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STEADY-STATE AND LOSS-OF-PUMPING ACCIDENT ANALYSES OF THE SAVANNAH RIVER NEW PRODUCTION REACTOR REPRESENTATIVE DESIGN

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ABSTRACT

This document contains the steady-state and loss-of-pumping accident analysis of the representative design for the Savannah River heavy water new production reactor. A description of the reactor system and computer input model, the results of the steady-state analysis, and the results of four loss-of-pumping accident calculations are presented.

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ABBREVIATIONS

AC	alternating current
C	Centigrade
DBA	design basis accident
DC	direct current
delta-p	pressure difference
DOE	Department of Energy
ECCS	emergency core cooling system
F	Fahrenheit
ft	feet
gpm	gallons per minute
hp	horsepower
HWR	heavy water reactor
kPa	kilopascal
Li-Al	Lithium-Aluminum
LOCA	loss-of-coolant accident
LOPA	loss-of-pumping accident
m	meters
MW	megawatts
NPR	new production reactor
p	pressure
psi	pounds per square inch
psia	pounds per square inch absolute
psid	pounds per square inch difference
rad	radians
s,sec	second
SNL	Sandia National Laboratories
SRS	Savannah River Site
TRAC	Transient Reactor Analysis Code
U-Al	Uranium-Aluminum
WSRC	Westinghouse Savannah River Company

ACKNOWLEDGEMENTS

The authors wish to gratefully acknowledge the support provided by Westinghouse Savannah River Company staff. In particular, the technical assistance and program oversight contributed by Wade Bickford and Harry Hootman were greatly appreciated.

SUMMARY

Westinghouse Savannah River Company (WSRC) is responsible for coordination of confirmatory thermal-hydraulics and safety engineering development in support of the heavy water new production reactor (HW-NPR). In support of that effort, Sandia National Laboratories (SNL) has used an existing thermal-hydraulics computer code (Transient Reactor Analysis Code, or TRAC) to develop a computational model of an HW-NPR, based on a representative system defined by WSRC.

The primary objective of the analyses in this report is to provide the NPR program with a computational tool for evaluation of key thermal-hydraulic phenomena which have severely limited reactor power in existing Savannah River reactors, and which will influence safety behavior in the HW-NPR. In the future, this capability will be integrated with associated programs for evaluation of thermal-hydraulic codes and experimental verification of code predictive capabilities. The key thermal-hydraulic variables or phenomena of interest which must be evaluated under the new upflow conditions introduced by the HW-NPR include:

- bounds to natural circulation capability,
- onset of two phase flow (boiling),
- critical heat flux,
- onset of flow instability and recovery from flow instability, and
- material temperature limits.

This report describes the first phase of this program, the development of a steady-state model. Included are descriptions of the representative reactor system used as a basis for the TRAC input and the resulting computational model, and a description of results used to indicate proper behavior of, and hence validation of, the model. Also given are results of loss-of-pumping accident (LOPA) analyses, in which natural circulation is a significant factor. The remaining variables of interest (critical heat flux, flow instability, etc) will be the subject of a future report, for which the TRAC model described here will be further developed to allow simulation of bounding design basis conditions during a large break loss-of-coolant accident (LOCA).

The HW-NPR is expected to include numerous features to improve the thermal-hydraulic safety margin under historical design basis accident (DBA) conditions, while still allowing efficient power operation for the production of special nuclear materials. These features include upflow through the core, an internal downcomer in the reactor tank, and adequate elevation difference between the core and heat exchangers, all of which promote improved natural circulation capability. The design may also include pressurizer/accumulators and high volume emergency coolant injection, which promote sustained coolant flow through the core during a LOCA. The representative

system defined by WSRC for development of the TRAC input model includes such improvements in safety design, the goal being to maintain solid water in the core under all postulated DBA conditions. The TRAC model developed by SNL fully represents this system, including fuel assemblies, reactor vessel piping, pumps, heat exchangers, and pressurizer/accumulators.

The TRAC model assumes that a heavy water reactor (HWR) operating at 2500 MW deposits 94%, or 2350 MW, in the fuel assemblies. The remaining heat is deposited in the moderator space and structural components. The core contains 438 assemblies, each consisting of 3 concentric Uranium/Aluminum (U-Al) fuel tubes and 2 concentric Lithium/Aluminum (Li-Al) target tubes. The average flow through each assembly is 0.021 m3/s (325 gpm), with a 248 kPa (36 psi) pressure drop. The elevation of the top of the heat exchangers is approximately level with the top of the reactor vessel. The secondary side cooling system includes a storage pool, with elevation above the primary heat exchangers to allow for gravity flow on loss of secondary pumps.

The scope of this report includes analysis of steady-state behavior and LOPA analysis for the following four cases:

1. DC primary pumps on, 100% secondary flow

2. DC primary pumps off, 100% secondary flow

3. DC primary pumps on, 10% secondary flow (i.e., gravity flow)

4. DC primary pumps off, 10% secondary flow

The results indicate that the steady-state model performs as expected, based on performance of production reactor designs of a similar nature, and will provide an acceptable basis from which to begin evaluation of thermal-hydraulics code capability and HW-NPR behavior under more challenging conditions.

The results of the LOPA studies indicate that the HW-NPR can be configured to maintain a full water-solid condition during transition to natural circulation for the four scenarios analyzed, and that the TRAC computer code is fully capable of simulating such transients

1. INTRODUCTION

The heavy water new production reactor (HW-NPR) program is using a triad approach to assuring full achievement of safety in the new reactor design. This includes 1) design contractor integration of stated Department of Energy (DOE) safety requirements, 2) oversight by a DOE national laboratory dedicated to NPR safety (Los Alamos National Laboratory), and 3) confirmatory development for the HWR engineering program coordinated by Westinghouse Savannah River Company (WSRC), the contractor operator to the Savannah River Site (SRS). Technical working groups have further been established by the DOE to provide guidance in the evaluation of existing analytical techniques in the areas of physics and thermal-hydraulics, and the need for additional code development and/or experimental confirmation.

Sandia National Laboratories (SNL) is assisting WSRC in the development and initial evaluation of thermal-hydraulic computational tools for confirmation of HW-NPR behavior. The previous experience which makes this program important is based on the current generation of SRS reactors. A re-evaluation of accident assumptions and modeling found that, if operated at their design limit of 2500 MW, boiling and flow instability would occur in fuel channels under postulated large break loss-of-coolant accident (LOCA) conditions. Such complex thermal-hydraulic phenomena are extremely difficult to model with computer simulation to a high degree of confidence, and the uncertainty has led to a policy to reduce power to assure water-solid conditions are maintained for all postulated accidents. Such a finding for the NPR program late in the design phase could seriously impact program schedule, or could seriously compromise the production capability of the new reactor.

The goal of this analysis is to provide a starting point for the evaluation of thermal-hydraulic computer simulations under the new conditions introduced by a representative HW-NPR. This includes upflow of coolant through the core, and an emphasis on passive natural circulation capabilities for accident mitigation. The initial predicted behavior of the representative system will also provide the HWR development program and technical working groups with feedback on the likely expectations for behavior of the HW-NPR under postulated accident conditions, and the safety margin under those conditions. Analysis of the code capability and possible safety margin together will give an initial indication of the need for additional development work. If water-solid conditions can be maintained in the NPR under the postulated conditions, the need for development of more sophisticated thermal-hydraulic tools may not be necessary for confirmation of behavior under DBA conditions. The ultimate objective is to provide assurance that all thermal-hydraulic phenomena expected in a HW-NPR during bounding DBAs will be well within the capabilities of computer simulation (and experimental data upon which the code correlations are based), and that no previously unidentified safety issue will be raised late in the design phase of the program.

The scope of work planned under this program has several phases. The first phase, the results of which are reported here, involves the development of a representative HWR system and a computer simulation of that system using an existing thermal-hydraulics code. The simulation is then exercised under steady-state and loss-of-pumping accident (LOPA) conditions, and evaluated to provide validation that the code is functioning correctly. Later phases of this work will then apply to more sophisticated transient analyses, including a large break LOCA.

This report is divided into five sections. Following the introduction, a description of the physical characteristics of representative HW-NPR with upflow is presented, followed by a description of the computer input model used to perform the analyses. The results of the steady-state analysis and the LOPA analysis are then presented, followed by conclusions of the study.

2. SYSTEM DESCRIPTION

In this section, the reactor vessel and primary cooling system of the heavy water new production reactor (HW-NPR) are described and the steps of the subsequent modeling process are detailed. The HW-NPR system configuration presented in this report includes improved safety design features intended to avoid or assist in mitigating severe accidents.

Section 2.1 provides a description of the assumptions made in the process of developing a HW-NPR representative design. Section 2.2 provides a description of the reactor vessel, fuel assemblies, and other major components comprising the primary cooling system.

2.1 Assumptions

Numerous assumptions were necessary to arrive at a representative design for the HW-NPR. Consideration was given to pertinent NPR design requirements and design limitations of the current SRS production reactor configuration in determining a representative design. The most influential design assumptions are given below, including a rationale where appropriate.

a. Information from the Heavy Water Reactor Preliminary Design Report (ARD-88-10)[1], which describes design characteristics of the NPR, applies in this report except where noted.

b. The NPR vessel will have flow up through the core, unlike the current production reactor design. An upflow design aids the natural tendency of heated, more buoyant (lower density) fluid to seek a higher level, or rise within the core. A downflow design opposes this tendency.

c. The NPR vessel will have an internal downcomer design, also unlike the current production reactor design. This allows a significant volume of fluid to reside upstream of the lower plenum, which will aid in keeping the core flooded during some accident scenarios.

d. Each loop will have a pressure control and injection component that functions as both a pressurizer and an accumulator located downstream of the pump (between the pump and reactor vessel). This location enhances pressure stability in the core region since pressurizer/accumulator injection would be directly upstream of the reactor vessel. To avoid flow instability problems, maintaining system pressure in the core region immediately following an accident is critical. Decreasing pressure decreases the saturation temperature, which leads to voiding in the core and interruption of assembly coolant flow, or flow instability. An ECCS system will also provide injection to the vessel. However, ECCS injection, which has no effect on prevention of or recovery from flow instability in the first seconds of a LOCA, is not modeled in this study.

e. The top of the heat exchangers will be below the top of the upper plenum. This is to ensure that no positive head exists at the top of the reactor vessel, which allows the top of the vessel to be removed for refueling or maintenance without spilling inventory.

f. The elevation of the pump and pressurizer/accumulator will be approximately the same as that of the heat exchangers. This reduces the overall length of loop piping, brings the pump above lower floor levels that could flood and affect the pump's performance, minimizes the possibility of siphoning effects on lower elevation piping, and simplifies the overall loop piping.

g. The fuel assemblies will be positioned as far below the top of the downcomer annulus as possible. In this configuration, since the water level in the downcomer always remains above that of the fuel assemblies, positive gravity head will tend to keep the fuel covered with water.

h. The moderator system and the primary system are fully separate to minimize the effects of severe accidents. If the primary and moderator systems were coupled and if fuel were to melt and relocate, it could enter the moderator space and add to the reactivity.

2.2 System Components

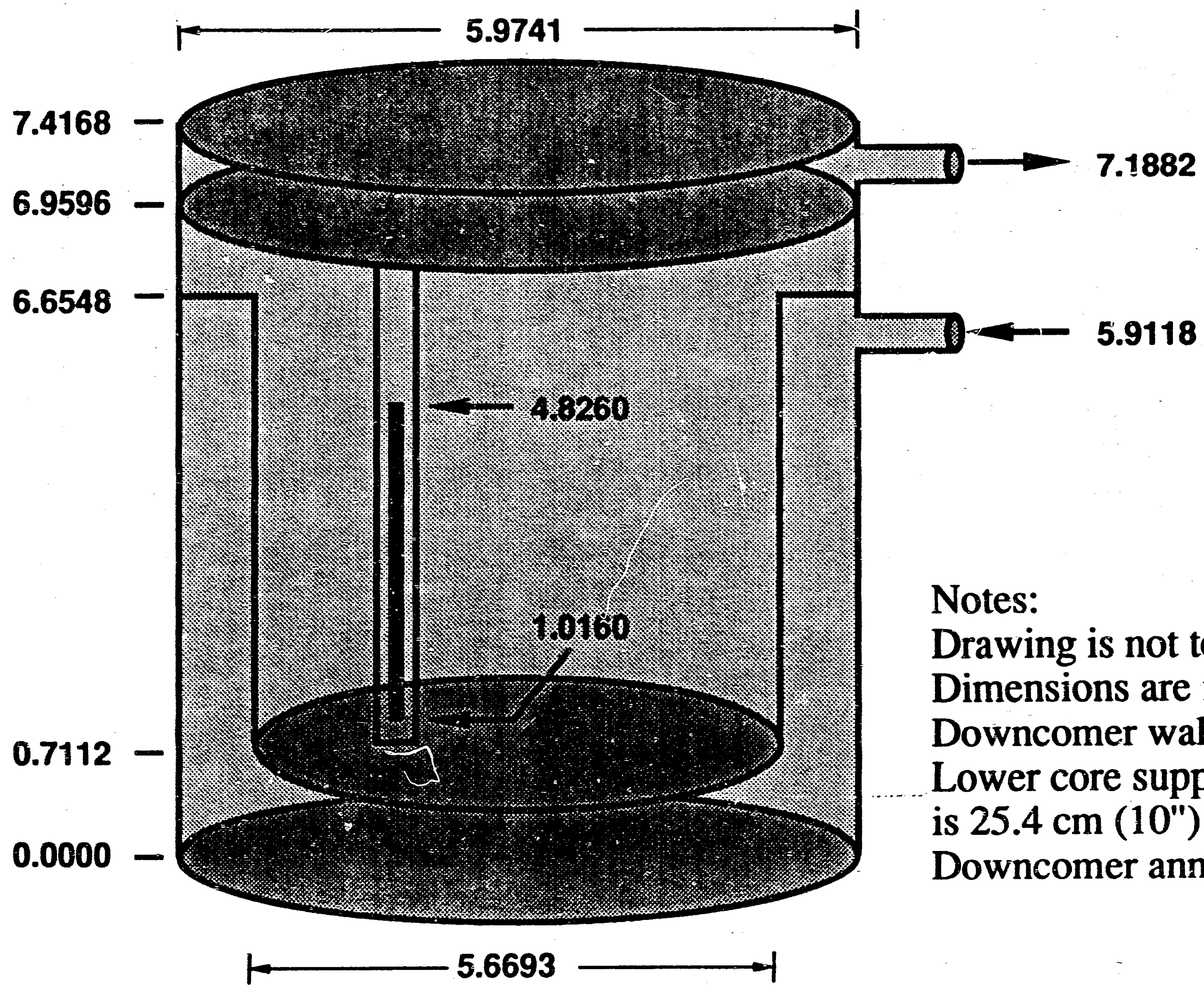
In this section, the components that comprise the HW-NPR primary cooling system, including the fuel assemblies and reactor vessel, will be briefly described. Information on these components related to the thermal-hydraulic modeling process will be included.

2.2.1 Reactor Vessel

The reactor vessel will incorporate several design changes from the current production reactor design. The two most important changes are the upflow direction of coolant flow and the addition of a downcomer within the vessel. All dimensions of the vessel and pertinent elevations within the core are given in Figure 2-1.

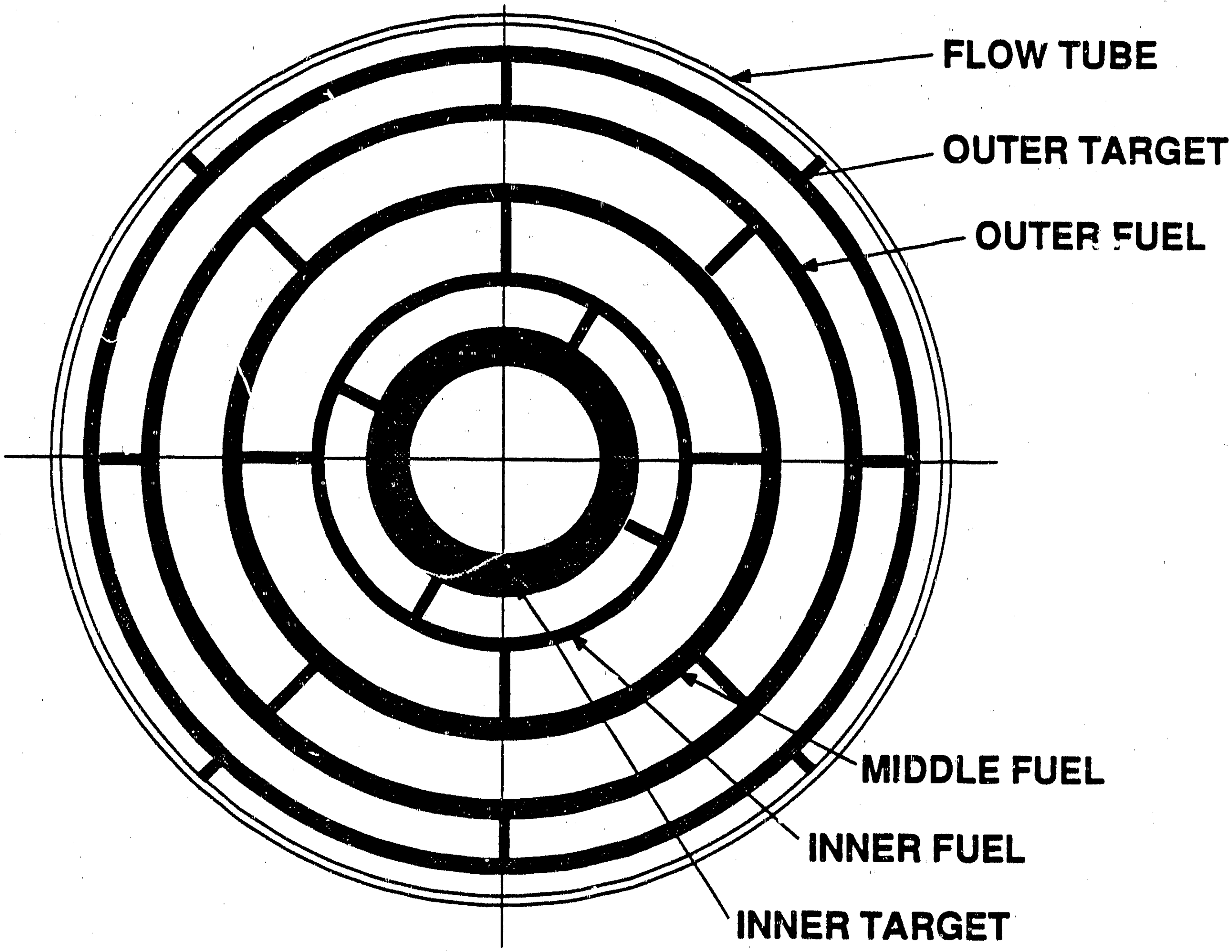
2.2.2 Fuel Assemblies

The proposed NPR reactor vessel design contains 438 fuel assemblies. Assuming a total reactor maximum power of 2500 MW, each fuel assembly will have a power of approximately 5.7 MW. The fuel assemblies are modeled as a five tube design (2 Li-Al target tubes and 3 U-Al fuel tubes) rather than the four tube design (2 Li-Al target tubes and 2 U-Al fuel tubes) currently utilized in the production reactors. Figure 2-2 shows the cross-section of the proposed HW-NPR fuel assembly.



Notes:
Drawing is not to scale.
Dimensions are in meters.
Downcomer wall is 5.1 cm (2") thick.
Lower core support plate/lower shield is 25.4 cm (10") thick.
Downcomer annulus is 10.2 cm (4").

Figure 2-1. Reactor Vessel Dimensions.



Fuel Assembly Dimensions (cm)

Flow Tube	OD	13.051
	ID	12.797
Outer Target Tube	OD Clad	12.141
	OD Core	11.989
	ID Core	11.882
	ID Clad	11.730
Outer Fuel Tube	OD Clad	10.566
	OD Core	10.414
	ID Core	10.198
	ID Clad	10.046
Middle Fuel Tube	OD Clad	8.235
	OD Core	8.082
	ID Core	7.755
	ID Clad	7.602
Inner Fuel Tube	OD Clad	5.766
	OD Core	5.613
	ID Core	5.372
	ID Clad	5.220
Inner Target Tube	OD Clad	3.988
	OD Core	3.835
	ID Core	2.934
	ID Clad	2.781

Figure 2-2. NPR MK-22 Assembly Cross-section and Dimensions.

2.2.3 Primary Cooling System

The primary cooling system functions to remove heat from the reactor vessel to a secondary system. The four loops in the primary system exit the vessel from the upper plenum and return to the top of the downcomer. The vessel and a characteristic loop of the representative design are shown in Figure 2-3. Each loop contains three heat exchangers, a centrifugal pump, and a pressurizer/accumulator. ECCS inlet lines also enter each primary loop on the cold leg near the reactor vessel, although ECCS injection will not be included in the primary system model for this study because it has no effect on flow instability during a large-break LOCA. All primary system hot and cold leg piping consists of 0.508 meter (20 inch) inner diameter lines. The moderator volume and flow are separate from the primary system. Dimensions of primary cooling system components are given in Figure 2-4. Other important primary system operational parameters are given in Table 2.1.

Table 2.1

Primary System Parameters

Total Primary System Flow	8.96 m^3/s (142,000 gpm)
Total Secondary Flow	11.51 m^3/s (182,400 gpm)
Total Primary Water Volume	128.7 m^3 (34,000 gal)
Pressurizer/Accumulator Pressure	1207 kPa (175 psi)

<u>Heat Exchangers</u>

Each of the four primary cooling system loops is equipped with three counterflow heat exchangers in parallel to remove heat generated in the core. The secondary system coolant flows on the shell side to remove primary system heat. As previously mentioned, the elevation of the heat exchangers is slightly below that of the top of the reactor vessel to allow removing the vessel head without spilling primary coolant inventory. All pertinent heat exchanger parameters are given in Figure 2-5.

<u>Pumps</u>

The primary cooling system contains four pumps, one for each loop, to deliver sufficient coolant to the reactor vessel during operating conditions. Each pump's operational characteristics are given in Table 2.2 below.

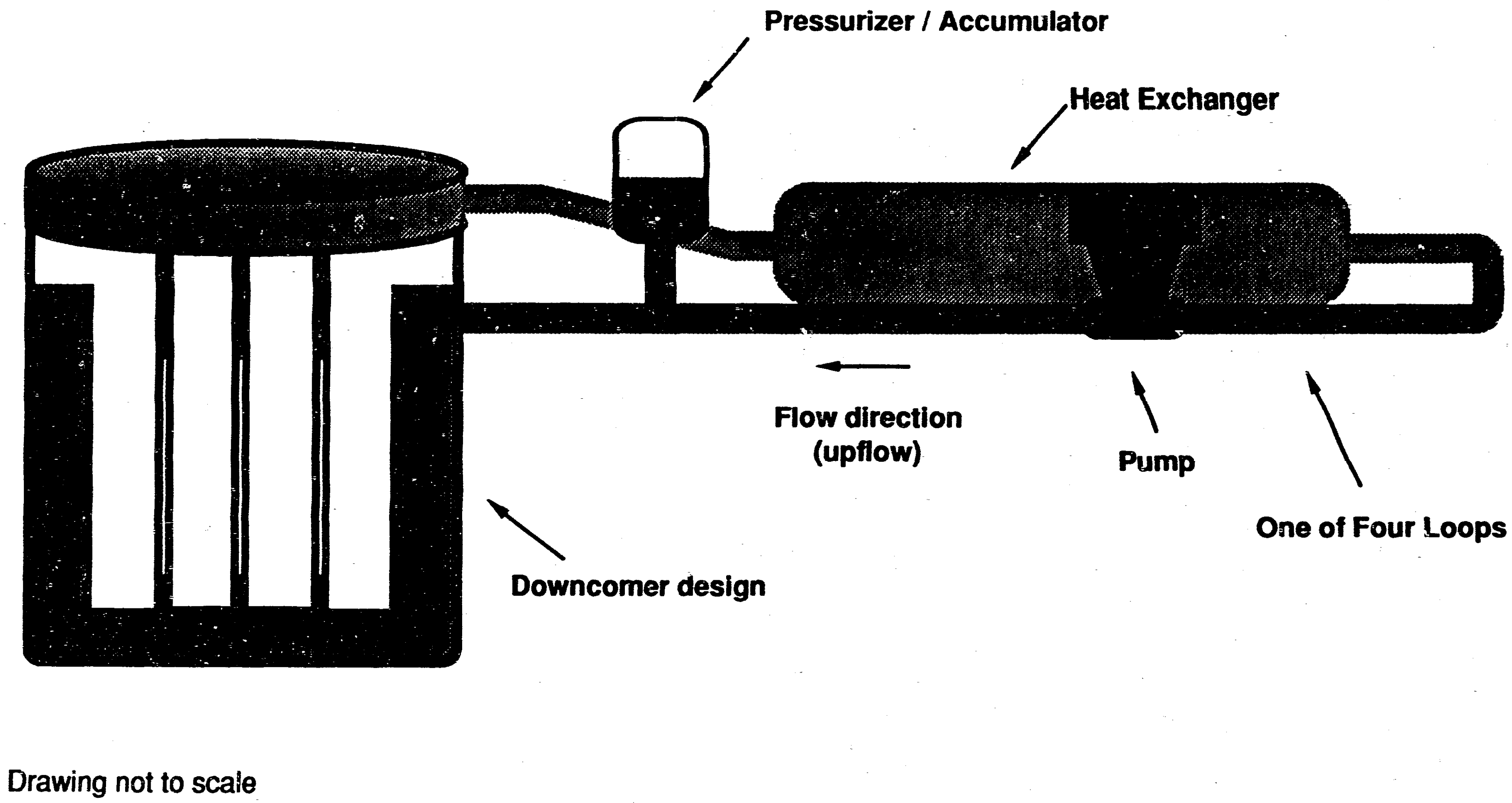


Figure 2-3. NPR Primary Cooling System Representative Design

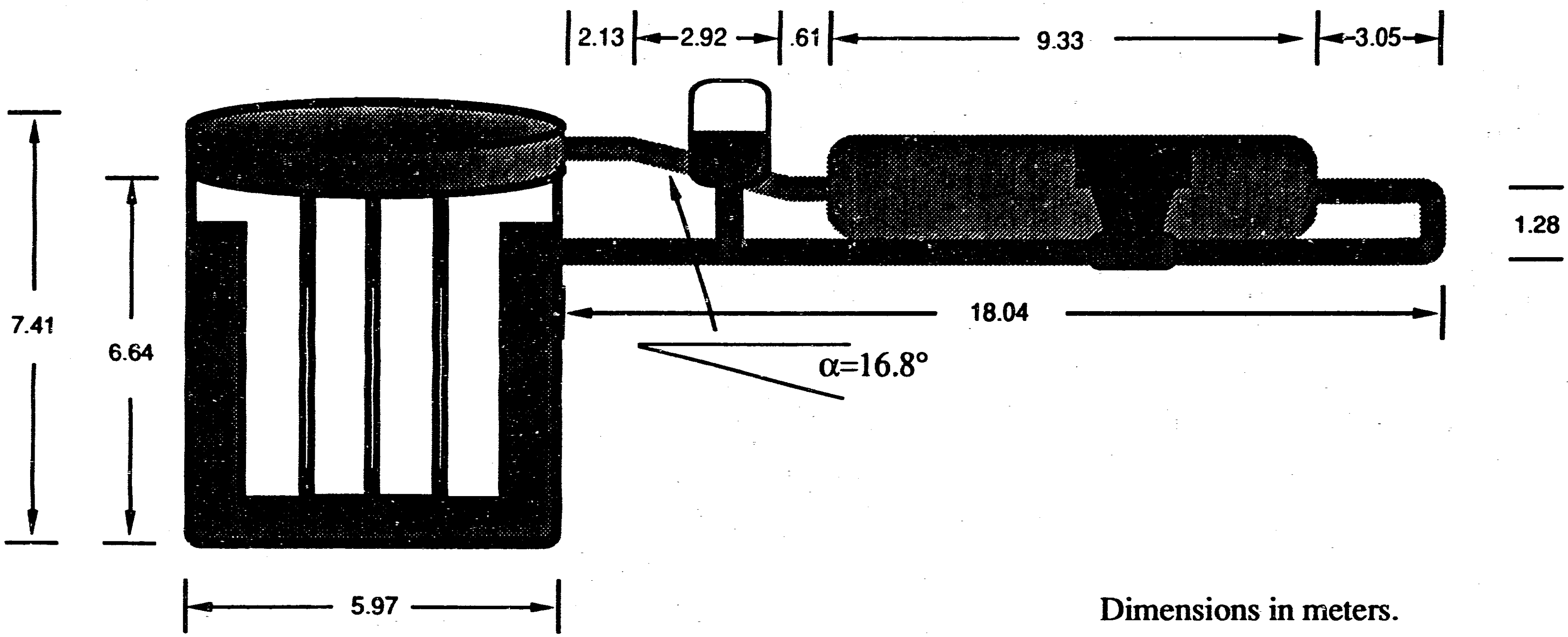
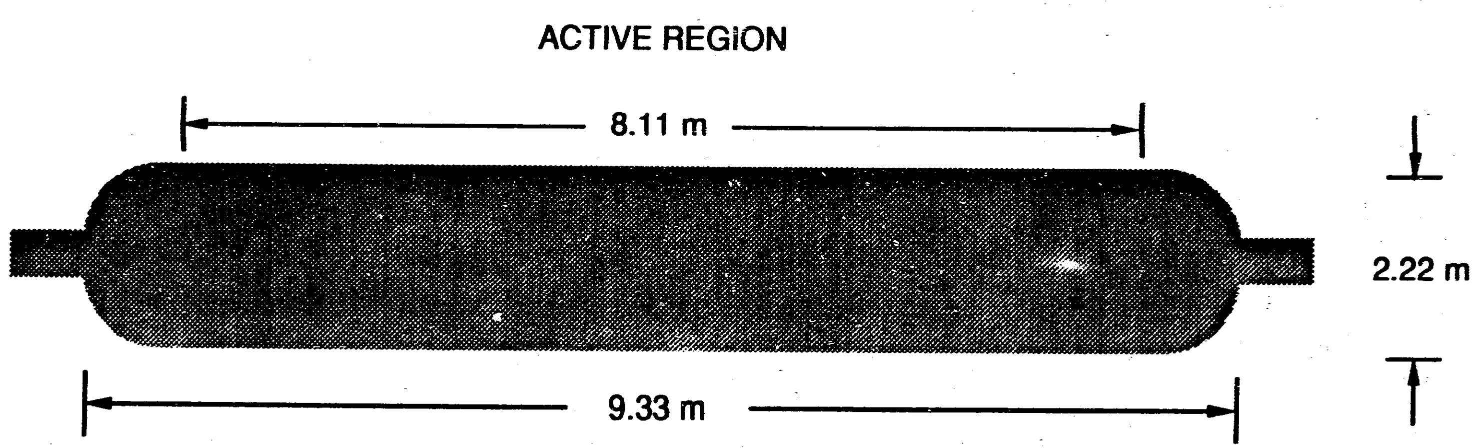


Figure 2-4. Dimensions of NPR Primary Cooling System.

HEAT EXCHANGER MODEL



Operational Characteristics:

primary flow = 0.747 m^3/s (11862 gpm)
secondary flow = 0.960 m^3/s (15200 gpm)
inlet temp (primary) = 106.7 °C (224.0 °F)
outlet temp (primary) = 43.3 °C (110.0 °F)
inlet temp (secondary) = 22.2 °C (72.0 °F)
outlet temp (secondary) = 71.0 °C (159.8 °F)
pressure drop (primary) = 515 kPa (74.7 psi)

Figure 2-5. Heat Exchanger Parameters.

Table 2.2

NPR Pump Parameters	
Power	3150 hp
Rated Head	91.4 m (300 ft)
Rated Speed	104.7 rad/sec
Rated Flow	2.25 m^3/s (35,587 gpm)
Pump Delta-P	888 kPa (128.8 psi)

Pressurizer/Accumulators

Each loop contains a pressurizer/accumulator component located between the pump and the reactor vessel on the cold leg. These components function to maintain pressure and volume control in the primary system by responding to abnormal pressure fluctuations. Each is capable of injecting 11.4 m^3 (3000 gal) of water into the cold leg near the inlet to the vessel.

3. MODEL DESCRIPTION

The following sections describe the development of a complete TRAC mod:l for the reactor vessel and primary cooling system. Prominent features of the model are given first, followed by a description of each component included in the model. The process of developing, testing, and validating each system component model separately before including all component models in the complete system model is described.

3.1 Model Features

Given below are significant features of the HW-NPR system TRAC model:

- the reactor vessel is modeled as a three-dimensional component,
- the fuel and loop components are modeled as one-dimensional components,
- the fuel assemblies are equally divided among core regions,
- pump models utilize existing Savannah River homologous pump curves, and
- the three parallel heat exchangers in each loop are lumped into one equivalent heat exchanger model.

3.2 Model Components

The complete steady-state model contains 129 components, which include 80 ROD (heat structure), 28 PIPE, 4 STGEN (heat exchanger), 4 PUMP, 4 TEE, 4 FILL (boundary condition), and 4 BREAK (boundary condition) components and 1 VESSEL component. These components are described in detail in Reference 2. The division of the reactor vessel, fuel assemblies and pipe components into separate TRAC component models, and further into the appropriate number of cells, was completed with the intention of including enough detail to accurately model all important processes without creating a computationally cumbersome (and costly) model. In the following subsections, the modeling process for each of the individual components of the primary cooling system is described.

3.2.1 Reactor Vessel

The reactor vessel is modeled using the TRAC 3-dimensional VESSEL component. The 3-dimensional nature of this component is important to model the reactor flow patterns of many postulated transients and accidents. The reactor vessel model is divided in the radial, axial and circumferential directions. Figure 3-1 shows the radial (5 segments) and axial (7 segments) cell division pattern for the vessel.

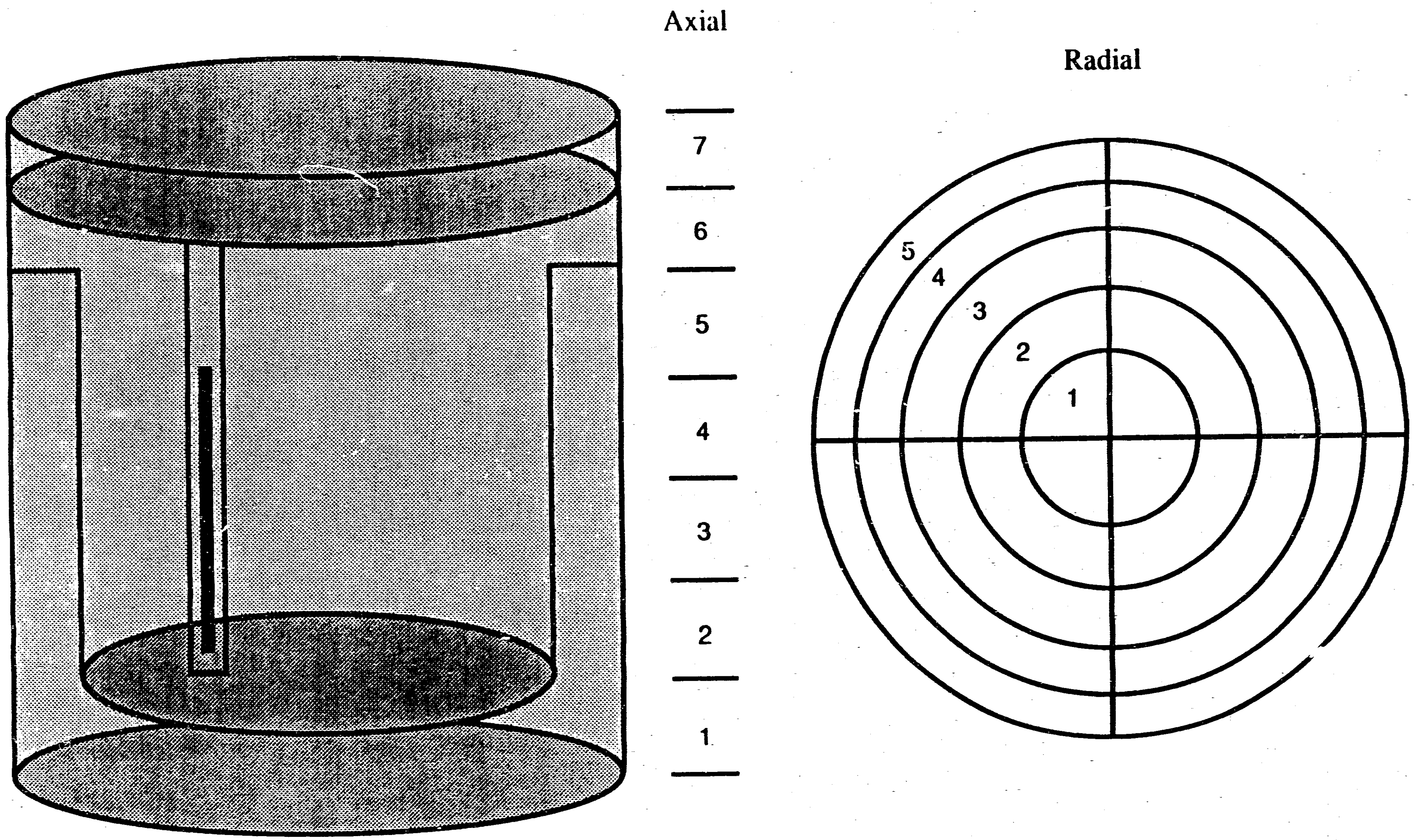


Figure 3-1. VESSEL Model Radial and Axial Cell Division.

Four 90-degree circumferential divisions are used to define quadrant numbers 1, 2, 3, and 4. These numbering conventions greatly simplify the process of assigning numbers to all components associated with the reactor vessel and fuel assemblies, as well as the coolant loops.

As shown in a cross-sectional view of the reactor vessel (Figure 3-2), the radial and circumferential segmentation creates 20 areas or vessel regions. The radius of each radial segment in the core region was chosen such that the innermost 16 areas (i.e., vessel regions 1 to 16 in Figure 3-2), representing the fuel assembly locations, are equal. These areas are numbered 01 to 04 from quadrant 1 to quadrant 4, respectively, at the cell center. This pattern continues outward to the outermost radial segment representing areas 17 to 20.

This segmentation of the reactor volume allows the modeler to conveniently assign certain reactor areas to segmented volumes. The lowest axial segment of the reactor (axial segment 1 in Fig. 3-1) represents the lower plenum, including the lower core support plate. The highest axial segment (axial segment 7 in Fig. 3-1) represents the upper plenum. The vessel outlet connections to each of the four loops are in the upper plenum. The internal downcomer is represented by the outermost radial segment of the reactor, areas 17 to 20 in Figure 3-2. Only axial segments 2 to 5 (see Figure 3-1) of this outermost radial segment model the downcomer. The four loop cold legs enter the vessel downcomer at axial segment 5. The active core region of the model is contained within the four innermost radial segments, vessel regions 01 to 16 in Fig. 3-2. The active core region includes only axial segments 2 through 5, since axial segments 1 and 7 represent the lower and upper plenums, respectively, and axial segment 6 represents the upper core shield region. These regions are graphically represented in Figure 3-3.

3.2.2 Fuel Assemblies

The fuel assemblies are modeled using the ROD and PIPE components of TRAC. Each of the 438 assemblies contains 3 U-Al fuel tubes and 2 Li-Al target tubes. The fuel tubes and target tubes are modeled using ROD (heat structure) components, which have a power producing option. However, this option was only utilized for the fuel tubes since the target tubes produce only negligible power. The flow channels between the fuel and target tubes were modeled using PIPE components. As a simplification, only one PIPE component was used to represent all the coolant flow channels in each fuel assembly modeled. This is shown graphically in Figure 3-4. The flow areas for all channels within the assembly were lumped together to form one flow area in the PIPE component, with the exception of the channels inside the inner target tube and outside the outer target tube, which are very low flow areas. The equivalent hydraulic diameter expression used for the PIPE component was four times the flow area divided by the wetted perimeter.

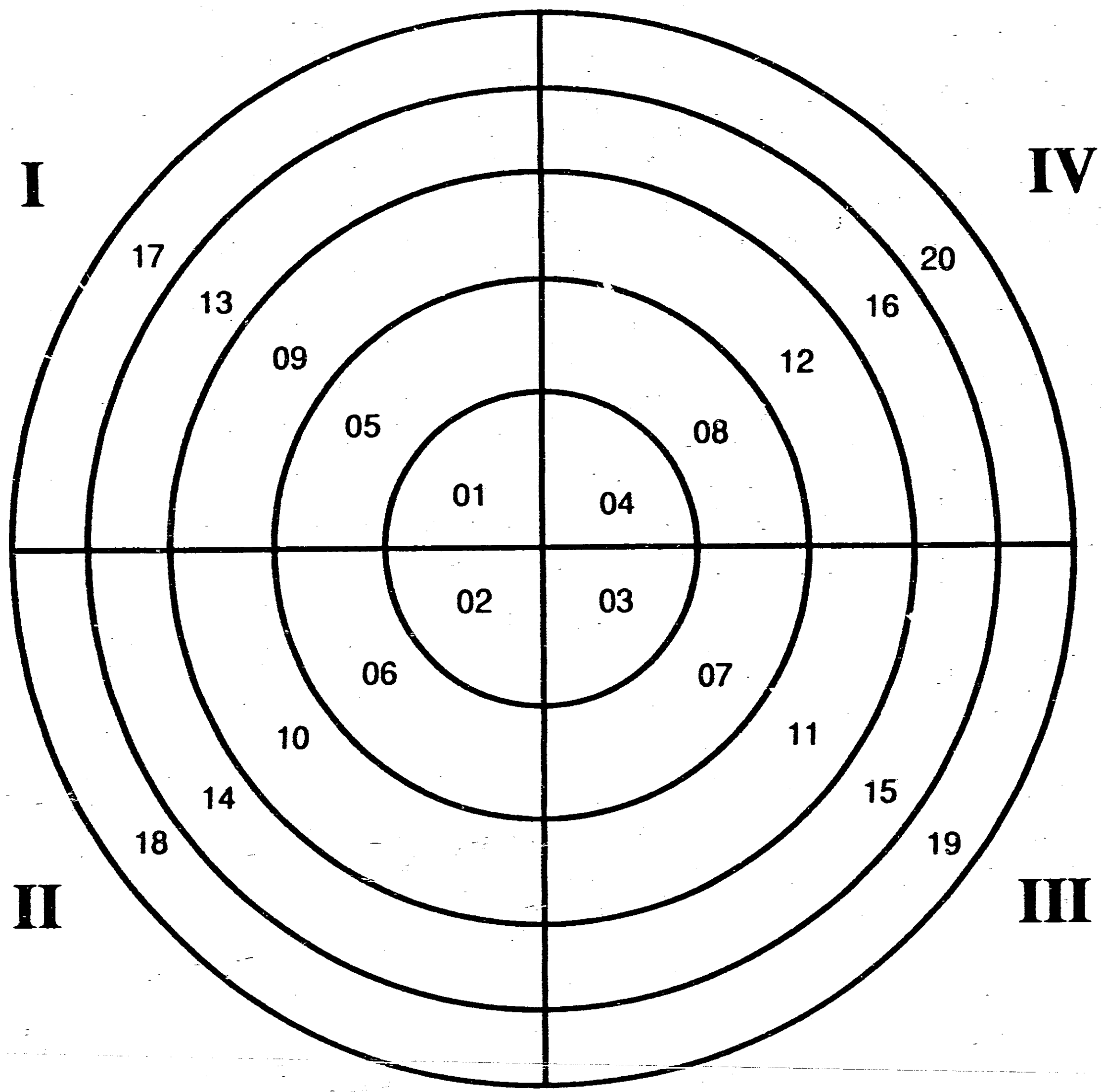


Figure 3-2. VESSEL Model Radial and Circumferential Cell Division.

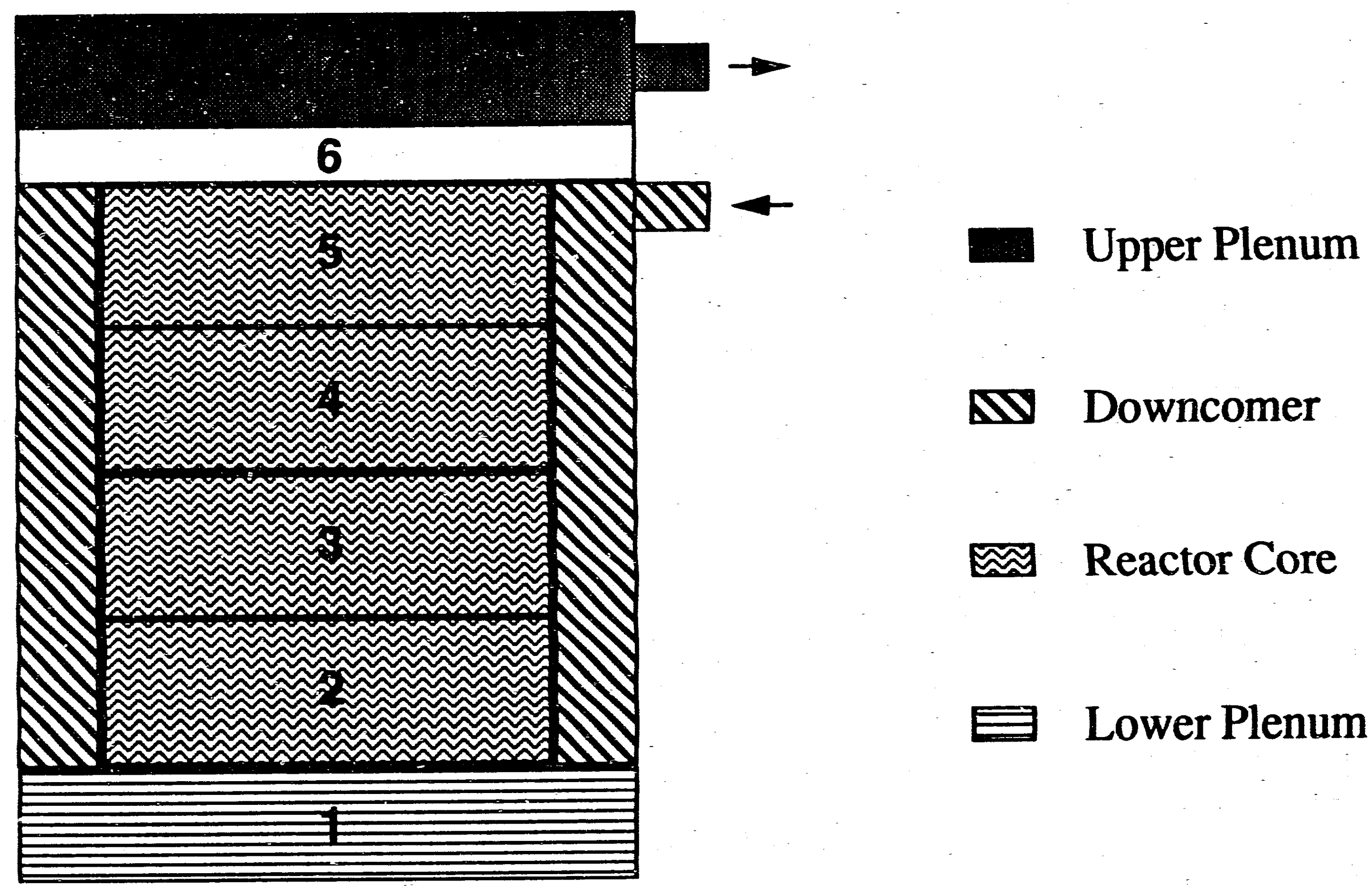


Figure 3-3. NPR VESSEL Model Regions.

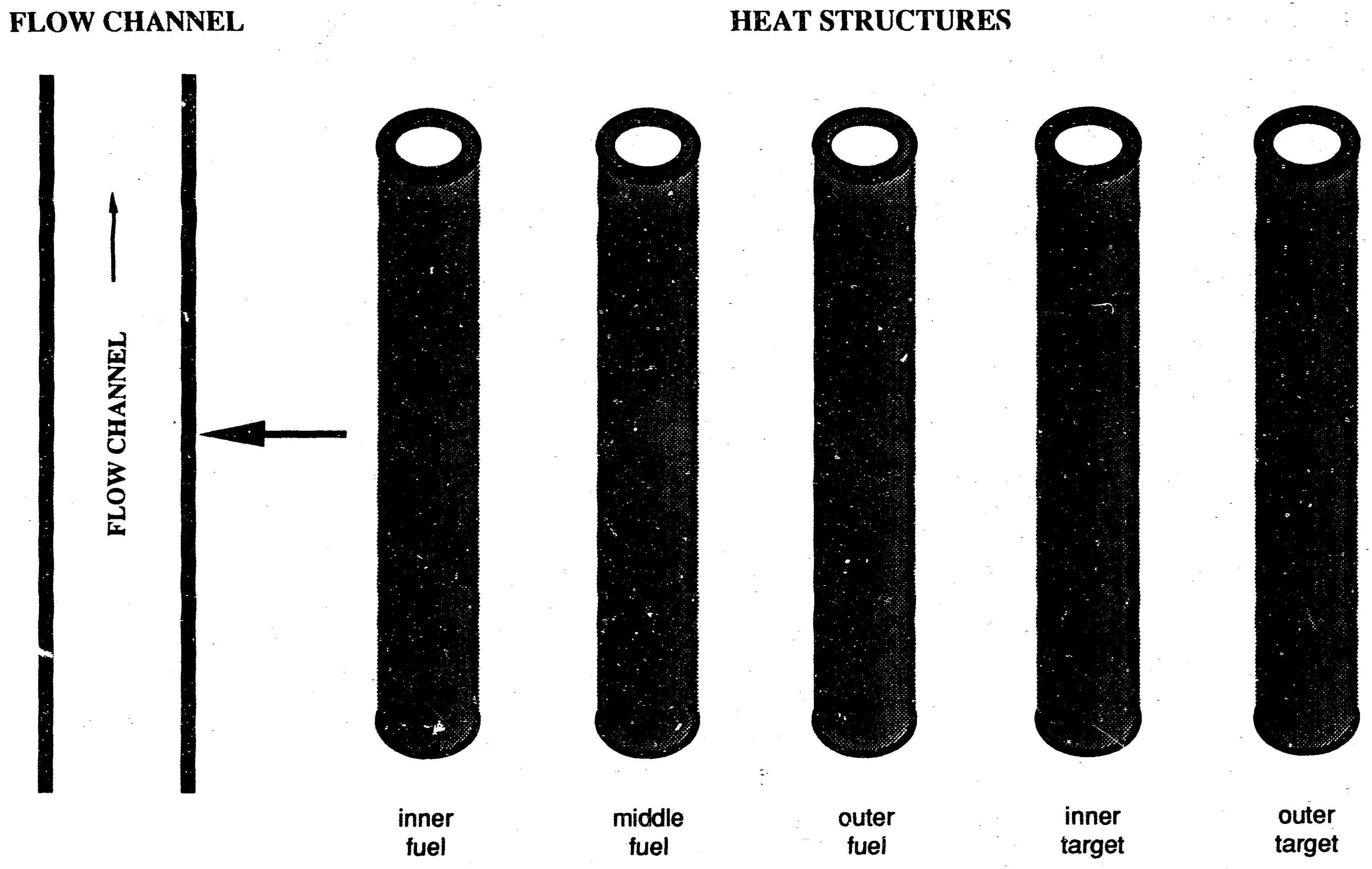


Figure 3-4. TRAC Fuel Assembly Components.

As described in the previous section, the active core region of the reactor vessel model is contained in axial segments 2 to 5 of the inner four radial segments, which represents 16 regions (see 'reactor core' in Figure 3-3). Each of these 16 vessel regions contains one representative assembly (i.e., one PIPE component and five ROD components), numbered by appending a '4' to the area number (01 to 16), resulting in the 16 vessel assemblies, 401 to 416, shown in Figure 3-5. The five ROD components in each assembly represent the three fuel tubes and two target tubes. The 438 actual fuel assemblies that reside within this region were divided equally among the 16 vessel regions that comprise the active core, since each of these regions has an equal cross-sectional area. This division resulted in 27.38 fuel assemblies for each of the 16 vessel regions. This modeling scheme results in 5 ROD components in each of the 16 vessel regions, or 80 ROD components in the vessel model. These ROD components are numbered 501 to 580 in ascending order with each vessel region (501 to 505 for region 01, 506 to 510 for region 02, etc.), as shown in Figure 3-6. The TRAC code contains effective multiplying factors to allow one fuel assembly model to represent 27.38 actual fuel assemblies.

The fuel and target tubes are nodalized as shown in Figure 3-7. Fifteen equally-spaced axial cells and three radial nodes in the tube wall were utilized. The flow channel in each assembly is modeled with 20 axial cells. Fifteen of these cells have the same length as that of the fuel and target tubes, representing the active core region. The remaining five axial cells represent sections of the flow channel above and below the fuel and target material. The cells of the fuel and target tubes (RODs) are coupled to the appropriate nodes of the flow channels (PIPE components) to model the core heat transfer. The axial power shape utilized in the model of the core region is taken from DPST-89-292[3] and is shown in Figure 3-8. Given in this figure are the fuel tube power fractions used in the TRAC input files. Table 3.1 gives the properties of aluminum, lithium-aluminum, uranium-aluminum, and stainless steel used in the TRAC heat transfer model.[4]

The 16 fuel assembly models, consisting of PIPE and ROD components, are connected to the VESSEL component in the core region previously described (see Section 3.2.1). The outlet of the uppermost cell of each flow channel is connected to the lower side of the appropriate volume of the upper plenum, represented by the highest axial segment of the VESSEL model. The inlet of the lowermost cell of each flow channel is connected to the upper side of the appropriate volume of the lower plenum, represented by the bottom axial segment of the VESSEL model. The space between the 16 flow channels in the core region model represents the moderator coolant fluid. This fluid and the structural components of the reactor core are assumed to receive six percent of the total reactor power, or 150 MW, in the form of gamma ray heating.

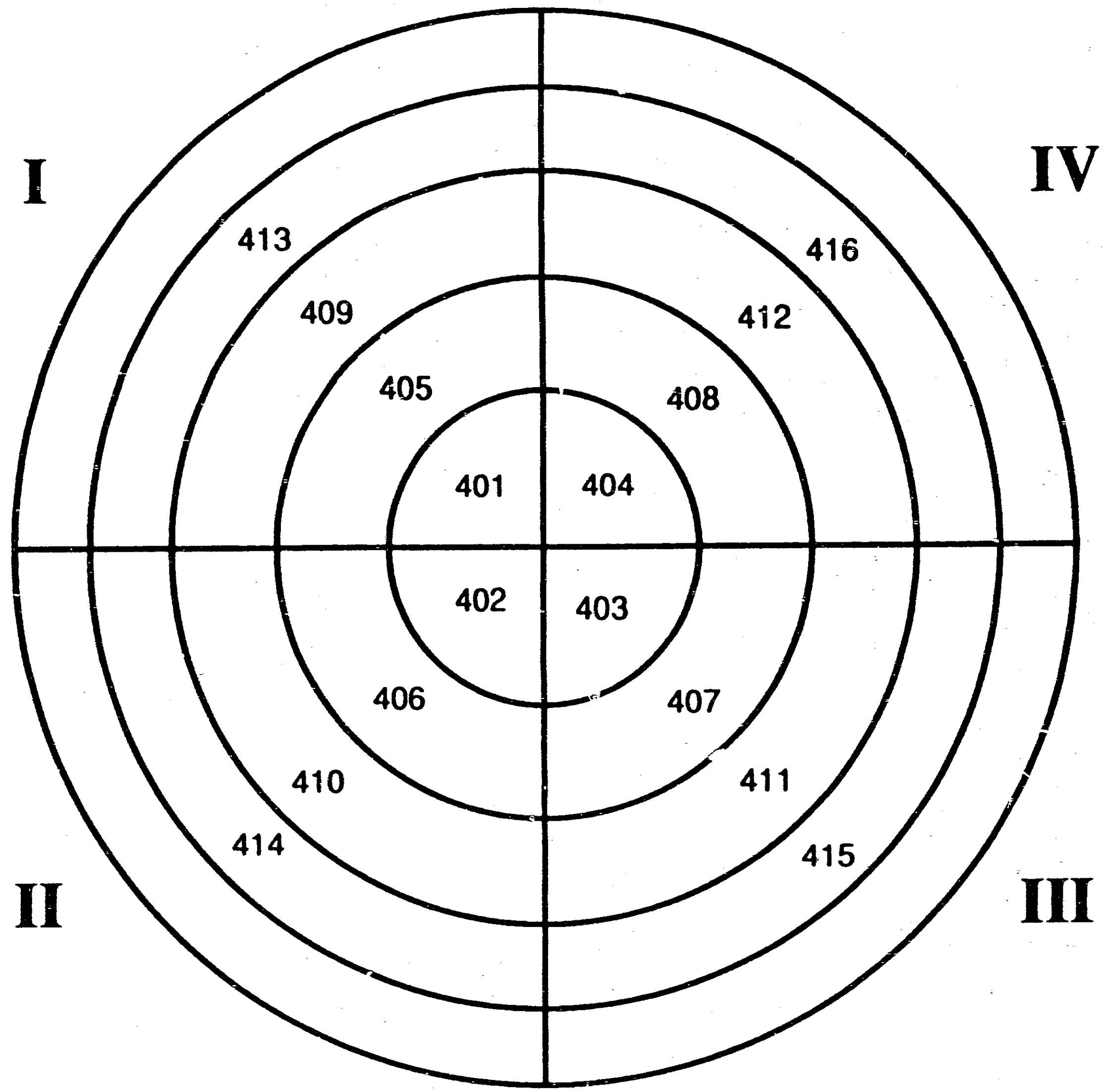


Figure 3-5. Fuel Assembly Numbers Within VESSEL Model.

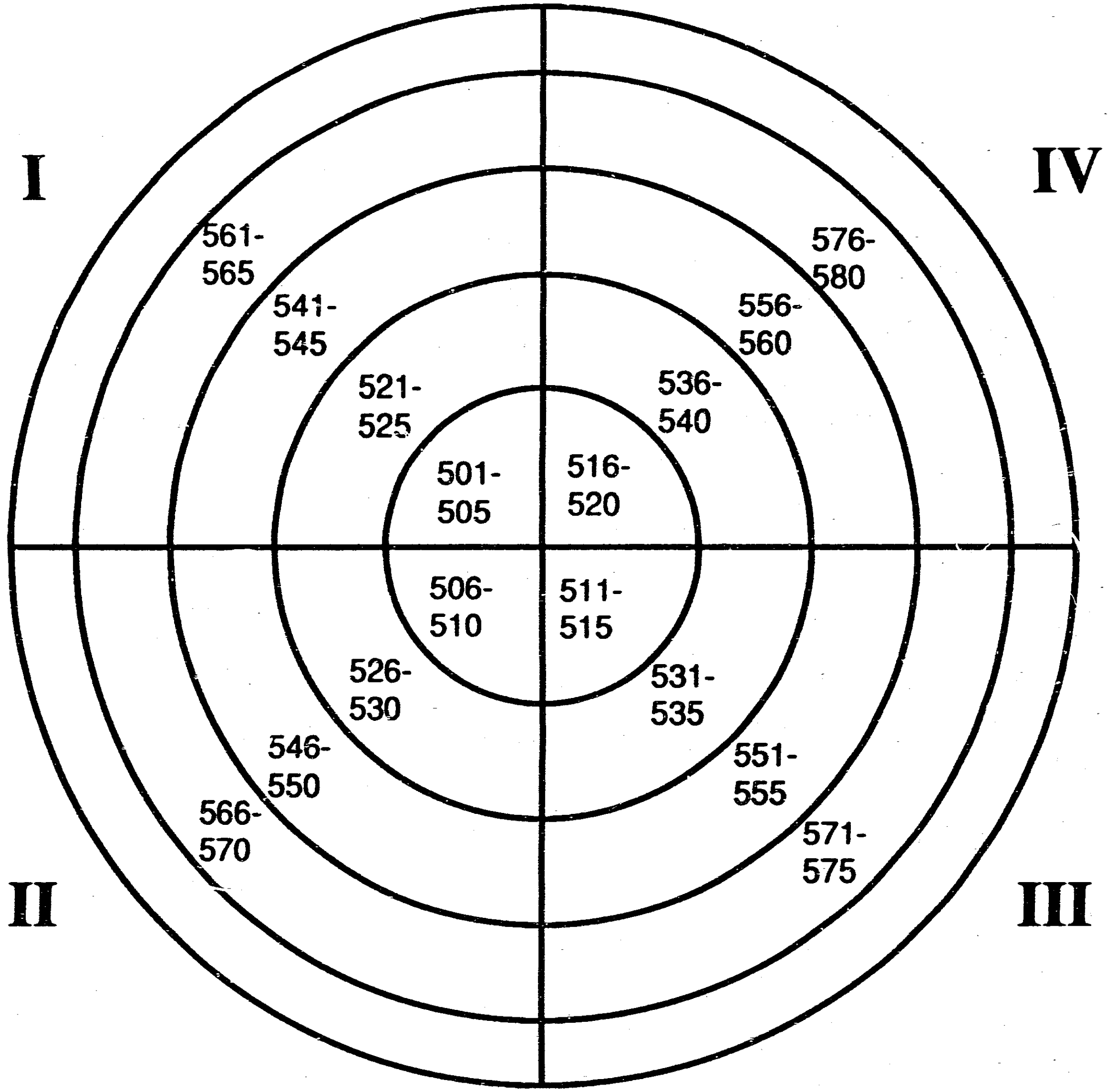


Figure 3-6. Heat Structure Numbers Within VESSEL Model.

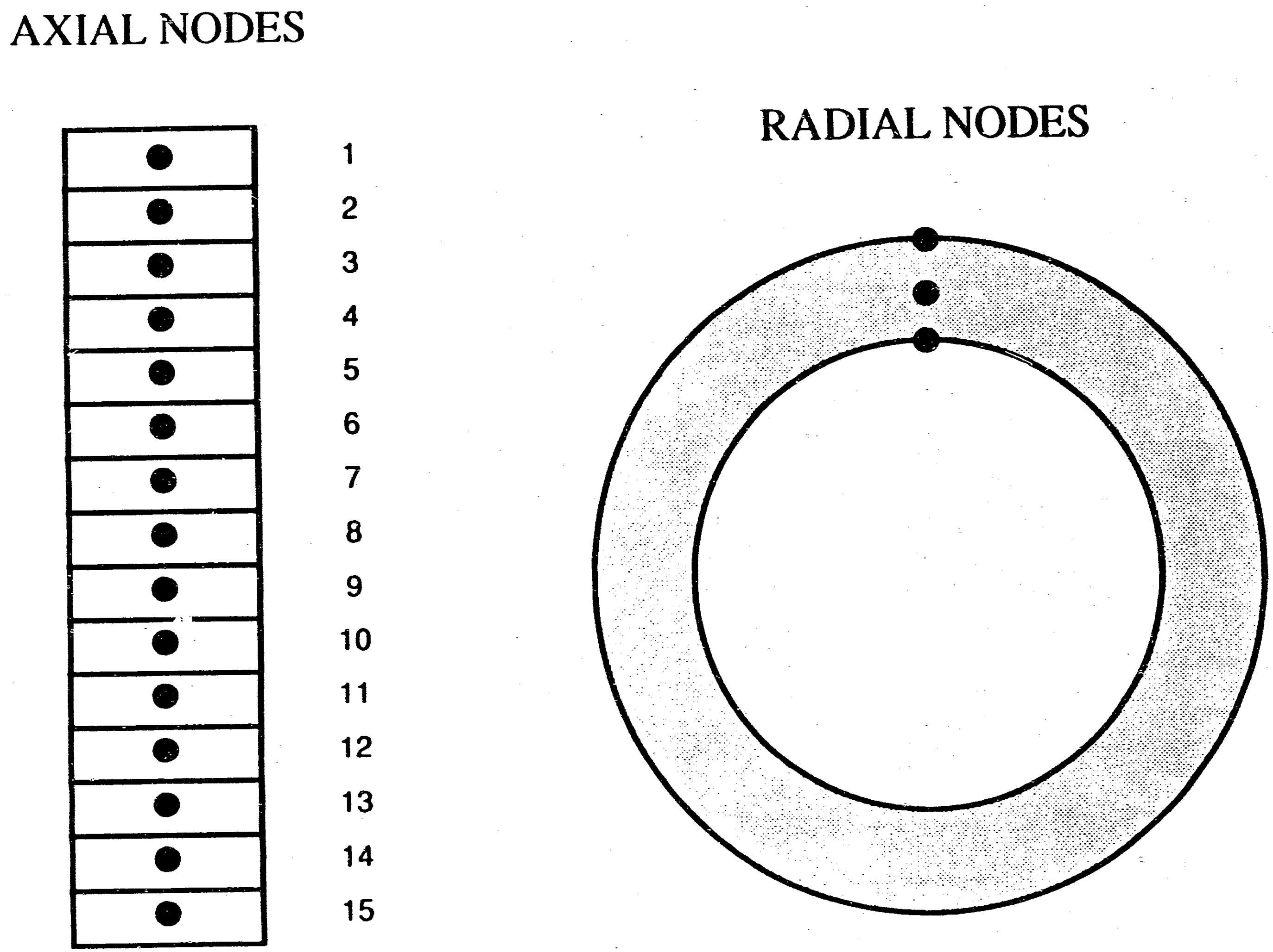
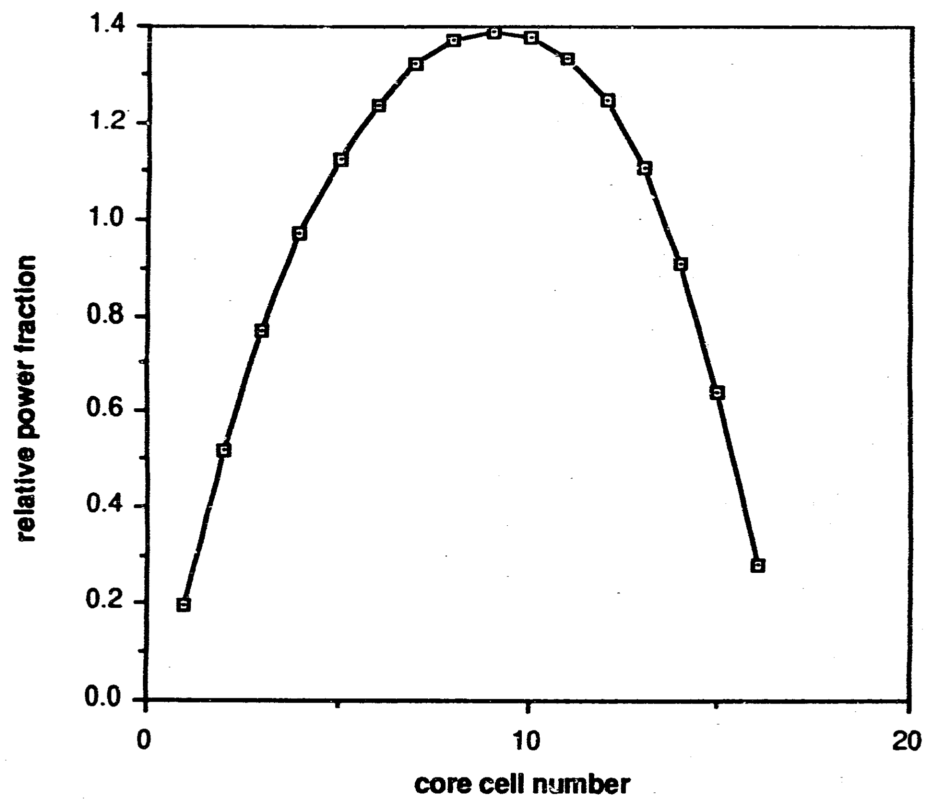


Figure 3-7. Fuel Tube and Target Tube Nodalization.

Axial Power Shape



FUEL TUBE POWER FRACTIONS:
Inner - 0.154
Middle - 0.3886
Outer - 0.4574

REF: DPST-89-292

Figure 3-8. Fuel Tube Axial Power Shape.

Table 3.1

Metal Properties Used in TRAC (SI Units).

ALUMINUM PROPERTIES

temp (c)	temp (k)	density	specific ht	conductivity
0.000000e+00	2.731500e+02	2.709489e+03	8.799064e+02	2.376412e+02
1.000000e+02	3.731500e+02	2.688659e+03	9.127124e+02	2.380540e+02
2.000000e+02	4.731500e+02	2.667830e+03	9.523942e+02	2.368006e+02
3.000000e+02	5.731500e+02	2.647000e+03	9.989518e+02	2.338809e+02
4.000000e+02	6.731500e+02	2.626170e+03	1.052385e+03	2.292950e+02
5.000000e+02	7.731500e+02	2.605340e+03	1.112694e+03	2.230429e+02
6.000000e+02	8.731500e+02	2.584510e+03	1.179879e+03	2.151245e+02
7.000000e+02	9.731500e+02	2.563680e+03	1.253940e+03	2.055399e+02
8.000000e+02	1.073150e+03	2.542850e+03	1.334876e+03	1.942891e+02

LI-ALUMINUM PROPERTIES

temp (c)	temp (k)	density	specific ht	conductivity
0.000000e+00	2.731500e+02	2.705157e+03	8.962193e+02	2.409149e+02
1.000000e+02	3.731500e+02	2.686568e+03	9.306894e+02	2.409149e+02
2.000000e+02	4.731500e+02	2.668150e+03	9.704774e+02	2.409149e+02
3.000000e+02	5.731500e+02	2.649899e+03	1.015583e+03	2.409149e+02
4.000000e+02	6.731500e+02	2.631814e+03	1.066007e+03	2.409149e+02
5.000000e+02	7.731500e+02	2.613894e+03	1.121749e+03	2.409149e+02
6.000000e+02	8.731500e+02	2.596136e+03	1.182809e+03	2.409149e+02
7.000000e+02	9.731500e+02	2.578538e+03	1.249187e+03	2.409149e+02
8.000000e+02	1.073150e+03	2.561099e+03	1.320882e+03	2.409149e+02

U-ALUMINUM PROPERTIES

temp (c)	temp (k)	density	specific ht	conductivity
0.000000e+00	2.731500e+02	2.755831e+03	7.267006e+02	2.215775e+02
1.000000e+02	3.731500e+02	2.739353e+03	7.540605e+02	2.235314e+02
2.000000e+02	4.731500e+02	2.723007e+03	7.874366e+02	2.227456e+02
3.000000e+02	5.731500e+02	2.706791e+03	8.268287e+02	2.192202e+02
4.000000e+02	6.731500e+02	2.690703e+03	8.722371e+02	2.129551e+02
5.000000e+02	7.731500e+02	2.674742e+03	9.236615e+02	2.039504e+02
6.000000e+02	8.731500e+02	2.658908e+03	9.811021e+02	1.922060e+02
7.000000e+02	9.731500e+02	2.643198e+03	1.044559e+03	1.777220e+02
8.000000e+02	1.073150e+03	2.627612e+03	1.114032e+03	1.604983e+02

STAINLESS STEEL (304L) PROPERTIES

temp (c)	temp (k)	density	specific ht	conductivity
0.000000e+00	2.731500e+02	8.027523e+03	3.829839e+02	1.477054e+01
1.000000e+02	3.731500e+02	7.982849e+03	3.977586e+02	1.628763e+01
2.000000e+02	4.731500e+02	7.938176e+03	4.125333e+02	1.773059e+01
3.000000e+02	5.731500e+02	7.893502e+03	4.273081e+02	1.909943e+01
4.000000e+02	6.731500e+02	7.848828e+03	4.420828e+02	2.039414e+01
5.000000e+02	7.731500e+02	7.804154e+03	4.568576e+02	2.161472e+01
6.000000e+02	8.731500e+02	7.759481e+03	4.716323e+02	2.276118e+01
7.000000e+02	9.731500e+02	7.714807e+03	4.864071e+02	2.383352e+01
8.000000e+02	1.073150e+03	7.670133e+03	5.011818e+02	2.483172e+01

REF: DPST-87-692

3.2.3 Primary Cooling System

The primary cooling system model consists of, for each loop, TRAC components for the pump (PUMP), heat exchangers (STGEN), pressurizer/accumulator (BREAK or ACCUM), and loop piping (PIPE). These components are shown for one coolant loop in Figure 3-9. The coordinates for the component positions with respect to the middle of the hot leg outlet from the reactor vessel, which give both length and elevation, are given in Figure 3-10. Components are numbered in each loop using two digit identifiers, the first number of which is the loop number in which the component resides (1X for loop 1, 2X for loop 2, etc.). The following subsections describe the modeling of each of these components.

Heat Exchangers

As previously mentioned, each of the four coolant loops has three heat exchangers. The three heat exchangers per loop were lumped into one per loop for a total of four heat exchangers in the primary system, modeled as one-dimensional STGEN components. The STGEN component is formed by representing the primary and secondary flow channels as PIPE components and representing the heat exchanger structure as a ROD (heat structure) component with no power production. The flow channels are modeled with twelve cells for both the primary and secondary sides. The secondary side of the heat exchanger was modeled as a PIPE component with boundary conditions. The secondary inlet to the heat exchanger is modeled with a FILL component, which imposes on the PIPE a constant inlet velocity boundary. The secondary outlet is modeled with a BREAK component, which imposes on the PIPE a constant outlet pressure boundary. The ROD component contains the same mesh cell spacing and the same number of cells as the PIPE components to facilitate thermal-hydraulic coupling of the mesh cells. The active heat transfer region of the STGEN component is represented by the middle ten cells. The heat transfer in these components is one-dimensional, between the ROD and PIPE mesh cells (i.e., no axial conduction). The STGEN components are numbered 12, 22, 32 and 42 for loops 1 to 4, respectively. Figure 3-11 shows the heat exchanger component.

Pumps

Each pump in the primary cooling system is modeled using the TRAC one-dimensional PUMP component. The pump model must consist of at least two cells, with the pressure rise applied between the first and second cells. The pump model in this analysis contains four cells. Homologous pump head curves for current Savannah River production reactors are used for the NPR pumps in this model. The version of TRAC used in this analysis contains no pump cavitation model. The PUMP components are numbered 14, 24, 34 and 44 for loops 1 to 4, respectively. Figure 3-12 shows the pump model.

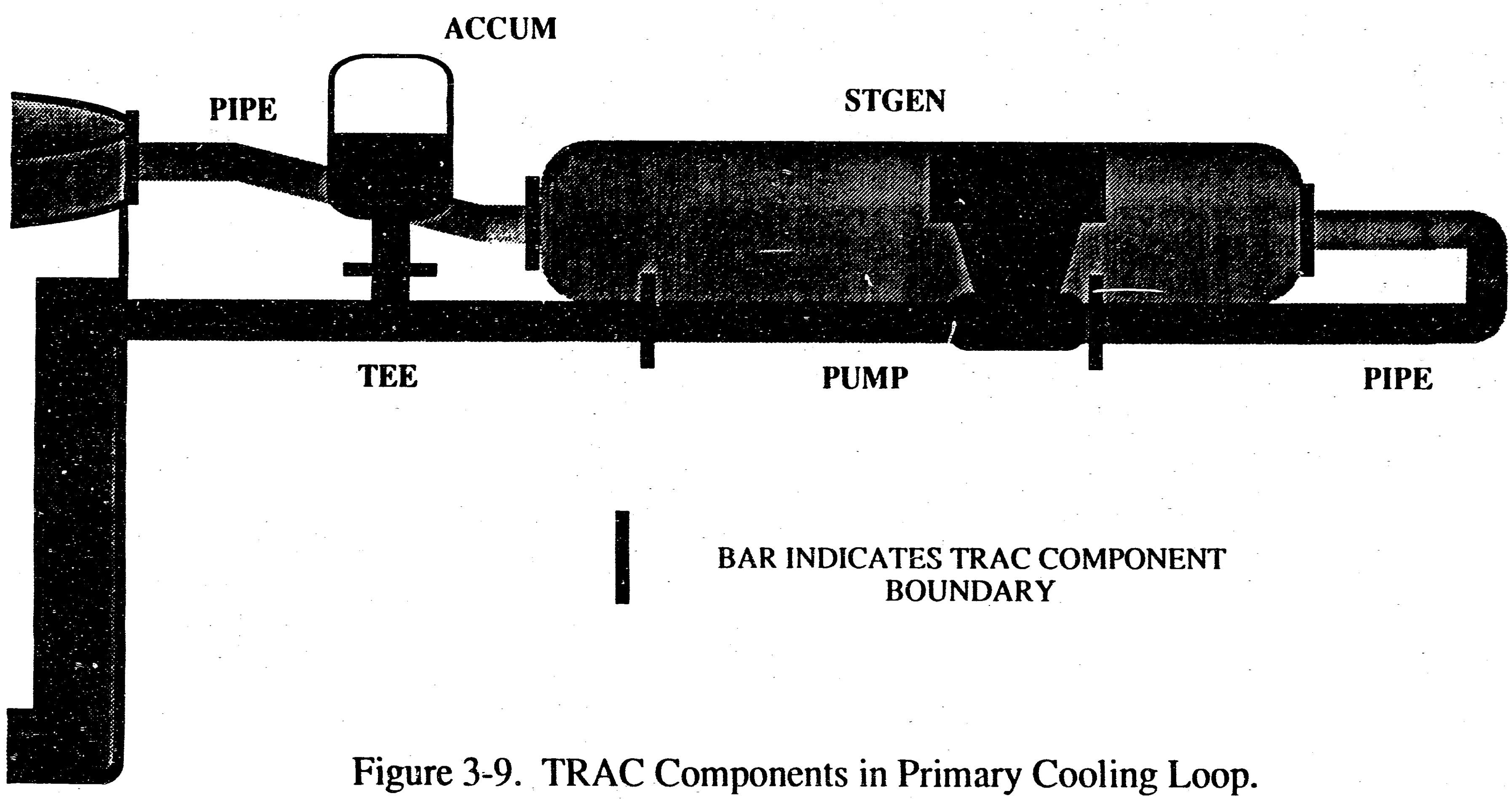


Figure 3-9. TRAC Components in Primary Cooling Loop.

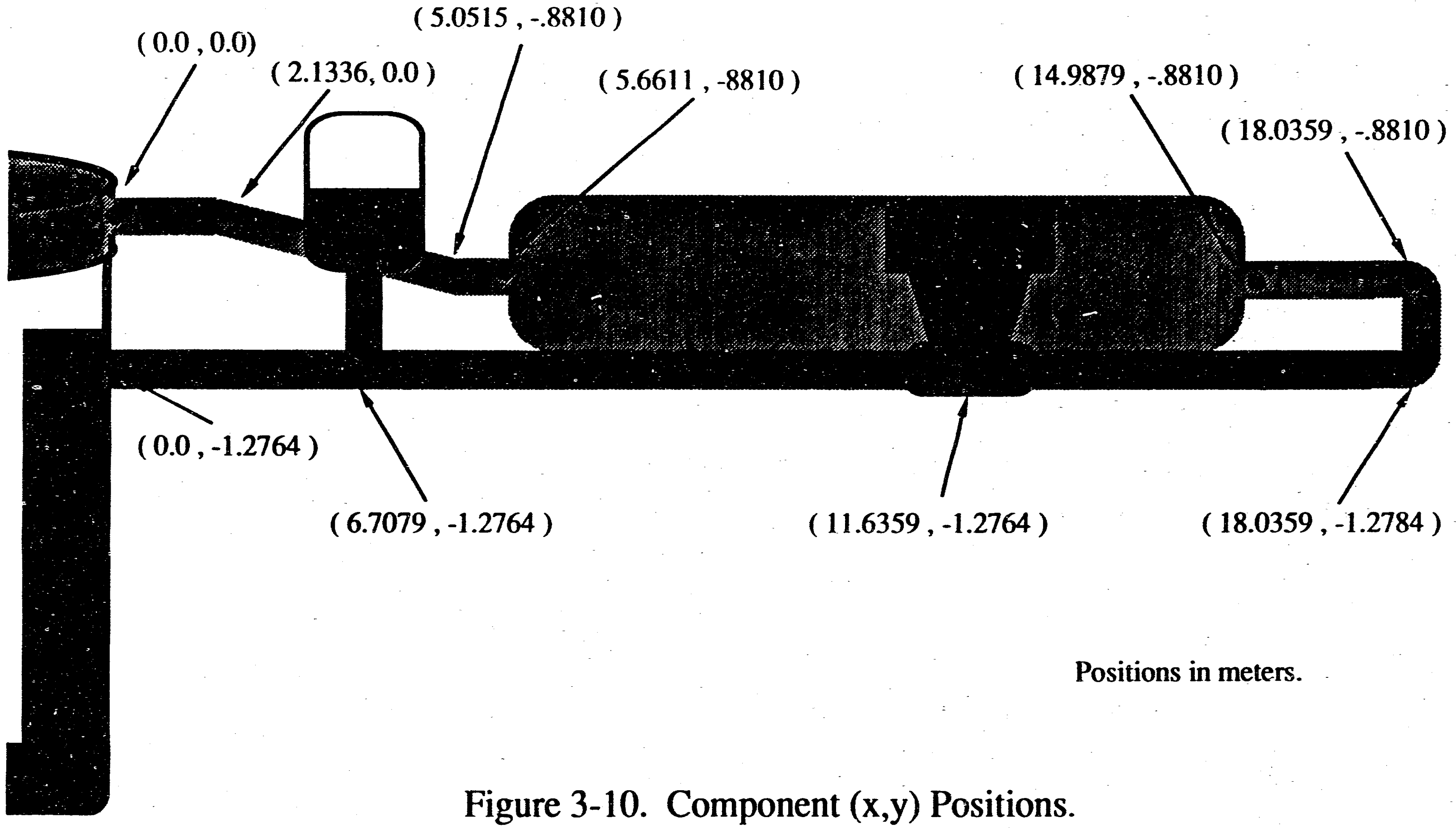


Figure 3-10. Component (x,y) Positions.

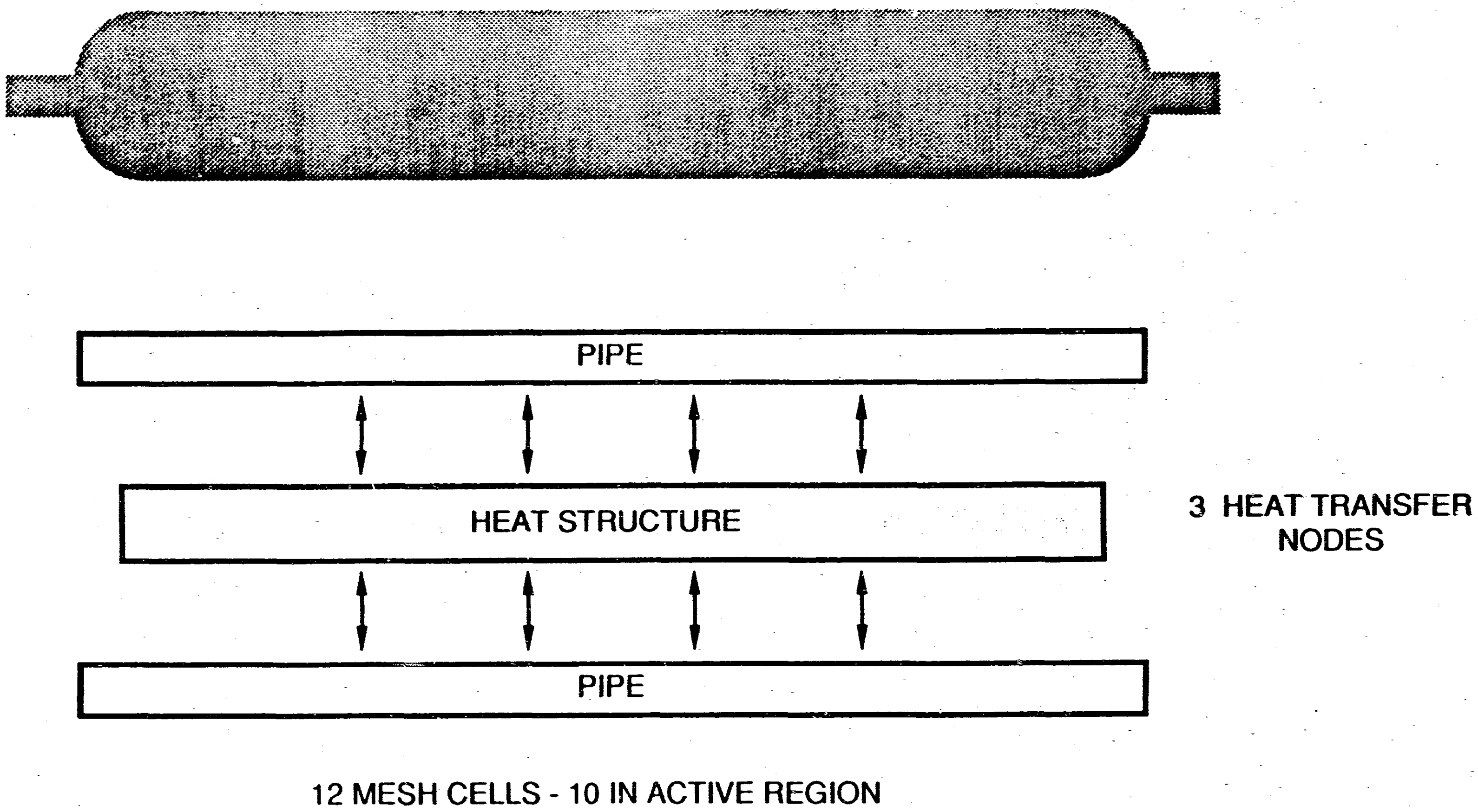
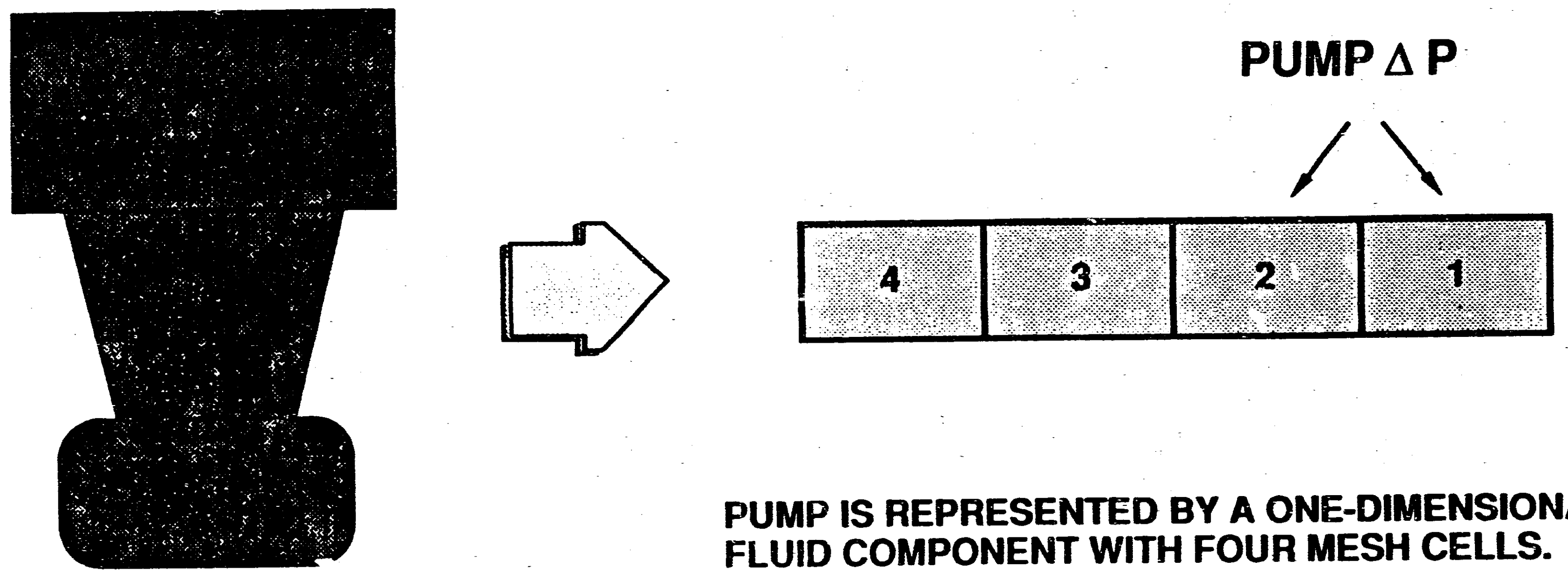


Figure 3-11. Heat Exchanger Noding.



PUMP IS REPRESENTED BY A ONE-DIMENSIONAL FLUID COMPONENT WITH FOUR MESH CELLS.

PRESSURE RISE IS APPLIED BETWEEN CELLS 1 AND 2.

Figure 3-12. Pump Noding.

Pressurizer/Accumulators

Each primary coolant loop will have a component to perform both the function of a pressurizer and an accumulator. This component will regulate the primary system pressure to a constant value and provide emergency coolant injection. It will be located on each loop between the pump and the cold leg inlet to the reactor vessel. The pressurizer is modeled using a TEE and a BREAK component. The TEE is an eight-cell, one-dimensional pipe with a two-cell vertical branch, or side pipe. The side pipe has a BREAK component connected to it to model the pressurizer action. The BREAK component is a constant-pressure device that is set at 175 psia to compensate for pressure fluctuations. The numbering for the TEE components is 15, 25, 35 and 45 for loops 1 to 4, respectively, and for the BREAK components is 150, 250, 350 and 450 for loops 1 to 4, respectively.

Loop Piping

All piping for the primary cooling system is 0.508 meters (20 inches) in inner diameter and follows the same numbering convention as for other loop components. Each loop has a seven-cell hot leg outlet pipe from the reactor and a twelve-cell cold leg pipe following the heat exchanger. For loops one to four, the hot legs are numbered 11, 21, 31 and 41, and the cold legs 13, 23, 33 and 43, respectively.

3.3 Model Testing and Validation

At each stage in the development of the complete model, code testing was performed to ensure that each component model was free from input errors prior to inclusion into the total model. In this manner of testing the model piece by piece, the final model should be free from major errors. This testing process was simplified by using BREAK components to realistically model boundaries of components.

The completed reactor vessel model was debugged by running the model with BREAK components applied to the four loop outlets and inlets. The BREAK components apply the desired fluid pressure at the above locations to simulate the effect of the fluid loops. BREAK components were applied to the ends of the fuel assembly models to simulate the effects of the upper and lower plenum of the reactor vessel. BREAKs were also used with the STGEN (heat exchanger) and PUMP components in the same fashion. Following individual loop component testing, PIPE components were used to simulate the vessel and form the four coolant loops. Each of the four loops was then individually tested.

Following the debugging of all individual component and partial system models, the complete primary system model was assembled and tested. Initially, all four pumps and the reactor power were turned off for a static test. During this test, the system should settle at a no-flow state. Any appreciable flow resulting in this case indicates a modeling error which introduces a momentum source, and hence, the undesired flow. Following successful completion of this static test case, the desired mass flow rate and pressure drop characteristics during steady state operation must be achieved.

Each NPR fuel assembly is currently designed to have 0.021 m^3/s (325 gpm) of reactor coolant flow at a pressure drop of 248 kPa (36 psi). This pressure drop does not include that specified for the upper and lower plenums (10.3 kPa (1.5 psi) each) or the orifice plate at the lower end of the assemblies (27.6 kPa (4 psi)). Friction factors were applied to the appropriate junctions in the PIPE component to represent the frictional losses due to the upper plenum, lower plenum, and orifice plate. The wall roughness within the PIPE component was adjusted to 5 micrometers (1.37×10^{-6} inches) to obtain the correct flow in the assembly. Much iteration was necessary in order to arrive at the correct flow at the desired pressure drop. Following this task, the complete model is ready to be utilized to simulate accidents and transients.

4. RESULTS

This section presents the results of both the steady-state analysis and the LOPA analysis. The steady-state results, in Section 4.1, include the final values of all salient system parameters during steady-state operation and plots of selected parameters. The LOPA results, Section 4.2, include a description of the parameters that were controlled to model LOPA conditions and plots of selected response parameters over the duration of the accident.

4.1 Steady-State

The desired characteristics of steady-state operation are found in an iterative process using the TRAC model. In this study, the desired steady-state values for the flow rate and pressure drop were 0.021 m^3/s (325 gpm) per assembly and 248 kPa (36 psi) per assembly, respectively.[5] By adjusting TRAC parameters, such as friction factors and pump head (see Section 3.3), the desired assembly flow at the proper pressure drop was achieved. Values for salient steady-state parameters are given in Table 4.1.

4.2 Loss-of-Pumping Accident

Following the creation of the desired steady-state input deck, which contains a "snapshot" of the system during normal operation, accidents and transients may be simulated using this input deck as a starting point. Four cases of varying levels of pumping loss were simulated using the NPR TRAC model, which include:

1. AC pumps coast down, DC motors on, 100% secondary flow.
2. AC pumps coast down, DC motors off, 100% secondary flow.
3. AC pumps coast down, DC motors on, 10% secondary flow.
4. AC pumps coast down, DC motors off, 10% secondary flow.

The subsections below describe the four loss-of-pumping accident cases and give the significant results for each.

4.2.1 Controlled Parameters

Three parameters were controlled within the transient TRAC input deck to simulate the occurrence of loss-of-pumping power and subsequent reactor scram. Control blocks within the input deck for reactor power, secondary flow, and primary pump speed were adjusted to achieve the desired effect. A power decay curve, shown in Figure 4-1, was input to scram the reactor for all LOPA cases. Two primary pump speed

Table 4.1

STEADY-STATE SYSTEM PARAMETERS

TOTAL REACTOR POWER	2500 MW (2350 MW to primary coolant)
AVERAGE ASSEMBLY FLOW RATE	0.021 m^3/s (325 gpm)
PRIMARY SYSTEM FLOW RATE	8.99 m^3/s (142,406 gpm)
SECONDARY SYSTEM FLOW RATE	11.5 m^3/s (182,403 gpm)
PRESSURIZER PRESSURE	1207 kPa (175 psi)
UPPER PLENUM PRESSURE	938 kPa (136 psi)
DELTA-P ACROSS PRIMARY PUMPS	889 kPa (129 psi)
DELTA-P ACROSS HEAT EXCHANGER	517 kPa (75 psi)
HEAT EXCHANGER PRIMARY INLET TEMP	107°C (224°F)
HEAT EXCHANGER PRIMARY OUTLET TEMP	44°C (111°F)
HEAT EXCHANGER SECONDARY INLET TEMP	22°C (72°F)
HEAT EXCHANGER SECONDARY OUTLET TEMP	71°C (160°F)

decay curves were used to simulate loss of pump power, one of which coasts the pumps to zero while the other coasts to DC power levels. Plots of pump speed fraction versus time for these cases are given in Figure 4-2. The secondary side heat exchanger flow rate is either at full flow or linearly ramped to 10% flow over 20 seconds. The 10% flow case corresponds to gravity feed (no power), and is shown graphically in Figure 4-3.

4.2.2 Case 1

This is the most favorable loss-of-pumping accident case, with the primary pumps powered by DC motors and with full secondary coolant flow. Throughout the accident, all primary fluid remains over 100 degrees subcooled and fuel temperatures sharply decline. The primary system pressure is maintained near 175 psia by the pressurizer/accumulators. The upper-lower plenum pressure difference steadily decreases from the operational value of 43 psid to approximately 13 psid during DC pump operation, after the AC pumps have coasted down. Plots of pressure, temperature, and mass flow rate in the vessel and coolant loops are given in Figures 4-4 to 4-7.

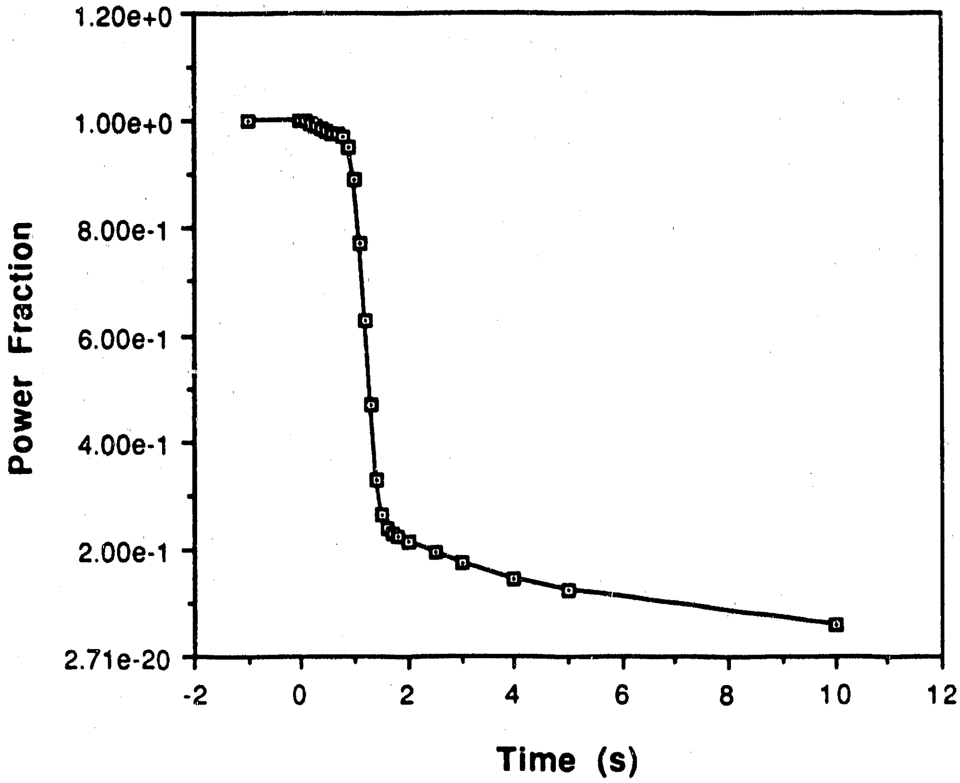


Figure 4-1. Reactor Power Decay Curve.

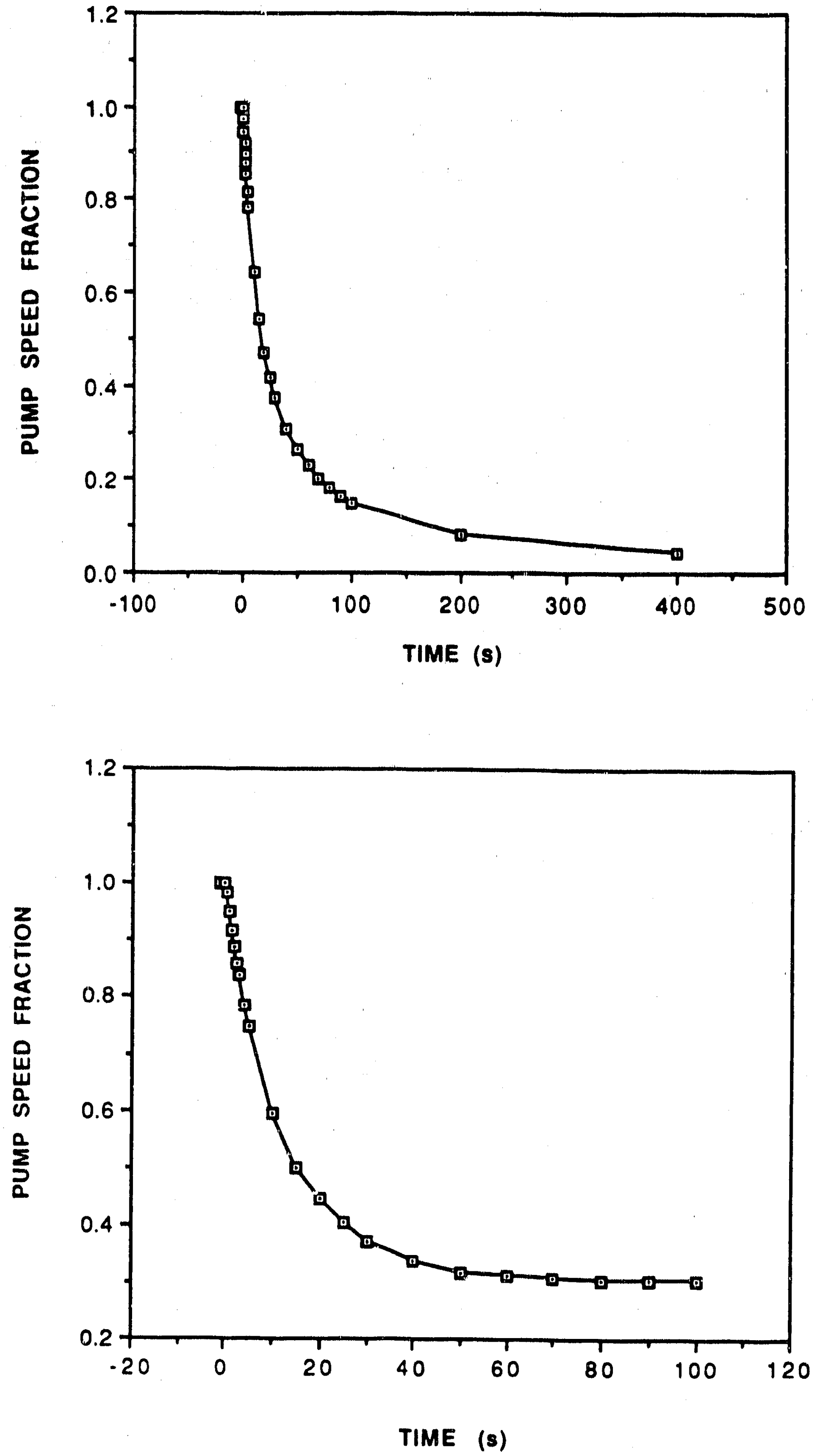


Figure 4-2. Pump Speed Decay Without (above) and With (below) DC Motors.

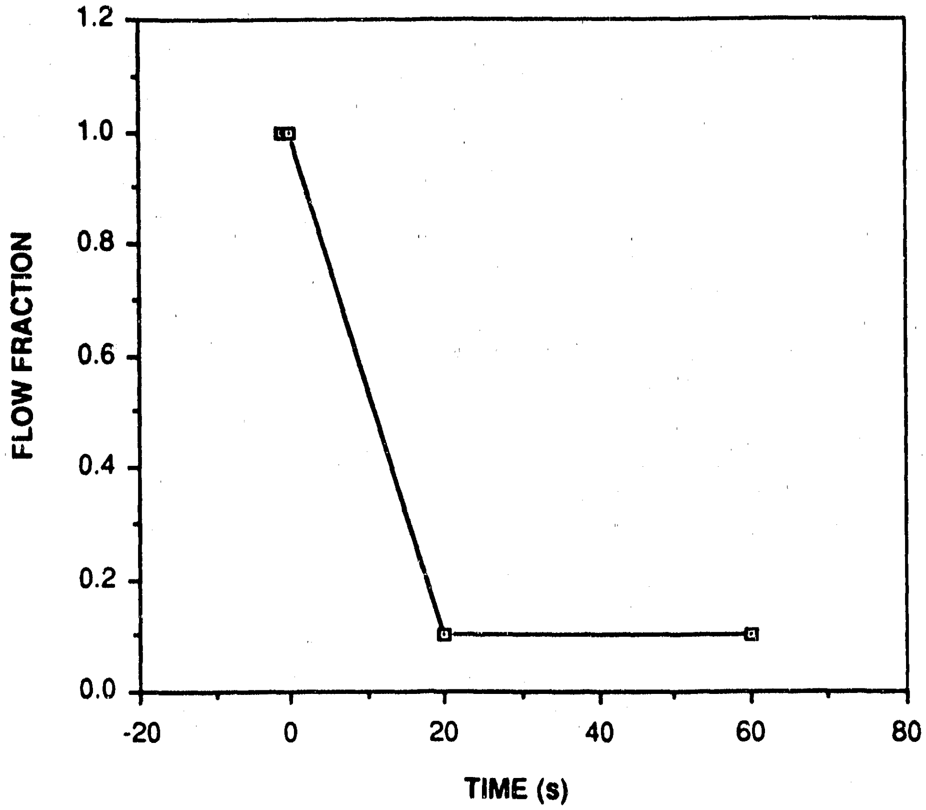


Figure 4-3. Secondary Flow Decay to Gravity Feed.

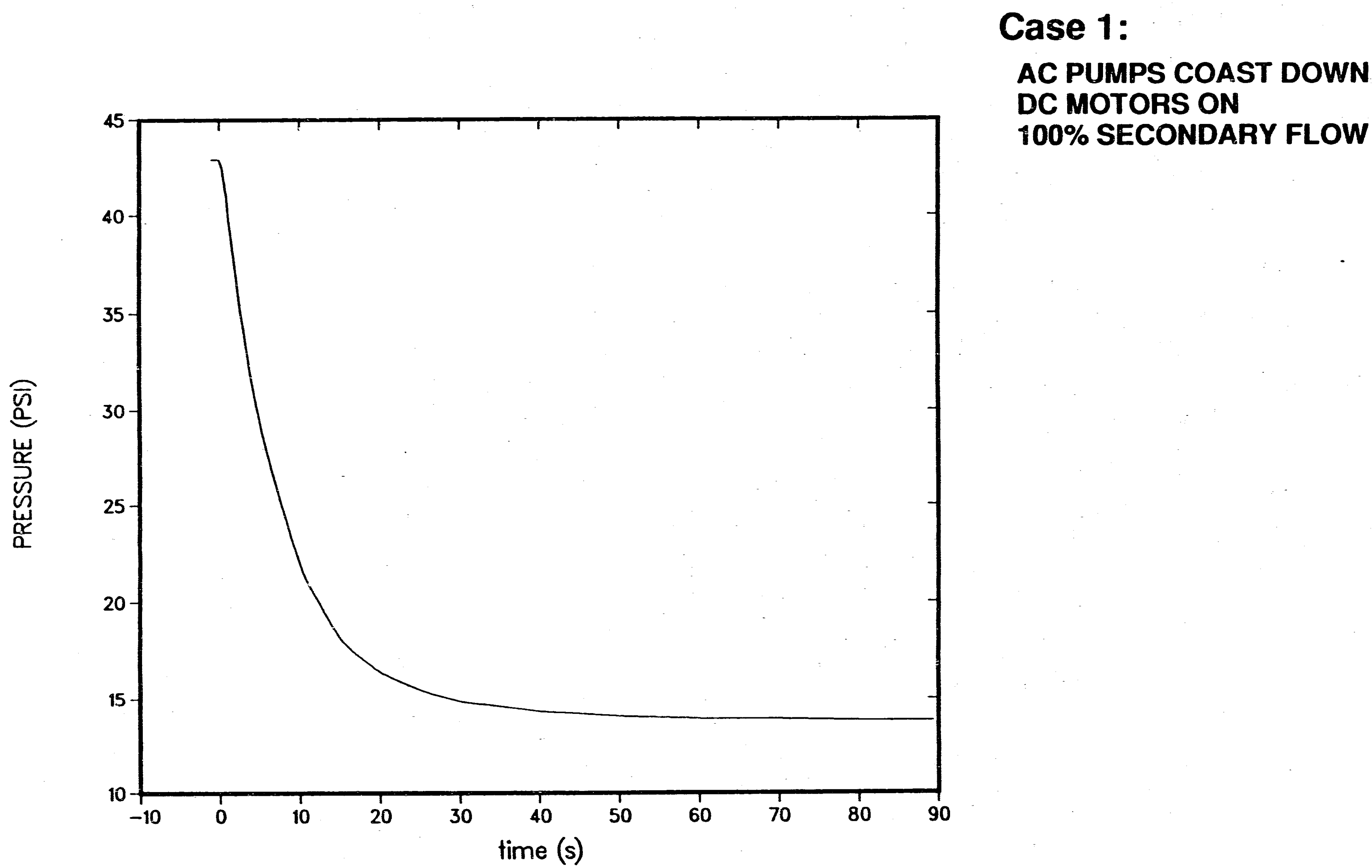


Figure 4-4. Case 1 - Upper-Lower Plenum Pressure Difference.

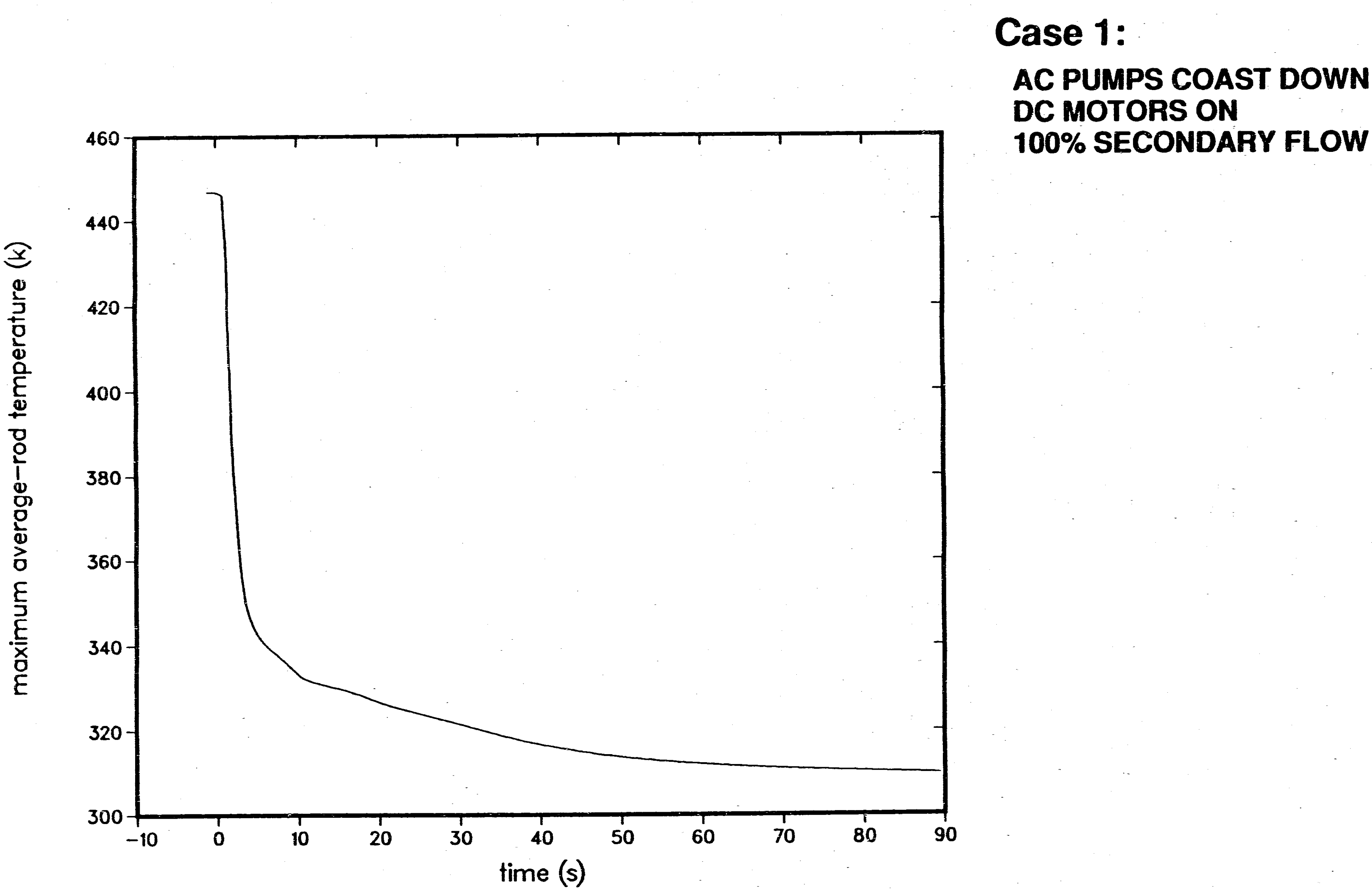
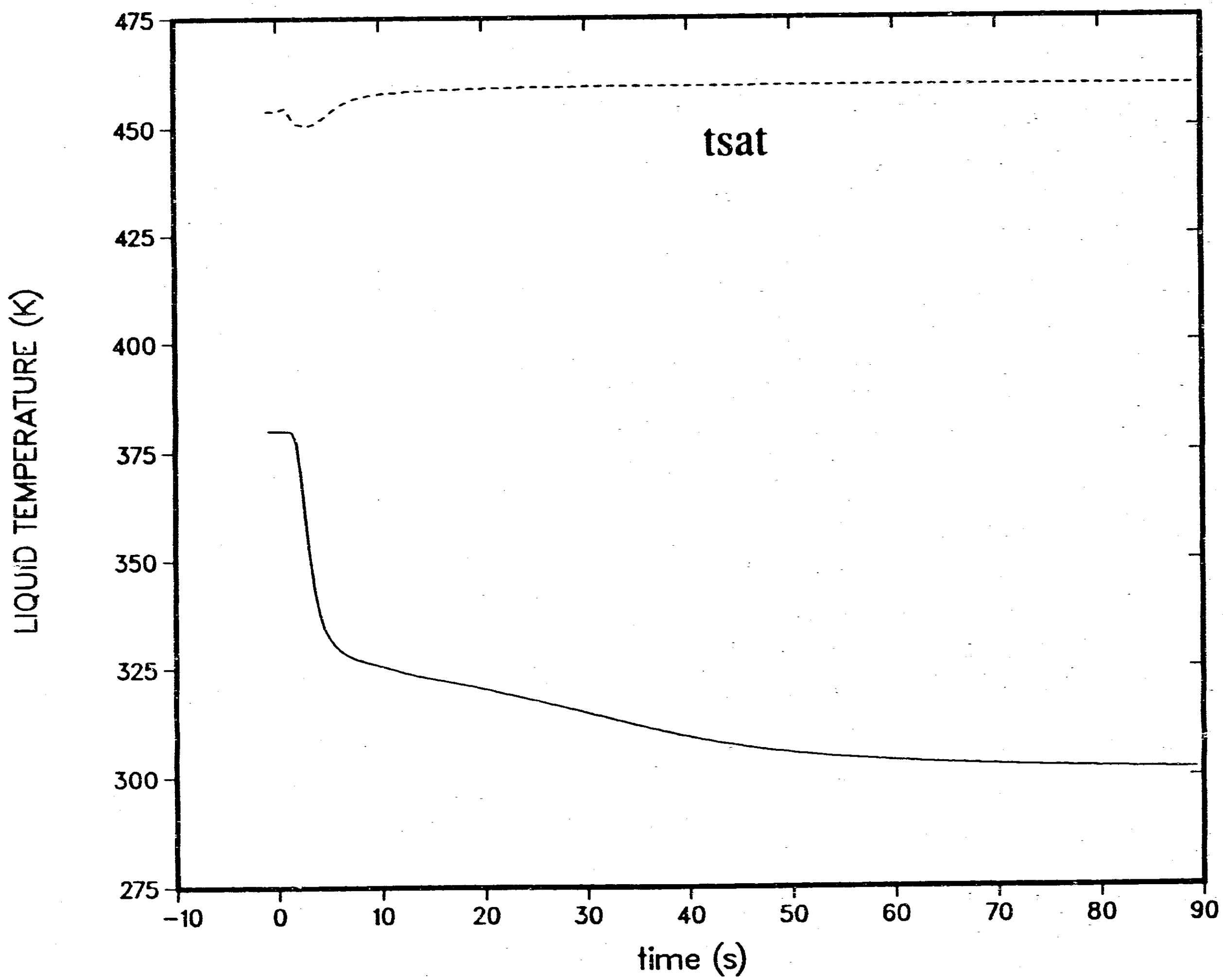


Figure 4-5. Case 1 - Maximum Middle Fuel Tube Temperature for Assembly 401.



Case 1:

AC PUMPS COAST DOWN
DC MOTORS ON
100% SECONDARY FLOW

Figure 4-6. Case 1 - Liquid Temperature at Assembly 401 Outlet.

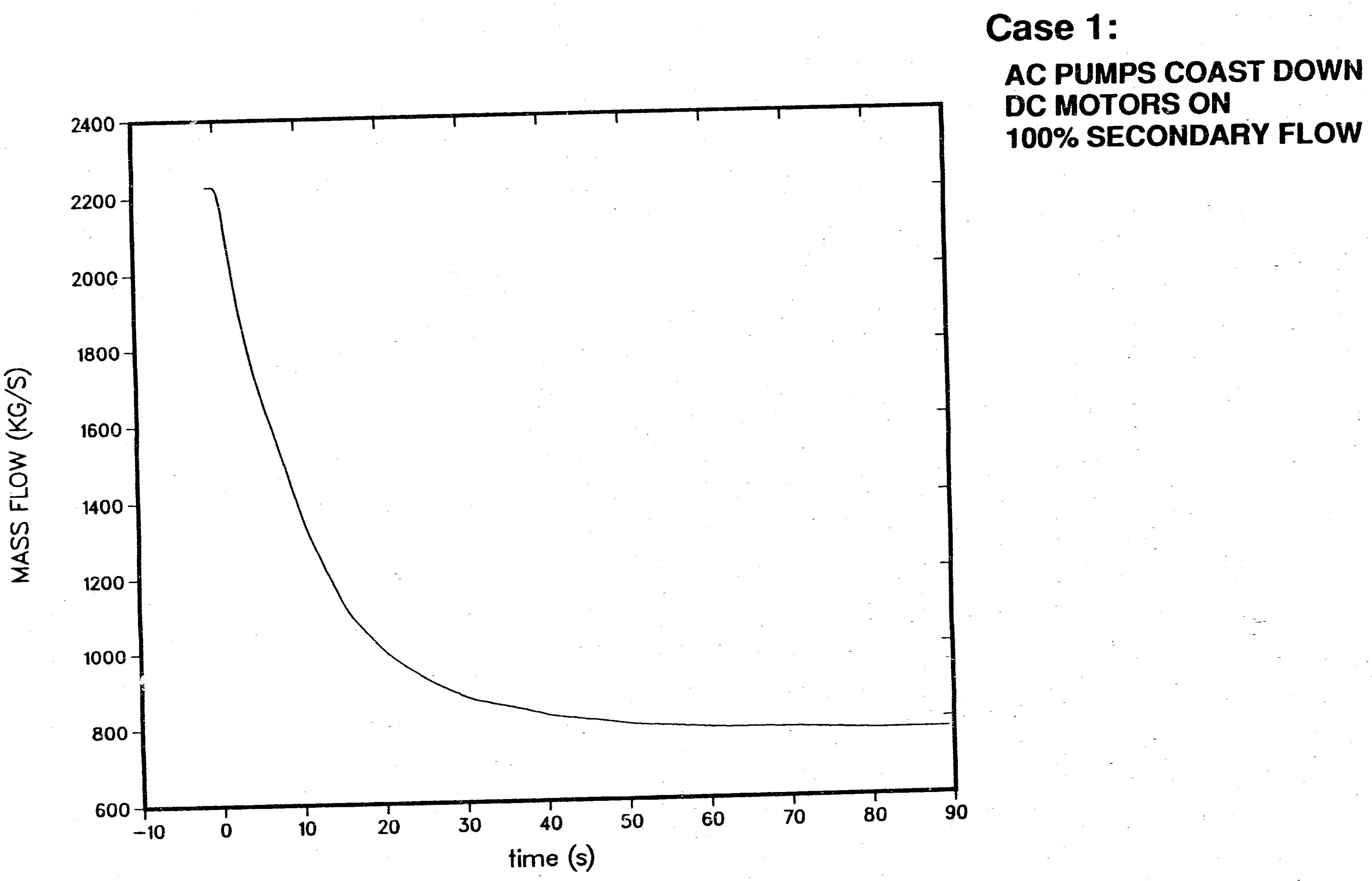


Figure 4-7. Case 1 - Loop 1 Hot Leg Mass Flow.

4.2.3 Case 2

This represents the first case of natural circulation occurring in the primary loop. All power is lost to the primary loop pumps, which coast down to zero flow after several hundred seconds. However, the secondary loop remains at full flow to remove heat from the primary. Following an initial sharp decline in temperature, the fuel and primary fluid both gradually heat up until approximately 800 seconds into the accident. At this point, natural circulation is established and the temperature profiles for the fuel and the primary coolant are flat. The primary fluid remains at least 100 degrees subcooled for this case. The pressurizer/accumulator components maintain system pressure at 175 psia. The upper-lower plenum pressure difference decreases from the operational value of 43 psid to 9 psid during natural circulation conditions. Pressure, temperature, and mass flow plots for this case are given in Figures 4-8 to 4-11.

4.2.4 Case 3

This case exhibits behavior similar to that of Case 1. The primary pumps are reduced to DC power levels and the secondary flow is reduced to gravity feed (10% of full flow). The temperature profiles for the fuel and primary fluid show an initial sharp decline followed by very gradual cooling to the end of the accident. The fuel and fluid temperatures for this nearly flat portion of the curve are slightly elevated from those for Case 1, due to the reduced flow on the secondary side of the heat exchanger. As before, the primary fluid remains subcooled by a large margin. The reduced secondary flow has negligible effect on the upper-lower plenum pressure difference, which, as in Case 1, steadily drops from the operational value to the lower value during DC pump operation. Pressure, temperature, and mass flow plots for this case are given in Figures 4-12 to 4-15.

4.2.5 Case 4

This represents the worst case of the analyzed loss-of-pumping accidents. No offsite or onsite power is available for primary or secondary pumps. The only means of accident mitigation for this case is natural circulation in the primary loop and limited secondary heat removal, which is reduced to that which can be obtained through gravity feed (10% of full flow). The results for this case are very similar to those for Case 2. The fuel and primary fluid temperatures, following an initial sharp decline, gradually rise and then flatten out at a safe level for the duration of the accident. The upper-lower plenum pressure difference, as in Case 2, drops steadily during transition from operational conditions to natural circulation conditions. Pressure, temperature, and mass flow plots for this case are given in Figures 4-16 to 4-19.

Figure 4-20 shows the maximum middle tube fuel temperature in assembly 401 for all LOPA cases.

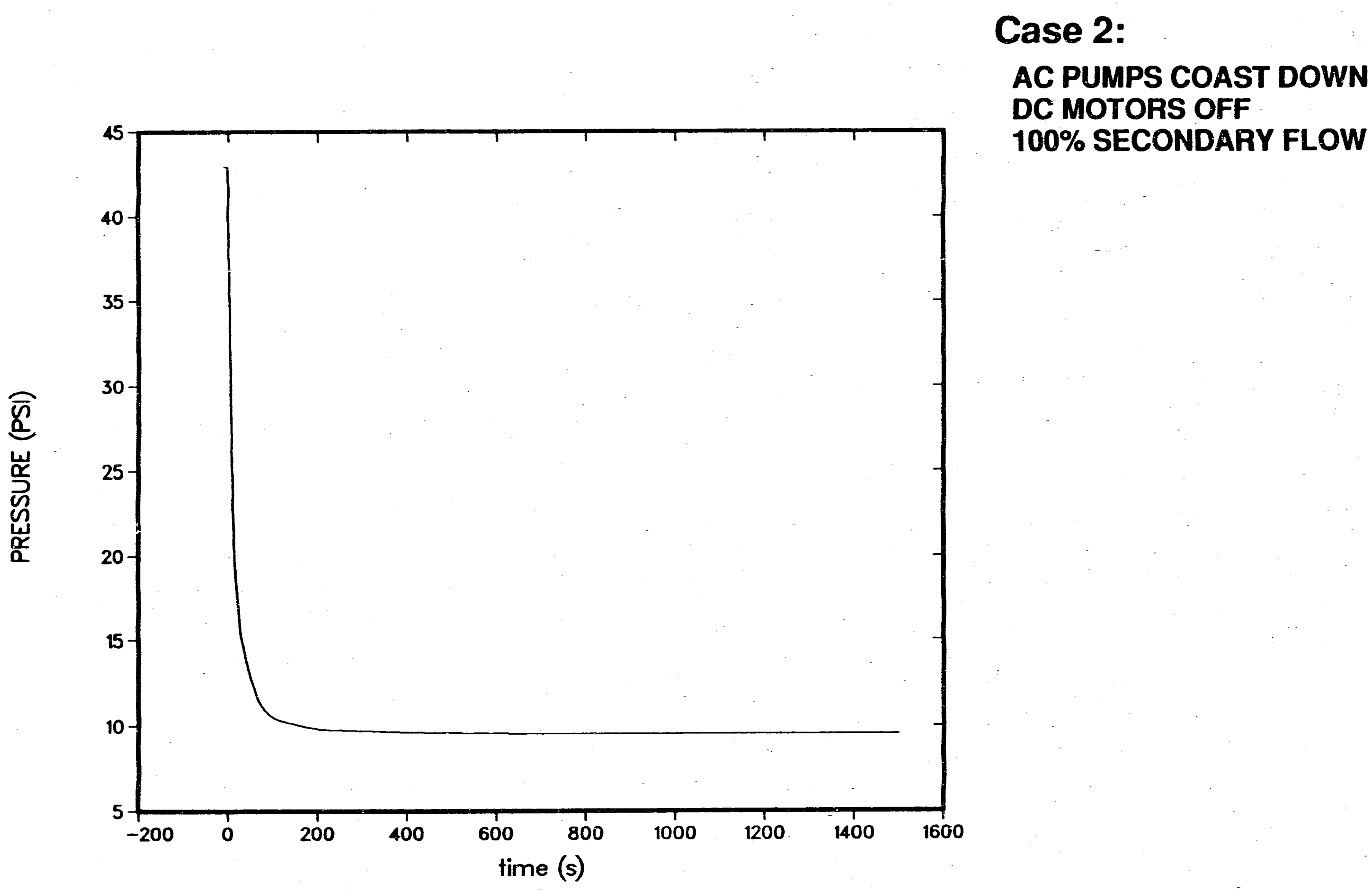


Figure 4-8. Case 2 - Upper-Lower Plenum Pressure Difference.

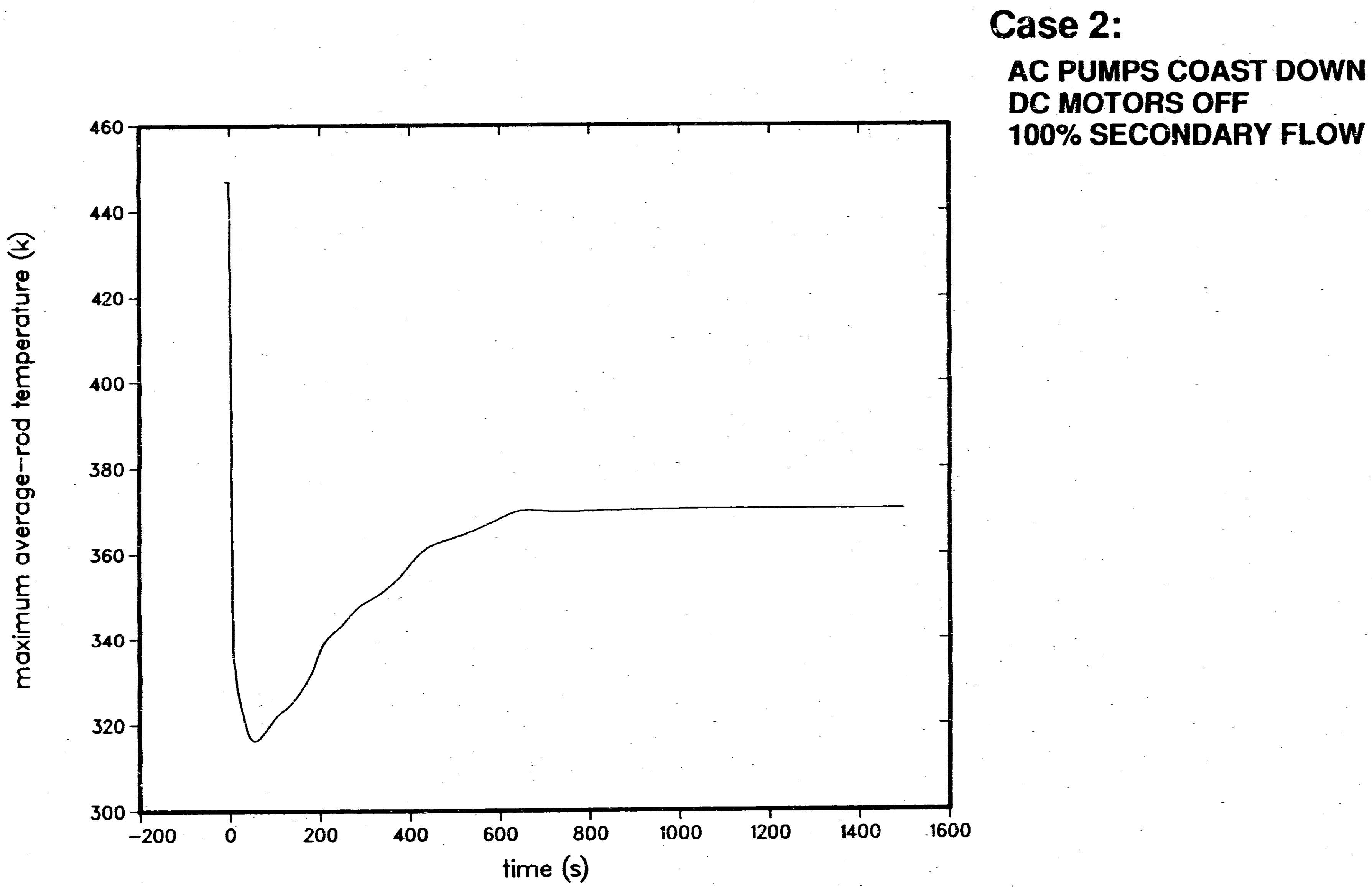


Figure 4-9. Case 2 - Maximum Middle Fuel Tube Temperature for Assembly 401.

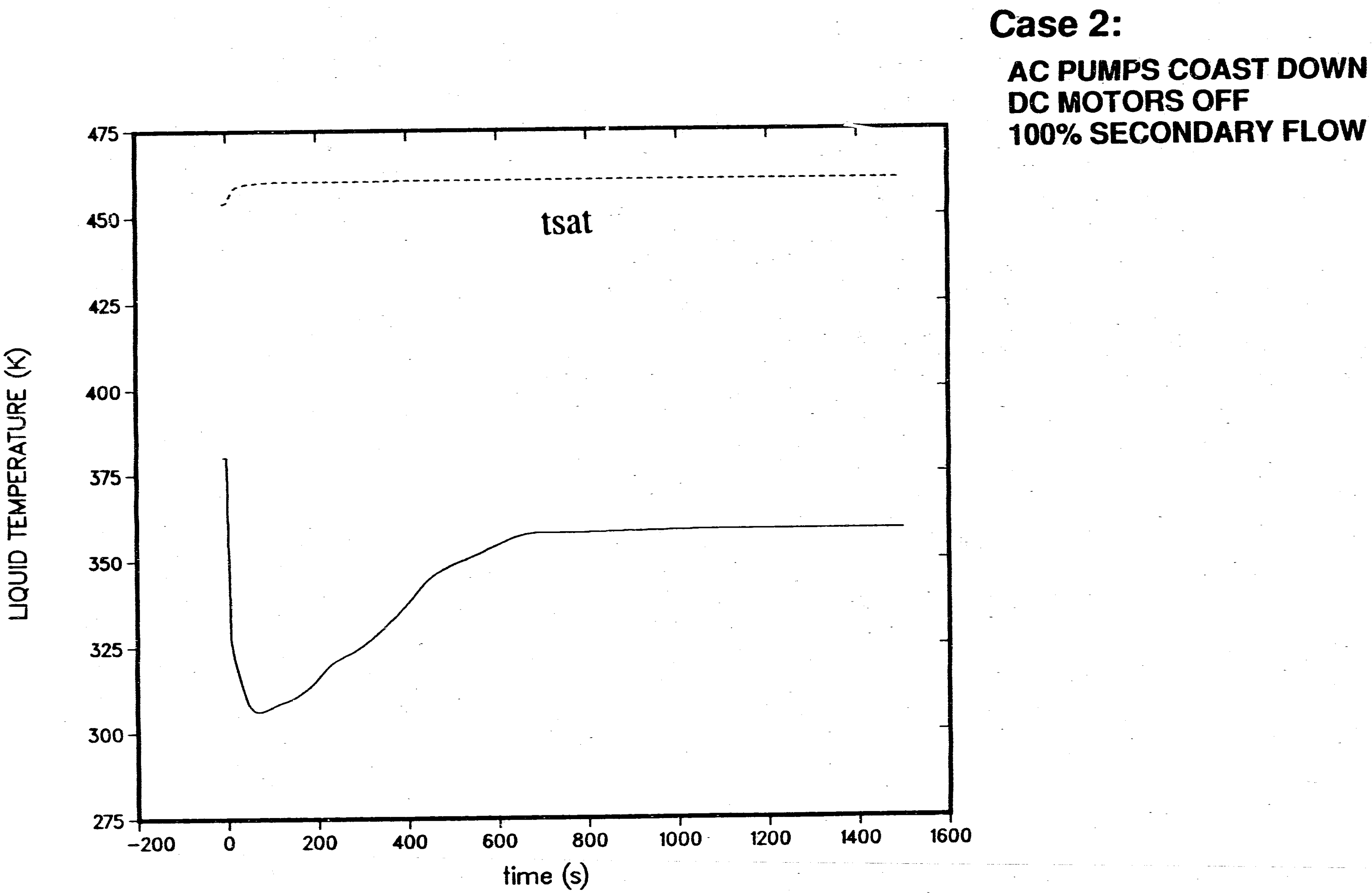


Figure 4-10. Case 2 - Liquid Temperature at Assembly 401 Outlet.

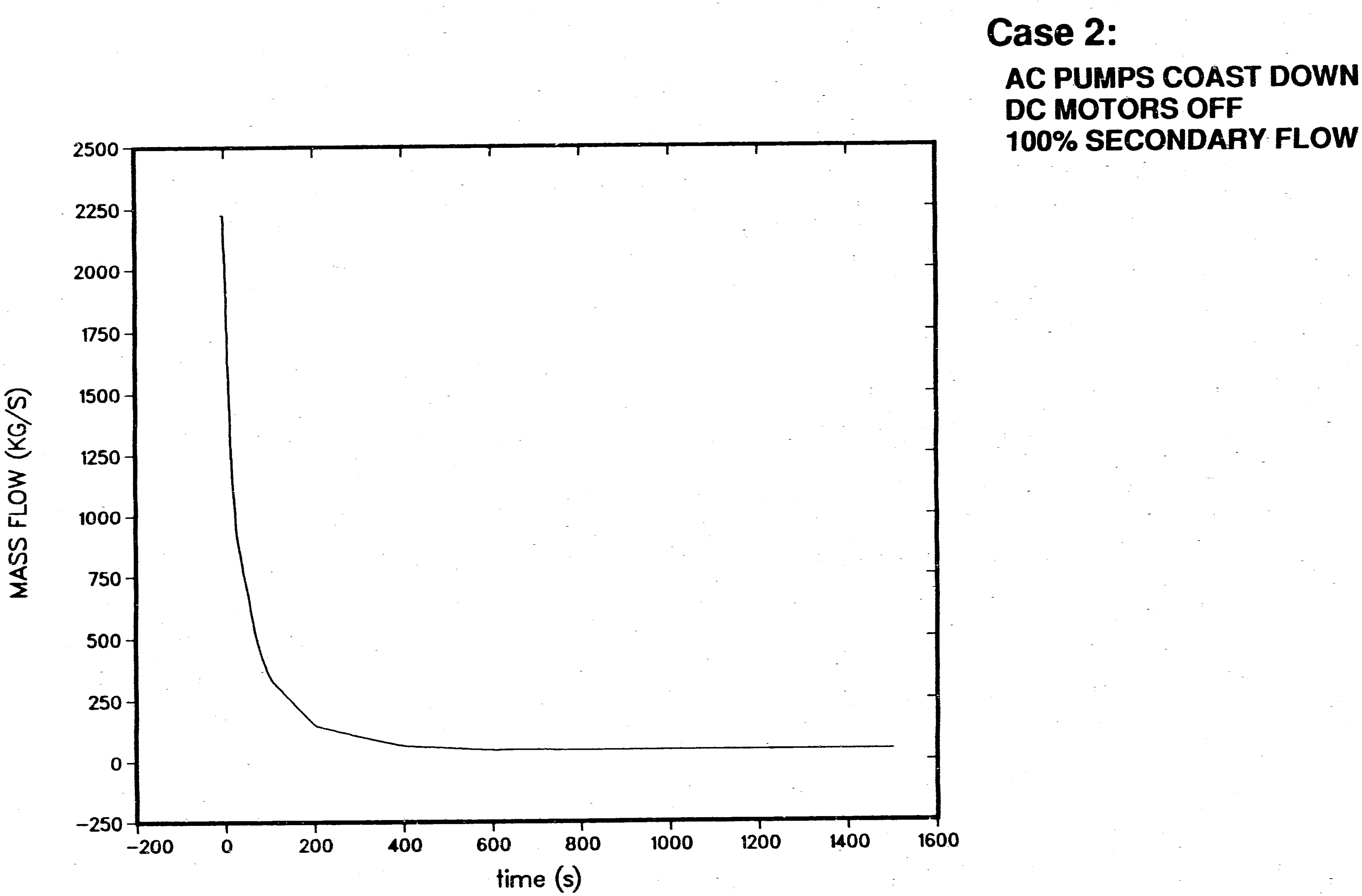


Figure 4-11. Case 2 - Loop 1 Hot Leg Mass Flow.

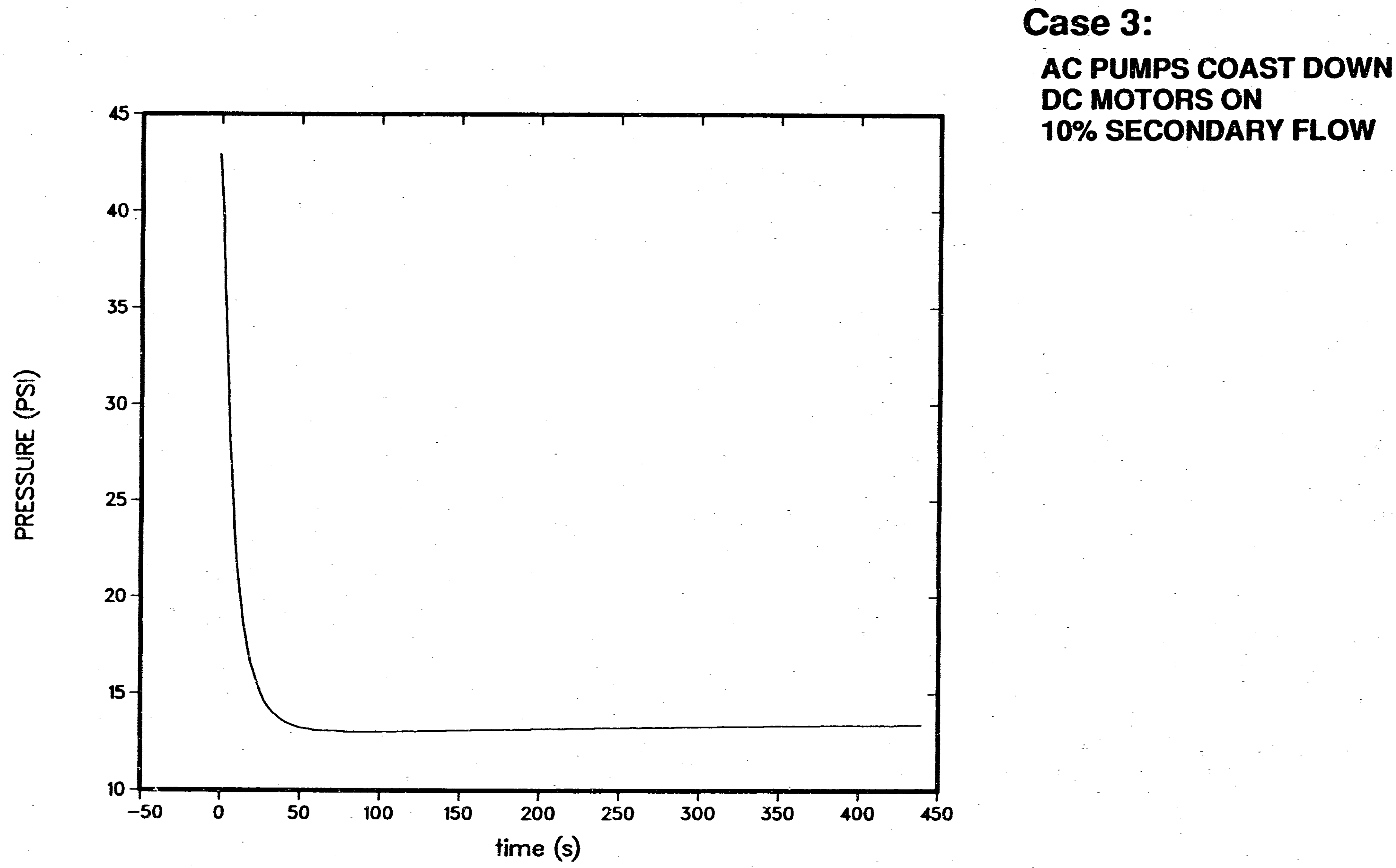


Figure 4-12. Case 3 - Upper-Lower Plenum Pressure Difference.

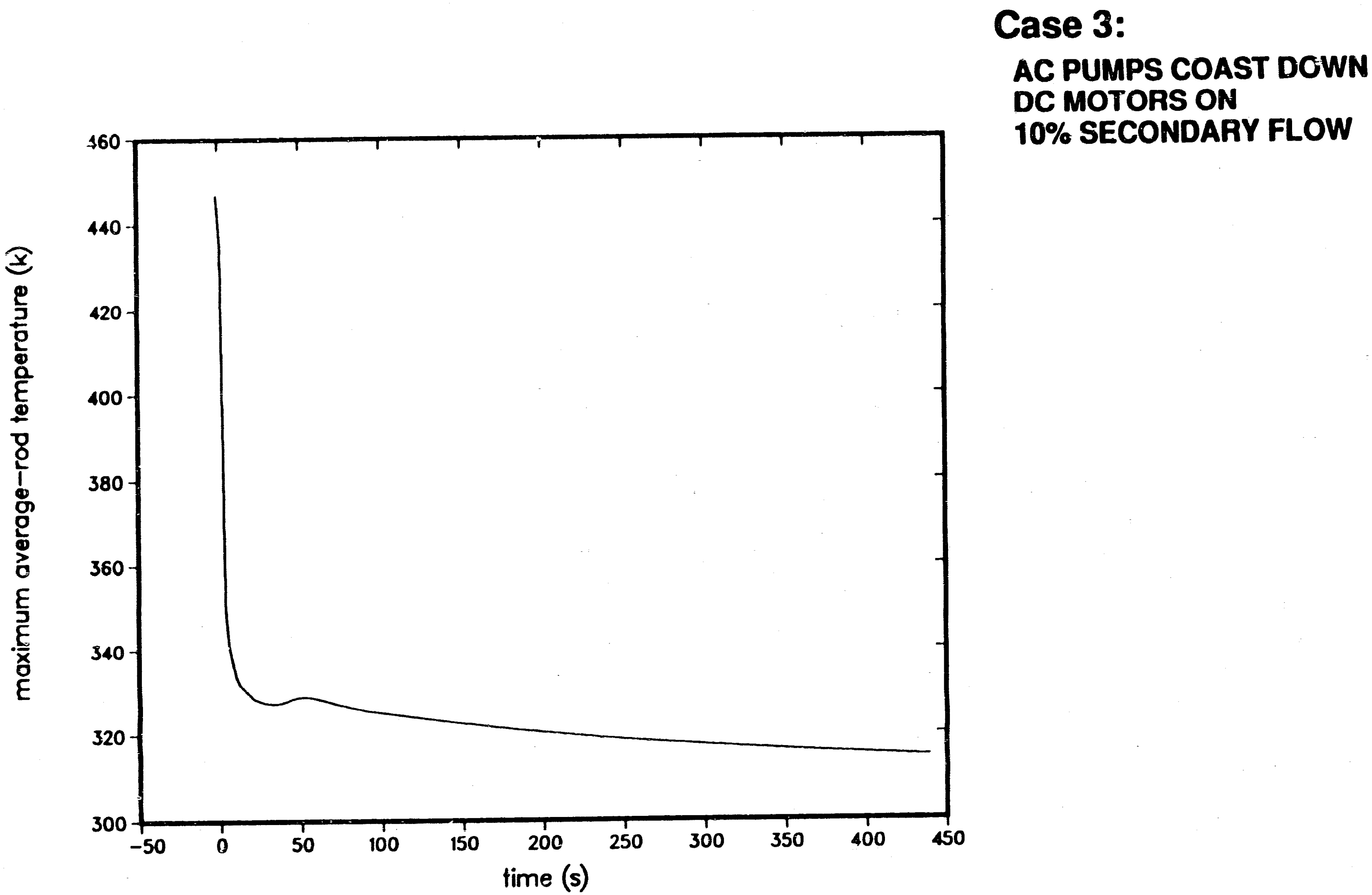


Figure 4-13. Case 3 - Maximum Middle Fuel Tube Temperature for Assembly 401.

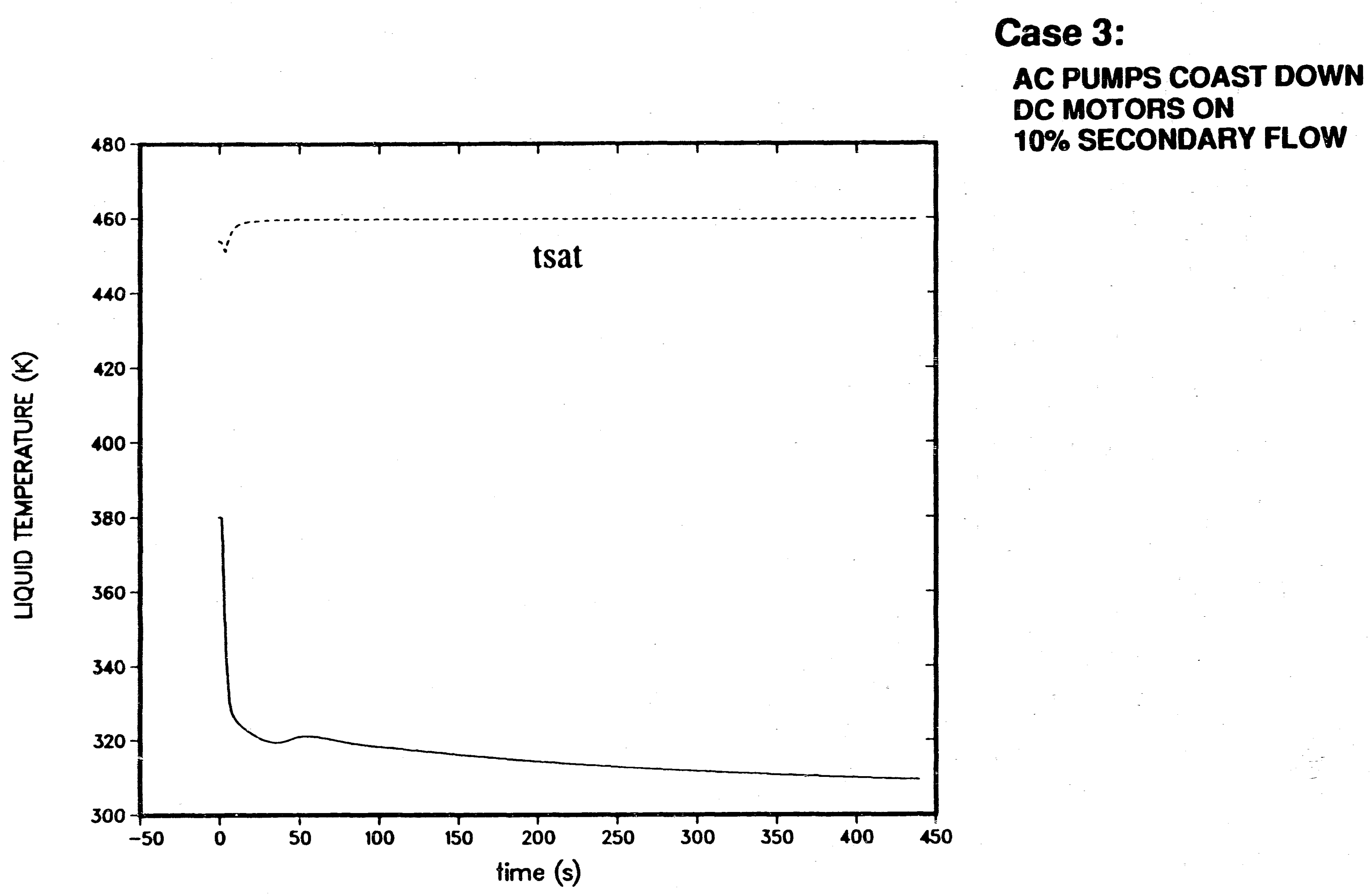
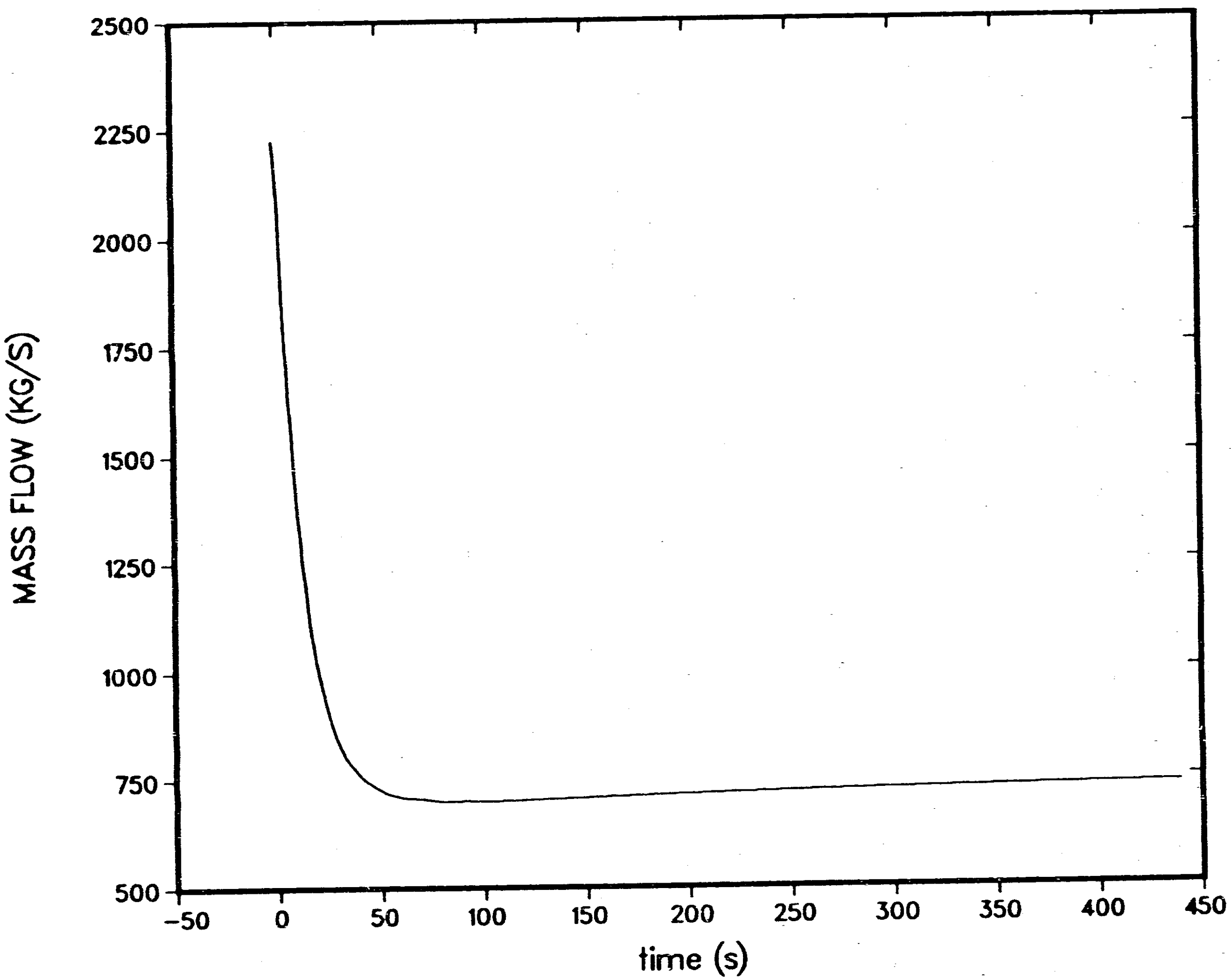


Figure 4-14. Case 3 - Liquid Temperature at Assembly 401 Outlet.



Case 3:

AC PUMPS COAST DOWN
DC MOTORS ON
10% SECONDARY FLOW

Figure 4-15. Case 3 - Loop 1 Hot Leg Mass Flow.

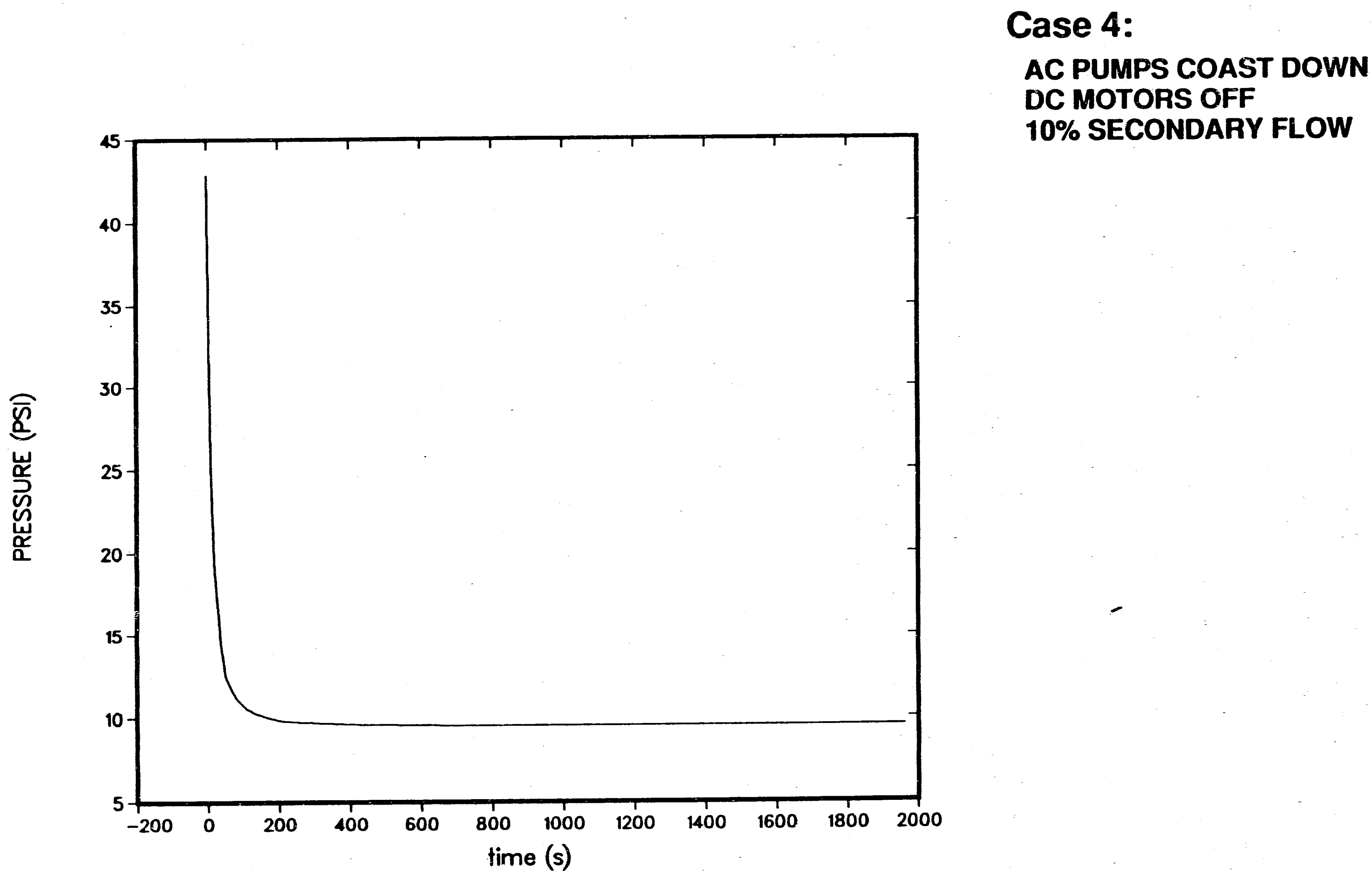


Figure 4-16. Case 4 - Upper-Lower Plenum Pressure Difference.

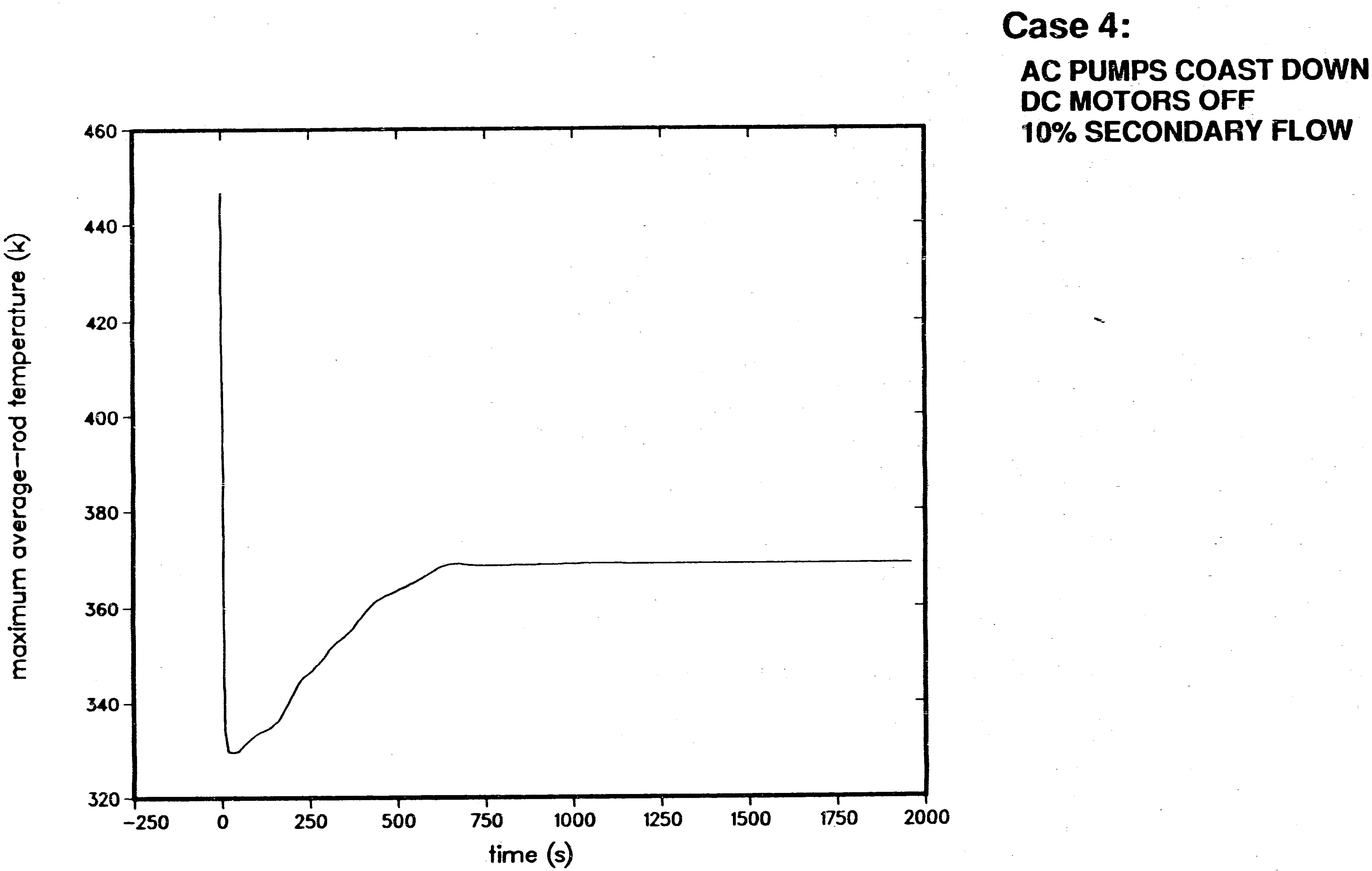


Figure 4-17. Case 4 - Maximum Middle Fuel Tube Temperature for Assembly 401.

Case 4:
AC PUMPS COAST DOWN
DC MOTORS OFF
10% SECONDARY FLOW

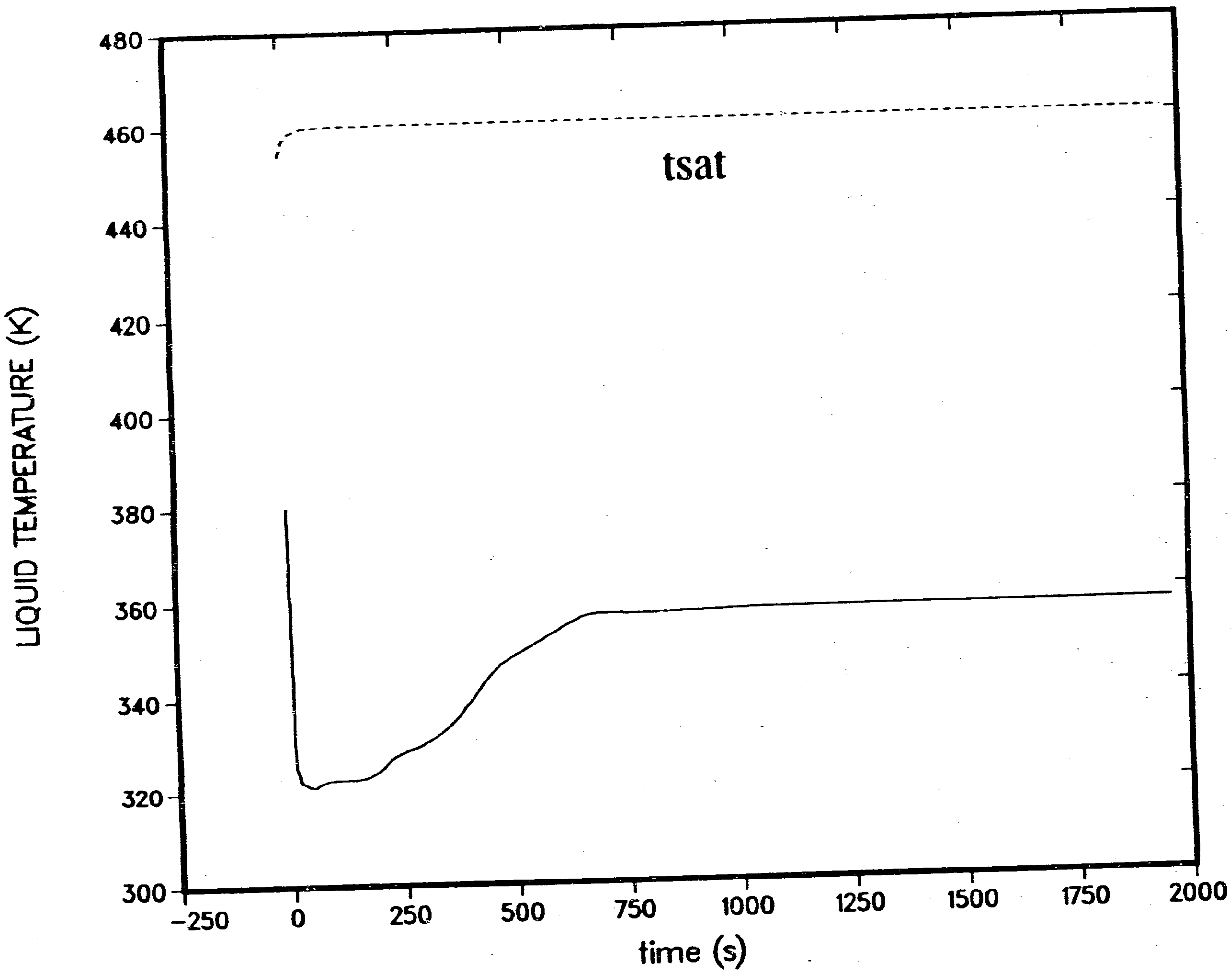


Figure 4-18. Case 4 - Liquid Temperature at Assembly 401 Outlet.

Case 4:

AC PUMPS COAST DOWN
DC MOTORS OFF
10% SECONDARY FLOW

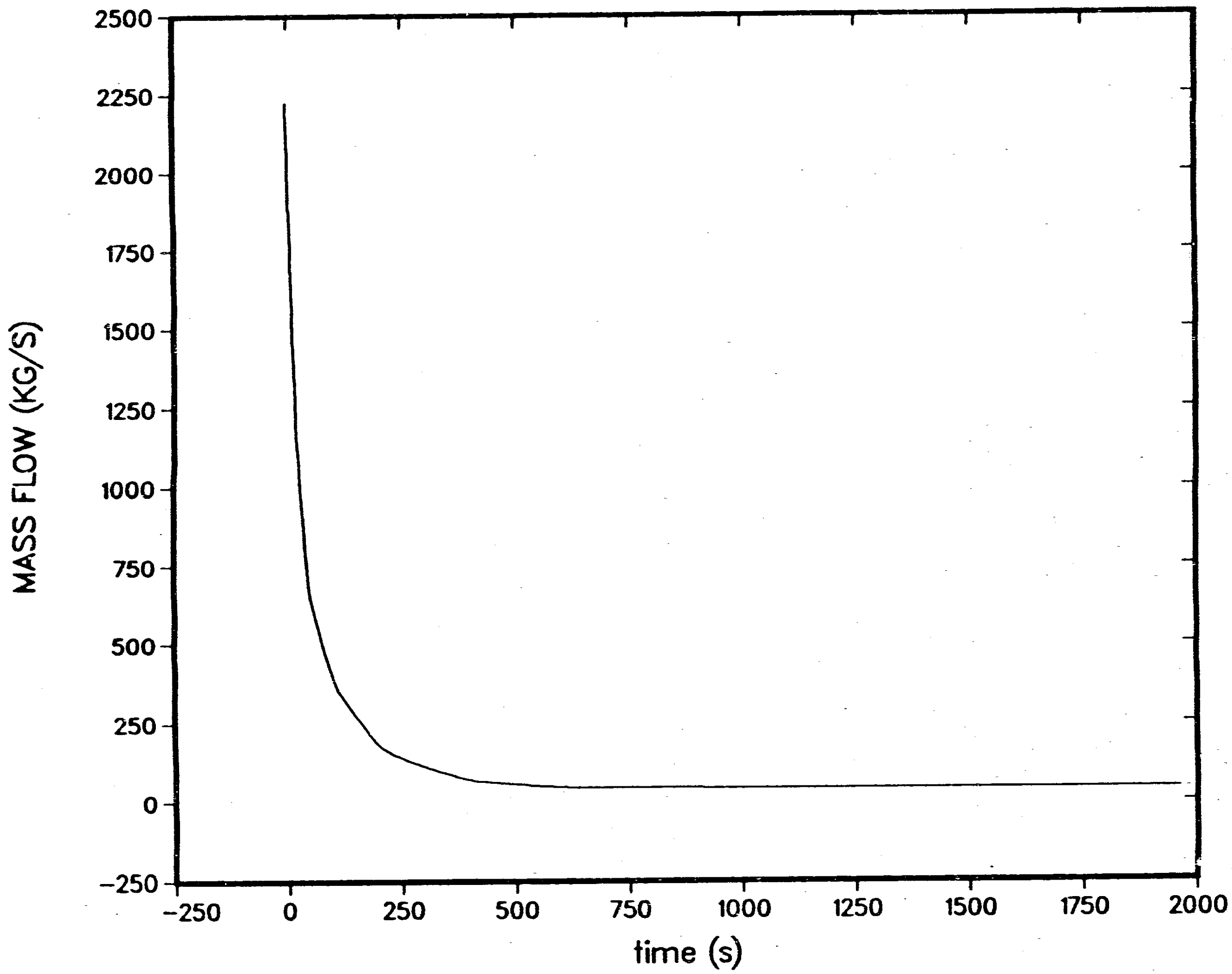


Figure 4-19. Case 4 - Loop 1 Hot Leg Mass Flow.

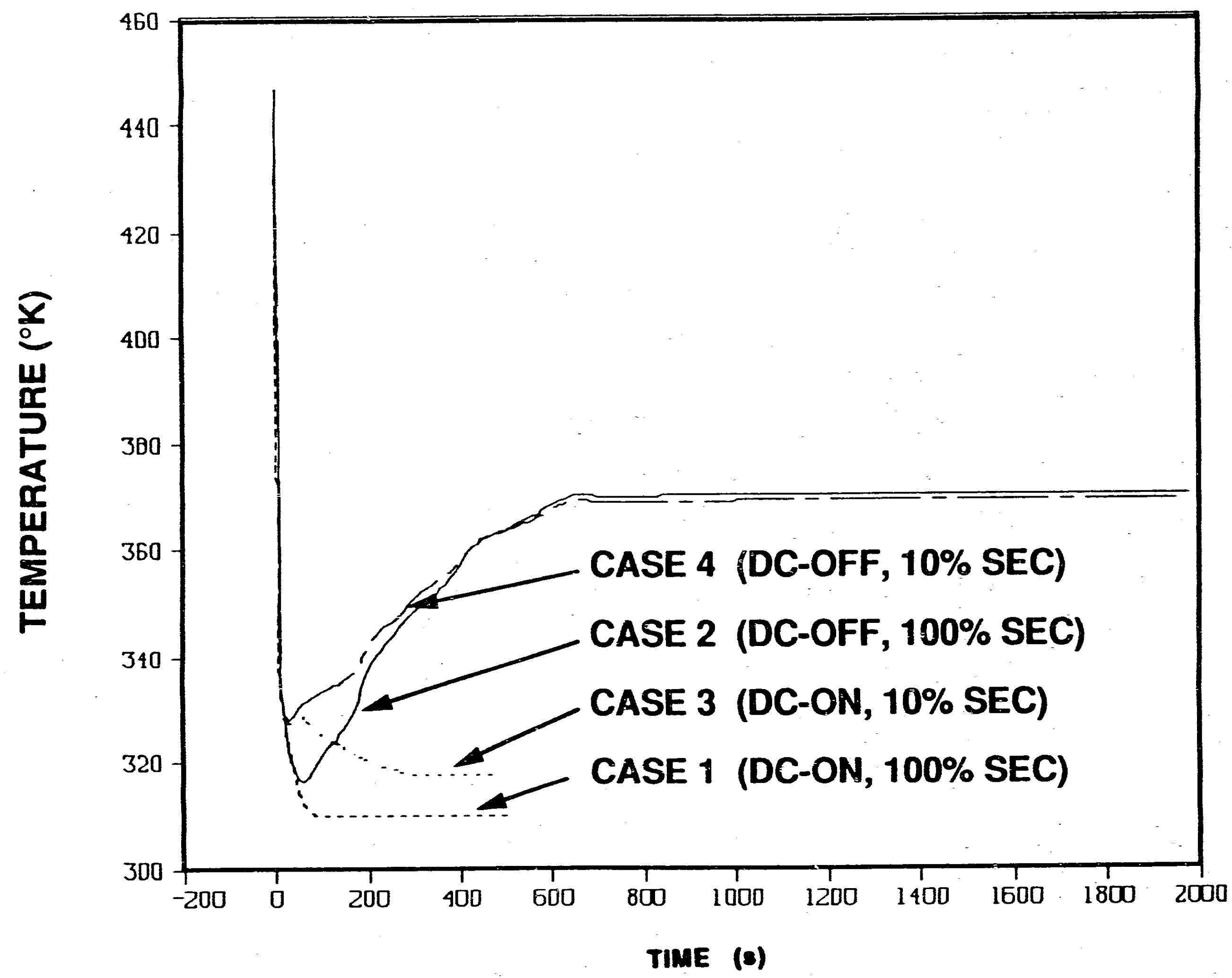


Figure 4-20. All LOPA Cases - Maximum Middle Fuel Tube Temperature for Assembly 401.

5. CONCLUSIONS

The results of this study indicate that the steady-state TRAC model of the representative HW-NPR design performs as expected, and will provide an acceptable basis from which to begin evaluation of thermal-hydraulics code capability and HWR behavior under more challenging conditions.

The results of the LOPA scenarios indicate that the HW-NPR, as modeled in this study, can maintain a full water solid condition during transition to natural circulation for the scenarios analyzed, and that the TRAC computer code is fully capable of simulating such transients. All cases of varying degrees of loss-of-pumping accidents resulted in safe transition to decay heat levels in the reactor with no cooling problems. This was observed even for the worst case of complete loss of power to both the primary and secondary pumps.

6. REFERENCES

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3. Massey, W.M., Sensitivity of Mark 22 Burnout Risk Limits to Axial Power Shapes, DPST-89-292, Savannah River Laboratory, Aiken SC, February 1989.

4. Nash C.A., Reactor Material Property Correlations for FLOWTRAN, DPST-87-692, Savannah River Laboratory, Aiken SC, September 1987.

5. Personal communication with Harry Hootman, WSRC.

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