Structural Integrity Assessment of the SMART Reactor Pressure Vessel during Heat-up and Cool-down

Jong Sung Kim [a*], S.G. Lee [b], S.K. Chung[b], M.J. Jhung [c], Y.H. Ryu [c]
[a]*Dept. of Mechanical Engineering, Sunchon National Univ., 413, Jungang-ro, Suncheon, 540-742*
[b]*KEPCO E&C Co., PERI, 8 Kumi-ro, Seongnam, 463-870*
[c]*KINS, 34 Kwahak-ro, Daejeon, 305-338*
[*]*Corresponding author: kimjsbat@sunchon.ac.kr*

1. Introduction

Recently, efforts have been devoted worldwide to expand the peaceful utilization of the nuclear energy other than electricity generation. Therefore, small/ medium size multi-purpose advanced reactor draws keen attention in consideration of its adaptive nature, simplicity of reactor design, and passive safety approach. It is expected that the demand for small/medium size reactor will arise for various applications such as small capacity power production, heat generation, and energy source for seawater desalination in the near future [1].

SMART of 330MWt is on the process of design [2]. The SMART RPV (reactor pressure vessel) contains internal components such as steam generators, reactor coolant pumps, pressurizer, UGS (upper guide structure), etc. in the reactor and space among the internal components is filled with reactor coolant [2]. The SMART RPV has different geometric and operating characteristics such as beltline weld location, neutron irradiation fluence level, etc. from the RPV of commercial nuclear power plants. So, structural integrity of the SMART RPV during heat-up and cool-down has to be assessed.

In this study, P-T limit curves of the SMART RPV were derived by utilizing ASME B&PV Code, Sec.XI, App.G[3] and FAVOR Code[4]. And then, structural integrity of the RPV during low temperature nuclear heating was evaluated using finite element analysis and ASME B&PV Code, Sec.XI, App.G.

2. P-T Limit Curves

2.1 Target Model and Method

Fig. 1 shows the SMART RPV and the beltline weld location. AS shown in Fig. 1, the beltline weld is located on the connection part between lower head and cylinder shell unlike the commercial RPV. Input data to derive P-T limit curves are as follows:.

·Material : SA508 Gr.3 Cl.1
·Maximum initial RT_{NDT}(beltline region) : -17.78℃
·Maximum initial RT_{NDT}(other region) : 4.44℃
·Chemical factor(base metal) : -6.67℃
·Chemical factor(weld metal) : -1.94℃
·Margin(base metal) : 27.78℃
·Margin(weld metal) : 31.11℃
·Initial Temperature : 21.1℃
·Neutron irradiation fluence (beltline region at 60year) : 1.0997×10^{14}n/cm^2
·Equivalent heat convection coefficient : 193.5～252.8 W/m^2 ^{o}C
·Crack depth ratio a/t : 1/4
·Crack shape ratio a/2b : 1/6
·Crack direction : axial

The P-T limit curves were derived via the assessment procedure of ASME B&PV Code, Sec.XI, App.G. Thermal stress intensity factors were calculated using FAVOR Code.

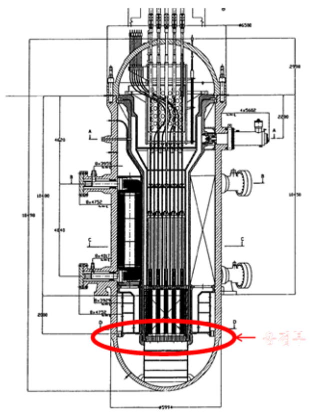


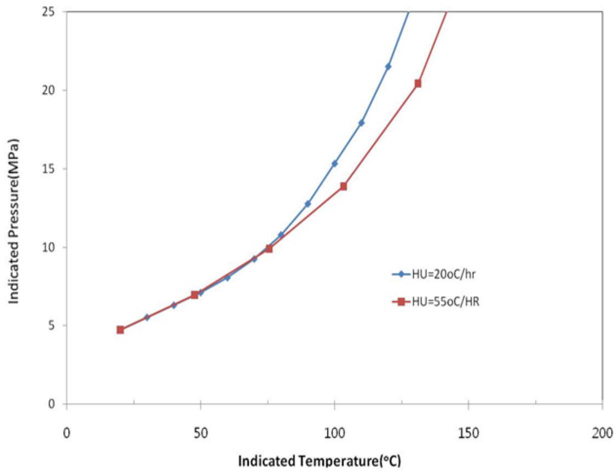
Fig. 1. Configuration of the SMART RPV and location of the beltline weld

2.2 Results

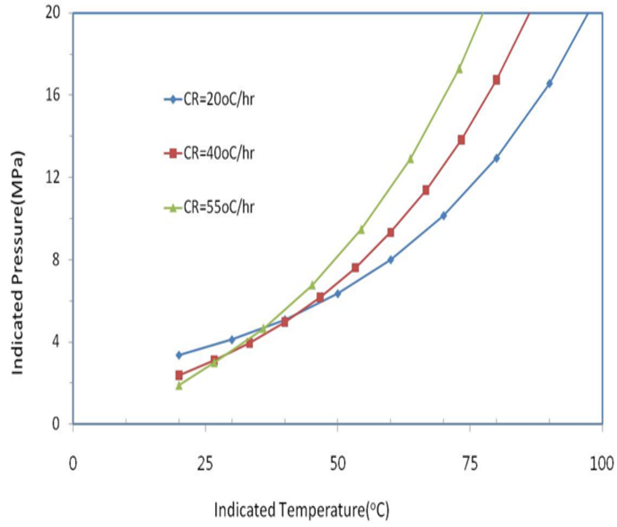
Fig. 2 depicts variation of the P-T limit curves vs. heat-up and cool-down rates. As depicted in Fig. 2, allowable operating temperature during heat-up increases under identical operating pressure with increase of heating rate. In general, allowable operating temperature during cool-down decreases under identical operating pressure with increase of cooling rate but there shows counter-tendency below 40^{o}C.

3. Structural Integrity during Low Temperature Nuclear Heating

3.1 Target Model and Method



(a) heat-up



(b) cool-down

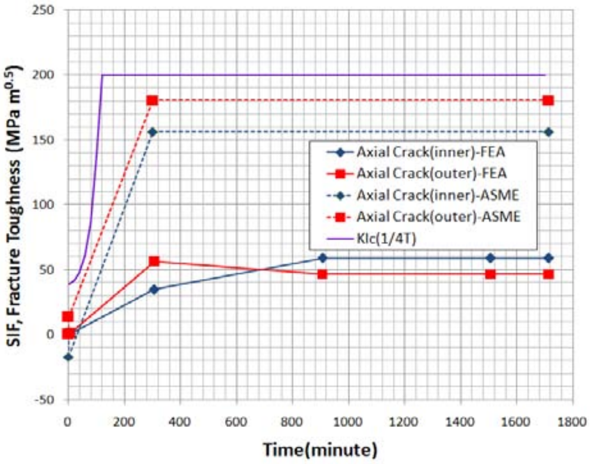
Fig. 3. P-T limit curves of the SMART RPV beltline weld

Finite element models of the SMART RPV with inner/outer axial and circumferential cracks were developed using FEA-Crack[5]. Low temperature nuclear heating was assumed. That is feedwater boiler was assumed to be failed at initial stage of heat-up. Crack depth ratio and shape ratio are 1/4 and 1/6, respectively. Fluid temperature contacted with inner surface of the RPV changes from 21°C to 299°C. Inner pressure linearly increases from 0MPa to 15MPa.

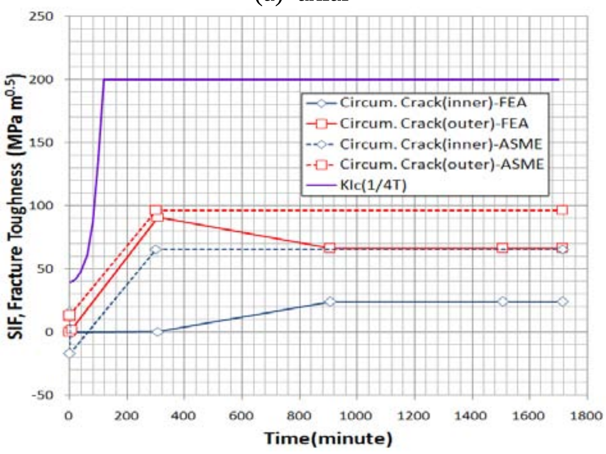
Structural integrity of the RPV during low temperature nuclear heating was evaluated via the finite element analysis using ABAQUS[6] and ASME B&PV Code, Sec.XI, App.G.

3.2 Results

Fig. 4 depicts fracture analysis results of cracks on the beltline weld during heat-up. As depicted in Fig. 5, fracture doesn't occur for all the cracks because stress intensity factors are lower than fracture toughness. That is, structural integrity of the RPV is maintained even during low temperature nuclear heating.



(a) axial



(b) circumferential

Fig. 4. Fracture analysis results of cracks on the beltline weld

3. Conclusions

The following conclusions are found via the structural integrity assessment of the SMART RPV during heat-up and cool-down:

- Effect of cooling rate on the P-T limit curves is relatively greater than commercial PWR,
- Structural integrity of beltline weld is maintained even during low temperature nuclear heating.

REFERENCES

[1] KINS, KINS/RR-689, 2009.
[2] KAERI, License Application Report for Standard Design of Integral Reactor,2010.
[3] ASME B&PV Code, Sec.XI, App.G, 2007.
[4] T.L. Dickson, FAVOR Code, ORNL/NRC/LTR/94/1, 1994.
[5] Quest Reliability, FEA-Crack, Ver.3.0, 2010.
[6]Simulia, ABAQUS User's Manuals, Ver.6.8, 2009.