

**Original Article****Investigation of Burst Pressures in PWR Primary Pressure Boundary Components***Ihn Namgung\** and *Nguyen Hoang Giang*<sup>1</sup>

Department of NPP Engineering, KEPCO International Nuclear Graduate School, 689-882 Haemaji-ro, Seosaeng-myeon, Ulju-gun, Ulsan 45014, Republic of Korea

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**ABSTRACT**

In a reactor coolant system of a nuclear power plant (NPP), an overpressure protection system keeps pressure in the loop within 110% of design pressure. However if the system does not work properly, pressure in the loop could elevate hugely in a short time. It would be seriously disastrous if a weak point in the pressure boundary component bursts and releases radioactive material within the containment; and it may lead to a leak outside the containment. In this study, a gross deformation that leads to a burst of pressure boundary components was investigated. Major components in the primary pressure boundary that is structurally important were selected based on structural mechanics, then, they were used to study the burst pressure of components by finite element method (FEM) analysis and by number of closed forms of theoretical relations. The burst pressure was also used as a metric of design optimization. It revealed which component was the weakest and which component had the highest margin to bursting failure. This information is valuable in severe accident progression prediction. The burst pressures of APR-1400, AP1000 and VVER-1000 reactor coolant systems were evaluated and compared to give relative margins of safety.

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**1. Introduction**

Since the Fukushima accident, safety of nuclear power plants (NPPs) is of utmost concern. One area of such concern is burst pressure assessment. ASME BPVC set up rules for design, operation and inspection of NPP components for the prevention of failure of pressure boundary components and for the safety of operation. However, the normal design process

following ASME BPVC does not include assessment of failure pressure of primary pressure boundary components or the balance between components such as margin of failure. It has become more pronounced if a severe accident is to be considered since severe accident progression is also related to the failure margin of components. The weakest component, location or position will fail first in the event of a severe accident.

\* Corresponding author.

E-mail address: [inamgung@kings.ac.kr](mailto:inamgung@kings.ac.kr) (I. Namgung).

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<sup>1</sup> This author is now with the National Research Institute of Mechanical Engineering, Ha Noi, Vietnam.

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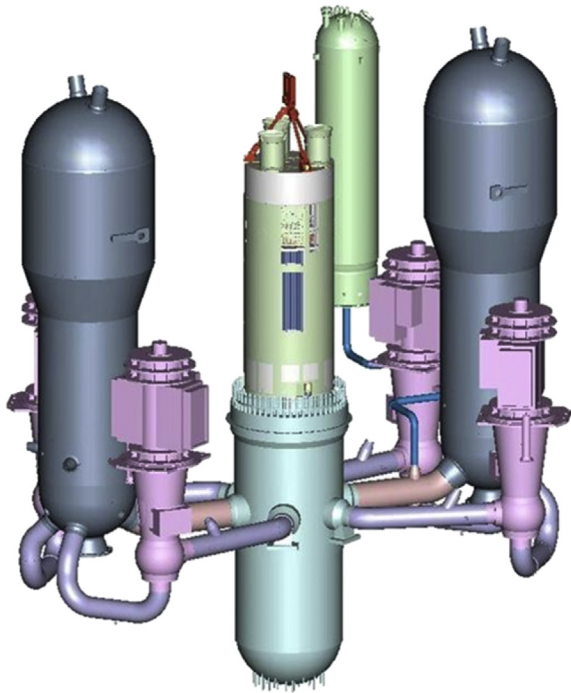


Fig. 1 – APR-1400 reactor coolant system.

The burst pressure estimate is usually regarded as part of beyond design basis accident (BDBA) evaluation and provides a scenario of accident progression at an initial stage. The knowledge and information on burst pressure can provide valuable information in emergency guides and preparation for accidents and also gives an indication of the location of the weak point. Hence the burst pressure estimation can screen out weak components in the system and quantifies the design margin to actual failure.

The knowledge of failure pressure of the primary reactor coolant system is also useful in optimizing the system design or planning accident progression. The designer can make components to have the same level of safety margin or purposely make the failure design margin smaller than other components so that failure progression can follow predefined route and minimize the impact. This is very useful in that a steam generator primary side head and heat transfer tube be designed with more margin so that any accident in the primary side can be confined within the primary loop or containment building. Moreover it prevents leakage of radioactive materials to the secondary side of the steam generator.

In this study, the burst pressure of pressure boundary components has been investigated by using an analytical formula including empirical relationship and finite element method (FEM) analysis using ANSYS, a general purpose FEM analysis software. A comprehensive review on theoretical formulation of burst pressure prediction models as well as comparison with test data was given by Christopher et al. 2002 [1], Law and Bowie. 2007 [2], and Zhu and Leis. 2012 [3]. In this paper, burst pressure of the pressurized water reactor (PWR) primary coolant system primary pressure boundary component of APR-1400, AP1400, and VVER-1000 was investigated and compared.

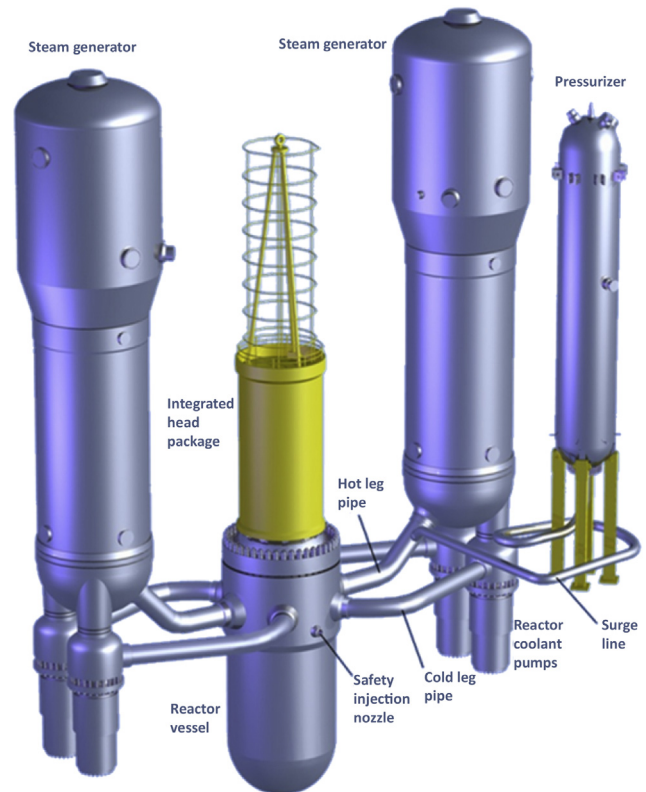


Fig. 2 – AP1000 reactor coolant system.

## 2. Review of PWR primary coolant system

### 2.1. APR-1400 reactor coolant system

APR-1400 is an advanced pressurized nuclear reactor, design by the Korea Hydro and Nuclear Power, Kyungju, Korea developed from the OPR-1000 design [4]. The APR-1400 reactor has a thermal power capacity of 4,000 MW and produces 1,400 MW gross electric power [4]. The reactor coolant system (RCS) includes a reactor vessel (RV) with two coolant loops. The major components of a loop include one steam generator (SG), two reactor coolant pumps (RCPs), a pressurizer and associated piping. All components above are located inside the containment building. The arrangement of the RCS is shown in Fig. 1.

Each of the two heat transfer loops of APR-1400 RCS piping contains five sections of pipes. One 1,066.8 mm inner diameter pipe connects the RV outlet nozzle and SG inlet nozzle, two 762 mm internal diameter pipes connect from the SG's two outlet nozzles to the RCP's suction nozzle, and two 762 mm inner diameter pipes connect the pumps discharge nozzle to the RV inlet nozzles. These pipes are referred to as the hot leg, the suction legs, and the cold legs, respectively. The pipes are made of steel SA 516 Grade 70.

### 2.2. AP1000 reactor coolant system

AP1000 is a pressurized water reactor designed by Westinghouse Electric Company, Pittsburgh, PA, USA [5]. The reactor

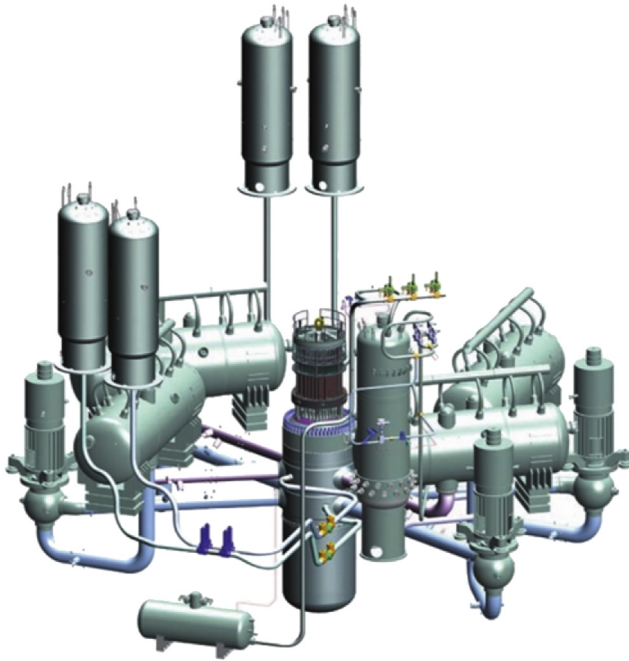


Fig. 3 – VVER-1000 reactor coolant system.

produces a net output of over 3,400 MW thermal, and nominal electrical output of 1,110 MW, from US NRC (2011). The AP1000 RCS consists of two heat transfer circuits. Each circuit includes an SG, two RCPs, and a single hot leg pipe and two cold leg pipes for circulation of reactor coolant material. In addition, the system includes the pressurizer, interconnecting piping, valves, and the instrumentation for operational control and safeguards actuation. Fig. 2 shows arrangement of the AP1000 RCS.

Each loop of AP1000 RCS piping includes sections of hot leg pipe and cold leg pipes interconnecting the RV, SGs, and RCPs. There are three sections of pipe in each heat transfer loop: one 787.4 mm inner diameter pipe between the RV outlet nozzle and SG inlet nozzle; two 558.8 mm inner diameter pipes from the RCP discharge nozzle to the RV inlet nozzle. RCS piping is fabricated from austenitic stainless steel SA 376 TP304, and fabricated according to ASME code, section III, class 1 requirements.

AP1000 RCS design is the latest development utilizing a canned RC pump. As shown in Fig. 2, the inlet to the RC pump is directly connected to the outlet of the steam generator; hence, the loop closure piping is unnecessary and simplifies the coolant pipe layout. This design utilizes a canned motor to simplify the loop design.

### 2.3. VVER-1000 reactor coolant system

VVER is a pressurized water reactor developed by Rosatom Corporation, Moscow, Russia. The reactor delivers about 3,000 MW of thermal power, and produces an output of 1,000 MW electricity. The VVER reactor coolant system consist of an RV with four heat transfer loops, each loop includes an

SG, RCP, a single outlet pipe (hot leg pipe), and an inlet pipe (cold leg pipe).

In each loop, the outlet pipe connects the RV outlet nozzle with the SG inlet collector, the suction pipe connects the SG outlet with the RCP inlet nozzle, and the inlet pipe connects the RCP outlet nozzle and RPV inlet nozzle. Four loops are installed at the same level. The angle between Loop 1 and Loop 2 is 55°, and the angle between Loop 2 and Loop 3 is 125°. The pair of Loops 1 and 2 are diagonally opposite the pair of Loops 3 and 4. Fig. 3 shows the arrangement of the VVER-1000 reactor primary coolant system.

VVER-1000 RCS piping comprises the circulation loop, each loop has three pipe sections. The section between the PWR outlet nozzle and the SG inlet collector is a hot leg pipe. The section between the SG outlet collector and inlet (intake) nozzle of the RCP set is a suction leg pipe. The section between the outlet nozzle of the RCP set and the PWR inlet nozzle is a cold leg pipe. The dimension of the internal diameter is 850 mm with a nominal thickness of 140 mm. The material used in RCS piping is steel 10GN2MFA (10ГН2МФА).

### 2.4. Burst pressure formulation for thick shell

When the thickness of the vessel is relatively large, the variation in the stress from the inner surface to the outer surface becomes appreciable. For a thick shell cylinder,  $\sigma_h$  is the hoop stress and  $\sigma_r$  is the radial stress. The stress in this thick shell can be shown as Eq. (1) [6].

$$\sigma_h - \sigma_r - r \frac{d\sigma_r}{dr} = 0 \quad \text{or} \quad \frac{\sigma_h - \sigma_r}{r} = \frac{d\sigma_r}{dr} \quad (1)$$

Note that  $\sigma_h$  and  $\sigma_r$  are maximum and minimum principal stresses respectively, thus by applying Tresca's yield criteria we can state when the maximum shear stress doubles and reaches yield strength, yielding occurs at the inside the vessel surface.

$$\tau_{max} = \tau_y = \frac{\sigma_h - \sigma_r}{2} \quad (2)$$

As the pressure continues to increase, plastic deformation penetrates farther into the wall of the vessel until it reaches the outer surface with an assumption that the material is perfectly plastic and shear stress is constant through the walled thickness. Thus every point in the walled thickness of the vessel is in the plastic state and reaches the shear stress value as in Eq. (2). Substituting Eq. (2) to Eq. (1) gives:

$$\frac{2\tau_y}{r} = \frac{d\sigma_r}{dr} \quad (3)$$

At the inner surface of the vessel,  $r = a$ , the radial stress is equal to the internal pressure, hence, integration of both sides of Eq. (4) gives:

$$\sigma_r = 2\tau_y \ln r + C \quad (4)$$

From the condition that at the outer surface of the vessel,  $r = b$ , the radial stress become zero. Applying a boundary condition, the integration constant is obtained as  $C = -2\tau_y \ln b$ . Thus,

$$\sigma_r = 2\tau_y \ln \frac{r}{b}$$

At the inner surface of the vessel,  $r = a$ , the radial stress equals the internal pressure, hence,

$$p = -\sigma_{r_a} = -2\tau_y \ln \frac{a}{b} \quad \text{or} \quad p = 2\tau_y \ln \frac{b}{a} \quad (5)$$

The pressure expressed in Eq. (5) is the required value of pressure to bring the entire wall of the vessel into a state of plastic flow. An assumption is made that upon reaching the burst pressure, the shearing stresses are uniform over the entire thickness and equal to the ultimate shearing strength of the material. The burst pressure is determined as:

$$p_b = 2\tau_u \ln \frac{b}{a} \quad (6)$$

The value  $\tau_u$  is difficult to obtain, so by applying the Tresca's criterion for material at a critical point,  $2\tau_u$  is equal to the ultimate tensile strength of material  $\sigma_u$ , and the burst pressure is determined by Eq. (6).

$$p_b = \sigma_u \ln \frac{D_o}{D_i} \quad (7)$$

Eq. (7) is known as the Turner Equation based on Tresca's yield criteria [4].

Svensson [7] presented a burst pressure prediction formula for material with strain hardening case based on von Mises criteria [1,7].

$$p_b = \sigma_u \left[ \frac{0.25}{(n + 0.227)} \left( \frac{e}{n} \right)^n \right] \ln k \quad (8)$$

where  $k$  is the ratio between the inner diameter and outer diameter and  $n$  is the straining hardening coefficient. These are defined as:

$$k = \frac{D_o}{D_i}$$

and

$$n = 0.224 \left( \frac{\sigma_{ult}}{\sigma_{ys}} - 1 \right)^{0.604}$$

Faupel and Fisher [8], carried out experiment and presented the burst pressure prediction formula incorporating yield and ultimate tensile strength based on von Mises criteria [2,3,8].

$$p_b = \frac{2}{\sqrt{3}} \sigma_y \left( 2 - \frac{\sigma_y}{\sigma_u} \right) \ln k \quad (9)$$

Bailey and Nadai [6] developed a burst pressure formula based on maximum shear stress plasticity relations.

$$p_b = \frac{\sigma_u}{2n} \left[ 1 - \frac{1}{(k)^{2n}} \right] \quad (10)$$

Soderberg [6] proposed a burst pressure prediction formula based upon assumption of uniform stress distribution throughout the wall by considering the average stress value and failure as a function of the significant stress or octahedral shear stress.

$$p_b = \frac{4}{\sqrt{3}} \sigma_u \left( \frac{k-1}{k+1} \right) \quad (11)$$

The 1962 edition of the ASME Boiler code section VIII gives the equation for allowable maximum pressure for a pressure vessel. The estimation of burst pressure is not included in the current ASME code.

$$p_b = \sigma_u \left( \frac{k-1}{0.6k+0.4} \right) \quad \text{for } k \leq 1.5 \quad (12)$$

The equations given above are used in the comparison with FEM results in this study.

## 2.5. Burst pressure prediction of NPP primary coolant system using FE analysis

In the FEM analyses, the reactor primary pressure boundary, reactor vessel, steam generator, RC pump, pressurizer, and main pipes, etc., are subjected to internal pressure. It is assumed that external pressure is negligible compare to internal pressure. The failure pressures are obtained by increasing the internal pressure from design pressure; at each increment in the value of internal pressure, the maximum stress is determined and compared with critical strength of the materials. Yield pressure is defined as the pressure value at which the material starts to yield, and burst pressure is defined as the pressure value at which entire wall thickness of the vessel reaches the magnitude of ultimate tensile strength. The increase in pressure was controlled so as not to exceed 0.1 MPa in the plastic region and the typical step size in pressure before the failure was 0.01 MPa to obtain more accurate results. The nonlinear large deflection analysis (plastic analysis) stops when it reaches burst conditions due to large translational movement of the element. In this instance, the stiffness matrix becomes singular and contains large translational deformation of the nodal vector that can be a linear combination of other nodal vectors. This is the instance when burst failure occurs.

## 2.6. Material properties and pressure boundary component models

Both APR-1400 and AP1000 reactor pressure vessels are made of the same SA-508 Gr3 Class1 low alloy steel. The material properties are given in Table 1. It has relatively high tensile strength at normal operating temperatures. Table 2 shows VVER-1000 reactor vessel material properties. In the analysis the cladding was not considered as structural material, hence it was ignored in the analysis. In the tables,  $\nu$  is Poisson's ratio,  $\rho$  is density,  $\sigma_y$  is yield strength, and  $\sigma_u$  is ultimate strength.

APR-1400 RCS piping consists of two hot leg pipes, four cold leg pipes and four loop closure pipes for a 2 + 4 loop

**Table 1 – Material properties of SA-508 Gr3 Class 1 low alloy steel for the reactor vessel of APR-1400 and AP1000.**

Temperature (°C)	$\nu$	$P$ (g/m <sup>3</sup> )	E (GPa)	$\sigma_y$ (MPa)	$\sigma_u$ (MPa)	$n = \sigma_y/\sigma_u$
21	0.3	7,833.4	186.8	345	550	0.63
343.3	0.3	7,833.4	173.1	286.2	550	0.52

**Table 2 – Material properties of 15Kh2MNA steel for the reactor vessel of VVER-1000.**

Temperature (°C)	$\nu$	$\rho$ (g/m <sup>3</sup> )	E (GPa)	$\sigma_y$ (MPa)	$\sigma_u$ (MPa)	$n = \sigma_y/\sigma_u$
21	0.3	7,860	205	622	725	0.86
350	0.3	7,778	185.5	441	539	0.82

configuration. The hot leg pipe inner diameter is 1,066.8 mm with a nominal thickness of 101.6 mm. The loop closure pipe and cold leg pipe inner diameter is 762 mm with a nominal thickness of 76.2 mm. The APR-1400 RCS piping is fabricated with SA-516 Grade 70 carbon steel. The normal operation temperature of hot leg pipes is 323.9°C and cold leg pipes is 290°C. Table 3 shows material properties of SA-516 at 21°C and 343°C [4].

AP1000 RCS piping includes three sections, i.e., one hot leg pipe and two cold leg pipes. The hot leg pipe has an inner diameter of 787.4 mm, and a nominal thickness of 82.6 mm. The cold leg pipes have inner diameters of 558.8 mm and nominal thicknesses of 65 mm AP1000 RCS piping is fabricated with SA-376 TP304 austenitic stainless steel. The normal operation temperature of hot leg pipes is 321°C and cold leg pipes is 279.44°C. The burst pressure analysis was performed based on design temperature of the RCS 343°C. Table 4 shows the main material properties of SA-376 TP304 steels at 21.1°C and 343°C [9].

VVER-1000 RCS piping includes three sections, a hot leg pipe, a loop-closure pipe, and a cold leg pipe. The main pipes have the same inner diameter of 850 mm and nominal thickness of 140 mm. The material used in the RCS piping is 10GN2MFA steel. At normal operating conditions, the temperature at the hot leg pipe is 318°C, and that for the cold leg pipe is 287°C. The burst pressure evaluation was done based on material properties at a temperature of 350°C. This will give slightly conservative results since at higher temperatures the yield and tensile stress is lower. Table 5 shows material properties of 10GN2MFA steel for analysis [5].

For the estimation of steam generator burst pressure, the heat transfer tube, commonly called the U-tube, is considered for burst pressure. The material for the APR-1400 steam generator heat transfer tube is Inconel alloy 690 (SB-163 N06609), and the material properties are given in Table 6. For the VVER-1000 SG heat transfer tube, the material is 08XH10T alloy and the properties are given in Table 7.

**Table 3 – Material properties of SA-516 Grade 70 steel for the RCS pipe of APR-1400.**

Temperature (°C)	$\nu$	$\rho$ (g/m <sup>3</sup> )	E (GPa)	$\sigma_y$ (MPa)	$\sigma_u$ (MPa)	$n = \sigma_y/\sigma_u$
21	0.3	7,850	202	262	482.6	0.54
343.3	0.3	7,850	182	194	482.6	0.4

RCS, reactor coolant system.

**Table 4 – Material properties of SA-376 TP304 steel for the RCS pipe of AP1000.**

Temperature (°C)	$\nu$	$\rho$ (g/m <sup>3</sup> )	E (GPa)	$\sigma_y$ (MPa)	$\sigma_u$ (MPa)	$n = \sigma_y/\sigma_u$
21	0.3	7,850	195	206.8	517.1	0.4
343.3	0.3	7,850	174	128	400	0.32
RCS.						

## 2.7. FEA models

The burst pressure assessment using FEA methods has been investigated widely [10–12]. Dwivedi et al [10] reviewed various types of methods, formulae, and theories for burst pressure calculation. Xue et al [11] demonstrated accurate calculation of the burst pressure using the FEM method employing nonlinear material behavior and geometric nonlinearity. Zhu and Leis [12] developed burst pressure prediction based on the average shear stress yield theory, proposed criteria of burst pressure, and made comparisons between experimental data and the results obtained by the FEM method.

Axisymmetric reactor vessel models for APR-1400, AP1000, and VVER-1000 were developed using ANSYS version 14.5 (ANSYS Systems Co.) [13]. The analysis models are shown in Figs. 4–6. The models include lower head and cylindrical shell sections only since nozzles such as cold leg, hot leg, control rod drive mechanism (CRDM) nozzles, and DVI nozzles are reinforced for the openings, therefore, the burst pressure is much higher. Also the vessel flanges and closure head flanges are reinforced to a bolted connection. These areas were not investigated in this study; however, these regions are reinforced as well.

Figs. 4–6 show where the maximum stress occurs. The inner surface of the model is set to inner pressure and the pressure varies from low to high. As the pressure increase, analysis should be carried out independently in a separate static analysis. From these analyses, the pressure load that produces yield stress and tensile stress through the wall can be recognized. Note also that from the figures, the shell section experiences higher stress than head as predicted by theory. This iterative analysis process is applied to other components as well.

In order to simplify the RCS primary pipe FEA analysis, straight section and elbow sections are modelled separately. Fig. 7 shows a typical model of straight section and various elbow sections of APR-1400 primary pipes. Also symmetry of geometry was taken into account to reduce the model size. In the case of AP1000 there is no loop closure pipe since the RC pump is directly connected to the SG. Also the radius of the

**Table 5 – Material properties of 10GN2MFA steel for the RCS Pipe of VVER-1000.**

Temperature (°C)	$\nu$	$\rho$ (g/m <sup>3</sup> )	E (GPa)	$\sigma_y$ (MPa)	$\sigma_u$ (MPa)	$n = \sigma_y/\sigma_u$
21	0.29	8,190	206.9	275	586	0.47
343.3	0.31	8,190	190.3	217	551.5	0.39

RCS, reactor coolant system.

**Table 6 – Material properties of the APR-1400 and AP1000 steam generator heat transfer tube (U-tube), Inconel Alloy 690.**

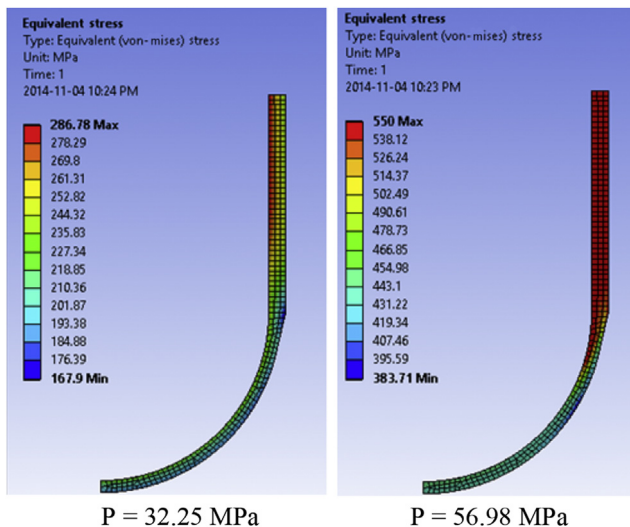
Temperature (°C)	$\nu$	$\rho$ (g/m <sup>3</sup> )	E (GPa)	$\sigma_y$ (MPa)	$\sigma_u$ (MPa)	$n = \sigma_y/\sigma_u$
21	0.29	8,190	206.9	275	586	0.47
343.3	0.31	8,190	190.3	217	551.5	0.39

**Table 7 – Material properties of 08XH10T alloy for the VVER-1000 steam generator heat transfer tube (U-tube).**

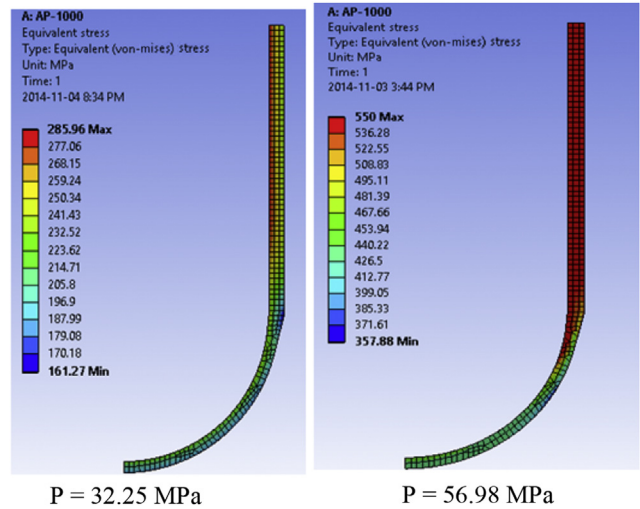
Temperature (°C)	$\nu$	$\rho$ (g/m <sup>3</sup> )	E (GPa)	$\sigma_y$ (MPa)	$\sigma_u$ (MPa)	$n = \sigma_y/\sigma_u$
20°C	0.3	7,860	200	345	540	0.64
350°C	0.3	7,778	181	295	490	0.6

elbow is much larger than the APR-1400 RCS primary pipe in order to reduce local stress concentration. For VVER-1000 RCS primary pipes, the inner diameter (ID) and thickness of pipe are the same for hot leg, cold leg, and loop closure pipes. Therefore the burst pressure at straight sections of the RCS pipe has the same burst pressure, hence the weakest location in VVER-1000 RCS pipes is at the inside elbow (crotch point) where the maximum stress occurs.

The steam generator heat transfer tube, which is called the U-tube, is composed of straight section and bended section. The bended section is fabricated by a bending operation which results in thinning of the outside bend wall and thickening of the inside bend wall. This change of wall thickness creates a stress distribution difference from that of the uniform thickness of the bend as in the case of the RCS bend, showing benign behavior. As the internal pressure increases, the straight section fails first, compared to the bended section [6]. Hence, the straight section of the SG heat transfer tube is modelled for FEM analysis and the FEM model is shown in Fig. 8.



**Fig. 4 – Finite element method model of reactor vessel of APR-1400 and stress distribution.**

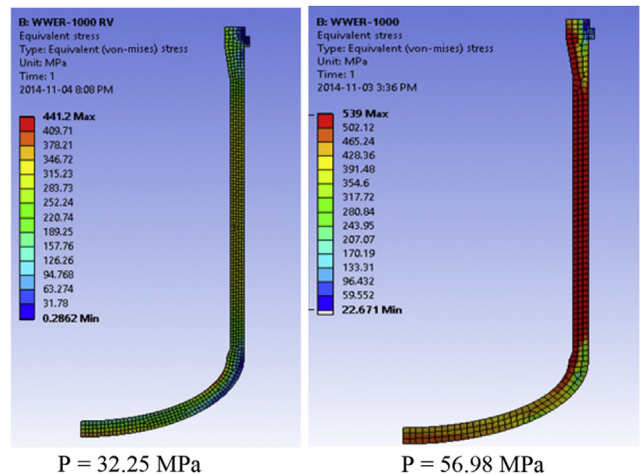


**Fig. 5 – Finite element method model of reactor vessel of AP1000 and stress distribution. P, pressure.**

**2.8. Burst pressure estimation of reactor pressure vessels**

In order to simplify the model, the symmetry of geometry is used and the symmetry plane is set to frictionless support. The analyses were performed with a large-deflection nonlinear option with bilinear plasticity. Table 8 summarizes the FEM analysis results of burst pressures for APR-1400, AP1000, and VVER-1000 reactor pressure vessels. In the process of calculating the burst pressure, yield pressures are also obtained by observing the pressure that produces 0.2% permanent plastic strain. The yield pressure is the point at which yield can occur in the reactor vessel’s inner surface, hence this pressure is the initiation point of plastic deformation as pressure increases.

Analytical solution to burst pressures of reactor vessels are given in Table 9, including the 1962 edition of ASME BPVC code estimates. The burst pressure is estimated using an analytic



**Fig. 6 – Finite element method model of reactor vessel of VVER-1000 and stress distribution.**

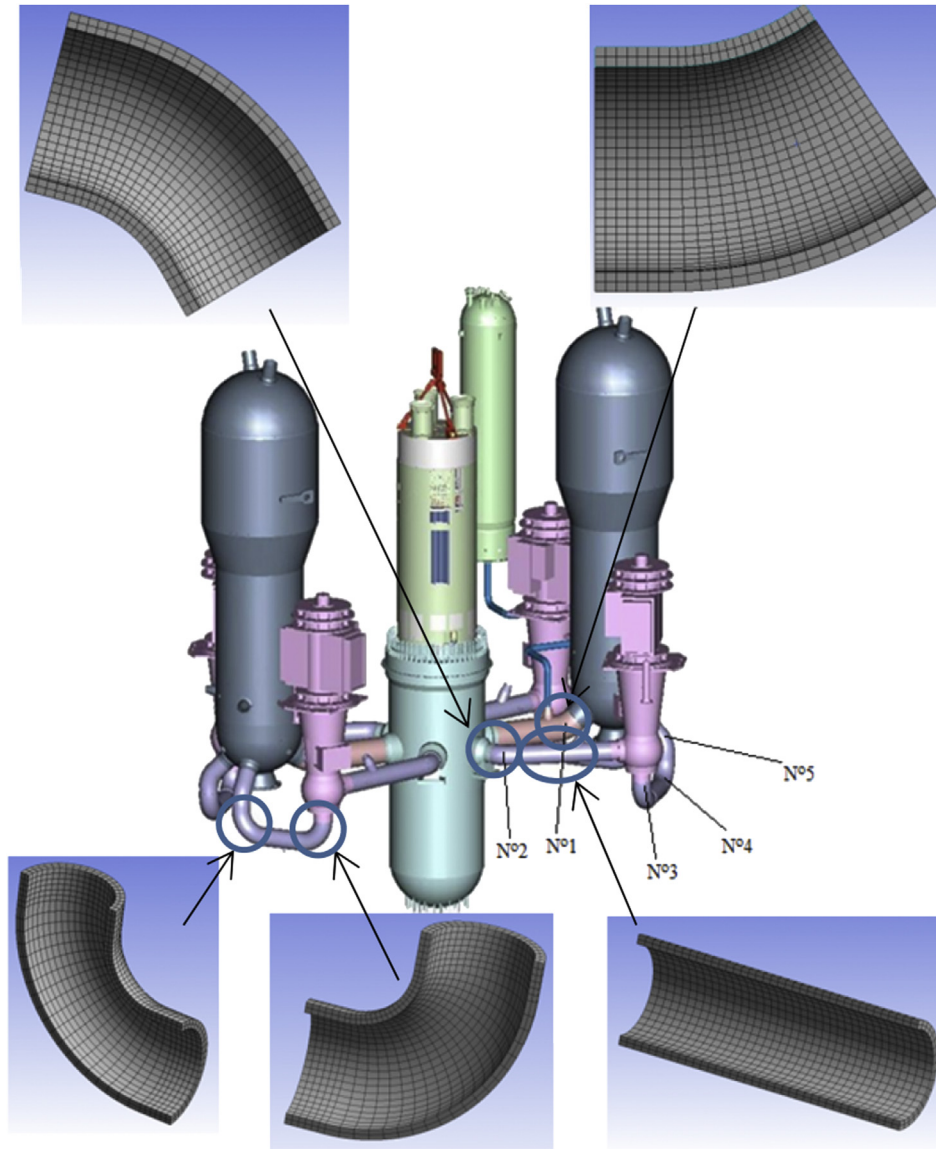


Fig. 7 – FEM model of APR-1400 RCS pipes.

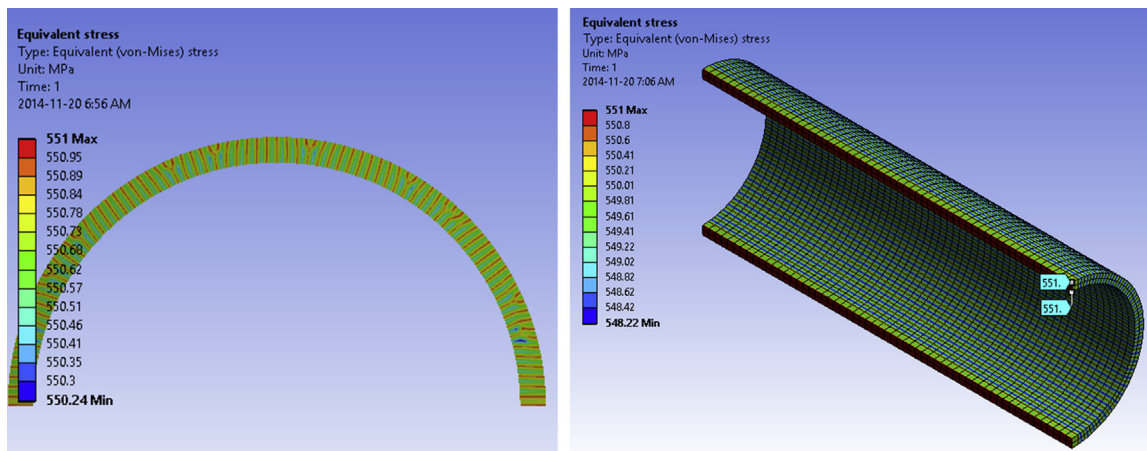


Fig. 8 – FEM model of AP-1000 steam generator U-tube and stress distributions.

**Table 8 – Yield pressure and burst pressure of reactor vessels of APR-1400, AP1000 and VVER-1000 by the finite element method.**

	Design pressure, $P_d$ , (MPa)	Yield pressure, $P_y$ , (MPa)	Burst pressure, $P_b$ , (MPa)	$P_y/P_d$	$P_b/P_d$
APR-1400	17.24	28.40	60.84	1.65	3.5
AP-1000	17.24	29.80	63.90	1.73	3.7
VVER-1000	19.70	32.25	56.98	1.64	2.9

formula that is smaller than that of FEM results; hence it gives somewhat more conservative solutions than FEM solutions. In particular, the Turner solution is the lowest value due to the perfect elastic and perfect plastic relationship it uses.

### 2.9. Burst pressure estimation of reactor coolant system main pipes

A reactor RCS pipe consists of a straight section and elbow section. The FEM analysis results are given in Tables 10–12. Comparing Table 8, the burst pressure of all types of reactor RCS pipes is higher than reactor vessels.

Table 13 shows the summary of the burst pressure of the RCS pipe straight section for APR-1400, AP1000, and VVER-1000 obtained from different formulations. The analytical

formulae was originally developed for the cylindrical section, hence it is applicable to straight sections of RCS pipes. Faupel's formula, Eq. (9), gives most conservative result while Soderberg's formula, Eq. (11), returns the highest value of burst pressure. This is attributed to the formulation as well as experimental materials they use in developing the formula. Overall the analytical solution returns more conservative burst pressure than the FEM solution. In particular, Faupel's solution gives the smallest value of burst pressure.

### 2.10. Burst pressure estimation of steam generator heat transfer tube

As shown in Fig. 8, a symmetry model was used to estimate the burst pressure of the SG heat transfer tube. Since one side

**Table 9 – Burst pressure results of reactor vessels by analytic formula.**

	Turner	Svensson	Faupel	Nadai–Bailey	Soderberg	ASME BPVC-1962
APR-1400	51.05	54.70	57.00	50.62	58.90	50.53
AP1000	53.99	57.85	60.29	53.51	62.29	53.41
VVER-1000	47.82	51.24	53.39	47.44	55.18	47.37

**Table 10 – Burst pressure estimation of APR-1400 RCS pipe.**

Location	Yield pressure at elbow (MPa)	Yield pressure at straight (MPa)	Burst pressure at elbow (MPa)	Burst pressure at straight (MPa)
Hot leg pipe	22.0	31.8	91.83	97.30
Cold leg pipe	22.6	33.0	92.00	101.5
Loop closure pipe	21.7	33.0	82.2	101.5

**Table 11 – Burst pressure estimation of AP1000 RCS pipe.**

Location	Yield pressure at elbow (MPa)	Yield pressure at straight (MPa)	Burst pressure at elbow (MPa)	Burst pressure at straight (MPa)
Hot leg pipe	20.5	25.8	79.80	87.70
Cold leg pipe	20.4	26.9	91.95	96.50

**Table 12 – Burst pressure estimation of VVER-1000 RCS pipe.**

Location	Yield pressure at elbow (MPa)	Yield pressure at straight (MPa)	Burst pressure at elbow (MPa)	Burst pressure at straight (MPa)
Hot leg & cold leg pipes	51.0	73.1	138.72	160.1
Loop closure pipe	51.0	73.1	136.5	160.1



**Table 13 – Burst pressure estimation of RCS pipe straight section by analytical formulae (MPa).**

	Turner	Svensson	Faupel	Nadai–Bailey	Soderberg	ASME BPVC-1962
APR-1400						
Hot leg	84.14	78.15	62.41	80.10	96.91	82.50
Cold leg	87.99	81.72	65.27	83.57	101.32	86.18
AP1000						
Hot leg	76.14	67.46	47.27	71.24	87.66	74.50
Cold leg	83.66	74.12	51.94	77.78	96.26	81.66
VVER-1000	139.52	140.28	135.59	132.81	160.03	134.77

RCS, reactor coolant system.

of the SG heat transfer tube is attached to a thick SG tube sheet, one end of the FEM model is assigned with fixed boundary condition and the other end is set to free (Fig. 8). In order to allow radial deformation, the symmetry surface is set to a frictionless boundary, hence it can expand freely. As for the loadings, the external pressure is set to zero for burst pressure calculation. Since bursting of the heat transfer tube is only possible, the secondary side pressure is assumed to be near zero pressure to make the loading condition more conservative. This condition represents conservative loading conditions during an accident. Table 14 shows the burst pressure obtained by the FEM analysis as well as according to analytical formula.

### 3. Results and discussion

Table 15 summarizes the burst pressure of a reactor pressure vessel, RCS pipe and SG heat transfer tube of APR-1400, AP1000 and VVER-1000, estimated by ANSYS. The burst pressure for RCS in elbow section of RCS pipe for all of APR-1400, AP1000, and VVER-1000 are smaller than straight section. The calculation shows clearly that the burst pressure of the reactor vessel is the smallest and the gap to the second smallest burst pressure is considerable. Therefore, it is not likely that other components will burst first. Because of this fact, for any accident that causes an internal pressure to increase, the reactor vessel will burst first. The burst pressure of the reactor vessel for VVER-1000 is lowest while that for other RCS pressure boundary components is highest for VVER-1000. This indicates there is more conservatism in the RCS pressure boundary component in VVER-1000 and there is room for improvement and optimization of design.

From the bursting failure calculation using a number of different theoretical formulae, the Soderberg's formula

**Table 15 – Summary of burst pressure of reactor coolant pressure boundary components using ANSYS (MPa).**

	APR-1400	AP1000	VVER-1000
Design pressure	17.24	17.24	19.7
Reactor vessel	60.84	63.9	56.98
RCS hot leg straight section	97.3	87.7	160.1
RCS hot leg elbow section	91.83	79.8	138.72
RCS cold leg straight section	101.5	96.5	160.1
RCS cold leg elbow section	92	91.95	136.5
Loop closure	82.2		136.5
SG heat transfer tube	84.1	85.5	106.55

RCS, reactor coolant system; SG, steam generator.

produces results that are closest to the FEM solution, but a slightly smaller value in all cases giving conservative results. Soderberg's solution gives a very good approximation of burst pressure. Also the reactor vessels have much smaller values of burst pressure.

Note also that in the case of VVER-1000 reactor coolant boundary components, the burst pressure for RCS pipes and SG heat transfer tubes is similar. Hence when the internal pressure exceeds burst pressure, either the RCS pipe or SG heat transfer tube will burst first depending on the higher local stress peak. In view of containing radioactive material within the RCS piping system, it is recommended to make the SG heat transfer tube thinner so that it bursts first. This is related to the safety management of reactor components and to the design optimization issues of RCS components.

In the case of APR-1400, the burst pressure of the loop-closure pipe is slightly lower than that of the SG heat transfer tubes, hence making it thicker will make the SG heat transfer tube burst first. Likewise, for AP1000, the burst pressure of the hot leg elbow is slightly lower than that of the SG heat transfer tubes, hence increasing the thickness of the hot

**Table 14 – Burst pressure estimation of SG heat transfer tube by finite element method and analytical formulae (MPa).**

	FEM	Turner	Svensson	Faupel	Nadai–Bailey	Soderberg	ASME BPVC-1962
APR-1400	84.1	67.16	62.11	49.02	64.84	77.46	66.27
AP-1000	85.5	68.17	63.04	49.76	65.78	78.62	67.26
VVER-1000	106.55	87.21	80.12	62.11	81.99	100.34	85.14

FEM, finite element method; SG, steam generator.

**Table 16 – Burst pressure of reactor pressure boundary components using Soderberg's formula (MPa).**

	APR-1400	AP-1000	VVER-1000
Reactor vessel	58.9	62.29	55.18
RCS hot leg	96.91	87.66	160.03
RCS cold leg	101.32	96.26	160.03
SG U-tube	77.46	78.62	100.34

leg elbow will make the SG heat transfer tube burst first. Table 16 shows summary of analytical results of burst pressures that are quite close to the FEM results. Analytical is a useful method that does not require FEM modeling, hence it can be used for obtaining initial approximation of burst pressure of a component.

In summary, the following was found and recommended from the calculation of the burst pressure and calculation of reactor pressure boundary components. (1) The burst pressure of the reactor pressure vessel is the smallest from RCS pressure boundary components. (2) The burst pressure of the VVER-1000 reactor is the smallest in comparison with APR-1400 and AP-1000. (3) The burst pressure of the VVER-1000 RCS components is highest in comparison with APR-1400 and AP-1000. (4) There is room for improvement in design optimization for VVER-1000. (5) The burst pressure of the APR-1400 closure pipe elbow and the AP-1000 hot leg elbow is lower than the SG heat transfer tube, hence increasing wall thickness is recommended for those components. (6) Soderberg's formula returns a burst pressure that is conservative and close to the FEM solution in comparison to other formulae, and it can be used to get approximate burst pressures without having to go through the FEM analysis or test. (7) It was shown that burst pressure can be used as a metric to determine the balance of system design. (8) It was also shown that the burst pressure estimation can be used for the purpose of design optimization.

### Conflicts of interest

The authors have no conflict of interest.

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