SAFETY CHARACTERISTICS OF THE U.S. ADVANCED LIQUID METAL REACTOR CORE

P.M. Magee, A.E. Dubberley, G.L. Gyorey, A.J. Lipps and T. Wu
GE Nuclear Energy, Advance Nuclear Technology
San Jose, CA 95153-5354

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ABSTRACT

The U. S. Advanced Liquid Metal Reactor (ALMR) design employs innovative, passive features to provide an unprecedented level of public safety and the ability to demonstrate this safety to the public. The key features employed in the core design to produce the desired passive safety characteristics are: a small core with a tight restraint system, the use of metallic U-Pu-Zr fuel, control rod withdrawal limiters, and gas expansion modules. In addition, the reactor vessel and closure are designed to have the capability to withstand, with large margins, the maximum possible core disruptive accident without breach and radiological release.

INTRODUCTION

In January 1989, the General Electric Company was awarded the Advanced Liquid Metal Reactor Program by the U. S. Department of Energy to develop an innovative advanced reactor concept aimed at improving safety, enhancing plant licensability, lowering plant costs, simplifying plant operation, and with a capability to utilize radioactive actinide waste products as fuel.[1] The concept referred to as PRISM (Power Reactor - Innovative, Small Module) is a small modular, pool-type sodium-cooled fast reactor. The basic nuclear steam system, one reactor module and steam generator, is shown in Figure 1. A PRISM power block, producing 465 MWe, consists of three modules operated in parallel and tied to a single turbine generator.

ALMR passive safety assures accommodation of anticipated transients without scram. A passive reactor vessel auxiliary cooling system (RVACS) assures safety-grade decay heat removal.[2] This system is always in operation and requires no operator action or power source. In addition to accommodating all design basis events and several anticipated transients without scram (ATWS) within performance limits, the ALMR reactor module is also designed to accommodate a whole core melt and disassembly accident without vessel breach. In addition, a low leakage, pressure retaining containment capability provides residual risk accommodation.

This paper focuses on four topics: (1) the basic ALMR safety philosophy, (2) reactor design features affecting safety, (3) performance during anticipated transients, without scram (ATWS), and (4) accommodation of positive sodium void worth.

ALMR SAFETY APPROACH

The safety goals and requirements established for the ALMR program include the conventional ones established in the past for sodium- cooled nuclear power plants, such as leak protection, fire mitigation, protection from natural phenomena, etc. However, in response to the recently evolving regulatory framework in the United States and the lessons learned from past experience, a number of additional safety goals were established for the ALMR program which may not have been used, or at lest not empha-

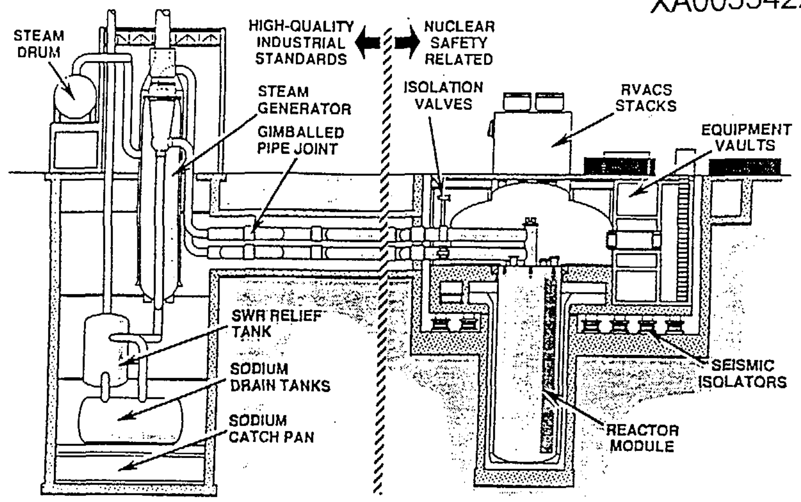


Figure 1. ALMR NUCLEAR STEAM SUPPLY SYSTEM

sized, in the past. In particular, the ALMR safety approach emphasizes accident prevention, preferably by using passive and natural processes, backed up by accident mitigation as required. This approach has led to the following safety goals and characteristics:

- Failure to scram to be less than 10^{-6} per demand; reactor protection system well separated from the plant control system.
- Strong inherent negative reactivity feedback for core reactivity control to maintain a safe state for ATWS events.
- No operator action required to reach and maintain a safe state.
- Passive decay heat removal, not vulnerable to operator errors.
- Operator action unable to inhibit or override safety actions.
- Very low core damage probability, below 10^{-6} per plant year.
- High margins in ultimate seismic capability.
- Accidental radiation release probabilities and characteristics such that detailed offsite evacuation exercises and early warning will not be required.
- Passive and other innovative safety characteristics to be demonstrable in a prototype without damaging the plant.

REACTOR DESIGN FEATURES FOR SAFETY

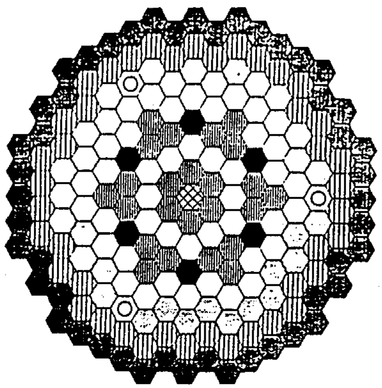
The core and reactor are designed specifically to support passive reactivity control and natural circulation decay heat removal with ample margins for public safety. The reactor is tall to enhance natural circulation in both the primary coolant and RVACS air circuits. The core power rating, 471 MWt, is sized to match the RVACS capabilities. The core average coolant outlet temperature, 485°C (905°F), and inlet temperature, 338°C (640°F), are low to enhance safety performance margins throughout the reactor. The reference fuel is metallic U-Pu-Zr alloy providing a relatively small cold-to-hot reactivity swing. The ferritic alloy HT9 is used for cladding and assembly ducts to minimize swelling associated with high irradiation fluence.

Core Design

The ALMR core nuclear design is largely governed by passive safety and reactivity control issues. The core configuration, shown in Figure 2, contains 42 fuel assemblies, 57 blanket assemblies, 3 gas expansion modules (GEMs), 6 control assemblies and a central ultimate shutdown assembly. Surrounding the core are 42 reflector and 48 shield assemblies. The core height is 134.6 cm (53 in) and the assembly pitch is 15.96 cm (6.282 in).

The core is designed to minimize reactivity changes during the cycle. Blanket assemblies are shuffled each outage, for three, four or five cycles, through a fixed pattern, starting in internal blanket positions and migrating to the radial blanket positions. This enhances internal conversion and reduces burnup swing. Shuffling has the added benefit of increasing the breeding capability and flattening the radial power profile. The fuel is not shuffled, but has three positionally symmetrical refueling batches. With this fuel management arrangement, there is no reactivity variation from cycle to cycle caused by refueling pattern variations. Burnup swing is small (about $0.50) for the equilibrium core, making metal fuel axial swelling effects the largest contributor to reactivity change during a cycle.

Reactivity control for normal operations of startup, load following and shutdown is accomplished by a system of six control rods. The control rod absorber assemblies have sufficient worth that any one of the six can shut down the reactor to a cold subcritical condition. Redundancy in worth and diversity in method of rod insertion makes the single shutdown system very reliable, such that the required scram reliability ($<10^{-6}$ failures to scram per demand) is estimated to be met with substantial margin. The ultimate shutdown system is included as a fully diverse means of scramming the reactor in the event of a complete failure of the normal scram system. The device consists of a canister of small boron carbide balls in the cover gas space under the reactor head connected by a flow tube to an empty duct in the center of the core. The



	Gas Expansion Module	*3*
	Shield	*48*
	Reflector	*42*
	Radial Blanket	*33*
	Driver Fuel	*42*
	Internal Blanket	*24*
	Control	*6*
	Ultimate Shutdown	*1*
	Total:	*199*

Figure 2. ALMR Configuration

balls can be released from the canister to drop down the tube into the core. This manually actuated scram follows reactivity self-limitation by core thermal feedbacks.

Several design features combine to make possible the single device scram capability of either scram system by contributing to a small scram worth requirement:

- Small core size for tight neutronic coupling
- Fast spectrum reactor for tight coupling
- Minimized reactivity swing
- Minimized reactivity design and fabrication uncertainties
- Minimized cycle to cycle reload reactivity variations

Rod stops are used to limit the potential reactivity insertion possible by an uncontrolled withdrawal of all six control rods. Mechanical stops in each control drive mechanism physically interfere with carriage out-motion and passively stop a rod runout. The amount of out-motion permitted by the stops is determined during reactor design by the core temperatures permitted during an overpower event. For cores with a potential for large reactivity changes during a cycle, such as the prototype test core, the stops can be moved during power operation under the control of a redundant, safety-grade, electronic controller. The gas expansion modules (Figure 3) are included in the core design to offset the reactivity insertion that occurs late in an unscrammed loss of cooling event. In such an event, the reactor vessel heats up and expands downward from the reactor head. Since the core is supported from the bottom by the vessel and the controls are supported from above by the head, this expansion lowers the core away from the controls and adds reactivity. The GEMs provide a fast negative reactivity feedback upon loss of primary flow to counter this effect. GEMs have been tested in a loss of flow safety test program in the Fast Flux Test Reactor.[3]

GEMs are empty assemblies, sealed at their top end and connected to the inlet coolant plenum at the bottom end. As a GEM is lowered into the reactor, it traps helium cover gas in its interior tank. The elevation of the sodium-gas interface inside the GEM is determined by the pressure of the sodium at the assembly lower inlet holes. The elevation of the seal at the assembly top end is selected to establish the height of the interface elevation just above the core at normal full power operating conditions. When the pumps are stopped, the gas expands against the reduced coolant pressure, pushes the sodium out of the assembly and lowers the gas-sodium interface to below the core. A core reactivity change is caused by the loss of sodium scattering back into the core when the device voids. For a small core with high radial leakage effects, GEMs are an effective reactivity feedback device. In the ALMR core, each of the three GEMs has a feedback worth of about 0.23$. The choice of metal fuel contributes to passive safety. Metal fuel with sodium thermal bonding between fuel and cladding results in low fuel temperatures during operation. Low operating temperatures and the harder spectrum of metal-fueled cores reduce the positive Doppler reactivity feedback in cooling the fuel from operating conditions to shutdown conditions. Thus less negative feedback from other passive means is needed to produce power self-control during loss of cooling events that include a failure to scram. Fuel axial thermal expansion also produces meaningful levels of reactivity feedback.

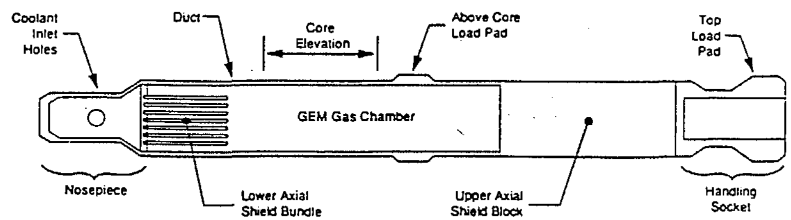


Figure 3. GAS EXPANSION MODULE

Reactor Design

Reactor design features that contribute to passive safety are the control drivelines and the core support structure. The control rod drivelines are located external to the upper internal structure so that they are immersed in the hot sodium coolant discharged from the core. The resulting rapid heatup of the driveline during over–temperature events provides a relatively rapid negative reactivity feedback as thermal expansion lowers the controls deeper into the core.

The core support structure contributes to reactivity feedbacks for self-regulation of power in several ways. The grid plate that spaces the lower ends of the core assemblies is made of a high thermal expansion coefficient austinitic steel. Over-temperature of the inlet coolant expands the plate and spreads the core, causing a negative reactivity feedback. The reactor structures surrounding this component are designed to accommodate this expansion and not interfere with the feedback. The assembly ducts include stiff load pads immediately above the core to control assembly spacing. Thermal expansion of these stiff pads forces core expansion and adds negative reactivity feedback. Finally, the lateral core restraint features are spaced at elevations promoting thermal bowing of the assemblies that expands the core during increases in power to flow ratio. The result is an increase in negative reactivity feedback from radial thermal expansion. Overall, about 3/4 of core radial expansion feedback is a result of thermal expansion of the above core load pads and the grid plate. The remaining 1/4 is a result of bowing distortion of the ducts.

ATWS SAFETY CRITERIA AND LIMITS

The ALMR safety design goal is to accommodate ATWS events, specifically unprotected transient overpower (UTOP), unprotected loss of flow (ULOF), and unprotected loss of heat sink (ULOHS), so that core damage leading to a safety challenge does not occur. The conservative criteria used to insure public safety during the ATWS events are the following:

Cladding Failure

High temperature cladding creep rupture is the principal fuel pin failure phenomenon during transients. The ferritic alloy HT9 has significant degradation in creep strength at elevated temperatures. For example, a typical end-of-life fuel pin at a 760°C (1400°F) peak cladding midwall temperature will fail by creep rupture in about 45 minutes, including the effects of cladding wastage from liquid-phase fuel-clad alloy.

A complication of the cladding creep rupture phenomena is internal cladding wastage caused by formation of a low melting temperature alloy of the metal fuel and cladding. Below the alloy melting temperature of 700°C (1300°F), the alloy formation is limited to a solid diffusion process and cladding degradation is extremely slow. Once the alloy has melted, the wastage rate increases rapidly. As a conceptual design limit, the cladding attack is limited to less than 10% of the wall thickness, or 0.051mm (0.002in). This also limits the amount of fuel involved in the liquid attack.

Local Sodium Boiling

To avoid local sodium boiling within the core, the peak coolant temperature in the core is limited to 954°C (1750°F). A conservative boiling point temperature for conditions in the core with the primary pumps not operating is 960°C (1760°F), and 1070°C (1960°F) is representative of the boiling point with the primary pumps operating at full flow.

Structural Integrity

The reactor vessel, internal structures and reactor components are protected from thermal creep damage by limiting the core average outlet temperature to a time-at-temperature criterion. For durations less than 1 hour, the limit is 760°C (1400°F). For times over 1 hour, 700°C (1300°F) is the limit.

Fuel Melting

Fuel central melting, per se, is not a cause of pin failure. TREAT tests, especially M5 and M6, have demonstrated that extensive fuel melting

(exceeding 80% of a given cross-section) does not affect the basic pin failure mechanism.[4] Failure by cladding creep rupture, with clad thinning by fuel-clad liquid phase formation, is the appropriate mechanistic cladding breach criterion even for pins with molten fuel in contact with the cladding.

The fuel melting temperature is well above the minimum temperature for formation of fuel-cladding liquid phase. Limitation of the amount and time duration of molten fuel (<50% pin cross-sectional area for <2 minutes) is provided to eliminate potential fuel motion reactivity effects.

ACCOMMODATION OF ANTICIPATED TRANSIENTS WITHOUT SCRAM

The analyses of ATWS events are performed with the events being initiated while the reactor is at nominal, full power (100%) conditions, with a core inlet temperature of 338°C (640°F) and a mixed mean outlet temperature of 485°C (905°F). The analyses are performed at beginning of equilibrium cycle conditions, when the power in the driver assemblies is the greatest. The peak assembly is representative of fresh fuel in the reactor; however, for conservatism, the fuel conductivity is based on irradiated fuel since the conductivity of fresh fuel drops rapidly during the first 1.5-2 atom % burnup.

The peak temperatures and maximum cladding attack by fuel/ clad liquid phase formation are summarized for the three ATWS events (UTOP, ULOF/LOHS, ULOHS) in Table 1. It is apparent that the conservative ATWS safety criteria are met with large margins in all cases. Performance details are discussed in the following paragraphs.

All-Rods Withdrawal Without Scram

This event postulates that a malfunction in the reactivity controller causes the shim motor to continue to withdraw the control rods until the driveline reaches the rod stop, and the Reactor Protection System (RPS) function of scramming the reactor is absent. Analysis of the withdrawal accident conservatively assumes a 0.40$ insertion limit. The rod stops are positioned to limit the reactivity insertion to 0.30$, less than the

Table 1
SUMMARY OF ATWS TRANSIENTS

	Peak Power	Peak Temperatures (°C/°F) Fuel	Cladding	Coolant	Core Ave. Outlet	Cladding Attack (mils)
Limits			700* (1300)	955 (1750)	700 (1300)	2.0
ATWS Events						
All-rods UTOP (0.40$)	175%	1030** (1884)	714 (1317)	685 (1265)	595 (1106)	0.011
ULOF/LOHS	100	800 (1470)	658 (1217)	647 (1196)	618 (1145)	0
ULOHS	100	790 (1454)	711 (1312)	711 (1312)	694 (1282)	0.00091

* Integrate cladding attack for time above 700 (1300)

**Greater than 955 (1750) for <2 minutes; limited molten fuel satisfies criterion

0.40$ limit even with appropriate margin. The reactivity insertion rate is 0.02$ per second, which corresponds to the maximum speed of the shiln this event, the rods are fully withdrawn to the rod stops (assumed to be at 0.40$) in 20 seconds. As shown in Figure 4, the power rises rapidly as the rods are withdrawn, and reaches a maximum of 175% of full power in 40 seconds. At this time, the negative reactivity feedback (shown in Figure 5), mostly Doppler and thermal expansion, has turned the power rise around. The power then drops over the next 100 seconds, and stabilizes at about 130% of full power. The fuel pin temperatures follow the power changes, with the peak fuel, cladding and coolant temperatures reaching maximums of 1030°C (1885°F), 715°C (1320°F) and 685°C (1265°F), respectively at 40 seconds. The cladding attack due to fuel/clad liquid phase formation during this event is less than .0025 mm, well below the 0.051 mm limit. The reactor mixed mean outlet temperature also peaks at 595°C (1105°F) in 40 seconds, and then levels off at about 563°C (1045°F). The temperatures during this event are shown in Figure 6. For conservatism during this event, it is assumed that the axial expansion of the fuel is based on the cladding temperature rather than the fuel temperature. All safety criteria are met with margin.The nominal 0.30$ all-rods withdrawal reactivity insertion, associated with the nominal rod stop setting, results in a peak fuel temperature of 975°C (1785°F), or 56°C (100°F) lower than the conservatively assumed 0.40$ reactivity insertion case.

Loss of Flow Plus Loss of Heat Sink Without Scram

The specified ALMR ATWS event is a loss of primary flow without scram. Normally, heat transfer out through the IHX/IHTS/steam generator train would continue. Even with loss of secondary and water-steam side pumping power (as in a station blackout), natural circulation would occur in the IHTS loop and heat would be removed through the IHTS pipe insulation and also into the stored water supply. For convenience in analysis and to provide additional conservatism, it has been assumed that no heat removal occurs through the IHX, IHTS and BOP. The transient is thus analyzed as an unprotected (no scram) loss of flow and heat sink (ULOF/LOHS) event. Adding further conservatism, the axial fuel expansion is based on fuel temperatures rather than on cladding temperatures.

In this event, the power and flow drop rapidly at the start of the transient, as shown in Figure 7, since the loss of flow activates the GEMs. As shown in Figure 8, there is some initial under-cooling of the fuel pins before the negative reactivity of the GEMs takes effect. The fuel, cladding and coolant temperatures peak at 800°C (1470°F), 660°C (1220°F) and 650°C (1200°F), respectively, at 3-5 seconds into the transient, and then the core starts to cool. As seen in Figure 9, there is little negative feedback other than GEMs during the early part of this transient because the GEMs rapidly reduce the power. However, as the primary pump coastdown ends, the coolant starts to heat up again. Since there is no heat sink other than RVACS, the vessel continues to heat up for a while. The heatup of the vessel causes the core to move away from the control rods, and the net effect is positive reactivity feedback due to thermal expansion. At about 1400 seconds into the transient, the effects of the control rod expansion along with the other positive feedback effects overcome the negative feedback of the GEMs and the power starts to rise. The core heats up during this slow power rise, and other feedback mechanisms (Doppler and core radial expansions) become negative and turn the small power excursion around. The mixed mean outlet temperature slowly increases as RVACS heat removal comes in balance with the decay power,and it peaks at about 635°C (1175°F) at 46000 seconds, as seen in Figure 10. The cladding attack during this event is less than .0025 mm.

Loss of Heat Sink Without Scram

In this event, it is assumed that the normal heat removal train (IHX, IHTS and BOP) is suddenly removed, e.g., due to a large sodium-water reaction causing the secondary sodium to be dumped. It is assumed that scram is not successful and the primary pumps continue to run until they are tripped by an over-temperature signal at 538°C (1000oF). The core performance during this event is summarized in Figures 11 through 14. Peak fuel, cladding and coolant temperatures are 790°C (1454°F), 711°C (1312°F), and 710°C (1312°F), respectively. Consistent with the other two ATWS events, all safety criteria are met with margin.

ACCOMMODATION OF POSITIVE SODIUM VOID REACTIVITY

The reactivity effect from core voiding at end-of-equilibrium cycle conditions is shown in Table 2. The principal contributors to the overall void worth are the interior assemblies (fuel and internal blankets). The control assemblies contribute the majority of the negative void reactivity due to the high leakage in these assemblies.The peripheral assemblies such as radial blankets, reflector and shield assemblies have a net effect of a small negative void worth. The negative effect results from the increase of neutron leakage that overrides the positive spectral effect

Table 2
SODIUM VOID WORTHS FOR THE ALMR REFERENCE METAL CORE

Assembly Voided	Void Worth ($) Full Length	Positive Region Only
Driver	2.95	3.87
Internal Blankets	2.47	2.72
Radial Blankets	-0.27	0.11
Fuel & Blankets	5.15	6.70
Control	-2.91	0.79
Ultimate Shutdown	-0.18	0.01
Gas Expansion Modules	-1.43	0.00
Reflectors & Shield	-0.20	0.00
Total Non-Fuel	-4.72	0.80
Whole Core	0.43	7.50

from sodium voiding. The total worth from the voiding of high power assemblies (fuel and blanket) is $5.15, assuming full assembly voiding.

The total worth of voiding all assemblies in the core is $0.43. It is noted that the reactivity effect of voiding control rods may be overestimated since no transport corrections were applied to the voided control assemblies.

The maximum void worth for a single assembly (full length) is about $0.16 for the inner fuel and $0.12 for the internal blankets. The void worth becomes less positive for those assemblies that are some distance from the core center and eventually becomes negative for the radial blanket and shield assemblies.

The axial distribution of void worth is shown in Figure 15 for the fuel and blanket assemblies and in Figure 16 for the non-fuel assemblies. In the axial traverse, the void worth is negative at the upper and lower ends of the fuel column due to pronounced neutron leakage effect and positive near the core midplane from the predominantly spectral effect. The cumulative fuel assembly void worth does not become positive until about the upper 30% of the active fuel region is voided.

It is important to realize that, for the ALMR, the ATWS events discussed above do not normally lead to fuel pin failures and core voiding and, in fact, have very large margins to severe core damage. The probability of a severe core accident is less than 2.5×10^{-7} per plant year or, equivalently, less than 2.5×10^{-8} per reactor year.

The large (>0.9g) earthquake contributes 86% of this severe core damage risk; all internal initiators contribute only 14%. The probability of an ATWS event resulting in a significant positive reactivity addition and severe core damage is less than 0.01; this combined with the probability of failure to scram of less than 4×10^{-7} per reactor year results in an ATWS severe core damage probability of less than 4×10^{-9} per reactor year. This probability is so low that severe core damage from ATWS events can clearly be treated as "residual risk".

In addition, the primary system is completely sealed during power operations, and provides a strong barrier designed to contain severe core disruptive events without leakage. Preliminary assessments indicate that the core support and primary systems structures can contain energetic events producing several hundred megajoules of mechanical energy and gross core melting. The expected energy release from a severe event in a small, metal-fueled core is not more than a few tens of megajoules, and metal fuel is expected to resolidify under most conditions in a porous, coolable form.

The containment surrounds the primary system and represents a separate pressure retaining, essentially leak tight boundary. It consists of the containment vessel which backs up the primary vessel, and the upper containment dome which backs up the reactor head closure. Even though the primary system boundary is designed to contain severe core disruptive events, the design basis for the containment is such an event with the simultaneous breach of the primary system boundary. For the containment design basis it is assumed that the cover gas escapes through the reactor head closure, carrying with it 100% of the fission gases and lesser fractions of the other radioactive materials, and that air enters the primary system, resulting in a pool fire which consumes all the oxygen available within the containment dome. The resulting pressures and temperatures are within the containment dome design levels of 1.7 bar (25 psig) overpressure and 370°C (700°F) temperature. The less than 1% volume per day design leak rate results in radiation dose levels less than 1 REM whole body at the plant boundary for the first 36 hours.

It has been concluded that the probability of core voiding in the ALMR is extremely low and the consequences, if core voiding were to occur, are acceptable because the maximum energetics produced are contained within the primary boundary and the core debris is subcritical and coolable. Additionally, a complete, separate containment boundary is provided around the primary boundary. Thus, the positive void worth of \$5.15 in the ALMR reference core is acceptable from the standpoint of public risk.

SUMMARY AND CONCLUSIONS

The ALMR is being designed to accommodate passively with little damage very unlikely events, down to a probability of occurrence less than 1 x 10^{-6} per plant year. The core design assures strong negative reactivity feedbacks such that anticipated transients without scram are safely accommodated. The primary boundary (reactor vessel and closure) is designed to accommodate the largest possible core disruptive accident without breach and radiological release; this approach fully accommodates, without performance or economic penalty, the \$5 positive void worth of the fuel plus blanket assemblies. The enhanced safety capabilities, combined with small size and cost, make possible a full scale prototype safety test and demonstration program. Licensing and public acceptance of this plant concept are thus significantly enhanced by the capabilities produced by designing for passive safety.

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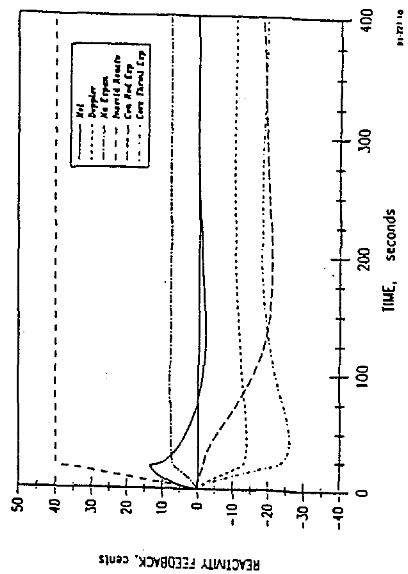

Figure 5. ALL RODS WITHDRAWAL WITHOUT SCRAM WITH FORCED COOLING: REACTIVITY FEEDBACKS

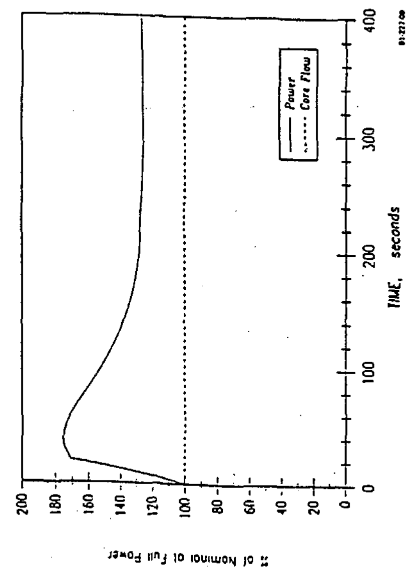

Figure 4. ALL RODS WITHDRAWAL WITHOUT SCRAM WITH FORCED COOLING: POWER AND FLOW

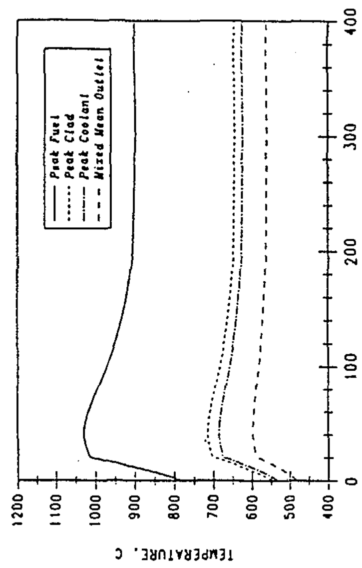

Figure 6. ALL RODS WITHDRAWAL WITHOUT SCRAM WITH FORCED COOLING: CORE TEMPERATURES

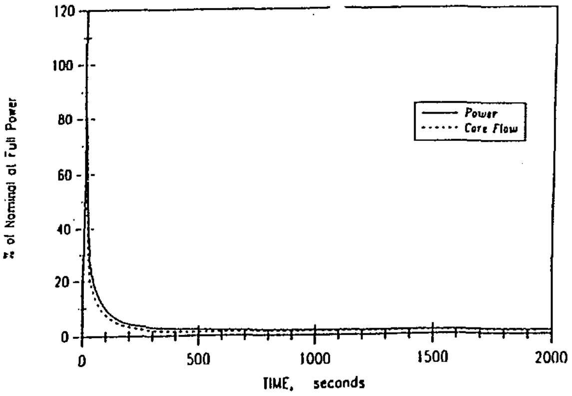

Figure 7. LOSS OF FLOW AND LOSS OF HEAT SINK WITHOUT SCRAM: POWER AND FLOW

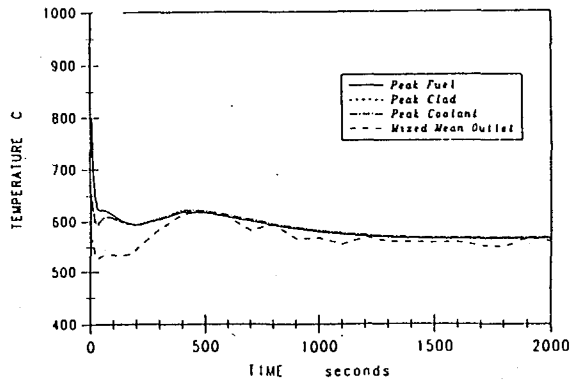

Figure 8. LOSS OF FLOW AND LOSS OF HEAT SINK WITHOUT SCRAM: NEAR-TERM CORE TEMPERATURES

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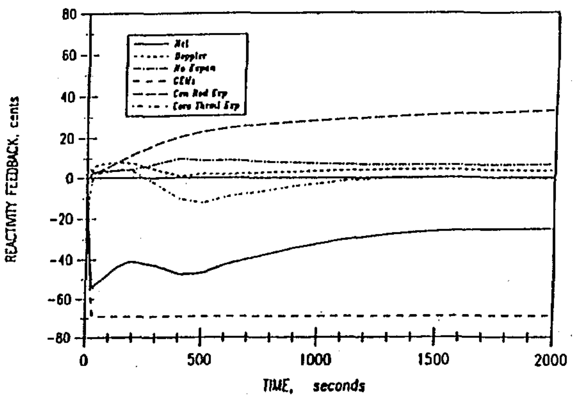

Figure 9. LOSS OF FLOW AND LOSS OF HEAT SINK WITHOUT SCRAM: REACTIVITY FEEDBACKS

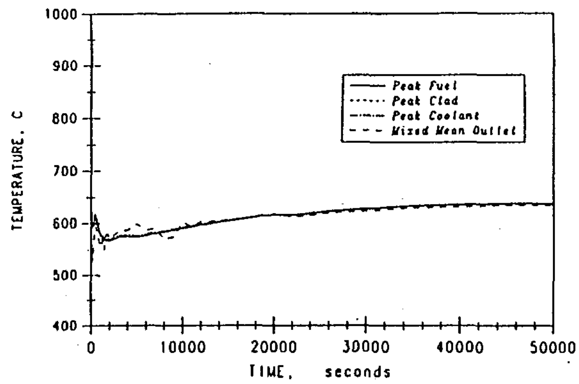

Figure 10. LOSS OF FLOW AND LOSS OF HEAT SINK WITHOUT SCRAM: LONG-TERM CORE TEMPERATURES

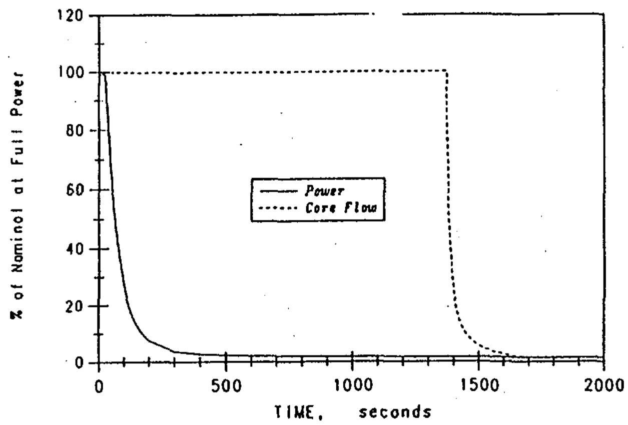

Figure 11. LOSS OF HEAT SINK WITHOUT SCRAM: POWER AND FLOW

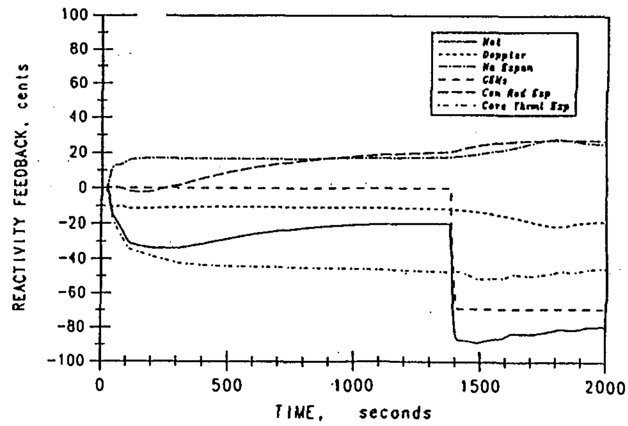

Figure 12. LOSS OF HEAT SINK WITHOUT SCRAM: REACTIVITY FEEDBACKS

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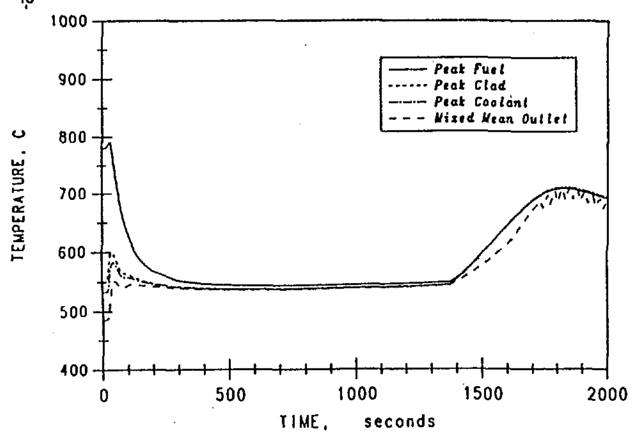

Figure 13. LOSS OF HEAT SINK WITHOUT SCRAM: NEAR-TERM CORE TEMPERATURES

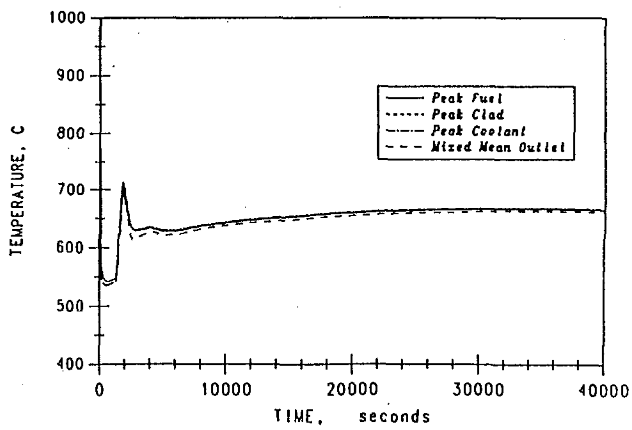

Figure 14. LOSS OF HEAT SINK WITHOUT SCRAM: LONG-TERM CORE TEMPERATURES

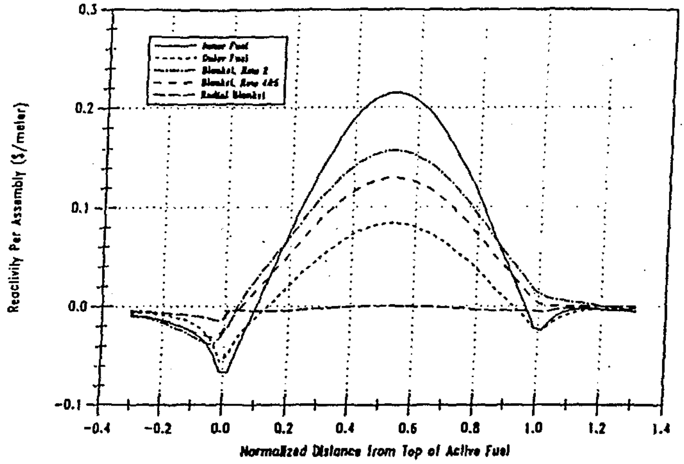

Figure 15. AXIAL VOID WORTH DISTRIBUTION FOR ALMR FUEL AND BLANKET ASSEMBLIES

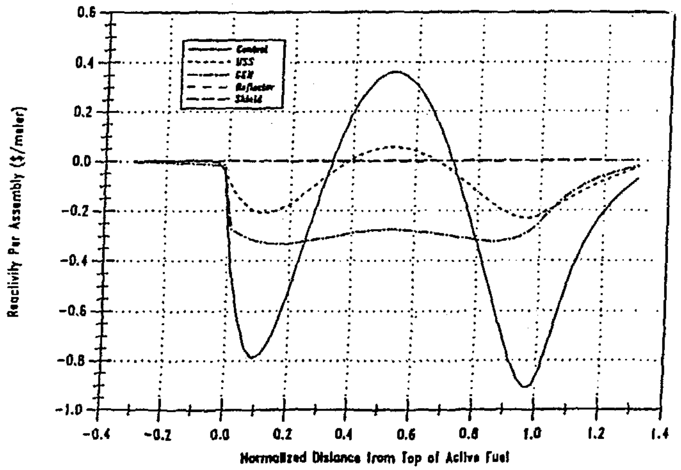

Figure 16. AXIAL VOID WORTH DISTRIBUTION FOR ALMR NON-FUEL AND BLANKET ASSEMBLIES