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Structural Integrity Evaluations of a Reactor Vessel for a Sodium-Cooled Fast Reactor Subjected to Elevated Temperatures

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1. Introduction

The elevated temperature structural design technologies, which include an accumulated inelastic strain, creep, creep rupture, and creep-fatigue, are being developed for a GEN-IV sodium-cooled fast reactor design as a part of an International Nuclear Energy Research Initiative (I-NERI) Project. In this paper, the tentative structural integrity of a reactor vessel is investigated and discussed for a normal operating condition in elevated temperatures. For the elevated temperature structural evaluation, the SIE ASME-NH program[1], which is a computer program implementing detailed rules of the ASME-NH code for class 1 components [2], is used.

2. Evaluations and Results

This section includes the design condition, a simple description, general assumptions used in the analysis, loading conditions, analysis model and boundary conditions, and the results and discussions in brief.

2.1 Design Conditions

The target reactor considered in this paper is a SMFR (Small Modular Fast Reactor) pre-conceptually designed by Argonne National Laboratory (ANL) [3]. The reactor power is 125MWt (50MWe). It is classified as a class 1 component and is categorized as a seismic class 1. The design lifetime is 60 years with 30 years core life without refueling. The maximum operating temperature is 510^{o}C and the maximum operating pressure is assumed to be 0.5MPa. The design material of the reactor vessel is 316 austenitic stainless steel.

2.2 Descriptions

The overall configuration of the reactor system is shown in Fig. 1. As shown in the figure, the reactor vessel is supported by the support skirt and the outer is surrounded by another vessel called the guard vessel. Insulation is provided on the exterior of the guard vessel to reduce the heat lost to the guard vessel cooling system.

The total height is 13.35m and the inner diameter is 5.57 m. The annulus gap between the reactor vessel and the guard vessel is 0.2m.

2.3 General Assumptions

- All the dead weight of the pool coolant is uniformly applied to the RV bottom head as an equivalent pressure.
- The weights of the Rx internals and core assemblies are uniformly applied to the RV bottom head as an equivalent pressure.
- Assumed a total weight of the internals (300 tons) and core assemblies (200 tons)
- Although insulation is provided on the exterior of the guard vessel to reduce the heat lost to the guard vessel cooling system, there is still a small heat loss to the exterior air. Then, it is assumed that the outer surface of the guard vessel has a thermal boundary condition with a constant (60^{o}C) temperature of the cooling air and a film coefficient of 0.5 W/^{o}C-m^{3}.
- It is assumed that the heat transfer between the reactor vessel to the guard vessel is only by radiation. The heat convection due to the inert gas filled in a gap is neglected.

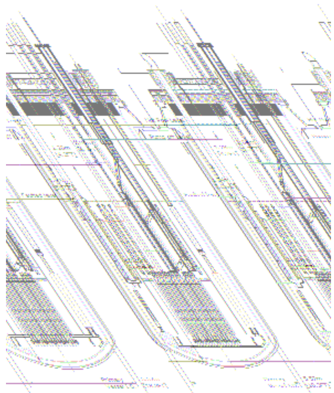

Fig. 1. Pre-conceptually Designed ABTR

2.4 Loading Conditions

The loading condition considered in this paper is a normal operating condition with a heat-up and a cool-down event. Figs. 2 shows an assumed thermal transient load cycle for the primary coolant and the inert gas filled in above coolant free surface. As shown in the figure, the duration of a heat-up and a cool-down is both 12 hours with a linear increment and decrement.

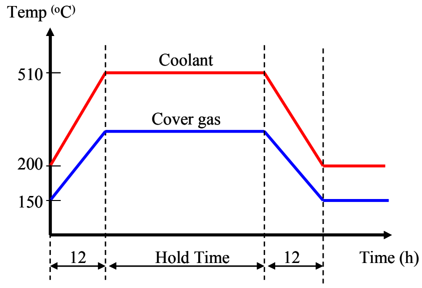

Fig. 2. Assumed Thermal Transient Loading Cycle

2.5 Analysis Model and Boundary Conditions

Fig. 3 reveals an axisymmetric finite element analysis model showing the thermal boundary conditions. As shown in the figure, it is conservatively assumed that the temperature of the top surface of the support flange and the support concrete surface is constant at 120°C and 21°C respectively. For the tentative evaluations, the model is prepared with the assumption that the primary coolant (510°C) is directly contacted with the whole reactor vessel inner surface. The dead weights of the primary coolant and the internal structures are applied to the reactor vessel bottom head with equivalent pressure loads as shown in Fig. 4.

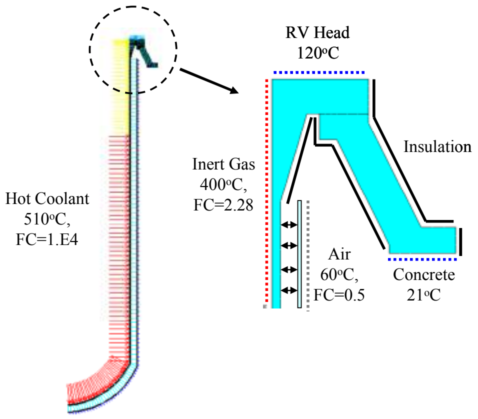

Fig. 3 Assumed Thermal Boundary Conditions

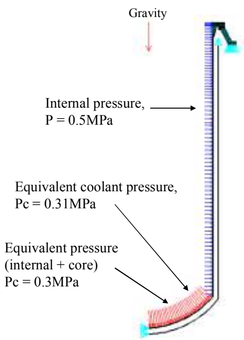

Fig. 4 Assumed Primary Load Boundary Conditions

2.6 Results and Discussions

For the design condition, the dead weights of the primary coolant and the internal structures, which exert a load on the bottom head, result in a significant stress concentration on a junction region between a side cylinder and the bottom head as shown in Fig. 5. The local membrane plus a bending stress intensity (225MPa) exceeds the design limit value of $1.5S_m$ (171MPa). Therefore, the optimal transition shape design of a junction part is strongly required.

Figs. 6 and 7 present the hoop stress time history responses for the heat-up and the cool-down respectively. From the results of the stress analyses, the maximum secondary stress range is determined at the times just after the ends of the heat-up and cool-down operations.

Table 1 presents the structural integrity evaluations at a section of the hot pool free surface region. As shown in the table, the primary stress limits are satisfied with enough margins but the creep ratcheting strain significantly exceeds the limit value for both rules of an elastic analysis and a simplified inelastic analysis. Therefore, the evaluation of the creep-fatigue damage rule of ASME-NH is not applicable in this design condition.

3. Conclusions

Tentative evaluation of a reactor vessel with a direct coolant contact temperature of 510°C can not satisfy the ASME-NH rules. Therefore, it is necessary to investigate a more detailed design condition and to perform an inelastic analysis with a verified constitutive model.

ACKNOWLEDGMENTS

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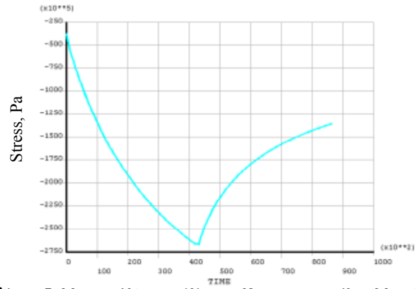

Fig. 5 Hoop Stress-Time Response for Heat-up

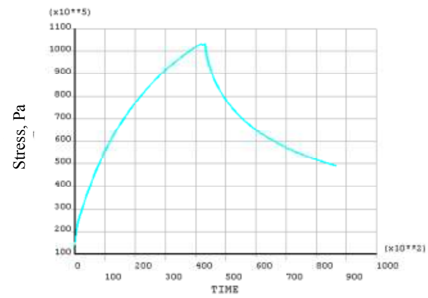

Fig. 6 Hoop Stress-Time Response for Cool-down

Table 1. Evaluation Results at Hot Pool Free Surface

Evaluation Items	Calculated	Limit value	Check
❑ Primary Stress Limits			
Membrane	29.4	106.9	OK
Membrane + Bending	29.6	109.6	OK
❑ Inelastic Strain Limits			
Elastic Analysis (X+Y)	3.17	1.0	Not OK
Simplified Inelastic Analysis (Creep ratchet strain	4.6 %	1.0 %	Not OK
❑ Creep-Fatigue Limits			
Fatigue	–	–	N/A
Creep	–	–	N/A