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THERMOHYDRAULIC RELATIONSHIPS FOR ADVANCED WATER COOLED REACTORS AND THE ROLE OF THE IAEA

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Abstract

Under the auspices of the International Atomic Energy Agency (IAEA) a Coordinated Research Program (CRP) on Thermohydraulic Relationships for Advanced Water-Cooled Reactors was carried out from 1995-1998. It was included into the IAEA's Programme following endorsement in 1995 by the International Working Group on Advanced Technologies for Water Cooled Reactors. The overall goal was to promote international information exchange and cooperation in establishing a consistent set of thermohydraulic relationships that are appropriate for use in analyzing the performance and safety of advanced water-cooled reactors.

Methods

The IAEA through CRPs provides a framework for international collaboration among institutes in industrialized and developing countries, typically 3 to 5 years in duration. The IAEA requires a significant contribution from participants in a CRP, e.g., experimental results, analytical efforts, and assignment of guest researchers of team.

The CRP on Thermohydraulic Relationships for Advanced Water-Cooled Reactors collected and peer reviewed relationships for critical heat flux (CHF), post-CHF heat transfer, pressure drops under low flow and low-pressure conditions. The CRP participants have combined databases, where possible, to prepare relationships for use in predicting these phenomena, and have provided information on the validation of these prediction methods. Organizations participating in the CRP have provided their experimental data to augment the database of the International Nuclear Safety Centre (INSC) at Argonne National Laboratories, which can be accessed at http://www.insc.anl.gov/thrmhydr/chf. The database includes: (1) look-up table for Critical Heat Flux (CHF) in 8-mm tubes, (2) CHF databank for VVER reactor applications, (3) look-up table for the post-dryout (PDO) heat transfer in tubes, (4) look-up table for CHF in VVER rod bundles, (5) CHF data from low power and low flow experiments.

Results

The final report of the CRP will provide: (1) a summary of important and relevant thermohydraulic phenomena for advanced water cooled reactors on the basis of previous work by the international community, (2) a state-of-the-art review and a recommended prediction method for the critical heat flux, which has been established through international co-operation and assessed within this CRP, (3) a state-of-the-art review and three methods for predicting film boiling heat transfer coefficients developed by institutes participating in this CRP, (4) a compilation of relevant pressure drop prediction methods and an assessment of these relations and the resulting recommendations, (5) a discussion on a methodology to select the range of interest for parameters affecting CHF, film boiling and pressure drop in advanced water cooled reactors, and (6) concluding remarks on the relationships referred to in (2)-(4) above and comments on future research needs in thermohydraulics of advanced water cooled reactors.

Discussion

The nuclear community has developed thermohydraulic codes for predicting the performance of water-cooled reactors under normal, transient and accident conditions. These codes are used for plant design, evaluation of safety margin, establishment of emergency procedures and operator training. The performance of these codes is dependent on the accuracy and consistency of the thermohydraulic relationships and thermophysical properties data contained in the codes. Extensive validation programmes have been carried out to demonstrate the applicability of the codes to plants, e.g., experimental data have been extensively compared with code predictions including International Standard Problems of the OECD Committee on the Safety of Nuclear Installations (CSNI) and IAEA standard problem exercises.

The objectives of the CRP were (1) to systematically list the requirements for thermohydraulic relationships in support of advanced water cooled reactors during normal and accident conditions, and provide details of their data base where possible and (2) to recommend and document a consistent set of thermohydraulic relationships for selected thermohydraulic phenomena such as CHF and post-CHF heat transfer and pressure drop. Hence this expanded the previous knowledge of these thermohydraulic phenomena by providing prediction methods having a wider range of validity including geometries being considered for Advanced Water-Cooled reactors.

Key collaborative activities of the participating institutes within the CRP include: (1) preparation of internationally peer reviewed and accepted prediction methods for CHF, post CHF heat transfer and pressure drop, and (2) establishment of a base of non-proprietary data and prediction methods available on the Internet.

This activity that includes documentation of the results of this international collaboration in an IAEA TECDOC has facilitated the transfer of knowledge of leading scientists in thermohydraulics to the next generation.

Conclusions

Evaluation of reactor performance under normal operation, accident and severe accident conditions require accurate representations of thermohydraulic relationships and thermophysical properties data. The results of a 4-year CRP have been documented and will be issued shortly as an IAEA TECDOC. The data and relevant thermohydraulic prediction methods contributed by the organizations participating in the IAEA's CRP has been stored in the INSC database, which is maintained by ANL.