PRESENT DAY DESIGN CHALLENGES EXEMPLIFIED BY THE CLINCH RIVER BREEDER REACTOR PLANT

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I. Introduction

The Clinch River Breeder Reactor Plant (CRBRP) is a sodium cooled fast spectrum breeder reactor. It is the Demonstration Plant for the future liquid metal fast breeder reactor (LMFBR) industry. It is being designed and developed primarily under ERDA sponsorship, based on guidelines set down by the Project Management Corporation (PMC), the organization that represents the contributing electrical utilities. The lead reactor manufacturer is Westinghouse Electric Corporation with major subcontracts with General Electric and Atomics International. The artchitect engineer is Burns and Roe, Inc. and the constructor is Stone & Webster.

The plant will be constructed on the Clinch River near Oak Ridge, Tennessee and will be operated by TVA. It will achieve criticality in 1983 and will supply over 350 MWe to the TVA grid when full power operation is achieved.

The objective of the LMFBR program, of which the CRBRP project is a part, is to increase our utilization of uranium by converting fertile uranium-238 into fissionable plutonium-239 through neutron capture at a rate faster than Pu-239 is consumed. In this way our presently limited uranium resources will be effectively expanded by a factor of roughly 70. The LMFBR will be a major contributor to the national goal of obtaining energy independence.

This paper presents a brief description of the CRBRP and presents an example of the design challenge facing the CRBRP engineer.

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II. CRBRP Description

To set the basis for the ensuing discussion of problems, let's review the status of the CRBR design. Figure 1 is an elevation view of the reactor and its enclosure in cross section. The reactor and enclosure system consist of the reactor vessel, thermal liner, and guard vessel; the head; the lower internals, which support the reactor in the vessel; the core barrel and restraint system; and the upper internals, which support the instrumentation and control rod driveline, provide a backup holddown system for the assemblies, and direct the sodium flow to achieve good mixing in the outlet plenum. In the center are the fuel assemblies and control assemblies, surrounded by the radial blanket and shield assemblies.

Figure 2 shows the reactor plan in which you can see the various reactor assemblies. There are 198 fuel assemblies in two zones with the outer two rows of fuel having a higher enrichment than that used in the inner zone to improve power flattening. The next 2 1/2 rows are radial blanket assemblies which contain depleted uranium as a fertile material and in which a portion of the breeding is accomplished. There is an axial blanket that performs the same function in the axial directions as the radial blanket does radially. Outside the radial blanket assemblies are four rows of shield assemblies. The outer shield assemblies react on the former rings of the core restraint system to provide radial position control of the core, blanket, and control assemblies. The control assemblies occupy 19 locations within the core. Of the six in row 4, two are start-up assemblies and four are secondary assemblies. These 6 control rods are withdrawn during operation. The secondary assemblies are being designed by General Electric on a different principle of operation from the Westinghouse design being used for the primary assemblies to provide a diverse shutdown capability. They have a completely separate control system for complete diversity of operation. The center assembly and the control assemblies on the flats move as one bank while the corner assemblies move as a second bank. The banks are actually moved by sequential step of one control assembly at a time, and the term bank should not be construed to imply movement of a group of assemblies at one time.

A fuel assembly is shown in detail in Figure 3. It's a hexagonal structure of 316 stainless steel (SS) about 4 1/2" across flats. The duct contains thickened regions called load pads. These pads enable the structure to be held rigidly in place while providing space in the core region for the assemblies to swell. The fuel assembly contains 217 fuel rods. Each rod is 0.23" in diameter and has a cladding thickness of 15 mils. The rods are spaced in an equilateral triangular array by wires wrapped on the rods. The core itself is 36" long, containing uranium-plutonium dioxide pellets slightly under stoichiometric in oxygen content.

Above and below the uranium-plutonium dioxide are 14" of depleted uranium dioxide pellets. These are the axial blankets used for breeding. There is a plenum volume 48" long to collect fission products.

Figure 4 shows a radial blanket assembly which is similar to a fuel assembly externally. Inside, however, it has only 61 rods as opposed to the 217 of the fuel assemblies. The rods are 0.506" in diameter and contain depleted uranium dioxide in 0.015" thick stainless steel cladding.

The control assembly, as shown in Figure 5, is quite different. It has an outer duct that remains in the core during operation. The inner duct which is connected to the control rod driveline, moves relative to the outer duct. Inside the inner duct we see again the familiar wire wrapped rods, only this time we call them pins, of 50-mils-thick, stainless steel cladding. Within these control pins are B_4C pellets, in place of fuel, and plenum space to collect helium generated from the neutron-boron reaction. The control assembly connects to the control rod driveline. The system can be disconnected at the top of the male assembly to allow a completely free plane just above the assemblies during refueling. The purpose of this free plane is discussed later.

The shield assembly, as shown in Figure 6, is a simple hex duct filled with stainless steel rods. Its load pads react against the former rings to provide the core restraint force.

All of these assemblies fit into inlet modules of the type shown in Figure 7. Each module holds seven assemblies and fits into the core support plate as shown in Figure 8.

The high-pressure flow in the inlet module is shown here forming a low-pressure plenum. The flow from this plenum flows to a region above the reactor support cone between the core barrel and the vessel. It then flows on into the region between the vessel and the vessel liner to hold the vessel at temperatures below 900°F while the sodium inside the liner is at 995°F to 1015°F.

The upper internals, Figure 9, are located above the reactor assemblies. This structure promotes mixing of core exit flow streams, provides backup holddown, supports instrumentation used for surveillance and control and guides and protects the control rod drivelines through the cross flow in the upper plenum region of the reactor vessel.

The fundamental purpose of the upper internals requires it to have structural members to withstand the upward force that could be generated by movement of fuel or other assemblies in the event of loss of hydraulic holddown. For this purpose, the strength requirements, along with the requirements to avoid flow induced vibration, demand relatively thick members. However, rather large thermal transients are occasionally generated in the core and blanket outlet stream. To survive these projected transients, very thin material is required. The compromise is achieved by directing the flow into a lined mixing chamber and then through 29 thin-walled chimneys. The sodium is then dumped into the upper plenum above the main structure of the upper internals. The design problem the upper internals present will be discussed later.

The control rod drivelines span the gap from the head through the upper internals down into the core region. During operation, the upper internals are keyed into the core barrel to assure alignment of the driveline with the tops of the control assemblies. During a refueling operation, the control rods are fully inserted and disconnected at this plane. The upper internals are then elevated about 9 inches. They are connected by four support columns to the intermediate rotating plug on the head. All rotating plugs are then free to turn, thus rotating the upper internals over the whole reactor area without interferring with any structure within the reactor system.

Figure 10 shows a 3-dimensional view of the head. It consists of three rotating plugs, an in-vessel transfer mechanism mounted in the small plug, and a fuel transfer port in the large plug. The intermediate rotating plug also contains the control rod drive mechanisms and upper internals supports. The rotation of this intermediate plug allows the area of the small plug to cover almost the total area of the intermediate plug. By rotating the large plug, the entire reactor area out to a radius of the transfer pots outside the core barrel is covered by the small plug and the in-vessel transfer machine. During refeuling the IVTM transfers the assemblies to and from the core and the transfer pot outside the core while an ex-vessel transfer machine moves the assemblies to and from the transfer pot, through the head, to ex-vessel storage.

Figure 11 shows a simplified flow diagram of the plant and the thermal hydraulic design parameters. The primary heat transport system includes the hot leg primary pump, the intermediate heat exchanger (IHX) with the primary sodium in the shell side, the cold leg check valve, the piping and guard vessels and the necessary monitoring instrumentation. The IHX is the barrier which precludes reaction between the radioactive primary sodium and the water in the steam generator.

The key performance parameters are given in Table 1.

Figure 12 shows a plan view of the piping arrangement in the reactor containment building. Key features of the arrangement are the expansion loops which prevent overstressing due to growth over the temperature operating range of almost 1,000°F, the larger diameter piping between the vessel and the primary pump inlet to minimize the pressure losses, shielding necessary to prevent neutron streaming and allow maintenance access within reasonable times, and the sizes of the components and guard vessels.

Table 1

Reactor Power	975 MWt
Steam Conditions	900°F, 1450 psig at throttle valve
Steam Flow	3.3 million pounds per hour
Primary Sodium Flow	13.8 million pounds per hour
Intermediate Sodium Flow	12.8 million pounds per hour
Primary Hot Leg Temperature	995°F
Primary Cold Leg Temperature	730°F
Intermediate Hot Leg Temperature	936°F
Intermediate Cold Leg Temperature	651°F
Feedwater Inlet Temperature	450°F
Recirculation Ratio	two to one
Gross Turbine Generator Heat Rate	8753 BTU/KW-Hr
Estimated Net Plant Efficiency	35.5%

Preliminary design of the reactor and heat transport systems are virtually complete. At this time, we are proceeding into the detailed final design. The CRBR has a 30-year life and the capability to upgrade to utilize fuel advances is available through replacement of the various reactor assemblies. The entire plant meets the guidelines set by PMC and provides a base for extrapolation to commercial size liquid metal fast breeder reactor plants.

III. A Specific Technical Challenge

The nuclear engineer is concerned in the CRBR design, and in future LMFBR designs, with achieving a high breeding ratio and a low doubling time. Breeding ratio is the rate of production of fuel divided by the rate of consumption of fuel, and must be greater than unity in a breeder. The doubling time is the length of time required for a breeder to produce enough excess fuel, after accounting for reprocessing losses, to provide fuel for a similar reactor, including in-reactor fuel and out-of-reactor fuel cycle support fuel.

The nuclear engineer's best method of achieving a high breeding ratio and short doubling time results in thermal/hydraulic operating conditions that place the maximum problems on the design engineer. How this situation arises is illustrated in the next few figures.

Figure 13 shows the variation of Breeding Ratio with fuel volume fraction for a typical core, the CRBR. Different core geometries would result in different curves, but the trend would remain the same. All this slide reveals is that if more uranium is packed into the core, more neutrons are captured by it, and plutonium is produced at a faster rate. The increase of uranium in the core increases the fuel volume fraction. The steel volume fraction stays relatively constant; so the increase in fuel volume fraction is accompanied by a decrease in sodium volume fraction. To carry away the same amount of power either the sodium flow velocity must increase and result in a greater pressure drop, or the temperature rise that the sodium is given as it flows through the core must be increased.

How these factors enter quantitatively is illustrated in Figure 14. This figure shows the variation of breeding ratio with the ΔT of the core, for a given pump characteristic and a constant power. While the pump capability can be considered a variable, in point of fact it is a fairly well fixed parameter. The point at which it is fixed is the limit of head that can be generated versus flow rate for the given operating temperature, using the present state-of-the-art.

Before discussing how this affects the reactor mechanical design, let us digress to another self-imposed technical problem. One advantage of sodium as a coolant is its high boiling point, over 1700°F. This allows operation of turbines in an efficient thermal regime so that plant thermal efficiencies on the order of 40% or greater are achievable as opposed to the low thirties percent efficiency of a water cooled reactor. A high thermal efficiency reduces fuel costs directly, and power costs both from lowered fuel costs and reduced capital investment in the form of smaller heat transfer areas in the IHX and steam generator, and less cooling tower capacity. In order to achieve temperatures that are likely to be of interest in future LMFBR's, the design temperature of the CRBRP steam cycle has been selected with a 900°F throttle temperature. That results in a 730°F inlet and a 995°F outlet for the reactor.

These high temperatures and high ΔT's impose severe problems on the designers of components that must operate in the environment.

The service life of metals subjected to high temperature operation is limited by a combination of creep strain from steady state mechanical loads, creep and fatigue cumulative damage resulting from the stresses generated during thermal transients, and high primary stresses from seismic events.

A complete evaluation of a component involves a complex inelastic analysis considering the effects on a specific geometry at a given steady state temperature and a "duty cycle" of various numbers of various transients with different rates of change of temperature and different magnitudes of temperature change.

The analysis of a component must consider the damage interaction, $\Sigma\, n/N_d + \Sigma\, t/T_d$. $\Sigma\, n/N_d$ is the sum of the ratios of the number of required cycles, n, to the allowable number of cycles at a given strain range, N_d, for the various events in the duty cycle. $\Sigma\, t/T_d$ is the sum of the ratios of time endured, t, divided by the total time to rupture at a sustained stress level. Figure 15 illustrates the envelope of acceptable magnitudes of the combined damage. Each transient results in contribution to the fatigue damage in the $\Sigma\, n/N_d$ term and a sustained residual stress during the hold-time between transients which contributes to the $\Sigma\, t/T_d$ term. The combination of the damage of both can never exceed the envelope shown in this figure. The intent of this figure is to provide perspective for some figures to come later.

As a general description of the difficulty of the design problem, it could be said that thick structures (stiff) are commonly required to sustain mechanical loads while the same thick structures cannot withstand a severe thermal environment. Thus, a stiff base structure is often required that has thinner protective structures enveloping it to mitigate the thermal effects on the base structure.

It would be impossible in the short time allowed to delve into this subject at any length, but the next three figures provide a feel for the problem. One could plot component lifetime versus various thermal duty cycles, but it is more informative to plot the damage factor sustained vs component thickness, parametric in thermal parameters. For illustrative purposes these curves are all developed for a right circular cylinder of 316 stainless steel, subjected to flowing fluid continuously for 20 years, and enduring 100 transients of the type described.

The first figure, Figure 16, illustrates the dramatic effect of a change in steady state operating temperature. At a 1" thickness the damage is increased about a factor of two for an increase in steady state conditions from 1000°F to 1050°F. Because the mixed mean outlet temperature of CRBR is nominally 995°F, much of the upper internals of the reactor must operate in the 1050°F steady state region. Note that these curves apply for a ΔT of -250°F and a rate of -25°F/sec. Figure 17 shows the effect of varying the rate of the transient. As noted earlier, a variation in the rate has little effect on thick members, because all rates become effectively step changes in the thickest members. There is a significant difference in the effects of different rates in very thin members. Figure 18 brings us to the point at issue. This figure shows the effect of the magnitude of a transient on the damage for any given material thickness. For a 2" thick steel cylinder, the damage is doubled for 100 transients of 300°F magnitude as compared with 100 transients of 200°F magnitude.

The magnitude of a large fraction of the transients imposed upon the CRBR upper internals is essentially the magnitude of the core ΔT, because that is the temperature difference that collapses following a scram. Thus, based upon the data presented in this paper, one could construct a plot of breeding ratio vs. upper internals material thicknesses. But that would be extending the simplistic analysis too far, and would not serve a purpose. Instead it should be noted that the core designs which improve breeding ratio do so at a penalty of the complexity of the upper internals design, for the upper internals cannot be made of only thin members. The requirements noted earlier for the upper internals to provide a backup holddown capability, align the control rods, provide for instrumentation, and fluid mixing cannot be met with a paper thin design. These internals must survive seismic events without loss of ability to scram, must sustain rather significant hydraulic loads, and must not undergo excessive flow induced vibration. The net result is that very clever mechanical design is required. Thin members are used as thermal shields is some areas. Inconel is substituted for stainless steel in places in spite of the cost of Inconel and welds or thick sections are tucked away in relatively quiescent areas. The upper internals are thus designed to survive the duty cycle with the largest ΔT's possible without incurring excessive costs or complexity that would compromise reliability.

The facts presented here that concentrated on the upper internals are also true for other plant components. For example, although the fuel assemblies contain significantly thinner structural components, they operate at temperatures well in excess of those of the upper internals, and the fuel lifetime is also significantly degraded by higher operating temperatures and greater ΔT's. The IHX can be made smaller with high system temperatures and high LMTD's, but the design is made more difficult.

The problems imposed on the designer are severe enough to test his cleverness, but they are to a great extent self-imposed. We could back off on requirements and ease our problems; of course too much back off would make us non-competitive with other sources of power. But we have not chosen our design goals just to provide a challenge. Rather, we have selected our design challenge in this demonstration plant in order to provide design solutions that will be utilized in numerous future LMFBR's to obtain high breeding ratios, high efficiencies, and low power costs.

IV. A Burgeoning Problem Area: The Licensing Impact on Design

Clearly, design is not just the performance of calculations and assemblage of hardware. Design includes meeting performance objectives, and these include cost, schedule, safety and licensing. Cost, schedule and safety have always been essential parts of the design engineers role, but let us examine a new and growing part of his job: licensing.

The question of CRBR safety has been discussed at length (Ref. 1), and we'll not go into it here except to point out that it involves three levels: first, inherent features of the design, such as the Doppler coefficient or low-pressure coolant; second, engineered safety features, such as two independent control rod systems; third, margin for unforeseen occurrences, such as extra-strength reactor vessel and head.

These three levels of safety result in the design necessary to make the reactor safe. That is not the same as the impact of licensing on design, which is what is needed to make the reactor licenseable. The impact of safety on design has led to what we call our "Reference Design". The impact of licensing on design has led to our "Reference Design" plus a "Parallel Design".

A. Reference Design

The Reference Design is the design we described above. From the safety standpoint, the most important considerations are its reliability and its design margins.

1. Reliability. To ensure that the probability of occurrence of a hypothetical core disruptive accident (HCDA) is satisfactorily low (less than 10^{-6} per reactor - year) that it need not be considered as a design basis, adequate reliability must be demonstrated in the reactor shutdown (control rod) system and in the decay heat removal system. The CRBRP is being designed and tested to demonstrate this. Two independent and diverse shutdown systems will be employed, each of which can shut the reactor down from full power even if one control rod is stuck in the operating system. Decay heat removal is possible thru three normal heat transport systems provided with diesel-generator powered pony motors, but also capable of natural convection heat removal. These three loops are backed up by a fourth auxiliary heat removal system.

2. Design Margin

Referring to the three levels of safety cited earlier (inherent features, engineered safeguards and margin), third level design margins (TLDM) exist to provide for unforeseen occurrences. Thus, although we design to prevent a hypothetical core disruptive accident (HCDA), we provide features to cope with it: the reactor vessel and nozzle walls are thickened; the reactor closure head is thicker and bolted on; the reactor vessel support ledge is made stronger; the core support structure is stronger; seals are put in the closure head to resist impact loading.

B. Parallel Design

To satisfy the desires of the Nuclear Regulatory Commission, another design is being carried along in parallel with the preceding one. In the event that we cannot convince the licensing authorities of the reliability of our reference plant, this would exist as an option. Its primary features are a sealed head access area, a primary inlet pipe sleeve and an ex-vessel core catcher.

1. Primary inlet pipe sleeve.
   An instantaneous, double-ended rupture of the reactor vessel inlet pipe could conceivably result in core melting. The primary coolant piping entering the vessel is made of stainless steel 304, and operates at temperatures less than 800°F. Its toughness, combined with a rigorous testing program, make it apparent to us that sudden occurrence of a large rupture is incredible. Crack propagation analyses and tests are being performed to corroborate this position. If the position cannot be accepted, a parallel design has been established which can tolerate a massive pipe rupture by placing a close fitting pipe around the primary pipe.

2. Sealed head access area.
   If an HCDA were to occur, and if the closure head seals did not accommodate it (they are designed to do so), an additional containment barrier could be provided by sealing a room directly above the reactor vessel closure head.

3. Ex-vessel core catcher.

   If the three primary coolant systems did not function, and if the overflow heat removal system did not function, and if natural convection did not work, and if the core melted thru the core support structure, reactor vessel and guard vessel; an ex-vessel core catcher can be placed beneath the reactor guard vessel to catch and cool the reactor core.

C. How extensively has licensing impacted design?

This impact can be judged by comparison with the Sandia Engineering Reactor. The Hazards Summary report for the Sandia Engineering Reactor (in 1958) was 3/4" thick, and the Final Hazards Summary report (in 1961) was 1" thick. The Preliminary Safety Analysis Report (PSAR) for CRBR (in 1975) is 54" thick (with another 15" for the CRBRP Environmental Report). Based on light water reactor experience, we expected about 270 Nuclear Regulatory Commission questions on the CRBRP PSAR; so far we've received 1000 NRC questions and are still counting. Replies to the 1000 questions will require many man-years of effort. We can compare that with about ten questions on the Sandia Engineering Reactor Final Hazards Summary from the AEC's Division of Operational Safety, which took three men two nights to answer.

Many of the questions on CRBRP result because this is a first-of-a-kind plant. Future LMFBR's will be easier to license. Nevertheless, the licensing atmosphere has changed, and licensing will continue to occupy more engineering effort than was the case in the past.

D. How should licensing impact design?

We must design and license a reactor that is safe. We believe the proper way to do so is to prevent the hypothetical core disruptive accident, rather than to cope with the consequences of a HCDA. One might object - why not do both? We would answer, as engineers, that all resources are limited and resources devoted to preventing the HCDA are at least a hundred-fold more valuable than resources devoted to coping with the HCDA. The really important work for the LMFBR is the work on reliability. We must work to do everything possible to prevent the HCDA, not work on accommodating one. Thus, we at Westinghouse favor work on reliability testing of control systems and heat removal systems, plus development of inherently safe design features. We oppose programs for the construction of huge facilities devoted to assessing the consequences of a HCDA because these can only detract from HCDA prevention.

V. Summary

The present day design challenges faced by the Clinch River Breeder Reactor Plant engineer result from two causes. The first cause is aspiration to achieve a design that will operate at conditions which are desirable for future LMFBRs in order for them to achieve low power costs and good breeding. The second cause is the licensing impact. Although licensing the CRBRP won't eliminate future licensing effort, many licensing questions will have been resolved and precedents set for the future LMFBR industry.

VI. References

1. J. Graham, ANS Topical Meeting on Fast Reactor Safety in Tucson, Arizona, "LMFBR Design Bases Accident and Its Accomodations", October, 1975.
2. ASME Boiler and Pressure Vessel Code Case 1592-1.
3. WARD-D-0048, "A Simplified Cumulative Damage Evaluation of Fast Reactor Thermal Transients to the Criteria of ASME Code Case 1331-8", May, 1975.

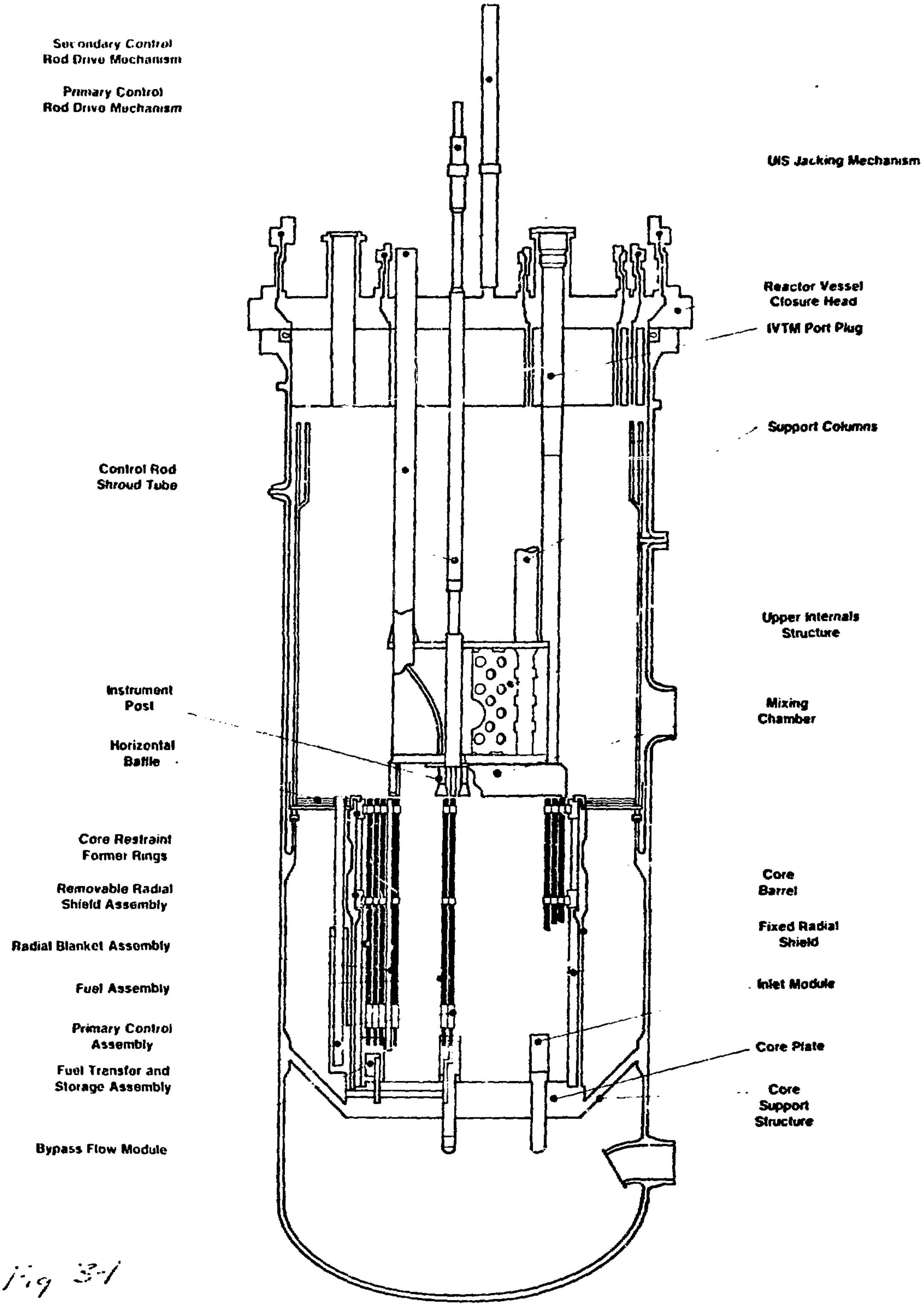

Figure 1. CRBRP Elevation

Figure 2.

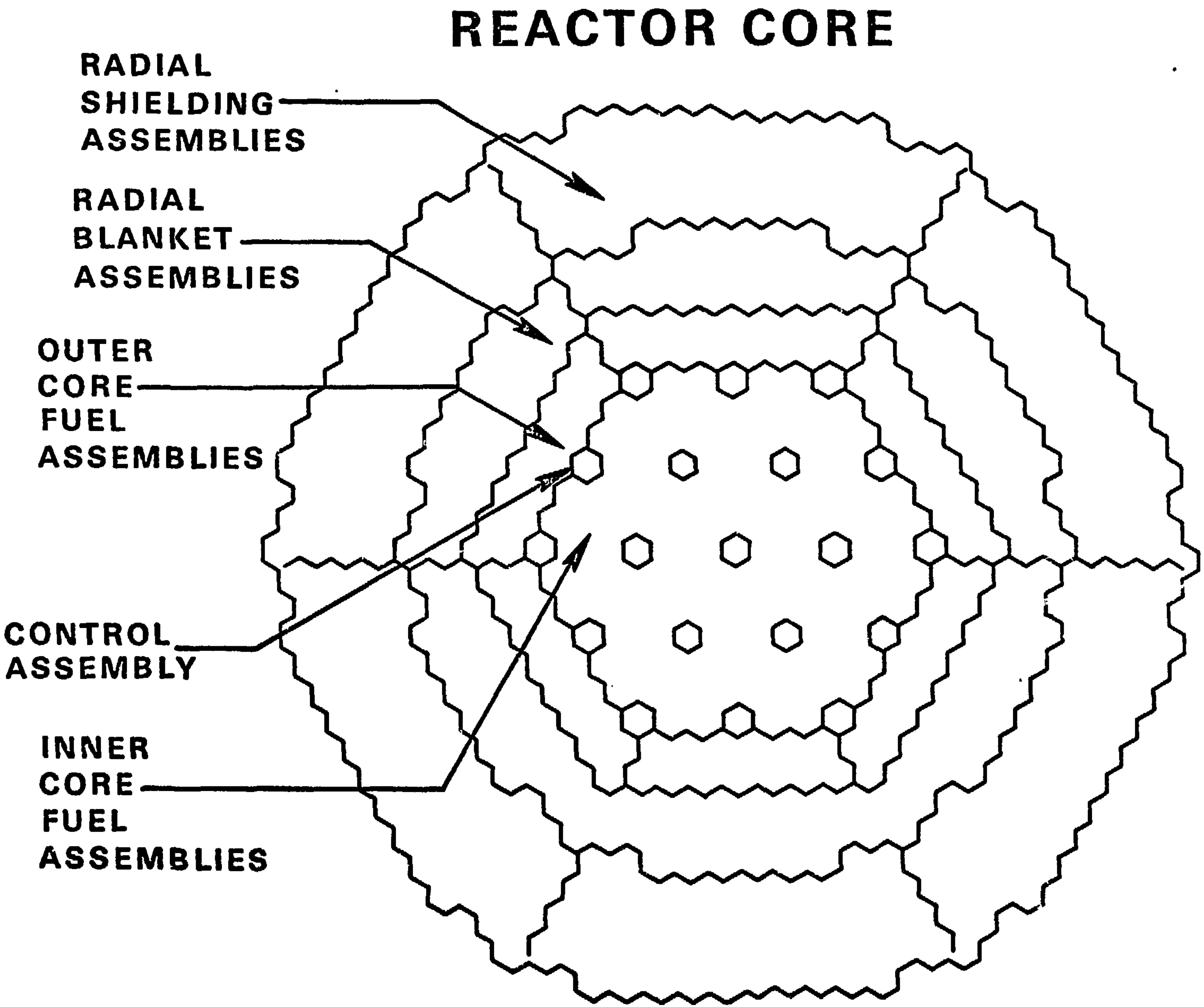
REACTOR CORE
RADIAL SHIELDING ASSEMBLIES
RADIAL BLANKET ASSEMBLIES
OUTER CORE FUEL ASSEMBLIES
CONTROL ASSEMBLY
INNER CORE FUEL ASSEMBLIES

Figure 3.

CRBRP FUEL ASSEMBLY

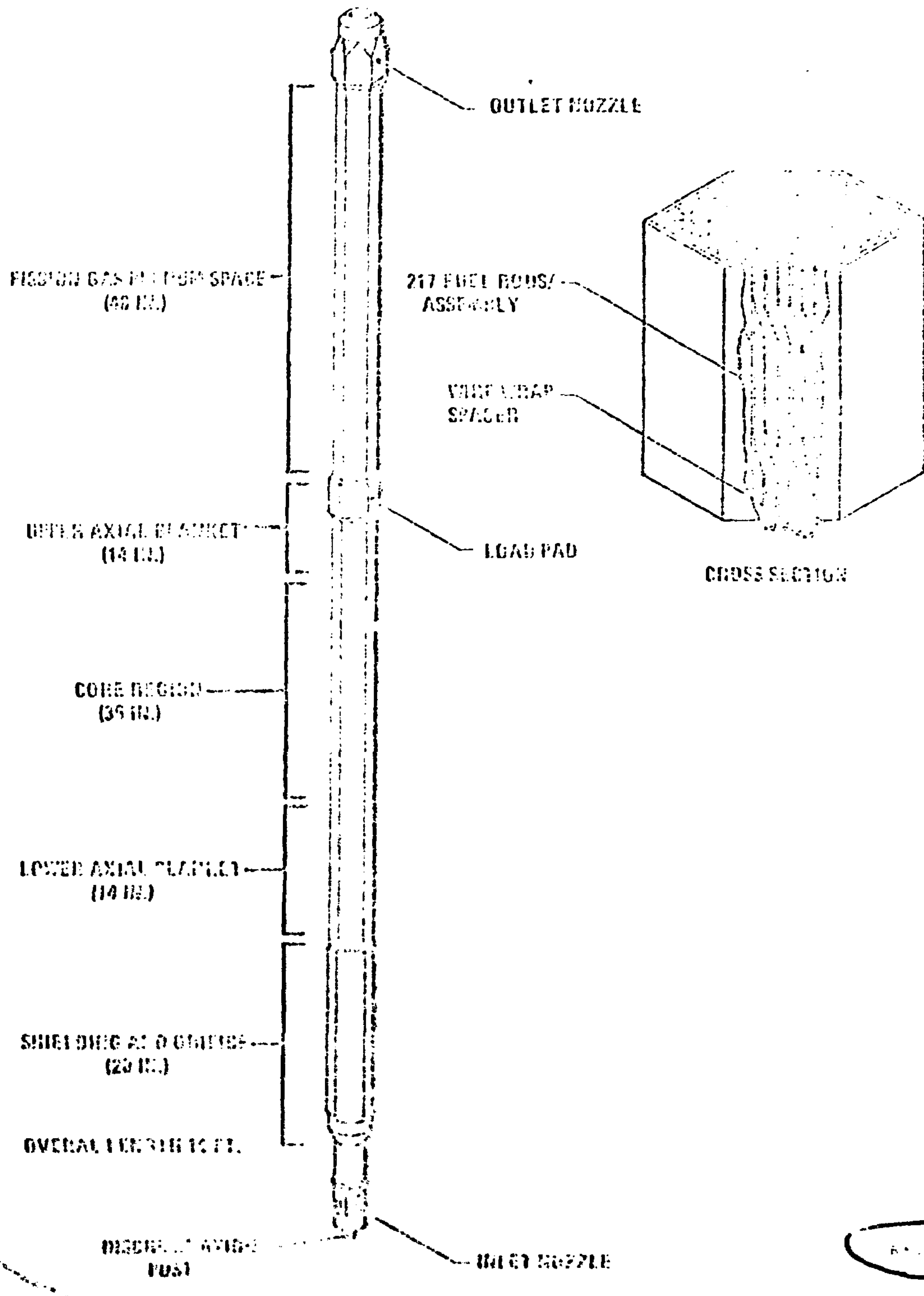

Figure 4.

CRBRP RADIAL BLANKET ASSEMBLY

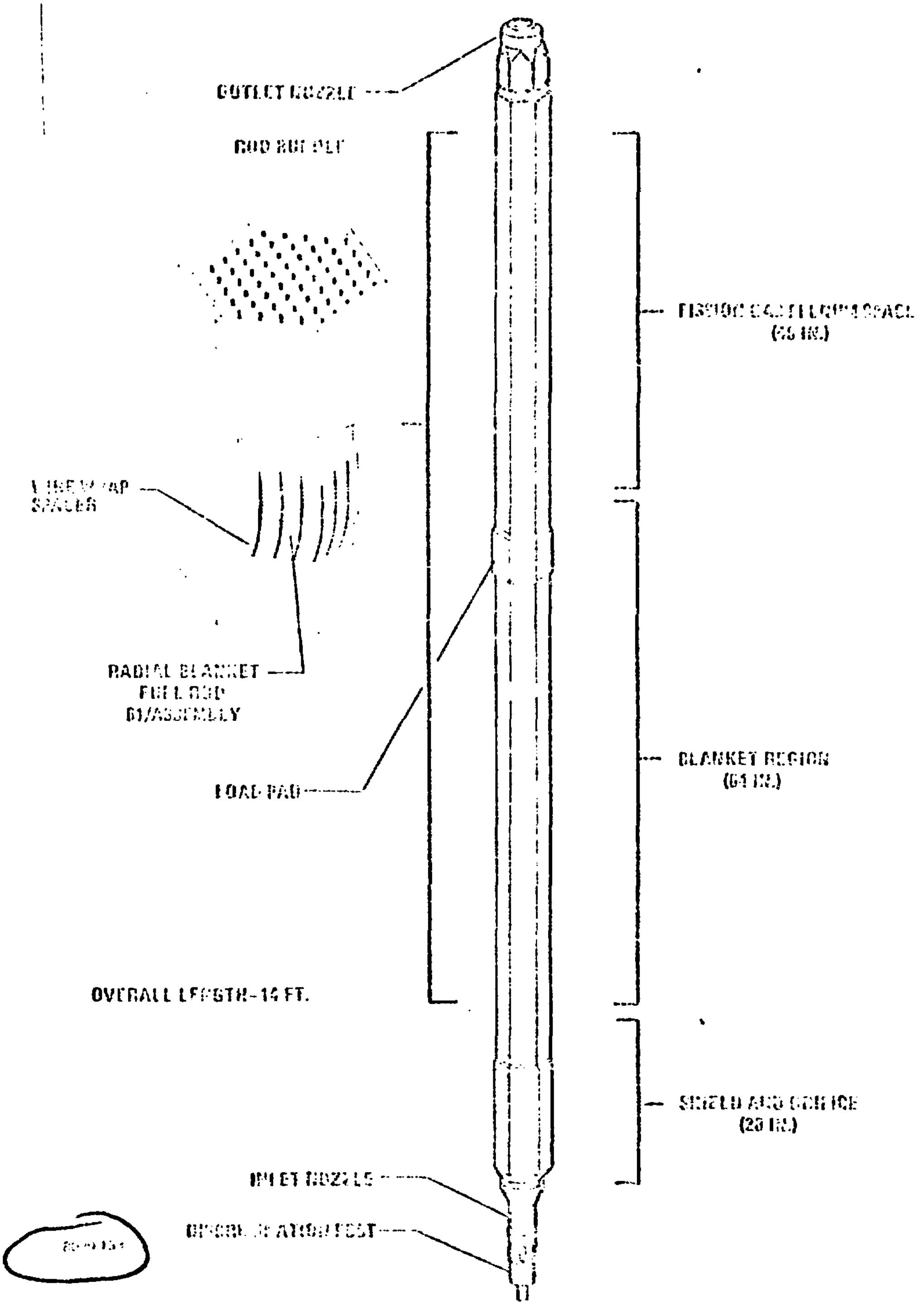

Figure 5.

PRIMARY CONTROL ASSEMBLY

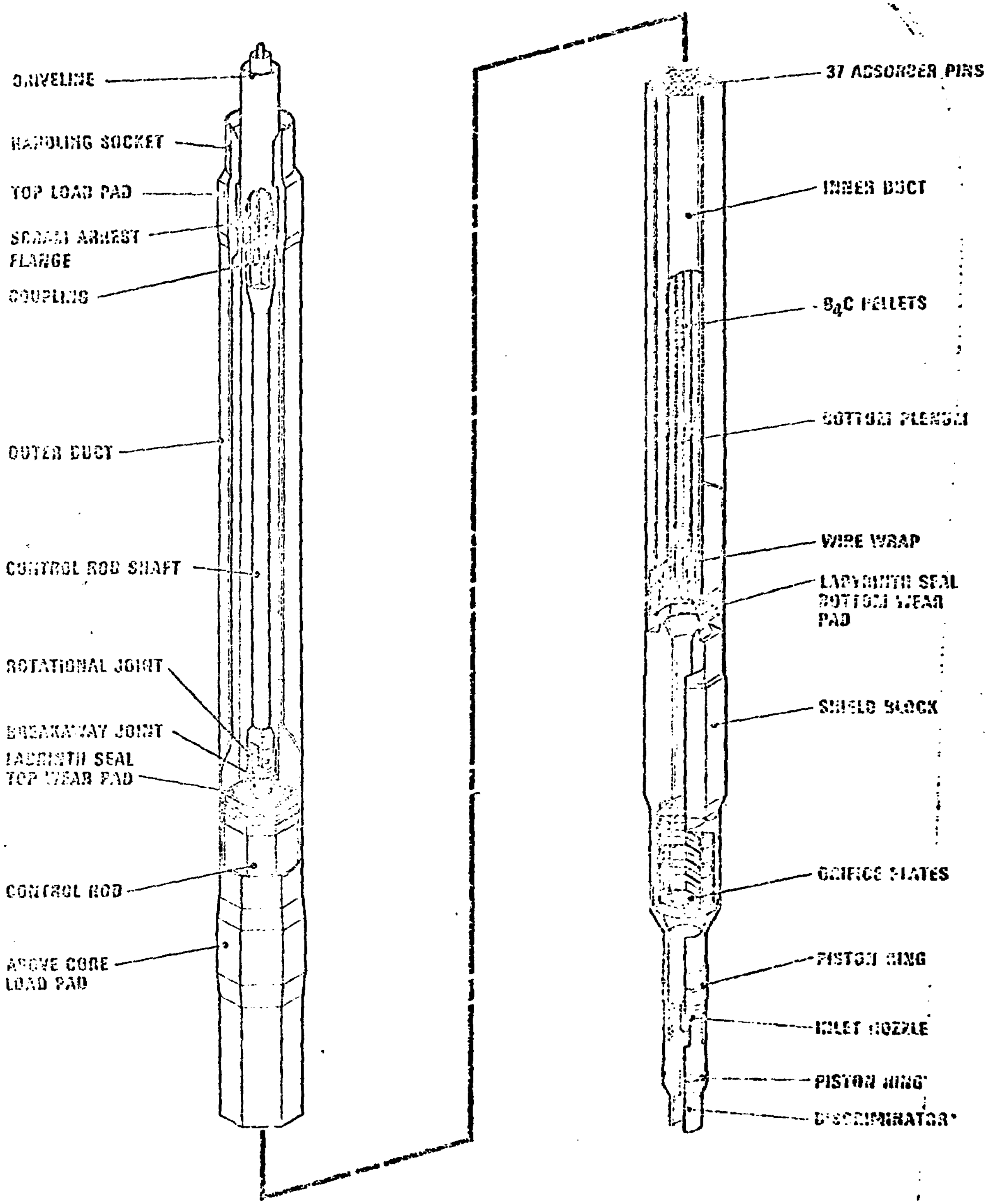

Figure 6

REMOVABLE RADIAL SHIELD

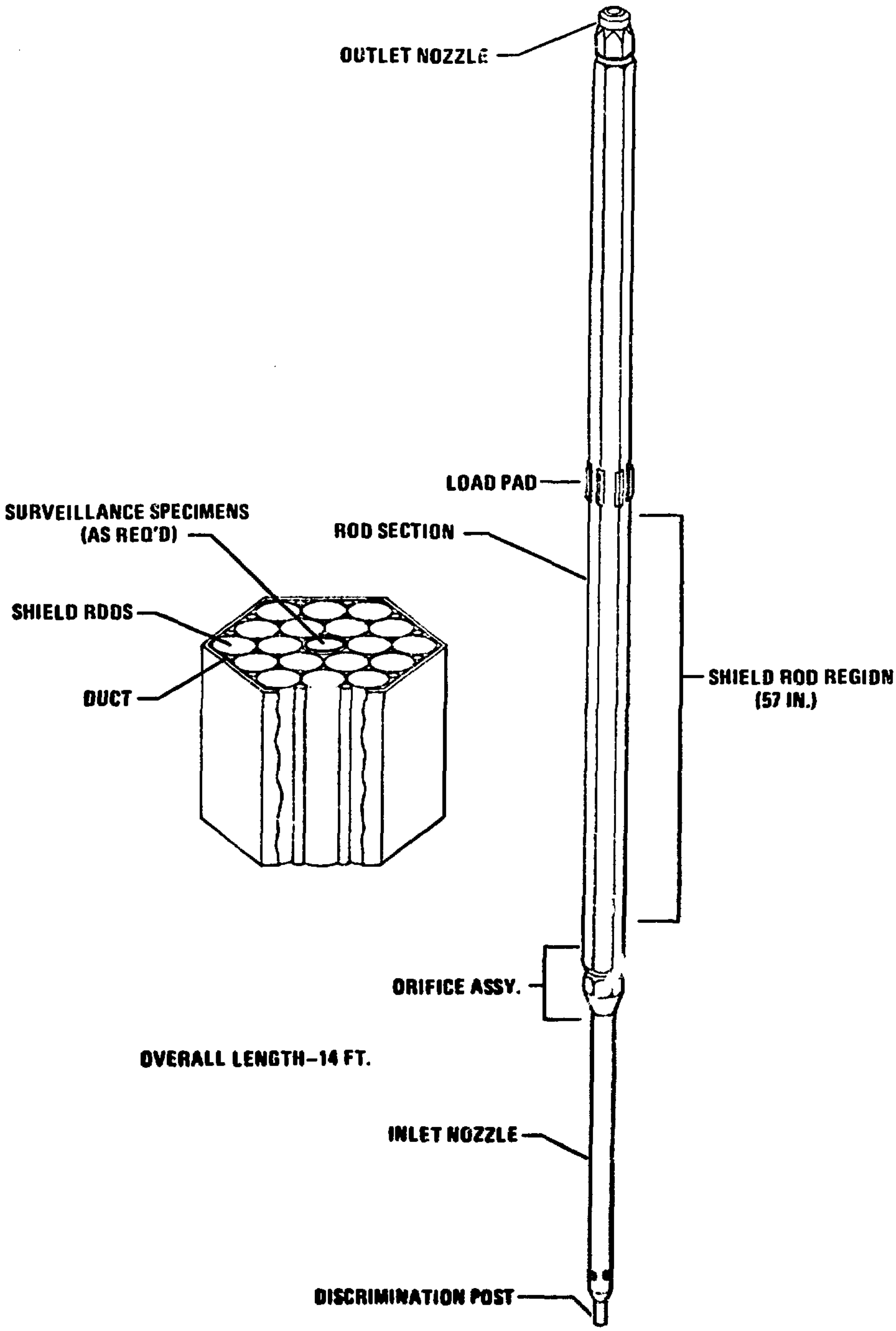

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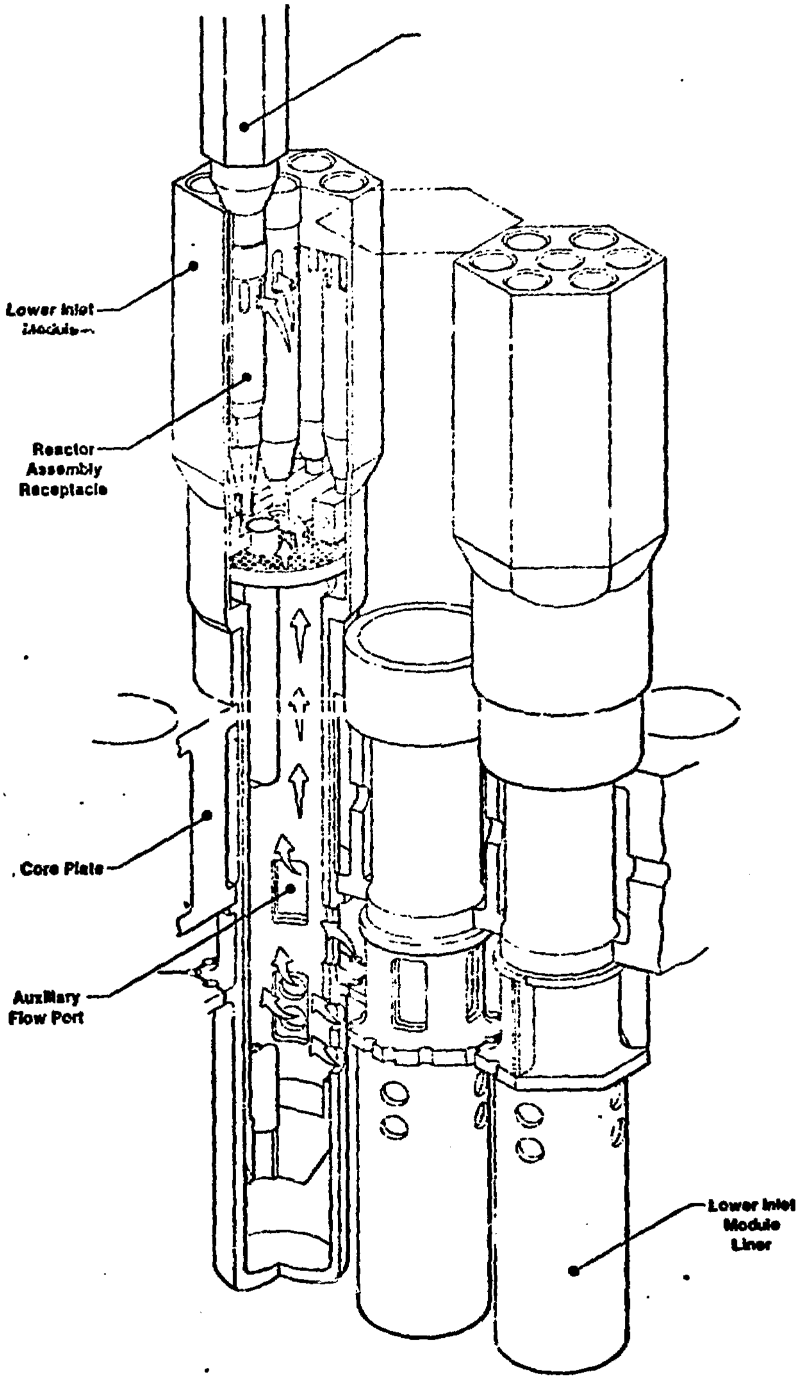
Figure 7. Peripheral Module Assembly



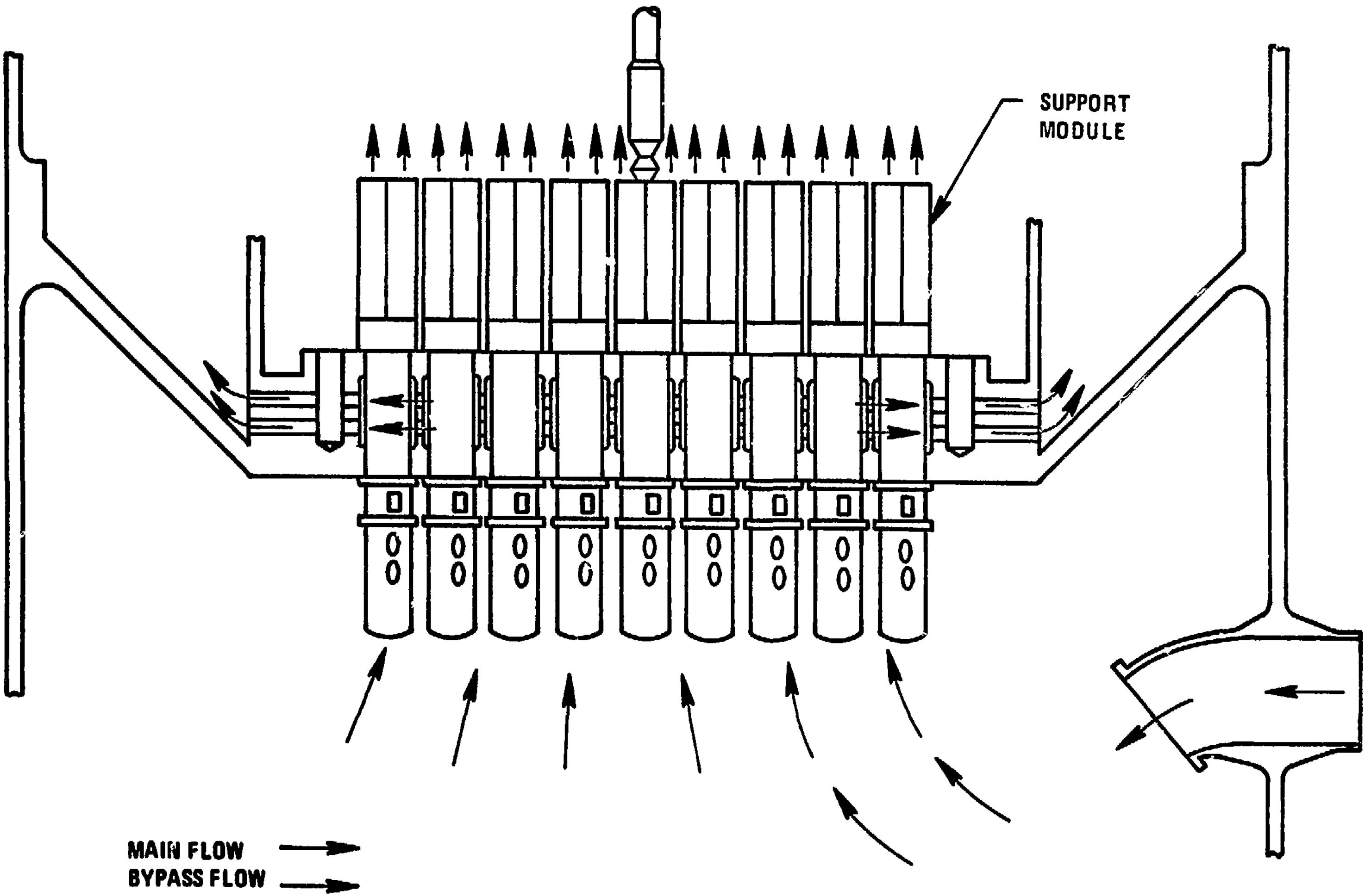


Figure 8 Modular CSS Assembly: Flow Paths

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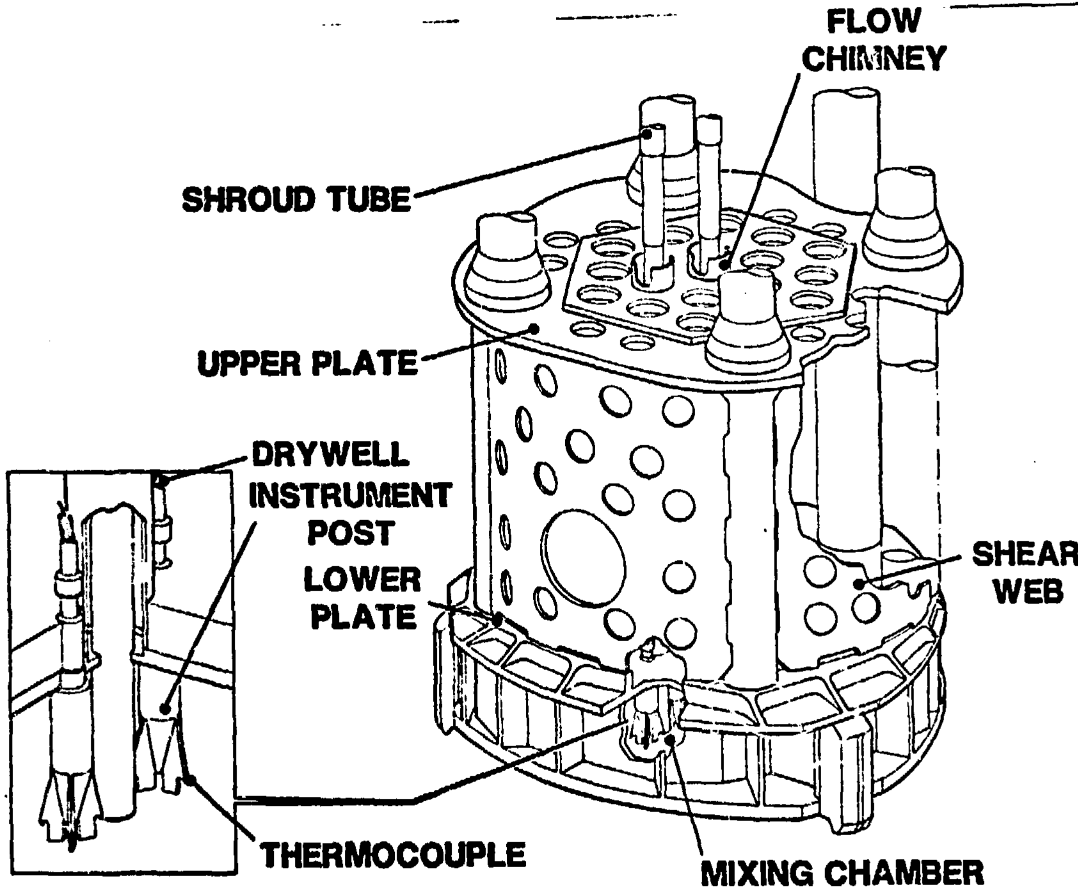

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Figure 9. Upper Internals

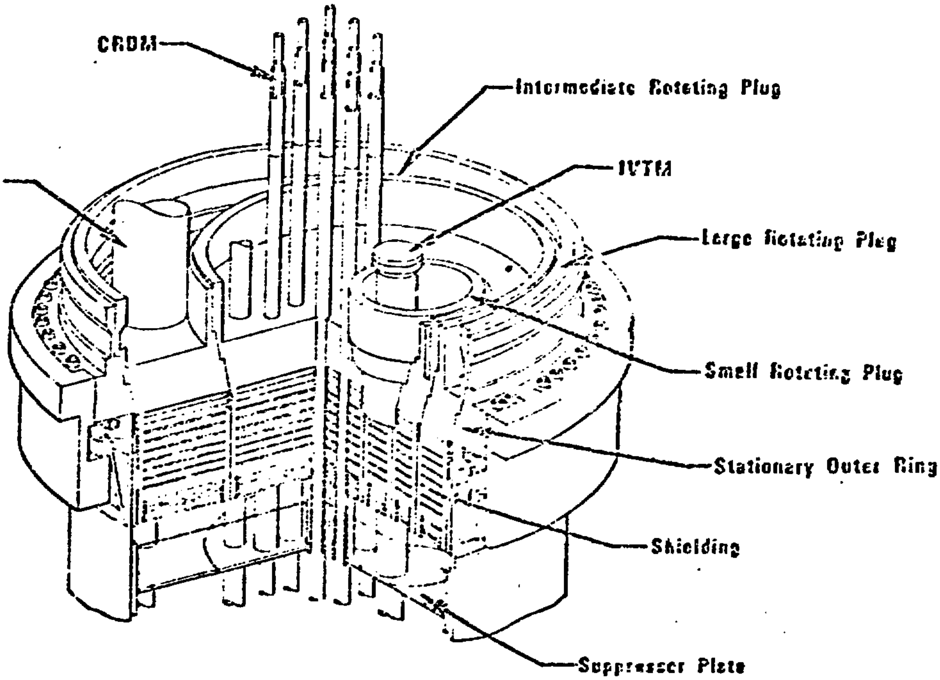


Figure 10 *Enclosure Head System*

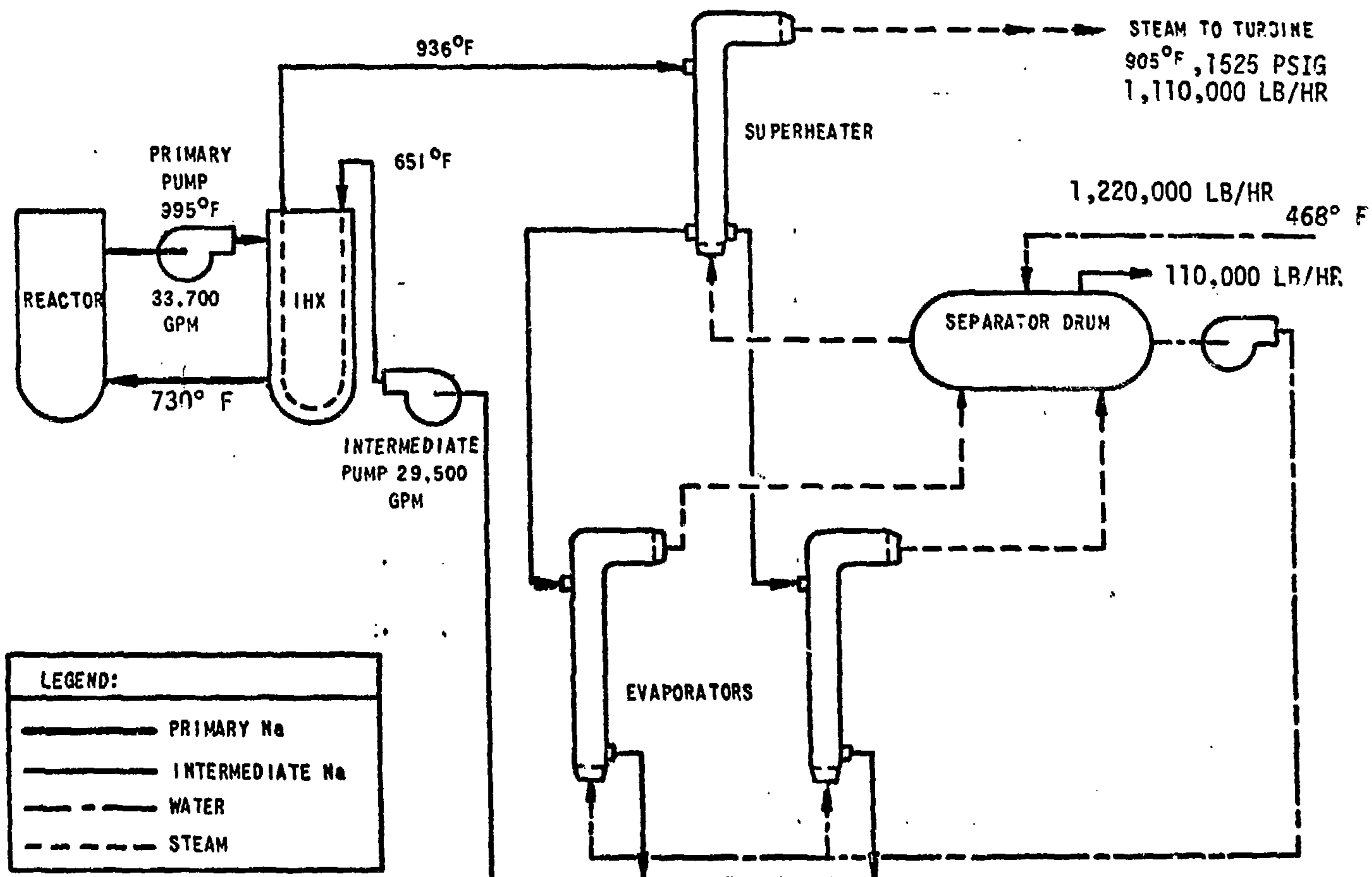


Figure 11: GENERAL CONFIGURATION OF ONE OF THREE HEAT TRANSPORT SYSTEM LOOPS.

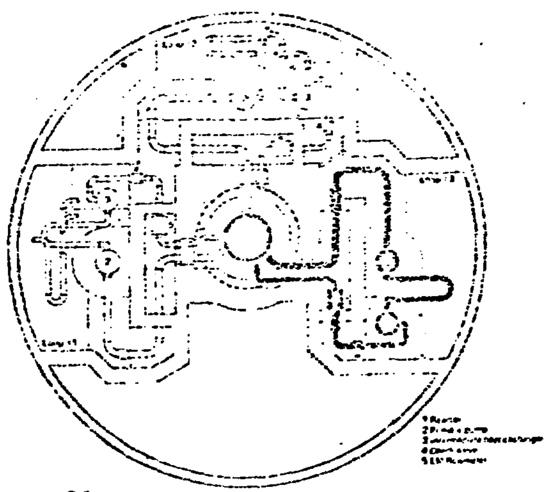
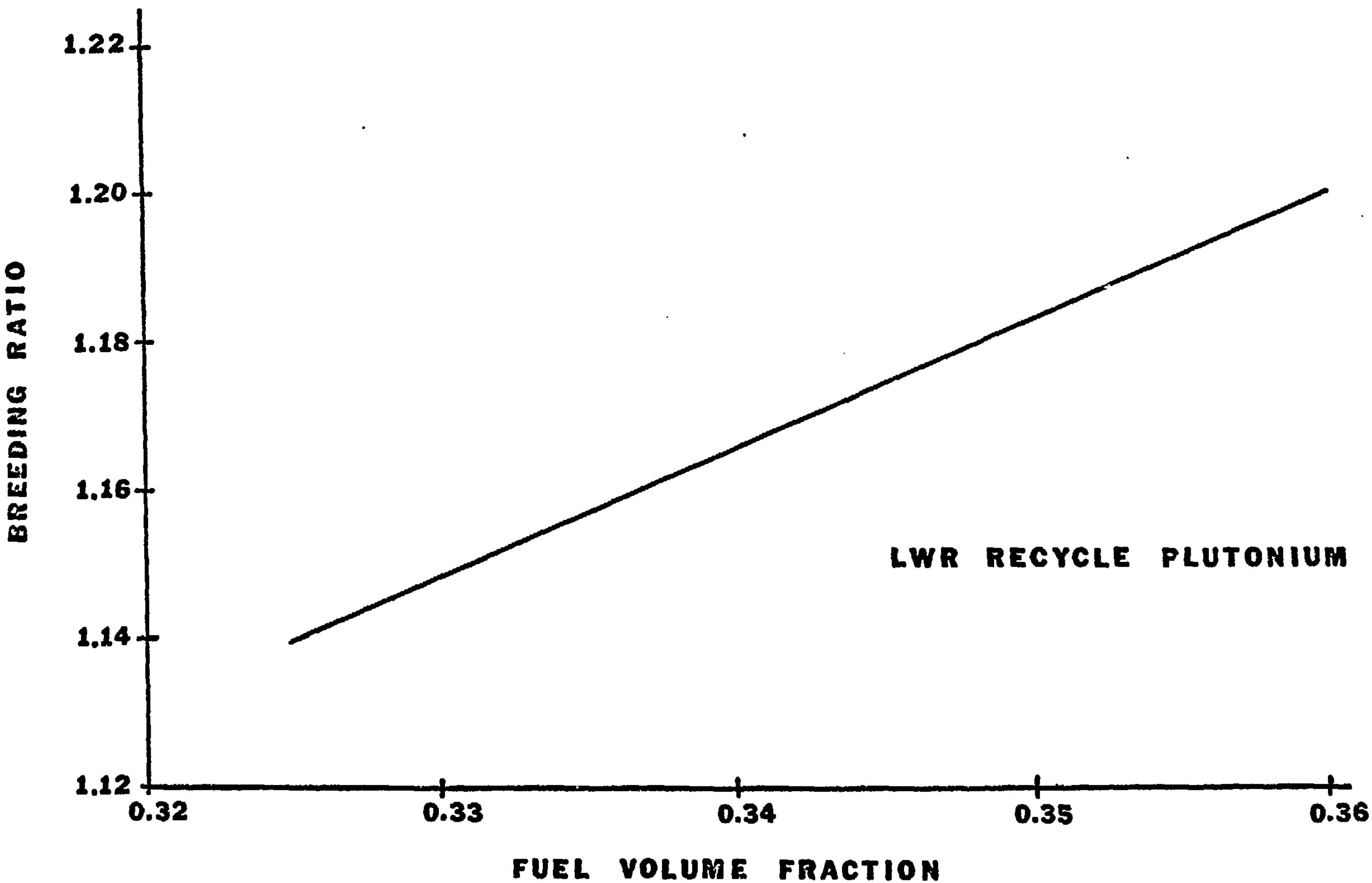
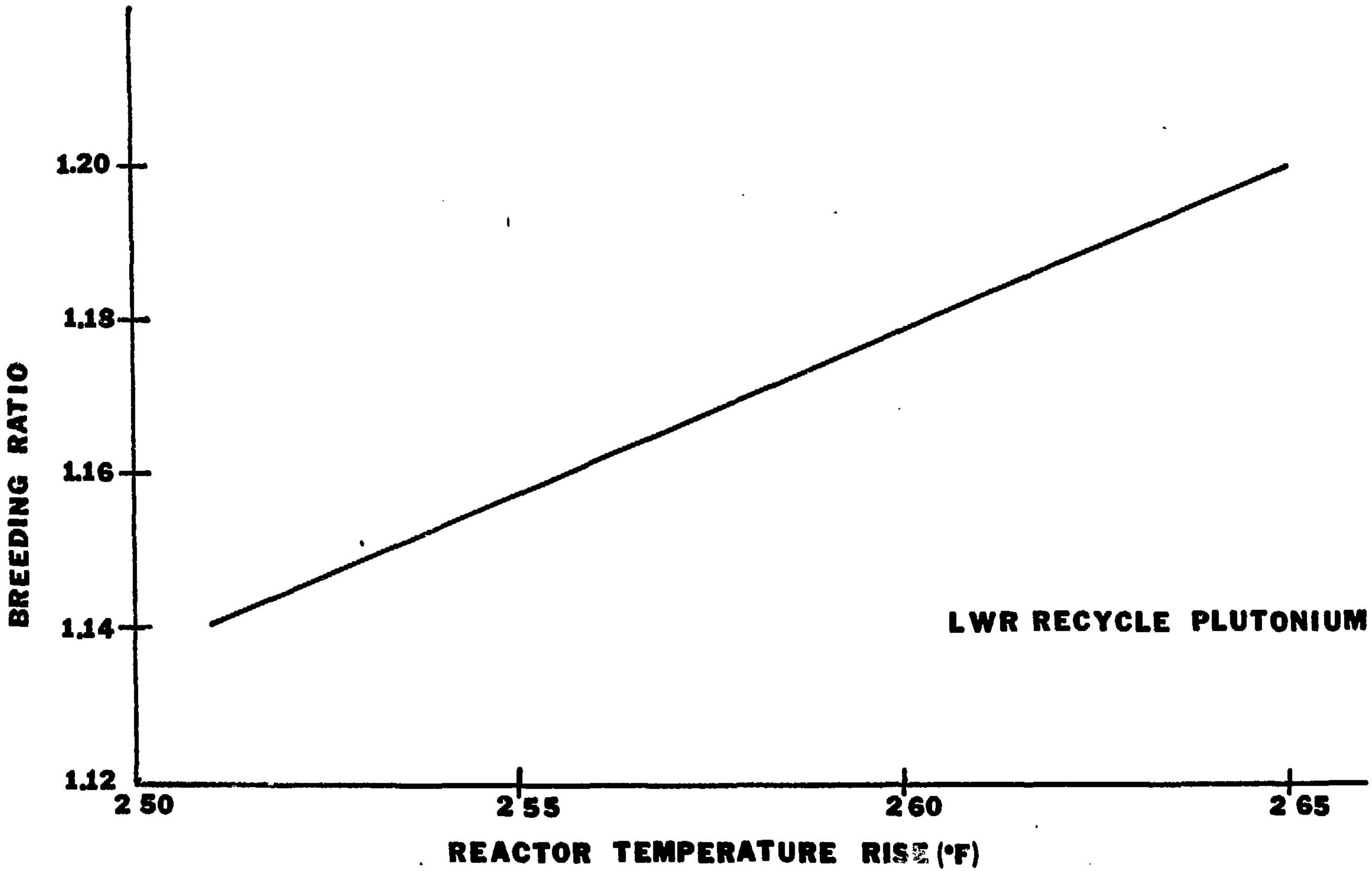


Figure 12 *Reactor Containment Building Primary Loop Layout*

Figure 13.
EQUILIBRIUM BREEDING RATIO
AS A FUNCTION OF FUEL VOLUME FRACTION



EQUILIBRIUM BREEDING RATIO AS A FUNCTION OF REACTOR TEMPERATURE RISE AT MAXIMUM PRESSURE DROP



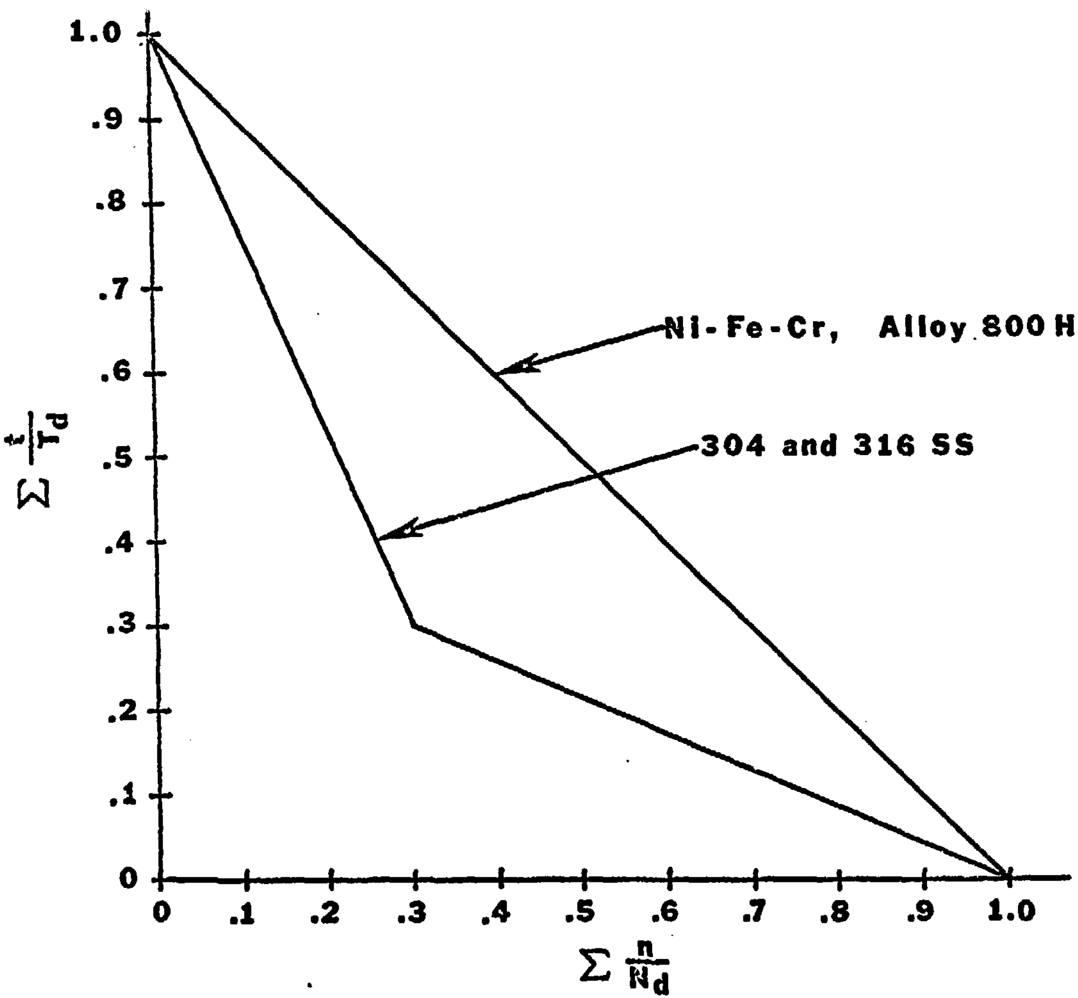


Figure 15 **Creep Fatigue Damage Envelope** (Ref. 2)

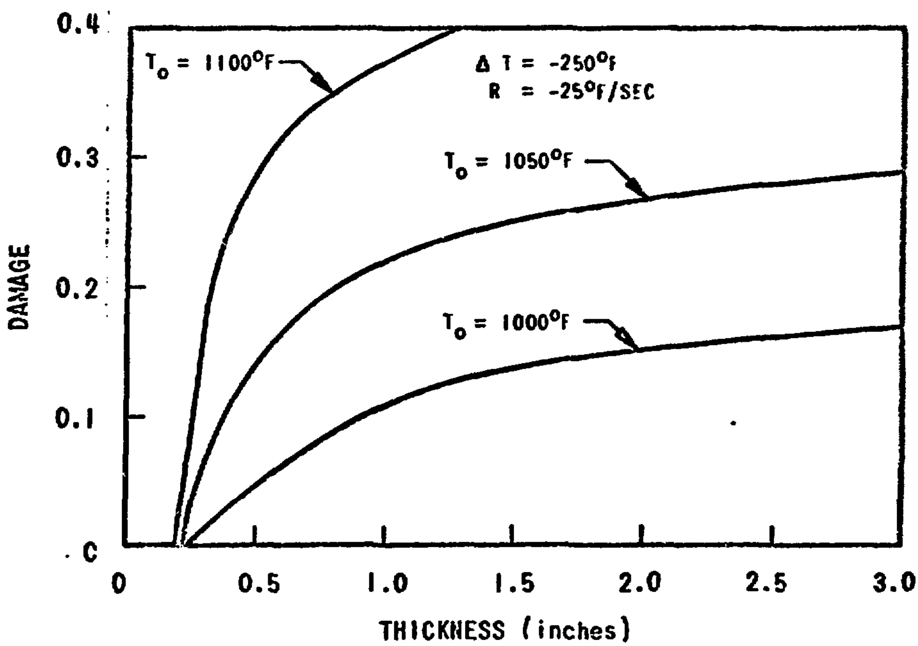


Figure 16. The Effect of Steady State Temperature on Creep/Fatigue Damage (Ref. 3)

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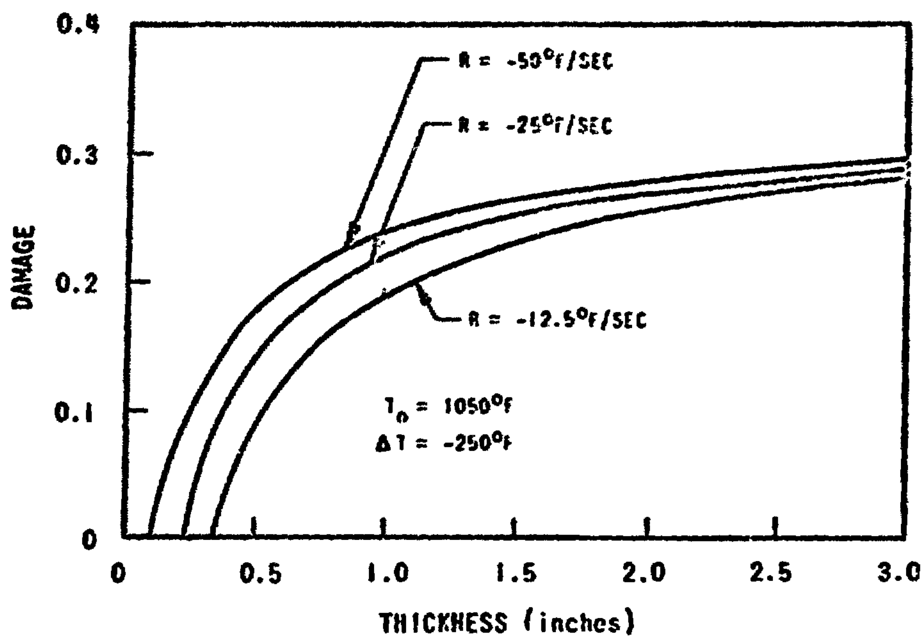

Figure 17. The Effect of Ramp Rate on Creep/Fatigue Damage (Ref. 3)

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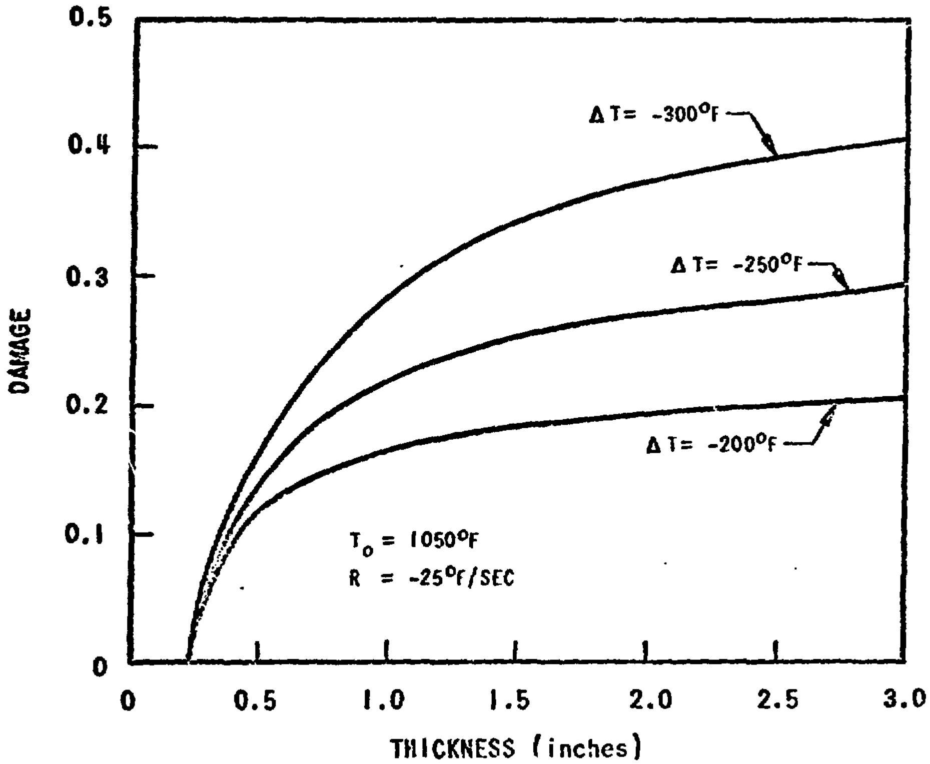

Figure 18. The Effect of Fluid ΔT on Creep/Fatigue Damage (Ref 3)