Development of core and structural materials for fast reactors

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Abstract. This paper summarizes ongoing efforts in Japan Atomic Energy Agency on the development of core and structural materials for sodium-cooled fast reactors. For core materials, oxide dispersion strengthened (ODS) steels and 11Cr ferritic steel (PNC-FMS) will be used for the fuel pin cladding and wrapper tube, respectively. As for ODS steel, 9Cr- and 11Cr-ODS steels have been extensively developed. Their laboratory-scale manufacturing technology has been developed including reliability improvement in tube microstructure and strength homogeneity. Large-scale manufacturing technology development and mechanical testing for codification of material strength standard are on-going. As for the PNC-FMS wrapper tube, the development of a dissimilar joining technique with type 316 steel and properties evaluation of dissimilar welds have been carried out. For structural materials, codification of 316FR stainless steel and Modified 9Cr-1Mo steel is ongoing. Acquisition and collection of long-term data of base metal and welded joints are continued and evaluation methodologies are being developed to establish a technical basis for 60-year design.

Key Words: ODS steel cladding tube, PNC-FMS wrapper tube, Mod.9Cr-1Mo steel, 316FR steel.

1. Introduction

For the development of fast reactors with higher safety and reliability with cost-effectiveness, the development of high performance materials is of significance. Japan Atomic Energy Agency (JAEA) has conducted basic research on core materials and structural materials and identified promising materials for next generation fast reactors. This paper describes the latest development of the materials [1, 2].

JAEA has developed oxide dispersion strengthened (ODS) steels for fast reactor (FR) long life fuel cladding tube[2-12], the use of which can be conducive to volume and hazardousness reduction of radioactive waste. The ODS steel is noticed as an advanced alloy having a highly swelling-resistant ferritic/martensitic matrix, in which nano-sized oxide particles are dispersed to improve high-temperature strength, thus achieving a beneficial combination of radiation resistance and high temperature strength [2, 3]. In the designing of the JSFR (Japan sodium-cooled fast reactor) [13], ODS steel is a candidate cladding tube material, which can resist to the temperature (midwall hot spot temperature) of 973K and peak neutron dose of 250dpa in normal operation. The modified type 316 steel cladding tube [14] having plenty of industrial experience has already been applied in fast reactor core applications worldwide. However the use of this kind of cladding tube limits the fuel burn-up level due to the onset of void swelling by high-dose neutron irradiation. Thus, several research institutes have tried to develop ODS steels for use as long life core material in generation IV nuclear fission reactors and fusion reactor [3, 15-18]. A technologically crucial problem for practical use of ODS steel cladding tube is lack of adequate industry experience; the important tasks towards the practical realization of ODS steel cladding tube should be fabrication technology development from laboratory scale to mass production scale, and accumulation of in-pile and out-of-pile mechanical properties data for establishment of a material strength standard. The operation temperature of wrapper tube material is assumed to be lower than 843K in normal operation. Thus, the 11Cr high strength ferritic-martensitic steel (PNC-FMS) [19] produced by conventional melt processes with satisfactory industrial experience can be used as wrapper tube material applied in a high-burn up fuel assembly. Since application of type 316 steel to entrance nozzle and handling head is assumed, joining technology between PNC-FMS and SUS316 tubes is a developmental task. The candidate joining technologies are dissimilar fusion welding process using electron beam (EB) welding, and mechanical joint technique. Therefore, there are two important tasks in PNC-FMS wrapper tube development: dissimilar joining technology development, and accumulation of in-pile and out-of-pile mechanical properties data for establishment of material strength standard [2]. This paper reviewed the current status on development of fast reactor long life core materials (i.e ODS steel cladding tube and PNC-FMS wrapper tube) in JAEA.

For the reactor vessel and internal structures of fast reactors, the most promising material is 316FR, an austenitic stainless steel with superior creep-fatigue resistance. 316FR was developed in Japan based on SUS316 of the Japanese Industrial Standard which is equivalent to Type 316SS [20]. Chemical composition was optimized to improve creep properties by reducing carbon content and adding nitrogen. A unique point with 316FR is that Phosphorus is also added to improve creep properties. This material has already been applied to the intermediate heat exchanger of the Japanese experimental fast reactor Joyo. For the coolant systems including primary and secondary piping, intermediate heat exchangers and steam generators, Mod.9Cr-1Mo steel, which is equivalent to ASME Grade 91 steel, is of prime interest [21]. The combination of high thermal conductivity and low thermal expansion in this material significantly contributes to simplifying plant design. The focus of the development of these materials is establishing materials strength standards for a 60-year design, extending the current time-dependant allowable stresses to 500,000 hours. From this viewpoint, the acquisition and collection of long-term creep data have been continued [22-26]. The most important failure mode to be prevented in the design of fast reactors is creep-fatigue failure, and suitable evaluation methods have been established for 316FR [27] and Mod.9Cr-1Mo steel [27]. The evaluation of environmental effects, irradiation effects and sodium environmental effects is also of significance, and the applicability of the evaluation methods developed for conventional materials [29] has been confirmed. Another important item to develop is creep-fatigue evaluation methods for welded joints [30, 31]. In the case of Monju, the Japanese prototype fast reactor, welded joints were not in high stress regions of the components, but in the next generation reactors, it would not necessarily be the case. When applying those materials for a service period of 60 years, it is necessary to identify suitable weld materials and clarify strength reduction that may occur during a service period. The technical developments will be incorporated in the Japan Society of Mechanical Engineers (JSME) Code for design and construction of fast reactors [32, 33].

2. Core materials for fuel assemblies

2.1.Oxide Dispersion Strengthened Steel

2.1.1. Material Development

As described in the previous report [2], ODS steel cladding tubes for FR application developed by JAEA are categorized into two types: martensitic ODS steel with ferritic-martensitic duplex matrix and ferritic ODS steel with recrystallized ferritic matrix. JAEA has focused on development of martensitic ODS steel cladding tube, which is advantageous over ferritic ODS steel in terms of irradiation resistance and tube manufacturability [2-4]. Both types of ODS steels contains high population of Y-Ti-O complex oxide particles dispersed in matrix for high-temperature strength enhancement [2-6]. As for ODS martensitic steel, 9Cr and 11Cr-ODS steels have been developed while 12Cr-ODS steel for ODS ferritic steel. Figure 1 shows schematic view of required properties for high burn-up fuel cladding tube. Taking into account the operation conditions of FR fuel cladding tube (i.e. high-temperature and high neutron dose environments, fuel-cladding chemical interaction, flowing sodium corrosion, nitric acid corrosion), high burnup fuel cladding tube should possess corrosion resistance in addition to high-temperature strength and irradiation resistance. Based on plenty of studies on high-temperature strength and neutron irradiation effects in conventional ferritic steels, Cr concentration of 9wt% has been considered as the best choice to produce creep rupture strength improvement and suppression of ductility loss caused by irradiation hardening. On the other hand, selection of Cr concentration higher than 9wt% would be profitable for corrosion resistance improvement. Based on the extensive research on high-temperature strength of 9Cr-ODS steels, whose optimized composition is Fe-0.13wt%C-9Cr-2W-0.2Ti-0.35Y_2O_3, microstructural factors controlling high-temperature strength and microstructure stability are shown to be high population of nano-oxide particles dispersed in matrix and residual-α ferrite as reinforcement phase [2-6, 10, 11]. In the light of knowledge on microstructure and mechanical properties of 9Cr-ODS steel, a new specification of high Cr ODS tempered martensitic was determined with the views described below [2, 9, 12]. Increasing Cr concentration can improve the corrosion resistance, however too much Cr concentration would produce a risk of irradiation embrittlement; the Cr concentration was selected to be 11wt%, which is the same as that of PNC-FMS having good irradiation resistance. The basic composition of 11Cr-ODS steel was selected as Fe-0.13wt%C-11Cr-1.4W-0.4Ni-0.2Ti-0.35Y_2O_3, where concentrations of Ni and W were chosen to produce the optimized ferritic-martensitic duplex microstructure (optimized fraction of residual-α ferrite) equivalent to that of 9Cr-ODS steel, and to keep the Ac1 temperature higher than 1123K. Figure 2 shows the thermodynamic calculation results for selecting Ni and W concentrations in 11Cr-ODS steel using FactSage code and FSstel data base [34]. In ODS steels, it is difficult to simply predict the phase state on the basis of calculated phase diagram due to the presence of nano-oxide particles which pin the motion of phase boundary [4, 9]. The driving force for α to γ phase transformation was shown to be a useful indication for controlling residual-α ferrite proportion [9]. Concentrations of minor alloying elements such as Ti, excess oxygen (Ex.O) and Y_2O_3 were set to the same as those of 9Cr-ODS steel for nano-structure control. Excess oxygen is defined as the oxygen concentration estimated by subtracting oxygen coupled with Y in the form of Y_2O_3 from total oxygen in steel [2-6]. The microstructure of 11Cr-ODS steel including the duplex matrix and nano-sized oxide particle dispersion was successfully controlled equivalent to that of 9Cr-ODS steel [9, 12], thus having good tube manufacturability. For enhancing the flexibility of ODS martensitic steel development, 11Cr-ODS steel is ranked as a prospective choice of FR fuel cladding tube in addition to 9Cr-ODS steel in JAEA.

2.1.2. Fabrication and verification

Figure 3 indicates the schematic view of manufacturing process of ODS steel cladding tube. As reported in several papers [3, 15-18], most of research institutes including JAEA have utilized powder metallurgy process while there are some differences between their fabrication procedures and conditions in detail. In general, Y_2O_3 cannot be in solid solution in Fe-based matrix at equilibrium state; only high energy mechanical alloying process can yield the enforcement of Y_2O_3 dissolution in Fe-based matrix. In the subsequent hot consolidation process such as hot-extrusion (HE) and hot isostatic pressing (HIP), the forcibly dissolved Y_2O_3 reprecipitates as nano-sized Y-Ti-O oxide particles as shown in Figure 4 [2-6]. JAEA has comprehensively advanced the development of ODS steel cladding tube fabrication technologies, which includes mother tube fabrication using powder metallurgy, tube manufacturing technology using cold-work and inspection technology, thus established those for high strength tube production in laboratory scale [2-5]. In-pile and out-of-pile mechanical tests of 9Cr and 12Cr-ODS steel cladding tubes were conducted to demonstrate their superior properties and accumulate a data base for material strength standard [2, 3]; post irradiation examination results of 9Cr, 12Cr-ODS steel cladding tube irradiated using material irradiation rig in Joyo (693–1108K to <33dpa) demonstrated the notable mechanical properties and microstructure stability after irradiation [2]. Raw material powder used for these tube productions were either pre-mix powder or partially pre-alloyed powder, where pre-mix powder means the mixture of elemental powder and Y_2O_3 powder. Partially pre-alloy powder means the mixture of pre-alloy powder, Y_2O_3 powder, and elemental powder for minor control of composition. Using BOR-60, the experimental fuel assemblies with 9Cr, 12Cr-ODS steel cladding fuel pins were irradiated up to peak burn-up of 112 GWd/t and peak neutron dose of 51 dpa [7, 8]. Superior properties of the ODS claddings such as fuel compatibility, dimensional stability under irradiation were confirmed. However, in this irradiation test, a fuel pin rupture occurred in a 9Cr-ODS steel pin while there was no rupture occurred in 12Cr-ODS steel. The 9Cr-ODS steel cladding tubes used for the BOR-60 irradiation test were fabricated with pre-mix powder, while 12Cr-ODS steel tubes partially pre-alloy powder. The utilization of pre-mix powder caused formation of matrix Cr heterogeneity (metallic Cr inclusion) in the 9Cr-ODS steel cladding tubes, which induced microstructure instability and fuel pin rupture in combination with high-temperature irradiation exceeding the designed maximum irradiation temperature, 973K [7, 8]. It was shown that metallic Cr inclusions are difficult to be detected by ultrasonic inspection technique because acoustic impedance of Cr roughly equals to that of Fe [8]. The investigation results indicated that homogeneity improvement of ODS steel tube products by suppressing metallic Cr inclusions was indispensable for demonstration of their intrinsic irradiation resistance and high temperature microstructure stability. In response to these results, JAEA has started modifying the fabrication process for homogeneity improvement and assurance of the cladding tube products. Mother tube fabrication process (Figure 3 (1)-(4)), the disseminating process of the cladding tube cleanness, was modified. JAEA designated this modified new process “the full pre-alloy process”. The raw material powder used in the full pre-alloy process is high cleanliness pre-alloy powder and Y_2O_3 powder; elemental powder is completely excluded all through the mother tube fabrication process. Additionally in this process, strict management system in powder metallurgy process (e.g. powder handling procedure, operation and management procedure of attritor ball mill,…, etc.) was determined. Using the full pre-alloy process, two lots of 9Cr-ODS steel cladding tubes were successfully fabricated without any cracking during cold-rolling; total number of fabricated tubes were 63, and tube size was 6.9mm in outer diameter, 6.1mm in inner diameter, and 1800mm in length. One hundred and fifty metallographic samples were randomly extracted and observed by optical microscope and scanning electron microscope (SEM), where the observation area in each sample is approximately 2 cm^2. In the observation, there were no metallic Cr inclusions and any detrimental heterogeneity. Figure 5 shows quantitative estimation results using image analysis technique of non-metallic inclusions formed in full pre-alloy and pre-mix 9Cr-ODS steel cladding tubes. An example of SEM view is also displayed in the Figure. Number of non-metallic inclusions is counted in 10 SEM views (size of a view is 45μm x 60 μm). It was clearly seen that the full pre-alloy process can decrease non-metallic inclusions which would act as initiation sites for fracture. Three lots of full pre-alloy 11Cr-ODS steel cladding tubes were also successfully manufactured without any cracks; total number of manufactured tube were 36. Pressurized resistance welding

(PRW) test between 11Cr-ODS steel cladding tube and 9Cr-ODS steel end-plug proved that 11Cr-ODS steel has adequate weldability equivalent to 9Cr-ODS steel [35].

The internally pressurized creep rupture test results of 9Cr and 11Cr-ODS steel cladding tubes are shown in Figure 6, where data of 9Cr-ODS steels are for full pre-alloy lot, partially pre-alloy lot, and pre-mix lot, and those of 11Cr-ODS only for full pre-alloy lot. Full pre-alloy 9Cr-ODS steels have good creep rupture strength much higher than pre-mix 9Cr-ODS steels. Modification of ODS steel production process to full pre-alloy process is originally motivated by the presence of detrimental metallic Cr inclusion in pre-mix lot; this modification can achieve not only suppression of metallic Cr inclusions but also reduction of non-metallic inclusions, thereby increasing high-temperature creep strength. For full pre-alloy 9Cr,11Cr-ODS steel cladding tubes, out-of-pile mechanical tests including long term creep rupture tests are on-going to accumulate the data for preparation of material strength standard. Neutron irradiation tests of these cladding tubes in Joyo are planned. The current production technology is in the laboratory-scale. Roughly 10 kg powder is processed in a mechanical alloying; mother tube size is 18 mm in outer diameter, 12mm in inner diameter, and 200mm in length. Towards the development of mass production process, JAEA restarted the research and development (R&D) to scale up the mechanical alloying process and mother tube size. The technological knowledge on full pre-alloy process will be incorporated into this development. Moreover, for accumulating basic knowledge conducive to the progress of scaling up R&D, JAEA has systematically studied the effects on nano-structure of 9Cr-ODS steel produced by thermo-mechanical process to consolidate MA powder [10], and the effects on homogeneity (type and amount of inclusions) and mechanical properties brought by choice of manufacturing procedure [11]. So far, the parameters dominating the nano-structure of 9Cr-ODS steel have been thought to be minor chemical compositions (Ti, Y_2O_3, and Ex.O concentrations), and hot-extrusion temperature [3-6]. The recent study [10] revealed that thermo-mechanical processing played a crucial role in nano-structure control. It was clearly shown the definite predominance of full pre-alloy process against pre-mix process in mechanical properties including Charpy impact properties and creep rupture strength [11]. ODS steel is a highly strengthened steel by nano-sized oxide dispersion. It is of engineering concern whether or not its increased yield stress could degrade the toughness. It was clearly shown that the full pre-alloy process can largely reduce the inclusions, thus endowing ODS steels with adequate impact toughness equivalent to PNC-FMS (Figure 7 [11]). For the establishment of a reliable production process of high strength tube in the future mass production, basic and systematic studies will be continued and reflected to the selection of mother tube fabrication conditions.

2.2.11Cr Ferritic/martensitic steel

The material development of PNC-FMS had already been completed. Chemical composition of PNC-FMS is defined as Fe-0.12C-11Cr-0.5Mo-2W-0.4Ni-0.2V-0.05Nb-0.05N in mass%. JSFR designing assumed that type 316 steel is applied to handling head and entrance nozzle of the fuel subassembly [2, 19] since it has substantial toughness, and can prevent sodium leakage due to difference of thermal expansion coefficient between entrance nozzle and core support plate. Thus, developmental tasks currently executed on PNC-FMS wrapper tube are mechanical data accumulation for establishment of material strength standard, and development of dissimilar joining technology between PNC-FMS and type 316 steel. As for the dissimilar joining, there are two options: dissimilar welding and mechanical joint. JAEA has optimized the electron beam (EB) welding condition on PNC-FMS - type 316 steel weld joints, and succeeded in suppressing δ-ferrite formation [19]. Long term thermal aging tests and mechanical tests of these joints are in progress.

3. Structural materials for reactor vessels and piping

3.1.Data acquisition and evaluation of long-term properties

The acquisition and collection of long-term creep data of 316FR and Mod.9Cr-1Mo steel are continued to support establishing time-dependent allowable stresses up to 500,000 hours [22-26]. An example of the current database is shown in Figure 8(1) for 316FR and Figure 8(2) for Mod.9Cr-1Mo steel, respectively. Time to rupture was formulated as a function of stress and temperature using the Larson-Miller Parameter. For Mod.9Cr-1Mo steel, in order to improve the accuracy of regression in a long-term region, the region splitting analysis method [36] has been adopted; the rupture curve was determined for the short-term and long-term regions respectively using a second-order polynomial function. The intercept of the two curves was determined as a half of yield strength at temperature. Although the region splitting analysis method is used only for Mod.9Cr-1Mo steel it is considered to improve the accuracy of prediction for 316FR, too.

3.2.Evaluation methods for welded joints

3.2.1.Similar welded joints

Establishing structural integrity evaluation methods of welded joints is one of the most important subjects in the design of next generation fast reactors. Welded joints would be used in high stress regions of components, contrary to the case of Monju, the Japanese prototype fast reactor. Developing creep-fatigue evaluation methodologies on the improved understanding of long-term creep properties is necessary. For the welded joints of 316FR, two candidate filler metals, similar metal and 16-8-2 type, are being investigated in light of long-term creep properties [37] as shown in Figure 9(1). The one with superior long-term properties will be chosen for actual plant applications. Creep-fatigue evaluation methodologies are developed taking account of the difference among base metal, weld metal and heat affected zone (HAZ) in terms of strength and inelastic behavior [38]. For the welded joints of Mod.9Cr-1Mo steel, the most important point is the evaluation of creep strength in long-term regions where “Type IV” cracking could occur [21]. As is the case with base metal, the region splitting analysis method is useful for welded joints of this steel [36]. Generally, Type IV cracking occurs in the long-term region where stress is below a half of 0.2% yield strength, and it is considered that the region splitting method reasonably represents this phenomenon. Studies have been done on the derivation of welded joint strength reduction factors using the region splitting analysis method, as shown in Figure 9(2) [39]. A significant amount of research and development on creep-fatigue evaluation methodologies has also been conducted [31, 40]. The base metal of Mod.9Cr-1Mo steel cyclically softens, while the weld metal cyclically hardens. Creep-fatigue evaluation should take account of the discontinuity both in strength and inelastic deformation properties. The location of failure could vary just like the case of creep rupture. It is desirable to obtain long-term creep data at temperatures as close as possible to the operating temperature of the component. Evaluation methods have been proposed for the welded joints of Mod.9Cr-1Mo steel [39].

3.2.Dissimilar welded joints

When ferritic/martensitic steels are used for structural components, usually dissimilar welded joints with austenitic stainless steels are used, too. When they are subject to elevated temperatures, creep property evaluation and creep-fatigue evaluation are needed. In the case of dissimilar welds, close attention should be paid to the location of failure when developing evaluation methodologies [40, 41].

3.3.Fabrication technologies

Next generation fast reactors could adopt various innovative designs, and the geometry or configuration of the components could be beyond the conventional application. For example, in a past preliminary design study of a sodium-cooled fast reactor [42], there were a couple of components that could need new fabrication technologies. Those include large diameter forged pipes, which are to increase the capacity of cooling per loop, a very thick forged plate for the tubesheets of steam generators, and very thin walled tubes of steam generators, which is to maximize heat transfer efficiency. Each of these needs could require some innovation in the fabrication process and may need the installment of new equipment.

There have been some studies in this direction. Nagae et al studied the mechanical properties of very thin walled heat transfer tubes and very thick forgings made for tubesheets [43]. For the former, tensile and creep tests were performed using arc-shaped specimen machined from thin-walled tubes with the thickness of approximately 1.5 mm. The properties were equivalent to thicker conventional tubes and the effects of thickness were considered negligible. With regards the thick forging, they investigated the effect of thickness on creep properties using forging with the thickness range of 250 to 550 mm. Generally, in the center of thickness in forging material, because of low cooling rate in the temper treatment, it is probable that the recovery of lath structure and the coarsening of precipitates easily occur. This could lead to the reduction of creep strength. Specimens for creep tests were machined from the center in thickness of the forgings, and it was shown that the creep strength was of the same level as that of plate products. The results indicate that product forms that were not common in conventional applications could also be feasible to the design of next generation fast reactors.

3.4.Codification in JSME

For the design and construction of next generation fast reactors in Japan, the JSME Code for design and construction of fast reactors will be used. The latest 2016 edition incorporates 316FR and Mod.9Cr-1Mo steel. A set of allowable stresses were prepared in the same manner as conventional materials with time-dependent ones up to 300,000 hours, which will be extended to 500,000 hours in a near-future edition. Equations such as creep rupture equation, creep strain equation, equations for low-cycle fatigue life and so on were also determined. An evaluation method for neutron irradiation effect was incorporated for 316FR. Sodium environmental effects are incorporated for both 316FR and Mod.9Cr-1Mo steel. In the course of the adoption of the new materials, a number of structural tests were performed and applicability of the elevated temperature design methods to the new materials was verified [44, 45].

4. Future Perspective

4.1. Core materials

There are two important tasks for ODS steel development in JAEA: development of large scale manufacturing technology for future mass production, and establishment of material strength standard. Towards the mass production process development, scaling up R&D of the mechanical alloying process has restarted. The knowledge on full pre-alloy process and basic research results on nano-structure control will be reflected to optimization of large scale manufacturing process. For PNC-FMS, the dissimilar joining technology with type 316 steel will be established. The irradiation performance of both steels will be evaluated through material, fuel pin and bundle subassembly irradiation test in Joyo to demonstrate in-reactor properties of 9Cr,11Cr-ODS steel and PNC-FMS for use as FR long life core materials. These data will be used for upgrading of the material strength standards of cladding and wrapper tubes.

4.2.Structural materials

Currently, extensive efforts are being made to extend the time-dependent allowable stresses of 316FR and Mod.9Cr-1Mo steels in the JSME Code to 500,000 hours [33]. The JSME Code provides not only time-dependent allowable stresses but also creep rupture equations and creep strain equations. Therefore, accuracy of the equations is also carefully being ensured. For the implementation of a 60-year design, on top of establishing allowable stresses and associated material strength standards, ensuring structural integrity by means of non-destructive monitoring would be desired. Research and development in this direction would be of significance in future.

5. Summary

Application of long life core materials to FR core can contribute to volume and hazardousness reduction of radioactive waste in conjunction with improvement of economic performance of FR. JAEA has developed 9Cr,11Cr-ODS steels and PNC-FMS as the most promising materials for long life fuel claddings and wrapper tubes, respectively. Development of mass production process of ODS steel tubes should be advanced in the light of technological knowledge derived through the full pre-alloy process development. Accumulation of mechanical properties data of both steels for preparation of material strength standard will be continued towards the practical realization of high burn-up core. It is planned that in-pile data of these core materials will be derived using Joyo.

316FR and Mod.9Cr-1Mo steels are promising candidates for the components such as pipes and vessels of next generation fast reactors. Extensive R&Ds have been conducted within JAEA and a fairly large amount of material data is available. Structural design codes have also been developed by JSME. A major item to be further explored for next generation fast reactor application is the extrapolation of elevated temperature properties to achieve a 60 year lifetime of a plant. For this purpose, databases and evaluation procedures are being developed. Code qualification is also ongoing and the JSME Code would soon implement time-dependent allowable stresses to 500,000 hours.

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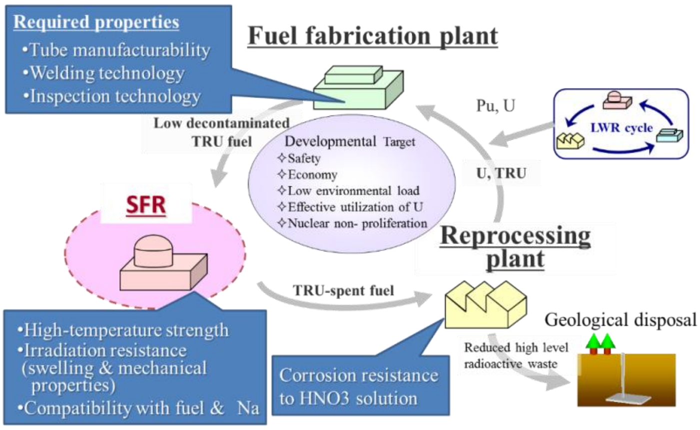

FIG. 1. Schematic view of required properties for high burn-up fuel cladding tube

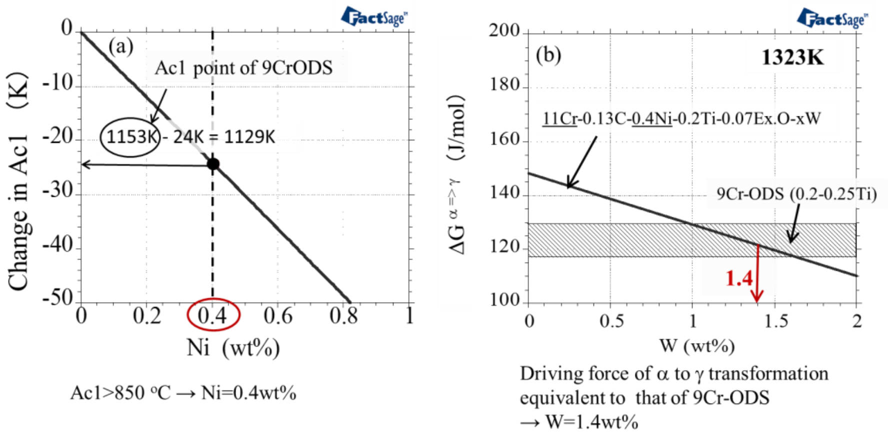

FIG. 2. Thermodynamic calculation results for selection of Ni and W concentrations in 11Cr-ODS steel using FactSage code and FSstel data base [34], (a) effect of Ni concentration on change in Ac1 point, (b) chemical driving force for α to γ reverse phase transformation at 1323K versus W concentration.

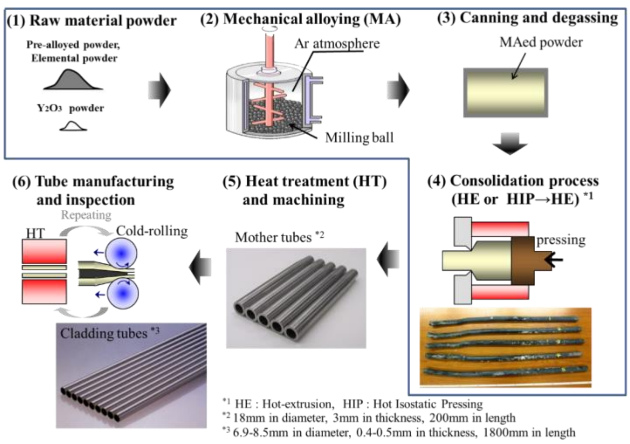

FIG. 3. Schematic view of manufacturing process of ODS steel cladding tube.

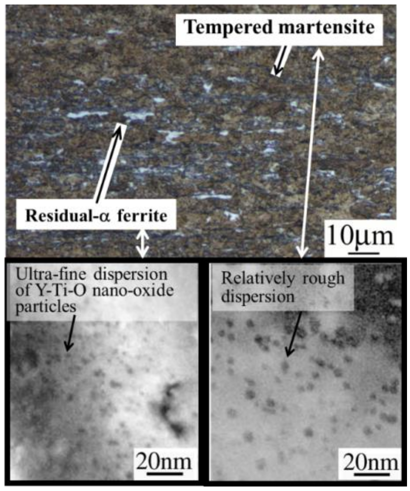

FIG. 4. Typical microstructure (transmission electron microscope image, bright field view) of 9Cr-ODS steel cladding tube.

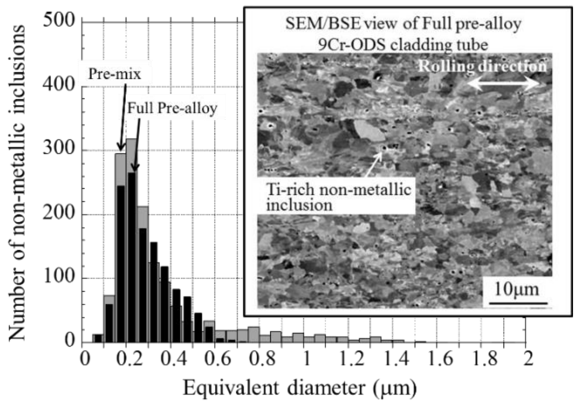

FIG. 5. Image analysis result of metallographic observation for counting non-metallic inclusions in full pre-alloy and pre-mix 9Cr-ODS steel cladding tubes. Number of non-metallic inclusions were counted in 10 SEM observation views, whose sizes are 45 μm x 60 μm for a view.

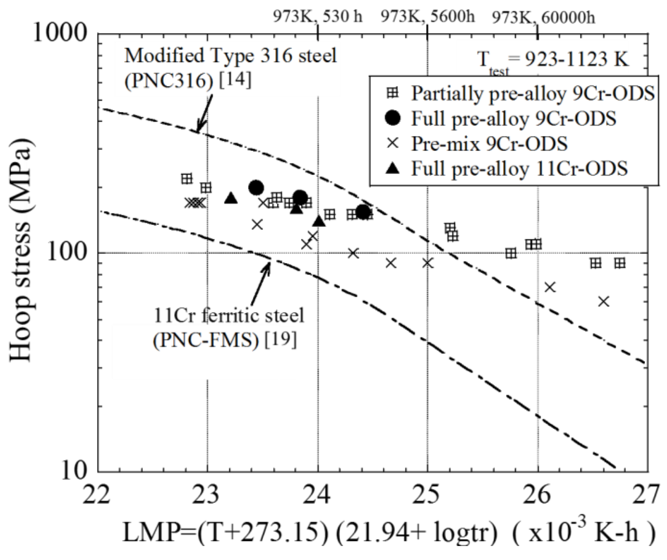

FIG. 6. Internally pressurized creep rupture test results of 9Cr and 11Cr-ODS steel cladding tubes.

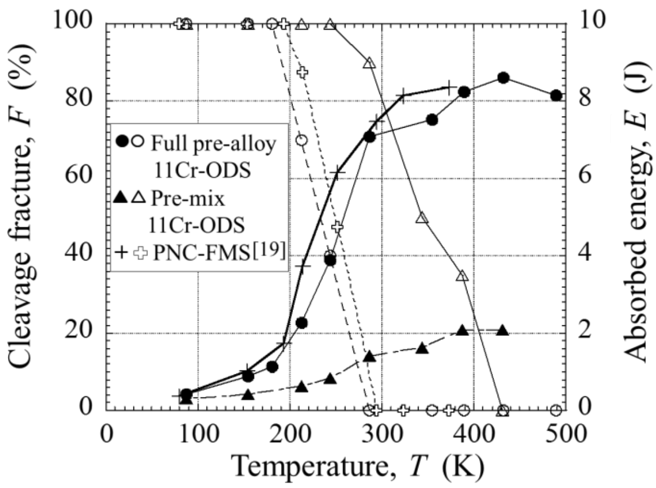

FIG. 7. 1/3-size Charpy impact properties of full pre-alloy and pre-mix 11Cr-ODS steels and 11Cr-ferritic steel (PNC-FMS) [11].

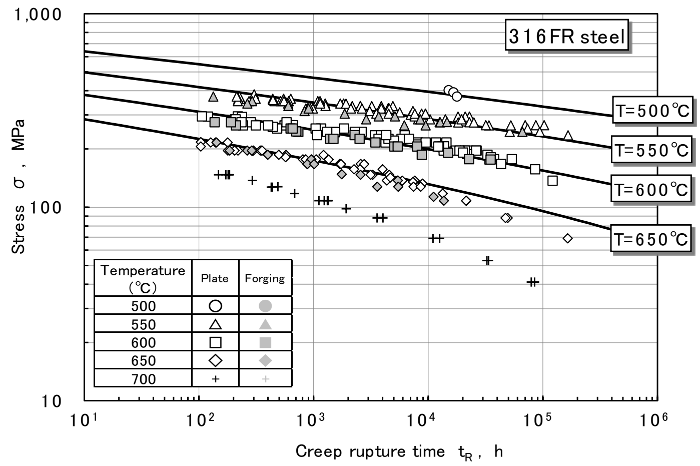

FIG. 8 (1) Acquisition and collection of long-term creep data (316FR) [22-24].

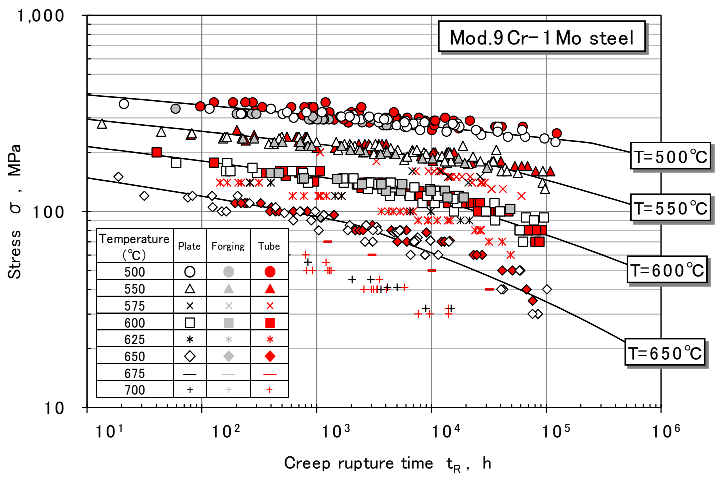

Figure 8 (2) Acquisition and collection of long-term creep data (Mod.9Cr-1Mo steel) [25,26].

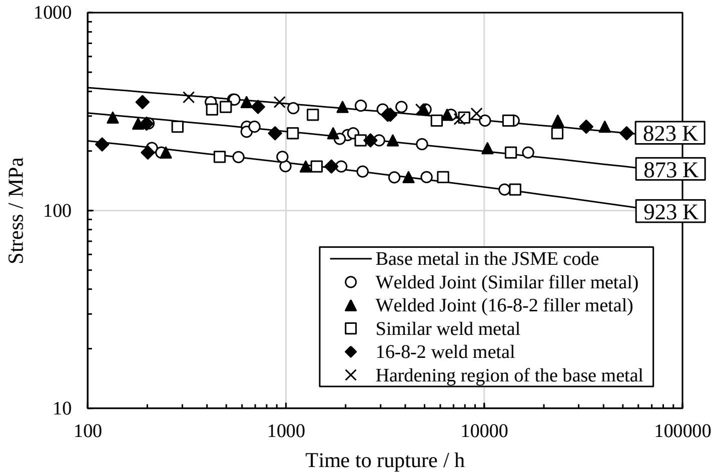

FIG. 9(1) Acquisition and collection of creep data of welded joints (316FR) [37].

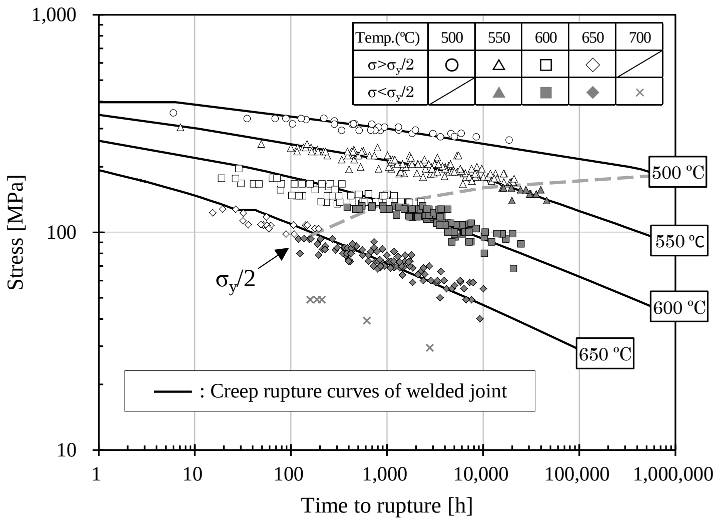

FIG. 9(2) Acquisition and collection of creep data of welded joints (Mod.9Cr-1Mo steel) [39].