ELEVATED TEMPERATURE DESIGN AND INTEGRITY EVALUATION OF A LARGE-SCALE SODIUM TEST FACILITY, STELLA-2

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ABSTRACT

A large-scale sodium test facility of STELLA-2 is to be constructed by 2019 for integral effect tests, and a detailed design of the test facilities is underway at the Korea Atomic Energy Research Institute (KAERI). Elevated temperature design (ETD) evaluations were conducted for the model reactor assembly of 316L stainless steel and intermediate heat exchanger (IHX) of Gr.91 steel according to the ETD codes of RCC-MRx and ASME Subsection NH. The integrity of the sodium test facilities was confirmed based on the design evaluations according to the design codes, and conservatism in the design codes was compared.

INTRODUCTION

The objectives of the STELLA-2 sodium test facility are integral effect tests for verification of the dynamic plant response associated with a synthetic review of the key safety issues and safety analysis code validation with the obtained test database. The reference plant of STELLA-2 is a PGSFR (Prototype Generation IV Sodium-cooled Fast Reactor) (Lee, 2017, IAEA, 2012). A detailed design of STELLA-2 is underway at KAERI (1). Securing the structural integrity for the intended severe design transients is very important because the planned tests in the sodium test facility can be successfully conducted only in a mechanically safe system.

The sodium test facility will be subjected to high thermal loads with a core outlet temperature of 550°C at atmospheric pressure. The main components subjected to high temperature are the model reactor internals of 316L stainless steel and an intermediate heat exchanger (IHX) of Mod.9Cr-1Mo (ASME Grade 91 steel). An elevated temperature design (ETD) of the STELLA-2 system was conducted as per RCC-MRx rather than ASME Section III Subsection NH (hereafter ASME-NH) because high-temperature material properties are available for 316L stainless steel in RCC-MRx, and not available in ASME-NH (Lee, 2016). 316L stainless steel is a material used in high-temperature test facilities because it is a readily available and economical heat-resistant material.

For ETD evaluations of the high-temperature components, a web-based design evaluation program of the HITEP_RCC-MRx (Lee et al., 2017) was developed for a design-by-analysis for ETD evaluations according to RCC-MRx. The material properties of RCC-MRx are embedded in the program, and a third-party program verification was conducted.

DESIGN EVALUATION OF THE STELLA-2 COMPONENTS

The overall design features of the STELLA-2 sodium test facility are shown in Fig. 1, which shows that it is practically a scaled-down sodium reactor system with a primary and secondary heat transport system. The main components of STELLA-2 are the model reactor vessel, reactor internals, reactor head and support system, five heat exchangers (IHX, DHX, AHX, FHX and UHX), tanks (reservoir, expansion tank

and sodium storage tank), and piping systems. Design evaluations of the piping system in STELLA-2 were conducted in a previous study by Son et al. (Son, 2016).

The thermal design transients for the model RV and RI are shown in Fig. 2, which has been selected in a conservative manner, and the transients for IHX are the same with those of the model RI.

Figure 1. Schematics of the STELLA-2 test facility.

Figure 2. Design transients of model RV, RI and IHX

A 3D solid model of the model reactor assembly with a number of main components supported on the reactor head is shown in Fig. 3(a). The 3D model of IHX is shown in Fig. 4(a). Heat transfer analyses and thermal analyses for both components were conducted. The profiles of the stress intensities for the reactor assembly and IHX are shown in Fig. 3(b) and Fig. 4(b), respectively. The critical locations of the redan corner in Fig. 3(c) and secondary sodium nozzle part of Fig. 4(b) subjected to the high temperature of the creep regime were selected, and elevated temperature design evaluations were conducted according to RCC-MRx.

Design evaluations for the redan corner point were conducted according to RCC-MRx and ASME-NH. The material of redan is 316L stainless steel. The material properties of 316SS were used in ASME-NH because 316L properties are not available in ASME-NH unlike RCC-MRx. The evaluation results for creep-fatigue damage show that both creep damage and fatigue damage were negligibly small, as shown in

Fig. 5. Although the materials for redan are different in the two codes, the evaluation results show that the evaluation results of RCC-MRx are less conservative than those of ASME-NH.

The design evaluation results of IHX are shown in Table 1 for the secondary outlet nozzle part of IHX. As shown in Table 1, all design limits of the load-controlled stress, inelastic strain, and creep-fatigue were satisfied. The material of IHX is Gr.91 steel, and it is well known that the design evaluations of ASME-NH are overly conservative, especially for a creep-fatigue damage evaluation in Gr.91 steel (Lee et al., 2012). As shown in Table 1, creep damage and fatigue damage were negligibly small under the intended design transients of Fig. 2.

Figure 3. Schematic of model RV & RI, profiles of stress intensities under thermal loads

Figure 4. Schematic of model RV & RI, profiles of stress intensities under thermal loads

Figure 4. Evaluation results of creep-fatigue damage for redan corner

Table 1: Design evaluation results of IHX nozzle part according to RCC-MRx.

HITEP_RCC-DBA				
Material	Gr.91			
Service Level	Level A			
Max. Temp	530			
Total hold Time	15000			
Hold Temperature	530			
Evaluation Items		Calculated	Limit value	Remark
Load-Controlled Limits	$P_m < S_m$	13.679	134	OK!!
	$P_t < 1.5S_m$	13.679	201	OK!!
	$P_L + P_b < 1.5S_m$	14.458	201	OK!!
	$U(\Omega P_m)$	3.9927E-7	1	OK!!
	$U(P_m + P_b)$	5.4922E-7	1	OK!!
	$W(1.35P_m)$	7.1116E-7	1	OK!!
	$W[1.35(P_m + \Phi P_b)]$	9.7487E-7	1	OK!!
	J-integral	350	No data (A3)	-
Inelastic Strain Limits	$P_1 < 1.3 S_m$	0.0472	174.2	OK!!
	$P_2 < 1.3 \times 1.5 S_m$	0.0485	261.3	OK!!
	$\text{Max}(P_m + P_b) + \text{Max } q(j,j') < 3 S_m$	162.96	402	OK!!
	$\epsilon_{\text{plastic}} + \epsilon_{\text{creep}} (1.25 P_1) < 1 \%$	7.9144E-15	0.5	OK!!
	$\epsilon_{\text{plastic}} + \epsilon_{\text{creep}} (1.25 P_3) < 2 \%$	8.6905E-15	1	OK!!
Creep-Fatigue Damage	Fatigue Damage	Negligible	5.7888E-6	OK!!
	Creep Damage	0.0329	0.9999	OK!!

CONCLUSION

Elevated temperature design (ETD) evaluations for a model reactor assembly and IHX in a large-scale sodium test facility of STELLA-2 were conducted. The integrity of the main components was confirmed in terms of the load-controlled stress limits, inelastic strain limits, and creep-fatigue damage based on the design evaluations mainly according to RCC-MRx because high-temperature material properties of 316L stainless steel are available in RCC-MRx but not in ASME-NH. A design evaluation program of the

HITEP_RCC-MRx was developed for reliable and efficient computations on the design limit checking according to RCC-MRx.

ACKNOWLEDGEMENTS

This work was supported by an International Research & Development Program Foundation NRF grant (2013K1A3A7A03078195) and by a NRF grant (2012M2A8A2025635) funded by the Korea government (MSIP). Yong-Sun Ju of KOASIS Inc. has contributed in 3D finite element analysis of the STELLA-2 components in this study.

NOMENCLATURE

P_1 : effective primary stress intensity
 P_2 : effective primary stress intensity of the sum of primary stresses
 P_L : local primary membrane stress
 P_b : local primary bending stress
 P_m : primary membrane stress
 S_m : design stress intensity at a given temperature
 U : creep usage fraction
 W : creep rupture usage fraction
 Φ : coefficient on the geometry of the cross section

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