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WORKING MATERIAL

IAEA Workshop (Training Course) on Codes and Standards for Sodium Cooled Fast Reactors

Beijing, China,

6 – 8 July 2010

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Summary

Introduction

The Workshop (training course) was organized by the IAEA Nuclear Power Technology Development Section within the framework of the IAEA Technical Cooperation (TC) Project CPR/4/032 "Enhancing the Capabilities of National Institutions Supporting Nuclear Power Development". It was held from 6 – 8 July 2010 at the Centre of Regulation and Standards for Nuclear Power (Centre 4, ISNI) in Beijing, China.

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The training course consisted of lectures and Q&A sessions. The lectures dealt with the history of the development of Design Codes and Standards for Sodium Cooled Fast Reactors (SFRs) in the respective country, the detailed description of the current design Codes and Standards for SFRs and their application to ongoing Fast Reactor design projects, as well as the ongoing development work and plans for the future in this area. Annex 1 contains the detailed Workshop program.

Assessment of International Design Codes and Standards for SFRs

The summary of the experts' assessment of the international Codes and Standards for SFRs is given in Table 1. The categories dealt with in this assessment summary table are safety, materials, design, and construction.

TABLE 1. SUMMARY OF THE EXPERTS' ASSESSMENT OF THE INTERNATIONAL CODES AND STANDARDS FOR SFRS

Sl.No	Aspects	India (RCC-MR)	Japan	Russia
		I. Safety Criteria		
I.1	Safety approach, safety goals	AERB safety criteria formulated in 1990 for the pool type reactors CDA as beyond design basis event meeting the respective sight boundary dose and containment signed for sodium release under moderate energy release. Safety systems designed for two levels of EQ (SSE and OBE). For the future SFRs, the safety criteria has been totally revised (which is under finalization) giving due considerations on present level of knowledge to achieve enhanced safety with improved safety. The reliability of shutdown systems has been enhanced by one order	Passive shutdown system and re-criticality free core, for example, are now investigated	
I.2	Design approaches and technologies to meet safety goals	For the current design, design features incorporated from operating feedback of operating experiences. Mechanical design as RCC-MR 2002 for PFBR. For future designs, lessons learned from PFBR, R&D outcome, improvements in RCC-MR-2007, enhanced ISI and repair strategies would be adopted		
I.3	Regulatory documents	Documents are under preparation	AESJ (Atomic Energy Society of Japan) is responsible to prepare the documents.	Very comprehensive. List of documents is given in Annex 3. Russian regulatory base is complete enough to regulate SRs in safe manner
		II. Materials		
II.1	Comprehensiveness	Core not covered	Clad and wrappers are not covered	
II.2	Data for Design	Available	Available (304SS, 316SS, 321SS(tube), 316FR, 2.25Cr-1Mo(plate and tube) and Mod.9Cr-1Mo	

II.3	Data for Analysis	Available except for welds. Data is to be added for G91 steel	Available except for welds	
		III. Design		
III.1	Comprehensiveness	Issues not addressed in Annex 1	JSME (Japan Society of Mechanical Engineers) rules on design & construction for FR components will be revised by 2016	
III.2	High temperature design rules	Comprehensively addressed	Under discussion in JSME code committee	
III.3	Design Rules for welding	Creep fatigue damage interaction diagram to be provided for welds	For Mod.9Cr-1Mo steel, creep strength evaluation method has been given (taking "Type-IV" damage into account). Creep fatigue strength assessment technique is under discussion	
III.4	Design rules for seismic analysis	ASME Appendix-N is adopted	Under discussion in JEA (Japan Electric Association)	
III.5	Design rules for LBB	Appendix-A16 provides	Under discussion in task group of JSME	
III.6	Rules for core	Not available. Interim guidelines have been formulated	Not available	
III.7	Rules for Mechanisms	Volume K provides rules	Intermediate dwelling model, for example, is now developed	
III.8	Rules for Class-1 Pipings	RB 3600 provides	ETD-4000 of JSME rules on design & construction for FR components has been provided	
III.9	Rules for Box Type Structures	RB 3800 provides	Not available	
III.10	Rules for Heat exchangers	RB 3900 provides	ETD-3000 of JSME rules on design & construction for FR components is applicable	
		IV. Construction		
IV.1	Comprehensiveness	Issues not addressed in Annex 2	JSME rules on design & construction for FR components has been provided and revised by 2016	
IV.2	Rules for manufacturing deviations	Rule for Crack like defects are provided in A16 Allowable form tolerances are provided Acceptable weld configurations are provided	Taken into account in JSME rules on design & construction for FR components	
IV.3	Rules for ISI	Not available. Rules of ASME-Sec IX are adopted	JSME rules on Fitness-for-service will be provided by 2016.	

Annex 1

Program of the IAEA Workshop (Training Course) on
Codes and Standards for Sodium Cooled Fast Reactors
Centre of Regulation and Standards for Nuclear Power (Centre 4, ISNI), Beijing, China, 6 – 8 July 2010

Time	Tuesday 6 July 2010	Wednesday 7 July 2010	Thursday 8 July 2010
9:00		Design Codes and Standards for the Prototype Fast Breeder Reactor (PFBR): ASME-Sec III NH, RCC-MR, indigenous results and material data, revisions to RCC-MR 2007, experiences in the design and construction of PFBR (Perumal Chellapandi, IGCAR)	Comparison of Indian, Japanese and Russian Experience (Perumal Chellapandi, IGCAR; Takashi Wakai, JAEA; Nikolay Khrennikov, SECNRS)
9:30	Opening (ISNI and IAEA)		
10:00	Status of Fast Reactor Technology Development: General Overview (Alexander Stanculescu, IAEA)	Design Codes and Standards for the Prototype Fast Breeder Reactor (PFBR) …, cont. (Perumal Chellapandi, IGCAR)	General Discussion: Way Forward in Developing Design Codes and Standards for Fast Reactors, Q&A Session with Lecturers (Perumal Chellapandi, IGCAR; Takashi Wakai, JAEA; Nikolay Khrennikov, SECNRS)
11:00	Break	Break	Break
11:20	Fast Breeder Reactor Development and Deployment Programs in India (Perumal Chellapandi, IGCAR)	Design Codes and Standards for Sodium Cooled Fast Reactors in Japan (Takashi Wakai, JAEA)	General Discussion: Way Forward in Developing Design Codes and Standards for Fast Reactors, Q&A Session with Lecturers, continued (Perumal Chellapandi, IGCAR; Takashi Wakai, JAEA; Nikolay Khrennikov, SECNRS)
12:20	Lunch	Lunch	Lunch
14:00	Fast Breeder Reactor Development and Deployment Programs in Japan (Takashi Wakai, JAEA)	Design Codes and Standards for Sodium Cooled Fast Reactors in Japan, cont. (Takashi Wakai, JAEA)	IAEA: International Atomic Energy Agency IGCAR: Indira Gandhi Centre for Atomic Research JAEA: Japan Atomic Energy Agency SECNRS: Scientific and Engineering Centre for Nuclear and Radiation Safety
15:00	Fast Breeder Reactor Development and Deployment Programs in the Russian Federation (Nikolay Khrennikov, SECNRS)	Design Codes and Standards for Sodium Cooled Fast Reactors in the Russian Federation(Nikolay Khrennikov, SECNRS)	
16:00	Break	Break	
16:20	General Discussion, Q&A Session with Lecturers	Design Codes and Standards for Sodium Cooled Fast Reactors in the Russian Federation, cont. (Nikolay Khrennikov, SECNRS)	

Annex 2

Rules for Issues Related to Materials not Addressed in RCC-MR

1.0 Effect of sodium

The effects of reactor quality sodium on the mechanical properties of SS 316 LN, SS 304 LN and modified 9 Cr 1 Mo (T91) steel base metal are negligible. Hence material data generated in air could be safely applied for the design of components with wall thickness larger than 2 mm. The influence of corrosion, carburization, decarburization, formation of ferritic substructure layers and sensitization, are considered for thin walled components (t < 2 mm) [1]. For the thickness <2 mm, for e.g. IHX tubes (thickness is 0.8 mm), since the loss of thickness due to corrosion may be significant, extra thickness is provided to consider the effects of corrosion. The loss of thickness due to sodium corrosion is quantified for PFBR in references [2].
It is worth mentioning that the corrosion of SG tubes (nominal wall thickness is 2.3 mm) in the steam side is considered in the thickness selection [3].

2.0 Weld strength reduction factor for modified 9Cr 1 Mo

The stress and strain analysis is carried out as if the structures were made of homogenous material and weld strength reduction factors, also called J factors (fatigue and creep strength reduction factors) are then applied to the parent material failure properties to allow for any weakening effect of the welds. Comprehensive creep and fatigue strength reduction factors are provided in RCC-MR-A9, 1J for SS 316 LN and 2J for SS 304 LN. (For 304 LN, fatigue strength reduction factor is not provided). But factors are not given for modified 9Cr 1 Mo. However, the weld strength reduction factors which are recommended for EFR design are adopted [4]. Summary of the interim recommendation is as follows:

J_m	=	1	to be used with S_m
J_r	=	J_t = 0.9	for temperatures between 698 K - 798 K
J_f	=	1.25	for full penetration welds subjected to full Inspection.

3.0 316LN data for 60 years thermal aging

Under preparation.

4.0 Material data for welds for the application of A16 rules

Under preparation.

References

1. Recommendations made in the concluding session of IWGFR / IAEA specialists meeting on properties of structural materials in LMFBR, Karlsruhe, Germany, Jan 1991.
2. S. Rajendran Pillai et.al., ‘High Temperature Corrosion of clad and Structural Materials in Sodium’, PFBR/MCG/CSTD/Dec. 1999
3. S. Rajendran Pillai et.al., ‘Corrosion Loss of SG tubes of Modified 9Cr-1Mo steel’, PFBR/MCG/CSTD/R-2, Dec. 1999.
4. Escaravase M et al, "Weldment design rules for the EFR", Seminar on high temperature weldments, I Mech, London, 1990. Picker, C. UK Development of a Strain Based Creep F*atigue Assessment Procedure for Fast Reactor Design*, IAEA-TECDOC-933, “Creep Fatigue Damage Rules for Advanced Fast Reactor Design’, Technical Committee Meeting, Manchester, UK, 11-13, June 1996.

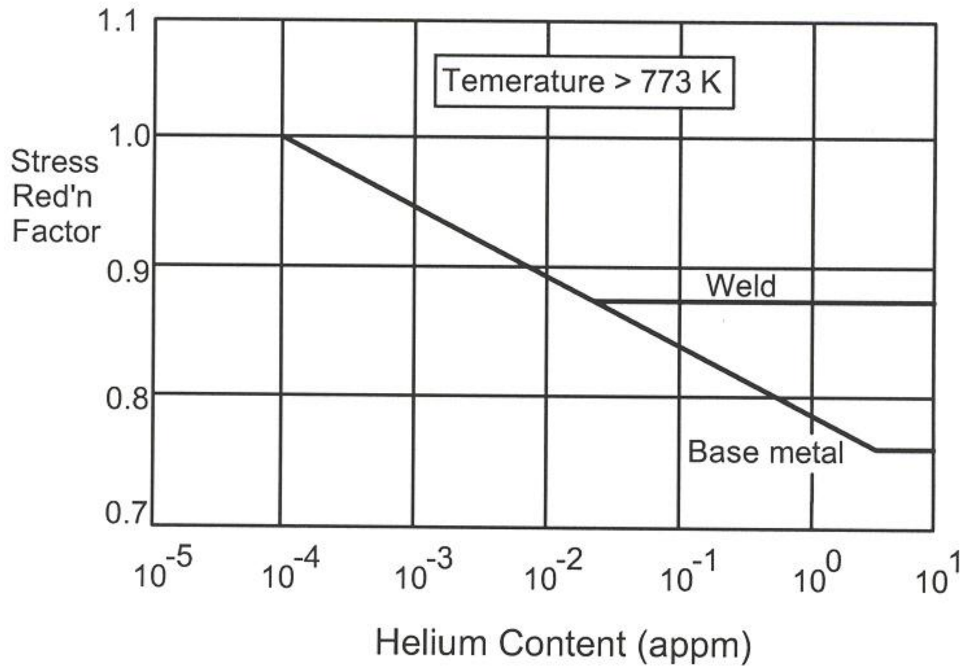
Annex 3

Criteria for the Aspects Related to Design, not Covered in RCC-MR

For the aspects not covered either in RCC-MR or in ASME, design rules for considering the effects of low as well as high dose irradiation, thermal striping, core disruptive accident, leak before break procedure for sodium boundary components, etc, certain `ad hoc' rules given in the literature are used. The summary of these rules are as follows:

1.0 Effect of low dose irradiation

The effect of neutron dose level less than 1 dpa is negligible for the cold pool components. The main and inner vessels, grid plate and core cover plate are exposed to relatively low neutron dose levels less than 1 dpa. The critical component in this respect is grid plate which however, experiences dose level less than 1 dpa, i.e. 0.33 dpa for 40 yrs. The thermal neutron irradiation produces helium, mainly by the reaction with boron present in the steel at high temperature. Helium affects creep rupture strength and ductility of austenitic stainless in the creep regime. The interim recommendation for design activities is to apply a stress reduction factor which varies as a function of helium content as indicated in the Fig. 1 [1].



2.0

Fig. 1. Stress reduction factors for 316 LN to allow for irradiation.

2.0 Effect of high dose irradiation

For the design of core components which are subject to high dose of neutron irradiation (> 5 dpa), a set of design rules with associated set of mechanical properties has been formulated based on limited information available on the practice followed in other countries and documented in [2]. The salient features of recommendations are given below.

The reduction of ductility of the material due to neutron irradiation has the maximum impact on the design rules. Irradiation induced creep and swelling of material should also be included in the analysis for deformation of components. Two types of analyses can be followed – elastic/in-pile creep/swelling analysis (EICSA) or inelastic analysis (IEA). For EICSA, limits are stipulated on stresses while for IEA the limits are on strains. The failure mechanisms and associated design rules for EICSA and IEA are given in Tables 1 and 2.

TABLE 1. FAILURE MECHANISMS & ASSOCIATED DESIGN RULES FOR EICSA

Failure mode	Governing parameter	Critical value	Factor of safety	Design rule
Tensile instability	P_m	S_y or S_u	1.18 & 1.7 for category 1 &2 0.9 & 1.25 for category 3	P_m<0.85 S_y or < 0.6 S_u For category 1 & 2 P_m<1.1 S_y or < 0.8 S_u For category 3
Failure in bending	P_m+ P_b	K S_y or S_u	Same as above	Same as above
Ratcheting	P_m+P_b+Q	Shakedown limits or S_u	1.7 on S_u for category 1&2 and 1.25 on S^u for category 3	P_m+P_b+Q < 0.6 S_u for category 1 &2 P_m+P_b+Q < 0.8 S_u for category 3 loads
Localized rupture	P_m+P_b+Q+F	S_u	1.25 for category 1&2 1.1 for category 3	P_m+P_b+Q+F< 0.8 S_u for cat. 1 &2 loads P_m+P_b+Q+F,< 0.9 S_u $_f$or category 3
Damage due to creep & fatigue	Time duration ,Δt and no. of cycles ,n	Rupture life t_d and fatigue limit N	4 on CDF for category 1 & 2 1.3 for cat.1,2 & 3	$\Sigma\Delta t/t_d + \Sigma n/N$ < 0.25 for cat. 1 and < 0.5 for cat. 1 & 2 and < 0.75 for cat. 1,2 &3
Brittle fracture	K_I	K_{IC}	----	$K_I < K_{IC}$

Note : Factors of safety given for damage are on cumulative damage fraction (CDF).

TABLE 2. FAILURE MECHANISMS & ASSOCIATED DESIGN RULES FOR IEA

Failure mode	Governing parameter	Critical value	Factor of safety	Design rule
Tensile instability & ratcheting	ε_m^p	$\varepsilon_u/2$	3 for cat. 1 & 2 1.5 for cat. 1,2 &3	$\Sigma\varepsilon_m^p/ (\varepsilon_u/2)$ <0.33 for category 1&2 $\Sigma\varepsilon_m^p/ (\varepsilon_u/2)$ <0.66 for category 1,2 &3
Localized rupture	$\varepsilon^p{}_t$	ε_f/TF	2 for cat. 1 & 2 and 1.25 for cat. 1,2 &3	$\Sigma\varepsilon^p{}_t/(\varepsilon_f/TF)$<0.5 for category 1&2 $\Sigma\varepsilon^p{}_t/(\varepsilon_f/TF)$<0.8 for category1,2 & 3
Damage due to creep & fatigue	Time duration ,Δt and no. of cycles ,n	Rupture life t_d and fatigue limit N	4 on CDF for cat. 1 , 2 for cat. 1 & 2 and 1.3 for cat. 1,2 &3	$\Sigma\Delta t/t_d + \Sigma$ n/N < 0.25 for cat. 1 and < 0.5 for cat. 1 & 2 and < 0.75 for cat. 1,2 &3
Brittle fracture	J-integral	J_c	----	J < J_c

3.0 Design rules for thermal striping

The thermal cycles are caused by a special phenomenon called 'thermal striping' which occurs mainly at the core cover plate of control plug and also at mixing 'Tee' junctions in the sodium pipelines. The frequency of fluctuation ranges from 0.01 to 10 Hz, which yields cumulative cycles in the order of 1.26×10^9 for a typical plant, designed for 4 y with 75% load factor (LF). These temperature fluctuations cause high cycle fatigue damage (HCF) on the component wall. There are a few critical locations particularly in the hot pool where both LCF and HCF cycles are simultaneously imposed on the metal wall for which matured design rules are not available to ensure the structural integrity. The damage becomes very severe for the welded structures particularly with defects even though the defects are within acceptable limits of the applicable inspection codes.

Thermal striping limits are established by specifying acceptable ΔT in the sodium jets in the vicinity of structural wall surface, delinking the thermal hydraulics of the sodium system. Associated parameters are chosen to make the recommended limits applicable to most of the current and evolving SFR designs. Accordingly, the structural material considered is austenitic stainless steel type SS 316 LN, which is the general choice of current and future SFR designs. From thermal striping considerations, alternate material is considered for above core structure parts to allow higher thermal striping limits. However, in this case, thermal striping would not be of concern. Further, the hot pool is at isothermal temperature at 823 K during normal operating condition. This temperature has been selected for the new designs such EFR (France), JSFR (Japan) and BN-series of reactors (Russia). In view of high heat transfer coefficient of sodium, the structural wall surface, where the damage is maximum, sees the temperature of hot pool sodium (823 K). Since, the maximum temperature should be considered for the creep fatigue damage assessment as per design codes such as ASME Sec III NH and RCC-MR, the temperature of 823 K is considered for the damage assessment. For computing the through-wall temperature distribution, heat transfer coefficient of sodium in the vicinity of wall surface is required. Since the sodium has excellent heat transfer properties, its heat transfer co-efficient is high, generally varying from 10'000 to 40'000 W/m^2K. For the computation of structural wall temperatures and stresses, higher heat transfer coefficient value yields conservative results and hence 40'000 W/m^2K is used for the analysis. As regards the wall thickness, the maximum thickness of hot pool components is 30 mm. It has been shown that beyond this thickness, it does not have any significant effect. Apart from the above mentioned material, temperature and wall thickness, the other input data required is the accumulated creep and fatigue damage at the specified locations, which is the parameter in the present study. Accordingly, allowable thermal striping limits are recommended as the function of accumulated creep fatigue damage (Fig. 2).

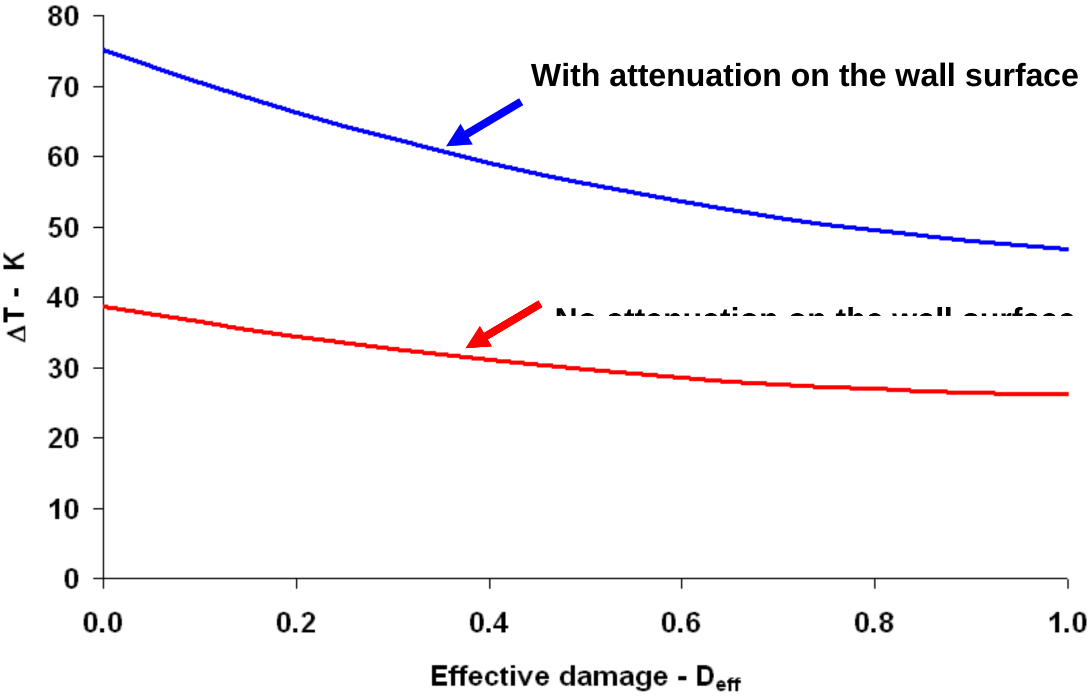


Fig. 2. Permissible thermal striping limits for FBR structures with SS 316 LN.

For deriving thermal striping limits, it is assumed that the structural wall has accumulated an effective creep fatigue damage, termed as 'D_{eff}', due to the major load cycles caused by variations of normal and thermal transient conditions. Further, it is also assumed that the structural wall has an initial part-through crack size of 0.1 mm which might not have been detected by the existing NDE techniques, which could grow due to the imposed major load cycles, depending upon the value of D_{eff}. As per the applicable design codes for the structural design of SFR components, the maximum permissible value for D_{eff} is equal to 1. Thus the structure can have a value for D_{eff}, varying from 0 to 1. As per the approach, proposed by Picker [3], when the D_{eff} value varies from 0 to 1, the crack grows from 0.1 mm to 0.5 mm. This implies that we are permitting / restricting a maximum crack size of 0.5 mm in the design stage itself, which can be detected confidently during periodic in-service inspection stage, employing with the appropriate NDE techniques. The plate can have any crack size (a) within the range of 0.1 – 0.5 mm, depending upon the accumulated damage (D_{eff}) at that location. With the presence of thermal fluctuations imposed by thermal striping, it is restricted that the crack should not further grow under due to thermal striping. This condition is ensured by satisfying the fatigue damage design criteria recommended by A16 through revised "σ-d approach" [4].

The maximum range of temperature fluctuations can be as high as 70 K where the accumulated creep fatigue damage is insignificant (< 0.1) and 45 K is acceptable where the creep fatigue is very significant (0.9). These limits are found to be consistent with the broad limits extrapolated from the failure experiences of international SFRs and sodium facilities.

The following are the specific limits:

- For the core cover plate of control plug wherein the low cycle fatigue damage is negligible (~0.0), the acceptable thermal striping limit is 70 K. For the lower portion wherein the damage is ~0.2, the acceptable limit is 60 K.

- For the locations on inner vessel or main vessel wherein the accumulated fatigue damage may be moderate (0.5), the acceptable thermal striping limit is 50 K.
- Acceptable amplitude of oscillation of stratified sodium layers on the inner vessel redan surface is ± 300 mm.
- Acceptable amplitude of free level oscillation on upper shell of inner vessel is ± 60 mm.

4.0 Design limits for the components under Core Disruptive Accident (CDA)

As per the safety criteria for Design of PFBR (April 1990), the core disruptive accident has been considered as BDBA. However, following the philosophy of defence-in-depth, the structural integrity of primary containment under CDA (energy release of about 100 MJ) has been assessed. Based on the analysis, the sodium release to RCB through the gaps of the penetrations on the top shield is estimated. The temperature and pressure rise, consequent to resulting sodium fire form input data for the RCB design.

Since design codes do not provide rules for this event, integrity is assured by limiting the accumulated plastic strain to an acceptable strain limit for the structural material SS 316 LN. The acceptable strain limit is derived from uniform elongation after taking into account effects of welds, irradiation, thermal ageing, accumulated creep and fatigue damage, strain rate on the ductility [5]. Based on these metallurgical considerations, the acceptable strain limit for main vessel can be about 15%. However, in order to avoid any mechanical interactions between main vessel and safety vessel (the minimum inter vessel space is 250 mm), the radial deformation is limited to 4 mm.

5.0 Leak Before Break (LBB) procedure for sodium boundary components

LBB is justified for the main vessel and secondary sodium piping as per French Procedure A16 (2002) which provides necessary material data and analysis methods based on fracture mechanics concepts for austenitic stainless steel sodium boundary components under creep and non-creep conditions [6]. It is worth mentioning that operating experience of FBTR and sodium loops demonstrates the successful operation of sodium leak detection systems [7, 8, 9]. Similar systems are to be employed for PFBR.

References

1. Escaravase C. et al, "EFR-DRC proposal to introduce in design work in low dose neutron irradiation effects", proc of IAEA specialists meeting on influence of low dose irradiation on the design criteria of fixed internals in fast reactor, IAEA-TEC DOC-817.
2. S Govindarajan, "Structural Design Criteria for High Dose Neutron Irradiation", PFBR/31100/DN/1007/R-1, March, 2000.
3. Picker, C. *UK Development of a Strain Based Creep Fatigue Assessment Procedure for Fast Reactor Design*, IAEA-TECDOC-933, "Creep Fatigue Damage Rules for Advanced Fast Reactor Design', Technical Committee Meeting, Manchester, UK, 11-13, June 1996.
4. P. Chellapandi, "Thermal Striping Limits", PFBR/31250/DN/1043/R-1, Nov 2000. PDSC MoM 35 (clause 3.2)].
5. Mahapetro, H.K. et al, "Acceptable strain limits for main vessel under DBA",PFBR / 31050 / DN / 1010 / R-C, 1995.
6. A16: Guide for Defect Assessment for Leak Before Break Analysis", Dec 1995.
7. FBTR Annual Report: 1989-2003
8. K. Gurumurthy, et al. "Report on Minor Sodium Leak in NaX of LCTR", Internal report IGC/EDG/STD/99129/SRUOR/3004, May 1999.
9. K. Gurumurthy, et al. "Report on Failure of Bellow Sealed Valve in LCTR", Internal report IGC/EDG/STD/99129/SRUOR/3006, June 2003.

Annex 4

List of Safety Documents Prepared in Russia

1. General Safety Provisions for Nuclear Power Plants OPB-88/97 NP-001-97
2. Nuclear safety rules of power plant reactor facilities NP-082-07
3. Regulations on seismic design of nuclear plants NP-031-01
4. Nuclear plant siting. Basic safety criteria and requirements NP-032-01
5. External natural and technogenic impacts on nuclear energy facilities NP-064-05
6. Requirements to the QA Program for NPP NP-011-99
7. Safety rules for nuclear fuel on-site storage and transportation at nuclear power facilities NP-061-05
8. Basic health rules of radiation safety ensuring OSPORB-99
9. Sanitary rules on nuclear plant design and operation SP AS-03
10. NPP Fire protection. General requirements NPB-113-03
11. NPP Fire protection. Design requirements NPB-114-2002
12. Strength Calculation Norms for Components and Pipelines for NP Installations PNAE G-7-002-87
13. Rules on Design and Safe operation of Components and Pipelines PNAE G-7-008-89
14. Pipeline armature for nuclear power plants. General technical requirements. NP-068-05
15. Rules for layout and operation of important for safety ventilation systems of nuclear plants. NP-036-05
16. Regulations on design and operation of localization safety systems of nuclear plants. NP-010-98
17. Requirements for control systems important for NPP safety. NP-026-04
18. Hydrogen explosion protection rules for nuclear power plants. NP-040-02
19. Safety rules for the handling of radioactive wastes from nuclear power plants. NP-002-04
20. Collection, Reprocessing and Storage Solid Radioactive Waste. Safety Requirements. NP-020-2000
21. Collection, Reprocessing and Storage of Liquid Radioactive Waste. Safety Requirements. NP-019-2000
22. Gaseous Radioactive Waste Management. Safety Requirements. NP-021-2000.
23. Rules on design of emergency power supply systems for nuclear plants. PNAE G-9-027-91.
24. Main Requirements for Service Life Extension of NPP Power Unit. NP-017-2000
25. Rules for physical protection of radiation sources, storage facilities, and radioactive substances. NP-083-07
26. Safety in Radioactive Waste Management. General Provisions. NP-058-04
27. Rules for physical protection of radioactive radiation sources on transportation. НП-073-06
28. Guidelines for the In-Depth Safety Assessment of Operational NPP Units (OUOB AS) RB-001-05
29. Requirements for a Content of Safety Justification Report for NPP with Fast-Breeder Reactors. NP-018-05
30. Guidelines for Periodic Safety Assessment of NPP Unit. RB-041-07
31. Accident managements programmes in NPPs. A guidebook. Technical reports series. # 368, IAEA, 1994
32. Implementation of accident management programmes in NPPs. Safety reports series. # 32, IAEA, 2004
33. Requirements to the content of decommissioning program for NPP units RB-013-2000

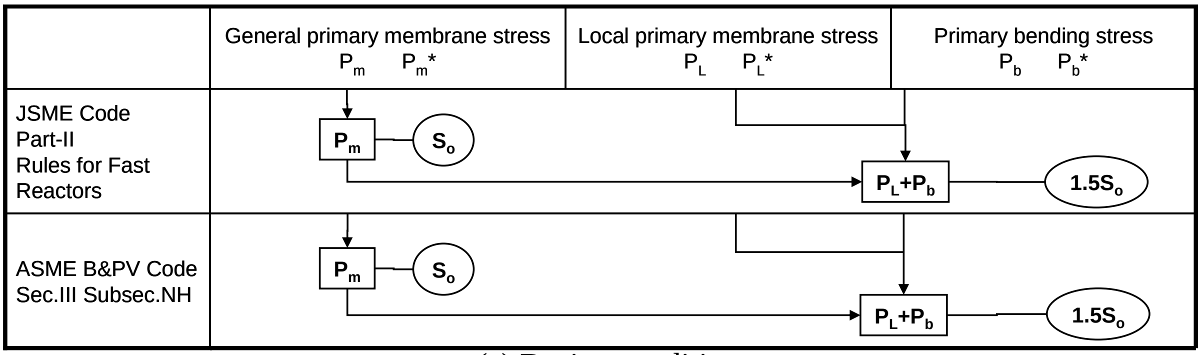
34. Regulations on procedure for investigation and account of violations in NPP operation. NP-004-08
35. Rules of the procedure for emergency situation announcement, expedition data communications and arrangement of urgent assistance to nuclear plants in case of radiation hazardous situation. NP-005-98
36. Standard content of the plan of personnel protection measures in case of accident at nuclear power plant. NP-015-2000
37. Development and review of plant specific emergency operating procedures. Safety reports series # 48. IAEA, 2006

Annex 5

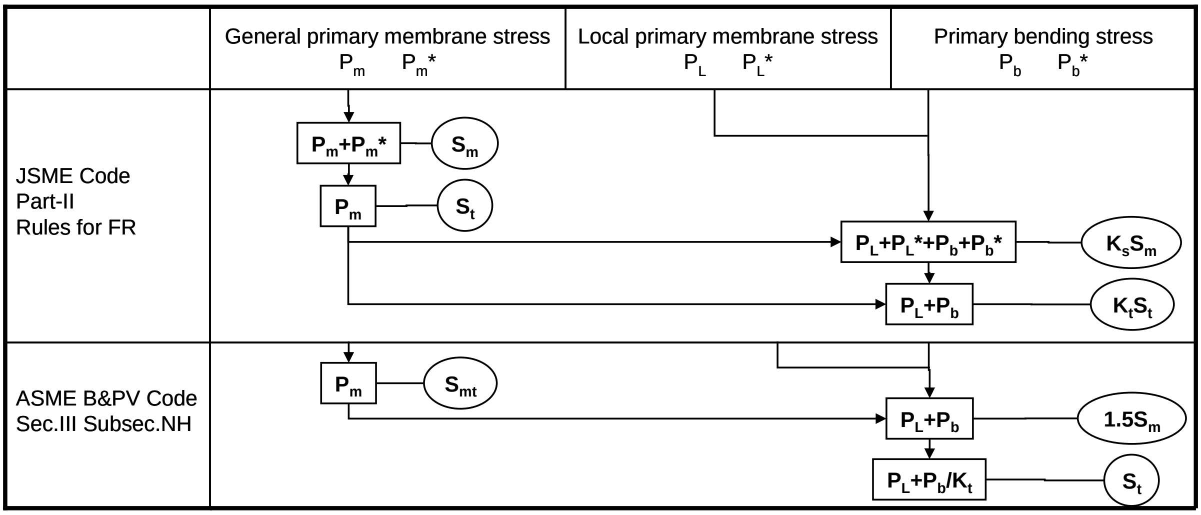
Comparison between ASME and JSME codes

Basically, “Codes for Nuclear Power Generation Facilities - Rules on Design and Construction for Nuclear Power Plants - Part II Rules for Fast Reactors” published by JSME (Japan Society of Mechanical Engineers) is established with reference to ASME Boiler and Pressure Vessel Code Case N-47. The code case N-47 was inherited by the ASME Boiler and Pressure Vessel Code Sec.III Dev.1 Subsec.NH.

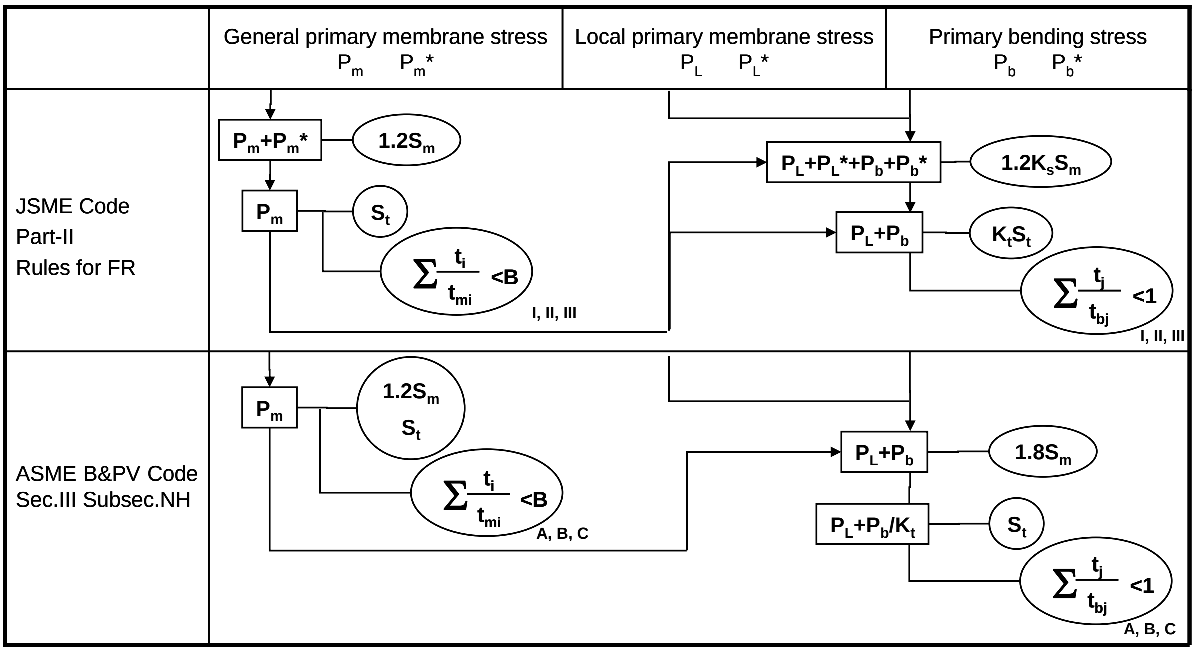
This Annex summarizes in Figures 1 and 2 the major differences between ASME Boiler and Pressure Vessel Code Sec.III Dev.1 Subsec.NH and JSME Codes for Nuclear Power Generation Facilities - Rules on Design and Construction for Nuclear Power Plants - Part II Rules for Fast Reactors.



(a) Design condition

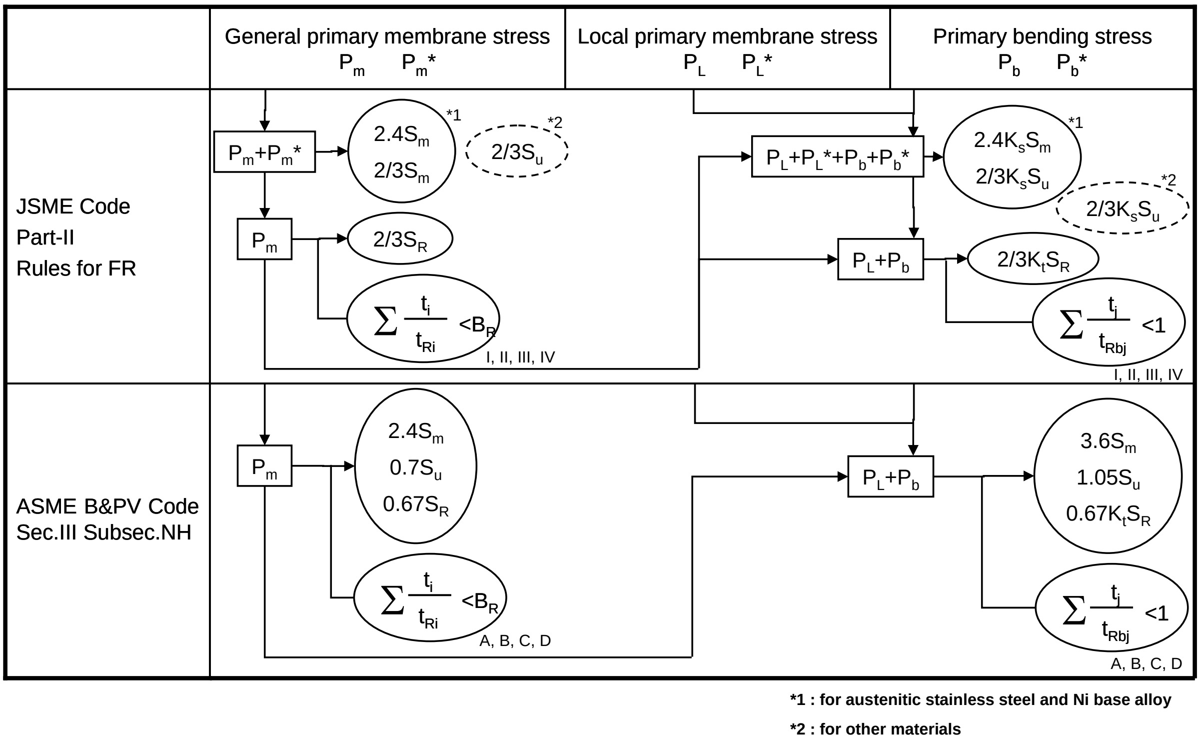


(b) Level A, B service conditions



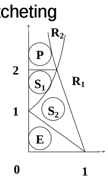
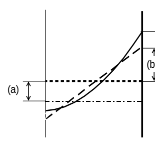
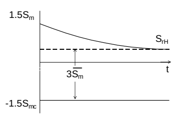
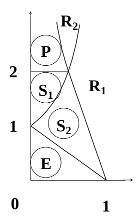
(c) Level C service condition

Fig. 1 Limits of primary stress



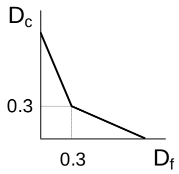
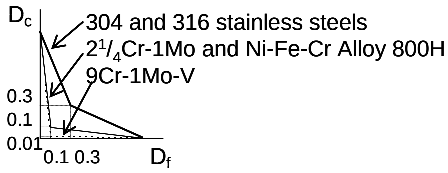
(d) Level D service condition

Fig. 1 (continued) Limits of primary stress

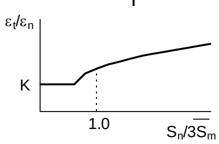
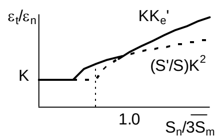
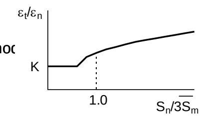
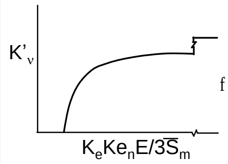
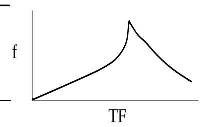
	Service condition	Limits for inelastic strains	Satisfaction of strain limits using elastic analysis		
			Limits for shake-down		Limits for ratcheting
JSME Code Part-II Rules for FR	Level A, B, C service and testing conditions	(a) Strains averaged through the thickness : 1% (b) Strain at the surface, due to an equivalent linear distribution of strain through the thickness : 2%	Creep effect is insignificant	$S_n^* \leq 3S_{mH}$ $\left\{ S_n \leq 3\overline{S_m} \text{ or } S_n \leq \beta_s(3\overline{S_m}),\ S_n' \leq 3\overline{S_m} \right.$	Limits of thermal ratcheting $\left\{ \right.$ Miller model, Bree model 
			Long-term primary stress is small		Limits of creep ratcheting O'donnell & Porowski method 
			General requirement	$\{P_L+P_L^*+P_b+P_b^*/Kt\}_{max}+\langle Q+Q^*\rangle_R<S_a$ S_a is representative value of S_y considering creep ※	
ASME Code Sec.III Subsec.NH		(a) Strains averaged through the thickness : 1% (b) Strains at the surface, due to an equivalent linear distribution of strain through the thickness : 2% (c) Local strains at any point : 5%	Test No.A-1		
			Test No.A-2	$\{P_L+P_b/Kt\}_{max}+\langle Q\rangle_R<S_y$ ※	
			Test No.A-3	$S_n<\text{Min.}[3S_m, 3\overline{S_m}]$	
			Test No. B-1,B-2, B-3	—	

※These limits are only to permit elastic strain.

Fig. 2 Limits of strain

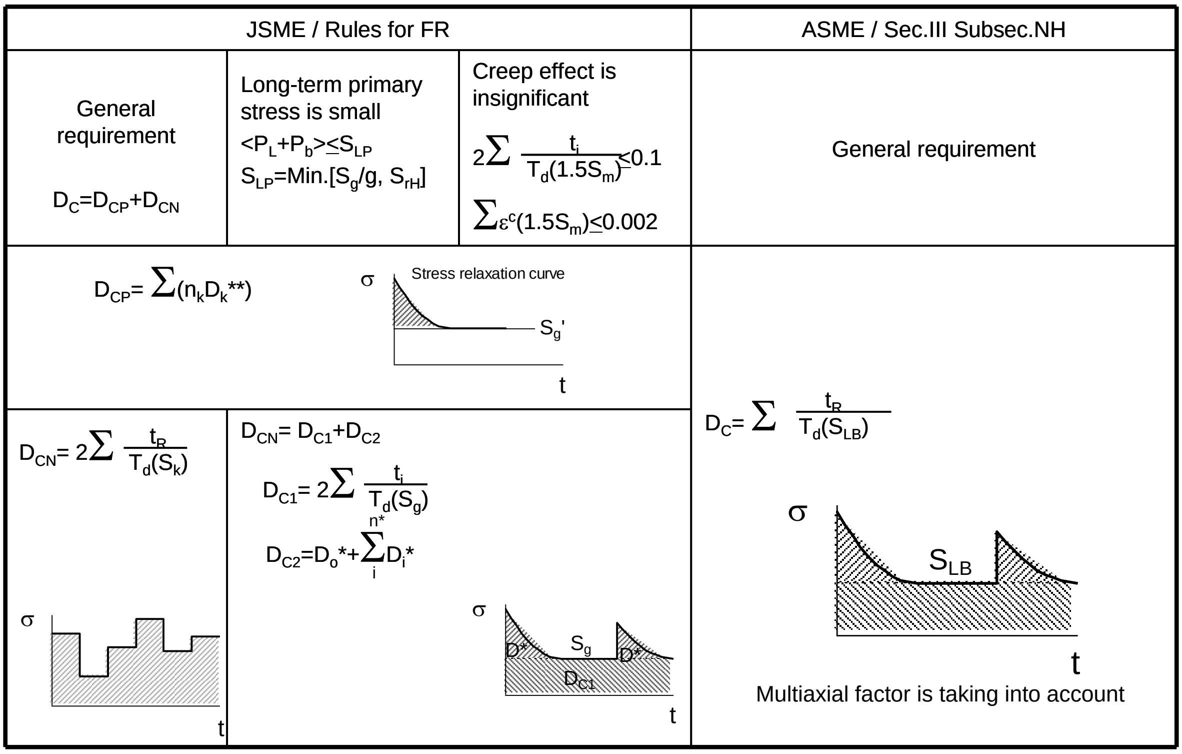
JSME Code Part-II Rules for Fast Reactors	ASME B&PV Code Sec.III Subsec.NH
$D_f+D_c\le D$ $D_f=\sum\frac{n_i}{N_{di}}$ $D_c=2\int\frac{dt}{T_d(\sigma)}$ $D_C=D_{CP}+D_{CN}$ 	$D_f+D_c\le D$ $D_f=\sum\frac{n_i}{N_{di}}$ $D_c=\int\frac{dt}{T_d(\sigma/K')}$ K'=0.67 for Austenitic stainless steel, Ni-Fe-Cr (Alloy 800H) and $2^1/_4$Cr-1Mo. K'=0.87 for 9Cr-1Mo-V 

(a) Limits of creep fatigue (C/F) damage

JSME / Rules for FR			ASME / Sec.III Subsec.NH
General requirement	Long-term primary stress is small $\langle P_L+P_b\rangle\le S_{LP}$ S_{LP}=Min.[S_g/g, S_{rH}]	Creep effect is insignificant $2\sum\frac{t_i}{T_d(1.5S_m)}\le 0.1$ $\sum\varepsilon^c(1.5S_m)\le 0.002$	General requirement
$\varepsilon_t=(S^*/S)K^2\varepsilon_n+K\varepsilon_c+K_T\varepsilon_F$ 	$S_n\le 3\overline{S}_m$ $\varepsilon_t=(S'/S)K^2\varepsilon_n+K_T\varepsilon_F$ $S_n>3\overline{S}_m$: ε_t=Max.[$(S'/S)K^2\varepsilon_n+K_T\varepsilon_F$, $KK_e'\varepsilon_n+K_T\varepsilon_F$] 		$\varepsilon_t=K_v\varepsilon_{mod}+K\varepsilon_c$ $\varepsilon_{mod}=(S^*/S)K^2\varepsilon_n$ $\varepsilon_{mod}=K^2S^*K\varepsilon_n/\sigma_{mod}$ $\varepsilon_{mod}=K_eK\varepsilon_n$    $K_v=1.0+f(K'_v-1.0)$ Multiaxial factor is taken into account
	Design fatigue strain range taking strain rate effect into account		Design fatigue strain range

(b) Fatigue damage

Fig. 3 Limits of creep fatigue (C/F) damage



(c) Creep damage

Fig. 3 (continued) Limits of creep fatigue (C/F) damage