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RECENT DEVELOPMENTS FOR FAST REACTOR STRUCTURAL DESIGN STANDARD (FDS)

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ABSTRACT

For realization of reliable and economical fast reactor (FR) plants, Japan Nuclear Cycle Development Institute(JNC) and Japan Atomic Power Company(JAPC) are cooperating on “Feasibility Study on Commercialized FR Cycle Systems”. To certify the design concepts through evaluation of their structural integrity, the research and development of “Elevated Temperature Structural Design Guide for Commercialized Fast Reactor (FDS)” is recognized as an essential theme. FDS focuses on particular failure modes of FRs such as ratchet deformation and creep fatigue damages due to cyclic thermal loads. To evaluate these modes, three main developments are in progress. One is “Refinement of Failure Criteria” for particular modes of FRs. Next is development of “Guidelines for Inelastic Design Analysis” in order to predict elastic plastic and creep behaviors. Furthermore, efforts are being made toward preparing “Guidelines for Thermal Load Modeling” for FR component design where thermal loads are dominant. These studies were performed under the sponsorship of the Ministry of Economy, Trade and Industry of Japanese government.

KEY WORDS: Fast Reactor, Structural Design Standard, Failure Criteria, Inelastic Analysis, Thermal Load

1. INTRODUCTION

The Japan Nuclear Cycle Development Institute (JNC) and Japan Atomic Power Company (JAPC) are cooperating to conduct a research and development program called the “Feasibility Study on FBR Cycle” (Feasibility Study) for analyzing the practical application of the fast breeder reactor (FBR) cycle [1]. This study aims at the development of a reactor system that implements the concept of a plant with superior safety and economy. For this purpose, such plant designs are tried as the compact and simple nuclear reactor structure, shortening of the piping, and the use of new structures such as the intermediate heat exchanger combined with the pump. To achieve drastic rationalization of above design, sophistication of the structural design standard was required. Therefore, the JNC and JAPC have been conducting research and development program for developing “Elevated Temperature Structural Design Guide for Commercialized Fast Reactor (FDS)” since 2000 on a contract basis with the Ministry of Economy, Trade and Industry. This report describes the recent progress of this joint research.

The first established fast reactor structural design code in Japan is the “Elevated Temperature Structural Design Guide for Class 1 Components of Prototype Fast Breeder Reactor (BDS) [2]” which was developed by extension of the ASME B&PV Code Sec. III CC1592[3]. Introducing additional developments of materials and structural analysis methods to BDS, the “Elevated Temperature Structural Design Guide for Demonstration Fast Breeder Reactor (DDS) [3]” was developed.

Furthermore, commercialized fast reactors have different design needs from prototype and demonstration fast breeder reactors. To satisfy above needs, new design methods are developed and integrated with DDS into

FDS with two sets of guidelines for inelastic design analysis and for thermal load modeling.

2. MAIN R&D ISSUES FOR STRUCTURAL DESIGN OF COMERCIALIZED FAST REACTOR

Particular characteristics of fast reactor structural design are low pressure and high temperature conditions due to the use of liquid metal as coolant. Under a high-temperature condition, the thermal stress induced by the temperature difference in the structure increases; further, the elastic plastic and creep deformation easily occur due to a decrease in the yield strength of the material. Additionally, under a low-pressure condition, the possibility of ductile fracture and creep failure is low. Consequently, ratchet deformation and creep-fatigue damage due to cyclic thermal stress become dominant failure modes in fast reactor components (Figure 1).

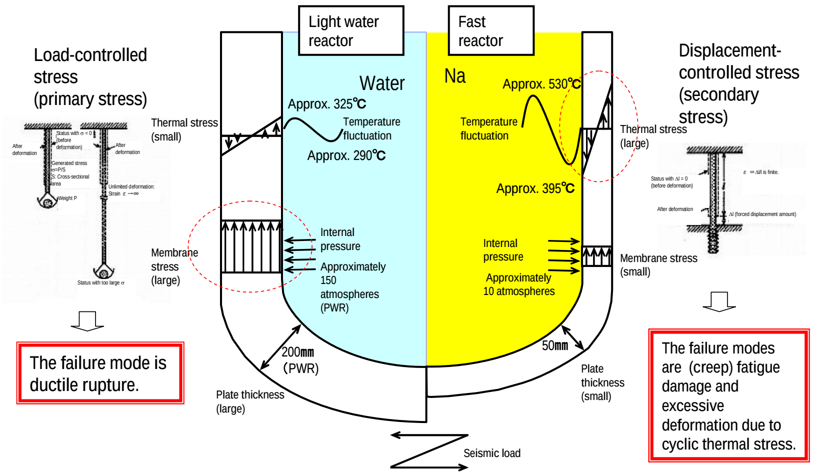


Figure 1 Dominant failure modes in FR plants

Design research is being conducted on a sodium-cooled commercialized reactor as feasibility study. A compact and simple plant design shown in Figure 2 is examined for improving its reliability and economy from prototype and the other available demonstration reactors. Abbreviation of protection equipments for the reactor vessel wall (reactor wall cooling system, etc.) increases the thermal stress on the relevant section. Further, in a small, thin-walled vessel, the primary stress for supporting the core weight overlaid on the reactor vessel wall becomes larger than that observed in previous fast reactors. These loading conditions enhance ratchet deformation and (creep) fatigue damage at a vessel wall around liquid free surface (①) and at a lower part of vessel (②).

With regard to the cooling system, adoption of an integrated primary pump and intermediate heat exchanger reduces the heat capacity of coolant against power output whereas the flow velocity increases, resulting in a severe thermal transient load (③). Reduction in the number of loops enhances heat transfer coefficient from high flow rate. It increases risk of high cycle thermal fatigue of pipes which is typical problem at mixing zone of high and low temperature fluid (④).

In order to overcome above structural design problems, FDS introduces several new technologies such as Figure 3. Some severe parts of reactor vessel and IHX locate in cold leg where normal operating temperature is under creep regime. To treat these parts as non-creep design area, classification method of high-temperature design area is improved (A). When both ratchet strain and fatigue damages increase, there is a possibility of those interaction. Failure criteria considering above interaction is investigated (B). To predict inelastic response of structures preciously, inelastic constitutive equations(C) and design evaluation method based on inelastic analysis results (D) are recommended. In spite of severer thermal loads, conventional design guides for FRs has no rule for thermal load modeling. Load modeling methods are studied for system thermal transient load (E) and thermal striping load (F).

Above developments are integrated with DDS into FDS which consists of “Elevated temperature structural design guide”, “Guidelines for Inelastic Design Analysis” and “Guidelines for Thermal Load Modeling”.

“Elevated temperature structural design guide” is DDS plus (A) and (B) with other improvements related with failure criteria. (C) and (D) are related with analysis methods and are different category from failure criteria, therefore those are arranged as “Guidelines for Inelastic Design Analysis”. (E) and (F) are concerning load modeling, so that those are put into “Guidelines for Thermal Load Modeling”. These two set of guidelines are referred from “Elevated temperature structural design guide”.

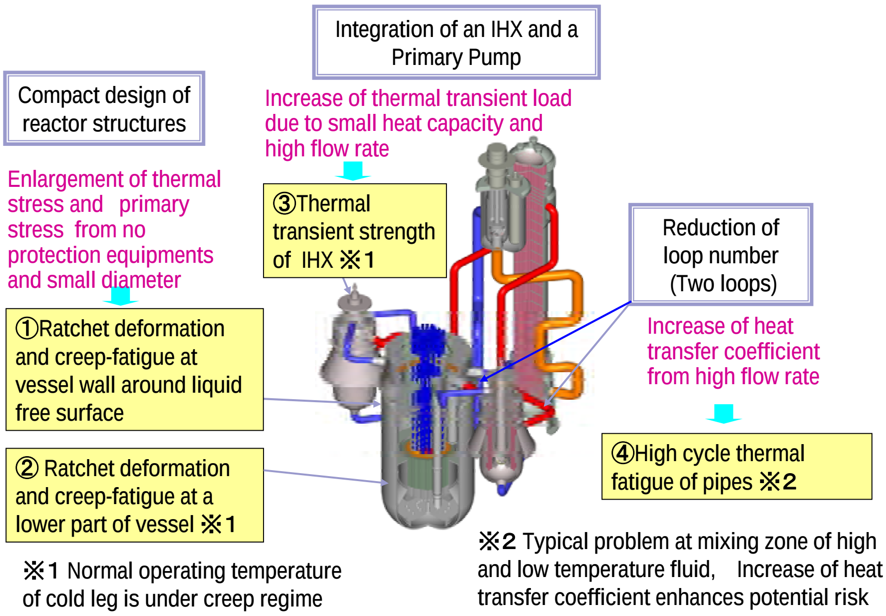


Figure 2 Structural design problems of commercialized fast reactor

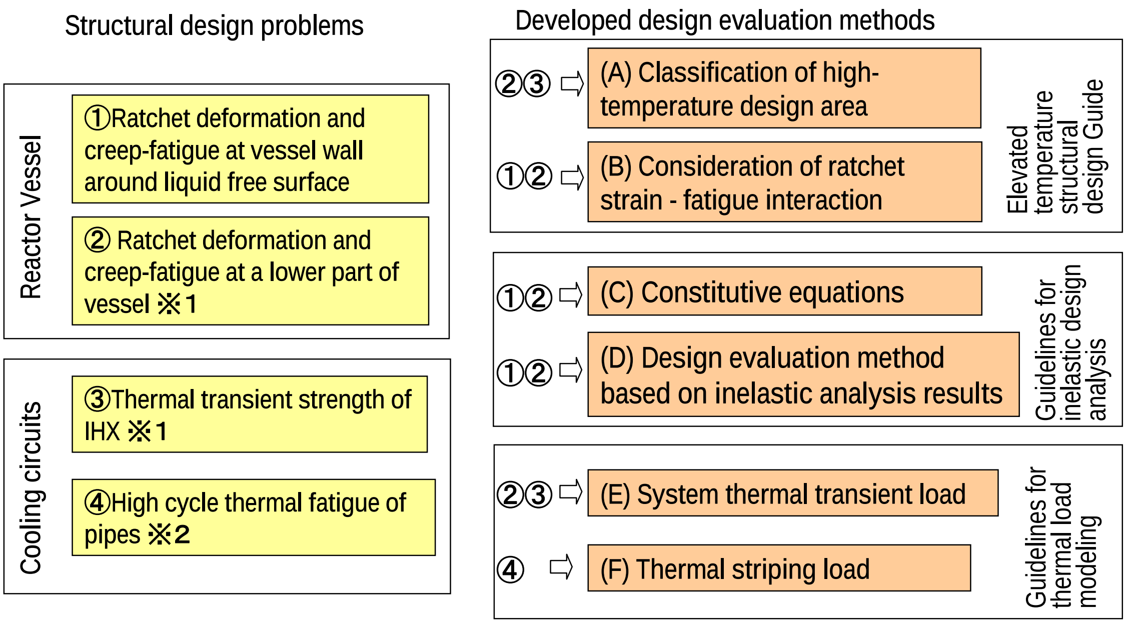


Figure 3 Developed design evaluation methods to overcome above problems

3. ELEVALTED TEMPERATURE STRUCTURAL DESIGN GUIDE

3.1 Classification of high-temperature design area

In the existing BDS[2] and DDS[4], the creep design area in a plant system is judged based on a comparison of the highest temperature during operation and a uniform limiting value (425°C for austenitic steel such as 316FR steel and 375°C for ferrite steel such as 12Cr steel) regardless of operating time. The influence of creep is frequently over-estimated in sections for which operation time at a high temperature is short. There are critical components that are normally used in a relatively low-temperature area and for which the high-temperature hold time is short (※ in Figure 2). Therefore, it needs to develop rational classification method for the high-temperature design area.

The French RCC-MR[5] code adopts a more rational method of judgment than the Japanese BDS and DDS by combining temperature and time. The actual creep strength depends on the temperature, time, and stress. In the current research, a rational method for judging the creep design area is examined based on the combination of temperature, time, and stress.

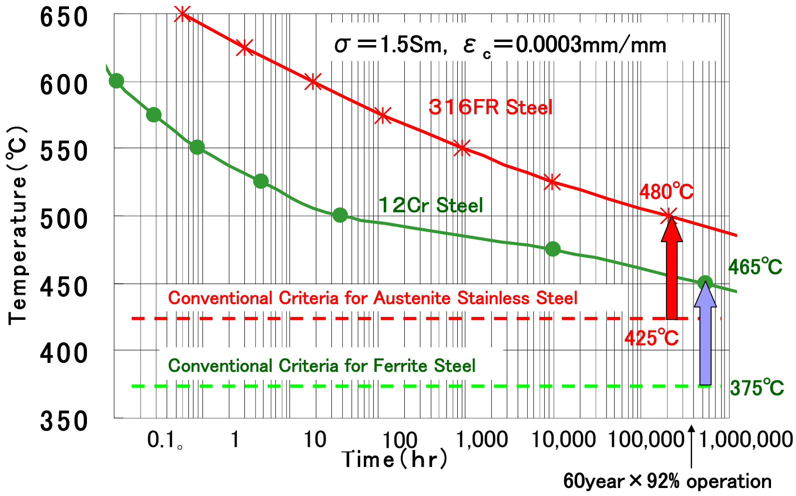


Figure 4 Negligible Creep) curve

3.2 Consideration of ratchet strain - fatigue interaction

In FDS, a rational limit is assigned to ratchet strain. On the other hand, recent researches suggest that a ratchet strain of even 1% to 2% may influence fatigue life or creep-fatigue life. Consequently, it becomes necessary to understand the influence of the ratchet on creep-fatigue life in the operation conditions of fast reactors. So that, ratcheting-fatigue tests were conducted to examine the influence of ratchet on creep-fatigue strength.

Figure 5 shows the concept underlying the test. The x-axis indicates the number of cycles of strain, while the y-axis indicates the accumulated amount of strain. In the ratchet-fatigue test, cyclic strain at a constant amplitude and ratchet strain (accumulated strain) are overlaid. The following load patterns can be considered depending on the method of applying ratchet strain in the overall life. (I) Strain is given as pre-strain in the initial stage. Based on the existing knowledge, a decrease in lifetime due to the existence of mean strain is not found. (II) In the overall life, strain is increased gradually. Recent tests indicate a significant decrease in lifetime. (III) As an intermediate between the above two patterns, ratchet strain is gradually increased until the number of cycles reached No; subsequently, saturation occurs. For load pattern (III) described above, we obtained the result of a preliminary test using the number of cycles until saturation occurs as a parameter.

Test data obtained until now are shown in Figure 5. The x-axis of the figure indicates the ratio of the number of cycles until saturation of ratchet strain occurs N_0 to the lifetime at fatigue test Nf_0. The y-axis indicates the ratio of the lifetime at the ratcheting-fatigue test N_f to the lifetime at fatigue test N_{f0}. Values smaller than 1 along the y-axis indicate a decrease in fatigue strength due to the ratchet. In the test, we changed N_0 and examined the variation in strength due to the change. Other conditions such as material (316FR steel), temperature (550°C), strain range (0.5%), strain rate (0.1 %/s), and ratchet strain (1.41%) were held constant.

From test results of Figure 5, we judged the ratchet strain lower than 2% has no influence on the fatigue lifetime when N0=20, 1000. N0=1000 is conservative condition compared with design conditions where thermal transient number is less than several hundreds. On the other hand, fatigue strength decreases if ratchet strain is applied during the overall life.

Since above results, FDS limits accumulated inelastic strain within 2% considering ratchet strain - fatigue interaction.

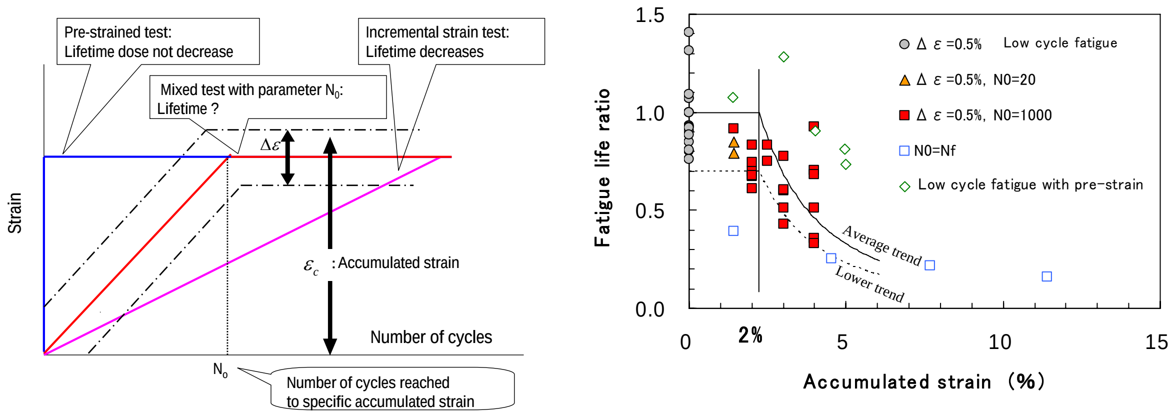


Figure 5 Summary of ratcheting-fatigue test conditions and results

4. GUIDELINES FOR INELASTIC DESIGN ANALYSIS

4.1 Basic policy and scope

Although the application of the inelastic analysis methodology to design has been tried since 1970[6] little progress has been made. One of the major reasons for this is lack of definitive constitutive equations for describing elastic-plastic-creep behavior of materials. Furthermore, the detailed constitutive equations that were developed for the purpose of realistic modeling of inelastic behavior of materials can accurately simulate test results, for which the material characteristics and load history are known. However, these conditions are uncertain in the design phase, which results in insufficient use of the advantage. That is another major reason.

The basic policy of developed guidelines is more rational evaluation result than that obtained by existing elastic analysis, and a conservative to design conditions, including essentially uncertain factors [7]. Since inelastic analysis has many influence factors on the results, it is difficult to secure maintenance of general conditions. Therefore, the scope of this set of guidelines is limited by considering the features of materials and load of fast reactors to avoid this problem.

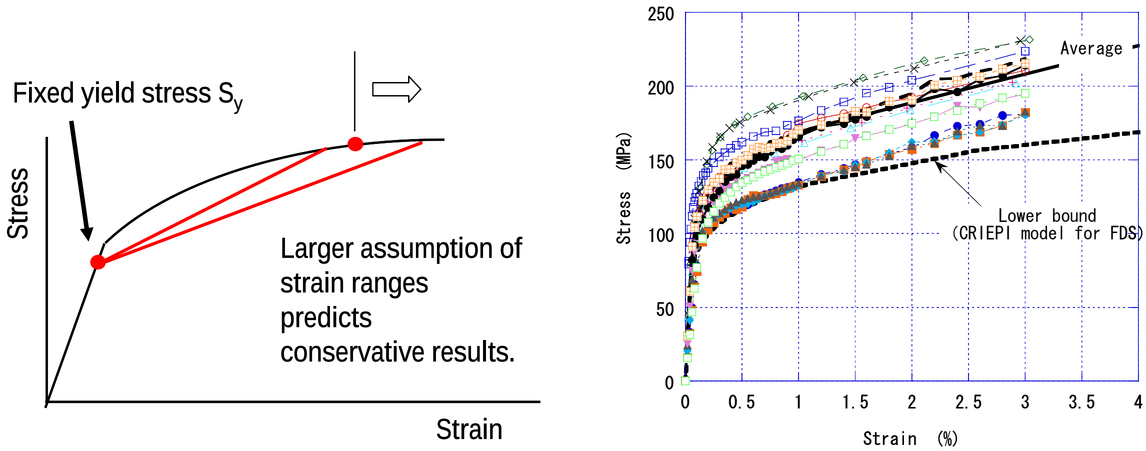
The material used is assumed to be austenitic stainless steel (ex.316FR). Loading conditions are low primary stress ones such as design conditions in sodium systems of fast reactors that receives basically displacement-controlled load. Failure modes to be evaluated are assumed to be ratchet deformation and (creep) fatigue damage.

4.2 Constitutive equations

For fast reactor design by inelastic analysis, the two kinds of constitutive equations are recommended. One is “Bi-linear model” for easy treatment of inelastic calculation. Another is “CRIEPI model for FDS” to achieve precious estimation. Ratchet deformation and (creep) fatigue damage mainly depend on strain. Therefore, above constitutive equations are made use to predict larger (conservative) strain, since there are no constitutive equations that can evaluate both conservative stress and strain.

“Bi-linear model is one of the familiar constitutive equations. Therefore, it is easy to be used and often is provided as a standard option in many popular analysis codes such as ABAQUS, MARC and FINAS. This classic model can be simply described, which enables easy understanding. Main influence factors on strain of this model are work hardening, cyclic hardening and Bauschinger’s effect. All of these factors are modeled to predict larger strain. Bi-linear approximation of stress-strain curve is made as Figure 6 for conservative consideration of work hardening and cyclic hardening. When an assumed strain range is larger than an obtained result, it is a conservative solution. For conservative treatment of Bauschinger’s effect, a kinematic hardening model with restriction of yield surface moving is adopted.

“CRIEPI model” is a precious constitutive equation which can consider nonlinear stress-strain relationships, cyclic hardening and temperature dependency [8]. Its model is modified into “CRIEPI model for FDS” to simulate lower bound of both monotonic and cyclic stress-strain curves with 99% reliable range.



(a) Bi-linear model (b)CRIEPI model for FDS

Figure 6 Recommended Constitutive Equations

4.3 Design evaluation method based on inelastic analysis results

(1) Modeling of the load history

Since inelastic analysis results depend on the load history which is undetermined in the design stage. Design load history should be modeled to bound capable loading conditions of actual plants.

The realistic load modeling procedure is proposed for conservative prediction as shown in Figure7. Conventional design codes based on elastic analysis, usually define unit load cycle composed of load events that can determine stress range for ratcheting and fatigue evaluation. The first step of load modeling procedure is extraction of unit load cycles, whose stress ranges exceed shakedown limit. Furthermore, kinds of unit loads are reduced by enveloping small unit loads with larger ones. The second step is to envelop all unit loads with the largest one. This assumption is appropriate for the Reactor vessel adjacent to coolant surface level, where the design-based event making stress over the shake-down limit is only “Plant start-up”, although various design-based events exist. That eliminates the difficulty of the load history effect. When above assumption becomes over conservative, more than two kinds of unit load cycles are necessary to be considered. Step3 is arrangement of order of different unit loads to predict larger strain by minimizing number of stabilized stress-strain cycles. Minimization of repeat numbers of the same unit load cycle, for example, can meet this requirement.

(2) Creep fatigue damage evaluation method

Creep fatigue damage is evaluated by both stress and strain, however there are no constitutive equations which provide both conservative stress and strain. Therefore, only strain is utilized from inelastic analysis results and stress is estimated indirectly from strain.

Fatigue damage is estimated by the equivalent strain range and the fatigue curves with Miner’s rule. The equivalent strain range is obtained from inelastic analysis results.

Creep damage is evaluated based on stress with the time fraction rule. It is difficult to estimate creep damage directly from inelastic analysis of which constitutive equation tends to underestimate stress. Therefore, a conservative stress evaluation method based on the cyclic stress-strain relationship is introduced as in figure 8 to evaluate conservative stress from inelastic calculated strain. Since initial stress of relaxation is not always peak value in actual design condition. Stress evaluation method in figure 8 can consider such situation.

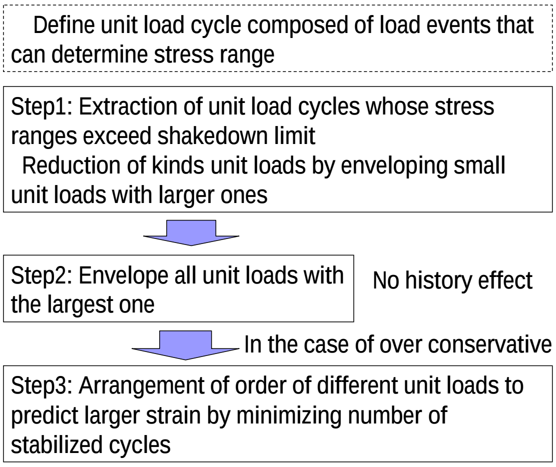


Figure 7 Modeling of the load history

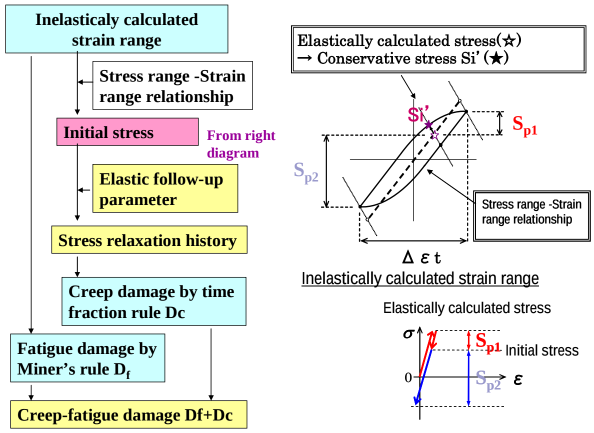


Figure 8 Creep-Fatigue Evaluation Procedure

5. GUIDELINES FOR THERMAL LOAD MODELING

5.1 System thermal transient load

Thermal stress generated by the temperature variation of the coolant that is caused by the operation status of plants is called "system thermal transient load". It is one of the principal loads for FR components. This load is affected by many kinds of parameters such as "system parameters". Since each effective factors has a variable range as in figure 9, "system thermal transient load" should consider the most severe case among combinations of effective factors.

Conventional design procedures evaluate thermal-hydraulic and structural characteristics individually. For estimating the former, tendency of the coolant temperature change to the variation of the effective factors is obtained from parametric thermal-hydraulic analysis. In order to envelop the above tendency, coolant temperature histories are conservatively approximated by multi-linear curves with design factors. This thermal transient conditions are then passed to the structure design side, and geometries possible in this condition are identified from the structural analysis in this procedure (Multi-linear approximation in Figure 10).

On the other hand, a method for directly obtaining the relation between the "effective factors" and generated thermal stress, based on thermal-hydraulic-structure total analysis has been developed [9]. This procedure can model "system thermal transient load" without conservative design factors (Design of experiments in Figure 10). Basic idea is to find the combination of effective factors which generate the maximum thermal stress among the all of there combinations. This idea requires to grasp relationship between effective factors and thermal stress, therefore difficulties are enormous number (more than several thousand) of thermal-hydraulic-structure total analyses. In order to reduce the number of analysis cases and computational time, "Design of experiments (variation of Taguchi method)[10] " and thermal-hydraulic-structure total analysis code are adopted. "Design of experiments" is a technique of using the orthogonality of condition allocation and obtaining the influence of multiple factors on the results with a reduced number of trials. Further, the influence of the factors on the results can be quantitatively evaluated with the function of sensitivity analysis. The thermal-hydraulic-structure total analysis code executes the response calculation of the coolant temperature variation and thermal stress using the input of various system parameters.

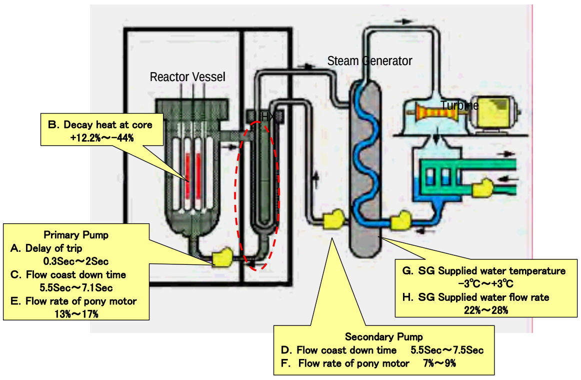


Figure 9 Effective factors on system thermal transient load

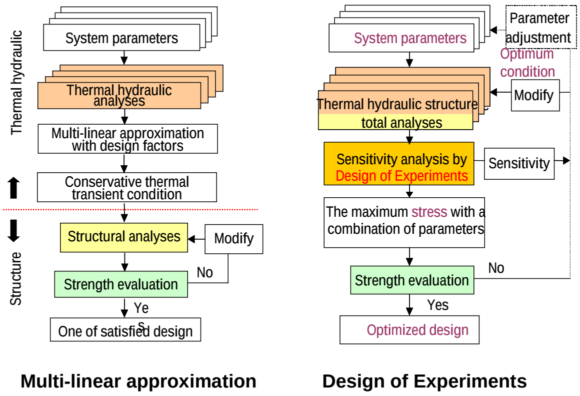


Figure 10 Evaluation Procedure for load modeling

5.2 Thermal striping load

Recent investigation on thermal striping has revealed that amplitude of temperature fluctuation was attenuated during a series of fluids mixing to structural response by (1)turbulent diffusion, (2)heat transfer and (3)heat conduction as in figure 11 [11]. Above attenuations depend on frequency of temperature fluctuation. A wall model subjected to sinusoidal fluctuation of fluid temperature can explain this mechanism[12]. If a frequency of fluctuation is very low, whole temperature of the wall easily respond to fluid temperature, because thermal diffusivity homogenizes structural temperature. Therefore, low frequency fluctuations do not induce large thermal stress that is caused from temperature gradients in structures. On the other hand, a wall surface cannot respond to very high frequency fluctuation, since a structure has a finite time constant of thermal response due to heat capacity. Therefore, high frequency fluctuations do not lead to large thermal stress. As a result, there are special frequency area that are damageable.

“Thermal striping load” is evaluated by 4 step screening rules considering above attenuation factors. The first step is simple and conservative method without consideration of attenuation factors. The second step takes attenuation by only turbulent diffusion into account. The third one considers (1)turbulent diffusion, (2)heat transfer and (3)heat conduction, however conservatively bounds frequency characteristics. The forth one makes consideration of all attenuation factors with frequency characteristics. Among them, the first, second and the third steps are based on the same idea as the JSME guideline for light water reactors [11]. The forth one is explained here since it is new proposal.

In order to consider frequency effect, power spectrum density function (PSD) quantified frequency characteristics of fluid temperature and thermal stress fluctuations. Figure 12 illustrates the thermal stress evaluation procedure based on PSD. Required input data is ordinal design specifications such as flow rates, diameters, material properties and wall thickness of pipes ①. According to momentum ratio between main and branch pipes which is calculated from flow rates and diameters, flow patterns are classified into three patterns such as wall jet, deflecting jet, and impingement jet [13] ②. For each pattern, design chart gives Non-dimensional PSD of fluid ③. Frequency transfer function from fluid temperature to thermal stress can be theoretically derived from heat transfer coefficient, material properties, wall thickness and constraint condition [12] ④. This function transfers PSD of fluid to PSD of stress ⑤. By using inverse Fourier transformation, PSD of stress is transformed into time history of stress. In this step, random phase is assumed in order to simplify the mixing database in fluid ⑥. The rain flow wave counting method extracts stress ranges and cycles from time history of stress ⑦. From above data, fatigue damage factor is evaluated based on Miner’s rule ⑧.

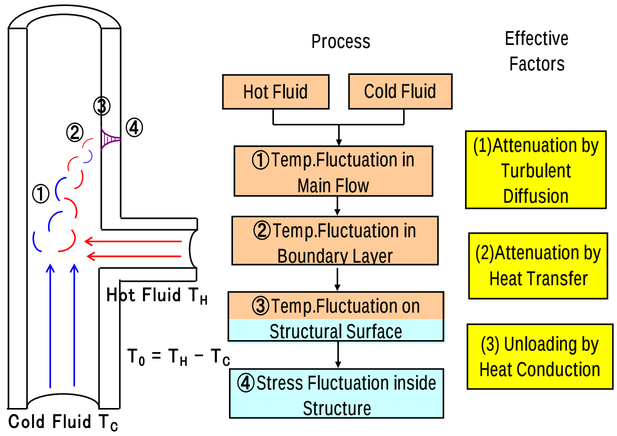


Figure 11 Thermal Fatigue Mechanism Induced by Fluid Temperature Fluctuation

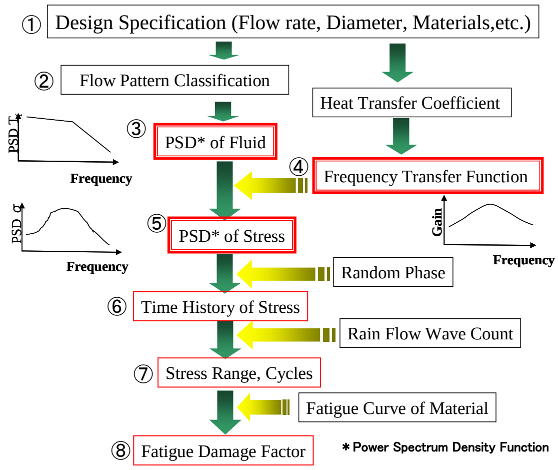


Figure 12 Thermal Stress Evaluation Procedure *by Power Spectrum Density Function*

6. CONCLUSIONS

Japan Nuclear Cycle Development Institute(JNC) and Japan Atomic Power Company(JAPC) are cooperating to develop “Elevated Temperature Structural Design Guide for Commercialized Fast Reactor (FDS)”. New design methods to satisfy design requirements of commercialized fast reactors are integrated with “Elevated Temperature Structural Design Guide for Demonstration Fast Breeder Reactor (DDS)” into FDS which consists of “Elevated temperature structural design guide”, “Guidelines for Inelastic Design Analysis” and “Guidelines for Thermal Load Modeling”.

“Elevated temperature structural design guide” is DDS plus (A) classification method of high-temperature design area and (B) failure criteria considering ratchet strain - fatigue interaction, with other improvements related with failure criteria. “Guidelines for Inelastic Design Analysis” recommends (C) constitutive equations and (D) design evaluation method based on inelastic analysis results. “Guidelines for Thermal Load Modeling” provides modeling methods for (E) system thermal transient load and (F) thermal striping load.

Adequacies of new developments were proved for particular design cases. To extend these applicable area, experimental validation are continued. Details and back ground of above guidelines will be presented by associated papers.

7. AKNOWLEDGEMENT

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