FEASIBILITY STUDY FOR DESIGN OF FUEL ROD IN JSCWR

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Abstract

On the basis of the core and fuel design, as well as for safety analyses and material development established in the JSCWR, the thermo-mechanical behavior of typical fuel rods was evaluated to study the feasibility for design of fuel rods. Analyses showed that stress, cumulative creep damage on the cladding and fuel centerline temperature were able to be within the respective design limits. It was found that the long term cumulative creep damage effect in normal operating played an important role in the feasibility evaluation because of high temperature of the cladding and stress from large pressure difference between inner and outer of the fuel.

1. Introduction

The Supercritical Water-Cooled Reactor (SCWR) is a high-temperature, high-pressure water-cooled reactor that operates above the thermodynamic critical point of water (374°C, 22.1 MPa), which enables combination of a once-through reactor and a direct cycle system.

Under the financial support of Ministry of Economy, Trade and Industry (METI) of Japan, several R & D activities leading to the Japanese Supercritical Water-Cooled Reactor (JSCWR) development were carried out. The Generation IV International Forum (GIF) collaboration follows the efforts for material development for fuel cladding in SCWR [1, 2]; several design studies of the SCWR core, fuel and safety system to meet requirement of higher economical advantage and higher reliability [1, 3, 4]; and evaluation of thermo-mechanical behavior of fuel rod to investigate its feasibility and to specify material requirements [5]. Through these studies, the feasibility of design of the fuel concept has been discussed and confirmed [6, 7].

As described by Yamada et al. [8], the reactor core of the JSCWR is operated at 25.0MPa. The feed water temperature is 290°C, and the average core outlet coolant temperature is 510°C. Both the pressure and temperature are much higher than those of current light water reactors (LWRs). But the basic concept of the fuel shape and its configuration in the reactor core is quite similar to the LWRs. The fuel rod contains UO_2 pellets like the LWR fuel which are loaded in a long and thin cladding tube made of a stainless steel. Since the conditions of pressure and temperature are so severe for the fuel, the thermo-mechanical behavior of the fuel rod must be a key factor for feasibility of the core. In this paper, based on the latest core and fuel design of the JSCWR, the thermo-mechanical behavior of fuel rods is evaluated to study the feasibility for the design concept.

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2. Thermo-mechanical design criteria of fuel rod

In previous studies for thermo-mechanical behaviors of SCWR fuel rod [5, 10, 11], one can see a basic feature, i.e. the cladding suffered from heavy compressive stress in its early stage due to the system pressure as high as 25MPa. To compensate the system pressure, a pre-pressure of helium (He) gas in the fuel has to be introduced.

The fuel cladding shall keep its mechanical integrity in both normal operating and anticipated operational occurrences (abnormal transients) in accordance with the fuel design criteria of LWRs and Fast Breeder Reactors (FBRs). From general considerations for fuel rod behavior in SCWR from the previous studies [5, 9, 10], the following should be taken into account to the failure mechanism at least.

(a) Buckling collapse
(b) Stress rupture
(c) Excessive deformation
(d) Creep damage

A design criterion must be that the cladding geometry should be able to avoid (a) buckling collapse. The design in elastic zone is introduced to satisfy (b) and (c) as a criterion in this study. For the mechanisms (c) and (d), cumulative creep damage may be a good index as a criterion. Cumulative damage by creep is evaluated by the cumulative creep damage fraction (CDF) defined by Eq. (1) [11].

$$CDF = \sum_k \left(\frac{\Delta t}{t_d} \right)_k \tag{1}$$

where Δt is the time interval that the cladding stays at k^{th} condition of temperature and stress on the cladding; and t_d is the allowable time duration determined from the stress-to-rupture curve of the cladding material for a given stress and temperature. Since there is no consideration of fatigue in the study because of no detailed cyclic pattern evaluated, CDF should be sufficiently smaller than 0.3, which is the limitation from the view point of creep-fatigue damage [12]. Furthermore, the fuel pellet centerline temperature is better to be under the pellet melting point because the fuel keeps its figure.

Consequently, fuel rod design criteria are set as follows.

(1) Thickness of cladding tube shall be sufficient to avoid buckling collapse.
(2) Von Mises equivalent stress in cladding shall be lower than the yield strength (S_y) of the cladding material in normal operation and abnormal transient.
(3) Cumulative creep damage fraction (CDF) based on the cladding material shall be smaller than 0.3 in normal operation and abnormal transient.
(4) Pellet center line temperature shall be lower than the melting point of fuel with effect of burnup in normal operation and abnormal transient.

(5) In the case of over inner pressure by fission product (FP) gas release with burnup, out going creep displacement shall be prevented so that the pellet-cladding gap can keep its integrity from thermal feedback.

Needless to say, as the criteria were derived from scarce knowledge by theoretical speculations, these should be modified in the future by feedback from more detail studies of JSCWR fuel behavior including experimental works.

3. Calculation procedure

3.1 Calculation method

The thermo-mechanical behavior of a fuel rod was studied by using a computer program for LWR fuel rod analysis, FEMAXI-6, which was developed by Japan Atomic Energy Agency [13]. Austenitic stainless steel, stainless steel type 304 (SS304), is one of the materials for the cladding treated in the code as well as Zircalloy.

The fuel pellet was divided into 10 rings. The fuel active length was axially divided into 24 nodes (node 1 for the bottom and node 24 for the top in the core); and upper and lower plenums were attached. In the JSCWR the main plenum is the lower one because the lower part of the core is at lower temperature, so that the inner pressure of the fuel due to thermal expansion of gas in the plenum is lower.

The study was conducted as follows [5]. The diameter and thickness of the cladding were firstly confirmed to avoid buckling collapse. The design is going with an idea in which any plastic strain cannot be allowed. Initial helium pressure, pellet-cladding gap and gas plenum volume should be tuned up to make a proper inner pressure during the fuel life. If the pressure cannot be kept lower than the system pressure, out-going creep deformation and an increase of the pellet-cladding gap should be avoided. Otherwise a thermal feedback could be caused, resulting in a higher temperature in pellet and higher fission product gas release (FGR).

3.2 Calculation conditions

3.2.1 Fuel rod specification

The specification of a fuel rod in the core is shown in Table 1. The listed values for the cladding were determined from pre-analysis considering the general behavior and the study of flow described in the previous sections.

The material for the fuel cladding is still under development. A material that has yield stress and creep strength of austenitic stainless steel type 316 (SS316) has shown to be feasible to become a candidate for a SCWR fuel cladding in the previous study [5, 7]. However, SS316 showed relatively larger corrosion in SCW condition in studies of a previous Japanese project [2, 14]. Meanwhile modified stainless steel type 310S (SS310S) with Zr doping seems to have preferable properties in general corrosion, stress corrosion cracking, yield strength, irradiation stability and manufacturing of long tube [14]. The data base of the material is scarce compared to commercial base materials in this time. Therefore, in this study, temperature and stress were tentatively calculated on FEMAXI-6 model base, SS304, and some important properties of the modified SS310S were taken into account for the evaluation of design limit as described in the next subsections.

Table 1 Fuel rod specifications for thermo-mechanical analysis

Item	Value	Comment
Cladding		
outer diameter	7.0mm	
Thickness	0.44mm	
Material	FEMAXI-6 model*	or modified SS310S *
Pellet		
outer diameter	5.9mm	
Fuel active length	4200mm	
Plenum volume ratio to fuel volume	0.08 to 0.16	parameter
Initial He pressure	7.5MPa to 8.5MPa	parameter

*Tentative; mechanical properties in FEMAXI-6 is based on SS304.

3.2.2 Cladding material modelling

The 0.2% proof stress of the candidate modified SS310S, as the yield strength, is shown in Fig. 3. The temperature dependency of SS316 [12] was renormalized to the modified SS310S data at 550°C, because of scarce knowledge of the temperature dependency of the material. The yield strength is significantly higher than SS316.

The creep rupture time is modeled as shown in Fig. 4. This model was made from measured data at 700°C and 800°C with Larson-Miller parameter fitting.

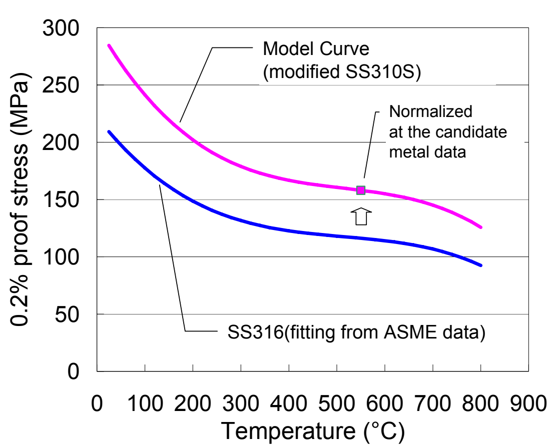

Figure 3 Yield strength (0.2%proof stress) model for the modified SS310S

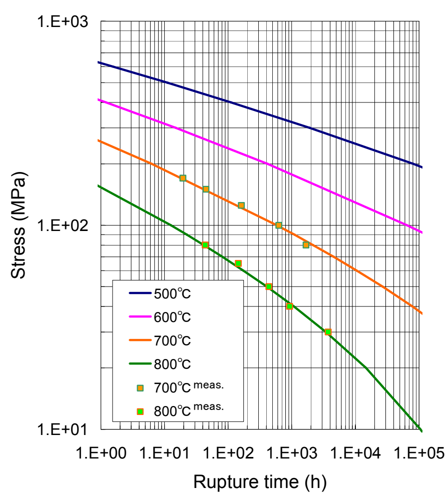

Figure 4 Creep rupture model of the modified SS310S

A candidate material in Ref. [14] has been evaluated as reduction of 5.9μm/1000h at 700°C by general corrosion. A simple calculation using these data has been performed to include this effect, as will be described later.

3.2.3 Power and cladding temperature history in normal operation

Based on the core and fuel design [9], linear heat generation rate (LHGR), fast neutron flux and cladding surface temperature (CST) were determined as input conditions for the evaluation of thermo-mechanical behavior of the fuel rods as functions of time and axial position. The peak value of LHGR is 42kW/m, fast neutron flux 2.3×10^{18}n/m^2/s (proportional to LHGR) and cladding surface temperature 610.7°C in nominal among all fuel rods in the core. The residence time is 21454h (894 days).

From the time dependent data of the all fuel rods in the core, typical fuel rods were selected as fuels that should be analyzed to meet the design criteria. Considering thermo-mechanical behavior of SCWR fuel rods, there are two important physical values to prevent the fuel rods from failure. One is the CDF and the other one is the stress of the cladding. CST is expected as high as 700°C and the material properties are steeply degrading with increasing temperature, especially creep strength. LHGR affects fuel pellet temperature and thus internal gas pressure, pellet-clad mechanical interaction (PCMI) and/or temperature difference between inner and outer surface of cladding. Hence high LHGR causes high cladding stress. The two typical fuel rods were, therefore, selected as those that have experienced the maximum CST rod (MCST-rod for short) and the maximum LHGR rod (MLHGR-rod) in the whole core.

For the MCST-rod, nominal CST and LHGR as a function of time are shown in Figs. 5 and 6, respectively. The position and time are not coinciding in the peak of CST (node19) and LHGR (node11) in this case. For the MLHGR-rod, Figs. 7 and 8 show CST and LHGR as a function of time, respectively. The axial node and time where and when the highest CST appeared are not in coincident with those for higher LHGR.

The MCST during the entire life of the fuel rods in the whole core is 610.7°C in nominal case. Considering the statistical effect from engineering uncertainties, the MCST becomes 696.5°C within 99.99% probability with 95% upper confidential level. The analysis of fuel thermo-mechanical behavior was, therefore, carried out in a way in which the difference of the two temperatures (85.8°C) was added to the nominal CST history during the entire life of the MCST-rod. In the core, 42.2kW/m in LHGR was the highest value. For this fuel rod, called as MLHGR-rod, the statistical effect concerning the CST was also added, like for the MCST-rod.

The MCST-rod is treated as the most challenging fuel rod in the following discussion from consideration of licensing procedure in which one typical fuel rod should be analyzed. As discussed below, creep damage is the essential to be discussed for the fuel integrity, thus the MCST-rod should be a candidate. In this study, fuel behavior is evaluated with the statistical effect from engineering uncertainties for the CST as described above. According to the core design [9], approximately one fuel rod would have the highest temperature at a certain time on a position during the whole plant life. This statistical treatment, therefore, can be significantly conservative, and hence the evaluation of the MCST-rod should be appropriate to discuss the fuel rod feasibility.

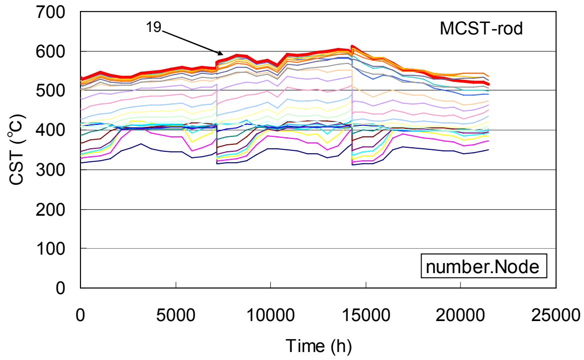

Figure 5 Nominal CST of MCST-rod

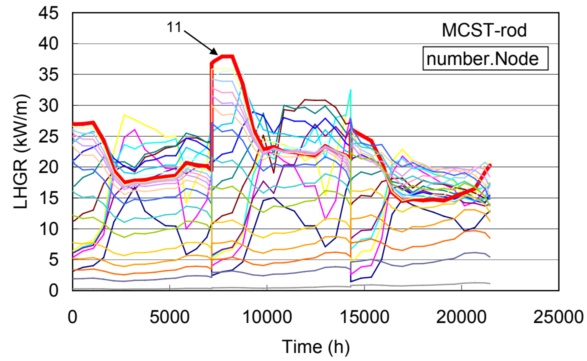

Figure 6 LHGR of MCST-rod

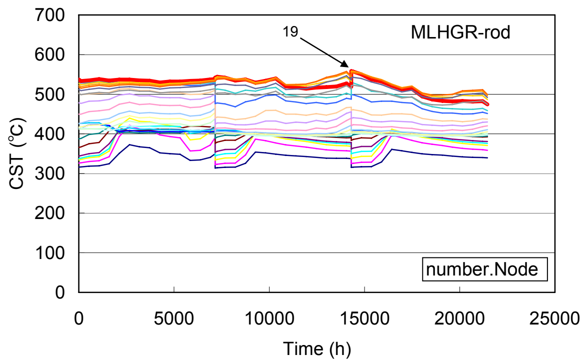

Figure 7 Nominal CST of MLHGR-rod

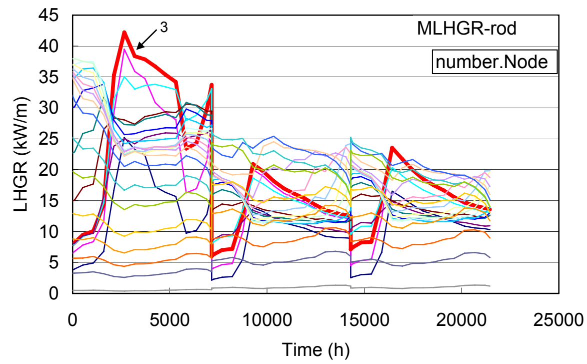

Figure 8 LHGR of MLHGR-rod

3.2.4 Transient condition

Conditions in transient analysis for fuel rod integrity were extracted from the safety analysis report for the JSCWR system in Ref. [15]. Among these events, the largest impact on fuel rod integrity would happen in the event of uncontrolled CR withdrawal at normal operation, due to its relatively large and simultaneous increasing of CST (ΔCST: CST change from the beginning of transient) and LHGR as shown in Fig. 9.

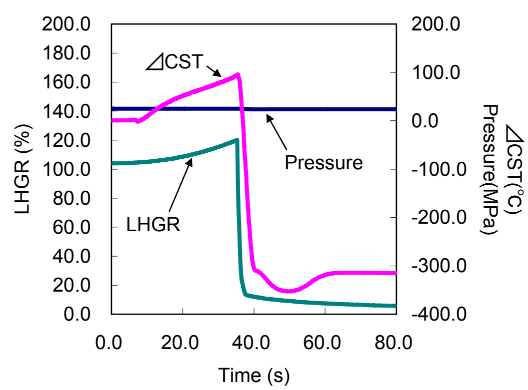

Figure 9 ΔCST, LHGR and Pressure change in transient of
'Uncontrolled CR withdrawal at normal operation'

In this event, main parameter changes are as follows. These changes were added to the normal operation history of the rods at appropriate time points.

- ΔCST: c.a. +100K
- LHGR: 115%; Furthermore, a factor 1.04 of local effect of control rod which is adopted from consideration of neutronics calculations
- Pressure: +0.3MPa
- Period: c.a. 35sec until scram

4. Results and discussions

4.1 Behavior in normal operation

In the following case, an initial pressure inside the fuel rod P_{ini} of 8.5MPa and plenum volume ratio to fuel (V_p/V_f) of 0.16 were firstly specified. The equivalent stress (σ_{eq}), CDF and pellet centerline temperature (T_C) are shown in Figs. 10, 11 and 12, respectively. These figures are shown for certain axial nodes in which the maximum value appeared in respective evaluated items. All values are evidently below the design allowable limits. For the stress, it also seems that SS316 is feasible if the allowable stress is taken from Fig. 3 and when considering that the CST is less than 700°C.

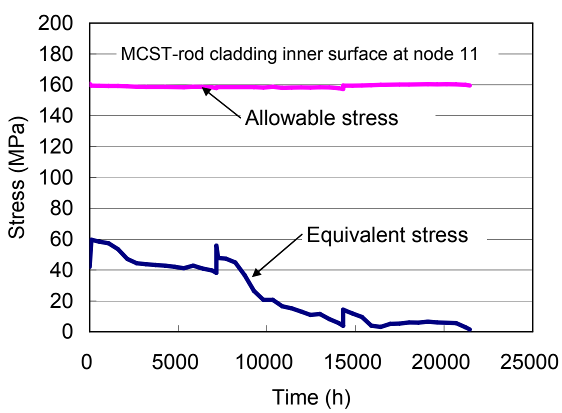

Figure 10 Equivalent stress at peak node ($MCST$-rod;P_{ini}=8.5 MPa, V_p/V_f= 0.16)

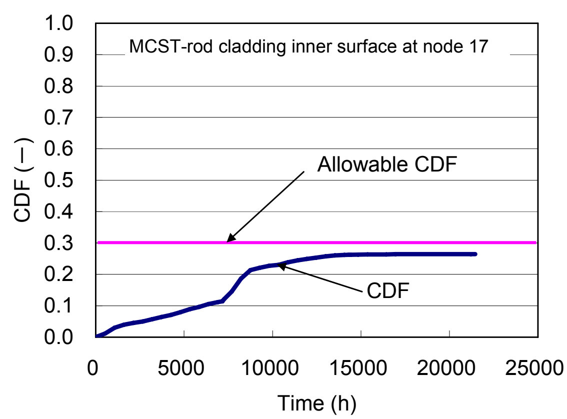

Figure 11 CDF at peak node (MCST-rod;P_{ini}=8.5 MPa, V_p/V_f= 0.16)

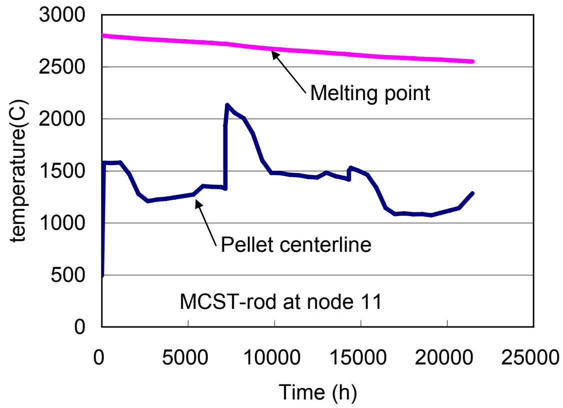

Figure 12 Fuel temperature at peak node
(MCST-rod;P_{ini}=8.5 MPa, V_p/V_f= 0.16)

As shown in Figs 5 and 6, MCST appeared in node 19 and MLHGR in node 11. Figures 10 and 12 show that both the peak of equivalent stress and the peak of fuel centerline temperature coincided with the MLHGR point (node 11). Meanwhile, the maximum CDF did not appear in the MCST point (node 19), but appeared in node 17 instead. The maximum CDF was recorded for the inner surface of the node. There are the facts that the temperature at the inner surface is generally higher than that at the outer one due to generation of heat inside the cladding. The LHGR in the node 17 was higher than that in the node 19. The two facts caused higher inner surface temperature in the node 17, hence the higher CDF than the one in the node 19.

The maximum temperature at the outer and inner surface of the cladding did not coincide at the same node in this case. But the MCST-rod can be expected to have a high temperature compared with the outer surface of the other rods and hence the inner surface temperature can be expected as the highest in the other node in the rod. It is proven by the result that the node 11 had only very low CDF despite that the node recorded the highest temperature inside the cladding by the highest LHGR, because the inner surface temperature was suppressed in the node due to suppressed CST. Nonetheless, the representativeness of the MCST- and MLHGR-rod is so important that further studies should be necessary.

The inner pressure of this fuel case becomes higher than the system pressure, 25MPa, at the end of the 2nd cycle as shown in Fig. 13. This over pressure may change the cladding creep behavior from in-going to out-going. If the creep out-going rate is high enough to stimulate a ‘thermal feedback’ as mentioned in previous subsection, the cladding may desperately be deformed. Figure 14 shows the gap and creep displacement as a function of time. The cladding creep causes an inward deformation until 15000h and then keeps a certain value to the end of life, but it never goes outward at the node 12, at half axial position. The creep behaviors in the other axial positions have a same tendency. There were some creep changes after around 15000h in some axial nodes, but the changing rate was as small as the order of 1×10^{-6}μm/h. Therefore, it does not cause violation of the minimum gap and the fuel design for the core can be considered to be feasible. Nevertheless, the overpressure effect should be also confirmed by tests.

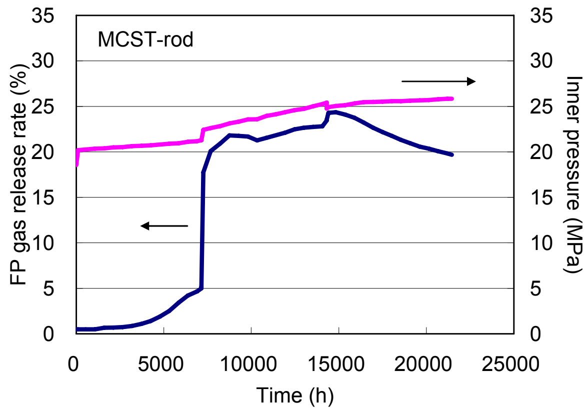

Figure 13 FGR and inner pressure
(MCST-rod;P_{ini}=8.5 MPa, V_p/V_f= 0.16)

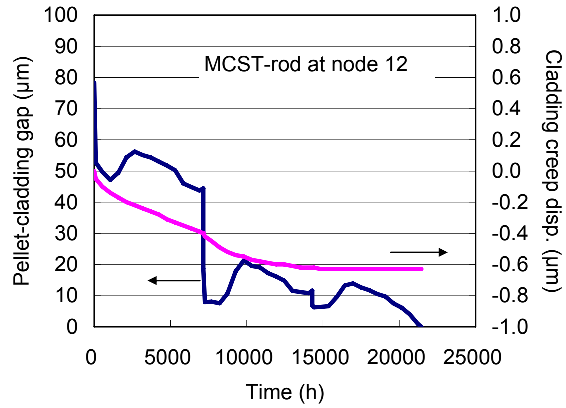

Figure 14 Gap and creep displacement
(MCST-rod;P_{ini}=8.5 MPa, V_p/V_f= 0.16)

In case of the smaller plenum volume specified by P_{ini}=8.5MPa and V_p/V_f=0.08, it was found that CDF was within the allowable level. The smaller V_p/V_f ensured the smaller CDF by fission gas release (FGR) even in the earlier stage of the life. Furthermore for the lower initial helium pressure case, P_{ini}=7.5MPa and V_p/V_f=0.08, one observes also a similar tendency in CDF. CDF increased at beginning of the 2nd cycle, but the smaller V_p/V_f made a larger inner pressure that causes the compressive stress to be lower and then the CDF keeps at a lower level until the end of life. The larger V_p/V_f could not make sufficient inner pressure to compensate the system pressure by FGR in the earlier stage of the life. The small plenum volume, therefore, is needed in the case of P_{ini}=7.5MPa. In both cases mentioned above, there are concerns about overpressure effect because either case makes actually larger inner pressure by FGR due to the smaller plenum volume, thus those have overpressure in later stage of the life. The overpressure effect will be more severe than the first design of the fuel rod.

For the MLHGR-rod, the statistical effect concerning the CST was also added, like for the MCST-rod. In accordance with the results, it was found that σ_{eq}, CDF and T_C were able to be kept lower than the allowable levels with the same specification as the MCST-rod.

Consequently, the initial helium pressure of 8.5MPa and a certain volume of fission gas plenum are needed. If manufacturability of this quite highly pressurized fuel rod can be ensured, this fuel rod will be feasible. For the initial pressure of 7.5MPa, it would be easier to manufacture but overpressure effects should be tested.

4.2 Behavior in abnormal transient

A transient condition was assumed from results of a safety analysis of the plant [15]. An abnormal control rod withdrawal during rated power operation was considered as the most severe event for the fuel rod as described in Ref. [7].

Calculations were carried out based on an assumption that the transient occurred once in a whole life at the maximum point of either CST or LHGR in a certain cycle. The fuel is assumed to be a feasible design case, namely P_{ini} = 8.5MPa and V_p/V_f = 0.16. The stress, CDF and pellet centerline temperature for MCST-rod and MLHGR-rod are listed as ratios to the design limits in Table 2.

Table 2 Evaluated ratios to the design limits at transient of abnormal CR withdrawal during rated power operation

		ratio to design limit					
		1st cycle		2nd cycle		3rd cycle	
		Max. CST	Max. LHGR	Max. CST	Max. LHGR	Max. CST	Max. LHGR
MCST-rod	stress	0.374	0.443	0.374	0.374	0.374	0.374
	CDF	0.351	0.166	0.868	0.621	0.875	0.875
	centerline temp.	0.600	0.600	0.780	0.800	0.780	0.780
MLHGR-rod	stress	0.416	0.518	0.416	0.416	0.416	0.416
	CDF	0.322	0.261	0.328	0.322	0.328	0.328
	centerline temp.	0.790	0.800	0.790	0.790	0.790	0.790

It is found clearly that all values are within the design limits. Because the transient time is as short as 35s, it has no significant impact on the creep damage. And this design of fuel is not expected to cause PCMI, thus a large stress is not expected with increasing LHGR. It is concluded that the cladding keeps its mechanical integrity even in transient conditions.

4.3 Effects of variation factors

There can be different conditions from those in the previous discussions. Calculations were performed only for the MCST-rod in normal operation because the condition was considered as a representative to discuss the integrity of the fuel rods. The variations to be investigated were selected such that long term effects can be evaluated, namely a reduction of wall thickness of the cladding by general corrosion and FP gas release. The fuel is selected according to the feasible design, namely P_{ini} = 8.5MPa and V_p/V_f = 0.16.

As long term general corrosion tests are still on-going and any corrosion model has not yet been established, a simple assumption is set here from data in Ref. [14]. A candidate material has been evaluated with a reduction of wall thickness of 5.9μm/1000h at 700°C. As the time dependency is not clear now, a linear time dependence has been assumed neglecting the actual temperature of the cladding for a conservative evaluation. Thus 127μm in reduction of the wall thickness after multiplied resident time, 21454h, is set for the calculation. Moreover, this reduction is adopted from the beginning for simplicity. Thinner walls caused higher stress but the change is not so large. This effect appeared in CDF to make it larger, due to the cumulative phenomenon, reaching up to 0.29, just under the limit. A large out-going creep was not observed. Remembering that the wall thickness was already reduced from the initial condition, the fuel design is again considered to be feasible, even with a margin. More studies with a corrosion model with concrete wall thickness reduction data by long term general corrosion tests should be necessary.

The FP gas released inside the fuel rod works to compensate the initial difference between the inner and the system pressure. The time dependency of the FP gas release rate depends on the evaluation model that is based on a well known thermally activated diffusion mechanism of FP gas atom in a grain of the fuel pellet crystal with a diffusion coefficient (D) [13]. The model parameters were tested to find the effects on the fuel behavior by the following three cases.

(1) Larger diffusion coefficient (3×D): larger FGR in a whole life would cause lower stress in early life of the fuel but higher stress in late life.
(2) Smaller diffusion coefficient (0.5×D): smaller FGR in a whole life would cause higher stress in earlier period of the fuel but lower stress in later period.
(3) Larger open porosity percentage (50%) and smaller diffusion coefficient (0.8×D): Larger FGR in early life but little bit lower inner pressure at the end of life.

For (1) larger FGR case, the stress was not notably changed but became lower in early life of the fuel and higher in late life. CDF became lower. The inner pressure exceeded the system pressure. Although the gap was still stable, the out-going creep became larger.

For (2) smaller FGR case, CDF exceeded the limit. The inner pressure was a little higher than the system pressure in this case.

For (3) mixed FGR case, CDF was not notably changed although FGR was recorded larger in the earlier period. It is reasonable because the FP gas contained in the pellet was small in the earlier period and the release rate itself was not so large, to cause a large effect on the inner pressure.

By these parameter surveys of FGR, it is found that a certain amount of FGR is necessary to compensate the system pressure from the view point of CDF. But overpressure from a large amount of FGR causes out-going creep of cladding. Therefore, the FGR behavior at this temperature condition and overpressure effects should be examined carefully in the future work.

5. Conclusion

Analyses by a computer code for the fuel rod behavior showed that stress, cumulative creep damage on the cladding and fuel centerline temperature were able to be within the respective design limits.

It was found that the long term cumulative creep damage effect in normal operating conditions played an important role in the feasibility evaluation. In order to compensate the pressure difference, not only a large initial gas pressure but also a certain amount of fission product gas release was needed. This has lead to a larger pressure at the end of fuel life and caused overpressure in the fuel. In addition, general corrosion may cause the creep damage to be larger via reduction of cladding wall thickness as a long term effect.

It is concluded also that there are further studies needed in the future as follows:

For the reliability of the analysis methods of fuel behavior.

- Study of detailed fission gas release
- Study of detailed overpressure behavior
- Appropriate modeling of the above properties into a calculation code and its validation

For the reliability of the design procedure.

- Confirmation of representativeness of the selected fuel rods in the licensing procedure
- Well-developed data base of the cladding material, especially long term properties like creep and general corrosion
- Validation of the integral design method by a Fuel Qualification Test program

6. References

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