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Scientific Basis for Nuclear Waste Management XXXVII

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PREFACE

The 37th International Symposium on the Scientific Basis for Nuclear Waste Management (Materials Research Society Symposium Proceedings Volume 1665) was held in Barcelona (Catalonia, Spain), September 30–October 3, 2013. The symposium was officially opened by Dr. Antoni Gurgui, commissioner of Consejo Seguridad Nuclear (Nuclear Safety Council) in Spain. About 80 attendees from 12 countries listened to 51 presentations and discussed 29 posters during the three and a half days of scientific sessions.

The symposium covered the following topics: national and international programs; performance assessment/geological disposal; radionuclide solubility, speciation, sorption and migration; corrosion studies of zircaloy, container and carbon steel; high level waste and ceramic and advanced materials.

Special posters highlighted the First Nuclides EC project and the EC Pooled Facilities called TALISMAN.

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December 2013

Acknowledgments

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Treatment of Irradiated Graphite to Meet Acceptance Criteria for Waste Disposal: Problem and Solutions

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ABSTRACT

An overview is given of an International Atomic Energy Agency Coordinated Research Project (CRP) on the treatment of irradiated graphite (i-graphite) to meet acceptance criteria for waste disposal. Graphite is a unique radioactive waste stream, with some quarter-million metric tons worldwide eventually needing to be disposed of. The CRP has involved 24 organizations from 10 Member States. Innovative and conventional methods for i-graphite characterization, retrieval, treatment and conditioning technologies have been explored in the course of this work, and offer a range of options for competent authorities in individual Member States to deploy according to local requirements and regulatory conditions.

INTRODUCTION

Graphite is a porous, chemically inert material, highly conductive and resistant to corrosion, in general retaining its properties after exposure to an intense radiation field and at high temperatures. It does, however, undergo structural changes as a result of exposure to fast neutrons, resulting in dimensional change of components and significant changes in their mechanical and physical properties. In addition, certain impurities, together with the 1.1% of ^{13}C naturally present in the graphite, become activated through interactions with slow neutrons and thus present a significant radiation hazard post-exposure which must be accommodated in subsequent dismantling and disposal. Finally, in reactors where the graphite has been exposed to an oxidising coolant, some degree of oxidation of the material induced by the ionizing radiation field will have taken place, potentially affecting its strength.

Graphite is used in reactors as a neutron moderator and reflector, a structural material, and a fuel-element matrix material. It has been deployed in about 250 uranium- (or UO_2)-graphite reactors such as the United Kingdom Magnox and Advanced Gas-Cooled Reactors (AGR), the French UNGG, a small number of high-temperature reactors (HTRs), the Soviet-era RBMKs, and in numerous 'production' reactors and materials-testing reactors. Most of those reactors are now quite old, with many already shutdown.

The ability to dismantle and remove graphite stacks has already been demonstrated in a small number of different reactor designs *e.g.* at Fort St. Vrain (prototype HTR) in the USA, the air-cooled Brookhaven research reactor, also in the USA, and the GLEEP research reactor and Windscale prototype AGR in the UK. The resulting irradiated graphite waste (often referred to within the industry as *i-graphite*) then awaits acceptable treatment and disposal solutions whilst