CHAPTER 1
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SUMMARY

This chapter introduces the heat transport system and associated systems by a discussion of design requirements and engineering considerations which guide the design of systems to transfer fission heat to coolant for the production of steam. This is followed by a brief description of these systems for the purpose of establishing, in the mind of the reader, some appreciation of the purpose, function, layout and size of the major components. Finally, an overview of the main features of the heat transport system is given to acquaint the reader with the dominant fluid and heat transfer processes taking place.

1. INTRODUCTION TO DESIGN AND ANALYSIS

1.1 Design Requirements and Engineering Considerations

The fissioning process results in heat generation in the nuclear fuel and surrounding media. This thermal energy can be utilized to produce electricity or process steam by the use of a heat transport medium, the coolant. Here we will discuss some of the thermal-hydraulic features which characterize the CANDU system.

The main objectives of the heat transport system are to provide heat transfer at high thermal efficiency and to allow the maximum amount of energy to be extracted from the fuel without surpassing safe limits.

The requirements for such a system can be summarized as follows:

a) Due to the decay heat produced by the fuel even when the reactor is shut-down, continuous coolant flow must be provided. This leads to the requirement for pumps, pump flywheels, standby cooling, thermosyphoning etc.

b) Costs should be minimized with due regard for the other requirements. This usually leads to trade offs between, for example, heavy water (D_20) costs, pumping power costs, equipment and piping size and costs, layout and engineering constraints.

c) Layout should minimize man-rem exposure and maximize maintainability and accessibility within the constraints of other considerations.

d) Provision must be made for pressure and inventory control of the heat transfer system. Excessively high pressure could damage the fluid boundaries (pipes, etc). Low pressure could lead to high coolant voiding and possible fuel damage and to pump damage from cavitation. Low inventory jeopardizes coolant circulation and pressure control.

e) The system must be sufficiently reliable since downtime leads to high replacement energy costs, high man-rem exposure and repair costs.

f) The design should provide high process efficiency.

g) The system should exhibit ease of constructibility to reduce initial costs and time of construction, and to enhance maintainability.

h) The system should meet and preferably surpass all safety and licensing requirements.

Various coolants can be used in the CANDU design to achieve the above objectives and requirements.

Any nuclear station design employs a tradeoff in design features to best achieve the lowest cost power within the safety limits. The U.S. nuclear industry, for instance, because of the availability of enriched uranium from existing UF_6 diffusion plants, chose to use enriched uranium and H_2O coolant. Canada, with a good supply of natural uranium but without uranium enrichment facilities, chose to enrich the H_2O coolant in order to achieve the necessary neutron economy.

From a neutron economy viewpoint, the medium surrounding the fuel, ie. the coolant and the moderator, must not absorb neutrons and must moderate the neutron energy by a minimum of collision interactions. D_2O is by far the best moderator/coolant from this viewpoint. The cost, however, is high at approximately $300/kg in 1980 dollars.

Using H_2O as the coolant, as in the CANDU-BLW, Gentilly-1, gives poorer neutron economy than the CANDU-PHW and requires booster rods for startup until the positive void coefficient of reactivity adds a sufficient positive reactivity to maintain criticality. Because of this and because of reactivity control difficulties associated with the large void coefficient of reactivity, no new commercial CANDU-BLW's are planned. Organic coolant, Monsanto OS-84, requires slightly enriched fuel (1.2 to 2.4 wt%). This option was found feasible but, due to the success of the CANDU-PHW, no commercial OCR's are planned.

Another nuclear consideration is that the coolant should have a low induced radioactivity. Both H_2O and D_2O produce N-16 and 0-19 which emit γ's in the 6-7 MeV range. This leads to reduced accessibility and maintainability while on power. The short half life (<1 minute) allows shutdown accessibility. Tritium, H^3 or T, has a 12 year half life and represents a major dose commitment for the station. Since tritium is a β emitter, the problem is one of leakage, leading to possible absorption/ingestion by humans. Organic coolant has very little induced reactivity and aids in ease of operations, accessibility, etc.

The coolants should also be stable in a radiation environment. At the high system pressure of the heat transport systems of H_2O and D_2O, radiolysis is not a problem. However, since hydrogen and deuterium have a tendency to diffuse through the pipework, the heat transport system becomes concentrated in oxygen and enhances corrosion. Supplying an excess of hydrogen or deuterium prevents this occurrence by driving the chemical equilibrium balance towards the associated state.

Organic coolant is more susceptible to radiolysis and requires degassing and makeup.

The choice of coolant also depends on other factors, such as pumping power, heat capacity, heat transfer coefficients, flowrates, pressure drop, boiling point, freezing point, corrosion, flammability, thermal stability, and cost.

Water (both D_2O and H_2O) is an attractive heat transport fluid since it offers a good balance of the above considerations. The specific heat, density and thermal conductivities are high compared to alternatives such as N_2, CO_2 and OS-84 (organic). Since pumping power is given by:

Pumping power = pressure drop x volumetric flow rate,

water requires less pumping power for a given heat removal.

For the Bruce reactors, approximately 24 MW's of pumping power are required for each reactor. Of this 24 MW, roughly 90% (or 21.5 MW) ends up in the primary heat transport system as heat due to friction. At an overall station efficiency of 30%, the net unit load for pumping power is 24 - 21.5 MW (bearing and windage losses) plus 21.5 x .7 = 15 MW (rejected energy) for a total of 18.5 MW. This represents over 2% of the electrical power generated. Since every MW saved here by reducing pumping power is gained as electrical output, considerable emphasis is placed on lowering pumping power.

Limiting flowrates for water depend on many factors such as temperature, the presence of boiling, water chemistry, geometry and flow regime. Fretting considerations have led to a 10 m/sec limit on fuel channel velocity in single phase water. Erosion/corrosion considerations have led to 4.3 to 6.1 m/s (14 to 20 ft/s) in the steam generator tubes and 16.8 m/s (55 ft/s) in heat transport piping. These limits may change as more is learned about the limiting phenomena.

The fuel distribution in the coolant is such to maximize the surface to volume ratio of the fuel so that the highest heat transfer surface can be exposed to the coolant for maximum heat transfer without drying out the fuel surface. However, if carried to extremes the fuel volume in the core will be lower than optimum and parasitic neutron absorption due to the sheath will increase. Present designs employ 37 or 28 elements in a fuel bundle. Work is under way in the nuclear industry to further optimize the design given the varying economics of fuel, D_2O, pressure drop, etc.

The use of boiling in the coolant permits higher heat transfer due to the high heat transfer coefficient of post-nucleate boiling.

Ideally, the coolant temperature should be as high as possible for maximum overall thermal efficiency. Thus a high boiling point, low vapour pressure liquid is desirable so that the heat transport system can be at the lowest possible pressure. This reduces the thickness of the pressure boundary and thus is important for reducing the parasitic burnup in the core. Organic coolant is far superior to water from this point of view and is being considered for application to the tar sands where high temperature and pressure steam is needed for oil extraction.

For the case of organic coolant, the secondary side H_2O pressure is higher than the primary side OS-84 pressure. Thus boiler tube leaks will cause a water leak into the primary coolant system.

Freezing point concerns for H_2O and D_2O are minor. For OS-84 provision must be made to prevent freezing while shutdown and cold. Continuous coolant makeup reduces this problem.

Corrosion of the heat transport system materials must be minimized because of possible deterioration, flow restrictions and contamination with active isotopes.

The CANDU-PHW heat transport system has water coolant, low cobalt carbon steel piping, stainless steel end fittings, zircalloy pressure tubes and Monel or Incoly steam generator tubes. A pH of 10.2 to 10.8 is maintained by lithium hydroxide. Hydrogen gas is added to keep the dissolved oxygen content low to help minimize corrosion. The intent is to produce and maintain a continuous and adherent film of magnetite on all the carbon steel surfaces. Corrosion with organic coolant is a lesser problem, controlled by degassing, by using N_2 cover gas, and by a dechlorinator system.

No flammability or thermal stability problems exist with water (except for the possible Zr-water reaction producing H_2 during a LOCA giving the potential for H_2 explosion) but organic coolant is combustible, although it will not sustain combustion on its own. Organic coolant is also not as thermally stable as water.

The current cost of D_2O ($\gtrsim$ \$300/kg - 1980 dollars) is high, making it the more expensive coolant. This contributes to a high capital cost for the CANDU-PHW but a low operating cost due to the efficient use of natural U.

1.2 Heat Transport Systems Description

Since the principles of all CANDU heat transport systems (HTS) are the same, the CANDU 600 MW(e) will be used for illustrative purposes. The major differences between stations is discussed in Chapter 2.

1.2.1 Fundamentals of the CANDU Nuclear Steam Supply System

The CANDU reactor is contained within a low pressure tank called the calandria (Figure 1). The fuel channel assemblies run through the calandria and contain the bundles of natural uranium fuel. The calandria is filled with heavy water (D_2O) which moderates or slows the fast neutrons, making a chain reaction possible. The heat of fission generated within the fuel is removed by the pressurized heavy water coolant which is pumped through the fuel channels. This hot coolant is passed through the steam generator where heat is transferred to light water to generate steam.

The pressure tube forms the pressure boundary of the heat transport system (Figure 2); the heavy water coolant passes through and around the bundles of natural uranium fuel located within the pressure tube. The calandria tube is in contact with the moderator. The annular space between the pressure tube and the calandria tube provides thermal insulation between the hot heat transport system coolant and the cool moderator.

The portions of the fuel channel assemblies external to the calandria (Figure 3) are known as the end fittings; the end fittings have connections to the feeders which feed coolant into and out of the fuel channels.

1.2.2 Heat Transport System (HTS)

The CANDU 600 MW(e) reactor has 380 fuel channels arranged in a square array within the calandria. The heat transport system is arranged into two circuits, one to each side of the vertical centre line of the reactor core, with 190 fuel channels in each circuit.

The circuits are shown in Figure 4; each circuit contains 2 pumps, 2 steam generators, 2 inlet headers and 2 outlet headers in a 'figure-of-eight' arrangement. Feeders connect the inlet and outlet of the fuel channels to the inlet and outlet headers, respectively. The flow through the fuel channels is bidirectional (i.e., opposite directions in adjacent channels).

The arrangement of the HTS within the reactor building is illustrated in Figures 5 and 6. The steam generators, HTS pumps and headers are located above the reactor; this permits the heat transport system coolant to be drained to the header elevation for maintenance of the HTS pumps and steam generators, and also facilitates thermosyphoning (natural circulation) when the HTS pumps are unavailable.

A typical HTS pump is shown in Figure 7 and a typical steam generator is shown in Figure 8.

1.2.3 Pressure and Inventory Control System

The inventory of the HTS (Figure 9) is controlled by 'feeding' D_2O into, or 'bleeding' D_2O out of, the HTS system. At power, the HTS pressure is controlled by a pressurizer connected to the two HTS circuits. Heat is added to the pressurizer via electric heaters to increase pressure and is removed via steam bleed to reduce pressure. The inventory control system can also provide pressure control at low power (less than 5%) when the pressurizer may be isolated. The pressurizer also serves to limit the magnitude of HTS pressure transients by supplying coolant to the heat transport system when pressure is decreasing. Typical heat transport system transients are discussed later in this chapter.

Valves that discharge D_2O from the heat transport system (HTS relief valves, pressurizer steam bleed valves and relief valves) connect to the degasser condenser. The relief devices of the degasser condenser are set above the normal HTS operating pressure, thereby limiting the discharge of D_2O from the HTS in the event that any of PHTS or pressurizer discharge valves fail open.

1.2.4 Shutdown Cooling System

The shutdown cooling system (Figure 10) can be utilized to remove decay power following a reactor shutdown. Two independent shutdown cooling system circuits are provided, one at each end of the reactor core. D_2O is taken from the outlet header, passed through a pump and heat exchanger, and returned to the inlet header. Since there are no valves in the HTS circuits, a portion of the shutdown cooling system flow passes from the outlet header to the inlet header via the steam generators. The shutdown cooling system can also be operated utilizing the HTS pumps; in this mode of operation, the shutdown cooling system flow bypasses the shutdown cooling system pumps. This system is also effective with the HTS depressurized and the D_2O level lowered to the elevation of the headers; this facilitates maintenance of the steam generators and the HTS pumps.

1.2.5 Heat Transport System Purification

The accumulation of active materials in the CANDU HTS is inherently very low. This is primarily due to restrictions placed on materials used in the HT system (for example, very low cobalt levels are permitted), and the absence of failed fuel during reactor operation (in the event fuel failures do occur, they are detected and removed). To further minimize the accumulation of active deposits within the HTS, the coolant is continuously filtered and purified. The head of one HTS pump in each circuit is utilized to provide a flow of HTS coolant through the purification system (Figure 11). An intercooler is utilized to minimize heat losses. Flow through the filters and ion exchange columns is cold and pressurized.

1.3 Main Process Features

The previous sections laid out the purpose of the HTS along with some schematics of typical designs indicating the main components and their functions. Also described were the desirable properties of these components. This section lays out the fundamental principles governing the mass and heat transfer, setting tne scene for subsequent detailed investigations.

Performing an energy balance around the reactor, the energy out of the reactor equals the energy going in plus the reactor energy generation. Thus:

$$\dot{M}\, h_o = \dot{M}\, h_i + Q \text{ or } Q = \dot{M}(h_o - h_i), \tag{1}$$

where $\dot{M}$ = coolant mass flowrate (kg/s);
h_o = core exit enthalpy (kJ/kg);
h_i = core inlet enthalpy (kJ/kg);
Q = reactor power transferred to the coolant (kJ/s or kW).

Neglecting minor factors such as pump heat, piping heat losses, pump gland seal leakage and miscellaneous heat losses via auxiliary systems, the power transferred to the steam generator is Q kW. The heat transfer at any point in the steam generator is given by Newton's law of cooling:

$$dQ = U\,(T_p - T_s)\,dA \tag{2}$$

where U = overall heat transfer coefficient (kJ/m^2 °C),
A = heat transfer area (m^2),
T_p = primary (D_2O) side temperature (°C),
T_s = secondary side (H_2O) temperature (°C).

U is a function of flow, temperature, the amount of boiling (quality), the physical layout, heat exchanger tube material and the degree of crudding or fouling in the steam generator. Thus the total heat transfer is

$$Q = \int_Q dQ = \int_A U(T_p - T_s)\,dA \tag{3}$$

However the D_2O and H_2O temperatures are not constant throughout the steam generator. A schematic representation of the variation is shown in Figure 12.

Using the 600 MW CANDU as an example, demineralized feedwater (H_2O) enters the preheating section of the steam generator at roughly 175°C and gains heat from the exiting D_2O (∿265°C at ∿5 MPa) and the H_2O begins to boil. The temperature then remains essentially constant as the H_2O travels through the boiler (left to right in Figure 12). The D_2O (primary fluid) enters the boiler section of the steam generator at roughly 310°C at 10 mPa with 4% quality (i.e., 4% by weight of steam). The heat transfer to the secondary side condenses the steam and the temperature subsequently drops as the D_2O travels through the steam generator tubes (right to left in Figure 12).

For the purposes of discussion, we will simplify equation 3 by assuming a temperature distribution as shown in Figure 13. Thus we have ignored the preheating section (where the H_2O temperature is less than saturation) and have assumed that no boiling occurs on the primary side. Further we assume that U is constant. These are crude approximations but adequate for discussion purposes. Thus, equation 3 becomes:

$$Q = UA \left[\frac{T_o + T_i}{2}\right] - T_s \qquad 4$$

This can be related to enthalpy by noting that

$$h \simeq C_p T + \text{CONSTANT}, \qquad 5$$

where C_p is the heat capacity of water, Equation 4 then becomes:

$$Q = \frac{UA}{C_p} \left[\frac{h_o + h_i}{2}\right] - h_s \qquad 6$$

if we assume the same properties for H_2O and D_2O.

A final primary heat transport system relation is needed to complete this approximate picture. The primary side flow is determined by a balance between the head generated by the primary pumps and the circuit head losses due to friction.

$$\Delta P_{pump} = A_o + A_1\dot{M} + A_2(\dot{M})^2 + \ldots = \Delta P_{circuit} = K(\dot{M})^2 \qquad 7$$

where K can be a complex function of material properties and pipe geometric details. Typical shapes for Equation 9 and 10 are shown in Figure 14. The intersection of the two curves is the operating point, equation 7.

The primary heat transport approximate conditions are set, then, by the simultaneous solution of the energy balance at the core, the energy balance at the steam generator and the momentum balance around the circuit.

Equations 1 and 6 can be rearranged (eliminating h_o) to give:

$$h_i = Q/\dot{M}\left[\frac{C_p\dot{M}}{UA} - \frac{1}{2}\right] + h_s \ . \qquad 8$$

Thus we see that since all parameters, Q, $\dot{M}$, C_p, A, U, etc., are positive quantities, the reactor inlet enthalpy (and hence the inlet temperature will rise up as flow rises, will rise as secondary side temperature and enthalpy rise and may go up or down as power changes.

The reactor outlet enthalpy, h_o, is directly related to h_i by equation 1. Thus:

$$h_o = Q/\dot{M} + h_i = Q/\dot{M} \left[\frac{C_p \dot{M}}{UA} + \frac{1}{2}\right] + h_s \qquad 9$$

The average enthalpy in the core and the steam generator is:

$$\bar{h} = \frac{h_o + h_i}{2} = (Q/\dot{M}) \left(\frac{C_p \dot{M}}{UA}\right) + h_s = \frac{QC_p}{UA} + h_s \qquad 10$$

This result is worth remarking since it shows that $\bar{h}$ is not a direct function of flow. Given h_s, C_p/UA as fixed for a given secondary side temperature and steam generator geometry, $\bar{h}$ is a simple linear function of the reactor power, Q. Figure 15 illustrates this point and also shows the spread or variation in h about $\bar{h}$ given by:

$$h_o - \bar{h} = Q/\dot{M} \left(\frac{C_p \dot{M}}{UA} + \frac{1}{2}\right) + h_s - \frac{Q}{\dot{M}} \frac{C_p \dot{M}}{UA} - h_s = Q/\dot{M} \qquad 11$$

Similarly,

$$\bar{h} - h_i = Q/\dot{M}. \qquad 12$$

From equation 12 we see that the primary side enthalpy floats on top of the secondary side with just enough Δh to transfer Q kW of power. Also, given a rough estimate of flow for the calculation of U (not a strong function of flow since $\dot{M}$ is large and turbulent - most of the resistance to heat transfer is due to conduction through the tubes and crud layer) we can calculate $\bar{h}$ and estimate the spread in h ($\bar{h} \pm Q/2\dot{M}$). This gives a good first estimate of the temperatures and enthalpies and indicates whether boiling will occur in the primary circuit or not. With this enthalpy, temperature and hence density estimate, the circuit losses can be calculated and compared to the available pump head at that flow. The flow estimate can be updated and the whole prodcedure repeated until convergence is reached. A sample calculation follows.

1.4 Sample Heat Balance for CANDU 600

Parameters:

$Q \simeq 2000$ MW(th) $= 2 \times 10^6$ kW(th) (given)
$\dot{M} \simeq 8000$ kg/s (guessed)

$T_s \simeq 265°C$ (given)
$C_p \simeq 4.25$ kJ/kg (guesses) $\Bigg\} \Rightarrow h_s \approx 1125$ kJ/kg
$U \simeq 21.25$ kJ/s°C m^2
$A \simeq 3200$ m^2 $\Bigg\} \Rightarrow \frac{UA}{C_p\dot{M}} \simeq 2$ (guessed)
P_{ROH} = 10 MPa (given)

Thus from equation 8

$$h_i = 250\ [\frac{1}{2} - \frac{1}{2}] + h_s = h_s = 1125 \text{ kJ/kg.} \qquad 13$$

and

$$h_o = h_i + Q/\dot{M} = h_i + 250 = 1375 \text{ kJ/kg}. \qquad 14$$

The saturation enthalpy at the outlet header is roughly 1370 KJ/kg. Hence our prediction of the primary outlet conditions is that the D_2O should just come to saturation. In fact, the detailed design calculations give the outlet quality at 4% with an enthalpy of ~1415 KJ/kg.

From Equation 8

$$\frac{\partial \dot{M}}{\partial h_i} = \frac{Q}{2(h_i - h_s - \frac{QC_p}{UA})}2 = \frac{Q}{2(Q/2\dot{M})}2 = \frac{2\dot{M}^2}{Q}, \qquad 15$$

and

$$\frac{\partial(\dot{M}/\dot{M}_o)}{\partial(h_i/h_{io})} = 9.0\ . \qquad 16$$

Thus if we had chosen h_i to start our iterative calculation, and our guess was in error by 25% the flow will subsequently be in error by 9.0 x 25% or 225%. Thus huge swings in estimated flow will accompany the search for the right h_i.

If, instead, we guess at flow and are out by 25%, the h_i calculated will be out by only 2.8%. Convergence will, thus, be much better behaved. Figure 1.16 illustrates the calculational procedure.

The proper iterative procedure, then, would take full advantage of these sensitivities. The key parameters are fixed or guessed: Q is usually given, M is guessed from, say, a single phase circuit loss calculation at any reasonable enthalpy, U is calculated based on empirical correlations, A is usually given and T_s is usually given. The enthalpies can then be readily calculated. Depending on the nature of the correlations used for U, the iteration may involve an inner loop on U and h to converge on a self-consistent heat transfer given the flow. The flow, then, is updated based on the circuit loss calculation until convergence is reached.

1.5 Transient Behaviour

Consideration of transient behaviour of the Heat Transport System is essential since the system is composed of many interdependent items which must be modelled simultaneously for efficient design. The increasing emphasis upon transient and safety related analyses dictate the use of a comprehensive model such as SOPHT, incorporating the many facets of a nuclear generating station.

It is instructive to illustrate transient behaviour by looking at a few transients using Darlington A as an example.

(a) A typical transient, say a reactor trip, starting from the steady state at 100% full power (F.P.) is caused by the rapid insertion of the shutdown control rods in the reactor core. The cause of the trip could be a real or spurious signal to the control system.

The resulting system transient is shown in Figure 17. The neutron power drops to decay heat levels within 2-4 seconds. The power to coolant drops more slowly due to the heat transport lag. The power drop causes a depressurization of the primary circuit and consequently a large flow from the surge tank. As the PHT system pressure drops, so does the temperature of the D_2O. This causes a boiler cooldown and depressurization as shown. Finally the governor valve closes down in an attempt to maintain the boiler pressure at the setpoint. This transient gives insight into the size of the pressurizer and pressurizer interconnect piping needed to keep the HT system pressure high enough to prevent the main pumps from cavitating.

(b) A turbine trip is initiated by a real or spurious signal, indicating an inability of the electrical generator, the electrical grid or the turbine to accept the steam power. The trip signal, in this case, closes the emergency stop valves and causes the transient shown in Figure 18. The trip signal automatically initiates a reactor stepback, i.e., a reduction in power. The primary system pressure drops due to the power reduction while the secondary flow reduction causes an increase in boiler pressure. The surge tank provides an outsurge to relieve the pressure drop in the primary system while the boiler safety and steam discharge valves relieve the high boiler pressure. This case gives information on relief valve sizing for the secondary side as well as giving the consequences of the control philosophy used.

(c) The rapid cooldown of the boiler is initiated by tripping the reactor and opening the discharge valves. This results in a rapid depressurization, and hence cooldown of the boiler. The transient, as shown in Figure 19, is used in the stress analysis of the boiler, as are the other transients.

(d) An example of manoeuvring is given in Figure 20. Here, starting from an 80% full power steady state solution, the reactor is ramped up in power at the specified rate of 0.8%/sec as demanded via the demand power routine (DPR). The resulting PHT pressure transient proved too much and the control scheme automatically initiated a reduction in reactor power or stepback. Such a result indicates that the rate of 0.8%/sec is too high for the system configuration studied.

(e) The final example, loss of Class IV power (the main electrical grid), represents an operational transient needed for the stress analysis of the steam generators. Figure 21 shows, as a function of time, two of the system parameters affecting the stress analsyis. The loss of Class IV power occurs at time t=0 and causes a loss of power to the main heat transport pumps, feedwater pumps, reheater drains pumps and turbine condenser cooling water pumps. This immediately initiates a trip signal to the turbine governor valves. The subsequent sequence of events is shown in the figure. The loss of the turbine causes a fast controlled power reduction stepback, to 60% full power. This signal clears at t=2 seconds, but a high drum pressure causes a further slow reduction in power via a setback, and low primary flow finally trips the reactor. The use of the resulting transient in the stress analysis may lead to the conclusion that the transients are unacceptable. In that event, the action taken may be to resize the pressurizer interconnect lines.

1.6 Critical Heat Flux, CHF

As a fuel bundle power is raised, the closer the bundle comes to a heat transfer crisis. This critical heat flux represents an upper bound for design purposes. Figure 22 shows typical CHF data as a function of quality. Implicit in this data is a functional dependence on pressure and flow. For low power, high quality situations (as might occur under low flow conditions), dryout of the fuel pencil surface occurs. The surface temperature rises dramatically and the fuel sheath may melt, causing fuel failure. For high power, low quality situations (as might occur for over-power conditions), fuel centerline melting occurs first, before surface dryout. In that event, fission product release in the event of a fuel pin rupture would be large.

Thus, for economic and safety reasons a lower bound CHF correlation was chosen below all data points for 37 element fuel.

$$CHF = 0.57 \times 2{,}798.76 \exp(-3.2819\ x)$$

where CHF = kw/m^2. This correlation is constantly under scrutiny and revision. It has served a useful purpose, however, for a number of years, dating back to Bruce A design. Recent testing at CRNL and Westinghouse attempts to account for non-uniform heat flux in segmented bundles.

A typical channel power and channel quality profile is given in Figure 2 3. If replotted as power vs. quality, Figure 24 results. Superimposed on the CHF curve, we see the relationship between the operating heat flux and the critical heat flux. To calculate the reserve, the power-quality curves are calculated for increasingly higher powers until the power-quality curve touches the CHF-quality curve (as shown by the dotted line in Figure 25). This power, then, is the maximum power achievable without dryout or melting. The ratio of this maximum power to the nominal operating power is the critical power ratio, CPR.

Typical reserve margins (to allow for instrumentation error, simulation of neutron flux error, refuelling rupple, and operating margin) require a CPR of 1.35 to 1.40. The primary heat transport design conditions are set, in part, based on this restraint.

1.7 Overview

This explains in very simple terms the main features of the CANDU heat transport system (HTS). The main actors in the interplay of processes in the PHT are:

HTS Flows
HTS Temperatures (Quality)
Circuit Resistances
Pump Head
Boiler Heat Transfer
Power

These govern the steady state and transient operation of the HTS. If the quality is non-zero, it is important to know by how much since it influences U and K to a large degree. Also the relationship between the quality and void fraction is important for determining the swell and shrink during transients.

The transient behaviour is important because flow, temperature and pressure swings can be damaging to the components. Accurate models of the system are required so that detailed analysis of normal and abnormal events of plant operation can be analysed and accounted for in the design of the plant. The accuracy required depends on the design margin. Tight margins require high accuracy while crude models will suffice for robust designs.

Plant control systems, as discussed later, ensure that process parameter (flow, temperature, pressure) swings are limited and hence the plant is protected. Plant control also governs the controlled manoeuvring of power output as required by electrical or steam production demand.

Accurate assessments of pressure drops, two-phase flow behaviour, heat transfer details, pump head curves, etc., are required for design. Detailed equations for mass, energy and momentum balances are required throughout the system and they must be solved simultaneously in the steady state and the transient. Empirical correlations must be found to account for complex processes like pressure drop in pipes under single and two-phase conditions, heat transfer (boiling and non-boiling) etc. These topics are covered in the following chapters. Detailed assessments of layout, maintenance and man-rem considerations, economy, technical feasibility, component life, stress analysis, safety, and controllability must all be considered before a design is finalized.

Although the PHTS is very simple in concept, the details make it quite complex indeed. It's size, use of expensive D_2O, and necessary provision for cooling to the fuel <u>at all times</u> make it expensive. The system is designed to very high standards because the first line of defence to ensure safety in the CANDU system is the integrity of the primary boundary (pipes) and the proper design of the processes within that boundary.

Subsequent chapters discuss in varying degrees the details involved in satisfying the above.

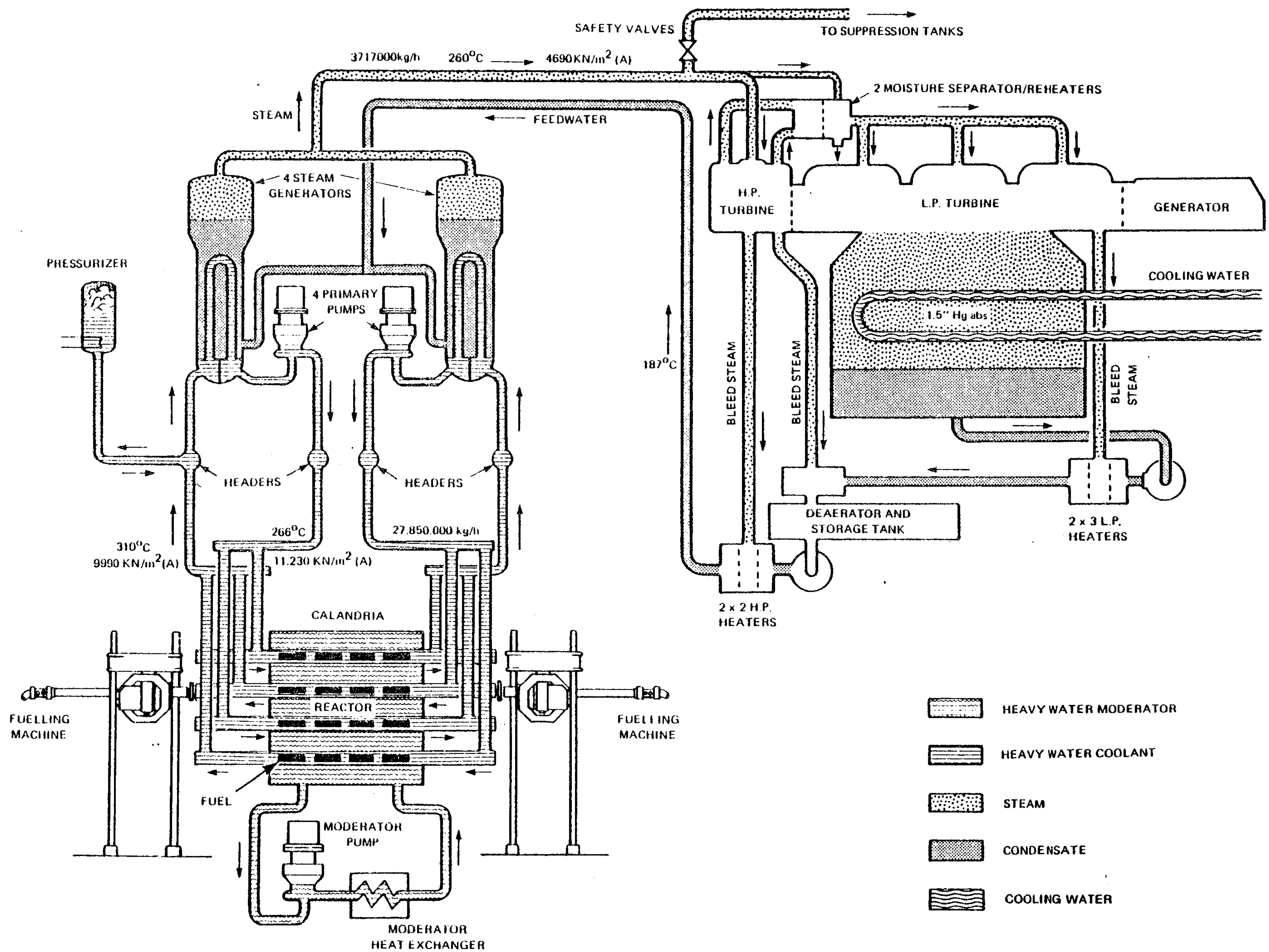


FIGURE 1 CANDU NUCLEAR POWER SYSTEM

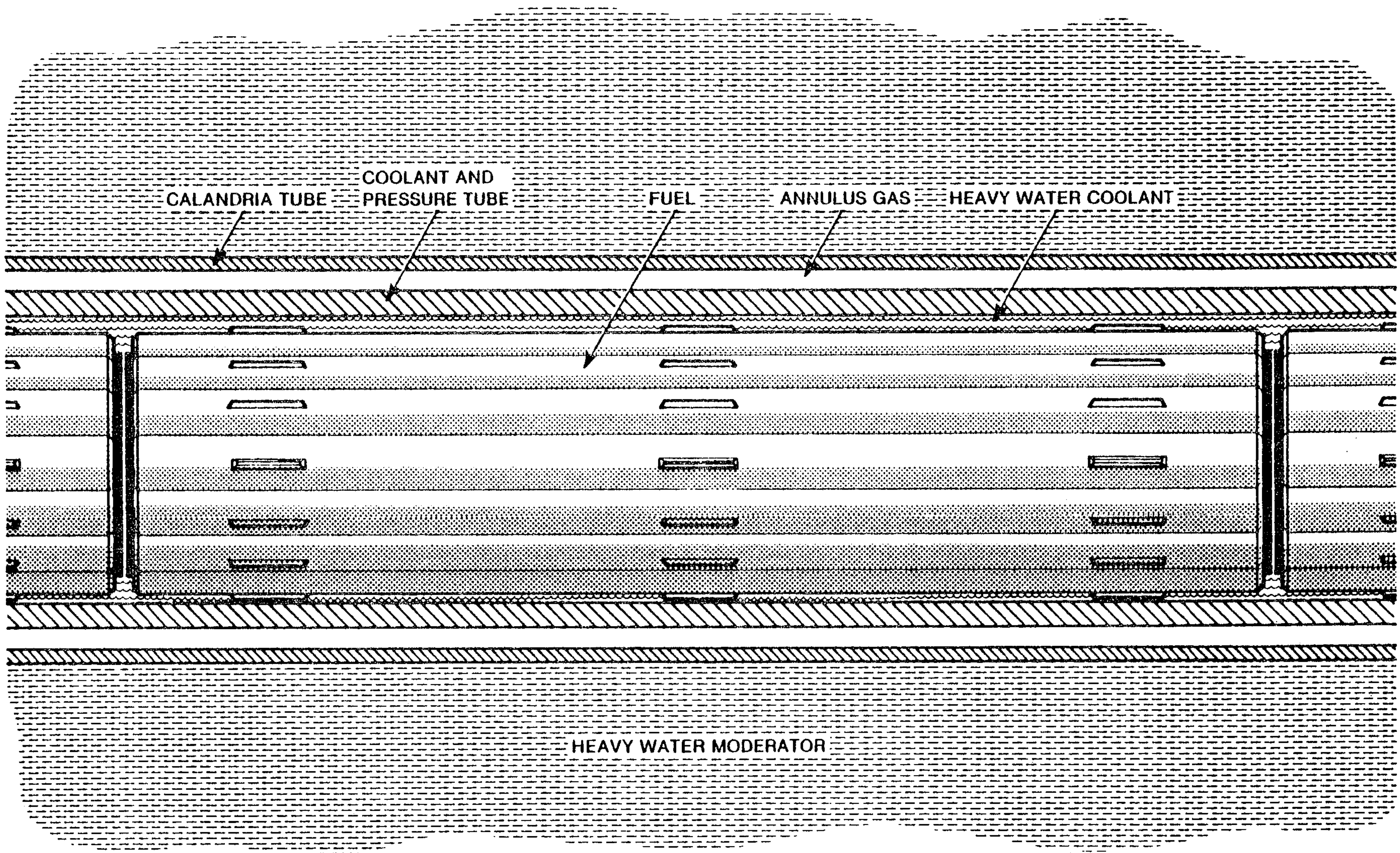


FIGURE 2 FUEL CHANNEL ARRANGEMENT

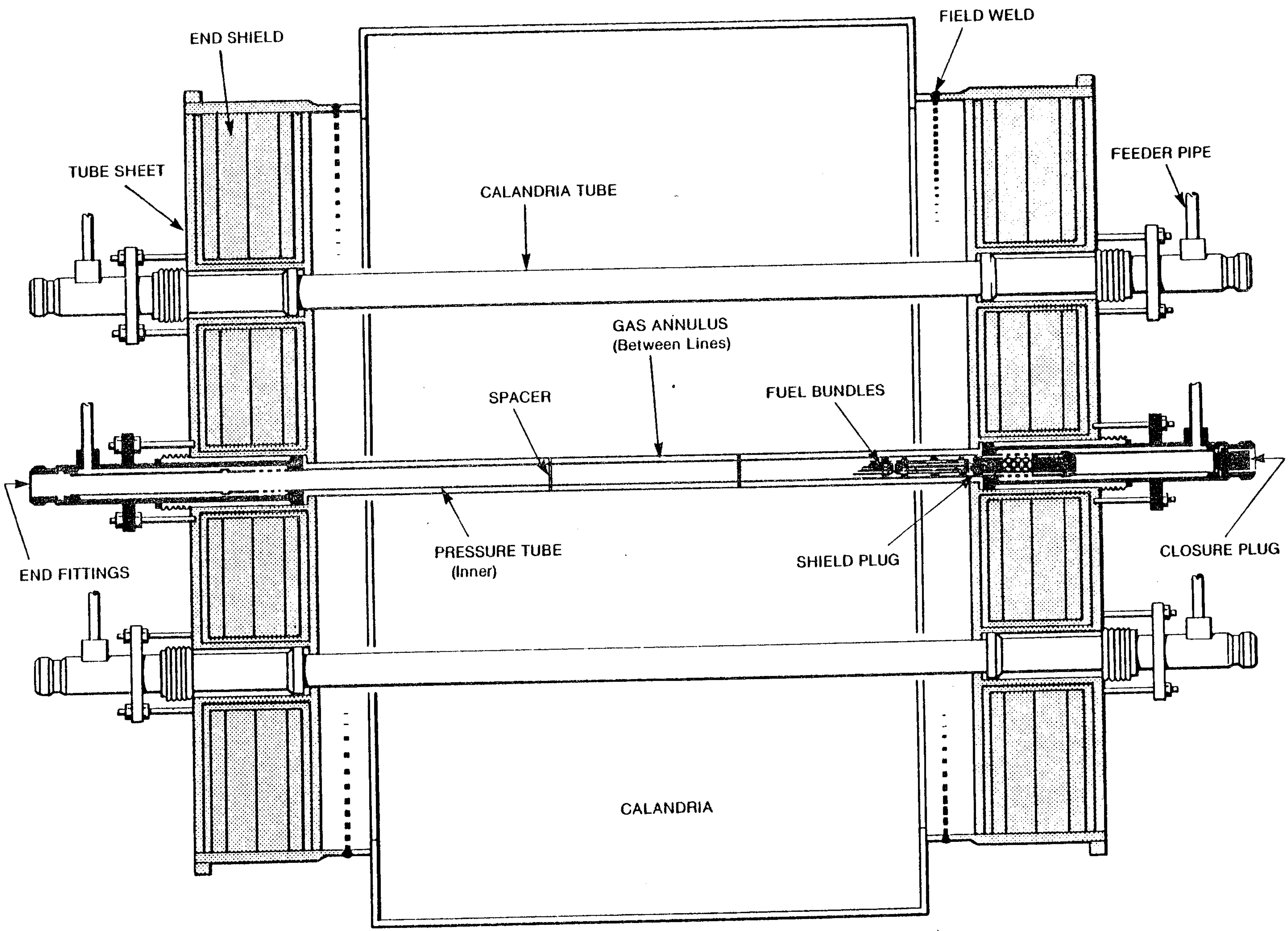


FIGURE 3 REACTOR CORE SCHEMATIC

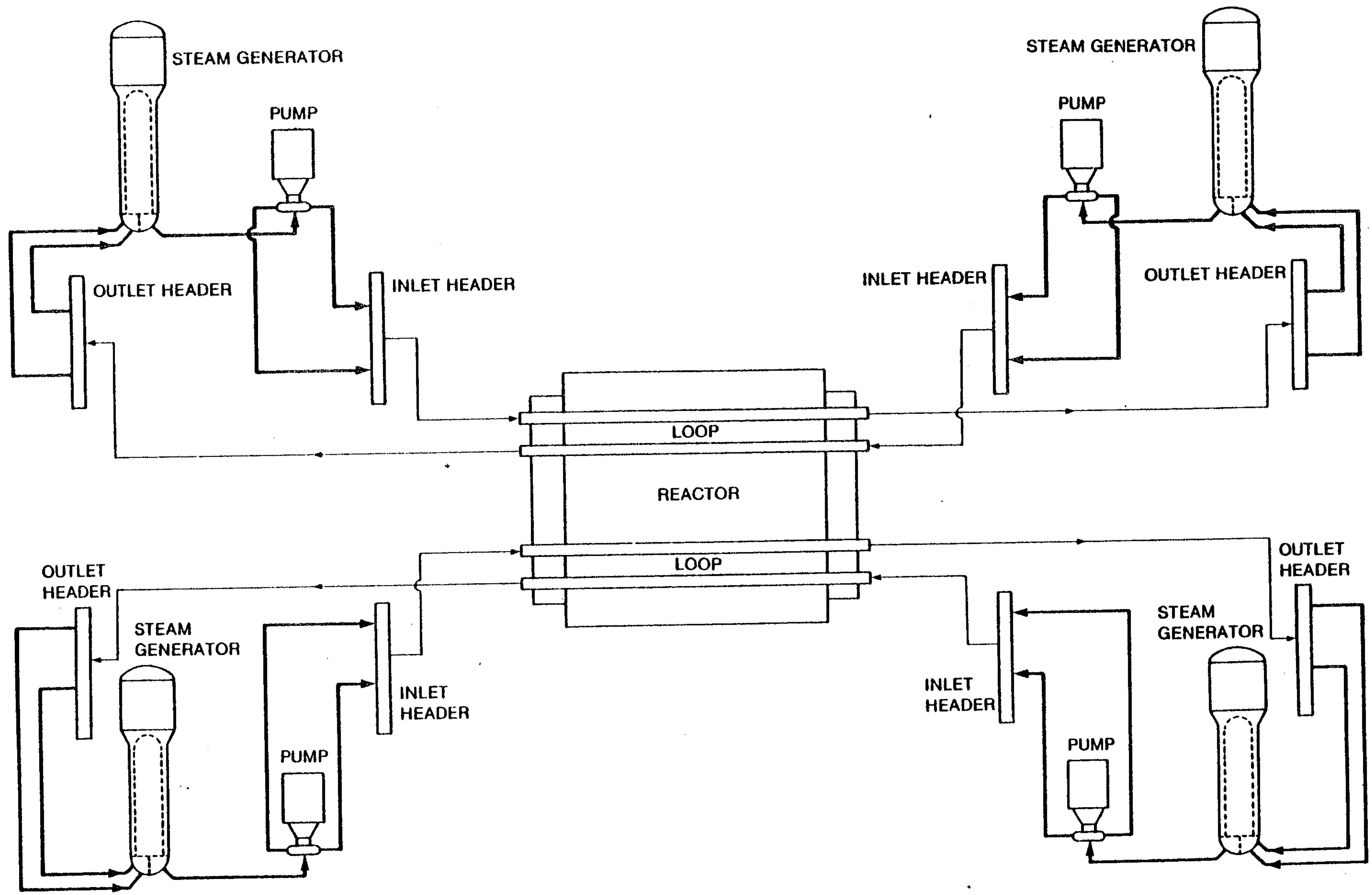
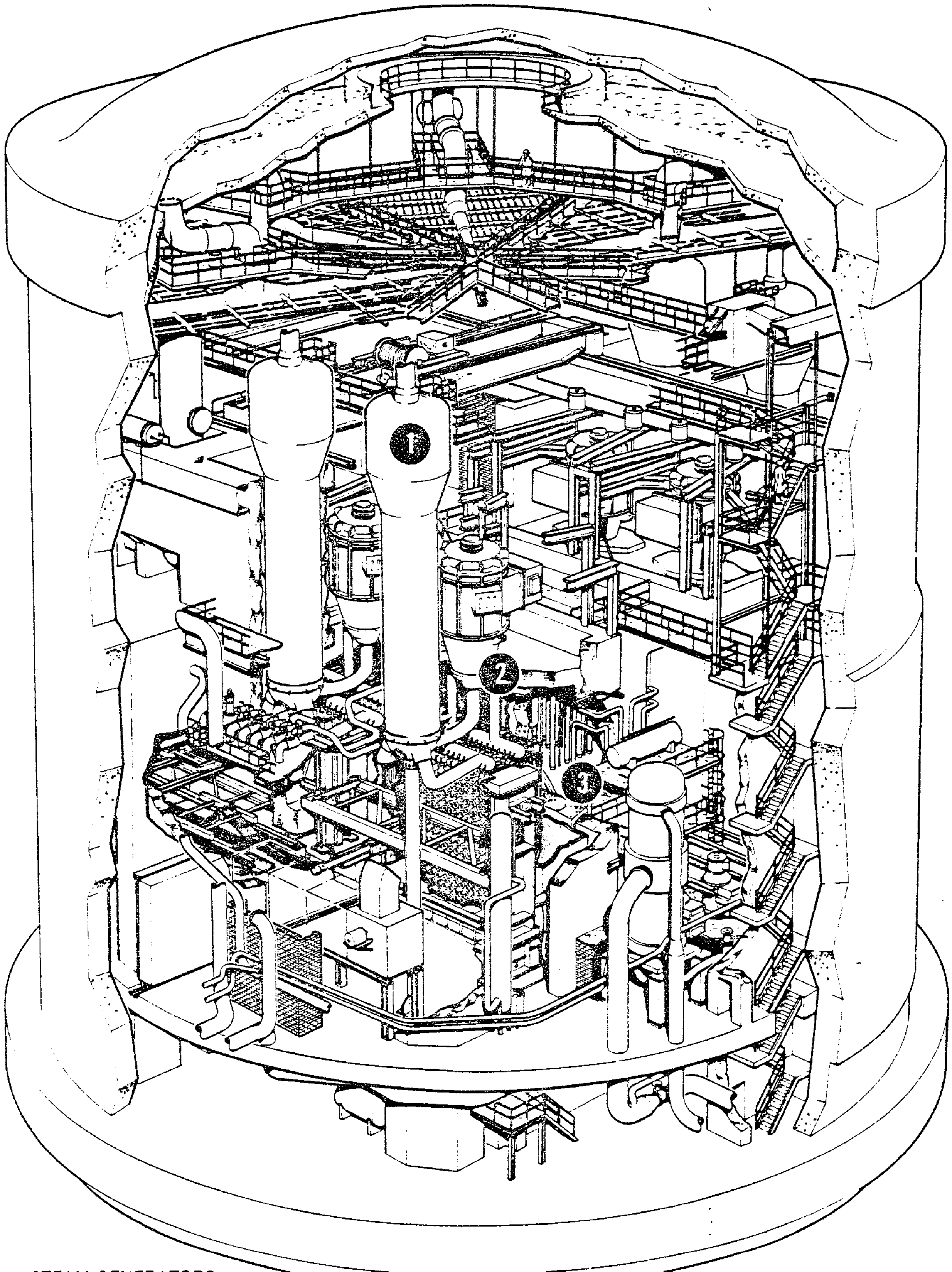
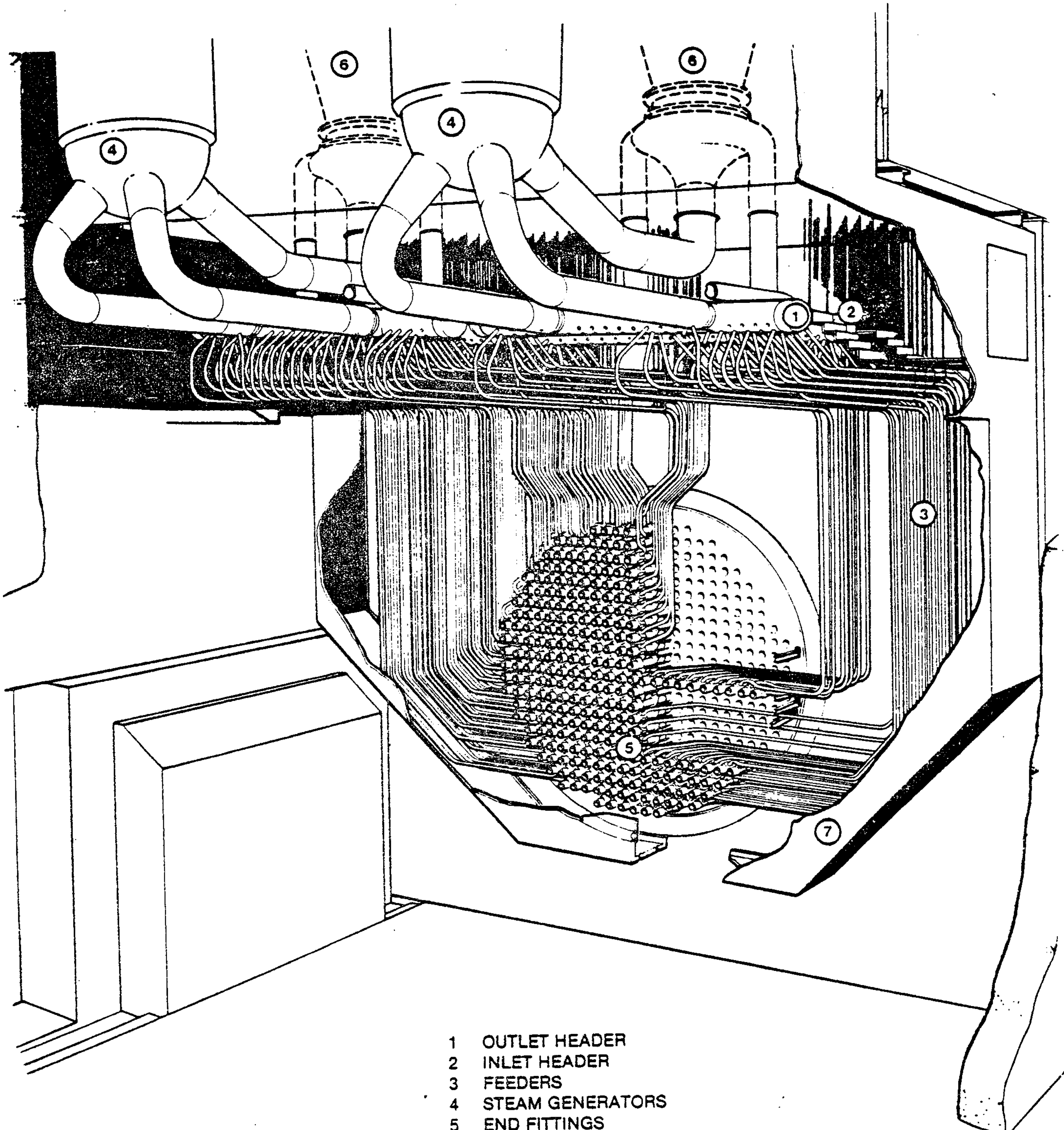


FIGURE 4 A HEAT TRANSPORT SYSTEM



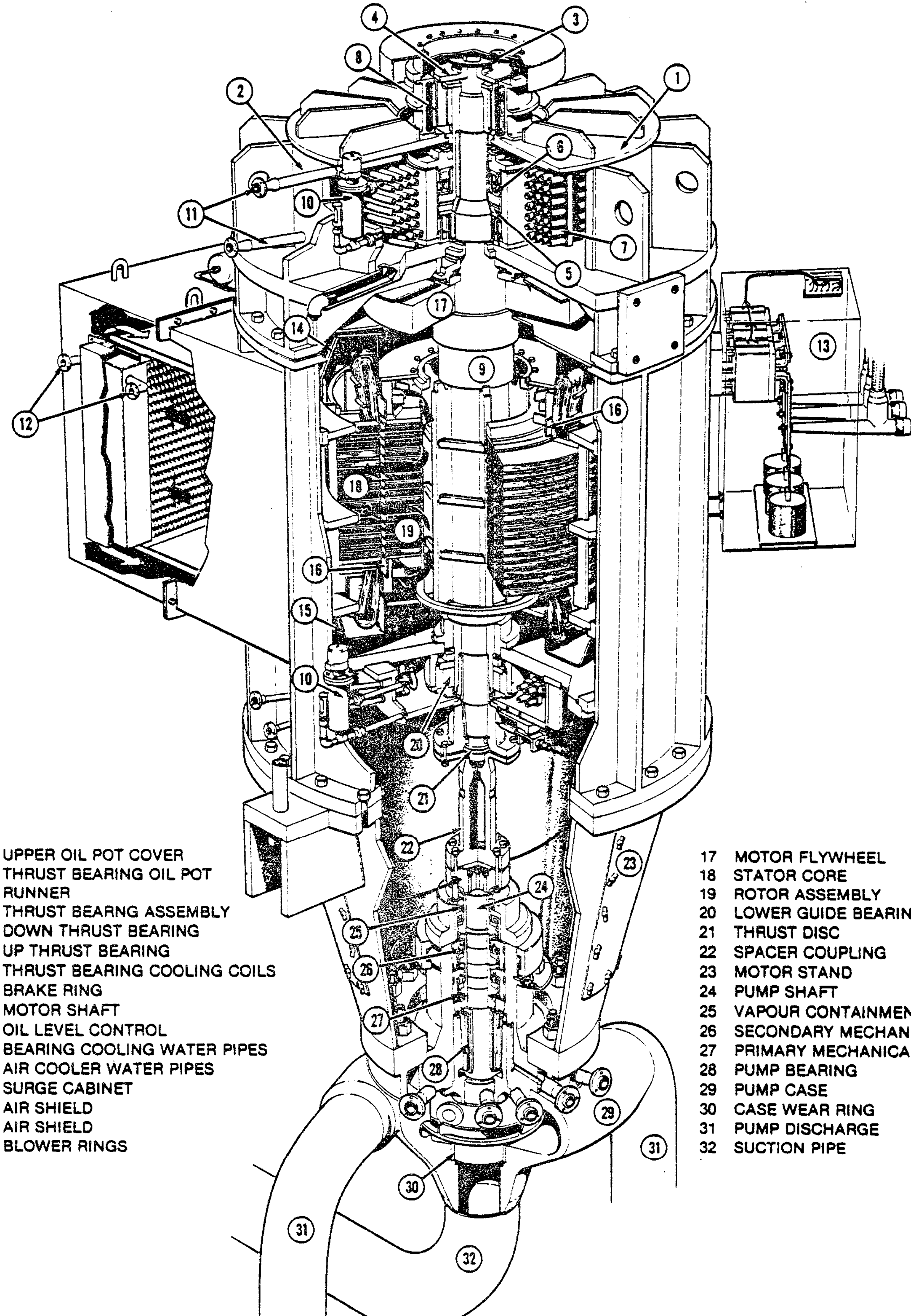
1 STEAM GENERATORS
2 HTS PUMPS
3 REACTOR

FIGURE 5 LOCATION OF HEAT TRANSPORT SYSTEM EQUIPMENT



1 OUTLET HEADER
2 INLET HEADER
3 FEEDERS
4 STEAM GENERATORS
5 END FITTINGS
6 HEAT TRANSPORT PUMPS
7 INSULATION CABINET

FIGURE 6 FEEDER AND HEADER ARRANGEMENT



1 UPPER OIL POT COVER
2 THRUST BEARING OIL POT
3 RUNNER
4 THRUST BEARNG ASSEMBLY
5 DOWN THRUST BEARING
6 UP THRUST BEARING
7 THRUST BEARING COOLING COILS
8 BRAKE RING
9 MOTOR SHAFT
10 OIL LEVEL CONTROL
11 BEARING COOLING WATER PIPES
12 AIR COOLER WATER PIPES
13 SURGE CABINET
14 AIR SHIELD
15 AIR SHIELD
16 BLOWER RINGS
17 MOTOR FLYWHEEL
18 STATOR CORE
19 ROTOR ASSEMBLY
20 LOWER GUIDE BEARING
21 THRUST DISC
22 SPACER COUPLING
23 MOTOR STAND
24 PUMP SHAFT
25 VAPOUR CONTAINMENT SEAL
26 SECONDARY MECHANICAL SEAL
27 PRIMARY MECHANICAL SEAL
28 PUMP BEARING
29 PUMP CASE
30 CASE WEAR RING
31 PUMP DISCHARGE
32 SUCTION PIPE

FIGURE 7 HEAT TRANSPORT SYSTEM PUMP

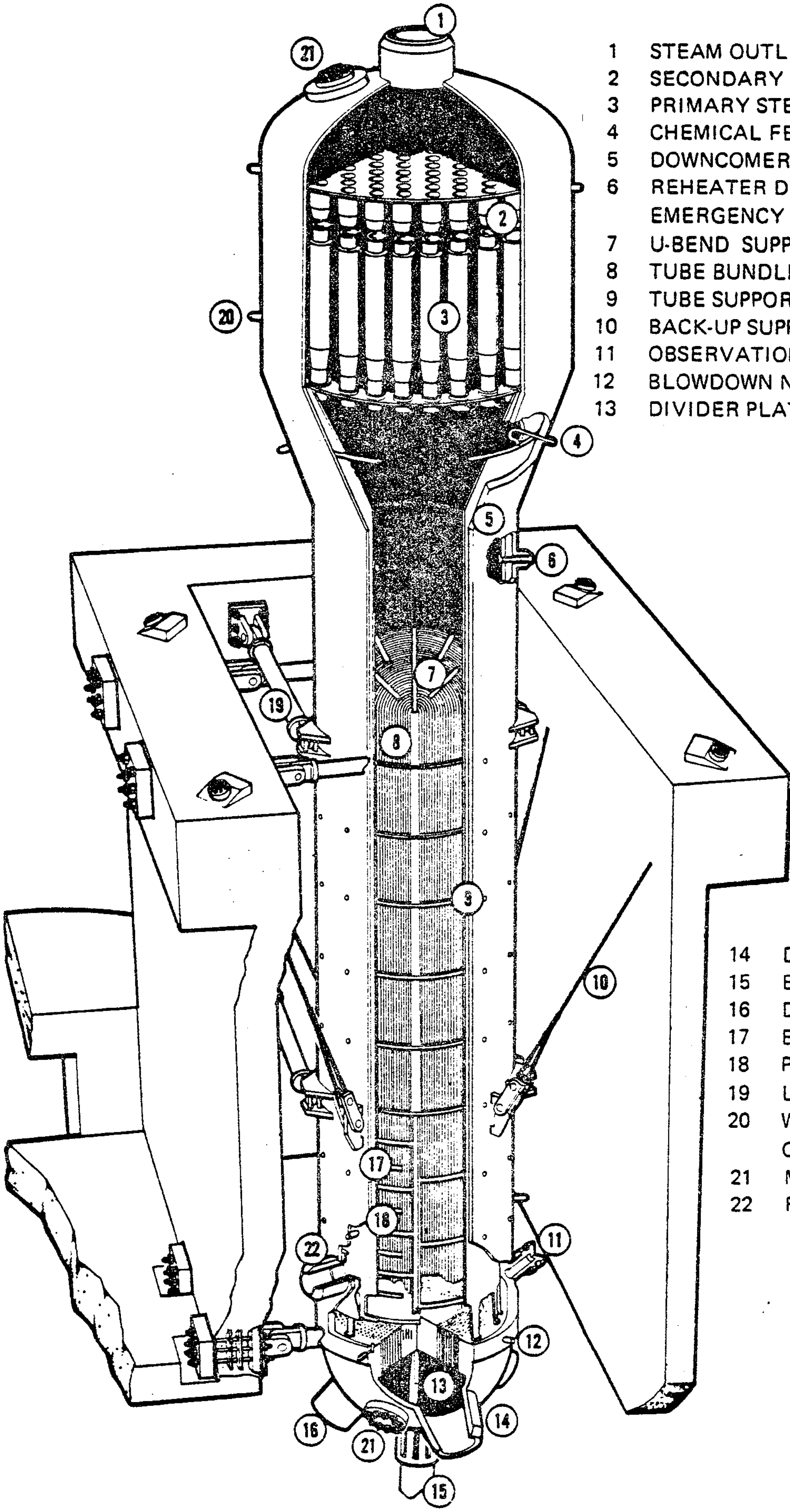


FIGURE 8 600 MW STEAM GENERATOR

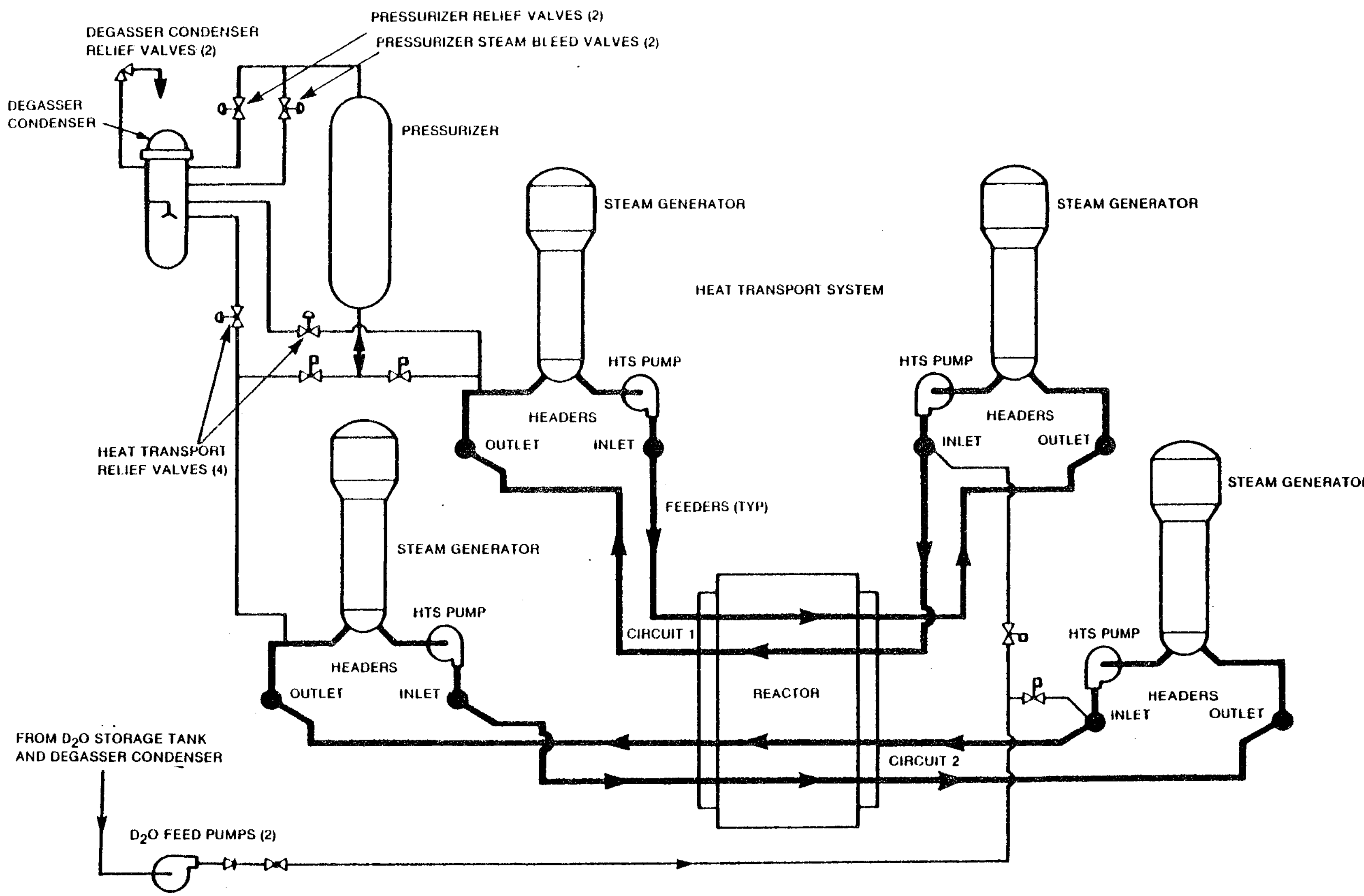


FIGURE 9 HEAT TRANSPORT PRESSURE AND INVENTORY CONTROL SYSTEM

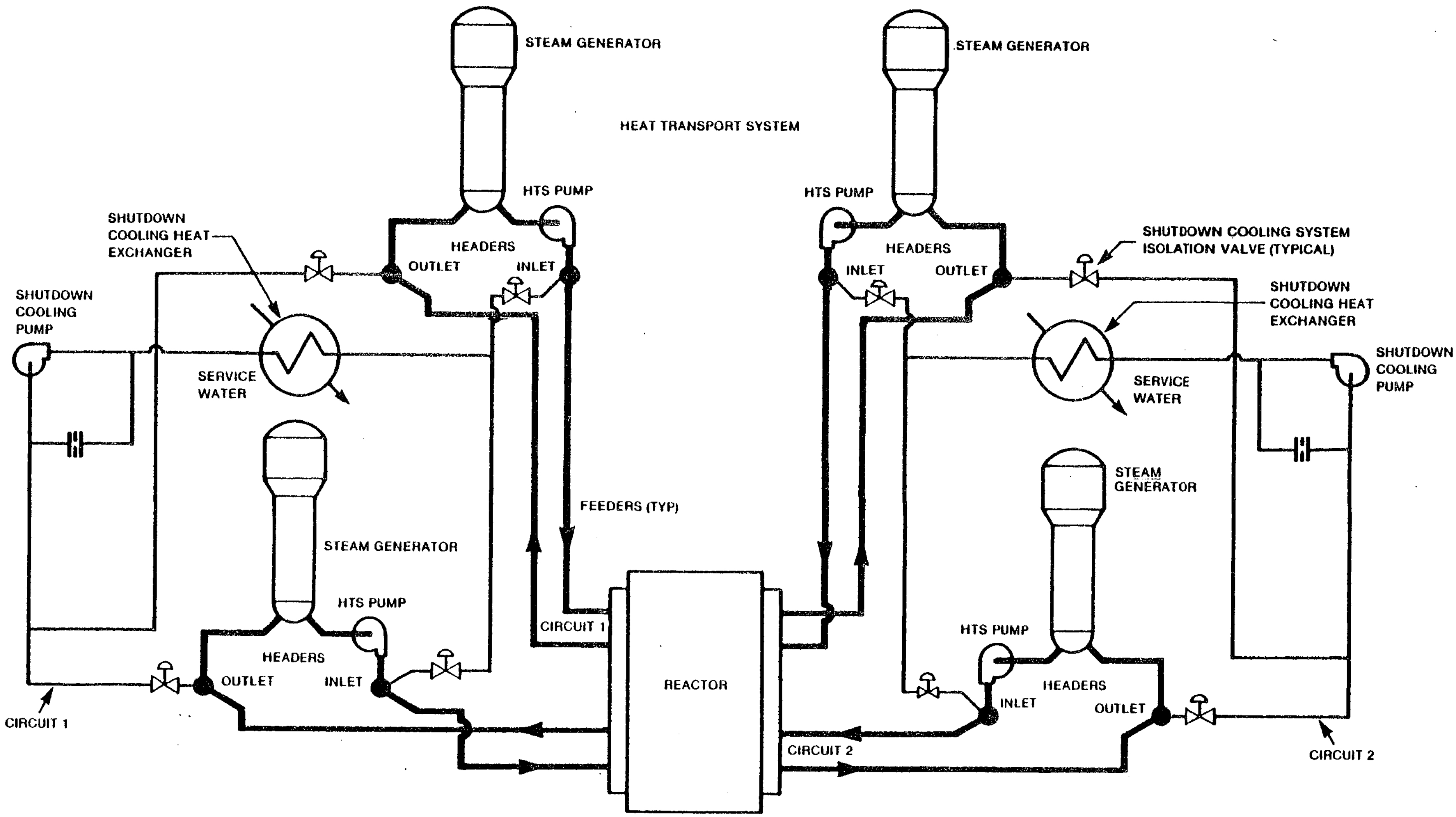


FIGURE 10 SHUTDOWN COOLING SYSTEM

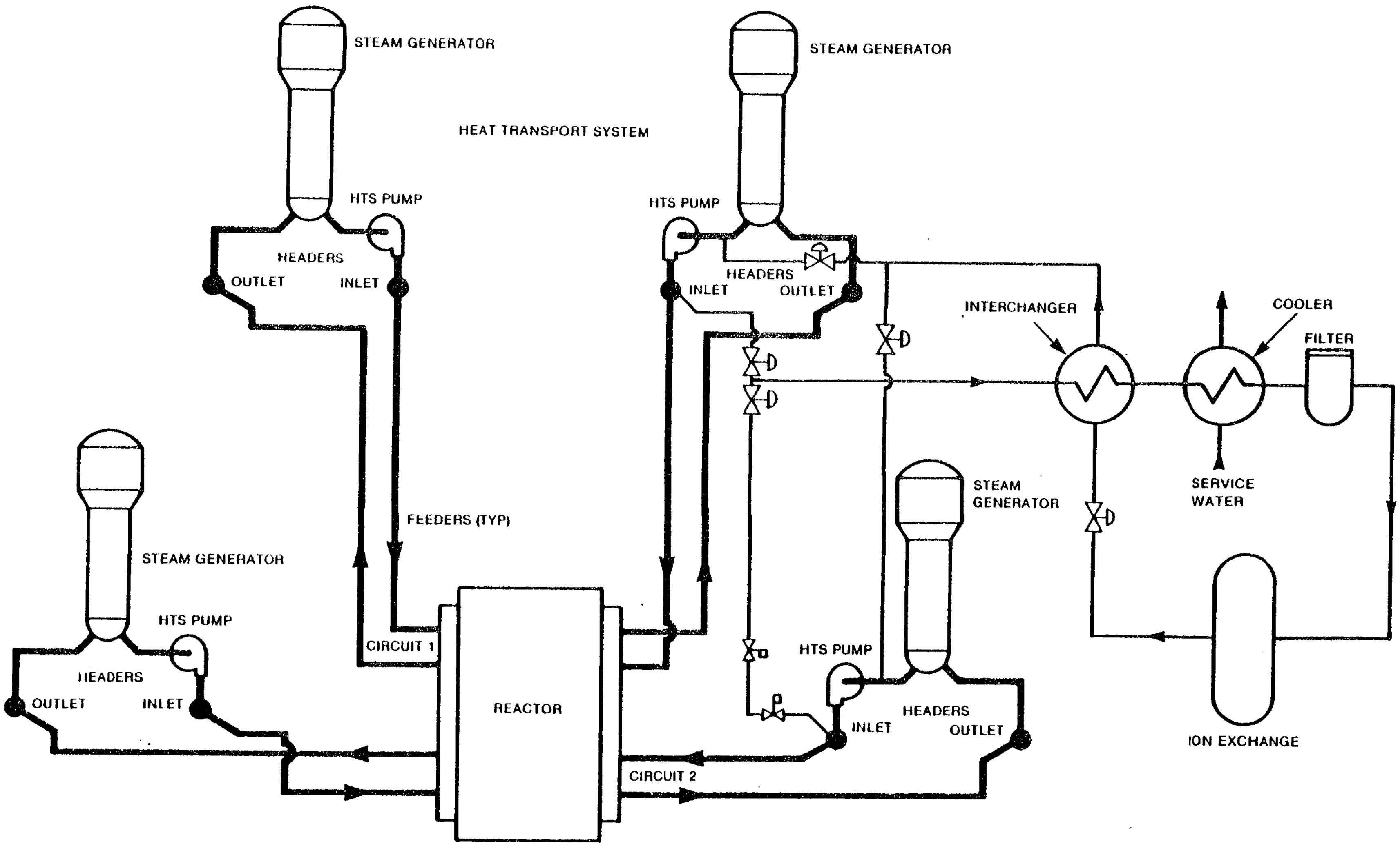


FIGURE 11 HEAT TRANSPORT PURIFICATION SYSTEM

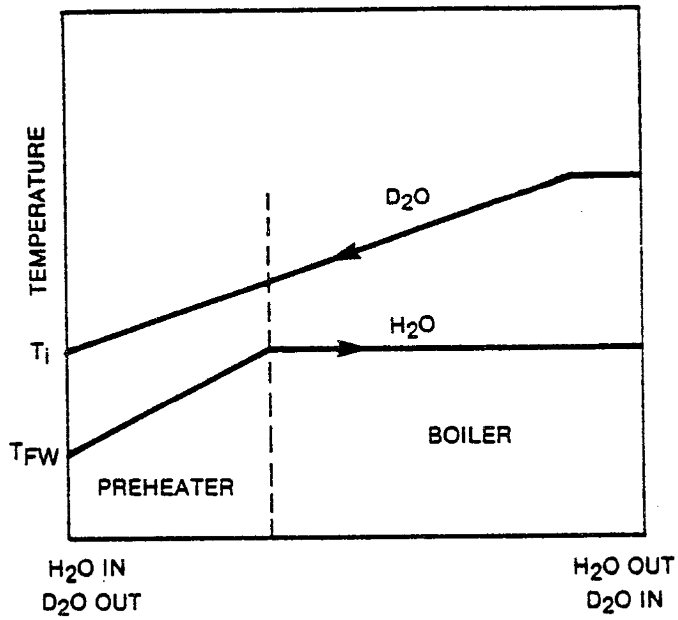


FIGURE 12 STEAM GENERATOR TEMPERATURE DISTRIBUTION

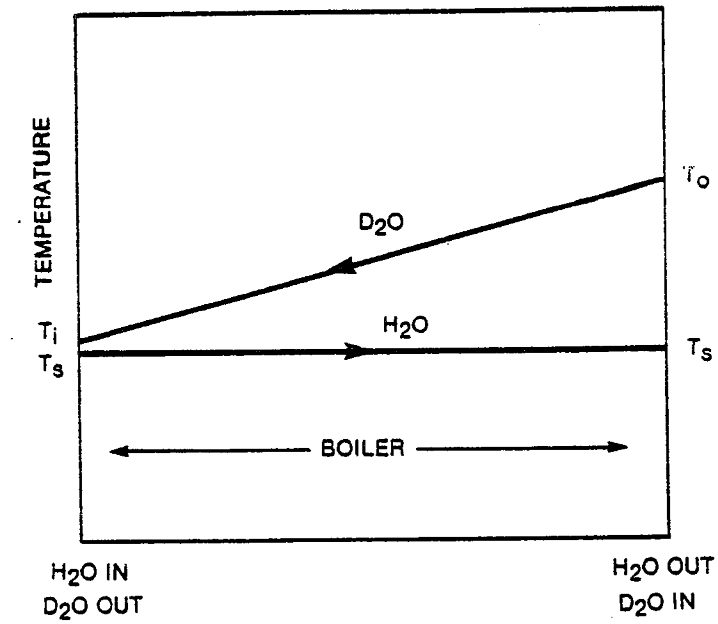


FIGURE 13 SIMPLIFIED STEAM GENERATOR TEMPERATURE DISTRIBUTION

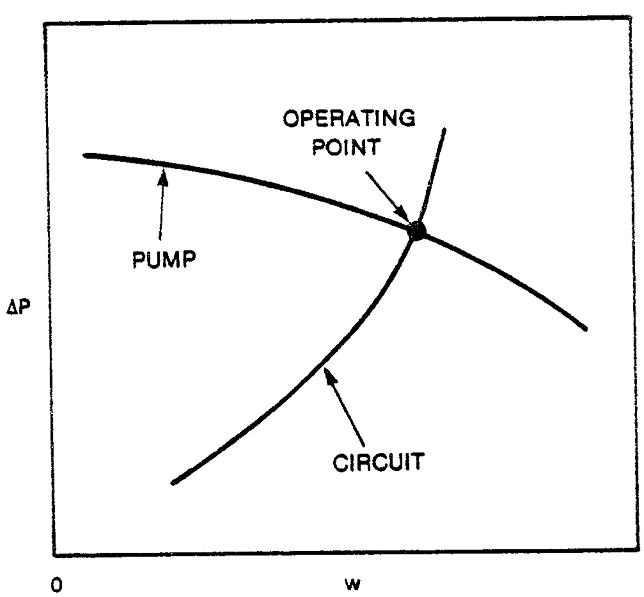


FIGURE 14 CIRCUIT LOSSES AND PUMP HEAD VS. FLOW

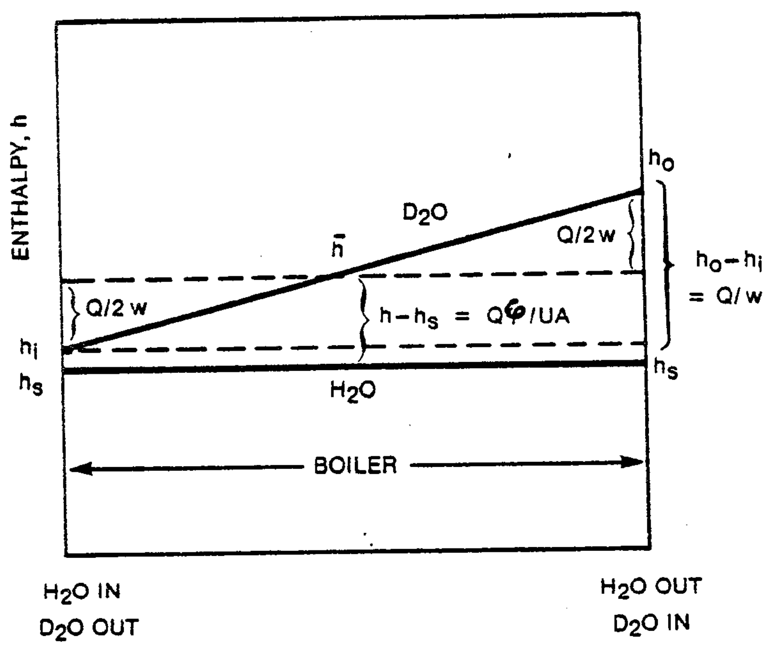


FIGURE 15 ENTHALPY VARIATIONS

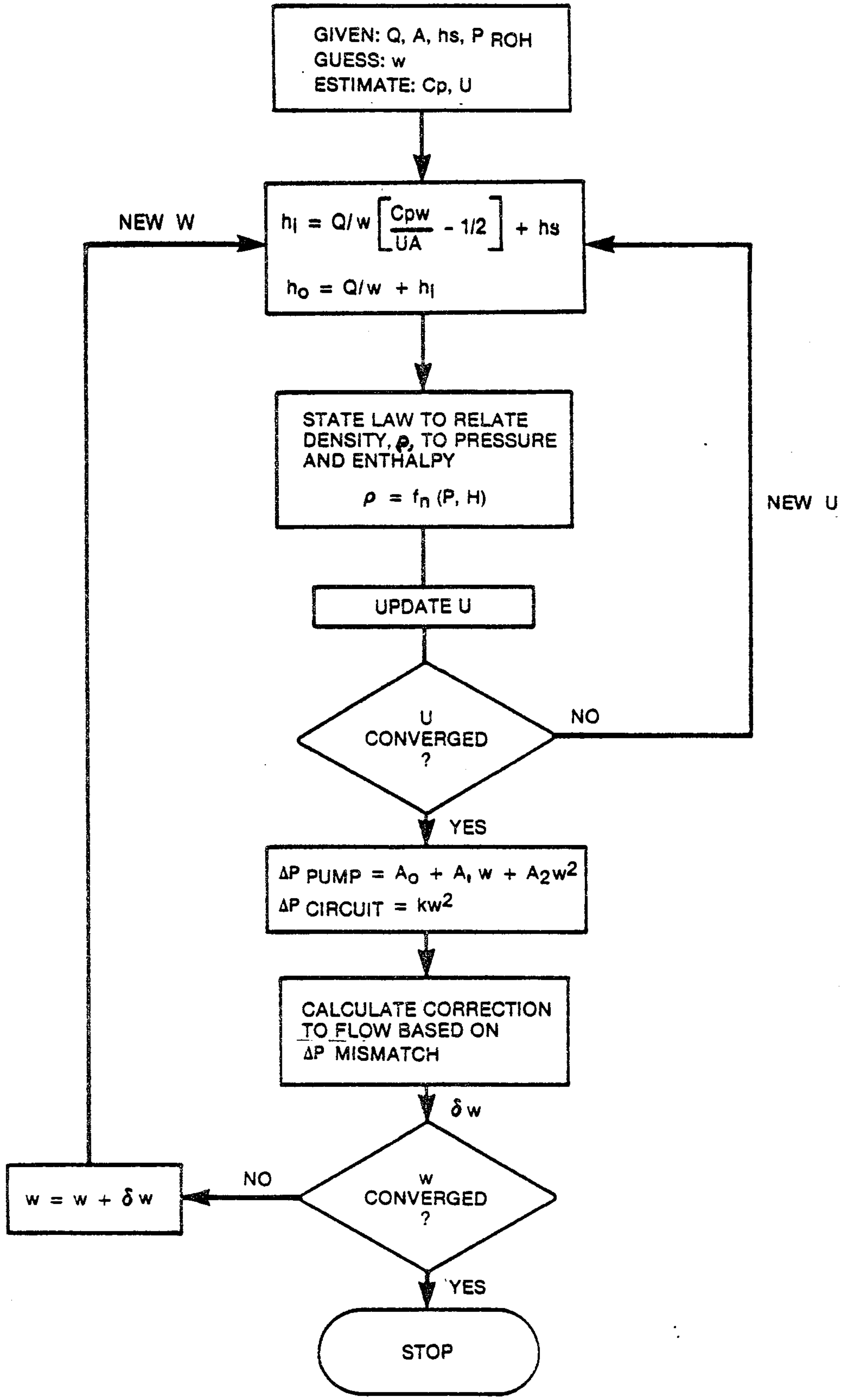


FIGURE 16 FLOW CHART FOR HTS CALCULATIONS

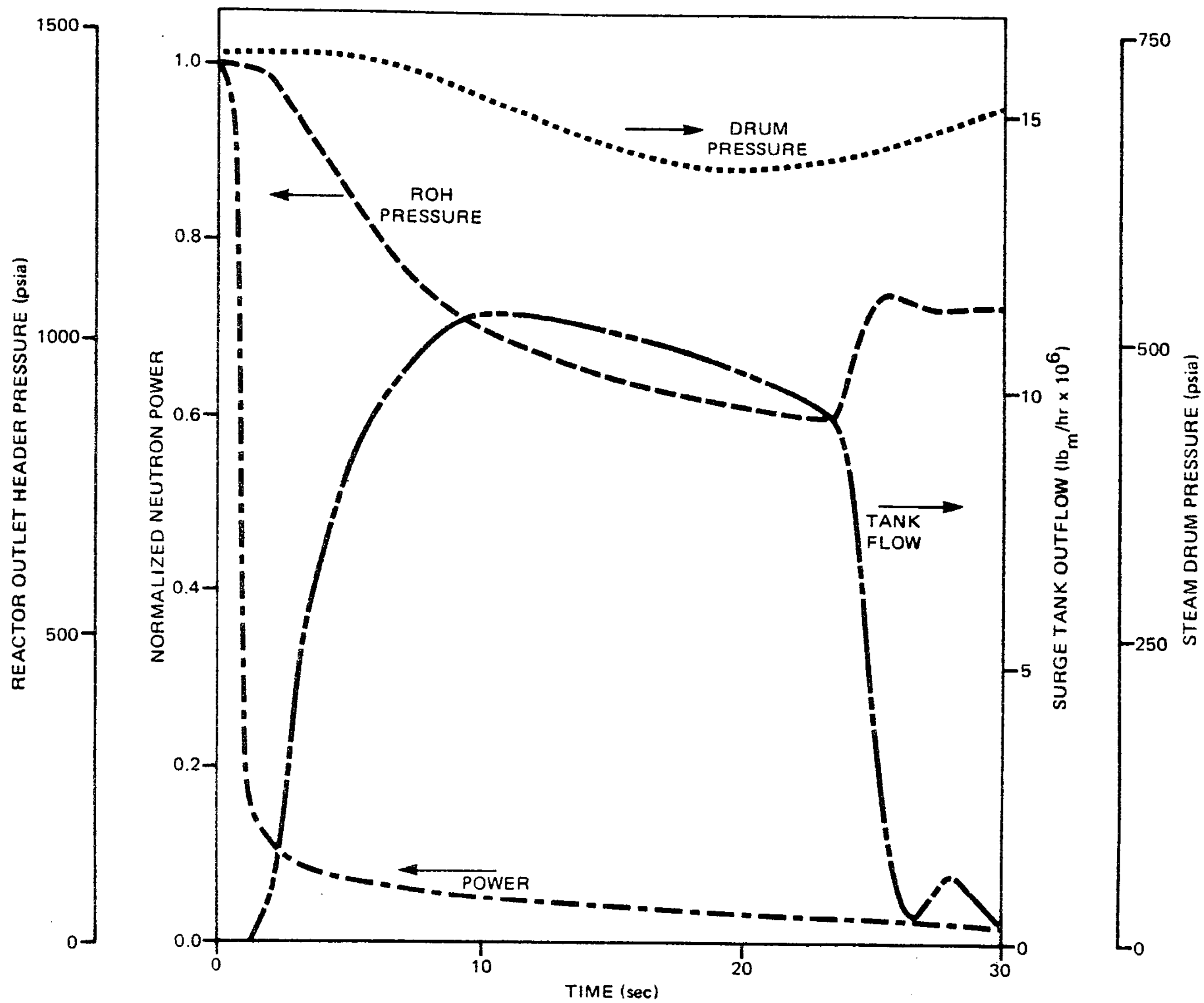


FIGURE 17b STEAM DRUM PRESSURE, SURGE TANK FLOW AND PRIMARY SYSTEM PRESSURE AND POWER AS A FUNCTION OF TIME FOR A REACTOR TRIP

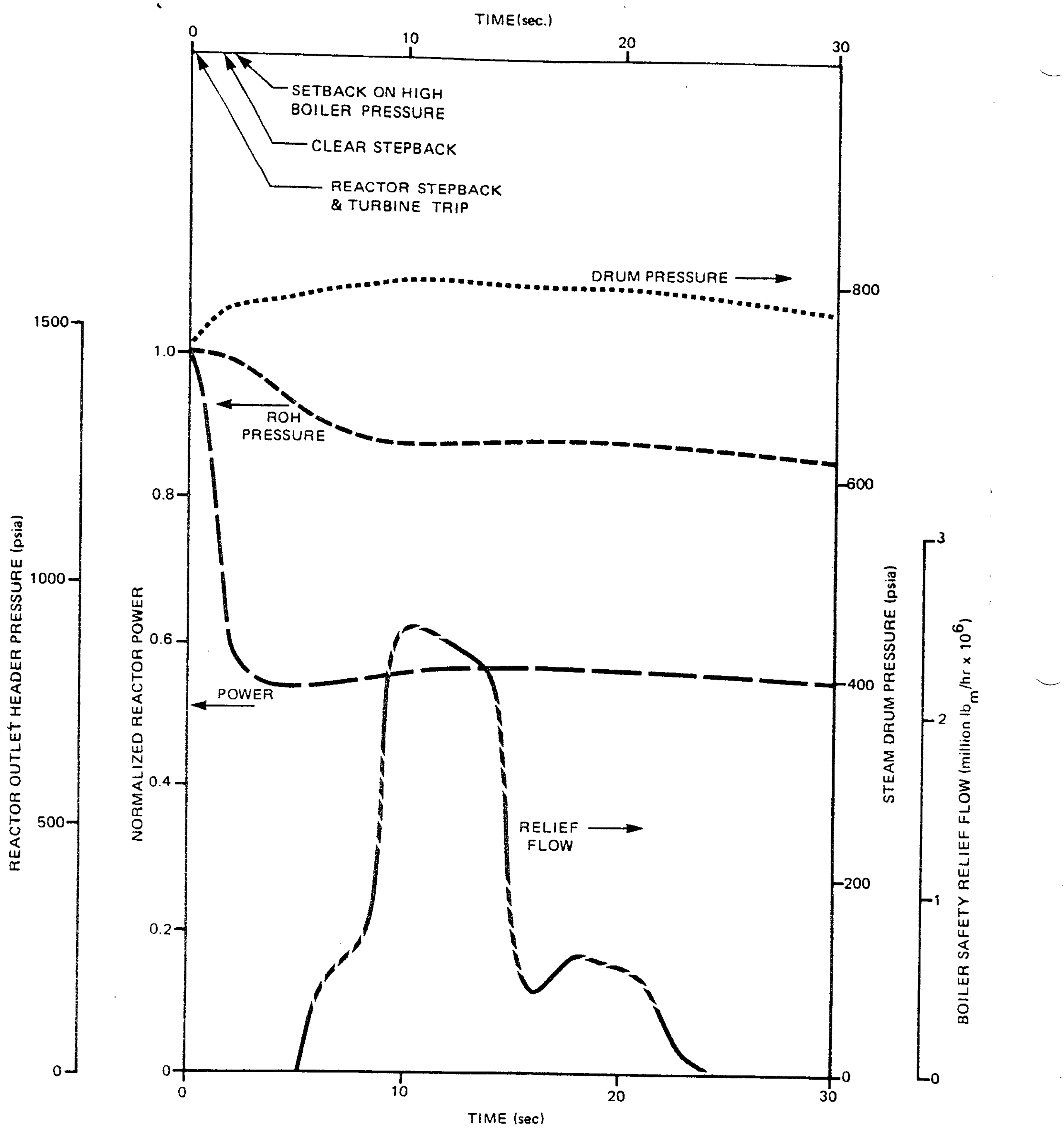


FIGURE 18 STEAM DRUM PRESSURE, BOILER SAFETY RELIEF FLOW AND PRIMARY SYSTEM PRESSURE AND POWER AS A FUNCTION OF TIME FOR A TURBINE TRIP

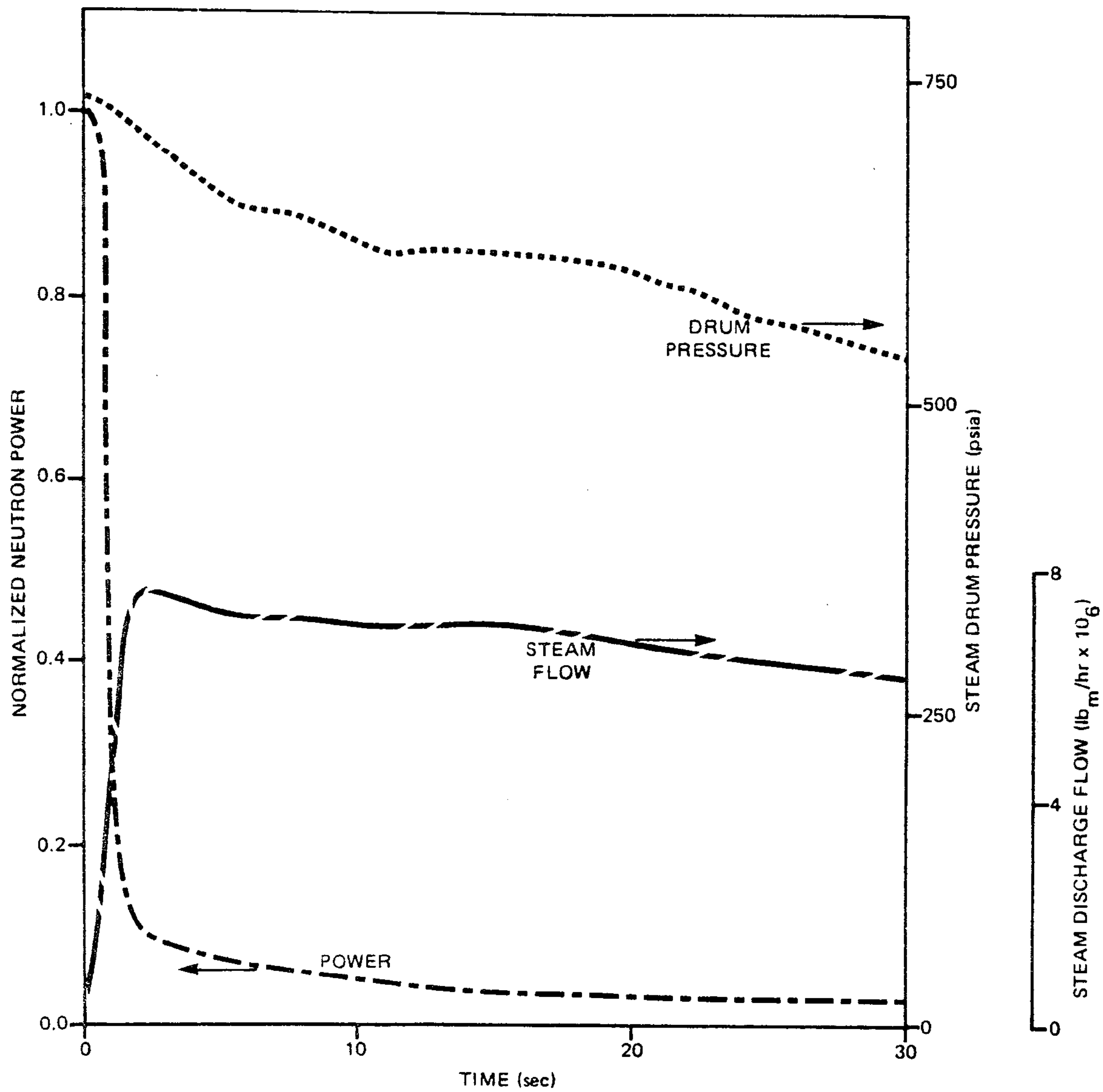


FIGURE 19 STEAM DRUM PRESSURE, STEAM DISCHARGE FLOW AND PRIMARY SYSTEM POWER AS A FUNCTION OF TIME FOR A RAPID COOLDOWN

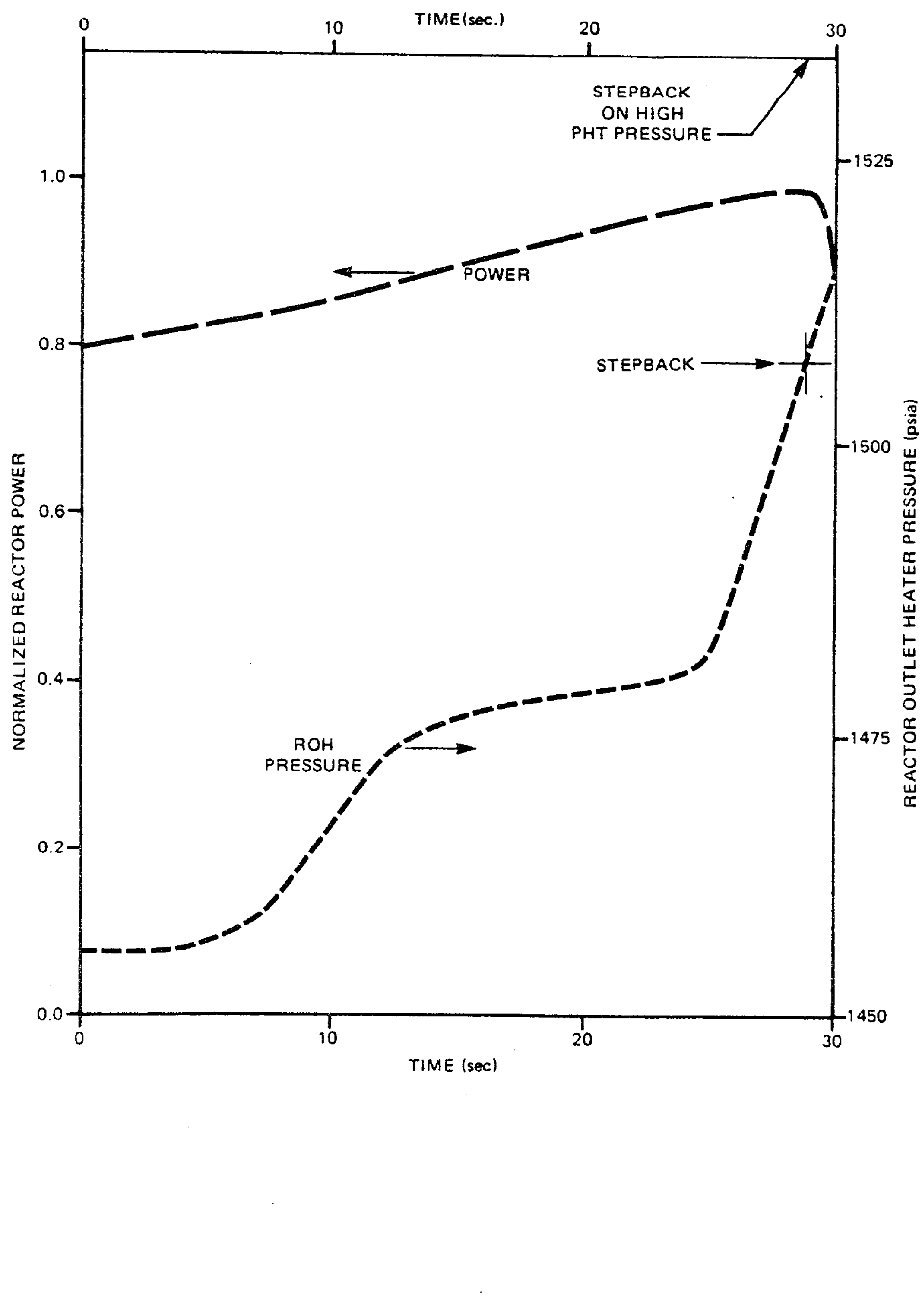


FIGURE 20 PRIMARY SYSTEM POWER AND PRESSURE AS A FUNCTION OF TIME FOR MANEUVERING FROM 80% TO 100% FULL POWER

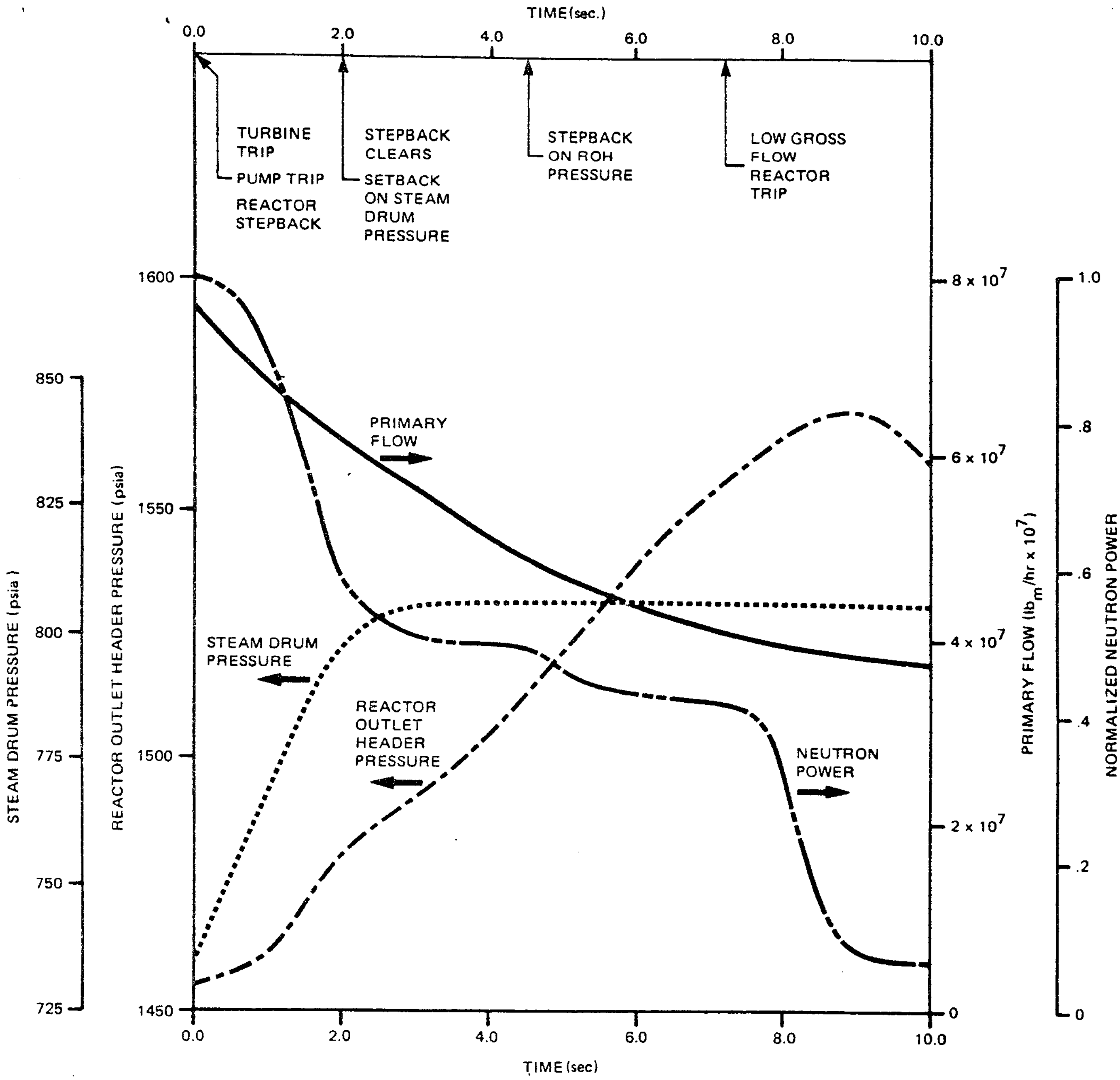


FIGURE 21 STEAM DRUM PRESSURE AND PRIMARY SYSTEM PRESSURE, FLOW AND POWER AS A FUNCTION OF TIME FOR A TOTAL LOSS OF CLASS IV POWER FAILURE

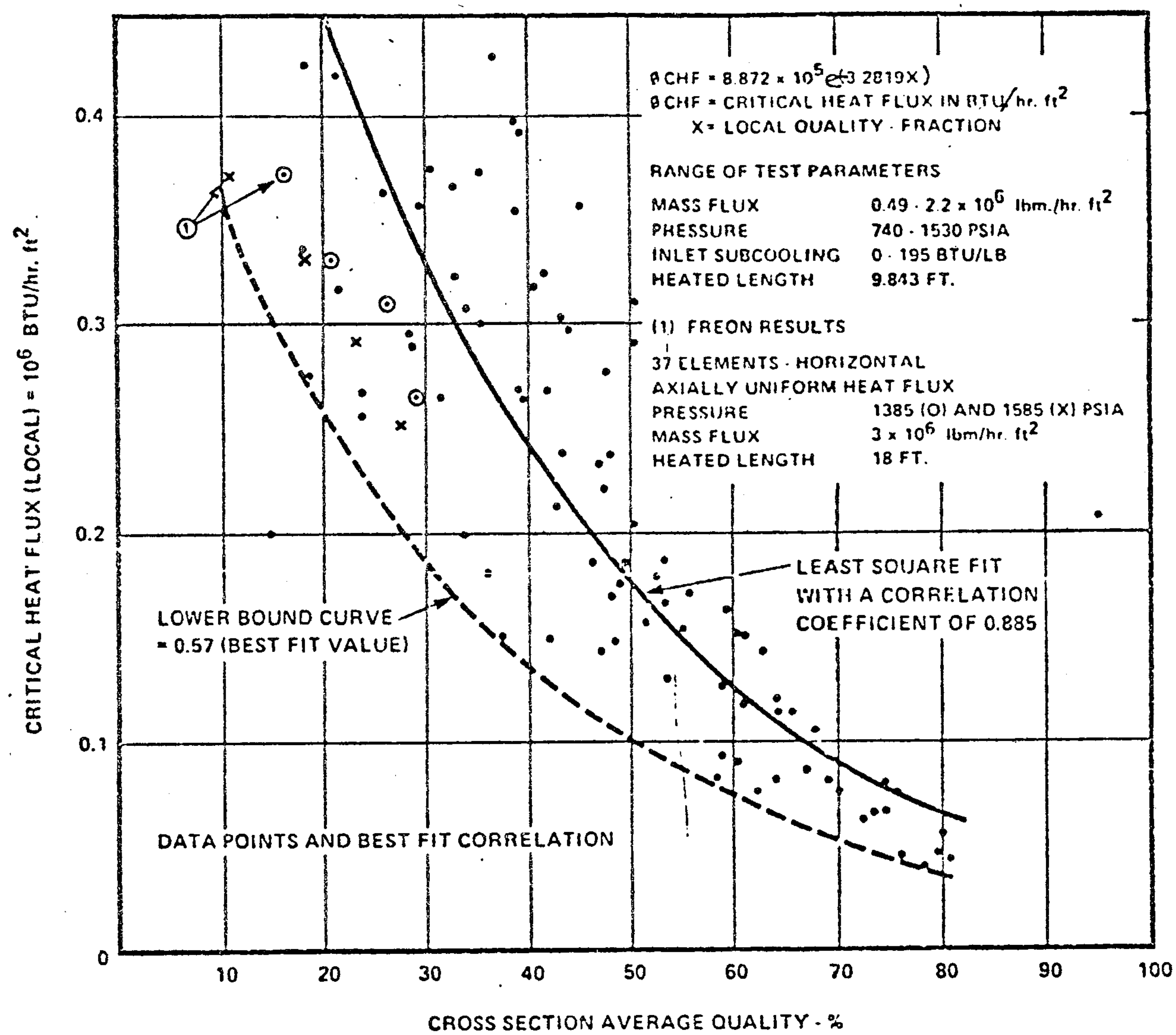


FIGURE 22 "IN REACTOR" CRITICAL HEAT FLUX MEASUREMENTS

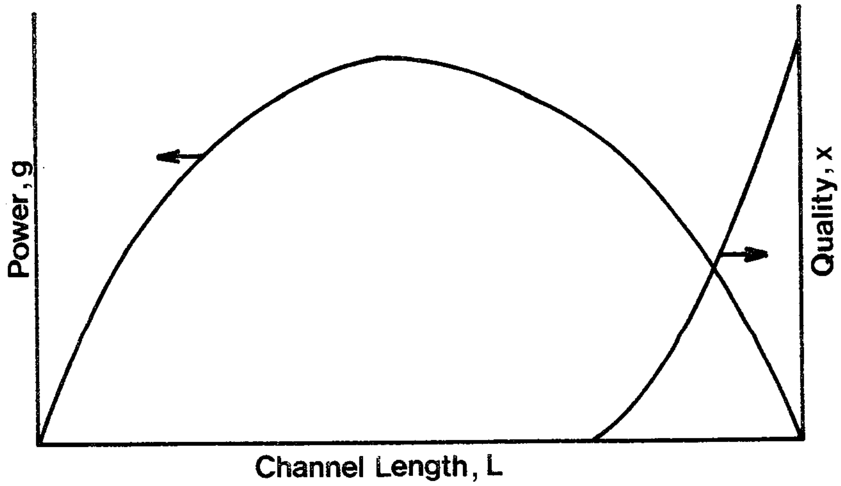


Fig. 23: Power and quality vs. length along a fuel channel.

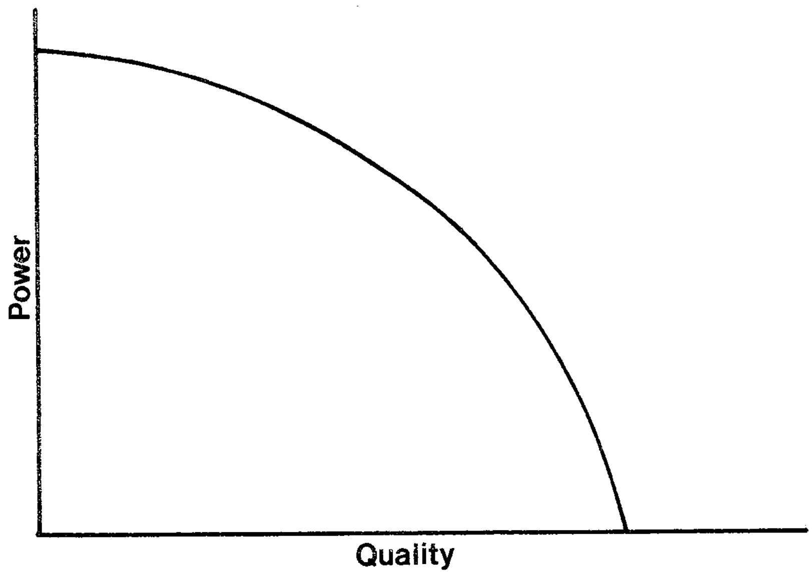


Fig. 24: Power vs. quality.

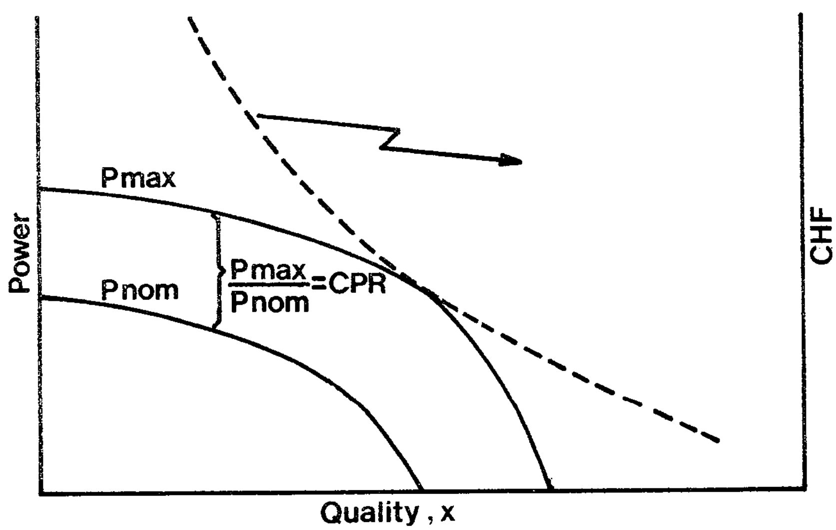


Fig. 25: CPR determination.