

Structural Integrity Evaluation of Reactor Enclosure System for PGSFR

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1. Introduction

The reactor vessel of PGSFR(Prototype Gen-IV Sodium Fast Reactor) is the container for the primary sodium and the support for the reactor internal structures. The reactor vessel bottom head provides the support flange for the core support structure. The vessel should carry the weight of reactor internal structures and core transferred to the core support flange through the core support structure. The vessel also carries those loads of contained primary sodium, and its own weight in tension up to its integral connection with the reactor support structure. During normal and abnormal conditions, the reactor vessel is subjected to an elevated temperature higher than 425 °C that can cause creep damage for Type 316 stainless steel. Therefore ASME Boiler and Pressure Vessel Code, Section III Division 5 is applicable for elevated temperature design.

The RES(Reactor enclosure system) including reactor vessel, containment vessel, reactor head and reactor support structure shall be designed to withstand all of the pressures, temperatures and forces which are likely to be imposed on them. Especially, reactor vessel containing elevated temperature sodium coolant suffers a rapid temperature gradient in the upper cover gas region. This causes high thermal stress at the upper region of reactor vessel, and therefore a design for reduction of thermal stress is indispensable. In this paper, a design for mitigating the thermal stress of RES was presented. And the structural integrity of RES was verified in case of design condition and service level A of ASME Boiler and Pressure Vessel Code, Section III-Subsection NH [1].

2. Finite element analysis

2.1. Design modification

In the case of past design of reactor vessel, excessive thermal stress was occurred at the upper end of reactor vessel due to rapid temperature decrease of short cover gas section and high rigidity of reactor supports. To solve this problem, a method to lower the sodium free surface level to 1.5m was proposed. And a design for lowering the stiffness of the reactor supports was also proposed as shown in Fig. 1.

2.2. Geometry

The height and thickness of RV are 15.574 m and 50 mm, respectively. The shape of bottom head is torispherical type. And there are no penetration and attachment. The front view of RV is given in Fig. 2.

2.3. FE analysis

To perform FE(Finite Element) analysis of RV, axisymmetric FE model of RES was modeled as shown in Fig. 3.

FE analysis was performed by using ANSYS 15.0 and PLANE183(8-Node Structural Plane element)element and PLANE77(8-Node Thermal Plane element)element were used for structural and thermal analysis, respectively [2].

A total of four primary loads were considered as Table I, and the stress distribution for each load case was shown in Fig. 4. As can be seen from the results in Fig. 4, the stress due to the weight of reactor internal structure and fuel assembly was the highest at 64.4 MPa and occurred on the internal support flange.

During steady state operation, the temperature of the reactor vessel rose to 518 °C near the cold-pool free surface level as shown in Fig. 5(a). In this case, the thermal stress distribution was shown in Fig. 5(b) and the maximum stress was calculated as 326 MPa above the reactor support.

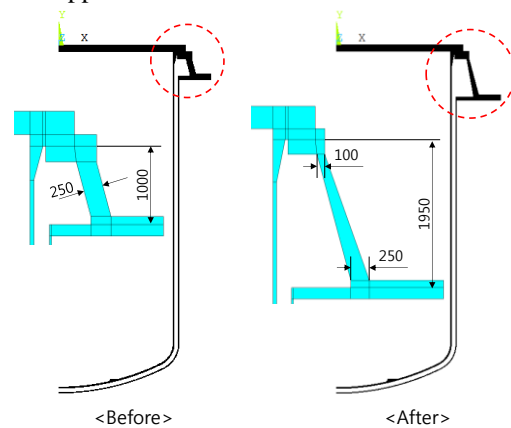


Fig. 1. Design modification of reactor vessel support structure

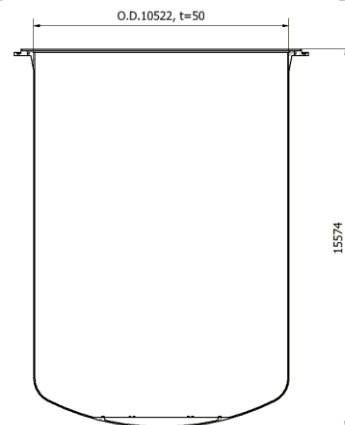


Fig. 2. Front view of the reactor vessel

Table I : Primary load cases applied to reactor vessel

Load case	Load
1	Weight of reactor vessel
2	Weight of both components supported by reactor head and 2 nd sodium
3	Weight of reactor internal structure and fuel assembly
4	Hydrostatic pressure of sodium coolant

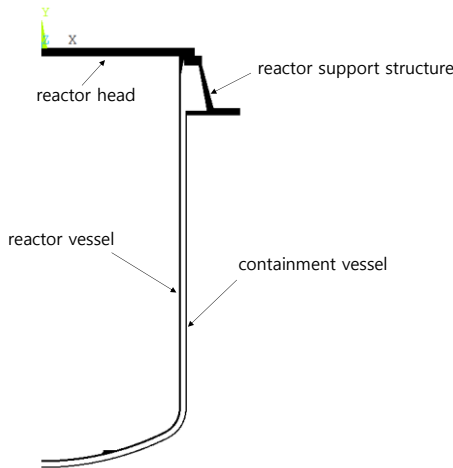


Fig. 3. FE model of reactor enclosure system

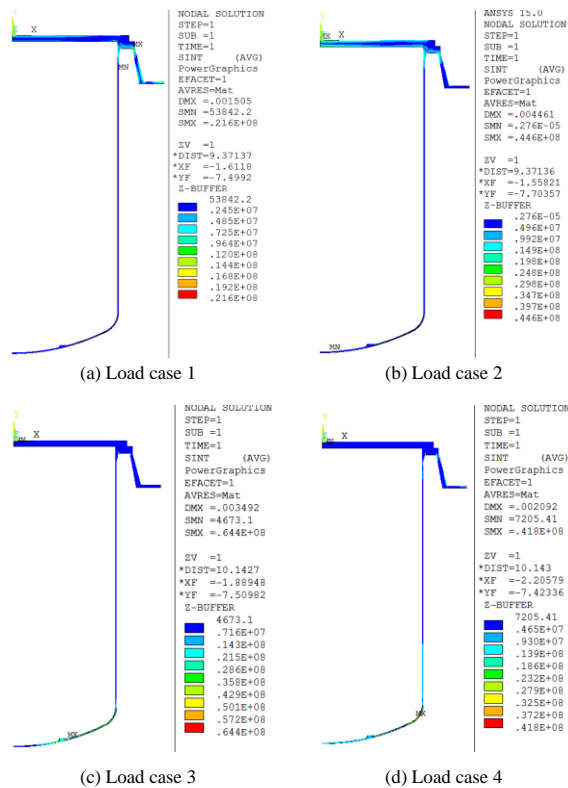
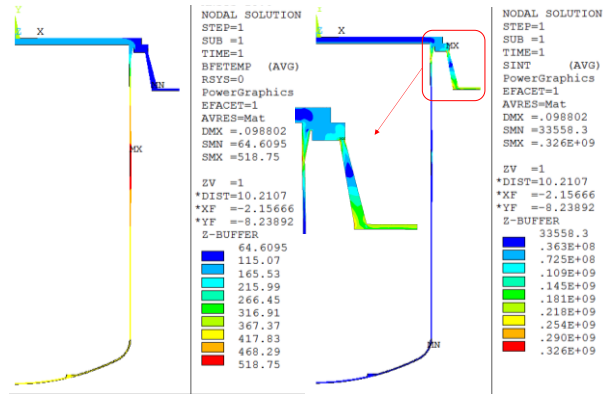


Fig. 4. FE analysis result of primary load



(a) Temperature distribution (b) Thermal stress distribution

Fig. 5. FE analysis result of thermal load

3. Structural integrity evaluation

Structural integrity evaluation according to ASME Boiler and Pressure Vessel Code, Section III-Subsection NH was performed for 5 sections which were selected as high structural and thermal stress points as shown in Fig. 6.

Linearized primary stresses for design condition were given in Table II. And linearized primary and thermal stresses for service level A were given in Table III and Table IV, respectively.

From the evaluation results of design condition given in Table V, the minimum design margin which was defined in equation (1) was evaluated as 0.42 at bottom head connecting point. In case of service level A, the minimum design margin was evaluated as 0.27 at upper part of reactor support structure.

$$\text{Design margin} = \frac{\text{Allowable Stress}}{\text{Calculated Stress}} - 1 \quad (1)$$

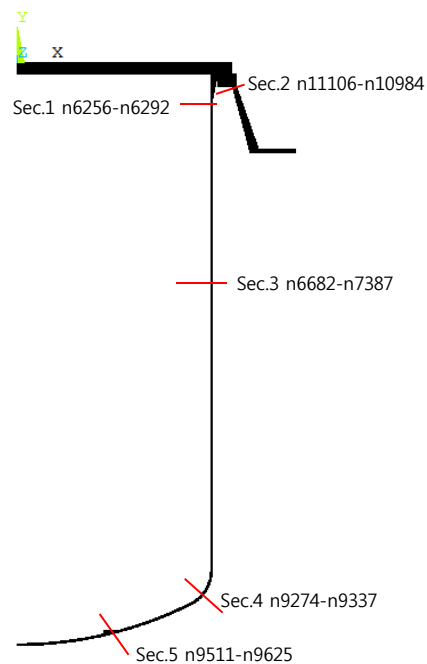


Fig. 6. Structural integrity evaluation sections

