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Performance evaluation of passive pulse generator for auto depressurization system of Advanced Heavy Water Reactor

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ABSTRACT

The Advanced Heavy Water Reactor (AHWR) is a 300 MWe pressure tube type boiling light water cooled, heavy water moderated reactor. It has natural circulation based main heat transport system with integral steam drums. Recently, work has been initiated on a novel passive safety system named Passive Auto Depressurization System (PADS) for AHWR. This system is a part of shutdown cooling system and in case of occurrence of small break loss of coolant accident; PADS system mitigates chance of reactor core temperature rise. It actuates automatically at low steam drum level and rapidly depressurizes main heat transport system so that water injection from gravity driven water pool can be initiated to maintain coolant inventory for a sufficiently long time.

The main component of PADS is Passive Pulse Generator (PPG), a passive device which is actuated at low steam drum level and generates high pressure signal within a short time span. The PPG generates a passive pressure pulse of around 11 bar amplitude with 8 min response time with threshold steam drum level fall of 100 mm. This pressure pulse is used for opening a passive valve which initiates decay heat removal through isolation condenser leading to depressurization of main heat transport system. An experimental facility has been designed towards the development and performance evaluation of PPG for PADS system of AHWR.

This paper deals with the design of the experimental facility describing its piping layout, steam drum & heat exchanger vessel and associated instrumentation & control system. Test results along with RELAP5 analysis are discussed to validate PPG performance for PADS in AHWR.

1. Introduction

Post-Fukushima, several design improvisations in the operational nuclear power reactors are being considered and implemented. The need to incorporate passive or inherent safety features which require no active controls, is being recognized. These safety features may rely on gravity, natural convection or resistance to high temperature. The Indian Advanced Heavy Water Reactor (AHWR) has been designed incorporating these safety features. Some of the passive safety systems in AHWR are natural circulation based main heat transport system, isolation condenser system for decay heat removal, emergency core cooling system, passive poison injection system, passive containment cooling system and automatic depressurization system (Sinha and Kakodkar, 2006). A simplified schematic arrangement of the AHWR and its various systems are shown in Fig. 1.

Design basis accidents in a nuclear reactor are classified by type of initiating events. Loss of Coolant Accident (LOCA) includes a full or partial break of group distribution header and break of main steam line in different locations. The safety criteria prescribed in 10 CFR 50.46 (U.S. NRC) is applicable to both large and small break LOCAs (SBLOCA). Safety of the reactor core in LOCA scenarios can be taken care of by introducing systems like Auto Depressurization System (Mehedinteanu, 2013), residual heat removal system (Ayhan and Sökmen, 2016) or decay heat removal system.

Abbreviations: ADPV, Auto Depressurization Passive Valve; AHWR, Advanced Heavy Water Reactor; BARC, Bhabha Atomic Research Centre; DN, Diametre Nominel; I^2S-LWR, Integral Inherently Safe Light Water Reactor; LOCA, Loss of Coolant Accident; PADS, Passive Auto Depressurization System; PID, Proportional-Integral-Derivative; PPG, Passive Pulse Generator; PPPT, Passive Pressure Pulse Transmitter; RELAP, Reactor Excursion and Leak Analysis Program; SBLOCA, Small-Break Loss of Coolant Accident; SS, Stainless Steel; SWR, Siedewasserreaktor; VVER, Vodo-Vodyanoi Energetichesky Reaktor.

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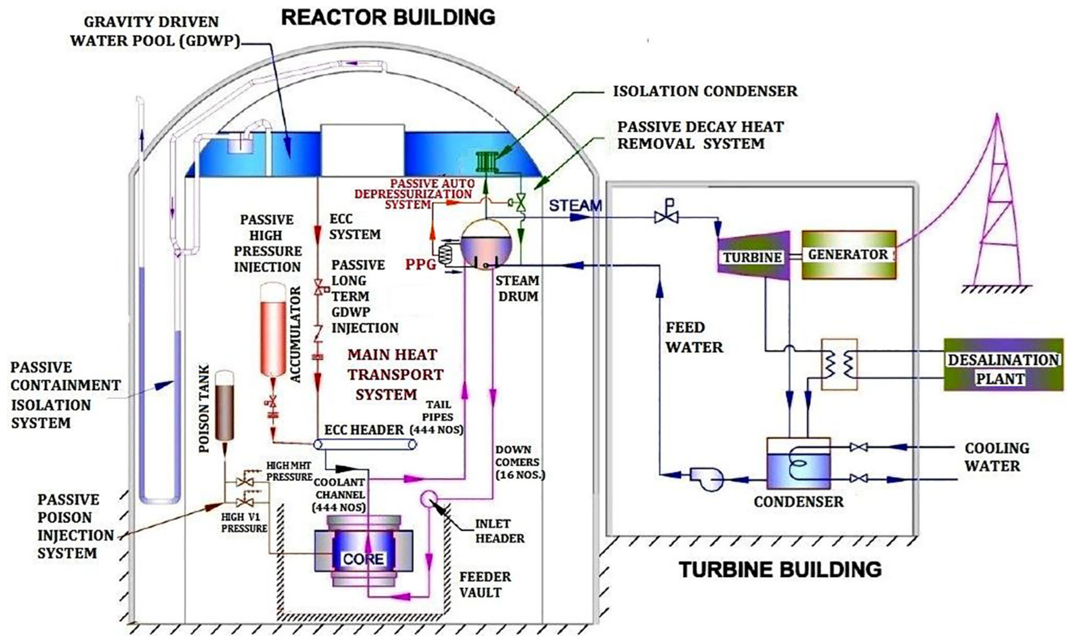

Fig. 1. Simplified schematic arrangement of AHWR.

Consequent to the accident at Three Mile Island Unit 2 reactor, plant operational transients received major attention (Aksan, 2008). In Gen-III/III + reactors namely, Siedewasserreaktor (SWR-1000), heat exchangers called Passive Pressure Pulse Transmitter (PPPT) are connected to reactor pressure vessel to initiate multiple safety-related switching operations like reactor scram, automatic depressurization and containment isolation (Stosic, 2006). These passive safety systems need to be evaluated before their deployment on a commercial scale (IAEA, 2012). Accordingly, the PPPT for SWR-1000 was tested in different variants using the NOKO test facility and Integral Test Facility Karlstein of AREVA (Morozov and Soshkina, 2008). Initially, PPPT systems were built for KERENA containment in full height & level scale and volume scaling of 1:24 was adopted (Leyer and Wich, 2012). Subsequently, full-scale single component tests were carried out in this facility, while the pre-test calculations were carried out with the thermal hydraulic code RELAP5 (Szijártó, 2015). However, no published literature is available regarding PPPT design and test results.

Passive Auto Depressurization System (PADS) in AHWR is designed to mitigate the consequence of SBLOCA. In AHWR, SBLOCA corresponds to break size of 1–5% of reactor inlet header area. RELAP5 analysis indicates that in such a scenario with reactor shutdown condition and completion of high-pressure accumulator injection, main heat transport system pressure does not fall below 2 bar even after 2 h. As per design intent, this pressure should be below 2 bar to initiate water injection from gravity driven water pool which is situated at the top of the reactor core inside the containment. The unavailability of water injection for a longer time from large size gravity driven water pool, may lead to rise in clad temperature due to insufficient coolant inventory from accumulators (Contractor et al., 2015). Therefore, to achieve faster depressurization of main heat transport system, PADS has been incorporated in the AHWR design. This system not only adds a redundant safety feature in the reactor but also makes it inherently safe in small break LOCA scenario by its passive mode of operation. PADS mechanism will be implemented for the first time in an Indian reactor and hence it is important to generate considerable experimental data to have confidence in this system.

The novelty of the proposed system lies in its functioning as a passive level sensing mechanism for steam drum vessel and generating a pressure pulse of significant amplitude. The passive sensing is carried out by Passive Pulse Generator (PPG) which is a small helical coil heat exchanger connected to reactor steam drum and the final control element is an in-house developed passive valve (Sapra et al., 2015). Once the steam drum level falls below a threshold level, it generates pressure pulse signal passively. This pressure signal is used to open a passive valve which depressurizes main heat transport system by actuating isolation condenser based decay heat removal system. A test facility has been designed, constructed & commissioned and extensive experimentations have been carried out to experimentally analyze PPG along with the steam drum at rated temperature & pressure conditions of AHWR. RELAP5 model of PPG test facility has been developed and validated with experimental data which will assist in analyzing actual PADS performance for AHWR.

The RELAP5/MOD3.2 code has been successfully used worldwide to analyze large and small break LOCAs and natural circulation studies. It has been validated for natural circulation based experimental facilities, such as the High-Pressure Natural Circulation Loop and the Parallel Channel Loop installed and operating at BARC (Mangal et al., 2012). It has also been validated for startup transients and power ramping of natural circulation loop (Lakshmanan et al., 2009). RELAP5 has also been used for natural circulation phenomena in VVER-1000 (Mousavian et al., 2004). A study on the instability of two-phase natural circulation flow under passive external reactor vessel cooling using RELAP5 has also been performed (Guozhi et al., 2013). An important aspect of PPG vessel simulation is modeling of the helical coil. Literature survey shows modeling of helical-coil steam generator has been successfully carried out by RELAP5-3D (Hoffer et al., 2011). RELAP5 has been used to model the Integral Inherently Safe Light Water Reactor (I^2S-LWR) reactor pressure vessel and containment passive systems like Passive Reactor Cavity Cooling System which uses a helical coil heat exchanger (Wang et al., 2018). Also, RELAP5/Mod.3.3 code has been used to validate MATLAB based thermal-hydraulic model of helical coil (Caramello et al., 2014). This work highlights the use of RELAP5 (Fletcher and Schultz, 2001) for modeling PPG vessel and PPG test facility.

This article covers the details of PADS system for AHWR, brief design & description of PPG test facility and test results of PPG unit.