Lead Cooled Fast Neutron Reactor Brest Nikiet

ALEXIS JANELLE

Vol. 1: Nuclear Engineering Fundamentals; Vol. 2: Reactor Design; Vol. 3: Reactor Analysis; Vol. 4: Reactors of Generations III and IV; Vol. 5: Fuel Cycles, Decommissioning, Waste Disposal and Safeguards

International Atomic Energy Agency

For the first time a book has been written on the technological and scientific knowledge, acquired during, buiding , operation and even dismantling of the Superphenix plant. This reactor remains today the most powerful sodium fast breeder reactor operated in the world.(1200 MWe). The last fast breeder reactor operated in the world is BN 800 in Russia that reached his nominal power (800 MWe) in 2016. Joel Guidez began his career in the field of sodium-cooled fast reactors after leaving Ecole Centrale-Paris, in 1973. He has held various positions at Cadarache, Phenix and Superphenix, including as the head of the thermal hydraulic laboratory conducting tests for Phenix, Superphenix and the EFR European Fast Reactor project. He was also head of the OSIRIS research reactor, located at SACLAY, and of the HFR European Commission reactor, located in the Netherlands and spent two years as nuclear attaché at the French embassy in Berlin. His 2012 book "Phenix: the experience feedback" was translated into English and republished in 2013, and this new book on Superphenix is in the same spirit of thematic analysis of a reactor experience feedback. Gérard Prêle graduated from the Ecole Centrale-Lyon and entered EDF and the field of sodium-cooled fast reactors in 1983. In 1985 he joined Superphenix, where he was a duty engineer and was later in charge of safety. He has held various positions at Superphenix and Phenix and was a fast neutron reactor (SFR) engineer at the EDF Centre Lyonnais d'Ingénierie (CLI). He worked as Safety Security Environment and Radiation Protection Mission head in Superphenix at the beginning of dismantling and then in the field of PWR for two years. Since 2006 he has been involved in the Gen IV and the SFR/Astrid projects. Today, as an SFR/system and operations expert, one of his major roles is assisting the CEA in the preliminary design of the ASTRID reactor.

Generation of Nuclear Data for Lead-cooled Fast Reactors Using the Monte Carlo Method Elsevier

Thermal Hydraulics of Water-Cooled Nuclear Reactors reviews flow and heat transfer phenomena in nuclear systems and examines the critical contribution of this analysis to nuclear technology development. With a strong focus on system thermal hydraulics (SYS TH), the book provides a detailed, yet approachable, presentation of current approaches to reactor thermal hydraulic analysis, also considering the importance of this discipline for the design and operation of safe and efficient water-cooled and moderated reactors. Part One presents the background to nuclear thermal hydraulics, starting with a historical perspective, defining key terms, and considering thermal hydraulics requirements in nuclear technology. Part Two addresses the principles of thermodynamics and relevant target phenomena in nuclear systems. Next, the book focuses on nuclear thermal hydraulics modeling, covering the key areas of heat transfer and pressure drops, then moving on to an introduction to SYS TH and computational fluid dynamics codes. The final part of the book reviews the application of thermal hydraulics in nuclear technology, with chapters on V&V and uncertainty in SYS TH codes, the BEPU approach, and applications to new reactor design, plant lifetime extension, and accident analysis. This book is a valuable resource for academics, graduate students, and professionals studying the thermal hydraulic analysis of nuclear power plants and using SYS TH to demonstrate their safety and acceptability. Contains a systematic and comprehensive review of current approaches to the thermal-hydraulic analysis of water-cooled and moderated nuclear reactors Clearly presents the relationship between system level (top-down analysis) and component level phenomenology (bottom-up analysis) Provides a strong focus on nuclear system thermal hydraulic (SYS TH) codes Presents detailed coverage of the applications of thermal-hydraulics to demonstrate the safety and acceptability of nuclear power plants

Sodium Fast Reactors with Closed Fuel Cycle CRC Press

Molten Salt Reactors is a comprehensive reference on the status of molten salt reactor (MSR) research and thorium fuel utilization. There is growing awareness that nuclear energy is needed to complement intermittent energy sources and to avoid pollution from fossil fuels. Light water reactors are complex, expensive, and vulnerable to core melt, steam explosions, and hydrogen explosions, so better technology is needed. MSRs could operate safely at nearly atmospheric pressure and high temperature, yielding efficient electrical power generation, desalination, actinide incineration, hydrogen production, and other industrial heat applications. Coverage includes: Motivation -- why are we interested? Technical issues – reactor physics, thermal hydraulics, materials, environment, ... Generic designs -- thermal, fast, solid fuel, liquid fuel, ... Specific designs – aimed at electrical power, actinide incineration, thorium utilization, ... Worldwide activities in 23 countries Conclusions This book is a collaboration of 58 authors from 23 countries, written in cooperation with the International Thorium Molten Salt Forum. It can serve as a reference for engineers and scientists, and it can be used as a textbook for graduate students and advanced undergrads. Molten Salt Reactors is the only complete review of the technology currently available, making this an essential text for anyone reviewing the use of MSRs and thorium fuel, including students, nuclear researchers, industrial engineers, and policy makers. Written in cooperation with the International Thorium Molten-Salt Forum Covers MSR-specific issues, various reactor designs, and discusses issues such as the environmental impact, non-proliferation, and licensing Includes case studies and examples from experts across the globe

Storage and Hybridization of Nuclear Energy Woodhead Publishing

" 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. "

Thermal-Hydraulics of Water Cooled Nuclear Reactors Woodhead Publishing

Nuclear Reactor Technology Development and Utilization presents the theory and principles of the most common advanced nuclear reactor systems and provides a context for the value and utilization of nuclear power in a variety of applications both inside and outside a traditional nuclear setting. As countries across the globe realize their plans for a sustainable energy future, the need for innovative nuclear reactor design is increasing, and this book will provide a deep understanding of how these technologies can aid in a region’s goal for clean and reliable energy. Dr Khan and Dr Nakhabov, alongside their team of expert contributors, discuss a variety of important topics, including nuclear fuel cycles, plant decommissioning and hybrid energy systems, while considering a variety of diverse uses such as nuclear desalination, hydrogen generation and radioisotope production. Knowledge acquired enables the reader to conduct further research in academia and industry, and apply the latest design, development, integration, safety and economic guidance to their work and research. Combines reactor fundamentals with a contemporary look at evolving trends in the design of advanced reactors and their application to both nuclear and non-nuclear uses Analyses the latest research and uses of hybrid systems which bring together nuclear technology with renewable energy technologies Presents applications, economic factors and an analysis of sustainability factors in one comprehensive resource

Fundamentals of Nuclear Reactor Physics Woodhead Publishing

This report discusses the status of Lead-Cooled Fast Reactor (LFR) research and development carried out during the first half of FY 2008 under the U.S. Department of Energy Generation IV Nuclear Energy Systems Initiative. Lead-Cooled Fast Reactor research and development has recently been transferred from Generation IV to the Reactor Campaign of the Global Nuclear Energy Partnership (GNEP). Another status report shall be issued at the end of FY 2008 covering all of the LFR activities carried out in FY 2008 for both Generation IV and GNEP. The focus of research and development in FY 2008 is an initial investigation of a concept for a LFR Advanced Recycling Reactor (ARR)

Technology Pilot Plant (TPP)/demonstration test reactor (demo) incorporating features and operating conditions of the European Lead-cooled SYstem (ELSY) (almost equal to) 600 MWe lead (Pb)-cooled LFR preconceptual design for the transmutation of waste and central station power generation, and which would enable irradiation testing of advanced fuels and structural materials. Initial scoping core concept development analyses have been carried out for a 100 MWt core composed of sixteen open-lattice 20 by 20 fuel assemblies largely similar to those of the ELSY preconceptual fuel assembly design incorporating fuel pins with mixed oxide (MOX) fuel, central control rods in each fuel assembly, and cooled with Pb coolant. For a cycle length of three years, the core is calculated to have a conversion ratio of 0.79, an average discharge burnup of 108 MWd/kg of heavy metal, and a burnup reactivity swing of about 13 dollars. With a control rod in each fuel assembly, the reactivity worth of an individual rod would need to be significantly greater than one dollar which is undesirable for postulated rod withdrawal reactivity insertion events. A peak neutron fast flux of 2.0 x 1015 (n/cm2-s) is calculated. For comparison, the 400 MWt Fast Flux Test Facility (FFTF) achieved a peak neutron fast flux of 7.2 x 1015 (n/cm2-s) and the initially 563 MWt PHENIX reactor attained 2.0 x 1015 (n/cm2-s) before one of three intermediate cooling loops was shut down due to concerns about potential steam generator tube failures. The calculations do not assume a test assembly location for advanced fuels and materials irradiation in place of a fuel assembly (e.g., at the center of the core); the calculations have not examined whether it would be feasible to replace the central assembly by a test assembly location. However, having only fifteen driver assemblies implies a significant effect due to perturbations introduced by the test assembly. The peak neutron fast flux is low compared with the fast fluxes previously achieved in FFTF and PHENIX. Furthermore, the peak neutron fluence is only about half of the limiting value (4 x 1023 n/cm2) typically used for ferritic steels. The results thus suggest that a larger power level (e.g., 400 MWt) and a larger core would be better for a TPP based upon the ELSY fuel assembly design and which can also perform irradiation testing of advanced fuels and materials. In particular, a core having a higher power level and larger dimensions would achieve a suitable average discharge burnup, peak fast flux, peak fluence, and would support the inclusion of one or more test assembly locations. Participation in the Generation IV International Forum Provisional System Steering Committee for the LFR is being maintained throughout FY 2008. Results from the analysis of samples previously exposed to flowing lead-bismuth eutectic (LBE) in the DELTA loop are summarized and a model for the oxidation/corrosion kinetics of steels in heavy liquid metal coolants was applied to systematically compare the calculated long-term (i.e., following several years of growth) oxide layer thicknesses of several steels.

Technologies for Separations and Transmutation IOS Press

Thermal Hydraulics Aspects of Liquid Metal cooled Nuclear Reactors is a comprehensive collection of liquid metal thermal hydraulics research and development for nuclear liquid metal reactor applications. A deliverable of the SESAME H2020 project, this book is written by top European experts who discuss topics of note that are supplemented by an international contribution from U.S. partners within the framework of the NEAMS program under the U.S. DOE. This book is a convenient source for students, professionals and academics interested in liquid metal thermal hydraulics in nuclear applications. In addition, it will also help newcomers become familiar with current techniques and knowledge. Presents the latest information on one of the deliverables of the SESAME H2020 project Provides an overview on the design and history of liquid metal cooled fast reactors worldwide Describes the challenges in thermal hydraulics related to the design and safety analysis of liquid metal cooled fast reactors Includes the codes, methods, correlations, guidelines and limitations for liquid metal fast reactor thermal hydraulic simulations clearly Discusses state-of-the-art, multi-scale techniques for liquid metal fast reactor thermal hydraulics applications

Nuclear Power Warts and All Springer Science & Business Media

This collection presents papers from the 149th Annual Meeting & Exhibition of The Minerals, Metals & Materials Society.

Polonium Extraction Techniques for a Lead-bismuth Cooled Fast Reactor CreateSpace

This book covers the fundamentals of thermodynamics required to understand electrical power generation systems, honing in on the application of these principles to nuclear reactor power systems. It includes all the necessary information regarding the fundamental laws to gain a complete understanding and apply them specifically to the challenges of operating nuclear plants. Beginning with definitions of thermodynamic variables such as temperature, pressure and specific volume, the book then explains the laws in detail, focusing on pivotal concepts such as enthalpy and entropy, irreversibility, availability, and Maxwell relations. Specific applications of the fundamentals to Brayton and Rankine cycles for power generation are considered in-depth, in support of the book’s core goal- providing an examination of how the thermodynamic principles are applied to the design, operation and safety analysis of current and projected reactor systems. Detailed appendices cover metric and English system units and conversions, detailed steam and gas tables, heat transfer properties, and nuclear reactor system descriptions.

Plentiful Energy Springer Nature

The LEADER project goal is to improve and develop a scaled demonstrator of the LFR technology, ALFRED. The work in this thesis is focused in the ALFRED project framework and its mission is to obtain few-group cross section data for LFRs. Cross sections The neutron transport problem is crucial in nuclear engineering and nuclear reactor physics. Neutron transport theory and the diffusion theory applied to neutron reactions are briefly described, including their principles and hypothesis. The two different computational approaches to solve the neutron transport problem are summarized. The software used to obtain the data is based on a modification of the Monte Carlo method. Thus, some basic probability theory concepts are introduced. This section follows with the discussion of the Monte Carlo method and its principles, and how it can be applied to solve the neutron transport problem. Afterwards, the Serpent code is explained, as well as its features and characteristics. The process of creating a 2-dimension model of ALFRED fuel assembly and the elaboration of Serpent input files are detailed. Cross section data for five neutron energy groups and at different material temperatures is obtained by running several simulations using Serpent. The last section includes a brief description of LFR technology and some specific ALFRED features. Some advantages and disadvantages of LFRs are included, along with some proposals to solve the disadvantages. The last part of this section illustrates the proposed ALFRED core scheme, using the data publicly available to date.

Improving Fuel Cycle Design and Safety Characteristics of a Gas Cooled Fast Reactor Handbook of Generation IV Nuclear Reactors

An history on the development of Nuclear Power, types of reactors, fission theory and a detailed look at how nuclear accidents happened. This book covers; NUCLEAR POWER EARLY DEVELOPMENT details the contributions of noteworthy scientist: THE ATOM details the forces with the Atom: RADIOACTIVITY describes the types of radiation: how it is measured and different sources: NUCLEAR REACTIONS describes Fusion and Fission, how to increase rate of fission by moderation and enrichment. Describes enrichment techniques: HOW RADIATION EFECTS THE HUMAN BODY describes how cancer occurs by effects on chromosomes discuses natural sources of radiation Relative Biological Effectiveness, Radiation Hormesis, Neoplasm: TYPES OF NUCLEAR REACTORS; Describes different types of Thermal Reactors, and Fast Reactors including Pressure Water Reactors, Gas Cooled Reactors and advance variant, Water Cooled Water Moderated Power Reactor (WWER), Pressurised Heavy Water Reactors, Boiling Water Reactors and advanced variant, Pebble Bed Reactors, Aquesous Homogeneous Reactors, Fast Breeder Reactors, Liquid Metal Fast Breeder Reactors, Sodium Cooled Reactors, Lead Cooled Reactors. Development Stage Reactors these include Integral Fast Reactors (IFR), High Temperature Gas Cooled Reactor, Small Sealed Transportable Autonumous Reactor, Clean and Environmentally Safe Advanced Reactor. Design Stage Reactors, Reduced Moderation Water Reactor, Hydrogen Moderated Self Regulating Nuclear Power Module, Subcritical Reactors, Energy Amplifier, Thorium-based Reactors, Advanced Heavy Water Reactor, Kalpakkarn Mini.DESIGN FACTORS FOR AGR AND PWR discusses fuel, coolant, moderators control rods, chemical compatability, fuel clad.EFFECTS OF REACTIVITY discusses measurement of irradiation, isotopic changes in the fuel, change in Fuel from Uranium to Plutonium and its conversion ratio, refuelling options. Reactor poisons Xenon, Samarium Cadmium, Europium, Gaddolinium, Krypton and Technetium and their significance. TEMPERATURE EFFECTS considers fuel and moderator coefficients. REACTOR CONTROL Describes the various types of control rods i.e grey, coarse, safety and the requirement of superariculated rods in case of distotion. Back up control methods include the use of Nitrogen. Reactivity faults are desribed the protection methods and measurement. GETTING THE RIGHT NEUTRON TO MODERATOR BALANCE including the required level of enrichment. EFFECTS OF REACTOR COMPOSITION ON REACTIVITY : HOMOGENEOUS AND HETEROGENEOUS REACTOR affects of size and shape on neutron leakage, flux distribution in various shaped reactors, Jo Berssel Envelope: RADIAL FLUX, CHANNEL POWER AND REACTION POWER Channel power variations, alterating reactivity by enrichment, neutron absorbtion, differential irradiation, nuetron reflection. NEUTRON KINETICS Delta K, International reactivity units, Neutron Multiplication, Effective neutron multiplication factor, Delayed neutron lifetime, Codd & Wells Table. OPERATIONAL VALUES OF DELTA K Explin terms defining excess reactivity resulting doubling times then prompt criticality, defines operational limits on Delta K: REACTIVITY BALANCES, Built in reactivity, Reactivity Build Up, Xenon, Control Rods: ALL NUCLEAR SITES WORLD WIDE a list of peacefull nuclear power site that are operational, under construction or shutdown: NUCLEAR FUEL TRANSPORTATION AND DISPOSAL OF WASTE Spent fuel and fission by-products, Diffinition of Low, Intermediate, and High levels of waste, waste storage, Details of storage facilities world wide, nuclear reprocessing plants world wide, vitrification, plutomium oxide, storage flasks A, B and C, transportation of radioactive substances, bespoke sea transportation, transport flask tests, UK NUCLEAR SAFETY RECORD Windscale Fire: NUCLEAR CATASTROPHIES Three mile island meltdown containment, Why chernoble's reactor went prompt critical , Chernobyl Investigatin Conclusions, Fukushima triple meltdown, Acheiving optimum nuclear safety, WANO:

Thermodynamics In Nuclear Power Plant Systems National Academies Press

The compatibility of structural materials, such as steels with lead and lead-bismuth eutectic, poses a critical challenge in the development of heavy liquid metal (HLM) cooled fast reactors. Factors such as the high temperatures, fast neutron flux and irradiation exposure and corrosiveness provide a severe environment for the materials in these advanced reactor systems. The compatibility of liquid coolant with structural materials is critical for the development of innovative nuclear energy systems. To understand the current status of the research and development in this area as well as to provide a forum to exchange information on structural materials for HLM cooled reactors at the national and international levels, the IAEA organized a technical meeting. This resulted in the current publication which presents the summaries of the technical and the group sessions, conclusions and recommendations, and the papers presented at the event.

Experimental Facilities in Support of Liquid Metal Cooled Fast Neutron Systems KIT Scientific Publishing

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.

Fractional-Order Models for Nuclear Reactor Analysis Academic Press

Since 2002, the Department of Energy's (DOE's) Generation IV Nuclear Energy Systems (Gen IV) Program has addressed the research and development (R & D) necessary to support next-generation nuclear energy systems. The six most promising systems identified for next-generation nuclear energy are described within this roadmap. Two employ a thermal neutron spectrum with coolants and temperatures that enable hydrogen or electricity production with high efficiency (the Supercritical Water Reactor-SCWR and the Very High Temperature Reactor-VHTR). Three employ a fast neutron spectrum to enable more effective management of actinides through recycling of most components in the discharged fuel (the Gas-cooled Fast Reactor-GFR, the Lead-cooled Fast Reactor-LFR, and the Sodium-cooled Fast Reactor-SFR). The Molten Salt Reactor (MSR) employs a circulating liquid fuel mixture that offers considerable flexibility for recycling actinides and may provide an alternative to accelerator-driven systems. At the inception of DOE's Gen IV program, it was decided to significantly pursue five of the six concepts identified in the Gen IV roadmap to determine which of them was most appropriate to meet the needs of future U.S. nuclear power generation. In particular, evaluation of the highly efficient thermal SCWR and VHTR reactors was initiated primarily for energy production, and evaluation of the three fast reactor concepts, SFR, LFR, and GFR, was begun to assess viability for both energy production and their potential contribution to closing the fuel cycle. Within the Gen IV Program itself, only the VHTR class of reactors was selected for continued development. Hence, this document will address the multiple activities under the Gen IV program that contribute to the development of the VHTR. A few major technologies have been recognized by DOE as necessary to enable the deployment of the next generation of advanced nuclear reactors, including the development and qualification of the structural materials needed to ensure their safe and reliable operation. The focus of this document will be the overall range of DOE's structural materials research activities being conducted to support VHTR development. By far, the largest portion of material's R & D supporting VHTR development is that being performed directly as part of the Next-Generation Nuclear Plant (NGNP) Project. Supplementary VHTR materials R & D being performed in the DOE program, including university and international research programs and that being performed under direct contracts with the American Society for Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, will also be described. Specific areas of high-priority materials research that will be needed to deploy the NGNP and provide a basis for subsequent VHTRs are described, including the following: (1) Graphite: (a) Extensive unirradiated materials characterization and assessment of irradiation effects on properties must be performed to qualify new grades of graphite for nuclear service, including thermo-physical and mechanical properties and their changes, statistical variations from billot-to-billot and lot-to-lot, creep, and especially, irradiation creep. (b) Predictive models, as well as codification of the requirements and design methods for graphite core supports, must be developed to provide a basis for licensing. (2) Ceramics: Both fibrous and load-bearing ceramics must be qualified for environmental and radiation service as insulating materials. (3) Ceramic Composites: Carbon-carbon and SiC-SiC composites must be qualified for specialized usage in selected high-temperature components, such as core stabilizers, control rods, and insulating covers and ducting. This will require development of component-specific designs and fabrication processes, materials characterization, assessment of environmental and irradiation effects, and establishment of codes and standards for materials testing and design requirements. (4) Pressure Vessel Steels: (a) Qualification of short-term, high-temperature properties of light water reactor steels for anticipated VHTR off-normal conditions must be determined, as well as the effects of aging on tensile, creep, and toughness properties, and on thermal emissivity. (b) Large-scale fabrication process for higher temperature alloys, such as 9Cr-1MoV, including ensuring thick-section and weldment integrity must be developed, as well as improved definitions of creep-fatigue and negligible creep behavior. (5) High-Temperature Alloys: (a) Qualification and codification of materials for the intermediate heat exchanger, such as Alloys 617 or 230, for long-term very high-temperature creep, creep-fatigue, and environmental aging degradation must be done, especially in thin sections for compact designs, for both base metal and weldments. (b) Constitutive models and an improved methodology for high-temperature design must be developed.

Thermal Hydraulics Aspects of Liquid Metal Cooled Nuclear Reactors Woodhead Publishing

The construction of nuclear power plants in the United States is stopping, as regulators, reactor manufacturers, and operators sort out a host of technical and institutional problems. This volume summarizes the status of nuclear power, analyzes the obstacles to resumption of construction of nuclear plants, and describes and evaluates the technological alternatives for safer, more economical reactors. Topics covered include Institutional issues--including regulatory practices at the federal and state levels, the growing trends toward greater competition in the generation of electricity, and nuclear and nonnuclear generation options. Critical evaluation of advanced reactors--covering attributes such as cost, construction time, safety, development status, and fuel cycles. Finally, three alternative federal research and development programs are presented.

Updated Generation IV Reactors Integrated Materials Technology Program Plan, Revision 2 Woodhead Publishing

Physics of Nuclear Reactors presents a comprehensive analysis of nuclear reactor physics. Editors P. Mohanakrishnan, Om Pal Singh, and Kannan Umasankari and a team of expert contributors combine their knowledge to guide the reader through a toolkit of methods for solving transport equations, understanding the physics of reactor design principles, and developing reactor safety strategies. The inclusion of experimental and operational reactor physics makes this a unique reference for those working and researching nuclear power and the fuel cycle in existing power generation sites and experimental facilities. The book also includes radiation physics, shielding techniques and an analysis of shield design, neutron monitoring and core operations. Those involved in the development and operation of nuclear reactors and the fuel cycle will gain a thorough understanding of all elements of nuclear reactor physics, thus enabling them to apply the analysis and solution methods provided to their own work and research. This book looks to future reactors in development and analyzes their status and challenges before providing possible worked-through solutions. Cover image: Kaiga Atomic Power Station Units 1 – 4, Karnataka, India. In 2018, Unit 1 of the Kaiga Station surpassed the world record of continuous operation, at 962 days. Image courtesy of DAE, India. Includes methods for solving neutron transport problems, nuclear cross-section data and solutions of transport theory Dedicates a chapter to reactor safety that covers mitigation, probabilistic safety assessment and uncertainty analysis Covers experimental and operational physics with details on noise analysis and failed fuel detection

Technical and Scientific Achievements Springer

The Integral Fast Reactor (IFR) is a fast reactor system developed at Argonne National Laboratory in the decade 1984 to 1994. The IFR project developed the technology for a complete system; the reactor, the entire fuel cycle and the waste management technologies were all included in the development program. The reactor concept had important features and characteristics that were completely new and fuel cycle and waste management technologies that were entirely new developments. The reactor is a “fast” reactor – that is, the chain reaction is maintained by “fast” neutrons with high energy – which produces its own fuel. The IFR reactor and associated fuel cycle is a closed system. Electrical power is generated, new fissile fuel is produced to replace the fuel burned, its used fuel is processed for recycling by pyroprocessing – a new development – and waste is put in final form for disposal. All this is done on one self-sufficient site.The scale and duration of the project and its funding made it the largest nuclear energy R and D program of its day. Its purpose was the development of a long term massive new energy source, capable of meeting the nation's electrical energy needs in any amount, and for as long as it is needed, forever, if necessary. Safety, non-proliferation and waste toxicity properties were improved as well, these three the characteristics most commonly cited in opposition to nuclear power.Development proceeded from success to success. Most of the development had been done when the program was abruptly cancelled by the newly elected Clinton Administration. In his 1994 State of the Union address the president stated that “unnecessary programs in advanced reactor development will be terminated.” The IFR was that program.This book gives the real story of the IFR, written by the two nuclear scientists who were most deeply involved in its conception, the development of its R and D program, and its management.Between the scientific and engineering papers and reports, and books on the IFR, and the non-technical and often impassioned dialogue that continues to this day on fast reactor technology, we felt there is room for a volume that, while accurate technically, is written in a manner accessible to the non-specialist and even to the non-technical reader who simply wants to know what this technology is.

Fundamentals, Types, and Benefits Explained Woodhead Publishing

This book is a complete update of the classic 1981 FAST BREEDER REACTORS textbook authored by Alan E. Waltar and Albert B. Reynolds, which , along with the Russian translation, served as a major reference book for fast reactors systems. Major updates include transmutation physics (a key technology to substantially ameliorate issues associated with the storage of high-level nuclear waste), advances in fuels and materials technology (including metal fuels and cladding materials capable of high-temperature and high burnup), and new approaches to reactor safety (including passive safety technology), New chapters on gas-cooled and lead-cooled fast spectrum reactors are also included. Key international experts contributing to the text include Chaim Braun, (Stanford University) Ronald Omberg, (Pacific Northwest National Laboratory, Massimo Salvatores (CEA, France), Baldev Raj, (Indira Gandhi Center for Atomic Research, India) , John Sackett (Argonne National Laboratory), Kevan Weaver, (TerraPower Corporation) ,James Seinicki(Argonne National Laboratory). Russell Stachowski (General Electric), Toshikazu Takeda (University of Fukui, Japan), and Yoshitaka Chikazawa (Japan Atomic Energy Agency).

Nuclear Wastes Woodhead Publishing

This report provides an update on development of a pre-conceptual design for the Small Secure Transportable Autonomous Reactor (SSTAR) Lead-Cooled Fast Reactor (LFR) plant concept and supporting research and development activities. SSTAR is a small, 20 MWe (45 MWt), natural circulation, fast reactor plant for international deployment concept incorporating proliferation resistance for deployment in non-fuel cycle states and developing nations, fissile self-sufficiency for efficient utilization of uranium resources, autonomous load following making it suitable for small or immature grid applications, and a high degree of passive safety further supporting deployment in developing nations. In FY 2006, improvements have been made at ANL to the pre-conceptual design of both the reactor system and the energy converter which incorporates a supercritical carbon dioxide Brayton cycle providing higher plant efficiency (44 %) and improved economic competitiveness. The supercritical CO2 Brayton cycle technology is also applicable to Sodium-Cooled Fast Reactors providing the same benefits. One key accomplishment has been the development of a control strategy for automatic control of the supercritical CO2 Brayton cycle in principle enabling autonomous load following over the full power range between nominal and essentially zero power. Under autonomous load following operation, the reactor core power adjusts itself to equal the heat removal from the reactor system to the power converter through the large reactivity feedback of the fast spectrum core without the need for motion of control rods, while the automatic control of the power converter matches the heat removal from the reactor to the grid load. The report includes early calculations for an international benchmarking problem for a LBE-cooled, nitride-fueled fast reactor core organized by the IAEA as part of a Coordinated Research Project on Small Reactors without Onsite Refueling; the calculations use the same neutronics computer codes and methodologies applied to SSTAR. Another section of the report details the SSTAR safety design approach which is based upon defense-in-depth providing multiple levels of protection against the release of radioactive materials and how the inherent safety features of the lead coolant, nitride fuel, fast neutron spectrum core, pool vessel configuration, natural circulation, and containment meet or exceed the requirements for each level of protection. The report also includes recent results of a systematic analysis by LANL of data on corrosion of candidate cladding and structural material alloys of interest to SSTAR by LBE and Pb coolants; the data were taken from a new database on corrosion by liquid metal coolants created at LANL. The analysis methodology that considers penetration of an oxidation front into the alloy and dissolution of the trailing edge of the oxide into the coolant enables the long-term corrosion rate to be extracted from shorter-term corrosion data thereby enabling an evaluation of alloy performance over long core lifetimes (e.g., 30 years) that has heretofore not been possible. A number of candidate alloy specimens with special treatments or coatings which might enhance corrosion resistance at the temperatures at which SSTAR would operate were analyzed following testing in the DELTA loop at LANL including steels that were treated by laser peening at LLNL; laser peening is an approach that alters the oxide-metal bonds which could potentially improve corrosion resistance. LLNL is also carrying out Multi-Scale Modeling of the Fe-Cr system with the goal of assisting in the development of cladding and structural materials having greater resistance to irradiation.

Development, Validation, and Application Academic Press

Handbook of Generation IV Nuclear Reactors presents information on the current fleet of Nuclear Power Plants (NPPs) with water-cooled reactors (Generation III and III+) (96% of 430 power reactors in the world) that have relatively low thermal efficiencies (within the range of 32 36%) compared to those of modern advanced thermal power plants (combined cycle gas-fired power plants – up to 62% and supercritical pressure coal-fired power plants – up to 55%). Moreover, thermal efficiency of the current fleet of NPPs with water-cooled reactors cannot be increased significantly without completely different innovative designs, which are Generation IV reactors. Nuclear power is vital for generating electrical energy without carbon emissions. Complete with the latest research, development, and design, and written by an international team of experts, this handbook is completely dedicated to Generation IV reactors. Presents the first comprehensive handbook dedicated entirely to generation IV nuclear reactors Reviews the latest trends and developments Complete with the latest research, development, and design information in generation IV nuclear reactors Written by an international team of experts in the field