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1000-MWe LMFBR ACCIDENT ANALYSIS
AND SAFETY SYSTEM DESIGN STUDY

— Topical Report —
Candidate Protective Features

by
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PREFACE

This report was originally prepared as a technical note to document the work performed in a specific contract activity as soon as the work was completed. The technical editing was limited in order to meet the objective of timely reporting. The report was issued for USAEC-ANL use only, and the intent was to update and consolidate the information from all technical notes in a comprehensive phase report before final publication for public distribution at the end of Phase II.

This plan was changed when the contract was terminated in October 1970 for the convenience of the government. Instead, a final summary report will be prepared, and the previously issued technical notes will be published as formal topical reports. In accordance with the modified plan, this technical note is being published in its original form without further editing or modification except for minor technical corrections and changes in the title and date of issue. Even without updating and technical editing, the report provides detailed information that should be helpful in evaluating and resolving LMFBR safety questions in related areas.

M. W. Croft
Lynchburg, Virginia
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1000-MWe LMFBR Accident Analysis and
Safety System Design Study
Candidate Protective
Features

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ABSTRACT

This report describes the work performed to complete Activity 241 (Review and Identify Candidate Protective Features) of the 1000-MWe LMFBR Accident Analysis and Safety System Design Study. A literature survey was conducted to determine the approach to protection taken by other LMFBR designers, and to identify protective features (especially instruments) that may be available. An updated estimate of possible protective requirements was prepared to help guide the effort. An initial assessment resulted in identification of several possible candidate features. The major unresolved problem area is in determining and providing the protection needed for local accidents that may initiate within a fuel assembly.

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1. INTRODUCTION

1.1. Background

Babcock & Wilcox (B&W) is conducting the Accident Analysis and Safety System Design Study to develop a better understanding of the influence of safety requirements on the design of large LMFBRs. The work is being carried out on a cost-sharing basis under Argonne National Laboratory contract No. 31-109-38-2339; utility companies are participating under separate agreements with B&W.

Although the Follow-On Study provided first-step information on industry-favored LMFBR conceptual designs, the scope of the study did not provide for in-depth resolution of many safety-oriented problems or for detailed guidance of specific research and development (R&D) requirements. To achieve this objective, B&W is analyzing the accidents that determine the design bases for certain protective systems and safety features and preparing conceptual designs for these systems and safety features. The reference design from B&W's 1000-MWe Follow-On Study has been selected as the basis for the studies. The comprehensive work plan divides the total effort into a series of parallel and sequential work activities. This activity is entitled, "The Identification of Candidate Protective Features."

1.2. Objective

The objective of Activity 241 is to identify features that can improve reactor protection. This is the first effort in a series of activities that will subsequently establish functional requirements, select reference systems from among the candidates, perform conceptual design and evaluation of the reference systems, and define Research & Development requirements.

1.3. Scope

Reactor protection ensures the integrity of the fuel by preventing or limiting accidents that might endanger the reactor core. Reactors are normally protected against a set of accidents somewhat more severe than might reasonably be expected to occur, but the emphasis is on credible rather than hypothetical accidents. Protection may be passive, in the form of passive safety devices intended primarily to prevent accident initiation, or it may be active. Sensors monitor plant conditions, logic compares the monitored conditions with predetermined safety limits, and actuators reduce power and/or provide supplementary cooling as needed for fuel protection. The sensing and logic functions are performed by the protection system, power reductions by the reactivity control system, and supplementary cooling by the decay heat removal systems.

In identifying candidate protective features in this activity, the core protection system was emphasized. Less effort was devoted to the other aspects of protection: passive safety devices, reactivity control, and decay heat removal. As indicated in the work plan, the Follow-On Study designs in these latter areas will be carried forward unless significant improvements are identified.

1.4. Follow-On Study Design¹

Traditionally reactor protective systems and devices are designed to prevent fuel damage by any credible accident. However, in the Follow-On Study it appeared that our reference 1000-MWe LMFBR design was incapable of meeting this condition for severe local flow blockage accidents in the core. Passive safety devices were provided to guarantee that a regularly shaped object could not completely block the coolant flow into a fuel assembly. However, it was assumed that a suitably flexible or irregular object might cause sudden and complete blockage of a single fuel assembly. Since such blockage could lead to sodium voiding and fuel damage in the blocked assembly, the design criterion adopted for a single-assembly accident stated that protection must prevent the spread of fuel damage beyond the initially affected assembly. In support of the workability of this criterion, an accident analysis showed that voiding of even the seven most reactive fuel

assemblies in the core would not lead to an unacceptable reactivity insertion. Other initiating conditions for this accident were not analyzed in detail, nor were other consequences.

The criterion for preventing the propagation of fuel damage from a local core accident was the most severe requirement for the protection system of the follow-on design. The normal technique of using a single detector to monitor the initiation of the accident, regardless of where it occurred in the plant, did not appear satisfactory. In most areas of the core, the reactivity worth of the sodium voided from an assembly would be sufficient to cause a scram signal from the nuclear instruments, and such a scram was included in the reference design; however, in the outermost row of fuel assemblies the sodium void coefficient would be too small for this technique to be effective. Other integral instrumentation techniques appeared to be less effective and/or impractical, so that the alternate approach of individually instrumenting all 288 fuel assemblies for protection was adopted.

In B&W's reference design a single thermocouple monitors the coolant temperature at the outlet of each fuel assembly. A partial or complete flow blockage in an assembly would increase the outlet sodium temperature, causing an increase in the thermocouple's signal. Heat transferred radially to adjacent assemblies would produce smaller increases in the outputs of their thermocouples. This concomitant signal from neighboring thermocouples should provide the redundancy required of protective instruments; if not, redundant thermocouples could be used for each assembly. Similar pseudocoincidence would not be available with flowmeters, and the prospect of redundant assembly flowmeters is unattractive in terms of arrangement, installation, maintenance, and economics.

The remainder of the protective instrumentation in B&W's reference design is conventional, including the specification of out-of-core nuclear instrumentation for basic power measurement. These instruments monitor for the bulk loss-of-flow and reactivity insertion accidents typical of power reactors.

Digital signal processing and logic is used in the reference protection system because the individual monitoring of 288 fuel assemblies increases the number of protective instrument channels by an

order of magnitude over that in existing reactors. Digital protection has not been used in an American reactor, and a number of technical and political problems concerning it were identified (but not solved) in the Follow-On Study. The selected system has two large multiprogrammed main frames operated in parallel; it is regarded as a tentative selection pending further work.

Reactivity is controlled in the reference design by 25 identical tantalum absorber rods. Seven are designated as safety rods and are fully withdrawn during operation; the remaining 18 rods are used for shim control and regulation. Each of these two control rod groups has enough reactivity to shut down the reactor.

Each reference control rod drive unit has two synchronous motors which drive the rods through an epicyclic planetary gear train. Both motors normally run continuously, and rod motion is controlled by applying power to one of three sets of windings in one of the motors to select insertion, withdrawal, or no motion. Scram is accomplished by simultaneously opening a magnetic latch and de-energizing and braking one of the drive motors. The other motor provides a power follow for the normal gravity scram.

The continuous operation of the control rod drive motors provided the opportunity for protection-grade reactivity insertions less dramatic than the maximum-capability scram. Controlled, guaranteed power reductions are attractive for moderate conditions that do not require a scram because less severe transients are produced and in some cases the plant can continue to operate safely at reduced power. The reference design has two protective power reduction actions in addition to scrams: fast setbacks, which are controlled insertions of the 18 non-safety rods; and rundowns, which are controlled insertions of three regulating rods at the normal maneuvering limit of 2% power per minute. These graduated protective actions are applied to the various accidents as required.

The reference design does not include an alternate or secondary shutdown means. Such a system was considered superfluous because its requirements would be virtually the same as for the primary shutdown system. The equivalent of a secondary shutdown system is

provided by primary system features such as extra control rods and powered and inertial followup to gravity rod drop.

Two normal decay heat removal loops are provided. Each is fully capable of safely cooling the shutdown core by transferring heat from the primary sodium through a closed NaK loop to the atmosphere. These loops are independent of the plant's intermediate heat transfer loops, but one is used for all normal shutdowns. When decay heat removal is required as a protective function, the protection system ensures that one of the two loops fulfills that function.

The protection system of the reference design also initiates closure of secondary containment penetrations as required for public safety. Since the safety element of the LMFBR program restricts the scope of protection to the core, isolation of the containment is not regarded as a protective function in the Accident Analysis and Safety System Design Study.

Electric power is required for operation of the active protective elements: instrumentation, logic, control rod drives, and decay heat removal pumps and valves. Multiple vital buses backed by battery-and-charger sets or diesel generators ensure the availability of power for protection.

2. SUMMARY

2.1. Fast Reactor Design Survey

The approach to protection taken by other fast reactor designers was examined in a literature survey. All designs generally followed the traditional water reactor approach to accidents involving the whole core. Considerable variation was noted in defining local core accidents that require protection, and in the design of that protection. Designs farthest from the construction stage generally have the most severe requirements and the most complex systems of protection, but the approaches taken for reactors under construction also vary.

2.2. Instruments

The major survey effort was devoted to identifying candidate instruments for initiating protection against local core accidents, since this is the area of greatest uncertainty in LMFBR protection. Several types were designated as candidates for further evaluation. The candidates fall into two categories: local instruments assigned to individual fuel assemblies, and integral instruments where relatively few sensors monitor all fuel assemblies. Complete assurance that any or all of the candidates (including the Follow-On Study reference design) can prevent fuel damage for all credible local accidents cannot be provided because of uncertainties in both accident descriptions and instrument capabilities.

The instrumentation included in the Follow-On Study design for whole-core protection was reviewed in the light of results from the literature survey and accident analyses performed under this contract. This instrumentation was found to be basically satisfactory and practical in terms of the state of the art anticipated in 1980.

2.3. Logic

A review of logic circuit requirements indicated that the potential requirement for numerous core instrument protection channels continues to support the examination of digital protection systems. A limited effort identified several alternate features beyond the two large digital main frames identified in the Follow-On Study. Minicomputers appear to be attractive candidates.

2.4. Reactivity Control

The Follow-On Study reference design for the reactivity control system was assessed in terms of the additional needs being generated under this contract; it is still considered satisfactory. Minor design changes were made to improve the reliability of the control logic systems.

2.5. Normal Decay Heat Removal

Although this is not a primary work area in the contract, two alternates to the Follow-On Study design were identified; both take increased advantage of natural circulation in the primary tank. The follow-on design continues to appear acceptable as well.

2.6. Passive Safety Devices

A candidate design was developed for flow guards that can significantly reduce the probability of severe flow blockages in fuel assembly inlets.

2.7. Protection Requirements

The set of requirements that the reactor protection system is expected to meet was updated. The objectives cannot be fully defined until we obtain a better understanding of local core accidents, but two tentative conclusions were reached: (1) fuel damage should not result from credible accidents affecting the core as a whole, or involving sudden flow blockage of fuel assemblies by foreign objects; (2) insufficient basis exists for defining the protection required for gradual blockage of flow to assemblies or for accidents initiating within an assembly. It appears likely that local fuel damage may not be preventable in the latter case.

3. DESCRIPTION OF WORK

3.1. Fast Reactor Design Survey

The first effort in Activity 241 was a literature survey to determine the protection provided in other fast reactor designs.

3.1.1. Experimental and Test Reactors

EBR-II^{2,3} is a sodium-cooled fast reactor with metallic uranium fuel. The coolant's void coefficient is negative throughout the core. Most protection measurements are bulk rather than local. A 2-of-4 coincidence scram is obtained from thermocouples at the outlet of four selected fuel assemblies, but this scram serves as a bulk core outlet temperature scram. There is no specific protection for individual fuel assemblies. Although EBR-II is a pot-type plant, there is sufficient piping between the core outlet plenum and the intermediate heat exchanger inlets for metering primary flow (with electromagnetic flow meters backed up by orifice meters). Various provisions for nonprotective core instrumentation, including instrumented fuel assemblies, have been and continue to be developed.

FARET⁴ was to have been a sodium-cooled fast test reactor, but the project has been cancelled. Although no individual protection of fuel assemblies was planned, a strong effort was made to provide in-core instrumentation for experiments.

FERMI⁵ is a sodium-cooled fast reactor with metallic uranium fuel and a negative sodium void coefficient. Protection measurements as in the EBR-II are restricted to bulk properties. Since the 1966 fuel melting incident, however, FERMI has developed⁶ and installed a Malfunction Detection Analyzer with ultimate (but not present) potential for protecting against similar flow-blockage incidents. The Analyzer uses a digital computer in conjunction with existing instruments to determine the following: anomalous reactivity, fuel assembly outlet

temperature difference (but only 25 of 91 assemblies are instrumented), and cover gas fission product activity.

SEFOR⁷ is an oxide-fueled, sodium-cooled fast reactor experiment. Partial sodium voiding can produce a small positive reactivity insertion, but the plant is designed to withstand severe transients, including prompt criticality, without damage. Again, no protective instrumentation is used for individual fuel assemblies, but a development program was conducted to provide instruments for in-core experimental data.

The FFTF^{8,9,10} is an oxide-fueled, sodium-cooled fast test reactor which is being designed for construction in the 1970s. The out-of-core protective instrumentation now planned is quite conventional. Documents reviewed to date indicate plans for extensive protective instrumentation for individual fuel assemblies as well as for in-core test loops; this is in contrast to the other reactors just discussed. Such instrumentation is facilitated by providing individual coolant flow ducts for the fuel assemblies. Each fuel assembly is expected to have both thermocouples and a flowmeter connected to the scram system. We have not seen identification of the logic circuits to be used with these instruments.

Tentative protective criteria for local core accidents in the FFTF are based on the results of accident studies and on general operating considerations. The inlet nozzle of the fuel assembly must limit the amount of flow reduction from credible blockages to at least the level at which active protection can prevent coolant boiling; to reach this level requires a reduction of about 60 to 80%. Even a complete blockage at shutdown would be detected by a flowmeter (but not by thermocouples). Other local accidents—local blockage within a fuel assembly, release of fission gas, molten fuel-sodium interaction, and perhaps others—have not been defined well enough to permit specification of meaningful detailed criteria. All such accidents, however, are subject to the basic design objectives: disturbances initiated locally should not prevent shutdown, should not progress mechanically or hydraulically beyond the initially affected assembly, and (through nuclear propagation) should not cause more than moderate damage to adjoining fuel assemblies. As defined for the FFTF, moderate damage is that requiring

repair or replacement of only the failed components, within the duration of a normal refueling cycle. Moderate damage is expected to occur only a few times during the life of the reactor.

The FFTF recently dropped¹¹ plans for a Failed Element Detection and Location system based on monitoring coolant from individual fuel assemblies. Other sophisticated methods of core instrumentation have been reported as under consideration, but their status is not definite.

3.1.2. 1000-MWe Follow-On Studies

The General Electric Company (GE) Follow-On Study^{12,13,14} considered two designs—conservative and advanced. GE's work has indicated that the protection required for these two designs is about the same except that the advanced design requires more reliability to ensure that failure of protection is highly unlikely. GE's analyses have indicated that damage from reasonably credible bulk accidents (reactivity insertion and coolant flow reduction) can be prevented by conventional means.

GE analyzed locally initiated accidents in the core in more detail than did the other follow-on contractors. For sudden flow blockage of a single assembly, protection was considered possible if the flow reduction was less than 50 to 80%, depending on the response time of the sensing instrument (infinity to zero). This range is similar to the 60-to-80% range given above for the FFTF. GE states that rapid flow reductions exceeding this range should be prevented by the design of the fuel assembly's inlet region. GE prefers the use of flowmeters for sensing an assembly's reduction in flow. Thermocouples are a second choice because the response time is longer and because the available temperature signal was shown to vary inversely with increasing blockage; however, thermocouples appear adequate for flow reductions of 50% or possibly more. Other instrumentation was not considered to respond fast enough to prevent sodium boiling, and in case that blockage did lead to boiling and violent expulsion of sodium from the assembly, GE considered noise detection superior to boiling detection.

GE qualitatively considered accidents originating within a fuel assembly. For a pin failure contributing debris that blocked coolant channels (assumed to occur in 0.25 second), it was calculated that

damage would propagate from the center to the wall of the assembly in 5 seconds; some 10 or more seconds would be required before propagation through the wall to an adjacent assembly. Depending on the specific type of blockage, the initiation might not be detectable until the wall failed; however, detection and protection with flowmeters and/or thermocouples was considered readily attainable before more than seven assemblies failed. GE concluded that voiding and melting of up to seven assemblies might be a suitable containment design basis accident for their advanced design approach; stated conversely, that accident may be regarded as the maximum unprotectable accident.

The foregoing discussion does not consider the possibility of an explosive energy release from sodium vapor. GE acknowledged the possibility of such an explosion, expressed the hope that further experiments would show that it would be highly unlikely to generate sufficient energy to damage the walls of assemblies, and suggested design measures to minimize the probability of finely divided fuel being suddenly dispersed into the sodium. These design measures include the use of vented fuel with a negligible pressure differential. GE's calculations also indicated that in the absence of power excursions, local failures cause little fuel melting before failure of the cladding.

Combustion Engineering, Inc. (CE) also provides the standard bulk protection instrumentation in its follow-on design.^{15,16,17} An unusual addition is entrained gas detection instrumentation in the primary loops, which can initiate scram; however, CE acknowledges that the plant design should eliminate the possibility of significant reactivity changes caused by gas bubbles.

As protection against local core accidents, CE provides a reactivity meter, coolant and cover gas fission product monitors, and outlet temperature thermocouples for several individual assemblies. These instruments cannot prevent limited propagation of fuel assembly damage if it should begin because the first lacks sensitivity, the second lacks speed, and the third lacks completeness. Two thermocouples are located at the outlet of each fuel assembly (the protective thermocouples are selected from among these). These thermocouples are scanned by a computer to alert the operator to possible gradual plugging; CE proposes a power oscillation technique to confirm that information.

Thus, CE attempts to detect gradual flow reductions in individual assemblies on a nonprotective basis, and a sudden large blockage is first protected against after limited propagation. The pertinent design criterion is that localized failure may occur in a few assemblies, but the resultant reactivity change may not be as large as inherent negative feedbacks such as Doppler. Credible accidents involving the core as a whole should not lead to sodium voiding or fuel melting.

The Atomics International (AI) Follow-On Study reports^{18,19,20,21} present relatively little information on protection. The treatment of bulk accidents appears to be conventional; protection against local core accidents does not appear to have been reported. AI chose to contain, rather than to protect against, core disassembly accidents, and indicated that an "alternate" approach providing additional core protection (of unspecified nature) could improve plant safety. AI's stated protection criterion is that for each credible accident that could result in fuel melting or in cladding yielding, there should be primary and backup protection. It is not clear that their protection meets this criterion for locally initiated accidents.

The Westinghouse Follow-On Study reports^{22,23} present very little information on protection. The list of scrams developed in Tasks II and III is conventional except for a combined temperature-flow scram and a sodium boiling scram; steam generator sodium-water reactions receive considerable attention. Presumably, the four-module core introduces some complication into the nuclear instrumentation. There are 44 outlet thermocouples in each module—one for each four fuel assemblies—whose stated purpose is to indicate hot fuel assemblies and provide early warning of control rod malfunction or local flow blockage. The relation of these thermocouples to reactor protection is not specified, nor are local core accident analyses presented. No specific criteria for protection were found in the Westinghouse reports although the Task V report states that the protection system will be designed to trip the reactor when necessary to limit fission product release from the core to that normal for vented fuel, and to limit energy release to an unspecified value. The Task V report also states that the core protection system has not reached the stage where the design can be discussed in detail.

3.1.3. Foreign Reactors

The Dounreay Fast Reactor²⁴ is an operating 60-MWt fast reactor experiment in Great Britain. Its protection is conventional except that the center channel outlet temperature of the core is used instead of the bulk outlet temperature. There is no specific protective instrumentation for local core accidents although for information nine additional thermocouples are distributed under the downflow core and 12 more are located under the "breeder."

The Prototype Fast Reactor^{25,26} (PFR) is a 250-MWe fast reactor under construction in Great Britain. Complete information on the protection planned for the PFR has not been obtained, but several interesting features have been reported. The bulk protection instrumentation is believed to be conventional except for a loss of flow measurement in the core. Since the operating primary loop's flow monitors are thermal noise transit time meters with required sampling times of 2 minutes or more, a separate method with greater speed is needed for protection. The selected method is to mount flow monitors on three dummy fuel assemblies at the core's periphery; scram is based on the rate of change in flow.

The PFR will have thermocouples at the outlets of most fuel assemblies, and tubes mixed through an orifice or rotational vanes have been suggested for achieving mixed-mean samples at the thermocouples. These outlet thermocouples are apparently not intended for protection. The basic means of protective detection for local core accidents will be the delayed neutron monitor, with a delay time of 10 to 30 seconds. Since the British desire to locate failed fuel by the same measurement principle used for detection, sodium samples will be taken from each of 186 fuel assemblies. Also considered was an alternate method of detecting the fission gas release that would result from rupture of the PFR's nonvented fuel. The rate of change of the power range nuclear instrument signal and/or reactivity has been proposed to detect extreme events such as large movements of molten fuel.

Two systems that might be capable of detecting local faults before gross cladding failure occurs were reported in 1966 as being studied for application to the PFR. One method uses noise

analysis of the fuel assembly's outlet thermocouple to identify hot coolant slugs associated with one or a few plugged channels in the subassembly; this method was reported to be 10 times more sensitive than the mixed-mean temperature method. The other method under study monitors the acoustic noise associated with coolant boiling, but evidently signal-to-noise problems exist. It is not known whether either system will be installed in the PFR.

The PFR's criterion for local accidents is that a fault condition must be detected as soon as possible after it occurs and certainly, if it can propagate, before damage spreads beyond the affected fuel assembly.

Little information has been obtained on the protection systems of the two French fast breeder reactors—the 20-MWt Rapsodie now in operation, and the 600-MWt Phenix which is under construction. In Rapsodie,^{27,8} outlet thermocouples for individual fuel assemblies are connected to a digital computer system used for protection. This method is reported to have detected a 10% assembly flow blockage at full flow and only 25% power, with a ΔT of only 2 C. Rapsodie has both delayed neutron and cover gas fission product monitors, but they are not protective.

The Phenix^{28,29,8} designers attempt to detect all local deterioration of the core. As for Rapsodie, the principal detection method consists of individual assembly outlet thermocouples, again connected to a digital computer system. The French consider reactivity measurements useful for local accident monitoring, but it is not known whether reactivity will be used as a protection parameter. The Phenix is also expected to have both delayed neutron and cover gas fission product monitors, with some capability for failed fuel location, but it is not known whether these instruments are protective. Neither Rapsodie nor Phenix uses assembly flowmeters for protection.

Information obtained on the protection of other foreign fast reactors is negligible.

3.2. Instruments

The Follow-On Study CSDD¹ and R&D³⁰ reports provide a survey of many candidate instruments; the R&D report alone lists 40 pertinent

references. The LMFBR Program Plan³¹ presents a broader survey of instruments. Since these three documents cover information through early 1968, a bibliography of over 100 additional sources covering information through early 1970, most of it on instruments, is presented in the Bibliography.

The B&W³⁰ and ANL³¹ R&D documents stress the need for upgrading the reliability and environmental resistance capabilities of detectors. This need is tacitly assumed for all of the types of instruments considered and is a prime target of the Program Plan.

Candidate protective instruments are discussed under two headings—those applicable to local accidents and those applicable to whole-core accidents. In this study the first category was emphasized for several reasons: (1) local accidents are not completely defined, so that great latitude is needed in identifying candidate instruments; (2) there are no time-proven instruments that are clearly satisfactory for local accidents (as is the case for most whole-core accidents); (3) the large number of instruments that may be required, and possible problems in installing them near the core, further complicate the identification of satisfactory methods; (4) the B&W Follow-On Study reference design is considered incomplete for local accidents, whereas the instrumentation specified for whole-core accidents continues to appear reasonably satisfactory.

3.2.1. Local Accidents

3.2.1.1. Sequence of Events

Since the mechanisms for initiating and executing local core accidents are not fully understood, the search for candidate instruments to detect these accidents must initially be very broad. As descriptions of the accidents improve, the list of candidate instruments will be refined. Table 1 was prepared to illustrate the general course of a locally initiated accident, the associated variables that could be monitored for protection, and the time sequence of events. The first column describes the events that would occur following sudden blockage of the coolant flow to a fuel assembly. As noted, this sequence applies with some modification to all the other locally initiated accidents. Two variations of the sequence shown in Table 1 are believed reasonable

for the sudden blockage case. First, the order of initial sodium boiling and initial cladding failure (steps 3a and 3b) may be reversed, depending largely on the time scale involved. Second, if the blockage is near the severe end of the range considered, then the temperature increase of step 2 may produce sodium superheating and slug ejection, which directly yields the substantial voiding of step 6 (followed by cladding failure and escape of molten fuel). The slug ejection caused by severe blockage also may result in "chugging" as liquid sodium returns to the voided region and is again ejected.

A flow blockage occurring during a shutdown period is a special case of the sequence shown in Table 1. The accident sequence would be similar, but the step 1 flow reduction would exist before significant power was generated, and the step 2 temperature increase would be spread out over the startup time. A gradually built up flow reduction would also lead to the sequence shown in Table 1, but again the time scale of the first two steps would be extended.

The final case to be considered is local accidents initiating within the fuel assembly. The accident course would again be similar to that in Table 1, but steps 1 and 2 would be on a few-channel basis and the next few steps might occur in a different geometric pattern. Also, the initiating event for the accident might provide a variable that could be monitored directly (for example, a gas bubble).

The first column of Table 1 provides a general guide to the variables that can be monitored and that might accompany any credible local core accident. The last two columns of the table list instruments that might be used to detect the corresponding events in the first column. Column 2 identifies instruments that could be applied to the individual fuel assemblies. Similarly, column 3 lists instruments that could be used on an integral or whole-core basis to detect the corresponding events in individual fuel assemblies. The remainder of this section discusses the protective potential of these various instruments.

Table 1. Possible Instrumentation for
Sudden Local Flow Blockage

Condition	Local Instruments	Integral Instruments
1. Gross assembly flow decrease.	1. Flowmeter (level, rate).	1. Sodium hammer pressure/acoustic noise.
2. Gross assembly temperature increase.	2. Outlet thermocouple (level, rate, compare; noise analysis if asymmetric).	
3a. Sodium boiling and limited voiding.	3. Bubble probe (if severe), ultrasonic boiling detector.	3a. Boiling detector, noise (acoustic, reactivity, nuclear instruments).
3b. Cladding failure begins.		3b. Failed fuel detector (esp DN monitor), gross activity (if severe).
4. Molten fuel begins to escape.		
5. Substantial sodium voiding.		5. Reactivity (from sodium), nuclear instruments (level, rate).
6. Sodium vapor explosion.		6. Pressure, noise (acoustic).
7. Local material redistribution, severe sodium voiding, structural damage (one or a few assemblies).		7. Reactivity.

3.2.1.2. Local Instruments for Local Accidents

Flowmeters

Individual assembly flowmeters can provide the earliest level of protective signals for almost all local accidents. However, no existing fast reactor uses them for protection, and only the FFTF and the GE Follow-On, among the designs surveyed in section 3.1, indicate fairly definite plans for them. This paradox does not arise entirely from the present lack of fully proven protection-grade instruments; there seems little doubt that such instruments could have been developed, and still can be. The biggest problem with assembly flowmeters is associated with installability, maintainability, and redundancy.

Installation of individual assembly flowmeters is difficult, particularly with our under-the-plug refueling concept. Some of the difficulties are cited on page 12-107 of the B&W Follow-On CSDD.¹ Another severe problem appears to exist in providing redundancy, since space for three or more flowmeters per assembly does not appear to be available without a prohibitive penalty. As far as we can determine, not even the FFTF or GE has proposed multiple flowmeters for the individual fuel assemblies. There are plans to provide the FFTF one flowmeter and several thermocouples per assembly. Evidently the rationale for this combination is to obtain the benefits of both flowmeters and redundancy, but the course to be followed if a flowmeter fails is open to question. For example, if protection is still adequate, then the flowmeter is not necessary. The argument that most assemblies still have flowmeter protection, and that the core is thus safer than if no flowmeters were used, appears to be untenable short of a completely probabilistic approach for which data are unavailable. The result of failure of a single flowmeter would appear to be shutdown for maintenance or replacement. As noted, this would require much time.

Conversely, individual assembly flowmeters are attractive in significant respects in addition to the early warning of a local accident. They can detect a shutdown blockage before start-up; it is only necessary to turn on the primary pumps. The same holds

true for blockages that occur during startup or other power transients that could obscure the signal from other instruments. They always tend to read in the proper direction—greater blockage means less flow.* Assurance of operability and some level of calibration can be obtained at any time. For whole-assembly blockages their sensitivity and response time are excellent. However, because of installation and redundancy problems, individual assembly flowmeters are unattractive.

The Follow-On Study CSDD¹ and R&D report³⁰ indicate that the most suitable types of flow detectors are the permanent-magnet flow-through meter (if it can be developed to withstand the outlet temperature above 1100 F, or if it is located at the assembly inlet) and the electromagnetic flow-through meter. One new type was considered in the present contractual work—a probe-type eddy current meter³² being developed for the FFTF. This probe, which is 1 inch in diameter in prototype models, could be inserted into the fuel assembly outlet in somewhat the same manner as a thermocouple is, but redundant installation does not appear feasible.

Thermocouples

A thermocouple is the only type of temperature sensor considered for individual assembly measurements, primarily because of the high probability of successful operation and the relative ease of installation and replacement. As indicated in Table 1, thermocouples can sense the increased temperatures associated with the second level of most local accidents. B&W's Follow-On CSDD discusses many aspects of using individual assembly outlet thermocouples for local accident detection. That material is not repeated here in detail, but several modifications to our reference design should be considered.

The reference design calls for only one thermocouple per assembly, on the premise that the neighbors of a voided assembly will receive heat transferred radially and that their thermocouples will indicate abnormally high temperatures. Pending

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* Thermocouples may indicate a temperature decrease for large sudden blockages.

calculations, it is estimated that the concomitant signal from neighboring thermocouples may not be available until an assembly reaches perhaps the fifth level of Table 1 (substantial sodium voiding), depending on the time scale of the accident. If scram were delayed that long, a prescram local sodium vapor explosion would appear to be possible. It is further noted that assemblies in the outermost row of fuel zone 3 have only two or three neighboring assemblies, so that six concomitant signals are not available if one of them voids. This shortcoming is significant because these are the same assemblies whose voiding is least likely to be detected by the nuclear instruments, since their sodium void reactivity worth is near zero. Finally, very good reliability is required to keep the thermocouples from adversely affecting plant availability through failure. Section 3.7.3. indicates that adequate reliability may be possible, but if three thermocouples were used in each assembly, then the plant availability would likely improve.

The time response required of the thermocouples has a major effect on the expected reliability. The Follow-On CSDD¹ designates an outlet thermocouple scram delay time of 0.3 to 1.3 seconds, consisting of a 0.15 second rod insertion delay time, a 0.15 second delay in the thermocouple's response to a large step increase in sodium temperature, and a 0 to 1 second delay to account for scanning by the computer once each second. Certainly the scanning frequency can be increased, so that the scan delay time can be reduced to 0.1 second or less, and the rod delay (which is quite short) could be reduced somewhat. But the 0.15-second thermocouple delay is a best value; it could be much longer if the temperature change was not a large step. There are four ways to increase the 10-second response time of the reference design's thermocouples: (1) the dry well in which the thermocouples are installed can be removed or opened (this also seriously affects replacement); (2) the thermocouple's junction can be changed from ungrounded to grounded; (3) the diameter of the thermocouple (now 1/16 inch) and its wires can be decreased; (4) the thermocouple's junction can be exposed to the sodium. Each of these changes would reduce the reliability of the thermocouple.

If further analyses indicate that the thermocouple's response time must be reduced, more than one thermocouple

would probably be installed for each fuel assembly. Even if the 10-second response time proves adequate, multiple thermocouples are a definite alternate.

Examination of the capability to calibrate assembly outlet thermocouples in the Follow-On CSDD¹ indicated that various means could be used to ensure their suitability for sudden flow blockages. Concern was expressed about the capability for calibrating them well enough to detect long-term progressive blockages or blockages that occur while the reactor is shutdown. Given decay heat in the core, it appears that shutdown primary flow could be varied to produce significant changes in outlet temperatures. The temperature change between one or two pony motors and six pumps at full speed would be about 100 F. Shutdown tests involving large flow changes would appear to offer a prospect of detecting significant blockages. Observation of the thermocouples' response during these tests, as well as during startups and shutdowns, would also provide periodic information on the response time. It is not clear that these shutdown data can be used to provide automatic protective action, but administrative procedures can probably be used to prevent startup in potentially dangerous situations.

We continue to prefer a change in thermocouple output for a scram signal, as opposed to trips based on level or on comparison of the outputs of several thermocouples, primarily because of the reduced dependence on accurate calibration in the temperature range of interest.

The British have reported³³ a very interesting technique intended to detect flow blockage in a single coolant channel within a PFR fuel assembly. A thermocouple is located in the assembly outlet plenum at a point where turbulence permits all of the individual channel outlet flows to reach it in random time sequence, but where mixing has not yet smoothed temperature differences. Noise analysis of the thermocouple's output identifies hot sodium from channels adjacent to a blocked channel. A thermocouple with a time constant of 1 second or less was reported to meet the objective of detecting full-height blockage of one channel. Sensitivity is reduced by normal power asymmetries, such as a radial power gradient across the assembly, and by core inlet temperature "noise." Multiple thermocouples can be used on

one assembly, with each seeing some of the channels, to improve sensitivity to small flow blockages. The analyses are supported by water tests but not by sodium tests. The time required for signal processing was not reported but appears to be on the order of seconds. An important aspect of this method is that the thermocouple's signal is available for other purposes, such as the scram signal used in B&W's reference design; separate thermocouples are not required. Reference 33 was published in 1963, and the method was reported as still under consideration for the PFR in 1966.²⁵ We have found no record of further work on the method, or of recent intentions regarding PFR application.

Bubble Probes

When the accident progresses to the level of coolant boiling and voiding within the fuel assembly, bubble detectors in the outlet coolant flow may detect the accident. (They could also be applied, on a more general level, to gas bubbles or voids in the inlet plenum.)

Two bubble probes tested by the Japanese³⁴ in water reactors seem suitable in principle for sodium reactors. Both techniques involve placing a probe in the coolant flow so that entrained bubbles or voids impinge directly on it. One type, similar to a solenoid, senses the change of impedance of a coil. The other resembles the end of a coaxial cable, with the center conductor protruding slightly, and produces a pulse when coolant is displaced from the exposed insulator. Both types worked well in water reactor tests. The solenoid type is temperature sensitive but has a larger signal. The Danish³⁵ developed and tested a parallel plate probe whose electrical impedance (resistance and capacitance) varies with the void content of water flowing through it. Temperature (to 95 C) and conductivity compensation were accomplished with a third electrode parallel to the second, with solid water between them.

These three devices generally exemplify the type of probe that could be used for bubble or void detection in an LMFBFR. They have the advantages of simplicity and installation ease, and the major disadvantages of limited information output and difficulty in testing to prove operability. They also detect the accident at a

relatively late level. They may be useful as secondary or tertiary scram instrumentation, but one is reluctant to consider them as the only means of detecting local core accidents.

Ultrasonic Boiling Detection

Aeroprojects, Inc., developed an ultrasonic detector³⁶ that utilizes the only known method of detecting incipient sodium boiling. A purely mechanical rod penetrates the reactor vessel and extends to the desired measurement point. Ultrasonic power is applied to the rod outside the vessel. The amount of energy required to induce sodium cavitation at the end of the rod (detected by change in the rod's vibrational mode) serves as a measure of the sodium's margin to boiling. A great many such devices would be needed to monitor a complete core, and attendant arrangement and installation problems would be unusually difficult. Additional questions concerning response time (presumed to be minutes), accuracy, and drift are unanswered. The device does not appear to be suitable for protective instrumentation.

Summary Comments on Local Instruments

Local—individual assembly—instruments share the basic disadvantages of being required in large numbers at hostile, relatively inaccessible locations. They offer the advantage of detecting local core accidents in their earliest stages, with maximum sensitivity. When credible local accidents are better defined and analyzed, flowmeters may be shown to have a small performance advantage over thermocouples. However, it is almost a certainty that thermocouples will be much easier to install and maintain. We know of no other detection method that is competitive with these devices for primary local detection of local accidents.

3.2.1.3. Integral Instruments for Local Accidents

Sodium Hammer Pressure—Acoustic Noise

The equivalent of water hammer may provide the earliest opportunity to detect local core accidents with instruments that need not be assigned to individual fuel assemblies. If the hydraulic shock associated with blockage is severe enough, then a few

fast-response pressure or acoustic transducers in a relatively accessible location such as the core inlet plenum may produce significant signals. This approach entails the dual problem of determining the signals associated with various blockages and separating these signals from noise that may originate elsewhere in the plant. Certainly there will be a threshold blockage level below which a usable signal is not available. Thus, the method does not appear suitable for protection against all local accidents, but may be valuable for severe sudden blockages.

If a locally initiated accident progresses to the point of severe coolant slug ejection and possibly a vapor explosion, this technique again becomes significant. However, at this level the onset of the accident is no longer detected; rather, the information supplied is that the accident has occurred.

Boiling and Noise Detection

Two approaches are available for detecting the noise associated with sodium boiling: direct monitoring of the acoustic noise, or indirect detection of its nuclear effects by analysis of reactivity or neutron flux noise. Saxe³⁷ has reviewed both methods and Marciniak³⁸ has considered neutronic techniques. Both methods are considered feasible, and data of both types have been obtained on operating reactors; however, there is much uncertainty about the interpretation of the data. Background noise from such sources as pumps, fuel and control rod vibration, and turbulence is difficult to characterize, and so is the signal. Sensors are simple and relatively well developed, and circuitry needs probably can be satisfied by minicomputers (unless cross-correlation techniques are used to extract subtle information). The feasibility of these methods for boiling detection probably will not be established for several years—both theoretical and experimental work are required—but proponents of the techniques are enthusiastic. (See references 37 and 38 and the bibliography).

Failed Fuel Detection

Once the accident progresses to the point of cladding defect (or if it is directly initiated by fuel failure), it can be detected by monitoring for direct evidence of fuel failure. This

measurement method is discussed here as an "integral" method because we doubt the need for complex individual assembly monitoring.

The delayed neutron monitor selected for monitoring failed fuel in B&W's Follow-On Study¹ is still our preferred method. We have not incorporated this instrument into the protection system primarily because of the long signal delay (≈ 10 seconds minimum) caused by coolant transport delay between the core and the sensor. This method was selected instead of the cover gas monitor, also commonly used in fast reactors, for the reasons stated in reference 1.

Porges³⁹ provides a penetrating discussion of the application of delayed neutron monitors to LMFBR plants. However, the capabilities he presents are partially dependent on the initial "burst" effect from bonding sodium in the fuel pin. The use of unbonded vented fuel such as that used in B&W's reference design considerably reduces the suitability of the technique for incorporation in an automatic protection system. Note, however, that the British place primary emphasis on the delayed neutron monitor for automatic protection against local accidents and for failed fuel detection in the PFR (see section 3.1.3).

The bubble probes discussed earlier offer some possibility for detecting fission gas released from failed fuel pins. However, this approach may be usable only for relatively large-scale failures.

A major effort of the FFTF has been to establish the feasibility of xenon isotope tagging for identification of specific failed fuel assemblies (reference 40, for example). In this concept a unique mixture of two or three xenon isotopes is implanted in each fuel assembly during fabrication. A cladding failure releases the xenon to the coolant and cover gas, where it is chemically concentrated and isotopically analyzed to identify the failed assembly. Unfortunately, the concept is not compatible with fuel vents. Anyway, it would not be used as a protective device, but rather as an operator aid.

Reactivity and Nuclear Instruments

Reactivity meters and nuclear instruments come into play in detecting local accidents only after the accident is well along its course (with one exception discussed below). For

assemblies near the center of the core a major portion of the assembly must void (or worse, most of the fuel must move) before the change is detectable. For peripheral assemblies, more than one assembly would probably have to fail grossly to be detectable.

APDA published⁶ a detailed analysis of the anomalous reactivity function of the Malfunction Detection Analyzer recently installed in Fermi. This analyzer repeatedly calculates anomalous core reactivity based on reactivity effects due to changes in the average sodium temperature, fuel heating, control rod positions, delayed neutron kinetics, net burnup of fissionable material, and base condition of the core (the latter is a defined low-power condition). A trial using plant data from a 1966 run during which no anomalous reactivity was believed present, yielded an indicated range of answers from +0.7 to -1.8 cents. This indicates that a threshold sensitivity of about ± 2 cents is available in Fermi. Presumably, a new plant with instruments designed for the purpose could improve that value. Fermi can perform reactivity balances as often as every 1.5 to 2.0 seconds; the frequency is limited by the computer, an IBM 1800.

In comparison, the peak sodium void worth of a fuel assembly in B&W's reference core is about 10 cents, where a cent is worth about half of a Fermi cent because of the different fuels and core designs. Thus, a reactivity meter with the same sensitivity as Fermi's is about 2-1/2 times more sensitive than the reactivity change from completely voiding our most reactive fuel assembly. Outer assemblies could void without being detected by this method.

The anomalous reactivity meter is a relatively complicated device, and some of the instruments required for its use—notably rod position—are not traditionally used as protective instruments. B&W concurs in APDA's planned program for obtaining on-line operating experience before connecting the device to the automatic shutdown system. Fermi's experience will be significant in determining prospects for this and other complex devices.

Much simpler reactivity instruments are available for determining shutdown reactivity. These instruments are valuable operator aids in ensuring that the fuel loading is correct and that anomalous effects such as fuel ratcheting have not developed.

The value of the nuclear instruments in protecting against local accidents is strongly dependent on the time scale. An insertion of a few cents must occur in a few milliseconds for a rate scram to be effective; otherwise, a delay of several seconds occurs before the level scram setpoint is reached. Thus, during small or slow voiding, more than one assembly is likely to be damaged before the scram unless the propagation of damage between assemblies is quite slow.

Summary Comments on Integral Instruments

Integral instruments share the basic advantage of simplicity: fewer detectors in more accessible locations and fewer protection channels. Paradoxically, most share the disadvantage of complexity; that is, noise analysis and reactivity calculation circuits are complicated. In balance, this combination may represent an improvement over local instruments since more of the complex channels can be ganged in coincidence (2 of 4 instead of 2 of 3) to allow for the higher failure rate expected from the complex circuits. Further, failures may experience a net transfer from detectors in the reactor vessel to accessible equipment cabinets. Ironically, both cases require digital computers: to handle the many protective channels of the local instruments and the complex calculations of the integral instruments.

Integral instruments generally require an additional demonstration of usefulness for local accident detection, especially for sensitivity and data interpretation. From Table 1 it can be seen that they tend to detect the accident at a later stage than do local instruments.

Pending further definition of their capabilities and of the characteristics of local accidents, boiling and noise detectors, and possibly the reactivity meter and delayed neutron monitor, must be regarded as candidate integral protective instruments for local accidents. The nuclear instruments should also be considered since they are already available and may well be usable for this purpose.

3.2.2. Whole-Core Accidents

Topical Reports covering analyses of whole-core flow reductions⁴¹ and reactivity insertions⁴² have been issued under this contract. These are the main classes of whole-core accidents. The results indicate that the reference design protection¹ of the Follow-On Study is satisfactory with one exception: Flow coastdown studies showed that the core's bulk outlet temperature sensors must be placed close to the fuel assembly outlets to avoid excessive scram time delay. Thus, the mixing needed to determine the mean bulk outlet temperature must be obtained by averaging the signals from perhaps a few dozen sensors, since the delay in waiting for sodium mixing in the plenum would probably be unacceptable. The change required is relatively minor.

The whole-core protection for the reference design is well within 1980 capabilities, and the resultant safety continues to appear satisfactory. Therefore, only a small effort was devoted to seeking additional candidate instruments for whole-core protection. Two areas in which possible improvements were identified are described below.

3.2.2.1. Primary Flow Measurement

In B&W's reference design, the only primary loop piping is short runs between the pump outlets and the core inlet plenum. Accurate measurement of primary flow—an important plant parameter—is difficult. Before delivery to the reactor site, the primary pumps are calibrated in a test loop. After installation, the pumps are used to calibrate differential pressure detectors connected across elbows in the primary pipes. These detectors thereafter serve as the reference measurement of primary flow although their output signal is expected to be quite small. For the core power-to-flow scram, flow is measured by differential pressure detectors, which are connected to the core's inlet and outlet plenums and are periodically calibrated against the elbow detectors. The protective signal for loss of a primary pump is obtained from pump power monitors which require little calibration (or alternatively from tachometers). Although the protective instrumentation associated with primary flow is considered satisfactory, improvement in the reference method for primary flow measurement

is desirable. The reference method is important, for it is used in the basic calorimetric heat balance measurements that determine the power calibration of the nuclear instruments, in addition to serving directly as the calibration standard for the protective instruments indicating the differential pressure of the core.

The Liquid Metal Engineering Center studied⁴³ the placement of both venturi and magnetic flowmeters in their Sodium Pump Test Facility. A literature search showed that the desirable uninterrupted flow path from the pump to a venturi meter is 25 to 30 pipe diameters, which would appear to rule out venturi meters for our application. Placement tests of magnetic flowmeters, on the other hand, showed that such meters can be located very close to elbows with negligible error if they are properly oriented. However, the effect of a slightly upstream pump was not reported, and difficult installation and maintenance problems are still involved in attempting to use magnetic flowmeters in the primary piping.

Two further concepts for measuring primary flow may be applicable. One is the transit time flowmeter; the other uses standard flowmeters in small-diameter bypass lines.

The transit time flowmeter, proposed for measuring primary flow in the pot-type PFR,²⁵ has been investigated by AI in this country.⁴⁴ The meter is based on measuring the transport time of a thermal noise pattern between two thermocouples spaced a known distance apart. (The noise is inherent in the system.) The British report that the thermocouples need only be spaced a few pipe diameters apart, and that measuring times of 100 to 1000 seconds are required. AI extended the concept to envelope two banks of thermocouples, each distributed across the pipe's radius, to measure average velocity. The thermocouple's time constant must be less than the transit time, and variations among thermocouples in a bank must be much less than the transit time. Because of these needs, AI used 20-mil, grounded-junction thermocouples with custom-filed tips to accomplish matching. Additional experiments are needed to establish the usefulness of the transit time method in our reactor.

The French have published two papers^{45,46} describing flow measurements by meters installed in 1/2- and 1-inch

lines bypassing the 12-inch main piping. In one case, the pump head was measured by bypassing a downstream labyrinth; in the other, flow was measured by bypassing a check valve. Experimentation with several bypass connections showed that the best method tried—carefully selected pitot tube taps—showed only a 2% ripple (attributed to roughness in the bypass lines). Absolute calibration is reported as required experimentally. With further development and experimentation, this method may be valuable for our application.

3.2.2.2. Leak Detection

Although sodium leakage does not directly endanger our core, it may lead to chemical reactions that do. Pending near-final plant design, it is premature to spell out the types and locations of leak detectors to be used beyond the detail level of the Follow-On CSDD.¹ The Liquid Metal Engineering Center recently issued a state-of-the-art study of sodium leak detection methods,⁴⁷ which will be of significant value in future design work. Two findings are of interest:

1. The main problem with all contemporary instruments is getting the leaking sodium to the detector.
2. Methods that depend on sodium reaction to produce a detectable effect appear more promising than do contemporary instruments.

3.2.2.3. Summary Comments on Whole-Core Accident Instruments

In contrast to the major problems of LMFBR safety, existing protection against whole-core accidents appears generally satisfactory. Problems do exist in such areas as precise primary flow measurement and leak detection, but they do not heavily influence the basic safety or the feasibility of the LMFBR concept.

3.3. Logic

Protection logic circuits are less important than instruments or actuators in determining the feasibility of protecting LMFBR plants, but they will be of equal significance in the design and licensing stages because of their importance relative to performance, reliability, and

availability. The literature indicates that B&W may stand alone in the U. S. LMFBR community in believing that digital computers may be a necessary part of the protection for large fast-breeder reactors. Accordingly, as a part of the identification of candidate logic features, we have reviewed the thinking that led us to regard digital protection as both inevitable and feasible.

Section 3.2.1 reviews the instruments available for initiating protection against core local accidents, and section 3.7 indicates the probable protective requirements as they appear today. Three alternatives, however ludicrous, emerge from comparing these sections.

First, the need for protecting against local accidents may go away. In this case, the required protection logic would be little different than for current water reactors.

Second, protection against local accidents may be satisfactorily accomplished without assigning protective instruments to each individual fuel assembly (and the utilities may accept whatever economic risk remains). In this case, the required protection logic would again be about the same as for existing water reactors. Digital computers, if needed, would only have to perform functions internal to an instrument channel, such as noise analysis.

Third, it may be necessary to assign protective instruments to each individual fuel assembly, and these instruments may have to be redundant and/or diverse. In this case, the number of protective channels will be greater than in existing water reactors by one or two orders of magnitude. Complex calculations may or may not be required in processing the information from these channels.

If either of the first two alternatives ultimately proves to be correct, then digital protection logic would be used only by choice—not of necessity. If the third alternative should turn out to be the proper one, then digital protection appears necessary. Even more to the point, however, is the need to recognize that the third alternative is very possible and cannot readily be handled by conventional analog-relay logic systems. Stated differently, the first two possibilities can be covered sufficiently for present purposes with no action at all, but candidate means must be identified for the third.

Arguments of this sort convince us that identification of candidate digital protection logic features is necessary at this time. Conversely, the literature describing operating experience and plans for reactor on-line digital computers is growing rapidly. (References 48 and 49 are two of many examples, and each includes several pertinent case histories.) There appears little doubt that digital protection will be commonplace in reactors built in 1980; it is already used in France, and is very close to reality in other countries including the U. S. We conclude that digital protection will also be feasible in reactors built in the 1980's.

The B&W Follow-On Study used two large digital-computer main frames for both protection and control. With this arrangement, questions on the separation of protection and control are difficult to answer; prudence suggests an alternate candidate design with separate computer systems for these two functions. Similarly, although the use of only two main frames can probably be defended against most criticism, problems do result if one main frame fails: Not only does the computer scram coincidence revert to 1-of-1, but testing of many protection system components must be stopped until the first main frame is back in service. Thus, another alternate candidate would use three digital-computer main frames in 2-of-3 coincidence.*

Minicomputers can provide many candidate alternates to large general-purpose machines. A number of available machines priced below \$10,000 makes it practical to use several in a protection system. Reliability claims are becoming impressive; for instance, MIL standard calculations for the Nova minicomputer indicate a mean time between failures at 25 C of more than 4900 hours.⁵⁰ The increasing use of medium and large-scale integrated circuits contributes to reliability and quick repairability. The entire circuitry of some minicomputers is contained in three or four circuit boards which can be replaced rapidly.

Figure 1 shows an alternate protection system logic section based on the use of minicomputers; it includes several other refinements to

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* Coincidences discussed here are for computer main frames, not instrument channels. See the Follow-On Study CSDD¹ for discussions of the distinction, and other aspects of the reference design.

the reference design. The use of a single outlet thermocouple per fuel assembly was assumed; the 288 thermocouples are distributed among eight scanners, so that a scanner failure will not cause a loss of signal from two adjacent thermocouples. The minicomputers function primarily as special-purpose machines rather than as general-purpose computers. Logic functions are clearly separated—two voter stages are used—to facilitate testing and fault location and to reduce the propagation of faults through the system. A general 2-of-4 coincidence is used. Finally, the system feeds dual scram buses (see section 3.5.1). Many of these features are also applicable to systems with large-scale computers.

The nuclear industry has begun to examine the methods used by the aerospace industry to achieve high system reliability. The Nuclear Safety Information Center at ORNL now includes aerospace and reliability sources in the material it reviews in compiling abstract cards. A group at Pennsylvania State University led by M. A. Schultz, an experienced protection system designer, is investigating the application of these sophisticated techniques to reactor protection system design.⁵¹ They present a number of techniques, such as increased emphasis on multiple-point static testing rather than elaborate dynamic testing, that further enhance the attractiveness of digital protection. These techniques, however, are beyond the scope of this Activity.

Contractual limitations prevent full exploration of the various candidate protection logic features. Looking ahead to Phase III, our objectives are to evaluate the overall performance of the protection, to estimate the effect of failures on plant availability, and perhaps to estimate roughly the reliability of the protection. We expect to use material from the Follow-On Study CSDD, this study, and the continuing effort on Activity 251 to prepare a representative design by the end of Phase II. Also, the required protective instrumentation is expected to be further defined as the work progresses; this will aid significantly in determining the necessary logic.

Figure 1. Minicomputer Protection System Logic

3.4. Reactivity Control

3.4.1. Reference System

The reactivity control system for B&W's Follow-On Study reference design is briefly summarized in section 1.3; the Follow-On Study CSDD¹ describes the system further. The work plan for the present contract specifies that the basic design shall remain unchanged from the Follow-On Study, and that interfaces, performance requirements, and the graduated protective action concept shall be reassessed.

A single-fault analysis performed during the Phase I Malfunction and Fault Survey revealed a potentially unsafe common-mode fault in the scram circuitry. Figure 2 shows the potential fault. If the proper voltage is applied to either wire designed X, then the power-interrupt action of the scram breakers will be defeated and no rods will scram. (Voltage applied to wire Y is not serious since scram of only a single rod is affected.) Although application of such voltage is unlikely, and could be guarded against to a high degree of certainty by design and testing, a better solution would probably be to eliminate the source of the potential fault. This is feasible since either the group of seven safety rods or the group of 18 shim-regulating rods contains sufficient reactivity to scram the reactor. Thus, the control rods can be divided into two groups, each with its own scram breakers and primary motor power supply. Both groups would be commanded to scram by a scram signal. But if either group failed because of the type of fault discussed above, then the scram would still be effected by the other rod group. Figure 1, which shows a conceptual minicomputer protection system, illustrates the interface between the dual scram group and the protection system.

Figure 2 illustrates another minor undesirable feature of the reference design. If the upper vital bus failed, then scram breakers 1 and 3 would open before another vital source could be connected to them, and the rods would spuriously scram; the same effect would result through the action of breakers 2 and 4 if the other vital bus failed. This type of spurious scram can be prevented if breakers 1 and 2 share one vital bus and breakers 3 and 4 share another, or if the four breakers are distributed among four vital buses.

Figure 2. Rod Power and Control

No other desirable changes have been identified for the reference reactivity control system. However, the possibility remains that local core accidents could require faster scrams than the 150 ms delay specified for the reference system. Also, future, more detailed analyses of other accidents might create a need for faster fast setbacks. Such needs will be considered as they arise.

3.4.2. Secondary Shutdown System

As indicated in the summary of section 1.3, the Follow-On reference design has no identifiable alternate or secondary shutdown system. The reason is that no set of shutdown criteria basically different from those of the primary shutdown system has been identified for the LMFBR. Our approach here is consistent with the traditional, since two recent surveys^{52,53} indicate that no fast or liquid-metal-cooled reactor has had a secondary shutdown system. Whenever such systems have been judged to be of value, the requirements—response speed, initiating signals, actuation means, control method—have been very different from those of the primary system. Light water reactors often have secondary shutdown systems to ensure that a scrammed reactor remains shutdown; these systems typically consist of a slow, manually initiated injections of poison solutions to offset core cooling and xenon and samarium decay effects. There is no significant comparable effect in an LMFBR.

3.5. Normal Decay Heat Removal

The reference decay heat removal system for B&W's Follow-On Study is summarized in section 1.3 and fully described in the Follow-On CSDD.¹ As was the case for reactivity control, the effort devoted to normal decay heat removal is limited by the work plan of the contract. The effort is restricted to design changes that provide clear benefits and to areas that preserve the integrity of other work done under the contract.

In the normal decay heat removal system, pony motors power the primary pumps to circulate primary coolant through the core. A high degree of redundancy is provided because any one of the six pumps can supply the required coolant flow. In addition, the geometry of the

system provides a natural circulation potential for which no credit was taken in the reference design.

Two possibilities have been identified for taking greater advantage of the inherent natural circulation capability of primary sodium to improve the reliability of the system. Both possibilities involve simplifying the coolant path between the core and the heat exchanger. This path is now as follows: After passing through the core, the hot sodium must reverse its direction (owing to the static head) and proceed down through the inoperative intermediate heat exchangers. Again, it must reverse its direction (owing to the density difference) and flow to the inlet of the emergency NaK coolers. The sodium is then cooled in the shell side of the heat exchanger, flows out into the cold leg of the primary system, and returns to the primary pump suction.

The first possibility involves the removal of decay heat directly through the intermediate heat exchangers rather than through separate heat exchangers. Under these conditions the steam generators would reject the heat to the condensers. The feasibility of this alternate depends on the capability of the steam generators to perform at low power without introducing attendant hydraulic instabilities at low flows.

The second possibility requires relocation of the NaK heat exchangers to the hot leg of the primary system. The prime factors influencing this arrangement are (1) establishing flow communication between the heat exchanger outlet and the core inlet without unduly penalizing full-power operations, and (2) ensuring operability of the heat exchangers under all conditions short of a DBA.

Complete natural circulation in the NaK system is not practical,⁵⁴ mainly because it would require unduly large NaK-to-air heat exchangers. However, if the air heat exchangers were replaced by NaK-water heat exchangers, it might indeed be possible to meet this objective.

In addition to examining the feasibility of natural NaK circulation, investigation⁵⁴ of the emergency decay heat removal system has produced information pertinent to normal decay heat removal. As a result it was concluded that NaK is the preferred coolant, mainly because of its materials compatibility and low freezing point. Also, one of the candidate emergency systems now being considered is completely independent of the normal system; this eliminates the isolation valves and the

sharing of NaK-air heat exchangers as specified in the reference design. Possible improvement of the normal system's reliability will be considered when the alternate emergency system is evaluated. Other work on the emergency system has tended to reinforce our satisfaction with features of the normal system design established during the Follow-On Study.

3.6. Passive Safety Devices

If primary coolant flow to the LMFBR fuel assemblies were blocked, then the fuel and cladding could become overheated and possibly result in a serious accident. This problem was considered during the Follow-On Study in order to find a means of preventing blockage by regularly shaped objects such as the plates involved in the Fermi accident. The solution was to machine a castle-type configuration in the lower end of each core assembly's support adapter in the lower grid plate region. Thus, if a regularly shaped object lodged over any of the support adapters, coolant could still flow to the assembly through the slots in the castle-type configuration.

Additional study has been performed to determine the potential sources of foreign objects and the shapes that might be expected. In the reference pool design many components are located within the primary tank: the primary pumps, IHXs, decay heat removal heat exchangers, coolant cold traps, neutron shield assemblies, fuel assemblies, blanket assemblies, reactivity control assemblies, fuel storage drums, and the core internal structures. Since all of these are completely or partially submerged in the primary coolant, they are potential sources of objects that could affect coolant flow. Although component failure is obviously required first, many irregular objects, including even some that are flexible, are potentially available for circulation through the system. Thus, additional engineered safety features must be incorporated into the reference design to minimize the possibility of flow blockage.

The design of the lower grid plate and the method of supporting the core assembly permit a fairly simple means of providing the protection required for flow blockage. Flow guards could be added to each of the flow holes in the lower plate of the grid to prevent objects from

passing into the plenum space above. Numerous small flow holes would be machined into each of the guards to maintain coolant flow with a low pressure drop. The proposed concept is shown in Figure 3, in which similar flow guards have been added to the bypass flow holes that provide coolant to the radial shield assemblies. It should be noted that the castle-type configurations on the support adapters would no longer be required; instead, side slots would be added to the control rod support adapters to permit crossflow from the plenum.

In operation, primary coolant enters the core vessel through the inlet nozzles shown in the figure, flows upward through the holes in the flow guards and into the plenum space formed between the plates of the lower grid structure, and then into the fuel assemblies. Any foreign objects entering or originating in the core vessel would be trapped against the flow guards. Regardless of the size or shape of these objects, flow to the core assemblies would not be interrupted because the many unblocked flow guards would permit crossflow through the plenum above. Foreign objects small enough to pass through the holes in the flow guards would present a different problem. Considering that the holes limit the size of such objects to fractions of an inch, the probability of reducing flow to any one core assembly is greatly reduced, if not completely eliminated. Even if small objects did enter one core assembly, they would be trapped against the lower grid in the fuel assembly. Crossflow is again available above that grid and dangerous blockage is extremely unlikely.

This proposed change to the reference design is considered a very attractive candidate protective device. Its use would reduce the probability of flow blockage by foreign objects to an extremely low level and virtually eliminate this as an accident cause.

3.7. Protection Requirements

3.7.1. Background

To help direct the search for candidate protective features, the contract's work plan calls for preliminary identification of required protective actions and functional requirements. These are based on the Follow-On Study, which devoted insufficient formal effort to establishing firm requirements, as well as on the Phase I and early

Figure 3. Primary Coolant Flow Guards

Phase II work done under the present contract. The Malfunction Survey and Fault Trees generated in Phase I and the Accident Analysis and Initiating Condition studies generated in Phase II are pertinent. The results reported here are strictly preliminary since protective functional requirements will be developed under Activity 231.

Hanauer and Walker⁵⁵ distinguish two important elements of reactor protection: (1) reliability, or the probability of a system functioning when called upon; and (2) performance, or the ability of a system to provide protection when it functions as designed. Thus, reliability is associated with the ability to function as called for by the design, and performance is associated with the degree to which the design provides the intended protection. A third element is vital in defining protective requirements for the present contract. This element, which might be called "criteria," "objectives," or "philosophy," involves the definition of intended protection. Hanauer and Walker deplore the lack of available standards for the first two elements, yet at the present stage of LMFBR development the greatest uncertainties exist in the area of defining objectives. Our present status in defining required LMFBR protection will be discussed in terms of these three elements, but in reverse order: objectives, performance, and reliability.

3.7.2. Objectives

Walker and O'Brien⁵⁶ describe the objectives of reactor protection as follows:

The purpose of a protection system is to prevent the reactor from reaching unsafe conditions or to mitigate the consequences of such conditions....It would not be practical to build a protection system that could cope with every conceivable accident situation, so in practice a set of accidents somewhat more severe than might reasonably be expected to occur is selected as the design basis for the protection system....However, nuclear plant protection systems are not designed to provide assured protection for extremely severe accidents with magnitudes greater than those selected, on the assumption that the chances of such occurrences are extremely remote.

Thus, protection traditionally is required for credible accidents.

The Phase I fault trees⁵⁷ and early Phase II analyses—for loss of flow⁴¹ and reactivity insertions⁴²—indicate that the Follow-On reference design provides satisfactory protection for accidents involving the core as a whole. It appears reasonably certain that this protection requirement can be met:

- Fuel damage shall not result from credible accidents affecting the core as a whole.

There is considerable uncertainty about accidents initiating locally within the reactor core; it seems reasonable to divide them into two classes: (1) flow blockages that involve no more than a few fuel assemblies and occur at the assembly inlet (Fermi-type accident), and (2) accidents that initiate within a single fuel assembly. These two classes are similar in that both entail fuel damage, but they are different in a number of other respects.

There appear to be reasonable prospects for the successful argument that design techniques, including passive safety devices, can prevent fuel damage by sudden large-scale blockage of flow to any fuel assembly. In this context "large-scale blockage" means a blockage that reduces flow to the extent that fuel damage in the affected assembly cannot be prevented; the GE and FFTF studies cited in section 3.1 indicate that the region of concern may be a flow reduction of roughly 60 to 70%. For smaller sudden blockages, either the temperature or the flow instrumentation for the individual fuel assemblies apparently can initiate suitable protection. Thus, it also appears reasonably certain that the following protection requirement can be met:

- Credible accidents caused by sudden blockage of flow to fuel assemblies by foreign objects shall not cause fuel damage.

Slower blockage of flow to individual fuel assemblies is a different problem. Possible mechanisms for such blockage include the buildup of foreign objects, corrosion, and other methods that are difficult to identify clearly. The instrumentation requirements for this type of blockage are also more severe than for a partial sudden blockage, for a sudden large signal change is now replaced by a slow change that may be indiscernible from instrument drift. Thus, both the

nature of the problem and the ability to cope with it are more uncertain than for the sudden blockage. Establishment of a protection requirement appears premature at this time:

- The requirement for protection against gradual blockage of flow in a fuel assembly is dependent on further identification of the problem. As a goal, such accidents shall not credibly cause fuel damage.

Finally, the protection required for accidents originating within a fuel assembly must be defined. This area is receiving major attention under the present contract, as indicated in the quarterly progress reports.⁵⁸ Additional work is needed before credibility judgments can be made in this area, and we wish to defer such judgments for several months. It is clear, however, that LMFBRs will not be marketable if there is a reasonable probability that such accidents can extensively damage an investor's plant, even if the public safety is preserved by the containment. A compromise protection requirement, subject to change as knowledge of the problem increases, is as follows:

- Credible accidents originating within a fuel assembly shall at worst damage only a few fuel assemblies. Two goals are also desirable: Fuel damage shall be confined to the initially affected assembly, and the integrity of the primary containment shall not be endangered.

3.7.3. Performance

During the Follow-On Study, the protection against whole-core accidents was based on the present state of the art, apart from details such as the development of instruments and actuators having the necessary reliability and environmental resistance. The accident analyses performed thus far in Phase II indicate that that design provides satisfactory multilevel protection for whole-core accidents. Thus, for the degree of finality needed at this time, there appears to be no need to change the whole-core accident requirements given in the Follow-On CSDD.¹ More detailed analyses in Phase III may lead to minor quantitative changes.

Definition of credible local core accidents depends on further work in the Accident Analysis and Initiating Conditions activities. The use of one thermocouple per assembly with a 10 second time constant, as presented in the Follow-On Study, may not be satisfactory for certain types of credible accidents. Subsequent work in Phase II and probably in Phase III will be required for a reasonable definition of local protective criteria.

3.7.4. Reliability

Since most potential reliability performance requirements relate to details of the design, it would be premature to discuss them before Phase III. It is our intention to satisfy the reliability criteria given in the Follow-On Study CSDD,¹ which reflect the 70 AEC General Design Criteria and the draft IEEE Standard No. 279.

Closely related to reliability is the effect of safe failures and spurious shutdowns on plant availability. In response to an ANL comment on the Follow-On Study CSDD, the effect of failure of an assembly's outlet thermocouple on plant availability was briefly investigated. A rather optimistic value (0.0062 per year) was selected as the thermocouple failure rate based on the first 22 months of Rapsodie operation,²⁹ during which time only one out of 84 outlet thermocouples failed. BNWL⁸ has tabulated the probability of at least x of y units functioning for one year as a function of failure rate. This corresponds to our Follow-On Study requirement: Of the seven thermocouples in an assembly and its six contiguous neighbors, six must be operable to protect the assembly. The tabulated case closest to our situation is for six of seven units with a failure rate of 0.01 per year, which is quite comparable. For that case the probability of success is 0.998. Raising this number to the 41st power ($288 \div 7$), the annual probability of not shutting down for thermocouple replacement in one year is 0.921. Assuming that turbine maintenance provides an annual period for replacing all failed thermocouples, it is estimated that the plant would have to be shut down for thermocouple replacement only once in about 13 years. Many assumptions and some optimism are inherent in this estimate, but it is believed to indicate that individual assemblies can be protected without seriously affecting availability.

4. CONCLUSIONS AND RECOMMENDATIONS

4.1. Candidates for Further Evaluation

Based on the identification and preliminary assessment of the protective features reported in section 3, the following candidate features will be evaluated further under Activity 251:

1. Local Instruments for Local Accidents

Thermocouples (several possibilities).
Flowmeters.
Bubble probes.

2. Integral Instruments for Local Accidents

Fast-response pressure and acoustic detectors.
Boiling detection by acoustic or nucleonic noise.
Delayed neutron monitors.
Anomalous reactivity techniques.
Nuclear instruments.

3. Instruments for Whole-Core Accidents

The only departures from the present design intended for evaluation are primary flow measurement and sodium leak detection.

4. Logic Circuits

Various combinations of the digital features identified in section 3.3.

5. Reference Reactivity Control System

No significant departure from the present design is planned.

6. Alternate Shutdown System

No alternate shutdown system is planned.

7. Normal Decay Heat Removal

The only change to the present design intended for evaluation is the concept of natural circulation in the primary tank.

8. Passive Safety Devices

The only new passive safety device intended for evaluation is the flow guard concept discussed in section 3.6.

Other attractive candidates that emerge during this activity will be considered.

4.2. Problem Areas

Difficulties in defining protection for accidents initiating locally in the reactor core constitute by far the largest problem area. Uncertainties exist at the following levels:

1. The local accidents that are regarded as credible and as requiring protection must be identified.
2. These credible accidents must be described in detail, analytically and/or experimentally.
3. The response of the various candidate instruments to these accidents must be determined, analytically and/or experimentally.
4. In many cases the potential performance capabilities of the candidate instruments must be defined by Research & Development work.

Other problems are much smaller than the local accident problems cited above, particularly concerning their impact on the safety and economic feasibility of the LMFBR concept. Primary flow measurement and implementation of digital logic components are perhaps the most significant.

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October 20, 1970

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PROCESSING OF 1000-MWe LMFBR SAFETY STUDIES

We have been asked by L. W. Fromm, Manager, 1000-MWe Studies LMFBR Program Office, ANL, to print and distribute subject reports as a logical continuation of our involvement in the Follow-on Study Program. A total of 12 reports have been generated by BAW. There may also be a single report from each of 3 contractors, AI, GE and Westinghouse. Each report will be cleared for publication before being sent to DTIE.

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February 19, 1971

PRO:K:029

Mr. Robert L. Shannon, Director
Division of Technical Information Extension
U.S. Atomic Energy Commission
P.O. Box 62
Oak Ridge, Tennessee 27830

Subject: 1000-MWe LMFBR Safety Studies -
Publication of Babcock & Wilcox Topical Reports

Reference: Letter, L. W. Fromm to R. L. Shannon, "1000-MWe LMFBR
Safety Studies - Publication of Contractors' Phase and
Topical Reports," October 16, 1970

Dear Mr. Shannon:

In the reference letter I advised you that we would be transmitting to you for publication a total of twelve Babcock & Wilcox Company Phase and Topical Reports, and possibly three reports from other contractors, generated under the AEC-sponsored 1000-MWe LMFBR Safety Analysis Studies program. With that letter I enclosed one B&W report (BAW-1344, which you have since published), and advised that the remainder would be transmitted to you for publication when received and patent-cleared.

I am enclosing herewith one copy of each of the eight B&W Topical Reports listed below, all of which are now patent-cleared and ready for publication. The covers for these reports should be that used for the previously-issued BAW-1344, except for the changes noted in the table below. The letters heading the columns of the table are keyed to the markings on the attached xerox copy of the cover for BAW-1344.

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BAW-1342	TOPICAL REPORT	Accident Analysis Methods
BAW-1349	TOPICAL REPORT	Candidate Secondary Containment Support Systems
BAW-1350	TOPICAL REPORT	Accident Initiating Conditions Part 1 - Flow Abnormalities
BAW-1351	TOPICAL REPORT	Candidate Emergency Decay Heat Removal Systems
BAW-1352	TOPICAL REPORT	Candidate Primary Containment Safety Features
BAW-1354	TOPICAL REPORT	Candidate Protective Features
BAW-1355	TOPICAL REPORT	Effects of Irradiation-Induced Metal Swelling on the Reference Design
BAW-1360	TOPICAL REPORT	Accident Initiating Conditions Part 2 - Reactivity Insertions

All other parts of the front covers for these reports should remain the same as the cover for BAW-1344.

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Mr. R. L. Shannon, Director
February 19, 1971

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In the reference letter I stated that there would be twelve B&W reports, and possibly three from other contractors. This has now been revised downward to eleven B&W reports and one report from Atomics International. The single remaining B&W report and the AI report will be transmitted to you when received and patent-cleared.

Thank you again for your excellent cooperation in publishing these reports. If there are any questions, please contact me on FTS extension 312/739-2971 or 312/739-4844.

Very truly yours,

L. W. Fromm, Manager
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