Review of Design Margins for Structural Integrity Evaluation of SALUS Reactor Internal Structures according to ASME Section III Division 5

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1. Introduction

The conceptual design of the Generation-IV advanced small reactor SALUS(Small, Advanced, Long-cycled and Ultimate Safe SFR)[1] with a long-term reactor core has been under development and it is based upon the structural concept and design specifications of the PGSFR[2, 3]. The preliminary structural design was carried out and its structural integrity against the design loads was confirmed to meet the long-term core and thermal fluid requirements[4].

In this study, the structural integrity of SALUS reactor internal structures subjected to normal operating loadings was evaluated per ASME Code, Section III, Division 5 (ASME Div.5) procedures. ASME Div.5[5] provides several methods to evaluate high temperature structural integrity, of which complexity and conservatism differ. The structural integrity margins according to these various evaluation procedures were compared and analyzed.

2. Structural Analysis of SALUS Reactor Internal Structures

The design of SALUS reactor internal structures has been changed from those of PGSFR to improve the structural integrity. A complex peanut-shaped redan structure was changed to a simple cylindrical upper inner vessel and a horizontal separation plate was changed to a conical middle inner vessel as shown in Fig.1. Support cylinders for main components such as Pump, IHX, and DHX are welded upward to the middle inner vessel. The core support structure has a skirt shape and is welded to the lug installed on the core support flange of the lower head of the reactor vessel, and the concept of the PGSFR is applied as it is.

Structural analysis was performed for the SALUS reactor internal structures with respect to the normal operating loadings shown in Fig.2 and the weight of the reactor structures and the pressure of sodium coolant were applied as primary loadings. The structural integrity assessment was carried out per ASME Div.5. The number of heat-up and cool-down transients shown in Fig.2 is assumed to be 39 during design life of 60 years.

Structural analysis model

ANSYS Version 2019[6] was used for a finite element analysis to evaluate the structural analysis of the reactor internal structures, and an analysis model was prepared as a half symmetrical model considering the symmetry of the structure. All the reactor internal structures are made of Type 316 Stainless Steel and its material properties are shown in Table 1. It is assumed that the physical properties of the core shielding structure are the same as that of the structural material Type 316SS, and only the density is set to 70% of the Type 316SS.

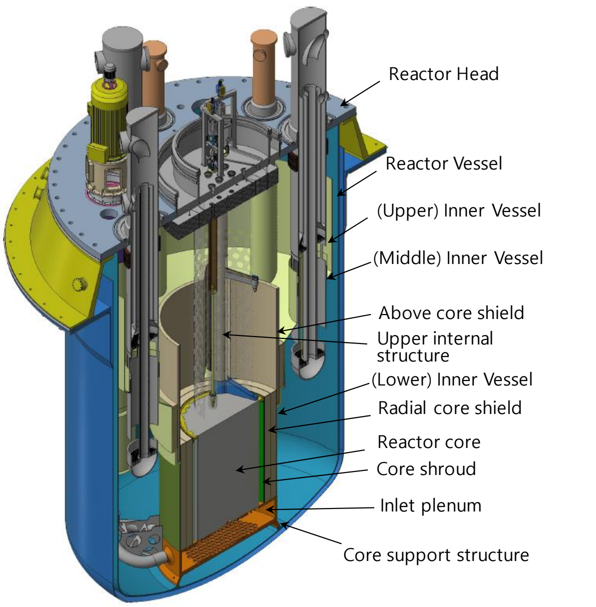

Fig.1 Reactor vessel and internal structures (SALUS)

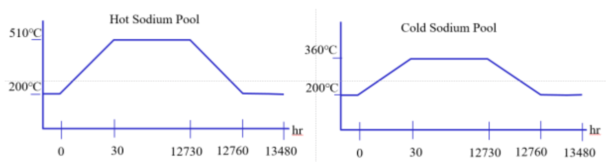

Fig.2 Level A Normal Operating Condition (1 cycle, Hot and Cold Sodium Pools)

Table 1. Material properties of Type 316 Stainless steel

Temp. (℃)	Thermal Conductivity (W/(m℃))	Thermal Expansion (Sequent) (m/m/℃)×10^{-6}	Specific Heat (J/(kg℃))	Elastic Modulus (GPa)	Density (kg/m^3)	Poisson's Ratio
20	14.1	15.3	491.9	195	8030	0.31
100	15.4	16.2	511.4	189	8030	0.31
200	16.8	17.0	525.7	183	8030	0.31
300	18.3	17.7	540.0	176	8030	0.31
400	19.7	18.1	552.5	169	8030	0.31
500	21.2	18.4	566.5	160	8030	0.31
600	22.6	18.8	574.4	151	8030	0.31

Fig.3 shows a finite element analysis model of the reactor internal structures. The element used for the analysis was Solid185 (3D 8-node structural solid) element and the total number of elements and nodes are 105,076 and 153,169, respectively.

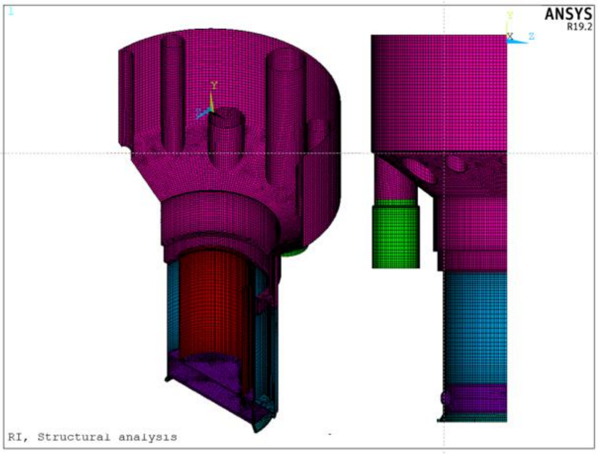

Fig.3 Finite element model of reactor internal structures

Structural analysis result

Fig.4 shows the stress distribution as a result of structural analysis of the reactor internal structures after 30 hours of heat-up loadings. As shown in Fig.4, the structural integrity was evaluated by selecting 4 major regions (Cut A ~ Cut D) showing the large stresses. As a result of the evaluations, the structural integrity was secured by satisfying the requirements of ASME Div.5 in all regions. Among them, the most interested section Cut C, which is the junction area between the inner vessel and the extended cylinder for a fuel transport port, was selected to assess the margins of various evaluation procedures in ASME Div.5.

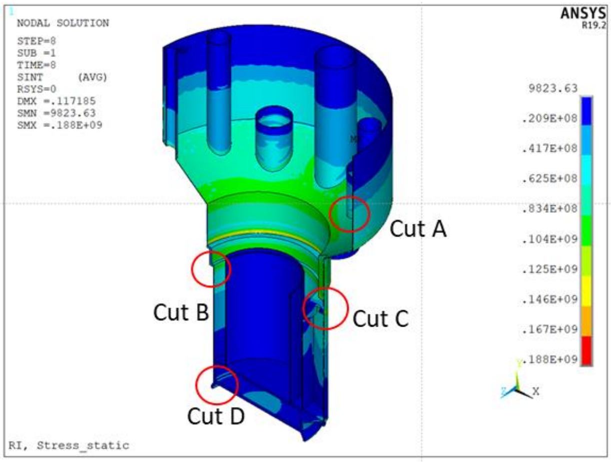

Fig.4 Stress intensity distribution of reactor internal structures

3. Margin Assessment for Structural Integrity of SALUS Reactor Internal Structures

When evaluating the structural integrity of a reactor structure, it is sufficient to satisfy the allowable stress requirements and the fatigue damage requirement below the temperature at which the structure does not undergo creep damage. However, at high temperature condition where creep occurs in a structure, the inelastic strain requirements and creep-fatigue damage requirement that occur during its lifetime should be additionally examined following the ASME Div.5 procedures in addition to the stress requirements.

Structural integrity assessment procedure in ASME Div.5

In principle, when we apply ASME Div.5 to the design and analysis of reactor internal structures, Subsection HGA and HGB need to be applied. Particularly, HGB is for the high temperature design and analysis of reactor internal structures including core support structure, its actual design rule is pretty similar to HBB though there are some subtle differences and both HGB and HBB use HBB-T when evaluating strain requirements and creep-fatigue damage requirements. In this study, HBB and HBB-T were considered for the purpose of comparative analysis with consistency.

HBB-3223 in ASME Div.5 describes the stress requirements for Level A loadings, which is similar to ASME Section III, Subsection NB. In addition, the creep-fatigue damage limit varies depending on the metallic materials when creep damage and fatigue damage occur at the same time, following the ASME Div.5 HBB T-1400 procedure.

Typical and complicated requirement is the inelastic strain requirements at high temperature conditions. The average inelastic strain of 1% and the bending strain of 2% should be satisfied during its lifetime. HBB T-1320 presents 3 procedures for satisfaction of strain limits using elastic analysis; Test A-1, A-2, and A-3. If one of the requirements of Test A-1, A-2, and A-3 is satisfied, it is determined that the strain requirement is satisfied. In addition, HBB T-1332 presents 2 procedures for satisfaction of strain limits using simplified inelastic analysis; Test B-1 and B-2, and if one of these is satisfied, it is determined that the strain requirement is satisfied.

The finite element analysis results for Cut C of reactor internal structure were applied to above 5 procedures (Test A-1, A-2, A-3, B-1, and B-2), and corresponding structural integrity margins were compared and analyzed. The key contents of those evaluation procedures are briefly described as follows:

- HBB-T-1322 Test No. A-1

$$\mathrm{X} + \mathrm{Y} \le S_a/S_y$$

where X is the primary stress index and Y is the secondary stress index.

- HBB-T-1323 Test No. A-2

$$\mathrm{X} + \mathrm{Y} \le 1$$

- HBB-T-1324 Test No. A-3

$$(a)\ \sum_i \frac{t_i}{rt_{id}} \le 0.1,$$

$$(b)\ \sum_i \varepsilon_i \le 0.2\%,$$

$$(c)\ 3\overline{S_m} = 1.5S_m + S_{rH}$$

- HBB-T-1332 Test Nos. B-1 and B-2

HBB-T-1332 provides a process of directly obtaining the inelastic strains over its lifetime by using an isochronous stress-strain curve and comparing it with the allowable values. The most important step in the procedure for evaluating the inelastic strains is to calculate the effective creep stress σ_c as shown in the following equation.

$$\sigma_c = Z \cdot S_{yL}$$

Here, Z is the effective creep stress parameter to be obtained by applying X and Y values to the Figure HBB-T-1332-1 (Test B-1) or Figure HBB-T-1332-2 (Test B-2). The total inelastic strain (called as creep ratcheting strain) was obtained by using the isochronous stress-strain curves of Figure HBB-T-1800 by applying $1.25\sigma_c$.

Structural integrity margin assessment

The structural integrity was evaluated for the reactor internal structures according to the ASME Div.5 by applying finite element analysis results.

Table 2 shows that the design margins for both inner and outer surfaces of Cut C region in the inner vessel subjected to a normal heat-up and cool-down loadings. Stress intensity values and creep-fatigue damage values satisfies the allowable values with sufficient margins.

Table 2. Structural integrity assessment (stress and creep-fatig ue damage)

Section	Location	Temp(℃)	Evaluation Item	Calculated Values	Allowable Values	Margin (%)	Code and Standard
Cut C	Inner	200/ 449.9	P_m	0.831E+01(MPa)	0.108E+03(MPa)	1200	ASME Sec.III Div.5 **HBB-3223**
			P_L+P_b	0.759E+01(MPa)	0.162E+03(MPa)	2030	
			P_L+P_b/K_t	0.761E+01(MPa)	0.139E+03(MPa)	1730	
			Fatigue Damage	0.612E-08	<0.3	0.146E+06	ASME Sec.III Div.5 **HBB T-1400**
			Creep Damage	0.684E-03	<0.3		
	Outer	200/ 427.7	P_m	0.832E+01(MPa)	0.110E+03(MPa)	1220	ASME Sec.III Div.5 **HBB-3223**
			P_L+P_b	0.121E+02(MPa)	0.165E+03(MPa)	1260	
			P_L+P_b/K_t	0.109E+02(MPa)	0.143E+03(MPa)	1210	
			Fatigue Damage	0.163E-05	<0.3	0.841E+03	ASME Sec.III Div.5 **HBB T-1400**
			Creep Damage	0.106E+00	<0.3		

Table 3 shows the comparison of design margins for strain requirements with respect to various evaluation procedures such as Test A-1, A-2, A-3, B-1, and B-2. As can be seen from the previous explanation, Test A-1 and A-2 procedures are the simplest and easiest to apply, but as shown in Table 3, the margin was the least compared to Test A-3, B-1, and B-2 procedures. Even the outer surface of Cut C was found to be not satisfied with structural integrity because of its excessive conservatism. Test B-1 and B-2 showed the same result with infinite margin instead of complicated application procedures, confirming that it is the least conservative evaluation method.

It can be summarized as follows from the result of this study.

- The order of simplicity of the application procedure is Test A-1/A-2 ＞ Test A-3 ＞ Test B-1/B-2.
- The order of conservatism of evaluation result is Test A-1/A-2 > Test A-3 ＞ Test B-1/B-2.

Table 3. Structural integrity assessment (inelastic strains)

Section	Location	Temp(℃)	Evaluation Item	Calculated Values	Allowable Values	Margin (%)	Code and Standard
Cut C	Inner	200/ 449.9	X + Y	0.59842	1.0	67.1	ASME Sec.III Div.5 **HBB (Test A-1)**
			X + Y	0.59842	1.0	67.1	ASME Sec.III Div.5 **HBB (Test A-2)**
			3Sm	0.729E+02(MPa)	0.363E+03(MPa)	398	ASME Sec.III Div.5 **HBB (Test A-3)**
			Thermal Stress Ratchet	0.729E+02(MPa)	0.176E+04(MPa)	231	
			UFS(Ti/Tim)	0.322E-05	0.1	3.11E+06	
			Creep Strain	0.000E+00(%)	0.200E+00(%)	Infinity	
			Creep Ratcheting Strain	0.000E+00(%)	1.0 (%)	Infinity	ASME Sec.III Div.5 **HBB (Test B-1)**
			Creep Ratcheting Strain	0.000E+00(%)	1.0 (%)	Infinity	ASME Sec.III Div.5 **HBB (Test B-2)**
	Outer	200/ 427.7	X + Y	1.3406	1.0	-25	ASME Sec.III Div.5 **HBB (Test A-1)**
			X + Y	1.3406	1.0	-25	ASME Sec.III Div.5 **HBB (Test A-2)**
			3Sm	0.170E+03(MPa)	0.366E+03(MPa)	115	ASME Sec.III Div.5 **HBB (Test A-3)**
			Thermal Stress Ratchet	0.170E+03(MPa)	0.179E+04(MPa)	953	
			UFS(Ti/Tim)	0.409E-55	0.1	2.44E+56	
			Creep Strain	0.000E+00(%)	0.200E+00(%)	Infinity	
			Creep Ratcheting Strain	0.000E+00(%)	1.0(%)	Infinity	ASME Sec.III Div.5 **HBB (Test B-1)**
			Creep Ratcheting Strain	0.000E+00(%)	1.0(%)	Infinity	ASME Sec.III Div.5 **HBB (Test B-2)**

4. Results and Discussion

In this study, the high temperature structural integrity of SALUS reactor internal structures subjected to normal operating loadings was evaluated per various procedures in ASME Di.5. As a result of the evaluations, the structural integrity was secured by satisfying the requirements of ASME Div.5 in all regions. Among them, interested section (Cut C) was selected to assess the margins of various evaluation procedures for inelastic strain requirements of ASME Div.5. The result of this study indicates that the order of simplicity of the application procedure is (Test A-1/A-2 ＞ Test A-3 ＞ Test B-1/B-2) and the order of conservatism of evaluation result is (Test A-1/A-2 > Test A-3 ＞ Test B-1/B-2). And it is recommended to use Test B-1 or B-2 procedures for structural integrity assessment since they have the lowest conservatism.

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