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Assessment study of the coupled code RELAP5/PARCS against the Peach Bottom BWR turbine trip test

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Abstract

The modeling of complex transients in nuclear power plants (NPP) remains a challenging topic for best estimate three-dimensional coupled code computational tools. This technique is, nowadays, extensively used for simulating transients that involve core spatial asymmetric phenomena and strong feedback effects between core neutronics and reactor loop thermal–hydraulics. In this framework, the Peach Bottom BWR turbine trip experiment 2 is considered. The test involves a rapid positive reactivity addition into the core generated by a water hammer load. To perform a numerical simulation of such phenomenon a reference case was calculated using the coupled code RELAP5/PARCS. An overall data comparison shows good agreement between calculated and measured pressure wave trend in the core region. However, the predicted power response during the excursion phase did not match correctly the experimental tendency. For this purpose, a series of sensitivity analyses have been carried out to identify the most probable reasons of such discrepancy. It was found out that the uncertainties related to the cross-sections modeling and to the thermal–hydraulic closure relationships are the main source of the incorrect power feedback response during the transient.

1. Introduction

Evaluation of the nuclear power plant performance during transient conditions has been the main issue of safety researches since the beginning of the exploitation of nuclear energy. During the last decades, safety analyses were performed using a variety of conservative and best estimate (BE) thermal–hydraulic and kinetic models. Generally, these computational tools were developed in parallel ways and could not take into account the global phenomena involved during a given transient. Nowadays thanks to continuous

Abbreviations: BE, best estimate; BPV, by-pass valve; BWR, boiling water reactor; CRISSUES, critical issues in nuclear reactor technology: a state of the art report; CS, cross-section; 3D, three-dimensional; EC, European Commission; IAEA, International Atomic Energy Agency; NEA, Nuclear Energy Agency; NPP, nuclear power plant; OECD, organization for economic co-operation and development; PVM, parallel virtual machine; TSV, turbine stop valve; TT, turbine trip; VALCO, validation of coupled neutronics/thermal–hydraulics codes for VVERs; VVER, water-cooled water-moderated energy reactor

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progress of computer technology, the coupling between thermal–hydraulic system codes and 3D neutron kinetic codes become feasible. The main features of these advanced codes are exploited to get more accurate simulation of complex transients in NPPs, and more particularly when strong feedback interactions between kinetic and thermal–hydraulic aspects are involved.

Generally, the coupled code method uses validated and qualified computational tools. Notwithstanding this fact, the resulted "coupled code" should be assessed through the general procedure of code validation process (D'Auria et al., 2001). For this purpose, several international activities involving organizations as OECD/NEA, EC (projects CRISSUES (D'Auria, 2003) and VALCO (Weiß, 2003)) and IAEA (IAEA TECDOC, in press), have been completed or are in progress aiming to gauge the capabilities of coupled codes and the new frontiers for the nuclear technology that could be opened by this technique (realistic safety analyses of existing plants, design of new reactors, relaxation of the current safety margins allowing higher operating power and extending the fuel cycles) (Bousbia Salah, 2004).

The current study provides a valuable contribution to the assessment and validation of coupled code technique through the well-defined turbine trip (TT) experiments performed at the end of cycle 2 of the Peach Bottom BWR NPP (Carmichael and Niemi, 1978). The test was defined by the OECD/NEA as a benchmark exercise for the coupled code computational tools (Solis et al., 2001). The transient under consideration involves both asymmetric phenomena and core cooling loop interactions. It is characterized by a positive reactivity insertion into the core induced by a sonic pressure wave originated in the steam lines. Due to the inherent feedback mechanisms, the core power exhibits a prompt excursion in response to the rapid reduction of the void inventory. In order to perform a BE simulation of such complex interactions, the coupled thermal–hydraulic system code RELAP5/Mod3.3 (Ransom et al., 1990) and the 3D neutronic code PARCS/2.4 (Joo et al., 1998) are used.

A reference code run, based upon the skeleton input deck of the Benchmark specifications (Solis et al., 2001) as well as the provided set of cross-sections library, was performed. The calculated results were afterwards compared with the available experimental data. A good overall agreement between the calculated and experimental pressure wave amplitude and propagation is observed. However, the calculated power response exhibited less conformity with the measured one. For this purpose, sensitivity studies have been carried out in order to identify the most probable reasons of such discrepancies.

2. Peach Bottom turbine trip test description

Turbine trip and low-flow stability transient tests were performed at Peach Bottom BWR Unit 2 prior to shutdown for refuelling at the end of cycle 2 in April 1977. TT tests were conducted to provide experimental database for computational tools as well as to investigate the effects of pressure transients generated in the reactor vessel following turbine trips from three different reactor power levels: 47.4, 61.6 and 69.1% rated power (Carmichael and Niemi, 1978). For each test, a total of 160 measurements were recorded for the resulting database.

The experiment under consideration was carried out at an operating power level equal to 61.6% of its nominal value. The transient is initiated by closing the turbine stop valve (TSV) in a very short time of about 0.096 s. As a result, a pressure wave is generated in the main steam piping and propagates at sound velocity with relatively little attenuation toward the reactor core. The pressure wave reaches the core zone following two different paths: through the steam separator filled with a mixture of water and steam and through the vessel downcomer filled with subcooled water. This double slightly unsynchronized effect, results in a rapid reduction of the core void inventory. The inherent core feedback mechanisms make the reactor power to experience a rapid exponential rise. After 0.06 s of the TSV closure, the steam by-pass valve (BPV) opens automatically to relax the water hammer loads. The TSV signal that activates the reactor scram initiation was intentionally delayed to allow a relative high-neutron flux effect to take place in the core. The scram is activated when the reactor power exceeds 95% of its nominal value.

3. Calculation models and hypothesis

To perform a numerical simulation of the TT transient the coupled code RELAP5/Mod3.3/PARCSV2.4