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Establishment of evaluation method for Integrity of Fuel assembly in PGSFR, and Preliminary analysis

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1. Introduction

Sodium-cooled fast reactor (SFR) design technologies have been developed in Korea since 1997 under a National Nuclear R&D Program to achieve an enhanced safety, an efficient utilization of uranium resources, and a reduction of a high-level waste volume. In 2015, the preliminary specific design of the Prototype Gen-IV Sodium-cooled Fast Reactor (PGSFR) was completed, which is a pool-type SFR with the thermal power of 392.2 MWt and uses metallic fuel of U-10%Zr for a core having inherent reactivity feedback mechanisms and high thermal conductivity. The PGSFR consists of the primary heat transport system (PHTS), the intermediate heat transport system (IHTS), the steam generators (SGs) including balance of plant, and the decay heat removal system (DHRS).

The PGSFR fuel assembly is much harder but thinner than the PWR assembly. It operates under much higher temperature conditions of liquid sodium (>500 ℃, at the fuel assembly exit) than the PWR conditions of water coolant of around 320 ℃. The nuclear fission is caused by fast neutrons rather than a thermal neutrons of the PWR. Therefore, stability must be evaluated and demonstrated through safety analysis under all conditions. For safety analysis, it must be possible to classify and analyze expected transients. Basis accidents are designed that may occur in the fuel assembly. It is predicted that nuclear fuel and system damage. It should be shown that it is operating normally or has sufficient safety margin expected. Further, the temperature range of the PGSFR system is very wide, 400-800 ℃, and it is essential to ensure structural integrity in this range. G. H. Koo and S. K. Kim performed structural design and integrity evaluations for reactor vessel of PGSFR [1]. D. W. Lee and H. Y. Lee performed high temperature structural analysis of test fuel assembly for reactor core sub-channel flow test [2].

However, there are no applicable design criteria to ensure structural integrity. The ASME Section III Division 5 [3] and RCC-MRx [4] codes are easily accessible for applying design criteria for high temperature reactor components. However, since it does not contain the material of the PGSFR fuel assembly. It cannot be used for the mechanical design of the PGSFR fuel assembly. It must present new design criteria for PGSFR fuel assembly. K.H. Yoon and H. k. Kim et al. performed component design and accident analysis of fuel assembly for PGSFR [5]. H. k. Kim, K. H. Yoon, Y. H. lee et al. performed mechanical design of a sodium cooled fast reactor fuel assembly in Korea: Normal operation condition [6].

This study presented the establishment of evaluation method and evaluation criteria for the structural integrity of high temperature fuel assembly in PGSFR.

2. Procedure of assessment integrity

The procedure for thermal stress analysis on fuel rods is shown in Fig. 1. First, the analysis target is modeled in detail. In this step, it is necessary to model a fuel rod containing a wire wrap. Other components are excluded likewise the duct of the fuel assembly, the nose piece the upper and lower reflectors, etc. Second, using MARS code and the computational fluid dynamics (CFD) code analysis, temperature and pressure distribution are derived. Then, the results are verified by comparing two results. It is entered into finite element analysis program to proceed with the structural analysis. ANSYS Mechanical is used to perform finite element analysis. In finite element analysis, boundary conditions are given to structure. Lastly, the structural integrity is evaluated by comparing primary and secondary stresses against the stress limit.

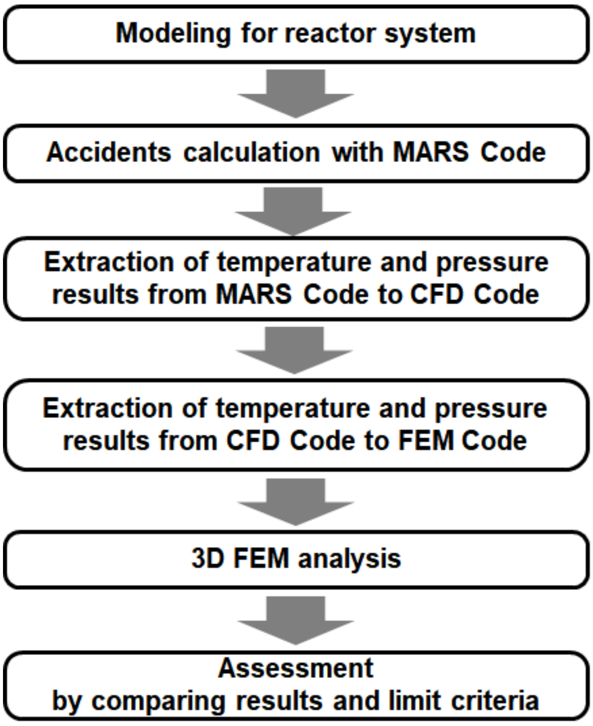

Fig. 1. Flow chart of system related finite analysis

3. Design criteria and stress limits

Since ASME Section III Division 5 was developed for high temperature reactors and their components, it sought to apply design methods and standards early on. However, there was a problem that it did not provide the material properties of HT-9. Alternatively, the RCC-MRx code was investigated and it was developed for the design of high temperature reactor components. However, this also cannot be used with the same problem as ASME code. Therefore, new standards for PGSFR were proposed with references developed specifically for SFR [6].

The stress limit consists of the yield or ultimate strength of the structural material at the operating temperature, which is multiplied by several numerical factors for safety margins. The evaluated stresses are linearized into primary membrane stress (P_m), primary bending stress (P_b), and secondary stresses (Q) in the order P_m, P_b, Q. After all, similar to ASME code section III Division 1, structural integrity can be guaranteed if P_m, ($P_m + P_b$), and ($P_m + P_b + Q$) are less than the stress limit. Table I summarizes the newly constructed stress limits. This is the criteria that we have selected the most conservative criteria from the references [7, 8]. The more you go from Level A to D, the less likely it is to occur.

Table I: The stress limits for each category events for PGSFR fuel assembly mechanical design

Level	P_m	$P_m + P_b$	$P_m + P_b + Q$
A	$0.55\ \sigma_u$		$0.6\ \sigma_u$
B	$0.6\ \sigma_u$		$0.6\ \sigma_u$
C	$0.75\ \sigma_u$		$0.8\ \sigma_u$
D	$0.9\ \sigma_u$		$0.9\ \sigma_u$

Table I shows that only extreme strength was used for the stress limit. This occurs when the yield and extreme intensity of the HT-9 are multiplied by the specific factors provided in the bibliography. We used the relevant formulae of σ_y and σ_u of HT-9 [9], which are given as Eqs (1) and (2) below [6]. The calculated values of σ_y and σ_u of HT-9 were verified through a comparison with the experimental data provided by the ANL [7], which is given in Fig. 2.

• Formula of the yield strength , σ_y in MPa

$$\ln\left(\frac{\sigma^* F}{\sigma_y}\right) = A\left[\frac{\left(\frac{\sigma^*}{G}\right)^M}{\dot{\varepsilon}\exp\left(\frac{Q}{RT_k}\right)}\right]^\lambda \qquad (1)$$

$where\ F = B\sqrt{1 - 0.5\tanh\left(\frac{T_i - D}{90}\right)} + C$, and F=1 for unirradiated condition. For the other parameters, A=20,000, M=5, λ=0.172, Q=82,000, R=1.98726, G= 89,640-53.78T_i (the shear modulus in MPa), $\sigma^* = 430 - 190\tanh(\frac{T_i - 640}{225})$, B= 1-0.02 ϕ_t, C=0.02, D=425+10 ϕ_t, $T_i = T_k - 273$,$T_i = T_l - 273$, whence T_l is the irradiation temperature in Kelvin, ϕ_t is the first neutron flux ($< 11 \times 10^{22} n/cm^2$), and $\dot{\varepsilon}$ is the strain arte (s^{-1}, $10^{-5}s^{-1} \ll 10^{-2}s^{-1}$).

• Formula of the ultimate strength, σ_u in MPa

$$\sigma_u = \sigma_y[1.1 - 0.1\tanh\left(\frac{\Delta T - 200}{200}\right)] \qquad (2)$$

where $\Delta T = T_k - T_l\ and\ \Delta T = T_k - 848.$

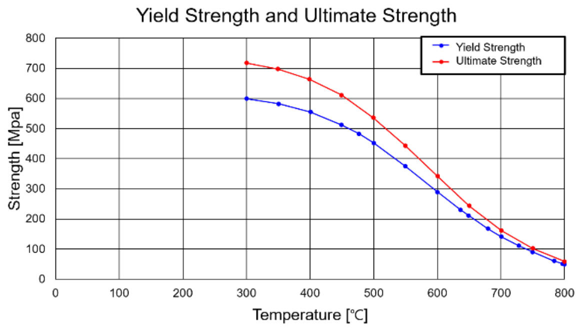

Fig. 2. Calculated yield and ultimate strength of HT-9

4. Analysis for normal operation conditions

4.1 Design of analysis target

A fuel rod is composed of a wire-wrapped cladding tube in which a U-10Zr fuel slug with a Na bonding material is initially installed. Later, the fissile material will include TRU in addition to U-Zr. The fuel rod material, developed at KAERI, was HT-9. The actual PGSFR nuclear fuel assembly consists of 217 pin fuel rods. But in this study, the structural evaluation methodology was applied to the 7 pin fuel rods. Some of the key design parameters of the 7 pin fuel rods are provided in Table II [10].

Table II: Geometric parameters of 7-pin fuel assembly

Geometric parameters	Values
Number of fuel fins	7
Pin diameter	6.629 mm
Pin pitch	7.9 mm
Pitch to diameter ratio	1.192
Pin axial length	1317 mm
Heated length	450 mm
Heat flux distribution	Uniform
Cladding thickness	0.55 mm
Duct inner flat to flat distance	23.6 mm
Coolant	Sodium

4.2 Preparation of inputs

The fuel rods are covered with a long hexagonal duct with a nominal thickness of 3 mm and are connected to the upper handling socket and lower nosepiece. The fuel rod's lower end-caps are mounted in the mounting rails, which are vertically held within the hexagonal duct. Fig. 3 showed the fuel rod's lower end-caps and mounting rails. Therefore, it is assumed that the lower end of the fuel rod is a fixed surface. Also, since the upper end of the fuel rod is connected to the fuel expansion gap, it is assumed to be a free surface instead of a fixed surface. Fig. 4 showed upper side of fuel rods. Since the fuel rod and wire wrap do not contact, it is assumed that there is no contact force.

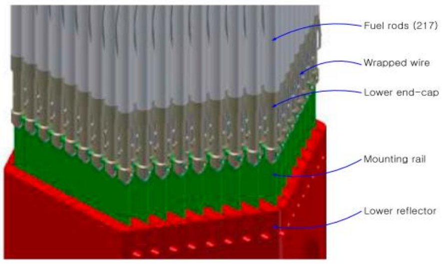

Fig. 3. Fuel rod bundle with the mounting rail and lower reflector [5]

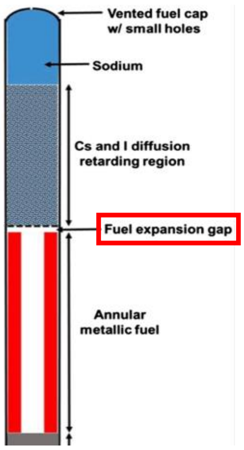

Fig. 4. Fuel rod design [11]

It is necessary to have the temperature and pressure distribution data along the fuel assembly height to pick up the largest stress value. The distribution from MARS Code and CFD code are used to perform the finite element analysis

4.3 Assessment

The stresses at each elevation were linearized to obtain $\mathrm{P_m}$, $\mathrm{P_b}$, and Q. The maximum values of $\mathrm{P_m}$, $(\mathrm{P_m} + \mathrm{P_b})$, and $(\mathrm{P_m} + \mathrm{P_b} + Q)$ at elevation were then calculated and compared with the stress limits of the non-operation and normal operation conditions, which were obtained for the maximum temperature at elevation.
Fig.5 and Fig. 6 illustrates the maximum values of $\mathrm{P_m}$, $(\mathrm{P_m} + \mathrm{P_b})$, and $(\mathrm{P_m} + \mathrm{P_b} + Q)$, and 0.55 σ_u and 0.6 σ_u corresponding to the maximum temperature at elevation of 7 pin fuel rods. The maximum $(\mathrm{P_m} + \mathrm{P_b} + Q)$ of 7 pin fuel assembly is 68 MPa. These are far below the minimum stress limit, i.e., 171 MPa. A sufficient margin is found between the data of 0.6 σ_u and evaluated stresses for both fuel assemblies. Therefore, it is soundly concluded that the structural integrity of the PFSFR fuel assembly components is guaranteed during normal operation from a strength standpoint.

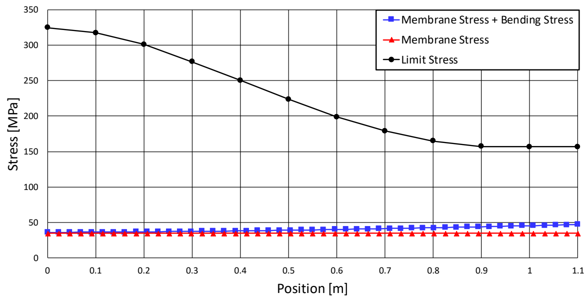

Fig. 5. $\mathrm{P_m}$, $(\mathrm{P_m} + \mathrm{P_b})$, and Limit Stress(0.55 σ_u) at position

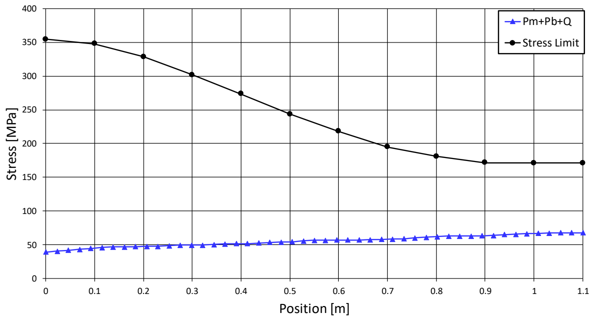

Fig. 6. $(\mathrm{P_m} + \mathrm{P_b} + Q)$ and Limit Stress(0.6 σ_u) at position

5. Conclusion

In this study, the structural integrity was evaluated in consideration of the geometrical shape and dynamic flow conditions of the nuclear fuel assembly. HT9 was used as the quality of the aggregate in which the methodology for structural integrity evaluation was constructed and the structural integrity was evaluated by applying the methodology to the 7 pin fuel bundle. In addition, sufficient stability was proved by conservatively setting and evaluating stress limit standards. The maximum stress strength when a machine and a thermal load is indicated by 68 MPa, which is a sufficient difference from the stress limit standard 171 MPa. Health assessment of strain limits under high temperature conditions is currently underway. Using this methodology, it is going to evaluate the structural integrity of the 217-pin fuel bundle.

ACKNOWLEDGMENTS

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