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INVESTIGATION OF APR1400 REACTOR VESSEL SUPPORT

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Abstract*—The APR1400 is a pressurized water nuclear reactor designed and developed by Korea. Since its introduction if 2007, the reactor has been used in a number of commercial applications. Although having been successfully deployed, the new reactor has been continuously modified to improve its performance and safety features. This paper aims to investigate a safety improvement for the reactor at the support system. Currently the APR1400 reactor is supported by 4 columns positioned at the cold leg nozzles. This system challenges at seismic events due to the height of the columns and also it inhibits ex-vessel coolant flow in the mitigation of severe accidents. This paper studies an alternative support system for the reactor vessel while eliminates the use of support columns. A plate-type support positioned at the upper flange of the reactor vessel is finalized. This plate-type support is designed with specific cross section. Conducted thermal and static structural analysis shows that the subject is adequate for an initial design concept. Design limits are defined by ASME code and the design follows ASME rules. From the outcome of this paper, further studies such as dynamic analysis are expected to improve the concept design quality. A successfully design of the plate-support is hoped to contribute to the perfection of the APR1400 Reactor.

***Index Terms*—Nuclear Reactor Support, ASME Code Stress Evaluation, Column Support, Plate Support.**

I. INTRODUCTION

The Advanced Power Reactor 1400 [MWe] – APR1400 is an advanced pressurized water reactor designed by the Korea Electric Power Corporation (KEPCO). At the moment, there are 12 APR1400 projects are being deployed, of which 8 projects are located in Korea (Shin-Kori and Shin-Hanul) and 4 in UAE (Barakah). Within the number of projects, SKN3 (Shin-Kori Unit 3) is planned to connect to the grid in 2016, and SKN4 and BNPP1 (Barakah NPP Unit 1) are planned to connect to the grid in 2017. The BNPP2 to 4 and SHN1 and 2 (Shin-Hanul NPP 1 and 2) are planned to successively connect to grid each year from 2018 and the rest are under construction or in planned stage.

Currently in the design of the APR1400, the reactor vessel is supported by 4 vertical support columns at the cold leg locations. This support system has been approved by the Nuclear Regulatory Commission (NRC) and has already been applied on the current projects. However, there is opportunity for improvement of the design since the column supports act as energy storage for the horizontal seismic motion and may increase dynamic load to the reactor core. In case of earthquake events, the tall columns transfer amplified movement to the reactor, resulting in higher instability to the reactor core. In another scenario, during severe accident events where coolant is filled into the reactor cavity for ex-vessel cooling, the columns act as hindrance to coolant flow since it occupies a large space of cavity and also inhibit evaporated steam flow.

In this paper, a plate-type support concept is developed in order to propose a possible alternative for the reactor support system. The proposed design change is analyzed using ANSYS V17.0. The work covers steady-state thermal analysis and static structural analysis for the design condition. From analysis result, primary membrane and bending stress intensity (P_m and P_b) are evaluated to verify with allowable design limits. Also the column support is reassessed in term of stress intensity and buckling load to show more insight of the Reactor Vessel support system.

II. SURVEY OF APR1400 REACTOR VESSEL CURRENT SUPPORT SYSTEM

A. *Overview*

APR1400 reactor vessel support system consists of four vertical columns installed under cold leg nozzles. The columns support the weight of the reactor vessel including reactor core and internals as well as coolant within reactor vertically. Each column has rectangular cross section of 762 x 279.4 mm (30 x 11 in) and height of around 5850 mm (230 in) (APR1400 DCD, 2014). The column system of support allows horizontal thermal expansion in events such as reactor operation, shut-down, normal operation, heat-up or cooldown, etc. As stated in the APR1400 Design Control Document Tier 2 Chapter 5, the supports are designed to "accommodate normal, seismic, IRWST discharge and BLPB loads". Fig. 1 illustrates the column support system.

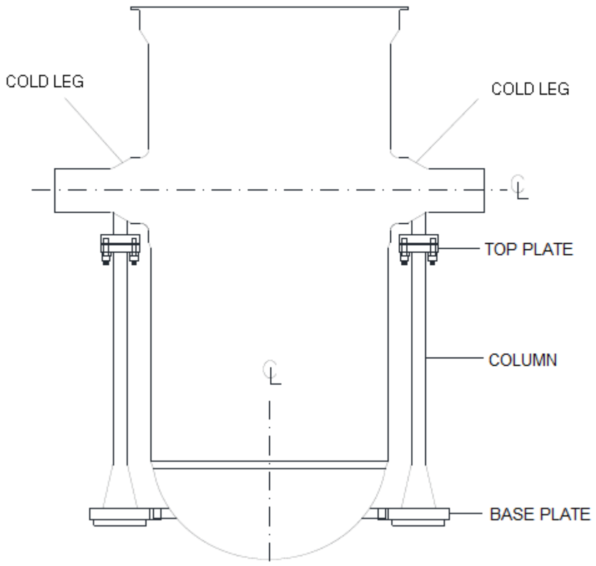

Fig 1: Column support system (APR1400 DCD, 2014)

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B. Reassessing stress in column support structure

Before looking into the new concept for the support system, it is necessary to understand the current system so that comparison can be made. In order to investigate the stress within the column support under loading condition, a 3D model is created using ANSYS. The model consists of one support column and a portion of the reactor vessel that it is connected to. The portion modeled in this paper os approximate 11% of the total volume of the reactor vessel. Figure 2 shows the model created for analysis. Thermal conditions are applied to the model to perform steady-state thermal analysis. The result of thermal analysis is combined with mechanical loads to perform static structural analysis. Loadings and conditions are presented in Table 1.

Table 1: Design loading condition

Loading/ Condition	Value	Unit
Reactor Internal temperature[1]	343.3	°C
Inside pressure	17.237	MPa
Reactor internals load[2]	16.23	MPa
Reactor closure head load[3]	0.1103	MPa

1. The reactor internal temperature was take from the hot-leg which is higher than cold-leg temperature for conservative estimation of temperature distribution
2. The reactor internal load includes reactor internals, reactor core and coolant weight that are applied to reactor vessel support surface. The table shows the converted value of the load into pressure.
3. The weight of reactor closure head is applied to the mating surface of reactor vessel. The value shown in the table represents the closure head weight converted to pressure.

Assumptions are made to simplify the model for the reduction of problem size. Perfectly insulation and frictionless support is applied on the symmetry cut planes in the thermal and structural analysis to ensure the continuity condition of temperature and stress distribution respectively. Insulation condition is assumed in other areas where insulations were actually placed.

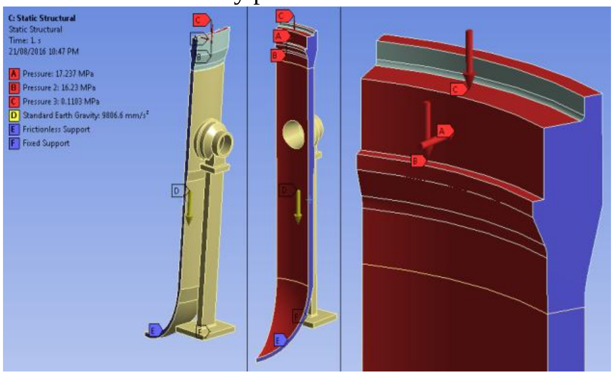

Fig 2: 3D analysis model of column support

By definition of the ASME code, column support is a linear-type support. The column support is categorized as Class 1 and shall conform to Article NF-3320 "Design by linear elastic analysis for Class 1". The column stress limit shall be considered for Stress in Compression case. The allowable compressive stress for column is calculated depending on the value of the largest effective slenderness ratio versus the slenderness ratio separating elastic and inelastic buckling.

The largest effective slenderness ratio is calculated by:

$$\frac{Kl}{r} = \frac{Kl}{\sqrt{I/A}} \quad (ASME\ NF-3320)$$

Where K is the effective length factor, equals to 1.2 in this case.
l is the column actual unbraced length of the member.
r is the governing radius of gyration

In this case,

$$\frac{Kl}{r} = \frac{1.2 \times 5878}{\sqrt{\frac{762 \times 279.4^3/12}{762 \times 279.4}}} = 81.4$$

The slenderness ratio separating elastic and inelastic buckling C_c is calculated by:

$$C_c = \sqrt{\frac{2\pi^2 E}{S_y}} \quad (ASME\ NF-3320)$$

Where E is the Young's modulus of material
S_y is yield strength at design temperature ($S_y = 3S_m$)

Then in this case,

$$C_c = \sqrt{\frac{2\pi^2 172000}{3 \times 191}} = 76.98$$

For Kl/r > C_c, the maximum allowable compressive stress is given by:

$$F_a = \frac{12\pi^2 E}{23(Kl/r)^2} \quad (ASME\ NF-3320)$$

The maximum allowable compressive stress is 133.7 MPa for the column support.

The failure theory used for linear-type support in ASME code is the maximum stress theory. In the maximum stress theory, the controlling stress is the maximum principle stress, as stated in ASME code. Stress paths were created along the height of the column model to investigate the maximum principle stress. The paths locations are shown in Fig. 3.

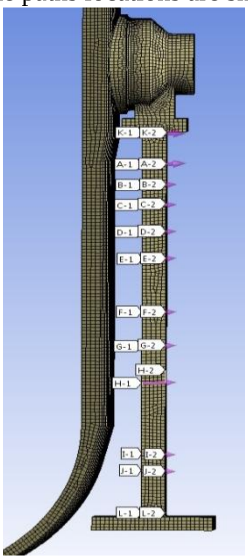

Fig 3: Location of paths on column support model

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The result of maximum principle stress distributed on the column is presented in Table 2.

Table 2: Maximum principle stress for paths

Path	Maximum principle stress (MPa)	Allowable stress (MPa)
1-A	4.98	133.7
2-B	1.17	133.7
3-C	0.25	133.7
4-D	0.31	133.7
5-E	0.48	133.7
6-F	0.02	133.7
7-G	0.03	133.7
8-H	0.04	133.7
9-I	0.04	133.7
10-J	13.9	133.7
11-K	59.6	133.7
12-L	127.21	133.7

Analyzing the results, the maximum principle stress peaks at the top and bottom part of the column however it still stay within the allowable limit. The principle stress value along the column is much lower than at the ends. Therefore the column supports is reproved to have adequate compressive strength.

C. Buckling analysis

When a column or plate is under compressive stress, it is often the case that the buckling load might be lower than yield load. Because of this phenomenon, the reactor vessel support column is also examined for the buckling load. In order to verify the buckling strength of the column, linear buckling theory is applied. Linear buckling theory usually tends to overestimate the buckling load compare to non-linear theory. However the linear buckling analysis is useful when the maximum principal stress is much lower than allowable stress as shown in Table 2, and it simplifies analysis time and effort. In this study, the buckling loads are evaluated and possible buckling modes are calculated.

The linear buckling theory predicts the minimum buckling load for a column as:

$$P_{buckling} = \frac{\pi^2 EI}{L^2}$$

Where E is Young's modulus of material (MPa), I is area moment of inertia (mm^4), and L is the length of the beam (mm).

The support column material is SA-508 Grade 2 Class 1. Design temperature is 343.3^{o}C. From ASME Section II Part D, the Young's modulus value is 172 GPa. Then the minimum predicted buckling load is:

$$P_{buckling} = \frac{\pi^2 \times 172{,}000 \times \frac{762 \times 279.4^3}{12}}{5{,}828^2}$$

$$= 80{,}490{,}307\ (N) \cong 80{,}500\ kN$$

While the vertical weight of reactor vessel and all internals per column is calculated to be ~6100 kN, the actual load is far below the buckling limit. The difference also compensates the overestimate problem of the linear theory. ANSYS analysis is used to verify the equation and investigate buckling modes. Fig. 4 shows buckling modes with associated occurring load. Detailed loads are shown in Table 3.

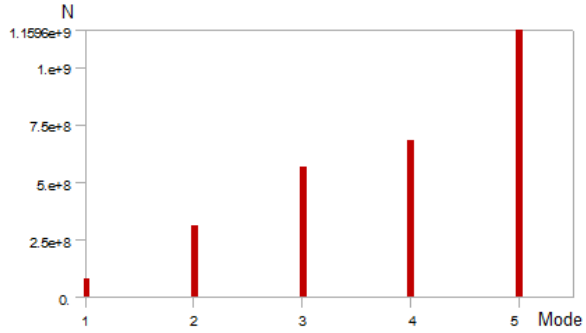

Fig 4: Buckling modes with associated loading

Table 3: Associated load analyzed for buckling mode

Mode	Load (N)
1	7.8675×10^7
2	3.0941×10^8
3	5.6446×10^8
4	6.7717×10^8
5	1.1596×10^9

For verification purpose of this paper, the first buckling mode with the lowest buckling load is the only concern. The Mode 1 is presented in Fig. 5 below.

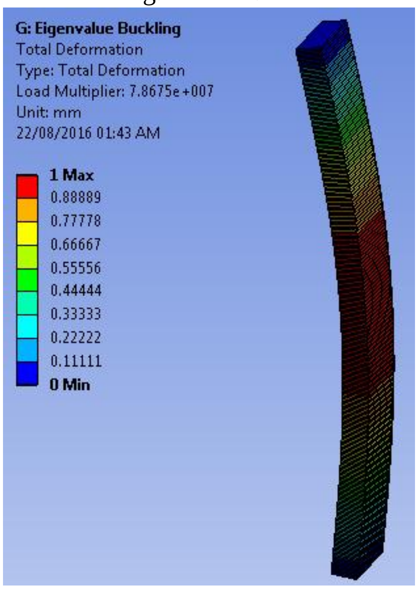

Fig 5: Buckling mode 1

III. COLUMN LESS CONCEPT

The column less support system investigated in this study is in form of a ring support. This design allows uninhibited cooling of reactor vessel in case of severe accident and prevent blockage of ex-vessel coolant flow thereby increasing the safety and probability of containing radioactive material within primary loop. The column less support shall support the reactor vessel from the top flange and must allow thermal expansion of the vessel during heat-up and cooldown. A number of concepts were developed to the mentioned requirements. Each concept possesses a set of unique parameters which defines the cross

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section geometry. By using parameter set function in ANSYS, the iterative process of design optimization was automated. A table of variable design parameters is created for each concept. From each table of parameters, the resulting equivalent stresses are evaluated following re-meshing and recalculating the stresses at specified locations. The optimum value of design parameter for each different design can be observed and found by this process. Upon find near optimum design parameters, a fine tuning of design parameter is done to find near optimum value of design parameters.

In the proposed design, support location is changed from cold-leg to reactor vessel flange to have larger cavity volume. This change may requires the relocation of steam generator support location and RC pump support location as well if this design change would be incorporated. And the reactor vessel is modified with thick flange to transfer load to the support. Fig. 6 presents axisymmetric modeling of the shell-type concepts that were investigated but were discarded due to the un-satisfactory result of stress level. Therefore shell-type support ideas are eliminated in earlier stage. A plate-type support was investigated and has shown decent and is described in this study.

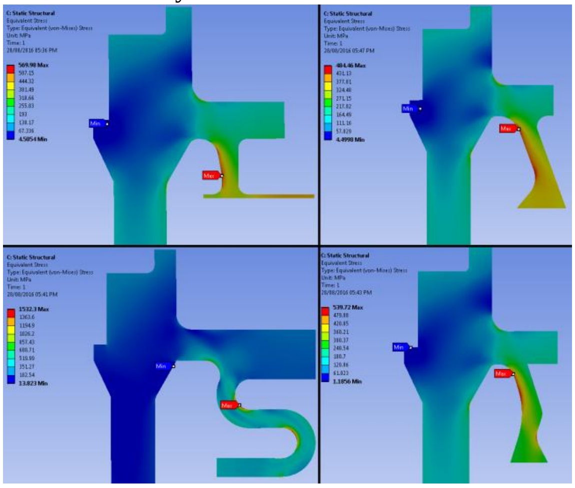

Fig 6: Shell-type support concepts

IV. DESIGN AND ANALYSIS OF THE PLATE-TYPE SUPPORT

A. Design limits for plate-type support

The design limits are defined in ASME Section III Article NF-3000 Design. The support is categorized as plate-type support and shall follow the “Design rules for plate- and shell-type supports” as stated in Article NF-3200. Since this is the reactor vessel support, the support class is defined as Class 1 and shall conform to Article NF-3220 “Design by analysis for Class 1”. In accordance with Article NF-3220, the stress intensity limits that must be satisfied for design loadings are:

a. *General primary membrane stress intensity P_m*
b. *Primary membrane plus primary bending stress intensity $P_m + P_b$*

The allowable value for requirement a. is S_m at design temperature and b. is $1.5S_m$ where S_m is material stress intensity. The material of the support is selected to be the same material with the reactor vessel, which is SA-508 Grade 3 Class 1. From ASME Section II Part D, the S_m value is found to be 191MPa.

Table 4: Allowable stress intensity limits of SA508 Grade 2 Class 1 (ASME, 2010)

Stress category	Allowable stress intensity (MPa)
P_m	191
$P_m + P_b$	286.5

The support base shall be anchored onto concrete base in accordance with Regulatory Guide 1.199 of U.S.NRC.

B. Design description

The plate support is in the form of a thick flat ring connected to the upper part of the reactor vessel.

Fig. 7 shows location of the support. The base plate is positioned under the support.

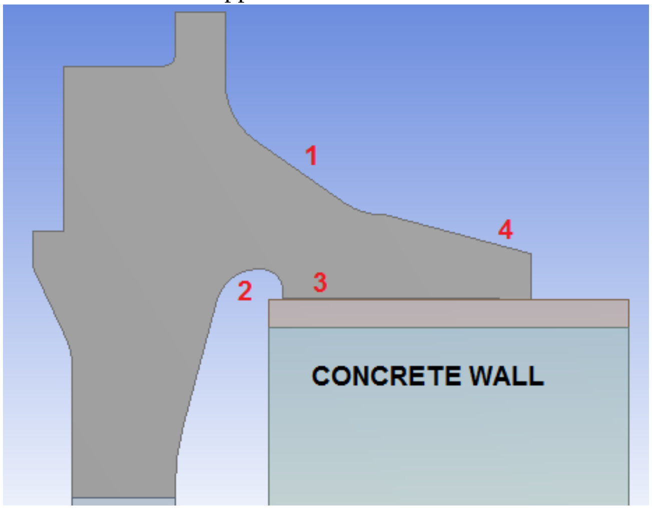

Fig 7: Reactor vessel flange cross section (upper part) with plate support

The design has the following key features:

- The contact surface between reactor vessel flange and support plate have enough surface to distribute weight load.
- Material is added at location 1 to increase stiffness of reactor vessel flange.
- Location 2 is rounded to reduce stress concentration
- The bottom surface (location 3) is slightly angled to shape uniform distribution of support pressure.
- At location 4, thickness is reduced to save unnecessary material.

C. Thermo-mechanical analysis

The plate support is desired to be placed on low friction surface to allow thermal expansion. For the purpose of analysis convenience, number of cases are analyzed that in one case the contact between reactor flange support and the base plate is frictionless and the other it is frictional. The base plate is fixed on concrete base and is assumed to stay un-deformed under load.

Design Loads are listed in Table 5 and analysis boundary conditions showing location of loads are given in Fig. 8 and 9 for thermal and static analysis, respectively.

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Table 5: Design loads

Load	Value	Unit
Thermal condition		
Design temperature	343.3	C
Convection	0.00001	W/mm^{2}C
Static structural		
Reactor internals weight*	16.23	MPa
Closure head weight*	0.1103	MPa
Design pressure	17.237	MPa

**Weight is divided by applied area to get pressure value.*

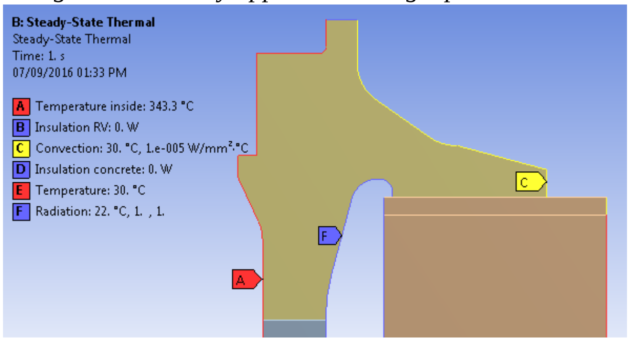

Fig 8: Steady-State Thermal condition

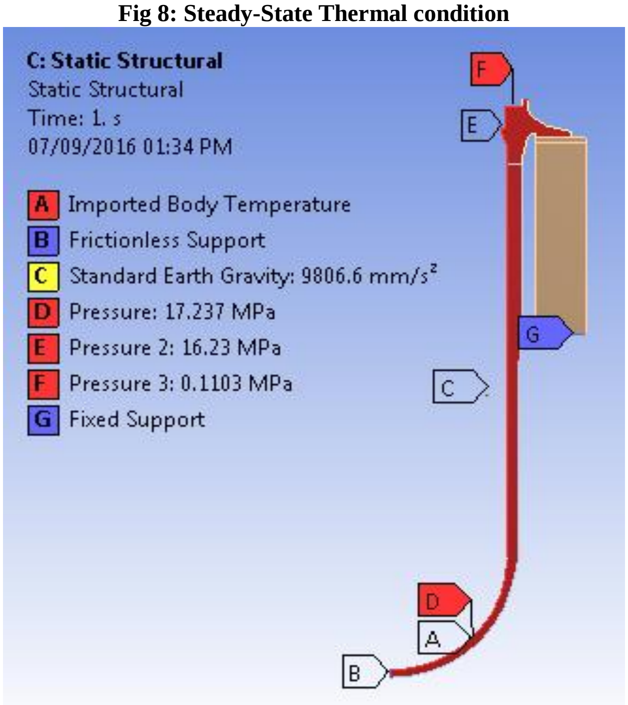

Fig 9: Static Structural condition

To investigate stress intensity within the structure, stress paths are created from the maximum equivalent stress location (See Fig. 10).

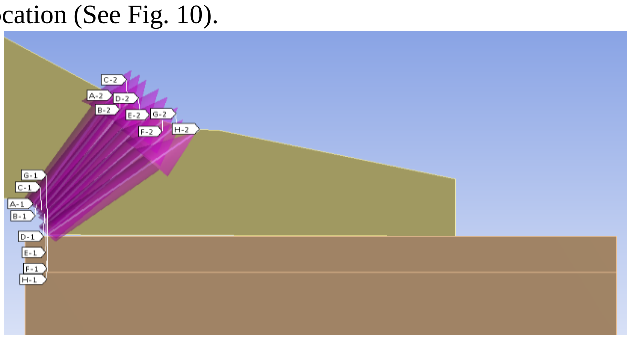

Fig 10: Linearized stress paths for primary membrane and bending stress intensity

The resulting primary membrane and primary bending stress intensities are summarized in Tables 6, 7 and 8 for frictionless, frictional with f=0.3 for dry friction simulation and fixed case for simulating bolted condition, respectively.

The frictionless and fixed support cases utilize the same model to simulate where the only difference is contact condition between the bottom surface of the support plate and the top surface of the base plate. The analysis procedure is Modelling – Meshing – Thermal analysis – Structural analysis. For frictional support case, to simulate friction, the procedure is modified as Modelling – Meshing – Structural analysis (mechanical load only) – Thermal analysis – Structural analysis (thermal stress included).

Table 6: Stress intensity analysis result for frictionless case

Stress path	P_m (MPa)	$P_m + P_b$ (MPa)	SA-508 S_m (MPa)	SA-508 $1.5S_m$(MPa)
1-A	155.8	186.23	191	286.5
2-B	159.17	190.80	191	286.5
3-C	164.77	211.85	191	286.5
4-D	170.06	226.01	191	286.5
5-E	174.39	238.22	191	286.5
6-F	178.24	248.23	191	286.5
7-G	183.37	259.60	191	286.5
8-H	185.03	257.65	191	286.5

Table 7: Stress intensity analysis result for frictional with f=0.3 case

Stress path	P_m (MPa)	$P_m + P_b$ (MPa)	SA-508 S_m (MPa)	SA-508 $1.5S_m$ (MPa)
1-A	106.82	122.10	191	286.5
2-B	101.39	127.05	191	286.5
3-C	98.62	128.71	191	286.5
4-D	90.82	127.50	191	286.5
5-E	86.20	124.21	191	286.5
6-F	82.36	138.84	191	286.5
7-G	79.58	150.27	191	286.5
8-H	82.27	156.03	191	286.5

Table 8: Stress intensity analysis result for bolted case

Stress path	P_m (MPa)	$P_m + P_b$ (MPa)	SA-508 S_m (MPa)	SA-508 $1.5S_m$ (MPa)
1-A	125.62	466.34	191	286.5
2-B	131.87	467.27	191	286.5
3-C	138.06	450.54	191	286.5
4-D	146.66	431.38	191	286.5
5-E	151.95	398.92	191	286.5
6-F	156.63	347.87	191	286.5
7-G	154.01	257.09	191	286.5
8-H	176.45	266.06	191	286.5

Fig. 11 shows temperature distribution and Fig. 12 present equivalent stress intensity (showing thermal stress) for frictionless case. Fig. 13 and 14 show the temperature distribution and thermal stress contour for frictional support case with friction coefficient of 0.3. Finally Fig 15 and 16 show the temperature distribution and thermal stress contour for fixed support case simulating bolted support case.

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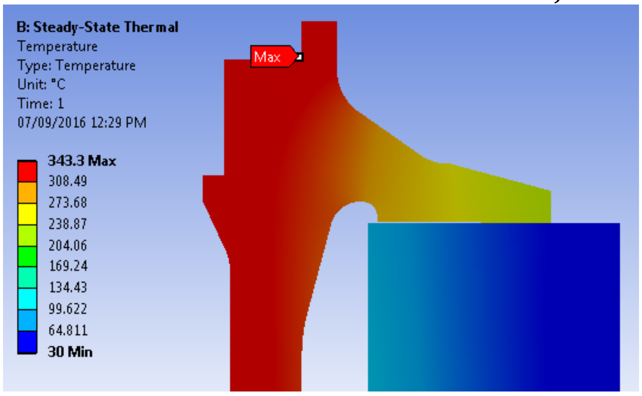

Fig 11: Temperature distribution from thermal analysis – frictionless case

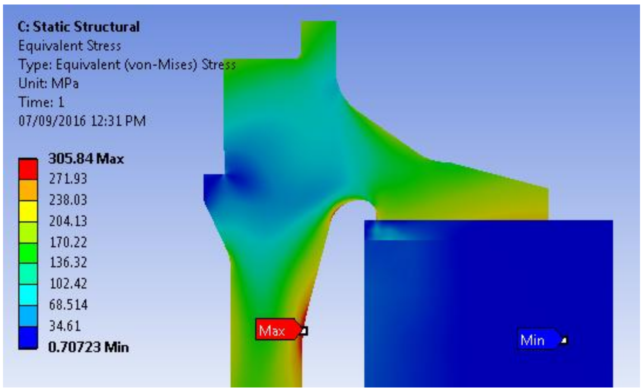

Fig 12: Stress distribution from static structural analysis – frictionless case

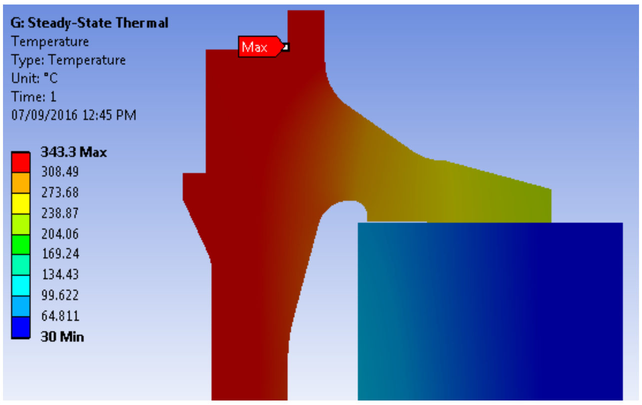

Fig 13: Temperature distribution from thermal analysis – frictional f=0.3 case

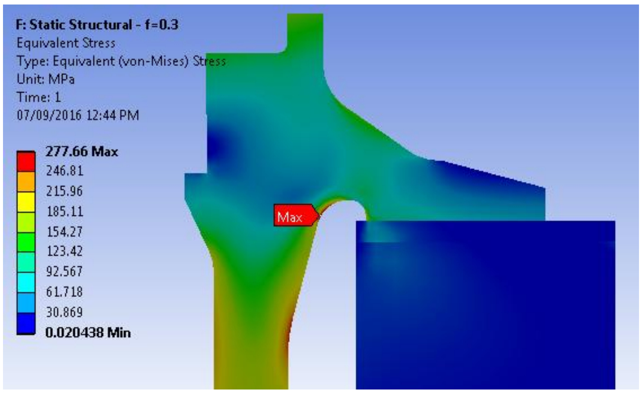

Fig 14: Stress distribution from static structural analysis – frictional f=0.3 case

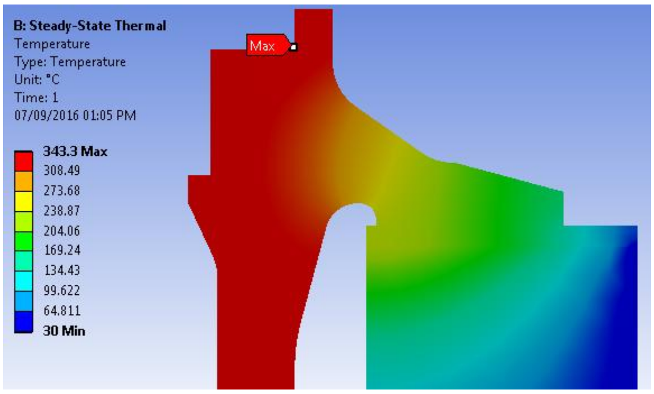

Fig 15: Temperature distribution from thermal analysis – bolted support case

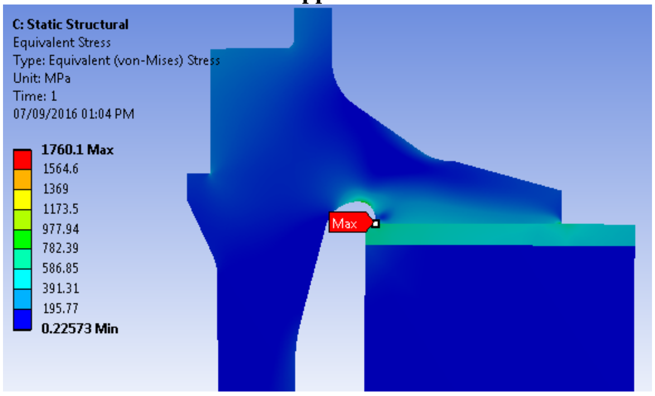

Fig 16: Stress distribution from static structural analysis – bolted support case

V. RESULT AND FURTHER CONSIDERATION

Stress condition in frictionless and frictional contact between the support plate and the base plate cases can satisfy design condition is examined and shows that both can satisfy design condition. Observing the resultant stress map shows that the maximum equivalent stress is located at the curved geometry between the plate support and the Reactor Vessel body. This location is where failure will most likely to occur. Allowable linearized stress limit shall be defined in accordance with ASME code. This stress value is investigated across the connection region between the support and the body.

The frictional case produces a more realistic model and creates more accurate result. The stress difference between frictional and frictionless cases can be explained by the frictional force on the bottom surface of the plate support acts as a reaction force that effect the resultant stress. In reality, this reaction force contains not only friction. The base plate under the weight load of the reactor vessel will plastically deform to some degree and will contribute to the resultant stress within the support.

The stress level in fixed support case is significant. The fixed support prevents thermal expansion movement therefore a vast amount of stress was observed. This indicates that the fixed support will not satisfy the design requirement and is not suitable in this case.

Table 9 presents the general analysis result for 3 above cases.

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Table 9: Analysis result of maximum linearized stress intensity

Stress category	Frictionless case	Frictional case f=0.3	Fixed case	Allowable value
P_m	185.03	106.82	176.45	191
$P_m + P_b$	257.65	156.03	466.34	286.5

The result presented in this paper can be useful as a first step of developing a column less support for reactor vessel.

Further enhancements that could be done on the design in this paper that can be considered:

- Secure system design to hold the Reactor Vessel in place.
- Further study on concrete behavior under transferred heat.
- Further optimization for the design.
- Economic feasibility study for the design.

For the development of this topic, there are a number of further considerations:

- Dynamic analysis such as operational vibration and verifying seismic response.
- Material related topics such as radiation effect, creep, fatigue… on the plate support.
- Method of installing the support on concrete wall around the reactor vessel.
- Further study on friction between plate support and the base plate; base plate local deformation.
- Ex-vessel cooling capacity improvement to improve current column design.

VI. CONCLUSION

In this paper, the column support system was reevaluated in term of stress and buckling analysis. The stress condition in column support shows that the current column support can satisfy the requirement stress of material. The buckling load is significantly lower than the first buckling load, which indicates that the column support can withstand the weight of the reactor vessel. Linear method was used to calculate and analyze the buckling load, although non-linear method shall be used for better accuracy, the analysis result showed much lower value, which is acceptable in this case.

A number of alternative designs for reactor vessel support has been studied and analyzed by FEA method using ANSYS Workbench. The shell –type support designs have shown not suitable for supporting the reactor vessel from the top of the vessel but plate type support shows high potential for further study.

A plate-type support has been developed with specially modified geometry. Thermal and static structural analysis was conducted for three cases including frictionless, frictional and fixed support. The analysis result was compared with ASME code limits to show that it follows the requirements. Friction condition was assessed and the result shows that stress on the support was lowered and transferred to other area within the reactor vessel body.

The paper has verified a highly potential replacement for the current support system at an early stage. Although the study is primitive at this point, further considerations are recommended to develop more detailed design. With such advanced studies, a more advantageous reactor support system is expected to be very possible.

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