Structural Design and Evaluation of a Steam Generator in PGSFR

Chang-Gyu Park[1], Hoe-Woong Kim[1], Jae-Hun Cho, Jong-Bum Kim[1], Sung-Kyun Kim[1]

[1]Korea Atomic Energy Research Institute (KAERI), Daejeon, South Korea

E-mail contact of main author: chgpark@kaeri.re.kr

Abstract. A once-through steam generator in sodium-cooled fast reactor converts the sub-cooled feedwater to superheated steam by transferring heat from sodium in the intermediate heat transport system to water/steam and provides superheated steam under normal power conditions. It is a heat exchanger as well as a structural barrier between liquid sodium and water/steam and thus is regarded as one of the most critical components in the sodium-cooled fast reactor deciding the plant reliability and availability. The prototype Gen-IV sodium-cooled fast reactor(PGSFR) employs a vertical once-through shell-and-tube heat exchanger which has a sodium-to-water counterflow with single-walled straight heat transfer tubes. Its construction material is the 9Cr-1Mo-V steel for high heat transfer performance and high allowable strength at elevated temperature. It uses the straight tube so that it is possible to apply the single piece tube without any weld for intertube joint. Since straight tubes do not provide enough flexibility to accommodate the longitudinal expansion difference among tubes, flow distributors are applied for uniform flow at inlet shell. And the expansion bellows joint was joined on the main shell to provide the large flexibility to compensate for differential thermal expansion between shell and tube bundle. In this study, structural design for a PGSFR SG was described and structural integrities against representative operating duty cycle event were evaluated by ASME B&PV Section III Division 5. From the evaluation results, the structural design issues were drawn and feasibility of design improvement was discussed.

Key Words: PGSFR, Steam Generator, Structural Design, Structural Integrity

1. Introduction

A sodium-cooled fast reactor(SFR) has been widely recognized as a technical alternative to effectively manage the spent nuclear fuel. In Korea, the Korea Atomic Energy Research Institute(KAERI) has been developing SFR design technology and the national project to develop the prototype Gen-IV sodium cooled fast reactor(PGSFR) was initiated in 2012[1]. The PGSFR is a pool-type reactor having two independent intermediate heat transport system(IHTS) loops with two steam generators.

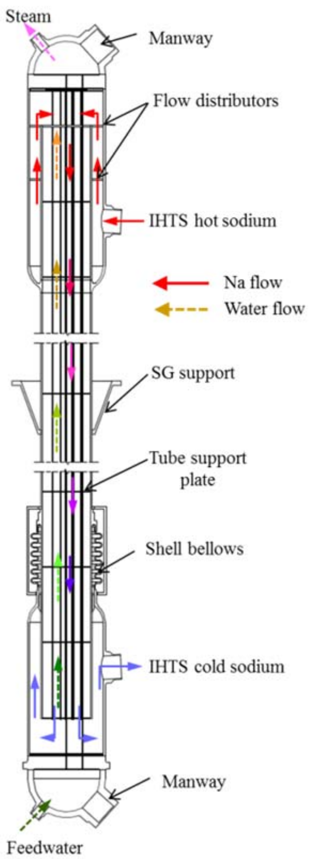
Steam generator is a heat exchanger as well as a structural barrier between liquid sodium and water/steam and thus it is regarded as one of the most critical components in SFR system because there is the possibility of a sodium-water reaction(SWR). Thus, high reliability is required. Various kinds of SG concepts including heat transfer tube shapes have been studied in view of structural integrity, hydraulic performance and economics[2]. Unfortunately, unanimity on design of SFR steam generator has not yet reached unlike inverted U tube steam generator in thermal nuclear reactor plants. In general, straight tube SGs provide advantages such as the absence of tube-to-tube welding and easier tube inspection compared with long helical coiled tube SGs. And, it would be more economic considering the simple manufacturing process. The difference in thermal expansion between a tube bundle and a SG shell structure, which is a potential drawback of the straight tube SG, could be resolved by applying a shell bellows in the bottom region of the SG cylindrical shell.

In this study, a steam generator for PGSFR is selected as s monolithic integral once-through type component with single-walled straight tube through several design option studies, while providing a couple of future considerations for structural shapes, fabrication process, and inspection methods. For the structural integrity evaluation, design loads for service condition were classified and the corresponding structural analyses for each design load were carried out. Then, structural integrities under the service levels were evaluated according to ASME Division 5 Code rule and the results were discussed.

2. Design Features of PGSFR SG

2.1.Structural Design

The PGSFR SG is a vertically oriented, shell-and-tube type heat exchanger as shown in FIG.1. The function of the SG is to transfer the heat from secondary sodium system to feedwater to generate superheated steam. The SG is a once-through integrated type with single wall straight tubes. Feedwater enters inside of tubes at the bottom chamber of the SG through the feedwater nozzles, flows through the orifice at the tubesheet and flows upwards in the counter-flow direction to the downcoming sodium. There is a need to absorb the differential expansion between shell and tubes and the design concept employing the shell bellows accommodates the expansion difference between them. It is located in the bottom portion of SG shell above the sodium outlet nozzle so as to permit higher allowable stress. The tubes are supported and guided at various locations by tube support plates. Tube support plates are supported by 12 tie-rods which are fixed to upper tubesheet. Upper and lower tubesheets are protected by thermal shields and a manhole is provided on each water chamber and steam header to perform the inservice inspection for tubes and tube plugging if required.



Design parameters	Values
Heat Capacity [MWt]	196
Feed Water Flow rate [kg/s]	86.85
Feed Water Temperature [℃]	240
SG Inlet Pressure [MPa]	18
SG Outlet Pressure [MPa]	16.7
Steam Exit Temperature [℃]	503
Sodium Flow rate [kg/s]	782.45
Sodium Inlet Temperature [℃]	528
Sodium Outlet Temperature [℃]	332
Tube Type	Single wall straight
Tube Material	9Cr-1Mo-V
Tube OD/ID [mm]	17.3/12.7
Number of Tubes	769
Tube Length [m]	25.38
Heat Transfer Area [m^2]	1060.85
Tube Pitch to Diameter Ratio	1.867
Tube Side Pressure Loss [kPa]	281.99
Shell Side Pressure Loss [kPa]	212.28
Tube Plugging Margin [%]	10

FIG.1. Structural concept and design parameters of a PGSFR steam generator

The unit is designed for 196 MWth and generates steam at 16.7 MPa and 503˚C. The intermediate sodium flow rate is 782.15 kg/s and the inlet and oulet sodium temperatures are 528˚C and a 332˚C, respectively. The flowrate of feedwater from feedwater system to SG is 86.85 kg/s at 240˚C and superheated steam is generated by passing through straight tubes. The overall size of the unit is about 28.08 m in height and 1.75 m in outer diameter. Total dry weight is around 95 tons per unit. SG sizing parameters are listed in FIG.1 [3].

The SG unit is fully filled with secondary sodium. The IHTS expansion tank, containing cover gas region, is installed at IHTS cold piping to accommodate the sodium expansion or contraction due to a rapid change in the sodium temperature and pressure pulse associated with a potential SWR and provide the space for the installation of the leak detection instruments.

2.2. Material Selection

The PGSFR nuclear steam supply system(NSSS) is designed, fabricated, and examined in compliance with ASME B&PV Section III Division 5, which is a construction code for high temperature reactors[4]. One of the design principles in selecting the design materials is based on the NSSS design to avoid the dissimilar metal weld joints in entire safety related NSSS components.

The steam generator in PGSFR is designed according to Class A component design rule. Based on the ASME Code, available design structural materials are Type 304 stainless steel, Type 316 stainless steel, 2.25Cr-1Mo steel, Alloy 800H, and 9Cr-1Mo-V steel. The design materials given above are recommended to be used as the candidate structural materials for Gen-IV systems at elevated temperature conditions.

Specially, heat transfer tubes are exposed at 528℃ and 18.0 MPa pressure under normal operating conditions. Thus, the time-dependent allowable stress intensity as well as the thermal conductivity for heat transfer is important design properties. FIG. 2 shows the mechanical properties of materials implemented in Division 5 Subsection HB. 2.25Cr-1Mo and 9Cr-1Mo-V steels show good performance in thermal conductivity compared with other 3 materials above. And 9Cr-1Mo-V steel has much higher allowable stress intensity compared with 2.25Cr-1Mo steel around 525℃ which is the normal operating temperature of SG. Moreover, it is well known that 9Cr-1Mo-V steel has high resistance to stress corrosion cracking and is advantageous to overcome the problem of carbon transfer associated with 2.25Cr-1Mo steel. In this respect, 9Cr-1Mo-V tube is selected as a material for steam generator tube. Shell structures and tubesheets are also made of the same material to avoid the dissimilar metal weld joints.

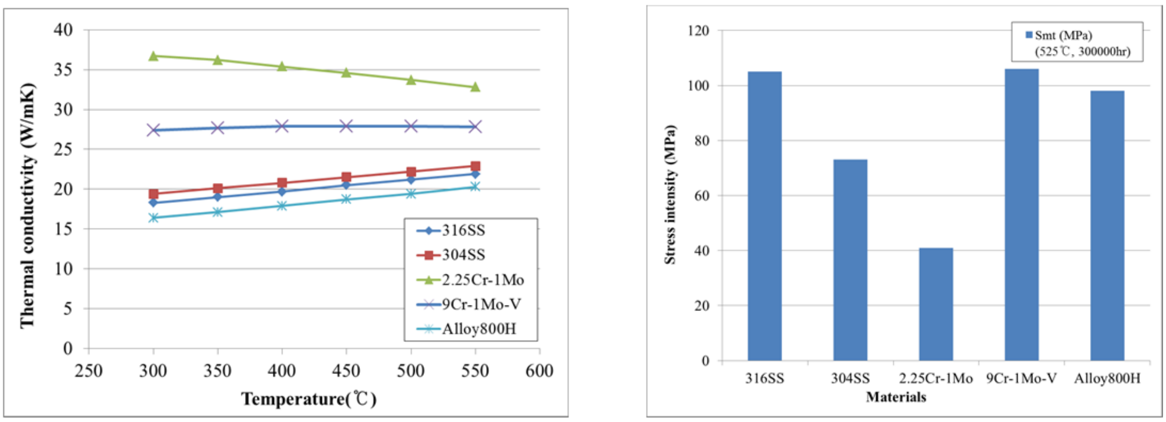


FIG. 2. Mechanical properties in ASME Section III Division 5 Subsection HB

2.3.Tube Type selection

Typically, heat transfer tube shape in SFR SG can be a straight tube, helical coil, J-tube or bended tube. Recently, The SG tubes were narrowed down to the helical coil tube and the straight tube except bended tube in India[2,5]. For the helical coil steam generator, it is advantageous to accommodate the axial thermal expansion because of its flexibility. However, these tubes are very long and have different lengths. Besides, there is also a worry about cross flow and vibration of the tubes. The clamps are therefore necessary to hold the various tubes in their place to prevent unnecessary movement of the tubes. But these clamps need to accommodate expansion and contraction during heating and cooling of the SG. Inservice inspection(ISI) of the steam generator may be problematic due to the long length of the helical tube. During fabrication, these long tubes are fabricated from smaller length tubes that are welded together. It is critical that these tubes are aligned correctly during welding to ensure that the ISI instrument can pass through this part of the tube. For sodium-heated steam generators, it is preferable to have boiling occurring at the same elevation in the SG but it will occur at different elevations because the tubes have different lengths.

On the other hand, a straight-tube sodium-heated steam generator has a larger number of tubes but the tube length is shorter. And the tubesheet is thicker than that of a helical coil tube SG. One of important problems in a straight tube SG is how to join the tubes to the tubesheet. Some have used a boss on the tubesheet that has the same dimensions as the tube. The uniform flow paths for the water and steam are required to ensure the stable steam exiting the top of the sodium-heated SG. Eventually, it can enhance the tube integrity by generating the uniform axial expansion for all heat transfer tubes. The other issue in straight tube SG is that there is thermal expansion difference between the tubes and the shell. It is expected to be accommodated by implementing a bellows that was machined out of a forging and welded to the SG shell bottom region.

PGSFR adopts the single wall straight tube for a steam generator based upon the several case studies. To apply the straight tubes in PGSFR SG, two technical issues are shell bellows development for 9Cr-1Mo-V steel and the uniformity of tubes' outlet temperatures under all operating conditions. From the thermal hydraulic analysis results, the uniformity of tubes' outlet temperature will be evaluated in the structural analysis.

The PGSFR SG adopts the straights tube without any mechanical features to accommodate the non-uniformity of temperature distribution among tubes. Whereas, the PFBR SG in India has straight tubes with an area where tubes are bent, which gives some flexibility to the tubes in case of various thermal expansions among them. This could be a future option for the PGSFR SG in case of difficulty to demonstrate thermal uniformity among tubes.

2.4.Tube-to-Tubesheet Joint

The tube-to-tubesheet joint is an important concern in SFR SG because joint region is a boundary between sodium and water and welding section which might be a potential defect region is unavoidable. The tubes in steam generator can be generally welded to tubesheet by expansion joint with seal weld, expansion joint with spigot side fillet weld or in-bore weld joint with raised spigot. Currently, the steam generator tubes in PGSFR are welded to tubesheet by expansion joint, especially hydraulic expansion joint as shown in FIG. 3. This method yields simple manufacturing of tubesheet and provides easy accessibility for manufacturing and inservice inspection of tube to tubesheet weld joints.

In conventional PWR's SG, this method provides the minimized residual stress and reliable joint with leak tightness which is already well proven. But the hydraulic expansion and weld joint from PWR technology needs to be studied in more details because tube-to-tubesheet weld joint will need post-weld heat treatment and the temperature of operation for the top weld is in creep region. Moreover, since the tube material of 9Cr-1Mo-V steel which has high yield strength with lower elongation property, it is required higher expansion pressure and liable to get cracks during expansion. Applying this method in 9Cr-1Mo-V steel tube has not been proven yet and the mock-up test needs to be followed to establish the manufacturing parameters and quality assurance. Welding the tube to the raised boss on the tubesheet maybe an alternative option for tube-to-tubesheet joint.

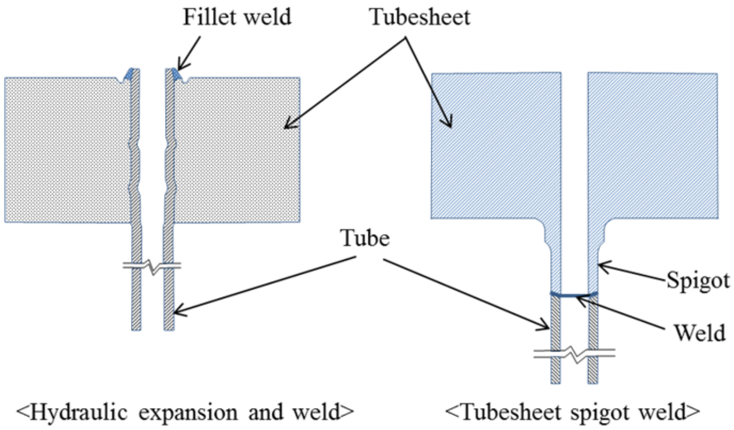


FIG. 3. Tube-to-tubesheet joint

2.5.Inservice Inspection

Since the SG provides a boundary between sodium and water, a serious accident can occur due to the potential SWR if there is any sodium leakage. Therefore, inservice inspection (ISI) of the SG to detect any defect to prevent SWR especially in SG tubes is quite important. For ISI of SG tubes, the volumetric examination and continuous monitoring are conducted according to ASME B&PV Code Sec.XI Div.3 [6]. To enhance the safety, the combined inspection sensor which can perform remote field eddy current testing (RFECT) and magnetic flux leakage testing (MFLT) simultaneously will be employed for the volumetric examination of ferromagnetic 9Cr-1Mo-V steel tubes. For the circumferential groove as shown in FIG. 4, even 10% defect was well detected. For the continuous monitoring, hydrogen meters will be installed near the sodium outlet nozzle of the SG and at expansion tank [7].

For ISI of tube-to-tubesheet joint, the volumetric examination is conducted. If tube-to-tubesheet is connected by the expansion joint, the combined inspection sensor used for ISI of SG tubes will be employed and thus the tube and tube-to-tubesheet can be inspected together. But for the seal welds, if exist, the ultrasonic testing will be additionally conducted. On the other hand, the insertion-type phased array or rotating ultrasonic sensor may be employed for the volumetric examination if tube-to-tubesheet is connected by the spigot welding.

The visual examination (VTM-2) and continuous monitoring are conducted for ISI of the SG shell according to ASME B&PV Code Sec. XI Div. 3. The continuous monitoring will be conducted by sodium leak detectors, such as a spark plug and a smog detector, installed outside the shell. For ISI of the supporting structure of the SG, the visual examination is conducted and the surface examination such as liquid penetration method is additionally conducted to enhance the safety[8]. ISI of supporting structure will be conducted for all welds in each inspection interval and the insulation should be removed during inspection. The dimensional gauging to investigate the tilting and position change of the SG should be conducted during the visual examination.

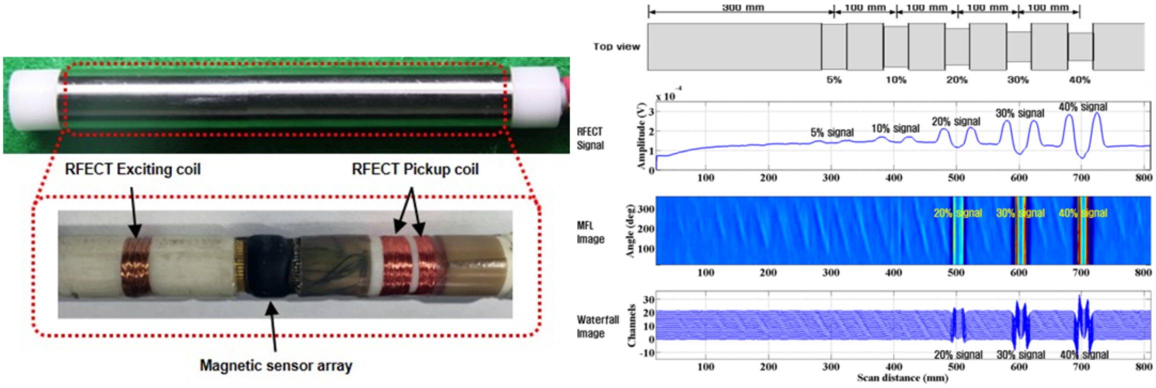


FIG.4. Combined inspection sensor and performance test results for circumferential grooves

3. Structural Integrity Evaluations

3.1.Evaluation Models

The structural analyses are carried out by finite element method using ANSYS software[9]. If full 3-D model is applied for analysis, it increases computing time because of lots of tube and large geometry with thin wall. To simplify the analysis model, two approaches are employed; part models for critical portions are applied and the tubes are not included in analysis model. Note that the mass of tubes is reflected on the tubesheet as a mechanical load. The selected parts for structural analyses are upper head, support structure, bellows and lower head. Upper head includes the steam header, upper tubesheet and main shell, and tubesheet has hundreds of holes for tube insertion. Lower head is almost same as upper head except hemisphere for steam and feedwater. An axisymmetric element is applied for support structure and bellows but 3-D element is applied for upper head and lower head because of non-symmetric tube arrangement. FIG.5 shows the analysis models for structural analysis of a steam generator. Element types used in heat transfer and stress analyses are PLANE55, PLANE182 for axisymmetric model and SOLID70, SOLID175 for solid model, respectively.

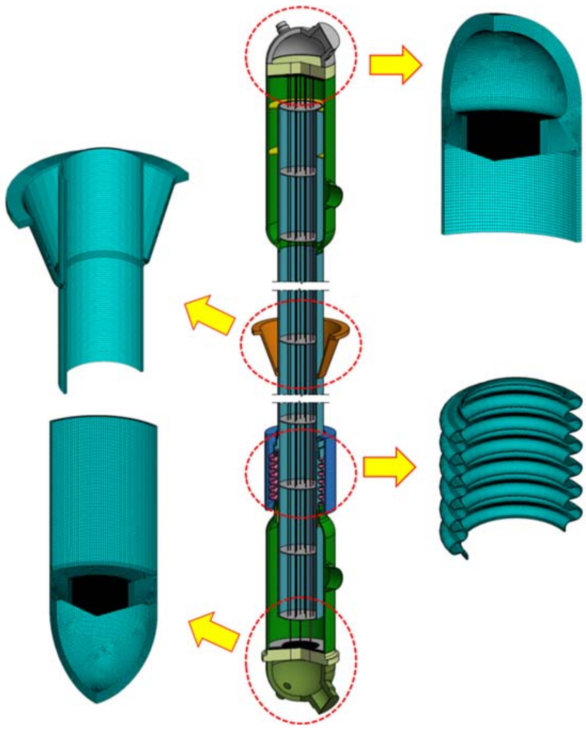


FIG.5 Finite element models for structural analysis of a steam generator

3.2.Design Loading Condition

To ensure the structural integrity, the component should satisfy the limits of design loading condition in advance. The primary loads on SG for design loading are dead weight and design pressure. The design pressure on SG shell is set to be 3.5 MPa by virtue of potential SWR pressure. Other design pressures in steam header and feedwater chamber are 18.4 MPa and 20.0 MPa, respectively. FIG. 6 shows the stress intensity distributions for design loading condition. The structural integrity was evaluated for critical sections according to ASME Section III Division 5 Subsection HB rule and results are summarized in TABLE I. The most critical section against design loading condition is a steam header but all sections satisfy the design criteria with design margin over 25% at least.

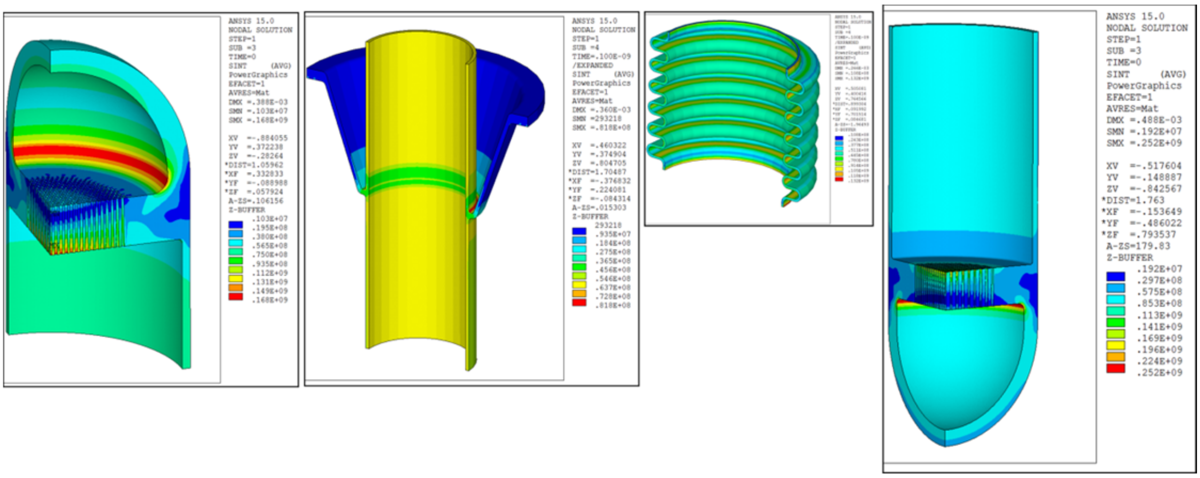


FIG. 6. Stress intensity distributions for design loading condition

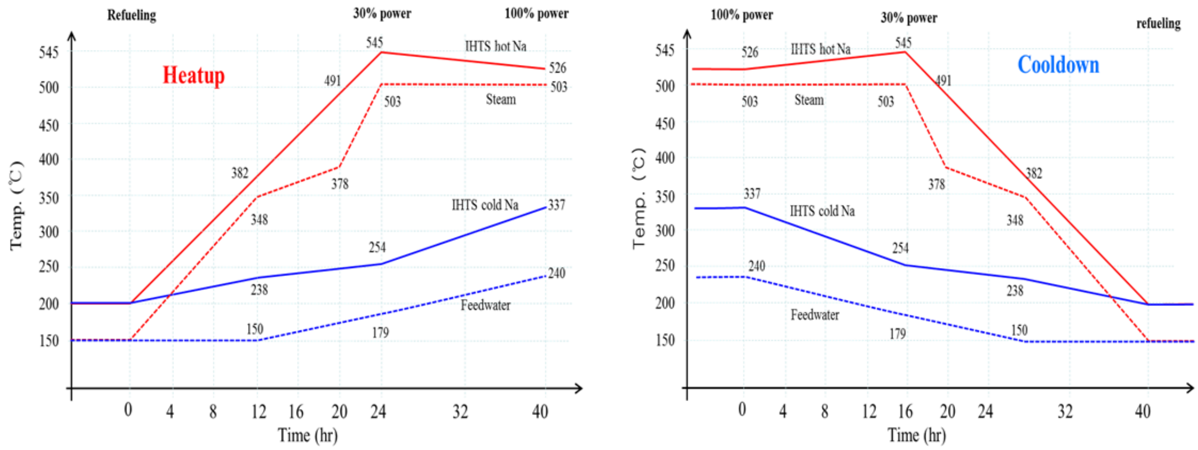
TABLE I. STRUCTURAL INTEGRITY EVALUATION RESULTS FOR DESIGN LOADING

Sections		Criteria	Calculated (MPa)	Allowable (MPa)	Margin
Steam Header	Inner	P_m	47.4	97.8	1.06
		$P_L + P_b$	116.7	146.7	0.26
	Outer	P_m	47.4	97.8	1.06
		$P_L + P_b$	21.9	146.7	5.07
Main Shell	Inner	P_m	60.3	97.8	0.62
		$P_L + P_b$	63.4	146.7	1.31
	Outer	P_m	60.3	97.8	0.62
		$P_L + P_b$	57.2	146.7	1.56
Support	Inner	P_m	57.0	97.8	0.72
		$P_L + P_b$	58.9	146.7	1.49
	Outer	P_m	57.0	97.8	0.72
		$P_L + P_b$	55.3	146.7	1.65
Water Chamber	Inner	P_m	53.9	170.3	2.16
		$P_L + P_b$	144.9	255.5	0.76
	Outer	P_m	53.9	170.3	2.16
		$P_L + P_b$	37.7	255.5	5.78

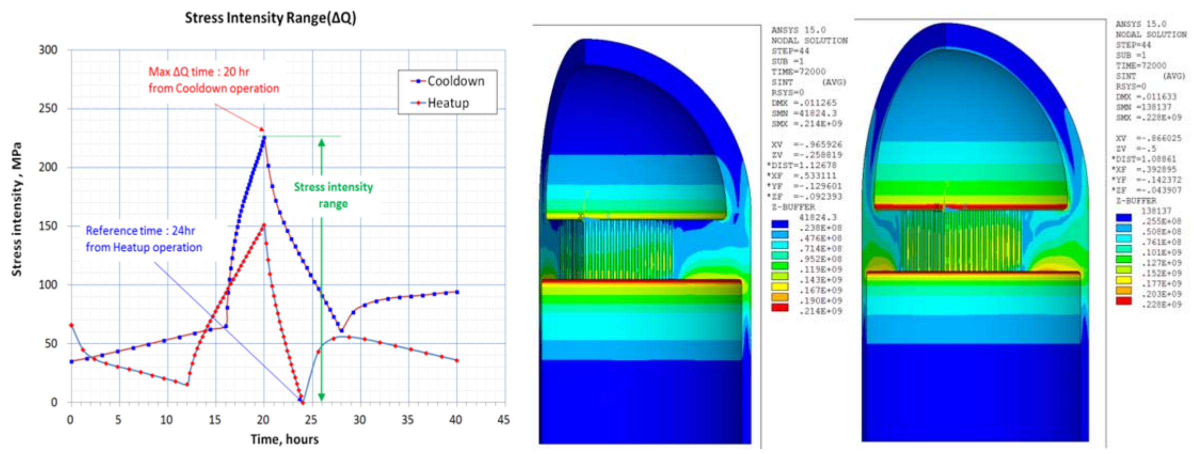
3.3.Level A Service Condition

Apart from design loading condition, operating cycle event of Level A service is also included in structural integrity evaluation. In this evaluation, refueling cycle event is regarded as an operating cycle of Level A service. Because upper and lower tubesheets are the thermal barrier between sodium and water, thermal load is more severe at both heads rather than support structure and bellows. Upper head is selected as an evaluation region for thermal transient analysis by considering the fluid temperature behavior. From the heat balance diagram, temperature and pressure for sodium and feedwater/steam are 528℃, 503℃ and 16.7 MPa, 0.5 MPa respectively at full power condition. FIG.7 shows the temperature time histories of heatup and cooldown operations for refueling cycle and thermal stress intensity at the time of maximum stress range. The cooling rate for cooldown operation is assumed to be the same as that of heatup operation with opposite sign.

Design lifetime and its availability is assumed to be 60 years and 80%, respectively. TABLE II shows the results of the structural integrity evaluations for refueling cycle. The steam header section satisfies the design criteria for primary stress, inelastic strain, use-fraction sum with primary stress, and creep-fatigue damage. For the creep damage evaluation, its design margin is not so high because of its long lifetime at high temperature because the heating rates were assumed to be very conservative for heatup and cooldown operation in this analysis.



(a) Time histories of fluid temperature for heatup and cooldown operations



(b) time history of stress intensity range *(c) stress distribution at max. stress intensity range*

FIG. 7. Thermal stress analysis results for refueling cycle event

TABLE III. STRUCTURAL INTEGRITY EVALUATION RESULTS FOR LEVEL A SERVICE

Section		Criteria	Calculated	Allowable	Margin
Steam header	Inner	P_m (MPa)	44.5	121.9	1.74
		UFS (t/t_m)	420480	1987700	0.21
		$P_L + P_b$ (MPa)	116.8	217.7	0.86
		$P_L + P_b/K_t$ (MPa)	102.4	121.9	0.19
		UFS (t/t_b)	420480	815620	0.52
		Strain limits	0.61	1.0	0.67
		Fatigue damage	0.00	0.02	>>10
		Creep damage	0.95	1.0	0.05
	Outer	P_m (MPa)	44.5	118.1	1.65
		UFS (t/t_m)	420480	1935900	0.22
		$P_L + P_b$ (MPa)	28.8	215.4	3.60
		$P_L + P_b/K_t$ (MPa)	14.8	118.1	6.98
		UFS (t/t_b)	420480	2546900	0.17
		Strain limits	0.42	1.0	1.38
		Fatigue damage	0.00	0.36	>>10
		Creep damage	0.27	1.0	2.70

4. Conclusions

The structural design for a PGSFR steam generator with 196 MWth capacity is established through the comparison studies for several design options. It is a once-through heat exchanger made of 9Cr-1Mo-V steel and has straight heat transfer tubes to remove the intertube joint. The tubes can be jointed to tubesheet by hydraulic expansion and strong fillet weld, whereas raised spigot weld is alternative option for it. The combined inspection sensor, which can perform remote field eddy current testing and magnetic flux leakage testing simultaneously, is under development and it will be employed for the volumetric examination of ferromagnetic 9Cr-1Mo-V steel tubes.

The structural analyses of a PGSFR steam generator are carried out and its structural integrities under the given service levels are evaluated in accordance with ASME Code rule. The design loads according to design condition and operating cycle event are classified and stresses calculated from structural analyses are linearized and summarized in their stress components. As a result, the SG structure satisfies the design criteria for both service levels even though the very conservative operating assumption. And, additional evaluation considering various operating events will be followed.

Acknowledgements

This work was supported by the Nuclear Research &Development Program of the National Research Foundation with a grant funded by the Korean Ministry of Science, ICT and Future Planning (No. 2012M2A8A2008157).

References

[1] Jaewoon Yoo, J. Chang, J.-Y. Lim, J.-S. Cheon, T.-H. Lee, S.K.Kim, K.L.Lee, and H.-K. Joo, “Overall System Description and Safety Characteristics of Prototype Gen IV Sodium Cooled Fast Reactor in Korea,” Nuclear Engineering and Technology, Vol.48, pp.1059-1070 (2016).

[2] N. Kisohara, T. Moribe, T. Sakai, “Temperature and Flow distributions in sodium-heated large straight tube steam generator by numerical methods,” Nuclear Technology, Vol.164, pp.103-118 (2008).

[3] Steam Generator Sizing Calculation for PGSFR, SFR-330-DF-302-004, Korea Atomic Energy Research Institute (2016).

[4] ASME Boiler and Pressure Vessel Code, Section III Division 5, High Temperature Reactor (2011).

[5] R. Nandakumar, P. Selvaraj, S. Athmalingam, V. Balasubramaniyan, S. C. Chetal, “Thermal simulation of sodium heated once through steam generator for a fast reactor,” Int. J. Adv. Eng Sci Appl Math Vol.4(3), pp.127–137(2012).

[6] ASME Boiler and Pressure Vessel Code, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components”, Division 3 (2006).

[7] Jong-Bum Kim, Ji-Young Jeong, Tae-Ho Lee, Sungkyun Kim, Dong-Jin Euh, and Hyung-Kook Joo, “On the Safety and Performance Demonstration Tests of Prototype Gen-IV Sodium-Cooled Fast Reactor and Validation and Verification of Computational Codes,” Nuclear Engineering and Technology, Vol.48, pp.1083-1095 (2016).

[8] ASME Boiler and Pressure Vessel Code, Section V, “Nondestructive Examination”, (2011).

[9] ANSYS Users manual, Release 15.0, ANSYS Inc..