transient two-phase flow

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TRANSIENT TWO-PHASE FLOW

Proceedings of the Third CSNI Specialist Meeting

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TRANSIENT TWO-PHASE FLOW

Preface

This book contains the proceedings of the third CSNI specialist meeting on transient two-phase flow. The meeting and the book have been structured to examine various areas of study, experimental and analytical, that lead to code development and verification:

Session I—Measurement Techniques covers a wide range of novel measurement approaches and provides a good basis for initiating an experimental effort. The six papers represent only a small sample of a much larger international effort. Through their extensive bibliographies one can have access to the many different research efforts.

Session II—Experimental Studies deals mostly with reflood phenomena. This is to be contrasted with the earlier meetings in which blowdown was emphasized. Analysis of the data shows that there are still many interesting problems for future research before all physical phenomena are fully described. One of the papers brings out the importance of entrainment at the quench front.

Session III—Fundamentals of Transient Two-Phase Flow contains a number of contributions to our understanding of fundamentals. Some very useful work is described in which experimental data were analyzed in a detailed manner to provide an insight into interfacial effects. This type of work could lead to a generalized method of specifying flow regimes. An interesting and cohesive group of papers dealing with the governing equations and closure laws form a major part of this session.

Session IV—Numerical Methods discusses methods for obtaining solutions to mist flow, flow boiling, and LWR systems. A number of different computational procedures are described and some of the difficulties in their implementation are discussed. Results for LWR system analysis using TRAC-BD1 and RELAP5 are also found in this session.

Session V—Code Application and Assessment covers topics ranging from detailed analysis of separate effects experiments to current achievements of major code systems. Much progress was made during the five-year period between the first and third meetings of this group. In 1976 all codes were based primarily on homogeneous equilibrium considerations. New codes are appearing that include most of the major features needed to model two-phase flows. The need to develop measurement techniques for some physical phenomena arises when code verification is considered. The advanced codes such as TRAC, RELAP5, CATHARE, and others allowing unequal velocities and temperatures must now mature through application and assessment.

Panel Discussion: "How Good Do Codes Have to Be?" generated a great deal of lively discussion. The need for a measure of "goodness" was discussed and a number of opinions were given. This brings up the important question of how does one assess or verify a given code. The importance of the code user was emphasized by each of the panelists. It was generally agreed that code accuracy and modeling of phenomena were good enough for the large-break LOCA. For small-break LOCAs and operational transients, more representative experiments as well as improvements in secondary side modeling were believed to be needed.

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The CSNI Specialist Meeting

The Third Specialist Meeting on Transient Two-Phase Flow, held in Pasadena, California, March 23–25, 1981, was sponsored by the OECD Nuclear Energy Agency (NEA) Committee on the Safety of Nuclear Installations (CSNI) and was hosted by the California Institute of Technology and the U.S. Nuclear Regulatory Commission. A continuation of the first (Toronto, August 3–4, 1976[1]) and second (Paris, June 12–14, 1978[2]) specialist meetings, the third meeting was attended by 100 engineers and scientists from 12 countries who represented many disciplines: analysts and experimenters, code and instrument developers.

The main objectives of the meeting were to bring together specialists in two-phase flow modeling, numerical analysis, experiments and instrumentation so that they could exchange information, compare progress made since the previous specialists' meetings, and indicate the directions in which more research should be done and what approach would give the most fruitful results.

The Nuclear Energy Agency (NEA) Committee on the Safety of Nuclear Installations (CSNI) is an international committee made up of scientists and engineers who have responsibilities for nuclear power plant safety research and licensing. The Committee was established in 1973 to develop and coordinate the NEA work on nuclear safety matters. The Committee's purpose is to foster international cooperation in nuclear safety among the OECD member countries by setting up working groups or task forces and arranging specialist meetings on certain specific subjects.

The technical areas at present covered by these activities are as follows: particular aspects of safety research relative to water reactors, fast reactors and high-temperature gas-cooled reactors; probabilistic assessment and reliability analysis, especially with regard to rare events; siting research as affects protection against external impacts; fuel cycle safety research; the safety of nuclear ships; various safety aspects of steel components in nuclear installations; licensing of nuclear installations; and a number of specific exchanges of information.

The specialist meetings are organized to provide a forum for discussing fundamental aspects of transient two-phase flow processes. The latter play a crucial role in nuclear safety evaluations of loss of coolant accidents and/or of plant transients. The meetings focused on two-phase flow models including the governing equations, numerical methods for solving them, experiments used for the validation of the mathematical modeling, and measurement techniques required to obtain the necessary experimental results.

[1] Published as Banerjee, S., and Weaver, K. R. (eds.), *Transient Two-Phase Flow: Proceedings of the First CSNI Specialist Meeting*, Atomic Energy of Canada Ltd., Toronto, 1978.

[2] Published as Réocreux, M., and Katz, G. (eds.), *Transient Two-Phase Flow: Proceedings of the Second CSNI Specialist Meeting*, Commissariat à l'Energy Atomique, Paris, 1980.

A meeting that brings together researchers from a large number of different countries requires a great deal of effort. The successful organization and operation of the Conference was to a very large degree the result of the efforts by Eric Johnson of the CSNI and by Helen Burrus of the California Institute of Technology. We also appreciate the support given us by the California Institute of Technology which gave generously of its facilities and personnel.

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Contents

FUNDAMENTALS OF TRANSIENT TWO-PHASE FLOW

NUMERICAL METHODS

CODE APPLICATION AND ASSESSMENT

PANEL DISCUSSION: HOW GOOD DO CODES HAVE TO BE?

MEASUREMENT TECHNIQUES

Modern Measurement Techniques for Inadequate Cooling of Nuclear Reactor Cores

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ABSTRACT

A reliable and unambiguous monitoring of reactor vessel liquid level, which can detect inadequate core cooling (ICC), was an important item from the Three Mile Island (TMI) Lessons Learned survey.

A multitude of methods was proposed by many parties involved. They could be divided in two main groups: intrusive and nonintrusive, regarding the penetration of the reactor pressure boundary. Some of the methods considered are proven in similar or nonnuclear systems, such as static head/differential pressure, heated thermocouples, sonic/ultrasonic devices, microwaves, etc., while others are in a developing stage, although based on proven principles. Several methods from this second group would be: gamma attenuation, neutron diffusion (both locally applied), neutron/gamma activation (n-gamma and gamma-n) reactions, traversing probes signal and noise analysis, etc.

Applicability of various methods should be considered in conjunction with factors, such as reactor operating state (steady power, transients, or shutdown) and sensing field (upper plenum and core region).

This paper lists all realistic candidate methods in a systematic way and evaluates their relative merits and drawbacks based on several criteria, such as: feasibility study and test results, power plant retrofit possibility/ convenience, mounting cost and time needed, instrument longevity/survival in the adverse conditions of the reactor environment, operability, reliability, unambiguity of data, serviceability, and other features.

1. INTRODUCTION

Until recently, the water inventory in a pressurized water reactor (PWR) vessel was not directly measured. Various investigation groups [], 2 , 3] identified this as a major contributor to the TMI-2 accident of March 1979. In order to protect the health and safety of the public, the U.S. Nuclear Regulatory Commission (USNRC) set requirements for both new and existing PWRs to provide direct, unambiguous methods to detect inadequate core cooling (ICC) in PWR vessels. Although there are several promising techniques, very few have been adequately evaluated. It is not known to what extent they meet NRC requirements or whether they can be backfitted in a practical manner. To meet this need, the NRC, Division of Reactor Safety Research (RSR), organized several meetings to explore feasible methods and sponsor limited evaluation studies.

The basic approach of evaluating the various candidates for liquid level measurement is to formulate a set of criteria against which the performance of each method can be judged as to its feasibility for application as a power plant instrument. These criteria are data quality, survivability, reliability, retrofit, and operation.

Once the criteria is established, the merit and shortcoming of each proposed method will be evaluated. Several staff meetings were held, and experts were solicited to establish a concensus of evaluation on those more established methods. In addition, a specialist's meeting was held in October 1980 at the WRSR information meeting where proposers of more novel approaches were invited to present their cases for peer review [4].

A few methods considered more feasible are being subject to bench and system tests in NRC experiment facilities under the conditions simulating those expected to exist in reactor accidents or transient.

This paper summarizes several techniques that may be used to measure the core cooling ablity of power reactors. Evaluations of these techniques, based on testing and engineering judgment, are also presented.

It should be emphasized that evaluations reported in this paper reflect only the technical opinion of some of the instrumentation staff in the Reactor Safety Research Division. As such, it should in no way be considered as the NRC official position.

2. CRITERIA FOR EVALUATION

Although many proposed techniques appear good in principle, one must carefully consider many important factors before actually installing them in nuclear power plants. Some of these factors are NRC licensing requirements, data quality, reliability, ability to survive during operation and abnormal conditions, and impacts on existing reactors and plant operations when retrofitted. These considerations can be used as criteria to evaluate proposed techniques.

2.1 NRC Licensing Requirements

The NRC licensing requirement for the ICC instrument is part of the post-TMI Action Plan [5], Section II.F.2. A follow-on document provides guidance on how one can meet this requirement [6]. Some of the points worth mentioning are: The measurement must be unambiguous and easy to interpret under various phenomena with the coolant pumps on and off; it must cover the entire length of the vessel; it must give advance warning of the approach of ICC; and it must meet all of the qualification requirements for safety-related electrical equipment. However, NRC allows combining the new instrument with existing in-vessel instruments, such as core-exit thermocouple and subcooling meters, if the new instrument cannot meet all the requirements by itself.

2.2 Data Quality

Inherent measurement characteristics. Each technique has its own inherent advantage and limitations; these should be carefully recognized in the feasibility evaluation. The measurement should be unambiguous, the need for data inference should be minimal, and its function should not rely too much on other measurements. It should cover the normal operation conditions of the reactor and abnormal conditions. This means the instrument should perform