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# ON
# ‘ADVANCED FUEL PELLET MATERIALS AND FUEL ROD DESIGNS FOR WATER COOLED REACTORS’

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# SESSION 1

Fabrication and Design

# Innovative Process Techniques to Optimize Quality and Microstructure of UO2 Fuel for PHWRs in India

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Nuclear Fuel Complex ( NFC) is the sole manufacturer of $UO_2$ fuel for the pressurized heavy water reactors (PHWRs) in India and currently supplying fuel to PHWRs-220 MWe and PHWRs-540 MWe operating in the country. High density (94-96%TD) and defect free $UO_2$ fuel pellets are produced from $UO_2$ powder, which is derived through ammonium di-uranate (ADU) precipitate route. The PHWRs fuel pellets are fabricated using process steps such as pressure agglomeration of $UO_2$ powder, admixing of lubricant, die compaction of green pellets followed by high temperature (1700 ℃) solid state sintering of green pellets.

The in-pile performance of the collapsible clad fuel is largely dependent on pellet density, physical integrity and microstructure. The physical and chemical characteristics of $UO_2$ powder, powder morphology and pellet fabrication technique are mainly responsible for these attributes of the fuel pellets. Higher amounts of impurity anions like sulphate, carbonate, etc. in magnesium-di-uranate (MDU) raw material and larger fraction of coarser particles in mono-modal particle size distribution of $UO_2$ powder were found to be the prime reasons for low sintered density and in-homogenous microstructure in $UO_2$ sintered pellets. Nitrogen pick up was found to be predominant in low density sintered pellets having duplex grain structure having larger grains, larger pores in the core of the pellet. End chipping and end cap defect in sintered pellet was encountered for some powder lots. The pellet integrity was improved by doping 3-5 % $U_3O_8$ powder with $UO_2$ granules. Doping fine grade ammonium-di-uranate (ADU) with $UO_2$ granules is found to bring encouraging results for solving low density and microstructure related problems. An innovative technique was also developed to retrieve the acceptable density pellets from non-conforming sintering boats having sintered pellets of variable sintered density.

The paper highlights the incidences of low density, end defects and heterogeneous microstructure encountered in sintered $UO_2$ fuel pellets of certain lots and discusses the problem solving processes adopted in pellet production line.

# Studies on the Sintering Behavior of $UO_2$-$Gd_2O_3$ Fuel Pellets

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The incorporation of gadolinium directly into nuclear power reactor fuel is important from the point of reactivity compensation and adjustment of power distribution enabling thus longer fuel cycles and optimized fuel utilization. The $UO_2$-$Gd_2O_3$ poisoned fuel is being proposed to be implanted in Brazil according to the future requirements established for Angra II nuclear power plant. The incorporation of $Gd_2O_3$ powder directly into the $UO_2$ powder by dry mechanical blending is the most attractive process because of its simplicity. Nevertheless, processing by this method leads to difficulties while obtaining sintered pellets with the minimum required density. This is due to blockages during the sintering process. There is little information in published literature about the possible mechanism for this blockage and this is restricted to the hypothesis based on formation of a low diffusivity Gd rich $(U,Gd)O_2$ phase. The objective of this investigation has been to study the blockage mechanism in this system during the sintering process, contributing thus, to clarify the cause for the blockage. Experimentally it has been shown that the blocking mechanism is based on pore formation because of the Kirkendall effect, instead the formation of law diffusivity phases. The formation of a solid solution during the intermediate stage of sintering leads to formation of large pores, which are difficult to remove in the fined stage of sintering. The phenomenon is better characterized as a concurrence between formation and elimination of pores during sintering than as a sintering blockage

# Evolutionary Design Studies of PHWR Fuel Rods

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Argentina has two nuclear power plants, Embalse (CANDU 6,600 MWe) and Atucha 1 (PHWR, 357 MWe). Both reactors operate with natural uranium fuel ($UO_2$) and are moderated and cooled with pressurized heavy water.

Currently it is in its final stage the construction of a third reactor of the same type (Atucha 2, 745 MWe) and is planned to build new plants in accordance with a growing demand and the need to diversify the national energy matrix.

Fuel assemblies for these power plant are manufactured in Argentina and over the years their designs have been improved product of operational experience, fabrication evolution and technical-economic needs.

The changes in both fuel designs have been evolving in some cases, such as increasing the mass of the fuel pellets. Other changes have represented a total conversion of the nuclear reactor operation; this is the case of the use of slightly enriched uranium (0.85 %) in Atucha 1, since 2001, allowing a higher burnup and therefore an important reduction on fuel consumption.

At the present the initial design of the Atucha-2 fuel assembly is completed and the first core is currently under construction. Future evolution of this fuel might be based on the previous experience and also will consider the information from the operation.

To study the changes that could be implemented on the Atucha-2 fuel rod the effects of some design parameters have been studied. Aspects like pellet density, pellet porosity, pellet geometry, dishing volume, plenum volume, pellet-cladding gap size and initial pressurization were considered and their effects on the thermo-mechanical behavior of the entire fuel rod in terms of safety and performance were analyzed. Fuel fabrication constrains and potential improvements were also considered.

In this paper are presented the studies of the evolutionary design changes proposed for Atucha-2 fuel rod and the analysis for their implementation. Safety limits that might be affected are identified and safety margins evaluated.

# Application of Probabilistic Methods to Fuel Rod Design Evaluation

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The safety concept of nuclear power plants fuel rods relies on a system of multiple barriers to preclude the release of fuel containing uranium and/or plutonium or highly radioactive solid and gaseous fission products generated during operation of the reactor. The gas-tight welded fuel rods (FRs) represent a first and important barrier of this concept. As a consequence, FR design must demonstrate that systematic failures are precluded by applying appropriate codes and methods. In particular, this means that critical physical quantities (e.g. temperatures, strains, internal pressure) must not exceed their design respectively defect limits.

Traditional proofs in the field of nuclear engineering are based on a conservative-deterministic approach which makes use of an unfavorable combination of input parameters with respect to interesting quantities.

However, there have been significant drivers during the last years which triggered the advancement of more realistic methodologies. For instance, power uprates of plants, increased enrichments and burnups, and the pertaining challenging experience feedback from PIE and pool measurements push forward to regimes which were not preconceived when the traditional concepts in fuel rod design evaluation were established. Thus, all stakeholders of the licensing process start to reflect the potential of the traditional approaches in the light of current and future demands, i.e. in particular to improve fuel utilization and reduce the amount of radioactive waste:

- What is the degree of conservatism of the conservative-deterministic approach? Is it over-conservative, are sufficient margins left at more challenging demands? Or might there even be a lack of conservatism possible?
- Is the selection of input parameters within the deterministic approach still justified? Are there - dependent on the situation – other combinations conceivable with more unfavorable results? Is it possible to map the possible nonlinear response of state-of-the-art codes?

To answer these questions it was proposed to apply an advanced approach based on the combination of probabilistic methodologies, best-estimate codes, and a comprehensive, reliable database what allows for a controlled way to approach the technical limits by a realistic assessment ("best-estimate scenario") as well as by a conservative result with a well defined degree of conservatism in terms of an appropriate quantile. Thus, the new approach is able to fulfill all requirements of licensing scenarios and, thus, can be used as a direct licensing tool or as a benchmark for the traditional methods.

# Fuel Assembly Designs for Achieving High Burnups in 220 MWe Indian PHWRs

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Presently 19-element natural uranium fuel bundles are used in 220 MWe Indian PHWRs, The core average design discharge burnup for these bundles is 7000 MWD/TeU and maximum burnup for assembly goes up to of 15,000 MWD/TeU.

Use of slightly enriched uranium in place of natural uranium in 19-element fuel bundles, in 220 MWe PHWRs is being investigated. Due to higher fissile content these bundles will be capable of delivering higher burnup than the natural uranium bundles. The maximum burnup possible with these bundles is 30,000 MWd/TeU.

In PHWR fuel elements no plenum space is available and the cladding is of collapsible type. The additional fission product swelling and gas release due to use of SEU fuel in PHWRs, needs to be accommodated within the fuel elements taking into account these factors. Studies have been carried out for different fuel element target burnups with different alternative concepts. Modification in pellet shape and pellet parameters is considered. These studies for the PHWR fuel elements/assemblies have been elaborated in this paper.

# Research on Welding Defects of Pressure Resistance Welding for Heavy Water Reactor Fuel Element

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QS-3 HWR in China uses Canadian CANDU-6 technology, noted for 2 x 728 MW total capacity, and the design capacity factor for 85 %, life expectancy is 40 years. The project formally started on June 8, 1998. The first load supplied by ZPI (Zircatec Precision Industries Inc.), and formally put into commercial operation on December 31, 2002. In order to meet the reload demand, BNFP (Baotou Nuclear Fuel Plant) and ZPI signed CANDU-6 technology transfer contract in December 1998, the fuel plant was completed and put into operation in December 2002, the first reload made by BNFP was put into reactor on 27 March 2003. The first reload fuel assemblies manufactured by BNFP were loaded in reactor on March 27th, 2003. Until to June 30th, 2009, 56209 fuel assemblies manufactured by BNFP have been loaded, 49059 fuel assemblies have been unload and among them 19 fuel assemblies have leakage. All the leaking fuel assemblies were manufactured before 2005. From October 2005 to June 2009, there was no leakage for 45 months. It is proved that the fuel assemblies manufactured by BNFP have good performance in reactor.

This article introduces BNFP pressure resistance welding process, and classifies the defects into different types, then analyzes the root cause for the welding defects with abnormal phenomenon, by making artificial defects to simulate the defects condition to further verify the cause, and then it draws the conclusions of the major cause of the defects. During the formal production, precaution for decreasing the defects are implemented, thus gained valuable practical manufacturing experience which could improve the fuel quality and determine the direction of control and improvement for HWR fuel element end plug welding with pressure resistance welding process.

# Current Status of Research and Development of Nuclear Fuel Elements for PWR in Indonesia

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The energy need of Indonesia is increasing due to the population growth and for the economic progress. The government of Indonesia intends to apply an optimum energy mix comprising all viable prospective energy sources. The Government Regulation No. 5 year 2006 indicates the target of energy mix until 2025 and the share of nuclear energy is about 2 % of primary energy or 4 % of electricity (4000 MWe). The first two units of NPP is expected to be operated before 2020 as stated in Act No. 17 year 2007 on National Long Term Development Planning 2005-2025. The first NPP to be operated in Indonesia is PWR type with capacity of 1000 MWe/unit. One of the strategies to strength and increase national capacity in the program for NPP introduction is domestication industry for nuclear fuel. To reach this purpose, the activities of research and development is focus on nuclear fuel production technology for PWR. Currently research and development activity in Indonesia is to produce prototype of nuclear fuel element for PWR in the form of test fuel pin or mini pin. In this paper we will presenting the pelletization and fabrication technology development. The existing facility was designed for PHWR fuel element of CIRENE type. Development of pelletization technology is carried out by modifying the compacting machine. Parameters of compacting and sintering are determined based on both compressibility and compactibility of the pellet as indicated by density and mechanical strength of the $UO_2$ green pellets. The sintering parameters to be determined are temperature, heating rate, and soaking time. Currently, the fabrication process is under experiment. All of the data resulted from the experiment that will be presented in this meeting.

# SESSION 2

Advanced Fuels

# High Burnup $UO_2$ Fuel Pellets with Dopants for WWER

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The currently achieved level of design and technology developments provided for the implementation of the fuel cycle (4x1) in WWER at the maximal design burnup of 56 MWd/kgU per FA. Presently in Russia the program is under way to improve the technical and economic parameters of WWER fuel cycles characterized by an increased fuel usability.

To meet the requirements placed on the new fuel that ensures the reliable operation under conditions of higher burnups complex activities are under way to optimize the composition and microstructure of fuel pellets as applied to WWER.

This paper describes a general approach to providing the stimulated composition and microstructure of fuel via introducing various dopants. Aside from this, the paper presents the experimentally results of studies into the main technologic and operational characteristics of dopant containing fuel pellets including higher grain sizes, pores distribution and oxygen to metal ratio.

The results of the experiments made it possible to work out the pilot commercial process of the modified fuel fabrication, to manufacture pellet batches to be semi-commercially operated at NPP with WWER.

# Westinghouse Advanced Doped Pellet – Characteristics and Irradiation Behavior

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There are a number of trends in the nuclear power industry, which put additional requirements on the operational flexibility and reliability of nuclear fuel, for example power uprates and longer cycles in order to increase production, higher burnup levels in order to reduce the backend cost of the fuel cycle, and lower goals for activity release from power plant operation. These additional requirements can be addressed by increasing the fuel density, improving the FG retention, improving the PCI resistance and improving the post-failure performance. In order to achieve that, Westinghouse has developed ADOPT (Advanced Doped Pellet Technology) $UO_2$ fuel containing additions of chromium and aluminium oxides. The additives facilitate pellet densification during sintering, enlarge the pellet grain size, and increase the creep rate. The final manufactured doped pellets reach about 0.5 % higher density within a shorter sintering time and a five times larger grain size compared with standard $UO_2$ fuel pellets. Fuel rods with ADOPT pellets have been irradiated in several light water reactors (LWRs) since 1999, including two full SVEA Optima2 reloads in 2005. ADOPT pellets has been investigated in pool-side and hot cell Post Irradiation Examinations (PIEs), as well as in a ramp test and a fuel washout test in the Studsvik R2 test reactor. The investigations have identified three areas of improved operational behaviour: Reduced Fission Gas Release (FGR), improved Pellet Cladding Interaction (PCI) performance thanks to increased pellet plasticity and higher resistance against post-failure degradation. The better FGR behaviour of ADOPT has been verified with a pool side FGR gamma measurement performed at 55 MWd/kgU, as well as transient tests in the Studsvik R2 reactor. Creep measurements performed on fresh pellets show that ADOPT has a higher creep rate which is beneficial for the PCI performance. ADOPT has also been part of a high power Halden test (IFA-677). The results from this test will be presented in this paper.

# Washout Behaviour of Chromia-doped $UO_2$ and Gadolinia Fuels in LWR Environments

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Current economic pressure has forced nuclear utilities to demand more of the fuel average discharged burnups, linear heat rating increases or operating cycle extensions. This general trend is accompanied with uncompromising towards safety margins.

In this context, the AREVA chromia-doped fuel proved to bring significant margins in mitigation of Pellet-clad Interaction risk of clad failure as well as to the reduction of Fission Gas Release and induced rod internal pressure at the end of life. The optimized $Cr_2O_3$-doped characteristics provide additional fuel robustness by a reduced pellet chipping susceptibility and reliability features. This latter concerns fuel washout behaviour in case of defective rods after the occurrence of primary defects occasionally observed under operation.

This paper presents results of the testing program carried out to clarify the washout behaviour of both chromia-doped $UO_2$ and chromia-doped gadolinia fuel pellets with comparison to non-doped standard fuels. The pure reaction with oxygen has been first investigated by thermogravimetry studies under dry conditions. For a better understanding, pellets with different chromia additions and grain sizes were tested. Distinctly, increasing the matrix grain size and additionally the fuel density improved the fuel oxidation resistance. For non-doped fuels, optical examinations revealed the phenomenon proceeded by surface oxidation quickly followed by intergranular cracking and spalling of oxidized grains. On the contrary, with the AREVA large grain chromia doped fuels, the formation of an outside layer delays oxygen diffusion and decelerates the oxidation rate. In a second study, fuel corrosion behaviour was studied by means of autoclave leaching tests under representative LWR conditions. The wash-out rate of the $Cr_2O_3$-doped pellets has been shown to be reduced up to a factor of 5 in comparison to non-doped fuel types. The doped sample surface remains nearly intact following the formation of a compact surface layer.

The entire testing program clearly establishes the improved resistance to oxidation and corrosion of the AREVA optimised chromia doped $UO_2$ and gadolinia fuels. Such an enhanced operational behaviour is desired to struggle against disintegration of the fuel in case of defective rods and combined consequences to LWR primary coolant contaminations.

# Development of Burnable Neutron Absorber (BNA) Pellets

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Certain fuel designs require the presence of burnable neutron absorbers (BNA). Historically, BNA have been added to the $UO_2$ fuel meat to control reactivity in fresh fuel but in AECL's Advanced CANDU Reactor®™[†] fuel design, BNA is used to achieve a reduced coolant void reactivity. For this application it is desirable to have the BNA separate from the uranium. Zirconium oxide was chosen as a candidate inert matrix material to host the BNA due to its irradiation stability, compatibility with the BNA oxides, and material properties. This paper documents AECL's fabrication development activities and gives preliminary results of irradiation testing.

# SESSION 3

Innovative Fuel Designs

# Zirconia Inert Matrix Fuels for Plutonium and Minor Actinides Burning in Reactors

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The radio toxicity of the $UO_2$ spent fuel is dominated by plutonium and minor actinides (MA): Np, Am and Cm, after decay of the short life fission products. Zirconia ceramics containing Pu and MA in the form of an inert matrix fuel (IMF) could be used to burn these actinides in light water reactors or in high temperature reactors. Optimization of the fuel designs dictated by properties such as thermal, mechanical, chemical and physical must be performed with attention for their behavior under irradiation. Zirconia must be stabilized by yttria to form a solid solution such as $MA_zY_yPu_xZr_{1-y}O_{2-\zeta}$ where minor actinide oxides are also soluble. MA may act as a burnable poison reducing the reactivity at the beginning of life and yielding fertile nuclides improving the reactivity at the end of life. These zirconia cubic solid solutions are stable under heavy ion irradiation. The retention of fission products in zirconia, under similar thermodynamic conditions, is a priori stronger, compared to $UO_2$, the lattice parameter being larger for $UO_2$ than for $(Y,Zr)O_{2-x}$ $(Er,Y,Pu,Zr)O_{2-x}$ in which Pu contains 5% Am was successfully irradiated in the Proteus reactor at PSI, in the HFR facility, Pelten as well as in the Halden reactor. These tests support potential irradiations of such IMF in a commercial reactor. This would allow later a commercial deployment of such a zirconia fuel for Pu and MA utilization in a last cycle. The fuel forms namely pellet-fuel, cercer, cermet or coated particle fuel are discussed considering the once through strategy. For this strategy, low solubility of the inert matrix is required for geological disposal. As spent fuels these IMFs must be excellent materials from the solubility point of view, this parameter was studied in detail for a range of solutions corresponding to groundwater under near field conditions. Under these conditions the IMF solubility is about 10 times smaller than glass, which makes the zirconia material very attractive for deep geological disposal. The desired objective would be to use IMF to produce energy in reactors, opting for an economical and ecological solution.

# Designing a Thorium Fuel Irradiation Experiment

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A Norwegian company, Thor Energy, has been planning an irradiation experiment in which a thorium-plutonium oxide fuel will be tested in the Halden reactor in Norway. As part of this project a set of specifications has been prepared to govern the manufacture of the test batch of thorium-plutonium oxide fuel pellets. Composition, microstructural and shaping parameters were considered carefully – as outlined in this paper. Establishing specifications for the fuel ceramic stoichiometry and for certain impurity limits proved to be of specific importance for the thorium-plutonium test pellets.

# Current Status of a Development Project on Erbia Credit Super High Burnup Fuel

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In order to reduce the number of spent fuel assemblies and then to improve fuel economics, it is effective to develop super high burnup fuel with higher enrichment than 5 wt%. However, the current nuclear facilities are restricted with the limit of fuel enrichment 5 wt% or less at the viewpoint of criticality safety. Therefore, the large investment such as a design modification and a reconstruction, a license acquisition and remodeling each process are needed. Adding this, a decrease of handling amount of uranium results an aggravation of manufacturing efficiency as well.

To solve such issues, "Erbia credit Super High Burnup (Er-SHB) Fuel" concept is proposed by the authors. In Er credit concept, by a means of adding low content ( > 0.2 wt% ) of erbia into all $UO_2$ powder, a reactivity of high enrichment ( > 5 wt% ) fuel is suppressed under that of current fuel assemblies, i.e. below 5 wt% enrichment. Since erbia is mixed into $UO_2$ powder just after the re-conversion process, we can take the advantage of negative reactivity credit of erbia against for the criticality safety issues appearing in the front-end stream.

A development program of the Er-SHB fuel had launched in 2005, under the support project of Ministry of Economy, Trade and Industry (METI) for Innovative and Viable Nuclear Energy Technologies (IVNET). The development program covers a wide aspect of the development of LWR fuel as follows:

- Critical experiments of fully erbia loaded core whose $^{235}U$ enrichment is 5 to 10 wt%
- Development of an uncertainty reduction technique for neutronics parameters
- Criticality safety analysis using erbia credit
- Fabrication test and physicochemical properties measurement of erbia-bearing fuel pellet
- Core design using the Er-SHB fuel assemblies
- Applicability of burnup credit for the Er-SHB fuels
- Effect on the back-end stream such as disposal of high-level radioactive waste.

In this paper, (l) outline of the development project, (2) measurement and analysis results of critical experiment, (3) ECOS (Erbia Content Of Sub criticality judgment) diagram resulting from a series of criticality safety analysis are presented.

# Minimization of Inner Diametric Tolerance of Annular Pellet for Dual Cooled Fuel

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A dual cooled fuel consists of internal and external cladding tubes in which annular pellets are stacked and cooling water flows in both internal and external coolant passages. It is recently being reconsidered as a promising option for a power uprate of a pressurized water reactor fuel assembly because an annular fuel shows a lot of advantages from the point of a fuel safety and its economy due to an increased heat transfer area and a thin pellet thickness.

Many technical issues might cause a serious problem to adopt the dual cooled annular fuel to the commercial PWR reactors. One of the most important issues is a heat flux split toward an internal cladding and an external cladding due to the gap conductance asymmetry which results from a preferential expansion of a fuel pellet toward the outside during an irradiation. Gap conductance is directly related to the inner and outer gap thicknesses. Initial gap thicknesses can vary with a pellet's dimensions which are affected by a reactor operation condition. Recently, it is suggested that a fuel rod with a smaller inner gap and a larger outer gap can reduce this gap conductance asymmetry. This approach can be effective only after precise tolerance technology is achieved.

Because of an inhomogeneous green density distribution along the compact height, an hour-glassing usually occurred in a sintered cylindrical PWR fuel pellet fabricated by a conventional double-acting press. Thus, a sintered pellet usually undergoes a centerless grinding process in order to secure a pellet's specifications. In the case of an annular pellet fabrication using a conventional double-acting press, the same hour-glass shape would probably occur. The outer diameter tolerance of an annular pellet can be controlled easily similar to that of a conventional cylindrical PWR pellet through a centerless grinding. However, it appears not to be simple in the case of an inner surface grinding. It would be the best way to satisfy the specifications for the inner diameter in an as-fabricated pellet.

This paper deals with several approaches that we have tried to reduce the diametric tolerance of the sintered annular pellet. We focused on the tolerance of the inner diameter in as-sintered annular pellet without a post grinding.

# A New Uncertainty Reduction Method for Fuel Fabrication Process and PWR Cores with Erbia-bearing Fuel

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The concept of a fuel fabrication system with erbia bearing high burnup fuel has been proposed. The erbia is added to all fuel with over 5 % $^{235}$U enrichment to retain the neutronics characteristics to that within 5 % $^{235}$U enrichment. There is a problem of the prediction accuracy of the neutronics characteristics with erbia bearing fuel because of the short of experimental data of erbia bearing fuel. The purpose of the present work is to reduce the uncertainty. A new method has been proposed by combining bias factor method and the cross section adjustment method. The cross section adjustment method is applied only to erbium (Er). And the neutronics characteristic except Er is improved by the bias factor method. The main contribution of Er is neutron capture. To evaluate the accuracy of $k_{eff}$, the erbia worth is the suitable experimental data. Therefore, the prediction accuracy of neutronics characteristics of Er is improved through the cross section adjustment using the sample worth. In the case of the blending machine (H/U=0) for the fuel fabrication process, the uncertainty reduction, which shows the rate of reduction of uncertainty, of the $k_{eff}$ is 0.604 by the present method and 0.555 by the conventional bias factor method. In the case of H/U=1.0, the uncertainty reduction by the present method was 0.760. Using the bias factor method, the uncertainty reduction was 0.593. The prediction uncertainty was significantly reduced by the present method. For the PWR core, the uncertainty reduction, which shows the rate of reduction of uncertainty, of the $k_{eff}$ is 0.865 by the present method and 0.801 by the conventional bias factor method. Thus the prediction uncertainties are reduced by the present method compared to the bias factor method.

# Investigation of Liquid-metal-bonded Hydride Fuels for LWRs

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U-Zr hydride fuels have been utilized in a variety of reactor designs in the past few decades. The fuel, which consists of metallic uranium particles embedded in a zirconium hydride matrix, contains sufficient hydrogen to act in addition to water as a moderator. The presence of hydrogen in the fuel provides significant benefits; most notably the strong negative fuel-temperature reactivity feedback. As opposed to oxide fuel, the neutron flux spectrum in hydride fuel is essentially constant throughout the fuel pellet (due to the presence of moderator), which eliminates the rim effect. Benchmarked against oxide fuels with the same enrichment, hydride fuels achieve higher uranium burnups. The thermal conductivity of $U\text{-}ZrH_{1.6}$ is five times that of uranium oxide. Hydride fuels also exhibit excellent fission gas retention characteristics. In comparison with oxide fuels however, hydride fuels exhibit large swelling at low burnups if the temperature exceeds ~ 700 ℃. Zirconium-based LWR cladding is highly susceptible to hydrogen embrittlement which could occur due to proximity to hydride fuel. To mitigate this difficulty, (a) operating temperatures are kept less than 650 ℃, which does not restrict the linear heat rate during operation and (b) a fuel-cladding gap larger than that in oxide fuels that is filled with a liquid-metal alloy (lead-tin-bismuth) has been proposed to prevent fuel-cladding contact (PCMI) to the final burnup. Due to the alloy's large thermal conductivity, the liquid-metal bond eliminates the temperature drop across the gap.

The work at UC Berkeley during the previous years has been focused on addressing the need for fundamental knowledge regarding this fuel system. The studies include:

- Fabrication and characterization of U-Th-Zr hydride fuels.
- Development of codes for predicting the behaviour of hydride fuel under typical LWR steady-state and transient conditions.
- Experimental determination of the diffusion coefficient of hydrogen in Th-Zr hydrides by neutron scattering.
- Investigation of the compatibility of hydride fuels with Zircaloy cladding with and without a liquid-metal bond in the fuel-cladding gap.
- Measurement of the kinetics of hydrogen desorption and adsorption from Zr hydride at elevated temperatures.

Application for a grant has been made to irradiate mini-hydride fuel elements in the Advanced Test Reactor (INL) under typical LWR conditions.