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**A RISK CHARACTERIZATION OF SAFETY RESEARCH AREAS
FOR INTEGRAL FAST REACTOR PROGRAM PLANNING***

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Abstract

This paper characterizes the areas of Integral Fast Reactor (IFR) safety research in terms of their importance in addressing the risk of core disruption sequences for innovative designs. Such sequences have traditionally been determined to constitute the primary risk to public health and safety. All core disruption sequences are folded into four fault categories: classic unprotected (unscrammed) events; loss of decay heat; local fault propagation; and failure of critical reactor structures. Event trees are used to describe these sequences and the areas in the IFR Safety and related Base Technology research programs are discussed with respect to their relevance in addressing the key issues in preventing or delimiting core disruptive sequences. Thus a measure of potential for risk reduction is obtained for guidance in establishing research priorities.

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Introduction and Summary

Through emphasis on inherent safety, the Integral Fast Reactor (IFR) design concept has attained extremely low levels of risk from core disruption sequences, which have traditionally constituted the primary risk of nuclear reactors to public health and safety. This paper develops a risk framework for these sequences which demonstrates the importance of the key elements of IFR safety research in having established the low risk posture of the innovative designs (SAFR and PRISM) using metal fuels. Its purpose is to provide a qualitative measure of the potential for still further risk reduction as a guide to the establishment of research priorities for future design innovations.

An overview of the top-level types of safety research can be obtained by looking at the main elements of the project Work Breakdown Structure:

IFR Safety

1. In-pile experiments are used to determine the margins to failure for and validate the analysis of the transient behavior of metal fuel;
2. safety modeling and analyses investigate the safety aspects of metal fuels over a wide range of reactor sizes and are applied to the innovative reactor designs (PRISM and SAFR);

3. out-of-pile tests on both unirradiated and irradiated fuel study key phenomenology in fuel behavior under accident conditions; and
4. analyses of operational transients and local faults determine margins of safety for metal fuels.

Also addressing IFR safety issues are programs in criticals testing, decay heat experimentation and modeling, and seismic and structural analysis.

To assist in highlighting the risk reduction potential of the various research areas, all core disruption sequences are folded into four fault categories: classic unprotected (unscrammed) events; loss of decay heat; local fault propagation; and failure of critical reactor structures. Event trees will be used to help relate the aforementioned research areas to specific issues in preventing or delimiting core disruptive sequences.

Unprotected Events

The first fault category, generic unprotected (unscrammed) initiators, covers events involving loss of flow (LOF), reactivity insertion leading to transient overpower (TOP), and degradation or loss of the heat removal capability or heat sink (LOHS). Because these events can be initiated by occurrences such as severe earthquake that could conceivably result in, for example, simultaneous loss of power to the pumps and severe structural damage to the balance of plant and heat removal systems, combinations of these generic sequences must also be considered.

Figure 1 characterizes accident progression ranging to severe core damage and whole core melting. A 10^{-6} /demand scram failure probability is representative of values used for scram failure in past LMR risk assessments and is consistent with current innovative design values for the primary scram system. For reference, NRC used a value of 3×10^{-5} /demand for automatic scram failure frequency as a guideline for evaluation of LWR scram systems during their ATWS studies. However, secondary scram systems may have significant effect. Depending on design and common-cause considerations invoked by the transient, automatic scram function failure probabilities on the order of 10^{-9} /demand have been calculated. Since the inherent shutdown feature of the IFR designs would allow many hours before sustained high

Figure 1. Probabilities of Shutdown Mechanisms and Resulting Core Damage
States for Unprotected Initiators

temperatures led to incipient fuel failure or major core/vessel structural failures, manual scram considerations are also important.

The emphasis on so-called inherent shutdown characterizes a principal difference in the approach to safety for the innovative designs as opposed to earlier LMR concepts such as CRBR. Operational transient analysis is being used to establish an operating safety strategy that optimizes the use of inherent control with simplifications in the plant protection and control systems and concomitant reductions in event frequencies and scram demands. Important reductions can be achieved in the number of safety channels by taking advantage of the inherent response to the LOHS because upsets in the balance of plant essentially need no waterside signals. Testing of viable strategies will be carried out at the EBR-2 reactor under the auspices of the Inherent Safety/Operability Testing (ISOT) program. Design parameters affecting not only the feedback mechanisms but also physics and thermal-hydraulic values used in accident analyses, can then be more accurately set to optimize the design in terms of risk/benefit.

Inherent shutdown strictly refers not to true shutdown (i.e. permanent subcriticality) but to the reduction of fission power to equilibrium levels below that from decay heat. It is accomplished by the natural (or inherent) response of the reactor's negative reactivity feedback mechanisms to the temperature fields that arise in an unscrammed event. The low risk of core melt sequences for unprotected transients is basically due to the inherent shutdown success probability, which is a function of safety margins and their uncertainties [1]. Accordingly, current research emphasizes enhanced knowledge of the reactivity feedback mechanisms, the reactor transient temperatures and associated structural and thermal hydraulic responses, fuel failure temperatures, and the temperatures at which coherent fuel failure will defeat inherent shutdown. A discussion of each of these research categories is given below.

Reactivity feedback investigations include basic criticals testing of metal fuel full flow transients to provide a benign mechanism for fuel removal and shutdown. In the high temperature range of the time-at-temperature failure regime, TREAT tests provide data on the transient temperature margins to configurations to better define Doppler and sodium void feedback. In-pile TREAT testing [2,3] tracks fuel expansion with the hodoscope to provide a

measure of pre-pin-failure fuel expansion and elongation upon fuel melting. TREAT M-series tests determine differences in the response of ternary and binary fuel types, and D9 and HT-9 cladding types to conditions characteristic of severe TOPs. In addition, irradiated fuel segments are heated in out-of-pile tests in the Fuel Behavior Test Apparatus (FBTA) tests to study the compatibility of irradiated fuel and cladding at elevated temperatures and to determine the axial distribution of fission gas retention in irradiated fuel. This in turn provides insight into the effect of retained fission gas on short-term fuel swelling and elongation and empirical correlations on which to base mechanistic axial expansion modeling in the benchmark and whole core codes. Advanced methods are being developed to investigate fuel assembly motion and core restraint and grid plate expansion to better predict core radial expansion, the most important component of inherent feedback.

Data on failure of fuel at different burnup levels and transient time-at-temperature regimes are provided by the FBTA and M-series tests. In addition, out-of-pile heating tests of whole irradiated pins are being considered for future work. The most likely mechanism for high-temperature fuel pin failure appears to be fission-gas driven rupture, near the top of the fuel column, of cladding thinned by low-melting alloy ("eutectic") formation with the fuel. Investigations are being conducted on the effect of different compositions of uranium, plutonium, and zirconium on the rates of eutectic formation with HT9 and D9 cladding. Fuel behavior benchmark modeling is directed to understanding the failure mechanisms, especially phenomena sensitive to the composition of the ternary alloy and design features affecting fission gas pressurization and transport.

The next shutdown mechanism branch point shown in Fig. 1 involves localized pre-clad-failure axial fuel relocation or post-clad-failure fuel sweepout. The high temperatures of sodium necessary to promote eutectic penetration and failure, and the compatibility of metal fuel and sodium suggest that the eutectic liquid would be incoherently expelled from the fuel pins in small quantities and be swept out in clad failure and on pre- and post-clad-failure fuel motion. These tests have shown extensive relocation of failed fuel to above the active fuel region. The out-of-pile heating of irradiated fuel segments and associated fuel modeling efforts are directed to lower temperatures in this regime.

The SAS code [4] incorporates the aforementioned experimental and modeling insights to provide a comprehensive tool for calculating system response to the unprotected initiators. Significant improvements to the existing SAS models have been made to properly simulate the various feedback mechanisms, especially with respect to the core radial expansion and deformation. These models are being used to characterize the inherent response of various designs of LMR and are being validated by analysis of reactor transients in EBR-II and FFTF. Models are also being constructed that address the phenomena concerned with in-pin fuel relocation, eutectic penetration and fuel failure. These will be embodied in the DEFORM-5 and PINACLE modules of SAS4A. Finally, investigations are under way with a view to applying SAS fuel motion models, PLUTO-2 and LEVITATE, to the metal fuel system so that the sweepout mechanism can be further investigated and the impact of flow rate assessed.

The next shutdown mechanism branch point inhibiting progression to whole core involvement is that of fuel relocation in severely damaged subassemblies. The closeness of temperatures of fuel-clad eutectic melting and of the core region sodium during severe accidents suggests extensive intrasubassembly fuel relocation before blockage formation. The M-series tests have prompted these insights; given the debris coolability characteristics discussed below, permanent subcriticality prior to whole core involvement is probable. Subassembly-to-subassembly propagation mechanisms are also being investigated as a subset of local faults efforts.

In-vessel retention is the next branch point in accident progression. Downward melt relocation through steel and sodium is specifically investigated as part of out-of-pile debris testing. Sizes, shapes, and packing fractions of debris formed by quenching molten metal in sodium are also being studied to form a basis for assessing the intrinsic coolability of such debris beds by natural circulation within the reactor vessel. Higher observed porosities than those typically expected for oxide debris beds suggest full core inventories of metal fuel will be coolable in-vessel. However, the range of alloys that would fall below the core has not been investigated. The ultimate objective of the debris characterization tests is to show how this can be achieved, given the continuing availability of vessel heat rejection.

Despite its low likelihood there is still some small residual risk of vessel breach. Accordingly, limited analysis has been initiated to characterize for risk evaluations core melt fractions, release fractions and energetics for those disruption initiators which cannot be essentially precluded by design.

Generic Loss of Decay Heat Removal Initiators

The second category of sequences covers all initiators leading to the loss of decay heat removal capability. In these sequences, the normal heat removal path through the plant steam loop is assumed to fail and the capabilities of auxiliary reactor vessel cooling systems are evaluated. Because of the independence, passive nature, high reliability, and robustness of the auxiliary decay heat cooling systems of the advanced designs and the large heat capacity of the pool configurations, extremely long grace periods exist for recovery. Thus only extremely improbable catastrophic initiators appear capable of credibly leading to severe core damage.

In the recent CRBR and LSPB oxide-fueled LMR designs, the most probable cause of core melt was loss of decay heat removal capability, with failure probabilities of 10^{-4} and 10^{-5} /yr respectively [5,6]. CRBR relied on several active systems including 1) a forced convection NaK-to-air heat exchanger in the secondary sodium loop and 2) a cooling system in the ex-vessel storage tank. LSPB, without a reactor vessel cooling system, relied mainly upon a two-unit (one in the hot pool and one through the redan) passive Direct Reactor Auxiliary Cooling System (DRACS) to back up the normal heat removal path through the steam generator. The SAFR and PRISM concepts rely on advanced passive means of decay heat removal in addition to the normal decay heat removal path. The significant improvement of these designs is the addition of the Reactor Vessel Auxiliary Cooling System (RVACS-SAFR and RVACS-PRISM) which is basically a large chimney cooling the guard vessel. Decay heat transport includes convection to the reactor vessel and radiation to the guard vessel ("containment"), which is convectively cooled by atmospheric air. As a result significantly lower probabilities of loss of decay heat removal capability are predicted - 10^{-6} to 10^{-8} depending on design and initiating event.

A major threat to the integrity of the reactor vessel cooling systems is failure of the reactor vessel/guard vessel combination. Seismic events have been shown to be important to both designs and the structural/functional fragility of the heat removal paths (reactor vessel, guard vessel, stacks) in earthquakes must be better assessed. Research to assess the nominal performance and reliability characteristics of these systems is ongoing in the Natural Circulation Shutdown Heat Removal Test Facility (NSTF) at ANL. A circumferential segment of the guard vessel/duct wall natural convecting air system has been fabricated. It consists of a 1.3 meter wide by 6.7 m. tall heated zone with a 18 m. insulated chimney. It is capable of operating with guard vessel temperatures to 900 K and heat fluxes to 20 kw/m² (conditions in excess of demands of PRISM and SAFR). Tests of the PRISM configuration have been completed and tests of the SAFR system are presently underway.

Except of course for reactivity feedback and recriticality the core-related phenomenology issues in melt relocation and debris coolability for protected loss-of-decay heat sequences are similar to those for the unprotected events. Figure 2 characterizes the probability of reaching the same levels of damage as described earlier. The branch point probabilities of ~1.0 reflect the small windows in time that would be associated with the indicated transitions through the damage states. Assessment of the probabilities of recovery requires a knowledge of the time it takes for the accident to progress through the indicated events as a function of different levels of decay heat removal. Thus, SASSYS modeling of decay heat removal is used to properly characterize the system response under longer-term (relative to the unprotected scenarios) transient conditions. In this domain it is particularly important that the core temperature be correct and appropriate fuel behavior be modelled. A major concern is the structural response of core support and reactor vessel structures under time-at-elevated-temperature conditions that suggest creep as a dominant failure mechanism. This analysis requires a good characterization of the feedback components of the system as well as detailed models of the various decay heat removal systems. Analysis of reactor experiments will help determine the quality of the simulation.

Figure 2. Probabilities of Heat Sink Recoveries and Resulting Core Damage
States for Protected Loss of Decay Heat Initiators

Generic Local Faults and Subassembly Initiators Local Faults

The third category includes those initiators which cause local power/flow disturbances in either a single subassembly or a local region of the core. The concern is a sequence of events in which fuel failure may propagate first within a subassembly envelope and finally cause loss of coolable geometry in adjacent subassemblies. These events do not have initiators which are readily detectable by instrumentation and amenable to automatic shutdown system inputs; as a result the probabilities associated with shutdown have not been assessed. Initiators generally can be classified as those which increase cladding stress or reduce cladding strength. Due to the reduced stresses in the cladding with metal fuel there is a lower expectation of cladding failure. Local faults analysis also requires consideration of long term operation with failed fuel and stochastic clad failure [7]. Oxide fuel technology [8,9] addressing causes and consequences initiated from outside the fuel pin is applicable to metal fuels. Risk issues which were essentially resolved during the oxide program include blockages, subassembly location errors (flow reduction or excessive power), fission gas blanketing, wire wrap failure, excessive cladding or wire wrap swelling and duct/bundle differential swelling. Although the mechanisms are well established, detailed applications to metal fuel designs are required to ensure adequacy of margins to failure.

The differences between metal and oxide fuel materials include metal fuel compatibility with sodium, high metal fuel thermal conductivity, fission gas retention in metal fuel to about 25 to 30% swelling (about 2 atom % burnup when the porosity becomes interconnected and gas is readily transported to the fission gas plenum), and low structural strength of uranium metal-especially in the porous condition. These features allow use of a sodium bond, and result in low fuel temperatures and a higher power to melt margin than for oxide fuels. However there is a potential for a eutectic to form during off-normal conditions. There is a large metal fuel data base from the EBR-II driver fuel program which used uranium-fissium alloys in Mark I and II fuel designs. Although it is not directly applicable to the latest advanced fuel designs, it does provide significant insights into metal fuel performance.

The likelihood of local fuel pin breach (derived from a "significant local blockage") or of molten fuel released during operation (derived from a "significant over-enrichment event") leading to CRBR core melt was estimated to be 10^{-8} per year, of which 10^{-4} represented the probability of occurrence of

the initiator; another 0.1 represented the Failed Fuel Monitoring Systems (FFMS) failing to initiate a PPS action (by prompting the operator); and the remaining 10^{-3} corresponds to failure to maintain coolable geometry. A bound on the total uncertainty range was estimated to be 300 ($10 \times 3 \times 10$).

For metal fuel faults in advanced reactors, both of the above initiators could be postulated to lead to molten fuel or molten eutectic "ejection" from the pin although the core melt probability is expected to be smaller than for CRBR. Loss of Bond (LOB) must also be considered, either as a result of fabrication defects or stochastic cladding failures. Investigation would involve a range of LOB characterization studies with respect to bond defect, LOB after cladding failure, potential for eutectic formation and rate of cladding penetration, cladding failure mode, mobility of eutectic inside and outside the pin, and interface features (layers) which limit or control the processes. Work on some of these areas are in process. Previous LOB experiments with EBR-II Mark I fuel resulted in no clad failure even in elements assembled without bond sodium in the upper half of the element.

Other areas for study include the effects of cladding material selection, and the fuel/cladding interface, and the capabilities of FFMS to detect failure and interpret the resultant signals in the presence of previous failures. These phenomena are considered in a limited risk context and programs are only necessary to determine the variation in consequences around one or two base cases of interest. Subassembly design is also important, in particular, the degree to which an outlet blockage is considered in the design of the outlet nozzle. If the likelihood of an outlet blockage-induced event can be reduced to that of an inlet blockage failure event, the overall probability of a local fault entry to a whole core accident is greatly reduced.

The purpose of the above research is to demonstrate the inherent tolerance of the design to potential local fault initiators, process inherencies inhibiting propagation, and engineering inherencies restricting the impact of initiators; to enhance the other aspects of inherency; and to assure redundant and timely FFMS signals to the operator. The overall fuel response is such that entry to whole core involvement from local faults appears extremely improbable.

Generic Initiators Involving Failure of Critical Reactor Structure

The fourth category of sequences includes all initiators leading to severe structural failures within the reactor/guard vessel envelope. Reactor structures have five main functions:

1. Maintain reactor coolant flow geometries (e.g. grid plate, redan structure, reactor vessel, air-side RACS/RVACS channel, pump).
2. Maintain boundaries for radioactivity release (e.g. reactor vessel, head, guard vessel).
3. Provide stable support for reactor internals (e.g. pump, core).
4. Promote inherent negative reactivity feedbacks (e.g. above core load pad expansion, grid plate expansion).
5. Minimize reactivity insertion as a result of relative core/control rod movements (e.g. reactor head, reactor vessel).

Items 3, 4 and 5 relate to reactivity considerations both for normal operation and inherent response to unprotected events.

The primary initiators of catastrophic structural failures are:

1. Random failure of components due to design and manufacturing flaws.
2. Longer term effects due to corrosion, premature aging, or creep.
3. Seismic induced failures/ movements.

The random failure of flow structures, in particular the reactor vessel, has been assessed to have a probability of failure on the order of 1×10^{-7} /year and may be important to risk. Premature aging may occur through thermal cycling and irradiation over plant lifetime, degrading the structural capabilities of the core support and other structures within the reactor vessel. Creep-induced failure of the grid plate would distort the core and affect scram reliability. High stress concentrations could lead to vessel failure. Although temperatures are somewhat higher, the LMR pool configurations appear less vulnerable to catastrophic failure than LWR vessels because of the lack of sustained high pressures.

The impact of seismic events upon reactor structures inducing failures or excessive deflection (reactivity effects) is small, but the fragility of the

reactor vessel in a seismic event is a major factor in the overall risk relative to loss of decay heat removal capability. The impact of the seismic event upon pump coastdown and control rod insertion must also be considered.

In summary, the available assessments of critical reactor structures indicate that more attention must be placed on both failure to perform design function and catastrophic structural failure in the development of advanced LMRs. The structural mechanics methods and codes are being developed along with extension or modification of existing codes to properly address the areas listed above.

Summary of Risk Characterization Results and Conclusions

Table 1 presents a summary matrix highlighting the applicability of each area of research relevant to the various stages of accident progression shown in the event trees.

Innovative designs with metal-fueled cores follow past LMR design practices wherein widespread core melting and disruption were made exceedingly unlikely by designing and constructing essential equipment to be highly reliable and by providing redundant and diverse scram systems. Advanced LMRs improve the risk posture by incorporating the unique feedback characteristics of the IFR concept to permit these reactor designs to avoid core meltdown under unprotected transient conditions. Further, because of the low stored energy in IFR fuel (an order of magnitude less than in oxide fuel at similar rating), these systems can survive a sudden and complete rupture of any part of the high pressure core coolant supply system if a normal scram is accomplished.

The resulting exceedingly low probability of core meltdown in IFR systems is illustrated by Table 2 which compares the risk postures of past and current LMR designs vis-a-vis generic initiators capable of leading to severe core damage. A summary of the key reasons for the improved risk posture of advanced metal fueled reactors is as follows:

Unprotected transients: Inherent shutdown provides a high probability (>.999) of reducing total power to decay heat levels, allowing the reactor to equilibrate at temperatures with large margins to fuel failure or structural damage, thereby allowing many hours (vs seconds or minutes) for effecting permanent shutdown.

Table 1. Summary of Research Issues and Programs vis-a-vis Risk Issues

Risk Issue Likelihood of	Research Issues	Research Programs
Entry into potential severe core damage sequences for unprotected events	Initiator frequencies <ul style="list-style-type: none"> - internal event initiators - external (seismic) initiators Operational analysis Scram, flow, and heat sink system capabilities and reliabilities <ul style="list-style-type: none"> - nominal environment - seismic environment 	Vendor initiatives Structural analysis and modeling Vendor initiatives EBR-II and FFTF transient tests Operational transient Analysis Local faults investigations Decay heat removal tests and modeling Structural analysis and modeling
Pre-failure inherent shutdown	Reactivity feedback <ul style="list-style-type: none"> - Doppler and sodium density measurements and modeling - radial/core expansion modeling - fuel axial expansion measurements and modeling - modeling of integrated effects Core power, flow, and temperature response Fuel failure criteria and temperatures	Criticals testing EBR-II and FFTF transient tests Fuel behavior tests and modeling Structural analysis and modeling SASSYS modeling Fuel behavior modeling SASSYS modeling and analysis EBR-II and FFTF transient tests Fuel failure tests and modeling
Shutdown with localized fuel relocation	Fuel failure mechanisms <ul style="list-style-type: none"> - temperature margins - in-pin fuel relocation - in-channel relocation 	TREAT tests Fuel behavior tests and modeling SAS analysis and modeling
Shutdown with intrasubassembly fuel relocation	Intrasubassembly fuel melt and relocation Debris characterization and coolability	TREAT Tests Out of pile debris tests and modeling SAS analysis and modeling
Shutdown with in-vessel containment	Melt breakup, quench, and solidification Debris characterization and coolability	Out-of-pile debris tests and modeling
Entry into/recovery from potential severe core damage sequences for loss of decay heat events	Fuel melt and relocation Debris characterization and coolability Fuel failure-long term creep Decay heat removal Out-of-pile NSTF experiments	Fuel behavior Tests and Modeling
Entry into/recovery from potential severe core damage sequences from failure of reactor structures	Failure to perform design function <ul style="list-style-type: none"> - Loss of flow geometries - Loss of boundary - Reactivity insertion - Impact on inherent shutdown Catastrophic failure Seismic response	Structural Analysis Programs <ul style="list-style-type: none"> - Dynamic - Static/thermal response Seismic technology center Seismic isolation Sloshing
Entry into potential severe core damage sequences from local faults	Adequacy of margins to failure for events initiated outside element Applicability of existing Mark I and Mark II data base for IFR designs Sodium bond defect - lack and loss of bond Inlet and outlet blockages Failure modes and detection <ul style="list-style-type: none"> - cladding material behavior - failure signals to FFMS and FFMS capabilities - operation with failed fuel 	Applications of methodologies developed for oxide program Confirmatory performance and transient testing of Mark III and IV fuels Test program to include fuel distribution, fuel/clad interaction and material mobility for IFR fuel designs Vendor design review and verification Range of limited experimental activities to study areas of specific interest in a risk context, confirming or developing from previous data bases

Table 2. Comparative Overview of IFR Risk Posture vs Earlier Concepts

<u>Generic Initiators</u>	<u>Characteristic Frequencies of Progressing into Fuel Damage Sequences (1)</u>	
	Innovative Designs	Earlier Concepts
<u>Internal Events</u>		
ATWS (LOF, TOP, LOHS)	$<10^{-9}$	10^{-6}
Protected Loss of Decay Heat	$<<10^{-7}$ (2)	10^{-4} - 10^{-5} (2)
Local Faults	TBD	10^{-8}
Failure of Critical Reactor Structures	10^{-6} - 10^{-7}	-- (4)
<u>Seismic Events</u>		
Unprotected LOF, TOP, and LOHS	TBD	--
Protected Loss of Decay Heat	$<10^{-6}$ (2)	--
Failure of Critical Reactor Structures	10^{-6} (3)	--

(1) An NRC safety goal for "large releases" is 10^{-6} /yr.

(2) Numerical values shown for protected loss of decay heat initiators reflect frequencies of loss of all decay heat removal capability. Inequality signs reflect probabilities of recovery efforts preventing fuel damage, very high for internal events and somewhat lower for seismic events.

(3) Value shown for critical structural failures assume a conditional fuel damage probability of 1.

(4) Dashed lines indicate information was either inapplicable or not available.

Loss-of-decay heat: The large thermal mass of a pool reactor and passive systems anchored by the reactor vessel cooling systems provide much higher reliabilities for decay heat removal than for past designs and provide much longer grace periods for response to cooling system faults that might arise.

Local faults: The compatibility of metal fuel and sodium, the much lower temperature fields, and the much-reduced intra-pin temperature gradients combine to reduce both the likelihood of entry into and the consequences of these sequences relative to the oxide designs.

Failure of critical reactor structures: Fragility and seismic isolation analyses are being used to evaluate the designs against the effects of earthquake--the main precursor to such failures. Traditional failure mechanisms, such as creep, are also being investigated.

Despite the improved safety posture, a theoretical possibility of meltdown remains from sudden loss of flow without scram, from complete long-term loss of all decay heat removal systems, or from major structural failures that cause these losses. Accordingly, the Safety Program includes study of phenomena related to core melt sequences, although efforts continue to demonstrate their negligible likelihood of occurrence and risk.

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