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SUPERPRISM METAL CORE MARGINS TO SEVERE CORE DAMAGE

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ABSTRACT

The SuperPRISM metal-fueled modular reactor is designed to accommodate a set of beyond design basis accident initiator events that in prior LMRs led to coolant boiling and core melting. These "Accommodated Anticipated Transients Without Scram" (A-ATWS) are chosen to bound the higher probability core disruptive accident (CDA) initiators. Their explicit accommodation by design, with some amount of core and reactor life reduction, greatly reduces the probability of reactor failure, containment challenge and public endangerment. These events, all with an assumed failure of the safety scram system, include:

- unprotected all-rods withdrawal to the rod stop limit of 0.30$
- unprotected loss of primary flow
- unprotected loss of heat sink
- unprotected loss of flow and heat sink
- unprotected safe shutdown earthquake

Three events which are well beyond the design basis have been selected to assess the available margins to sodium voiding. These margin assessment events are:

- unprotected all-rods withdrawal to greater than the rod stop limit of 0.30$
- loss of flow with multiple safety system failures
- control rod ejection

In addition, four types of very low probability severe accident events which have the potential to void a portion or all of core have been selected to assess the potential core damage and energetics of very low probability residual risk events. These events include:

- Flow blockage of a single assembly
- Sweep-through of gas entrained outside of core
- Unprotected all-rods withdrawal without rod stops
- Unprotected loss of flow with degraded flow coastdown

Calculations indicate that essentially all severe, "beyond design basis" and even the less probable "residual risk" core accidents result in zero or negligible energetics. None of the extremely severe voiding accident types appear to have the potential to bring the S-PRISM metal core to prompt critical (e.g., result in 1$ or more of net positive reactivity). However, additional experimental confirmation is needed for the unprotected loss of flow accident without coastdown on all four pumps.

Finally, one event has been selected in which large scale core melting will occur. This event allows assessment of the progression of a CDA and the potential for recriticalities. The event is:

- Loss of normal heat sink with increased core decay heat and reduced shutdown heat removal capability

The analysis of this event indicates very low energetics potential with metal fuel and suggests further design studies or modifications may be useful in assuring no recriticalities may occur as the molten metal fuel drains through the core lower axial shield zone. Once the molten fuel is into the lower inlet plenum, it is contained and cooled in a dispersed geometry.

INTRODUCTION

This paper provides an assessment of the ability of the SuperPRISM reactor system to accommodate a wide spectrum of events that are well beyond design basis. Events are analyzed to assess the available margins to coolant voiding, and the potential core damage and energetics. A large scale CDA event is analyzed to assess the potential for recriticalities. Each of the postulated events are described and assessed in the following section.

First, however, a few special considerations about metal fuel failures are outlined to aid in understanding the event progressions.

METAL FUEL BASICS

Metal fuel is a ternary alloy of uranium, plutonium and zirconium. At typical driver fuel enrichments, the conservative alloy melting temperature is about 1750°F (954°C). Breeding blanket alloy does not include Pu and has a higher melting temperature of about 1880°F (1025°C). The fuel is formed in solid metallic slugs and sealed inside steel cladding tubes. The slugs occupy about 75% of the internal cross-section of the pins, and the annulus between the slug and cladding is filled with sodium for thermal bonding between the two. During irradiation, fission gas formed in the fuel forms pores throughout the slug. The fuel swells into tight contact with the cladding within the first few atom percent of burnup. The low initial fuel smeared density allows extensive volumetric fuel swelling, such that the pores interconnect and allow fission gas to escape to the plenum. At operating temperatures, the fuel is relatively soft compared to the cladding and fuel-cladding mechanical interaction (FCMI) is not a significant pin failure mechanism.

The voids form a reservoir of trapped gas distributed throughout the fuel to drive an expansion if the fuel melts. Molten fuel is quite viscous and a molten foam so formed does not tend to drain back down into a compacted state. The axial in-pin foaming expansion causes a large negative reactivity feedback that can serve to end an event.

The fuel and blanket alloys can form a eutectic with the iron in the pin cladding by diffusion. Experiments using irradiated and unirradiated fuel and blanket materials indicates the eutectic alloy formed in the interdiffusion zone has a melting temperature of 1300°F (704°C). This is significantly below the bulk fuel melting temperature and tends to be the major pin failure mechanism in severe transients. Once the eutectic liquefies, interdiffusion accelerates and cladding wastage leads to rapid failure.

Eutectic attack of the cladding occurs at temperatures well below the boiling point of the sodium coolant, which is typically higher than 1750°F (950°C). Thus, for transients that fail the cladding while sodium is still flowing, the molten fuel and fuel-cladding eutectic are expelled into cooler sodium and can be swept out of the core. The expelled fuel tends to solidify into ribbons or stringers that are dispersed and coolable where they settle out of the flow in the upper plenum.

For pin failures without high velocity sodium to cause sweep-out, the molten fuel will drain downward through the stagnant sodium or sodium vapor. If it resolidifies in the low flux regions of the lower axial shielding, lack of cooling will soon remelt it and allow it to continue draining towards the coolant inlet plenum. Once in the inlet plenum, the fuel remains a coolable liquid layer until it erodes enough reactor structure and inlet orificing module iron to raise the alloy melting temperature above the local temperatures. The CDA then ends with a solid fuel and steel mass in the inlet plenum. The geometry and enrichment are well below critical and the debris is coolable by the cold pool below the inlet plenum. In SuperPRISM, the inlet plenum and core support structures are specifically designed to provide this core retaining and cooling function.

MARGIN ASSESSMENT EVENTS

Unprotected All-rods Withdrawal to Greater than 0.30$ Limit

The rod stops are set to limit the total reactivity addition from all nine primary rods withdrawal from the full power operating position to 0.20$. For conservatism, the reactivity insertion is assumed to be 0.30$. In this accommodated unprotected transient overpower (UTOP) event, one of the ATWS events, the power increases rapidly and peaks at about 180% power near the end of the rod withdrawal stroke. The power then decreases in response to core reactivity feedbacks to quasi-steady-state of about 130% power.

For margin assessment, the UTOP reactivity insertion is assumed to be up to 0.60$ at the maximum control drive stepper motor speed. Table 1 summarizes calculated temperatures for this series of UTOPs in the PRISM reactor. As the inserted reactivity increases, the peak centerline fuel temperature, the time some fuel is molten, and the time the peak cladding temperature is above the minimum liquid phase formation temperature of 1300°F increase. However, even at 0.60$ reactivity addition, the cladding wastage is less than 0.3 mil and the margin to boiling in the peak channel is greater than 400°F. A small number of fuel pins would be expected to fail with molten fuel injected into the coolant channels and carried out of the core by the sodium flow momentum. The temperature of the molten metal fuel is almost the same as the sodium temperature and there is no chemical reaction nor any energy generated upon fuel-coolant contact. Long-term cladding temperatures remain below 1300°F even for 0.60$ reactivity addition.

Table 1 provides the time period when molten fuel is present in peak fuel pins. The melting temperature is based on a nominal solidus temperature of 1815°F for U-2Zr-20Pu. The peak assemblies are fresh fuel assemblies having an enrichment of 18%; however, it is also assumed that these assemblies have undergone Zr redistribution leaving a region near the centerline of 2% Zr.

This type of melt behavior has been experimentally demonstrated in a series of TREAT transient tests using metal fuel pins irradiated in EBR-II. The pins were subjected to rapid reactivity addition transient. The pins typically failed at 4 to 4.8 times nominal peak power, that is, 45 to 65 kW/ft peak. Significant amounts of fuel were molten over the upper 1/3 of the pins at the time of cladding failure. The fission gas within the pin plenum and fuel then expelled the molten fuel into the coolant stream where it rapidly froze in stringers and was carried out of the core. There were no interactions between the molten (and later solid) metal fuel and the sodium.

Reactivity losses associated with various amounts of molten fuel ejection from a single PRISM assembly are summarized in Table 2, as calculated by the nuclear perturbation method.

Extrapolation of these results to the SuperPRISM reactor considers the higher core temperature and total core power and the larger size of the uprated design. Higher core power is not strongly significant as the coolant flow returns the core to the original temperature rise and the pin linear power is about the same. The small core size increase will impact reactivity feedbacks slightly, but a resulting transient temperature change will be small. The higher core inlet and outlet temperatures act to increase the transient temperatures by about the same amount as the operating temperature rise, 20°F (11°C) As a result, the times the fuel is molten and above 1300°F increases and additional pins will be expected to fail. However, the change is small and not expected to significantly affect the number of fuel failures or change the overall conclusions.

If the transient power increase is fast enough to melt significant amounts of fuel along the centerline, along with the surface eutectic melting, the molten fuel column will foam upwards within the pin and produce a large negative feedback from in-pin fuel relocation. If sufficient fuel is molten at the time of clad rupture, sweep-out of ejected fuel is an effective mechanism for reactivity loss and permanent shutdown.

For pins at lower temperature, fuel-clad liquid phase will be slowly formed. This liquid phase, located between the fuel column and the cladding, will act as a lubricant and "unlock" the fuel column from the cladding. The fuel is then expected to extrude, adding a small negative reactivity. Eventually, the cladding will fail in a region containing liquid phase. The liquid phase will flow into the sodium and be swept out of the core. When sufficient fuel, in the form of molted fuel or fuel-clad liquid phase, is swept out of the core, the reactivity will permanently decrease and the transient will be terminated.

Note that, in addition to assuming the control system and scram system have failed, no credit has been taken here for the Secondary Shutdown System. This system is activated by a second independent safety grade Reactor Protection System (RPS) and by passive Curie point magnetic latches. The latches provide a "Self Actuated Shutdown System" (SASS) capability in the event that the RPS fails to request a scram when needed.

Unprotected Loss of Flow with Multiple Safety System Failure

Unprotected loss of primary flow (ULOF) events are characterized by loss of electrical power to the four primary pumps followed by coastdown controlled by the electrical feedback input from the synchronous motors. As core temperatures increase, reactivity feedbacks come into play and the power decreases, but not as rapidly as the flow.

There are very large margins to local boiling in ULOF events in PRISM, as shown in Table 3. The higher operating temperature in SuperPRISM (+20°F (11°C)) will increase these peak transient temperatures, but will not change the overall conclusions. For margin assessment, events are assumed with increasing severity of multiple safety system failures. Table 3 tabulates the consequences of multiple failures of three safety systems: coastdown failures in one or more primary EM pumps, scram failure of the nine primary control rods and 3 secondary control rods and failure of the six gas expansion modules to provide negative reactivity feedback. To reach local boiling in the peak power fuel assemblies requires one of the following sets of multiple failures: 1) loss of coastdown on three or four pumps plus no scram or 2) loss of coastdown on two pumps plus no scram plus no more than three GEMs effective

Control Rod Ejection Accident

An investigation of the S-PRISM control system was made to estimate the probability of a control rod ejection accident as traditionally analyzed for LWRs. This investigation led to the conclusion that the unique design of the S-PRISM control system reduces the probability of all postulated scenarios to such a low value that analysis of the consequences is not required. This is due to the fact that criticality cannot be attained with just one secondary or two primary control assemblies inserted.

The rod accident analyzed for LWRs assumes a scenario in which a control assembly is left "hung up" in the core during control rod withdrawal and is then released and ejected from the core after the reactor is critical or at power. This results in a rapid ramp input of the total worth of the stuck rod. A similar event in the S-PRISM is extremely unlikely due to the safety design of the control system. The S-PRISM has nine primary and three secondary control assemblies. The reactor can be shut down to hot standby conditions with insertion of one secondary or two primary rods. A rod ejection accident therefore requires that eleven or more control assemblies are simultaneously ejected.

The reactor control system withdraws the assemblies one at a time in small increments so that all nine rods are always at approximately the same elevation. Accidents involving inadvertent withdrawal of control assemblies have been analyzed and are acceptable. During normal scram the control assemblies are followed into the core by the drive lines driven by special high-torque drive-in motors to assure full insertion in 18 seconds. Following a scram the control assemblies could not be suddenly ejected to the elevation required to achieve criticality since the drivelines would be fully inserted.

The design of the control system requires the assemblies to fall into the core by gravity in two seconds with full coolant flow. To assure that this is achieved, they are designed so that the pressure drop across the assembly at normal coolant flow is low enough that there is a safety factor of nine that they will not be lifted by hydrodynamic forces. Thus, even if the drivelines did not rapidly follow the assemblies into the core after a scram, the assemblies could not be affected by normal coolant flow.

Only one extremely improbable scenario has been identified that could result in ejection of multiple assemblies. This scenario assumed that during refueling, with the control drive lines withdrawn to allow rotation of the UIS, the primary pumps are turned on at full power. Since the control assemblies have a safety factor of nine against flotation at full coolant flow, this alone would not eject the assemblies. It must be further assumed that the control assemblies are sufficiently plugged to provide the lifting force required to eject them from the core. Since rod drop tests are part of the cycle restart tests, the probability of this sequence of events is so low that evaluation of this scenario is not required.

CONSEQUENCES OF CORE VOIDING ASSESSMENT EVENTS

Four types of severe accident events which have the potential to void part or all of the S-PRISM core have been identified. These are:

- Flow blockage of a single assembly
- Sweep-through of gas entrained outside of core
- Unprotected all-rods withdrawal with reactivity addition greatly in excess of that set by the rod stops
- Unprotected loss of flow with degraded flow coastdown

An extreme example of each type of voiding event is analyzed in the following four subsections.

Total Flow Blockage of a Single Fuel Assembly

Reactor startup is initiated from a bulk primary sodium temperature of 550°F with 90% core flow. The power is increased to 25% full power in no less than 30 minutes. When the power reaches about 4 - 8 % of full power, the blocked assembly will rapidly void, due either to fission gas blanketing or sodium boiling. In either case, the voiding of the assembly will add no more than 0.15$ reactivity. Centerline fuel melting will also occur at about this time.

The maximum possible reactivity additions due to assembly voiding and subsequent fuel slumping will not result in rapid propagation to other assemblies at these low power levels. As in the Fermi reactor blockage accident, molten fuel movement is expected to generate a net reduction in reactivity.

The event will be terminated automatically upon a flux trip due to an increase in power resulting from the initial voiding or from molten fuel compaction within the blocked assembly, provided the core power increase is greater than the flux scram set point. Without the scram, fuel ejection from the assembly will terminate the local power excursion. Alternatively, the event can be terminated by operator action either (1) after observation of an unexpected reactivity change, (2) on response to delayed neutron signals resulting from refluxing and repeated expulsion of sodium, with failed fuel particles, from the blocked assembly, or (3) in the worst case, in response to a delayed neutron signal resulting from penetration of molten fuel-clad alloy into edge channels of an adjacent assembly with flowing sodium. Even if no action is taken, the propagation into an adjacent assembly will allow sufficient sodium flow into the blocked assembly to cool it. In any case, the event terminates in a benign configuration, with minimal core damage. The above analysis and general conclusions are strongly supported by the FERMI flow blockage incident, in its initiation, development, termination and final configuration.

It should also be noted that flow blockages will not occur in the S-PRISM active core region during power operation because of the lack of chemical reaction between the metal fuel and sodium. This is a distinct safety advantage of metal fuel relative to oxide fuel.

Massive Sweep-through of Gas Entrained Outside of Core

Four potential sources of gas which could be swept through the reactor core have been identified: fission gas in fuel pins, argon in GEMs, sodium free surface entrainment, and nitrogen released from an EM pump stator.

At end of life, there are approximately 4 in^3 of gas in a fuel pin, or a total of 0.77 ft^3 in a fuel assembly of 271 pins. Any gas released from a pin weld failure or rupture of the cladding will be carried up out of the core, through the fuel assembly and into the outlet plenum. The gas will be significantly de-entrained around the flow loop, and any remaining gas will be in the form of fine bubbles before entering the core. The reactivity effect of sweep-through of such fine bubbles will be insignificant. A similar argument holds for release of argon from a gas expansion assembly (GEM).

Gas entrainment (by vortices) at the sodium free surface of the outlet plenum has been considered in all previous LMR designs. This possibility can easily be eliminated by proper design, supported by water flow tests.

The nitrogen gas in the EM pump stators is viewed as the only significant source of gas for entrainment and sweep-through. There is about 30 ft^3 of nitrogen at 700°F and 14 psi in each pump. The pumps are located in the primary cold leg, downstream of the IHXs. Flow from the pumps is ducted directly to the core inlet plenum. A pump stator failure which would release gas is expected to be a small leak, not a major rupture; no mechanism for a major rupture has been identified. Upon leak, because of the lower pressure within the stator, sodium will flow into the stator rather than gas out of the stator. It can be further anticipated that the stator failure will fail the pump, terminate the pumping action, and, consequently, result in flow reversal through the pump. Thus, any gas release will not travel directly to the core but to the common plenum feeding the four pumps.

As a limiting event, it was assumed that all the gas from one EM pump is released as a massive bubble and travels directly to the core. Even in this case, it is expected that the bubble will be broken up in the core inlet plenum, due to the "forest" of inlet modules, and will preferentially pass through the outer region of the core (reflectors and shields), which has negative void worth.

An even more conservative assumption is that the massive gas bubble passes uniformly through the entire core. In this extreme case, the gas will have an average volume (within the core) of 9460 in^3 and will extend over 9% of the active core height. The middle 9% of the core has about 20% of the total void worth, or 0.8\$ at EOC. Based on the maximum sodium flow velocity of 21 ft/sec, the maximum reactivity addition rate is 54 \$/sec to a peak of 0.8\$. This very rapid, sub-prompt-critical reactivity addition and removal only increases the peak coolant temperature to less than 1250°F.

Thus, it is concluded based on a simple limiting event analysis that the worst possible gas sweep-through event is tolerable, prompt criticality is not reached, and boiling is not initiated.

Unprotected All-rods Withdrawal to Maximum Possible Limit

The total amount of reactivity suppressed by the nine primary control rods at BOEC, including uncertainties, is about 2.25\$ for the SuperPRISM metal core. During an unprotected transient overpower event, this maximum amount of reactivity can be inserted only if the rod stops on all nine drivelines fail. As a limiting case to envelope other potential core designs, ARIES calculations were run to a transient overpower with the rods being withdrawn at two times the maximum speed of the stepper motor, up to 8\$ total insertion. The method used for scoping this transient does not model fuel movement or coolant boiling, thus the analysis is followed only to boiling. This transient rapidly leads to cladding rupture and fuel dispersal. The peak cladding temperature reaches 1990°F, the point of very rapid liquid phase cladding attack and essentially instantaneous cladding rupture, at 20.2 sec., with the reactivity insertion reaching about 0.9\$. Boiling has not occurred at this time and expelled fuel will be swept from the assembly. If fuel dispersion did not occur, the analysis would predict boiling in the peak coolant subchannels at about 22.4 sec.

It is concluded that any transient overpower event of sufficient magnitude to result in assembly voiding leads, by means of molten fuel and/or fuel-clad liquid phase sweep-out, to benign termination of the transient. There will be negligible energetics and work potential, no challenge to the containment and solid, coolable, subcritical debris.

Unprotected Loss of Flow Without Coastdown

An abrupt loss of flow without scram remains as the most obvious path to initiate energetics sufficient to threaten primary system integrity. Simultaneous, complete failure of the coastdown systems on all four primary EM pumps, coupled with a failure to scram at least one control rod, leads to large scale coolant boiling and voiding in the low-pressure primary system. The positive coolant voiding reactivity feedback results in elevated reactor power and causes wide-spread fuel melting and cladding failures.

This issue has been investigated for the PRISM core by Argonne National Laboratory. An initial, scoping study was performed which considered cores with total void worth of 10\$ or less. The accident sequence was initiated by complete, instantaneous loss of coolant flow. A voiding-driven power transient with peak power a few tens of nominal followed. Expansion of fuel within cladding then returned the power to a few times nominal. Fuel dispersal in the coolant channels following cladding failures resulted in neutronic shutdown. In toto, the energy released was estimated to be about an order of magnitude lower than would be required to fail the reactor vessel boundary.

This scoping study was followed by an updating of the SAS4A severe accident analysis code to include molten fuel motion models, and calculations were run for the unprotected loss of flow without coastdown event in the PRISM core. The core was modeled with nine channels representing 126 driver fuel assemblies and three channels representing 108 internal and radial blanket assemblies. A reduced motor/generator inertia resulted in a very rapid flow coastdown, with 180 ms halving time. No fission gas migration was allowed during the transient, and in-pin molten fuel motion was permitted when the molten cavity reached the top of the fuel pin.

Sudden flow stoppage leads to rapid heating of the sodium coolant in the core, a consequence of continued power generation, not heat stored in the fuel. Boiling begins in the highest power assemblies within a few seconds. Coolant voiding in some (high powered) assemblies results in a positive net reactivity which leads to a power increase. Boiling continues to spread in response to the power increase. Fuel melting within the pin occurs simultaneously with boiling.

Axial upward motion of molten fuel within the pin into the fission gas plenum occurs, driven by fission gas retained in the fuel, introducing substantial negative reactivity feedback, terminating the power rise. Fuel pin (cladding) failures occur, initially causing a small positive reactivity feedback. The resulting molten fuel dispersal upward by gas escaping from the pins into the coolant channels shuts down the reactor.

The SAS4A results indicate that the PRISM core responds non-energetically to an unprotected loss of flow event with loss of coastdown on all four primary pumps. For an assumed uniform 5 at.% fuel burnup, peak fuel temperatures were below 3140F. There was a relatively long delay between the start of in-pin fuel motion and cladding failures and a large energy deposition due to decreased initial in-pin reactivity feedback rates and longer transient times. When the calculation terminated, the energy release had reached nearly 7 full-power-seconds and the peak fuel temperature was less than 2600°F. Assuming an uniform 15 at.% burnup resulted in increased sensitivity to timing of in-pin motion and cladding failures but did not result in a significant threat to the reactor vessel. Cladding failures occurred before in-pin fuel motion. Peak fuel temperatures were about 4760°F.

Extrapolation of these studies to the higher temperature SuperPRISM core will not materially change the conclusions. Additional experimental tests and detailed analyses are needed to fully define the safety-related behavior of this class of severe transients, particularly the movement of molten fuel within the core in the later stages of the transient.

ASSESSMENT OF GROSS CORE MELTING AND REDISTRIBUTION

Gross core melting in SuperPRISM will be similar to that of the PRISM core. The peak and average linear powers are similar, the structures are the same, and the core mean temperatures are only increased by 20F (11C).

PRISM safety analyses indicate that the probability of core melt by extensive fuel-clad liquid phase formation is extremely low, well below the level of the safety goal. (The probability of core melt by actual fuel melting is even lower, because fuel melting temperatures are at least 400°F greater than the minimum temperature for fuel-clad liquid phase eutectic to form.) ATWS transients have been shown to be benign and do not to lead to core voiding or meltdown. In addition, preliminary tests suggest that blockages in the metal core will be coolable and will not lead to significant amounts of fuel liquefaction.

Therefore, a hypothetical, low probability (residual risk) event has been postulated in order to evaluate the capability of the SuperPRISM primary system boundary to accommodate a large (whole-core) melt-down. The hypothetical event is initiated by:

- Loss of normal heat sink and ACS heat removal
- Higher than expected decay heat
- Reduced capability of RVACS

Core decay heat is increased to an upper 4 design-sigma value. In addition, the airside heat transfer coefficient and thermal emissivity are decreased to their respective 4-sigma levels. This hypothetical event results in an estimated maximum average core outlet temperature of 1350 °F at 30 hours into the transient. At sodium temperatures above 1300°F, a slow interaction occurs between the ternary metal fuel and the cladding, resulting in the formation of a fuel-clad liquid phase. Fuel-clad liquid phase formation and cladding attack will begin at approximately 16 hours into the transient. High burnup (end of 3rd cycle) fuel pins will rupture after about 20 hours at the top of the fuel columns, where the fuel surface temperatures are highest. Following cladding rupture, the liquefied fuel-clad mixture has the potential to relocate downwards in the fuel assembly and into the inlet plenum. As it passes through the core region, the fuel increases core reactivity and increases the equilibrium shutdown temperature of the core. If at any location the fuel solidifies by cooling from the inlet sodium or assembly structure, the local flow restriction will remelt the fuel and it will continue to the inlet plenum.

The evaluation focused on assessing the feasibility of containing the liquefied fuel within the reactor vessel, specifically within the core inlet plenum. Scoping analyses were performed for a set of melt compositions and volumes to assess the potentials for recriticality, temperatures in the liquid phase, the capability of the plenum structure to accommodate the steel consumed by the liquid phase, and the capability of the plenum to contain the fuel-clad mixture.

Based on the analyses, recriticality does not appear to be possible. Recriticality could occur only if all of the fuel in the core slumped, at the same time, into the shield regions of the fuel assemblies. However, due to different fission gas pressures in pins with different burnups, the times of cladding failure and release of liquefied fuel-clad into the sodium coolant vary so widely from third to second to first cycle assemblies that no more than one-third of the fuel would be relocated in the shield region at any given time. Thus recriticalities during the melt progression through the assembly structures is not expected. Recriticality is also not expected for any composition after the liquefied fuel-clad mixture spreads out in the inlet plenum. The geometry of the melt pool and the fuel composition result in a subcritical arrangement.

The analysis of the state of core containment within the inlet plenum was performed with the thermal hydraulic model used for the RVACS analysis, with appropriate modifications to include relocation of the core to the inlet plenum as indicated in Figure 1. Figure 2 shows the predicted temperature distribution in the fuel-clad mixture, the inlet plenum structures at the peak temperature conditions, 29.9 hours after shutdown. It can be seen that, at this time, a portion of the 15-cm thick bottom support plate would be above 1500 °F, the temperature at which the ASME Code assumes it to have no strength, although in reality it would still have some structural capability. However, the entire 5 cm thick backup plate located directly below the bottom support plate (lower portion of the high pressure inlet plenum) is maintained at temperatures well below the 1500°F ASME Code temperature limit. With this configuration, the majority of the temperature gradient is taken across the 15-cm thick bottom support plate leaving the backup plate temperatures within the ASME Code temperature limits. Analyses indicate that the fuel-clad mixture will consume at most 2.5 cm of the 15 cm bottom plate of the inlet plenum, with no effect on the backup plate underneath the plenum, and that the fuel-clad mixture can be retained from the entire core (fuel and blankets) in the inlet plenum. Analysis also indicates that the reactor cover gas pressure increase due to fission gas released from failed pins and decay heat loads can be accommodated within the primary system boundary with comfortable design margins against ASME Code design limits.

An additional, less severe, thermal analysis was performed assuming that only 50 percent of the core (fuel and blanket) was relocated to the inlet plenum, and that the relocation process occurred in stages at times corresponding to the estimated pin failure times for first, second and third cycle assemblies. The results of this analysis showed that all structural temperatures are well below the 1500 °F limit during the entire transient.

Thus, the scoping analyses indicate that for the hypothetical case in which it is assumed that a major core melt occurs, the molten fuel/clad eutectic would be retained within the lower internal structures and away from the reactor vessel. Evaluations to better define the core fuel-clad compositions, geometries and temperatures in order to validate the above conclusion could be performed if desired. The confirmatory program would consist of ex-reactor experiments including:

- downward relocation of fuel-clad liquid phase in the assembly
- fuel-clad liquid phase breakup, quench and solidification in the sodium-filled regions under the core
- the effect of iron in the liquid phase mixture composition ranging from UFe_2 to various composition of U-Fe-Zr,
- coolability of core debris accumulated on horizontal surfaces in the sodium pool
- fuel dispersal in a transient overpower event retention of fuel and fission products within the sodium.

CONCLUSIONS

These assessments have shown that essentially all severe, "residual risk" core accidents which could lead to core voiding result in zero or negligible energetics. None of the extremely severe voiding accident types appear to have the potential to bring a metal core to prompt critical (e.g., result in 1$ or more of net positive reactivity) as summarized below.

Flow blockages can not occur in the active core region during power operation because of the lack of chemical reaction between the metal fuel and sodium. This is a distinct safety advantage of metal fuel relative to oxide fuel. Flow blockage of an assembly can occur due to a fabrication error. Preliminary analysis of core startup with an undetected blocked assembly indicates that the event will terminate in a benign configuration, with minimal core damage. This conclusion is strongly supported by the FERMI flow blockage incident.

The bounding case of voiding of the core due to massive gas sweep-through results in void reactivity addition of less than 54$/sec, to a peak of 0.86$. The resulting peak coolant temperatures does not exceed 1250°F. All possible gas sweep-through events are benign, boiling is not initiated, and prompt criticality is not reached.

Unprotected transient overpower events, up to the maximum possible 2.25$ which could be inserted by the nine control rods at BOEC, will terminate with benign sweepout of molten fuel and/or fuel-clad eutectic from the active core. There will be no energetics as the molten fuel does not interact with the sodium coolant. The relatively cold temperatures within the reactor vessel (other than in the core) result in the formation of subcritical, coolable, solid fuel "stringers" within the reactor vessel.

SAS4A results indicate that the core responds non-energetically to an unprotected loss of flow event with loss of coastdown on all four primary pumps. The accident does not lead to energetic events in the initiating phase. At the end of the SAS4A calculations, after some dispersal in the coolant channels, the peak fuel temperature is less than 2600°F (one to two seconds after fuel pin failure). Additional work is needed to verify these results; modeling improvements may provide further insight, but experimental support is required.

Table 1. RESULTS FOR UNPROTECTED TRANSIENT OVERPOWER EVENTS OF INCREASING SEVERITY AT BOC

Magnitude (cents)	Peak Fuel Temp (F)	Time Fuel Molten (sec)	Time Eutectic Molten (sec)	Peak Coolant Temperature (F)
30 (Nominal)	1792	0	0	1213
40	1926	49	49	1296
50	2083	60	60	1397
60	2253	70	70	1511

Table 2. REACTIVITY LOSSES ASSOCIATED WITH MOLTEN FUEL EJECTION

	Reactivity Loss ($/Assembly)		
Percent of fuel ejected	10%	50%	100%
Inner ring fuel assembly	- 0.06	- 0.87	- 1.87
Outer ring fuel assembly	- 0.04	- 0.54	- 1.18

Table 3. MARGINS TO LOCAL BOILING IN LOSS OF FLOW EVENTS

Pumps With Coastdown (Of 4)	Control Rods Inserted (Of 12)	GEMs Operational (Of 6)	Consequences
4	10	6	Expected
4	0	6	Nominal
4	0	0	No boiling
3	0	0	No boiling
2	1	0	No boiling
2	0	4 - 6	No boiling
2	0	0 - 3	Local boiling
1	1	0 - 6	No boiling
1	0	0 - 6	Local boiling
0	1	0 - 6	No boiling
0	0	0 - 6	Local boiling

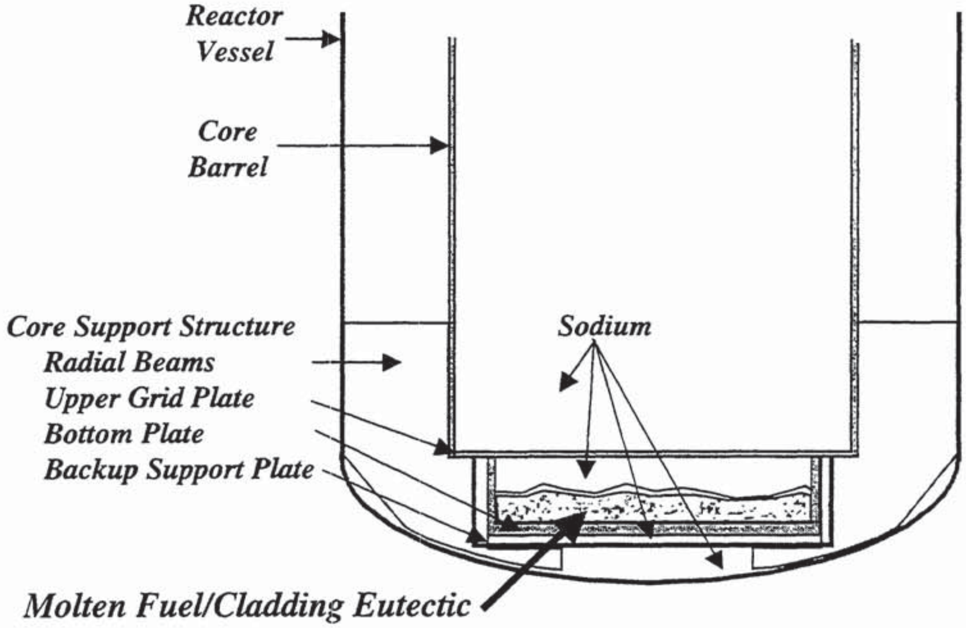

Figure 1. CORE EUTECTIC ACCOMMODATION MODEL

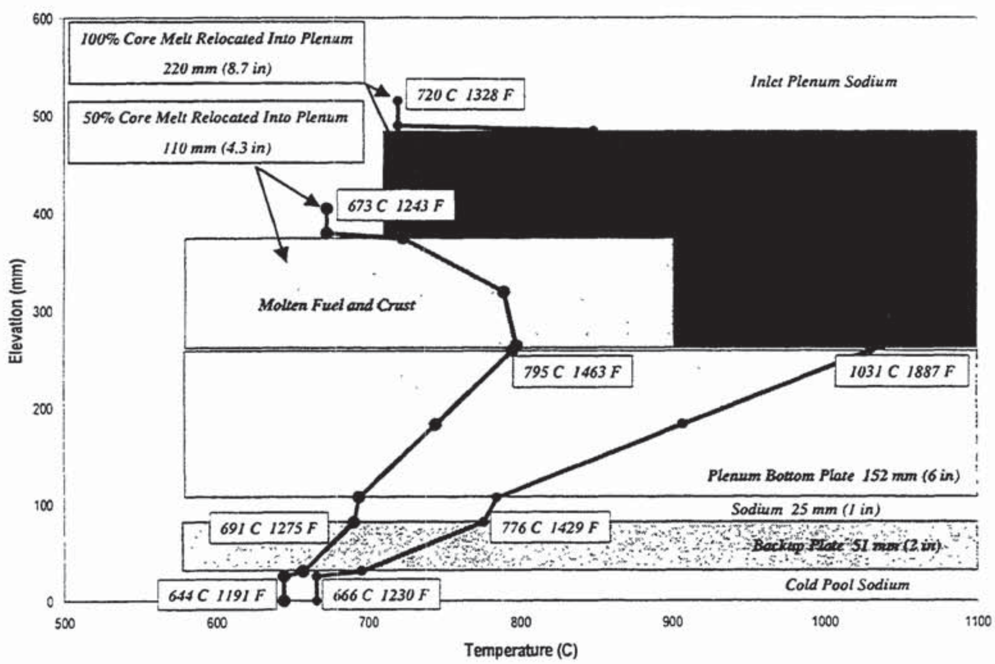

Figure 2. TEMPERATURE PROFILE IN MELT AND INLET PLENUM STRUCTURES AT 30 HOURS (PEAK TEMPERATURES)