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STRUCTURAL MATERIALS AND CODE DEVELOPMENT FOR JAPANESE SODIUM-COOLED FAST REACTORS

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ABSTRACT

This paper describes the latest status on the development of elevated temperature materials and structural codes for Japanese sodium-cooled fast reactors (SFRs). Based on the extensive research and development activities in the last decades in Japan, two materials, 316FR and Modified 9Cr-1Mo steels were recently incorporated into the 2012 Edition of Fast Reactor Design and Construction Code of the Japan Society of Mechanical Engineers (JSME). Structural design methodologies are continuously being improved towards the next major revision planed in 2016 Edition where methodologies for a 60-year design of Japanese demonstration fast reactor will be provided. Codes and guidelines for fitness-for-service, leak-before-break evaluation and reliability assessment are concurrently being developed utilizing the System Based Code concept aiming at establishing an integrated code system that encompasses a life cycle of SFRs.

INTRODUCTION

For the commercialization of innovative reactors such as Generation IV nuclear reactors, the development of materials with superior elevated temperature properties is one of the most crucial aspects. Also of critical importance is to provide schemes that allow taking fully advantage of the superior materials while ensuring structural integrity over a long period of design life; those include maneuvers to monitor actual margins in the component during operation, various elevated temperature design methods that are reliable particularly in long-term regions, and inservice inspection requirements most appropriately derived taking the characteristics of sodium cooled fast reactor design in consideration.

This paper introduces the ongoing activities on the above perspective in Japan. Various research and developments are conducted by the Japan Atomic Energy Agency (JAEA) and its partners, and the results are codified in the Fast Reactor Codes published by the Japan Society of Mechanical Engineers. Activities are going on for both the prototype fast breeder reactor Monju, and the demonstration fast reactor JSFR (Japan Sodium-cooled Fast Reactor) which is in the phase of conceptual design.

Regarding new materials development, the 2012 Edition of the Fast Reactor Design and Construction Code of the Japan Society of Mechanical Engineers (JSME FR Code) has implemented 316FR and Mod.9Cr-1Mo steels [1] for expected application to JSFR. The former is the fast reactor grade of Type 316 Stainless Steel with higher creep properties, and the latter is a material equivalent to the Grade 91 steels in the Boiler and Pressure Vessel Code of the American Society of Mechanical Engineers (ASME). The time dependent allowable stresses are determined for up to 300,000 hours in the 2012 Edition, and the extension to 500,000 hours, which corresponds to a design life of 60 years, is planned in the 2016 Edition where the next major revisions are expected.

As to the framework of code requirements that allows to fully take advantage of structural materials for fast breeder reactors, the System Based Code (SBC) concept, the concept currently being extensively elaborated for the application to next generation reactors, can be utilized [2-5]. The SBC concept was proposed based on the recognition that existing codes and standards structure had become unnecessarily rigid in the long history of development and through a number of revisions. The SBC concept presents a new framework that is flexible enough to reflect the state-of-the-art technologies and operation experiences of existing reactors. It encompasses a whole plant life cycle and is flexible enough to allow modification of design margins based on inspection methods when it is fully implemented. The discussion in this paper, however, focuses on the determination of inservice inspection requirements for a given combination of material, component design and operation conditions, which is an issue of current interest.

Subsequent sections of this paper describe the above aspects in detail.

CODIFICATION OF 316FR AND MOD.9CR-1MO STEEL IN 2012 EDITION OF JSME FAST REACTOR CODE

The JSME FR Code had included a limited number of materials which were used in Monju at elevated temperatures [1]. Those were SUS304 (SA-240 304), SUS316 (SA-240 316), tube of SUS321 and normalized and tempered 2 1/4Cr-1Mo steel (SA-387 22 cl.1). The notation in parentheses refers to the equivalent material in the ASME Boiler and Pressure Vessel Code Section II.

The 2012 Edition of the JSME FR Code recently incorporated 316FR and Mod.9Cr-1Mo steel. 316FR is an austenitic stainless steel developed from conventional SUS316 (Type 316 Stainless Steel) by modifying the chemical composition to improve creep properties. The carbon content was reduced (0.020 % maximum), and the nitrogen content was increased (0.06-0.12 %). The phosphorous content was also adjusted to lie in the range 0.020-0.045 %. Mod.9Cr-1Mo steel is equivalent to ASTM/ASME Grade 91 steel.

A large materials database (Structural Materials Database, SMAT) has been developed for these two materials based on research and development that had been conducted in Japan since the 1990s. Creep rupture data of which time exceeds 100,000 hours were obtained for both materials. Creep-fatigue tests have also been extensively performed, and for 316FR, the maximum time to failure is over 80,000 hours. Tests using aged materials, tests in sodium environment, tests on pre-irradiated specimens have also been conducted and results were accordingly analyzed.

The 2012 Edition of the JSME FR Code implemented guidelines for the incorporation of new materials which define requirements regarding the number of heats, tests to be performed, conditions under which tests should be carried out, etc. 316FR and Mod.9Cr-1Mo steel were codified following these guidelines. The allowable stresses in the JSFR Code are determined basically in the same approaches as those in the ASME Boiler and Pressure Vessel Code Sections II and Subsection NH of Section III, but there are some differences in detail. In the 2012 Edition, the time dependent allowable stresses were determined for up to 300,000 hours.

The details in the determination of the minimum stress to creep rupture are as follows [6, 7]. In the case of 316 FR steel, two product forms, plate and forging, were codified. Referring the database SMAT, for plate, 250 creep rupture data from 9 heats obtained in the temperature range of 550-700 C were utilized. For forging, 50 data from 3 heats obtained in the temperature range of 550-600 C were used. In selecting the heats, those with less than 10 data or those without data at 550 C and 600 C were excluded to ensure reliability. The maximum time to rupture was 121,757 hours and 66,562 hours, for plate and forging, respectively. Since both product forms exhibited the same creep behavior, they were formulated into an identical set of curves as shown in Fig. 1.

Two product forms, plate and forgings, were also tested for Mod.9Cr-1Mo steel. 330 data points were obtained in the temperature range 550-700 C. The heats selected for analysis were subject to the same restrictions as described for 316FR steel. The maximum time to rupture was 120,500 hours. This material also demonstrated similar creep behavior in plates and forgings, and both product forms were combined into a single set of curves, as shown in Fig. 2. Those curves were formulated using the region splitting analysis method [8] that the authors consider is the most appropriate approach for representing the creep rupture curves of high-chromium steels.

Other allowable stresses and allowable strain range (design fatigue curves) were also determined in a similar manner.

The code name of 316FR is “JSME F-01” and that of Mod.9Cr-1Mo steel is “JSME F-02”, where “F” designates fast reactor use. For F-02, ASME Grade 91 steel has been assigned as an equivalent material.

EXTENSION OF ALLOWABLE STRESSES TOWARDS 60-YEAR LIFE DESIGN

The next major revision for the JSME FR Code is scheduled in 2016. In this edition, the time dependent allowable stresses will be extended to 500,000 hours. At the same time, the elevated temperature design rules will be updated where necessary to ensure that they are applicable to a 60-year design with maintaining appropriate margins.

To ensure materials integrity for 60 years, a “three-principle approach” is being pursed. The first principle is to obtain as much reliable long-term creep data as possible by the time allowables are established. The second principle is to justify (inevitable) extrapolation using metallurgical observations, and the third is to conduct long-term creep tests that run in parallel with plant operation to demonstrate actual creep margins at various points in the service life of the plant. The following paragraphs provide more details regarding the three principles just described.

More long-term creep rupture testing than indicated in Figs. 1 and 2 has been initiated to support the planned extension of the allowables to 500,000 hours. Some of these tests will produce data to support the 2016 Edition of the JSME FR Code and will be used to reduce the required extrapolation from the test data to the 500,000-hour service life to less than 5 times. The number of the newly generated data will be limited, however, since there is not so much time before we will be drafting the 2016 Edition of the JSME FR Code. Therefore, in order to further justify the extrapolation, information on some of the continuing tests will also be utilized. If ongoing tests, particularly those estimated to be close to rupture, are treated as if they have already ruptured, the allowable stresses could be determined conservatively yet avoiding being too much so. After the specimens have actually ruptured, the margins on the allowable stresses could be revisited and revised in the upward direction if relevant.

Metallurgical investigations also help justify extrapolation based on temperature acceleration. The target temperature range for acceleration is 50 C, that is, we assume that we can evaluate stress-to-rupture at 550 C based on data generated at 600 C utilizing the Larson-Miller Parameter method. In order to verify this, various metallurgical properties such as hardness, grain size, dislocation density, precipitates, inclusions, etc. are being analyzed. The results will be displayed in a time-temperature-precipitation format, for example.

The third point is monitoring the properties of the specific materials actually used in components of the plant. The simplest way is to start creep tests with materials and stress levels equivalent to those in the plant component before the plant operation and continue them in parallel to plant service. This will serve to quantify actual margins for a particular component and reflect the information obtained to plant operation and code revisions when necessary. Tests have already been planned and partially have started for this purpose. Another possible way is to put coupons in the plant in a way similar to we implement coupons to monitor irradiation effects in existing reactors. We would need to discuss this point in further detail because the installment of such equipment may affect component or system design. Moreover, methods for evaluating effects of loads on aging must be developed, because coupon specimens are most likely installed stress-free. However, once realized, this method would be a very powerful tool to determine actual margins.

To summarize, the first two principles out of the three proposed for the realization of a 60-year design will be implemented simultaneously with the development of the 2016 Edition of the JSME FR Code, and the third one will be further elaborated in longer time efforts.

CREEP-FATIGUE AND WELEDED JOINTS

In the previous section, we discussed the most basic element in realizing a 60-year design: capturing and extrapolating creep rupture strength of base metals. The most significant concern in the design of sodium-cooled reactors is, however, creep-fatigue interaction accompanied by stress relaxation during steady state operation. Accurate creep rupture properties are a prerequisite for a creep-fatigue evaluation, but their usefulness for predicting the long-term failure life under cyclic loading conditions has to be demonstrated by comparing the predicted failure life with creep-fatigue test data. Producing creep-fatigue test data requires more sophisticated machines and associated resources compared to the case of creep; the maximum duration available for creep-fatigue tests would inevitably be shorter. Even with this difficulty, a creep-fatigue test database in which the maximum failure time is over 80,000 hours (316FR steel) has been established in the course of research and development in Japan. Based on the database, the authors have conducted a detailed study and have demonstrated for both 316FR and Mod.9Cr-1Mo steels that adequate conservatism is maintained in long-term regions by using the time-fraction approach based creep-fatigue damage evaluation method as shown in Figs. 3 and 4 [9, 10].

Another point which could be of concern is welded joints. Again, data would be limited compared to base metals, therefore, welded joint properties have to be evaluated with taking advantage of information obtained on base metals. The data set used for the determination of weld strength reduction factors for Mod.9Cr-1Mo steel is shown in Fig. 5 [11]. Those factors were also determined utilizing the region splitting analysis method that was used for the formulation of the creep rupture equations of base metal.

The relevance of a 60-year design of sodium-cooled fast reactors ought to be judged taking into account of factors such that we discussed above; the careful extension of allowable stresses is an indispensable part of it yet not the whole of it. Important issues like creep-fatigue and welded joints have to be addressed with the same intensity. And, one more item to be dealt with equally cautiously is inservice inspection, which will be discussed in the next section.

DERIVATION OF INSERVICE INSPECTION REQUIREMTNS BASED ON SYSTEM BASED CODE APPROACH

One of the things the authors would like to emphasize in this paper is that the job for material engineers does not terminate either with setting allowable stresses or with starting long-term creep tests. To ensure the integrity of materials over a long design life without imposing unnecessary burden or restrictions on other aspects of plant design and operation, considerations from a broader perspective becomes critical. This paper discusses a specific issue of how we are able to determine the most appropriate inservice inspection requirements for sodium-cooled fast reactors, utilizing the SBC concept.

The System Based Code (SBC) concept was proposed by Asada [2] and extensively studied not only in JSME [2-5] but also in ASME [13] as a new framework that would realize the most adequate allocation of margins to various technological aspects of plant design and operation. The SBC concept consists of three parts: 1) design to target reliability that must be met throughout the service life, 2) margin exchange among the various technical areas of concern such as design, inspection, fabrication, and fitness for service, and 3) expand technical options by the timely adoption of newly developed technologies that are not in current codes and standards.

As described above, the SBC concept allows the interaction of design and inspection in terms of margins. We are able to determine inservice inspection requirements based on design specifications, and vice versa. In this paper, however, we limit our discussion on the first case; the design is left intact and we derive the most appropriate inservice inspection requirements for it

A representative logic flow diagram based on the System Based Code concept is given in Fig. 6 [12]. When we leave design (rules) intact, we are to perform evaluations in two stages; the first stage is structural reliability evaluation that include failure modes that have not been explicitly addressed in the design code (an example is stress corrosion cracking in light water reactors), and the second stage is evaluation from the stand point of plant safety. In this stage, safety related aspects, such as the function of the component in terms of maintaining safety, defects postulated in the safety analysis of the plant, the availability of mitigation systems, the detectability of defects that might grow to a critical size, etc. are considered as input. Based on such input, the most appropriate inservice inspection requirements are derived with the bottom line being maintaining the safety of the plant.

Specifics for the second stage evaluation would vary component to component. Here is an example of a component that forms a sodium boundary such as pipe of the primary coolant system. The essence in this case is that if we are able to demonstrate that structural integrity is sufficiently high (the first stage evaluation), and that the possibility of such catastrophic failures that affect plant safety can be practically eliminated (the second stage evaluation), then inservice inspection may be relaxed to simple system monitoring. The specific technology used for the second stage evaluation in this case is leak-before-break evaluation. Since sodium-cooled fast reactors are operated in a low pressure and their components are made of ductile materials, leak-before-break holds naturally for most of the components.

Leak-before-break evaluation procedures have been published for light water reactors [13, 14]. RCC-MR [15] provides technologies available for leak-before-evaluation of sodium-cooled fast reactors. Extensive research and development have also been conducted on the development of leak-before-break procedures for the piping systems of sodium cooled fast reactors in Japan [16-18]. The authors are developing a procedure for leak-before-break evaluation of sodium-cooled fast reactor components as one of the codes that materialize the SBC concept. The procedure would provide basis for the determination of inservice inspection requirements for sodium boundaries. In order to highlight the characteristics of the procedure we are developing, it would be helpful to compare it to one of the existing ones, for example, the Section 3.6.3 of NUREG-0800 [13]. Those two have both common aspects and different natures. They are common in that both present specific procedures according to which we are able to evaluate if unstable fracture is the case or not under specific conditions (note that procedures themselves are independent and they adopt separate approaches). At the same time, they have quite different natures, besides that NUREG-0800 is issued by a regulatory body. NUREG-0800 provides leak-before-evaluation procedures as a tool that can be used only for judging if the dynamic effects of the pipe ruptures postulated in Standard Review Plan section 3.6.2 has to be considered. It requires relevant inservice inspections as a prerequisite for applying the procedure. On the contrary, the procedure the authors are developing is for the determination of inservice inspection requirements.

The leak-before-break procedures will be tied in a set of inter-related codes for sodium-cooled fast reactors; design and construction code, welding code, fitness-for-service code, leak-before-break evaluation code, and the guidelines for structural reliability evaluation of passive components, as shown in Fig. 7. The issuance of those codes is expected in JSME in 2016. The structural reliability evaluation guidelines would be established based on elevated temperature fracture mechanics that allow such evaluations that are required in the second stage to the SBC process.

CODE DEVELOPING ACTIVITIES

The extension of the allowable stresses of 316FR and Mod.9Cr-1Mo steels up to 500,000 hours is planned in the 2016 Edition of the JSME FR Code. The issuance of a set of codes for sodium-cooled fast reactors that allows the utilization of the SBC concept in a manner described in this paper is also expected in JSME in 2016. Activities to develop appropriate inservice inspection requirements are concurrently going on in the ASME Boiler and Pressure Vessel Code Committee. The JSME/ASME Joint Task Group for System Based Code, established in 2012, is working on developing a Code Case that provides alternative requirements to the current Section XI Division 3, inservice inspection requirements for liquid metal reactors. The Code Case is being developed utilizing the SBC concept. The essence of it will in harmonization with the inservice inspection requirements implemented in the JSME fitness-for-service code for fast reactors.

CONCLUSIONS

A set of codes are being developed in the Japan Society of Mechanical Engineers to allow a 60-year design of sodium-cooled fast reactors. They consist of design and construction code, welding code, fitness-for-service code, leak-before-evaluation code, and the guidelines for structural reliability evaluation of passive components. Allowable stresses for up to 500,000 hours will be determined based on a database that covers long-term regions so that the magnitude of extrapolation would not be too large. Schemes to identify actual margins of the component of the plant would be implemented. Issues that could be critical in sodium-cooled fast reactors such as creep-fatigue damage assessment and the evaluation of welded joints are most carefully addressed with obtaining as many data as possible in long-term regions. Moreover, inservice inspection requirements which are most appropriate to sodium-cooled reactors in terms of safety and reliability would be derived by utilizing the System Based Code concept.

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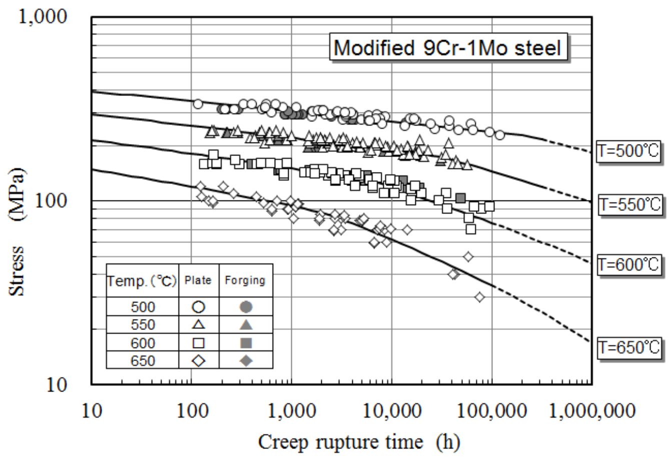

Fig. 1 Formulation of creep rupture time for Mod.9Cr-1Mo steel

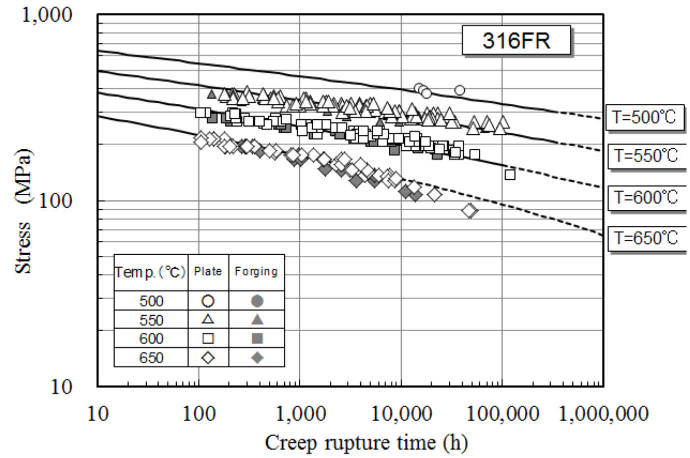

Fig. 2 Formulation of creep rupture time for 316FR steel

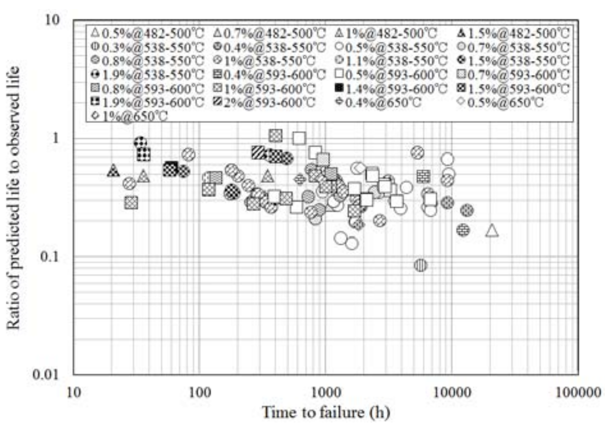

Fig. 3 Margins in creep-fatigue evaluation of Mod.9Cr-1Mo steel

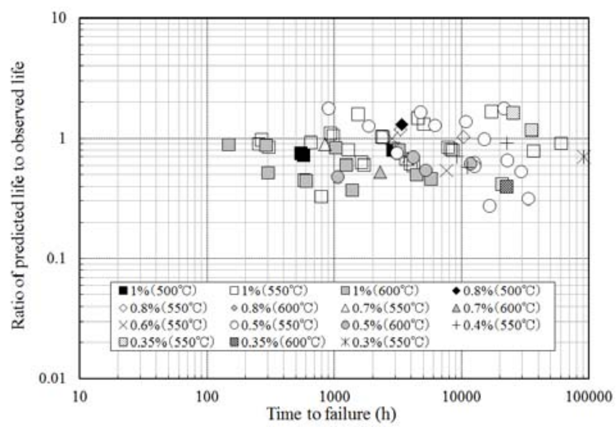

Fig. 4 Margins in creep-fatigue evaluation of 316FR steel

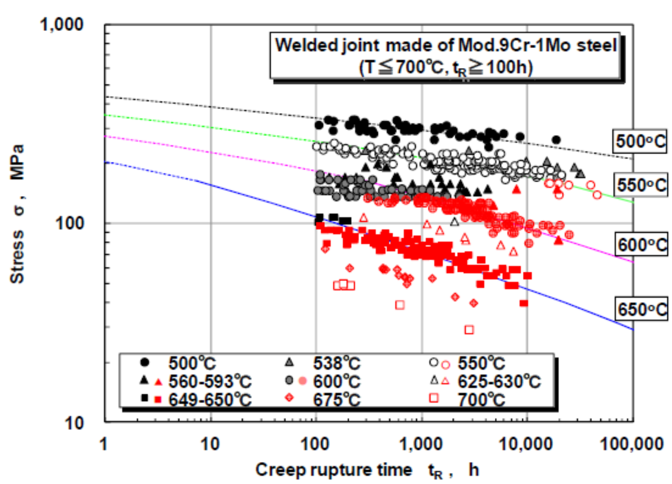

Fig. 5 Data used for the determination of welded joint strength reduction factors

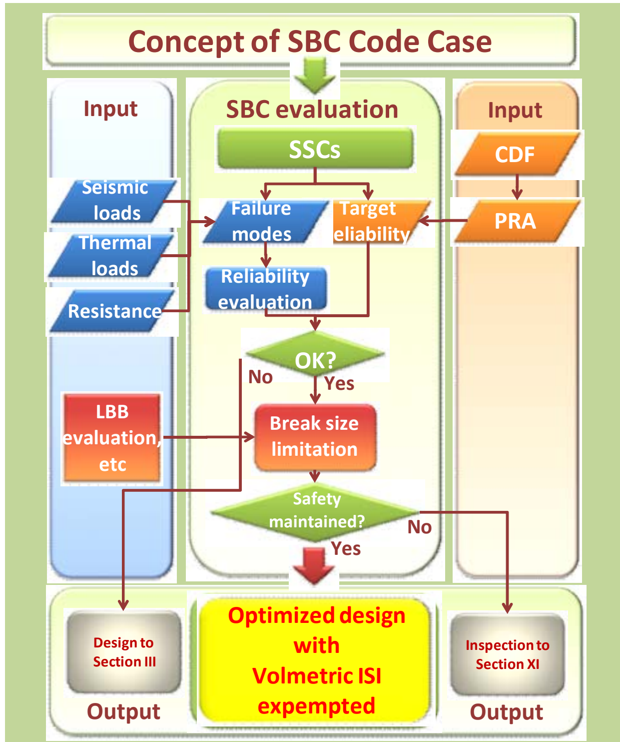

Fig. 6 A logic flow diagram for the determination of ISI requirements based on the SBC concept

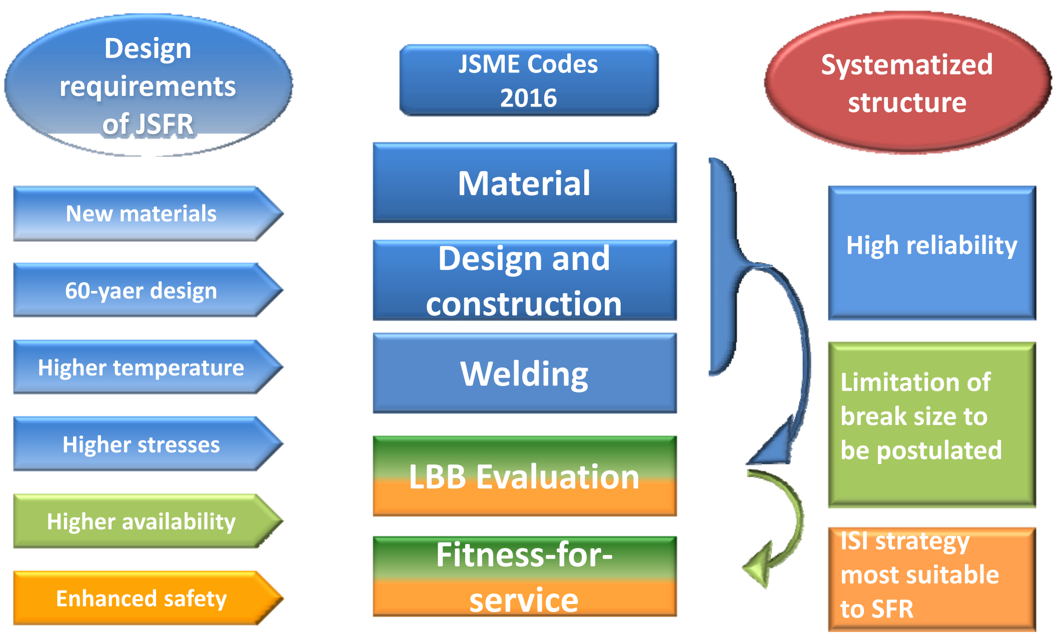

Fig.7 Code structure planned in the 2016 Edition of the JSME Fast Reactor Codes