RELAP5/MOD3.2 POST TEST CALCULATION OF THE PKL-EXPERIMENT PKLIII-B4.3 pdf

1: Pucchi www.enganchecubano.com www.enganchecubano.com www.enganchecubano.com Et - www.e

In test B, the influence of non-condensables on heat transfer in the steam generators during reflux condenser conditions was investigated. This report presents the results of a post-test analysis of PKL III-B using RELAP5/Mod

This is a special issue published in Science and Technology of Nuclear Installations. All articles are open access articles distributed under the Creative Commons Attribution License, which permits unrestricted use, distribution, and reproduction in any medium, provided the original work is properly cited. Del Nevo, and F. This is an open access article distributed under the Creative Commons Attribution License, which permits unrestricted use, distribution, and reproduction in any medium, provided the original work is properly cited. A considerable amount of resources has been devoted at the international level during the past three decades for establishing and conducting experimental programmes in scaleddown integral test facilities ITFs. These were aimed at solving open issues for current nuclear power plant NPP , demonstrating the technical feasibility of innovative designs, and generating reference databases to support code development and assessment. Since the end of the nineties, the maintenance of the competences of experienced research teams as well as the creation of a new generation of professionals in the area of the nuclear safety is also a priority objective of operating research facilities and promoting experimental programmes. Tens of ITFs have been built and operated so far all over the world. Few of them, related to existing water reactor technology, are currently in operation e. Some others are constructed or under design and are focused on innovative water reactor concepts e. The experimental data from such facilities are applicable to full-scale nuclear plant conditions; if the test facilities and the initial and boundary conditions of experiments are properly scaled, for example, the scaling will not affectthe evolution of physical processes important for the postulated accident scenario. This evaluation determines whether the data may be used in nuclear plant safety analyses of a postulated accident. On the other side, the experimental data are fundamental for supporting the development and demonstrating the reliability of computer codes in simulating the behavior of an NPP during a postulated accident scenario: The reliance on computer codes, among the other things, is based on the reason that, very often, one cannot directly apply results from test facilities to a plant, including the reference plant of the facility design. Applications of computer codes to accident analyses require the implicit assumptions that these codes have the capabilities to scale up phenomena and processes from test facilities to full-scale plant conditions. However, the different scale, in terms of geometry, characterizing any facility and a nuclear plant does not ensure a priori that a code, which is able to reproduce a generic transient in a scaled facility, is also able to calculate with the same accuracy the same transient in NPP. The aforementioned topics involve a number of key activities, in which the research is taken up. This special issue documents the present scientific and technical status and recent advances in relation to the ITF, the experimental programmes, the issues connected with the code assessment, and the scaling issue, which is related both to the representativeness of the phenomena in the facilities as well as the applicability of the codes. The special issue collects 7 papers, which are divided into 4 main groups, according to the following rationale: The paper by K. The authors point out important results achieved in improving the level of understanding of the PWRs system response under accident conditions as well as of the thermal-hydraulic phenomena relevant for the nuclear reactor safety and the importance for code validation. The description of relevant phenomena of the transient and the results of assessment of the code are reported in the paper. Annunziato illustrate the LOBI facility project. The paper provides a historical perspective and summarizes major achievements of the research programme which has represented an effective approach to international collaboration in the field of reactor safety research and development. Focus is also given to the issue of the management of research data. The paper of J. Manera addresses the issue of the validation process of system codes for the transient analyses of PWR. The paper underlines the relevance of the validation activity in relation to the modeling of an NPP for safety analysis purposes. Bousbia Salah and J. The tests are devoted to the study of the cooldown procedures operated after the reactor trip in order to bring the primary side temperature and pressure to the residual heat removal system RHRS operating conditions. The objective is to assess the impact of a chosen cool-down strategy upon the occurrence of natural circulation interruption NCI and the capabilities of CATHARE2 code in predicting such process. Relevance is given to the interaction between the key parameters governing the transient and the phenomena involved. The selected experiments are LOCA scenarios of different break sizes and with different availability of safety injection components. The objective of the analysis is to improve the knowledge of the phenomena reproduced in the facility in order to use them for nodalization qualification purposes of nuclear power plants or for establishing accuracy databases for uncertainty methodologies. The code selected is RELAP5, widely used all over the world for safety analysis of nuclear power plants. Del Nevo et al. The objective is to collect, analyze, and document the numerical activity pre- and posttest performed by the participants, describing the performances of the code simulations and their capabilities to reproduce the relevant thermal-hydraulic phenomena observed in the experiment. It discusses also the safety significance of the PMK-2 projects. The paper by V. The paper of D. The applications cover integral containment response tests, component tests, primary system tests, and separate effect tests. The paper provides an overview of the research programs performed in relation to BWR containment systems and those planned for PWR containment systems. The authors discuss the facility features and capabilities, the instrumentation systems, and the experimental program. The paper outlines the integral system tests, which will be performed to simulate transients and LOCA loss of coolant accident scenarios. Focus is given to the countercurrent flow limiting CCFL phenomenon. The activity includes the application of the propagation of input errors PIEs method, which is used to perform the uncertainty analysis and to identify the input parameters having more influence on the figure of merit selected in this case the peak cladding temperature. The first paper of the innovative integral-type water reactor concepts group by F. The multiapplication small lightwater reactor MASLWR is the scaled-down model of a small modular PWR, relying on natural circulation during both normal and accident conditions. The paper includes a review of the main characteristics of the facility and summarizes the tests already executed and the related code validation activity. The paper of A. In conclusion, the scientific and technical contributions from the authors provide the readers with useful information related to 4 experimental facilities and cover a broad spectrum of the recent past and current activities dedicated to this special issue. Acknowledgments The guest editors acknowledge all authors who have submitted papers to this special issue. Special thanks are due to our colleagues for the kind collaboration in reviewing these papers. The PKL facility models the entire primary side and significant parts of the secondary side of a pressurized water reactor PWR at a height scale of:. Volumes, power ratings and mass flows are scaled with a ratio of: The experimental facility consists of 4 primary loops with circulation pumps and steam generators SGs arranged symmetrically around the reactor pressure vessel RPV. The investigations carried out encompass a very broad spectrum from accident scenario simulations with large, medium, and small breaks, over the investigation of shutdown procedures after a wide variety of accidents, to the systematic investigation of complex thermal-hydraulic phenomena. This paper presents a survey of test objectives and programs carried out to date. It also describes the test facility in its present state. Some important results obtained over the years with focus on investigations carried out since the beginning of the international cooperation are exemplarily discussed.. Introduction Complex thermal-hydraulic system codes are used for the analysis of accident sequences in pressurized water reactors. The PKL test facility has been in operation since ; however, in the meantime, the objectives of the experiments performed at the PKL test facility have changed considerably with the result that the test rig has been refitted many times to suit the additional and ongoing tasks and also to match latest developments, for example, in the fields of measuring instrumentation and data processing. Since the commencement of experiments at the PKL test facility, the various phases of the experiments have always reflected and given priority to current safety issues. The primary objective of all PKL experiments has been and remains the experimental investigation of thermalhydraulic processes in PWRs with respect to the response of the overall system. To some extent the investigations also include the behavior of individual components and subsystems during the simulation of operational transients and accidents. The tests performed to date in total more than 5 integral experiments have altogether contributed to a better understanding of the sometimes highly complex thermal-hydraulic processes involved in various accident scenarios and to a better assessment of the countermeasures implemented for accident control. In addition, they have supplied valuable information regarding safety margins available in the plants. The test results have also found concrete application in the validation and further development of thermal hydraulic computer codes, the so-called system codes. PKL Test Programs Reactor safety research in the seventies centered above all on the theoretical and experimental analysis of large-break loss-of-coolant accidents LOCAs , focusing on verifying the effectiveness of the emergency core cooling system 12 2 Science and Technology of Nuclear Installations Participation of German government Additional participation of German utilities Additional participation of international partners Large and small break loss-of-coolant accidents st PKL test Jan. ECCS required for controlling such events. The large-break LOCA experiments were interrupted in in the wake of the accident at three-mile island unit 2 TMI-2 for the performance of experiments at the PKL test facility designed to contribute to gaining information as quickly as possible on issues raised by this event. The investigations focused on demonstrating the safety margins of the operating units through the experimental verification of the effectiveness of the engineered safety features in the event of large- and small-break LOCAs were covered within the test programs PKL I and II [, 2]. While the first test series within PKL III covered design-basis accidents and cooldown procedures detailed in the operating manual, the main interest was then focused on beyond-design-basis accidents and the experimental verification of accident-management procedures [3]. The major topics covered by the experiments between 2 and 27 were i boron dilution events following SB-LOCA, ii loss of residual heat removal under shut-down conditions. Scaling and operating parameters. The entire primary side and the most significant components of the secondary side excluding turbines and condenser , including the appropriate system technology, are represented. Because the essential construction principles of the western types of PWRs are similar, it is possible to make statements concerning the behavior of other companies plants. In any case, the analysis of plant-specific reactor transients must then be made with the help of computer codes. Following the scaling concept, all geodetic heights are represented in a: The entire volume of the primary side and, as far as possible, the partition of the individual Figure 3: PKL steam generators top view. For some components, the exact volume scaling was not applied in order to simulate certain thermal-hydraulic phenomena, for example, countercurrent flow limitation in the hot legs. This allowed dimensionless numbers e. The single-phase pressure losses, correspond to a large extent to the values in a PWR. Together with the thermal losses they have been determined in detail for every component and section for the entire load and temperature ranges reactor coolant pump RCP operation and NC conditions under cold and hot conditions. The core is modeled by a bundle of 34 electrically heated rods with a total power of 2. The core geometry is, like the SG geometry, constructed as an actual section ; that is, the individual heated rods and U-tubes have the actual geometry, but the number of heated rods in the core and the number of U-tubes in the SG are reduced by the scaling factor: The PKL heater rod bundle has a uniform axial power profile, and the heater rods are arranged in three concentric zones heated independently of one other which enables radial power profiles across the test bundle to be simulated. Boron concentration and loop-flow measurements. The RPV downcomer is modeled as an annulus in the upper region and continues as two stand pipes connected to the lower plenum. This is of importance with respect of the removal of stored heat from the walls during cool down. In general, the structure masses resulting from the limitation of pressure to 45 bar return the relation between heat capacities of structures and coolant masses to be in good accordance with the reference plant. The symmetrical arrangement of the 4 loops around the RPV also allows investigating the individual effects of multiple system failures. Experiments on the behavior of a 3-loop 2-loop plant can also be conducted by simply isolating one two loop s. Each of the primary side loops contains active RCPs, which are equipped with speed controllers to enable any pump characteristics to be simulated. Preserving the frictional pressure losses in the SGs and in the core region, the integral pressure loss for the entire primary system is also very similar to that of the actual plant. PKL is also equipped with all relevant engineered safety and operational systems on the primary and secondary side. On the primary side, four independent high- and low-pressure safety injection systems connected to both, the hot and cold legs, the residual heat removal system, 8 accumulators, the pressurizer pressure control system, and the chemical and volume control system are simulated. On the secondary side the feed, water system, the emergency feed water system and the main steam lines with all control features of the original systems are modeled.

2: www.enganchecubano.com www.enganchecubano.com - www.enganchecubano.com

Are you sure you want to remove RELAP5/MOD post test calculation of the PKL-experiment PKLIII-B from your list?

3: Integral Test Facilities and Thermal-Hydraulic System Codes in Nuclear Safety Analysis - PDF

NUREG/IA International Agreement Report RELAP5/MOD Post Test Calculation of the PKL-Experiment PKLIII-B Prepared by L Kamer Siemens Energierzeugung (KWU).

4: RELAP5/MOD post test calculation of the PKL-experiment PKLIII-B (edition) | Open Library

Get this from a library! RELAP5/MOD post test calculation of the PKL-experiment PKLIII-B [L Karner; U.S. Nuclear Regulatory Commission. Office of Nuclear Regulatory Research.;.

RELAP5/MOD3.2 POST TEST CALCULATION OF THE PKL-EXPERIMENT PKLIII-B4.3 pdf

Classroom assessment mcmillan 7thh edition The exhibitions at Harvard College founded prior to 1800. Ten commandments of faith and fitness Healthy Gluten-free Cooking Vikings (Interfact) First Steps in Quilting Leslie Linsley Renewal of the promise : prophets of the seventh century BC Sword of revolution. Making her way alone From consultation to confrontation History of Tasmania I lost my keys and my car Hyper Police Volume 4 (Hyper Police) A selected bibliography on the hydrology of the Platte River Basin in Nebraska through 1991 Anglo-Japanese relations during the First World War Coral reef feeders The Phallic Deity 91 Challenges in the Middle East Sport and the media William Carries on (William) Legend of korra book Ebook davis farmers market cookbook Olivers alphabets Pictorial introduction to the Bible Vernon Fishers File 00 Pathways to the present Revisionist mystery Build Your Own Army of Web Bots Within 24 Hours (Army of Web Bots Series, 1) Authorizing Policy Glorious Destiny of Marshall Nnikon Nniku (Ubu Repertory Theater Publications,) Americas first traitor Tuscan folk-lore and sketches Sample journalism cover letter Jennifer Johnston Aggressive introvert; a study of Herbert Hoover and public relations management, 1912-1932. The early years of Madame Royale. Essay on faith, reason, and human nature Prayer declaration point for your family Raymond Chandler in Hollywood Aptitude with answer