Nuclear Engineering and Design 330 (2018) 250–264

Contents lists available at ScienceDirect

Nuclear Engineering and Design

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

Fragility comparison analysis of CPR1000 PWR containment subjected to internal pressure

Lei Zhou[a,b], Jianbo Li[a,b,*], Hong Zhong[a,b], Gao Lin[a,b], Zhongcheng Li[c]

[a] State Key Laboratory of Coastal and Offshore Engineering, Dalian University of Technology, Dalian 116024, China
[b] Institute of Earthquake Engineering, Dalian University of Technology, Dalian 116024, Liaoning, China
[c] China General Nuclear Power Group, Shenzhen 518029, Guangdong, China

ARTICLE INFO

Keywords:
CPR1000 PWR containment
Numerical modeling
Fragility analysis
Internal pressure

ABSTRACT

Evaluating the reliability of the containment structure of a nuclear power plant under an accident internal pressure is of importance in Design Basis Accident and Design Extension Condition analyses. To study the impact of the diverse requirements concerning material properties in various Nuclear Power Plant (NPP) design codes on the reliability of the CPR1000 PWR containment structure, its detailed full-sized three-dimensional Finite Element (FE) model with nonlinear material constitutions, accurate geometric features, and a complex reinforcement system is constructed, and a set of material property data is obtained through practical measurements at a certain CPR1000 power plant for comparison. Based on the Monte Carlo method, random properties of the main material of the containment are applied to produce 100 samples using two sets of NPP design codes (Chinese and European) and one set of practically measured data. Various damage patterns of the samples generated owing to the uncertainty in the material properties are distinguishable. Generally, a concrete crack near an equipment hatch causes containment functional failure and when pre-stressed tendons begin to yield, the containment structure would not continue to maintain its integrity. Based on the functional and structural failure criteria, the corresponding fragility curves are obtained, and the results suggest that irrespective of the differences in the definition of material properties in various NPP design codes, the calculated reliability is practically the same. The difference of internal pressures corresponding to 5% failure probability is less than 2.6%. However, the containment based on practically measured data shows a slightly higher and slightly lower reliability than that based on Chinese and European design codes, respectively.

1. Introduction

Following the 2011 Fukushima disaster in Japan, to establish a new safety target and fully consider the operation condition of a nuclear power plant (NPP) in terms of design and accident control, the Design Extension Condition (DEC) was established and made a part of the Beyond Design Basis Accident (BDBA) conditions (IAEA, 2012), in which the reliability of a containment structure under high internal pressure when severe accidents occur is of vital importance.

Currently, CPR1000 PWR is the most popular reactor in mainland China. It is a new mega-kilowatt class pressurized water reactor improved at the Daya Bay Nuclear Power Station based on European civil design codes and nuclear power specifications (China General Nuclear Power Corporation, 2006). Since the import of the CPR1000 PWR prototype in the 1980s, the Chinese nuclear power industry has been developing rapidly for more than 20 years. By October 2016, in mainland China, 35 nuclear power reactors were in operation and 20 under construction, and the construction of more reactors is about to start to double the nuclear capacity to at least 58 GWe by 2020 or 2021, including six operating CPR1000 PWR units (e.g., at Hong Yanhe, Fang Chenggang, and Ling Ao) by October 2015 and two under construction (Schneider, 2015). The reliability assessment of the CPR1000 PWR containment is becoming more and more necessary and momentous in terms of the public safety, the environmental protection and the long-term development of nuclear power industry.

The CPR1000 PWR containment structure is a typical pre-stressed reinforced concrete structure with a steel liner inside. Based on the strategy of defense in depth, the containment should be able to endure a disruptive core explosion and the subsequent hyperthermia, overpressure and debris, and simultaneously be able to prevent the radionuclides from leaking into the external environment. The prototype of CPR1000 PWR was designed according to French NPP design codes that have evolved into a collection of matured codes including the EN (European Norm) series regulated by the European Committee for

* Corresponding author at: Institute of Earthquake Engineering, Dalian University of Technology, Dalian 116024, Liaoning, China.
E-mail addresses: hoomar@mail.dlut.edu.cn (L. Zhou), jianboli@dlut.edu.cn (J. Li), hzhong@dlut.edu.cn (H. Zhong), gaolin@dlut.edu.cn (G. Lin).

https://doi.org/10.1016/j.nucengdes.2018.02.005
Received 6 August 2017; Received in revised form 30 January 2018; Accepted 4 February 2018

Standardization (CEN) and European Committee for Electrotechnical Standardization, EPR (European Pressurized Reactor) series set by the Electricity De France (EDF), and RCC (Règles de Conception et de Construction) series formulated by AFCEN (Association Française pour les règles de Conception, de construction et de surveillance en exploitation des matériels des Chaudières Electro Nucléaires). The prestressed reinforced concrete containment structure of CPR1000 PWR mainly follows the Design and Construction Rules for Civil Works of PWR Nuclear Islands (RCC-G) and the Technical rules for Design and Calculations Relating to Reinforced Concrete Structures and Buildings (CCBA68) of French. Besides, at the same time the CPR1000 PWR containment has to fulfil the requirements of the Chinese NPP design codes reflecting the different demands on the design, construction, operation, inspection et al. that should be specifically considered in the safety assessment of CPR1000 PWR.

To assess the ultimate bearing capacity (UBC) of a large-scale engineering structure of critical importance such as a containment, there are basically two approaches. One is deterministic, i.e., modeling the structure numerically or experimentally with highly accurate material property parameters, geometric features, load inputs, boundary conditions, reasonable assumptions and appropriate simplifications. In view of the enormous size of a containment structure, a scaled model test is a reasonable substitute for the original structure. Considerable studies have employed scaled model experiments for determining the UBC of a containment under the internal pressure. Koenig (1986) studied a LWR steel containment with a 1:8 model test. Twidale and Crowder (1991) carried out an 1:10 model of the Sizewell 'B' PCCV building in order to validate a NPP design code. Zhang and Shu (2001) designed an 1:10 model test of a Pre-stressed Concrete Containment Vessel (PCCV). The Nuclear Power Engineering Corporation (NUPEC) of Japan conducted a series 1:8 proof tests of a reinforced concrete containment (RCC) with pressure and seismic load (Hirama et al., 2005). Hessheimer et al. (1997) illustrated an 1:4 PCCV model test at Sandia National Laboratories (SNL) that is one of the most rigorous and comprehensive model tests. The security redundancy when destructive earthquakes occur is studied thoroughly as well. Except the model tests by NUPEC, Qian (2007) studied the aseismic performance of the CNP1000 with an 1:10 pseudo-dynamic model test. Wang et al. (2014) carried out a shanking table tests of an 1:15 containment structure. Recently, the largest model test of the containment at present was accomplished by the Electricite De France (EDF) (Mazars et al., 2016). This test not only payed attention to the UBC under over pressurization and earthquake but also the performance degradation of the containment. Liu et al. (2016) carried out a shaking table test of an 1:20 CPR1000 PWR containment model considering the concrete part only. Although the effectiveness and accuracy can be satisfied by applying appropriate similarity laws, still, if it is allowed, the scale factor should be as close to one as possible. It is difficult to simulate the ultimate state of a scaled model with the same mechanical parameters of its prototype.

As for deterministic numerical analysis, to eliminate the difficulty in modeling complex structures and decrease the computation efforts, numerical models must be well simplified to some extent. Fardis and Nacar (1984) studied nonlinear response of a steel-lined RCC with a axisymmetric model neglecting the presence of penetrations. Akbar and Gupta (1986) firstly performed a nonlinear analysis of a RCC subjected to inertia force due to earthquake with shell and membrane elements. Radulescu et al. (1997) studied the UBC of the CANDU 6 containment structures under pressure load with the openings, penetrations and the airlocks considered separately. Hu et al. (2000, 2006, 2016) studied the UPC of a RCC with 1/8 part of the structure by using ABAQUS. Ray et al. (2003), Song et al. (2009) and Chakraboty et al. (2017) studied the UBC of the containment structure (RCC, PCCV) with shell elements. Zhang et al. (2017) simplified the PCCV to a 1/2 axisymmetric model with integral constitutive model, improving the calculation efficiency. Tavakkoli et al. (2017) built a quite elaborate finite element model of a PCCV under internal pressure with ANSYS, focusing on the influence of

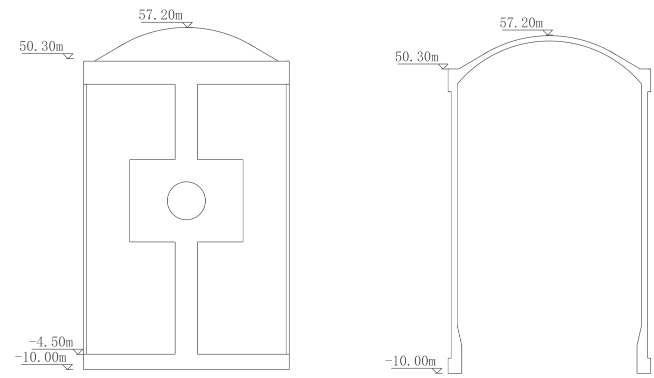

Fig. 1. Schematic of the CPR1000 PWR containment structure.

the prestressing tendon system on the UBC of the containment. The containment destruction pattern and overall response derived via numerical models with various numerical software, constitutions and simulation methods are basically consistent with the results of experimental modeling (Chen and Qian, 2002; Yonezawa et al., 2002; Xia et al., 2002; Basha et al., 2003; Anderson, 2008; Shokoohfar and Rahai, 2016; He et al., 2017).

The other approach is indeterminate, i.e., in terms of probability statistics, considering the uncertainty in material properties, load inputs or boundary conditions to form fragility curves to describe the reliability of the structure in the ultimate state (Pandey, 1997). In the nuclear industry, fragility curves have been widely utilized to evaluate the ability of several types of reactors to resist a certain earthquake degree or the efficiency of a vibration isolation system (Cho and Yang, 2005; Ozaki et al., 1998). Fragility curves can be attained by judgmental (Rossetto and Elnashai, 2003), empirical (Kiureghian, 2010), experimental (Banerjee et al., 2013), analytical or hybrid (Frankie, 2013) methods, among which the analytical methods are the most widely used due to its advantages of high flexibility, increasing reliability and extensive applicability. However, considering the disadvantages of highly time-consuming computational cost, the fragility analysis by analytical methods is mostly performed with simplified numerical models. Choi et al. (2008) used a lumped mass model to perform the fragility analysis of NPP structures. Zentner (2010) studied a reactor coolant system with a stick model. Hariri-Ardebili and Saouma (2016) developed a probabilistic seismic demand model for concrete dams with 2-D simplified model. So far it is very rare applying elaborate nonlinear analysis to get the fragility curves of complex structures.

In terms of reliability, the NPP design codes of USA, Europe, and China take into consideration the uncertainty such as that in the material properties and load. However, they differ in the corresponding specific regulations, possibly because of the differences in manufacturing level, technology level, and expected service life (Ning et al., 2010). For CPR1000 PWR, it is of importance to analyze the effect of these differences between European and Chinese NPP design codes on the reliability of a containment in case of an accident. However, this is not yet fully studied.

In this study, we aim to evaluate and compare the reliability of the CPR1000 PWR containment, via the Finite Element (FE) method and Monte Carlo method, based on random material mechanical parameters recommended by Chinese and European NPP design codes when some accidents occur generating an internal overpressure. With stochastic distributed parameters of material mechanical properties that can be derived from the corresponding specifications in the NPP design codes, a relatively large number of detailed numerical models are generated in which the nonlinear material properties and structural features are well preserved. In addition, practically measured data are used for comparison. Based on the presupposed failure criteria and single ultimate bearing capacity analysis and statistical analysis of the numerical models, fragility curves are obtained. Finally, we develop some insights