Progress in Nuclear Energy 70 (2014) 159–166

Contents lists available at ScienceDirect

Progress in Nuclear Energy

journal homepage: www.elsevier.com/locate/pnucene

Assessment of passive residual heat removal system cooling capacity

J. Zou [a], Q. Li [b], L.L. Tong [a], X.W. Cao [a,] *

[a] School of Mechanical Engineering, Shanghai Jiao Tong University, Shanghai 200240, China
[b] Department of Design Management, Shandong Nuclear Power Co. Ltd., Shandong 265116, China

ARTICLE INFO

Article history:
Received 29 January 2013
Received in revised form
8 September 2013
Accepted 12 September 2013

Keywords:
Passive residual heat removal system
Loss of normal feedwater
Main feedwater line rupture
Sensitivity analysis

ABSTRACT

Advanced passive PWR relies on passive safety systems to provide core cooling capacity and deal with design basis accidents and beyond design basis accidents. However, the passive safety system is lack of practical operating experience and their performance is heavily influenced by other systems. The cooling capacity of passive residual heat removal system (PRHR), which is designed to remove decay heat when normal heat removal approach is not available, requires specific assessment during different accidents. In this study, a detail model of advanced passive PWR, including Reactor Coolant System (RCS), simplified secondary side and Engineered Safety Features (ESF), has been built using mechanism accident analysis code. The plant transient has been simulated, and cooling capacity of PRHR been analyzed during loss of normal feedwater and main feedwater line rupture. Conservative assumptions were made specially based on different accident scenarios and one of the two fail-open valves arranged in parallel at the PRHR heat exchanger (HX) outlet line was assumed not open, as the worst single failure. The progress of the two accident sequence is calculated and the thermalhydraulic behavior of RCS is investigated and the main transient parameters are obtained, including primary side pressure, steam generator pressure, pressurizer water level. The cooling power and system response are calculated. The results show that PRHR, with CMT injection, can remove the decay heat from RCS to IRWST, keeping the pressures of RCS and steam generators remaining below 110 percent of the design values and the pressurizer overfilling is prevented. Sensitivity study has been performed to study the system resistance effects on the capacity of PRHR, which shows that increase in system resistance coefficient reduces the cooling capacity of PRHR.

1. Introduction

In advanced passive pressurized water reactor (PWR) design passive systems are widely implemented to deal with design basis accidents (Schulz, 2006). As reported in AP1000 Design Control Document (DCD) (Westinghouse Electric Company, 2007), the passive residual heat removal (PRHR) system is designed to provide reactor coolant system cooling and prevents reactor coolant system (RCS) overpressure during loss of normal feedwater flow or main feedwater line break events. The main system functional component is passive residual heat removal heat exchanger (PRHR HX). The vertical C-shaped tubes of PRHR HX are submerged in the in-containment refueling water storage tank (IRWST) and located above the elevation of the RCS loops. Whenever the normal heat removal paths are unavailable, PRHR will be actuated by density contrast between cold water in C-shaped tubed cooled by IRWST and hot water in RCS loop. The core decay heat is removed continuously by natural circulation in PRHR and transferred to IRWST via PRHR HX.

Compared with conventional PWR, PRHR is an innovative design (IAEA, 2005) and has been implemented in several nuclear reactors (Park et al., 2009). However, there has not been a practical experience of the PRHR based on passive natural circulation in any active PWR, hence investigation is required. Thermalhydraulic phenomena related to PRHR, including natural circulation, thermal stratification, entrainment, etc., are complicated (IAEA, 2000). A series of integral systems tests of advanced passive PWR nuclear steam supply system was performed at the APEX-1000 test facility (Wright, 2007.), and results show that the performance of other systems in reactor coolant system, such as reactor coolant pump and CMT, has significant influence on PRHR cooling capacity, it varies with different accident sequences and specific assessment is required.

Mechanism accident analysis code has been widely used in passive safety system analysis and secondary side response analysis of PWR. Advanced passive PWR passive core cooling system pre-operational tests was simulated by means of RELAP5 (Liocea

* Corresponding author. Tel./fax: +86 21 34205495.
E-mail address: caoxuewu@sjtu.edu.cn (X.W. Cao).

0149-1970/$ – see front matter © 2013 Elsevier Ltd. All rights reserved.
http://dx.doi.org/10.1016/j.pnucene.2013.09.011

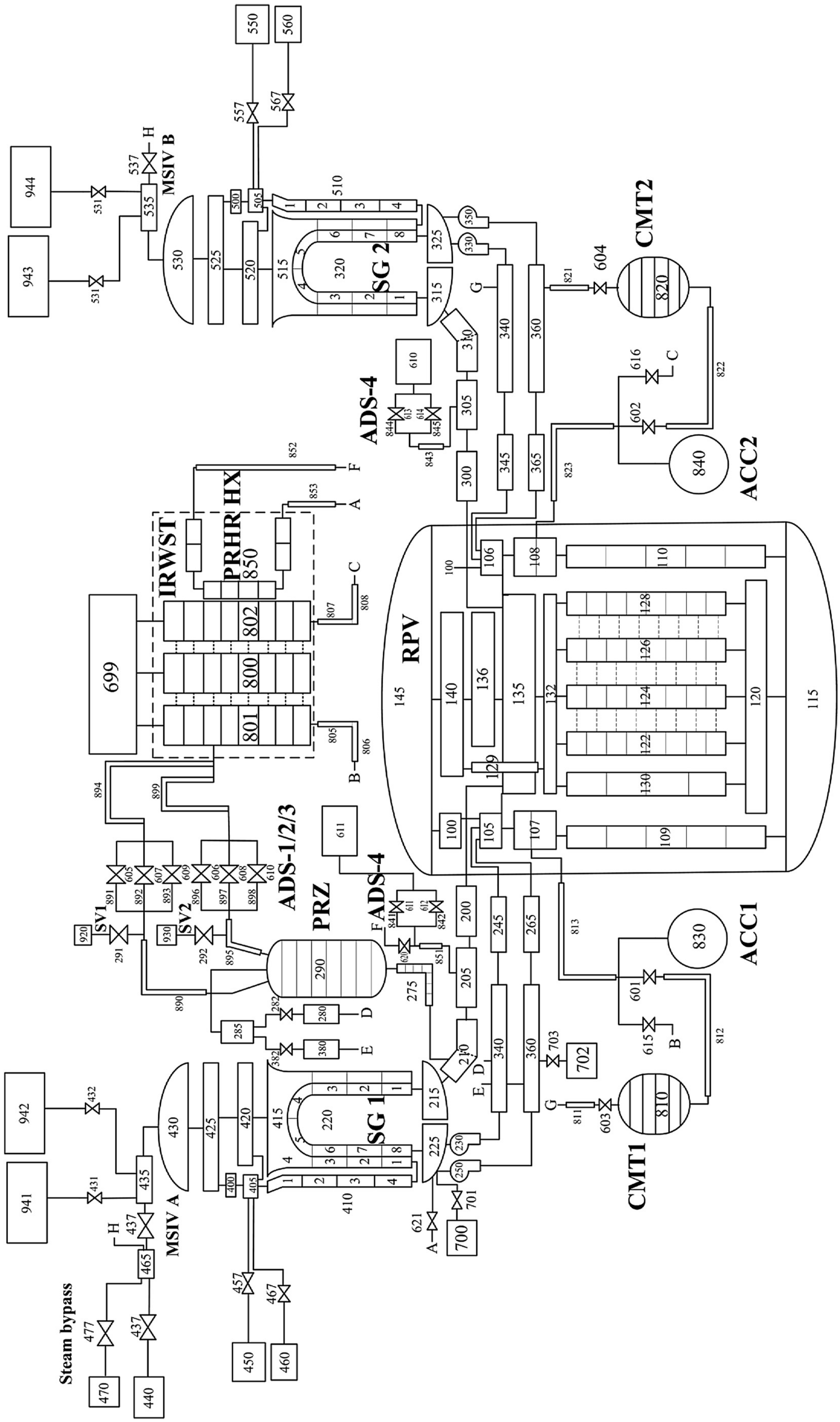

Fig. 1. Mechanism accident analysis model.