High-temperature Mechanical Analysis of Discharge Pipeline of Pressurizer

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Abstract. The severe accident is the beyond basis design accident due to several initial faults. The discharge pipeline of pressurizer is mainly used for pressure relief and evacuation of hydrogen in the primary circuit in severe accident. It can reduce the possibility of high-pressure core melt and the explosion of hydrogen. But in severe accident, the highest temperature of gas in the pipeline can reach 1200℃, which far exceeds the design temperature of pressurized water reactor. So it may cause the buckling or rupture of the pipeline. The pipeline may loss its discharge function. In this passage, we study the resistance of the discharge pipeline to high temperature in severe accident from the aspects of severe accident situation, high-temperature mechanical properties of pipeline material, the temperature field and stress of the pipeline, the evaluation of the stress and so on. In order to demonstrate the function of a certain nuclear power plant discharge pipeline, RCC-MR which is suitable for the high-temperature evaluation, is choose to evaluate the stress, the creep and the buckling of the pipeline. The results of the evaluation show that the discharge pipeline in this nuclear power plant can meet the functional requirements in the severe accidents.

Keywords: Supercharger, piping, mechanical analysis.

1. General

Pressurizer discharge pipeline is mainly used for pressure relief and hydrogen discharge in the primary circuit in severe accidents. It can safely convert the Guaranty core from high-pressure melt pile to low-pressure melt pile, and prevent the high-radioactivity water vapor and explosive hydrogen in the core from being discharged into the containment on a large scale [1]. In the severe accident of high temperature melting reactor in nuclear power plant, the steam temperature in Pressurizer discharge pipeline can reach 1200 ℃ at the highest. Such high temperature will affect the mechanical capability of the pipeline to a great extent, and may cause the discharge pipeline to bend, lose stability or break, and lose its discharge function.

This paper mainly studies whether the Pressurizer discharge pipeline of a nuclear power plant can safely relieve pressure without serious functional loss in the period before the end of hydrogen production in the core in a severe accident. In this paper, FLUENT is used to analyze the temperature field distribution of the pipeline under severe accident, then PIPESTRESS is used to calculate the Bends stress of the pipeline, and finally RCC-MR is used to evaluate the mechanics of the discharge pipeline to evaluate whether the pipeline meets the specification requirements.

2. Severe accident scenario analysis

Severe accidents are Beyond Design Basis Accident caused by superposition of multiple initial events [2]. The change of steam temperature and flow rate at the entrance of Pressurizer discharge line with time under the most severe severe accident conditions of its nuclear power plant is shown in Figure 1 and Table 1.

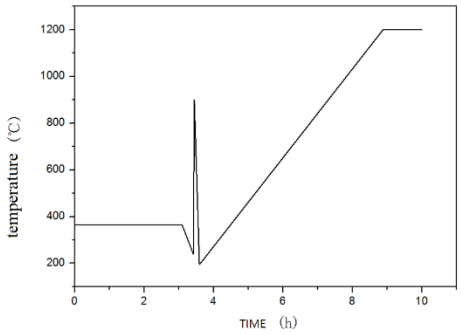


Figure 1. Fluid temperature.

Table 1. Fluid Flow.

time (h)	3.5	4	5	7.5	7.8	10
Fluid Flow (kg/s)	130	5	10	20	0.05	0.05

3. Structure and functional requirements of discharge pipeline

Pressurizer discharge system of a nuclear power plant (Figure 2) is mainly composed of three parts: three rows of Safety valve groups, two rows of relief valve groups, relief tank and corresponding discharge pipelines. In a severe accident, when the Reactor vessel is overpressurized, the Pressurizer Safety valve will open, and the fluid in the primary circuit will be discharged into the pressure relief tank through the Pressurizer and the discharge pipeline to maintain the core pressure at a reasonable level; When the temperature of the primary circuit is too high (more than 650 ℃) and the Safety valve fails, the pressure relief valve is opened in severe accidents, and a large amount of fluid enters the pressure relief tank through the discharge pipeline to achieve the purpose of rapid pressure reduction.

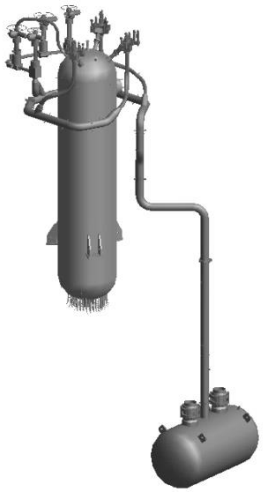


Figure 2. Pressurizer Discharge System.

Pressurizer discharge pipeline is nuclear level 1 pipeline. The material of the discharge line in front of the relief valve is 316LN Stainless Steel, and the material of the line behind the valve is 304L Stainless Steel. The mechanical properties of 316LN and 304L Stainless Steel are similar, and their strength decreases with the increase of temperature [3] [4]. At high temperature, they all have good ductility and creep after a certain time.

4. Pipeline heat transfer analysis

According to the steam temperature change curve at the inlet of the discharge pipeline, the finite element model of the pipeline is established, and the heat transfer analysis is carried out by using SIMPLE steady-state analysis method of FLUENT, and the temperature change curve of the pipe wall is obtained for mechanical evaluation.

According to the change of steam temperature and flow rate with time in Figure 1 and Table 1, the temperature field distribution of pipeline under the following two scenarios is analyzed:

Scenario 1: 7.5 hour after the accident, the inlet steam temperature is 950 ℃, the flow rate is 20kg/s, and the pressure is 2 MPa.

Scenario 2: 9 hour after the accident, the inlet steam temperature is 1200 ℃, the flow rate is 0.05 kg/s, and the pressure is 2MPa.

In this paper, the discharge pipeline is simplified as a straight pipe in Fig. 3 for heat transfer analysis, and the grids at the edges of the inner and outer walls of the pipeline are encrypted to make the heat transfer analysis of the wall more accurate, as shown in Fig. 4.



Figure 3. Finite element mesh model of discharge pipeline.

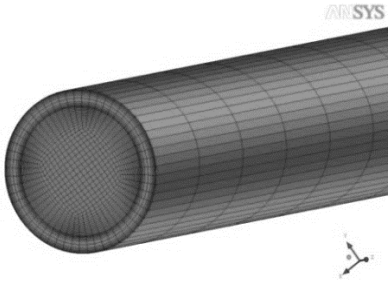
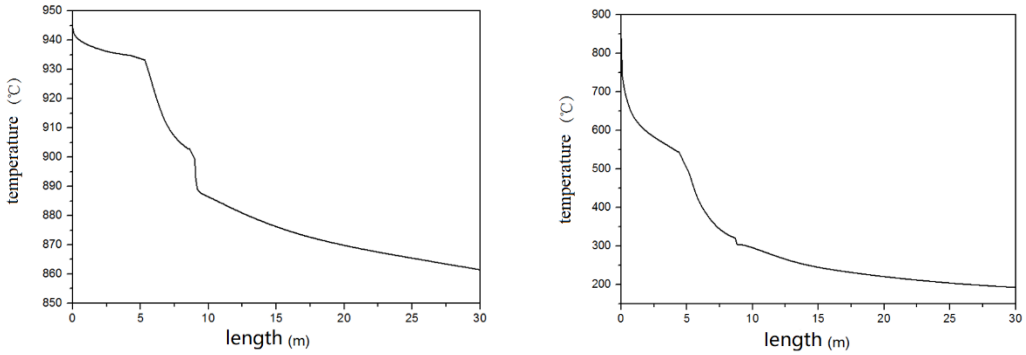


Figure 4. Finite element mesh at the inlet of discharge pipeline.

Fluid-structure coupling heat transfer coefficient is adopted in the inner wall of the pipeline, and convection heat transfer and radiation heat transfer should be considered comprehensively in the heat transfer coefficient of the outer wall of the pipeline [5]. Among them, the equivalent heat transfer coefficient [6] of the outer wall of the pipeline is:

$$\mathrm{h} = h_1 + h_2 = \frac{\lambda \cdot \mathrm{C}(GrPr)^n}{l} + \varepsilon \cdot \sigma\left(\frac{T_w^4 - T_f^4}{T_w - T_f}\right) \quad (1)$$

h_1 is the convective heat transfer coefficient in large space, h_2 is the radiation heat transfer coefficient,T_w is the tube wall temperature and T_f is the fluid temperature.



Scenario 1 Scenario 2

Figure 5. Temperature distribution on the inner wall of pipeline.

After calculation, the variation curve of pipeline inner wall temperature along pipeline axis in Scenario 1 and Scenario 2 is shown in Figure 5. After 7.8 h, because the steam flow rate in the discharge pipeline is basically zero, even if the inlet steam temperature is close to 1200 ℃, the pipe

wall temperature is below 800 ℃, which is lower than the pipe wall temperature in scenario 1. Therefore, the pipe wall temperature reaches the maximum in scenario 1, and after that, the pipe wall temperature gradually drops below 800 ℃ due to the sharp decrease of steam flow rate.

Therefore, according to the conservative principle, the pipe wall temperature in scenario 1, namely 950 ℃, is taken for stress calculation and evaluation.

5. Evaluation rules of RCC-MR

RCC-MR is a nuclear power Technical Specifications designed and built for high temperature nuclear facilities components formulated and published by French AFCEN. Volume B of RCC-MR is mainly suitable for the design and evaluation of high temperature nuclear grade 1 components [7].

When the temperature of the pipe is above the negligible creep curve, it indicates that the creep is significant, and the evaluation method of significant creep must be used.

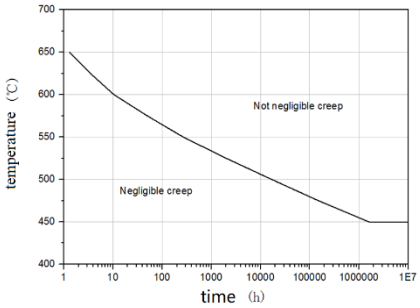


Figure 6. Negligible creep curve of pipe material.

The formula for evaluating significant creep caused by constant or uniformly increased load is as follows,

$$\overline{P_m} \le Min[2S_m; 0.7(R_m)_{min}] \quad (2)$$

$$\overline{P_m + P_b} \le Min[3S_m; 0.9(R_m)_{min}] \quad (3)$$

$$\overline{S_m + S_b} \le 1.9S^* \quad (4)$$

$$S^* = Min\left[0.8 \cdot \left(R^t_{p0.2}\right)_{min}; 0.4 \cdot (R_m)_{min}\right] \quad (5)$$

Among them, $\overline{P_m}$ is the film stress caused by pressure, $\overline{P_b}$ is Bends stress, S_m is the allowable stress strength of the material, R_m is the tensile strength of the material and $R^t_{p0.2}$ is the yield strength of the material.

$$W_{A,C,D}[1.35 \cdot \overline{P_m}] \le 1 \quad (6)$$

$$W_{A,C,D}[1.35 \cdot (\overline{P_m + \phi P_b})] \le 1 \quad (7)$$

W is the creep service coefficient, which is mainly used to consider the influence of time on the capability of materials. Its expression is:

$$\mathrm{W} = \sum_k (\frac{t_k}{T_k}) \quad (8)$$

Among them, T_k is the maximum allowable time under the characteristic stress, which is found in Table 2 according to the material type, temperature and the size of the characteristic stress. t_k is the time corresponding to the k-th time period.

Table 2. Maximum permissible time of 316LN Stainless Steel at high temperature.

Temperature (℃)	1h	10h	30h	100h
450	443	443	443.1	443.1
550	401	386	379.7	372.5
600	351	316	300.3	284
650	298	250	229.3	209
700	257	213	195.2	164.6
750	230	164	138.4	114.6
800	169	116	96.3	78.2
850	122	81	65.8	52.3

900	86	55	44.1	34.3
950	60	37	29	22

6. Carry out mechanical evaluation of Pressurizer discharge pipeline

The temperature of the discharge pipeline in Pressurizer of a nuclear power plant in a severe accident is above the negligible creep curve, so the mechanical evaluation of the discharge pipeline in Pressurizer is carried out according to the remarkable creep evaluation formula of RCC-MR. The evaluation results of the largest pipeline in stress are shown in Tables 3 ~ 5.

Table 3. Evaluation results of film stress.

Pipeline No.	Material	P_m (MPa) Straight pipe	P_m(MPa) Elbow	Permissible stress (MPa)
3RCP7115/7117TY	304LStainless Steel	19.27	23.94	37.2

Table 4. Evaluation results of Film stress + Bends stress.

Pipeline No. (Material)	Straight pipe (MPa)			Elbow (MPa)			Permissible stress (MPa)
	P_m	P_b	$P_m + P_b$	P_m	P_b	$P_m + P_b$	
3RCP6310/6320TY (316LN)	3.22	36.13	39.35	3.22	12.27	15.49	71.9

Table 5. Evaluation results of Sm + Sb.

Pipeline No. (Material)	Straight pipe (MPa)			Elbow (MPa)			Permissible stress (MPa)
	Sm	Sb	Sm + Sb	Sm	Sb	Sm + Sb	
3RCP6310/6320TY (316LN)	0	36.13	36.13	0	12.27	12.27	60.72

According to Table 4, $(1.35 \cdot (\overline{P_m + \phi P_b}))_{max}$=53.13Mpa, Therefore, the maximum allowable time corresponding to each temperature range in Table 6 can be obtained.

$$W = \sum_k (\frac{t_k}{T_k}) = 0.003 + 0.03 + 0.3 = 0.333 < 1$$

Table 6. Calculation results of creep service coefficient.

Temperature (℃)	450-500	500-550	550-600	650-700	700-750	750-800	800-850	850-900	900-950
t_k(h)	0.3	0.3	0.3	0.3	0.3	0.3	0.3	0.3	0.3
T_k(h)	>100	>100	>100	>100	>100	>100	100	10	1
W_k (h)	~0	~0	~0	~0	~0	~0	0.003	0.03	0.3

7. Conclusion

The heat transfer analysis of a Pressurizer exhaust pipeline in a nuclear power plant is carried out. The results show that the highest wall temperature in severe accidents is 945 ℃, which is far lower than the gas temperature at the inlet of the pipeline.

RCC-MR is used to evaluate the mechanics of the pipeline under severe accident. The evaluation results show that the stress and creep coefficient of the pipeline are less than the allowable values in the code.

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