Creep-Fatigue Damage Evaluation of a Model Reactor Vessel and Reactor Internals of Sodium Test Facility according to ASME-NH and RCC-MRx Codes

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1. Introduction

The Sodium-cooled Fast Reactor (SFR) is one of Generation IV nuclear reactor designs, which are aimed for great improvements in safety and nuclear fuel economy. The STELLA-2 (Sodium Test Loop for Safety Simulation and Assessment) in Fig. 1 is a sodium testing facility designed by Korea Atomic Energy Research Institute (KAERI), and is the second phase of the program from the STELLA-1 [1]. STELLA-2 is a 1/5 scaled-down model of a prototype Gen-IV SFR (PGSFR) [2] in its length scale. The objective of the STELLA-2 is to support the specific design approval for PGSFR by synthetic reviews of key safety issues and code validations through the integral effect tests. Due to its high temperature operation in SFRs (and in a testing facility) up to 550 ℃, thermally induced creep-fatigue damage is very likely in components including a reactor vessel, reactor internals (interior structures), heat exchangers, pipelines, etc. In this study, structural integrity of the components such as reactor vessel and internals in STELLA-2 has been evaluated against creep-fatigue failures at a concept-design step. As 2D analysis yields far conservative results [3], a realistic 3D simulation is performed by a commercial software. This paper mainly focuses on a reactor vessel and its internals in STELLA-2, as it contains important function modules such as a core, and gives a high temperature gradient. The evaluation of creep-fatigue damage is performed according to the ASME Section III Subsection NH [4] and RCC-MRx [5] codes. The results are briefly contrasted as well.

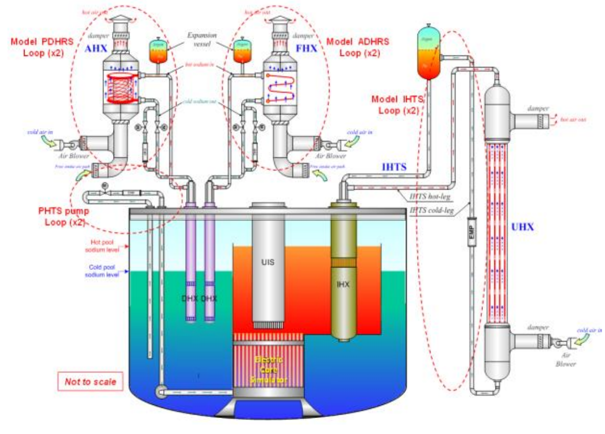

Fig. 1. Conceptual Configuration of component arrangements in STELLA-2

2. Finite Element Analysis

A 3D finite element model for STELLA-2 was constructed from a 3D Computer-Aided Design geometry of the vessel with internals. Since the given structure is axis-symmetry, a half 3D Ansys [6] model shown in Fig. 2 is used having cut surfaces constrained. The FE model takes 3D element types of SOLID70 and SOLID185 for heat transfer and thermal stress analyses, respectively. The figure shows the half model of the vessel. The yellow curved inner container with a green separate plate called Redan divides hot and cold sodium pool. Long purple columns are an intermediate heat exchanger (IHX) which delivers thermal energy to a steam generator. A red flange supports the whole structure held stationary from the ground. A magenta cylinder contains a nuclear core simulator. A dummy body is placed for inner sub-components considering only their weight and pressure as a 1D element (MASS21).

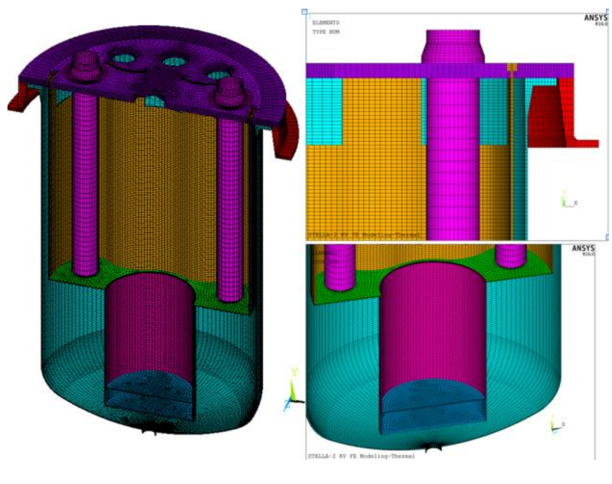

Fig. 2. Reactor Vessel Finite Element Model with inner structures as solid mesh elements

The reactor vessel, internals, and vessel head of STELLA-2 are designed of 316L stainless steel instead of 316SS, which is the actual design material of reactor structures in PGSFR, for the sake of the procurement. With the material selection, though, the property of 316L is substituted by 316SS as in Table I to follow the high-temperature evaluation in case of the ASME-NH.

For this study, a transient operational condition—heat-up for 3.5 hours, steady-state for 72 hours, and cool down for another 3.5 hours—is assumed (Fig. 3).

From a steady-state when two pool temperatures are equal to 200℃, a core outlet reaches up to 550℃ while a cold plenum rises only to 390℃. During the operation, cold sodium around the exits of IHXs is heated up around the core simulator, and sent to Redan. Through the heat exchangers, hot sodium is cooled down, and discharged as cold at the exits. In a real plant, this procedure is repeated indefinitely.

Table I: Material Property for High-Temp. Evaluation

Temp [°C]	**S** [MPa]	**Sy** [MPa]	**E** [GPa]	**α** [10^{-6}mm /mm-°C]	**TC** [W/m-°C]	**TD** [$10^{-6}m^2/s$]
20	138.0	207	195	15.3	14.1	3.57
200	134.0	148	183	17.0	16.8	3.98
300	119.0	132	176	17.7	18.3	4.22
400	111.0	123	169	18.1	19.7	4.44
500	107.0	118	160	18.4	21.2	4.66
550	105.0	116	156	18.6	21.9	4.78
600	80.3		151	18.8	22.6	4.90

* S: Strength, TC: Thermal Conductivity, TD: Thermal Diffusivity
The Poisson ratio ν 0.31 and density ρ 8,030 kg/m^3 are taken the same for all the temperature range

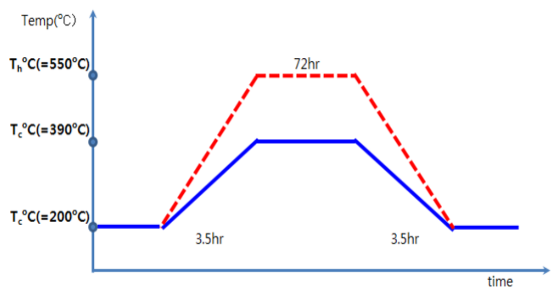

Fig. 3. Thermal Loading Conditions for Heat Transfer Analysis

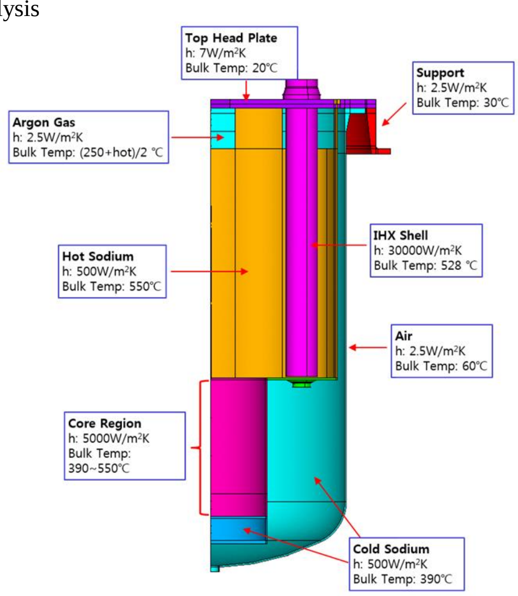

Fig. 4. Thermal boundary conditions of the vessel, of which only the half is shown here, are determined by pre-processed thermo-fluid dynamic numerical analysis.

For the sake of brevity of this paper, only the thermal stress analysis is considered, because pressure due to gravity–the components' weight–is far less significant than thermal loads. An SFR plant typically operates in low pressure and high temperature. Fig. 4 shows thermal boundary conditions in a simplified but conservative model.

Creep-fatigue damage evaluation for selective critical points of the object is performed based on the most recent edition of ASME-NH [4] and RCC-MRx [5] codes, and results are compared in between. The linear damage summation rule is applied to both codes as given in Eq. (1).

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

where p and q is a number of different cycle types and a time interval for creep damage calculation, respectively. Indices, j and k, indicate a cycle type. n and Δt is a number of applied repetitions of cycle type j and a hold time applied for one creep-fatigue load of k, respectively. N_d is a number of design allowable cycles for j, and T_d is an allowable time duration determined from the stress-to-rupture curves during the time interval k. D is total creep-fatigue damage.

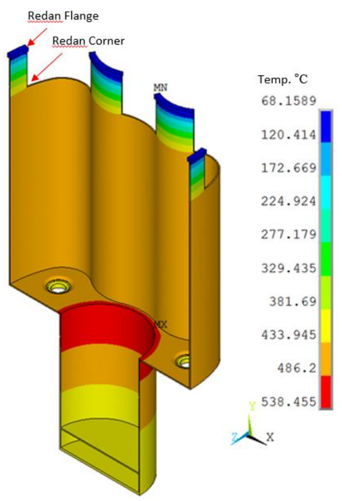

Fig. 5. Heat Transfer Analysis Result of the Redan structure: the temperature at the core is the highest, and gets colder as it goes to the head.

3. Results

The structural integrity of the reactor vessel and internals of STELLA-2 designed regarding creep-fatigue damage was evaluated according to the high temperature design codes: ASME-NH and RCC-MRx. A few critical locations–reactor vessel, reactor head, separate plate, reactor support, Redan–were inspected in lieu of the whole structure. Among them, Redan flange and corner sites were found as the most severe stress state with a secondary (thermal) stress of up to 328 MPa which is about 10 times greater than its primary stress due to mechanical loads. Even if Redan flange stress 328 MPa exceeds Redan corner 258 MPa as shown in Fig. 5, the temperature at the flange reaches up to only 250℃, while it is 470℃ at the corner after heating up for 4.25 hours. Thus only the Redan corner is focused on for the high temperature evaluation.

The stress components are linearized at sites of interest, and three terms are decomposed: membrane, bending and peak stress. At the Redan corner, the stress ranges from 90.2 (outside of Redan) to 285.5 MPa (inside) as shown in Fig. 6. (The material used at the site, 316SS, allows up to the stress level of 321 MPa without failure.) A red line in the figure shows the path for the stress linearization. The linearized stress is summarized in Table II.

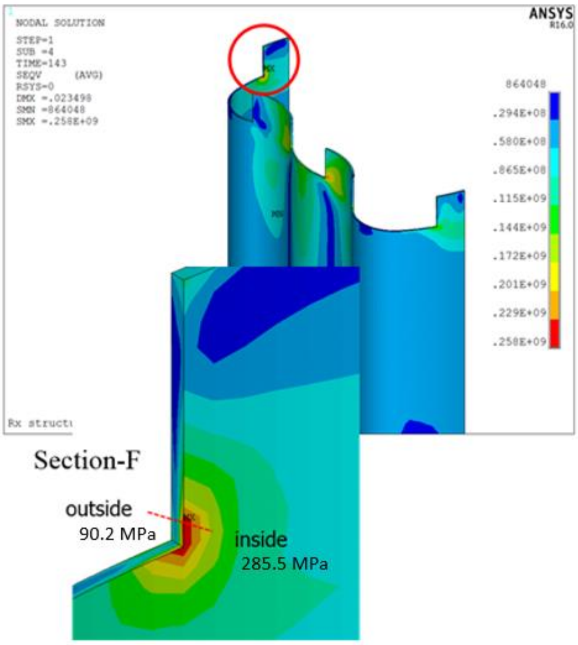

Fig. 6. Thermal-Structural Analysis result of a Redan corner: the figure shows the stress gradient from inside 285.5 MPa to outside 90.2 MPa.

Table II: Stress linearization results (Primary + Secondary loads, Redan corner, Unit: MPa)

	S1	S2	S3	SINT	SEQV	Note
Membrane	8.1	-12.6	-64.6	72.7	64.8	
Membrane +bending	23.1	-9.4	-223	246	231	In
	96.7	-1.3	-24.5	121	111	Out
Peak	9.1	6.1	-1.0	10.1	9.0	In
	0.1	-6.0	-7.8	7.7	7.2	Out
Total	25.0	-2.9	-217	242	229	In
	91.3	-7.3	-26.7	118	110	Out

i) ASME-NH: During the operation, strains due to mechanical and thermal loads are calculated to evaluate creep-fatigue damage. The first step is to obtain total strain range ε_t (0.317 %), which is derived from an equivalent strain range $\Delta\varepsilon_{equiv,i}$ in time at an inspection point i (which is the Redan corner as described). The total strain range is subsequently used to enter a design fatigue curve in ASME-NH [4], and the number N_d of a design allowable life was determined as 8,788 cycles.

To evaluate creep damage, total number of hours at elevated temperatures (as defined in NH), t_H (30 hours in this study), and the hold-time temperature T_{HT} (470℃). From the time-independent isochronous stress-strain curve in [4] corresponding to T_{HT} and the expected minimum stress-to-rupture curve in [4], the allowable time duration T_d was found as 300,000 hours. These findings result in damage factors due to the fatigue and creep (D_{f1} and D_{c1} respectively) according to ASME-NH in the following equations.

$$D_{f1} = \frac{500}{8788} = 0.057 \qquad (2)$$

$$D_{c1} = \frac{30}{300000} = 0.0001 \qquad (3)$$

ii) RCC-MRx: A similar evaluation process is taken using the French high temperature design code, RCC-MRx. The total strain range ε_t was determined for the same investigation point as 0.168% by the sum of the elastic-plastic strain and creep strain range as shown in Eq. (4).

$$\Delta\bar{\varepsilon} = \Delta\bar{\varepsilon}_{el+pl} + \Delta\bar{\varepsilon}_{cr} = 0.168(\%) \qquad (4)$$

$$D_{f2} = \frac{500}{2910806} = 0.000172 \qquad (5)$$

$$D_{c2} = \frac{30}{1573450} = 0.01020 \qquad (6)$$

Subsequently, the allowable cycle number and time duration were determined as 2,910,806 cycles and 1,573,450 hours, respectively. Eqs. (5, 6) show D_{f2} and D_{c2} according to the RCC-MRx code followed from (4). As a result, the RCC-MRx results have longer allowable fatigue cycle but sooner creep damage failure than the ASME-NH.

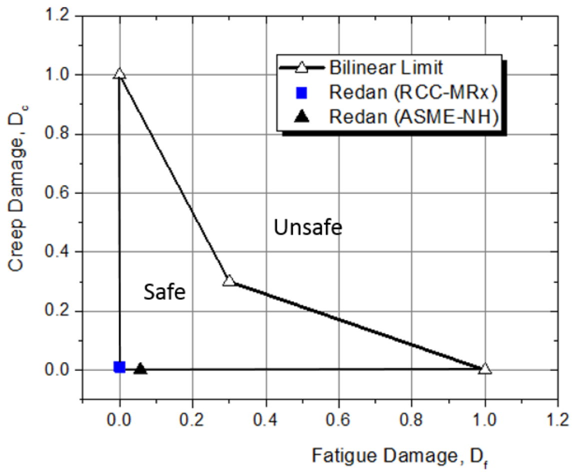

Fig. 7. High Temperature Creep-Fatigue Evaluation of the Redan Corner: ASME-NH appears as conservative compared with RCC-MRx.

The Fig. 7 visualizes two final evaluation points based on two codes in the safety-limit envelope (bold black lines with triangle corners). Two results are all far inside of the safe region. The major difference of two codes is that the RCC-MRx employs peak stress intensities (creep laws are directly used) and the ASME-NH takes strain components from finite element analysis with isochronous curve data. From the damage factor location to the limit, a safety margin can be calculated. Safety factors of two results are turned out to be very close, but it appears that the ASME-NH result is slightly more conservative than the RCC-MRx.

4. Conclusions

A design integrity guarding against a creep-fatigue damage failure operating at high temperature was evaluated for the reactor vessel with its internal structure of the STELLA-2. Both the high temperature design codes were used for the evaluation, and results were compared. All the results showed the vessel as a whole is safely designed at the given operating conditions, while the ASME-NH gives a conservative evaluation. Both the design codes utilize a finite element analysis model to calculate damage factors. A 3D finite element simulation study gave a realistic evaluation.

ACKNOWLEDGEMENTS

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