LMFBR SAFETY TESTING NEEDS AND
THE CONCEPTUAL DESIGN OF A NEW SAFETY RESEARCH EXPERIMENT FACILITY

by

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1. Introduction

A substantial fraction of the U.S. LMFBR Safety Program is devoted to in-pile testing. Recognition of the importance of this testing has led to continuing studies of the need for new and improved facilities. These studies have shown the need for new facilities and based on these needs ANL is currently engaged in the conceptual design of a new safety research experiment facility (SAREF). This new facility would have the capability of performing safety tests on fuel arrays of up to a full fuel assembly (and possibly larger) in size. The screening of reactor concepts which would meet the experiment needs concluded that a mixed-spectrum reactor of the TREAT type (that is, a system capable of generating shaped power transients without the need for significant heat removal during the test) would complement the available testing facilities so that all of the currently identified safety testing needs could be met. A final decision on the type of system to be built will be made shortly.

2. Experiment Needs

In any discussion of needs for specific in-pile tests, it should be kept in mind that the solutions to specific fast reactor safety problems require more than in-pile tests. A complete discussion of safety research needs is beyond the scope of this paper, but it should be noted that extensive analysis and out-of-pile testing are also required. It must also be kept in mind that new facilities must be viewed as part of the family of facilities that support the program. Therefore, a discussion of new facilities must account for the capabilities and limitations of complementary existing facilities. Although this has been considered in the studies described herein, a detailed description of the testing programs planned for existing facilities is beyond the scope of this paper.

The specific requirements for in-pile tests depend, to a certain extent, upon the approach to fast reactor safety. We have assumed, in outlining the need for a new testing facility, a balanced approach to the three levels of safety. These three levels can be characterized in the following way:

First Level - Design for maximum safety in normal operation and maximum tolerance for safety malfunctions.

Second Level - Provide protection systems designed to assure that incidents leading to off-normal operation will be prevented or accommodated safely.

Third Level - Provide special features in the design as additional assurance that protection to the public is provided even for extremely unlikely or unforeseen circumstances.

The major need for in-pile testing at the first level of safety concern is in establishing fuel performance limits, since this determines the envelope of operating limits for the reactor. Testing of this type will constitute a large fraction of the testing load in the available or planned testing facilities.

At the second level of safety concern, detection and protection will play a major role. While operating experience in steady-state reactors will be a major part of in-pile testing, tests of systems in specific off-normal modes might be required in a new facility.

The needs for in-pile testing at the third level of safety concern are for tests to understand and predict reactor behavior in severe low-probability accident conditions, to establish and confirm bounds on the consequences of such conditions, and to provide a basis for designing and evaluating consequence-limiting systems. This class of tests, which primarily focuses on core behavior during core disruptive accidents, forms the basis for much of the need for new facility capability. This is because information needed for the first two levels can largely be obtained with existing or planned steady-state reactor facilities such as EBR-II and FFTF.

3. Core Disruptive Accidents

The course of an LMFBR whole-core disruptive accident is, in general, controlled by reactivity effects. In turn, the reactivity history depends on a number of interrelated phenomena and, most importantly, those involving core material displacement. In a large LMFBR, sodium voiding can add sufficient reactivity to lead to a rapid power excursion. A fast reactor also is very sensitive to fuel motion, and convective motion can also produce an excursion. In situations in which the reactor

protective system is assumed not to be effective, some form of dispersive fuel motion (reactor disassembly) is required to provide sufficient negative reactivity to terminate a given accident sequence.

The whole-core accidents usually considered in evaluating the safety of LMFBRS can be classified in two broad categories: (1) transient undercooling (usually loss-of-flow) without scram accidents, and (2) reactivity-insertion (or transient overpower) without scram accidents. These accidents can be further subdivided into an initiating phase and a core-disruptive phase. The accident can either terminate at the end of the initiating phase with a largely intact core, or proceed into complete core disruption either by gradual melting of the whole core or by a severe prompt-critical excursion which directly disassembles the core. An unprotected loss-of-cooling accident can itself lead potentially to the addition of significant amounts of reactivity. However, during the initial stages, important differences exist between loss-of-cooling and reactivity-insertion accidents. One basic difference lies in the temporal sequence in which the temperatures of the fuel and cladding rise. In the case of a reactivity-insertion accident, it is the fuel temperature that rises first.

3.1 Transient Overpower Accidents

Inasmuch as coolant temperatures rise relatively slowly in a transient overpower accident, considerable fuel melting can occur before sodium voiding and, at some point, fuel-pin failures (cladding ruptures in the core region) can occur. The reactor protection system should be designed to prevent fuel failures for anticipated transients. After fuel-pin failure, rapid sodium expulsion from the coolant channels potentially can be induced because of the ejection of molten fuel and fission-product gas (for irradiated fuel) from the fuel pins. Particularly in large LMFBRS, sodium voiding can lead to rapid reactivity insertions. However, the expulsion of fuel from the reactor core region through sweepout by sodium vapor and liquid, or through expansion of the fission gas-fuel mixture, can counteract with negative reactivities and tend to terminate the excursion. If this is not effective, the sodium void reactivity, in combination with the initiating reactivity, possibly can lead to a prompt-critical excursion with a resultant core disassembly due to high fuel vapor pressures. The interplay between the positive and negative reactivity effects is complex and depends on the fuel-pin failure times, locations, and modes, and on the magnitudes of the fuel and sodium reactivity worths.

The key problems in the initiating phase of transient overpower accidents that require in-pile testing are:

- (1) An understanding of transient fuel performance and fuel failure thresholds to determine operating design margin.
- (2) The nature and location of fuel pin failures.
- (3) The rate and direction of fuel and coolant movements following fuel pin failure.
- (4) The characteristics of extended motion of fuel, including any freezing or plugging and ultimate coolability of the core, if the initiating phase of the accident does not lead directly to a mechanical disassembly.

3.2 Loss-of-Flow Accidents

Whole-core loss-of-flow accidents potentially can be initiated in several ways. However, the initiating event usually considered is that of loss of off-site power with a resultant coastdown of all pumps and with an assumed failure of the protection system. In such a flow coastdown without scram accident, the first substantial temperature increase occurs in the coolant and the cladding. If no protective action is taken, this leads to sodium boiling near the top of the core. After sodium boiling and ejection from an assembly, a thin liquid film remains on the fuel pin, but the film dries quickly. Following this sodium-film dryout, essentially no heat transfer occurs from the fuel. The fuel temperature increases rapidly while simultaneously the temperature distribution within each pin tends to become flat. In the meantime, the cladding melts and can be followed by fuel melting and slumping, which may add significant amounts of reactivity and lead to a prompt-critical excursion, or by fuel dispersal which will rapidly reduce power.

For a small LMFBFR, the reactor power may remain fairly low during the voiding phase of a flow-coastdown accident, and, as noted above, fuel temperatures may increase little until after the cladding melts. However, for larger reactors, the larger positive sodium-void worths can lead to fairly rapid increases in power because of sodium voiding in the hotter assemblies. For the large portion of the core that remains unvoided, the accident appears much like a reactivity-insertion accident, and fuel melting with consequent fuel failures may occur before sodium voiding in the unvoided channels. Again, the interplay of reactivity effects will strongly determine the detailed sequence of events.

The key problems in the initiating phase of a loss-of-flow accident that require in-pile testing are:

- (1) The rates of sodium voiding and clad movement in the first assemblies to boil.
- (2) The rate and direction of early fuel movements in the first assemblies to melt and specifically whether clad and fission gases can cause early fuel dispersal.
- (3) The rate and direction of fuel and coolant movements in unvoided channels.
- (4) The nature and extent of any cladding or fuel plugs which would inhibit fuel dispersal.

3.3 Transition Phase

For both loss-of-flow and transient overpower accidents, the initial motion of core material may result in a neutronic shutdown. If stable cooling of the partially disrupted core can be established at that point, then the accident would be terminated prior to gross core disruption. If not, the accident will proceed to either a complete and rapid disassembly of the core, if reactivity insertion magnitudes and rates are high, or to a more gradual meltdown.

In recent years, as the methods used to analyze whole-core disruptive accidents have become more detailed, the initial excursions have been shown to be milder than anticipated. The net result of this is that the accident sequence tends to proceed into a gradual meltdown of the core, instead of ending in an early disassembly excursion. This does not mean that a disassembly excursion cannot be induced at some point in the accident, but merely that it is less likely in the initial phase. This is particularly true for smaller reactors in which sodium voiding reactivity is not dominating.

As the subassemblies successively progress through the stage of coolant voiding and melting, the pressures generated may be too low to cause a massive dispersal of molten fuel from the core region. The subassembly duct walls are quickly melted and growing regions of molten fuel and steel begin to form in the hottest portions of the core. This stage of the accident can be accompanied by a number of mild excursions induced by continued fuel motion in other parts of the core, or possibly by the reentry of fuel that was temporarily dispersed by mild pressurizations. This state of the accident has been termed the "transition phase".

Once an accident has entered the transition phase, it appears likely to proceed into whole-core involvement. Thus, ensuring an end to the neutronics events requires attainment of a subcritical configuration through the permanent dispersal of a large fraction of the core. This dispersal can result either from an energetic excursion ending in a classical disassembly, or from more gradual mechanisms such as boil-out and melt-out. It is currently believed that the most likely termination of a core disruptive accident of this type is a gradual fuel ejection.

A common point to all fast reactor whole-core accident scenarios in which loss of active shutdown devices is assumed is that irreversible dispersive fuel motion must occur in order to provide permanent neutronic shutdown. Thus, without regard to initiating events, three generalized accident scenarios involving somewhat different fuel dispersion processes can be identified, although these may overlap.

- (1) Situations in which small amounts of fuel can be removed permanently through hydraulic forces or mild internal pressurization from fission products, cladding vapor, and/or fuel vapor.

Subassembly geometries could be retained and in-place cooling maintained or established under low fission power or decay heat conditions.

(2) Situations similar to (1) above, but in which either permanent fuel removal or in-place cooling are not maintained or established early in the accident sequence and which proceed to the "transition" phase involving a gradual loss of subassembly geometry. Fuel removal and neutronic shut-down would occur either through a series of mild pressurization events or through gradual melt-out and boil-out processes.

(3) Situations in which gross fuel removal occurs due to a general "mechanical" disassembly. In this case, sufficient internal pressurization is implied such that the core is well dispersed. The damage consequences from this type of event can vary significantly depending on whether the pressurization that causes the disassembly is by fuel vapor pressure or gaseous fission products.

During and following the events leading to termination of the accident, the thermal energy generated and stored in core materials can be converted to work energy. Work can come from either the expansion of the core materials or the interaction of these materials with the sodium coolant. Once the energy partition is established, the response of the system can be determined. After the work energy is released and imparted to the system structure and containment, the whole-core disruptive accident enters its final phase involving long-term decay heat removal from the fuel. The final disposition of core materials and the measures required to assure cooling of the core debris are, of course, dependent on the reactor size and design.

The key problems in the transition phase of a core-disruptive whole-core accident which require in-pile tests are:

(1) The nature of extended motion of core materials following a nonenergetic meltdown. It is currently believed these are dispersive and would essentially preclude severe secondary criticalities.

(2) In cases where an accident proceeds to disassembly with sodium in the core, the rate of transfer of heat from the fuel to sodium must be established.

(3) Whether fission gases can exert pressures to disassemble the core at low fuel temperatures must be established.

(4) The ultimate coolability and containment of core debris.

4. Requirements for New In-Pile Testing Facilities

The ideal safety testing reactor would incorporate features that would permit complete prototypical simulation of the LMFBR environment for a wide variety of accident conditions. It has been concluded, after extensive study, that the ideal testing system that can "do everything" is neither practical nor required. The experiment needs can be met in realistic and achievable test systems that provide prototypicality of only those test parameters that are required to meet specified experiment objectives. The following specific test reactor requirements were established as a basis for evaluating and selecting practical test reactor systems.

(1) The system should be able to test at least one full fuel assembly and preferably more than one.

(2) The system must be able to produce shaped power transients of the type shown in Fig. 1.

(3) The test fuel assembly must have a very small power depression across it, which implies a hard spectrum in the test region (but not necessarily elsewhere).

(4) The system should have inherent neutronic shutdown which can automatically compensate for unanticipated reactivity changes in the test fuel assembly.

(5) A short period is desirable for some tests (initial periods of 1-3 msec).

(6) The system should have instrumentation such as a neutron hodoscope to measure fuel motion.

(7) The system must be able to achieve the required energy depositions in the test fuel assembly without damage to itself.

To establish required test fuel energy depositions and evaluate the capability of the design to carry out the required experimental program, seven classes of safety experiments were defined. These seven classes of experiments cover the key problems previously noted.

The required test fuel energy depositions were established by calculating the energy required to conduct each class of test. Two typical power transients are shown in Fig. 1. The first is typical of a loss-of-flow test. The energy deposition in the test fuel required for this test is illustrated in Table 1.

In this test the fuel is first preheated to normal operating temperatures, the flow decay is then initiated and, after fuel has melted, decay heat is simulated for another two minutes. An alternate would be to add a burst at the end before the decay heat simulation to simulate a burst caused by sodium voiding.

Table II shows the test fuel energy requirements typical of a transient overpower test. In this test, flow is maintained constant. The test assembly is brought to normal operating conditions and then the transient overpower is simulated.

Test fuel energy deposition requirements for each of the seven classes of safety experiments are shown in Table III. Tests of transient fuel performance, fuel failure thresholds, and mild overpower transients should be conducted with fuel of normal enrichment (e.g., as discharged from FFTF). Other tests can use full enrichment of the UO_2 in the fuel. Since the test fuel energy deposition is approximately in proportion to the fissionable atom density in the test fuel, the tests on fuels of normal enrichment provide the most severe performance demands.

It would be desirable also to have a short period available for fuel-coolant interaction and effective equation-of-state tests.

5. Choice of Facility Type

New facility testing capabilities will be required to accommodate experiments in most of the areas of concern. This new capability would include testing fuel arrays up to full subassembly size (and possibly larger) with flat power distributions, and rated power test fuel operation for at least 30 sec or, alternatively, decay power levels for longer periods. In consideration of these experiment needs beyond the range of existing facility capabilities, a comprehensive screening and evaluation of test reactor concepts was performed to identify the generic types of reactors, and specific concepts within each type, that can best serve the testing program.

Three generic types of reactors were studied. The first (designated Class I) was defined as a system capable of generating shaped power transients without the need for significant reactor heat removal during the test. An additional requirement that can be imposed on the test reactor is some limited form of heat removal during a test which would permit longer term operation, particularly at lower power levels, in addition to shaped power transients. This type of facility was designated Class II. A further extension of the previous types to provide long-term steady-state operation (for days or weeks) in addition to Class I and Class II capabilities was designated as a Class III facility. Based on these studies a mixed-spectrum Class I safety research experiment facility with the flexibility of changing core loadings to accommodate different types of experiments was chosen to pursue through conceptual design.

6. Conceptual Design of a New Safety Research Experiment Facility

6.1 General

The reactor concept chosen has a central test hole surrounded by a neutron spectrum hardening zone of UO_2 -Cr fuel. A second converter zone composed of $UC \cdot ZrC \cdot C$ (graphite) fuel surrounds the inner converter zone. An outer driver zone is composed of UO_2 -BeO fuel. Helium is circulated through the reactor in closed loops in order to maintain fuel containers cool during test operation and to remove reactor energy after the test. The reactor and primary coolant system (three loops required plus one standby) are embedded in a prestressed concrete reactor vessel of the type and general geometry used in HTGRs.

Each loop has a gas circulator and heat exchanger whose capacity is 66% of that required to cool the fuel containers after a peak power transient. The circulators are downstream of the heat exchangers and pump the helium coolant back into the reactor at a peak temperature of 65°C (i.e., immediately after a transient).

A secondary coolant system transfers heat from the side-arm loop helium coolers to the heat rejection systems outside of the containment buildings.

Figure 2 shows an overall elevation of the reactor facility. The reactor building consists of a steel enclosure containment building structure surrounded by a 0.9 m thick missile protection shield. The steel-liner containment structure extends approximately 34 m above grade and 21 m below grade. Two major openings, one approximately 14.3 m wide by 4.9 m high large freight door which is used to permit large trucks to bring equipment into the building, and a smaller 2.4 m wide by 3.1 m high freight door which is adequate for smaller trucks and handling vehicles to enter the building, are provided. In addition, a personnel air lock door and an emergency air lock door are provided. These building access openings are designed to meet the 2.35 atm internal pressure requirement of the building. The operating floor of the reactor building is at grade level, thus permitting vehicles to be driven directly into the reactor building and to unload equipment at the reactor working floor level. The reactor vessel structure is below grade and extends to the foundation of the building. Intermediate floors are located below grade at elevations where required, i.e., hodoscope room, control rod drive room, and other facility equipment and auxiliary service areas. It is intended that the free space above the reactor floor be maintained as large as practical, and that most areas of the 31 m dia operating floor be serviced by the two 40 (short) ton hoist rotary bridge cranes which have a minimum hook height of about 15 m above the operating floor.

In addition to the reactor building, an assembly building for handling experiments, a reactor services building, and control and support buildings are provided. The assembly building is located adjacent to the reactor building such that the reactor building opens directly into the corridor which passes through this building. This building, approximately 21 m wide and 38 m long, houses the assembly facilities including the 6.1 m wide truck air lock. It is expected that major experiment disassembly operations will be performed at some other facility. This building also houses the toilet-shower-personnel decontamination facilities.

6.2 Reactor Core Configuration

A cross section of the reactor core is shown in Fig. 3. The reactor consists of a two-zone converter surrounded by a nominal 843 mm thick driver fuel zone. Twenty-six control rods are used to control the reactivity of the system and to establish the transient mode of operation. Nominal 600 mm thick axial and radial reflector zones are provided around the core volume.

The reactor vessel for the closed helium system uses a Prestressed Concrete Reactor Vessel (PCRV) which acts as a reinforcement to a steel liner. This also serves as the reactor bulk biological shield. A rectangular cavity in the core, reflector, and in the PCRV is provided so that a neutron hodoscope capable of "viewing" a full-length 217-pin bundle of LMFB-type fuel rods can be placed against the reactor vessel liner and be operated in the facility. A 228 mm OD central test hole, which is part of the converter, provides the reference experiment test space. Provision has been made in the design for replacement of the converter with other converters with larger or smaller central test holes and with correspondingly different fuel loadings capable of testing other test configurations.

Reactor coolant enters the top of the core at a nominal pressure of 6.8 atm and at 65°C. Flow is through the coolant channels on the outside of the converter fuel containers, and in the coolant channels surrounding each of the driver fuel elements.

The converter consists of two zones of different fuel types. The inner converter consists of a nominal 200 mm thick zone of close packed hexagonal $\text{UO}_2\text{-Cr}$ cermet rods, each 12.7 mm across flats. The outer converter consists of a nominal 290 mm thick zone of close packed hexagonal UC-ZrC-C (graphite) composite fuel rods, each 12.7 mm across flats. These fuel rods are illustrated in Fig. 4.

The cermet composition is nominally 10 v/o UO_2 and 90 v/o Cr, and the carbide-graphite composition is nominally 15 v/o UC-ZrC and 85 v/o graphite, heat treated to provide the desired micro-structure. Radially varying fissile loadings are employed in the reactor in order to obtain optimal neutronic characteristics.

The converter and driver zones function as a temporary energy storage device during transient test operation. One of the major requirements for the fuel rod design is that the fuel material be able to sustain repeated high temperature cyclic operation. The fuel also should have as high a heat capacity as possible since test fuel energy deposition is determined by the heat capacity of the reactor fuel. The UO_2 -Chromium cermet, the UC-ZrC-C (graphite) composite fuel, and the UO_2 -BeO fuel meet these requirements best of those fuel materials considered.

The inner converter fuel zone is designed to reach a peak temperature in a transient of 1600°C . The outer converter fuel zone is designed to reach a peak temperature of 2500°C . Each converter fuel zone is surrounded by pyrolytic graphite insulation. This insulation material is capable of providing the necessary retardation of heat flow out of the fuel zone to the helium-cooled converter fuel-container walls (stainless steel), and is also one of the highest temperature refractories of all insulating materials that are compatible with the fuel and meet the physics requirements.

Each converter fuel zone and insulation arrangement is loaded into an individual fuel container. These concentric cylinder fuel containers provide for (a) complete encapsulation of all the fuel material in each converter zone, (b) appropriate clear space for the hodoscope slot or window, (c) coolant channels on all exposed structure surfaces that are in close proximity with the high temperature fuel and reflector materials.

The driver fuel elements are also shown in Fig. 4. They consist of blocks of UO_2 -BeO inside Zircaloy cans, each 96.5 mm square. The cans are separated from the UO_2 -BeO fuel material by insulation. Twenty-six fuel element positions are occupied by control rod guide elements. These elements contain central control rod guide sleeves that allow the control rods to pass through and be guided by the modified fuel sections in these elements.

Surrounding the core is a radial reflector of stacked machined graphite blocks that essentially fill the space between the driver fuel elements and the reactor vessel.

6.3 Reactor System

A vertical section of the reactor system is shown in Fig. 5.

The reactor vessel consists of a Prestressed Concrete Reactor Vessel (PCRVR) similar to that used in the high temperature gas-cooled reactors (HTGR). It consists of a 12.7 mm thick steel liner which is backed up by a thick section of reinforced concrete in the form of a large cylindrical concrete monolith. Reinforcing tendons pass through U-shaped pipes in the concrete as is the practice in the gas-cooled reactor design. The steel liner cylindrical portion consists of a 3.5 m diameter, 12.7 mm thick, carbon steel vessel. The bottom head contains the 26 control rod drive nozzles which extend down through the lower concrete reinforcing segment of the PCRVR into the control rod drive room below the vessel, which is also pressurized to 6.8 atm and therefore a part of the concrete reactor vessel system.

Since the control rod drive room is also designed to be pressurized to the rated reactor coolant pressure (i.e., 6.8 atm), it is also lined with a steel shell and has provision for a personnel and equipment access hatch mounted in one wall of this cylindrical section below the reactor.

As shown in Fig. 5, the reactor vessel also makes provision for the heat exchangers and gas circulators to be located in side arm loops which mount off from the reactor liner. The design of these loops, their closures and the equipment placed in these loops follow closely the proven design of the auxiliary heat exchanger and circulators used in the HTGR designed by General Atomic.

Four side arm heat removal system loops are provided. Each contains a heat exchanger and a circulator capable of handling 66% of the heat load and the reactor flow rate. Three units would

always be in operation and a fourth would be available for standby. Each side arm loop has 0.6 m inlet and outlet lines which penetrate the reactor vessel and are connected to a 1.5 m dia equipment chamber into which the heat exchanger and the circulator are placed.

A schematic of the heat rejection system is shown in Fig. 6.

6.4 Reactor Performance

Figure 7 shows a cross section of a typical single assembly test loop in place in the reactor. The reactor is designed to provide a single central test hole to accommodate experiments of this type. The experiment containment tube also serves as the inner wall of the converter vessel. The tube is designed to withstand the reactor pressure on the outside and to provide a third level of containment for plutonium bearing-experiment capsules and test loops. Each capsule or test loop also would have its own primary and secondary containment system.

When experiments of various sizes are tested, the converter and experiment containment tube can be replaced, since it is the intent of this design to allow the experimenter to "tailor" the reactor configuration to his needs.

While design work is not yet complete, calculations of the reference system have shown that the 7000 joule/gm input to a full test fuel assembly with fully enriched pins can be achieved, and perhaps bettered, with max/min power depressions across the test bundle of less than 1.15 and max/min power depressions across any one test pin of less than 1.05.

The 2500 joule/gm requirement for pins of prototypic enrichment (FF: F driver fuel enrichments) also can be met.

The three zone reactor cores described above would have kinetics capabilities characterized by initial periods of 10 to 20 msec. Intermediate spectrum core loadings also are being investigated which would provide an initial period capability of 1 to 3 msec.

In summary, the three zone Class 1 safety research experiment facility of the type described appears to be able to meet the experiment needs that can now be foreseen. It has the ability to provide the required energy depositions, the test assembly power distributions, and inherent neutronic shutdown which will automatically compensate for unanticipated reactivity changes. It is designed to be easily accessible and flexible to accommodate a wide range of experiment types.

Table I
Energy Requirements for a Loss-of-Flow Test

Event	Duration, sec	Power, W/g	Energy, J/g
Preheat	2	400	800
Coastdown at Normal Power	8	225	1800
Fuel and Clad Melting at Normal Power	6	225	1400
Decay Heat Simulation	120	25	3000
Total			7000

Table 11
Energy Requirements for
Transient Fuel Performance and Fuel Failure Threshold Tests

Event	Duration, sec	Power, W/g	Energy, J/g
Preheat	2	400	800
Normal Power	2	225	450
Fuel Failure Transient	-	-	1000
Contingency	-	-	250
Total			2500

Table III
Energy Requirements for Each Type of Test Needed

Type of Test	Energy Requirement
Transient Fuel Performance and Fuel Failure Thresholds	2500 Joules/gm
Mild Overpower Transients	2500 Joules/gm
Loss-of-Flow Tests	7000 Joules/gm
Extended Fuel Motion Following Core Disruption (Transition Phase)	7000 Joules/gm
Fuel-Coolant Interaction Tests	5000 Joules/gm
Effective Equation-of-State (Fission-gas Effects)	5000 Joules/cm
Postaccident Heat Removal Tests	7000 Joules/gm

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CLASS I SAREF
MAIN BUILDINGS CUTAWAY

CLASS I SAREF
 CORE - CROSS SECTION

DRIVER FUEL ELEMENT

CLASS I SAREF

CLASS I SAREF
SECTION THROUGH
REACTOR VESSEL (PCRv)