Kyoto University
Research Reactor Institute

International Symposium on

Nuclear Back-end Issues and the Role of Nuclear Transmutation Technology after the accident of TEPCO's Fukushima Daiichi Nuclear Power Stations

28 November 2013
The Shiran Hall, Kyoto University

KURRI Research Program for Scientific Basis of Nuclear Safety

Programme
&
Abstracts

Preface

The nuclear accident at the TEPCO's Fukushima Daiichi NPP in March 2011, which followed an earthquake and a tsunami, brought serious impacts on society. This accident inevitably underlined the necessity of establishing new and comprehensive scientific research for promoting nuclear safety. Facing this situation, the Kyoto University Research Reactor Institute (KURRI) developed a new research program called the "KUR Research Program for Scientific Basis of Nuclear Safety" from 2012. In this program, it is planned to hold an annual series of international symposia along with identified research activities for promoting nuclear safety.

The second annual symposium in this series deals with nuclear back-end issues and the role of nuclear transmutation technology after the accident of the TEPCO's Fukushima Daiichi NPP, following the first one for the radiological effect of the accident on the public. As well as on the impacts of the accident, the accident has called us to focus our attention on a large amount of spent nuclear fuels stored in NPPs. In fact, public anxiety regarding the treatment and disposal of high-level radioactive wastes that require long-term control is now growing, while the government policy on the back-end of nuclear fuel cycle is unpredictable in the aftermath of the accident. The issues may not be simply technical, but are critically important not only for settling the accident but also for pursuing nuclear energy production in the world. It is thus needed to address the current status of the back-end issues and to discuss the future direction of research and development on radioactive waste treatment and disposal.

Approximately 9 invited lectures and 54 scientific papers will be presented in the symposium. Almost all the presentations have been written as original scientific papers, peer reviewed by specialists of the relevant research fields, and included in the proceedings booklet. On behalf of all the participants of the symposium, I sincerely hope that this symposium and its proceedings will contribute to the reduction of public anxiety and will promote further progress in the research on nuclear safety.

Hirotake Moriyama
Director, Kyoto University Research Reactor Institute
Chair of the organizing committee

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Kyoto University Research Reactor Institute (KURRI)

Support organization

Atomic Energy Society of Japan

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Program

9:00-10:00 Registration, Poster Reception

10:00-10:15 **Opening Remarks** Prof. Hirotake Moriyama, Director of KURRI

10:15-10:45 **Decommissioning Activity at Fukushima Daiichi NPS**
Speaker : Mr. Hideki Masui (TEPCO)
Chair: Prof. Ken Nakajima (KURRI)

10:45-11:15 **Issue of the High-Level Radioactive Waste**
Speaker : Prof. Kenji Yamaji (RITE)
Chair: Prof. Ken Nakajima (KURRI)

11:15-11:45 **Roles of Nuclear Fuel Cycle Technologies on Geological Disposal**
Speaker : Prof. Joonhong Ahn (University of California, Berkeley)
Chair: Prof. Yoshiaki Kiyanagi (Nagoya University)

11:45-12:15 **Expectation to Nuclear Transmutation**
Speaker : Dr. Akito Arima (Musashi Academy)
Chair: Prof. Yoshiaki Kiyanagi (Nagoya University)

12:15-14:00 Lunch and poster session

14:00-14:30 **OECD / NEA Activities Related to the Nuclear Fuel Cycle**
Speaker : Dr. Thierry Dujardin (OECD)
Chair: Dr. Kazufumi Tsujimoto (JAEA)

14:30-15:00 **Contribution of the European Commission to a European Strategy for HLW Management through Partitionning & Transmutation**
- Presentation of MYRRHA and its in the European P&T strategy -
Speaker : Prof. Hamid Aït Abderrahim (SCK・CEN)
Chair: Prof. Hironobu Unesaki (KURRI)

15:00-15:30 **ADS Study in JAEA** Speaker : Dr. Hiroyuki Oigawa (JAEA)
Chair: Prof. Hironobu Unesaki (KURRI)

15:30-16:00 Coffee break and poster session

16:00-16:30 **Accelerator-Driven System (ADS) Study in Kyoto University Research Reactor Institute (KURRI)** Speaker : Prof. Cheol Ho Pyeon (KURRI)
Chair: Prof. Jun-ichi Katakura (Nagaoka University of Technology)

16:30-17:00 **Nuclear Data Study for Nuclear Transmutation**
Speaker : Prof. Masayuki Igashira (Tokyo Institute of Technology)
Chair: Prof. Jun-ichi Katakura (Nagaoka University of Technology)

17:00-18:00 **Panel Discussion** Chair: Prof. Ken Nakajima

18:00-18:15 **Closing Remarks** Prof. Hajimu Yamana

18:15-19:40 Reception

Poster sessions

Topics 1-a Nuclear transmutation - Reactor system –

1 Lu ZHANG, Kun FENG, Yucui GAO, Deliang FAN, Yongwei FANG
Physical Analysis of LBE Spallation Target for CIADS

2 Kun FENG, Lu ZHANG, Deliang FAN, Zhuang WU, Yongwei YANG
Analysis of Thermal-hydraulics for the LBE Spallation Target of ADS

3 Takanori KITADA, Vu Thanh MAI, Noboru DOBUCHI
Study on Neutron Spectrum of Pulsed Neutron Reactor

4 Toshihiro YAMAMOTO
Theory of power spectral density and Feynman-*a* method in Accelerator Driven System and their higher order mode effects

5 W.F.G. VAN ROOIJEN
Neutron requirement for transmutation of MA isotopes

6 Tetsushi HINO, Masaya OHTSUKA, Renzo TAKEDA, Junichi MIWA, Kumiaki MORIYA
Application of the Resource-renewable Boiling Water Reactor for TRU Management and Long-Term Energy Supply

7 Takanori SUGAWARA, Kenji NISHIHARA, Kazufumi TSUJIMOTO, Toshinobu SASA
Hironari OBAYASHI, Yuji KURATA, Hiroyuki OIGAWA
Investigations of Lead-Bismuth cooled Accelerator-Driven System

8 Toshinobu SASA, Hayanori TAKEI, Shigeru SAITO, Hironari OBAYASHI, Kenji NISHIHARA, Takanori SUGAWARA, Hiroki IWAMOTO, Kazufumi TSUJIMOTO, Hiroyuki OIGAWA
J-PARC Transmutation Experimental Facility

9 Kyoko ISHII, Mitsuaki YAMAOKA, Yasuyuki MORIKI, Takashi OOMORI, Yasushi TSUBOI, Kazuo ARIE, Masatoshi KAWASHIMA
Development of Uranium-Free TRU Metallic Fuel Fast Reactor Core with Enhanced Doppler Reactivity Feedback

10 Akihiro SASAHARA, Tetsuo MATSUMURA
Chemical Isotopic Analysis and Computational Evaluation of Nuclide Composition in PWR-UO_2 and MOX Spent Fuels Using JENDL, ENDF/B, JEF, and JEFF

11 Kenji KONASHI, Tsugio YOKOYAMA
Enhancement of Transmutation of Minor Actinides by Hydride Target

12 Hiroki SONO , Kotaro TONOIKE , Kazuhiko IZAWA , Takashi KIDA , Fuyumi KOBAYASHI , Masato SUMIYA , Hiroyuki FUKAYA , Miki UMEDA , Kazuhiko OGAWA, Yoshinori MIYOSHI
Project to Modify the STACY Critical Facility for Experimental Study on Fuel Debris Criticality Control

13 Naoto AIZAWA, Tomohiko Iwasaki, Tatsuro TAKANI, Takashi TANI, Shunsuke KANEMOCHI, Motomu SUZUKI, Fumito KUBO, Yasuaki WATANABE
ADS study in Tohoku University

14 Toshikazu TAKEDA, Koji FUJIMURA
Method Development for Calculating Minor Actinide Transmutation in a Fast Reactor

15 Daisuke ITO, Kazuki HASE, and Yasushi SAITO
Heat transfer study for ADS solid target
-surface wettability and its effect on a boiling heat transfer-

16 Gen ARIYOSHI, Daisuke ITO, and Yasushi SAITO
Experimental study of flow structure and turbulent characteristics in lead-bismuth two-phase flow

17 Kyoko Mukaida, Naoto Yasumatsu, Masanori Heta, Akira Ohtaki, Hiroki Shiotani, Kiyoshi Ono, Masaru Hirata
Scenario study for the transition from thermal reactors to fast reactors in the world

18 Tetsuya MOURI, Taira HAZAMA, Hiroshi NISHI
Current Status and Future Prospect of MA Irradiation Tests in Monju

19 Shigeo OHKI, Tsutomu OKUBO, Tomoyuki ABE
Study on Actinide Burning by Fast Reactor

20 Seung-Woo HONG, Sang-In BAK, Jong-Seo CHAI, J. P. Curbelo, Yacine KADI, Claudio TENREIRO
Neutronic analysis and transmutation performance of Th-based Molten Salt Fuels

Topics 1-b Nuclear transmutation - Fuel cycle –

1 Kenji NISHIHARA, Kazufumi TSUJIMOTO, Hiroyuki OIGAWA
Transmutation Scenarios after Closing Nuclear Power Plants

2 Kotaro TONOIKE, Hiroki SONO, Miki UMEDA, Yuichi YAMANE, Teruhiko KUGO, Kenya SUYAMA
Principle Options of Fuel Debris Criticality Control in Fukushima Daiichi Reactors

3 Seung-Woo HONG, Sang-In BAK, Masoud BEHZAD, Jong-Seo CHAI, Yacine KADI, Vijay K. MANCHANDA, Tae-Sun PARK, Claudio TENREIRO, Chirag K. VYAS
ADS Research Activities in Sungkyunkwan University

Topics 1-c Nuclear transmutation - Material science –

1 Shizuka TAKAI, Kouichi HAGINO
Nuclear transmutation of long-lived nuclides with laser Compton scattering : Quantitative analysis by theoretical approach

2 Kento YAMAMOTO, Keisuke OKUMURA, Kensuke KOJIMA, Tsutomu OKAMOTO
Sensitivity analyses of initial compositions and cross sections for activation products of in-core structure materials

3 Hiroaki MUTA, Toshiaki KAWANO, Yuji OHISHI, Ken KUROSAKI, Shinsuke YAMANAKA
Thermophysical properties of thorium oxide

4 Toshiaki KAWANO, Hiroaki MUTA, Yuji OHISHI, Ken KUROSAKI, Shinsuke YAMANAKA
Characterization and thermophysical properties of $RE_2Zr_2O_7$ and $Nd_2Ce_2O_7$ precipitate in ThO_2-based fuel

5 Minoru TAKAHASHI, Yun GAO, Marion GUIHOT, Asril PRAMUTADI
Overview of Studies on Lead-bismuth Technology in Tokyo Institute of Technology

6 Hironari OBAYASHI, Hayanori TAKEI, Hiroki IWAMOTO, Toshinobu SASA
Evaluation of structural integrity of beam window for TEF target by thermal/fluid-structure interaction analysis

7 K. HIROSE, T. OHTSUKI, Y. SHIBASAKI, N. IWASA, J. HORI, S. SEKIMOTO, K. TAKAMIYA, H. YASHIMA, K. NISHIO, and Y. KIYANAGI
Fission cross-section measurement of minor actinides using a lead slowing-down neutron spectrometer KULS at KUR

8 Tsuyoshi NISHI, Yasuo ARAI, Masahide TAKANO, Hirokazu HAYASHI, Masaki KURATA
Development of TRU nitride database for designing ADS fuel

9 M. KOIZUMI, F. KITATANI, H. TSUCHIYA, H. HARADA, J. TAKAMINE, M. KURETA, H. IIMURA, M. SEYA, B. BECKER, S. KOPECKY, W. MONDELAERS, P. SCHILLEBEECKX
Recent progress in research and development on the neutron resonance densitometry (NRD) for quantification of nuclear materials in particle-like debris

10 Marion GUIHOT, Minoru TAKAHASHI
Status of Studies and Future Subjects of Material Corrosion for Lead-bismuth Cooling System

11 Shigeru SAITO, Kenji KIKUCHI, Dai HAMAGUCHI, Shinya ENDO, Kouji USAMI, Naotoshi SAKURABA, Hiromitsu MIYAI, Katsutoshi ONO, Hiroki MATSUI, Masayoshi KAWAI, Yong DAI
Mechanical properties of beam window materials for ADS irradiated in a spallation environment

12 Jun-ichi HORI, Tadafumi SANO, Yoshiyuki TAKAHASHI, Hironobu UNESAKI, Ken NAKAJIMA
Development of non-destructive assay to fuel debris of Fukushima Daiichi NPP (1) Experimental validation for the application of a self-indication method

13 Tadafumi SANO, Jun-ichi HORI, Yoshiyuki TAKAHASHI, Hironobu UNESAKI, Ken NAKAJIMA
Development of non-destructive assay to fuel debris of Fukushima Daiichi NPP (2) Numerical validation for the application of a self-indication method

14 Shoji NAKAMURA
Precise Measurements of Neutron Capture Cross-Sections for LLFPs and MAs

15 Kenya SUYAMA, Gunzo UCHIYAMA, Hiroyuki FUKAYA, Miki UMEDA, Toru YAMAMOTO, Motomu SUZUKI
Development of the Method to Assay Hardly Measurable Elements in Spent Nuclear Fuel and Application to BWR 9×9 fuel

16 M. Yamawaki, T. Terai, T. Koyama, Y.Arita1, Y. Sekiguchi, K. Uozumi and M.Kinoshita
Evaluation of volatile FP elements behavior for severe accident analysis of molten salt reactor

Topics 2-a Back-end cycle issues - Radioactive waste treatment/disposal -

1 Nobuyoshi ISHII, Shinichi OGIYAMA, Shinji SAKURAI, Keiko TAGAMI, Shigeo UCHIDA
Environmental parameters for assessing the behavior of radiocarbon in the paddy soil-to-rice plant system

2 Jian ZHENG, Tagami KEIKO, Wenting BU, and Shigeo UCHIDA
Development of ICP-MS based analytical method for the determination of radioactive Cs isotopes

3 Asako SHIMADA, Mayumi Ozawa, Yutaka KAMEO, Takuyo YASUMATSU, Koji NEBASHI, Takuya NIIYAMA, Shuhei SEKI, Masatoshi KAJIO, Kuniaki TAKAHASHI
Development of rapid analytical method of ^{129}I in the contaminated water and woods arising at Fukushima Daiichi Nuclear Power Station

4 Shigeo UCHIDA, Keiko TAGAMI
Determination of low-level technetium-99 in the environmental samples by inductively coupled plasma mass spectrometry

5 Guosheng YANG, Jian ZHENG, Keiko TAGAMI, Shigeo UCHIDA
A simple and rapid method for separation and preconcentration of Ra in water samples

6 Hiromi TANABE, Kuniyoshi HOSHINO
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7 Takayuki SASAKI, Akira KIRISHIMA, Yuu TAKENO, Kohei FUKUDA, Nobuaki SATO
A simulation of Sr leaching in aqueous solution mixed with seawater and preparation of U/Zr oxides at high temperatures

8 Yoko FUJIKAWA, Hiroaki OZAKI, Hiroshi TSUNO, Pengfei WEI, Aiichiro FUJINAGA, Ryouhei TAKANAMI, Shogo TANIGUCHI, Shojiro KIMURA
Volume reduction of municipal solid wastes contaminated with radioactive cesium by ferrocyanide coprecipitation technique

9 Takao TSUBOYA
A Prescription toward to the Long-term Isolation of High Level Radioactive Waste in Japan

10 Shinji YOSHIKAWA, Kouichi USHIRODA, Michiaki OHKUBO, Yutaka HISHIKI, Kiyoshi HOMMA
MEXT-Sponsored Research Activities for Environmental Load Reduction

11 Masahiko SAITO, Satoru SUZUKI, Isamu SETO
NUMO's activities after the Great East Japan earthquake

12 Akira KITAMURA, Yukio TACHI
Migration Parameters and their Evaluation and Estimation Methodologies of Safety-relevant Radionuclides for Performance Assessment of Japanese Geological Disposal of HLW and TRU Waste

13 Takamitsu ISHIDERA, Seiichi KUROSAWA, Masanori HAYASHI, Yasuyuki SUZUKI
Evaluation of Distribution Coefficients for Radionuclides Sorption on Bentonite Colloid

14 Yuya Takahashi, Hitoshi Nakamura, Akira Yamada, Koji Mizuguchi, Reiko Fujita
The treatment process of simulated debris from a severe accident using molten salt system

Topics 2-b Back-end cycle issues - Public acceptance -

1 Akemi YOSHIDA
Considering Geological Disposal Program of High-level Radioactive Waste through Classroom Debate

Abstracts

Decommissioning Activity at Fukushima Daiichi NPS

Hideki Masui

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The presentation will provide the recovery works from the accident which occurred at Fukushima Daiichi Nuclear Power Station on March 11, 2011.
First, the focus is given to the countermeasures conducted until the cold shutdown state was accomplished in December 2011, which was the condition where the temperature within the reactor was below 100 degrees Celsius and the radiation level at the station premise was significantly suppressed. For instance, alternative circulating cooling systems, contaminated water processing facilities and primary containment vessels (PCVs) gas control system were established to stabilize the plants.
Then, the roadmap toward the decommissioning for Units 1 to 4 will be introduced. The roadmap divides the term of decommissioning into the following three phases.

1. The beginning of fuel removal from the SFPs
2. The beginning of fuel debris removal
3. The end of decommission activities

To achieve this roadmap, a variety of activities is ongoing. The examples include:

- The investigations on internal conditions of PCVs are imperative before fuel debris removal from damaged reactors. The visual inspection, radiation dosimetry and measurement of accumulated water level inside PCVs for Units 1 and 2 are being conducted by inserting an industrial endoscope.
- It was planned to install building covers for Units 1, 3 and 4 where the upper parts of the reactor buildings were blown during the accident. In preparation for this, dismantling of rubbles from top floors of reactor buildings is under way in Unit 3, and has been completed in Unit 4.

Above all, the inventory control (contaminated water management) is the most pressing issue. Since the ground water volume flowing into reactor buildings is significant, the volume of contaminated water increases steadily. It causes a big challenge to storage. To cope with it, the measures are being considered and tested, such as decontaminating the incoming water by using multi-nuclide removal equipment, or constructing a groundwater bypass from the upstream to the sea so that incoming flow can be reduced. Special attention will be given to the latest situation and countermeasures of water leak incidents.

In addition to these activities, R&D works toward the decommissioning is now being developed. This presentation will be finally summarized with the issues we are facing and the countermeasures we are considering for future recovery activities.

Keywords
Fukushima, decommissioning, contaminated water

Issue of the High-Level Radioactive Waste

Kenji Yamaji

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In relation to disposal of high-level radioactive waste (HLW), related government organizations and the Nuclear Waste Management Organization of Japan (NUMO) have been making efforts aimed at beginning a literature survey, a first step of HLW disposal according to fundamental policies and final disposal plan based on the "Designated Radioactive Waste Final Disposal Act". However, a difficult situation continues in which responses from municipalities, which are necessary for beginning literature survey, are not being made.

In September 2010 the Science Council of Japan (SCJ) received a deliberation request from the Chairman of the Japan Atomic Energy Commission, and SCJ formed a Review Committee for Disposal of High-Level Radioactive Waste. The Review Committee made a Reply on Disposal of High-Level Radioactive Waste in September 2012. In this paper, an outline of the current HLW disposal policy and the contents of the Reply are introduced.

Until now, the Final Disposal Act formulated in June 2000 has been the policy framework for Japan's HLW problems, and NUMO was established in October 2000 as an organization to implement those policies.

The following is an outline of policies stipulated based on the Final Disposal Act.

- Conduct final disposal by burying it, after confirming safety, in a geological layer at least 300 meters underground.
- The Ministry of Economy, Trade and Industry (METI) is in charge of supervising and regulating disposal projects, and the Minister of METI will stipulate a fundamental policy on final disposal based on the law and will, every five years, stipulate a 10-year plan for final disposal.
- Selection of a place for final disposal of high-level radioactive waste will be conducted through the following 3-stage procedures: (1) selection of a "preliminary investigation area," (2) selection of a "detailed investigation area," and (3) selection of a "site for construction of final disposal facilities."
- NUMO is established as the main party for conducting geological disposal.
- When selecting preliminary investigation areas, NUMO will accept offers from all cities, towns, and villages in Japan.
- A party that constructs a nuclear reactor for generating power must contribute the expenses required for final disposal of high-level radioactive waste. The Radioactive Waste Management Funding and Research Center will be in charge of managing those funds as a reserve fund, and will provide the funds to NUMO after receiving approval from the Minister of METI.
- If necessary, the law will be reconsidered when 10 years have passed since it was enacted.

On the other hand, the Review Committee of SCJ pointed out the following six proposals in order to search for a path toward consensus formation: 1) Fundamental reconsideration of policies related to disposal of high-level radioactive waste; 2) Awareness of the limits of scientific and technical abilities and securing scientific autonomy; 3) Rebuilding a policy framework centered on temporal safe storage and management of the total amount of HLW; 4) Necessity of persuasive policy decision procedures for fairness of burdens; 5) Necessity of multiple-stage consensus formation by establishing opportunities for debate; and 6) Awareness that long-term persistent undertakings are necessary for problem resolution.

Keywords
High-level radioactive waste, temporal safe storage, management of total amount

Roles of Nuclear Fuel Cycle Technologies on Geological Disposal

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Spent fuel management has been a challenge in any nuclear country. Even for countries with advanced technological capability and diverse geological and societal conditions, finding an actual site for a geological repository has proven very difficult politically and societally. Almost all countries that tried to advance from their feasibility study stage to siting stage had a major setback, such as Canada's Seaborne Commission in 1998, President Obama's decision in 2009 for the Yucca Mountain Project in the United States, and most recently the recommendation made by the Science Council's committee to the Atomic Energy Commission of Japan.

By learning from numerous failures and a few successful cases, such as Finland's Olukiluoto, Sweden's Forsmark, and US WIPP for TRU wastes, the concept of geological disposal has evolved significantly. In 1980s and earlier, the concept of geological disposal was just about a mined geological repository. Through discussions in the international community during the last decade, the contemporary geological disposal concept has expanded significantly. It now includes wide variety of options, ranging from treatment and interim storage prior to final geological disposal, resulting in great flexibility to accommodate people's values, which also range widely and evolve with time. Variety and flexibility are crucial to implement reversibility in public-participatory, adaptive staged approach for decision-making process.

To achieve a breakthrough for deadlocked situation in spent nuclear fuel management, historically various concepts of partitioning and transmutation (P&T) have been proposed and developed. While the effects of P&T on reduction of toxicity of long-lived minor actinides (MA) have been well recognized and studied, it has been pointed out mainly by the repository community that such reduction of MA would not directly affect the safety of a geological repository. This conflict of views on effects of P&T has resulted from the fact that the repository performance is measured and regulated solely by the radiological safety, i.e., annual dose for the public by intake of radionuclides that are released from the repository.

As the understanding about repository mechanisms and importance of the flexibility in decision-making process backed by variety of options has achieved remarkable progress, new roles of P&T seem to have emerged, which can be observed from two points. One is the convergence of the spent fuel management. As far as the nuclear power utilization continues, stockpile of nuclear materials after their utilization will increase. This ever-increasing, diverging tendency makes the safety and safeguards of spent fuel management worse, and thus affected the public perception adversely. P&T can contribute to enhance convergence of nuclear material stockpile under sustainable nuclear-power utilization. The second point is the variety and flexibility. While the geological repository has moderate variety in technological options, they are limited basically by adjusting the time for cooling and choosing the location of the interim storage and repository. P&T enhances the technological variety significantly by augmenting options of waste forms and radionuclide inventory in waste forms, which would also open a possibility of making a repository more compact and robust.

Keywords
Convergence, Reversibility, Geological disposal

Expectation to Nuclear Transmutation

Akito Arima

Musashi Academy

1. 一次エネルギー、電力の需要はますます伸びる
   The demand for primary energy and electricity is increasing year by year.

2. 地球温暖化は一層深刻化する
   The global warming is becoming a more serious problem.

3. 再生可能エネルギーは開発せよ
   しかし時間と金がかかる
   The development of renewable energy has to be promoted. However, it will require sufficient resources of time and budget.

4. 人類は原子力も使わざるをえない
   Human beings cannot help depending on nuclear energy as well as other energy resources.

5-1 原子力の将来のための安全技術を開発せよ
   The safety technology of nuclear energy has to be developed for the future.

5-2 バック・エンド技術を強化し、最終処理場を早く決めよ
   それは国の責任
   The technology for the back-end of nuclear fuel cycle has to be enhanced. The site for final disposal of nuclear wastes has to be determined as soon as possible, which is a responsibility of the Government of Japan.

5-3 ADSのような技術を研究開発し、最終処理をやり易いようにせよ
   The research and development of innovative technologies such as Accelerator Driven System have to be promoted in order to encourage the progress of final disposal.

5-4 廃炉、安全技術、バックエンド技術etc. で国際協力を強力に行え
   The research and development of nuclear technologies for reactor decommissioning, safety technology, and back-end, etc., have to be promoted intensively through international cooperation.

Contribution of the European Commission to a European Strategy for HLW Management through Partitionning & Transmutation Presentation of MYRRHA and its in the European P&T strategy

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Presently, the European Union relies for 30% of its electric power production on Generation II-III fission nuclear reactors leading to the annual production of 2500 t/y of used fuel, containing 25 t of Plutonium, and High Level Wastes (HLW) such as 3.5 t of minor actinides (MA), namely Neptunium (Np), Americium (Am) and Curium (Cm) and 3 t of long-lived fission products (LLFPs). This MA and LLFP stocks need to be managed in an appropriate way. The used fuel reprocessing (closed fuel cycle) followed by the geological disposal or the direct geological disposal (open fuel cycle) are today the envisaged solutions depending on national fuel cycle options and waste management policies. Required time scale for the geological disposal exceeds our accumulated technological knowledge and this remains the main concern of the public. The Partitioning and Transmutation (P&T) has been pointed out in numerous studies as the strategy that can relax constraints on the geological disposal, and reduce the monitoring period to technological and manageable time scales. Therefore a special effort has to be made to integrate P&T in advanced fuel cycles and advanced options for HLW Management. Transmutation based on critical or sub-critical fast spectrum transmuters should be evaluated, in order to assess the technical and economical feasibility of this waste management option, which could ease the development of a deep geological storage.

Despite diverse strategies and policies pursued by European Member States concerning nuclear power and envisaged fuel cycle policy ranging from the once through without reprocessing to the double-strata fuel cycle ending with the ADS as the ultimate burner or Gen-IV fast critical reactors multi-recycling all transuranic (TRUs), P&T require an integrated effort at the European and even worldwide level. Even when considering the phase out of nuclear energy, the combination of P&T and a dedicated burner such as ADS technologies, at a European scale, would allow to meet the objectives of both types of countries, the ones phasing out the nuclear energy as well as countries favouring the continuation of the nuclear energy development towards the deployment of new fast spectrum systems.

After nearly twenty years of basic research funded by national programmes and EURATOM framework programmes, the research community needs to reach a position of being able to quantify indicators for decision-makers, such as the proportion of waste to be channelled to this mode of management, but also issues related to safety, radiation protection, transport, secondary wastes, costs, and scheduling.

From 2005, the research community on P&T within the EU started structuring its research towards a more integrated approach. This resulted during the FP6 into two large integrated projects namely EUROPART dealing with partitioning and EUROTRANS dealing with ADS design for transmutation, development of advanced fuel for transmutation, R&D activities related to the heavy liquid metal technology, innovative structural materials and nuclear data measurement. This approach resulted in a European strategy given in introduction based on the so-called “four building blocks” at engineering level for P&T that will result in identification of the costs and the benefits of partitioning and transmutation for European society.

The MYRRHA project contributes heavily to the third building block of this European strategy and in this paper we will focus on the ADS programme in the EU through the MYRRHA project.

OECD / NEA Activities Related to the Nuclear Fuel Cycle

Thierry Dujardin

OECD / NEA
Acting Deputy Director-General
Deputy Director, Science and Development

The OECD Nuclear Energy Agency is an intergovernmental organisation which mission is:

- To assist its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as

- To provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

The OCDE/NEA is conducting a series of projects related to the Nuclear Fuel Cycle, mainly through three of its standing technical committees: the Nuclear Science Committee, the Nuclear Development Committee, and the Committee on the Safety of Nuclear Installations which methods of work will be briefly described.

Scientific activities related to fuels, materials, physics, separation chemistry and fuel cycle scenarios are being undertaken, mostly focussing on advanced nuclear systems.

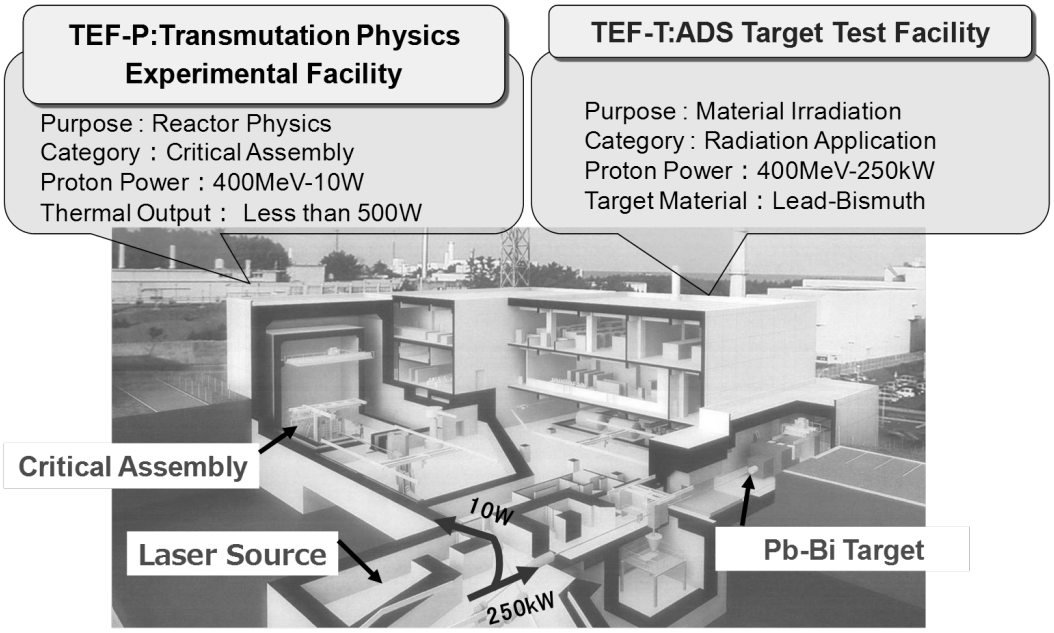
The recent study on the Economics of the Back-End of the Fuel Cycle and its policy recommendations will also be presented.

ADS Study in JAEA

Hiroyuki OIGAWA

Japan Atomic Energy Agency

Management of radioactive wastes is one of the most critical issues to use nuclear power. By transmuting long-lived minor actinides (MA) contained in high-level radioactive wastes to short-lived or stable nuclides, reduction of the burden of nuclear waste disposal can be foreseen. Aiming at effective transmutation of MA, the Japan Atomic Energy Agency (JAEA) is conducting research and development on an Accelerator-Driven System (ADS), which is a subcritical nuclear reactor coupled with an intense spallation neutron source driven by a proton accelerator. One unit of ADS with thermal power of 800 MW can transmute 250 kg of MA annually, which corresponds to the amount generated in 10 units of light water reactors with electric power of 1 GW per year. In JAEA, a variety of technical areas are being studied to verify feasibility of ADS. A superconducting proton LINAC is being developed, and its reliability is being investigated. Lead-bismuth eutectic (LBE) is considered as a primary candidate of a spallation target and a core coolant, though we do not have sufficient experience on it. We are, therefore, developing the LBE technology from viewpoints of material corrosion, thermal hydraulics, activation products, and so on. For the safety of a subcritical core, we are performing transient analysis on various accidental conditions. Moreover, construction of the Transmutation Experimental Facility (TEF) is being planned as a phase-2 program of the J-PARC project to perform basic experiments for transmutation technology. TEF consists of two facilities: the ADS Target Test Facility (TEF-T) and the Transmutation Physics Experimental Facility (TEF-P). TEF-T is a material irradiation facility by using 400 MeV – 250 kW proton beam and an LBE spallation target. In TEF-T, we expect to accumulate important experimental data for beam window of ADS. TEF-P is a critical / subcritical assembly with maximum thermal power of 500 W. TEF-P can accept a 400 MeV – 10W proton beam. MA-bearing fuel can be loaded in TEF-P. We will investigate physics properties of transmutation systems in TEF-P.



Keywords
minor actinides, high-level radioactive wastes, transmutation, accelerator driven system, spallation target, superconducting proton accelerator, subcritical core, Transmutation Experimental Facility, J-PARC

Accelerator-Driven System (ADS) Study in Kyoto University Research Reactor Institute (KURRI)

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The experimental studies on the accelerator-driven system (ADS) are being conducted for an application to nuclear transmutation techniques with the combined use of the Kyoto University Critical Assembly (KUCA; A-core: solid-moderated and -reflected core) and the fixed-field alternating gradient (FFAG) accelerator, in the Kyoto University Research Reactor Institute. At KUCA, the spallation neutrons generated by 100 MeV proton beams from the FFAG accelerator had been successfully injected into the uranium-loaded core for the first time in the world, and also into the thorium-loaded core. For the ADS experiments with 100 MeV protons, the static conditions and kinetic behaviors were examined in aiming at the establishment of measurement techniques of reactor physics parameters, including the reaction rates, the neutron spectrum, the neutron multiplication, the neutron decay constants and the subcriticality, and the confirmation of numerical simulation precision by the Monte Carlo approach.

An upcoming ADS at KUCA could be composed of highly-enriched uranium-fueled and Pb-Bi-reflected core, in consideration of actual ADS designed by the Japan Atomic Energy Agency. Especially the neutronic characteristics of Pb-Bi are considered importantly analyzed experimentally from in the viewpoint of reactor physics: neutron yield and neutron spectrum by Pb-Bi target; uncertainties of Pb-Bi cross sections in the core. Also, Pb-Bi liquid behaviors are expected to be examined with the use of Pb-Bi liquid loop equipments and test facilities for the investigation of Pb-Bi liquid characteristics from the viewpoints of heat transfer and thermal hydraulics in two-phase flow. In the future, as preliminary study on the Pb-Bi characteristics, the sample worth experiments on Pb, Bi and Pb-Bi could be conducted to investigate the uncertainties of these cross sections with the use of the solid plates. Furthermore, irradiation experiments of the minor actinides (^{237}Np and ^{241}Am) could be carried out in hard spectrum core at KUCA to conduct the feasibility study on conversion analyses (^{237}Np/^{238}U and ^{241}Am/^{235}U).

Keywords
ADS, KUCA, FFAG accelerator, Pb-Bi, Nuclear transmutation

Nuclear Data Study for Nuclear Transmutation

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The current national policy in Japan for the management and disposal of Minor Actinides (MAs: Np-237, Am-241, Am-242m, Am-243, Cm-244, Cm-245, Cm-246, etc.) and Long-Lived Fission Products (LLFPs: Se-79, Zr-93, Tc-99, Pd-107, Sn-126, I-129, Cs-135, etc.) generated in fission reactors is a sequence of vitrification, interim storage, and then depositing underground together with other nuclear wastes. Alternatively, if MAs and LLFPs are extracted and transmuted into stable nuclides, the environmental loading in the geological repository becomes very small. Moreover, the ethical problem that MAs and LLFPs are undesirable property for our far descendants will be solved.

From this viewpoint, the nuclear transmutation of MAs and LLFPs is a very attractive subject, and neutron capture and/or fission reactions are the most promising transmutation reactions. Therefore, databases on neutron capture and/or fission reaction cross sections are indispensable for developing the transmutation technology. However, the accuracy of nuclear databases for MAs and LLFPs is quite poor both in quality and quantity at the present time. Therefore, the improvement of the database accuracy is an urgent task.

In this presentation, the present status of the accuracy of nuclear databases such as JENDL-4.0 and ENDF/B-VII.1 is explained for some important nuclides of MAs and LLFPs. Then, a general view of activities of nuclear data measurements in the world such as those of n_TOF at CERN, LANSCE at LANL, and ANNRI at J-PARC/JAEA is taken. Finally, a Japanese nuclear data project entitled "Systematic Study on Neutron Capture Reaction Cross Sections for the Technological Development of Nuclear Transmutation of Long-Lived Nuclear Wastes" is briefly described. The objective of the project is to contribute to the improvement of the database accuracy, by making the precise measurements of capture cross sections of MAs and LLFPs, analyzing the measured results theoretically, elucidating the capture reaction mechanism of MAs and LLFPs, and supplying reliable calculated capture cross sections for all MAs and LLFPs in the whole neutron energy region.

This work was supported by JSPS KAKENHI Grant Number 22226016.

Keywords
Nuclear transmutation, Minor actinides, Long-lived fission products, Neutron capture, Neutron fission, Cross sections

Poster session

1a-1

Physical Analysis of LBE Spallation Target for CIADS

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For the project of the Chinese Initiative Accelerator Driven Sub-critical system(CIADS), the Lead-Bismuth-Eutectic(LBE) spallation target is one of the two alternatives, which has high quality of thermal performance, mature technology, and other advantages. In the ADS, the high intensity proton beam bombards on the spallation target through the beam window and produces neutrons, then the neutrons are used to driven the sub-critical system. Physically, the design of the spallation target determines the production and the utilization of the neurons, as well as the performance of the sub-critical reactors and other key issues.Focusing on the energy deposition and the neutron flux distribution of the target and the beam window, we got the curve of target with different geometric parameters and beam sizes, through a series of calculations using the Monte Carlo program-MCNPX. Analysis of the target performance coupled to the sub-critical reactor was done, from the point of view of physics. And targets with different locations and different geometries in the core of sub-critical reactor were studied considering the energy deposition rate of the target and windows. In the mean time, the profile of neutron flux and the neutron energy spectrum for different positions were calculated. The relationship of keff and the power amplification factor of the couple system were also studied. The energy deposition rate can give a heat source for the thermal-hydraulics analysis. And other parameters can be used for the optimization of the LBE target design.

Keywords: LBE Target , Energy Deposition Rate , Target coupled with the reactor , keff

1a-2

Analysis of Thermal-hydraulics for the LBE Spallation Target of ADS

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The spallation target is an important part of the Acceleration Driven sub-critical System (ADS). A loop-type Lead-Bismuth-Eutectic(LBE) spallation target was studied through the investigation of the existing LBE target concepts . The code MCNPX was used for the physics simulation of the spallation process, the CFD code FLUENT was used for the thermal-hydraulics analysis. The methodology was verified and validated by repeating the work mentioned in some literatures. This paper mainly introduces to our work related to the analysis of thermal-hydraulics of the spallation target. The thermal-hydraulic behavior of the target has been studied by using FLUENT. A series of the results were obtained by changing the inlet velocity of LBE, the diameter of proton spot, and the intensity of proton beam under the condition that the thickness of target window is fixed, and the allowable intensities of beam under the different thickness of target windows were obtained. The operating condition which meet the criterion of thermal-hydraulics with enough margin was given .The results are valuable for the parameter choice of the LBE spallation target for the Chinese Initiative Acceleration Driven sub-critical System (CIADS)

Keywords
LBE spallation target, ADS, Thermal-hydraulics

1a-3

Study on Neutron Spectrum of Pulsed Neutron Reactor

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Accelerator Driven Subcritical Reactor (ADSR) is considered as one of the best candidate to annihilate the radioactivity of nuclear waste, and has been investigated in many institutes for many years. ADSR is operated by the pulsed proton beam as an ignition of spallation reaction to produce many neutrons. Kyoto University Critical Assembly (KUCA) is one of the facilities to demonstrate ADSR by using accelerated proton for spallation reaction or deuteron for DT reaction.

The experiments at KUCA modeled thermal reactor with DT reaction had been performed in order to clarify the validity of reactivity measurement through the fitting of neutron decay constant at various positions in subcritical and critical states. For the analysis of the experiments, continuous energy Monte Carlo code: MVP-II with JENDL-4 library was used in eigenvalue mode and time-dependent mode.

The neutron distribution evaluated by time-dependent mode is changing with time after neutron ignited into the system, and the neutron distribution in energy and space becomes almost stable after 1 micro-second of neutron ignited. This paper focuses on the neutron spectra of two modes in subcritical state, although there are many specific points to be considered in subcritical system.

There is a remarkable difference in neutron spectra between two results of eigenvalue and time dependent modes. The neutron spectrum (at 1 micro-second after neutron ignited) evaluated in time dependent mode is softer than that in eigenvalue mode, and the difference is more remarkable as the subcriticality is larger. This fact can be understood by considering the phenomena caused in the subcritical system described in the following. The ignited neutron causes many types of reactions; fission, radiative capture, scattering, etc. In subcritical system, the magnitude of neutron is decreasing with time but thermal neutron decrease after fast neutron starts to decrease. Thermal neutron needs additional time to be moderated in the system, in other words, although thermal (moderated) neutron decreases after moderation process, fast (un-moderated) neutron decreases without moderation process. This causes the time-difference of decreasing between fast and thermal neutrons, and thus neutron spectrum is softer in time-dependent mode compared to that in eigenvalue mode.

Eigenvalue mode is usually used to evaluate k-eigenvalue, and k-eigenvalue is useful to check the subcriticality of the system. It is widely known that the neutron distribution in space obtained by k-eigenvalue mode is different from that by fixed-source mode, therefore there are many suggestions about how to evaluate the effective-subcriticality. In addition to this, this paper emphasizes the necessity of the consideration to evaluate the effective-subcriticality that the neutron distribution in energy obtained by k-eigenvalue mode is also different from that by time-dependent mode. When the neutron spectrum of pulsed neutron reactor is evaluated, the alpha-eigenvalue mode instead of k-eigenvalue mode is recommended in order to match the neutron spectrum during the decreasing in time after pulsed neutron is ignited to the subcritical system.

Keywords
ADSR, Neutron spectrum, k-eigenvalue mode, time-dependent mode, alpha-eigenvalue

1a-4

Theory of power spectral density and Feynman-α method in Accelerator Driven System and their higher order mode effects

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It is important to monitor the subcriticality of an accelerator driven system (ADS) for maintaining its nuclear criticality safety. Reactor noise techniques are promising candidates for its purpose since the techniques can be installed with the minimum disturbance of the reactor operations. This paper focuses on the methods of power spectral density and Feynman-α as reactor noise methods for monitoring the subcriticality of an ADS where pulsed spallation neutrons are injected with a constant time-period. That is, the neutron source in an ADS does not have a Poisson character. Thus, the conventional reactor noise theory, which does not account for either periodically pulsed neutron source or for its non-Poisson character, cannot be applied to the noise measurements in ADSs. Many studies have been performed for noise theory in ADSs. Through these studies we know that the Feynman *Y* function is composed of the sum of a correlated and uncorrelated component. On the other hand, the power spectral density is also composed of two components: a correlated component and an uncorrelated component of an infinite number of the Dirac delta function peaks. Yamamoto very recently developed a Monte Carlo method to calculate a series of the Dirac delta function peaks by solving the frequency domain neutron transport equation using the complex-valued weight Monte Carlo method[1)]. In this paper, the formulae of the Feynman *Y* function and power spectral density are presented for an infinite homogeneous multiplying system where neither spatial-higher order modes nor energy-higher order modes are excited. These formulae are verified with comparison to their Monte Carlo simulations.

In a realistic ADS, the spatial or energy- higher order mode effects cannot be neglected because ADSs are operated with a certain level of subcriticality and their dimensions are finite. To obtain accurate information on the subcriticality, higher order mode effects in the Feynman *Y* function and power spectral density need to be evaluated and thus the theory on the higher order modes of the reactor noise techniques in an ADS is needed. This paper will present the space- and energy-dependent formulae of the Feynman *Y* function and power spectral density. These formulae have been applied to numerical test calculations for an ADS with one-dimensional infinite slab geometry. The Feynman *Y* function and power spectral density are obtained by the Monte Carlo simulations for noise measurements in the ADS. The results of the Monte Carlo simulations are compared with the theoretical formulae. Both results agree very well, which may show verification of the theoretical formulae. The Feynman *Y* function and power spectral density are decomposed into the sum of the fundamental mode, which we seek to know, and higher order modes. Generally speaking, the higher order mode effects are more significant in the uncorrelated component of the Feynman *Y* function and power spectral density. Especially, it is found that the first higher order mode is almost comparable to the fundamental mode in the uncorrelated component. Thus, an α value obtained by fitting the uncorrelated component of the power spectral density to a conventional formula for power spectral density, $A/(\alpha^2+\omega^2)$, largely deviate from the fundamental mode α value. On the other hand, the correlated component of the power spectral density is less contaminated by the higher order modes. We can obtain an α value that is much closer to the fundamental mode α value by using the correlated component.

1) Toshihiro Yamamoto, "Higher order mode analyses of power spectral density and Feynman-α method in accelerator driven system with periodically pulsed spallation neutron source," *Ann. Nucl. Energy* (in press).

Keywords
Rector noise, power spectral density, Feynman-α, ADS, higher order mode

1a-5

Neutron requirement for transmutation of MA isotopes

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In the aftermath of the accident at the Fukushima-1 Nuclear Power Station, a reorientation is occurring in Japan with respect to the future of nuclear power. One of the topics that is gaining interest is the process of actinide transmutation to reduce the amount of long-lived nuclear waste (LLW). The objective of transmutation is to fission the LLW nuclides, because fission product isotopes in general have a relatively short half-life.

Minor actinide (MA) nuclides remaining in spent nuclear fuel are by definition not fissile, so one expects to require a minimum of roughly two neutrons for transmutation: one neutron to turn the non-fissile isotope into a fissile isotope, and a second neutron to cause fission. If the first or second neutron does not cause fission, one needs even more neutrons for transmutation. When the fission occurs, ν new neutrons are released into the system. One needs to calculate how many neutrons are needed to fully transmute a given isotope and all of its possible descendants. If the balance is positive, full transmutation can be achieved with a (series of) critical reactor(s). If the balance is negative, external neutrons are needed – provided for instance by an Accelerator Driven System (ADS) or a nuclear fusion reactor (fission – fusion hybrid).

The total number of neutrons released in a transmutation chain can be calculated from elementary considerations as a summation over the probability that transmutation reaction *x* occurs, multiplied by the number of neutrons released in that reaction (-1 for capture, ν -1 for fission, (N-1) for (n,Nn), 0 for radioactive decay, etc). The governing formula is derived in [1]:

$$\bar{n}_{r,I} = \sum_{j1} P_{I\to j1}\left\{\bar{n}_{r,j1} + \sum_{j2} P_{j1\to j2}\left[\bar{n}_{r,j2} + \sum_{j3} P_{j2\to j3}\cdots\right]\right\}$$

This equation is somewhat cumbersome to evaluate on a computer (not in the least because the transmutation chain is infinitely long in principle). Therefore, we propose to use a somewhat different approach. Suppose one has a system with a constant neutron flux, to which one isotope is added continuously. After sufficient time, the mixture will reach a steady state composition. The composition reflects the probabilities of the transmutation reactions. Thus neutron balance for transmutation is related to the infinite multiplication factor of the hypothetical steady state mixture. This will be detailed in the full paper.

A computer program based on this approach was created, using the transmutation matrix from ORIGEN-S. Some results are given in the table. It is clear that complete transmutation, in general, cannot be achieved with thermal reactors alone; fast reactors are needed. The most important isotope is obviously U-238, as this is the source isotope for all MA.

Isotope	Thermal	Fast
U-238	-0.51	0.72
Pu-239	0.23	1.54
Np-237	-1.44	0.61
Am-241	-1.54	0.66
Cm-244	-1.28	1.45

Depending on the future availability of fast reactors, external neutrons (ADS) will be required for transmutation purposes..

[1] M. Salvatores, I. Slessarev, M. Uematsu, "A global physics approach to transmutation of radioactive nuclei", Nucl. Sc. Eng. **116**, 1-18 (1994)

Keywords
Nuclear transmutation, neutron economy, ADS, transmutation reactor

1a-6

Application of the Resource-renewable Boiling Water Reactor for TRU Management and Long-Term Energy Supply

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An adaptation of the BWR has been conceived that has the potential to breed and consume transuranium elements (TRUs) using a multi-recycling process. This is the resource-renewable BWR (RBWR), and it can be used as a long-term energy supply, while greatly reducing the negative environmental impact that TRUs cause as they are otherwise long-lived radioactive wastes.

A major characteristic of the BWR is “boiling” in the core which includes water that has the roles of being a moderator and a coolant. The neutron energy spectrum can be hardened by reducing the hydrogen-to-uranium ratio (H/U) using the two-phase flow and combining with the hexagonal tight fuel lattice, so that the transmutation of U-238 to fissile plutonium is promoted with increasing resonance absorption. This enables the multi-recycling process of both breeding and consuming TRUs.

The rated electric power, core pressure and diameter of the pressure vessel of the RBWR are 1350 MWe, 7.2 MPa and 6200 mm, respectively, and these values are identical to those of the latest commercial BWR, the advanced ABWR (ABWR). The core configuration in the horizontal direction is composed of 720 hexagonal fuel bundles and 223 Y-type control rods. The height configuration uses a parfait core concept in which an internal blanket of depleted uranium oxide is placed between the upper and lower fissile zones of the TRU oxides. Cores with a compatible fuel bundle each other, but for different purposes, have been proposed so far; they are the TRU burners (RBWR-TB/TB2) and the break-even reactor (RBWR-AC). The RBWR has a capability to make the transition between the TRU burners and the RBWR-AC only by changing fuel bundles.

The RBWR-TB is the TRU burner that is designed to fission almost all the TRUs, leaving only a minimum critical mass of TRUs, by repeating the recycling of them to remove the concern that TRUs will be long-lived radioactive wastes when their usefulness has been exhausted. The RBWR-TB2 is another type of the TRU burner, and it is designed to be suitable to burn TRUs from LWR spent fuels, while the RBWR-TB is designed as a burner for TRUs from the RBWR-TB itself. The fuel bundles of the RBWR-TB and TB2 have a relatively larger H/U value than that of the RBWR-AC to transmute fertile minor actinides into fissile fuel elements and fission them. The numerical calculation showed the fission efficiencies of TRUs are more than 50% in the RBWR-TB and more than 40% in the RBWR-TB2. The fission efficiency is defined as the net decrease in TRUs divided by the total amount of fissioned actinides for the total fuel life. The fission efficiencies in the RBWR-TB and RBWR-TB2 respectively correspond to more than twice and nearly twice the amount of TRUs accumulated in the ABWR per generating power.

The RBWR-AC is the break-even reactor, which has a breeding ratio of 1.01. This ratio is defined as the number of atoms of fissile plutonium left in the discharged fuel bundles per fissile plutonium in the initial fuel bundles. The RBWR-AC burns depleted uranium by using TRUs extracted from the spent fuel bundles of LWRs without decreasing the amount of the TRUs. The fuel bundle of the RBWR-AC is composed of 271 fuel rods in a triangular lattice.

All of the above RBWR cores achieve negative void coefficients. In addition the isotopic composition of TRUs will be preserved through the multi-recycling process by adjusting the neutron energy spectrum. This is necessary to continue the fission and recycling process of TRUs while maintaining the criticality and meeting the various constraints, such as the negative void coefficient.

Various fast reactors have already been proposed and developed for the TRU management so far, but it is thought to be also valuable to investigate achieving breeding and consuming of TRUs by LWRs with proven technologies.

Keywords
BWR, TRU, burner, break-even, void coefficient, multi-recycle

1a-7

Investigations of Lead-Bismuth cooled Accelerator-Driven System

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The Japan Atomic Energy Agency (JAEA) has been conducting research and development on an accelerator-driven system (ADS) to transmute minor actinides (MAs) included in high-level waste (HLW) into short-lived nuclides. The ADS proposed by JAEA is a lead bismuth eutectic (LBE) cooled-tank-type ADS. The system consists of a high-intensity proton accelerator with 1.5GeV beam energy, a subcritical core with 800MW thermal power and an LBE spallation target placed at a center of the subcritical core. Since the ADS is a hybrid system of an accelerator and a nuclear reactor, there are various inherent issues in the research and development of the ADS. In this study, two important issues are introduced from the viewpoint of the ADS plant investigation and the latest results are presented.

The first issue is a design of a beam window. The beam window forms a boundary between the accelerator and the spallation target. It is obvious that the beam window should be tolerant to the following severe conditions: external pressure by the LBE, heat generation by the proton beam, creep deformation at high temperature, corrosion in the LBE and irradiation damage by neutrons and protons. Through the calculations of particle transport, thermal hydraulics and structural analysis, a feasible beam window concept, which was an ellipse one with 2mm thickness as minimum, was presented with the consideration of the external pressure, the heat generation and the corrosion effect. The consideration of the creep deformation and the irradiation damage will be the future work for the beam window design.

Another issue is a safety of the ADS. It is considered that the ADS is safer than critical reactors because the ADS is operated in a subcritical state. Since the system is operated under the subcritical condition, its shutdown is easy and a possibility of a critical accident is smaller than the conventional critical reactors. Considering such characteristics of the ADS, a preliminary investigation for the safety was performed by the Level 1 PSA (Probabilistic Safety Assessment) and transient analyses were carried out for the typical accidents for the ADS. In the case of beam window breakage, as a typical ADS accident, it was confirmed that the LBE flowed into a beam duct and the spallation neutrons in the subcritical core decreased. Then, the core power decreased. In the case of protected loss of heat sink (PLOHS), as one of the severest accidents, the calculation result indicated that there was a possibility of the core damage because the cladding tube temperature reached at its melting point after 20h. As a next step, it is required to design a safety system of the ADS to decrease these frequencies and to ease the accidents.

Keywords
Accelerator-Driven System (ADS), Beam window, Transient analysis, Safety, Lead-bismuth eutectic (LBE)

1a-8

J-PARC Transmutation Experimental Facility

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After the Fukushima accident, nuclear transmutation got much interested as an effective option of nuclear waste management. Japan Atomic Energy Agency (JAEA) proposes the transmutation of minor actinides by accelerator-driven system (ADS) using lead-bismuth eutectic alloy (Pb-Bi) as a spallation target and a coolant of subcritical core. Current design of ADS has 800MW_{th} of rated power, which is driven by 20MW proton LINAC, to transmute minor actinides generated from 10 units of standard Light Water Reactors.

To obtain the data required for ADS design including European MYRRHA project, JAEA plans to build a Transmutation Experimental Facility (TEF) within the framework of J-PARC project. TEF consists of two buildings, one is an ADS target test facility (TEF-T), which will be installed high power Pb-Bi spallation target, and the other is Transmutation Physics Experimental Facility (TEF-P), which set up a fast critical/subcritical assembly driven by low power proton beam. TEF will be located at the end of 400 MeV LINAC of J-PARC and accepts 250kW proton beam with repetition rate of 25 Hz.

As a major research and development items of TEF-T, irradiation test for candidate ADS structural materials, engineering tests for Pb-Bi target operation and experiments to determine the effective lifetime of proton beam window will be performed. Several kinds of Pb-Bi target will be used according to above mentioned research purposes. Reference design parameter, that considers operation condition of ADS transmutor, was determined by thermal-hydraulic analyses and structural analyses. When the target operates with full power beam, fast neutron spectrum field is formed around the target and it is possible to apply multi-purpose usage. Various research plans have been proposed and layout of the experimental hall surrounding the target are underway. Basic physics application such as measurements of nuclear reaction data is considered as one of the major purposes.

In the presentation, roadmap to establish the ADS transmutor and design activities for TEF construction will be summarized.

Keywords

Transmutation, Accelerator-driven System, J-PARC, Transmutation Experimental Facility

1a-9

Development of Uranium-Free TRU Metallic Fuel Fast Reactor Core with Enhanced Doppler Reactivity Feedback

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For sustainable nuclear power deployment, not only ensuring its enhanced safety but also reduction of environmental burden associated with radioactive waste management is a challenging issue for the international community. History has shown that getting public support is difficult for waste management plans which involve mass disposal of radioactive waste with a half-life of tens of thousands of years. Therefore, as one of the solutions Toshiba has been developing a system, taking into account that for the time being light water reactors (LWRs) play a leading role in commercial nuclear power plants, which enables toxicity and radioactivity of high level waste to be reduced to the same as those of natural uranium within a few hundred years.

This system is characterized by a closed fuel cycle which encircles following main facilities: fuel manufacturing plants which fabricates uranium-free transuranium (TRU) metallic fuel and long-lived fission products (LLFPs) target from TRU and LLFPs extracted from LWR spent fuel, fast reactors to burn those fuel, and recycling facilities to reprocess and refabricate the spent fuel from fast reactors by pyroprocessing. Use of TRU metallic fuel without uranium makes it possible to maximize TRU transmutation rate in comparison with uranium and plutonium mixed oxide fuel, because it prevents the fuel itself from producing new plutonium and minor actinides, and furthermore because metallic fuel has much smaller capture-to-fission ratios of TRU than those of mixed oxide fuel. Besides, adoption of metallic fuel enables recycling system less challenging even for uranium-free fuel because conventional scheme of fuel recycling by electrorefining and injection casting is applicable. Although substances remained after reprocessing are finally to be disposed outside the cycle, their toxicity and radioactivity are diminished to the same level as those of natural uranium by establishment of technologies to enhance burning and processing rates and storing them for a few hundred years within the system.

Safety prospects for uranium-free TRU fuel, however, are yet to be ensured since Doppler reactivity feedback of the core is anticipated to decrease seriously in the absence of uranium. The purpose of this paper is to analyze the effect of following measures taken, respectively, to enhance Doppler reactivity feedback of the core: (1) use of the elements with relatively larger resonance neutron capture cross section under fast neutron spectrum circumstances, e.g. Zr, Nb, Mo, etc., and (2) use of neutron moderators, e.g. Be, ^{7}Li, ^{11}B, H, etc., to enhance negative Doppler reactivity feedback due to even-plutonium isotopes. Assuming a typical prototype fast reactor core, analysis was performed using in-house two dimensional neutron diffusion calculation code. The results show that the equal or much greater negative Doppler coefficient compared to that of the metallic fuel containing uranium can be obtained when using elements such as Nb or Mo as a metal fuel alloy or moderators such as BeO, $^{7}Li_2O$, $^{11}B_4C$, or ZrH_2. Consequently, it is confirmed that aforementioned measures have potentiality to enhance Doppler reactivity feedback of the uranium-free TRU metallic fuel core.

Keywords
Uranium-free, Fast Reactor, Metallic Fuel

1a-10

Chemical Isotopic Analysis and Computational Evaluation of Nuclide Composition in PWR-UO_2 and MOX Spent Fuels Using JENDL, ENDF/B, JEF, and JEFF

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Isotopic composition in the spent fuels is basic data to evaluate criticality of the spent fuels as well as the source intensity such as radiation emission and heat generation. Especially, at the accident of the nuclear reactors, the estimation of the inventory of the nuclides in the fuels loaded into reactors or spent fuel storage pools is quite important. Therefore, the verification and the improvement of the accuracy for the calculated nuclide compositions in spent fuels are required.

In this study, chemical isotopic analyses of actinides and fission products were carried out on a high burn-up PWR-UO_2 and a high burn-up MOX fuel. Furthermore, computational analyses were performed using the integrated burn-up calculation code SWAT. The differences between the amounts by the chemical isotopic analysis and the SWAT calculation using JENDL-3.2, JENDL-3.3, ENDF/B-VI.5, ENDF/B-VI.8, JEF-2.2 and JEFF-3.0 were evaluated as the ratios of the calculated values to the experimental ones (C/E ratios). Regarding actinides, the calculated ^{244}Cm amount, which is an important nuclide as a major neutron source in spent fuels, was underestimated. The main sensitive path to ^{244}Cm was therefore investigated by a simple depletion calculation for actinides and the cause of its underestimation was discussed. Fission products ^{88}Sr, ^{90}Sr, ^{89}Y, ^{106}Ru, ^{133}Cs, ^{135}Cs and ^{144}Nd in the PWR-UO_2 fuel, and ^{88}Sr, ^{90}Sr, ^{106}Ru, ^{133}Cs, ^{134}Cs and ^{135}Cs in the PWR-MOX fuel, which contribute to gamma sources, decay heat, neutron absorption or a burn-up indicator in spent fuels, were further investigated to improve their C/E values by using simplified burn-up chains of fission products using JENDL-3.3, and consequently, the corrected value for the fission yield or the capture cross section of fission products was estimated by using the sensitivity coefficients.

As the results of the analysis for the PWR-UO_2 and the PWR-MOX fuels. The calculation accuracy of ^{244}Cm will be quite improved by the reevaluation for both capture cross sections of ^{243}Am and plutonium isotopes. For ^{88}Sr, ^{90}Sr and ^{106}Ru, the correction for their own fission yields improved their C/E ratios. The correction for the ^{89}Sr fission yield improved the C/E ratio for ^{89}Y, and that for the ^{133}Xe fission yield improved the C/E ratio for ^{133}Cs and ^{134}Cs. For ^{135}Cs, it suggests that the underestimation of the fission yield or the overestimation of the capture cross section of ^{135}Xe results mainly in the underestimation of the calculated ^{135}Cs amount, and in the case of ^{144}Nd, it suggests that the underestimation of the ^{144}Ce fission yield results mainly in the underestimation of the calculated ^{144}Nd amount.

The effect of the power history on nuclide composition was also investigated on the PWR-UO_2 fuel. In the comparison of the calculated amounts with the detailed power history and the constant one, the amounts of ^{241}Am, ^{242m}Am and ^{155}Gd were slightly affected by the power history.

Keywords
high-burnup UO_2 fuel, high-burnup MOX fuel, isotopic analysis, actinides, fission products, sensitivity analysis, JENDL-3.2, JENDL-3.3, ENDF/B-VI.5, ENDF/B-VI.8, JEF-2.2, JEFF-3.0

1a-11

Enhancement of Transmutation of Minor Actinides by Hydride Target

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High level wastes generated after reprocessing spent nuclear fuels include long-lived radioactive nuclides of minor actinides (MA), i.e. ^{237}Np, ^{241}Am, ^{243}Am, ^{244}Cm and so on. The transmutation of MA by the Fast Breeder Reactor (FBR) has intensively studied to reduce radioactivity of the wastes. Core design study with a driver fuel containing MA shows limitation on maximum allowable MA content to avoid the impact of reactivity coefficients. Another loading mode of MA was studied, where MA is separately located at radial blanket region. The latter mode has no impact on main reactor core parameters but a transmutation rate is small due to lower neutron fluxes. A hydride target including MA is able to enhance the transmutation rate in FBR. Fast neutrons generated in the core region of FBR are moderated in the hydride target assemblies and then produce high flux of thermal and epithermal neutrons. The neutron reaction cross sections in the thermal energy region are large compared with that in the fast energy region. In this paper, a design study of core with MA hydride targets shows an advantage in transmutation of MA comparing with core with a driver fuel containing MA.

Previously, a number of transmutation using moderated subassemblies in FBR were proposed. In the case that Zr hydride and metal MA are separately located in the transmutation subassemblies, neutrons are moderated in the region of Zr hydride without absorption and cause power peaking in the fuel assemblies adjacent to the targets. The MA hydride target improves the power peaking problem, since the moderated neutrons are mostly absorbed in the MA hydride target. The MA hydride target has also an advantage on the increase of the MA loading content, since MA and MA-Zr alloy absorb hydrogen such as Zr.

Transmutation performances of the hydride MA target in fast reactors have been evaluated for the core of Japanese prototype fast reactor Monju, where the thermal power is 710MWt and the diameter of the active core is about 1800mm and the height is 930mm. A total of 54 MA hydride target assemblies are located at the most inner row of the three radial blanket rows. Each assembly contains 61 MA hydride target pins, where the diameter of the pellets is 10.4mm and the stack length is 930mm. The MA hydride target pin has hydride pellets consisting of $(MA_{0.3}, Zr_{0.7})H_{1.6}$,where the composition of MA is derived from LWR discharged fuel. The calculations with a metal MA target pin were also done for comparison.

The calculation results show that the transmutation amount of actinides is equivalent to the annual production of that from eight 1000MWe-class LWRs. The reduction ratio, which is the ratio of the transmuted MA to the loaded MA after one year, is 0.266. The effective half life, which is defined as the time required to be half of the loaded amount of MA is only 2.24 full power years, which is one fifth times smaller than that of the metal MA case. The hydride target shows excellent performance for transmutation of the wastes. We propose the energy system combining LWRs and FBRs with the hydride targets to enclose the long-lived nuclear wastes in the fuel cycle.

Keywords
hydride, minor actinide, fast breeder reactor

1a-12

Project to Modify the STACY Critical Facility for Experimental Study on Fuel Debris Criticality Control

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In the severe accident of Fukushima Daiichi Nuclear Power Stations (NPS), most of the fuel loaded in the Unit 1, 2 and 3 cores was seriously damaged and melted, resulting in a considerable amount of fuel debris. It is believed that some parts of fuel debris involve molten structural materials such as zircaloy, stainless steel and concrete. Toward the decommissioning of Fukushima Daiichi NPS, the whole fuel debris should be retrieved from each pressure vessel and containment vessel. The fuel debris, of which chemical composition and physical state are uncertain, needs to be treated with great care from criticality safety's viewpoint. In particular, large blocks of fuel debris can change in physical state such as size and water content when they are broken into fragments to retrieve in cooling water. Criticality control measures for the fuel debris, therefore, will have to be established by the time the fuel debris begins to be retrieved from Fukushima Daiichi NPS (scheduled for 2020 by the governmental council for the decommissioning of Fukushima Daiichi NPS).

For dealing with such a serious problem on criticality control of fuel debris, Japan Atomic Energy Agency (JAEA) has been planning to modify Static Experiment Critical Facility (STACY) and to pursue critical experiments on fuel debris. STACY, a facility using solution fuel, is to be converted into a thermal critical assembly using fuel rods and light water moderator. It is notable for the modified STACY that its core has a wide range of neutron energy spectrum between thermal-reactor spectra and intermediate-reactor ones. The core spectrum can cover a relatively hard spectrum of the fuel debris likely to become critical. A series of critical experiments will be conducted in the modified STACY using simulated fuel debris samples. The simulated fuel debris samples (sintered pellets) are to be manufactured by mixing uranium oxide and reactor structural materials (Zr, Fe, Si, Gd, B, etc.) with various chemical compositions. The manufacturing and analytical equipments for the simulated fuel debris samples are to be installed in glove boxes in the experimental building adjoining the modified STACY.

The STACY modification is under safety review by the regulatory authority of Japan to comply with new safety standards for research reactors enforced in December 2013. The first criticality of the modified STACY is scheduled for 2018. The modified STACY will provide benchmark data for criticality safety of fuel debris to validate the criticality control measures applicable to Fukushima Daiichi NPS.

Keywords
Fukushima Daiichi, Fuel debris, Criticality control, Criticality safety, Critical facility, STACY, simulated fuel debris sample

1a-13

ADS study in Tohoku University

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Accelerator-Driven System (ADS) has inherent problems arising from the combination of an accelerator and a subcritical core. Since ADS is stably-operated by the control of an accelerator, the change of beam profiles influences the core and its components. For example, if the beam incident position is shifted from the center of the target, the thermal power distribution will be varied. Moreover, the material strength of a beam duct will be worsened as the burnup proceeds because the beam current is gradually increased in a cycle to compensate the decrease of k_{eff} and maintain the thermal power. On the basis of these issues, we work through two researches: a reactivity control for the reduction of beam current, and transient analyses on the variation of beam condition such as shape and position.

With respect to a reactivity control, the introduction of burnable poison is studied, and the metal hydride is employed as the first candidate for burnable poison. We have investigated the use of the burnable poison assembly with Gd-hydride in the past, and the burnup swing is reduced by 74% in the case of ADS with the initial multiplication factor k_{eff} = 0.97 proposed by Japan Atomic Energy Agency (JAEA) as a result of the optimization of the specifications such as the ratio of hydride and loading position. Safety parameters such as Doppler coefficient and void reactivity coefficient are almost the same as the original core model.

As for transient analyses, ADS dynamics calculation code ADSE has been developed in Tohoku University to analyze the beam transient which accompanies with the variation of the beam condition as shape and position. ADSE is neutronics-thermohydraulics coupled code, and can deal with the change of the external neutron source. The variation of neutron flux, the thermal power, and temperatures of fuel, cladding and coolant are obtained from ADSE. The soundness of cladding is also able to be evaluated by the use of the programs of the thermoelastic analysis and the creep analysis. ADSE is applied to the former design of ADS with k_{eff} = 0.95 proposed by JAEA, and beam transient is analyzed. The result of the analysis indicates that the stress acting on cladding exceeds the failure criteria when the incident beam centroid moves large distance from the center of the target.

Keywords
ADS, reactivity control, hydride, transient, beam behavior,

1a-14

Method Development for Calculating Minor Actinide Transmutation in a Fast Reactor

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Fast reactors are one of the most promising devices to transmute minor actinides which have long-lived toxic because of presence of a large number of neutrons and hard neutron spectrum. Here we first define transmutation of each MA nuclide fuelled in a reactor core. The transmutation rate is composed of two terms; first is the amount of incineration rate by fission and the second is the net transmutation rate to fuel (Uranium and Plutonium). The first fission rates of individual nuclides contain the direct fission of the relevant nuclide plus the fission of other nuclides transmuted by decays or neutron reactions. It was found that the indirect fission contribution by Pu-238 and Pu-239 is remarkably large for nuclides Np-239 and Am-241. The net production rates of U and Pu are calculated from the difference between the production rate of U and Pu from the relevant MA nuclide and the MA production from the initial U and Pu.

The calculation method of the nuclide-wise transmutation rate has been developed. In this method we first perform the conventional burnup calculations, and store the burnup-dependent flux in each region, which is used in the second step calculation. In the second step, we consider only the relevant MA in each region, and perform burnup calculations using the flux obtained in the first step. To transmute MA we consider the homogeneous loading and the heterogeneous loading of MAs. In the homogeneous loading we optimize the transmutation rate of MAs and the safety aspect of a FR core. The sodium void reactivity worth and the Doppler reactivity is considered as the safety parameters. To reduce the sodium void reactivity worth we consider a FR core with sodium plenum on the top of fissile region and an internal blanket region. In the heterogeneous loading of MAs we consider special fuel assemblies with MAs and moderator materials. The merit of the heterogeneous loading is to decrease the number of special fuel assemblies with MAs et al.

In addition to the improvement of calculation method, we will develop the estimation method of calculation uncertainty based on the burnup sensitivity analysis. Also to improve the calculation accuracy of the sodium void worth the cross section adjustment method and the bias factor method will be improved.

Keywords
Minor actinide, Transmutation, Fast reactor

1a-15

Heat transfer study for ADS solid target
- surface wettability and its effect on a boiling heat transfer-

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Accelerator driven system (ADS) has been developed as a next-generation nuclear energy system. High energy particles and high density heat are generated by irradiating a proton beam to a spallation target. Although a lead-bismuth liquid target is the leading candidate as a spallation target for transmutation, however a solid target should be also investigated if ADS is used for a neutron source like a research reactor. Therefore, it is essential to establish a heat removal technique in the high energy radiation field from the solid target cooled by water. Such high energy radiation may affect the surface wettability and the boiling heat transfer. Recently some researchers have reported that surface wettability effect on a boiling heat transfer by using ultraviolet or gamma ray, however, there is no research on the relation between the proton beam irradiation and the surface wettability. Thus, in this study, the effect of the proton beam irradiation on the surface wettability was investigated by comparing with those of ultraviolet and gamma ray irradiations. Various material samples (titanium dioxide, stainless steel, copper, tungsten and tantalum) were irradiated by the Fixed Field Alternating Gradient (FFAG) accelerator in KURRI. As a result of the proton beam irradiation, the wettability enhancement was confirmed by measuring a contact angle on the surface before and after irradiations. This effect shows a similar tendency as that for ultraviolet and gamma ray. Additionally, the influence of water radiolysis to the radiation-induced surface activation was also suggested by comparing between the irradiations in water and air. On the other hand, the effect of the surface wettability on boiling heat transfer was investigated by applying photo- and radiation-induced surface activations. Copper heating surface coated with titanium dioxide layer was used and the contact angle was changed by ultraviolet irradiation. The contact angle of another copper surface was changed by gamma ray irradiation. Then, boiling heat transfer experiments in a pool boiling setup were performed with and without irradiation at various subcooling conditions. The heating surface temperature and heat flux were estimated from measured temperature distribution by assuming one dimensional heat conduction. The boiling regime transits from natural convection to nucleate boiling as the wall temperature increases. And then, microbubble emission boiling (MEB) occurred after critical heat flux (CHF). MEB shows higher cooling characteristics than CHF and was observed at all of the subcooled conditions at the present experimental conditions. Experimental results show that a nucleate boiling curve moves to higher wall superheat side with the irradiated surface, however the heat transfer enhancement in the CHF and MEB was not obvious at present experimental conditions.

Keywords
Surface wettability, Proton beam, Ultraviolet, Gamma ray, Microbubble emission boiling

1a-16

Experimental study of flow structure and turbulent characteristics in lead-bismuth two-phase flow

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For severe accident in an accelerator driven system, a gas-liquid two-phase flow with a large liquid-to-gas density ratio might appear, such as a steam leakage into hot lead-bismuth flow. It is still difficult to predict such phenomena because there are no available flow models for two-phase flow with large density ratio compared to ordinary two-phase flows such as an air-water two-phase flow. Therefore, two-phase flow model should be developed based on the experimental data of two-phase flows with large density ratio. In this study, a liquid metal two-phase flow was measured by using a four-sensor electrical conductivity probe and a miniature electro-magnetic probe in order to establish the experimental database for lead-bismuth flow structure. A lead bismuth eutectic (LBE) test loop was used to measure the flow structure and turbulent characteristics of the liquid-metal two-phase flow in a vertical pipe with a diameter of 50 mm and a length of 2,000 mm. Nitrogen gas was injected to the test section. So the density ratio between LBE and nitrogen was about 10,000, which is about 10 times larger than that of air-water two-phase flow. In the measurement with the four-sensor probe, the radial profiles of void fraction, bubble frequency, interfacial area concentration, gas velocity and bubble size were measured at different axial positions. The results show that the void fraction profile changes from wall-peak to core-peak with an increase of the superficial gas velocity at the present flow conditions. The void fraction could be predicted by one-dimensional drift-flux model. However, existing correlations overestimate interfacial area concentrations. Experiments were also performed to understand the turbulent structure in a liquid-metal two-phase flow by using the electro-magnetic probe. Radial profiles of the axial liquid velocity and turbulence intensity were measured. The turbulence intensity was increased as the superficial gas velocity increases. From the data measured by both four-sensor and electro-magnetic probes, it is shown that the turbulence intensity at the pipe center was proportional to the void fraction to the power of 0.8 for higher void fraction. This result represented the similar tendency as previous data in air-water two-phase flows.

Keywords
Lead bismuth eutectic, Two-phase flow, Four-sensor probe, Electro-magnetic probe

1a-17

Scenario study for the transition from thermal reactors to fast reactors in the world

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1. INTORODUCTION

Nuclear energy is expected to continue playing an important role in solution for covering the increasing demand for energy mainly in developing countries, while curbing greenhouse gas emissions. To achieve global sustainable energy future, deployment of fast reactor (FR) and its fuel cycle system and transition from thermal reactors will be significant. This paper introduces the results of evaluation carried out to explore the effects obtained by FR deployment in the world through several assumptions and scenarios using the comprehensive nuclear system evaluation method named nuclear supply chain model (SCM) developed by Japan Atomic Energy Agency (JAEA).

2. EVALUATION METHOD

SCM is a comprehensive evaluation system capable of evaluating key indicators, such as uranium resources, radioactive waste generation, non-proliferation, on long-term scenarios by simulating of nuclear supply chain from uranium supply to waste disposal. In this study, the world nuclear energy capacity was set up at 2500GW in 2100 and it will be continued at same level after the year and the share of heavy water reactor in the world was set at 6% in 2100. Moreover, it was assumed that three type world nuclear policies in homogeneous world, *i.e.* (a) directly disposal of spent fuels (OT), (b) reprocessing spent fuels and recycling plutonium in light water reactors (LWRs) (PT), and (c) reprocessing spent fuels and recycling plutonium in LWRs and FRs (FR). In case of (c), it was also assumed that the case of only 40% of the country in the world pursues recycling policy (FR40) in addition to heterogeneous world model. The capacity and the share were decided with reference to the scenario standardized by IAEA/INPRO/GAINS project, and the designs of FR were based on the Fast Reactor Cycle Technology Development Project (FaCT Project) Phase I by JAEA

3. Results

The result from the evaluation shows the plutonium recycling in LWR and LWR/FR are effective in reducing spent fuel stock piles, the disposal volume of high-level radioactive waste, plutonium inventories compared with the direct disposal of spent fuels. In addition, as more realistic approach, when only 40% of the country in the world pursues recycling policy, the results indicates the reduction of spent fuel stock piles, the disposal volume of high-level radioactive waste, plutonium inventories by plutonium recycling in LWR/FR although the effect is smaller than homogeneous world model. Therefore, this study concludes that plutonium recycling in LWR/FR will contribute to achieving global sustainable energy future, and JAEA will contribute to achieve them through R&D of FRs including MONJU and its fuel cycle while international cooperation.

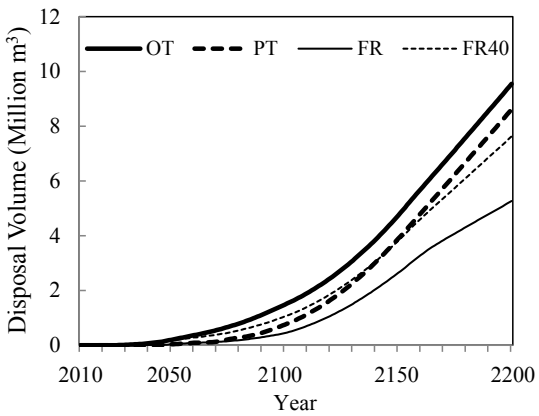

Fig. HLW disposal volume

Keywords

Fast Reactor Nuclear Fuel Cycle, Comprehensive Nuclear System Evaluation Method, Global Scenario Study

1a-18

Current Status and Future Prospect of MA Irradiation Tests in Monju

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I. Introduction

This paper overviews the current status and future prospect of the high-level radioactive wastes (HLWs) reduction technology by FBRs, especially using Monju. Homogeneous MA loading is assumed from the viewpoint of efficient power generation and not requiring additional specific fuel fabrication facilities.

II. Current status

Preproduction, characterization and material property measurements, and irradiation tests in Joyo have been performed for the MA (Np and Am) -bearing fuel pellets, as a basic research for future licensing in Monju. Deterioration of thermal conductivity and melting point, due to Am blending up to around 3wt%, has been found to be in the same order of Pu. Migration behavior of Am during irradiation has been identified to be nearly the same as Pu.

A useful finding on the reactor physics for MA (Am)-bearing fuel has been obtained from the System Start-up Tests (SST) in Monju. Monju restarted its operation in 2010, after 14-year's interruption. The core average content of ^{241}Am in the restart core was 1.5wt%, which accumulated by the ^{241}Pu decay and three times larger than the core before the interruption. Major reactor physics parameters, such as criticality, control rod worth, isothermal temperature coefficient, and feedback reactivity were measured. Precise evaluation of the core characteristics measurement and analysis has been performed. As a result, it was confirmed that the revision of the ^{241}Am cross section in JENDL-4.0 nuclear data library provides consistent results between the cores before and after the interruption. This implies the adequacy of the revision for the ^{241}Am cross section from JENDL-3.3 to JENDL-4.0 libraries, which contributes to the precise prediction of the MA (Am) incineration ratio.

III. Future prospect

The development of the radioactive toxicity reduction technology of HLWs has been identified as one of the most important missions for Monju at the Working Group on Monju Research Plan, established in the Japanese government. The GACID (Global Actinide Cycle International Demonstration) Project, which is aiming at conducting MA-bearing fuel irradiation tests in Monju, from on a pin-scale to a bundle scale under collaboration of CEA, DOE and JAEA, has been also positioned as one of the major activities to be conducted in the future Monju.

The material property measurements and irradiation behavior modeling, based on the Joyo irradiation test, is ongoing for the licensing in Monju. The fuel fabrication, transportation, irradiation and post-irradiation examinations are scheduled after the licensing. The first irradiation test is scheduled from the 5th cycle of the operation after restarting Monju. Irradiation tests with Cm-bearing fuel and on a bundle-scale are the future plan to be implemented.

Preliminary tests are also being planned prior to the GACID Project. Some of the current driver fuels, which contains ^{241}Am up to around 2 wt% at the maximum, are planned to be examined after irradiation. This amount of Am content can be recognized as significant, when compared with the MA content of 1.1 to 1.2 wt% for the future multiple-recycled fuel in the FBR system. Moreover a pin-scale MA-bearing fuel irradiation test is being planned to be started from the 2nd cycle for an early start of irradiation under the current licensing. These preliminary tests will also contribute to the MA-baring fuel technology development. Early restart of Monju is essential to complete its missions at the earliest occasion.

Keywords
Monju, GACID, Fast reactor

1a-19

Study on Actinide Burning by Fast Reactor

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After the accident at Fukushima Daiichi Nuclear Power Plant, re-examination of national energy strategy including nuclear energy has proceeded in Japan. Additionally, there are much more interests about the management of radioactive waste from the nuclear power generation.

Regarding this situation, the Fast Breeder Reactor (FBR) with its fuel cycle system has a possibility of playing the following important roles:

- Acquiring domestic energy resources is still an important issue for Japan which currently depends on most of energy resources from abroad. Even if the power generating capacity of nuclear energy is reduced, it seems important to keep an option of the sustainable energy resource by the plutonium (Pu) recycling in FBR cycle system.

- The FBR cycle system can also be utilized for the management of actinide nuclides (i.e. Pu and minor actinides (MAs)) in the spent fuel. The core design flexibility enables either breeding or burning of Pu, depending on the needs. (In the latter case, the reactor is referred to as Fast Reactor (FR).) MAs can be transmuted and eventually burnt by fission in both FBR and FR systems.

Under the above-mentioned background, “the study for reducing the quantity and toxicity of radioactive waste” has also recently been discussed in relation to the experimental planning of the prototype FBR “Monju.” Based on the accumulated knowledge of the conventional FBR research and development, the study on actinide burning FR has been performed in JAEA.

This paper describes the present status of the actinide burning FR core design study in JAEA. The basic characteristics in reactor physics are compared between FBR and FR. Based on the result of parametric survey calculations on FR core specifications, the relationships between core performances and Pu/MA transmutation characteristics are discussed. The foreseen experimental program using “Monju” etc. in support of the actinide burning FR core design is also explained briefly.

Keywords
Plutonium, Minor actinide, Transmutation, Fast reactor, Core design

1a-20

Neutronic analysis and transmutation performance of Th-based Molten Salt Fuels

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Thorium fuel and thorium fuel cycles are attractive for the long-term nuclear energy production with low radioactive waste. The utilization of thorium in various reactors ranging from thermal to fast neutron energies has been successfully demonstrated, but substantial R&D is required before commercialization becomes possible. The molten-salt reactor (MSR) systems present a very special feature of a liquid fuel. MSR concepts, which can be used as efficient burners of TRU from spent LWR fuel, have also a breeding capability in any kind of neutron spectrum (from thermal to fast), when using the thorium or fast spectrum U-Pu fuel cycle. In both options, they have an interesting potential for the minimization of radiotoxic nuclear waste. We have worked on the transmutation performance of a thorium-based Accelerator Driven System with Molten Salt Fuels. In this study, we investigate characteristics of a subcritical ADS with Molten Salt Fuels by considering three types of fuels based on Thorium and compare the results on neutronic analysis, transmutation of Minor Actinides, and breeding ratios, etc. The transmutation potential of different Molten Salt fuels, the breeding potential for Th-based fuels, and the safety characteristics of such systems are discussed.

Keywords
Accelerator Driven System, Thorium, Molten Salt Fuels, Transmutation

1b-1

Transmutation Scenarios after Closing Nuclear Power Plants

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When we think of the end of the nuclear power generation in the future of a certain country, there will be considerable amount of nuclear material that will be no longer burnable in the commercial nuclear power reactors. Some countries may stop nuclear power generation after they operate only the light water reactor (LWR) with the use of the uranium fuel. They will have depleted uranium, plutonium (Pu) and small amount of minor actinides (MA) in the spent fuel. Others may stop after introducing a fleet of the fast breeder reactor (FBR), and they will have larger amount of Pu and MA in the FBR spent fuel than those in the LWR one. When they stop the nuclear power generation, they have to choose among directly-disposing of the spent fuels, transporting them to the other country and burning them in the dedicated transmutation systems. The first choice might be superior in economics aspect, but difficult to find and construct the repository owing to large size and possibility to recover Pu for nuclear weapon. The second one is reasonable if neighbouring countries need them, but is not available for the last country that stops the nuclear power generation. The third one is costly but effective to minimize the burden of repository.

In the transmutation strategy, Pu has to be transmuted prior to MA because of its amount and usability as the nuclear weapon. For the country without the FBR technology, Pu must be transmuted in the LWR dedicated for the transmutation in forms of the MOX fuel or the ROX fuel. The ROX fuel that stands for “rock-like oxide” fuel contains mainly Pu with less amount of ^{238}U to reduce Pu production and to transmute about 90% of fissile Pu. The spent ROX fuel is so stable to confine radio-nuclides and it is difficult to recover Pu after disposal. Both transmutation methods in the LWR can reduce the emission of radio-nuclides to the environment and the opportunity to use Pu in the repository as the weapon. However, the potential radio-toxicity is not reduced due to remaining Pu and MA. If the transmutation systems such as fast reactor or accelerator-driven system are introduced in the nuclear fuel cycle, very small amount of Pu and MA are contained in the high-level waste, and, the potential radio-toxicity is reduced by one to two orders of magnitude.

In the present study, above mentioned transmutation methods were investigated quantitatively in terms of necessary number of transmutation systems and period after closing the nuclear power with the use of the analysis code for nuclear material flow. Moreover, the comparison of repository size and its potential radio-toxicity were made along the transmutation scenarios.

Keywords
Transmutation, Scenario study, ADS

1b-2

Principle Options of Fuel Debris Criticality Control in Fukushima Daiichi Reactors

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Japan Atomic Energy Agency

Since March 2011, Japan Atomic Energy Agency (JAEA) has been conducting studies on criticality control issues of the fuel debris in the damaged reactors of the Fukushima Daiichi site. The studies are both qualitative, referring the existing criticality control principles for normal nuclear facilities, and quantitative, computing expected criticality characteristics of the fuel debris. The current situation of the reactors is also considered in the study.

In normal nuclear facilities, the goal of criticality control is to secure a subcritical condition of fissile material. The goal is achieved by regulating composition, mass, and/or geometry of fissile material. In the accident of Three Mile Island Unit 2 reactor (TMI-2), heavily damaged and melted fuel assemblies formed fuel debris of a large amount whose geometry is uncertain, whose mass is larger than the minimum critical mass, and whose composition is unknown except the possible highest ^{235}U enrichment. Moreover, the fuel debris had to be cooled and shielded by water. To overcome this highly uncertain situation, the coolant water was borated with a concentration more than 4350 ppm that was enough to secure the subcritical condition.

The situation of Fukushima Daiichi reactors is severer than TMI-2. Most major difference from the viewpoint of criticality control is that the coolant water flow is practically "once through". Boron must be endlessly added in the coolant water to keep an appropriate concentration, which is not feasible. The water is not borated relying on the circumstantial evidence that the xenon gas monitoring in the containment vessels does not show a sign of criticality. Although the fuel debris will not be touched for a while, its condition may change due to gradual drop of its temperature or geometry change by aftershocks. The condition can be intentionally changed when the retrieval work of fuel debris starts. Such condition change may lead criticality of the fuel debris.

To avoid the criticality, or the environmental impact due to the criticality, a certain principle of criticality control must be established. There may be several options: prevention of criticality by the coolant water boration, prevention of criticality by the neutronic monitoring, prevention of environmental impact due to criticality, etc. Every option has both merits and demerits.

It is necessary to understand the actual condition of the fuel debris toward its certain criticality control by choosing an appropriate principle from those options. Observation, sample taking and analysis of the fuel debris must be conducted adequately.

Keywords
Fukushima Daiichi, Fuel debris, Criticality control

1b-3

ADS Research Activities in Sungkyunkwan University

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We present research activities on the Thorium based Accelerator Driven System (ADS) conducted in the Department of Energy Science at Sungkyunkwan University. The activities include computational modeling, partitioning and transmutation, reactor core design, and life cycle assessment analysis. For the computational modeling, we have been developing nuclear data library based on the ENDF, which is a new prototype of a nuclear data manipulation package (TNudy) based on the ROOT system (http://root.cern.ch).. An accelerator-based production of radioisotopes and transmutation has also been studied based on the concept of the TARC (Transmutation by Adiabatic Resonance Crossing) which is a neutron converter target. Thermodynamic analyses of a multi-channel helium-cooled device are performed with the Computational Fluid Dynamics (CFD) code CFX. Designs of compact reactor cores, with and without MA fuel elements have been studied. A new protocol has been developed to recover ^{90}Sr from irradiated ^{232}Th. As a strategy for the management of high level nuclear waste actinide partitioning studies have been conducted using 'Green' extractants. For the purpose of assessing the possible acceptance of the ADS technology by our society, we have been conducting Life Cycle Assessment analyses for estimating the cost of the conventional and modified fuel cycles of nuclear power. The presentation will describe the current activities in these research fields.

Keywords
ADS, Thorium, Nuclear Data, Partitioning & Transmutation, Life Cycle Assessment

1c-1

Nuclear transmutation of long-lived nuclides with laser Compton scattering : Quantitative analysis by theoretical approach

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One of the major problems of fuel cycle is the disposal of high-level radioactive waste which contains long-lived nuclides such as ^{129}I and high-decay heating nuclides such as ^{137}Cs. After the severe accident of Fukushima Daiichi Nuclear Power Plant, there is also a problem of ^{137}Cs having been concentrated by treatment of contaminated water. Transmuting such nuclides into short-lived or stable nuclides is one possible way to resolve this problem. Neutron capture reactions have been proposed for transmutations of such fission products. However, neutron capture cross sections differ significantly from nuclide to nuclide, and this transmutation method is not effective for nuclides with small neutron capture cross sections such as ^{137}Cs.

Recently, the photo-neutron (γ,n) reactions with laser Compton scattering γ-rays has been suggested as an alternative method for nuclear transmutations. The main point of this transmutation is to use γ-rays generated by laser photons backscattered off GeV electrons as well as photonuclear reactions via electric giant dipole resonance (GDR), which in general has a large cross section. The GDR is a collective excitation of a nucleus which decays mainly by the emission of neutrons and its total cross section is a smooth function of mass number. Therefore, this method is expected to be effective for transmuting fission products regardless of isotopes.

In this paper, we investigate in detail the transmutation of ^{137}Cs which is one of high-decay heating nuclides and has a small neutron capture cross section. We calculate (γ, γ), (γ,n) and (γ,2n) reaction cross sections for ^{137}Cs, ^{136}Cs, ^{135}Cs based on the Hauser-Feshbach theory using the TALYS code and evaluate quantitatively the effectiveness of this transmutation method. Since the (γ,2n) reaction transmutes ^{137}Cs ($T_{1/2}$= 30 y) into ^{135}Cs ($T_{1/2}$= 2.3×10^{6} y), we optimized the γ-ray energy for the transmutation to control the (γ,2n) reaction.

Figure 1(a) shows the number of ^{137}Cs as a function of irradiation time under various photon fluxes. The transmutation of ^{137}Cs is effective for the photon flux over 10^{18}/s ; irradiation with photon flux 10^{18}/s for 24 hours results in 10% reduction of the initial amount of ^{137}Cs. However, achievable photon flux is 2×10^{12}/s at present maximum accelerator systems where the reduction rate of ^{137}Cs is two orders of magnitude smaller than the natural decay rate of ^{137}Cs as shown in Figure 1(b). Although the transmutation of ^{137}Cs with laser Compton scattering is theoretically possible, it is actually difficult in the present condition.

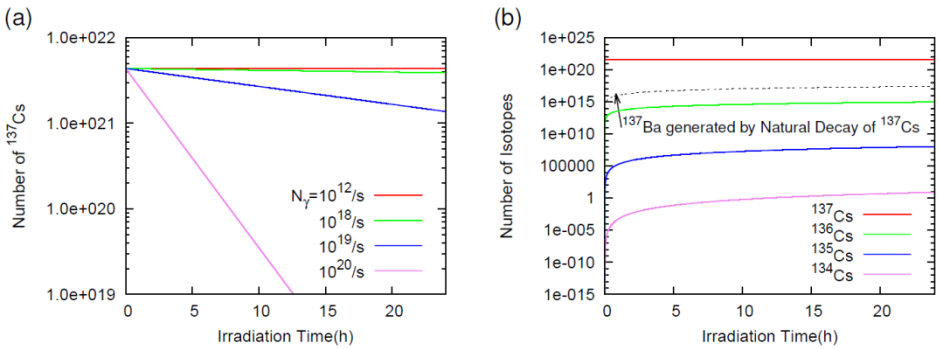

Fig.1 (a) Dependence of the photon flux N_γ $=10^{12}$, 10^{18}, 10^{19}, 10^{20}/s on the reduction of ^{137}Cs of 1g as a function of irradiation time. (b) The numbers of isotopes when ^{137}Cs of 1g is irradiated with photon flux 2×10^{12}/s.

Keywords
Transmutation, Photo-neutron reaction, Laser Compton scattering

1c-2

Sensitivity analyses of initial compositions and cross sections for activation products of in-core structure materials

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Improving the accuracy of predicting concentrations of activation products is important in the field of nuclear back-end for various evaluations. A quantitative investigation about the elements and the nuclear reactions leading to generation of activation products is of much help to achieve it. To clarify quantitatively the source elements and the nuclear reactions dominating generation of activation products, sensitivity analyses of initial compositions and cross sections were conducted.

In the analyses, ORIGEN2.2 code was used with ORLIBJ40, which is a set of the 1-group cross section libraries based on JENDL-4.0. A sensitivity coefficient is defined as the ratio of the amount of the concentration change of the target activation product to the amount of the change of an initial composition or a cross section. Activations of cladding tubes, end-plugs and spacers of fuel assemblies and channel boxes in BWR, whose materials are zirconium alloy, SUS304 stainless steel, and nickel-chromium-based alloy, were analyzed. The typical BWR irradiation history and flux level were applied in these analyses. From about 50 representative radioactive nuclides, several nuclides were selected as the targets of sensitivity analyses because of their large concentrations in the target materials.

The results clarified quantitatively the source elements and the nuclear reactions dominating generation of activation products. It was remarkable that the dominant generation pathways were clarified even for the nuclides generated through complicated pathways. These examples are shown in Figure 1 and Figure 2. In conclusion, the results of sensitivity analyses could be utilized to select the objective of elements for measurements of impurities in the materials and of nuclear data for improvement of accuracy.

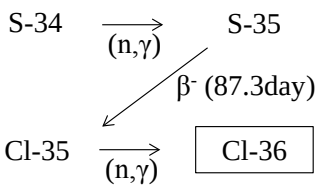

Figure 1 An example of an activation product generated through two nuclear reactions and a nuclear decay (Cl-36 in SUS304).

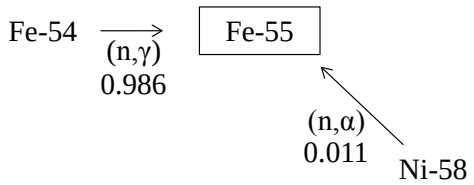

Figure 2 An example of an activation product with two contributing generation pathways (Fe-55 in SUS304). The values in the figure show the sensitivity coefficients of the cross sections.

Keywords
activation products, sensitivity study, burnup calculation, ORIGEN2.2, ORLIBJ40

1c-3

Thermophysical properties of thorium oxide

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Thorium oxide (ThO_2) has been considered as a nuclear fuel for various reactors, such as light water reactors, heavy water reactors, high-temperature gas-cooled reactors, and accelerator-driven systems. Additionally, ThO_2 is a matrix candidate for plutonium incineration. Numerous studies have been undertaken on the physical properties of $(Th,U,Pu)O_2$ as loading fuels of initial state. FP behavior has also been evaluated through the synthesis and observation of ThO_2-based SIMFUEL. However, the effects on the thermopysical properties have been rarely evaluated. In the present study, high-density pellets of FP element–containing ThO_2 were prepared using the Spark plasma sintering (SPS) method, and the effects on lattice parameter and thermal conductivity were investigated. These effects, including the oxygen defect, were quantitatively evaluated and are applicable to adiabatic FP elements.

The powder of $Th_{1-x}M_xO_{2-y}$ (M=Y, La, Ce, Nd, Gd, U; x=0.05, 0.10) were synthesized by solid-state reaction. The raw powders were mixed by wet ball-milling for 12 h and then cold pressed and heated at 1873 K for 72 h in a 4% hydrogen/argon gas flow. The powders were then ball-milled for another 12 h, packed to graphite die, and sintered in an SPS apparatus in a flow of argon gas. The sintering temperature and pressure were 1873 K and 50 MPa, respectively, for all the samples. The sintering temperature was controlled by the electric current. After the sintering, the samples were heat-treated to adjust the M/O ratio. The sample compositions are assumed be $Th_{1-x}M_xO_2$ for M=Ce, U, and $Th_{1-x}M_xO_{2-x/2}$ for M=Y, La, Nd, and Gd. This means that all Ce, Th, and U ions are tetravalent and other rare earth elements are trivalent in the samples.

The relative sample density reaches above 90% of the theoretical density. ThO_2 is known to be sintering-resistive, it generally requires a special granulation technique, sintering additives, and a high sintering temperature for fabrication of the high density pellets. Here, relatively high density pellets are obtained by applying the SPS technique at moderate sintering temperature and without any additive.The lattice parameter appears to change linearly with mol fraction of dopants for all the samples.Trivalent elements, such as Y, La, Nd, and Gd additions, significantly decrease the thermal conductivity near room temperature. This decrease is caused by the phonon-point defect scattering by the solute ions in addition to the scattering by the oxygen defect. Klemens and Callaway's model is used to estimate the lattice thermal conductivity for solid solutions. From the analysis, we can estimate the thermal conductivity of various ThO_2 that have rare earth elements added.

Keywords
Thorium oxide, FP, Thermophysical properties

1c-4

Characterization and thermophysical properties of $RE_2Zr_2O_7$ and $Nd_2Ce_2O_7$ precipitate in ThO_2-based fuel

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Thorium fuel cycle is contributed to reduce radioactive waste in the spent fuel because of the less long lived minor actinides yielding than Uranium-Plutonium fuel cycle. Additionally the fuel can be applied to accelerator driven system for nuclear transmutation. In spite of these merits, there are few reports on ThO_2-based fuel, especially focusing on the FPs chemical state and influences on fuel properties, as compared to UO_2. As for precipitate FPs in ThO_2-based SIMFUEL, pyrochlore-type compound $Nd_2(Zr,Ce)_2O_7$, which is not observed in irradiated UO_2 and $(U,Pu)O_2$ fuel, were detected. The existence of precipitates is possible to affect the fuel properties greatly, therefore, it is essential to comprehend the FPs behavior in nuclear fuel exactly. However, the physical properties and the precipitate mechanism are not elucidated completely.

In this study, the pyrochlore oxides $RE_2Zr_2O_7$ (RE=Y, La, Pr, Nd, Eu, Gd, Dy, Er) and $Nd_2Ce_2O_7$ were synthesized by solid-state reactions and pelletized by spark plasma sintering in order to make the high-density pellets. We examined the phase states and lattice parameters of them by means of X-ray diffraction and SEM/EDX analysis. The lattice parameters of $RE_2Zr_2O_7$ depended on the ionic radii of rare earth ion. The heat capacity, thermal conductivity and coefficient of linear thermal expansion were also measured. It was confirmed that the thermal conductivities were independent of temperature and significantly lower than those for ThO_2. The elastic constant and Vickers hardness was also investigated. In addition, to reveal the precipitate mechanism of pyrochlore-type compound, solid solutions between ThO_2 and Nd_2O_3, ZrO_2, were synthesized by solid-state reactions respectively and annealed under controlled oxygen potential, -60, -210,-260, -310 kJ. The phase states of the obtained samples were estimated by SEM/EDX analysis and lattice parameter.

Keywords
Thorium oxide, FP, Pyrochlore-type compound

1c-5

Overview of Studies on Lead-bismuth Technology in Tokyo Institute of Technology

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The early phase of experimental study using lead alloy in our laboratory was initiated by simulation of thermal-hydraulic phenomena in molten fuel-coolant thermal interaction. Wood's metal (48Bi-26Pb-13Sn-13Cd) was used for the experiment. It was performed from 1975 till 1980. One of the experiments was a vapor explosion test in the discharge of a jet of hot liquid metal into a cold water pool. Another one was a basic test of interfacial heat transfer and boiling in hot Wood's metal-water direct contact stratified flow in a horizontal duct. for the development of the accelerator driven system (ADS) and lead alloy-cooled fast reactors (LFR) Since 1999, basic experimental studies on the technology of lead-bismuth eutectic (LBE, 45Pb-55Bi) have been performed for the development of ADS and lead-bismuth-cooled fast reactor (LFR). Based on the studies of the LBE technology, an innovative concept of LBE, i. e. PBWFR (150MWe/450MWt) was formulated and proposed. The present paper overviews the basic studies on the LBE technology that have been performed in our laboratory. The subjects of the studies are classified into the following items:

(1) Compatibility of structural and cladding materials with LBE
- Corrosion of existing steels and steels containing Si and Al
- Corrosion of refractory metal-coated and Fe-Al alloy-surface-coated steels
- Corrosion of ceramic materials
- Corrosion of steels under bending and tensile stresses
- Corrosion of cold worked steels and welded steels
- Erosion of steel in LBE flow with low oxygen concentration

(2) Oxygen control oxygen sensor for LBE flow circuit
- Oxygen control using Ar-H_2-H_2O gas mixture and solid PbO
- Oxygen sensor with air reference and the reference of Bi and Bi oxide mixture

(3) Gas-LBE two-phase flows
- Natural circulation of LBE in single-phase flow
- 2-d Ar gas bubbly two-phase flow in LBE in a rectangular duct
- Visualization of N_2 gas-LBE bubbly two-phase flow with gamma ray radiography
- Ar-LBE and N_2 gas-LBE two-phase flow
- Removal of LBE particles in gas flow with Chevron dryer and electrostatic precipitator

(4) Phase changes in LBE systems
- Direct contact boiling two-phase flow of water in hot LBE flow
- Violent boiling in direct-contact of hot LBE droplet in cold water
- Freezing of lead-alloy on cold metals

(5) Property of LBE
- Diffusion coefficients of metal impurities in LBE

Keywords
Lead-bismuth eutectic, Corrosion, Oxygen Sensor, Thermal-hydraulics, Particle Flow, Diffusion Coefficient

1c-6

Evaluation of structural integrity of beam window for TEF target by thermal/fluid-structure interaction analysis

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Japan Atomic Energy Agency

Japan Atomic Energy Agency (JAEA) has been researching and developing an accelerator-driven system (ADS) as a dedicated system for the transmutation of long-lived radioactive nuclides. The ADS proposed by JAEA uses the lead-bismuth eutectic (LBE) alloy as a target material and a coolant. In the various R&D for ADS, construction of the Transmutation Experimental Facility (TEF) is planned under the framework of the J-PARC project as a preceding step before the construction of demonstrative ADS. In the development of the TEF, the estimation of the thermal-fluid properties of the flowing LBE is an important issue, because the LBE has the tendency to cause the corrosion/erosion to the structural materials. And the beam window of ADS, which separates the proton accelerator and the LBE subcritical core vessel, is exposed to the high temperature environment induced by the incidence of proton beam, and the embitterment of materials by the irradiation occurs. Therefore, the feasibility of the beam window is the most important factor for the realization of TEF. The objective of this study is to evaluate the feasibility of a designed beam window of TEF target by the numerical analysis with a 3D model. The analysis was performed by considering (1) the current density and shape of the proton beam, (2) the thermal-fluid behavior of LBE around the beam window as a function of the flow rate and inlet temperature, (3) the material and the thickness of the beam window, (4) the structural strength of the beam window.

In the reference case, the current density and the profile of the proton beam were 20 μA/cm^2 and a Gaussian shape, respectively. The flow rate of LBE and temperature at the inlet were 1 ℓ /sec and 350 ºC. The material of a beam window was type 316 stainless steel with the 2 mm thick. In this case, the maximum velocity of LBE and the maximum temperature located at the top of the beam window were about 1.2 m/sec and 477 ºC. By increasing the flow rate of LBE up to 4 ℓ /sec, the maximum temperature of a beam window was reduced around 420 ºC. The maximum shear stress was 190 MPa, which was observed at the center on the outside surface of a beam window. The value of analyzed stress in the reference case was lower than the tolerance level of the stress strength of the material given by the legal limitation which is applied to the fast reactor, and hence the feasibility of a designed beam window was confirmed.

Keywords
Accelerator-Driven System (ADS), Lead Bismuth, Beam window, Star-CD, ABAQUS,

1c-7

Fission cross-section measurement of minor actinides using a lead slowing-down neutron spectrometer KULS at KUR

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Management of minor actinides (MAs) and long-lived fission products is one of the most important problems for the use of nuclear power. Accurate data for neutron-induced reactions are required for nuclear transmutation technique of minor actinides using an accelerator-driven subcritical system or other innovative reactors.

We have performed a cross-section measurement of fission of MAs (Np, Am and Cm isotopes) using a lead slowing-down neutron spectrometer (KULS) at Kyoto University Research Reactor Institute. KULS is a neutron spectrometer assembled with lead blocks (1.5x1.5x1.5 m^3) with a neutron production target in the center. Neutrons produced by irradiating the target with electron beams are slowed down in the lead material and finally reach to a sample placed in a hole provided in KULS. The energy of the slowed neutron is determined by a relation $E_n = K/(t - t_0)^2$, where K is a slowing-down parameter and calibrated as 178 keVμs^2. Because of a short distance between the production target and the sample, typically 40 cm, KULS gives a high flux neutron field and enables us to measure a small cross-section with small amount of sample.

The fission cross-sections were measured using a multi-layered parallel plate avalanche counter [1] in which two samples can be set back-to-back. Fission fragments from a sample of MA and a ^{235}U sample were simultaneously detected. Then, the fission cross-section of MA can be obtained relatively to that of ^{235}U. In Fig.1, the results for ^{241}Am obtained in the present work is compared with the other works [2,3]. The solid line is the evaluation of JENDL-4.0 [4] which is broadened according to the neutron energy resolution of KULS.

In the symposium, the results for ^{237}Np, $^{241, 242m, 243}$Am and $^{245, 248}$Cm will be presented.

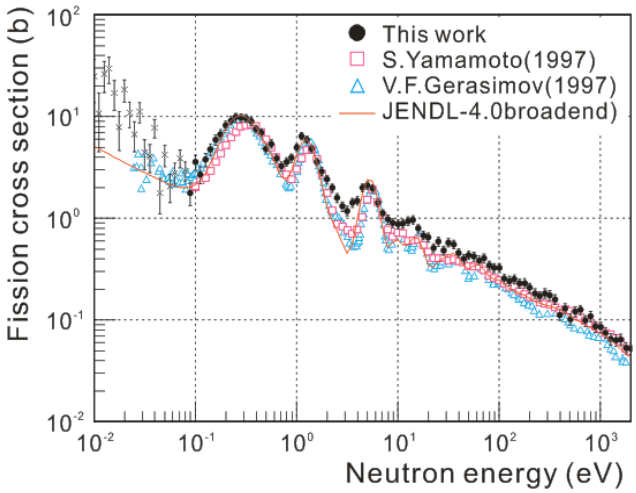

Fig. 1 The cross-section of ^{241}Am(n,f) obtained in the present work.

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Keywords
Minor actinides, fission cross section, lead slowing-down neutron spectrometer

1c-8

Development of TRU nitride database for designing ADS fuel

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Nuclear Science and Engineering Directorate, Japan Atomic Energy Agency

Spent fuels arising from nuclear power plant contain significant amounts of long-lived minor actinides (MA: Np, Am, Cm) and thus, proper management of MA is one of the key issues for sustainable nuclear energy use. Nuclear transmutation technology using sub-critical Accelerator Driven System (ADS) potentially offers a promising option to overcome the concern in the future nuclear fuel cycle [1]. Nitride solid solutions containing MA and Pu, hereafter called as transuranium (TRU) nitrides, are proposed as a candidate material of ADS fuel, in which TRU nitrides are used by diluting with ZrN. It is necessary to develop the property database of nitrides for designing ADS fuel. Experimental data or estimated values of extensive properties for TRU nitrides are being summarized in Japan Atomic Energy Agency (JAEA). Also, from a view point of practicality for fuel design, the properties are being formulated, in which error evaluation is also made as much as possible.

Several important properties on TRU nitrides have been measured recently in JAEA. Nevertheless, some property data for TRU nitrides are still often lacking. In our database, the properties of uranium nitride, UN, and uranium-plutonium mixed nitride, (U,Pu)N, are substitutionally given in such cases. The data for UN are mostly cited from the efforts in USA [2-7], in which UN had been examined as a potential fuel for space reactors. The data for (U,Pu)N are mostly cited from two distinguished monographs by Matzke [8] and Blank [9] of Institute for Transuranium Elements (ITU), in which (U,Pu)N as well as (U,Pu)C had been investigated in detail as an advance fuel for fast reactors. Besides, a few property data on TRU nitrides had been collected under the Advanced Accelerator Assisted (AAA) and the Advance Fuel Cycle Initiative (AFCI) programs of USA [10-11].

At present, most of the thermophysical properties on TRU nitrides, such as thermal conductivity [12] and lattice thermal expansion [13], have been available with enough accuracy for practical use in fuel design of ADS. On the other hand, most of the mechanical properties and irradiation behavior, which are still limited to those of UN and (U,Pu)N, should be examined hereafter by experiments or reliable simulations. Further, variation in properties with burnup progressing also should be evaluated, because high burnup (~20 at.%) is necessary for TRU-transmutation in ADS.

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Keywords
Accelerator-Driven-System (ADS), TRU nitride database, Thermophysical property

1c-9

Recent progress in research and development on the neutron resonance densitometry (NRD) for quantification of nuclear materials in particle-like debris

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Quantifying nuclear materials (NM) in stock of debris of melted fuel (MF) formed in a severe accident is considered to be difficult because of their variety of size, shape, unknown compositions and their strong radio activity. Although, techniques of non-destructive assay (NDA) are indispensable for evaluation of NM in debris, quantification methods have not been established so far. In the cases of TMI-2 or Chernobyl-4, nuclear material was accounted based on some estimation.

We have proposed a technique called neutron resonance densitometry (NRD) to quantify NM in particle-like debris of MF. We assume particle-like debris is produced in rapid cooling processes of severe accidents. Small pieces are alto produced when MF are cut or broken down to be taken out of the damaged reactors.

NRD is a method of a combination of neutron resonance transmission analysis (NRTA) and neutron resonance capture analysis (NRCA) or prompt gamma-ray analysis (PGA). In NRTA, neutron transmission rate is measured as a function of neutron energy with a time-of-flight (TOF) technique. Characteristic neutron absorption peaks of Pu and U isotopes are observed in the neutron energy range of 1-50 eV. Measurements of these transmission spectra, therefore, can be carried out with a short-flight-path TOF system. The NRTA measurements are not interfered by strong gamma-ray radiation from MF samples. However, composing isotopes with large total cross-section (H, B, Cl, Fe, etc.) distort the obtained neutron absorption spectra. Accordingly, accurate measurement could not be performed. The quantity of these containing isotopes could not be determined by NRTA measurements, because these isotopes do not resonantly absorb neutrons in this energy range. To identify and quantity the composing isotopes, NRCA/PGA method is introduced. In NRCA/PGA, prompt-gamma rays are measured with a high resolution and S/N gamma detection system. Development of a special gamma-ray spectrometer for NRCA/PGA is in progress. Information from the NRCA/PGA is used to optimize the sample thickness and measurement time, and to support NRTA analysis.

In order to examine the NRD method, studies have started under the agreement between JAEA and EURATOM in the field of nuclear materials safeguards research and development. Experiments studying the effects of sample properties, such as target thickness, particle size, temperature and the achieved accuracy, are going on at the TOF-facility GELINA of EC-JRC-IRMM.

In this presentation, we briefly describe the concept of NRD, overview the research project of development of NRD, and explain the recent progress of the study. This research was supported by JSGO/MEXT.

Keywords

Neutron resonance densitometry, NRD, transmission, capture, Fukushima, severe accident, nuclear security

1c-10

Status of Studies and Future Subjects of Material Corrosion for Lead-bismuth Cooling System

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One of the most crucial issues in the development of lead alloy-cooled reactors and accelerator-driven transmutation system is the compatibility of structural materials with corrosive lead-bismuth eutectic at high temperature up to 650℃. In this paper, the current studies about static corrosion in lead-bismuth on various types of steel, in which different stresses are applied (heat treatments, strengths…) and under different experimental conditions (temperature, oxygen concentration…) are reviewed. Corrosion kinetics and mechanisms in lead-bismuth are explored. We first examine the importance of oxygen concentration and the different means of measure and control. We then discuss the influence of oxygen concentration and temperature on corrosion and the effect of stresses on the surface of corrosion. Current apparatus are described and existing experimental results are summarized. Finally, based on the present results, some recommendations for future experiments are proposed to complete the knowledge about corrosion in lead-bismuth coolant system before its industrial applications.

Keywords
Lead-bismuth eutectic, Corrosion, Oxygen control

1c-11

Mechanical properties of beam window materials for ADS irradiated in a spallation environment

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In several institutes, the research and development for an accelerator-driven transmutation system (ADS) to transmute minor actinide (MA) have been progressed. ADS is composed of a high-intensity proton accelerator, a spallation target and a sub-critical reactor core loaded with MA fuel. As a coolant and a target material, LBE (Led-bismuth eutectic) is selected. The spallation target is placed at the center of the sub-critical reactor core. The proton beam accelerated at the high-intensity accelerator is injected into the target. The sub-critical core is driven by the spallation neutrons produced in the target and MA is transmuted to short-lived isotopes. The bean window of ADS will be used as the boundary between the proton accelerator and the reactor core. The beam window submerged in the reactor core will be subjected to high-energy proton and spallation neutron irradiation. To evaluate the lifetime of the beam window, post irradiation examination (PIE) of the STIP (SINQ target irradiation program, SINQ; Swiss spallation neutron source) specimens has been carried out.

The specimens tested in this study were made from the austenitic steel JPCA (Japan Primary Candidate Alloy). JPCA is a modified austenitic stainless steel by the addition of Ti to AISI type 316 SS in order to improve swelling resistance. The specimens were irradiated at SINQ Target 4 (STIP-II) with high-energy protons and spallation neutrons. The irradiation conditions were as follows: the proton energy was 580 MeV, irradiation temperatures ranged from 100 to 430℃and displacement damage levels ranged from 7.0 to 19.5 dpa. Tensile tests were performed in air at room temperature (R.T.), 250℃and 350℃. Bend-fatigue tests were performed in air at room temperature under deflection control mode. The wave form of the control signal was a sine curve with a frequency of 26 Hz. Fracture surface observation after the tests was done by SEM.

Results of the tensile tests performed at R.T. showed the extra hardening of JPCA at higher dose compared to the fission neutron irradiated data. At the higher temperatures, 250℃and 350℃, the extra hardening was not observed. Degradation of ductility bottomed around 10 dpa, and specimens kept their ductility until 19.5 dpa. All specimens fractured in ductile manner. Results of the bend-fatigue tests showed that the numbers of cycles to failure (Nf) were not changed by irradiation. Dpa dependence of Nf was not clearly seen in the irradiation conditions. In spallation environment, He atom production ratio is very high and most of He atoms are retained in the materials. In spite of large number of He atoms, all JPCA specimens show transgranular fracture surface.

Keywords
ADS, Beam window, Austenitic steel, JPCA, Proton irradiation, Tensile property, Bend-fatigue property

1c-12

Development of non-destructive assay to fuel debris of Fukushima Daiichi NPP (1) Experimental validation for the application of a self-indication method

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It is surmised that melted fuel debris are existing in the core at units 1, 2 and 3 of Fukushima Daiichi NPP. Identifying the fuel debris status in the reactors is one of the most important issues for decommissioning. Therefore, we need to study how to analyze the properties of actual debris collected in advance of removal work. As debris contain melted fuel, cladding tube and structure materials in addition to a salt content heterogeneously, non-destructive assaying the distribution of nuclear material in debris is absolutely essential for nuclear material accountancy and critical safety.

Neutron Resonance Densitometry (NRD) based on Neutron Resonance Transmission Analysis (NRTA) and Neutron Resonance Capture Analysis (NRCA) with the Time-of-Flight (TOF) technique is a promising way to characterize debris. However, there are two difficulties to apply the method to fuel debris. In the NRTA, many resonances of other nuclides which are contained in debris may make it difficult to identify and quantify the target nuclide. In the NRCA, it is expected that the intense decayed gamma rays from debris give rise to high background and large dead time of gamma-ray detector. In this work, we proposed a new concept of “self-indication method” as a complementary assay to overcome those difficulties. In the self-indication method, we set an indicator consisted of target nuclide with a high purity beam-downstream from a sample. By detecting the reaction products such as neutron capture gamma rays or fission products from the indicator with the TOF method, the transmission neutron can be measured indirectly. The self-indicator is a transmission neutron detector which has a high efficiency around the objective neutron resonance energies of target nuclide, so that it enables us to quantify effectively the amount of resonance absorption of the target nuclide. Moreover, it is hard to be affected by the decayed gamma rays from the debris.

In order to verify the method, we performed the experiments using a 46-MeV electron linear accelerator at the Kyoto University Research Reactor Institute. A sample and an indicator were set at distances of 11.0 and 12.7 m from the neutron source, respectively. Pulsed neutron beam was well-collimated to 24 mm in diameter. The indicator was surrounded by a detection assembly $Bi_{12}Ge_3O_{12}$ (BGO) scintillators, which consists of 12 scintillation bricks of each $5\times5\times7.5$ cm^3. At first, we measured the densities of gold foils with 10, 20, 30, 40, 50 μm in thickness by area analysis at the 4.9-eV resonance region. A ^{10}B plug with 8 mm in thickness or a gold foil with 50 μm in thickness was used as an indicator. Since the former thick indicator can absorb almost neutrons with energies below epi-thermal region, it was equal to the conventional NRTA. In the latter case, it was the self-indication measurement. It was confirmed that the densities of the target nuclide can be determined by both methods within 3% accuracy. At the next step, we added a silver foil with 50 μm to a gold foil with 10 μm as a sample. It is worth noting that silver has a large resonance at 5.0 eV close by 4.9-eV resonance of ^{197}Au. It was experimentally shown that the contribution from the other nuclide can be remarkably suppressed by applying the self-indication method. Finally, we have demonstrated non-destructive assay to nuclear material using a mixture composed of natural uranium foils, sealed minor actinide samples of ^{237}Np and ^{243}Am. In the symposium, the results of those experiments will be presented.

This work is supported by JSPS KAKENHI 24760714.

Keywords
fuel debris, pulsed-neutron source, TOF, self-indication method, neutron resonance absorption

1c-13

Development of non-destructive assay to fuel debris of Fukushima Daiichi NPP (2) Numerical validation for the application of a self-indication method

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Kyoto University Research Reactor Institute

In order to perform decommissioning of Fukushima daiitchi NPP safely, it is very important to measure the component of fuel debris. Therefore, a new non-destructive assay to identify and quantify the target nuclide in fuel debris using a pulsed neutron source is under development in Kyoto University Research Reactor Institute.

We use the self-indication method for the non-destructive assay. This method is one of neutron transmission method. The neutron transmission method is focused on resonance reactions (i.e. capture, fission) at the target nuclide. In the conventional neutron transmission method, a sample is irradiated by a pulsed neutron beam and the energy distribution of transmitted neutrons from the sample is measured by the Time-of-Flight technique. Then, the target nuclide in the sample is identified and quantified by using transmitted neutrons in resonance energy region. This is a remarkably effective method to identify and quantify the target nuclide. However, if the energy spectrum of transmitted neutron has many dips caused by resonance reaction of other nuclides, it is difficult to identify and quantify the target nuclide in the sample.

In the self-indication method, the transmitted neutrons from the sample are injected into an indicator. The indicator is consisted of a high purity target nuclide. The transmitted neutrons are obtained via resonance reactions in the indicator. The self-indication method has high signal to noise (S/N) ratio than the conventional method.

In this study, numerical validation for the self-indication method to identify and quantify nuclides in a BWR-MOX pellet is described. The burnup of MOX pellet is 0GWd/t, 10GWd/t, 20GWd/t, 30GWd/t, 40GWd/t and 50GWd/t. The 12m measurement line in KUR-LINAC is simulated as a calculational geometry. Numerical calculations are carried out by continuous energy Monte-Carlo code MVP2[1] with JENDL-4.0[2] as the nuclear data library. The burnup calculations of BWR-MOX pellet are performed by deterministic neutronics code SARC2006 [3] with JENDL-4.0.

For example, a numerical result to identify I-129 in the MOX pellet is described. The bunup of MOX pellet is 20GWd/t. I-129 has only 4 resonances in the energy region of 0.1eV to 100eV. The resonance peaks are 41eV, 73eV, 75eV and 97eV. The numerical result shows that the transmitted neutrons are easily obtained via I-129 resonance absorption reactions in the indicator.

Numerical validation for application of self-indication method is carried out. As the results, it is described that the self-indication method has good S/N than the neutron transmission method to quantify amount of target nuclides in the fuel debris.

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Keywords Self-indication method, Non-destructive assay, Resonance, MOX pellet, Numerical validation

1c-14

Precise Measurements of Neutron Capture Cross-Sections for LLFPs and MAs

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To evaluate the feasibility of development of nuclear transmutation technology and advanced nuclear system, precise nuclear data of neutron capture cross-sections for long-lived fission products (LLFPs) and minor actinides (MAs) are indispensable. Precise nuclear data for stable isotopes of LLFPs are also needed because of their relatively large fission yields.

In the poster session, we will present our research activities for the measurements of the neutron capture cross-sections for LLFPs and MAs.

Keywords

neutron capture cross-section, minor actinides, long-lived fission product

1c-15

Development of the Method to Assay Hardly Measurable Elements in Spent Nuclear Fuel and Application to BWR 9×9 fuel

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In fission products in used nuclear fuel, there are several stable isotopes which have large neutron absorption effect. For the evaluation of the neutronics characteristics of the nuclear reactor, amount of such isotopes should be evaluated by using burnup calculation codes. To confirm the correctness of such data obtained by calculation codes is important to assure the precision of the evaluation of the neutron multiplication factor of used nuclear fuel. However, it is known that there are several hardly measurable elements in such important fission products. The data of the amount of the hardly measurable elements in used nuclear fuel are scarce in the world.

Japan Atomic Energy Agency (JAEA) had been developed the method to assess the amount of fission products which are hardly measurable and have large neutron capture cross section, under the auspices of the Japan Nuclear Energy Safety Organization. In this development, the measurement method was developed, combining a simple and effective chemical separation scheme of fission products from used nuclear fuel and an Induced Coupled Plasma Mass Spectrometry with high-sensitivity and high-precision. This method was applied to the measurement program for used BWR 9×9 fuel assembly. This measurement method is applicable to the required measurement for the countermeasure to the accident of the Fukushima Dai-ichi Nuclear Power Plants of Tokyo Electric Power Company (TEPCO). JAEA has a measurement plan for not only BWR but also PWR fuel.

This presentation describes the measurement method developed in the study as well as the future measurement plan in JAEA.

Keywords
Post Irradiation Examinations, Fission Products, Isotopic Composition

1c-16

Evaluation of volatile FP elements behavior for severe accident analysis of molten salt reactor

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The recent interests on molten salt reactor stem from the very high operation temperature, the passive safety feature against unexpected core accident and possible applicability for nuclear transmutation of radioactive waste, while the retention of fission products (FPs) during the accident event is not well understood. Especially for the volatile FPs, such as iodine and cesium, the vapor pressure nor the chemical form of the vapor species during the accident condition have not been studied yet. Hence, the authors have started to study high temperature vaporization behaviors from the molten salt containing FPs under moisture and air.

First, the vapor pressures of compounds in gaseous phase in equilibrium with LiF-NaF-KF (46.5-11.5-42.0 mol%) molten salt containing 3.26 mol% of CsF and 1.0 mol% of CsI were calculated using a thermodynamic database "MALT for Windows". The ratio of CsF to CsI was determined according to the fission yield of ^{233}U. At the elevated temperature of 900K under 0.1MPa of argon gas containing neither air nor moisture, the vapor pressures of the dominant gaseous compounds containing the FP were calculated as 36 Pa for CsF and 27 Pa for CsI. In addition to these compounds, it was estimated that some other compounds also exhibit high vapor pressures under air. Actually, the vapor pressures in air without moisture were calculated as 78 Pa for $CsNO_3$, 30 Pa for I (iodine in atomic form) and 24 Pa for I_2, while those in air containing 1 mol% H_2O were 177 Pa for $CsNO_3$, 12 Pa for I and 4 Pa for I_2. Therefore, it was suggested that the vaporization of FPs from the molten fluoride salt is significantly affected by the ambient atmosphere condition.

According to the above estimation, experiments to measure the actual vaporized amounts of iodine and cesium from the molten fluoride salt have been planned. The molten fluoride salt containing CsF and CsI, whose composition is the same as described above, is to be loaded in a nickel boat. The nickel boat is placed in a 1-inch diameter tube made of nickel, in which the experimental gas flows. The nickel tube is to be horizontally placed in an electric furnace so that the inside of the tube can be heated up to 1100K. The vaporized material caused by the heating will flow to the downstream with the argon gas flow and be captured in scrubbers filled with water. The content of the vaporized material is to be estimated by analyzing the amount of the material collected in the scrubber solutions as well as in washed solutions of the apparatus, such as inside of the nickel tube and connecting tubes. By changing the experimental temperature (800 K though 1100 K), the experimental gas (pure argon, dry air, or wet air), and the gas flow rate (up to 200 sccm), the behaviors of volatile FPs on severe accidents of molten salt reactors, such as leakage of the molten fluoride salt, are to be be evaluated.

In addition, equilibrium vapor pressure measurement is to be undertaken by means of the Knudsen cell mass spectrometry to evaluate the thermodynamic activities of the volatile elements, Cs and I, in the pseudo-binary systems such as LiF-CsF and LiF-LiI. The obtained activity data will be applied to the analysis of the behaviors of volatile FPs as described above.

Part of this work was supported by Chubu Electric Power Company through "Nuclear Safety Research Laboratory-Research Grant".

Keywords
Molten salt reactor, Fission product, Vaporization, Molten fluoride salt,

2a-1

Environmental parameters for assessing the behavior of radiocarbon in the paddy soil-to-rice plant system

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Transuranic (TRU) waste contains a variety of radionuclides, e.g., Np, Pu and long-lived radionuclides like ^{14}C and ^{129}I. In Japan, TRU waste is categorized into four groups in accordance with physical properties and the concentration of radioactive materials. Group 2 waste of these groups includes hull and end piece wastes with relatively high amounts of ^{14}C, and leaching of low molecular weight ^{14}C-organic materials from simulated hull wastes has been reported. The ^{14}C-organic materials have very little sorption properties to clay and rock, and thus the possible migration of the ^{14}C from a TRU repository site to the biosphere through groundwater presents some concern. Indeed, potential wastes from the Fukushima Daiichi nuclear power plants expected to have ^{14}C. Therefore, it is necessary to understand the behavior of ^{14}C in the human habitation sphere for the safe disposal of radioactive wastes.

In our research project, soil-soil solution distribution coefficients (K_d) and soil-to-rice plant transfer factors (TFs) of ^{14}C have been determined to clarify the behavior of ^{14}C in Japanese rice paddy fields. In addition, ^{14}C partitioning in solid, liquid, and gas phases was determined. As a source of ^{14}C, ^{14}C-sodium acetate was used because the chemical form would be occurred the highest amounts among the potential ^{14}C-organic materials. These environmental parameters commonly used for mathematical dose assessment models.

The K_d values were obtained by tracer experiments and then calculated from activities of the ^{14}C directly measured in solid and liquid phases at the end of 7 days incubation period. From the results, the mean of K_d value was 140 ± 77 L Kg^{-1}. This value was higher than expected from its chemical form. The degradation of the ^{14}C-sodium acetate and the accumulation of the ^{14}C by soil microbes caused the relatively high value of K_d, and also the ^{14}C gas was emitted from the culture (approximately 60% of total added) due to the microbial activity.

TFs were determined by laboratory and field experiments. In the field experiments, TF of carbon was estimated by using stable carbon isotope ratios, and the average TF value was 0.11 ± 0.04 for the edible part. On the other hand, the average TF of ^{14}C, which was obtained by tracer experiments, was 6.8 ± 2.4; however, it seemed that the value was overestimated. In the tracer experiments, rice plants also cultivated in the soil without ^{14}C contamination as a negative control, and interestingly a portion of the ^{14}C added to other pots was detected in the negative control white rice. The results suggest that the ^{14}C gas, which was emitted from the contaminated soil as observed in the K_d experiments, was assimilated by the negative control plants through photosynthesis.

From these results, it was assumed that acetic acid as a ^{14}C source in irrigated paddy soils was rapidly incorporated and decomposed by soil microbes. Although the most of the ^{14}C was released into the air as gas forms, some of the ^{14}C was assimilated into the microbial cells. The main chemical species of ^{14}C gas released into the air was $^{14}CO_2$ according to our additional study, and a part of the $^{14}CO_2$ could be used by photosynthesis of the rice plants. Only a small amount of the ^{14}C was absorbed through the roots: however, the soil origin ^{14}C to the rice plants was detected. Human expose to ^{14}C through rice intake must be considered in the safety assessment of TRU waste disposal.

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Keywords
TRU waste, C-14, rice paddy, Kd, TF

2a-2

Development of ICP-MS based analytical method for the determination of radioactive Cs isotopes

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Radioactive Cs isotopes are high yield fission products up to 6.535% and 6.236% for ^{135}Cs and ^{137}Cs, respectively, from thermal neutron fission of ^{235}U. They can be released into the environment in nuclear events, such as the detonation of a nuclear weapon or nuclear reactor accident. In the fission chains for ^{135}Cs and ^{137}Cs, a shielding of ^{135}Cs occurs due to neutron capture of its precursor, ^{135}Xe, to form ^{136}Xe, whereas production of ^{137}Cs is unaffected. This process causes a high degree of variance of ^{135}Cs/^{137}Cs isotope ratio with source. Thus, the ratio of ^{135}Cs/^{137}Cs will be characteristic of the reactor operation and shutdown conditions. The determination of ^{135}Cs by radiometric methods is very difficult and quite insensitive because of its long half-life and low-energy β-decay. Neutron activation analysis (NAA) and thermal ionization mass spectrometer (TIMS) have been investigated for ^{135}Cs activity and ^{135}Cs/^{137}Cs isotope ratio analysis in environmental samples, but these techniques are expensive and not widely available. Recently, the analytical potential of ICP-MS for ^{135}Cs, ^{137}Cs activities and ^{135}Cs/^{137}Cs isotope ratio have been discussed in the studies of environmental radioactivity and nuclear forensics, but the analysis of environmental samples remains to be a great analytical challenge. In Japan, there has been no report on the analysis of ^{135}Cs/^{137}Cs isotope ratio in environmental samples since no method has been established for ^{135}Cs analysis in the MEXT environmental radioactivity analytical method.

Here we report a newly established ICP-MS based analytical method for the determination of ^{135}Cs, ^{137}Cs activities and ^{135}Cs/^{137}Cs isotope ratio in environment samples with an aim to obtain isotopic composition of radioactive Cs isotopes released from the Fukushima Daiichi nuclear power plant (FDNPP) accident. We propose that ^{135}Cs/^{137}Cs can be used as a powerful tracer for long-term estimation of the environmental mobility of the FDNPP accident related fission products since the currently used ^{134}Cs/^{137}Cs activity ratio will become unavailable in the future for identification of the FDNPP contamination due to the short half-life of ^{134}Cs. The developed analytical method can be applied to the analysis of radioactive wastes for the accurate radiation dose assessment by ^{135}Cs, and to nuclear fuel analysis to estimate nuclear fuel burn-up in the damaged reactors for reactor decommissioning in the future.

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Keywords
Radioactive wastes, Radioactive Cs isotopes, ICP-MS

2a-3

Development of rapid analytical method of ^{129}I in the contaminated water and woods arising at Fukushima Daiichi Nuclear Power Station

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Introduction

At the moment, contaminated water has been arising in the reactor and turbine buildings at Fukushima Daiichi Nuclear Power Station. The accumulated water processing facility was setup to decontaminate and to desalinate. The number of the tank storing the treated water has been increasing. Woods in the site were cut to get the space to set the tanks. These woods also became radioactive waste. Iodine-129 is one of the important nuclides of which radioactivity has to be evaluated for the safety disposal of the waste. Although I^- is considered major species of ^{129}I generated in the reactor, IO_3^- and I_2 are possibly generated depending on the condition. In addition to this, isotope exchange reaction between natural $^{127}IO_3^-$ and reactor generated $^{129}I^-$ could be taken place. Therefore, analytical condition to determine each species of I in a solution was investigated. Furthermore, conventional analytical method of ^{129}I in concrete was optimized to apply it to woods.

Experimental

$^{129}I^-$ (0.1 Bq) and $^{127}IO_3^-$ (1 μg) were added into 3 M NaOH solution or HCl solution (pH=2) with and without reductant ($NaHSO_3$) to study the isotope exchange reaction and behavior of these iodine species in the analysis using Anion SR. The operation of Anion SR was based on reference [1].

Known amount of I^- or IO_3^- was added into the bark of pine, representative of woody material taken at the establishment of Japan Atomic Energy Agency, and it was put in a wet oxygen gas line set in an electric furnace. The rate of temperature increase was optimized for woody sample. The vaporized I was trapped in an alkaline solution. Because insoluble organic material was deposited in the gas line and the trap, oxidant (hopcalite II) was set in a downstream of the sample to decompose it. Iodine was measured by Inductively Coupled Plasma Mass Spectrometry measurement with Dynamic Reaction Cell.

Results and discussion

Table 1 shows recovery of I from examined solutions. From 3 M NaOH solution, ^{129}I was recovered and ^{127}I wasn't recovered regardless the existence of the reductant. That is, ^{127}I and ^{129}I showed different behavior. This result supports that I^- is extracted and IO_3^- isn't extracted by the Anion SR, Reductant doesn't work, and reaction between $^{127}IO_3^-$ and $^{129}I^-$ is negligible. On the other hands, ^{129}I and ^{127}I were recovered with the reductant and were not recovered without the reductant from the HCl solution. Similar behavior of ^{127}I and ^{129}I was observed in this case. This implies that IO_3^- is dominant species in HCl solution at pH 2, IO_3^- is reduced to I^- by the reductant in this experimental condition. Since the oxidation process from I^- to IO_3^- is including the reaction with I^-, I_2 and IO_3^-, isotope exchange would be taken place. Consequently, both of I^- and IO_3^- are analyzed at pH 2 condition and I^- is analyzed at 3 M NaOH condition. Accordingly, speciation of I^- and IO_3^- is available.

Table 1 recovery of I (%)

diluent	Without $NaHSO_3$		With $NaHSO_3$	
	^{127}I	^{129}I	^{127}I	^{129}I
3 M NaOH	2.3±1.9	71±2.9	2.1±0.2	86±4.3
HCl (pH2)	8.6±6.3	19±4.8	64±7.3	60±3.1

For woody material, approximately 90% recovery was obtained for both of I^- and IO_3^- in the alkaline trap of combustion method. Anomalous combustion was avoided by the stepwise and slow rate of temperature increase.

[1] Y. Kameo, *et al.*, JAEA-Data/Code, (2010) 2010-028

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Keywords ^{129}I, Accumulated Water, Anion SR

2a-4

Determination of low-level technetium-99 in the environmental samples by inductively coupled plasma mass spectrometry

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Technetium-99 (half-life: 2.11×10^5 y) is a pure beta emitter and is produced in the fission of ^{235}U and ^{239}Pu with a relatively high fission yield of ca. 6% like ^{137}Cs. The amount of ^{99}Tc has been increasing due to the operation of nuclear power plants. For waste management of spent nuclear fuels and high level radioactive wastes after reprocessing, it is necessary to consider the fate of this radionuclide in the environment. Indeed, internationally, the long-lived isotopes ^{129}I, ^{79}Se and ^{99}Tc have been identified as key radionuclides for waste repositories. The dominant Tc species in natural aqueous solution in equilibrium with the atmosphere is the highly soluble chemical form pertechnetate, TcO_4^-. In the terrestrial environment, $^{99}TcO_4^-$ has a high geochemical mobility and availability for plants. We have also carried out $^{99}TcO_4^-$ uptake study by plant roots grew in hydroculture, and found fairly high Tc bioavailability (unpublished data). However, it was not clear whether ^{99}Tc was also mobile in the actual environment or not.

To understand the fate in the environment, global-fallout ^{99}Tc can be a good tracer. However, the concentrations were in low levels in environmental samples, therefore, it was necessary to develop more sensitive methods. In this study, we focused on the use of inductively coupled plasma mass spectrometry (ICP-MS) to measure ^{99}Tc. The important points to measure ^{99}Tc by ICP-MS were (1) to remove interferences (elements having isobars and polyatomic ions) at mass of 99, (2) to concentrate ^{99}Tc in the separation processes, and (3) to use suitable yield tracers. For (1), removal of Ru and potential $^{98}MoH+$ ion interferences needed to be clarified. For (2), liquid-liquid extractions with cyclohexanone and chromatographic extraction with TEVA resin (Eichrom) were compared. The suitable yield tracers would be ^{99m}Tc, ^{95m}Tc and Re (chemically similar to Tc) were considered. Finally, we have developed several methods, i.e. soil, plants, river water, seawater, and seaweeds. In this presentation, we will summarize the information for ^{99}Tc analysis by ICP-MS. The method is applicable to measure ground water from monitoring wells near radioactive waste disposal sites.

Keywords
Technetium-99, Analytical methods, Environmental samples, ICP-MS

2a-5

A simple and rapid method for separation and preconcentration of Ra in water samples

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Radium-226 is one of the important radionuclides for dose assessment of radioactive waste disposal, particularly, for uranium waste. For this uranium waste, shallow land disposal will be considered and it is necessary to carry out environmental monitoring around the site, for example, well water (and ground water) monitoring. Since Ra-226 is a progeny of U-238, continuous monitoring of its concentration in water samples is of great importance. It is difficult, however, to detect low concentration of Ra-226 in water samples. Therefore, preconcentration of Ra in water samples is necessary. Recently, application of sector field inductively coupled plasma mass spectrometry (SF-ICP-MS) for determination of long-lived radionuclides present at ultraltrace level in the environment is considered as one of the popular and reliable methods; this method has several advantages, i.e., smaller sample size, higher precision, and time saving compared to those of the conventional radiometric methods. However, in order to measure ultratrace-levels of Ra-226 by SF-ICP-MS, some separation steps for Ra from alkaline and alkaline earth element in the sample matrix is needed. Furthermore, the removal of any potential sources of interference (molecular ions and isobaric interferences) that may result in incorrect results is of crucial importance, especially the polyatomic interference $^{88}Sr^{138}Ba^{+}$ because of the relatively high concentrations of Sr and Ba in water samples. Due to their similar chemical property, the separation of Ba-Ra is difficult; the effective removal of Sr from Ba (Ra) is more practical for ^{226}Ra determination by SF-ICP-MS. In this study, several ion-exchange chromatographic separation methods were compared and the most suitable condition for Sr separation from Ba (Ra) in simulated river water will be reported.

Keywords
Ra-226, SF-ICP-MS, chromatographic separation, preconcentration, water sample

2a-6

Consideration on treatment and disposal of secondary wastes generated from treatment of contaminated water

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Background

There is a significant volume of highly contaminated water generated from the accident of the Fukushima Daiichi Nuclear Power Station. Several methods have been applied to decontaminate radioactivity of the water, such as the Areva's device Curion's device, the SARRY, the ALPS, and so on. After the decontamination treatments using these devices, various secondary wastes such as sludge and spent adsorbents have been generated. These wastes are now tentatively stored at the site but the further treatment shall be applied to produce appropriate waste forms to be stored and disposed of in a repository finally.

Author has been studying about how to manage Hull wastes, i.e. Zircaloy, Stainless-steel and high Ni alloy, generated from a reprocessing plant. For this purpose, author has been pursuing waste characterization work, has developed various treatment technology options and has developed a disposal concept (i.e. the first TRU report). According to these experiences, authors consider how to promote the treatment and disposal technology in safety and efficient way.

Needs of inventory list and its online management system

Establishing waste inventory including chemical and physical form, radionuclide inventory, source term, location of waste is the fundamental issue and the first step for pursuing further consideration of treatment and disposal. However it is very difficult to fully establish it from the beginning. So the stepwise development and evaluation are important. A management system of the waste inventory should be established, and newly obtained information should be shared through the management system. There have been several and intensive waste characterization works and these results should be added in the waste inventory as soon as possible.

Development strategy of waste treatment, storage, transport and disposal technique

Establishment of the criteria for development of each technology is the fundamental issue and the first step. For example, for treatment technology, simple system, applicability for various waste, volume reduction factor, economy, minimized secondary waste, and difficulty of residual R&D are important. The total cost of treatment and disposal should be also evaluated.

Regarding the stability of waste form, low leaching rate is needed especially for the waste containing long-lived radionuclides. If identification of radionuclides of waste through the treatment process becomes easier, it will be also considered to be advantage.

Technology options for treatment, storage, transportation and disposal have been proposed from domestic and international organizations. These options should be integrated and managed in a data base of technology options. R&D result should be also included in the data base as soon as possible.

Organization of evaluation team

To evaluate the feasibility of each technology and waste inventory information needed for each technology, it is important to organize a team of system engineers, researchers, implementer and other related stakeholder. The team should evaluate the feasibility of each technology together as stepwise manner.

Needs of long-term knowledge management

Because the actual treatment and disposal of wastes will be pursued after a few tens of years, it should be also considered about knowledge management, training and education of next generation.

2a-7

A simulation of Sr leaching in aqueous solution mixed with seawater and preparation of U/Zr oxides at high temperatures

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After the station black out of Fukushima Dai-ichi NPP in March, 2011, thousands of tons of seawater (SW) for cooling the high temperature reactor cores was injected. The highly water-soluble radionuclides such as cesium and iodine were released into the cooling water and the air from the melted core and debris which might be reacted with a certain amount of zirconium alloy of the cladding of the fuel and/or its oxide. Additionally, the other FPs and TRU were possibly transferred to the cooling water mixed with SW under high temperature condition in the pressure vessel. As reported by TEPCO, most of elements have been and will be recovered by several water treatment systems. However, the real SW, which is different from the ideal and clear artificial solution in lab, contains many kinds of matrix ion, suspended matters and impurities. We have no enough knowledge of the kind and amount of radionuclides (except cesium and iodine) presented at the bottom of RPV/PCV etc. and stacked in water treatment facilities such as tanks, adsorption towers and waste containers, so far. The simulation of the radionuclide behavior in the real SW will be helpful to explain the forthcoming analysis data about the contents of minor FPs and TRU in contaminated water and solid waste, and to suggest more effective and safer decommissioning and waste management. In the present study, 1) the fuel debris composed of oxides and metals of uranium and zirconium as main constituents in the reactor was prepared and the phase relations analyzed, and in parallel, 2) the apparent chemical behavior of strontium in SW was preliminary investigated prior to the other FPs.

After the heat treatment of the mixture of UO_2 and ZrO_2 in vacuum, the UO_2 solid solution phase such as $Zr_yU_{1-y}O_{2+x}$ seemed to be formed as well as the small amount of the monoclinic ZrO_2 phase at temperatures lower than 1273 K. However, phases with higher oxidation state such as, U_3O_7 and U_3O_8 seemed to appear at 1073 - 1473 K with significant amount of the ZrO_2 phase. Over this temperature, a new phase similar to the UO_2 phase seemed to be formed showing the decomposition of U_3O_8 phase at high temperature. Also, the tetragonal ZrO_2 was also observed in the same sample since phase transition from monoclinic to tetragonal occurred at high temperature. When the mixture of zirconium to uranium oxides was treated by mechanochemical method under reducing condition, the X-ray pattern of the products showed that they were the same fluorite structure and the lattice parameter of this phase was linearly decreased with increasing zirconium ratio. This suggests that the UO_2 and ZrO_2 form the solid solution from low Zr/U ratio to high one. These results were also discussed with the phase diagram.

Sr-90 in contaminated water has been detected at a quite low molar concentration level but high radioactivity, while natural non-radioactive strontium ion exists in SW at around 10^{-4} mol/dm^3 (M). In the thermodynamic point of view, the “additional” tracer strontium must be dissolved in the cooling water completely. However, in the recovery test of Sr-85 (ca. 10^{-10} M) from SW (sampled at the seashore in Minamisoma) at the neutral pH and 25/90 °C, 5 % of Sr-85 was filtered out by 3 kDa molecular weight cut-off filter, while 100% Sr-85 was recovered through 10 kDa to 0.45μm filters. In the SEM-EDX analysis, the filtrate on the 3 kDa filter was tiny, but a sort of silica colloid, aluminosilicate, and organic matter were assigned. The interaction with these materials may imply a different behavior of Sr from the ideal solution chemistry.

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Keywords
Fuel debris, Uranium/Zirconium oxides, Fission product, Strontium, Seawater

2a-8

Volume reduction of municipal solid wastes contaminated with radioactive cesium by ferrocyanide coprecipitation technique

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The estimated remaining sustainable years of the final disposal sites for municipal solid waste (MSW) in Japan were only 18 years as of the end of FY2008. The waste avoidance, waste volume reduction and recycling of the MSW, therefore, had been the national policy. However, the pollution of the environment from the TEPCO (Tokyo Electric Power Company) Fukushima Daiichi nuclear reactor accident created an entirely new problem. Since waste incineration and water treatment are by their nature the processes that concentrate pollutants like radioactive cesium (rad-cesium hereafter) in the ashes and treatment sludge, MSW containing high concentration of rad-cesium started to be produced in some areas. The recycling of MSW as concrete material and compost, etc. has become difficult, and when the rad-cesium concentration in the waste was below the clearance level, rumors often prevented its reuse. Also most citizens are in hard opposition to the disposal of rad-cesium-containing wastes even if the radioactivity of the waste is below the governmental limit to the disposal in the landfills with leachate collection systems (i.e., 8,000 Bq/kg of Cs-134+Cs-137). Inevitably, undisposed treatment residues are now piling up in many treatment facilities in Fukushima Prefecture, which may eventually jeopardize the treatment itself and exert a serious effect to everyday life.

We have studied the applicability of the technique that uses insoluble ferrocyanide (Fer hereafter) compounds to selectively remove rad-cesium from the extracts of MSW, which are the complex mixture of various organic and inorganic compounds. For the present study, we collected waste materials (both sludge and fly ash from sewage plants, drinking water treatment plants, refuse incinerators, and melting furnaces), some of them contaminated with rad-cesium, and some not contaminated. The samples were extracted with water or oxalic acid. As for the samples originally not contaminated, we first added stable cesium (100 μg/L) or Cs-137 (8,000 Bq/L), added metal salt if necessary, adjusted pH to the desired value, and then added soluble ferrocyanide salt to produce insoluble Fer-metal compounds that co-precipitate cesium. Samples with rad-cesium contamination were extracted and tested likewise but without addition of Cs tracer. The extract before and after the coprecipitation were analyzed for rad-cesium and other constituents using a Ge semiconductor detector, ICP-MS and ion chromatography. The precipitate was subjected to X-ray fluorescence analysis and FT-IR spectroscopy.

In most of the samples, indigenous metals (mixtures of Fe(III), Zn(II), Cu(II), and Mn(II)) in the extract formed precipitates with Fer and effectively removed Cs. However, Cs removal was reduced when high concentration of Zn (e.g., 10 mM) was contained in the extract because Fer-Zn precipitate has low Cs distribution factor compared with other Fer-metal precipitates. Concerning the extracts of waste materials that had been treated to reduce the leaching of heavy metals, it was necessary to add Fe(II), Fe(III), Cu(II) or Ni(II) salt to produce the Fer-metal precipitate in them. When Fer -Ni was precipitated in the extract of rad-cesium-contaminated fly ash from a refuse incinerator, the removal of rad-cesium was low but it was improved when excess Fe(II) was used. On the other hand, rad-cesium removal was high for Fer-Ni and Fer-Fe(III) co-precipitation when the extract of rad-cesium-contaminated fly ash from a melting furnace was tested. The difference may be explained by postulating the existence of colloidal, non-ionic rad-cesium (e.g., sorbed on suspended particles) in the incinerator fly ash extract. In case of oxalic acid extracts, oxalic acid had to be diluted to 0.05M or lower to produce the Fer precipitate with 0.1 mM of Fer. On the whole, if the type of metal salts and pH in the waste extract was optimized, Cs removal was higher than 95 percents when Fer concentration was 0.1 mM (c.a. 30 mg/L of precipitate in the extract).

Keywords ferrocyanide co-precipitation, cesium removal, municipal solid waste, decontamination

2a-9

A Prescription toward to the Long-term Isolation of High Level Radioactive Waste in Japan.

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After the Fukushima-Daiichi Nuclear Power Plants accident, existing spent fuels at reactor sites cause a new arguments regarding with the high level radioactive waste issue together with vitrified wastes. It seems the high level radioactive waste issues is one of the main reason of public concern on nuclear utilization.

The technical basis for the long-term isolation of high level radioactive waste (HLW) have been established for geological disposal in Japanese islands by year 2000 based on 25 years intensive research as the H12 Project. On the other hand, a report was compiled focus to the social and institutional aspect on HLW disposal by 1998. Then, the Final Disposal Law including provisions e.g., role of the implement body as non-governmental organization, the funding system based on "polluter pays" principle, and consent base site selection processes was enacted in 2000. It is to say Japan enters into the social implementation phase of geological disposal of HLW since 2000.

Although the implement body makes every endeavor to select survey sites, the beginning step for implementation phase, however, the candidate sites has not yet been realized up to now from difficulty of gaining public confidence.

This paper will try to discuss on those issues from both technical view points and institutional ones of the final disposal of HLW in Japan, then will try to show key factors may exist in governance of the implement body as well as machinery of site selection system.

This paper proposes to reform the framework of the Final Disposal Law, e.g., the establish of implement body as the governmental organization, the introduction of a new decision-making process grounded on public participation.

Keywords

High Level Radioactive Waste, HLW, Long-term Isolation, Geological Disposal, Final Disposal Law, Governance, Implement Organization, Decision-making Process, Public Participation

2a-10

MEXT-Sponsored Research Activities for Environmental Load Reduction

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Ministry of Education, Culture, Sports, Science and Technology in JAPAN(MEXT) sponsors research activities relevant to innovative nuclear systems, which aim at overcoming problems of existing commercial nuclear power plants in safety, economy, proliferation-resistance and other issues. This sponsorship is implemented as a framework of competitive research fund, entitled "Innovative Nuclear Research and Development Program." Within this framework, a new public recruitment for research proposals began in FY2013 with specific purpose of environmental load reduction of nuclear spent fuel.

The framework of the competitive research fund and the concept of this research recruitment are explained, as well as the contents of the 6 adopted projects.

Keywords
spent nuclear fuel, environmental load reduction

2a-11

NUMO's activities after the Great East Japan earthquake

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Nuclear Waste Management Organization of Japan (NUMO)

NUMO (Nuclear Waste Management Organization of Japan) is the organization responsible for implementing geological disposal of HLW and TRU waste. A three-stage site selection process was defined by the Specified Radioactive Waste Final Disposal Act in June 2000 and open solicitation of volunteer municipalities commenced in December 2002. However, to date, no local municipality has come forward as a candidate for an initial literature survey. Against this backdrop, "Radioactive Waste Working Group" organized by Japanese Government is currently addressing issues related to geological disposal.

NUMO takes various consensus building measures including hosting symposia, workshops as well as supporting classroom debates at universities. In addition, it conducts nationwide researches on the public attitude on the geological disposal twice a year. The latest research in February 2013 resulted that 72% of the people understood the need for geological disposal but only 17% favored to construct a geological disposal facility in their home town. Their major concern was that safe disposal is not assured considering the large earthquakes such as the March 2011 Great East Japan earthquake and the subsequent accidents at the Fukushima Daiichi nuclear power plant. In order to improve this situation, NUMO has started studies on the potential impacts of such catastrophic events on the operational and post-closure safety of a geological disposal system.

With respect to the operational safety, it has to be ensured that, despite various incidents which could occur during the repository construction and operation phases, the overpack has sufficient physical strength and heat resistance to prevent the release of any radionuclides. The study concluded that 1 even though the accidental conditions beyond the design-basis were assumed, the physical and thermal durability of metal overpack would be warranted.

The identification and characterization of faults is an important issue to be addressed for ensuring post-closure safety. In fact, previously undiscovered faults have caused M7-class earthquakes in Japan during the last decade. It is, however, expected that the application of the combined techniques of surface exploration and seismic prospecting would identify such faults during preliminary and detailed site investigations. In addition, the possible impact of groundwater level changes associated with a major earthquake on radionuclide migration was also assessed. Although a radionuclide release spike appears in response to such an earthquake, the calculated maximum dose is still significantly smaller than the safety limits of international dose standards.

The above studies have re-confirmed the safety in both the operational and post-closure phases. NUMO, however, will continue to increase the confidence of safe geological disposal by applying state-of the art technologies and lessons learned from the great earthquake, as well as to communicate with public to solve their concerns.

Keywords
geological disposal, HLW, earthquake

2a-12

Migration Parameters and their Evaluation and Estimation Methodologies of Safety-relevant Radionuclides for Performance Assessment of Japanese Geological Disposal of HLW and TRU Waste

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Many radionuclides have been selected as safety-relevant for performance assessment of Japanese geological disposal system of HLW and TRU waste as shown in Fig. 1. Reliable migration parameters of these radionuclides, e.g. solubility in porewater in buffer material, distribution coefficient onto buffer material and rocks, and effective diffusion coefficient in buffer material and rocks, should be evaluated and estimated. Evaluation and estimation of solubility are based on thermodynamic calculation using originally-developed thermodynamic database. Distribution coefficient and effective diffusion coefficient are evaluated and estimated based on accumulating many appropriate experimental values and using sorption and/or diffusion models, though there are various methodologies. Sorption and diffusion databases have been developed by accumulating experimental values with various conditions as many as possible, assigning their reliability based on some criteria, and selecting appropriate values from the accumulated values. Examples of the proposed methodologies on evaluation and estimation of migration parameters will be presented. Tremendous efforts have been made to evaluate and estimate the reliable parameters as described above. Required resources generally increase with increasing the number of elements of interest. Therefore, reduction of safety-relevant radionuclides using nuclear transmutation may be effective to reduce the efforts.

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18
1	H																	He
2	Li	Be											B	**C** C-14	N	O	F	Ne
3	Na	Mg											Al	Si	P	S	**Cl** Cl-36	Ar
4	K	Ca	Sc	Ti	V	Cr	Mn	Fe	**Co** Co-60	**Ni** Ni-59,63	Cu	Zn	Ga	Ge	As	**Se** Se-79	Br	Kr
5	Rb	**Sr** Sr-90	Y	**Zr** Zr-93	**Nb** Nb-93m,94	**Mo** Mo-93	**Tc** Tc-99	Ru	Rh	**Pd** Pd-107	Ag	Cd	In	**Sn** Sn-126	Sb	Te	**I** I-129	Xe
6	**Cs** Cs-135,137	Ba		Hf	Ta	W	Re	Os	Ir	Pt	Au	Hg	Tl	**Pb** Pb-210	Bi	Po	At	Rn
7	Fr	**Ra** Ra-226,228		Rf	Db	Sg	Bh	Hs	Mt	Ds	Rg	Cn						

La	Ce	Pr	Nd	Pm	**Sm** Sm-151	Eu	Gd	Tb	Dy	Ho	Er	Tm	Yb	Lu
Ac Ac-227	**Th** Th-228,229, 230,232	**Pa** Pa-231	**U** U-233,234, 235,236,238	**Np** Np-237	**Pu** Pu-238,239, 240,241,242	**Am** Am-241, 242m,243	**Cm** Cm-244, 245,246	Bk	Cf	Es	Fm	Md	No	Lr

Fig. 1 Safety-relevant radionuclides for performance assessment of Japanese geological disposal of HLW and TRU waste

Keywords

High-level radioactive waste (HLW), TRU waste, geological disposal, performance assessment, safety-relevant radionuclide, solubility, distribution coefficient, effective diffusion coefficient

2a-13

Evaluation of Distribution Coefficients for Radionuclides Sorption on Bentonite Colloid

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In the geological disposal of high-level radioactive waste (HLW) in Japan, the vitrified HLW surrounded by overpack and compacted bentonite is disposed of in deep underground to ensure the long-term safety. Compacted bentonite saturates with groundwater after repository closure. Radionuclides contained in HLW may dissolve into porewater of compacted bentonite and migrate through the compacted bentonite and host rock. Compacted bentonite effectively retards the radionuclides by sorption with low permeability. However, colloidal particle of bentonite (bentonite colloid) might be generated by the erosion of the compacted bentonite. Bentonite colloid sorbs less onto the minerals in host rock, so migration of radionuclides is possibly facilitated by radionuclide sorption on bentonite colloid. Therefore, potential impact of bentonite colloid on the migration of radionuclides needs to be evaluated in the performance assessment of geological disposal.

Japan Atomic Energy Agency (JAEA) has been investigating sorption behavior of radionuclides on the bentonite colloid and developing the evaluation model for the radionuclide sorption behavior. In our previous studies, the distribution coefficients of radionuclides on bentonite colloid were found to be higher than those on non-colloidal bentonite particle. This difference was interpreted to be due to the higher surface complexation site capacities in bentonite colloid. Considering this site capacities, the sorption behavior of Am was well interpreted based on a previously reported sorption model for non-colloidal bentonite[1)]. The calculated distribution coefficients by this evaluation method were also good agreement with the experimental results of Cs and Am sorption on bentonite colloid in 'solution - bentonite colloid - granite' ternary system, which is a simulated condition of radionuclides migration in host rock [2)].

To apply the evaluation model for the performance assessment, selectivity coefficients of safety-relevant radionuclides are necessary for each surface complexation site. Therefore, the applicability of evaluation model should be validated systematically for the safety-relevant radionuclides. In this context, we have been carrying out the sorption experiment of Cs, Am and Np on bentonite colloid to verify the applicability. In the presentation, the evaluation model for distribution coefficient of radionuclides sorption on bentonite colloid will be introduced in detail and the application result of the evaluation model to the sorption behavior of Np on bentonite colloid will be shown.

1) K. Iijima et al., (2008). Radiochim. Acta 96, pp.721-730.
2) K. Iijima et al., (2010). Radiochim. Acta 98, pp.729-736.

Keywords
Geological Disposal, Performance Assessment, Distribution Coefficient, Bentonite Colloid, Sorption Model

2a-14

The treatment process of simulated debris from a severe accident using molten salt system

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In a severe nuclear accident, there is a risk of the nuclear fuel melting and forming debris. The debris contains uranium and highly radioactive nuclides, and therefore needs to be handled with great care in terms of both critical safety and high radiation safety. The debris also contains a large amount of weakly radioactive Fe and Zr, and it would be beneficial to separate the debris into each material group. However, the composition of the debris differs from that of spent fuel, and it is not possible to employ conventional reprocessing technology involving dissolution with nitric acid, because the debris contains a UO_2-ZrO_2 stable insoluble oxide mixture. In addition, it is difficult to employ metallurgical processes because of the high melting points and because the reactivity of Zr with water becomes higher above 900 °C. On the other hand, some molten salts are known to exhibit high solubility, good Zr recovery, and the ability to decontaminate Fe components. We speculated that this approach would be useful for separating Fe and Zr from the debris to produce low-level radioactive waste. In this study, therefore, we propose the concept of the treatment process and demonstrate the approach of the treatment process of debris. The simulated debris was prepared and pyrochemical electrolysis of Fe and Zr in NaCl-KCl molten salt was employed to separate Fe and Zr from debris.

The simulated debris was made from Zr, Fe, and CeO_2. The CeO_2 was used for simulating stable UO_2-ZrO_2. The simulated debris was prepared by using an arc furnace. The composition of the simulated debris was 20 wt% Zr, 20 wt% Fe, and 60 wt% CeO_2. The reagents of the simulated debris were placed in a crucible and melted with the arc furnace in an Ar atmosphere. This simulated the melting debris at the severe accident nuclear power plant. The cross-section of the simulated debris was analyzed by SEM and EDX, and the oxide phase was analyzed by XRD. After the simulated debris was melted in the arc furnace and solidified, it was separated into a metal phase and an oxide phase. The metal phase formed as spheres in the oxide phase. The two phases did not exhibit wettability with each other. From EDX analysis, the metal phase was a uniform alloy of Zr and Fe, and the oxide phase was Ce oxide. From the XRD analysis, the organic phase was identified as CeO_2.

Electrorefining of the simulated debris was conducted in molten NaCl-KCl containing 6 mol% $ZrCl_4$. In this study, NaCl-KCl molten salt was selected from the viewpoint of the wide potential window and properties of this salt, which is not a deliquescent salt. NaCl-KCl-6mol%$ZrCl_4$ (melting point of NaCl-KCl is 658 °C) was heated to 700 °C in an alumina cell to carry out electrorefining. The simulated debris was placed on a Ni plate, serving as the anode. Stainless steel was used as the cathode. In the electrorefining process, metal species in the simulated debris were dissolved electrically from the anode and deposited on the cathode. After electrorefining, 89% of the metal phase was separated from the simulated debris. The metal species in the simulated debris was separated as a deposit on the cathode.

In this study, we successfully separated Zr and Fe from simulated debris using NaCl-KCl molten salt electrolysis, and we selectively recovered the Zr and Fe. With this approach, it should be possible to reduce the volume of the debris by recovering metals, which can then be treated as low-radioactivity groups.

Keywords
severe accident, debris,pyrochemical, electrolysis, molten salt

2b-1

Considering Geological Disposal Program of High-level Radioactive Waste through Classroom Debate

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1. Introduction

Debate, in a narrow sense, i.e. "classroom debate", is conducted for the purpose of education and training. Perceived generally as a means of enhancing communication skills, debate also provides other benefits. Debate is an excellent way to develop students' ability to collect, organize and understand information since they need to deal with a vast amount of information in order to grasp issues associated with the resolution. In addition, students are allowed to take multiple,objective and critical perspectives through the experiences of taking both affirmative and negative positions, as well as refuting arguments in debate games.

Against this backdrop, "Introduction to Debate" courses offered at Sugiyama Jogakuen University and Aichi Shukutoku University in the first semester of 2013, focused on "High-level Radioactive Waste Management Program", a topic of current importance. Yet students majoring in liberal arts, in particular, had poor knowledge of this topic. This paper will describe and analyze the process and results of the coursework. It will also examine the advantages of debate in understanding and sharing the said topic.

2. Process and results of the coursework

At the outset, manners, techniques and rules commonly used in debate were introduced to students. Then, students repeatedly practiced model debates, which led them to get accustomed to debating. After wards, explanation about the resolution was given.

"High-level Radioactive Waste Management Program" was set as the subject for debate this year. In order to help students deepen their understanding of the issues and prepare for debate games, Professor Yamana of Kyoto University was invited to give lectures to students. With the cooperation of the Nuclear Waste Management Organization of Japan (NUMO), students also had the opportunity to visit Chubu Electric Power Company's Hamaoka Nuclear Power Station Exhibition Center as well as Japan Atomic Energy Agency's Underground Research Laboratory in Tono Geoscience Center.

Debate games were conducted under the resolution: "Should Japan abolish the geological disposal program and mandate permanent waste management?" Carefully written and debated last year in Professor Fujikawa's course at Chiba University, the resolution was appropriate for use in this year's debate at Sugiyama and Aichi Shukutoku. In each class of the two universities, two or three groups were formed to conduct the research necessary for debate games. As students would take both affirmative and negative positions, they explored arguments on both sides of the issue.

In debate games, some groups took full advantage of their preparation efforts while others did not. In the end, all seemed to realize the importance of preparation.

3. Evaluating effects of the coursework.

Through lectures by experts and tour to relevant facilities, students successfully gained a deep understanding of the topic which, at the outset, was almost unknown to them.

Participating in debate games also motivated students to fully grasp detailed issues since superficial knowledge is not enough. As students took both affirmative and negative positions, debate was effective in dealing with the controversial issue of nuclear waste management.

This year, students before joining the course, were almost indifferent to the resolution, which must be examined regardless of Japan's policy over nuclear power - , either it will continue or it will be abolished. They became interested in the topic at the end. As a result, the goal of the course was accomplished.

Keywords

Debate, Radioactive Waste