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THERMOHYDRAULICS IN A HIGH-TEMPERATURE GAS-COOLED REACTOR PRIMARY LOOP DURING EARLY PHASES OF UNRESTRICTED CORE-HEATUP ACCIDENTS*

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INTRODUCTION

In High Temperature Gas Cooled Reactor (HTGR) siting considerations, the Unrestricted Core Heatup Accidents (UCHA) are considered as accidents of highest consequence, corresponding to core meltdown accidents in light water reactors. Initiation of such accidents can be, for instance, due to station blackout, resulting in scram and loss of all main loop forced circulation, with none of the core auxiliary cooling system loops being started. The result is a slow but continuing core heatup, extending over days. During the initial phases of such UCHA scenarios, the primary loop remains pressurized, with the system pressure slowly increasing until the relief valve setpoint is reached.

The major objectives of the work described here were to determine times to depressurization as well as approximate loop component temperatures up to depressurization. The time of depressurization is the earliest time for some of the circulating inventory of fission products from the primary loop to enter the containment building, thus affecting the relatively minor consequences of some accident scenarios. Only in the extremely unlikely case of simultaneous containment isolation failure do significant consequences arise at this time. The temperature distribution at the time of depressurization also serves as input to long term analyses of UCHA with a change in models at that time, since heat transport by convection is significant in the high density gas prior to depressurization, but becomes negligible in the low density gas subsequent to depressurization. Furthermore, the early temperature history is required to establish the maximum time that is available to restore forced circulation (MTRC) without incurring any permanent damage to the equipment.

The investigations reported here are for the General Atomic "Base Line Zero" reactor[1] of 2240 MW(th). Most of the results apply at least qualitatively to all large HTGR steam cycle designs.

SYSTEM DESCRIPTION

A typical HTGR is shown in Figure 1. The core is located in the main cavity at the center of a prestressed concrete reactor vessel (PCRV). It consists of about 4000 fuel elements, which are hexagonal graphite blocks containing embedded fuel rods and coolant holes surrounding the fuel rods. The four main-loop side cavities contain the steam generators, the main loop isolation valves (MLIV), and the circulators. Three further side cavities contain three independent auxiliary cooling loops, constituting the core auxiliary cooling system (CACS).

The core and side cavities are thermally separated from the colder PCRV by the thermal barrier which consists of steel cover plates, a layer of thermal insulation, and the liner cooling system (LCS), the latter being a system of cooling tubes welded to the back of a steel liner.

During normal full-power operation about 9×10^6 lb/hr of helium circulate as coolant through the primary loop at a circulator outlet pressure of 1050 psia, entering the core with about 600 F and entering the steam generators with about 1270 F, transferring approximately 2240 MW th.

ACCIDENT DESCRIPTION

Initiation of UCHAs requires loss of all main loop cooling and lack of starting of any of the CACS loops. Following a scram and loss of forced circulation (LOFC), all main loop circulators coastdown to a full stop. The reactor considered here employs gravity operated flapper valves as main-loop isolation valves, so that under UCHA conditions the main coolant flow through the steam generator side cavities decreases rapidly to virtually zero. Since decay heat generation and orificing vary between different refueling regions of the core, natural convection currents will arise between the various regions, with the coolant rising in the hotter regions and falling in cooler regions. There is also some bypass flow around seals and through buffer lines in the side cavities, but the in-core natural circulation is the most significant remaining coolant flow. However, as the relatively hotter and cooler regions of the core are at the same elevation, these flows are extremely weak and tend to die out as temperature differences are equalized.[2]

During this initial period of the accident, core pressures will rise until the relief valve set point of 1123 psia is reached. This can typically occur in about 1 - 6 hr; the exact time being very sensitive to details of the steam generator performance during the initial part of the accident. As the relief valve setpoint is reached, partial depressurization and subsequent valve closure could occur. However, since the helium reaching the valves from the upper plenum is now at relatively high temperatures, it has become customary to assume that the valve(s) fail at this time, resulting in full and permanent depressurization.[3] Even if the valve would reseat itself a few times, it would eventually fail due to rising helium temperatures, and the slightly later depressurization would have little effect on the long-term accident scenario.

The pressure history is one of the main variables of interest during this early phase of the accident. Subsequent to LOFC, the pressure is virtually uniform throughout the primary loop. From energy conservation for a fluid in a closed loop at uniform pressure, one obtains for the system pressure[4]

$$\dot{P}(t) = \frac{\kappa-1}{V_o} \sum_{i=1,N} Q_i(t)$$

where V_o is the total primary coolant volume, P is the loop pressure, Q_i is the net component-to-helium heat flow within component i, N is the number of components, and κ the specific heat ratio.

This relationship shows that the loop pressure history is only a function of the net heat transfer into or out of the coolant throughout the loop. In integrated form this can be expressed as

$$\frac{V_o}{\kappa-1} \quad (p_{blow}-p_o) = \int_o^{t_{blow}} \sum_{i=1,N} Q_i(t)\ dt$$

where p_o is the initial system pressure, p_{blow} the relief valve setpoint and t_{blow} the time to reach the relief valve setpoint.

For an operating system pressure of 1050 psia and a relief valve setpoint of 1125 psia, the loop coolant can absorb about 1580 MJ before depressurization would occur. At steady state, full-power operation, the coolant energy is 9710 MJ (referred to an assumed LCS sink temperature of 200 F). Thus, the additional energy that the coolant can absorb before depressurization, is only 16% of its initial energy. The steady state energy transfer from the core to the steam generators through the coolant is 2400 MW. These data show that the energy addition to the coolant that could cause depressurization is extremely small with respect to the energy transferred between core and steam generator through the coolant during normal operation, as well as during the coastdown to accident conditions. Thus, the times to depressurization can be strongly affected by relatively short-term effects during early parts of the transient, as will be demonstrated below.

METHOD OF ANALYSIS

A modified version of the RATSAM code[5] was employed for the analysis of these early primary loop transients. This code effects a simultaneous flow and thermal analysis of the complete primary loop with core and side cavities, including steam generator and CACS cooling, as well as thermal liner cooling. Transient solutions of the mass, momentum and energy balances for the coolant are coupled with transient solid energy balances for the core and side cavity structures. The code is limited to a set of about 30 to 50 fluid nodes representing the primary loop coolant, plus the same number of solid nodes representing core and side cavity solids. A transient solution of the momentum equation is not essential for the current problem, except possibly for the first 30 s while the circulators coast down and the MLIVs close. However, transient mass and energy conservation were essential for modeling of the coolant in the various large cavities with very slow flows.

A schematic of the nodalization used is shown in Figure 2. The core was divided into two parallel flow regions of about equal numbers of refueling regions, with the refueling regions lumped into the "hot" core channel having about 2/3 of the decay heat, and the "average" core channel having the remaining 1/3 of the decay heat. The parallel flow path representing the four steam generators as well as the parallel flow path representing the CACS were essentially dead ended after valve closure, however, with some net mass inflows as the side cavities remain cooler than the core. An additional parallel flow path represents the remaining steam generator bypass flow.

The possible in-core nodalization (two parallel flow regions and five axial nodes for each flow region) was less than desirable, but the accurate prediction of the weak in-core natural circulation flows is difficult and apparently requires higher nodalization in particular in the axial direction than currently available from any HTGR code.[2,6] Furthermore, accurate in-core flow predictions at these low flow rates would require better information on frictional resistance in particular through the top reflector orifices and through the irregular bottom reflector flow passages at very low Reynolds numbers.

Prior to application, the code, originally developed by GA, was extensively reviewed and revised to ensure an independent and reliable assessment of early UCHA transients. All resulting model changes have been documented in Appendix A of Reference 4. In particular, it was found that very early secondary side transients in the steam generator could have a pronounced effect on the pressure history. Therefore, mass and energy conservation equations for the steam generator secondary side heat transfer were added. Details of these secondary side models are given in Reference 7. The computation of convective heat transfer coefficients between the coolant and the various solid structures was improved using, in particular, more representative Reynolds numbers.

While some model improvements remain desirable, on balance, the code in its revised form was considered to represent the best currently available tool for initial assessments of primary loop transients during early UCHA sequences.

All results to be reported here did assume a functioning liner cooling water system (LCS). In some scenarios, like in the case of station blackout, this would not be so. Considering the heat transfer through the thermal barrier to the liner coolant tubes, with the kaowool insulation of the thermal barrier being the dominant resistance, in case of the water coolant flow coming to a halt, the surrounding concrete would provide a sufficient heat sink for several hours before it begins to heat up sufficiently to reduce the helium to thermal barrier heat transfer. Thus, the effect of the LCS is fairly small for the first few hours of UCHA transients, and results at most in slightly earlier depressurization which is well within the accuracy limits of the current results.

CODE APPLICATIONS AND RESULTS

During the process of modifying and applying the code, numerous applications were made for various conditions, in particular using different thermal liner and steam generator cooling options. Only a few typical results will be reported here.

The base case assumes a functioning LCS with an average coolant temperature of 200 F. Steam generator secondary side cooling is based on a feedwater flow ramp at scram from design flow to 15% of design flow over 170 s, based on GASSAR data for DBA practice,[8] and with a final drop to zero feedwater flow at 1000 s.

The transient is initiated by a circulator trip, followed by scram. The circulators coast down to zero speed at 24 s, with main loop isolation valve closure at 31 s. The resulting loop mass flows are shown in Figure 3. Following the initial coastdown, the flow reverses in parts of the core, and a natural circulation flow of about 10 lb/s is gradually established between hotter and cooler regions of the core. The steam generator bypass flow persists at about 1 lb/s throughout the transient, while CACS bypass flow rates are about 0.2 lb/s. The natural circulation flows between different regions of the core are clearly the most significant remaining flows.

The heat exchange between the helium and the core, and the steam generators, and the thermal barrier is shown in Figures 4 and 5. Very early in the transient, the total core-to-helium heat transfer initially exceeds the decay heat, as shown in Figure 4, and the steam generators absorb more heat from the helium than the helium receives from the core, resulting in a decrease of energy in the coolant and a corresponding drop in pressure (Figure 6). About 1100 MJ are removed from the helium during the first 30 s of the transient (Frame (b) of Figure 4), almost doubling the initial margin to depressurization of 1580 MJ. As shown in Figure 5, from about 200 s on, most of the heat transferred to the helium in the active core regions is absorbed in the reflectors, with the net transfer from the core to the helium being on the order of 2 to 3 MW. Absorption of heat by the steam generators drops to about 0.3 MW after 1500 s. The heat flow to the thermal barrier slowly builds up to about 1 to 2 MW. With decreased energy removal in the steam generators, net accumulation of energy in the coolant begins at about 750 s, with the net heat input from the core exceeding the heat transfer to the thermal barrier and to the steam generator by only 0.5 to 1 MW. However, the continuing net accumulation of this relatively small amount of power in the coolant leads to depressurization at 6400 s. The corresponding system pressure is given in Figure 6, showing a minimum at about 750 s, and reaching the relief valve setpoint of 1125 psi at about 6400 s.

Typical system temperatures are shown in Figures 7 through 9. At decay heat levels, the temperature gradient in the fuel elements is small, and the fuel can be considered to be essentially at the average solid region temperature. Fuel failures are not expected at these early times. Nor have any failure temperatures been reached for any of the other components, except for the relief valves which are conventionally assumed to fail with initial depressurization.

The most important conclusion from this Base Case was that the early heat removal from the coolant primarily to the steam generators was very essential in decreasing the overall internal energy of the coolant, thus significantly delaying the time of depressurization.

It should be noted that not only the core graphite represents a significant thermal capacitance to which the primary loop coolant is exposed, but that the steam generator thermal capacitances are of the same order. Thus, early secondary side heat transfer and the thermal capacitance of the steam generators have a very significant effect on the early primary loop transients.

The steam generator cooling assumed in the Base Case, of 15% of feedwater flow up to 1000 s cannot be maintained in certain accident scenarios, in particular in cases of station blackout.

While over most of the Base Case, discussed above, heat removal by the steam generators was small, its initial contribution over the first 700 s was very significant, and since the integral of net heat flow into the coolant determines the blowdown time, early but strong heat removal by the steam generators can have drastic long term effects on the transient. Therefore, as a limiting case, a computation was made with zero heat removal from the secondary side of the steam generators.

The mass flows for this case are shown in Figure 10. The early mass flow decreases are identical to the Base Case data. However, the long term mass flows are slightly higher in this case, due to larger buoyancy forces between hot and average core channels.

The heat flows to and from the coolant are shown in Figures 11 and 12. The early heat flows to and from the helium are in essence similar to those of the Base Case, with the heat flow to the steam generator initially being absorbed in its solid capacitances. However, the very slightly lower amount of heat being absorbed in the steam generator accounts for the energy in the helium to remain approximately constant, in contrast to the Base Case, where a significant initial decrease in fluid energy was observed (Frames (b) of Figures 4 vs. 11). Similarly, the longer term heat flows (Figures 5 vs. 12) are qualitatively similar to those of the Base Case. However, the heat absorption in the steam generators in the first 1500 s is now significantly lower than in the Base Case, and as a result, the coolant energy decrease from 30 to 310 s is much less than in the Base Case, and the loop pressure begins to rise after only 310 s, as shown in Figure 13, resulting in a faster pressure increase and leading to depressurization after only 2500 s, compared to 6500 s in the Base Case.

It should be noted that the assumption of zero secondary side heat transfer is a limiting one, as there will always be some heat transfer to the remaining secondary side coolant. However, in station blackout initiated accidents the heat absorbed by the secondary side coolant might indeed be very small and depressurization times close to this limiting value cannot be excluded without further, more detailed evaluations.

The various component temperatures for this case remained very close to the corresponding temperature histories of the Base Case. The upper plenum temperatures were about 100 F higher in this case, and the lower plenum temperatures were slightly lower. These slightly higher gas temperature difference across the core increased the buoyancy forces and resulted in larger mass flows (Figure 10).

SUMMARY AND CONCLUSIONS

A modified version of the RATSAM code was applied to analyze early primary loop heatup transients subsequent to loss of all forced circulation. Some future improvements in the models remain desirable, in particular finer core nodalization and better information on core flow resistances at very low flow rates. With the current capabilities of the code, a few specific cases were analyzed. It was found that the steam generator secondary side flow conditions during the first few 100 s of the transient can have a very significant effect on the depressurization times which can range from 2000 to 8000 s.

Depressurization times previously reported by others have been significantly longer.[3,9] The major differences between those computations and the ones reported here are due to our more detailed secondary side modeling as well as due to some improvements in design input data, heat transfer coefficients and similar items. Some of our changes in input data might not be applicable for certain specific situations. However, they do represent cases of interest during UCHA scenarios.

Thus, with the uncertainties of the analyses in mind, it is concluded that while the longer depressurization times from previous work may be representative for certain cases, the shorter ones obtained here are at least as likely for many scenarios of interest.

Reference

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8. "General Atomic Standard Safety Analysis Report," (GASSAR), General Atomic Co.
9. Direct Communication with GA.

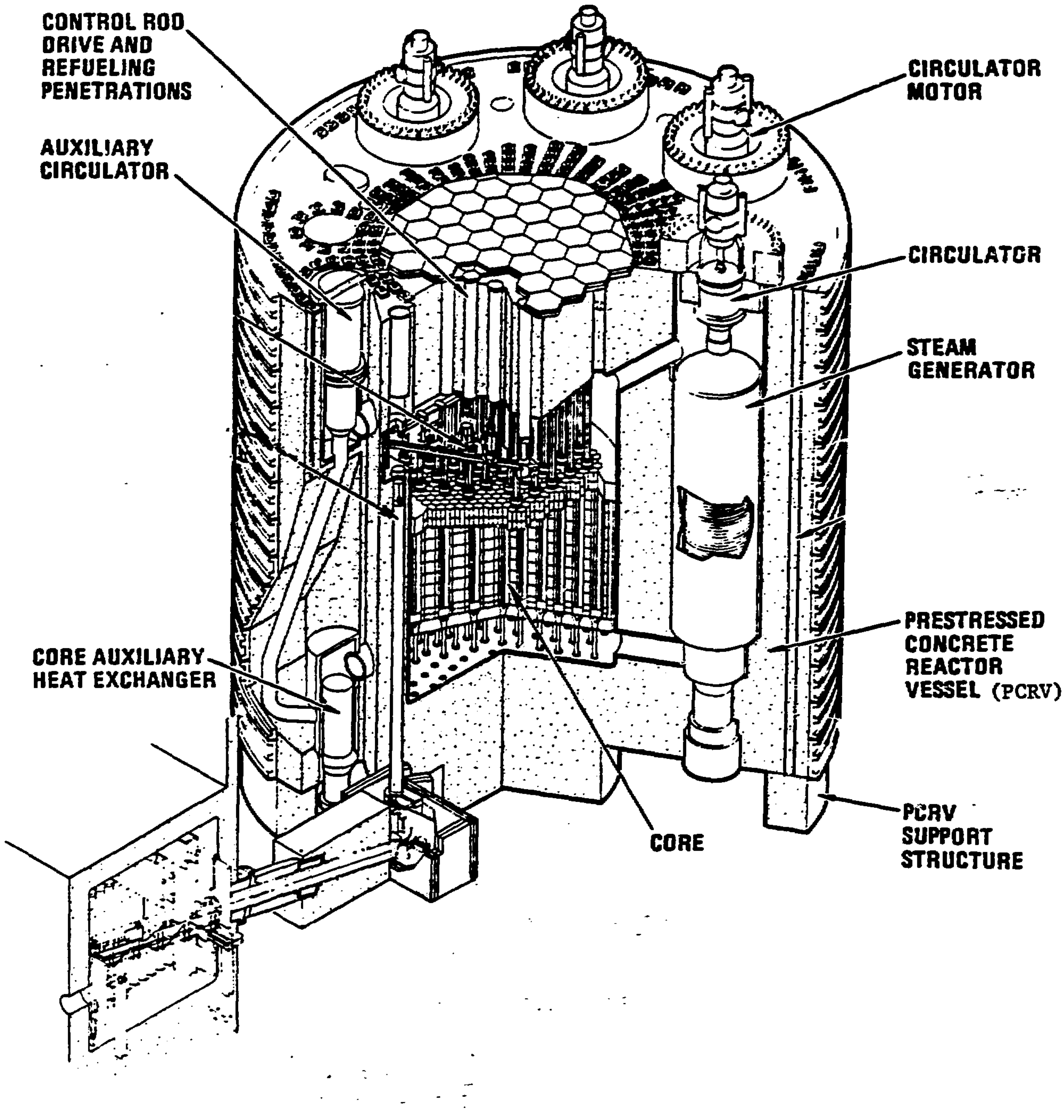


Figure 1 Typical HTGR Nuclear Steam Supply System (from GA Document PC-000040).

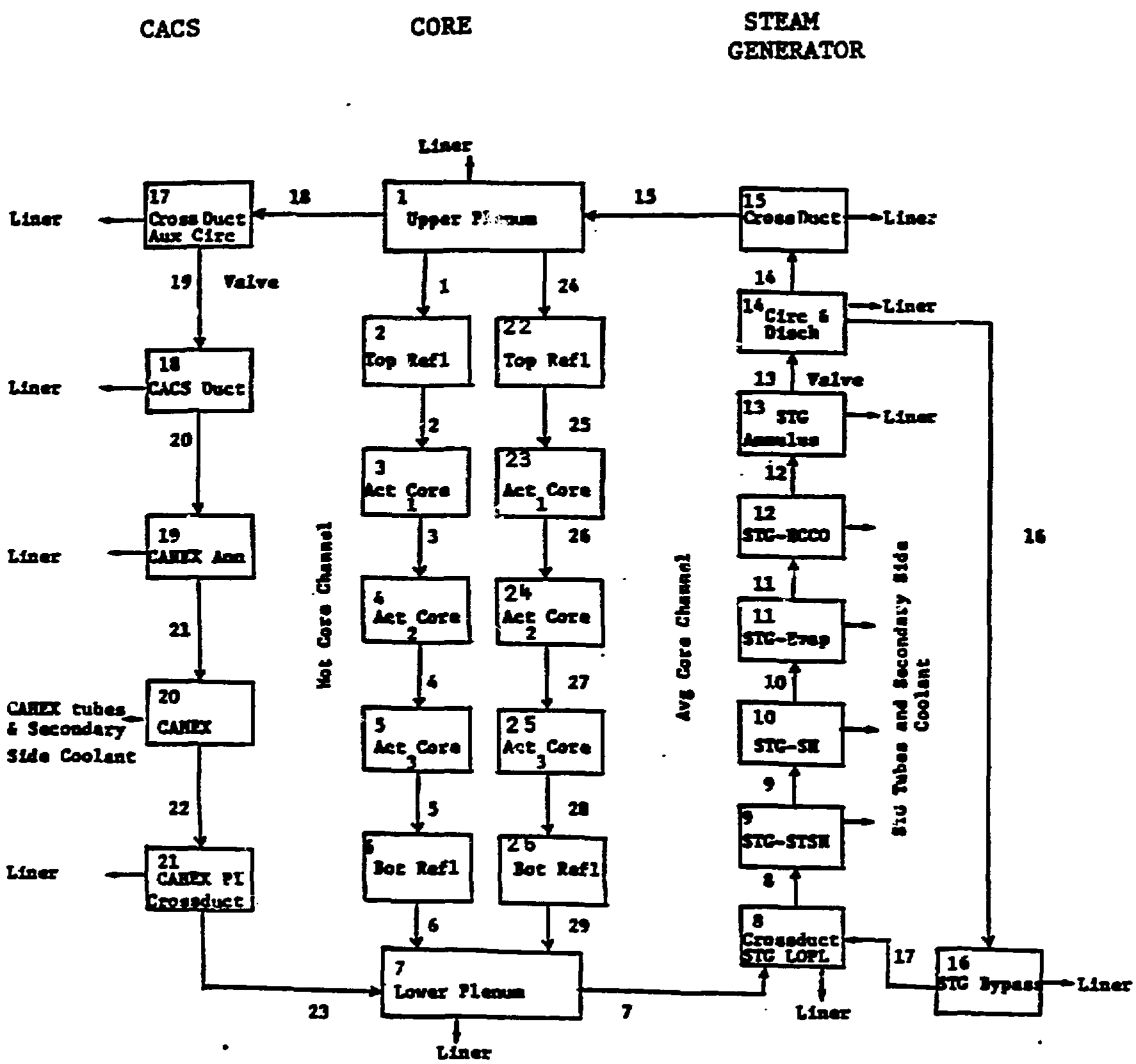


Figure 2 . Schematic of Nodalization for UCHA Analysis.

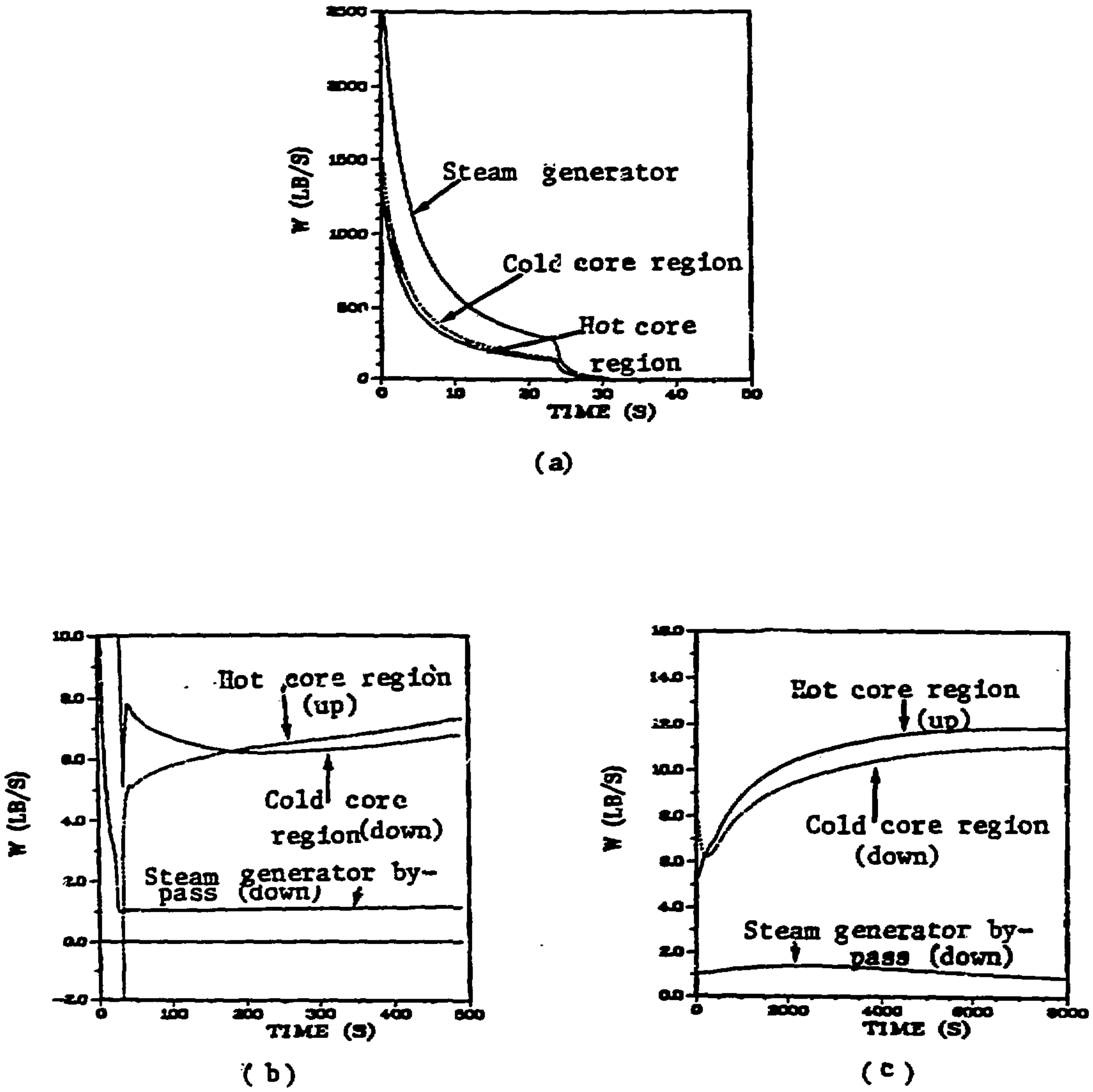


Figure 3. Mass flows (W) in the primary loop during a UCHA transient prior to depressurization (all time scales begin at trip time).

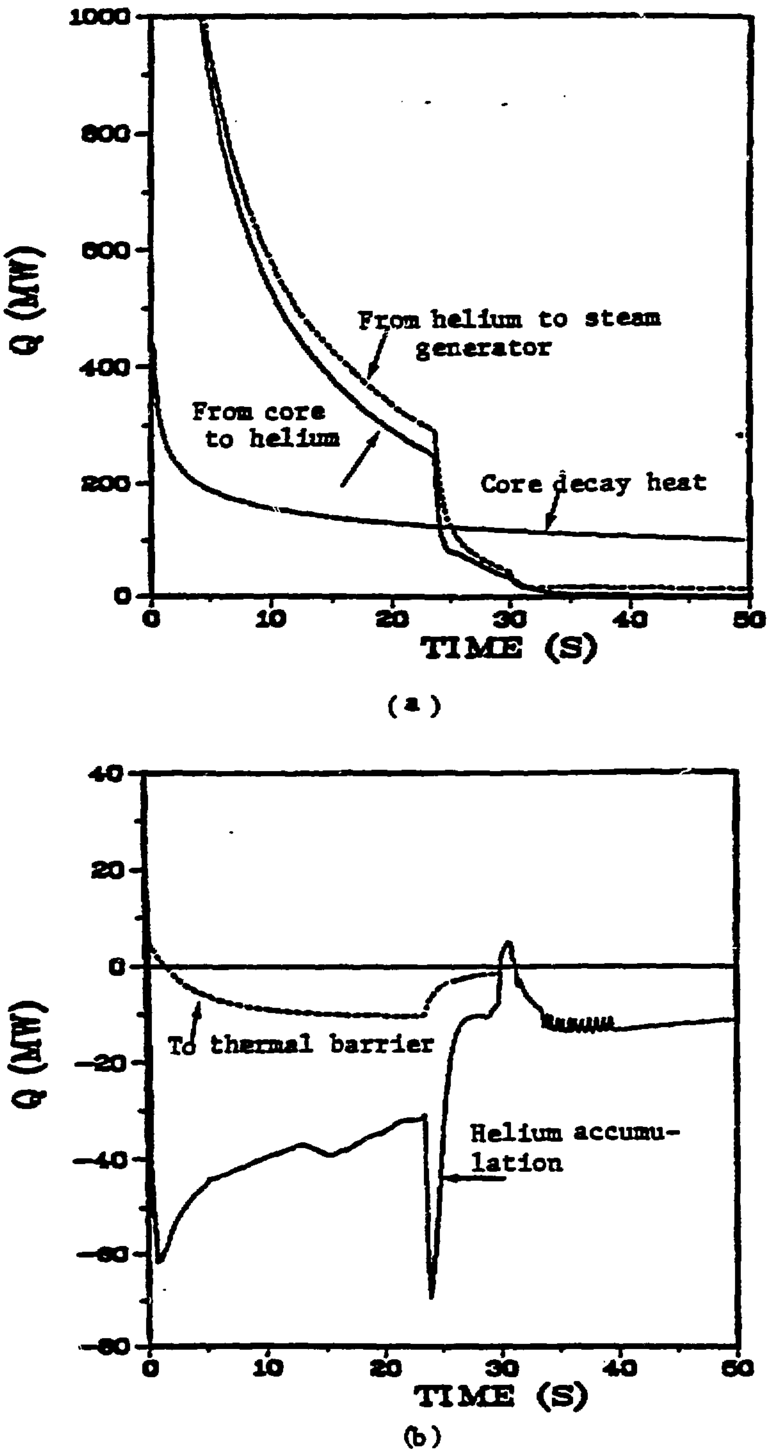


Figure 4. Early helium heat flows (Q) in the primary loop during a UCHA transient prior to depressurization (all time scales begin at trip time).

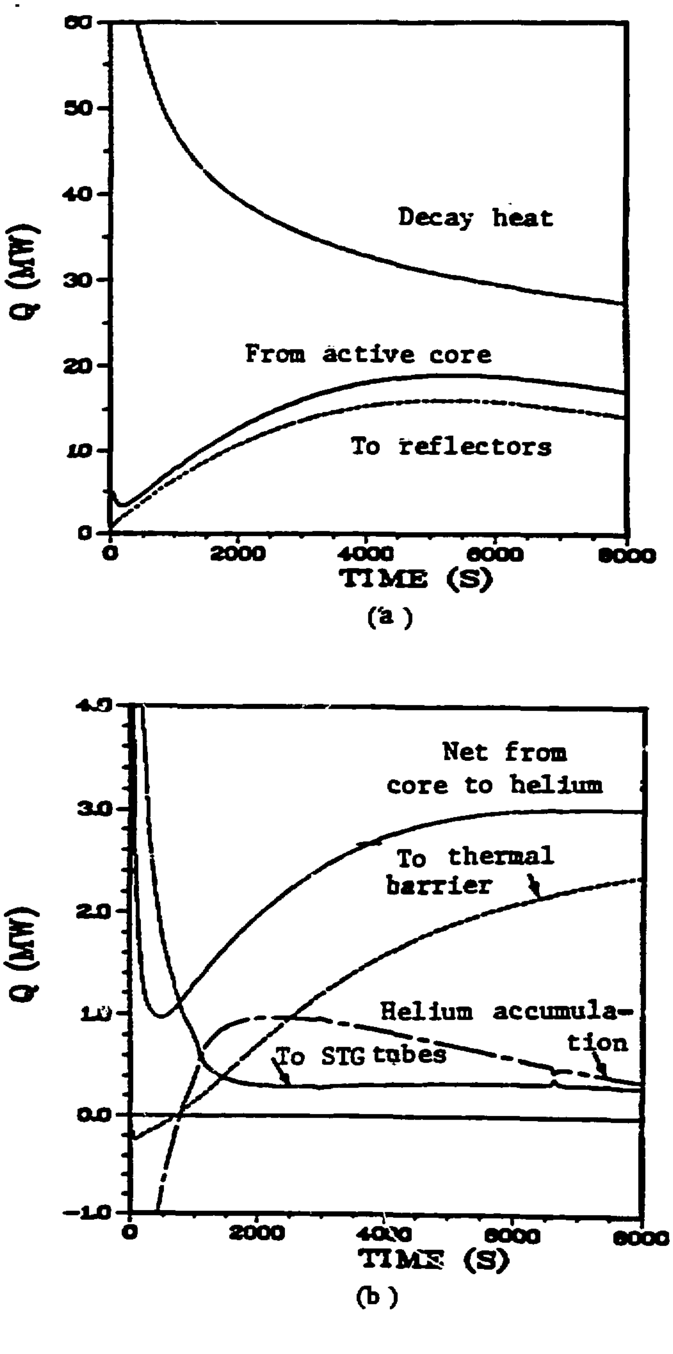


Figure 5. Long-term helium heat flows (Q) in the primary loop during a UCHA transient prior to depressurization (all time scales begin at trip time ; STG = steam generator).

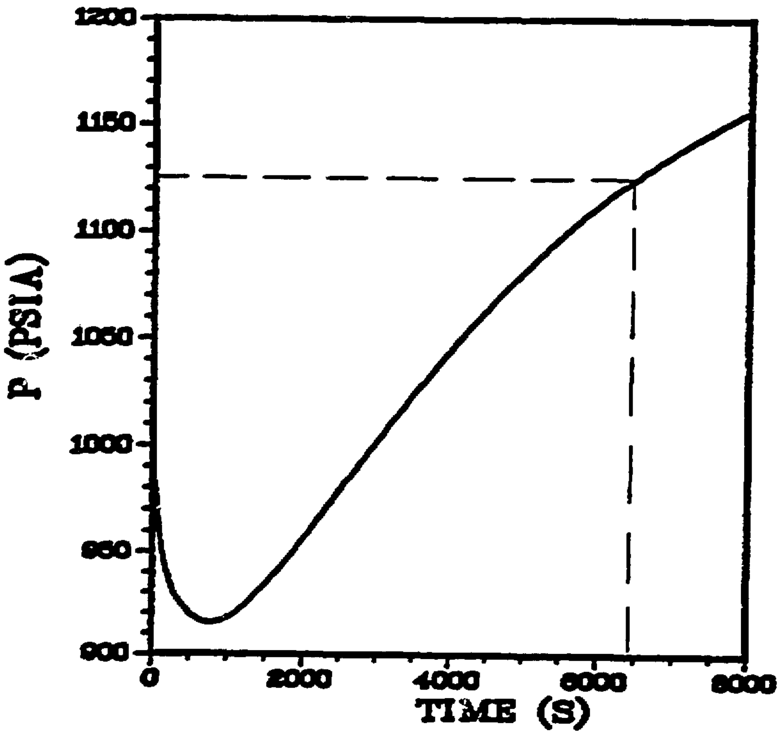
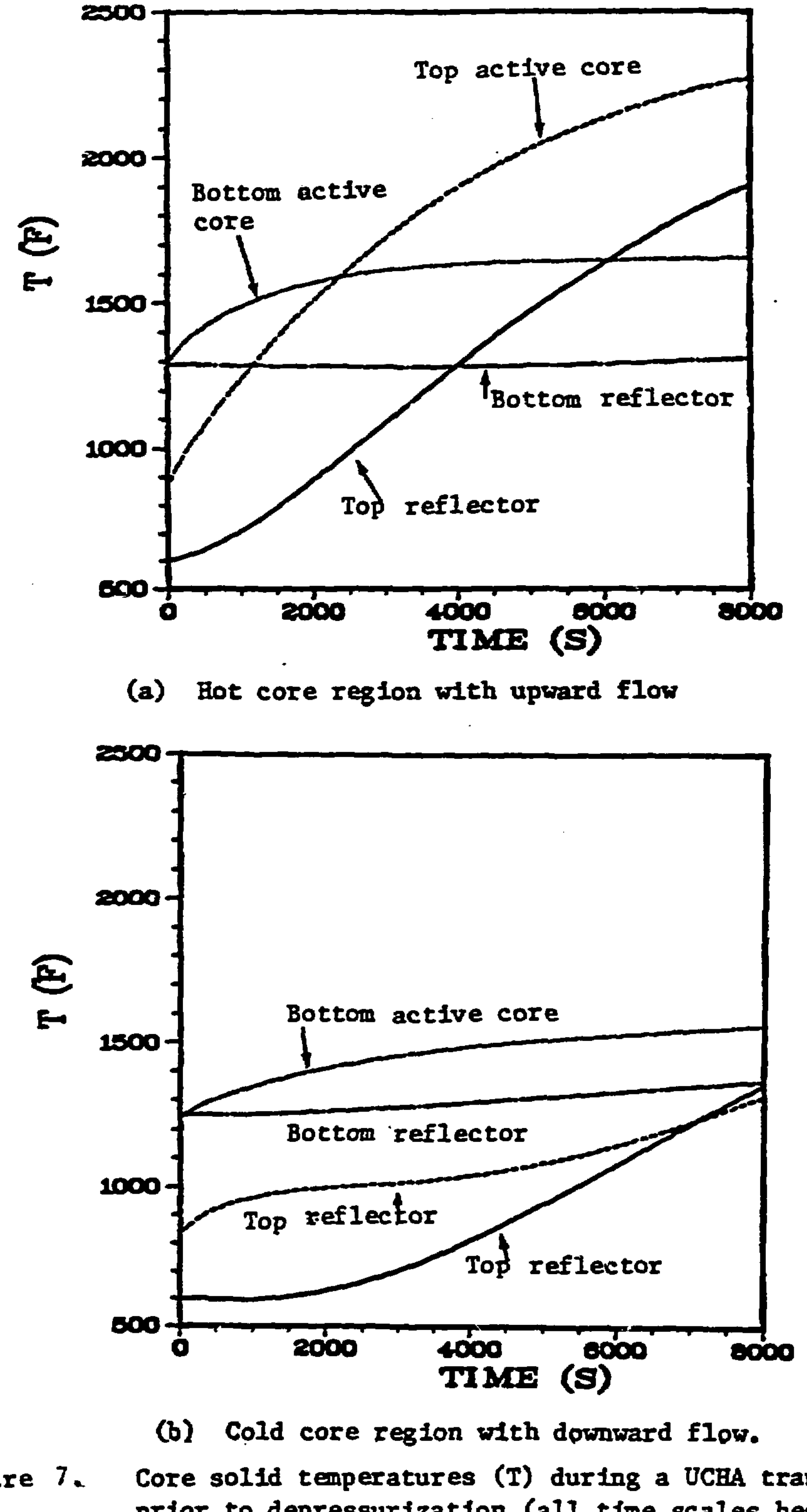


Figure 6. System pressure (P) in the primary loop during a UCHA transient prior to depressurization (all time scales begin at trip time).



(a) Hot core region with upward flow

(b) Cold core region with downward flow.

Figure 7. Core solid temperatures (T) during a UCHA transient prior to depressurization (all time scales begin at trip time).

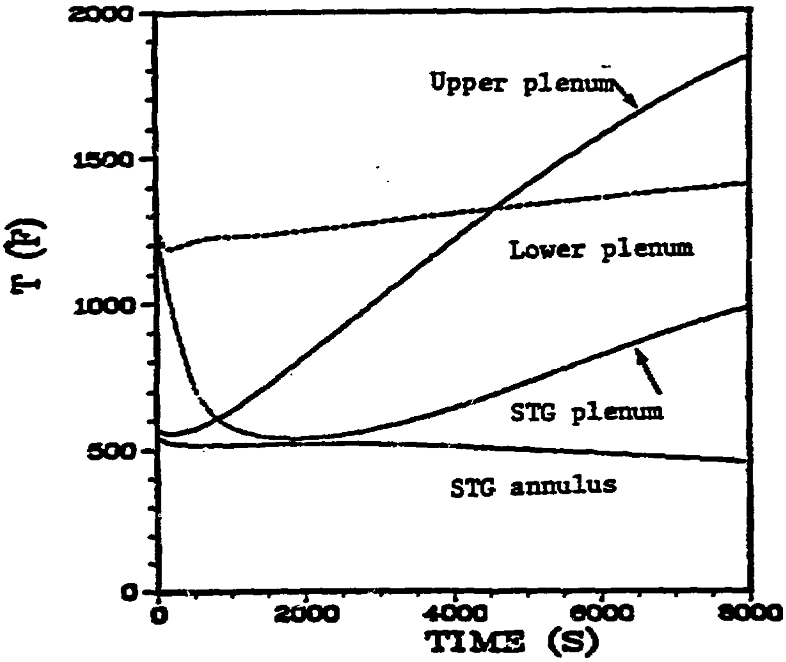


Figure 8. Loop helium temperatures (T) during a UCHA transient prior to depressurization (all time scales begin at trip time).
(STG = Steam Generator)

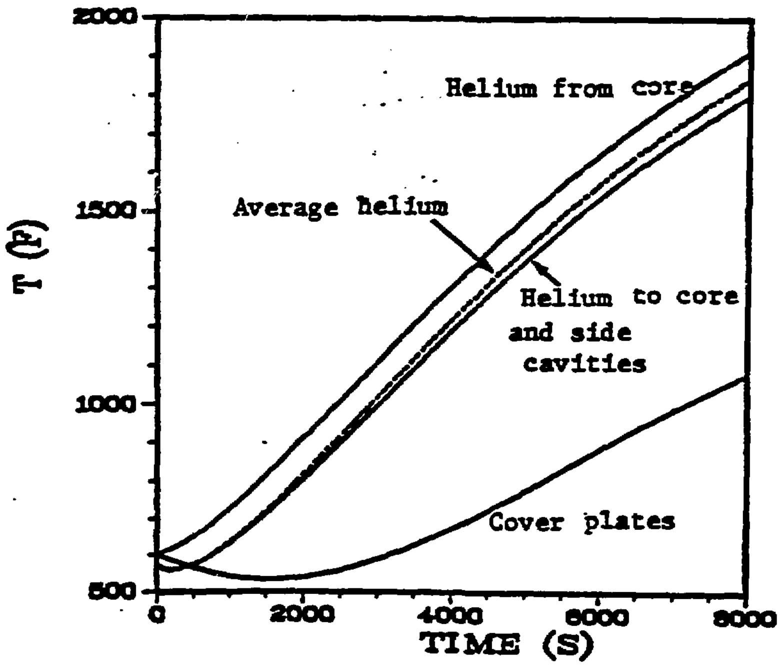


Figure 9. Upper plenum temperatures (T) during a UCHA transient prior to depressurization (all time scales begin at trip time).

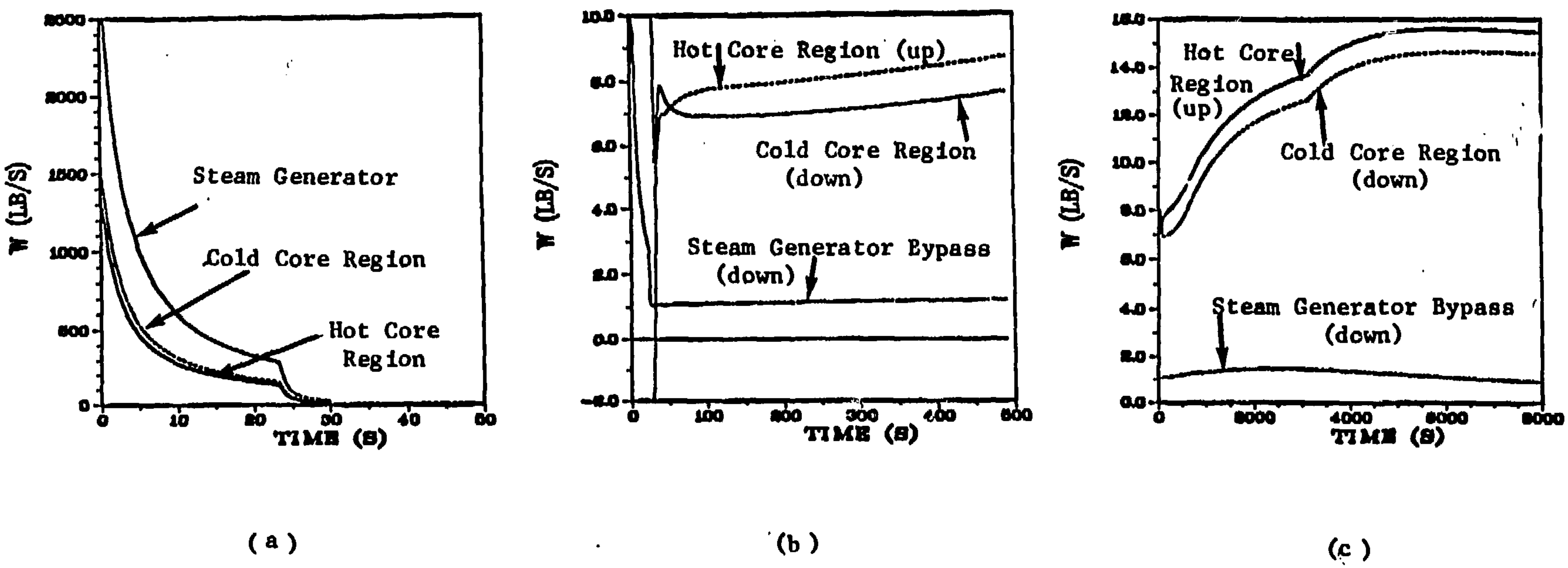
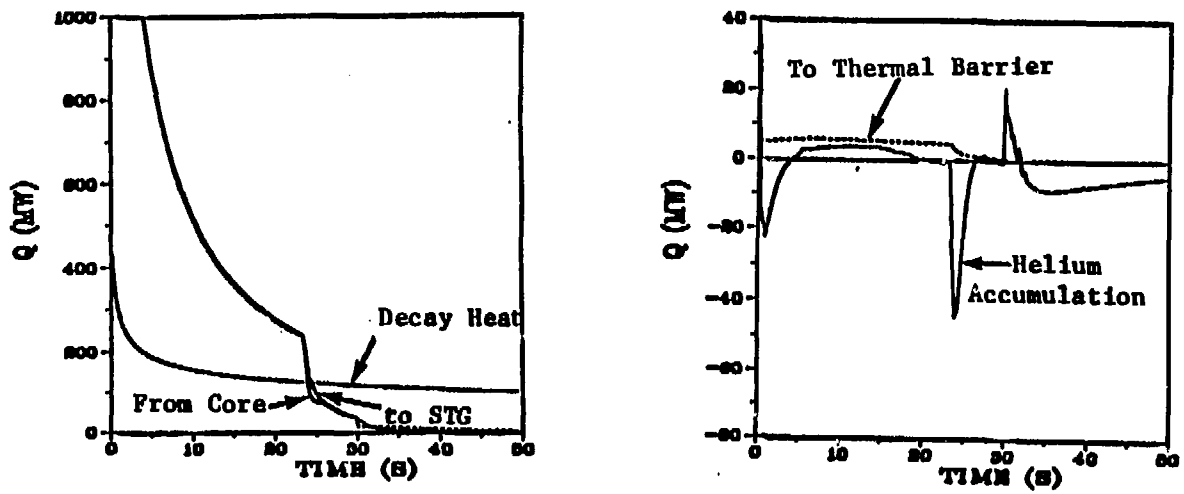
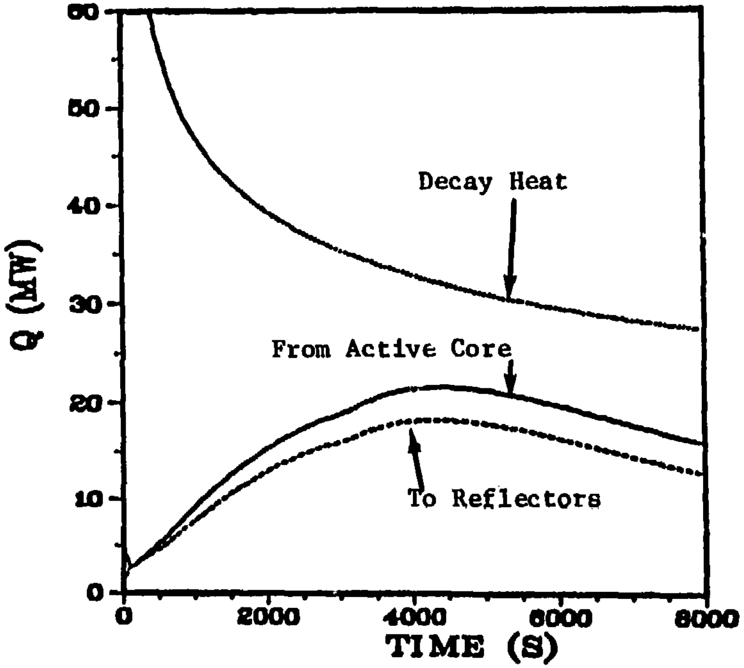


Figure 10. Mass Flows in the Primary Loop During an UCHA Transient Prior to Depressurization; No steam generator secondary side cooling (All time scales beginning at trip time.

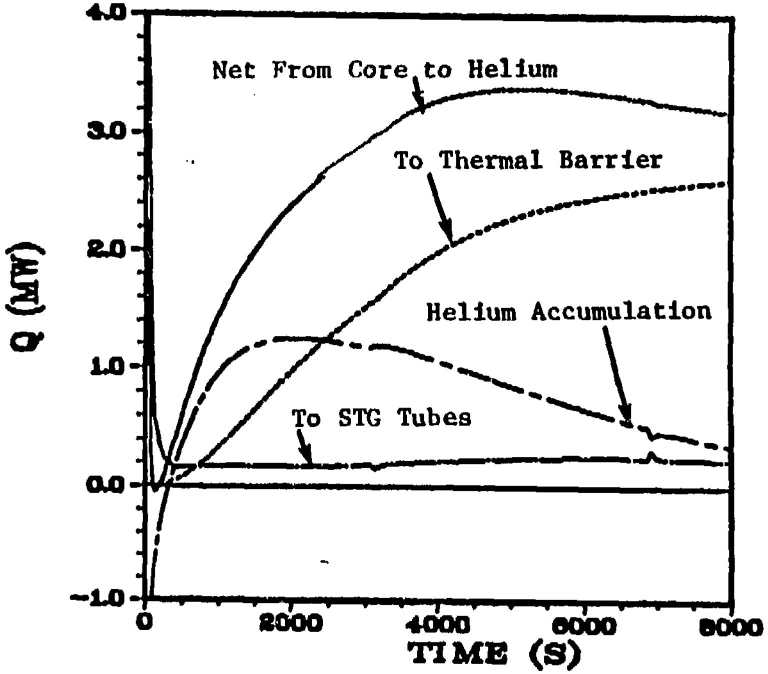


(a) (b)

Figure 11. Early Helium Heat Flows in the Primary Loop During an UCHA Transient Prior to Depressurization; No Steam Generator Secondary Side Cooling, (All Time Scales Beginning at trip Time; STG - Steam Generator).



(a)



(b)

Figure 12. Long Term Helium Heat Flows in the Primary Loop during an UCHA Transient Prior to Depressurization; No Steam Generator Secondary Side Cooling. (All Time Scales Beginning at Trip Time; STG = Steam Generator).

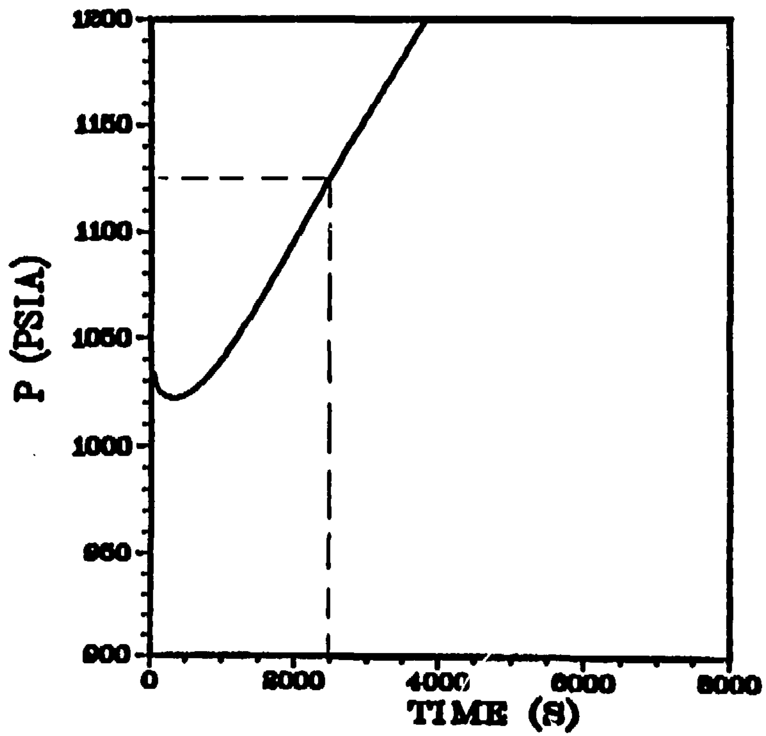


Figure 13. System Pressure in the Primary Loop During an UCHA Transient Prior to Depressurization; Case 2. (All time scales beginning at trip time).