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Fracture risk assessment for the pressurized water reactor pressure vessel under pressurized thermal shock events

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HIGHLIGHTS

- The PTS loading conditions consistent with the USNRC's new PTS rule are applied as the loading condition for a Taiwan domestic PWR.
- The state-of-the-art PFM technique is employed to analyze a reactor pressure vessel.
- Novel flaw model and embrittlement correlation are considered in the study.
- The RT-based regression formula of NUREG-1874 was also utilized to evaluate the failure risks of RPV.
- For slightly embrittled RPV, the SO-1 type PTSs play more important role than other types of PTS.

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ABSTRACT

The fracture risk of the pressurized water reactor pressure vessel of a Taiwan domestic nuclear power plant has been evaluated according to the technical basis of the U.S.NRC's new pressurized thermal shock (PTS) screening criteria. The ORNL's FAVOR code and the PNNL's flaw models were employed to perform the probabilistic fracture mechanics analysis associated with plant specific parameters of the domestic reactor pressure vessel. Meanwhile, the PTS thermal hydraulic and probabilistic risk assessment data analyzed from a similar nuclear power plant in the United States for establishing the new PTS rule were applied as the loading conditions. Besides, an RT-based regression formula derived by the U.S.NRC was also utilized to verify the through-wall cracking frequencies. It is found that the through-wall cracking of the analyzed reactor pressure vessel only occurs during the PTS events resulted from the stuck-open primary safety relief valves that later reclose, but with only an insignificant failure risk. The results indicate that the Taiwan domestic PWR pressure vessel has sufficient structural margin for the PTS attack until either the current license expiration dates or during the proposed extended operation periods.

1. Introduction

The reactor pressure vessel (RPV) is the most important pressure-boundary component of a nuclear power plant. During the power generation, the RPV shells subject to the neutron irradiation, leading to the localized embrittlement of vessel steel and shell weld materials in the beltline region corresponding to the reactor core. If an existing flaw of critical size exists in an embrittled RPV and some severe system transients were to occur, the flaw could initiate and then propagate rapidly through the vessel, leading to a through-wall crack and challenging the integrity of the RPV. To ensure the nuclear safety and structural integrity of RPVs in Taiwan, the licensee is strictly required by the regulatory body to perform the inservice inspection (ISI) periodically and thoroughly on all shell welds according to the requirement of ASME Boiler and Pressure Vessel Code, Section XI. In addition, the surveillance program also needs to be executed regularly to monitor the neutron fluence and radiation embrittlement of RPV materials.

The structural integrity of RPV is frequently determined by fracture mechanics that assumes the flaws have existed in the vessel wall. When evaluating the structural integrity of RPV, some key parameters such as chemical composition of metal materials, neutron irradiation, and flaw characteristics are difficult to determine directly and accurately because of the uncertainties due to measurement or other technique limits. In order to envelop the uncertainties as possible, evaluation based on deterministic approach usually considers conservative assumptions and thus inevitably produces the over-conservative result. Hence, over the recent decades, the probabilistic fracture mechanics (PFM) has been developed and gradually applied to evaluate the structural integrity

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Nomenclature

a_{KIC}	location parameter of Weibull probability function ($ksi\sqrt{in.}$)
b_{KIC}	scale parameter of Weibull probability function ($ksi\sqrt{in.}$)
c_{KIC}	shape parameter of Weibull probability function
cpi	instantaneous conditional probability of initiation
K_I	applied stress intensity factor ($ksi\sqrt{in.}$)
K_{Ia}	arrest fracture toughness ($ksi\sqrt{in.}$)
$RT_{MAX\text{-}XX}$	maximum reference temperature of beltline region materials (R)
RT_{NDT}	reference temperature of nil-ductility transition (°F)
T	crack tip temperature (°F)
α_{XX}	parameters of regression formula for beltline region materials
β	parameters of regression formula for vessel thickness
τ	time step of each transient (minute)
AW	axial weld
BWR	boiling water reactor
BWRVIP	BWR Vessel and Internal Project
B&W	Babcox & Wilcox
CE	Combustion Engineering
CPI	conditional probability of initiation
CPF	conditional probability of failure
CW	circumferential weld
ECCS	emergency core cooling system
EFPY	effective full power year
EOL	end-of-license
EPRI	Electric Power Research Institute
FAVOR	Fracture Analysis of Vessels-Oak Ridge
FCI	frequency of crack initiation
IGA	initiation-growth-arrest
ISI	inservice inspection
LOCA	loss-of-coolant accident
MSLB	main steam line break
ORNL	Oak Ridge National Laboratory
PFM	probabilistic fracture mechanics
PL	plate
PNNL	Pacific Northwest National Laboratory
PRA	probabilistic risk assessment
PTS	pressurized thermal shock
PWR	pressurized water reactor
RPV	reactor pressure vessel
SAW	submerged arc welding
SGTR	steam generator tube rupture
SIFIC	stress intensity factor influence coefficient
SMAW	shielded metal arc welding
SO-1	stuck-open primary safety relief valves
SO-2	stuck-open secondary safety relief valves
TWCF	through-wall cracking frequency
U.S.NRC	United States Nuclear Regulatory Commission
WPS	warm-prestress

of RPV. At first, the PFM was employed to evaluate the fracture risk of pressurized water reactor (PWR) pressure vessels subjected to pressurized thermal shocks (PTSs) for rule making (Dickson, 1995; U.S.NRC, 1982) by the United States Nuclear Regulatory Commission (U.S.NRC). Then, in 1990s, the BWR Vessel and Internal Project (BWRVIP) and U.S.NRC had evaluated the fracture probabilities for boiling water reactor (BWR) pressure vessel shell welds and finally concluded that the ISI on the circumferential shell welds can be relieved conditionally (BWRVIP, 1995; U.S.NRC, 1998a,b). On the other hand, the U.S.NRC's new PTS screening criteria for the PWR pressure vessels were also determined by the PFM analysis results (EricksonKirk et al., 2007; EricksonKirk and Dickson, 2010). Recently, the PFM has been widely employed for some operational issues such as pressure-temperature limits (Chou and Huang, 2014; Dickson et al., 2010, 2011a; EPRI, 2009), or the fracture phenomena investigation such as constraint effect, laminar flaws, and so forth (González-Albuixech et al., 2014; Qian and Niffenegger, 2015).

For a PWR, the severe transients of concern which challenge the integrity of RPV are known as the PTS events characterized by a rapid cooling of the downcomer and internal RPV surface, associated with potential re-pressurization of the RPV. The accidents or malfunctions that may cause the reactor coolant water temperature to be decreased rapidly and then trigger the PTS events can be mainly categorized by following reasons (EricksonKirk et al., 2007; EricksonKirk and Dickson, 2010):

(1) Pipe breaks in the primary system: Namely the loss-of-coolant accidents (LOCAs) resulted from the pipe breaks in the primary system. When the LOCA happens, the water level and pressure decrease and then the emergency core cooling system (ECCS) activates to make up the reactor water, thus resulting in rapid cooling of inner vessel wall. The controlling feature of LOCA type PTS transient is the size of the break. The larger the break, the faster the PTS proceeds, and the more severe the thermal shock occurs. However, the pressure is not significant in the type of PTS events.
(2) Stuck-open primary safety relief valves (SO-1): The SO-1 scenarios are similar to a small LOCA, but the later valve reclosure will induce the immediate re-pressurization of RPV. Therefore, for this kind of PTSs, both low temperature and high pressure take place.
(3) Breaks in the secondary system: Secondary side breaks include the main steam line break (MSLB) and the stuck-open valves in the steam system (SO-2). The breaks of secondary system saturate the secondary side water in the steam generator, which decreases the temperature of the primary side water flowing through the steam generator and then circulating back to the RPV. During the type of PTS events, the temperature would be higher than LOCA and SO-1 (at least 212–250 °F), but the primary side may still be in high pressure state, which challenges the integrity of the RPV.
(4) Others: Such as steam generator tube rupture (SGTR), mixed primary and secondary breaks, and so on.

The original PTS screening criteria of 10 CFR 50.61 prescribes that the maximum values of reference nil-ductility transition temperature (RT_{NDT}) at the end-of-license (EOL) are 270 °F for the axial welds and plates, and 300 °F for the circumferential welds, which was on the basis of the acceptable through-wall cracking frequency (TWCF) of 5×10^{-6}/year (Dickson, 1995; U.S.NRC, 1982). In 2010, the U.S.NRC issued the risk-informed revision of PTS screening criteria (10 CFR 50.61a) based on the allowable TWCF of 1×10^{-6}/year (EricksonKirk et al., 2007; EricksonKirk and Dickson, 2010). A series of researches including the radiation embrittlement (Eason et al., 2007), flaw distribution (Simonen et al., 2003), probabilistic risk assessment (PRA) (Whitehead et al., 2004; Wood et al., 2008), thermal hydraulic analysis (Arcieri et al., 2004; Chang et al., 2004), and PFM (Williams et al., 2007) were performed in more detail for the revised PTS rule. The technical basis was described in the NUREG-1806 and NUREG-1874 reports (EricksonKirk et al., 2007; EricksonKirk and Dickson, 2010).

In the paper, we aim at evaluating the structural reliability of a Taiwan domestic PWR pressure vessel under various PTS events referred from the technical basis of the U.S.NRC's new PTS rule. At first, a model which includes axial welds, circumferential