Progress in Nuclear Energy 77 (2014) 344–351

Contents lists available at ScienceDirect

Progress in Nuclear Energy

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

Study on pressure relief system design for high temperature gas cooled reactor

Wang Yan

Institute of Nuclear and New Energy Technology of Tsinghua University, The Key Laboratory of Advanced Reactor Engineering and Safety, Ministry of Education, Beijing 100084, China

A R T I C L E I N F O

Article history:
Received 15 October 2013
Received in revised form
13 December 2013
Accepted 1 January 2014
Available online 23 January 2014

Keywords:
High temperature gas-cooled reactor
Steam generator
Pressure relief system

A B S T R A C T

Under some postulated accidental conditions, the secondary loop of the reactor needs to release its pressure for consequent system action. During the pressure relief transient, the residual fluid mixture of water and the steam will pass through the heat-exchange tubes of the steam generator and the live-steam pipeline, which decreases the steam generator pressure and the steam generator temperature due to taking the heat stored in the components away. The pressure and the temperature during the transient needs to be controlled to protect the steam generator from the potential damage. A 250 MWth high temperature gas cooled reactor is selected to evaluate this countermeasure effect during the pressure relief transient in this paper. A thermal-hydraulic system analysis code is used for modeling and simulating the pressure relief transient after the accident occurs. The temperature change of the fluid and the components in the steam generator are relative to the pressure relief transient, which is dominated by the design of pressure relief valve. Larger diameter of pressure relief valve will result in more intensive blown-down transient which causes faster pressure decrease and stronger heat transfer between the inside walls of the components in the secondary loop and the fluid. The designed pressure value where the pressure relief valve closes will affect the transient also. The designed pressure value dominates the time of the pressure relief transient which affects the amount of the fluid from the pressure relief valve. Different amount of the fluid will take away different quantity of heat from the components in the secondary loop of reactor. The pressure relief transients with a series of different diameter of pressure relief valve under some typical designed pressure value were simulated and compared. The results indicate that the pressure and the temperature of the steam generator during the pressure relief transient could be controlled effectively with reasonable design on the pressure relief system, which will provide protection and benefit for the steam generator after the postulated accident.

1. Introduction

After a postulated accident in high temperature gas cooled reactor (HTGR) is detected, for example the primary loop loses its pressure, the emergency shutdown system will be started and the residual heat removal system will take the reactor decay away to cool down the reactor. The steam generator (SG) which still remains high pressure and high temperature level also needs to release its pressure for the subsequent system action.

The pressure relief valve (PRV) installed on the live-steam pipeline (LP) will open to release the pressure of the SG secondary side to a lower designed value. During the pressure relief transient, the fluid mixture of water and steam in the secondary side of the SG will flow through the SG heat-exchange tubes (HT) and the live-steam pipeline, and the wall surfaces of these components will transfer heat with the fluid. Thus, accompanying the pressure decrease, the temperature of the components in the SG and the live-steam pipeline will decrease during the pressure relief transient. The change of the pressure and temperature during the transient should be controlled to protect the SG.

In this paper, a 250 MWth high temperature gas cooled reactor of HTR-PM is selected for the study on the pressure relief transient. For better understanding, the brief descriptions on the HTR-PM and its pressure relief transient are given in the following sections.

2. The HTR-PM reactor

2.1. General description

The inherent safety concept has been widely accepted among the nuclear community for its well-known safety features (Lohnert,

E-mail address: wangyanfcw@tsinghua.edu.cn.

0149-1970/$ – see front matter © 2014 Elsevier Ltd. All rights reserved.
http://dx.doi.org/10.1016/j.pnucene.2014.01.006

1990). It is considered that the modular HTR has the capability to realize the safety target that the consequences of all conceivable assumed severe accidents should not result in notable offsite radiation impacts so that the need for offsite emergency can be technically eliminated. Therefore, the modular HTR becomes one candidate for the Generation IV nuclear energy system technology.

The research on the modular high temperature reactor (HTR) has been ongoing in China. After the successful design, construction and operation of the 10 MW high temperature gas-cooled test reactor (Xu and Zuo, 2002), the 200 MWe high temperature gas-cooled reactor pebble-bed module (HTR-PM) project is ongoing with the support of the Chinese government. The HTR-PM plant consists of 2 × 250 MW module pebble-bed reactors of one-zone cylindrical core, which are designed with standardization and modularization technology (Zhang and Sun, 2007).

In order to reduce engineering technical risks, the conventional steam-turbine cycle is selected instead of helium turbine cycle, because the latter is still under development and there are many unsolved problems for commercial plant application. The two module reactors are connected to one steam-turbine generator with a generating efficiency of about 42%. The reactor core is a loose packed bed with an average height of 11 m and a diameter of 3 m. Each reactor module basically consists of the reactor pressure vessel (RPV), the steam generator pressure vessel (SGPV), and the connecting horizontal coaxial hot-gas duct pressure vessel (HDPV). The main helium blower is mounted on the upper part of the SGPV. The hot helium from the outlet of the reactor core with an average temperature of 750 °C transfers heat to the secondary loop water in the SG to produce high-pressure superheated steam and then cooled down to 250 °C, meanwhile the feed water of the secondary side of SG in the heat-exchange tubes bundle flows upwards from the bottom to the top and is heated from 205 °C to 571 °C (Yanhua et al., 2011). The general design parameters are shown in Table 1, and the cross-cut diagram of the structure of one reactor unit is illustrated in Fig. 1.

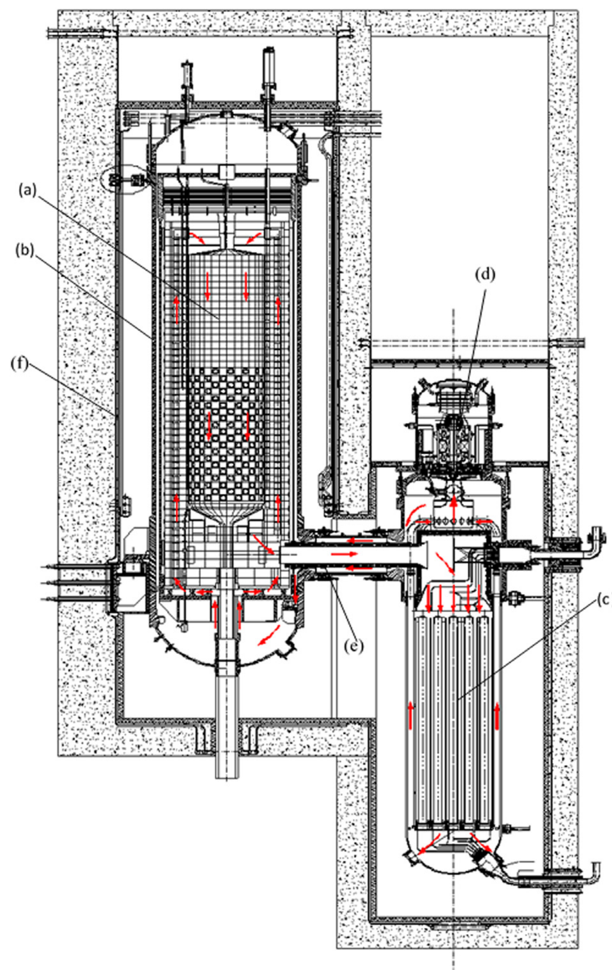

Fig. 1. Illustration of the HTR-PM Core (a) reactor core (b) reactor pressure vessel (c) steam generator (d) helium blower (e) hot-gas duct pressure vessel (f) water-cooling panel.

2.2. Relevant components

2.2.1. Reactor

The reactor core with a mean power density of 3.22 MW/m^3, located inside the RPV, is a loose packed bed with an average height of 11 m and a diameter of 3 m, which consists of about 420,000 spherical fuel elements in the equilibrium state. There are about 12,000 "TRISO" coated fuel particles of 0.92 mm diameter scattered in the inner graphite matrix with a diameter of 5.0 cm in the fuel zone of each fuel element, and the outer shell with a thickness of 0.5 cm is a fuel-free zone.

The hot helium with an average temperature of 750 °C from the reactor is fed into the SG through the central channel of hot gas duct, and flows around the outside of the heat-exchange tubes from the upside to downside. The cooled helium turns reversely at the bottom of the SG, and flows upwards between the SG shroud and the inner wall of the pressure vessel of SG.

The cold helium with an average temperature of 250 °C circulated by the helium blower installed on the top space of the pressure vessel of SG into the RPV through the outer annular channel of the heat gas duct. The main part of the helium flows upwards to the top of the core through 30 coolant boreholes in the side reflector and is collected in the cold helium plenum located in the upper part of the top reflector, and then it flows down mainly through the pebble bed and is heated up to an average temperature of 750 °C flowing out of RPV.

Table 1
Primary design parameters of HTR-PM.

Parameters	Designed value
Reactor power (MWth)	2 × 250
Helium pressure of primary loop (MPa)	7
Helium mass flow rate (kg/s)	96
Inlet helium temperature (°C)	250
Outlet helium temperature (°C)	750
Main steam pressure (MPa)	13.9
Main feed-water temperature (°C)	205
Main steam temperature (°C)	571
Feed-water flow rate for one reactor steam generator (kg/s)	98

2.2.2. Steam generator

The simplified SG secondary loop of HTR-PM is illustrated in Fig. 2. The SG is designed as a once-through assembly type of helical tube steam generator placed below the core in elevation. The hot helium is fed into the top of the SG above the heat-exchange tube bundle, then flows around the heat-exchange tube and transfers its heat to the 205 °C subcooled feed-water in the tubes to produce a 571 °C superheated steam, whereby it cools down from approx. 750 to 250 °C. The feed-water flows through the helical tubes from the bottom to the top in the SG. Then all the superheated steam is collected at the upper plenum and passes through the live-steam pipeline to the turbine hall. There is a low plenum and a high plenum on the two ends of the heat-exchange tubes. The flow