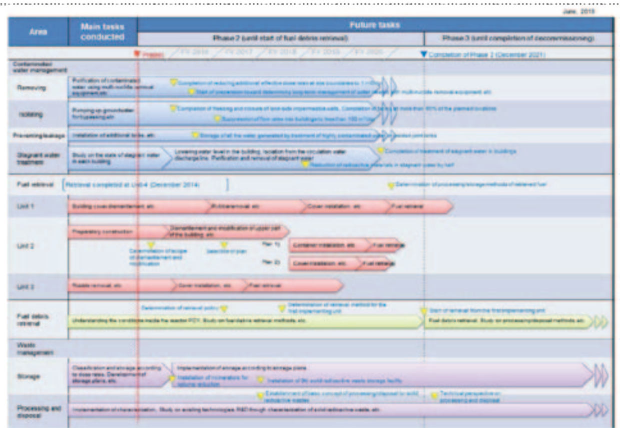
Main Processes for the Decommissioning of Fukushima Daiichi NPS



Source: Mid-and-Long-Term Roadmap towards the Decommissioning of TEPCO's Fukushima Daiichi Nuclear Power Station, June 12, 2015, Inter-Ministerial Council for Contaminated Water and Decommissioning Issues

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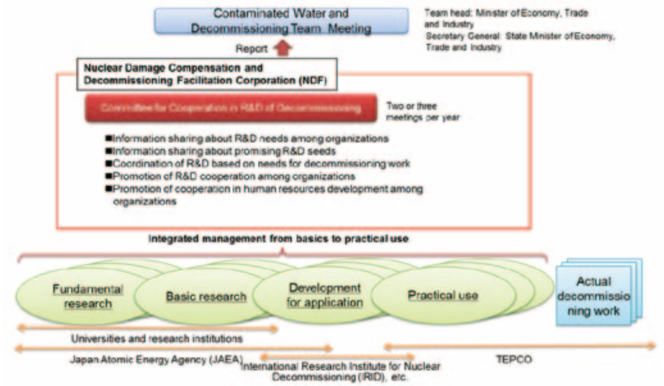
Integrated Management from Fundamental/Basic Science to Practical Use



Source: Mid-and-Long-Term Roadmap towards the Decommissioning of TEPCO's Fukushima Daiichi Nuclear Power Station, June 12, 2015, Inter-Ministerial Council for Contaminated Water and Decommissioning Issues

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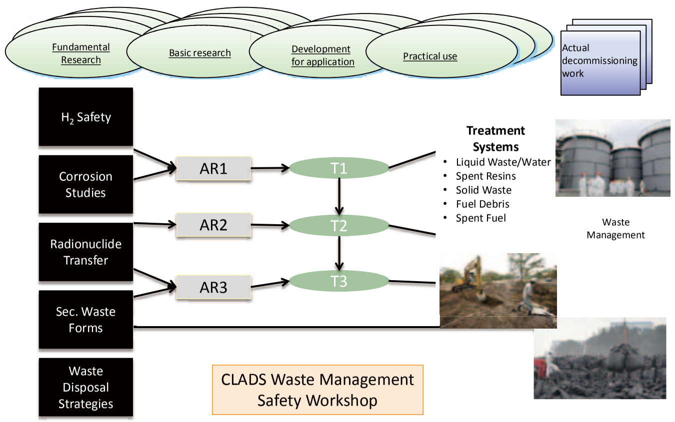
Research Maps help bridge the gap between Basic Science & Application



CLADS Waste Management Safety Workshop

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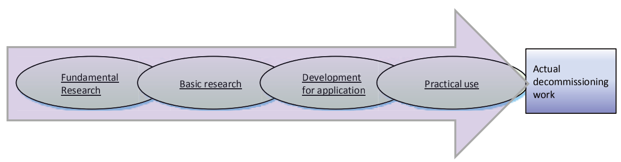
Research Mapping Value Proposition

- Solution-oriented method for aligning R&D with operational needs
- Illustrates the integration of Science, Technology and Practical Application
- Defensible basis for an R&D investment strategy
- Flexible to support decision-makers, scientists, general public
- Opportunity for broad, transparent stakeholder consensus building
- Applicable to long-term timelines



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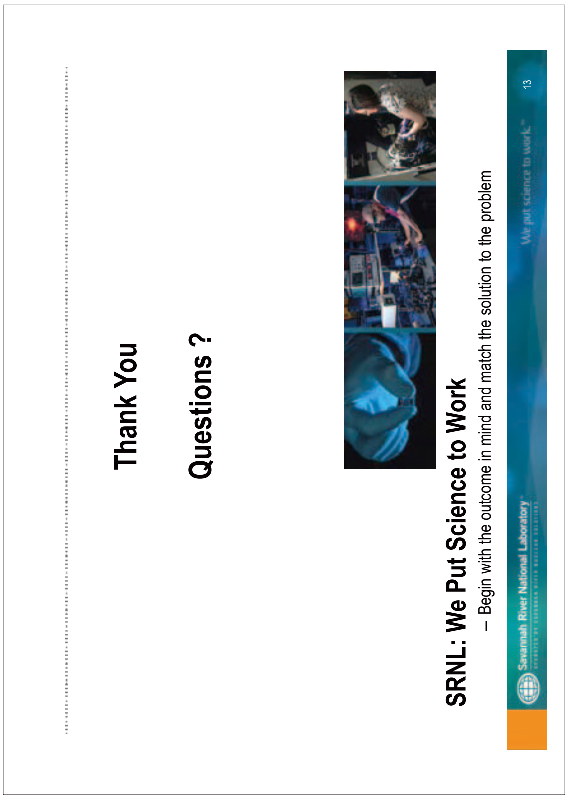
Thank You
Questions ?
SRNL: We Put Science to Work
– Begin with the outcome in mind and match the solution to the problem
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13

Appendix 1: Conference Program

November 7th, 2016, LATOV 6F

12:00 - Registration
13:00 -13:05 Opening Remarks Toru Ogawa (JAEA)

13:05-13:35 Management of Solid Waste Arising from Fukushima Daiichi Decontamination and Decommissioning
Tatsuro Kobayashi (Tokyo Electric Power Company Holdings Inc.)

13:35-13:55 Activities of CLADS Waste Management Division
Takashi Ashida (JAEA)

13:55 -15:25 Session 1 Experiences of Waste Management in Countries
Facilitator: Kevin M. Kostelnik (Savannah River National Laboratory)

- Legacy Waste Management in the United States
Roger Seitz（Savannah River National Laboratory）
- Waste Management for Decommissioning in the UK
Robin Orr (UK National Nuclear Laboratory)
- Plan and Implementation of Nuclear Waste Management in Finland
Kari Rasilainen (VTT Technical Research Center of Finland Ltd.)

15:25-15:40 Coffee break

15:40-17:10 Session 2 Research Activities of Universities
Facilitator: Syunichi Suzuki (The University of Tokyo)

- Recovery and Volume-reduction Solidification of Radioactive Elements from Solid Waste
Kenji Takeshita (Tokyo Institute of Technology)
- Rapid Analysis of Sr-90
Yoshiotaka Takagai (Fukushima University)
- Activity of National Institute of Technology, Fukushima College
Shigekazu Suzuki (National Institute Technology, Fukushima College)

17:10-18:00 Poster Presentation

18:30- Reception (LATOV 3F La•Pa•Pa)

Appendix 2: Seminar Program

November 8th, 2016, LATOV 6F

9:30 - Registration

9:50 -12:00 Session 3 Radiolytic Hydrogen and Corrosion

Facilitator: Takafumi Motooka (JAEA)

- Gamma-Radiolysis Induced Corrosion

  Jungsook Clara Wren (Western University)
- Corrosion Risk Management in Fukushima Dai-ich NPS

  Syunichi Suzuki (The University of Tokyo)
- Radiolytic Corrosion of Carbon Steel and Irradiation Facilities

  Hiroyuki Inoue (Osaka Prefecture University)
- Applications and Insights from Radiolysis Modeling at Pacific Northwest National Laboratory

  Richard S. Wittman (Pacific Northwest National Laboratory)

12:00 -13:00 Lunch

13:00 -14:30 Session 4 Hydrogen Safety Research

Facilitator: Jungsook Clara Wren (Western University)

- Activities of the U.S. Department of Energy Facility Contractors Group (EFCOG) to Promote Hydrogen Safety

  Kevin R. O'Kula (AECOM Nuclear and Environment Technical Services)
- Hydrogen Safety Technology: Application of the Automotive Catalyst in a Hydrogen Recombiner

  Hirohisa Tanaka (Kwansei Gakuin University)
- Hydrogen Safety Research in Europe

  Ahmed Bentaib (Institute de Radioprotection et de Surete Nucleaire)

14:30 -15:00 Coffee break

15:00 -16:30 Session 5 Research and Facility Needs for Collaboration

Facilitator: Isao Yamagishi (JAEA)

- Steady Irradiation Equipment and its Application to WM Researches

  Ryuji Nagaishi (JAEA)
- Research Maps - Tools for Aligning Fundamental and Applied Research with Operation

  Kevin M. Kostelnik (Savannah River National Laboratory)
- Wrap-up for International Collaboration

16:30 -16:40 Closing Takashi Ashida (JAEA)

Appendix 3: Abstracts of Conference (In Japanese)

2.1 福島第一原子力発電所の廃止措置で発生する固体廃棄物の管理について

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1. 福島第一原子力発電所の廃止措置で発生する廃棄物
福島第一原子力発電所(以下「福一」)の廃止措置においては、通常の原子力発電所における廃止措置とは異なり、大量かつ多種多様な放射性物質廃棄物(気体・液体・固体)が発生している。これらは、基本的には放射性物質を固体状に濃集し敷地内で安全に管理して環境への移行を抑制している。

2. 固体廃棄物の発生から保管までの流れ
これらの固体廃棄物は、処理・処分までの間、敷地内の一時保管施設で管理し、順次、固体廃棄物貯蔵庫に収納を行っていくが、貯蔵庫の容量にも限りがあることから、必要な技術開発を行い可能な限り減容し、そしてより安定化していく必要がある。

3. 固体廃棄物の一時保管の状況
現在、廃止措置で発生した約 30 万m^3の瓦礫等(瓦礫類、伐採木、使用済放射線防護具)について、(1)瓦礫類(汚染金属、コンクリート、雑可燃物、土壌など)は、表面線量率に応じて屋外集積、シート養生、容器収納、覆土式保管施設、または固体廃棄物貯蔵庫等で一時保管、(2)伐採木(幹、根、枝葉)は、火災リスクを減少させるために屋外集積と保管槽で一時保管、(3)使用済放射線防護具(保護衣、マスク、靴、手袋等)は、焼却されるまでの間、容器収納して一時保管している。
また、汚染水処理で発生する水処理二次廃棄物のうち、(4)使用済吸着塔類は、必要となる放射線遮蔽に応じてラックまたはボックスカルバートに収納して一時保管、(5)廃スラッジ、および濃縮廃液は、既存の建屋内やタンクに一時保管している。

4. 今後の取組み
瓦礫等の発生量は 2028 年頃までに約 75 万m^3に達すると見積もられており、構内の一時保安施設の容量を超過する。このため、廃棄物量の削減に向け、汚染レベルの低い金属やコンクリートを福一の敷地内において、再使用・再利用できるようにしていく。さらに、震災以降に得られた知見や経験を活かして、瓦礫等の物量をできるだけ精度よく予測していく。この物量予測に基づき廃棄物の減容処理設備(増設雑固体廃棄物焼却炉、低線量瓦礫減容施設など)や増設固体廃棄物貯蔵庫を設計し、計画的に導入していく。そして、新たに導入される増設固体廃棄物貯蔵庫が操業を開始した以降、構内の一時保管を段階的に解消していく。
また、水処理二次廃棄物については、大型保管庫を建設して屋内に保管していく。しかしながら、それでもなお漏えいリスクは無くならないことから、多核種除去設備で発生するスラリー状の廃棄物を脱水し、安定化することについて、国際廃炉研究機構(IRID)を中心に模擬装置を用いながら研究開発が進められている。
さらに、将来の処理・処分に向けては、多種多様な固体廃棄物の性状把握は不可欠であるため、JAEA殿おいて、福一の敷地隣接地に新たな分析施設の整備が進められている。

2.2 CLADS 廃棄物処理処分ディビジョンの活動

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1. 廃炉国際共同研究センターの概要

福島第一原子力発電所（1F）の廃止措置に関する研究開発を推進するため、日本原子力研究開発機構に廃炉国際共同研究センター（CLADS）が 2015 年に設立された。CLADS は、次の 4 つのディビジョンから構成されている。

(1)廃棄物処理処分ディビジョン
(2)燃料デブリ取扱・分析ディビジョン
(3)事故進展挙動評価ディビジョン
(4)遠隔技術ディビジョン

CLADS の具体的な取り組みは、次の通りである。

(1)国内外の叡智を結集する場（プラットフォーム）の整備
(2)国内外の廃炉研究の強化
(3)中長期的な人材育成機能の強化
(4)情報発信機能の整備

さらに、国際的な研究開発拠点として、「廃炉国際共同研究棟」を CLADS の中核として建設している。

2. 廃棄物管理に関する研究開発

1F の事故やその後の復旧作業で、多種多様の廃棄物が発生した。この廃棄物の性状は、通常の原子力発電所から発生する廃棄物と異なっている。そのため、廃棄物の性状把握、保管、処理、処分に関する研究開発が中長期ロードマップ[1]に基づいて実施されている。その主な内容は、以下の通りである。

(1)性状把握
- 廃棄物試料の分析
- 放射能量等の評価
- 基礎基盤データの収集
- 難測定核種の分析法の開発

(2)保管
- セシウム吸着装置からの吸着塔の健全性評価
- 廃棄物の保管に関する安全技術の開発

(3)処理
- 処理・廃棄体化技術の調査
- 処理・廃棄体に関する基礎試験
- 技術の絞り込みの検討

(4)処分
- 処分概念の特徴整理・調査
- 性状に応じた処分概念の検討
- 新たな処分概念等の検討

これらの研究成果は、中長期ロードマップに示されている「廃棄物処理・処分に関する基本的考え方」に反映される。

参考文献

[1] 廃炉・汚染水対策関係閣僚等会議、東京電力㈱福島第一原子力発電所の廃止措置等に向けた中長期ロードマップ、平成 27 年 6 月 12 日

2.3.1 米国におけるレガシー廃棄物の管理
（米国における過去の遺産としての廃棄物の管理）

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1. 序章

環境修復作業は結果として膨大な量の廃棄物を生じさせる。この作業は，処理，貯蔵，処分方法の考慮を含む包括的な廃棄物の管理戦略を必要とする。一般公衆に対して情報を提供するという役割を担う規制当局の方法と同様に，これらの方法においては，廃棄物の受け入れ基準（waste acceptance criteria: WAC），適切な性状把握，多様な廃棄体，収納容器，輸送方法に対する考慮が含まれる。複数のUSDOE（United States Department of Energy）の施設において専用の処分用施設が整備されるとともに，新たな処分を可能とする施設が民間の低レベル放射性廃棄物(LLW)処分施設において整備されている(図 1 参照)。化学的有害物質を含む廃棄物の処分のために設計された施設もまた，環境修復作業の結果として生じる低レベルの放射性廃棄物の処分に使用されている。多くの USDOE の施設や民間の施設は化学的有害物質を含む廃棄物の処分に対する規制要件に合致するように設計されている。この結果，多くの USDOE の施設や民間の施設では，放射性毒物と非放射性毒物からなる混合廃棄物（mixed wastes）の処分も可能となっている。 これらの施設は， 個々に最大で1500 万トンになるような処分容量を有している。

図 1：USDOE Hanford(左)とIdaho(右)のサイトにおける遮水設備および集水排水設備を伴う施設。これらの施設は環境修復作業に伴い発生する廃棄物の処分に用いられる。Idaho の施設は，他の様々な環境修復に伴って生じる廃棄物と同様にサイトでの試験施設の廃止措置によって生じえる圧力容器の処分に使用されている。(Courtesy: USDOE)

2. 規制の観点からのアプローチ

環境修復作業と混合廃棄物の処分に関する意思決定は USEPA（United States Environmental Protection Agency）の規制手順に則って実施される。この手順は他の厳密な代替評価手法を含むものである。この手法は公共事業における住民参加制度（外部からの規制監督の意味を含む）に対する要件に沿って実施される。一般的に適用される手法は，意思決定のために構築された体系に含まれる 9 つの基準を含んでいる(表 1 参照)。この手順は，手法の評価や選定された代替案の利点を理解すること，主要なサイトにおいて処分施設を含むことにおいて極めて効果的であることが証明されている。USEPA は多くの規範的な要求事項（処分施設の設計，廃棄物処理など）を示している。これらの要求事項は，規制との整合性や公衆の受容性に関して利点を持っている反面，サイト固有の廃棄物に適用することにおいて課題を提起するものとなっている。処分施設に対する規範的な設計は公衆の受容性に関しては極めてうま

く機能している。けれども，廃棄物処理の規範的なアプローチが，その標準的な処理方法が特定の廃棄物に対して必ずしも最適とならないような状況に導く場合がある。このことは，異なったアプローチの採用を必要とするものである。性能評価において，いくつかの廃棄体の性能をモデル化する場合，モデル化を支援するパラメータの整備が必要になる。有害廃棄物の処分を支持するための性能評価に必要な詳細なデータは通常は取得されていないことから，このことは，新たな挑戦となる。

表 1：意思決定に用いられる9つの基準

規制
人間の健康の保護
連邦及び合衆国の規制との合致
最適化
長期の有効性（WAC に基づく実施や安全評価）
毒性，移動性，含有物の量の低減（処理，廃棄体化）
短期の有効性（作業者の安全の保全，輸送安全）
サイトでの実現可能性（WAC に基づく実施）
経済的有効性
利害関係者
規制の観点からの受容性
社会的受容性

3．これまでに扱ったことのない廃棄物の取りあつかい

環境修復作業は結果として多様な廃棄物を生じさせることになる。そして，このことは管理のための戦略的なアプローチの構築を要求することになる。多量に存在する主要な廃棄物はこれまで定義されているような比較的な単純な廃棄物ではあるものの，膨大な量の処分を要求するものである（土壌とか，軽微に汚染された瓦礫類など。けれども様々な仕様があり，これまで扱ったことのない廃棄物もまた生じている（炉材やイオン交換樹脂など）。これらのより特徴的な廃棄物には，適切な処理，処分方法を同定する検討が必要となる。また，これらは，標準化された規制のアプローチの十分性を試すかもしれない。イオン交換樹脂のような固体二次廃棄物は，合衆国の環境修復作業に伴って生じる廃棄物に関連した様々な戦略的な検討を例示するものである。この発表では，廃棄体をモデル化するために要求される入力条件や仮定を同定するための検討と同様に，いくつかの現実的な取り扱いや処理の検討を提示する。

研究の優先順位を効率的に同定するために，性能評価の活動と分析研究チームの活動を統合することが重要である。初期の性能評価のモデル化は分析結果に対して重要と考えられる仮定やパラメータを同定するために用いられる。研究者および性能評価者は必要のあるキーとなる分野を同定するために共同すべきである。このことにより，検討を効果的に実施す（有意な必要性に焦点を当てた研究の実施等）ことが可能になる。また，このことは，研究者が実施する試験や性状把握の研究の妥当性を提供することにおいて助けとなる。

4．まとめ

処理，貯蔵，処分の代替案を検討する場合，国の戦略において取り扱われる多くの要素を考慮する必要がある。それらは，規制的な検討と技術的な検討の組み合わせである。規制的な検討は，以下のことを含んでいる。

・規範と性能に基づいたアプローチのバランスを取ること
・処理処分施設の WAC を確立すること
・複数の基準を考慮可能な意思決定のアプローチを実施すること

・公共事業における住民参加制度に関連した透明性のあるアプローチを考慮すること。

技術的な検討は，概して，そのほとんどが特異的な廃棄物に対してなされる。技術的な検討では，輸送要件，廃棄体や容器に対する要件などの処理，貯蔵，処分施設に係る WAC を考慮しなければならない。規制や技術的な考慮事項は，廃棄物の最終処分方法に影響を与える可能性のある重大な仮定を同定できるようにするために，環境修復作業に対する全体計画と結び付けられるべきである。モデル化のための膨大な量の入力条件と仮定が与えられた場合，なされるべき意思決定に対して重大な意味を持つデータや仮定に関する検討を最適化するために，データ収集とモデル化検討を統合することは重要である。

（翻訳:CLADS　大井　貴夫）

2.3.2 英国における廃止措置のための廃棄物管理

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英国は民生発電のエネルギーソースとして原子力を採用した最初の国の一つである。実際に，英国のコールダーホール原子炉は 1956 年に世界で最初に商業スケールにおいて発電を行った。最終的には，26 のマグノックス原子炉（黒鉛減速炭酸ガス冷却型原子炉：現在閉鎖）と 14 の改良型ガス冷却炉(AGR)と1つの加圧水型炉(PWR)が英国において建設された。英国における原子力発電施設の開発と操業を支援するため，ウラン濃縮プラント，燃料製造施設，使用済み燃料貯蔵施設，使用済み燃料再処理施設だけでなく，研究炉，原型炉，そして，研究施設が建設された。

研究炉，原型炉と他のプラントの幾つかは，既に英国内において廃止措置されているが，多くの施設は残ったままである。セラフィールドサイトは英国において最も大きな原子力施設サイトであり，最大の廃止措置対象である。廃止措置と環境回復は 2100 年以降にわたっても継続されるだろう[1]。施設の廃止措置は膨大な量の廃棄物を発生させ，その大部分は，低レベル放射性廃棄物（LLW）に分類され，浅地中処分されることになる [2] 。 けれども廃止措置を行うことによって，廃棄物の取り出しや中レベル放射性廃棄物の管理，遺産的な施設に現在貯蔵されている核物質に対しての特別な検討が必要であることがわかる。

1980 年代の初頭，英国は，使用済み燃料再処理施設から生じる固体及びスラリー状の中レベル放射性廃棄物を処理するプログラムをスタートさせた。これによって，実験室スケールの試験から施設運転までの段階的な開発手法が確立された。この研究から，再処理施設から生じる操業副産物を処理するために，4 つの廃棄物セメント固化施設がセラフィールドに建設された。これらの施設は現在もなお，25 年以上にわたって操業中で，5 万以上の廃棄物パッケージを生産している。廃棄物のセメント固化，処分の実現性に関する基準の開発[3]，廃棄体の性能の例示に関して，英国は，かなりの経験を蓄積している。この経験は，廃止措置によって生じる廃棄物の管理のための足場となっている。

セラフィールドにおいて最も優先順位の高いタスクは，過去の燃料再処理事業から生じ，現在高レベルの廃棄物と物質を貯蔵している，高度に汚染された遺産的な施設を廃止措置することである[1]。現在セラフィールドの廃棄物施設で処分される廃棄物とは異なり，遺産的な施設から発生する中レベル放射性廃棄物は，不均質な材料の混合物で，異なるレベルの毒性を示しており，これらの特徴は，廃棄物の取り出し時期によって変化する。加えて，これらの廃棄物の毒性を緩和することが可能な技術に関して重大な制限がある。これらの検討は，前述の廃棄物処理プロセスにおいてよりも，より柔軟で実用的なアプローチを要求する。

この発表では放射性廃棄物の管理に関する英国の体系の概要を提供する。英国の遺産的な施設から廃止措置によって取り出される中レベル放射性廃棄物に課せられた様々な検討についての議論に先立ち，廃棄物処理の開発や再処理廃棄物のセメント固化から得られた経験が要約される。 そして，英国のセラフィールドサイトの廃棄物に対する戦略の例と関連する研究が議論される。

References

[1] Nuclear Decommissioning Authority, Strategy Effective from April 2016.
[2] Nuclear Decommissioning Authority, Radioactive Wastes in the UK: A summary of the 2013 Inventory.
[3] Geological Disposal Generic Waste Package Specification, Radioactive Waste Management Limited.

（翻訳：CLADS　大井　貴夫）

2.3.3 フィンランドにおける廃棄物管理計画と実施状況

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1. 序論

フィンランドの原子力廃棄物管理計画は、低及び中レベル廃棄物と使用済み原子燃料の廃棄物管理に関して今日世界で最も先進的である。30年前に策定した計画と工程をわずかに変更はしたものの今日まで運用している。フィンランドの成功は、計画における、科学的研究との連携、意思決定、規制要求、事業者要求にわたっている。処分場の選定やその調査において積み重ねられた実績は、各国の関心を呼ぶと思われるが、直ちに他の国に応用できる方法、手法ではない。フィンランドの手法の特徴は、理解のために安全評価を用いて、サイトの調査や処分概念の開発を段階的に深化してきている。

2. 低及び中レベル廃棄物管理

フィンランドでは、現在、運転中の原子力発電所 Loviisa と Olkilinoto に 2 つの低及び中レベル廃棄物処分場の許可が下りている。それらは、地下約 100 m の結晶質岩の地層処分場である。将来のそれぞれの発電所の廃止措置廃棄物を処分するために処分場を拡張する、よく練られた計画がある。

3. 高レベルの使用済み原子燃料管理

フィンランドはおそらく世界で初めて使用済み原子燃料の深地層処分を許可した国である。処分場は KBS-3V 概念に基づくものとなり、使用済み燃料集合体をキャニスタに入れ、それぞれを垂直な穴にひとつずつ入れて処分する。使用済み燃料処分の体系的な計画は、Fig. 1 に示すように 30 年以上継続している。

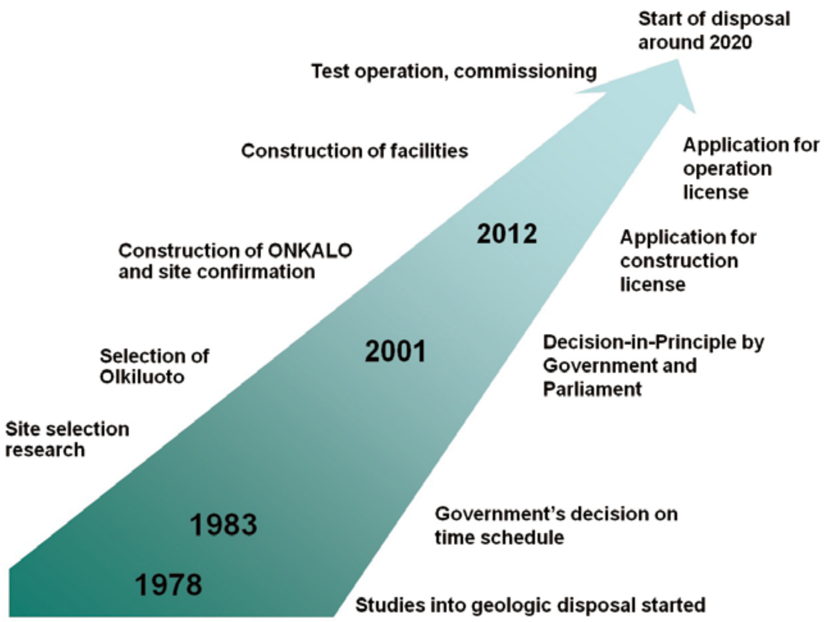

Fig. 1. フィンランドにおける使用済み原子燃料処分の工程 [1].

4. 原子力廃棄物管理のコスト

現在稼働している、また、これから建設される原子炉にて発生する、全ての原子力廃棄物管理の全費用

は、2012 年に 65 億ユーロと見積もられた。うち 35 億ユーロは使用済み燃料の費用である。この推算は、運転廃棄物、使用済み燃料と廃止措置廃棄物を含んでいる。既存（2015 年末の時点で）の原子力廃棄物の将来の管理に要する推定コストは、約 24.5 億ユーロである（割引なし）。原子力廃棄物管理の費用は、原子力廃棄物の発生者から徴収し、国の原子力廃棄物管理基金として管理される。これにより、原子力廃棄物の発生者が破産した場合においても、既存の原子力廃棄物を管理するために十分な費用を確保する。

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（翻訳：CLADS 駒 義和）

2.4.1 固体廃棄物からの放射性元素の回収と減容固定化

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2011 年 3 月に発生した福島第一原子力発電所事故により大量の放射性物質が環境中に放出され、放射性物質により広範囲に環境が汚染され、大量の固体廃棄物が発生している。福島第一原子力発電所のサイト内でも環境系の固廃棄物量だけても、伐採木が 8 万 m^3 以上、瓦礫と土壌が 15 万 m^3 以上と大量である。こうした放射性廃棄物はこれまで取り扱った経験がなく、合理的な最終処分が可能になるよう新たな処理・処分技術の開発が必要である。2015 年 3 月に JAEA によって実施された環境系廃棄物の放射線量の測定によれば、事故炉建屋近くの土壌や瓦礫では ^{137}Cs で数万～数百万 Bq/kg と高い放射能量が計測され、少量ではあるが ^{90}Sr の存在も確認されている。

こうした現状を踏まえて我々の研究グループでは環境系固体廃棄物からの放射性核種を回収し、減容固化する処理システム（英文要旨の Fig.1）を提案している。放射性 Cs（^{134}Cs,^{137}Cs）は土壌や伐採木の焼却灰などに安定に存在し、特に土壌では粘土鉱物に強く吸着されている。本研究では最初に汚染土壌の処理について検討を進めた。土壌から粘土鉱物を含む細粒分が分級法によって分離され、有機酸を含む亜臨界水で洗浄された。粘土鉱物の結晶構造は有機酸により分解され、強吸着された Cs が水相に回収された。Cs はプルシアンブルーに選択吸着され、プルシアンブルーは 300ºC で熱分解された。分解残渣中の Cs は水洗浄及び希硝酸洗浄により濃縮回収され、ホウ珪酸ガラスに固定化された。これらの実験結果から、50,000Bq/kg の Cs 汚染土壌の処理を対象として最終廃棄物（ガラス固化体）の体積を推定すると、ガラス固化体の体積は元の土壌体積の 2 万分の 1 程度と評価された。本技術の導入により大量に発生した汚染土壌中の Cs はガラス固化体として高減容固化することができ、処理された土壌廃棄物は再利用が可能になる。

2.4.2 ^{90}Sr の迅速分析法

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東京電力福島第一原子力発電所事故では，東日本の広範囲が放射性物質で汚染される深刻な問題となった。^{90}Sr 等のβ線核種は，放射性 Cs 等のγ線核種と比較して，簡易モニタリングが技術的に難しい。従来の分析法（ミルキング法）では，分析時間が2週間かかるために十分な調査には至っていない。迅速で精度の高い分析方法で，実験者の安全（放射線防護）を確保できる分析手法が望まれている。本研究では，高周波誘導結合プラズマ-質量分析装置（ICP-MS）を用いた新しい ^{90}Sr 分析法を開発した。具体的には，ICP-MS における ^{90}Sr 分析で最も困難な「同重体分離」と「検出感度」の問題を，装置内酸化（コリジョン）反応とカラム濃縮分離を組み合わせることで，これらを解決した。

種々の検討の結果，オンラインカラム濃縮と ICP-MS 装置内での金属酸化反応を兼備することで，^{90}Sr の同重体干渉の影響を全く受けない Sr90 特化型の ICP-MS 分析を実現できた。検出下限値は，質量濃度で 0.06 ppq (0.3 Bq/L)であった。分析に要する時間は，20 分以内で，繰り返し再現性は 2.9% (10 ppq ^{90}Sr(n = 10)) であった。また，福島県避難区域のある地点の土壌を測定した結果，公定法では 74 Bq/kg，本法では 89 Bq/kg と値がほぼ一致し，非常に良好な結果を得ることができたので，この分析手法について紹介する。

参考文献

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2.4.3 福島工業高等専門学校における活動

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1. 緒言

福島第一原子力発電所廃止措置は国家課題であり，中長期的な視点からの人材育成が求められている．福島高専は福島第一原子力発電にもっとも近い国立の工学系高等教育機関であることから，廃止措置に係る人材育成を積極的に行い，全国高専のハブとなる役割が求められている．平成 27 年度に文部科学省の「国家課題対応型研究開発事業　英知を結集した原子力科学技術・人材育成推進事業　廃止措置研究・人材育成等強化プログラム」の採択を受けて，「廃炉に関する基盤研究を通じた創造的人材育成プログラム　-高専間ネットワークを活用した福島からの学際的なチャレンジ-」を開始した．

2. 廃止措置人材育成高専等連携協議会

「廃炉に関する基盤研究を通じた創造的人材育成プログラム」の開始に伴い，福島高専から全国の高専や研究機関，大学に廃止措置に関する基盤研究と人材育成の連携を呼びかけた．その結果，26高専と協力体制を築くことができ，廃止措置人材育成高専等連携協議会を設立した．初代会長には中村福島高専校長が就任した．2016 年 4 月現在で，高専 34 校，大学 9 校，自治体関係 3 機関，民間企業 6 社が連携協議会に加盟している．

3. 廃炉創造学習プログラム

政府及び東京電力は，「東京電力福島第一原子力発電所1～4号機の廃止措置等に向けた中長期ロードマップ」に基づいて，各種の取組が進められてきている．その中長期ロードマップの中で，「中長期の視点での人材育成及び大学・研究機関との連携」に取り組むことが記載されており，本教育プログラムはそれに対応している．また，本教育プログラムは，基盤研究，教育・人材育成，施設利用促進の 3 つの柱で構成されている．

2015 年には 3 年生対象の「廃炉と社会」と 4 年生対象の「廃炉工学」を開講した．2016 年には廃止措置関連インターンシップと 2 年生対象の「放射線基礎」，3 年生対象の「廃炉ロボット概論」を開講し，2017 年には 1 年生対象の「原子力発電基礎」，5 年生対象の「原子力事故総論」を開講予定である．

インターンシップは廃止措置関連企業と自治体の 2 箇所で実施する複合型で行っている．

4. 廃炉創造ロボコン

ロボット製作を通じて学生に廃炉に関する興味を持たせると同時に学生の創造性教育に貢献し，課題解決能力のみならず課題発見能力の養うことを目的として実施する．平成 28 年 12 月 3 日に日本原子力研究開発機構楢葉遠隔技術開発センターを会場として開催する．競技フィールドには，モックアップ階段と標準テストフィールドの何れかを各チームが選択する．

単なるロボット製作だけで終わらせずに，廃炉についての理解や興味を高めるために，参加学生を対象として楢葉サマースクールを 9 月 2 日，3 日に開催し，福島第一原子力発電所の視察と廃炉に関する講義を行った．

2.5 ポスター発表

2.5.1 X 線吸収分光法による水素再結合反応中の貴金属微粒子触媒のその場観測

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要 旨
原子力発電施設のシビアアクシデント時や、放射性廃棄物の長期保管時に発生する水素を、貴金属微粒子触媒による水素再結合(水生成)反応により、安全な水に変換する研究が注目されている。本研究では、SPring-8にてエネルギー分散型X吸収微細構造解析(DXAFS)を利用し、貴金属微粒子の挙動のその場時分割観測を行い、一酸化炭素の被毒作用の機構解明を目指した。結果、水素再結合反応に重要な役割を果たしている表面酸化膜の形成が、一酸化炭素によって阻害されることが活性低下の原因のひとつだとわかった。さらに、酸素吸蔵能力を持つセリア系担体を用いることで、一酸化炭素による阻害作用を軽減できることがわかり、触媒設計に有益な指針を得ることができた。

2.5.2 水素再結合への自動車触媒の応用

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要 旨
福島第一原子力発電所のシビアアクシデント時に、炉心冷却のため多くの海水が注入された。この多量の水を含んだ核廃棄物を隔離保管する際、放射線により水が分解され水素と酸素を発生することが問題となる。容器内に水素が充満することを防ぐため、酸素と再結合(酸化)させる高活性な触媒が求められている。我々は、インテリジェント自動車触媒が、室温から水素を酸化する能力が高いことを見出した。当研究室では、水素酸化反応を阻害する一酸化炭素などの触媒毒や水蒸気の影響を調べ、これら共存ガスの影響を受けにくい触媒材料の改良を進めている。高活性な再結合触媒の実現を目指している。

2.5.3 放射性廃棄物長期保管容器内に発生する水素ガスの自然対流挙動に関する予備解析の結果

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要 旨

放射性廃棄物保管容器では、放射線による水分解によって発生する水素ガスの濃度が時間とともに増大するため、保管容器の安全性向上に対しては密閉容器内での水素ガスの挙動を正確に把握することが重要である。水素ガスは、容器内に蓄積される放射性物質によって加熱されるため、自然対流を形成することが考えられる。そこで、廃棄物長期保管容器内での放射性物質の崩壊熱に起因する水素ガスの自然対流挙動を定量的に把握することを目的として、廃棄物長期保管容器を縮小簡略モデルで模擬する小型実験装置の体系に対して予備解析を実施した。本研究の結果、廃棄物保管容器内の自然対流現象を数値的に予測できる見通しが得られた。

2.5.4 水素ガス濃度を低減させるPAR付き放射性廃棄物長期保管容器の性能評価のための基礎実験計画

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要 旨

原子力発電所の廃炉においては、燃料デブリ等の放射性廃棄物の長期に亘る管理が不可欠であり、その過程では水の放射線分解によって可燃性ガスである水素が発生する。したがって、廃棄物保管容器の安全性を確保するためには、発生する水素ガスの濃度を低減し、かつ爆発限界（4%）未満に維持させることが重要である。そこで、水素処理装置（PAR）を利用した可燃性ガス濃度低減機構を保管容器内に設置し、その有効性を実験的に調べることを立案した。本報では小型モデル実験装置による研究計画について述べる。

2.5.5 土中微生物の代謝による土壌稠密化の放射性核種拡散・移流の抑止 –その技術的原理と実際応用の展望-

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要 旨
土中や環境水中に遍在する微生物（主として枯草菌類など）の嫌気性呼吸により発生する二酸化炭素は，元来土壌中に豊富に含まれるカルシウムイオンと化合し、カルサイト（炭酸カルシウムの最安定相）を生成する。これはバイオセメンテーションとよばれる現象でコンクリート建造物の自己修復効果などを誘引するしくみとして近年低環境負荷型土木技術の一原理として着目されている。人為的にカルシウムイオンと微生物の代謝のための栄養分を地上から適切な速度で供給して土中バイオセメンテーションを誘引すれば、遮蔽構造物の土中埋設などの大規模・高価な工事を行うことなく廃炉エリアでの流出放射性核種の制御不能な土中物質移動を抑止できる。

2.5.6 福島事故後の原子力に対する日仏の若い世代の意識調査

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要 旨
今後の原子力災害の収束には、およそ40年の歳月を要するといわれている。そのため、長期間の作業に携わる人材の育成は重要な課題である。今後の人材育成のあり方について検討するため、日本の高専生とフランスの学生を対象に，100名規模の原子力に対する意識調査を行った。

2.5.7 水熱処理法用いた風化黒雲母の層間内に飽和されたセシウムの脱離

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要 旨

福島第一原子力発電所事故によって大量に飛散した放射性セシウムが周辺の土壌を汚染した。現在、福島で除染作業が進められているが、除染で取り除いた廃棄土壌の処理が大きな課題になっている。土壌からの高効率セシウム脱離が実現できれば、セシウム除去後の土壌はリサイクル可能となり、廃棄物の大幅減容化が可能になる。そこで本研究ではアルカリ・アルカリ土類イオンを含む亜臨界水による風化黒雲母粘土からのセシウム脱離試験を行い、水熱処理法（亜臨界水洗浄）によるセシウムの高速脱離プロセスの成立性について検討した。結果として、セシウムを飽和させた風化黒雲母に対して、亜臨界水洗浄による風化黒雲母からのセシウムの除去率は、アルカリ金属イオン(Na^+, K^+)を用いた場合には250℃加熱で約20%だったのに対し、アルカリ土類金属イオン(Mg^{2+}, Ca^{2+})を用いると、同じ温度圧力条件で脱離率は100%であった。各加熱温度での風化黒雲母粘土のX線回折により結晶相の同定を行った結果、処理温度が高いほど黒雲母層間内でのアルカリ土類金属イオンの拡散速度が大きくなり、セシウムとの高速イオン交換が可能なることが分かった。

国際単位系（SI）

表１．SI 基本単位

基本量	SI 基本単位	
	名称	記号
長さ	メートル	m
質量	キログラム	kg
時間	秒	s
電流	アンペア	A
熱力学温度	ケルビン	K
物質量	モル	mol
光度	カンデラ	cd

表２．基本単位を用いて表されるSI組立単位の例

組立量	SI 組立単位	
	名称	記号
面積	平方メートル	m^2
体積	立方メートル	m^3
速さ，速度	メートル毎秒	m/s
加速度	メートル毎秒毎秒	m/s^2
波数	毎メートル	m^{-1}
密度，質量密度	キログラム毎立方メートル	kg/m^3
面積密度	キログラム毎平方メートル	kg/m^2
比体積	立方メートル毎キログラム	m^3/kg
電流密度	アンペア毎平方メートル	A/m^2
磁界の強さ	アンペア毎メートル	A/m
量濃度(a)，濃度	モル毎立方メートル	mol/m^3
質量濃度	キログラム毎立方メートル	kg/m^3
輝度	カンデラ毎平方メートル	cd/m^2
屈折率(b)	（数字の） 1	1
比透磁率(b)	（数字の） 1	1

(a) 量濃度（amount concentration）は臨床化学の分野では物質濃度（substance concentration）ともよばれる。
(b) これらは無次元量あるいは次元１をもつ量であるが、そのことを表す単位記号である数字の１は通常は表記しない。

表３．固有の名称と記号で表されるSI組立単位

組立量	SI 組立単位			
	名称	記号	他のSI単位による表し方	SI基本単位による表し方
平面角	ラジアン(b)	rad	1 (b)	m/m
立体角	ステラジアン(b)	sr(c)	1 (b)	m^2/m^2
周波数	ヘルツ(d)	Hz		s^{-1}
力	ニュートン	N		m kg s^{-2}
圧力，応力	パスカル	Pa	N/m^2	m^{-1} kg s^{-2}
エネルギー，仕事，熱量	ジュール	J	N m	m^2 kg s^{-2}
仕事率，工率，放射束	ワット	W	J/s	m^2 kg s^{-3}
電荷，電気量	クーロン	C		s A
電位差（電圧），起電力	ボルト	V	W/A	m^2 kg s^{-3} A^{-1}
静電容量	ファラド	F	C/V	m^{-2} kg^{-1} s^4 A^2
電気抵抗	オーム	Ω	V/A	m^2 kg s^{-3} A^{-2}
コンダクタンス	ジーメンス	S	A/V	m^{-2} kg^{-1} s^3 A^2
磁束	ウエーバ	Wb	Vs	m^2 kg s^{-2} A^{-1}
磁束密度	テスラ	T	Wb/m^2	kg s^{-2} A^{-1}
インダクタンス	ヘンリー	H	Wb/A	m^2 kg s^{-2} A^{-2}
セルシウス温度	セルシウス度(e)	℃		K
光束	ルーメン	lm	cd sr(c)	cd
照度	ルクス	lx	lm/m^2	m^{-2} cd
放射性核種の放射能(f)	ベクレル(d)	Bq		s^{-1}
吸収線量, 比エネルギー分与, カーマ	グレイ	Gy	J/kg	m^2 s^{-2}
線量当量, 周辺線量当量, 方向性線量当量, 個人線量当量	シーベルト(g)	Sv	J/kg	m^2 s^{-2}
酸素活性	カタール	kat		s^{-1} mol

(a)SI接頭語は固有の名称と記号を持つ組立単位と組み合わせても使用できる。しかし接頭語を付した単位はもはやコヒーレントではない。
(b)ラジアンとステラジアンは数字の１に対する単位の特別な名称で、量についての情報をつたえるために使われる。実際には、使用する時には記号rad及びsrが用いられるが、習慣として組立単位としての記号である数字の１は明示されない。
(c)測光学ではステラジアンという名称と記号srを単位の表し方の中に、そのまま維持している。
(d)ヘルツは周期現象についてのみ、ベクレルは放射性核種の統計的過程についてのみ使用される。
(e)セルシウス度はケルビンの特別な名称で、セルシウス温度を表すために使用される。セルシウス度とケルビンの単位の大きさは同一である。したがって、温度差や温度間隔を表す数値はどちらの単位で表しても同じである。
(f)放射性核種の放射能（activity referred to a radionuclide）は、しばしば誤った用語で"radioactivity"と記される。
(g)単位シーベルト（PV,2002,70,205）についてはCIPM勧告2（CI-2002）を参照。

表４．単位の中に固有の名称と記号を含むSI組立単位の例

組立量	SI 組立単位		
	名称	記号	SI 基本単位による表し方
粘度	パスカル秒	Pa s	m^{-1} kg s^{-1}
力のモーメント	ニュートンメートル	N m	m^2 kg s^{-2}
表面張力	ニュートン毎メートル	N/m	kg s^{-2}
角速度	ラジアン毎秒	rad/s	m m^{-1} s^{-1}=s^{-1}
角加速度	ラジアン毎秒毎秒	rad/s^2	m m^{-1} s^{-2}=s^{-2}
熱流密度，放射照度	ワット毎平方メートル	W/m^2	kg s^{-3}
熱容量，エントロピー	ジュール毎ケルビン	J/K	m^2 kg s^{-2} K^{-1}
比熱容量，比エントロピー	ジュール毎キログラム毎ケルビン	J/(kg K)	m^2 s^{-2} K^{-1}
比エネルギー	ジュール毎キログラム	J/kg	m^2 s^{-2}
熱伝導率	ワット毎メートル毎ケルビン	W/(m K)	m kg s^{-3} K^{-1}
体積エネルギー	ジュール毎立方メートル	J/m^3	m^{-1} kg s^{-2}
電界の強さ	ボルト毎メートル	V/m	m kg s^{-3} A^{-1}
電荷密度	クーロン毎立方メートル	C/m^3	m^{-3} s A
表面電荷	クーロン毎平方メートル	C/m^2	m^{-2} s A
電束密度，電気変位	クーロン毎平方メートル	C/m^2	m^{-2} s A
誘電率	ファラド毎メートル	F/m	m^{-3} kg^{-1} s^4 A^2
透磁率	ヘンリー毎メートル	H/m	m kg s^{-2} A^{-2}
モルエネルギー	ジュール毎モル	J/mol	m^2 kg s^{-2} mol^{-1}
モルエントロピー, モル熱容量	ジュール毎モル毎ケルビン	J/(mol K)	m^2 kg s^{-2} K^{-1} mol^{-1}
照射線量（Ｘ線及びγ線）	クーロン毎キログラム	C/kg	kg^{-1} s A
吸収線量率	グレイ毎秒	Gy/s	m^2 s^{-3}
放射強度	ワット毎ステラジアン	W/sr	m^4 m^{-2} kg s^{-3}=m^2 kg s^{-3}
放射輝度	ワット毎平方メートル毎ステラジアン	W/(m^2 sr)	m^2 m^{-2} kg s^{-3}=kg s^{-3}
酵素活性濃度	カタール毎立方メートル	kat/m^3	m^{-3} s^{-1} mol

表５．SI 接頭語

乗数	名称	記号	乗数	名称	記号
10^{24}	ヨタ	Y	10^{-1}	デシ	d
10^{21}	ゼタ	Z	10^{-2}	センチ	c
10^{18}	エクサ	E	10^{-3}	ミリ	m
10^{15}	ペタ	P	10^{-6}	マイクロ	µ
10^{12}	テラ	T	10^{-9}	ナノ	n
10^{9}	ギガ	G	10^{-12}	ピコ	p
10^{6}	メガ	M	10^{-15}	フェムト	f
10^{3}	キロ	k	10^{-18}	アト	a
10^{2}	ヘクト	h	10^{-21}	ゼプト	z
10^{1}	デカ	da	10^{-24}	ヨクト	y

表６．SIに属さないが、SIと併用される単位

名称	記号	SI 単位による値
分	min	1 min=60 s
時	h	1 h =60 min=3600 s
日	d	1 d=24 h=86 400 s
度	°	1°=(π/180) rad
分	'	1'=(1/60)°=(π/10 800) rad
秒	"	1"=(1/60)'=(π/648 000) rad
ヘクタール	ha	1 ha=1 hm^2=10^{4}m^2
リットル	L，l	1 L=1 l=1 dm^3=10^{3}cm^3=10^{-3}m^3
トン	t	1 t=10^3 kg

表７．SIに属さないが、SIと併用される単位で、SI単位で表される数値が実験的に得られるもの

名称	記号	SI 単位で表される数値
電子ボルト	eV	1 eV=1.602 176 53(14)×10^{-19}J
ダルトン	Da	1 Da=1.660 538 86(28)×10^{-27}kg
統一原子質量単位	u	1 u=1 Da
天文単位	ua	1 ua=1.495 978 706 91(6)×10^{11}m

表８．SIに属さないが、SIと併用されるその他の単位

名称	記号	SI 単位で表される数値
バール	bar	1 bar=0.1MPa=100 kPa=10^{5}Pa
水銀柱ミリメートル	mmHg	1 mmHg≈133.322Pa
オングストローム	Å	1 Å=0.1nm=100pm=10^{-10}m
海里	M	1 M=1852m
バーン	b	1 b=100fm^2=(10^{-12}cm)^2=10^{-28}m^2
ノット	kn	1 kn=(1852/3600)m/s
ネーパ	Np	SI単位との数値的な関係は、対数量の定義に依存。
ベル	B	
デシベル	dB	

表９．固有の名称をもつCGS組立単位

名称	記号	SI 単位で表される数値
エルグ	erg	1 erg=10^{-7} J
ダイン	dyn	1 dyn=10^{-5}N
ポアズ	P	1 P=1 dyn s cm^{-2}=0.1Pa s
ストークス	St	1 St =1cm^2 s^{-1}=10^{-4}m^2 s^{-1}
スチルブ	sb	1 sb =1cd cm^{-2}=10^{4}cd m^{-2}
フォト	ph	1 ph=1cd sr cm^{-2} =10^{4}lx
ガル	Gal	1 Gal =1cm s^{-2}=10^{-2}ms^{-2}
マクスウエル	Mx	1 Mx = 1G cm^2=10^{-8}Wb
ガウス	G	1 G =1Mx cm^{-2} =10^{-4}T
エルステッド(a)	Oe	1 Oe ≙ (10^3/4π)A m^{-1}

(a) ３元系のCGS単位系とSIでは直接比較できないため、等号「 ≙ 」は対応関係を示すものである。

表10．SIに属さないその他の単位の例

名称	記号	SI 単位で表される数値
キュリー	Ci	1 Ci=3.7×10^{10}Bq
レントゲン	R	1 R = 2.58×10^{-4}C/kg
ラド	rad	1 rad=1cGy=10^{-2}Gy
レム	rem	1 rem=1 cSv=10^{-2}Sv
ガンマ	γ	1 γ=1 nT=10^{-9}T
フェルミ		1 フェルミ=1 fm=10^{-15}m
メートル系カラット		1 メートル系カラット= 0.2 g = 2×10^{-4}kg
トル	Torr	1 Torr = (101 325/760) Pa
標準大気圧	atm	1 atm = 101 325 Pa
カロリー	cal	1 cal=4.1858J（「15℃」カロリー），4.1868J（「IT」カロリー），4.184J（「熱化学」カロリー）
ミクロン	µ	1 µ =1µm=10^{-6}m

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