ANFC 2017

The 3rd Asian Nuclear Fuel Conference

Innovation by Collaboration

Jeju, Korea

September 8-9, 2017

Ramada Plaza Jeju

http://nuclear.kaist.ac.kr/ANFC2017
Welcome Message

Dear Colleagues,

The Asian Nuclear Fuel Conference (ANFC) is the tri-annual international meeting, which provides an international and academic forum for fundamental aspects of nuclear fuels, with collaborations among Korea Nuclear Society, Atomic Energy Societies of Japan, and China Nuclear Society.

In September 2017, the ANFC 2017 will be held in Jeju Island, which is registered as a UNESCO World Natural Heritage. I look forward to seeing you in Jeju in ANFC 2017.

Prof. Dong Seong SOHN
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ANFC 2017
Registration Fee

For KNS/AESJ/CNS members:
• 200 USD (220,000 KRW) till Aug.15
• on site 300,000 KRW

For Students:
• 100 USD (110,000 KRW) till Aug.15
• on site 150,000 KRW

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# Program
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Invited Speakers

• Prof. D.S. Sohn (UNIST)
  - U-Mo/Al Research Reactor Fuel Performance

• Mr. Zengguang Lei (CNNC)
  - Nuclear Fuel Cycle in China

• Dr. M. Osaka (JAEA)
  - Results and Progress of Fundamental Research on FP Chemistry

• Dr. M. Steinbrück (KIT-IAM)
  - High-temperature oxidation behavior of classical zirconium alloys and new ATF cladding materials

• Prof. M. Uno (Univ. Fukui)
  - Preparation of Sodium Uranates

• Prof. K. Kurosaki (Osaka Univ.)
  - Heat Capacity Analysis of Zirconium Hydride and Deuteride
Oral Presentations
The U-Mo/Al research reactor fuel has been under development for the conversion of HEU (high enriched uranium) fueled core to LEU (low enriched uranium) fueled core as a part of the Reduced Enrichment for Research and Test Reactors (RERTR) program, since 1978. In this paper, we present briefly our research achievements on the irradiation performance modelling for U-Mo/Al dispersion fuel: fission-induced meat swelling and creep, pore formation in the fuel meat by the mechanistic manner, thermal conductivity of U-Mo/Al fuel meat, and establishment of the PRIME code for the performance analysis. These achievements are described with the review of the physics and mechanisms involved in them.
Green Nuclear Energy Promoting the Sustainable Development of Energy (invited)

Authors: Zengguang Lei
Affiliation and address:
1. China National Nuclear Corporation

Presenting author: Zengguang Lei
Position: Chief Engineer
A fundamental research on fission product (FP) chemistry is underway at Japan Atomic Energy Agency. The purpose is to establish an FP chemistry database in each region of a LWR (RPV, RCS, PCV) under severe accident conditions. Improvement of FP chemical models based on this database is another important task of the research. Research outputs are reflected to the research and development of decommissioning of Fukushima Daiichi Nuclear Power Station (1F) and the enhancement of LWR safety. Four research items have thus been conducted considering the specific issues of 1F and the priority in the source term research area, as follows:
- Effects of boron (B) release kinetics and chemical conditions on FP behavior,
- Cesium (Cs) chemisorption and reactions with structural materials,
- Establishment of a thermodynamic and thermophysical properties database for FP compounds,
- Development of experimental and analytical techniques for the reproduction of FP behavior and for measurement techniques of chemical form of FP compounds.

In this presentation, the results and progress of the research are presented. The main body of the present research is reproductive tests for FP release and transport, which aims at acquiring experimental information for FP chemistry by direct measurements of chemical forms of FP compounds in deposits and gas-phase samples. Concept and specification of the test facility named TeRRa (Test Bench for FP Release and TRansport) are presented. Some preliminary test results conducted using CsI samples under inert and steam-containing atmospheres have shown the effects of steam on the aerosol formation.
and growth behavior. Another important issue is to obtain basic knowledge on Cs chemisorption onto structural material, specifically stainless steel (SS). The chemisorbed Cs can become a long-term radiation source having significant impact on the management of workers’ radiation dose in the 1F decommissioning work. Basic test results for the Cs chemisorption have shown that Cs formed a complex oxide of Cs-Si-Fe-O by chemical reactions with the component of SS at relatively high temperatures over 1000 K. Results of a systematic test by varying test conditions have shown that temperature, atmosphere and elements contained in the SS were the main factors influencing the Cs chemisorption behavior. It is of note that a revaporization of Cs chemisorbed compounds, CsSiFeO₄, was promoted under a steam starvation atmosphere. Thus it was found that the chemistry for the Cs chemisorption phenomena should be taken into account for the Cs chemisorption model.
High-temperature oxidation behavior of classical zirconium alloys and new ATF cladding materials (invited)

Authors: Martin Steinbrück, Chongchong Tang
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Presenting author: M. Steinbrück
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Zirconium alloys are used as fuel cladding materials in light water reactors (LWR) for decades. They show excellent properties during operation but pose the risk of strong degradation, hydrogen and heat release during severe accident scenarios due to the serious high-temperature interaction of the zirconium alloys with water steam. The talk will give a brief summary of comprehensive work done at KIT on high-temperature oxidation of zirconium alloys in various atmospheres before it addresses recent developments on ATF claddings with focus on their high-temperature oxidation behavior. Examples will be given for silicon carbide ceramic-matrix composites, coated zirconium alloys and FeCrAl alloys.
Paper Title: Enhanced Oxidation Resistance of a Zirconium Alloy Cladding Tube by High-Temperature Pre-Oxidation

Authors: C.M.Lee¹, T.W.Cho¹, G.Y.Jeong¹, M.J.Kim¹, H.-J.Lee¹, J.-H.Kim¹, D.-S.Sohn*¹

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After the Fukushima accident, the researches of accident tolerant fuels (ATF) are being conducted with concentrated interests in all over the world. The main goal of ATF is to retain adequate ductility during and after an accident such as loss of coolant accident (LOCA), so that the fuel claddings are not failed, and the radioactive materials within the fuel claddings are not released to the outside. Thereby, newly developed ATF should achieve low oxidation rate at high temperature to minimize the concentration of oxygen in the zirconium substrate and the formation of brittle phases such as zirconium dioxide (ZrO₂) and oxygen-stabilized alpha zirconium (α-Zr(O)). In addition, they should also have a high resistance to the onset of the breakaway oxidation, which is known to cause the rapid absorption of hydrogen and oxygen to fuel cladding, resulting in the cladding to become brittle in a relatively short time. One of the possible methods which can be used as the future ATF is the high-temperature pre-oxidized zirconium alloy. So far, most of the pre-oxidation studies were performed at normal operating temperature (360 °C). However, it was found that the pre-oxidation at high temperature (800-1200 °C) is significantly more effective to reduce the oxidation rate and to enhance the breakaway oxidation resistance, and these findings will be presented in this presentation.
A Study on the Corrosion Behavior of Zirconium Alloys with Anodic Nanoporous Oxide in High Temperature Severe Accident Environment

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Presenting author: Y.J.Park
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In the case of zirconium alloys, which are widely used in the field of nuclear engineering, explosive generation of hydrogen, a by-product of reaction of zirconium and steam, was pointed out as a cause of the accident in Fukushima Nuclear Power Plant in 2011. Therefore, research on the accident-tolerant fuel cladding material (ATF) has been attempted to increase the safety of the zirconium alloy cladding material in the reactor. The main research flow is an attempt to completely replace the zirconium alloy with ceramic materials using SiC or other metal materials such as Mo and FeCrAl. These studies provide improved corrosion resistance instead of abandoning the powerful advantages of zirconium. The authors proposed an anodization method to fabricate a nanostructured zirconium oxide film on the surface of the alloy to improve the corrosion resistance at a lower cost. In this paper, after the anodization of Zr-Nb-Sn alloys to produce nanostructured oxide films, the corrosion resistance of specimens was evaluated by thermo-gravimetric analyzer under high temperature air and steam environment and compared with pristine Zr-Nb-Sn alloys. As a result, the nanostructured oxide film formed by anodization shows excellent corrosion protection.
Fabrication of oxide pellets containing lumped Gd$_2$O$_3$ for the application of new burnable absorber fuel design

Authors: Qusai M. Mistarihi$^1$, Wooseong Park$^2$, Kyungseok Nam$^2$, Ho Jin Ryu$^1$*

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The performance of Light Water Reactors (LWRs) can be improved by effectively controlling the excess amount of reactivity added at the beginning of the fuel cycle (BOC) in order to have a longer fuel cycle lengths and higher fuel burnups. Burnable Absorbers is one of the reactivity control mechanisms for the excess amount of reactivity. UO$_2$-Gd$_2$O$_3$ fuel in the form of UO$_2$ homogeneously mixed with Gd$_2$O$_3$ is one of the most commonly used BA fuel. Since the Gd$_2$O$_3$ has a high neutron absorption cross section, its burnout rate is very high and it will be consumed quickly at the BOC. The effectiveness of UO$_2$-Gd$_2$O$_3$ fuel can be improved by the lumping of Gd$_2$O$_3$ into the UO$_2$ matrix in order to reduce its burnup rate through having a higher self-shielding factor and therefore better reactivity control. However, the lumping of Gd$_2$O$_3$ into the UO$_2$ matrix will introduce some challenges to the fabrication process in terms of the formation of interfacial cracks due to the different sintering rate and thermal expansion coefficients of UO$_2$ and Gd$_2$O$_3$ respectively. None the less, these interfacial cracks can be eliminated by changing the sintering rate and thermal expansion mismatch through mixing or by changing the initial density of the lumped Gd$_2$O$_3$. In order not to affect the properties of Gd$_2$O$_3$, changing the initial density approach seems to be more effective. 8 wt. Y$_2$O$_3$ stabilized ZrO$_2$ (8YSZ) pellet, a surrogate for UO$_2$, containing lumped Gd$_2$O$_3$ mini-pellet, sphere, or rod were fabricated by the combination of uniaxial pressing and cold isostatic pressing and sintered by the microwave sintering at temperatures up to 1500°C. Gd$_2$O$_3$ spheres were fabricated by the drip casting method and the optimized parameters of the fabrication process were investigated. Spherical beads with Gd$_2$O$_3$ content of 75 wt. % were produced when the Na alginate concentration was 4 wt. %, the dropping distance was 3-8 cm. The effect of the initial density of the lumped Gd$_2$O$_3$ on the fabric-ability of oxide pellets containing lumped Gd$_2$O$_3$ was investigated. The initial density of the lumped Gd$_2$O$_3$ mini-pellets were controlled by changing the pre-sintering temperature of the Gd$_2$O$_3$ mini-pellets before the insertion in the 8YSZ pellets and then further sintering by MW at 1500°C. It was found that by controlling the initial density of the lumped Gd$_2$O$_3$, the interfacial cracks can be eliminated. In addition, the mini-pellets after the insertion and further sintering by MW showed a higher densification than the mini-pellets without further sintering by MW due to the uniform microstructure formed at the initial step sintering.
Mechanical and Thermal properties of $\text{U}_3\text{Si}_2$

Authors: Afiqa Mohamad$^1$, Yuji Ohishi$^1$, Hiroaki Muta$^1$, Ken Kurosaki$^{1,2}$, Shinsuke Yamanaka$^{1,3}$

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$\text{U}_3\text{Si}$ and $\text{U}_3\text{Si}_2$ have been explored as alternative candidate nuclear fuel materials for accident tolerant fuel due to the higher thermal conductivity and higher uranium density compared with those of $\text{UO}_2$. However, $\text{U}_3\text{Si}$ swells considerably under irradiation, and a phase transition occurs above 1000 K$^1$. From this perspective, $\text{U}_3\text{Si}_2$ is a promising candidate due to its favorable properties. To utilize $\text{U}_3\text{Si}_2$ as alternative nuclear fuels, accurate understanding of the thermal and mechanical properties of $\text{U}_3\text{Si}_2$ is important for design of the fuel. There are no experimental data regarding the mechanical properties of $\text{U}_3\text{Si}_2$ and there are discrepancies in the reported thermal properties$^2$. Therefore, in the present study we aim to report thermal and mechanical properties of $\text{U}_3\text{Si}_2$.

The nominal composition of $\text{U}_3\text{Si}_2$ was prepared from natural uranium and Si with 2 wt% of excess Si by arc melting in Ar atmosphere. The $\text{U}_3\text{Si}_2$ bulk sample was then synthesized by spark plasma sintering at a pressure of 75 MPa and a temperature of 1123 K followed by annealing at 1000 K for 72h. X-ray diffraction measurement showed that almost single phase of tetragonal $\text{U}_3\text{Si}_2$ was obtained. The pellet density exceeded 96% of the theoretical density. The experimental values of thermal and mechanical properties obtained from the present study were compared with the available literature regarding the properties of $\text{U}_3\text{Si}_2$. The results of thermal and mechanical properties will be discussed in the presentation.

Preparation of Sodium Uranates (invited)

Authors: Masayoshi UNO\textsuperscript{1}, Keisuke YOKOYAMA\textsuperscript{1,2} and Yukihiro MURAKAMI\textsuperscript{1,3}
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Presenting Author: Masayoshi UNO
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In order to estimate the fuel behavior during a severe accident in a fast reactor it is necessary to make pure sodium uranates and to measure their properties. In the present study, sodium uranates (Na\textsubscript{x}U\textsubscript{y}O\textsubscript{z}) were synthesized at various reaction temperatures for various durations using UO\textsubscript{2} and Na\textsubscript{2}O or Na\textsubscript{2}CO\textsubscript{3} as starting materials. The purity of the products was estimated by an X-ray diffraction method. We succeeded in synthesizing nearly pure sodium uranate such as Na\textsubscript{2}U\textsubscript{2}O\textsubscript{7}, α-Na\textsubscript{2}UO\textsubscript{4}, NaUO\textsubscript{3}, and Na\textsubscript{4}UO\textsubscript{4}. Pure Na\textsubscript{2}U\textsubscript{2}O\textsubscript{7} was obtained by heating a mixture of UO\textsubscript{2} and Na\textsubscript{2}CO\textsubscript{3} (UO\textsubscript{2}:Na\textsubscript{2}CO\textsubscript{3}=2:1) at 1123K for 50 hours in flowing dry air. Nearly pure a-Na\textsubscript{2}UO\textsubscript{4} was obtained by heating a mixture of UO\textsubscript{2} and Na\textsubscript{2}CO\textsubscript{3} (UO\textsubscript{2}:Na\textsubscript{2}CO\textsubscript{3}=1:1) at 1123K for 20 hours in flowing dry air. Nearly pure NaUO\textsubscript{3} was obtained by heating a mixture of UO\textsubscript{2} and Na\textsubscript{2}CO\textsubscript{3} (UO\textsubscript{2}:Na\textsubscript{2}CO\textsubscript{3}=2:1) at 1073K for 22 hours in flowing Ar. Pure Na\textsubscript{4}UO\textsubscript{4} was obtained by heating a mixture of UO\textsubscript{2} and Na\textsubscript{2}CO\textsubscript{3} (UO\textsubscript{2}:Na\textsubscript{2}CO\textsubscript{3}=2:1) at 773K for 5 hours in flowing 5%H\textsubscript{2}-Ar. Furthermore, near pure a-Na\textsubscript{2}UO\textsubscript{4} was obtained by heating a mixture of a-Na\textsubscript{2}U\textsubscript{2}O\textsubscript{7} and Na\textsubscript{2}CO\textsubscript{3} (Na\textsubscript{2}U\textsubscript{2}O\textsubscript{7}:Na\textsubscript{2}CO\textsubscript{3}=1:1) at 1073K 12 hours in flowing Ar. Nearly pure NaUO\textsubscript{3} was obtained by heating a mixture of Na\textsubscript{2}UO\textsubscript{4} and UO\textsubscript{2} (Na\textsubscript{2}UO\textsubscript{4}:UO\textsubscript{2}=1:1) 1073K for 18 hours in flowing Ar. The phase transition temperature from the α-phase to the β-phase of sample of Na\textsubscript{2}U\textsubscript{2}O\textsubscript{7} was also clarified to be 658.2K by thermal expansion measurement.
Phase Relation of Simulated MCCI Debris at High Temperatures

Authors: N. Sato\textsuperscript{1}, D. Akiyama, A. Kirishima, T. Sasaki\textsuperscript{2}

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By the severe accident occurred at Fukushima Dai-ichi Nuclear Power Station (NPS) in March 2011, several types of fuel debris were formed by the reaction of fuel materials with various reactor structure materials at high temperatures. They are classified as oxide, metal, MCCI, powder and secondary debris. The MCCI debris was formed by the interaction between molten core and concrete. To accomplish the decommissioning of Fukushima Dai-ichi NPS, it is essential to understand the phase relationship between the fuel material and the concrete components in MCCI debris. In this study, we focused on the reaction of uranium dioxide (UO\textsubscript{2}) and concrete components such as CaO and SiO\textsubscript{2} at high temperatures under oxidizing and reducing conditions. The mixed powder of UO\textsubscript{2} and each reactant materials were heated at 1200 and 1400°C in Ar + 2\%O\textsubscript{2} (oxidizing condition) or Ar + 10\%H\textsubscript{2} (reducing condition) atmospheres with a gas flow rate of 20 ml/min for reaction time. After the heat treatment, the phase relation of the products was analyzed by powder X-ray diffraction (XRD). In the UO\textsubscript{2}-CaO system, several types of U-Ca-O compounds were formed below 1400°C and (Ca,U)O\textsubscript{2+x} solid solution appeared at 1600°C under the reducing condition. On the other hand, CaUO\textsubscript{4} phase was observed as well as Ca\textsubscript{y}Si\textsubscript{1-y}O\textsubscript{2+x} solid solution one under oxidizing condition. For the UO\textsubscript{2}-SiO\textsubscript{2} system, no significant phase change was observed under reducing and oxidizing conditions. In the case of the UO\textsubscript{2}-CaO-SiO\textsubscript{2} ternary system, UO\textsubscript{2} and CaSiO\textsubscript{3} phases were observed under reducing condition. In case of oxidizing condition, (Ca, Si, U)O\textsubscript{2+x} solid solution and Ca\textsubscript{2}SiO\textsubscript{4} phases coexisted at 1200°C, while only (Ca, Si, U)O\textsubscript{2+x} solid solution was observed over 1300°C For the UO\textsubscript{2}-CaO-SiO\textsubscript{2} systems, the lattice parameters of UO\textsubscript{2} solid solution phase decreased with increasing temperature indicating the increase of solubility of these elements. From these results, it should be noticed that reaction behavior of uranium oxides and concrete components is different at high temperatures.
Selective fluorination and molten salt electrolysis for fuel debris – HF fluorination of UO₂ and ZrO₂

Authors: H. Matsuura¹, T. Ono¹, T. Uchiyama¹, A. Nezu², N. Sato³
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An on-site pyrochemical treatment process for fuel debris has been proposed. The procedure proposed is: 1) selective fluorination, 2) selective dissolution into molten salt, and 3) molten salt electrolysis. As the first step, fluorination treatment by dry hydrogen fluoride gas has been investigated, since uranium and trans uranium elements should not be vaporized as hexavalent form, but kept as tetravalent form. In this step, the best condition of selective fluorination of uranium must be found. According to thermodynamic studies on various composition of uranium – zirconium oxide under the treatment of dry HF gas, pre-treatment by hydrogen gas reduction is considered to be very effective to selective fluorination of uranium in the debris containing in uranium rich composition. However, if the fuel debris contains zirconium rich, pre-oxidation treatment would even better to avoid proceeding the fluorination of zirconium.
Zirconium (Zr) hydride and deuteride are considered as a candidate of neutron reflector materials in fast reactors (FRs). When Zr hydride and deuteride are installed into FRs, the thermal properties such as the heat capacity ($C_p$) need to be understood. According to the Zr-H binary phase diagram, up to the hydrogen content $x$ of 2.0 in ZrH$_x$, there exist two phases, viz., the $\delta$-phase ($x \sim 1.6$) with a cubic structure and the $\epsilon$-phase ($x \sim 1.9$) with a tetragonal structure. Although several data on the thermal properties of $\delta$-phase Zr hydride and deuteride have been reported [1], those of $\epsilon$-phase Zr hydride and deuteride are limited. The motivation of the present study is to investigate the $C_p$ of $\epsilon$-phase Zr deuteride, in order to contribute to practical application of Zr hydride/deuteride neutron reflector materials in FRs. Bulk samples of $\epsilon$-ZrD$_{1.90}$ were fabricated and their temperature dependence of $C_p$ was evaluated. The obtained data were compared with those of $\delta$-ZrH$_{1.66}$ to understand the effects of both the kind and the amount of hydrogen isotopes on the $C_p$ of Zr hydride and deuteride [2].

Interaction behavior between U-Zr-Ce and yttria coated graphite crucible during casting for SFR metallic fuel development

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Rare-Earth(RE) elements in the metallic fuel derive several negative effects such as strong reactivity, FCCI, and distribution of other contents in the metallic fuel. Therefore, it is necessary to improve distribution of RE in the U-Zr matrix and to prevent interaction between RE and casting components. In order to discuss the interation between crucible and U-Zr melt containing RE elements, U-Zr-Ce alloy was analyzed as a surrogate of U-Zr-RE fuel. The property of Ce represents RE elements and it is possible to simplify the interaction. U, Zr, Ce elements were charged into a yttria coated graphite crucible to cast the U-Zr-Ce fuel. The yttria was plasma spray coated on the graphite crucible to prevent the interaction. Physical and chemical property of melt residue which was fabricated via injection casting was investigated by SEM, EDS, and XRD. The coating layer was damaged and reacted with melt, but the crucible was not damaged during the casting process. Carbon was penetrated into the melt, and Ce was agglomerated at the penetration site. Investigation of coating layer at the penetration site was not possible, but it is possible to assume that the interaction was started at the crack site of coating layer.
A uranium-molybdenum (U-Mo) alloy fuel has been developed for a next research and test reactor fuel due to its high uranium density. The most favorable design has been considered as a dispersion type of plate fuel where U-Mo particles are distributed in the Al matrix. Thermal conductivity is one of the most important variables determining fuel temperature and performances. The major microstructure change during irradiation is the formation of the interaction layer (IL) between U-Mo particles and Al matrix. The formation of IL and consumption of Al matrix cause a severe degradation in the thermal conductivity. However, the available data are very scarce in the literature due to the difficulty of the sample fabrication and measurement. In this study, we measured the thermal conductivities of as-fabricated U-7Mo/Al and U-7Mo/Al-5Si dispersion fuel having uranium loading of 5.0, 6.5, and 8.0 gU/cm³. We also measured the thermal conductivities of heat-treated U-7Mo/Al to investigate the effects of IL. The thermal diffusivity was measured by laser flash method and heat capacity was measured using differential scanning calorimeter (DSC). The thermal conductivity was then calculated using measured data and expressed as a function of temperature and U-Mo and IL volume fraction. As expected, the Si addition to Al matrix causes a decrease in the thermal conductivity. As the U-Mo and IL volume fraction increase, the thermal conductivity decreases monotonically. These overall trends are accord with previous literature. It is also compared with previous theoretical models and the models give an over-estimation for both as-fabricated and heat-treated cases.
Uncertainty and Sensitivity Analysis for U-Mo/Al Dispersion Fuel

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A large number of the operating research reactors still use highly-enriched uranium (HEU) as fuel to provide high neutron fluxes that are superior for their applications. However, apprehensions on the use of HEU significantly increase. Therefore, the development of low-enriched uranium (LEU) fuels has been pursued for research reactors to replace the use of highly-enriched uranium (HEU) to improve proliferation resistance of fuels and fuel cycles. U-Mo particles dispersed in an Al matrix (U-Mo/Al dispersion fuel) is a promising fuel for the conversion of research reactors that currently use HEU fuels to LEU-fuelled reactors due to its high density and good irradiation stability.

Uncertainties of key parameters may influence a significant impact on the fuel temperature since fuel performance, represented by swelling, fission gas release, and interaction layers formation, is affected by fuel temperature and vice versa. These parameters and their uncertainties were obtained from mainly ATR or IAEA documents and AFIP-1 data of RERTR program. In addition, some parameters and their uncertainties have been calculated through the basic equations and uncertainty propagation methods to be used for the uncertainty and sensitivity analyses.

In this study, the uncertainty ranges of the reactor operation conditions, fuel fabrication, fuel properties, and the changes of fuel during irradiation, such as the thermal conductivity of irradiated fuel, oxide layer thickness and pH value uncertainties, are used to determine the probable fuel temperature ranges with a confidence level of 95%. The combined uncertainty effect of these parameters on the fuel temperature range is also determined using the propagation of uncertainty and probabilistic sensitivity analysis (Monte Carlo simulation) by randomly sampling numbers following the Gaussian distribution. Lastly, Pearson correlation coefficient, a commonly used statistical coefficient in nuclear engineering, was used as the sensitivity analysis parameter to show the relative influence of these parameters on the operating temperature.

The results provide the operating temperature profile of U-Mo/Al dispersion fuel as well as the upper and lower bounds of the temperature as the overall effect of the uncertainties is applied. On the other hand, Pearson correlation coefficient shows the heat flux and the heat transfer coefficient have the highest relative impact on controlling the temperature of the fuel as well as their effect on other parameters as they have the largest values of the coefficient. These sensitivity analyses may provide technical advantages in the fuel performance evaluation, computational safety analyses, and the licensing process. The uncertainty and sensitivity analyses can be used for any reactor with any different operation conditions if the required parameters for the temperature calculations are provided.
Irradiation performance modelling for U-Mo/Al dispersion fuel in the PRIME code

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The PRIME (PRedIction code for thermo-MEchanical performance of research reactor fuel) computational program has been developed to predict plate-type fuel performance and assess thermomechanical integrity in the irradiation conditions. It is a collection of the FORTRAN-based modules that are coupled dependently to each other. The PRIME code has a capability to calculate the meat swelling distribution profile with consideration of fission-induced creep, and porosities by not only bubbles and pores formed both in the inside and outside of fuel particles. In this paper, the irradiation performance modelling and implementation scheme are described, and the demonstration of the PRIME code is performed.
Poster Presentations
A study of cesium chemisorption onto surface of stainless steel

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Cesium distribution in Fukushima Daiichi Nuclear Power Station (1F) is of crucial importance for the 1F decommissioning R&D. Our focus is insoluble and stable Cs compounds deposited on the reactor structural materials formed by the chemisorption phenomena, i.e. a chemical reaction between Cs vapor and stainless steel (SS) structural material at relatively high temperature of around 1000 K simulated the temperature of inside of RPV at 1F accident. Although a basic knowledge for the Cs chemisorption phenomena had been obtained in the ‘80s, it is insufficient for the 1F having a variety of chemical conditions.

The objective of this study is, therefore, to establish the Cs-chemisorption model for Cs distribution estimation by a severe accident (SA) analysis code, considering chemical properties and knowledge of the chemisorbed compounds that would be obtained in this study. The obtained knowledge is also useful for an accurate prediction of chemistry related phenomena, not only chemisorption but also revaporation, leaching, long-term stability, and so on.

We have conducted preliminary tests for Cs chemisorption onto SS under varying chemical conditions of temperatures (1073 and 1273 K), atmosphere (H₂/H₂O>1) and elemental compositions in stainless steel (Mo and Si contents). It was confirmed from these tests that Cs was chemisorbed by the reaction with Si to form Fe including Cs-Si-O compounds like as that reported in the previous study. The chemisorbed amount was found to depend not only on temperature but also on atmosphere and Si contents. Of note is the implication of revaporization of Cs chemisorbed compounds from the SS surface. It can be said from this result that the Cs chemisorption behaviors should be considered in relation to chemical properties of compounds formed.

In order to model the Cs chemisorption behavior based on the above-mentioned results, the applicable base models were surveyed and two types of models were found. One is that incorporated in the present SA analysis codes; the Cs-chemisorbed amount is expressed as a function of only temperature. The other is that can treat the chemical properties of various compounds.

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The Analysis of New Burnable Absorber Using MCNP6

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A new soluble boron free small modular reactor (SBF-SMR) is required to have new BAs which may suppress the reactivity for long cycle depletion with slow depletion characteristics. In order to satisfy such requirements, a new design of a discrete burnable absorber (BA) which consists of double-layers of B₄C-Al₂O₃ material is introduced to suppress reactivity. The BA is mainly used to suppress the most of excess reactivity and it is preferable because the frequently inserting control rod would cause power distorting and change shape, which makes it difficult to operate reactors. The SBF-SMR is being designed to achieve 3 to 5 years cycle length and the maximum excess reactivity is estimated around 150 to 200 ppm with a flattest reactivity profile for entire cycle. For the analysis, the WH 17x17 fuel assembly is chosen with enrichment of 5 wt% U-235. In order to obtain the optimum BA, several sensitive studies are carried out by changing design parameters of the BA such as radius, composition, and loading number. All lattice depletion analyses are done by MCNP6 code which is a well-known Monte Carlo particle transport code and is widely used for application of the burnup dependent criticality analysis including radiation shielding analysis.
Semi-implicit Numerical Calculation of Fission Product Migration in a Tri-structural Isotropic Coated Fuel Particle

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A stiff ODE (ordinary differential equation) solver was applied to calculate the migration of fission products within a TRISO (tri-structural isotropic coated fuel particle) of a high temperature reactor (HTR). An HTR fuel element called a compact contains a very large number of TRISOs in it. An analysis on the fission product migration in a compact requires lots of fast and accurate calculations of the fission product releases from a TRISO at various locations of a compact. Five Fickian diffusion equations were set up for the five regions of a TRISO which consists of a kernel at its center and four surrounding coating layers. There were no analytical solutions even when the diffusivity of a fission product was constant. Two numerical methods were developed by applying finite element and finite difference schemes to the diffusion equations. In Euler or Crank-Nicholsen numerical method, its time increment had to be unrealistically tiny in order to maintain the numerical stability of the method. Fifth-order Runge-Kutta method with monitoring of local truncation error to ensure accuracy and adjust step size was very time-consuming. Semi-implicit extrapolation method for integrating stiff ODEs, with monitoring of local truncation error to adjust step size, was successfully used to calculate the releases of fission products, $^{110m}$Ag, $^{137}$Cs, $^{90}$Sr, and $^{85}$Kr, from a TRISO.
The Sintering Behavior of Gd$_2$O$_3$ and Characteristics of Metal Matrix Composite Reinforced with the Sintered Gd$_2$O$_3$

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Neutron absorption is one of the most important components in nuclear engineering because without absorbing neutrons, some important parts of nuclear engineering cannot exist. For example, the reactivity of nuclear a reactor cannot be controlled, protection from irradiation of neutron cannot be performed, and maintaining sub-criticality in the storage system of spent fuel become impossible.

To achieve better neutron absorption ability, finding a new material is necessary because all the current using neutron absorption materials have its shortage. Alloys containing neutron-absorbing elements, such as B, Gd, Cd have low neutron absorption ability by its solubility limit. Therefore, alloys must contain a small amount of neutron absorption element, which means limited neutron absorption ability. Boron containing polymers have also shortage such as low neutron absorption by limited contents neutron absorbing element and degradation issue. Another neutron absorber B$_4$C dispersed MMC has a better characteristic in terms of neutron absorbing ability rather than alloys and polymers because it has no contents limitation. However, it has also shortage, when the fraction of B$_4$C becomes higher, the neutron absorption ability becomes higher, but the ductility of absorber becomes lower.

A Gd$_2$O$_3$ dispersed MMC is a promising candidate for new neutron absorber because it has the advantage of B$_4$C dispersed MMC, and it can overcome the shortage of B$_4$C dispersed MMC, the brittleness issue. The brittleness of B$_4$C MMC is originated from small neutron capture cross-section so that large amount of B$_4$C should be contained in MMC to get proper neutron absorbing ability. However, Gd has 4000 times larger neutron capture cross-section, so with the small amount of Gd$_2$O$_3$, same neutron absorption ability can be achieved. In other words, 15 vol.% B$_4$C MMC has same neutron absorption ability as 1 vol.% of Gd$_2$O$_3$.

In this study, sintering behavior of Gd$_2$O$_3$ was investigated at 1600°C with various time to get sound Gd$_2$O$_3$ particle as reinforcement. Then, B$_4$C powder and Gd$_2$O$_3$ powder are mixed with aluminum powder with two fraction 1.5vol.% and 20vol.% and mixed through a 3d mixer at 40rpm for 5 hours. After that, the mixed powders are sintered for 5 minutes at 450°C and 40 MPa by using SPS. With the fabricated MMCs, Microstructure analysis was performed by SEM and hardness test and bending test was performed to analyze mechanical properties.
Electrochemical Behavior of Zirconium in Molten Chloride Coexisting Fluoride for Fuel Debris Treatment

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To develop processing nuclear fuel debris that has been caused by the nuclear accident at the Fukushima Daiichi Nuclear Power Plant with the Great East Japan Earthquake in 2011. As an on-site processing, molten salt electrolysis can be applied. This study was dedicated to using molten chloride bath. Partly fluorinated zirconium is target in this study. Electrochemical behavior of zirconium has been investigated in molten chloride with addition of zirconium fluoride. As a result, two steps reduction of zirconium was observed, according to both cyclic voltammetry and square wave voltammetry. Therefore it would be considered that zirconium fluoride was dissolved in molten chloride bath by addition of fluoride. Furthermore, reduction potential of Zr⁴⁺/Zr²⁺ was negatively shifted with increasing concentration of zirconium fluoride. This fact would be related to stability of divalent zirconium due to coexisting fluoride ions in melt bath.
Preliminary study of irradiation performance of nuclear graphite moderator/reflector in research reactor

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A nuclear graphite has been known as a typical material for a moderator and reflector due to its excellent neutronic properties such as low neutron absorption cross-section. It also has been used for a core structure material in a nuclear reactor such as VHTR. During the reactor lifetime, it is important to make the graphite structure maintained sufficiently strong and undistorted. In general, however, there is severe damage when it is exposed to high neutron fluence and temperature. Therefore, it is important to evaluate and predict the stress in graphite components for the safety assessments and estimation of core lifetime. Some studies were carried out for the case of VHTR where the operating temperature is very high (> 800˚C). In this study, we are considering the nuclear graphite to be used as a moderator and reflector in the research reactor where the operating temperature is low (< 200˚C). For the safety assessment, a preliminary study was conducted using ABAQUS FEA and its user sub-routine, UMAT. The material database was constructed from the irradiation data of IG-110. A rectangular design of graphite block canned in Al cladding was considered. The thermal, mechanical, and neutronic boundary conditions were given conservatively. The constitutive equation follows Maxwell-Kelvin model where the total strain includes the strains of elastic, thermal, irradiation induced dimensional change, and creep. The analysis was carried out up to 16 effective full power year (EFPY). The maximum temperature is in a range of 75 ˚C. The dimensional changes along x- and y-axis are only 0.02 and 0.06 cm where the gaps are 0.5 and 1.5 cm, respectively. Therefore, we could conclude that there are enough safety margins in this condition.
Laser Surface Treatment to Form an ODS Layer on Zircaloy-4

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In this study, surface treatment was performed by a laser beam to form a dispersed oxide layer in Zircaloy-4. Laser beam scanning of a tube coated with yttrium oxide ($Y_2O_3$) resulted in the formation of a dispersed oxide layer in the tube's surface region. $Y_2O_3$ particles penetrated the Zircaloy-4 during the laser treatment and were distributed uniformly in the surface region. The oxide dispersion strengthened (ODS) layer increased the mechanical strength of Zircaloy-4. The tensile strength of Zircaloy-4 increased by 10–20% with the formation of the dispersed oxide layer. The compressive yield stress also increased, by more than 15%. It is noted that the strength increase was evident with the formation of a thin dispersed oxide layer whose thickness less than 10% of that of the Zircaloy-4 substrate. The strengthening of Zircaloy-4 by the ODS layer was very effective at elevated temperature. The brittle fracture observed at room temperature was changed in ductile at elevated temperatures. The formation of ODS layer is useful for manufacturing Zr alloy tubes with enhanced safety and accident tolerance because of the increased strength up to high temperatures. In addition, the investigated technique of surface ODS layer formation is advantageous for application to final products or materials since surface treatment has the advantage that it can be performed to obtain the desired properties at the desired site.
Migration behavior of rare earth inclusions in Uranium-Zirconium alloy

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Uranium-zirconium-rare earth (U-Zr-RE) metallic fuel for a sodium-cooled fast reactor (SFR) has been developed owing to its high thermal conductivity, simple fabrication procedures, good neutron economy, excellent compatibility with a Na coolant and inherent passive safety. The metallic fuel slugs of U-Zr-RE were soundly fabricated using an injection casting method, and the results have already been reported by the Korea Atomic Energy Research Institute (KAERI), but it was usually focused on the manufacturing characterization of fuel slugs, even if the chemical composition of RE in the fuel slugs has differences in comparison with the charging amount of RE. The reason for this mismatched composition has not yet been clarified. Thus, to determine the reason, it is extremely important to understand clearly the migration behavior and formation of RE inclusions in U-Zr alloy. We determined that the melt-residues and fuel slugs of U-10Zr-xRE (x=5, 7, 10wt.%) can be characterized through various analytical methods such as the density, optical microscopy (OM), scanning electron microscopy (SEM), energy-dispersive x-ray spectroscopy (EDX), and inductively coupled plasma atomic emission spectroscopy (ICP-AES). In particular, SEM data indicate that spherical RE inclusions are formed and irregularly distributed in a U-Zr matrix. Moreover, the RE segregation layer was formed at the upper region of melt-residues of U-10Zr-7RE and U-10Zr-10RE, but not in U-10Zr-5RE. Therefore, we found that the formation of a segregation layer depends on the RE composition when the RE content is higher than 5wt.% and the RE elements are not dissolved in U-Zr alloys owing to the immiscibility in U-Zr alloys. We discussed that the formation and migration of RE inclusions on the basis of temperature gradient, density, surface energy, and thermodynamically stable state.
Development of melting and casting technology for fuel slug with high contents of rare earth

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For the development of rare-earth fuel manufacturing process, the validity of fuel fabrication feasibility by using improved injection casting technique which is developed by KAERI(Korea Atomic Energy Research Institute) is necessary. However, the evaluation of the recovery rate of the high-content Rare Earth(RE)-fuel slug, the density and the chemical analysis of the high-content rare earth specimens showed a difference between the RE loading amount and the RE content in the fuel core. It was necessary to identify the cause.

In order to investigate the cause of the decrease of the rare earth content in the fuel slug for developing the rare earth high content fuel casting technology, casting in which the rare earth content was increased to 10 wt.%, 7 wt.%, 5 wt.% was carried out to evaluate the recovery rate, density, chemical analysis, and microstructure analysis of the fuel slug and debris, and their properties were evaluated.

As a result of analysis of high content rare earth fuel residue and fuel slug, when RE content exceeding RE 5% was charged at the production of rare earth high amount fuel slug, only 5% RE was used in melt casting, and the other RE was floated and the RE rich layer was formed due to the difference in density. The RE rich layer was thicker as the rare earth charge content was larger. This layer was considered to have a larger reaction with the crucible.
Paper Title: Development of remote end-cap welding technology for SFR Fuel cladding

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The development of remote fuel fabrication technology for a sodium-cooled fast reactor (SFR) has been started that can be directly applied to a remote fabrication process of TRU fuel. The remote welding technology is one of remote fuel fabrication process that used to a fuel rod end-cap welding and fuel rods bundle assembly in a highly radioactive environment, so the direct human access is not possible. This technology can also be applied to practical use and commercialization of TRU nuclear metal fuel in near future. The purpose of this study is to find out the most suitable welding technology for the end-cap remote welding process and optimize its welding condition. In this study, candidate technologies for remote welding technology development were investigated and analyzed, and the advantages and disadvantages of each technology were analyzed. Finally, the final candidate technology was selected to investigate the welding conditions and welding results. As a candidate technology for remote welding, GTAW, friction welding, and disc / fiber laser welding technology were selected, and characteristics of each technology were analyzed and final candidate technology was selected. As a result of the technical review, it was evaluated that the laser welding technique is the most suitable for the remote welding technique. Among them, the fiber laser welding method was evaluated as the most suitable technique for the remote welding of the seal stopper. Therefore, laser welding was performed by varying the welding output and the welding linear velocity in order to set the optimal conditions for the fiber laser welding technique. The microstructural (OM) and mechanical properties (hardness test, tensile test) were evaluated for each condition, respectively. In the case of laser welding, the welding depth is reduced as the welding power is reduced to 250 W or less, and the welding is not sufficiently performed. When the welding power is increased by 250 W or more, the welding part is widened more than necessary and some oxidation occurs on the surface. In case of welding speed, if welding speed is increased by 53 rpm or more, welding is not sufficiently performed, and if it is reduced to 53 rpm or less, melting of welded part is increased. The average hardness of the welds increased about twice as much as that of the base metal, which is thought to be due to the formation of martensite structure with rapid cooling rate after welding.
Venue

Ramada Plaza Jeju

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- Hotel Robero Jeju

ANFC 2017
Ramada Plaza Jeju

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- 10 minutes by Taxi
  - Fare will be approximately 7000 KRW (won)
Ora Hall (8th Floor)
## Technical Program

### September 8 (Friday) 2017

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<td>C. M. Lee (UNIST)</td>
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<td>Y. J. Park (KAIST)</td>
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<td>Q. M. Mistarihi (KAIST)</td>
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<td></td>
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<td>A. Mohamad (Osaka Univ.)</td>
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<td></td>
<td><strong>Lunch</strong></td>
<td>1&lt;sup&gt;st&lt;/sup&gt; Floor TAMMORA</td>
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<td></td>
<td><strong>Break</strong></td>
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<tr>
<td>14:20 ~ 15:30</td>
<td>Fuel Behavior and Waste</td>
<td>M. Uno (Univ. Fukui) <em>Invited</em></td>
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<td></td>
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<td>N. Sato (Tohoku Univ.)</td>
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<td></td>
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<td>H. Matsuura (Tokyo City Univ.)</td>
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<td></td>
<td><strong>Break</strong></td>
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<tr>
<td>15:50 ~ 17:30</td>
<td>Fuel and Materials for FR and RR</td>
<td>K. Kurosaki (Osaka Univ.) <em>Invited</em></td>
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<td>S. W. Kuk (KAERI)</td>
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<td>T. W. Cho (UNIST)</td>
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<td>F. Sweidan (KAIST)</td>
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<td>G. Y. Jeong (UNIST)</td>
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<td><strong>Adjourn</strong></td>
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<tr>
<td>18:30 ~</td>
<td>Banquet</td>
<td>Ora Hall (8F)</td>
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### September 9 (Saturday) 2017

<table>
<thead>
<tr>
<th>Time</th>
<th>Session</th>
<th>Details</th>
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</thead>
<tbody>
<tr>
<td>08:00 ~ 09:00</td>
<td>Meeting</td>
<td>Collaboration Plan Discussion Meeting (ANFC Organizers)</td>
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<tr>
<td></td>
<td>Breakfast</td>
<td>Invited Speakers</td>
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<tr>
<td>09:00 ~ 12:00</td>
<td>Team Discussion</td>
<td>Students’ activity</td>
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</tbody>
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**ANFC 2017**