FEASIBILITY OF AN INTEGRATED STEAM GENERATOR SYSTEM IN A SODIUM-COOLED FAST REACTOR SUBJECTED TO ELEVATED TEMPERATURE SERVICES

GYEONG-HOI KOO* and JAE-HAN LEE
Korea Atomic Energy Research Institute
1045 Daedeok-daero, Yuseong-gu, Daejeon, 305-353, Korea
*Corresponding author. E-mail : ghkoo@kaeri.re.kr

Received January 21, 2009
Accepted for Publication April 15, 2009

As one of the ways to enhance the economical features in sodium-cooled fast reactor development, the concept of an integrated steam generator and pump system (ISGPS) is proposed from a structural point of view. And the related intermediate heat transfer system (IHTS) piping layout compatible with the ISGPS is described in detail. To assure the creep design lifetime of 60 years, the structural integrity is investigated through high temperature structural evaluation procedures by the SIE ASME-NH computer code, which implements the ASME-NH design rules. From the results of this study, it is found that the proposed ISGPS concept is feasible and applicable to a commercial SFR design.

KEYWORDS : Sodium-Cooled Fast Reactor, Integrated Steam Generator, ASME-NH, SIE ASME-NH, High Temperature Structural Design, Creep Design

1. INTRODUCTION

One of main issues in sodium-cooled fast reactor (SFR) design is enhancement of economic factors for construction of a commercial plant [1,2]. However, it is not easy to accomplish this goal, especially from a conceptual design stage, without reference reactors. Furthermore, the estimation of degree of economic enhancement for the proposed design concept is also very difficult work due to the lack of construction and operation experience for commercial SFRs. There can be many items from which to extract economical parameters, such as a R&D program cost, fuel cycle cost, engineering cost, construction and operation cost, power generation cost, and so on. From the aspect of a structural contribution in pursuit of an economical advantage, a simplification of the main components can be one way to engineer the design, fabrication, and maintenance.

In general, there are two types of SFR systems selected as candidate GEN-IV reactors, loop-type and pool-type. The former is a representative SFR under development in Japan [3] and the latter is a KALIMER [4-6] under development by KAERI in Korea. For the SFR safety design, there is an Intermediate Heat Transfer System (IHTS) consisting of piping and secondary pump to prevent irradiated primary sodium from direct contact with feed water of the steam generator during tube rupture accidents. As for the secondary pump, a mechanical or electro-magnetic pump can be used. The latter is known to have an advantage for a small capacity reactor (less than 600MWe). For a larger capacity, it is necessary to introduce a mechanical pump due to overall performance. When using a mechanical pump for a commercial plant, there may be several problems that need to be resolved for secondary sodium leaks and maintenance, vibrations, complex pipe layout, and so on.

To overcome the disadvantages in the conventional SFR concepts and to enhance the economical design features, several concepts of the ISGPS (Integrated Steam Generator and Pump System) are proposed and the feasibility of the proposed concept is discussed in this paper from a structural point of view. Also, the related IHTS piping layout and the Intermediate Heat Exchanger concepts are described in detail. One of the main concerns with the new concept is a high temperature structural issue related with the thermal expansion of the proposed IHTS piping layout. To check on the high temperature structural integrity of the piping design, creep design evaluations are carried out for two assumed representative operating types such as the steady state condition and the 72 hours without an operator's action accident condition. For evaluation of a 60 year design lifetime by the ASME-NH code [7], providing a limited set of design data for 3×10^5 hours ($\approx$34 years), the SIE ASME-NH computer code [8], which can extrapolate design data, is used.

2. DETERMINATION OF THE TARGET STRUCTURAL DESIGN GOALS

To propose a new component concept for a sodium-cooled fast reactor that is satisfying from an economical aspect, it is necessary to specify the design target goals that can guarantee a reasonable profit, especially those that relate to the economic enhancement. These design goals can be addressed in terms of nuclear core, thermal hydraulic design advantage, safety, mechanical structures, and so on. In this paper, the structural point of view is mainly considered as the design goal in developing the new design concept of the main components.

The main components considered in this paper include the Integrated SG and Pump System (ISGPS), the IHTS Piping system, and the IHX (Intermediate Heat Exchanger). The concept of the ISGPS affects the layout of the IHTS piping system and, sequentially, the IHX configurations. Therefore, the structural design goals are prepared to satisfy these components.

The new design concept of the ISGPS will meet the specified top tier target structural design goals such as

- New Advanced Design Concept
- Simplifications
- 60 Year Design Lifetime
- Seismic Integrity
- Fabrication Availability
- In-service Inspection Guarantee
- Resolution of Structural Design Issues

The goal of the new advanced design concept will be to establish a creative design concept compared with those of worldwide sodium-cooled fast reactors. The simplification and the 60 year design lifetime of the components and the systems are very important, which is closely related with the economics. The assurance of the seismic integrity is one of the important factors in the sodium-cooled fast reactor due to an unavoidable thin shell structural adoption for a component design in an elevated temperature service. The fabrication of a large size pipe or a new material application for the components should be available in the new concept development. Furthermore, the new concept should guarantee an in-service inspection during an overhaul period. As a final target goal of the new concept, the structural design issues related with the elevated temperature service, which may cause excessive deformation, creep rupture, creep-fatigue damage, and creep buckling must be resolved by the ASME Boiler and Pressure Vessel Code Section III, Subsection NH [7] rules with an adequate structural design avoiding structural discontinuities, dissimilar welding, elastic follow-up regions, and so on.

Table 1 presents a comparison of the design features among commercial size SFRs being developed in the world [9].

Table 1. Inter-Comparison of the Design Features among Commercial Size SFRs

	Net Power (MWe)	No. of Primary Pump	No. of IHTS Loop	IHTS HL OD/TK (mm)	IHTS HL Piping material	RV ID/TK (m/Cm)	RV Height (m)	RV Material	Core Size (m)
Super-Phenix 1	1242	4 (M)	4	700/11	Cr18Ni12Mo 2.5Mn1.8Si	21/(2.5.6.0)	17.3	316	3.7
Super-Phenix 2	1440	4 (M)	4	760/-	316	20/(2.0.3.5)	16.2	316	3.97
SNR 2	1497	4 (-)	8	800/-	304	15/-	-	304	4.3
DFBR	660	3 (M)	3	711/12.7	304	10.4/5	16	316FR	2.99
CDFR	1500	4 (M)	4	864/10	316	19.22/2.5	18.1	316	3
BN-1600	1600	3 (M)	6	820/13	Cr18N9	17/2.5	14	Cr18Ni9	4.45
BN-800	800	3 (M)	3	630(820)/12	Cr16Ni11Mo3	12.9/3	14	Cr18Ni9	2.561
EFR	1580	3 (M)	6	711/11	Cr18Ni13	17.2/3.5	15.9	316	4.051
ALMR	303	1 (E)	1	711/13	316	9.118/5.1	19.355	316	2.164
SVBR-75/100	101.6	2 (M)	None	None	None	4.13/3.5	7	Cr18Ni9	-
BN-1800	1800	3 (M)	6	820/12	TBD	17/2.5	19.95	Cr18Ni9	5.167
BREST-1200	1200	4 (M)	None	None	None	9/5	18.6	Cr16Ni10	4.75
JSFR	1500	2 (M	2	1117.6/14.3	12Cr	10.7/3	21.2	316FR	5.38
G4SFR*	1200	3(2) (M)	3(2)	900/10	9Cr-1Mo	12/3	18	316	5

* : Tentative name of sodium-cooled fast reactor under conceptual development in Korea
M : Mechanical Pump
E : Electric Pump
- : Unknown

3. NEW COMPONENT DESIGN CONCEPT

3.1 Integrated Steam Generator and Pump System (ISGPS)

The main design features of the newly proposed structural design concept for the sodium-cooled fast reactor are the ISGPS. This system adopts a canned motor pump (CMP) as a secondary pump of the IHTS. The casing of the pump is welded directly to the steam generator channel head; then, the crossover leg to the steam generator is eliminated. The key advantages of the CMP adoption are 1) a simplification of the IHTS piping system, 2) no concern of cavitation due to having hydraulics above a pump, 4) no-leakage of secondary sodium coolant and no seal maintenance, 5) elimination of flywheel system, 6) very small vibration level, and so on.

Fig. 1 shows a basic concept of a steam generator system with a mechanical pump in the cold leg and a horizontal plane layout of the IHTS pipes. This design concept is simple but a great deal of chamber space is required in a lower part of the steam generator where the cold leg pipe is directly connected to it. And the possibility of vibrations due to pump operations may cause structural damage to the IHTS piping system. A maintenance problem

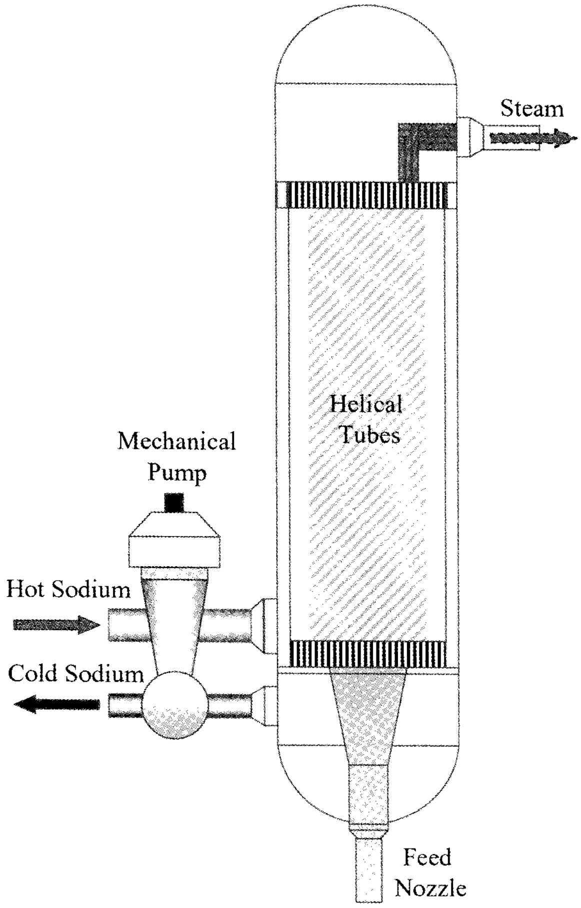


Fig. 1. General Concept of the Steam Generator with Mechanical Pump

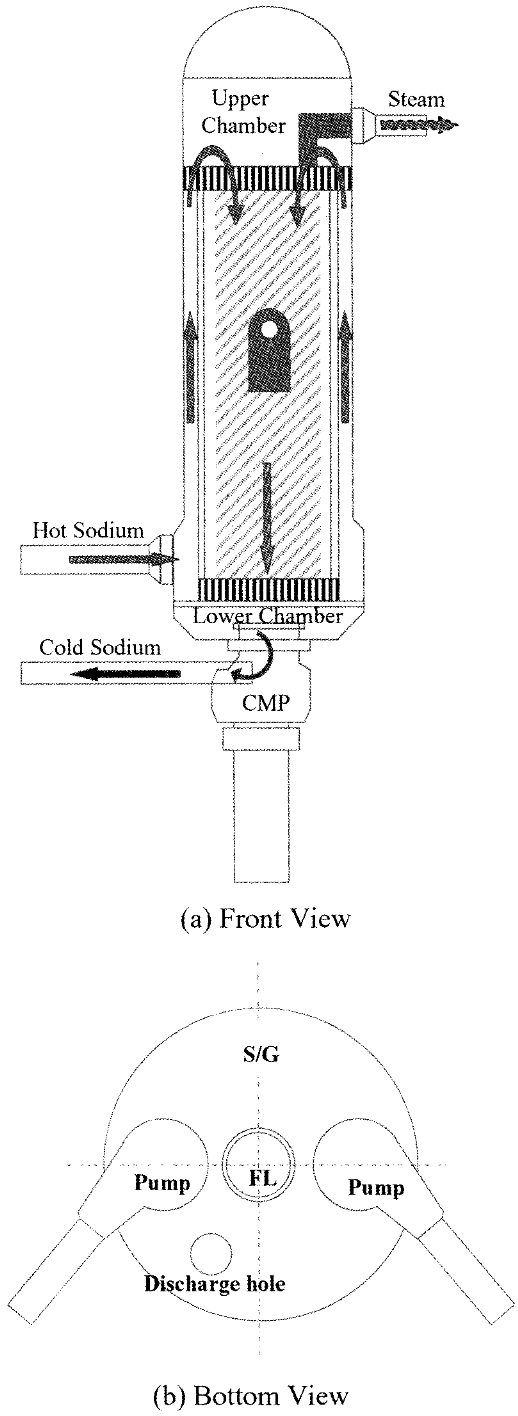


Fig. 2. Schematic of the ISGPS Concept

for the leakage seal may also exist for this design concept. Therefore, it seems that there is no innovative advantage except for the shortened IHTS cold leg pipe length.

To improve on the above basic design concept, the ISGPS is proposed, with an application of a canned motor pump, as shown in Fig. 2. As shown in the figure, the casing of the pump is directly welded to the steam generator channel head by eliminating the crossover leg. From the bottom view of the ISGPS in Fig. 2(b), two canned motor pumps are attached symmetrically to the SG bottom head and a feed line nozzle is connected vertically to the center of the bottom head. A sodium discharge hole is laid alongside the bottom head. The heat exchange tube of the steam generator is a helical type so that the height of the steam generator can be minimized. The design material for the outer shell is a modified 9Cr-1Mo steel, which is chosen to eliminate a dissimilar weld with the IHTS pipe made from the same material.

The flow path in the ISGPS is somewhat different from that of a conventional PWR steam generator. The feed water flows upward from a steam generator bottom head through the tubes while a secondary sodium coolant comes into the system from the hot leg nozzle welded to a lower part of a steam generator and flows upward through an annuls between co-cylinder of outer shell and inner shell to upper part of a steam generator chamber. From the upper chamber, the secondary sodium coolant is distributed uniformly and flows down to the lower chamber of the steam generator through the gaps between the tubes.

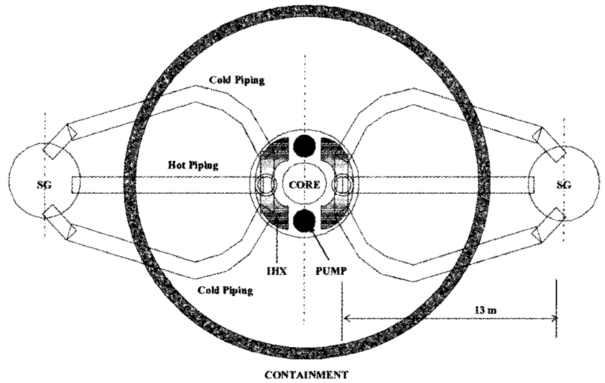
3.2 Layout of IHTS Piping System

The concept of the IHTS pipe layout, which can simplify the reactor coolant system and bring a distinguishable economic benefit, especially from thestructural and maintenance points of view, is basically concerned with the minimization of the total pipe length and with the simplification of the geometric layout.

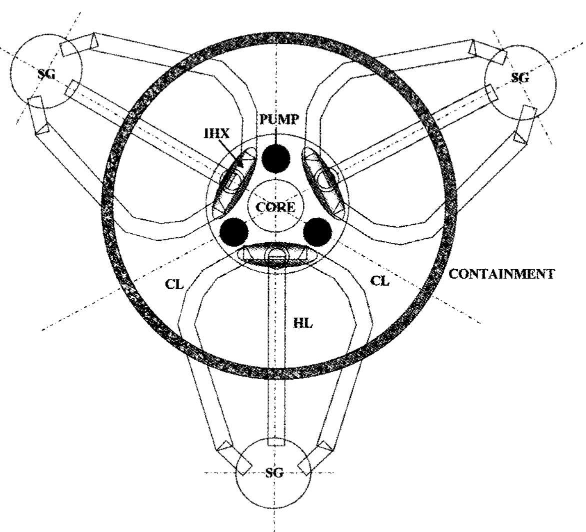
Fig. 3 presents a two loop and a three loop system, along with the proposed ISGPS. As shown in the figure, a plane layout concept is introduced for the IHTS piping system, which will allow it to substantiate a large size pipe design of about or over 1.0m in diameter for the hot leg side. A loop consists of two cold legs with curved sections and one hot leg with a straight run pipe. This IHTS pipe layout is very simple and has already been used for the light water reactor design of the AP1000 [10]. One of the important design benefits of this design concept is that it will be able to eliminate a branch pipe connection used in the KALIMER design [6] in the cold leg side, which is done to shorten the total pipe length but may cause flow-induced pipe vibrations and additional in-service inspection costs.

For a commercial sized sodium-cooled fast reactor with a capacity of 1200MWe, the diameter of an IHTS pipe is expected to be large, at about 1.0 m in the case of a two loop system design. According to our engineering experience, as the pipe size becomes larger, the two loop system might have advantageous features due to an easy space arrangement design around the reactor. To come up with a large size pipe, the concept of plane arrangement of the IHTS pipes is apparently beneficial, but the issue related with different thermal expansion displacements between the hot leg and the cold leg pipes is one of the design concerns that needs to be resolved with a connection of the main components, especially in high temperature operations.

For the IHTS pipe material, the modified 9Cr-1Mo-V steel, which is an ASME-NH design material and reveals relatively very small thermal expansion characteristics compared with other design materials, is selected to minimize the thermal expansion of the IHTS pipes and then to accomplish a feasible horizontal plane layout concept.



(a) Two Loop System



(b) Three Loop System

Fig. 3. Concept of the Reactor Loop System

3.3 Structural Integrity Evaluations of an IHTS Piping

To investigate the thermal expansion concerns for the proposed IHTS pipe design with a concept of a horizontal plane arrangement, structural analyses are carried out both for the steady state condition and the assumed severe accident condition for a 72 hours no-operator action.

For the analysis, a finite element program of the ANSYS commercial code [11] is used. The element type used in the analysis is a PIPE16 elastic pipe element which is a uniaxial element with tension-compression, torsion, and bending capabilities. Fig. 4 shows a finite element model with load boundary conditions and a coupled condition at a conjunction with the ISGPS. As shown in the analysis model, the junction nodes at the ends of the ISGPS side are modeled to be coupled at all degrees of freedom by an assumption of a rigid ISGPS lower part.

The support of the ISGPS is hinged forward in the direction of the thermal expansion of the IHTS pipes and can then accommodate an expansion displacement by means of a hinged movement from the original tilted position of the ISGPS, as shown in Fig. 5. To consider the initial load effect of the dead weight of the ISGPS due to the initial tilted position, a thermal expansion analysis is performed to find the maximum expansion displacement, which can result in a setting point for the initial tilting angle of the ISGPS. The calculated maximum thermal expansion displacement is 0.048m for a steady state operation of 545°C and 390°C for the hot leg and cold leg, respectively. With the assumption that the location of a hinged support from a horizontal plane of pipe layout is 5.0m, the initial tilting angle of the ISGPS shown in Fig. 5 can be calculated approximately as follows: $\theta = \tan^{-1}(0.048/5.0) = 0.55°$. Corresponding to the tilting angle, the initial horizontal load exerted on the IHTS pipes due to the tilting is calculated approximately with an assumption of a total ISGPS dead weight of 300 tons, as follows; $F_x = 300$ tons $\times \sin(0.55) \approx 29$kN.

By including the initial load effect of the ISGPS, the stress analyses are performed both for a steady state condition (545°C and 390°C for the hot leg and cold leg temperature, respectively) and a 72 hours accident condition (650°C and 500°C for the hot leg and cold leg temperature, respectively). Fig. 6 shows the results of the tresca stress intensity distributions for a steady state condition with a maximum value of 114MPa occurring at the cold leg junction with the Reactor. For the 72 hours accident condition, the maximum stress intensity is 115MPa at the same location. In the case of the 72 hours accident condition, we can assume a more severe temperature condition in which both the hot leg and the cold leg might be subjected

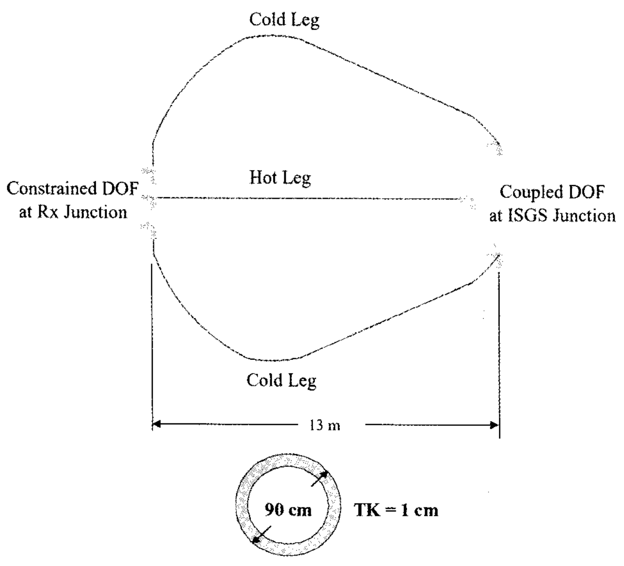


Fig. 4. Finite Element Model of the IHTS Piping

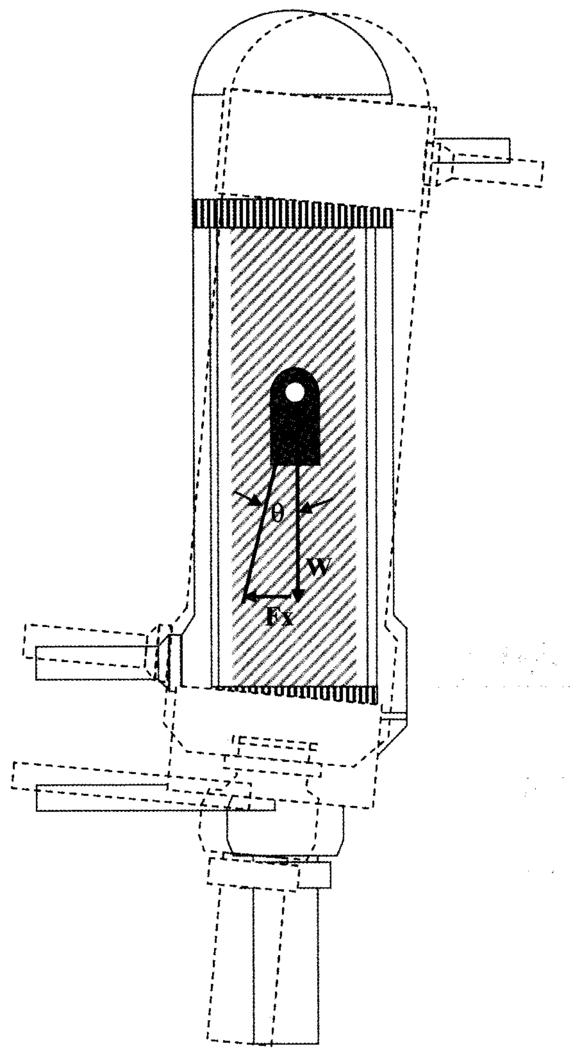


Fig. 5. Concept of the ISGS Initial Tilting Position

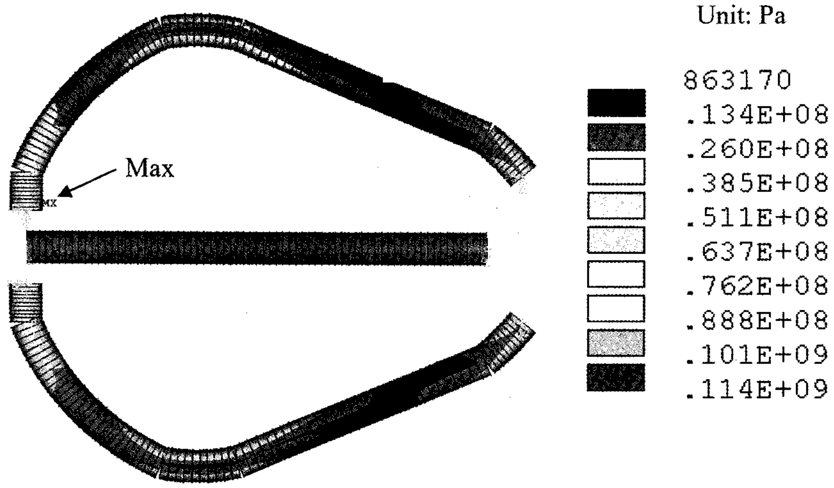


Fig. 6. Stress Intensity Contour for the Steady State Condition

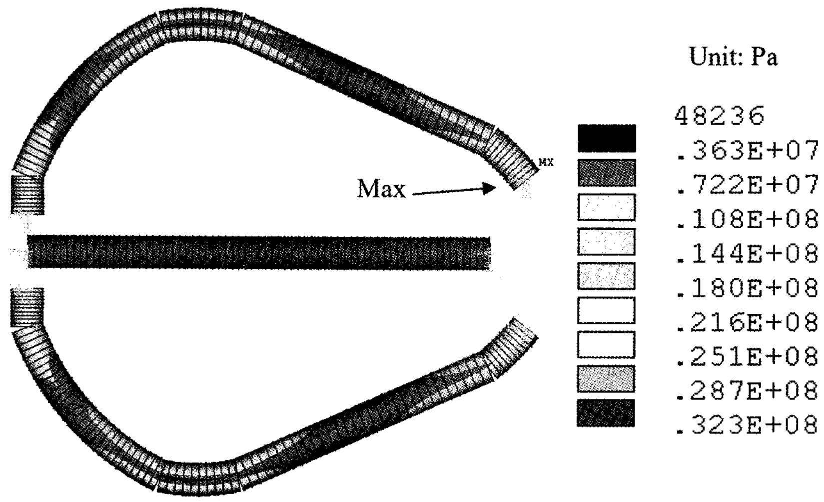


Fig. 7. Stress Intensity Contour for the Transient Condition: HL=CL=650°C

Table 2. Summary of the Stress Analyses

	Steady State Condition	72 Hours Accident Condition 1	72 Hours Accident Condition 2
Hot Leg Temperature, °C	545	650	650
Cold Leg Temperature, °C	390	500	650
Maximum Displacement, cm	4.8	6.1	6.3
Maximum Stress MPa	114	115	32

to the same high temperature of 650°C. In this case, the maximum stress intensity value is significantly reduced to 32.3MPa, which occurred at the cold leg junction with the ISGPS, as shown in Fig. 7.

Table 2 presents a summary of the results of the above stress analysis. From the table, we can see that the thermal expansion difference between the hot leg and the cold leg results in significant thermal stresses. Since the locations of the calculated maximum stress are subjected to an elevated temperature service, if the creep design features are acceptable for this stress level, the proposed IHTS piping layout will be feasible from the point of view of the structural design concept. Fig. 8 shows the marked allowable creep lifetime points corresponding to the calculated maximum stress values for the allowable creep

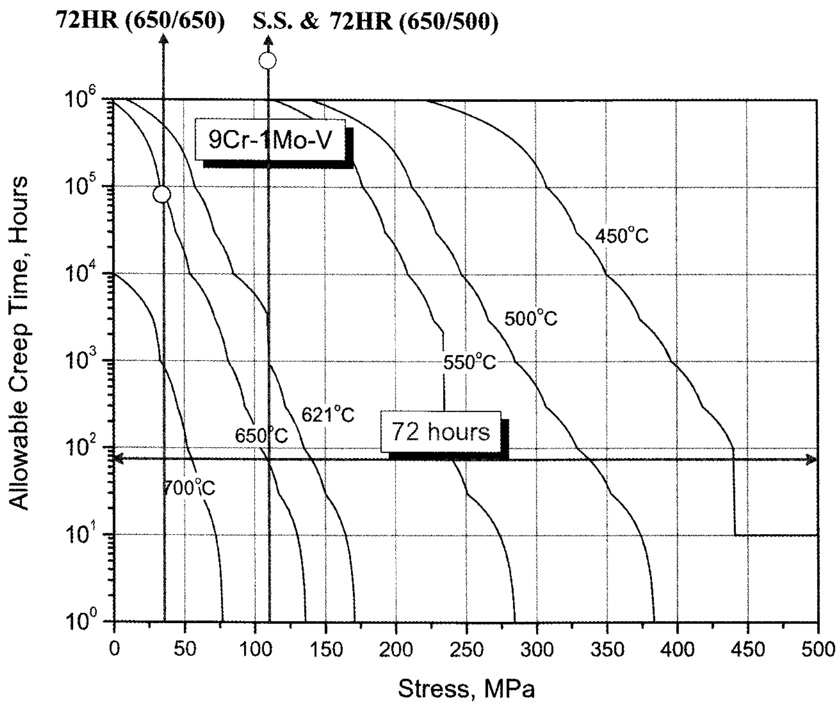


Fig. 8. Results of the Allowable Creep Life Time

time versus stress curves obtained from the expected minimum stress-to-rupture values provided by the ASME-NH rules for various temperature ranges. As shown in the figure, the allowable creep time is over 60 years for the steady state condition and the 72 hours accident condition 1. For the case of the 72 hours accident condition 2, the allowable creep time is about 1×10^5 hours, while the hold time is 72 hours. Therefore, through a preliminary evaluation of its high temperature structural integrity, the proposed concept of the IHTS piping system turned out to be acceptable.

4. ALTERNATIVE MODIFIED DESIGN CONCEPTS

4.1 Integrated Steam Generator System

As an alternative design concept for the ISGPS, the hot leg pipe can be extended to an upper part of the ISGPS with curved sections, as shown in Fig. 9. This concept can allow the system to avoid the complex design concept of an internal flow path through an annulus between a co-cylinder of the outer shell and the inner shell, and the initial tilting of the ISGPS to accommodate the thermal expansion displacement of the IHTS piping system. As shown in the figure, the main different features from the above design concept are 1) the hot leg pipe consisting of straight run sections and two curved sections, 2) the elimination of an internal annulus providing a path for hot sodium upstream, and 3) skirt support.

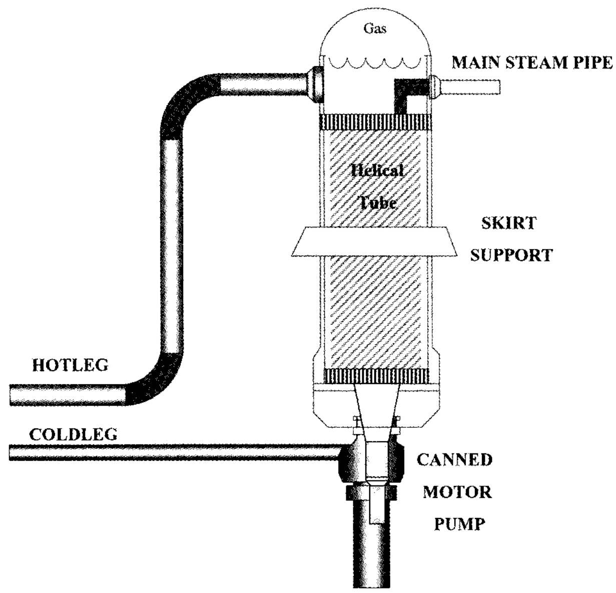


Fig. 9. Alternative Modified Concept of the ISGS with the Helical Tube

The concept of a curved hot leg system connecting to the upper part of the ISGPS has the benefit of eliminating the internal annulus structure inside the ISGPS and providing a flow path of hot sodium coolant. In addition, this concept has the location of a junction part of a curved

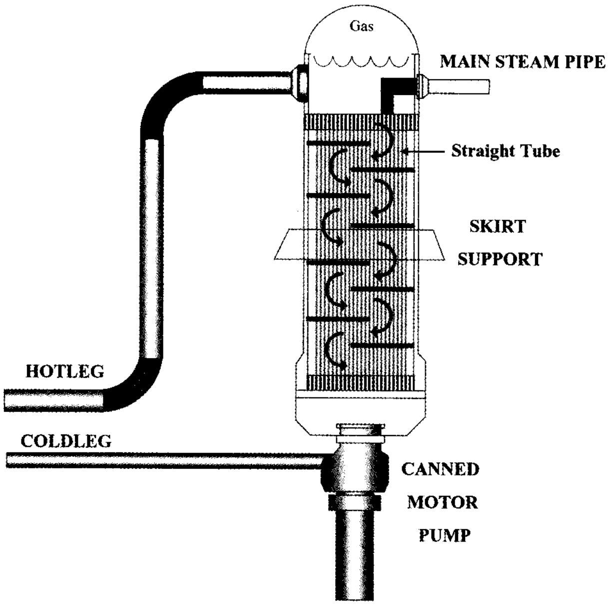


Fig. 10. Alternative Modified Concept of the ISGS with a Straight Tube

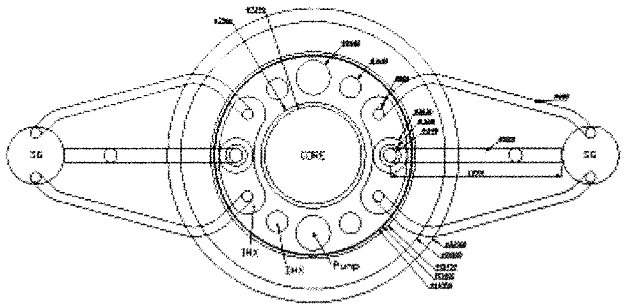


Fig. 11. Arrangement of the Reactor Coolant System with the ISGS Concept

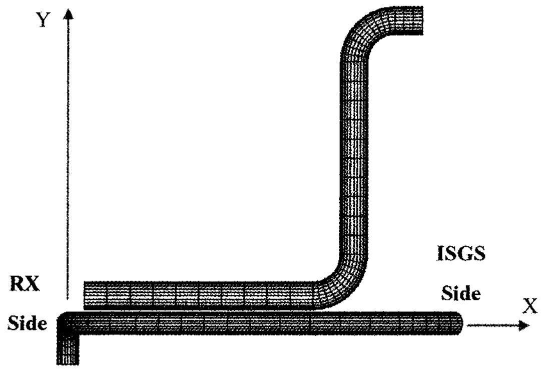
hot leg apart from that of the cold leg. Therefore, the problem of the invoking of high stress at the junction parts due to a thermal expansion difference between the hot leg and cold leg can be greatly improved because the curved hot leg pipe may absorb the thermal expansion difference.

The introduction of a helical steam generator tube can shorten the total height of the ISGPS; however, this modification involves several serious practical problems such as the fabrication and the in-service inspections. As another alternative, a modified concept is presented in this paper, as shown in Fig. 10. As shown in the figure, a straight tube is introduced with intermediate grid plates that guide a zigzag flow path, as used for the IHX. In this concept, the height of the straight tube type may be larger than that of a helical tube type, but it is expected to be minimized by an adequate design of the intermediate grid plates, satisfying the requirements for heat transfer area.

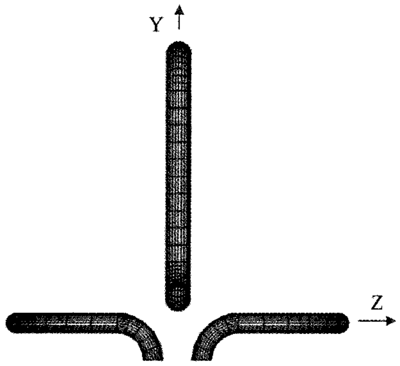
4.2 Overall Arrangement of the Reactor Coolant System

With the design concept of the ISGPS, the overall arrangement drawing of the reactor coolant system is presented in Fig. 11 for a two-loop system. As shown in the figure, the IHTS piping system consists of one hot leg and two cold legs for a loop and each cold pipe is separately connected to the IHX. Fig. 12 shows the 3-dimensional arrangement views of the IHTS piping system.

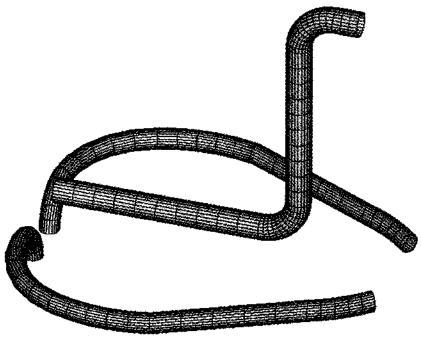
To establish a consistent equipment design connected to the IHTS piping layout, the kidney type IHX concept is proposed, as shown in overall arrangement drawing of



(a) Side View



(b) Front View



(c) Iso-View

Fig. 12. Arrangement of the IHTS Piping System

Fig. 11. More detailed schematic of the IHX concept is shown in Fig. 13. The secondary cold sodium flows into the downcomer pipes and gathers in a lower chamber. The gathered secondary sodium is distributed uniformly through a lower tubesheet and then flows through inside tubes making a heat exchange with the hot primary sodium flowing down the tube's outside. The hot secondary sodium gathered in the upper chamber exits the IHX through the hot leg nozzle. The proposed concept of a kidney type IHX has several advantages such as the elimination of a complex co-axial pipe, a reduction in the numbers of IHX, no branch pipe, an enhanced support stiffness against flow induced vibration and seismic load, and so on.

4.3 Structural Integrity of the hot Leg Pipe

One of the main concerns related with the proposed ISGPS design concept is whether the curved sections of the large size hot leg pipe can withstand the high temperature structural damage during a total design lifetime of 60 years. In this paper, the high temperature structural integrity for the hot leg pipe is investigated by the structural evaluation with the rules of the ASME-NH code, which is for class 1 components in elevated temperature service. In present ASME-NH rules, there are no specific design-by-formula evaluation rules for pipe components, which is available in the ASME Subsection NB-3650 (analysis of piping products). Therefore, a 3-diemsional solid finite element modeling is used for a structural analysis of the hot leg pipe. The element type used in the analysis is a SOLID70 for the thermal analysis and a SOLID 45 for the stress analysis provided by the ANSYS program [11].

The used pipe dimensions for the hot leg are a 1.0m outer diameter and a 1.0cm thickness. The length of the run pipes are 9m for the horizontal section and 7m for the vertical section; the radius of the curved section is 2.5m. The material is modified 9Cr-1Mo-V steel. As boundary conditions at the pipe ends of the junction parts with components, a degree of freedom corresponding to the axial thermal expansion direction is constrained as fixed and the radial thermal expansion of the pipe is allowed to be free.

4.3.1 Results and Discussions for the Steady State Conditions

Fig. 14 shows the results of the tresca stress intensity distributions for a steady state operation at 545°C. As shown in the figure, the maximum stress occurs at a lower

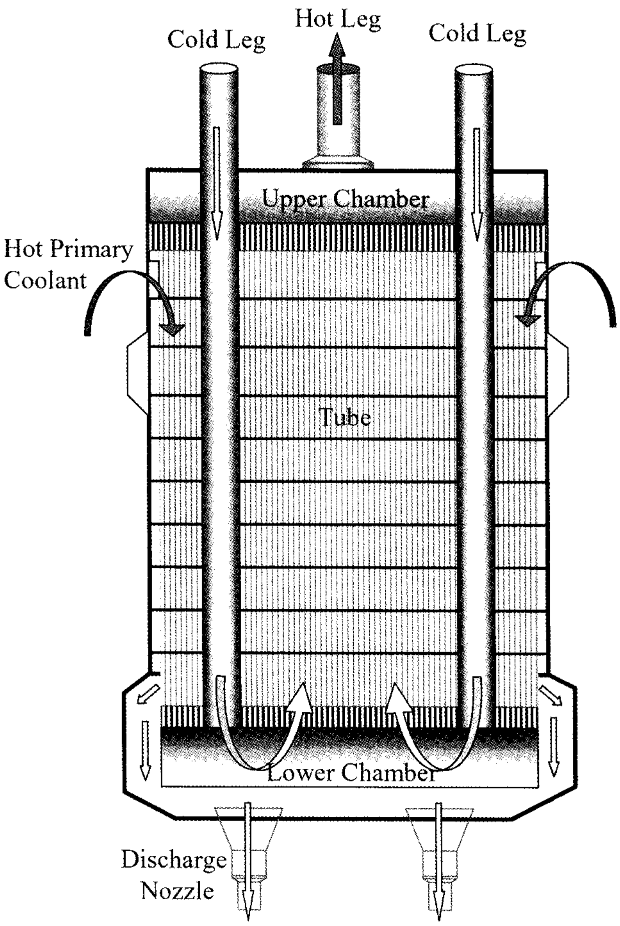


Fig. 13. Schematic of the Kidney Type IHX Concept

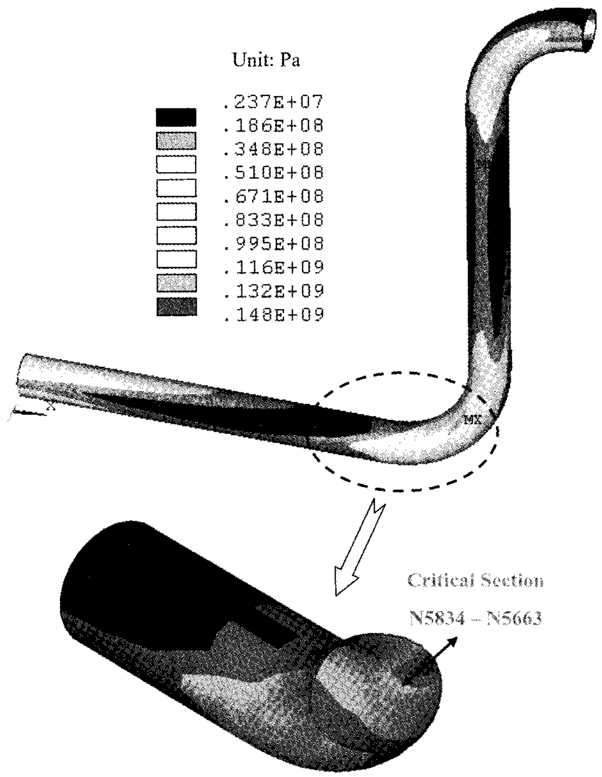


Fig. 14. Stress Intensity Contour for the S.S. Condition and Evaluation Section

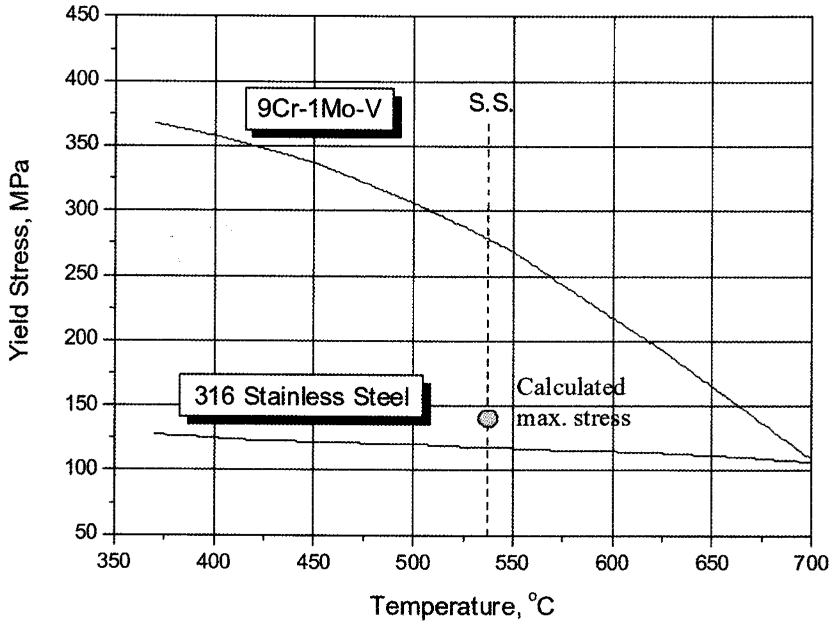


Fig. 15. Comparison of the Calculated Stress with the Yield Strength Value

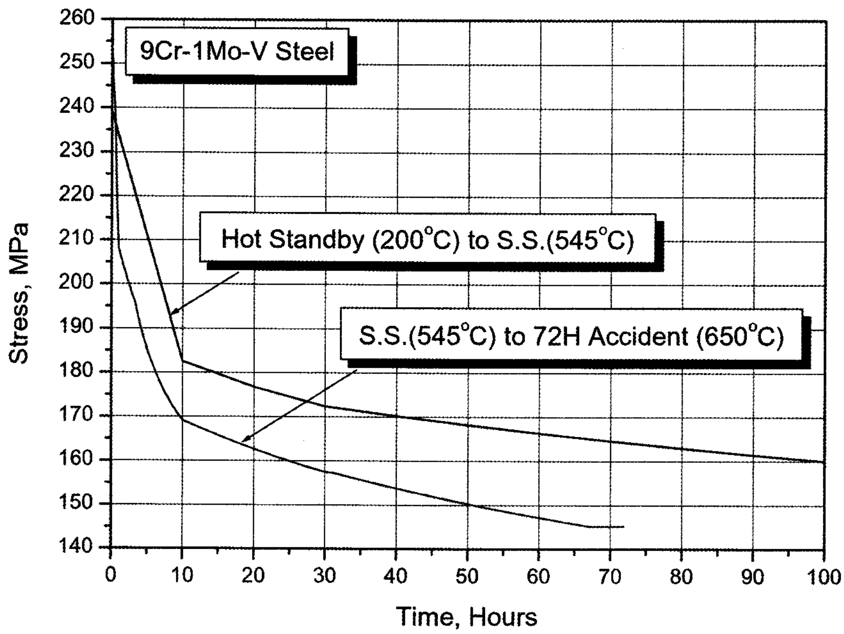


Fig. 16. Obtained Stress Relaxation Curves

curved section with 148 MPa, as expected. Due to the thermal expansion of the horizontal straight run pipe, a bending load occurs at the lower curved section and results in an oval deformation, as shown in the detailed sectional investigation. To check on the creep design integrity for the steady state operation, the allowable creep time corresponding to the calculated maximum stress value of 148 MPa and the temperature of 545°C is about 502,857 hours, obtained from the allowable creep time versus stress curves shown in Fig. 8. Therefore, we can see that it is acceptable when the total hold time design is less than 57 years.

From the view of the structural deformation, we can see that the curved section is entirely subject to elastic behavior during the steady state operation, as shown in Fig. 15. The calculated maximum stress of 148 MPa is a

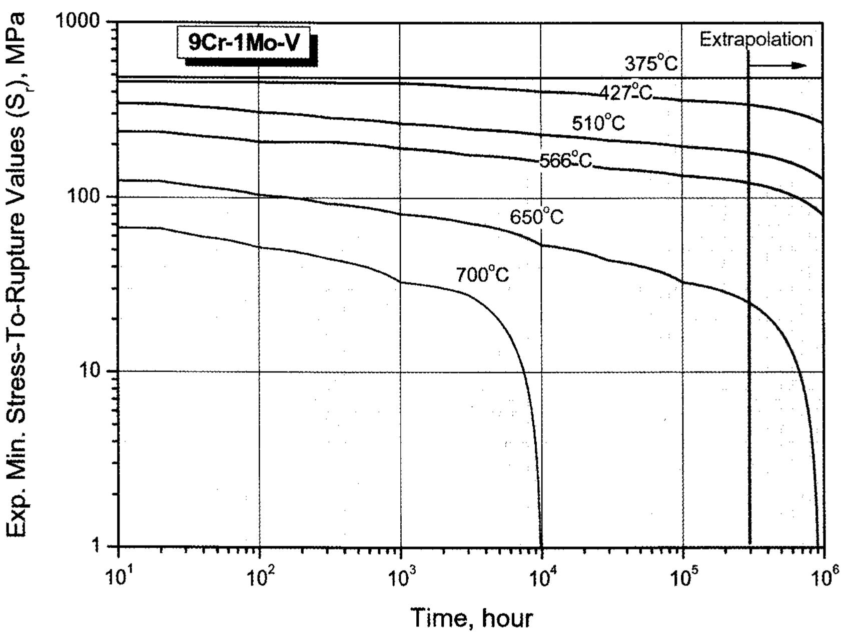


Fig. 17. Expected Minimum Stress-To-Rupture Values of ASME-NH

very low value compared with the yield stress, 273MPa, of the modified 9Cr-1Mo-V steel. Therefore, we can see that the use of the modified 9Cr-1Mo-V steel is a very big benefit for the piping design in this range.

4.3.2 Results and Discussions for the Transient Conditions

In this paper, two representative transient conditions are considered to evaluate the high temperature structural integrities, as follows:

- Transient type 1 : Hot Standby Condition (200°C) to Full Power Operation (545°C)
- Transient type 2 : Full Power Operation (545°C) to 72 Hours Accident Condition (650°C)

The used design hold time is 500,000 hours for creep damage evaluation. The critical section for evaluation is located at a lower curved section with a maximum stress as shown in Fig. 14. By using the SIE ASME-NH computer code, the stress relaxation curves, which are used for a calculation of the creep damage, are obtained for each transient as shown in Fig. 16. In the figure, we can see that the obtained initial stresses before the relaxation are very high, at over 240 MPa and 254 MPa for transient types 1 and type 2, respectively. And the initial stress of transient type 2, which has a much higher hold temperature than that of transient type 1, is much more rapidly relaxed than that of the transient type 1. The condition of high stresses during the initial period results in a very large creep damage. The calculated creep damage for transient type 1, which has a hold temperature of 545°C, is 0.896; this satisfies the design limit of a 1.0 value. For transient type 2, the initial dwell stress is 254 MPa; this exceeds the design values corresponding to the hold temperature of 650°C, as shown in Fig. 17 of the expected minimum stress-to-rupture curves. Therefore, it is found that the creep evaluation for transient type 2 is not applicable for the ASME-NH elastic analysis procedures. Even though it is known that the ASME-NH rules are very conservative, especially for the modified 9Cr-1Mo-V steel, an effort to reduce the maximum initial stress for a 72 hours accident is needed so that the proposed alternative IHTS piping concept can be accepted instead of changing the design rules.

5. CONCLUSIONS

In this paper, as a new component concept, an integrated steam generator and pump system (ISGPS) and its related IHTS piping and IHX are proposed to enhance the economical features through a simplification of a large size sodium-cooled fast reactor design. From the investigation of the design features and the high temperature structural evaluations of the related IHTS piping layout, it has been found by a preliminary analysis that the proposed ISGPS is feasible for construction and that the related IHTS piping design can satisfy the ASME-NH design rules for a 60 year elevated temperature service. For an alternative modified ISGPS design concept, even though this could make the steam generator design too simple, it is found from the structural evaluations that the related curved hot leg pipe design is a big issue that needs to be resolved to accommodate large thermal expansions, especially during severe accident conditions.

ACKNOWLEDGMENTS

This project has been carried out under the Nuclear R & D Program by MEST.

REFERENCES

[1] C.E. Boardman, M.Hui, and D. Carroll, "A Fast Track Approach to Commercializing the Sodium Cooled Fast Reactor," 7th Int. Conf. on Nuclear Engineering, ICONE7-7003, 1999.

[2] C.E. Boardman, A.E. Dubberley, and M. Hui, "Optimizing the Size of the Super-PRISM Reactor," 8th Int. Conf. on Nuclear engineering, 2000.

[3] M.Ichimiya, T. Mizuno, and S. Kotake, "A Next Generation Sodium-Cooled Fast Reactor Concept and Its R&D Program," Nuclear Engineering and Technology, Vol.39, No.2, 2007.

[4] G.H. Koo and B. Yoo, "Elevated Temperature Design of KALIMER Reactor Internals Accounting for Creep and Stress Rupture Effects," Journal of the Korean Nuclear Society, Vol. 32, No. 6, pp.566-594, 2000.

[5] G.H. Koo and B. Yoo, "Evaluation of Creep-Fatigue damage of KALIMER Reactor Internals Using the Elastic Analysis Method in RCC-MR," Journal of the Korean Nuclear Society, Vol. 33, No.6, pp.566-584, 2001.

[6] D.H. Hahn, Y.I. Kim, et.al, KALIMER-600 Conceptual Design Report, KAERI/TR-3381/2007, KAERI, 2007.

[7] ASME Code Section III, Subsection NH, 2007.

[8] G.H. Koo and J.H. Lee, Computer Program of SIE ASME-NH Code, KAERI/TR-3526/2008, KAERI, 2008

[9] Fast Reactor Database, IAEA-TECDOC-866, IAEA, 1996.

[10] W.E. Cummins, M.M. Corletti, and T.L.Schulz, "Westinghouse AP1000 Advanced Passive Plant," Proceedings of ICAPP03, 2003.

[11] ANSYS User's Manual for Revision 5.6, Volume I, II, III.