MASTER

1000 MWe

LMFBR

Liquid Metal Fast Breeder Reactor

FOLLOW-ON STUDY

COMBUSTION ENGINEERING

TASK II AND III REPORT

**A Reference Conceptual Design
of a 1000 MWe LMFBFR**

Volume 3

Safety and Control Analyses

Prepared for ARGONNE NATIONAL LABORATORY

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Volume III of III

REACTOR TECHNOLOGY (TID-4500)

1000 Mwe LIQUID METAL FAST BREEDER REACTOR FOLLOW-ON STUDY:
REPORT OF TASKS II AND III

A REFERENCE CONCEPTUAL DESIGN OF A 1000 Mwe LMFBR

VOLUME III: SAFETY AND CONTROL ANALYSES

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Approved by :

W. P. Staker
W. P. Staker,
Project Manager

NUCLEAR POWER DEPARTMENT
COMBUSTION ENGINEERING, INC.
WINDSOR, CONNECTICUT

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The material in this report represents the work of numerous personnel on the Liquid Metal Fast Breeder Reactor Project. Those involved and their areas of responsibility are:

M. G. Andrews - Fuels and Materials
W. P. Chernock - Fuels and Materials
J. C. Gilbertson - System Studies
W. J. Mordarski - Fuels and Materials
H. O. Muenchow - Reactor Support Systems
R. C. Noyes - System Studies and Technical Guidance
J. J. Prabulos, Jr. - System Studies
R. S. Stankiewicz - Mechanical Design Studies

W. P. Staker,
Project Manager

1000 MWE LIQUID METAL FAST BREEDER REACTOR
FOLLOW-ON STUDY: TASK I REPORT

PRELIMINARY STUDIES FOR A
REFERENCE CONCEPTUAL DESIGN
OF A 1000 MWE LMFBR

Contract 31-109-38-1999
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Authorized Classifier Date

Approved

W. P. Staker
W. P. Staker, Project
Manager

NUCLEAR POWER DEPARTMENT
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I. INTRODUCTION

The 1000 Mwe Liquid Metal Fast Breeder Reactor Follow-on Study is an extension of 1000 Mwe design studies performed previously for the Atomic Energy Commission.⁽¹⁾ The objective of the Follow-on program is to provide information which will be of assistance to the AEC in the furtherance of the Liquid Metal Fast Breeder Reactor (LMFER) Program Plan. The scope of the work for Combustion Engineering is in the area of the heat generation system of the nuclear steam supply system. The objective of this study of the heat generation system has been to develop a reference design and to identify the research and development programs which are required to achieve the selected design by 1980.

The work to be performed has been divided into four tasks. In Task I, Combustion Engineering began development of a heat generation system concept observing established ground rules and on the basis of an estimation of the state of technology which may be attainable between 1975 and 1985. In formulating this concept, Combustion Engineering considered the effects of fissile and fertile isotope costs, safety and other pertinent factors. Conceptual system design descriptions were prepared for major systems on the basis of preliminary evaluation studies which were used to establish the concept which merited detailed study.

In Task II, the reference concept was refined and developed further by preparation of the remaining system design descriptions for the reference system selected in Task I.

Task III has included an evaluation of the reference design and several parametric studies in order to optimize the design. The report of the work in Tasks II and III is the final report for the reference core design with all applicable systems design descriptions. It includes all descriptive information on the reference design. This report is being issued in three volumes. Volume I contains the Conceptual System Design Descriptions. Volume II, Static Design and Performance Analyses, contains the material in the core system, vessel system and reactor support systems areas leading to the reference design. Volume III, Safety and Control Analyses, contains the material in these areas which serves as a basis for the reference design.

The fourth task will comprise the preparation of a report outlining the recommended research and development, component testing, and new data or information required for design and construction of the selected reference design. In addition to identifying new information, and the research and development program needed to obtain this information, an estimate will be made of the probability of successful achievement in the research and development areas formulated.

This volume of the final report on the development of the reference core covers the Safety and Control Analyses. The trade-off of a cylindrical vs. a modular core called for in the contract has been summarized here because safety appears to be the overriding factor in the trade-off. Much of the physics and economics performance was presented in the Task I report.⁽²⁾ However, although these factors indicated a clear choice for a cylindrical geometry, a final decision could be made only after it was verified through the safety studies that it could meet our safety criteria and that it had no significant disadvantages relative to a modular design.

The reference core safety analyses contain studies of both protected accidents, where the safety system functions, and hypothetical accidents. As a portion of the Task V work, a review was made of safety considerations and assumptions made in selecting the reference concept, and a fault tree has been developed for credible accidents. Included in the fault tree are the initiating conditions for the accidents, the accidents which could result, and the potential consequences.

Other Task V work which is included in this volume of the report of Task II and III work is a review of the AEC General Design Criteria for Nuclear Power Plant Construction Permits, published July 11, 1967. A corresponding set of general safety criteria applicable to LMFER plants has been developed by modification of the original criteria in those areas studied under this contract. As discussed later, the modifications suggested by AIF to the original criteria have been accepted in most cases. In addition to the rationale of the LMFER criteria as developed, specific criteria have been developed for the reference design in order that it will comply with each general criterion.

II. SUMMARY

A trade-off study of cylindrical and modular cores has been made in order to determine the relative merits of each type. The cores compared are optimized to be as realistic as possible. However, they are all unrodded and unmoderated so the effect of these materials is not included. Among items considered are fuel cycle economics, breeding and safety, for both credible and hypothetical accidents.

The results of the trade-off indicate the existence of a definite advantage of the cylindrical core over the modular design in the areas of fuel cycle costs, breeding and mechanical design. The fuel cycle costs for the cylindrical core are approximately 0.3 mill/kwh less than for the optimum modular core. Compound doubling time of the cylindrical core is approximately two-thirds that of the modular core.

In the safety area, there appears to be no clear advantage of either cylindrical or modular core type, since both types meet basic safety criteria and the design requirements for their safety systems are essentially the same. Furthermore, the modular arrangement was found to have a theoretical inherent disadvantage in its response to certain meltdown accidents which tends to balance its theoretical inherent advantage in response to certain voiding accidents. The study resulted in two basic conclusions pertaining to safety:

- a) A flat cylindrical core with a moderately positive sodium void coefficient can be designed to meet the basic safety criteria which are required to obtain an acceptably safe and reliable reactor;
- b) The comparison of the reference cylindrical and modular concepts indicates that it will not be significantly easier to make the modular concept an acceptably safe and reliable reactor, despite its slightly negative sodium void coefficient.

Based on the foregoing, the optimum core selected for the reference 1000 Mwe reactor is a short cylindrical core with a moderately positive sodium void coefficient.

The reference core safety analysis was divided into two major areas: the investigation of protected accidents and the investigation of hypothetical accidents. The analysis involved a detailed investigation of several specific accidents included under these two general types in order to determine their effects on the reactor. These effects include such things as the reactor's inherent reactivity feedback, power perturbations, temperature changes in the fuel, clad, and coolant and any core damage.

Protected accidents are defined as those accidents which are initiated by some identifiable failure and in which some part of the protective system does not fail to respond. One class of protected accidents includes the loss of flow accidents which result from loss of all primary pumps, loss of one primary pump, local subassembly plugging, or a primary coolant boundary failure. Another class of protected accidents results from reactivity ramps resulting from such things as control rod ejection, control rod withdrawal, or gas bubbles traversing the core.

In order to illustrate most clearly the interrelationship of cause and effect for various accidents, a fault-tree diagram has been developed. Based upon the reference core safety studies, this fault-tree graphically displays the accidents with respect to the malfunctions which could occur during operation. It shows the initiating conditions, the resulting accidents and the potential consequences. Such an analysis could serve as a tool for the application of probability analysis since each possible fault is defined.

A number of conclusions have been derived from the analysis of protected accidents described. The main conclusion is that for all the protected accidents studied, the accident normally will terminate without core damage, or where damage is unavoidable, the accident will be terminated before damage can propagate beyond a local region in the core.

The hypothetical accidents studied included massive and rapid core voiding situations and coherent core meltdown situations. For the idealized limiting cases for these accidents in the reference design, reactivity ramp rates of \$20/sec and \$65/sec were calculated for voiding and meltdown respectively. A limited investigation was made of more realistic models for voiding and meltdown and it was found that these models yield considerably lower reactivity insertion rates for the same accidents.

A review has been made of the 70 AEC General Design Criteria for Nuclear Power Plant Construction Permits to determine their appropriateness for application to an LMFER plant of a type similar to the C-E reference design. Where necessary, the criteria were modified as necessary for an LMFER plant. Following the development of each criterion to be suitable for an LMFER, specific criteria were developed for the reference design in order to insure that it meets the more general criterion. Some criteria in areas not covered in the scope of this contract, such as engineered safeguards, are not commented upon.

The control requirements for the reference design are calculated to be approximately 2.72% $\Delta k/k$ for the shim-regulating system and 3.24% $\Delta k/k$ for the safety rod system. These values are based upon having two independent systems for shutting down the reactor while hot. In addition, the safety system has enough reactivity control that it alone will be sufficient even if the reactor is reduced to its refueling temperature. To meet these requirements, there are 13 shim-regulating assemblies each made of ten 1.25 in. OD tantalum rods arranged tangentially in a circle. The safety assemblies, which are normally out of the reactor during operation, are made of eighteen 1.21 in. OD rods containing $B_{10}C$. The safety subassemblies in the inner seed region of the core are of natural boron $B_{10}C$ whereas those in the outer seed region are enriched to 33% Boron-10. Rod heating calculations were made for all rods to determine the amount of cooling necessary.

III. SAFETY STUDIES

Evaluation of reactor safety considerations has led to the general requirement that the combination of (a) the reactor's inherent characteristics, (b) an appropriate protective system, and (c) properly designed containment will yield a system which satisfies the following basic safety criteria:

- 1) Barring complete failure of the plant control and protective system, no credible accident will produce either meltdown of the fuel or significant voiding of the core.
- 2) Even if complete failure of the plant control and protective system occurs, the maximum hypothetical accident will be contained within the shielded region of the primary system.

Safety studies, therefore, are divided into two major areas: (1) study of protected accidents, and (2) study of hypothetical accidents. The first area required study of the response of the reactor and its protective system to credible incidents. The objective was the establishment of the functional requirements of the protective system and the inherent feedback requirements of the reactor which together will satisfy the first criterion stated above.

Study of the response of the reactor to hypothetical incidents is required in order to establish the functional requirements of the containment system and the inherent feedback requirements of the reactor which together will satisfy the second criterion above.

Although it is generally apparent that most fast reactor designs can incorporate a safety system which will meet the above two criteria, it is not obvious whether some of the designs might be inherently safer than others or might meet the requirements with much less design effort or cost penalty. Specifically, there is a question whether any significant safety advantage exists for a core designed to have a zero sodium void coefficient. A trade-off study was made of the safety characteristics during protected and hypothetical accident situations of an optimized zero void effect modular core and a cylindrical design, in an effort to answer this question.

The final part of the Follow-on Safety Studies requires that a review be given the AEC General Design Criteria for Nuclear Power Plant Construction Permits⁽³⁾ and that a revised set of criteria be prepared which are applicable to LMFER Plants based on the experience gained in the design of the reference core.

A. CYLINDRICAL vs. MODULAR CORE TRADE-OFF STUDY

The reactor core trade-off study was performed on two core types, cylindrical and modular, that were surveyed and analyzed during Task I of the 1000 Mwe Liquid Metal Fast Breeder Reactor Follow-on Study. The cores compared were the optimized designs that are discussed in the core survey portion of the Task I report.⁽²⁾ Neither of these designs contained control rods or moderating material; thus, they were compared on a relatively equal basis. The trade-off study involved a comparison of those portions of the reactor that were covered in the follow-on study contract. Among items considered were fuel cycle economics, breeding, safety (credible and hypothetical accidents) and mechanical design of the core.

1. Summary of Results

The results of the study indicate the existence of a definite advantage of the cylindrical core over the modular design in the areas of fuel cycle costs, breeding and mechanical design. In the safety area, there appears to be no clear advantage of either core type, since both cores meet basic safety criteria and the design requirements for their safety systems are essentially the same. Furthermore, the modular arrangement was found to have a theoretical inherent disadvantage in its response to certain meltdown accidents which tends to balance its theoretical inherent advantage in response to certain voiding incidents. This has resulted in the following two basic conclusions pertaining to safety.

- a) A flat cylindrical core with a moderately positive sodium void coefficient can be designed to meet the basic safety criteria which are required to obtain an acceptably safe and reliable reactor.
- b) The comparison of the reference cylindrical and modular concepts indicates that it will not be significantly easier to make the modular concept an acceptably safe and reliable reactor, despite its slightly negative sodium void coefficient.

Therefore, it is evident that there is little incentive for choosing a modular design for the reason of obtaining a non-positive sodium void coefficient or for any other reason, and this has led to the choice of the flat cylindrical core with a moderately positive sodium void effect for the reference 1000 Mwe reactor.

2. Economic Comparison

The economic comparison was completed early in the study and has been discussed in considerable detail in the Task I Report.⁽²⁾ In that study, a series of cores for both geometries was analyzed which resulted in a determination of an optimum cylindrical and a modular core design based on fuel cycle costs, breeding and safety. The core optimization procedure and the results are briefly summarized in the following sections.

a. Optimization Procedures

While Fast Breeder Reactor core optimizations may be approached in several ways, all approaches follow the same basic procedure. Starting from the definition of a basic reactor type (specified power, fuel, coolant, clad, and core arrangement), several specific geometries are established; their performance is evaluated in terms of fuel cycle cost, void effect and breeding performance; and, finally, the arrangement with the best overall performance is selected.

Neglecting control and moderating materials, the core may be characterized for the purpose of physics calculations, by the following parameters: core height and diameter, fuel fraction, sodium fraction, and steel fraction. Specification of four of these five parameters is sufficient since the volume fractions must sum to 1.0. Arbitrary selection of values from this set of parameters presents some difficulties for optimization calculations because quantities such as fuel power rating, fuel pin diameter, and outlet temperature cannot be related explicitly to volume fractions and core diameter. The approach selected for this study avoids this difficulty by adopting core height, coolant velocity, fuel pin diameter, and core temperature rise as the four basic free parameters.

Applying the usual methods of thermal, hydraulic and structural analysis, and taking into consideration the engineering and performance limitations of the core materials, a unique set of core parameters (diameter and volume fractions) can be determined for each set of the above four parameters. Additionally, the fuel power rating and the inlet and outlet temperatures can be determined and much of the basic data for cost analysis can be generated. The process is illustrated schematically in table I.

Table I.
Core Optimization Procedure

FBR Type	Free Parameters	Material and Engineering Limitations	Reactor Characterization	Reactor Analysis	Reactor Performance
Fuel	Core Height	Fuel Conductivity, Density, Max. Temp., Burnup	Fuel Power Rating	Physics	Fuel Cycle Cost
Clad	Fuel Pin Dia.	Clad Conductivity, Max. Temp., Corrosion.	Core Dia.	Thermal	Doppler
Coolant	Coolant Velocity	Fuel Assembly Stress, Corrosion, Vibration, Pressure Drop.	Volume Fractions (Fuel, Steel, Coolant)	Costs	Void Effect
Power	Core Temp. Rise	Hot Channel Factors	Inlet & Outlet Temp.		Pu Doubling Time
Core Arrangement					Specific Inventory

b. Results of Optimization Study

Generally speaking, cores which have a high fuel fraction have good breeding and fuel cycle economic performance. Using appropriate material and engineering limitations, thermal, hydraulic and structural calculations were made to determine the fuel fraction for sodium-bonded carbide cores

as a function of the four free parameters. Results of these calculations are shown in figure 1. Core height and fuel pin diameter were found to be the most important parameters; therefore, the effect of these parameters on reactor performance was studied in the greatest detail.

The safety characteristics of both the cylindrical and modular cores are considered and discussed in the next section. However, some safety considerations were taken into account in the optimization of each core type and, therefore, they are reflected in the fuel cycle costs. The sodium void coefficient is of particular importance for the modular core since the primary incentive for the concept is the possibility of achieving a zero or negative void coefficient. Thus, the optimum modular core was designed to meet this criterion independently of the effect it might have on fuel cycle costs. For the cylinder, it is advantageous to keep the void effect as small as possible as long as its effect on the fuel cycle costs is not significant. Both these criteria were considered in choosing the two respective optimum cores.

A number of cores having different combinations of height and fuel pin diameter were studied for both the cylindrical and modular concepts. A third parameter, radial blanket enrichment was later introduced and included in the optimization. Height ranged between 1.5 ft and 6.0 ft, fuel slug diameter varied between 0.2 inch and 0.4 inch, and radial blanket enrichments were changed from about 2.0 percent to 6.0 percent. The characteristics of the optimum core resulting from each survey are summarized in table II.

Based on fuel cycle costs and the breeding performance of each design, it is apparent that the cylindrical design is superior. The better performance of the cylinder is not unexpected. There was a question, however, as to what the magnitude of the difference would be, and the present results indicate that it is quite large. As mentioned previously, neither system has been completely optimized with respect to all parameters. As the modular arrangement is more sensitive to changes in the radial blanket and permits, in a sense, a wider range of variations than the cylindrical arrangement (the number of modules could be considered a free parameter), it would not be unreasonable to expect the module to have a larger improvement potential in these areas. On the other hand, the survey was performed on equilibrium

Figure 1
Fuel Fraction Variation For 1000 Mwe Carbide - Fueled LMFB

Table II.

Comparison of Optimum Modular and Cylindrical Cores

	Cylindrical	Modular
Fuel Cycle Cost (mill/kwh)	0.29	0.60
Compound Doubling Time (years)	5.9	9.0
Total Breeding Ratio	1.55	1.55
Seed Conversion Ratio	0.996	0.712
Sodium Void Coefficient ($\% \Delta k/k$) (seed plus axial blanket)	+1.70	-0.14
Isothermal Doppler Coef. (Tdk/dT)*		
With Sodium	-0.0046	-0.0054
Without Sodium	-0.0031	-0.0030
Seed Specific Power (kwt/kg)	114.0	141.0
Seed Plus Axial Blanket		
Power Fraction (%)	88.0	61.0
Specific Inventory (kg fiss/Mwe)	3.22	5.81
Core Height (ft)	2.0	4.0
Fuel Slug Diameter (in.)	0.35	0.30
Radial Blanket Fissile Pu		
Composition (%)	5.0	4.5

* See Section III-3-c.

cores at midcycle compositions; no control rods were assumed and the cores were always just critical, i.e., $k_{eff} = 1.0$. The effect on breeding ratio and doubling time of burnup over a fuel cycle (i.e., the amount of excess reactivity needed at the beginning of the cycle) has not been factored into the survey calculations. This effect is a function of the internal conversion ratio and the blanket fissile compositions and volumes with respect to the seed. Preliminary calculations indicate that the modular cores need a larger initial excess $\Delta k/k$ requirement than the cylindrical cores. This would imply

a larger cost penalty for the modular core and might offset any improvement obtained by changing the modular arrangement or its blankets. (See section III-A-4 for a discussion of these variables.) As a result, the fuel cycle cost difference between the modular and cylindrical cores might become even higher. Thus, there is no doubt that the cylindrical arrangement has a substantial economic and breeding advantage over the modular arrangement.

The question that remains is: How much importance should be attributed to the difference in sodium void effect between the two arrangements? This is answered in the following section which discusses safety considerations.

3. Safety Comparisons

The safety system of a Liquid Metal Cooled Fast Breeder Reactor (LMFER) is comprised of the protective system, the emergency cooling system and the containment system. For all LMFER's, the design of these systems combined with the inherent safety characteristics of the core must satisfy the basic safety criteria given previously in the introduction to the Safety Studies (section III). Satisfying these criteria serves two functions. First of all, it guarantees that the nuclear power plant does not constitute a significant hazard to the public; secondly, it reduces to an acceptable level the risk of financial loss from accidental damage to the plant.

Although it is generally apparent that current fast reactor designs can incorporate a safety system which will meet these criteria, it is not obvious whether some of the designs might be inherently safer than others or might meet the requirements with much less design effort or cost penalty. The question now appears to be this: Does a core designed with a zero or negative sodium void coefficient have a significant safety advantage over a core with a small positive coefficient? Obtaining a negative sodium void coefficient in a practical 1000 Mwe core with unspoiled geometry does not seem possible. This forces the core designer to consider cores with spoiled geometries such as modular, annular, or extreme pancake types which provide a large neutron leakage to the blankets to obtain a zero or negative sodium void coefficient. In order to answer the above question, a comparison was made of the safety characteristics during protected and hypothetical accident situations of an optimized zero void effect modular core and a cylindrical design. The cylindrical

core had a positive sodium void coefficient ($+1.7\% \Delta k/k$ for total voiding of seed and axial blankets) and the modular core had a slightly negative sodium void coefficient ($-0.14\% \Delta k/k$, total seed and axial blankets). Results of accident studies indicate that only four of the many possible accidents are noticeably affected by the change in core design. These are:

- a) Credible Accidents: Local subassembly plugging and gas bubbles;
- b) Hypothetical Accidents: Core meltdown and gross core voiding.

Even though there are some differences in the effect of these accidents on the reactor core, as discussed below, it appears as if the safety system in each case will ultimately be the same; that is, there is nothing to be gained in the design of the safety system by choosing a core with a zero or negative sodium void coefficient. The effects of other credible accidents (i.e., where sodium voiding is prevented) and the resulting safety requirements are unchanged between the two core designs. In both cores, the safety system adheres to basic water reactor design philosophy; that is, coolant must always be maintained in the core.

The required Doppler coefficient is based primarily on limiting the explosive yield of the maximum hypothetical accident and requires a value of approximately -0.004 Tdk/dT (sodium removed) for either core. Neutron spectrum softening by some moderator addition is necessary in both cores to meet this Doppler requirement. Thus, the Doppler coefficient can be made about equal for the two cores, resulting in similar inherent responses to accidents where sodium voiding is prevented. The response to credible accidents with no sodium voiding combined with the results discussed below for both credible and hypothetical accidents involving the sodium void coefficient, has led to the conclusion that no significant safety advantages are obvious from a core designed with a negative sodium void coefficient and this, combined with the lower fuel cycle costs (about 0.3 mill/kwh difference) and simpler design of the cylindrical core geometry, indicates that the cylindrical core has a definite advantage over the modular design.

a. Credible Accidents

The two credible accidents in which the sodium void coefficient has any effect on the nature of the malfunction are: 1) the blocking of coolant flow to one or a few subassemblies, subsequently resulting in channel voiding, and 2) the passage of relatively large gas bubbles transported by the flowing coolant through the core. The first type, local subassembly plugging, will most likely occur over long periods of time and would be detected by local outlet temperature thermocouples before a voiding condition is approached. However, if it is undetected, as in the Fermi incident, it would cause sodium boiling and subsequent subassembly voiding. Events preceding sodium boiling are similar for either reactor design. Only the reactivity change following voiding is dependent on the sodium void coefficient and this is very small ($+0.015\% \Delta k$ to -0.007%) for a single subassembly in both core designs. The effect is so small that the voiding of one subassembly can probably not be detected by any of the gross core parameter monitoring systems (i.e., mixed-mean outlet temperature, reactor power, reactivity) (See figure 2). Thus, the first indication of such an incident will be the reactivity effect of melting and displacement of the fuel in the subassembly(s). Therefore, in the event of undetected local subassembly plugging, fuel damage is probably unavoidable in either core and the reactivity effect is small enough so that, even if positive, its overall effect on the core is not detrimental. If the number of plugged subassemblies is increased the elimination of local fuel damage is still not possible. However, early detection will become more probable because of a larger sodium void effect. This will greatly increase the chances of shutting down before any further failure propagation. It can be argued that a positive sodium void effect is more advantageous in this case than a zero or slightly negative value because the gross core parameter monitoring systems will now register a significant change and activate the safety circuits. A negative void coefficient will also cause changes in the core parameters, but for the reference modular core studied these changes are not large enough ($\sim 0.05\% \Delta k/k$ for seven subassemblies) to automatically initiate protective action or significantly reduce the reactor power level.

Figure 2
LOCAL SUBASSEMBLY PLUGGING ACCIDENT-Instantaneous Flow Blockage

It might further be proposed that a modular core with a very large negative sodium void coefficient (regardless of fuel cycle cost) would increase the safety characteristics following a plugging accident. Again, no matter what the magnitude (negative) of the coefficient, the blocked subassemblies will be damaged and the important safety function is to prevent propagation. The only permanent automatic trip in the protective system for a decrease in power (negative reactivity insertion) is the reactivity meter which can confidently detect a change in reactivity of $\geq 0.1\% \Delta k/k$ and scram the reactor. For a single voided subassembly to feed back this much reactivity, a total void effect of $\sim -2.0\% \Delta k/k$ (seed and axial blanket) is required. This is at least twice the value of any of the cores analyzed in the survey.⁽²⁾ Although it is certainly possible to design a core to attain this limit and more, the fuel cycle costs would suffer greatly (i.e., fuel cycle costs of between 0.85 -1.0 mill/kwh compared to 0.30 mill/kwh for the best cylinder). More importantly, it is doubtful whether anything is gained in safety over the reference cylindrical core. An equivalent positive value (i.e., $\sim +0.1\% \Delta k/k$) would not only initiate trip signals for reactivity but also for high power, subassembly outlet temperature, mixed-mean outlet temperature and power/flow. The negative feedback effect without scram will reduce the reactor power $\sim 20\%$, which does not approach a shutdown condition; thus the possibility of failure propagation still exists. Furthermore, to increase the void effect so that the voiding of one or a few subassemblies will introduce enough reactivity ($> -0.5\% \Delta k/k$) to inherently shut the reactor down, will require an overall void effect of between -8 to $-10\% \Delta k/k$. This would be impractical from a fuel cycle cost viewpoint, and probably not feasible technically. Therefore, for the plugging accident, there does not appear to be any reason from a safety aspect to choose a modular core over the cylindrical design.

The gas bubble accident is somewhat different in that no sodium boiling occurs unless the bubble is large and remains in the core long enough to elevate the clad temperature beyond the coolant boiling point. The combined effect of the reactivity change plus the temporary loss of the fuel pin heat sink caused by the passage of the gas bubble determines the maximum clad temperature during the transient. Because the reactivity

change can have a significant effect on the resulting temperatures, especially for large bubbles, it is necessary to investigate several bubble sizes. Furthermore, a positive sodium void coefficient will produce higher temperatures for most size bubbles than either a zero or negative coefficient. This would appear to be a decided safety advantage for the core with a zero or negative void coefficient. However, as already stated, the dimensions of the bubble are extremely important and it appears that a very large bubble is necessary to create a positive reactivity addition large enough to generate conditions which are hazardous to the core. Figure 29 (section III E.1.b.3) shows the results of the analysis of gas bubble incidents and indicates that a bubble large enough in diameter to instantaneously enter at least 19 fuel subassemblies and maintain a void height of about 15 inches in the core is necessary to create possible failure conditions with the least volume of gas. The void height of 15 inches creates the most reactive condition axially which occurs when approximately $\frac{2}{3}$ of the seed region is voided. This means that the actual bubble size at the core inlet must be about 2.8 ft in diameter by 0.5 ft high with all portions passing evenly through the core.

By proper reactor design, i.e., gas disentrainment equipment, flow baffles and gas detection equipment, the probability of bubbles this large ever reaching the core can be eliminated for practical purposes. It also should be pointed out that gas bubbles of a similar volume to that discussed above but in a different shape can also create hazardous conditions in a core with a zero or negative void coefficient. For example, a bubble with the same volume (height = 2.3 ft) and a diameter of about 1.1 ft that covers four subassembly inlets will cause clad or fuel damage in a core with a zero or slightly negative sodium void coefficient, as well as in the core with a positive effect. This bubble creates an ~6.0-ft high void in these four elements and causes failure because of the prolonged loss of heat sink. In conclusion, it is obvious that large gas bubbles must be eliminated from all fast reactor designs and that ability to do this must be designed into the safety system of any plant. Thus, the effect of a positive sodium void coefficient has little significance on the safety systems required for the plant.

b. Hypothetical Accidents

The analysis of hypothetical accidents included a comparison of the response of two core types; cylindrical and modular, to the maximum hypothetical accident. Details of this study are presented in section III.B.2, Hypothetical Accidents, and thus only the results will be summarized here. The events leading to the maximum hypothetical accident (MHA) are not described but could be initiated by any of several credible accidents discussed in the previous section provided all of the protective action failed. For example, the loss of all primary pumps with no scram will eventually create coolant temperatures high enough to cause gross core voiding and subsequent gross core meltdown, i.e., the maximum hypothetical accident. Therefore, the MHA involves two physical changes in the core, resulting in two possible reactivity insertions. The first reactivity effect can be either positive, zero or negative depending on the sodium void coefficient of the core design in question. The next, or meltdown reactivity insertion is usually assumed to result in a net positive effect. The meltdown model used determines the magnitude of this effect and the resulting design requirements for the containment system. Calculations were performed employing the commonly used homogenous model, which is usually considered to be quite conservative for most cores, and an improved or a more realistic model.

The results of the evaluation of hypothetical situations involving gross voiding and core meltdown, indicate that the cylindrical arrangement with its positive void effect is certainly no less safe than the modular arrangement and, furthermore, that it may well have distinct safety advantages. That is, using the ordinary homogenized meltdown model results in reactivity additions and subsequent energy releases which are about equal for the two cores (i.e., 70 - 100\$/sec). The positive reactivity addition from sodium voiding preceding meltdown, in the cylindrical core produces considerably less energy than does a homogenized meltdown and would disperse the core less violently, reducing the consequences of the MHA. Furthermore, an attempt to make a more realistic approach to the meltdown calculation has predicted a considerable difference in the response of the two cores following meltdown and indicates

a theoretical safety advantage for the cylindrical design. The cylindrical core does not achieve criticality during the rapid compaction phase of this meltdown because of the thin high leakage geometry, however, the modular core can become critical rapidly on meltdown and can compact at ramp rates of anywhere from 0 to 1800\$/sec.

c. Doppler Effect

The limiting Doppler requirement for either core is determined by the energy release characteristics following the meltdown accident or MHA. This limitation appears to be the same for both reactors and requires an isothermal Doppler coefficient of approximately -0.004 Tdk/dT with sodium removed. A comparison of these coefficients for various core regions is shown in table III and indicates that this requirement is not satisfied in either basic design. This means that both cores will require some modification to meet this criteria. A reduction in the mean neutron energy best accomplishes this goal and can be obtained most easily by the addition of a small amount of moderating material in the seed or radial blanket region or both. The cylindrical reference design has been modified (see Volume II) and requires about 4.5 v/o FeO distributed homogeneously throughout the seed region. The modular core design was not refined to this point; however, because the sodium removed Doppler coefficient is about the same for both cores without control rods ($\sim -0.0030 \text{ Tdk/dT}$), an equivalent amount of FeO distributed throughout the seed and radial blanket should attain the same effect. Alternate ways of distributing the moderator, and alternate moderator materials have been proposed for the modular design but it is not obvious that one has any cost or technical incentive over another. For example, the moderator could be graphite and placed in one or more (if required) rows of the radial blanket.

Table III illustrates the contribution to the overall isothermal Doppler coefficient from the seed, radial blanket and axial blanket. If the Doppler coefficient with sodium removed, i.e., seed plus blankets, is used when comparing cores to meet the meltdown criteria ($\sim -0.0040 \text{ Tdk/dT}$), it appears that the cylindrical design requires slightly less modification (i.e., moderator). However, it is not certain how important the contribution is from the axial blanket since it operates at a very low power density and in relatively low

Table III.

Isothermal Doppler Coefficient Comparison

<u>Condition</u>	<u>Reactor Type*</u>	
	<u>Modular</u>	<u>Cylindrical</u>
With Sodium (Tdk/dT)		
Seed Only	-0.0026	-0.0043
Seed and Radial Blanket	-0.0053	-0.0046
Seed, Radial Blanket and Axial Blanket	-0.0054	-0.0057
Without Sodium (Tdk/dT)		
Seed Only	-0.0005	-0.0029
Seed and Radial Blanket	-0.0030**	-0.0031**
Seed, Radial Blanket and Axial Blanket	-0.0030	-0.0036

* These calculations are for cores without moderator or control rods

** estimated

flux regions. For conservatism then, this contribution was ignored and the cores were compared on the bases of seed and radial blanket coefficients. It might be said that this same reasoning could also be applied to the radial blanket; however, that is not the case. In the modular core, the large radial blanket has a fairly high average plutonium concentration (4.5 percent) and operates at relatively high power densities, much of it in regions of high flux importance (i.e., between modules). Therefore, the radial blanket will readily respond in Doppler feedback following any disturbance in the core. During core meltdown, it is assumed that the radial blanket will experience physical changes similar to the seed region and will be available to contribute to the reactivity feedback effects. The cylindrical cores' radial blanket will act in much the same way; however, its contribution to the overall Doppler coefficient is small and almost insignificant in respect to determining the degree of modification needed to meet the meltdown criteria.

The Doppler coefficient is, of course, extremely important in protected accidents as well as the hypothetical type (meltdown) discussed above. The coefficients to be considered are those for a core with sodium. The important problem is to determine the effective Doppler coefficient, or in other words, the amount that the blankets contribute to the overall Doppler feedback effect

during a transient or accident situation. This is dependent on the temperature changes in the blanket regions relative to the seed. Temperature changes in the axial blankets are small and respond slowly, or more as a function of the coolant temperature change than the change in power. Again, for conservatism, these effects could be ignored. The radial blanket does experience a significant fuel temperature change during an excursion, especially in the modular design and must be included in the effective Doppler coefficient. Although its inclusion is not very important in the cylindrical design (-0.0043 to -0.0046 Tdk/dT) it is extremely important in the modular core (-0.0026 to -0.0053 Tdk/dT). The exact weighting of the radial blanket contribution for the modular core has not been calculated but it is estimated that it would result in an effective Doppler coefficient about equal to that in the cylindrical design, or -0.0045 Tdk/dT. Therefore, the Doppler response of both cores during any protected accident is about the same. It should be mentioned that in the protected accident studies the Doppler coefficient was varied from -0.0046 to -0.0060 Tdk/dT and found to make no difference in the protective system requirements. Therefore, as long as either core satisfies the meltdown criteria, the resulting Doppler coefficient (sodium in) is more than sufficient to satisfy the protected accident requirement. The obvious conclusion is that both the modular and cylindrical cores will require about equal modification to obtain the limiting Doppler coefficient of -0.0040 Tdk/dT, and thereafter are very similar in these Doppler characteristics.

4. Other Comparisons

Reiterating that the primary incentive for going to a modular design is to obtain a zero or negative sodium void coefficient, it must be determined what physically must be changed in the core to accomplish this and what effect (other than those effects already discussed above) such changes have on the reactor. The principle of the modular design is to divide the reactor into several semi-isolated small cores in one vessel with each core designed to produce a negative sodium void coefficient. The sodium void coefficient is a function of the neutron leakage from the seed region to non-seed regions and it becomes more negative as the leakage

is increased. Therefore small, high leakage cores can be sized to give a negative coefficient. Furthermore, in order to make such a reactor an efficient economical breeder, these non-seed regions which separate the various seed regions must logically be blanket material or depleted uranium. This creates several potential problem areas, other than those already mentioned, which are either different and additional or similar but expanded relative to a cylindrical core design.

Isolating each small core or module with one or more rows of radial blanket subassemblies inserts a large volume of blanket material into the reactor in comparison with the cylindrical design (i.e., about 2 to 3 times more). Furthermore, most of this additional blanket material is located in radial regions of high flux importance and this affects greatly the overall thermal-hydraulic and mechanical design of the core. An investigation of the factors influencing the mixed-mean outlet temperature has indicated the necessity of providing a variable orifice arrangement on the blanket subassemblies of the modular core design. Because of the large radial blanket volume, combined with its high plutonium content, the radial blanket generates almost 40 percent of the total reactor power at equilibrium conditions. At the end of each refueling cycle both seed and blanket subassemblies are replaced. The seed region receives fuel that is only slightly higher in Pu enrichment than that which is removed. Thus, the power density and coolant outlet temperature is about the same for the new and old subassemblies, other things being equal. On the other hand, the radial blanket subassemblies are replaced with those containing only depleted uranium and thus vary a great deal in power density over their lifetime as plutonium is bred. This plutonium content varies from 0 percent to about 7 percent over the blanket lifetime. Unless the inlet nozzles are designed to change the coolant flow to these subassemblies over their lifetime, the mixed-mean outlet temperature of the core will be reduced considerably by this blanket dilution effect. This will cause a considerable economic penalty to the plant and would justify developing a variable orificing technique for the blanket subassemblies in the modular design. Not nearly so strong an incentive exists for the cylindrical core because of the smaller relative radial blanket volume (power fraction equals ~ 0.05) and its reduced effect on the mixed-mean outlet temperature.

The determination of these dilution effects from both power and flow was discussed in the Task I Report⁽²⁾ for both the modular and cylindrical designs. These results are summarized in table IV and indicate the effect of using a variable orifice in the radial blankets of both cores; that is, the variable orifice effectively eliminates the dilution due to the power swing in the blanket subassemblies.

Table IV.

Dilution Factors

	Modular Core		Cylindrical Core	
	D.F.	ΔT_{mix}	D.F.	ΔT_{mix}
Without Variable Orifice in Radial Blanket	1.38	221	1.20	273
With Variable Orifice in Radial Blanket	1.20	254	1.16	282

D.F. = Dilution Factor = $\frac{\text{Peak Coolant } \Delta T}{\text{Mixed-mean Coolant } \Delta T}$

ΔT_{mix} = Reactor Mixed-mean ΔT

The potential improvement in mixed-mean outlet temperature that can be obtained from use of blanket variable orifices is indicated by the differences in the values of ΔT_{mix} , namely 33°F for the modular reactor and 9°F for the cylindrical reactor. These values would increase or decrease in proportion to the nominal temperature rise for the system.

Development of the variable orifice would not appear to be warranted for the cylindrical reactor; however, a modular reactor should be provided with this feature. The incentive for this is demonstrated by the results of the outlet temperature economic study of Schneider and Stoker,⁽⁴⁾ which indicate that for outlet temperatures in the range of 1050°F to 1150°F, a 33°F increase in the outlet temperature of a 1000 Mwe FER would be worth \$230,000/year the life of the plant. This is equivalent to \$3,400 per radial blanket subassembly at the normal rate of subassembly replacement

in a modular reactor, including the initial loading. In view of this economic advantage, and the reduction of thermal bowing problems associated with wide temperature swings in the blanket, it appears that a variable orifice will be required in the blanket of modular reactors.

These large radial blanket volumes and high plutonium contents are also reflected in the fuel cycle costs of the modular design. The greatest and most significant effect is in the inventory charge on plutonium generated in the radial blanket which is credited to the plant as soon as it is produced. This additional charge outweighs the gain from increased breeding and is the main reason for the greater fuel cycle costs of the modular design over the cylinder. Table V illustrates the breakdown of the fuel cycle costs for the two reference designs studied.

Table V.

Detailed Fuel Cycle Costs

	Optimized Reference Cores	
	Modular	Cylindrical
Seed Height (ft)	4.0	2.0
Fuel Slug Dia (in.)	0.30	0.35
Radial Blanket Pu Comp. (%)	4.5	5.0
Fuel Cycle Costs (mill/kwh)		
Fabrication		
Seed and Axial Blanket	0.106	0.174
Radial Blanket	0.092	0.019
Total	0.198	0.193
Interest on Fabrication		
Seed and Axial Blanket	0.013	0.023
Radial Blanket	0.041	0.014
Total	0.054	0.037
Inventory		
Seed and Axial Blanket	0.313	0.409
Radial Blanket	0.507	0.127
Total	0.820	0.536
Shipping		
Seed and Axial Blanket	0.023	0.043
Radial Blanket	0.017	0.005
Total	0.040	0.048
Processing Costs	0.113	0.099
Pu Credit	0.625	0.624
Total Fuel Cycle Costs	0.600	0.287

At first glance, it might be thought that this cost difference could be decreased or eliminated by either changing the blanket more often or using a thinner blanket. Radial blanket burnup (or Pu composition) was treated as a variable in the survey calculations. The results indicated (figure 3) that the fuel cycle costs tend to decrease as the plutonium concentration or radial blanket burnup increases. Since the average radial blanket plutonium concentration is directly related to the frequency it is changed (assuming it initially starts as depleted uranium), changing the radial blanket more often does not decrease the fuel cycle cost difference between the modular and cylindrical cores. The thickness of the radial blanket was not treated as a variable. It was held somewhat constant, with three rows of subassemblies between modules (1 1/2 per module) and three to four rows of subassemblies on the outer boundary. This resulted in blanket thicknesses of 1.1 to 1.4 ft and 1.3 to 1.75 ft respectively, depending on the subassembly size. These were probably not the most optimum blankets for all cores studied; however, from rough estimates, gains of the order of ~ 0.05 mill/kwh are the most that could be expected from changing the thickness. Furthermore, reducing the blanket thickness increases the sodium void coefficient since the modules became less isolated from each other and tend to behave more like a large single core. Therefore, any decrease in fuel cycle cost might be nullified by having to choose a lower enriched blanket to maintain a zero sodium void coefficient. Thus, it appears that the fuel cycle cost difference between the modular and cylindrical core is realistic and cannot be easily improved.

Reactivity changes due to burnup over the fuel cycle is another area which differs considerably for the two designs. Assuming both reactors are refueled at intervals of 1/3 the design total burnup of 100,000 Mwd/MT, or 33,000 Mwd/MT, the required excess reactivity to compensate for burnup at the start of the cycle is approximately 0.0 % $\Delta k/k$ for the cylinder and 4.5 % $\Delta k/k$ for the modular design. This requirement is largely a function of the internal conversion ratio of the seed region which is about 1.090 and 0.712, respectively. It should be pointed out that neither of these optimized cores contain any control material in these calculations, so

Figure 3

Modular Cores- Fuel Cycle Costs as a Function of Radial Blanket Composition

the absolute reactivity requirements would be expected to increase somewhat. However, the relative difference will remain about the same and this effect will increase the difference in fuel cycle cost between the modular and cylindrical designs; that is, the cycle average breeding ratio of the modular design will decrease more than in the cylinder. Approximate values were obtained, which indicate a change in the breeding ratio of ~ 0.09 or from 1.46 to 1.55 over the modular core fuel cycle. This converted into a fuel cycle cost differential, amounts to 0.133 mill/kwh. The cylindrical core has no change, because of a conversion ratio close to unity. The core surveys discussed in the Task I Report ⁽²⁾ did not include the reactivity effects of burnup and were all calculated at equilibrium conditions with a K_{eff} equal to one.

B. REFERENCE CORE SAFETY ANALYSIS

The reference core safety analysis was divided into two major areas: the investigation of (a) protected accidents and (b) hypothetical accidents. The analysis involved a detailed investigation of many specific accidents included under these two general types in an effort to determine their effect on the reactor. These effects include such things as: the reactor's inherent reactivity feedback, power perturbations, temperature changes in the fuel, clad and coolant and any core damage, local or gross. From this information, design requirements are established for the protective and containment systems such that the previously mentioned* basic safety criteria are satisfied.

1. Protected Accidents

Protected accidents consist of all identifiable credible accidents and it is a design requirement that these accidents will result in either (1) no damage to the reactor core, or (2) no damage to the reactor core greater than certain localized failures. Localized failure may occur as a result of plugging of one or several fuel subassemblies; however, such failures shall not produce reactivity changes which are large with respect to the inherent negative reactivity feedbacks of the system. Also, it is required that local

* Introduction to Safety Studies, Section III.

fuel failures or voiding be prevented from propagating to adjacent unaffected subassemblies by the action of the protective system and the mechanical arrangement of the core. Accidents which produce large reactivity insertions by such mechanisms as sudden voiding or compaction of large portions of the core are considered hypothetical and are not included in the protected accident studies. The analysis of the protected accidents and their resulting effect and use in the design of the protective system is discussed in the following sections. The design and the design requirements of the protective system resulting from this analysis is presented in the Safety System Description which is in Volume I of this report.

Protected accidents in the primary system fall into two general categories: (1) loss of flow accidents, which include loss or change of primary coolant flow to the core as a whole and the loss of flow to individual subassemblies, and (2) reactivity ramp accidents consisting of control rod withdrawal and ejection, fuel bundle drop and entrained gas bubbles transiting the core. The effects of accidents which have their source in either the secondary coolant loop or in the steam system have not been studied although they are also considered credible accidents and would eventually be included in the protective system design requirements. Although important, their ultimate feedback to the reactor core is expected to be less severe than the primary system accidents. Thus, the only changes in the protective system will be the addition of several reactor trip signals generated from the secondary and steam systems (i.e., secondary coolant flow, feedwater flow, turbine trip, seismic trips, etc.).

a. Loss of Flow Accidents

(1) Loss of All Primary Pumps

The complete loss of plant power and all emergency supplies could result in the loss of power to all pump motors simultaneously. Assuming these circumstances, the coolant flow would coast down or decay over approximately 15 to 30 seconds to some low flow rate (~6 to 7% full flow) based on natural convection cooling. During this period of flow decay, several reactor trip signals will be generated from various different reactor parameters, any of which will scram the reactor if functioning correctly. In the protected

accident studies, the protective system is assumed to operate on at least one of the reactor trips provided for each accident. The purpose of the study, then, is to determine the response of the reactor to this accident (and all protected accidents) in order to provide the design requirements for the protective system: that is, the changes in various reactor parameters, coolant temperature, flow, reactor power, reactivity, etc., are monitored to give a definite indication of approaching hazardous conditions to the plant operation. These early signals provide ample time to either correct the situation by manual or automatic procedures, or to automatically shut down the reactor before any damage occurs. The design philosophy of the protective system is to provide scrams on a minimum of three separate reactor parameters for any given accident, with the last trip acting early enough to scram the reactor before coolant or bond boiling, or fuel melting occurs.

The flow decay following a complete loss of all pump power is determined by the momentum in the pump impellers and in the flowing coolant. Since the pumps and complete primary system were not a part of this study contract, the flow decay characteristics were estimated from Fermi test data.⁽⁵⁾ This information, combined with the expected natural convection characteristics of the reactor, is shown in figure 4 and represents the flow characteristics during this accident. The FORE⁽⁶⁾ code was used to analyze this accident and the others. The reference core model was constructed and used as input into FORE. The input also included the flow decay curve and the various reactivity feedback effects. Table VI lists the core design data including the reactivity coefficients used in the various portions of the accident analysis. Since some of the safety studies were initiated before the final reference design was completed, the control rod worths used for determining reactivity rates (table VI-Ie) in the control rod withdrawal accidents were smaller than the final values listed in table VI-Ia&b. These changes have no effect on the results of the accident analysis.

Figure 4
ESTIMATED PRIMARY COOLANT FLOW DECAY CHARACTERISTICS

Table VI.

Reactor Design Data for Accident Analysis

I. Control Rods**

(a) Shim-Regulating Rods

Average (one rod)	0.223% $\Delta k/k$
Peak (one rod)	0.225% $\Delta k/k$
Total (13 rods)	2.904% $\Delta k/k$

(b) Safety Rods

Average (one rod)	0.400% $\Delta k/k$
Peak (one rod)	0.400% $\Delta k/k$
Total (9 rods)	3.600% $\Delta k/k$

(c) Worth as a Function of Axial Position

(see figure 9)

(d) Worth as a Function of Time After Scram

(see figure 10)

This data is based on control drives which supply a pneumatic assist at the start of scram which gives at least a total of 1g acceleration. (i.e., it counterbalances friction and other opposing forces.)

(e) Assumed Normal Control Rod Withdrawal Speeds (maximum) and Reactivity Rates

A. Withdrawal Speeds

1. Shim - 10 in./min
2. Safety - 5 in./min

B. Reactivity Ramps

Reactivity Rate ($\Delta k/k/\text{sec}$)*

1. All Rods Running at Their Maximum Speeds	1.800×10^{-4}
2. All Safety Rods	0.660×10^{-4}
3. All Shim Rods	1.142×10^{-4}

* Based on a preliminary control rod design. The change caused by updating the design has no effect on the results of the accident analysis.

** See Section IV for details

Table VI. (cont.)

(f) Control Rod Ejection Worths (Rod Ejection Accident)

	Ejection Time	Reactivity Ramp
Shim (peak)	0.186 sec	3.5\$/sec
Safety (peak)	0.063 sec	18.1\$/sec
$(\beta_{\text{eff}} = 0.00356)$		

II. Reactivity Coefficients

(a) Isothermal Doppler Coefficient ($T \frac{dk}{dT}$)

Sodium In	=	-0.00589
Sodium Removed	=	-0.00396

(b) Structure Expansion Coefficient (core radius)

$$\Delta k/k/^{\circ}\text{F} = -6.307 \times 10^{-6}$$

$$R_t \frac{dk}{dR_t} = -0.635$$

(c) Sodium Density Coefficient

$$\Delta k/k/^{\circ}\text{F} = +5.0 \times 10^{-7}$$

$$\rho_c \frac{dk}{d\rho_c} = +0.00316$$

(d) Sodium Bond Coefficient

$$\Delta k/k/^{\circ}\text{F} = +1.791 \times 10^{-8}$$

$$\rho_b \frac{dk}{d\rho_b} = +0.00099$$

(e) Cold Startup to 100% Power = $-0.523\% \Delta k/k$

III. Fuel Subassembly Worths

(Worth of subassembly in center core location)

	(% $\Delta k/k$)
(a) Outer Seed Zone Subassembly	0.401
(b) Inner Seed Zone Subassembly	0.263

IV. Pump Flow Coastdown Data

This information is based on actual FERMI pump tests⁽⁵⁾ at the reference core design temperatures.

(see figure 4)

The results of the analysis are presented in figures 5, 6, 7 and 8. Calculations were made for the reference core design and again for the same core but with a structural feedback coefficient that was doubled and equal to zero. Protective action trip points are marked at the time the trip signal is produced and delays are not included. A delay of about 0.1 second is expected from initiation of the trip signal to the actual beginning of control rod movement. In most of the calculations the actual scram was not included since its effect is always the same: that is, approximately 0.1 second after reactor trip, the control rods begin moving into the core, assisted with an initial pneumatic assist.* This almost immediately (0.1 to 0.2 sec) halts all power and temperature increases, turns them around and reduces everything towards a shutdown condition. This scram is shown on several of the curves simply to verify this statement and give added information. In these cases a scram of the safety rods only is assumed, or about $-3.6\% \Delta k/k$ (see Table VI). A normal scram would also include a pump trip for the purpose of reducing thermal shock effects, however, this was not incorporated in this analysis. To complete the input information, the reactivity worth of a control rod as a function of position and as a function of time after scram is illustrated in figures 9 and 10. Referring again to figures 5, 6 and 7, it is seen that a significant structural feedback coefficient, such as is in the reference core ($\Delta k/k/^{\circ}F = 6.31 \times 10^{-6}$), is quite helpful as it slows down the temperature increases (the coolant in particular) and provides time for additional protective action. This reactivity effect is incorporated into the total feedback effect by FORE as a function of the structural temperature changes. Therefore, it takes account of all time delays produced by the heat transfer properties. If the point of clad damage occurs at a coolant outlet temperature of 1600 $^{\circ}$ F to 1800 $^{\circ}$ F (due to coolant or bond boiling in the hottest channel), a difference of about 3 seconds (figure 6) is evident between the no structural feedback case and the reference case. Furthermore, it is apparent that if boiling did not take place at these temperatures, but at some higher temperature ($\sim 2200^{\circ}$ F), the coolant temperature would peak and start to decrease as a result of the feedback effects alone (structural feedback), possibly before any fuel damage. This would indicate that perhaps a larger structural

* The pneumatic assist has been added to the control drive mechanism as an extra safety feature. None of these studies indicates that a failure of mechanism would result in a hazardous situation.

Figure 5
LOSS OF FLOW ACCIDENT PRIMARY PUMPS
-Reactor Power-

Figure 6

LOSS OF FLOW ACCIDENT PRIMARY PUMPS
-Peak Coolant Outlet Temp-

Figure 7
LOSS OF FLOW ACCIDENT PRIMARY PUMPS
-Peak Fuel Temp-

Figure 8
LOSS OF FLOW ACCIDENT PRIMARY PUMPS
-Feedback Reactivity for Reference Design-

Figure 9

ESTIMATED 1000MWe REFERENCE CYLINDRICAL CORE
CONTROL ROD WORTH AS A FUNCTION OF POSITION

Figure 10

ESTIMATED 1000MW_e REFERENCE CYLINDRICAL CORE CONTROL ROD WORTH
AS A FUNCTION OF TIME AFTER SCRAM SIGNAL

feedback coefficient might inherently turn the accident around before a coolant temperature of between 1600°F - 1800°F is attained. This was investigated and it was found that a structural expansion coefficient of more than twice the reference core value would be required (see figure 6, $\Delta k/k/^{\circ}\text{F} = -12.61 \times 10^{-6}$). If feasible at all, this would require some special radial expansion amplifying device placed between each subassembly and is presently believed to be an unwarranted complication for a small gain in safety. The reference reactor already indicates at least five protective action trips preceeding coolant boiling, which appears to be more than sufficient to ensure the safety of the reactor in such an unlikely event. Figure 8 illustrates a detailed breakdown of the feedback effects in the reference core design. The significance and magnitude of the large structural expansion feedback reactivity is obvious. Also shown are the effects of the Doppler feedback and the contribution from the coolant sodium density change plus the squeezing of the sodium bond out of the fuel-clad gap. The positive sodium bond effect is small compared to the positive sodium density feedback. It contributes only about 5.0 percent of the total positive reactivity contribution. The Doppler coefficient used in this calculation was -0.0046 Tdk/dT but was later changed to -0.0060 for the remaining accidents after the reference design was updated. This small change has a negligible effect on the loss of pumps accident.

One additional parameter studied in the loss of primary flow accidents was the effect of varying the coolant flow decay time. If needed, this time could be lengthened by attaching a large fly wheel to the pump shaft, creating a longer coastdown. Although possible and, although it does allow more time for safety action, it appears that additional time, beyond that already provided by the structural feedback effect and the normal pump and coolant flow decay, is unnecessary and does not produce any significant safety advantages.

(2) Loss of One Primary Pump

The loss of a single primary pump due to power failure or a broken or frozen shaft does not appear to create conditions which are in any way detrimental to the reactor core. In reference to figures 5, 6 and 7, it

is observed that the reactor power and the fuel temperature drop to a slightly lower level caused by the negative structural feedback. The coolant outlet temperature increases slightly because of the reduced coolant flow rate and is the cause of the negative structural feedback reactivity. None of the conditions change enough to cause any scram action; however, alarms would be activated on loss of pump power and reduced flow. This would alert the operator to the malfunction and any corrective action would be accomplished manually.

The investigation of the simultaneous loss of more than one pump (but less than all six of them) was unnecessary. If all pumps operate from separate power buses, the probability of such a simultaneous occurrence is negligible. But, even if it did happen, more than ample protection is available because the worst condition (all pumps failed) is already included in the protective system design requirements.

(3) Local Subassembly Plugging

The localized simultaneous blocking of coolant flow to one or a few adjacent subassemblies might be caused by some primary system contaminant plating or crudding the subassembly surfaces over a period of time as in the Sodium Reactor Experiment (SRE)⁽⁷⁾ incident, or by a complete rapid plugging of the inlet port(s) by a relatively large size solid object carried there by the sodium flow, as in the FERMI⁽⁸⁾ incident. With proper design of the subassembly inlet nozzle (several inlet holes per nozzle), rapid total blockage by a transported object can be avoided; however, partial blockage of some of the inlet holes could still take place. Thus, the most credible plugging accidents would be (1) a gradual surface crudding over time periods of days or hours, possibly leading to total blockage if undetected, or (2) the rapid partial blockage (maximum flow reductions of 20 to 30 percent) caused by plugging of one or more nozzle inlet holes.

Several different plugging conditions listed below have been analyzed to determine how much protection can be provided by the protective system.

- 1) Complete instantaneous blockage
- 2) Partial instantaneous blockage
- 3) Partial blockage over a long time period of hours to days
- 4) Complete blockage over a long time period.

The first type, complete instantaneous blockage, is the worst case and presents an upper limit to the plugging accidents. As discussed above, it is difficult to realize just how this condition could present itself in a subassembly with proper inlet nozzle design. However, if partial blockage by crudding was allowed to continue undetected for a long period of time, it might eventually result in a complete plug and loss of all flow through the subassembly (case 4 above). Thus, complete blockage is a conceivable situation and as much protection as possible should be provided. Figures 11, 12 and 13 show results for the complete instantaneous flow blockage of a single subassembly, a seven and a nineteen subassembly cluster, all in the center of the core. The curves indicate the effect on the whole core as a result of the reactivity addition from voiding of the subassembly(s). The reactivity effect of the subsequent fuel melting and repositioning in these voided subassemblies is not included. Complete melting of the fuel subassembly is expected about 6 seconds after voiding. Furthermore, as the fuel melts, it is expected to reduce the reactivity either by falling out of the seed region by gravity or being forced upward and out of the seed region by sodium flow. As the results indicate (figures 12 and 13), the core as a whole (or the unvoided subassemblies) could withstand the reactivity effects of voiding of at least seven subassemblies since no coolant boiling or fuel melting take place. However, the blocked subassemblies will be severely damaged after voiding when fuel melting begins. The loss of coolant occurs in such a short time that no detection appears possible before some fuel is damaged in the blocked subassemblies. Therefore, the function of the detection system in this accident is to react early enough to limit the damage to the blocked subassemblies, preventing any hazard to the public.

It is evident from the results (figures 11, 12 and 13), that the fewer the subassemblies simultaneously blocked, the more difficult it becomes to make prompt detection. Thus, the blockage of a single subassembly is more difficult to detect and may result in a longer delay than the blockage of several. This occurs because the positive reactivity effect from sodium voiding of one subassembly is so small ($\sim 0.014\% \Delta k/k$) that it may be

Figure 11
SUBASSEMBLY FLOW BLOCKAGE
-Reactor Power-

Figure 12
SUBASSEMBLY FLOW BLOCKAGE
- Peak Coolant Temp-

Figure 13
SUBASSEMBLY FLOW BLOCKAGE
-Peak Fuel Temperature-

undetectable until fuel meltdown takes place a few (3 to 6) seconds later. Since the reactor power or outlet temperature do not increase significantly, they probably won't detect the malfunction. A reactivity meter⁽⁹⁾ might separate the change from the normal background noise and expected normal changes; however, a scram on this signal alone is probably not feasible. It appears that the reactivity change must be used in coincidence with another low limit signal such as fission product activity in the primary sodium or in the reactor vessel cover gas which would be detected within about 10 to 20 seconds, respectively, after the first gross clad failure. A more definite reactivity signal might occur after the fuel melts and begins moving out of the seed region. This could produce a decrease in reactivity of up to $-0.26\% \Delta k/k$ for one subassembly if all the fuel were removed. A reactivity change of $\pm 0.1\% \Delta k/k^*$ is assumed large enough to be confidently detected with the reactivity meter following any accident situation and thus can be used as a protective action scram signal. Therefore, if less than 40 percent of the fuel exited the seed region after melting (in ~ 6 seconds), a protective action trip would be initiated.

The flow blockage, and subsequent voiding of several subassemblies may avoid these problems because the positive reactivity addition from voiding is large enough to be confidently detected by the reactivity meter. Furthermore, it results in power changes large enough to cause high power and outlet temperature level trips. In such cases, the reactor will shut down before much fuel has melted in the blocked subassemblies, preventing any failure propagation.

The calculation of the subassembly voiding characteristics as a function of time was performed using ANTE, a one-dimensional sodium boiling and expulsion code assuming a liquid superheat of 215°F . Figure 14 illustrates the results of this calculation for the complete instantaneous blockage of a subassembly at full power. The reactivity addition as a function of void size was calculated with a MACH-1⁽¹⁰⁾ synthesis procedure and combined with the ANTE results to obtain the reactivity addition as a function of time (figure 15). This information was then used as input into the FORE code for the calculation of the reactivity effects on the reactor core. (Further discussion of voiding calculations are presented in the discussion of hypothetical accidents.)

* This value is probably high and can be reduced with better knowledge of the reactivity meter operation.

Figure 14
SUBASSEMBLY VOID BOUNDARIES FOR COMPLETE
INSTANTANEOUS FLOW BLOCKAGE AT FULL POWER
(Liquid Superheat = 215°F)

Figure 15
REACTIVITY EFFECT FOR THE AXIAL VOIDING OF A FUEL SUBASSEMBLY

If local plugging does take place, it is more likely to occur as partial blockage of one or a few subassemblies over a significant length of time (minutes, hours or days). Under these conditions, the subassembly(s) would operate at a reduced coolant flow rate and higher outlet temperatures. Partial blockage producing up to approximately 50% reduction in flow can be tolerated before sodium boiling of the bond is initiated (see figure 16). Boiling of the coolant sodium occurs with a slightly greater decrease in flow. Once boiling occurs in the coolant, the accident proceeds in a way which is identical to the complete blockage situation discussed above. However, for local flow reductions of less than 50%, there is a probability of detection before the situation gets worse. This protection is provided by the two outlet thermocouples placed in each individual subassembly. A discussion of how this detection system might function is given in Appendix A. This detection system is not part of the overall protective system but is basically there to provide information which would assist the operator in early prevention of a local plugging accident.

(4) Primary Coolant Boundary Failure

Primary coolant boundary failures can basically be separated into two groups, the major and minor rupture. Major ruptures would include any gross failure or sodium leak and can be represented by some limiting example such as the double-ended pipe break. This example will be used here and discussed as to its effects on the design of the reference core and the protective system. This accident or basic malfunction is the same as the maximum credible accident (MCA) for light water reactors.

The minor rupture covers a wide range of other small leaks which could eventually result in some loss of coolant and a decrease in sodium level. Early protective action would be available on signals from the sodium level and leak detectors. However, if this failed to correct the cause, the sodium level might continue to drop to the primary coolant outlet nozzles. At this point, the coolant flow would begin to decrease and gas might become entrained. The accident would then proceed in a manner similar to accidents already considered, that is, the loss of flow by pump failure (a (1) & (2)) and the gas bubble accident (b (3)). Because of this, no further consideration

Figure 16
PEAK FUEL SUBASSEMBLY
STEADY-STATE TEMPERATURES AT REDUCED FLOW

or discussion of the minor rupture is presented at this time, and the only primary coolant boundary failure analyzed was the limiting double-ended pipe break.

A major rupture of a primary pipe creates the greatest potential hazard if it occurs on the inlet side between the check valve and the reactor vessel. If the rupture occurs upstream of the check valve (and the check valve is functional) the result may be a major sodium spill in the pipe galleries; however, there is little immediate effect on the core. Siphon breaker and pump trip systems would be designed to automatically isolate the affected loop, thus minimizing loss of sodium from the system. Under these circumstances, the only possible hazard to the core comes from the possibility of excessive loss of sodium if either the siphon breaker or pump trip systems do not function properly and eventual entrainment of gas and loss of flow in all loops. However, at full flow for one loop, a minimum of 5 seconds is available before a dangerous low level occurs in this case, and the scram on reactor vessel sodium level would occur within 2 seconds.

A break downstream of the check valve is more of a hazard because of the direct communication between the point of rupture and the inlet plenum to the reactor. Since the hydraulic resistance of a major rupture in this location is considerably less than the resistance of the core, a large fraction of the primary flow would be rather quickly diverted out the break thus creating a sudden loss of flow situation in the core.

A simplified analysis of this accident has been made based on the following conservative assumptions:

1. An inlet pipe is completely severed instantaneously downstream of the check valve.
2. The inlet plenum pressure drops instantaneously to the equilibrium value as determined by steady state analysis of flow of coolant from the 5 remaining loops.
3. The primary pumps have a steep characteristic curve providing essentially constant flow to the system.
4. The pressure differential across the break is determined only by the change of coolant momentum.

5. The core pressure drop is proportional to the square of the flow through core.
6. The break occurs while the reactor is operating at full flow and full power.
7. Flow in the ruptured loop is cut off.
8. Flow from emergency reserve tanks is a small fraction of the loss out the ruptured pipe.

For the reference design (see figure 17) these assumptions yield the following conditions immediately after the break occurs:

1. Core flow = 40% of full flow
2. Inlet plenum pressure = 12 psi
3. Flow out the rupture = 43% of full flow
4. Reactor vessel level is decreasing at a rate of 1.1 ft/sec.

Transient analysis of the response of the core to step decreases of flow have been made with the FORE code and results are shown in figure 18. These calculations indicate that a considerable margin to boiling exists at the equilibrium flow of 40% and that several scram signals will initiate reactor shutdown within ~ 0.5 seconds after the break occurs. However, only 1 to 2 seconds are available before the sodium level in the primary vessel reaches the outlet nozzles and the remainder of the primary flow would be lost. This situation has been analyzed conservatively with FORE assuming the sodium level scram occurs at 0.5 sec. and primary flow is lost at 1.0 sec. Results of this analysis are shown in figure 19. This indicates that scrams on any of the first three available protective action signals; subassembly outlet temperature, sodium level or reactivity, will shutdown the plant before failure conditions are approached. That is, the peak coolant outlet temperature is limited to less than 1500°F. Should the protective systems fail to respond, the accident would ultimately lead to boiling of the coolant and extensive damage to the core. This accident is no different than most others in this respect except that only ~ 2 seconds from the start of the accident are available before boiling starts compared to a minimum of 6 seconds for other accidents.

Figure 17
EMERGENCY COOLING SYSTEM

Figure 18
EFFECT OF REDUCED CORE COOLANT FLOW FOLLOWING THE PIPE
RUPTURE ACCIDENT ASSUMING SODIUM LEVEL REMAINS
ABOVE OUTLET NOZZLES

Figure 19
REFERENCE CORE PIPE RUPTURE ACCIDENT WITH SCRAM
ON THE SODIUM LEVEL SIGNAL

There are several relatively straight forward modifications to the reference system that could be made to lengthen the time available for protective action to be comparable to that of other accidents. However, before discussing these modifications, the credibility of this accident ought to be examined. Consideration of this accident is prompted by the existing precedent for its consideration as the "maximum credible accident" in pressurized water reactors. There are fundamental differences between PWR's and LMFER's, however, which raise substantial questions as to appropriateness of assigning the same order of probability for this sort of failure in the LMFER as in the PWR. The most prominent differences are:

1. In the areas around the reactor vessel and piping to the check valve, the reference LMFER has a redundant primary coolant boundry while the PWR does not. Therefore this accident in the LMFER requires the simultaneous failure of two pipes rather than one.
2. For the LMFER, piping stresses due to internal coolant pressure are low relative to the strength of the pipe and sudden increases in the pressurization of the coolant system are not possible as a result of credible accidents as is the case with the PWR.

Within the limited scope of the present study, it is not practical to argue this point in detail, however, it is apparent that dogmatic application to LMFER's of the PWR precedent on "double-ended pipe ruptures" is not a good idea. Its occurrence borders on the incredible and, as indicated by the above analysis, its consequences are not particularly severe unless the protective system fails completely. But this is also the case with other accidents which are much more likely, such as loss of pumping power.

In light of the above, it is tentatively concluded that no special modifications to the reference system are necessary because of considerations of this accident. This conclusion is subject to reexamination after development and analysis of details of the primary system. Should this conclusion change, the sort of modifications that are contemplated are:

1. Increase diameter and height of the reactor vessel to increase the amount of stored coolant and the distance between the free surface and outlet nozzles.
2. Move the check valves to a position inside the reactor vessel.
3. Use a combined surge-tank-primary-pump arrangement for storage of reserve coolant.

Each of these modifications has some advantage other than those associated with the pipe rupture accident and would, therefore, be considered as a matter of course in the development of the primary system design.

b. Reactivity Ramps

(1) Control Rod Ejection

In the Control Rod Ejection Accident, it must be assumed that somehow a control rod is subject to a large upward driving force with little or no restraint at a time when the reactor is at full power operation (the worst case). To conceive of such an event, several compounding errors and malfunctions must be postulated. One example of the compounding of events leading to such an accident is as follows:

- a) The reactor is taken to full power operation with an undetected stuck control rod completely inserted;
- b) The orifice insert in that control rod location was unknowingly removed during shutdown so the control rod is now subject to the full inlet plenum pressure. (Normally, the inlet pressure force to the control rod is smaller than the gravity force acting downward.);
- c) During the time that the reactor was taken to full power, the control rod drive mechanism for this stuck rod was withdrawn unattached to the control rod;
- d) While the reactor is operating at full power, the stuck rod suddenly becomes free and moves out of the seed region with a driving force equal to the inlet pressure.

Since a safety rod is worth more in reactivity than the shim-regulating rods and because it weighs less and will be ejected faster (see Table VI), it has the highest reactivity insertion rate and was used in this analysis. The possibility of two or more control rods experiencing similar malfunctions simultaneously is even more inconceivable and was not considered.

The effect of this accident on the reactor core is summarized in figures 20, 21 and 22. The reactivity insertion rate for the safety rod with maximum worth ejected by a 75 psi inlet plenum pressure is 18.0 β /sec for a total of 0.40% $\Delta k/k$. The total ejection time is 0.063 second. As the curves (E-reference design) indicate, the core and fuel will be undamaged following

Figure 20
CONTROL ROD EJECTION ACCIDENT AT FULL POWER
Reactor Power

Figure 21
CONTROL ROD EJECTION ACCIDENT AT FULL POWER
Peak Coolant Temperature

Figure 22
CONTROL ROD EJECTION ACCIDENT AT FULL POWER
Peak Fuel Temperature

this accident, provided that at least one of the five available trip signals scrams the reactor. It should be reiterated that, by definition, a credible accident requires that if only one out of three separate protective action signals is operable, the plant will shut down with no gross core damage. Thus, the protection is somewhat redundant for this accident.

Scram on any of the first four protective action signals will limit the peak fuel and coolant temperatures to 3200°F and 1460°F, respectively; which is well below any impending damage limits. Even scram on the fifth signal, mixed mean outlet temperature, will prevent damage although fuel melting and coolant boiling limits are in near proximity in the peak channel. As in most of the accidents, the reactor conditions following scram were not calculated because in all cases the power and temperatures will decrease rapidly, within approximately 0.1 second after the initial signal (see scram for Loss of Flow Accident-Loss of All Primary Pumps, Section III.E.1.a.1). The two other curves (A&C) are shown for comparison purposes and indicate the response for different control rod worths.

(2) Control Rod Withdrawal

The Control Rod Withdrawal Accident is a result of a withdrawal at normal drive speeds of one or more control rods, starting from some normal reactor condition but continuing unstopped, eventually causing abnormal operating conditions. Two initial conditions are usually considered in this type of accident: withdrawal from full power operation and withdrawal from cold or hot startups. These accidents were analyzed for the reference core design using the reactivity coefficients and reference design data given in Table VI. The reactivity insertion rate was the same in all cases and was based on the withdrawal of all control rods (safety and shim-regulating) from their most reactive position (approximate axial core center) at the maximum drive speed. This amounts to an insertion rate of $1.8 \times 10^{-4} \Delta k/k/\text{sec.}^*$ Again, the results of the analysis are used to predict the time available before any thermal damage occurs in the reactor, and to determine the safety measures that must be provided by the protective system.

* This value is based on a preliminary control rod design and does not represent the final design rod worths given in table VI. However, this change in total worth does not affect the conclusions drawn from these accidents.

(a) Withdrawal at Full Power

Although the safety rods would be fully withdrawn during this operating condition, the reactivity insertion rate used in the calculation assumes that they are moving through their most reactive position and includes their contribution. This adds another degree of conservatism to the calculation (i.e., $1.8 \times 10^{-4} \Delta k/k/\text{sec}$ compared to $1.14 \times 10^{-4} \Delta k/k/\text{sec}$ based on shim-regulating rods only) but does not change the conclusion: that is, the rate of change of power and temperature is increased but the protective action trips remain at the same limit and do not change with respect to each other. Only the time interval before the limits are reached is decreased slightly. The effect on the reactor system parameters is shown in figure 23. Protective action trip signals are indicated where exceeded, and even with the conservative reactivity insertion rate of $1.8 \times 10^{-4} \Delta k/k/\text{sec}$, more than sufficient time is available for protective action.

Although reduced initial power cases were not analyzed (other than the startup case discussed below), the results are expected to be similar. However, there will be more lapsed time before the respective trips are activated. The maximum power and temperature obtained, if all but the final scram is inoperable, would be about the same. Earlier scrams on reactivity and power/flow mismatch would stop the excursion, in most cases, even before full power is attained.

(b) Withdrawal at Startup

The control rod withdrawal from startup conditions, usually identified as the startup accident, is a somewhat more interesting accident since the results change depending on the startup conditions assumed. Four different basic startup conditions are possible. Two of these consist of a startup from the cold (375°F) refueling condition with coolant flows of either 100 percent full flow or 6 percent full flow. The other two are startups from a hot condition (800°F), for example, following a reactor scram, with either flows of 100 percent full flow or approximately 6 percent of full flow. Two of these startup conditions were chosen for the analysis: one, considered the worst case, is the hot startup at low flow (800°F at 6 percent

Figure 23
CONTROL ROD WITHDRAWAL ACCIDENT AT FULL POWER

flow) and the other, a hot startup with full flow. These two conditions essentially cover the complete range of startup accident possibilities and enable a determination of all the required protection.

Figures 24, 25 and 26 show the results of these accidents and indicate the protective action available. All calculations were started from a subcritical condition of $k = 0.99$ with a source of 1.0×10^4 neutrons/cm²-sec. The insertion rate, as already discussed earlier, was 1.8×10^{-4} $\Delta k/k/\text{sec}$. The startup accident from the hot low-flow condition (800°F at 6 percent flow) appears to be slightly more serious than other startup conditions because of fewer available trip signals which are somewhat delayed. As indicated (figure 24), the Doppler feedback stops the initial power burst following criticality at a peak of about 460 Mwt or considerably less than the permanent 125 percent full power scram. This produces a short-lived decrease in power, eventually followed by a gradual increase which is proportional to the control rod withdrawal rate. Many seconds later (~ 75.0 sec), the full power trip would be reached but too late to prevent damage to the core. However, several other protective action trips are available, some of them permanent set limits, so there appears to be no lack of protection. Special trips on reactor period and 1 percent full power are provided for the startup operations only. It has not been established yet whether trips on reactivity and power/flow can be used during startup, but the permanent scrams from subassembly and mixed mean outlet temperature are available as final backup. This same case was rerun with a reactivity insertion rate of 1/2 the maximum or 0.9×10^{-4} $\Delta k/k/\text{sec}$ and starting with a $K_{\text{eff}} = 0.995$. As indicated in figure 24, the power peak is reduced to about 260 Mwt but the general slope of the curves is the same, with the same conclusions for available protective action. The temperature increases are slower because of the slower power rise, but the same backup scrams are available.

The startup accident initiated from hot conditions at 100 percent flow exhibits a somewhat different behavior. Because of the better heat removal (100 percent flow), the fuel and coolant do not heat up as fast, thus reducing the reactivity feedback effects. Instead of completely stopping the

Figure 24
CONTROL ROD WITHDRAWAL ACCIDENT AT STARTUP CONDITIONS
-Reactor Power-

Figure 25
CONTROL ROD WITHDRAWAL ACCIDENT AT STARTUP CONDITIONS
-Peak Coolant Outlet Temperature-

Figure 26
CONTROL ROD WITHDRAWAL ACCIDENT AT STARTUP CONDITIONS
-Peak Fuel Temperature-

initial power burst, the Doppler coefficient only decreases the rate of change of power. Therefore, after the initial burst, the power continues to rise (figure 24) but at a reduced rate. Under these conditions, the 125 percent full power trip is exceeded long before failure conditions are produced, thus adding to the protective system action available. Furthermore, the trips discussed above will also be activated at the points indicated in figures 24 and 25. The three curves shown for this case illustrate the effect of changing the Doppler and the radial expansion feedback coefficients. None of these changes have a significant effect on the available protection of the reactor core. For added information, a scram was calculated (curve E) for the trip signal from the 125 percent full power limit.

(3) Gas Bubbles

Accidents arising from entrained gas bubbles in the primary sodium coolant can be initiated from several sources: (1) large accumulations of fission gas, (2) large bubbles from cover gas entrained at the pumps or at other free surfaces, and (3) vortexing in the reactor vessel, etc. The entrained gas, if not detected or removed, could conceivably combine into large bubbles or be created as large bubbles and pass through the reactor core at coolant velocities, temporarily halting the heat removal capacity for a group of fuel pins, a whole fuel subassembly, or even several fuel subassemblies, depending on the bubble size. It is felt that the core can be designed employing baffles, mixers and gas removal equipment which will eliminate the formation of large single bubbles. However, these designs and the necessary equipment have not been developed yet, thus the effect of this accident on the reactor core was determined for the inclusion into the protective system requirements, since it represents an upper limit.

The analysis of this accident consists of a parametric study of the gas bubble dimensions, length and diameter, to determine its relationship to fuel damage. The bubbles are assumed to transit the core at the average coolant velocity of 26.9 ft/sec. All calculations are performed assuming the bubble is centered around the core centerline and includes both reactivity and heat transfer effects. The coolant void reactivity is estimated using

MACH-1. A typical reactivity response for a 2.0 ft bubble which voids 1, 7 and 19 subassemblies is shown in figure 27. This void reactivity is then inserted into FORE to determine the thermal effects on the reactor core. A start-restart routine is used with FORE since the heat transfer coefficient cannot automatically be changed as a function of time.

The range of bubble sizes investigated included lengths of from >0.0 ft to approximately 6.0 ft. This latter dimension is an upper limit in that a bubble this length or greater may cause failure even with a zero reactivity effect as is essentially the situation for a single voided subassembly. Bubble lengths greater than this will produce dangerous clad temperatures from the prolonged loss of heat sink alone. Bubble diameters are determined by the number of subassemblies affected, which varied from one up to approximately 39. Bubbles with larger diameters than this may cause failure no matter what their length is. For the range of bubbles investigated, any damage produced by the temporary loss of cooling plus any reactivity effect is caused by exceeding the clad temperature limits as opposed to any other possible fuel failure limits such as fuel melting, high fuel vapor pressures, etc. The clad failure limit is assumed to be in the range of 1600 to 1800°F, which is determined by the temperature that sodium boiling would be expected if coolant follows the bubble into the subassembly. Thus, it is also assumed that if this sodium enters the subassembly and comes in contact with clad surfaces at these temperatures or higher, it will vaporize and maintain a void until the fuel pins actually fail. A typical example indicating the peak surface clad temperature caused by the passage of a 2.0 ft bubble is illustrated in figure 28. Bubble diameter variations are represented by the number of subassemblies voided. Data similar to this was generated for several different bubble sizes and was transformed into the parametric illustration shown in figure 29. This figure illustrates the data calculated for bubbles over the complete range of length and diameter. For cases where a significant positive reactivity effect is present (i.e., ≥ 7 subassemblies voided), an initial temperature peak is evident. This is produced by an axial peak reactivity effect which occurs when about 2/3 of the center core length is voided. Bubbles longer than this have a smaller reactivity effect

Figure 27
REACTIVITY EFFECT OF 2.0 FT BUBBLE
TRANSITING THE REFERENCE CORE

Figure 28
PEAK CLAD TEMPERATURE AS A RESULT OF A 2.0 FT BUBBLE
TRANSITING THE CORE

Figure 29
GAS BUBBLE ACCIDENT- The Effect of Void Size on the Clad Temperature

and tend to decrease the peak clad temperature. However, after a certain bubble length is attained, the loss of heat sink (coolant) over-compensates for the reduced reactivity and the peak clad temperature again increases until at some given combination of length and diameter fuel pin failures may occur.

An analysis of the curves shown in figure 29 gives an indication of the range of bubble sizes necessary to cause fuel damage. Due to the reactivity peak at $2/3$ the core length, a short flat bubble voiding a region of about 1.2 ft in height by 19 subassemblies will create clad temperatures approaching the failure range. This requires a bubble with a volume of approximately 2.3 ft^3 . Voiding the same number of subassemblies but increasing the voided length will decrease the clad temperature up to a length of about 3.75 ft. At this length, the clad failure range is again attained but with a bubble volume of at least 7.0 ft^3 . If the voided length is increased even further, it is still possible to remain below the clad failure range provided the bubble diameter or number of voided subassemblies is decreased (i.e., reducing the reactivity effect). The limiting case is reached at a voided length of about 6.25 ft with one subassembly voided, which has essentially a zero reactivity effect but causes clad damage by an extended loss of coolant. A bubble long or narrow enough to create such a void is, of course, quite improbable. However, bubbles which void 3 or 4 subassemblies over an ~ 5.5 ft length are more square in form or are of a much more probable geometry. This bubble again has essentially a zero void effect with a volume of $\sim 2.5 \text{ ft}^3$.

The analysis of the bubble accident data, therefore, indicates that single coherent gas volumes of approximately 2.3 ft^3 and greater could initiate fuel failure if formed in the proper configuration. Furthermore, the positive reactivity effect caused by the gas void increases the number of possible bubble sizes which might cause failure. However, even with no positive sodium void effect, or essentially zero, there are bubbles with geometries of high probability (Vol. $\sim 2.3 \text{ ft}^3$) that can cause failure simply by the prolonged loss of heat sink. This leads to the conclusion that coherent bubbles of fairly large sizes can definitely be a serious

safety hazard to the reactor for both a zero and a positive void effect and thus should be eliminated wherever possible and protected against. This should be accomplished by (1) eliminating the sources of gas, and (2) providing design features in the primary system which will remove the gas once it is present. Detection equipment must also be provided on the primary loops to monitor and protect against any large gas build-up.

(4) Subassembly Drop

The analysis of the subassembly or fuel bundle drop accident historically has assumed that by some series of compounding mistakes and malfunctions, the reactor is brought unknowingly to a near critical condition prior to the insertion of the last fuel bundle. Furthermore, it is assumed that this condition exists with all control rods fully inserted (and detached from their drives) into the core. This last fuel bundle is then accidentally dropped into its location creating an excursion which takes the reactor through criticality, reaches a peak power which is limited by the inherent reactivity feedback, and eventually levels off at some power determined by the worth of the dropped fuel bundle. Unless some additional shutdown reactivity were available, i.e., cocked rods, soluble poison, etc., the reactor would remain at this power level until the fuel could be removed. An example of this accident for the reference core maximum worth fuel bundle is illustrated in figures 30 and 31. This calculation assumes that an outer zone fuel bundle ($\sim 0.4\% \Delta k/k$) is dropped into the center core location.

After considering the number and kinds of malfunctions that must occur, and furthermore must be compounded upon each other, it was felt that this accident is extremely improbable. Sufficient protection against this accident is provided by the numerous checks and safety precautions taken during fuel fabrication, through intermediate handling procedures and finally through criticality, combined with a large shutdown margin provided by all available poison (control and safety rods) inserted into the core during refueling. This is in general agreement with the safety philosophy of current water reactor design.

Figure 30
FUEL SUBASSEMBLY DROP ACCIDENT
-Reactor Power-

Figure 31
FUEL SUBASSEMBLY DROP ACCIDENT
-Peak Fuel and Coolant Temp-

c. Fault Tree Analysis for Credible Accidents

Task V of the contract includes a safety review of the reference core concept. Based upon the reference core safety studies which have been described in the previous section, a fault-tree analysis of the reference concept reactor accidents has been developed to graphically display the credible accidents with respect to the malfunctions which could occur during operation. It illustrates the casualty and the consequences as well as the interrelationships.

The method of displaying the fault-tree is similar to that used in aircraft safety analysis.⁽¹¹⁾ The fault-tree is basically a graphic method of tracing and analyzing the potential of one or more malfunctions to produce some final undesirable event. Physically the fault-tree starts at the bottom, indicating the many malfunctions and initiating conditions, and then leads into the consequences and other possible malfunctions; finally ending at the top with the ultimate undesirable event. Standard symbols are used to represent the fault event and the various conditional inputs and alternate consequences possible. It is an excellent tool for the application of probability analysis since each possible fault is clearly defined.

The fault-tree analysis as applied to the credible accident survey of the 1000 Mwe reference core concept is illustrated in figure 32. It is somewhat different from most standard safety fault-trees, in that, in addition to showing the malfunctions and consequences in the reactor itself, it further includes the protection available and the consequences of its malfunctioning. The basic philosophy behind this technique is (1) to force a detailed investigation of all events following any given malfunction, reducing the possibility that something might be overlooked, and (2) to obtain a display of all possible faults including those in the protection system, which will allow a determination of the probability of each malfunction causing the ultimate final consequence. This will provide an evaluation of the risks involved and to some degree the amount of protection available. This information can then be factored back into the reactor design if deemed necessary. The probability analysis has not been performed as part of this study; however, the fault tree shown can readily be used in this capacity.

FAULT TREE ANALYSIS OF CREDIBLE ACCIDENTS FOR CE 1000 MWE LMFR REFERENCE CONCEPT

Figure 32

The graphic illustration of the fault tree is largely self explanatory and does not require much additional discussion. The basic malfunctions are those accidents named and analyzed in parts (a) and (b) of this section, i.e., the Protected Accident Studies. The "initiating condition for the accident" is the major effect that the basic malfunction has on the reactor parameters. Following the initiating condition, are the initial consequences which are primarily changes in other reactor parameters that are caused by the initiating condition (i.e., changes in reactivity, power, period and temperature). Other initial consequences include physical changes in the coolant such as local sodium boiling and voiding. Indicated next, is the reaction of the protective system which is designed to respond and shut down the reactor as the initial consequences are detected. At this point, the effects of the accident should be halted, provided the protective system functions normally. However, if all available protective action for any given accident fails, the conditions in the reactor core will proceed to those described in the final consequences portion of the fault tree. Some minor accidents (basic malfunctions) such as loss of one primary pump or small gas bubbles do not result in any core damage following failure of the protection system. But, most accidents will lead to some local failure and can ultimately cause complete destruction of the core under these circumstances. The final fault event or "reactor core destroyed" is not developed any further since the follow-on study did not include a containment analyses. This would be one starting point or basic malfunction for a fault tree development of the containment safety system.

As an aid in reading the fault tree display, the mechanics and symbol meanings are explained here briefly. The tree is oriented and read from the bottom up. At the bottom, the basic malfunctions are displayed and are enclosed in diamonds. These diamonds represent an undeveloped fault event which means that this event or malfunction could be preceded by a fault tree of its own. Each of these basic malfunctions (or accidents) is given an identification symbol which is represented by a letter enclosed by a circle. This is only used to aid in following any given accident through the fault tree. Other fault events or conditional inputs which are essentially

developed or complete as defined are designated by rectangular boxes. Many of these events may follow any given basic malfunction. The logic of the fault tree is determined by the various decision gates and the conditional input. "And gates" and "or gates" indicate a choice of routes that the accident might follow. For these gates only, a slight variation in the standard symbolism was used. That is, with the gates either upright or reversed, the "and" or "or" meaning remains valid. The "inhibit gate" represents a block in the path of the accident, in that a given conditional input must always be satisfied before the accident can proceed up a particular route. A hexagon is the symbol used for the "inhibit gate" and represents the locations throughout the fault tree in which a probability of occurrence could be applied. The transfer symbol is used to eliminate the duplication of event chains. If a transfer is made for any given accident, the chain should always remain with that accident. For example, after the primary coolant boundary failure proceeds to a loss of coolant and decrease in sodium level or route (A), some initial protective action is available and is indicated by a transfer to 14. After this protective action chain of events is completed, a transfer is again made back to the original accident route (A) by the symbol 15. The accident then continues on according to the logic presented. Thus the entire chain of events can likewise be followed and the faults identified for any given basic malfunction or accident.

d. Conclusions from Accident Analysis

A summary of the conclusions derived from the analysis of the protected accidents is included herein.

(1) In all accidents, the last available scram occurs in sufficient time to 1) terminate the accident without damage, or in those accidents where some damage is unavoidable, it will 2) terminate the accident before the failure can propagate.

(2) Most accidents are not inherently terminated by internal reactivity feedback mechanisms although the severity may be significantly reduced. External scram is needed to completely terminate the accident.

- (3) The results of the accident are not affected by the available scram reactivity worth as long as it is sufficient to make the reactor sub-critical.
- (4) The slight reduction in insertion time provided by the pneumatic assist on the control rods while scramming does not gain any protection, thus a gravity drop is sufficient. However, this assist may help guarantee that the rods will scram if they are stuck or misaligned.
- (5) The instantaneous voiding of at least nineteen center subassemblies can be tolerated with no further damage to the core other than that in the voided subassemblies.
- (6) No major differences have been discovered between the safety characteristics of the reference core, and cores of this type analyzed by others.
- (7) A core exhibiting a zero or slightly positive or slightly negative sodium void coefficient may experience damage during a flow blockage accident which is more severe than in a core exhibiting a significant negative or positive (as in the reference core) coefficient. A zero or negligible reactivity effect from voiding would go undetected until significant fuel damage occurred; whereas, a large positive or negative effect would be immediately indicated followed by the appropriate protective action, possibly limiting the amount of damage.
- (8) The analysis of the bubble accident has indicated the need to eliminate the possibility of large gas bubbles from any fast reactor design. Bubbles containing gas volumes of $\geq 2.3 \text{ ft}^3$ and with the proper configuration can cause fuel damage. It is felt that by proper design, i.e., gas disentrainment equipment, flow bubbles and gas detection equipment, the probability of bubbles this large ever reaching the core can be eliminated for all practical purposes.
- (9) A bubble configuration that causes no reactivity effect can still cause fuel damage due to the prolonged loss of heat sink (coolant).
- (10) The double ended pipe rupture, which was used as the maximum credible primary coolant system boundary failure in this study and is the maximum credible accident (MCA) for light water reactors, can be prevented from causing any core damage. This is accomplished by a combination of proper primary system design and the available protection from the protective system.

2. Hypothetical Accidents

The conditions leading to the maximum hypothetical accident (MHA) involve two physical changes in the core resulting in two possible reactivity insertions. These changes include: (1) voiding of most or all of the core sodium followed by (2) gross fuel meltdown and compaction of the core fuel, clad and structure. The first reactivity addition can be either positive or negative depending on the sodium void coefficient of the core design in question. The next reactivity insertion (due to meltdown) is usually assumed to result in a net positive effect. These core meltdown reactivity ramp rates are ordinarily calculated by assuming that meltdown and compaction occur homogeneously over the entire core under gravity acceleration. However in reality, it is more likely that core melting would lead to some kind of inhomogeneous collapse. An idealized model for such a melting process was developed and reactivity ramps were calculated with this model for both cylindrical and modular cores. Upper limit values for reactivity insertion rates due to sudden, coherent voiding of the cylindrical core have also been calculated using a sodium ejection model which accounts for superheating of the coolant. Finally, Bethe-Tait type calculations were made to determine the upper limit for explosive energy yield from the highest reactivity ramp calculated for the reference design.

a. Description of Methods

The analyses of conditions associated with hypothetical accidents, i.e., core voiding, fuel meltdown and Bethe-Tait explosion, were carried out with the three one-dimensional program codes, ANTE, MACH-1, and WEAK EXPLOSION.

The code ANTE is a slightly modified version of ETNA⁽¹²⁾ which deals with the thermal-hydraulic behavior of a liquid coolant in one specific coolant channel. It allows boiling of the coolant and subsequent ejection from the channel. The power shape, various heat transfer coefficients, friction factors, and the variation of the hydraulic diameter are entered pointwise along the channel. Heat loss from the coolant channel to the surroundings is taken into account by considering the adjacent temperatures. During a specific run, the power, inlet and outlet pressure and also the adjacent temperature can be varied with time, the output being a chronological representation of fuel, shroud, and coolant temperatures, coolant velocity, coolant vapor pressure, and the pressure in the channel.

The MACH-1 program⁽¹⁰⁾ is a one-dimensional diffusion code. Because voiding patterns and meltdown configurations are complicated two-dimensional arrangements, MACH calculations in one direction were synthesized with the orthogonal calculations by manually transferring group- and region-dependent bucklings. Thus, synthesized two-dimensional results were obtained for the reactivity calculations.

The code WEAK EXPLOSION⁽¹³⁾ was used to calculate the energy released from the maximum hypothetical accident. It performs a coupled neutronics-hydrodynamics calculation for a spherically symmetric reactor core to obtain the time-dependent energy release that results from the insertion of reactivity at a given rate. The neutronics are determined from polynomial distributions of power and material worth; the Doppler effect and delayed neutrons being taken into account. The output is a chronological listing of energy, power, multiplication constant and Doppler reactivity.

b. Core Voiding

The most straightforward sort of accident leading to gross coolant voiding of the core and subsequent fuel overheating and meltdown is a loss of pressure in the inlet plenum without any protective action. This can be caused either by pump failure, an inlet line breakage, or a reactor vessel rupture. The histories of the first two of these accidents have been analyzed and discussed in section E-1. The third is treated here as the limiting worst case.

The following assumptions were used in calculating this limiting case for the reference core:

- 1) The inlet coolant plenum is instantaneously depressurized at full power due to a sudden, large reactor vessel rupture.
- 2) Protective systems fail.
- 3) All fuel channels begin to boil when the sodium reaches a liquid superheat of 215°F.
- 4) The sodium does not reenter any voided channels.

Results from ANTE calculations for the sudden pressure loss accident are represented in figures 33 and 34 showing how changes in the power production of the fuel and the superheat of the coolant affect the time, after failure,

that boiling sets in and the coolant in the channel is completely ejected. Assuming the superheat to be constant throughout the reactor, voiding patterns for the core can be constructed from the radial power shape as shown in figure 35. However, it is virtually impossible that the physically unstable phenomenon of liquid superheating should occur in exactly the same way throughout the core. Experiments have shown that when boiling is initiated in sodium during a transient, the degree of superheat varies from zero to several hundred degrees in a more or less random manner.⁽¹⁴⁾ This would lead to voiding of individual subassemblies in a random pattern throughout the core. This behavior can be represented approximately by a homogeneous voiding pattern.

Figure 36 shows the results of calculations of the reactivity as a function of percent of reactor volume voided for both voiding models. Figure 37 shows the change of reactivity from the time that boiling sets in. It might be noted that with the "homogeneous" model the time corresponding to the different voiding steps is rather arbitrary. In the graph (figure 37), it was assumed that both models would void to 80% in the same time (about 400ms).

c. Core Meltdown

The usual method for calculating reactivity ramps due to fuel meltdown and core compaction (meltdown and crashdown in our terminology) is to assume a homogeneous distribution of the resulting geometry and density changes over the whole of the reactor core. In effect, this means that the accident is simulated by increasing the fuel density while at the same time the core volume is reduced. Accordingly, the results show a continuously increasing reactivity, the ramp of which is determined by assuming an appropriate model for the compaction. This is usually assumed to be acceleration of the core center of gravity by one g. Calculations using this model resulted in a ramp of $\sim 65\$/\text{sec}$ for both the cylindrical and modular designs.

It is difficult to visualize conditions where an actual meltdown and compaction of a core would be represented realistically by this homogeneous model. A more probable situation is that the molten material will, under force of gravity, drip down and collect either in the lower plenum or lower

Figure 33

INFLUENCE OF ENERGY PRODUCTION ON COOLANT EJECTION

Figure 34

INFLUENCE OF SUPERHEAT ON COOLANT EJECTION

Figure 35
SPREADING OF VOID REGIONS (ANTE RESULTS)

Figure 36
REACTIVITY vs VOIDED CORE VOLUME

Figure 37
REACTIVITY vs TIME

axial blanket. Also, the fuel lava could, upon reaching the lower axial blanket, resolidify thus clogging up the coolant channels keeping the rest of the melting fuel from collecting further down: the fuel lava would remain in a region of relatively high importance, namely the lower part of the seed region.

In this model, calculated axially as mentioned before with a strictly one-dimensional diffusion program, melting was initiated at the midplane of the seed region (see figure 38). The molten material included 40 percent of the structural material and was distributed evenly in the coolant channels starting at the boundary between seed and lower axial blanket working upward. Sixty percent of the structure was left in the region voided by fuel to support the remaining upper seed and axial blanket; the 40/60 ratio of melting to remaining structure was picked arbitrarily.

The process of meltdown is carried on until the buildup of fuel lava reaches the top of the still unmolten lower fuel (part c of figure 38). At this point crashdown is assumed to occur, i.e., the remaining part of the upper fuel together with the upper blanket under force of gravity drops down onto the lava pool. Here, two extremes have been analyzed to determine the effect of the remaining structure; in one case, the structural material was distributed homogeneously throughout the seed material (without densification of the structure); in the other case, it is thought of as forming a layer between the lava pool and the upper part of the core (with densification of the structure).

In figure 39 the reactivity for the cylindrical reference design is plotted as a function of the void height H_v . The arrows indicate whether meltdown or crashdown is taking place. Figure 39 also shows results of the same analysis for one module of the modular core design. A comparison shows that the results are very strongly geometry dependent with the cylindrical core exhibiting a definite advantage over the taller, small diameter modular design.

For the cylindrical, 2-ft high core, an initial decrease in reactivity is experienced as the center molten material falls to the edge of the seed region. This is followed by an increase in reactivity as the top of the core falls to the bottom. However, because of the increased leakage of the

Figure 38
REACTOR CORE MELTDOWN MODEL

Figure 39
DESIGN BASIS ACCIDENT - Reactivity as a Function of Void Height

new geometry (i.e., a thinner core with fuel concentrated near the boundary), the reactor does not become critical during the rapid phase of compaction. The modular, 4-ft high core experiences a similar initial decrease in reactivity. However, during the process of meltdown, the buildup of lava reaches a point where the decrease in reactivity, due to the effect of bringing seed material from a high importance, low leakage to a low importance, high leakage region, is finally overcompensated by the reactivity increase, due to the densification of fuel material. From there on, the reactivity rises steadily. During the first part of the crashdown there is no noticeable change in the overall reactivity, because the two reactor systems (upper fuel and lower fuel lava) are far enough separated to be uncoupled. However, later on, the falling upper part acts as a strong reflector for the high density fuel lava region. The reactivity, therefore, increases tremendously. It should be reiterated, that the point at which crashdown occurs has been picked rather arbitrarily, and that it could actually occur anytime after meltdown. The important thing in determining the amount of energy released in the subsequent explosion is the reactivity ramp rate when the reactor goes supercritical. As seen in figures 37 and 38, this depends on when crashdown occurs and could result in reactivity ramps of anywhere between ~ 0 /sec to $\$1800$ /sec. The maximum or most conservative estimate is used in the energy release calculations.

In figure 40 the reactivity increase during crashdown is plotted as a function of time for both the cylindrical and modular core design. Here it was assumed that the upper part in each case is uniformly accelerated by 1 g. With this approach we come up with maximum reactivity insertion rates in the order of $\$1800$ /sec for the modular core. This value must be compared to the homogeneous compaction value of $\$65$ /sec for the cylindrical core since the cylinder does not become critical during the rapid compaction phase of this meltdown model.

Our evaluation of hypothetical situations involving gross voiding and core meltdown, therefore, indicates that the cylindrical arrangement with its positive void effect is certainly no less safe than the modular arrangement and, furthermore, that it may well have distinct safety advantages*. That is, using the ordinary homogenized meltdown model results in reactivity additions

*See following page for further comments.

Note Added In Final Printing

Since the first printing of this report, an error has been found in the calculations for the nonhomogeneous meltdown case for the cylindrical core. Rather than end in a non-critical configuration, the final configuration has a reactivity of plus 2.5% $\Delta k/k$. As a result, the reactivity ramp for this model is greater than for the cylindrical core with homogeneous compaction although less than for the modular core with nonhomogeneous compaction. Consequently, the conclusion still holds that the modular arrangement with a small negative void effect is not significantly safer than the cylindrical arrangement with its positive void effect.

Effective
Multiplication
Constant,
 K_{eff}

Note: Upper Carbide and Blanket Accelerated by 1g

Figure 40

Reactivity as Function of Falling Time During Crashdown

and subsequent energy releases which are about equal for the two cores. The positive reactivity addition from sodium voiding, preceding meltdown, in the cylindrical core produces considerably less energy than does a homogenized meltdown and would disperse the core less violently, reducing the consequences of the MHA. Furthermore, an attempt to make a more realistic approach to the meltdown calculation has predicted a considerable difference in the response of the two cores following meltdown and indicates a possible safety advantage for the cylindrical design.

d. PuC Equation of State

The equation of state derived for PuC is based on the LASL vapor pressure data as reported by Miller.⁽¹⁵⁾ Using the Clausius-Clapeyron equation, this data gives an estimated heat of vaporization of 70 Kcal/Mole at the melting point (1920°K). This data was used to estimate the critical temperature by assuming that the molar entropy of vaporization of PuC behaves like other materials in terms of reduced temperature.⁽¹⁷⁾ The critical volume proposed by Kubit⁽¹⁸⁾ was used together with an assumed critical compressability of 0.27 to obtain the critical pressure. Critical constants derived in this manner are given in the table below where they are compared to those proposed by Kubit.

Critical Constants for PuC

	This Analysis	Kubit
Temperature (Tc) (°K)	4470	3100
Pressure (Pc) (atm)	1490	1036
Volume (Vc) (cm ³ /mole)	67	67

Erout's method⁽¹⁶⁾ was used to calculate pressure and temperature as a function of total energy density and to obtain an estimate of the upper limit of the mechanical work energy available from expansion of the vaporized fuel to atmospheric pressure. A value of 0.87 was used for the initial value of reduced volume which is conservative for the reference design. Results of these calculations are shown in figure 41 where the total energy density is normalized to zero at a reduced temperature of 0.50 which is near the normal boiling point obtained from these calculations (2200°K).

e. Bethe-Tait Analysis

Calculations with the WEAK EXPLOSION code were made using the pressure and temperature relationships shown in figure 41. The excursions were initiated from normal full power conditions using an equivalent spherical representation of the core. The reactivity ramp of \$65/sec obtained from homogeneous compaction of the core at one-g acceleration was used for calculations in which the Doppler constant was the parameter. Results of these calculations are shown in figure 42.

The important role of the Doppler effect in reducing the available work energy from such an excursion is shown by these calculations. An order of magnitude reduction is obtained with a Doppler constant greater than about -0.002. For the reference design, the upper limit of the explosive yield is about 300 Mw-sec which is safely below the limit of 1000 Mw-sec established in the design criteria. As would be expected from the similar assumptions and models used, these results are not unlike those obtained from studies of other reactor systems of similar size.⁽¹⁷⁾

It can be seen from figure 42 that the Doppler constant for the voided core need not be large to obtain a satisfactory reduction in explosive yield. It is only necessary to be safely to the right of the knee in the curve which for this system occurs in the region near $T_{dk}/dT = -0.002$. The minimum calculated value of the Doppler constant for the reference design without moderator is -0.0021 with considerable uncertainty due to uncertainties in physics analysis. The need for the BeO moderator in this core is then in some doubt. It is felt; however, that the present design is appropriately conservative in this respect in light of the relatively large uncertainties in both the Bethe-Tait and physics analyses.

Figure 41
PUC EQUATION OF STATE, $V_r = 0.87$

Figure 42
AVAILABLE WORK ENERGY

C. REVIEW OF GENERAL DESIGN CRITERIA

The 70 AEC General Design Criteria for Nuclear Power Plant Construction Permits⁽³⁾ have been reviewed to determine their appropriateness for application to an LMFER plant of a type similar to the C-E reference design. Because of the limited scope of the present contract, not all of the criteria were considered in detail. The review was limited to the more general criteria and those relating to the core, the protective system, control rods, fuel handling and the reactor vessel. Criteria omitted on this basis are Numbers 4, 5, 17, 38-43, 45-48, 50-65 and 70. These omitted criteria cover either overall plant requirements or details of engineered safeguards. Some of the overall criteria are discussed where it was felt useful comments could be developed and the more general criteria on engineered safeguards are also discussed.

Because C-E is in general agreement with the modifications suggested by the Atomic Industrial Forum⁽¹⁹⁾ (AIF) to the original 70 AEC Criteria, the AIF set of criteria were used as the starting point for this review. This simplifies the review to some extent by eliminating the need to reintroduce and reexplain the AIF modifications. The rationale for the AIF changes is explained in reference 19, portions of which are attached as Appendix B.

AIF recommended eliminating criteria 8, 21, 22 and 24 and in this review, C-E has recommended eliminating criteria 13 and 31. Except for No. 8, these criteria were eliminated because the requirements they state are covered by other criteria. However, these are only the easiest and most obvious criteria which can be eliminated on this basis. It was found that repetition of the same or a very similar requirement in several criteria occurs quite frequently within the set of 70 criteria. This is especially true with respect to requirements related to engineered safeguards. At such time as a fairly general consensus is obtained on LMFER criteria, it is suggested that these criteria be restructured and consolidated to eliminate some of this redundancy.

One of the most basic considerations used in formulating the criteria is the choice to state the criteria in the mandatory form (e.g., "The reactor shall be so designed. . .") rather than in an advisory form. The benefits of this form of statement are recognized, but it imposes certain limitations which must be recognized if one is to avoid arriving at untenable positions. These limitations are enumerated below.

1. The criteria must be derived from what has evolved as acceptable design practice over the years, otherwise there is no assurance that the criteria can be met in practice. It is gathered from the preamble of the statement of the proposed criteria that the intent was to follow this approach, for the statement is made "The General Design Criteria reflect the predominating experience to date with water reactors. . . ." This is an entirely acceptable approach so long as it is adhered to consistently.
2. Criteria stated in the mandatory form cannot be constructed in such a way as to promote the upgrading of safety design with time: for either the criterion must be written so that it can be met by current design practice, in which case it has no upgrading effect, or it must be written to demand a higher standard of safety, in which case it cannot be met by current practice and the result can only be unpleasantness and frustration in licensing activities.
3. Mandatory criteria cannot reasonably be written to cover considerations which are not yet fully understood, or considerations which are too broad to have a precise relation to safety, definable in clear straight-forward terms. When understanding is incomplete, it is impossible to state a general criterion in a definite and unambiguous way (even though the understanding may be quite sufficient for an evaluation of the safety of a particular plant), and a mandatory criterion which is indefinite and ambiguous is a contradiction in terms. When the concepts considered are so broad (e.g., reactivity coefficients) that they can affect the safety of different reactors in different ways, and can interact with other safety considerations, they are not suitable subjects for mandatory criteria. The attempt to regulate them by criteria will either penalize some plants unnecessarily or will prove to set inadequate limitations on other plants.

In formulating the proposed criteria for an LMFBR, the above inherent limitations of this mandatory form were taken into account to the extent practical considering the present state of development for these systems.

To a certain extent, however, it was necessary to attempt to project the future evolution of practical design practice for LMFER's. Certainly, mistakes may have been made in this projection; however, it was felt that this exercise would be of little value if the suggested criteria reflected only the present state-of-the-art. Therefore, the intention here is to indicate what appears at this time to be the best direction in which designs and systems ought to evolve as fast reactors are developed. Despite the fact that the mandatory form has been retained, there is no intention to state hard and fast criteria which must be met before the LMFER can be considered safe.

There is no single path to nuclear plant safety. The trick is to select that path which does provide assurance of safety, and which is compatible with practicability. This means that criteria cannot be imposed arbitrarily, but must be consistent with what is known to be practicable. Since the practicality of a number of LMFER safety criteria has not yet been demonstrated, it is expected that considerable evolution of these criteria will occur as designs are developed in more detail.

Specific comments on criteria are listed below together with the preferred version of the criteria. Specific criteria have been developed for the C-E reference design in order that it will comply with each general criterion; however, this was not necessary in all cases since some criteria are sufficiently specific to require no further elaboration. These specific criteria are sufficient to guarantee meeting the general criteria, and therefore are sufficient to guarantee safety, but are not regarded as necessary since other such criteria could probably be developed which would also be sufficient to guarantee meeting the general criteria.

1. Criterion 1 - Quality Standards (Category A)

Those systems and components of reactor facilities which are essential to the prevention, or the mitigation of the consequences, of nuclear accidents which could cause undue risk to the health and safety of the public shall be indentified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be

performed. Where generally recognized codes and standards pertaining to design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance criteria to be used shall be identified. An indication of the applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance criteria used is required. Where such items are not covered by applicable codes and standards, a showing of adequacy is required.

Comment: Unchanged from AIF.

The following sections are intended to apply only to those areas studied by C-E in the LMFER Follow-on Study Contract.

a. C-E Reference LMFER Criteria

1.1 Systems and components of the reactor will be designed and fabricated in accordance with established codes and/or standards as applicable to assure that their quality is in keeping with safe functioning of the reactor facility.

1.2 To assure such quality will be applied, the design, construction and testing of the reactor system will conform to the following general codes, as applicable:

- a) ASME Boiler and Pressure Vessel Code Sec. III, Nuclear Vessels;
- b) ASA Code for Pressure Piping, E-31.7;
- c) ASME Boiler and Pressure Vessel Code, Sec. VIII, Unfired Pressure Vessels;
- d) National Electrical Code;
- e) Uniform Building Code;
- f) U. S. Code of Federal Regulations, Title 10, Part 20, Protection against Radiation.

The codes listed above represent the minimum requirements and may be supplemented as necessary for the unique LMFER application.

1.3 Additional quality standards requirements may be specified in the following:

- a) Chemistry, heat treatment, and physical properties of alloys, including stainless steels, used in primary and secondary sodium service (ASTM standards).
- b) Welding procedures and qualifications for primary and secondary sodium systems and containment or confinement systems. Inspection and qualification may extend to sources of material for critical components and system (ASME Codes and ASTM Standards).

1.4 To assure adequate quality in design, construction and performance of the associated electrical systems or subsystems, the following shall be considered:

- a) NEMA Standard, Industrial Control IC-1-1965;
- b) Insulated Power Cable Engineers Association Standards;
- c) IEEE Standards for Instrumentation.

1.5 Shielding will be provided to reduce personnel radiation exposures to levels consistent with AECM-0524, or AEC Standards for Protection against Radiation 10 CFR 20.

1.6 The system will be appropriately designed to limit the release of radionuclides to a level which will result in compliance with AEC Site Criteria 10 CFR 100.

2. Criterion 2 - Performance Standards (Category A)

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect public health and safety or to mitigation of their consequences shall be designed, fabricated, and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena such as earthquakes, tornadoes, flooding conditions, winds, ice and other local site effects. The design bases so established shall reflect: (a) appropriate consideration for the most severe of these natural phenomena that have been recorded for the

site and the surrounding area, and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

Comment: Unchanged from AIF.

The following specific criteria refer to the reactor vessel, reactor core and reactivity control systems, since these were the basic areas covered in the C-E LMFER Follow-on Study.

a. C-E Reference LMFER Criteria

2.1 The structures, systems and components of the LMFER facility essential to the prevention or mitigation of accidents which could affect the public health and safety will be designed, fabricated and erected to withstand without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena such as earthquakes, tornadoes, floods, winds, ice and other local site effects.

2.2 All structures and elements of the plant reactor vessel and internals, control rods and core will be designed in accordance with sound engineering practice and will be designed to be capable of withstanding seismic forces corresponding to ground accelerations derived from operational earthquake and/or design basis earthquake conditions. The operational earthquake is an earthquake having a spectrum similar to that of the El Centro, California, earthquake of May 18, 1940, except that the maximum horizontal ground acceleration shall be 0.1 g acting simultaneously with a maximum vertical acceleration of 0.075 g. The design basis earthquake has twice the acceleration of the operational earthquake, i.e., 0.2 g horizontal acting simultaneously with 0.15 g vertical.

2.3 Class A structures, systems and components shall be designed to function following the design basis earthquake. Class A items are those which are designed to ensure that, in the event of an earthquake, the plant will maintain its capability of shutting down, removing decay heat, and being safely contained, and that credible malfunctions will not tend to increase reactivity. Earthquake loads for this case shall be appropriately combined with other applicable loads and the allowable stresses for

this case shall not exceed the yield stresses of the materials, unless it can be shown by the principles of limit analysis that the combined loadings do not exceed 90 percent of the lower bound limit for yield collapse load.

2.4 Class E structures, systems and components (those items which are required to ensure normal, safe plant operation) shall be designed to withstand and operate through loads from the operational earthquake, which are appropriately combined with other applicable loads and the resultant stresses shall not exceed normal working stresses. Both Class E and A structures, systems and components shall be capable of operating through the operational earthquake.

2.5 Earthquake loads and calculated stresses for Class C structures, systems and components (those items which are not required for normal, safe plant operation) shall be determined in accordance with the requirements of the Uniform Building Code for Zone II.

b. Reference Design Features Related to Criteria

2.1 Structures and components of the reactor vessel and internals, reactivity control mechanisms and core are designed so that stresses due to normal loads plus the operational earthquake will not exceed those design values permitted in the applicable codes (ASME Sec. III and Sec. VIII), while stresses due to normal loads plus the design basis earthquake will not exceed the yield stress of the affected materials. Earthquake stresses will be based on a horizontal ground acceleration and a vertical ground acceleration of two-thirds of the horizontal, with the two acting simultaneously.

2.2 The reactor vessel, vessel internals and reactivity control systems and components are Class A items. These systems, structures and components thereof will be equipped with motion limiting devices and snubbers to prevent distortion or permanent damage arising from operational or design basis earthquake loadings.

2.3 The reactor vessel is fixed by top mounted tie rod hanger supports to allow for thermal expansion and equipped with motion limiting snubbers at the lower portion of the vessel body to limit the effect of ground accelerations for dislocating the vessel relative to the heat transport system piping. The piping system will be supported by constant force hangers and will be provided with expansion loops to allow for thermal expansion, in addition to snubbers and shock absorbers to limit relative motions in the pipe and other attached equipment under seismic loads.

2.4 Allowable stresses in the piping system and reactor vessel nozzles will be in conformity with code provisions. Where the codes are not applicable, stresses will be in conformity with the general requirement that either yielding will not be permitted, or that the amount of yielding will be less than the available ductility in the system so as to avoid danger from fracture or tearing.

2.5 The support of valves and other equipment will be designed so that they do not produce unduly high stresses on the reactor vessel body, because of eccentricity of their heavy masses on the pipe runs.

2.6 The reactivity control drives and any other portion of the system affecting the ability of the control rods to scram are Class A items. Deflection of the control rod drives and guide tubes will be minimized by use of secondary support structures. Guide tubes will be supported from lateral movement by a structural support grid above the core. Reactivity drives will be supported by a reactor vessel top closure superstructure. Control rod bundles are provided with generous clearances and guide surfaces within the core lattice to allow for minor deflection of the control rod drive or top closure under seismic conditions corresponding to the design base earthquake.

2.7 The reactor core is to be retained fixed by means of a core locking band so as not to cause any increase in reactivity under seismic loads. The core band is a static, rigid device and is capable of withstanding loads from the design basis earthquake.

3. Criterion 3 - Fire Protection (Category A)

A reactor facility shall be designed such that the probability of events such as fires and explosions and the potential consequences of such events will not result in undue risk to the health and safety of the public. Noncombustible and fire resistant materials shall be used whenever necessary to preclude such risk throughout the facility, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features.

Comment: Unchanged from AIF.

Reactor systems with combustible coolants (e.g., sodium) ought to have one or more special criteria which are concerned with the particular safeguards required for coolant fire protection. Coolant fire protection criteria would most appropriately appear as a separate subsection in section VII - Engineered Safety Features - rather than in section I - Overall Plant Requirements. The above criterion should be retained with an added reference to special criteria for combustible reactor coolants. Criteria and systems for coolant fire protection for the C-E concept are not fully developed and are outside the scope of the present contract.

4. Criterion 6 - Reactor Core Design (Categories A & E)

The reactor core with its related controls and protection systems shall be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits which have been stipulated and justified. The core and related auxiliary system designs shall provide this integrity under all expected conditions of normal operation with appropriate margins for uncertainties and for specified transient situations which can be anticipated.

Comment: Unchanged from AIF.

a. C. E. Reference LMFER Criteria

6.1 The fuel pins and reactor control and protective systems shall be designed for an expected clad failure rate, due to all causes, of 0.3% of the pins per core over the fuel lifetime assuming the first level of

protection is always functional. (The design expected failure rate is established at a level which permits operation with some failed fuel but which will allow substantial increases in clad failures to be detected before the equivalent of more than one subassembly is involved. Since failures are most likely at end-of-life and only 1/3 of the core is near end-of-life at any one time, the expected peak number of failed pins is <0.1%.)

6.2 The fuel pin and fuel subassembly design shall be such that clad failures do not tend to propagate from pin to pin and do not tend to cause loss of the sodium bond between clad and fuel.

6.3 The fuel subassembly shall be designed such that total coolant inlet plugging by single objects is not possible and such that coolant boiling and/or fuel damage from partial or localized plugging is prevented from propagating to adjacent subassemblies before appropriate protective system action is accomplished.

6.4 The protective system shall be designed with redundant back-up trips which will prevent fuel melting or boiling of the bond and/or coolant should the first level of protection fail to properly respond to equipment malfunctions or operational errors. (The question of a proper definition of "acceptable fuel damage limits" is involved in all of the subcriteria above as well as in other criteria considered below. This definition is particularly difficult to establish for the sodium bonded, carbide fueled LMFER because the fuel does not operate near its melting point nor the sodium near its boiling point. It is not primarily a matter of avoiding DNE or critical heat flux as is the case with a LWR, but rather it is a matter on the one hand of avoiding extreme temperatures in those rare cases where a failure of the first level of protection is coincident with some other failure or error and on the other hand of establishing a combination of fuel operating conditions, core design and a control and protective system which, over the life of the core, limits clad failures to manageable levels.)

b. Reference Design Features Related to Criteria

6.1 The fuel pin design minimizes stresses on the clad by absorbing fuel swelling in a large radial sodium filled gap and venting fission gasses to the coolant. Based on available data, the hottest, highest burnup fuel has a low probability of touching the clad at the end of its life. The normal operating condition for the coolant is highly subcooled such that the minimum critical heat flux is at least 10 times the normal peak heat flux.

6.2 The fuel pin vent maintains the bond at a lower pressure than the reactor coolant, thus preventing escape of the bond through a clad failure. This also prevents fission gasses from escaping from clad failures and interfering with cooling of adjacent pins. The normally unstressed portion of the clad adjacent to the vent is manufactured with stress concentrating grooves so that the clad will fail safe in the event of clogging of the vent. Positive pin-to-pin spacing is provided to prevent bowing or excessive swelling of defective fuel from causing significant interference with cooling of adjacent pins.

6.3 The inlet hardware of the fuel subassembly has several inlet openings with different orientation. A solid wrapper tube with outside cooling surrounds each subassembly to prevent radial flow of sodium vapor and to retard radial propagation of melting. The moderator and filler components of the subassembly further inhibit the progress of radial melting. The protective system has three subsystems of different principles to detect local plugging and shutdown the reactor to prevent propagation.

6.4 For each type of equipment failure or operational error, the protective system provides three different kinds of trip signals and three independent sets of poison rods that can shutdown the reactor. Each of these systems has further internal redundancy.

5. Criterion 7 - Reactor Stability

Considering credible temperature, pressure, flow, power, geometric configuration, and other anticipated changes that could occur in the power range as a result of normal or abnormal operations, the reactor system with its related controls and protection systems shall be designed

with appropriate margins for uncertainties to be sufficiently stable to prevent exceeding acceptable fuel damage limits during transients.

Comment: Criterion 7 has been broadened to proscribe all designs which may be subject to various kinds of instabilities, including power oscillations, that could produce unacceptable fuel damage. In agreement with AIF Criterion 8 - Overall Power Coefficient, has been eliminated because Criterion 7 (as stated above) better expresses LMFER requirements in this area. The mechanical - thermal instability of the EER-I core is an example of an instability unrelated to power oscillations or the normal power coefficient which this criterion would cover.

a. C.E. Reference LMFER Criteria

7.1 Design of the reactor control system shall be based on dynamic analysis of the coupled reactor and heat transfer systems using nominal and conservative limiting values of feedback effects in various combinations which tend to produce the most unstable conditions. (This criterion is intended to ensure that the control system itself does not contribute any instabilities to the overall system and to ensure that feedback effects which may be unreliable (e.g. fuel expansion) are not required for stability or do not produce instabilities if their actual values are considerably different than expected or change over the life of the core.)

7.2 The physics and thermal design of the reactor shall be such that the Doppler negative reactivity effect is several times larger than any prompt positive reactivity effects from corresponding changes of power and temperature. The magnitude of the Doppler effect shall also be sufficient to limit power and temperature excursions resulting from the most severe credible single equipment failures or operational errors to values which yield temperatures less than the boiling point of the coolant or bond and the melting point of the fuel. (This criterion reflects a basic philosophy of reliance on the Doppler for inherent stability of the core for prompt effects and also reliance on the Doppler for inherent protection against single credible faults which could produce a rapid

reactivity insertion, e.g., control rod ejection. This is a corollary of criterion 32. The reference to the fuel melting point and the coolant boiling point does not imply that approaching these temperatures is acceptable on a routine basis, but only as a result of the most severe accident resulting from a single equipment failure.)

7.3 The core and reactor internal mechanical design shall be such that structural distortions due to temperature and/or coolant pressure or flow gradients or changes do not cause unmanageable reactivity effects. Mechanical restraint of the core and fuel shall be such that natural radial and axial thermal expansion is not unduly inhibited. (The intent here is to require positive mechanical control of the core configuration, to avoid positive reactivity effects (especially those due to coolant pressure or cross flow) and at the same time to take maximum advantage of the natural negative temperature coefficient due to fuel and structural expansion.)

b. Reference Design Features Related to Criteria

7.1 Dynamic analysis of the type indicated has not been undertaken because of the limited scope of the contract; however, the inherent stability of the core due to the negative Doppler and structural coefficients should make the system easy to control by conventional methods.

7.2 The only prompt positive effect in the reference design is the sodium density effect and this is an order of magnitude lower than the negative Doppler effect. The structural expansion effect is also negative and is of the same order as the Doppler for isothermal temperature changes. The integrated Doppler effect from full power to the initiation of the first fuel failures is $-0.6\% \Delta k/k$ which exceeds the highest worth of a safety rod ($0.5\% \Delta k/k$).

7.3 The reference core is designed with hydraulic axial holddown backed up by the upper instrument structure. Positive radial clamping is provided at the elevation of the seed region to ensure a tight core assembly. The core clamp is spring loaded to allow for radial thermal expansion. Because of the large radial gap between fuel and clad, at least 2/3 of the fuel should always be completely free to expand axially. Appropriate experiments ought to demonstrate that this fuel expansion will provide a reliable increase to the negative power coefficient.

6. Criterion 9 - Reactor Primary Coolant System Boundary (Category A)

The reactor primary coolant system boundary shall be designed, fabricated and constructed so as to have an exceedingly low probability of gross rupture or significant uncontrolled leakage throughout its design lifetime.

Comment: Unchanged from AIF except for substitution of "primary coolant system boundary" for "coolant pressure boundary". This change reflects differences in nature and function between the LWR and LMFER primary coolant system boundaries. In the LMFER the concern is not so much with high internal pressure and pressure stresses but rather with thermal stresses, loss of coolant from large or small leaks, fires from coolant leaks and uncontrolled radioactive cover gas leakage from the primary system. The following specific criteria relate to the reactor vessel and the upper shield assembly.

a. C.E. Reference LMFER Criteria

9.1 The primary coolant system boundary shall be provided with internal thermal shock barriers in all areas where rapid coolant temperature changes can occur due to anticipated transients, where mixing of streams of different temperature can occur, or where coolant streams of different temperature are in close proximity. (Since it will not be practical to demonstrate the exact requirements for liners in all areas, the conservative approach is to require liners unless it can be shown that they are not necessary.)

9.2 All penetrations of the primary coolant system boundary which require mechanical or bellows seals shall be made above the coolant free surface level. Redundant seals shall be provided for these penetrations and provisions for leak testing and replacement of these seals shall be made: (This requirement avoids mechanical seals in sodium service and minimizes the number of under-sodium penetrations. Unavoidable exceptions would be allowed, e.g., pressure transducer installations.)

9.3 The integrated neutron exposure of the primary coolant system boundary shall be limited by appropriate shielding or other means to prevent significant deterioration of its mechanical properties over its design life.

b. Reference Design Features Related to Criteria

9.1 The reactor vessel is protected with thermal shock liners in all areas which could be contacted with outlet sodium. The flow arrangement is such that most of the vessel is contacted by inlet sodium which is relatively cool and not subject to rapid temperature transients.

9.2 All instrument and control rod penetrations are made through the top closure head and are provided with redundant seals. (See Vol. I.)

9.3 With the steel shielding provided, the estimated maximum exposure of the reactor vessel is 3×10^{19} nvt >1.0 Mev.

7. Criterion 10 - Reactor Containment (Category A)

Reactor containment shall be provided. The containment system shall be designed to sustain without undue risk to the health and safety of the public the immediate and long term effects of gross equipment and/or system failures which have been stipulated and justified as being reasonable limiting cases with respect to loads imposed on each component of the containment system. Performance of the containment system shall be evaluated conservatively, considering each such failure to occur independently.

Comment: The concept of containment is somewhat more complex for the LMFBR than it is for the LWR because of the potential of Bethe-Tait type incidents as well as coolant release and/or sodium fire incidents. The former may require specialized containment features and energy absorption shields while the latter requires containment of a more conventional nature. The reference to the pipe rupture has been deleted because it is not, in most systems, an appropriate initiating event for a Bethe-Tait incident. Also, the maximum sodium fire incident would not necessarily be associated with a pipe rupture, this would depend on system design. It is not reasonable to require the containment system to sustain a maximum Bethe-Tait incident and maximum coolant release and/or fire incidents at the same time unless it can be shown that either of these events tends to lead to the other. Details of the containment system for the C.E. concept are not fully developed at the present time.

8. Criterion 11 - Control Room (Category E)

The facility shall be provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit continuous occupancy of the control room under any credible post-accident condition or as an alternative, access to other areas of the facility as necessary to shut down and maintain safe control of the facility without excessive radiation exposures of personnel.

Comment: Unchanged from AIF.

9. Criterion 12 - Instrumentation and Control Systems (Category E)

Instrumentation and controls shall be provided as required to monitor and maintain within prescribed operating ranges those reactor facility operating variables essential to avoid undue risk to the health and safety of the public.

Comment: Unchanged from AIF.

10. Criterion 13 - Fission Process Monitors and Controls (Category E)

Comment: Delete Criterion 13. This requirement adequately covered by Criterion 12.

11. Criterion 14 - Core Protection Systems (Category E)

Core protection systems, together with associated equipment, shall be designed to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.

Comment: Unchanged from AIF.

a. C.E. Reference LMFER Criteria

Criteria 11 through 14 are all concerned with control and protective systems and thus can be treated as a group. Operating variables which require monitoring for safety reasons (Criterion 12) are stated and discussed in section IV of this Volume and in Volume I, section 5. Core protection systems (Criterion 14) are also discussed in the above sections of this report where specific design requirements and criteria are given. The overall functional criterion for control and protective systems is not completely separable from general safety criteria for the reactor system and therefore is mentioned under Criteria 6 and 7 above. A more complete statement of this criterion is given below.

14.1 Accidents shall be considered credible which involve no more than three identifiable and independent system failures or operational errors, each of which has a relatively low probability of occurrence. These may include failures of various components of the control and protection systems. No credible accident shall produce unmanageable reactivity changes from sodium voiding or fuel failure and/or meltdown. Localized voiding or fuel failure in a few fuel subassemblies may occur as a result of localized failures (e.g., plugging); however, such failures shall not produce reactivity changes which are large with respect to the inherent negative reactivity feedbacks of the system (Doppler, structural, etc.). Additionally, local fuel failures or voiding shall not propagate to adjacent, unaffected subassemblies. Credible accidents which involve the core as a whole (e.g., loss of the primary pumps, or control rod ejection) shall not result in any sodium voiding or fuel melting.

b. Reference Design Features Related to Criteria

14.1 Protective systems for the reference design are described and discussed in section IV of this volume and in Volume I, section 5. Twenty-two poison type control subassemblies are provided. The insertion of a maximum of only three of these is required to bring the reactor to zero power for the maximum credible reactivity insertion accident. The rods are divided into three subsystems, each of which can perform essential protective action independently.

12. Criterion 15 - Engineered Safety Features Protection Systems (Category B)

Protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features.

Comment: Unchanged. For application to FER's, this criterion would be concerned with sodium fire control systems, containment building pressure and activity control systems and emergency core cooling systems. The former areas are not being discussed in detail as indicated previously; criteria related to emergency core cooling are given under Section IV - E of these criteria which treats this subject specifically.

13. Criterion 16 - Monitoring Reactor Coolant Leakage (Category E)

Means shall be provided to detect significant uncontrolled leakage from the reactor primary coolant system boundary.

Comment: Unchanged from AIF except for substitution of "primary coolant system" for "coolant pressure" for the reason noted previously. The following criteria apply to the reactor vessel and top closure.

a. C.E. Reference LMFBR Criteria

16.1 Means shall be provided for circulating and monitoring the atmosphere between the reactor vessel and its secondary containment. In addition, leak detectors of a different principle shall be placed in this region at several appropriate locations. (This reflects a dual approach to leak detection which entails the use of local sodium detectors (e.g., spark plug type) at points where leaking sodium should be expected to collect and also the monitoring of gas spaces for the presence of sodium vapor or sodium oxide.)

16.2 A system of frequent or continuous sodium inventory accounting shall be established for the reactor primary coolant system. (This system would involve the use of level measurements, temperature measurements and other appropriate information and a computer to keep track of the amount of coolant in the primary system. It serves as a backup to the other leakage detection systems and also guards against inadvertent transfer or leakage of primary coolant to storage tanks.)

16.3 The upper face of the vessel closure head shall be enclosed by a gas tight secondary barrier and the space within this barrier shall be monitored for leakage of cover gas. (There are several reasons for requiring a "top hat" over the closure head; one of the more important reasons is providing an effective means for controlling and monitoring leakage through its numerous penetrations.)

b. Reference Design Features Related to Criteria

The criteria for monitoring leakage are relatively specific as stated and therefore need no further elaboration or explanation. Detailed designs can be established to meet these criteria by obvious and straight-forward means.

14. Criterion 18 - Monitoring Fuel and Waste Storage (Category E)

Monitoring and alarm instrumentation shall be provided for fuel and waste storage and associated handling areas for conditions that might result in loss of capability to remove decay heat and to detect excessive radiation levels.

Comment: Unchanged from AIF. The following criteria apply to systems associated with fuel handling only.

a. C.E. Reference LMFBR Criteria

18.1 Direct Gamma radiation area monitors shall be provided in all spent fuel handling and storage areas.

18.2 Continuous atmospheric monitors of fission products shall be provided in all spent fuel handling and storage areas.

18.3 All decay heat removal systems shall be instrumented such that their thermal performances can be assessed at any time.

b. Reference Design Features Related to Criteria

18.1 No instrumentation designs for these criteria have been completed.

Reliability and Testability of Protection Systems

15. Criterion 19 - Protection Systems Reliability (Category E)

Protection systems shall be designed for high functional reliability and in-service testability necessary to avoid undue risk to the health and safety of the public.

Comment: Unchanged from AIF.

a. C.E. Reference LMFBR Criteria

19.1 Protection systems shall use only proven components and principles of operation. (This criterion requires that designs of protection systems be based on applicable prior experience in similar service and/or that they have appropriate proof testing prior to service in the operating reactor.)

19.2 The reactor system shall be designed for periodic scram and other emergency procedure testing from full power and other appropriate conditions. Neutron, temperature, flow and other essential detection instruments shall be designed for periodic calibration either in place or by removal and testing. Protective System logic shall be testable by appropriate simulation of accident conditions.

b. Reference Design Features Related to Criteria

19.1 The control rod system is of a conventional design for sodium systems operated mechanically from above the top closure head. FERMI type drive mechanisms are used. Protective systems will use logic and components proved reliable in water reactors.

19.2 Details for testing of protective systems are not fully developed; however, in most areas there appears to be no difficulty in meeting this criterion. Most instruments will be replaceable (thermocouples, neutron detectors, etc.) and thus should not present calibration problems. Primary flowmeters can be calibrated in-place by use of a heat balance. There is some question about how to check and calibrate fission product detectors and entrained gas detectors. It is believed; however, that satisfactory answers can be found for these questionable areas.

16. Criterion 20 - Protection Systems Redundancy and Independence (Category B)

Redundancy and independence designed into protection systems shall be sufficient to assure that no single failure or removal from service of any component or channel of such a system will result in loss of the protection function. The redundancy provided shall include, as a minimum, two channels of protection for each protection function to be served. Different principles should be used to increase the assurance of true independence of redundant components when such different principles can be found, are practicable, and increase the overall safety of the system.

Comment: In this single case C.E. disagrees with the recommendations of the AIF, which would delete the reference to the use of different principles to achieve redundancy. It is believed that the use of redundant protective systems of different principles is an important means of increasing safety in some cases, although one can always visualize other cases in which the

second principle might be so poor as to be unacceptable. The statement in the criterion: "Different principles shall be used where necessary to achieve true independence. . ." is self-defeating. The main advantage of using systems of different principles is to prevent systematic design (or operating) errors from rendering the entire protection system useless. Since the circumstance to be protected against is one which is unforeseen, it is clearly impossible to recognize the necessity for protecting against it. For these reasons, the final statement of the criterion has been retained in the modified form indicated above. Because of the close connection between Criterion 20 and 23, C.E. criteria and design features related to these criteria are discussed together under Criterion 23.

17. Criterion 21 - Single Failure Definition (Category E)

Comment: Delete this criterion per AIF.

18. Criterion 22 - Separation of Protection and Control Instrumentation Systems

Comment: Delete this criterion per AIF.

19. Criterion 23 - Protection Against Multiple Disability for Protection Systems (Category E)

The effects of adverse conditions to which redundant channels or protection systems might be exposed in common, either under normal conditions or those of a nuclear accident, shall not result in loss of the protection function or are tolerable on some other basis.

Comment: Unchanged from AIF. This criterion is retained tentatively with the reservation that it may be necessary to modify it because it is not clear that there are practical methods to meet this criterion considering the state of art anticipated for first generation LMFBR's. It will be difficult to prove that the common exposure of control and safety rods and/or their operating mechanisms to extraordinary thermal or hydraulic conditions or major structural faults in regions in or above the core or in the upper structures cannot result in the loss of their protective function. This is not to say that there is reason to believe that control rods and their operating mechanisms cannot be designed to safely sustain the effects of anticipated adverse conditions to which they are exposed in common, but

rather to indicate that the best proof against unanticipated or poorly understood conditions is the use of redundant protective systems of different principles. The above remarks under Criterion 20 are relevant in this respect as is Criterion 27.

a. C.E. Reference LMFER Criteria

23.1 Each control rod drive, operating mechanism and poison subassembly, to the greatest practical extent, shall be individually isolated and protected from mechanical, thermal, hydraulic and other potentially damaging conditions to which they may be exposed.

23.2 Power supplies, lubricant systems, detectors, electronic systems and other systems essential to the proper functioning of control rods shall be separated into three physically independent subsystems, one subsystem operating the shim-regulating rods and the other two subsystems operating two separate banks of safety rods. The shim-regulating subsystem and the safety subsystem shall use systems and components of different principle when such different principles can be found, are practicable and increase overall safety. (This system of triple redundancy has been adopted in response to the previously indicated requirement (Criterion 14.1) to provide triple protection against all credible system failures and operational errors.)

23.3 The safety rod system shall not share components with any other systems. (The shim-regulating rod system functions primarily as the normal operating control system for the reactor but is also a part of the protective system in that it provides protective set-backs and also scrams under the same conditions as the safety rod system. It is therefore a dual purpose system. By contrast, the safety rod system is required to be a single purpose system.)

b. Reference Design Features Related to Criteria

23.1 The principal design features which isolate and protect control rods and drives are, (1) the individual covers on the drives and the "top hat" cover for all drives, (2) the control rod guide tubes and the structure that supports them and (3) the control subassembly wrapper. Removal of all fuel from control subassemblies provides a double barrier between fuel and control assemblies.

23.2 The division of the shim and safety rod systems in the manner required in this criteria is part of the general specifications and design requirements for the reference system. Physical details have not been worked out. It is anticipated that some sharing of detectors (e.g., flow meters) between the two systems of safety rods will be required; however, it will still be possible to maintain protection at the required level.

23.3 It appears that there will be no difficulty in meeting this requirement with the exception of the sharing of structural support for the safety rods and guide tube with other systems attached to the closure lead and upper shield. This structural sharing and the similarity of mechanical design among all shim and safety rods will be subject to careful examination to assure that these systems are well protected against common faults.

20. Criterion 25 - Demonstration of Functional Operability of Protection Systems (Category E)

Means shall be included for suitable testing of the active components of protection systems while the reactor is in operation to determine if failure or loss of redundancy has occurred.

Comment: Unchanged from AIF. This criterion can be met without difficulty provided its interpretation does not include testing of actual insertion of control subassemblies during operation. Sensors can be tested by cross-checking during operation and trip circuits and logic can be tested with simulated signals. No major differences between LWR and LMFBR practices are anticipated.

a. Reference Design Features Related to Criterion

The portions of the protection system to which this criterion applies are based on LWR design practice and employ the same testing techniques.

21. Criterion 26 - Protection Systems Fail-Safe Design (Category E)

The protection systems shall be designed to fail into a safe state or into a state established as tolerable on a defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or adverse environments (e.g., extreme heat or cold, fire, steam, or water) are experienced.

Comment: Unchanged. Again this criterion can be met without difficulty provided, as is the case with LWR's, it is not applied to the control subassemblies themselves. (Certain types of failures unavoidably tend to cause ejection of the control element in either case; however, this type of failure is extremely unlikely.)

a. Reference Design Features Related to Criterion

The portion of the protective system to which this criterion applies is based on LWR design practice and employs the same type of fail-safe features.

22. Criterion 27 - Redundancy of Reactivity Control (Category A)

Two independent reactivity control systems, preferably of different principles, shall be provided.

Comment: Unchanged from AIF. Although the C.E. control systems are not of basically different principles, the "preferably of different principles" phrase has been retained because it is agreed that a different principle is preferable for the second system if such a system can be developed for LMFER's.

a. C.E. Reference LMFER Criterion

27.1 For discussions of "independence" and "different principles" for these control systems the reader is referred to the discussions under Criteria 20 and 23.

b. Reference Design Features Related to Criterion

27.1 The reference design of the control rods are described in the control rod design section of this report.

27.2 The reference design provides a shim-regulating rod system and a safety rod system. The total number of control rods for both systems is twenty-two. The selection of such a number of rods and actuators was done to obtain maximum reliability of the overall reactivity control system, and to minimize the maximum reactivity worth of any single control rod.

27.3 The thirteen shim-regulating rods and much of the detection equipment are used in a secondary scram system. The instrumentation and remaining detection equipment are separate. All thirteen shim regulating rods are designed to scram during malfunctions, in the same manner as the safety rods. These design features provide one level of mechanical and instrumentation redundancy within the overall reactivity control system.

27.4 The safety rod system is internally divided between two subsystems, viz., safety rod systems numbers one and two in order to provide another level of redundancy within the safety rod system.

27.5 The above described design features provide a maximum of independence of function of the reactivity control systems and also make available several levels of overall redundancy.

23. Criterion 28 - Reactivity Hot Shutdown Capability (Category A)

Each of the reactivity control systems provided shall be independently capable of making and holding the core subcritical from any hot standby or hot operating condition.

Comment: The words "Each of" and "independently" have been inserted in the AIF wording in order to explicitly require each system to have the capacity to shutdown the reactor. Reactivity calculations and rod worths are reported in the control rod design section of this volume.

a. C.E. Reference LMFBR Criterion

28.1 Both the shim-regulating rod system and the safety rod system shall have the capability of supplying a shutdown margin of $-1.0\% \Delta\rho$ at a primary system sodium temperature of 800°F , with the core in its most reactive configuration. (This criterion meets the redundancy reactivity requirements for shutting the core down from hot standby or hot operating conditions. The shutdown margin design value of $-1\% \Delta\rho$ at a primary system temperature sodium temperature of 800°F was provided to ensure the redundant shutdown margin requirements of the criterion on both the shim-regulating rod system and the safety rod system.)

b. Reference Design Features Related to Criterion

28.1 Both the shim-regulating rod system and the safety rod system are designed to control the following reactivity allowances:

Allowances	Value ($-\Delta\rho\%$)
a. Shutdown Margin	1.0
b. Integral power coefficient from zero power to full power	0.3
c. Burnup	0.9
d. Maneuvering and error allowance	<u>0.5</u>
Total	2.7

24. Criterion 29 - Reactivity Shutdown Capability (Category A)

Each of the reactivity control systems provided shall be capable of making the core subcritical under any anticipated operating condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits.

Comment: This criterion has been changed from the AIF version to require both systems to have rapid shutdown capability. The changes in this criterion, together with those in No's. 27 and 28, reflect a more stringent requirement for the redundancy of the control systems than is necessary for LWR's. In a number of important situations, the two control systems provided in LWR plants are complementary rather than redundant. This complementary function is appropriate for LWR's because of the potential for relatively large but slow reactivity insertions from Xenon and temperature transients. This potential does not exist in LMFER's; available excess reactivity in LMFER's is much less than in LWR's; however, the potential insertion rate is relatively high for most of the excess reactivity that is present. A stuck rod requirement has been eliminated because of the increases in redundancy required by this criterion.

a. C.E. Reference LMFER Criterion

29.1 The design scram rates of the shim-regulating rods shall be the same as the design scram rates of the safety rods; viz., at least 0.9 g. (This criterion results in core operating conditions remaining within the acceptable fuel damage limits used in this design for all of the

accidents thus far analyzed. The analysis of the presently recognized possible operating conditions are reported in this volume.)

b. Reference Design Features Related to Criterion

29.1 The control drive mechanism design has been reported elsewhere in the technical notes; however, the drive is basically an electromechanical ball-nut drive unit for rod positioning, and a pneumatic cylinder for scram assist. A shock absorber is also provided for rod deceleration. The other customary design features are included, viz., control circuitry, dual position indication, latching mechanisms, scram releases, etc. These actuators drive thirteen stainless steel clad tantalum shim-regulating rod bundles, and nine boron carbide safety rod bundles.

25. Criterion 30 - Reactivity Holddown Capability (Category E)

One of the reactivity control systems provided shall be capable of making the core subcritical under credible accident conditions with appropriate margins for contingencies and limiting any subsequent return to power such that there will be no undue risk to the health and safety of the public.

Comment: Along with previous remarks, this criterion has been changed from the AIF version to specifically require one of the control systems to have the capability to cover all excess reactivity available. The requirement to meet this criterion even with the sticking of the most reactive rod has been replaced by the requirement for rapid shutdown capability of both reactivity control systems.

a. C.E. Reference LMFER Criterion

30.1 The safety rod system shall be designed to be able to make the core subcritical under the "worst credible accident" condition. This condition shall be equivalent to having all of the rods of the shim-regulating stuck out of the core with the reactor core cooled down to refueling temperature but not unclamping. The safety rods shall be considered fully inserted in the core. The core shall be considered to be a fresh core.

b. Reference Design Features Related to Criteria

30.1 The safety rod system is designed to control the following reactivity:

Allowance	Value ($-\Delta\rho\%$)
a. Heatup allowance from refueling temperature (375°F) to hot standby (800°F)	0.5
b. Integral power coefficient from zero power to full power	0.3
c. Burnup	0.9
d. Maneuvering and error allowance	0.5
e. Shutdown margin (at 375°F)	<u>1.0</u>
Total	3.2

These allowances were felt to be quite reasonable, when one considers the simultaneous malfunctions of thirteen shim-regulating rods and their drives, and the relative inaccessability of additional reactivity under such circumstances. In such an event, as postulated by this criterion the core would be shutdown by at least 1.0% $\Delta\rho$.

30.2 An allowance for the most reactive rod sticking out of the core did not seem appropriate since the thirteen shim rods with a total reactivity worth of -2.9% $\Delta\rho$ are more than equivalent to a single stuck rod.

30.3 A design allowance of -0.5% $\Delta\rho$ was provided for the safety rod system to cover presently recognized and unrecognized uncertainties and to provide core control near the end of core lives. This results in a safety rod system worth of -3.2% $\Delta\rho$. This negative reactivity along with that of the shim-regulating rod system (-2.9% $\Delta\rho$) provides a refueling shutdown margin of -4.2% $\Delta\rho$.

26. Criterion 31 - Reactivity Control Systems Malfunction (Category E)

The reactor protection systems shall be capable of protecting against any single malfunction of the reactivity control system, such as unplanned continuous withdrawal (not ejection or dropout) of a control rod, by limiting reactivity transients to avoid exceeding acceptable fuel damage limits.

Comment: Unchanged from AIF. However, this criterion does not seem appropriate in light of the modifications incorporated into Criterion 29, i.e., the insistence of both control systems possessing equal speed capabilities. Hence, this criterion is not considered necessary. The results of analyses of continuous withdrawal of all control rods are reported in this volume.

27. Criterion 32 - Maximum Reactivity Worth of Control Rods (Category A)

Limits, which include reasonable margin, shall be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot cause boiling of the reactor coolant or fuel pin failures sufficient to prevent adequate emergency core cooling.

Comment: "Boiling of the reactor coolant or fuel pin failures. . ." has been substituted for "rupture of the reactor coolant pressure boundary or disrupt the core. . ." to reflect the severe effects of large reactivity insertions in LMFER's which are comparable to those stated for the LWR.

a. C.E. Reference LMFER Criteria

32.1 Core negative reactivity available in movable control rods shall be divided among a number of rods, in order to minimize individual rod worths.

32.2 Reactor core power level, rate of change of power level, reactor coolant flow and reactor coolant temperature shall be monitored for the detection of adverse power-to-flow conditions.

32.3 The primary reactor coolant system shall be designed such that large accumulations of gas-filled sodium voids can be reliably detected and/or removed from the system.

b. Reference Design Features Related to the Criteria

32.1 The total negative reactivity available in movable control rods is divided among 22 control rods, with a maximum worth of $0.4\% \Delta k/k$. This design provision limits the available control reactivity, and its insertion rate for the prevention of excessive power-to-flow conditions.

32.2 The design of the reactor protection system is described under nuclear and radiation controls section of these criteria. The analysis of several accident situations that could possibly lead to coolant boiling have been reported in this volume, e.g., control rod withdrawal incident from both startup and full power condition, rod ejection incidents; loss of flow incidents; etc. The fuel damage limits used in this design are not exceeded.

Reactor Primary Coolant System Boundary

General Comments: Criteria 33, 34 and 35 as presented originally or as rewritten by AIF are rather specific requirements aimed quite obviously at LWR systems. Literal interpretation of these criteria for LMFER's would yield trivial requirements which would not be sufficient to properly define what is needed for the safety of these systems. On the other hand, criteria 33-35 might be more liberally interpreted to require the primary system to sustain without rupture the effects of the maximum hypothetical Bethe-Tait accident. This is not a desirable interpretation since some LMFER containment systems specifically allow primary system rupture in order to relieve blast pressures in a predictable and desirable manner. Furthermore, maximum accident containment criteria ought to be specifically identified as such and grouped with other Engineered Safety Features criteria.

Under the circumstances contemplated by these criteria (normal operation and/or control rod ejection, cold coolant slug, etc.) LMFER systems are not subject to either "reactivity induced loadings" or "rapidly propagating" type of failures; therefore, these criteria (33-35) ought to be eliminated. With the exception of redundancy (new criterion 33) and surveillance (criterion 36), other requirements for the primary system are adequately covered by criterion 9.

28. Criterion 33 - Redundance of Reactor Primary Coolant System Boundary

An independent secondary boundary shall be provided for critical portions of the reactor primary coolant system which, together with appropriate complementary systems, will prevent any primary coolant system rupture from resulting in loss of coolant sufficient to prevent adequate emergency core cooling.

Comment: This criterion formalizes the generally accepted requirement that some sort of double wall arrangement be used for those portions of the primary system in which leaks could otherwise drain the sodium below safe levels. "Appropriate complementary systems" refers primarily to siphon breakers and reserve supplies of sodium. Although a number of considerations are involved in establishing this requirement, perhaps the most fundamental reason is the fact that only limited supplies of primary coolant are available and no readily available substitute coolant can be used in an emergency. This situation demands that loss of coolant be positively controlled and that adequate reserve supplies of sodium will be available.

a. C.E. Reference LMFER Criteria

33.1 Portions of the reactor vessel, emergency coolant system and primary piping in the reactor cavity region which are below the normal coolant level shall be of double wall design appropriate for control of sodium leakage.

33.2 Siphon breakers shall be provided in the primary system which will prevent reactor vessel drainage below the level of the inlet nozzles in the event of a primary system rupture at a level below this point.

33.3 The volume of the primary system sodium, including reserve tanks if necessary, shall be such that any rupture in the reactor vessel or emergency cooling system will not lower the sodium level below a safe level above the primary outlet nozzles. (This criterion, in combination with criteria 33.1 and 33.2 establishes a double level criterion aimed at ensuring emergency cooling by either the emergency cooling system or the primary system, depending upon where the leak occurs. A low level leak in the reactor vessel or one of the emergency cooling systems will flood the space between the primary system and its secondary containment. This flooding would constitute an unavoidable common fault for the two emergency cooling systems. Although one of these systems should remain operable (if the other is leaking) it should not be completely relied on for this situation.)

b. Reference Design Features Related to Criteria

33.1 The reactor vessel primary piping and emergency cooling system are designed with relatively close fitting secondary jackets which extend above the free surface level on the vessel and to the check valves on the primary piping. This secondary containment completely encloses the primary emergency cooling loops.

33.2 The core barrel which separates inlet and outlet coolant is not sealed to the inside of the vessel but is free to move at the top to compensate for relative thermal expansion and to allow for its removal along with other vessel internals. The clearance provided for these purposes also serves as a siphon breaking path which will effectively prevent siphoning below the level of the inlet nozzles.

33.3 Primary system reserve tank volumes have not been established at this point; however, a reserve sodium volume requirement considerably in excess of that required for this criterion has been established from consideration of Bethe-Tait Accidents. In the event of such an accident, the reactor vessel would be designed to rupture above the free surface level; however, neither it nor its jacket would be relied upon to survive without leaks at lower levels. Reserve coolant capacity is therefore required to be sufficient to fill the reactor cavity to a level above the emergency cooling nozzles.

29. Criterion 36 - Reactor Primary Coolant System Boundary Surveillance (Category A)

Reactor primary coolant system boundary components shall have provisions for inspection, testing, and surveillance of critical areas by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming with current applicable codes shall be provided.

Comment: Unchanged from AIF except for substitution of "primary coolant system" for "coolant pressure".

a. Reference Design Features Related to Criterion

Inspection of the reactor vessel from the inside is provided for by designing all vessel internals to be removable. The refueling scheme provides for wet transfer and storage sufficient to accommodate a complete core loading so that the time required for such an inspection is minimized. This arrangement also allows relatively detailed inspection of internal structures.

Engineered Safety Features

General Comment: Engineered Safety Features constitute four major systems, viz: (E) emergency core cooling, (C) containment, (D) containment pressure reducing, and (E) containment air cleanup. Consideration of all of these systems was specifically excluded from the scope of the present contract except in respect to their general relationship to the reactor safety system. For this reason, details of these systems are not available and, therefore, a complete review of criteria in this category is not appropriate. However, certain selected criteria have been reviewed in light of the safety studies that were performed and comment on these is presented below.

30. Criterion 37 - Engineered Safety Features Basis for Design (Category A)

Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant system, and their protection systems. Such engineered safety features shall be designed to cope with the effects of gross equipment and or system failures which have been stipulated and justified as being reasonable limiting cases with respect to loads and conditions imposed on each engineered safety feature.

Comment: The reference to the pipe rupture incident has been eliminated for the reasons indicated in the comments on criterion 10. For LMFBR's, it is not practical or logical to refer the design basis for all engineered safeguards to a single design basis accident. At least two such accidents (the maximum fire and the maximum Bethe-Tait incident) need to be considered and probably also a third. This third accident might be called: "the maximum primary (radioactive) sodium spill without a fire". This accident would be the design basis for primary coolant fire protection engineered safeguards. The more specific requirements related to this general criterion are covered by other criteria discussed below.

Emergency Core Cooling Systems

31. Criterion 44 - Emergency Core Cooling System Capability (Category A)

An emergency core cooling system with the capability for accomplishing adequate emergency core cooling shall be provided. This core cooling system and the core and protective systems shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function for all credible equipment failures including pump failures, loss of the main heat sink and all sizes of breaks in the reactor coolant piping up to the equivalent of a double-ended rupture of the largest pipe. The performance of such emergency core cooling system shall be evaluated conservatively in each area of uncertainty.

Comment: The AIF preference for a single emergency core cooling system has been retained even though the C.E. conceptual design has two systems. The main loops must be relied upon to handle the initial effects of a major pipe rupture or other cooling fault that occurs while the reactor is at power. After the system is shutdown, operable main loops can obviously handle most emergency cooling requirements either by natural convection or with pumps operated by emergency power. It is not unreasonable to assume that for certain designs it can be shown that one or more of the main loops, operating in an emergency mode, can handle all anticipated emergency cooling requirements and that there is sufficient redundancy in the main system to provide the required assurance of cooling.

However in the present C.E. design, it does not appear that it is possible to guarantee that the level in the primary system cannot drop to a point that will render the main system inoperable. Also, it is not at all obvious that there are not conditions when it would be essential or at least highly desirable to deliberately drain the main loops. In this case, the emergency loops would be essential. For these reasons, the C.E. design provides two emergency cooling systems (emergency mode of main loops and emergency loops).

Containment

32. Criterion 49 - Containment Design Basis (Category A)

The containment structure, including access openings and penetrations, and any necessary containment heat removal systems shall be designed so that the leakage of radioactive materials from the containment structure will not result in undue risk to the health and safety of the public under conditions of pressure, temperature and airborne activity resulting from various gross equipment and/or system failures which have been stipulated and justified as being reasonable limiting cases with respect to loads imposed on each component of the containment system. Failures that shall be considered include, but are not limited to, a major primary pipe rupture, a severe power flow mismatch in the core leading to coolant boiling and a major system failure leading to exposure of primary coolant to the containment building atmosphere. The consequences of such failures shall be evaluated conservatively considering each to occur independently and assuming coincident failure of a single active component of the protective system.

Comment: As is the case with the original criterion, this is a more specific restatement of criterion 10. The comments made with respect to criterion 10 are therefore relevant. As indicated in the comments on Criterion 37, there are three kinds of incidents which influence containment design. As is the case with SEFOR and many proposed LMFER designs, the C.E. arrangement employs double containment; the first (inner) barrier completely encloses the primary system with biological shields and a gas tight shield liner; the second (outer) barrier is the containment building. The inner containment is complementary rather than redundant in that its most important function is to greatly attenuate and suppress the pressure, temperature and radiation effect of major accidents. A major primary coolant fire in the containment building requires breaching the inner containment as well as coolant release. Since the inner containment would be opened deliberately for refueling or repair only after reactor shutdown and sodium decay for a prescribed period of time, and since the inner containment would be specifically designed with considerable margin to resist the pressures and temperatures of the maximum coolant release and Bethe-Tait incidents. It would not be reasonable to require the Bethe-Tait incident to be considered to occur simultaneously with either of the other major accidents.

Fuel and Waste Storage

Comments have necessarily been restricted within this section to considerations of the criteria from the standpoint of the fuel handling system design work that has been performed for the heat generation system. The scope of work has not included any considerations of waste storage systems.

33. Criterion 66 - Prevention of Fuel Storage Criticality (Category E)

Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls.

Comment: Unchanged.

a. C.E. Reference LMFBR Criterion

66.1 The new and spent fuel storage facilities shall be designed to prevent criticality by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls.

b. Reference Design Features Related to Criterion

66.1 Prevention of criticality in the spent fuel storage pools is prevented in the reference design by monitoring emitted radiations, by geometrical spacings, and by the use of mechanical boron poison plates. Fuel storage is designed on 12-inch centers with borated plates added between bundles for redundancy of criticality control. The present scope of work has not allowed detailed designs for fuel storage racks from the mechanical standpoint (earthquake stresses, etc.).

66.2 Preliminary design calculations indicate no fuel criticality difficulties while the fuel bundles are in the fuel transfer system.

66.3 Preliminary design calculations have been performed to provide the design for the new fuel storage racks. Also, in this instance, geometrical spacing and boron plates are used to provide a redundancy of criticality control even in the event of flooding by water.

34. Criterion 67 - Fuel and Waste Storage Decay Heat (Category E)

Reliable decay heat removal systems shall be designed to prevent damage to the fuel in storage facilities that could result in radioactivity release which would result in undue risk to the health and safety of the public.

Comment: Unchanged from AIF.

a. C.E. Reference LMFER Criterion

67.1 Decay heat removal systems shall be designed to prevent damage to the fuel in storage facilities that could result in radioactivity releases. Several levels of coolant redundancy shall be provided to insure reliability of functioning.

b. Reference Design Features Related to Criterion

67.1 The fuel transfer system has been designed such that fuel is cooled by three independent loops which remove heat from the sodium pot by NaK coolant and which transfer the decay heat to the containment building atmosphere. Each of the three independent loops is designed to handle the maximum heat load of the hottest core subassembly. Each of the three loops is designed to function either with natural circulation or with the triple redundant power supplies that are provided. The fuel handling machine systems have been designed to be fail safe from the fuel melting standpoint in the event of loss of power.

67.2 All fuel storage pools have been designed with a double containment and with leak detection monitoring between the two walls. With this design, loss of the sodium coolant level becomes impossible. The present design calls for storage capabilities for one reactor loading, including core, blanket, and reflector subassemblies (300 locations). All cover gases are monitored for any rise in fission product activity (refer to criterion 18).

67.3 The ex-reactor-vessel fuel storage system is comprised of the tank, a sodium heat transport loop with a pump, a heat dump of 5 Mwe maximum capacity, and a sodium clean-up system.

35. Criterion 68 - Fuel and Waste Storage Radiation Shielding (Category E)

Adequate shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities.

Comment: Unchanged from AIF.

a. C.E. Reference LMFER Criterion

68.1 Adequate shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities.

b. Reference Design Features Related to Criterion

68.1 Shielding has been provided in the CE-LMFER reference design to meet the intent of this criterion. Further detailed shielding analysis will be required as more layouts of these facilities are completed.

36. Criterion 69 - Protection Against Radioactivity Release from Spent Fuel and Waste Storage (Category E)

Provisions shall be made in the design of fuel and waste storage facilities such that no undue risk to the health and safety of the public could result from an accidental release of radioactivity.

Comment: Unchanged from AIF.

a. C.E. Reference LMFER Criterion

69.1 Design provisions shall be made in the design of fuel and waste storage facilities such that no undue risk to the health and safety of the public could result from an accidental release of radioactivity.

b. Reference Design Features Related to Criterion

69.1 Redundant containment of fuel is provided in the reference design from the time the fuel is irradiated. When the fuel is in the fuel handling machine transfer system, two degrees of containment are afforded, viz., the machine containment and the reactor containment building. When the fuel is stored in the spent fuel storage pools, two degrees of containment are provided, viz., the pool containment and the containment building. When the fuel has been prepared for shipping, three degrees of containment are available to it, viz., the potting compound (a lead-bismuth compound), the seal-welded-transfer pot that the fuel elements are potted in, and the

shipping cask that the potted fuel elements are placed in for shipment. These containment systems have been designed to account for the vented fuel element design. Although only a few considerations have been made for the handling and shipment of damaged fuel, it is felt at this time that such fuel elements would be handled separately from the ordinary sequence of handling for undamaged fuel. Further design considerations would be necessary for such fuel.

IV. CONTROL RODS

This section describes the control rod arrangement selected for the reference design and the engineering analysis leading to its development. The two basic principals which guided selection among alternate control rod arrangements were: (1) simplicity and reliability to minimize difficulties during normal operation, and (2) redundancy, isolation and independence to assure adequate emergency response. Table VII summarizes the main characteristics of the control system selected.

Table VII

Control Rods

Summary of Main Characteristics

1. Arrangement	a) Each control subassembly occupies a full fuel position. Control subassemblies contain no fuel.b) There are two kinds of control subassemblies which function independently: 1. Shim-regulating subassemblies.2. Safety subassemblies.c) See figure 43 for location of control subassemblies in the core.
2. Control rod bundle design	a) Control rods are solid cylinders sized for adequate cooling and assembled in clusters attached to a central support tube. Each shim-regulating bundle consists of 10 stainless steel clad, 1.25 in. OD rods. Each safety bundle consists of 18 stainless steel clad, 1.21 in. OD rods.b) Empty control positions are filled with sodium.
3. Shroud design	a) Shrouds are externally identical to fuel subassemblies except for upper hardware.b) A barrier is provided as part of the shroud to thermally isolate shim-regulating rod cooling sodium from fuel cooling sodium.

Table VII (Cont.)

Control Rods

Summary of Main Characteristics

4. Materials	a) Shim-regulating rods: stainless clad tantalum. b) Safety rods: stainless clad boron carbide. c) Structure: stainless steel.
5. Method of cooling	a) Each control subassembly is supplied with a flow of sodium from the high pressure plenum controlled by inlet orifices. Shim-regulating orifices are fixed and safety orifices have two positions - open and closed. b) Sodium flow to each control subassembly is sufficient to handle the heat load of the fully inserted control rod bundle at 100% power and normal core mixed-mean temperature rise. (Safety rod bundles are not designed for normal operation in the seed region; however, the design is such that the reactor can be operated temporarily with an inserted safety rod.)
6. Reactivity control requirements (% $\Delta k/k$)	
a) Shim-Regulating System	
1) Burnup - one cycle	0.863
2) 100% to 0% power	0.355
3) Maneuvering and Error Allowance	0.500
4) Shutdown margin	<u>1.000</u>
Total operational reactivity control requirement for shim system	2.718
b) Safety System	
1) Burnup - one cycle	0.863
2) 100% to 0% power	0.355
3) Maneuvering and Error Allowance	0.500
4) Shutdown margin	1.000
5) 800°F to 375°F	<u>0.521</u>
Total operational reactivity control requirement for safety system	3.239

Table VII (Cont.)

Control Rods

Summary of Main Characteristics

7. Reactivity worth of control subassemblies ($\% \Delta k/k$)

a) Shim-Regulating Rod Bundles

- 1) Worth per bundle fully inserted in critical equilibrium core

Average	0.223
Peak	0.225

- 2) Total worth fully inserted ('safety rods out')

2.90

b) Safety Rod Bundles

- 1) Worth per bundle fully inserted in critical equilibrium core

Average	0.400
Peak	0.400

- 2) Total worth fully inserted (shim-regulating rods out)

3.60

8. Thermal and hydraulic characteristics of control rods

	Shim-Regulating Half Inserted	Normal	Safety Fully Inserted
a) Heat generation rate (kw/ft/rod)			
Peak	55	5	17
b) Peak center temperature including hot channel factors F	1650	1360	2340
c) Peak surface heat flux (Btu/hr-ft ²)	5.7×10^5	0.52×10^5	1.8×10^5
d) Peak flow for a subassembly (lb/hr)	3.0×10^4	2.3×10^4	
e) Peak velocity (ft/sec)	2.6	1.7	
f) Pressure drop (psi)	<1.0	<1.0	

- LEGEND**
- CONTROL - 13
 - ⊠—SAFETY - 9
 - ▨—INNER SEED - 111
 - OUTER SEED - 108
 - BLANKET - 60
 - ⊞—REFLECTOR - 66
 - * EQUIVALENT DIA. 8.87'

Figure 43
CORE ARRANGEMENT

A. REFERENCE DESIGN CONTROL REQUIREMENTS

1. Calculation of Operational Reactivity Changes

The first step in the selection of an appropriate control assembly design for the reference core was the determination of the operational reactivity control requirements using MACH-1 and CYCLE-2 calculations.

MACH-1 eigenvalue calculations were completed for determining the reactivity change due to the core thermal-structural expansion in the reference design. The mode of expansion assumed was free expansion from 375°F (refueling temperature) to 800°F, clamping at 800°F, followed by expansion* to 940°F. Changes in the seed sodium volume fraction and seed diameter for each configuration change of the model are given in Table VIII together with the calculated eigenvalue differences.

Table VIII
Core Thermal-Structural Expansion

Configuration Change	Temperature Change °F	Fractional Change in Seed Sodium Volume Fraction	Percent Change in Seed Diameter	Eigenvalue Difference Δk
Cold to Operating at Zero Power	375 - 800	1.00410	+0.407	-0.00247
Application of Core Clamp	800 - 800	0.98439	-0.466	+0.00305
Zero Power to Full Power (Clamped)	800 - 940	0.99688	+0.138	-0.00080

Previously computed values of the Doppler coefficient were used to estimate the isothermal Doppler reactivity changes between the refueling and operating temperatures at zero power and between zero power and full power.

The CYCLE-2 program was used to obtain an estimate of the reactivity change due to burnup over one operating interval which corresponds to an incremental burnup of 33,333 Mwd/Mt between refuelings. A reactivity decrease of $-0.863\% \Delta k/k$ was obtained.

* The restraining core clamp is designed to be sufficiently elastic to allow full thermal expansion while the core is clamped.

2. Formulation of Control Requirements

Control requirements for the reference design were formulated from the calculated operational reactivity changes in accordance with applicable provisions of the seventy USAEC General Design Criteria (see section III-C). The total control requirement for the safety-rod system was determined assuming the core clamp is not removed and the core is in its most reactive condition. Results are given in Table IX.

Table IX
Reactivity Effects and Associated
Control Requirements for Reference Design

	Reactivity Requirement $\% \Delta k/k$	
	Shim-Regulating System	Safety Rod System
Burnup-One Cycle	0.863	0.863
Full to Zero Power		
Expansion Effect	0.080	0.080
Fuel Doppler Effect	0.275	0.275
Maneuvering and Error Allowance	0.500	0.500
Shutdown Margin	1.000	1.000
Operating to Cold at Zero Power		
Expansion Effect	--	0.247
Fuel Doppler Effect	--	0.274
Total Control Requirement	2.718	3.239

3. Rod Worth Calculations

Assembly worth calculations described herein were performed by the "poison cell" method in which the reactor seed region containing N control subassemblies is assumed to be composed of N identical cells. Two different MACH eigenvalue calculations with reflective boundary conditions are carried

out for each cell: (1) with sodium present in the assembly position; and (2) with the fully inserted subassembly. Subtraction of the two eigenvalues for the cell gives the total worth of N fully inserted assemblies in the complete core.

a. Preliminary Calculations

Some preliminary calculations were first performed to aid in the design of two control systems which meet the requirements discussed in the preceding section. The volume fractions and enrichment used in these preliminary calculations, however, were not modified to allow for the presence of the control assemblies themselves. Both annular and cylindrical rods were studied. The relative worth of natural E_4C rods (70% theoretical density) and tantalum rods were considered as well as the difference between annular geometry and cylindrical geometry rods. These preliminary results are summarized in Table X.

Table X

Comparison of Worth of Equivalent E_4C and Ta Rods

Geometry	Diameter (in.)	Number Rods	Cell Radius (cm)	Worth, $\% \Delta k/k$		Relative Worth E_4C/Ta
				E_4C	Ta	
Cylindrical	4.48	13	36.99	3.884	2.476	1.57
Cylindrical	4.04	16	33.35	4.313	2.705	1.59
Cylindrical	3.61	20	29.84	4.766	2.942	1.62
Cylindrical	3.23	25	26.66	5.212	3.173	1.64
Annular	$D_o=4$ $D_i=3$	13	36.90	2.209	1.296	1.70

The relative worths given in the table indicate that the boron carbide rods are worth about 1.6 times as much as equivalent rods composed of tantalum. This is because at high energies, the capture cross sections of boron-10 are greater than those for tantalum.

The results of an annular "poison cell" calculation are also given in Table X for both E_4C and tantalum. The worth per unit cross-sectional

area in the annular geometry is about 1.5 times that in cylindrical geometry for boron carbide and 1.4 times that in cylindrical geometry for tantalum which indicates a slightly greater self-shielding effect for the boron carbide material, as is expected because of its greater worth.

In order to verify the adequacy of diffusion theory for assembly worth calculations in the reference design, the preliminary cell calculations for 4.48 inch diameter tantalum rods were repeated with S_4 transport theory using the DTF-IV code.⁽²¹⁾ A worth of 2.682% $\Delta k/k$ was obtained to be compared with 2.476% $\Delta k/k$ obtained with diffusion theory. Although the worths are underestimated slightly, the comparison indicates that diffusion theory is satisfactory for approximate calculations such as these.

b. Final Calculations

The control configuration chosen to meet the requirements of the reference design consists of thirteen tantalum shim-regulating assemblies, and six enriched boron carbide (70% theoretical density) and three natural boron carbide (63% T.D.) safety assemblies. Tantalum was selected for the shim-regulating assemblies in spite of its lower relative worth because of its longer lifetime, better thermal properties and because of the absence of gas generation during operation. Since the boron carbide safety assemblies are intended to be used only during shutdown, the higher worth material is preferred.

Although the preliminary calculations reported in Table X indicate that the control requirements could be satisfied with a relatively small number of large (i.e., >4 inch) diameter rods, the calculation of the heating rate due to neutron capture for a 4.5 inch diameter tantalum rod yielded a peak linear rate in excess of 300 kw/ft. Consequently, smaller diameter rods were used in the two types of control subassemblies. An annular arrangement of ten 1.25 inch diameter rods was chosen for the thirteen shim-regulating assemblies. This design is depicted in figure 44. Eighteen 1.21 inch diameter rods were chosen for the nine safety-rod assemblies depicted in figure 45.

"Poison cell" diffusion calculations were used to obtain a first estimate of the total worth of the fully inserted assemblies. The worth of the tantalum shim-regulating assemblies was calculated with the boron carbide safety

CONTROL ROD WITH HELICAL SPACER
SCALE NONE

CONTROL & FUEL SUBASSEMBLIES
SCALE: 1/8

CONTROL SUBASSEMBLY
SCALE: 1/8

SECTION A-A
SCALE: 1/2

SECTION B-B
SCALE: 1/2

SECTION C-C
SCALE: 1/2

SECTION D-D
SCALE: 1/2

1-DIM SHOWN ARE AT OPERATING TEMP

Figure 44
CONTROL SUBASSEMBLY-SHIM-REGULATING

CONTROL ROD WITH HELICAL SPACER
SCALE: 1/12

CONTROL & FUEL SUBASSEMBLIES
SCALE: 1/8

CONTROL SUBASSEMBLY
SCALE: 1/16

1-DIM SHOWN ARE AT OPERATING TEMP

Figure 45
CONTROL SUBASSEMBLY - SAFETY

assemblies out and vice versa. Natural boron was assumed for the safety rods. The radial distribution of the neutron importance function shown in figure 46 was then used to correct the average assembly worths obtained from the poison cell calculations for the actual position of each control subassembly in the seed region. These positions are indicated on the figure together with the average value of $\phi\phi^*$. Results of the worth calculations are given in Table XI.

Table XI

Calculated Worths of Reference Design Control Subassemblies

	Worth Fully Inserted $\% \Delta k/k$
a. Tantalum Shim-Regulating Assemblies - (13)	
1. "Poison Cell" Worths	
Total Worth	2.378
Average Assembly Worth	0.183
2. Importance Corrected Worths	
Total Worth	2.904
Average Assembly Worth	0.223
Peak Assembly Worth	0.225
b. Natural B_4C Safety Assemblies (9)	
1. "Poison Cell" Worths	
Total Worth	3.253
Average Assembly Worth	0.361
2. Importance Corrected Worths	
Total Worth	3.398
Average Position Worth/Assembly	0.378
Peak Inner Position Worth/Assembly	0.442 (70% dense)
	0.400 (63% dense)
Outer Position Worth/Assembly	0.360

After importance corrections for position in the core, a total worth of $2.90\% \Delta k/k$ was obtained for the shim-regulating assemblies which is sufficient to meet the requirements for that system given in Table IX. The natural boron carbide safety assemblies with a worth of $3.40\% \Delta k/k$, do meet the formulated requirement; however, the peak worth of $0.442\% \Delta k/k$ for

Figure 46

RADIAL IMPORTANCE DISTRIBUTION IN REFERENCE DESIGN

this arrangement would be unnecessarily high. A more satisfactory arrangement was obtained by reducing the boron carbide density in the inner assemblies and enriching the outer assemblies in boron-10.

To determine the enrichment required, additional "poison cell" calculations were carried out with varying percentages of boron-10. The results of these additional calculations are plotted in figure 47, which is a plot of the fractional increase in rod worth with boron-10 enrichment. From this figure, it was determined that an enrichment of 33% is sufficient to meet the formulated requirement for the safety system with some margin for error.

B. ROD HEATING CALCULATIONS

1. Tantalum Shim-Regulating Rods

The heating rate in a single 1.25 inch diameter rod, ten of which comprise each of the 13 shim-regulating assemblies, was estimated as the sum of the following three contributions: (1) absorption of radiative capture gammas; (2) absorption of core gammas; and (3) beta and gamma heating resulting from radioisotope decay. The computation of each of these component effects is discussed below.

a. Absorption of Radiative Capture Gammas

The volumetric capture rate in the rods was calculated using the multigroup capture cross sections for tantalum together with the peak multigroup flux obtained from a radial MACH calculation. For the peak flux of 7.08×10^{15} neutrons/cm²-sec which occurs at the center of the seed region, the corresponding capture rate (i.e., $\Phi \Sigma_c$) is 19.3×10^{13} captures/cm³-sec. Flux depression within the rods was neglected in determining this rate. The volumetric heating rate in Mev/cm³-sec is obtained from the capture rate using the relation,

$$Q = \Phi \Sigma_c \sum_i E_{\gamma i} \left(\frac{\mu_e}{\mu_T} \right)_i (1 - f_i) \quad (1)$$

where the summation is taken over all i groups of the radiative capture gamma energy spectrum and where $E_{\gamma i}$ is the average energy of the radiative capture gamma photon emitted in group i , $\left(\frac{\mu_e}{\mu_T} \right)_i$ is the ratio of energy absorption and total linear attenuation coefficients for tantalum in group i , and f_i

Figure 47
FRACTIONAL INCREASE IN SAFETY ROD WORTH
WITH BORON-10 ENRICHMENT

is the fraction of the gamma radiation in the i th group which escapes from the rods, and hence, does not contribute to the heating rate. In view of the lack of more complete data, the energy spectrum of the radiative capture gammas was assumed to be the same as that for thermal neutron capture. Data used in the calculation are given in Table XII. The computed volumetric heating rate is 59×10^{13} Mev/cm³-sec or 94.5 watts/cc. A linear heat rate of 22.8 kw/ft was obtained by multiplying the volumetric rate by the cross-sectional area of the single rod.

b. Absorption of Core Gammas

The heating rate due to the absorption of core gammas is given by the expression,

$$Q = ES_v \frac{\mu_e}{\mu_T} \left(1 - e^{-\mu_T R_o} \right) \quad (2)$$

where E is a buildup factor for scattered photons, S_v is the volumetric gamma flux in Mev/cc-sec, μ_e and μ_T are the energy and total linear absorption coefficients in cm⁻¹ and R_o is the outer radius of the rod in cm. The core gamma specific flux level used in the calculation of this contribution to the total heating rate was assumed to be the same as that measured in the SNAP-10 reactor ground tests.⁽²⁴⁾ These and other data used are given in Table XIII. The volumetric gamma fluxes used in equation (2) were obtained by multiplying the gamma fluxes in Table VII by the seed region power density of 590 watts/cc. The final result of the calculations is a linear heat rate of 13.9 kw/ft due to the absorption of core gammas.

c. Radioisotope Decay Heating

The physical process for decay heating in tantalum control rods is depicted below:

Table XII

Energy Spectrum of Radiative Capture
Gammas for Tantalum*

Quantity	Group i				
	1	2	3	4	5
Average Gamma Photon Energy (Mev)	0.5	1.5	2.5	4.0	6.0
Average Energy per Capture E_{γ_i} (Mev/Capture)	0.685	1.483	1.650	2.200	0.300
$\frac{\mu e}{\mu T}$ i	0.631	0.547	0.672	0.772	0.85
f_i	0.13	0.28	0.32	0.32	0.32

* See References (22) and (23).

Table XIII

Data Used in Calculation of Heat Rate Due to
Absorption of Core Gammas in Tantalum

Group	Gamma Energy (Mev)	Gamma Flux (Mev/watt-sec)	μT (cm^{-1})	Buildup Factor B	$\frac{\mu e}{\mu T}$
1	0.5	1.83×10^{11}	2.04	2.15	0.714
2	1.5	1.79×10^{11}	0.84	2.00	0.776
3	2.5	8.74×10^{10}	0.70	1.60	0.662
4	3.5	3.93×10^{10}	0.67	1.40	0.751
5	4.5	1.31×10^{10}	0.67	1.25	0.799
6	5.5	4.28×10^9	0.67	1.20	0.842
7	6.5	2.21×10^9	0.67	1.15	0.861
8	7.5	8.06×10^8	0.67	1.15	0.880
9	8.5	3.72×10^8	0.67	1.10	0.901
10	9.5	1.55×10^8	0.67	1.10	0.904

This process is described mathematically by differential equations and solutions for the two tantalum isotopes:

$$\text{Ta}^{181} \left\{ \begin{array}{l} \frac{dN_1}{dt} = -\phi\sigma_c N_1 \\ N_1 = N_{10} e^{-\phi\sigma_c t} \end{array} \right. \quad (3)$$

$$\text{Ta}^{182} \left\{ \begin{array}{l} \frac{dN_2}{dt} = \phi\sigma_c N_1 - \lambda_2 N_2 \end{array} \right. \quad (5)$$

$$N_2 = \frac{N_{10}}{\frac{\lambda_2}{\phi\sigma_c} - 1} \left[\begin{array}{c} e^{-\phi\sigma_c t} \\ -e^{-\lambda_2 t} \end{array} \right] \quad (6)$$

In the above series of equations, N_{10} is the amount of tantalum present in a clean rod.

The maximum heating rate due to radioisotope decay occurs at the time when the concentration of Ta^{182} is greatest. This time is obtained by setting equation (5) equal to zero and solving for t . Using the peak capture rate determined in the analysis of neutron capture heating (i.e., $\phi\sigma_c$), the time obtained for the maximum concentration is 512 days. The disintegration rate at this time is then 16.6×10^{13} disintegrations per cm^3 per second.

The heating rate due to the absorption of gammas from these disintegrations is then

$$Q = \lambda_2 N_2 \sum_i \left(1 - f_i \right) \left(\frac{\mu_e}{\mu_T} \right)_i g_i E_{\gamma_i} \quad (7)$$

where g_i is the fraction of the disintegrations producing gammas in the i th group. Decay data for tantalum-182⁽²⁵⁾ is given in Table XIV together with other quantities used in solving equation (7). A linear heat rate of 5.9 kw/ft was computed from this source.

The beta heating rate from these disintegrations was calculated assuming that each disintegration produces a 0.5 Mev beta.⁽²⁶⁾ All of the kinetic energy of the beta particles was assumed to contribute to rod heating (i.e., $Q = 0.5 \lambda_2 N_2$) yielding a linear rate of 3.2 kw/ft.

The total linear rate from radioisotope decay heating is then the sum of the beta and the gamma contributions, or 9 kw/ft.

2. Boron Carbide Safety Rods

Since radioisotope decay does not occur in boron carbide rods, the heating rate in the 1.21 inch diameter rods comprising each of the nine safety assemblies was calculated from the contributions due to neutron capture and absorption of core gammas.

Table XIV

Data Used in Calculation of Heat Rate Due to
Absorption of Gammas from Decay of Ta¹⁸²

Group	$E\gamma_i$ (Mev)	g_i	$\left(\frac{\mu e}{\mu T} \right)_i$	f_i
1	0.05	0.24	1.000	0.00
2	0.25	1.234	1.000	0.00
3	0.75	0.320	0.4375	0.11
4	1.25	0.946	0.5699	0.26

a. Neutron Capture Heating

Heating in boron carbide rods occurs as a result of absorptions in the boron-10 isotope:

The heating rate is given by,

$$Q = \Phi \Sigma_c \left[(0.06)(2.79) + (0.94)(2.31) + (0.94)(1 - f)(0.478) \right] \quad (8)$$

where $\Phi \Sigma_c$ is the volumetric capture rate in the boron-10 isotope and f is the fraction of the 0.478 Mev gammas which escape from the rods ($f = 0.77$ for 1.21 inch diameter rods).

Using the peak multigroup fluxes and cross sections for boron-10, $\Phi \Sigma_c = 21.7 \times 10^{13}$ captures/cm³-sec yielding a linear heat rate of 16.1 kw/ft.

b. Absorption of Core Gammas

The calculation of this quantity was essentially the same as that described previously for the tantalum rods. A linear heat rate of 0.9 kw/ft was obtained. Linear absorption coefficients used are listed in Table XV.

Table XV

Data Used in Calculation of Heat Rate Due to
Absorption of Core Gammas in Boron-10

Group	Gamma Energy (Mev)	μT (cm ⁻¹)	$\frac{\mu e}{\mu T}$	Buildup Factor E
1	0.5	0.148	0.392	1.0
2	1.5	0.092	0.478	1.0
3	2.5	0.072	0.514	1.0
4	3.5	0.060	0.550	1.0
5	4.5	0.060	0.550	1.0
6	5.5	0.060	0.550	1.0
7	6.5	0.060	0.550	1.0
8	7.5	0.060	0.550	1.0
9	8.5	0.060	0.550	1.0
10	9.5	0.060	0.550	1.0

3. Summary

Calculated peak linear heating rates obtained for the tantalum shim and B₄C safety rods are summarized in Table XVI. The rates for partially withdrawn rods were obtained by multiplying the inserted total by 1.2 based on an estimate of the lightening rod effect.

Table XVI

Calculated Peak Heating Rates for 1.25 inch Diameter
Tantalum Shim and 1.21 inch Diameter Enriched
Boron Carbide Safety Rods

	Peak Linear Rate (kw/ft)	
	Tantalum	E ₄ C
Neutron Capture	22.80	16.1
Core Gammas	13.92	0.9
Radioisotope Decay		
Beta	3.21	0.00
Gamma	5.86	0.00
Total for Inserted Rod	45.79	17.0
Total for Partially Withdrawn Rod	54.95	

C. CONTROL ROD COOLING METHODS

Several cooling methods were considered for control rod bundles. Two major problems are involved: heat transfer from the poison to the sodium coolant and heat transport out of the subassembly via the sodium. The first problem is easily solved by adjusting the diameter of the poison rods to make center temperatures and surface heat fluxes within acceptable limits.

The second problem has a number of possible solutions, none of which are completely satisfactory. The principal difficulty is the radial temperature gradient between control rod coolant and fuel coolant in adjacent subassemblies that tends to develop when the control rods are withdrawn. For example, consider a central shim assembly. The total heat load for the assembly fully inserted is 2.7×10^6 Btu/hr (0.8 Mw). Coolant flow required to handle this heat at a ΔT of 300°F (nominal, excluding hot channel factors) is 3.0×10^4 lb/hr. When the assembly is fully withdrawn, the principal heat source for the control position is the heat conducted in from adjacent fuel subassemblies. If the coolant flow for the shim position is held constant, this heat conduction will raise the sodium temperature by only 60°F. The temperature increase in adjacent fuel subassemblies can be as high as 375°F, thus creating a radial gradient of 315°F or about 150°F across each shroud wall. The thermal stresses in the shroud walls with a 150°F ΔT are excessive. Furthermore, these stresses inevitably must be distributed unevenly around the shroud.

Some of the solutions considered are listed below and illustrated in the sketches in figure 48.

- (1) Use a large number of small control bundles or single rods dispersed throughout the core. In this case, the heat load at any single position is small enough that radial heat conduction through sodium and steel will remove the heat without excessive temperature gradients. Self-shielding is also reduced in this arrangement.
- (2) Use fueled control rod followers which generate approximately the same amount of heat as the poison material. This would reduce the swing in heat load as the rod is withdrawn.
- (3) Put one row of fuel rods in the shroud assembly and promote radial sodium mixing between the fuel region and the poison region. This amount of fuel would generate about three times the heat as the inserted poison assembly and would therefore be the controlling factor in determining sodium temperatures.
- (4) Develop a variable orifice scheme for matching sodium flow in control subassemblies to the heat load at any position of withdrawal.
- (5) Build a thermal barrier into the shroud assembly to spread out radial temperature differences and reduce temperature gradients in critical structural elements to an acceptable level.
- (6) Various combinations of the above.

The thermal barrier scheme (5) was adopted for the shim-regulating assemblies because it is the simplest and therefore most reliable. Scheme (1) is also simple in principle but it would increase the required number of rod drives by a factor of at least 3 and would complicate the fuel subassembly design. The fueled follower scheme (2) requires an increase in the overall length of all subassemblies and the fuel handling system, increases the mechanical complexity of the control subassembly and increases the chance of fuel failures causing jamming of control rods. Concern for such interaction between fuel and control also makes the advisability of using scheme (3) questionable. The variable orifice arrangement (4) is the most attractive alternate and is used for the safety assemblies, where its application is relatively simple, i.e., on-off control only. However, because of the close tolerances required for proportional control, it doesn't appear practical for shim rods unless a double plenum arrangement is used.

Figure 48
CONTROL ROD COOLING METHODS

Since the thermal barrier scheme will have a fixed orifice for flow control, its principal drawback is the cold sodium that will exit at the top of the core when the rod is fully withdrawn - the usual situation during operation. This will require a carefully designed (and tested) sodium mixing region above the core to equalize temperatures. This should not present too difficult a design problem because the control rod sodium flow is relatively small.

D. CONTROL SUBASSEMBLY MECHANICAL DESIGN

Operational control is accomplished through vertical movement of 13 shim-regulating rod bundles positioned in the core as shown in figure 43. The 13 shim-regulating bundles and drives are physically identical and may act both as a regulating and shim system. Normally, one to three of these assemblies would be in use at any one time for regulating. The nine safety rod bundles shown are used to provide an independent back-up safety system for scram and also normal reactor shutdown.

Shim bundles are capable of being moved through a 36-inch stroke; safeties have a 32-inch stroke. Electromagnetic scram devices are provided for both shim-regulating and safety bundles. Safety bundles are normally operated ganged together and are either fully withdrawn or fully inserted.

Control rod bundles are actuated by electromechanical drive systems mounted on the reactor vessel closure head, as shown in figure 2-2 of Vol. I. The control subassemblies, shown in figures 44 and 45 are held down within the core by gravity, with hydraulic forces being balanced at the lower end. The drive mechanism guide tube backs this up.

Each control rod bundle is free to move in response to its respective drive mechanism. A gripper mechanism is used to engage the control rod bundle to the drive mechanism main shaft. This mechanism is engaged at all times except during refueling phases, when refueling machine clearance is required.

The reference design control rod subassembly shown in figures 44 and 45 consists of a circular shroud similar in dimension to a fuel subassembly and an internally movable bundle assembly. General design configuration of the shim-regulating and safety subassemblies are the same, differing basically in material, number, and sizes of rods and cooling scheme (tantalum for shim-regulating control and E₄C for safety).

Each of the rods forming the bundle is fixed at both ends by means of a spider type end ring which in turn is attached to a central stainless steel support tube. A handling head and a guide shaft are provided respectively, at the upper and lower extremes of the support tube. Vertical and lateral structural loads on the control rod bundle are carried by this support tube.

Fixed orifice plates are used at the shroud inlet nozzle to achieve appropriate control rod cooling. Sodium enters through the inlet nozzle slots, passes through the orifice plates, flows through the rod bundle and out through the shroud upper end. Perforations in the upper portion of the shroud assembly are utilized to mix the cooler sodium of the control subassemblies with that of the adjacent fuel subassemblies. Control subassembly handling and removal is accomplished by grappling the internal shoulder of the shroud upper end. Since fuel subassembly handling is accomplished by external gripping of the larger shaped fuel handling head, chance of inadvertent pickup of a control rod is nil.

An alternate shim arrangement to permit variable orificing of coolant flow is shown in figure 49. This arrangement is a modification of scheme 4, figure 48. In this arrangement a portion of sodium leakage flow from the fuel subassembly grid plate seals is fed via an intermediate plenum to the control rod inlet. Movement of the lower control rod guide bearing occurs with rod position, thereby varying the open area of slots or holes in the inlet nozzle. The net effect will be to obtain relatively constant outlet sodium temperature from the control rod in all operating positions. This offers the advantage of reducing the shroud thermal stress level and also makes more efficient use of normally bypassed sodium leakage from the fuel subassemblies.

Figure 49
CONTROL SUBASSEMBLY
VARIABLE INLET ORIFICE
Scale: 1/8

REFERENCES

1. "Liquid Metal Fast Breeder Reactor Design Study", CEND-200 (Jan. 1964).
2. "1000 Mwe Liquid Metal Fast Breeder Reactor Follow-on Study: Task I Report, Preliminary Studies for a Reference Conceptual Design of a 1000 Mwe LMFER", CEND-322 (December 1967).
3. "AEC General Design Criteria, for Nuclear Power Plant Construction Permits", Appendix A, 10 CFR 50, July 11, 1967.
4. G. Schneider and D. Stoker, "Steam Cycle Influence on Fast Breeder Reactor Design", Nuclear Engineering and Design, 4, pp. 352 - 354, (1966).
5. R. H. Costello, M. M. Fiedler, J. J. Flis, R. H. Jones, T. E. Lempges, J. Matte, G. Vanmassenhove, "APDA Reactor Components Test", APDA-147.
6. P. Greebler, D. E. Sherer, N. H. Walter, "FORE - A Computational Program for the Analysis of Fast Reactor Excursions", GEAP-4090, October, 1962.
7. R. L. Ashley, et al, "Sodium Reactor Experiment Fuel Element Damage - Interim and Final Report", NAA-SR-4488 and Supplement, November 1959 and June 1961.
8. W. J. McCarthy, "A Review of the FERMI Reactor Fuel Damage Incident and a Preliminary Assessment of its Significance to the Design and Operation of Sodium Cooled Fast Reactors", V-2, International Conference on Fast Reactor Safety, Aix-en-Provence, France, September 19-22, 1967.
9. G. Chauvet, E. Toumier, P. Villain, "Reactimeter - Its Utilization", V-5, International Conference on Fast Reactor Safety, Aix-en-Provence, France, September 19-22, 1967.
10. D. A. Meneley, L. C. Kvitek, D. M. O'Shea, "MACH-1, A One-Dimensional Diffusion-Theory Package", ANL-7223, June 1966.
11. R. J. Fentz and T. A. Woldeck, "The Application of Fault Tree Analysis to Dynamic Systems", Boeing Company, System Safety Symposium, Seattle, Washington, June 8-9, 1965.
12. R. C. Noyes, "A Discussion of Methods for Theoretical Analysis of Coolant Ejection in FER's, Including a Description of the ETNA Code", RTi/TR(67)F, February 1967 (Comitato Nazionale Energia Nucleare, C.S.N. della Casaccia, Rome, Italy).
13. J. W. Stephenson, R. E. Nicholson, "Weak Explosion Program", ASTRA 417-6.0, February 1961.

14. G. Grass, "Measurements of the Superheating and Studies About Boiling Phenomena in Liquid Metals", International Conference on Fast Reactor Safety, Aix-en-Provence, France, September 1967, paper II E-4.
15. David Miller, "A Critical Review of the Properties of Materials at the High Temperatures and Pressures Significant for Fast Reactor Safety", ANL-7120, October 1965.
16. R. H. Erout, "Equation of State and Heat Content of Uranium", APDA-118, February 1957.
17. R. A. Meyer, E. Wolfe, N. F. Friedman, R. Seifert, "Fast Reactor Meltdown Accidents Using Bethe-Tait Analysis", GEAP-4809, January 1967.
18. C. J. Kubit, et al, "High-Temperature Thermodynamic Equation of State of Plutonium Carbide", Trans ANS, Vol. 8, No. 1, June 1965.
19. "Comment on AEC's Proposed General Design Criteria by the Atomic Industrial Forum", Atomic Energy Clearing House Review, Vol. 13, No. 41, pp 9-19, October 9, 1967.
20. "Nuclear Reactors and Earthquakes" USAEC Report TID-7024 (August 1963).
21. K. D. Lathrop, DTF-IV, A Fortran-IV Program for Solving the Multigroup Transport Equation with Anisotropic Scattering, Los Alamos Scientific Laboratory, LA-3373 (1965).
22. E. Elizard and L. Abbott, Reactor Handbook, Vol. III, Part E, SHIELDING, 2nd Ed. Interscience Publishers, 1962, p. 46.
23. E. L. Field, "Estimation of Self-Absorption Within a Homogeneous Cylindrical Source", Nucleonics, 11 (9), 66 (1953).
24. J. Belcher and L. Zoller, "OTT Laminated - Shield Configuration Analysis Method", DC 61-4-15, February 1960.
25. Revised Activation Handbook for Aircraft Engineers, General Dynamics Corporation, NARF-57-50T, 1957, p. 137.
26. Rockwell, Reactor Shielding Design Manual, D. Van Nostrand Co., Inc., 1956, p. 464.

APPENDIX A

Method for Detecting Partial Subassembly Plugging

This detection system is based on information provided by the two outlet thermocouples in each individual subassembly. This temperature information can be used to detect local coolant flow changes in two different ways. The first and most obvious determination is any increase in outlet temperature created by the decreased flow. If the thermocouple is not malfunctioning, this will indicate a change in subassembly flow and if this change is detected only locally, it further indicates partial plugging. A change in the outlet thermocouple reading might also be caused by a malfunctioning thermocouple. Back-up to this thermocouple is provided by the other thermocouple in each subassembly; thus, there is an excellent possibility of eliminating the malfunctioning thermocouple as the cause for the temperature change indication. However, one of the thermocouples could previously have failed or both could be malfunctioning. Therefore, in order to determine if the flow is actually decreased or if it is a thermocouple going bad, a second method of detection is desirable for confirmation that flow blockage has occurred. This involves programming a power oscillation into the reactor and comparing the thermocouple response of the suspected subassembly with the response of a similar normally operating subassembly. If the temperature response in the affected subassembly(s) indicates a significant time lag or change in amplitude, some blockage would be confirmed and protective action could be taken. If no lag or amplitude change is evident, thermocouple malfunction is probably the cause and reactor operation would be continued. The alternate possibility is that the thermocouples are malfunctioning because of a local voiding situation that is already in progress. At this point, this early detection is too late anyway to prevent damage to that subassembly and the protective system (as discussed in the complete local plugging section III-B-1-a-3) is designed to prevent failure propagation to the adjacent assemblies.

Calculations were performed to determine the minimum flow reduction that can confidently be recognized by the power oscillation technique. A power oscillation of $\pm 5\%$ over ~ 5 seconds was programmed into the reactor, operating at full power as shown in figure A-1. It appears that if an oscillation of this type is continued over several cycles, flow reductions of the order of 10 - 20% can be resolved. Thus, if the thermocouples are, in fact, indicating the occurrence of local partial plugging, this test can be used for varification.

Figure A - 1
TEMPERATURE RESPONSE TO A 5% POWER OSCILLATION
UNDER VARIED FLOW CONDITIONS

All of the temperature data from the subassembly coolant outlet temperature thermocouples will be continuously monitored and analyzed by a computer system. If any high readings are detected, the comparisons and tests will immediately be made to determine the cause. The power oscillation test is automatically programmed into the control rods by the computer. Comparisons to the backup and other adjacent thermocouples are also automatically performed by this computer. From this data, the operator must then decide if any corrective action is necessary or if it is simply a thermocouple failure. The computer and its data analysis is not considered an integral part of the protective system but is used only to obtain operating information and to increase the probability of detecting local plugging before it has a chance to damage the fuel subassembly. With proper core and subassembly design, the most realistic and probable type of plugging will be slow, which should allow plenty of time for detection. This detection system is tailored for that type of blockage. As already pointed out, instantaneous complete or partial blockage which immediately causes channel voiding, cannot be detected early enough to be prevented by any detection system. It is recognized that a small percentage of these outlet thermocouples may be inoperable during a year's refuelling cycle (they can be replaced during refueling). Therefore a few subassemblies may be unprotected by this system at any given time. There appears to be no economically feasible way of avoiding this problem. Therefore the basic philosophy is to provide highly reliable thermocouples which will insure one good measurement in a very high percentage of the subassemblies. This system is designed primarily to increase operating efficiency and to protect the plant owner's investment by reducing the risk of extended outages. It is not intended as part of the safety system required for the protection of the public; however, by reducing the probability of local fuel failures, it increases the overall safety of the system.

APPENDIX E

Comments on AEC's Proposed General Design Criteria
for Nuclear Power Plant Construction Permits
By the Atomic Industrial Forum

Published in the Atomic Energy Clearing House Review,
Vol. 13, No. 41, pp 9-19 (Oct. 9, 1967)

Note: Criteria omitted from discussion of Section III-C are indicated
by title only

1. Criterion 1 - Quality Standards (Category A)

Those systems and components of reactor facilities which are essential to the prevention, or the mitigation of the consequences, of nuclear accidents which could cause undue risk to the health and safety of the public shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes and standards pertaining to design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance criteria to be used shall be identified. An indication of the applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance criteria used is required. Where such items are not covered by applicable codes and standards, a showing of adequacy is required.

In the first sentence we have modified "accidents" with "nuclear" and substituted the phrase "cause undue risk to the health and safety of the public" to more precisely reflect what we believe was the AEC's intent. In the last sentence of the original draft, we have dropped the word "sufficiency" since we do not believe that it should be the responsibility of the applicant to document this unless the sufficiency of some specific item is in question. If for any reason the AEC questions the adequacy or sufficiency of a code or standard, it should take this matter up with the appropriate code drafting committee. Note that we have added a sentence requiring a showing of adequacy where there is no applicable code. The balance of the suggested changes are editorial in nature.

2. Criterion 2 - Performance Standards (Category A)

Those systems and components of reactor facilities which are essential to the prevention or to the mitigation of the consequences of nuclear accidents which could cause undue risk to the health and safety of the public shall be designed, fabricated, and erected to performance standards that will enable such systems and components to withstand, without undue risk to the health and safety of the public the forces that might reasonably

be imposed by the occurrence of an extraordinary natural phenomenon such as earthquake, tornado, flooding condition, wind or ice. The design bases so established shall reflect: (a) appropriate consideration of the most severe of these natural phenomena that have been officially recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

The changes in the first sentence are in line with those suggested for Criterion 1. We have deleted the word "additional" on the premise that it is not reasonable to ask the applicant to consider the simultaneous or cumulative forces of more than one extraordinary natural phenomenon.

3. Criterion 3 - Fire Protection (Category A)

A reactor facility shall be designed such that the probability of events such as fires and explosions and the potential consequences of such events will not result in undue risk to the health and safety of the public. Noncombustible and fire resistant materials shall be used whenever necessary to preclude such risk throughout the facility, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features.

These changes are consistent with the objective of assuring that there will be no undue risk to the health and safety of the public

4. Criterion 4 - Sharing of Systems (Category A)

5. Criterion 5 - Records Requirements (Category A)

6. Criterion 6 - Reactor Core Design (Categories A & E)

The reactor core with its related controls and protection systems shall be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits which have been stipulated and justified. The core and related auxiliary system designs shall provide this integrity under all expected conditions of normal operation with appropriate margins for uncertainties and for specified transient situations which can be anticipated.

We assume that "acceptable fuel damage limits" will be based on "undue risk to the health and safety of the public", not on economic grounds. The latter consideration is a matter for the licensee to decide. Further, these limits will depend on the circumstances leading to the damage. The example "transient situations" have been deleted since they may not be applicable in certain cases and they might also tend to prejudice design innovations.

7. Criterion 7 - Suppression of Power Oscillations (Category E)

The design of the reactor core with its related controls and protection systems shall ensure that power oscillations, the magnitude of which could cause damage in excess of acceptable fuel damage limits, are not possible or can be readily suppressed.

See comment on Criterion 6 with respect to "acceptable fuel damage limits".

8. Criterion 8 - Overall Power Coefficient (Category E)

We recommend deletion of this criterion since it is not applicable to certain reactor types. It is possible for the overall power coefficient resulting from a sum of components with different time constants to be positive without causing any serious safety problem. For example, in a sodium graphite reactor the coefficient has a prompt negative component together with a positive component with a long time constant. This results in an overall positive coefficient, but the negative part of the coefficient is large enough and fast enough to assure satisfactory control and safety. Safety problems relating to reactivity considerations are adequately covered in Criteria 6 and 7.

9. Criterion 9 - Reactor Coolant Pressure Boundary (Category A)

The reactor coolant pressure boundary shall be designed, fabricated and constructed so as to have an exceedingly low probability of gross rupture or significant uncontrolled leakage throughout its design lifetime.

It is important to characterize the leakage as "uncontrolled". Our only other suggested change is insertion of the word, "fabricated".

10. Criterion 10 - Reactor Containment (Category A)

Reactor containment shall be provided. The containment structure shall be designed (a) to sustain without undue risk to the health and safety of the public the initial effects of gross equipment failures, such as a large reactor coolant pipe break, without loss of required integrity and (b) together with other engineered safety features as may be necessary, to retain for as long as the situation requires the functional capability of the containment to the extent necessary to avoid undue risk to the health and safety of the public.

To avoid any ambiguity, "containment" should be characterized as "reactor containment". The statutory requirement of the licensee and the AEC is "to avoid undue risk to the health and safety of the public", not "to protect the public". It would be helpful to cross reference this criterion to Criterion 37 to indicate what the AEC means by "engineered safety features". Consistent with our comments on Criterion 37, we have substituted "pipe" for "boundary" on the premise that an applicant should not be required to consider a design basis accident more conservative than the instantaneous double-ended, circumferential rupture of a large coolant pipe.

11. Criterion 11 - Control Room (Category B)

The facility shall be provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit continuous occupancy of the control room under any credible post-accident condition or as an alternative, access to other areas of the facility as necessary to shut down and maintain safe control of the facility without excessive radiation exposures of personnel.

As originally drafted, this criterion could be interpreted as requiring a second control room. Not only would such a requirement be inconsistent with current practice, we believe that the complexities introduced could adversely affect overall plant safety. We believe it possible to design and equip a control room to assure continuous occupancy under all circumstances, including fire. We have deleted reference to 10 CFR 20 since the radiation exposure limits set forth therein apply to normal operating conditions, not accident conditions. Compliance with the radiation exposure limits of 10 CFR 20 under accident or post-accident circumstances is neither necessary nor reasonable. We have deleted the last sentence of the original draft since it is unnecessary and contradictory with the requirement of continuous occupancy of the control room.

12. Criterion 12 - Instrumentation and Control Systems (Category E)

Instrumentation and controls shall be provided as required to monitor and maintain within prescribed operating ranges those reactor facility operating variables essential to avoid undue risk to the health and safety of the public.

We have modified this criterion to more accurately and precisely reflect its intent.

13. Criterion 13 - Fission Process Monitors and Controls (Category E)

Means shall be provided for monitoring or otherwise measuring and maintaining control over the fission process throughout core life under all conditions that can reasonably be anticipated to cause variations in reactivity of the core.

We have dropped the two examples since they are measures of reactivity rather than the fission process.

14. Criterion 14 - Core Protection Systems (Category E)

Core protection systems, together with associated equipment, shall be designed to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.

We have deleted the phrase "act automatically" since manual action will prove adequate, indeed desirable, in some instances.

15. Criterion 15 - Engineered Safety Features Protection Systems (Category E)

No change suggested.

16. Criterion 16 - Monitoring Reactor Coolant Leakage (Category E)

Means shall be provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary.

We have assumed the intent of this criterion is to assure that leakage from the primary system will be detected, not that the entire reactor coolant pressure boundary will be monitored. The latter requirement would be inconsistent with current practice and unnecessary. Also, consistent with Criterion 9, we believe that the leakage should be characterized as significant and uncontrolled.

17. Criterion 17 - Monitoring Radioactivity Releases (Category E)

18. Criterion 18 - Monitoring Fuel and Waste Storage (Category E)

Monitoring and alarm instrumentation shall be provided for fuel and waste storage and associated handling areas for conditions that might result in loss of capability to remove decay heat and to detect excessive radiation levels.

We believe that the modified language as indicated above more accurately and precisely reflects the intent of the criterion.

19. Criterion 19 - Protection Systems Reliability (Category E)

Protection systems shall be designed for high functional reliability and in-service testability necessary to avoid undue risk to the health and safety of the public.

The suggested change is in line with our comment on Criterion 1.

20. Criterion 20 - Protection Systems Redundancy and Independence (Category E)

Redundancy and independence designed into protection systems shall be sufficient to assure that no single failure or removal from service of any component or channel of such a system will result in loss of the protection function. The redundancy provided shall include, as a minimum, two channels of protection for each protection function to be served.

The significant change we have made here is to delete the last sentence of the original draft. It would appear preferable to provide duplicates of the best system or component rather than going to an inferior system or component based on a different principle.

21. Criterion 21 - Single Failure Definition (Category E)

We recommend deletion of this criterion since it is more of a definition than a criterion and since the implied requirement is adequately covered by Criterion 23.

22. Criterion 22 - Separation of Protection and Control Instrumentation Systems (Category E)

This criterion should be deleted inasmuch as its requirements, to the extent they should be included in general criteria, appear to be adequately covered by Criterion 20.

23. Criterion 23 - Protection Against Multiple Disability for Protection Systems (Category E)

The effects of adverse conditions to which redundant channels or protection systems might be exposed in common, either under normal conditions or those of a nuclear accident, shall not result in loss of the protection function or are tolerable on some other basis.

The suggested change here includes modifying the word, "accident" with the word, "nuclear" and adding to the criterion the phrase, "or are tolerable on some other basis".

24. Criterion 24 - Emergency Power for Protection Systems (Category E)

We recommend deletion of this criterion since it would appear preferable to focus all requirements for emergency power in Criterion 39. Note that "protection systems" has been incorporated in Criterion 39 to accommodate this deletion.

25. Criterion 25 - Demonstration of Functional Operability of Protection Systems (Category E)

Means shall be included for suitable testing of the active components of protection systems while the reactor is in operation to determine if failure or loss of redundancy has occurred.

The reason for the changes here is that the licensee should be given some latitude in determining when and how such tests should be carried out. Further, he should be required only to test the active components of a protection system in contrast, for example, to a rupture diaphragm which could only be tested at the expense of destroying it. Also, certain tests might permit the licensee to determine if failure or loss of redundancy has occurred, but they might not permit him to demonstrate it.

26. Criterion 26 - Protection Systems Fail-Safe Design (Category E)

No change suggested.

27. Criterion 27 - Redundancy of Reactivity Control (Category A)

Two independent reactivity control systems, preferably of different principles, shall be provided.

The phrase, "At least" which prefaced the original criterion suggests a possible escalation of requirements which we do not believe was intended.

28. Criterion 28 - Reactivity Hot Shutdown Capability (Category A)

The reactivity control systems provided shall be capable of making and holding the core subcritical from any hot standby or hot operating condition.

Deletion of the preface phrase, "At least two of" is based on the comment made on Criterion 27. We have deleted the examples at the end of the original criterion since they could be interpreted to indicate a requirement for two fast reactivity shutdown mechanisms. This requirement is unnecessary when there is sufficient redundancy in one of the reactivity control systems to assure shutdown.

29. Criterion 29 - Reactivity Shutdown Capability (Category A)

One of the reactivity control systems provided shall be capable of making the core subcritical under any anticipated operating condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margin should assure subcriticality with the most reactive control rod fully withdrawn.

Deletion of the preface phrase, "At least", is consistent with the comments on Criteria 27 & 28. The other editorial changes are for purposes of clarification.

30. Criterion 30 - Reactivity Holddown Capability (Category E)

The reactivity control systems provided shall be capable of making the core subcritical under credible accident conditions with appropriate margins for contingencies and limiting any subsequent return to power such that there will be no undue risk to the health and safety of the public.

Deletion of the preface phrase, "At least one of", is consistent with the comments on Criteria 27, 28 & 29. Further, the public health and safety will not be compromised by a return to low power.

31. Criterion 31 - Reactivity Control Systems Malfunction (Category E)

The reactor protection systems shall be capable of protecting against any single malfunction of the reactivity control system, such as unplanned continuous withdrawal (not ejection or dropout) of a control rod, by limiting reactivity transients to avoid exceeding acceptable fuel damage limits.

We believe the criterion should preserve its original objective and at the same time acknowledge that one of the functions of the reactor protection system is to protect against certain control system malfunctions.

32. Criterion 32 - Maximum Reactivity Worth of Control Rods (Category A)

Limits, which include reasonable margin, shall be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to lose capability of cooling the core.

We believe substitution of "reasonable" for "considerable" and the substitution of "lose capability of cooling the core" for "impair the effectiveness of emergency core cooling" more precisely reflects the intent of the criterion. The re-wording also correctly implies that emergency core cooling will generally be required only if the reactor coolant pressure boundary is breached.

33. Criterion 33 - Reactor Coolant Pressure Boundary Capability (Category A)

The reactor coolant pressure boundary shall be capable of accommodating without rupture the static and dynamic loads imposed on any boundary component as a result of an inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

We have deleted the phrase, "and with only limited allowance for energy absorption through plastic deformation", on the premise that it is not helpful.

34. Criterion 34 - Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention (Category A)

The reactor coolant pressure boundary shall be designed and operated to reduce to an acceptable level the probability of rapidly propagating type failures. Consideration shall be given (a) to the provisions for

control over service temperature and irradiation effects which may require operational restrictions, (b) to the design and construction of the reactor pressure vessel in accordance with applicable codes, including those which establish requirements for absorption of energy within the elastic strain energy range and for the absorption of energy by plastic deformation and (c) to the design and construction of reactor coolant pressure boundary piping and equipment in accordance with applicable codes.

The detailed requirements contained in the original version are not appropriate for general criteria.

35. Criterion 35 - Reactor Coolant Pressure Boundary Brittle Fracture Prevention (Category A)

With the re-writing of Criterion 34 as indicated above, this criterion can and should be deleted.

36. Criterion 36 - Reactor Coolant Pressure Boundary Surveillance (Category A)

Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance of critical areas by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming with current applicable codes shall be provided.

It should not be necessary to inspect or maintain surveillance over all portions of the coolant pressure boundary; hence, we have inserted the phrase, "of critical areas". We believe that both the applicant and the AEC are in a better position to take advantage of developing technology and code refinement if these general design criteria refer to "current applicable codes" rather than to specifically designated codes.

37. Criterion 37 - Engineered Safety Features Basis for Design (Category A)

Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. Such engineered safety features shall be designed to cope with any size reactor coolant piping break up to and including the equivalent of a circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends.

Deletion of the phrase, "As a minimum", and substitution of "piping" for "pressure boundary" are both intended to eliminate the implication that the applicant should be required to consider a design accident basis more conservative than the instantaneous, double-ended, circumferential rupture of the largest pipe in the primary system. On this premise, retention of the original language introduces a vagueness which tends to defeat the objective of the criterion.

- 38. Criterion 38 - Reliability and Testability of Engineered Safety Features (Category A)
- 39. Criterion 39 - Emergency Power (Category A)
- 40. Criterion 40 - Missile Protection (Category A)
- 41. Criterion 41 - Engineered Safety Features Performance Capability (Category A)
- 42. Criterion 42 - Engineered Safety Features Components Capability (Category A)
- 43. Criterion 43 - Accident Aggravation Prevention (Category A)
- 44. Criterion 44 - Emergency Core Cooling System Capability (Category A)

An emergency core cooling system with the capability for accomplishing adequate emergency core cooling shall be provided. This core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to acceptable amounts for all sizes of breaks in the reactor coolant piping up to the equivalent of a double-ended rupture of the largest pipe. The performance of such emergency core cooling system shall be evaluated conservatively in each area of uncertainty.

In our opinion, one emergency core cooling system which incorporates a sufficient redundancy of active components and covers the full range of postulated breaks should be adequate. Our modification of this criterion reflects this consensus. For this reason, we have omitted the last sentence of the original criterion.

- 45. Criterion 45 - Inspection of Emergency Core Cooling System (Category A)
- 46. Criterion 46 - Testing of Emergency Core Cooling System Components (Category A)
- 47. Criterion 47 - Testing of Emergency Core Cooling System (Category A)
- 48. Criterion 48 - Testing of Operational Sequence of Emergency Core Cooling System (Category A)

49. Criterion 49 - Containment Design Basis (Category A)

The containment structure, including access openings and penetrations, and any necessary containment heat removal systems shall be designed so that the leakage of radioactive materials from the containment structure under conditions of pressure and temperature resulting from the largest credible energy release following a loss-of-coolant accident, including energy from metal-water or other chemical reactions that could occur as a consequence of failure of any single active component in the emergency core cooling system, will not result in undue risk to the health and safety of the public.

The objective of this criterion, in our opinion, should be that under the circumstances of an accident the integrity of the containment should be such as to prevent leakage of radioactive material which would result in an undue risk to the health and safety of the public. Since the maintenance of containment integrity is based on effective functioning of the emergency core cooling system, it appears unreasonable in this criterion to assume the complete failure of the emergency core cooling system; hence we have assumed a failure of a single active component. Consistent with this assumption, we believe that the pressure and temperature to be withstood should be characteristic of those anticipated from the largest credible energy release associated with a loss-of-coolant accident, including the calculated energy from metal-water and other chemical reactions. Acceptance of the "failure of a single active component" concept is consistent with Criterion 41.

50. Criterion 50 - NDT Requirement for Containment Material (Category A)

51. Criterion 51 - Reactor Coolant Pressure Boundary Outside Containment (Category A)

52. Criterion 52 - Containment Heat Removal Systems (Category A)

53. Criterion 53 - Containment Isolation Valves (Category A)

54. Criterion 54 - Initial Leakage Rate Testing of Containment (Category A)

55. Criterion 55 - Periodic Containment Leakage Rate Testing (Category A)

56. Criterion 56 - Provisions for Testing of Penetrations (Category A)

57. Criterion 57 - Provisions for Testing of Isolation Valves (Category A)

- 58. Criterion 58 - Inspection of Containment Pressure-Reducing Systems (Category A)
- 59. Criterion 59 - Testing of Containment Pressure-Reducing Systems Components (Category A)
- 60. Criterion 60 - Testing of Containment Spray Systems (Category A)
- 61. Criterion 61 - Testing of Operational Sequence of Containment Pressure-Reducing Systems (Category A)
- 62. Criterion 62 - Inspection of Air Cleanup Systems (Category A)
- 63. Criterion 63 - Testing of Air Cleanup Systems Components (Category A)
- 64. Criterion 64 - Testing of Air Cleanup Systems (Category A)
- 65. Criterion 65 - Testing of Operational Sequence of Air Cleanup Systems (Category A)
- 66. Criterion 66 - Prevention of Fuel Storage Criticality (Category E)

No change suggested.

- 67. Criterion 67 - Fuel and Waste Storage Decay Heat (Category E)

Reliable decay heat removal systems shall be designed to prevent damage to the fuel in storage facilities that could result in radioactivity release which would result in undue risk to the health and safety of the public.

We have substituted "which would result in undue risk to the health and safety of the public" for "to plant operating areas or the public environs" since we believe the first phrase more accurately describes the responsibility of the licensee.

- 68. Criterion 68 - Fuel and Waste Storage Radiation Shielding (Category E)

Adequate shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities.

The suggested change permits the criterion to accommodate radiation limits as may be specified which may differ from those set forth in 10 CFR 20.

- 69. Criterion 69 - Protection Against Radioactivity Release from Spent Fuel and Waste Storage (Category E)

Provisions shall be made in the design of fuel and waste storage facilities such that no undue risk to the health and safety of the public could result from an accidental release of radioactivity.

We have avoided the use of the word, "containment" because of its possible ambiguous connotation. The licensee may rely on some means other than containment to meet the requirements of the criterion. The other suggested changes are consistent with our comments on Criterion 67.

70. Criterion 70 - Control of Releases of Radioactivity to the Environment
(Category E)