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《Technical Note》

Conceptual Safety Design Analyses of Korea Advanced Liquid Metal Reactor

S.D. SUK and C.K. PARK
Korea Atomic Energy Research Institute
150 Dukjin-dong, Yusong, Taejon, 305-353 - Korea
sdsuk@nanum.kaeri.re.kr

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Abstract

The national long-term R&D program, updated in 1997, requires Korea Atomic Energy Research Institute(KAERI) to complete by the year 2006 the basic design of Korea Advanced Liquid Metal Reactor(KALIMER), along with supporting R&D work, with the capability of resolving the issue of spent fuel storage as well as with significantly enhanced safety. KALIMER is a 150 MWe pool-type sodium cooled prototype reactor that uses metallic fuel. The conceptual design is currently under way to establish a self-consistent design meeting a set of major safety design requirements for accident prevention. Some of the current emphasis includes those for inherent and passive means of negative reactivity insertion and decay heat removal, high shutdown reliability, prevention of and protection from sodium chemical reaction, and high seismic margin, among others. All of these requirements affect the reactor design significantly and involve extensive supporting R&D programs. This paper summarizes some of the results of conceptual engineering and design analyses performed for the safety of KALIMER in the area of inherent safety, passive decay heat removal, sodium water reaction, and seismic isolation.

Key Words : inherent safety, seismic isolation, passive decay heat removal, sodium-water reaction, KALIMER

1. Introduction

From the viewpoint that liquid metal reactors (LMRs) have the potential of enhanced safety utilizing inherent safety characteristics and of resolving spent fuel storage problems through proliferation-resistant actinide recycling, LMRs appear to be the most promising nuclear power option of the future. In this context, the KALIMER development program was launched as a national long-term R&D program in 1992 and has been carried out by Korea Atomic Energy Research Institute(KAERI) since then. As such, the objective of the KALIMER Program was set to develop an inherently and ultimately safe, environmentally friendly, proliferation-resistant and economically viable fast reactor concept.

At the early phase of the conceptual design, an emphasis has been made to establish a self-consistent design satisfying a set of the major safety design requirements to avoid unusual occurrences, or to arrest them. One of the major requirements of current emphasis is that KALIMER shall be of inherent passive means of negative reactivity insertion and decay heat removal, sufficient to place the reactor system in a safe stable state for bounding ATWS(anticipated transient without scram) events without significant damage to the core or reactor system structure. Even with the inherent reactor shutdown requirement, the reactivity control and shutdown systems are required to result in extremely high shutdown reliability. As for all the other sodium cooled reactors, the structures, systems, and components of KALIMER are to be designed and located to minimize the probability and consequences of sodium chemical reactions. Seismic isolation is also required to achieve high seismic margins. All these safety design requirements affect the design significantly and demand supporting R&D programs of substance.

For the analysis of KALIMER' s inherent safety, a plant-wide transient analysis code, SSC-K, is being developed. Models for reactivity feedback effects and pool thermal-hydraulics have been developed into the code and a preliminary analysis of UTOP(unprotected transient overpower)and ULOF/LOHS (unprotected loss of flow and loss of heat sink)performance has been attempted. Design alternatives have been investigated to improve the decay heat removal capability by passive means, for which functional testings are to be done. Seismic base isolation is shown to reduce the seismic response of building and structures significantly and ,therefore, provide a great advantage in safety as well as economy for the structural design of nuclear power plants. Substantial progress has been made in developing and validating the methodologies, and engineering analyses for the structural design of KALIMER are currently under way. Engineering and design analyses are also being made to improve the intermediate heat transport system(IHTS) configuration against a sodium chemical reaction.

In the following sections, the major design features of KALIMER are briefly described, and some of the results from the safety design analyses and supporting R&D programs are summarized.

2. Major Design Features of KALIMER

KALIMER is a 150 Mwe pool-type, sodium-cooled prototype fast reactor. One of the major design feature of KALIMER is that the fuel is metallic, which brings potential benefits over the oxide fuel in improved inherent safety, reduced burdens of nuclear waste, and unique proliferation resistance. The choice of the metallic fuel was made, mainly based on the concept of the Integral Fast Reactor(IFR) developed by Argonne National Laboratory(ANL) in US[1].

Fundamental to the superior safety and fuel cycle characteristics of the IFR is the metallic U-Pu-Zr alloy fuel. The inherent safety potential of metallic fuel was demonstrated by the tests conducted with EBR-II in 1986[2]. The fuel test had been performed to achieve burnup levels of 20 at.% in the lead test assemblies. The IFR program had been complemented by the Advanced Liquid Metal Reactor(ALMR) program by an industrial team, led by the General Electric Company, who had developed a conceptual design of the Power Reactor Inherently Safe Module(PRISM)[3]. With the use of metallic fuels, therefore, the basic design concept of KALIMER is in line with that of the IFR and PRISM. However, there are a number of major differences between the design of KALIMER and that of

Table 1. KALIMER Key Design Parameters

OVERALL			
Net plant Power, Mwe	150	**PHTS**	
Core Power, MWt	392	Reactor Core I/O Temp,°C	386.2 / 530.0
Gross Plant Efficiency, %	41.5	Total PHTS Flow Rate, kg/s	2143.1
Net Plant Efficiency, %	38.2	Primary Pump Type	electromagnetic
Reactor	Pool Type	Number of Primary Pumps	4
Number of IHTS Loops	2		
Safety Shutdown Heat Removal	PSDRS		
Seismic Design	Seismic Isolation Bearing		
		IHTS	
CORE		IHX I/O temp., °C	339.7 / 511.0
Core Configuration	Radially Homogeneous	IHTS Total Flow Rate, kg/s	1803.6
Core Height, mm	1000	IHTS Pump Type	Electromagnetic
Maximum Core Diameter, mm	3447	Number of IHXs	4
Fuel Form	U-10% Zr Alloy	Number of SGs	2
Enrichments (IC/OC) for Equilibrium Core, %	14.4 / 20.0		
Assembly Pitch, mm	161.2	**Steam System**	
Fuel/Blanket Pins per Assembly	271 / 127	Steam Flow Rate, kg/s	175.5
Cladding Material	HT9	Steam Temperature., °C	483.2
Refueling Interval, months	12	Steam Pressure, MPa	15.50

PRISM in such areas as core configurations, core support and reactor internal structures, reactor system operating conditions, IHTS(Intermediate Heat Transport System) configuration, and the methods of seismic isolations, to name a few[4]. Table 1 summarizes some of the major design parameters of KALIMER, which is currently in the conceptual design phase. A salient feature of its key system designs is briefly described in the following[5,6].

2.1. Core and Fuel Assembly

The KALIMER core system is designed to generate 392MWt of power. The reference core utilizes a homogeneous core configuration in the radial direction with two driver fuel enrichment zones, surrounded by a layer of blanket assemblies. The core layout, shown in Figure 1, consists of 96 driver fuel assemblies, 42 radial blanket assemblies, 6 control rods, 1 ultimate shutdown system(USS) assembly self-actuated by a Curie point electromagnet, 6 gas expansion modules(GEMs), 48 reflector assemblies, 54 B_4C shield assemblies, 72 shield assemblies, and 54 in-vessel storage(IVS) of fuel assemblies, in an annular configuration. There are no upper or lower axial blankets surrounding the core. The reference core has an active core height of 100 cm and a radial equivalent diameter(including control rods) of 172 cm. The physically outermost core diameter of all assemblies is 344.7cm .

The core structural material is HT-9(ferretic martenstic steel). Its low irradiation swelling characteristics permits adequate nuclear performance in a physically small core. The fuel pin is made of sealed HT-9 tubing containing metal fuel slug in columns. The fuel is immersed in sodium for thermal bonding with the cladding. A fission gas plenum is located above the fuel slug

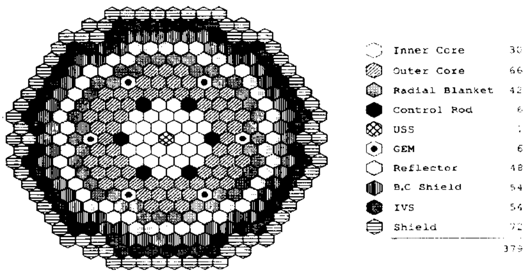


Fig. 1. KALIMER Core Layout

and sodium bond. The bottom of each fuel pin is a solid rod end plug for axial shielding. The driver fuel, blanket fuel, reflector, and shield assemblies use identical structural components(nosepiece, duct, and handling socket etc.), with only the pin bundle inside the duct and its mounting grid changing from one assembly type to the other. In all assemblies, the pins are in a triangular pitch array. The bottom end of each assembly is formed by the nosepiece which provides the lower restraint function and the coolant inlet. Surrounding the pin bundle and welded to the nosepiece is a hexagonal cross-section duct , which provides structural tie between the top and bottom end hardware of the assembly and isolates each pin bundle from its neighbors[5].

2.2. Reactivity Control and Reactor Shutdown

Reactivity and power are controlled by means of the control rod system in the driver fuel region of the core. The control rod design satisfies both the one rod stuck condition and the unit control rod worth condition against the unprotected transient over-power(UTOP) event. The gas expansion modules(GEMs), the concept of which was initially proposed for the design of PRISM(Power Reactor Innovative Small Module) by General Electric[3], are passive reactivity feedback assemblies that insert negative reactivity into the core during a loss of flow. The Self-Actuated Shutdown System(SASS) located at the center of the core is designed as an ultimate shutdown system by using a Curie point electromagnet which loses its magnetic force holding the shutoff rod when the temperature of the primary sodium reaches the curie point, hence a passive shutdown can be achieved. This temperature-dependent SASS had been widely explored in the past for implementing into the LMR design by a variety of organizations in the countries like US, France and Japan, the details of which are described in Ref.7.

2.3. Residual Heat Removal System

In KALIMER, the shutdown heat removal system is designed with the emphasis on system reliability to achieve a higher level of plant safety. Safety grade heat removal is achieved by the Passive Safety Decay Heat Removal System (PSDRS), which consists of the air path around the containment vessel, as shown in Figure 8. The decay heat from the core is transferred to the reactor vessel by the natural circulation of the sodium in the reactor pool and the heat is in turn transferred by conduction and radiation to the containment vessel. The hot containment vessel wall heats the surrounding air in the air channel and the density difference between the air inside and that outside the channel induce the pressure difference. This difference forms air flow and the heat is transferred from the containment vessel to the air, which dissipates the transferred heat finally to the plant environment.

The concept of PSDRS is found in the Reactor Auxiliary Cooling System(RACS) of the Rockwell International' s SAFR design[8]and the Reactor Vessel Auxiliary Cooling System(RVACS) of PRISM[3], but their performance characteristics are not well publicized in the open literature.

Normally the decay heat is removed by steam

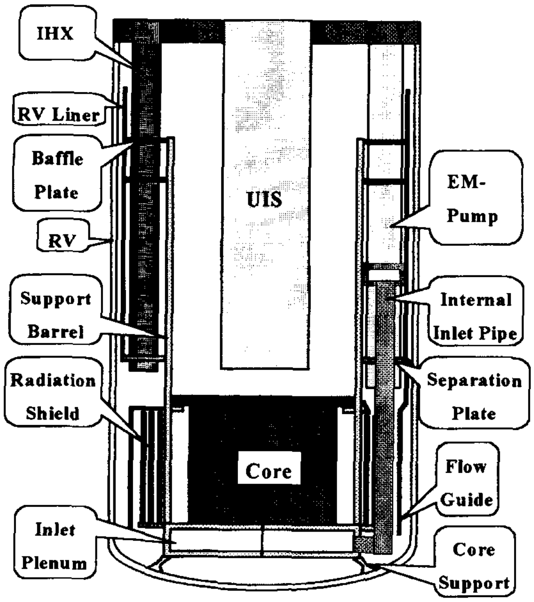


Fig. 2. Conceptual Diagram of Reactor Internal Structure

generators and the condenser. During the maintenance of any IHTS, heat is removed by the remaining IHTS loop. Also there is the Steam Generator Auxiliary Cooling System(SGACS) to aid the decay heat removal. The SGACS, as in PRISM design, induces natural or forced circulation of atmospheric air past the shell side of steam generator. Intensive analysis on the system performance and design parameters is under progress for system level design optimization.

2.4. Reactor Structure

The reactor vessel has the overall dimensions of 17.6m height, 7.02m diameter, and 5cm thickness in preliminary concept design and is composed of a cylindrical shell with an integral hemispherical shell bottom head. The structural integrity and safety of the reactor vessel have been achieved by eliminating the penetration nozzle and attachments other than the core support structure.

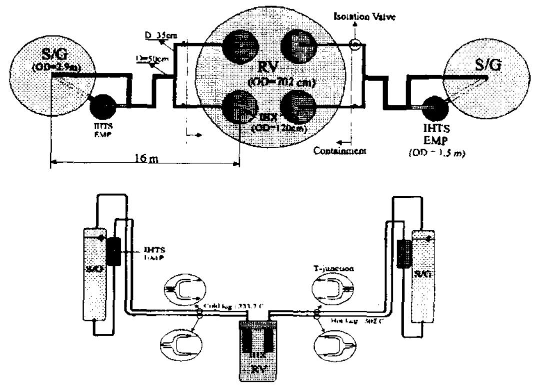


Fig. 3. General Arrangement of NSSS

The shape of the core support structure is a skirt-type. All equipment, like the intermediate heat exchanger(IHX), elctromagnetic(EM) Pump, in-vessel transfer machine(IVTM), and upper internal structure(UIS) are supported by a reactor head, and a rotating plug is adopted for the refueling operation. The support barrel, which is a major component of reactor internal structures, serves as a redan to separate the hot sodium pool and cold sodium pool and as a support of internal structures, including the reactor core. Figure 2 illustrates a conceptual diagram of the reactor internal structures of KALIMER. The containment vessel, which encloses the reactor vessel, is easy to access from the reactor vault so that the inspection and maintenance of the vessel can be easily accomplished. General arrangements for the NSSS and the reactor building are tentatively developed as shown in Figures 3&4.

The seismic base isolation for the reactor building using high damping rubber bearings has been adopted to achieve the sufficient structural integrity and economic design of KALIMER when subjected to a design basis earthquake such as a horizontal Safe Shutdown Earthquake of 0.3g. The high damping laminated rubber bearings(HLRBs) are installed between the upper

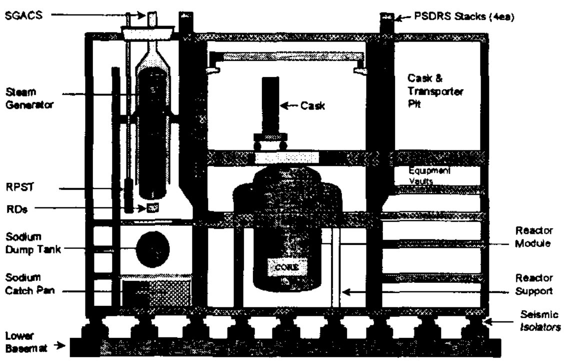


Fig. 4. General Arrangement of Reactor Building

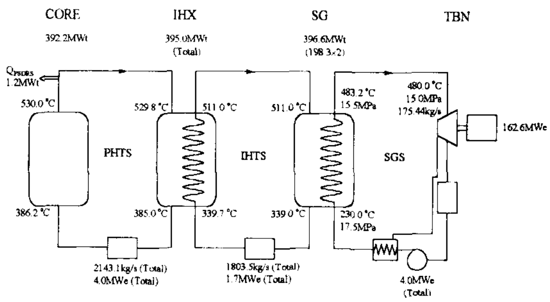


Fig. 5. KALIMER Plant Heat Balance

basemat of the reactor building and the lower basemat on the ground. The concrete upper basemat on which whole reactor building is seated is about 40m wide, 60m long, and 2m deep. The gap between upper and lower basemat is about 2m, where 176 isolators or so will be installed and maintained. Horizontal seismic base isolation system is excellent to reduce the horizontal seismic responses but tends to amplify the vertical responses. The development of a design concept adopting 3 dimensional seismic base isolation is under consideration to reduce both horizontal and vertical seismic responses.

2.5. Heat Transport System

A superheat steam cycle is implemented to have a high plant efficiency, noting that high thermal efficiency reduces the heat discharge from the plant, resulting in less impact to the environment. The IHTS consists of two loops, and each loop is equipped with one steam generator unit. For safety, large system thermal inertia is achieved by using a pool-based primary system. Strong emphasis has been given to the prevention and mitigation of possible sodium-water reaction events for the IHTS piping routing. Valves for the isolation of IHX from the sodium-water reaction products are installed at each IHTS piping penetrating the containment. The system reliability is improved by using EM pumps, which do not have moving parts, for both the primary and intermediate coolant pumping. The low momentum inertia of the EM pump is compensated for by using an auxiliary device which keeps a certain amount of rotating kinetic energy when the EM pump runs normally but supplies electricity from the rotating kinetic energy to the EM pumps when the electricity supply to the pumps is interrupted.

The operating temperature and primary component size were determined to make the net plant thermal efficiency higher than 38%. For the purpose of establishing a self-consistent system and component configuration, the preliminary plant heat balance diagram has been set up simply applying the first law of thermodynamics , as shown in Figure 5[9].

3. Safety Design Analyses

3.1. Inherent Safety Analyses

A plant-wide transient analysis code is being developed for the analysis of KALIMER' s inherent safety and for assistance in the development of design, where new design features will frequently demand not just new data but new models. A transient and safety analysis code, SSC-K, is under

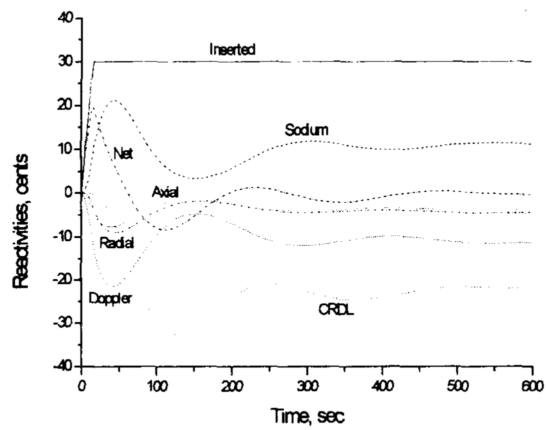


Fig. 6. Reactivity During UTOP Event

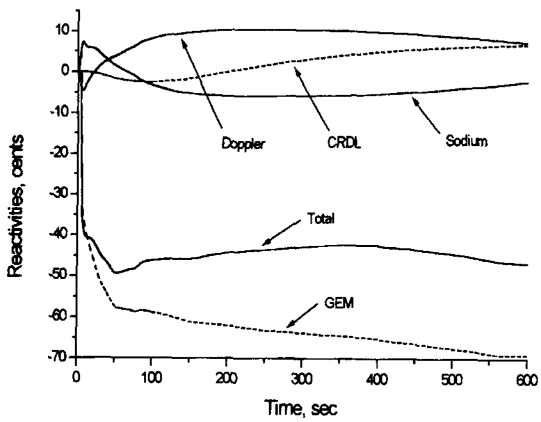


Fig. 7. Reactivity During ULOF/LOHS Event

development based upon the SSC-L code[10], which was developed by BNL for the analysis of loop type LMRs with oxide-fueled core. Models modified and newly developed and implemented into the code so far include models for reactivity feedback effects, pool thermal-hydraulics, and passive decay heat removal, among others. In order to verify the logic of the models developed, and to assess the effectiveness of the inherent safety features based upon the negative reactivity feedback in achieving the safety design objectives of passive safety, a preliminary analysis of UTOP and ULOF/LOHS performance has been attempted.

Inadvertent withdrawal of the control rod at a reactivity insertion rate of 2 cents/second was assumed for the simulation of UTOP. The reactor power increases initially due to the positive reactivity arising from the removal of control rods, but turns downward once the negative reactivity feedback effect increases enough to counter the positive insertion, causing the total reactivity to become slightly negative and stabilizing the power at a level higher than that of the initial steady state. As shown in Figure 6, the Doppler effect is an instantaneous and important feedback for UTOP, and the net reactivity increases initially and then decreases to negative values due to feedback effects.

Trip of all primary pumps with coast-down and the loss of IHX heat removal capability due to sodium water reaction in the steam generator is assumed for the ULOF combined with a LOHS event. Reduction of the core flow is due to the coast-down of primary electromagnetic pumps, and the reactor power decreases to about 6% of the rated power due to negative reactivities. When there were no GEMs in the core, a sodium boiling occurred since the reactor power decreases rather slowly and the power-to-flow ratio increases. As shown in the Figure 7, the net reactivity is always negative during the course of the transient due mainly to the largest contribution from GEMs[11].

According to the preliminary evaluation of the inherent safety characteristics, there is a large safety margin even under severe unprotected event conditions. In order to validate the SSC-K code for safety analysis, code-to-code comparison calculations and/or calculation against experimental data need to be performed. Even though EBR-II experiments have shown the possibility of inherent safety of small metallic cores, there needs to be an investigation in extending the result to larger cores. Coast-down characteristics of electromagnetic pumps have a significant effect on the core safety under loss of flow events, and the performance of a synchronous machine for inertia needs to be evaluated. The effect of the fluctuating sodium level inside the GEM on reactivity, and the effect of GEM reactivity insertion due to the restart of pumps at low power operation needs also to be investigated.

The probability of HCDA occurrence is extremely low due to the inherent safety characteristics of KALIMER, and mechanistic analysis is not planned during the conceptual design stage. However, depending upon the decision of the licensing authority, there may be a developmental effort for the mechanistic approach in the future. A simple model, based upon the Modified Bethe-Tait model, is being developed for the estimation of energy release and available work under HCDA for the analysis of ultimate safety.

3.2. Passive Decay Heat Removal Analysis

In the operation of the PSDRS, the core decay heat is transferred to the containment vessel, as shown in Figure 8, and the heat from the containment vessel is dissipated to the air flow which is generated by the natural circulation from the density difference between the air channel and the environment. The heat dissipation to the air flow is made of two paths. One is the direct convection heat transfer from the containment wall surface and the other is an indirect path to the air. In the indirect path, heat is first transported from the containment vessel surface by radiation to the air separator which separates the hot air from the incoming cold air. Then the heat is dissipated to the air flow by convection.

The main resistance in the heat transfer from the core to the air in a system like the PSDRS is at the path from the containment vessel to the air. The improvement of the heat removal capacity of the system comes to heavily depend on the improvement of the heat transfer in the air channel. Two types of works have been made to improve the heat removal capacity. One is the modification of the wall surface to enhance the convection heat transfer coefficient and the other is modification of the air channel configuration itself. In this study, a new channel configuration is introduced and the feasibility of the heat transfer enhancement of the new channel configuration is examined for black body surfaces.

As a first step to establish a new channel configuration for enhancing the channel heat transfer, the effects of the conventional cross-rod radiation structure were investigated changing the number of structures and their gap sizes. It was observed from these preliminary analyses that reducing the pressure loss through the radiation structure is most important to enhance the heat transfer and the structure needs to completely block radiation rays reaching the structure region wherever they may come from[12].

The new channel configuration, as shown in Figure 9 , is provided with lateral compact heat transfer surface structures of identical vanes. The new channel configuration is different from the conventional cluster-rod-matrix configurations in that it uses a compact heat transfer surface of vanes and the surface is directed toward annular cross-section of the channel. The vertical distance of the vane structure is 0.04m and the size of gap between each vane is 0.006m. With this configuration, radiation rays are completely blocked, resolving a concern of possible leakage problem with the previous radiation structure of cross-rod matrix. Also the geometry of the upper and lower parts of the vane maintains symmetry to radiation heat transfer so that all of its heat transfer surface can be effectively utilized.

Analysis of the air channel cooling with the new

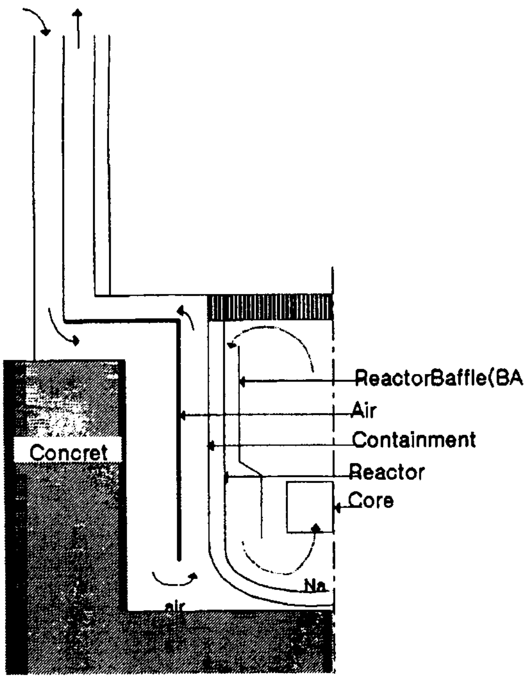


Fig. 8. Analysis Domain & Heat Transfer Path

radiation structures revealed substantial heat transfer enhancement, and the feasibility of the heat transfer enhancement with the new channel configuration design has been confirmed. For the geometry of the air channel of black surface and fixed wall temperature condition, for instance, the overall heat transfer is predicted to increase up to 6 times more than the heat transfer rate of the same gap size , and up to about two times more than the rate at the optimum gap size without radiation structures[12].

3.3. Seismic Isolation Study

3.3.1. Laminated Rubber Bearing(LRB) and Shake Table Test

For the rubber specimen and LRB tests, various effects such as the shear strain, the loading rate, the cyclic loading, and so on are investigated. In

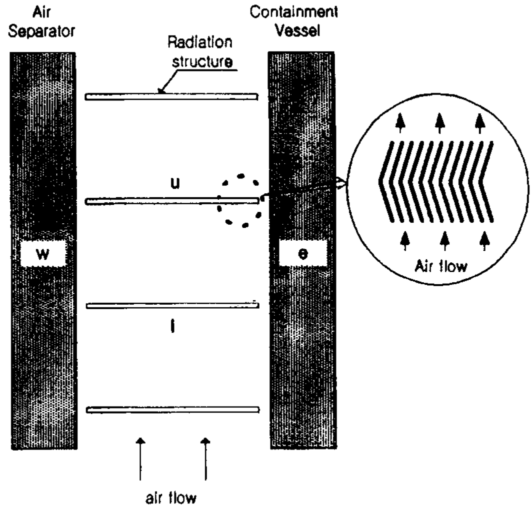
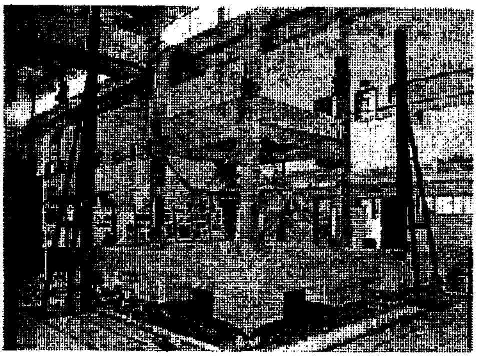


Fig. 9. Radiation Structure

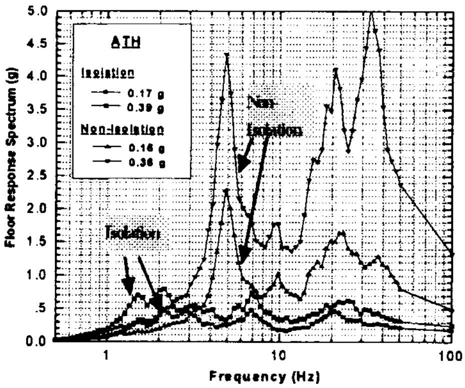
these tests, the LRB being developed in KAERI shows good mechanical characteristics applicable to KALIMER. In the shaking table tests for the seismically isolated structure, it is confirmed that the seismically isolated structure produces significant reductions in the seismic responses compared with the case of the non-isolated structure. Structural dynamic tests for base isolated structures equipped with 2 dimensional 4-1/8 scale high damping rubber bearings were performed using 30 ton-6 dof shaker. Figure 10 shows the test model structure and seismic response results at the upper slab of the test model to the artificial time history(ATH) input of SSE 0.3g[13,14].

3.3.2. Reactor Building and Reactor' Internal Structure Analyses

To obtain the time history of the seismic responses of a reactor building, a lumped-mass beam model was developed. The model is composed of two sticks; one is 9 beam elements for the reactor building and the other is 3 beam



Test Model



Test Results of Isolated and Non-isolated Test Model to Artificial Time History 0.3g & 0.15g

Fig. 10. KALIMER Seismic Isolation Test Program

elements for the reactor support structure. The time history response analyses for the non-isolated and isolated reactor buildings have been carried out, using the ABAQUS code[15], for an ATH earthquake generated by using the seismic design spectrum curve of US NRCRG1.60. Design basis earthquakes for KALIMER are SSE 0.3g for the horizontal and 0.2g for the vertical direction, and OBE 0.15g for horizontal and 0.1g for the vertical direction, respectively.

The isolation frequency of the reactor building is 0.5 Hz, and the equivalent damping of LRB is 12%. The horizontal stiffness of the total isolator is 5.77×10^{8}N/m and the damping is 4.356×10^{7}N.sec/m. The total weight of the reactor building is about 68,000 tons.. The maximum acceleration responses of the non-isolated and isolated reactor buildings for the horizontal and vertical earthquake data are shown in Table 2[16].

The time history responses for x-direction displacement are presented in Figure 11, and the response spectra at major locations are represented in Figure 12.

The maximum peak acceleration is reduced to 0.177g for the isolated condition, while it is 1.46g for the non-isolated condition. The maximum displacement becomes larger, up to 15.0cm, for the isolated condition. The maximum acceleration for the vertical earthquake of 0.208g ZPA(zero period acceleration) is amplified to 0.848g for the isolated condition, while the maximum acceleration is amplified to 0.577g for the non-isolated condition. This agrees with the general trend that the horizontal isolation of the structure can amplify the vertical responses.

Seismic analysis work for the reactor internal structures was also carried out through the modal analysis, the seismic time history analysis, and the equivalent seismic stress analysis. The detailed local stiffness analyses were performed to construct the lumped-mass seismic analysis model from the 3-dimensional finite element model of the reactor internal structures of KALIMER. Results show that the seismic responses of the reactor structures of the seismically isolated KALIMER are significantly reduced in their accelerations and relative displacements in the horizontal direction. However, for the vertical direction, significant response amplifications occur

Table 2. Accelerations and Displacements of Reactor Building Under ATH Earthquake

Location	X-Direction(g)		Y-Direction(g)		Z-Vertical (g)	
	Non-isolated	Isolated	Non-isolated	Isolated	Non-isolated	2D isolator
Base	0.30	0.175	0.30	0.177	0.205	0.321
Top	1.461	0.177	1.609	0.179	0.577	0.848
RV support	0.583	0.173	0.676	0.175	0.362	0.558
Max. Relative Displacement(cm)	0.638	14.94	0.832	14.94	0.069	0.102

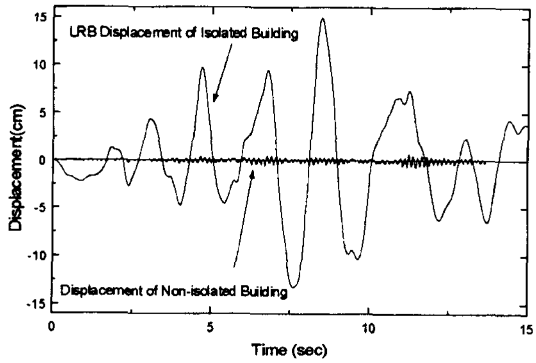


Fig. 11. Displacement Responses of Reactor Building

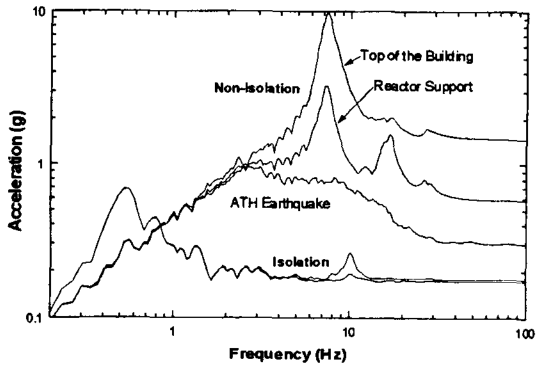


Fig. 12. Comparison of Acceleration Response Spectra of Reactor Building(ATH,X-dir. 0.3g)

in whole structures. This is due to the vertical structural frequency of 8.1Hz located in the dominant excitation frequency band of input motion[16]. The development of a design concept adopting 3 dimensional seismic base isolation is under consideration to reduce both horizontal and vertical seismic responses[17]

Seismic margin was also evaluated for each of KALIMER reactor internal structures including the reactor vessel and containment vessel. It is shown that containment vessel, reactor vessel, inlet plenum, and core support have large seismic stress margins but reactor vessel liner, support barrel, separation plate, and baffle plate have small margins. The maximum stress occurs in reactor vessel liner parts connected with the separation plate due to the vertical seismic loads.

3.3.3. Core Seismic Response

Application of seismic isolation to the LMR design brings about a significant reduction in core seismic responses. For the general investigation of core seismic responses, the SAC-CORE code has been developed, which consists of a number of modules such as SAC-MODAL for modal analysis , SAC-CORE for analysis of nonlinear time history, SAC-FRS for calculation of floor response spectrum ,and other processing modules [18]. In the code, the basic seismic excitation mechanism is modeled using the Runge-Kutta numerical algorithm and the solution procedures are

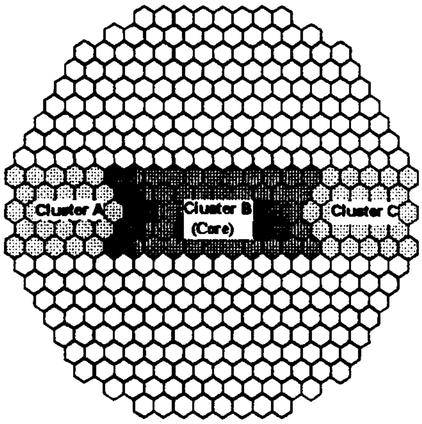


Fig. 13. Clustering of LMR Core Assemblies

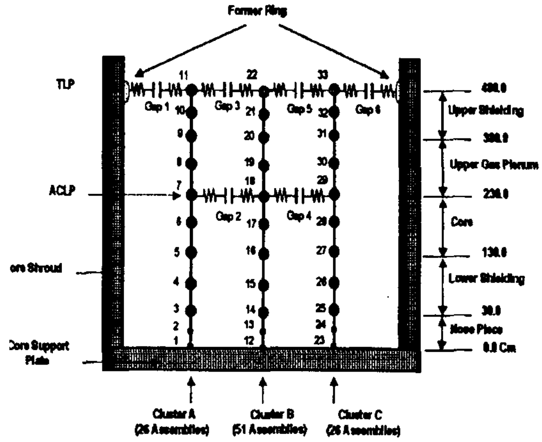


Fig. 14. Simplified Core Seismic Analysis Model

developed considering multi-gap impacts at the fuel assembly load pads. For the verification of the code, the seismic test results with the RAPSODIE core mock-up facility were analyzed and compared[19].

The seismic analysis of LMR core structures is a complex problem involving the dynamic interaction of many hundreds of individual fuel, blanket, and shield assemblies in a sodium environment. To simplify the core seismic problem, the cluster modeling technique shown in Figure 13 for a diametral row of the core is used. The clusters of assemblies are assumed to have no relative motion between the assemblies within a cluster. The diametral row modeling approach gives conservative results, and it is easier to evaluate the core seismic behavior compared to a full core model using the cluster technique.

In the present analyses, the 3-clusters row model, in which cluster B represents fuel assemblies and clusters A and C represent the shield, blanket, reflector, and etc. as shown in Figure 13, is used to simplify the core seismic problem. Clusters A and C have 26 assemblies each and cluster B has 51 assemblies. The core seismic model used in the analyses contains 30-beam elements and 6-impact gap elements, as shown in Figure 14. Preliminary designed impact locations are modeled between the assemblies at the top load pad(TLP), the above-core load pad(ACLP), and between the outer shield assemblies and the former ring at the top. The material and sectional properties of ducts, gap properties and other data used in the analyses are described in Ref. 20.

The results of modal analysis show that the fundamental frequency of the LMR core is 4.3 Hz and the second natural frequency is 24.3Hz. These natural frequencies of the core will show non-linear behavior during impact at load pads. For the general investigation of core seismic responses, the harmonic excitations subjected to the rigid core shroud and core support plate are used in the analyses considering conservative excitation conditions. Table 3 shows the input loading conditions. The excitation frequency of 4.3 Hz is the fundamental frequency of the core , obtained from the modal analysis of the core seismic model shown in Figure 14. Other values of acceleration and excitation frequency were obtained from detailed seismic analysis model for a reactor system[21].

The results of the core seismic response analyses show that , load case 4, which is the case

Table 3. Load Cases Used in Core Seismic Analyses

Load Case	Core Support Excitation for SSE Conditions (0.3g)		
	Acc.	Freq.	Remarks
1	1.28g	8.1 Hz	Non-Iso, RI Freq.
2	1.28g	4.3 Hz	Non-Iso,Core Freq.
3	0.22g	4.3 Hz	Iso., Core Freq.
4	0.22g	0.7 Hz	Isolation Freq.

of a seismically isolated LMR, gives significantly reduced seismic responses compared with those of load cases 1 and 2, which are for the cases of the non-isolated LMR. The seismic responses for load case 3 , which may give a limit design case of the seismic isolation frequency for the core, show little reduction in seismic responses(Figure 15). When the seismic isolation frequency(0.7Hz) is much lower than the core fundamental frequency (4.3Hz), a good isolation performance is observed in terms of core seismic responses. Figure 16 illustrates the impact load at gap 3(TLP).

From these results, we can conclude that seismic isolation provides great reductions of the impact loads as well as the number of contacts at the load pads of core assemblies at the former ring. This can allow the simple design of a core control system. And it is expected that the requirements of the UIS(upper internal structure)/core relative deflection limits, and the core compaction and reactivity insertions at the SSE load condition can be easily satisfied when an efficient seismic isolation is adapted for the LMR design.

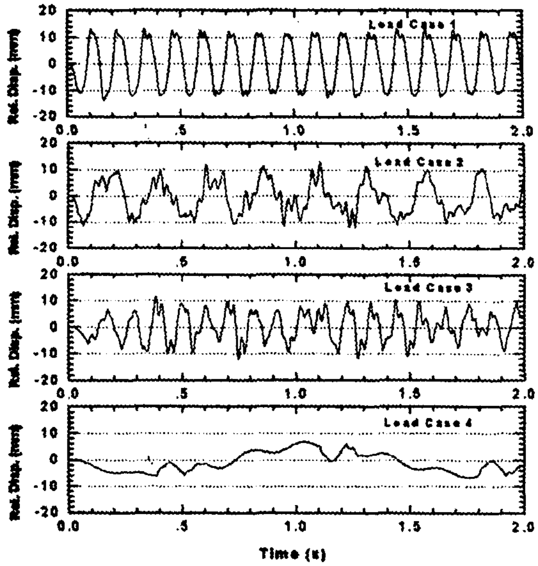


Fig. 15 .Relative Displacement at Node 22(TLP)

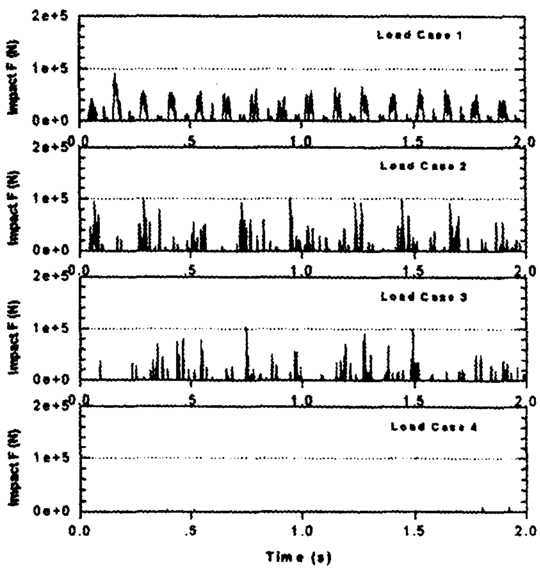


Fig. 16. Impact Loads at Gap3 (TLP)

3.4. Sodium Water Reaction Analysis

Large scale water leakage into the sodium side due to the failure of tubes in LMR steam generators leads to an increase in the pressure and temperature by hydrogen and the heat of reaction, and may significantly affect the structural integrity of the intermediate heat transport system(IHTS). Prior to designing the IHTS and the steam generator, a pre-estimate of the pressure effects for this system should be conducted. As a general trend of pressure change, when water leakage occurs, a relatively high pressure is formed within milliseconds and is called the initial spike pressure.

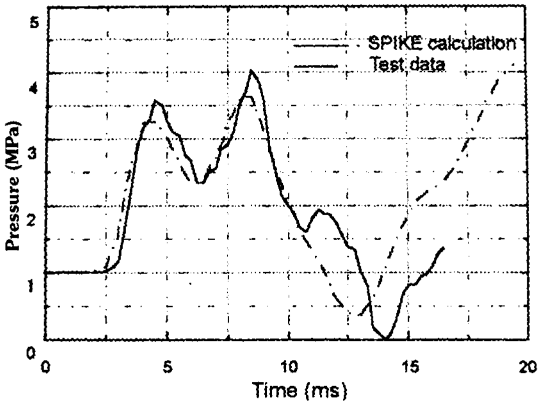


Fig. 17. Comparison of Pressure Changes

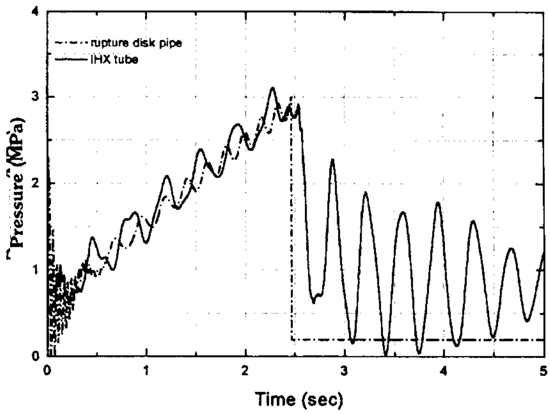


Fig. 18. Pressure Transient at the IHTS

After this peak pressure, a lower secondary pressure follows and decreases slowly because pressure change is not sensitive to time. This step is called the quasi-steady state. The intensity of the initial spike pressure depends on the internal structure of the steam generator and the transient characteristics of the sonic waves. The intensity of the pressure depends on the inertia constraints of the IHTS.

A computer code, SPIKE, has been developed for analyzing the various characteristics of the IHTS resulting from initial spike pressure[22]. The sodium flow in the IHTS is assumed to be a compressible, one dimensional, unsteady viscous flow. As part of an effort to check the performance of the code, the test results from the JNC(Japan Nuclear Corporation)'s experimental facilities for modeling the IHTS[23]were simulated using the code. A sample comparison of the calculated results with the experimental values at the IHX inlet is shown in Figure 17. The figure shows that the calculated results are consistent with the experimental values[22]. The code will be further verified by imulation experiments at KAERI's test facility, which is scaled down from KALIMER to the ratio of 1/256 (the heat load scale-down ratio, about 1/6 of the linear scale-down ratio). The SG-model has a diameter of 420mm (O.D.) and length of 2,750mm with 5 layer helical coil tubes, of which the total length is about 280m and the material is stainless steel 304 without welding. To assure safety from accidents caused by large water leakage in KALIMER steam generators, studies on leak propagation, their simulation, and a pressure change estimation by computer codes have been carried out. The computer codes, HOPRE and DIPRE , are being developed to analyze the quasi-steady-state pressure[24].

The SPIKE code has been applied to investigate the pressure transients at various points of the IHTS of KALIMER. As shown in Figure 3, KALIMER is of two IHTS loops, each loop consisting of a steam generator, two intermediate heat exchangers, a pump and pipes which are connected with several fittings. The IHTS of KALIMER was modeled as a network having 40 branches and 39 junctions for the analyses. Results show that pressure transients or peak pressures are rather sensitive to such design parameters as leak rate, the distance between steam generator and IHX, and the distance between the lower plenum of steam generator and the rupture disc installed at the piping connecting the bottom part of the SG to the sodium expansion tank. Excessive pressure in the SG bursts open the rupture disc, dumping sodium and reaction products to the sodium expansion tank. It was also observed that pressure transient behavior was quite sensitive to the size of the sodium expansion tank[25]. Fig. 18 shows the effect of the rupture disc, which is set to burst at the system pressure of 3.0 MPa . The solid curve represents the pressure at the IHX, which is the major equipment to be protected, should the sodium-water reaction occur. It can be observed that the rupture disc is quite effective in relieving the pressure build up in the IHTS.

4. Conclusions

At the early phase of the conceptual design of KALIMER, emphasis has been placed upon establishing a self-consistent design satisfying a set of the major safety design requirements for accident prevention. Some of these requirements include those for inherent and passive characteristics of negative reactivity insertion and decay heat removal, high shutdown reliability, high seismic margin, and prevention of sodium chemical reaction, among others.

In this context, a number of up-front engineering and design analyses have been performed to evaluate the conceptual design of KALIMER from the viewpoint of some of the major safety design requirements. For the analysis of the inherent safety of KALIMER, a plant-wide transient analysis code SSC-K has been developed. Models for reactivity feedback effects, pool thermal-hydraulics, and passive decay heat removal have been developed into the code and a preliminary analysis of UTOP and ULOF/LOHS performance has been performed. The results show that net reactivity stays negative during the transients analyzed. Design alternatives also have been investigated to improve the passive decay heat removal capability of KALIMER. Analysis of the air channel cooling with a new radiation-convection structure revealed substantial heat transfer enhancement.

In the meantime, seismic base isolation is shown to significantly reduce the seismic response of the reactor building and reactor internal structures of KALIMER and, therefore, provide a great advantage in safety for the structural design of nuclear power plants. Engineering and design analyses were also performed to evaluate the IHTS configuration against sodium water reaction. The SPIKE code has been developed to investigate the pressure transients at various points of the IHTS of KALIMER. The results show that pressure transients or peak pressures are rather sensitive to such design parameters as leak rate, distance between the lower plenum of steam generator and rupture disc, and the distance between steam generator and IHX.

Substantial progress has been made in developing and validating the methodologies and engineering analyses of the conceptual design of KALIMER. An investment is also being made on the key design features testing, such as electromagnetic pump, self-actuated shutdown system, and fuelling machine in reactor vessel, and passive decay heat removal system. Effort will be continued to be made down the road to accomplish the mandate; that is, to complete the basic design of KALIMER as well as the supporting R&D work by 2006.

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