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Structural Integrity Evaluation of KALIMER-600 Reactor Internal Structures for Normal Operating Loads

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1. Introduction

KALIMER-600(Korea Advanced LIquid MEtal Reactor, 600Mwe)[1] is a pool type sodium-cooled liquid metal reactor and its normal operating temperature is 545℃. Because the primary heat transfer system is operating in a reactor vessel and thus a severe thermal load can be induced, the reactor internal structures should be installed in a reactor vessel as a thin shell type to minimize the bending stress.

This study is on the structural integrity evaluation of the reactor internal structures based on the FE(Finite Element) structural analysis for a normal operating condition. The structural integrity of the reactor internal structure is evaluated by ASME Boiler and Pressure Vessel Code Section III, Subsection NG, in the case where the metal temperature does not exceed 427℃(800°F). But in the case of exceeding 427℃, the ASME Code Case N-201-4 is used. This paper deals with the stress limits, the accumulated inelastic limits and a creep-fatigue damage evaluation for the most critical part of the reactor internal structures.

2. Reactor Internal Structures

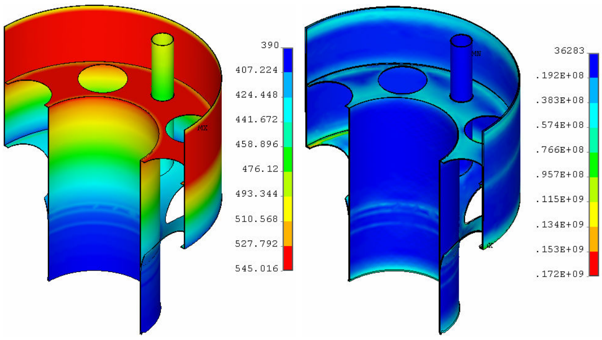
2.1 Design Features

KALIMER-600 reactor internal structures have 3-main functions for providing 1) a core support, 2) a primary coolant flow path, and 3) a component support. Basically all the reactor internal structures are designed to meet these functional requirements[2].

The KALIMER-600 reactor internal strucrtures are mainly composed of the Core Support Structure(CSS), the Inlet Plenum(IP), the Support Barrel(SB), the Baffle Plate(BP), the Reactor Baffle (RB) and the Reactor Baffle Support. KALIMER-600 adopts the baffle structure type to protect the RV from a high temperature thermal loading and the detached CSS to eliminate the excessive thermal stresses caused by a thermal expansion difference. The BP which forms the boundary between the hot coolant and the cold coolant is predicted to induce the thermal stress concentration. For the material data of the reactor internal structures, 316 SS is used now.

2.2 Heat Transfer and Stress Analysis

In this study, the tempeature distribution can be calculated from Ref.[3] by using ANSYS 9.0 finite element analysis software[4] for a normal operating condition. Fig. 1(a) shows the calculated temperature distribution of the half axisymmetric model. As shown in Fig. 1(a), the hot sodium bulk temperatures inside the support barrel are 545℃ for the upper region of the support barrel elevation and 405℃ for the lower region of the support barrel elevation. The cold sodium bulk temperature is 390℃ which is applied to the outer surface of the SB and the inner surface of the RV.



(a) Temperature Distribition (b) Stress Distribiton
Figure 1. Finite Element Analysis Results of the Reactor Internal Structures for Normal Operating Condition.

Fig. 1(b) shows the stress analysis result under the thermal load and structural dead weight for normal operation. The stresses are significantly occurred at the junction part between the support barrel and baffle plate, and the juntion part between the reactor baffle and baffle support structure.

3. Structural Integrity Evaluation

In a previous research work, the structural integrity evalutions for the lower parts of the reactor internal structures were carried out by using ASME Subsection NG. In this paper, ASME Code Case N-201-4[5] for an elevated temperature design of reactor internal structures is used to evaluate the structural integrity. In evaluating the structural integrity for an elevated temperature, it can be evaluated by the following three quantities such as the load-controlled stresses, the total accumulated inelastic strain and creep-fatigue damage.

3.1 Load-Controlled Stress Limits

The stress calculations required for the analysis of Level A and B Service Loadings are based on a linearly-elastic material model. The calculated stress intensity values calculated in the range of an elevated temperature should satisfy the following conditions according to the classification of a stress intensity.

$$P_m \le S_{mt} \quad (1)$$

$$P_m + P_b \le KS_m \quad (2)$$

$$P_m + P_b / K_t \le S_t \quad (3)$$

$$K_t = (K+1)/2 \quad (4)$$

where, the factor K is 1.5 for a shell type structure and thus the K_t is 1.25 in Eq. (4).

The total service life of the KALIMER-600 reactor is 60 years and the refueling interval is a minimum of 18 months. Therefore, the number of the cyclic history occurring from a hot standby to a steady state operation is assumed conservatively as 60 during the reactor life time. The total numer of hours of the normal operation is assumed as 525,600 hours and thus the average cycle time becomes 525,600/60=8,760 hours.

From the stress analysis result, the structural integrity by the ASME design code at the most critical section is evaluated. Table 1 shows the evaluation results at the reactor baffle support. From Table 1, the stress value satisfies the rules of the service limits with enough margins.

Table 1. Stress Limit Checks (MPa)

Section No.	$P_m \le S_{mt}$		$P_m + P_b \le KS_m$		$P_m + P_b / K_t \le S_t$	
	P_m	S_{mt}	$P_m + P_b$	KS_m	$P_m + P_b / K_t$	S_t
No.1	31.4	102	45.7	153	42.8	140

3.2 Deformation and Strain Limit

The deformation and strain limits for a structural integrity are evaluated by the inealstic strain limits and the strain limits by using elastic analysis. The strain limits by using the elastic analysis are considered to have been satisfied if the limits of any one of Test No.A-1, A-2, A-3 which are described in ASME Code Case N-201-4 are satisfied. In this study, the strain limit is evaluated by the strain limits by using the elastic analysis, specially Test No.A-1.

$$X + Y \le S_a / S_y \quad (5)$$

$$X = (P_L + P_b / K_t)_{\max} \div S_y \quad (6)$$

$$Y = (Q_R)_{\max} \div S_y \quad (7)$$

Where, $(Q_R)_{\max}$ is the maximu range of the secondary stress intensity during the cycle being considered and S_y is the average of the S_y values at the maximum and minimum wall averaged temperature during the cycle. From the stress analysis result and Eqs.(5) through (7), the strain limit is evaluated and it satisfies the limit with a very small design margin for a normal operation.

$$(X + Y = 0.988) \le (S_a / S_y = 1) \quad (8)$$

3.3 Creep-Fatigue Damage Evaluation

The accumulated creep and fatigue damage should satisfy the following relation for a combination of the Level A, B and C Service Loadings.

$$\sum_{j=1}^{p}\left(\frac{n}{N_d}\right)_j + \sum_{k=1}^{q}\left(\frac{\Delta t}{T_d}\right)_k \le D \quad (9)$$

where, D, P, $(n)_j$, $(N_d)_j$, q, $(T_d)_k$ are defined in Ref.[5] and the total damage should not exceed the creep-fatigue damage envelope curves[5].

Considering the operation life time, the evaluated total damage value is about 0.31 and this is located inside the creep-fatigue damage envelope as shown in Fig. 2.

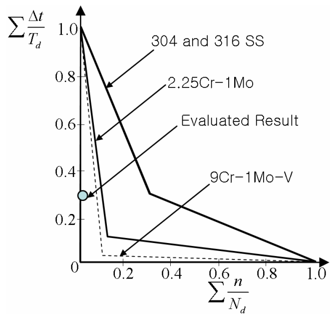


Figure 2. Creep-Fatigue Damage Envelope and Result

4. Conclusion

This paper is on the structural integrity evaluation of the KALIMER-600 internal structures for a normal operation by using the ASME Code Case N-201-4.

From the evaluation results, the maximum stress part of the internal structures satisfies the code rules for the stress limits with a enough margin. But, for a satisfaction of the deformation and strain limit, the selected part satisfies the rules by using the simplified inealstic analysis method with a small design margin. From the evaluation of the creep-fatigue damage, the fatigue damage is negligible and the selected critical part satisfies the creep-fatigue damage.

In addition to this study, we will make efforts to verify the structural integrity of several other regions of the reactor internal structures.

ACKNOWLEDGMENTS

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