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TRANSIENT PERFORMANCE OF S-PRISM

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ABSTRACT

S-PRISM is an advanced Fast Reactor plant design that utilizes compact modular pool-type reactors sized to enable factory fabrication and an affordable prototype test of a single Nuclear Steam Supply System (NSSS) for design certification at minimum cost and risk. Based on the success of the previous DOE sponsored Advanced Liquid Metal Reactor (ALMR) program GE has continued to develop and assess the technical viability and economic potential of an uprated plant called SuperPRISM (S-PRISM).

S-PRISM retains all of the key ALMR design features including passive reactor shutdown, passive shutdown heat removal, and passive reactor cavity cooling that were developed under the earlier DOE program. Verification that the passive performance characteristics still function adequately to support licensing following the power uprating is necessary.

This paper presents the results of transient analyses performed to assess the ability of S-PRISM to accommodate severe accident initiator events. A unique safety capability of S-PRISM is accommodation of the "higher probability" accident initiators that led to core melt accidents in prior large LMRs. These events, the Anticipated Transients Without Scram (ATWS) events, are thus the focus of passive safety confirmation analyses. The events included in this assessment are: Unprotected Loss of Flow, Unprotected Loss of Heat Sink, Unprotected Loss of Flow and Heatsink, Unprotected Transient Overpower and Unprotected Safe Shutdown Earthquake.

The transient analyses employ the ARIES-P plant dynamics code which explicitly models the reactor and internals, core, coolant loops and heat exchangers, BOP, plant control and protection systems and decay heat removal systems. Results indicate the passive plant response to the ATWS events safely carries the plant through the accident initiator without leading to coolant boiling or fuel relocation, thus the ATWS events do not challenge public safety or investment risk.

1 Introduction

The PRISM plant concept has been uprated to improve economics and reduce excessive conservatism. Each reactor produces 1000 MWt, with two reactors combined into a power block that supplies steam to a single turbine-generator. Since a single reactor design is intended to support commercialization under differing national environments, metal and mixed oxide (MOX) fuel are accommodated within a common geometric envelope, and a breeding ratio from about 0.8 to 1.3 is supported. Based on prior studies under the USDoE-sponsored Advanced Liquid Metal Reactor (ALMR) Program, mixed oxide fuel produces the least advantageous passive safety performance. For this reason, a MOX core is selected to assess the passive safety performance of the uprated reactor.

The MOX core used in the transient analyses is shown in Figure 1. The core layout is a radially heterogenous configuration with 162 fuel and 73 internal and 60 radial blanket assemblies. Blanket shuffling is employed to reduce peaking and improve breeding. The driver fuel is 1372 mm (54 in) tall. Upper and lower axial blanket zones are each 305 mm (12 in) tall. The cycle average breeding ratio is 1.15. With MOX fuel, this is approximately the maximum breeding ratio possible within the core geometric envelope. However, with metal fuel, the breeding ratio capability is greater than 1.3 in the same reactor geometric and safety envelope.

2 Analysis Model

Transient analysis of S-PRISM uses the ARIES-P computer program. The reactor and plant models are specific to the PRISM-type arrangement of equipment, structures and control system logic.

The model consists of two reactors connected to a single BOP (turbine, generator, feedwater, steam system and support systems.) Figure 2 illustrates the overall plant arrangement modeled. First-principle models are used for physical phenomena. The plant (power block) supervisory control system (PCS), the reactor protection system (RPS), the EM pump thermal shutoff system (TSS), the reactivity control system, and all valve and pump local controllers are included in the model.

Figure 3 illustrates the equipment arrangement and coolant flow path in the reactor module.

3 Transient Event Classification, Criteria And Limits

Transient events are grouped into three categories based on the probability of the event's occurrence: 1) design basis events (DBEs), 2) accommodated anticipated transients without scram (A-ATWS) and 3) residual risk events.

3.1 Design Basis Events

The first category, design basis events, consists of events to be considered in designing of the subsystems and components of the nuclear steam supply system. These events are normally expected to occur regularly during the lifetime of the plant and include startup, normal operations, load following and shutdown. Events caused by equipment failures in non-safety-grade systems are also included.

Design basis events are evaluated on a conservative two-sigma basis. Component lifetime, economics and performance are the focus of the evaluations of these events. ASME Service Levels "A", "B" and "C" apply to these events, depending upon the probability of each event and its potential damages. The goal is to ensure long-life, high reliability components and structures and to ensure their potential failure modes do not unduly shorten plant or equipment life. Tables 1 and 2 summarize criteria and limits applied to these events.

3.2 Accommodated Anticipated Transients Without Scram

The second event category is a portion of the traditional beyond design basis events that are of such low probability that they are normally precluded from direct consideration in conjunction with the design requirements. They represent the subset of beyond design basis events with three characteristics in earlier LMRs;

- in prior LMR designs these accidents lead rapidly into coolant boiling, fuel melting and core disassembly
- the root initiating event has a high probability of occurrence during the life of the plant
- the reactor protection system (RPS) is assumed to fail to terminate the event following the initiating event

Since the initiator events are of high probability (about 1 x 10^{-2}), these events are "anticipated" to occur during the life of the plant. The further assumed failure of the RPS causes them to progress into a core disassembly accident. Since the probability of failure of the RPS is less than 1 x 10^{-6}, the probability of the accident event is on the order of 1 x 10^{-8} or less. In prior LMRs, these events are termed anticipated transients without scram (ATWS) and are not accommodated by the plant except by containment of the accident debris.

The S-PRISM top-level requirements require this subset of beyond design basis events to be accommodated without loss of reactor integrity or radiological release, using passive or inherent natural processes such as natural circulation. Reliance on inherent natural processes ensures the accommodation of the event will occur with a very high probability. A loss of functionality or component life-termination following the event is acceptable since the goal is public protection. Since they are beyond design basis events that are accommodated by the plant within a selected damage envelope, they are renamed accommodated anticipated transient without scram (A-ATWS) events for the S-PRISM plant design.

The A-ATWS events are evaluated on a nominal basis like other beyond design basis events.

The focus of the evaluation of the A-ATWS events in S-PRISM is evidence of passive limitation of the severity of the event such that reactor and core integrity is maintained.

ASME Service Level "D" limits are applied to these events to ensure continuity of critical equipment and structural functions during the event. Tables 1 and 3 list the core temperature criteria and limits applicable to these events.

3.3 Residual Risk Events

The third category is that of residual risk. The probability of a severe core accident is less than 10^{-7} per plant year, or equivalently for the modular S-PRISM, less than 10^{-8} per reactor year. These probabilities are so low that the events fall into a "residual risk" classification.

One application of some these events is use as the design basis for containment. In this case, the equipment operational performance is not of interest, so long as its failure does not jeopardize containment performance.

While the plant has no performance criteria related to these events, some events may be used to determine the "safety-robustness" of the plant design. Robustness represents the tolerance for failures beyond those associated with the DBEs and A-ATWS events. Transient performance analyses thus use subsets of the residual risk events to look for conditions or sequential failures that can push the plant past the criteria for A-ATWS event accommodation. These events are not considered in this paper.

4 Transient Analysis Results

Table 4 summarizes maximum temperatures reached during the transients analyzed. Each event is discussed below. Since Design Basis events, by design, do not pose a significant challenge to public or investment risk, only one severe event is presented for illustrative purposes. The five important A-ATWS events are presented in detail since they are used to judge passive safety performance.

4.1 Design Basis Events

4.1.1 Over-power To Scram

Design basis events typically end with a fast power runback under PCS control or, if more severe, with a scram under control of the RPS. Since the setpoints of the RPS are selected to limit damage to the reactor and core to low levels, these events primarily affect component lifetime and are not a challenge to safety. Among the more severe of these events is a transient overpower in which power continues to increase until the RPS executes a scram. In ARIES, this event is modeled as a rod withdrawal ending in a scram based on the RPS setpoints.

Figure 4 plots reactor performance parameters during the overpower and scram. The rod runout begins at 2 seconds at the physical maximum rate of 0.02$/sec from a rod runout, with the reactor initially operating at 100% power and flow.

Figure 4(a) plots core fission power and primary coolant flow rate during the event. The RPS scram setpoint is 113% of full power, including uncertainties, so this upper bound scram setting is used for conservatism. Sensing, logic and initial rod motion impose a short delay that allows power to overshoot to 114% of full power before the rod insertion starts reducing power. Scram shutdown is rapid. Power shutdown from rod motion begins 7.4 seconds after the start of the runout.

There is a short additional delay before the primary pumps are tripped. The RPS delays the pump trip until it verifies the beginning of the shutdown through observation of decreasing flux. The pump trip begins 9.1 seconds after the start of the runout. Primary flow drops to pony-motor flow, about 10% of full flow.

The core outlet temperature peaks at 515 °C when the core is at 114 % power, at 8.8 seconds after the rod runout begins, as shown in Figure 4(b).

The fuel temperature decreases as a result of shutdown of fission power. The cooling fuel causes a positive reactivity insertion from Doppler feedback of 1.09$. The primary pump trip causes the GEMs to void through the core elevation and produce a 1.4$ negative reactivity insertion. The control rod insertion dominates these core feedbacks with more than 13$ of negative reactivity insertion.

Figure 4(c) plots the peak temperatures in the hottest core fuel assembly, including +2 sigma hot channel factors. The hot subchannel temperature in this assembly peaks at 701 °C at 8 seconds after the runout begins. At this time, the pumps are still operating and the local saturation temperature is about 1066 °C. An ample margin to coolant boiling is maintained.

The fuel centerline temperature peaks at 2266 °C. The melting temperature of the fuel is about 2680 °C, thus an ample margin to fuel melting exists.

4.2 A-ATWS Transient Performance

4.2.1 Loss Of Flow Without Scram

A loss of flow without scram event (ULOF) is initiated by a failure of power to the primary EM pump power supplies. The supply of bulk power to the EM pump power supplies is not a safety-grade system, thus this event is of high enough probability to be considered "anticipated". The EM pumps, pump control system and flow coastdown synchronous machines are safety-grade and are thus available to provide a flow coastdown when the power is lost. Since continuing power is unavailable at the end of the safety-grade coastdown, pony motor operation of the EM pumps is not possible and natural circulation of the primary sodium removes core power. To create the A-ATWS ULOF event from the LOF, the RPS is assumed failed.

Figures 5 plots reactor performance parameters during the ULOF. The power to the primary pumps is terminated at 2 seconds with the reactor operating at 100% power and flow.

Figure 5(a) plots core fission power and primary coolant flow rate during the event. Since the loss of flow includes a loss of pressurization of the core inlet high-pressure plenum, the GEMs rapidly void and cause a 1.4$ negative reactivity feedback. The fission power is rapidly terminated by the action of the GEMS.

During the first 100 seconds of the event, the power ratio is greater than the flow ratio and coolant temperatures increase. This is shown in Figure 5(b). Core shutdown by the GEMs soon over-cools the core and reactivity feedbacks increase core power to balance the heat being removed by the IHTS. This equilibrium state is fully established by about 1500 seconds.

The fuel temperature decreases as a result of the GEM shutdown of fission power. The cooling fuel causes a positive reactivity insertion from Doppler feedback, as shown in Figure 5(c). The initial increase in core outlet temperature heats the control driveline and thermal expansion extends the control absorbers farther towards the core, however core cooling and vessel heating move the core farther from the drives at the same time. The net effect is a small positive reactivity feedback from control-core relative motion during the early stage of the event.

During the first few hundred seconds of the event, the positive feedbacks, mostly fuel Doppler and axial feedbacks, are offset by the GEM negative feedback and the core achieves a rapid subcritical shutdown.

Over the longer term, the growing positive feedback from vessel heating and control driveline cooling combines with the positive feedbacks from core axial thermal compaction and Doppler to fully offset the GEM shutdown. This condition indicates the core is over-cooled compared to the equilibrium temperature of a zero-power critical state. The net reactivity becomes slightly positive and increases fission power to generate enough heat to compensate for the IHTS cooling. The slight increase in power causes the Doppler and control driveline feedbacks to reduce and bring the core back to a critical state that is in equilibrium with the IHTS heat removal capability.

Under this long-term equilibrium, power and flow are close to 9% of normal operating conditions and primary loop heat transport is by natural circulation. The core outlet temperature is less han 500 °C. If heat removal capability is reduced by steam generator dryout or by loop flow reductions, the core power will drop maintain equilibrium with heat removal capability.

The hot subchannel temperature peaks at 797 °C at 11.5 seconds after the pump power is terminated. Since the boiling temperature of sodium in the core under natural circulation conditions is approximately 950 °C, an ample margin to coolant boiling is maintained. Since the core power is very low by this time in the transient, the cladding temperature is almost the same as the subchannel coolant temperature. The duration of the high temperature spike is short and damage accrual (inelastic creep) to core components is minimal. Over the long term, core temperatures remain low and core damage is negligible. Operator actions are eventually required to finally insert the controls and end the event with a cold shutdown.

4.2.2 Loss Of Heat Sink Without Scram

A loss of heat sink without scram event (ULOHS) is initiated by a failure of the IHTS or BOP. Because the power to the pump controls is not safety-grade, a failure to provide power to the IHTS EM pumps is of high enough probability to be considered "anticipated" and is more severe than the loss of feedwater or steam generator. The IHTS does not include flow coastdown synchronous machines and thus no flow coastdown occurs when the power is lost. To create the A-ATWS ULOHS event from the LOHS, the RPS is assumed failed.

Since the primary EM pump power is available, a loss of heat sink initiator event would normally cause a reactor scram and a trip of the primary EM pumps after reducing flux indicates a shutdown is in progress. However, since the RPS is assumed failed, both the control rod insertion and primary pump trip are disabled. The EM pumps remain operating and adding heat to the primary loop until the safety-grade EM pump thermal shutoff system (TSS) executes the pump trip. Following this action, the event becomes fundamentally identical to an event with RVACS and ACS cooling and with normal shutdown, with performance as predicted by the decay heat removal analyses. This long-term performance characteristic is the result of passive reactor shutdown by core feedbacks when the RPS fails to act.

Figure 6(a) plots the core power and flow fractions during the ULOHS. The LOHS begins at 2 seconds from full power operation. Initially, the primary EM pumps continue operation and hot coolant quickly passes to the cold pool. The increasing core inlet temperature causes negative reactivity feedback that begins reducing power at a rate of about 0.4 %/sec. After the heatup has progressed for about 200 seconds, the coolant temperature at the EM pump inlet has reached the trip temperature setpoint of the TSS. This safety-grade system terminates power to the EM pumps and a flow coastdown begins. Since power is available to the system, the pumps coast down to pony motor flow and then remain at constant power.

As shown in Figure 6(b), the core outlet temperature reaches a maximum of 649C at 240 seconds. This occurs during the power to flow imbalance during the flow coastdown phase, where the GEMs are shutting down fission power more slowly than the flow coastdown is occurring.

Figure 6(c) plots core reactivity feedbacks. At the start of the LOHS, the increasing core temperature causes a small control driveline thermal expansion and negative feedback. Core component and restraint thermal expansion causes a growing negative feedback from core radial thermal expansion feedback. It is primarily the core radial thermal expansion that offsets a positive reactivity feedback from sodium density feedback and causes the early power decrease while the primary pumps continue to operate.

At 200 seconds, GEM negative feedback begins to dominate the core power state as the pumps coastdown. A general cooling of the reactor continues for about 1200 seconds. A decreasing fuel temperature that accompanies GEM-driven shutdown creates an increasing positive Doppler feedback. The heating vessel lengthens and causes an additional positive feedback from changes in the relative positions of the core and control absorbers.

The increasing positive feedback offsets the GEM negative insertion by around 1200 seconds and core power then increases to reach an equilibrium with the heat being removed by RVACS and natural circulation of the IHTS. The long-term equilibrium state is about 8% power and 11% pony motor flow as enhanced by the buoyant head. The equilibrium core outlet temperature is about 490C. The peak subchannel coolant temperature peaks at 224 seconds at 739 °C. An ample margin to coolant boiling, about 950 °C, remains. Operator actions are eventually required to finally insert the controls and end the event with a cold shutdown.

4.2.3 Loss Of Flow And Heat Sink Without Scram

A loss of flow and heat sink without scram event (ULOFLOHS) is initiated by a failure of power to the primary and IHTS EM pump power supplies. The supply of power within the plant is not a safety-grade system, thus this event is of high enough probability to be considered "anticipated". This initiator event is similar to a loss of site power (site blackout event). The safety-grade primary EM pump synchronous machines provide flow coastdown for the primary coolant. The IHTS does not include flow coastdown synchronous machines and thus no flow coastdown occurs when the power is lost. Both heat transport loops continue to function by natural circulation. To create the A-ATWS event, the RPS is assumed failed.

Figure 7(a) plots the core power and flow fraction during the event. The loss of primary pump power results in an immediate flow coastdown to natural circulation. GEMs insert about 1.4$ of negative reactivity and cause rapid fission power shutdown. Shutdown results in falling fuel temperatures, as shown in Figure 7(b). At the same time, the loss of heat sink causes coolant and structure temperatures to increase. The core outlet temperature peaks at about 600 °C over an extended interval.

Falling fuel temperature causes a positive Doppler reactivity feedback, as shown in Figure 7(c). Increasing coolant temperature causes a positive insertion from sodium density feedback. The rapid vessel heatup causes a positive reactivity insertion. These positive insertions are balanced by the negative GEM insertion, such that the core remains subcritical at the elevated temperatures. The vessel and control drivelines are heating and increasing in length. Thus they tend to offset each other to some degree.

The coolant temperature in the peak subchannel reaches a maximum of 792 °C at about 16 seconds. The margin to local boiling (950 °C) is acceptable. Operator actions are eventually required to finally insert the controls and end the event with a cold shutdown.

4.2.4 Transient Overpower Without Scram

A transient overpower without scram (UTOP) is initiated by a failure of the non-safety-grade control system. Incorrect control rod setpoints cause the runout of the control rods until they impact against the rod blocks that are part of the safety-grade rod stop system (RSS). The RSS limits the maximum reactivity insertion to 0.30$. To create the A-ATWS event, the RPS is assumed failed.

Figure 8(a) plots fractional power and flow. As the rods are withdrawn at the maximum rate of 0.02$/sec, power increases to a maximum of 124.5% at 16 seconds after the start of the rod runout. During the first 200 seconds, core reactivity feedbacks offset some of the control rod insertion and reduce power to a steady-state of 115%. Flow remains constant.

System temperatures are plotted in Figure 8(b). The core outlet temperature increases to 537 °C and then remains roughly constant at the new operating point.

Core reactivity feedbacks are plotted in Figure 8(c). The control insertion is offset by Doppler feedback from the fuel and by control driveline thermal expansion and core axial thermal expansion. The power increase leads to a fuel temperature to cause both the axial expansion and the Doppler feedbacks. Over a much longer interval, core radial expansion becomes more active in providing negative feedback. The increase in core outlet temperature heats the control driveline and thermal expansion provides a negative feedback. The vessel also slowly heats and reduces the amount of negative feedback the control driveline expansion creates.

Fuel centerline temperature increases to 2291 °C and is well below the melting temperature of 2680 °C. The peak subchannel coolant temperature increases to 677 °C. Since the primary pumps are operating, the local saturation temperature is 1066 °C and an ample margin to boiling exists.

This event does not result in a fission power shutdown. Rather, the plant is able to continue carrying the power produced by the reactor. Operation at the long-term steady-state 115% power is tolerated by the plant and equipment, although with some amount of lifetime reduction. Under normal plant supervisory controls, the unaffected reactor in the power block will be assigned a power setpoint of 85% to offset the excess power from the affected reactor. However, the plant can accommodate the higher power of both reactors experiencing a UTOP for an extended period. Operator actions are eventually required to finally insert the controls and end the event with a cold shutdown.

4.2.5 Safe Shutdown Earthquake Without Scram

A safe shutdown earthquake without scram (USSE) is initiated by a safe shutdown earthquake. For S-PRISM, this is an earthquake with a 0.5 g ZPA ground motion. To create the A-ATWS event, the RPS is assumed failed.

During the A-ATWS, the impact on the core is primarily a cyclic reactivity insertion caused by horizontal and vertical vibration.

Horizontal vibration: The reactor module is on a platform that is seismically isolated in the horizontal direction. The horizontal natural frequency of the NSSS is about 0.75 Hz. Thus, the reactor is moving horizontally with a 0.75 Hz frequency. The lateral natural frequency of each assembly is about 10 Hz, thus the core sways horizontally with a 0.75 Hz frequency with compaction at that frequency when the core impacts the top former ring at the end of each sideways sway. Finite element analyses of reactor dynamics indicates the compaction causes at most a radial reactivity feedback insertion of +- 0.30 $.

Vertical vibration: The reactor module is not seismically isolated in the vertical direction, however, the small diameter of the reactor makes it stiff in this direction. The head, carrying the controls, and the core support structure, carrying the core, have natural frequencies of approximately 10 Hz. Assuming these two structures oscillate out of phase, the maximum reactivity insertion of the controls and core moving relative to each other is +- 0.16 $.

The two reactivity insertions (from horizontal and vertical oscillations) are simplified into sine waves at 0.75 and 10 Hz and summed. For analysis, the oscillations are assumed to last 20 seconds. The reactivity oscillation starts at 2 seconds with the reactor operating at full power.

Figure 6.6-1 plots the core power and flow fractions during the USSE. Flow remains constant. Since the RPS fails to release the rods when the large plant acceleration is detected, the control rods and core sway cause a reactivity oscillation. The reactivity oscillation causes power to also oscillate, with peak instantaneous powers reaching 180% of full power. The instantaneous power closely follows the reactivity insertion oscillation. The power spikes are of short duration and the heat capacity of the fuel absorbs and smoothes them into a rising average fuel temperature. The event is of short duration and has little effect on overall coolant and structure temperatures, as shown in Figure 9(b).

As shown in Figure 9(c), the principal reactivity is the seismic insertion. Heating fuel causes a small negative Doppler feedback that tends to slightly reduce the overall power peaks with time. Although the seismic event causes a small sodium elevation oscillation within the GEM assemblies, it is not within the core axial region and has no feedback effect. The small change in overall core and reactor temperatures cause little control or vessel length change and thus no reactivity feedbacks from these phenomena.

The power oscillations cause an oscillation in the fuel inner temperatures, but the heat capacity of the fuel smooths the effects in the cladding and coolant. These temperatures remain near the normal full power operating state. In spite of the large power oscillations, the fuel centerline temperature increases less than 200°C above normal temperatures in the peaks and does not approach melting.

5 Conclusions

An LMR plant concept that is based on a passive safety approach to protection of the public and plant investment risk reduction uses an approach to CDA countermeasures very different from prior conventional LMR designs. First and foremost, the focus of the design is to reduce the probability of a CDA to such a small value that it does not cause extreme features or systems to be added to the plant specifically to accommodate it. The second focus is then to ensure that if a CDA were to occur, it would not result in energetics sufficient breach the reactor boundary and thus not be a significant challenge to containment.

These goals are best satisfied by use of passive features of the LMR that are, by nature, very difficult to defeat. If the probability of a CDA is negligible because reactivity control, heat removal and containment are provided by non-active systems using natural laws, as opposed to active machines or operator actions, then special CDA accommodation features are not needed for licensing. Further, if conservatively estimated energetics from an assumed CDA are insufficient to breach the vessel or containment, then special features to reduce energetics are not needed for licensing.

The S-PRISM plant employs a wide array of passive features to ensure reactivity control, heat removal and containment. Overall, the passive safety features employed in S-PRISM form the most effective CDA countermeasures practically achievable. Collectively, they reduce the probability of a CDA to a negligible value and then accommodate the event if it is assumed to occur.

Significant improvement in safety, licensing and owner risk are achieved at acceptable capital cost if the ATWS events are passively accommodated by the plant within performance limits that well assure 1) the integrity of the reactor boundary, 2) the control of fission power and 3) the removal of heat from the reactor. The transient analyses indicate acceptable performance during the ATWS events, with large margins to coolant boiling or fuel melting.

Table 1 Core Temperature Limits For Reactor Structural Integrity

Parameter	Temperature Limit	
ASME Service Level A:		
Core Average Outlet Coolant	950 °F steady state	510 °C steady state
Peak Assembly Discharge Coolant	1100 °F <1 hr.	593 °C <1 hr.
Assy Discharge Coolant Striping Potential (Delta Temp)	370 °F	206 °C
ASME Service Level B:		
Core Average Outlet Coolant	1100 °F <1000 hr.	593 °C <1000 hr.
ASME Service Level C:		
Core Average Outlet Coolant	1250 °F	677 °C
ASME Service Level D:		
Core Average Outlet Coolant	1400 °F <1 hr. 1350 °F >1 hr.	760 °C <1 hr. 732 °C >1 hr.

Table 2 Limits For Core Component Integrity In Design Basis Events

Parameter	Steady State Operation	Transient Operation
Peak Subchannel Coolant Temp. (°F)		
Under Full Flow	1950 °F 1066 C	1950 °F 1066 °C
Natural Circulation	1750 °F 954 C	1750 °F 954 °C
Cladding Damage Accrual		
Creep Rupture Damage Fraction (CDF)	<0.001	<0.2
Thermal Creep Strain	0.01	-
Total Strain	0.03	-
Swelling	0.05	-
Peak Fuel Temperature(Mixed Oxide)		
Fuel (BOL and EOL)	4856 °F 2680 °C	4856 °F 2680 °C
Blanket (BOL)	5156 °F 2847 °C	5156 °F 2847 °C
Blanket (EOL)	5101 °F 2816 °C	5101 °F 2816 °C

Table 3 Limits For Core Component Integrity In A-ATWS Events

Parameter	Limit	
Subchannel Coolant Local Boiling		
During Full Flow Conditions	1950 °F	1066 °C
During Natural Circulation Flow	1750 °F	954 °C
Fuel Melting (Mixed Oxide)		
Fuel (BOL and EOL)	4856 °F	2680 °C
Blanket (BOL)	5156 °F	2847 °C
Blanket (EOL)	5101 °F	2816 °C
Area Percent Allowed Melting	50%	50%

Table 4 Peak Transient Temperature Summary

	Peak Core Outlet Temperature		Peak Subchannel Coolant Temperature		Peak Fuel Centerline Temperature	
	(sec)	(°C)	(sec)	(°C)	(sec)	(°C)
Temperature Limits (C)						
Saturation Limit - Pumps On		1066		1066		
Saturation Limit - Pumps Off		954		954		
Fuel Melting Temperature						2680
Design Basis Events						
Transient Overpower To Scram	8.2	517	7.8	701	7.4	2266
A-ATWS Events						
Loss Of Flow Without Scram (ULOF)	14.5	593	11.5	792	0.0	1938
Loss Of Heatsink Without Scram (ULOHS)	211.0	653	208.0	739	0.0	1938
Loss Of Flow And Heatsink Without Scram (ULOFLOHS)	14.5	594	11.5	793	0.0	1938
Transient Overpower Without Scram (UTOP)	85.0	538	21.0	677	20.0	2291
Safe Shutdown Earthquake Without Scram (USSE, 0.5g ZPA)	88.0	534	53.0	641	3.0	2142

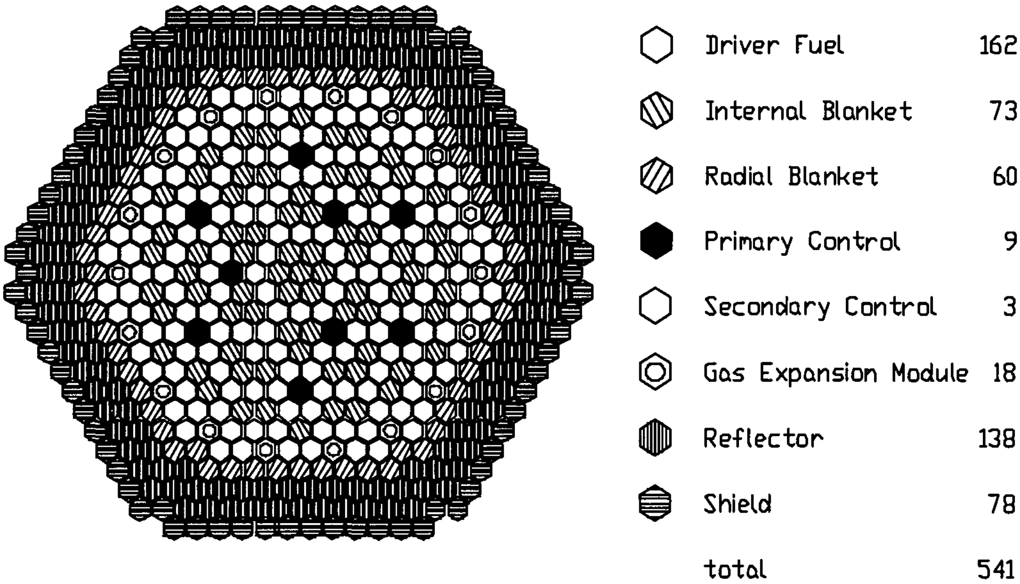

Figure 1 Radial Heterogeneous MOX Core Configuration

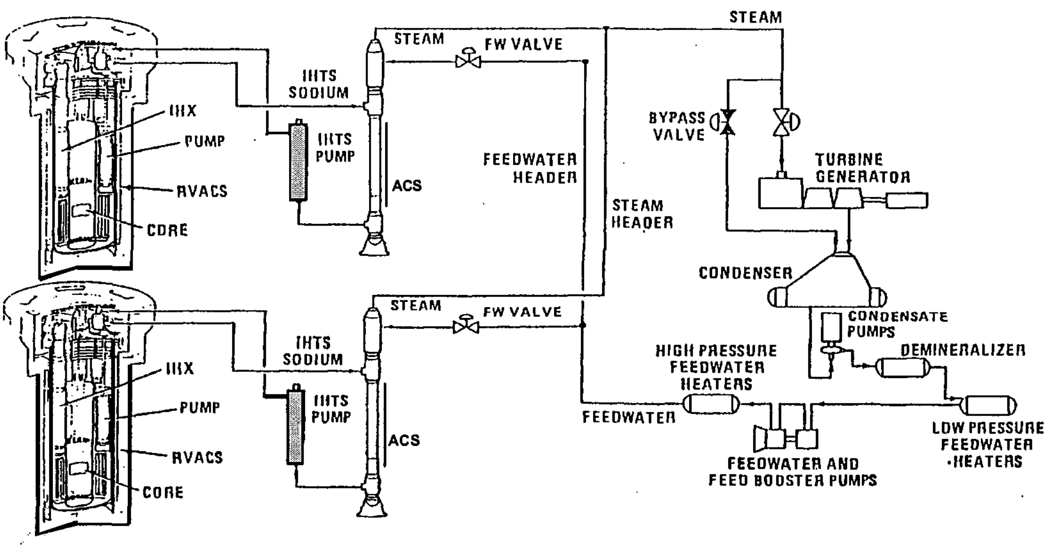

Figure 2 ARIES-P Model

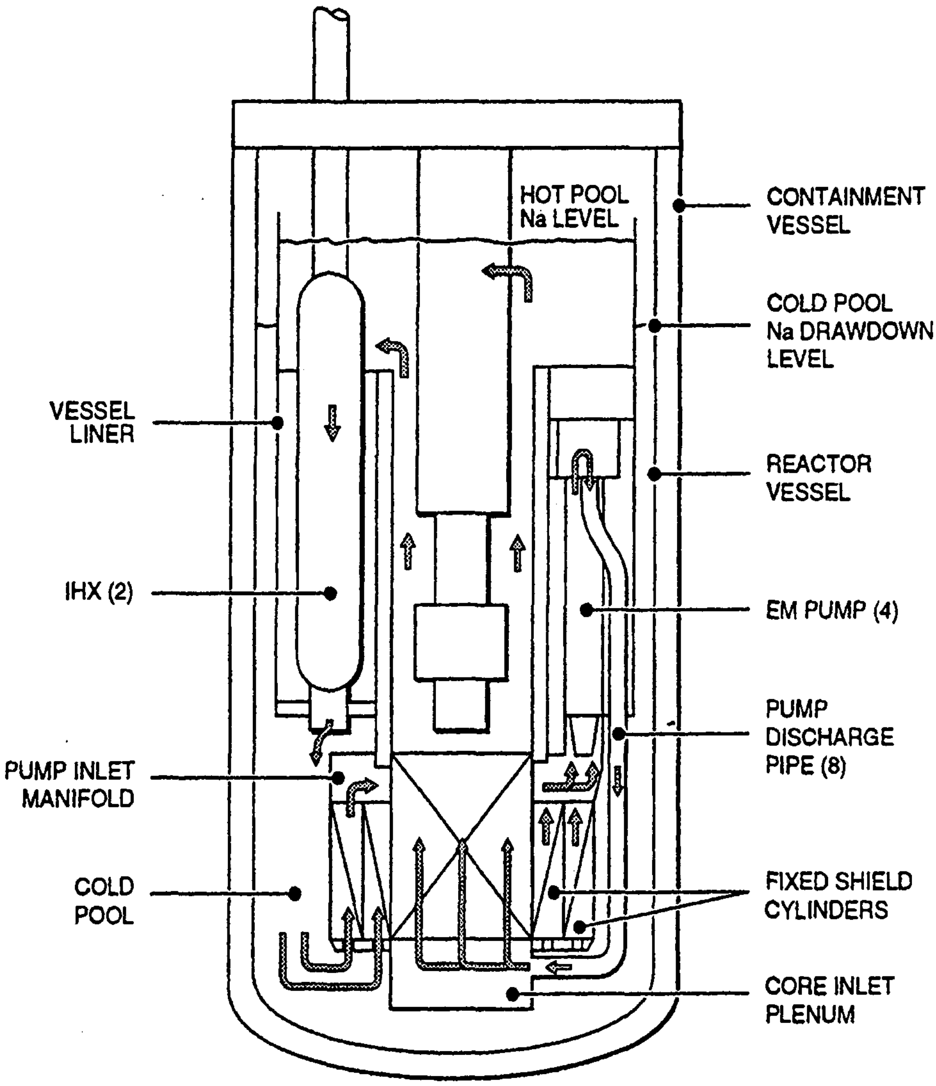

Figure 3 Reactor Primary Coolant Flow Path

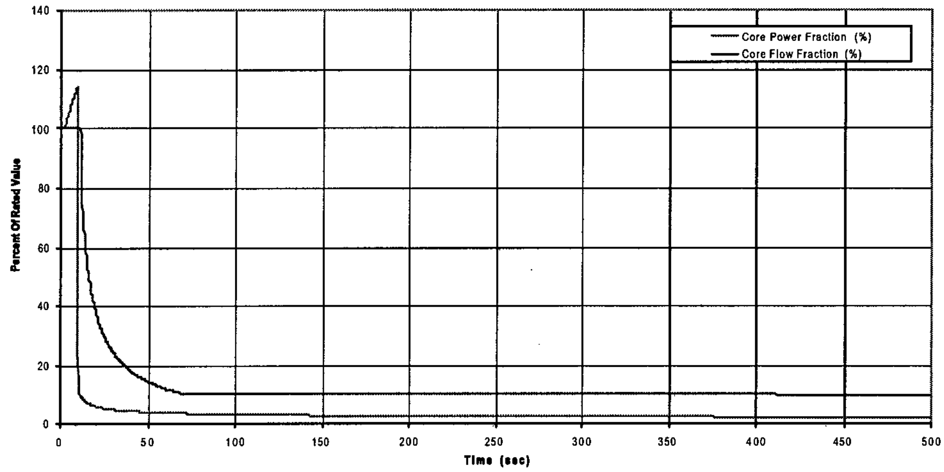

Figure 4(a) Over-Power To Scram - Power And Flow

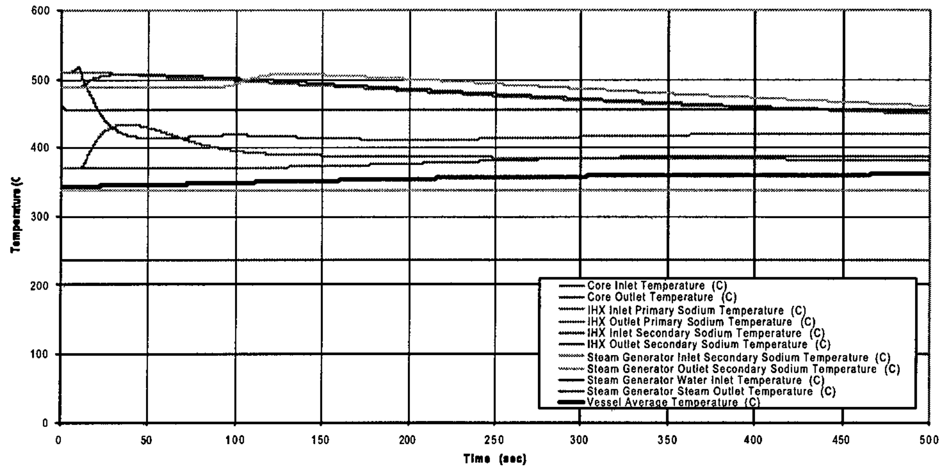

Figure 4(b) Over-Power To Scram - System Temperatures

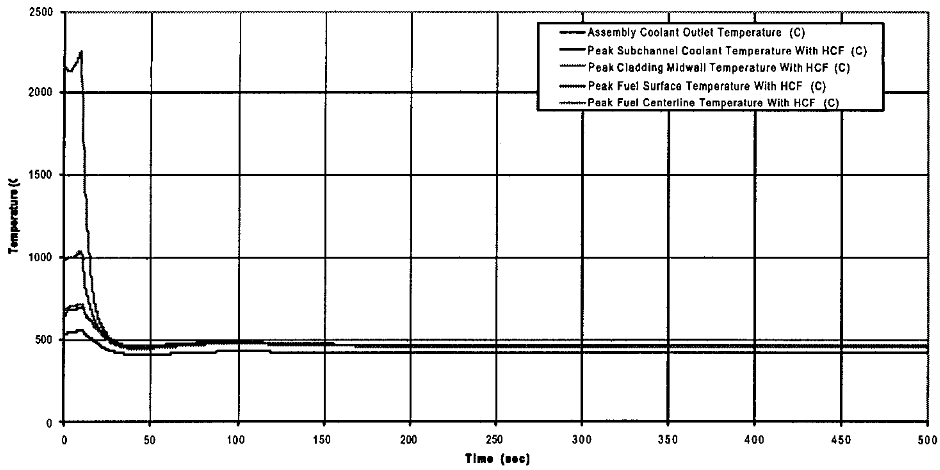

Figure 4(c) Over-Power To Scram - Peak Assembly Peak Temperatures

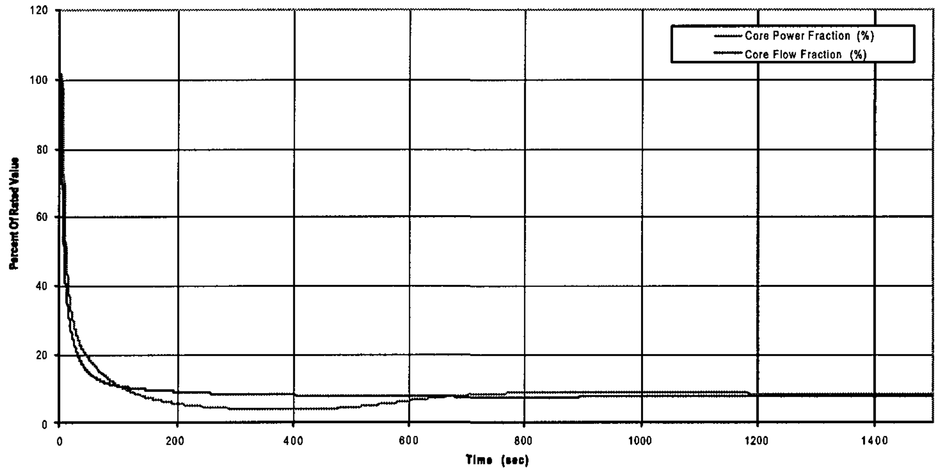

Figure 5(a) ULOF - Power And Flow

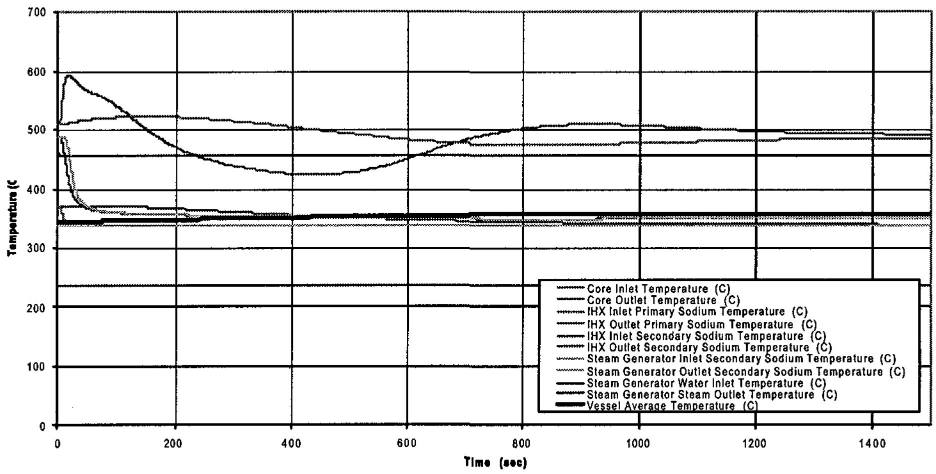

Figure 5(b) ULOF - System Temperatures

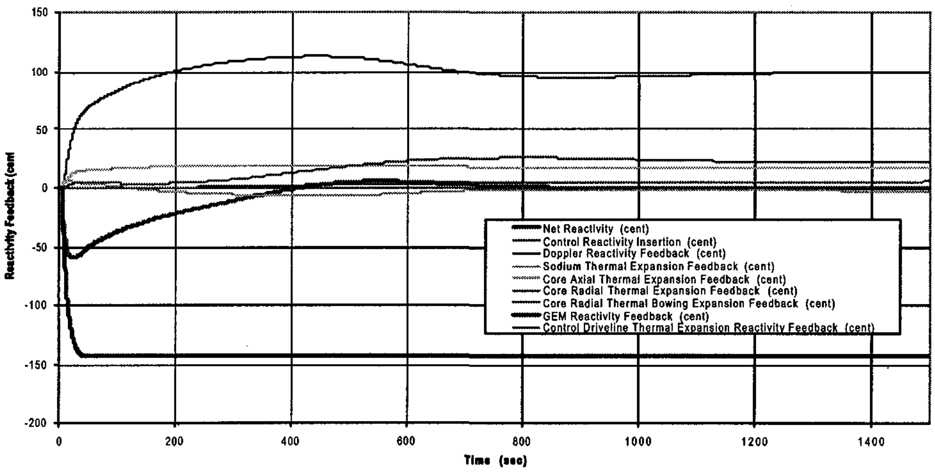

Figure 5(c) ULOF - Reactivity Feedback

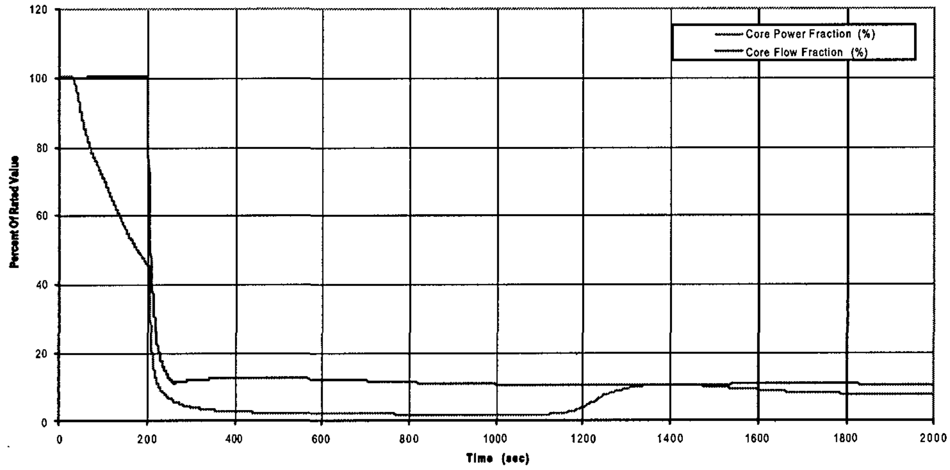

Figure 6(a) ULOHS - Power And Flow

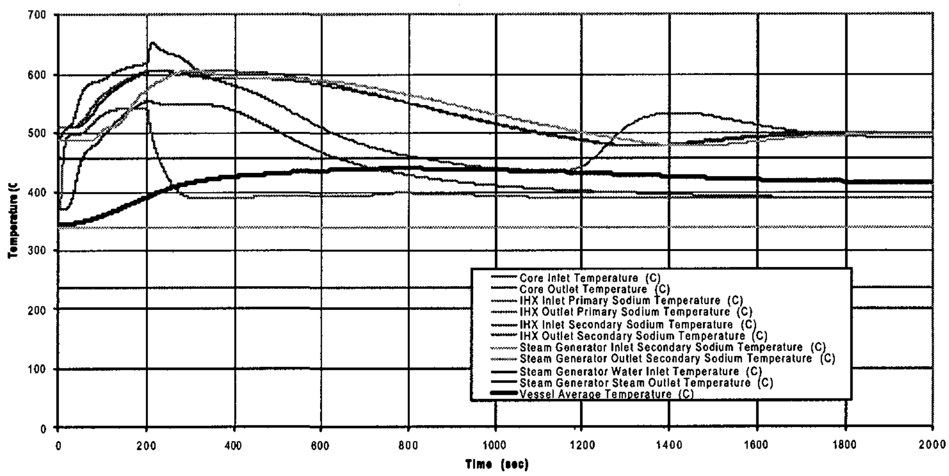

Figure 6(b) ULOHS - System Temperatures

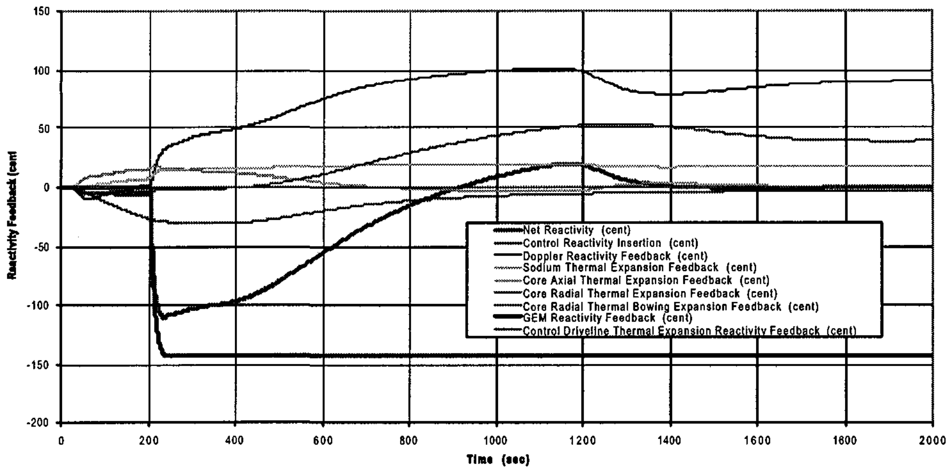

Figure 6(c) ULOHS - Reactivity Feedback

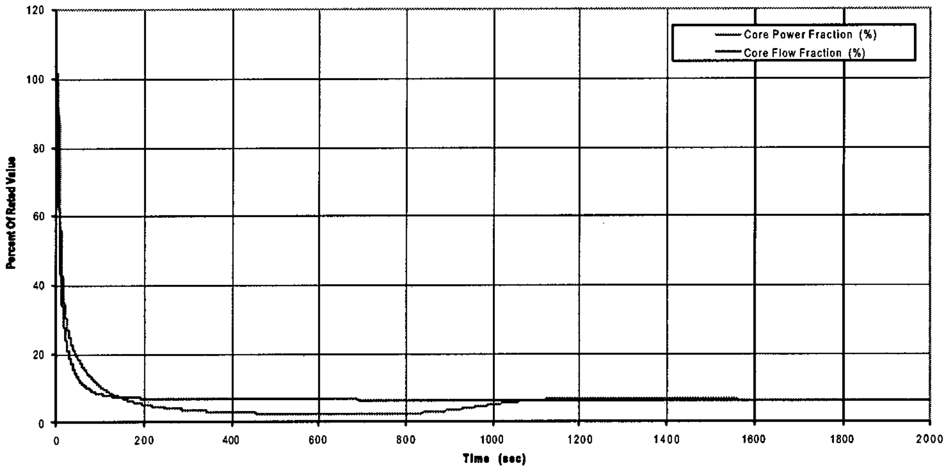

Figure 7(a) ULOFLOHS - Power And Flow

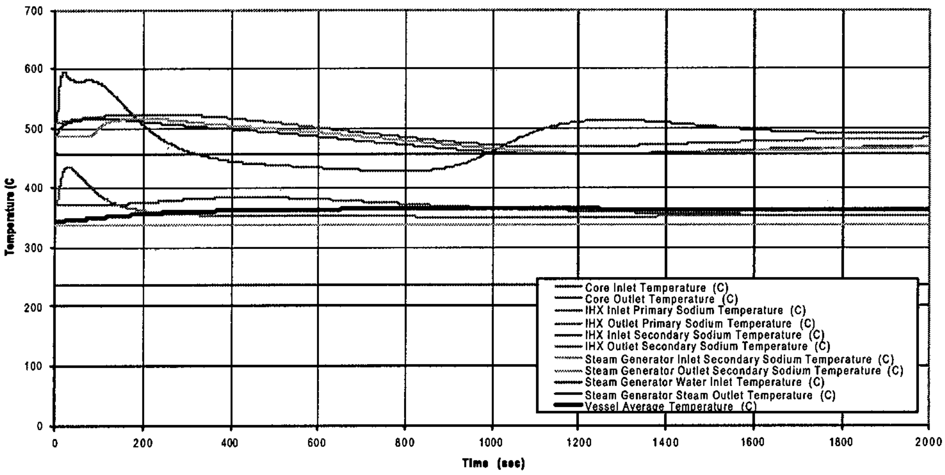

Figure 7(b) ULOFLOHS - System Temperatures

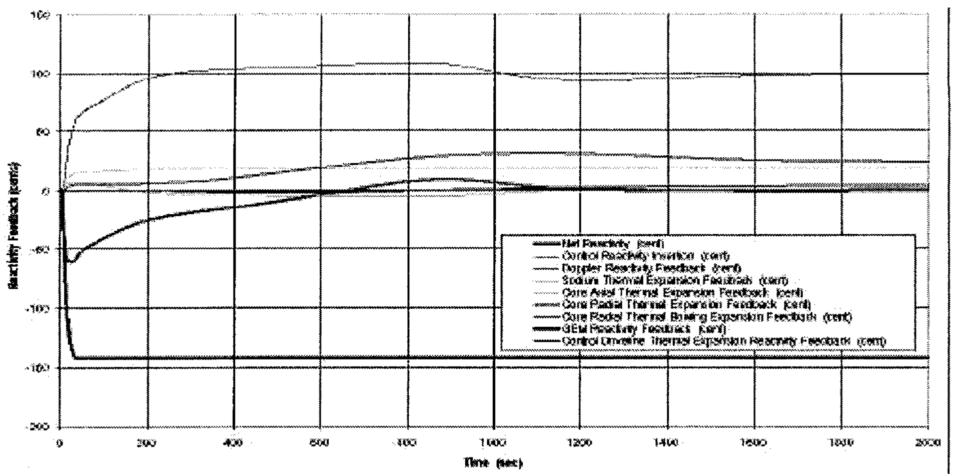

Figure 7(c) ULOFLOHS - Reactivity Feedback

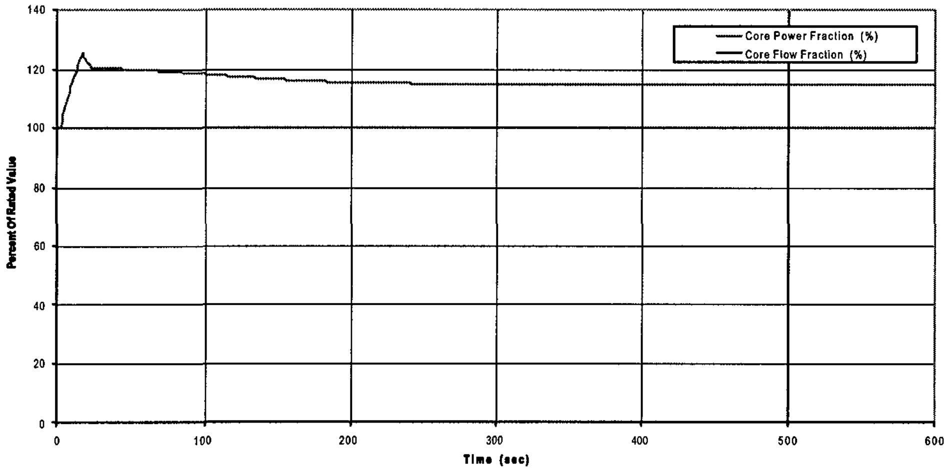

Figure 8(a) UTOP - Power And Flow

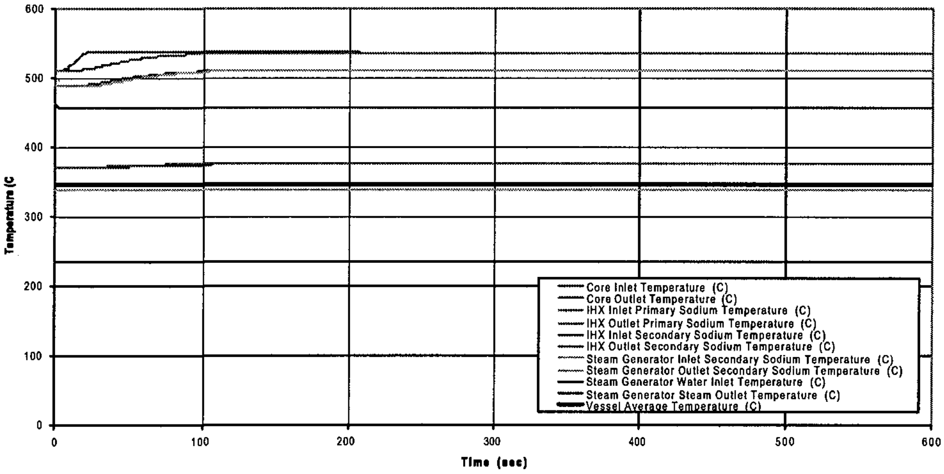

Figure 8(b) UTOP - System Temperatures

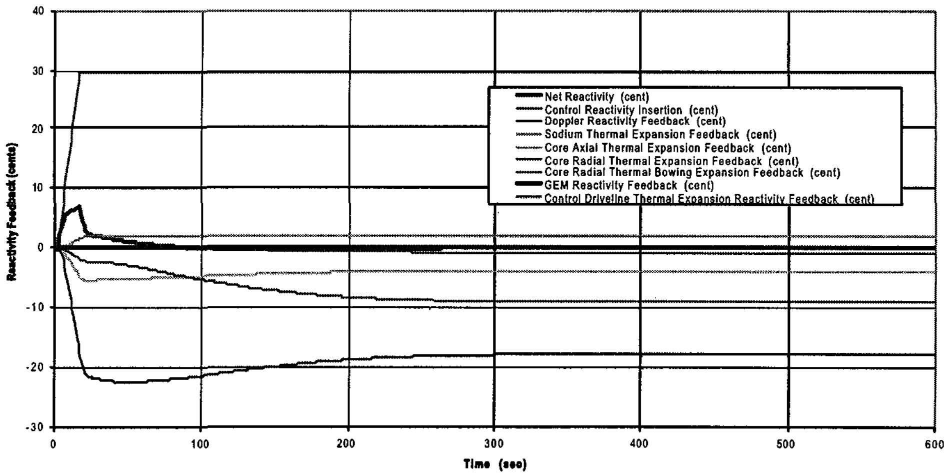

Figure 8(c) UTOP - Reactivity Feedback

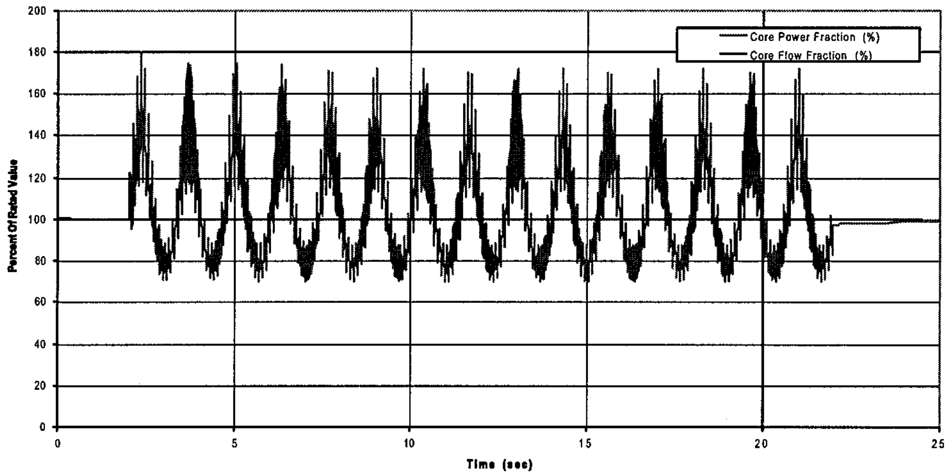

Figure 9(a) USSE - Power And Flow

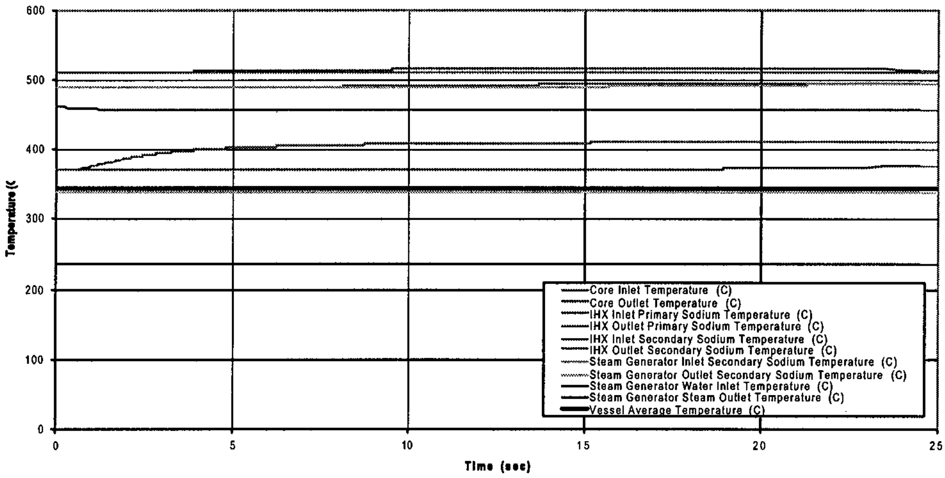

Figure 9(b) USSE - System Temperatures

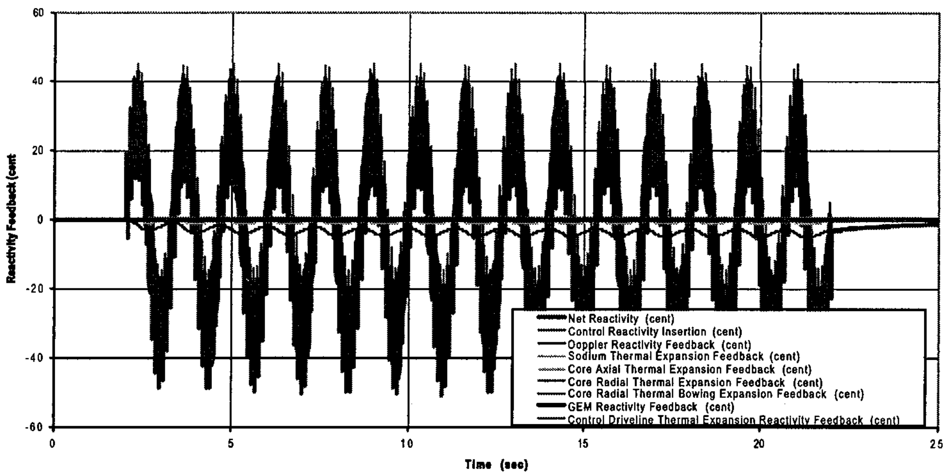

Figure 9(c) USSE - Reactivity Feedback