Physics and Safety Features of the AMBIDEXTER Nuclear Energy Complex

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ABSTRACT – Adopting the closed-fuel-cycle, multiplex-energy-conversion, and regenerative-radwaste-management systems requirements, the concept of AMBIDEXTER* Nuclear Energy Complex has been scrutinized. The key elements and attributes of the design embody the nuclear self-sustainability of the molten (Th-U-Pu)F_4 fuel cycle system, integrated with pyro-chemical cleanup processings.

This paper intends to explore its eligibility for a Generation IV nuclear power system, by means of simulating the physics and safety behaviors of the 250 MW_{th} proto-type during various design and off-design conditions. Based upon the lumped parameter approximation method, AMBISM, a dynamic model of the fluid-fueled reactor system, the 3-loop heat/energy transport circuits, and the on-line material/radiation conversion circuits of the AMBIDEXTER-NEC were developed and incorporated into the MATLAB/Simulink package.

The physics design bases were established under the conditions of that long-term reactivity adjustments should be achieved by varying material concentrations in the fuel salt stream during operation and of that short-term reactivity perturbations due to temperature and load changes should be digested as much as possible without regulating system interventions. In conjunction with bounding conditions for evaluating the self-regulation criteria, MATLAB/AMBISM simulations were performed for those initiating events associated with mismatch in balance between the heat generation and removal, and classified into anticipated operational occurrences or emergency conditions, but without scram.

Results of the simulations demonstrate that, even for accidents with moderate occurrence frequencies, the inherent safety characteristics of the AMBIDEXTER reactor system could limit transients not to overrun to an unsafe condition and return the reactor to another safe state in a short period. Upon the detailed physics and thermal-hydraulic analyses for broader spectrum of events, this self-regulation potential should be tuned to the quantified design requirements for the basic design of the AMBIDEXTER-NEC.

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* Advanced Molten-salt Break-even Inherently-safe Dual-missioning EXperiment and TEst Reactor

I. Introduction

Adopting the closed-fuel-cycle, multiplex-energy-conversion, and regenerative-radwaste-management systems requirements, the concept of AMBIDEXTER Nuclear Energy Complex has been scrutinized.[1)] It does require neither external feeding nor bleeding of fissile materials to maintain the reactor operation. Availability of high temperature heat produced from and carried by molten salt fuel makes it possible to expand design choices for thermodynamic cycles combined. Fission products separated from fuel stream at the on-line reprocessing facilities can be reused for radio-isotope or radiation sources.

The key elements and attributes of the design embody the nuclear self-sustainability of the molten (Th-U-Pu)F_4 fuel cycle system, integrated with pyro-chemical cleanup processings. For the given fractions of the LiF and BeF_2 compound as the base material for fuel salt, the composition fractions of thorium, uranium, and plutonium were estimated subject to the condition that the reactor system should maintain both the effective multiplication factor and the conversion ratio being equal to unity with minimum margin for operational loss, over the plant life. Optimization of this self-sustainability requirement can be tuned by introducing differential reprocessing rates for the chemical groups.

According to recently identified Generation IV technology goals, the sustainability, safety & reliability, and economics should be considered as the direct measures of the Generation IV qualification, that are very similar to the general design requirements of the AMBIDEXTER-NEC already established. This paper, therefore, intends to demonstrate how to achieve its eligibility, mainly in safety aspects, for a Generation IV nuclear energy system, by means of simulating its physics and safety behaviors of the 250 MW_{th} proto-type during various design and off-design conditions.

Based upon the lumped parameter approximation method, MATLAB/AMBISM, a dynamic simulation model for the fluid-fueled reactor system, the 3-loop heat/energy transport circuits, and the on-line material/radiation conversion circuits of the AMBIDEXTER-NEC was developed for and incorporated into the MATLAB/Simulink package.[2)] Its neutronics model represents multi-region core and includes circuit-transit effects for the delayed neutron precursors. Multi-node heat transfer models for major equipments are derived by assuming that thermodynamic properties, except the heat transfer coefficients, of heat transfer media are invariant to operating conditions.

As the objective of this study was focused to investigating the self-regulation capabilities of the AMBIDEXTER, simulation scenarios interested were initiating events associated with mismatch in balance between the heat generation and removal, and classified into anticipated operational occurrences or emergency conditions, for example, loss of fluid flows, abnormal insertions of positive reactivity, and failures in a piping network. Transients forcedly terminated by a reactor scram were not considered in the models.

II. Nuclear Characteristics of the AMBIDEXTER-NEC

II.1 System Descriptions

As shown in Fig. 1, the AMBIDEXTER-NEC is essentially comprised of 3 system groups, the reactor systems, the energy/heat transport systems, and the material/radiation transport systems. The $^{7}LiF\text{-}BeF_2\text{-}(Th, U, Pu)F_4$ fuel salt in liquid phase, is endowed with dual functions of fission source and heat carrier during flowing up with designed velocities through vertical fuel channels in graphite moderated core.

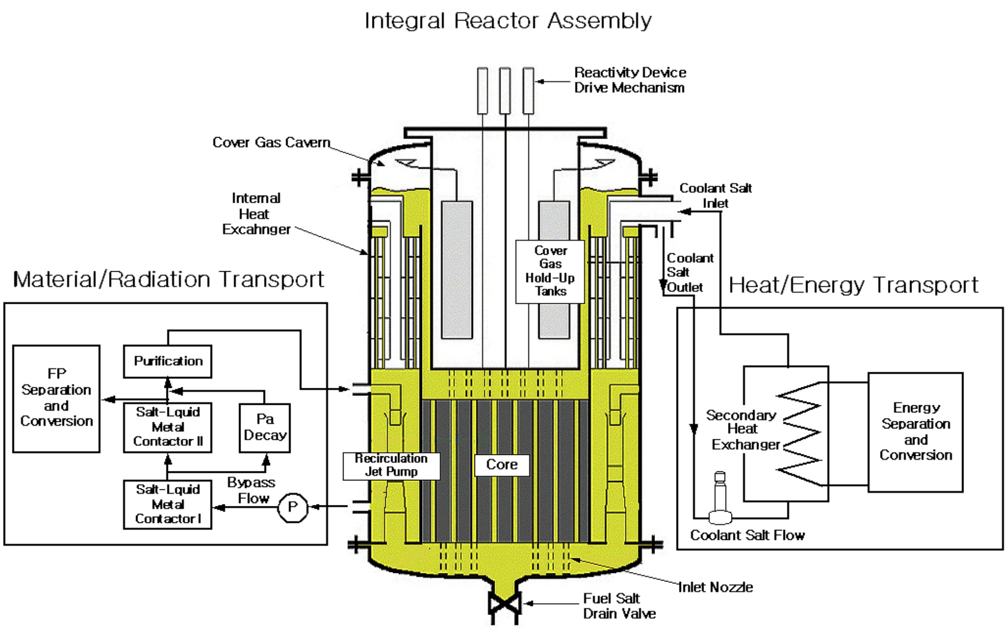

Figure 1. Schematic Diagram of the AMBIDEXTER-NEC

Radioactively dirty fuel salt is bounded on tube walls of the intermediate heat exchangers, placed above the reactor core and inside the reactor vessel, by countercurrent flow of clean coolant salt. The intermediate heat transport loop, then, separates the low-pressure reactor system from the high-pressure steam cycle system. In spite of its low viscosity, high specific heat, and negligible vapor pressure at operating temperature, as the fluoride salt compounds have relatively high liquidus temperature and material cost, the AMBIDEXTER heat transport systems is configured to reduce required piping volume and active components.

The material/radiation transport systems, interconnected with the reactor system via the bypass flow line at the intermediate heat exchanger exits, have function of continuously separating fission and activation products dissolved in fuel salt, among which the Pa and Np isotopes are converted into the U and Pu, respectively while stored in the decay tank. After supplemented with fresh thorium and natural uranium, the converted U and Pu return to the inlet of the reactor.

Important attributes of the AMBIDEXTER-NEC design are depicted in Fig. 2. None of the front- and back-end fuel cycle processes handling non-natural heavy elements should be allowed to cross the plant site boundary for lifetime operation of the equilibrium core. Under this circumstance, required amounts of fissile loadings for steady power operations can be internally managed when relying on the break-even conversion ratio criterion, but not on external supply. On-site storages of the HLW, after partitioned, processed and stabilized at the material/radiation transport systems, can be designed to facilitate necessary provisions for radiation and radioactive material sources. This ideally closed fuel cycle system is unique for the AMBIDEXTER concept.

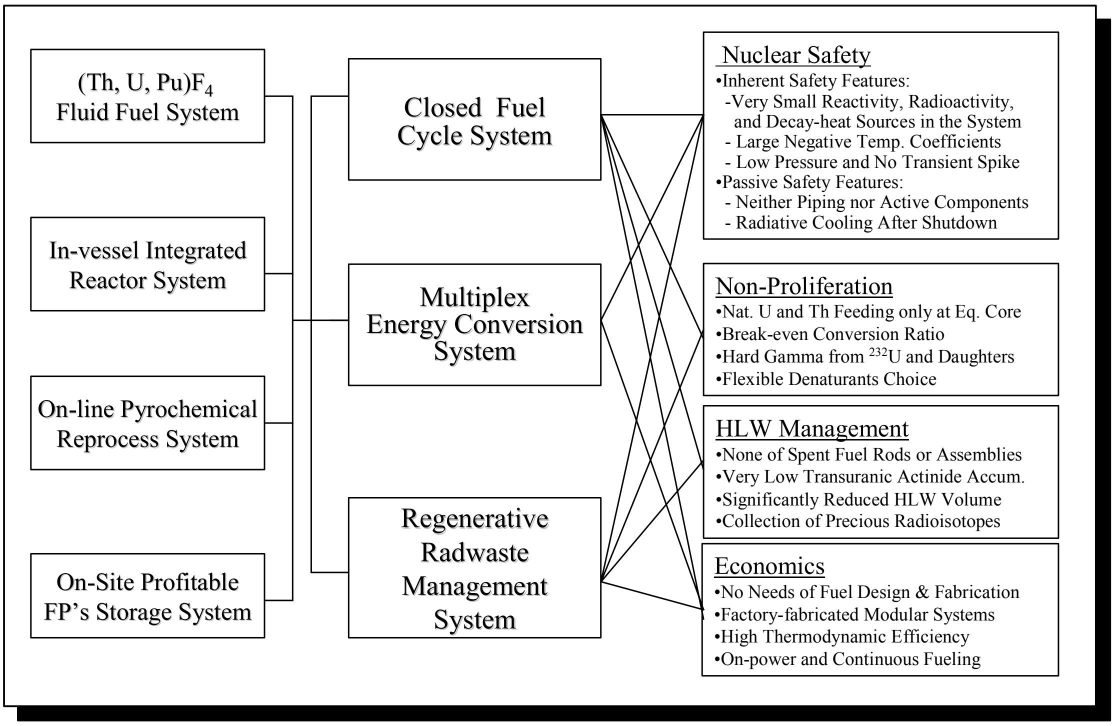

Figure 2. Attributes of the AMBIDEXTER-NEC Concept

II.2 Reactor Physics Design Characteristics

The self-sustainability of nuclear reactors is a measure of internal balances both in neutron loss and generation and in fissile consumption and production. To fulfill this stringent requirement for design, accurate control and estimation of material compositions over an operation period is important. Modified with new nuclear data sets, ORIGEN2/HELIOS/SQUID computing system was established and tested if it would be acceptable for conceptual design studies. An ORIGEN2 and HELIOS synthesis algorithm[3)] was developed to approximate the quasi-equilibrium nuclide compositions for the continuously reconditioned fuel salt system.

As the basic elements of the AMBIDEXTER reactor core, seed and blanket lattices are made of the hexagonal graphite cylinder with differently sized central channels for molten fuel salt flows. The salt-to-graphite volume ratio of the seed lattice is considered as the primary contributor to the neutron multiplication property of the core having the burn-up independent fuel composition, meanwhile that of the blanket lattice to the conversion ratio. The core is, then, constructed of three co-centric annular zones that are the inner zone with seed lattices, the outer zone with blanket lattices, and the outermost zone with reflector blocks.

Table 1 summarizes the preliminary design characteristics of AMBIDEXTER reactor for the quasi-equilibrium fuel condition. To minimize parasitic absorption loss of neutrons, ^{7}Li enrichment of ^{7}LiF in fuel salt was taken to 99.995% and the density of nuclear-grade graphite 1.87 g/cm^3. Fissile enrichment, 3.33 w/o was estimated from the total weight ratio of fissile to heavy element nuclides in the fuel salt.

Regarding the self-sustainability criteria, the k_{eff} and the conversion ratio are found to be 1.008 and 1.017, respectively, which corresponds to the explanation why the daily feed rate of total heavy elements in the table is slightly underestimated. Definitions of neutron fission and utilization factors in the table are equivalent to the conventional η and f, but extended to include contributions from non-thermal neutrons. The isothermal temperature coefficient of reactivity at operating condition is about -2.0×10^{-5} δk/k/□, which inherently provides the core with negative reactivity feedback during events inducing system temperature increases.

As heat-up by fission incline the molten fuel salt to be homogenized into a thermodynamic equilibrium via convective heat transfer during slowly flowing up in the fuel channel, the power density and form factors were approximated by assuming uniform over the cross-section area of the channel. Therefore, according to the table, the local power peaking factor was estimated as high as 2.89, but it can be said that safety concerns associated with the fuel temperature profile are no more valid.

Table 1. Design Characteristics of the AMBIDEXTER Reactor

<u>Fuel Salt</u>:	
Composition(m/o);	
$^7LiF\text{-}BeF_2\text{-}(Th, U, Pu)F_4$	
70.0-16.0-(11.0, 2.98, .01)	
Fissile Enrichment, w/o;	3.33
Total Core Loadings, kg;	29,500
Loading per Channel, kg;	
11.7(seed)/128.7(blanket)	
Daily Feeding Rates, g/d;	
175(Th)/71.5(Nat. U)	
<u>Neutronics</u>:	
k_{eff}/Conversion Ratio;	1.008/1.017
Neutron Fission Factor, η;	1.135
Neutron Utilization Factor, f;	0.882
Peak Neutron Fluxes, $\times 10^{14}$ nv;	
Thermal (≤1.0 eV)	3.70
Fast (≥50.0 kev)	3.39
Reactivity Coefficient, $\delta k/k/$□;	
Fuel Temperature	-2.30×10^{-5}
Graphite Temperature	3.02×10^{-6}
Prompt Neutron Lifetime, s;	2.589×10^{-4}

Delayed Neutron Yield, β_{eff};	0.00302
<u>Thermal-hydraulics</u>:	
Total Thermal Power, MW;	250.0
Power Density, W/(cm^3 of fuel salt);	
Peak	744.3
Average	28.1
Form Factors;	
Radial	0.547
Axial	0.632
Channel Flow Velocity, cm/s;	129.0
Inlet/Outlet Temperature, □;	
Fuel Salt in Core	1150/1300
Graphite in Core	1160/1310
Coolant Salt in IHT	850/1050
<u>Chemical Reprocessing</u>:	
Processing Rates per Core;	
Noble Gases	10 s
Noble Metal	50 s
Rare Earth & Halogen	50 d
Semi-Noble Metal	200 d
Pa, Np	3 d

II.3 Principles in Safety Design

In order to develop the reactor system concept to fulfill the safety and reliability goals for Generation IV nuclear energy systems, advantages of inherent and passive safety characteristics should be implemented in safety design. Inherent features incorporated in the AMBIDEXTER-NEC safety design are very low excess reactivity, large negative reactivity feedback, very low inventories of releasable radioactivity and decay heat sources, and low system pressure and negligible vapor pressure at operating conditions. And the reactor safety is reinforced by passive natures of the low-density decay-heat sources distributed inside piping- and valve-free reactor system, which should make it easy to design natural cooling provisions for removing decay power.

Thermodynamic and hydrodynamic properties of molten salts, and mechanical and chemical stability of structural materials have been observed not to induce marginal changes during short-term overload operations even under high neutron and radiation environments. Nevertheless, heuristics in determining the operational safety limits suggested that the uppermost temperature uniformly acceptable for the molten fuel and coolant salts, graphite block, and Hastelloy vessel would not be higher than 1350□[4]. On the other hand, the lower temperature limits for the molten salts should be greater than their liquidus points.

III. Self-Regulation Properties of the AMBIDEXTER-NEC

Combined effects of large negative temperature coefficient and broad operable temperature range of the molten salt fuel should positively contribute to simplifying control requirements for the AMBIDEXTER reactor system, if the stability of the system is ensured. To investigate the property of self-regulation responding to various perturbation insertion spectra, the system dynamic model, MATLAB/AMBISM was developed. It consists of multi-node equipment models representing neutronic and thermal-hydraulic behaviors of the reactor core, and thermal-hydraulics for the intermediate heat exchangers, supercritical steam generators and superheated steam reheaters. Time delayed effects accounting relatively slow velocities of salt flows in the equipments and piping network are also included in the models.

A simplified flow diagram of the MATLAB/AMBISM model is shown in Figure 3. The neutronics model not shown in the figure is based on the typical point kinetics equation with 6 delayed-neutron precursor equations modified to include the effects of core-resident and loop-recirculation times. The thermo- and hydro-dynamic properties of heat transfer media in the equipment models were assumed to be invariant over the operating temperature and pressure conditions. Heat transfer coefficients at interfaces were approximately expressed as proportional functions of an exponent of the momentary mass flow rate ratio.

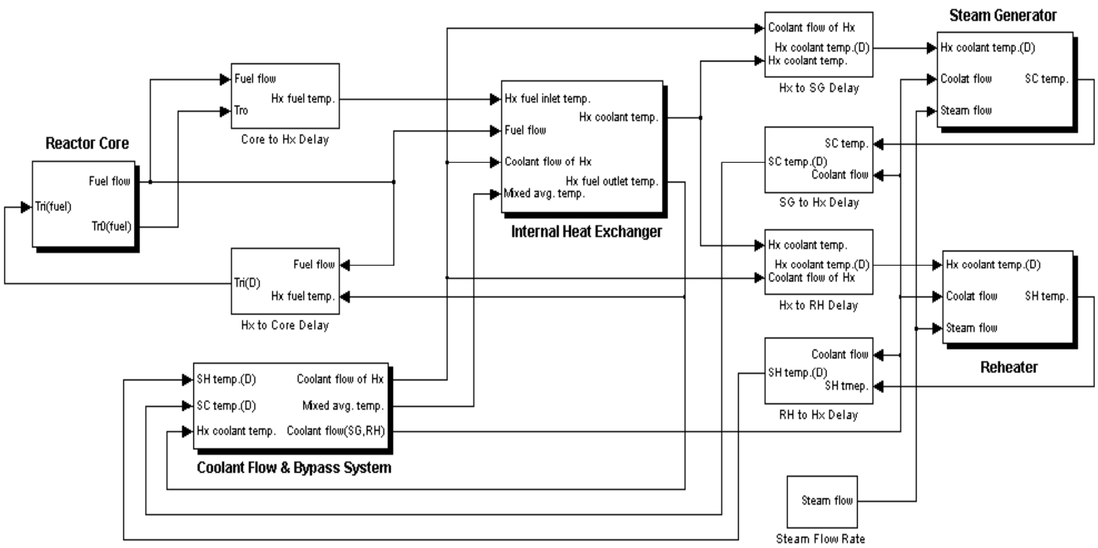

Fig. 3 MATLAB/AMBISM Model of the Reactor and Energy/Heat Transport Systems

Simulation scenarios selected were initiating events associated with mismatch in balancing between heat generation and removal capabilities but classified into anticipated operational transients or emergency conditions without scram. Only the process parameters assumed not changing during the incidents, were the feed-water temperatures inlet to steam generators and to reheaters, 700 and 650□, respectively.

III.1 Loss of Flow in Loops

The AMBIDEXTER energy/heat transport systems configure a 3-loop system in each of whose loops forcedly circulate different heat carrying fluid, for instance, fuel salt in the reactor loop, coolant salt in the intermediate loop, and supercritical steam in the steam cycle loop. Then a sudden loss of power or mechanical failures in pumps or valves would initiate an abnormal flow reduction in the corresponding loop.

For dynamic simulations of the AMBIDEXTER fluid systems, flow-induced circumstances considered were 10, 30, and 50% flow reductions either in the reactor system or in the intermediate loop, initiated at the 100% full power operation and propagated with the rate of 5%/s. To test if it is a practicable alternative for regulating the thermal power, similar flow reductions in the coolant salt loop, due to a valve failure on the steam generator by-pass line were also simulated. Additionally, 10, 30, and 40% mass flow rate losses of supercritical steam with the rate of 1%/s, were also simulated to evaluate the potential for load following operation.

Fig. 4 shows the reactor power and reactivity transients for the cases of 30% loss of different fluid flows. And, as a function of the flow rate loss, changes in peak and saturated values of the gross thermal power and the core outlet temperature of the fuel salt are illustrated in Fig. 5. Other simulation results such as the saturated power and fluid temperatures at the equipment outlet are also found in Table 2.

As shown in the figures and table, both the powers and the fuel temperatures at up to 50% loss of coolant salt flow or at up to 40% loss of steam mass flow incidents, remain below the steady full power conditions. Transients for all the cases were smoothly terminated and stabilized about 250 seconds after perturbations inserted. Thus, it can be concluded that the reactor would not be imposed at an unsafe state in case of partial loss of flow in the heat/energy transport circuits. Nevertheless, transients induced by the loss of fuel salt flow show different behaviors of that, as a slowing down of the fuel flow increases in the in-core resident time of the fuel salt, followed by increases in the fuel temperature and delayed-neutron precursor density, the reactor can maintain a steady state critical condition with this extra delayed-neutron source. Also, the simulation model of loss of fuel salt flows ignored the effects of accumulation of fission products poisoning and interruption of fissile feeding, which should have introduced additional negative reactivity into the reactor.

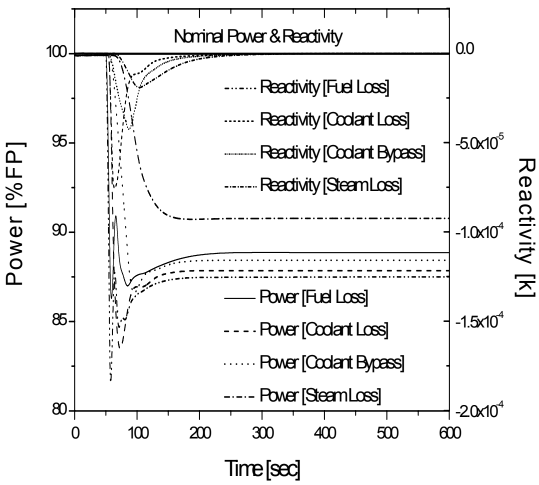

Figure 4. Power and Reactivity Changes During 30% Loss of Flow Transients

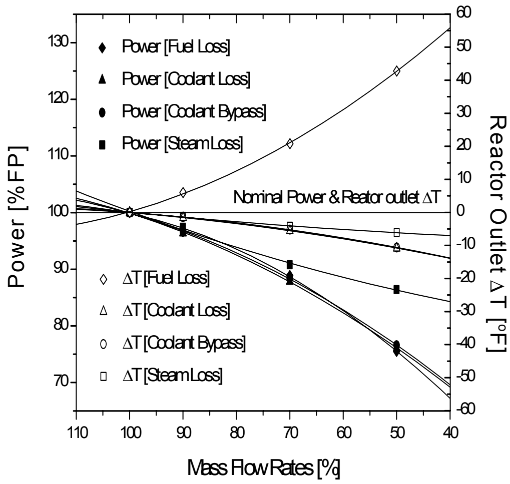

Figure 5. Maximum Power and Fuel Outlet Temperatures During Loss of Flow

Table 2. MATLAB/AMBISM Simulations for Loss of Flow Incidents

	Loss of Flow Rate [%]	Saturated Power [%FP]	Maximum Temperature [^oF]				
			Graphite	Fuel Salt Core-Outlet	Coolant Salt IHX-Outlet	Steam SG-Outlet	Steam RH-Outlet
Loss of Fuel Salt Flow	10	97.0	1316.0	1306.7	1039.9	991.4	991.3
	30	88.9	1331.5	1321.5	1011.1	966.7	966.1
	50	75.5	1355.7	1343.5	963.5	925.9	924.4
Loss of Coolant Salt Flow	10	96.4	1308.0	1299.1	1060.7	996.7	997.3
	30	87.9	1303.4	1295.2	1086.0	986.5	985.5
	50	76.1	1296.9	1289.9	1119.7	968.8	958.0
Bypass of Coolant Salt Flow	10	96.6	1308.2	1299.2	1060.1	997.5	998.0
	30	88.4	1303.7	1295.5	1079.7	988.4	987.5
	50	76.6	1297.2	1290.1	1107.9	970.9	960.1
Loss of Steam Flow	10	97.4	1308.6	1299.5	1055.5	1016.8	1014.5
	30	90.8	1305.0	1296.5	1069.0	1055.0	1045.0
	40	86.4	1302.6	1294.6	1077.9	1077.3	1061.7

III.2 Uncontrolled Reactivity Insertions

The physics design bases were established under the conditions of that long-term reactivity adjustments should be achieved by varying material concentrations in the fuel salt stream during operations and, of that short-term reactivity perturbations due to temperature and load changes should be internally absorbed as much as possible without regulating system interventions. And, thus, MATLAB/AMBISM simulations were performed to investigate the bounds of allowable reactivity against any unsafe consequences at loss of regulation events.

Uncontrolled reactivity insertion scenarios used for the simulations were 1 second ramp and step insertions of 0.03, 0.15, 0.3, and 0.5 mk reactivity. For the ramp insertion cases, Fig.6

shows time behaviors of the reactor power and reactivity during transients. And the peak power and fuel temperature reached are illustrated as a function of the total reactivity inserted in Fig. 7. The step insertion cases also show similar to but slightly higher than the ramp insertion cases.

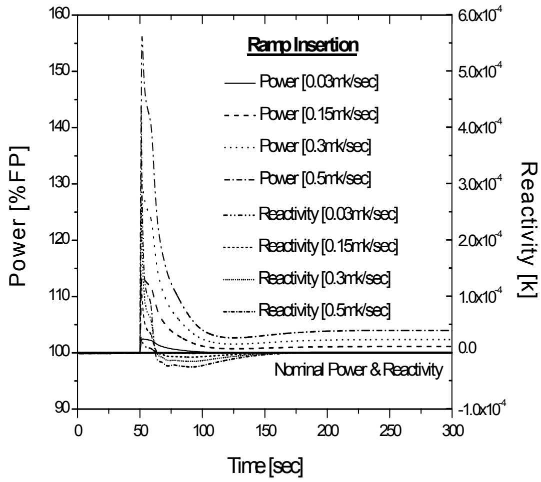

Figure 6. Power and Reactivity Changes During RIA's

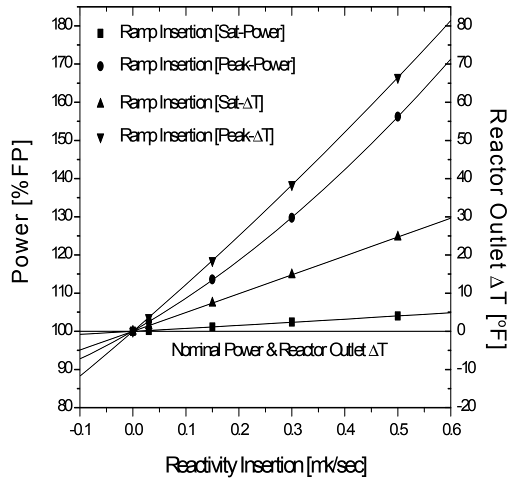

Figure 7. Maximum Power and Fuel Outlet Temperatures During RIA's

The figures show that, due to the large negative temperature coefficient of the fuel salt, up to 0.5 mk insertion of positive reactivity should not cause the fuel temperature spike of above 50□, the proposed operational safety limit, and the corresponding power peaking is 1.56. In general, after ~50 seconds overpower duration, the reactivity-induced transients of the AMBIDEXTER reactor become spontaneously saturated to a stable but slightly higher power level than 100% FP., without any regulating system interventions.

IV. Conclusion

The sustainability, safety & reliability, and economic goals of the Generation IV nuclear energy system are very similar to the general design requirements of the AMBIDEXTER-NEC already established. By means of simulating its physics and safety behaviors of the 250 MW_{th} proto-type during various design and off-design conditions, the eligibility, mainly in safety aspects, of the AMBIDEXTER for a Generation IV nuclear energy system has been demonstrated.

The MATLAB/AMBISM code, using the MATLAB/Simulink package, consists of multi-node equipment models representing neutronic and thermal-hydraulic behaviors of the reactor core, and thermal-hydraulics for the intermediate heat exchangers, supercritical steam generators and superheated steam reheaters, and piping networks. Simulation scenarios selected were initiating events associated with mismatch in balancing between heat generation and removal capabilities but classified into anticipated operational transients or emergency conditions without scram.

Results of the simulations demonstrate that, even for accidents with moderate occurrence frequencies, the inherent safety characteristics of the AMBIDEXTER reactor system could limit transients not to overrun to a unsafe condition and return the reactor to another safe condition in a short period. Upon the detailed physics and thermal-hydraulic analyses for broader spectrum of events, this self-regulation potential should be tuned to the quantified design requirements for the basic design of the AMBIDEXTER reactor and regulating systems.

In conclusion, according to the present study, the self-regulation characteristics of the AMBIDEXTER-NEC would not require a reactor regulating system designed to protect the reactor system against most of generic design-based incidents, unlikely for the conventional power reactors.

Acknowledgements

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