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SAFETY ASPECTS OF LMR CORE DESIGN*

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ABSTRACT

Features contributing to increased safety margins in liquid metal-cooled reactor (LMR) design are identified. The technical basis is presented for the performance of a pool-type reactor system with an advanced metallic alloy fuel in unprotected accidents. Results are presented from analyses of anticipated transients without scram, including loss-of-flow (LOF), transient overpower (TOP), and loss-of-heat-sink (LOHS) accidents.

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INTRODUCTION

The goals of the fast reactor program sponsored by the U.S. Department of Energy are focused on development of a Liquid Metal Reactor (LMR) system in which investment costs are minimized and safety margins are maximized (1). The emphasis on economics and safety is in response to the realization that commercial acceptance of the LMR reactor concept depends on its adaptability to near-term marketing and licensing requirements. The twin goals of improved economics and enhanced safety are being accomplished in large part by the adoption of a safety philosophy which emphasizes utilization of natural, or inherent thermal, mechanical, hydraulic, and neutronic responses to normal and off-normal operating conditions. This philosophy, and the design choices which implement it, provides enhanced safety margins and permits reduction of the number and complexity of engineered, safety-grade systems, leading to a corresponding reduction in plant investment costs.

The generic LMR design is particularly amenable to the inherent safety philosophy, due to its superior performance characteristics. The LMR's coolant, molten sodium, operates at near-atmospheric pressures, with a margin to boiling greater than 400 K (700°F), eliminating the need for thick-walled pressure vessels. Liquid sodium exhibits high thermal conductivity and specific heat capacity, and enables an LMR to operate at decay heat levels under natural circulation, without the need for forced flow. The high breeding gain in an LMR reduces the burnup cycle reactivity swing, and the required external control reactivity. In over-power conditions, the prompt negative Doppler reactivity feedback limits the power rise. All of these inherent mechanisms contribute to the superior safety performance of an LMR.

While the concept of inherent safety is not new,* recent developments in the LMR program have high-lighted the inherent safety performance potential of advanced LMR's. In particular, the Integral Fast Reactor (IFR) program at Argonne National Laboratory (2) has pointed out the superior inherent safety and economic potential of pool-type LMR's with advanced metallic fuel designs. In the pool-type LMR design, all primary system components (including the core, pumps, and intermediate heat exchangers) are submerged in liquid sodium in a single reactor vessel, with no pipes connecting components other than the pump outlet to the core inlet. This assures that, in the unlikely event of a severe accident, the core will remain submerged in liquid sodium, and natural circulation flow paths will be maintained. Furthermore, the large heat capacity of the pool provides long times for corrective operator action in the event of decay heat removal system failure.

In an LMR, metallic fuel provides enhanced safety performance due to its high thermal conductivity. At normal operating conditions, the high conductivity of metallic fuel results in a relatively shallow radial temperature gradient in the fuel pin. In any accident situation, this minimal temperature gradient yields a reduced positive Doppler reactivity feedback to be compensated during the power reduction to decay heat levels. In protected transients, external control requirements are reduced, and in unprotected transients, system temperature rises are reduced.

Because of their traditional prominence in fast reactor safety, unprotected accidents, or anticipated transients without scram (ATWS), have received considerable attention within the inherent safety framework. Following discussions of 1) inherent safety performance characteristics, 2) metallic and oxide fuel properties, and 3)

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*Earlier this year, a series of inherent safety integral proof tests were successfully conducted in the EBR-II reactor at Argonne National Laboratory's Idaho site. The EBR-II reactor plant design was initially proposed to the U.S. Atomic Energy Commission in 1953.

ATWS analysis methods, the response of a representative metallic-fuelled reactor design to unprotected loss-of-flow, transient overpower, and loss-of-heat sink accident initiators will be presented and discussed.

LMR INHERENT SAFETY PERFORMANCE CHARACTERISTICS

While they are nominally not considered to be part of the reactor design basis, the consequences of unprotected (i.e. without scram) accidents have traditionally played a very significant role in the evaluation of safety performance and the determination of containment requirements for licensability of LMR's. This comes about due to the potential in an LMR for recriticality events following core disruption, resulting in energy releases which could challenge containment integrity and present some measure of risk to the public health. Concern over the potential consequences of unprotected accidents has led LMR designers to develop comprehensive, redundant, engineered safety systems, with the reliability of these systems assured by design to reduce the likelihood of any unprotected accident to an acceptably low level. However, because every engineered system has some residual failure probability, in LMR safety this has led to a continuing, open-ended dialog between LMR safety analysts and regulators.

The essence of the inherent safety idea is to provide for intrinsic LMR performance characteristics which maintain the balance between reactor cooling capability and power production and prevent core disruption, especially in instances when engineered safety systems have failed. These response characteristics must therefore be based on the inherent thermal, mechanical, hydraulic, and neutronic reactor system properties, which can be determined by the choice and arrangement of reactor materials. In the full spectrum of unprotected accidents, three specific initiators have emerged to serve as quantifiers of safety margins. They are: 1) the loss-of-flow (LOF) accident, in which power to the coolant pumps is lost, 2) the transient overpower (TOP) accident, in which a single, inserted control rod is withdrawn, and 3) the loss-of-heat-sink (LOHS) accident, in which feedwater supply to the steam generators is lost. For all three initiators, it is also assumed that the plant safety system fails to insert the shutdown control rods. The key to successful prevention of core disruption under these conditions is the provision in the design for reactor performance characteristics which 1) limit mechanisms leading to reactor damage, and 2) promote mechanisms responding to the upset condition and acting to restore the reactor power production/cooling balance. An example of the first is the minimization of individual control rod worths, to limit the inserted reactivity in the TOP accident. For economy, it is desirable to limit the number of control rods, so the objective is to reduce the total burnup reactivity swing, without introducing burnable poisons which degrade fuel cycle economics. This is achieved in an LMR by maximizing the breeding potential and conversion of fertile uranium into fissile plutonium.

An example of a mitigating mechanism in ATWS accidents is core radial expansion. As the outlet coolant temperature rises during an ATWS transient, heat is transferred to the above-core load pads, which expand and increase the mean core diameter. The negative reactivity effect associated with load pad heating (or with core support grid expansion for cases with inlet coolant heating) acts to reduce the reactor power level and restore equilibrium between the power and the heat rejection rate, at an elevated system temperature. Another mechanism which can act to restore equilibrium is differential thermal expansion of control rod drives and the core support structure to yield a net insertion of the control rods.

For load pad thermal expansion to be effective, it is necessary that the radial core restraint system be configured to provide contact at the load pad plane during normal operation. In addition, provision must be made to allow for thermal expansion during the transient. A low-tension core restraint system allows for duct expansion and load pad growth as the coolant temperature rises, providing a negative reactivity feedback. To enhance differential control rod expansion, core outlet coolant flow may be ducted around rod drive-lines, and core-support members can be located in regions that heat relatively slowing during an accident. This assures that the differential movement of control rods early in the transient acts to insert control material into the reactor.

Because of the time required for core outlet sodium to travel to the control rod drive elevation and for heat to be transferred from the hot sodium into the subassembly load pads, it may be necessary to provide for elongation of the natural primary pump flow coastdown to avoid power-to-flow mismatches resulting in near-term coolant boiling. Specific designs might employ flywheels geared to the pump shaft or battery-fed power supplies on the pump pony motor.

Radial core expansion and control rod drive elongation provide the overall negative reactivity feedback to lower the reactor power during an unprotected loss-of-flow event. As the accident proceeds other reactivity effects that must be considered are fuel Doppler feedback, coolant density feedback, and fuel thermal expansion. As the power decreases, the fuel temperatures will drop, yielding a prompt positive reactivity effect. The heatup of the coolant causes a corresponding coolant density decrease, adding a positive reactivity mechanism. Finally, the chilling fuel will contract, and the fuel density increase will add positive reactivity.

In a transient overpower event, rod withdrawal introduces positive reactivity, which leads to fuel heating and prompt negative Doppler feedback. As the fuel expands, the density decrease also yields negative reactivity. Coolant heating leads to load pad and control rod driveline expansion (negative feedback) and coolant density reduction (positive feedback).

In the loss of heat sink accident, the temperature of the core inlet coolant rises, heating the core support structure and spreading the core radially, reducing the reactivity and the core power level.

In all three ATWS accidents, the key to avoidance of short-term core disruption is to maintain the coolant outlet temperature below its boiling point. At normal operating conditions, the core inlet temperature is around 600 K (620°F), and the average coolant temperature rise through the core is around 150 K (270°F). To avoid coolant boiling, the transient, normalized power-to-flow ratio must be kept below about 4 in order to keep core-average coolant temperatures below the boiling point of sodium at around 1200 K (1700°F). In the long term, the overall negative feedback will tend to bring the reactor power into equilibrium with the available heat rejection rate, and the system will approach an asymptotic temperature distribution. To avoid core disruption in the long term, it is necessary that the peak asymptotic temperatures in strategic components (reactor vessel, core support structure, fuel cladding) be maintained below levels at which creep could cause failures. Avoidance of both short- and long-term core disruption in ATWS events depends on 1) providing sufficient negative reactivity feedback to overcome the power-to-cooling mismatch and return the system to equilibrium at slightly elevated system temperatures, or alternately 2), reducing the positive reactivity feedback components acting to resist the transition to system equilibrium. It is by this second mechanism that metallic fuel provides superior inherent safety performance in ATWS events, due to its thermal and neutronic properties.

METALLIC FUEL PROPERTIES

Early LMR designs employed metallic fuel designs because a) metallic fuel is chemically compatible with sodium, b) metallic fuel offers superior thermal and neutronic performance, and c) metallic fuels were well known and understood at the time (3). When demands for higher burnups and coolant outlet temperatures were applied, emphasis on metallic fuel lessened, and ceramic fuels, particularly (U, Pu) O_2, became favored. In more recent times, LMR system designs have featured reduced coolant temperatures, and metallic fuel designs have been developed which are capable of reliable performance at high burnups. Combined with the inherent safety advantages of metallic fuel, these factors have prompted a renewed interest in metallic-fueled reactor designs, and have served as the technical basis for the development of the Integral Fast Reactor concept at ANL (2).

The IFR metallic fuel design is an advanced concept developed as a result of experience with metallic fuels in EBR-II and other reactors (3). In the IFR fuel design, the fuel is cast as a uranium-plutonium-zirconium alloy. Some of the properties of the IFR metallic fuel are compared with a typical oxide fuel in Table I. As the data in Table I show, metallic fuel is denser than oxide, with a thermal conductivity higher by an order of magnitude, and a lower specific heat. The thermal expansion coefficient of metallic fuel is higher than oxide, and the melting point is much lower. To allow for fuel swelling upon irradiation, the IFR metallic fuel design features an as-fabricated smear density of 75%. Since the U-Pu-Zr alloy is chemically compatible with sodium, the fuel rod is submerged in liquid sodium inside the cladding. The bond-gap sodium, together with the high thermal conductivity, give the metallic fuel pin an order-of-magnitude faster thermal response time compared to the lower conductivity, gas-bonded oxide fuel.

The high thermal conductance provided by the bond-gap sodium lowers the fuel surface temperature of metallic fuel compared to oxide fuel, and due to its higher thermal conductivity, metallic fuel exhibits relatively small radial temperature gradients. Metallic fuel therefore operates at much lower temperatures than oxide fuel, and the amount of stored heat at normal operating conditions is reduced correspondingly.

Table I. LMR Fuel Properties

	Oxide	Metal
Nominal Composition	UO_2 - 20% PuO_2	U - 15% Pu - 10% Zr
Density, g/cc	11.0	15.8
Thermal Conductivity, W/cm-°C	0.023	0.22
Specific Heat, J/g-°C	0.34	0.19
Thermal Expansion Coefficient, $°C^{-1}$	1.1×10^{-5}	2.0×10^{-5}
Melting Point, °C	2750	1106
Fuel Pin Thermal Time Constant, sec.	~3	~0.3

ATWS ANALYSIS METHODS

The primary computational tool used for the analysis of ATWS events in the IFR program at ANL is the SASSYS LMR systems analysis code (4). The SASSYS code has been designed to analyze a wide range of reactor and balance-of-plant transients, from normal operational transients through severe transients leading to coolant boiling. In addition to a point kinetics neutronics formulation with first order perturbation theory reactivity feedbacks, SASSYS performs a detailed thermal hydraulic analysis of the reactor core, inlet and outlet coolant plena, primary and intermediate heat transport systems, intermediate heat exchangers, steam generators, and decay heat removal systems. Recently, a simulation of the plant control and protection systems has been added. SASSYS is capable of analyzing both loop and pool designs, with any arrangement of components. With its efficient numerical methods and data management, SASSYS is fast running, usually faster than real time on typical mainframe computers.

The SASSYS core fuel pin heat transfer model uses a two-dimensional (axial/radial) spatial representation for a pin and its associated coolant and duct wall. A single, average pin is used to represent all the pins in a subassembly, and like subassemblies are grouped into channels. Multiple channels, as many as one per subassembly, are used to represent the entire core consisting of fuel, blanket, and reflector subassemblies. The SASSYS core hydraulics model is a one-dimensional (axial) channel-wise treatment, with all channels hydraulically coupled to the inlet and outlet plena.

For the primary and intermediate loop thermal hydraulic models, SASSYS uses a generalized geometry featuring volumes filled with compressible liquid or gas connected by segments. The volumes are assumed to be well-mixed, while heat flux and temperature distributions are allowed along the length of the segments. Heat conduction between components is treated, as are radiative heat losses at the reactor vessel wall. A detailed steam generator model is available for rapid transients, and an air dump heat exchanger is modeled for decay heat removal studies.

For ATWS event simulation, SASSYS computes reactivity feedbacks from the fuel Doppler effect, coolant density changes, fuel axial expansion (or contraction), core radial expansion, and control rod motions due to differential thermal expansion of control rod drivelines and the core support structure. A separate accounting is made of core fission power and channel-by-channel decay heat.

LMR ATWS ANALYSIS

In order to quantify the inherent safety margins provided by the IFR design features, the SASSYS computer code was applied in the analysis of unprotected loss-of-flow (LOF), transient overpower (TOP), and loss-of-heat-sink (LOHS) accidents in a representative metallic-fueled LMR design. The plant analyzed here is rated at 900 MWt, and has a pool-type primary system configuration with two primary coolant pumps and four intermediate heat exchangers. The core layout is heterogeneous, with core-internal breeder subassemblies. Table II tabulates some of the important safety performance characteristics of the reactor.

Table II. Metallic-Fuelled Reactor Safety Performance Characteristics

	Driver Fuel	Breeder Fuel
Linear Power Rating, kW/ft	12.4	11.3
Doppler Coefficient, T dk/dT	1.2×10^{-3}	-2.0×10^{-3}
Coolant Void Reactivity, $	3.2	2.3
Core Expansion Coefficient, $/cm		
Axial	-0.61	
Radial	-1.63	
Single Primary Rod Runout Worth, $	0.08	

For the LOF accident, the assumed initiator is loss of power to the primary and intermediate coolant pumps without scram. In this plant design, the primary pump coastdown produces an initial flow halving time of six seconds. As the flow decreases, the core outlet temperature rises. The heating and expansion of the above-core load pads spreads the core radially, causing negative reactivity feedback that reduces the reactor power. As the power falls, the coolant outlet temperature peaks and then also begins to decrease. Following stoppage of the primary pumps, natural circulation is established, and a transition to a near-equilibrium state is made. Table III summarizes reactor conditions at the time of the coolant temperature peak and in the longer term, near-equilibrium state. The analysis shows that coolant boiling is avoided with substantial margin in the short-term transient. In the long term, system temperatures remain below levels at which load-stress-induced creep could result in structural failures. The long-term inherent safety margin is provided mainly by the relatively small positive Doppler reactivity feedback, which comes about due to the high thermal conductivity of the metallic fuel.

Table III. Unprotected LOF Results

	Peak P/F	Asymptotic
Time, sec	45	800
Coolant Temperature, K Peak S/A	1115	915
Non-Boiling Margin, K	95	290
Power, P/P_0	0.29	0.045
Flow, F/F_0	0.11	0.031
Reactivities, $		
Coolant	+0.15	+0.06
Core Expansion		
Axial	+0.00	+0.08
Radial	-0.41	-0.15
Doppler	-0.04	+0.04
CRD Expansion	-0.06	-0.04
Net	-0.36	-0.01

For the TOP accident, the assumed initiator is an uncompensated withdrawal of a single, maximum-worth control rod. As shown in Table IV, for this metallic-fueled core with its high breeding gain and low cycle burnup reactivity swing, this amounts to an insertion of eight cents of reactivity. In the resulting transient, the reactor power rises to 12% above nominal, followed by a very slight heating of the coolant which introduces sufficient negative reactivity to return the reactor power gradually to equilibrium with the assumed nominal heat rejection at the steam generators. These results show that the low control rod worth made possible by the high breeding gain in the metallic core results in only slight over-temperature conditions in the single rod TOP ATWS accident.

For the LOHS accident, it is assumed that feedwater supply to the steam generators is lost, yielding a gradual heating of the intermediate and primary coolant systems and an increase in the core inlet temperature. Heating of the core support grid spreads the core radially, introducing negative reactivity which reduces the

Table IV. Unprotected TOP Results

	Peak P	Asymptotic
Time, sec	75	800
Peak S/A Coolant Temperature, K	847	850
Non-Boiling Margin, K	353	350
Power, P/P_o	1.12	1.01
Reactivities, $		
Coolant	+0.01	+0.03
Core Expansion		
Axial	-0.02	-0.02
Radial	-0.04	-0.06
Doppler	-0.02	-0.02
CRD Expansion	-0.01	-0.01
CRD Withdrawal	+0.08	+0.08
Net	+0.01	-0.00

reactor power. In the long term, the reactor power equilibrates to any available heat sink with the inlet temperature elevated above the initial state. Table V summarizes conditions at the initial state and at 800 seconds into the transient for the case of a total loss of heat sink. The results show that for the metallic-fueled design, the negative reactivity feedbacks reduce the reactor power as the core inlet temperature rises, with peak temperatures only slightly elevated above nominal conditions.

SUMMARY

An increased emphasis on economics and safety has focused renewed attention on metallic fuel for liquid metal-cooled reactors.[1] The superior thermal and neutronic performance characteristics of metallic fuel provide inherent mechanisms for improved safety margins, and permit the reduction or elimination of costly engineered safety systems. In particular, the performance of metallic-fuelled reactors during

Table V. Unprotected LOHS Results

	Initial	Asymptotic
Time, sec	0	800
Peak S/A Coolant Temperature, K	824	844
Core Inlet Temperature, K	630	831
Non-Boiling Margin, K	376	356
Power, P/P_o	1.0	0.07
Reactivities		
Coolant	0	+0.16
Core Expansion		
Axial	0	+0.04
Radial	0	-0.18
Doppler	0	-0.03
CRD Expansion	0	-0.01
Net	0	-0.02

anticipated transients without scram (ATWS) reduces accident consequences to the level of accommodation within a slight extension of normal design basis margins. This eliminates the need for operator intervention or automatic activation of engineered safety systems to prevent core disruption or system structural failures in unprotected transients.

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