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액체금속로 KALIMER 중대사고 시 용융연료 거동 해석

Analysis of Molten Fuel Behavior in Coolant Channel during Severe Accidents in KALIMER

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제 출 문

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과 제 명 : 액금로 안전해석 기술개발

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요 약 문

KALIMER-600 은 설계 고유 안전성으로 ATWS 사고와 같은 설계기준 사고 시, 냉각재 비등이나 핵연료 용융이 발생치 않는다는 것이 예비안전해석 결과에 의해 입증된 바 있다. 그러나 미래형 설계에 대해서는, 격납용기 요건 및 안전 성능 평가를 위해 발생 확률이 극히 낮은, 계통 3중 실패로 인한 노심 용융 사고도 고려하여야 한다. 중대사고 발생 시, 노심 용융물은 원자로 용기 내에 제한해야 하며 (In-vessel retention), 이 문제의 해결에 주요한 현상이 노심 상.하부 구조물에서의 노심 용융물의 응고 및 폐색(freezing and plugging) 현상이다.

본 연구에서는 KALIMER-600 노심에서 냉각재 상실 사고 시, 분출된 용융 연료가 냉각재 유로에서 응고되어 유로를 막기 전에 이동할 수 있는 거리를 계산하였다. 해석 모델에서는 용융연료가 핵연료 용융점 보다 온도가 낮은 냉각유로 구조물과 접촉할 때, 연료는 고화하여 각(solid crust)을 형성한다고 가정한다. 해석 결과 용융연료는 KALIMER 노심 하부의 차폐(lower shield) 부위와 상부의 핵연료 기체 plenum 영역에서 응고되어 냉각유로가 막히게 되는 것으로 나타났다.

SUMMARY

Preliminary safety analyses of the KALIMER-600 design have shown that the design has inherent safety characteristics and is capable of accommodating double fault initiators such as ATWS events without boiling coolant or melting fuel. For the future design of liquid metal reactor, however, the evaluation of the safety performance and the determination of containment requirements may require consideration of tripe-fault accident sequences of extremely low probability of occurrence that leads to fuel melting. For any postulated accident sequence which leads to core melting, in-vessel retention of the core debris will required as a design requirement for the future design of LMR. For sodium-cooled core designs with metallic fuel, one of the major phenomenological modeling uncertainties to be resolved is the potential for freezing and plugging of molten metallic fuel in above- and below-core structures and possibly in inter-subassembly spaces.

In this study, scoping analyses were carried out to evaluate the penetration depths in the coolant channels by molten fuel mixture during the unprotected loss-of-flow accidents in the core of the KALIMER-600.It is assumed in the analyses that a solid fuel crust would start to form upon contact with the coolant channel structure temperature of which is below the fuel solidus. The analysis results predict that the coolant channels would be plugged by the freezing molten fuel in the inlet lower shield as well as in the outlet, fission-gas-plenum region for the KALIMER-600 design.

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1. INTRODUCTION

KALIMER-600 is a sodium-cooled, pool-type reactor with electrical capacity of 600MWe. The development of the reactor system was made to satisfy the design targets of enhanced safety, competitive economics, proliferation resistance and environmental friendliness. The reactor core is breakeven and consists of metal fuel with no radial and axial blanket. The other main design features of the reactor include pool type primary heat transport system(PHTS), two-loop intermediate heat transport system(IHTS) and steam generator system(SGS), totally passive decay heat removal system and seismically isolated reactor building. The preliminary safety analyses of the KALIMER-600 design have shown that the design has inherent safety characteristics and is capable of accommodating anticipated transient without scram(ATWS) events. The self-regulation of power without scram is mainly due to the inherent and passive reactivity feedback[1,2].

Extensive analysis of the response of metallic fueled reactors to double fault initiators such as loss of flow, transient overpower, or loss of heat sink with failure to scram have shown that the reactors passively adjust their power to match the available heat rejection capability without boiling coolant or melting fuel[3,4,5]. The analysis results have been confirmed in full-scale tests in the EBR-II reactor for the loss of flow and loss of heat sink sequences[6,7]. However, the evaluation of the safety performance and the determination of containment requirements may require consideration of tripe-fault accident sequences that lead to fuel melting. Such multiple fault initiators have extremely low probability of occurrence, however. Most likely candidates identified for the accident initiators leading to a partial or whole core melting are the essentially unlimited rod bank runout unprotected transient overpower(UTOP) and the abrupt unprotected loss of flow(ULOF).

For any postulated accident sequence which leads to core melting, in-vessel retention of the core debris will required as a design requirement for the future design of LMR. For sodium-cooled core designs which feature metallic fuel, some phenomenological modeling uncertainties will need to be resolved to a level which is required for commercial licensing. One such data need is the potential for freezing and plugging of molten metallic fuel in above- and below-core structures and possibly in inter-subassembly spaces. An important issue is the quantity of fuel removed from the core region to assure subcriticality.

Refreezing and plugging of molten reactor fuel has been studied extensively, both theoretically and experimentally--mostly for uranium dioxide fuels. Uranium dioxide, with a melting temperature about 1500℃ above that of steel, typically has been found to melt and entrain a layer of cladding as the flow front passes, forming a slug of molten claddign traveling ahead of the molten fuel. Plugging then occurs by refreezing this molten cladding to form a blockage farther down the channel. In contrast, molten metallic uranium, with a melting temperature below that of steel, seldom, if ever, would become hot enough to generate a contact temperature above the steel solidus and so would be expected to melt and entrain little, if any, cladding material near the flow front. Rather, the process would be one of desuperheating and refreezing of the fuel itself.

In this study, scoping analyses were carried out to evaluate the penetration depths in the coolant channels by molten fuel mixture during the unprotected loss-of-flow accidents in the core of the KALIMER-600. It is assumed in the analyses that a solid fuel crust would start to form upon contact with the coolant channel structure temperature of which is below the fuel solidus. Crust growth would be limited by the convective heat transfer as long as significant melt superheat persisted. When superheat neat the flow front had been reduced sufficiently, conductive heat transfer from the crust would exceed the convective heating from the melt and the crust could grow to plug the channel completely. The analysis results predict that the coolant channels would be plugged by the freezing molten fuel in the inlet lower shield as well as in the outlet, fission-gas-plenum region for the KALIMER-600 design.

2. KALIMER Design Features

2.1 Reactor Core Design

The KALIMER breakeven core, generating 1589.3 MWt, is a homogeneous, metal alloy fuel design with 703 assemblies: 102 inner driver fuel assemblies, 126 middle driver fuel assemblies, 108 outer dirver fuel assemblies, 12 control rods, 1 ultimate shutdown system (USS) assembly, 72 reflector assemblies, 78 B_4C shield assemblies, 90 shield assemblies, and 114 in-vessel storages (IVSs). This configuration is shown in Figure 2.1. An ultimate shutdown system (USS) is included as a means to bring the reactor to cold critical conditions in the event of a complete failure of the normal scram system and after the inherent reactivity feedbacks have brought the core to a safe, but critical state at an elevated temperature[1,2].

In this design, the blanket assemblies are completely removed in the core so as to exclude the production of the weapon-grade plutonium. The inlet temperature is 366.2 ^oC and the bulk outlet temperature is 510.0 ^oC, with an average temperature rise of 143.8 ^oC. The active core height is 100.0 cm and there are no axial blankets in the core. The core structural material is HT9. Equivalent diameter of the core(including control rods) is 318.0 cm. The physically outermost core diameter including all the assemblies is 471.5 cm. Table 2.1 shows the overall core design parameters

Table 2.2 shows the fuel assembly design parameters. The fuel form is U-TRU-10%Zr ternary alloy. The duct pitch is 16.21 cm. The driver fuel assembly includes 267 fuel pins and 4 moderator pins. The moderator pins are used to reduce the coolant void reactivity worth by softening the core neutron spectrum. The driver fuel has smeared density of 75 %. At equilibrium, the design basis refueling interval follows 18 months of operation at 85 % capacity factor, with one-third of the driver fuel assemblies, being replaced during each outage. The fuel assemblies are not shuffled, but remain in position for the entire cycles. Following removal from the core, they decay for one operating cycle in the IVS positions..

The driver fuel assembly includes 271 pins. Of these pins, 267 pins are fuel and 4 pins are moderator. Fuel pins are made of sealed tubing containing fissile and fertile materials in columns. The fuel is immersed in sodium for thermal bonding with the cladding. Surrounding the pin bundle and welded to the nosepiece is a hexagonal crosssection duct. The duct functions to control the coolant flow and isolate each pin bundle from its neighbors. It is also the structural tie between the top and bottom end hardware of the assembly. A thickened duct section, the above core load pad, serves to maintain assembly spacing and prevent core compaction.

Figures 2.2 and 2.3 show the overview of the fuel rod design and fuel assembly duct along with key section views , respectively. Neutron and gamma shielding is provided integral with the pin bundle above and below the core in the form of end plugs and/or fission gas plena. All the assemblies use wire wrap pin spacing, the same handling socket design, and the same nosepiece design except for the discrimination post. The handling socket provides inner and top surfaces and radial holes that mate with the refueling machinery grapple and two outer hexagonal surfaces that form the top load pad and the insertion clearance for the in-vessel fuel storage hanger restraint.

Flow orificing is part of the receptacle, not the assembly. It is provided by a stack of perforated disks in the lower portion of the receptacle. The core assemblies use common structural components with only the assembly internals being changed to form the various assembly types. The discrimination socket in the bottom of the assembly nosepiece also changes with assembly type.

2.2 Reactor Vessel and PHTS

The reactor vessel is the boundary of the primary heat transport system and performs support and container functions during all temperature, pressure, and load variations which occur during the operating lifetime. The reactor vessel, which is made of type 316 stainless steel, has overall dimensions of 18m height, 11.41m outer diameter, and 0.05m thickness in conceptual design. The reactor vessel is attached to the reactor head and supports the reactor internal structures, the reactor core, and primary sodium, including IHX, PHTS pumps and IVTM. The IHX and PHTS pumps are installed through the holes of the baffle plate and separation plate. The space inside the reactor vessel is thermally divided into two regions, hot region and cold region. The support barrel, the baffle plate, separation plate and the reactor baffle form the boundary of the two regions.Two PHTS pumps, four IHXs and two DHXs are arranged in the annular space between the support barrel and reactor baffle in a reactor vessel. The PHTS pump is a centrifugal type mechanical pump which is located vertically and circulates the primary sodium to transfer core heat to IHTS through the IHX. Figure 2. 4 shows a schematic diagram of the reactor vessel and PHTS of the KALIMER-600 design[1,2].

The major elements of the core support structure are as follows:

1) The simple skirt type structure,
2) The primary sodium inlet plenum which contains the receptacles for the subassembly nosepieces,
3) The support barrel and the former rings.

The core assemblies are held (1) by their nosepieces in the receptacles, and (2) by the load pads near the top of the assemblies which are surrounded by the former ring attached to the support barrel. The separation of the assemblies is maintained by an intermediate plane of load pads at an elevation above the active core. Positioning of the handling sockets is also maintained by the top load pads. The intermediate load pads above the core are not restrained by the former ring attached to the support barrel. Thus, the core assemblies are free to bow as dictated by temperature differences and their metallurgical condition. Load transfer is through the core assembly load pads to the former ring and the support barrel.

In the vertical direction, core restraint is provided by the combination of assembly weight and hydraulic balance. Hydraulic balance is a method for reducing the upward-acting hydraulic forces on the assemblies. The bottom ends of the receptacles for these assemblies have hydraulic communication with the low pressure region under the inlet plenum. High pressure sodium entering the receptacles and core assemblies from the sides pushs down on their inside bottom ends. To maintain the differential pressure, the receptacles and the core assembly nozzles have piston ring seals above and below their inlet ports. Additionally, backup holddown may be provided by nosepiece seal/lock rings. The lower nosepiece hydraulic seal rings may be used to supplement the normal assembly holddown. The seat which fits into in the nosepiece receptacle has a conic ramps that the seal rings must be compressed past for assembly removal. The enhanced friction force generated as the rings slide up the conic sections provides the supplemental holddown.

The inlet plenum, located in the central region of the core support structure and below the core, receives primary sodium from the primary pipes and distributes it to the core via the nosepiece receptacles. The receptacles are located in a triangular pitch to match the core array map. The receptacles participate in the core orificing. The depth of the inlet plenum is established by the space required for the inlet piping nozzle forging welds and for the radial flow area necessary to assure uniform flow distribution to all the core assemblies. This flow distribution is further enhanced by the design of the receptacles which are necked down on their lower end to increase the available flow area.

2.3 Residual Heat Removal System

The residual heat removal system of KALIMER-600 provides highly reliable heat removal capability in case of unavailability of the main heat transport path consisting of PHTS, IHTS, and SGS. When the normal heat transport path is not available, the safety related PDRC (Passive Decay heat Removal Circuit) system provides sufficient decay heat removal (DHR) capability and it relies exclusively on natural convection heat transfer.

PDRC system comprises independent two heat removal loops, and each loop is equipped with one sodium-sodium decay heat exchanger (DHX), one sodium-air heat exchanger (AHX) and the heat removing sodium piping connecting DHX with AHX as shown in Figure 2.5. DHX is a shell-and-tube type counter-current flow heat exchanger, and is placed at the position higher than the cold pool free surface during the normal plant operation, and thus it is not directly contact with the hot sodium. AHX placed above the reactor building has the function of rejecting the system heat to the environment, and the counter-wounded heat transfer tubes are helically arranged. A direct heat exchange is performed between the tube-side hot sodium and the shell-side atmospheric air through the AHX sodium tube surface.

The heat removal of DHX in a normal steady-state condition is achieved only by thermal radiation heat transfer process, and this feature makes the heat loss through PDRC system at the normal plant operation pretty small. In contrast, under transient conditions such as the loss of heat sink accident, the liquid level difference between the hot and the cold pool is eliminated and then the hot pool sodium is expanded due to a continuously generated core decay heat.

The hot pool sodium overflows into the cold pool through the inner space of the shell-side DHX, and the natural circulation path of pool sodium is formed. As natural circulating sodium flow rate in the shell-side DHX increases, the heat removal rate of

DHX is also increased. The system is self-regulating since the heat removal capacity of PDRC is directly proportional to the pool sodium temperature variation. To this end, the core decay heat can be continuously discharged into the final heat sink, i.e., the atmosphere, without either operator action or any active component actuation[1,2].

Table 2.1 Core Design Parameters

Operating Conditions	
Core Thermal Power (MWt)	1589.3
Core Electric Power (MWwe)	600
Core Mixed Mean Inlet/Outlet Temperature (^{o}c)	366.2/ 510.0
Total Flow Rate (kg/s)	kg/s
Plant Thermal Efficiency (%)	
Plant Capacity Factor (%)	85.0
Refueling Interval (months)	18
Effective Full Power Day (EFPD)	465
Number of Batches	
Inner Driver Fuel	3
Middle Driver Fuel	3
Outer Driver Fuel	3
Core Design Parameters	
Core Configuration	Homogeneous
Number of Core Enrichment Zones	3
Active Core Height (cm)	100.0
Maximum Core Diameter (cm)	471.5
Axial Blanket Thickness (cm)	None
Feed Fuel Composition	LMR Recycled
Number of Assemblies	
Inner Driver Fuel	102
Middle Driver Fuel	126
Outer Driver Fuel	108
Reflector	72
Control Rod	12
USS (SASS)	1
B_4C/Radial Shield	78/90
IVS	114
Total	703
Core Structural Material	HT9

Table 2.2 Fuel Assembly Design Parameters

Fuel /Moderator Material	U-TRU-10Zr/ZrH_2
Smeared Density (%)	75
Active Fuel Length (cm)	100.0
Fuel Element Length (cm)	366.8
Overall Assembly Length (cm)	462.2
Duct Pitch (mm)	162.1
Duct Gap (mm)	4.0
Duct Wall Thickness (mm)	3.7
Pins per Fuel Assembly (Fuel/Moderator)	267/4
Pin Outer Diameter (mm)	7.60
Pin P/D Ratio	1.184
Upper Fission Gas Plenum Length (/Na Filled) (cm)	152.5(/ 25.0)
Core Structural Material	HT9
Volume Fractions (%)	
Driver Fuel	
Fuel Slug	29.56
Coolant(including Bond)	45.56
Structural Material	24.30
Moderator	0.59
Control Rod	36.07
Pin Material(20% enriched B_4C)	36.61
Coolant	27.32
Radial Shield(/with B_4C Shield)	
Pin Material(smeared)(/90% enriched B_4C)	75.43/58.10
Coolant	15.82/18.24
Structural Material	8.75/23.66

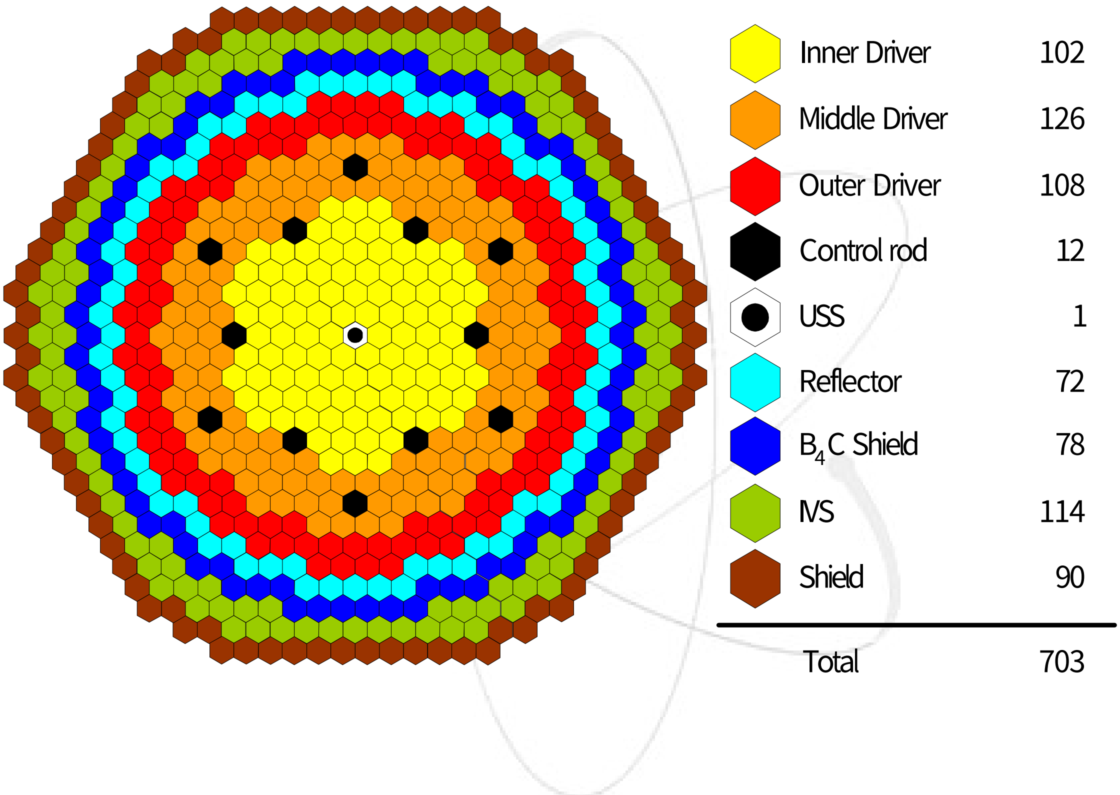

Figure 2.1 Core Design Configuration

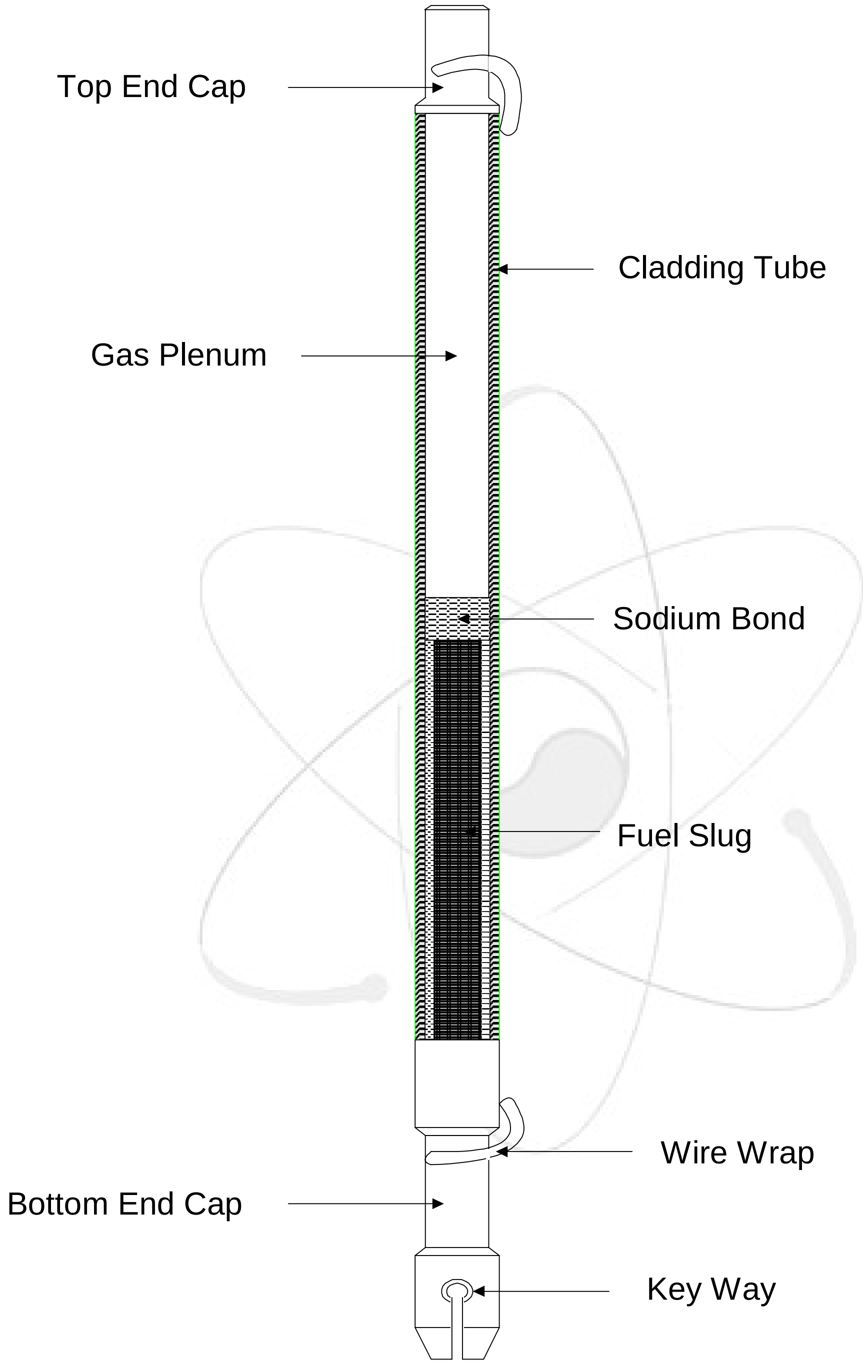

Figure 2.2 Schematic of the KALIMER DriverFuel Pin

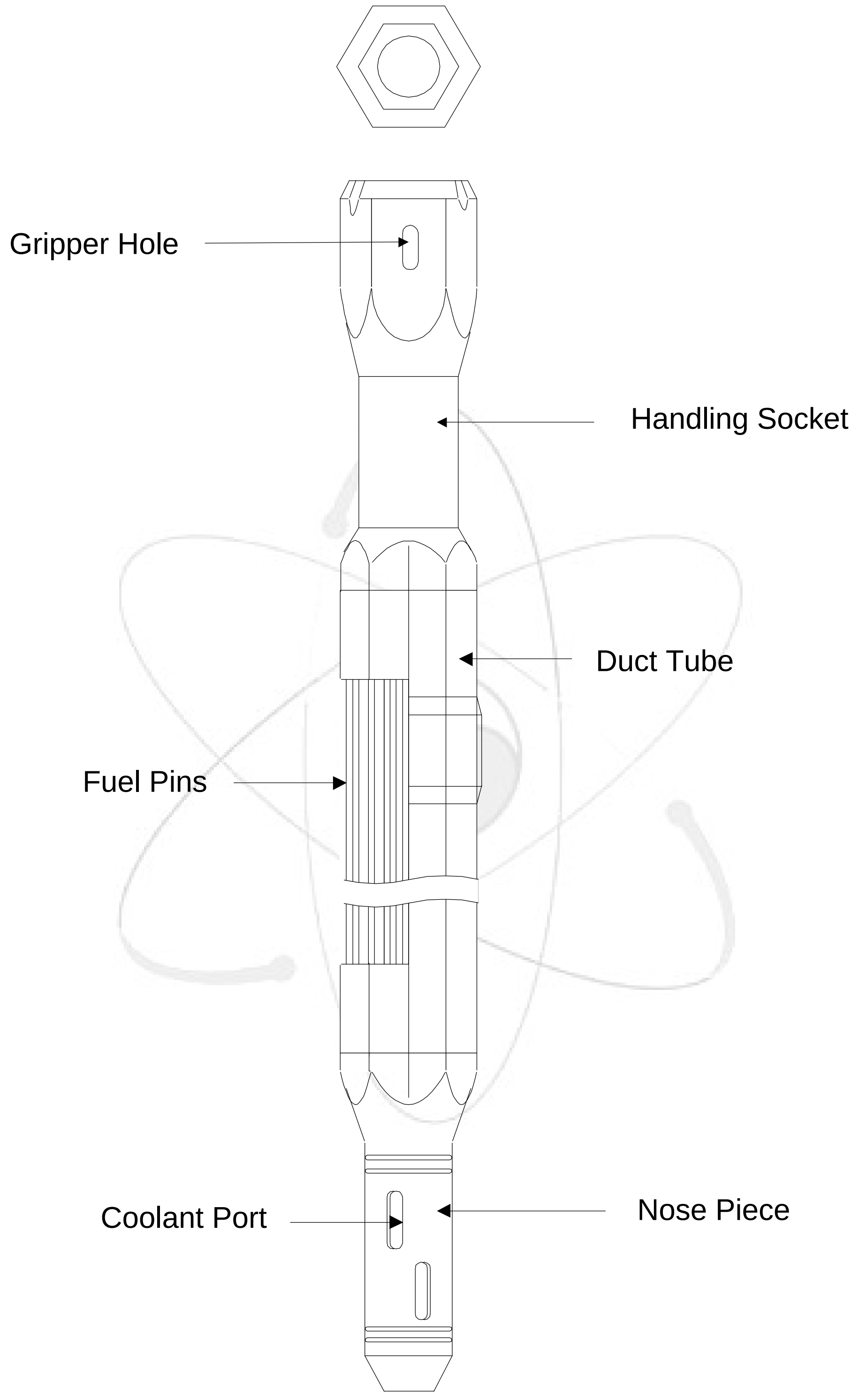

Figure 2.3 Schematic of the KALIMER Fuel Assembly

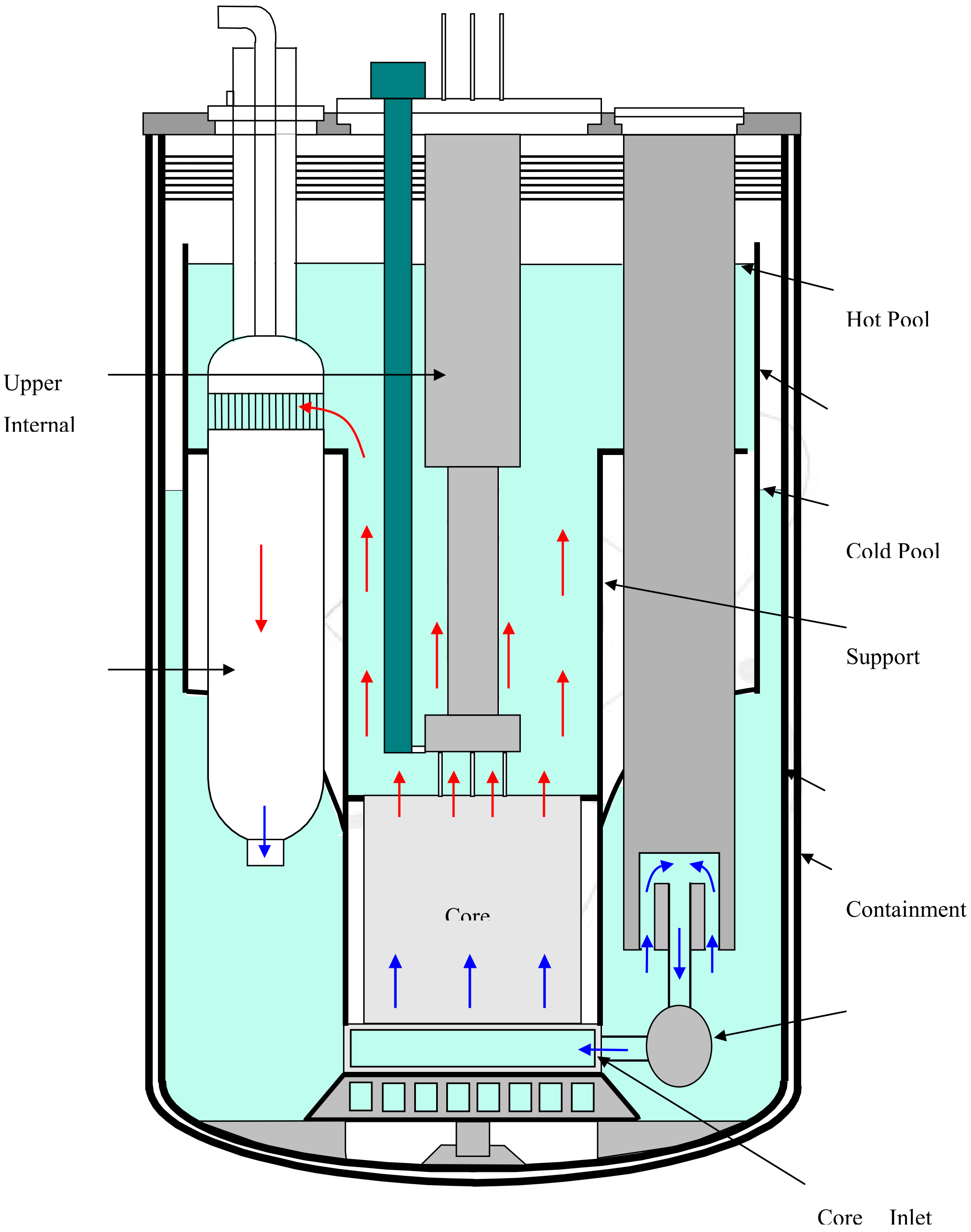

Figure 2.4 KALIMER Reactor Vessel and PHTS

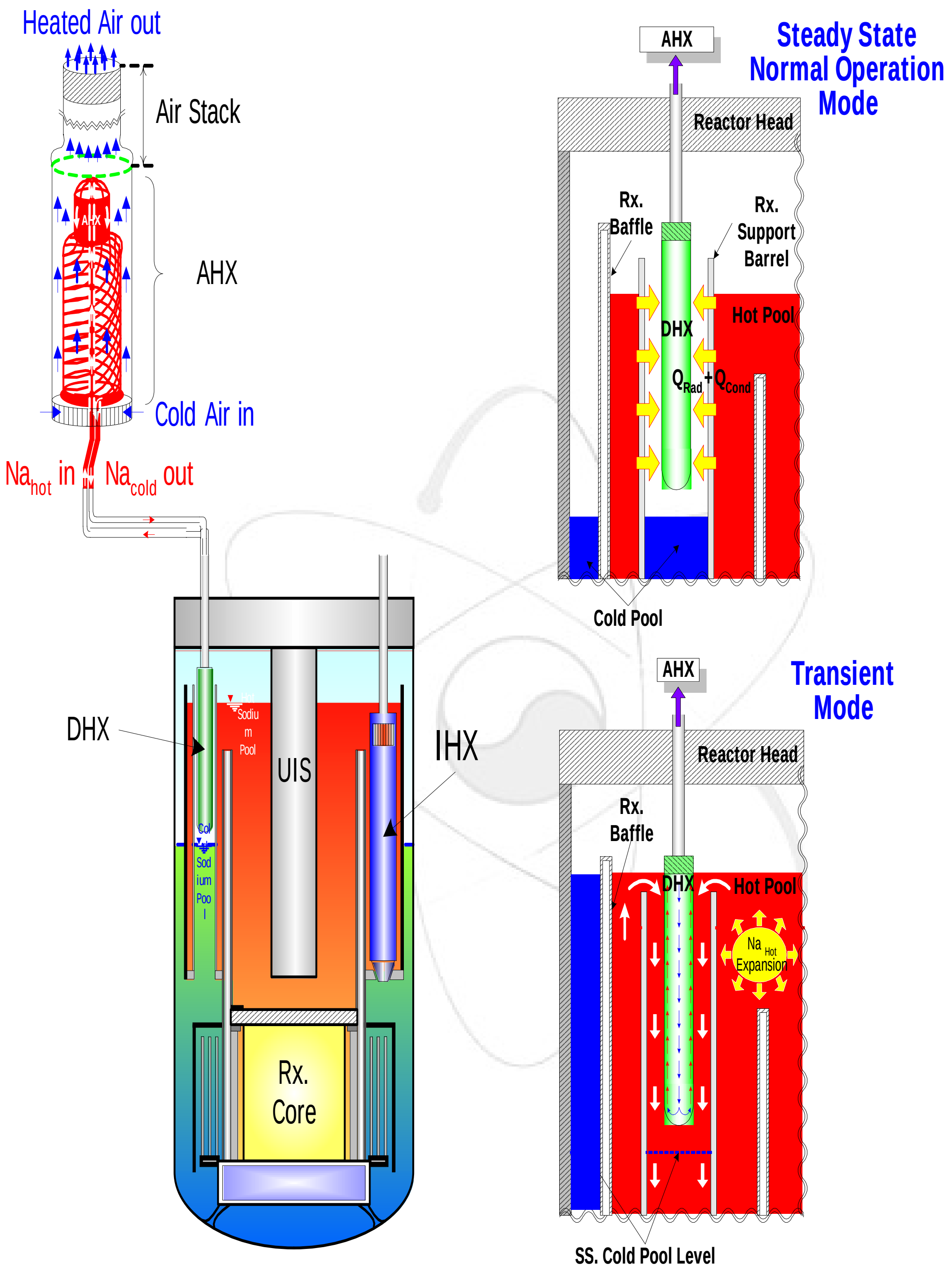

Figure 2.5 KALIMER Residual Heat Removal System

3. SEVERE ACCIDENT SEQUENCES

For the coolant boiling and subsequent fuel melting to be initiated in the KALIMER reactor core, a third fault initiator must be assumed. Such multiple fault initiators have extremely low probability of occurrence, however. Most likely candidates identified for the accident initiators leading to energetic core disruption are the essentially unlimited rod bank runout unprotected transient overpower(UTOP) and the abrupt unprotected loss of flow(ULOF)[2,5,6].

3.1 Unprotected Transient Overpower

The unlimited rod bank runout UTOP can be caused by: (1) a cluster of control rods withdraw at the same time resulting in the addition of large amount of reactivity enough to bring about fuel melting, or (2) a catastrophic earthquake causing the core support structure to fail, allowing the core to fall away from the control rods[6].

Sequence of the accident, predicted by several analyses[2,3,4,5,6] and the tests in EBR-II[7], is as follows: The control rod bank withdrawal may add reactivity to the core in the range of several dollars per second. As reactivity increases, the core power and temperature rise. The increasing temperature generates reactivity feedback in opposition to the power increase, but continued addition of reactivity from the withdrawing control rods increases the reactor power to the point where the coolant at core exit is approaching boiling.

At the same time, temperatures in fuel pins are at or above the fuel melting point, while still below the clad melting temperature. Fuel temperatures tend to rise no more than a few hundred degree above the fuel melting point due to the reactivity brought about by the upward in-pin motion of the molten fuel occurring prior to cladding failure. The combination of Doppler and axial expansion feedback and the negative feedback associated with the in-pin fuel relocation prevents the reactivity from reaching prompt critical. The peak temperature of the cladding remains skewed toward the core outlet because the sodium continues to flow, carrying heat from the lower portion of the fuel along with it, and because the high thermal conductivity and efficient heat transfer assure that the temperatures of the coolant and adjacent cladding are not very different.

The analysis showed that dispersive fuel motion follows cladding failure, with the continued operation of the coolant pumps causing the coolant to sweep fuel upward and out of the fuel assemblies toward the outlet plenum. At lower reactivity insertion rates, cladding failure occur near the top of the fuel pins, and post failure fuel relocation contributes uniformly negative reactivity feedback. At higher reactivity insertion rates, cladding failures occur closer to the axial midplane of the core, but by this time, in-pin fuel relocation has contributed enough negative reactivity feedback to prevent the initially positive, post-failure fuel relocation from driving the reactor prompt critical.

The predicted consequences for IFR-type fuel of the triple-fault transient overpower accident have been confirmed in experiments in Argonne National Laboratory's Transient Reactor Test Facility (TREAT) [8]. Those experiments demonstrated that coolant voiding and cladding failure occur almost simultaneously at a power-to-flow ratio about four times nominal. The failure always occurred at or near the interface between the bottom of the gas plenum and the top of the fuel. Irradiated fuel has been swollen by fission gas that is trapped at high pressure (up to 100 atm.) in interconnected pores that occupy up to 30% of the volume of the distended fuel. As the fuel melts, the trapped gas forms high-pressure bubbles. Upon failure of the cladding, much or all of the molten center of the fuel pin (extending even to the bottom of the pin) froths through the breach at the top, driven by the expanding fission gas.

The TREAT experiments showed that the froth of molten fuel then travels dispersively upward out of the core region; no blockages are formed in the coolant channel, and coolant flow is maintained. Thus, remelting and reentry into the core of the relocated fuel material is not an issue. This low temperature dispersive behavior of the fuel provides massive negative reactivity feedback, which overwhelms all other reactivity feedback effects and takes the core to deep subcriticality, terminating the transient after the failure of only a small number of assemblies. Involvement of the whole core is averted, energy deposition is low, prompt criticality is never achieved, and there are no shock waves.

3.2 Unprotected Loss of Flow

For the triple-fault abrupt loss of flow without scram(LOFWS), the assumed initiator is almost instantaneous depressurization of the inlet plenum, due either to pump failure compounded by a blockage that prevents a normal (~10 s) flow coastdown, or to simultaneous rupture of all inlet pipes, perhaps in a massive earthquake. Failure to scram is assumed as well (scram would preclude fuel melting and boiling). The expected scenario for this case, confirmed by extensive analyses with the SAS$4A code, has the following sequence of events[4,5].

The almost instantaneous loss of pressure in the inlet plenum causes the flow to stop within hundreds of milliseconds. Power does not diminish as quickly, because of delayed-neutron holdback and only modest feedbacks from structural expansion. Coolant voiding in the core begins in a few assemblies, causing flow reversal and subsequently adding reactivity at the rate of approximately 10$/sec from the positive sodium void effect. The power rises sharply and causes voiding to occur in other assemblies. This causes the net reactivity to increase but the core does not reach prompt critical due to the fast-acting negative feedbacks from Doppler effect and axial fuel expansion.

At this point in the accident, the fuel is molten within intact cladding, and has made the entrapped fission gas much hotter than the gas in the plenum. The mixture of fuel and gas expands upward within the intact cladding , introducing some negative reactivity feedback. This drives the reactor subcritical. Subsequent voiding in other assemblies at different powers thereafter keeps the net reactivity in the range of a few tens of cents positive until failures in pin cladding occur.

Just prior to initial cladding failure, approximately two-thirds of the fuel assemblies could be completely voided of coolant. With no sodium flow, cladding failures occur just above the midplane, with each assembly contributing a short pulse of positive reactivity as the froth of fission gas and fuel moves axially within the cladding toward the midplane failure location. The froth of fuel and gas then exits at the failure site and travels axially upward and downward(dispersive motion)in the largely voided coolant channel, introducing negative reactivity. Depending on assembly location, these pulses can be from a few cents to a few tens of cents of reactivity. Differences in assembly power coupled to the narrowness of these pulses relative to time for the accident to proceed assures that the cladding failures are sufficiently separated in time that the compactive fuel motion is not able to take the reactor above, or even near, prompt critical. The subsequent ex-pin fuel motion is sufficient to drive the reactor deeply subcritical, terminating the initiating phase of the transient.

The transient is over within several seconds. Only a few full-power-seconds of energy are generated prior to termination, and peak fuel temperatures remain below 2000 K, considerably less than the temperature required to produce significant pressure of fuel vapor. Super prompt criticality has been avoided by the fuse-like action of the fuel itself, fuel vapor is not generated, and any liquid fuel the mixes with sodium is at much too low a temperature to cause a vapor explosion.

Parametric studies have shown that the inherent non-energetic termination of the initiating phase in a triple-fault abrupt LOF without scram is sensitive to the timing of within-pin movement of fuel into the fission-gas plenum prior to fuel pin failure, but is insensitive to most of the other parameter. Axial relocation prior to cladding failure is an established phenomenon, having been observed in the above-referenced TREAT tests; however, an experimental simulation of the entire abrupt LOF sequence has not been performed in TREAT [6].

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4. FUEL PENETRATION ANALYSIS

4.1 Reference Sequence

The TOP accident sequence does not appear to post any credible threat to the integrity of the reactor vessel either during the initiation phase of the accident or during the post accident heat removal phase with the collection of core debris in the lower structure of the reactor vessel. Analysis results for TOP accidents initiated by reactivity insertion rates up to 10 $/s show that the net reactivity does not reach prompt critical. After cladding failure, the pressure drop caused by the continued operation of the coolant pumps causes fuel ejected into the coolant channels to be swept upward out of the core, leading to reactor shutdown[6].

Some of the fuel may freeze on the cooler structures in the upper part of the assembly, but complete blockages are not expected and some of the fuel may be swept into the outlet plenum. Reestablishment of single-phase coolant flow in the channel is expected. This postfailure behavior of the fuel is supported by the M-series TREAT tests.[8] The tests showed that following cladding failure, the fuel moved a substantial distance downstream from the original fuel zone. Recriticality considerations indicate that even if the entire core fuel inventory were to collect on the lower core support plate, the resulting configuration would not be critical. It is therefore unlikely that the small amount of fuel which might be swept into the outlet plenum could collect anywhere into a critical configuration. Thus, all the fuel, whether in the core or relocated outside the core can be expected to be in coolable, subcritical configurations.

The sequence of the accident initiated by an abrupt loss of flow accompanied by failure of all reactor scram system is used as the reference accident sequence in this study for the analysis of molten fuel freezing and plugging in a sodium voided coolant channel. Upon cladding failure in a sudden loss-of-flow transient, molten fuel released at near midplane elevations would be blown biaxially through coolant channels toward the core ends. Cladding within the core boundaries generally would be so hot as to preclude refreezing at those elevations. Upon entering cooler regions above and below the core, rapid cooling, refreezing and plugging in principle could occur.

Refreezing and plugging of molten reactor fuel has been studied extensively, both theoretically and experimentally--mostly for uranium dioxide fuels. Uranium dioxide, with a melting temperature some 1500℃ above that of steel, typically has been found to melt and entrain a layer of cladding as the flow front passes, forming a slug of molten claddign traveling ahead of the molten fuel. Plugging then occurs by refreezing this molten cladding to form a blockage farther down the channel. In contrast, molten metallic uranium, with a melting temperature below that of steel, seldom, if ever, would become hot enough to generate a contact temperature above the steel solidus and so would be expected to melt and entrain little, if any, cladding material near the flow front. Rather, the process would be one of desuperheating and refreezing the fuel.

4.2 ANALYSIS MODEL

For temperatures of interest the fuel-wall contact temperature would be below the fuel solidus such that a solid fuel crust would start to form upon contact. So long as significant melt superheat persisted (more than as few tens of degrees) crust growth would be limited by the convective heat transfer. Later, when superheat neat the flow front had been reduced sufficiently, conductive heat transfer from the crust would exceed the convective heating from the melt and the crust could grow to plug the channel completely.

The rate of change of fuel temperature near the flow front may be written as

$$\rho c_p V \frac{dT}{dt} = -hA[T(t) - T_{sol}] \tag{4.1}$$

where,

ρc_p = fuel volumetric heat capacity

V = fuel channel volme

A = fuel channel surface area

h = convective heat transfer coefficient

T_{sol} = fuel solidus

Equation (4.1) may be converted to a spatial form by use of $dt = dx/u$, where u is the velocity of the melt front,

$$\frac{dT}{dx} = -\frac{h}{(V/A)\rho c_p u}[T(x) - T_{sol}] \tag{4.2}$$

Defining the characteristic length x_0 such that Equation (4.2) can be rewritten as,

$$\frac{dT}{dx} = -\frac{1}{x_0}[T(x) - T_{sol}] \tag{4.3}$$

then it may be expressed with the use of Equation (4.6) as

$$x_0 = \frac{\rho c_p u d}{4h} \tag{4.4}$$

where d is the channel hydraulic diameter: $d = 4(\frac{V}{A})$

The convective heat transfer may be described by a conventional liquid metal heat transfer correlation,

$$Nu = a + bPe^{0.80} \tag{4.5}$$

where a = 7, b=0.025.

It can be shown that the characteristic penetration distance depends mildly upon the flow velocity so that it may be approximated as a constant with the flow velocity evaluated at the time the flow first exits the original core boundary. Assuming the characteristic length constant, Equation (4.3) may be integrated to get

$$T(x) - T_{sol} = [T(0) - T_{sol}]e^{-x/x_0}; T(x) \geq T_{liq} \tag{4.6}$$

$$T(x) - T_{sol} = [T_{liq} - T_{sol}]e^{-\frac{x - x_1}{x_0^{'}}}; T_{liq} \geq T(x) \geq T_{sol} \tag{4.7}$$

where

$T(0)$ = initial temperature of the melt
T_{liq} = fuel liquidus temperature
x_1 = penetration distance at which liquidus is reached

$x_0^{'}$ is the characteristic length in the mushy zone of molten fuel between liquidus and solidus temperature.

The penetration distance at which fuel liquidus is reached can be obtained by inverting Equation (4.7) with *T(x)* equated to T_{liq} as follows:

$$x_1 = x_0 \ln[\frac{T(0) - T_{sol}}{T_{liq} - T_{sol}}] \tag{4.8}$$

Meanwhile, the penetration distance at which crust growth to full channel diameter can occur, x_2, may be expressed by defining T_{sol}^{+} and equating $T(x)$ to it:

$$x_2 = x_1 + x_0^{'} \ln[\frac{T_{liq} - T_{sol}}{T_{sol}^{+} - T_{sol}}] \tag{4.9}$$

where T_{sol}^{+} is the temperature incremented above solidus at which full diameter frozen crust can form. T_{sol}^{+} may be estimated by equating the rate of heat conduction from the crust to the wall equals that of heat convection from the melt to the crust, namely

$$(k/\delta)(T_{sol} - T_w) = h[T^{+}{}_{sol} - T_{sol}] \tag{4.10}$$

where,
δ = the conductive growth to fill the channel(=d/2)
T_w = wall temperature

Rearranging Equation (4.10) to get T_{sol}^{+},

$$T_{sol}^{+} = T_{sol} + \frac{2k}{hd}(T_{sol} - T_w) = \frac{2}{Nu}(T_{sol} - T_w) \tag{4.11}$$

The melt front penetrates further after desuperheating and the penetration distance can be expressed in terms of the time to channel closure by the melt front, which may be estimated by solving a transcendental equation.

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4.3 Analysis Results

Using the methods developed above, the penetration depths in the coolant channels by molten fuel mixture were estimated during the unprotected loss-of-flow accidents in the core of the KALIMER-600. At the time of cladding failure, a large fraction of the fuel in the failed pin is molten, and as this fuel enters the coolant channel, about half the fuel moves upward and remainder moves downward. Previous analyses show that the fuel moves past the upper end of the core with a velocity less than 10 m/s and with fuel temperatures around 1300℃. Temperatures near the center of the core are less than 1500℃ when fuel begins to leave the core. Cladding within the core boundaries generally would be so hot as to preclude refreezing at those elevations. Upon entering cooler regions above and below the core, rapid cooling, refreezing and plugging in principle could occur.

The current KALIMER-600 design features a continuation of pin geometry below and above the active core as in the KALIMER-150 design. The inlet/outlet channels are defined by a continuation of the in-core cladding to form shield rods and fission gas plena, which are typical of many LMR designs. Using the pin channel data listed in Table 4.1, the channel hydraulic diameter at the top and bottom ends of the active core of the KALIMER-600 is estimated to be 4.15 mm. Meanwhile, a special, large-flow-diameter endfittings are employed for the inlet shielding in the IFR design, for which the hydraulic diameter is as much as 2.5 cm.

Table 4.2 shows the results of parametric studies of plugging depths calculated using the model described in Section 4.2. Major parameters of variation include the initial melt velocity, pin channel configuration and melt temperature, among others. First, two different initial velocities of the melt were used; 10m/s and 1.0 m/s , representing the upper limit of melt velocity in the abrupt loss of flow and gravity flow of the melt, respectively. Also, a number of cases were analyzed for the different pin channel conditions of the IFR and KALIMER designs. Initial fuel temperatures corresponding to radially peak, average, and minimum power location were used: 1,500, 1,320, and 1,140 ^{o}C , respectively. Structural wall temperatures used for channel inlet and outlet are 350 and 500 ^{o}C , respectively.

Data used for the material properties of the melt are:

ρ_0 = 14 g/cm^3

k_0 = 0.2 W/cm.K

c_p =0.2 J/g.K

l_f = 40 J/g

T_{liq} =1,300 ^{o}C

T_{sol} =1,100 ^{o}C

Meanwhile, the following data were used for the material properties of the steel structure:

ρ_0 = 7.5 g/cm^3

k_0 = 0.26 W/cm.K

c_p =0.6 J/g.K

As for the friction factor used to get the penetration distance after desuperheating, following conventional correlation for turbulent flow was utilized where applicable,

$$f = 0.184\,\mathrm{Re}^{-0.2} \tag{4.12}$$

These results predict that the coolant channels would be plugged by the freezing molten fuel in the inlet , lower shield region with its length about 112 cm as well as in the outlet, fission-gas-plenum region (length ≈ 150 cm) for the KALIMER-600 design. For the IFR design, the outlet region of the same length and hydraulic diameter as the KALIMER design would be also plugged but complete penetration of the inlet, large-diameter lower-shielding region (length ≈ 50 cm) is highly likely for all radial regions.

The initial fuel temperatures are characteristic of those predicted for a sudden loss-of-flow transient. Any milder transient which leads to a cooler, lower velocity melt would be more plugging prone, even in large diameter inlet channels.

4.4 Post Freezing/Plugging Behavior

There are two limiting scenarios depicting molten core debris flow downward from the core region. If the below-core structure has large diameter channels, the molten core debris can flow rapidly from the core, reaching the lower core structures a few seconds after cladding failure. However, if the lower core structure has very small diameter channels, the core debris will freeze and plug the channels.

For the case with large diameter channels, the melt from a given subassembly will descend rapidly into sodium-filled areas driven by fission gas pressure. This could result in incomplete fragmentation or fragmentation with very small particles. The vigor of the melt-sodium contact may improve melt spreading in lateral direction, but the melt will have a high rate of decay-heat generation. Also, the melt should contain relatively little dissolved steel because of the rapid downflow.

For the case with small diameter channels, the melt will probably emerge from the assembly by meltthrough of the duct wall. The flow will then be within the spaces between ducts. In these spaces, the fuel melt will tend to displace sodium and may flow to the level of the upper core plate without plugging. The melt will then require considerable time to melt through another duct wall below the level of the pin structure and flow downward through the sodium inlet channels. Considering the difficult flow path for the melt, the melt will reach the core support plates after the decay heat is reduced to some extent. Thus, the corresponding amount of decay heat need be assumed in considerations of fuel coolability on the lower core support plates for this case. The fission gas pressure driving the melt would have been relieved sidewise by duct failure or upward long before the melt reaches below the wire-wrapped region so that there would be a gradual entry into the sodium. Thus, the fragmentation should be relatively complete because of the gradual entry into the sodium, but the degree of spreading may be small. The melt will probably contain considerably more steel than the eutectic composition of the fuel because of having to essentially melt through much of the lower structure.

Table 4.1 KALIMER Pin Channel Design Parameters

	KALIMER 600	KALIMER150
Core Thermal Power(MWt)	1589.3	392.2
Core Electric Power(MWe)	600	150
Core Inlet/Outlet Temperature(^{o}C)	366.2/510.0	386.2/530.0
PHT Pump	2 Centrifugal	4 EM pumps
Core Configuration	Heterogeneous	Homogeneous
Core Structural Material	HT9	HT9
Number of Assemblies		
Driver Fuel (Inner/Middle/Outer)	102/126/108	54
Inner Blanket	None	24
Radial Blanket	None	48
GEM/Control Rod/USS	0/12/1	6/6/1
Reflector/Shield/IVS	72/68/114	48/126/54
Driver Fuel Assembly		
Duct Pitch(mm)	162.1	161.0
Duct Gap(mm)	4.00	4.00
Duct Wall Thickness(mm)	3.70	3.70
Duct Outer/Inner Flat to Flat(mm)	158.09/150.70	157.0/149.6
Overall Assembly Length(mm)	4,621.7	4,756
Pins Per Assembly	271	271
Pin Data		
Fuel Type	U-TRU-10Zr	U-TRU-10Zr/U-10Zr
Pin Pitch(mm)	9.0	8.9
Pin Outer/Inner Diameter(mm)	7.60/6.54	7.4/6.3
Pin P/D Ratio	1.184	1.203
Fuel Slug Diameter(mm)	5.66	5.46
Cladding Thickness(mm)	0.53	0.55
Overall Pin Length(mm)	3,668	3,708
Fuel Slug Height(mm)	1,000	1,000
Upper Gas Plenum/Na Filled(mm)	1,525/25.0	1,565/25.0
Lower Shield(mm)	1,117.6	,117.6

Table 4. 2 Freezing Model Parametric Study Results

Melt Velocity	Channel Configuration	Penetration (cm)	Melt Temperatures			$T_{sol}^{+} - T_{sol}$ (°C)
			1,500 °C	1,320 °C	1,140 °C	
10 m/s	*KALIMER Inlet* d= 0.4 cm T_w=350 °C	X_1	14.0	1.93	0.0	36.2
		X_2	56.9	44.8	2.52	
		x_c	7.47	7.47	7.47	
		x_t	64.4	52.3	10.0	
	KALIMER/IFR Outlet d= 0.4 cm T_w=500 °C	X_1	14.0	1.93	0.0	28.9
		X_2	62.5	50.4	8.11	
		x_c	7.47	7.47	7.47	
		x_t	70.0	57.9	15.6	
	IFR Inlet d= 2.5 cm T_w=350 °C	X_1	145	20.0	0.0	9.60
		X_2	895	769	352	
		x_c	137	137	137	
		x_t	1,032	906	489	
1.0 m/s	*KALIMER Inlet* d= 0.4 cm T_w=350 °C	X_1	4.67	0.642	0.0	120
		X_2	10.0	5.97	-	
		x_c	1.92	1.92	1.92	
		x_t	11.9	7.89	-	
	KALIMER/IFR Outlet d= 0.4 cm T_w=500 °C	X_1	4.67	0.642	0.0	96.3
		X_2	12.3	8.32	-	
		x_c	1.92	1.92	1.92	
		x_t	14.3	10.2	-	
	IFR Inlet d= 2.5 cm T_w=350 °C	X_1	74.2	10.2	0.0	48.9
		X_2	267	203	-	
		x_c	35.8	35.8	35.8	
		x_t	303	239	35.8	

Note: X_1 = penetration distance at which liquidus is reached

X_2 = penetration at which crust growth to full channel diameter can occur

x_c = distance melt front travels after desuperheating

x_t = total penetration distance ($x_t = X_2 + x_c$)

$T_{sol}^{+} - T_{sol}$ = temperatureincrement above solidus at which full-diameter crust can form

5. CONCLUSION

In the KALIMER-600 design, the wire-wrapped pin bundle structure continues for 112 cm below the core region with solid steel rods and extends above the core with fission gas plenum region for about 150 cm. The hydraulic diameter of its pin channel amounts only to 4 mm such that the core debris will freeze and plug the channel above and below the core. This is also the case for the KALIMER-150 design. If the below-core structure has large diameter channels as in the IFR design, the molten core debris can flow rapidly from the core, reaching the lower core structures a few seconds after cladding failure.

One major modeling uncertainty is the possible effects of sodium in the channels on the freezing and plugging process. Virtually all previous work in this area involved the freezing and plugging of molten uranium oxide in LMR and LWR fuel channels. In that case, the very high temperature of the fuel caused the sodium to move ahead of the melt. However, the temperature of fuel/steel melts is not far from the boiling point of sodium. This may result in the fuel mixing with the sodium coolant, which could promote freezing and plugging.

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서 지 정 보 양 식

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초록(15-20 줄내외)

KALIMER-600 은 설계 고유 안전성으로 ATWS 사고와 같은 설계 기준 사고 시, 냉각재 비등이나 핵연료 용융이 발생치 않는다는 것이 예비안전해석 결과에 의해 입증된 바 있다. 그러나 미래형 설계에 대해서, 격납용기 요건 및 안전 성능 평가를 위해 발생 확률이 극히 낮은 노심 용융 사고(계통 3중 실패로 인한)도 고려 되어야 한다. 중대사고 시 노심 용융물을 원자로 용기 내에 제한해야 하며(In-vessel retention), 이 문제의 해결에 주요한 현상이 노심 상.하부 구조물에서의 노심 용융물의 고화 및 폐색(freezing and plugging) 현상이다.

본 연구에서는 KALIMER-600 노심에서 냉각재 상실 사고 시, 분출된 용융 연료가 냉각재 유로에서 이동하는 거리를 계산하였다. 해석 모델에서는 용융연료가 핵연료 용융점 보다 온도가 낮은 냉각유로 구조물과 접촉할 때, 연료는 고화하여 각(solid crust)을 형성한다고 가정한다. 해석 결과 용융연료는 KALIMER 노심 하부의 차폐(lower shield) 부위와 상부의 핵연료 기체 plenum 영역에서 고화되어 냉각유로가 막히게 되는 것으로 나타났다

주제명키워드
(10 단어내외)

액체금속로, 노심용융사고, in-vessel retention, 용융연료 고화, 냉각유로 plugging

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Analysis of Molten Fuel Behavior in Coolant Channel during Severe Accidents in KALIMER

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Abstract (15-20 Lines)

Preliminary safety analyses of the KALIMER-600 design have shown that the design has inherent safety characteristics and is capable of accommodating double fault initiators such as ATWS events without boiling coolant or melting fuel. For the future design of liquid metal reactor, however, the evaluation of the safety performance and the determination of containment requirements may require consideration of tripe-fault accident sequences of extremely low probability of occurrence that leads to fuel melting. For any postulated accident sequence which leads to core melting, in-vessel retention of the core debris will required as a design requirement for the future design of LMR. For sodium-cooled core designs with metallic fuel, one of the major phenomenological modeling uncertainties to be resolved is the potential for freezing and plugging of molten metallic fuel in above- and below-core structures and possibly in inter-subassembly spaces.

In this study, scoping analyses were carried out to evaluate the penetration depths in the coolant channels by molten fuel mixture during the unprotected loss-of-flow accidents in the core of the KALIMER-600.It is assumed in the analyses that a solid fuel crust would start to form upon contact with the coolant channel structure temperature of which is below the fuel solidus. The analysis results predict that the coolant channels would be plugged by the freezing molten fuel in the inlet lower shield as well as in the outlet, fission-gas-plenum region for the KALIMER-600 design.

Subject Keywords (About 10 words)	Liquid Metal Reactor, KALIMER severe accident, fuel freezing and plugging,