Nuclear Fuel Behaviour Under Reactivity Initiated Accident Ria Conditions

This is an authoritative compilation of information regarding methods and data used in all phases of nuclear engineering. Addressing nuclear engineers and scientists at all levels, this book provides a condensed reference on nuclear engineering since 1958.

An analysis of fuel rod behavior during reactivity initiated accidents is presented. The calculational approach is described, predictions are discussed and compared to data from the SPERT power excursion tests, and the sensitivity of maximum cladding temperatures to initial reactor conditions, enthalpy insertions, and other experiment design parameters is investigated. 10 references. (auth).

This book presents select proceedings of the International Conference on Future Learning Aspects of Mechanical Engineering (FLAME 2018). The book discusses interdisciplinary areas such as automobile engineering, mechatronics, applied and structural mechanics, bio-mechanics, biomedical instrumentation, ergonomics, biodynamic modeling, nuclear engineering, agriculture engineering, and farm machineries. The contents of the book will benefit both researchers and professionals.

A new generation of nuclear reactor designs is being developed in order to meet the needs of the 21st century. In the short term, the most important objective is to improve competitiveness in the deregulated market. For this purpose evolutionary light water reactors are being developed and promoted actively. In the longer term, other requirements related to long-term sustainability will emerge, including the need to minimise the environmental burden passed on to future generations, the need to establish sustainability of the fuel and the need to minimise stocks of separated plutonium and their accessibility. At this workshop, information on R&D activities for advanced reactor systems was exchanged and research areas in which international co-operation could be strengthened were identified, in particular the roles that could be played by existing experimental facilities and the possible needs for new infrastructure.

This book presents the state of the art on thermophysical and thermochemical properties, fabrication methodologies, irradiation behaviours, fuel reprocessing procedures, and aspects of waste management for oxide fuels in general and for thoria-based fuels in particular. The book covers all the essential features involved in the development of and working with nuclear technology. With the help of key databases, many of which were created by the authors, information is presented in the form of tables, figures, schematic diagrams and flow sheets, and photographs. This information will be useful for scientists and engineers working in the nuclear field, particularly for design and simulation, and

for establishing the technology. One special feature is the inclusion of the latest information on thoria-based fuels, especially on the use of thorium in power generation, as it has less proliferation potential for nuclear weapons. Given its natural abundance, thorium offers a future alternative to uranium fuels in nuclear technology. In closing, the latest information on conventional uranium and plutonium fuels is also provided.

The transient in-reactor fuels testing workshop was held on May 4-5, 2009 at Idaho National Laboratory. The purpose of this meeting was to provide a forum where technical experts in transient testing of nuclear fuels could meet directly with technical instrumentation experts and nuclear fuel modeling and simulation experts to discuss needed advancements in transient testing to support a basic understanding of nuclear fuel behavior under off-normal conditions. The workshop was attended by representatives from Commissariat à l'Énergie Atomique CEA, Japanese Atomic Energy Agency (JAEA), Department of Energy (DOE), AREVA, General Electric - Global Nuclear Fuels (GE-GNF), Westinghouse, Electric Power Research Institute (EPRI), universities, and several DOE national laboratories. Transient testing of fuels and materials generates information required for advanced fuels in future nuclear power plants. Future nuclear power plants will rely heavily on advanced computer modeling and simulation that describes fuel behavior under off-normal conditions. TREAT is an ideal facility for this testing because of its flexibility, proven operation and material condition. The opportunity exists to develop advanced instrumentation and data collection that can support modeling and simulation needs much better than was possible in the past. In order to take advantage of these opportunities, test programs must be carefully designed to yield basic information to support modeling before conducting integral performance tests. An early start of TREAT and operation at low power would provide significant dividends in training, development of instrumentation, and checkout of reactor systems. Early start of TREAT (2015) is needed to support the requirements of potential users of TREAT and include the testing of full length fuel irradiated in the FFTF reactor. The capabilities provided by TREAT are needed for the development of nuclear power and the following benefits will be realized by the refurbishment and restart of TREAT. TREAT is an absolute necessity in the suite of reactor fuel test capabilities TREAT yields valuable information on reactivity effects, margins to failure, fuel dispersal, and failure propagation Most importantly, interpretation of TREAT experiment results is a stringent test of the integrated understanding of fuel performance.

Performance of Trasuranic-Loaded Fully Ceramic Micro-Encapsulated Fuel in LWRs Interim Report, Including Void Reactivity Evaluation

Handbook of Nuclear Engineering
IAEA Yearbook
Characterization of Exposure-dependent Eigenvalue Drift Using Monte Carlo Based Nuclear Fuel Management
Analysis of Fuel Behavior During Reactivity Initiated Accidents
State of the Art Report
Program approach and analysis of results
Nuclear Fuel Behaviour Under Reactivity-initiated Accident (RIA) Conditions
Nuclear Materials Science
Reactivity Initiated Accident Test Series Test RIA 1-2 Fuel Behavior Report

The "VOLGA" conferences, hosted in odd-numbered years by the Department of Theoretical and Experimental Reactor Physics of the Moscow Engineering Physics Institute (MEPhI), are some of the most prestigious technical meetings held in Russia. Traditionally, these conferences present the opportunity for reactor physicists from around the world to gather at MEPhI's holiday camp on the banks of the Volga river (near Tver) to exchange ideas and explore innovative concepts related to nuclear power development. In 1997, NATO became involved in the "VOLGA" meetings for the first time by co-sponsoring "VOLGA97" as an advanced research workshop. This workshop broke with tradition a bit in that the venue was moved from MEPhI's holiday camp to a location nearer Moscow. The workshop program was effectively organized in order to cover a broad range of topics relating to the theme of the meeting. Generally, the papers concerned safety related questions associated with utilizing both weapons-grade and reactor-grade plutonium in the nuclear fuel cycle, including facility requirements, licensing issues, proliferation risks, and a variety of advanced concepts for alternative fuel cycles. The program contained a total of ninety-nine papers presented in five days of sessions.

Dynamics and Control of Nuclear Reactors presents the latest knowledge and research in reactor dynamics, control and instrumentation; important factors in ensuring the safe and economic operation of nuclear power plants. This book provides current and future engineers with a single resource containing all relevant information, including detailed treatments on the modeling, simulation, operational features and dynamic characteristics of pressurized light-water reactors, boiling light-water reactors, pressurized heavy-water reactors and molten-salt reactors. It also provides pertinent, but less detailed information on small modular reactors, sodium fast reactors, and gas-cooled reactors. Provides case studies and examples to demonstrate learning through problem solving, including an analysis of accidents at Three Mile Island, Chernobyl and Fukushima Daiichi Includes MATLAB codes to enable the reader to apply the knowledge gained to their own projects and *research Features examples and problems that illustrate the principles of dynamic analysis as well as the mathematical tools necessary to understand and apply the analysis Publishers Note: Table 3.1 has been revised and will be included in future printings of the book with the following data: Group Decay Constant, li (sec-1) Delayed Neutron Fraction (bi) 1 0.0124 0.000221 2 0.0305 0.001467 3 0.111 0.001313 4 0.301 0.002647 5 1.14 0.000771 6 3.01 0.000281 Total delayed neutron fraction: 0.0067*

The present volume in our annual review series reviews a wide range of developments, giving a broad interpretation to the "technology" of our title. Starting at the beginning, Science, we have the review of basic nuclear physics data of Walker and Weaver for reactor kinetics, particularly, there fore, delayed neutron data. In the search for better and better accuracy, it is being realized that this involves the closest scrutiny of fundamental data, given to us here from the Birmingham school. Associated with this review of data is the review from Italy by Professor Pacilio and his co workers of the theory of reactor kinetics in the stochastic form, and a valuable compilation of the theory underlying a wide range of practical techniques. Tending more to technology come the papers by Jervis, reviewing the application of digital computers to the control of large nuclear power stations as developed in both the united Kingdom and Canada, Pickman's review of the design of fuels for heavy water reactors, and the account by Ishi kawa and Inabe of the new Japanese Research Reactor Program, itself initially directed largely to fuel element studies. The balance of the volume is made up of more philoso phical contributions to the practicalities of nuclear power.

MOX fuel, a mixture of weapon-grade plutonium and natural or depleted uranium, may be used to deplete a portion of the world's surplus of weapon-grade plutonium. A number of reactors currently operate in Europe with one-third MOX cores, and others are scheduled to begin using MOX fuels in both Europe and Japan in the near future. While Russia has laboratory-scale MOX fabrication facilities, the technology remains under study. No fuels containing plutonium are used in the U.S. The 25 presentations in this book give an impressive overview of MOX technology. The following issues are covered: an up to date report on the disposition of ex-weapons Pu in Russia; an analysis of safety features of MOX fuel configurations of different reactor concepts and their operating and control measures; an exchange of information on the status of MOX utilisation in existing power plants, the fabrication technology of various MOX fuels and their behaviour in practice; a discussion of the typical national approaches by Russia and the western countries to the utilisation of Pu as MOX fuel; an introduction to new ideas, enhancing the disposition option of MOX fuel exploitation and destruction in existing and future advanced reactor systems; and the identification of common research areas where defined tasks can be initiated in cooperative partnership.

High-performance alloys that can withstand operation in hazardous nuclear environments are critical to *presentday in-service reactor support and maintenance and are foundational for reactor concepts of the future. With commercial nuclear energy vendors and operators facing the retirement of staff during the coming decades, much of the scholarly knowledge of nuclear materials pursuant to appropriate, impactful, and safe usage is at risk. Led by the multi-award winning editorial team of G. Robert Odette (UCSB) and Steven J. Zinkle (UTK/ORNL) and with contributions from leaders of each alloy discipline, Structural Alloys for Nuclear Energy Applications aids the next generation of researchers and industry staff developing and maintaining steels, nickel-base alloys, zirconium alloys, and other structural alloys in nuclear energy applications. This authoritative reference is a critical acquisition for institutions and individuals seeking state-of-the-art knowledge aided by the editors' unique personal insight from decades of frontline research, engineering and management. Focuses on in-service irradiation, thermal, mechanical, and chemical performance capabilities. Covers the use of steels and other structural alloys in current fission technology, leading edge Generation-IV fission reactors, and future fusion power reactors. Provides a critical and comprehensive review of the state-of-the-art experimental knowledge base of reactor materials, for applications ranging from engineering safety and lifetime assessments to supporting the development of advanced computational models.*

The Linear Reactivity Model (LRM) is a simple nuclear fuel management model that comes with a diskette containing three programs. Consisting of a collection of algorithms and methods, the LRM describes complex core behavior, but it is simpler than the complex programs developed for design calculations. This makes the LRM particularly useful as a teaching tool to explain the basic principles of nuclear fuel management. The LRM mainly focuses on the pressurized water reactor, but it is also directly applicable to the boiling water reactor. Application of the LRM to the CANDU reactor is also covered.

Light water reactor fuel response during reactivity initiated accident experiments
Quarterly Progress Report on the Nuclear Safety Research Reactor (NSRR) Experiments (1), October 1975 - March 1976
Technical Review
Select Proceedings of FLAME 2018
Phase 1
Future Transient Testing of Advanced Fuels
Nuclear Safety
Thoria-based Nuclear Fuels
Advances in Nuclear Fuel
Improving Fuel Cycle Design and Safety Characteristics of a Gas Cooled Fast Reactor

The ability to accurately predict the multiplication factor (k eff) of a nuclear reactor core as a function of exposure continues to be an elusive task for core designers despite decades of advances in computational methods. The difference between a predicted eigenvalue (target) and the actual eigenvalue at critical reactor conditions is herein referred to as the "eigenvalue drift." This dissertation studies exposure-dependent eigenvalue drift using MCNP-based fuel management analysis of the ORNL High Flux Isotope Reactor core. Spatial-dependent burnup is evaluated using the MONTEBURNS and ALEPH codes to link MCNP to ORIGEN to help analyze the behavior of k eff as a function of fuel exposure. Understanding the exposure-dependent eigenvalue drift of a nuclear reactor is of particular relevance when trying to predict the impact of major design changes upon fuel cycle behavior and length. In this research, the design of an advanced HFIR core with a fuel loading of 12 kg of 235 U is contrasted against the current loading of 9.4 kg. The goal of applying exposure dependent eigenvalue characterization is to produce a more accurate prediction of the fuel cycle length than prior analysis techniques, and to improve our understanding of the reactivity behavior of the core throughout the cycle. This investigation predicted a fuel cycle length of 40 days, representing a 50% increase in the cycle length in response to a 25% increase in fuel loading. The average burnup increased by about 48 MWd/kg U and it was confirmed that the excess reactivity can be controlled with the present design and arrangement of control elements throughout the core's life. Another major design change studied was the effect of installing an internal beryllium reflector upon cycle length. Exposure dependent eigenvalue predictions indicate that the actual benefit could be twice as large as that originally assessed via beginning-of-life (BOL) analyses.

During the last century, nuclear power has been established as a reliable source of energy in the major industrialised countries. It has recently enjoyed a revival in attention and research due to the environmental concerns surrounding current conventional energy sources. Issues of regulation and safety are at the forefront of all discussions involving nuclear power, and will govern its place in the future. The Future of Nuclear Power takes a technical and comprehensive look at the current and future status of nuclear power throughout the world. The 17 chapters are divided into two main sections: a review of all current generation plants, and concepts for new advanced reactor design and safety. The broad-ranging topics covered by this publication, coupled with the current revival of interest in nuclear energy, make it a timely reference for all nuclear scientists. Reviews the issues surrounding the future operation of existing commercial nuclear plants Several chapters dedicated to the extensive research programs in place concerning safe and reliable operation Compares nuclear and non-nuclear options for energy needs in the future; evaluating the benefits and risks of both

This thesis reports a series of investigations examining the corrosion process of used nuclear fuel under permanent disposal conditions. The motivation of the project is that the safety assessment of deep geological disposal of spent nuclear fuel requires a fundamental understanding of the processes controlling fuel corrosion which could lead to the release of radionuclides to the geosphere from a failed container. One primary objective of this project was to develop a computational model in order to simulate fuel corrosion under the disposal conditions. The mathematical model was developed using COMSOL Multiphysics based on the finite element method. The chemical engineering module and the diluted species transportation module of the software are suitable for the simulations required. Literature research of the model development on the radiation-induced spent fuel corrosion revealed many key features required in modelling radiolytic corrosion (in particular for -radiation). These features were incorporated into the model presented in the thesis along with the recently available kinetics data and mechanisms. Evaluation of different model setups and sensitivity tests of different parameters were performed. A series of simulations were designed and developed to determine the influence of redox conditions, with the emphasis on -radiolysis and steel vessel corrosion products, on the corrosion rate of spent fuel. The model presented in the thesis takes into account the -radiolysis of water, the reaction of radiolytic H_2O_2 with UO_2 both directly and via galvanic coupling with noble metal particles, the reaction with H_2 via galvanic coupling, the Fenton reaction and other redox reactions involving H_2O_2 and H_2. The calculated fuel corrosion rate is very sensitive to $[Fe2]_{bulk}$ produced by corrosion of the steel vessel. When the $[Fe^{2+}]_{bulk}$ is greater than 4.2 mol L^{-1} even the radiolytically produced H_2 alone can suppress fuel corrosion without assistance from external H_2 for CANDU fuel with an age of 1000 years or larger. The ability of H_2 to suppress fuel corrosion is shown to be sensitive to fuel burnup (density of noble metal fission products) and a complete suppression of corrosion can be achieved at bulk H_2 concentrations in the order of 0.1 mol L^{-1}. This approach is 1-dimensional and considers only the corrosion of a planar fuel surface. It will act as a preliminary step in the eventual development of 2-D and 3-D models involving the customized geometries necessary to account for the fractured nature of the spent fuel and the complex fuel bundle geometry. A second objective of this project was to develop a more detailed understanding of the H_2O_2 decomposition process and its influence on UO_2 corrosion. Several variables (potential, pH, carbonate/bicarbonate, and fission products) can influence the reactivity of H_2O_2. Their influence on the surface composition and electrical conductivity of UO_2 will affect surface redox reaction rates and significantly alter the overall fuel corrosion rate. Electrochemical methods were used to separate a corrosion reaction into its two constituent half reactions allowing the determination of the rate dependence on potential for each half reaction. The primary electrochemical techniques used were cyclic voltammetry (CV) to examine a system in general, cathodic stripping voltammetry (CSV) to determine the consequences of a period of oxidation, corrosion potential (E CORR) measurements to monitor redox conditions, linear polarization resistance (LPR) measurements to calculate corrosion rates, and electrochemical impedance spectroscopy (EIS) measurements to monitor changes in uranium oxide film properties. Since the changes in surface condition also have a significant impact on the H_2O_2 reactivity, the surface/solution analytical techniques were used to link the electrochemical/chemical processes to the compositional and structural changes observed on a UO_2 surface. These techniques included scanning electron microscopy (SEM) to analyze surface morphologies, X-ray photoelectron spectroscopy (XPS) to etermine the oxidation states of UO_2 surface, and inductively coupled plasma atomic emission spectroscopy (ICP-AES) to measure the dissolved U in solutions. In this thesis, the mechanisms of H_2O_2 decomposition on fuel surface and the consequent effect on UO_2 dissolution have been investigated under various conditions (pH, carbonate/bicarbonate). At the lower pH values both the anodic oxidation and decomposition reactions are almost completely blocked by a thin surface layer of U VI oxide. At higher pH this layer becomes more soluble and anodic oxidation occurs on the sublayer of $U^{IV}_{1-2x}U^{V}_{2x}O_{2+x}$, but is partially controlled by transport through a permeable, chemically dissolving UVI oxide/hydroxide layer. At positive electrode potent ial, approximately 70% of the anodic current is consumed by H_2O_2 oxidation the remaining 30% going to produce soluble UO_2^{2+}. At higher pH values peroxide decomposition occurs on an unblocked $U^{IV}_{1-2x}U^{V}_{2x}O_{2+x}$ surface and the pH dependence of the reaction suggests HO_2▯ is the electroactive form of peroxide. The anodic behaviour of simulated nuclear fuel (SIMFUEL) in solutions containing H_2O_2 and HCO_3▯/CO_3^2▯ has been studied electrochemically and using surface analytical techniques, in particular XPS. Two anodic reactions are possible, the oxidative dissolution of UO_2 and H_2O_2 oxidation. The rate of both reactions is controlled by the chemical release of U VI surface species, and the rates can both be increased by the addition of HCO_3▯/CO_3^2▯. Under anodic conditions the dominant reaction is H_2O_2 oxidation, although UO_2 dissolution may also be accelerated by the formation of a uranylperoxycarbonate complex. Similarly, under open circuit (corrosion) conditions both UO_2 corrosion and H_2O_2 decomposition are also controlled by the rate of release of U VI surface species which blocks access of H_2O_2 to the underlying conductive $U^{IV}_{1-2x}U^{V}_{2x}O_{2+x}$ surface. A series of electrochemical experiments has been conducted on SIMFUEL electrodes containing different dopants with the primary purpose of determining the relative importance of the UO_2 and ▯-particle surfaces in the balance between UO_2 oxidation/dissolution and H_2O_2 decomposition. On the electrode containing both rare earth elements and noble metal particles, the anodic current is increased at high potentials, which is absence on the electrode containing only rare earth elements. The direct anodic oxidation of H_2O_2 occurs on -particles is interpreted at high potentials, making H_2O_2 oxidation the dominant reaction, the UO_2 surface being partially blocked by the presence of U VI surface species.

Presents brief descriptions of 20 fuel-related safety criteria along with both the rationale for having such criteria and possible new design and operational issues which could have an effect on them.

Worldwide there are more than 430 nuclear power plants operating and more plants are being constructed or planned for construction. For nuclear power to be sustainable the nuclear fuel must be sustainable and there should be adequate nuclear fuel waste management program. Continuous technological advances will lead towards sustainable nuclear fuel through closed fuel cycles and advance fuel development. This focuses on challenges and issues that need to be addressed for better performance and safety of nuclear fuel in nuclear plants. These focused areas are on development of high conductivity new fuels, radiation induced corrosion, fuel behavior during abnormal events in reactor, and decontamination of radioactive material.

The Hanford Spent Nuclear Fuel Project focuses its efforts on determining how to safely move the degraded N-Reactor spent fuel from water-stored basins to a dry storage facility. Based on the laboratory data, the project chose to use a conservative enhancement factor in analyzing the oxidation behavior of the spent metallic fuel. However, there is a need for the project to increase the fuel throughput for the drying treatment process by implementing certain design optimization steps. The study discussed in this paper re-evaluated the previous laboratory data in conjunction with the cold vacuum drying (CVD) process experience and determined whether the built-in level of conservatism could accommodate the potential changes in the process without compromising public and worker safety. An established oxidation reaction-rate constant was used to accurately determine the reactive surface areas of corroded N-Reactor fuel elements. The surface areas calculated for 6 different N-Reactor elements that were stored in the K-West Basin and shipped to Pacific Northwest National Laboratory for drying studies ranges from as low as 0.0018 m2 for a broken element to 8.1 m2 for a highly corroded SNF element 5744U. The SNF element 0309M that was a clean broken piece was used to calibrate the calculation method. The result using the SNF reaction rate constant (i.e., kSNF) gave a very good (i.e., 0.0018 m2) agreement with the geometrical value of 0.0015 m2. Having established that the hydrogen generation can be used to determine the exposed surface area of these irregular corroded SNF elements, the calculations was extended to provide a good estimate of the exposed uranium surface area of SNF elements loaded into the multi-canister overpacks (MCOs).

In the late 1980s, the National Cancer Institute initiated an investigation of cancer risks in populations near 52 commercial nuclear power plants and 10 Department of Energy nuclear facilities (including research and nuclear weapons production facilities and one reprocessing plant) in the United States. The results of the NCI investigation were used a primary resource for communicating with the public about the cancer risks near the nuclear facilities. However, this study is now over 20 years old. The U.S. Nuclear Regulatory Commission requested that the National Academy of Sciences provide an updated assessment of cancer risks in populations near USNRC-licensed nuclear facilities that utilize or process uranium for the production of electricity. Analysis of Cancer Risks in Populations near Nuclear Facilities: Phase 1 focuses on identifying scientifically sound approaches for carrying out an assessment of cancer risks associated with living near a nuclear facility, judgments about the strengths and weaknesses of various statistical power, ability to assess potential confounding factors, possible biases, and required effort. The results from this Phase 1 study will be used to inform the design of cancer risk assessment, which will be carried out in Phase 2. This report is beneficial for the general public, communities near nuclear facilities, stakeholders, healthcare providers, policy makers, state and local officials, community leaders, and the media.

Calculations have been performed to assess the neutronic behavior of pins of Fully-Ceramic Micro-encapsulated (FCM) fuel in otherwise-conventional Pressurized Water Reactor (PWR) fuel pins. The FCM fuel contains transuranic (TRU)-only oxide fuel in tri-isotropic (TRISO) particles with the TRU loading coming from the spent fuel of a conventional LWR after 5 years of cooling. Use of the TRISO particle fuel would provide an additional barrier to fission product release in the event of cladding failure. Depletion calculations were performed to evaluate reactivity-limited burnup of the TRU-only FCM fuel. These calculations showed that due to relatively little space available for fuel, the achievable burnup with these pins alone is quite small. Various reactivity parameters were also evaluated at each burnup step including moderator temperature coefficient (MTC), Doppler, and soluble boron worth. These were compared to reference UO2 and MOX unit cells. The TRU-only FCM fuel exhibits degraded MTC and Doppler coefficients relative to UO2 and MOX. Also, the reactivity effects of coolant voiding suggest that the behavior of this fuel would be similar to a MOX fuel of very high plutonium fraction, which are known to have positive void reactivity. In general, loading of TRU-only FCM fuel into an assembly without significant quantities of uranium presents challenges to the reactor design. However, if such FCM fuel pins are included in a heterogeneous assembly alongside LEU fuel pins, the overall reactivity behavior would be dominated by the uranium pins while attractive TRU destruction performance levels in the TRU-only FCM fuel pins is. From this work, it is concluded that use of heterogeneous assemblies such as these appears feasible from a preliminary reactor physics standpoint.

Concerns around global warming have led to a nuclear renaissance in many countries. Meanwhile the nuclear industry is already warning of a need to train more nuclear engineers and scientists who are needed in a range of areas from healthcare and radiation detection to space exploration and advanced materials, as well as for the nuclear power industry. Here Karl Whittle provides a solid overview of the intersection of nuclear engineering and materials science at a level approachable by advanced students from materials, engineering and physics. The text explains the unique aspects needed in the design and implementation of materials for use in demanding nuclear settings. In addition to material properties and their interaction with radiation, the book covers a range of topics including reactor design, fuels, fusion, future technologies and lessons learned from past incidents. Accompanied by problems, videos and teaching aids the book is suitable for a course text in nuclear materials and a reference for those already working in the field.

Results are presented from preliminary tests designed to investigate the behavior of preirradiated fuel rods under reactivity initiated accident (RIA) conditions. The tests were conducted in 1970 as part of the SPERT/Capsule Driver Core (CDC) program. The report was intended to be published in a series of Idaho Nuclear Corporation Interim Technical Reports (IN-ITRs); however, the CDC program was terminated before the report could be released. In September 1975, the Nuclear Regulatory Commission concluded that the data contained in the report could be a valuable reference in planning future water reactor safety program tests and requested its release.

Experimental results from six recent Power Burst Facility (PBF) reactivity initiated accident (RIA) tests are compared with data from previous Special Power Excursion Reactor Test (SPERT), and Japanese Nuclear Safety Research Reactor (NSRR) tests. The RIA fuel behavior experimental program recently started in the PBF is being conducted with coolant conditions typical of hot-startup conditions in a commercial boiling water reactor. The SPERT and NSRR test programs investigated the behavior of single or small clusters of light water reactor (LWR) type fuel rods under approximate room temperature and atmospheric pressure conditions in capsules containing stagnant water. As observed in the SPERT and NSRR tests, energy deposition, and consequent enthalpy increase in the PBF test fuel, appears to be the single most important variable. However, the consequences of failure at boiling water hot-startup system conditions appear to be more severe than previously observed in either the stagnant capsule SPERT or NSRR tests. Metallographic examination of both previously unirradiated and irradiated PBF fuel rod cross sections revealed extensive variation in cladding wall thicknesses (involving considerable plastic flow) and fuel shattering along grain boundaries in both restructured and unrestructured fuel regions. Oxidation of the cladding resulted in fracture at the location of cladding thinning and disintegration of the rods during quench. In addition, swelling of the gaseous and potentially volatile fission products in previously irradiated fuel resulted in volume increases of up to 180% and blockage of the coolant channels within the flow shrouds surrounding the fuel rods.

" The Generation IV Forum is an international nuclear energy research initiative aimed at developing the fourth generation of nuclear reactors, envisaged to enter service halfway the 21st century. One of the Generation IV reactor systems is the Gas Cooled Fast Reactor (GCFR), the subject of study in this thesis. The Generation IV reactor concepts should improve all aspects of nuclear power generation. Within Generation IV, the GCFR concept specifically targets sustainability of nuclear power generation. The Gas Cooled Fast Reactor core power density is high in comparison to other gas cooled reactor concepts. Like all nuclear reactors, the GCFR produces decay heat after shut down, which has to be transported out of the reactor under all circumstances. The layout of the primary system therefore focuses on using natural convection Decay Heat Removal (DHR) where possible, with a large coolant fraction in the core to reduce friction losses. "

Nuclear Fuel Safety Criteria
Progress in Nuclear Energy
Data Base on the Behavior of High Burnup Fuel Rods with Zr-1% Nb Cladding and UO2 Fuel (VVER Type) Under Reactivity Accident Conditions
Advances in Nuclear Science and Technology
Behavior of Fuel with Zirconium Alloy Cladding in Reactivity-Initiated Accident and Loss-of-Coolant Accident
Reactive Behavior of K-Basin Spent Nuclear Fuel
Effects of Burnup on Fuel Failure. Power Burst Tests on Fuel Rods with 13,000 and 32,000 MWd/MTU Burnup
Nuclear Materials for Fission Reactors
The Linear Reactivity Model for Nuclear Fuel Management
Advanced Reactors with Innovative Fuels

Materials in a nuclear environment are exposed to extreme conditions of radiation, temperature and/or corrosion, and in many ca the combination of these makes the material behavior very different from conventional materials. This is evident for the four major technological challenges the nuclear technology domain is facing currently: (i) long-term operation of existing Generation II nuclea power plants, (ii) the design of the next generation reactors (Generation IV), (iii) the construction of the ITER fusion reactor in Cadarache (France), (iv) and the intermediate and final disposal of nuclear waste. In order to address these challenges, engineer and designers need to know the properties of a wide variety of materials under these conditions and to understand the underlying processes affecting changes in their behavior, in order to assess their performance and to determine the limits of operation. Comprehensive Nuclear Materials 2e provides broad ranging, validated summaries of all the major topics in the field of nuclear material research for fission as well as fusion reactor systems. Attention is given to the fundamental scientific aspects of nuclear materials: fuel and structural materials for fission reactors, waste materials, and materials for fusion reactors. The articles are written at a level that allows undergraduate students to understand the material, while providing active researchers with a ready reference resource of information. Most of the chapters from the first Edition have been revised and updated and a significant number of new topics are covered in completely new material. During the ten years between the two editions, the challenge for applications of nuclear materials has been significantly impacted by world events, public awareness, and technological innovation Materials play a key role as enablers of new technologies, and we trust that this new edition of Comprehensive Nuclear Materials has captured the key recent developments. Critically reviews the major classes and functions of materials, supporting the selectio assessment, validation and engineering of materials in extreme nuclear environments Comprehensive resource for up-to-date an authoritative information which is not always available elsewhere, even in journals Provides an in-depth treatment of materials modeling and simulation, with a specific focus on nuclear issues Serves as an excellent entry point for students and researchers to the field

This volume brings together 47 papers from scientists involved in the fabrication of new nuclear fuels, in basic research of nuclea materials, their application and technology as well as in computer codes and modelling of fuel behaviour. The main emphasis is o progress in the development of non-oxide fuels besides reporting advances in the more conventional oxide fuels. The two current performed large reactor safety programmes CORA and PHEBUS-FP are described in invited lectures. The contributions review basic property measurements, as well as the present state of fuel performance modelling. The performance of today's nuclear fue hence UO_2, at high burnup is also reviewed with particular emphasis on the recently observed phenomenon of grain subdivision the cold part of the oxide fuel at high burnup, the so-called "rim" effect. Similar phenomena can be simulated by ion implantation i order to better elucidate the underlying mechanism and reviews on high resolution electron microscopy provide further informatio The papers will provide a useful treatise of views, ideas and new results for all those scientists and engineers involved in the spe questions of current nuclear waste management.

Along with our colleagues at the Japan Atomic Energy Research Institute (JAERI) and Japan Atomic Energy Agency (JAEA), we have performed extensive research programs for more than 2 decades and have developed a better understanding of fuel behav under accident conditions. Data and findings from these research programs have provided the technical basis directly and indirec for regulatory criteria in Japan and other countries. This paper reviews and summarizes the major outcomes from the research programs performed at JAERI and JAEA facilities and identifies further research needs. The programs aim to generate data and models to (1) evaluate the adequacy of present safety criteria and safety margins; (2) create a database for the future regulation higher-burnup UO2 and mixed-oxide (MOX) fuels, new cladding, and pellets; (3) provide adequately mechanistic computer codes regulatory application; (4) promote a better understanding of phenomena that appear in high-burnup regions and MOX fuels, such as rim effect and an effect of plutonium agglomerates in MOX fuels, and to evaluate those effects on fuel behavior in accident conditions; and (5) to assess fuel behaviors with higher fuel duty, such as plant power uprates, longer operating cycles, and wate chemistry changes. The programs comprise reactivity-initiated accident studies, including pulse-irradiation experiments in the Nuclear Safety Research Reactor, cladding mechanical tests, and the development and verification of RANNS; loss-of-coolant accident tests, including integral thermal shock tests, oxidation rate measurements, and cladding mechanical tests; development verification of a computer code FEMAXI-6 that simulates high-burnup fuel behavior under normal operating and abnormal transie conditions; and studies on phenomena specific to high-burnup regions, including thermal property measurements, experimental a analytical simulation of rim structure formation, cladding mechanical tests on effects of hydrides, etc.

The current focus of the Deep Burn Project is on once-through burning of transuranice (TRU) in light water reactors (LWRs). The fuel form is called Fully-Ceramic Micro-encapsulated (FCM) fuel, a concept that borrows the tri-isotropic (TRISO) fuel particle design from high-temperature reactor technology. In the Deep Burn LWR (DB-LWR) concept, these fuel particles would be press into compacts using SiC matrix material and loaded into fuel pins for use in conventional LWRs. The TRU loading comes from the spent fuel of a conventional LWR after 5 years of cooling. Unit cell calculations have been performed using the DRAGON-4 code order assess the physics attributes of TRU-only FCM fuel in an LWR lattice. Depletion calculations assuming an infinite lattice condition were performed with calculations of various reactivity coefficients performed at each step. Unit cells containing typical UO2 and MOX fuel were analyzed in the same way to provide a baseline against which to compare the TRU-only FCM fuel. Load of TRU-only FCM fuel into a pin without significant quantities of uranium challenges the design from the standpoint of several key reactivity parameters, particularly void reactivity, and to some degree, the Doppler coefficient. These unit cells, while providing an indication of how a whole core of similar fuel would behave, also provide information of how individual pins of TRU-only FCM fuel would influence the reactivity behavior of a heterogeneous assembly. If these FCM fuel pins are included in a heterogeneous assembly with LEU fuel pins, the overall reactivity behavior would be dominated by the uranium pins while attractive TRU destruction performance of the TRU-only FCM fuel pins may be preserved. A configuration such as this would be similar to CONF assemblies analyzed in previous studies. Analogous to the plutonium content limits imposed on MOX fuel, some amount of TRU- FCM pins in an otherwise-uranium fuel assembly may give acceptable reactivity performance. Assembly calculations will be performed in future work to explore the design options for heterogeneous assemblies of this type and their impact on reactivity coefficients.

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