FOREWORD

All fast reactor fuel assemblies and heat exchangers have rod bundle geometry. Fluid flow and heat transfer in a rod bundle and complex phenomena and basic understanding of these phenomena is essential for developing designs that optimize performance during normal operating conditions and that maintain structural integrity during abnormal operation.

Extensive experimental and analytical studies on liquid metal fluid flow distribution and heat transfer in fuel pin and heat exchanger rod-bundles have been performed in several countries with fast reactor programmes (notably in France, Germany, India, Japan, the Russian Federation, the United Kingdom and the United States of America) over the past decades. The validity of the computer codes and design approaches was proven by comparison of code results with measured velocity, pressure and temperature distributions in rod-bundles cooled/heated by liquid metal, usually sodium.

Considerable experimental and theoretical studies on various aspects of LMFR thermohydraulics have been done at the Institute of Physics and Power Engineering (IPPE), Obninsk, Russian Federation. The IAEA's International Working Group on Fast Reactors (IWGFR) recommended that IPPE should generalize its thermohydraulic studies as well as other countries' results that have been published in journals and in proceedings of international meetings.

This report was prepared in response to the recommendation from IWGFR and includes the methodology and philosophy of the analytical and experimental investigations in their application to the core and heat exchanger thermohydraulic design of LMFRs.

The IAEA officer responsible for this work was A.A. Rinejski of the Division of Nuclear Power.

INTRODUCTION

The information given in this report is concerned with liquid metal fast breeder reactors, some of which are in operation (France, Japan, Russian Federation), others under construction. Comprehensive thermal hydraulic research applied to such reactors has been carried out in the SSC IPPE.

It should be noted that liquid metal reactors can now be considered as the most safe and promising type of reactor in nuclear power engineering. However, in order for fast reactors to be widely used in industry it is necessary to improve their economic indicators, as well as to make their performance more reliable. This requires analysis of the problems of the reactor performance under deviations of the subassembly geometry from nominal (deformation), as well as under the transient conditions to be resolved, as well as various accident situations.

The problems noted above can be resolved by both experiments and predictions. The information accumulated must be processed, firstly for steady state operating conditions with the ensuing extension to the transient behavior.

The authors believe that the material given below will be useful for the further advanced thermal hydraulic analysis of fast reactors in the situations closely approximating the in-pile conditions.

The codes developed up to now are rated in order of governing equations, calculation procedures, system of closing relations and their accuracy. However, all modern codes use the momentum and energy transport factors, heat transfer coefficients, pressure drop and others to close relation between velocity and temperature fields. These factors should be evaluated with the use of relationships presented here, so they have been derived on processing the data gained in the special experiments.

Detailed analysis and rating of the thermal hydraulic codes which have been developed in the IPPE, as applied to reactor subassembly and heat exchanger, are given in this publication. Also, results of experimental investigations and numerical modelling of velocity and temperature distribution are illustrated. Experience of fast reactor performance (Russian Federation, France, Kazakhstan) suggests that subassemblies are subject to great deformation in campaigns associated with swelling and creeping. The authors have obtained considerable experimental data on temperature behaviour in deformed bundles, that have allowed this data to be generalized and used in the complex code TEMP-MIF. Considerable attention has been given to the problems of the equalization of temperature behaviour in the fast reactor subassembly, for which purpose an influence of various parameter is analysed.

On the whole, the authors demonstrate advanced approaches, new experimental techniques and numerical procedures, hydraulic and thermal constants required for the thermal physical validation of liquid metal reactor core and heat exchanger.