Preliminary Structural Integrity Evaluation of a Reactor Internal Structure for a SMFR

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1. Introduction

A preliminary structural analysis for the reactor internal structure is performed and the results are evaluated to confirm the structural integrity in accordance with ASME B&PV Code Section III, Subsection NH. A reactor internal structure is a single integrated unit that separates the hot pool from the cold pool, and provides for a communication of the hot sodium from the discharge of the reactor core to the inlet of the intermediate heat exchanger. The coolant temperature of the hot and cold plenum are 510 ^{o}C and 355 ^{o}C respectively at nominal steady state condition. The material of a reactor internal structure is 316 stainless steel, which is classified as a class 1 component in this report. A reactor internal structure is evaluated for the given thermal and mechanical loads in Service Level A condition. The adequacy of a reactor internal structure is evaluated by checking its structural responses with the ASME code stress limits and the structural deformation limits at elevated temperature[1].

2. Short Description and Modeling

A reactor internal structure of a SMFR(Small Modular Fast Reactor) consists of multiple plates welded together that form a contoured shape around the intermediate heat exchangers and the upper internal structure. A reactor internal structure contains the hot sodium from the core outlet and helps to minimize leakage from the hot sodium to the cold sodium sides of a reactor internal structure. It is supported vertically by the lower internal structure and seal welded core barrel. The main dimensions are 11.1 m in height and 0.02 m in thickness.

A finite element analysis was performed using FEA software ANSYS. The simple three dimensional model without a mechanical seal between the IHX and a reactor internal structure is presented in Fig. 1 and three critical locations shown in such figure were evaluated for the structural integrity check. Fig. 2 shows the boundary conditions of reactor internal structure during the steady state condition.

The resulting F.E model was calculated to three sets of mechanical and thermal load cases. The mechanical load considered the self weight and the thermal load considered three cycle types based on the duty cycle events of a SMFR given in Table 1. The total design life time for the structural evaluation is assumed as 60 years. Fig. 3 shows the coolant temperature distribution of cycle type 1 from refueling to full power.

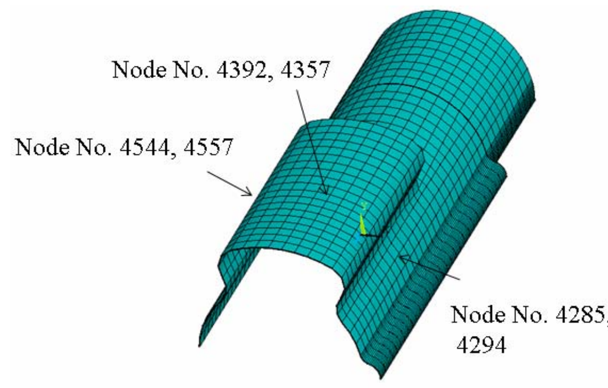

Fig. 1 Finite element model of a reactor internal structure

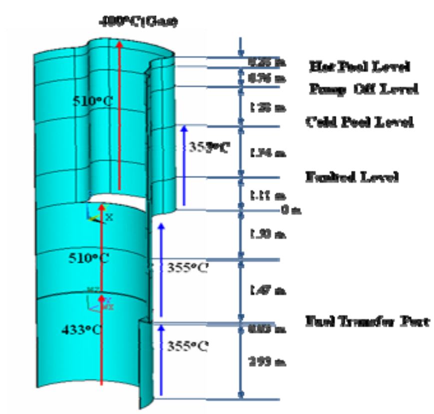

Fig. 2 Boundary conditions of a reactor internal structure during the steady state condition.

In this figure, the maximum temperature of the hot region in the cycle type 1 is 510 ^{o}C and the time duration is 424 hour with 180 cycles.

Table 1. Duty cycle events(Thermal load)

Events	Event Name	No. of Cycle	Time Duration	Life Time	Remarks
Level A				526528 hr	
Cycle Type 1	Start up from refueling temp. to full power	180	424 hr	76320 hr	
Cycle Type 2	Start up from Hot standby to full power	851	152 hr	129352 hr	
Cycle Type 3	Daily loading and unloading	13369	24 hr	320856 hr	

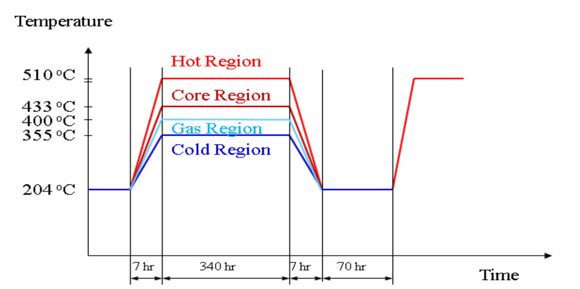

Fig. 3 Coolant temperature distribution from the refueling to full power(cycle type 1)

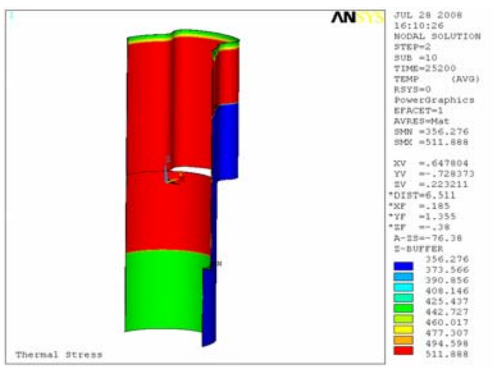

Fig. 4 Temperature distribution of reactor internal structure during the steady state condition

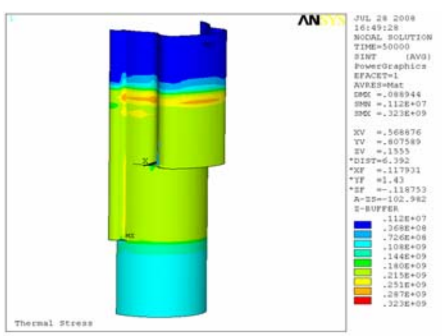

Fig. 5 Stress intensity distribution of reactor internal structure during the steady state condition

3. Results and Discussions

A reactor internal structure is evaluated for the given thermal and mechanical loads in Service Level A condition. Fig. 4 and Fig. 5 show the temperature and stress intensity distributions of reactor internal structure during the steady state condition, respectively. The stress calculation result shows the most highly loaded region to be located in the upper part of a reactor internal structure at the cold pool free surfaces[2].

Total six points at those surfaces were calculated for the structural integrity check, and the most critical position was located at the node 4294 of the hot pool region for a reactor internal structure. Table 2 shows the structural integrity check results in such position. The membrane and bending stresses due to the primary stresses are very small and the primary stresses are acceptable with great design margins. In the case of the inelastic strain results, the elastic approach does not satisfy the limit value of 1.0 because the effect due to the secondary stress is great, but the simplified inelastic approach satisfies the limit value of 1.0%. For the check results of creep fatigue limit, the calculated fatigue and creep damage is 0.0052 and 0.3147, respectively. The calculated mechanical and thermal stresses and strains are acceptable with enough margins against the allowable stress limits.

Table 2. Structural integrity check results(node 4294)

Evaluation Items	Calculated	Limit value	Check
❑ Primary Stress Limits			
Membrane	0.9037	109.3	OK
Membrane + Bending	0.8247	142.7	OK
❑ Inelastic Strain Limits			
Elastic Approach	1.6363	1.0	NOT OK
Simplified Inelastic Approach	0.00 %	1.0 %	OK
❑ Creep-Fatigue Limits			
Fatigue Damage	0.0052	0.2937	OK
Creep Damage	0.3147	0.9879	OK

4. Conclusions

For the given event duty cycles of a SMFR, the structural integrity of the reactor internal structure is confirmed with the proper margins per the design criteria of the subsection NH. All the check items with the primary stress evaluation, inelastic strain evaluation, and the creep-fatigue damage evaluation are below the corresponding allowable limits required the subsection NH.

Acknowledgements

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REFERENCES

[1] ASME Boiler and Pressure Vessel Code Section III, Subsection NH, ASME, 2004.
[2] ASME Boiler and Pressure Vessel Code Section II, Part A, Part D, ASME, 2004.