Title	Determination of in-service inspection requirements for fast reactor components using System Based Code concept
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Title. Determination of In-service Inspection Requirements for Fast Reactor Components Using System Based Code Concept

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Abstract

In our previous study (Takaya, et al. 2015a), we proposed a new process for determining the in-service inspection (ISI) requirements using the System Based Code concept. The proposed process consists of two complementary evaluations, one focusing on structural integrity and the other on plant safety. First, the structural reliability of a specified component is evaluated considering potentially active degradation mechanisms, including those that are not explicitly addressed in the design codes. If the structural reliability meets the requirement, the second evaluation can be conducted, which assesses the detectability of defects before they can grow to an unacceptable size, taking plant safety into account. If there is any feasible way to detect defects, it is adopted as an ISI requirement. Otherwise, a structural integrity evaluation would be required under a sufficiently conservative hypothesis. In other words, if the additional requirements are met, detectability is not an obligation.

In this study, the ISI requirements for a reactor guard vessel (RGV) and core support structure (CSS) of a prototype sodium-cooled fast breeder reactor in Japan (Monju) were investigated using the proposed process. Creep-fatigue and fatigue were chosen as the potentially degradation mechanisms for the RGV and CSS, respectively. The Stage I evaluations using the Monte-Carlo method showed that both components had sufficient reliability if these degradation mechanisms were considered. At Stage II, the reliability levels of the components were evaluated assuming initial fully circumferential cracks with a depth equal to 10% of the thickness as additional requirements because there was no available inspection method for the components. It was shown that both components had sufficient reliability even with the additional requirement based on conservative hypothesis. The failure occurrences of these components were practically eliminated. Hence, it was concluded that no ISI requirements were needed for these components.

The proposed process is expected to contribute to the realization of effective and rational ISI by properly taking into account plant-specific features.

Keywords.

In-service inspection, Reliability, Creep-fatigue, Crack initiation, Crack propagation

1 Introduction

In-service inspection (ISI) is important for the safety and stable operation of nuclear power plants. ISI requirements should be determined while taking into account the specific facilities and safety design of each plant. Sodium-cooled fast reactors (SFRs) have some features that are different from those of conventional light water reactors, such as operation at temperature above which creep effect occurs, hereinafter, referred to as elevated temperature, a low internal pressure, and almost negligible corrosion in the purity-controlled sodium. Efforts to develop rules for ISI in liquid-metal-cooled plants were made alongside the Clinch River Breeder Reactor project, and resulted in Section XI, Division 3 of the American society of mechanical engineers (ASME) boiler and pressure vessel (B&PV) code, hereinafter referred to as Section XI, Division 3 (ASME 2001). However, because of the cancellation of the Clinch River project, no major revisions have occurred since the first publication in 1980, and some parts of the code such as the acceptance standards for the examination of Class 1 and 2 components still need to be prepared. Therefore, the Japan society of mechanical engineers (JSME)/ASME Joint Task Group for System Based Code (SBC) was established in 2012 by the ASME B&PV Code Committee, and is working to develop alternative requirements to Section XI, Division 3, utilizing the SBC concept (Asayama, et al. 2014).

In our previous study, a new process for determining ISI requirements was proposed on the basis of the SBC concept (Takaya, et al. 2015a). The SBC concept was proposed in the course of the research and development for Japanese fast breeder reactors in the 1990s (Asada, et al. 2002a, Asada, et al. 2002b, and Asada 2006). One of the key concepts is margin optimization, which provides a new framework intended to allow the optimum allocation of margins for the structural integrity of components encompassing various technical aspects in a plant life cycle, such as the material, design, fabrication, installation, and inspection, as well as the repair and replacement. By fully taking these technical characteristics into account, the SBC concept pursues improved reliability and economy while meeting the plant safety goals. The logic flow diagram of the proposed process is shown in Figure 1. It consists of two complementary evaluations, one focusing on structural integrity (Stage I) and the other on plant safety (Stage II). In Stage I, the structural reliability of a specified component is evaluated considering potentially active degradation mechanisms, including those that are not explicitly addressed in the design codes. For example, outer surface corrosion of low-alloy steels in air might have to be considered as a potentially active degradation mechanism in some cases although it is not covered by the design codes. A general measure of structural reliability is probability of structural integrity loss, for example, a breach of coolant boundary. If the structural reliability meets a requirement, it is possible to proceed to Stage II, where an assessment is made of the detectability of defects before they grow to an unacceptable size in consideration of plant safety. If there is any feasible way to reliably detect service degradation, it is adopted as an ISI requirement. Otherwise, a structural integrity evaluation would be required while employing a sufficiently conservative hypothesis. In other words, if the additional requirements are met, detectability is not an obligation.

This study investigated the ISI requirements for a reactor guard vessel (RGV) and core support structure (CSS) of a prototype sodium-cooled fast breeder reactor in Japan (Monju), based on the proposed SBC process.

2 Evaluation Object Components

2.1 RGV

Figure 2 shows the main components of the Monju reactor. A RGV is one of the unique components of a SFR. It envelops the reactor vessel (RV) and some parts of the components of the primary heat transfer system (PHTS). The volume between the RGV and the components inside it was designed to maintain the reactor coolant surface level required for removing the decay heat in the event of a coolant leakage from either the RV or the PHTS. In addition, the area for the PHTS is filled with purity-controlled nitrogen gas to prevent sodium fires. The material of the RGV is SUS304, which is equivalent to type 304 stainless steel. The maximum temperature at a high stress region during normal operations is approximately 445°C. The main loads are the dead load and that due to thermal transients during the start-ups and shutdowns.

Section XI, Division 3 gives RGVs a Class 2 classification, and Visual Testing for liquid-Metal cooled plants (VTM)-3 examinations of welds are required for austenitic and low-alloy steel vessels. A VTM-3 examination is a visual examination, which is used to determine the general mechanical and structural conditions of components and their supports and to detect discontinuities and imperfections. On the other hand, even a small through-wall crack leading to coolant leakage is not allowable for an RGV because of the required safety function of maintaining the reactor coolant surface level. In general, some physical displacements are assumed around regions where large cracks exist. Detection of physical displacements by VTM-3 will contribute to finding the cracks. Thus, a VTM-3 examination is considered to be effective to some extent. However, it is not certain whether VTM-3 examinations are suitable to prevent the penetration of cracks of any length because physical displacement might be too slight to be detected especially for small cracks. Therefore, the ISI requirements for an RGV are derived according to the proposed process on the basis of the SBC concept.

The first step is to determine potentially failure modes, which have to be exhaustively analyzed even if they are not explicitly addressed in the design code. The RGV is filled with and surrounded by purity-controlled nitrogen gas. Thus, corrosion can be excluded. Because the flow of nitrogen gas is low velocity, and there are no vibration sources, vibration fatigue can also be excluded. This leaves creep-fatigue interaction damage because the RGV is used at elevated temperature, and cyclic loads are produced during the reactor start-ups and shutdowns.

A target reliability is also needed for the determination of the ISI requirements according to the proposed process. Kurisaka et al. (2011) proposed a method for determining the required structure- and component-level reliabilities from the quantitative safety design requirements for the core damage frequency (CDF) and containment failure frequency (CFF) utilizing probabilistic safety assessment (PSA) models. They applied the proposed method to the Japan Sodium-cooled Fast Reactor (JSFR), which was a demonstration reactor being developed. The quantitative safety design requirements for the CDF and CFF for the JSFR were assumed to be 10^{-5}/site-year and 10^{-6}/site-year, respectively (Kotake et al. 2008). Kurisaka et al. (2011) assumed that there were 10 reactors at a single site. Thus, the CDF and CFF requirements for a single reactor were found to be 10^{-6}/reactor-year and 10^{-7}/reactor-year, respectively. The boundary failure of the

RGV is related to a loss of the reactor coolant surface level required for the decay heat removal, which is one of the typical event sequences in a SFR leading to core damage. The derived target reliability of the RGV was 2×10^{-5}/reactor-year. Some of the design features of the JSFR are different from those of Monju (Yamano et al. 2012), and there is only a single reactor at the Monju site. However, the previously stated target reliability for the RGV is used as an example in this paper. The design life of Monju is 30 years. Thus, the target reliability of the RGV for the entire design life is 6×10^{-4}/30 years.

2.2 CSS

A CSS supports a core support plate, core barrel, and other internal components, and transfers their weight to the RV through a mounting arm with attachment bolts. The required safety function is to maintain the core configuration. The material is SUS304. The CSS is used in a purity-controlled liquid sodium environment. The maximum temperature during normal operations is approximately 400°C, where the creep effect is negligible.

Section XI, Division 3 gives CSSs a Class 1 classification, and VTM-3 examinations are required. Either a literal visual method (e.g., periscope and light) or a combination of under-sodium scanning and dimensional gauging is assumed for the VTM-3 examination of the CSS. However, it is impossible to apply a literal visual method to the CSS under a sodium environment. It is also difficult to apply the combination of under-sodium scanning and dimensional gauging because the appendix for under-sodium scanning is still in the course of preparation. In addition, although significant efforts have been made to develop under-sodium scanning systems, they does not reach a practical-use level yet. Therefore, it is necessary to determine the ISI requirements for the CSS using the proposed procedure based on the SBC concept.

Failure modes for the CSS must also be determined. The environment of the CSS contains purity-controlled sodium and neutron irradiation. The corrosion in purity-controlled sodium is negligible (Furukawa et al. 2009). Material properties change as a result of neutron irradiation. For example, a decrease in ductility is one of the concerns. However, a surveillance program during operation is already planned for Monju to confirm the neutron irradiation effects. The loosening of the attachment bolts may be another concern, but preventive measures against the rotation of the bolts have been taken. The temperature is low enough to neglect creep damage. This leaves only fatigue damage due to cyclic loads during the reactor start-ups and shutdowns.

A target reliability for the CSS of the JSFR has been proposed by Kurisaka et al. (2011). Just a single failure of the CSS would result in severe core damage and the loss of the containment function. Therefore, a target reliability of 2×10^{-10}/reactor-year was proposed for the CSS, which was much smaller than the target level of core damage sequences expressed by combining the initiating events with the loss of the mitigation system. The target reliability of the CSS for the entire design life is 6×10^{-9}/30 years in this study.

3 Evaluation Procedures

3.1 Crack Initiation Evaluation

The following limit to the creep-fatigue damage in the JSME design and construction code for fast reactors (JSME FR Code; JSME 2013) is used as a crack initiation criterion:

$$D = f\left(D_f, D_c\right) \tag{1}$$

where D is the criterion value that connects points (D_f, D_c) = (1, 0), (0.3, 0.3), and (0, 1). D_f and D_c are the fatigue damage and creep damage, respectively. It is assumed that a fully circumferential crack with a depth of 1 mm is initiated when the fatigue-creep interaction damage reaches D.

D_f and D_c are basically evaluated according to the JSME FR Code. The creep-fatigue damage evaluation method in the JSME FR Code was originally based on the design methods for Monju, which was explained by Iida et al. (1987). D_f is calculated as follows:

$$D_f = n / N_f\left(\varepsilon_t\right) \tag{2}$$

where n is the number of cycles, and N_f is the number of cycles to crack initiation for the total strain range, ε_t.

D_c is calculated as the sum of the creep damage under a constant stress of S_g, those under the relaxation of a secondary stress, D_0^* for an initial monotonic loading and D^* for successive cyclic loadings, and that under the relaxation of a peak stress D^{**} as follows:

$$D_c = \frac{2t}{t_R\left(S_g\right)} + D_0^* + n^* D^* + n^{**} D^{**} \tag{3}$$

where t is the total time at elevated temperature (h), and t_R is the creep rupture time (h). In this paper, S_g is assumed to be $3(P_L+P_b)$, where P_L and P_b are the long-term local primary membrane and bending stresses (MPa), respectively.

The JSME FR Code provides the values of D_0^*, D^*, and D^{**} in charts. However, in this paper, they are evaluated using the following equations:

$$D_0^* = 2\left[\int_0^{t'} \frac{dt}{t_R(\sigma)} - \frac{t'}{t_R\left(S_g\right)}\right] \tag{4}$$

$$D^* = 2\left[\int_0^{t'} \frac{dt}{t_R(\sigma)} - \frac{t'}{t_R\left(S_g\right)}\right] \tag{5}$$

$$D^{**} = \int_0^{t'} \frac{dt}{t_R(\sigma)} - \frac{t'}{t_R\left(S_g\right)} \tag{6}$$

where t' was originally the time required until the relaxation stress becomes S_g, but here it is modified to be either this time or the holding time at the elevated temperature for each cycle (h), whichever is shorter, to avoid an excessive overestimation of D_c.

The initial stress for each creep damage is determined as follows:

$$S_i = \begin{cases} P_L + P_b + Q, & for\ D_0^* \\ MIN\left[\frac{1}{2}S_n, S_n - 1.5S_{mC}\right], & for\ D^* \\ MIN\left[\frac{1}{2}\Delta\sigma_R(\varepsilon_t), E\varepsilon_t - 1.5S_{mC}\right], & for\ D^{**} \end{cases} \tag{7}$$

where Q is the long-term secondary stress (MPa), S_n is the intensity of the primary plus secondary stress range (MPa), S_{mC} is the design stress intensity at the compression side (MPa), E is the modulus of longitudinal elasticity (MPa), and $\Delta\sigma_R$ is the stress range in a cyclic stress–strain curve. In the JSME FR Code, the same value is used for SUS304 as S_i for D^* regardless of the intensity of the actual induced stress. However, here, the S_i value for D^* is evaluated using S_n based on the method for ferritic steels in the JSME FR Code to obtain a more realistic value.

The relaxation of the stress is calculated using the following equation and the strain hardening theory:

$$\dot{\sigma} = -\frac{E\dot{\varepsilon}_c}{q} \tag{8}$$

where q is the elastic follow-up parameter (equal to 3 for D_0^* and D^*, and 1 for D^{**}), and ε_c is the creep strain. The dots represent time derivatives.

3.2 Crack Propagation Evaluation

The following equation was used to calculate the crack propagation due to the creep-fatigue damage (Fujioka et al. 1995):

$$\frac{da}{dn} = C_f \Delta J_f^{\ m_f} + C_c \Delta J_c^{\ m_c} \tag{9}$$

where a is the crack depth (mm), and C_f, C_c, m_f, and m_c are material constants. ΔJ_f and ΔJ_c are the J-integral ranges of the fatigue and creep, respectively. The reference stress method was used to calculate the J-integral ranges (Miura et al. 2000).

A depth of half the thickness is simply chosen as the critical crack depth in this paper, although a deeper crack could be accepted by conducting detailed evaluations of unstable fractures.

3.3 Reliability Evaluation Conditions

A direct Monte-Carlo method is used for the probabilistic evaluations. The total numbers of trial samples for the RGV and CSS are 10^7 and 10^9, respectively, corresponding to the target reliabilities.

Cross sections for the evaluations of the RGV and CSS are shown in Figure 3. The portion where the load is highest during normal operations is chosen as the cross section for the RGV evaluation because the RGV structure is relatively simple. On the other hand, the mounting arm from the RV is chosen as the cross section to evaluate the reliability of the CSS instead of the CSS itself because there are multiple load transfer paths, except in this section. Therefore, this is a critical section so that the required safety function of the CSS can be maintained. SUS304 is used as the material for both the RV and CSS.

Tables 1 and 2 list the evaluation parameters and random variables, respectively. The evaluation parameters are based on the design values for operating conditions I and II. The thermal stress factor χ is chosen as a random variable to consider the potentially scattering of the temperature range during start-ups and shutdowns. This factor is multiplied by S_n and Q in the Monte-Carlo simulation. The creep rupture time factor α_R, creep strain factor α_c, and fatigue life factor α_f are also chosen as random variables, as in our previous study (Takaya et al. 2015a). Their distribution types and variations are based on material test results used for developing material strength standards in the JSME FR Code (Takaya et al. 2015b). In the JSME FR Code, the design creep rupture times and creep strains are determined using α_R = 10 and α_c = 3, respectively. The values of the reduction factors are also based on material test results. In this paper, α_R and α_c are treated as random variables. Thus, the equations for the average creep rupture time and creep strain in the JSME FR Code are used as they are to evaluate the average values. On the other hand, the design fatigue life curves are determined by applying a reduction factor of two to the mean strain amplitude or 20 to the mean cycles, whichever factor leads to the greatest conservatism. It has been stated that a factor of 20 can be regarded as the product of three subfactors: the scatter of the material test data (=2.0), size effect (=2.5), and other effects such as the surface finish and atmosphere (=4.0) (Chopra and Shack 2003). The factor of two corresponds to the 5% point of the failure probability for the material test data used to derive the value in Table 2. Thus, the value of the factor for the scatter of the material test data is reasonable. In addition, the other factors should also be treated as random variables. However, there is not yet enough information. Therefore, in this paper, the best-fit curves in the JSME FR Code lowered by a factor of two for the strain or 10 for the cycles, whichever is more conservative, is used as the average fatigue life curves for the reliability evaluation. The variations in the coefficients for crack growth were determined by assuming that the 95% point corresponds to twice the average because most of the experimental data were within the factor of two (Fujioka et al. 1995).

4 Results and Discussions

4.1 Stage I Evaluations

Figure 4 shows the probability distributions of fatigue damage and the ratio of the creep-fatigue damage at the end of the design life to the criterion value for crack initiation calculated by A/B in Figure 5. The numbers of crack initiation samples were just 1 out of 10^7 and zero out of 10^9 for the RGV and CSS, respectively. The crack initiation cycle of the RGV sample was 458, and the crack depth at the end of the design life, 460 cycles, was just 1.00115 mm, which was far from the failure criterion of 20 mm. These results show that both the RGV and CSS have sufficient structural reliability compared with the targets, 6×10^{-4}/30 years and 6×10^{-9}/30 years, respectively.

4.2 Stage II Evaluations

At Stage II, the detectability and/or probability of a break with the maximum allowable size is assessed in terms of the required safety function. When such a break is not practically detectable, the reliability is evaluated using an additional requirement based on conservative hypothesis. If the reliability with the additional requirement meets the target reliability, the detectability, in other words, ISI, is not necessary.

In this study, a crack with a depth of half the thickness was simply chosen as the failure criterion for both components. Section XI, Division 3 requires a VTM-3 examination for the RGV, but these examinations will not be effective for cracks without general deformation, as previously mentioned. Devices for volumetric tests are being developed for the RV of Monju (Tagawa et al. 2007). Although these are potentially expected to be applicable to the RGV, it was assumed in this study that no sufficiently effective methods or technologies are currently available to detect service induced cracks in the RGV. As previously explained, no crack detection methods are available for the CSS. Therefore, reliability evaluations with additional requirements were conducted for them.

The additional requirements shall be determined to demonstrate that the occurrence of unallowable breaks can be eliminated using defensible technical basis. The requirements shall be correlated to the highest consequence failure mode, and shall be set in a way that disregards the functionality associated with design and operation of the component in the postulated failure mode. These requirements can be categorized into four groups: load, resistance, environment, and configuration. For example, an additional requirement related to resistance can be established as a decrease in resistance caused by loss of strength-enhancement mechanisms or metallurgical stability. Applying only one of these is deemed sufficient. If a requirement does not naturally fall into one of the categories of load, resistance, and environment, the fourth category, configuration, shall be selected, and unanticipated defects or distortions shall be postulated. In this study, a fully circumferential crack with a depth of 10% of the thickness was assumed as an initial defect. Such a large initial defect could surely be detected by manufacturing inspections or the pre-service examinations. Thus, it is considered to be a sufficiently conservative requirement.

Figure 6 shows the probability distributions of the crack depth at the end of the service life with the additional requirement. The number of failure samples was four out of 10^7 for the RGV, which meant its failure probability was approximately 4×10^{-7}. The target reliability for the RGV is 6×10^{-4}/30 years. Therefore, it was confirmed that the RGV has sufficient reliability even with the additional requirement. On the other hand, the number of failure samples was zero out of 10^9 for the CSS. The maximum depth in all the samples was about 14.8 mm, whereas the failure criterion depth was 70 mm. The initial crack depth was 14 mm. Thus, the cracks hardly propagated in the case of the CSS. It is difficult to estimate the failure probability of the CSS, but it is obviously much lower than the reliability target, 6×10^{-9}/30 years, even if the additional requirement is considered.

As result, the occurrence of failures in both the RGV and CSS can be practically eliminated. Hence, no ISI requirements are needed for these components.

5 Conclusion

The proposed process for ISI requirement determination based on the SBC concept was applied to the RGV and CSS of a prototype sodium-cooled fast breeder reactor (Monju) in Japan. The proposed process consists of two stages: a structural reliability evaluation for potentially degradation mechanisms (Stage I) and defect detectability assessment or reliability evaluation with an additional requirement (Stage II). The potentially degradation mechanisms were the creep-fatigue and fatigue for the RGV and CSS, respectively. The Stage I evaluations using the Monte-Carlo method showed that both components have sufficient reliability. At Stage II, it was assumed that the conventional ISI techniques were not sufficient to detect service induced cracks in the RGV or the CSS. Therefore, reliability evaluations were conducted with an additional requirement consisting of an initial fully circumferential crack with a depth of 10% of the thickness. As a result, it was shown that both components had sufficient reliability, even with the additional requirement. The occurrences of failures of these components were practically eliminated. Hence, it was concluded that no ISI requirements were needed for these components.

Acknowledgment

The authors thank Mr. Naoto Sasaki of Ascend Co. Ltd for performing the calculations of the reliability evaluations.

Tables

Table 1: Evaluation parameters

		RGV	CSS
Outer diameter (mm)		7,840	6,720
Thickness (mm)		40	140
Operating temperature (°C)		445	405
Number of cycles		460	460
Total time at elevated temperatures for the entire service life (h)		210,000	0
P_L+P_b for creep damage evaluation (MPa)		3	-
Q for creep damage evaluation (MPa)		133	-
S_n (MPa)		238	107
Stress range perpendicular to a crack (MPa)	Primary membrane	0	7
	Primary bending	0	53
	Secondary membrane	2	0
	Secondary bending	236	15
Shakedown range (MPa)		301	323
Strain rate (mm/mm/s)		1E-8	1E-6

Table 2: Random variables

	Distribution type	Logarithmic standard deviation
Thermal stress factor, χ	Log-normal	0.113 (RGV), 0.258 (CSS)
Creep rupture time factor, α_R	Log-normal	0.560
Creep strain factor, α_c	Log-normal	0.800
Fatigue life factor, α_f	Log-normal	0.420
Coefficient for creep crack growth, C_c	Log-normal	0.422
Coefficient for fatigue crack growth, C_f	Log-normal	0.422

Figure captions

Figure 1. Logic flow diagram of SBC process

Figure 2. Main components of Monju reactor

Figure 3. Cross sections for evaluation and dimensions

(a) RGV, (b) Mounting arm of RV for CSS

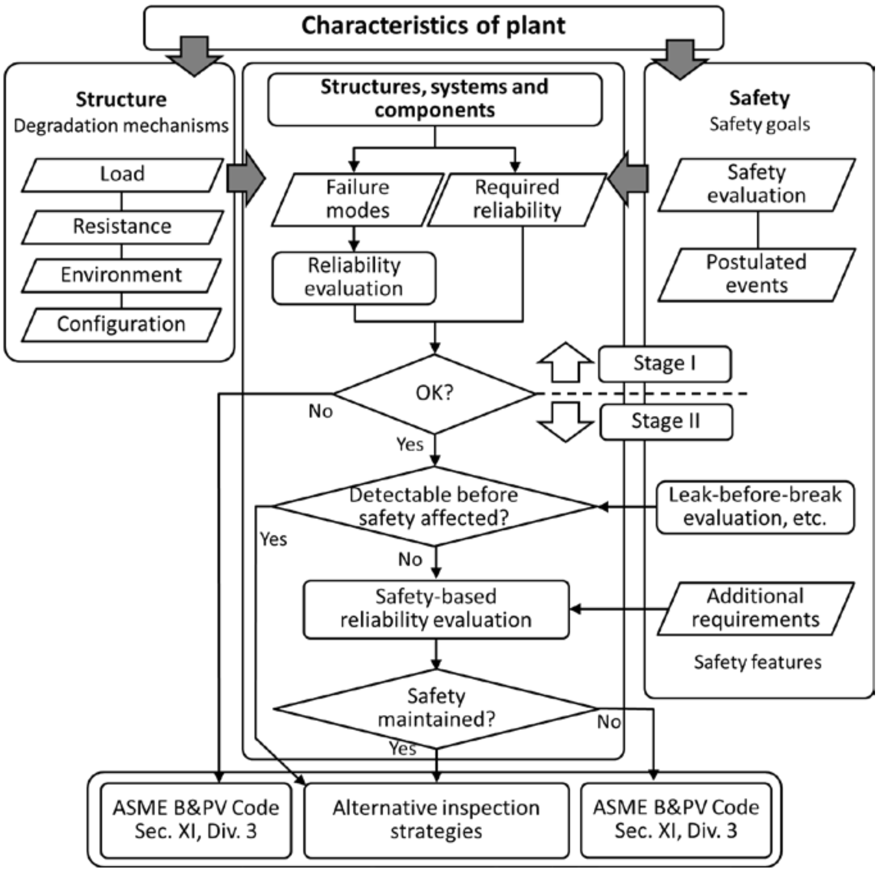
Figure 4. Probability distributions of fatigue damage and ratio of creep-fatigue damage at end of service life to criterion value for crack initiation

(a) RGV, (b)CSS

Figure 5. Distances from origin to creep-fatigue damage at end of service life and criterion value for crack initiation

Figure 6. Probability distributions of crack depth at end of service life with additional requirement that initial cracks with a depth of 10% of the thickness were assumed

(a) RGV, (b)CSS



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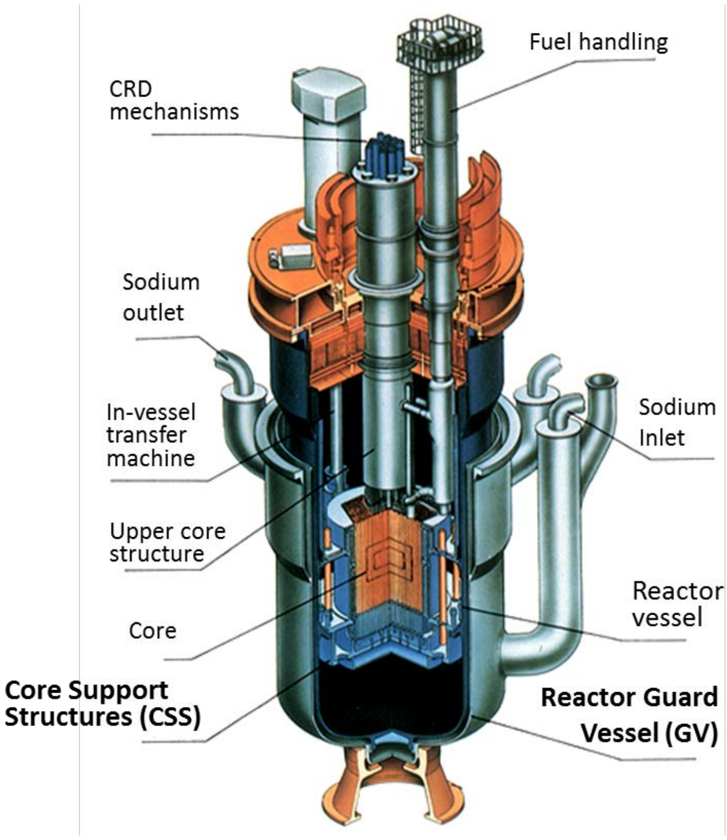
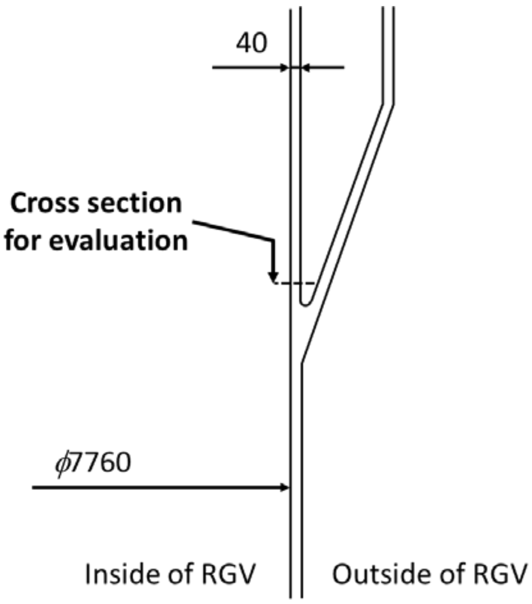
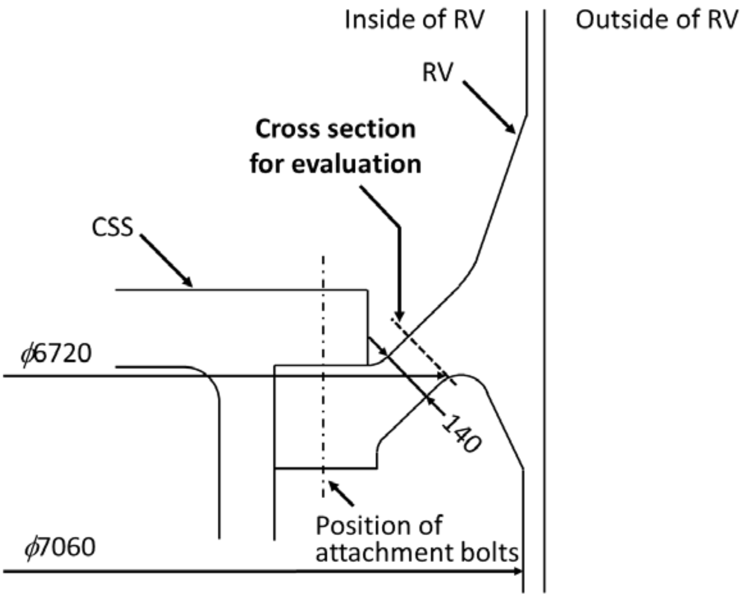


Figure 2. Main components of Monju reactor

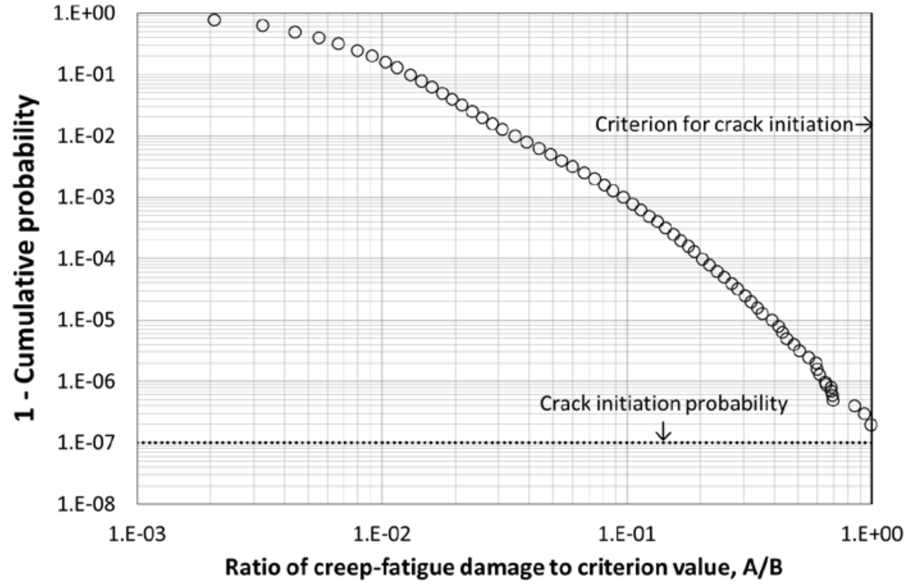


(a) RGV

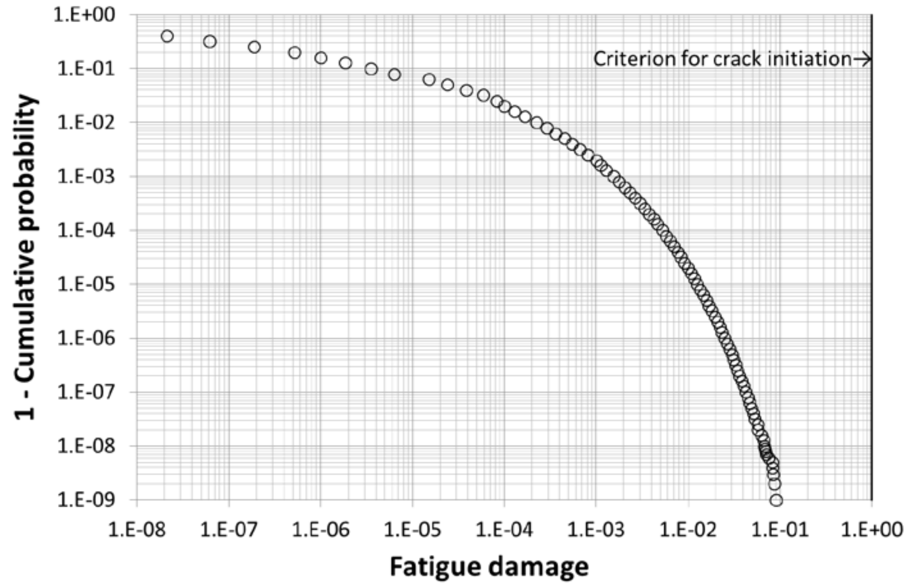


(b) Mounting arm of RV for CSS

Figure 3. Cross sections for evaluation and dimensions



(a) RGV



(b) CSS

Figure 4. Probability distributions of fatigue damage and ratio of creep-fatigue damage at end of service life to criterion value for crack initiation

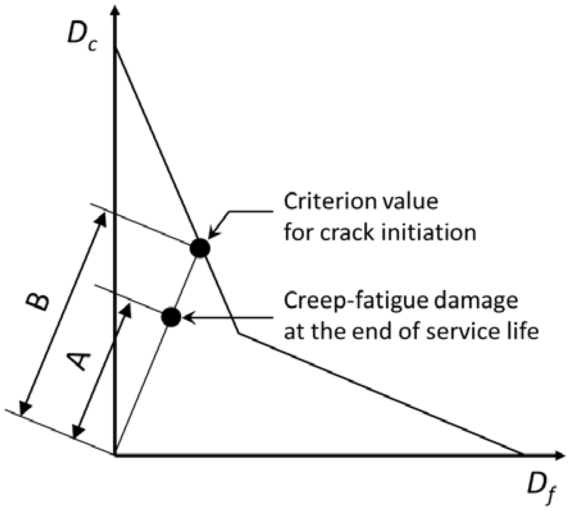
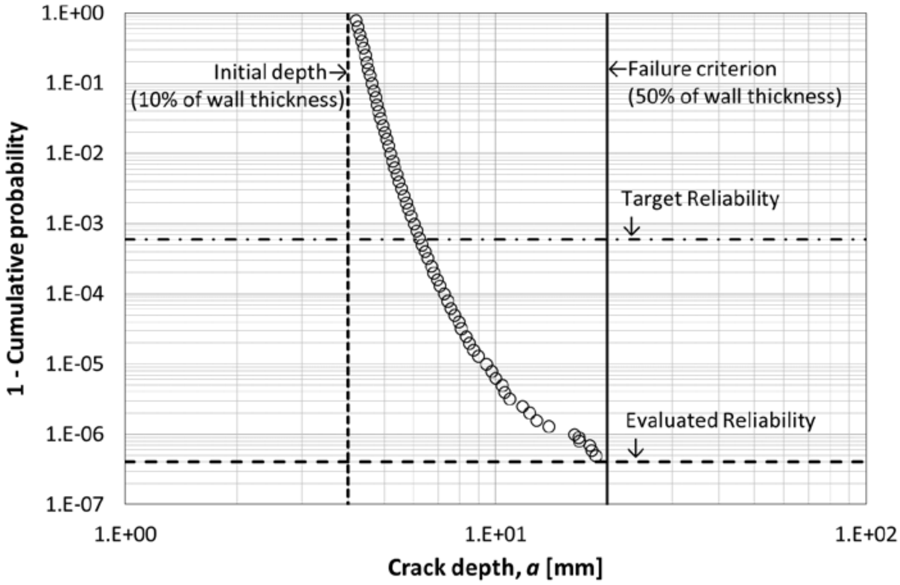
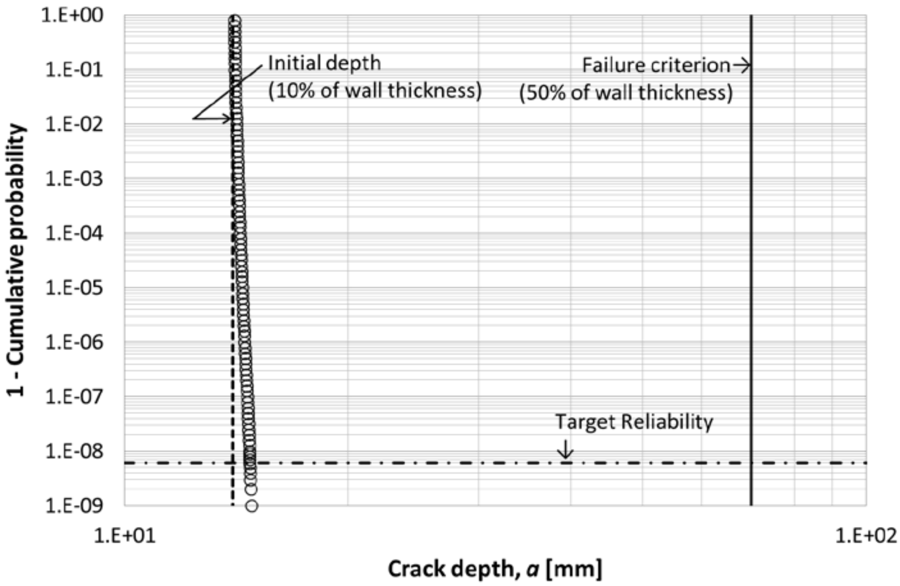


Figure 5. Distances from origin to creep-fatigue damage at end of service life

and criterion value for crack initiation



(a) RGV



(b) CSS

Figure 6. Probability distributions of crack depth at end of service life with additional requirement that initial cracks with a depth of 10% of the thickness were assumed

References

American Society of Mechanical Engineers (2001). “Boiler and Pressure Vessel Code, Section XI, Division 3, Rules for Inspection and Testing of Components of Liquid-Metal-Cooled Plants,” ASME, New York, NY.

Asada, Y., Tashimo, M. and Ueta, M. (2002a). “System Based Code—Principal Concept,” *Proc., 10th International Conference on Nuclear Engineering*, paper 22730.

Asada, Y., Tashimo, M. and Ueta, M. (2002b). “System Based Code—Basic Structure,” *Proc., 10th International Conference on Nuclear Engineering*, paper 22731.

Asada, Y. (2006). “Japanese Activities Concerning Nuclear Codes and Standards—Part II,” *Journal of Pressure Vessel and Technology*, 128, 64–70.

Asayama, T., Takaya, S., Morishita, M., and Schaaf, F. (2014). “Elaboration of the System Based Code Concept –Activities in JSME and ASME– (4) Joint Efforts of JSME and ASME,” *Proc., 22nd International Conference on Nuclear Engineering*, ICONE22-30572.

Chopra, O.K. and Shack, W.J. (2003) “Review of the Margins for ASME Code Fatigue Design Curve–Effects of Surface Roughness and Material Variability,” U.S. NRC, Washington, DC, NUREG/CR-6815.

Fujioka, T., Shimakawa, T., Miura, N., and Kashima, K. (1995) “Development and Verification of an Evaluation Method for Creep-Fatigue Crack Propagation in FBR Components,” *PVP*, ASME, 305, 395–402.

Furukawa, T., Kato, S., and Yoshida, E. (2009) “Compatibility of FBR Materials with Sodium,” *Journal of Nuclear Materials*, 392, 249–254.

Iida, K., Asada, Y., Okabayashi, K., and Nagata, T. (1987). “Simplified Analysis and Design for Elevated Temperature Components of Monju,” *Nuclear Engineering and Design*, North-Holland, 98, 305–317.

Japan Society of Mechanical Engineers (2013). “Codes for Nuclear Power Generation Facilities, Rules on Design and Construction for Nuclear Power Plants, Part II: Fast Reactor Standards,” JSME, JP, JSME S NC2-2013, *in Japanese*.

Kotake, S., Mihara, T., Kubo, S., Aoto, K., and Toda, M. (2008). “Development of Advanced Loop-Type Fast Reactor in Japan (1): Current Status of JSFR Development,” *Proc. 2008 International Congress on Advances in Nuclear Power Plants*, Anaheim, CA, paper 8226.

Kurisaka, K., Nakai, R., Asayama, T., and Takaya, S. (2011). “Development of System Based Code (1) Reliability Target Derivation of Structures and Components,” *Journal of Power and Energy Systems*, JSME, JP, 5, 19–32.

Miura, N., Shimakawa, T., Nakayama, Y., and Takahashi, Y. (2000) “Simplified *J*-Evaluation Method for Flaw Evaluation at High Temperature – Part 1: Systematization of Evaluation Method,” *PVP*, ASME, 410-1, 111–118

Tagawa, A., Ueda, M., and Yamashita, T. (2007) “Development of the ISI Device for Fast Breeder Reactor MONJU Reactor Vessel,” *Journal of Power and Energy Systems*, JSME, JP, 1, 3–12.

Takaya, S., Asayama, T., Kamishima, Y., Machida, H., Watanabe, D., Nakai, S., and Morishita, M. (2015a), “Application of the System Based Code Concept to the Determination of In-Service Inspection Requirements,” *Journal of Nuclear Engineering and Radiation Science*, 1 paper 011004.

Takaya, S., Sasaki, N., and Tomobe, M. (2015b) “Statistical Properties of Material Strength for Reliability Evaluation of Components of Fast Reactors,” Japan Atomic Energy Agency, JP, JAEA-Data/Code 2015-002.

Yamano, H., Kubo, S., Shimakawa, Y., Fujita, K., Suzuki, T., and Kurisaka, K. (2012). “Safety Design and Evaluation in a Large-Scale Japan Sodium-Cooled Fast Reactor,” *Science and Technology of Nuclear Installations*, Vol. 2012, paper 614973.