NUREG/CR-4019

A New Steam Cooled Reactor

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Prepared for
U.S. Nuclear Regulatory Commission

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NUREG/CR-4019
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Manuscript Completed: August 1984
Date Published: November 1984

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NRC G-04-84-007

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BY

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I. INTRODUCTION

As of February 1984, most of the nuclear power community has come to the realization that drastic changes will be required if the industry is to survive. The cancellations of the Zimmer plant, the Marble Hill plant and the denial of a license for Byron etc. have finally convinced the community that "business as usual" is simply not going to work. Prior to these events the prevailing wisdom was that all that was required to revive the nuclear industry was to continue patiently along the path of making small evolutionary safety improvements and operating without major incident. And in time with increased electrical demand, higher oil prices, and coal generated environmental problems, the financial, institutional, and public restraints would be reduced sufficiently to allow the industry to prosper again.

The new spirit of changing attitudes was typified by the Congressional hearing testimony of C. F. Jones, a prominent member of the nuclear establishment, before the Subcommittee on Energy Research and Production, Feb. 7, 1984. Mr. Jones' thesis principally was that the utility industry was living in the past and was simply not set up to live in a regulatory atmosphere. His solutions called for a basic

* The purpose of this report is to document the work accomplished by the authors on the steam cooled reactor from 1982 to 1984. The authors are grateful to the Nuclear Regulatory Commission for their assistance.

revision of the way the electric industry operated requiring new designs of nuclear plants and new methods of procurement and manufacturing.

Even these changes, however, will not be enough. At a conference (1) aimed at defining what was wrong with nuclear power and how one would go about fixing the industry's problems, the following statements were made:

> "The major safety issues--waste disposal and reactor safety--have not been fully demonstrated. Second, until they are, it is unlikely that the industry will gain public acceptability. The perception of uncertainty about the safety of nuclear power is the single most important problem in public acceptability."

Changes in public perception are most difficult to achieve particularly in the face of continual adverse media coverage for every minor incident occurring in the industry.

It thus appears that a good case can now be made for starting over and coming forth with a new nuclear power plant that can be proven safe, thus allaying public fears and perceptions. Such a new plant will take a long time to develop, construct and prove out, but as there are no new orders for nuclear plants in the U.S., or are there likely to be any in the next several years, there is time to start thinking about 21st century plants.

II. DESIRABLE CHARACTERISTICS OF A NEW NUCLEAR PLANT

If one could start over in the nuclear industry, knowing what we know now after 30 years of experience, what would be some of the desirable characteristics and criteria that would be built into the new plant? In a Jan. 1984 presentation, J. D. White of ORNL listed 26 plant features to be studied to arrive at the desired criteria for "Revitalization of the Nuclear Power Option in the U.S."

Here we list 11 semi-obvious characteristics that appear to be achievable at least in a new steam cooled reactor plant.

1. <u>The plant should be ultra safe, having a walk away from capability and being able to demonstrate proof of this safety feature.</u>

Such a plant could have all of its power supplies and coolant supplies cut off and still be able to sit in place indefinitely without any meltdown or release of fission products. A further desirable corollary of this principle is that the plant should possess this characteristic regardless of plant size or rating.

Currently there appears to be a talking trend toward smaller plants. The basic argument is that the industry originally pushed the art too far and too fast in escalating from 100 MWE plants to 1000 MWE plants in just 2-3 years. The argument goes that we should have taken the "bugs" out of the plants at possibly a 200-300 MWE level rather than having larger problems on larger plants. And with smaller forecast demand growth rates it appears imprudent for many utilities (domestic and foreign) to add capacity in too large blocks. Furthermore, it has been known for a long time that the walk-away-from characteristic could easily be achieved by reducing the power density of a plant. Consequently, a smaller rated plant with essentially a fixed mechanical structure, fulfills this desirable safety characteristic.

Before joining the trend toward smaller plants, however, there are at least two arguments in favor of developing 500-2500 MWE plants with the walk-away feature. The first argument is the original one of economy of scale. The cost of a plant is not directly proportional to its power rating, but rather varies approximately as the square root of the rating. Hence a 1000 MWE plant would be expected to cost roughly 40% more than a 500 MWE one.

Secondly, if there is to be any nuclear power future, it must provide a reasonable fraction of the electrical demand. As we are looking toward a long term future, if 50% of our electricity were to be supplied by nuclear power by the year 2030 and we predicted only a modest growth rate in demand of 3% year, then 14685 reactors of 100 MWE rating might be required for replacement and to supply the added load. The concept of a regulatory agency monitoring 14685 reactors is somewhat staggering. (Fortunately with a walk-away-from plant, one would anticipate that the current core meltdown probabilility of one chance in ten thousand per reactor year would not apply.)

2. <u>The Sample Plant Should Accommodate Either a Burner or a Breeder Reactor.</u>

In retrospect and with 20-20 hindsight, it now appears silly to have two kinds of basic plants in the U.S. economy. That is, why do we need a light water thermal reactor plant for burning uranium and a sodium cooled fast reactor plant for breeding? The situation simply grew historically and evolved from state-of-the-art knowledge at the time decisions had to be made. One crucial decision was made by the then Capt. Rickover when he selected the STR (Submarine Thermal Reactor) over the SIR (Submarine Intermediate Reactor) for further development with AEC and Navy funds. Rickover's decision was made on maintenance considerations as the radioactive sodium primary system of the SIR would be much more difficult to service than the "cooler" water primary system of the STR. (And besides he was having sodium heat exchanger problems at the time).

The breeder reactor suffers from the stigma that it is perceived by the public as being less safe than the burner. But if criteria (1) is upheld breeders and burners alike would be ultra safe and hopefully the stigma would be removed. The present day argument is that breeders are not necessary in that we have plenty of uranium fuel. However, if both burners and breeders could be made equally safe, and the breeder cost approximately the same amount or less, using an identical plant, the burner appears to be unnecessary. However, even if a once through cycle is used at first, the plutonium still exists in the spent fuel. Ultimately the natural sources of uranium would be used up and it would then become economic to reprocess the fuel elements for the plutonium. And the best way of disposing of plutonium with its long half-life is not to bury it in the ground but to burn it up and convert the residue into shorter half-life products.

So with public acceptance the above argument suggests that in the long term we would need only breeders. However, over the short and intermediate terms we will probably need burners to provide the correct ratio of plutonium isotopes to best feed an overall fuel management program. Thus criterion (2) insists that both types of operation occur in the same ultra safe plant.

3. The same plant should use either enriched uranium as fuel or plutonium and breed with selected isotopes of both materials.

As indicated above there appears to be no reason at present for the U.S. to reprocess its spent fuel to extract its plutonium isotopes, particularly in view of possible added cost and possible complications with the non-proliferation treaty. Nevertheless, the spent fuel elements exist as does their plutonium content. New fuel elements could be manufactured using U-235, U-238 mixtures, or U-238 and Pu-239, Pu-240, Pu-241, Pu-242 mixtures in the ratios normally obtained by burning the fuel elements over a reasonable lifetime. The U.S. Has a large inventory of U-238. (The statement has been made that we have more energy stored in cannisters at Oak Ridge than the Saudis have in their oil reserves.) It seems reasonable to assume that ultimately we will be into a mixed oxide economy. The intent of criteria (3) is to insure that regardless of the fuel isotopes used, a high conversion ratio, or breeder reactor can be operated in the new plant.

4. Fuel Lifetime Should be Long, Possibly 3-5 Years Without Opening the Vessel.

If the reactor has a high conversion ratio, or is a breeder, the reactivity lifetime of the fuel should be very long. The fuel lifetime then becomes a metallurgical lifetime and will require that fuel elements be improved to stand up for say 200,000 MWD/tonne. Test specimens have been made to last that long in test loops, and EBR, FFTR and Phoenix fuel has been show to be capable of lasting metallurgically for greater than 100,000 MWD/tonne. So it is likely that some fuel element development will be required if the fuel is to be balanced to have its metallurgical lifetime match the reactivity lifetime.

There are two obvious reasons why one desires this long fuel lifetime. First, the cost of fuel goes down the longer it is burned at a fixed power output. It is necessary in some reactors, such as the steam cooled reactor, to offset the cost of added enrichment or plutonium extraction, against longer life.

Secondly, the largest contributor to non-availability is the lost time at refueling. And if refueling can occur once every 3 to 5 years instead of once a year, then lifetime plant availability can be substantially increased.

5. Plant Technology Should be Water Based.

It is tempting to regard the grass on the other side of the fence as greener particularly when one is in trouble. White hopes such as pebble bed reactors are currently regarded very prominently (2). The truth of the matter is that most of the U.S. experience is water based. We have 4 major vendors, several architect engineers, dozens of nuclear engineering departments at universities, and thousands of individuals who have expertise in the design, construction, operation or regulation of water based plants. It seems logical to call upon this experience for the new reactor of the future. However, this statement implies that a decision on the new reactor must be made soon. If the new enterprise is not started soon, many of these people will be dispersed, and the oldest experienced personnel who have been in the business since its inception will have died out.

6. The Plant Should Use a Thermodynamic Cycle That Produces Reasonable Efficiencies at Low Pressures. (33% at 800 psi) 35% at 1000 psi).

The quest in engineering has always been toward higher efficiencies. Higher efficiency means lower operating cost and the economic incentives are very great in a large power plant. This quest has led to higher temperatures and inevitably metallurgical limits have been reached. Early coal fired steam plants attempted to push operating temperatures to 1200-1300 degrees F. Materials problems arose and plants with high availability are rare that have operating temperatures higher than 900 degrees F.

In a nuclear plant the safety issue becomes pervasive. The higher the pressure, the higher the temperature, and in water plants the greater the stored energy, with the potential of more damage in the event of a system break. It seems reasonable to attempt to equalize operating costs by swapping efficiency against lower steel costs in a low pressure plant. At the moment, capital costs of nuclear plants are extremely high, whereas operating costs are lower than comparable coal-fired plants. The selection of a low pressure plant would reduce the steel requirements of the plant considerably, but would probably increase somewhat the turbine-generator costs. In accounting terms one can amortize the capital costs over the plant lifetime and translate them into operating costs. It is not at all clear whether the overall operating costs of a low pressure plant would be much lower than a similar high pressure plant, but the increased safety seems obvious.

7. The Plant Should be Capable of Being Inherently Designed for Sabotage Resistance.

Up until fairly recently the method of protecting a nuclear plant against sabotage was to build a normal plant and surround it by a small private army. Any special built-in security features, such as barbed wire fences, anti-intrusion electornics, etc. had little or no connection with the nuclear portion of the plant.

The PIUS reactor design pioneers a new approach to the problem. The pressure vessel head was designed to withstand a direct 1000 lb. bomb hit. After the vessel head was to be put in place, the only crane capable of removing it was to be disassembled and stored. Three days would be required to reassemble the crane, etc. It is clear many other anti-sabotage features could be designed into the main portions of future plants ahead of time if it were deemed desirable. Unfortunately, acts of terrorism seem to be on the rise, and it appears prudent in considering design features to select the ones most sabotage resistant.

It turns out that the walk-away-from feature of the new plants also serve an anti-sabotage function in that the threat of sabotage is in many instances as effective as the sabotage itself. One scenario calls for a group of terrorists taking over a nuclear plant and threatening to blow it up if their compatriots are not released from jail. In a walk-away-from design the operator simply throws the power switch to off as he leaves the control room. The reactor plant can now sit unharmed in place while negotiations with the saboteurs ensue.

8. The Plant Should Be Easy to Maintain.

It is now recognized that maintainability is a most important characteristic that should be considered in the earliest concept and design phases. Steam generators must be considered as leaky and provisions made to eliminate them or provide easy means for fixing leaks. Maintenance of items inside the pressure vessel is difficult often requiring special tools and large crews because of the factor of people burnout. Placement of key components in shielded compartments that allow sufficient maintenance access are also considered in recent designs. And finally some reactor concepts are inherently easier to maintain than others.

9. The Plant Should Have High Availability.

Criteria (4) and (8) partially address this point. The plant must have a long cycle time and ease of maintenance. The second largest contributor to hign availability is meticulous scheduling of operations, down time and repairs. Essentially this step requires a large computer and a conscientious planning staff capable of analyzing the effect on availability of preventive and corrective maintenance. Finally, availability can be improved by a reexamination of the safety system limits in conjunction with a new type of digital control system. The control system now can be extremely fast and would catch most transients before they can reach the safety system settings. In this way the plant would hardly ever scram (most probably on the failure of safety system components) and again availability should be increased.

10. The Plant Should Be Economic.

Again, this criterion is obvious and is partially contained in criterion (6). The only point to be made is that there may be a trade-off between economics and safety or perceived safety without getting into the argument of how safe is safe enough. Clearly, one would be willing to pay somewhat more for a plant that could be perceived as being safe by the general public. In the end such a plant might prove to be more economic in that less time delays involved with public concerns and interventions might ensue. However, as a general statement, the new nuclear plant must be competitive or lose out to other forms of electrical generation.

Finally plant costs vary roughly with the amount of steel used. A low pressure plant should use less steel, and hence has a good shot at being more economic than higher pressure plants.

11. <u>The Plant Should Have a Built-in Clean-up System and All Paths of Radioactivity Release Should Be Through This Clean-up System</u>.

At Three Mile Island, soon after the accident, radioactive water found its way from the pressurizer overflow tank to the containment building sump. The sump pump automatically began pumping this liquid into the auxiliary building thus breaking the confinement of the vapor container and permitting radioactivity escape routes not contemplated by the plant designers. It now appears possible to think out pipe breaks and valve openings in advance and devise a system that will enforce cleanup of gas and water before any releases occur. The setup is particularly attractive to a reactor that can't melt down and that operates at low pressure.

III. THE REACTOR

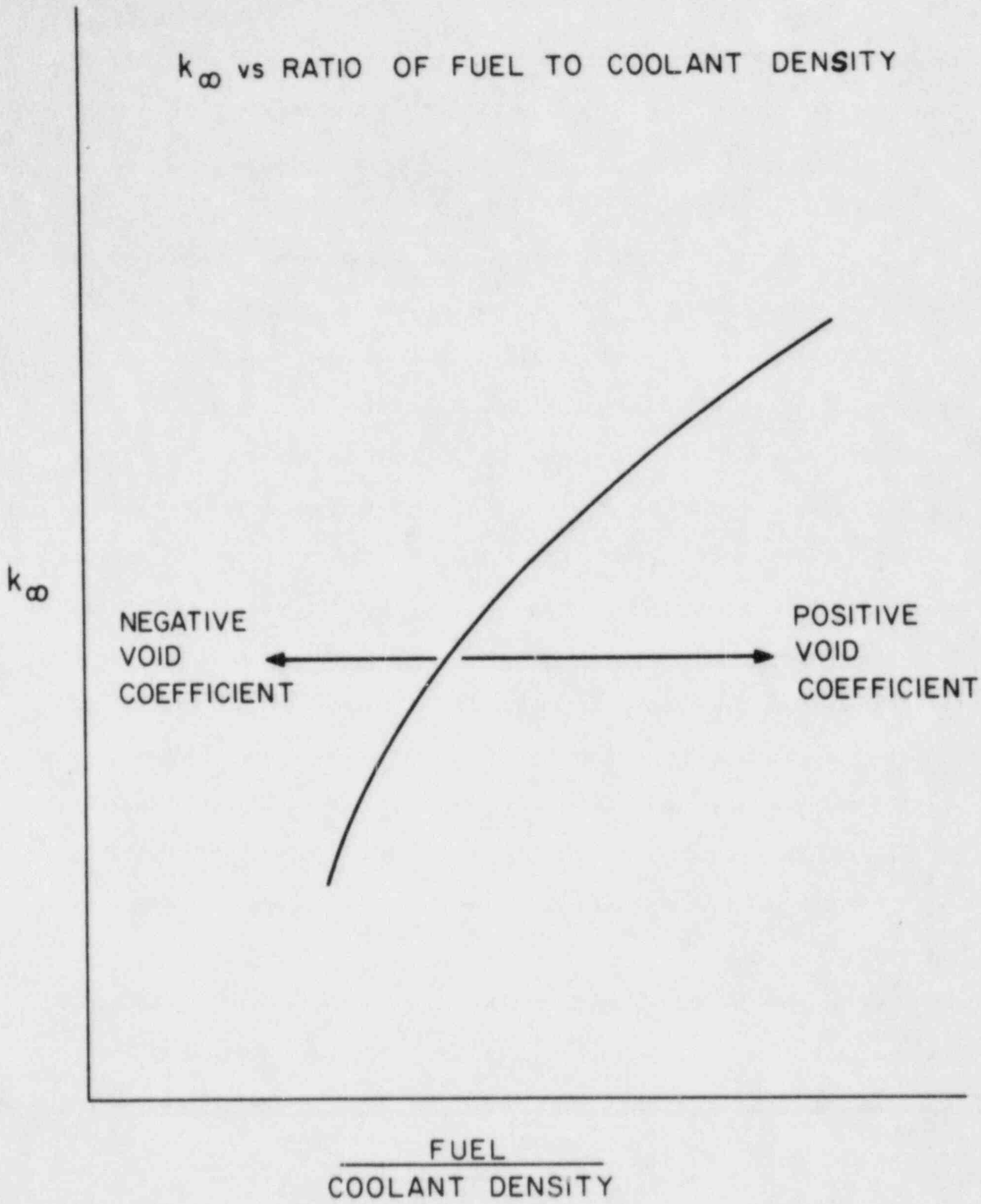
1. General

For the 21st century nuclear plant we propose a new steam-cooled reactor that fulfills the 11 criteria just described. The steam-cooled reactor had its origin in the 1950s when American Gas and Electric Company provided development funds to Babcock and Wilcox to work on an advanced reactor. The concept at the time called for a steam-cooled breeder, using mixed oxides of uranium and plutonium as fuel with a mixture of approximately 85 percent uranium isotopes and 15 percent plutonium isotopes. The original designs were recently updated with the realization that a curve similar to Fig. 1 existed and could be used to shape reactor characteristics.

The curve indicates that there are regions in which a negative void coefficient can be obtained. The curve has a similar shape for water cooled or steam reactors. This realization removed one of the principal objections to early steam-cooled reactor designs, namely the fact that the reactivity increased as the core voided.

Thus, a steam-cooled reactor can be designed such that if one attempts to draw more steam (or more power) from the reactor it will tend to shut itself down and protect itself. For low coolant densities, reactor geometry is chosen such that fast neutron leakage will occur and tend to reduce the neutron multiplication. Thus, the positive void coefficient of previous steam-cooled reactors would be avoided geometrically by increasing leakage.

An additional feature of the new concept is that the reactor could be designed using either conventional fuel elements or new more rugged elements. In either case, the fuel elements or the core structure would contain thermal neutron absorbers like borinated stainless steel or tungsten such that as the steam density increased or the core was flooded the resulting higher density thermalizes the neutron spectrum and the reactivity is reduced by the absorption of neutrons by the thermal poisons. Early versions of fast steam-cooled reactors suffered from the fact that when the reactor was flooded with water, it wanted to



k_∞ vs RATIO OF FUEL TO COOLANT DENSITY
k_∞
NEGATIVE VOID COEFFICIENT
POSITIVE VOID COEFFICIENT
FUEL / COOLANT DENSITY

FIGURE 1

go critical as a thermal reactor. This problem is now avoided by the judicious introduction of thermal neutron and resonance absorbers.

2. Uranium Fueled Reactors

The steam cooled reactor can be used either as a converter or a breeder. When fueled with U-235 and U-238, it operates as a high efficiency converter. When fueled by Pu isotopes and U-238 it is capable of breeding. As a result of the design features a reactivity vs. density curve of the shape of Fig. 2 can be obtained for a uranium fueled reactor. Here the density is allowed to vary from roughly the density of air to the density of water. A peak in reactivity results somewhere within this range. In the case of Fig. 2, this peak occurs at 1000 psia corresponding to a steam density of 0.0314 grams/cc. Fig. 2 is for a uranium fueled reactor having an enrichment in U-235 of 10.17%.

This reactor has a steady state conversion ratio of 0.85 and as a result requires only 0.021 in excess reactivity in its initial loading to achieve its theoretical lifetime. With this loading and conversion ratio, calculations have indicated a reactivity lifetime of greater than 125,000 MWD/tonne.

The mass of U-235 required for this reactor is 7.74×10^3 Kg. The reactor fuel elements are loaded to 113 grams of uranium dioxide/ft and there are 600,000 ft of rods employed. The total amount of uranium in the core is 67,800 kg.

a. Lifetime

With these constants and a reactivity lifetime of 125,000

REACTIVITY vs DENSITY

URANIUM FUELED REACTOR

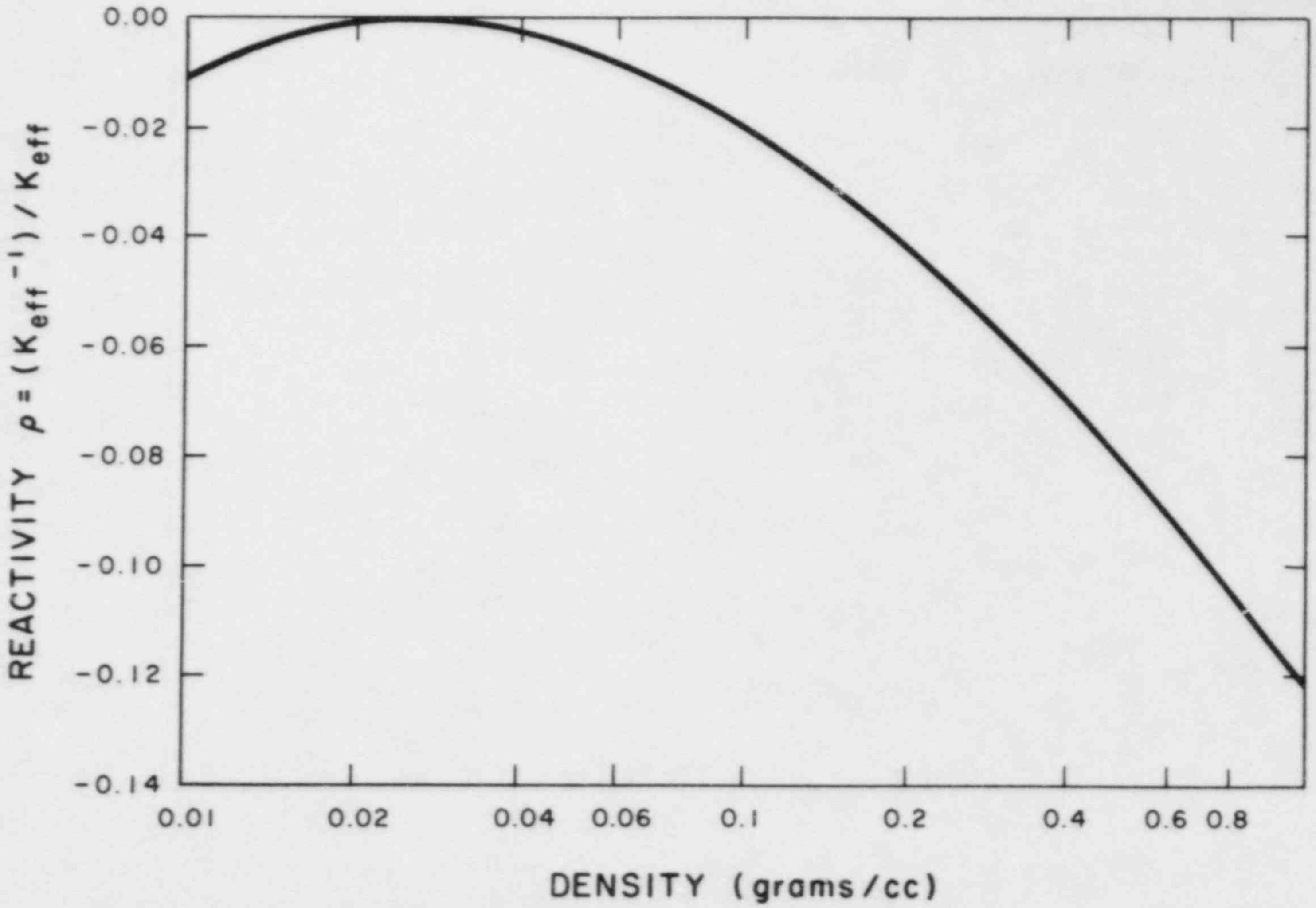


Figure 2

MWD/tonne, one can theoretically expect a reactivity core life of approximately 9 years. As previously indicated metallurgical lifetimes in the neighborhood of 100,000 MWD/tonne have been achieved in EBR and now in FFTF, so a metallurgical lifetime for these cores might be expected to be 7.2 years. Uranium fueled cores have been calculated for steam cooled reactors with reactivity lifetimes in excess of 200,000 MWD/tonne, so with improved metallurgical development, it is quite conceivable that the vessel head might not have to be removed more often than about once every 5 years.

b. Reactivity Worths

The doppler coefficient for this reactor is approximately -4.3 x 10^6 $\Delta k/F^o$. In going from an initial cold shutdown to power, the change in reactivity is only 0.003. Actually, normal shutdown temperature of the reactor will be near the saturation temperature of the water (for our prototype case 544 F^o at 1000 Psi). Consequently the doppler reactivity change from normal shutdown to full power would be quite small.

The excess reactivity requirement for lifetime thus would dominate the design. The required shutdown reactivity of 0.021 + 0.003 = 0.024 should have some added margin of safety. (The W. Zinn edict after the SL-1 accident would add another 2%.) For this reactor an added 1% is probably sufficient, making the total rod worth say between 0.034 and 0.044. This amount of worth can be obtained by using less than 2% of the core volume for control rods in a suitable geometry. It would be anticipated that boron carbide rods or equivalent would be used and either natural or enriched B-10 could be employed depending on metallurgical considerations.

3. Plutonium Reactors

The shape of the reactivity curves can be modified at will. Fig. 3 indicates the curves of certain mixed oxide fueled reactors. Here five different cases are shown having the dimensions and characteristics given in Table 1.

Table 1

Dimensions and Characteristics of Five Reactor Design Cases

Cases	Pitch Between Rods	Fuel Rod Outer Radius	Fuel Rod Inner Radius	Core Height	Core Diameter	Boron-10 Concentration
	(cm)	(cm)	(cm)	(cm)	(cm)	(wt% clad or equiv.)
I	1.350	0.500	0.450	220	400	1.16
II	1.350	0.500	0.450	220	400	1.39
III	1.350	0.500	0.450	151	400	1.39
IV	1.350	0.500	0.450	244	400	1.39
V	3.175	0.9525	0.800	220	487	0.79

Cases I, II, and V operate with the reactivity peaking at the same coolant density of 0.0314 gms per cubic centimeter which corresponds to a saturated steam pressure of 6.89 MPa (1000 psia). Case III peaks at a higher pressure and Case IV at a lower one. The reactor can be made to

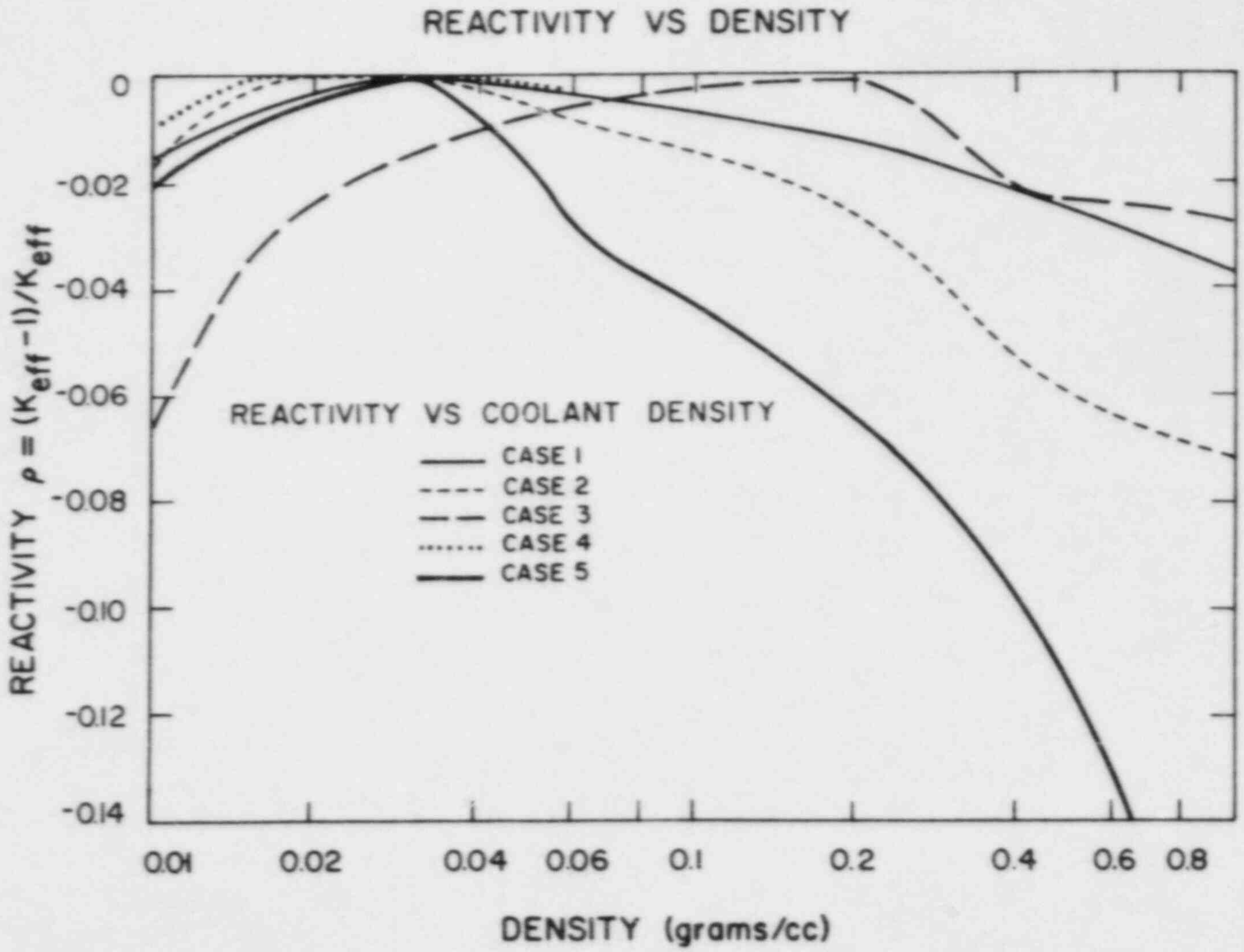

REACTIVITY VS DENSITY
REACTIVITY $\rho = (K_{eff} - 1)/K_{eff}$
0
-0.02
-0.04
-0.06
-0.08
-0.10
-0.12
-0.14
REACTIVITY VS COOLANT DENSITY
CASE 1
CASE 2
CASE 3
CASE 4
CASE 5
0.01
0.02
0.04
0.06
0.1
0.2
0.4
0.6
0.8
DENSITY (grams/cc)

FIGURE 3

have its peak in reactivity anywhere from 3.45 to 22.1 MPa (500 to 3200 psia) steam.

Similarly each case of Fig. 3 has a different breeding ratio, ranging from about 1.15 and a compound doubling time of 39 years, down to a breeding ratio of 1.06 and a compound doubling time of 165 years.

The heavy curve of Fig. 3 is for a reactor that employs large diameter fuel rods and exhibits large swings in reactivity as a function of density and will be used as an example. When flooded with pure water, the shutdown reactivity reaches 18 percent.

The reactor physics design of Case V is a "hold-your-own breeder" with the breeding ratio being only 1.06. The mass of fissile plutonium is rather large, about 6.40 metric tons, and the doubling time is very long. Of the heavy metal in the core, 8.9 percent is plutonium and 91.1 percent is uranium. The uranium used is depleted uranium.

The core lattice is quite open; lattice rods are 1.875 cm (0.75 inches) in diameter with 0.1588 cm (1/16 inch) stainless steel cladding and have an open spacing of 1.27 cm (0.50 inches) on a triangular lattice. The core is 4.87 m (16 feet) in diameter and 2.44 m (8 feet) high. Other designs have used conventional LWR fuel rods with the geometry adjusted to provide the required leakage.

Lattice parameters were calculated using the VIM code, which is a continuous energy Monte-Carlo code that originated at Atomics International and was later improved at Argonne. It can compute all reactor constants given enough time, and is one of the best codes available for reactor design. It uses the cross-sections of Brookhaven

END/B-IV from the Nuclear Cross Section Center at Brookhaven. VIM cross-section libraries for all of the heavy isotopes are also available at 1000 degrees K. Thus, there is reasonable assurance that if such a core were built, its performance would closely approximate the computer design. Appendix A describes the usage of VIM in some detail and Appendix B indicates the operation of RFD-2 its companion code.

The example core has the following characteristics:

(1) The shape of reactivity curve is given in Fig. 3, Case V. It will be noted that a very high density coefficient exists, particularly in the dry coolant area of the curve. This negative coefficient is in the order of 0.05 percent change in reactivity per percent change in density. This high negative coefficient should be very effective in squelching transients. The coefficient will come into operation in less than 1 second.

(2) The fraction of plutonium isotopes used is given in Table II.

Table II

Fraction of Pu Isotopes

	Fraction
Pu-239	0.58
Pu-240	0.24
Pu-241	0.13
Pu-242	0.05

It will be noted that this mixture of plutonium isotopes is approximately the isotopic distribution obtained by operating an LWR over its normal life cycle of roughly 40,000 MWD/tonne.

(3) The ratio number densities of B-10 to stainless steel is 0.0079.

(4) The results of typical VIM calculations are shown in Table III from which k_{eff} and ρ can be calculated.

Table III

Typical Results from VIM Calculations

Run I: Density of water = 31 kg/m^3

Run II: Density of water = VOID

Run	Group	E_{lower}	D	$\sum a$	$\sum 1 \to 0+1$	$\sum f$
I	1	0.8298 Mev	3.489	0.00329	0.01788	0.008233
I	2	5.5308 kev	2.094	0.00252	0.00439	0.002648
I	3	1.0×10^{-5}ev	1.451	0.01699	------	0.01317
II	1	0.8208 Mev	3.793	0.00322	0.01696	0.007890
II	2	5.5308 kev	2.578	0.00269	0.00113	0.002501
II	3	1.0×10^{-5}ev	1.686	0.000949	------	0.006101

(5) The fuel constants, breeding ratios and doubling times are given in Table IV.

Table IV

Masses, Breeding Ratio, and Doubling Times of Example Reactor

Total Fuel Volume	=	10.175×10^6 cm^3
Loading of Heavy Metal	=	71.6 Metric Tons (MT)
Loading of Fissile Plutonium	=	6.40 MT
Breeding Ratio	=	1.06
Excess Plutonium (Fissile Pu/year)	=	0.0351/Year
Compound Doubling Time	=	165 Years

Table V

Comparison of Steam at 1000 psi and Sodium as a Fast Breeder Coolant

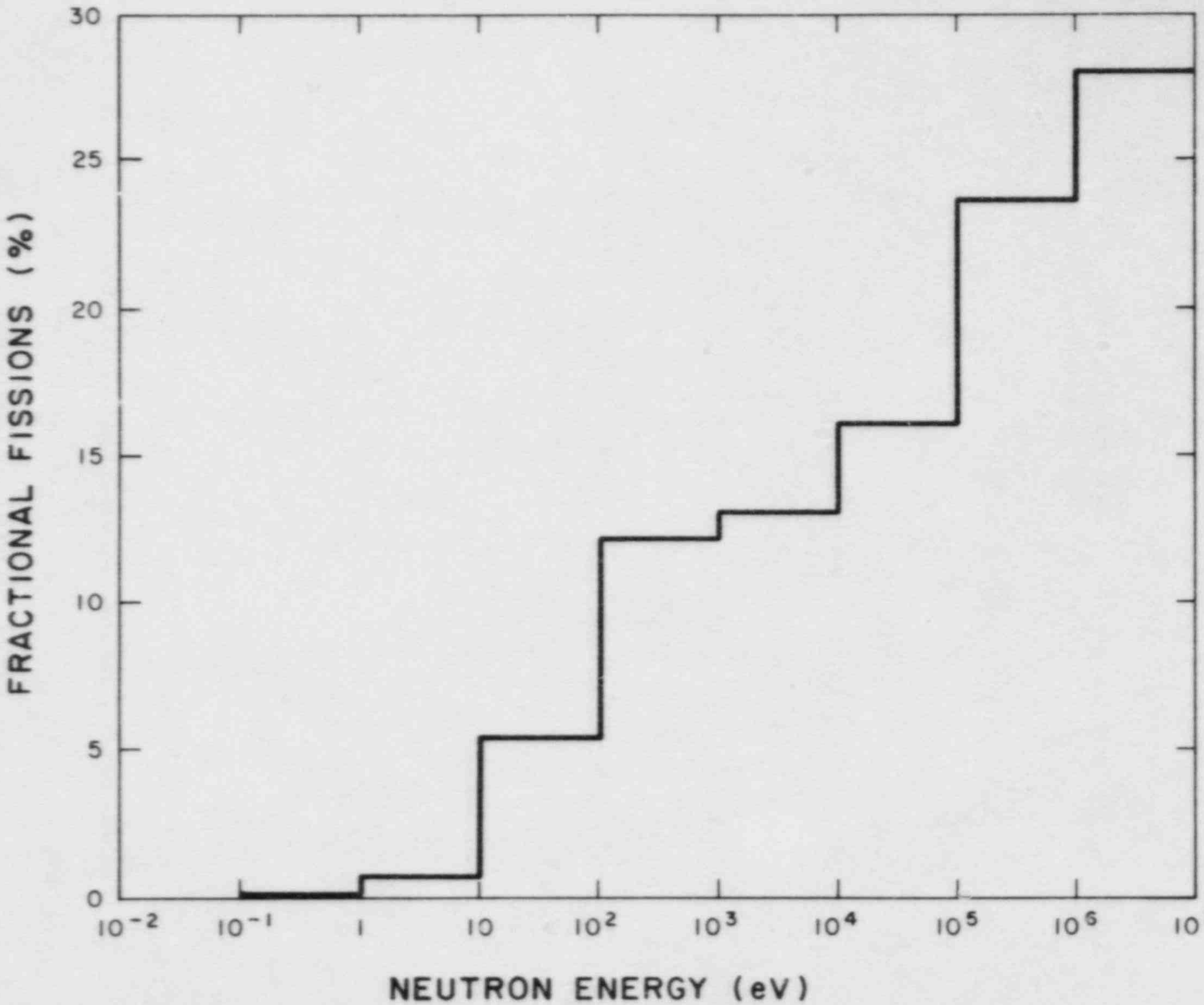
		STEAM COOLED FAST BREEDER	SODIUM COOLED FAST BREEDER
	η	2.147	2.283
	ε	0.399	0.356
	$\eta+\varepsilon-1$	1.546	1.639
NEUTRON LOSSES			
Structure		0.074	0.158
Coolant		0.003	0.010
B-10		0.014	---
Fission Products		0.075	0.055
Leakage		0.046	0.046
Pm-241 Decay		0.031	0.031
TOTAL LOSSES		0.243	0.308
NET NEUTRONS FOR BREEDING		1.303	1.331

Table IV indicates the breeding ratio of 1.06 for the example reactor Case 5 of Fig. 3). The other reactors of Fig. 3 show varying breeding ratios up to 1.15. Figure 3 was presented to indicate the various shapes of reactivity curves that could be obtained by initial design and made no attempt to optimize the breeding ratio. Table V indicates a comparison of an optimized steam-cooled breeder as compared with an early version of the sodium cooled Clinch River Breeder Reactor.

It will be noted that the bottom line of net neutrons available for breeding is essentially the same for the two reactors. In other words, if the objective of the plant were to breed, then the steam-cooled breeder as described would breed about as well as a sodium-cooled breeder.

The question also arises as to the shift in the peak of the reactivity curve as a function of lifetime, or fission product generation. Our calculations to date indicate that leakage effects dominate the situation, and that only a small peak shift would be encountered over the core lifetime. As will be indicated later, the control system will set the operating point at the peak even though this peak may shift a few psi over the lifetime.

(6) Figure 3a indicates the fractional fissions as a function of neutron energy for a typical reactor of this design. It will be observed that this is a modestly fast reactor with substantial fissions occurring in the intermediate range.



ENERGY SPECTRUM

FIGURE 3a

There are negligible fissions at thermal energy and there should be no problem with xenon poisoning.

(7) The thermal constants used in the example calculations are as follows:

a. Thermal Power Output = 1800 MW

b. Core Surface Area = 3,882 m^2

c. Heat Flux = 4636 kw/m^2

(8) Fuel lifetime is metallurgically limited. Reactivity limit is greater than 150,000 MWD/1000 kg for this particular reactor. Calculations for other similar reactors indicate reactivity lifetimes in excess of 200,000 MWD/tonne.

(9) Control rod calculations were not made for this reactor, but it appears intuitively that the excess reactivity requirements will be small for a reactor that barely breeds.

b. Fuel Elements

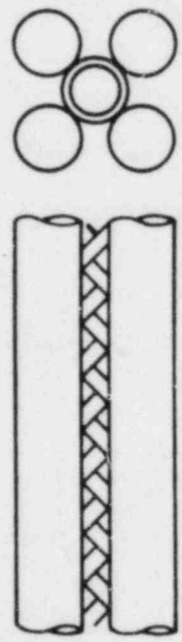
The fuel elements for this reactor can be of conventional design and they may be mounted between the usual type of grid plates in the normal bundles or clusters. For the steam cooled breeder example selected somewhat larger than conventional diameter elements were chosen with thicker cladding to form an extremely rugged element. An element of this sort would be expected to run quite hot in the center, possibly above the melting point of the mixed oxide fuel. Consequently the center portion of the fuel was removed and replaced by a tungsten rod. This rod serves two purposes. First, it is a high temperature material, and second, it is a thermal neutron absorber and helps control the shape of the reactivity curve. However, it is to be emphasized that conventional fuel elements may be used and the required thermal absorbers can be placed in the cladding structure or in the heat transfer enhancers.

Control rods for the example can be similar to those used in LWRs with conventional drives. The shim rods are to be slow moving, serving only shimming and total shutdown functions with coolant density control serving as the normal regulating element and for safety shutdown. A fast regulator rod may be required as described later.

It is to be noted at this point that some form of heat transfer enhancement will be required in order to get the power out of these cores using steam cooling without excessive temperature rises. This enhancement has been considered in two configurations. The first scheme is indicated in Fig. 4a in which a fuel element is wrapped by a spoiler wire. Tests on such elements have been performed by KWU in Germany and wire wrapping has been suggested for this core by its affiliate Combustion Engineering.

Another scheme is one developed by NASA (3) in which fuel elements are separated by a spiral tape. Fig. 4b indicates four elements attached to a spiral spacer. As the elements in this reactor are stainless steel clad, the elements may be brazed or welded to the spacer depending on metallurgical suitability. Such an assembly of elements and tapes should be extremely rugged and provides a heat transfer enhancement by a factor greater than 4. The wire wrapping usually shows

a.) Wire Wrapped Element



b.) Spiral Tape Spacer

HEAT TRANSFER ENHANCEMENT

FIGURE 4

enhancement factors between 2 and 3. (4) The wire or tape would also, as suggested above, provide a convenient location for the thermal poison required by the core to produce the flooded reactivity characteristic.

The problem with the use of the tape is that it creates a greater pressure drop across the core than does the wire wrapping. In a system operating at 1000 psi, the heat transfer coefficient can be increased by a factor of 4, but the pressure drop will also increase by a factor of 3.5 to 4.

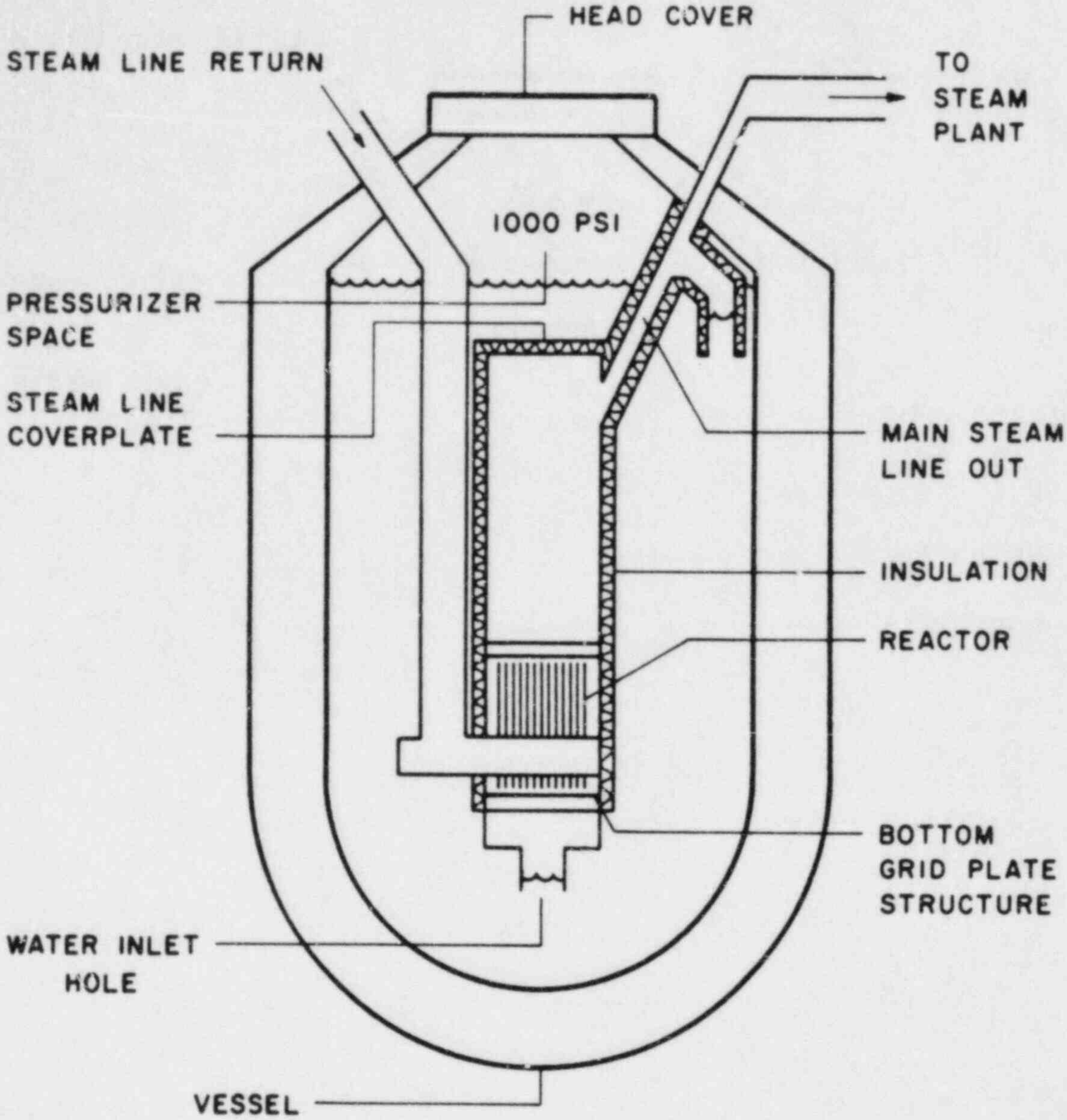
IV THE PLANT

1. General

The reactor as just described must be placed in some type of pressure vessel and connected to a plant. Four types of pressure vessels have been examined and will be described later, but for now let us place the reactor in a non-specific vessel as indicated in Fig. 5. For illustrative purposes the vessel is made of either steel or concrete and contains a single large volume capable of being pressurized to an example pressure of 1000 psi.

Inside the vessel, the reactor is placed in a thin walled steam pipe on top of a massive grid plate. Both the steam pipe and the water external to it are essentially at the same pressure. In this example, the steam pipe is also shown as being thermally insulated from the pool using a wet metallic insulation, but no insulation or only partial insulation may be required. The reactor is presumed to have a conventional set of control rods and control rod drives not shown.

In normal operation the steam pipe is filled with flowing steam of

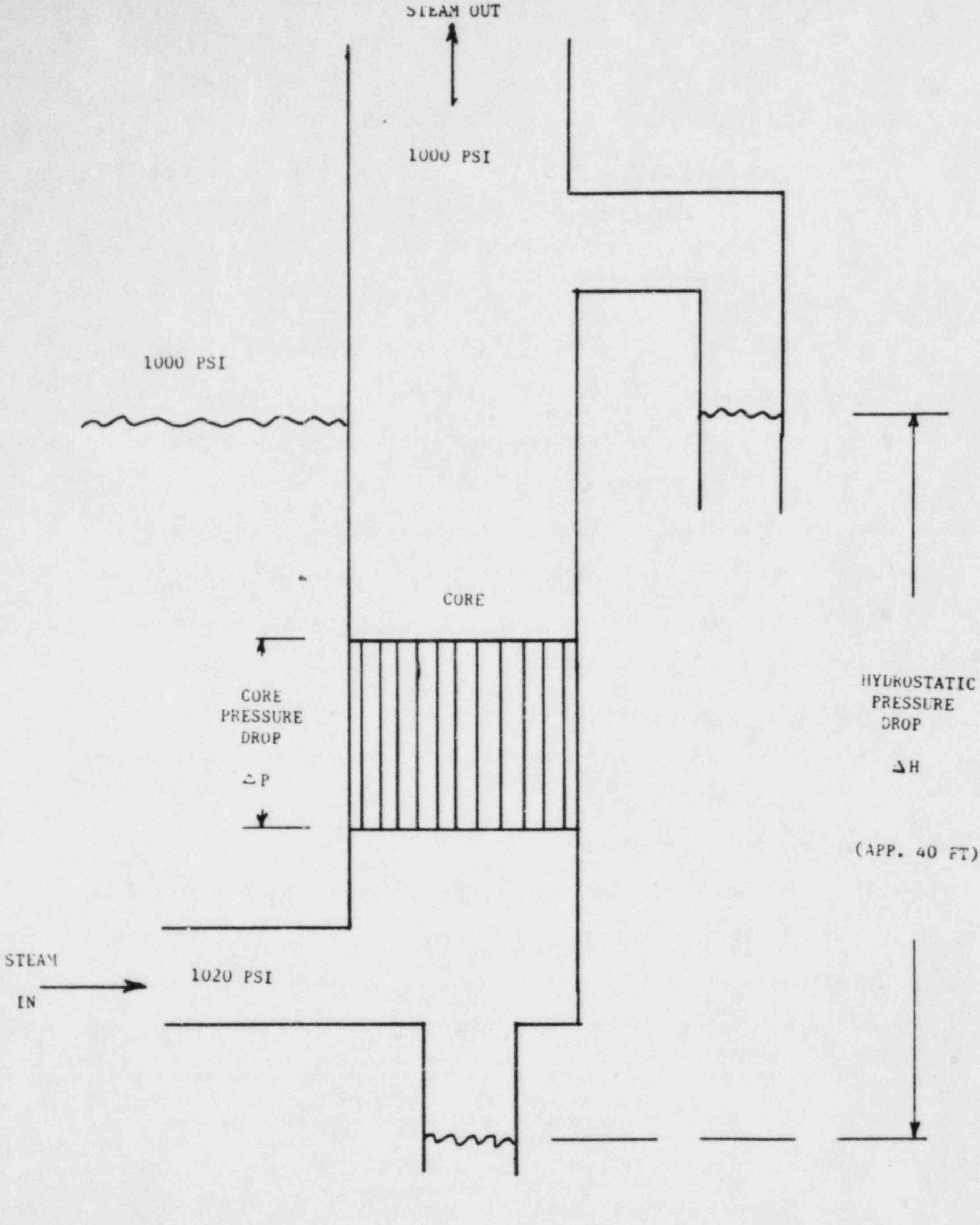


REACTOR SYSTEM IN VESSEL

FIGURE 5

the proper density to keep the reactivity on the peak of the curve as previously indicated. The steam pipe is connected to the water pool at two places, first at the bottom of the steam pipe and second near the top of the pool. A steam-water pressure balance is set up as indicated in Fig. 6. Here the steam is shown as entering the reactor from the plant via a return line at a pressure of 1020 psi. The pressure drop across the reactor is indicated as being 20 psi, and this drop is exactly matched by the hydrostatic head of the pool between the two openings. This balance precludes the entrance of water into the steampipe as long as steam flow is normal. Honeycomb type diffuser sections are presumed to be placed at the steam pipe openings to prevent crossflow and the steam from bubbling through the water.

If for any reason the pressure balance is upset, the water invades the steam pipe, and shuts the reactor off. The normal way of shutting down the reactor would be to turn off the steam blower and allow the steam pipe to flood. The incoming water may be partially heated by passing over the grid plate which has absorbed a few magawatts of gamma heat. Or the insulation may be removed from the steam line and hotter water than becomes available from the pool. Any steam line break internal or external to the vessel would cause the reactor to be flooded. The reactor can, of course, be also shut down by its control rod system, but it is anticipated that the simplest shutdown will be by turning off the blower. Water entry speed into the steam pipe would be controlled to any desired value by proper sizing of the interface openings. In addition, once the reactor and steam pipe are flooded,



PRESSURE BALANCE IN STEAM PIPE

FIGURE 6

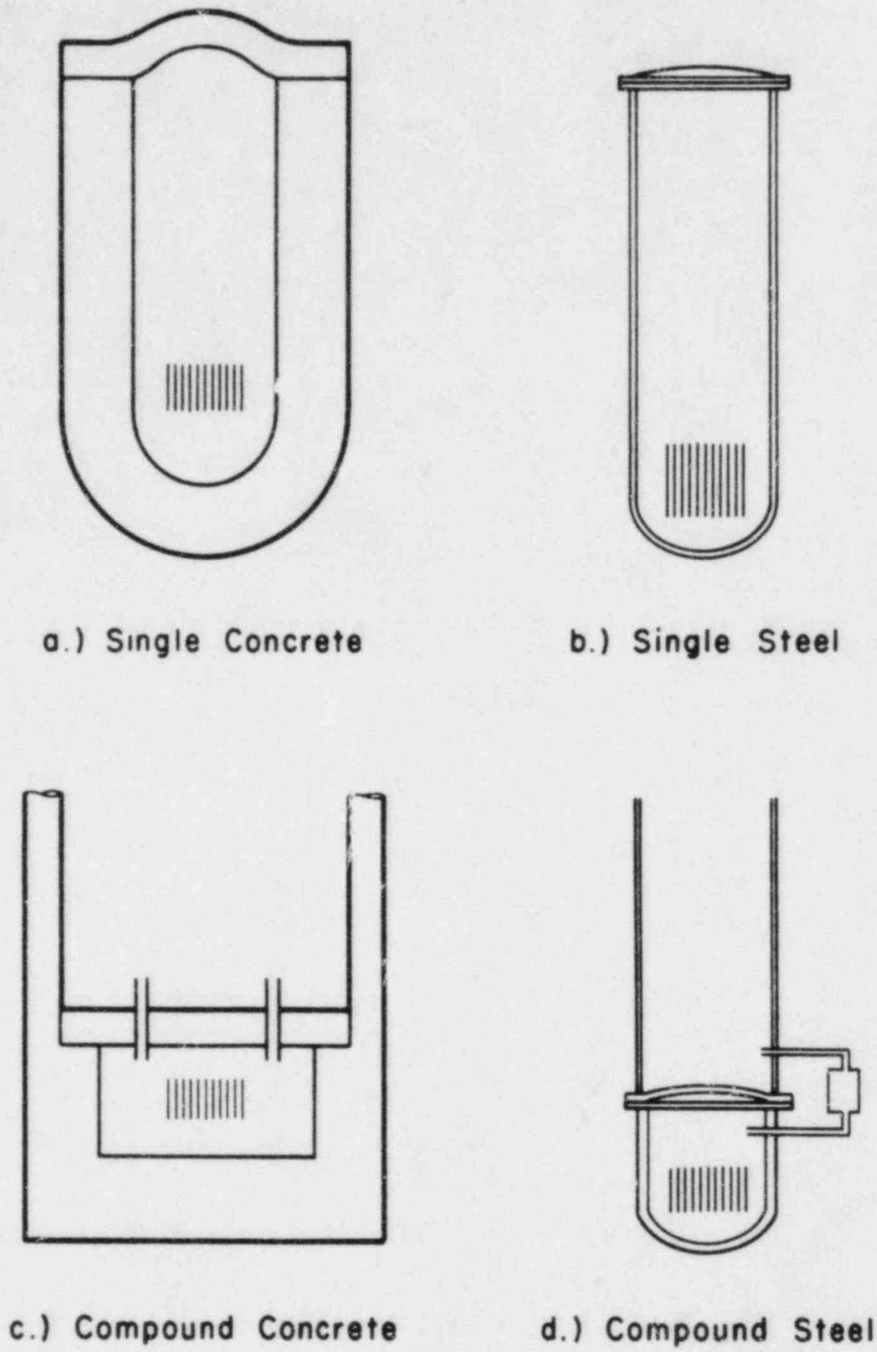
natural draft circulation is set up because the hot reactor is at the bottom of the steam pipe. In this way, the decay heat from the reactor is naturally transported to the pool of water.

2. Pressure Vessels

The four types of vessels that have been considered for this reactor are indicated on Fig. 7. The vessels are first divided into two groups, single volume or compound. The single vessels, as indicated in Fig. 5 are large volume vessels that are pressurized to a single operating pressure. The compound vessels consist of a minimum of two volumes, the first at operating pressure and the other volume(s) at atmospheric pressure. The principal purpose of the compound vessel is to cut down on the amount of high pressure volume so that the vessel system can be made less expensive. Each class of vessel can be constructed either in steel or reinforced concrete. A brief description of all 4 types follows:

a. Single Concrete Vessel

This vessel is typified by the design of the ASEA-ATOM PIUS reactor vessel as described by Hannerz (5,6). The PIUS is a large prestressed concrete vessel, 25 ft thick having an inside diameter of approximately 43 ft, and a height of 100 ft. The vessel has stainless steel liners, and a number of the requirements such as pressurization, feed throughs, anti-leakage provisions, etc. have been worked out. Pressurization is accomplished in a conventional manner, using heaters in a steam space at the top of the reactor



PRESSURE VESSEL TYPES

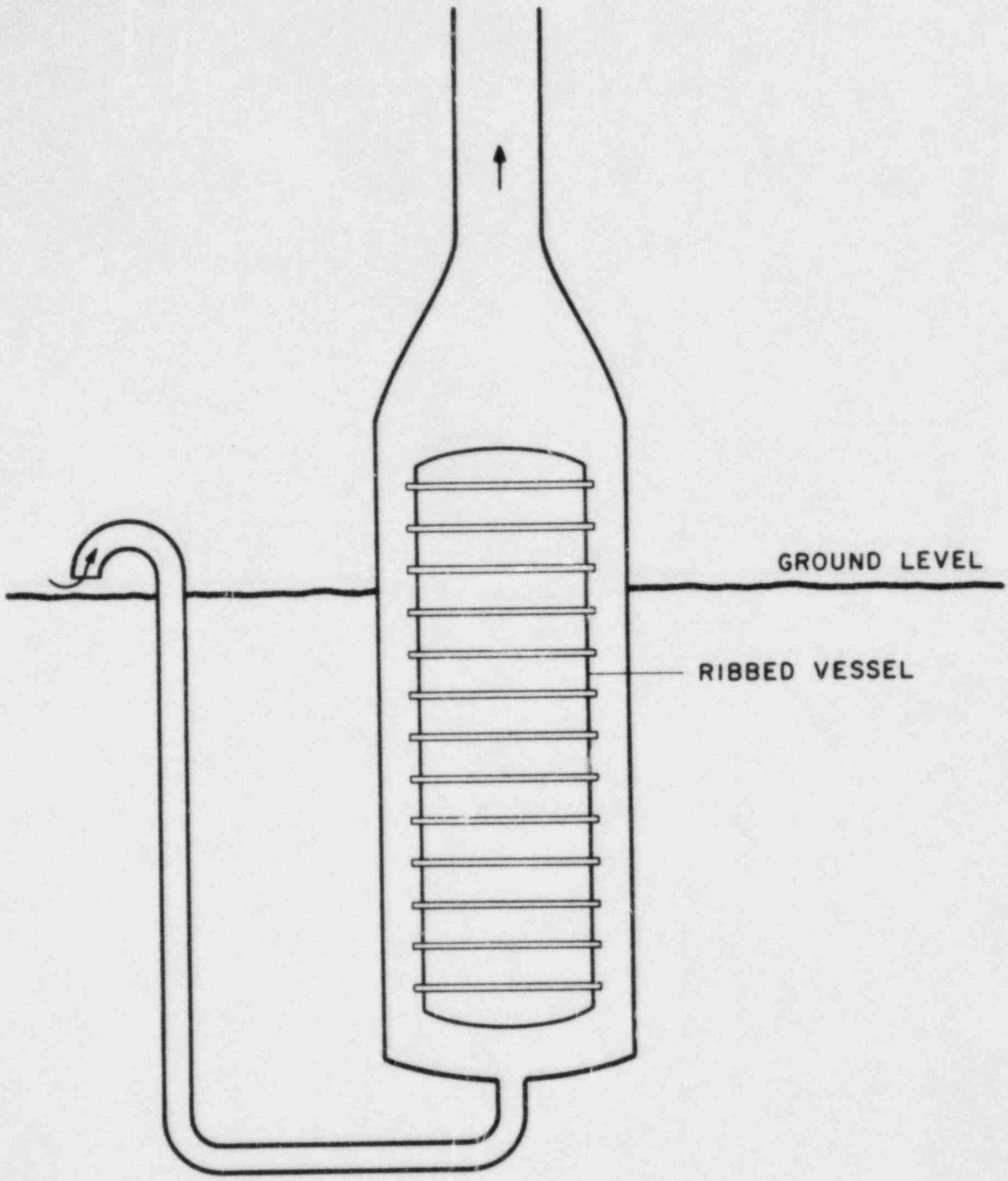
FIGURE 7

pool. This method of pressurization causes all free surfaces in the pool to assume saturation temperature (544 degrees F at 1000 Psia). This fact is important for the steam cooled reactor in that when the reactor is flooded the initial incoming water would be at saturation temperature somewhat alleviating the effect of thermal shock on the fuel elements. This vessel also has considered the problem of sabotage in its design, and has a heavy head closure capable of withstanding a direct 1000 lb bomb hit. Furthermore there is to be only one crane on the site capable of lifting the head, and after this head is in place, this crane is to be disassembled. Reassembly time is estimated at 3 days making it most difficult for a saboteur to get at the reactor. The principal visible problem with this vessel is its high cost, estimated at being approximately $70,000,000. A single concrete vessel for the steam-cooled reactor would not have to be as large as the PIUS vessel. In the steam-cooled reactor vessel, there would be no pumps, heat exchangers, or any of the large components used by PIUS. Only the reactor and steam pipe would be in the steam cooled reactor vessel. Second, the volume of water required in the PIUS vessel is set to some extent by the amount of vessel passive heat removal capacity and the desired water boil off time. In the steam cooled reactor, considerably more passive heat removal is provided, and boil-off is minimized, so that a smaller water volume can be tolerated. (See succeeding sections)

b. Single Steel Vessel

Because of the low operating pressure involved in the steam cooled reactor, it is feasible to consider constructing a large vessel out of steel. The vessel would have a diameter of about 25 ft, a height of about 125 ft and a thickness suitable to the pressure ultimately selected. Such a vessel could be shop fabricated, but might require solution of special transportation problems.

The interest in such a vessel is twofold requiring further study. First, the vessel is expected to be inexpensive. Second, the shape of the vessel leads one to consider the possibility of simple natural draft air cooling for the vessel as a means of removing the decay heat. Fig. 8 indicates a simplified sketch of this scheme. After the reactor is flooded and shutdown, the water adjacent to the reactor begins to heat up via the natural circulation loop and soon starts to transfer some of this heat to the vessel. Ambient air sets up a natural draft on the outside of the vessel creating a chimney like arrangement. The vessel is ribbed to increase heat transfer and to provide more surface area. Elementary calculations indicate that approximately 5 MW could be removed by the system at a surface temperature of 200 degrees F if both convection and radiation are considered. This amount of heat removal is not sufficient to take care of the first several hours of decay heat and if the vessel is depressurized, considerable water will boil away. Alternatively, if the vessel remains pressurized,



SELF COOLED SINGLE STEEL VESSEL CONCEPT

FIGURE 8

the temperature in the vessel and the wall will rise to approximately 578 degrees F at which temperature approximately 25 MW can be removed from the vessel by convection and radiation. As 578 degrees F is below the normal operating temperature of the reactor, this condition may well be feasible.

A third suggestion is to provide a blower and forced air at the inlet to the chimney system as a dedicated heat removal system from the vessel walls. This blower need only operate for a few hours after shutdown before natural draft can take over. If the blower fails, no great harm is done in that either the vessel temperature will rise to a safe value or some water will boil away still leaving the core covered. The blower would also serve other purposes in the cleanup system as will be indicated later.

(1) Vessel Heat Removal System

The above arguments consider only the decay heat removal problem after shutdown. A large single vessel system also has the problem of steady state heat removal from the pool under normal operating conditions. Even if the steam pipe is insulated as shown in Fig. 5, the thermal leakage from the steam pipe to the pool and the gamma heat will range from 0.5% to 1% of the thermal power depending on whether a once through or coaxial steam pipe is used (see Fig. 10). Thus in an 1800 MWT single pool system provisions must be made for removing 9 to 18 MW in steady state operation.

A more conventional way of handling this steady state and decay heat is indicated in Fig. 9. Here the vessel or the vessel liner is indicated as having cooling pipes wrapped around its outside surface. These pipes are arranged in sections with six sections each capable of removing 5 MW each planned in an early concept. Some sections are wound radially about the vessel and others are axially wound as indicated. The top section of cooling coils would be required for the concrete single vessel in any case as cooling would probably be needed for penetrations and seals.

Each cooling section is connected to some form of passive cooling system. For example, a cooling coil could be connected to an elevated air cooled radiator in a natural convection system. Another possibility if sabotage is a prime consideration, is to connect the cooling coils to an earth condenser. This device is a network of pipes buried a few feet below ground anywhere in the exclusion area. Prior calculations (7) have indicated that a field of roughly 100,000 linear feet of buried pipe would be required to dissipate 30 MW. In 1974 such a network of thin walled stainless pipe could have been installed for roughly $100,000. Undoubtedly this price would be considerably higher today.

Other passive shcemes for cooling the vessel or liner might employ heat pipes feeding a local pond or reservoir. In any event some form of steady state dedicated passive heat removal would be provided. A 30 MW system would handle the

PRINCIPAL VESSEL PENETRATIONS
SHOWING VESSEL AND POOL COOLERS

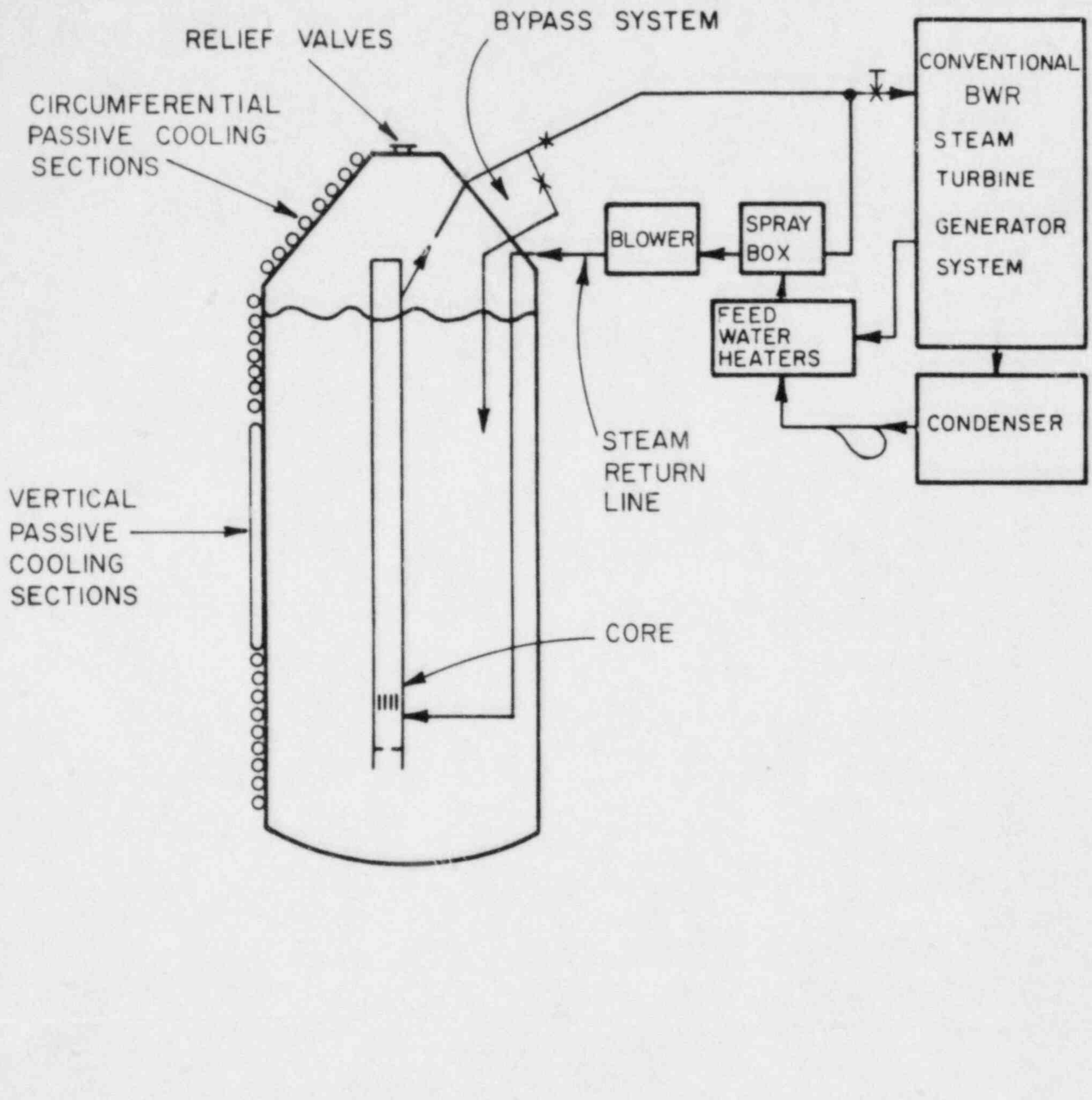


FIGURE 9

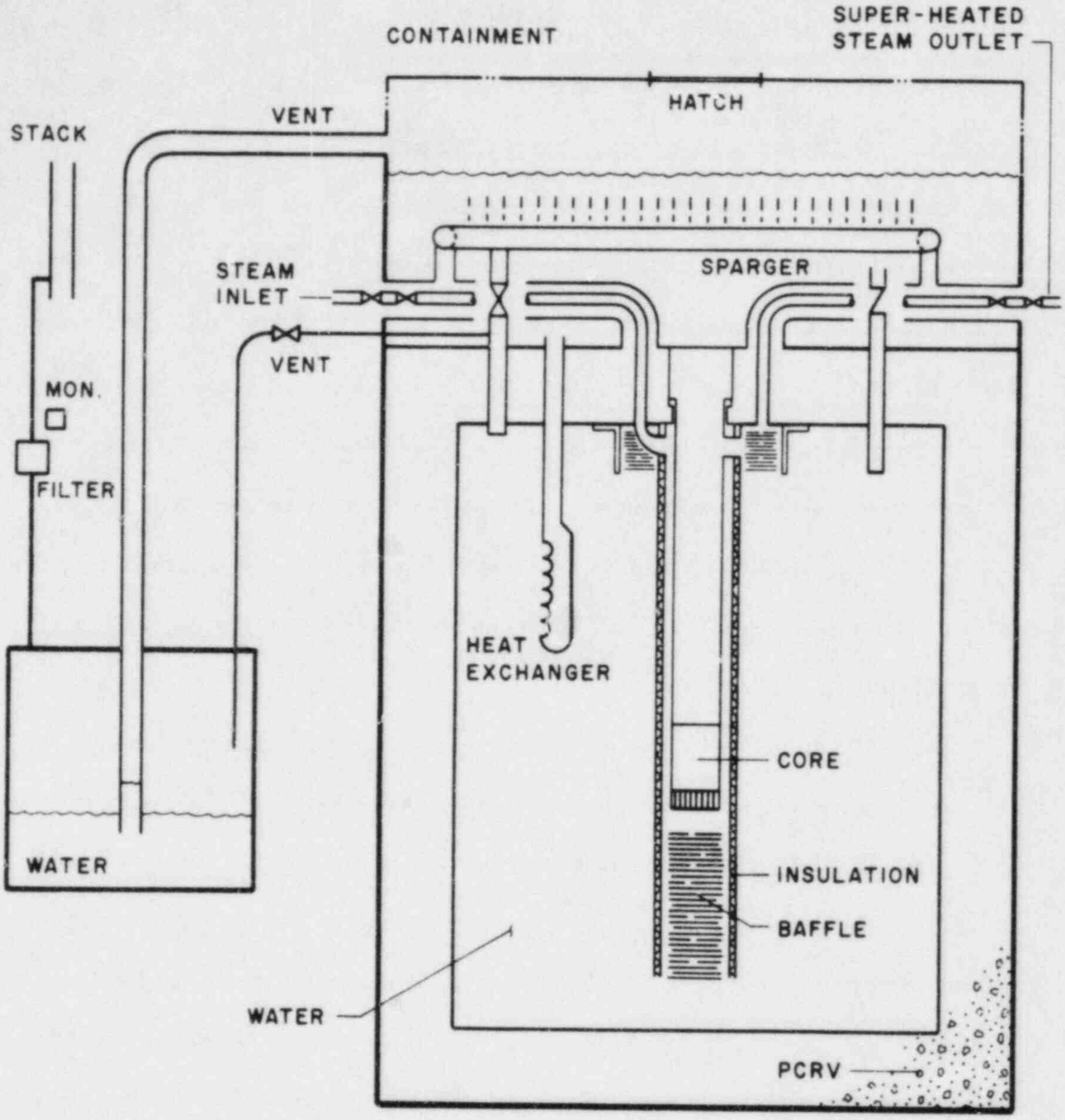
decay heat in less than 2 hours after shutdown.

Figure 9 also shows a version of the process system and the principal steam penetrations. It will be noted that in this version relief valves are located near the top of the vessel. Consequently these valves would be expected to blow off steam rather than water. Blow down would be to some form of external condensation pool in this concept.

c. Concrete Compound Vessel

Figure 10 indicates a version of this type vessel as conceived by C. Storrs of Combustion Engineering Co. His Mark II concept contains two coupled concrete pools with the bottom pool being pressurized to operating pressure and the top pool being at atmospheric pressure. Fig. 10 is not to scale in that the top pool contains 6 times the volume of the bottom pool. The steam pipe is coaxial and the return steam from the plant is first fed into the outer annulus of the pipe. Steam is then led through this annulus and is turned around at the bottom of the core, then fed up the steam pipe and out to the plant. The coaxial arrangement cuts in half the amount of steam pipe area that might be insulated.

The coupling between pools is indicated as being a conventional blow-down system containing relief valves and a sparger. Such systems are currently employed on BWR's and have the advantage of being already approved for licensing by NRC. They have the disadvantage, however, of containing relief valves which at best



STEAM COOLED REACTOR CONCEPT MK II

FIGURE 10

have a bad failure image. In fairness, however, the relief valves in this application are only required to open, and if they do not reseat, a TMI-like situation does not result. All that happens is that the reactor would be flooded and stay flooded. It simply could not be turned on again until the valve was properly seated.

A somewhat better form of relief valve is employed by PIUS. These valves are described in ref. 6 as follows:

"The concrete vessel is protected against overpressure by spring-loaded pressure release valve. These valves can also be used for depressurizing the concrete vessel, e.g., if the level in the latter sinks below a given value, without compromising their duty for protection against overpressure.

This is done by supporting the spring holding down the valve seal in the closed position on a cylindrical body which in turn is held in place by a pressure higher than that in the concrete vessel. This pressure is delivered by any one of four-battery backed centrifugal pumps taking suction from the minimum permitted level in the concrete vessel." (Obviously near the vessel top) "If the level falls below this, the pump head will be lost, the support cylinder can no longer be held in place and the valve opens independently of the pressure in the concrete vessel. When the concrete vessel pressure level is above the minimum, the valves will function as normal spring-loaded pressure release valves."

At first glance this device appears rather Rube Goldbergy, but it really seems to be well thought out and should be an improvement over a conventional power operated relief valve. The problem is that for an ultra-safe reactor that depends completely on coupling the two pools together, there just appear to be too many components. A diverse system in which some of the relief valves were of this type and others were conventional might be a solution, but some scheme that was more passive would be preferred. (see Fig. 12).

The piping marked "heat exchanger" in Fig. 10 is the equivalent of the vessel or liner cooling pipes in Fig. 9. The heat exchanger here is in direct contact with the pool water and may also be set up in sections with each section being coupled to some form of passive heat removal system as previously described.

Figure 10 also indicates the presence of some form of overflow tankage. The overflow tank is a convenient holdup tank for retention of fission products in the event of failed fuel elements and will be covered in more detail later.

d. Compound Steel Vessel

The single steel vessel is unconventionally large and as a self-cooling device suffers from a possible lack of heat transfer surface. The compound steel vessel offers a way around these problems plus it appears to have a number of other potential features that require further investigation. Figure 11 illustrates one interesting concept. Here the reactor is mounted in a conventional BWR type pressure vessel suitably modified to decay heat is indicated in Fig. 9. Here the vessel or the vessel liner is indicated as having cooling pipes wrapped around its outside surface. These pipes are arranged in sections with six sections each capable of removing 5 MW each planned in an early concept. Some sections are wound radially about the vessel and others are axially wound as indicated. The top section of cooling coils would be required for the concrete single vessel in any case as cooling would probably be needed for penetrations and seals.

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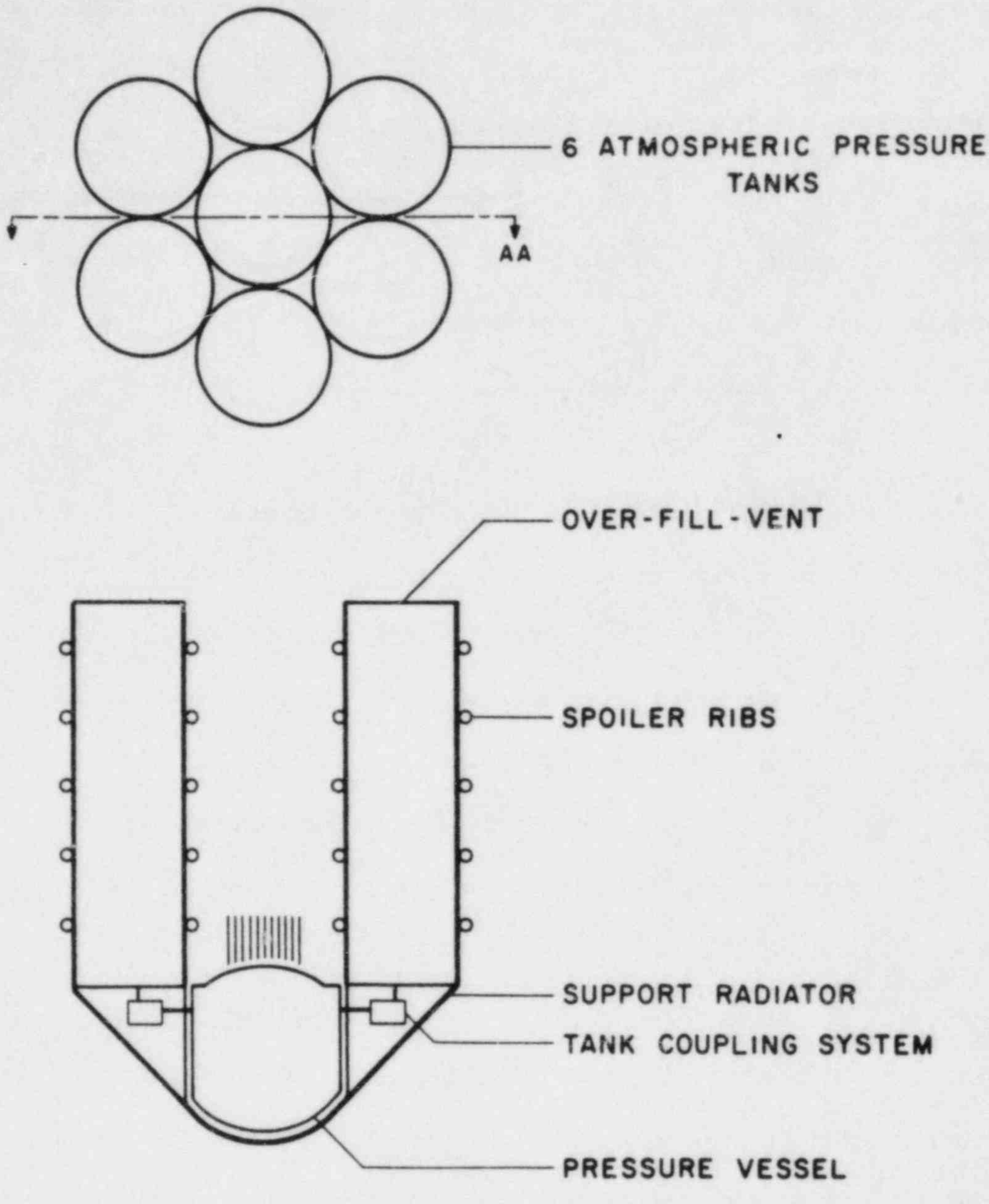


SELF-COOLING COMPOUND VESSEL SCHEMATIC

FIGURE 11

accommodate the steam lines and the lower pressure requirement.

Surrounding the main pressure vessel would be 6 atmospheric pressure tanks each approximately 25 ft in diameter and 50 ft high. For the same volume of water the surface area increases by a factor of about 2.5 and now this assembly in a self-cooling configuration can dissipate approximately 25 MW at 212 degrees F. This power is about all the heat to dissipation required in either steady state or shutdown mode. Again, if slightly more dissipation is desired all that is needed is more volume at atmospheric pressure or that the pool temperature be allowed to rise somewhat.

The six atmospheric pressure tanks are coupled to the main pressure vessel by some semi-passive form of coupling device. For example, a pressure balance similar to that of the main reactor system might be set up. A simplified scheme is illustrated in Fig. 12. Here the six tanks of Fig. 11 are shown coupled to the pressure vessel by one coupling device. Diverse other couplers may also be present. A multistage high pressure centrifugal pump or high leakage displacement pump is shown attempting to pump water into the pressure vessel. Flow exists between the two vessels until the pressure vessel is pressurized up to the pump's rated pressure capacity. The pump then rotates delivering no flow but providing a pressure barrier. In the event the pump is turned off, the high pressure in the main pressure vessel will first blow off through the pump's reverse leakage. After the pressures between the two tanks have equalized, the lower pressure vessel will be refilled by hydrostatic pressure through the forward leakage of the pump. (This scheme is indicated in a more complex form in reference (8) in which two opposing pumps are used with one pump supplying flow through the leakage of the other.)

This concept is interesting in that the multistage pump in effect becomes the system pressurizer as well as the relief valve. And on first thought this device provides a constant pressure system regardless of changes in the plant operating parameters. For example, if the pressure in the main pressure vessel tended to rise for any reason, this excess pressure would leak out by reverse flow past the fixed pressure capacity pump. Conversely, if the pressure in this vessel dropped, the rotating pump would soon bring the system back to the design pressure point.

A single multiple stage combination pump and pressurizer thus provides a simple scheme for overpressure relief and for automatically depressurizing the system when the power is turned off. It may be desirable to place this pump and the main blower on the same power supply. Thus when the blower is cut off the reactor automatically depressurizes and floods.

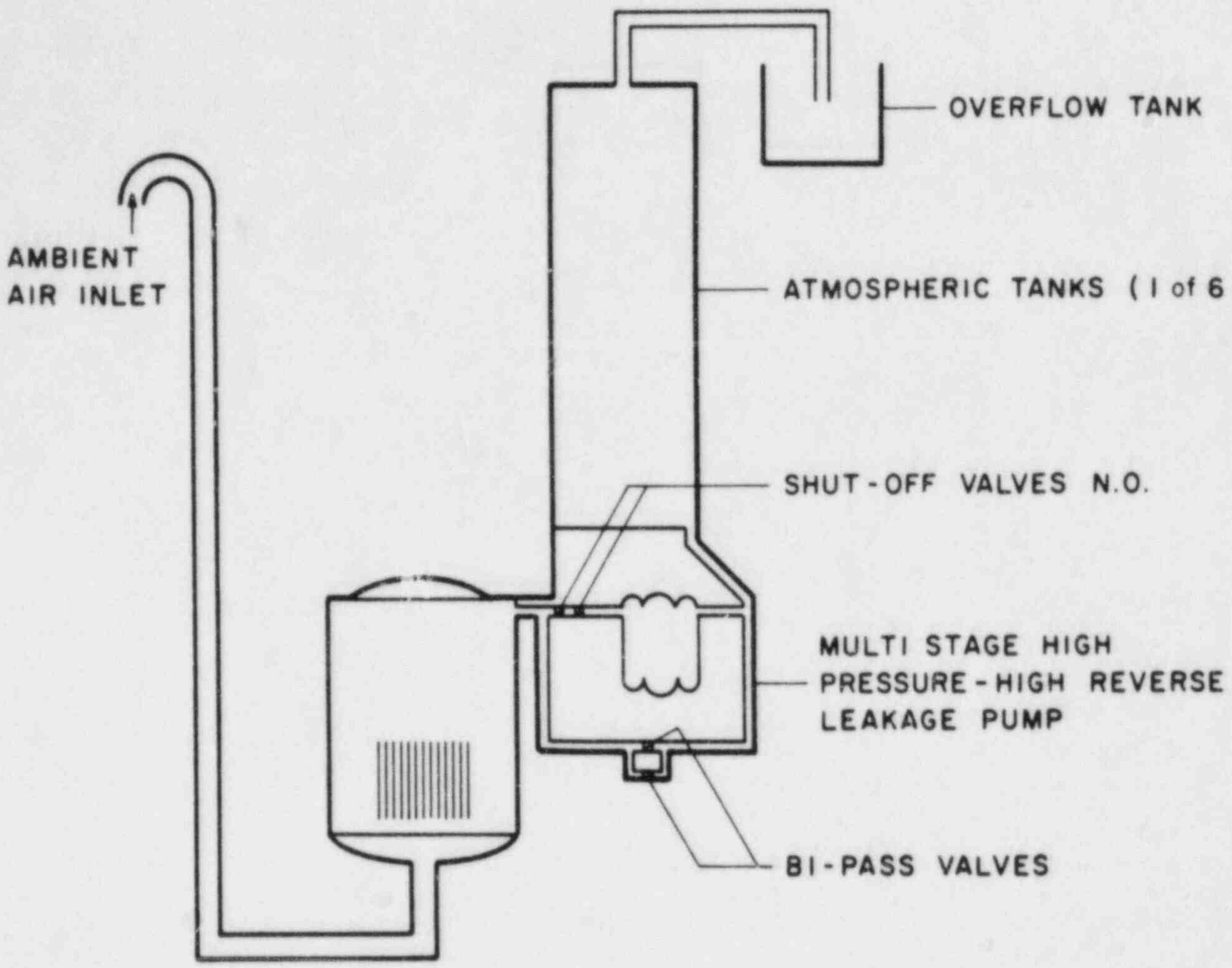
The coupling pump would, of course, be special and might require development. Interestingly, it does not appear that the pump should be very large. Theoretically the vessel could be pressurized through an 1/8" diameter line if desired. However, back leakage requirements would dominate the design as well as heating problems. A pump running constantly at zero flow could be expected to run quite hot and external cooling would possibly be needed.

Figure 12 indicates a scheme whereby six tanks might be coupled in separately with six pumps. Now the problem arises as to what happens if one pump fails. The other pumps would then proceed to attempt to empty their tanks through the leakage in the failed pump. Now blocking valves are required and bypass valves also appear desirable. And although the system would be quite reliable with multiple diverse valving, the system now has become complex and dependent upon engineered safeguards. It appears cleaner to use a single tank or to tie the tanks together in parallel and use a single pump and other diverse forms of relief. Further investigation is required as to the technical and economic feasibility of pressurizing and depressurizing in this manner.

Figure 12 also provides a hint as to the direction a cleanup and pipe leakage system might go. It is clear that coupling and potential mixing of high and low pressure water exists in the multiple vessels. The design of the pump outlet should be to maximize this mixing. Elaborate cleanup and decontamination systems can now be installed in either the atmospheric tanks or their overflow systems as shown later. The point is continuous cleanup can be obtained at low pressure and temperature even while the reactor is operating at high pressure.

3. Thermodynamic Cycle

The heat transfer considerations for power operation are presented



SELF-COOLING COMPOUND VESSEL PRESSURIZER
AND RELIEF SYSTEM

FIGURE 12

for the plutonium-fueled reactor having the reactivity characteristics of Fig. 3 case V. This is an 1800 MWT reactor operating at a pressure of 1000 psi, with a saturation temperature of 544 degrees F and a coolant density of 2.44 lbs/ft^3. Figure 13 indicates the thermodynamic cycle. A blower has been inserted in a recirculating steam line and a reheat cycle is being used. If an attempt were made to use natural convection cooling without reheat and blower two deterring effects occur. First, about one-third of the core surface area would be required to heat the water from the pool to saturation temperature. Second the steam velocity would be very low, about 5 feet per second. This slow speed causes the heat transfer coefficient to be extremely low (about 36 Btu/hr ft^2) and as a result the fuel element surface temperatures would have to be extremely high (2500 degrees F) in order to remove the required heat from the core. It has already been noted that some from of fuel element heat transfer enhancement will be required for this core.

As this core might employ a newly developed fuel element as well as a conventional element, a high temperature element might be considered. This element could have the capability of sitting in dry air at 800 degrees C (1472 degrees F). An element with this capability has already been designed for the TREAT UPGRADE reactor. However, for normal operations, metallurgical considerations would limit the operating temperature to about 1100 degrees F. We have elected instead to confine our studies to fuel elements operating in conventional temperature ranges to minimize development costs.

Figure 13 now indicates the steam cycle of the plant as conceived by A. Fraas of the Institute for Energy Analysis. The thermodynamic

STEAM FLOW CYCLE

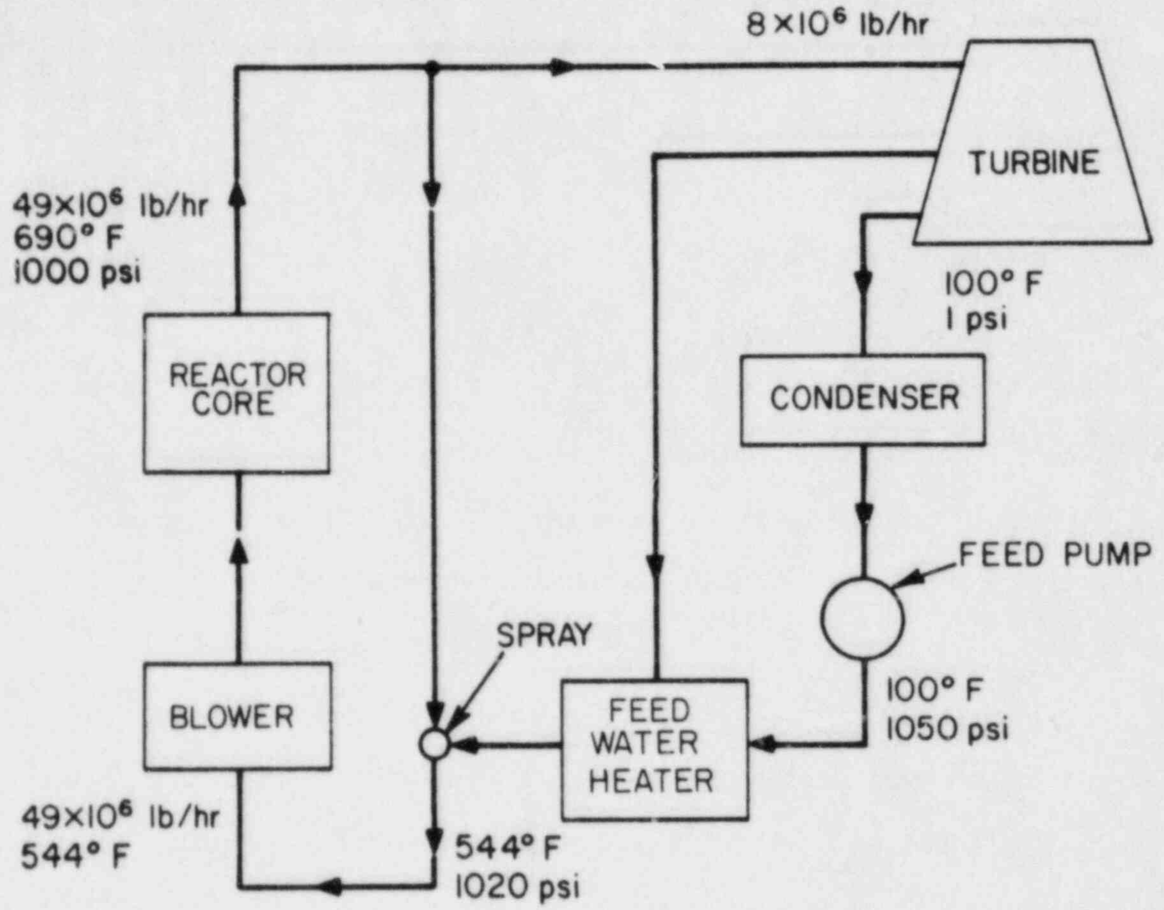
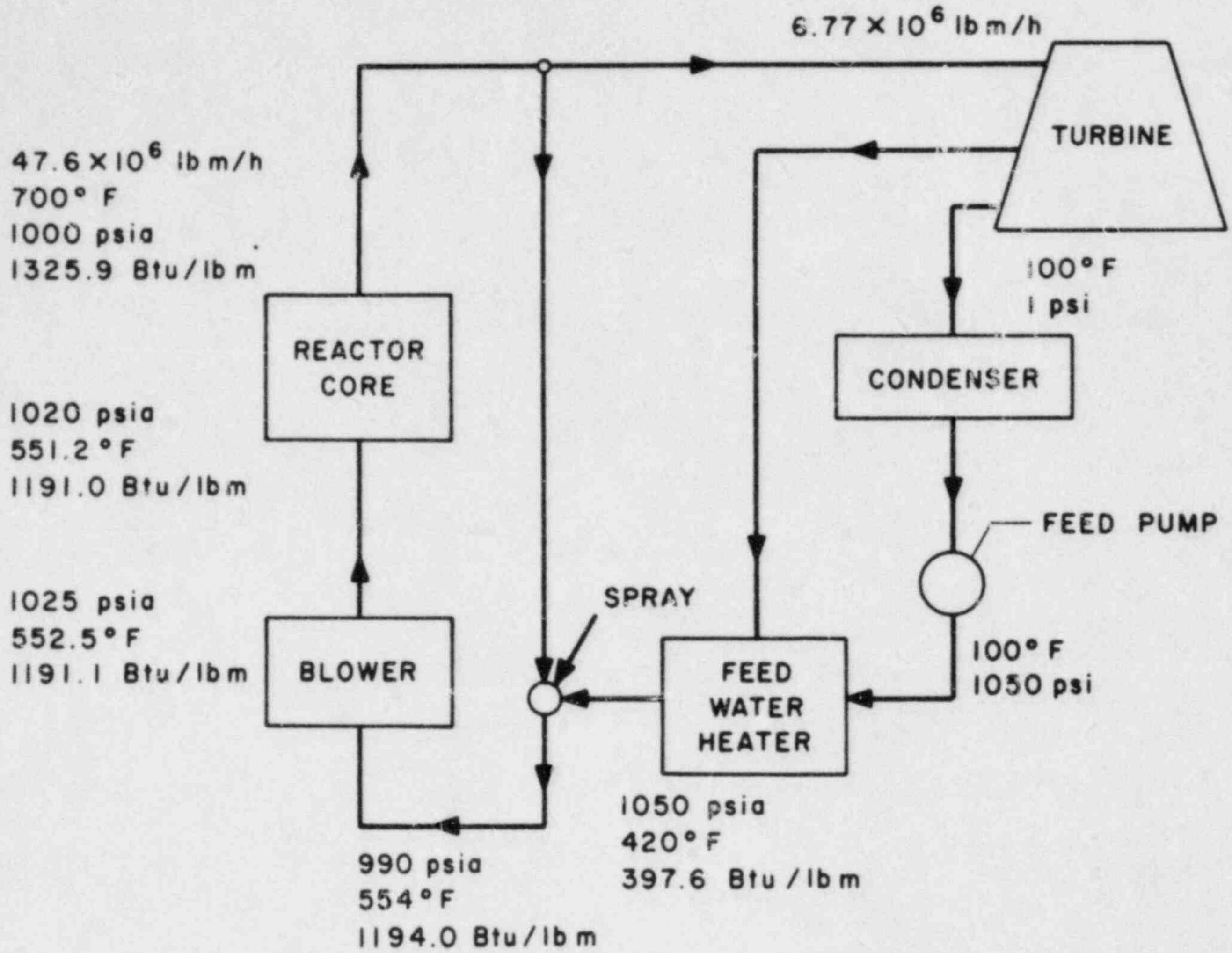


FIGURE 13

constants used are indicated in Table 6. The blower operates in the return line, where it is at the coolest possible temperature. Blower losses, of course, go into heating the recirculating steam. Approximately 16% of the recirculated gas is bled off and fed to the turbine. Not shown on Fig. 13 is also some form of steam dump direct to the condenser. The condensate feed line is now sprayed into the return steam line in a spray box (conventionally known as a desuper-heater) to where the resultant steam temperature is reduced to roughly the saturation temperature of 544 F^o. This steam is then returned to the core inlet where it is superheated to a core exit temperature of 690 F^o at full power.

The blower is to provide a velocity between 125 and 250 feet/second. As velocities of 400 ft/sec have been used in superheaters, this velocity should not provide any problems. At 250 ft/sec, the steam flow cycle time around the steam piping loop will probably be less than one second.

A more refined steady state thermodynamic loop is shown in Fig. 14. This loop was developed by G. F. diLauro of Combustion Engineering and now balances masses and enthalpies around the loop with special consideration to the reactor core inlet conditons. In order to avoid corrosive and erosive effects on core materials the steam entering the core should be slightly superheated to insure no water droplets in the steam. Sufficient superheat must be provided to the steam at the exit of the circulator-blower so that heat loss and pressure drop between the circulator and the core inlet can be accommodated. In addition, the inlet of the circulator should be, at least, dry saturated steam in order to avoid the erosive effects of water droplets. Figure 14 shows



STEAM FLOW CYCLE

FIGURE 14

the steam cycle conditions that meet these requirements. A typical feedwater temperature of 420 degrees F is assumed. The pressure of the spray box-circulator region is assumed to be at 990 psia. The spray box produces essentially saturated steam for the circulator which produces 1025 psia steam. This allows a 5 psi pressure drop to the core inlet. Heat loss from the blower and transport is also assumed.

Because water getting into the blower could destroy blower blades it is also essential that basic provisions be taken that no water can get into the blower. A simple arrangement would merely provide feed pumps too small to ever permit solid water to get into the system. Other precautions may also have to be provided.

V. OPERATION

1. Startup

Startup of the steam-cooled reactor is somewhat more complex than the startup of conventional LWR's and BWR's. The complexity stems from the fact that a shutdown steam-cooled reactor is presumed to be flooded and means must be provided to clear the steam lines of water and ultimately to arrive at the optimum steam density. At least two approaches appear possible.

Table 6 STEAM-COOLED BREEDER REACTOR

Thermodynamic Constants

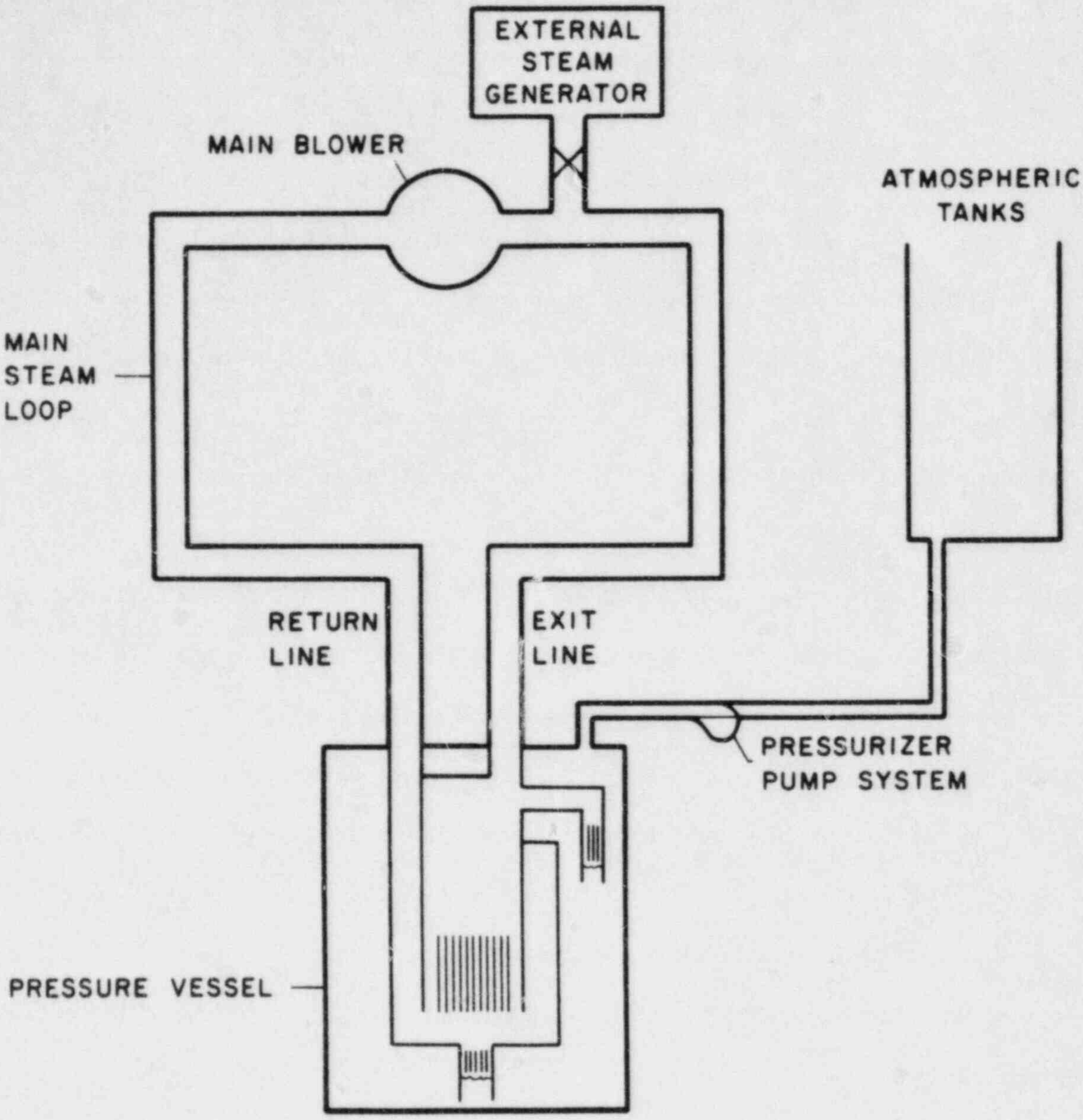
Power	1800 MWT = 6.13×10^9 Btu/hr
Specific power	22.5 kw/kg
Power density	58 w/cm^2
Pressure	1000 psi
Inlet temp. ^oF	544
Outlet temp, ^oF	690
Steam flow, lb/hr	7.9×10^6 for boiling and superheat
	49.1×10^6 for superheat only
Recirculated steam flow, lb/hr	41×10^6
Feedwater flow, lb/hr	8×10^6
Enthalpy - sat. water @ 544^{o}F	542 Btu/lb
- sat. steam	1192
- superheated steam	1318
Δh of superheat	1262
Fuel el. area, ft^2	33,411
Free flow ratio	.65
Equiv. passage dia., in	1.8
Flow area, ft^2	130.7
Core dia., ft	16
Core height, ft	8
Fuel el. dia., in	0.75
Q/A, Btu/hr-ft^2 (av.)	185,000
Heat transfer coef., Btu/hr ft^2 ^{o}F	2300
Film ΔT, ^{o}F	80

a. Line Blowout system

Figure 15 indicates a potential startup scheme based on the assumption that the compound steel vessel is being used in conjunction with concentric reactor steam lines and that the pump-pressurizer system is available. Modifications of this system would be used for other pressure vessel-pressurizer combinations. In Fig. 15, the initial assumptions are made that the system is depressurized, the reactor is flooded, shutdown, and that the control rods are in the core. The figure indicates a closed steam system with fragmentary reheat lines and main blower only indicated. An external source of steam is also capable of being coupled into the steam lines. The pressurizer pump is turned off.

The initial step is to couple in the external steam generator. This generator can produce steam up to the operating pressure, for example, 1000 psi. Full pressure is not required to blow the water out of the system. Roughly only twice the pressure drop across the core will suffice. The water will be blown out of the steam lines and steam will ultimately bubble out of the diffuser openings in the steam pipe.

The steam will exit the pressure vessel via the leaky pump and will bubble up through the atmospheric tank. As the water level is lowered in the steam lines it will approach the top of the core. When the level is a few feet above the core, the main coolant blower is turned on. The blower outlet pressure now will be higher than the static water head and an agitated water and steam mixture will be available to cool the core while the water level in the steam pipes is steadily reduced.



BLOWOUT STARTUP SYSTEM

FIGURE 15

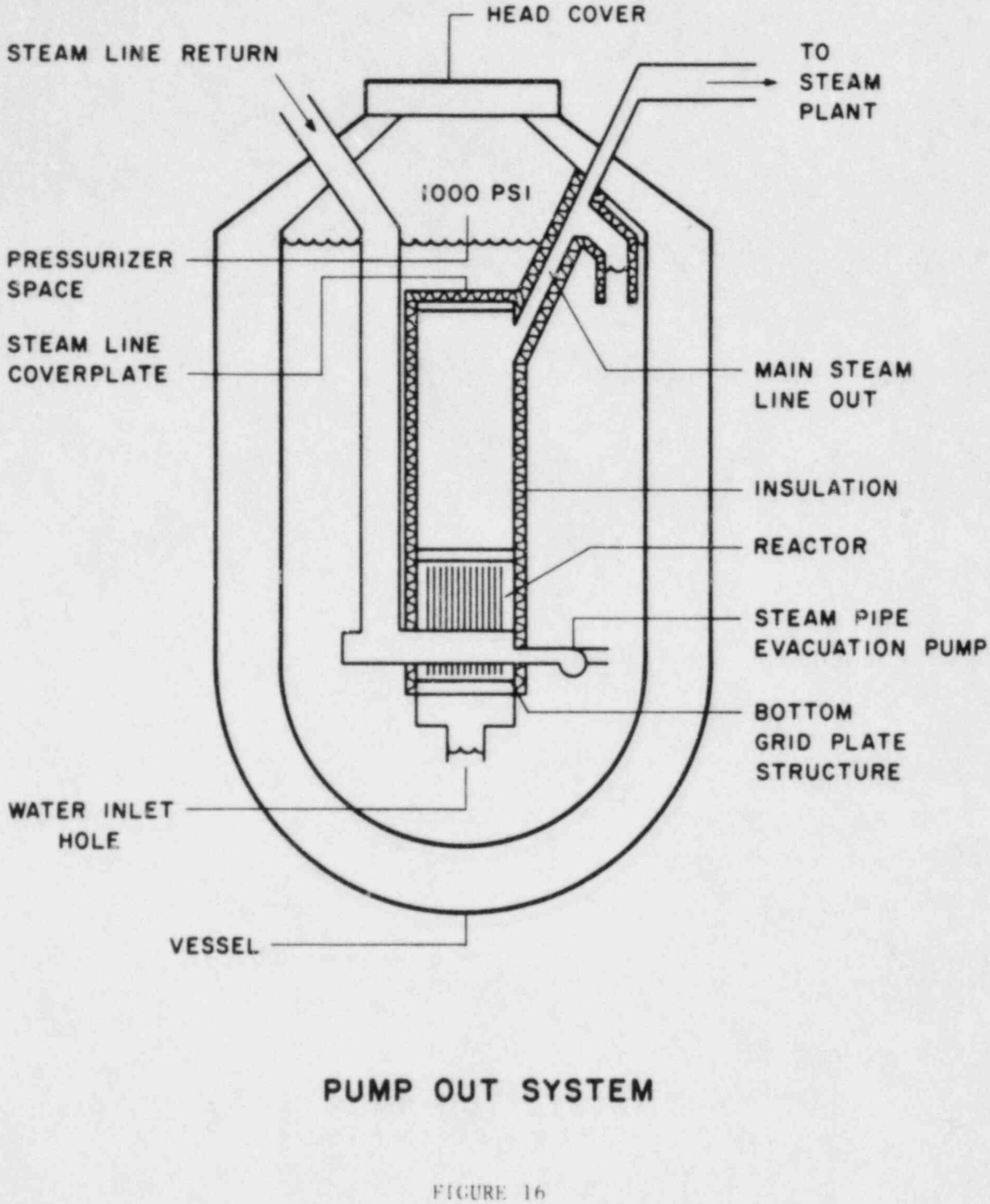
Ultimately, the pipes will empty and the correct pressure drop to match the hydrostatic head will be established. The pressurizer pump is now started and water pressure slowly increased to operating pressure.

In order to prevent some reflooding, it will be necessary to approximately match increases in water pressure with increases in external steam injection pressure. During this period the bubbling will have stopped and the interfaces will remain established. A minor point to be explored is the possibility of setting up a water oscillation while juggling both steam and water pressure. This condition can be avoided by the application of suitable damping in the water system if required.

Once the system operating pressure is reached the external steam generator is valved off and shut down. The steam lines are now clear, the reactor vessel system is pressurized, and the reactor can now be made critical via the control rods. The steam density is incorrect for power level operation, and it is assumed that either there is sufficient control rod worth to override the density coefficient, or the control syste can establish the proper denisty regardless of power level. The system is then brought to power automatically with the control system establishing proper parameters in all sub-systems. Obviously a startup system this complex must be highly automated with proper interlocks and safeguards to always assure core cooling.

b. Pumpout System

Fig. 16 shows a possible implementation of this scheme. Here a single vessel, either concrete or steel, is indicated, and again



PUMP OUT SYSTEM

FIGURE 16

simple variatons appear possible for the compound vessel. Although in and out steam pipes are indicated for the reactor an annular feed would probably be used as shown in Fig. 15. In this instance the vessel may be pressurized in a conventional manner to start, but the core is presumed to be flooded. From the figure it will be noted that a steam pipe evacuation pump has been added to the system connecting in under the reactor. The pump inlet is from the steam pipe and the exit is to the pool. In practice, the pump and its associated valve might be mounted external to the vessel for easy maintenance, and two high pressure vessel penetrations would be required.

The steam pipe evacuation pump is designed to have sufficient capacity to overcome the water pressure and inward leakage caused by the water head in the r actor vessel. That is with the pump operating it can pump out water to the pool faster than water can leak into the pool. Thus the steam pipe will be evacuated of water and the pump will be required to handle a mixture of steam and water for a while. Again, once the steam pipe is evacuated to slightly above core level, the main blower can be started and the pressure drop across the core established. The reactor can now be made critical via the control rods.

An external steam generator again will be rquired for the initial startup. It is not clear on subsequent startups whether this source of steam will be needed. Up to a few megawatts of gamma heat will be present for succeeding startups and this may generated sufficient steam to allow the control system to establish the correct reactivity density as before. In either case the steam generator would be available if required.

Neither of the two schemes outlined may work exactly as indicated without some additional valving. Once the proper geometry is established water flow paths can be examined more critically and refined techniques made available for evacuation, continuous core cooling and startup.

2. Power Level Operation

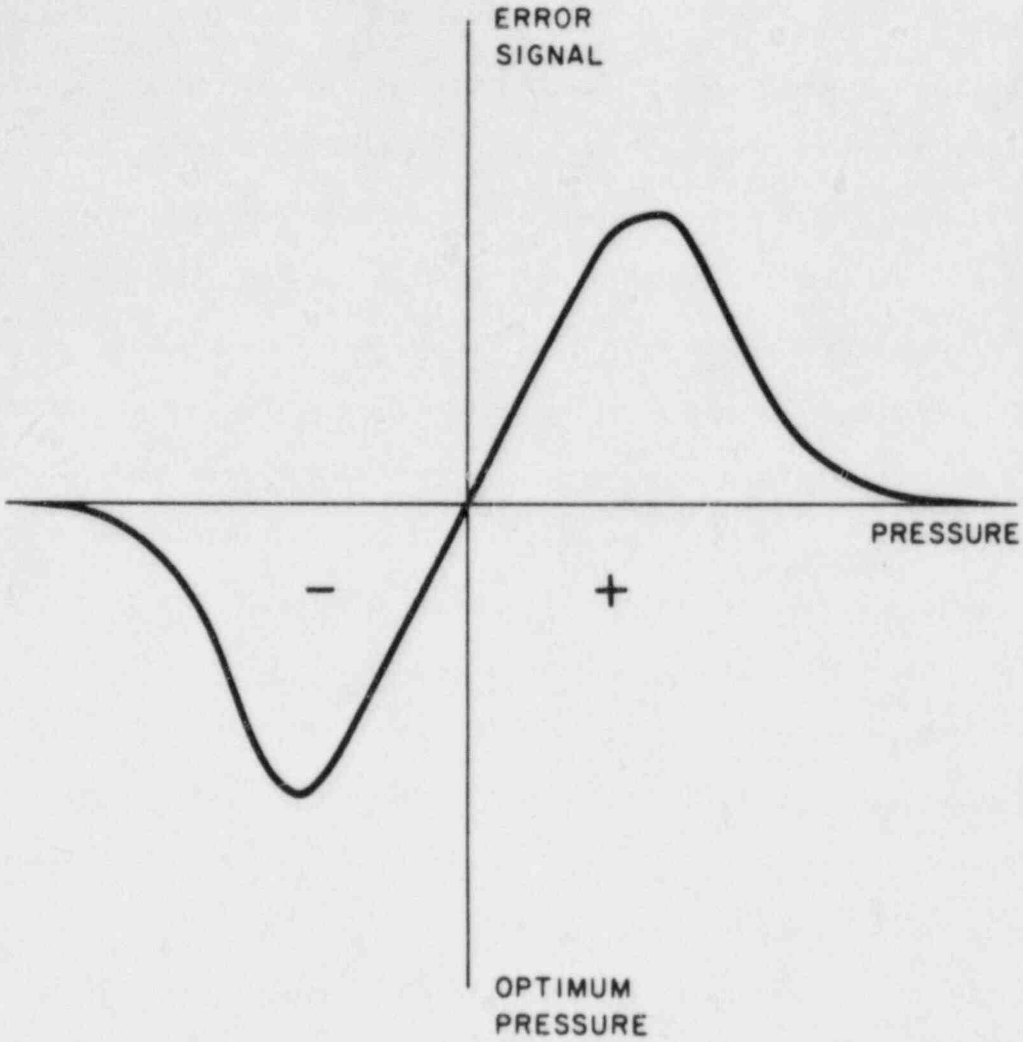
a. Reactor Stability

With complex reactivity curves as indicated in Figs. 2 and 3 it is prudent to inquire as to the stability of these reactors before attempting to control them at power. In particular the density coefficient, and hence density feedback, can be either positive or negative depending on which side of the reactivity peak the reactor operates. This feedback coupled with the doppler effect can provide two path feedback having different time constants and different signs and magnitude. The way to get around these stability restrictions is to always operate the reactor at or near the peak of the reactivity curve. At the peak, the density coefficient is zero. Hence there is only one feedback path present, the doppler feedback. Reactors having only one feedback path cannot oscillate, (9) therefore the reactor is always stable at power when operating at the peak of the curve or close to it. Way-off-peak operation is then limited to subcritical operation. And again, subcritical reactors, even with two feedback paths, are stable.

One can, however, envision off-peak operation at criticality. And although the control system will attempt to eliminate this situation, it must be considered. Appendix C indicates by a simplified analysis what the boundaries of the problem are. Crudely it appears that if the doppler feedback is greater in magnitude than the density feedback, the system again will always be stable. This situation can be achieved by shaping the reactivity curve so that it is relatively flat at the peak. More sophisticated analysis will be required, but it does not appear as though reactor stability will be a problem.

b. Reactor and Plant Control

The control system for a steam-cooled reactor plant will be more complex than that of a conventional LWR. Yet, because of the unique reactor characteristics, this complexity can yield greater performance without in any way compromising safety. To appreciate this fact, envision that the control is such that its normal reactor operating range is restricted. That is, the control system functions normally as long as the system pressure or density remains within a certain narrow range. If this range is exceeded in either direction, the control system would collapse and the reactor allowed to protect itself by its normal reactivity characteristic. This statement leads to a generalized error curve of the shape of Fig. 17. The reactor now is essentially invulnerable to control system failures. That is the design would be such that a component failure would cause the system to drive or demand either a very low density



GENERALIZED ERROR CURVE

FIGURE 17

or a very high density, in either case shutting the reactor down. This opportunity has not been available before in conventional reactors in that in one direction the drive potentially could be toward higher reactivity. So regardless of the detailed control scheme that would be adopted, this particular feature would be incorporated and precautions taken that failures cannot occur to drive the system to the center of its operating range--only two extreme end point failures permitted. (With a digital system this is not difficult. Components are either on or off.)

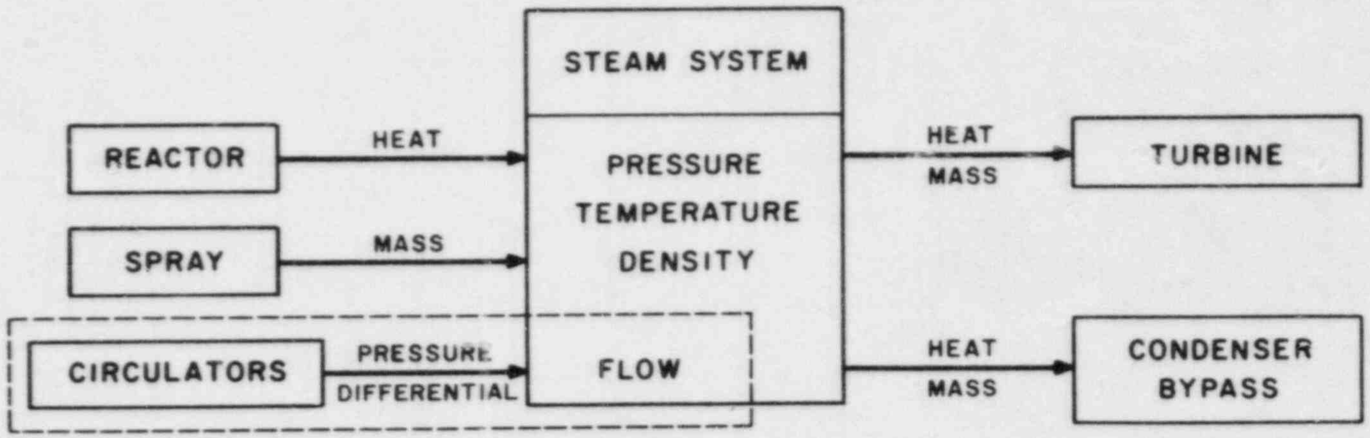
Another general control feature that should be adopted because of the uniqueness of the system is that the system response time should be much faster than in conventional control systems. This response is made possibly by the extremely small loop circulating time. Steam exiting the reactor goes around the recirculation loop in under two seconds because of its high velocity. Typical recirculation times in a conventional PWR might be 20-40 secs. This situation would permit the use of either a very fast spray control valve or a very fast regulating rod (possibly limited by spray mixing or neutron detector statistics). Such rods were developed in the 1950's at Oak Ridge and Argonne National Laboratories and had response times in the neighborhood of 0.1 seconds. For the rods of the steam-cooled reactor, this might require considerably more power for the regulator rod drive than do conventional drives.

With a reactor system responding in the neighborhood of a few tenths of a second and a plant that responded in a few seconds, transients would not have time to reach any appreciable magnitude before the control system caught them and turned them around. (This is in comparison with the few minutes response time before most conventional reactor transients are over.) And, of course, the amplitude of system transients would be small, and would not challenge the safety system at reasonable settings. If the safety system is required to shut the reactor down only on very rare occasions then availability is obviously improved.

These arguments lead one to a sophisticated modern digital control system of a type never before attempted for reactors. Fortunately the special characteristics of the steam-cooled reactor enhance such a design. Before proceeding to the philosophy of the digital design, we examine the reactor and plant control problems in conventional analog terms.

c. Conventional Control System Approaches

A description of a conventional control system using an interesting algorithm is presented by C. Storrs of Combustion Engineering. This approach continually measures the mass of steam in the circulating system and adjusts it to a desired mass. Although there are some difficulties with this approach as handling the required peak in density shifts with time, the following description in Storrs' language is useful in bounding the problem. "To understand the control of the steam-cooled reactor, please refer to Fig. 18. The steam system is represented by the central box, and characterized by the parameters of pressure, temperature, density and flow. (It is recognized that these are not independent.) The reactor adds heat to the system, the spray system adds mass, and the



CONTROL SYSTEM ELEMENTS

FIGURE 18

turbine removes heat and mass. The circulators create a presure differential which causes flow unless the system is blocked. Since constant-speed circulators are assumed, these need only be considered during stratup and anticipated transients, and will be omitted in discussions of normal operation for simplicity. Similarly, a bypass to the condenser is shown which is used only for startup and anticipated transients.

The behavior of the system is such that increasing reactor power will increase pressure and temperature; increasing spray will increase density but reduce pressure and temperature; and increasing flow to the turbine will decrease pressure, temperature, and density.

Figure 19 shows the same thing in greater detail, with a simplified control schematic added. The objective is to deliver the demanded turbine power while maintaining constant turbine inlet pressure and temperature.

The power demand is shown at the top right, and will incorporate limits on rate and maximum setting. The turbine control system compares this with the actual turbine power, and adjusts the turbine control valve accordingly.

The reactor control system (top left) compares the reactor power derived from neutron sensors with the turbine power. The output is limited and biassed by system temperature and pressure. The output is used to drive the control shims in or out. These shims have limited reactivity capability; thus control system failures cannot result in serious positive reactivity excursions.

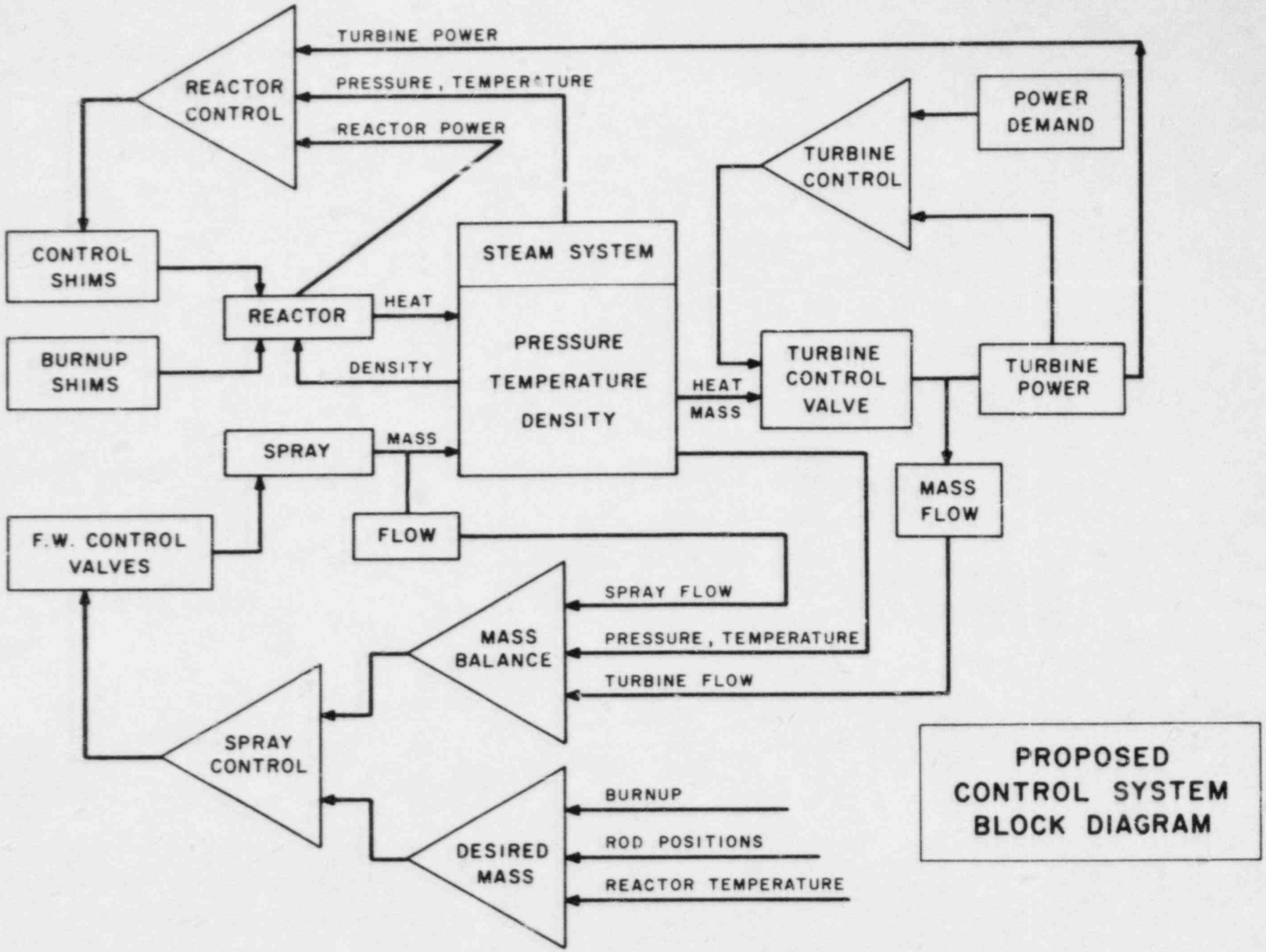

TURBINE POWER
REACTOR CONTROL
PRESSURE, TEMPERATURE
REACTOR POWER
POWER DEMAND
TURBINE CONTROL
CONTROL SHIMS
STEAM SYSTEM
REACTOR
HEAT
BURNUP SHIMS
PRESSURE
TEMPERATURE
DENSITY
DENSITY
HEAT
MASS
TURBINE CONTROL VALVE
TURBINE POWER
SPRAY
MASS
FLOW
MASS FLOW
F.W. CONTROL VALVES
SPRAY FLOW
MASS BALANCE
PRESSURE, TEMPERATURE
TURBINE FLOW
SPRAY CONTROL
DESIRED MASS
BURNUP
ROD POSITIONS
REACTOR TEMPERATURE
PROPOSED CONTROL SYSTEM BLOCK DIAGRAM

FIGURE 19

Other, manually-controlled shims are provided to compensate for burnup.

It may be preferable to use the power demand signal rather than the actual turbine power in order to anticipate changes. Although the reactor is trying to maintain constant outlet temperature and pressure, controlling directly from these parameters is likely to be more difficult due to the complexity and thermal inertia of the system.

The feedwater spray control is shown at the bottom of Fig. 19. It is basically trying to maintain constant system inventory by comparing mass flow to the turbine with feedwater flow. This signal is limited and biased by system temperature and pressure. In particular, the temperature should be kept above the saturation point to avoid core flooding and complete reactor shutdown. If the pressure or temperature become excessive, the spray will increase to reduce them and at the same time reduce reactor power. Presumably excess density will reduce reactivity more and faster than shim rod motion. Increased spray will also be demanded by an error signal from the reactor control showing that the reactor power exceeds the turbine power by more than a deisired amount.

The spray control system calculates the mass of steam in the system (from inlet and outlet flows and from thermodynamic parameters) and compares it with the desired mass (hence density). The desired mass is derived from an algorithm involving burnup, rod positions, reactor temperature, and any other parameters which affect the density for greatest reactivity. Because of the size, complexity and thermal inertia of the steam system it seems unlikely that dithering the feedwater control valves will result in an interpretable signal from the reactor. Periodic experimental verification of the density algorithm will be made over a limited interval.

It is interesting to note that during normal operation changing reactor power has no effect upon the density.

It is now possible to describe how to reduce power and to cope with a turbine trip. For a normal power reduction, the reduced power demand causes the turbine control valve to close, reducing mass flow to the turbine and turbine power output. The reactor control senses the reduced power and the increased pressure and temperature, and drives in shims to compensate. The spray control senses the reduced turbine flow and reduces spray to maintain system inventory and density.

If the shims cannot reduce reactor power fast enough, an error signal and/or excessive temperature and pressure will cause a feedwater spray override, increasing spray flow. This reduces pressure, temperature and reactivity. When pressure, temperature and reactor power are restored to their proper control range, the spray control resumes its normal function of restoring and maintaining system inventory.

On a turbine trip, feedwater spray and shim insertion will likely be inadequate to control the system. The bypass valves to the condenser open, removing heat and mass from the system."

The above description as previously stated is a conventional slow responding control system. From the previous section it does not take advantage of the fast recirculating loop transit time.

Another philosohical control scheme that employs a different algorithm and is capable of being designed for fast response is defined by the schematic diagram of Fig. 20 in analog terms. The system is similar to the previous one except now provisions are made to control on the peak of the reactivity density curve at all times by an actual calculation of the reactivity. As indicated in Fig. 20 a measurement is first made of the neutron output of the reactor. This neutron level is then fed into a fast reactivity computer and a reactivity calculation made. The calculation consists of using some form of the normal reactor kinetics equations as modified by changes in temperature and pressure. In the conventional kinetics equations solution, reactivity is usually the input variable and neutron level is then calculated. Here neutron level is measured and reactivity calculated. Such devices, previously called reactivity simulators, have been built in the past and successfully used at Brookhaven National Laboratory for rod calibration purposes etc.

It will also be noted from the diagram that some form of density measurement would also be required. The reactivity computer will indicate whether the reactivity is zero or not, but does not provide enough information to say whether more or less spray water is required. Some form of recent density history would be needed to provide a sign sensing error.

The remainder of the control system is quite conventional with a power demand signal originating at the generator and fed forward

SIMPLIFIED CONTROL SYSTEM

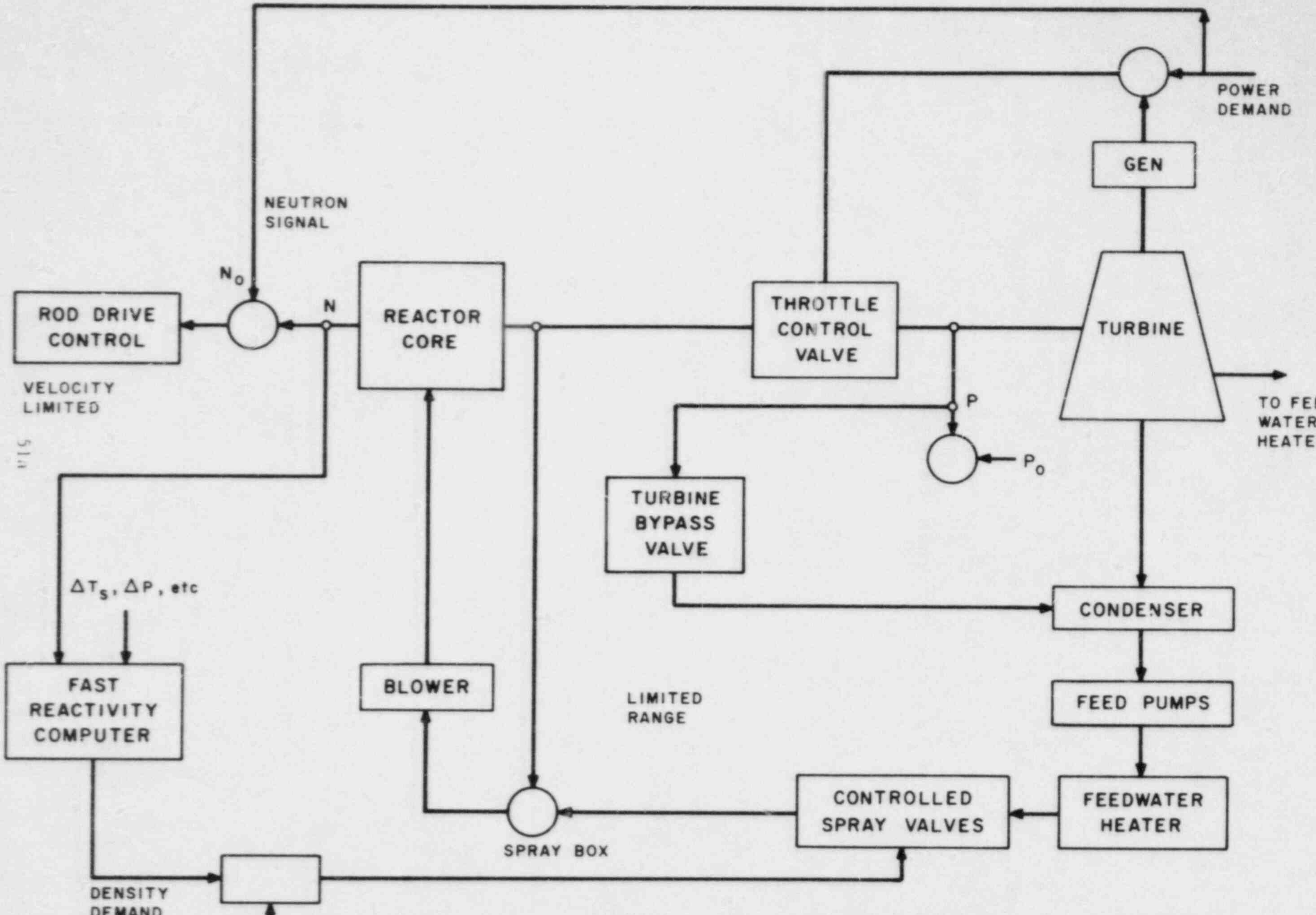


FIGURE 20

to the reactor control rod drives. This connection is in keeping with the fast control approach particularly in that this system cannot be expected to naturally load follow. It will, of course, load follow very well in the negative direction, but will clearly have to be forced in the case of higher power demand. Natural load following is desirable in conventional reactors, but in the case of the steam cooled reactor, it contributes very little in that the fast control system provides much quicker responses to load demand changes in a completely safe manner.

d. Digital Control Approach

Sophisticated digital control schemes have been available to industry for some years now, and many processes have used them. For example, Model Algorithmic Control (MAC) has been successfully used for Superheater Control (10), On Line Control of Steam Generators (11), European Transonic Wind Tunnel Control (12), F-100 Jet Engine Multivariable Control (13), and Fossil Fueled Electrical Plant Control (14), etc. And although there have been some studies, no serious attempts has been made as yet to incorporate these latest techniques into U.S. nuclear plants. The reasons are multifold and entwined. First, the existing types of control work and have a long history of successful operation. Secondly they are licensable, so why ask for more regulatory hassle. (Early pioneers in attempting to incorporate rather simple digital control concepts in control and safety systems were held up for three years on the question of proving that the software would cover all conceivable situations.)

Thirdly, the market currently would not sustain large development costs for an industry that is not selling its product. And finally, there still exists some fear of the technique. "Would you trust your life to a digital computer?"

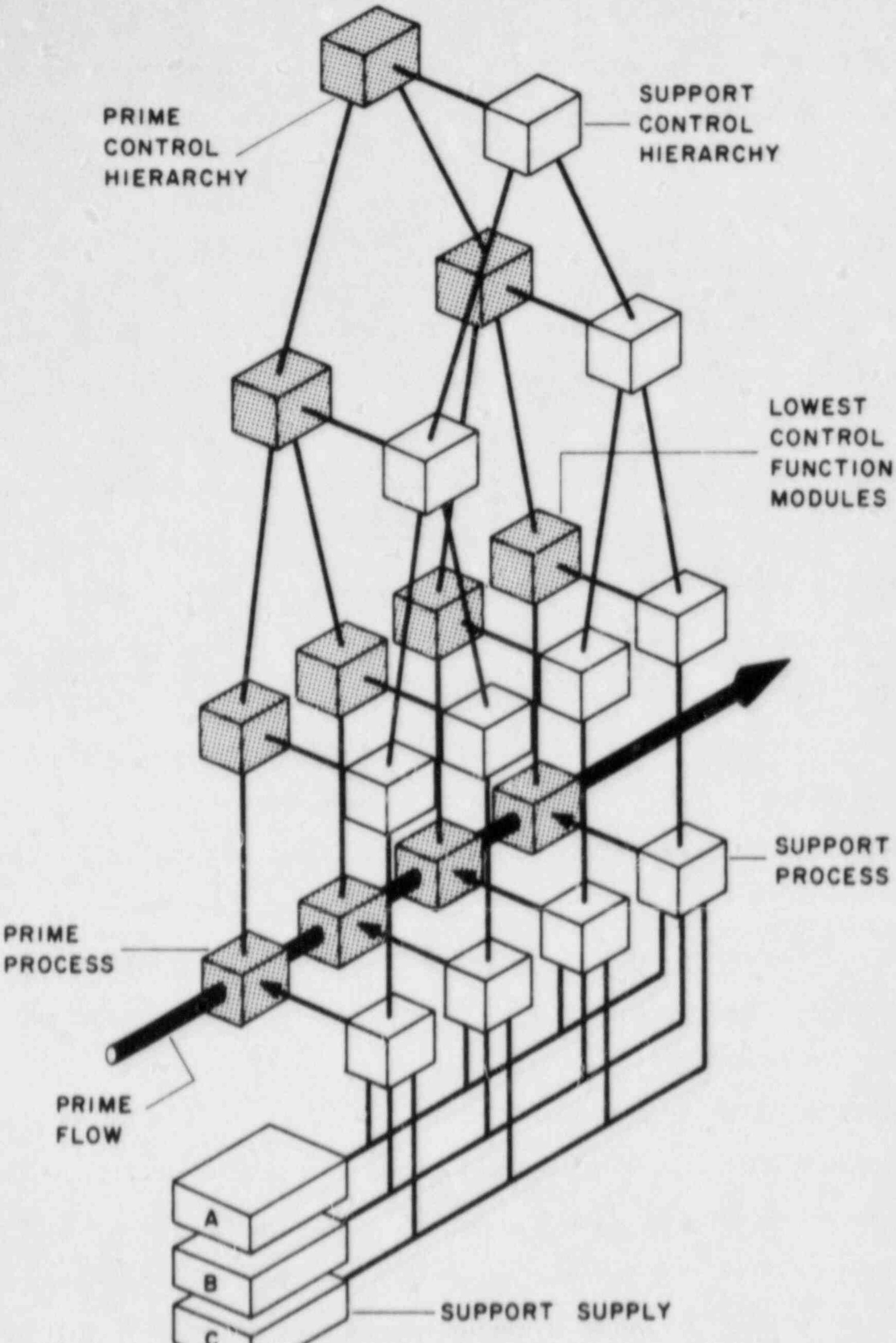
Work is currently going on at the Oak Ridge National Laboratory by J. Anderson and R. Kisner on a Structured Control approach that incorporates the latest control theories and would satisfy the requirements of the Steam-Cooled Reactor. From their work: "The classical approach to control system design in process industries involves mostly single input, single output (SISO) control algorithms. With this approach, a controller module is assigned to each major process component; however, global coordination of the controllers is often constrained because communication is local between a few controllers or altogether absent. Such communication, when present, is limited to discrete control signals (e.g., initiate, terminate, permit, and inhibit). The classical objectives for control and control system structure provide capability for normal operation but little flexibility for restructuring, as would be needed for degraded operation. In a similar fashion, operator displays have been limited to single variable readouts, or one sensor to one display design. Much of the limitations in the classical approach can be attributed to the limitations of analog technology. Contrast this with the modern or systems approach, which follows.

The systems approach to control system design involves the use of multiple input, multiple output (MIMO) algorithms as needed to accomplish the required functions. With this approach, a hierarchy of control modules is created for system-wide coordination of the entire process system, and rather than discrete control signals, data are passed between the modules. The resulting system is flexible and reconfigurable so that not only can complex optimizations be performed with the plant in normal conditions but also various stages of degradation of equipment can be accommodated while remaining on line. Operator displays can also follow the MIMO approach so that combined parameter displays can be provided that interact well with the operator's mental model and understanding of the plant. The expansion from the classical to the systems approach is a result of combining digital (computer) and analog technologies. A further refinement that adds structure to the systems approach can be made for the engineering of large-scale hierarchical control systems.

The structured approach is particularly suited for large-scale systems, where many groups of designers and analysts are interacting. The approach is used to impart consistency and commonality among design groups. To prevent the omission of less than obvious but required functions, categories of control functions and their associated data flows are provided to control engineers for creating a functionally oriented (rather than component or equipment oriented) system. The procedure for engineering a control system in this way is computer implementable, which can decrease analysis and design time and reduce communication errors."

The Anderson-Kisner approach envisions a hierarchal system similar to the process used in many biological systems. For example, to move an arm requires a local feedback system around the shoulder, elbow and wrist joints. These three local systems are in turn controlled by a group nerve hierarchy which is in turn coordinated by an overall brain directed body system function. In nuclear terms, the overall objective of producing power is broken down into a number of sub-functions each with its own local control system coordinated together by several levels of hierarchal objectives. Figure 21 indicates two structures, a prime process control and a support process control structure. Both of these structures are interconnected. It is presumed that an independent safety system structure is also present. Each block in the diagram can represent an optimal control, a multivariate control or simply an on-off switch. In the general case each block might be a small microcomputer element and the system is so structured as to be able to tolerate and command degraded operation if required.

A great deal more work must be accomplished both philosophically and technically before such a system would be available. What would be the level and type of redundancy required? What is the role of the operator in such a system? The best system is likely to be some mix of human and automatic control with a carefully thought out continuum at the interface. It seems time, however, for the greater portion of the supervisory control task to be assigned to automation and less to the man in the loop. With the unique self-protecting characteristics of the steam-cooled reactor, this seems like a good place to start injecting this type



CONTROL SYSTEM HIERARCHIES

FIGURE 21

of sophisticated control with the prime objective of producing safe power at high availability.

3. Shutdown

a. Normal shutdown

As has been previously indicated, the first step in a normal shutdown would be to turn off the blower and allow the reactor to flood. The control rods would also be inserted prior to any restart attempt. The shutdown heat removal depends on the type vessel chosen and the means of depressurization. By way of example, we select the reference system of Fig. 12 using a compound vessel. Here under normal conditions, the vessel would be cooled by a vessel blower system. The pump-pressurizer would be operative, and as soon as the steam pipes flooded, more water would be pumped into the main pressure vessel from the atmospheric tank. This new water would be somewhat cooler than the prior pressure vessel water. Several competing effects now occur. The reactor would be heating up the new and old mixed water toward the boiling point. The decay heat would be dropping fast. The water would be transferring some of the heat to the main vessel, which, in turn, would conduct this heat to its outside wall and be removed by air convection. Some heat would be removed from the walls of the atmospheric tank. It will take many minutes for a quasi-equilibrium to be set up and some local boiling may occur in parts of the core. The bubbles from the core may be suppressed by the cooler water at the top of the vessel. If the pressure builds up in the vessel it will be relieved by back-leakage through the pump and the gas would attempt to escape to the top of the atmospheric tank. As the decay heat decreases, and the heat transfer through the vessel walls is established, any boiling would stop and the system would be established at lower and lower equilibrium temperatures with time.

b. Abnormal shutdowns

(1) Failed Vessel Blower

In the event the vessel blower fails or is turned off, natural draft convection takes over, and as previously indicated, more approximately 25 MW can be removed. However, for the first few hours after shutdown, natural draft removal is inadequate and it is presumed that some bulk boiling will take place, the vessel pressure will rise and be relieved by steam blowing through the pressurizing pump leakage and escaping to the top of the atmospheric tank and into the overflow tank system. (See Fig. 24.) Now, depending on the geometry either a bubble will be formed at the top of the pressure vessel or the pressurizing pump will fill most of this void with water. In either case, the water level in the atmospheric tank would be expected to drop a few feet and be reestablished at its former level by the makeup system (See Section 3C). Even if the makeup system fails, the water level drop would be only a few feet before the natural draft forced a stable equilibrium level.

(2) Depressurization

Depressurization would occur by shutoff or failure of the connecting pressurizing pump. We will also assume failure of the vessel blower, as both these components are probably on the same power supply. Violent boiling will now occur and again the water level in the atmospheric tank will attempt to drop. Geometry of the connection between the pressure vessel and atmospheric tank now becomes crucial. If the geometry is such that a blocking bubble can occur in the main pressure vessel, the water level in this vessel can drop possibly 10-15 ft. (Not serious in a vessel 85 ft high with the core at the bottom). However, with a wide mouth vertical riser relief line and a carefully thought out pump containing no traps, it is inconceivable that some water will not be able to run down the outer rim of the pipe while steam is blowing up the center of the pipe. Thus the core section has available the water of the atmospheric tank, and the situation can only get better with time as the pressure is reduced and the decay heat slowly diminished. Obviously the dynamics are complex, but enough water would be available in the main vessel to withstand a few days of boiling even with failure of the vessel blower, and no make up available from the atmospheric tank. The beauty of the situation is that the situation is self-correcting even with power turned off.

c. Water Makeup Systems.

One of the advantages of the compound vessel is that makeup water can be supplied to the system at esentially atmospheric pressure. As it is highly desirable, but not essential, that water be held at above a minimum level in the atmospheric tank, several provisions would be made for supplying this water.

The first method would be a normal automatic system operating from a demineralized water tank via a conventional level gage and control valve. The automatic control valve would be backed up by a manual valving system. Thus the tank level would be kept between fixed limits.

A second demineralized water tank would be provided that operated via a mechanical float valve if the atmospheric vessel water level got down as far as, say, halfway down the main pressure vessel. This second water tank and piping would be underground and the float valve would be inside the second atmospheric tank. The float valve would be redundant and require no power.

A third line of defense would be the provision to inject raw city water, if the vessel level persisted in going lower than approximately three quarters of the way down. This system would probably be automatic as well as manually controlled with special provisions to prevent the automatic system from prematurely opening. And finally, the facility would own a fire engine complete with hoses and couplings to supply fire system raw water from prefereably a local pond or river via convenient hydrants. The fire engine hoses could also operate from the city water mains.

Thus with all of these provisions it is highly unlikely that the atmospheric tank would ever lose water. However, when we consider sabotage problems or severe seismic events we must assume that all external power to the plant is cut, all internally generated power is disabled, and all water mains, connections, etc. are broken. Hopefully, a saboteur won't find the demineralized water tank buried underground, but we can make the assumption that this tank has been discovered and blown up or drained. As long as the main pressure vessel remains intact we have no core melt down and sufficient water to handle several days of decay heat via natural draft and cooling. (And as will be shown later, even a pressure vessel rupture can be tolerated, particularly if power is available.

We continue the accident scenario now with a single vessel which may be the pressure vessel of the compound system or a large single vessel by itself.

d. Single Vessel--Abnormal Shutdown

We first consider a large undamaged single vessel by itself (PIUS equivalent water volume and plant rated output of 1800 Mwt) and will use the diagram of Fig. 9 as an example. Here six sets of passive cooling coils are shown for decay heat removal. Shutdown proceeds in a normal manner as previously described except that some conventional form of relief valve may be provided to enable blowdown to an external pool similar to PIUS.

The reactor is shutdown and flooded with all power turned off. The blower is not running and provisions have been made to depressurize passively with power off. (Actually PIUS depressurizes passively on low water level and over pressure.) If these provisions fail, the reactor decay heat creating increased system pressure will cause the system to depressurize by opening a spring loaded relief valve or a rupture disc.

The reactor is now sitting in the pool and boiling water at atmospheric pressure. The reactor power level is coming down on the normal decay heat curve, and the passive linear cooling system is presumed to be mostly disabled with only one out of 6 coils operative and thus removing 5 MW continuously. The water level has dropped 25 feet out of say 100 ft available in this tank. With pool water level at 25 feet below the normal surface level, the water temperature would begin to go below 212°F if the pool liner system would continue to operate as efficiently in steam as in water. This is not likely even though there is an excess of liner surface available for cooling. In preliminary scoping calculations it was assumed that the pool liner cooling system with steam cooling operated at roughly 20 percent of its normal efficiency and it continued to take out about 1 MW. There are also some conduction losses through the vessel walls (about 0.34 MW if all the vessel surface is considered available), but the scoping calculation considered only 1 MW of total cooling available. Under these conditions the water continues to boil and at the end of 40 days the vessel is dry and the decay heat is down to roughly 3.6 MW. Of course, if more than one passive cooler is operative, the vessel simply would not go dry for several months.

A more detailed look at the steam piping system is now called for. Figure 22 indicates the principal steam piping. It will be noted that two check valves have been placed in the steam return lines of his particular configuration. Their purpose is not for normal operation, but represent one way to seal off the vessel in the event of a steam line break external to the vessel. (Actually in the reference design we have elected to examine double wall piping.) Another valve has been added to the system in the steam return line. This is the gravity operated damper. This valve which may also be partially spring loaded, opens up when the pressure is removed from the system. It is assumed that this is a highly reliable high temperature valve with excellent

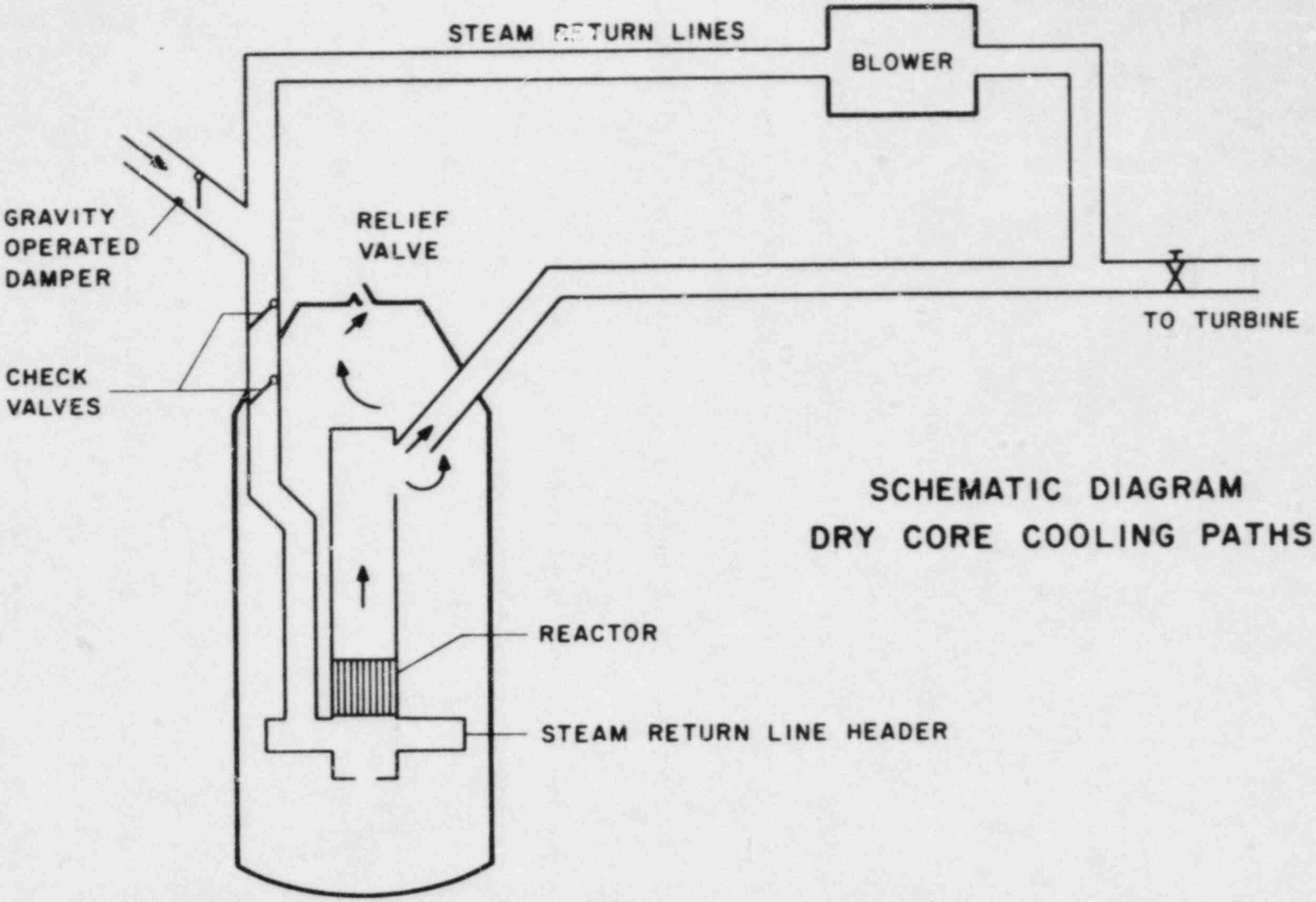

STEAM RETURN LINES
BLOWER
GRAVITY OPERATED DAMPER
RELIEF VALVE
TO TURBINE
CHECK VALVES
SCHEMATIC DIAGRAM
DRY CORE COOLING PATHS
REACTOR
STEAM RETURN LINE HEADER

FIGURE 22

sealed bearings. Under normal operating conditions the valve is shut by the pressure. The valve flap drops open when the pressure is greatly reduced. Other methods for automatically opening the system are also available. For example, the main turbine bypass line could be opened to the atmosphere or cleanup system upon loss of power.

So at shutdown dry conditions, an air flow path would exist as shown by the arrows in Fig. 22. Cool ambient temperature air enters at the damper from a protected air inlet, goes through the return line to the bottom header at the base of the shutdown box and is heated up going through the reactor. The warm air then exits the reactor and vessel via the relief valve. In any event, a free flowing natural convection system has been formed which is capable of removing a great deal of heat from the reactor.

There now are three principal avenues of removing heat from the dry reactor. These are, radiation to the cool air flowing through the core, direct convection to this flowing air, and radiation from the outer core edge to the walls of the vessel.

Scoping calculations were first made assuming that the core would be allowed to go up to its rated surface temperature of 1471°F. At this temperature it was found that if the core radiated to a cooler gas at an average temperature of 700°F, the radiation process alone could remove up to 170 MW (recall the core after 40 days shutdown is only putting out 3.6 MW). Similarly 6.85 MW of heat could be removed by convection using a heat transfer coefficient of only 1 BTU/hr ft^2°F. A further simplified conservative calculation indicated that the outside barrel of the core could also radiate 3.6 MW to the vessel wall.

As a result of these calculations, it would appear that in the actual situation of the reactor sitting dry 40 days after shutdown, that its temperature could be well below $400^{o}F$ and it would still dissipate its shutdown decay heat. Or alternatively the entire amount of water could be lost very quickly from the vessel, such as in a few hours, and the reactor fuel would not exceed its temperature rating. As it is difficult to conceive of a reactor vessel losing its water quickly (even setting up a huge pump would take a few hours) it appears that a dry reactor operating with some such air cooling scheme would sit indefinitely at a shutdown temperature well below its normal operating temperature.

We have partially considered the transition case of the core slowly going dry over the 30 to 40 day period. As previously indicated the reactivity is always negative. We have not examined the heat transfer aspects.

The type of design indicated above would be employed only if sabotage were an extremely important problem. For most normal catastrophies, the core would not go dry.

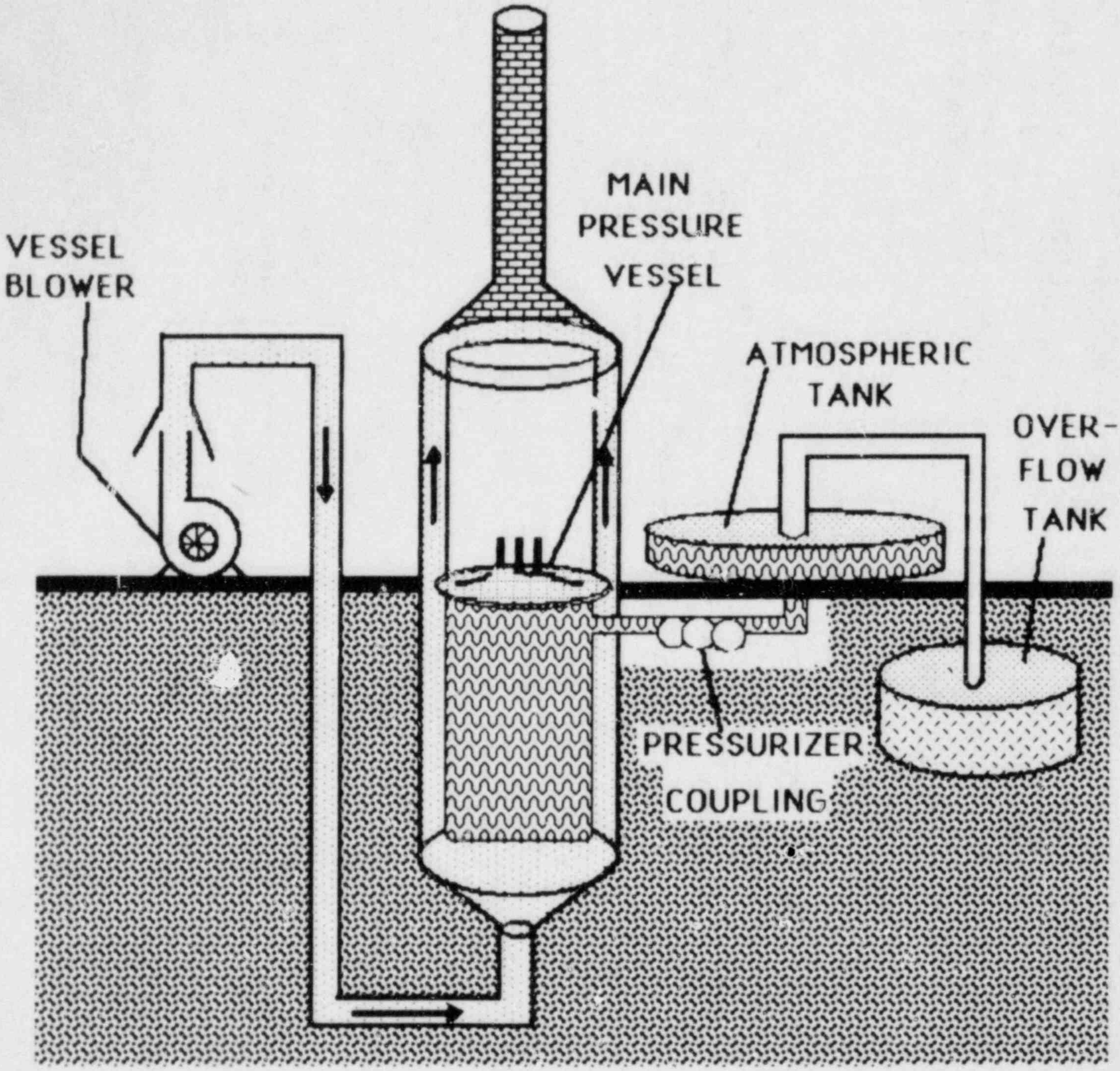
VI. SECONDARY CONTAINMENT AND CLEANUP SYSTEMS

The vessel cooling configurations of Figs. 7b and d lead to some interesting new concepts of handling secondary system confinement and fission product retention. It will be recalled that both of these vessels are made of steel, are mostly thin walled and are self-cooled by natural convection. As a start let us now add a blower to the natural draft intake of the vessel of Fig. 6d cooling system. And as has been indicated, natural draft is sufficient to adequately cool the vessel a few hours after shutdown. From a cooling standpoint it doesn't matter whether the blower is operative or not. Fig. 23 shows a simplified version of this portion of the concept. The purpose of the blower is simply to provide a large volume of air to be available during the first few hours and in the event radiation dilution is required.

The low pressure operation of the steam cooled reactor now provides the opportunity to examine double walled piping for the secondary system instead of the conventional single walled piping and containment vessel. With single walled piping the steam cooled reactor behaves similarly to the conventional BWR with respect to secondary pipe leaks or breaks, etc. The major difference is in the event of a secondary system break, the reactor would be flooded and only a small volume of radioactive steam would have to be dealt with. That is depending on the break location, the circulating reheat loop is presumed to be sealed off by the flooded reactor vessel and only the residual steam in the circulating line and turbine system would be released.

A double walled version is suggested by Fig. 24. Here the recirculating lines and all high pressure steam lines and systems are presumed to be double walled. The condenser is also indicated as being doubly encased, but this step may not be required.

The vessel blower now feeds air past the reactor vessel and atmosphere tank(s) and then up the stack as illustrated in the case of this compound vessel. The atmospheric tank is presumed to have a large overflow vessel associated with it and connections for partially emptying and refilling the atmospheric tank from the overflow vessel.



VESSEL COOLING AND PRESSURIZING SYSTEM

FIGURE 23

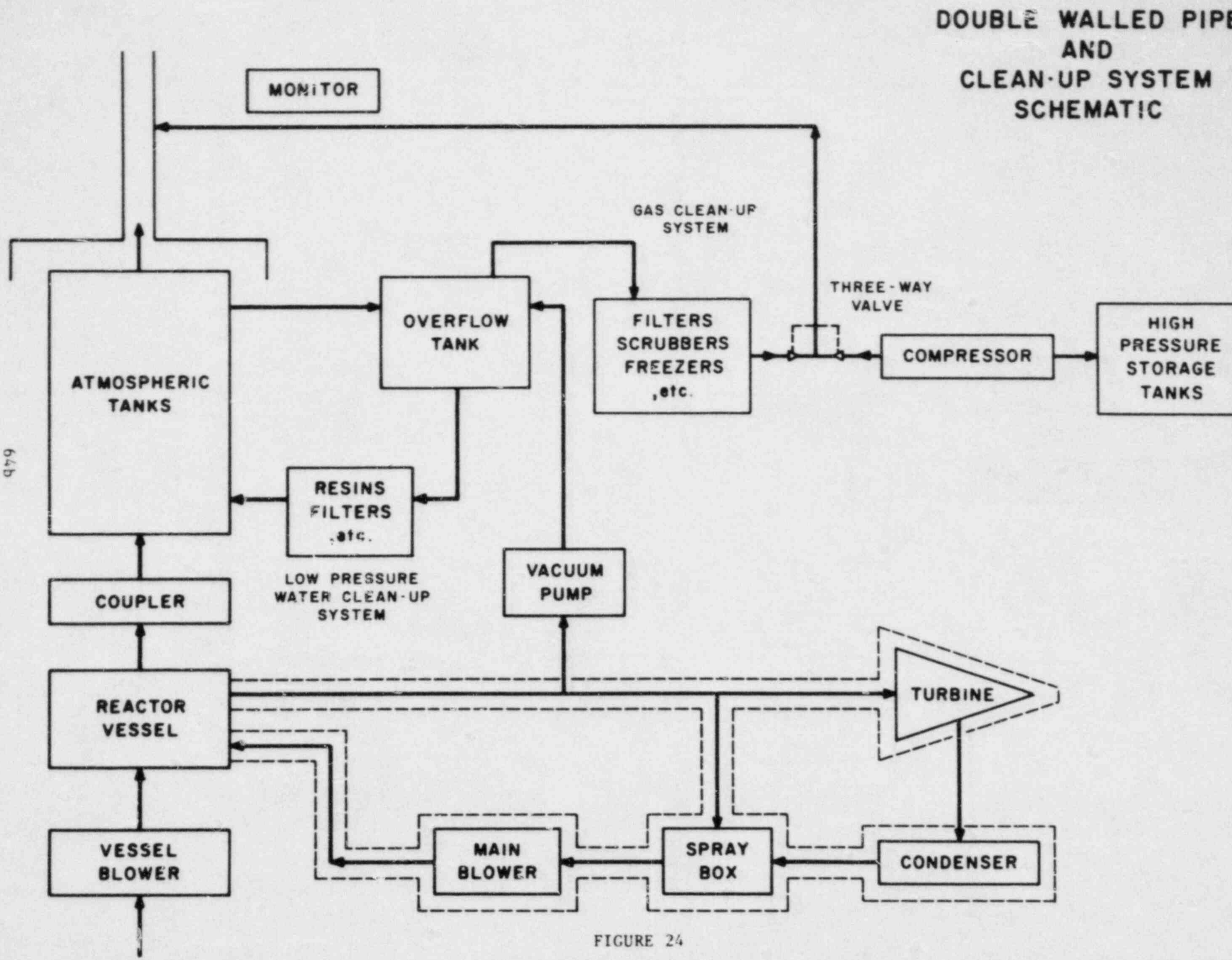

DOUBLE WALLED PIPE
AND
CLEAN-UP SYSTEM
SCHEMATIC
MONITOR
GAS CLEAN-UP
SYSTEM
THREE-WAY
VALVE
ATMOSPHERIC
TANKS
OVERFLOW
TANK
FILTERS
SCRUBBERS
FREEZERS
,etc.
COMPRESSOR
HIGH
PRESSURE
STORAGE
TANKS
RESINS
FILTERS
,etc.
LOW PRESSURE
WATER CLEAN-UP
SYSTEM
VACUUM
PUMP
COUPLER
REACTOR
VESSEL
TURBINE
VESSEL
BLOWER
MAIN
BLOWER
SPRAY
BOX
CONDENSER

FIGURE 24

The plant is now designed to have a complete on-line clean up system eminating from the overflow tank. Although the basic premise of this ultra-safe reactor implies that no meltdown is possible, it is still conceivable that defective fuel elements will be present. The leakage from these elements should be handled in a pre-thought out manner. In this plant all cleanup operations can be conducted at atmospheric pressure. In a normal PWR, for example, ion exchange fission product removal is normally conducted at high pressure. Here there is always some deliberate interchange of water between the reactor and atmospheric pressure tanks particularly at shutdown. Hence fission products in the atmospheric tank water from any cause may be removed continuously at low pressure. Resins also may be removed and replaced at low pressure even while the reactor is operating.

Radioactive gases similarly are transferred from the top of the overflow tank to the gaseous cleanup system and are processed by filters, scrubbers, freezers, etc. Radioactive iodine, cesium and other fission products will most likely be dissolved into the water by the large atmospheric tank volumes available. General Electric claims that bubbling these products through the suppression pool in the BWR-6 will remove over 99% of the iodine and particulate fission products from the vented gases (15). Krypton can also be frozen out in the cleanup system if desired. Any new technology that is developed can be employed in this series connection. Most of the noble gases have relatively short half-lives and in conventional PWR's the troublesome isotope is tritium, with a 13 year half-life. Tritium, however, is mostly generated from the borated water of the conventional plants. Here there is no borated water required and only a small amount of boron is encapsulated in the core structure. So tritium should not be much of a problem.

In any event after the gases are cleaned up by bubbling through the atmospheric pressure tank and by series systems they would normally be returned to the stack where the vessel blower would provide additional large dilutions before exit. In the event the stack radioactivity is too high to permit exit, the monitor will close the exit valve and retain the radioactivity in local removable high pressure tanks.

Secondary system leaks are handled in a similar manner. The space between the double walled pipes is continuously sucked out by a vacuum pump. This system provides two functions. First in normal operation, the vacuum created provides free pipe insulation. Secondly, in the event of a steam leak, the gas is channeled into the overflow tank where again it must pass through the extensive cleanup process before exiting.

For a larger inner pipe break where there conceivably might be water involved, good design would dictate the level of the various tanks and pipes such that the water would flow to the overflow tank by gravity.

As indicated above it may be possible to simply encase the secondary system in a double wall. The question then arises, as to what can be done about the primary system. From Fig. 5 it will be noted that a pipe break inside of the main pressure vessel does no harm in that such a break would only upset the pressure balance and cause reactor flooding. A break in the pressure vessel itself is more complex and depends on location and size.

First, from Fig. 23, it will be noted that the vessel system effectively has a wall around it that is open at the blower input and at the stack. A crack or break in this wall normally would permit ingress of ground water into the cooling space in that the water table would generally be far above the reactor level. A small break would cause moisture to run down the outer wall. This moisture probably would be evaporated by the flow of air past it from the blower. A huge break in the outer wall might cause the air cooling system to flood up to the water table line. As this line would be considerably higher than the reactor level, the blocked air passage would not permit air cooling, but instead the water in this passage would begin to boil from the vessel wall heat and this boiling could easily dissipate the reactor decay heat.

Similarly a large break in the pressure vessel or atmospheric tank would flood the air cooling space. Now the water level would depend on the break position. A break high in the atmospheric tank would only cause the water to run out to the break level. A break low in the atmospheric tank, or anywhere in the pressure vessel, would again cause the reactor vessel to be flooded both inside and outside and again the sytem would be cooled by boiling. A further discussion of a pressure vessel will be presented later.

Again the presumption is that no meltdown is possible and that minor amounts of gaseous fission product release would be handled by cleanup and air dilution.

There are obviously many questions and detail problems that arise in such a system. In particular, access to the secondary inner pipes and minor subsystems would have to be resolved. The point is, however, that a new look of this sort should provide a new system of continuous cleanup having far less impact on the public and on the plant operator in the event of minor mishap.

VII. PRELIMINARY REFERENCE DESIGN

1. General

Up until now we have presented a large number of concepts and alternate ways of exploiting and designing a steam-cooled reactor. In order to proceed with further analysis it now becomes necessary to choose a set of design conditions and configurations. At this stage the selection contains considerable subjectivity in that the selections have not been confirmed by technical or political analysis. An obvious example is the case of the reactor. The longest lived most economical reactor examined is a plutonium mixed oxide fueled breeder. Calculations have indicated a reactivity lifetime of greater that 200,000 MWD/tonne for one particular design. Yet, at the moment, the political climate for plutonium fueled breeders is unfavorable as we have plenty of uranium, and the perception of breeders is that they are unnecessary and possibly less safe. For these reasons we select a uranium fueled reactor as our initial prototype. We have calculations that indicate we might be able to design a reactor with a reactivity lifetime in excess of 150,000 MWD/tonne. Again we have no assurance that a fuel element can be obtained without extensive development that can meet this lifetime. As previously indicated, 100,000 MWD/tonne has been achieved in EBR and FFTF, so we assume their techniques are available and pragmatically select a 100,000 MWD/tonne reactor with somewhat high enrichment as our initial prototype. This reactor can be built today, even though we would prefer a longer lived more economical reactor that required some fuel element development. So our initial prototype reactor might be considered as a worst case.

2. Initial Prototype Reactor

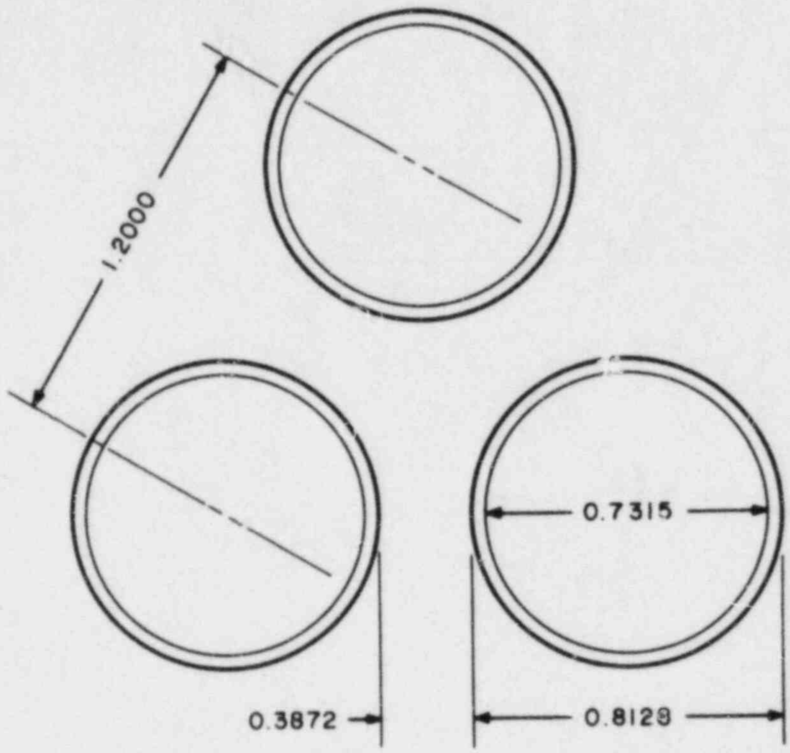
The first prototype reactor plant is to generate 3000 MWt. The lifetime of the core is limited by the materials of the cladding. The amount of excess reactivity required for 104,000 megawatt days/metric ton of heavy metal (MWD/MTHM) is only 0.032 plus a shutdown margin of say 0.02, which gives a total control rod reactivity worth of only 0.052. This reactivity will be obtained by B_4C rods.

The core dimensions, areas and volumes are identical for both the uranium and the late plutonium fueled cores. Thus, when plutonium becomes available, this reactor, if it were built, could be changed from a high conversion ratio reactor to a breeder reactor.

a. Core Dimensions and Layout

The core lattice is a hexagon and the equilateral triangle dimensions are given in Fig. 25. These elements are assembled into a hexagonal can of stainless steel as illustrated in Fig. 26. We will use the same technology as is being developed for liquid metal breeder reactors. The fuel rods are wire wrapped, to improve the heat transfer coefficient. There are 169 fuel rods in each assembly. One sixth of a core layout is given in Fig. 27. The fuel assemblies which have control rods within them will have 85 fuel rods, as well. There are 25 control rods which could control as

FUEL ELEMENT
TRIANGULAR LATTICE

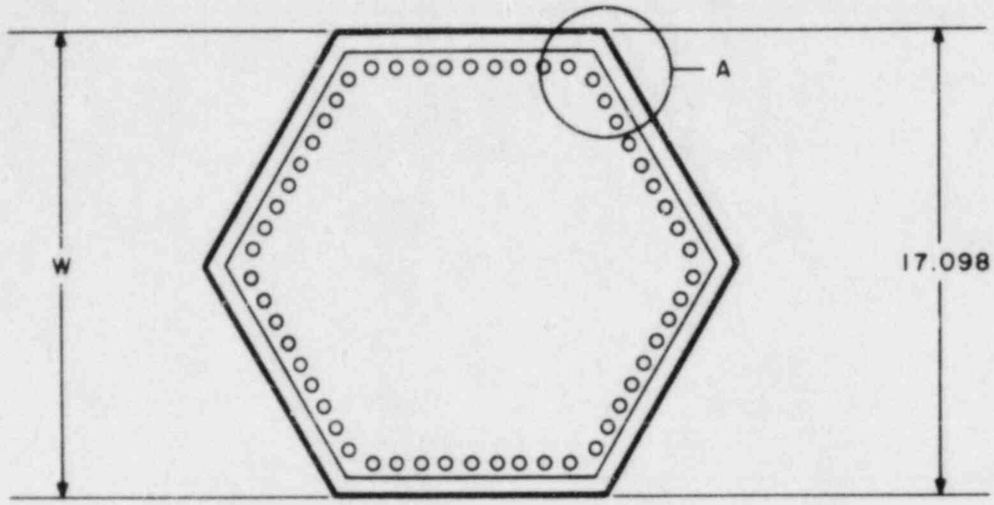


(dimensions in cm.)

FIGURE 25

HEXAGONAL FUEL ASSEMBLY

(dimensions in cm.)



$$W = \frac{\sqrt{3}}{2}\left[\,16 \times 1.200 + 1.20\,\right] + 0.40$$

$$W = 17.066$$

DETAIL A

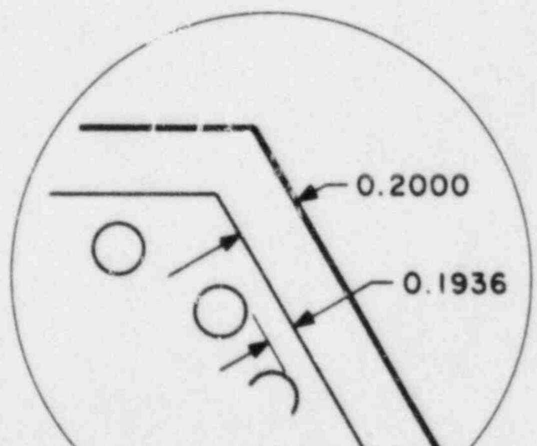


FIGURE 26

SCHEMATIC DIAGRAM
CORE LAYOUT

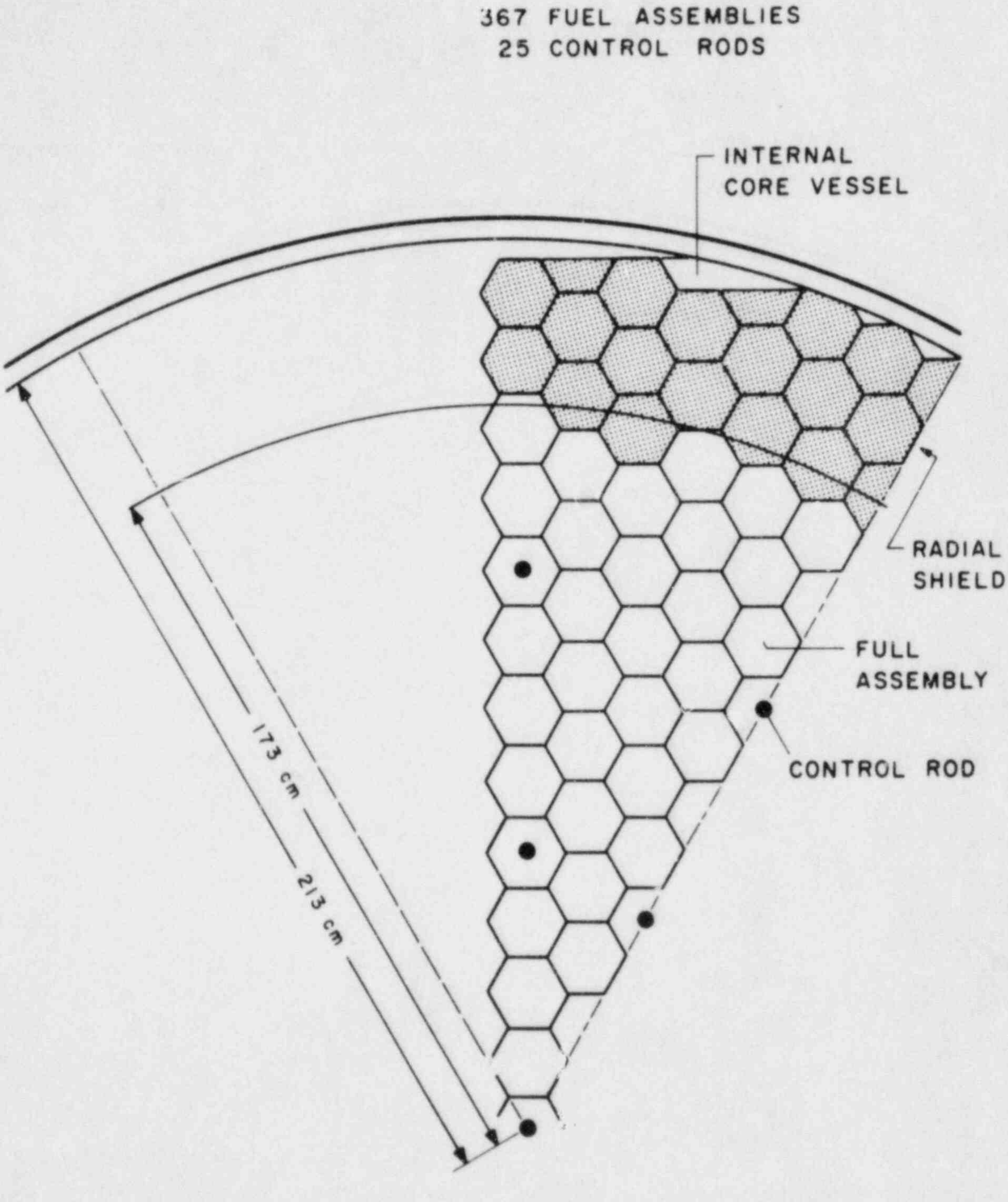


FIGURE 27

much as 12% in reactivity is required. A complete summation of dimensions and areas is given in Table 7.

b. Enriched Uranium Core Analysis

A 15 weight % enriched uranium core is calculated to give about 104,000 MWD/MTHM. The total uranium contained in the core is 60,000 kg and the thermal power is 3000 MW. Thus, assuming a capacity factor of 80%, a fuel reload will last about 7.12 years. Assuming that we reload 1/3 of the core at a time, the time between reloads is 2.37 years. This figure does not meet our original requirements, but recall this is a worst case.

TABLE 7

TABLE OF DIMENSIONS AND AREAS

DIMENSIONS	
Fuel Rod Outer Diameter	0.8128 cm
Fuel Rod Inner Diameter	0.7315 cm
Spacing between Fuel Rods	0.3872 cm
Pitch (Hexagonal Lattice)	1.2000 cm
HEXAGONAL LATTICE AREAS	
Cell	1.24704 cm^2
Total Fuel Rod	0.51887 cm^2
Inner Fuel Rod	0.42026 cm^2
Steam	0.69176 cm^2
Steel Wire Wraps	0.03641 cm^2
FUEL ASSEMBLY DIMENSIONS AND AREAS - using 169 Fuel Rods*	
Width from flat side to flat side	17.066 cm
Total Area	252.222 cm^2
Steam Area	149.512 cm^2
Total Fuel Rod Area	87.689 cm^2
Interior Area of Fuel Rods	71.024 cm^2
Area of Wire Wraps	6.153 cm^2
Area of Steel Can	8.868 cm^2
CORE DIMENSIONS AND AREAS - using 367 Fuel Assemblies	
Spacing between Assemblies	0.200 cm
Core Height	304.8 cm
Core Equivalent Radius	171.6 cm
Core Area	92565 cm^2

*Those assemblies which have control rods have 85 fuel rods.

The multiplication factor, k_{eff}, and the conversion ratio are shown as a function of burnup in Table 8. This calculation assumes the reload of 1/3 of a core every 2.37 years.

Table 8

Multiplication Factor and Conversion Ratio vs. Burnup

Burnup (1000^3MWD/MTHM)	k_{eff}	CR
0	1.0264	0.727
27	1.0202	0.757
53	1.0140	0.790
77	1.0075	0.820
104	1.0000	0.851

The relative variation of the U-235, Pu-239, Pu-240 and Pu-241 as a function of burnup are plotted in Fig. 28 and listed in Table 9.

Table 9

Relative Variation of U-235 and Pu Isotopes as a Function of Burnup

Burnup (1000^3MWD/MTHM)	U-235	Pu-239	Pu-240	Pu-241
0	0.9690	0	0	0
27	0.7406	0.1528	0.0065	0.0002
53	0.5651	0.2634	0.0023	0.0013
77	0.4308	0.3418	0.0440	0.0035
104	0.3222	0.4003	0.0697	0.0071

The relative neutron absorptions in fission products are given in Table 10.

Table 10

Ratio of Fission Product to Total Absorption as a Function of Burnup

Burnup (1000 MWD/MTHM)	$\Sigma_{FP}/\Sigma_{ATOT}$
27	0.0142
53	0.0270
77	0.0386
104	0.0510

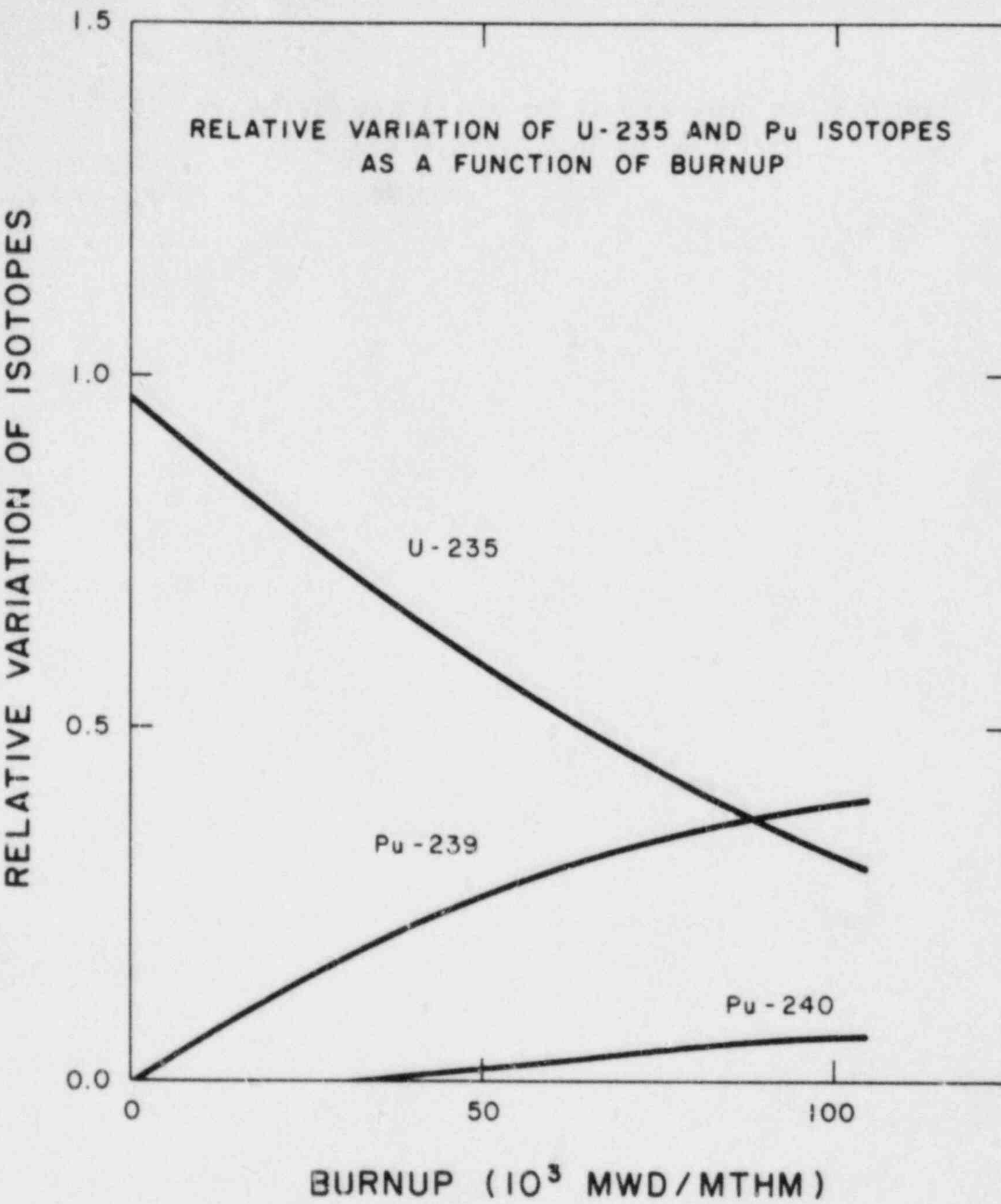

RELATIVE VARIATION OF U-235 AND Pu ISOTOPES
AS A FUNCTION OF BURNUP
U-235
Pu-239
Pu-240
RELATIVE VARIATION OF ISOTOPES
1.5
1.0
0.5
0.0
0
50
100
BURNUP (10³ MWD/MTHM)

FIGURE 28

The variation of k_{eff} with density created by voids is shown in Fig. 29. This calculation is important in that there are two situations in which the core may be partially voided. These are during startup and shutdown. It seems essential during shutdown that the core reactivity always be negative, regardless of whether the rods are in or out. During startup the rods would be in by administrative and interlocked controls. This calculation was made using the combinatorial properties of the VIM code. The calculation consists of a set of nested cylinders as illustrated in Fig. 30. Note that this is a three dimensional calculation. Each of the first two nested cylinders represent the core. The outside cylinder represents the radial, bottom, and upper shields. The inside cylinder can be voided and it's dimensions changed to simulate central voiding or partial core voiding. Complete core voiding also includes the shields. The results of the calculations are shown in Table 11.

Table 11

Void Coefficients

Portion of Core Voided	Δk_{eff}
Complete	-0.0200
Central 1.4%	-0.0016
Central 0.15%	-0.0020

It will be observed that the reactivity is negative for all cases.

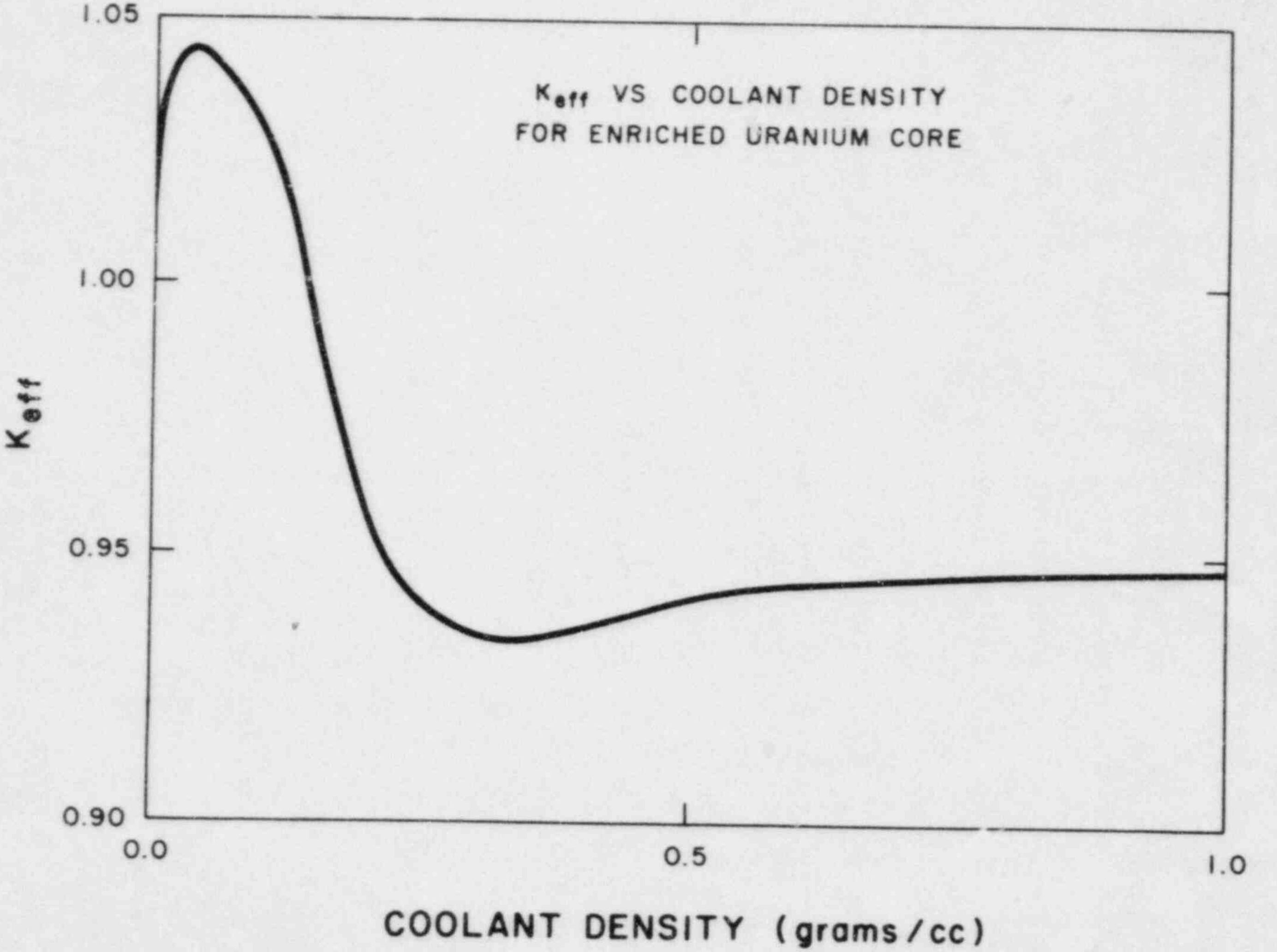
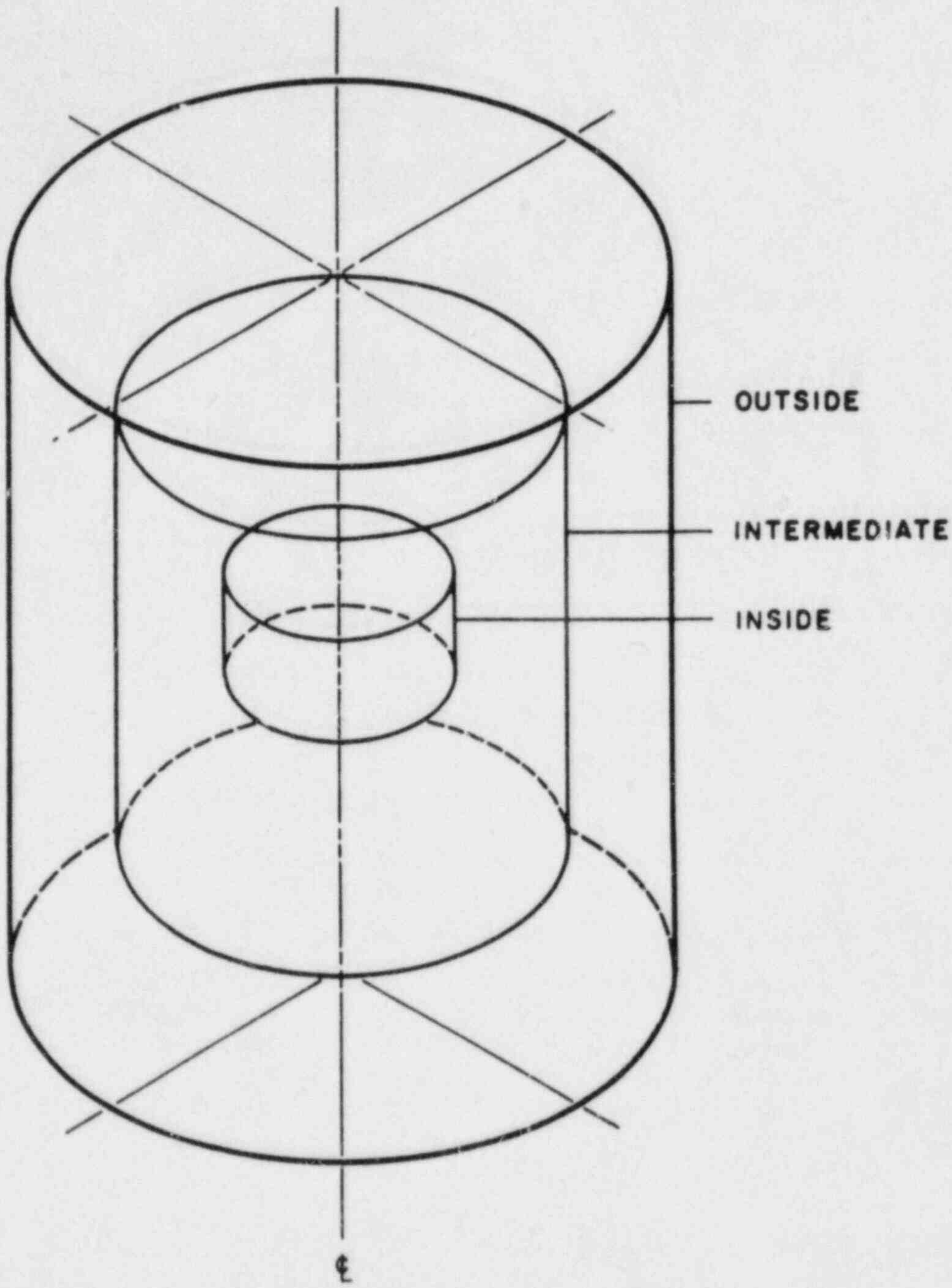

K$_{eff}$ VS COOLANT DENSITY
FOR ENRICHED URANIUM CORE
K$_{eff}$
1.05
1.00
0.95
0.90
0.0
0.5
1.0
COOLANT DENSITY (grams/cc)

FIGURE 29



LAYOUT OF NESTED CYLINDERS
USED IN VIM VOID CALCULATIONS

FIGURE 30

3. Thermal-Hydraulics Analyses

a. Introduction

The fast fluence (E>0.1 Mev) in the steam cooled reactor is 1.0 x 10^{23} neutron/cm^2 at an irradiation of 104,000 MWD/MTHM. From the data given in Waltar and Reynolds (16), it appears that the swelling of 20% cold worked 316 stainless steel is about 2.5% at a temperature of 450°C (842°F) at the neutron fluence. Furthermore, the irradiation creep of the same material below a temperature of 500°C is not affected by fluence. Since, 20% cold worked 316 stainless steel is considered to be the reference material for fast reactors operating with a maximum clad temperature of 620-650°C, we believe that backing off to a maximum clad temperature of about 500°C should certainly provide a good fuel element cladding.

Important information which is necessary to evaluate the thermal-hydraulics of this reference reactor are listed in Table 12.

Table 12

Hydraulic Diameter	0.0356 ft.
Velocity of the coolant	216 ft/sec
Density of the steam	1.91 lbm/ft^3
Number of fuel rods	59,923
Surface of fuel rods	50,200 ft^2

b. Average Steam Flows and Efficiency

Average steam flows in modern superheaters is in the range of 250-300 ft/sec. We are working with superheated steam at about 1000 psia and an average density of 1.91 lb-mass/ft^3. Thus, with an effective flow area of the steam of 57.1 ft^2 and a velocity of 216 ft/sec, the total mass flow through the core is,

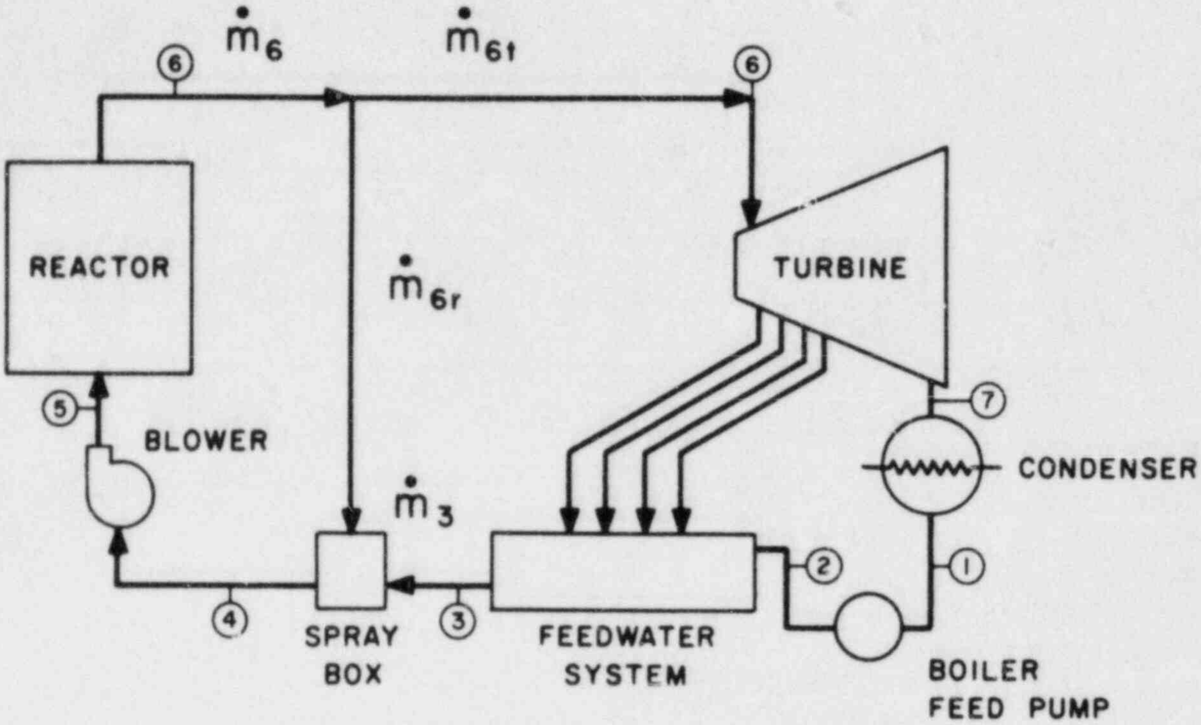
$$TMF = 1.91 \text{ lbm/ft}^3 \times 57.1 \text{ ft}^2 \times 216 \text{ ft/sec} \times 3600 \text{ sec/hr}$$
$$= 85.0 \times 10^6 \text{ lbm/hr.}$$

The average enthalpy coming into the reactor is 1210 BTU/lbm and steam leaves the reactor at $T = 700^oF$ and an enthalpy of 1325 BTU/lbm. Thus, the total enthalpy rise, or thermal power, is,

$$P(th) = 115 \text{ BTU/lbm} \times 85.0 \times 10^6 \text{ lbm/hr} = 9.78 \times 10^9 \text{BTU/hr}$$

The heat cycle is shown in Fig. 31. The return from the feedwater system is saturated liquid with an enthalpy of about 500 BTU/lb. In order that the return to the blower be pure vapor, we recycle steam from the reactor into the spray box. A typical summation of the parameters and the overall efficiency using feedwater heaters is shown Table 13. In this table the state positions refer to the numbered positions indicated in Fig. 31. The electrical power generated is about

$$E_p = 0.35 \times 9.78 \times 10^9 \text{BTU/hr/3413 BTU/hr/kwe}$$
$$= 1000 \text{ MW(e).}$$



$\dot{m}_6 = 85.0 \times 10^6$ lbm/hr $\leftarrow$ 85.0×10^6

$\dot{m}_{6t} = 13.7 \times 10^6$ lbm/hr $\leftarrow$ 13.7×10^6

$\dot{m}_{6r} = 71.3 \times 10^6$ lbm/hr $\leftarrow$ 71.3×10^6

THERMODYNAMIC CYCLE

FIGURE 31

Table 13

Typical Regenerative Rankine Cycle Calculation*

States	1	2	3	4	5	6	7
p (psia)	1	1090	1040	1000	1025	1000	1
T(^{o}F)	101.7	--	--	544	560	700	101.7
h (BTU/lbm)	69.7	74.7	499	1192	1210	1325	899
χ(quantity)	f	f	f	v	v	v	0.80
v(lbm/ft^3)	0.01614	--	--	--	--	--	--

* Efficiency of components: Turbine = 0.88, Boiler Feed Pump = 0.65, Blower = 0.91.

Recirculation ratio = 5.21

Overall efficiency = 35%

c. Heat Transfer Coefficient and Pressure Drop

First we need the equivalent fuel element diameter and the Reynolds number. The equivalent diameter is D_e = 0.0356 ft. The absolute viscosity is μ = 0.058 lbm/ft-hr. The density is 1.91 lbm/ft^3. The velocity is 216 ft/sec. The Reynolds number is therefore

$$Re = \frac{\rho V\ D_e}{\mu} = 0.912 \times 10^6.$$

The Dittus-Boelter equation for turbulent flow for smooth pipes modified by Weissman (17) for triangular rod bundles is,

$$h_s = 0.0324\ Re^{0.8}(c_p\mu)^{1/3}k^{2/3}/D_e,$$

where, h_s = heat transfer coefficient, BTU/hr-ft^2-^{o}F

c_p = specific heat = 0.76 BTU/lb-^{o}F

k = fluid thermal conductivity = 0.0304 BTU/hr-ft-^{o}F.

Therefore, for smooth pipes the heat transfer coefficient is,

$$h_s = 1946\ \text{BTU/hr-ft}^2\text{-}^o\text{F}.$$

The stainless steel wire wraps should augment the heat transfer by about a factor slightly higher than two according to A. P. Frass (4). Thus, the augmented heat transfer coefficient is about

$$h \cong 2\ h_s \cong 4000\ \text{BTU/hr}.$$

The pressure drop through the core for smooth tubes is,

$$\Delta P_c = f_s \rho V^2/2g_c \text{ where,}$$

ρ = mass density

V = velocity

g_c = lb mass-ft/lb force-sec^2

f_s = friction factor.

For smooth tubes, $f_s = 0.184\ Re^{-0.2} = 0.0118.$

Using the previous results we can obtain,

$$\Delta P_c = 4.36 \times 10^{-12} f_s \left(\frac{L}{D_e}\right) G^2 \ (\text{psi}) ,$$

where G is lbm/ft^2-hr and L is the length of the core in ft. Thus, L/De = 281 and $G = 1.485 \times 10^6\ lbm/ft^2$-hr and we find that,

$$\Delta P_c = 31.9 \text{ psi}.$$

We can now compare this smooth pressure drop with the wire wrapped element pressure drop using Rheme's method (18). The effective velocity depends on the rod bundle geometry and is given by

$$\left(V_{eff}/V\right)^2 = \left[\left(\frac{p}{d}\right)^{0.5} + \left[7.6\ \frac{d_s}{H}\left(\frac{p}{d}\right)^2\right]^{2.16}\right.$$

where:
- V = average fluid velocity
- p = pitch of the lattice
- d = outside diameter of fuel rods
- d_s = diameter of wire spacers
- H = helical-spacer wire pitch.

In this case, $p = 1.200$ cm, $d = 0.8128$ cm, $d_s = 0.3872$ cm and $H = 10$ cm and therefore,

$$V_{eff}/V = 1.175.$$

The modified Reynolds number is

$$Re' = \frac{\delta V_{eff} De}{\mu} = 1.07 \times 10^6,$$

and the modified friction factor is,

$$f' = \frac{64}{Re'} + \frac{1.07}{(Re')^{0.133}}$$

$$f' = 7.28 \times 10^{-5} + 0.0129 = 0.0129 .$$

Thus, the friction factor is increased by $f'/f_s = 1.09$ and, thus, the core pressure drop is about

$$\Delta p_c = 34.7 \text{ psi}.$$

Hot Channel Analyses

Our physics calculations show that the peak to average fluxes normal to the steam flow are remarkably constant. Over a core lifetime of 104,000 MWD/MTHM the variation is less than 10%. Thus, with fixed orificing for each fuel assembly we will assume, that the overall peak to average enthalpy change in each steam channel can be held to 1.20 or less.

The average enthalpy change as indicated above is 1375-1210 = 115 BTU/lb. and the average exit temperature is 700°C. The hot channel exit temperature is 1.20 x 115 + 1210 = 1348 BTU/lb, and the hot channel has an exit temperature of 734°F. The average heat flux is 9.78×10^9 BTU/hr/50,200 ft^2 = 195,000 BTU/hr-ft^2. The maximum heat flux in the core is about 500,000 BTU/hr-ft^2.

The temperature drop (T_w-T_B) between the tube wall and the bulk temperature of the steam is at most,

$$T_w - T_B = q''_{max}/h = 125^{\circ}F.$$

The temperature drop through the clad ($T_c - T_w$) is at most,

$$T_c - T_w = q''_{max} \quad \Delta t/k,$$

where Δt is the thickness and k the thermal conductivity of the clad. Using Δt = 0.00125 ft and k = 11 BTU/hr-ft^2, we find that,

$$T_c - T_w = 47^{\circ}F.$$

Thus, the maximum clad temperature is less than, 734°F + 124°F + 47°F = 906°F (486°C).

A summary of these calculations is given in Table 14.

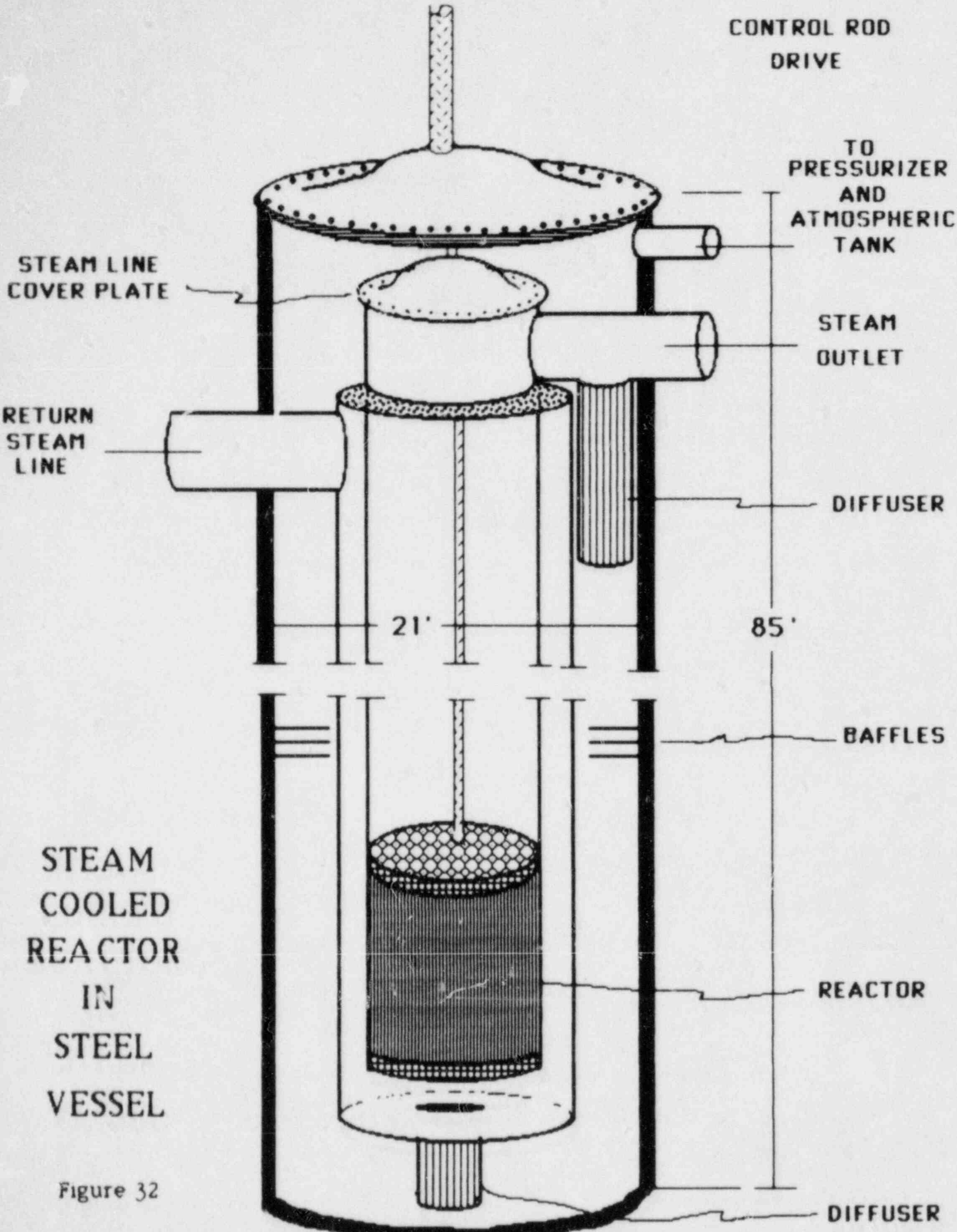
Table 14

Summary of Thermal-Hydraulics

Average linear power, k_w/ft	4.78
Peak linear power, k_w/ft	12.4
Average heat flux, BTU/hr-ft^2	195,000
Peak heat flux, BTU/hr-ft^2	500,000
Heat transfer coefficient of the steam, BTU/hr-ft^2 ^{o}F	4000
Pressure drop through the core, psia	34.7
Inlet temperature, ^{o}F	560
Average exit temperature, ^{o}F	700
Maximum clad temperature, ^{o}F	900

4. Pressure Vessel System

For the reference system we have selected a compound steel vessel system similar in concept to that shown in Fig. 23. The high pressure section is as shown in Fig. 32 and is roughly the size of a conventional BWR vessel being 85 ft high and 21 $^1/_2$ ft in diameter. However, the wall thickness to meet code requirements at 1000 psi is only 3.2 inches. We are suggesting the use of a clad vessel consisting of 302A Grade carbon steel clad with type 304 stainless steel, using conventional techniques. The vessel is placed underground with its cover roughly at ground level. This vessel connects to a single atmospheric tank of undetermined dimensions whose bottom again is roughly at ground level. The high pressure section has an outer lining approximately 6" away from its outer surface and has a natural draft and powered cooling system



STEAM COOLED REACTOR IN STEEL VESSEL

Figure 32

similar to Fig. 23. The double walled piping and cleanup system of Fig. 24 has also been selected.

Inside the vessel of Fig. 32, the steam pipes are indicated as being coaxial. The return steam enters the vessel in a 4.4 ft diameter pipe or its equivalent. That is, it may be desirable to bring in multiple smaller pipes into the return line annular header, and if two or more pipes are used they should be sized to produce the same area as the 4.4 ft pipe. From a simplicity point of view one large pipe is preferred. The steam enters the header and flows down the core barrel shell, then up through the core and into a similar exit header. For ease of pipe runs both the inlet and outlet pipes should be on the same side of the vessel. (See Fig. 33.) They are shown here on opposite sides of the vessel for clarity.

In both the incoming and outgoing pipe sections are the diffusers which provide the normal interface between the steam and water. Section AA indicates that the diffusers consist of a large number of small diameter pipes to prevent side to side bubbling effects. The diffusers also provide room to take up some error or changes in pressure drop across the core with time. They also furnish a required pressure drop during transients. The overall diameter of the diffuser section determines the rate of reactor flooding at shutdown. For our initial reference we will size the opening to cause complete flooding in 10 seconds.

It will be noted that the steam pipes have no insulation inside of the vessel. In this manner the water temperature in the vessel can rise to nearly the steam temperature. This situation eliminates the thermal shock problem that might occur if cold water were injected into the core at every shutdown. Of course, expensive metallic insulation is also eliminated. (Estimated at $1 million for PIUS.) The higher temperature water also makes it easier to transfer out shutdown heat by natural draft. The disadvantage is that there is more stored energy in the vessel water that makes it easier to boil off water. It is believed that the relief system coupled with the atmospheric tank overcomes this difficulty.

Once the main pressure vessel head is removed, it is necessary to also remove the steam line cover plate in order to get to the core. As the pressure inside and outside the steam lines in the vessel is roughly the same, the steam pipes and cover plate can be made of relatively thin material. (Probably being $^1/_2$ to 1" thick depending on structural considerations.) There is also some small pressure difference during startup and during transients that must be considered in the design.

A control rod shaft is also indicated in Fig. 32. This shaft must pass through a seal in the steam-line cover plate. Again because of the small pressure difference, these seals can be rudimentary and even then a small amount of leakage does no harm. No thought has as yet been given to ganging control rods, but at most only 25 such shafts and seals would be required.

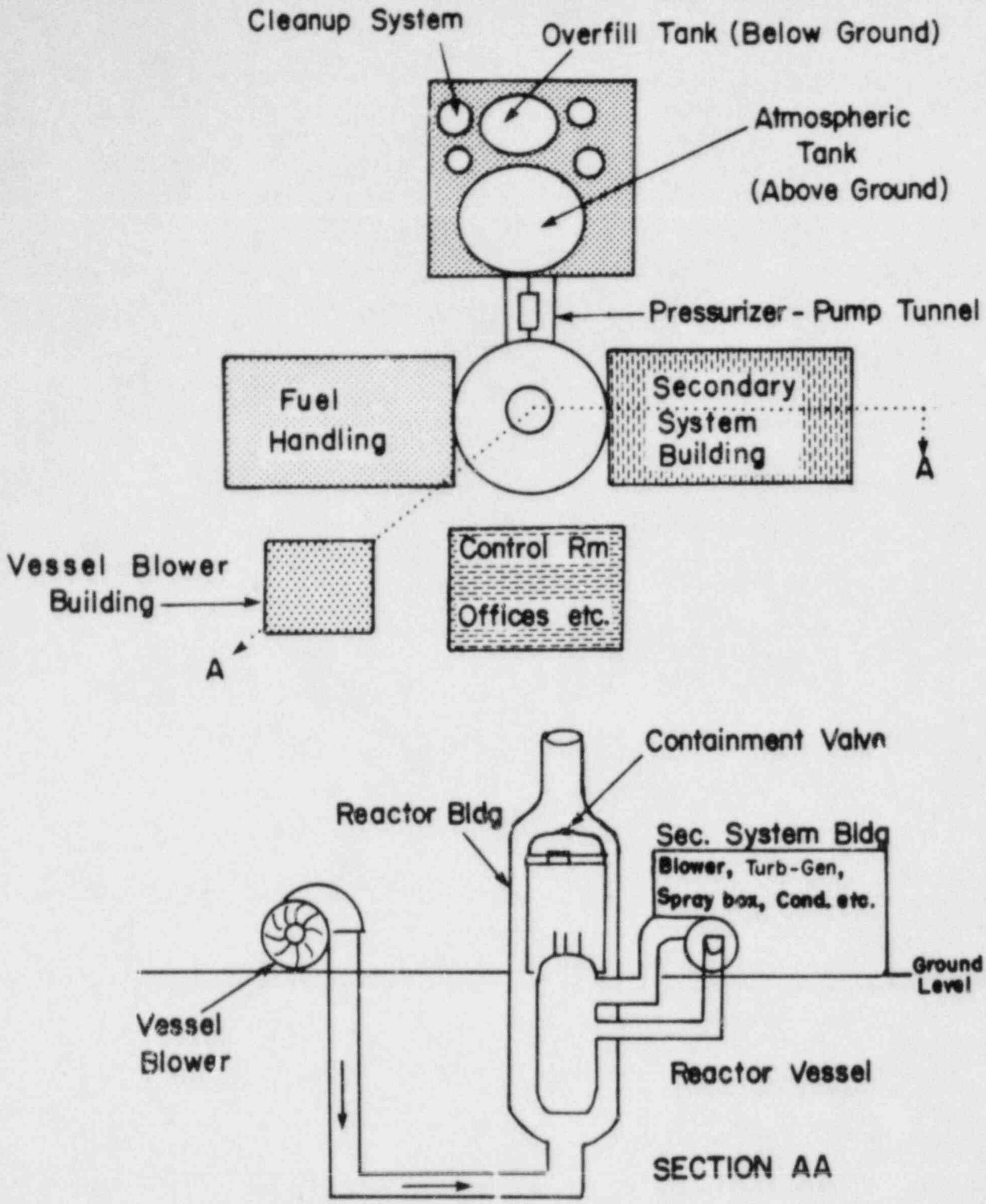
Fig. 32 also indicates a conventional control rod drive penetration through the main head. This drive could be an existing magnetic jack or linear motor, etc. Once the head was removed, the control rod shaft would be exposed and it is presumed this shaft would be in one or more sections as suggested by the figure. If this step were not taken the reactor building would have to be roughly 85 ft high and shaft handling would certainly be awkward. As is, some form or open lattice control rod support structure will probably be required inside the steam pipe to support these long multi-section shafts.

Another feature shown in the drawing are a series of baffle plates that may be needed during startup. The blow out startup technique suggested calls for successive gas pressure changes. Without some damping, oscillations might possibly be set up. The baffles simply indicate one way of providing this damping.

5. Plant Layout

A preliminary suggested building layout is shown in Fig. 33. The layout indicates the reactor in the center of a four main building cluster. Cooling towers and other auxiliary buildings are not shown. The reactor building is seen as being wrapped around by a chimney structure. From the sectional view it can be seen that main steam pipe runs are quite short and the amount of double walled large piping appears short. An interesting design arises if the coaxial piping is carried from the reactor to the blower. The blower then becomes part of the coaxial system.

In any event the layout indicates the below ground main pressure vessel and the above ground atmospheric tank connected by a pressurizer pipe tunnel in which would be mounted the pressurizer pump, providing



PRELIMINARY SUGGESTED BUILDING LAYOUT

FIGURE 33

for easy pump maintenance. The reactor building is, of course, part of the vessel cooling system, effectively providing the chimney for the natural draft. Provisions would be made to either open the reactor containment building to the chimney or close it off via a conventional containment vessel valve. Later studies may also indicate the desirability of increasing the blower elevation and making the atmospheric tank a large diameter shallow tank. This step might be taken after a study of all potential pipe breaks. The object is to prevent the possibility of ever flooding the blower. The same result could be obtained by valving.

An interesting solution is now seen in answer to the question, what happens if the main pressure vessel cracks or springs a large leak. From Fig. 33, a small leak emitting only a dribble of water would probably not depressurize the system. The water would be evaporated into the vessel blower airstream. A large break, however, would depressurize and could be expected to fill up the space between the vessel and the blower, particularly if the pressurizer pump remained operative. Whether the pump were operative or not, the contents of the atmospheric tank (and its make-up system) would be expected to ultimately fill the space between the vessel and its liner. The water would exit at ground level through the vessel blower. As long as water were flowing, there is more than enough surface area in the vessel to dissipate the decay heat. And even if the makeup system eventually runs out, (hardly likely with a connection to city and fire water mains) boiling will keep the vessel cool for a long time and the natural drafts would ultimately take over.

The design would now call for a drain in a sealed blower house to be connected back to the overflow tank of the waste disposal system so that no potentially contaminated water leaves the system. And with this addition to the makeup system, the probability of going dry becomes even more remote.

VIII. RELIABILITY AND MAINTAINABILITY

It is quite impossible to calculate the reliability of a concept without detailed design. However, several comments might be made with respect to the steam-cooled reactor preliminary prototype concept. First, reliability for systems with redundant components has been shown (19) to depend only on the mean time to repair (MTTR) and in today's reactor designs most key components can be expected to be in redundant configurations. In the steam-cooled reactor, only a minimum number of components are inaccessible on a day-to-day basis. Not including the core, only the control rod steam pipe cover feed through bearings are in the vessel and unavailable for immediate maintenance. Most of the auxiliary systems are at atmospheric pressure and consequently the MTTR should be lower for the steam cooled reactor.

In analyzing the reliability of a plant, several tools may be used. Usually some form of reliability analysis such as fault tree and event tree analysis are undertaken simultaneously to form a risk analysis. The event trees are obtained from some form of Failure Modes and Effects Analysis (FMEA). It has been found that the largest payoff is reliability and safety results when FMEA's are conducted at all

Table 15
Failure Mode and Effects Analysis
Steam Cooled Reactor Prototype Concept

Component or System	Failure Mode	Result	Comments or Fix
1. Main Blower	a. Fails On - Keeps running	a. Nothing at reactor power. If reactor is shutdown and flooded blower should not be able to suck in water to its impeller.	a. Either the pipe elevations should be high enough not permit water into the blower from reactor, or traps should be installed in steam lines. Make-up pumps should also be limited in capacity to prevent blower from sucking water from spray box.
	b. Fails off (1) Loss of power		
	(2) Impeller Blocked by foreign object	b. Normal method of reactor flooding	(2) Blower protected by breakers
2. Vessel	a. Fails on	a. Nothing in normal operation vessel continues to be cooled	a. Power can eventually be cut by using breakers closer to power source
	b. Fails off	b. Nothing in either normal or shutdown operation	b. Natural draft cooling takes over and although temperatures would rise somewhat no harm would be done.
	c. Chimney path blocked	c. Temperature of reactor vessel water would rise, ultimately increasing pressure and blowing off steam to atmospheric tank	c. It's very difficult to completely block entire chimney area except by deliberate sabotage. If accomplished, however, an active scram on high temperature water would shut down reactor and depressurize. Water would boil off for several days, but means are provided to automatically and manually add new water. Some radiation and conduction cooling would also occur and limit temperatures.
3. Steam Pipes	A main steam pipe breaks		
	a. Inside Vessel (1) Return steam line	(1) Pressure balance upset, reactor floods	(1) Reactor shutdown, normal cooling occurs
	(2) Steam Outlet line	(2) Same	(2) Same

	(3) Outside Vessel (Coaxial Plumbing)		
	(a) Inner pipe	(3)(a) The core becomes partially bypassed. Core temperature rises.	(3)(a) Core flooded by active high temperature system.
	(b) Middle pipe	(b) Possibly nothing	(b) The outer pipe takes over the role of the middle pipe. Temperatures in the cleanup system rise sharply.
	(c) Outer pipe	(c) Nothing happens reactor system	(c) The cleanup system is partially diabled, indicate by a loss of vacuum.
4. Pressurizer Pump	1. Fails to rotate	1. System loses pressure and starts to blow down to atmospheric tank	1. Reactor starts to shut down because control system cannot maintain correct quality steam. Reactor flood by turning off blower on high temperature and/or low pressure.
	2. Continues to rotate	2. System remains pressurized	2. No immediate problem either at power or shutdown
5. Spray Box Control Valve	1. Fails to open	1. Steam density higher than optimum	1. Reactor power level is steadily reduced because coolant density is too high.
	2. Fails closed	2. Steam density lower than optimum	2. Reactor power level is steadily reduced because coolant density is too low.
6. Atmospheric Tank Water Injection Systems	6. All systems fail to inject new water into tank at shutdown	6. Tank water slowly boils on sequences described in section V 3c	6. Many weeks are involved in the process providing time to bring in external water supplies if required. Reactor remains undamaged even if new water not supplied.
7. Condensate Makeup Water Systems	Supplies either too much or too little water	Same as item 5	Same as item 5. Feed water pumps will be sized so as to prevent water from getting into blower.
8. Control Rods and Drives	8. Fail to move (a) A full out position at power	(a) Normal Operation	(a) Not noticed as long as reactor is at power
	(b) At full out position at startup	(b) Nothing	(b) Reactor sequencing interlocks would prevent react startup unless rods were in core
	(c) At full in or intermediate position at power or startup	(c) Nothing	(c) Reactor cannot be made critical at any position other than essentially full out

9. Turbine	a. Breaks blades or jammed so that it cannot rotate	Steam lost from system via or bypass valves	No immediate effect on reactor. Turbine shutdown signal will ultimately produce reactor scram.
10. External Steam Generator	9. Unavailable for any reason		
	a. During power operation	Nothing	a. External steam generator not needed at power.
	b. At startup	b. Either cannot start up plant or plant can be started on gamma heat created steam.	b. There may be sufficient steam available from reactor bottom grid plate heating to start up reactor.
11. Gas Pressurizer Secondary System	Fails to deliver pressure	Water cannot be blown out of steam pipes	Reactor cannot be started.
12. Control System Computer	Fails in either direction	Reactor cannot be kept at power	Computer would be designed to fail calling for either too high or too low density steam
13. Nuclear Instrumentation System	Fails a. Power level channels fail--reading high	a. Reactor flooded on high level neutron signal.	a. Similar to scramming procedure in conventional reactor.
	b. Power level channels fail or reac low	b. If attempts are made to raise power level reactor shuts off because of improper coolant density.	b. Redundant safety channels should also catch this event.
14. Vital power bus	14. Loss of power	14. Reactor shuts down	14. Blower and pressurizer pump are on this line.
15. Crack in Reactor Vessel	15. Earthquake or sabotage	15. Water runs out of vessel into cooling s ce between vessel and liner	15. Water level equilibrates we'l above core level. If pressurizer pump 's on or off the atmospheric tank should dump its water into the cooling space and run out the vessel blower inlet. Reactor is protected by boiling at outside vessel walls and new water can be added at atmospheric tank.

stages of the design, from concept to detailed components. The analysis is refined at each stage as more and more is known about the systems and components. We are in a position on the basis of the material presented in this write-up to conduct a limited FMEA on the major obvious components of the prototype steam-cooled reactor. Table 15 shows such an analysis.

The FMEA points up the desirability of adding several small features to the plant for additional protection. For example, active scram (flooding) signals should be provided for vessel overpressure and for high temperature water in the pressure vessel dome. The FMEA of Table 15 is indicated for mostly single failures. Some double failure analysis has been accomplished, but not presented. In any event the plant pretty much protects the reactor against component failures, and only a limited safety system appears desirable mostly to protect other components such as the main blower.

IX. ECONOMICS

1. Capital Costs

It also appears impossible to obtain absolute capital costs on a preliminary concept that has not yet reached the design stage. Moreover, the capital costs of two similar nuclear plants located at different sites have varied by a factor of 3 to 1, because of local problems, labor problems, regulation induced changes during construction, etc. Also, capital costs have escalated extremely rapidly over the last ten years in the U.S. and there is no method of predicting how far they will go in the next 30 years. So obtaining an actual dollar cost in this situation, for a preliminary concept, is hopeless.

Therefore a more reasonable approach might be to make direct comparisons with existing or detailed designed plants and see what variations from these "normal" plants exist in the steam-cooled reactor. In this way a feel for relative costs may be obtained, and a determination made as to whether the new plant would be economic with respect to the older ones.

In this case two simple comparisons can be made, first with the ASEA-ATOM designed PIUS which contains a number of common items with the steam-cooled reactor, and second with a conventional BWR which also has some commonality.

PIUS has already been compared with a conventional PWR by Hannerz and the conclusion reached that on a scaled-down 500 MWe plant, the $/Mw captial costs were slightly lower for PIUS without a secondary containment building for the reactor system. For their plant, where all large primary components are buried in the concrete pressure vessel, Hannerz believes this is sufficient containment. Actual figures are confidential as of this date and nothing but the above conclusion has been publicly released.

On the other hand, if a containment building is required, the conventional PWR apparently becomes less expensive than PIUS. The steam cooled reactor with double walled piping might also well negate the need for conventional building containment. The necessity for a complete containment building undoubtedly would depend on the regulatory and political climates at the time the plants were built. We now also make the assumption that BWR's and PWR's are competitive cost-wise in that both types have been sold to similar customers.

It, therefore, becomes useful to crudely compare the steam-cooled reactor against PIUS. Table 16, showing this comparison, is simplified in that it assumes that the two plants are essentially the same except for the reactor and its immediate appurtenances.

There are, of course, other differences, but again it seems reasonable to assume that once development costs are completed there is a very good possibility that the steam-cooled breeder would cost less than PIUS, because of the lower vessel cost. And PIUS, according to its designers, should cost less than a conventional PWR or BWR plant. In any event all four plants appear to be in the same cost ballpark with no startling price differences apparent at this time.

The other comparison to be made in more detail is directly with a conventional BWR. A conventional BWR today contains 113 generic subsystems grouped into 7 major categories. (20) These categories are:

N. Nuclear Systems

S. Engineered Safety Systems

C. Containment Systems

E. Electrical Systems

P. Power Conversion Systems

W. Process Auxiliary Systems

X. Plant Auxiliary Systems

Table 16 STEAM-COOLED REACTOR COMPARISON WITH PIUS	
Steam-Cooled Reactor Deleted Items from PIUS	Steam-Cooled Breeder Counter Balancing Items
1. Fewer Control Rods and Drives	1. Added Tankage for fill and cleanup
2. No large primary pumps.	2. Somewhat larger internal liner
3. No large primary heat exchangers.	3. Addition of steam blower and special valving should not quite match-up cost-wise against reactor in-vessel primary loop components.
4. No electronic variable speed primary pump controls.	4. More complex control and startup system
5. No internal piping insulation	
6. No boron injection and cleanup system.	
7. No plant chemical building.	

It is clear that all of these systems are not of equal importance or equal cost. A simplified comparison is made in Appendix D of the principal items on a subsystem basis between the conventional BWR and the proposed steam-cooled reactor.

And again, this comparison shows several unpriced differences between the two plants, but it appears as though the bulk of the auxiliary systems would be essentially the same.

The real hope for capital cost reduction in the steam cooled reactor comes from two sources. First the lower pressure operation significantly reduces the amount of steel required by the plant. A partial counterbalance is in the added cost of a somewhat larger low pressure turbine. If we estimate that the plant costs are on a per pound of steel basis, cutting the operating pressure from 1500 to 1000 psi first reduces the steel cost crudely by 1.5. If the hardware cost of a conventional 1000 MW BWR plant is estimated at $\$500 \times 10^6$ then the savings might be $\$167 \times 10^6$, so there is considerable margin available for buying a larger turbine.

The larger factor appears to be the long construction time presently required for nuclear plants in this country. This long time period creates enormous interest charges and the opportunity for backfitting. Shorter construction time can result from efficient management (Note St. Lucie II), standardization and relaxation of regulations. If the conclusion is reached by NRC that a suitable economic ultra-safe reactor can be built, then, theoretically, they should require that no other type be built. They might then standardize in the extreme on one type of ultra-safe reactor. Their regulations should then be thoroughly reexamined in the light of the new plant and simplified. (It is our understanding that a preliminary study is indeed currently underway.) If the public, after proof and demonstration that the new plants are indeed far safer than existing plants, is convinced, protest and intervenor caused delays might also disappear.

2. Operating Costs

Operating costs consist of a number of items such as maintenance costs, fuel costs, legal services, etc. At this stage it can only be hoped that with design efforts based on experience that maintenance costs can be reduced. A better perception of the safety of the plant and a better regulatory climate might reduce the legal expenses, but for now there is no real basis for sharply reducing these operating costs.

The largest contributor to operating costs is likely to be fuel costs. As the prototype reactor with its high enrichment represents a worst case, let us examine its fuel costs in some detail.

a. Estimated Fuel Costs

The reference core would be initially loaded with one third at 10% enrichment, one third at 13% enrichment and one third at 15% enrichment. The reloads would also have 15% enrichment.

The initial core inventory of 60,000 kg of uranium has a cost which is calculated on the basis of 0.2 wt. % tails assay from the enrichment plant. The SWU and natural uranium feed per kg of enriched uranium are estimated as:

Enriched Uranium -	10%	15%
Feed	- 19.178	28.963
SWU	- 20.863	33.225

assuming a linear interpolation between 10% and 15% enriched uranium we get an average feed of 24.40 and SWU of 27.45. Then, the initial core costs are given in Table 17.

Table 17

Core Inventory Costs

Cost of Natural Uranium	@ \$60/kg
Cost of SWU	@ \$130/kg SWU
Cost of fabrication	@ \$195/kg

		Cost ($\$10^6$)
(1) Natural Uranium Cost		
24.40 kg/kg x 60,000 kg x \$60/kg	=	87.80
(2) SWU Cost		
24.40 kg/kg x 60,000 x \$130/kg	=	214.39
(3) Fabrication Cost		
60,000 x \$195/kg	=	11.70
Total Cost	=	\$313.89

We will assume that one third of the core will be reloaded at the end of every two years. This requires a capacity factor of 0.8654, which is not unreasonable. Hopefully, we might get between 0.90-0.95 capacity factor.

The cost of a reload is shown in Table 18.

Table 18

Cost of a Reload

		Costs ($\$10^6$)
Natural Uranium Cost		
28.963 kg/kg x 60,000/3 kg x \$60/kg	=	\$ 34.76
SWU Cost		
33.245 kg/kg x 60,000/3 kg x \$130/kg	=	86.41
Fabrication Cost		
60,000/3kg x \$195/kg	=	3.88
Total Cost	=	\$125.00

The fuel cost, in mills/kwh, is the ratio of the present value of all costs associated with fuel to the present value of the electricity generated over a thirty year period. Assuming that the cost of money is 12%/year, the cash outlay by the utilities and present value are shown in Table 19.

Table 19

Present Value of All Costs Over Thirty Year Period

End of Year	Cash Outlay ($\$10^6$)	Present Value	PVCO ($\$10^6$)
0	314	1.000	314
3	125	0.712	89
6	125	0.507	63.4
9	125	0.361	45.1
12	125	0.257	32.1
15	125	0.183	22.9
18	125	0.130	16.3
21	125	0.093	11.6
24	125	0.066	8.3
27	125	0.047	5.9
	Present Value of All Costs =		\$608.6

The sum of the present value over 30 years is 9.058. Thus, assuming that each year the plant generates 7.582×10^9kwh, we find the present value of all the electricity generated during the 30 year period to be,

$$\text{Energy kwh} = 9.058 \times 7.582 \times 10^9 = 68.67 \times 10^9 \text{kwh}$$

Thus, thirty year average present value of the fuel cost is,

$$m = \frac{\$608.6 \times 10^6 \times 10^3 \text{ mills/\$}}{68.67 \times 10^9 \text{kwh}} = 8.86 \text{ mills/kwh.}$$

If the cost of money were to drop or rise from the 12% figure assumed this cost would, of course, change.

The 8.86 mills/kwh is somewhat higher than conventional plant fuel costs as anticipated with the greatly enriched uranium reactor. If a plutonium reactor were used, the cost would be expected to fall about 4 mills/kwh. And when the metallurgy improves to where 200,000 MWD/tonne can be obtained from a fuel element, the fuel cost of the uranium reactor would also be approximately 4 mills/kwh in today's dollars.

X. SUMMARY AND CONCLUSIONS

On the basis of the material presented, it now appears that an ultra-safe steam-cooled reactor plant can be designed and built with a minimum of development. The reactor based on existing fuel element design techniques appears to be economically viable, and with future development could be considered "low cost."

The reactor and plant have a number of unique features that have been derived on the basis of the industry's 35 years of experience. Among these features are:

> The principle of operating a reactor on the peak of a reactivity curve such that the reactor goes subcritical if the coolant density decreases or increases away from the operating peak;
>
> The ultra-safe feature can be adapted to large size plants;
>
> The same plant accommodates either a burner or a breeder;

The plant can start out burning uranium and then can switch to using plutonium if the national fuel cycle needs call for breeding;

As a breeder the steam-cooled reactor is essentially equivalent to the liquid metal cooled breeder reactor;

Reactivity lifetime is long, possibly 3 to 5 years without opening the pressure vessel;

Plant technology is water based;

The plant operates at low pressure and conventional temperatures, but efficiency is competitive, being approximately 35% at 1000 psi;

The plant appears to be easy to maintain and should have high availability;

The plant contains a built in clean-up system that operates at atmospheric pressure;

A self-cooling system for the pressure vessel is used that is dedicated to the removal of decay heat even without power;

A new fast acting control system is provided to quickly limit transient excursions to where the operation of the safety system is rarely required;

Control system is highly automated with a minimum of man-machine interfaces;

The plant can use double walled piping;

The pressure vessel can be ruptured without endangering the reactor.

The safety of the reactor is provable for perception purposes. Either artificial pipe breaks may be demonstrated, and/or random control manipulation can be demonstrated with public participation. And finally, the perception of breeders is that they are less safe than burners, and now here is a breeder that is the safest of them all.

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APPENDIX A

DESCRIPTION OF THE VIM CODE

VIM is a continuous energy Monte Carlo code which was started at Atomics International (A-1) and completed at Argonne National Laboratory (A-2). The VIM code provides a wide geometrical capability and a neutron physics data base closely representing the ENDF/B-IV data from which is was derived.

VIM contains the combinatorial geometry package developed for the code SAM-CE developed by M. O. Cohen (A-3), which has been extended to include the description of repeating rectangular and hexagonal lattices. The use of combinatorial geometry permits a detailed description of complicated and irregular assemblies. An infinite, homogeneous medium option is also available to provide an efficient capability for data testing and cross section evaluation.

Cross section definition in VIM is by composition-independent microscopic data sets. Resonance and smooth cross sections are specified pointwise with linear interpolation to provide a continuous energy cross section description; unresolved resonances are described by the probability table method developed by L. B. Levitt (A-4). The reaction types fission, elastic scattering, discrete level inelastic scattering, inelastic continuum scattering, and (n,2n) reaction are specifically defined, while "capture" is defined as the remaining possible outcome of a neutron collision. Neutron trajectories and scattering are continuous in angle. Anisotropic elastic and discrete level inelastic scattering are described with probability tables derived from ENDF/B-IV data. For those materials with thermal scattering law data specified in ENDF/B, the data are run through a series of processing codes into a code to create the final library of thermal scattering probability tables and thermal inelastic cross sections which are used by VIM. A discussion of the techniques used to generate the VIM cross sections may be found in R. E. Prael (A-5) and R. E. Prael and H. Henryson (A-6).

An independent check of VIM resonance region cross sections was performed at VPI&SU by numerically integrating the data on the VIM cross section library to obtain infinitely dilute resonance integrals for the principal isotopes. These integrals are compared with experimental values in Table A-1. The good agreement shown here verifies the VIM data base for these isotopes.

The VIM code calculates eigenvalues by analog, collision, and track length estimation, and averaging c the various eigenvalue estimates is provided for variance reduction. Both collision and track length estimation are used to provide reaction rate estimates by region, group and isotope, while group and region integrated fluxes are given by track length estimation. Track length estimation of reaction rates and fluxes is used to provide estimates of microscopic cross sections over tally regions. A service code called RETALLY can be used to collapse these tally regions in space and energy as desired to provide things such as pin cell or unit-assembly averaged few group microscopic cross sections.

Table A-1

A Comparison of VIM Calculated and Experimentally Measured Resonance Integrals

Isotope	Calculated Capture Integral (barns)	Calculated Fission Integral (barns)	Measured Capture Integral (barns)	Measured Fission Integral (barns)
U235*	125	206	128 ± 5	208 ± 10
U238**	286	-	286 ± 8	-
Pu239*	167	215	167 ± 7	231 ± 14

* U235 and Pu239 measured values are above 3 ev and are taken from H. M. Eiland et al. (A-7).

** U238 measured value is above 0.5 ev and is taken from J. Hardy et al. (A-8).

A typical result of the VIM statistics are shown in Table A-2 for the 8.0 weight % fissile Pu infinite lattice case. Since the effective standard deviation is computed by VIM, we can use Student's t-distribution (A-9) with whatever degrees of freedom are quoted, to find the confidence limit of the calculation. We obtain the confidence limits for the mean value by using the percentage points of Student's t-distribution. Thus, the probability,

$$p(\bar{x} - s\, t_{\nu-1,\alpha}/\nu^{1/2} < \mu < \bar{x} + s\, t_{\nu-1,\alpha}/\nu^{1/2}) = 1 - \alpha \quad ,$$

defines $(1-\alpha)$ confidence interval for the mean, μ. s is the square root of the sample variance and ν are the degrees of freedom. Assuming a symmetric distribution and using the combined analog estimator and track length for the data in Table A-2 we get,

$$s\, t_{\nu-1,\alpha} = 0.00214\, t_{104,\alpha}\, .$$

For 99% confidence limits, $k_{eff} = 1.0781 \pm 0.00055$. Finally, we note that by reducing the confidence limits to 95%, we get $k_{eff} = 1.0781 \pm 0.00042$.

References for Appendix A

A-1. Levitt, L. B. and Lewis, R. C., "VIM-1, A Non-Multigroup Monte Carlo Code for Analysis of Fast Critical Assemblies," AI-AEC-12951, Atomics International (1970).

A-2. Prael, R. E. and Milton, L. J., "A User's Manual for the Monte Carlo Code, VIM," FRA-TM-84, Argonne National Laboratory (1976).

A-3. Cohen, M. O., "SAM-CE: A Three-Dimensional Monte Carlo Code for Forward Neutron and Forward and Adjoint Gamma Ray Transport Equations," MR-7021, Mathematical Applications Group, Inc. (1971).

A-4. Levitt, L. B., "The Probability Table Method for Treating Unresolved Resonances in Monte Carlo Critically Calculations," Trans. Am. Nucl. Soc. 14: 648 (1971).

A-5. Prael, R. E., "Cross Section Preparaion for the Continuous-Energy Monte Carlo Code VIM," Nuclear Cross Sections and Technology, p. 447 (1975).

A-6. Prael, R. E. and Henryson, H., "A Comparison of VIM and MC^2-1-Two Detailed Solutions of the Neutron Slowing Down Problems," Nuclear Cross Sections and Technology, p. 451 (1975).

A-7. Eiland, H. M., et al., "Epithermal Measurements of Capture and Fission Resource Integrals in O-235, U-233, Pu-239 and Pu-241," Nuclear Science and Engineering, 44, p. 180 (1971).

A-8. J. Hardy et al., "Influence of Electrical Binding Effects in Water on Measured Resonance Integrals," Nuclear Science and Engineering 27, p. 135 (1967).

A-9. R. A. Fischer and F. Yates, Statistical Tables, Cliver & Boyd, Ltd., London, England (1938).

APPENDIX B

USE of RFD-2

<u>Computation of Eigenvalues and Group Fluxes</u>

Following the strategy utilized in ODMUG, a one-dimensional criticality program developed by J. R. Thomas (B-1), the eigenvalues and four group fluxes are found by solving the following equation:

$$-\nabla \left[D_k(r)\nabla\phi_k(r)\right] + \Sigma_k(r)\phi_k(r) = \chi_k G(r)/\lambda + \Sigma^R_{k-1}(r) \tag{B-1}$$

where, k = energy group 1, 2, 3, or 4 (group 4 is the thermal group),

$D_k(r)$ = diffusion coefficient in group k at position r,

$\Sigma_k(r)$ = $D_k(r)B_k^{\,2}(r) + \Sigma^a_k(r) + \Sigma^R_k(r)$ is the total cross section,

$B_k^{\,2}$ = transverse buckling,

Σ^a_k = macroscopic absorption cross section in group k,

Σ^R_k = macroscopic removal cross section by scattering in group k,

Σ^p_k = macroscopic poison absorption cross section in group k,

χ_k = fraction of fission neutrons born in group k,

$G(r) = \sum_{k=1}^{4} \nu_k \Sigma^f_k \phi_k$ is the fission source, and

λ = eigenvalue related to k_{eff}.

RFD-2 automatically creates a set of linear equations from equation (B-1) through finite difference methods. These equations are conveniently set in the matrix form,

$$\underline{\Gamma}\,\underline{\phi} = (1/\lambda)\,\underline{X}\,\underline{F}\,\underline{\phi} + \underline{R}\,\underline{\phi} \qquad \text{(B-2)}$$

whereby the flux vector, $\underline{\phi}$, corresponding to the neutron flux at each spatial point r in the cylindrical reactor, is solved iteratively by Gaussian elimination. Since the matrix $\underline{\Gamma}$ is tridiagonal, no inner iterations are required. The outer iterations for eigenvalue convergence are accelerated by use of Chebyshev polynomials.

Fuel Depletion

The following are the decay chains used in RFD-2 for solving the fuel depletion equations:

$$\begin{aligned} n + \text{U-235} &\rightarrow \text{Fission} && \text{(B-3)}\\ &\rightarrow \text{U-239} + \gamma \end{aligned}$$

$$\begin{aligned} n + \text{U-238} &\rightarrow \text{Fission} && \text{(B-4)}\\ &\rightarrow \text{U-239} \rightarrow \text{Np-239} + \beta^- \rightarrow \text{Pu-239} + \beta^- \end{aligned}$$

$$\begin{aligned} n + \text{Pu-239} &\rightarrow \text{Fission} && \text{(B-5)}\\ &\rightarrow \text{Pu-240} + \gamma \\ n + \text{Pu-240} &\rightarrow \text{Fission} \\ &\rightarrow \text{Pu-241} + \gamma \end{aligned}$$

$$\begin{aligned} n + \text{Pu-241} &\rightarrow \text{Fission} && \text{(B-6)}\\ &\rightarrow \text{Pu-242} + \gamma \\ \text{Pu-241} &\rightarrow \text{Am-241} + \beta^- \end{aligned}$$

$$\begin{aligned} n + \text{Pu-242} &\rightarrow \text{Fission} && \text{(B-7)}\\ &\rightarrow \text{Pu-243} \rightarrow \text{Am-243} + \beta^- \end{aligned}$$

These decay chains can be written as a set of coupled first order differential equations in which U-236, U-239, Am-241, Pu-243 and Am-243 can be ignored for the cases to be studied. Furthermore, there is also no explicit representation of gamma and beta deposition energy in RFD-2.

$$dN_{25} \,/\, dt = - \sum_{k=1}^{4} \sigma_k^{a,25} \; \phi_k \; N_{25} \tag{B-8}$$

$$dN_{28} \,/\, dt = - \sum_{k=1}^{4} \sigma_k^{a,28} \; \phi_k \; N_{28} \tag{B-9}$$

$$dN_{49} \,/\, dt = \sum_{k=1}^{4} \sigma_k^{c,28} \; \phi_k \; N_{28} - \sum_{k=1}^{4} \sigma_k^{a,49} \; \phi_k \; N_{49} \tag{B-10}$$

$$dN_{40} \,/\, dt = \sum_{k=1}^{4} \sigma_k^{c,49} \; \phi_k \; N_{49} - \sum_{k=1}^{4} \sigma_k^{a,40} \; \phi_k \; N_{40} \tag{B-11}$$

$$dN_{41} \,/\, dt = \sum_{k=1}^{4} \sigma_k^{c,40} \; \phi_k \; N_{40} - [\lambda_{41} + \sum_{k=1}^{4} \sigma_k^{a,41} \; \phi_k \approx N_{41} \tag{B-12}$$

$$dN_{42} \,/\, dt = \sum_{k=1}^{4} \sigma_k^{c,41} \; \phi_k \; N_{41} - \sum_{k=1}^{4} \sigma_k^{a,42} \; \phi_k \; N_{42} \tag{B-13}$$

where, $\sigma_k^{a,m}$ = the microscopic absorption cross section of fuel isotope m for energy group k.

$\sigma_k^{c,m}$ = the microscopic capture cross section of fuel isotope m for energy group k.

N_m = the number density of fuel isotope m in atoms/cubic centimeter.

λ_{41} = decay constant for Pu-241 (all the other fuel isotopes have negligible decay constants).

The neutron energy spectrum in a particular core region can be taken as a constant over sufficiently small time intervals. Given this basic assumption, the rate equations outlined above can be solved analytically in terms of the neutron fluence, defined as the resonance fluence,

$$\Theta = \int_{o}^{t} \phi_3(t')dt' \qquad \text{(B-14)}$$

where the time period t is defined by the user in RFD-2 and is typically on the order of 700 hours. These non-homogeneous first order differential equations can be solved using standard methods. Their solutions are given by M. C. Edlund (B-2). These solutions for the number densities in atoms/unit volume as a function of fluence is given in Table B-1.

^{135}Xe and ^{149}Sm are treated individually and the remaining fission products are divided into three pseudo-fission product groups; 1) rapidly saturating, 2) slowly saturating and 3) non-saturating. There are eleven isotopes in the rapidly saturating group, five isotopes in the slowly saturating group and one hundred sixty five isotopes in the non-saturating group. These are given in Table B-2.

During June 1966 a fission product Cross Section Evaluation Group was formed at Brookhaven National Laboratory. They decided to represent fission product poisoning as a function of three lumped pseudo-fission products. Of course, ^{135}Xe and ^{149}Sm were treated individually. Data sources and theoretical methods were used in this evaluation and a complete listing of the data in the ENDF/B format is given in the work of W. A. Wittkopf (B-3). He included ^{113}Cd, ^{151}Sm, ^{155}Gd and ^{157}Gd in the rapidly saturating group and ^{95}Mo, ^{99}Tv, ^{103}Rh, ^{131}Xe, ^{133}Cs, ^{143}Nd, ^{147}Pm, ^{152}Sm and ^{153}Eu in the slowly saturating group. About eight years ago, in his early investigations of tight lattices, M. C. Edlund (B-4) decided to move six isotopes from the slowly saturating group to the rapidly saturating group because the estimated infinitely dilute integrals were some hundred times greater than those isotopes still remaining in the slowly saturating group. The reason for doing this is the increased importance of resonance absorptions in tight lattices.

The TOAFEW collapsing code and data file for ENDF/B-IV fission products are taken form the work of W. B. Wilson, T. R. England and R. J. LaBauve (B-5). The ENDF/B-IV fission product library includes radioactive decay, neutron reaction and fission yield data for 824 nuclides. The TOAFEW collapsing code allows the user to specify any neutron spectrum he wants. In these studies, the spectrum is quite different from a conventional PWR and this particular code is very useful.

Table B-1

<u>Variation of Fuel Isotope Densities with Fluence</u>

$$N_5\ (\Theta) = N_5\ (0)\ e^{-\mu_5\ \Theta}$$

$$N_6\ (\Theta) = N_6\ (0) + \frac{\gamma_5\ N_5\ (0)}{\mu_5 - \mu_6}\ e^{-\mu_6\ (\Theta)} + \frac{\gamma_5\ N_5\ (0)}{\mu_6 - \mu_5}\ e^{-\mu_5\ \Theta}$$

$$N_8\ (\Theta) = N_8\ (0)\ e^{-\mu_8\ \Theta}$$

$$N_9\ (\Theta) = N_9\ (0) + \frac{\gamma_8\ N_8\ (0)}{\mu_8 - \mu_9}\ e^{-\mu_9\ \Theta} + \frac{\gamma_8\ N_8\ (0)}{\mu_9 - \mu_8}\ e^{-\mu_8\ \Theta}$$

$$N_0\ (\Theta) = A\ e^{-\mu_0\ \Theta} + B\ e^{-\mu_9\ \Theta} + C\ e^{-\mu_8\ \Theta}$$

$$N_1\ (\Theta) = D\ e^{-\mu_1\ \Theta} + E\ e^{-\mu_0\ \Theta} + F\ e^{-\mu_9\ \Theta} + G\ e^{-\mu_8\ \Theta}$$

$$N_2\ (\Theta) = H\ e^{-\mu_2\ \Theta} + Q\ e^{-\mu_0\ \Theta} + R\ e^{-\mu_0\ \Theta} + S\ e^{-\mu_9\ \Theta} + T\ e^{-\mu_8\ \Theta}$$

where

$$A = N_0\ (0) + \frac{\gamma_9 \gamma_8 N_8\ (0)}{(\mu_0 - \mu_9)(\mu_9 - \mu_8)} + \frac{\gamma_9\ N_9\ (0)}{\mu_9 - \mu_0} + \frac{\gamma_9 \gamma_8 N_8\ (0)}{(\mu_8 - \mu_0)(\mu_9 - \mu_8)}$$

$$B = \frac{\gamma_9}{\mu_0 - \mu_9}\ N_9\ (0) + \frac{\gamma_8\ N_8\ (0)}{\mu_8 - \mu_9}$$

$$C = \frac{\gamma_9 \gamma_8 N_8\ (0)}{(\mu_0 - \mu_8)(\mu_9 - \mu_8)}$$

$$D = N_1(0) + \frac{\gamma_0}{\mu_0-\mu_1} A + \frac{\gamma_0}{\mu_9-\mu_1} B + \frac{\gamma_0}{\mu_8-\mu_1} C$$

$$E = \frac{\gamma_0}{\mu_1-\mu_0} A$$

$$F = \frac{\gamma_0}{\mu_1-\mu_9} B$$

$$G = \frac{\gamma_0}{\mu_1-\mu_8} C$$

$$H = N_2(0) + \frac{\gamma_1}{\mu_1-\mu_2} D + \frac{\gamma_1}{\mu_0-\mu_2} E + \frac{\gamma_1}{\mu_9-\mu_2} F + \frac{\gamma_1}{\mu_8-\mu_2} G$$

$$Q = \frac{\gamma_1}{\mu_2-\mu_0} E$$

$$R = \frac{\gamma_1}{\mu_2-\mu_1} D$$

$$S = \frac{\gamma_1}{\mu_2-\mu_9} F$$

$$T = \frac{\gamma_1}{\mu_2-\mu_8} G$$

$$\mu_i = \Big[\sigma_{ai}^{(1)} \frac{\phi_1}{\phi_3} + \sigma_{ai}^{(2)} \frac{\phi_2}{\phi_3} + \sigma_{ai}^{(3)} + \sigma_{ai}^{(4)} \frac{\phi_4}{\phi_3}$$

Except for μ_1 we add a term γ_{41}/ϕ_3 where γ_{41} is the radioactive decay constant for Pu-241.

γ_i = average radiative capture cross section for each isotope, for example;

$$\gamma_8 = \sigma_{\gamma 8}^{(1)} \frac{\phi_1}{\phi_3} + \sigma_{\gamma 8}^{(2)} \frac{\phi_2}{\phi_3} + \sigma_{\gamma 8}^{(3)} + \sigma_{\gamma 8}^{(4)} \frac{\phi_4}{\phi_3} .$$

Table B-2

FISSION PRODUCT ISOTOPES MAKING UP
EACH PSEUDO-FISSION PRODUCT GROUP

Fission Product Group	Isotope	Z	A
Rapidly Saturating	Rh	45	103
	Ag	47	109
	Cd	48	113
	In	49	115
	Xe	54	131
	Pm	61	147
	Sm	62	151
	Sm	62	152
	Eu	63	153
	Gd	64	155
	Gd	64	157
Slowly Saturating	Mo	42	95
	Tc	43	99
	Cs	55	133
	Nd	60	143
	Nd	60	145
Non-Saturating	165 Isotopes		
Total Number of Isotopes =	181		

Table B-3

INDIVIDUAL ESTIMATORS FOR K-EFFECTIVE

	ANALOG	TRACK LENGTH	COLLISION
ESTIMATED K-EFF	0.107971D+01	0.107516D+01	0.107961D+01
EST. STND. DEV.	0.250796D-02	0.318042D-02	0.260741D-02

COMBINED ESTIMATORS FOR K-EFFECTIVE

	ANALOG ESTIMATOR AND TRACK LENGTH	ANALOG ESTIMATOR AND COLLISION	TRACK LENGTH AND COLLISION
ESTIMATED K-EFF	0.107807D+01	0.107966D+01	0.107910D+01
CORRELATION COEF	0.171621D+00	0.349311D+00	0.760262D+00
EST. STND. DEV.	0.212557D-02	0.209920D-02	0.259651D-02
EFFECTIVE STANDARD DEVIATION (*)	0.214073D-02	0.209921D-02	0.265223D-02

(*) FOR OBTAINING CONFIDENCE INTERVALS USING STUDENT'S T-DISTRIBUTION WITH 105 DEGREES OF FREEDOM

SIMPLE AVERAGES OF ESTIMATORS FOR K-EFFECTIVE

	ANALOG ESTIMATOR AND TRACK LENGTH	ANALOG ESTIMATOR AND COLLISION	TRACK LENGTH AND COLLISION	ALL THREE ESTIMATORS
ESTIMATED K-EFF	0.107743D+01	0.107966D+01	0.107738D+01	0.107816D+01
EST. STND. DEV.	0.218762D-02	0.210101D-02	0.271674D-02	0.218884D-02

References for Appendix B

B-1. J. R. Thomas, "Reactor Statics Module, RS-8, Three Group Criticality Program" Report on National Science Foundation Grant GZ-2888, Aug. 1974.

B-2. M. C. Edlund, "Fuel Management Module, FM-1, Fuel Burnup in Slow Neutron Fission Reactors," Report on National Science Foundation Grant GZ-2888, 1974.

B-3. W. A. Wittkopf, "Lumped Fission Product Neutron Capture Cross sections for END/B, BAW-320,", Babcock & Wilcox, Lynchburg, Virginia (1966).

B-4. M. C. Edlund, "Physics of the Uranium Plutonium Fuel Cycle in Pressurized Water Reactors," Trans. Am. Nucl. Soc., 24, 509 (1976).

B-5. W. B. Wilson, T. R. England and R. J. LaBauve, "Multigroup and Few Group Cross Sections for ENDF/B-IV Fission Products; the TOAFEW Collapsing Code and Data File of the 154-Group Fission-Product Cross Sections," LA-7174-MS, Los Alamos Scientific Laboratory, Los Alamos, New Mexico (March 1978).

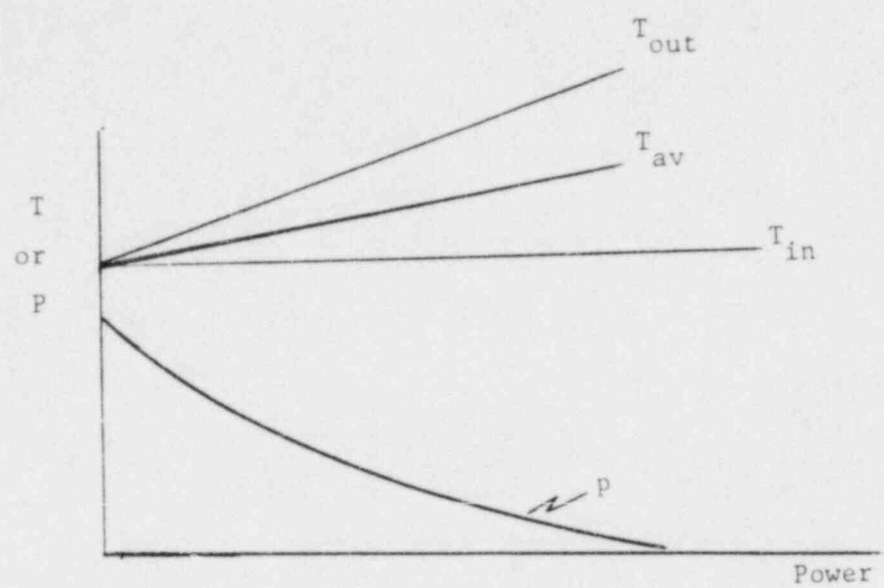
APPENDIX C

Text Reference (9) provides a simplified general solution of the reactor stability problem with two path feed back in LWR's. This analysis looks at Nyquist stability only, which implies small signal linear analysis. Small signal techniques have been extremely successful in reactor work providing good results for reactivity inputs as high as 0.010 k. There are several non-linear approaches that can be examined later if required. (La Grange, Liopunov, etc.).

The analysis in the reference is for an LWR having two feedback paths, one being in series with the other. This analysis will follow the LWR case by analogy. To start, first a signal disturbs the reactor. The fuel elements change their temperature and very quickly feed back reacti ity via the doppler coefficient α_f. After the fuel temperature changes, the change in heat is then transferred to the moderator. The moderator then, in turn, also feeds back a reactivity change to the reactor via a temperature coefficient α_w (water). The key point is that there is a time delay between the doppler effect and the moderator temperature coefficient effect. In the analysis this delay is represented by a single order time lag.

A similar situation exists in the steam-cooled reactor (SCR). The fuel temperature changes first, and reactivity is fed back via α_f, the doppler coefficient. The change in fuel temperature creates a change in power output that, in turn, affects the system pressure. Reactivity is fed back some time later via a pressure (or density) coefficient, α_p. Therefore, by analogy one merely substitutes α_p for α_w. Actually, the mechanism setting up a pressure change is more complex for the steam cooled reactor and probably contains two or more time lags. But as a first approximation, we make the assumption that pressure feedback occurs on the basis of a single order time lag from fuel temperature.

We now must inspect one other difference between the LWR and the SCR. This difference is created by the power program. The power program of the SCR is indicated below in which we plot temperature and pressures against power output.



It will be recalled that the inlet temperature to the reactor is held constant by the pressurizer arrangement and the secondary steam system. All free surfaces in the vessel are held at the saturation temperature (544° at 1000 psi). This fact now, immutably, controls the steady state values of T_{out}, T_{av}, and p as the power level is varied. The key point to note is that, although T_{av} rises with power level, p falls. In the LWR, the key variables are T_f and T_{av}. Note they always go up and down together with power. In the SCR, the key variables T_{av} and p go in opposite directions with power. This will mean that we have to watch the signs of the feedback reactivity in order to be able to use the analogy to the reference analysis on a one for one basis.

Without any further ado, we can then proceed to the solution to the problem given in table 5-11 p. 148 of Control of Nuclear Reactors and Power Plants (9). Three pertinent cases are involved as presented in table C-1.

Table C-1

Pertinent Cases in Stability Solution

Temperature Coefficient Range	Stability
Case I	
α_w negative	Completely stable
α_f negative	
under the condition that	
$$\left| \frac{\alpha_w}{\alpha_f} \right| < \left| \frac{\mu_w}{\xi} (\bar{r} - \bar{\lambda}) + \frac{\mu_w}{\mu_f} \right| \qquad \text{(C-1)}$$	
Case II	
α_w negative	Stability depends upon gain
α_f negative	
under the condition that	
$$\left| \frac{\alpha_w}{\alpha_f} \right| < \left| \frac{\mu_w}{\xi} (\bar{r} - \bar{\lambda}) + \frac{\mu_w}{\mu_f} \right| \qquad \text{(C-2)}$$	
Case III	
α_w positive	Completely stable, but may have poor transient response
α_f negative	
under the condition that	
$$\left| \alpha_f \right| > \left| \alpha_w \right| \qquad \text{(C-3)}$$	

In this table, temperature coefficient range translates into feedback coefficient range. α_w, and α_f are not to be interpreted as feedback coefficients, but rather as the reactivity created as far as sign is concerned. That is, α_f negative means the reactivity change $\alpha_f \Delta T_{av}$ is in the opposite direction as the effect that created the feedback, i.e., a positive change in reactivity.

The constants μ_f, μ_w, ξ, $\bar{r}$, $\bar{\lambda}$ have little meaning to us at this point in that they relate to the LWR reactor design and the time delay mechanism of the moderator temperature coefficient feedback. As the pressure coefficient in the SCR is set up by a completely different mechanism, we can't use equations (C-1) and (C-2) by analogy, but will approach them in a different manner later.

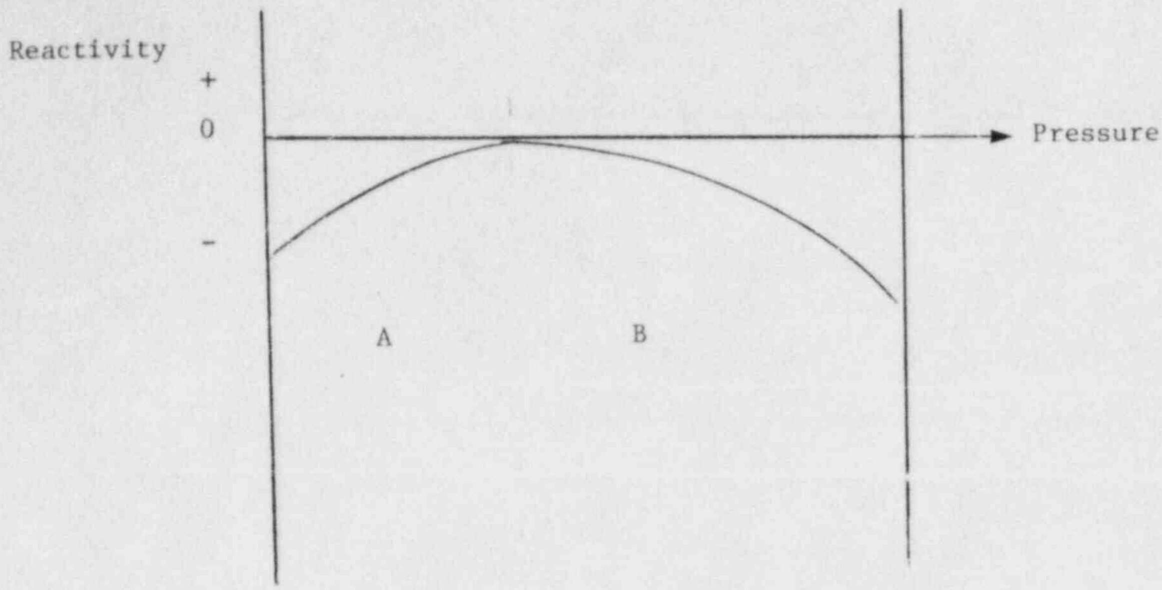
Gain refers to the open loop gain of the combined reactor and feeback system. The gain of the feedback system is simply a constant C. The gain of the reactor in simple approximation is $\frac{n_o \beta}{\ell^*}$ where n_o is the neutron power level, β the conventional fraction of delayed neutrons, and ℓ^* the mean neutron lifetime. Of interest here is the fact that the gain depends directly on powr level.

So the open loop gain is

$$\frac{C\ n_o \beta}{\ell^*} \qquad \text{(C-4)}$$

which merely implies that, if the power level is low enough, the reactor can't oscillate.

We now go back to the basic design curve of the SCR, and see what conditions apply to the various parts of the curve.



We make a table as to the direction of a change in the variables in regions A and B for a positive reactivity or power change.

region	δk	Power	T_{av}	$\alpha_f \Delta T_{av}$	p	$\alpha_p \Delta p$
A	↑	↑	↑	↓	↓	↓
B	↑	↑	↑	↓	↓	↑

We note that in region A, if the reactivity or power goes up, the temperature induced reactivity and the pressure induced reactivity go down. In other words, corresponding to the table cases, we have either Case I or Case II where both feedbacks are negative.

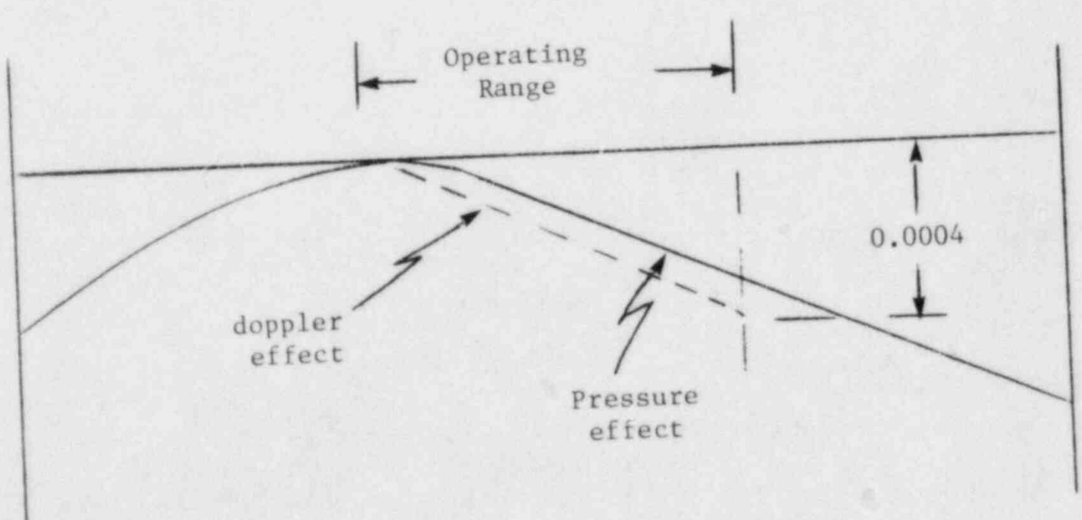
Similarly, in region B, if the reactivity or power goes up, the temperature feedback goes down, but the pressure feedback goes up. Or, in the table notation α_f is negative, α_p is positive -- Case III of the table.

Let us look at Case III, region B first. The constraint to stability is, in our case

$$| \alpha_f \Delta T_f | > | \alpha_p \Delta p |$$

In other words, the reactivity created by the density change must be less than the reactivity created by the doppler effect.

We can look at the full power swing in order to get a feel for numbers. The doppler coefficient in the SCR is about $-5.5 \times 10^{-6}/^oF$. T_{in} is constant at 544^oF, and T_{out} at full power is 690^oF. Therefore, the T_{av} swing from zero power to full power is 73^oF creating a reactivity difference of 0.0004 δk. So, crudely, we might want to limit the reactivity available by a density swing on the high pressure section of the curve to slightly less than 0.0004. This situation is shown graphically and exaggerated on the sketch below.



Actually, this picture should be very conservative. If we are operating near the peak of the curve, the slope of the pressure curve is far less than the slope of the doppler effect. In reality, only a small portion of the curve near the peak would be used for power operation. Much of the remainder of the curve would be for subcritical operation. And subcritical reactors can't oscillate under the two-feedback restriction. So, although the numbers are unrefined at this stage, clearly if a better analysis shows the need for a restriction of the sort $|\alpha_f \Delta T_f| > |\alpha_p \Delta p|$ arises, it can always be met by changing the values of the reactivity pressure curve by decreasing the slope in the operating range. (And this can be done by manipulating the thermal poison in the core.)

A similar situation holds with respect to the "A" region of the basic curve. If one decided to operate on the low density portion of the curve, the operating range again would be small. Table 5-11 of the reference indicates that when both feedbacks are negative, we can have either a completely stable situation or a conditionally stable situation depending on the ratio of $\frac{\alpha_p \Delta p}{\alpha_f \Delta T_{av}}$. This ratio can also be a function of the design of the reactor. (In water reactors it's about 10.) It is clear from the form of equation (C-1) and (C-2) that the reactor can be made completely stable, again if α_p is made small enough. And recall at the peak of the reactivity curve α_p is zero. The point in all cases is the same. As long as one can set the value of the two intercepts on the reactivity curve so as to limit the value of the density feedback reactivity over the operating range, one can insure that the reactor will be stable.

There is now the difference between Cases I and II in that Case II depends on gain. Both cases would produce "stiff" systems. That is they would respond quickly to inhibit transients. On the other hand, region B, while stable, would be under damped. Transients would be larger, and oscillations might take longer to die out.

It will be recalled that this analysis is for the general case where the reactor is permitted to go critical anywhere on the reactivity curve. In actual operation, criticality will be restricted to the vicinity of the peak of the curve. Here the conditions are much more favorable. At the peak of the curve α_p shrinks to zero. Hence we have only a single feedback path, the doppler effect. And single feedback paths around simple reactors cannot cause the system to oscillate. (In servo language the reactor has a 90° phase lag and the feedback has a 90° lag giving a total phase shift around the loop of 180°. Nyquist stability requires that the open loop phase shift be greater than 180° for an oscillation to be sustained.)

At this stage of the design it appears highly likely that the reactor will be stable.

APPENDIX D

Subsystem Comparison of Conventional BWR with Steam Cooled Reactor

	BWR	Steam-Cooled Reactor
NUCLEAR SYSTEMS--N		
N01	Reactor Core	Core more complex because of enhanced heat transfer. Grid plate heavier, steam pipes and diffuser openings additional
N02	Control Rod Drive System	Few rods simpler drive for shim rods
N02.A	Control Rod Drive Hydraulic System	Fast Regulating rod more complex
N03	Reactor Control System	More complex
N04	Reactor Recirculation System (including reactor vessel and internals)	No recirculation system, no jet pumps, no moisture separator and dryer. Partially compensated for by the addition of a blower, spray box, and valving.
N05	Standby Liquid Control System	Not used.
N06	Reactor Protection System	Similar
N07	Neutron Monitoring System	Similar, though a little simpler--no moving detectors.
N08	Residual Heat Removal/Low-Pressure Injection System	Different but roughly similar costs.
N09	Reactor Water Cleanup System	Similar but larger.
ENGINEERING SAFETY SYSTEMS--S		
S01	Reactor Core Isolation Cooling System	Not used
SO-2	Engineered safety features activated by a number of independent control systems.	Fewer
	BWR	Steam-Cooled Reactor

ENGINEERING SAFETY SYSTEM--S (continued)

S03	Engineered Safety Features	Not used
S03.A	High Pressure Core Injection/Spray System	Not used
S03.C	Low Pressure Coolant Injection is a functional mode of the Residual Heat Removal System	Tank coolant injection system used. Somewhat more complicated.
S03.D	Low Pressure Core Spray System	Not used
S03.E	Automatic Depressurization System	Simpler
S04	Remote Shutdown System	Probably not as complex.

CONTAINMENT SYSTEMS--C

C01	Primary Containment and Penetrations	May or may not be required compensated for by double walled piping.
C02	Reactor Building	Deeper underground structure
C03	Containment Heat Removal is a function of the Residual Heat Removal System	Probably not required.
C04	Containment Isolation System	Probably not required.
C05	Containment Purge System	Probably not required.
C06	Standy Gas Treatment System	Similar.
C07	Combustible Gas Control System	Probably not required.
C08	Containment Ventilation System	Probably not required.
C09	Reactor Building Ventilation System	Similar
C010	Containment Spray System BWR	Probably not required. Steam-Cooled Reactor

ELECTRICAL SYSTEMS--E

E01	Main Power System	Same
E01.A	Protective Relaying and Controls	Same
E02	Plant AC Distribution System	Same
E02.A	Essential Power System	Possibly smaller.
E02.B	Nonessential Power System	Same
E02.C	HPCS Power System	Not used.
E02.D	Protective Relaying and Controls	Same
E03	Instrumentation and Control Power Systems	Same
E03.A	DC Power System - Vital DC Power Subsystem - Plant DC Power Subsystem	Similar, possibly smaller.
E03.B	Instrument AC Power System - Vital Instrument AC Power Subsystem - Plant Instrument AC Power Subsystem	Same
E04	Emergency Power System	Possible smaller.
E04.A	Diesel-Generator Fuel Oil Subsystem	Same if used.
E04.B	Diesel-Generator Cooling Subsystem	Same if used.
E04.C	Diesel-Generator Air Subsystem	Same if used.
E04.D	Diesel-Generator Lubrication Oil Subsystem	Same if used.
E05	Plant Lighting System	Same
E05.A	Essential Lighting	Same
	BWR	Steam-Cooled Reactor

ELECTRICAL SYSTEMS--E (continued)

E05.B	Nonessential Lighting	Same
E06	Plant Computer	Similar, both upgraded state-of-the-art
E07	Switchyard	Same
E07.A	DC Control Power System	Same
E07.B	Protective Relaying	Same

POWER CONVERSION SYSTEMS--P

P01	Main Steam System	Similar, probably requiring throttle valve
P02	Turbine-Generator	Larger
P02.A	Electro-Hydraulic Control Subsystem	Same
P02.B	Turbine Gland Seal Subsystem	Same
P02.C	Turbine Lubrication Subsystem	Same
P02.D	Stator (Hydrogen) Cooling Subsystem	Same
P02.E	Hydrogen Seal Oil Subsystem	Similar
P03	Turbine Bypass System	Possibly larger to handle full power dumping
P04	Condenser and Condensate System	Same
P04.B	Turbine Condensate Cleanup/ Polishing System	Same
P04.C	Condensate Heater Drain	Same
P05	Feedwater System	Similar
P05.A	Feedwater Heater Drain Subsystem	Same
	BWR	Steam-Cooled Reactor

POWER CONVERSION SYSTEMS--P (continued)

P06	Circulating Water System	Simpler, no jet pumps.
P07	Steam Generator Blowdown System	Similar-may be connected to atmospheric tanks
P08	Auxiliary Steam System	Same, but another auxiliary system would be needed for initial reactor startup.

PROCESS AUXILIARY SYSTEMS--W

W01	Radioactive Waste system	similar
W01.A	Gaseous Radwaste System - Offgas subsystem	More complex, but at low pressure
W01.B	Liquid Radwaste System	More complex, but at low pressure
W01.C	Solid Radwaste System	Same
W02	Radiaton Monitoring System	Same
W02.A	Plant Area Radiation Monitors	Same
W02.B	Environmental Radiation Monitors	Same
W02.C	Process Radiaton Monitors	Same
W03	Cooling Water Systems	Same
W03.A	Reactor Building Cooling Water System	Same
W03.B	Turbine Building Cooling Water System	Same
W04	Service Water Systems	Same
W04.A	Demineralized Makeup Water System	Similar, probably larger storage tanks
	BWR	Steam-Cooled Reactor

PROCESS AUXILIARY SYSTEMS--W (continued)

W04.B	Station Service Water System - Essential Service Water System - Nonessential Service Water System	Same
W04.C	Chilled Water System	Same
W05	Refueling System	Same, but with two heads to remove
W06	Spent Fuel Storage System	Similar
W06.A	Fuel Pool Cooling and Clean-up System	Similar, if used
W07	Compressed Air System	Same
W07.A	Service Air System	Same
W07.B	Instrument Air System	Same
W08	Process Sampling System	Same
W09	Plant Gas System	Same
W09.A	Nitrogen System	Same
W09.B	Hydrogen System	Same if required.

PLANT AUXILIARY SYSTEMS--X

X01	Potable and Sanitary Water System	Same, but with connections for emergency pool cooling.
X02	Fire Protection System	Same, but with connections for emergency pool cooling.
X02.A	Water System	Same
X02.B	Carbon Dioxide System	Same
X03	Communications System	Same
X04	Security System	Same
X05	Heating, Ventilating, and Air Conditioning System	Same
	BWR	Steam-Cooled Reactor

PLANT AUXILIARY SYSTEMS--X (continued)

X05.A	Control Room Habitability System	Same
X05.B	Turbine Building Ventilation System	Probably smaller because of double wall containment
X05.C	Diesel Building Ventilation System	Same
X05.D	Auxiliary Building Ventilation System	Smaller
X05.E	Fuel Building Ventilation System	Same
X06	Nonradioactive Waste System	Same
X06.A	Gaseous Waste	Same
X06.B	Liquid Waste	Same
X06.C	Solid Waste	Same

NRC FORM 335 (7-77) U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET**	1. REPORT NUMBER (Assigned by DDC) NUREG/CR-4019
4. TITLE AND SUBTITLE (Add Volume No., if appropriate) A NEW STEAM COOLED REACTOR	2. (Leave blank)
	3. RECIPIENT'S ACCESSION NO.
7. AUTHOR(S) Melton C. Edlund and Monte Schultz	5. DATE REPORT COMPLETED MONTH August YEAR 1984
9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) Virginia Polytechnic Institute Blacksburg, VA 24061	DATE REPORT ISSUED MONTH November YEAR 1984
	6. (Leave blank)
	8. (Leave blank)
12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) Office of Nuclear Regulatory Research U. S. Nuclear Regulatory Commission Washington, D. C. 20555	10. PROJECT/TASK/WORK UNIT NO.
	11. CONTRACT NO. NRC-G-04-84-007
13. TYPE OF REPORT Final Report	PERIOD COVERED (Inclusive dates) 2/21/84-9/30/84
15. SUPPLEMENTARY NOTES	14. (Leave blank)

16. ABSTRACT (200 words or less)

A novel concept is described for a nuclear power plant that is ultra safe based on current knowledge of nuclear reactor safety. Both burner and breeder-type cores are studied. The concept utilizes steam-cooling during normal operation with automatic shutdown and heat removal by natural convection under off-normal conditions.

17. KEY WORDS AND DOCUMENT ANALYSIS 17a. DESCRIPTORS

Ultra safe; Reactor; Concepts, steam-cooling

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT	19. SECURITY CLASS (This report)	21. NO. OF PAGES
Unlimited	Unclassified	
	20. SECURITY CLASS (This page)	22. PRICE S

NRC FORM 335 (7-77)

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NUCLEAR REGULATORY COMMISSION
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