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Thermal–mechanical stress analysis of pressurized water reactor pressure vessel with/without a preexisting crack under grid load following conditions ☆

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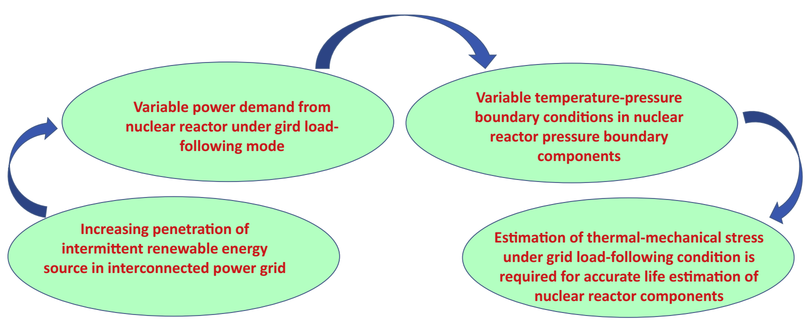
HIGHLIGHTS

- Use of intermittent renewable-energy source in power grid is becoming a trend.
- Gird load-following can leads to variable power demand from Nuclear power plant.
- Reactor components can be stressed differently under gird load-following mode.
- Estimation of stress–strain state under grid load-following condition is essential.

GRAPHICAL ABSTRACT



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ABSTRACT

In this paper, we present thermal–mechanical stress analysis of a pressurized water reactor pressure vessel and its hot-leg and cold-leg nozzles. Results are presented from thermal and thermal–mechanical stress analysis under reactor heat-up, cool-down, and grid load-following conditions. Analysis results are given with and without the presence of preexisting crack in the reactor nozzle (axial crack in hot leg nozzle). From the model results it is found that the stress–strain states are significantly higher in case of presence of crack than without crack. The stress–strain state under grid load following condition are more realistic compared to the stress–strain state estimated assuming simplified transients.

1. Introduction

Use of renewable energy such as solar and wind has increasingly become a worldwide goal to avoid catastrophic climate change. Widespread availability of clean, affordable, and reliable energy would also be a cornerstone of the world's increasing prosperity and economic growth (Chu and Majumdar, 2012). However, such renewable energy sources are only intermittently available and cannot reliably be used for base-load demand.

In many countries such as the U.S., France, South Korea, and Japan, nuclear energy is extensively used as a base-load source of electricity. However, when more and more renewable energy sources are connected to the electric grid, a question arises: do the nuclear power plants (NPPs) have the ability to adjust to a varying load from the interconnected grid, including daily and seasonal variations (Lokhov, 2011a,b; Bruynooghe et al., 2010; International Atomic Energy Agency, 2013; Savolainen, 2015;

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Ablay, 2013; Foley et al., 2010; Ingersoll et al., 2015)? Under the load-following mode, the pressure boundary components of NPPs may be subjected to additional thermal–mechanical cycles, particularly when the fluctuation in the gap between the grid demand and renewable energy supply is severe (in terms of both frequency and amplitude). Although most modern nuclear plants are designed to follow grid demands to a certain extent through consideration of a large safety factor, no study (at least none reported in the open literature) has been undertaken to determine the stress–strain state of reactor components under grid-load following. Most previous research on NPP component safety assessment, including our own (Mohanty et al., 2015a,b), is based on a stress analysis of components using simplified design transients. For accurate structural integrity assessment of NPP components, it is necessary to perform structural fatigue evaluation under more realistic loads (Wilhelm et al., 2013; Bergholz et al., 2012; Rudolph and Bergholz, 2008). In this regard, thermal–mechanical stress analysis of NPP components under the grid load-following mode might be necessary for accurate fatigue evaluation. In addition, since there are plans for increasing the life of current NPPs from their original design life of 40 years to the extended life of 80 years, aging-related material issues (Shah and MacDonald, 1993; Chopra and Stevens, 2014) can play additional detrimental role in the structural integrity of NPP components.

Argonne National Laboratory (ANL), under the sponsorship of the Department of Energy's Light Water Reactor Sustainability (LWRS) program, has been involved with extensive material testing (Mohanty et al., 2015c, 2014) and mechanistic modeling (Mohanty et al., 2015a,b; Mohanty et al., 2016) for assessing the structural integrity of NPP components under design and extended service conditions. In this paper we present detailed results from thermal mechanical stress analysis of a reactor pressure vessel (RPV) and its nozzle (both with and without preexisting crack) under typical reactor heat-up, cool-down, and load-following modes. The details of the proposed framework and related results are discussed below.

2. Finite element model of reactor pressure vessel and nozzles

We developed finite element (FE) models for both heat transfer analysis and for subsequent thermal–mechanical stress analysis. In our earlier work (Mohanty et al., 2015a,b) we presented a preliminary/skeletal FE model of an overall reactor consisting of a reactor pressure vessel, hot leg, cold leg, and steam generator. In this work, we present a detailed FE model of the reactor pressure vessel and its nozzle only. The major aim was to perform a stress analysis under realistic thermal–mechanical loading such as under grid load-following conditions and to study the stress–strain state of the RPV and its nozzles with/without the presence of crack. The details of the FE model are discussed below.

The models were developed by using commercially available ABAQUS FE software (Dassault Systèmes, 2014). The FE models are based on approximate geometry determined from publicly available literature (Shah and MacDonald, 1993; Schulz, 2006; Cummins et al., 2003; Westinghouse Electric, 2000, 2011). The RPV model includes a typical two-loop pressurized water reactor with two hot-leg (HL) nozzles and 4 cold-leg (CL) nozzles. Fig. 1 shows the outer/inner diameter (OD/ID) surface of the RPV and its HL and CL nozzles. For the requirement of modeling cracks, 3D models were developed and meshed by using eight nodded 3D brick elements. In our previous work (Mohanty et al., 2015a, b) we found that eight-node linear elements (DC3D8) were sufficient to model heat transfer compared to a computationally expensive counterpart of 20-node brick elements (DC3D10). For stress analysis, the corresponding C3D8, 8-node linear elements were used. Note that in our earlier work (Mohanty et al., 2015a,b), we considered other components such as the HL, CL, and steam generator; in the present work, however, we have not considered those components to reduce the computational burden by limiting the number of finite elements. Instead, we included detailed geometry of HL/CL nozzles and increased the number of elements along the thickness direction to allow modeling of preexisting cracks. Furthermore, in our previous work our intention was to perform preliminary system level FE model to estimate the typical stress–strain and displacement states under simplified design transients. However, in the present discussed work we present stress-analysis results under more realistic transients such as under grid load-following conditions and with/without presence of crack. In addition, compared to our earlier model (for which HL and CL nozzles were not included), we included detailed HL and CL nozzles to the FE model of RPV. Fig. 2 shows the OD and ID surface of the RPV and it nozzle FE mesh. A finer mesh was selected near the nozzle area for modeling possible stress hot spots arising due to the presence of openings, such as nozzles and preexisting cracks. The RPV and nozzle assembly have a total of 72,977 DC3D8 elements for heat transfer models or C3D8 elements for structural analysis models. The materials properties of 508 low alloy steel (508 LAS) is used for the FE modeling of all the sections of the RPV and its nozzles.

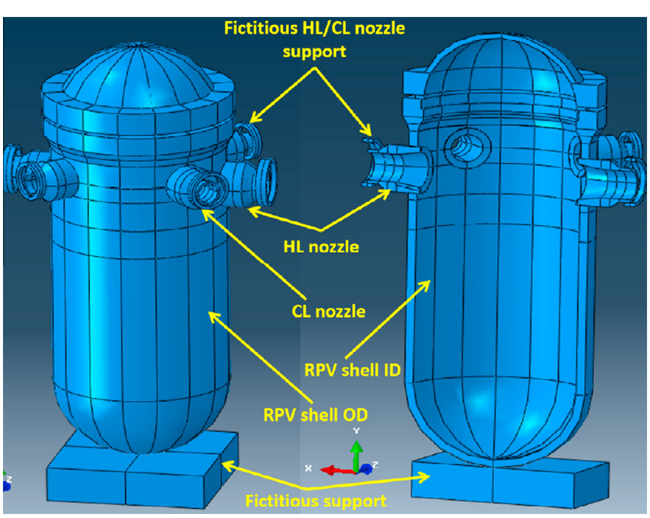

Fig. 1. Solid model of RPV and its nozzle.

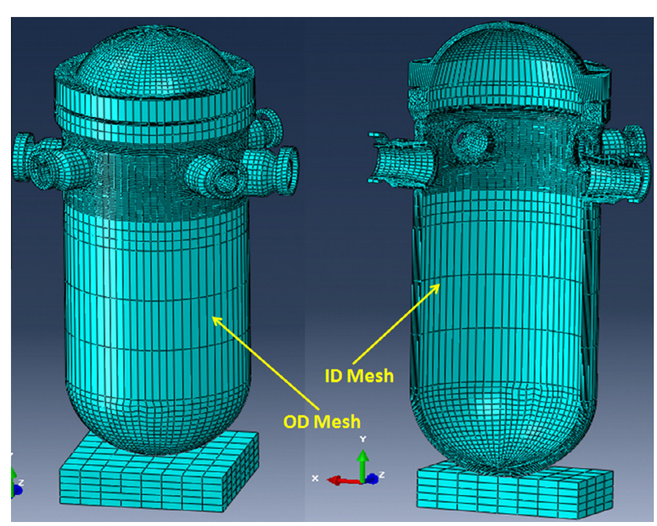

Fig. 2. RPV and its nozzle FE mesh.