♦특집♦ 가공과 진동

Analytical Structural Integrity for Welding Part at Piping Penetration under Seismic Loads

Heon-Oh Choi*, Hoon-Hyung Jung**, Chae-sil Kim[#]

(* C&K Eng., **, # Changwon National University)

지진하중이 적용되는 배관 관통부의 용접에 대한 구조 건전성 해석

> 최헌오^{*}, 정훈형^{**}, 김재실[#] (Received 3 February 2014; accepted 17 February 2014)

ABSTRACT

The purpose of this paper is to assess the structural integrity of piping penetrations for nuclear power plants. A piping qualification analysis describes loads due to deadweight, pressure difference acts normal to the plate, thermal transients, and earthquakes, among other events, on piping penetrations that have been modeled as an anchor. Amodel was analyzed using a commercial finite element program. Apiping penetration analysis model was constructed with an assembly of pipe, head fittings and sleeves. Normally, the design load, thus obtained, will consist of three moments and three forces, referred to a Cartesian coordinate system. When comparing the stress analysis results from each required cutting position, the general membrane stress intensities and local membrane plus bending stress intensities during a structural evaluation cannot exceed the allowable amount of stress for the design loads. Therefore, the piping penetration design satisfies the code requirements.

Key Words : Piping penetration(배관 관통부), Nuclear power plant(원자력 발전소), Seismic(지진)

1. Introduction

Currently, the likelihood of the occurrence of an

earthquake that exceeds the design basis of nuclear power plants in South Korea is very small. However, an accident due to leakage of radioactive matter can inflict catastrophic damage on the environment nearby. Therefore, more rigorous and precise seismic analysis in comparison to other industrial facilities was required. To resist large vibrations such as those that occur from earthquakes during normal operation and transient operation status, the piping penetration inside

^{*} C&K Eng.

[#] C. A. : Department of Mechanical Engineering, Changwon National University

E-mail : kimcs@changwon.ac.kr

^{**} Department of Mechanical Engineering, Graduate School, Changwon National University

structures of nuclear power plants need to conform to the guidelines of the design seismic qualifications[1]. This paper describes the thermal-hydraulic conditions experienced at various locations inside the safety injection system during design basis events. The results presented provide a conservative basis for the plant component design.

The nuclear steam supply system performance and safety related design bases events with the associated frequencies of occurrence are presented in Table 1. The frequency of occurrence is based on operating plant histories and engineering judgement and is intended for design purposes only. The values presented may exceed the actual expected number of operational occurrences. A steam line break, for example, is included as a design basis event but is not expected to occur in the life of the plant. A turbine power step change of about 10% is included as a weekly event although the actual frequency is expected to be significantly less than this value. The design frequency of occurrence reflects estimates of the yearly 40, monthly 500, weekly 2,000, daily 15,000 or three times per hour 1,000,000 operations over a 40-year plant life. Based on the frequency of occurrence the events are divided into normal, upset,

Table 1 Frequencies of occurrence

Categories	Thermal/Pressure mode description	Number of occurrences
Normal event	System shutdown conditions following plant normal/upset event	1100
Upset event	54	
Emergency event	System conditions following plant emergency event	20
Faulted event	System conditions following plant faulted event	30

emergency and faulted categories as defined in Table 1.

ASME Section III defines the relationship between the alternating stress and the allowable number of cycles for specific materials. The specified number of operational cycles divided by the allowable number of cycles is defined as the usage factor for a particular event. The sum of the usage factors for all normal, upset, emergency and faulted categories must be less than one over the design life of the component. The purpose of this paper is to assess the structural integrity for the piping penetration for the safety injection system for the nuclear power plant unit.

2. Structural analysis method

2.1 Modeling

The model was analyzed using the ANSYS finite element (FE) computer program [4]. The model was divided into elements as shown in Fig. 1 and Fig. 2. The element types used in this analysis were SOLID185 (3-dimensional 8-node structural solid) and MPC184 (multipoint constraint elements: rigid link/beam) elements. The SOLID185 elements were used in the pipe, head fitting and sleeve model. The MPC184 elements were used for applying pipe loads.

For the boundary conditions, supporting conditions of the piping penetration assembly were represented

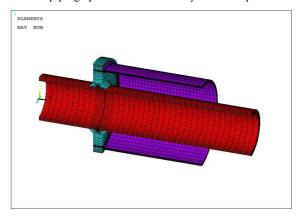


Fig. 1 FE modeling image of piping penetration

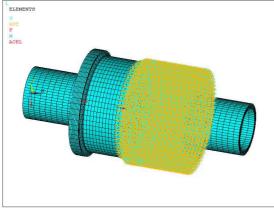


Table 2 Piping Loads

Loading conditions	Normal (A)	Upset (B)	Emergency (C)	Faulted (D)
Fx(kgf)	824	420	1612	1691
Fy(kgf)	350	291	555	600
Fz(kgf)	1737	332	3362	3407
Mx(kgf)	639	136	1287	1286
My(kgf)	1707	352	3431	3536
Mz(kgf)	624	359	1205	1233

Boundary Fig. 2 conditions image of piping penetration

by translational (UX, UY, UZ) and rotational (ROTX, ROTY, ROTZ) constraints for the boundary conditions in the numerical analysis.

2.2 Piping loads

The piping stress analysis describes loads due to thermal expansion, dead weight, thermal transients and earthquake, etc. on a seal plate that has been modeled as an anchor. If the line continues to the other side of the anchor and becomes a part of another piping system, the anchor loads from the second system must be combined with the first set of loads in the appropriate manner. This gives us resultant loads on the anchor due to thermal expansion of piping, deadweight and earthquake, etc. These loads, then, should be lumped together to give the worst possible combination of loads on the anchor. Normally, the design load, thus obtained, will consist of three moments and three forces referred to as a Cartesian coordinate system. The x-axis is from centerline of a component to the pipe, the y-axis is perpendicular to the x-axis in the vertical plane positive upward and the z-axis is perpendicular to the x-axis in the horizontal plane to form a right hand coordinate system. The piping loads are shown in Table 2. The piping loads in Table 2 are maximum absolute values

Table 3 System Criteria

Loading conditions	General membrane (PM)	Local membrane + Secondary(PL+PB)	
Nornal (A)	Sm	1.5Sm	
Upset (B)	1.1Sm	1.65Sm	
Emergency (C)	Larger of 1.2Sm or Sy	Larger of 1.8Sm or 1.5Sy	
Faulted (D)	Larger of 0.7Su or Sy+(Su-Sy)/3	Larger of 1.05Su or 1.5Sy+(Su-Sy)/2	

of each condition. However, if the designed thickness of the seal plate is abnormally high, or if the designer finds it difficult to meet the Code allowable stress values, the seismic loads can be considered separately. In that case, stresses due to seismic loads only satisfy the requirements of Subsection NE of ASME Section III.

2.3 Stress evaluations

The general membrane stress intensities (P_M) and the local membrane plus bending stress intensities $(P_{L}+P_{R})$ are linearized at the maximum stress location using the ANSYS postprocessor "POST1". The results are summarized below at the location shown in Fig.

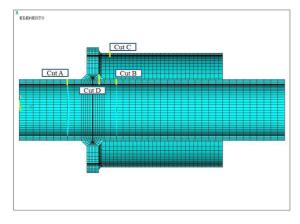
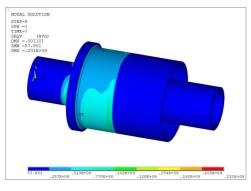
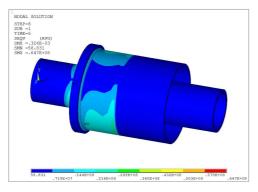


Fig. 3 Cutting locations of piping penetration



(a) Normal event



(c) Emergency event

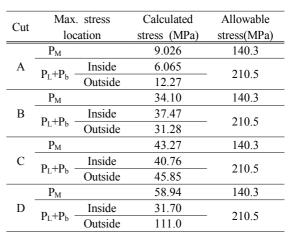
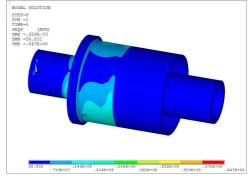
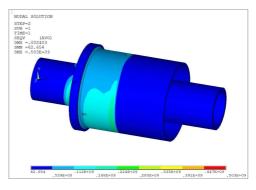


Table 4 Stress evaluation of normal event



(b) Upset event



(d) Faulted event

Fig. 4 Stress distributions

Max	. stress	Calculated	Allowable	
location		stress (MPa)	stress (MPa)	
P _M		5.003	140.3	
P _L +P _b	Inside	3.517	210.5	
	Outside	6.535		
P_M		9.676	140.3	
P _L +P _b -	Inside	9.148	210.5	
	Outside	10.64	210.3	
P_M		13.98	140.3	
P _L +P _b -	Inside	12.49	210.5	
	Outside	15.50	210.5	
$\mathbf{P}_{\mathbf{M}}$		14.81	140.3	
P _L +P _b -	Inside	18.87	210.5	
	Outside	25.58	210.5	
	$\frac{P_{M}}{P_{L}+P_{b}} - \frac{P_{M}}{P_{L}+P_{b}} - \frac{P_{M}}{P_{L}+P_{b}} - \frac{P_{M}}{P_{L}+P_{b}} - \frac{P_{M}}{P_{M}}$	$\begin{array}{c} P_{M} \\ \hline P_{L}+P_{b} & \hline Inside \\ \hline Outside \\ P_{M} \\ P_{L}+P_{b} & \hline Outside \\ \hline P_{M} \\ P_{L}+P_{b} & \hline Inside \\ \hline Outside \\ \hline P_{M} \\ \hline P_{L}+P_{b} & \hline Inside \\ \hline P_{M} \\ \hline P_{L}+P_{b} & \hline Inside \\ \hline \end{array}$	$ \begin{array}{c c c c c c c c c c c c c c c c c c c $	

Table 5 Stress evaluation of upset event

Table 6 Stress evaluation of emergency event

Cut	Max. stress		Calculated	Allowable	
	location		stress (MPa)	stress (MPa)	
	P _M		18.57	140.3	
Α	$P_{I}+P_{b}$	Inside	12.49	- 210.5	
	PL+Pb	Outside	25.30	210.3	
	P_M		71.73	140.3	
В	P _L +P _b -	Inside	79.18	210.5	
		Outside	65.35	210.5	
	P_M		89.92	140.3	
C	P _L +P _b -	Inside	84.73	210.5	
		Outside	95.25	210.5	
D	P _M		106.2	140.3	
	P_L+P_b -	Inside	71.68	210.5	
		Outside	197.5	210.5	

3. The stress distributions are presented in Fig. 4 through Fig. 7, and the stress evaluations are linearized at each cut as shown in Table 4 through Table 7.

2.4 Fatigue analysis

The stress concentration factor is applied to the linearized stresses at appropriate locations, that is cut A, B, C and D outside. The cumulative fatigue usage factor (U) is calculated at both sides of each cut as

Cut	Max. stress		Calculated	Allowable
Cut	location		stress (MPa)	stress (MPa)
	P _M		18.57	140.3
A	D I D	Inside	12.67	210.5
	P _L +P _b	Outside	25.32	210.5
В	P_M		72.97	140.3
	P _L +P _b -	Inside	80.60	210.5
		Outside	66.43	210.5
с _	P_M		90.41	140.3
	P _L +P _b	Inside	85.10	210.5
		Outside	95.87	210.5
D	P _M		108.7	140.3
	P _L +P _b -	Inside	73.43	210.5
		Outside	202.7	210.5

Table 7 Stress evaluation of faulted event

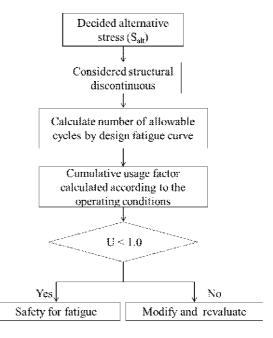


Fig. 5 Fatigue analysis procedures

required by ASME Code Sec. III. The fatigue analysis procedure is described by the following steps and is shown in Fig. 5. Step 1: Repeated the number of times of each form of stress cycles, n1, n2, ... nn. Step 2: The alternative stress intensity, Salt1,Salt2,...,

Cut	Node	Cumulative usage factor			
Cui	No.	Normal	Upset	Emergency	Faulted
	32295	0.00110	0.00225	0.00002	0.00003
A -	32087	0.00110	0.00225	0.00002	0.00004
В -	34658	0.00175	0.00225	0.00006	0.00009
	34450	0.00153	0.00225	0.00005	0.00007
C -	18784	0.00189	0.00225	0.00006	0.00009
	16832	0.00206	0.00225	0.00007	0.00010
D -	905	0.00205	0.00259	0.00007	0.00010
	34047	0.00405	0.00298	0.00013	0.00020

Table 8 Stress evaluation of faulted event

Saltn for each stress cycle form calculated as in the procedure above. Step 3: (N1,N2,...,Nn) the maximum allowable number of iterations for Salt1, Salt2, ..., Saltn using the appropriate design fatigue curves. Step 4: Usage factors,

$$\left(u_1=\left(\frac{n_1}{N_1}\right),u_2=\left(\frac{n_2}{N_2}\right),\ldots,u_n=\left(\frac{n_n}{N_n}\right)\right)$$

are obtained from each stress cycle form. Step 5: The cumulative usage factor U calculated with the following formula.

$$U = \sum_{S_{alt}} (n / N) < 1.0$$
 (1)

n : Design lifetime occurrence for Salt

N : Allowable occurrence

Salt : Alternative Stress Intensity (Salt=(1/2)aSp)

- Sp: Range of Peak Stress Intensity
- a : The ratio of the modulus of elasticity defined by NB-3219

The number of occurrences is shown in Table 1. The cumulative fatigue usage factors are shown in Table 8. A summary of the cumulative fatigue usage factors is given below:

3. Conclusion

The general membrane stress intensities and local membrane plus bending stress intensities at the structural evaluation do not exceed the allowable values for all conditions, and the cumulative usage factors do not exceed unity 1.0. Therefore, all structural and fatigue requirements were satisfied.

Acknowledgement

Special thanks go to C&K Eng. for lots of corporation in this paper.

References

- (1) Nuclear Power Engineering Committee of the IEEE Power Engineering Society, Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations, IEEE Std. 323, 1983.
- (2) Nuclear Power Engineering Committee of the IEEE Power Engineering Society, IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations, IEEE Std. 323, 1987.
- (3) ASME NQA-1, Quality Assurance Program Requirements for Nuclear Facilities.
- (4) Andrew D. Dimarogonas, Vibration for Engineers, Prentis Hall, 127-137, 1996.
- (5) A. C Hang and G. O Sankey, "The Containment of Disc Burst Fragments by Cylindrical Shells", Journal of Engineering for Power. 114-309, April 1974.
- (6) E. Oberg and F. D. Jones, Machinery's Handbook, 28th Edition, 224, 2009.
- (7) ASME OM Code, Code for Operation and Maintenance of Nuclear Power Plants, 2007 Edition with 2009 Addenda.
- (8) Korea Electric Power Industry Code, KEPIC-MN, Rules for Construction of Nuclear Power Plant

Component, ASME Boiler & Pressure Vessel Code Section III, Division 1. Subsection NB/MC, 2010 Edition with 2007 through 2009 Addenda.

⁽⁹⁾ ANSYS version 12.1