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THERMAL-HYDRAULIC ANALYSES OF OVERCOOLING SEQUENCES FOR THE H. B. ROBINSON UNIT 2 PRESSURIZED THERMAL SHOCK STUDY DRAFT

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ABSTRACT

Oak Ridge National Laboratory (ORNL), as a part of the Nuclear Regulatory Commission's (NRC's) pressurized thermal shock (PTS) integration study for the resolution of Unresolved Safety Issue A49, identified overcooling sequences of interest to the H. B. Robinson PTS study. For each sequence, reactor vessel downcomer fluid pressure and temperature histories were required for the two-hour period following the initiating event. Analyses previously performed at the Idaho National Engineering Laboratory (INEL) fully investigated a limited number of the sequences using a detailed RELAP5 model of the H. B. Robinson, Unit 2 (HBR-2) plant. However, a full investigation of all sequences using the detailed model was not economically practical. New methods were required to generate results for the remaining sequences. Pressure and temperature histories for these remaining sequences were generated at INEL through a process combining: (1) partial-length calculations using the detailed RELAP5 model, (2) full-length calculations using a simplified RELAP5 model, and (3) hand calculations. This report documents both the methods used in this process and the results.

The sequences investigated contain significant conservatisms concerning equipment failures, operator actions, or both. Consequently, care should be taken in applying the results presented herein without an understanding of the conservatisms and assumptions.

The results of the thermal-hydraulic analyses presented here, along with additional analyses of multidimensional and fracture mechanics effects, will be utilized by ORNL to assist the NRC in resolving the PTS unresolved safety issue.

FIN No. A6047--Code Assessment and Applications (Transient Analysis)

SUMMARY

Oak Ridge National Laboratory (ORNL) identified overcooling sequences of interest to the H. B. Robinson Unit 2 (HBR-2) pressurized thermal shock study. Sequence-initiating events include isolatable and non-isolatable primary system breaks, large and small secondary system breaks, steam generator tube ruptures, and control system failures following reactor trip. Some sequences involve combinations of primary and secondary system breaks.

Reactor vessel downcomer fluid pressure and temperature histories were required for the two-hour period following the initiating event in 183 sequences. The histories will be used by ORNL as boundary conditions for fracture mechanics analyses of the reactor vessel wall.

Analyses previously performed at the Idaho National Engineering Laboratory (INEL) fully investigated a limited number of the sequences using a detailed, quality-assured RELAP5 model of the HBR-2 plant.[1] Performing similar analyses for all 183 overcooling sequences was not economically feasible. Instead, pressure and temperature histories were generated at INEL through a process combining: (1) partial-length calculations using the detailed RELAP5 model, (2) full-length calculations using a simplified RELAP5 model, and (3) hand calculations.

The sequences were grouped at ORNL by initiating event (small steam line break, large steam line break, small primary break, etc.). These groupings were not convenient for thermal-hydraulic analyses. Therefore, the sequences were regrouped according to the controlling thermal-hydraulic conditions or phenomena. For example, a group titled "Steam Line Breaks With One Affected Steam Generator" includes sequences with initiating events of large and small steam line breaks at hot standby and full-power conditions. This group also includes sequences initiated by reactor trip, with subsequent failing open of one steam line power-operated relief

valve. All sequences in the group thus share thermal-hydraulic similarity because heat removal to the single affected steam generator is the controlling thermal-hydraulic mechanism.

Within each new group, the pressure and temperature histories were generated for each sequence by consistently applying the following general method. A simplified RELAP5 model was developed specifically to address the controlling phenomena for the group. The simplified model was then benchmarked against the detailed RELAP5 model to assure the validity of the simplified model. Pressure and temperature histories were then generated by combining results of a partial-length, detailed model RELAP5 calculation over the initial portion of the transient (typically 10-15 minutes) and a simplified model RELAP5 calculation over the remainder of the two-hour period. In general the detailed model was required during the early portions of transients to simulate relatively complicated phenomena such as the effects of reactor trip, initiation of high pressure injection, reactor coolant pump trip, and secondary steam relief. However, during later portions of the transients the phenomena are, in general, quasi-steady or slowly varying; benchmarks between detailed and simplified models show the simplified models adequately represent the phenomena under these circumstances.

In addition to pressure and temperature histories, information is provided on heat transfer coefficients on the inside surface of the reactor vessel wall. As requested by ORNL, INEL identified for each sequence the previously reported scenario that has the most representative heat transfer coefficient. Estimates of uncertainty in the pressure and temperature histories are also provided.

The results of the thermal-hydraulic analyses in this report represent part of the information required by ORNL for the assessment of the PTS issue. The results of this report are not to be used directly as an indication of pressurized thermal shock severity for the sequences investigated. Following additional analyses of multi-dimensional and

fracture mechanics effects, ORNL will integrate all results and publish a report that estimates the likelihood of reactor vessel failure and identifies important event sequences, operator and control actions, and uncertainties. Computer simulations of the sequences were performed using best-estimate conditions and assumptions. However, the sequence definitions themselves contain significant conservative assumptions concerning equipment failures, operator actions or omissions, or combinations of the two. Consequently, care should be taken in applying the results presented herein without an understanding of the conservatisms and assumptions.

Results of the analyses presented here indicate that the most severe sequences for PTS are those with secondary-side breaks. The larger the break and the longer auxiliary feedwater continues, the more severe the results. A failure of the charging to throttle makes these sequences even more severe.

Results for reactor trip and steam generator tube rupture sequences were relatively less severe, with results for the latter controlled mainly by the assumptions of operator action.

Medium break loss-of-coolant accidents (LOCAs) were found to produce not only very low temperatures but also low pressures which limited severity. Small break LOCAs were less severe than medium break LOCAs.

Combination primary- and secondary-side breaks were found to be less severe than the corresponding secondary-side-only breaks, due to the lower primary system pressures.

NOMENCLATURE

AFW	Auxiliary feedwater
ASG	Affected steam generator
DC	Downcomer
ECCS	Emergency core cooling system
HBR-2	H. B. Robinson, Unit 2
HPI	High-pressure injection
INEL	Idaho National Engineering Laboratory
LOCA	Loss-of-coolant accident
LPI	Low-pressure injection
MBLOCA	Medium break loss-of-coolant accident
MFW	Main feedwater
MSIV	Main steam isolation valve
NR	Narrow range
NRC	U. S. Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
P-I	Proportional-integral
PORV	Power-operated relief valve
PTS	Pressurized thermal shock
PWR	Pressurized water reactor
RCP	Reactor coolant pump
RCS	Reactor coolant system
RV	Reactor vessel
SBLOCA	Small break loss-of-coolant accident
SDV	Steam dump valve (turbine bypass)
SG	Steam generator
SGTR	Steam generator tube rupture
USG	Unaffected steam generator

CONTENTS

FIGURES

TABLES

1. INTRODUCTION

Rapid cooling of a reactor pressure vessel, accompanied by high coolant pressure, during a transient or accident is referred to as pressurized thermal shock (PTS). In late 1981 the U. S. Nuclear Regulatory Commission (NRC) designated PTS as an unresolved safety issue and developed a task action plan (TAP A-49) to resolve the issue.

The safety issue exists because rapid cooling at the inner surface of the reactor vessel wall produces thermal stresses within the wall. As long as the resistance to fracture of the reactor vessel is high, overcooling transients will not cause vessel failure. However, NRC staff analyses (SECY-85-465) showed that certain older plants with copper and other impurities in vessel weldments may become sensitive to PTS after several years as the nil-ductility transition temperature of the weld material gradually increases. The purpose of the thermal-hydraulic analyses presented in this report is to define the behavior of a plant during various kinds of postulated severe overcooling transients, with multiple failures of equipment and without operator corrective action. For each of these postulated transients, Oak Ridge National Laboratory (ORNL) will calculate both the reactor vessel temperature distribution and the stresses during the transient as well as the conditional probability of vessel failure if the transient should occur. ORNL will publish a report that integrates these results to estimate the likelihood of PTS driving a crack through the reactor vessel wall, and to identify important event sequences, operator and control actions, and uncertainties.

This series of analyses is intended to provide information to help the NRC staff confirm the bases for the screening criteria in the proposed PTS rule (proposed 10CFR 50.61) and to determine the content required for licensees' plant-specific safety analysis reports and the acceptance criteria for corrective measures.

The analyses presented in this report were performed using best-estimate modeling assumptions for both plant conditions and operator

responses to the events specified in the sequence descriptions. The reader is cautioned, however, that the sequence descriptions were based on extremely conservative assumptions regarding equipment malfunctions, operator actions and omissions, or combinations of these. Thus, while the analyses results represent best-estimate plant responses to the sequences as defined, they do not represent the most probable plant responses to the sequence-initiating events.

Analyses presented in this report were performed for the H. B. Robinson, Unit 2 (HBR-2) pressurized water reactor operated at Hartsville, South Carolina by Carolina Power and Light Company. The reactor is of Westinghouse three-loop design and is currently being operated at reduced power, due in part to steam generator tube plugging. Analyses presented here assumed a full rated thermal power of 2300 MW. This is the power level at which the plant is expected to operate following a replacement of steam generators. Other anticipated plant changes were also incorporated into the analyses.

This is the second plant for which PTS thermal-hydraulic analyses have been performed at INEL. Similar work, described in Reference 2, was performed for the Oconee-1 plant, a PWR of Babcock and Wilcox design.

This report is organized as follows: the sequences of interest to PTS, as defined at ORNL, appear in Section 2; the methods by which pressure and temperature histories were determined appear in Section 3; results of the analyses appear in Sections 4 through 13; estimates for heat transfer coefficients of the reactor vessel wall appear in Section 14; a discussion of uncertainties appears in Section 15; conclusions are stated in Section 16; and references appear in Section 17. Plotted results, showing reactor vessel downcomer pressure and temperature histories for each sequence, are presented in Appendix A.

2. SEQUENCE DEFINITIONS

Oak Ridge National Laboratory (ORNL) defined for detailed study sequences of interest to the HBR-2 pressurized thermal shock study. The sequence list includes those sequences expected to have the highest probability of vessel fracture. The sequences were primarily grouped by ORNL according to sequence-initiating event and operating power level. Tables 1 through 12 present the sequence definitions for Groups 1 through 12, respectively. Corresponding initiating events for each group are as follows:

Group 1	Small break LOCA at power
Group 2	Medium break LOCA at power
Group 3	Small break LOCA at hot standby
Group 4	Medium break LOCA at hot standby
Group 5	Small steam line break at power
Group 6	Large steam line break at power
Group 7	Small steam line break at hot standby
Group 8	Large steam line break at hot standby
Group 9	Reactor trip at full power
Group 10	Steam generator tube rupture at hot standby
Group 11	Isolatable small break LOCA at power
Group 12	Isolatable medium break LOCA at power

In the terminology used here: (1) a small break LOCA is initiated by a single, stuck-open pressurizer power-operated relief valve (PORV); (2) a medium break LOCA is initiated by a 0.0635-m (2-1/2 in.) diameter hot leg break; (3) a small steam line break is a single, stuck-open steam line PORV; and (4) a large steam line break is a double-ended rupture of one steam line downstream of the flow restrictor and upstream of the main steam isolation valve. The term "at power" means during 2300 MW full-power operation; "at hot standby" means 100 hours after reactor shutdown, assuming infinite operating time. The term "isolatable LOCA" refers to a loss-of-coolant accident in which the operator, after a delay, takes action

TABLE 1. SEQUENCE DESCRIPTIONS--SMALL BREAK LOCA AT POWER

NO	TURB TRIP	FW REG VLV	PORV CLOSE	SDV CLOSE	SI SGL GEN	HP INJ	MFIV CLOSE	MFW PMP TP
1	Occurs on demand	Occurs on demand	Close on demand	Close on demand	Generated on demand	Occurs on demand	Close on demand	Trip on demand
2	Occurs on demand	Occurs on demand	Close on demand	Close on demand	Generated on demand	Occurs on demand	Close on demand	Trip on demand
3	Occurs on demand	Occurs on demand	Close on demand	Close on demand	Generated on demand	Occurs on demand	Close on demand	Trip on demand
4	Occurs on demand	Occurs on demand	Close on demand	Close on demand	Generated on demand	Occurs on demand	Close on demand	Trip on demand
5	Occurs on demand	Occurs on demand	NA	1 fails to close	Generated on demand	Occurs on demand	Close on demand	Trip on demand
6	Occurs on demand	Occurs on demand	NA	2 fail to close	Generated on demand	Occurs on demand	Close on demand	Trip on demand
7	Occurs on demand	Occurs on demand	NA	3 fail to close	Generated on demand	Occurs on demand	Close on demand	Trip on demand
8	Occurs on demand	Occurs on demand	NA	5 fail to close	Generated on demand	Occurs on demand	Close on demand	Trip on demand
9	Occurs on demand	Occurs on demand	Fails on 1 line	Close on demand	Generated on demand	Occurs on demand	Close on demand	Trip on demand
10	Occurs on demand	Occurs on demand	Fails on 1 line	Close on demand	Generated on demand	Occurs on demand	Close on demand	Trip on demand
11	Occurs on demand	Occurs on demand	Fails on 1 line	Close on demand	Generated on demand	Occurs on demand	Close on demand	Trip on demand
12	Occurs on demand	Occurs on demand	Fails on 2 lines	Close on demand	Generated on demand	Occurs on demand	Close on demand	Trip on demand
13	Occurs on demand	Fails on 1 line	NA	Close on demand	Generated on demand	Occurs on demand	Close on demand	Trip on demand
14	Occurs on demand	Fails on 1 line	NA	Close on demand	Generated on demand	Occurs on demand	Close on demand	Trip on demand
15	Occurs on demand	Fails on 1 line	NA	Close on demand	Generated on demand	Occurs on demand	Close on demand	Trip on demand
16	Occurs on demand	Fails on 1 line	NA	1 fails to close	Generated on demand	Occurs on demand	Close on demand	Trip on demand
17	Occurs on demand	Fails on 2 lines	NA	Close on demand	Generated on demand	Occurs on demand	Close on demand	Trip on demand
18	Occurs on demand	Fails on 3 lines	NA	Close on demand	Generated on demand	Occurs on demand	Close on demand	Trip on demand

NO	MSIV CLOSE	AFW ACT	AFW AUTO	AFW SG ISO	ACC INJ	THR AFW	RHR ACT
1	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Prior to high level alarm	Occurs on demand
2	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Failure to throttle	Occurs on demand
3	NA	Actuates on demand	Overfeed	NA	Occurs on demand	Prior to high level alarm	Occurs on demand
4	NA	Actuates on demand	Overfeed	NA	Occurs on demand	Failure to throttle	Occurs on demand
5	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Prior to high level alarm	Occurs on demand
6	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Prior to high level alarm	Occurs on demand
7	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Prior to high level alarm	Occurs on demand
8	Close on demand	Actuates on demand	Auto controlled	NA	Occurs on demand	Prior to high level alarm	Occurs on demand
9	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Prior to high level alarm	Occurs on demand
10	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Failure to throttle	Occurs on demand
11	NA	Actuates on demand	Overfeed	NA	Occurs on demand	Prior to high level alarm	Occurs on demand
12	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Prior to high level alarm	Occurs on demand
13	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Prior to high level alarm	Occurs on demand
14	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Failure to throttle	Occurs on demand
15	NA	Actuates on demand	Overfeed	NA	Occurs on demand	Prior to high level alarm	Occurs on demand
16	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Prior to high level alarm	Occurs on demand
17	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Prior to high level alarm	Occurs on demand
18	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Prior to high level alarm	Occurs on demand

TABLE 2. SEQUENCE DESCRIPTIONS--MEDIUM BREAK LOCA AT POWER

NO	TURB TRIP	FW REG VLV	PORV CLOSE	SDV CLOSE	SI SGL GEN	HP INJ	MFIV CLOSE	MFW PMP TP
1	Occurs on demand	Occurs on demand	Close on demand	Close on demand	Generated on demand	Occurs on demand	Close on demand	Trip on demand
2	Occurs on demand	Occurs on demand	Close on demand	Close on demand	Generated on demand	Occurs on demand	Close on demand	Trip on demand
3	Occurs on demand	Occurs on demand	Close on demand	Close on demand	Generated on demand	Occurs on demand	Close on demand	Trip on demand
4	Occurs on demand	Occurs on demand	Close on demand	Close on demand	Generated on demand	Occurs on demand	Close on demand	Trip on demand
5	Occurs on demand	Occurs on demand	NA	1 fails to close	Generated on demand	Occurs on demand	Close on demand	Trip on demand
6	Occurs on demand	Occurs on demand	NA	2 fail to close	Generated on demand	Occurs on demand	Close on demand	Trip on demand
7	Occurs on demand	Occurs on demand	NA	3 fail to close	Generated on demand	Occurs on demand	Close on demand	Trip on demand
8	Occurs on demand	Occurs on demand	NA	5 fail to close	Generated on demand	Occurs on demand	Close on demand	Trip on demand
9	Occurs on demand	Occurs on demand	Fails on 1 line	Close on demand	Generated on demand	Occurs on demand	Close on demand	Trip on demand
10	Occurs on demand	Occurs on demand	Fails on 1 line	Close on demand	Generated on demand	Occurs on demand	Close on demand	Trip on demand
11	Occurs on demand	Occurs on demand	Fails on 2 lines	Close on demand	Generated on demand	Occurs on demand	Close on demand	Trip on demand

NO	MSIV CLOSE	AFW ACT	AFW AUTO	AFW SG ISO	ACC INJ	THR AFW	RHR ACT
1	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Prior to high level alarm	Occurs on demand
2	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Failure to throttle	Occurs on demand
3	NA	Actuates on demand	Overfeed	NA	Occurs on demand	Prior to high level alarm	Occurs on demand
4	NA	Actuates on demand	Overfeed	NA	Occurs on demand	Failure to throttle	Occurs on demand
5	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Prior to high level alarm	Occurs on demand
6	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Prior to high level alarm	Occurs on demand
7	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Prior to high level alarm	Occurs on demand
8	Close on demand	Actuates on demand	Auto controlled	NA	Occurs on demand	Prior to high level alarm	Occurs on demand
9	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Prior to high level alarm	Occurs on demand
10	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Failure to throttle	Occurs on demand
11	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Prior to high level alarm	Occurs on demand

TABLE 3. SEQUENCE DESCRIPTIONS--SMALL BREAK LOCA AT HOT STANDBY

NO	SDV CLOSE	SI SGL GEN	HP INJ	MFIV CLOSE	MFW PMP TP	MSIV CLOSE	AFW ACT
1	Close on demand	Generated on demand	Occurs on demand	Close on demand	Trip on demand	NA	Actuates on demand
2	Close on demand	Generated on demand	Occurs on demand	Close on demand	Trip on demand	NA	Actuates on demand
3	1 Fails to close	Generated on demand	Occurs on demand	Close on demand	Trip on demand	NA	Actuates on demand

NO	AFW AUTO	AFW SG ISO	ACC INJ	THR AFW	RHR ACT
1	Auto controlled	NA	Occurs on demand	Prior to high level alarm	Occurs on demand
2	Auto controlled	NA	Occurs on demand	Failure to throttle	Occurs on demand
3	Auto controlled	NA	Occurs on demand	Prior to high level alarm	Occurs on demand

TABLE 4. SEQUENCE DESCRIPTIONS--MEDIUM BREAK LOCA AT HOT STANDBY

NO	SDV CLOSE	SI SGL GEN	HP INJ	MFIV CLOSE	MFW PMP TP	MSIV CLOSE	AFW ACT
1	Close on demand	Generated on demand	Occurs on demand	Close on demand	Trip on demand	NA	Actuates on demand

NO	AFW AUTO	AFW SG ISO	ACC INJ	THR AFW	RHR ACT
1	Auto controlled	NA	Occurs on demand	Prior to high level alarm	Occurs on demand

TABLE 5. SEQUENCE DESCRIPTIONS--SMALL STEAM LINE BREAK AT POWER

NO	CKVL RP CL	SDV CLOSE	FW REG VLV	SI SGL GEN	MFIV CLOSE	MFW PMP TP	MSIV CLOSE
1	Closes on demand	Close on demand	Occurs on demand	Generated on demand	Close on demand	Trip on demand	NA
2	Closes on demand	Close on demand	Occurs on demand	Generated on demand	Close on demand	Trip on demand	NA
3	Closes on demand	Close on demand	Occurs on demand	Generated on demand	Close on demand	Trip on demand	NA
4	Closes on demand	Close on demand	Occurs on demand	Generated on demand	Close on demand	Trip on demand	NA
5	Closes on demand	Close on demand	Occurs on demand	Generated on demand	Close on demand	Trip on demand	NA
6	Closes on demand	Close on demand	Occurs on demand	Generated on demand	Close on demand	Trip on demand	NA
7	Closes on demand	Close on demand	Occurs on demand	Generated on demand	Close on demand	Trip on demand	NA
8	Closes on demand	Close on demand	Occurs on demand	Generated on demand	Close on demand	Trip on demand	NA
9	Closes on demand	Close on demand	Occurs on demand	Generated on demand	Close on demand	Trip on demand	NA
10	Closes on demand	Close on demand	Occurs on demand	Generated on demand	Close on demand	Trip on demand	NA
11	Closes on demand	Close on demand	Occurs on demand	Generated on demand	Close on demand	Trip on demand	NA
12	Closes on demand	Close on demand	Occurs on demand	Generated on demand	Close on demand	Trip on demand	NA
13	Closes on demand	Close on demand	Occurs on demand	Generated on demand	Close on demand	Trip on demand	NA
14	Closes on demand	1 fails to close	Occurs on demand	Generated on demand	Close on demand	Trip on demand	NA
15	Closes on demand	1 fails to close	Occurs on demand	Generated on demand	Close on demand	Trip on demand	NA
16	Closes on demand	1 fails to close	Occurs on demand	Generated on demand	Close on demand	Trip on demand	NA
17	Closes on demand	1 fails to close	Occurs on demand	Generated on demand	Close on demand	Trip on demand	NA
18	Closes on demand	1 fails to close	Occurs on demand	Generated on demand	Close on demand	Trip on demand	NA
19	Closes on demand	2 fail to close	Occurs on demand	Generated on demand	Close on demand	Trip on demand	NA
20	Closes on demand	2 fail to close	Occurs on demand	Generated on demand	Close on demand	Trip on demand	NA
21	Fails to close	Close on demand	Occurs on demand	Generated on demand	Close on demand	Trip on demand	NA
22	Fails to close	Close on demand	Occurs on demand	Generated on demand	Close on demand	Trip on demand	NA
23	Fails to close	Close on demand	Occurs on demand	Generated on demand	Close on demand	Trip on demand	NA
24	Fails to close	Close on demand	Occurs on demand	Generated on demand	Close on demand	Trip on demand	NA

TABLE 5. (continued)

NO	AFW ACT	AFW AUTO	AFW SG ISO	THR AFW	HP INJ	THR HPI CH
1	Actuates on demand	Auto controlled	Occurs as required	Prior to high level alarm	Occurs on demand	As required
2	Actuates on demand	Auto controlled	Occurs as required	Prior to high level alarm	Occurs on demand	Failure to throttle
3	Actuates on demand	Auto controlled	Occurs as required	Failure to throttle	Occurs on demand	As required
4	Actuates on demand	Auto controlled	Occurs as required	Failure to throttle	Occurs on demand	Failure to throttle
5	Actuates on demand	Auto controlled	Not isolated	Prior to high level alarm	Occurs on demand	As required
6	Actuates on demand	Auto controlled	Not isolated	Prior to high level alarm	Occurs on demand	Failure to throttle
7	Actuates on demand	Auto controlled	Not isolated	Failure to throttle	Occurs on demand	As required
8	Actuates on demand	Auto controlled	Not isolated	Failure to throttle	Occurs on demand	Failure to throttle
9	Actuates on demand	Overfeed	Occurs as required	Prior to high level alarm	Occurs on demand	As required
10	Actuates on demand	Overfeed	Occurs as required	Prior to high level alarm	Occurs on demand	Failure to throttle
11	Actuates on demand	Overfeed	Occurs as required	Failure to throttle	Occurs on demand	As required
12	Actuates on demand	Overfeed	Not isolated	Prior to high level alarm	Occurs on demand	As required
13	Actuates on demand	Overfeed	Not isolated	Failure to throttle	Occurs on demand	As required
14	Actuates on demand	Auto controlled	Occurs as required	Prior to high level alarm	Occurs on demand	As required
15	Actuates on demand	Auto controlled	Occurs as required	Prior to high level alarm	Occurs on demand	Failure to throttle
16	Actuates on demand	Auto controlled	Occurs as required	Failure to throttle	Occurs on demand	As required
17	Actuates on demand	Auto controlled	Not isolated	Prior to high level alarm	Occurs on demand	As required
18	Actuates on demand	Overfeed	Occurs on demand	Prior to high level alarm	Occurs on demand	As required
19	Actuates on demand	Auto controlled	Occurs as required	Prior to high level alarm	Occurs on demand	As required
20	Actuates on demand	Auto controlled	Not isolated	Prior to high level alarm	Occurs on demand	As required
21	Actuates on demand	Auto controlled	Occurs as required	Prior to high level alarm	Occurs on demand	As required
22	Actuates on demand	Auto controlled	Occurs as required	Prior to high level alarm	Occurs on demand	Failure to throttle
23	Actuates on demand	Auto controlled	Occurs as required	Failure to throttle	Occurs on demand	As required
24	Actuates on demand	Overfeed	Occurs as required	Prior to high level alarm	Occurs on demand	As required

TABLE 6. SEQUENCE DESCRIPTIONS--LARGE STEAM LINE BREAK AT POWER

NO	CKVL RP CL	MSIV RP CL	RM MSIV CL	SDV CLOSE	SI SGL GEN	FW REG VLV	MFIV CLOSE	MFW PMP TP
1	Closes on demand	NA	Close on demand	NA	Generated on demand	Occurs on demand	Close on demand	Trip on demand
2	Closes on demand	NA	Close on demand	NA	Generated on demand	Occurs on demand	Close on demand	Trip on demand
3	Closes on demand	NA	Close on demand	NA	Generated on demand	Occurs on demand	Close on demand	Trip on demand
4	Closes on demand	NA	Close on demand	NA	Generated on demand	Occurs on demand	Close on demand	Trip on demand
5	Closes on demand	NA	Close on demand	NA	Generated on demand	Occurs on demand	Close on demand	Trip on demand
6	Closes on demand	NA	Close on demand	NA	Generated on demand	Occurs on demand	Close on demand	Trip on demand
7	Closes on demand	NA	Close on demand	NA	Generated on demand	Occurs on demand	Close on demand	Trip on demand
8	Closes on demand	NA	Close on demand	NA	Generated on demand	Occurs on demand	Close on demand	Trip on demand
9	Closes on demand	NA	Close on demand	NA	Generated on demand	Occurs on demand	Close on demand	Trip on demand

NO	AFW ACT	AFW AUTO	AFW SG ISO	THR AFW	HP INJ	THR HPI CH
1	Actuates on demand	Auto controlled	Occurs as required	Prior to high level alarm	Occurs on demand	As required
2	Actuates on demand	Auto controlled	Occurs as required	Prior to high level alarm	Occurs on demand	Failure to throttle
3	Actuates on demand	Auto controlled	Occurs as required	Failure to throttle	Occurs on demand	As required
4	Actuates on demand	Auto controlled	Occurs as required	Failure to throttle	Occurs on demand	Failure to throttle
5	Actuates on demand	Auto controlled	Not isolated	Prior to high level alarm	Occurs on demand	As required
6	Actuates on demand	Auto controlled	Not isolated	Prior to high level alarm	Occurs on demand	Failure to throttle
7	Actuates on demand	Auto controlled	Not isolated	Failure to throttle	Occurs on demand	As required
8	Actuates on demand	Overfeed	Occurs as required	Prior to high level alarm	Occurs on demand	As required
9	Actuates on demand	Overfeed	Not isolated	Prior to high level alarm	Occurs on demand	As required

TABLE 7. SEQUENCE DESCRIPTIONS--SMALL STEAM LINE BREAK AT HOT STANDBY

NO	CKVL RP CL	SDV CLOSE	SI SGL GEN	MFIV CLOSE	MFW PMP TP	MSIV CLOSE	AFW ACT
1	Closes on demand	Close on demand	Generated on demand	Close on demand	Trip on demand	NA	Actuates on demand
2	Closes on demand	Close on demand	Generated on demand	Close on demand	Trip on demand	NA	Actuates on demand
3	Closes on demand	Close on demand	Generated on demand	Close on demand	Trip on demand	NA	Actuates on demand
4	Closes on demand	Close on demand	Generated on demand	Close on demand	Trip on demand	NA	Actuates on demand
5	Closes on demand	Close on demand	Generated on demand	Close on demand	Trip on demand	NA	Actuates on demand
6	Closes on demand	Close on demand	Generated on demand	Close on demand	Trip on demand	NA	Actuates on demand
7	Closes on demand	Close on demand	Generated on demand	Close on demand	Trip on demand	NA	Actuates on demand
8	Closes on demand	Close on demand	Generated on demand	Close on demand	Trip on demand	NA	Actuates on demand
9	Closes on demand	1 fails to close	Generated on demand	Close on demand	Trip on demand	NA	Actuates on demand
10	Closes on demand	1 fails to close	Generated on demand	Close on demand	Trip on demand	NA	Actuates on demand
11	Closes on demand	2 fail to close	Generated on demand	Close on demand	Trip on demand	NA	Actuates on demand
12	Fails to close	Close on demand	Generated on demand	Close on demand	Trip on demand	NA	Actuates on demand

NO	AFW AUTO	AFW SG ISO	THR AFW	HP INJ	THR HPI CH
1	Auto controlled	Occurs as required	Prior to high level alarm	Occurs on demand	As required
2	Auto controlled	Occurs as required	Prior to high level alarm	Occurs on demand	Failure to throttle
3	Auto controlled	Occurs as required	Failure to throttle	Occurs on demand	As required
4	Auto controlled	Not isolated	Prior to high level alarm	Occurs on demand	As required
5	Auto controlled	Not isolated	Prior to high level alarm	Occurs on demand	Failure to throttle
6	Auto controlled	Not isolated	Failure to throttle	Occurs on demand	As required
7	Overfeed	Occurs as required	Prior to high level alarm	Occurs on demand	As required
8	Overfeed	Not isolated	Prior to high level alarm	Occurs on demand	As required
9	Auto controlled	Occurs as required	Prior to high level alarm	Occurs on demand	As required
10	Auto controlled	Not isolated	Prior to high level alarm	Occurs on demand	As required
11	Auto controlled	Occurs as required	Prior to high level alarm	Occurs on demand	As required
12	Auto controlled	Occurs as required	Prior to high level alarm	Occurs on demand	As required

TABLE 8. SEQUENCE DESCRIPTIONS--LARGE STEAM LINE BREAK AT HOT STANDBY

NO	CKVL RP CL	MSIV RP CL	RM MSIV CL	SDV CLOSE	SI SGL GEN	MFIV CLOSE	MFW PMP TP	AFW ACT
1	Closes on demand	NA	Close on demand	Close on demand	Generated on demand	Close on demand	Trip on demand	Actuates on demand
2	Closes on demand	NA	Close on demand	Close on demand	Generated on demand	Close on demand	Trip on demand	Actuates on demand
3	Closes on demand	NA	Close on demand	Close on demand	Generated on demand	Close on demand	Trip on demand	Actuates on demand
4	Closes on demand	NA	Close on demand	Close on demand	Generated on demand	Close on demand	Trip on demand	Actuates on demand
5	Closes on demand	NA	Close on demand	Close on demand	Generated on demand	Close on demand	Trip on demand	Actuates on demand
6	Closes on demand	NA	Close on demand	Close on demand	Generated on demand	Close on demand	Trip on demand	Actuates on demand

NO	AFW AUTO	AFW SG ISO	THR AFW	HP INJ	THR HPI CH
1	Auto controlled	Occurs on demand	Prior to high level alarm	Occurs on demand	As required
2	Auto controlled	Occurs on demand	Prior to high level alarm	Occurs on demand	Failure to throttle
3	Auto controlled	Occurs on demand	Failure to throttle	Occurs on demand	As required
4	Auto controlled	Not isolated	Prior to high level alarm	Occurs on demand	As required
5	Overfeed	Occurs as required	Prior to high level alarm	Occurs on demand	As required
6	Overfeed	Not isolated	Prior to high level alarm	Occurs on demand	As required

TABLE 9. SEQUENCE DESCRIPTIONS--REACTOR TRIP AT FULL POWER

NO	TURB TRIPS	FW REG VLV	PORV CLOSE	SDV CLOSE	SI SGL GEN	MFIV CLOSE	MFW PMP TP
1	Occurs on demand	Occurs on demand	Close on demand	Close on demand	NA	NA	NA
2	Occurs on demand	Occurs on demand	NA	1 fails to close	Occurs on demand	Close on demand	Trip on demand
3	Occurs on demand	Occurs on demand	NA	1 fails to close	Occurs on demand	Close on demand	Trip on demand
4	Occurs on demand	Occurs on demand	NA	1 fails to close	Occurs on demand	Close on demand	Trip on demand
5	Occurs on demand	Occurs on demand	NA	1 fails to close	Occurs on demand	Close on demand	Trip on demand
6	Occurs on demand	Occurs on demand	NA	1 fails to close	Occurs on demand	Close on demand	Trip on demand
7	Occurs on demand	Occurs on demand	NA	1 fails to close	Occurs on demand	Close on demand	Trip on demand
8	Occurs on demand	Occurs on demand	NA	1 fails to close	Occurs on demand	Close on demand	Trip on demand
9	Occurs on demand	Occurs on demand	NA	2 fail to close	Occurs on demand	Close on demand	Trip on demand
10	Occurs on demand	Occurs on demand	NA	2 fail to close	Occurs on demand	Close on demand	Trip on demand
11	Occurs on demand	Occurs on demand	NA	2 fail to close	Occurs on demand	Close on demand	Trip on demand
12	Occurs on demand	Occurs on demand	NA	2 fail to close	Occurs on demand	Close on demand	Trip on demand
13	Occurs on demand	Occurs on demand	NA	2 fail to close	Occurs on demand	Close on demand	Trip on demand
14	Occurs on demand	Occurs on demand	NA	3 fail to close	Occurs on demand	Close on demand	Trip on demand
15	Occurs on demand	Occurs on demand	NA	3 fail to close	Occurs on demand	Close on demand	Trip on demand
16	Occurs on demand	Occurs on demand	NA	3 fail to close	Occurs on demand	Close on demand	Trip on demand
17	Occurs on demand	Occurs on demand	NA	3 fail to close	Occurs on demand	Close on demand	Trip on demand
18	Occurs on demand	Occurs on demand	NA	3 fail to close	Occurs on demand	Close on demand	Trip on demand
19	Occurs on demand	Occurs on demand	NA	5 fail to close	Occurs on demand	Close on demand	Trip on demand
20	Occurs on demand	Occurs on demand	NA	5 fail to close	Occurs on demand	Close on demand	Trip on demand
21	Occurs on demand	Occurs on demand	NA	5 fail to close	Occurs on demand	Close on demand	Trip on demand
22	Occurs on demand	Occurs on demand	NA	5 fail to close	Occurs on demand	Close on demand	Trip on demand
23	Occurs on demand	Occurs on demand	NA	5 fail to close	Occurs on demand	Close on demand	Trip on demand
24	Occurs on demand	Occurs on demand	NA	5 fail to close	Occurs on demand	Close on demand	Trip on demand
25	Occurs on demand	Occurs on demand	Fails on 1 line	Close on demand	Occurs on demand	Close on demand	Trip on demand

TABLE 9. (continued)

NO	MSIV CLOSE	AFW ACT	AFW AUTO	AFW SG ISO	HP INJ	THR HPI CH	THR AFW
1	NA	NA	NA	NA	NA	NA	NA
2	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	As required	Prior to high level alarm
3	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	As required	Failure to throttle
4	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Failure to throttle	Prior to high level alarm
5	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Failure to throttle	Failure to throttle
6	NA	Actuates on demand	Overfeed	NA	Occurs on demand	As required	Prior to high level alarm
7	NA	Actuates on demand	Overfeed	NA	Occurs on demand	As required	Failure to throttle
8	NA	Actuates on demand	Overfeed	NA	Occurs on demand	Failure to throttle	Prior to high level alarm
9	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	As required	Prior to high level alarm
10	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	As required	Failure to throttle
11	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Failure to throttle	Prior to high level alarm
12	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Failure to throttle	Failure to throttle
13	NA	Actuates on demand	Overfeed	NA	Occurs on demand	As required	Prior to high level alarm
14	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	As required	Prior to high level alarm
15	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	As required	Failure to throttle
16	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Failure to throttle	Prior to high level alarm
17	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Failure to throttle	Failure to throttle
18	NA	Actuates on demand	Overfeed	NA	Occurs on demand	As required	Prior to high level alarm
19	Occurs on demand	Actuates on demand	Auto controlled	NA	Occurs on demand	As required	Prior to high level alarm
20	Occurs on demand	Actuates on demand	Auto controlled	NA	Occurs on demand	As required	Failure to throttle
21	Occurs on demand	Actuates on demand	Auto controlled	NA	Occurs on demand	Failure to throttle	Prior to high level alarm
22	Occurs on demand	Actuates on demand	Auto controlled	NA	Occurs on demand	Failure to throttle	Failure to throttle
23	Occurs on demand	Actuates on demand	Overfeed	NA	Occurs on demand	As required	Prior to high level alarm
24	1 fails to close	Actuates on demand	Auto controlled	Occurs as required	Occurs on demand	As required	Prior to high level alarm
25	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	As required	Prior to high level alarm

TABLE 9. (continued)

NO	TURB TRIPS	FW REG VLV	PORV CLOSE	SDV CLOSE	SI SGL GEN	MFIV CLOSE	MFW PMP TP
26	Occurs on demand	Occurs on demand	Fails on 1 line	Close on demand	Occurs on demand	Close on demand	Trip on demand
27	Occurs on demand	Occurs on demand	Fails on 1 line	Close on demand	Occurs on demand	Close on demand	Trip on demand
28	Occurs on demand	Occurs on demand	Fails on 1 line	Close on demand	Occurs on demand	Close on demand	Trip on demand
29	Occurs on demand	Occurs on demand	Fails on 1 line	Close on demand	Occurs on demand	Close on demand	Trip on demand
30	Occurs on demand	Occurs on demand	Fails on 1 line	Close on demand	Occurs on demand	Close on demand	Trip on demand
31	Occurs on demand	Occurs on demand	Fails on 1 line	Close on demand	Occurs on demand	Close on demand	Trip on demand
32	Occurs on demand	Occurs on demand	Fails on 1 line	Close on demand	Occurs on demand	Close on demand	Trip on demand
33	Occurs on demand	Occurs on demand	Fails on 2 lines	Close on demand	Occurs on demand	Close on demand	Trip on demand
34	Occurs on demand	Occurs on demand	Fails on 2 lines	Close on demand	Occurs on demand	Close on demand	Trip on demand
35	Occurs on demand	Occurs on demand	Fails on 2 lines	Close on demand	Occurs on demand	Close on demand	Trip on demand
36	Occurs on demand	Occurs on demand	Fails on 2 lines	Close on demand	Occurs on demand	Close on demand	Trip on demand
37	Occurs on demand	Occurs on demand	Fails on 2 lines	Close on demand	Occurs on demand	Close on demand	Trip on demand
38	Occurs on demand	Occurs on demand	Fails on 2 lines	Close on demand	Occurs on demand	Close on demand	Trip on demand
39	Occurs on demand	Occurs on demand	Fails on 2 lines	Close on demand	Occurs on demand	Close on demand	Trip on demand
40	Occurs on demand	Occurs on demand	Fails on 2 lines	Close on demand	Occurs on demand	Close on demand	Trip on demand
41	Occurs on demand	Occurs on demand	Fails on 3 lines	Close on demand	Occurs on demand	Close on demand	Trip on demand
42	Occurs on demand	Occurs on demand	Fails on 3 lines	Close on demand	Occurs on demand	Close on demand	Trip on demand
43	Occurs on demand	Occurs on demand	Fails on 3 lines	Close on demand	Occurs on demand	Close on demand	Trip on demand
44	Occurs on demand	Occurs on demand	Fails on 3 lines	Close on demand	Occurs on demand	Close on demand	Trip on demand
45	Occurs on demand	Occurs on demand	Fails on 3 lines	Close on demand	Occurs on demand	Close on demand	Trip on demand
46	Occurs on demand	Occurs on demand	Fails on 3 lines	Close on demand	Occurs on demand	Close on demand	Trip on demand
47	Occurs on demand	Occurs on demand	Fails on 3 lines	Close on demand	Occurs on demand	Close on demand	Trip on demand
48	Occurs on demand	Fails on 1 line	NA	Close on demand	Occurs on demand	Close on demand	Trip on demand
49	Occurs on demand	Fails on 1 line	NA	Close on demand	Occurs on demand	Close on demand	Trip on demand
50	Occurs on demand	Fails on 1 line	NA	Close on demand	Occurs on demand	Close on demand	Trip on demand

TABLE 9. (continued)

NO	MSIV CLOSE	AFW ACT	AFW AUTO	AFW SG ISO	HP INJ	THR HPI CH	THR AFW
26	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	As required	Failure to throttle
27	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Failure to throttle	Prior to high level alarm
28	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Failure to throttle	Failure to throttle
29	NA	Actuates on demand	Overfeed	NA	Occurs on demand	As required	Prior to high level alarm
30	NA	Actuates on demand	Overfeed	NA	Occurs on demand	As required	Failure to throttle
31	NA	Actuates on demand	Overfeed	NA	Occurs on demand	Failure to throttle	Prior to high level alarm
32	NA	Actuates on demand	Overfeed	NA	Occurs on demand	Failure to throttle	Failure to throttle
33	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	As required	Prior to high level alarm
34	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	As required	Failure to throttle
35	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Failure to throttle	Prior to high level alarm
36	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Failure to throttle	Failure to throttle
37	NA	Actuates on demand	Overfeed	NA	Occurs on demand	As required	Prior to high level alarm
38	NA	Actuates on demand	Overfeed	NA	Occurs on demand	As required	Failure to throttle
39	NA	Actuates on demand	Overfeed	NA	Occurs on demand	Failure to throttle	Prior to high level alarm
40	NA	Actuates on demand	Overfeed	NA	Occurs on demand	Failure to throttle	Failure to throttle
41	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	As required	Prior to high level alarm
42	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	As required	Failure to throttle
43	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Failure to throttle	Prior to high level alarm
44	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Failure to throttle	Failure to throttle
45	NA	Actuates on demand	Overfeed	NA	Occurs on demand	As required	Prior to high level alarm
46	NA	Actuates on demand	Overfeed	NA	Occurs on demand	As required	Failure to throttle
47	NA	Actuates on demand	Overfeed	NA	Occurs on demand	Failure to throttle	Prior to high level alarm
48	NA	Actuates on demand	Auto Controlled	NA	Occurs on demand	As required	Prior to high level alarm
49	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	As required	Failure to throttle
50	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Failure to throttle	Prior to high level alarm

TABLE 9. (continued)

NO	TURB TRIPS	FW REG VLV	PORV CLOSE	SDV CLOSE	SI SGL GEN	MFIV CLOSE	MFW PMP TP
51	Occurs on demand	Fails on 1 line	NA	Close on demand	Occurs on demand	Close on demand	Trip on demand
52	Occurs on demand	Fails on 1 line	NA	Close on demand	Occurs on demand	Close on demand	Trip on demand
53	Occurs on demand	Fails on 1 line	NA	Close on demand	Occurs on demand	Close on demand	Trip on demand
54	Occurs on demand	Fails on 1 line	NA	Close on demand	Occurs on demand	Close on demand	Trip on demand
55	Occurs on demand	Fails on 1 line	NA	Close on demand	Occurs on demand	Close on demand	Trip on demand
56	Occurs on demand	Fails on 1 line	NA	Close on demand	Occurs on demand	1 fails to close	Fail to trip
57	Occurs on demand	Fails on 1 line	NA	1 fails to close	Occurs on demand	Close on demand	Trip on demand
58	Occurs on demand	Fails on 1 line	NA	1 fails to close	Occurs on demand	Close on demand	Trip on demand
59	Occurs on demand	Fails on 1 line	NA	1 fails to close	Occurs on demand	Close on demand	Trip on demand
60	Occurs on demand	Fails on 1 line	NA	1 fails to close	Occurs on demand	Close on demand	Trip on demand
61	Occurs on demand	Fails on 1 line	NA	1 fails to close	Occurs on demand	Close on demand	Trip on demand
62	Occurs on demand	Fails on 1 line	NA	2 fail to close	Occurs on demand	Close on demand	Trip on demand
63	Occurs on demand	Fails on 1 line	NA	2 fail to close	Occurs on demand	Close on demand	Trip on demand
64	Occurs on demand	Fails on 1 line	NA	2 fail to close	Occurs on demand	Close on demand	Trip on demand
65	Occurs on demand	Fails on 1 line	NA	2 fail to close	Occurs on demand	Close on demand	Trip on demand
66	Occurs on demand	Fails on 2 lines	NA	Close on demand	Occurs on demand	Close on demand	Trip on demand
67	Occurs on demand	Fails on 2 lines	NA	Close on demand	Occurs on demand	Close on demand	Trip on demand
68	Occurs on demand	Fails on 2 lines	NA	Close on demand	Occurs on demand	Close on demand	Trip on demand
69	Occurs on demand	Fails on 2 lines	NA	Close on demand	Occurs on demand	Close on demand	Trip on demand
70	Occurs on demand	Fails on 2 lines	NA	Close on demand	Occurs on demand	Close on demand	Trip on demand
71	Occurs on demand	Fails on 2 lines	NA	Close on demand	Occurs on demand	Close on demand	Trip on demand
72	Occurs on demand	Fails on 2 lines	NA	Close on demand	Occurs on demand	Close on demand	Trip on demand
73	Occurs on demand	Fails on 2 lines	NA	Close on demand	Occurs on demand	Close on demand	Trip on demand
74	Occurs on demand	Fails on 2 lines	NA	Close on demand	Occurs on demand	1 fails to close	Fail to trip
75	Occurs on demand	Fails on 2 lines	NA	1 fails to close	Occurs on demand	Close on demand	Trip on demand

TABLE 9. (continued)

NO	MSIV CLOSE	AFW ACT	AFW AUTO	AFW SG ISO	HP INJ	THR HPI CH	THR AFW
51	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Failure to throttle	Failure to throttle
52	NA	Actuates on demand	Overfeed	NA	Occurs on demand	As required	Prior to high level alarm
53	NA	Actuates on demand	Overfeed	NA	Occurs on demand	As required	Failure to throttle
54	NA	Actuates on demand	Overfeed	NA	Occurs on demand	Failure to throttle	Prior to high level alarm
55	NA	Actuates on demand	Overfeed	NA	Occurs on demand	Failure to throttle	Failure to throttle
56	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	As required	Prior to high level alarm
57	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	As required	Prior to high level alarm
58	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	As required	Failure to throttle
59	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Failure to throttle	Prior to high level alarm
60	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Failure to throttle	Failure to throttle
61	NA	Actuates on demand	Overfeed	NA	Occurs on demand	As required	Prior to high level alarm
62	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	As required	Prior to high level alarm
63	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	As required	Failure to throttle
64	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Failure to throttle	Prior to high level alarm
65	NA	Actuates on demand	Overfeed	NA	Occurs on demand	As required	Prior to high level alarm
66	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	As required	Prior to high level alarm
67	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	As required	Failure to throttle
68	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Failure to throttle	Prior to high level alarm
69	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Failure to throttle	Failure to throttle
70	NA	Actuates on demand	Overfeed	NA	Occurs on demand	As required	Prior to high level alarm
71	NA	Actuates on demand	Overfeed	NA	Occurs on demand	As required	Failure to throttle
72	NA	Actuates on demand	Overfeed	NA	Occurs on demand	Failure to throttle	Prior to high level alarm
73	NA	Actuates on demand	Overfeed	NA	Occurs on demand	Failure to throttle	Failure to throttle
74	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	As required	Prior to high level alarm
75	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	As required	Prior to high level alarm

TABLE 9. (continued)

NO	TURB TRIPS	FW REG VLV	PORV CLOSE	SDV CLOSE	SI SGL GEN	MFIV CLOSE	MFW PMP TP
76	Occurs on demand	Fails on 2 lines	NA	1 fails to close	Occurs on demand	Close on demand	Trip on demand
77	Occurs on demand	Fails on 2 lines	NA	1 fails to close	Occurs on demand	Close on demand	Trip on demand
78	Occurs on demand	Fails on 2 lines	NA	1 fails to close	Occurs on demand	Close on demand	Trip on demand
79	Occurs on demand	Fails on 2 lines	NA	2 fail to close	Occurs on demand	Close on demand	Trip on demand
80	Occurs on demand	Fails on 2 lines	NA	2 fail to close	Occurs on demand	Close on demand	Trip on demand
81	Occurs on demand	Fails on 3 lines	NA	Close on demand	Occurs on demand	Close on demand	Trip on demand
82	Occurs on demand	Fails on 3 lines	NA	Close on demand	Occurs on demand	Close on demand	Trip on demand
83	Occurs on demand	Fails on 3 lines	NA	Close on demand	Occurs on demand	Close on demand	Trip on demand
84	Occurs on demand	Fails on 3 lines	NA	Close on demand	Occurs on demand	Close on demand	Trip on demand
85	Occurs on demand	Fails on 3 lines	NA	Close on demand	Occurs on demand	Close on demand	Trip on demand
86	Occurs on demand	Fails on 3 lines	NA	1 fails to close	Occurs on demand	Close on demand	Trip on demand
87	Occurs on demand	Fails on 3 lines	NA	2 fail to close	Occurs on demand	Close on demand	Trip on demand

TABLE 9. (continued)

NO	MSIV CLOSE	AFW ACT	AFW AUTO	AFW SG ISO	HP INJ	THR HPI CH	THR AFW
76	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	As required	Failure to throttle
77	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Failure to throttle	Prior to high level alarm
78	NA	Actuates on demand	Overfeed	NA	Occurs on demand	As required	Prior to high level alarm
79	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	As required	Prior to high level alarm
80	NA	Actuates on demand	Overfeed	NA	Occurs on demand	As required	Prior to high level alarm
81	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	As required	Prior to high level alarm
82	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	As required	Failure to throttle
83	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Failure to throttle	Prior to high level alarm
84	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	Failure to throttle	Failure to throttle
85	NA	Actuates on demand	Overfeed	NA	Occurs on demand	As required	Prior to high level alarm
86	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	As required	Prior to high level alarm
87	NA	Actuates on demand	Auto controlled	NA	Occurs on demand	As required	Prior to high level alarm

TABLE 10. SEQUENCE DESCRIPTIONS--STEAM GENERATOR TUBE RUPTURE AT HOT STANDBY

All sequences below are initiated by the rupture of a single tube in Steam Generator A. The rupture occurs at the tubesheet on the cold-leg side with the reactor at hot standby conditions.

Sequence 1 (Base case, Scenario 13 in Reference 1)

The operator responds as follows:

1. If SIAS signal is generated, the operator will trip the reactor coolant pumps when RCS pressure reaches 1300 psig.

2. The operator will throttle AFW flow to maintain 40% SG level.

3. At 500 s the operator closes the affected steam generator MSIV.

4. At 10 minutes the operator fully opens 3 steam dumps and cools down the primary system until 45°F subcooling is attained. Subcooling is measured using the core outlet temperature and saturation temperature in the affected SG secondary.

5. When subcooling is attained, close the steam dumps.

6. Wait 260 s after event 5, then open one pressurizer PORV and depressurize the primary system.

7. When the pressurizer and affected steam generator steam dome pressures have equalized, close the PORV.

8. Wait 500 s after event 7, then open one pressurizer PORV and depressurize the primary system to 1000 psia. Close the PORV.

9. Wait 100 s after event 8, then secure HPI.

Sequence 2

Same as Sequence 1, except the steam dumps fail to close for 10 min after the subcooling requirement is met.

Sequence 3

Same as Sequence 1, except the pressurizer PORV sticks open for 10 min on first opening.

TABLE 10. (continued)

Sequence 4

Same as Sequence 1, except the second pressurizer PORV opening does not occur, and HPI and charging are throttled when pressurizer setpoint level is attained.

Sequence 5

Same as Sequence 4, except the HPI and charging are not throttled.

TABLE 11. SEQUENCE DESCRIPTIONS--ISOLATABLE SBLOCA AT POWER

Sequence 1 below is identical to Scenario 7 as reported in Reference 1. This scenario involves a stuck-open pressurizer PORV at full-power conditions which the operator isolates at 10 minutes into the transient. Sequences 2 through 4 below are defined as changes to Sequence 1.

1. Scenario 7 as previously defined by ORNL.[1]

2. Isolate at 10 minutes, charging fails to throttle.

3. Isolate at 20 minutes.

4. Isolate at 20 minutes, charging fails to throttle.

TABLE 12. SEQUENCE DESCRIPTIONS--ISOLATABLE MBLOCA AT POWER

The sequences below are based on Scenario 6 as presented in Reference 1. This scenario involved a 0.0635-m (2.5-in.) diameter hot leg break at full power which was not isolated. Sequences 1 through 4 below are defined as changes to Scenario 6.

1. Isolate at 10 minutes, charging fails to throttle.

2. Isolate at 10 minutes, charging throttled 3 minutes after pressurizer level setpoint reached.

3. Isolate at 20 minutes, charging fails to throttle.

4. Isolate at 20 minutes, charging throttled 3 minutes after pressurizer level setpoint reached.

to terminate the loss of primary system inventory. In the case of an "isolatable small break LOCA," for example, it is assumed that the operator closes the PORV block valve, thus terminating the primary-side break.

Each sequence is defined by the initial power level, the initiating event, and subsequent hardware and operator actions as defined in Tables 1 through 12. Column headings in these tables are abbreviations for events; Table 13 (for steam line break sequences) and Table 14 (for all other sequences) provide more detailed descriptions of the events. To simplify the identification of sequences, the following numbering scheme was adopted. Sequences are identified by two numbers: first, the group number, and second, the line number within the group. Sequence 1-4, for example, is the fourth sequence listed in Table 1. Sequence 1-4 represents a small break LOCA during full-power operation, with a hardware failure of the auxiliary feedwater (AFW) flow controller and the failure of the operators to manually throttle AFW based on steam generator levels.

For each sequence, ORNL required for its fracture mechanics analyses a reactor vessel downcomer fluid pressure and temperature history for the two-hour period following the initiating event, an indication of a representative heat transfer coefficient on the reactor vessel wall, and an estimate of uncertainties in the pressure and temperature histories. This report documents both the methods used to provide this information and the results.

TABLE 13. STEAM LINE BREAK EVENT DESCRIPTIONS

Heading	Description	Entries	Description
CKVL RP CL	Check valve on ruptured steam line closes on demand.	Closes on demand	Check valve on ruptured steam line closes to prevent backflow from remaining steam lines.
		Fails to close	Check valve on ruptured steam line fails to prevent backflow from remaining steam lines.
MSIV RP CL	MSIV on ruptured steam line closes on demand (not required for small break).	NA	Closure not required if check valve operates properly.
		Closes on demand	MSIV on ruptured steam line closes on demand; required if check valve fails to close.
RM MSIV CL	MSIVs on unaffected steam lines close on demand (not required for small break).	Close on demand	Remaining 2 MSIVs close on demand.
		1 fails to close	MSIV on 1 line fails to close.
		2 fail to close	MSIVs on 2 lines fail to close.
SDV CLOSE	Steam dump valves (SDVs) open on turbine trip and close when required. Turbine trip assumed to occur as part of the initiating event.	Close on demand	All 5 SDVs close when required.
		1 fails to close	1 SDV fails to close when required.
		2 fail to close	2 SDVs fail to close when required.
		3 fail to close	3 SDVs fail to close when required. Sequences involving 3 or more SDV failures coupled with 1 or more FW reg. valve failures were not considered.
		5 fail to close	5 SDVs fail to close when required. Since the failure probability for failure of 4 SDVs is the same, it was not considered as a separate sequence.
MFW PMP TP	MFW pumps trip on demand.	Trip on demand	Both MFW pumps trip on demand.
		Fails to trip	MFW pumps fail to trip on demand. Failure must be coupled with at least 1 MFIV failing to close.
AFW ACT	AFW actuates on demand.	Actuates on demand	AFW always actuates on demand and delivers minimum required flow.

TABLE 13. (continued)

Heading	Description	Entries	Description
AFW AUTO	AFW auto control.	Auto controlled	AFW auto control operates as required.
		Overfeed	AFW auto control fails resulting in steam generator overfeed at maximum rate.
AFW SG ISO	AFW isolation to affected steam generator.	Occurs as required	Operator isolates affected steam generator as required.
		Not isolated	Operator fails to isolate affected steam generator as required.
THR AFW	AFW throttled.	Prior to high level alarm	Operator throttles AFW prior to steam generator high level alarm.
		Failure to throttle	Operator fails to throttle AFW. For failure coupled with failure to throttle HPI and charging pumps, if the coupled failure leads to a sequence with a failure probability $\leq 10^{-7}$, the sequence was not considered.
HP INJ	HPI occurs on demand.	Occurs on demand	HPI occurs as required, given that SI signal has been generated.
THR HPI CH	HPI and charging pumps throttled.	As required	Operator throttles HPI and charging pumps to prevent repressurization.
		Failure to throttle	Operator fails to throttle HPI and charging pumps. For failure coupled with failure to throttle AFW, if the coupled failure leads to a sequence with a failure probability $\leq 10^{-7}$, the sequence was not considered.
FW REG VLV	FW reg. valves runback on turbine trip. Not included for sequences initiated at hot standby. Turbine trip assumed to occur as part of the initiating event.	Occurs on demand	FW reg. valves runback and modulate as required on 3 lines.
		Fails on 1 line	FW reg. valve remains open on 1 line. Sequences involving 1 or more FW reg. valve failures coupled with 3 or more SDV failures were not considered.
		Fails on 2 lines	FW reg. valve remains open on 2 lines.
		Fails on 3 lines	FW reg. valves remains open on 3 lines.
SI SGL GEN	SI signal generated when required.	Generated on demand	SI signal always generated when required, based on system response to initiator.

TABLE 13. (continued)

Heading	Description	Entries	Description
MFIV CLOSE	MFIVs close on demand.	Close on demand	All close on demand, given that SI signal has been generated.
		1 fails to close	1 MFIV fails to close on demand. Failure must be coupled with failure of MFW pumps to trip. Failure of MFIVs to close not considered when FW reg. valve runback occurs successfully.
RM PORV CL	PORVs open on turbine trip (>70% power) and those on the two unaffected steam lines close when required. Not included for sequences initiated at hot standby. Turbine trip assumed to occur as part of the initiating event.	Close on demand	Remaining 2 PORVs close on demand if opened.
		NA	Not required. If 1 or more FW reg. valves fail to runback, PORVs are not demanded. PORV failures are not significant if one or more SDVs fail to close.
		Fails on 1 line	PORV on 1 line fails to close.
		Fails on 2 lines	PORVs on 2 lines fail to close.

TABLE 14. REACTOR TRIP AND LOCA EVENT DESCRIPTIONS

Heading	Description	Entries	Description
TURB TRIPS	Turbine trips on demand. Not included for sequences initiated at hot standby.	Occurs on demand	Turbine always trips on demand. Steam side PORVs open (if trip is from >70% power) and SDVs open.
FW REG VLV	FW reg. valves runback on turbine trip. Not included for sequences initiated at hot standby.	Occurs on demand	FW reg. valves runback and modulate as required on 3 lines.
		Fails on 1 line	FW reg. valve remains open on 1 line. Sequences involving 1 or more FW reg. valve failures coupled with 3 or more SDV failures were not considered.
		Fails on 2 lines	FW reg. valve remains open on 2 lines.
		Fails on 3 lines	FW reg. valve remains open on 3 lines.
SDV CLOSE	Steam dump valves (SDVs) open on turbine trip and close when required.	Close on demand	All 5 SDVs close when required.
		1 fails to close	1 SDV fails to close when required.
		2 fail to close	2 SDVs fail to close when required.
		3 fail to close	3 SDVs fail to close when required. Sequences involving 3 or more SDV failures coupled with 1 or more FW reg. valve failures were not considered.
		5 fail to close	5 SDVs fail to close when required. Since the failure probability for failure of 4 SDVs is the same, it was not considered as a separate sequence.
SI SGL GEN	SI signal generated when required.	NA	SI signal not expected to be generated.
		Generated on demand	SI signal always generated when required, based on system response to initiator.
MSIV CLOSE	MSIVs close on demand.	NA	MSIV closure signal not expected. If signal generated, MSIV closure occurs. Failed open PORVs will not lead to demand for MSIVs to close.
		Occurs on demand	All MSIVs close on demand. MSIV closure probability not demanded unless at least four SDVs fail to close.

TABLE 14. (continued)

Heading	Description	Entries	Description
		1 fails to close	1 MSIV fails to close on demand. If evaluation shows that MSIV closure is not demanded, this sequence can be eliminated. SDVs will be manually isolated if MSIVs fail to close.
AFW ACT	AFW actuates on demand.	NA	AFW not demanded.
		Actuates on demand	AFW always actuates on demand and delivers minimum required flow.
AFW	AFW auto control	NA	AFW auto control not required.
		Auto controlled	AFW auto control operates as required.
		Overfeed	AFW auto control fails, resulting in steam generator overfeed at maximum rate.
THR AFW	AFW throttled.	NA	Not required.
		Prior to high level alarm	Operator throttles AFW prior to steam generator high level alarm.
		Failure to throttle	Operator fails to throttle AFW. For failure coupled with failure to throttle HPI and charging pumps, if the coupled failure leads to a sequence with a failure probability $\leq 10^{-7}$, the sequence was not considered.
ACC INJ	Accumulators discharge when required. Not included for sequences initiated by reactor trip.	Occurs on demand	Accumulators always discharge when required.
RHR ACT	RHR injection occurs on demand. Not included for sequences initiated by reactor trip.	Occurs on demand	RHR injection occurs as required, given that SI signal has been generated.
PORV CLOSE	PORVs open on turbine trip (>70% power) and close when required. Not included for sequences initiated at hot standby.	NA	Not required. If 1 or more FW reg. valves fail to runback, PORVs are not demanded. PORV failures are not significant if 1 or more TBVs fail to close.
		Close on demand	All 3 PORVs close on demand.
		Fails on 1 line	PORV on 1 line fails to close.

TABLE 14. (continued)

Heading	Description	Entries	Description
		Fails on 2 lines	PORVs on 2 lines fail to close. PORVs on 3 lines failing to close is equivalent to 1 SDV failing to close.
MFIV CLOSE	MFIVs close on demand.	NA	Remain open; closure not demanded.
		Close on demand	All close on demand, given that SI signal has been generated.
		1 fails to close	1 MFIV fails to close on demand. Failure must be coupled with failure of MFW pumps to trip. Failure of MFIVs to close not considered when FW reg. valve runback occurs successfully.
MFW PMP TP	MFW pump trips on demand.	NA	MFW pumps remain running; trip not demanded.
		Trip on demand	Both MFW pumps trip on demand.
		Fail to trip	MFW pumps fail to trip on demand. Failure must be coupled with at least 1 MFIV failing to close.
AFW SG ISO	AFW isolation to affected steam generator.	NA	AFW isolation not required unless MSIVs fail to close on demand.
		Occurs as required	Operator isolates affected steam generator as required.
HP INJ	HPI occurs on demand.	NA	HPI not demanded.
		Occurs on demand	HPI occurs as required, given that SI signal has been generated.
THR HPI CH	HPI and charging pumps throttled.	NA	Not required. Not included for LOCA initiators.
		As required	Operator throttles HPI and charging pumps to prevent repressurization.
		Failure to throttle	Operator fails to throttle HPI and charging pumps. For failure coupled with failure to throttle AFW, if the coupled failure leads to a sequence with a failure probability $\leq 10^{-7}$, the sequence was not considered.

3. METHODS

This section describes the general methods used to determine the reactor vessel pressure and temperature histories for the sequences shown in Section 2. Where application of the general methods was not feasible or inappropriate, details of alternate methods used appear in Sections 4 through 13.

3.1 Regrouping of Sequences by Controlling Conditions or Phenomena

As described in Section 2, the sequences were transmitted by ORNL in 12 groups, with sequences in each group sharing a common initiating event. To determine the thermal-hydraulic sequence responses, it was convenient to regroup the sequences according to controlling thermal-hydraulic phenomena rather than by initiating event. For example, all sequences involving only a secondary-side break affecting one steam generator were regrouped together into Group A (note: ORNL groupings are designated by a number while the INEL regroupings are designated by a letter). All Group A sequences share the controlling phenomena of heat removal to a single, affected steam generator. Sequences within Group A differ in break size, power level, and minor complicating failures which do not change the controlling phenomena. Group A thus contains all sequences in Groups 6 and 8 and part of the sequences in Groups 5, 7, and 9.

Table 15 presents the regrouping of sequences by controlling thermal-hydraulic phenomena. Groups A through C include sequences with only secondary-side breaks; Groups D and E include sequences with no primary- or secondary-side breaks, Groups F and I sequences with only primary-side breaks, Groups G and H sequences with combinations of primary- and secondary-side breaks; and Group J includes steam generator tube rupture sequences.

The purpose of regrouping as just described is to organize sequences in such a way that all sequences within a group share common controlling phenomena. By developing and qualifying a specific method to determine the pressure and temperature histories for any single sequence within a group,

TABLE 15. REGROUPING OF SEQUENCES BY CONTROLLING CONDITIONS OR PHENOMENA

Group	Controlling Conditions or Phenomena	Sequences
A	Secondary-side break, 1 affected steam generator	5-1, 6-1 through 6-9, 7-1 through 7-8, 8-1 through 8-6, 9-25 through 9-32.
B	Secondary-side break, 3 symmetrically affected steam generators	7-12, 9-2 through 9-23, 9-41 through 9-47.
C	Secondary-side breaks with 2 affected steam generators or with 3 symmetrically affected steam generators	5-14, 5-15, 5-17 through 5-20, 7-9 through 7-11, 9-33 through 9-40.
D	Reactor trip from full power, no primary- or secondary-side breaks	9-1, 9-49 through 9-55.
E	Main feedwater overfill	9-56.
F	Primary-side breaks	1-1 through 1-4, 2-1 through 2-4, 3-1, 3-2, and 4-1.
G	Primary-side breaks combined with symmetric secondary-side breaks	1-5 through 1-8, 2-5 through 2-8, 3-3.
H	Primary-side breaks combined with asymmetric secondary-side breaks	1-9 through 1-12, 2-9 through 2-11.
I	Isolatable primary-side breaks	11-1 through 11-4, 12-1 through 12-4.
J	Steam generator tube ruptures	10-1 through 10-5.

TABLE 16. THERMAL-HYDRAULICALLY EQUIVALENT SEQUENCES

Sequence(s)	Equivalent Sequence	Reason(s)[a]
1-13, 1-17, 1-18	1-1	B
1-14	1-2	B
1-15	1-3	B
1-16	1-5	B
5-2 through 5-13	5-1	D
5-16	5-14	F
5-21 through 5-24	5-1	D
9-24	9-19	E
9-48, 9-66, and 9-81	9-1	B
9-57, 9-75, and 9-86	9-2	A,B
9-58 and 9-76	9-3	A,B
9-59 and 9-77	9-4	A,B
9-60	9-5	A,B
9-61 and 9-78	9-6	A,B
9-62, 9-79, and 9-87	9-9	A,B
9-63	9-10	A,B
9-64	9-11	A,B
9-65 and 9-80	9-13	A,B
9-67 and 9-82	9-49	B
9-68 and 9-83	9-50	B
9-69 and 9-84	9-51	B
9-70 and 9-85	9-52	B
9-71	9-53	B
9-72	9-54	B
9-73	9-55	B
9-74	9-56	C

a. A--Same break size and location.

B--Feedwater regulating valve failure is inconsequential following a reactor trip, unless accompanied by a feedwater isolation valve failure on the same feedline.

TABLE 16. (continued)

C--Since only one feedwater isolation valve fails open, the failure of only 1 feedwater regulating valve is consequential.

D--Sequences initiated by a stuck-open steam line PORV at full power do not result in reactor or turbine trips. For further discussion see sequence 5-1 results in Section 4.1.

E--Main steam isolation valve (MSIV) closure signal is not generated because coincident high steam line flow (in 2 out of 3 lines) and low average temperature were not encountered. "MSIV Fails To Close On Demand" is therefore inconsequential. Since the MSIV did not fail to close, the "SG ISO" is changed from "Occurs As Required" to "NA". For further discussion see sequence 9-19 results in Section 4.2.

F--"Failure to Throttle" auxiliary fedwater (AFW) is inconsequential because steam generator narrow range levels had not recovered to 40% before AFW was isolated at 10 minutes.

an analyst has thus developed a method generally applicable to all sequences within that group. Furthermore, this approach assures that sequences with the same controlling phenomena are analyzed in a consistent manner.

3.2 Identification of Equivalent Sequences

The reader may have noticed that not all sequences defined by ORNL (Tables 1 through 12) appear in the regrouped list (Table 15). The reason for this is that, during the process of regrouping, many sequences were identified as being thermal-hydraulically equivalent to other sequences. In cases where this occurred, only one of the sequences in an equivalent set is shown in Table 15. For completeness, however, plotted results in Appendix A are given for every sequence in Tables 1 through 12. Plotted results for all sequences in an equivalent set were identical. Table 16 lists the equivalent sequences and the reasons for equivalency.

3.3 Application of Models

Sequences begin from the steady plant conditions associated with either full-power or hot standby operation. When the initiating event occurs, the plant experiences a transient defined by: (1) the initiating event, (2) operator or hardware failures specified in the sequence description, and (3) automatic and operator plant actions encountered as a result of changes in conditions due to (1) and (2). Such transients generally include an early phase, during which a complicated scenario of operator and automatic actions occur, and a late phase, during which such actions have ceased and the thermal-hydraulic plant conditions are determined by relatively simple thermal-hydraulic processes.

During an event with one or more steam dump valves failing open, for example, the initiating event would be expected to cause a rapid succession of events such as: reactor and turbine trips, safety injection and auxiliary feedwater initiation, main feedtrain isolation, letdown isolation, termination of pressurizer heater power, an increase in makeup

flow, and reactor coolant pump trip. After these events, however, the plant conditions are controlled by the relatively stable processes of core heat addition, natural circulation loop flow, and heat removal to the generators. Depending on the severity of the initiating event and subsequent failures, this later stable phase typically begins at from 5 to 20 minutes after the initiating event.

Note that the term "stable phase" does not mean "steady phase". In the above example, factors contributing to non-steady behavior during the stable phase include: continually decreasing core decay heat, secondary system pressure, and stored energy in metal components.

A limited number of sequences were previously investigated, using a detailed RELAP5 thermal-hydraulic and control system model of the HBR-2 pressurized water reactor (PWR).[1] Investigating all sequences in Table 15 over the two-hour periods following initiating events was not economically feasible using this method. The method generally used for this purpose was to apply the detailed RELAP5 model during the early stage and a simplified RELAP5 model during the later stage of a sequence. Using this approach, the detailed model calculation (1) defined plant response during the complicated early phase of a transient, (2) allowed confirmation that a stable stage had been reached, (3) defined the starting conditions for a simplified model calculation over the stable stage, (4) provided an understanding of the stable stage controlling phenomena that was required to assemble a valid simplified model, and (5) provided results against which the results from the simplified model could be compared and qualified. Once qualified, a simplified model calculation was used to determine the thermal-hydraulic response over the later stage of the sequence. For analyses presented in this report, the simplified models typically ran faster than the detailed models by a factor of about 200. Comparisons between detailed and simplified model calculation results show agreement adequate to justify applying the simplified model over the later stages of the sequences. As will be shown in Section 4, agreement ranged from fair for calculations of primary-side breaks to excellent for

secondary-side breaks. The stable portions of the sequences were therefore controlled by quasi-steady simple mass and energy balances, which could be well-represented with the simplified model.

The following subsections document the basic detailed and simplified models and variations of the simplified models used in these analyses.

Calculations with the simplified model were performed using an updated version of the RELAP5/MOD1.6, Cycle 18 computer code. Detailed model calculations were performed using this version, as well as RELAP5/MOD2.

3.3.1 Description of Detailed Model

This section describes the base detailed RELAP5 HBR-2 PWR model. Subsections following describe the thermal-hydraulic and control system components of the model.

The detailed model was quality-assured in four ways. First, the development of each model component was documented on worksheets which include references to the plant documents supporting the development. Second, the worksheets were independently reviewed by an analyst other than the one who developed them. Third, utility analysts, already familiar with design and modeling of the plant, reviewed both the model at various stages of completion and the calculational results. Fourth, the simulation of an actual plant transient was performed, and the completed model and results were compared with measured plant data. The comparison appears in Section 3 of Reference 1.

Thermal-Hydraulic Model

The detailed RELAP5 model is a representation of the HBR-2 plant, describing all the major flow paths for both primary and secondary systems, including the main steam and feedwater systems. Also modeled are primary and secondary power-operated relief valves (PORVs), and safety valves. The

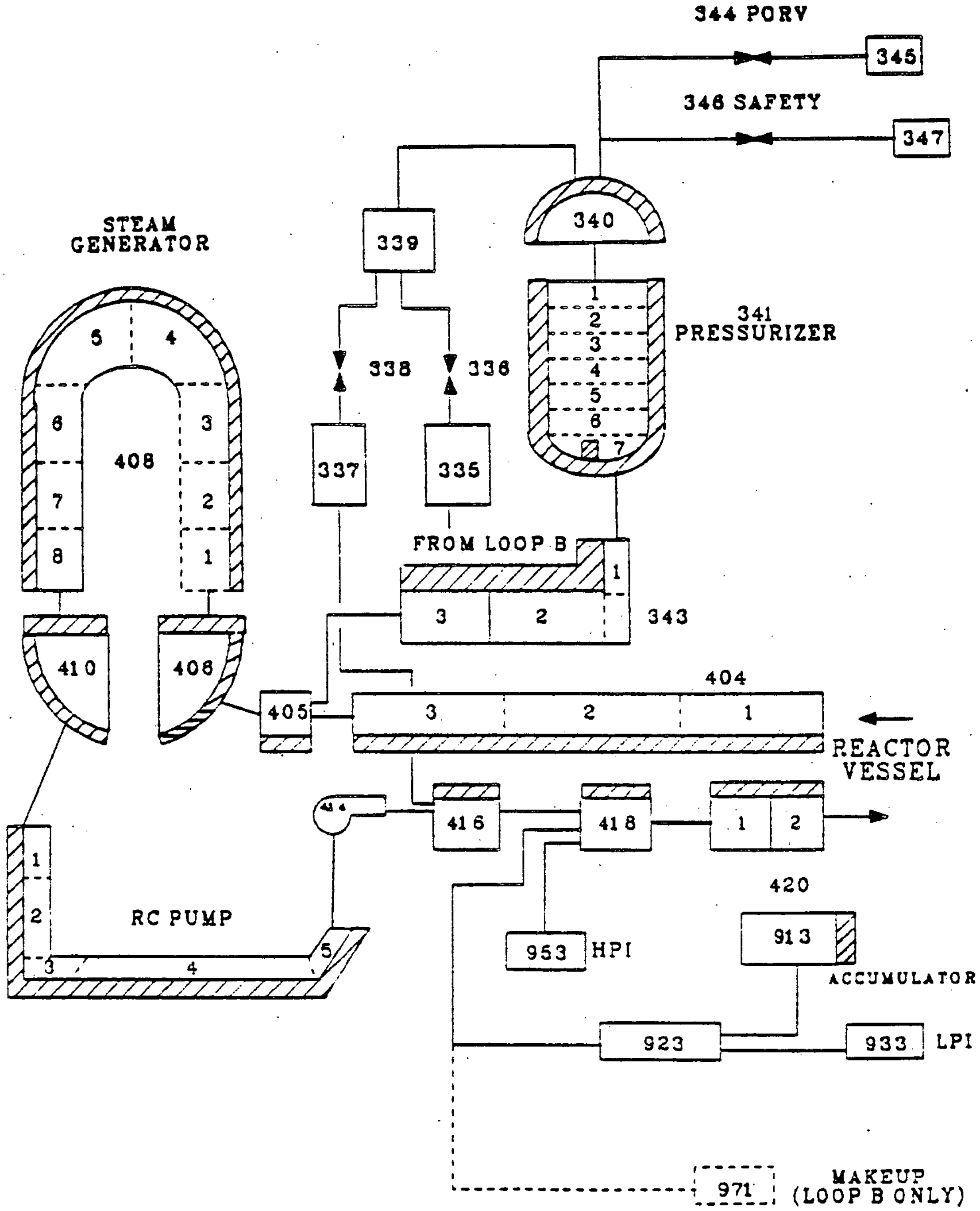


Figure 1. Detailed model nodalization of primary coolant loops (loop C shown).

emergency core cooling system (ECCS) is included in modeling the primary-side, and the auxiliary feedwater system is included in the secondary-side modeling. The model contains 224 volumes, 242 junctions and 218 heat structures. Descriptions of the primary and the secondary systems are presented in the following paragraphs.

Each loop of the HBR-2 plant's three primary coolant loops is represented explicitly in the detailed RELAP5 model. The loops are designated as A, B, and C. Each modeled loop contains a hot leg, U-tube steam generator, pump suction leg, pump, and cold leg as shown in Figure 1. The pressurizer is attached to the C loop, and the pressurizer spray lines are attached to the B and C loop cold legs. A low-pressure injection (LPI) port, an accumulator with its associated piping and a high-pressure injection (HPI) port are attached to each cold leg. The LPI and HPI models are set up to inject one-third of the total HPI and LPI flow into each loop. Also attached to the Loop B cold leg is the chemical and volume control system (CVCS). Makeup and letdown are modeled with a single junction. Heat structures are connected to each volume in the primary loops to represent the metal masses of the piping and steam generator tubes. Heat structures are also used to represent the pressurizer proportional and back-up heaters.

Figure 2 shows the RELAP5 nodalization used to represent the HBR-2 vessel. Represented in the model are the downcomer, downcomer bypass, lower plenum, core, upper plenum and upper head. The following leakage paths are represented in the vessel model: downcomer to upper head, downcomer to downcomer bypass, downcomer bypass to lower plenum, cold leg inlet annulus to upper plenum, and upper plenum to the upper head via the guide tube. Heat structures represent the core rods as well as the external and internal metal mass of the vessel. Decay heat is assumed to be at the ANS standard rate.

There are 130 volumes associated with the primary loops and 33 volumes associated with the vessel.

The detailed RELAP5 HBR-2 PWR secondary system model is shown in Figures 3 and 4. The steam generator secondary model, shown in

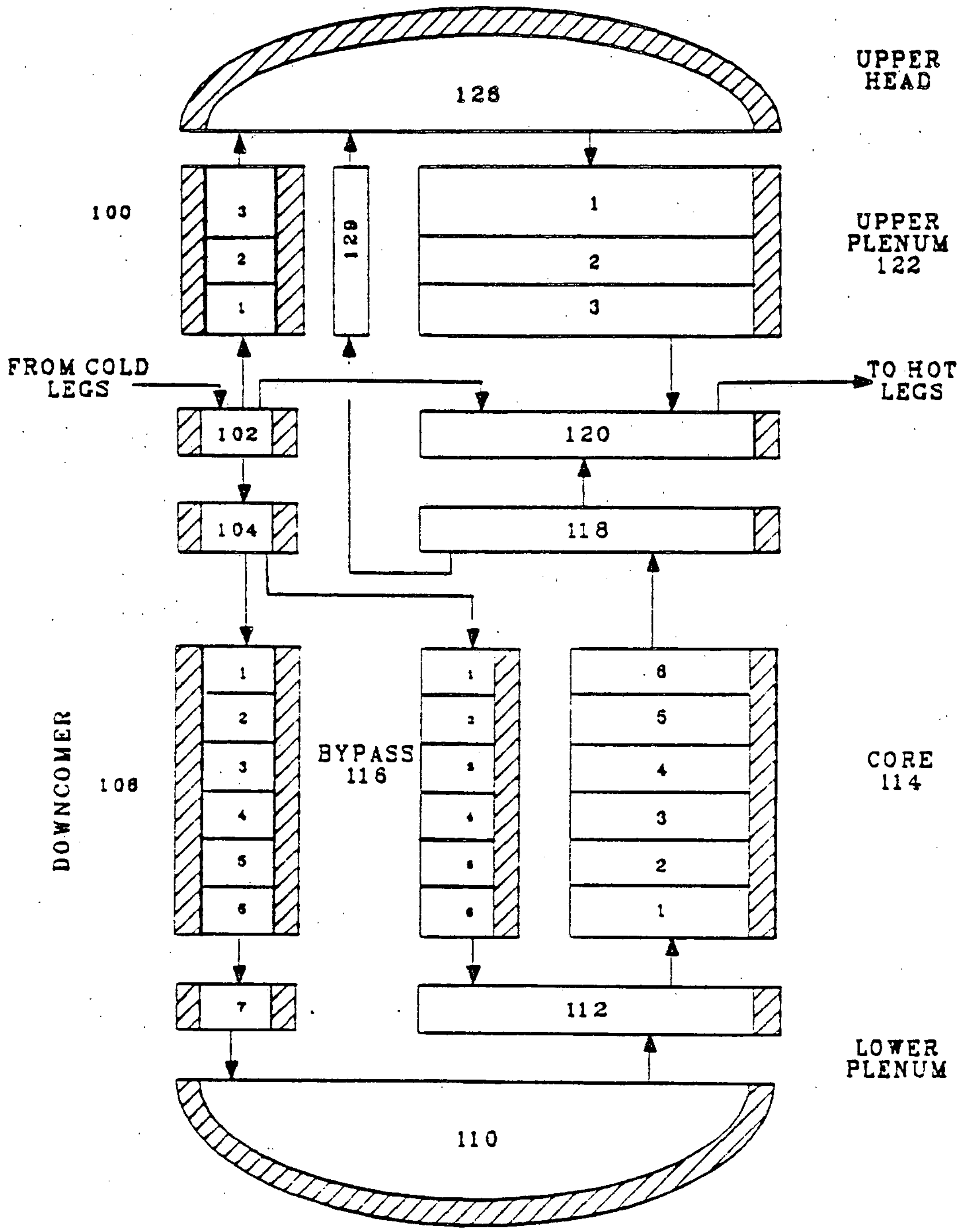
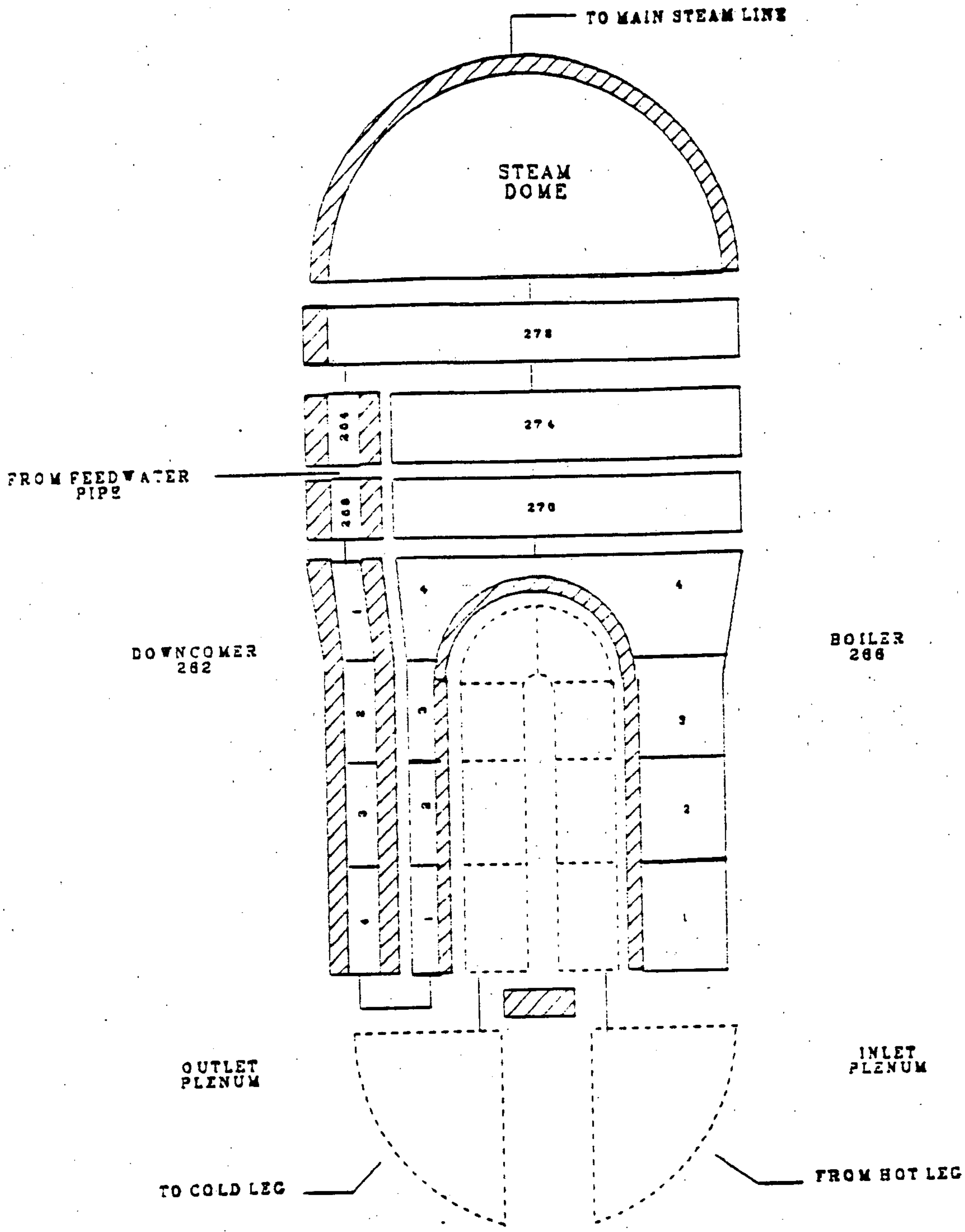


Figure 2. Detailed model nodalization of reactor vessel.



Loop	Component Numbers
A	2xx
B	3xx
C	4xx

Figure 3. Detailed model nodalization of steam generator (SGA shown).

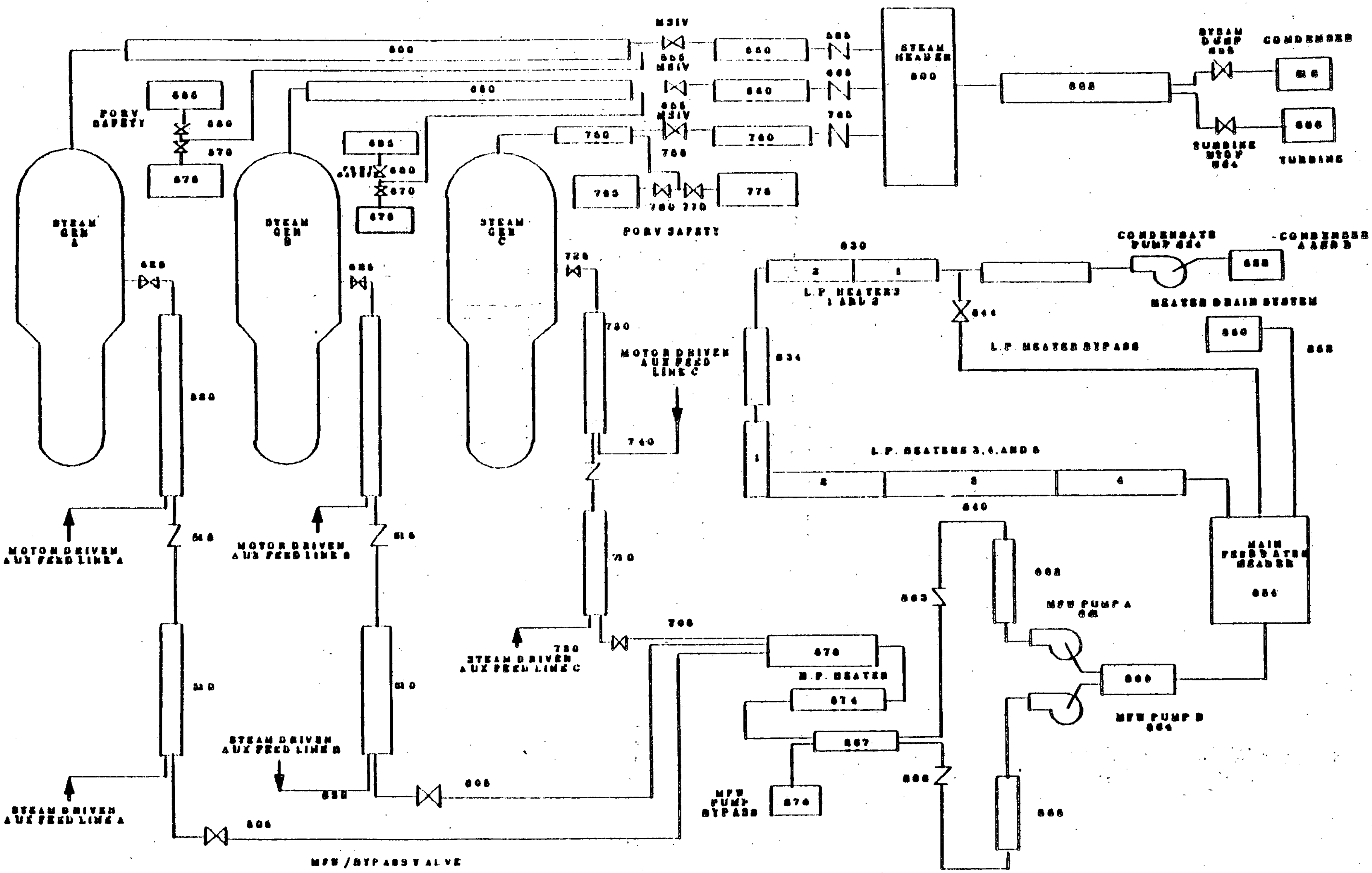


Figure 4. Detailed model nodalization of feedwater and steam systems.

Figure 3, represents the major flow paths in the secondary and includes the downcomer, boiler region, separator and dryer region, and the steam dome. Due to modeling constraints, the steam generator secondary separators and dryers were modeled within a single calculational volume. Separation in the model thus takes place at a single elevation rather than at two discrete elevations, as in the prototype steam generator. The effect of this difference is a perturbation of the flow field at the upper steam generator level tap, which affects the indicated level in a minor way. A further discussion of this effect appears in Section 3 of Reference 1.

The major flow paths of the steam line out to the turbine governor valves were modeled, and are shown in Figure 4. Each line from the steam generator secondary out to the common steam header was modeled individually and included a main steam isolation valve (MSIV), PORVs, a check valve, and safety valves. The flow restrictor was modeled in combination with the flow nozzle at the top of the steam dome. From the header to the turbine governor valves, the two steam lines in the plant were represented as one line in the model. The steam dump valve banks were modeled as one valve, with appropriate control logic to simulate the opening of each valve in the banks.

The major flow paths of the feedwater system were modeled and are shown in Figure 4. The feedwater system consists of the condensate system, main feedwater system, and the auxiliary feedwater system. The components included in modeling the condensate system were both condensate pumps, low-pressure feedwater heaters, low pressure heater bypass, heater drain system, and the main feedwater pump suction header. The condensers were modeled using a constant-pressure boundary condition. The components included in modeling the main feedwater system were both main feedwater pumps, main feedwater pump recirculation, high pressure feedwater heaters, main feedwater header tank, main feedwater/bypass valves, and piping to the steam generators, including the feedwater header ring. The auxiliary feedwater system modeling includes the motor-driven and turbine-driven systems, with a common header for each system and valves from the header to each feed line.

Heat structures for the secondary system include the internal and external metal for each of the steam generator secondaries, and the piping for both the steam and feedwater systems.

A total of 14 volumes represent the secondary for each steam generator. The steam line consists of 16 volumes, and the feedwater system contains 31 volumes.

Control System Models

The purpose of this section is to provide the reader with a general overview of the functions of the major control systems used in the detailed model. Information regarding the setpoints of the Westinghouse control systems will not be provided due to the proprietary nature of the various control system specifications. In general, the control systems were modeled as closely as possible and are considered to be good representations of the actual systems.

The steam dump control system will be described, followed by descriptions of the steam generator level control system, the pressurizer pressure control system, the pressurizer level control system, and additional systems.

Steam Dump Control System

The purpose of the steam dump control system (SDCS) is to:

1. Permit the nuclear plant to accept sudden losses of load without tripping the reactor

2. Remove stored energy and residual heat following a reactor trip and bring the plant to equilibrium no-load conditions without actuation of the steam generator safety valves

3. Permit control of the steam generator pressure at no-load conditions and permit a manually controlled cooldown of the plant.

The above tasks are accomplished by three modes of steam dump control. Requirements 1 and 2 are met by control of the primary system average fluid temperature, whereas requirement 3 is met by controlling the secondary system steam pressure. The SDCS is divided into three separate systems: Load Rejections Controller, Plant Trip Controller, and Steam Pressure Controller which will be described next.

The load rejections controller (LRC) is designed to control the primary system average temperature during periods of load rejection. Control of the primary system average temperature is performed by modulating the steam dump valves and, if the load rejection is greater than 70%, the steam line power-operated relief valves (PORVs). The turbine impulse stage pressure signal is linearly converted into the primary system average temperature setpoint. The filtered derivative of the turbine impulse stage pressure signal is used to determine whether or not a load rejection has occurred, and the size of the rejection when one does occur. Modulation of the steam dump valves is blocked if the condenser does not have sufficient vacuum, or if the primary system average temperature decreases below the minimum temperature setpoint. Other bistables exist in the real plant but were not modeled because they are not used in the various calculations presented here.

The function of the plant trip controller (PTC) is to bring the primary system average temperature down to the equilibrium no-load setpoint, 559 K (547°F) for the 2300 MW full-power case, after the turbine has been tripped. The PTC performs this function by modulating the steam dump valves. Unlike the LRC, the PTC does not have any control over the steam line relief valves. Modulation of the steam dump valves is blocked if the condenser does not have sufficient vacuum, or if the primary system average temperature decreases below the minimum temperature setpoint. Steam dump control system operation using the PTC is replaced with the steam pressure controller when the primary system average temperature is decreased to the no-load setpoint, with the additional constraint that 60 s must have expired since the plant was tripped. The 60 s delay is used to simulate the reactor operator's response time.

The steam pressure controller (SPC) is used to regulate the secondary system steam header pressure. This system is used when the plant is at no-load conditions, or to replace the PTC. The steam header pressure is controlled by modeling the steam dump valves. The setpoint pressure is 7.03 MPa (1020 psia). No modulation of the steam line PORVs is performed by this system. Modulation of the steam dump valves is blocked if the condenser does not have sufficient vacuum, or if the primary system average temperature decreases below the minimum temperature setpoint; however, unlike the LRC and PTC systems, the minimum temperature condition may be overridden by the plant operator to enable plant cooldown to cold shutdown conditions.

Steam Generator Level Control System

The steam generator level control system (SGLCS) is designed to regulate the liquid level in the steam generator (SG) downcomer. This control system uses three input signals to regulate the feedwater flow rate into each of the three steam generators. These three signals are: (1) the steam generator liquid level, (2) the steam flow rate, measured in the steam line at the SG outlet, and (3) the feedwater flow rate, measured downstream of the feedwater regulating valve. The SGLCS is used only when the plant load is above 15%.

The steam generator liquid level is determined by measuring the differential pressure between pressure taps in the SG downcomer. The liquid level is inferred from this differential pressure, and can be perturbed by events that influence these two taps in a nonsynchronous manner; for example, a main steam line break or turbine stop valve closure. The steam generator liquid level signal is compared to the setpoint level, which is a function of the turbine impulse stage pressure. The resulting error is then used as an input signal to a proportional-integral (P-I) controller.

To determine the feed-steam mismatch signal the feedwater and steam flow rate signals are compared. This signal is added to the level error signal, and the result is used as the input signal for another P-I controller. The output of this P-I controller is used to modulate the appropriate feedwater valve.

When the plant load is less than 15%, instead of using the SGLCS, the main feedwater valves are closed and feedwater control is performed manually, using the feedwater bypass valves to maintain the desired steam generator level. Additionally, one main feedwater pump and one condensate pump are used, instead of two of each, as in the full-power case.

The conditions that can result in main feedwater isolation in the plant have been incorporated into the SGLCS model. These conditions include plant trip, main feedwater pump trip, and initiation of the engineered safety features actuation signal (ESFAS).

Pressurizer Pressure Control System

The purpose of the pressurizer pressure control system (PPCS) is to maintain the desired primary system pressure. This function is performed using spray valves, relief valves, proportional heaters, and back-up heaters.

The pressurizer pressure is compared to its setpoint to determine the error. This error signal is used as the input signal to a P-I controller. The output of the P-I signal is used to control the function of both spray valves, the proportional and back-up heater source demands, and the valve area of one of the two pressurizer PORVs. The other PORV area is a function of the uncompensated pressurizer pressure signal.

The PPCS is modeled as accurately as possible and includes all the trips and setpoints in the actual plant, with two exceptions. First, the spray valves do not maintain a minimum flow as in the plant because of difficulties incurred due to thermal-hydraulic considerations. The minimum

flow requirement is imposed in the plant to maintain the spray line temperature at the temperature of the primary system cold legs. This is required to avoid the possibility of thermally stressing the spray lines when pressurizer spray is demanded. To compensate, the model spray lines were initialized at cold leg temperatures, and no heat losses from the lines to containment were considered. The second modeling exception is in the amount of power supplied to the proportional heaters during steady state operation. The heaters normally operate at 2000 kW to make up for plant heat losses and pressure decay due to the continuous minimum spray operation. Since the pressurizer tank walls were modeled as perfectly insulated heat structures, and the spray valves were completely isolated during the steady state initialization phase, this 2000 kW heater source was subtracted from the total possible proportional heater source of 4000 kW.

Pressurizer Level Control System

The purpose of the pressurizer level control system (PLCS) is to maintain the desired amount of liquid inventory in the primary coolant system. The amount of water inventory in the primary coolant system may be inferred from the liquid level in the pressurizer, which varies as a function of the primary system average coolant temperature.

The pressurizer setpoint level is a function of the primary system average coolant temperature. The setpoint level is subtracted from the actual level which is determined from a set of differential pressure taps located in the pressurizer. The pressurizer level error signal is used as the input signal in a P-I controller. The output of the P-I controller specifies the amount of change in the charging pump speed to effect the desired change in the primary system coolant inventory.

The level error signal is also used to actuate the back-up heaters when the pressurizer level error exceeds the setpoint level by 5%. Pressurizer heater demand is blocked when the pressurizer level becomes less than the low-level limit of 14%.

The PLCS is modeled to include both the reactor coolant pump seal injection contribution and the charging flow demanded by the compensated pressurizer level error signal.

Additional Control Systems

Included in the control system package are miscellaneous controllers and trips that are modeled to represent various system functions that cannot be classified in any of the aforementioned systems. These controllers perform functions such as: (1) feedwater recirculation to the condenser during periods of low feedwater demand, (2) low pressure feedwater heater bypass due to low main feedwater pump suction pressure, (3) specification of turbine impulse stage pressure as a function of steam flow rate and turbine governor valve area, and (4) control of the auxiliary feedwater systems.

3.3.2 Description of Base Simplified Model

This section describes the base simplified RELAP5 HBR-2 PWR model. The simplified model was developed primarily by combining calculational cells of the detailed model. As described in the last section, the detailed model was quality-assured in many ways. This philosophy was extended to the simplified model development; calculations supporting the combining of cells were independently checked by an analyst other than the one performing the calculations.

The simplified model was developed to address thermal-hydraulic plant phenomena during the later stages of sequences. The base model described here was specifically designed to address plant conditions where: (1) reactor and turbine trips have occurred, (2) safety injection and auxiliary feedwater flows have been initiated, and (3) reactor coolant pumps have been tripped. Furthermore, the model assumes that significant transient effects of the above actions have ceased and symmetric loop natural circulation continues.

The base model described here was generally applied, as is, to the later stages of sequences with stuck-open steam dump valves (most sequences in Group B) or small primary breaks (part of Group F). Variations on the base model, required to properly address sequences controlled by different phenomena, are discussed in the next subsection.

For sequences involving stuck-open steam dump valves, the later portions of the sequences are controlled by decay heat addition in the core, symmetric loop natural circulation, and symmetric heat removal through all steam generators. The symmetry exists because the steam dump valves are located on the portion of the steam line common to all three steam generators.

For sequences involving only a small primary break (defined in Section 2 as a single, stuck-open pressurizer PORV), the controlling phenomena during the later portions of the sequences are core heat

addition, virtually symmetric loop natural circulation, heat removal to steam generators, and energy and mass removal at the PORV. Minor loop asymmetries due to the break location are ignored.

A nodalization diagram, shown in Figure 5, of the base simplified model was developed to address the primary and secondary system mass and energy balances that control the phenomena described in the previous two paragraphs. Component 100 represents all fluid in the reactor coolant system (RCS), except for that in the pressurizer (Components 340 and 341) and that above the hot leg centerline within the reactor vessel (Component 126). Component 266 represents the fluid volume of the three steam generator secondaries. Auxiliary to the primary system are time-dependent junction Components 961, representing HPI and LPI flow as a function of primary system pressure, and 972, representing charging flow as a function of pressurizer level. The three accumulators are modeled by Component 911, and the pressurizer PORV by valve Component 344. Auxiliary to the secondary system are time dependent junction Components 534 and 538 representing motor- and turbine-driven auxiliary feedwater injection and time dependent junction Component 808 representing a secondary system steam break. The three heat structures shown represent the pressurizer heaters, the U-tubes of the three steam generators, and the core and passive heat structures. This latter heat structure represents the metal of the core, reactor vessel, loop piping, and the steam generator shells. Heat input is based on the ANS standard for decay heat and the time since reactor trip.

Auxiliary feedwater logic requires the calculation of steam generator narrow-range levels. Due to the simplicity of the model, this was not possible. Instead, results of detailed model calculations were reviewed during periods of similar plant behavior to determine the secondary masses corresponding to key, narrow-range level setpoints. The auxiliary feedwater control was then based on the mass setpoints rather than on level setpoints.

The steam break (Component 808) was represented with a time-dependent junction that calculated break steam flow based on the secondary system pressure, the break area, and homogeneous equilibrium model critical flow

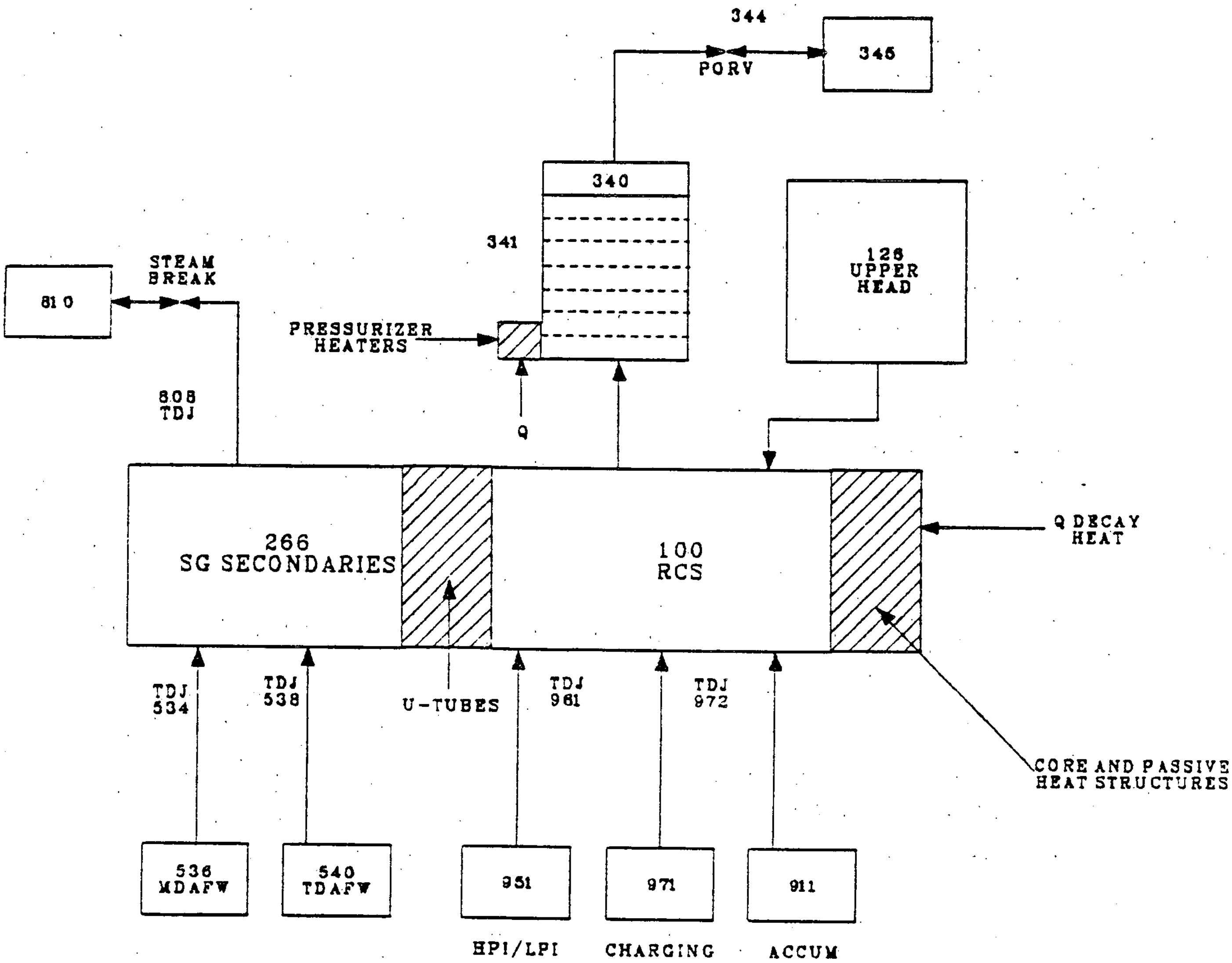


Figure 5. Base simplified model nodalization.

tables.[3] The steam generator secondaries were assumed to be at saturation conditions with inflow of subcooled auxiliary feedwater and outflow of saturated steam.

Due to the simplicity of the model, it was necessary to specify heat transfer coefficients on both sides of the U-tube heat structure. On the inside surface, the fluid in Component 100 of the simplified model is not flowing, while the fluid inside the U-tubes is flowing due to natural loop circulation. On the outside surface, the void fraction in Component 266 does not well-represent that in the lower boiler section. To circumvent these problems, constant heat transfer coefficients were specified at both locations based on representative coefficients calculated with the detailed model for similar conditions. In the same manner, a constant heat transfer coefficient was specified for the heat structure representing the core and other metal.

As was discussed earlier, initial conditions for the simplified model were derived from conditions calculated using the detailed model. As a result of qualifying the simplified model against the detailed model (results of these qualifications appear in Section 3.4), it was found that the best agreement was obtained if the RCS cell in the simplified model (Component 100) was initialized at a temperature consistent with the third U-tube cell of the detailed model (Cell 3 of Component 408 in Figure 1). The temperature of this U-tube cell generally differs from the reactor vessel downcomer fluid temperature only slightly [typically 2 K (3.6°F)]. Thus the simplified model RCS temperature generally was used directly as an indication of downcomer temperature. As will be discussed in Section 3.3.3 for the primary break sequences, natural loop circulation is slower and HPI flow higher as compared with the base model assumptions. In these cases, an alternate method (also discussed in Section 3.3.3) was used to calculate the downcomer temperature.

3.3.3 Simplified Model Variations

Variations to the base simplified model (from Section 3.3.2) were required to address various sequence-controlling phenomena; this section documents those variations.

Variation 1, Small Steam Break Affecting One Steam Generator, Reactor Coolant Pumps Tripped

For sequences involving a small break that affects only one steam generator, the base simplified model from Section 3.3.2 was first modified by reducing the size of the U-tube heat structure and the secondary volume to represent a single, affected steam generator. Next, the unaffected steam generator (USG) metal masses (but not an equivalent for the liquid masses) were added to the primary system heat structure. This was a modeling compromise to account for a slowdown in unaffected loop circulation flow over the period when the simplified model is applied. As the unaffected loop flow slows, the USGs become more loosely coupled to the primary system. This simplified model was benchmarked, for Sequence 7-4, against the detailed model; results of the benchmark comparison appear in Section 3.4.1.

Variation 2, Small Steam Break Affecting One Steam Generator, Reactor Coolant Pumps Operating

For sequences involving a small steam line break affecting a single steam generator where the reactor coolant pumps remain operating, the base simplified model from Section 3.3.2 was first modified by reducing the sizes of the U-tube heat structure and secondary volume to represent a single affected steam generator. Unlike Variation 1, in which the RCPs are tripped, the unaffected steam generators (USGs) remain closely coupled to the primary system. To account for this phenomenon, both the USG metal and liquid mass equivalents were added to the primary system heat structure. In addition, core decay heat was increased to compensate for the pump power due to continuous RCP operation, and U-tube and primary system heat structure inside surface heat transfer coefficients were increased (based on results using the detailed model) to account for forced primary system flow. This simplified model was benchmarked, for sequence 9-25, against the detailed model; results of the benchmark comparison appear in Section 3.4.2.

Variation 3, Steam Break Affecting One Steam Generator; Primary System Heat Removal Controlled by Unaffected Steam Generators

For sequences involving a steam break affecting a single steam generator but in which the unaffected steam generator heat removal is dominant, the following variation was made to the base simplified model described in Section 3.3.2. The secondary volume and U-tube heat structure were reduced in size to represent the two unaffected steam generators. This variation was typically applied over periods of a transient when, due to ASG AFW termination, the heat removal to the ASG was negligible and heat removal to the USGs was controlling. This could involve either: (1) unaffected loop natural circulation (due to very low USG temperatures), or (2) continued RCP operation, in which case the changes regarding RCP heat input and U-tube heat transfer coefficients, discussed under Variation 2, apply as well.

Variation 4, Large Steam Line Break Affecting One Steam Generator

For a large steam line break affecting only one steam generator, the unaffected loops stagnate completely early in the sequence. Due to the size of the break, the affected steam generator secondary pressure is very near atmospheric pressure. To account for the above distinctions, the base simplified model from Section 3.3.2 was first modified by reducing the sizes of the U-tube heat structure and secondary volume to represent a single affected steam generator. Next the RCS volume and its heat structure were reduced in size to remove the effects of the stagnant unaffected loop liquid and metal masses. Finally, the steam break critical flow model is not applicable because of the very low ASG secondary pressure; it was replaced by a friction-dominated steam break model. Friction flow was implemented by determining an effective loss coefficient from detailed model calculation results under similar flow conditions. This simplified model was benchmarked, for Sequence 8-4, against the detailed model. Results of the benchmark comparison appear in Section 3.4.3.

Variation 5, Small Steam Line Break Symmetrically Affecting Three Steam Generators, Reactor Coolant Pumps Operating

For sequences involving a small steam line break with three symmetrically affected steam generators, the base simplified model from Section 3.3.2 was modified by: (1) increasing the core decay heat to simulate pump heat addition, and (2) increasing the inside surface heat transfer coefficients for the U-tube and primary system heat structures, to account for forced RCS fluid circulation.

Variation 6, Steam Breaks Affecting Two Steam Generators

For sequences involving steam breaks affecting only two steam generators, the base simplified model was modified as follows. The steam generator secondary volume and the heat structure representing the U-tubes were reduced in size to correspond to two steam generators. Based on experience from the benchmarking of simplified model Variation 1 (presented in Section 3.4.1), the unaffected loop steam generator liquid and metal masses were removed from the model.

Variation 7, SBLOCA Sequences

The base simplified model was modified to represent the local effect of ECC on downcomer temperature for all the SBLOCA sequences (see Tables 1, 3, 4, and 11). The average RCS temperature was modified to compensate for both the mixing of ECC and loop flows and for the fluid transit time between the steam generator and the downcomer. Specifically, the downcomer temperature, T_D, was calculated as

$$T_D = \frac{m_{Loop}\,T + m_{ECC}\,T_{ECC}}{m_{Loop} + m_{ECC}} - \tau \frac{dT}{dt}$$

where T and T_{ECC} were the average RCS and ECC temperatures, respectively; m_{Loop} was the total flow due to natural circulation in all three loops; m_{ECC} was the sum of the HPI, LPI, accumulator and charging

flows; $\frac{dT}{dt}$ was the time derivative of the average RCS temperature; and T was the time required for the fluid to travel from the steam generator outlet to the downcomer. The loop flow, m_{Loop}, was estimated for each sequence, based on the results of a representative detailed model calculation. The transit time, T, was treated as a constant (80s). The local effect of ECC on downcomer temperature had to be estimated in the simplified model because results from the detailed model showed that the downcomer temperature could be more than 56 K (100°F) colder than the average RCS temperature. The LOCA sequences resulted in low RCS pressures and high ECC flow rates that were significant when compared to the loop flow due to natural circulation, and thus it was necessary to represent the effect of ECC mixing on downcomer temperature.

Variation 8, MBLOCA Sequences

The Variation 7 model was modified to represent MBLOCA sequences. A single junction, representing the break, and a time-dependent volume, representing the containment, were added to the Variation 7 model. The single junction connected the RCS (Component 100, Figure 5) to the time-dependent volume.

Variation 9, SBLOCA Sequences With One Stuck-Open Steam PORV

The Variation 7 model was modified to represent the asymmetric loop response resulting from one stuck-open steam PORV. First, the Variation 7 model was modified by reducing the size of the U-tube heat structure and secondary volume to represent a single affected steam generator. Second, Component 100 was modified to represent both the reactor vessel below the hot leg centerline and the single affected loop. An additional control volume was added to the model to represent the two unaffected primary loops. The additional volume was necessary because the unaffected loops respond differently than the rest of the RCS during sequences with

asymmetric steam breaks. The additional control volume accounted for flow stagnation in the unaffected loops, which results in flashing that can affect the pressure response of the RCS. Third, the primary system heat structure of the base model was modified by subtracting the metal mass of the unaffected loops. The metal in the unaffected loops would not cool as rapidly as the metal in the affected loop because the flow in the unaffected loops stagnates.

Variation 10, SBLOCA Sequences With Two Stuck-Open Steam PORVs

The Variation 7 model was modified to represent the asymmetric loop response resulting from two stuck-open steam PORVs. First, the Variation 7 model was modified by reducing the size of the U-tube heat structure and secondary volume to represent two affected steam generators. Second, Component 100 was modified to represent both the reactor vessel below the hot leg centerline and two affected loops. Third, the primary system heat structure of the base model was modified by subtracting the metal mass of the unaffected loop. The metal in the unaffected loop would not cool as rapidly as the metal in the affected loops because the flow in the unaffected loop stagnates. An additional control volume was added to the model to represent the unaffected primary loop. The additional control volume was necessary because the unaffected loop responds differently than the rest of the RCS during sequences with asymmetric steam breaks. The additional control volume accounted for flow stagnation in the unaffected loop, which results in flashing that can affect the pressure response of the RCS.

Variation 11, MBLOCAs With One Stuck-Open Steam PORV

The Variation 8 model was modified to represent the asymmetric loop response resulting from one stuck-open steam PORV. First, the model was modified by reducing the size of both the U-tube heat structure and the secondary volume, to represent a single affected steam generator. Second, Component 100 was modified to represent both the reactor vessel below the hot leg centerline and the single affected loop. The unaffected loops were modeled separately from the affected portion of the RCS because of the

asymmetric steam break. Third, the primary system heat structure of the base model was modified by subtracting the metal mass of the unaffected loops. Finally, the primary-side break was moved so that it was attached to the unaffected loops rather than the affected loop. This change more accurately represented the location of the break by allowing saturated water, rather than the highly subcooled water from the affected loop, to flow through the break.

Variation 12, MBLOCAs With Two Stuck-Open Steam PORVs

The Variation 8 model was modified to represent the asymmetric loop response resulting from two stuck-open steam PORVs. First, the model was modified by reducing the size of the U-tube heat structure and secondary volume to represent two affected steam generators. Second, Component 100 was modified to represent both the reactor vessel below the hot leg centerline and two affected loops. The unaffected loop was modeled separately from the affected portion of the RCS because of the asymmetric steam break. Finally, the primary system heat structure of the base model was modified by subtracting the metal mass of the unaffected loop.

The primary-side break was left attached to Component 100, which represents the affected portion of the RCS. This was different from the break location used in the Variation 11 model, but was justified by the assumption that the C loop contains both the primary-side break and one of the stuck-open PORVs.

Variation 13, Steam Generator Tube Rupture

For steam generator tube rupture sequences, the base simplified model was modified as follows. The existing steam generator secondary volume and tube heat structures were reduced in size to represent the two unaffected steam generators. A new volume, representing the affected steam generator secondary was connected by a break junction to the primary system volume. The break junction area was the total for both break flow paths. A loss coefficient which was consistent with the detailed model was chosen.

3.4 Simplified Model Benchmarks

This section shows the results of benchmark comparisons between simplified model and detailed model calculations of the same sequences. These comparisons lend confidence to the use of a method relying on calculations performed with a greatly simplified model.

3.4.1 Small Steam Break Affecting One Steam Generator at Hot Standby

Sequence 7-4 (see Table 7), starting from hot standby conditions, involves a single, stuck-open PORV. The detailed model (Section 3.3.1) and Variation 1 of the simplified model (Section 3.3.3) were run for this sequence over a common period from 1390 to 2490 s. Figures 6 and 7 show the results using the two models. The comparisons indicate that both the simplified model pressure and the temperature responses agree well with those of the detailed model. Differences of up to about 11 K (20°F) in the temperature comparison are believed to be caused by the inclusion of the USG metal mass in the primary system heat structure (see Section 3.3.1, Variation 1). Had the USG metal mass been deleted, as was the USG liquid mass, the comparisons would have been better. The drops in primary system pressure at about 1900 s are caused by termination of charging flow when the pressurizer setpoint level is attained. The drop in pressure when using the simplified model is smaller than when using the detailed model because the primary fluid thermal contraction rate is slightly smaller with the simplified model, and the primary system pressure drops until the HPI volumetric addition rate equals the thermal contraction rate.

3.4.2 Small Steam Break Affecting One Steam Generator at Full Power

Sequence 9-25 (see Table 9) involves a reactor trip from full power followed by a single, stuck-open steam PORV. For this sequence primary system pressure does not decline sufficiently to cause RCP trip. The detailed model (Section 3.3.1) and Variation 2 of the simplified model (Section 3.3.3) were run for this sequence over a common period from 75 to 800 s. Figures 8 and 9 show the results using the two models. The comparisons indicate that both the simplified model pressure and the

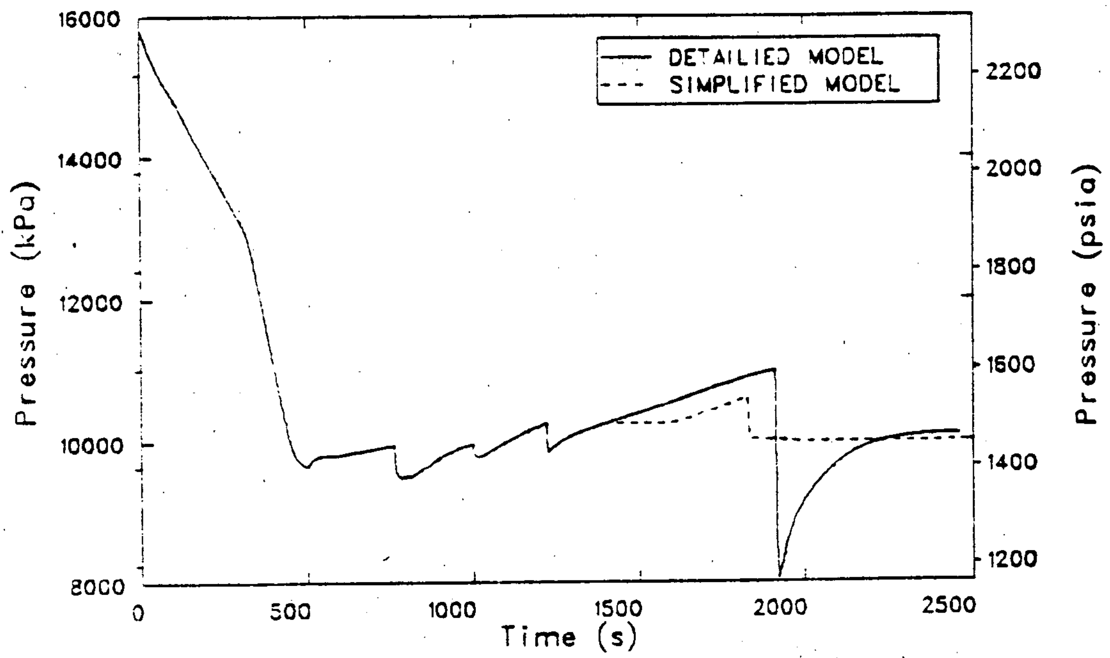


Figure 6. Sequence 7–4 simplified model pressure benchmark.

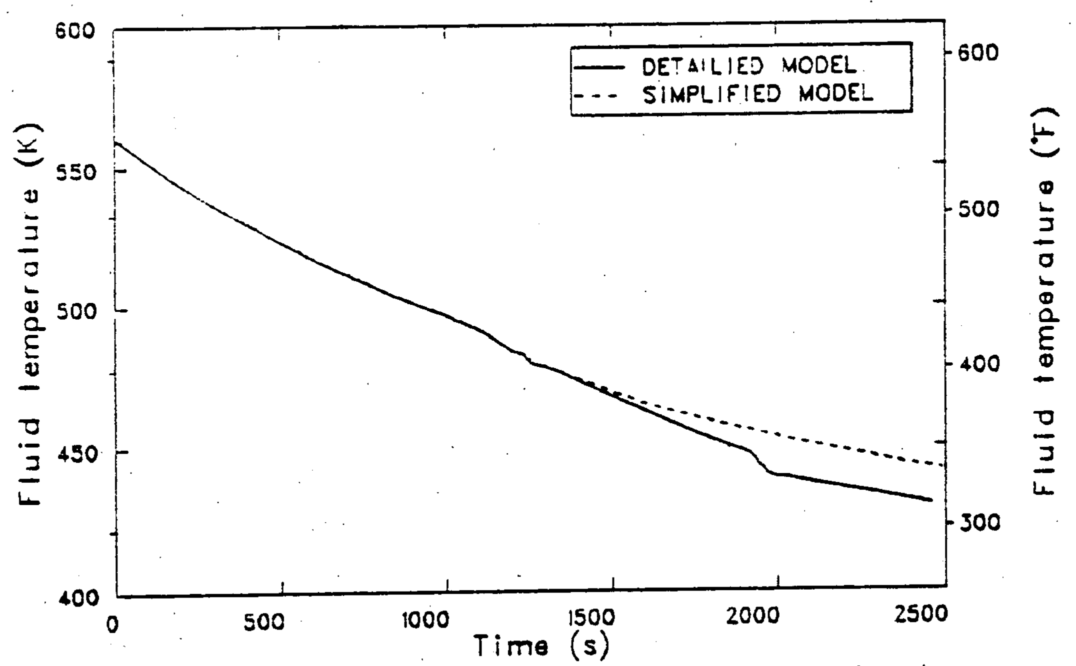


Figure 7. Sequence 7–4 simplified model temperature benchmark.

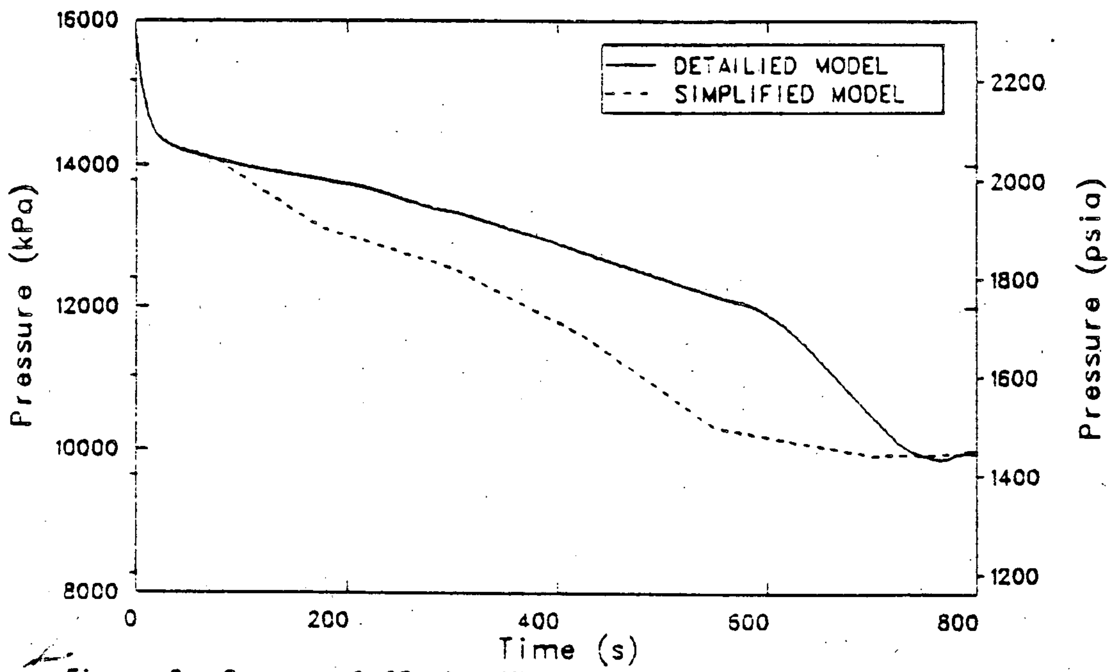


Figure 8. Sequence 9–25 simplified model pressure benchmark.

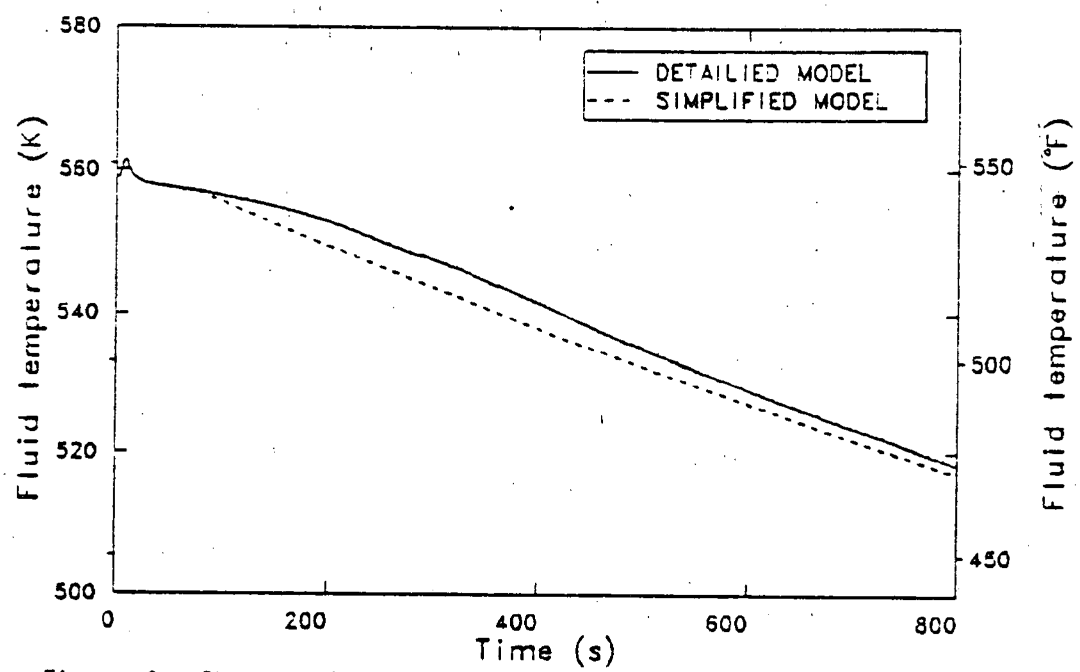


Figure 9. Sequence 9–25 simplified model temperature benchmark.

temperature responses agree well with those of the detailed model. The minor differences in the temperature responses shown in Figure 9 are caused by modeling the reactor vessel upper head as a dead-end flow path in the simplified model. In the plant, and in the detailed model (see Figure 2), minor flow paths into and out of the upper head tend to circulate upper head fluid when the RCPs are operating. With the simplified model, this circulation is not accounted for and the upper head fluid is not cooled, resulting in the colder primary system temperatures calculated with the simplified model. While the temperature differences are small [typically 4 K (7°F)], the extra fluid shrinkage due to these differences, caused the moderate discrepancy in primary system pressures shown in Figure 8.

3.4.3 Large Steam Break Affecting One Steam Generator at Hot Standby

Sequence 8-4 (see Table 8) involves a double-ended break of one steam line, starting from hot standby conditions. The detailed model (Section 3.3.1) and Variation 4 of the simplified model (Section 3.3.3) were run for this sequence over a common period from 200 to 1198 s. Figures 10 and 11 show the results using the two models. The comparisons indicate that both the simplified model pressure and the temperature responses agree well with those of the detailed model. Figure 11 indicates very good agreement for the reactor vessel downcomer temperature; the two curves differ by at most 8 K (14°F). The downcomer pressures, shown in Figure 10, are in adequate agreement with a maximum difference of 0.73 MPa (106 psi).

3.4.4 Steam Break Symmetrically Affecting Three Steam Generators at Full Power

Sequence 9-15 (see Table 9) involves the sticking open of three steam dump valves (SDVs) following a reactor trip from full-power conditions. The detailed model (Section 3.3.1) and the base simplified model (Section 3.3.2) were run for this sequence over a common period from 390 to 1790 s. Figures 12 and 13 show the results using the two models. The comparisons indicate that both the simplified model pressure and the temperature responses are in excellent agreement with those of the detailed

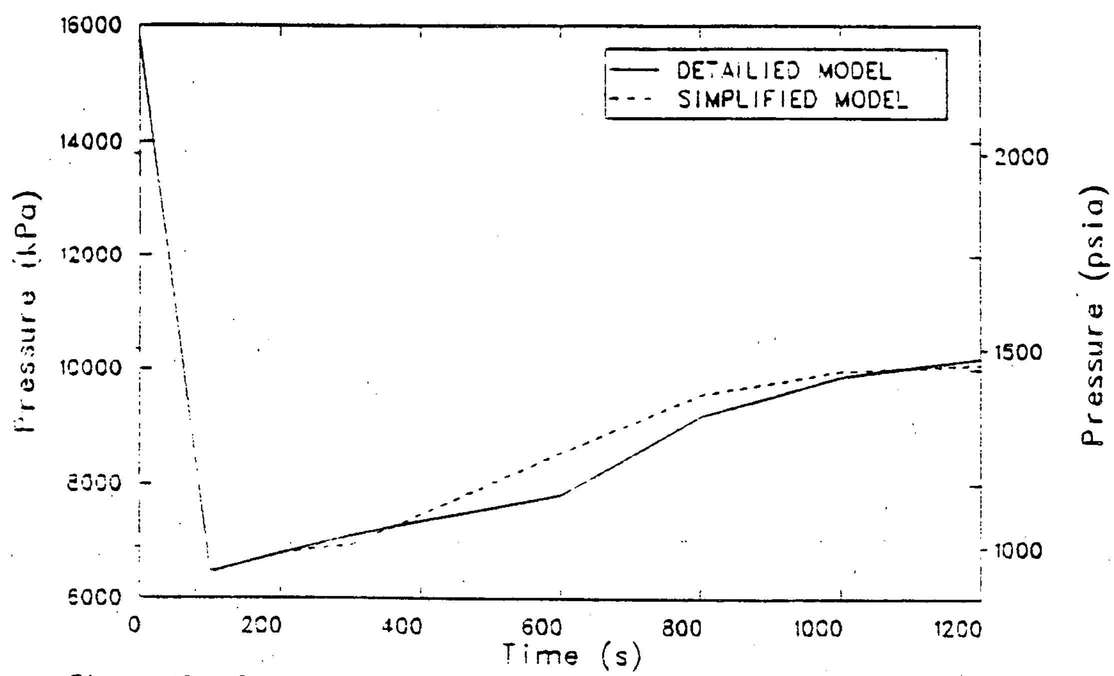


Figure 10. Sequence 8-4 simplified model pressure benchmark.

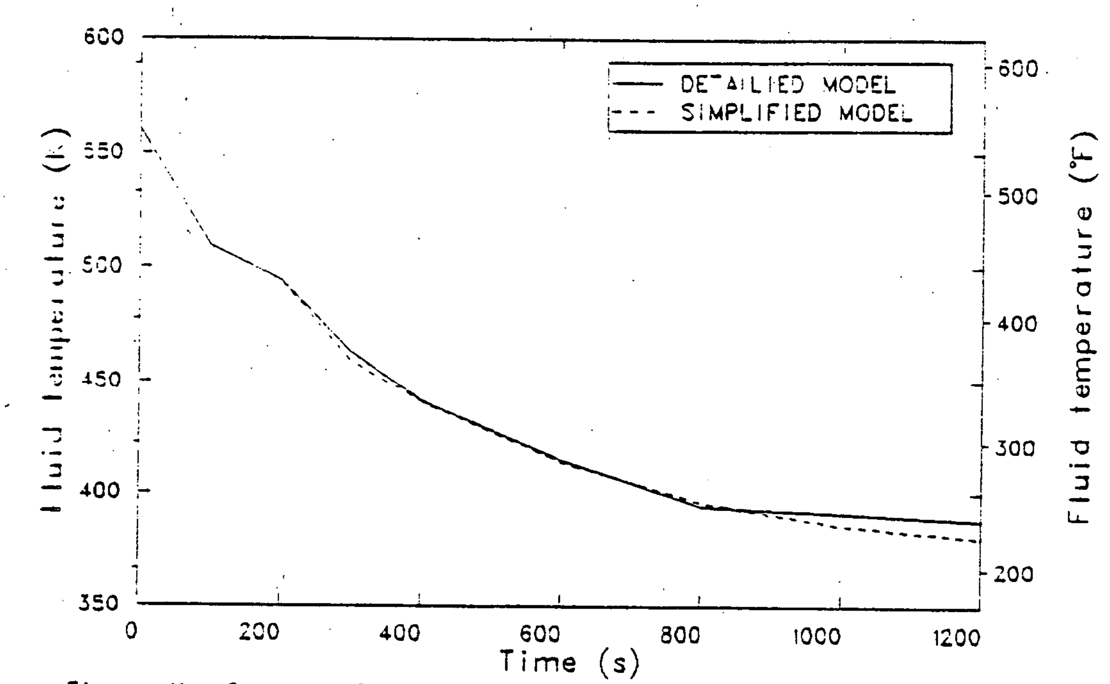


Figure 11. Sequence 8-4 simplified model temperature benchmark.

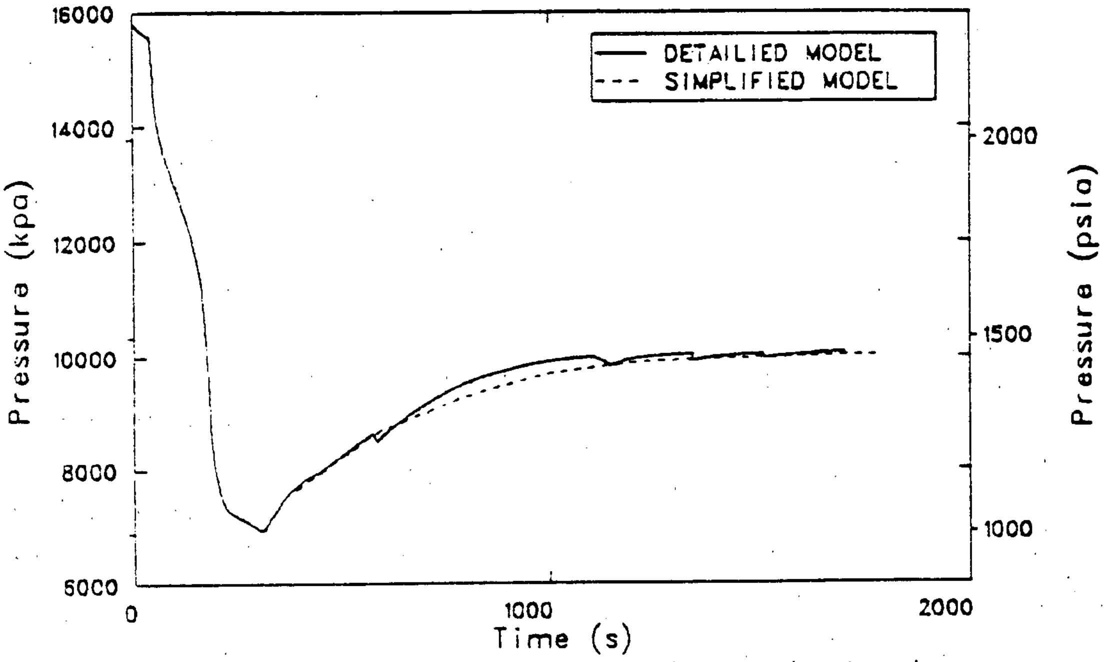


Figure 12. Sequence 9–15 simplified model pressure benchmark.

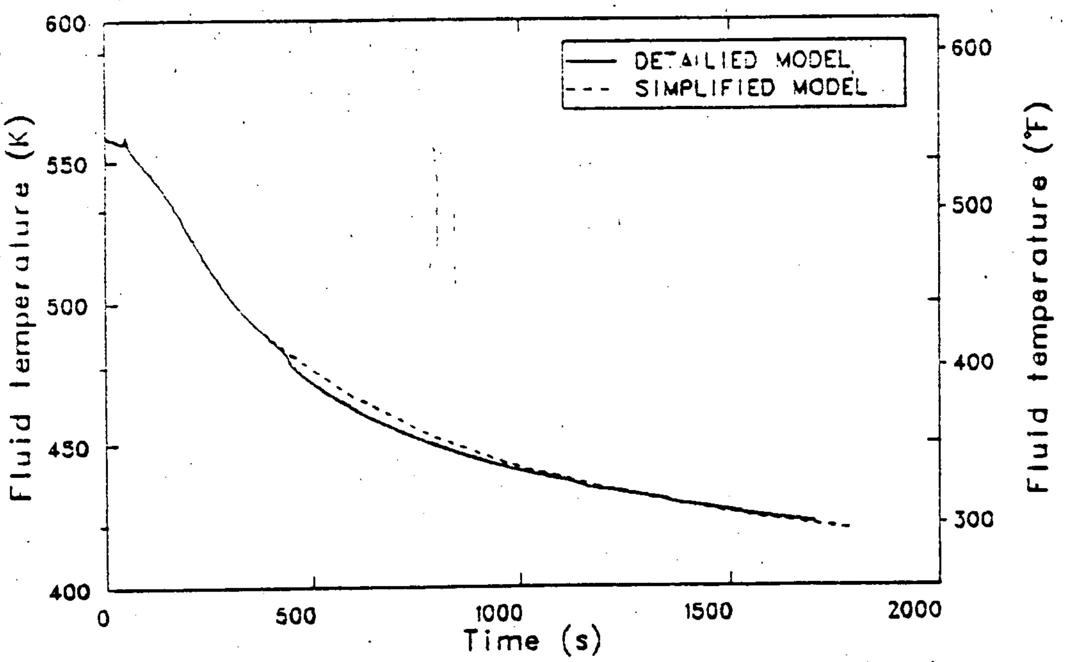


Figure 13. Sequence 9–15 simplified model temperature benchmark.

model. The pressure responses differ by at most 0.23 MPa (34 psi), and the temperature responses differ by at most 4 K (8°F). During most of the period of comparison, the agreement was much better than that indicated by these numbers.

3.4.5 SBLOCA at Hot Standby

Sequence 3-1 (see Table 3) involved a single, stuck-open pressurizer PORV at hot standby. The detailed model (Section 3.3.1) and Variation 7 of the simplified model (Section 3.3.3) were run for this sequence over a common period from 400 to 1725 s. Downcomer pressures calculated with the two models are shown in Figure 14. The pressure was relatively constant, between 400 and 1500 s. A gradual depressurization began near 1500 s in both calculations when the pressurizer filled with liquid and the flow out the PORV increased. Figures 15 and 16 show average RCS fluid temperatures and downcomer fluid temperatures, respectively. The temperature oscillations calculated with the detailed model were related to the low loop flows for this sequence. Similar oscillations were reported in Reference 1. Analysis indicated that those oscillations would not occur in the plant but, on the average, the results from the detailed model were reasonable. Figure 15 shows that the average temperature from the two models agreed closely. The calculated downcomer temperature with the simplified model (shown in Figure 16) was within the oscillations calculated with the detailed model, although the average cooldown rate was slightly smaller. The difference between the models was due to the assumption of a constant loop flow for the HPI mixing calculation in the simplified model. Although the comparison was thought to be reasonable, it illustrates the difficulty in calculating the effect of HPI mixing on downcomer temperature when using a simplified model. The results shown in Figures 14, 15, and 16 show generally good agreement between the detailed and simplified models and indicate that the simplified model can represent most of the important phenomena calculated with the detailed model.

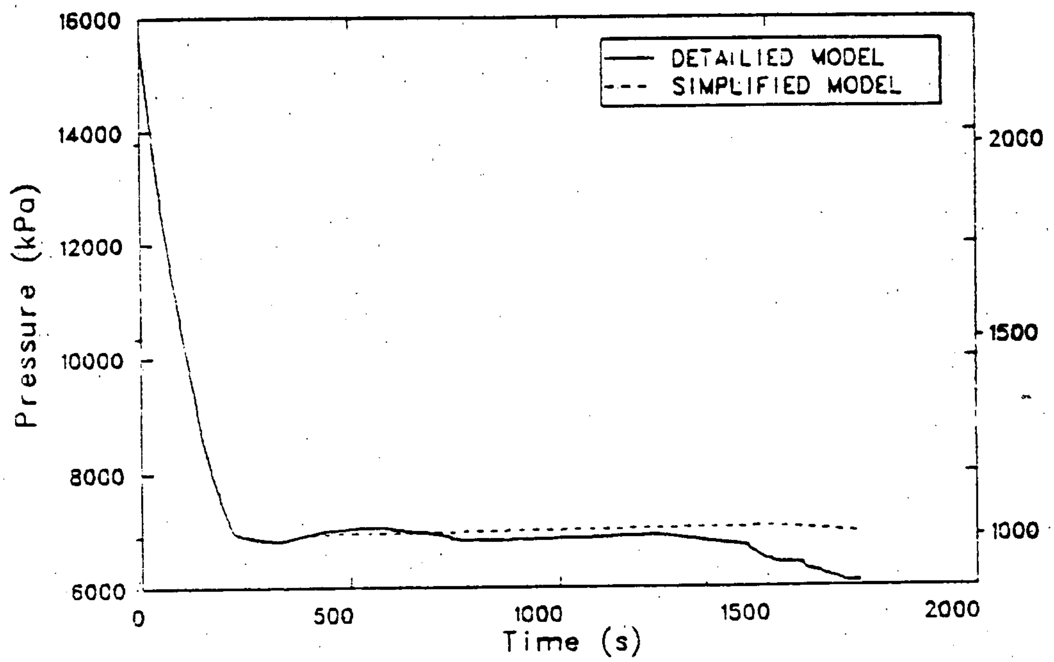


Figure 14. Sequence 3-1 simplified model downcomer pressure benchmark.

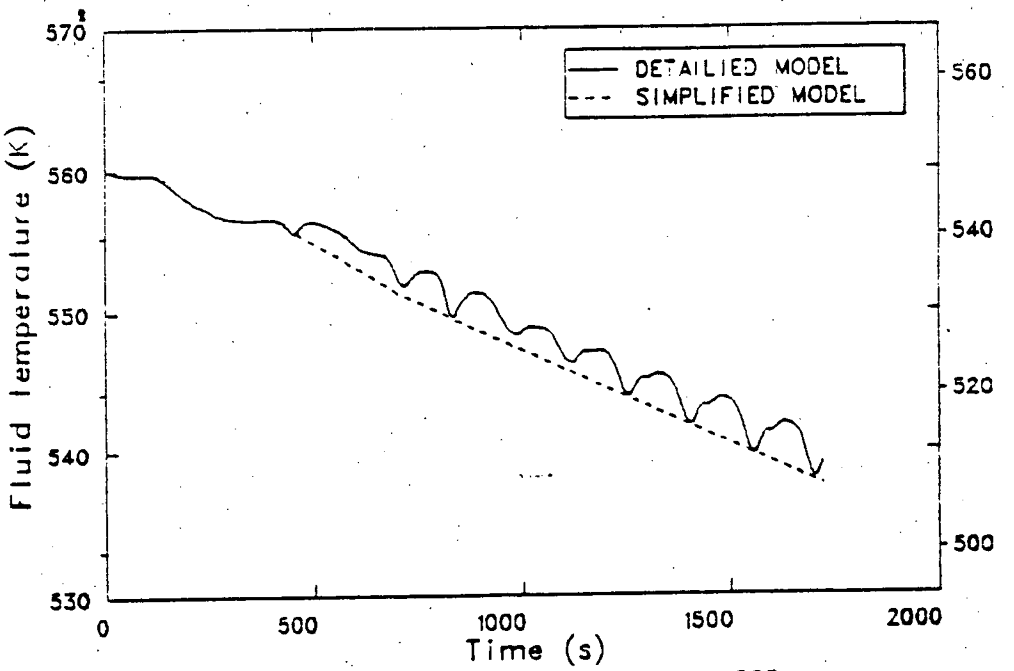


Figure 15. Sequence 3-1 simplified model average RCS temperature benchmark.

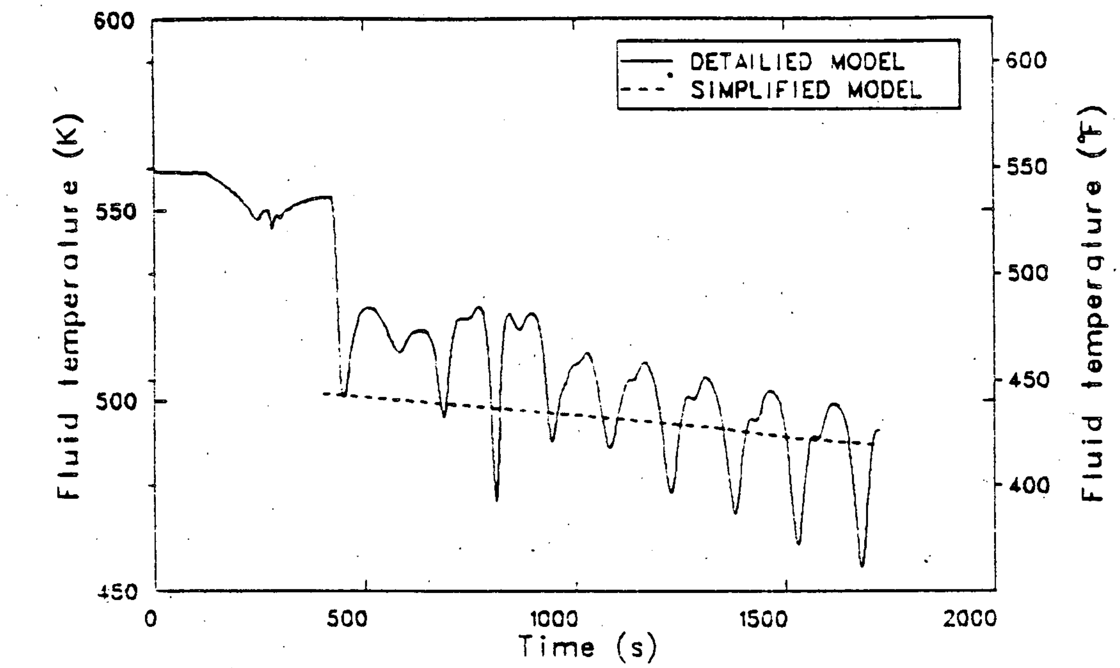


Figure 16. Sequence 3-1 simplified model downcomer temperature benchmark.

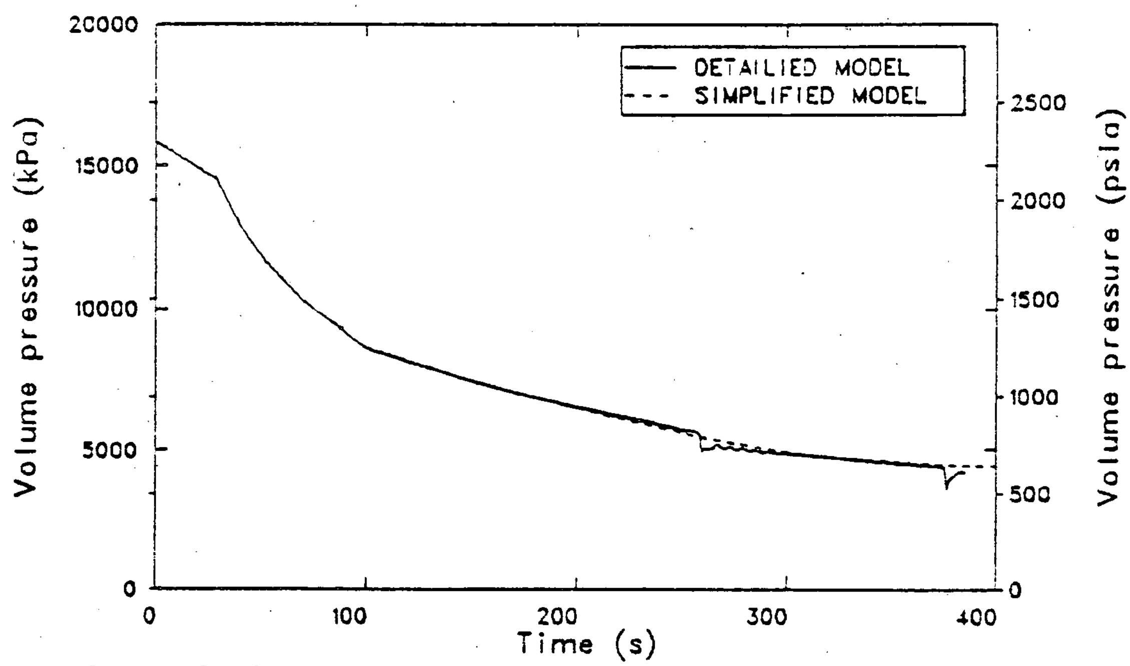


Figure 17. Sequence 1-8 simplified model downcomer pressure benchmark.

3.4.6 SBLOCA Combined with Steam Break, Symmetrically Affecting Three Steam Generators at Full Power

Sequence 1-8 (see Table 1) involved a single, stuck-open pressurizer PORV and five stuck-open steam dump valves at full power. The detailed model (Section 3.3.1) and Variation 7 of the simplified model (Section 3.3.3) were run for this sequence over a common period from 200 to 385 s. Downcomer pressures calculated with the detailed and simplified models are shown in Figure 17. The pressures show remarkably good agreement, even after accumulator injection was initiated near 330 s. The rapid depressurizations calculated with the detailed model at 265 and 375 s were caused by condensation spikes that were not reasonable but did not have a significant long-term effect on the results. Figures 18 and 19 show average RCS fluid temperatures and downcomer fluid temperatures, respectively. The comparison of the average RCS temperatures indicates that the simplified model calculated the overall cooldown rate well since the two temperatures were nearly parallel after 230 s. The effect of ECC mixing on downcomer temperature was also calculated well, even after the start of accumulator flow.

3.4.7 MBLOCA Combined With Small Steam Break, Affecting Two Steam Generators at Full Power

Sequence 2-11 (see Table 2) involved a 0.0635-m (2.5-in.) diameter hot leg break and two stuck-open steam PORVs at full-power conditions. The detailed model (Section 3.3.1) and Variation 9 of the simplified model (Section 3.3.3) were run for this sequence over a common period from 400 to 580 s. Downcomer pressures calculated with the detailed and simplified models are shown in Figure 20. The average depressurization rate was similar with the two models. Figures 21 and 22 show average RCS fluid temperatures and downcomer fluid temperature, respectively. The average RCS temperatures were within a few degrees shortly after the start of the benchmark calculation. The downcomer temperatures also showed fairly good agreement. The simplified model was also able to calculate the trend of the rapid cooldown following the start of accumulator injection near 450 s.

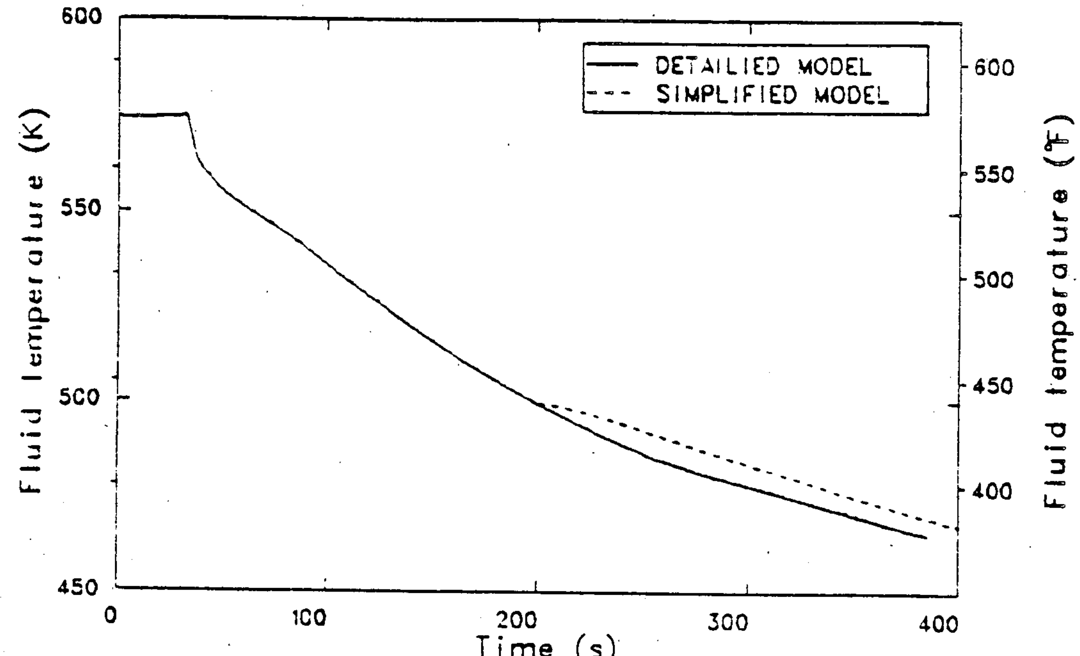


Figure 18. Sequence 1-8 simplified model average RCS temperature benchmark

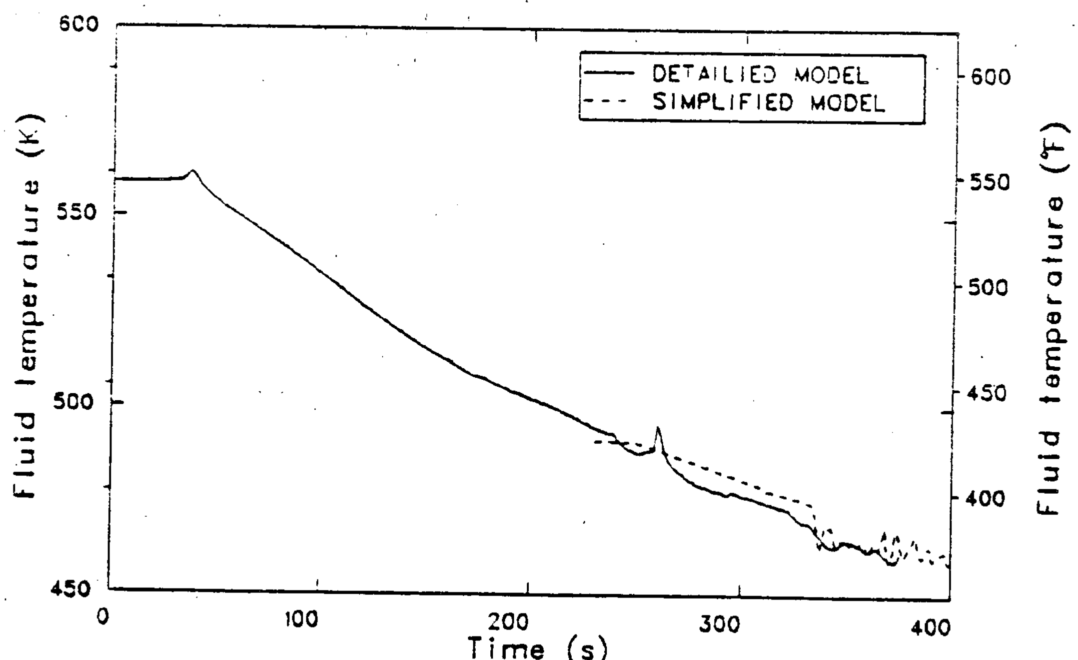


Figure 19. Sequence 1-8 simplified model downcomer temperature benchmark.

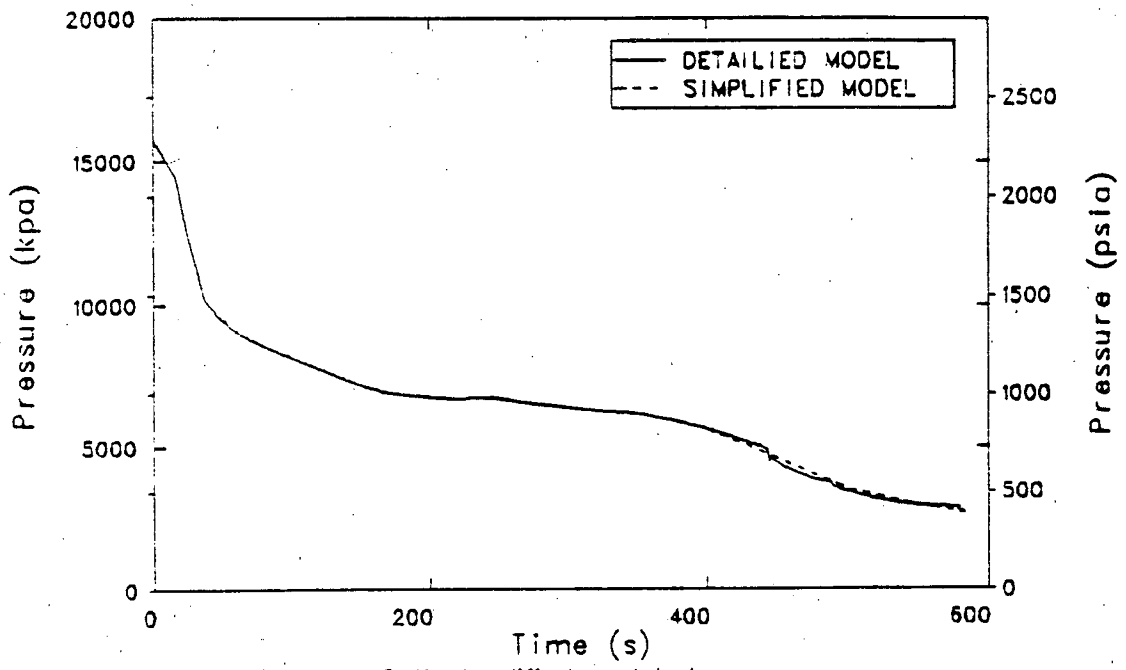


Figure 20. Sequence 2-11 simplified model downcomer pressure benchmark.

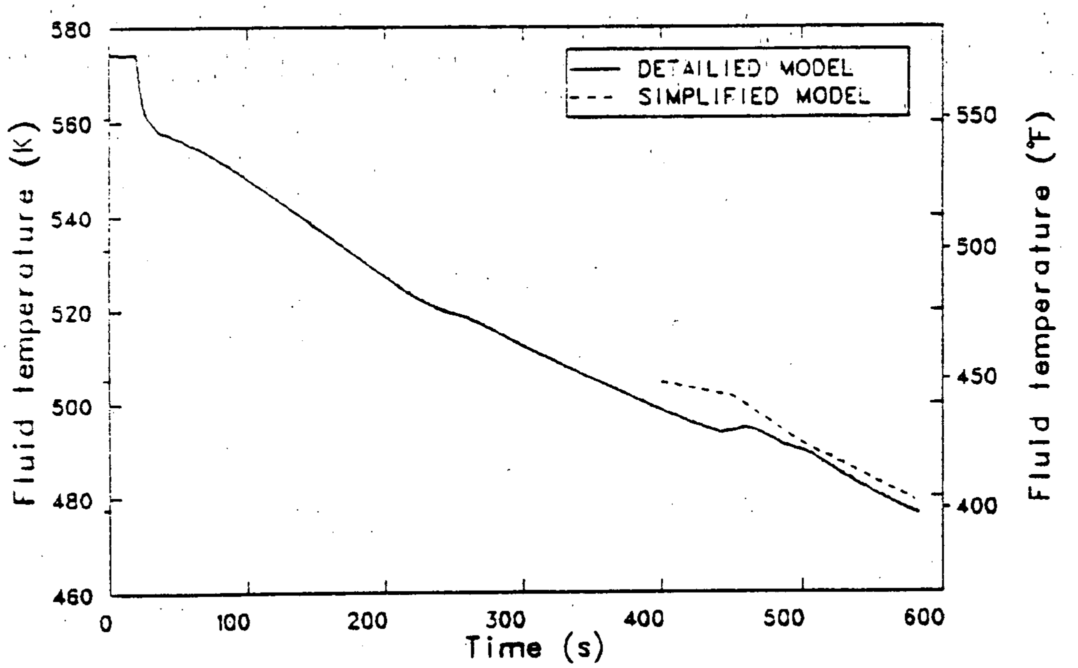


Figure 21. Sequence 2-11 simplified model average RCS temperature benchmark.

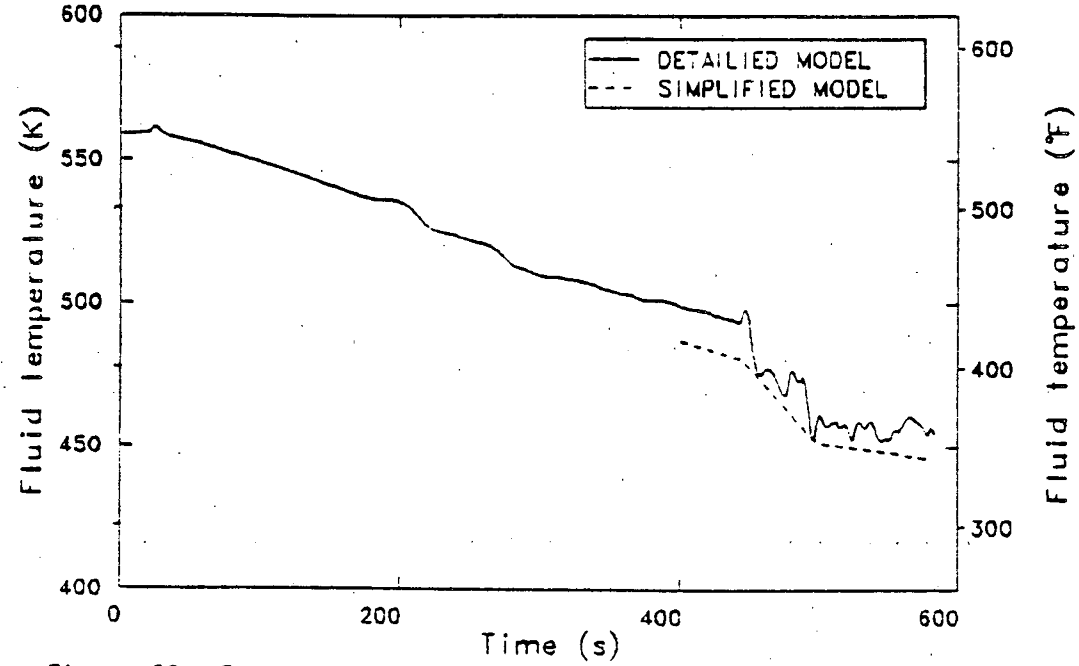


Figure 22. Sequence 2-11 simplified model downcomer temperature benchmark.

4. GROUP A RESULTS: STEAM LINE BREAK AFFECTING ONE STEAM GENERATOR

The following sections present the results of thermal-hydraulic analyses for sequences in Group A of Table 15, using the methods presented in Section 3 to determine reactor vessel downcomer fluid pressure and temperature histories. Section 4.1 defines the group and describes the controlling phenomena. Within Group A, sequences were organized by initiating event and power level as follows: (1) small break at hot standby, (2) small break at full power, (3) large break at hot standby, and (4) large break at full power. Sections 4.2 through 4.5 present the results for these subgroups. Conclusions covering Group A sequences appear in Section 4.6. To facilitate referencing of data, plotted results showing the pressure and temperature histories are organized in numerical sequence in Appendix A rather than appearing within this section. Due to the large number of sequences investigated in this report, detailed discussions of thermal-hydraulic processes for each sequence are not practical. Such discussions are documented for representative sequences in Reference 1.

4.1 Group A Definition

Group A includes all sequences controlled by a steam line break which affect only one steam generator. Within this section a "small" steam line break refers to a single, stuck-open steam line power-operated relief valve (PORV) and a "large" steam line break refers to a double-ended rupture of a single steam line. Both breaks are located downstream of the steam line flow restrictor and upstream of the main steam isolation valve (MSIV) and steam line check valve. So located, the break cannot be isolated by closure of the MSIV or affect the other two steam generators, due to the action of the check valve. Thus, all sequences in Group A are controlled by primary system heat removal to a single, affected steam generator (ASG) and heat addition from the two unaffected steam generators (USGs).

4.2 Small Steam Break At Hot Standby

Sequences involving small steam breaks at hot standby are numbered 7-1 through 7-8 (see Table 7). The sequences differ based on: (1) isolation of auxiliary feedwater (AFW) at 10 minutes, (2) AFW flow rate (normal or high) to the affected steam generator, (3) throttling of AFW to the unaffected steam generator, and (4) throttling of charging flow when pressurizer setpoint level is surpassed.

For these sequences it is possible to overfill the USGs only if AFW is isolated or throttled to the ASG. Due to very low ASG pressures, all AFW flows from the headers to the ASG, unless AFW valves to it are closed, in which case AFW can be delivered to the USGs.

Table 17 presents the methods used, and gives an indication of results for, sequences 7-1 through 7-8. Plotted results for all sequences appear in Appendix A.

For Sequences 7-1 through 7-8, turbine-driven AFW is not initiated because it requires two steam generators to have low level indications, and this condition was present only in the affected steam generator.

Calculations using the detailed model were performed for initial portions of Sequences 7-4 (Scenario 3 in Reference 1) and 7-7. Simplified models used were Variations 1, 2, and 3, as described in Section 3.3.3.

For Sequences 7-1, 7-2, and 7-7, AFW is terminated at 10 minutes; the affected steam generator subsequently drys out. Following dryout, the primary system heats up due to core decay heat, RCP power, and a loss of secondary heat sink. For convenience, hand calculations were performed over the heatup phases of these sequences. Using the simplified model results at the time of dryout, the primary system temperature was extended to 7200 s by calculating the effect of decay heat and RCP power on the temperature of the primary and secondary metal and liquid masses over this period. This temperature response was then used to calculate the pressure response through the assumptions of isentropic pressurizer behavior.

TABLE 17. RESULTS OF SMALL STEAM BREAKS AFFECTING 1 SG AT HOT STANDBY

Sequence Number	Initiating Event	Power Level	Failures	Detailed Model Calculation Used	Simplified Model Variation Used (From 3.3.3)
7-1	Small SLB	HSB	None	7-4	2
7-2	Small SLB	HSB	Charging not throttled	7-4	2
7-3	Small SLB	HSB	AFW overfill	7-4	3
7-4	Small SLB	HSB	AFW not isolated	7-4	1
7-5	Small SLB	HSB	AFW not isolated, charging not throttled	7-4	1
7-6	Small SLB	HSB	AFW not isolated, AFW overfill	7-4	1
7-7	Small SLB	HSB	AFW overfeed	7-7	2
7-8	Small SLB	HSB	AFW not isolated, AFW overfeed	7-7	1

	Methods Used, Periods in Seconds						
Sequence Number	Detailed Model	Simplified Model	Hand Calculations	Minimum RV DC Temperature (°F)	Maximum Subsequent RV DC Pressure (psia)	Figures Showing Plotted Results	Notes[a]
7-1	0 - 600	600 - 2216	2216 - 7200	412	2371	A-131, A-132	A,B,C
7-2	0 - 600	600 - 2268	2268 - 7200	409	2371	A-133, A-134	A,B,C
7-3	0 - 600	600 - 7200	---	369	2371	A-135, A-136	A,B,D
7-4	0 - 1290	1290 - 7200	---	283	1565	A-137, A-138	A
7-5	0 - 1290	1290 - 7200	---	270	2371	A-139, A-140	A
7-6	0 - 1290	1290 - 7200	---	283	1565	A-141, A-142	A
7-7	0 - 600	600 - 2400	2400 - 7200	393	2371	A-143, A-144	A,B,C
7-8	0 - 1000	1000 - 7200	---	288	1528	A-145, A-146	A

a. A--Turbine-driven AFW not initiated.

B--Reactor coolant pumps do not trip.

C--Hand calculation performed over the heatup phase following dryout of ASG.

D--Simplified model results were adjusted for ASG blowdown effects from 600 - 2216 s based on results in 7-1.

For Sequence 7-3, in which only the USG overfill was accounted for in the simplified model variation, the effects of the ASG blowdown from 600 to 2216 s (as calculated for Sequence 7-1) were superimposed on the simplified model results over this period.

The description for Sequence 7-6 is identical to that for Sequence 7-4 except that in Sequence 7-6 the USGs are overfilled with AFW, following the throttling of AFW to the ASG. In Sequence 7-4 this throttling occurred at 3318 s, a time when the primary system and the ASG were significantly colder than the USGs. As the USGs are overfilled with AFW they become colder; however, they do not become colder than the primary system. As a result, the USGs are not recoupled by natural circulation to the primary system, and the overfilling of the USGs has no effect on the primary system pressure and temperature. Therefore, results for Sequence 7-6 are identical to those for Sequence 7-4.

For Sequences 7-1, 7-2, 7-4, and 7-7, the AFW is terminated to the ASG at 10 minutes; by about 45 minutes, the ASG secondary was dry. As a result, in these sequences the reactor coolant pumps remained operating, and primary system temperatures increased after about 45 minutes. The resulting thermal expansion of the primary fluid drove the primary system pressures up to the PORV setpoint.

For Sequences 7-4, 7-5, 7-6, and 7-8 the AFW was continued until the time of spillover of liquid to the steam line from the ASG. After spillover, the AFW was throttled to maintain ASG secondary mass. Primary system temperatures declined continuously in these sequences, with the slope reduced following AFW throttling. Due to continuous thermal contraction, the primary system pressures were maintained below 11 MPa (1600 psia) except for Sequence 7-5 in which the failing of charging throttling drove the pressure to the PORV setpoint. This caused Sequence 7-5 to be the most severe sequence of the subgroup, with a minimum temperature of 405 K (270°F) and a maximum pressure of 16.35 MPa (2371 psia), both of which occurred at the end of the two-hour period.

4.3 Small Steam Break at Full Power

Sequences involving small steam breaks at full power are 5-1 and 9-25 through 9-32 (see Tables 5 and 9). Sequence 5-1 is initiated by the steam break, while Sequences 9-25 through 9-32 are initiated by a reactor trip followed by a steam PORV failing open. This distinction is important because a detailed model calculation of Sequence 5-1 indicated that, for an initiating event of a single, stuck-open steam PORV, an automatic reactor trip is not encountered. Instead, the steam and feed systems respond to the effects of the break and, after approximately 5 minutes, the plant reaches a new steady operating point, with the cold leg temperature 1 K (1.8°F) below the starting point. This drop in temperature is not sufficient to cause an automatic overpower reactor trip due to moderator temperature effects.

There is a possibility that the operator would respond to conditions present following a stuck-open steam PORV by manually tripping the reactor. For purposes of this study, it was assumed for Sequence 5-1 that this would not occur. If manual reactor trip did occur, Sequence 5-1 could be expected to be virtually identical to Sequence 9-25 in which the reactor trip was the initiating event.

Sequences 9-25 through 9-32 all assume no AFW isolation. Sequences 9-25 through 9-32 differ based on: (1) AFW flow rates (normal or high), (2) throttling of AFW to the USGs at 40% level, and (3) throttling of charging flow when pressurizer setpoint level is attained.

Table 18 presents the methods used, and gives an indication of results for, Sequences 5-1, and 9-25 through 9-32. Plotted results for all sequences appear in Appendix A.

For Sequences 9-25 through 9-32, both the motor- and turbine-driven AFW operate. Unlike the small steam break at hot standby, at full power the USGs relieve steam for a period following reactor trip. As a result, the USG levels decrease sufficiently to cause initiation of turbine-driven AFW.

TABLE 18. RESULTS OF SMALL STEAM BREAKS AFFECTING 1 SG AT FULL POWER

Sequence Number	Initiating Event	Power Level	Failures	Detailed Model Calculation Used	Simplified Model Variation Used (From 3.3.3)
5-1	Small SLB	Full	None	5-1	--
9-25	Reactor trip	Full	Stuck-open steam PORV	9-25	2
9-26	Reactor trip	Full	Stuck-open steam PORV, AFW overfill	9-25	2
9-27	Reactor trip	Full	Stuck-open steam PORV, charging not throttled	9-25	2
9-28	Reactor trip	Full	Stuck-open steam PORV, AFW overfill, charging not throttled	9-25	2
9-29	Reactor trip	Full	Stuck-open steam PORV, AFW overfeed	9-25	2
9-30	Reactor trip	Full	Stuck-open steam PORV, AFW overfeed, AFW overfill	9-25	2
9-31	Reactor trip	Full	Stuck-open steam PORV, AFW overfeed, charging not throttled	9-25	2
9-32	Reactor trip	Full	Stuck-open steam PORV, AFW overfeed, AFW overfill, charging not throttled	9-25	2

TABLE 18. (continued)

Sequence Number	Methods Used, Periods in Seconds: Detailed Model	Methods Used, Periods in Seconds: Simplified Model	Methods Used, Periods in Seconds: Hand Calculations	Minimum RV DC Temperature (°F)	Maximum Subsequent RV DC Pressure (psia)	Figures Showing Plotted Results	Notes[a]
5-1	0 - 337	--	337 - 7200	544	2296	A-65, A-66	A,B,C
9-25	0 - 75	75 - 7200	1877 - 7200	343	1675	A-215, A-216	A,D
9-26	0 - 75	75 - 7200	1877 - 7200	292	1647	A-217, A-218	A,D
9-27	0 - 75	75 - 7200	1877 - 7200	333	2371	A-219, A-220	A,D
9-28	0 - 75	75 - 7200	1877 - 7200	282	2371	A-221, A-222	A,D
9-29	0 - 75	75 - 7200	1047 - 7200	344	1623	A-223, A-224	A,D
9-30	0 - 75	75 - 7200	1047 - 7200	293	1602	A-225, A-226	A,D
9-31	0 - 75	75 - 7200	1047 - 7200	334	2371	A-227, A-228	A,D
9-32	0 - 75	75 - 7200	1047 - 7200	283	2371	A-229, A-230	A,D

a. A--Reactor coolant pumps do not trip.

B--Reactor does not trip.

C--Steady conditions at 337 s expected to continue through 7200 s.

D--Hand calculations performed to adjust simplified model results for AFW delivery to the USG.

In addition to Sequence 5-1, a detailed model calculation was performed for the initial portion of Sequence 9-25. A major finding of this calculation was that primary system pressure does not decline sufficiently to cause tripping of the reactor coolant pumps (RCPs). Simplified model calculations were performed using Variation 2, as described in Section 3.3.3.

For Sequence 5-1, the detailed model was run to 337 s, at which time a new steady state condition had been attained. This steady condition is expected to continue to 7200 s, so pressures and temperatures do not change between 377 and 7200 s. For Sequence 5-1 the absence of a reactor trip controls the response. Reactor vessel downcomer pressure and temperature remain near their full-power, steady state values.

For Sequences 9-25 through 9-32, the temperatures calculated using the simplified model were adjusted after the time the ASG was water-filled (1877 s with normal AFW delivery rate, and 1047 s at the overfeed rate). After that time, AFW is throttled to the ASG, and AFW flows to the USGs. The simplified model does not account for a cooldown of the USGs during this period. To adjust for this, the effect of the USG AFW demanded on the overall system temperature was determined by an energy balance hand calculation and this effect was superimposed on the simplified model calculation. Primary system pressure during this period was then adjusted accordingly, based on the primary system fluid shrinkage caused by the temperature adjustment.

For Sequences 9-25 through 9-32, the primary system temperatures continuously decline. The rate of decrease is significantly slowed when AFW is throttled to the ASG, at the time the ASG begins to spill liquid to the steam line. Primary system pressures are stabilized below 11.55 MPa (1675 psia), except in cases where the charging fails to throttle and the pressures are driven to the pressurizer PORV opening setpoint.

Sequence 9-28, involving failure to throttle AFW to the USGs and failure to throttle charging, was the most severe sequence of the subgroup,

with a minimum temperature of 412 K (282°F) and a maximum pressure of 16.35 MPa (2371 psia), both of which occurred at the end of the two-hour period.

4.4 Large Steam Break at Hot Standby

The sequences involving large steam breaks at hot standby are 8-1 through 8-6. These sequences differ based on: (1) AFW isolation to the ASG at 10 minutes, (2) AFW delivery rate (normal or overfeed), (3) AFW throttling to the USG at 40% level, and (4) throttling of charging when pressurizer setpoint level is attained.

Table 19 presents the methods used, and gives an indication of results for, Sequences 8-1 through 8-6. Plotted results for all sequences appear in Appendix A.

For these sequences it is possible to overfill the USGs only if AFW is isolated or throttled to the ASG. Due to very low ASG pressure, all AFW flows from the headers to the ASG, unless AFW valves to it are closed.

For these sequences, turbine-driven AFW is not initiated, because it requires two steam generators to have low level indications, and this condition was present only in the affected steam generator.

A detailed model calculation was performed for the initial portion of Sequence 8-4. Simplified models used Variation 4, as described in Section 3.3.3.

In Sequence 8-3 there is an AFW overfill of the unaffected steam generators that begins at 600 s, when the AFW is isolated to the ASG. At 2850 s, the USG temperatures fall below the primary system temperature, and natural circulation of the unaffected loops commence. Variation 4 of the simplified model does not include USG effects, and the following hand calculation was performed to address the time period from 2850 s to 7200 s. The primary system temperature response was determined by taking a mass-weighted average of the primary system temperature response from

TABLE 19. RESULTS OF LARGE STEAM BREAKS AFFECTING 1 SG AT HOT STANDBY

Sequence Number	Initiating Event	Power Level	Failures	Detailed Model Calculation Used	Simplified Model Variation Used (From 3.3.3)
8-1	Large SLB	HSB	None	8-4	4
8-2	Large SLB	HSB	Charging not throttled	8-4	4
8-3	Large SLB	HSB	AFW overfill	8-4	4
8-4	Large SLB	HSB	AFW not isolated	8-4	4
8-5	Large SLB	HSB	AFW overfeed	8-4	4
8-6	Large SLB	HSB	AFW not isolated, AFW overfeed	8-4	4

	Methods Used, Periods in Seconds						
Sequence Number	Detailed Model	Simplified Model	Hand Calculations	Minimum RV DC Temperature (°F)	Maximum Subsequent RV DC Pressure (psia)	Figures Showing Plotted Results	Notes[a]
8-1	0 - 600	600 - 7200	---	284	2371	A-155, A-156	A
8-2	0 - 600	600 - 7200	---	283	2371	A-157, A-158	A
8-3	0 - 600	600 - 2850	2850 - 7200	284	1792	A-159, A-160	A,B
8-4	0 - 200	200 - 7200	---	204	1947	A-161, A-162	A
8-5	0 - 200	200 - 7200	---	274	2371	A-163, A-164	A
8-6	0 - 200	200 - 7200	---	201	1995	A-165, A-166	A

a. A--Turbine-driven AFW does not initiate.

B--Hand calculation used to introduce effects of overfilling the USG with AFW.

Sequence 8-1 and the USG temperature response as it is overfilled. As a result, the primary system temperature for Sequence 8-3: decreases rapidly until AFW is isolated to the ASG at 600 s; increases until the USGs become colder than the primary system at 2850 s; decreases again until the USGs are filled at 3200 s; and then increases again when AFW is throttled to all steam generators.

Sequences 8-1 through 8-6 start with a rapid cooldown and depressurization of the primary system. In Sequences 8-1, 8-2, 8-3, and 8-5, AFW is isolated to the ASG at 10 minutes and the primary system heats up. The resulting primary system liquid thermal expansion drives the primary system pressure up to the pressurizer PORV opening setpoint (except for 8-3, where cooling to the USGs prevents it). In Sequences 8-4 and 8-6, AFW isolation does not occur until the ASG begins to spill liquid to the steam line (3700 s in 8-4 and 2452 s in 8-6). As a result, in Sequences 8-4 and 8-6, ASG cooling continues much longer and the primary system temperatures fall well below those for the other sequences. Due to additional primary system fluid shrinkage, however, the primary system pressures were maintained below 13.8 MPa (2000 psia) in Sequences 8-4 and 8-6.

All of Sequences 8-1 through 8-6 are severe. Sequences 8-1, 8-2, and 8-5 result in minimum temperatures of about 407 K (274°F) and maximum pressures of 16.35 MPa (2371 psia). Sequences 8-4 and 8-6 result in minimum temperatures of about 367 K (201°F) and maximum pressures of 13.8 MPa (2000 psia). Sequence 8-3 is slightly less severe, with a minimum temperature of 413 K (284°F) and a maximum pressure of 12.35 MPa (1792 psia).

4.5 Large Steam Break At Full Power

Sequences involving large breaks at full power are 6-1 through 6-9. These sequences differ, based on: (1) AFW isolation to the ASG at 10 minutes, (2) AFW delivery rate (normal of overfeed), (3) throttling of AFW to the ASG at 40% level, and (4) throttling of charging flow when pressurizer setpoint level is attained.

For these sequences it is possible to overfill the USGs only if AFW is isolated ot throttled to the ASG. Due to very low ASG pressures, all AFW flows from the headers to the ASG, unless AFW valves to it are closed. Both the motor- and turbine-driven AFW operate in these sequences because the USGs relieve steam for a period following reactor trip. As a result, the USG levels decrease sufficiently to cause initiation of turbine-driven AFW.

Table 20 both presents the methods used, and gives an indication of results for, Sequences 6-1 through 6-9. Plotted results for all sequences appear in Appendix A.

A detailed model calculation was performed for the initial portion of Sequence 6-1. Simplified models used Variations 3 and 4, as described in Section 3.3.3.

In Sequences 6-5, 6-6, 6-7, and 6-9, AFW is not isolated to the ASG at 10 minutes, and heat removal to the ASG dominated the sequence. In Sequences 6-1 through 6-4 and 6-8, however, AFW to the ASG was isolated at 10 minutes, and heat removal to the USGs became a dominant mechanism. For Sequences 6-1 through 6-4 and 6-8, therefore, detailed model results were used to 135 s; then Variation 4 simplified model results were used to 600 s, followed by Variation 3 simplified model results to 7200 s. In Sequences 6-1 and 6-8, the simplified model results were adjusted so that primary system temperatures did not exceed 554K (547°F). This is the saturation temperature corresponding to the opening setpoint pressure for the steam dump control system. Primary system temperature excursions would be limited by it.

Sequence 6-7 was found to be equivalent to Sequence 6-5. The two sequences differ only by USG AFW overfill criteria. The overfill of the USGs in Sequence 6-7 does not cause the USG temperature to fall below the primary system temperature calculated for Sequence 6-5. Therefore, unaffected loop natural circulation does not commence; the USGs remain decoupled from the primary system; and the USG overfill does not affect the primary system pressure and temperature responses.

TABLE 20. RESULTS OF LARGE STEAM BREAKS AFFECTING 1 SG AT FULL POWER

Sequence Number	Initiating Event	Power Level	Failures	Detailed Model Calculation Used	Simplified Model Variation Used (From 3.3.3)
6-1	Large SLB	Full	None	6-1	4,3
6-2	Large SLB	Full	Charging not throttled	6-1	4,3
6-3	Large SLB	Full	AFW overfill	6-1	4,3
6-4	Large SLB	Full	AFW overfill, charging not throttled	6-1	4,3
6-5	Large SLB	Full	AFW not isolated	6-1	4
6-6	Large SLB	Full	AFW not isolated, charging not throttled	6-1	4
6-7	Large SLB	Full	AFW not isolated, AFW overfill	6-1	4
6-8	Large SLB	Full	AFW overfeed	6-1	4,3
6-9	Large SLB	Full	AFW not isolated, AFW overfeed	6-1	4

	Methods Used, Periods in Seconds						
Sequence Number	Detailed Model	Simplified Model	Hand Calculations	Minimum RV DC Temperature (°F)	Maximum Subsequent RV DC Pressure (psia)	Figures Showing Plotted Results	Notes[a]
6-1	0 - 135	135 - 6532	6532 - 7200	302	2371	A-113, A-114	B,C
6-2	0 - 135	135 - 7200	---	302	2371	A-115, A-116	B
6-3	0 - 135	135 - 7200	---	302	2371	A-117, A-118	B
6-4	0 - 135	135 - 7200	---	302	2371	A-119, A-120	B
6-5	0 - 135	135 - 7200	---	229	1772	A-121, A-122	
6-6	0 - 135	135 - 7200	---	227	2371	A-123, A-124	
6-7	0 - 135	135 - 7200	---	229	1772	A-125, A-126	A
6-8	0 - 135	135 - 6600	6600 - 7200	299	2371	A-127, A-128	B,C
6-9	0 - 135	135 - 7200	---	229	1781	A-129, A-130	

a. A--Sequence 6-7 was determined to be equivalent to 6-5 because USGs do not become colder than the primary system.

B--Variation 4 simplified model used to 600 s, Variation 3 simplified model used for remainder of period shown.

C--Simplified model results were adjusted by limiting primary system temperature increases such that 547°F is not exceeded. 547°F is saturation temperature corresponding to opening setpoint pressure for the steam dump control system.

Because AFW is not isolated to the ASG at 10 minutes, primary system temperatures fall considerably lower in Sequences 6-5, 6-6, 6-7, and 6-9 than in the other sequences where AFW is isolated at 10 minutes. The extra primary fluid shrinkage caused by cooling to the lower temperatures, however, prevented the primary system pressure from reaching the pressurizer PORV opening setpoint pressure (except in Sequence 6-6 that included a failure to throttle charging).

Because the AFW was isolated at 10 minutes in Sequences 6-1, 6-2, 6-3, 6-4, and 6-8, primary system temperatures fell to only about 422 K (300°F). However, in all these sequences, the primary system pressure was later driven to the pressurizer PORV opening setpoint pressure by thermal expansion of the primary system fluid.

The most severe sequence of this subgroup was thus 6-6. The minimum temperature was 381 K (227°F), and maximum subsequent pressure was 16.35 MPa (2371 psia).

4.6 Conclusions

Group A includes 32 sequences involving steam breaks affecting one steam generator. These sequences were investigated in subgroups according to break size (small or large) and power level (hot standby or full-power initial conditions). General findings for the subgroups are summarized below.

For the small steam break (stuck-open steam PORV) at hot standby, Sequence 7-5 (no AFW isolation and a failure of charging throttling) was found to be the most severe sequence. This sequence had a minimum temperature of 405 K (270°F) and maximum pressure of 16.35 MPa (2371 psia). Other sequences in the subgroup were significantly less severe in temperature, pressure, or both.

For the small steam break (stuck-open steam PORV) at full power, Sequence 9-28 (failure to throttle AFW and charging) was the most severe sequence [412 K (282°F), 16.35 MPa (2371 psia)], and Sequence 9-32 was nearly as severe. An important finding for this group was that, if the

initiating event is the stuck-open PORV (Sequence 5-1), a reactor trip does not occur and an overcooling event is avoided, unless the operator manually trips the plant in response to the transient observed. If the initiating event is a reactor trip and the stuck-open PORV is a subsequent failure, the primary system pressure does not decline sufficiently to cause tripping of the reactor coolant pumps.

For the large steam break (double-ended main steam line break) at hot standby, all 6 sequences investigated were severe. Two classes of severity were found. The first class, with AFW isolated, has 3 sequences near 407 K (274°F), 16.35 MPa (2371 psia). The second class, with AFW not isolated, has 2 sequences near 367 K (201°F), 13.8 MPA (2000 psia).

For the large steam break at full power, the most severe sequence was 6-6 (no AFW isolation and a failure to throttle charging). Other sequences in the group were significantly less severe because either: (1) AFW is isolated at 10 minutes and the cooldown is stopped, or (2) AFW and the cooldown continue, but the resulting primary system fluid shrinkage prevents the primary system pressure from increasing to the pressurizer PORV opening setpoint pressure.

5. GROUP B RESULTS: SECONDARY-SIDE BREAK WITH THREE SYMMETRICALLY AFFECTED STEAM GENERATORS

The following sections present the results of thermal-hydraulic analyses for sequences in Group B of Table 15 using the methods presented in Section 3 to determine reactor vessel downcomer fluid pressure and temperature histories. Section 5.1 defines the group and describes the controlling phenomena. Section 5.2 discusses the results for a single, stuck-open steam line PORV (and steam line check valve failure) at hot standby conditions. Results for sequences initiated from full-power conditions are shown in Section 5.3. These sequences involve one or more stuck-open steam dump valves or three stuck-open steam PORVs. Conclusions covering all Group B sequences appear in Section 5.4. To facilitate referencing of data, plotted results showing the pressure and temperature histories are organized in numerical sequence order in Appendix A rather than appearing within this section. Due to the large number of sequences investigated in this report, detailed discussions of thermal-hydraulic processes for each sequence are not practical. Such discussions are documented for representative sequences in Reference 1.

5.1 Group B Definition

Group B includes all sequences controlled by a steam break that symmetrically affects all three generators and no primary system break. Sequences of this type fall into three categories: sequences involving (1) stuck-open steam line PORVs with coincident failure of the check valve on the same line, (2) three stuck-open steam line PORVs, and (3) one or more stuck-open steam dump valves (SDVs). In all of these cases, the steam break (or breaks) are located so that the steam flows exiting the steam generators are equal. As a result, the responses of the three steam generators are identical.

Sequences in Group B are controlled by equal primary system heat removal to each steam generator and heat addition from core decay heat and stored energy in metal heat structures.

For all Group B sequences, turbine-driven AFW is initiated because all three steam generator levels decrease sufficiently, due to loss of secondary inventory. Turbine-driven AFW is assumed to be unavailable when the steam generator secondary pressures are below 0.69 MPa (100 psia). Below this pressure, insufficient steam is available to run the turbine-driven AFW pumps.

5.2 One Stuck-Open Steam PORV and Steam Line Check Valve Failure at Hot Standby

Sequence 7-12 involves a stuck-open steam PORV on one steam line and a failure of the steam line check valve on the same line. With both of these failures, all three steam generators blow down through the steam PORV. As specified in the sequence description, all AFW is isolated at 10 min. As a result, the primary system pressure does not decline sufficiently to cause tripping of the reactor coolant pumps. Restoration of AFW will eventually be required (after the two hour period) to avoid drying out the steam generator secondaries.

Table 21 presents the methods used, and gives an indication of results for, Sequence 7-12. Plotted results for this sequence appear in Appendix A.

A detailed model calculation for the initial portion of Sequence 7-4 was used to predict the behavior for Sequence 7-12. Sequence 7-4 involves the same steam break without the check valve failure. The primary system heat removal was thus nearly the same for Sequences 7-4 and 7-12. Variation 5 of the simplified model (as described in Section 3.3.3) was used over the later stage of the sequence. A mass-weighted average of the steam generator secondary conditions (of the 1 ASG and 2 USGs) from the detailed model calculation for Sequence 7-4 was used to initialize the simplified model calculation for Sequence 7-12.

The minimum temperature of the fluid in the reactor vessel downcomer for Sequence 7-12 was 440 K (332°F), and the maximum subsequent pressure was 10.4 MPa (1505 psia). The severity of the sequence was limited by the isolation of AFW at 10 min.

TABLE 21. RESULTS OF ONE STUCK-OPEN STEAM PORV AND STEAM LINE CHECK VALVE FAILURE AT HOT STANDBY

Sequence Number	Initiating Event	Power Level	Failures	Detailed Model Calculation Used	Simplified Model Variation Used (From 3.3.3)
7-12	Small SLB	HSB	None	7-4	5

Sequence Number	Methods Used, Periods in Seconds: Detailed Model	Methods Used, Periods in Seconds: Simplified Model	Methods Used, Periods in Seconds: Hand Calculations	Minimum RV DC Temperature (°F)	Maximum Subsequent RV DC Pressure (psia)	Figures Showing Plotted Results
7-12	0 - 600	600 - 7200	---	332	1505	A-153, A-154

5.3 One or More Steam Dump Valves, or Three Steam PORVs, Stuck Open Following Reactor Trip From Full Power

Sequences 9-2 through 9-23 involve one or more stuck-open steam dump valves (SDVs) at full power. Sequences 9-41 through 9-47 involve three stuck-open steam PORVs at full power. For all these sequences, isolation of AFW is not required unless a main steam isolation valve (MSIV) fails to close on demand. The steam flow was not sufficient to demand the MSIVs with one, two, or three SDVs or three steam PORVs stuck open. With five SDVs stuck open, the steam flow was sufficient over the first few seconds of transient, but the additional requirement for low average temperature was not met until later in the transient. To demand the MSIVs, these two conditions must be satisfied at the same time; therefore, the MSIVs were not demanded for five SDVs stuck open as well. Since the MSIVs are not demanded, AFW is not isolated in these sequences. The sequences differ by: (1) AFW feed rates (normal or overfeed), (2) AFW throttling at 40% level or when liquid spills to the steam line, and (3) throttling of charging flow when the pressurizer setpoint level is attained.

Table 22 presents the analysis methods used, and gives an indication of results for, one SDV stuck open, Table 23 for two SDVs, Table 24 for three SDVs, Table 25 for five SDVs, and Table 26 for three steam PORVs. Plotted results for all sequences appear in Appendix A.

Sequences 9-2 through 9-8 involve one stuck-open SDV; 9-9 through 9-13 two SDVs; 9-14 through 9-18 three SDVs; and 9-19 through 9-23 five SDVs. Sequences 9-41 through 9-47 involve three stuck-open steam line PORVs. Detailed model calculations were performed for the initial portions of one sequence with each break size: 9-3, 9-9, 9-15, 9-19, and 9-41. Simplified model calculations used the base simplified model described in Section 3.3.2.

Sequences 9-6, 9-7, 9-8, 9-13, 9-18, 9-23, 9-45, 9-46, and 9-47 include an overfeed of AFW at a rate 48% above normal. The detailed model calculation used for these sequences did not include the effects of the overfeed. To account for this, the initial conditions for the simplified

TABLE 22. RESULTS OF ONE STUCK-OPEN STEAM DUMP VALVE AT FULL POWER

Sequence Number	Initiating Event	Power Level	Failures	Detailed Model Calculation Used	Simplified Model Variation Used (From 3.3.3)
9-2	Reactor trip	Full	1 stuck-open SDV	9-3	Base
9-3	Reactor trip	Full	1 stuck-open SDV, AFW overfill	9-3	Base
9-4	Reactor trip	Full	1 stuck-open SDV, charging not throttled	9-3	Base
9-5	Reactor trip	Full	1 stuck-open SDV, AFW overfill, charging not throttled	9-3	Base
9-6	Reactor trip	Full	1 stuck-open SDV, AFW overfeed	9-3	Base
9-7	Reactor trip	Full	1 stuck-open SDV, AFW overfeed, AFW overfill	9-3	Base
9-8	Reactor trip	Full	1 stuck-open SDV, AFW overfeed, charging not throttled	9-3	Base

Sequence Number	Methods Used, Periods in Seconds: Detailed Model	Simplified Model	Hand Calculations	Minimum RV DC Temperature (°F)	Maximum Subsequent RV DC Pressure (psia)	Figures Showing Plotted Results	Notes[a]
9-2	0 - 900	900 - 7200	---	312	1544	A-169, A-170	
9-3	0 - 900	900 - 7200	---	294	1553	A-171, A-172	
9-4	0 - 900	900 - 7200	---	305	2371	A-173, A-174	
9-5	0 - 900	900 - 7200	---	282	2371	A-175, A-176	
9-6	0 - 900	900 - 7200	---	306	1527	A-177, A-178	A
9-7	0 - 900	900 - 7200	---	297	1694	A-179, A-180	A
9-8	0 - 900	900 - 7200	---	308	2371	A-181, A-182	A

a. A--Adjustment made in simplified model initial condition to account for AFW overfeed from 0 to 900 s.

TABLE 23. RESULTS OF TWO STUCK-OPEN STEAM DUMP VALVES AT FULL POWER

Sequence Number	Initiating Event	Power Level	Failures	Detailed Model Calculation Used	Simplified Model Variation Used (From 3.3.3)
9-9	Reactor trip	Full	2 stuck-open SDVs	9-9	Base
9-10	Reactor trip	Full	2 stuck-open SDVs, AFW overfill	9-9	Base
9-11	Reactor trip	Full	2 stuck-open SDVs, charging not throttled	9-9	Base
9-12	Reactor trip	Full	2 stuck-open SDVs, AFW overfill, charging not throttled	9-9	Base
9-13	Reactor trip	Full	2 stuck-open SDVs, AFW overfeed	9-9	Base

Sequence Number	Methods Used, Periods in Seconds: Detailed Model	Simplified Model	Hand Calculations	Minimum RV DC Temperature (°F)	Maximum Subsequent RV DC Pressure (psia)	Figures Showing Plotted Results	Notes[a]
9-9	0 - 400	400 - 7200	---	264	1611	A-183, A-184	
9-10	0 - 400	400 - 7200	---	248	1518	A-185, A-186	
9-11	0 - 400	400 - 7200	---	257	2371	A-187, A-188	
9-12	0 - 400	400 - 7200	---	241	2371	A-189, A-190	
9-13	0 - 400	400 - 7200	---	267	1545	A-191, A-192	A

a. A--Adjustment made in simplified model initial condition to account for AFW overfeed from 0 to 400 s.

TABLE 24. RESULTS OF THREE STUCK-OPEN STEAM DUMP VALVES AT FULL POWER

Sequence Number	Initiating Event	Power Level	Failures	Detailed Model Calculation Used	Simplified Model Variation Used (From 3.3.3)
9-14	Reactor trip	Full	3 stuck-open SDVs	9-15	Base
9-15	Reactor trip	Full	3 stuck-open SDVs, AFW overfill	9-15	Base
9-16	Reactor trip	Full	3 stuck-open SDVs, charging not throttled	9-15	Base
9-17	Reactor trip	Full	3 stuck-open SDVs, AFW overfill, charging not throttled	9-15	Base
9-18	Reactor trip	Full	3 stuck-open SDVs, AFW overfeed	9-15	Base

	Methods Used, Periods in Seconds						
Sequence Number	Detailed Model	Simplified Model	Hand Calculations	Minimum RV DC Temperature (°F)	Maximum Subsequent RV DC Pressure (psia)	Figures Showing Plotted Results	Notes[a]
9-14	0 - 390	390 - 7200	---	234	1610	A-193, A-194	
9-15	0 - 390	390 - 7200	---	226	1613	A-195, A-196	
9-16	0 - 390	390 - 7200	---	225	2371	A-197, A-198	
9-17	0 - 390	390 - 7200	---	220	2371	A-199, A-200	
9-18	0 - 390	390 - 7200	---	258	1754	A-201, A-202	A

a. A--Adjustment made in simplified model initial condition to account for AFW overfeed from 0 to 390 s.

TABLE 25. RESULTS OF FIVE STUCK-OPEN STEAM DUMP VALVES AT FULL POWER

Sequence Number	Initiating Event	Power Level	Failures	Detailed Model Calculation Used	Simplified Model Variation Used (From 3.3.3)
9-19	Reactor trip	Full	5 stuck-open SDVs	9-19	Base
9-20	Reactor trip	Full	5 stuck-open SDVs, AFW overfill	9-19	Base
9-21	Reactor trip	Full	5 stuck-open SDVs, charging not throttled	9-19	Base
9-22	Reactor trip	Full	5 stuck-open SDVs, AFW overfill, charging not throttled	9-19	Base
9-23	Reactor trip	Full	5 stuck-open SDVs, AFW overfeed	9-19	Base

Sequence Number	Methods Used, Periods in Seconds: Detailed Model	Simplified Model	Hand Calculations	Minimum RV DC Temperature (°F)	Maximum Subsequent RV DC Pressure (psia)	Figures Showing Plotted Results	Notes[a]
9-19	0 - 400	400 - 7200	---	203	1594	A-203, A-204	
9-20	0 - 400	400 - 7200	---	201	1575	A-205, A-206	
9-21	0 - 400	400 - 7200	---	199	2371	A-207, A-208	
9-22	0 - 400	400 - 7200	---	196	2371	A-209, A-210	
9-23	0 - 400	400 - 7200	---	213	1669	A-211, A-212	A

a. A--Adjustment made in simplified model initial condition to account for AFW overfeed from 0 to 400 s.

TABLE 26. RESULTS OF THREE STUCK-OPEN STEAM LINE PORVS AT FULL POWER

Sequence Number	Initiating Event	Power Level	Failures	Detailed Model Calculation Used	Simplified Model Variation Used (From 3.3.3)
9-41	Reactor trip	Full	3 stuck-open SPORVs	9-41	Base
9-42	Reactor trip	Full	3 stuck-open SPORVs, AFW overfill	9-41	Base
9-43	Reactor trip	Full	3 stuck-open SPORVs, charging not throttled	9-41	Base
9-44	Reactor trip	Full	3 stuck-open SPORVs, AFW overfill, charging not throttled	9-41	Base
9-45	Reactor trip	Full	3 stuck-open SPORVs, AFW overfeed	9-41	Base
9-46	Reactor trip	Full	3 stuck-open SPORVs, AFW overfeed, AFW overfill	9-41	Base
9-47	Reactor trip	Full	3 stuck-open SPORVs, AFW overfeed, charging not throttled	9-41	Base

Sequence Number	Methods Used, Periods in Seconds: Detailed Model	Simplified Model	Hand Calculations	Minimum RV DC Temperature (°F)	Maximum Subsequent RV DC Pressure (psia)	Figures Showing Plotted Results	Notes[a]
9-41	0 - 400	400 - 7200	---	268	1615	A-247, A-248	
9-42	0 - 400	400 - 7200	---	251	1517	A-249, A-250	
9-43	0 - 400	400 - 7200	---	262	2371	A-251, A-252	
9-44	0 - 400	400 - 7200	---	244	2371	A-253, A-254	
9-45	0 - 400	400 - 7200	---	272	1490	A-255, A-256	A
9-46	0 - 400	400 - 7200	---	259	1632	A-257, A-258	A
9-47	0 - 400	400 - 7200	---	265	2371	A-259, A-260	A

a. A--Adjustment made in simplified model initial condition to account for AFW overfeed from 0 to 400 s.

model were adjusted by increasing the steam generator secondary mass for the extra AFW which would have been injected over the periods when the detailed model was applied.

Results of these sequences show a continuous cooldown of the primary system. The cooldown rate slows considerably when the turbine-driven AFW is terminated due to low secondary pressure. The cooldown rate slows even more when all AFW is throttled because of liquid spilling to the steam lines, or 40% level attainment, depending on the sequence description.

The continual primary system fluid shrinkage, caused by the cooldown, limits the repressurization to about 11.7 MPa (1700 psia) except in sequences involving a failure to throttle charging. For these sequences, the primary system pressure increases to the opening setpoint pressure of the pressurizer PORV, thus increasing the severity of the sequence.

The severity of the results usually varied directly with the size of the steam break. Minimum temperatures of the reactor vessel downcomer fluid within each break size category were calculated for the sequence involving failure to throttle AFW and charging. The smallest break size (one SDV) resulted in a minimum temperature of 412 K (282°F). As the break size increased, the minimum temperature decreased: 391 K (244°F) for three steam PORVs, 389 K (241°F) for two SDVs, 377 K (220°F) for three SDVs, and 364 K (196°F) for five SDVs.

The most severe Group B sequence was 9-22, involving five stuck-open SDVs with failures to throttle AFW and charging. The minimum temperature of the reactor vessel downcomer for this sequence was 364 K (196°F); the maximum subsequent pressure was 16.35 MPa (2371 psia).

5.4 Conclusions

Group B includes 30 sequences involving steam breaks that symmetrically affect all three steam generators. Sequences in this group were in general found to be severe because auxiliary feedwater continues for a significant time. The severity of results was found to be sensitive

to break size, with the larger break size producing the most severe results. Furthermore, sequences in which charging fails to throttle are more severe than those in which charging throttles, because primary system pressure is driven up to the pressurizer PORV opening setpoint pressure.

The most severe Group B sequence was 9-22, involving five stuck-open SDVs with failures to throttle AFW and charging. The minimum temperature of the reactor vessel downcomer for this sequence was 364 K (196°F); the maximum subsequent pressure was 16.35 MPa (2371 psia).

6. GROUP C RESULTS: SECONDARY-SIDE BREAKS AFFECTING TWO STEAM GENERATORS, OR THREE ASYMMETRICALLY AFFECTED STEAM GENERATORS

The following sections present the results of thermal-hydraulic analyses for sequences in Group C of Table 15 using the methods presented in Section 3 to determine reactor vessel downcomer fluid pressure and temperature histories. Section 6.1 defines the group and describes the controlling phenomena. Section 6.2 discusses the results for sequences containing combinations of stuck-open steam PORV and steam dump valves. Section 6.3 discusses results for sequences containing two stuck-open steam PORVs. Conclusions covering all group C sequences appear in Section 6.4. To facilitate referencing of data, plotted results, showing the pressure and temperature histories, are organized in numerical sequence order in Appendix A rather than appearing within this section. Due to the large number of sequences investigated in this report, detailed discussions of thermal-hydraulic processes for each sequence are not practical. Such discussions are documented for representative sequences in Reference 1.

6.1 Group C Definitions

Group C includes all sequences controlled by multiple steam line breaks that asymmetrically affect the steam generators. This includes two categories: (1) sequences involving one or more stuck-open steam PORVs and one stuck-open steam dump valve (SDV), and (2) sequences involving two stuck-open steam PORVs. In the first category, the open SDV affects all steam generators the same, but an open PORV affects only the steam generator on which it is located. In the second category, the steam generator without the open PORV is not affected, due to the action of the steam line check valves.

Sequences in Group C are controlled by primary system heat removal to each affected steam generator and heat addition from steam generators in flowing unaffected loops, core decay heat, and stored energy in metal heat structures.

For all Group C sequences, turbine-driven AFW is initiated because the levels in at least two steam generators decrease sufficiently, due to loss of secondary inventory. Turbine-driven AFW is assumed unavailable when all steam generator secondary pressures are below 0.69 MPa (100 psia). Below this pressure insufficient steam is available to run the turbine-driven AFW pumps.

6.2 Combinations of Stuck-Open Steam PORVs and Steam Dump Valves

Sequences 5-14, 5-15, 5-17 through 5-20, and 7-9 through 7-11 involve combinations of steam PORV and steam dump valve (SDV) breaks (see Tables 5 and 7). Sequences from Table 5 begin from full-power conditions, and sequences from Table 7 begin from hot standby conditions. The sequences further differ by: (1) AFW isolation at 10 min, (2) AFW feed rate (normal or overfeed), and (3) throttling of charging flow when the pressurizer setpoint level is attained. Each of the sequences involves one stuck-open SDV and either one or two stuck-open steam PORVs.

Table 27 presents the analysis methods used, and gives an indication of results for, sequences involving combination of SDV and steam PORV breaks. Plotted results for all sequences appear in Appendix A.

A detailed model calculation was performed for one sequence at each power level: Sequences 5-14, 5-19, 7-9, and 7-11. Simplified model calculations were performed, using the base simplified model described in Section 3.3.2.

The base simplified model contains a single secondary volume that represents three symmetrically affected steam generators. By using this model, it is assumed that the asymmetries in these sequences may be ignored. This approach was found to adequately model the phenomena present in these sequences because the total primary system heat removal rates calculated using the detailed and simplified models agreed to within 5%. The simplified model was initialized based on conditions present at the end of the detailed model calculation: the secondary system mass was conserved, and the secondary pressure was the average of the three steam generator pressures which had been calculated with the detailed model.

TABLE 27. RESULTS OF COMBINATION SDV AND STEAM PORV BREAKS

Sequence Number	Initiating Event	Power Level	Failures	Detailed Model Calculation Used	Simplified Model Variation Used (From 3.3.3)
5-14	Small SLB	Full	1 stuck-open SDV	5-14	Base
5-15	Small SLB	Full	1 stuck-open SDV, charging not throttled	5-14	Base
5-17	Small SLB	Full	1 stuck-open SDV, AFW not isolated	5-14	Base
5-18	Small SLB	Full	1 stuck-open SDV, AFW overfeed	5-14	Base
5-19	Small SLB	Full	2 stuck-open SDVs	5-19	Base
5-20	Small SLB	Full	2 stuck-open SDVs, AFW not isolated	5-19	Base
7-9	Small SLB	HSB	1 stuck-open SDV	7-9	Base
7-10	Small SLB	HSB	1 stuck-open SDV, AFW not isolated	7-9	Base
7-11	Small SLB	HSB	2 stuck-open SDVs	7-11	Base

Sequence Number	Methods Used, Periods in Seconds: Detailed Model	Simplified Model	Hand Calculations	Minimum RV DC Temperature (°F)	Maximum Subsequent RV DC Pressure (psia)	Figures Showing Plotted Results	Notes[a]
5-14	0 - 600	600 - 4400	4400 - 7200	311	2371	A-91, A-92	A
5-15	0 - 600	600 - 4000	4000 - 7200	309	2371	A-93, A-94	A
5-17	0 - 600	600 - 7200	---	277	1553	A-97, A-98	
5-18	0 - 600	600 - 5400	5400 - 7200	303	2371	A-99, A-100	A
5-19	0 - 600	600 - 3400	3400 - 7200	286	2371	A-101, A-102	A
5-20	0 - 600	600 - 7200	---	244	1618	A-103, A-104	
7-9	0 - 600	600 - 7200	---	239	1508	A-147, A-148	
7-10	0 - 600	600 - 7200	---	229	1470	A-149, A-150	
7-11	0 - 600	600 - 7200	---	209	1537	A-151, A-152	

a. A--Hand calculations performed over heatup phase following steam generator dryout.

System responses fell into two categories, depending on AFW isolation. The first category, sequences that include AFW isolation at 10 min, proceeded through a cooldown phase, followed by a steam generator dryout phase, and finally a heatup phase. For convenience, the responses during the heatup phases were hand calculated based on the effect of decay heat on the primary system liquid and metal masses. For sequences in this category, AFW restoration will eventually be required (after the two hour period of interest) for long-term plant cooling. For first-category sequences initiated from full power (5-14, 5-15, 5-18, and 5-19), steam generator dryout was calculated before two hours. For first-category sequences initiated from hot standby (7-9 and 7-11), steam generator dryout did not occur before two hours, so the heatup phase was not entered.

The second category, sequences where AFW is not isolated at 10 min (5-17, 5-20, and 7-10), proceeded through a continuous primary system cooldown. The cooldown rate slows considerably as AFW throttling begins when the steam generator narrow-range levels have recovered to 40%.

Due to the low decay heat and large break size, Sequence 7-11 (two steam PORVs and one SDV open at hot standby) reached the lowest reactor vessel downcomer temperature for the group, 371 K (209°F). The primary system repressurization for this (and other hot standby) sequences was limited because decay heat was insufficient to overcome the steam generator heat removal. Thus, the primary fluid was not heated; thermal expansion of the fluid was avoided; and only a slight repressurization was encountered.

Sequences 5-14, 5-15, 5-18, and 5-19 encountered heatup phases, and the resulting primary fluid thermal expansion was sufficient to force the primary system pressure to the pressurizer PORV opening setpoint pressure. Of these sequences, 5-19 (two steam PORVs and one SDV open at full power) had the lowest reactor vessel downcomer fluid temperature, 414 K (286°F).

The severity of sequences in this group is limited because significant primary system repressurization requires AFW termination, and this action terminates the cooldown. The sequences would be made much more severe if AFW isolation were delayed until 20 min, for example, thus allowing an extra 10 min of cooldown before the severe repressurization.

6.3 Two Stuck-Open Steam PORVs

Sequences 9-33 through 9-40 involve two stuck-open steam PORVs. All of these sequences are initiated from full-power conditions; AFW is not terminated by the operator at 10 min. The sequences differ by: (1) AFW delivery rate (normal or overfeed), (2) throttling of AFW to the unaffected steam generator (USG) at 40% narrow-range level or when liquid spills to the steam lines, and (3) throttling of charging flow when the pressurizer setpoint level is attained. For these sequences, AFW can be fed to the unaffected steam generators only when AFW is throttled to the ASGs; in all these sequences this throttling occurs when liquid spills from the ASG to the steam line. Due to the low ASG pressures, all AFW flows from the headers to the ASGs, unless the AFW valves to the ASGs are closed, in which case AFW can be delivered to the USG.

Table 28 presents the analysis methods used, and gives an indication of results for, sequences involving two stuck-open steam PORVs. Plotted results for all sequences appear in Appendix A.

A detailed model calculation was performed for the initial portion of Sequence 9-33. Simplified model calculations were performed using model Variation 6, as described in Section 3.3.3.

For sequences not involving AFW overfilling of the USG (9-33, 9-35, 9-37, and 9-39), it was found that the USG was not sufficiently cooled to cause heat removal from the primary system to the USG. In these sequences, the USG response does not affect the primary system response. The simplified model results were used to two hours.

For sequences involving AFW overfilling of the USG (9-34, 9-36, 9-38, and 9-40), the USG was found to affect the primary system response. For these sequences, hand calculations were performed, from the time of AFW throttling to the ASG to two hours after the initiating event, to predict the primary system response. The hand calculations first determined the USG temperature response, assuming no interaction between it and the primary system. Next the primary system temperature response was

TABLE 28. RESULTS OF TWO STUCK-OPEN STEAM PORVS

Sequence Number	Initiating Event	Power Level	Failures	Detailed Model Calculation Used	Simplified Model Variation Used (From 3.3.3)
9-33	Reactor trip	Full	2 stuck-open SPORVs	9-33	6
9-34	Reactor trip	Full	2 stuck-open SPORVs, AFW overfill	9-33	6
9-35	Reactor trip	Full	2 stuck-open SPORVs, charging not throttled	9-33	6
9-36	Reactor trip	Full	2 stuck-open SPORVs, AFW overfill, charging not throttled	9-33	6
9-37	Reactor trip	Full	2 stuck-open SPORVs, AFW overfeed	9-33	6
9-38	Reactor trip	Full	2 stuck-open SPORVs, AFW overfeed, AFW overfill	9-33	6
9-39	Reactor trip	Full	2 stuck-open SPORVs, AFW overfeed, charging not throttled	9-33	6
9-40	Reactor trip	Full	2 stuck-open SPORVs, AFW overfeed, AFW overfill, charging not throttled	9-33	6

Sequence Number	Methods Used, Periods in Seconds: Detailed Model	Simplified Model	Hand Calculations	Minimum RV DC Temperature (°F)	Maximum Subsequent RV DC Pressure (psia)	Figures Showing Plotted Results	Notes[a]
9-33	0 - 665	665 - 7200	---	295	1833	A-231, A-232	
9-34	0 - 665	665 - 3748	3748 - 7200	287	1782	A-233, A-234	A
9-35	0 - 665	665 - 7200	---	285	2371	A-235, A-236	
9-36	0 - 665	665 - 3049	3049 - 7200	280	2371	A-237, A-238	A
9-37	0 - 665	665 - 7200	---	298	1823	A-239, A-240	
9-38	0 - 665	665 - 2337	2337 - 7200	290	1816	A-241, A-242	A
9-39	0 - 665	665 - 7200	---	289	2371	A-243, A-244	
9-40	0 - 665	665 - 2340	2340 - 7200	283	2371	A-245, A-246	A

a. A--Hand calculation performed for period following AFW throttling to the ASGs.

determined by taking a mass-weighted average of the USG temperature response and the primary system temperature response, assuming no USG effects (note for Sequence 9-34 the latter response is available from the Sequence 9-33 simplified model calculation, etc.). The primary system pressure response was then calculated, using the primary system fluid thermal contraction rate, HPI and charging volume addition rates, and an adiabatic compression/expansion model of the pressurizer.

Results for all of these sequences indicate a continuous cooldown of the primary system until AFW throttling to the ASG occurs. After throttling, the temperature responses are virtually flat. The effect of overfilling the USG was found to be minor in all cases.

Minimum temperatures for all sequences were within 5 K (9°F) of 416 K (289°F). The more severe sequences were 9-35, 9-36, 9-39, and 9-40 in which the failure to throttle charging caused the primary system pressure to rise to the pressurizer PORV opening setpoint pressure.

6.4 Conclusions

Group C includes 17 sequences involving only steam breaks asymmetrically affecting the steam generators.

For sequences involving three asymmetrically affected steam generators, representing the phenomena with a single, simplified model secondary cell, using average conditions, was found to be a satisfactory method.

For sequences that specify AFW isolation at 10 min, AFW restoration will eventually be required to maintain long-term cooling. The restoration would occur after the two-hour period of interest for PTS. For these sequences, PTS severity would be increased significantly if AFW isolation were delayed, for example, to 20 min.

The sequences initiated from hot standby conditions were found to lead to colder reactor coolant temperatures than those initiated from full-power conditions. For these sequences, however, repressurization of the primary system is limited because decay heat is insufficient to overcome steam generator heat removal. Thus the primary system fluid is not heated, does not expand, and does not compress the pressurizer steam bubble which would increase the pressure. Had one or more of the hot standby sequences in this group specified a failure to throttle charging flow, the PTS severity of these sequences would be increased significantly.

The effects on the results of failure to throttle AFW to USGs and AFW overfeed (at a higher rate) were found to be minor.

7. GROUP D RESULTS: REACTOR TRIP FROM FULL POWER WITH NO PRIMARY- OR SECONDARY-SIDE BREAKS

The following sections present the results of thermal-hydraulic analyses for sequences in Group D of Table 15, using the methods presented in Section 3 to determine reactor vessel downcomer fluid pressure and temperature histories. Section 7.1 defines the group; Section 7.2 presents the results and conclusions for sequences in Group D. To facilitate referencing of data, plotted results showing the pressure and temperature histories are organized in numerical sequence in Appendix A rather than appearing within this section. Due to the large number of sequences investigated in this report, detailed discussions of thermal-hydraulic processes for each sequence are not practical. Such discussions are documented for representative sequences in Reference 1.

7.1 Group D Definition

Group D includes sequences not controlled by primary- or secondary-side breaks. These sequences are initiated by reactor trip from full-power conditions, with only minor failures within the auxiliary feedwater (AFW) or pressurizer level control systems. These sequences are controlled by near-normal decay heat removal to the steam generators including continued reactor coolant pump operation. Sequences 9-1, and 9-49 through 9-55 are included in Group D.

7.2 Results and Conclusions

Sequences 9-1 and 9-49 through 9-55 are initiated by a reactor trip from full-power conditions. The sequences differ by: (1) the AFW delivery rate (normal or overfeed), (2) AFW throttling at 40% steam generator level, or when liquid spillover to the steam lines occurs, and (3) throttling of charging flow when the pressurizer setpoint level is attained.

As indicated in Table 9, Sequences 9-49 through 9-55 include one main feedwater regulating valve (MFWRV) failing open. This failure is inconsequential since the main feedwater isolation valves (MFWIVs) close

automatically following a turbine trip. The MFWIVs are in series with the MFWRVs, so that the closure of either valve effectively isolates main feedwater to a steam generator.

Table 29 presents the analysis methods used, and gives an indication of results for, the Group D sequences. Plotted results for all sequences appear in Appendix A.

A detailed model calculation was performed for the initial portion of Sequence 9-1, a reactor trip from full power. By 900 s this sequence reaches a steady, post-trip condition expected to continue through 7200 s.

The hand calculations for Sequences 9-49 through 9-55 were used to modify the results calculated for Sequence 9-1 for effects of AFW overfill, AFW overfeed, and failure to throttle charging.

For AFW overfill, a new steam generator secondary temperature response was calculated, assuming no heat addition from the primary system. Next, a new primary system temperature response was calculated by taking a mass-weighted average of the primary system temperatures from Sequence 9-1 and the new secondary temperatures. Since the charging system flow capacity is sufficient to make up for the primary system fluid shrinkage rate caused by the cooldown, the primary system pressure response from Sequence 9-1 remains applicable.

For AFW overfeed, the results of Sequence 9-1 were modified to account for the faster AFW delivery rate. This modification affected only the timing of events.

For the failure to throttle charging, the primary system pressure response was modified, based on an adiabatic compression of the pressurizer steam bubble. The pressurizer inflow was assumed to be the net makeup flow, i.e., charging flow minus letdown flow.

TABLE 29. RESULTS OF REACTOR TRIP SEQUENCES WITH MINOR FAILURES

Sequence Number	Initiating Event	Power Level	Failures	Detailed Model Calculation Used	Simplified Model Variation Used (From 3.3.3)
9-1	Reactor trip	Full	None	9-1	--
9-49	Reactor trip	Full	AFW overfill	9-1	--
9-50	Reactor trip	Full	Charging not throttled	9-1	--
9-51	Reactor trip	Full	AFW overfill, charging not throttled	9-1	--
9-52	Reactor trip	Full	AFW overfeed	9-1	--
9-53	Reactor trip	Full	AFW overfeed, AFW overfill	9-1	--
9-54	Reactor trip	Full	AFW overfeed, charging not throttled	9-1	--
9-55	Reactor trip	Full	AFW overfeed, AFW overfill, charging not throttled	9-1	--

	Methods Used, Periods in Seconds						
Sequence Number	Detailed Model	Simplified Model	Hand Calculations	Minimum RV DC Temperature (°F)	Maximum Subsequent RV DC Pressure (psia)	Figures Showing Plotted Results	Notes[a]
9-1	0 - 900	---	900 - 7200	546	2296	A-167, A-168	A
9-49	0 - 900	---	900 - 7200	446	2371	A-263, A-264	B
9-50	0 - 900	---	900 - 7200	546	2371	A-265, A-266	B
9-51	0 - 900	---	900 - 7200	446	2371	A-267, A-268	B
9-52	0 - 900	---	900 - 7200	546	2296	A-269, A-270	B
9-53	0 - 900	---	900 - 7200	446	2371	A-271, A-272	B
9-54	0 - 900	---	900 - 7200	546	2371	A-273, A-274	B
9-55	0 - 900	---	900 - 7200	446	2371	A-275, A-276	B

a. A--Steady detailed model results at 900 s extrapolated to 7200 s.

B--Pressure and temperature responses calculated by hand based on detailed model results and effects of the AFW or charging failures. Specifics of these hand calculations appear in the text.

All Group D sequences are of limited severity for PTS. Only sequences involving an AFW overfill (9-49, 9-51, 9-53, and 9-55) result in a cooldown. The minimum primary system temperature for these sequences is 503 K (446°F). Primary system pressures rise to the pressurizer PORV opening setpoint pressure, except in Sequences 9-1 and 9-52, where they remain at the normal operating point.

8. GROUP E RESULTS: MAIN FEEDWATER OVERFILL

The following sections present the results of thermal-hydraulic analyses for the sequence in Group E of Table 15 using the methods presented in Section 3 to determine reactor vessel downcomer fluid pressure and temperature history. Section 8.1 defines the group, and Section 8.2 presents the results and conclusions for the sequence in Group E. To facilitate referencing of data, plotted results showing the pressure and temperature histories are organized in numerical sequence in Appendix A, rather than appearing within this section.

8.1 Group E Definition

Group E includes only Sequence 9-56 (see Table 9). This sequence is initiated by a reactor trip from full power, followed by failing open of the main feedwater regulating and isolation valves on one feed line, and a failure to trip main feedwater pumps until high steam generator level is reached.

Sequence 9-56 was considered by itself as a separate group because it is the only sequence from Tables 1-12 that involves the overfilling of the steam generators using the main feedwater system.

8.2 Results and Conclusions

Table 30 presents the analysis methods used, and gives an indication of results for, Sequence 9-56. Plotted results for this sequence appear as Figures A-277 and A-278 in Appendix A.

A detailed model calculation was run for the initial 440 s of Sequence 9-56. The affected steam generator level reached its high level setpoint at 60 s, at which time main feedwater pumps were tripped. After 440 s, the primary system pressure response assumes charging flow continues until the normal pressurizer level is attained. Afterwards, the pressurizer spray and heaters control the pressure normally. The primary

TABLE 30. RESULTS OF MAIN FEEDWATER OVERFILL

Sequence Number	Initiating Event	Power Level	Failures	Detailed Model Calculation Used	Simplified Model Variation Used (From 3.3.3)
9-56	Reactor trip	Full	MFWRV and MFWIV fail open on same line, MFW pumps do not trip after turbine trip	9-56	--

Sequence Number	Methods Used, Periods in Seconds: Detailed Model	Methods Used, Periods in Seconds: Simplified Model	Methods Used, Periods in Seconds: Hand Calculations	Minimum RV DC Temperature (°F)	Maximum Subsequent RV DC Pressure (psia)	Figures Showing Plotted Results
9-56	0 - 440	---	440 - 7200	544	2296	A-277, A-278

system temperature response assumes that the minor cooldown, underway at 440 s, continues until 559 K (547°F) is reached. Thereafter, the steam dump system is assumed to maintain that temperature.

Sequence 9-56 is of very low PTS severity. Significant cooling capability is not present during a main feedwater overfill transient because main feedwater is warm, relative to AFW, and because the main feedwater pump trip is reached quickly.

9. GROUP F RESULTS: LOCAS

The following sections present the results of thermal-hydraulic analyses for sequences in Group F of Table 15, using the methods presented in Section 3 to determine reactor vessel downcomer fluid pressure and temperature histories. Section 9.1 defines the group and describes the controlling phenomena. Within Group F, sequences were organized by break size into Small Break Loss-of-Coolant Accident (SBLOCA) and Medium Break Loss-of-Coolant Accident (MBLOCA) subgroups. Sections 9.2 and 9.3 present the results for these subgroups. Conclusions covering all Group F sequences appear in Section 9.4. To facilitate referencing of data, plotted results showing the pressure and temperature histories are organized in numerical sequence order in Appendix A, rather than appearing within this section. Due to the large number of sequences investigated in this report, detailed discussions of thermal-hydraulic processes for each sequence are not practical. Such discussions are documented for representative sequences in Reference 1.

9.1 Group F Definition

Group F includes all sequences controlled by a primary-side break that is not isolated. Within this section, a "small" break refers to a single stuck-open pressurizer PORV. A "medium" break refers to a 0.0635-m (2.5-in.) diameter hole located at the bottom of the C loop hot leg. The controlling phenomena are the mass and energy flow due to the break and the ECC, possible draining of the U-tubes which stops natural circulation, and the heat transfer to, or from, the steam generators.

9.2 SBLOCAs

Sequences 1-1 through 1-4 involve SBLOCAs at full power (see Table 1). Sequences 3-1 and 3-2 involve SBLOCAs at hot standby (see Table 3). The sequences differ based on: (1) initial power level, (2) AFW flowing at normal or high (overfeed) rates, and (3) AFW throttled or not throttled (overfill) at 40% narrow range level.

Table 31 presents the analysis methods used, and gives an indication of the results of, the SBLOCA sequences. Downcomer pressure and temperature results for all sequences are presented in Appendix A.

Detailed model calculations were performed for initial portions of Sequences 1-1 (see Scenario 7 in Reference 1) and 3-1. The two detailed model calculations provided a starting point for the simplified model, which was then used to calculate the remaining portions of the sequences. For each sequence, Table 31 shows the time periods calculated with the detailed model and with the simplified model. Variation 7 of the simplified model, which is described in Section 3.3.3, was used for all the SBLOCA sequences.

Both motor-driven and turbine-driven AFW were initiated in the SBLOCA sequences at full power (Sequences 1-1 through 1-4). Only motor-driven AFW was initiated in the sequences at hot standby (Sequences 3-1 and 3-2). The turbine-driven AFW was not initiated at hot standby because the large initial secondary mass and the small steam production prevented the steam generator levels from decreasing enough to actuate the turbine-driven system.

Both the simplified and detailed model calculations indicated that the ECC flow was generally as large as, or exceeded, the flow through the PORV. Consequently, the loops did not drain, the U-tubes were not voided, and natural circulation flow was maintained. Eventually the pressurizer filled with subcooled liquid, causing the flow out the PORV to increase and the RCS to depressurize. The energy removed by the combination of ECC and flow through the PORV eventually exceeded the decreasing core decay power. Consquently, the steam generators became heat sources rather than heat sinks. The steam generators became heat sources at 2000 s in Sequence 1-1 (SBLOCA at full power) and at 300 s in Sequence 3-1 (SBLOCA at hot standby). Even after the steam generators became heat sources, natural circulation continued in the liquid-filled loops because of the effects of core decay power, ECC, and loop transient times. At hot standby (Sequence 3-1), the natural circulation flow was about twice the HPI flow.

TABLE 31. RESULTS: SBLOCAS

Sequence Number	Initiating Event	Power Level	Failures	Detailed Model Calculation Used	Simplified Model Variation Used (From 3.3.3)
1-1	SBLOCA	Full	None	11-1	7
1-2	SBLOCA	Full	AFW overfill	11-1	7
1-3	SBLOCA	Full	AFW overfeed	11-1	7
1-4	SBLOCA	Full	AFW overfeed, AFW overfill	11-1	7
3-1	SBLOCA	HSB	None	3-1	7
3-2	SBLOCA	HSB	AFW overfill	3-1	7

Sequence Number	Methods Used, Periods in Seconds: Detailed Model	Simplified Model	Hand Calculations	Minimum RV DC Temperature (°F)	Maximum Subsequent RV DC Pressure (psia)	Figures Showing Plotted Results	Notes[a]
1-1	0 - 600	600 - 7200	---	474	924	A-1, A-2	
1-2	0 - 600	600 - 7200	---	382	578	A-3, A-4	
1-3	0 - 150	150 - 7200	---	473	919	A-5, A-6	
1-4	0 - 150	150 - 7200	---	381	576	A-7, A-8	
3-1	0 - 790	790 - 7200	---	279	679	A-59, A-60	A
3-2	0 - 400	400 - 7200	---	246	393	A-61, A-62	A

a. A--Turbine-driven AFW not initiated.

The downcomer pressure and temperature were significantly affected by the initial core power level, as shown in Table 31. As expected, the calculated downcomer pressures and temperatures at hot standby were lower than at full power. The minimum downcomer temperatures were from 76 to 108 K (136 to 195°F) lower at hot standby; the final pressures were about 1.4 MPa (200 psi) lower at hot standby. AFW overfill also significantly affected the downcomer pressure and temperatures. The final downcomer temperatures and pressures were from 18 to 51 K (33 to 92°F), and about 2 MPa (300 psi) lower when AFW was not throttled at 40% level. However, the AFW overfeed did not significantly affect either the downcomer temperature or pressure at two hours.

The SBLOCA sequences resulted in relatively low downcomer pressures and high downcomer temperatures. Consequently, the SBLOCA sequences are probably not severe PTS transients.

9.3 MBLOCAs

Sequences 2-1 through 2-4 (see Table 2) involve MBLOCAs at full power. Sequence 4-1 (see Table 4) involves a MBLOCA at hot standby. The sequences differ based on: (1) initial power level, (2) AFW flowing at normal or high (overfeed) rates, and (3) AFW throttled or not throttled (overfill) at 40% narrow range level.

Table 32 presents the analysis methods used, and gives an indication of the results for, the MBLOCA sequences. Downcomer pressure and temperature results for all sequences are presented in Appendix A.

Detailed model calculations were performed for the initial portions of the MBLOCA at full power with no other failures (Sequence 2-1), the MBLOCA at full power with AFW overfill (Sequence 2-2), and the MBLOCA at hot standby (Sequence 4-1). These sequences were extrapolated from the end of the detailed model calculation to two hours using hand calculations. The thermal-hydraulic conditions at two hours were based on steady state mass and energy balances. At steady state, the loops were assumed to be stagnated. The ECC was assumed to flow into the downcomer, past the core,

TABLE 32. RESULTS: MBLOCAS

Sequence Number	Initiating Event	Power Level	Failures	Detailed Model Calculation Used	Simplified Model Variation Used (From 3.3.3)
2-1	MBLOCA	Full	None	2-1	--
2-2	MBLOCA	Full	AFW overfill	2-2	--
2-3	MBLOCA	Full	AFW overfeed	2-1	--
2-4	MBLOCA	Full	AFW overfeed, AFW overfill	2-1	--
4-1	MBLOCA	HSB	None	4-1	--

	Methods Used, Periods in Seconds						
Sequence Number	Detailed Model	Simplified Model	Hand Calculations	Minimum RV DC Temperature (°F)	Maximum Subsequent RV DC Pressure (psia)	Figures Showing Plotted Results	Notes[a]
2-1	0 - 2800	---	2800 - 7200	100	144	A-37, A-38	A
2-2	0 - 1730	---	1730 - 7200	100	144	A-39, A-40	A
2-3	0 - 200	---	200 - 7200	100	144	A-41, A-42	B
2-4	0 - 200	---	200 - 7200	100	144	A-43, A-44	B
4-1	0 - 1740	---	1740 - 7200	100	144	A-63, A-64	A,C

a. A--Hand calculations based on extrapolations of detailed model results and steady state mass and energy balances.

B--Hand calculations based on engineering judgement and steady state mass and energy balances.

C--Turbine-driven AFW not initiated.

into the C loop hot leg, and out through the break. The break thus removed the heated ECC and consequently the core decay power. The results of Sequences 2-3 and 2-4 were based primarily on engineering judgement relative to the effects of AFW overfeed. The effect of AFW overfeed was to cool and depressurize the primary system more rapidly during the overfeed period, but the effect was assumed to diminish after AFW termination. Thus, the results of Sequence 2-3 (AFW overfeed) eventually converged with those of Sequence 2-1 (normal AFW flow). Similarly, the results of Sequence 2-4 (AFW overfeed and overfill) eventually converged with those of Sequence 2-2 (AFW overfill). The simplified model was not used to calculate any of the MBLOCA sequences shown in Table 32 because it was not thought to be adequate for these sequences. The break was big enough to cause voiding in the loops, but the simplified model could not calculate a mixture level within the RCS and hence could not reliably calculate the void fraction or flow at the break.

The MBLOCA was large enough to cause voiding in the loops, depressurize the RCS, and eventually remove the core decay power. Loop flows stagnated due to voiding in the U-tubes. The final conditions for all the sequences were the same. The final downcomer temperature was 311 K (100°F), slightly higher than the ECC temperature, due to bypass flows within the vessel. The final downcomer pressure was 1.0 MPa (144 psia), slightly below the LPI shutoff head. The effect of AFW was to depressurize and cool the RCS and to promote natural circulation flow. Consequently, AFW overfill significantly affected the pressure and temperature histories. The effect of AFW overfeed was judged to be insignificant. The sequences at full power were judged to be more severe relative to PTS than at hot standby because the RCS depressurized more slowly at full power.

9.4 Conclusions

The SBLOCA sequences are probably not severe relative to PTS. The MBLOCA sequences are more severe for PTS because the downcomer temperatures dropped nearly to the ECC temperature. However, the downcomer pressures were relatively low for the MBLOCA sequences.

Both the small and medium breaks were large enough to remove more energy than was generated by core decay heat within two hours. Consequently, the RCS could be cooled and depressurized with either break size. The thermal-hydraulic results were sensitive to AFW overfill, but not to AFW overfeed. Loop flow stagnation was calculated to occur only when the U-tubes were voided. The medium break was large enough to void the U-tubes but the small break was not. Loop flow stagnation did not occur when the loops were liquid full, even with heat transfer from the steam generators to the RCS.

10. GROUP G RESULTS: LOCA COMBINED WITH SYMMETRIC SECONDARY-SIDE BREAKS

The following sections present the results of thermal-hydraulic analyses for sequences in Group G of Table 15, using the methods presented in Section 3 to determine reactor vessel downcomer fluid pressure and temperature histories. Section 10.1 defines the group and describes the controlling phenomena. Section 10.2 presents the results for all the sequences in this group. Conclusions related to Group G sequences appear in Section 10.3. To facilitate referencing of data, plotted results showing the pressure and temperature histories are organized in numerical sequence order in Appendix A, rather than appearing within this section. Due to the large number of sequences investigated in this report, detailed discussions of thermal-hydraulic processes for each sequence are not practical. Such discussions are documented for representative sequences in Reference 1.

10.1 Group G Definition

Group G includes all sequences controlled by a primary-side break, combined with a secondary-side break that symmetrically affected all three steam generators. A SBLOCA refers to a LOCA initiated by a single, stuck-open pressurizer PORV. A MBLOCA refers to a LOCA initiated by a 0.0635-m (2.5-in.) diameter hole located at the bottom of the C loop hot leg. The secondary-side break was initiated when a given number of steam dump valves (SDVs) stuck open as they were demanded to open by the plant control system following reactor trip. The controlling phenomena were the mass and energy flows due to the primary-side break and ECC, and heat transfer to the steam generators.

10.2 Results

Sequences involving SBLOCAs combined with stuck-open SDVs are numbered 1-5 through 1-8 (see Table 1) for full power and 3-2 (see Table 3) for hot standby. Sequences involving MBLOCAs combined with stuck-open SDVs

are numbered 2-5 through 2-8 (see Table 2). The sequences differ based on primary-side break size, number of stuck-open SDVs, and initial core power level.

Table 33 summarizes the analysis methods used and gives the results of the LOCA sequences combined with stuck-open SDVs. Downcomer pressure and temperature results for all sequences are presented in Appendix A.

Detailed model calculations were performed for initial portions of Sequences 1-5, 1-8, 2-5, and 2-8. These sequences represented SBLOCAs and MBLOCAs with either one or five stuck-open SDVs which bounded the possible secondary break sizes. The detailed model calculations provided a starting point for the simplified model, which was used to calculate the bounding sequences to two hours. Variation 7 of the simplified model was used for the SBLOCA sequences, and Variation 8 was used for the MBLOCA sequences. These models were described in Section 3.3.3. Results for Sequences 1-6, 1-7, 2-6, and 2-7, which represented LOCAs with either two or three stuck-open SDVs, were obtained by interpolating the results of the appropriate bounding sequences. For example, results for Sequence 1-7 were obtained by interpolating Sequences 1-5 and 1-8, and Sequence 2-7 was interpolated from Sequences 2-5 and 2-8. Sequence 3-3, a SBLOCA at hot standby with one stuck-open SDV was not analyzed because of probabilistic considerations. SDVs are only occasionally demanded to open at hot standby. Once the PORV is opened, it can remove core decay power, and the SDVs would not be demanded to open. The probability that both the pressurizer PORV and a SDV would stick open at hot standby was thought to be relatively low. Consequently, Sequence 3-3 was not analyzed, and results for this sequence do not appear in either Table 33 or Appendix A.

The calculations showed that significant voiding of the RCS loops did not occur for the combinations of primary- and secondary-side breaks. Even in the MBLOCA sequences, only minor voiding occurred in the loops early in the transients. The loops filled with subcooled liquid due to the combined effects of ECC and the steam line break. With the loops liquid-full, natural circulation was maintained throughout the transient. In the SBLOCA sequences, a pressure plateau was reached near 5.5 MPa (800 psia) as the

TABLE 33. RESULTS OF LOCAS COMBINED WITH SYMMETRIC SECONDARY-SIDE BREAKS

Sequence Number	Initiating Event	Power Level	Failures	Detailed Model Calculation Used	Simplified Model Variation Used (From 3.3.3)
1-5	SBLOCA	Full	1 stuck-open SDV	1-5	7
1-6	SBLOCA	Full	2 stuck-open SDVs	--	--
1-7	SBLOCA	Full	3 stuck-open SDVs	--	--
1-8	SBLOCA	Full	5 stuck-open SDVs	1-8	7
2-5	MBLOCA	Full	1 stuck-open SDV	2-5	8
2-6	MBLOCA	Full	2 stuck-open SDVs	--	--
2-7	MBLOCA	Full	3 stuck-open SDVs	--	--
2-8	MBLOCA	Full	5 stuck-open SDVs	2-8	8
3-3	SBLOCA	HSB	1 stuck-open SDV	--	--

Sequence Number	Methods Used, Periods in Seconds: Detailed Model	Simplified Model	Hand Calculations	Minimum RV DC Temperature (°F)	Maximum Subsequent RV DC Pressure (psia)	Figures Showing Plotted Results	Notes[a]
1-5	0 - 750	750 - 7200	---	233	338	A-9, A-10	
1-6	---	---	0 - 7200	218	328	A-11, A-12	A
1-7	---	---	0 - 7200	203	317	A-13, A-14	A
1-8	0 - 350	350 - 7200	---	171	300	A-15, A-16	
2-5	0 - 550	550 - 7200	---	210	144	A-45, A-46	
2-6	---	---	0 - 7200	198	144	A-47, A-48	A
2-7	---	---	0 - 7200	185	144	A-49, A-50	A
2-8	0 - 540	540 - 7200	---	160	144	A-51, A-52	
3-3	---	---	---	---	---	---	B

a. A--Results interpolated from the sequences with 1 stuck-open SDV and 5 stuck-open SDVs.

B--Sequence deleted because of probability considerations.

pressurizer filled. The RCS then depressurized relatively rapidly until accumulator flow was intiated, at which time the rate of depressurization slowed. In the MBLOCA sequences, the pressure dropped relatively quickly to the LPI shutoff head and then remained constant. The downcomer temperatures were controlled by the heat transfer to the steam generators and by the ECC mixing. The cooldown rate always decreased when AFW was throttled. Table 33 shows that the minimum downcomer temperatures decreased as the break size on either the primary- or secondary-side increased. All the sequences were at or near steady state at two hours.

The sequences with combinations of LOCAs and steam line breaks had temperature histories that were similar to the corresponding steam line break, while the pressure histories were more similar to the corresponding LOCA. Because of the additional cooling due to the LOCA, the final downcomer temperatures in the combination LOCA and steam line break sequences were colder by 17 to 56 K (30 to 100°F) than in the steam line break of the same size described in Section 5. Because of the depressurization due to the LOCA, the final downcomer pressures in the combination LOCA and steam line break sequences were more than 8 MPa (1200 psi) lower than the steam line break of the same size. Because the pressures were much less, the combination LOCA and steam line break sequences are not as potentially severe PTS transients as the steam-line-break-only sequences.

10.3 Conclusions

The sequences with combined LOCAs and steam line breaks were probably not as severe relative to PTS as the corresponding steam line break transients described in Section 5.

The sequences with combined LOCAs and steam line breaks had temperature histories that were similar to a steam line break, while the pressure histories were more similar to those of a LOCA.

Natural circulation was maintained in these sequences because the RCS loops did not void.

11. GROUP H RESULTS: LOCA COMBINED WITH ASYMMETRIC SECONDARY-SIDE BREAKS

The following sections present the results of thermal-hydraulic analyses for sequences in Group H of Table 15, using the methods presented in Section 3 to determine reactor vessel downcomer fluid pressure and temperature histories. Section 11.1 defines the group and describes the controlling phenomena. Section 11.2 presents the results for all the sequences in this group. Conclusions related to Group H sequences appear in Section 11.3. To facilitate referencing of data, plotted results showing the pressure and temperature histories are organized in numerical sequence order in Appendix A rather than appearing within this section. Due to the large number of sequences investigated in this report, detailed discussions of thermal-hydraulic processes for each sequence are not practical. Such discussions are documented for representative sequences in Reference 1.

11.1 Group H Definition

Group H includes all sequences controlled by a primary-side break combined with an asymmetric secondary-side break. A small primary-side break refers to a single, stuck-open pressurizer PORV. A medium primary-side break refers to a 0.0635-m (2.5-in.) diameter hole located at the bottom of the C loop hot leg. The secondary-side break was initiated by either one stuck-open steam PORV on the A loop, or two stuck-open steam PORVs, one on the A loop and one on the C loop. The steam PORVs stuck open when the reactor tripped. The controlling phenomena are the mass and energy flows, due to the primary-side break and ECC, and heat transfer to the steam generators.

11.2 Results

Sequences 1-9 through 1-12 (see Table 1) involve SBLOCAs combined with stuck-open steam PORVs at full power. Sequences 2-9 through 2-11 (see Table 2) involve MBLOCAs combined with stuck-open steam PORVs at full

power. The sequences differ based on primary-side break size, number of stuck-open steam PORVs, AFW flowing at normal or high (overfeed) rates, and AFW throttled or not throttled (overfill) at 40% narrow-range level.

Table 34 summarizes the analysis methods used, and gives an indication of the results for, the LOCA sequences combined with stuck-open steam PORVs. Downcomer pressure and temperature results for all sequences are presented in Appendix A.

Detailed model calculations were performed for initial portions of Sequences 1-9, 1-12, 2-9, and 2-11. These sequences represented SBLOCAs and MBLOCAs with either one or two stuck-open steam PORVs. The detailed model calculations provided a starting point for the simplified model, which was then used to calculate all the Group H sequences to two hours. Four different simplified models were used, one for each combination of primary-side and secondary-side break size. These simplified models are described in Section 3.3.3.

The calculations showed that the affected RCS loop did not significantly void, even in the MBLOCA sequences. The affected loop subcooled, due to the combined effects of ECC and the steam line break. Natural circulation flow was maintained through the affected loop, but less flow passed through the unaffected loops, as described in Scenario 3 of Reference 1. The asymmetric steam line break caused a less rapid cooldown of the unaffected loops, which, when coupled with the low RCS pressure due to the LOCA, caused flashing in the unaffected loops. The flashing caused voiding in the U-tubes and stopped natural circulation in the unaffected loops.

In the SBLOCA sequences, a pressure plateau was reached near 5.5 MPa (800 psia) as the pressurizer filled. The RCS then depressurized relatively rapidly until the accumulator flow was initiated and the unaffected loops began flashing, slowing the rate of depressurization. In the MBLOCA sequences, the RCS pressure dropped quickly to the LPI shutoff head and then remained constant. The downcomer temperatures were controlled by heat transfer to the steam generators and by ECC mixing. The

TABLE 34. RESULTS OF LOCAS COMBINED WITH ASYMMETRIC SECONDARY-SIDE BREAKS

Sequence Number	Initiating Event	Power Level	Failures	Detailed Model Calculation Used	Simplified Model Variation Used (From 3.3.3)
1-9	SBLOCA	Full	1 stuck-open steam PORV	1-9	9
1-10	SBLOCA	Full	1 stuck-open steam PORV, AFW overfill	1-9	9
1-11	SBLOCA	Full	1 stuck-open steam PORV, AFW overfeed	1-9	9
1-12	SBLOCA	Full	2 stuck-open steam PORVs	1-12	10
2-9	MBLOCA	Full	1 stuck-open steam PORV	2-9	11
2-10	MBLOCA	Full	1 stuck-open steam PORV, AFW overfill	2-9	11
2-11	MBLOCA	Full	2 stuck-open steam PORVs	2-11	12

Sequence Number	Methods Used, Periods in Seconds: Detailed Model	Methods Used, Periods in Seconds: Simplified Model	Methods Used, Periods in Seconds: Hand Calculations	Minimum RV DC Temperature (°F)	Maximum Subsequent RV DC Pressure (psia)	Figures Showing Plotted Results
1-9	0 - 480	480 - 7200	---	224	368	A-17, A-18
1-10	0 - 480	480 - 7200	---	224	356	A-19, A-20
1-11	0 - 150	150 - 7200	---	224	368	A-21, A-22
1-12	0 - 475	475 - 7200	---	217	358	A-23, A-24
2-9	0 - 715	715 - 7200	---	190	144	A-53, A-54
2-10	0 - 625	625 - 7200	---	190	144	A-55, A-56
2-11	0 - 580	580 - 7200	---	203	144	A-57, A-58

downcomer temperatures were significantly affected by ECC mixing, since the downcomer temperatures could be more than 56K (100°F) lower than the average RCS temperatures. Table 34 shows that the minimum downcomer temperature decreased as the primary-side break size increased. For the SBLOCA, as expected, the minimum downcomer temperature was lower with two stuck-open steam PORVs than with one. However, for the MBLOCA, the minimum downcomer temperature was 7 K (13°F) lower with one stuck-open steam PORV than with two. The 7 K (13°F) temperature difference was less than the uncertainty in the results and may not be real. The average RCS temperature was colder with two stuck-open steam PORVs, but the calculated ECC mixing effect (see Variation 7, Section 3.3.3) was larger with one stuck-open steam PORV, because the total loop flow was assumed to be smaller with two stagnated unaffected loops than with one unaffected loop. All the sequences were at or near steady state at two hours.

The combination LOCA and stuck-open steam PORV sequences were probably not as severe PTS transients as were the corresponding steam line break sequences described in Sections 4.3 and 6.3. Because of the additional cooling due to the LOCA, the minimum downcomer temperatures were more than 56 K (100°F) lower in the combination sequences than in the corresponding steam line break sequences. However, because of the depressurization due to the LOCA, the final downcomer pressures were more than 9 MPa (1300 psia) lower in the combination sequences than in the corresponding steam line break sequences.

11.3 Conclusions

The sequences with combined LOCAs and stuck-open steam PORVs were probably not as severe relative to PTS as were the corresponding sequence with only stuck-open steam PORVs.

The combination of a LOCA and stuck-open PORVs resulted in sustained natural circulation in the affected loops. However, natural circulation in the unaffected loops stopped, due to voiding of the U-tubes.

12. GROUP I RESULTS: ISOLATABLE PRIMARY-SIDE BREAKS

The following sections present the results of thermal-hydraulic analyses for sequences in Group I of Table 15, using the methods presented in Section 3 to determine reactor vessel downcomer fluid pressure and temperature histories. Section 12.1 defines the group and describes the controlling phenomena. Section 12.2 presents the results for all the sequences in this group. Conclusions related to Group I sequences appear in Section 12.3. To facilitate referencing of data, plotted results showing the pressure and temperature histories are organized in numerical sequence order in Appendix A rather than appearing within this section. Due to the large number of sequences investigated in this report, detailed discussions of thermal-hydraulic processes for each sequence are not practical. Such discussions are documented for representative sequences in Reference 1.

12.1 Group I Definition

Group I consists of sequences initiated by a primary-side break which is isolated at a given time during the two-hour period. A "small" break refers to a single stuck-open pressurizer PORV. A "medium" break refers to a 0.0635-m (2.5-in.) diameter hole located at the bottom of the C loop hot leg. The controlling phenomena are the mass and energy flows due to the primary-side break and ECC, and heat transfer to the steam generators.

12.2 Results

Sequences 11-1 through 11-4 (see Table 11) involve isolatable SBLOCAs. Sequences involving isolatable MBLOCAs are numbered 12-1 through 12-4 (see Table 12). All the isolatable LOCA sequences were initiated at full core power. The sequences differ based on primary-side break size, time of LOCA isolation, and throttling of the charging flow. The charging flow was either throttled automatically by the pressurizer level control system, throttled manually 3 minutes after the level control should have throttled charging, or was not throttled.

Table 35 summarizes the analysis methods used and gives an indication of the results of the isolatable LOCA sequences. Downcomer pressure and temperature results for all sequences are presented in Appendix A.

Detailed model calculations were performed for initial portions of Sequences 11-1 (see Section 7 in Reference 1) and 12-3. These sequences represented a SBLOCA isolated at 10 min and a MBLOCA isolated at 20 min. The detailed model calculations provided starting points for the simplified models, which were used to calculate the Group I sequence to two hours. Variation 7 of the simplified model was used to calculate the SBLOCA sequences, and Variation 8 was used for the MBLOCA sequences. These models were described in Section 3.3.3. The results of simplified model calculations were augmented with hand calculations. The last 5000 s of Sequence 11-1 was based on the end point of the calculation, at which time the downcomer pressure and temperature were held constant by the pressurizer code safety valve and steam dump valves, respectively. The effect of maximum charging on SBLOCA Sequences 11-2 and 11-4 prior to 600 s (the end of the appropriate detailed calculation) was calculated by hand and added to the results of the detailed calculation. A corresponding hand calculation was not required for the MBLOCA sequence because maximum charging was calculated with the detailed model.

Table 35 shows that the isolatable LOCA sequences were not potentially severe PTS transients. The final downcomer pressures were relatively high, but the downcomer remained warm. The MBLOCAs that were isolated at 20 min were the most severe transients, but reached a minimum downcomer temperature of only 453 K (356°F). Because the downcomer temperature was decreasing at the time of break isolation, the MBLOCA sequences would have been more severe if the break had been isolated later. For example, if the break had been isolated at 40 min rather than 20 min, the minimum downcomer temperature would have been more than 56 K (100°F) lower.

In the SBLOCAs and MBLOCAs that were isolated at 10 min, the downcomer remained warm because natural circulation was maintained throughout the transients. Natural circulation was maintained because the RCS loops were not highly voided and the steam generators acted as heat sinks throughout

TABLE 35. RESULTS OF ISOLATABLE PRIMARY-SIDE BREAKS

Sequence Number	Initiating Event	Power Level	Failures or Conditions	Detailed Model Calculation Used	Simplified Model Variation Used (From 3.3.3)
11-1	SBLOCA	Full	Isolated at 10 min.	11-1	7
11-2	SBLOCA	Full	Isolated at 10 min., charging not throttled	11-1	7
11-3	SBLOCA	Full	Isolated at 20 min.	11-1	7
11-4	SBLOCA	Full	Isolated at 20 min., charging not throttled	11-1	7
12-1	MBLOCA	Full	Isolated at 10 min., charging not throttled	2-1	8
12-2	MBLOCA	Full	Isolated at 10 min., charging throttled manually	2-1	8
12-3	MBLOCA	Full	Isolated at 20 min., charging not throttled	12-3	8
12-4	MBLOCA	Full	Isolated at 20 min., charging throttled manually	12-3	8

Sequence Number	Methods Used, Periods in Seconds			Minimum RV DC Temperature (°F)	Maximum Subsequent RV DC Pressure (psia)	Figures Showing Plotted Results	Notes[a]
	Detailed Model	Simplified Model	Hand Calculations				
11-1	0 - 2200	---	2200 - 7200	517	2554	A-351, A-352	A,B
11-2	0 - 600	600 - 7200	150 - 600	501	2554	A-353, A-354	A,C
11-3	0 - 600	600 - 7200	---	516	2554	A-355, A-356	A
11-4	0 - 600	600 - 7200	150 - 600	506	2554	A-357, A-358	A,C
12-1	0 - 600	600 - 7200	---	463	2371	A-359, A-360	D
12-2	0 - 600	600 - 7200	---	463	1942	A-361, A-362	
12-3	0 - 1930	1930 - 7200	---	356	2371	A-363, A-364	D
12-4	0 - 1930	1930 - 7200	---	356	1914	A-365, A-366	

a. A--Maximum pressure determined by code safety valve opening setpoint.

B--Hand calculations based on extrapolations of detailed model results.

TABLE 35. (continued)

C--The effect of not throttling charging was calculated by hand and superimposed on results from the detailed model.

D--Maximum pressure determined by PORV opening setpoint.

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the transient. In the MBLOCAs that were isolated at 20 min, the steam generators were heat sources, the U-tubes were draining, and the loop flows were stagnating at the time of isolation. After isolation, HPI filled the loops, the RCS heated to above the steam generator temperatures, a large natural circulation flow was established, and the downcomer was warmed.

Prior to the isolation of the break, the calculated results were based on the LOCA sequences that were described in Section 9. All the sequences responded similarly after the isolation of the break. After break isolation, the cooling mechanism, due to energy flow through the break, no longer existed. The downcomer temperature increased after AFW was throttled, and the RCS pressure exceeded the HPI shutoff head. The temperature increased until the SDVs opened, and then held the temperature constant. The isolation of the break caused the downcomer pressure to increase as HPI and charging filled the RCS. The downcomer pressure was ultimately limited by the setpoint of either the pressurizer PORV or code safety valves. In the MBLOCA sequences with charging throttled, the RCS pressure did not reach the PORV setpoint within two hours. However, the PORV setpoint was reached in all the SBLOCA sequences. This difference between the SBLOCA and MBLOCA sequences was probably not real but was due to different calculation techniques. In the MBLOCA sequences, the pressure increased due to the compression of the steam bubbles in the pressurizer and upper head. In the SBLOCA sequences, the steam bubbles were condensed rather than compressed, and most of the pressure increase occurred after the RCS became liquid-full. The complete condensation of the steam bubbles in the SBLOCAs was probably not reasonable, and the uncertainty of these sequences, described in Section 15, was increased to compensate for this. The uncertainty in the calculated pressure of the isolatable SBLOCAs was not significant relative to PTS.

Throttling the charging significantly affected the downcomer pressure, but had only a minor effect on the downcomer temperature response. The primary effect of break size was on the amount of void in the RCS at the time of break isolation. Of course, the RCS was more highly voided in the MBLOCAs than in the SBLOCAs.

12.3 Conclusions

The isolatable LOCA sequences were not potentially severe PTS transients. The MBLOCAs that were isolated at 20 minutes were the most severe transients because the loops were voiding, causing loop flow stagnation and colder downcomer temperatures at the time of break isolation. The MBLOCA sequences would have been more severe if isolation of the break had been delayed. Failure to throttle charging significantly affected downcomer pressure response, but did not significantly affect the downcomer temperature response.

13. GROUP J RESULTS: STEAM GENERATOR TUBE RUPTURES

The following sections present the results of thermal-hydraulic analyses for sequences in Group J of Table 15, using the methods presented in Section 3 to determine reactor vessel downcomer fluid pressure and temperature histories. Section 13.1 defines the group, and Section 13.2 presents the results and conclusions for sequences in Group J. To facilitate referencing of data, plotted results showing the pressure and temperature histories are organized in numerical sequence order in Appendix A rather than appearing within this section. Due to the large number of sequences investigated in this report, detailed discussions of thermal-hydraulic processes for each sequence are not practical. Such discussions are documented for representative sequences in Reference 1.

13.1 Group J Definition

Group J includes Sequences 10-1 through 10-5 (see Table 10). All sequences involve the double-ended rupture of a single steam generator tube with the reactor operating at hot standby conditions. Sequence 10-1 (Scenario 13 in Reference 1) is a base sequence. Sequences 10-2 through 10-5 are modifications of Sequence 10-1. The sequences vary due to different operator responses to the events and to hardware failures.

The base sequence (10-1) requires the operator to isolate steam and feedwater to the affected steam generator (ASG) and cool the primary system to a specified temperature using the steam dump valves. The operator then equalizes the primary and secondary system pressures by opening the pressurizer PORV in two cycles. Sequences 10-2 through 10-5 assume failures such as stuck-open steam dump valves or a pressurizer PORV, or a failure to throttle HPI and charging.

13.2 Results and Conclusions

Table 36 presents the analysis methods used and gives an indication of results for all Group J sequences. Plotted results for all sequences appear in Appendix A.

TABLE 36. RESULTS OF STEAM GENERATOR TUBE RUPTURES

Sequence Number	Initiating Event	Power Level	Failures and Changes	Detailed Model Calculation Used	Simplified Model Variation Used (From 3.3.3)
10-1	SGTR	HSB	None	10-1	--
10-2	SGTR	HSB	SDV fails to close for 10 minutes	10-2	--
10-3	SGTR	HSB	PORV sticks open for 10 minutes on first opening	10-1	13
10-4	SGTR	HSB	Second PORV opening does not occur	10-1	--
10-5	SGTR	HSB	Second PORV opening does not occur and HPI/charging not throttled	10-1	13

Sequence Number	Methods Used, Periods in Seconds: Detailed Model	Simplified Model	Hand Calculations	Minimum RV DC Temperature (°F)	Maximum Subsequent RV DC Pressure (psia)	Figures Showing Plotted Results	Notes[a]
10-1	0 - 2500	---	2500 - 7200	386	1080	A-341, A-342	A
10-2	0 - 2500	---	2500 - 7200	277	1080	A-343, A-344	A
10-3	0 - 990	990 - 1590	1590 - 7200	394	950	A-345, A-346	B
10-4	0 - 1000	---	1000 - 7200	406	1020	A-347, A-348	A
10-5	0 - 990	990 - 2635	2635 - 7200	406	1401	A-349, A-350	B

a. A--Steady conditions at end of detailed model calculation were extrapolated to 7200 s.

B--Steady conditions at end of simplified model calculation were extrapolated to 7200 s.

A detailed model calculation was performed over the initial portions of Sequences 10-1 and 10-2. The detailed model was modified by representing the broken tube with four control volumes and two break paths. A discussion of this modification appears in Section 12.2 of Reference 1. To address the unique phenomena of a steam generator tube rupture event, Variation 13 of the simplified model was developed (see Section 3.3.3).

Steady temperature and pressure conditions were reached for all sequences, using either detailed or simplified model calculations. These steady conditions were extrapolated to continue to 7200 s.

For all tube rupture sequences, the effect of the break flow limits the potential for repressurization of the primary system. Even with a failure to throttle HPI and charging (Sequence 10-5), the repressurization was limited to 9.66 MPa (1401 psia). At that pressure, the break flow equaled the HPI and charging injection flow, and the pressure stabilized. For Sequences 10-1 through 10-4, the repressurization was limited by the action of the affected steam generator PORV and safety valves.

The cooldown rates in Sequences 10-1, and 10-3 through 10-5 were similar, with minor differences resulting from the different pressurizer PORV control cirteria. The cooldown in Sequence 10-2 was more severe as a result of the steam dump valve sticking open for 10 min. The minimum reactor vessel downcomer fluid temperature for Sequence 10-2 was 409 K (277°F).

Results for the steam generator tube rupture sequences were, in general, strongly affected by the assumptions for operator action. For example, if it is assumed the operator opens the steam dump valves and cools the primary system to a pre-specified temperature, then the temperature (but not its rate of change) is already defined by the assumption.

14. HEAT TRANSFER COEFFICIENTS

Another contractor has been enlisted by ORNL to determine, for the sequences presented in this report, the heat transfer coefficients on the inside surface of the reactor vessel wall. ORNL indicated, however, that it would facilitate their fracture mechanics analyses if a preliminary estimate of the heat transfer coefficient was made for each sequence. Reference 1 documents the reactor vessel wall heat transfer coefficient histories for twelve PTS sequences. INEL agreed to indicate, for each sequence, which of the Reference 1 scenarios has a heat transfer coefficient that is most representative.

Table 37 lists the scenario in Reference 1 that has the most representative heat transfer coefficient for each of the sequences presented in Sections 4 through 13 of this report. Representative heat transfer coefficients for the remaining sequences correspond to the sequence equivalency shown in Table 16. The assignments shown in Table 37 were made primarily by comparing the reactor vessel downcomer flow conditions in the detailed model calculations from this report with those from Reference 1, and selecting the Reference 1 scenario that most closely matched the conditions. Typically, this process resulted in the selection of the scenario having the most representative break size or initiating event, secondary heat sink behavior (source or sink), and reactor coolant pump trip control (trip and timing of trip). For sequences involving significant changes in the heat transfer coefficient as a transient progresses, more weighting was given to the end-state conditions in selecting the representative scenario.

TABLE 37. REPRESENTATIVE HEAT TRANSFER COEFFICIENTS

Heat Transfer Coefficient for Sequence	Is Best Represented By Reference 1 Scenario Number	Comments
1-1	11	Subtract 1 hr. from the time in Scenario 11
1-2	11	Subtract 1 hr. from the time in Scenario 11
1-3	11	Subtract 1 hr. from the time in Scenario 11
1-4	11	Subtract 1 hr. from the time in Scenario 11
1-5	3	Subtract 1000s from the time in Scenario 3
1-6	3	Subtract 1000s from the time in Scenario 3
1-7	3	Subtract 1000s from the time in Scenario 3
1-8	3	Subtract 1000s from the time in Scenario 3
1-9	3	Subtract 1000s from the time in Scenario 3
1-10	3	Subtract 1000s from the time in Scenario 3
1-11	3	Subtract 1000s from the time in Scenario 3
1-12	3	Subtract 1000s from the time in Scenario 3
2-1	6	
2-2	6	
2-3	6	
2-4	6	
2-5	6	
2-6	6	
2-7	6	
2-8	6	
2-9	6	
2-10	6	
2-11	6	
3-1	11	Subtract 1 hr. from the time in Scenario 11
3-2	11	Subtract 1 hr. from the time in Scenario 11
4-1	6	
5-1	5	RCPs not tripped
5-14	11	Subtract 1 hr. from the time in Scenario 11
5-15	11	Subtract 1 hr. from the time in Scenario 11
5-17	4	
5-18	11	Subtract 1 hr. from the time in Scenario 11
5-19	11	Subtract 1 hr. from the time in Scenario 11
5-20	4	
6-1	11	Subtract 1 hr. from the time in Scenario 11
6-2	11	Subtract 1 hr. from the time in Scenario 11
6-3	11	Subtract 1 hr. from the time in Scenario 11
6-4	11	Subtract 1 hr. from the time in Scenario 11
6-5	2	
6-6	2	
6-7	2	
6-8	11	Subtract 1 hr. from the time in Scenario 11
6-9	2	
7-1	5	RCPS not tripped
7-2	5	RCPS not tripped
7-3	5	RCPS not tripped
7-4	3	
7-5	3	

TABLE 37. (continued)

Heat Transfer Coefficient for Sequence	Is Best Represented By Reference 1 Scenario Number	Comments
7-6	3	
7-7	5	RCPS not tripped
7-8	3	
7-9	11	Subtract 1 hr. from time in Scenario 11
7-10	3	
7-11	11	Subtract 1 hr. from time in Scenario 11
7-12	3	
8-1	1	AFW isolated
8-2	1	AFW isolated
8-3	1	AFW isolated
8-4	2	
8-5	1	AFW isolated
8-6	2	
9-1	5	RCPs not tripped
9-2	4	
9-3	4	
9-4	4	
9-5	4	
9-6	4	
9-7	4	
9-8	4	
9-9	4	
9-10	4	
9-11	4	
9-12	4	
9-13	4	
9-14	4	
9-15	4	
9-16	4	
9-17	4	
9-18	4	
9-19	4	
9-20	4	
9-21	4	
9-22	4	
9-23	4	
9-24	4	
9-25	5	RCPs not tripped
9-26	5	RCPs not tripped
9-27	5	RCPs not tripped
9-28	5	RCPs not tripped
9-29	5	RCPs not tripped
9-30	5	RCPs not tripped
9-31	5	RCPs not tripped
9-32	5	RCPs not tripped
9-33	4	
9-34	4	

TABLE 37. (continued)

Heat Transfer Coefficient for Sequence	Is Best Represented By Reference 1 Scenario Number	Comments
9-35	4	
9-36	4	
9-37	4	
9-38	4	
9-39	4	
9-40	4	
9-41	4	
9-42	4	
9-43	4	
9-44	4	
9-45	4	
9-46	4	
9-47	4	
9-49	5	RCPs not tripped
9-50	5	RCPs not tripped
9-51	5	RCPs not tripped
9-52	5	RCPs not tripped
9-53	5	RCPs not tripped
9-54	5	RCPs not tripped
9-55	5	RCPs not tripped
9-56	5	RCPs not tripped
10-1	9	
10-2	9	
10-3	9	
10-4	9	
10-5	9	
11-1	7	
11-2	7	
11-3	7	
11-4	7	
12-1	6 ($t \leq$ 1000s), 7 ($t \geq$ 1000s)	
12-2	6 ($t \leq$ 1000s) 7 ($t \geq$ 1000s)	
12-3	6 ($t \leq$ 2500s), 7 ($t \geq$ 2500s)	
12-4	6 ($t \leq$ 2500s) 7 ($t \geq$ 2500s)	

15. UNCERTAINTIES

A comprehensive study of applicable uncertainties for all 183 sequences is beyond the scope of this work. Estimates of uncertainty are required, however, to facilitate the ORNL fracture mechanics analyses. This section presents estimates of uncertainty for the reactor vessel downcomer fluid temperatures and pressures documented in this report.

Estimates of uncertainties were developed separately for each category of sequence. First, for each category the controlling thermal-hydraulic phenomena were identified, and an estimate of uncertainty from the effects of each phenomenon was made. Second, an estimate of the uncertainty due to the use of the simplified model was determined primarily from the benchmark comparisons shown in Section 3.4. Third, a representative uncertainty due to measurements (including initial conditions) was determined. Next, the components of uncertainty (phenomena, simplified model, and measurement) were combined using the root-sum-square method. Finally, the uncertainty was limited based on physical constraints, if any. For example, an uncertainty range was not allowed to project a reactor vessel downcomer temperature below 305 K (90°F), the temperature of the HPI.

The uncertainties, in general, were stated as a function of the value of the variable itself. For example, the uncertainty in temperature is a function of temperature. This was found to be a useful approach since the components of uncertainty frequently varied in magnitude as a sequence progressed. As an example, during a large steam line break sequence the uncertainty in reactor vessel downcomer temperature is initially small and related to measurement error or minor uncertainties in initial conditions. As the transient begins, a relatively large uncertainty in the break flow dominates and the overall uncertainty rises as the temperature falls. After some time, the affected steam generator reaches a stable low-pressure operating point, with heat removal controlled by a stable steam generator secondary saturation temperature, and the overall uncertainty diminishes.

In situations where comparisons between experimental and code-calculated data were available, the comparisons were used directly as the indication of uncertainty due to controlling phenomena.

Table 38 shows the uncertainties resulting from the above evaluation. The maximum temperature uncertainty is ±39 K (±70°F) and the maximum pressure uncertainty is ±1.38 MPa (±200 psi). Uncertainty ranges shown in the table should be considered as 95% confidence ranges; that is, the actual value will usually fall within the stated range, but not always. For temperatures and pressures between values shown in the table, linear interpolation may be used.

TABLE 38. UNCERTAINTIES IN REACTOR VESSEL DOWNCOMER FLUID TEMPERATURE AND PRESSURE

Sequences Involving Small Steam Line Breaks Only

(All of Tables 5 and 7, plus Sequences 9-2 through 9-47, 9-57 through 9-65, 9-75 through 9-80, 9-86 and 9-87.

Temperature (°F)	(Error (°F)
550	± 5
400	± 25
200	± 25
100	± 5

Pressure (psia)	Error (psi)
2400	± 20
1800	± 125
1500	± 160
1400	± 165
1000	± 180
800	± 180

Sequences Involving Large Steam Line Breaks Only (All of Tables 6 and 8)

Temperature (°F)	(Error (°F)
550	± 5
400	+ 25, -40
200	+ 25, -50
100	± 5

Pressure (psia)	Error (psi)
2400	± 20
1800	± 125
1500	± 160
1400	± 165
1000	± 180
800	+ 180, -220

Sequences Not Involving Primary or Secondary Breaks (Sequences 9.1, 9-48 through 9-56, 9-66 through 9-14, and 9-81 through 9-85.)

Temperature Error + 5°F
Pressure Error ± 20 psi

TABLE 38. (continued)

Sequences Involving Non-Isolatable LOCAs or Steam Generator Tube Rupture (Tables 1, 2, 3, 4 and 10)

Temperature (°F)	(Error (°F)
550	± 5
400	± 40
300	± 70
160	± 70
100	+ 40, -10

Pressure (psia)	Error (psi)
2500	± 50
1000	± 50
800	± 160
400	± 160
200	± 75
140	± 25

Sequences Involving Isolatable Small Break LOCAs (Table 11)

Temperature (°F)	(Error (°F)
550	± 5
400	± 40
300	± 70
160	± 70
100	+ 40, -10

Pressure (psia)	Error (psi)
2500	+50, -50
2000	+100, -250
1200	+250, -100
1000	+250, -50
800	± 160
400	± 160
200	± 75
140	± 25

Sequences Involving Isolatable Medium Break LOCAs (Table 12)

Temperature (°F)	(Error (°F)
550	± 5
400	± 40
300	± 70
160	± 70
100	+ 40, -10

TABLE 38. (continued)

Pressure (psia)	Error (psi)
2500	± 25
2250	± 25
1800	± 200
800	± 160
400	± 160
200	± 75
140	± 25

16. CONCLUSIONS

Reactor vessel downcomer fluid pressure and temperature responses were determined for 183 sequences of interest to the pressurized thermal shock study for the HBR-2 Pressurized Water Reactor.

A unique economical method was developed to accomplish this effort. The method used may be generally applicable to the thermal-hydraulic analysis of other large groups of sequences sharing common characteristics, such as the output of fault tree analyses. The method involved the combination of partial-length calculations, using a detailed RELAP5 model of the plant; full-length calculations using a simplified RELAP5 model; and hand calculations. The simplified model was constructed specifically to address the controlling phenomena of a group of sequences, as determined from analyses of detailed model calculations. The simplified model was then benchmarked against the detailed model to assure the validity of the simplification process. Sequence responses were typically determined by joining results using the detailed model over the initial portion of the sequence (when relatively complicated transient phenomena predominate) with results using the simplified model over the later portion of the sequence (when steady or quasi-steady phenomena predominate). Once a simplified model has been developed and qualified for a particular type of sequence (for example, stuck-open steam dump valves at full power), the model was then applied to other sequences of that type. Thus, the thermal-hydraulic responses of many sequences may be determined at a considerable cost savings, compared with performing a detailed model calculation for each sequence.

As a convenience for the reader, a preliminary indication of the most thermal-hydraulically severe sequences for PTS are summarized in Table 39. Judgements concerning PTS severity require consideration of probability and fracture mechanics aspects not considered in this report. These judgements will be made later at ORNL. Table 39 lists all sequences which have both reactor vessel downcomer temperatures below 394 K (250°F) and pressures above 3.45 MPa (500 psia). These ranges were arbitrarily chosen only to limit the size of the table and are not to be considered as limits of PTS severity.

TABLE 39. SUMMARY OF SEVERE SEQUENCES

Sequence Number	Initiating Event	Power Level	Failures or Conditions	Minimum RV DC Temperature (°F)	Maximum Subsequent RV DC Pressure (psia)
8-4	Large steam break	HSB	AFW not isolated	204	1947
8-6	Large steam break	HSB	AFW not isolated, AFW overfeed	201	1995
6-5	Large steam break	Full	AFW not isolated	229	1772
6-6	Large steam break	Full	AFW not isolated, charging not throttled	227	2371
6-7	Large steam break	Full	AFW not isolated, AFW overfill	229	1772
6-9	Large steam break	Full	AFW not isolated, AFW overfeed	229	1781
9-10	Reactor trip	Full	2 stuck-open SDVs, AFW overfill	248	1518
9-12	Reactor trip	Full	2 stuck-open SDVs, AFW overfill, charging not throttled	241	2371
9-14	Reactor trip	Full	3 stuck-open SDVs	234	1610
9-15	Reactor trip	Full	3 stuck-open SDVs, AFW overfill	226	1613
9-16	Reactor trip	Full	3 stuck-open SDVs, charging not throttled	225	2371
9-17	Reactor trip	Full	3 stuck-open SDVs, AFW overfill, charging not throttled	220	2371
9-19	Reactor trip	Full	5 stuck-open SDVs	203	1594
9-20	Reactor trip	Full	5 stuck-open SDVs, AFW overfill	201	1575
9-21	Reactor trip	Full	5 stuck-open SDVs, charging not throttled	199	2371
9-22	Rector trip	Full	5 stuck-open SDVs, AFW overfill, charging not throttled	196	2371
9-23	Reactor trip	Full	5 stuck-open SDVs, AFW overfeed	213	1669
9-44	Reactor trip	Full	3 stuck-open steam PORVs, AFW overfill, charging not throttled	244	2371
5-20	Small steam break	Full	2 stuck-open SDVs, AFW not isolated	244	1618
7-9	Small steam break	HSB	1 stuck-open SDV	239	1508
7-10	Small steam break	HSB	1 stuck-open SDV, AFW not isolated	229	1470
7-11	Small steam break	HSB	2 stuck-open SDVs	209	1537

Significant findings of the analyses in this report are presented below, arranged by sequence type.

Secondary-Side Breaks

Seventy-nine sequences involving only secondary-side breaks were investigated. These sequences contained one or more stuck-open steam PORVs, one or more stuck-open steam dump valves, double-ended main steam line breaks, and combinations of these. General findings were: (1) the larger break size produced colder reactor vessel downcomer temperatures; (2) a failure to throttle charging results in high primary system pressures; and (3) sequences involving isolation of all auxiliary feedwater result in high primary system pressures. The most severe sequence investigated in this report was Sequence 9-22. This sequence involves five stuck-open steam dump valves and failures to throttle auxiliary feedwater and charging following a reactor trip from full power. For Sequence 9-22 the minimum reactor vessel downcomer fluid temperature was 364 K (196°F), and the maximum subsequent pressure was 16.35 MPa (2371 psia).

For sequences initiated by a stuck-open steam line PORV at full power, an automatic reactor trip does not occur, thus precluding PTS concern. In such an instance, the operator may manually trip the reactor. Thermal-hydraulic responses have been provided for these sequences as well.

For sequences initiated by a reactor trip and followed by a stuck-open steam line PORV, the primary system pressure does not decline sufficiently to cause tripping of the reactor coolant pumps, thus minimizing the PTS concern.

Sequences involving large (double-ended) main steam line breaks are, in general, severe. However, severity is limited because, if the cooldown continues (auxiliary feedwater not isolated), then the resulting primary fluid thermal contraction prevents a severe primary system repressurization. Thus only sequences involving a failure to throttle charging result in both a very low temperature and a pressure at the pressurizer PORV opening setpoint.

Reactor Trip With Minor Failures

Nine sequences initiated by reactor trip, followed by minor control system failures, and involving no primary- or secondary-side breaks were investigated. All nine of the sequences were found not to be severe. The minimum temperatures [503 K (446°F)] were calculated for sequences involving steam generator overfill using auxiliary feedwater.

Primary-Side Breaks

Nineteen sequences involving only primary-side breaks were investigated. Sequences with a medium break LOCA resulted in reactor vessel downcomer temperatures a few degrees above the HPI temperature; however, the resulting primary system pressures were very low, thus limiting PTS concern. Sequences with a small break LOCA resulted in higher temperatures and higher pressures than corresponding sequences with a medium break LOCA.

Both the medium and small break sizes were large enough to remove more energy than that generated by decay heat within two hours. Consequently, the reactor coolant system could be cooled and depressurized with either break size. Loop flow stagnation was calculated to occur only when the U-tubes were voided. The medium break was capable of voiding the U-tubes, but the small break was not. Loop flow stagnation did not occur when the loops were liquid full, even with heat transfer from the steam generators to the primary system.

Sequences involving isolation of a primary-side break resulted in high primary system pressures. As a result of isolation during the early stage of the sequence, however, severe overcooling of the primary system did not occur before the time of isolation, therefore minimizing PTS concern.

Combination Primary- and Secondary-Side Breaks

Sixteen sequences involving combination of primary- and secondary-side breaks were investigated. In general, these sequences are not as severe as the corresponding sequences involving only the secondary-side break. The

sequences with combined breaks have temperature responses similar to a corresponding secondary-side-break-only sequence, and pressure responses similar to a corresponding primary-side-break-only sequence.

For sequences with a symmetric, secondary-side break (stuck-open steam dump valves), natural circulation continued in all loops. For sequences with an asymmetric, secondary-side break (one or two stuck-open steam line PORVs), natural circulation continued only in the affected loop or loops.

Sequences involving a medium, primary-side break resulted in lower reactor vessel downcomer temperatures and pressures than did the corresponding sequences with small, primary-side breaks.

For all sequences, resulting temperatures are very low [344 K (160°F) for Sequence 2-8 is the lowest]. However, resulting pressures are also low [2.54 MPa (368 psia) for Sequences 1-9 and 1-11 is the highest], thus limiting PTS concern for combined breaks.

Steam Generator Tube Ruptures

Five sequences involving the rupture of a single steam generator tube with the reactor at hot standby conditions were investigated. Repressurization of the primary system is limited by the broken tube. The minimum temperatures are essentially determined by the operator actions and failures specified in the sequence description. For example, if the operator is assumed to open the steam dump valve until a subcooling criterion is met, the magnitude of the cooldown is specified by the assumption.

The steam generator tube rupture sequences were, in general, not severe, except for Sequence 10-2 which involved a 10-min period with a stuck-open steam dump valve. This sequence resulted in a minimum temperature of 409 K (277°F) and a maximum subsequent pressure of 7.45 MPa (1080 psia).

17. REFERENCES

1. C. D. Fletcher, et al., "RELAP5 Thermal-Hydraulic Analyses of Pressurized Thermal Shock Sequences for the H. B. Robinson Unit 2 Pressurized Water Reactor," EG&G Idaho, Inc., EGG-SAAM-6476, December 1983 (NUREG in progress).

2. C. D. Fletcher, et al., "RELAP5 Thermal-Hydraulic Analyses of Pressurized Thermal Shock Sequences for the Oconee-1 Pressurized Water Reactor," EG&G Idaho, Inc., NUREG/CR-3761, EGG-2310, June 1984.

3. D. G. Hall and L. S. Czapary, "Tables of Homogeneous Equilibrium Critical Flow Parameters for Water in SI Units," EG&G Idaho, Inc., EGG-2056, September 1980.

APPENDIX A
PLOTTED RESULTS OF REACTOR VESSEL DOWNCOMER FLUID PRESSURES AND TEMPERATURES

APPENDIX A

PLOTTED RESULTS OF REACTOR VESSEL DOWNCOMER FLUID PRESSURES AND TEMPERATURES

This appendix contains plotted results showing the reactor vessel downcomer fluid pressure and temperature histories for each sequence analyzed. Results are shown for the 2-hr period immediately following the initiating event.

Figures in this section are arranged by numerical sequence, first by table number and second by the line number within each table (see Section 2). For example, Sequence 7-6 is the sixth sequence shown in Table 7, and, in this Appendix, results for Sequence 7-6 are found following those for Sequence 7-5.

The analyses presented in this report were performed using best-estimate modeling assumptions for plant conditions and responses to the events specified in the sequence descriptions. The reader is cautioned, however, that the sequence descriptions were based on extremely conservative assumptions concerning equipment malfunctions, operator actions and omissions, or combinations of these. Thus, while the analyses results represent the best-estimate plant responses to the sequences as defined, they do not necessarily represent the most probable plant responses to the sequence initiating events.

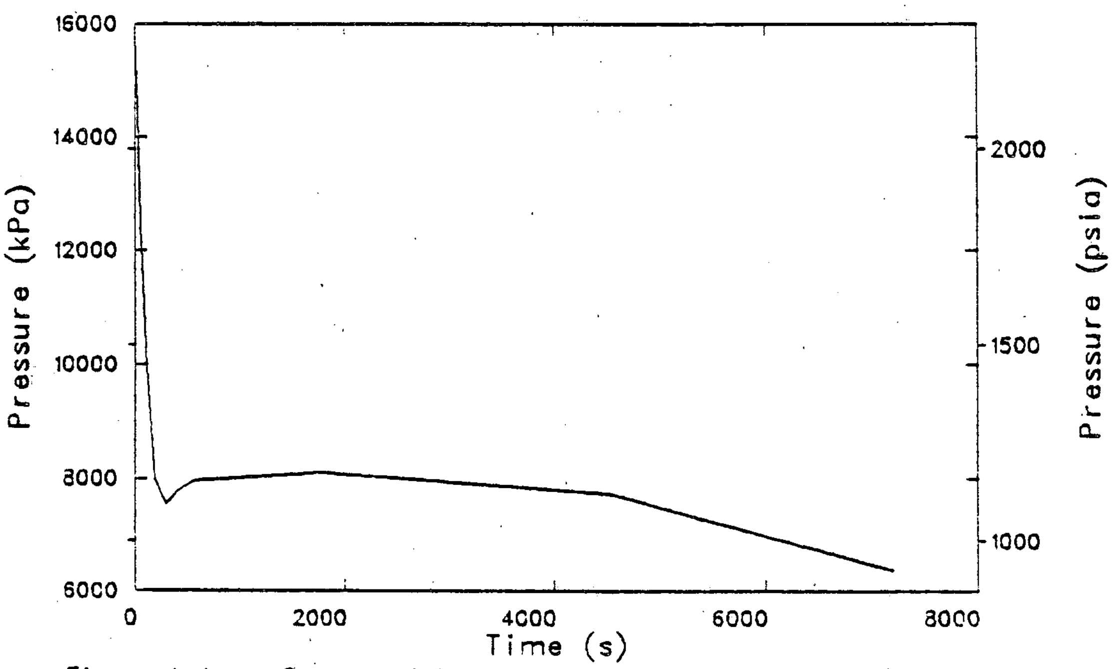


Figure A-1. Sequence 1-1 reactor vessel downcomer fluid pressure.

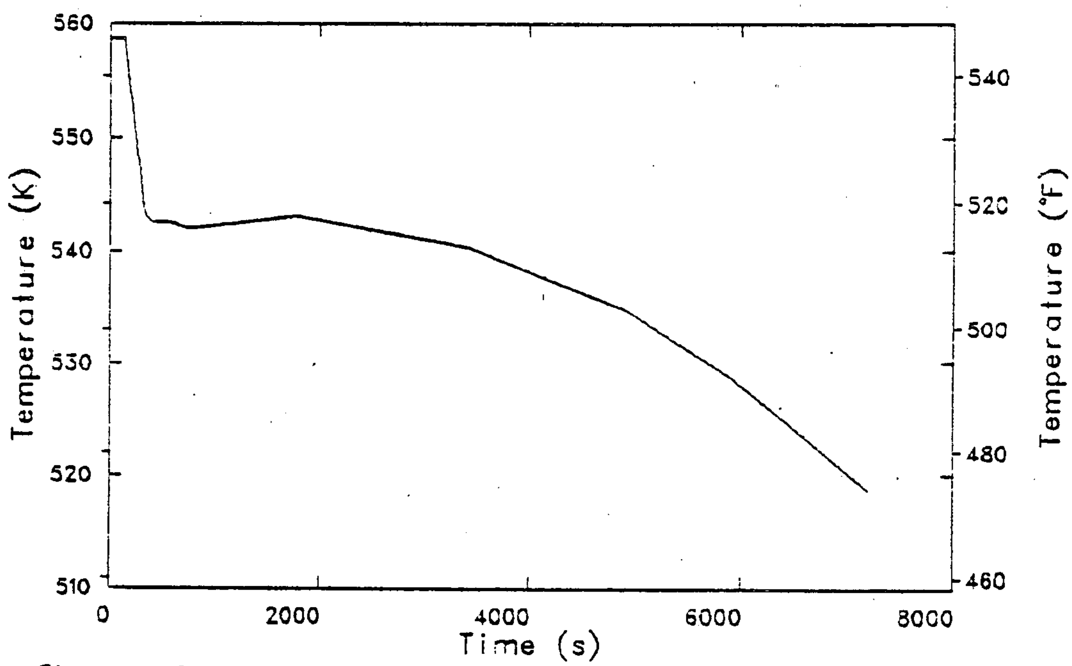
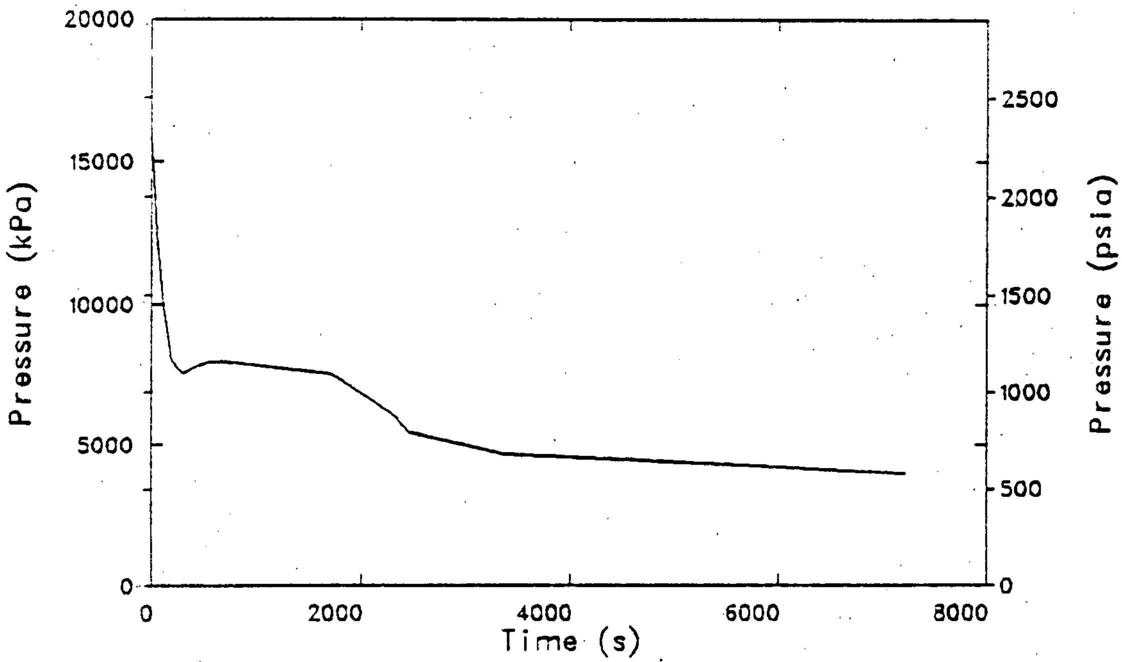


Figure A-2.. Sequence 1-1 reactor vessel downcomer fluid temperature.



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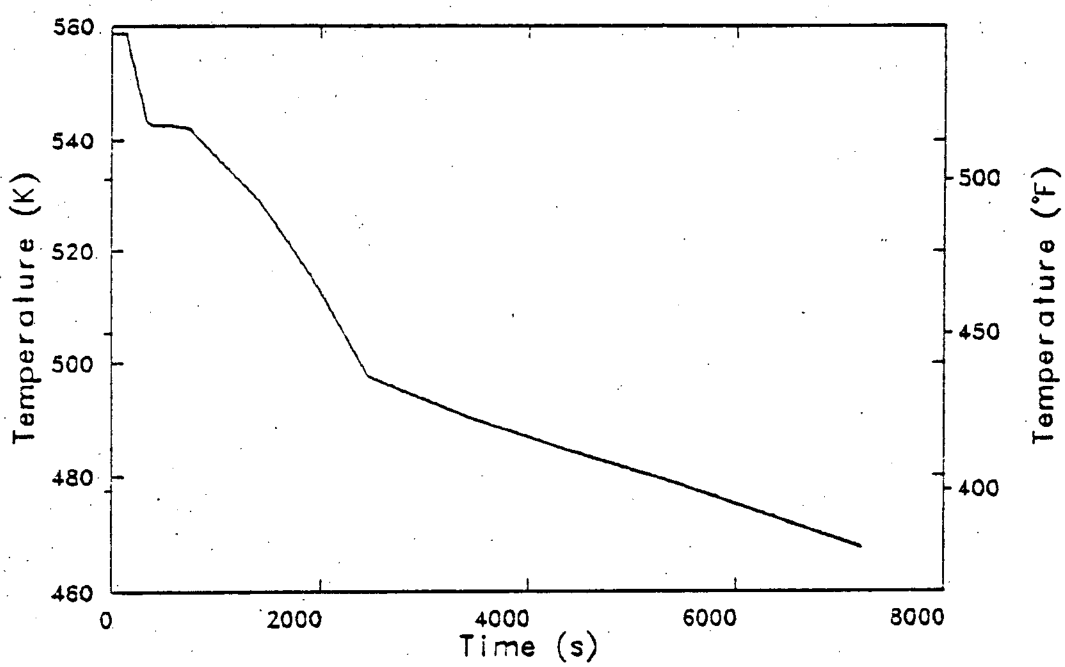


Figure A-4. Sequence 1-2 reactor vessel downcomer fluid temperature.

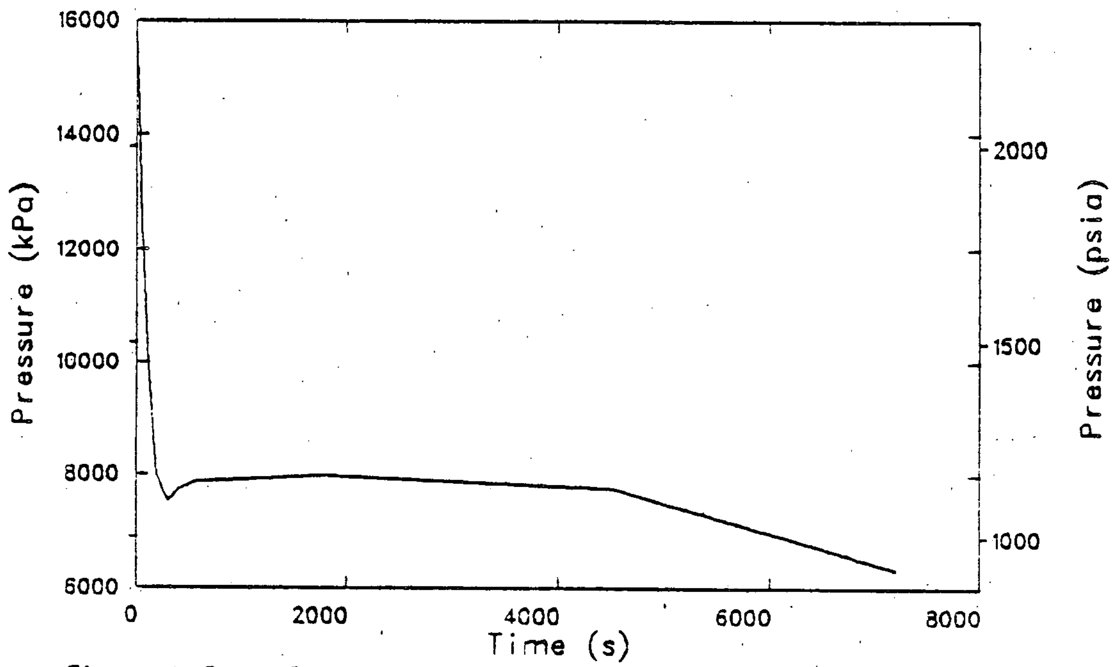


Figure A-5. Sequence 1-3 reactor vessel downcomer fluid pressure.

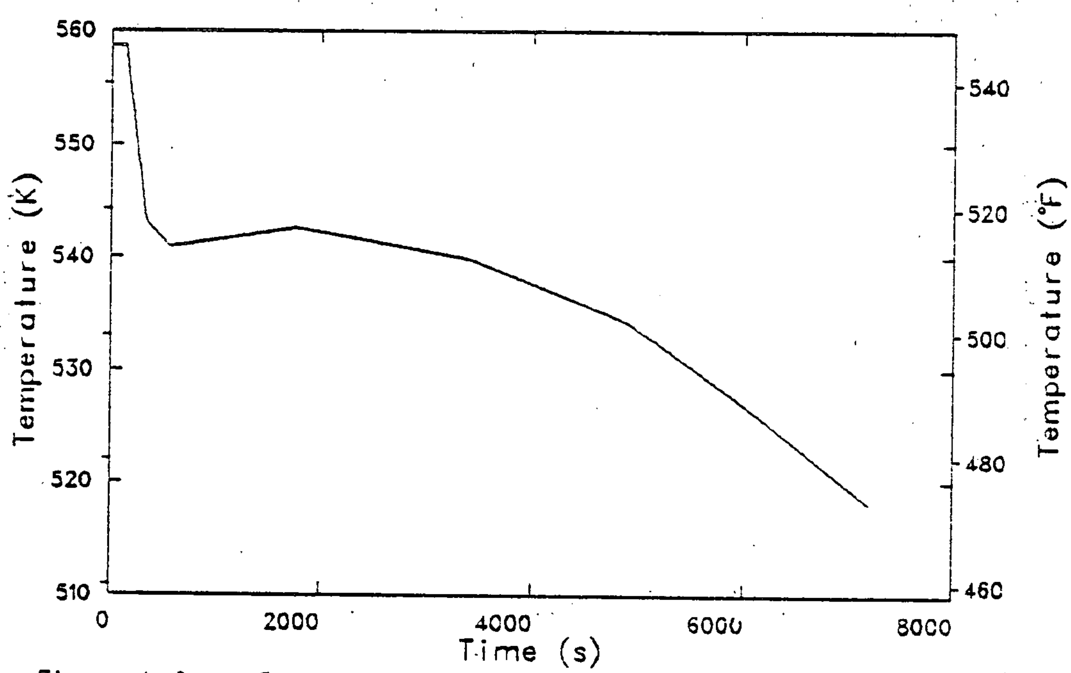


Figure A-6. Sequence 1-3 reactor vessel downcomer fluid temperature.

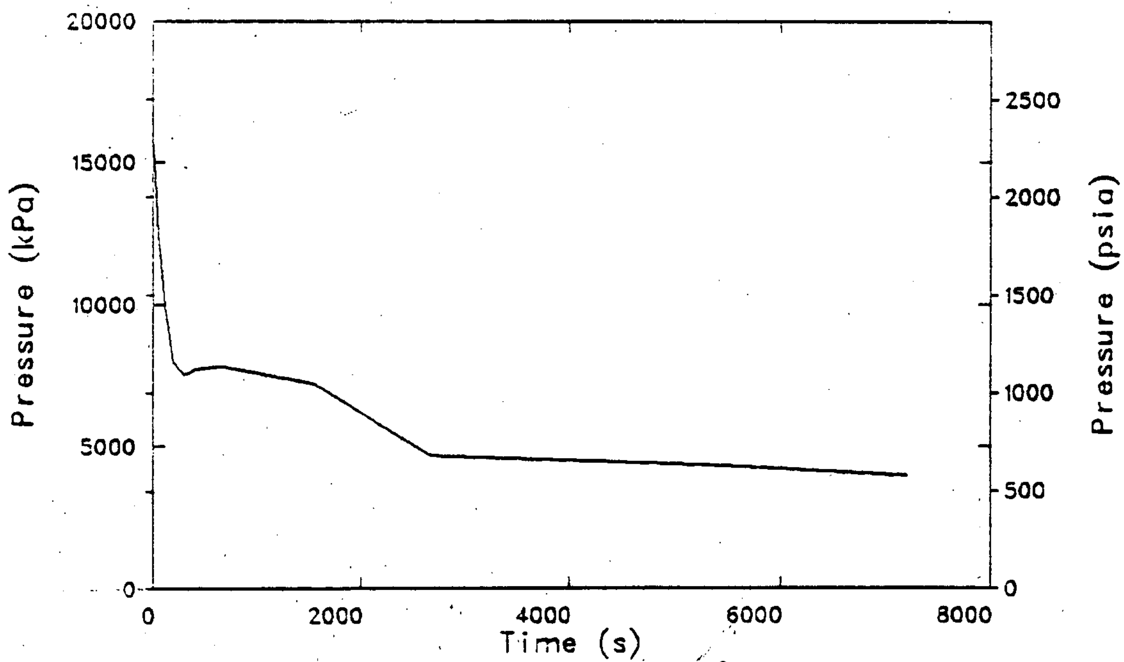


Figure A-7. Sequence 1-4 reactor vessel downcomer fluid pressure.

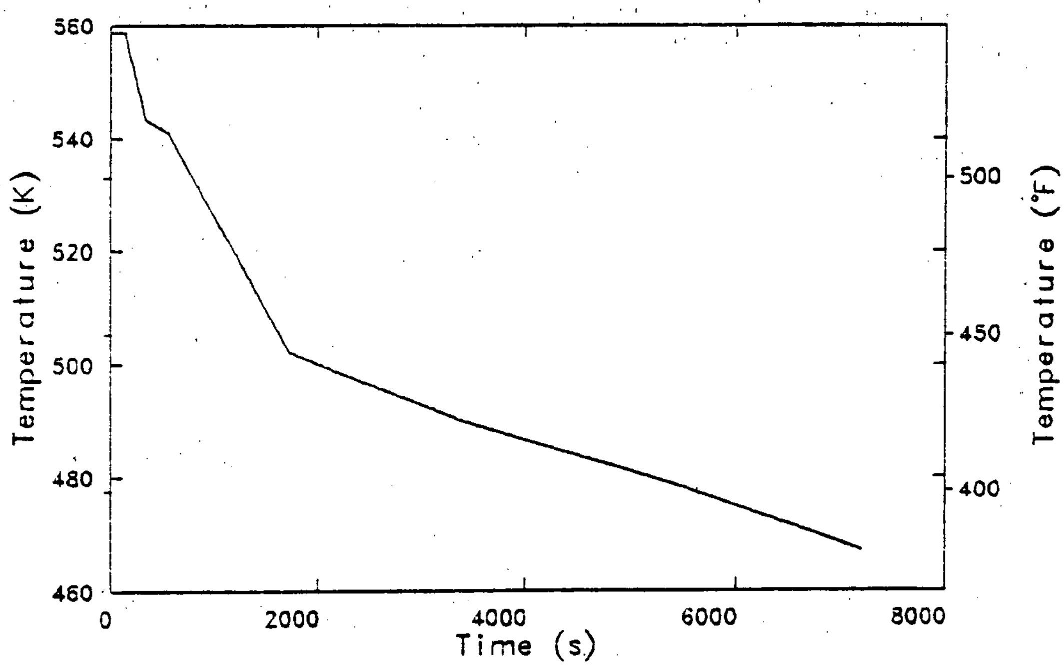


Figure A-8. Sequence 1-4 reactor vessel downcomer fluid temperature.

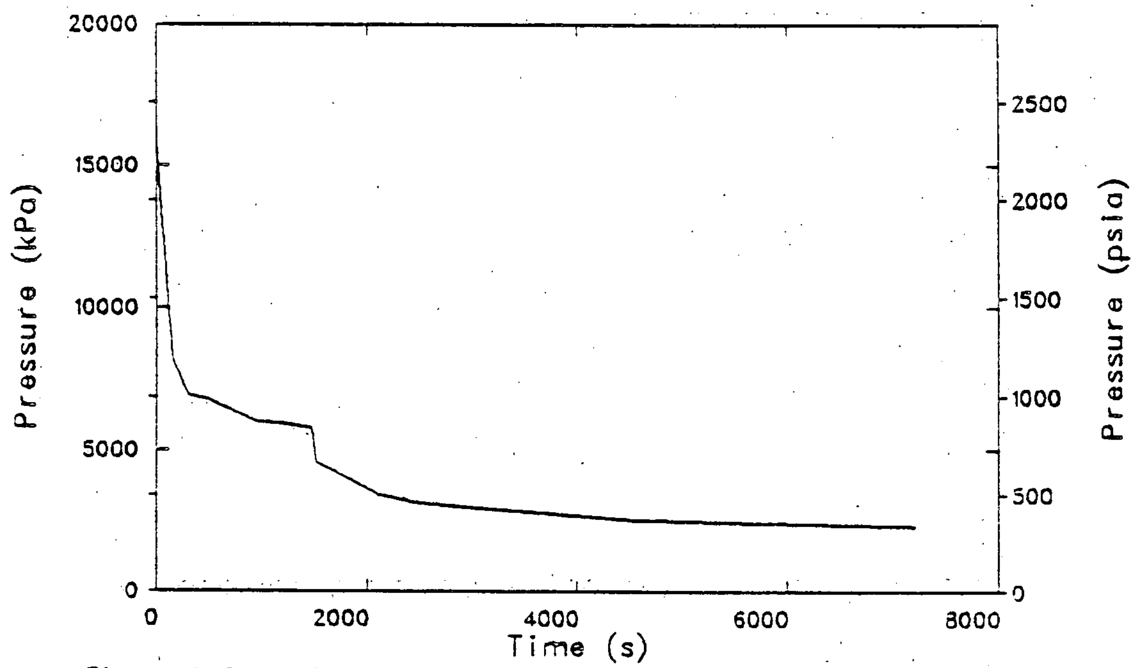


Figure A-9. Sequence 1-5 reactor vessel downcomer fluid pressure.

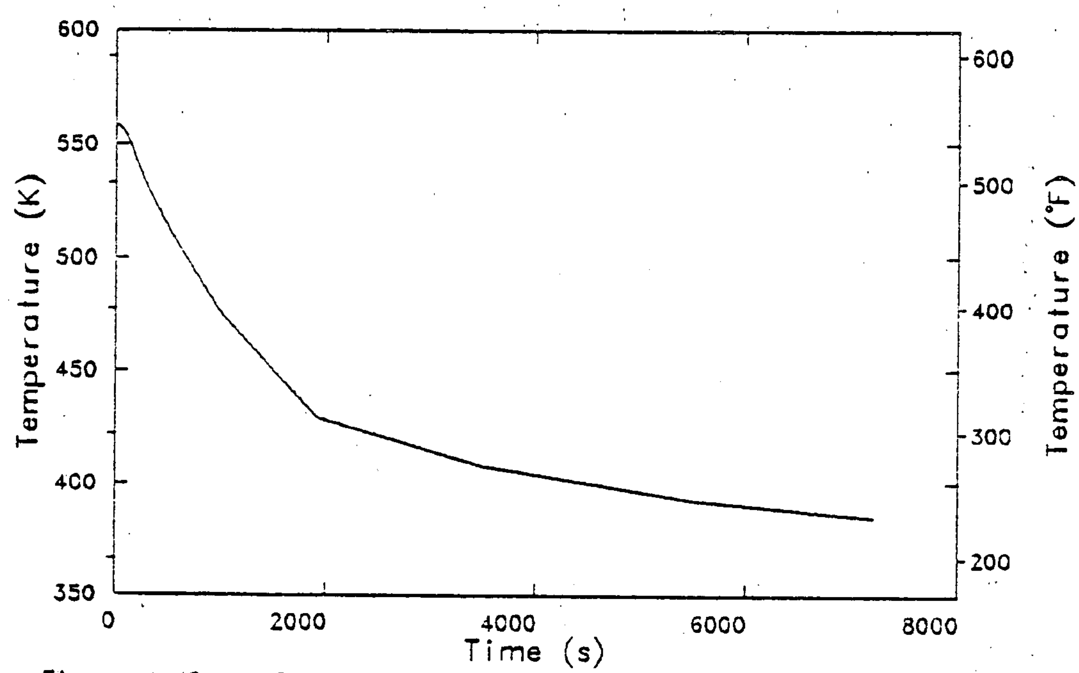


Figure A-10. Sequence 1-5 reactor vessel downcomer fluid temperature.

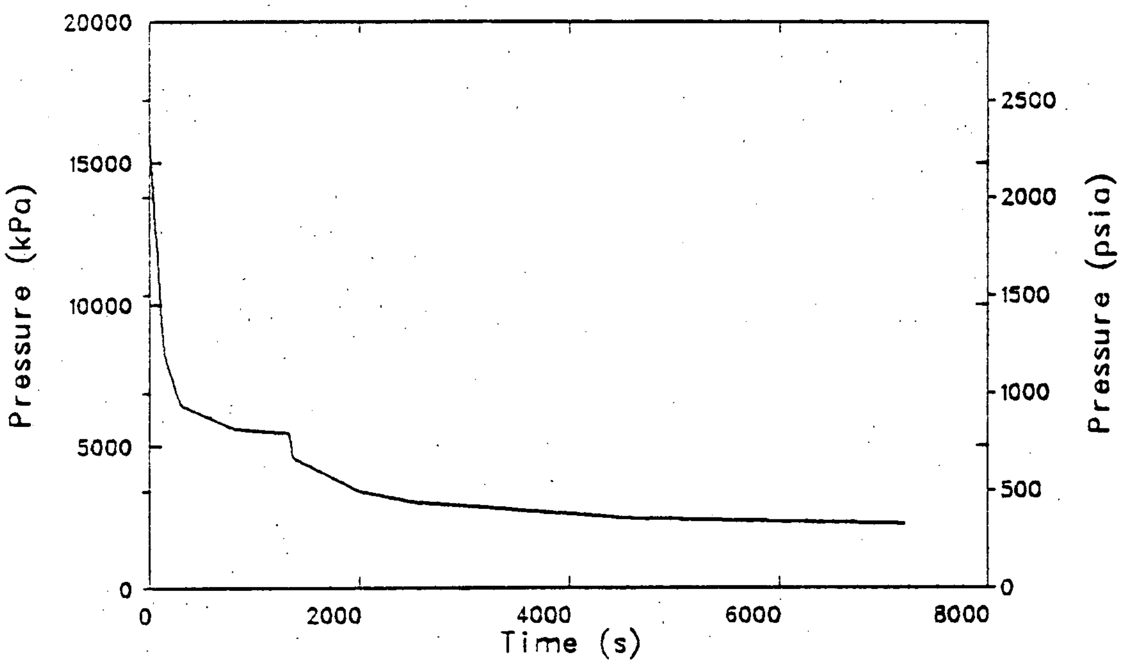


Figure A-11. Sequence 1-6 reactor vessel downcomer fluid pressure.

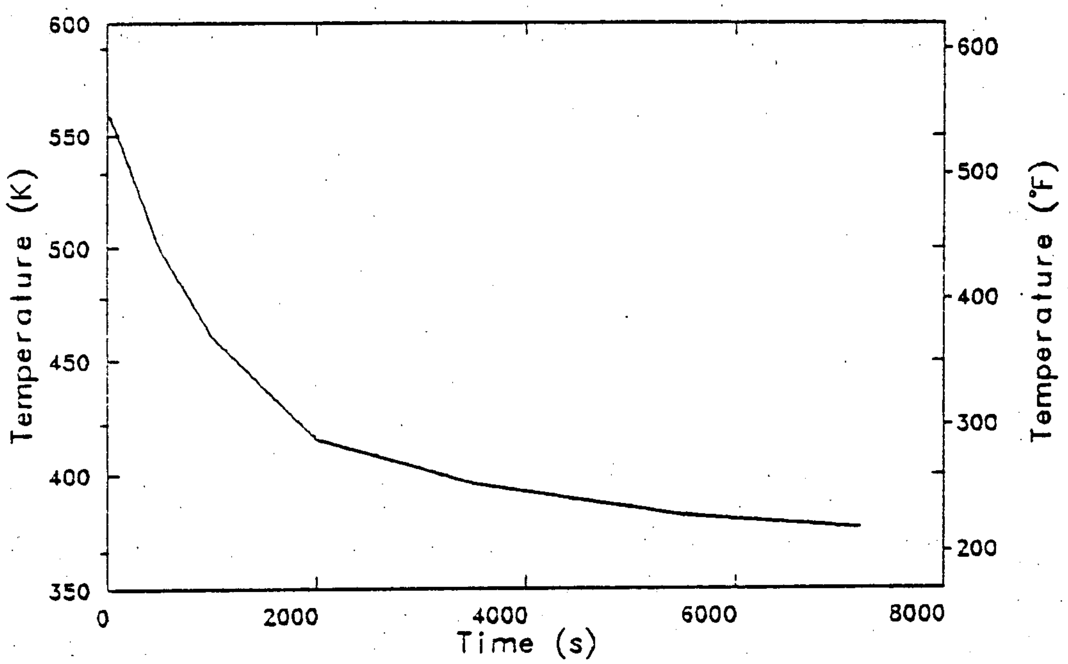


Figure A-12. Sequence 1-6 reactor vessel downcomer fluid temperature.

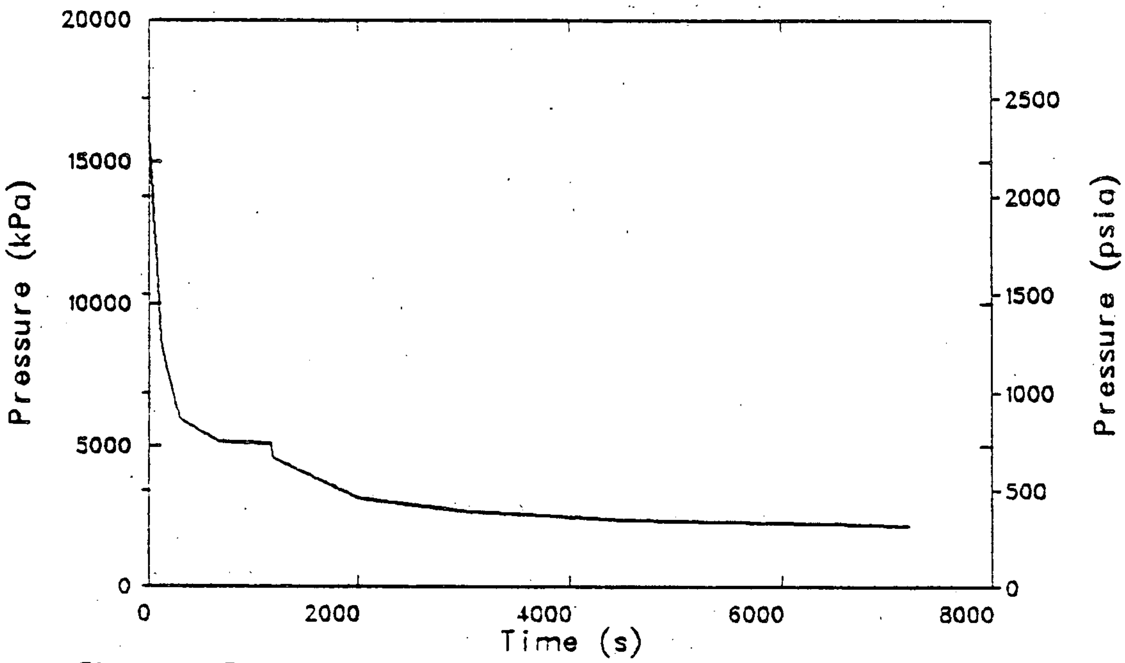


Figure A-13. Sequence 1-7 reactor vessel downcomer fluid pressure.

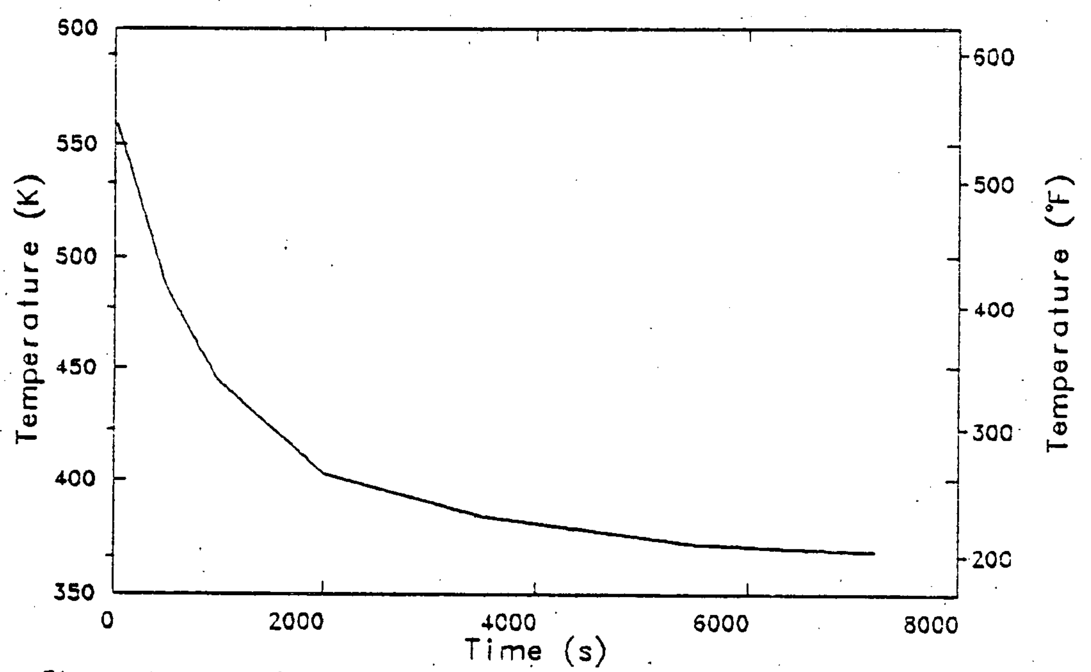


Figure A-14. Sequence 1-7 reactor vessel downcomer fluid temperature.

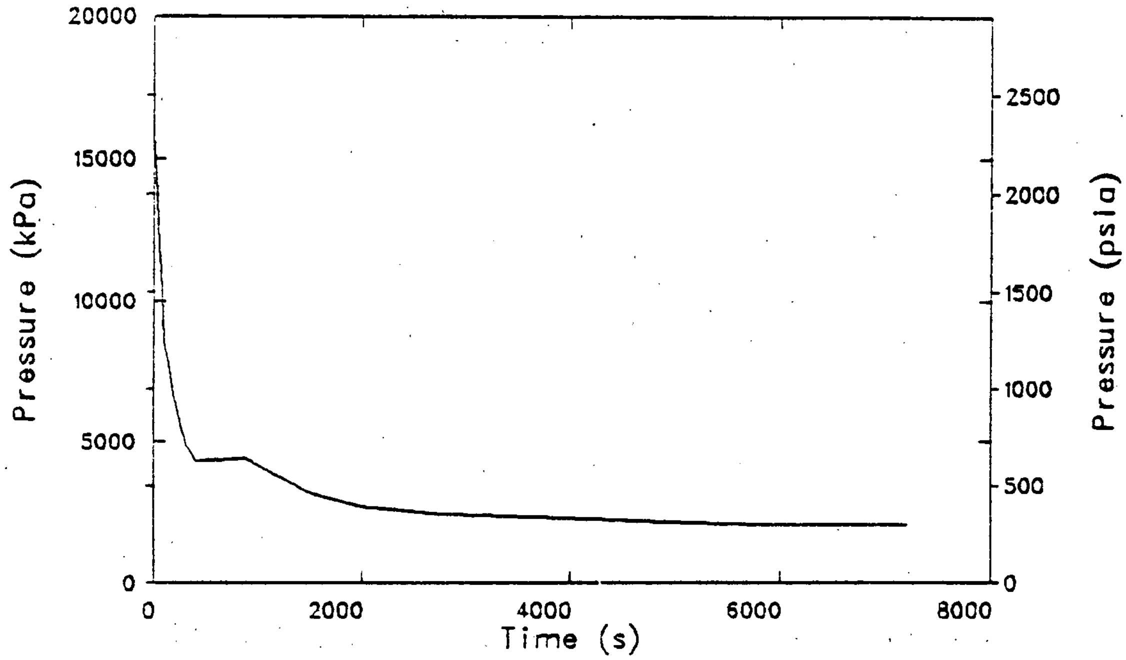


Figure A-15. Sequence 1-8 reactor vessel downcomer fluid pressure.

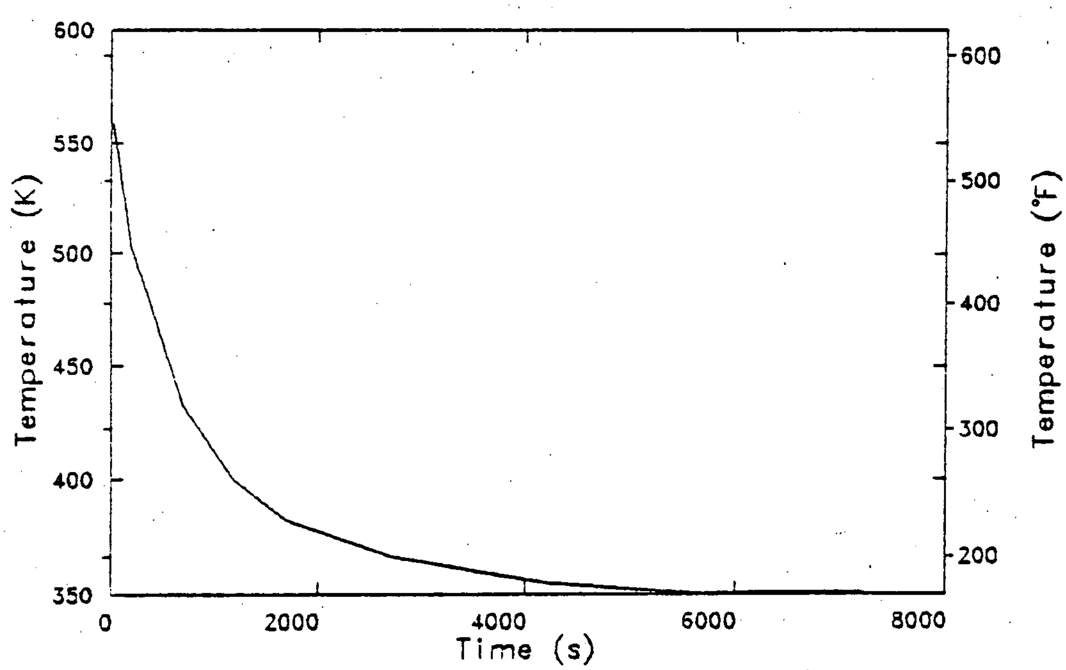


Figure A-16. Sequence 1-8 reactor vessel downcomer fluid temperature.

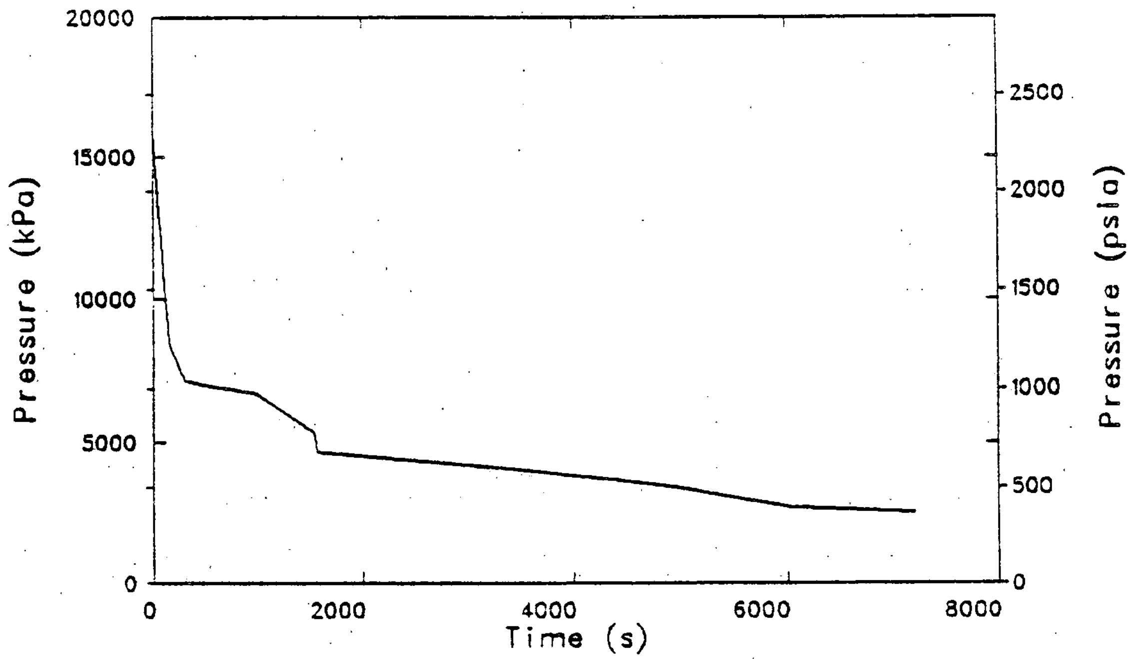


Figure A-17. Sequence 1-9 reactor vessel downcomer fluid pressure.

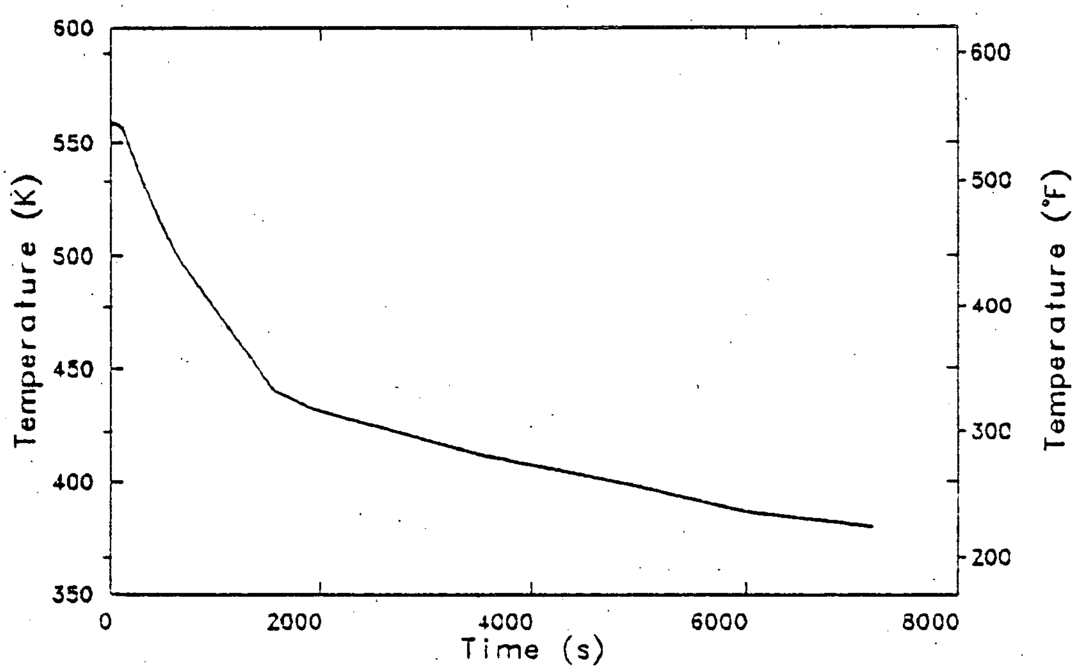


Figure A-18. Sequence 1-9 reactor vessel downcomer fluid temperature.

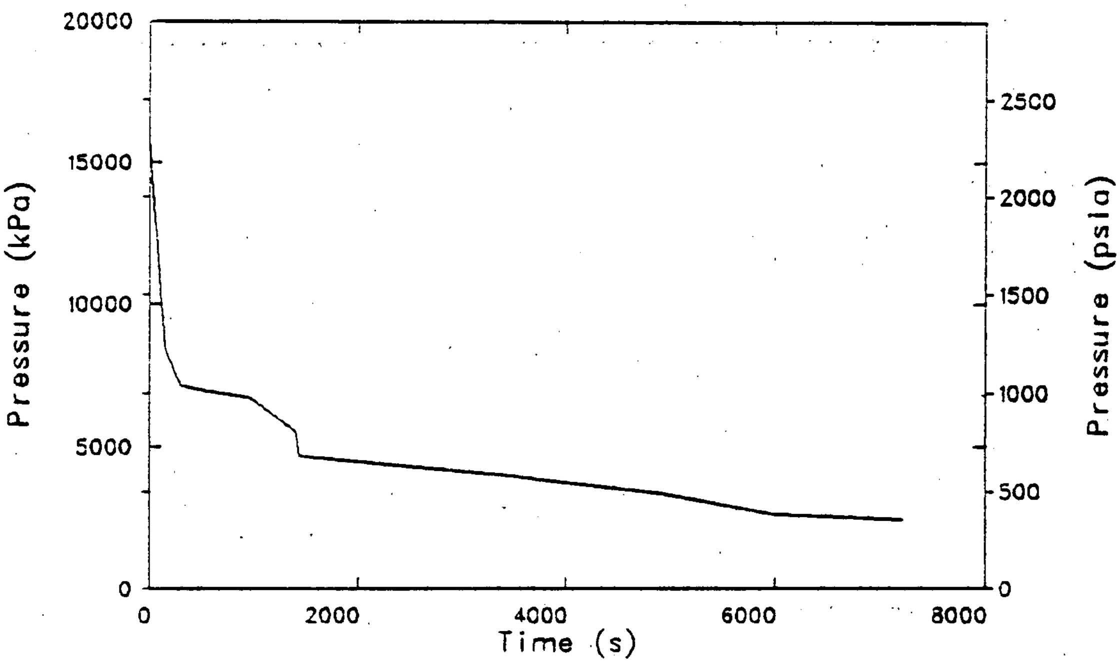


Figure A-19. Sequence 1-10 reactor vessel downcomer fluid pressure.

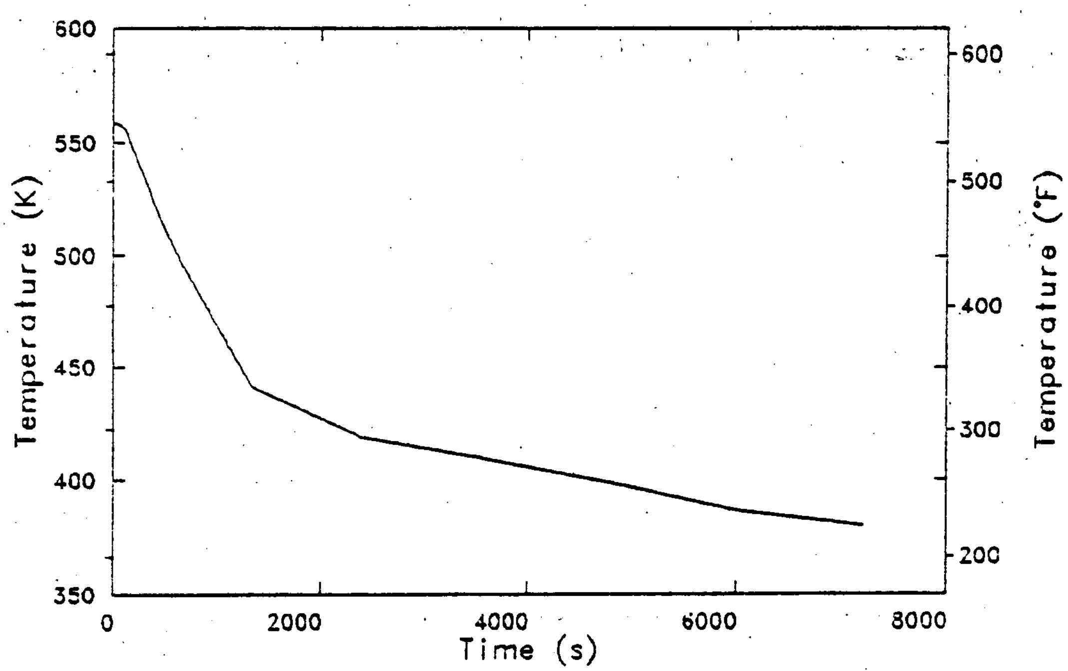


Figure A-20. Sequence i-10 reactor vessel downcomer fluid temperature.

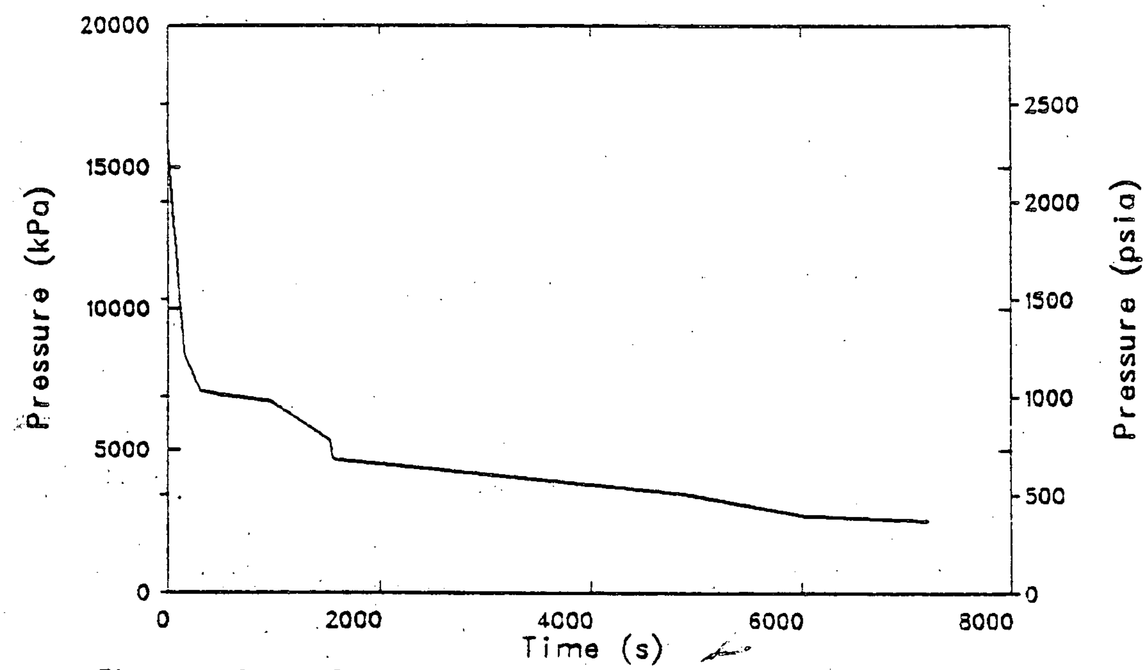


Figure A-21. Sequence 1-11 reactor vessel downcomer fluid pressure.

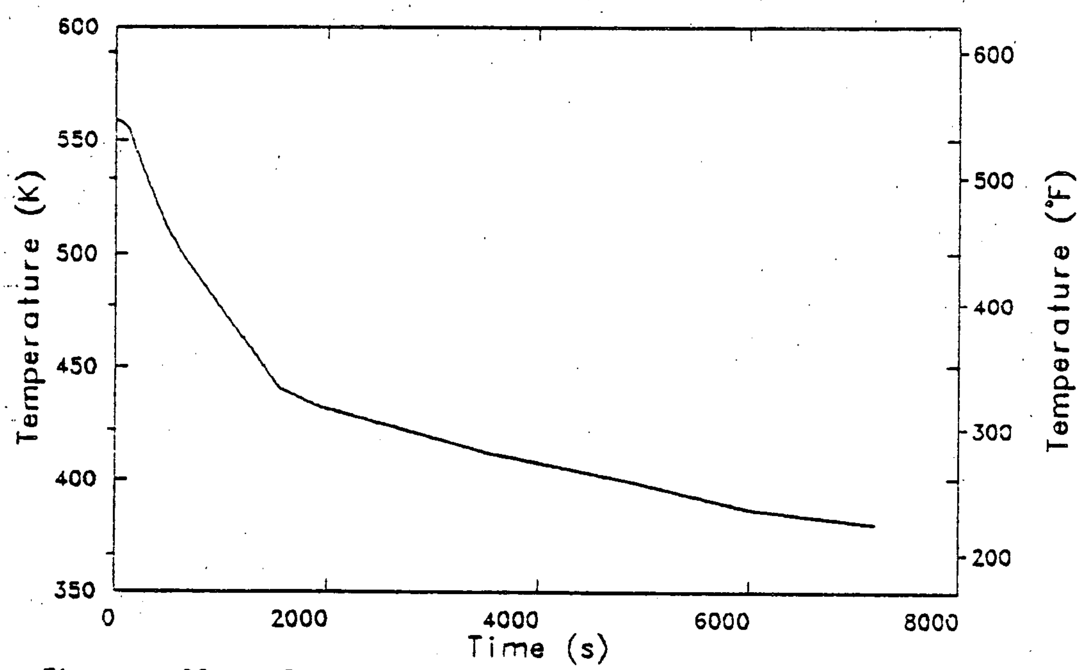


Figure A-22. Sequence 1-11 reactor vessel downcomer fluid temperature.

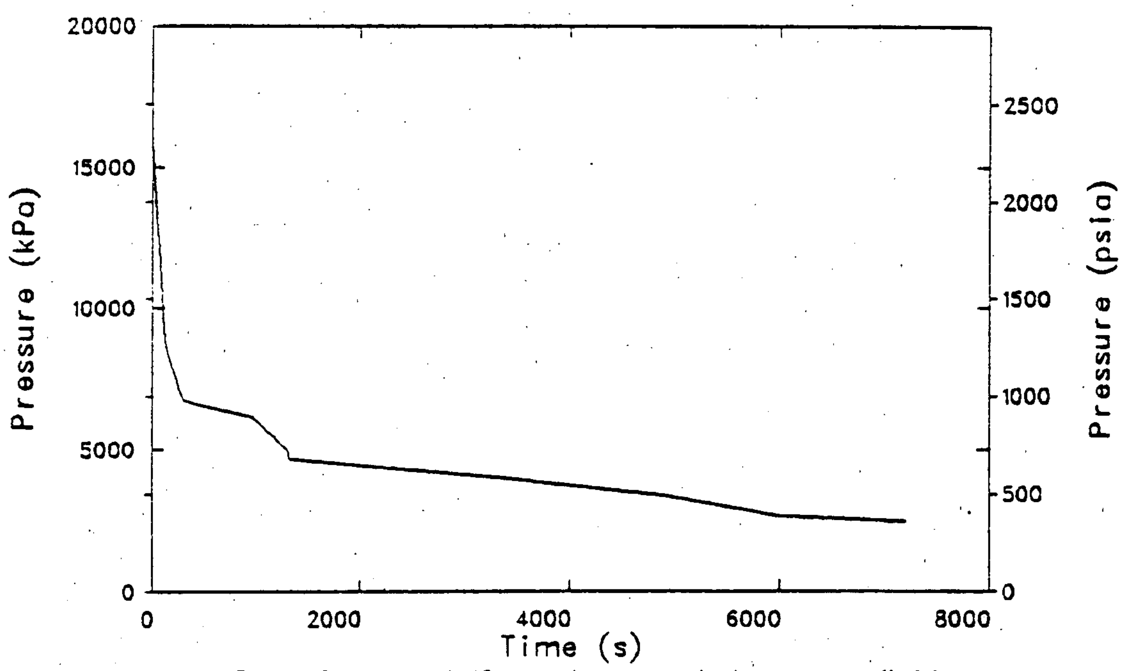


Figure A-23. Sequence 1-12 reactor vessel downcomer fluid pressure.

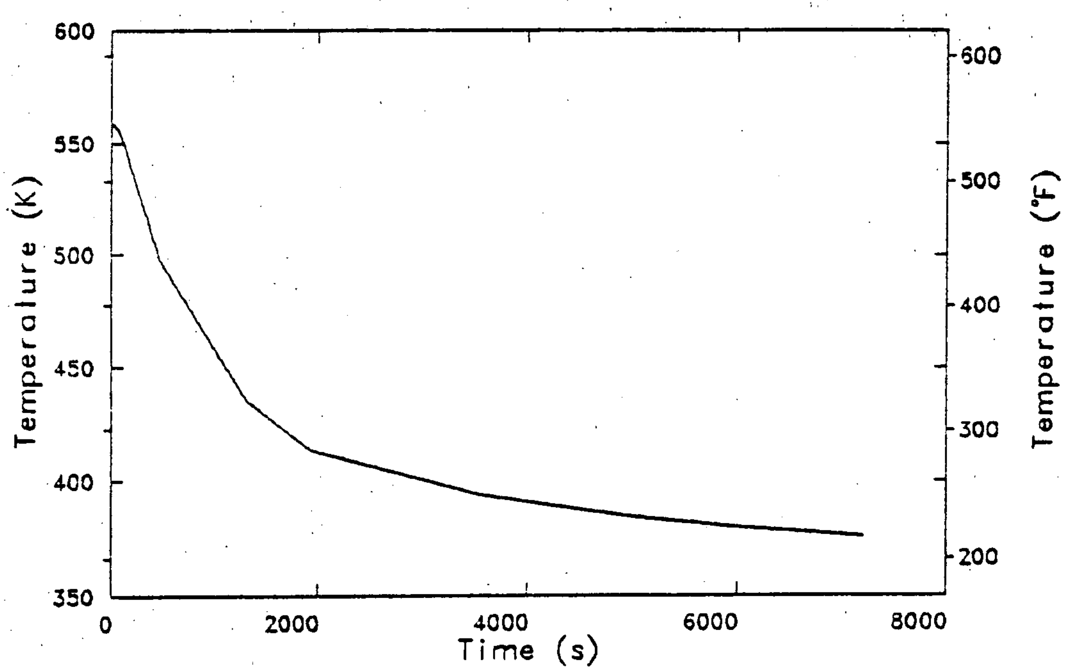


Figure A-24. Sequence 1-12 reactor vessel downcomer fluid temperature.

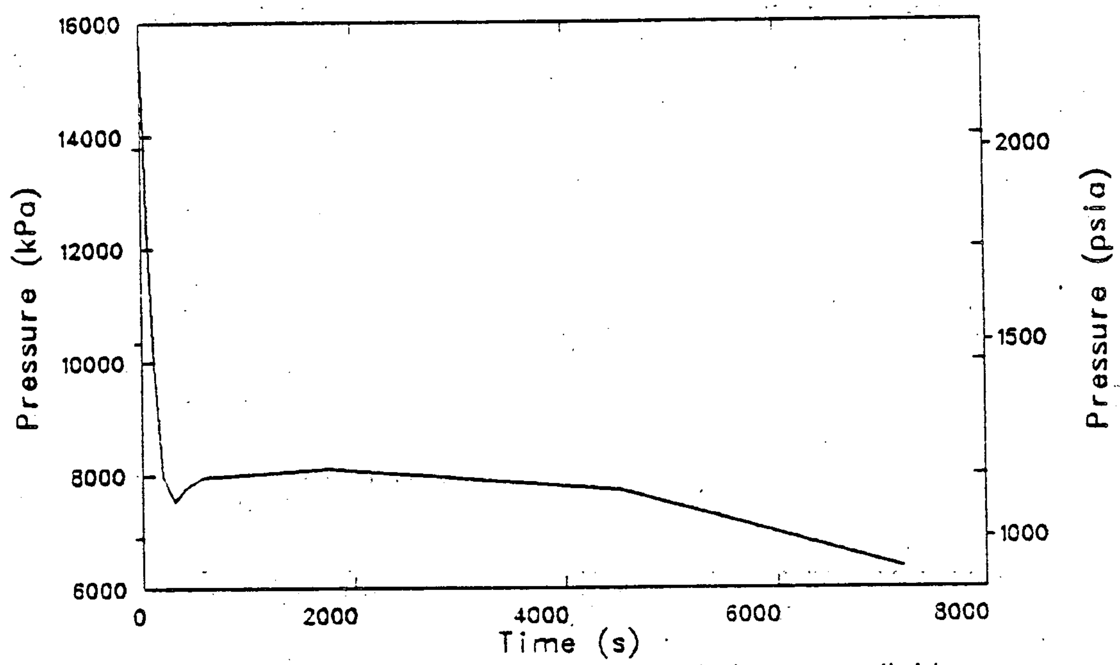


Figure A-25. Sequence 1-13 reactor vessel downcomer fluid pressure.

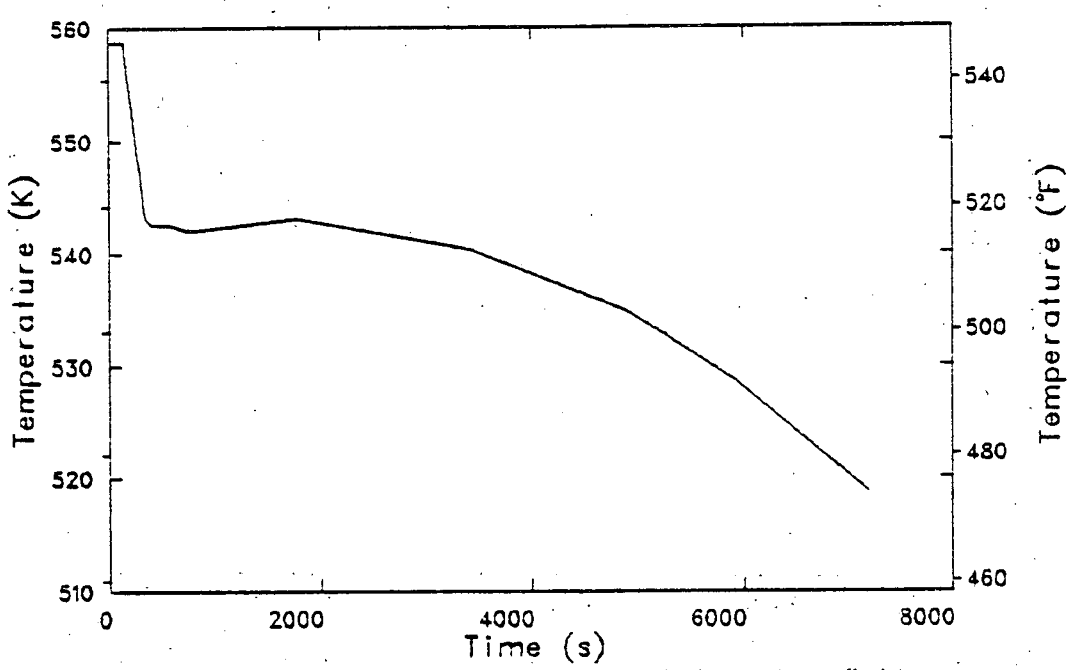


Figure A-26. Sequence 1-13 reactor vessel downcomer fluid temperature.

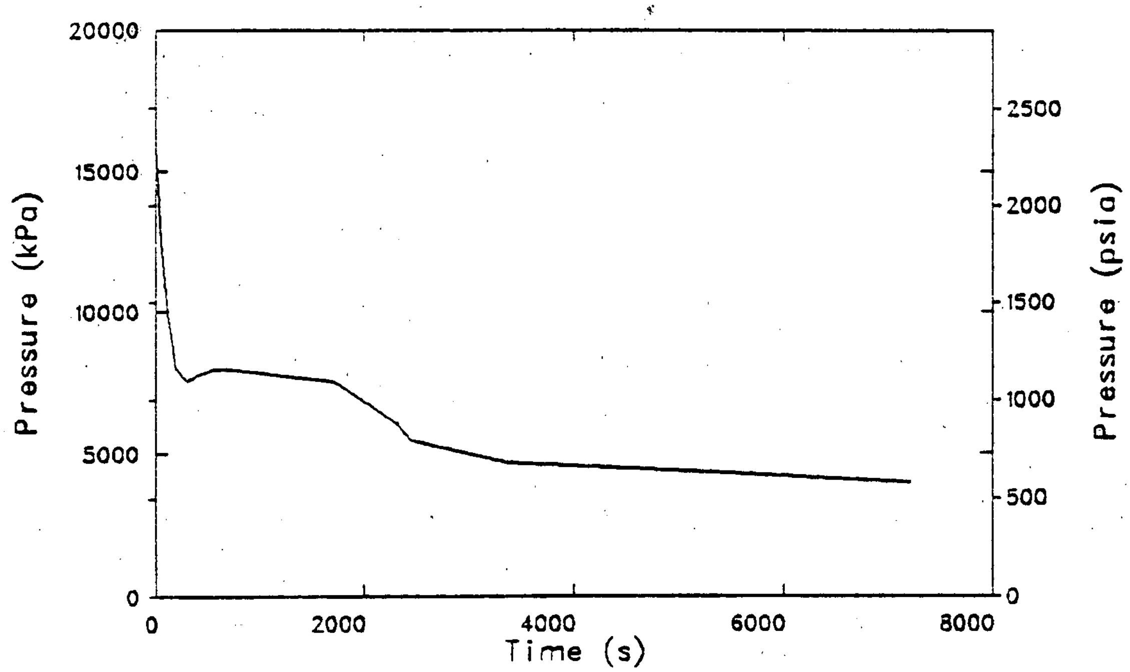


Figure A-27. Sequence 1-14 reactor vessel downcomer fluid pressure.

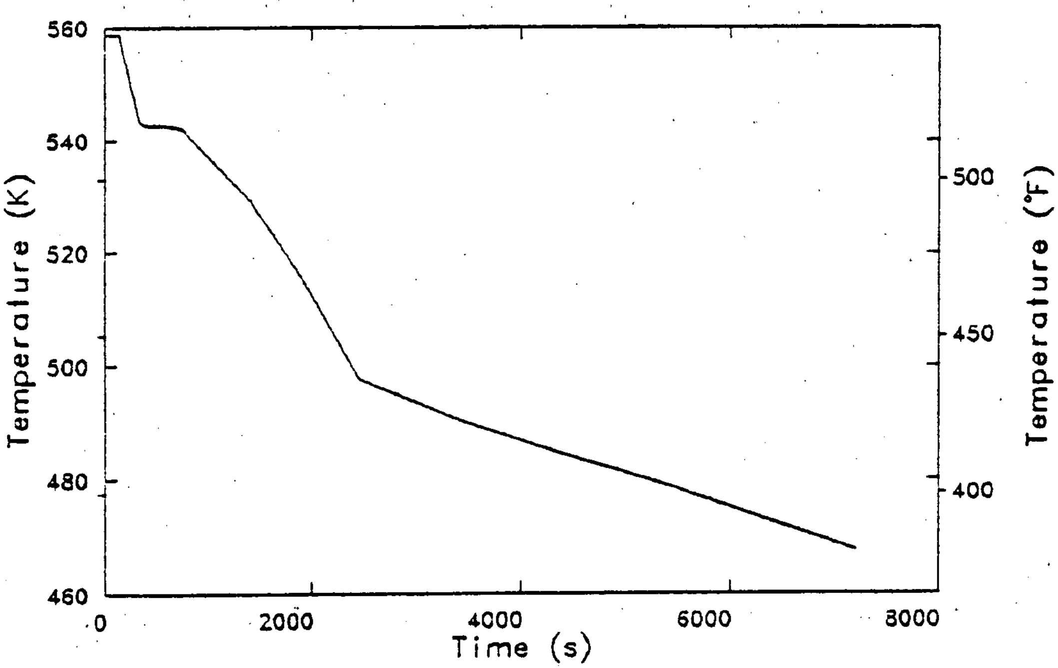


Figure A-28. Sequence 1-14 reactor vessel downcomer fluid temperature.

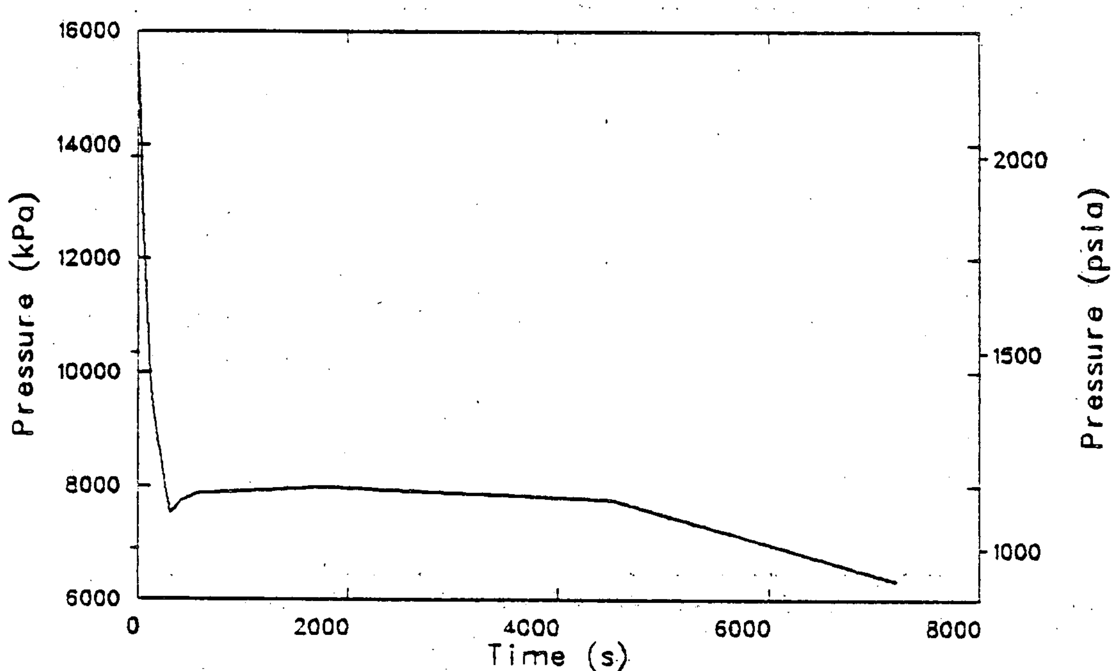
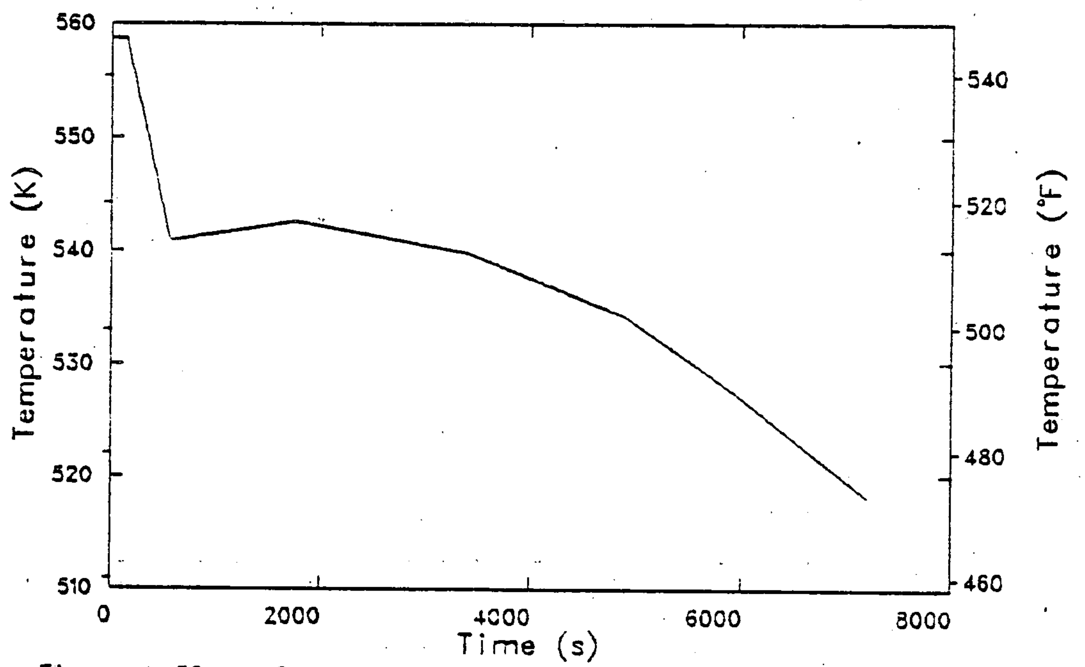


Figure A-29. Sequence 1-15 reactor vessel downcomer fluid pressure.



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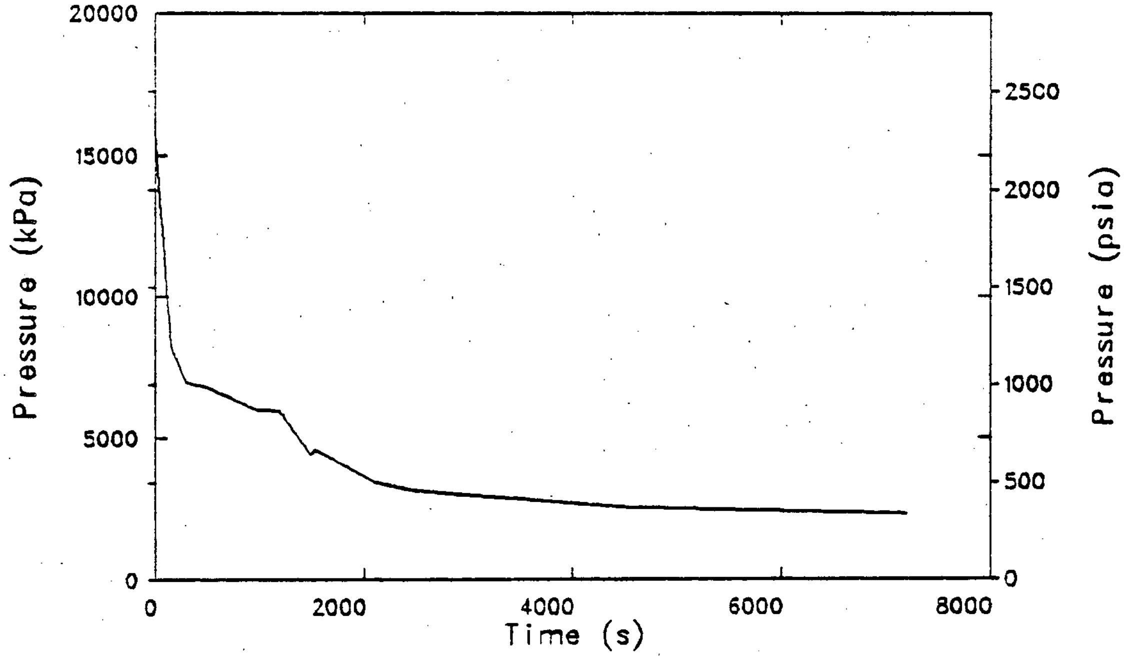


Figure A-31. Sequence 1-16 reactor vessel downcomer fluid pressure.

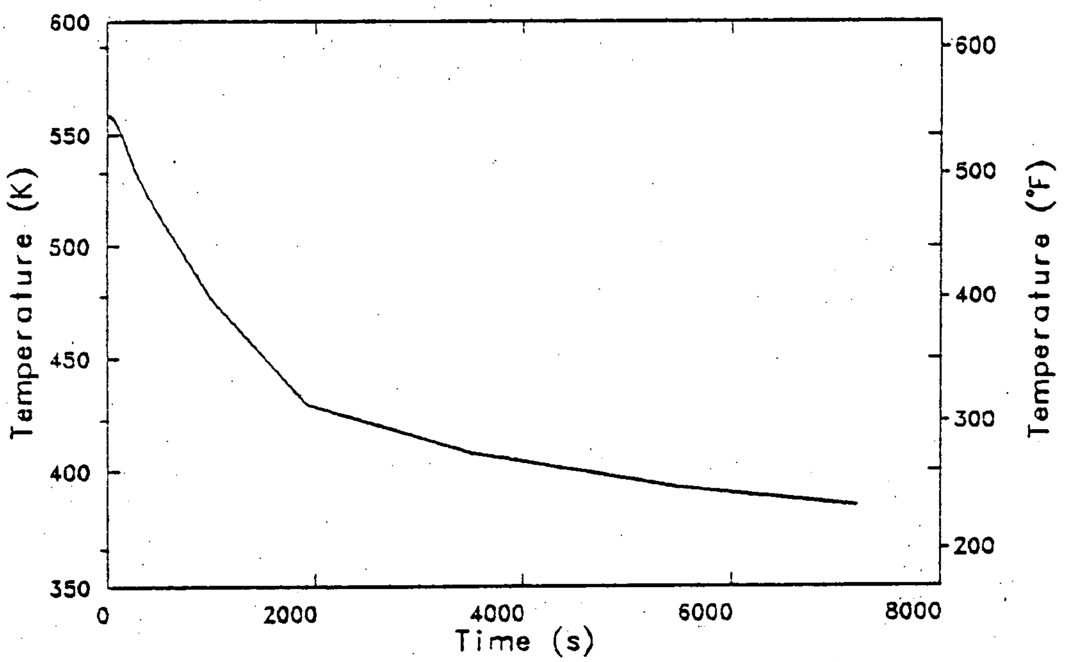
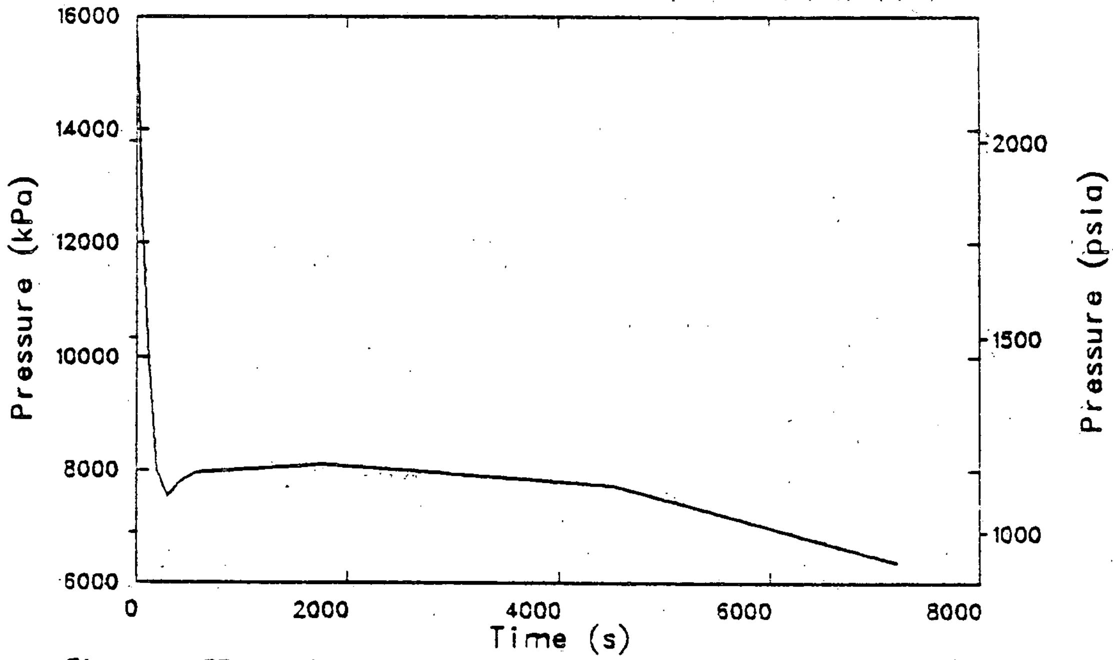


Figure A-32. Sequence 1-16 reactor vessel downcomer fluid temperature.



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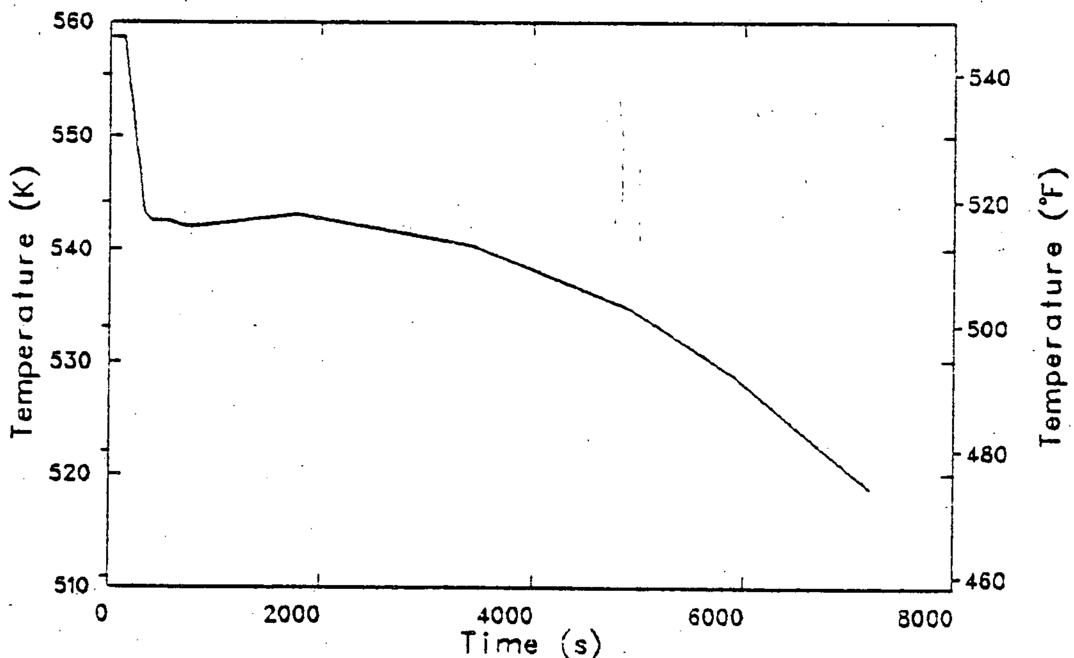

Figure A-34. Sequence 1-17 reactor vessel downcomer fluid temperature.

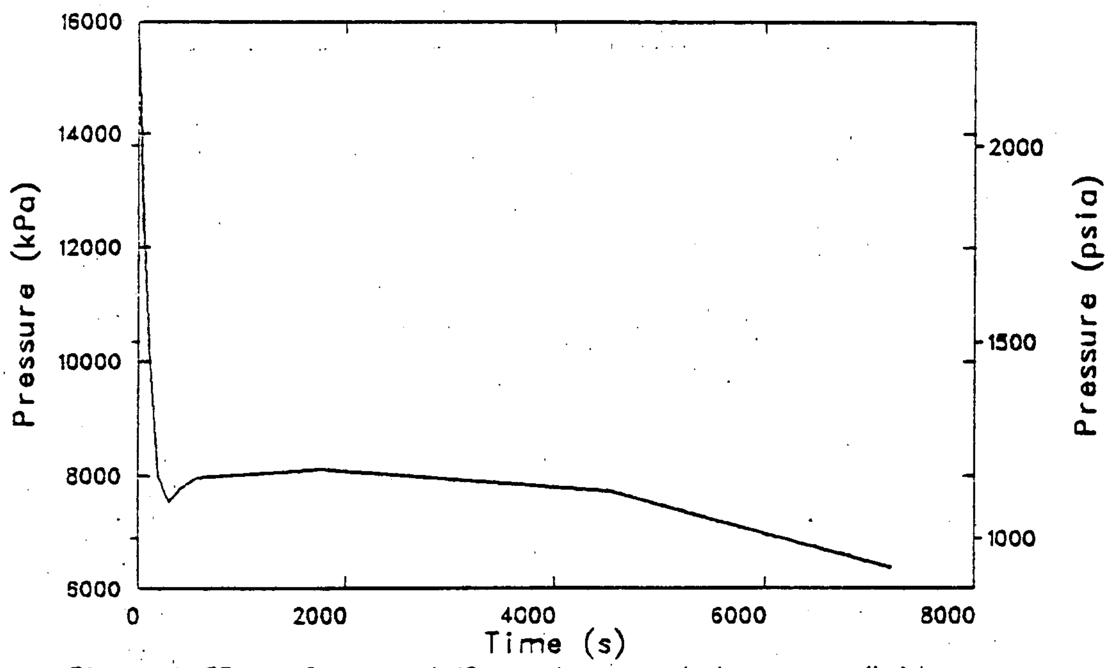

Figure A-35. Sequence 1-18 reactor vessel downcomer fluid pressure.

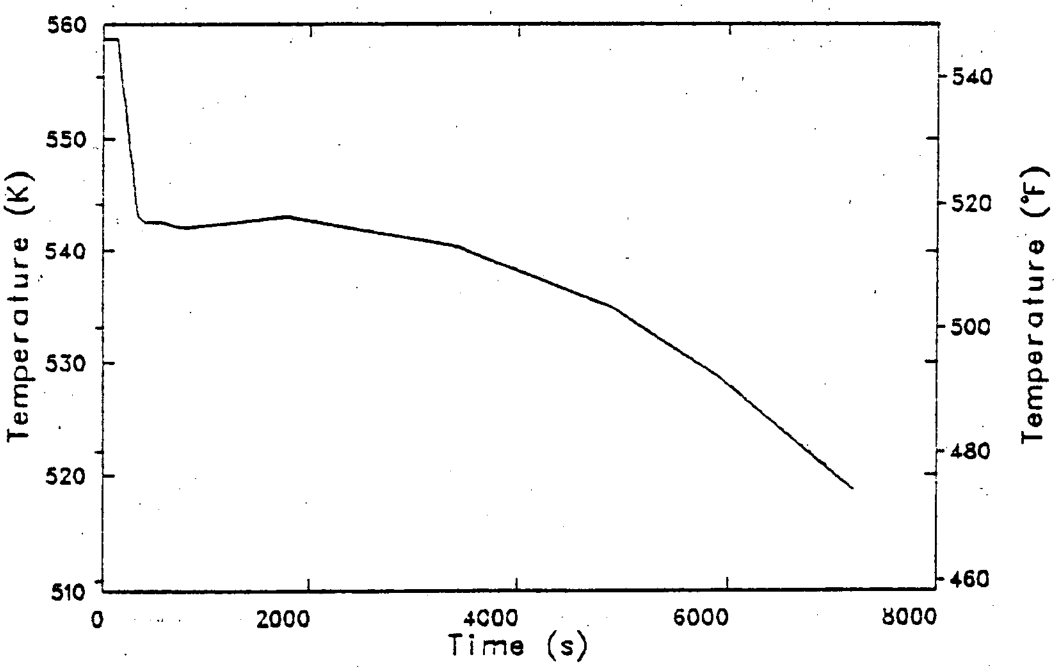

Figure A-36. Sequence 1-18 reactor vessel downcomer fluid temperature.

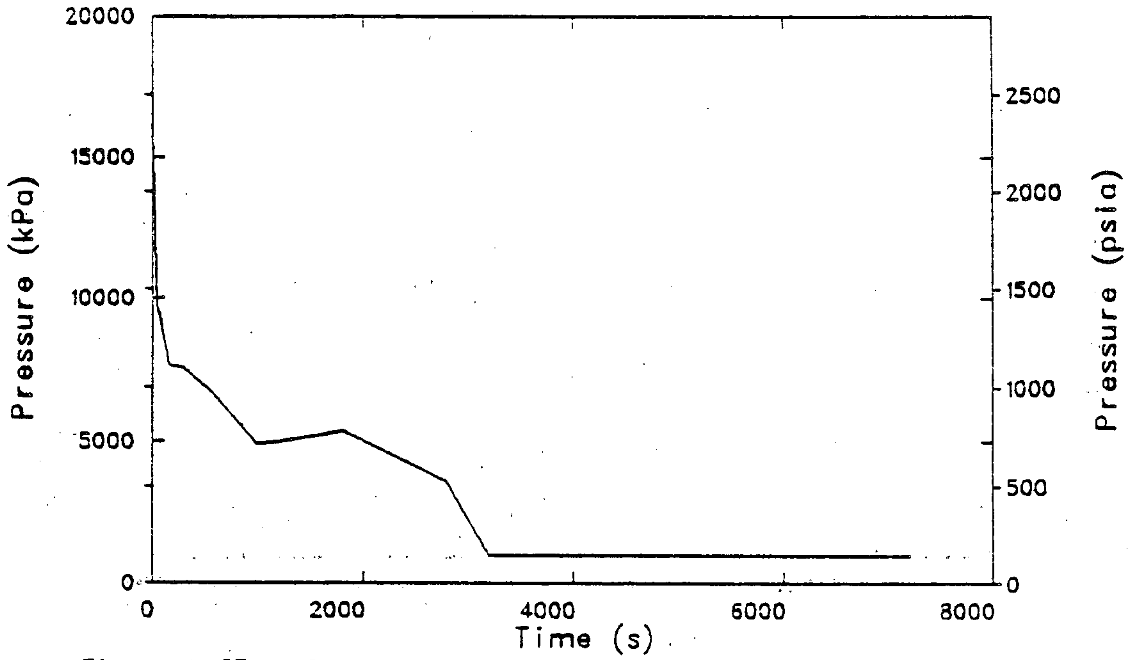

Figure A-37. Sequence 2-1 reactor vessel downcomer fluid pressure.

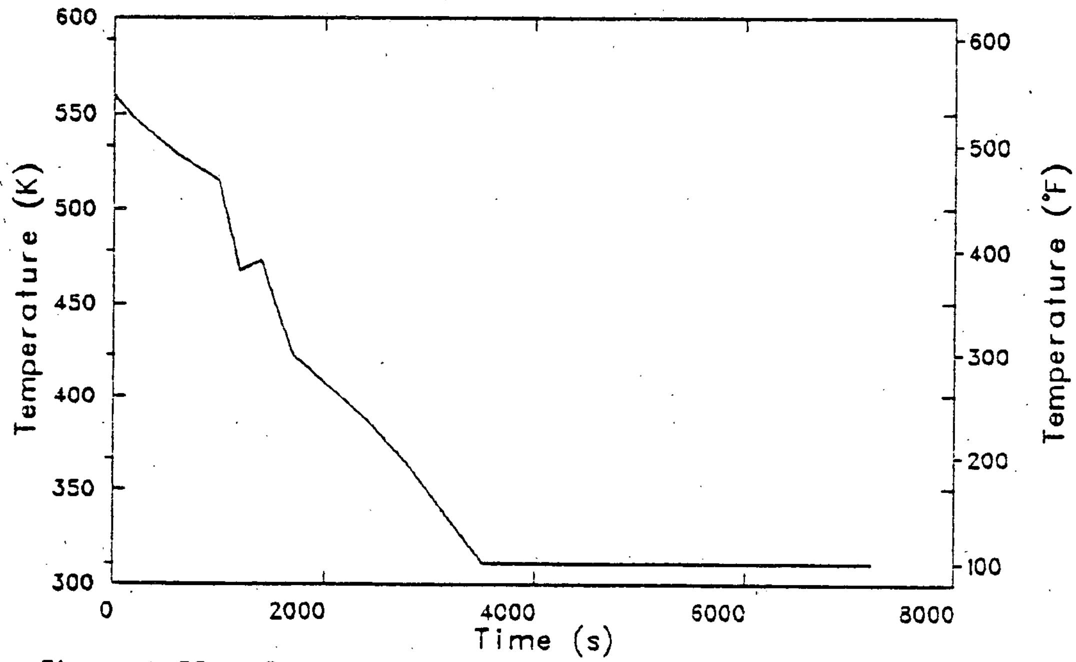

Figure A-38. Sequence 2-1 reactor vessel downcomer fluid temperature.

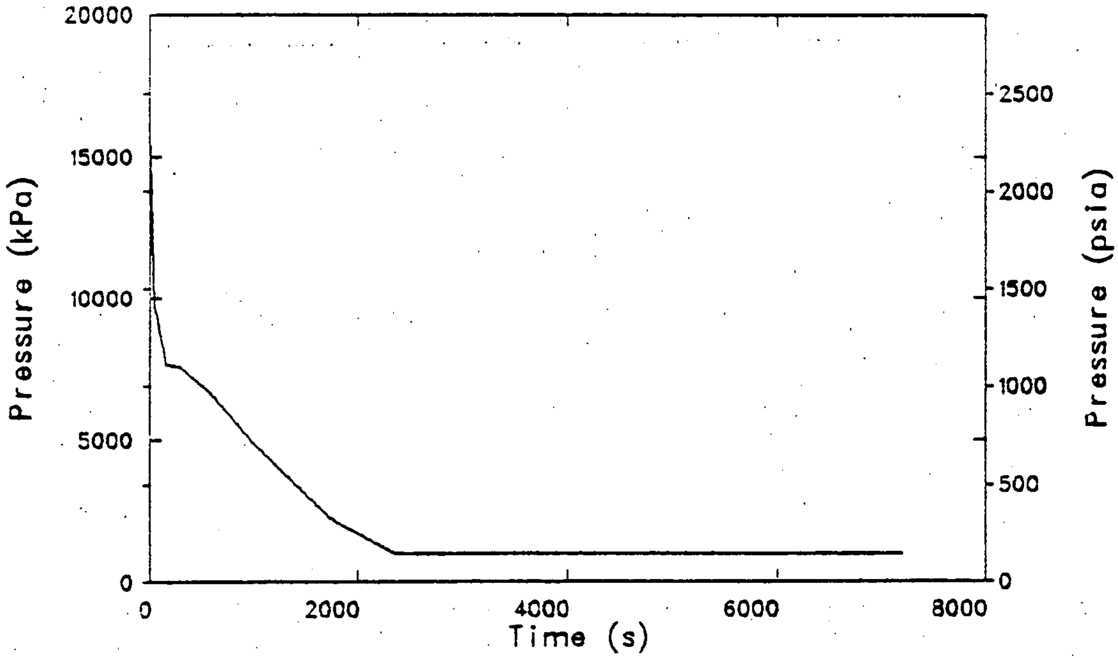

Figure A-39. Sequence 2-2 reactor vessel downcomer fluid pressure.

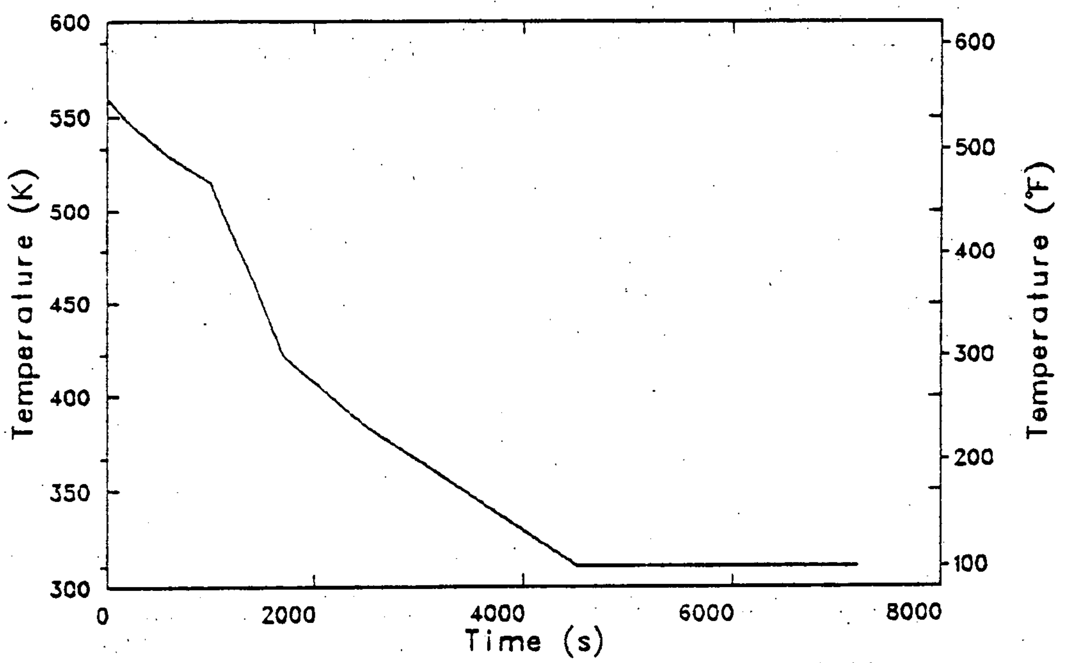

Figure A-40. Sequence 2-2 reactor vessel downcomer fluid temperature.

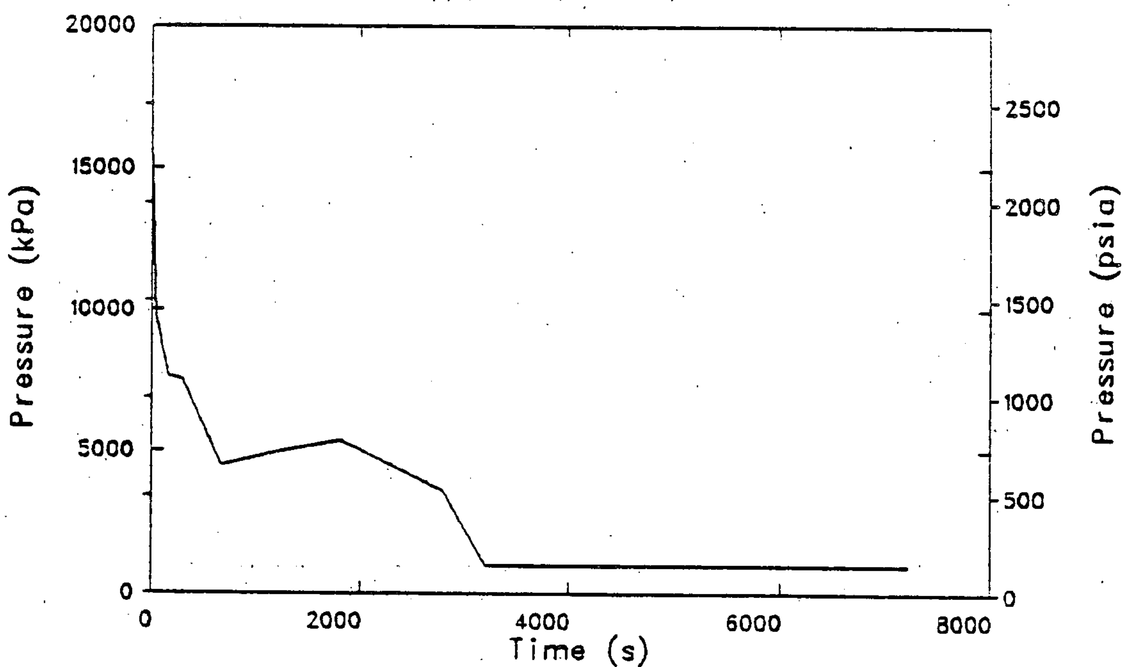

Figure A-41. Sequence 2-3 reactor vessel downcomer fluid pressure.

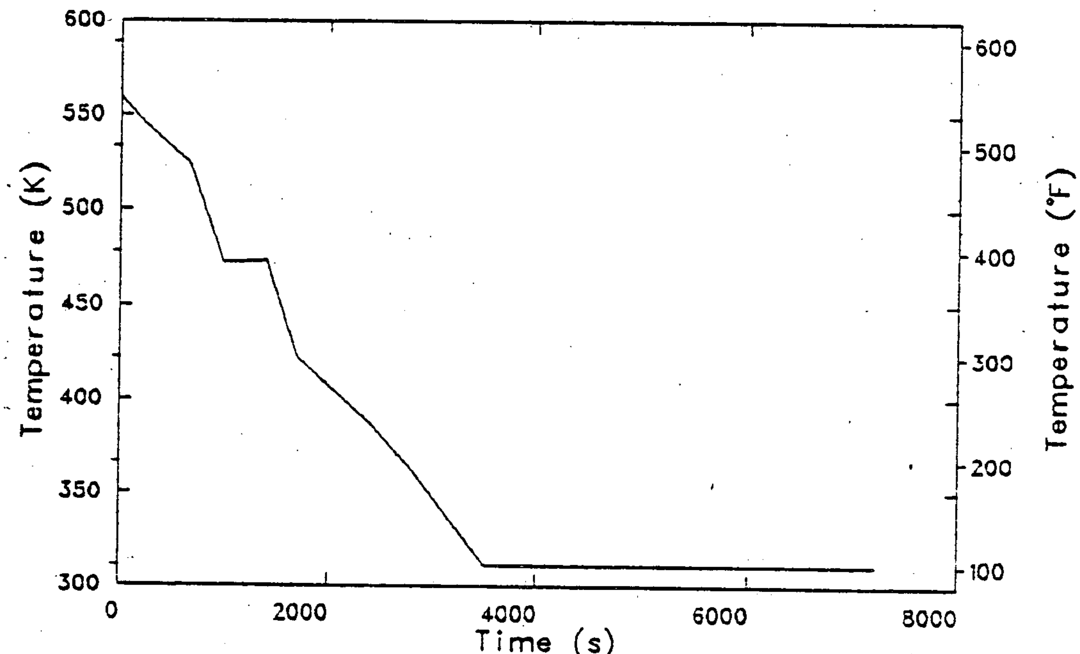

Figure A-42. Sequence 2-3 reactor vessel downcomer fluid temperature.

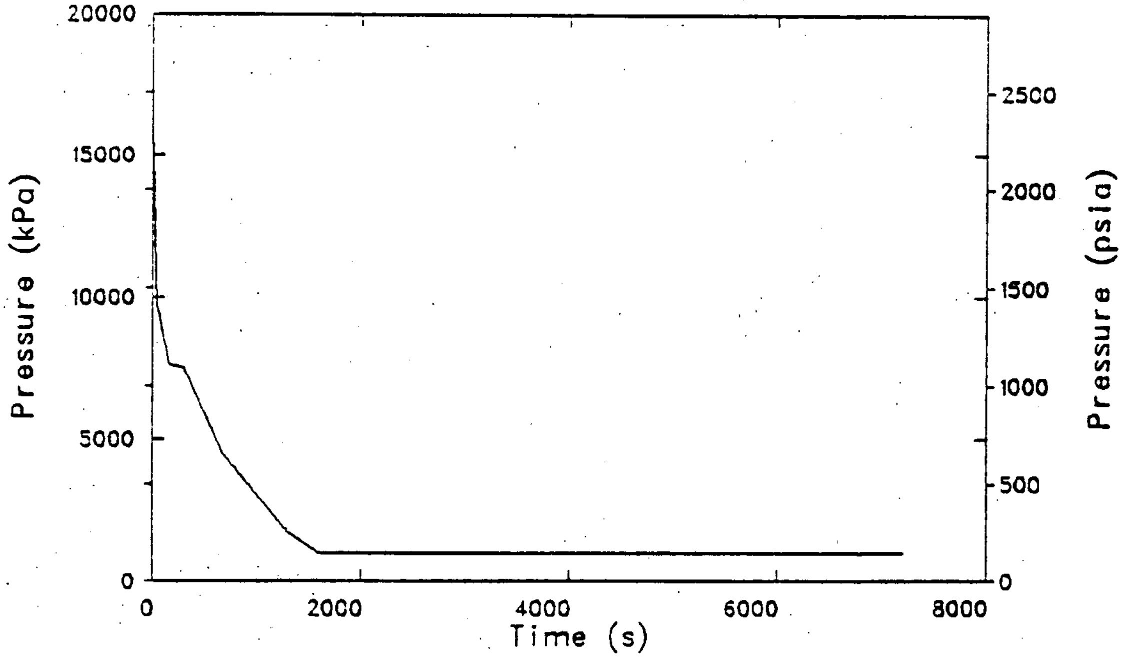

Figure A-43. Sequence 2-4 reactor vessel downcomer fluid pressure.

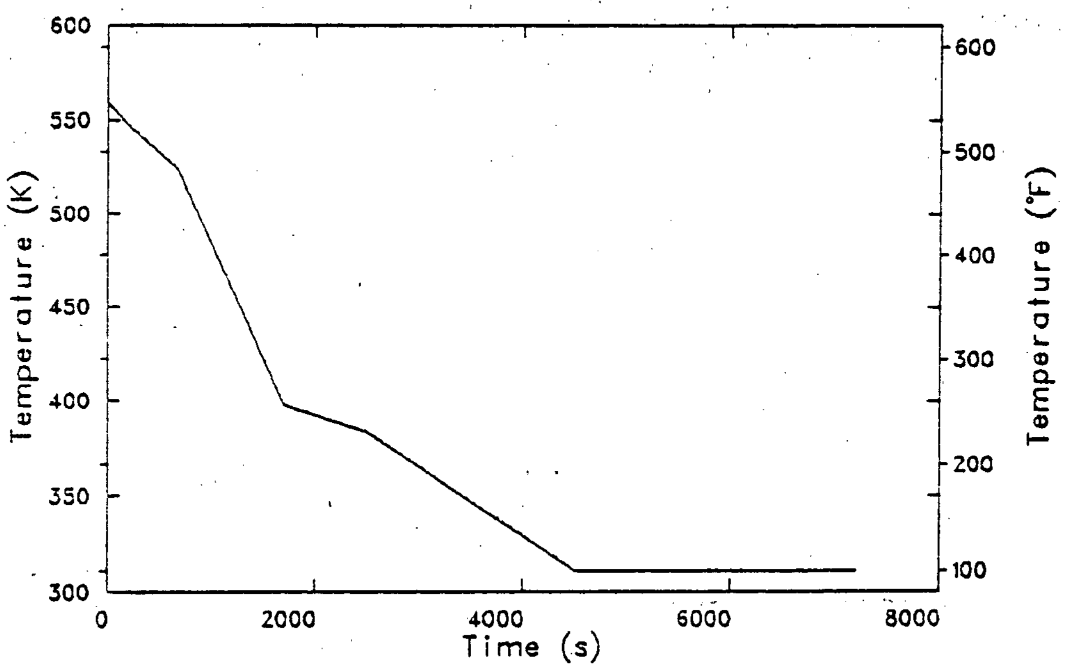

Figure A-44. Sequence 2-4 reactor vessel downcomer fluid temperature.

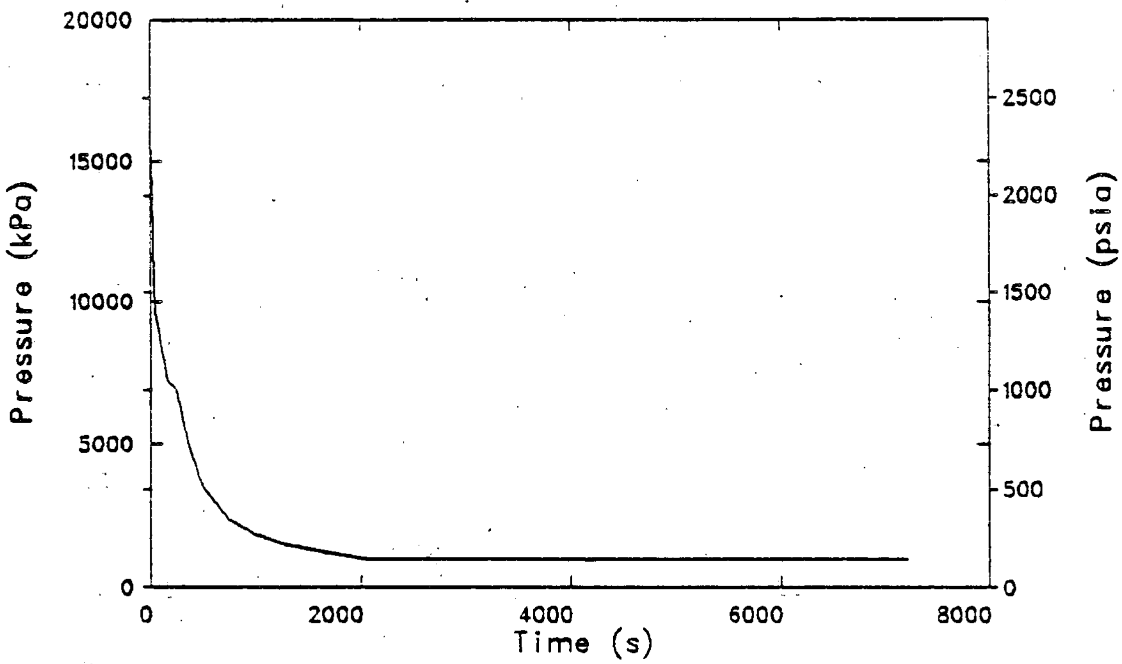

Figure A-45. Sequence 2-5 reactor vessel downcomer fluid pressure.

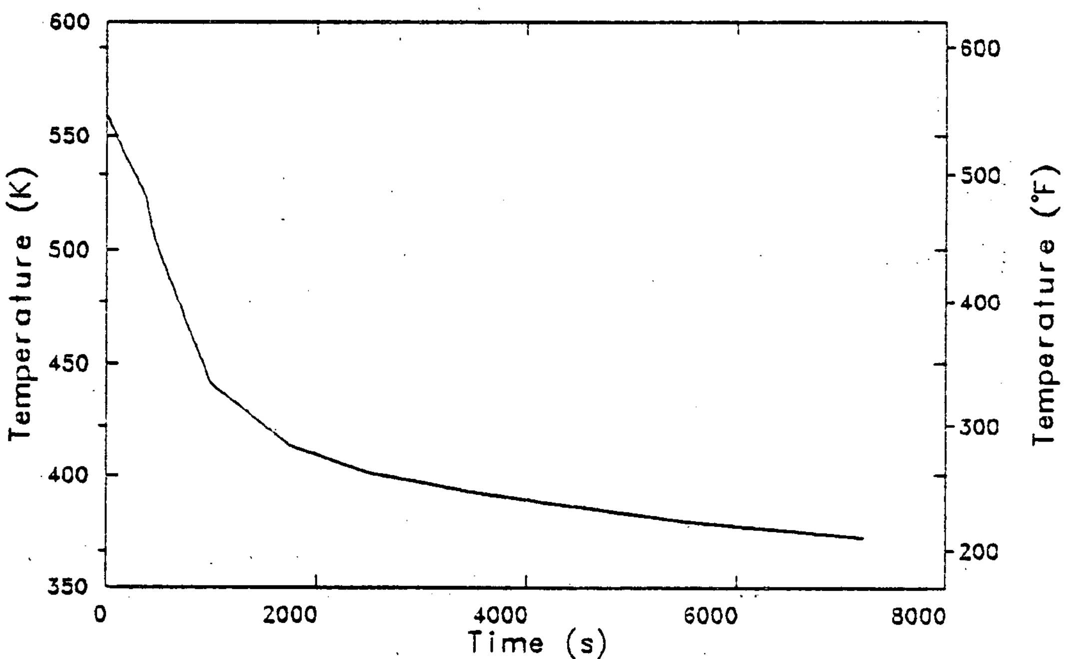

Figure A-46. Sequence 2-5 reactor vessel downcomer fluid temperature.

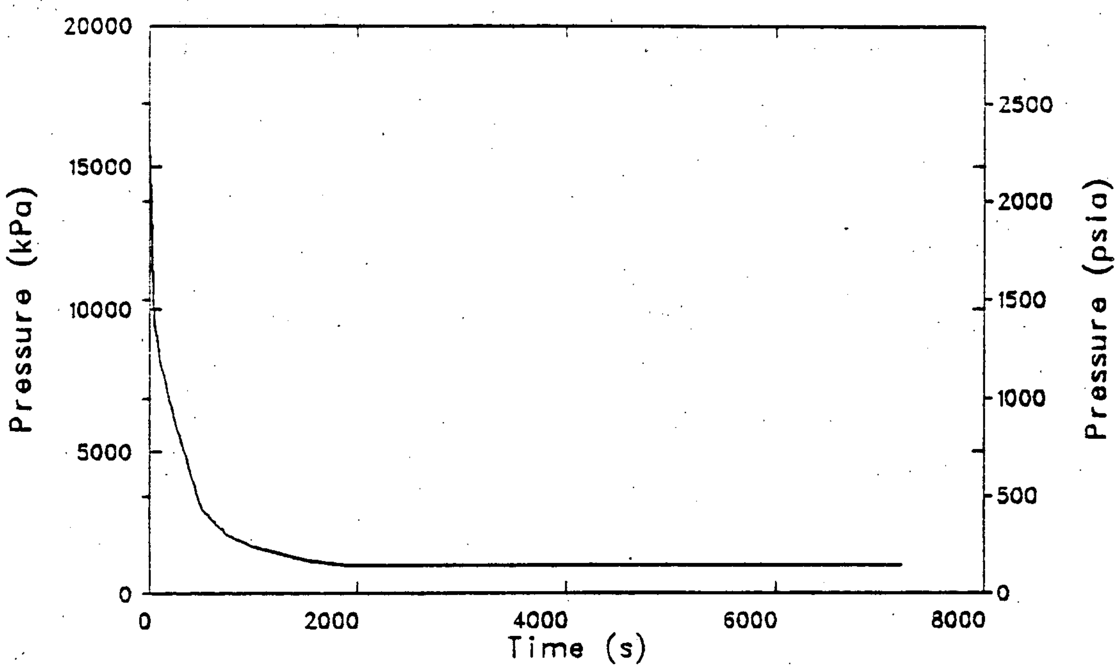

Figure A-47. Sequence 2-6 reactor vessel downcomer fluid pressure.

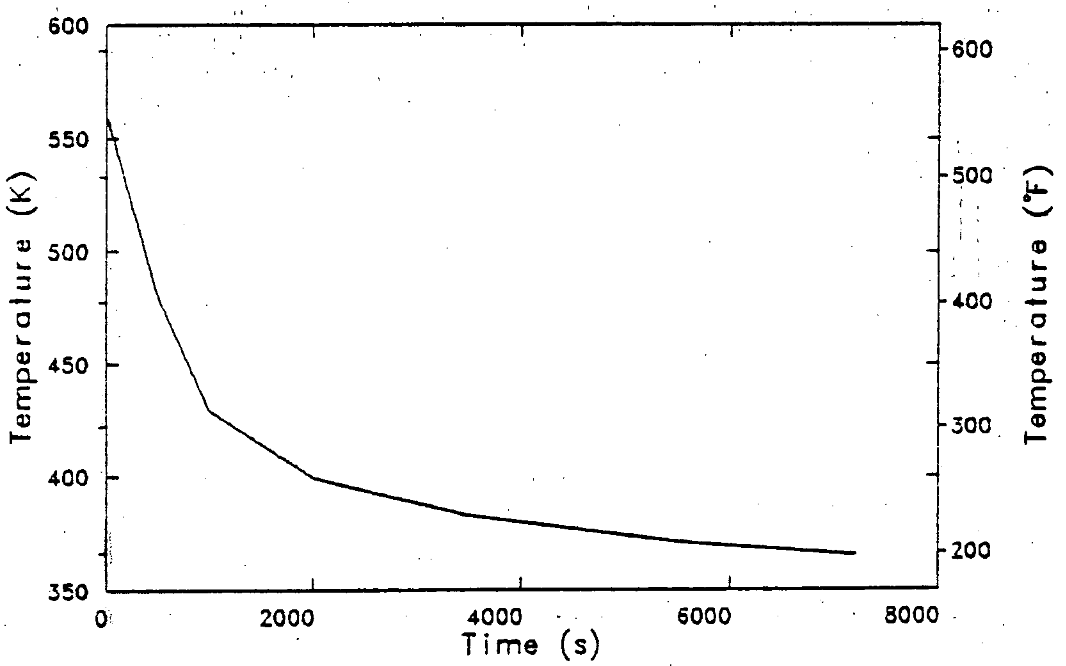

Figure A-48. Sequence 2-6 reactor vessel downcomer fluid temperature.

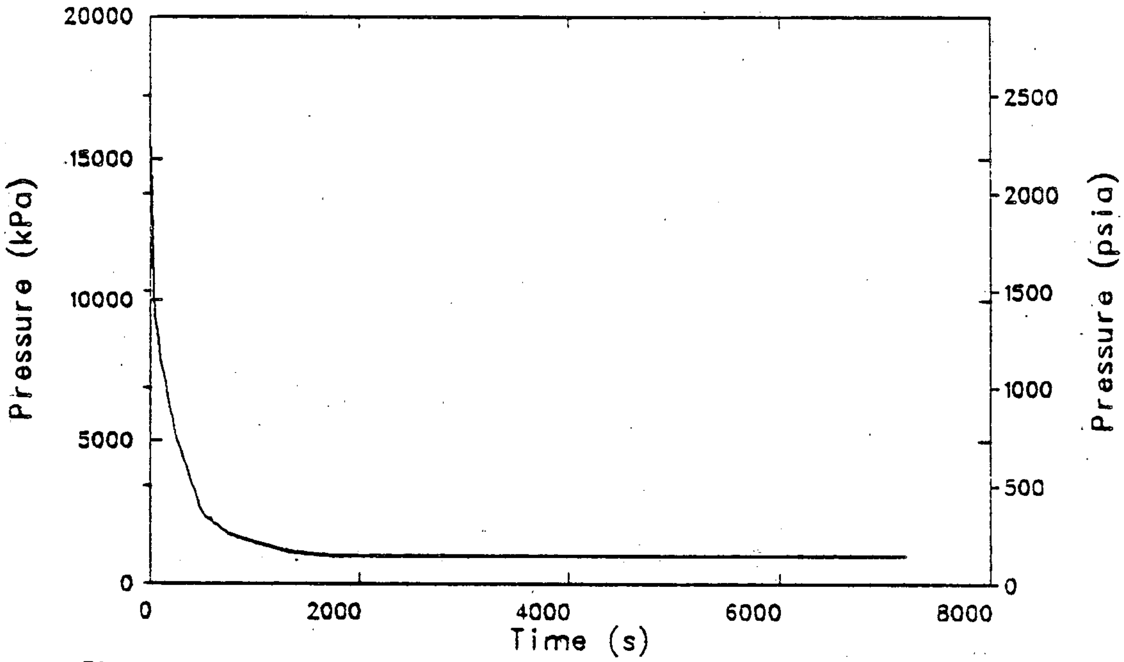

Figure A-49. Sequence 2-7 reactor vessel downcomer fluid pressure.

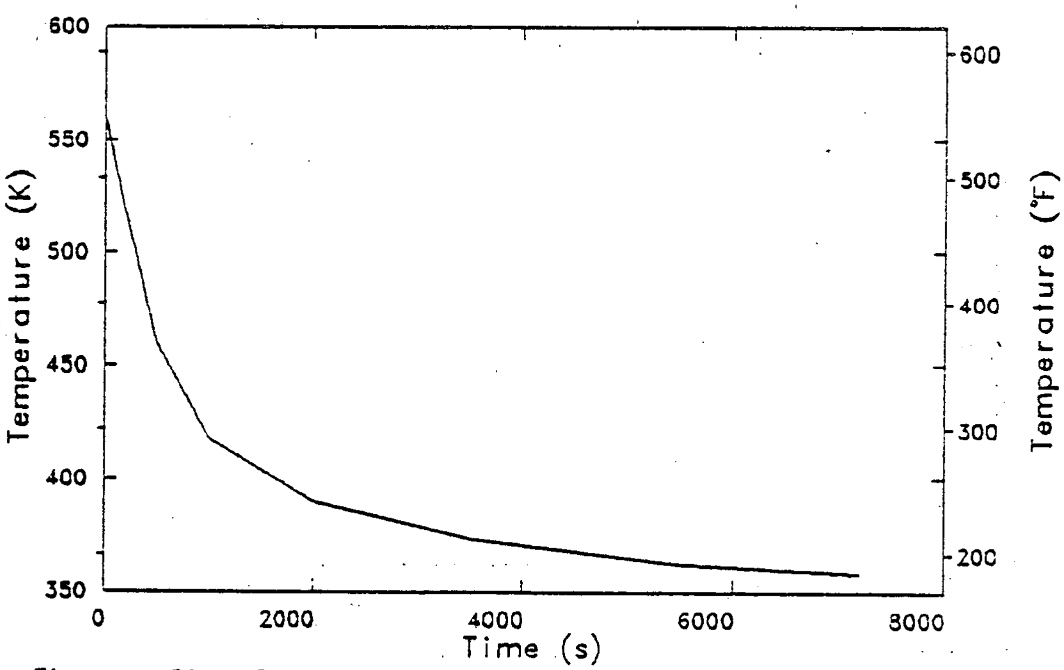

Figure A-50. Sequence 2-7 reactor vessel downcomer fluid temperature.

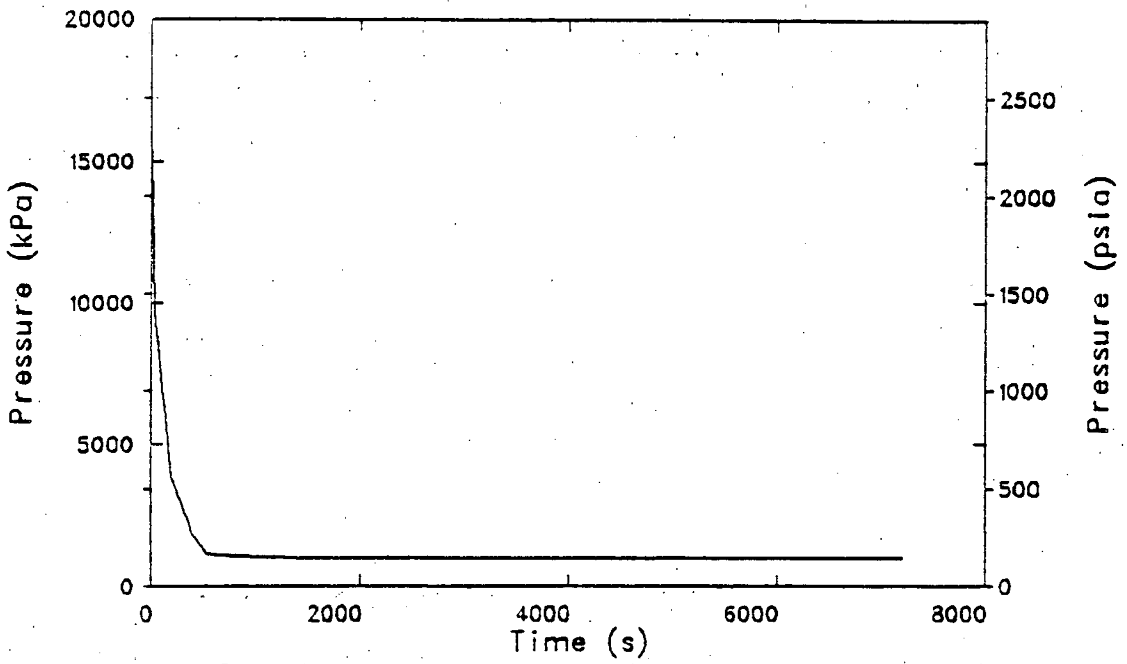

Figure A-51. Sequence 2-8 reactor vessel downcomer fluid pressure.

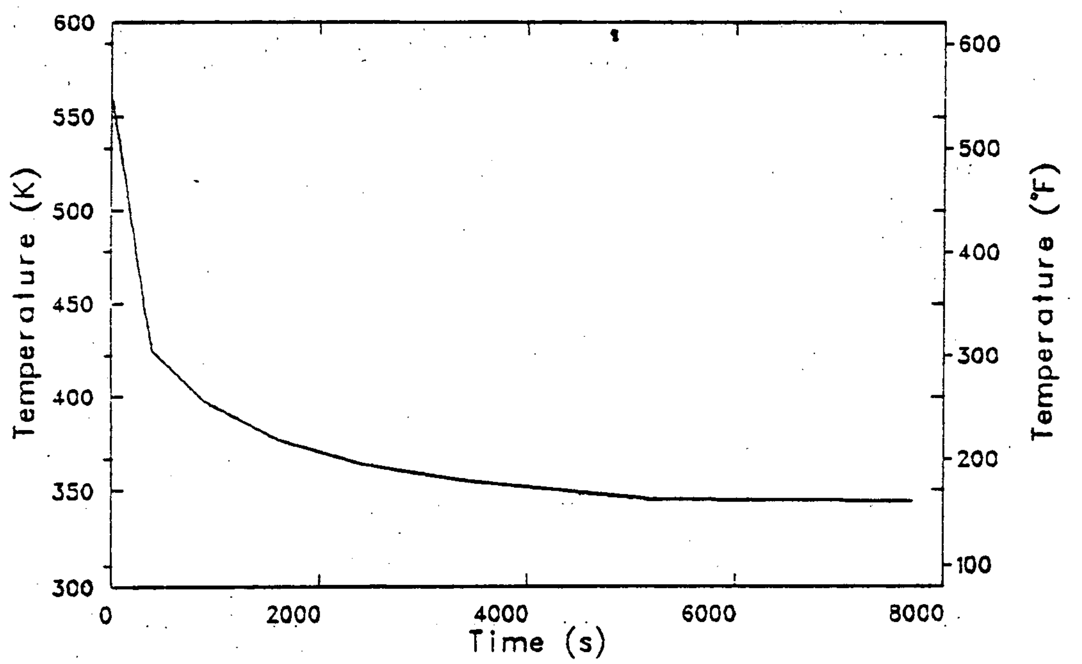

Figure A-52. Sequence 2-8 reactor vessel downcomer fluid temperature.

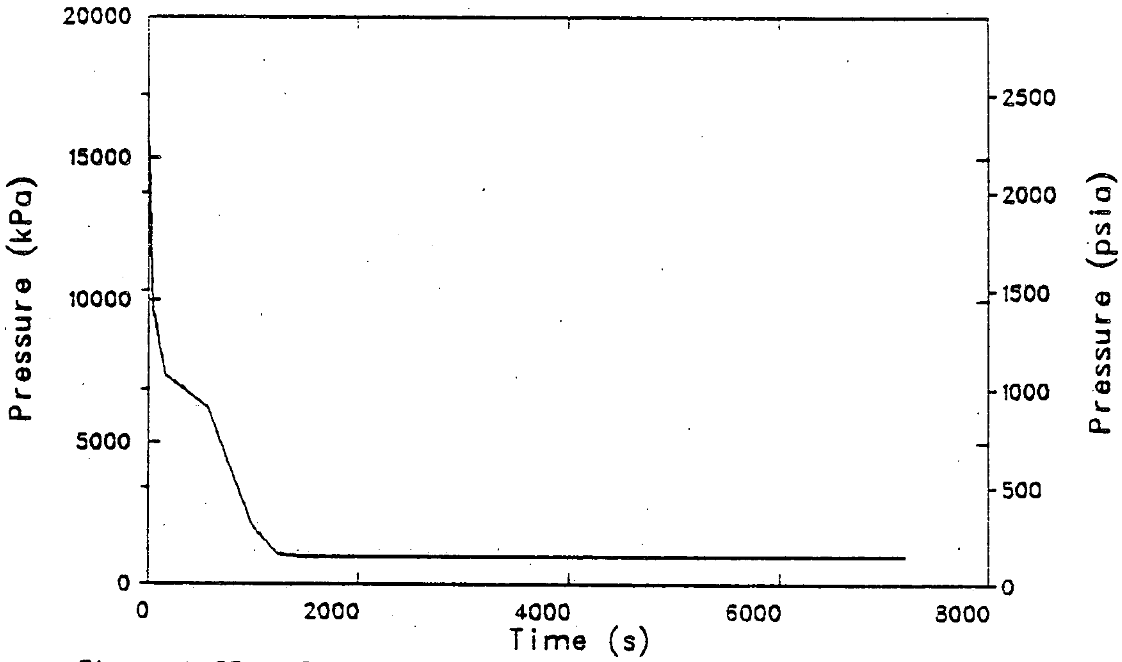

Figure A-53. Sequence 2-9 reactor vessel downcomer fluid pressure.

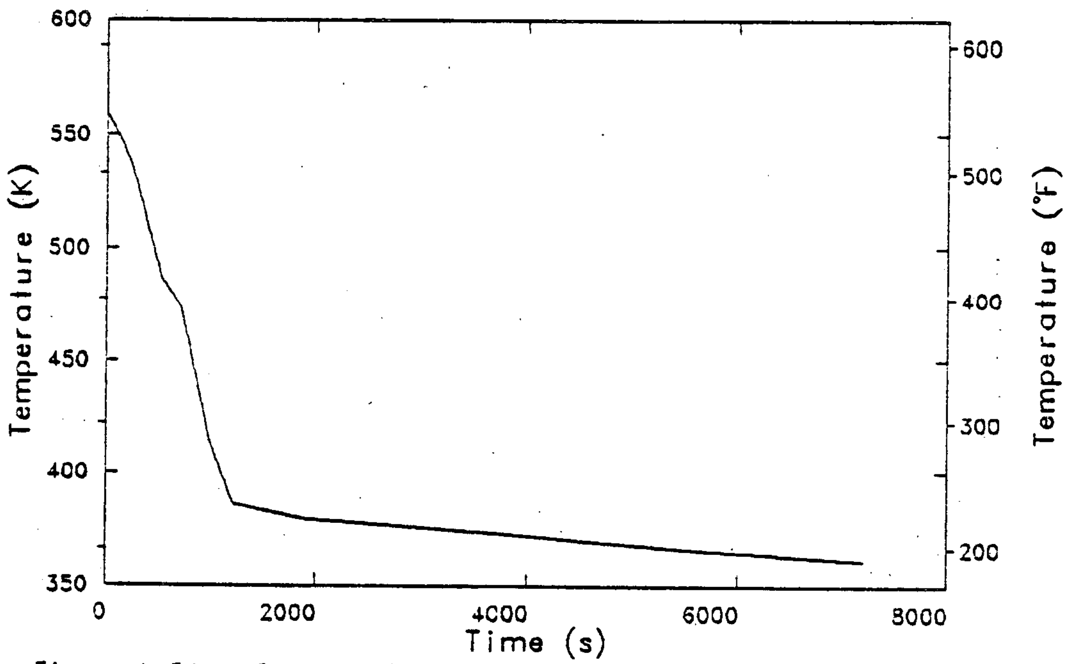

Figure A-54. Sequence 2-9 reactor vessel downcomer fluid temperature.

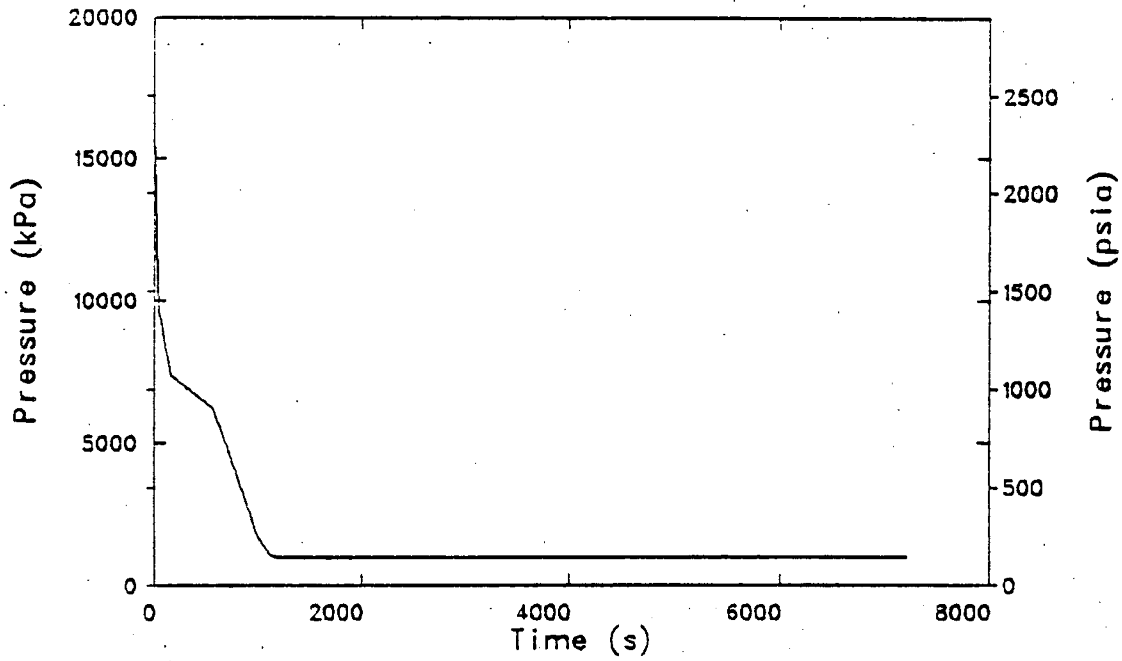

Figure A-55. Sequence 2-10 reactor vessel downcomer fluid pressure.

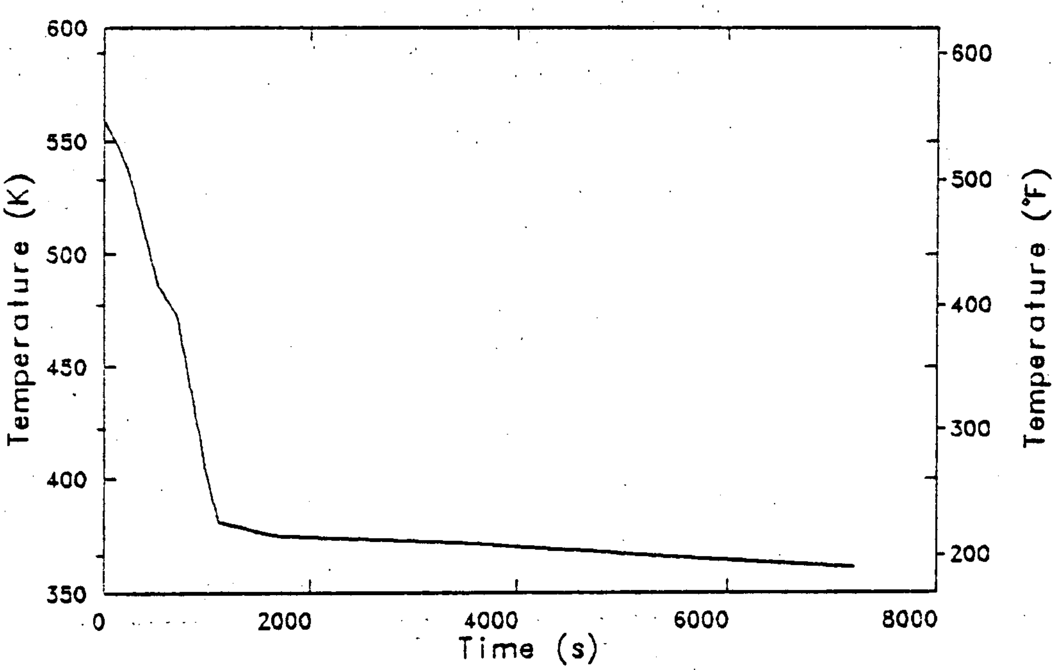

Figure A-56. Sequence 2-10 reactor vessel downcomer fluid temperature.

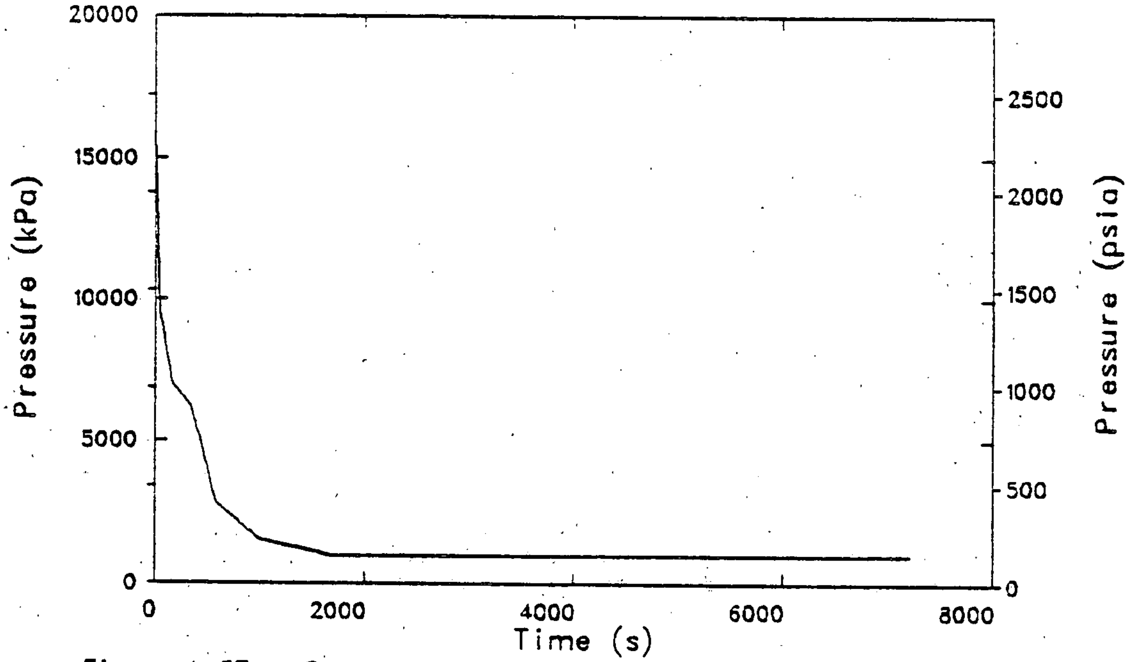

Figure A-57. Sequence 2-11 reactor vessel downcomer fluid pressure.

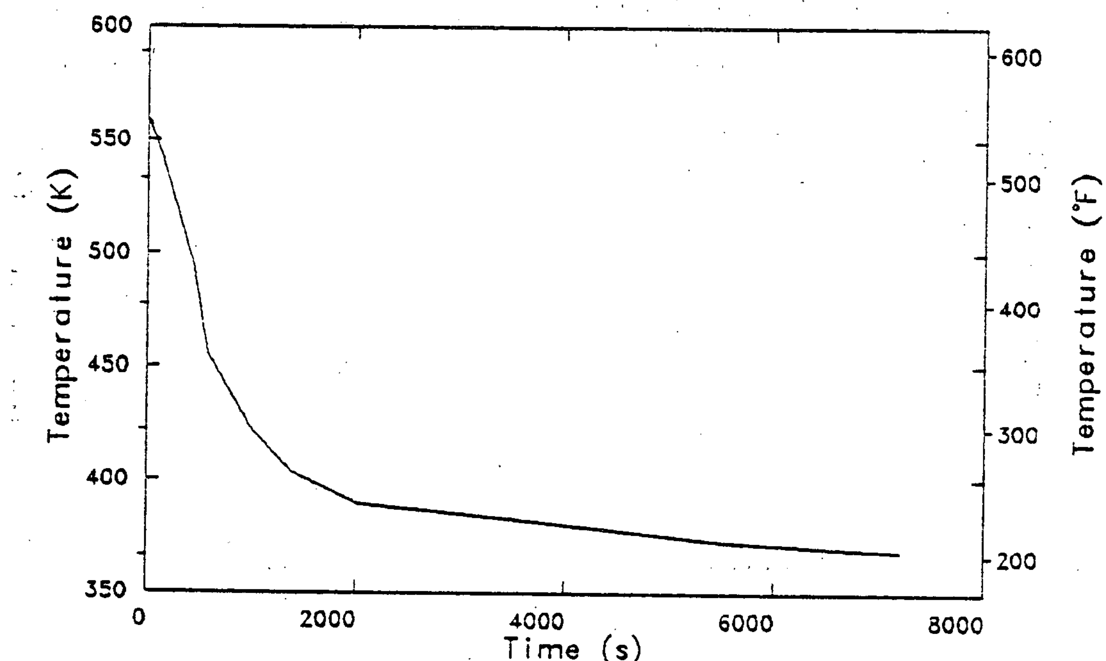

Figure A-58. Sequence 2-11 reactor vessel downcomer fluid temperature.

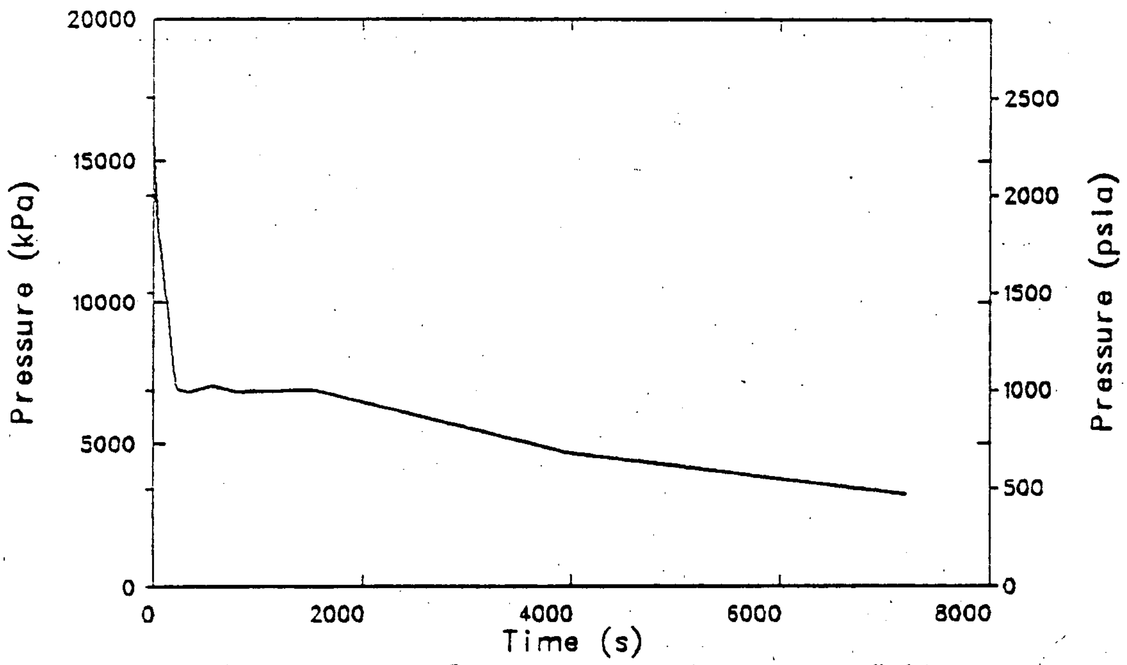

Figure A-59. Sequence 3-1 reactor vessel downcomer fluid pressure.

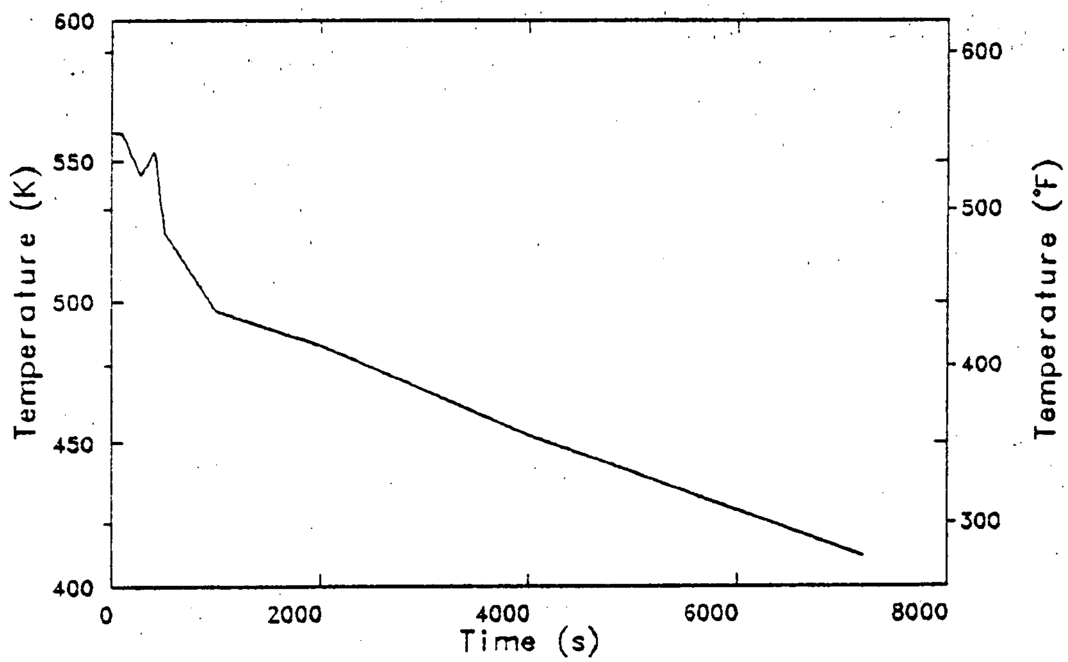

Figure A-60. Sequenca 3-1 reactor vessel downcomer fluid temperature.

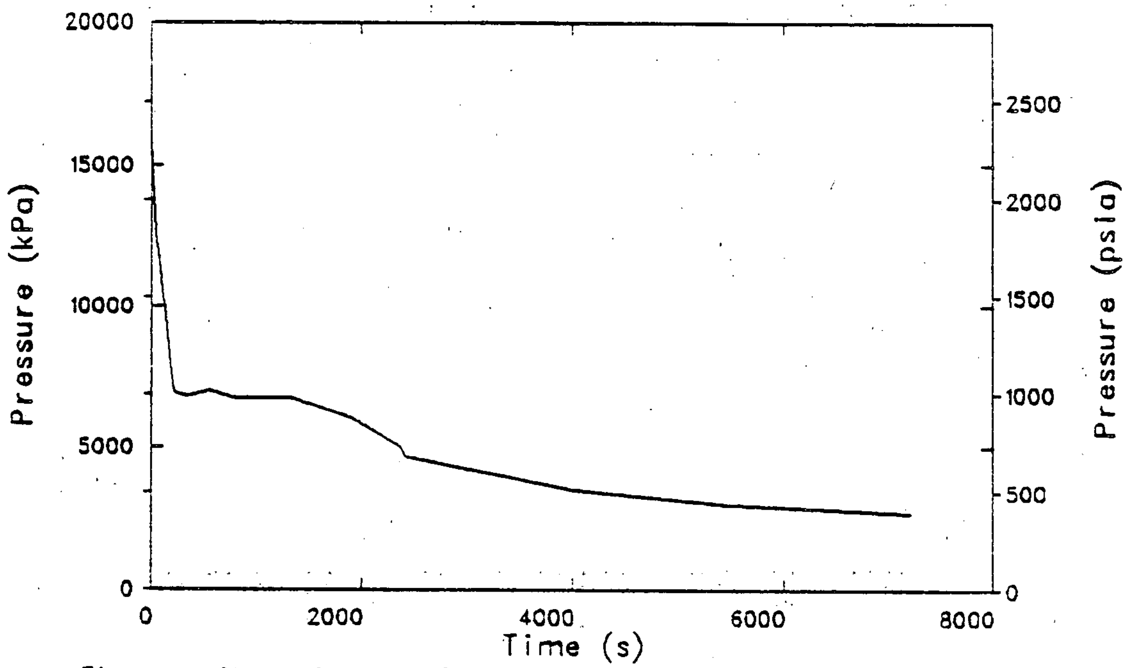

Figure A-61. Sequence 3-2 reactor vessel downcomer fluid pressure.

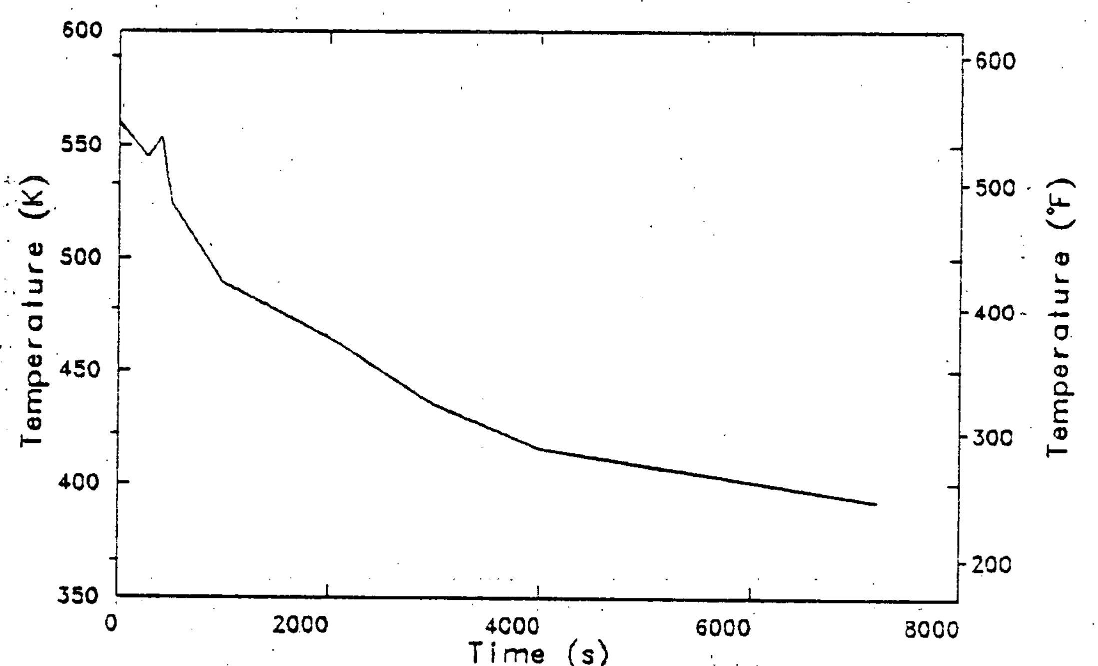

Figure A-62. Sequence 3-2 reactor vessel downcomer fluid temperature.

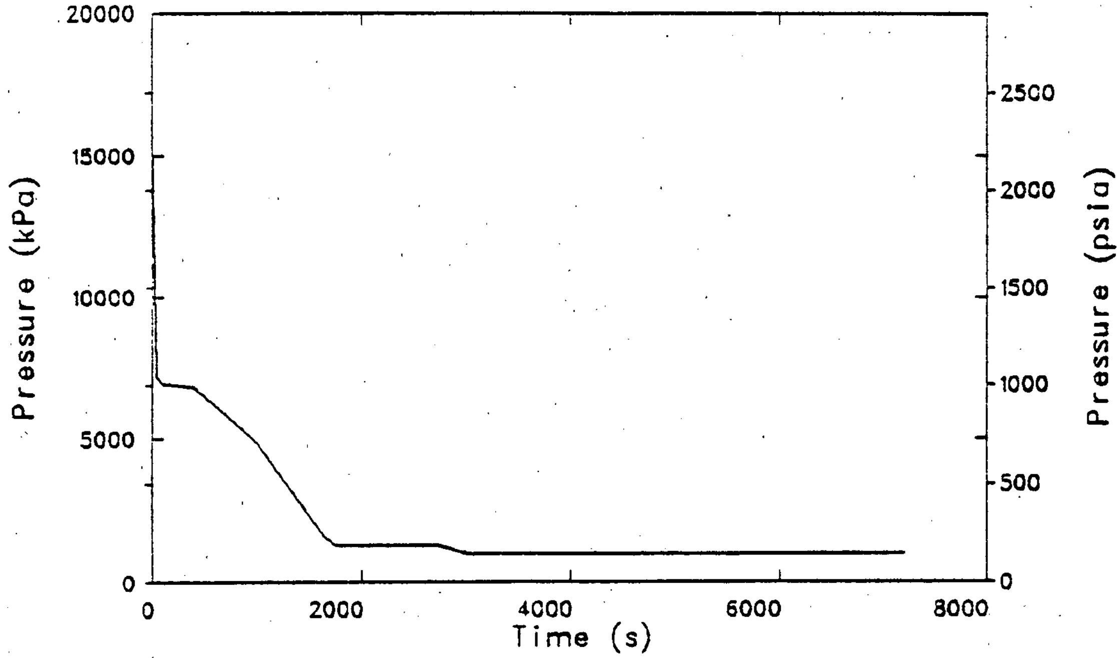

Figure A-63. Sequence 4-1 reactor vessel downcomer fluid pressure.

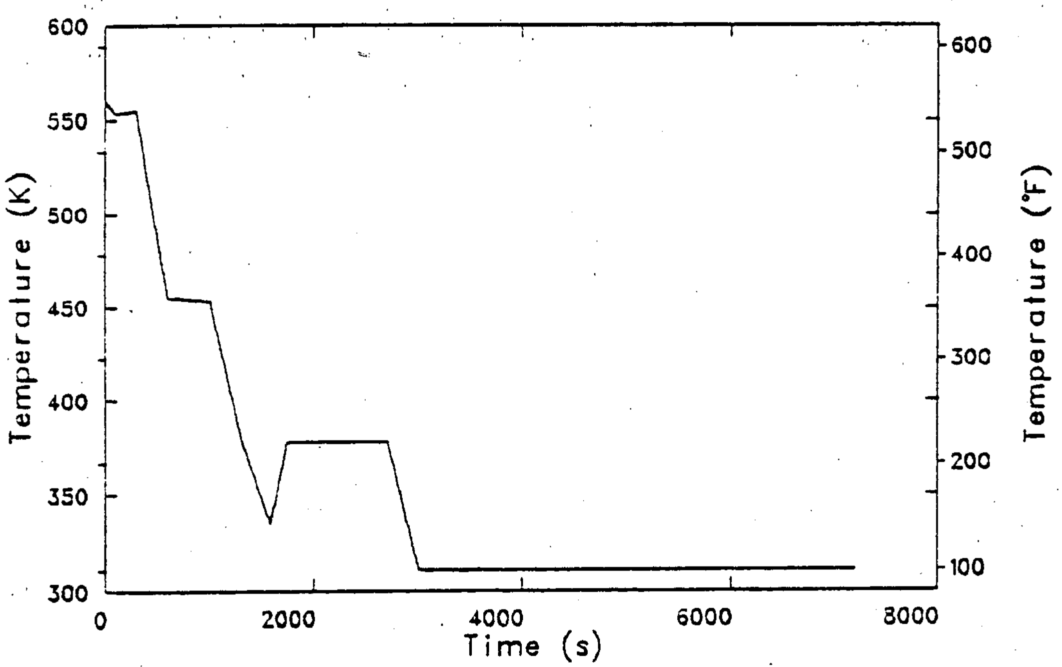

Figure A-64. Sequence 4-1 reactor vessel downcomer fluid temperature.

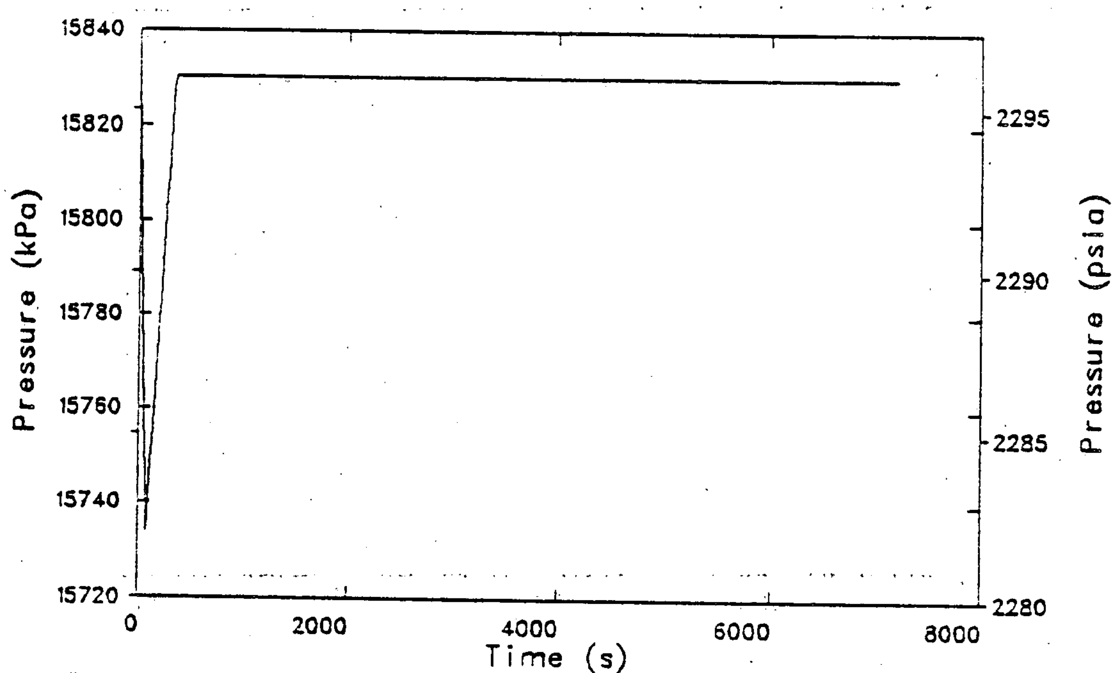

Figure A-65. Sequence 5-1 reactor vessel downcomer fluid pressure.

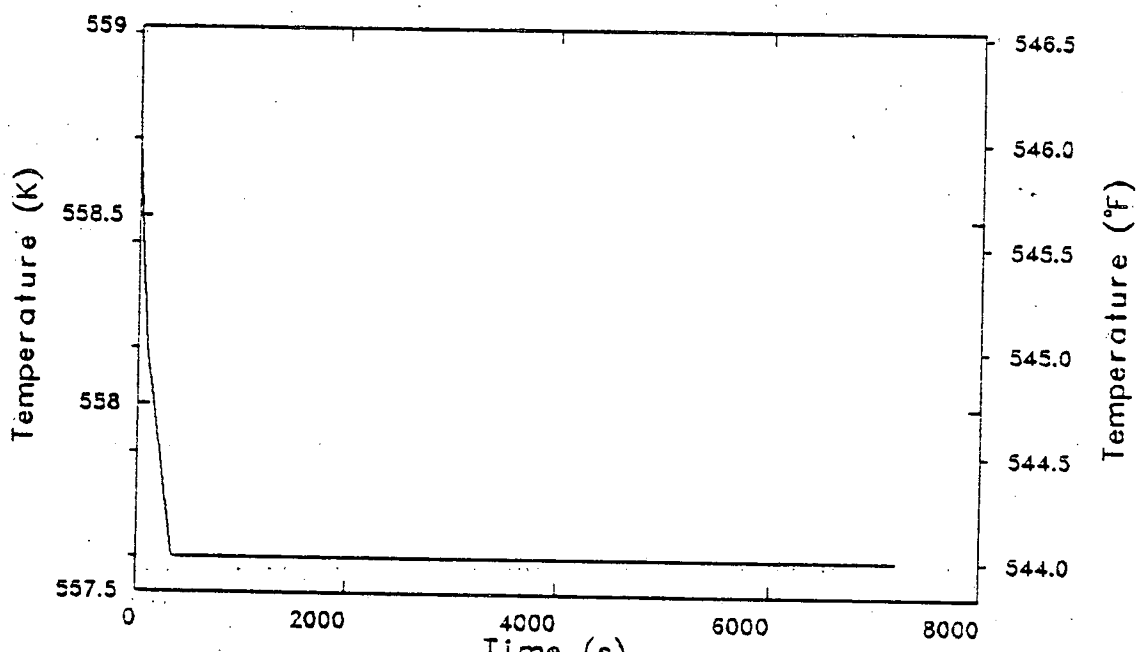

Figure A-66. Sequence 5-1 reactor vessel downcomer fluid temperature.

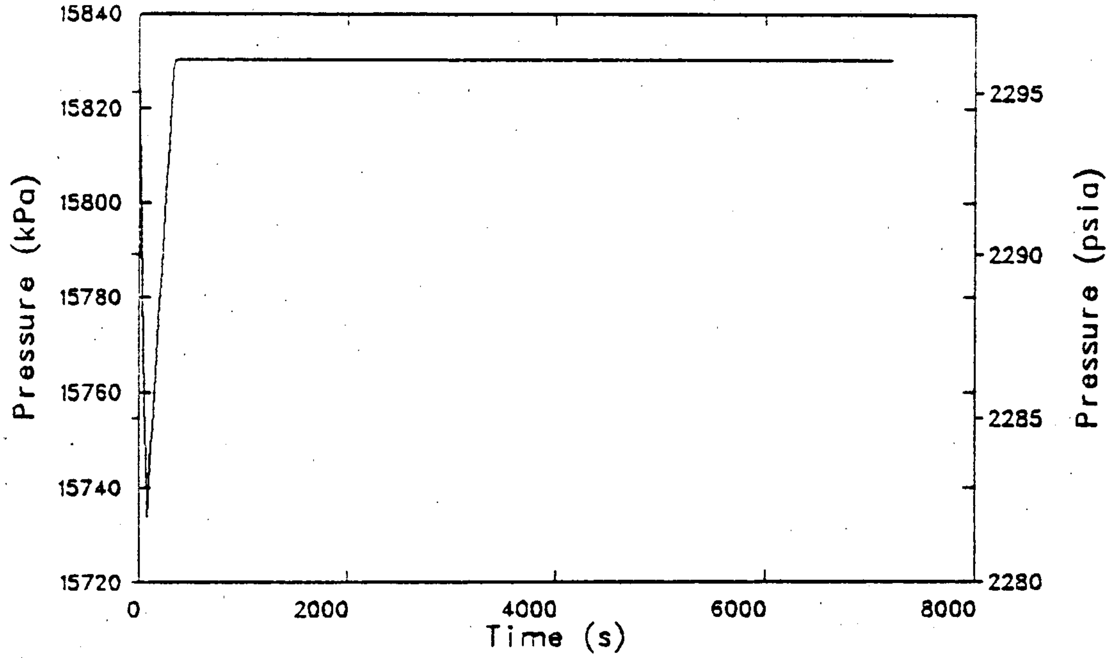

Figure A-67. Sequence 5-2 reactor vessel downcomer fluid pressure.

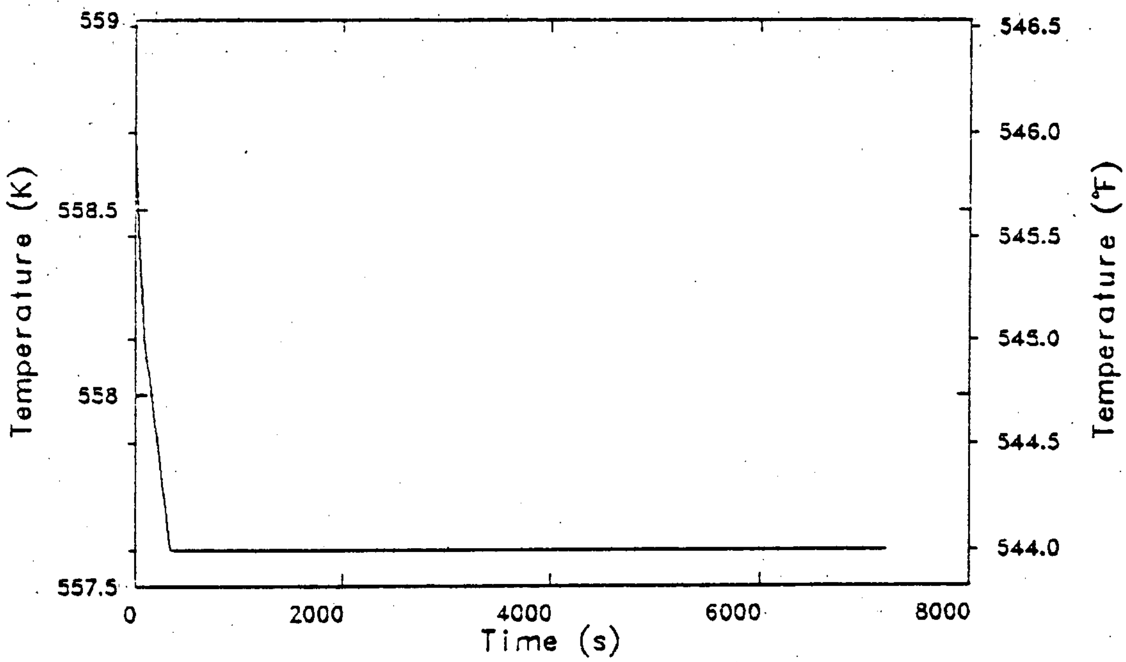

Figure A-68. Sequence 5-2 reactor vessel downcomer fluid temperature.

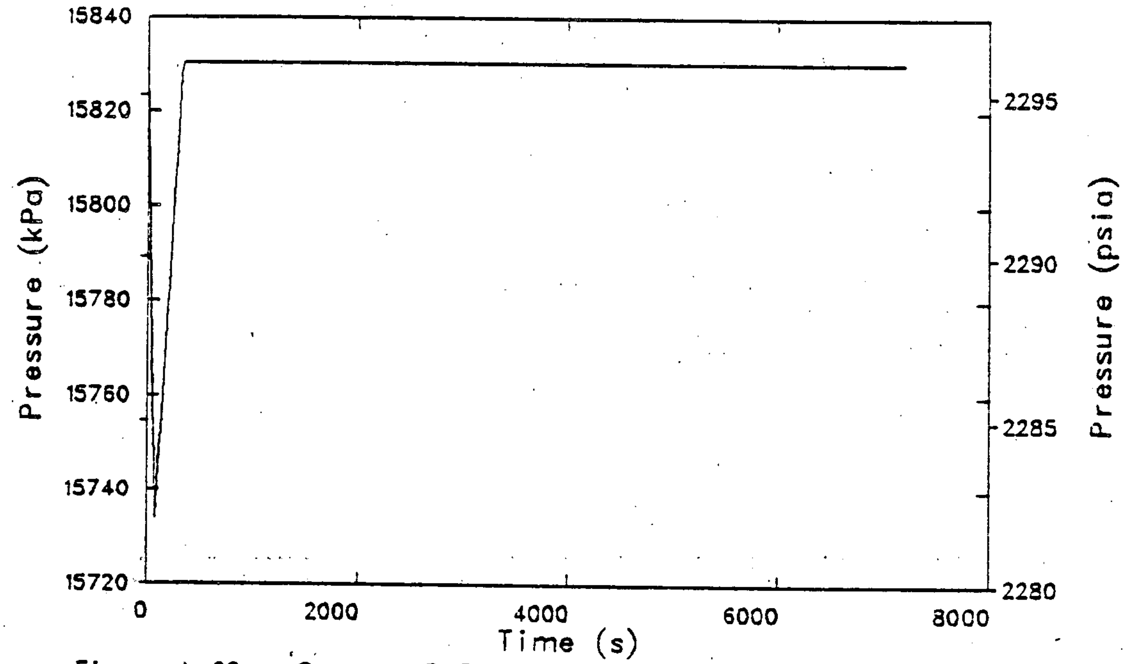

Figure A-69. Sequence 5-3 reactor vessel downcomer fluid pressure.

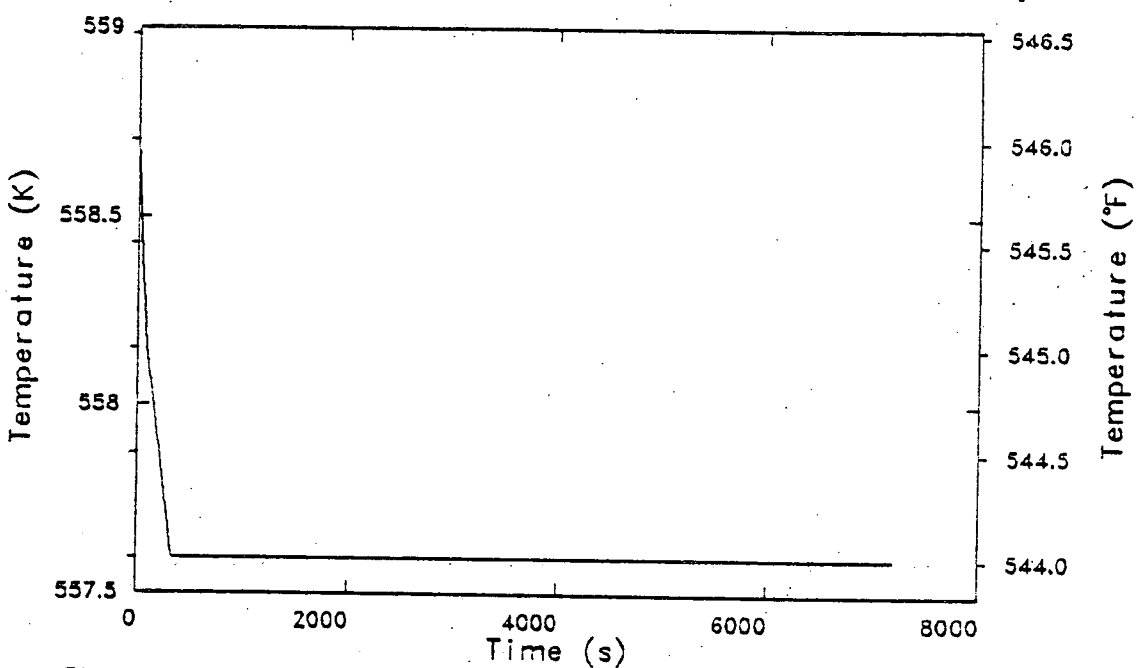

Figure A-70. Sequence 5-3 reactor vessel downcomer fluid temperature.

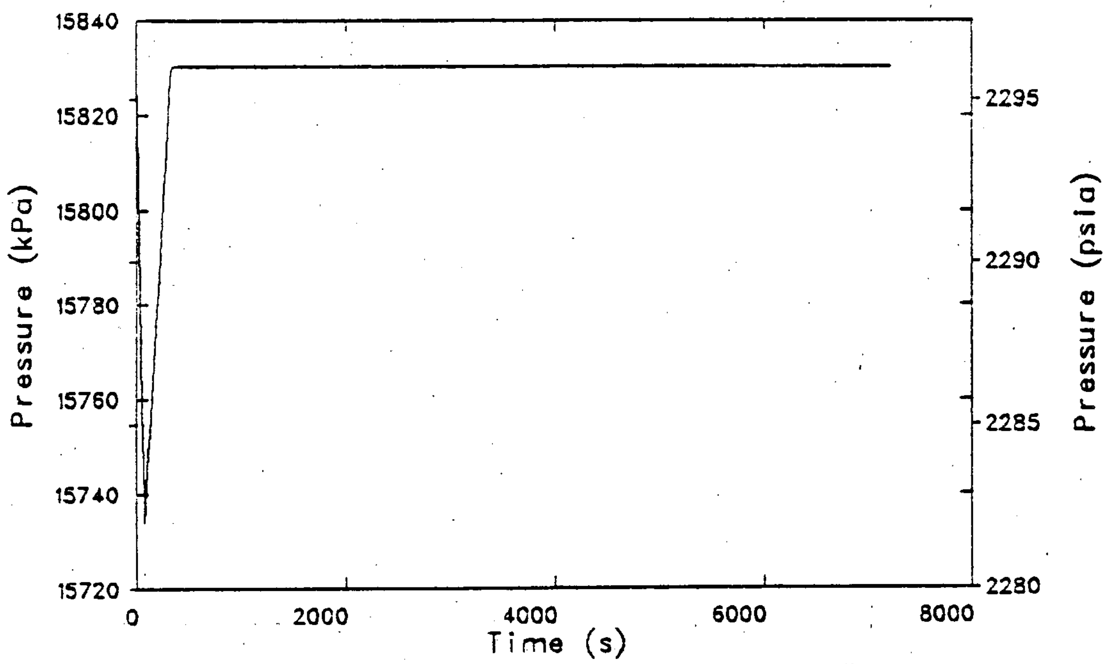

Figure A-71. Sequence 5-4 reactor vessel downcomer fluid pressure.

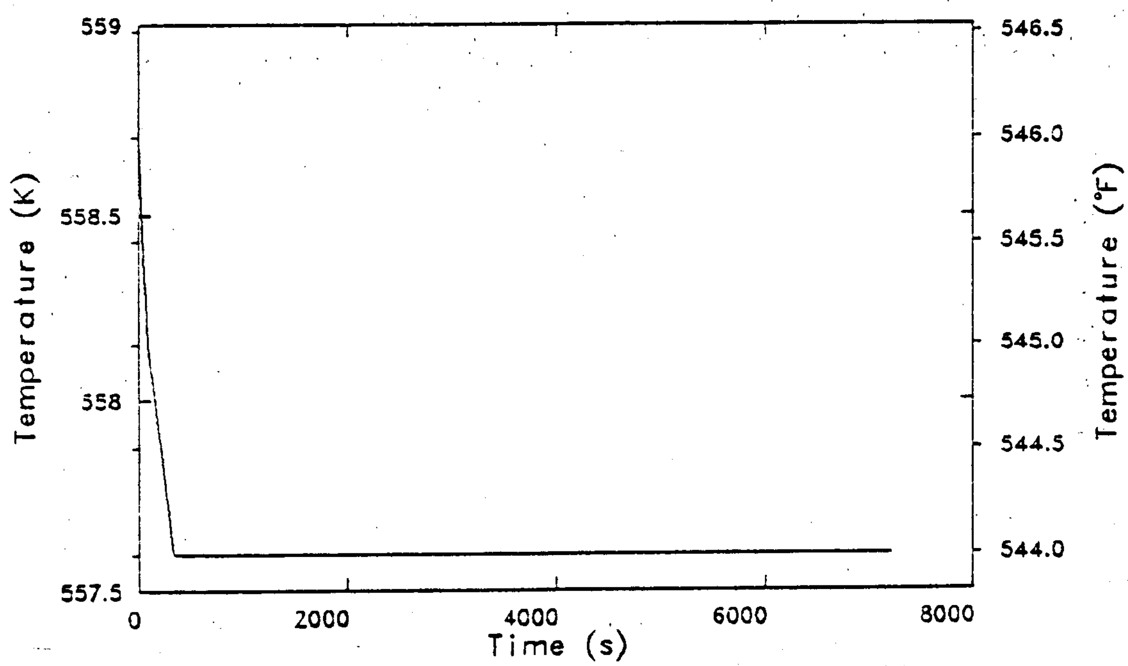

Figure A-72. Sequence 5-4 reactor vessel downcomer fluid temperature.

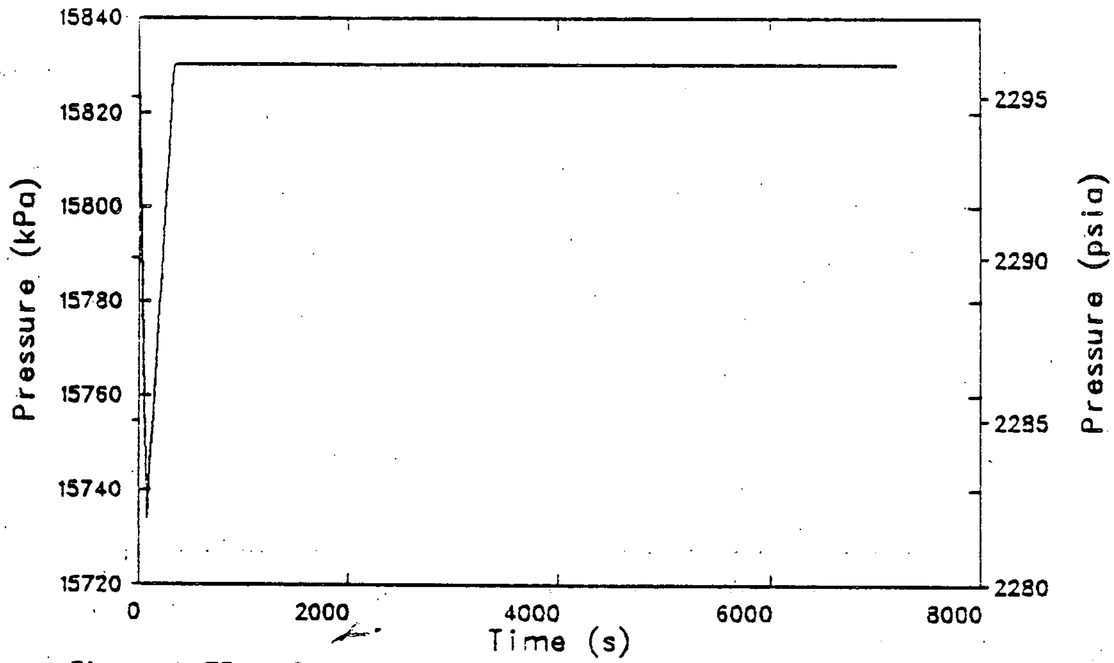

Figure A-73. Sequence 5-5 reactor vessel downcomer fluid pressure.

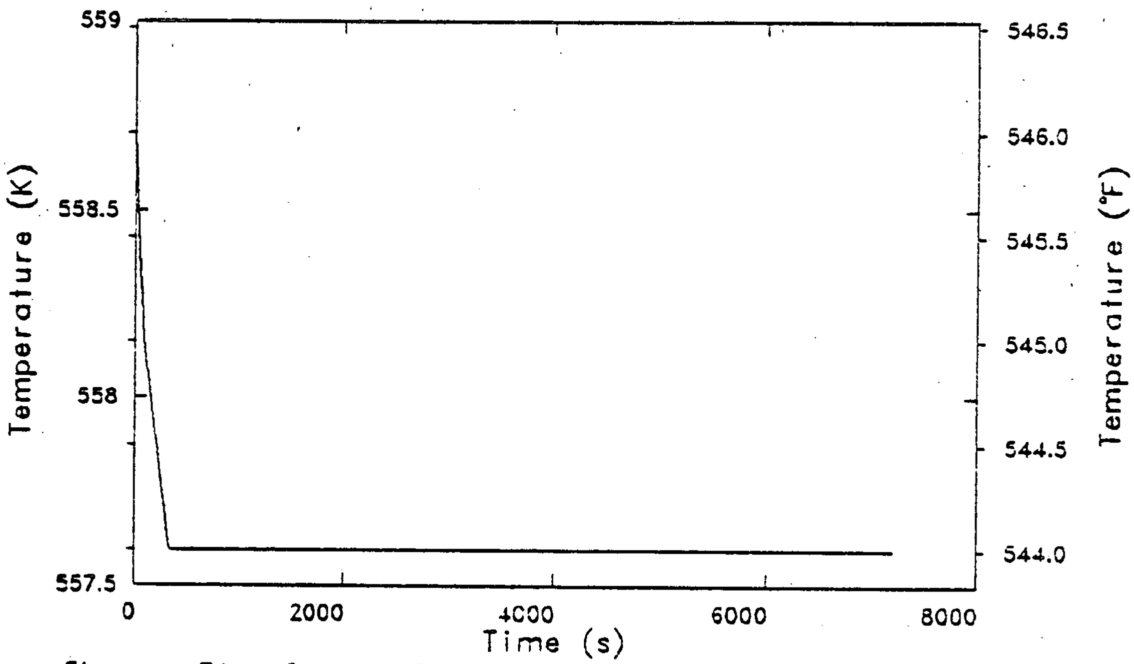

Figure A-74. Sequence 5-5 reactor vessel downcomer fluid temperature.

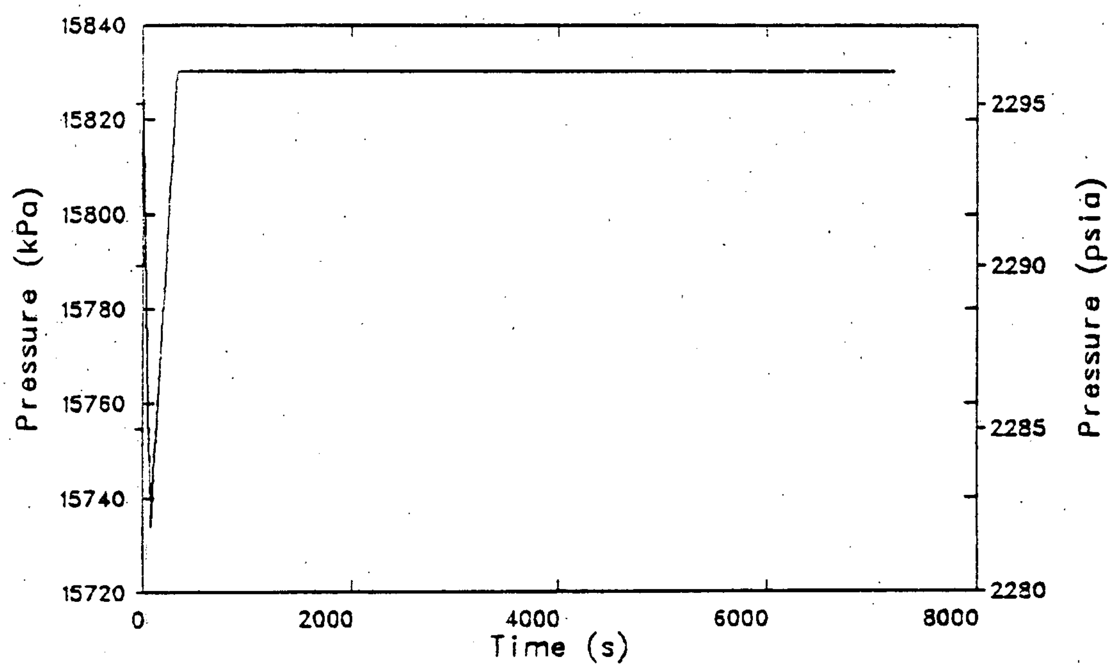

Figure A-75. Sequence 5-6 reactor vessel downcomer fluid pressure.

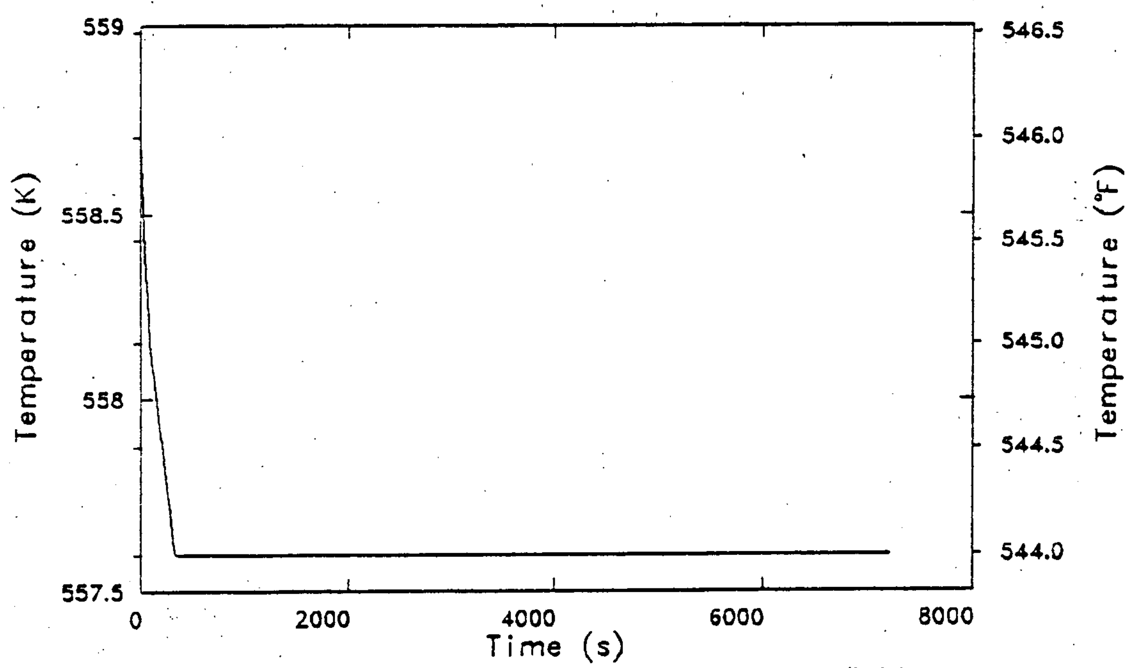

Figure A-76. Sequence 5-6 reactor vessel downcomer fluid temperature.

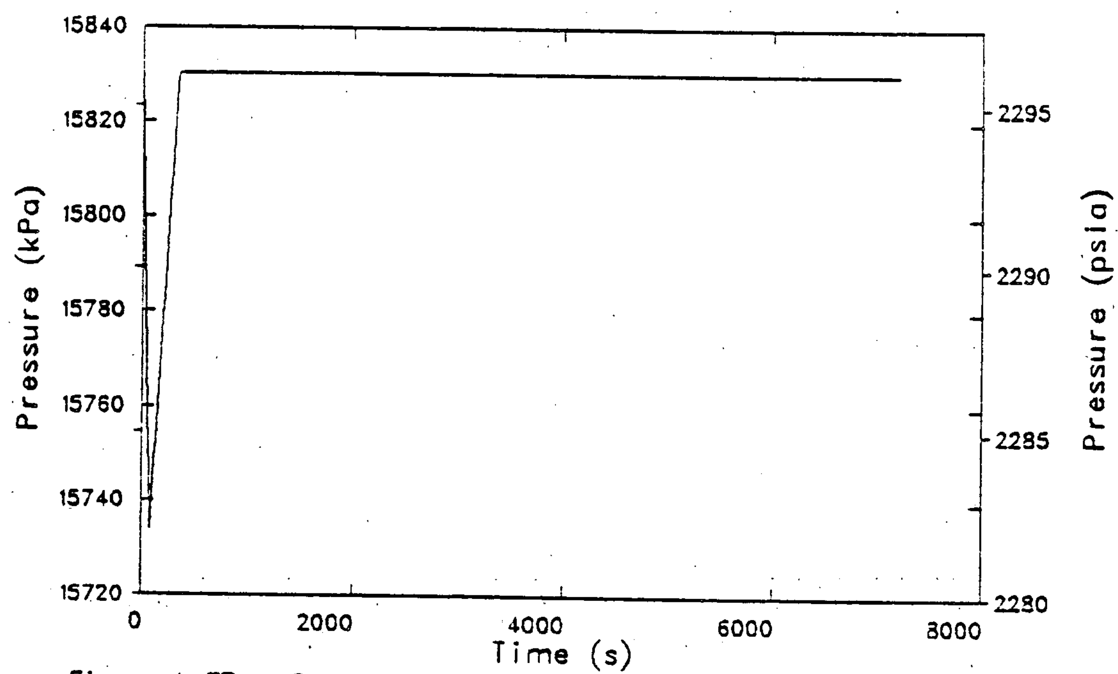

Figure A-77. Sequence 5-7 reactor vessel downcomer fluid pressure.

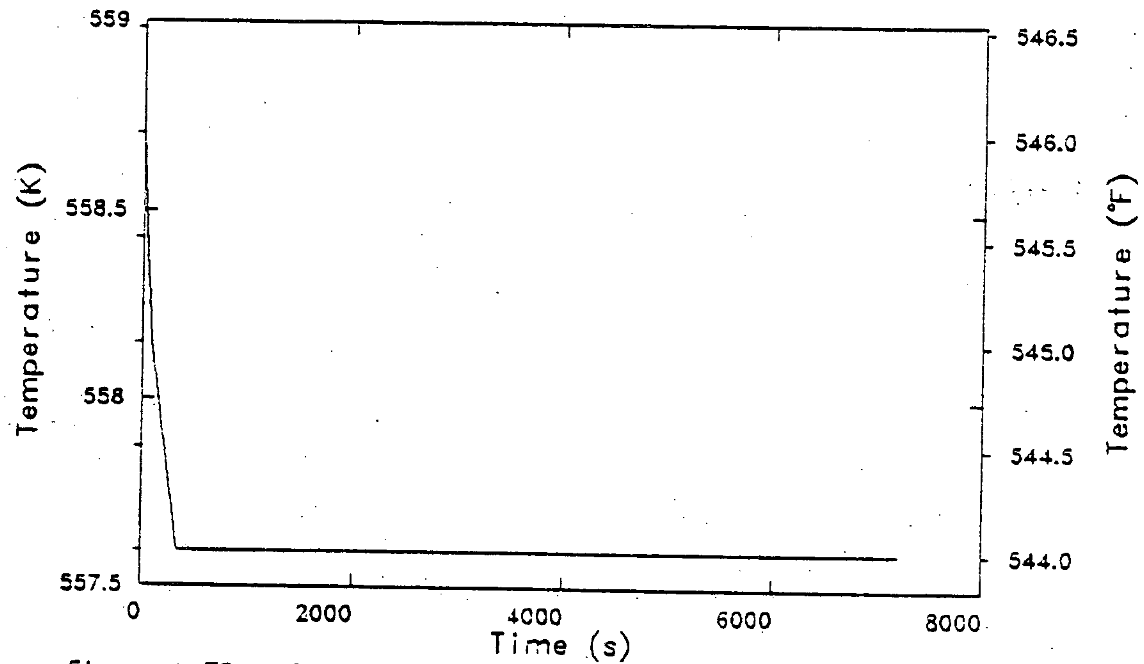

Figure A-78. Sequence 5-7 reactor vessel downcomer fluid temperature.

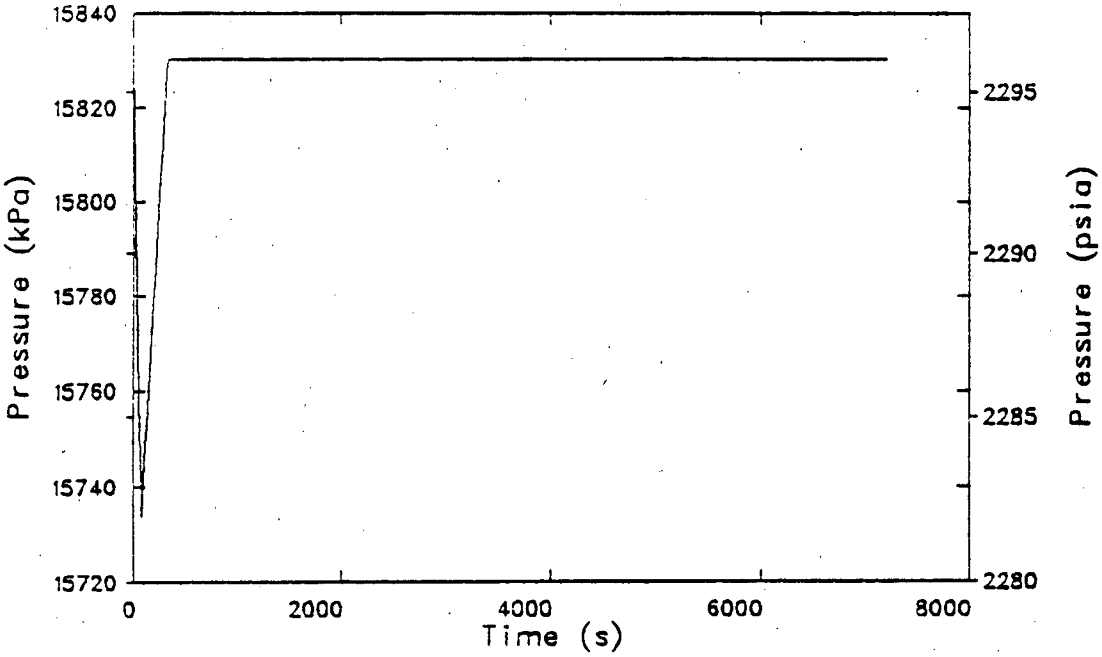

Figure A-79. Sequence 5-8 reactor vessel downcomer fluid pressure.

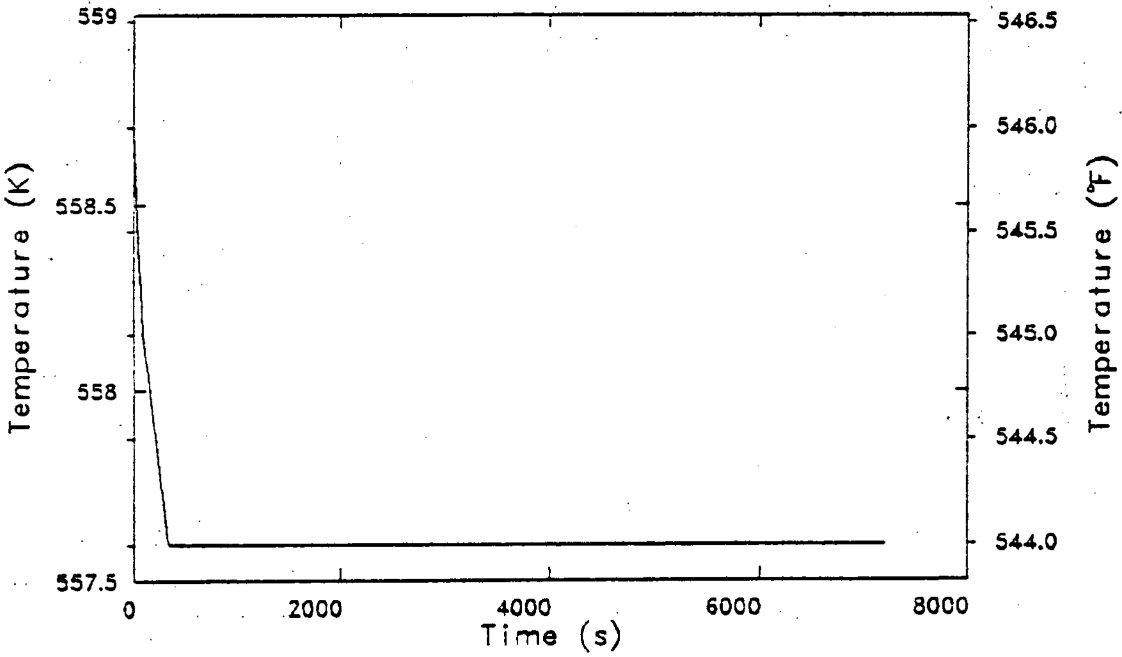

Figure A-80. Sequence 5-8 reactor vessel downcomer fluid temperature.

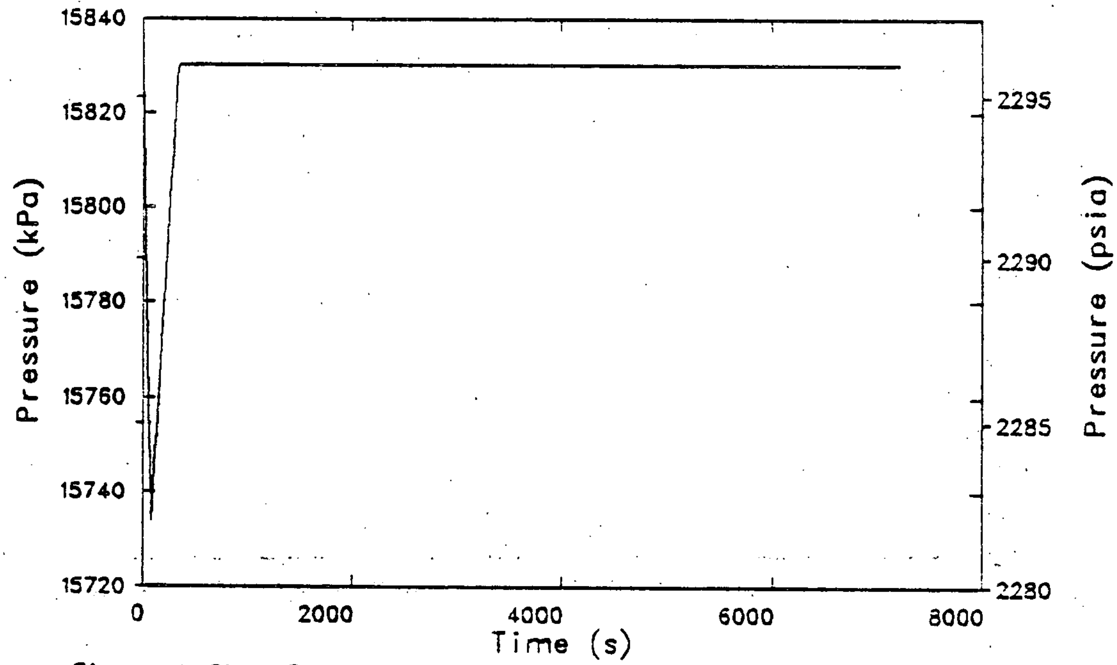

Figure A-81. Sequence 5-9 reactor vessel downcomer fluid pressure.

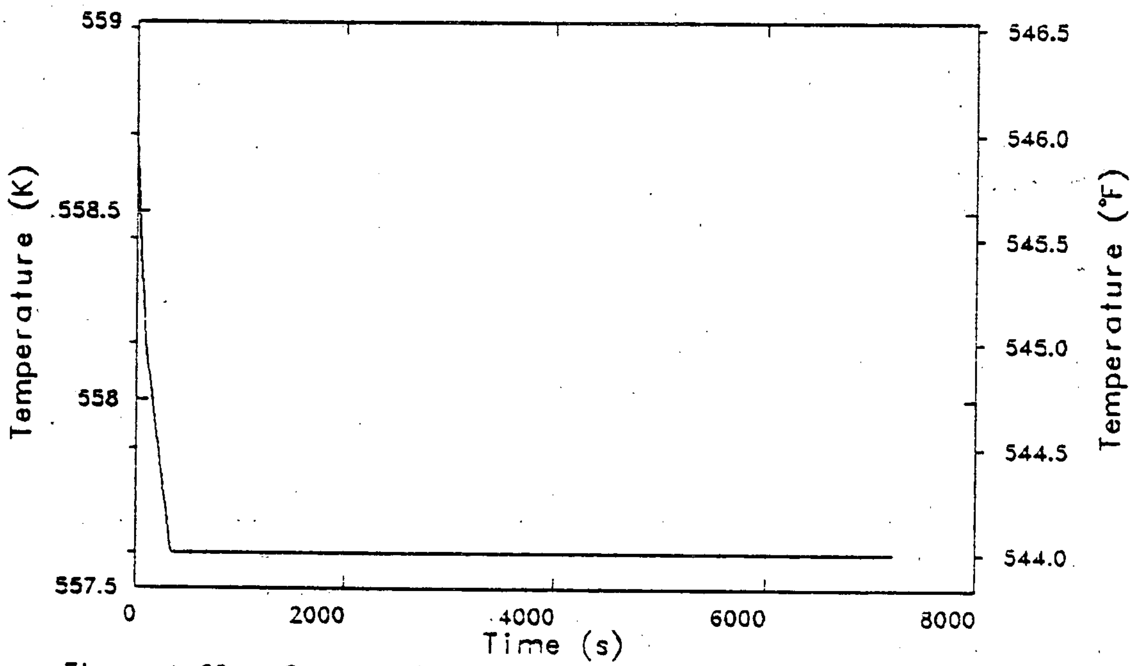

Figure A-82. Sequence 5-9 reactor vessel downcomer fluid temperature.

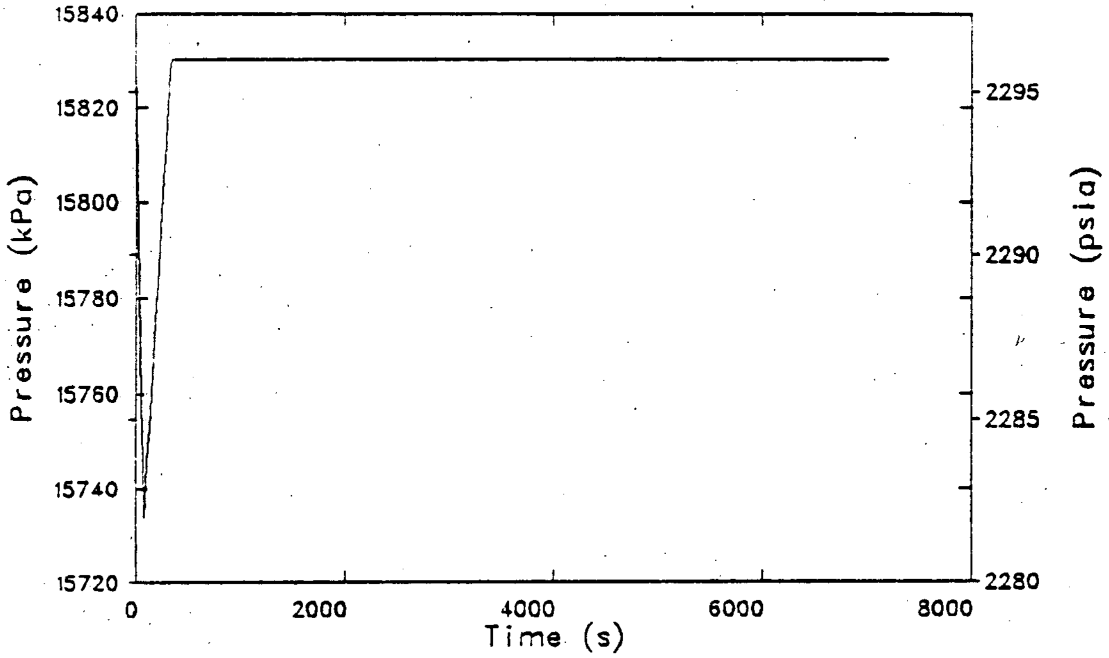

Figure A-83. Sequence 5-10 reactor vessel downcomer fluid pressure.

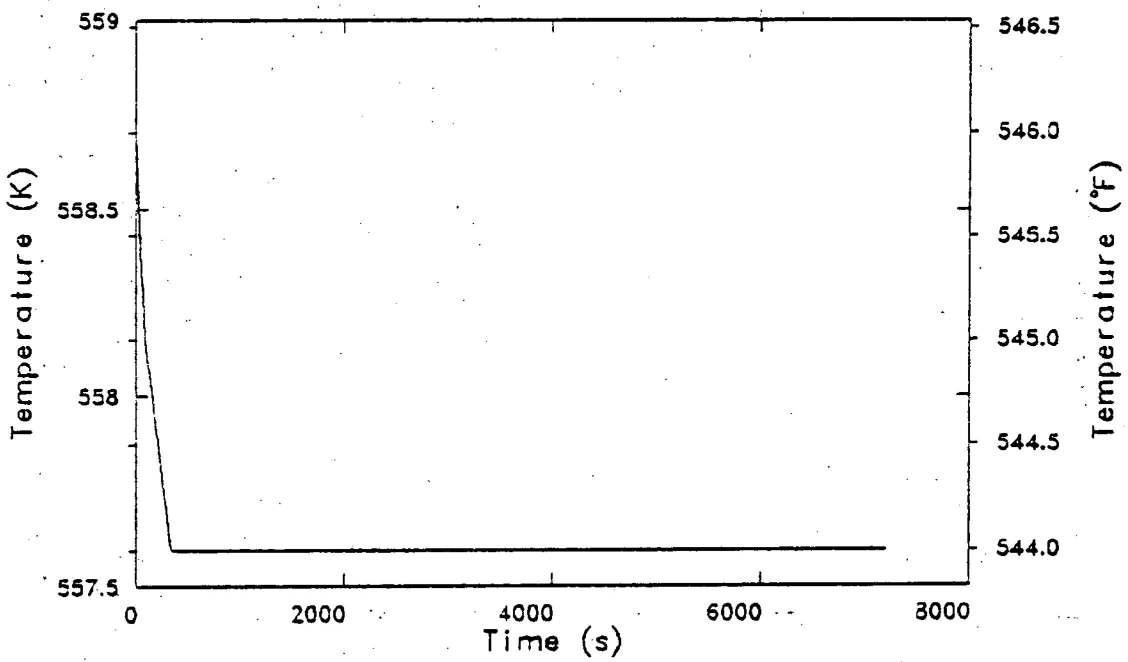

Figure A-84. Sequence 5-10 reactor vessel downcomer fluid temperature.

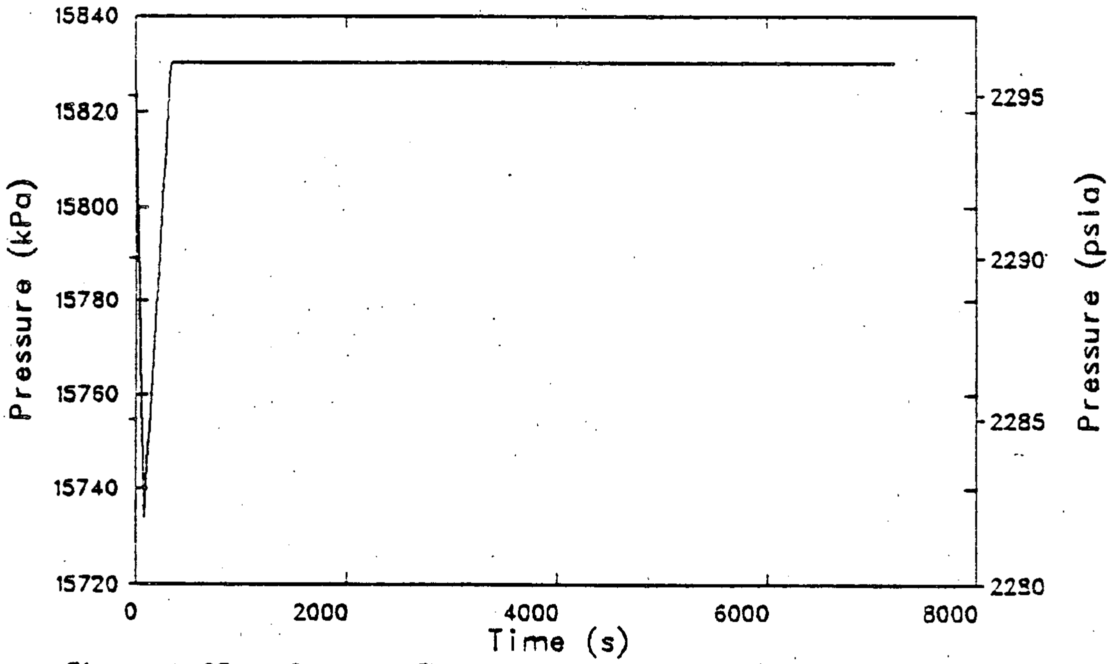

Figure A-85. Sequence 5-11 reactor vessel downcomer fluid pressure.

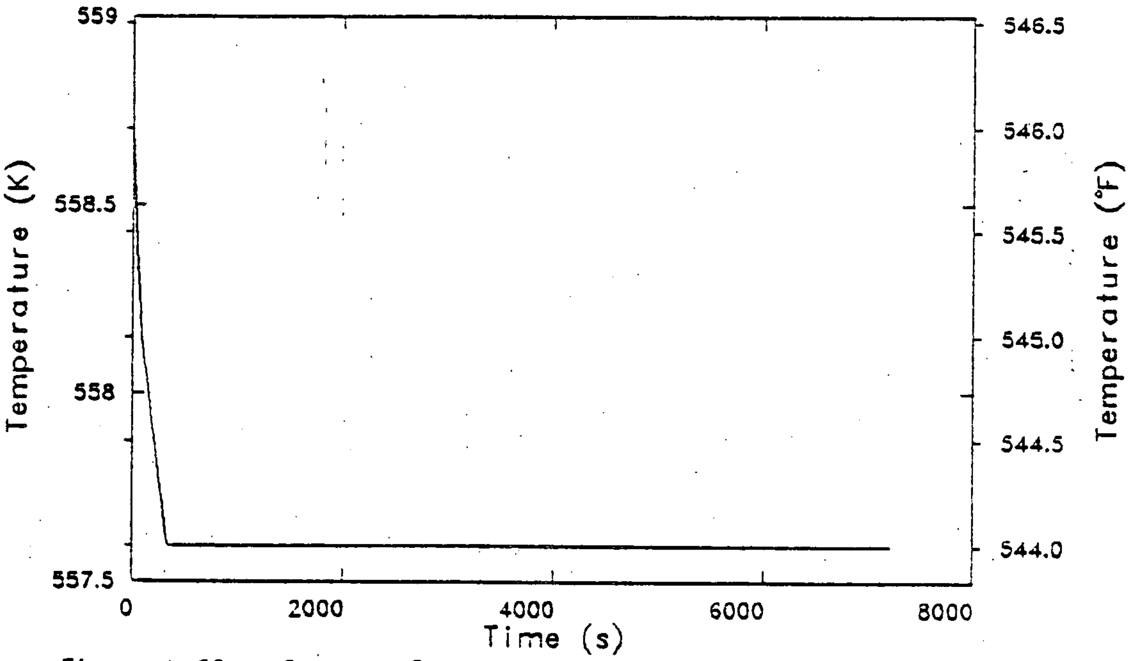

Figure A-86. Sequence 5-11 reactor vessel downcomer fluid temperature.

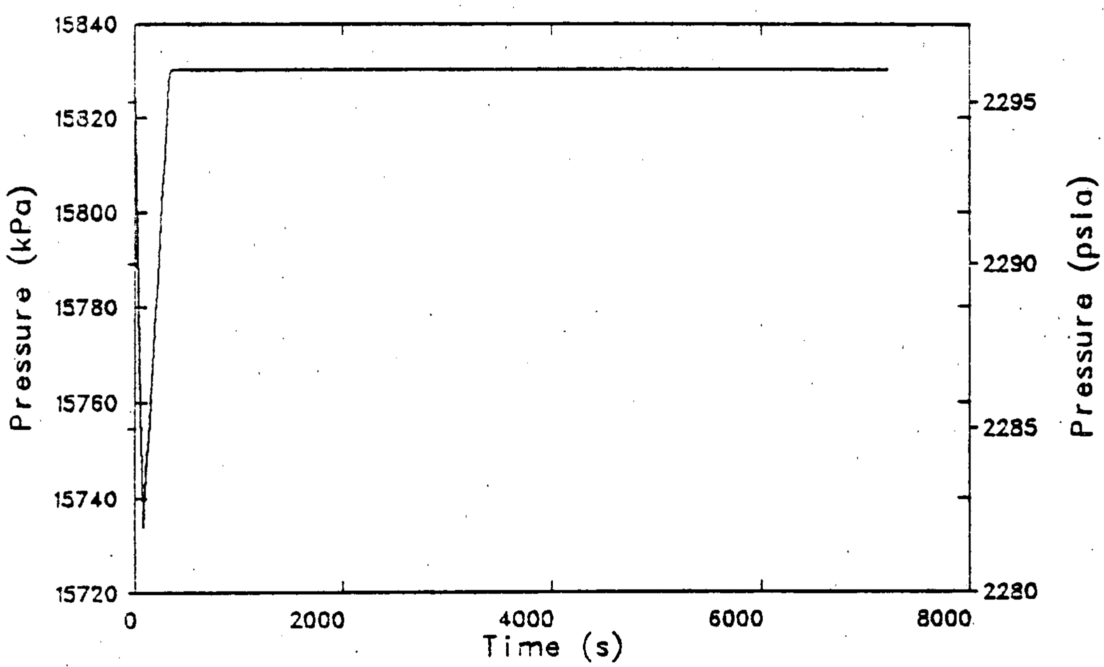

Figure A-87. Sequence 5-12 reactor vessel downcomer fluid pressure.

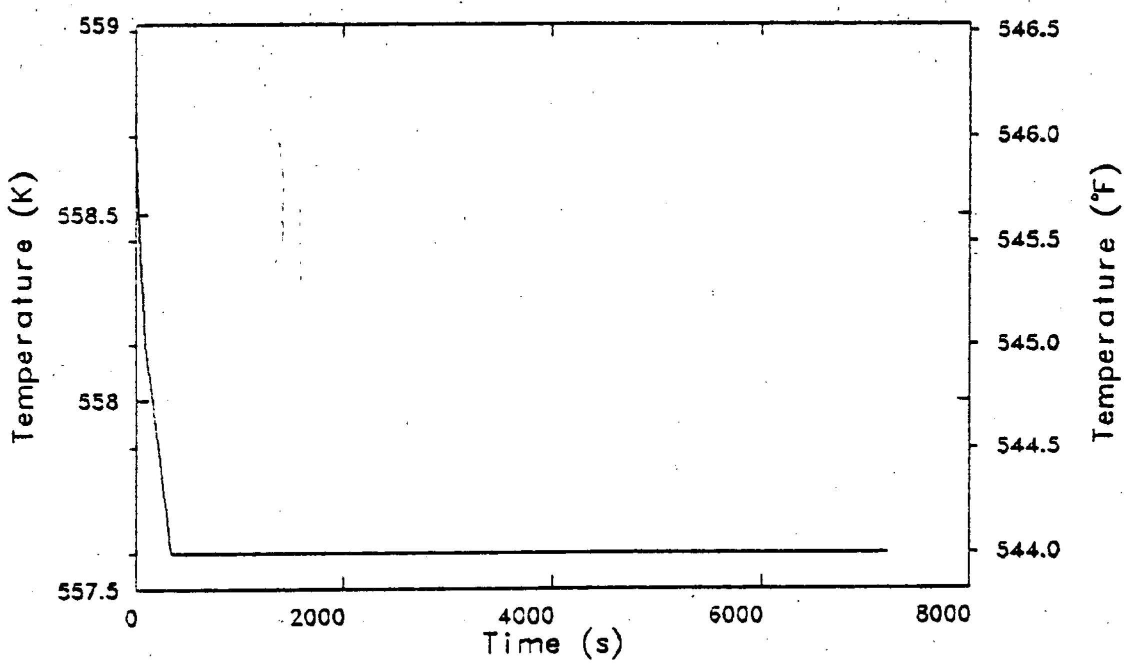

Figure A-88. Sequence 5-12 reactor vessel downcomer fluid temperature.

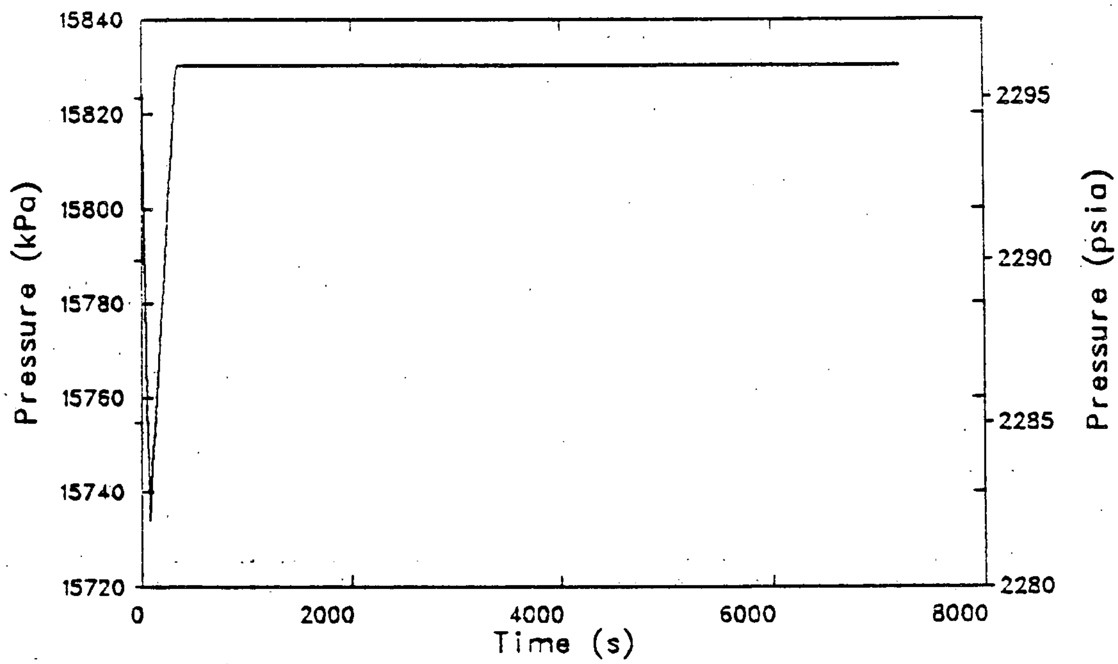

Figure A-89. Sequence 5-13 reactor vessel downcomer fluid pressure.

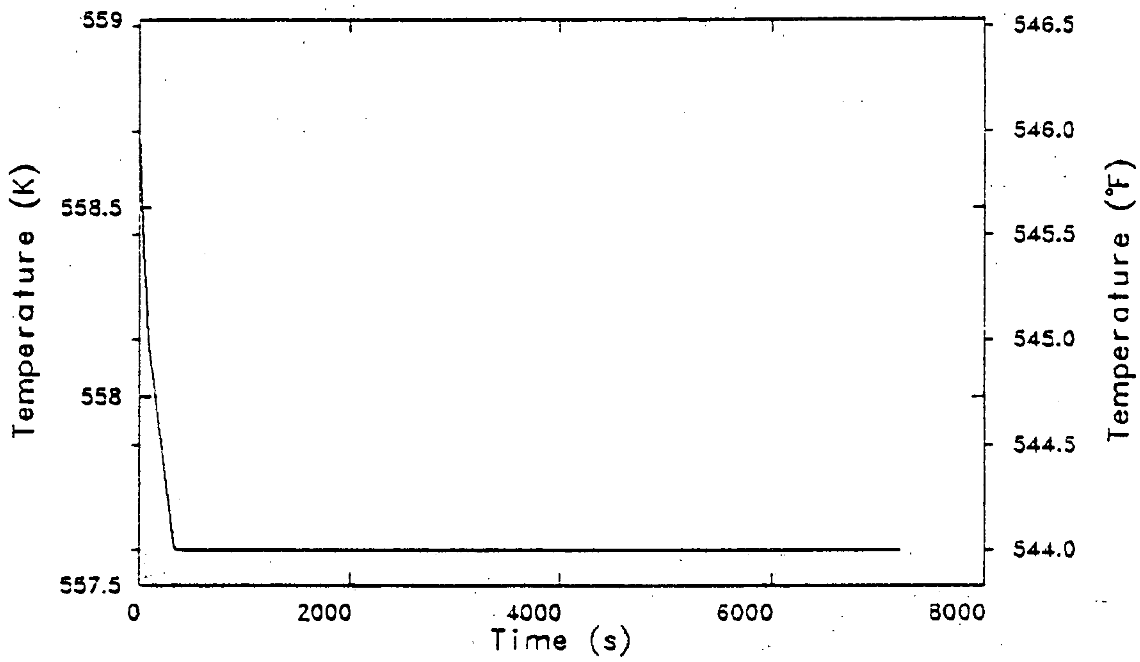

Figure A-90. Sequence 5-13 reactor vessel downcomer fluid temperature.

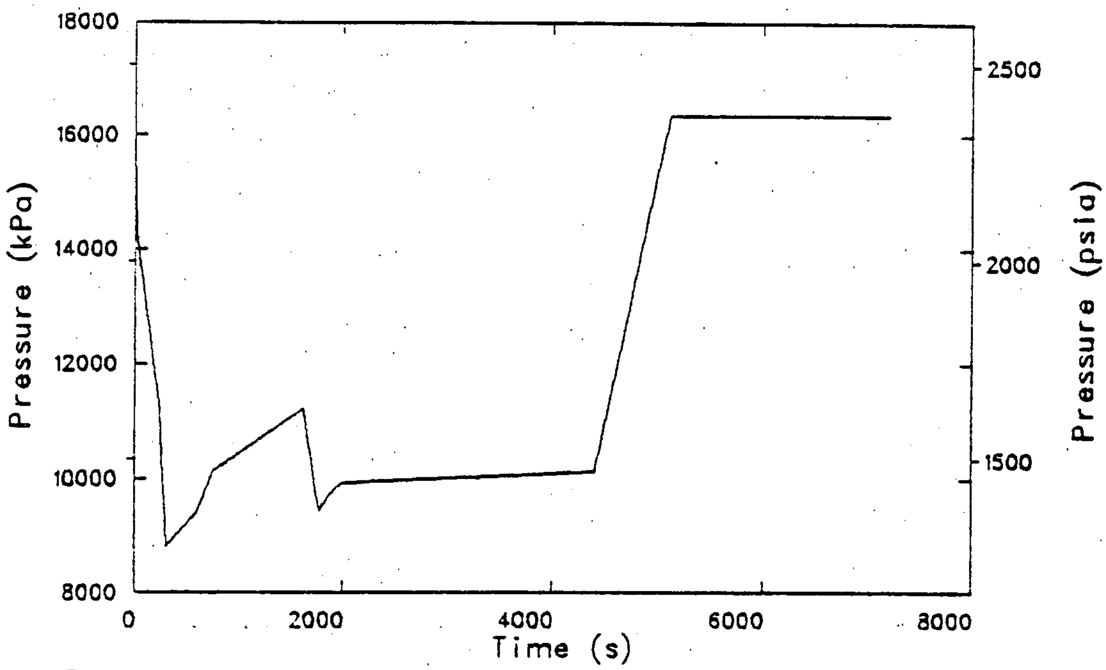

Figure A-91. Sequence 5-14 reactor vessel downcomer fluid pressure.

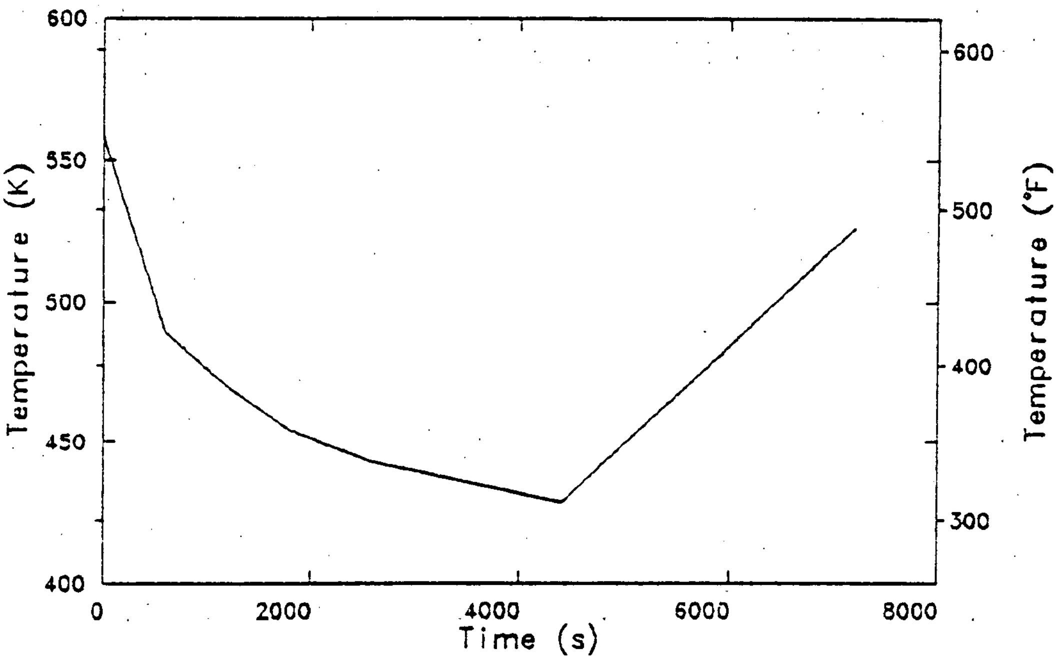

Figure A-92. Sequence 5-14 reactor vessel downcomer fluid temperature.

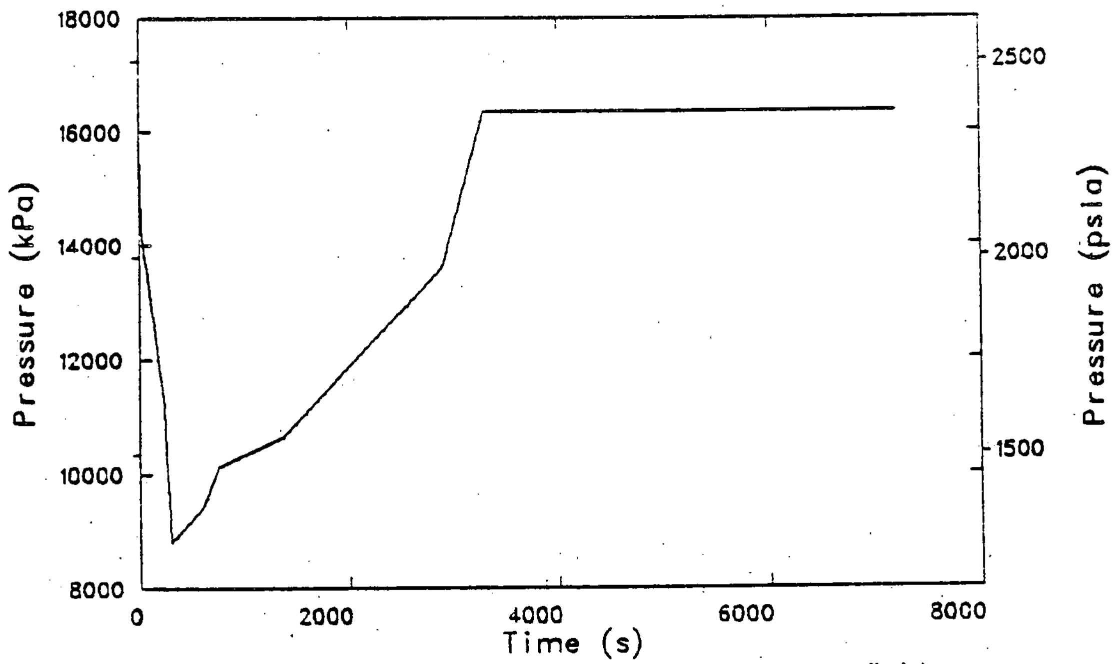

Figure A-93. Sequence 5-15 reactor vessel downcomer fluid pressure.

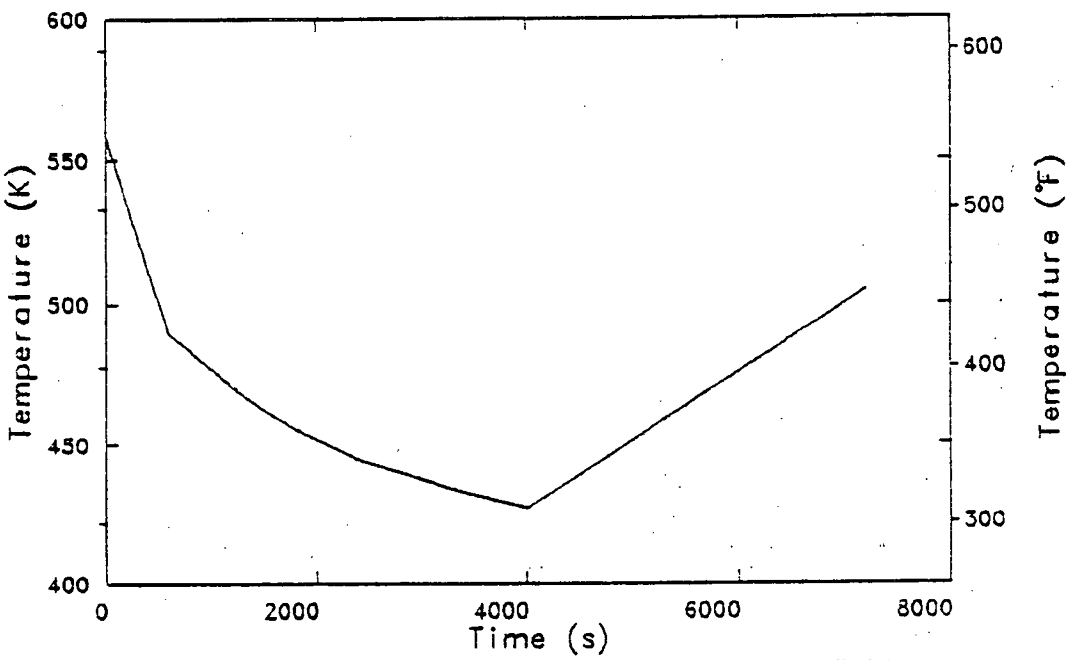

Figure A-94. Sequence 5-15 reactor vessel downcomer fluid temperature.

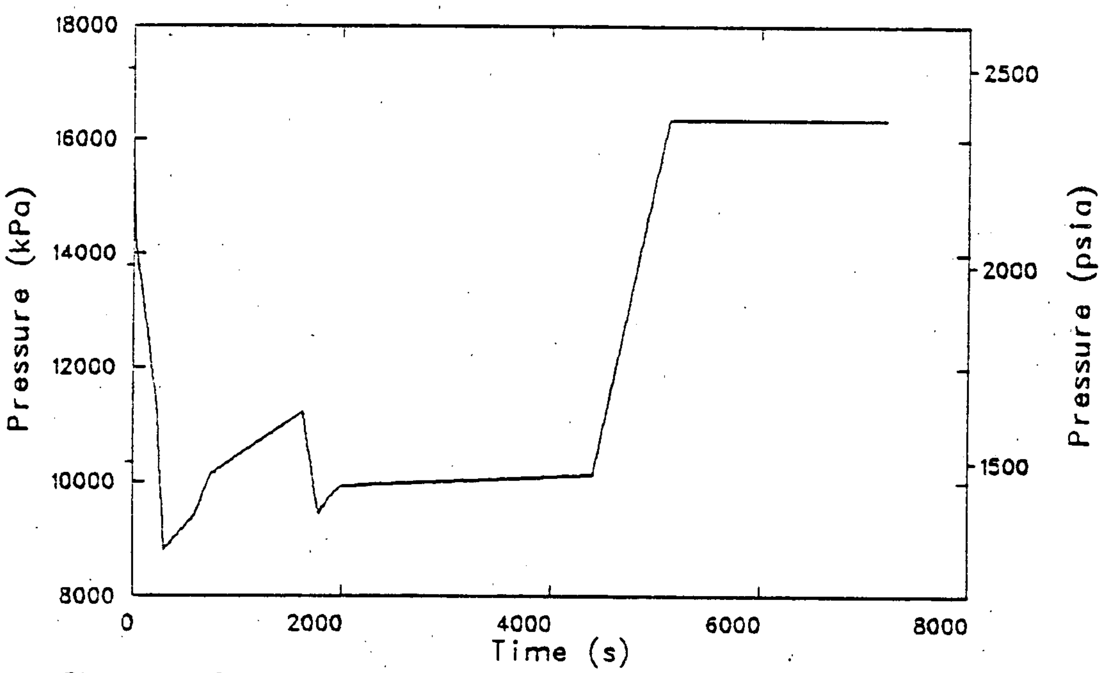

Figure A-95. Sequence 5-16 reactor vessel downcomer fluid pressure.

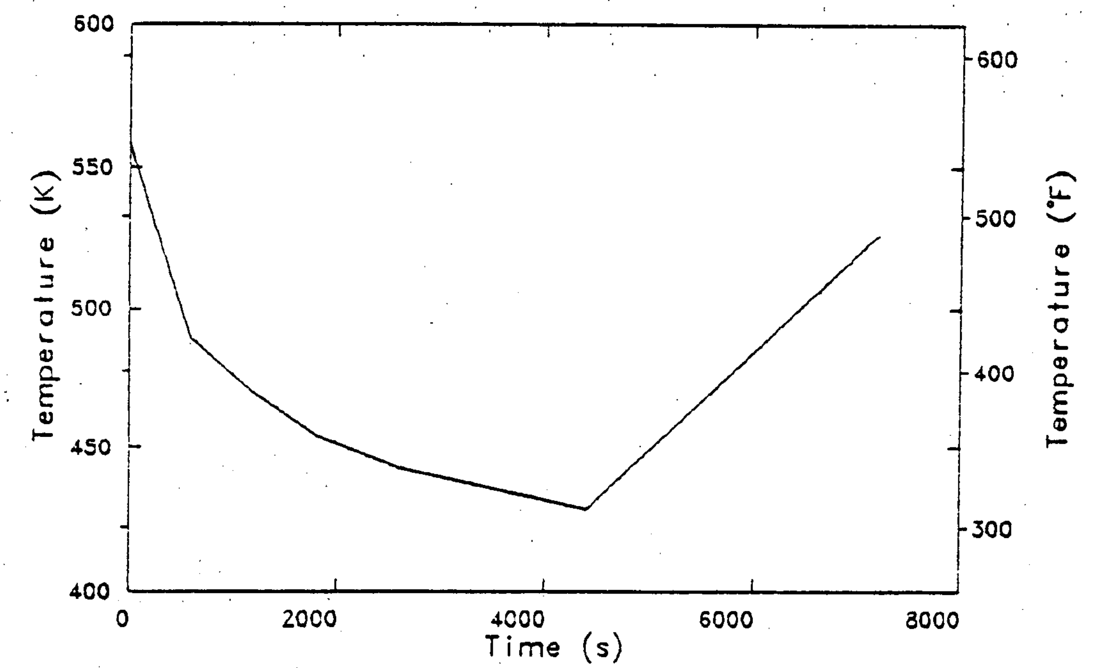

Figure A-96. Sequence 5-16 reactor vessel downcomer fluid temperature.

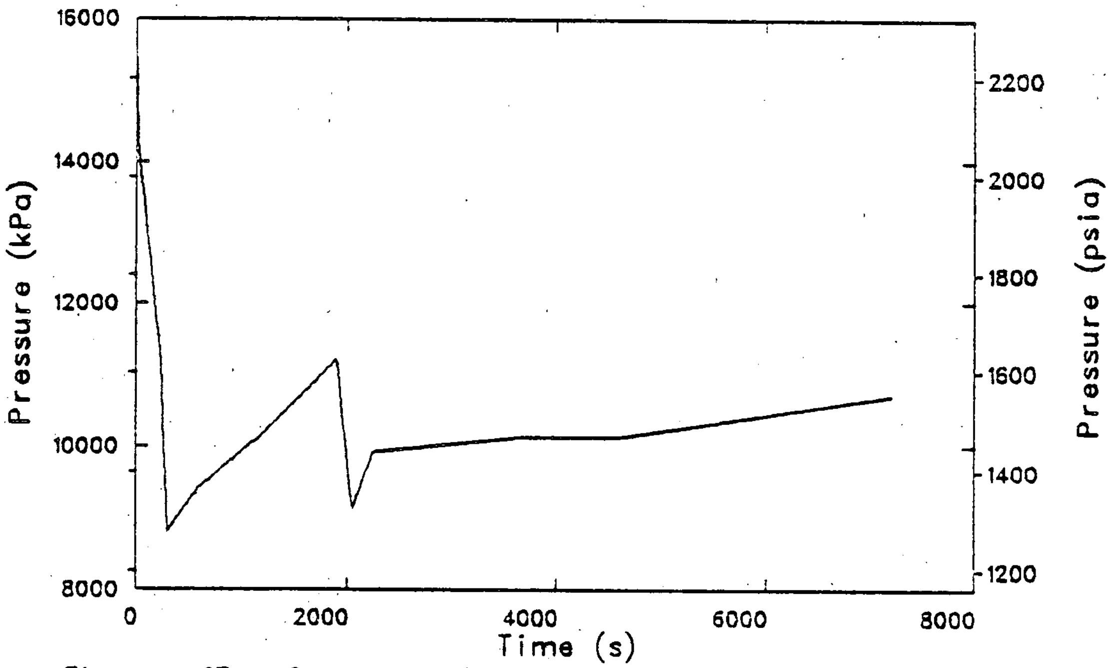

Figure A-97. Sequence 5-17 reactor vessel downcomer fluid pressure.

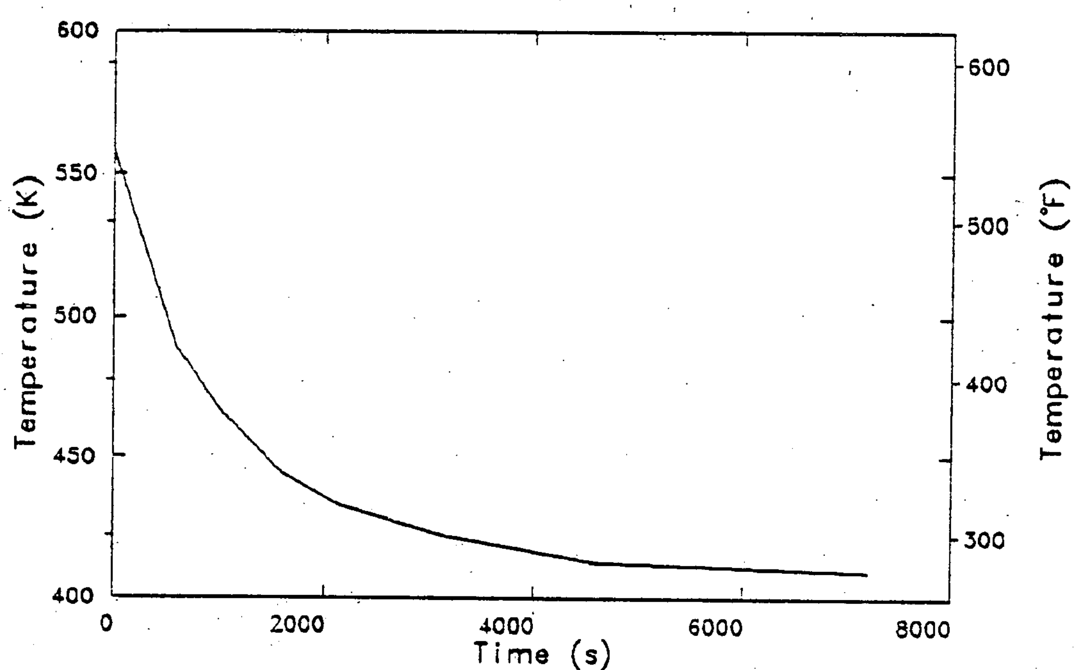

Figure A-98. Sequence 5-17 reactor vessel downcomer fluid temperature.

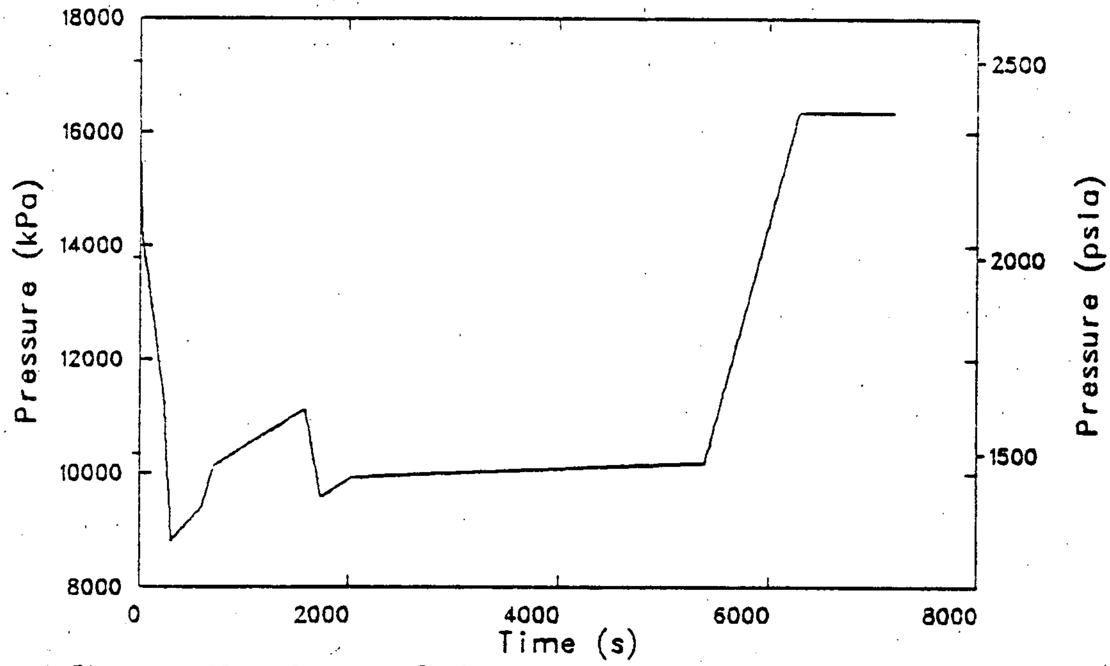

Figure A-99. Sequence 5-18 reactor vessel downcomer fluid pressure.

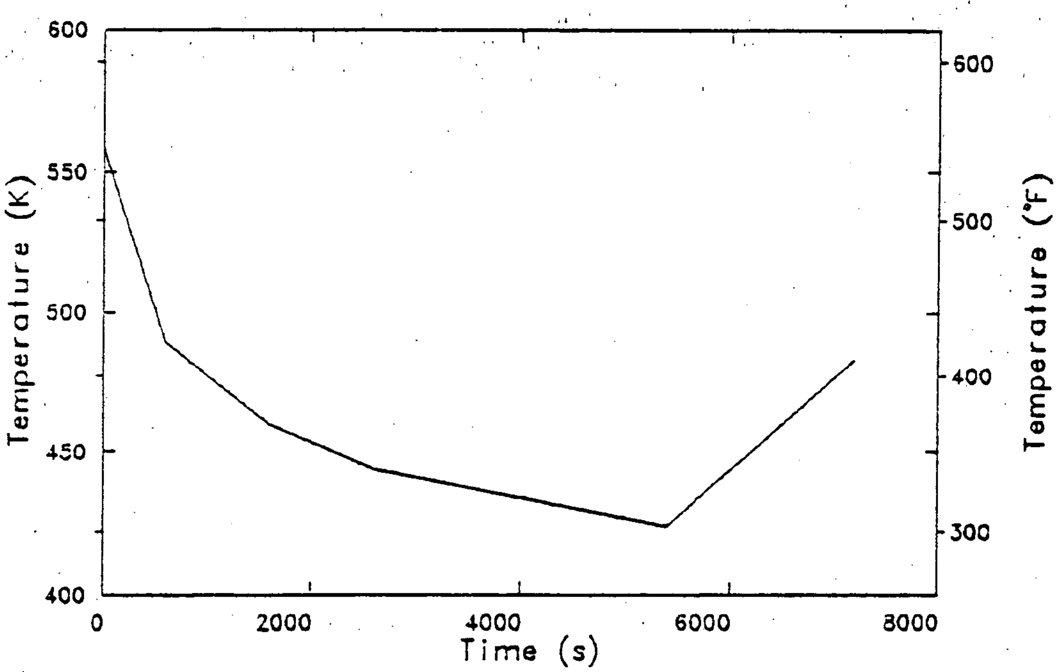

Figure A-100. Sequence 5-18 reactor vessel downcomer fluid temperature.

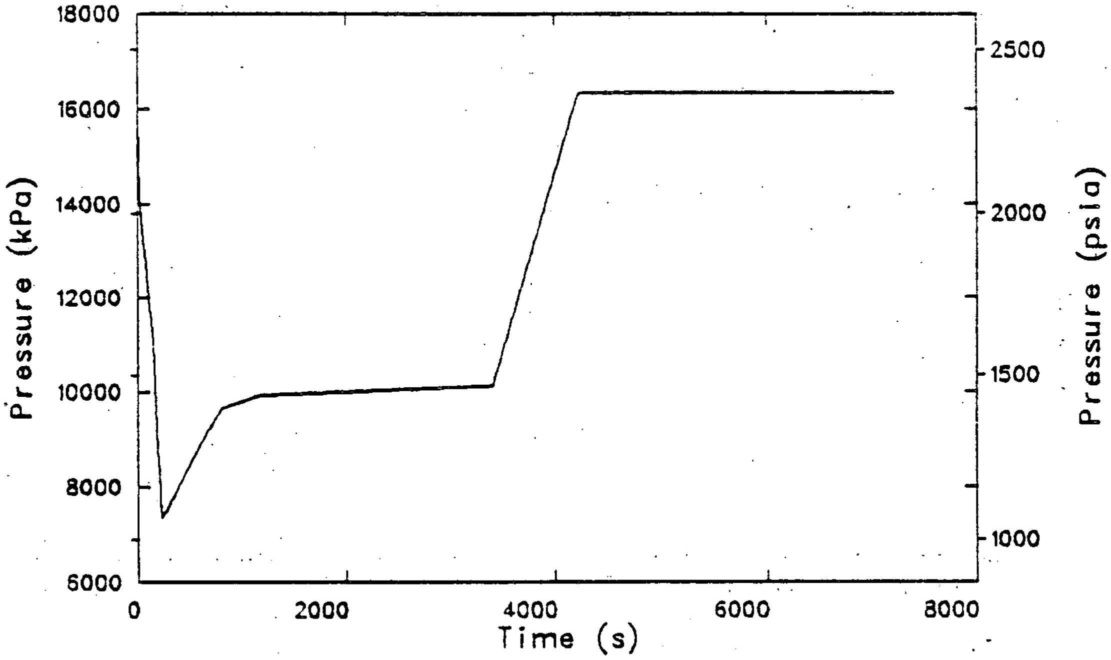

Figure A-101. Sequence 5-19 reactor vessel downcomer fluid pressure.

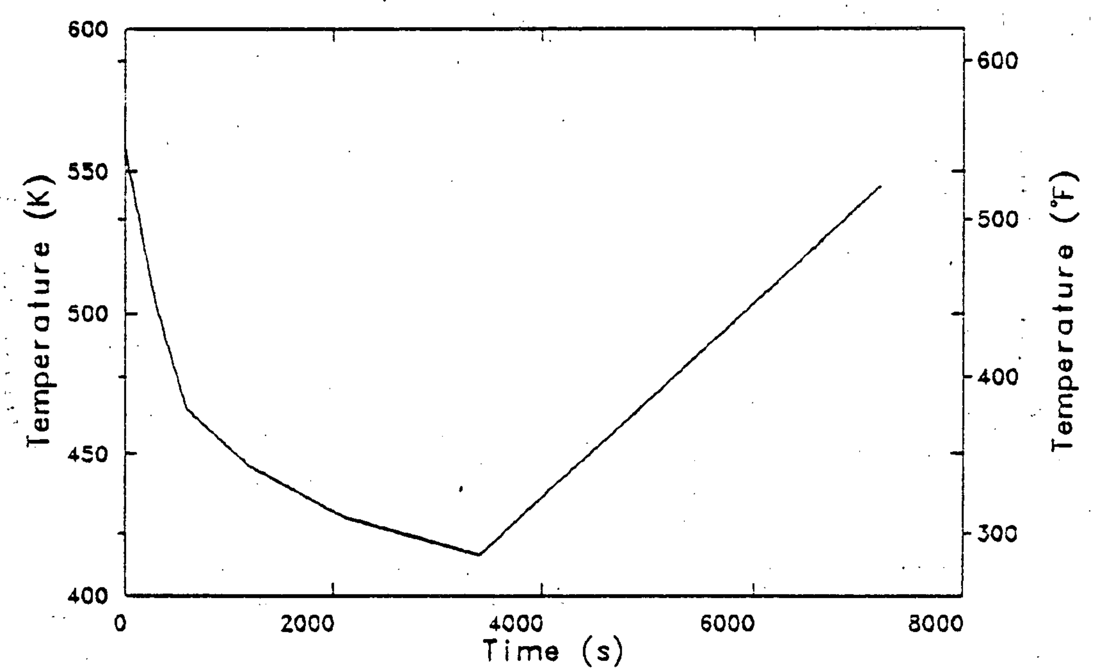

Figure A-102. Sequence 5-19 reactor vessel downcomer fluid temperature.

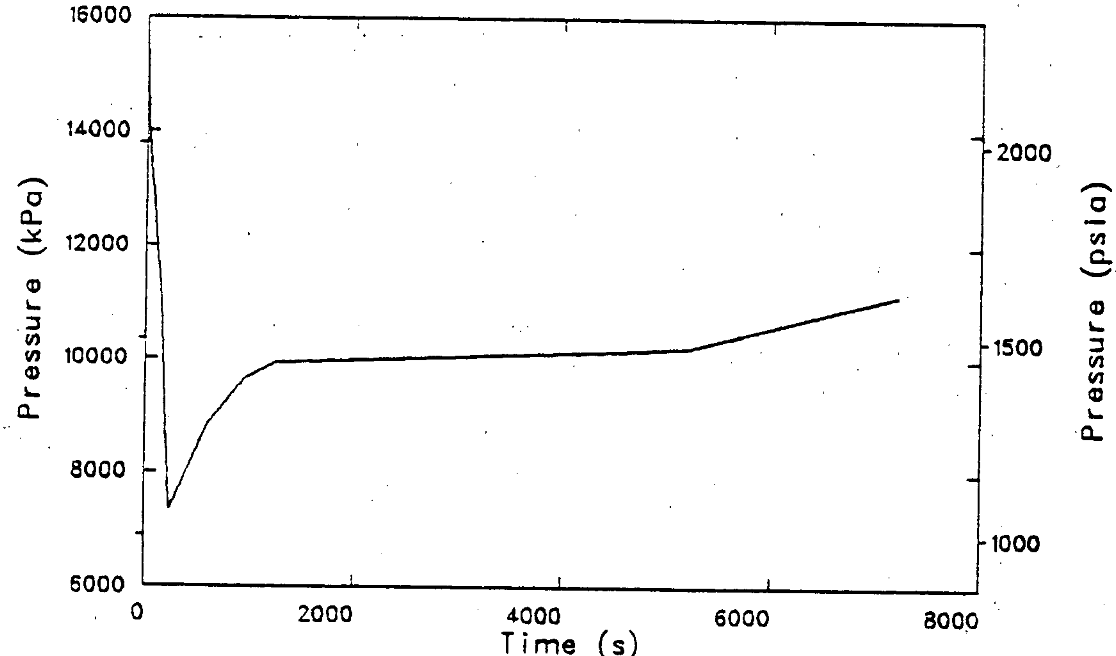

Figure A-103. Sequence 5-20 reactor vessel downcomer fluid pressure.

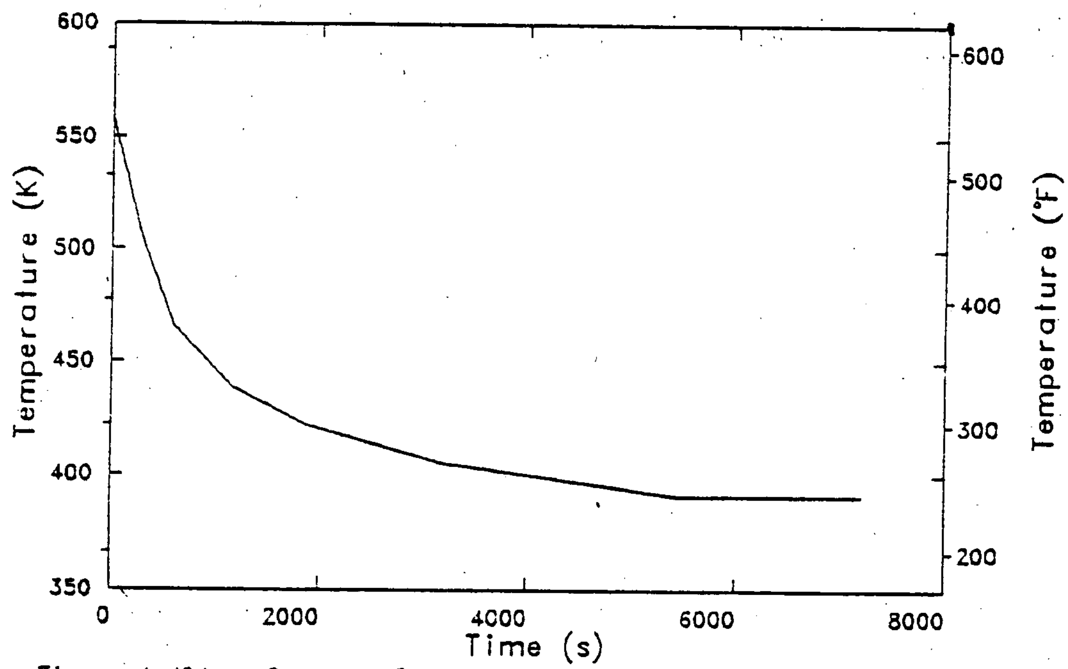

Figure A-104. Sequence 5-20 reactor vessel downcomer fluid temperature.

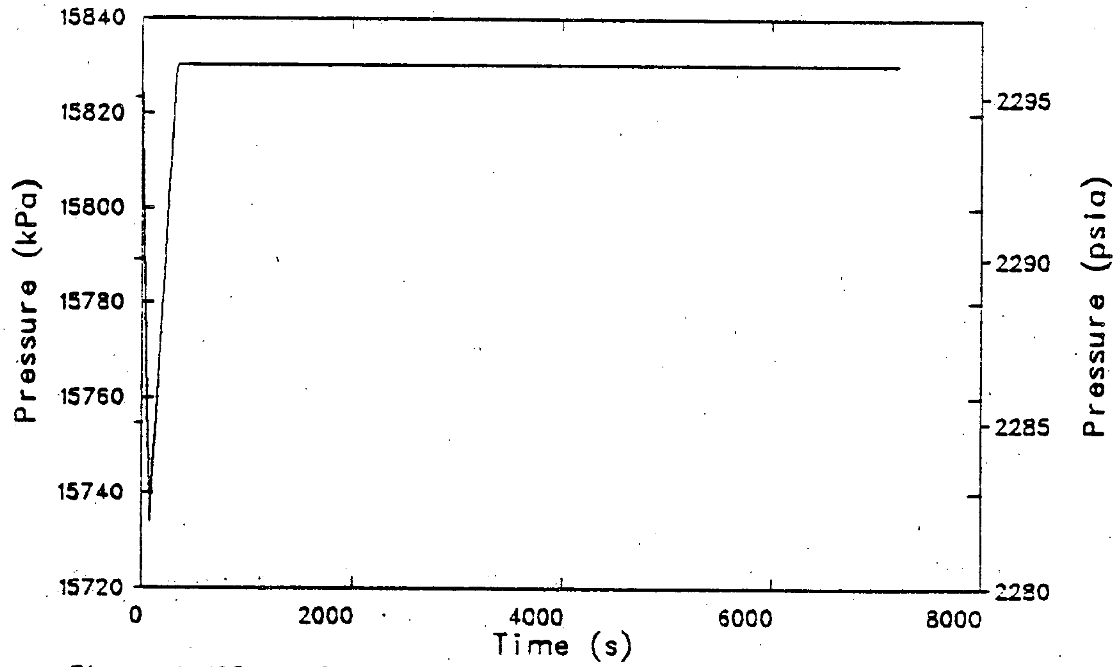

Figure A-105. Sequence 5-21 reactor vessel downcomer fluid pressure.

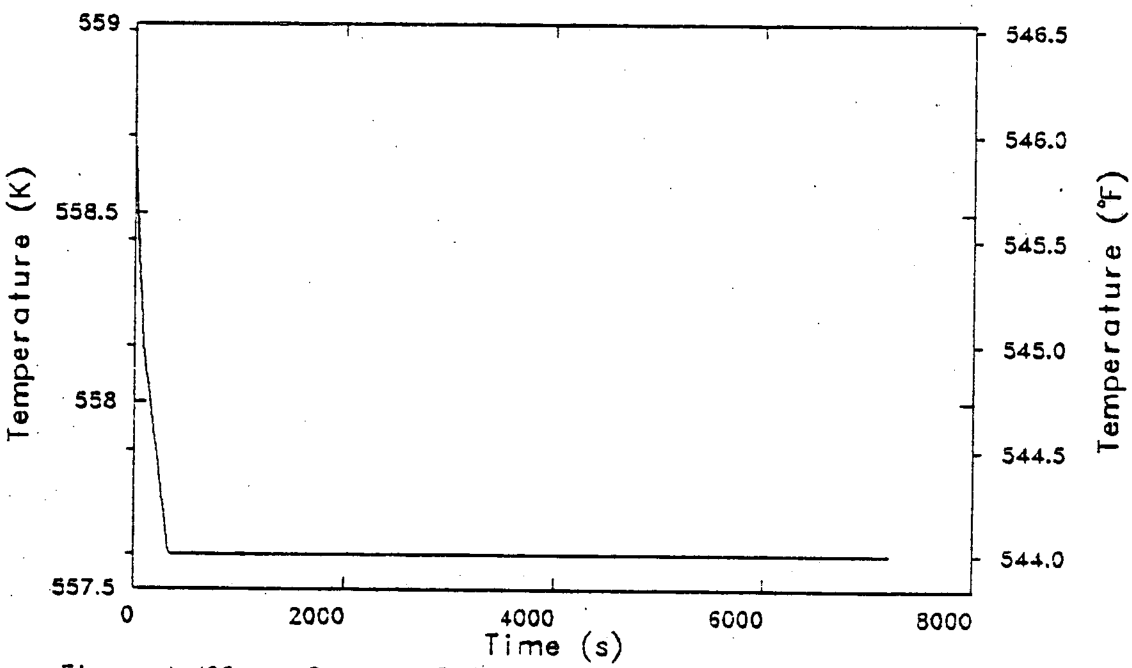

Figure A-106. Sequence 5-21 reactor vessel downcomer fluid temperature.

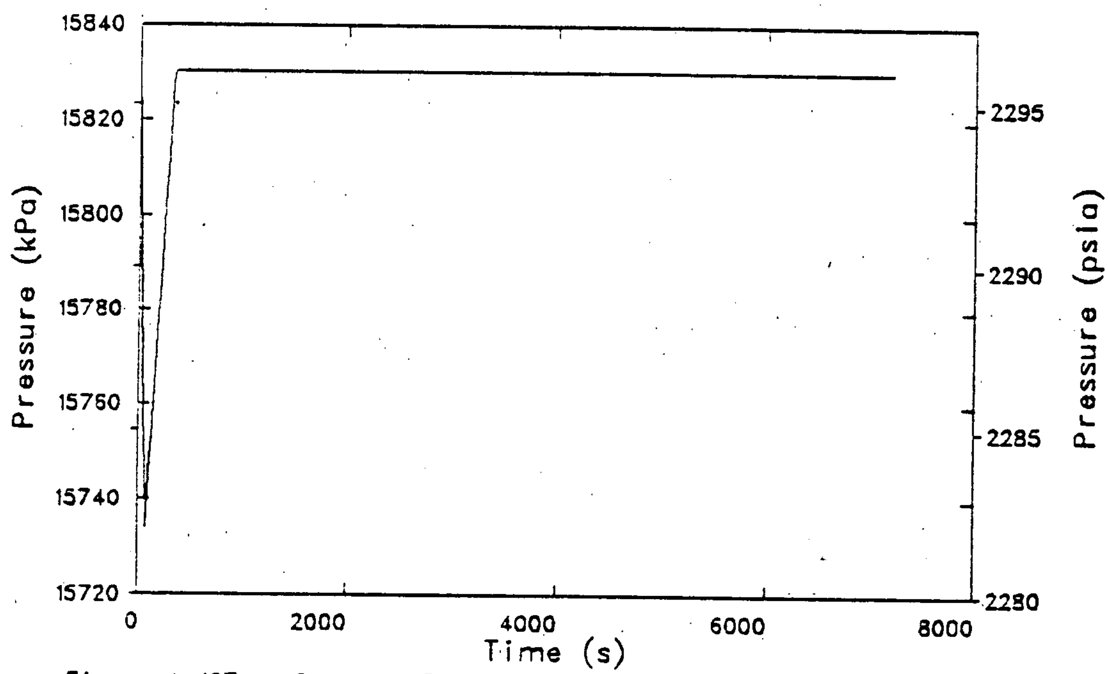

Figure A-107. Sequence 5-22 reactor vessel downcomer fluid pressure.

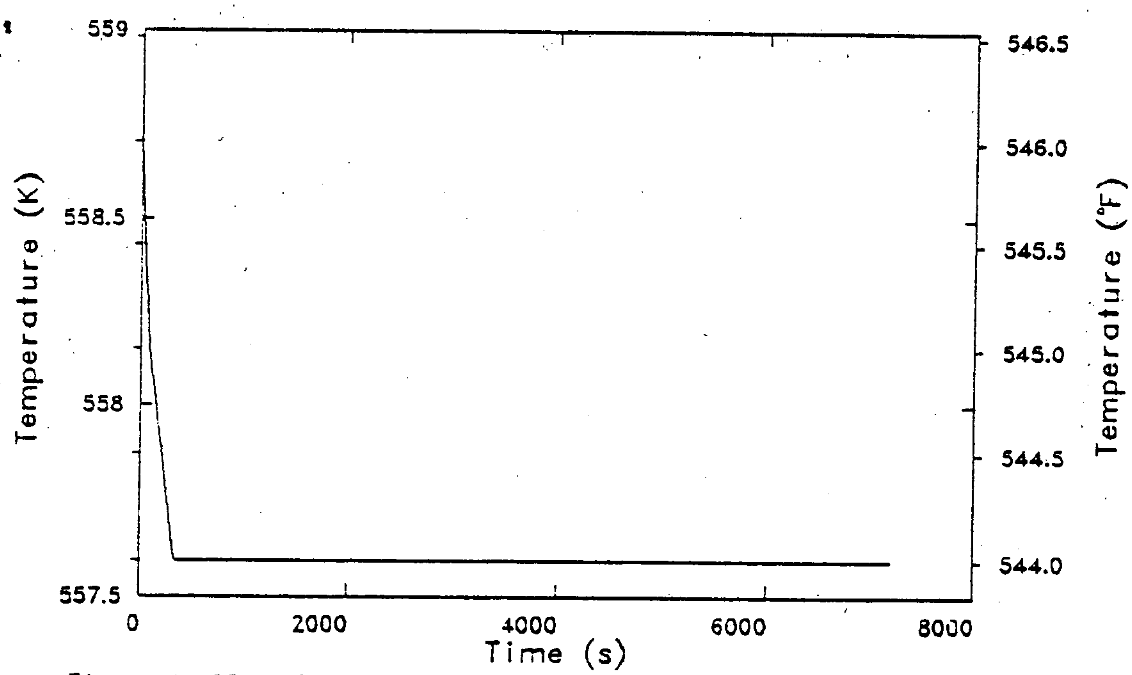

Figure A-108. Sequence 5-22 reactor vessel downcomer fluid temperature.

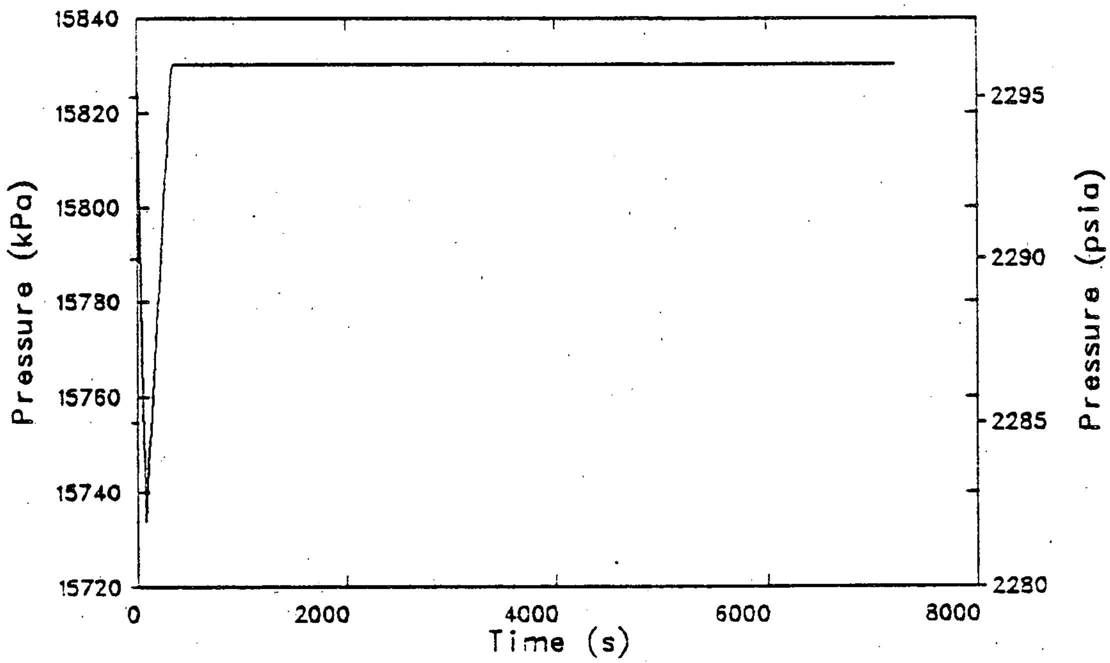

Figure A-109. Sequence 5-23 reactor vessel downcomer fluid pressure.

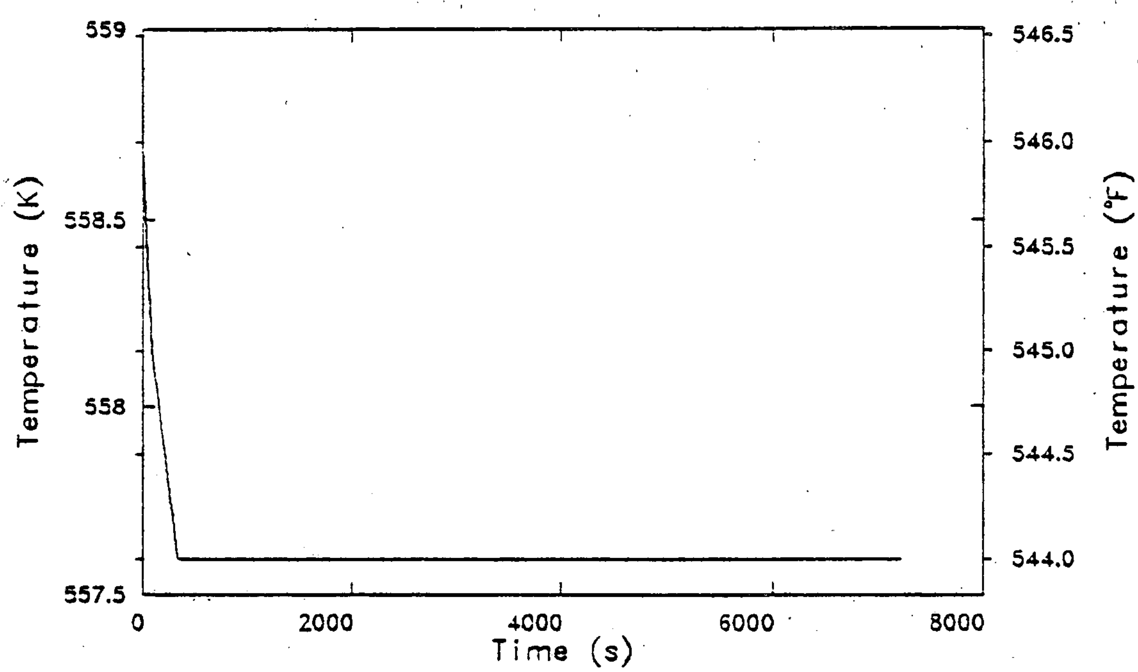

Figure A-110. Sequence 5-23 reactor vessel downcomer fluid temperature.

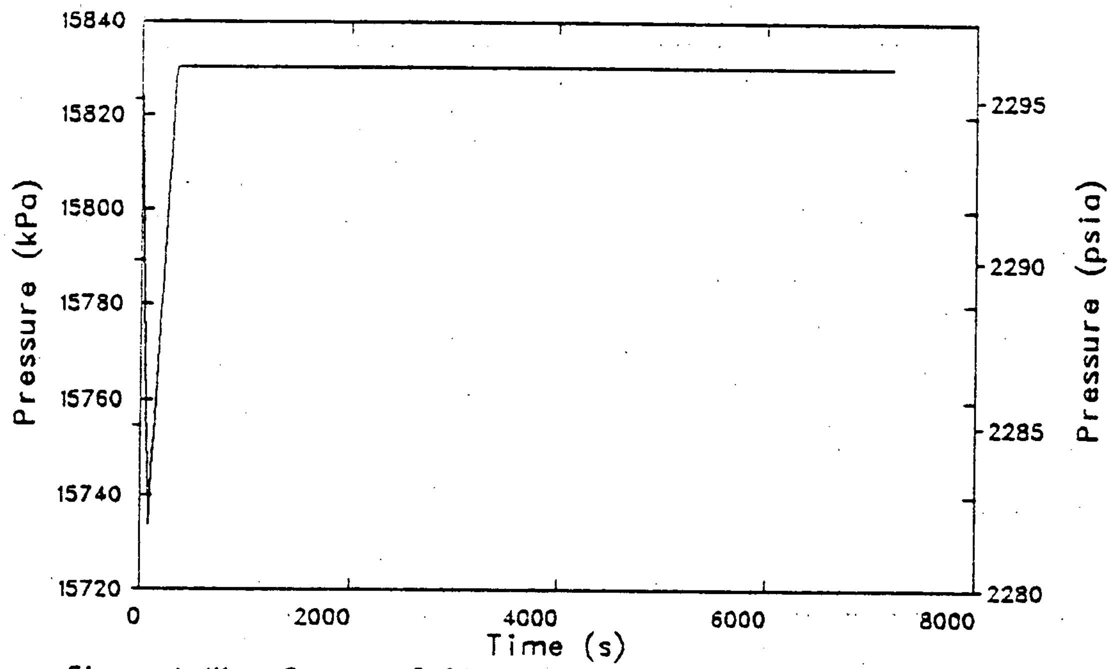

Figure A-111. Sequence 5-24 reactor vessel downcomer fluid pressure.

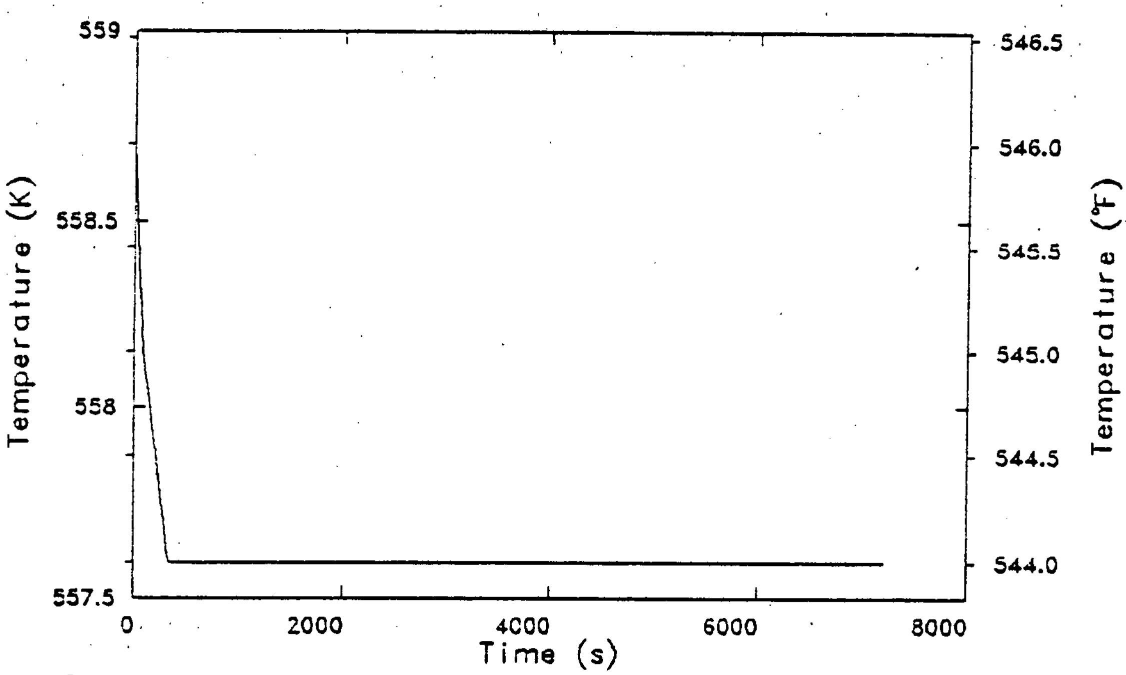

Figure A-112. Sequence 5-24 reactor vessel downcomer fluid temperature.

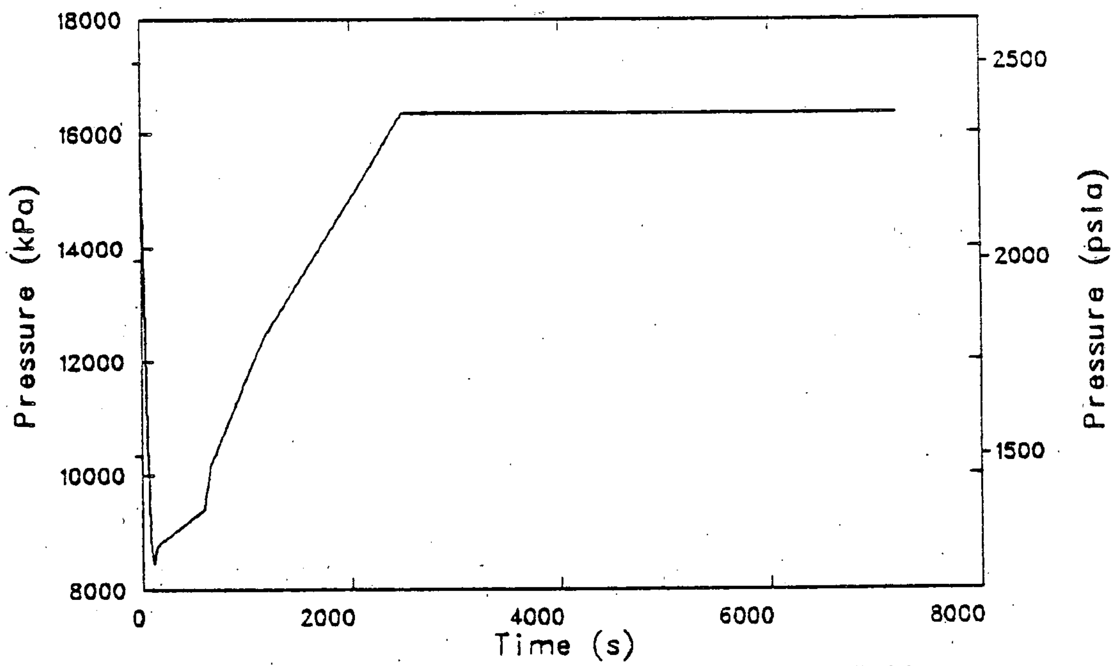

Figure A-113. Sequence 6-1 reactor vessel downcomer fluid pressure.

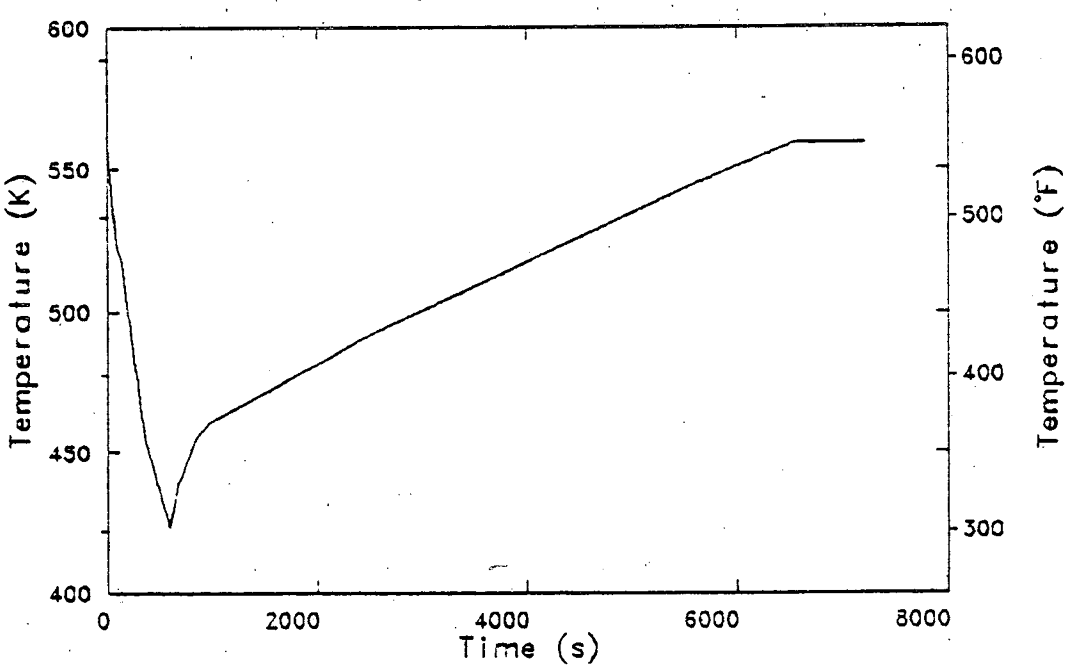

Figure A-114. Sequence 6-1 reactor vessel downcomer fluid temperature.

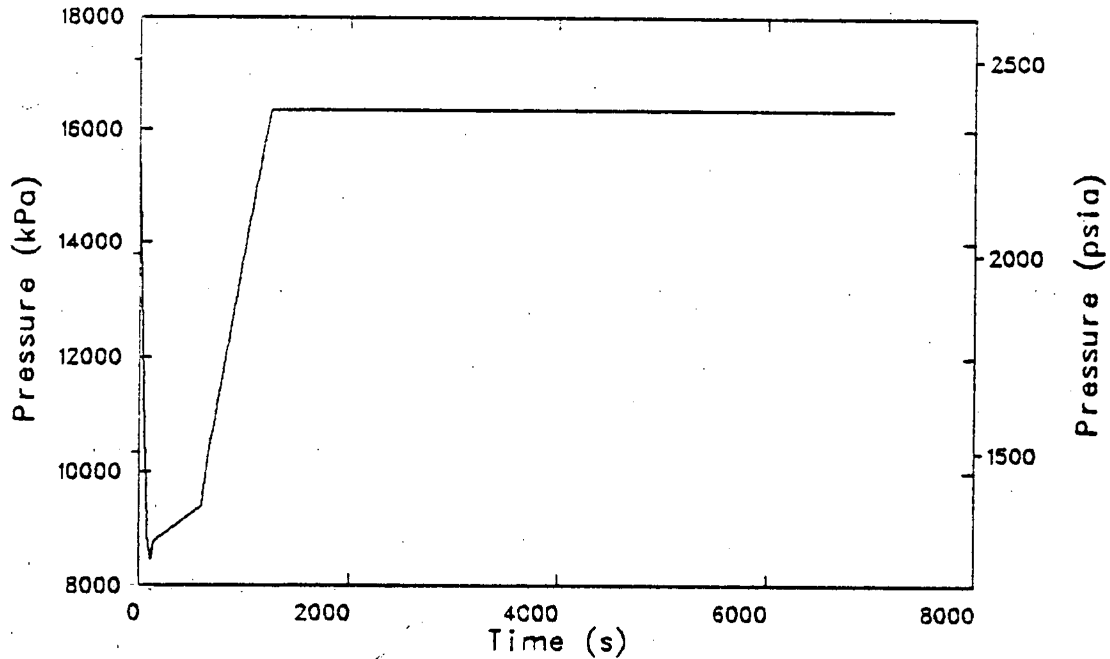

Figure A-115. Sequence 6-2 reactor vessel downcomer fluid pressure.

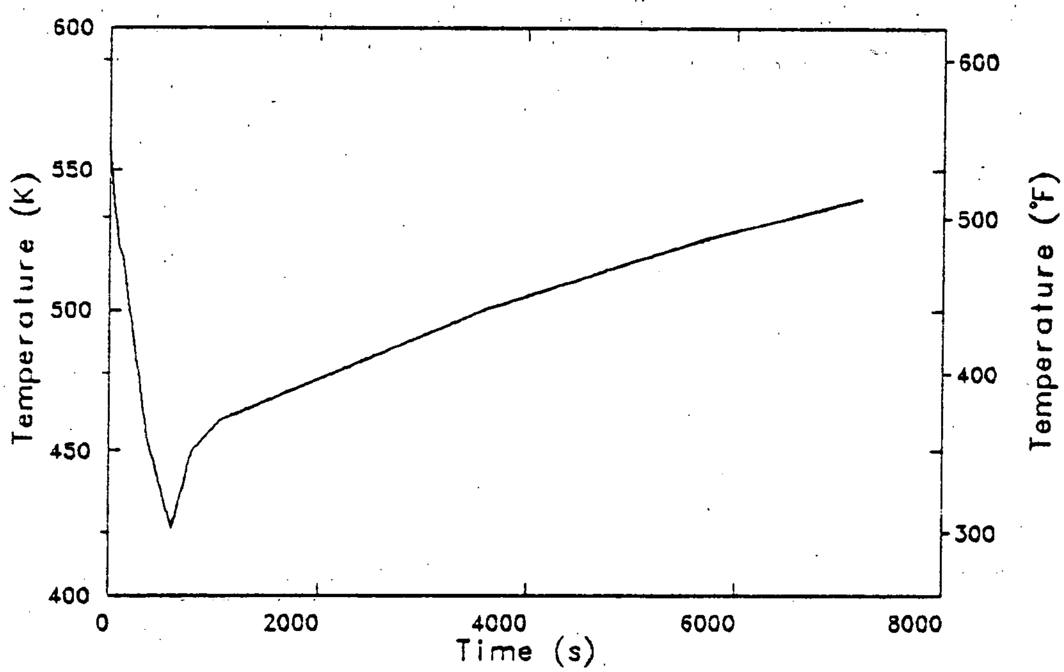

Figure A-116. Sequence 6-2 reactor vessel downcomer fluid temperature.

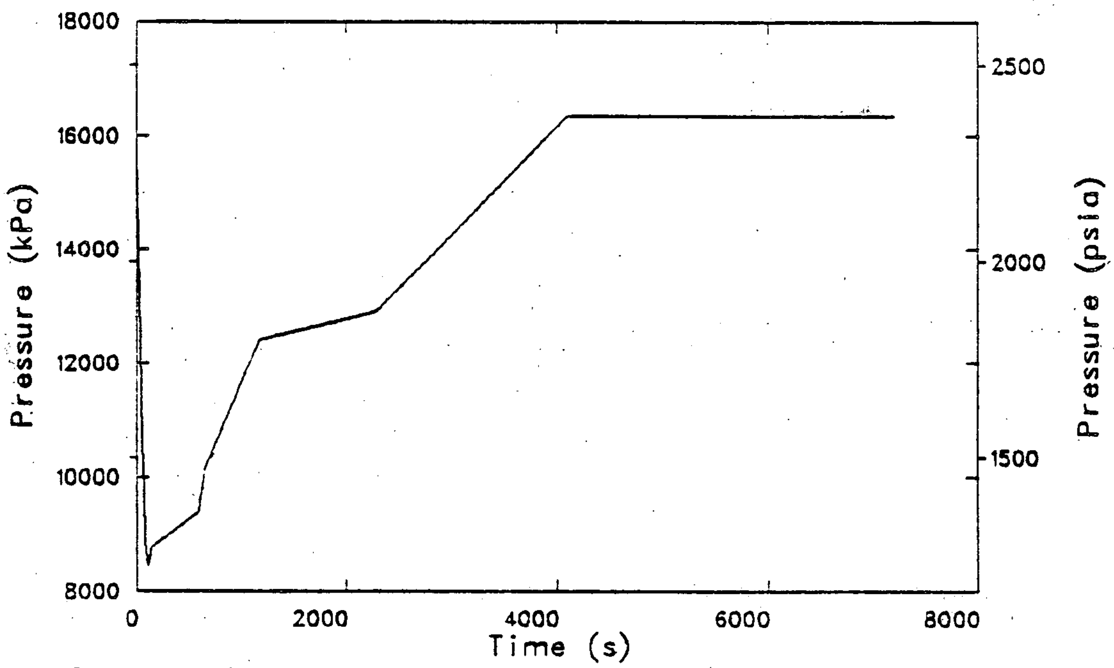

Figure A-117. Sequence 6-3 reactor vessel downcomer fluid pressure.

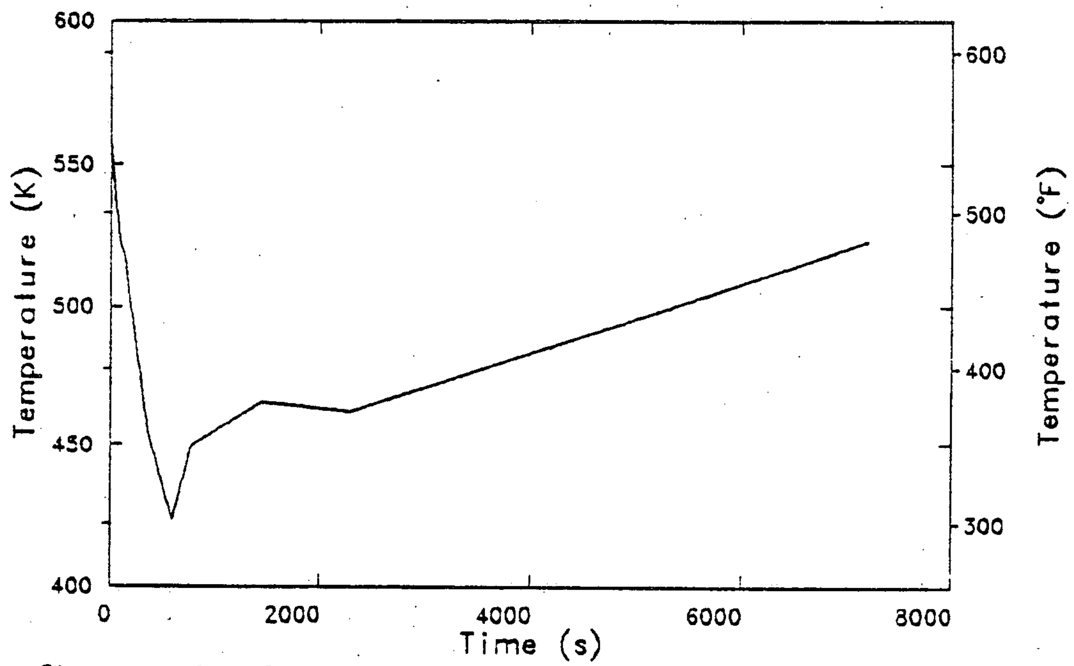

Figure A-118. Sequence 6-3 reactor vessel downcomer fluid temperature.

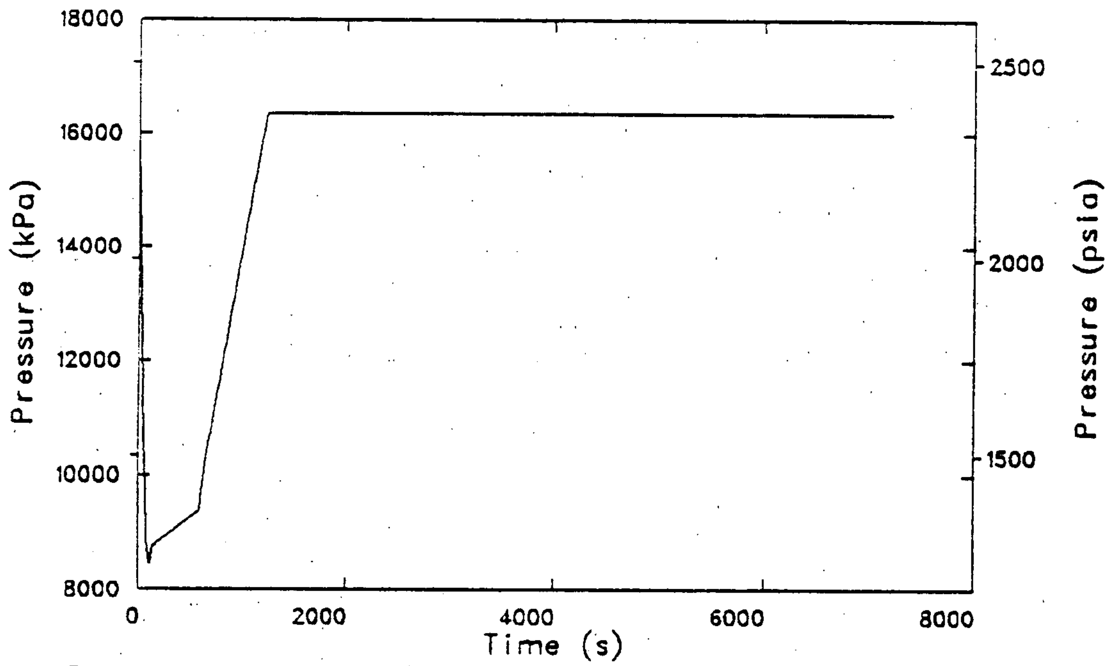

Figure A-119. Sequence 6-4 reactor vessel downcomer fluid pressure.

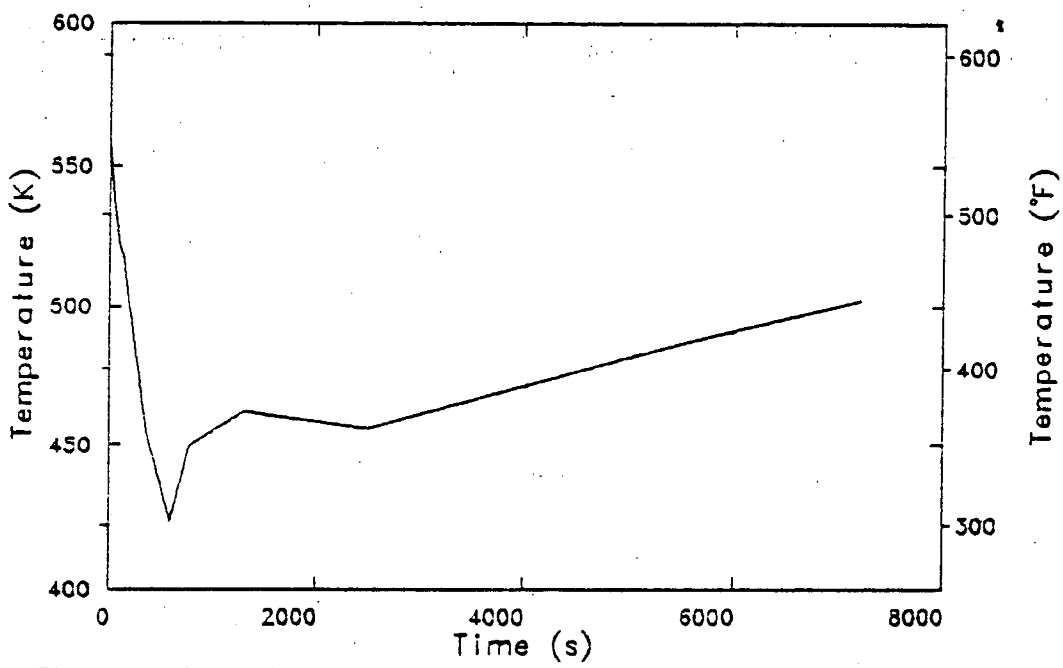

Figure A-120. Sequence 6-4 reactor vessel downcomer fluid temperature.

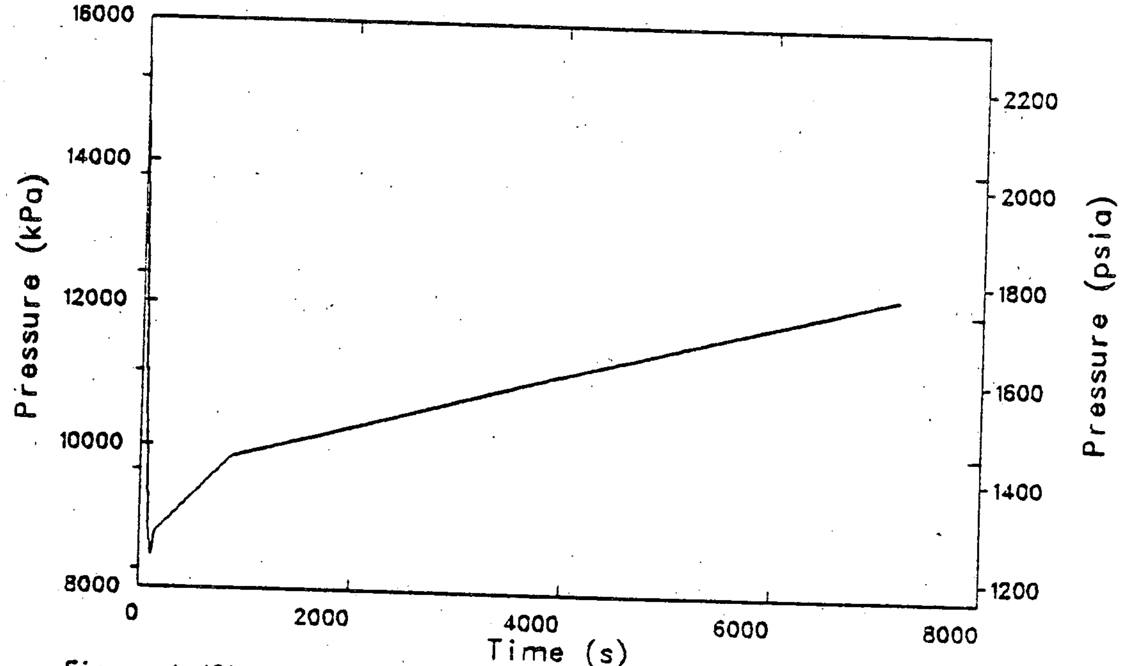

Figure A-121. Sequence 6-5 reactor vessel downcomer fluid pressure.

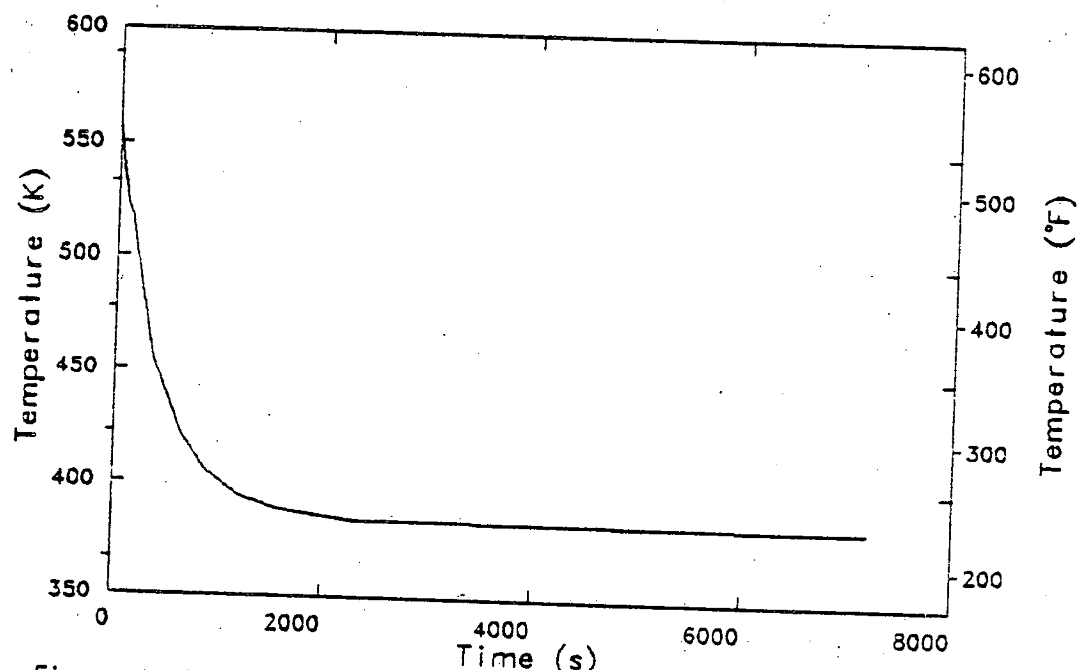

Figure A-122. Sequence 6-5 reactor vessel downcomer fluid temperature.

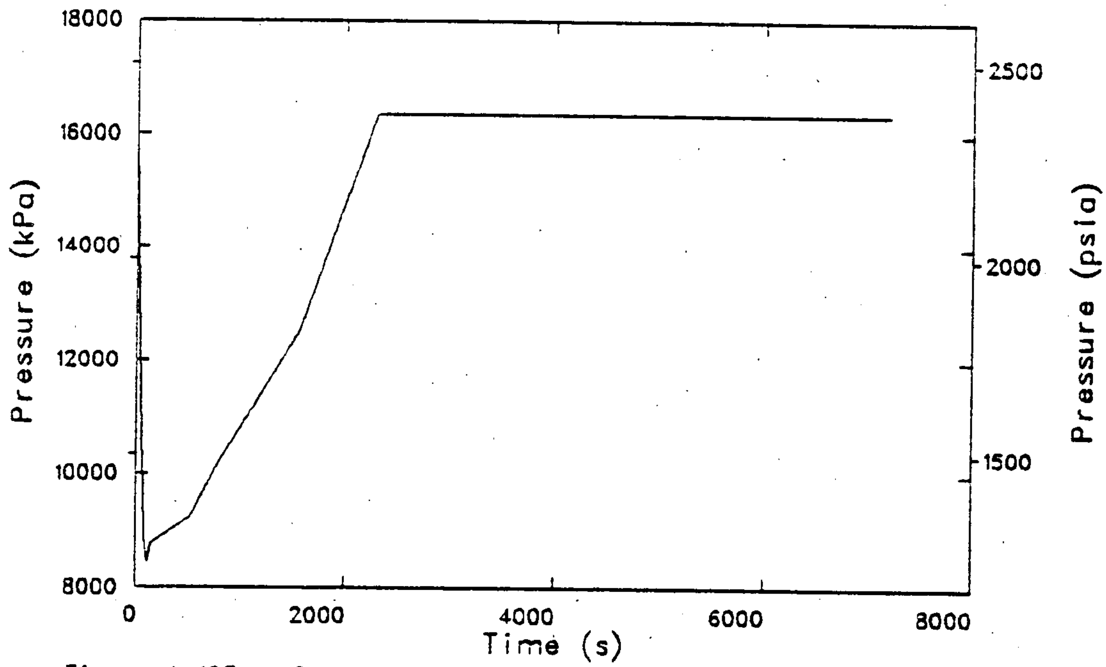

Figure A-123. Sequence 6-6 reactor vessel downcomer fluid pressure.

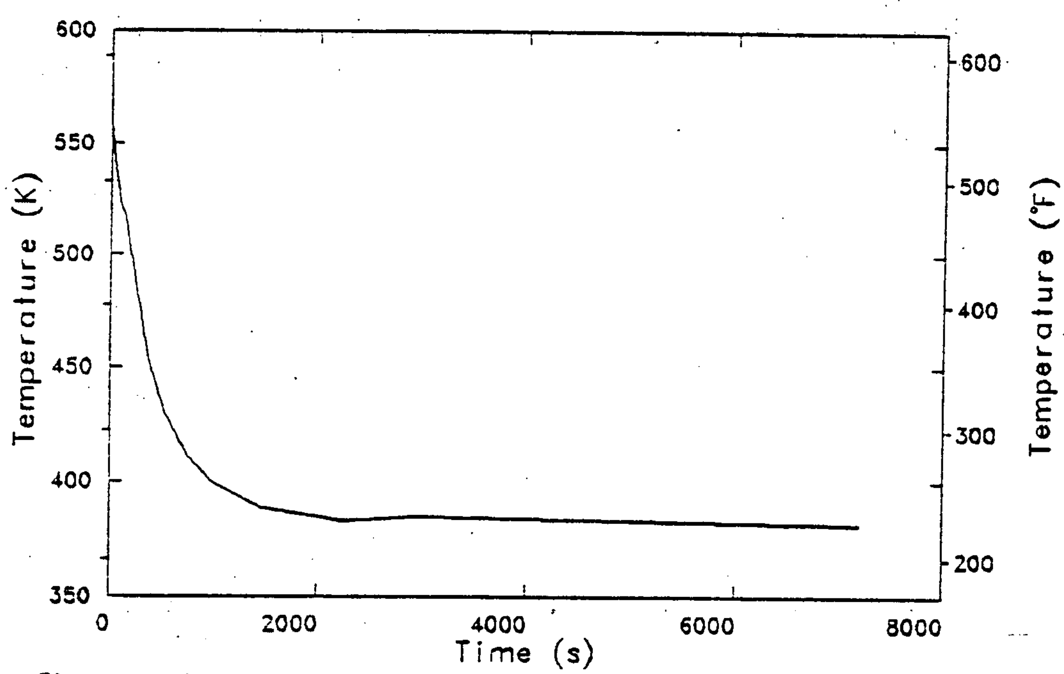

Figure A-124. Sequence 6-6 reactor vessel downcomer fluid temperature.

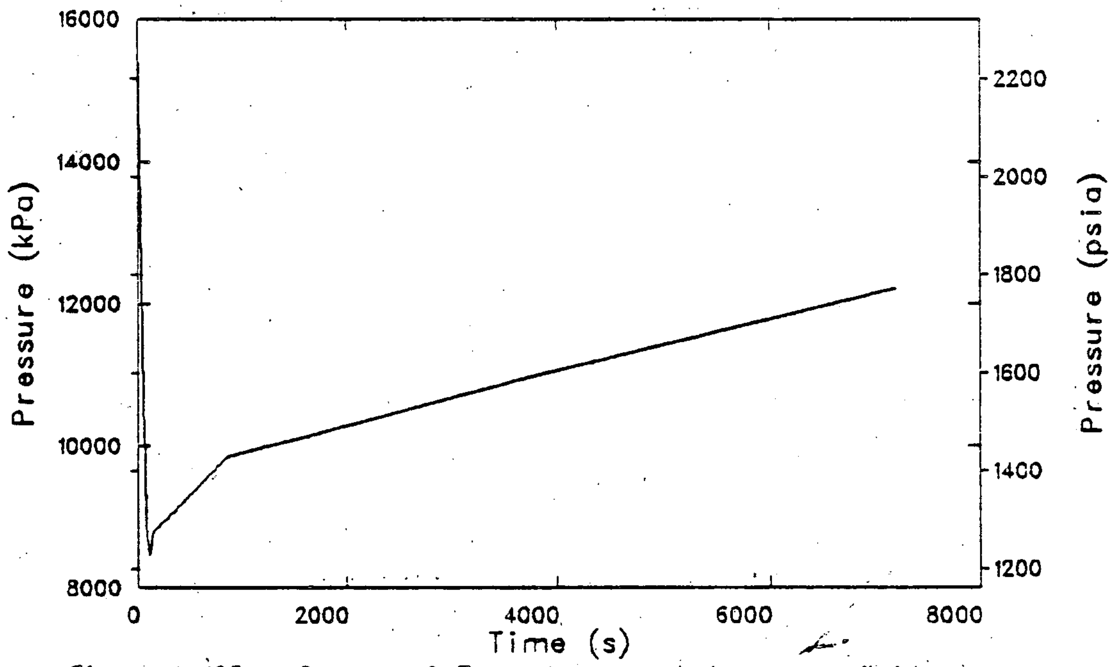

Figure A-125. Sequence 6-7 reactor vessel downcomer fluid pressure.

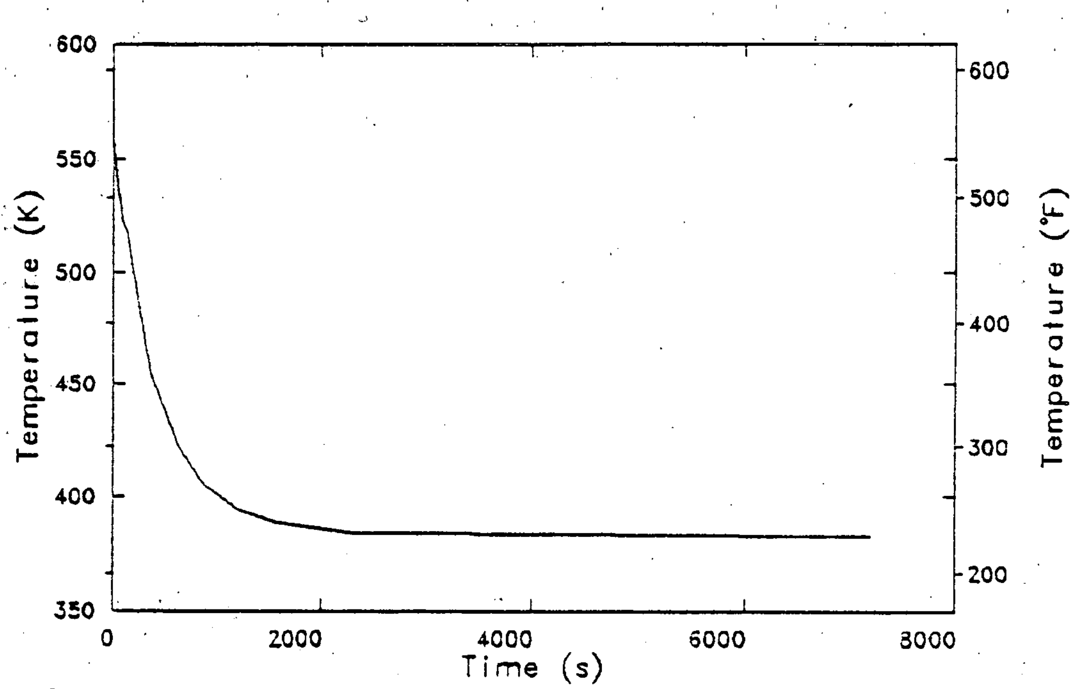

Figure A-126. Sequence 6-7 reactor vessel downcomer fluid temperature.

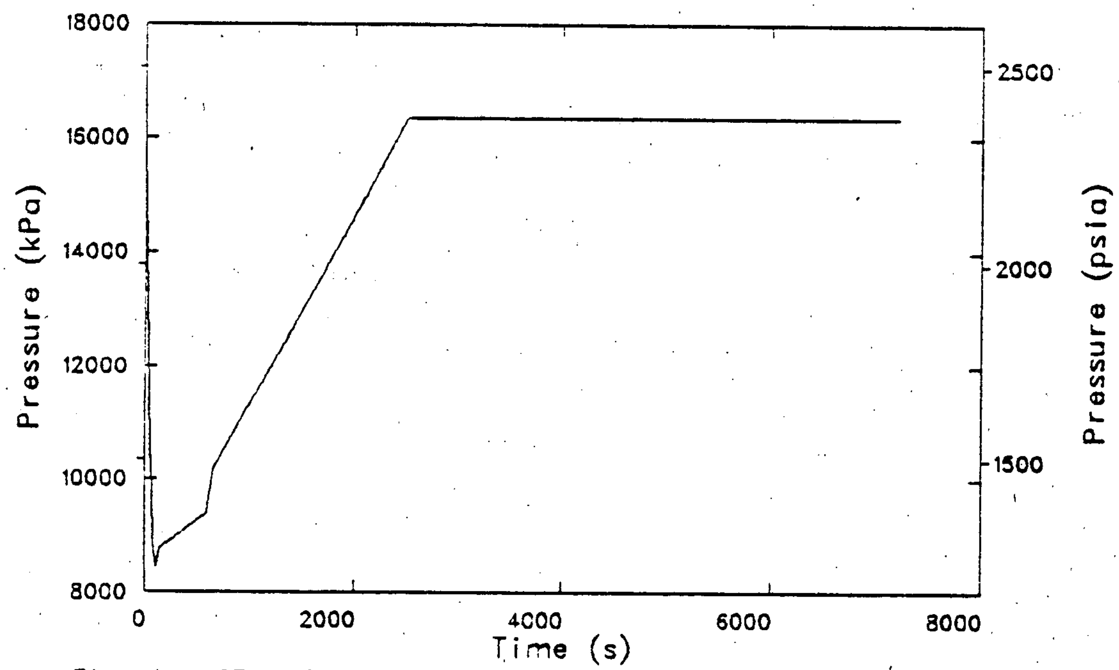

Figure A-127. Sequence 6-8 reactor vessel downcomer fluid pressure.

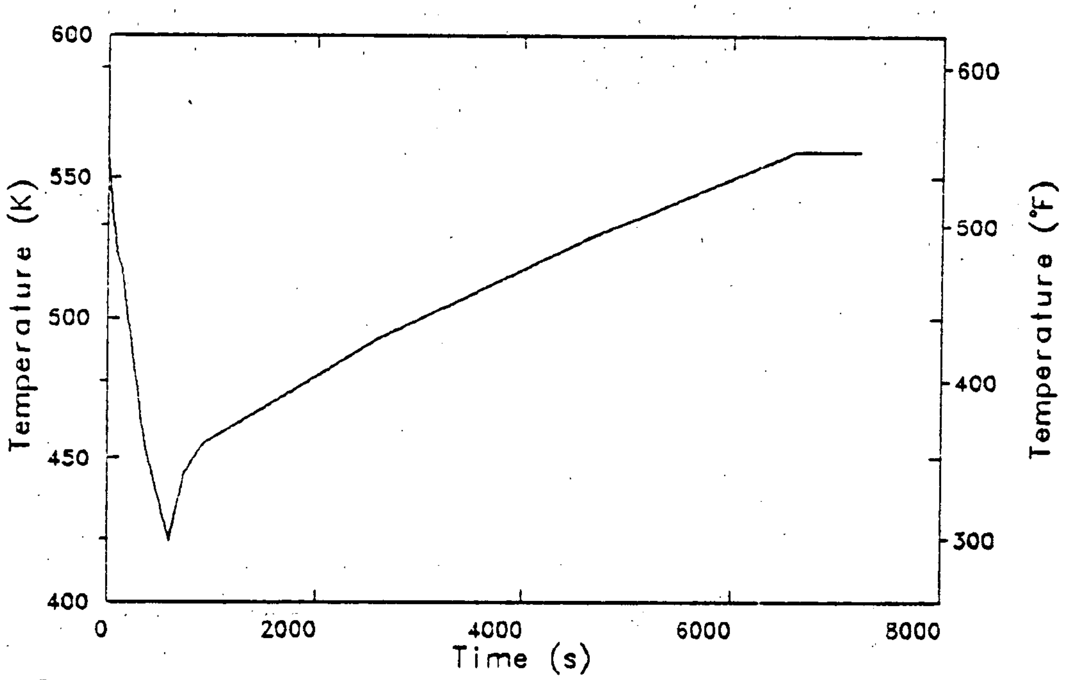

Figure A-128. Sequence 6-8 reactor vessel downcomer fluid temperature.

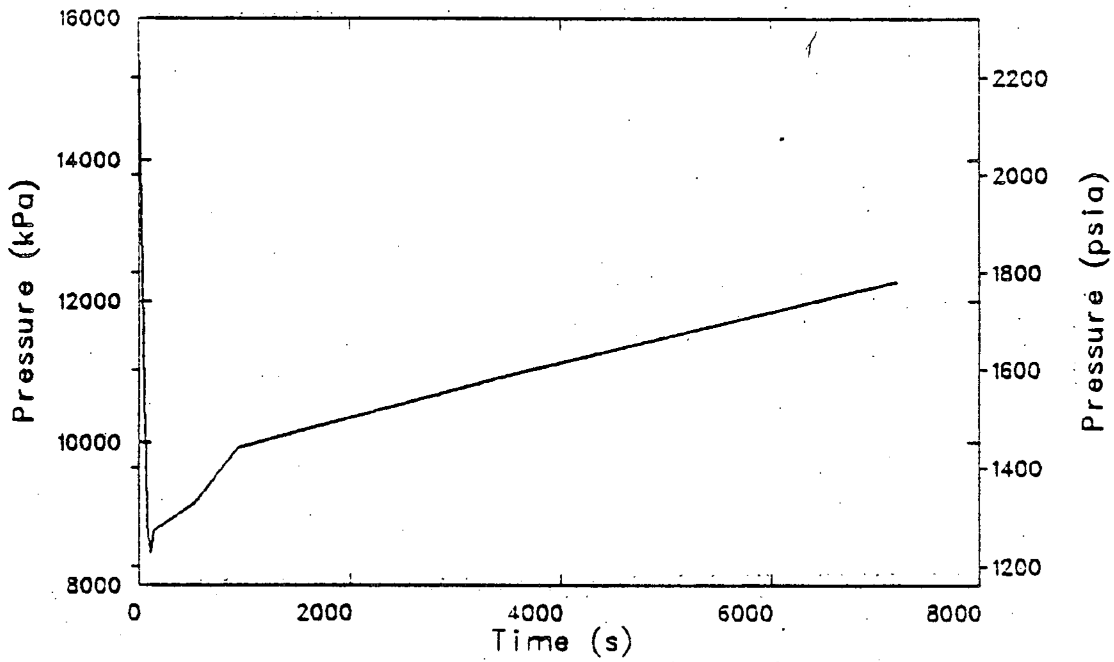

Figure A-129. Sequence 6-9 reactor vessel downcomer fluid pressure.

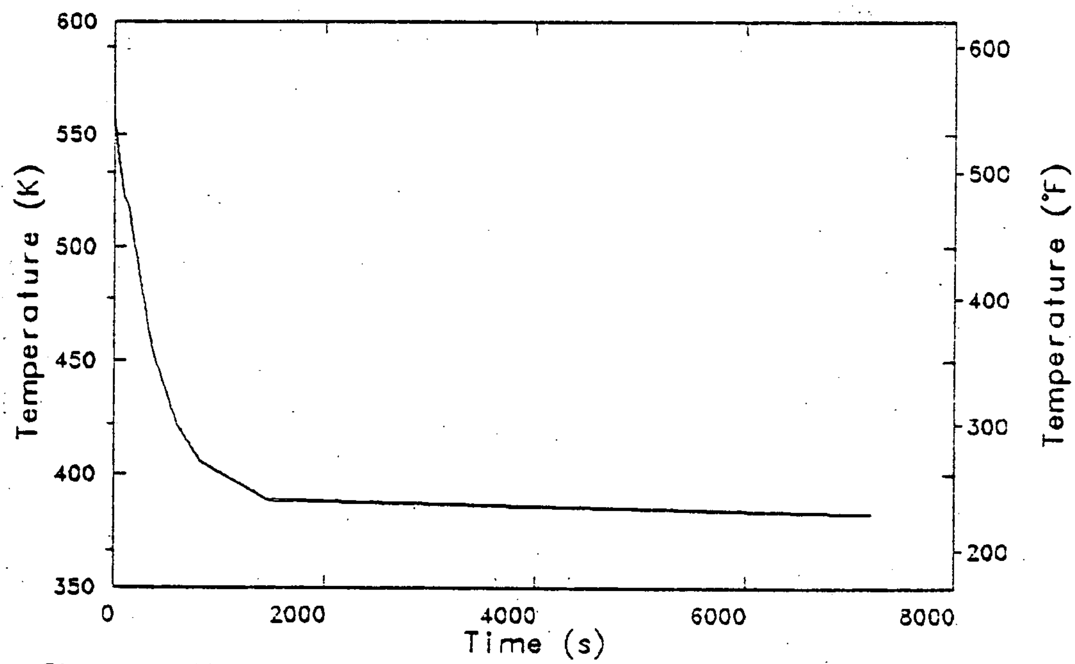

Figure A-130. Sequence 6-9 reactor vessel downcomer fluid temperature.

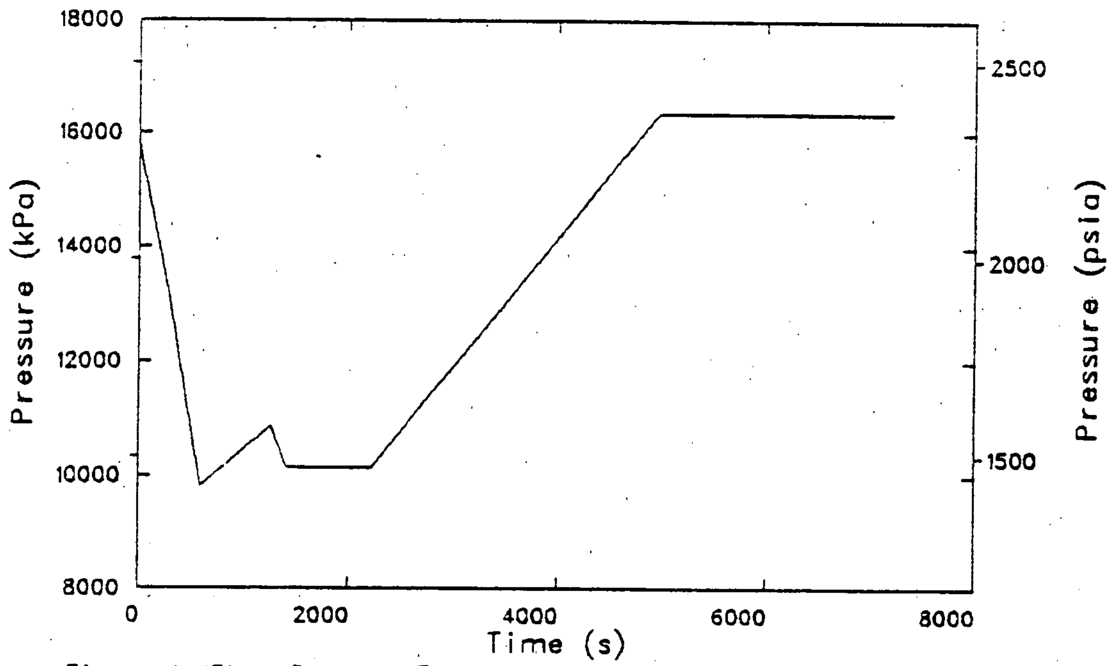

Figure A-131. Sequence 7-1 reactor vessel downcomer fluid pressure.

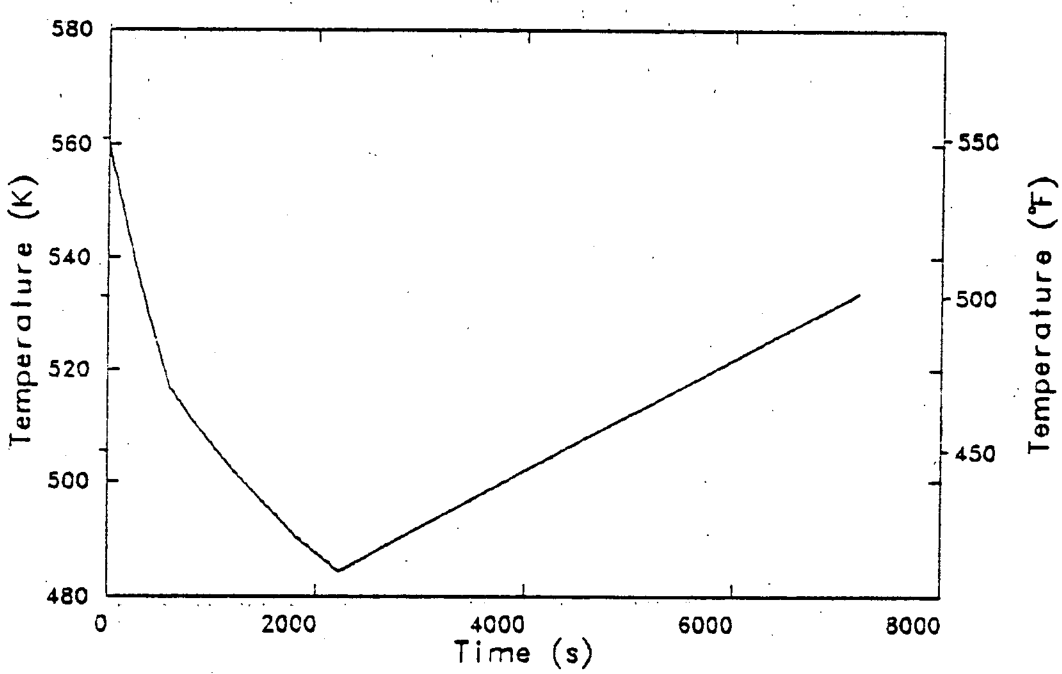

Figure A-132. Sequence 7-1 reactor vessel downcomer fluid temperature.

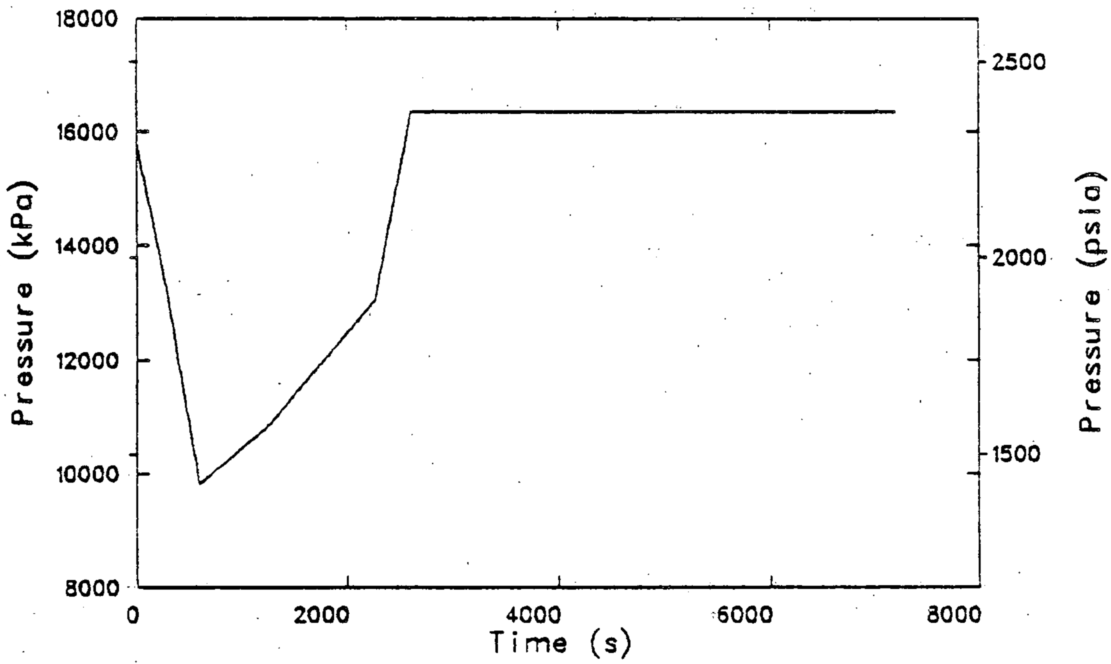

Figure A-133. Sequence 7-2 reactor vessel downcomer fluid pressure.

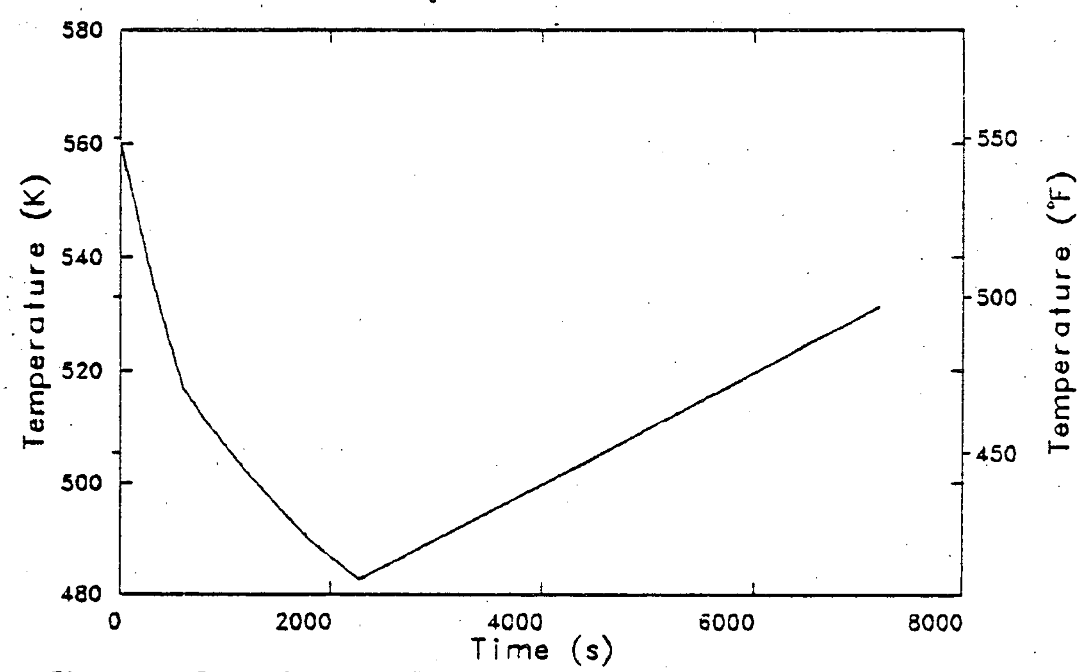

Figure A-134. Sequence 7-2 reactor vessel downcomer fluid temperature.

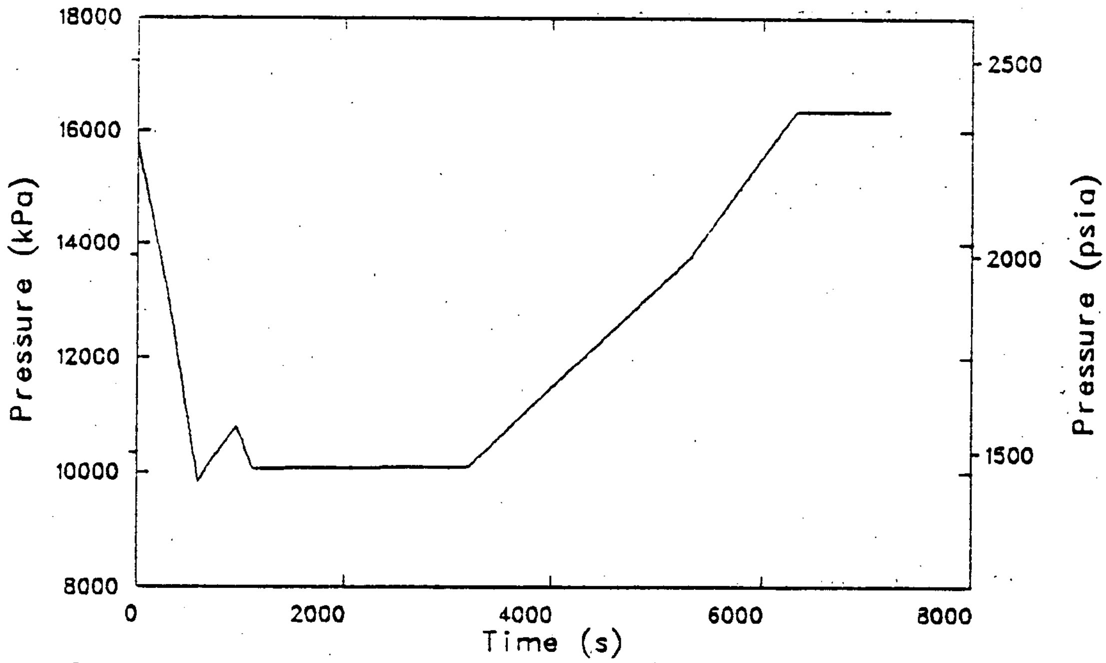

Figure A-135. Sequence 7-3 reactor vessel downcomer fluid pressure.

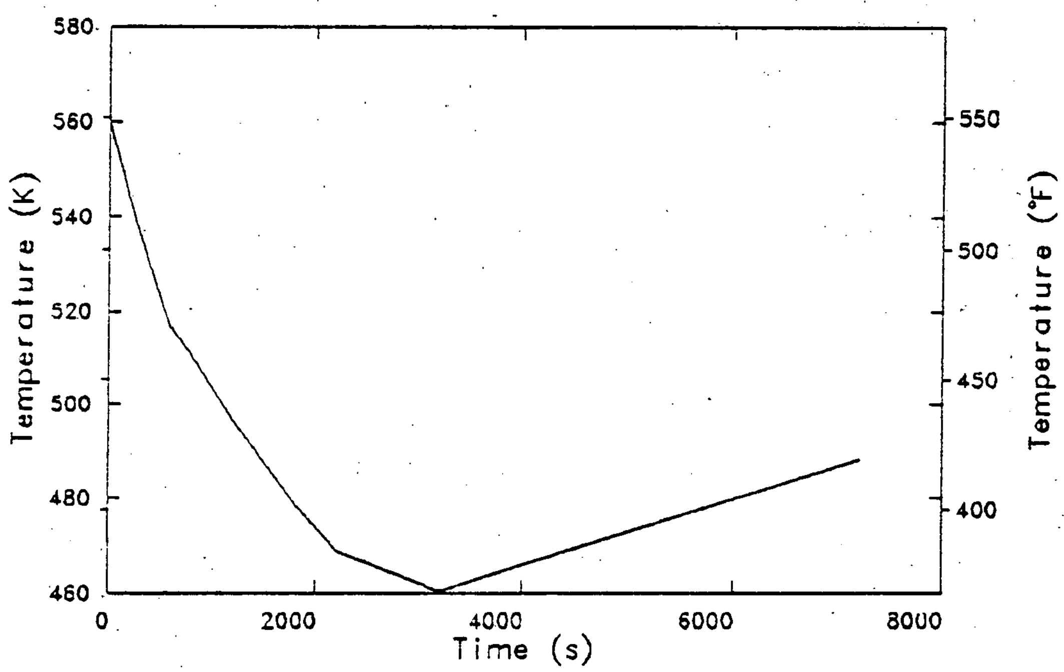

Figure A-136. Sequence 7-3 reactor vessel downcomer fluid temperature.

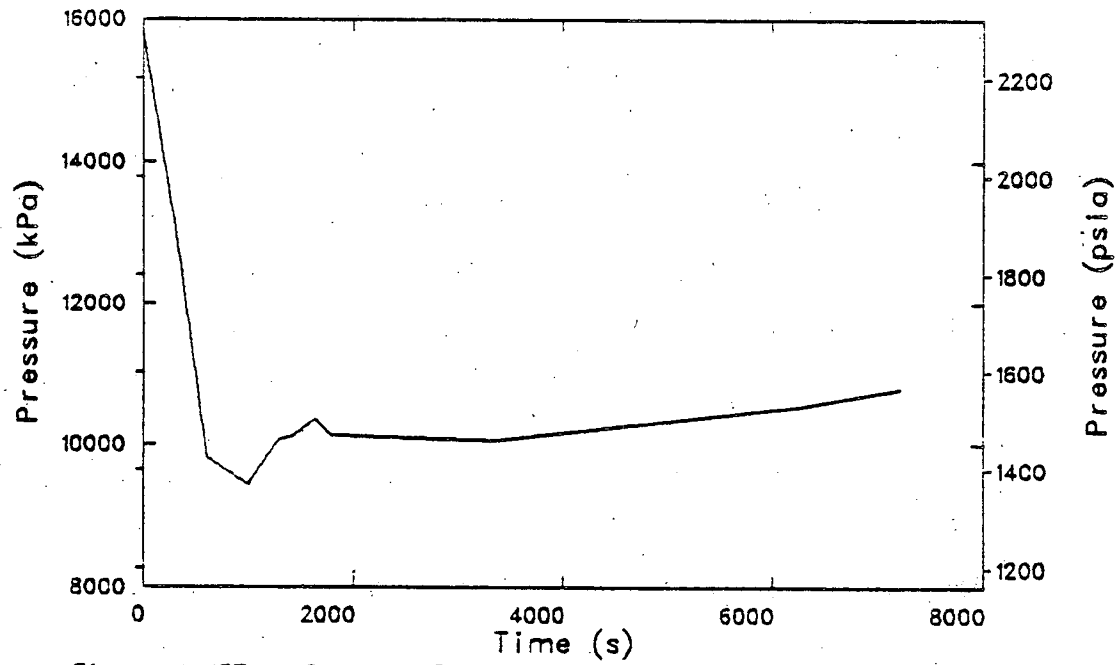

Figure A-137. Sequence 7-4 reactor vessel downcomer fluid pressure.

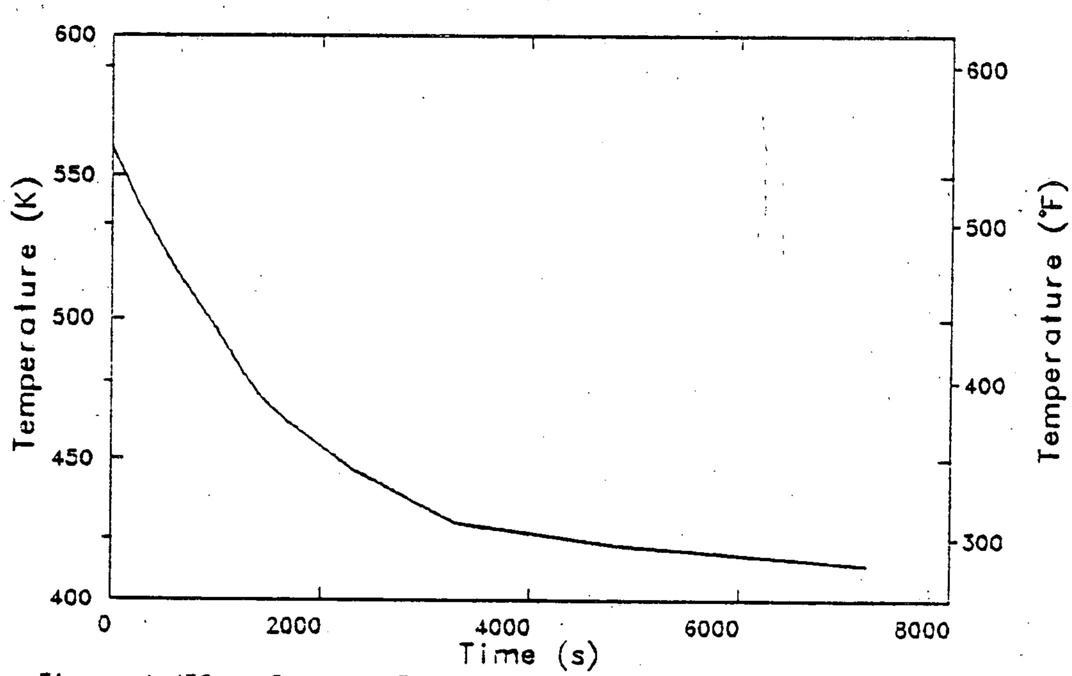

Figure A-138. Sequence 7-4 reactor vessel downcomer fluid temperature.

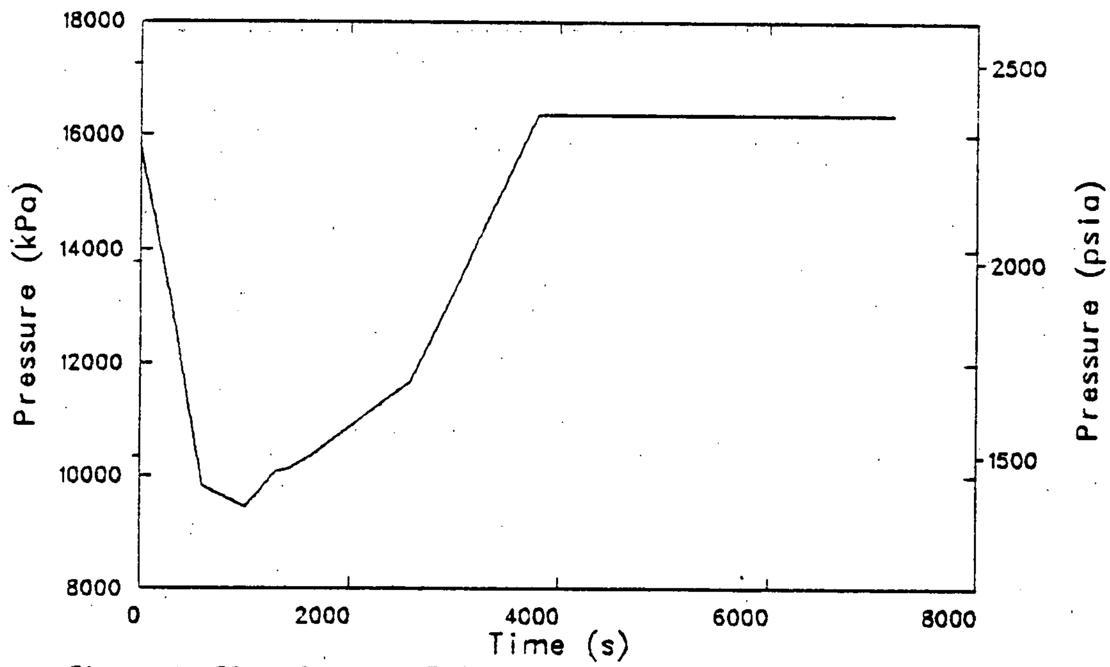

Figure A-139. Sequence 7-5 reactor vessel downcomer fluid pressure.

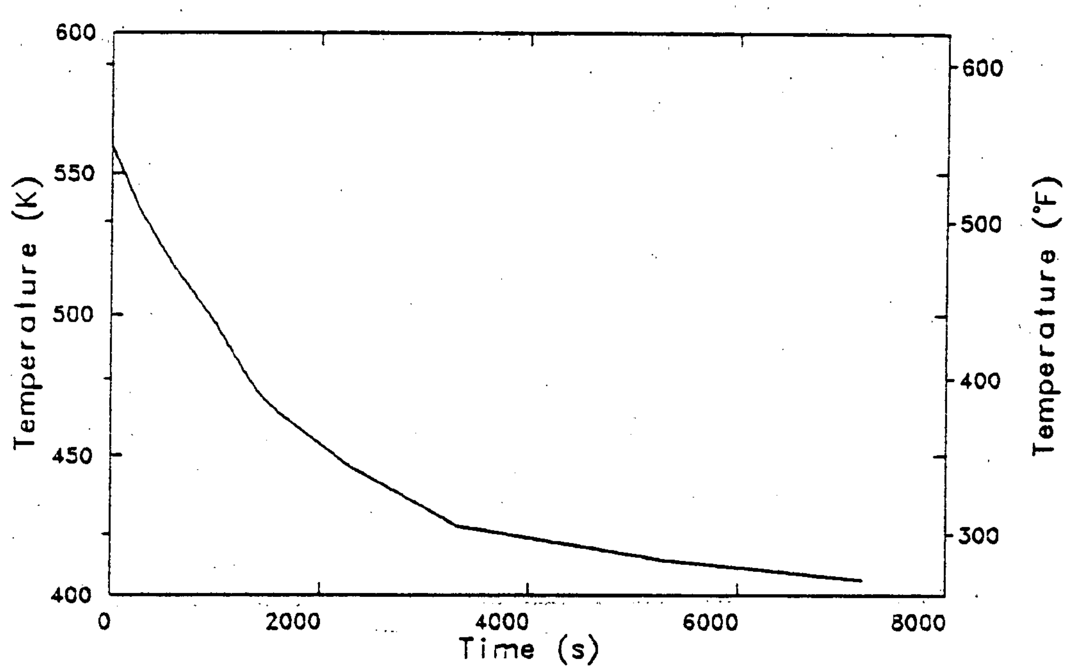

Figure A-140. Sequence 7-5 reactor vessel downcomer fluid temperature.

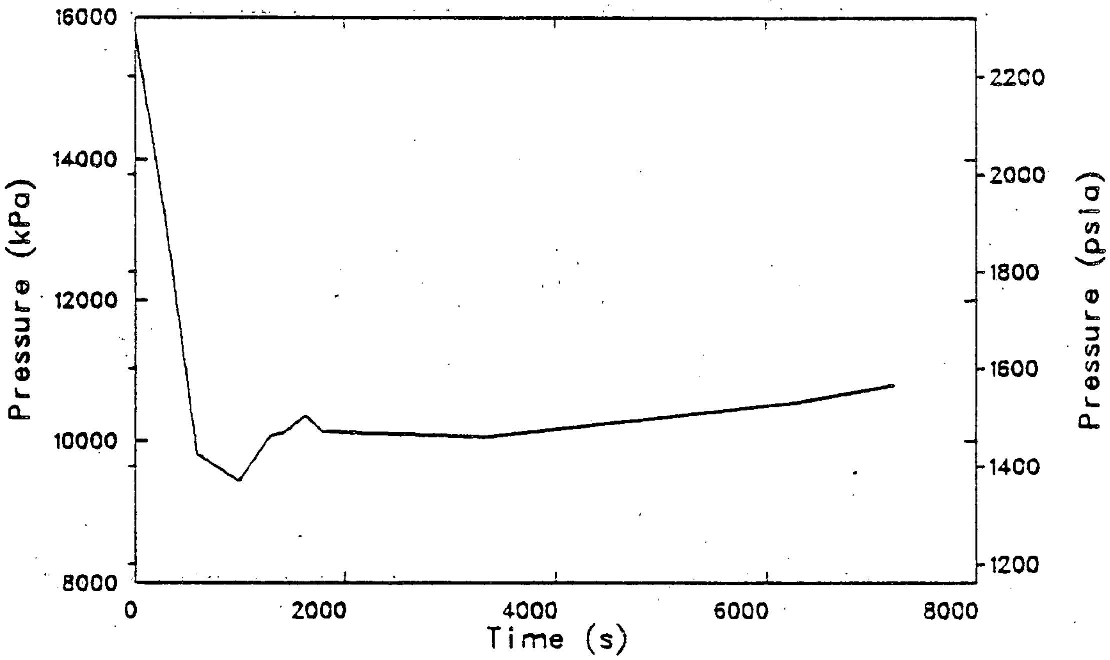

Figure A-141. Sequence 7-6 reactor vessel downcomer fluid pressure.

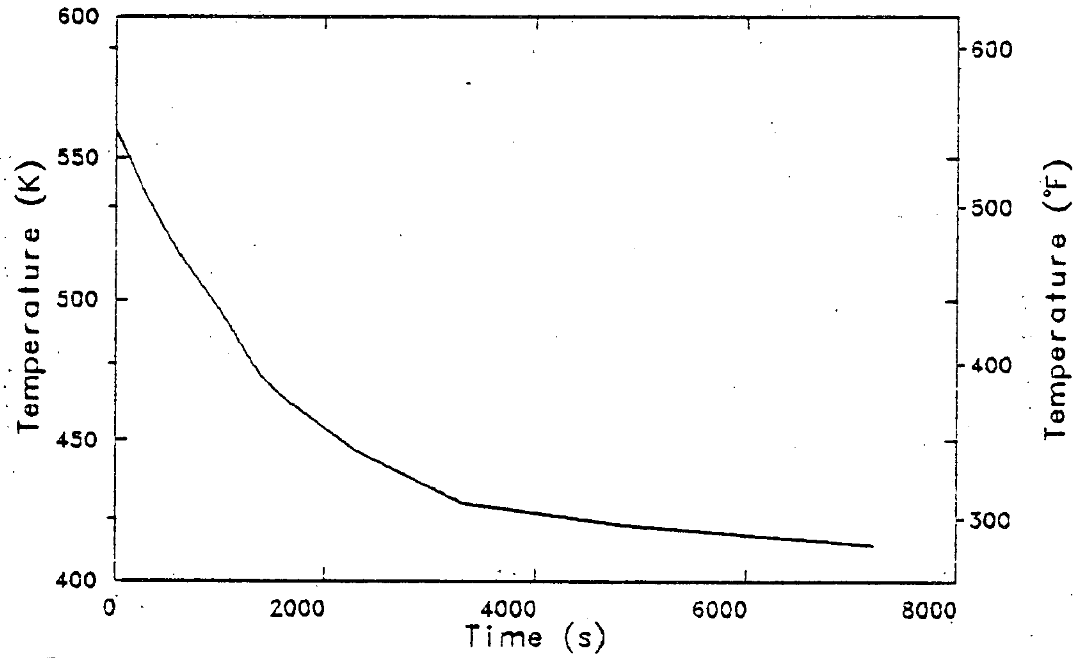

Figure A-142. Sequence 7-6 reactor vessel downcomer fluid temperature.

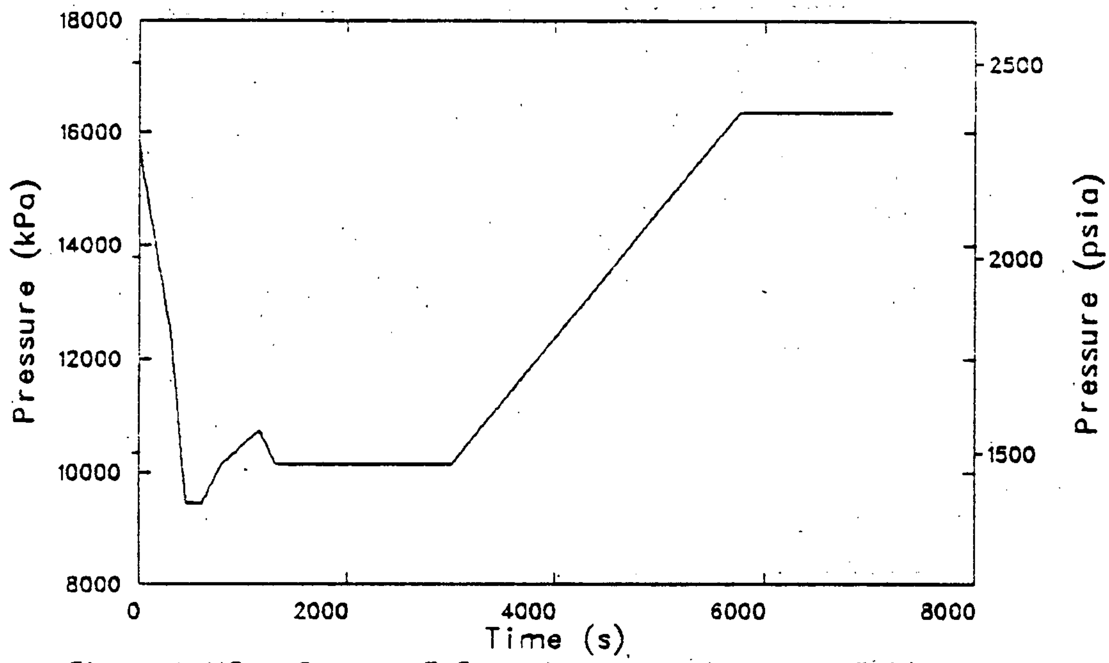

Figure A-143. Sequence 7-7 reactor vessel downcomer fluid pressure.

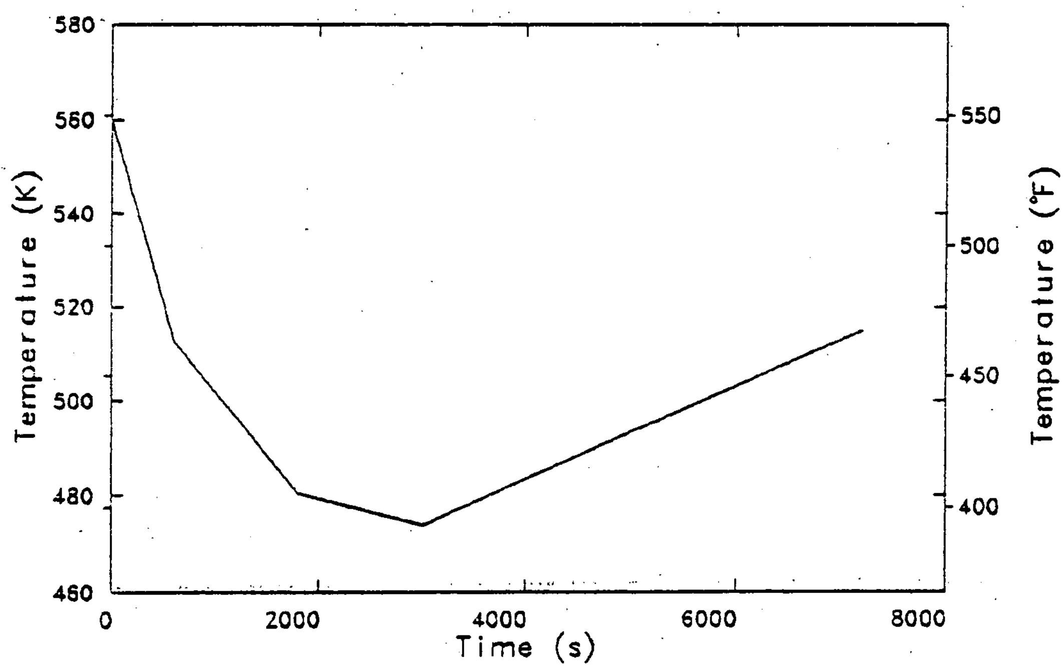

Figure A-144. Sequence 7-7 reactor vessel downcomer fluid temperature.

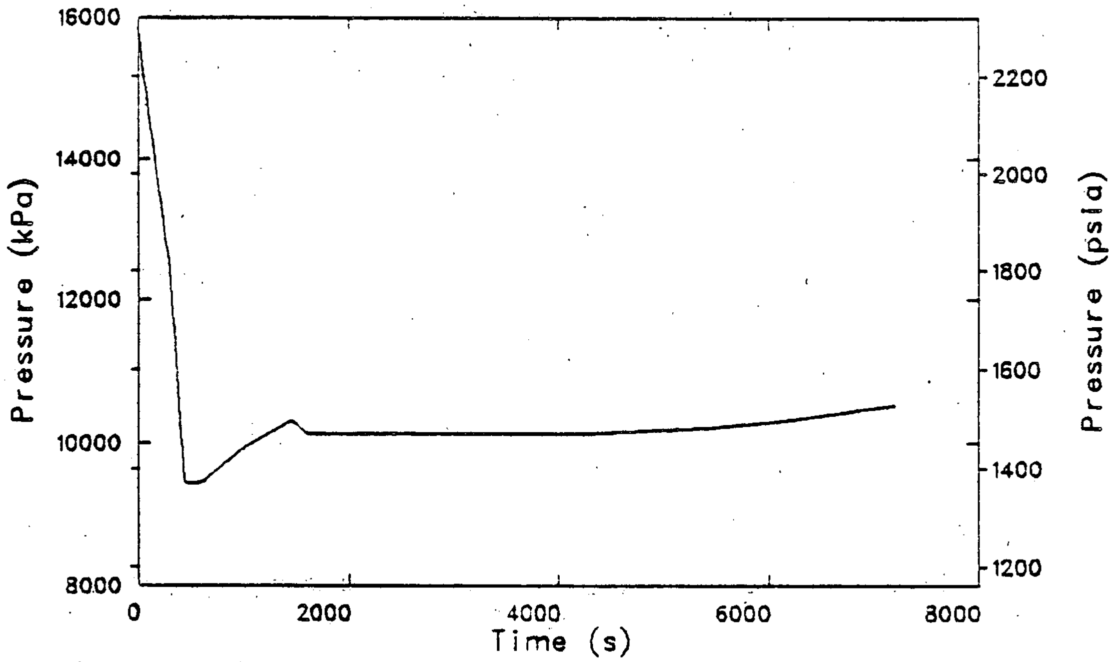

Figure A-145. Sequence 7-8 reactor vessel downcomer fluid pressure.

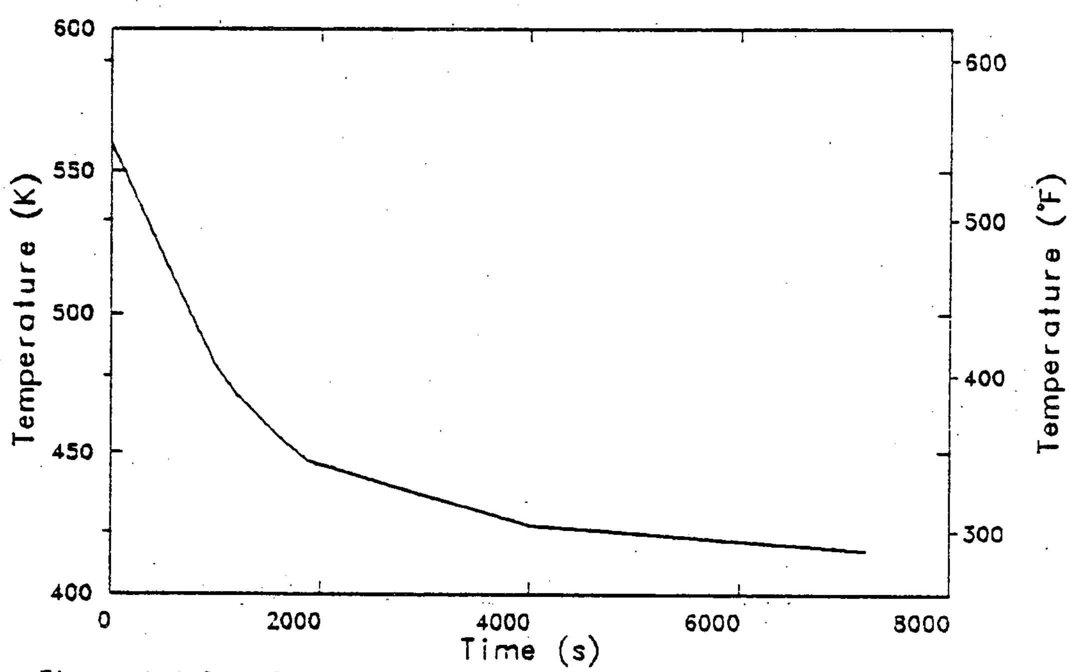

Figure A-146. Sequence 7-8 reactor vessel downcomer fluid temperature.

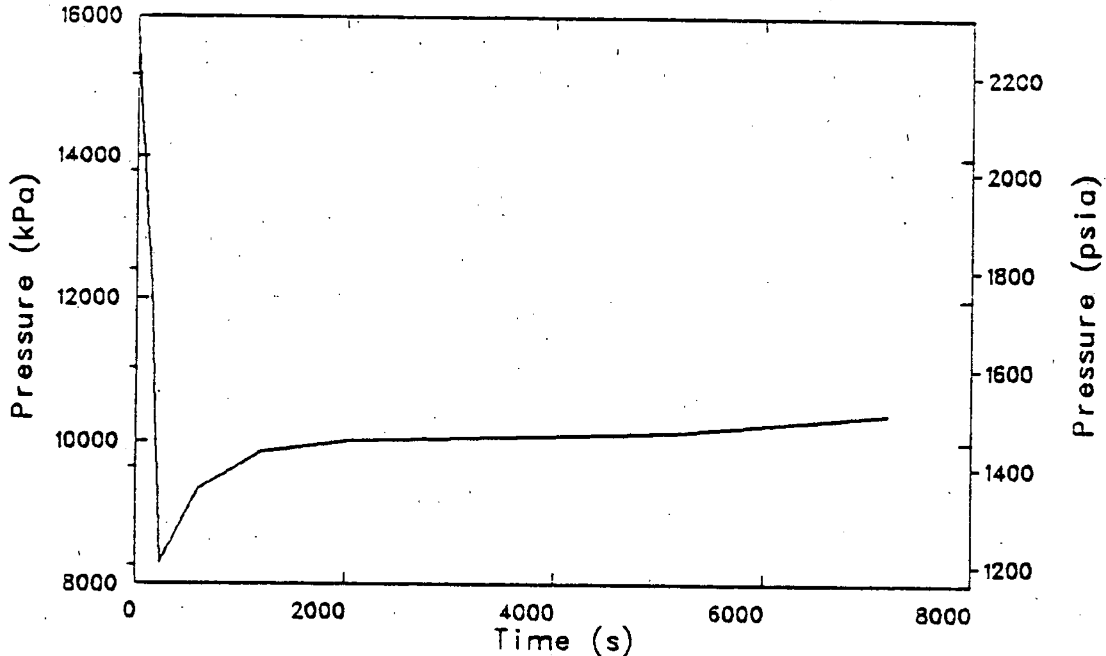

Figure A-147. Sequence 7-9 reactor vessel downcomer fluid pressure.

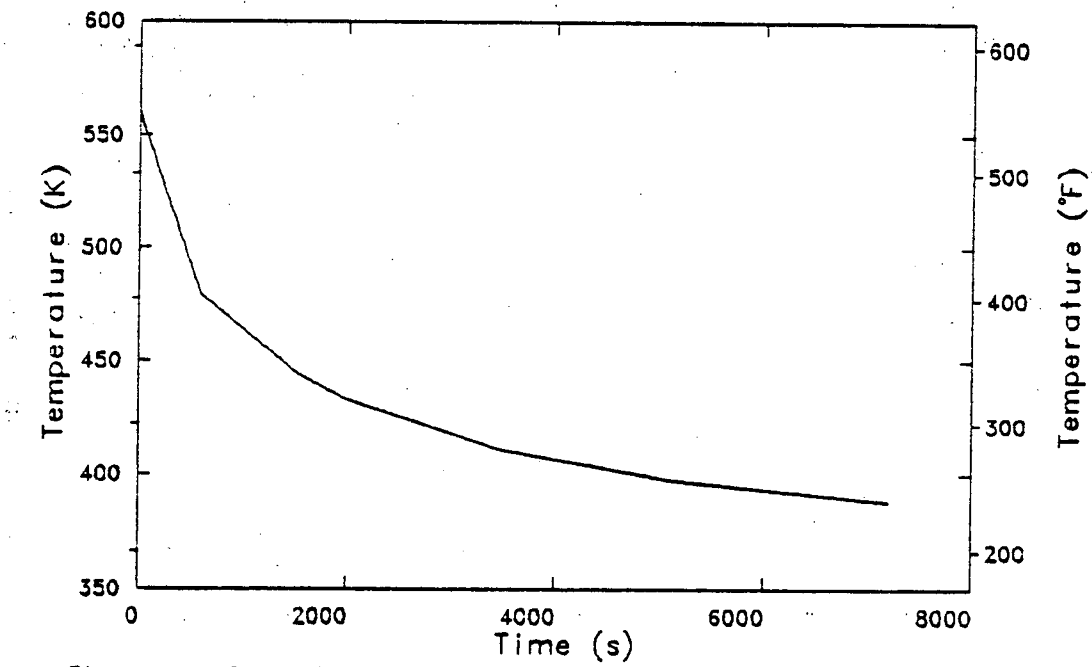

Figure A-148. Sequence 7-9 reactor vessel downcomer fluid temperature.

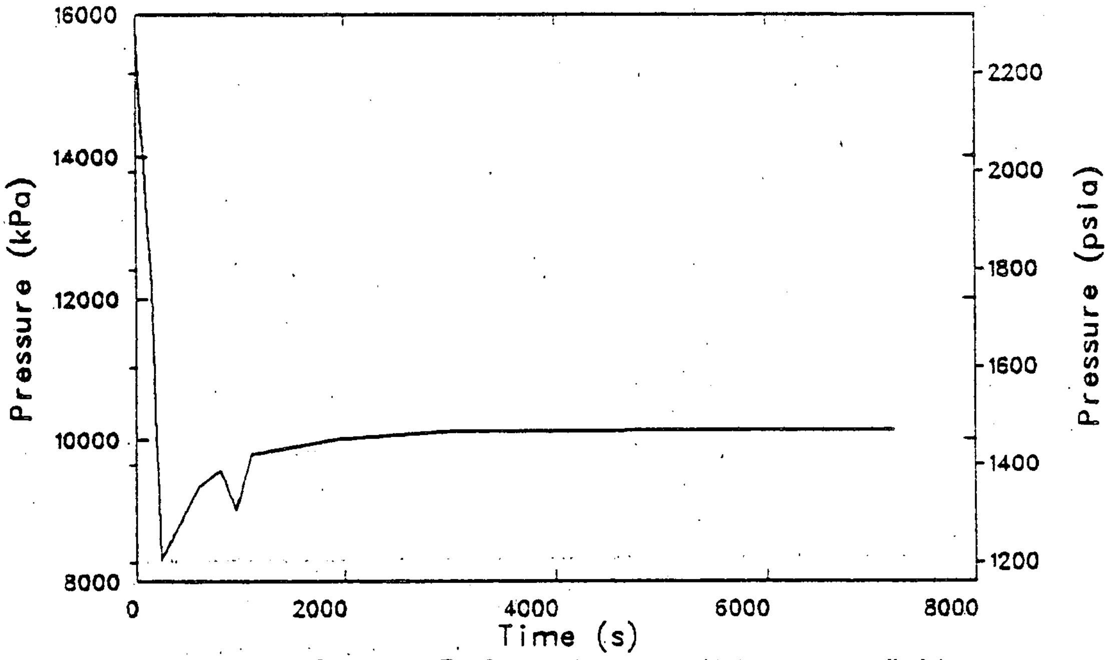

Figure A-149. Sequence 7-10 reactor vessel downcomer fluid pressure.

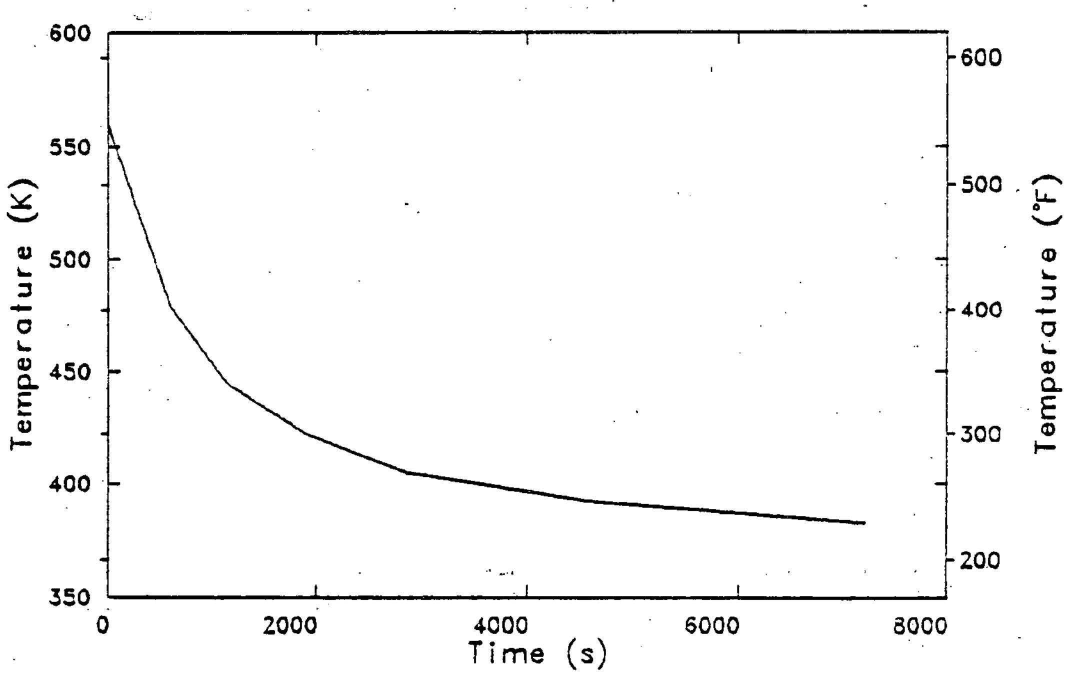

Figure A-150. Sequence 7-10 reactor vessel downcomer fluid temperature.

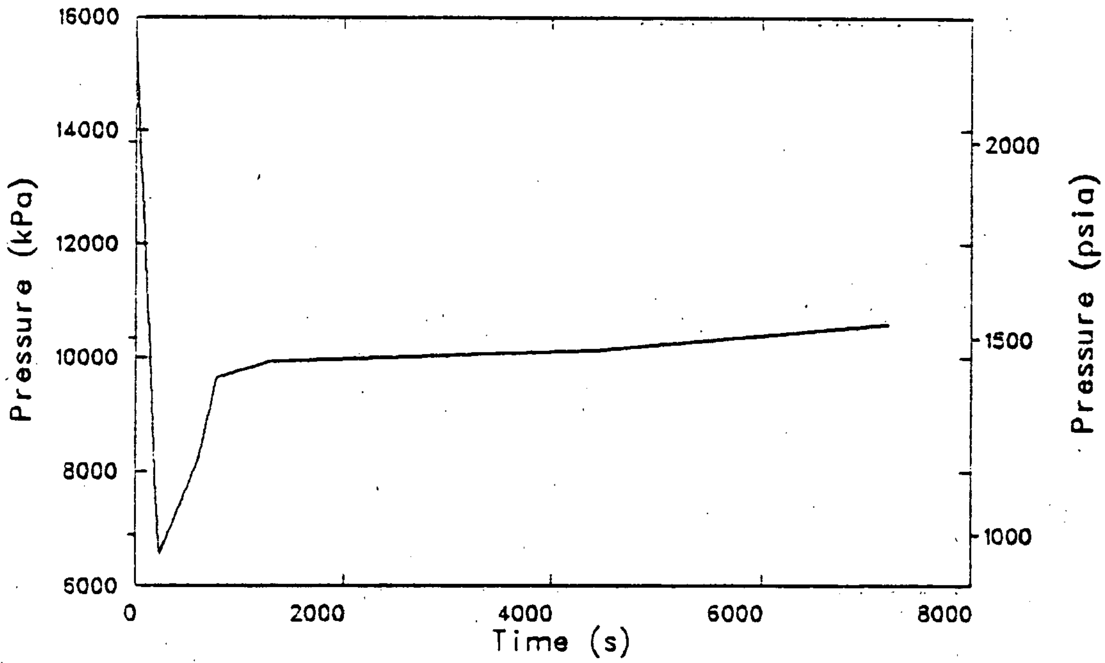

Figure A-151. Sequence 7-11 reactor vessel downcomer fluid pressure.

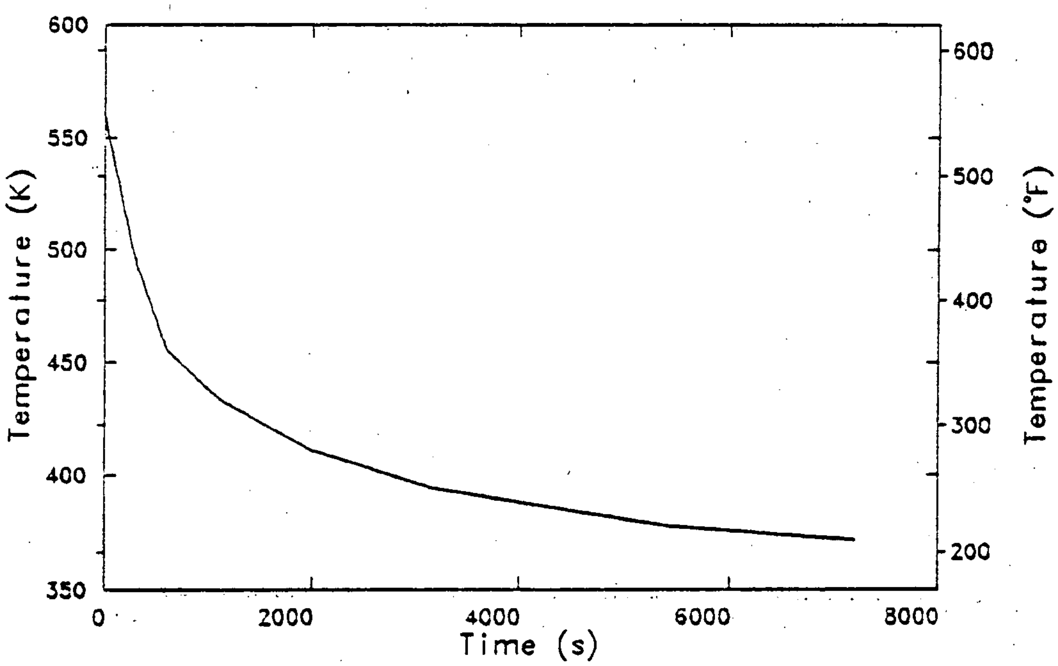

Figure A-152. Sequence 7-11 reactor vessel downcomer fluid temperature.

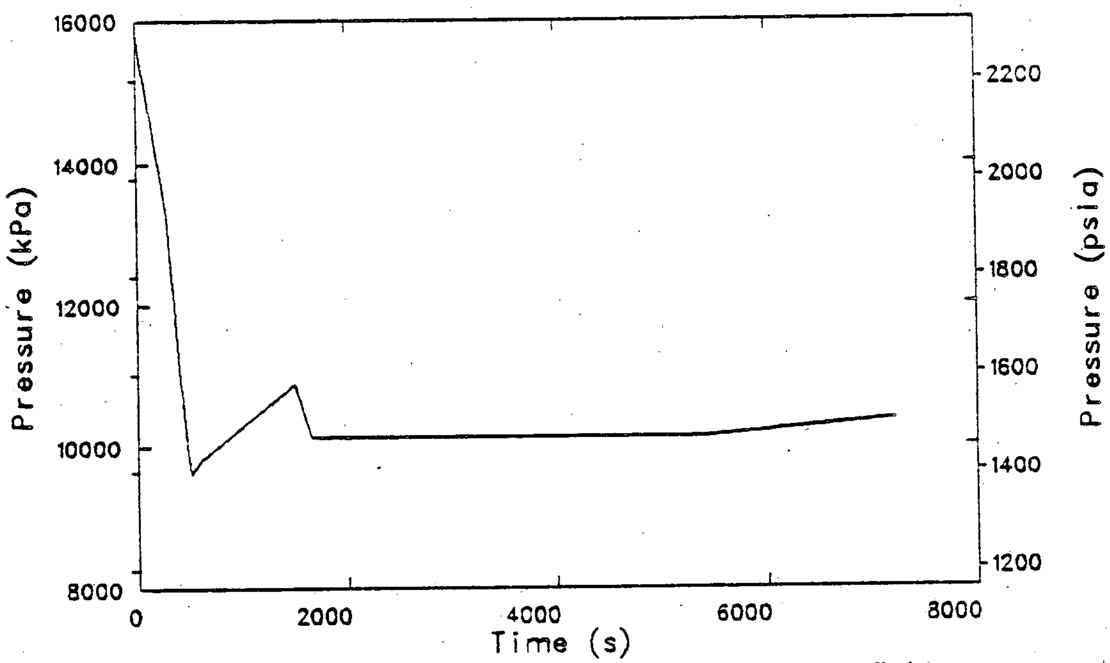

Figure A-153. Sequence 7-12 reactor vessel downcomer fluid pressure.

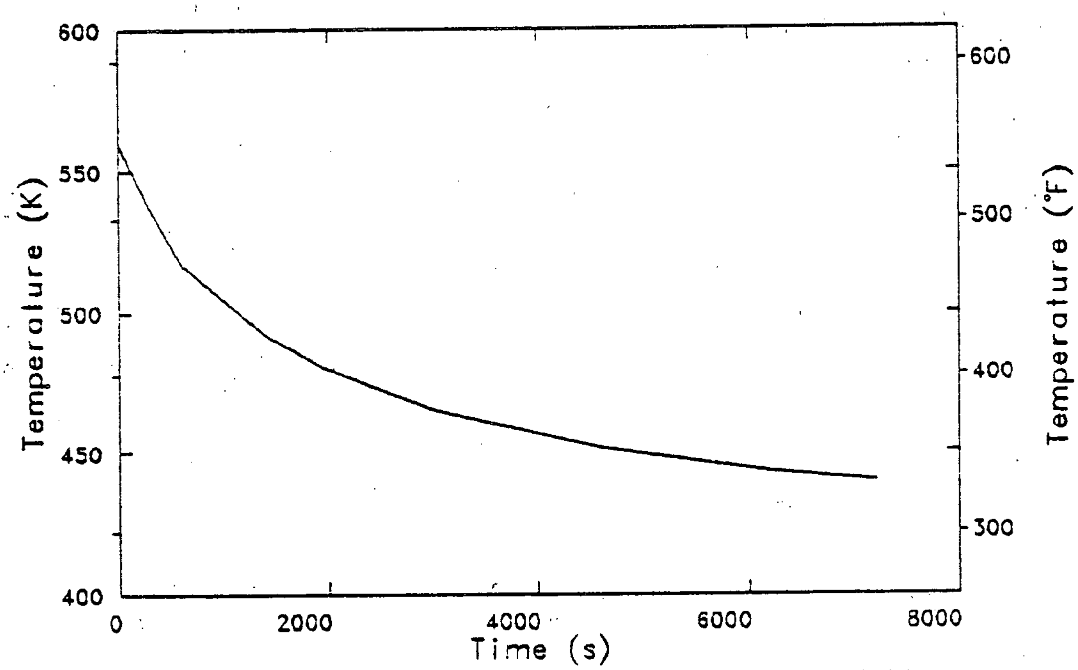

Figure A-154. Sequence 7-12 reactor vessel downcomer fluid temperature.

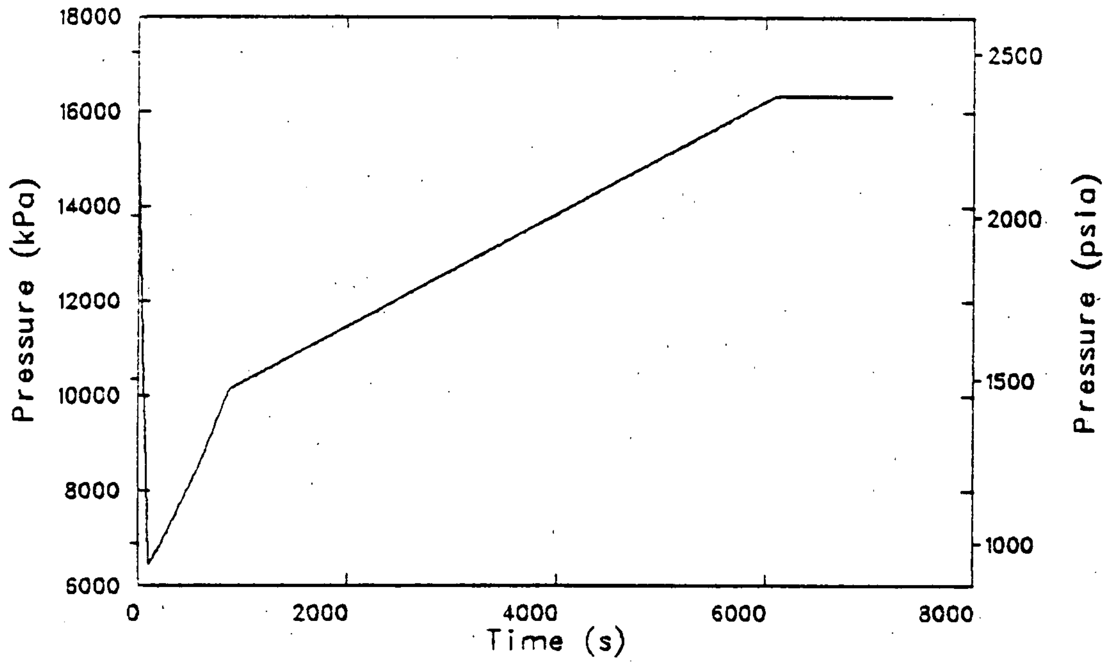

Figure A-155. Sequence 8-1 reactor vessel downcomer fluid pressure.

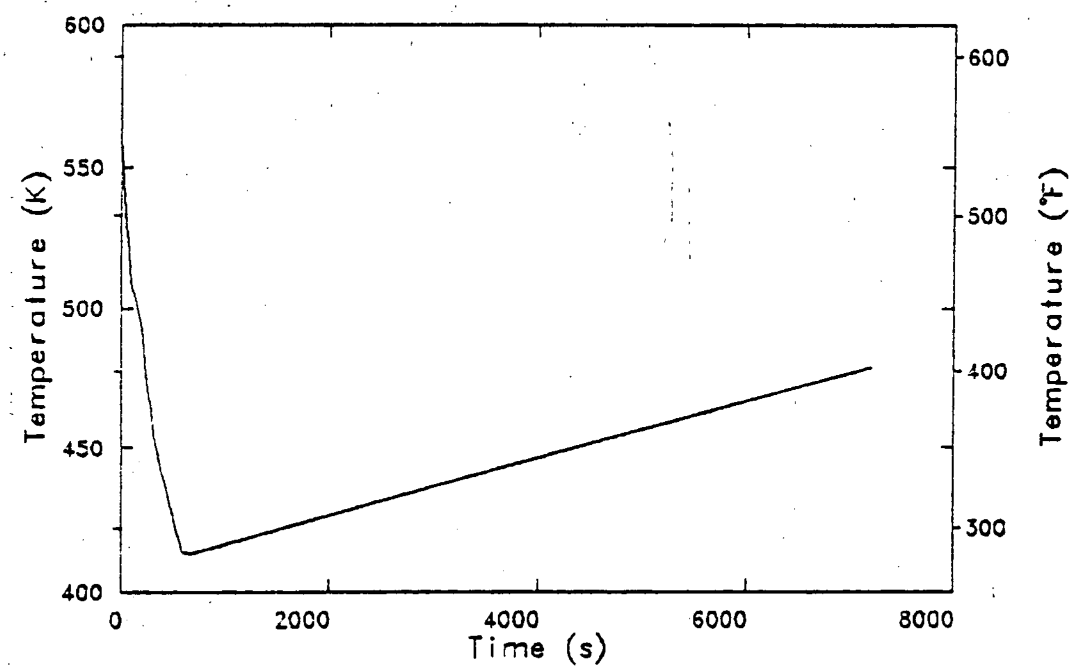

Figure A-156. Sequence 8-1 reactor vessel downcomer fluid temperature.

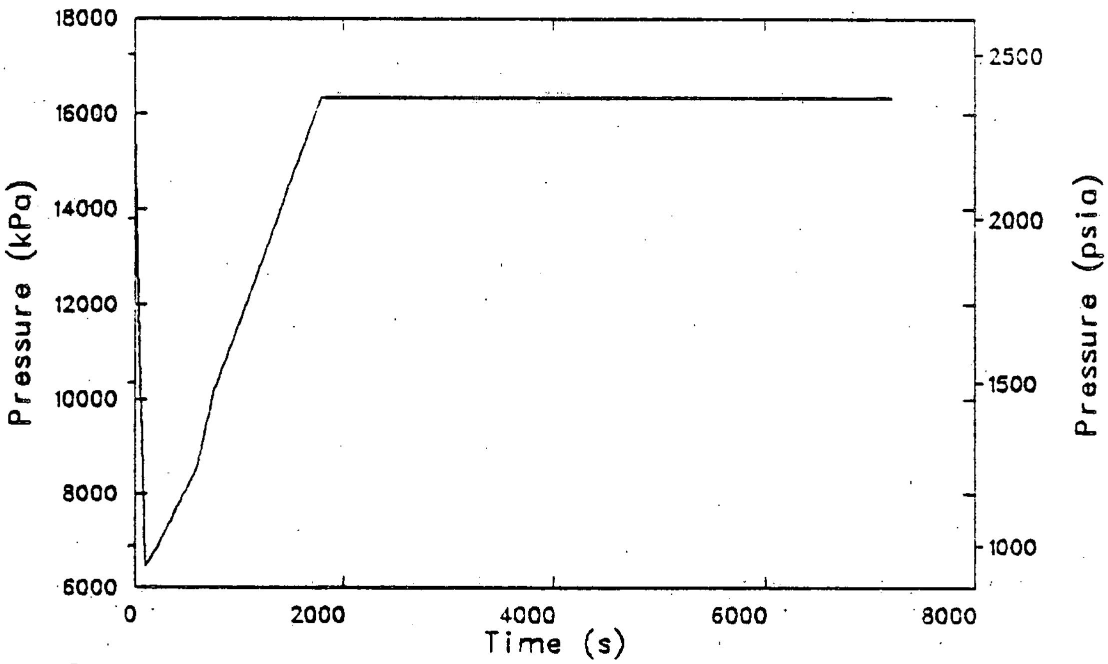

Figure A-157. Sequence 8-2 reactor vessel downcomer fluid pressure.

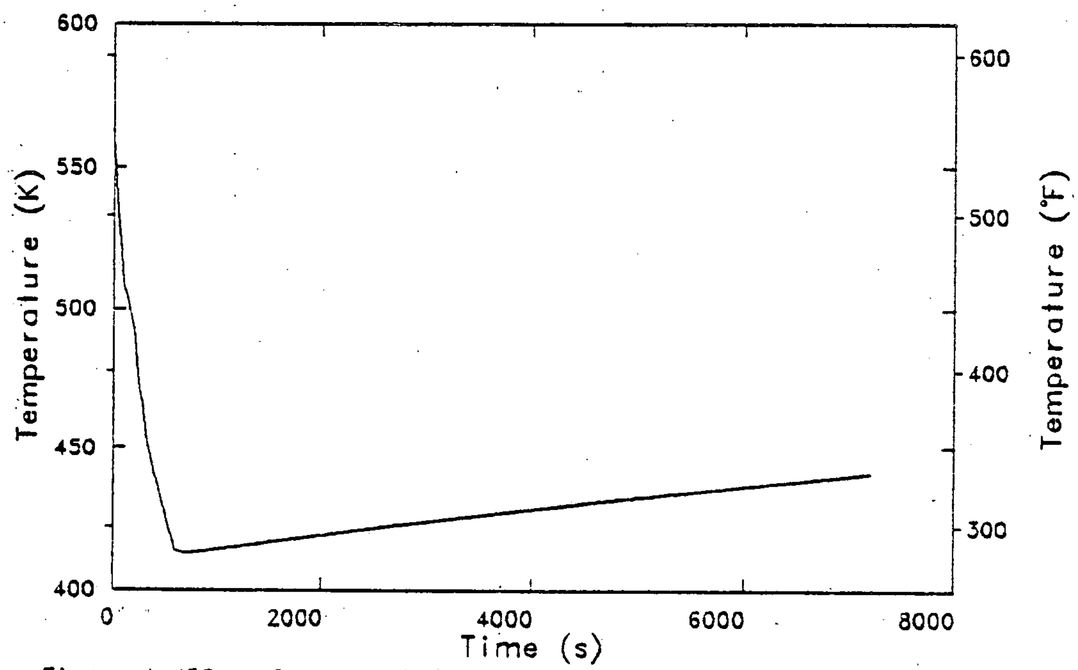

Figure A-158. Sequence 8-2 reactor vessel downcomer fluid temperature.

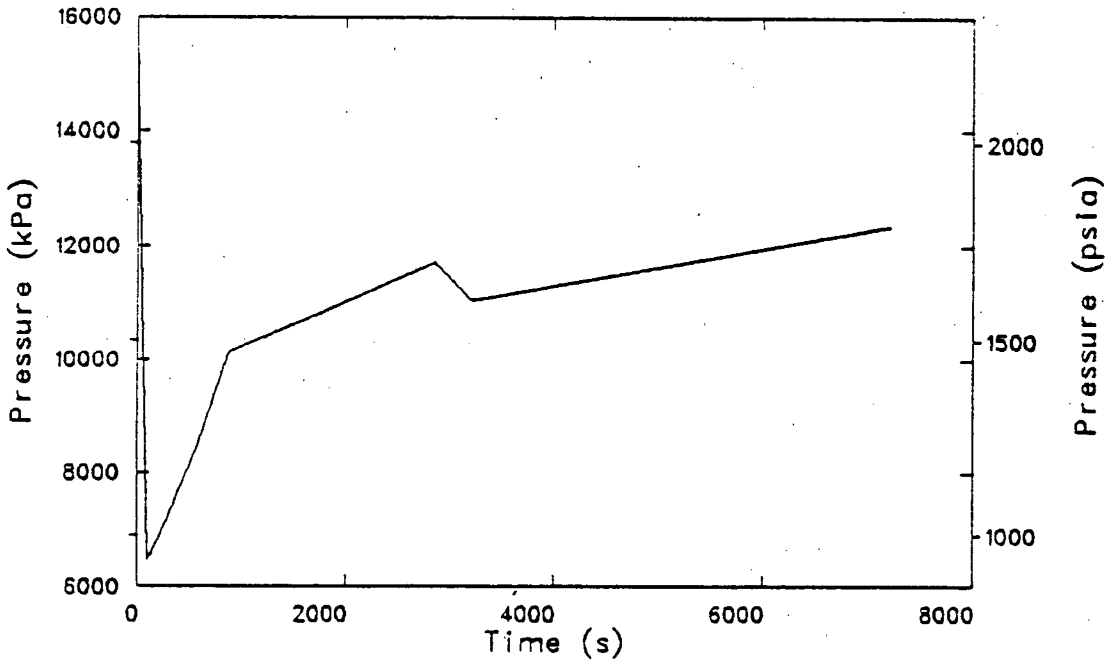

Figure A-159. Sequence 8-3 reactor vessel downcomer fluid pressure.

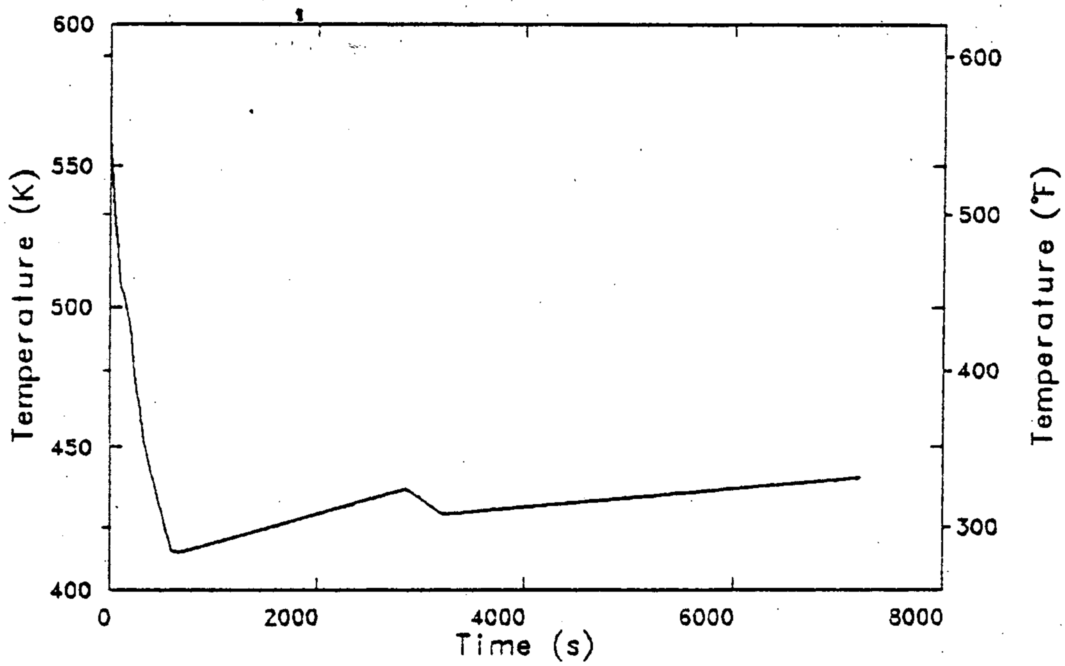

Figure A-160. Sequence 8-3 reactor vessel downcomer fluid temperature.

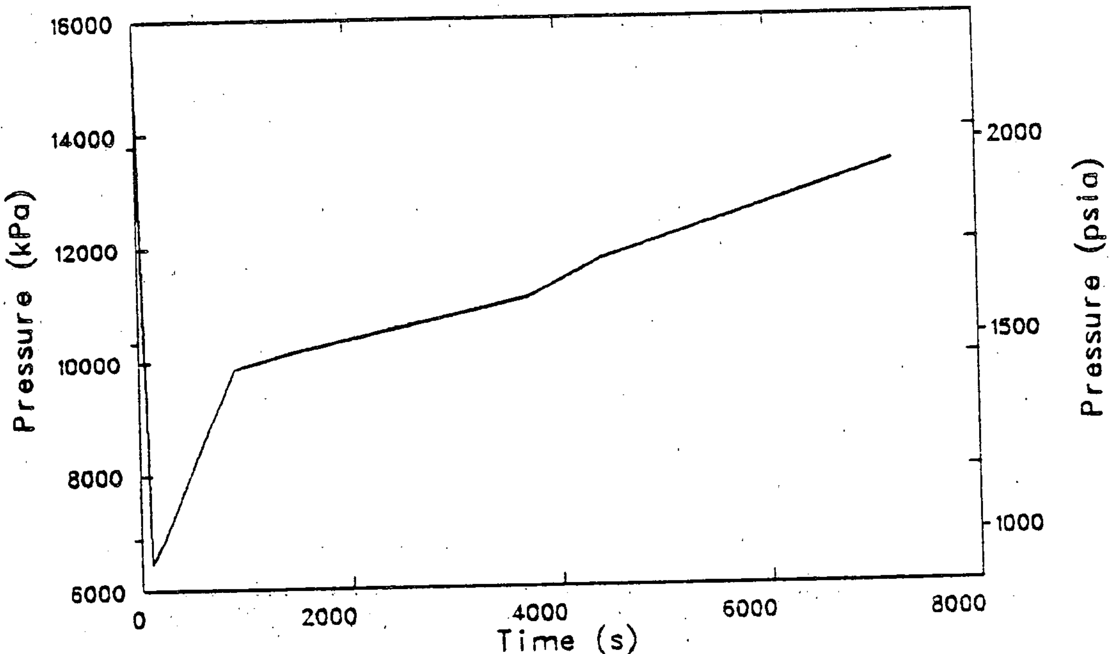

Figure A-161. Sequence 8-4 reactor vessel downcomer fluid pressure.

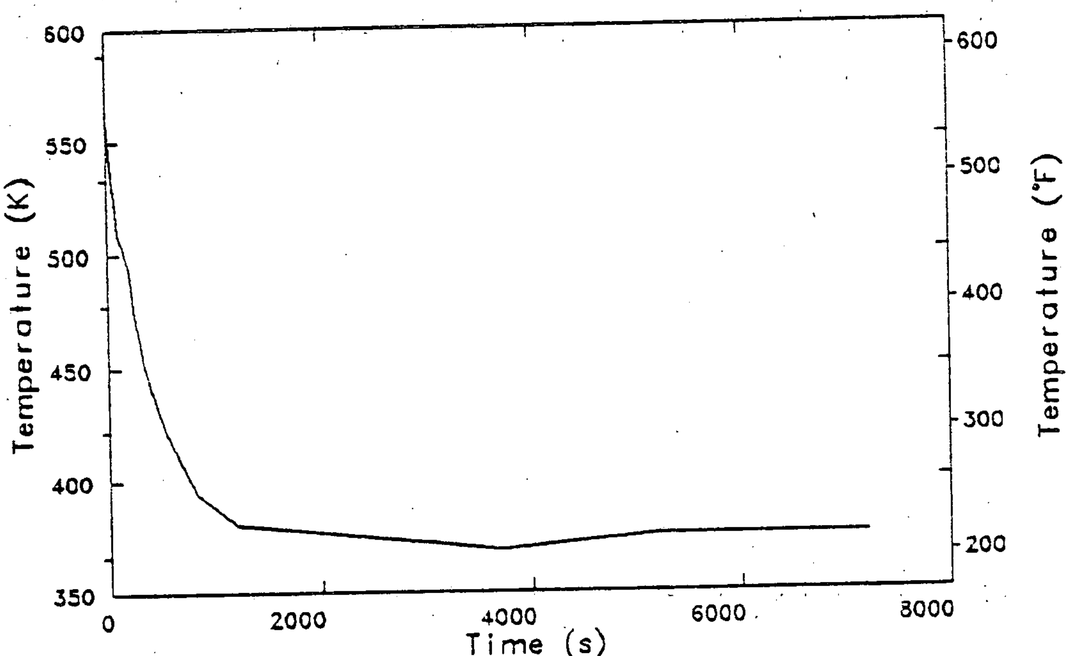

Figure A-162. Sequence 8-4 reactor vessel downcomer fluid temperature.

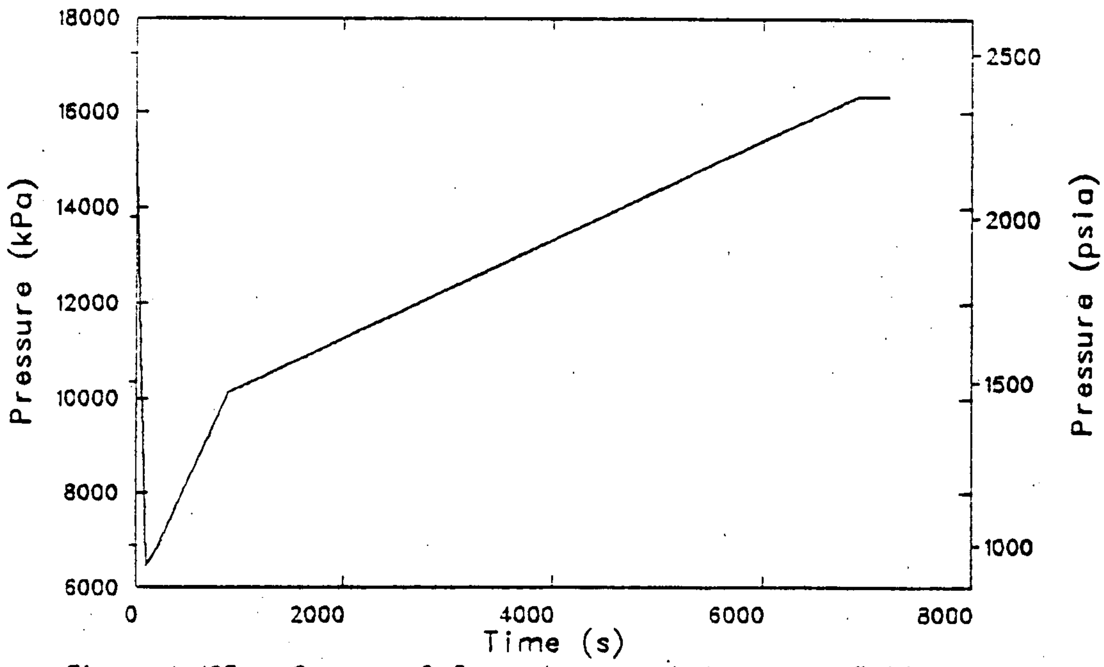

Figure A-163. Sequence 8-5 reactor vessel downcomer fluid pressure.

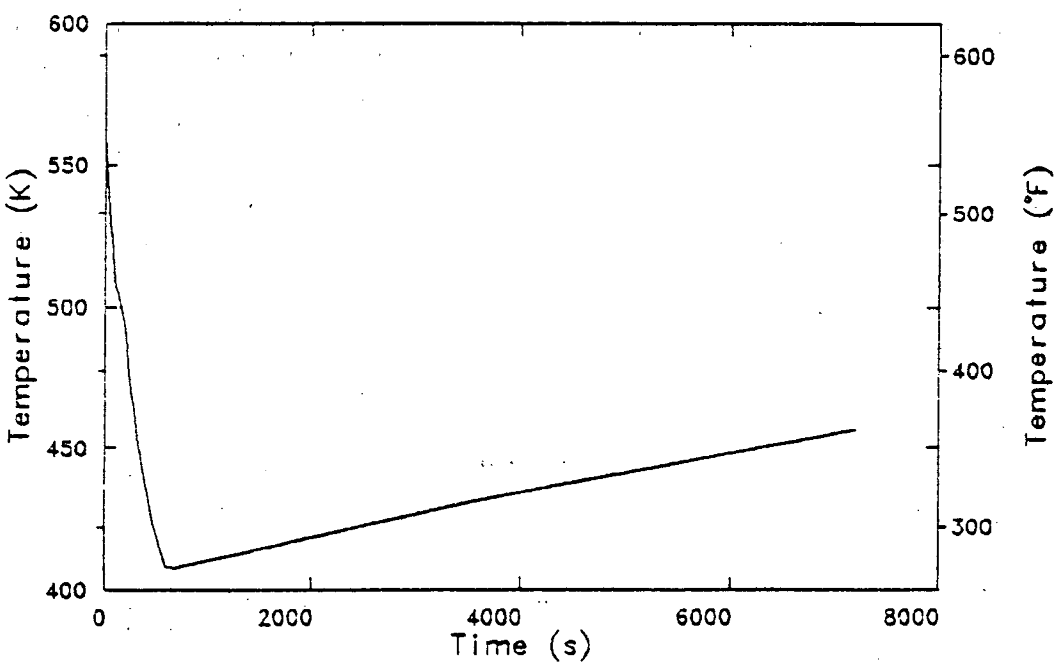

Figure A-164. Sequence 8-5 reactor vessel downcomer fluid temperature.

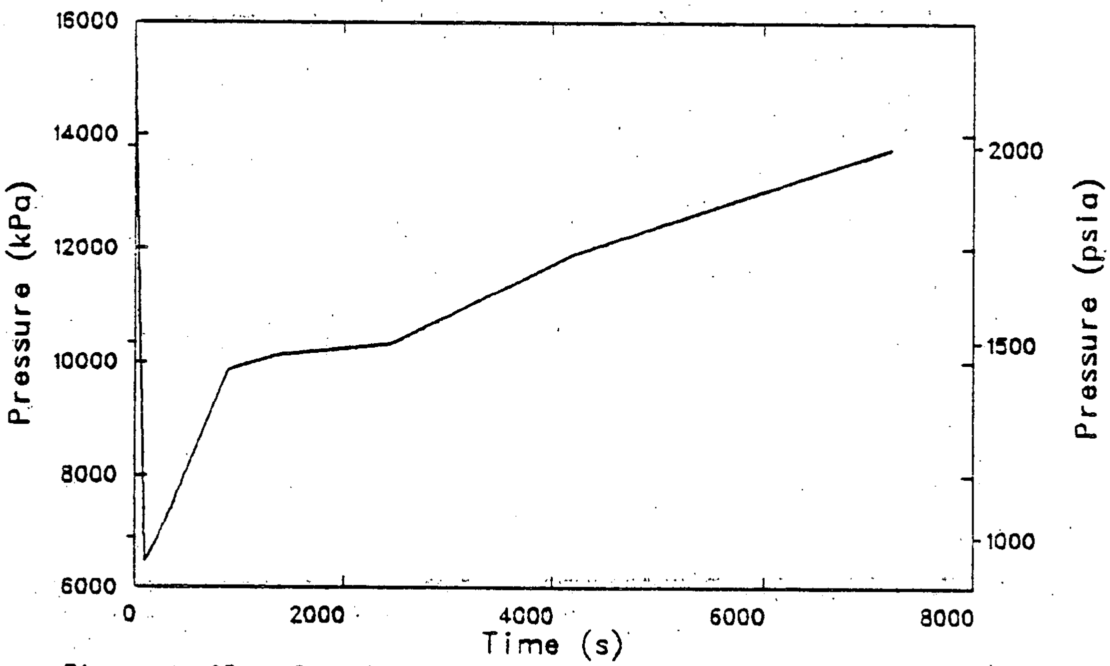

Figure A-165. Sequence 8-6 reactor vessel downcomer fluid pressure.

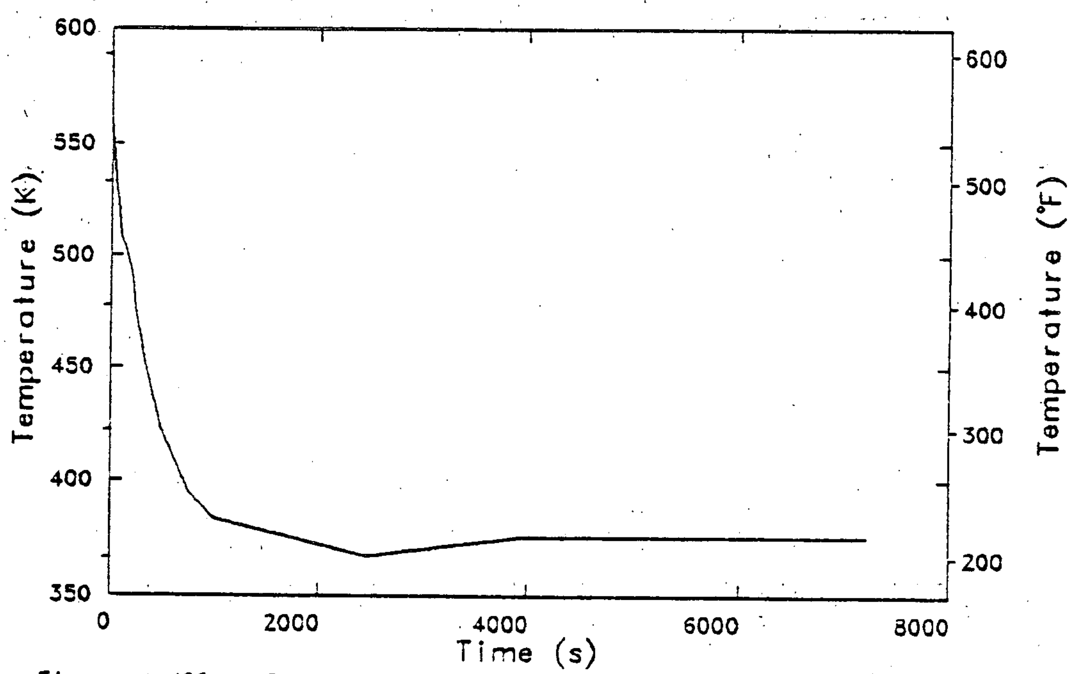

Figure A-166. Sequence 8-6 reactor vessel downcomer fluid temperature.

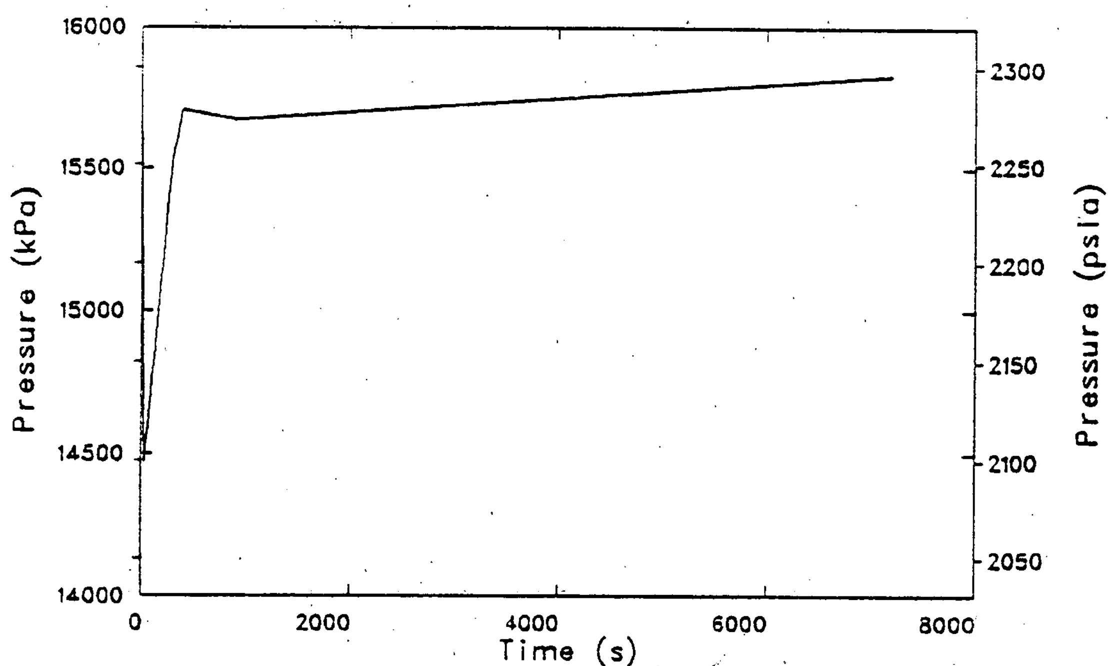

Figure A-167. Sequence 9-1 reactor vessel downcomer fluid pressure.

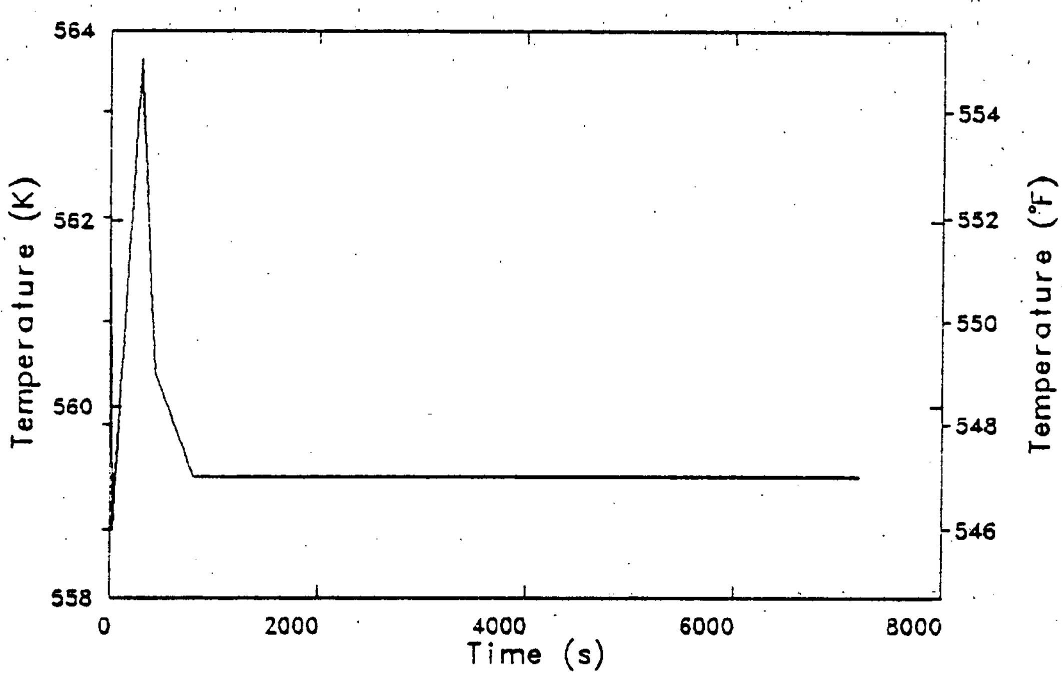

Figure A-168. Sequence 9-1 reactor vessel downcomer fluid temperature.

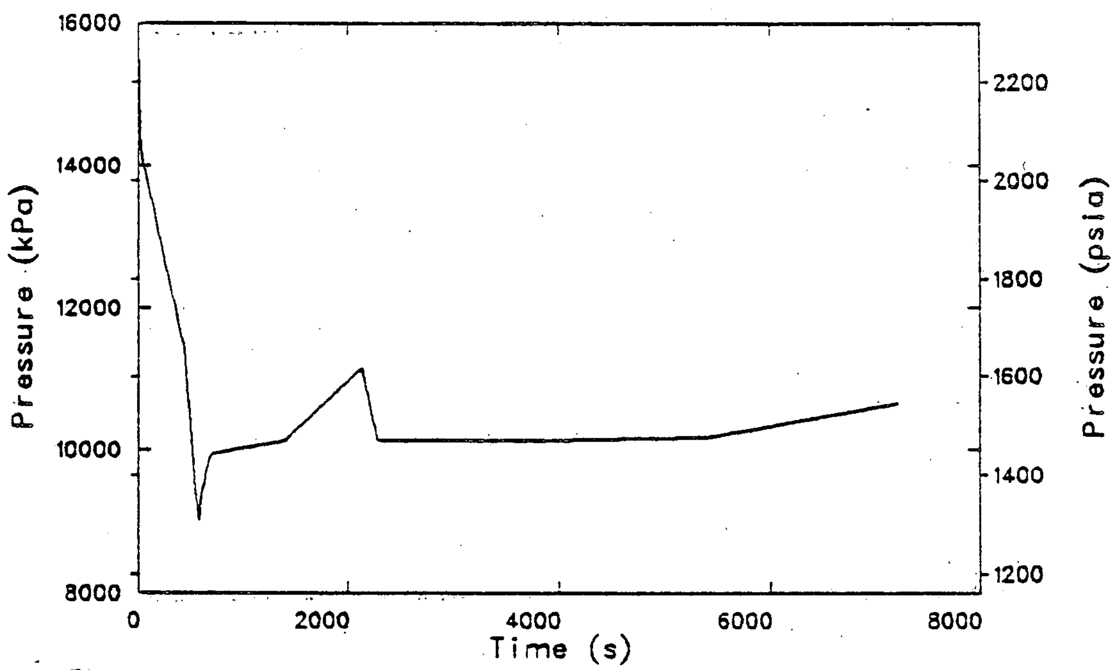

Figure A-169. Sequence 9-2 reactor vessel downcomer fluid pressure.

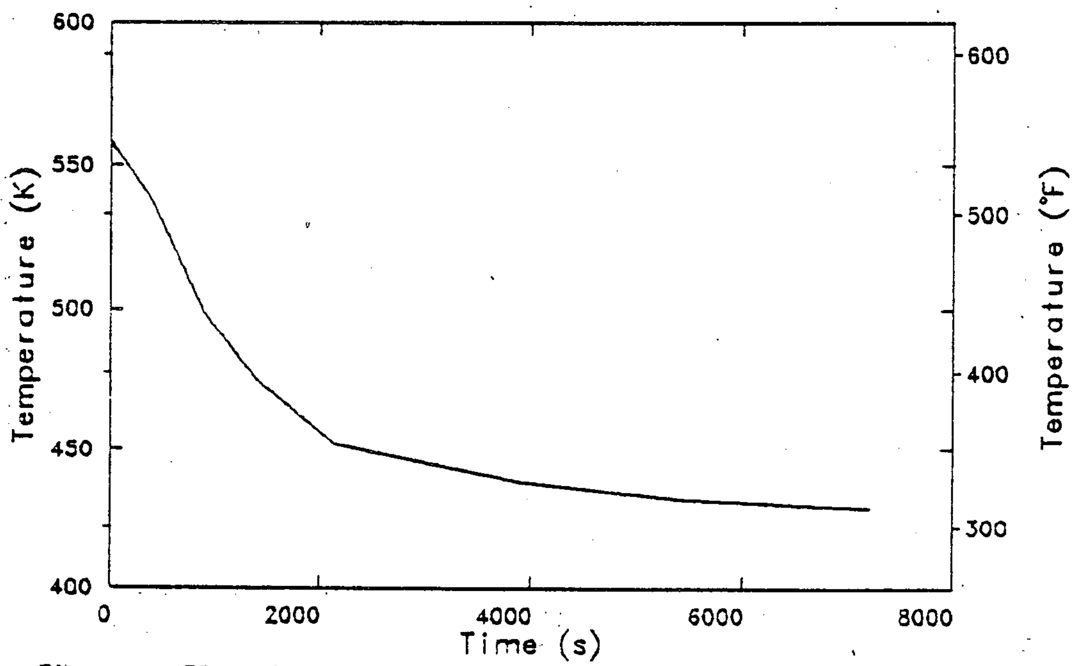

Figure A-170. Sequence 9-2 reactor vessel downcomer fluid temperature.

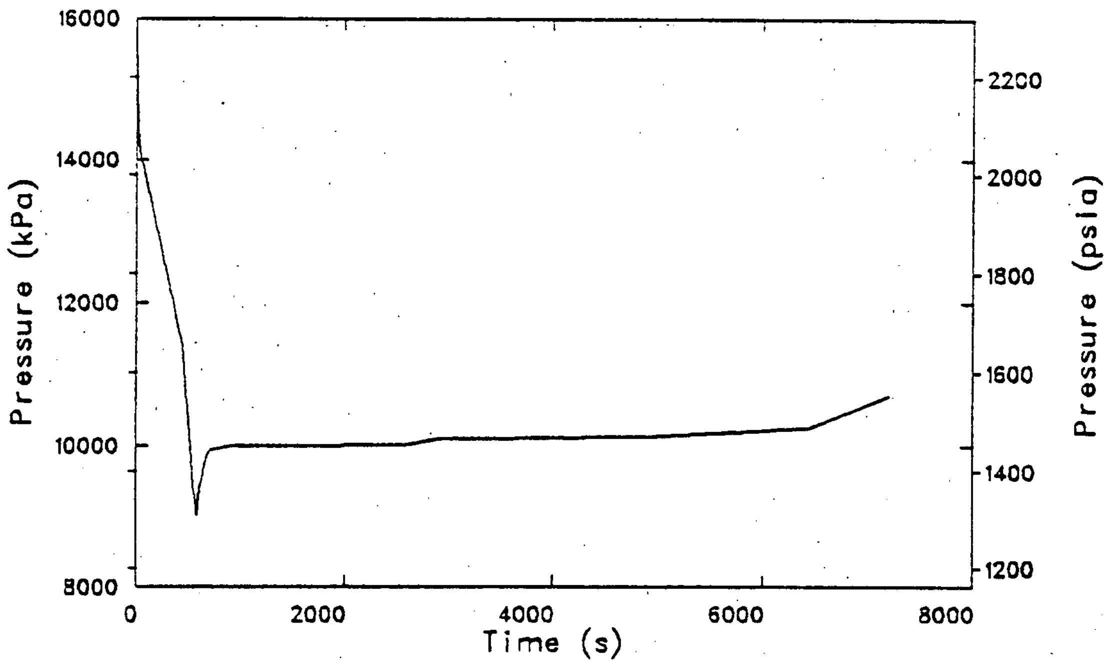

Figure A-171. Sequence 9-3 reactor vessel downcomer fluid pressure.

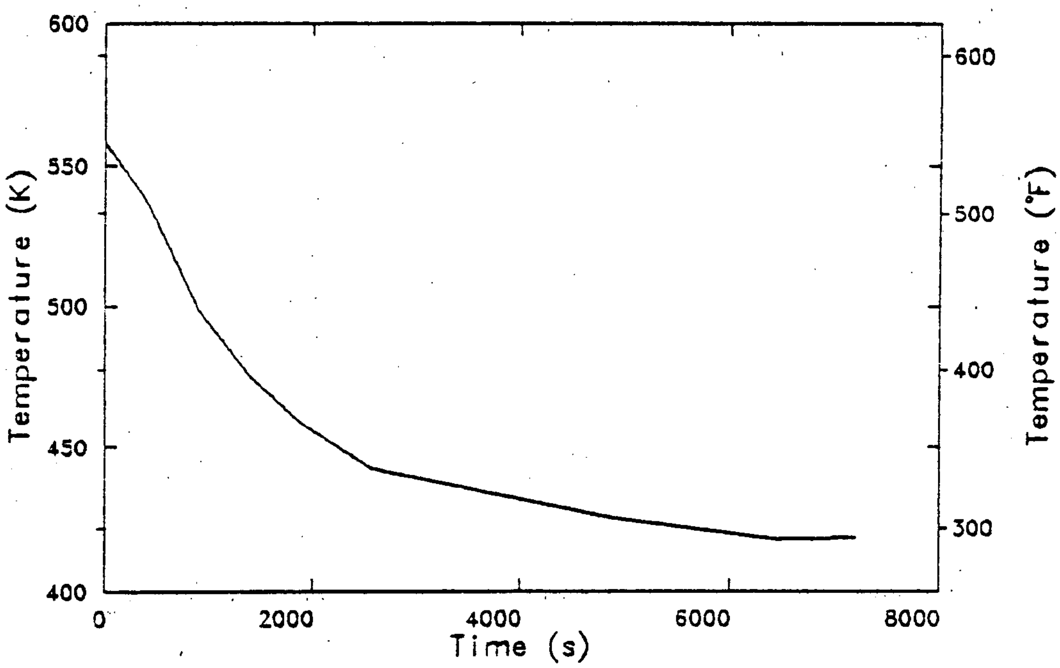

Figure A-172. Sequence 9-3 reactor vessel downcomer fluid temperature.

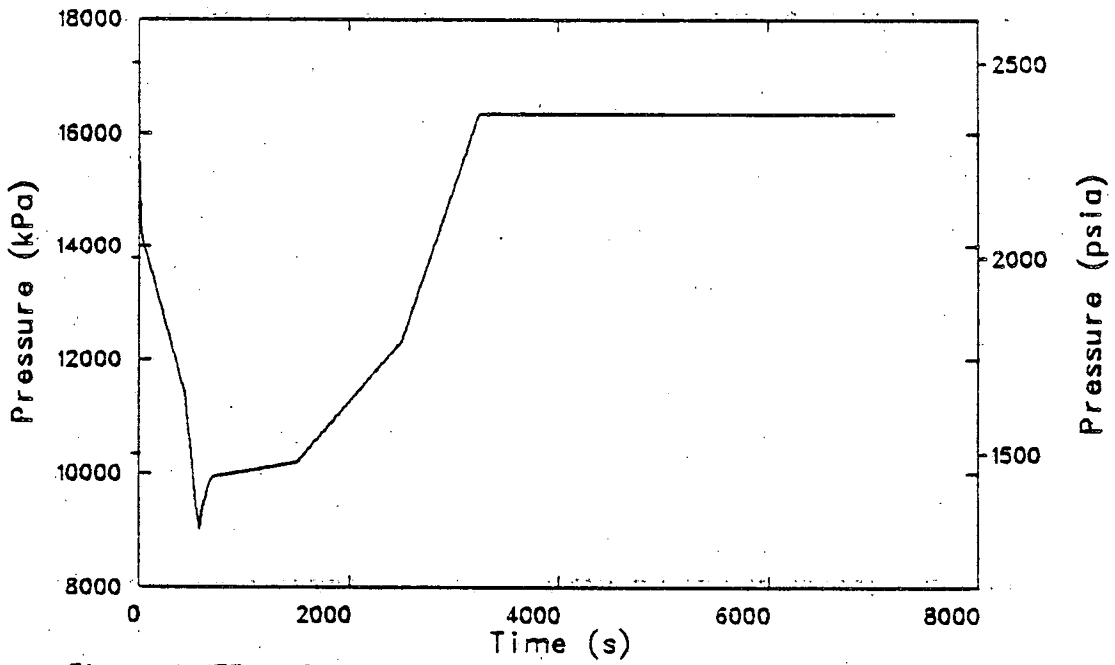

Figure A-173. Sequence 9-4 reactor vessel downcomer fluid pressure.

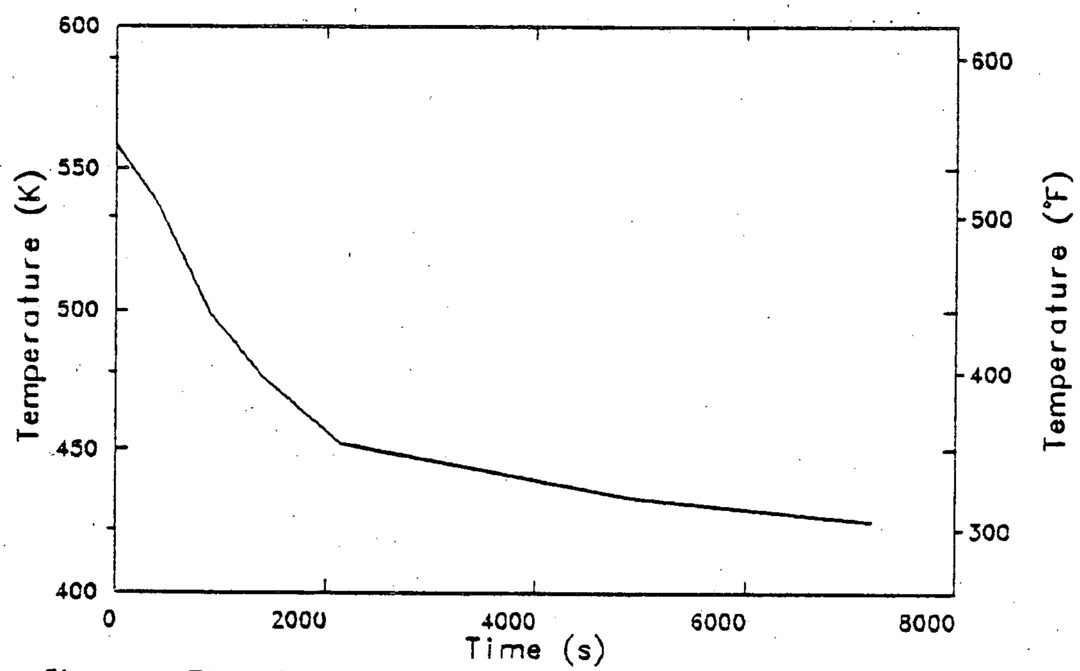

Figure A-174. Sequence 9-4 reactor vessel downcomer fluid temperature.

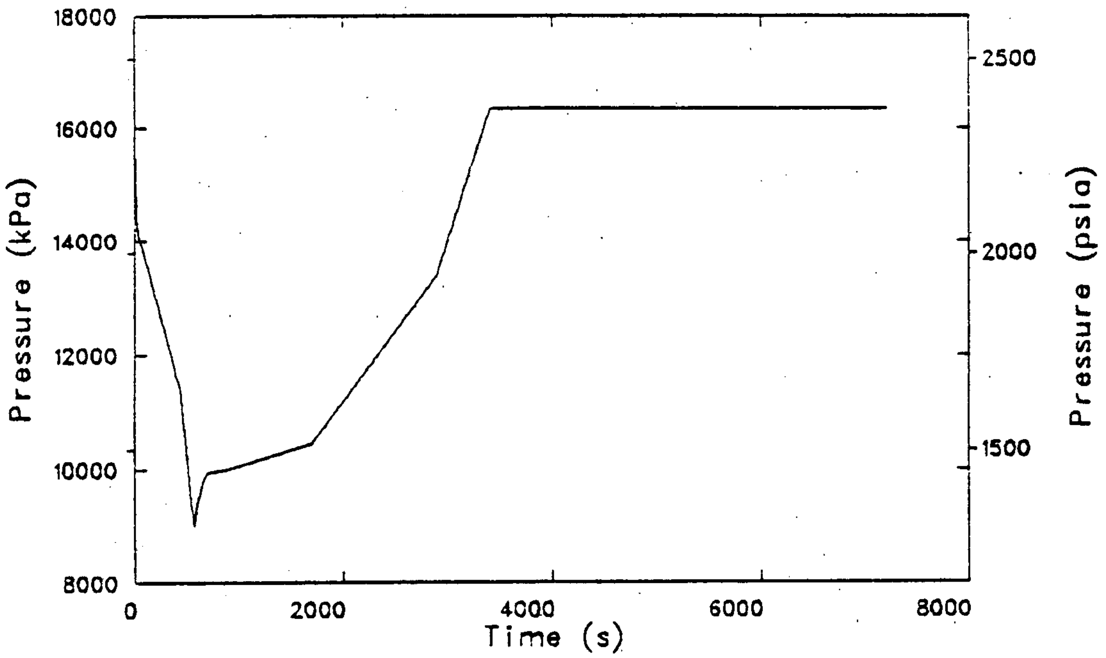

Figure A-175. Sequence 9-5 reactor vessel downcomer fluid pressure.

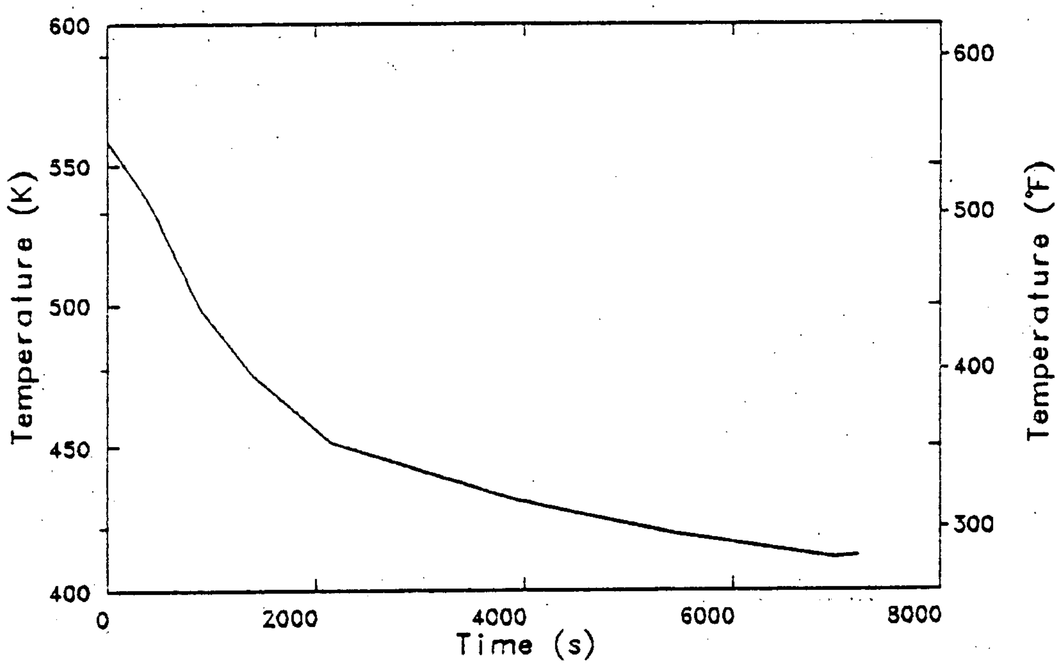

Figure A-176. Sequence 9-5 reactor vessel downcomer fluid temperature.

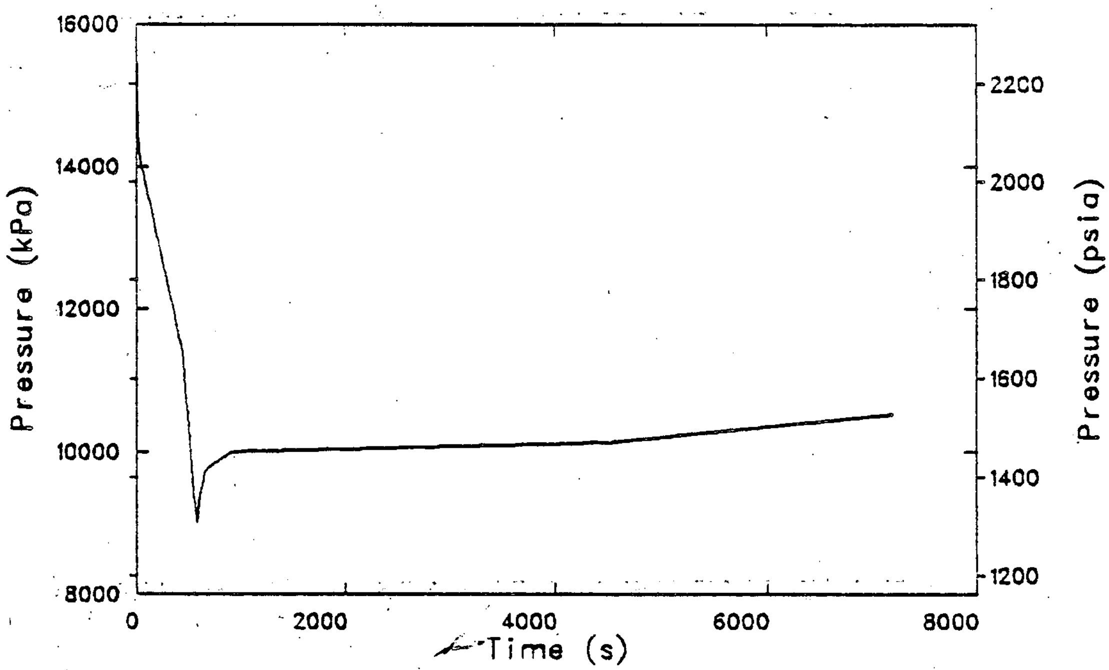

Figure A-177. Sequence 9-6 reactor vessel downcomer fluid pressure.

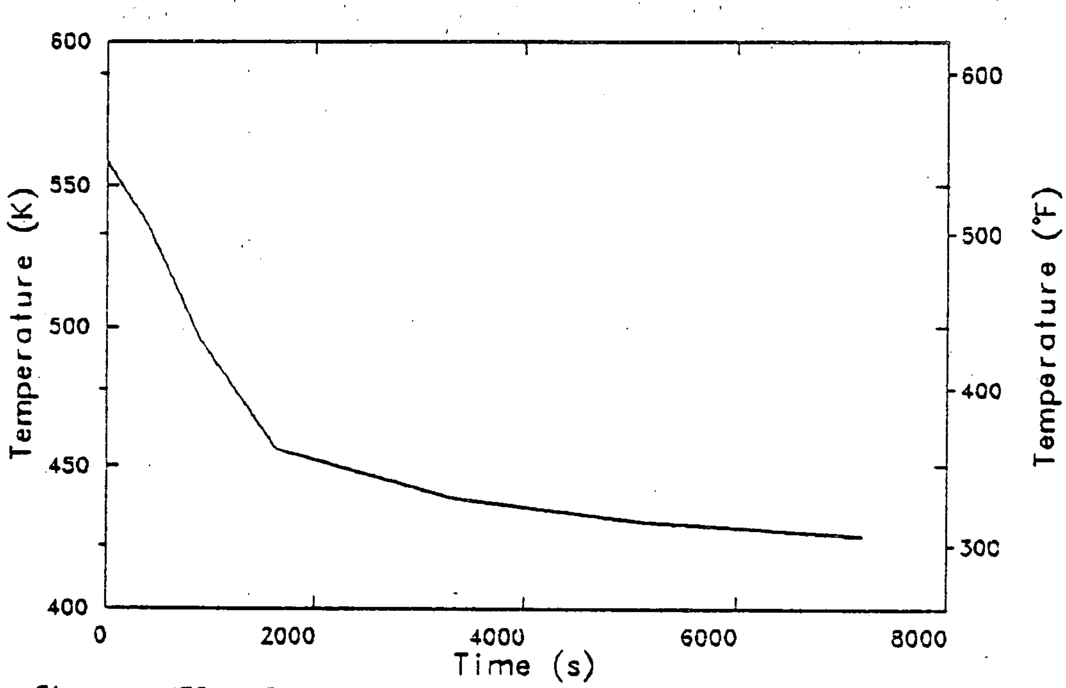

Figure A-178. Sequence 9-6 reactor vessel downcomer fluid temperature.

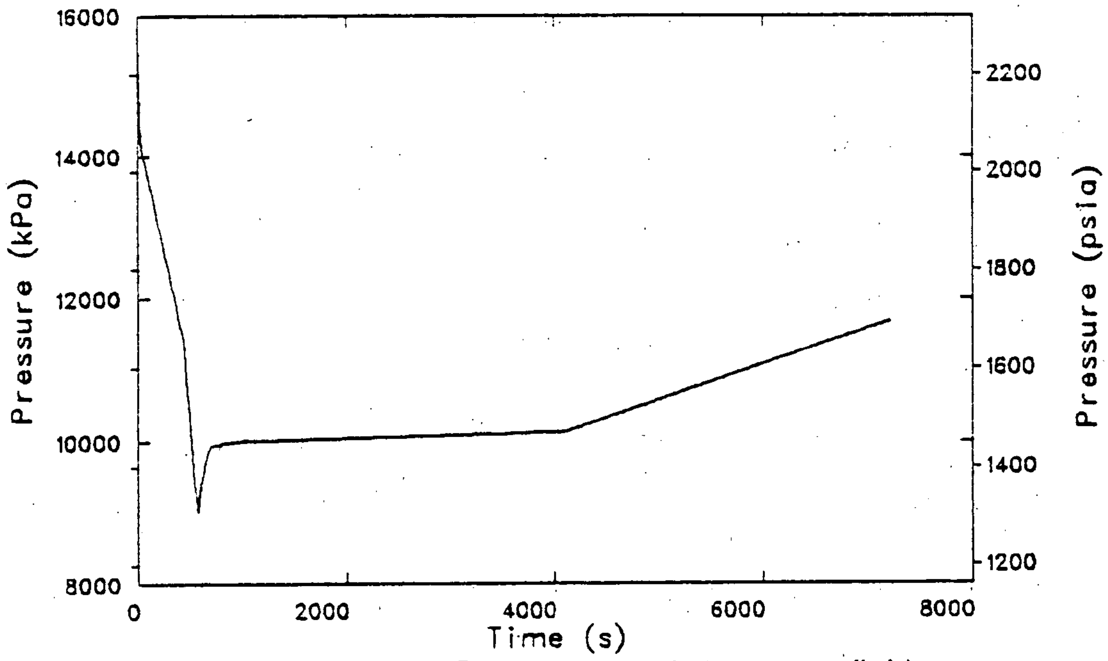

Figure A-179. Sequence 9-7 reactor vessel downcomer fluid pressure.

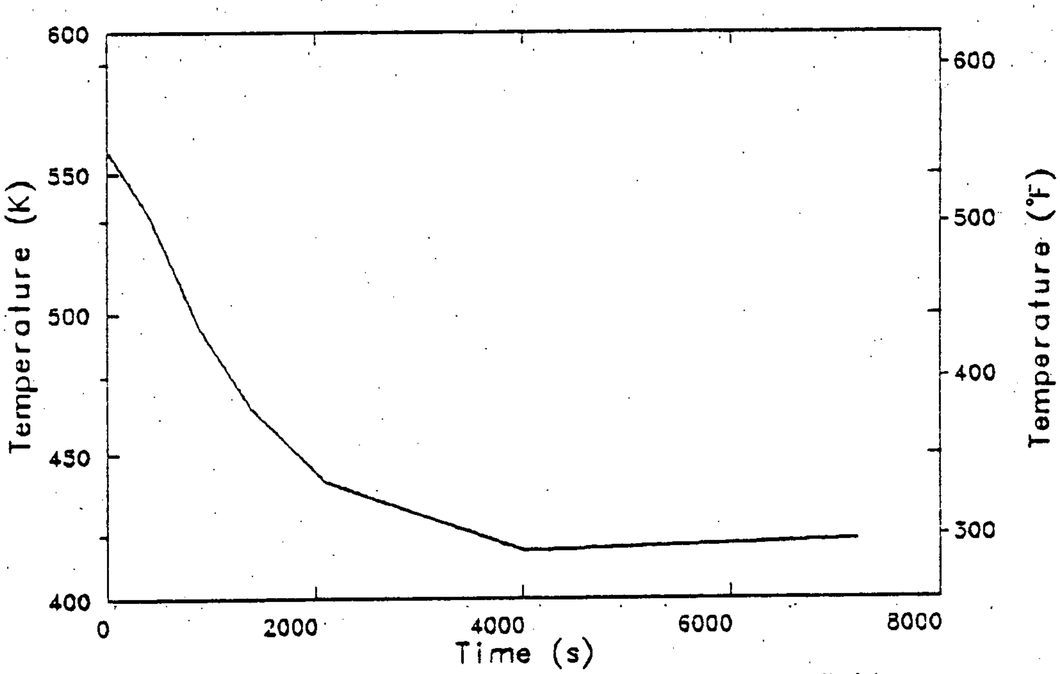

Figure A-180. Sequence 9-7 reactor vessel downcomer fluid temperature.

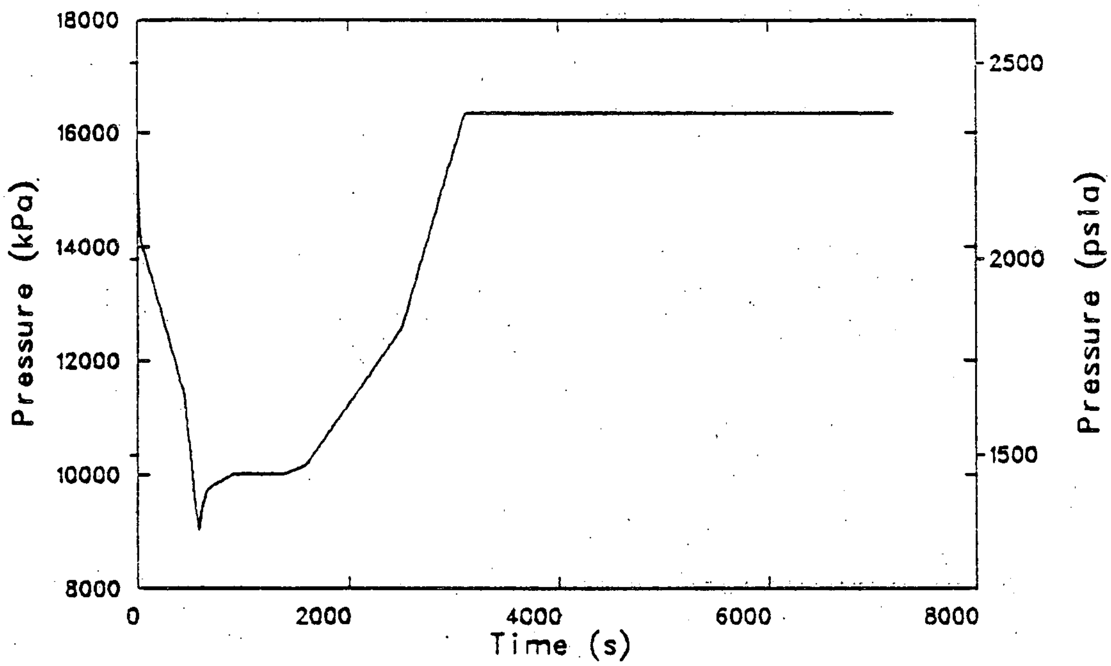

Figure A-181. Sequence 9-8 reactor vessel downcomer fluid pressure.

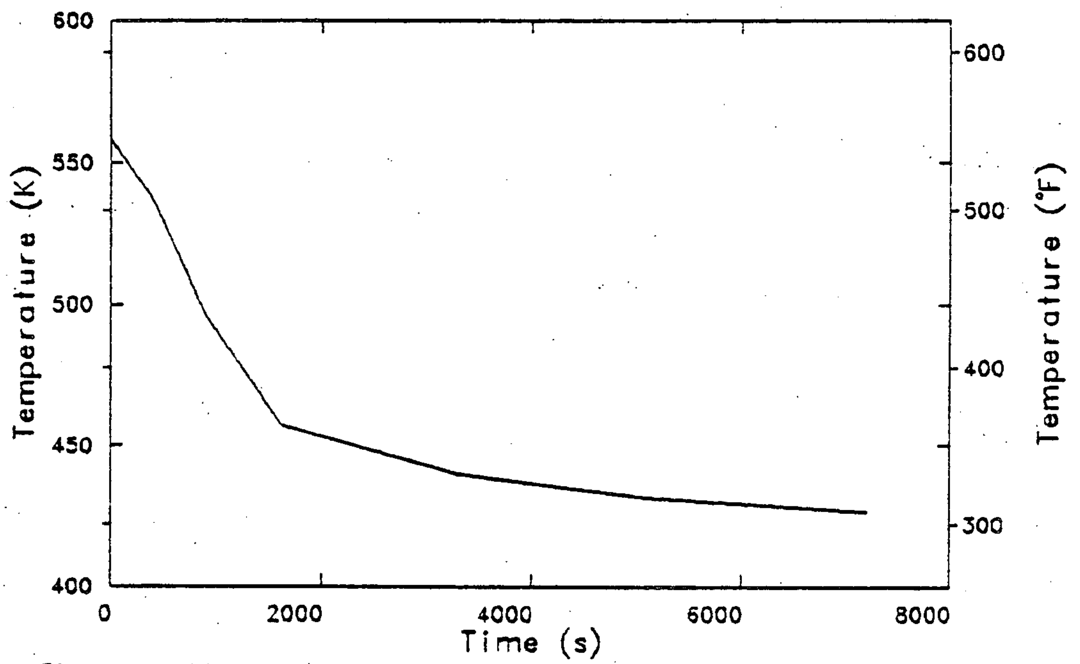

Figure A-182. Sequence 9-8 reactor vessel downcomer fluid temperature.

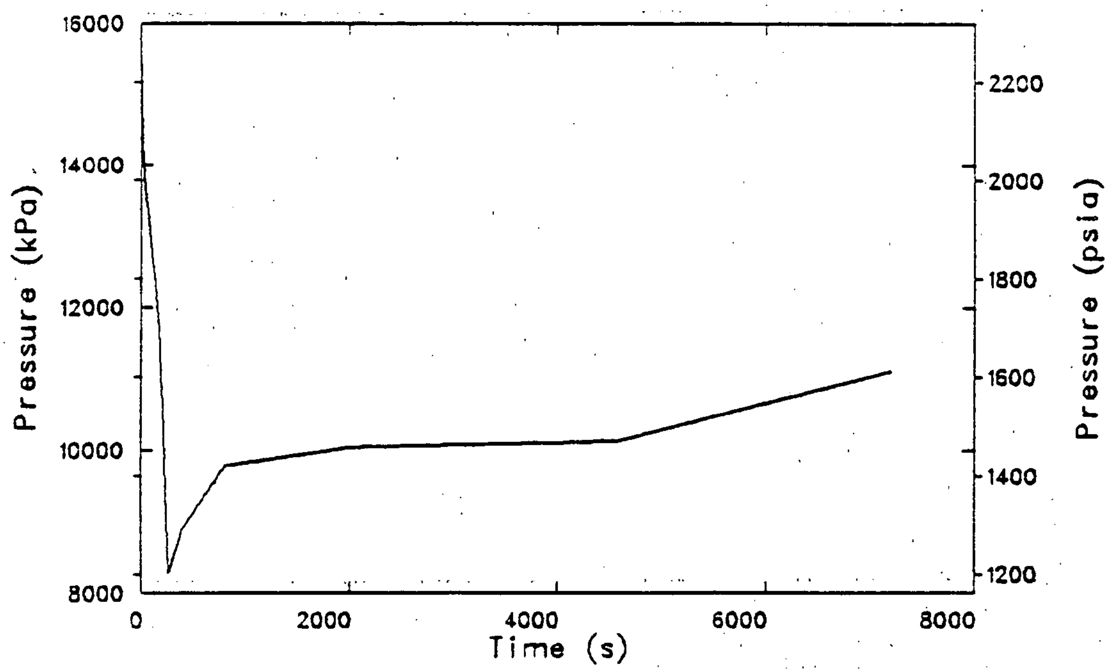

Figure A-183. Sequence 9-9 reactor vessel downcomer fluid pressure.

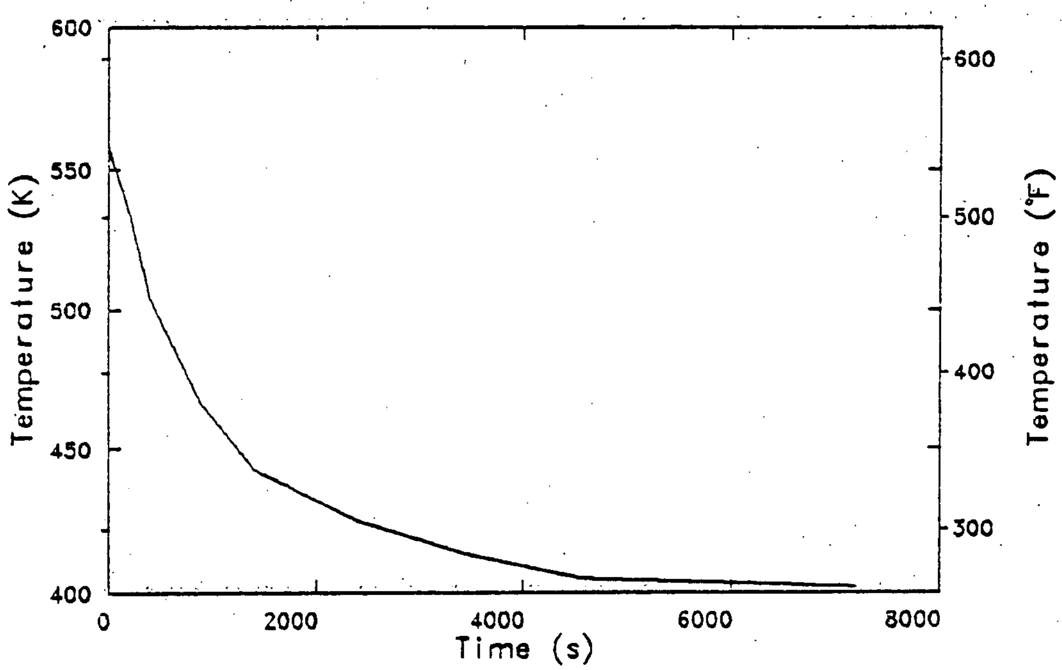

Figure A-184. Sequence 9-9 reactor vessel downcomer fluid temperature.

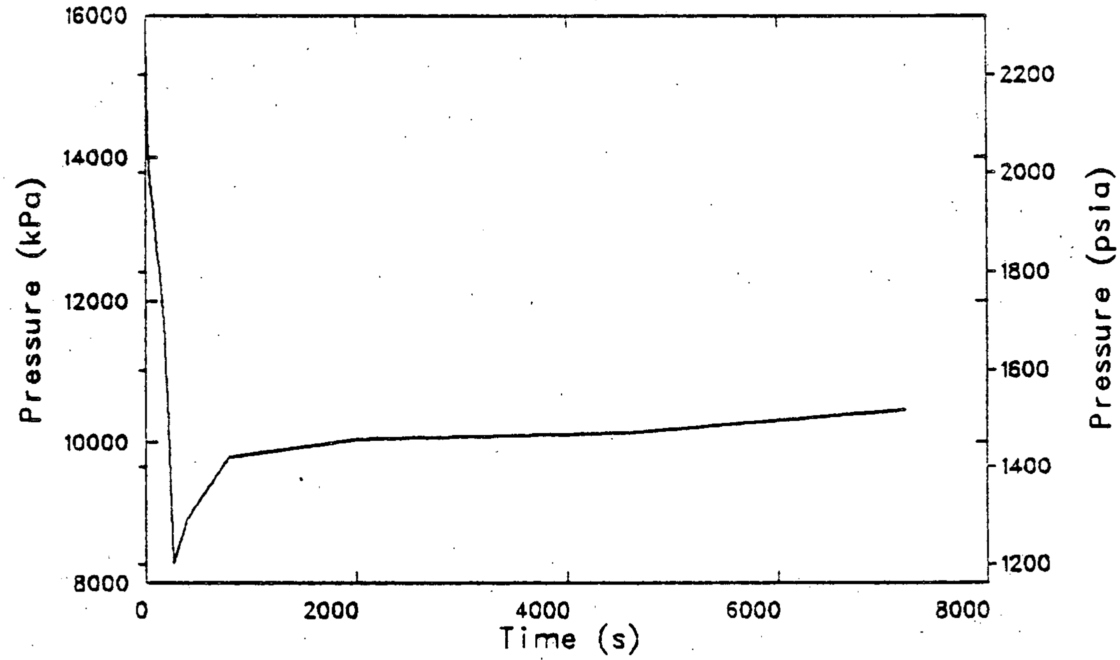

Figure A-185. Sequence 9–10 reactor vessel downcomer fluid pressure.

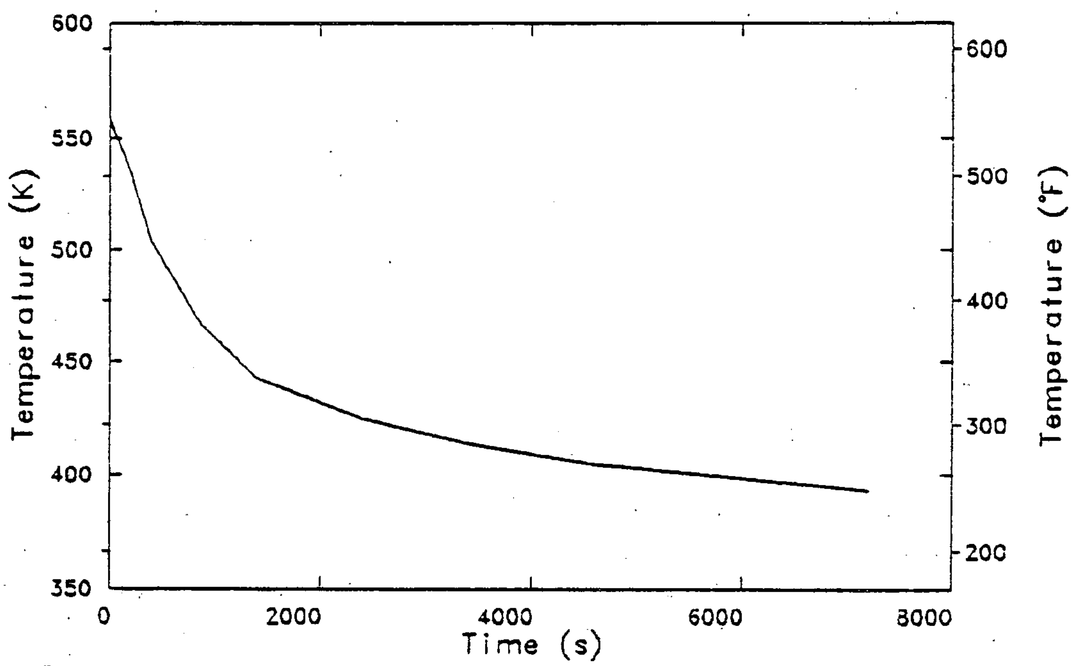

Figure A-186. Sequence 9–10 reactor vessel downcomer fluid temperature.

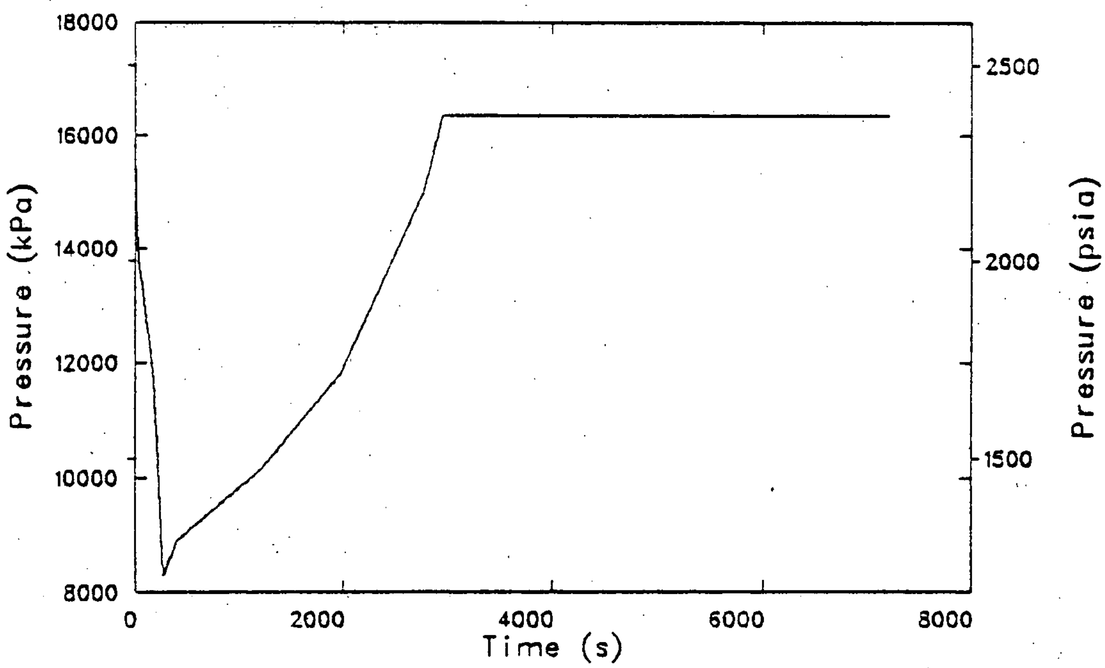

Figure A-187. Sequence 9-11 reactor vessel downcomer fluid pressure.

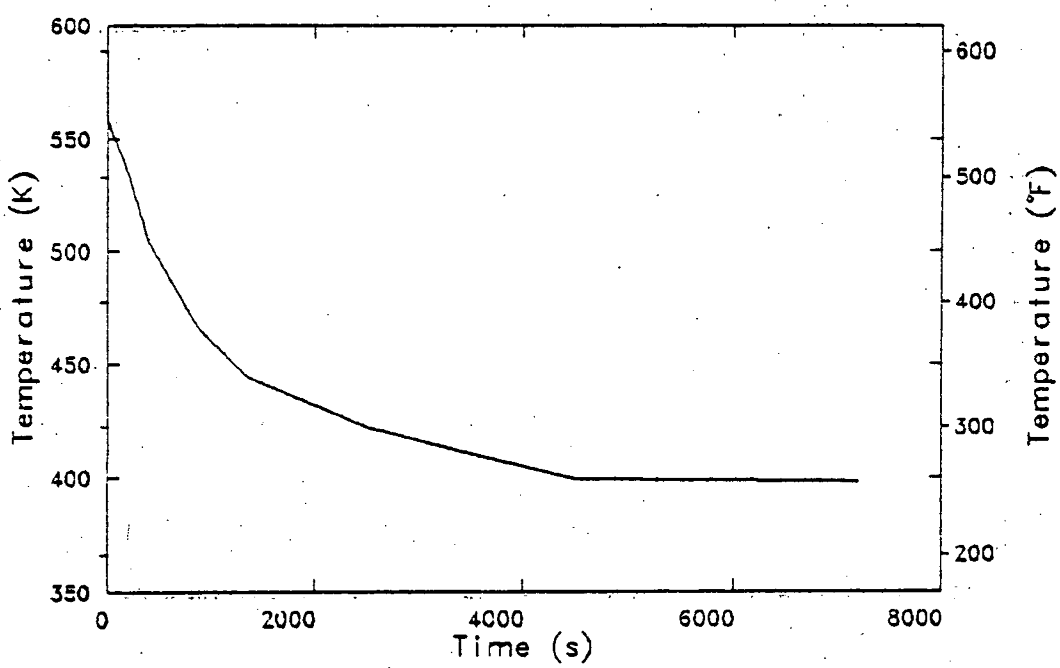

Figure A-188. Sequence 9-11 reactor vessel downcomer fluid temperature.

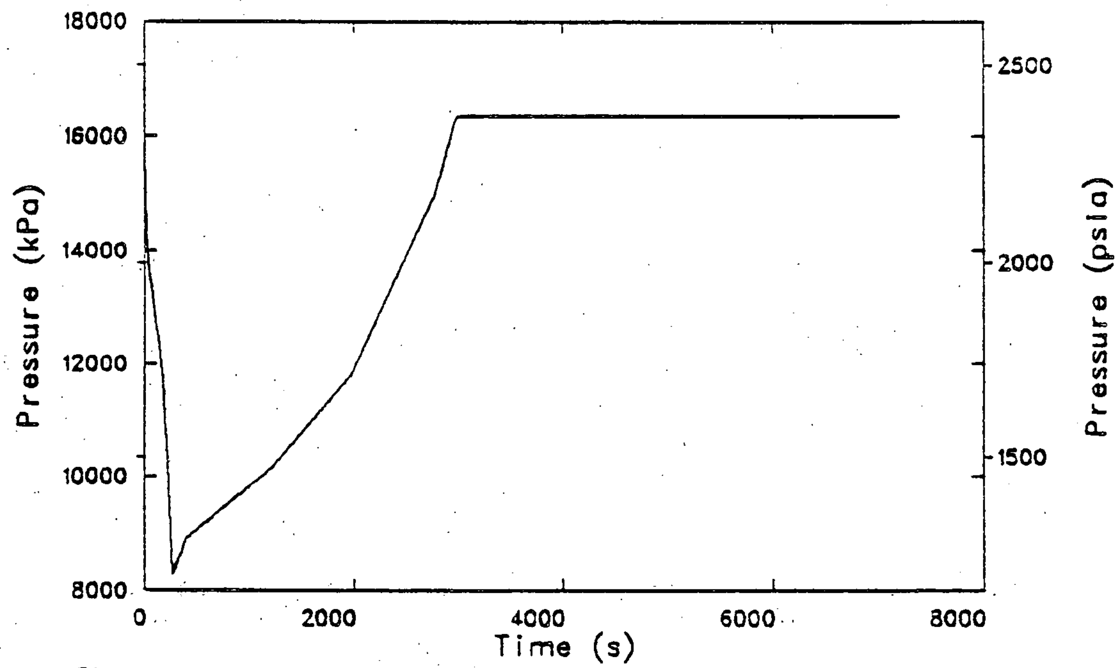

Figure A-189. Sequence 9-12 reactor vessel downcomer fluid pressure.

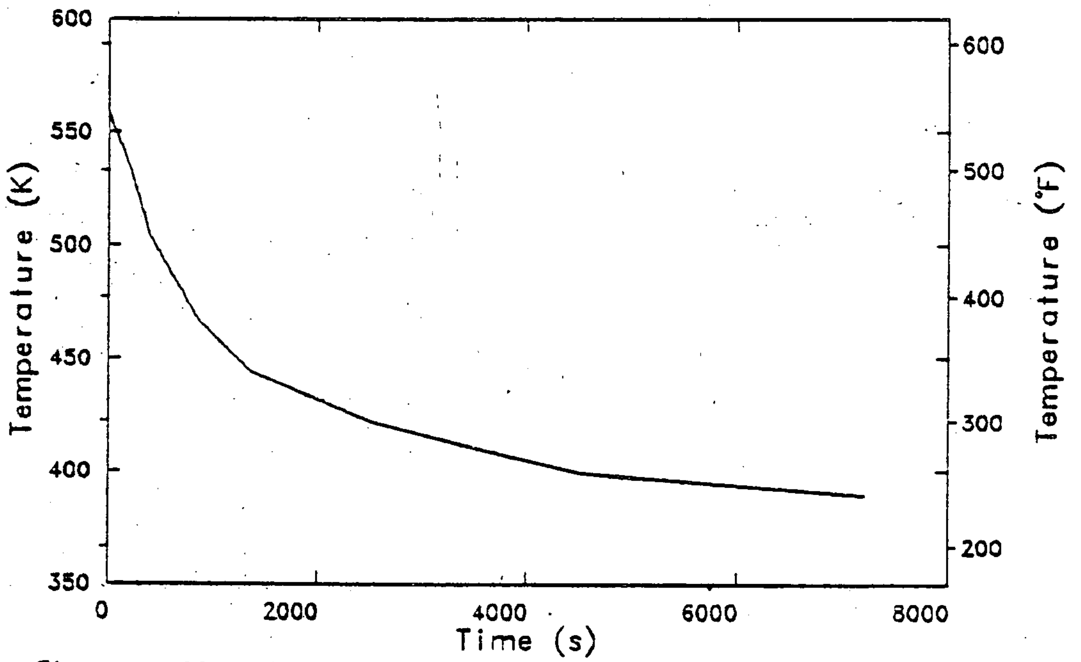

Figure A-190. Sequence 9-12 reactor vessel downcomer fluid temperature.

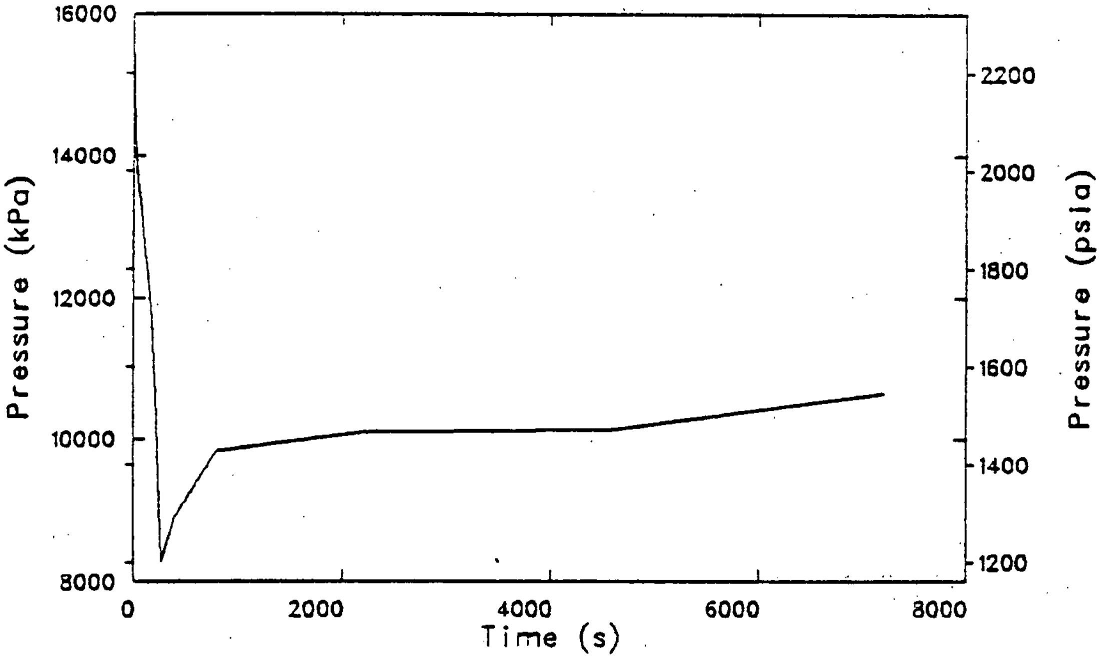

Figure A-191. Sequence 9-13 reactor vessel downcomer fluid pressure.

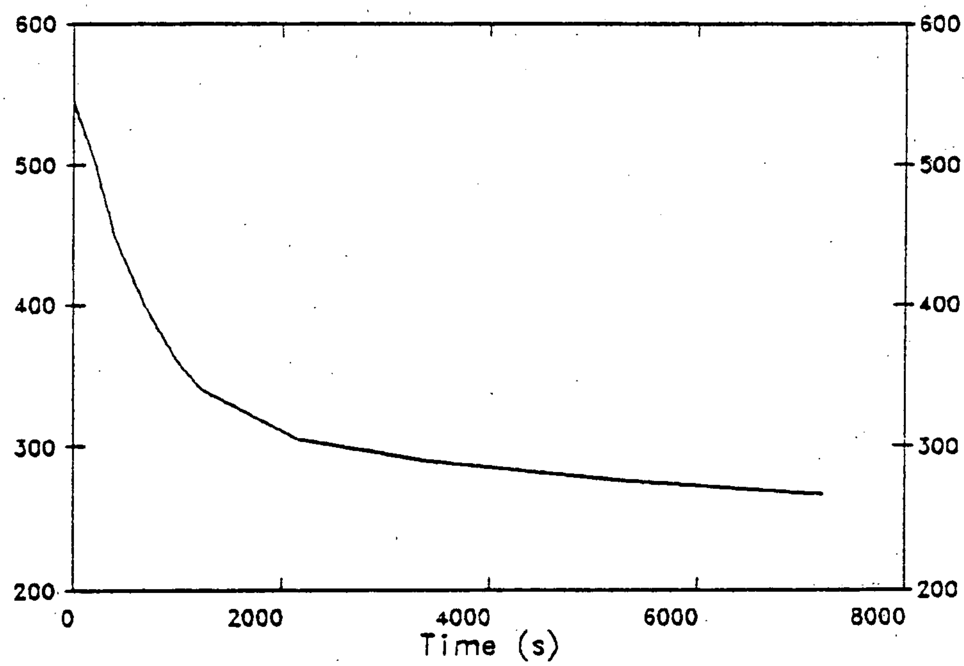

Figure A-192. Sequence 9-13 reactor vessel downcomer fluid temperature.

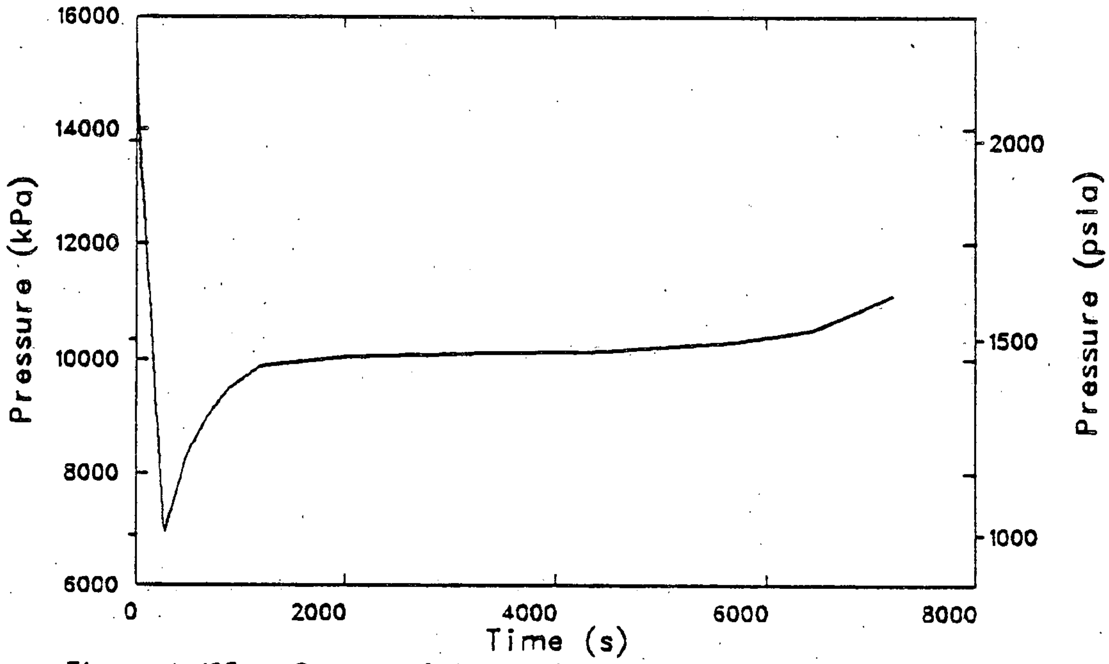

Figure A-193. Sequence 9-14 reactor vessel downcomer fluid pressure.

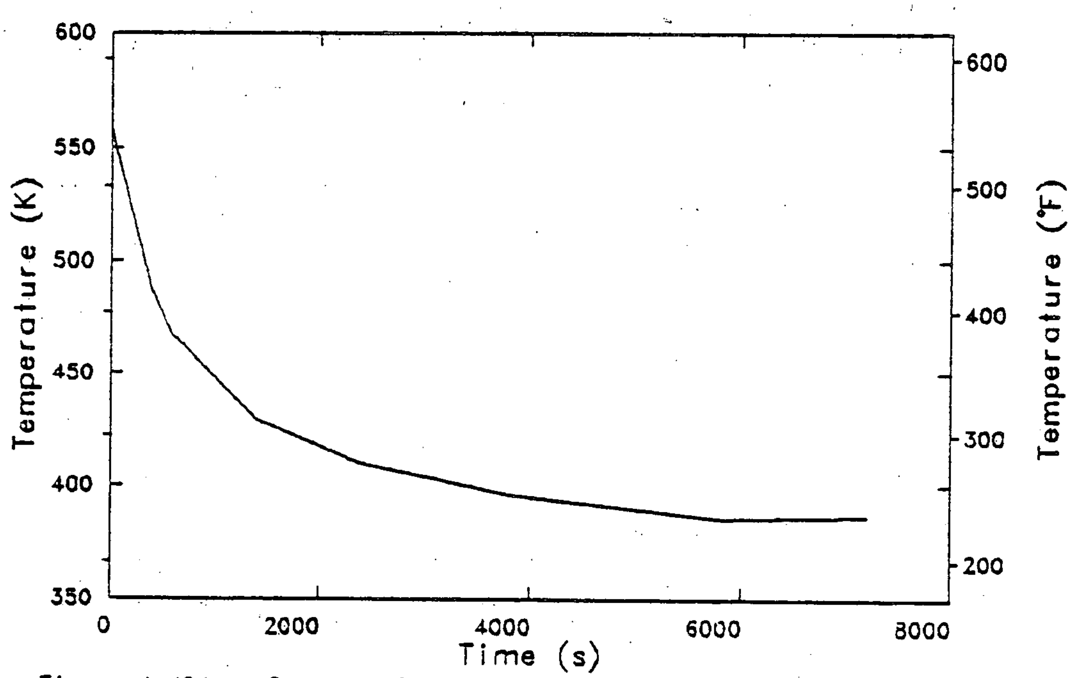

Figure A-194. Sequence 9-14 reactor vessel downcomer fluid temperature.

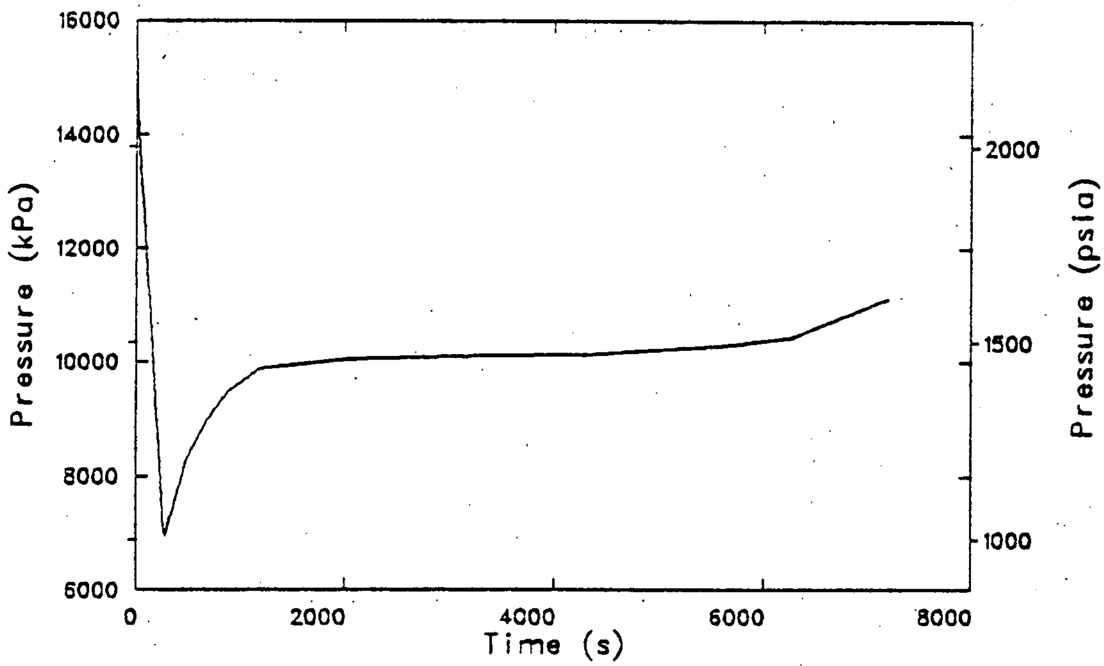

Figure A-195. Sequence 9-15 reactor vessel downcomer fluid pressure.

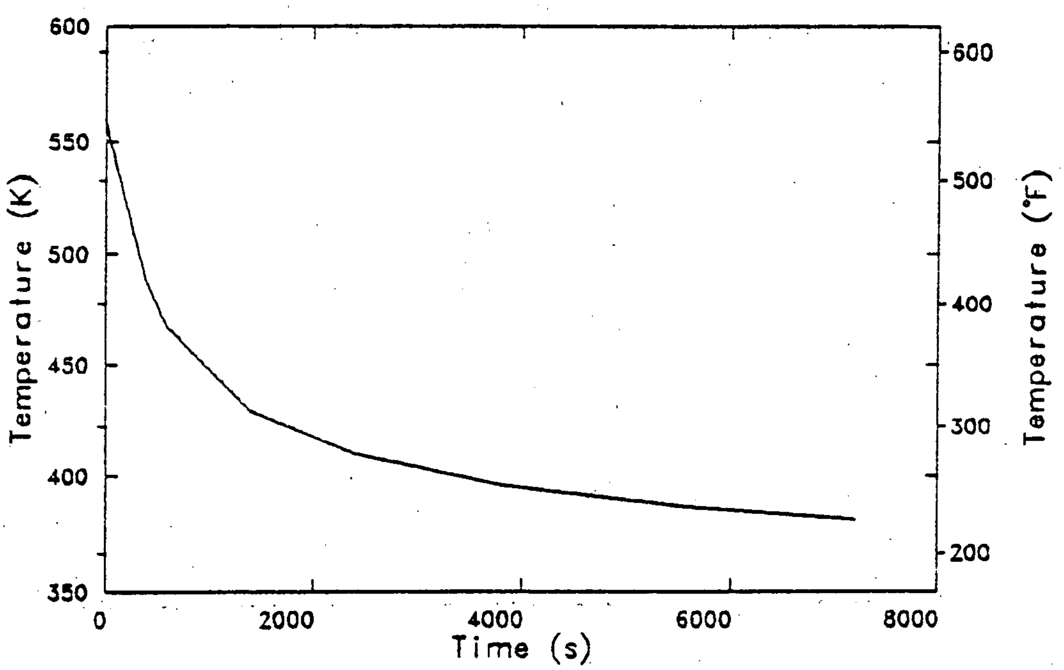

Figure A-196. Sequence 9-15 reactor vessel downcomer fluid temperature.

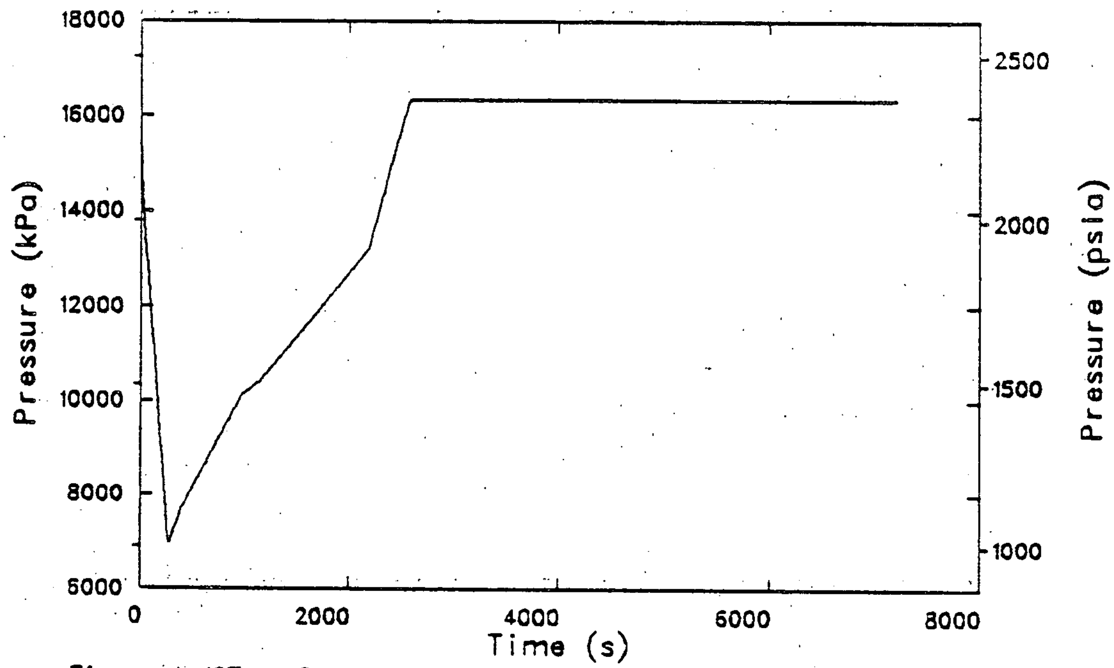

Figure A-197. Sequence 9-16 reactor vessel downcomer fluid pressure.

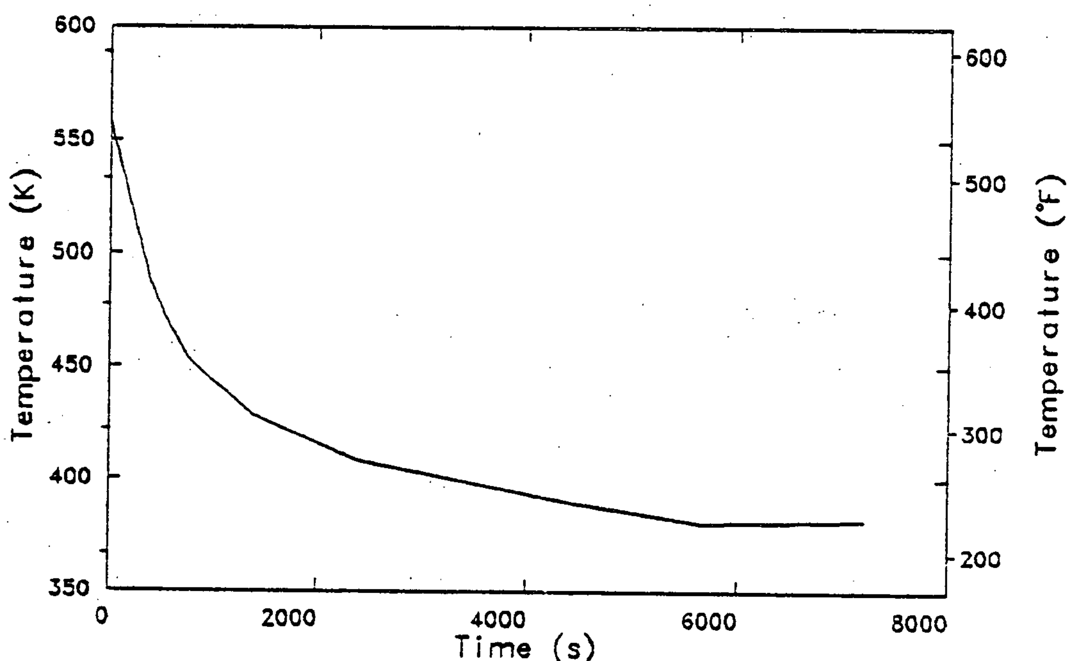

Figure A-198. Sequence 9-16 reactor vessel downcomer fluid temperature.

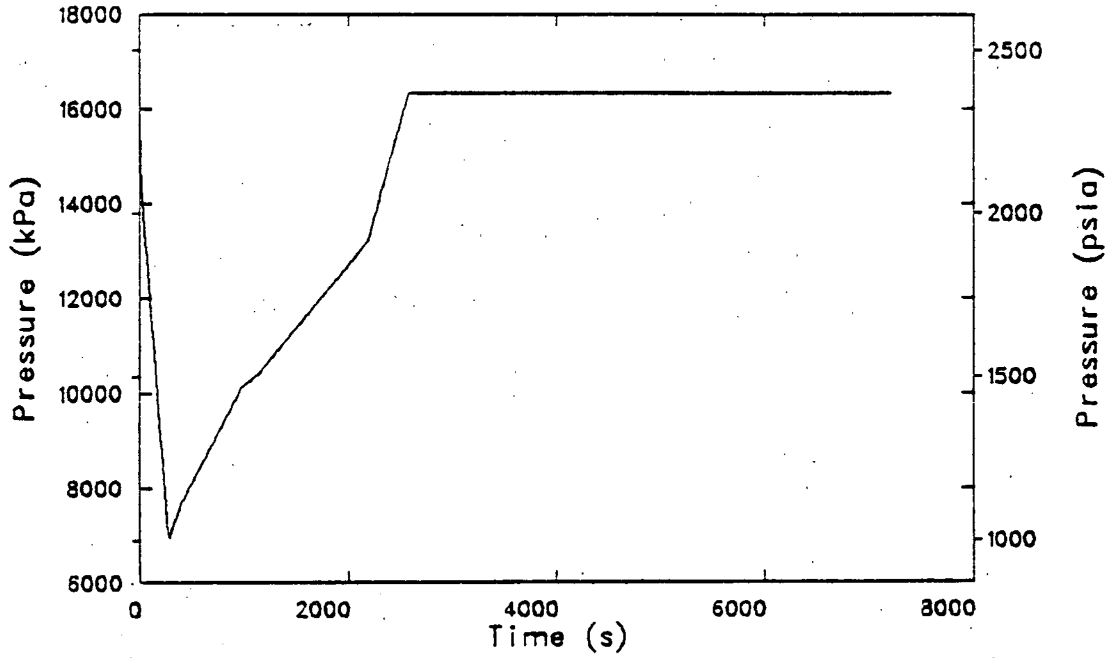

Figure A-199. Sequence 9-17 reactor vessel downcomer fluid pressure.

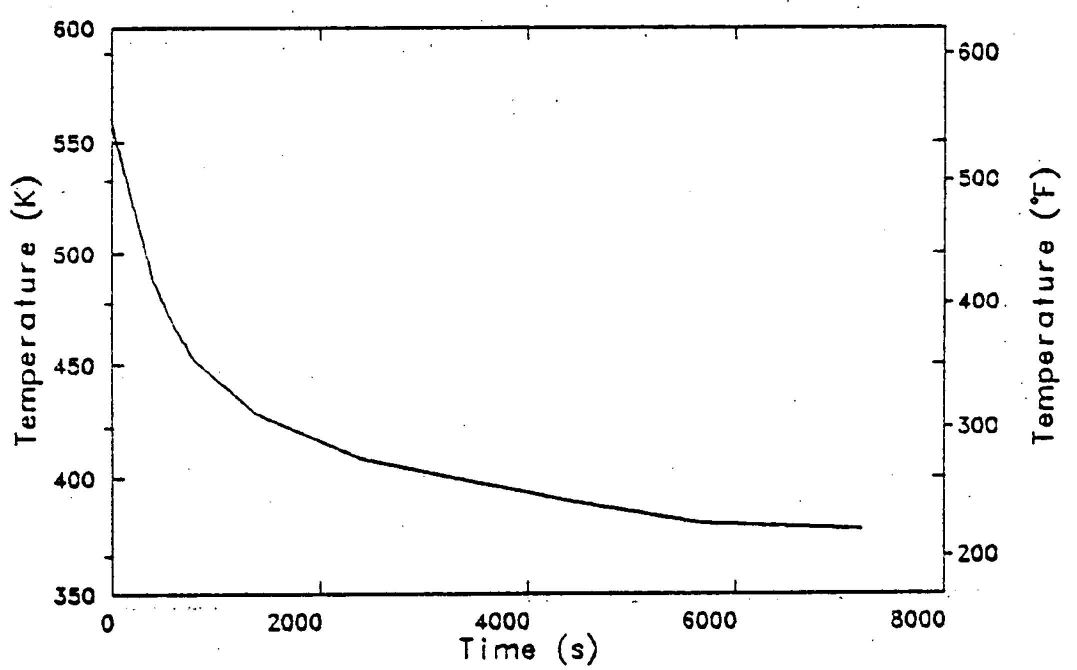

Figure A-200. Sequence 9-17 reactor vessel downcomer fluid temperature.

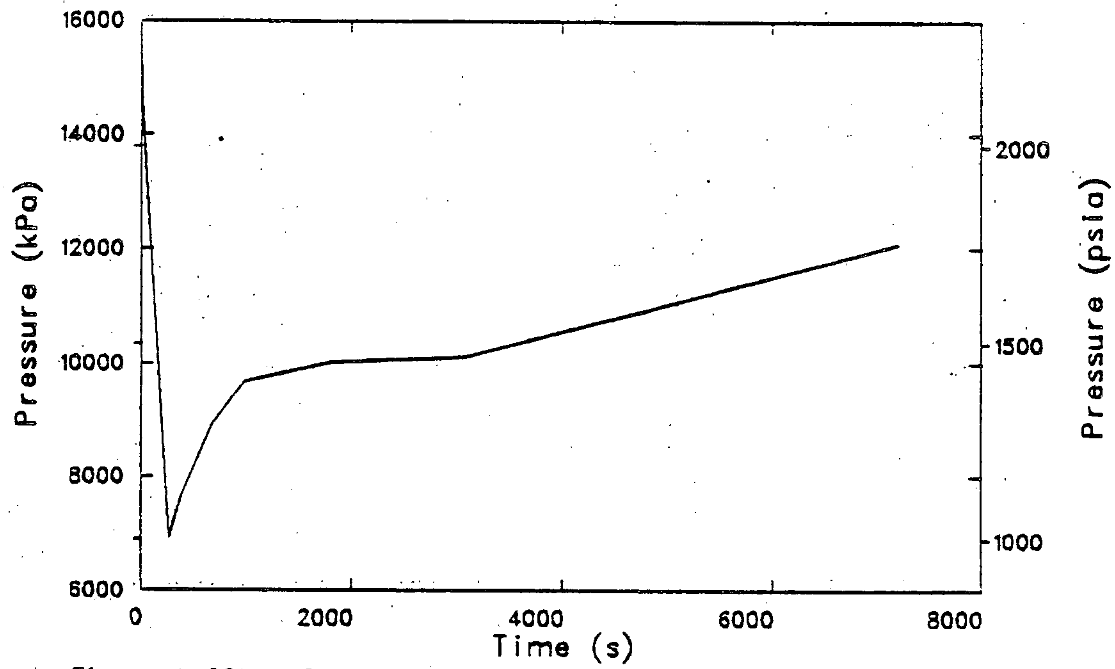

Figure A-201. Sequence 9-18 reactor vessel downcomer fluid pressure.

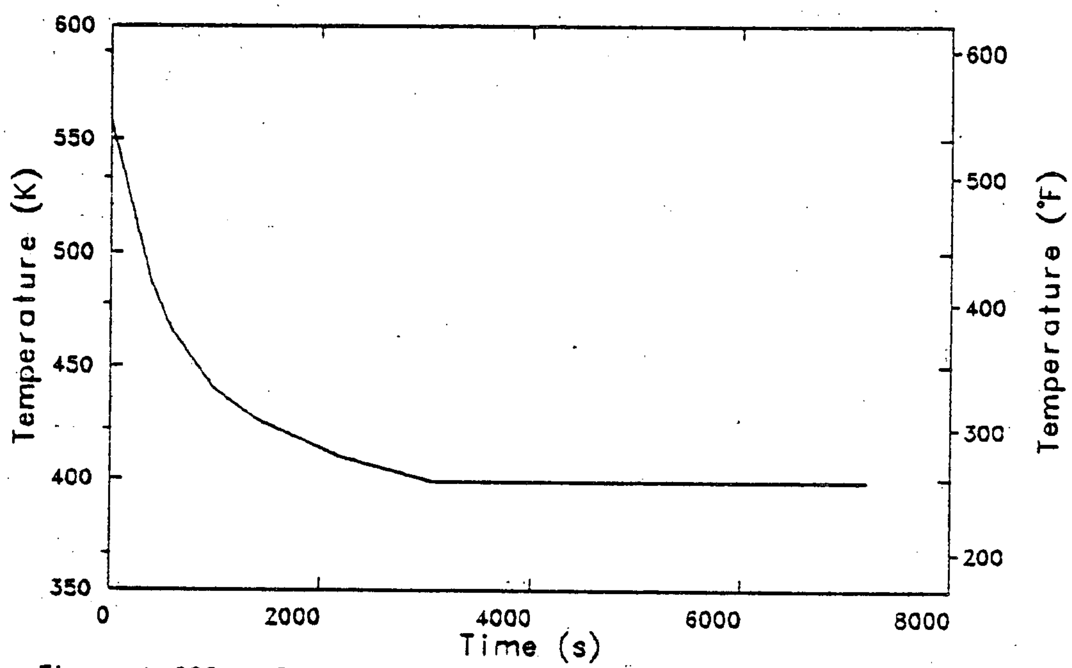

Figure A-202. Sequence 9-18 reactor vessel downcomer fluid temperature.

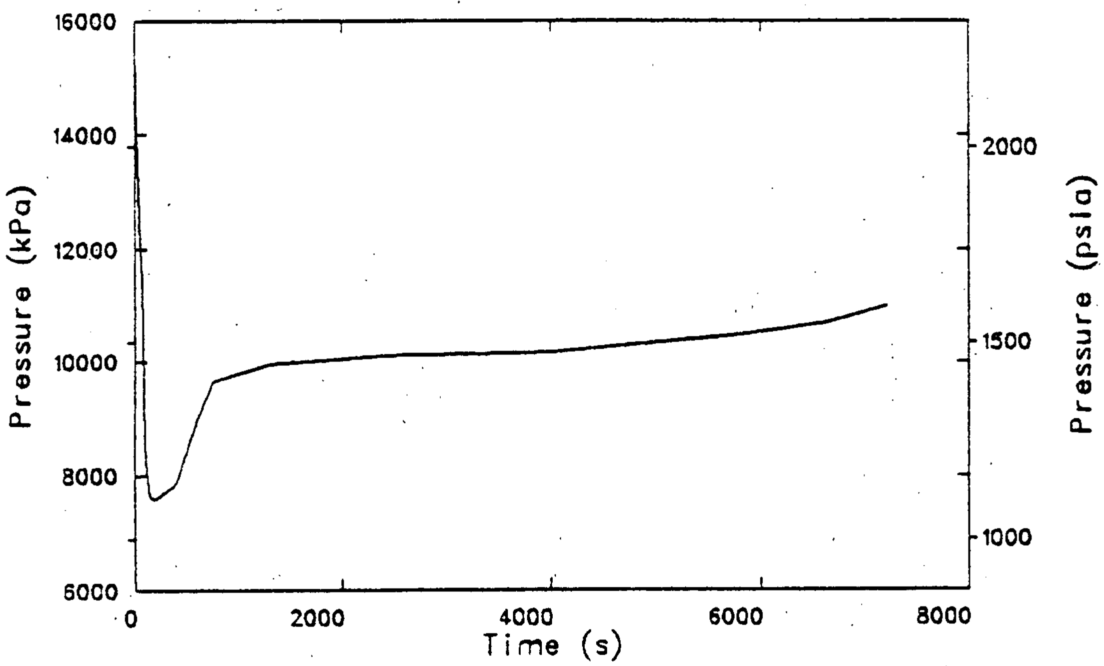

Figure A-203. Sequence 9-19 reactor vessel downcomer fluid pressure.

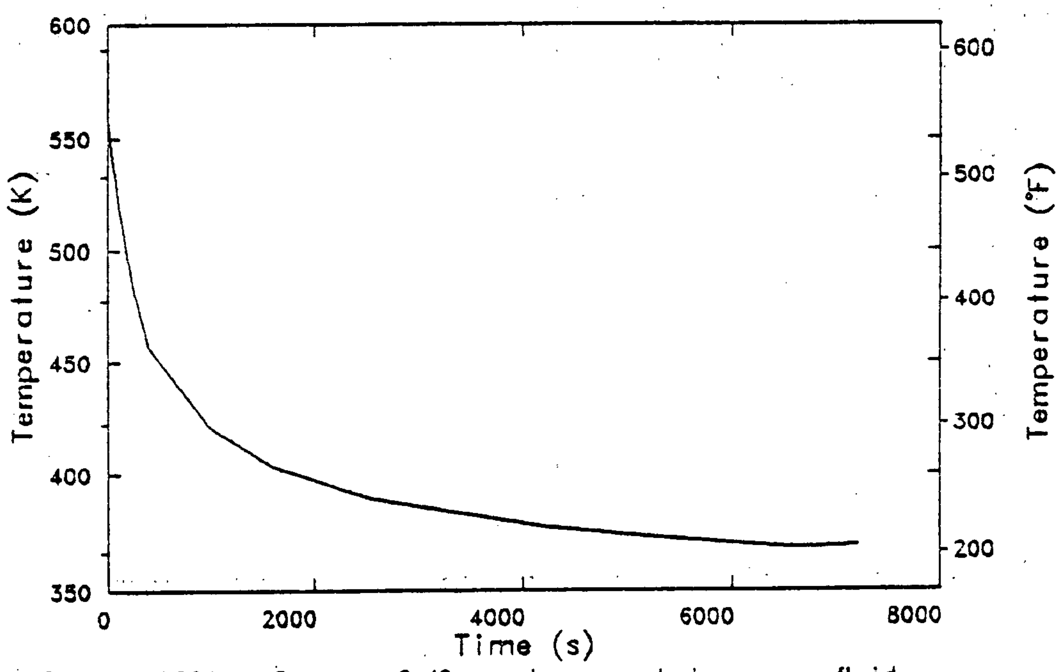

Figure A-204. Sequence 9-19 reactor vessel downcomer fluid temperature.

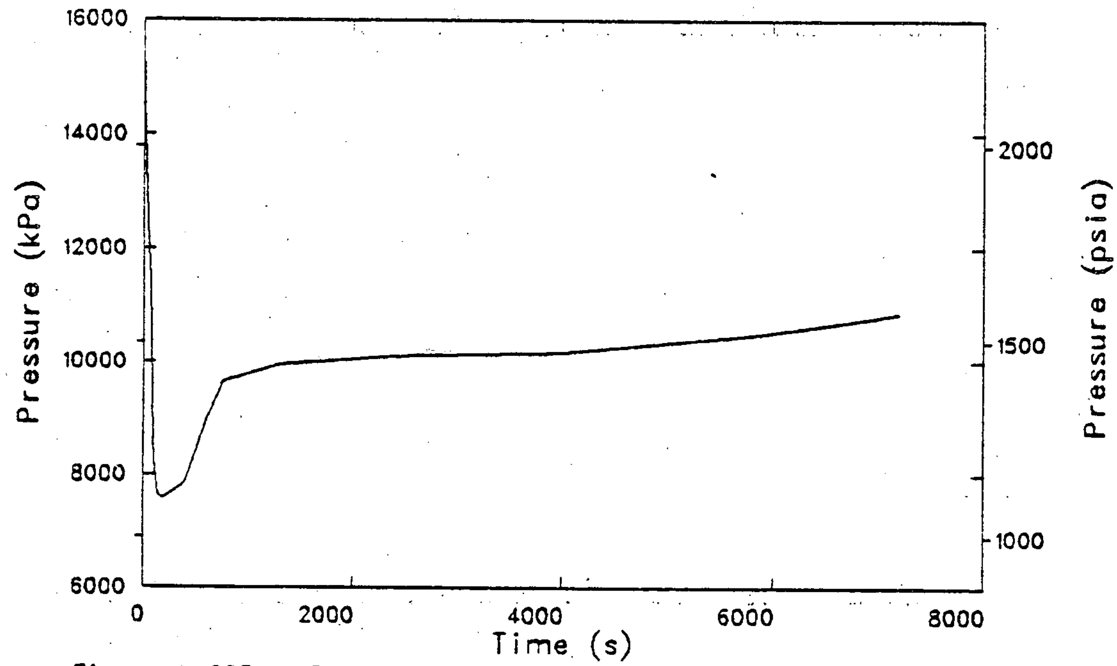

Figure A-205. Sequence 9-20 reactor vessel downcomer fluid pressure.

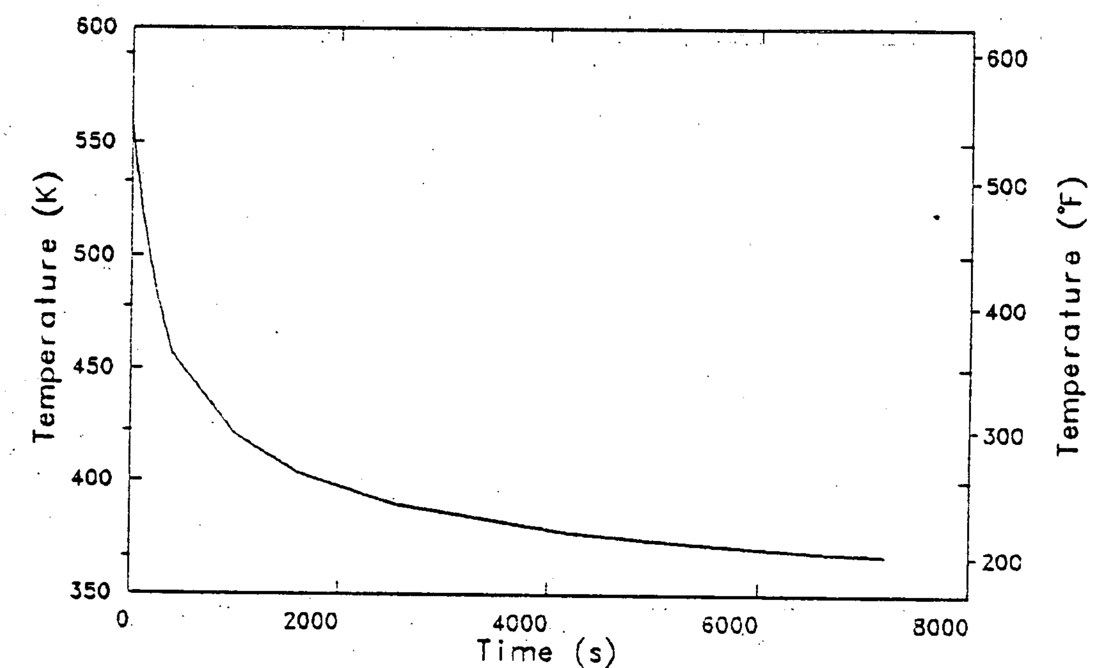

Figure A-206. Sequence 9-20 reactor vessel downcomer fluid temperature.

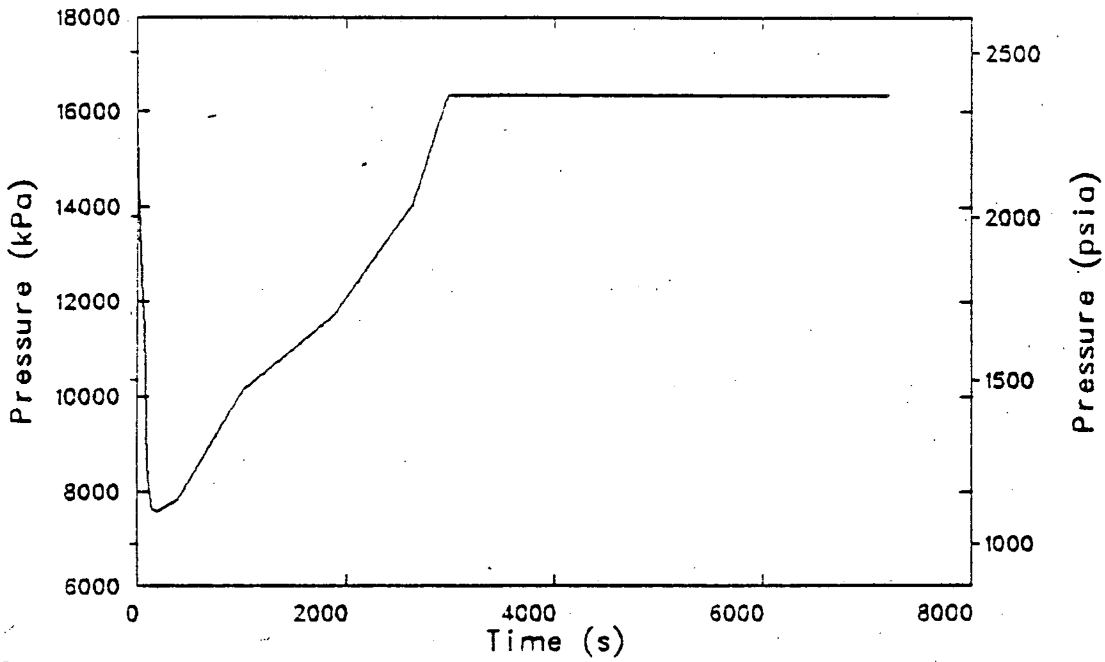

Figure A-207. Sequence 9-21 reactor vessel downcomer fluid pressure.

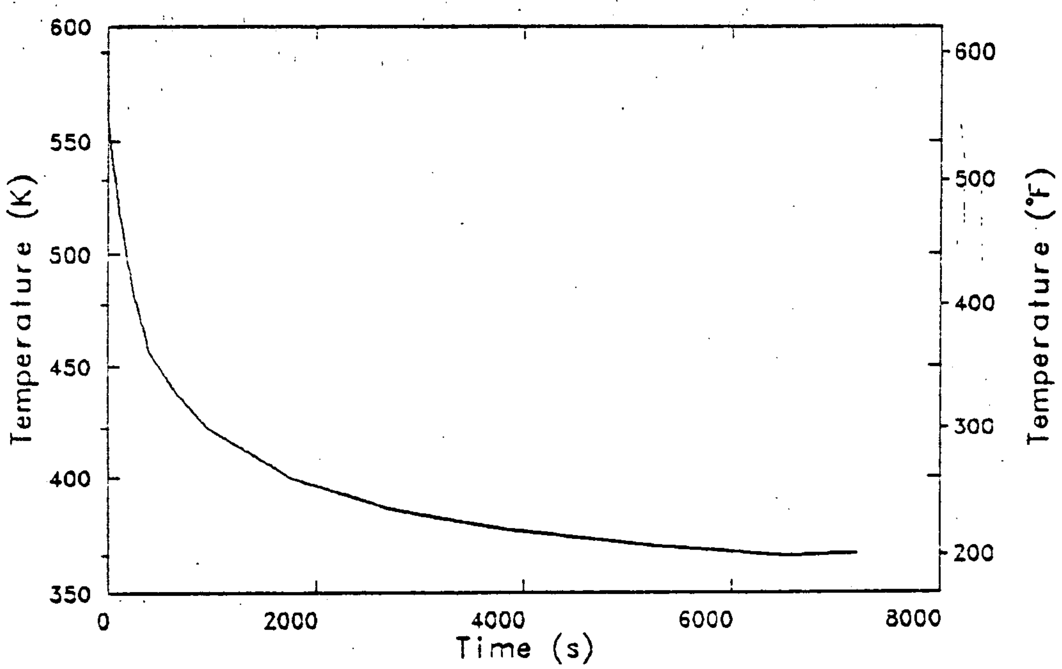

Figure A-208. Sequenca 9-21 reactor vessel downcomer fluid temperature.

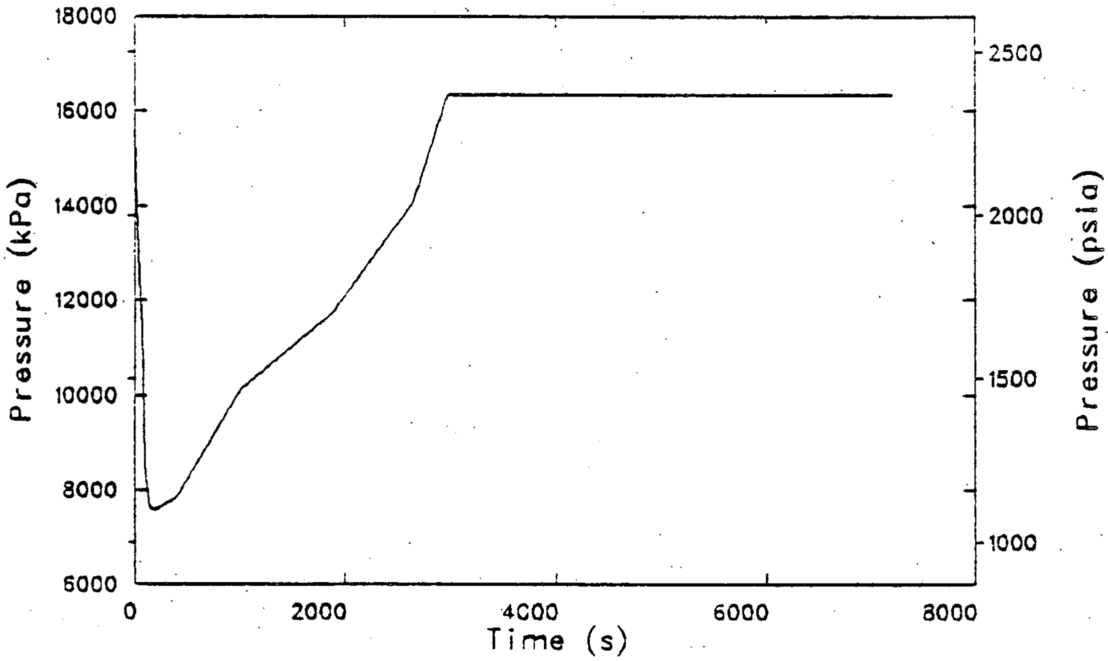

Figure A-209. Sequence 9-22 reactor vessel downcomer fluid pressure.

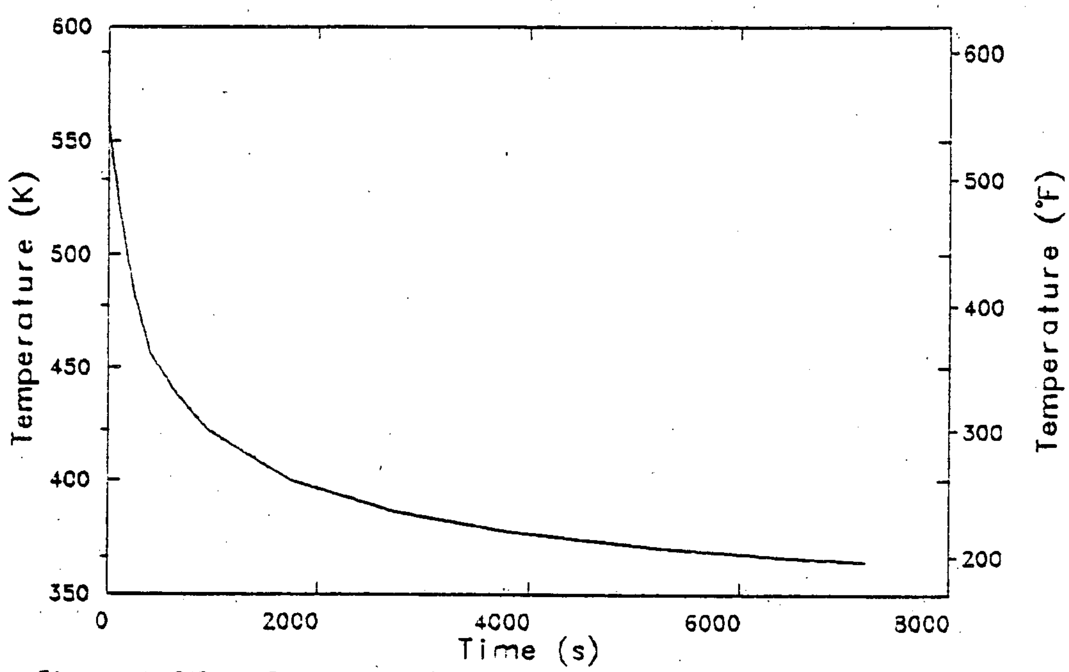

Figure A-210. Sequence 9-22 reactor vessel downcomer fluid temperature.

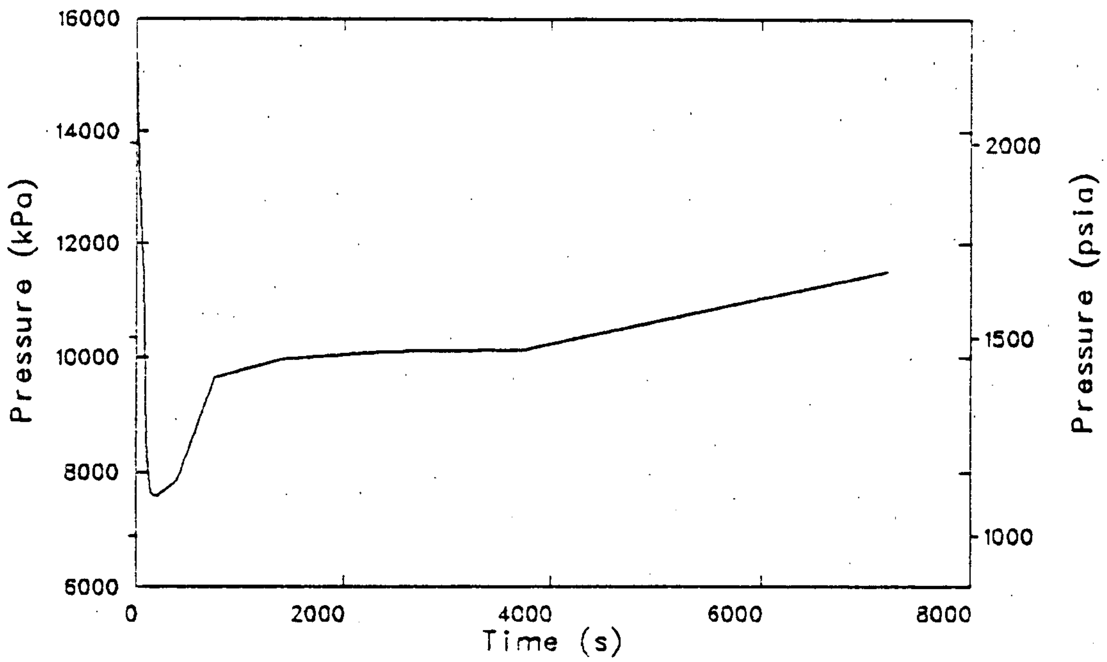

Figure A-211. Sequence 9-23 reactor vessel downcomer fluid pressure.

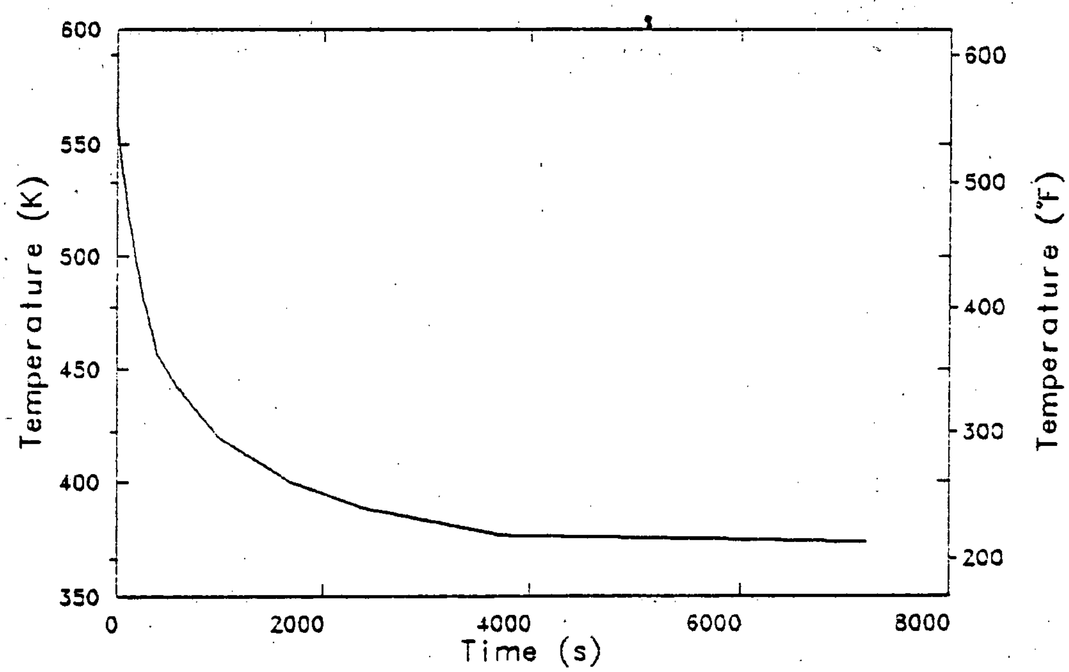

Figure A-212. Sequence 9-23 reactor vessel downcomer fluid temperature.

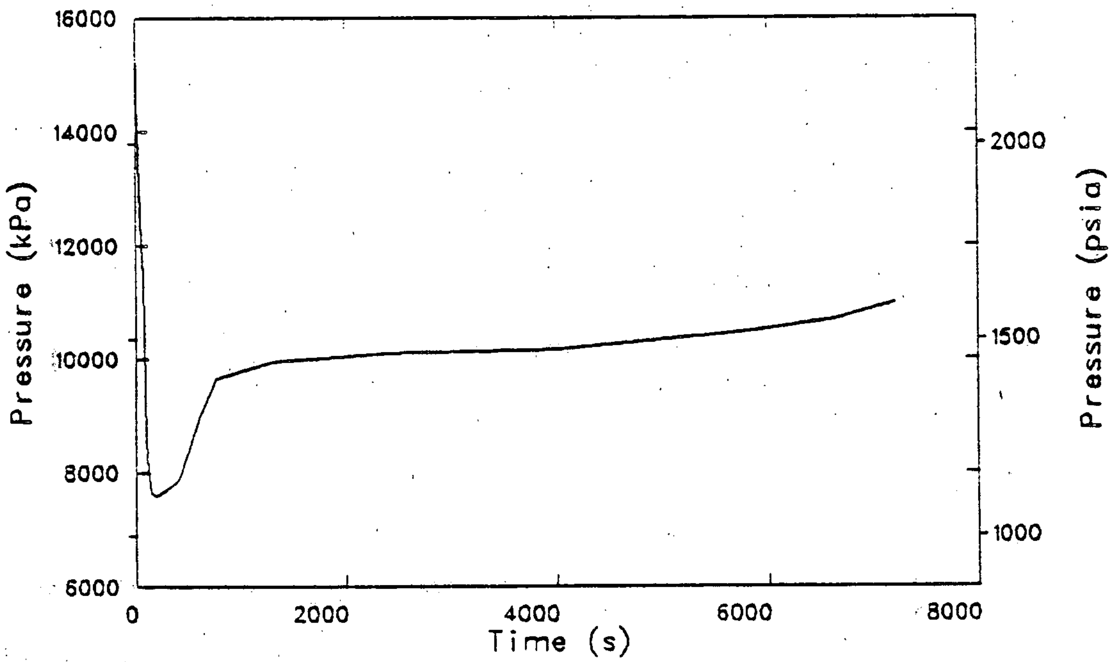

Figure A-213. Sequence 9-24 reactor vessel downcomer fluid pressure.

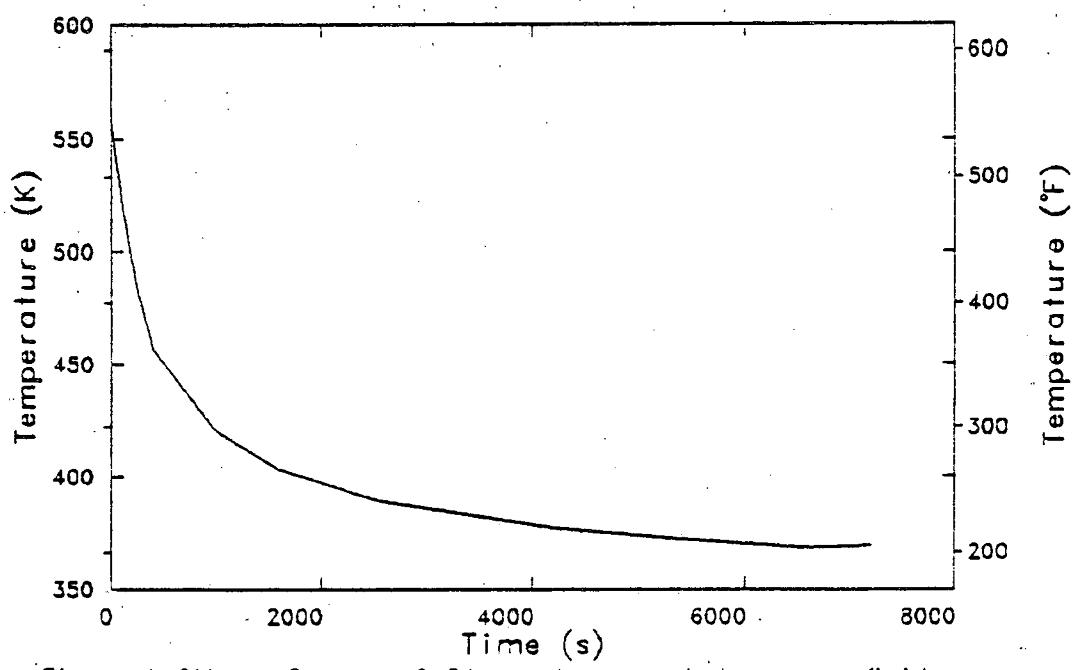

Figure A-214. Sequence 9-24 reactor vessel downcomer fluid temperature.

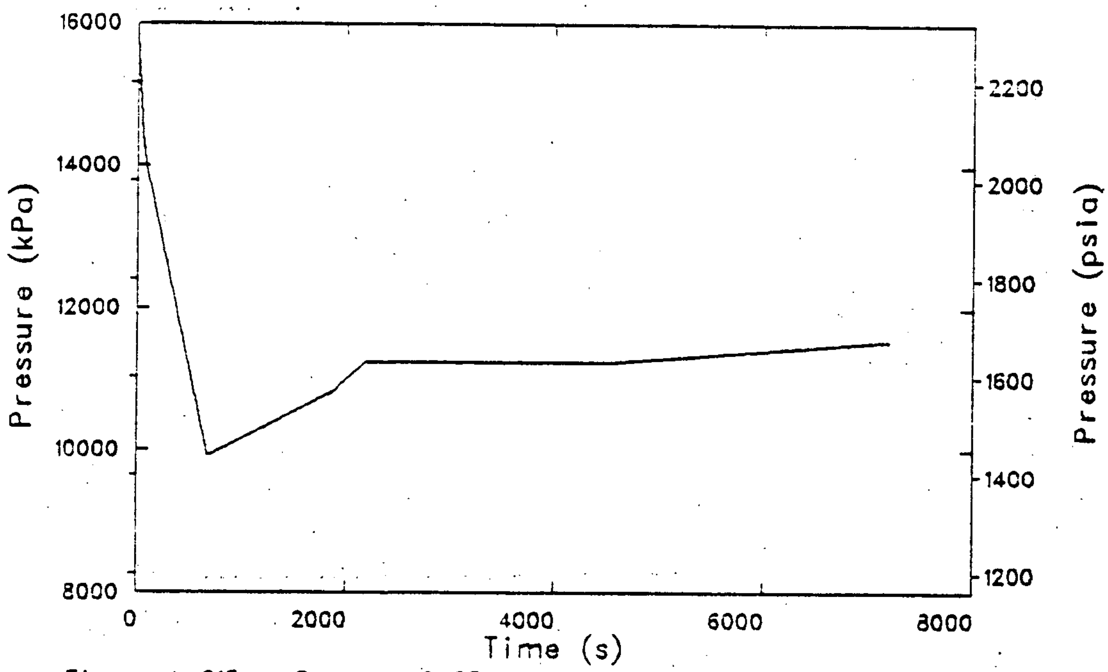

Figure A-215. Sequence 9-25 reactor vessel downcomer fluid pressure.

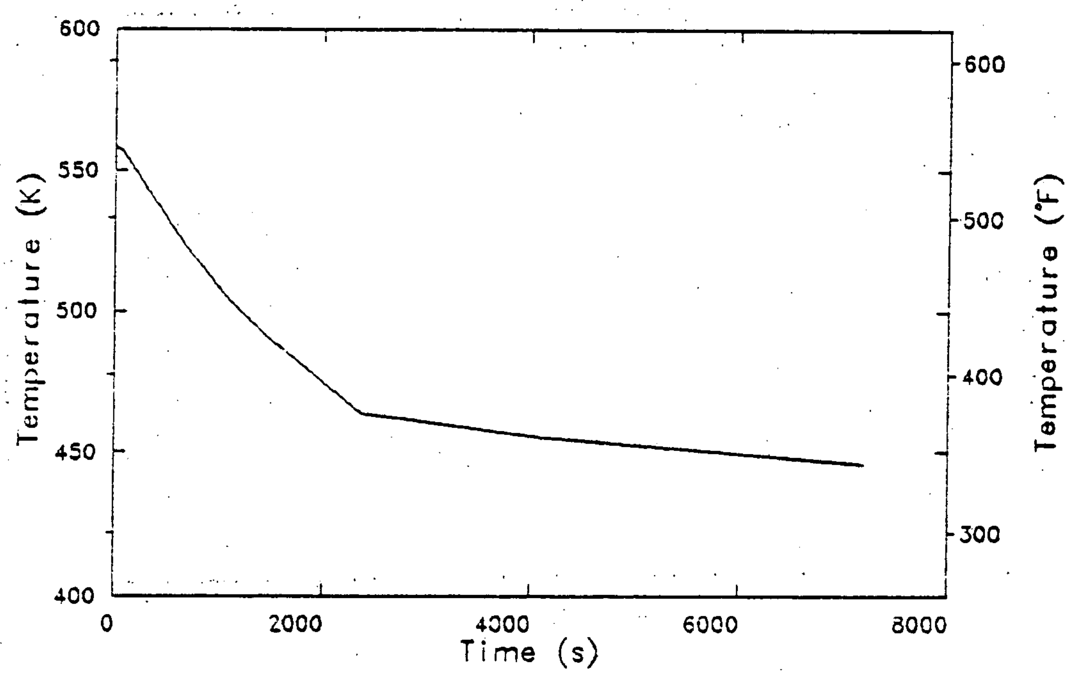

Figure A-216. Sequence 9-25 reactor vessel downcomer fluid temperature.

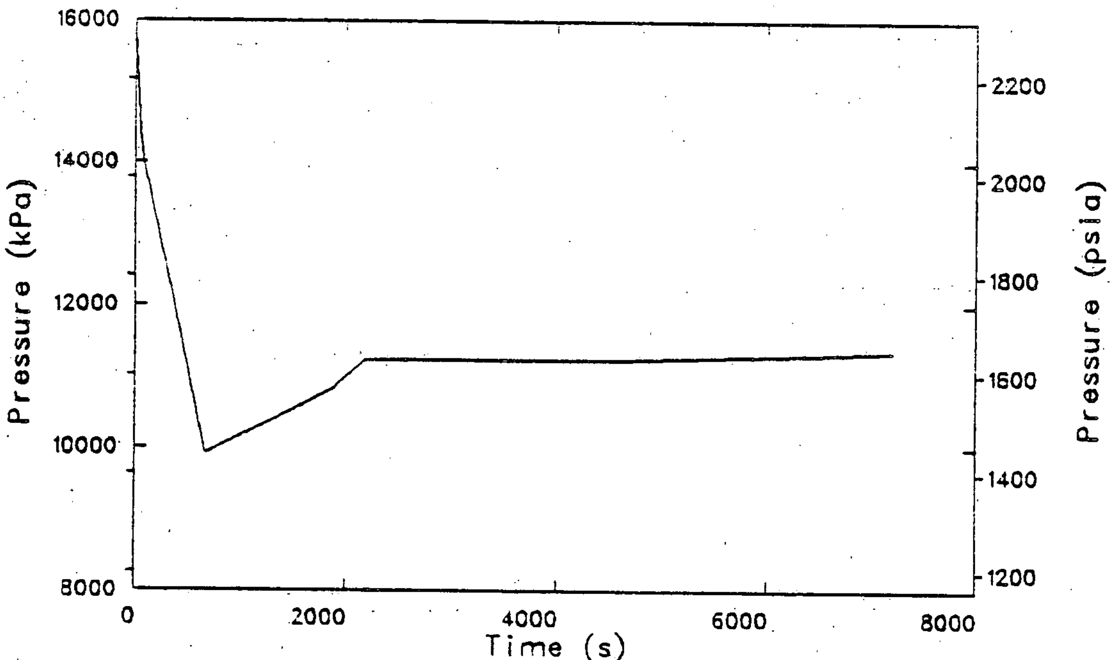

Figure A-217. Sequence 9-26 reactor vessel downcomer fluid pressure.

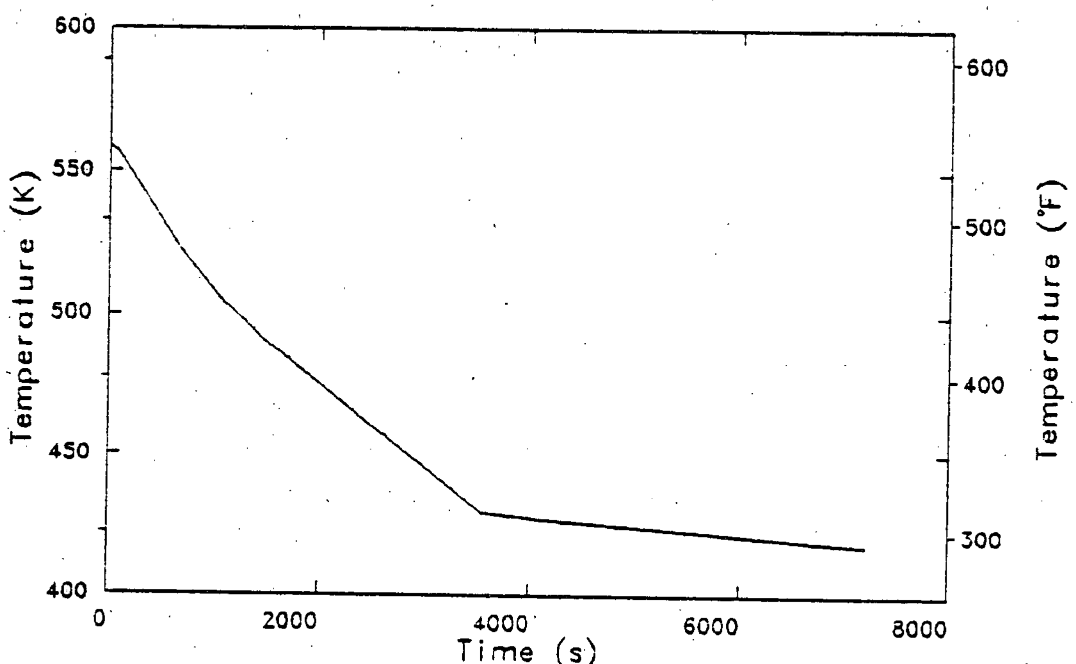

Figure A-218. Sequence 9-26 reactor vessel downcomer fluid temperature.

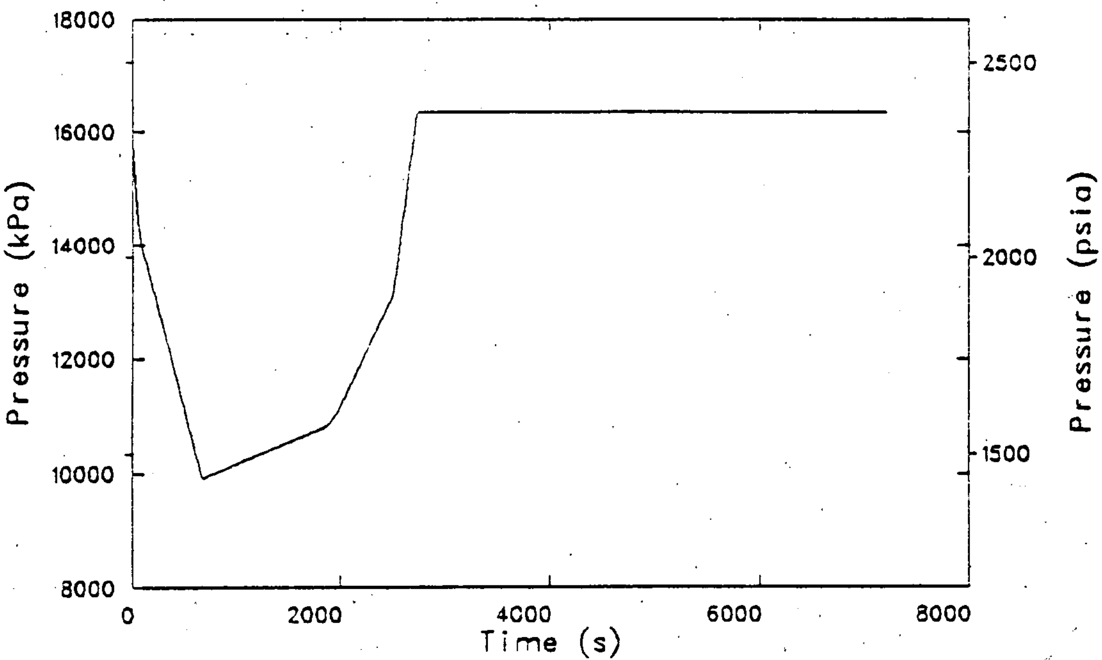

Figure A-219. Sequence 9-27 reactor vessel downcomer fluid pressure.

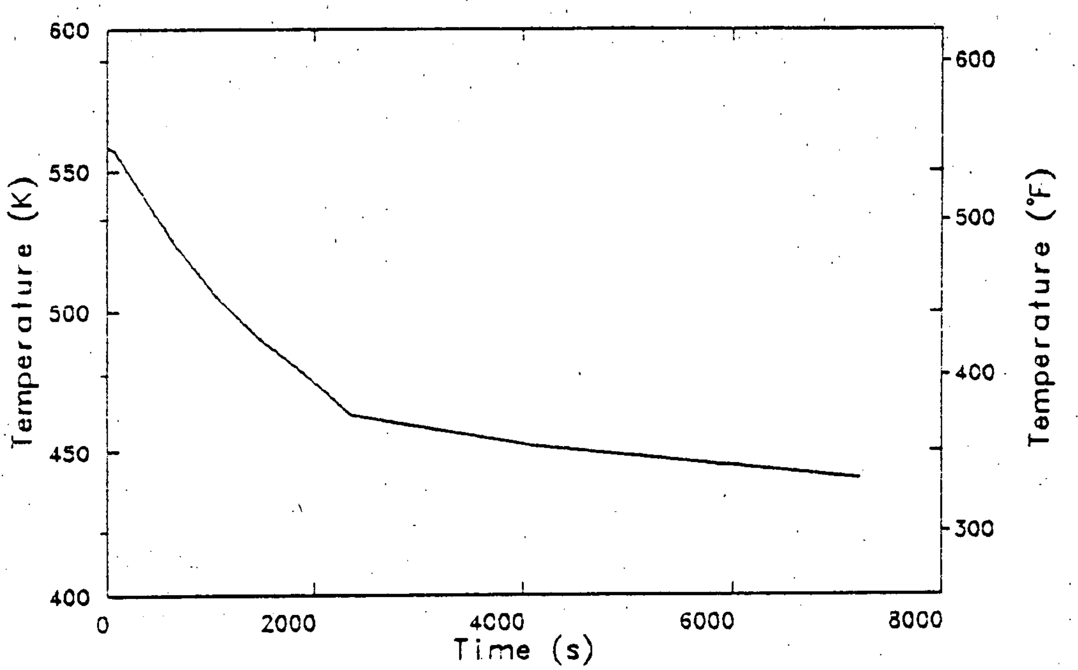

Figure A-220. Sequence 9-27 reactor vessel downcomer fluid temperature.

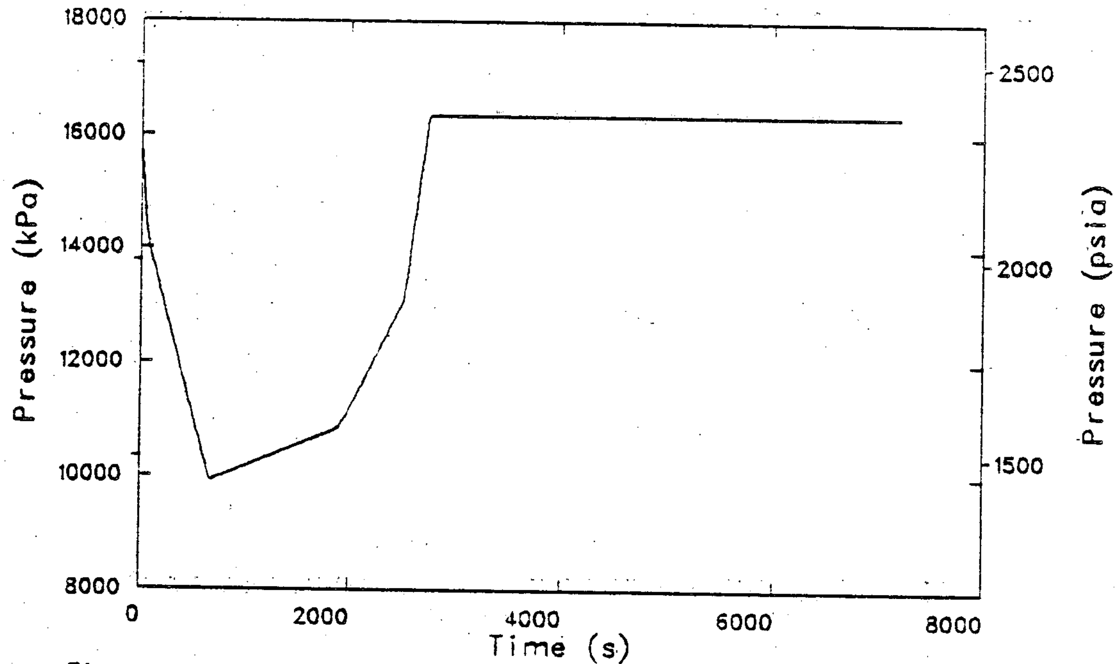

Figure A-221. Sequence 9-28 reactor vessel downcomer fluid pressure.

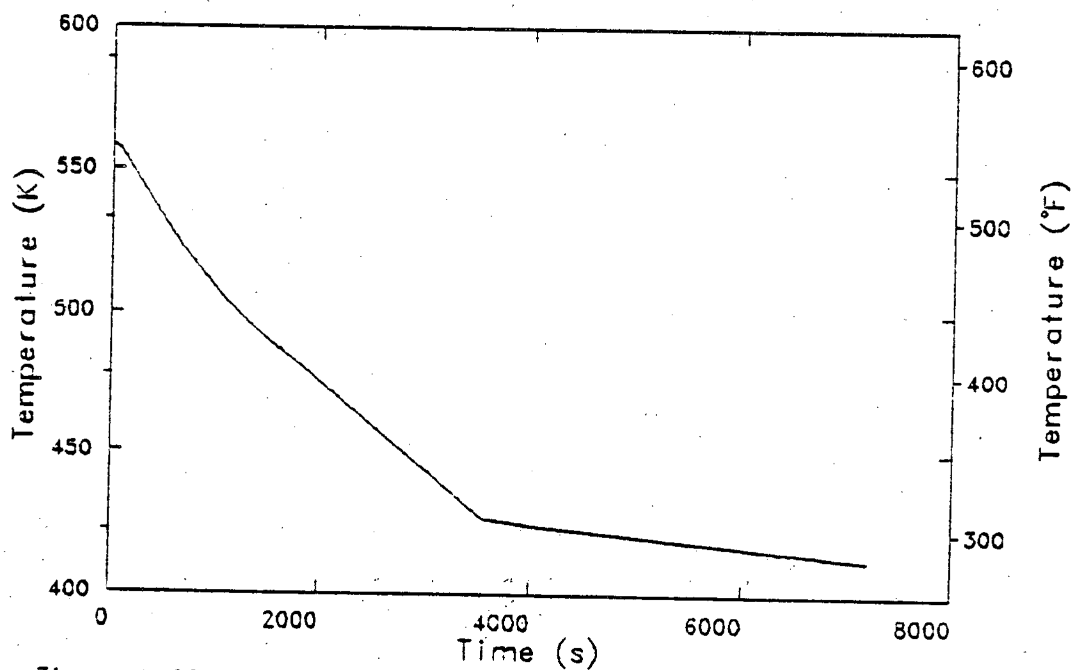

Figure A-222. Sequence 9-28 reactor vessel downcomer fluid temperature.

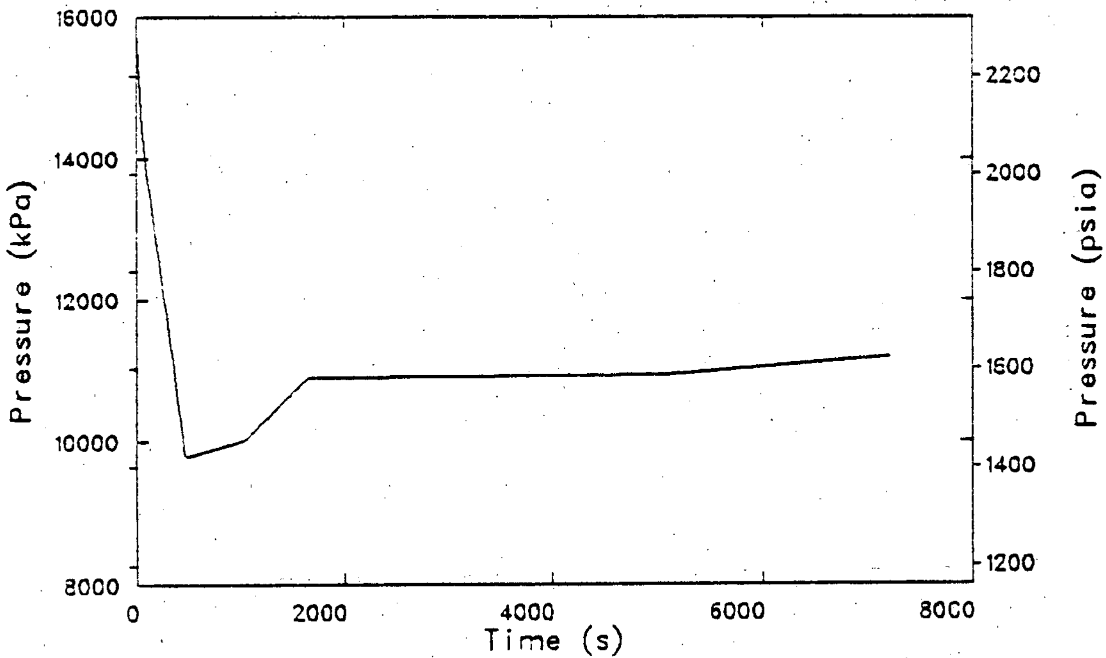

Figure A-223. Sequence 9-29 reactor vessel downcomer fluid pressure.

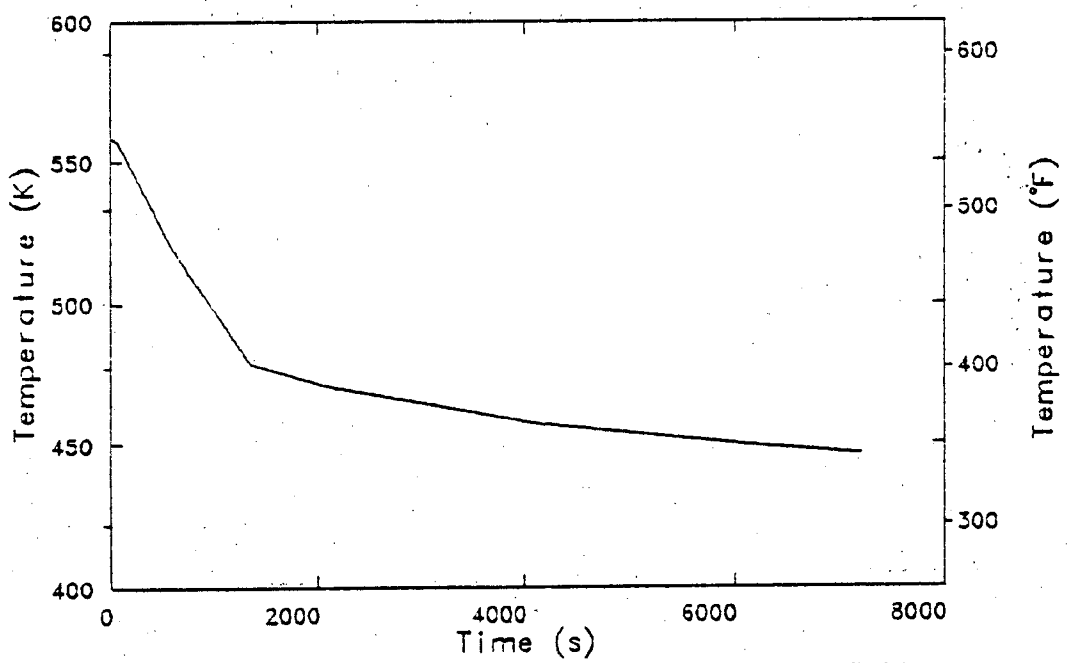

Figure A-224. Sequence 9-29 reactor vessel downcomer fluid temperature.

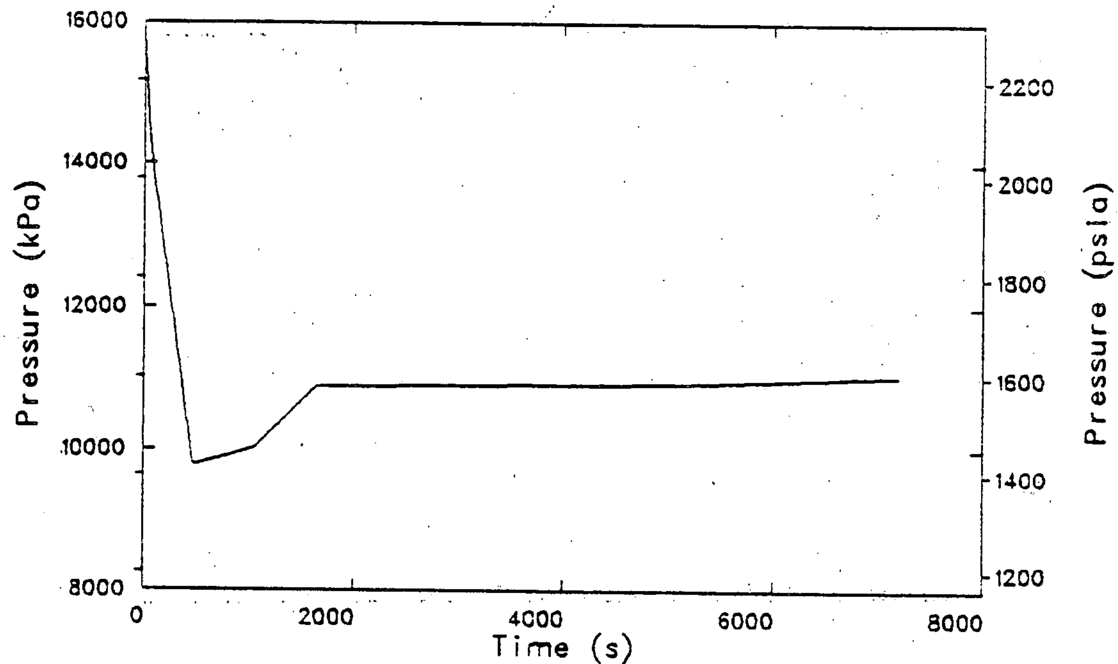

Figure A-225. Sequence 9-30 reactor vessel downcomer fluid pressure.

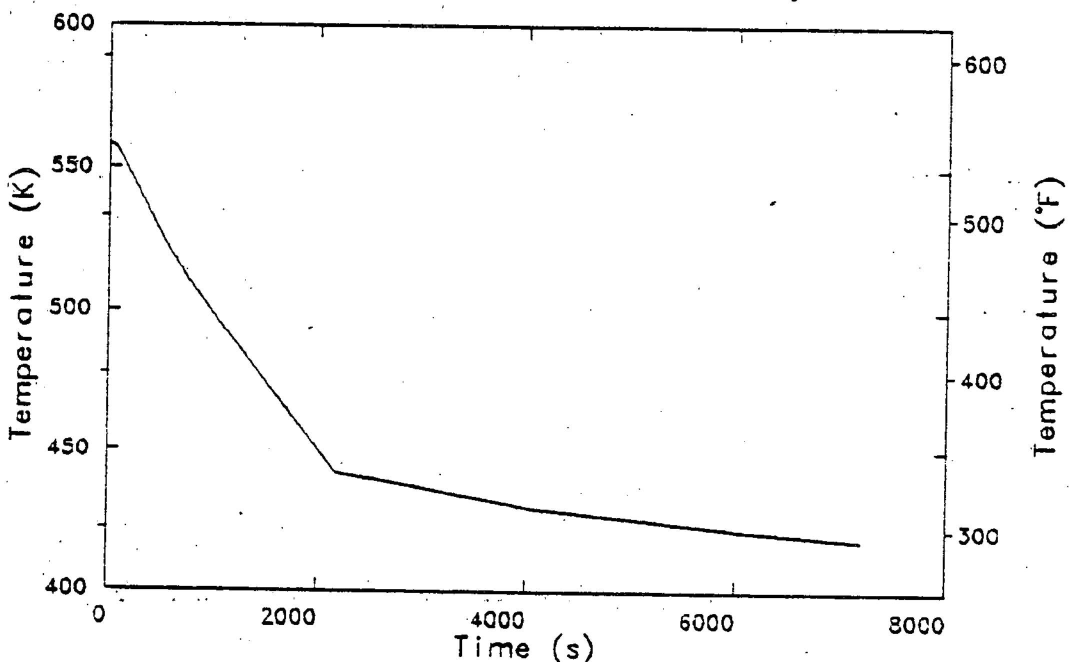

Figure A-226. Sequence 9-30 reactor vessel downcomer fluid temperature.

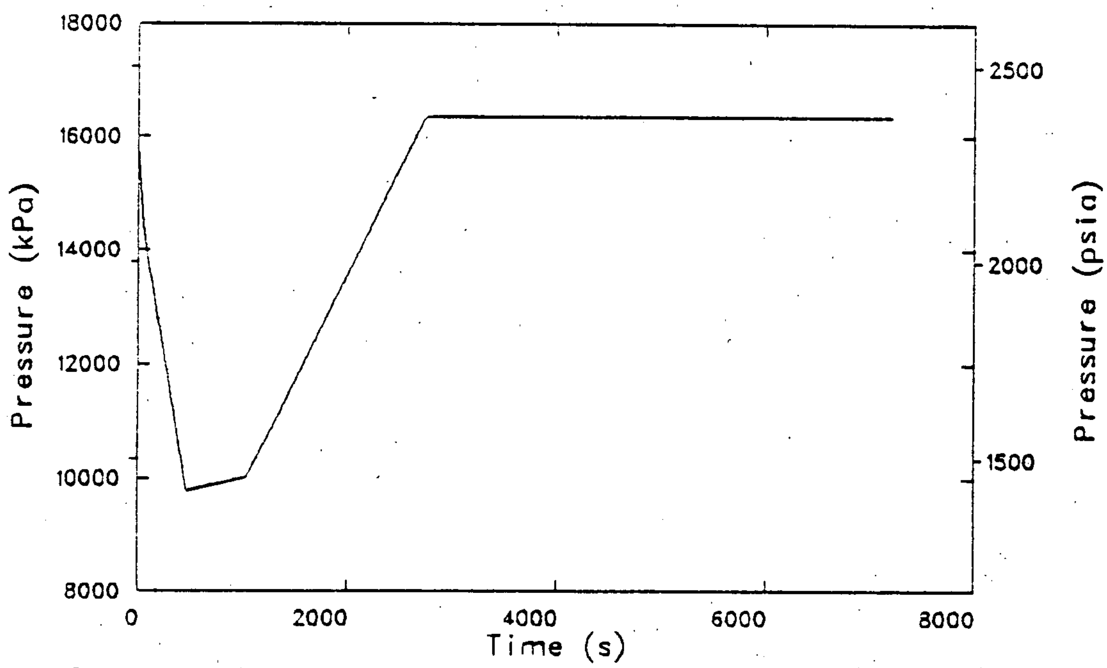

Figure A-227. Sequence 9-31 reactor vessel downcomer fluid pressure.

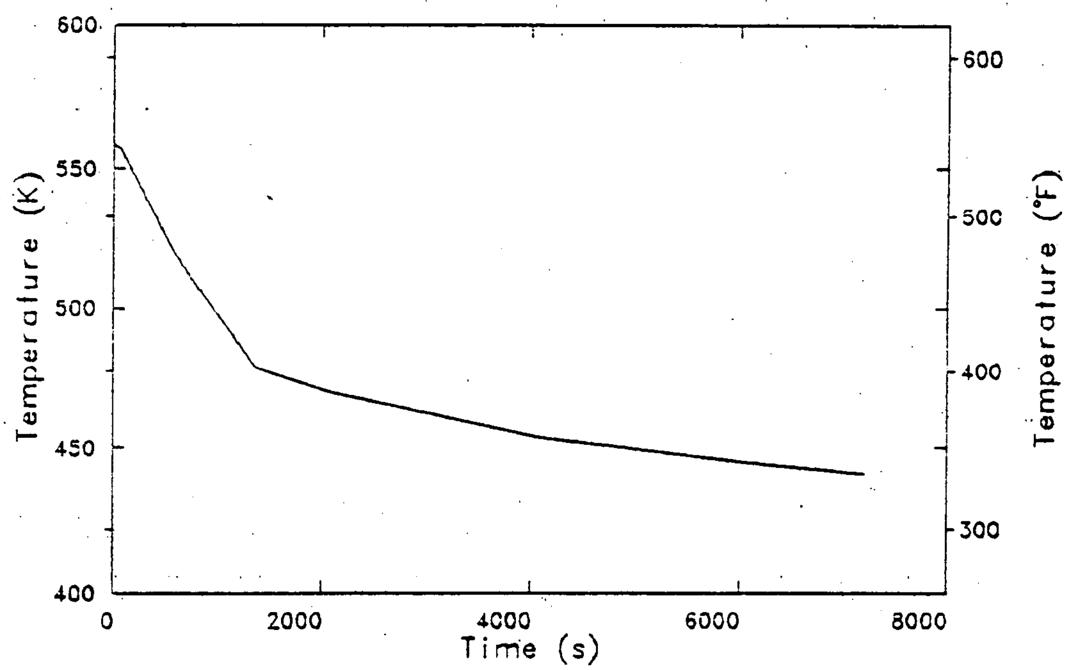

Figure A-228. Sequence 9-31 reactor vessel downcomer fluid temperature.

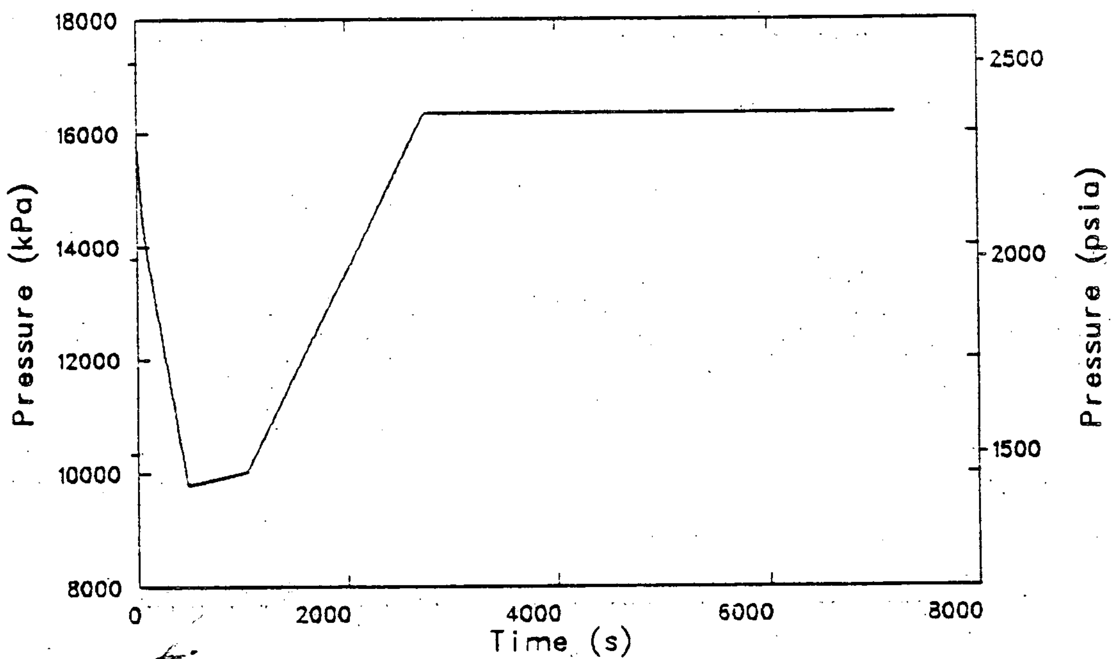

Figure A-229. Sequence 9-32 reactor vessel downcomer fluid pressure.

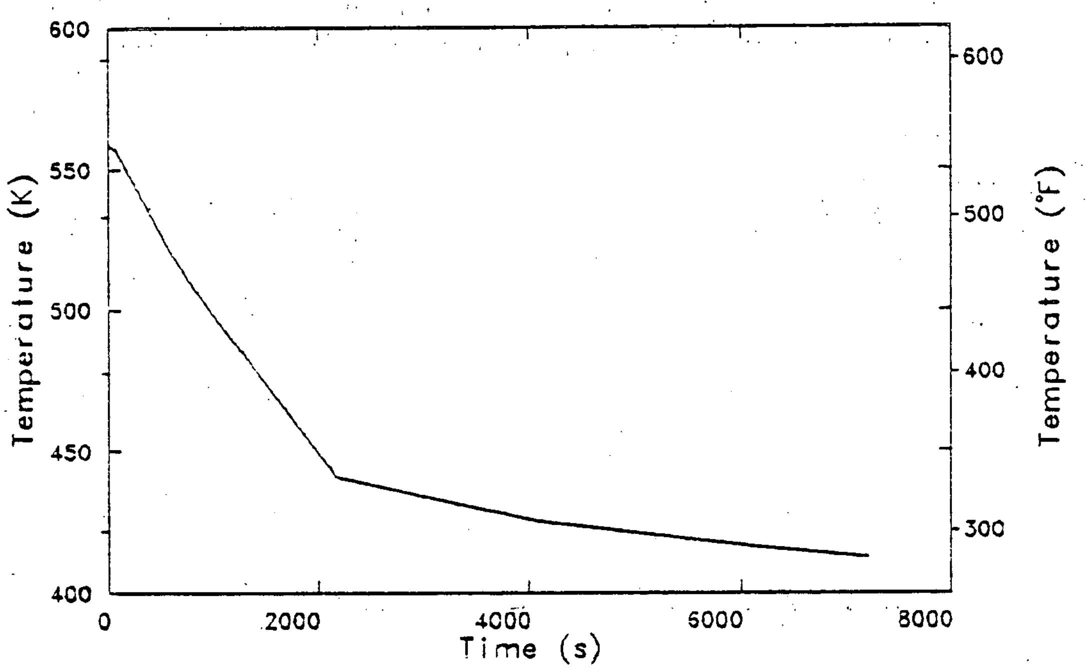

Figure A-230. Sequence 9-32 reactor vessel downcomer fluid temperature.

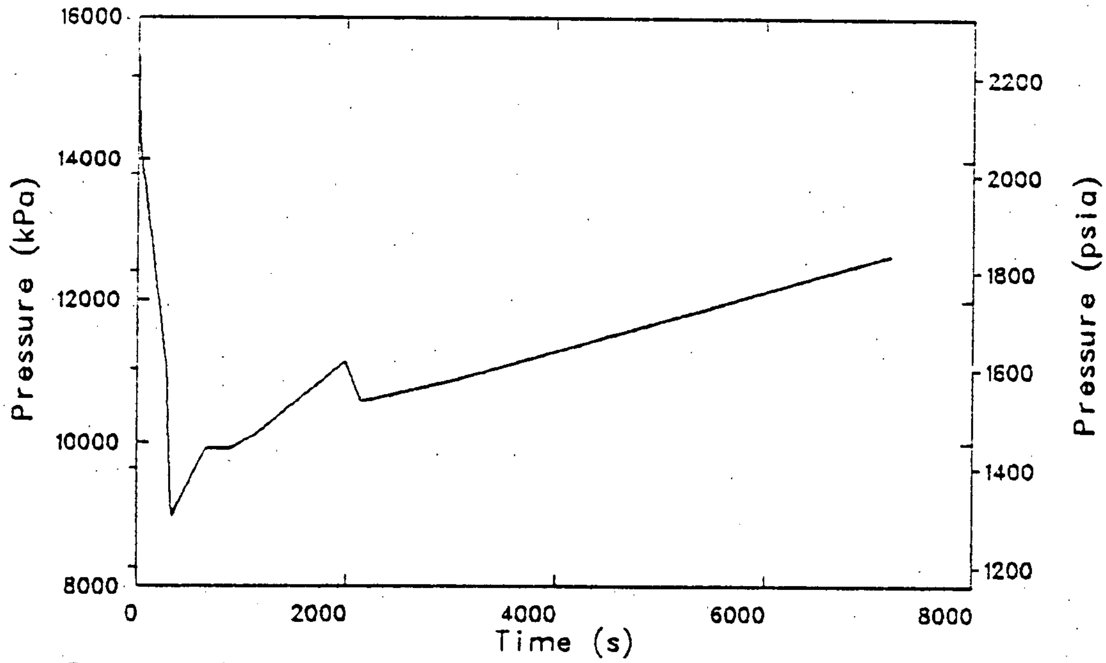

Figure A-231. Sequence 9-33 reactor vessel downcomer fluid pressure.

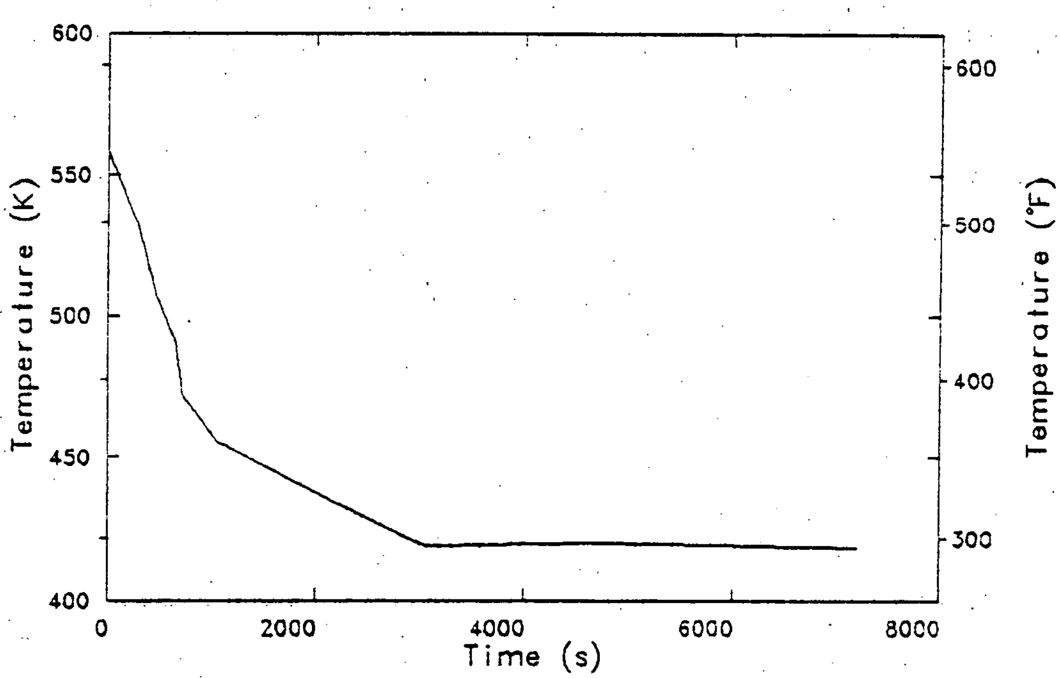

Figure A-232. Sequence 9-33 reactor vessel downcomer fluid temperature.

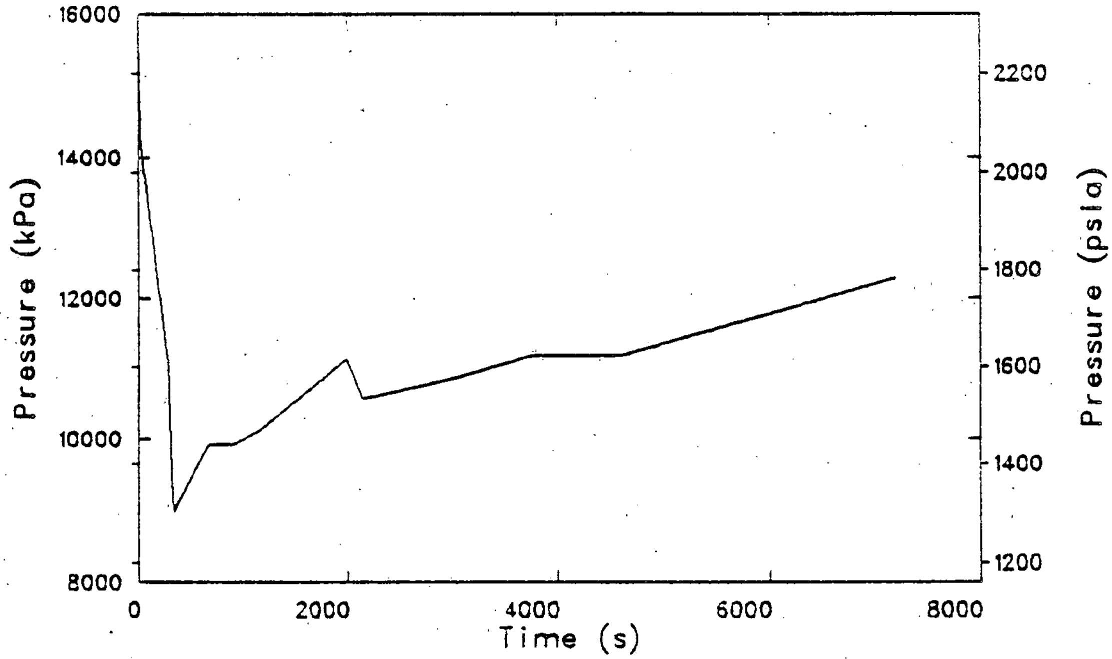

Figure A-233. Sequence 9-34 reactor vessel downcomer fluid pressure.

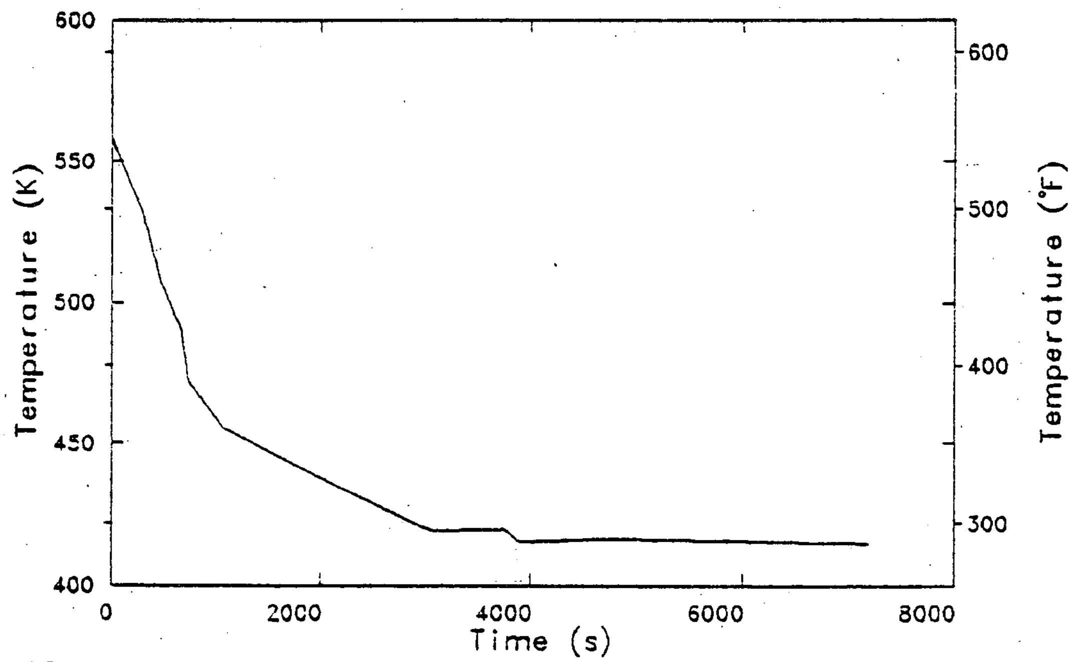

Figure A-234. Sequence 9-34 reactor vessel downcomer fluid temperature.

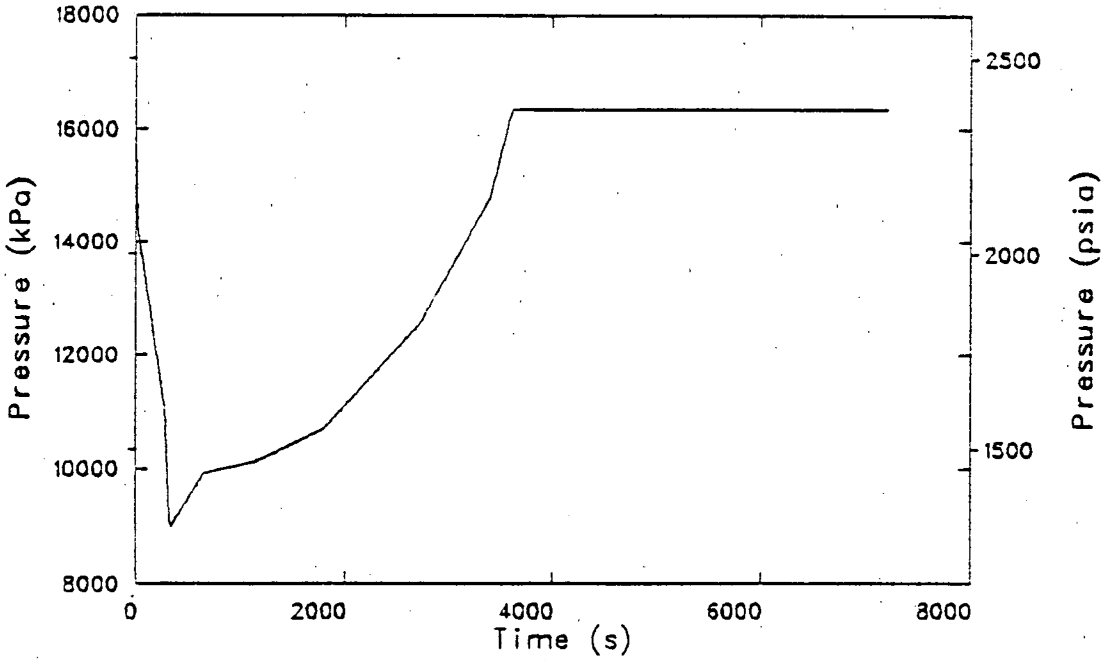

Figure A-235. Sequence 9-35 reactor vessel downcomer fluid pressure.

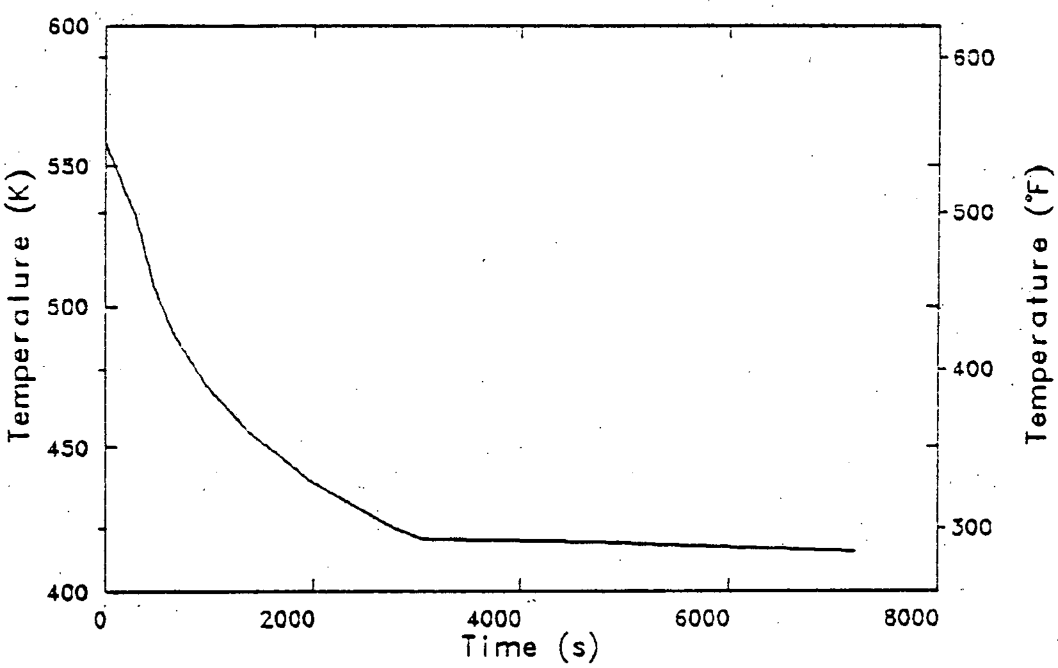

Figure A-236. Sequence 9-35 reactor vessel downcomer fluid temperature.

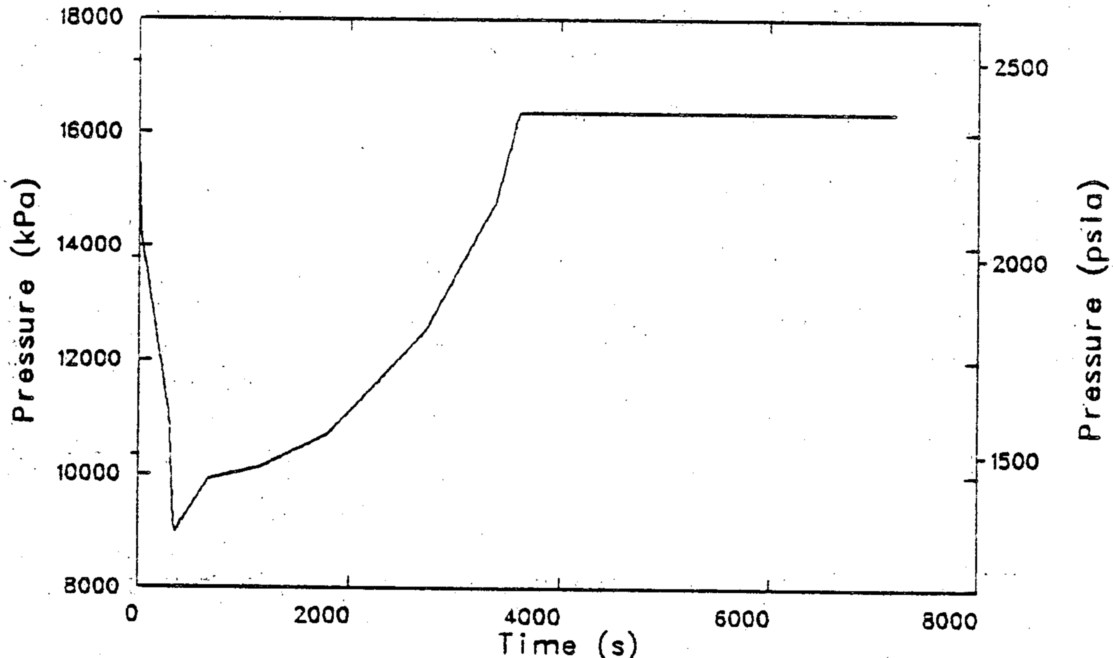

Figure A-237. Sequence 9-36 reactor vessel downcomer fluid pressure.

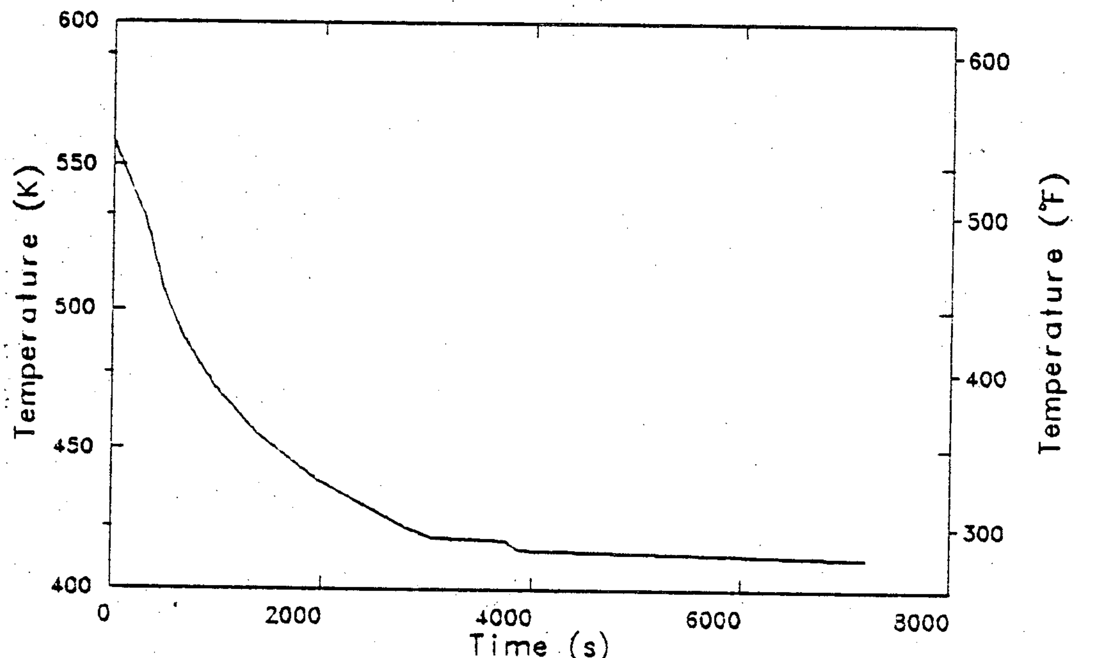

Figure A-238. Sequence 9-36 reactor vessel downcomer fluid temperature.

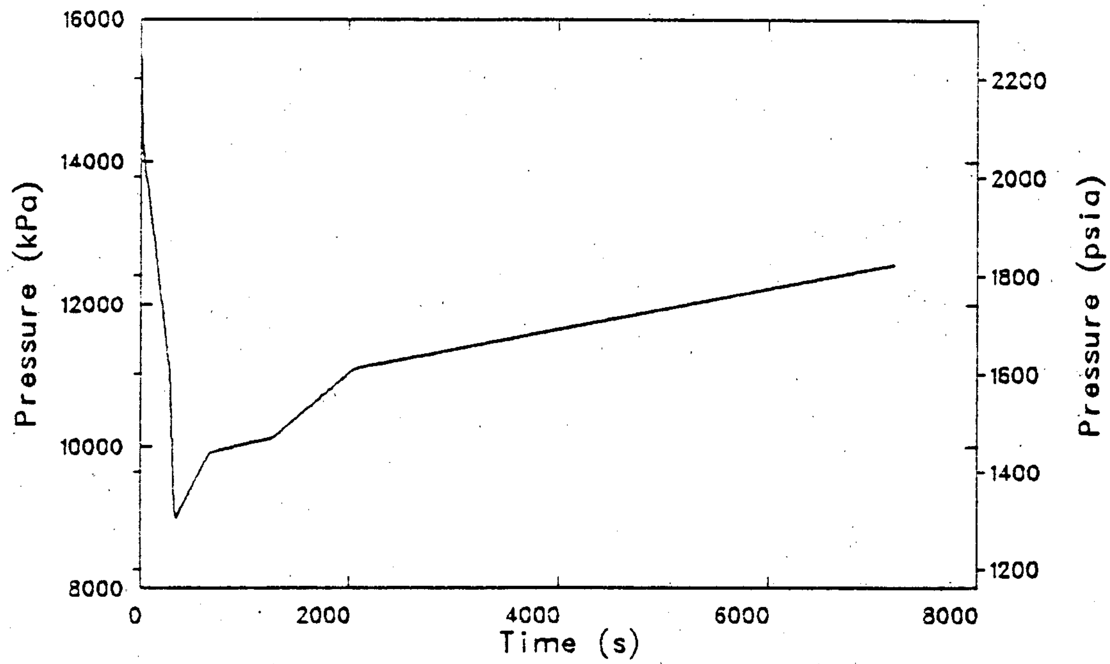

Figure A-239. Sequence 9-37 reactor vessel downcomer fluid pressure.

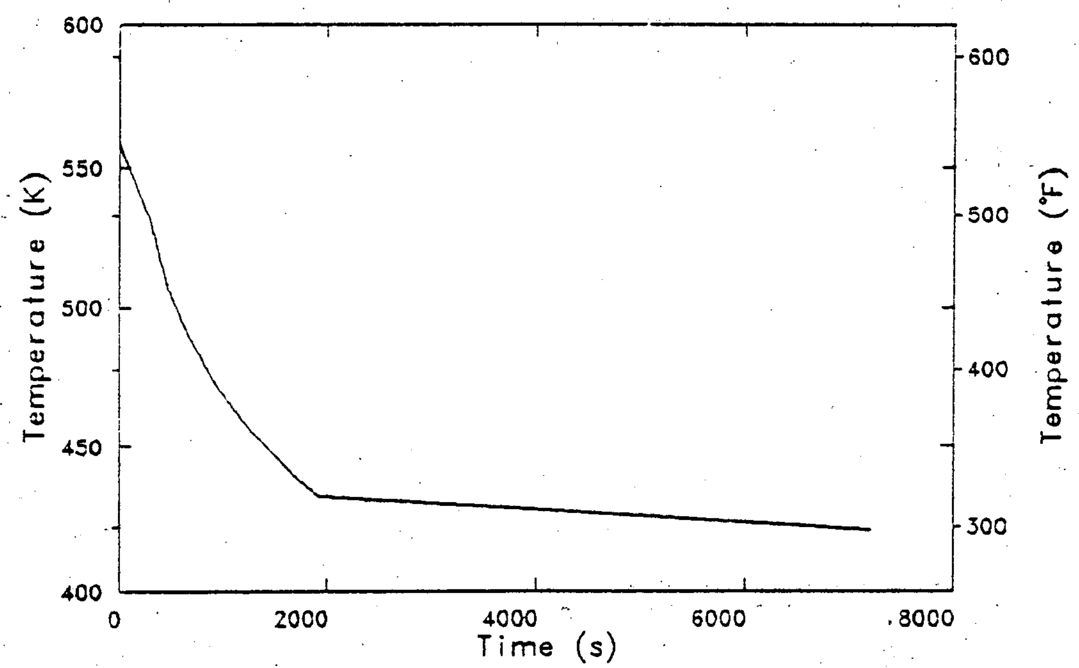

Figure A-240. Sequence 9-37 reactor vessel downcomer fluid temperature.

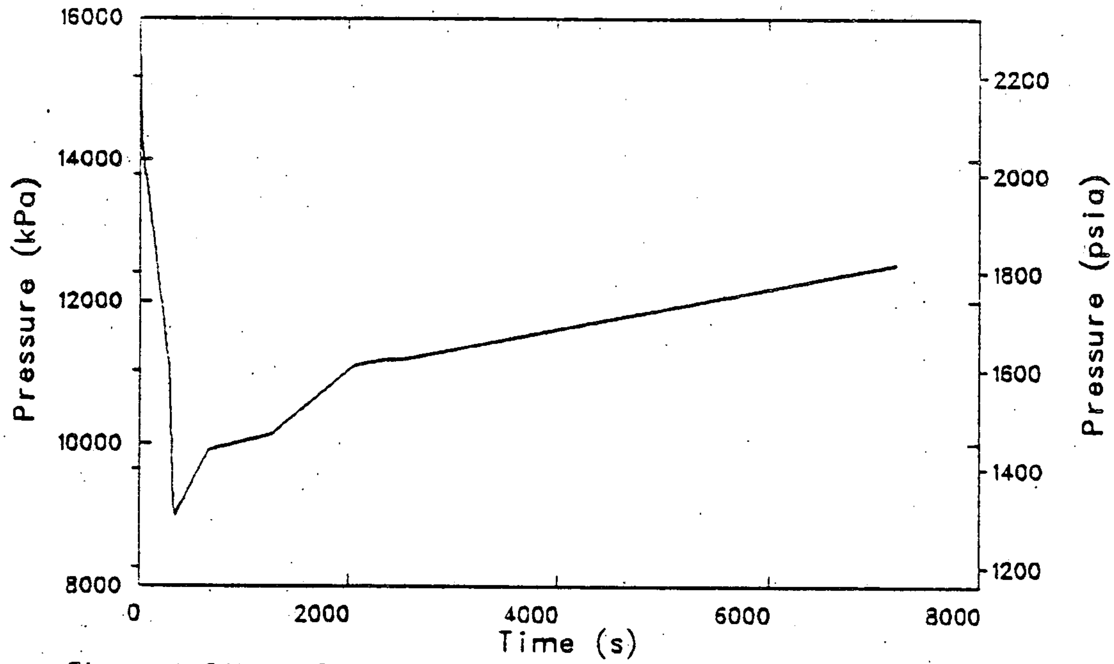

Figure A-241. Sequence 9-38 reactor vessel downcomer fluid pressure.

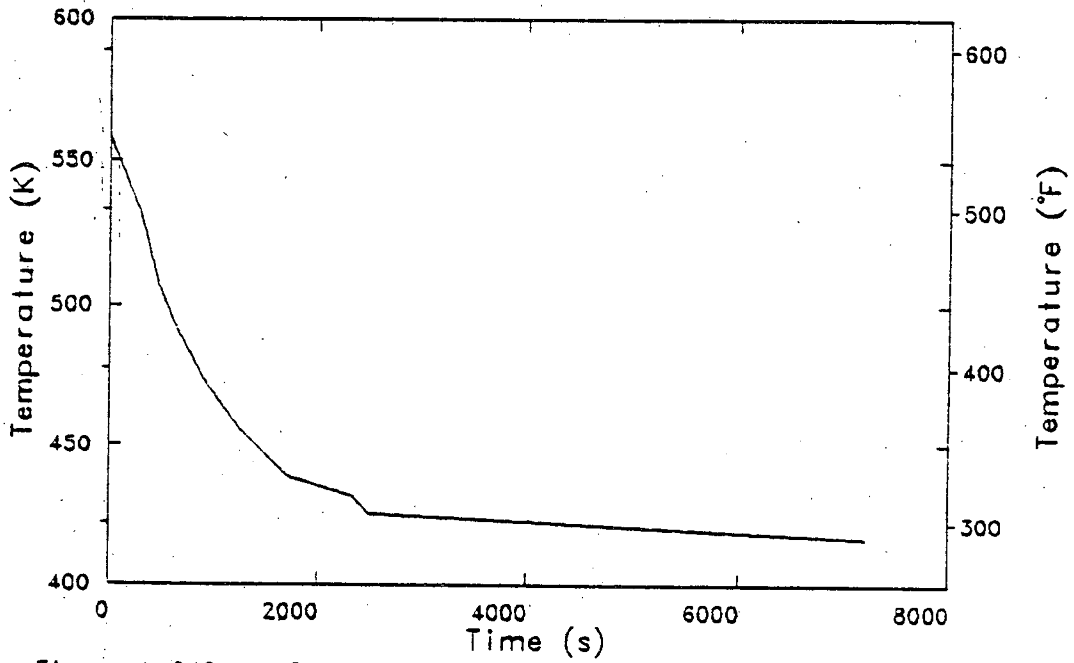

Figure A-242. Sequence 9-38 reactor vessel downcomer fluid temperature.

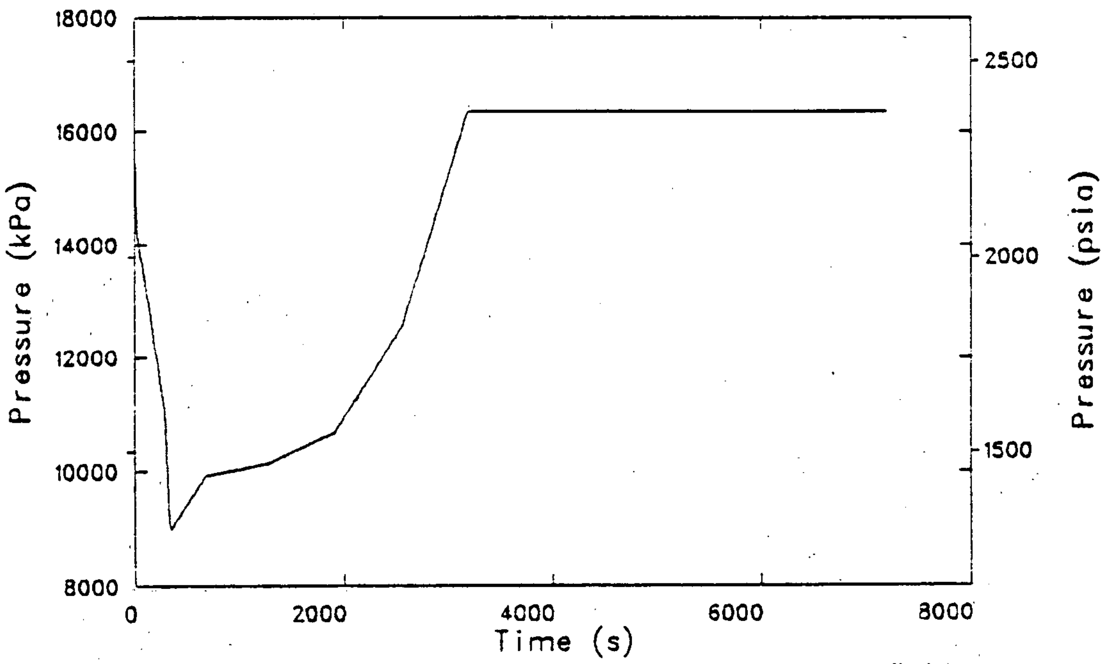

Figure A-243. Sequence 9-39 reactor vessel downcomer fluid pressure.

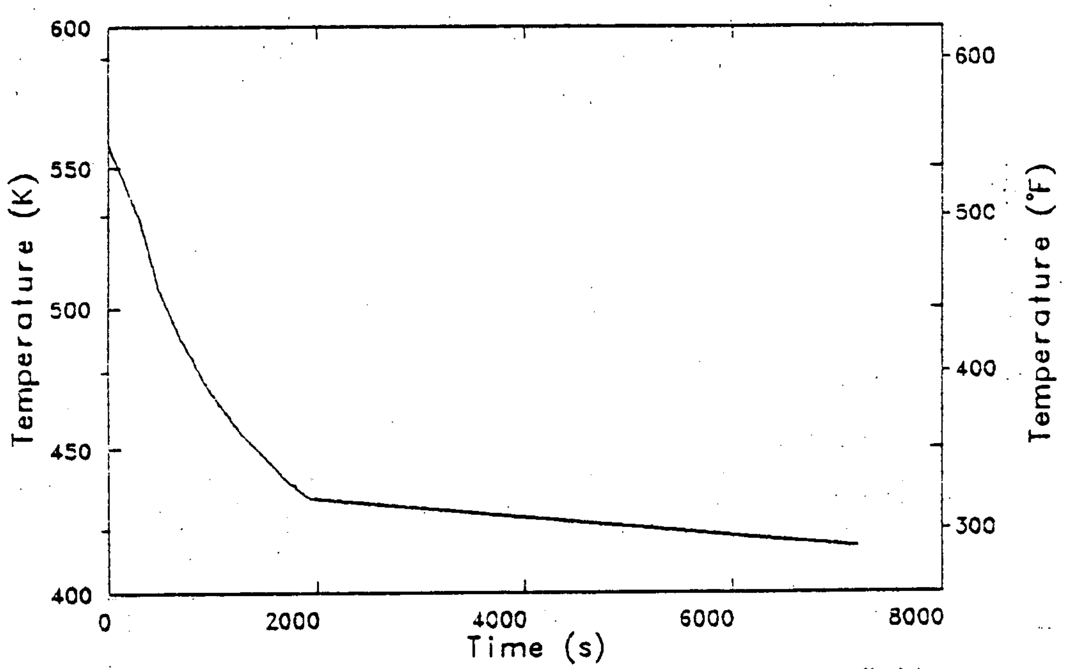

Figure A-244. Sequence 9-39 reactor vessel downcomer fluid temperature.

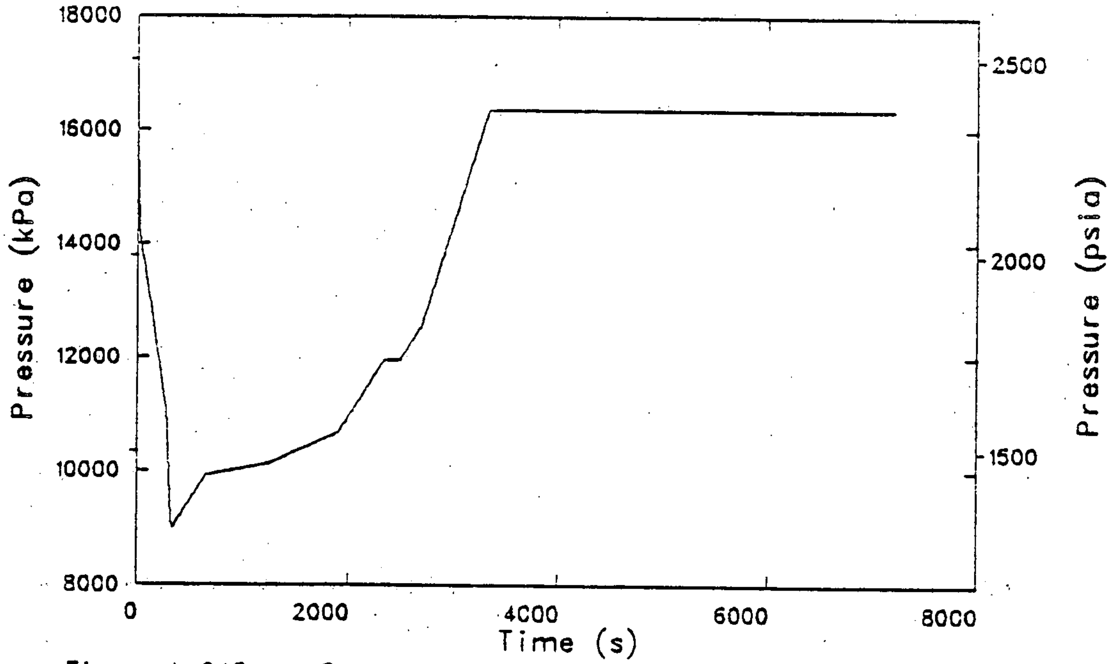

Figure A-245. Sequence 9-40 reactor vessel downcomer fluid pressure.

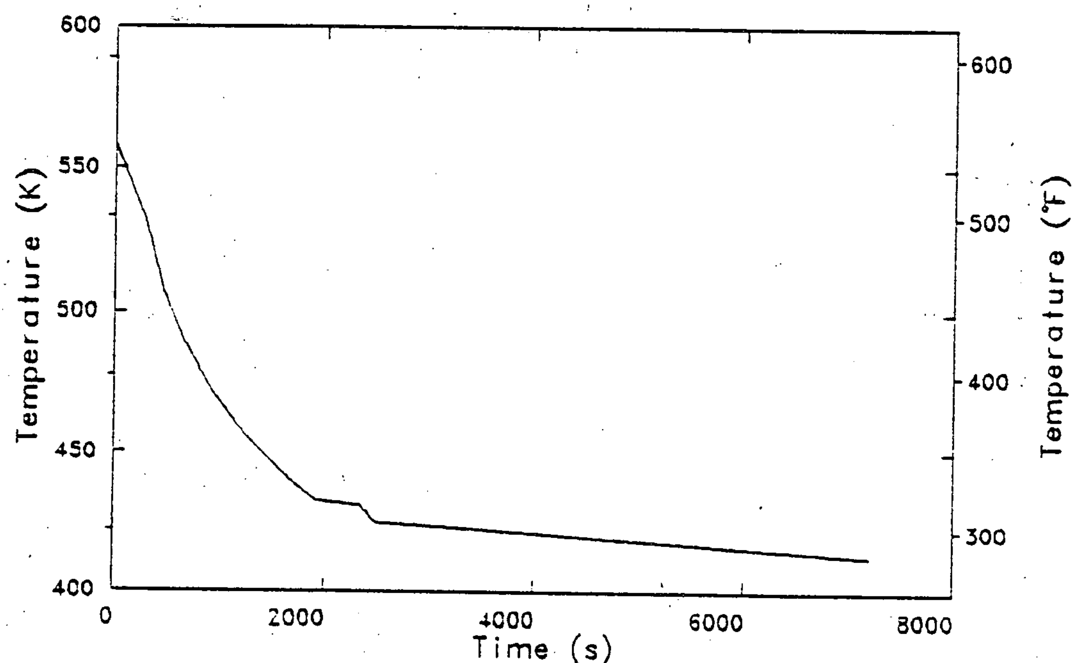

Figure A-246. Sequence 9-40 reactor vessel downcomer fluid temperature.

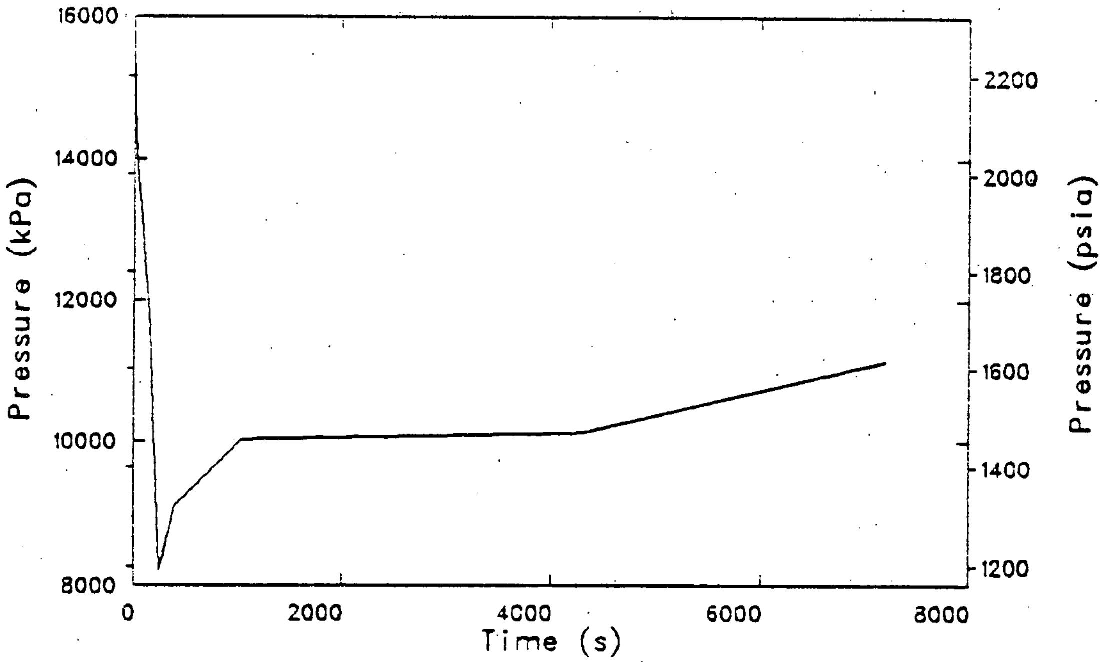

Figure A-247. Sequence 9-41 reactor vessel downcomer fluid pressure.

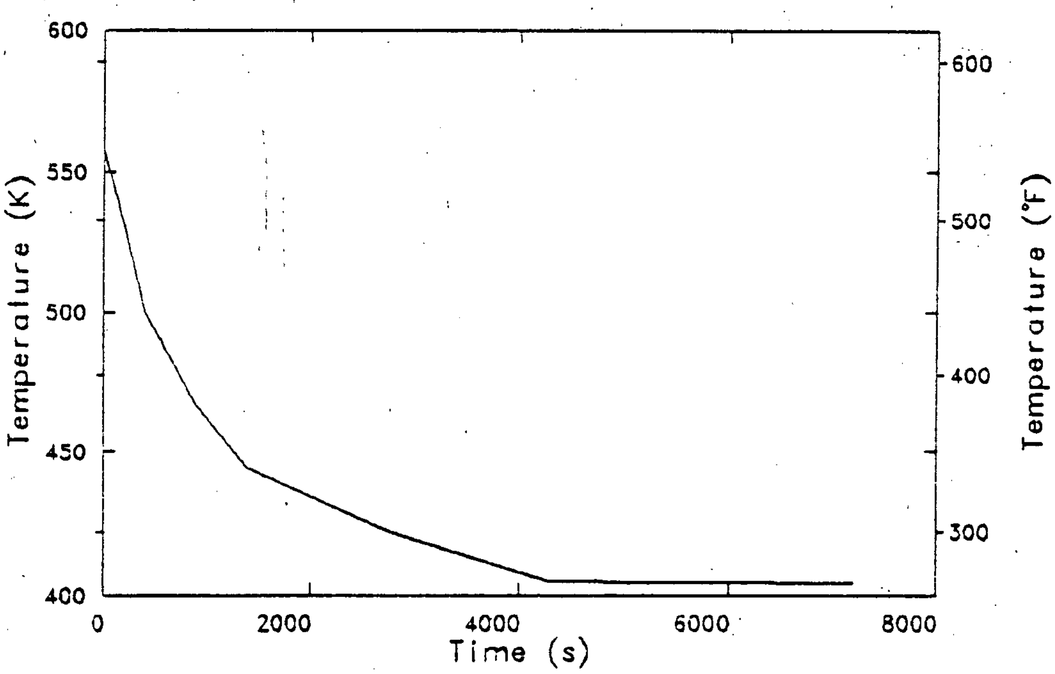

Figure A-248. Sequence 9-41 reactor vessel downcomer fluid temperature.

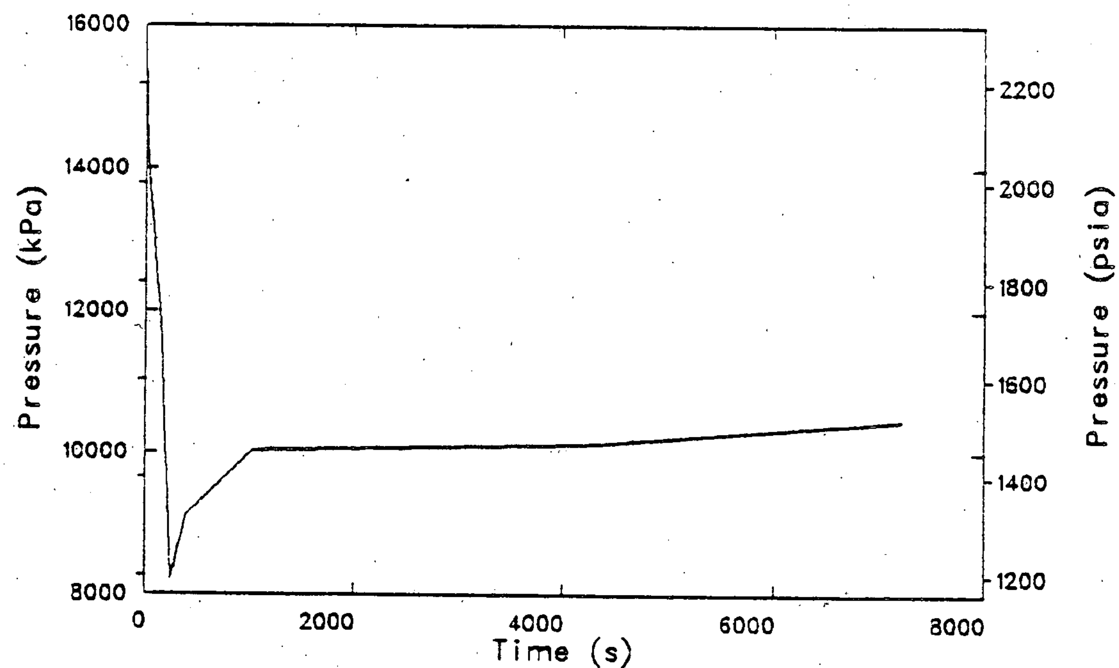

Figure A-249. Sequence 9-42 reactor vessel downcomer fluid pressure.

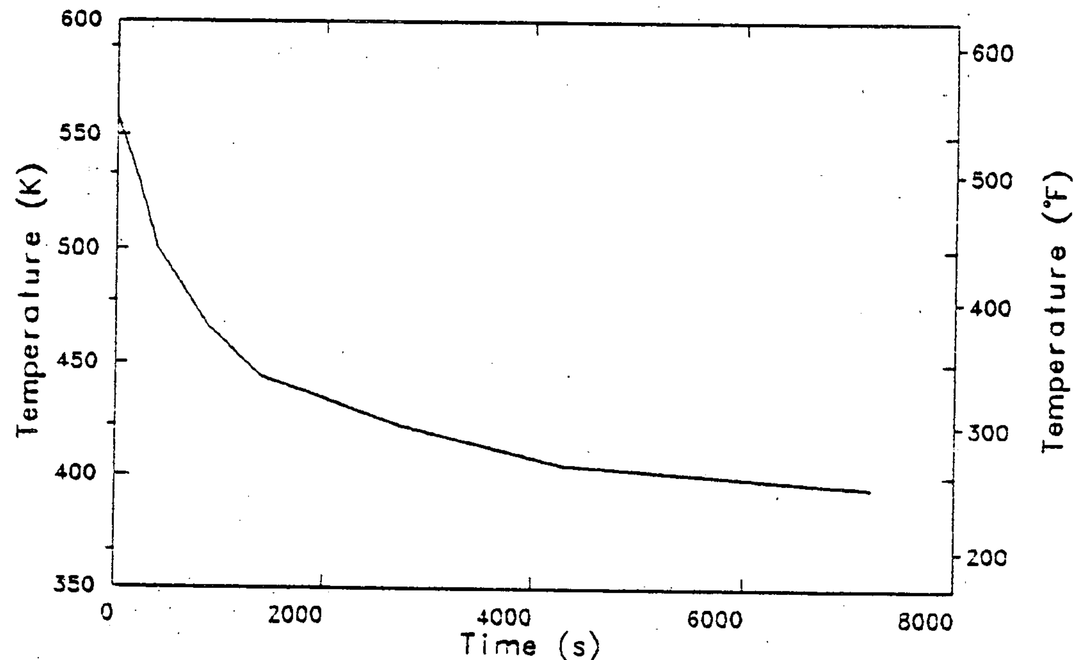

Figure A-250. Sequence 9-42 reactor vessel downcomer fluid temperature.

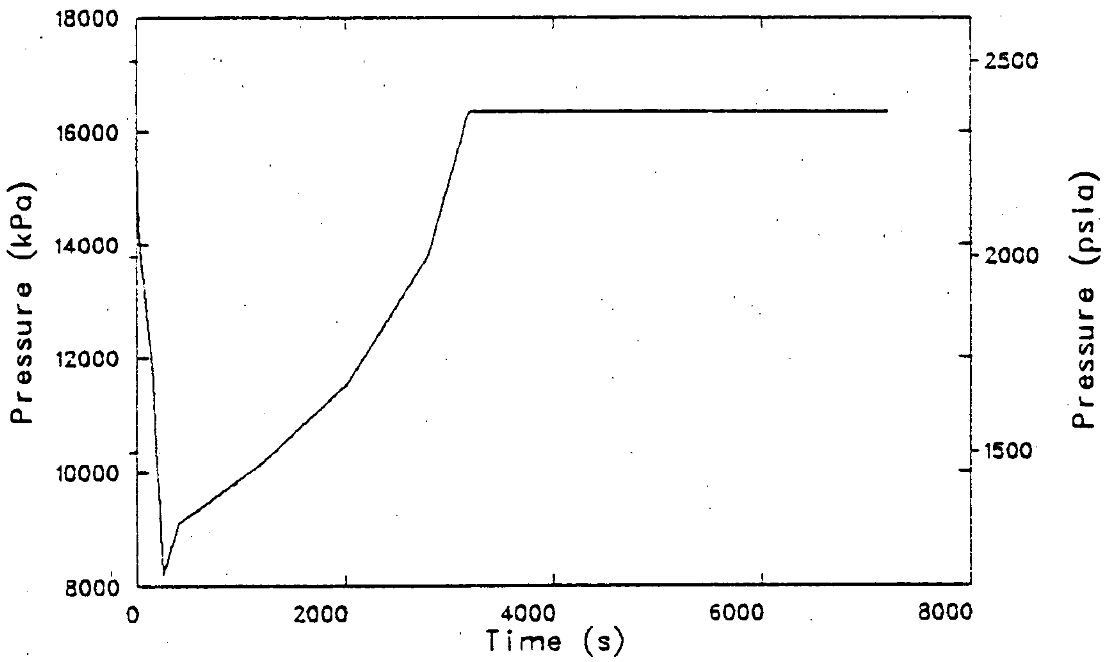

Figure A-251. Sequence 9-43 reactor vessel downcomer fluid pressure.

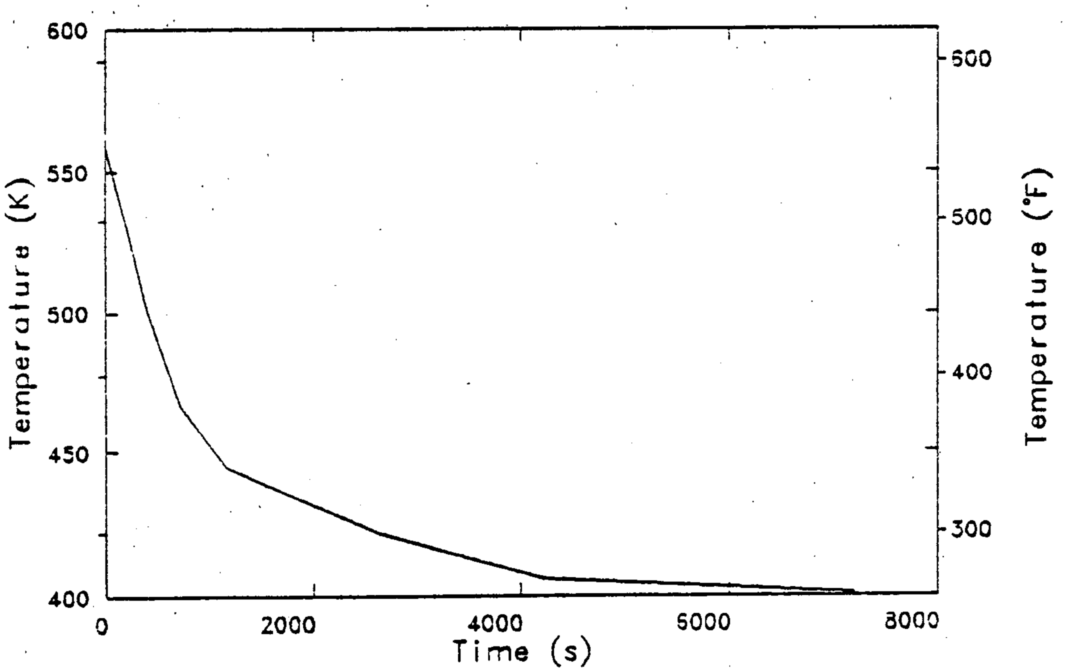

Figure A-252. Sequence 9-43 reactor vessel downcomer fluid temperature.

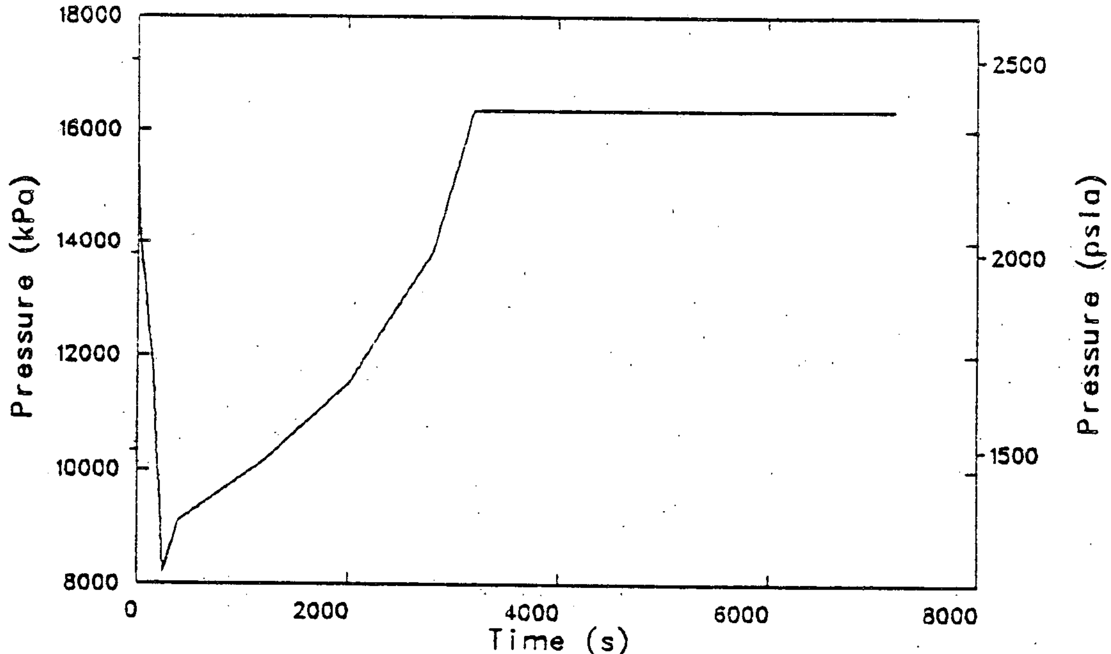

Figure A-253. Sequence 9-44 reactor vessel downcomer fluid pressure.

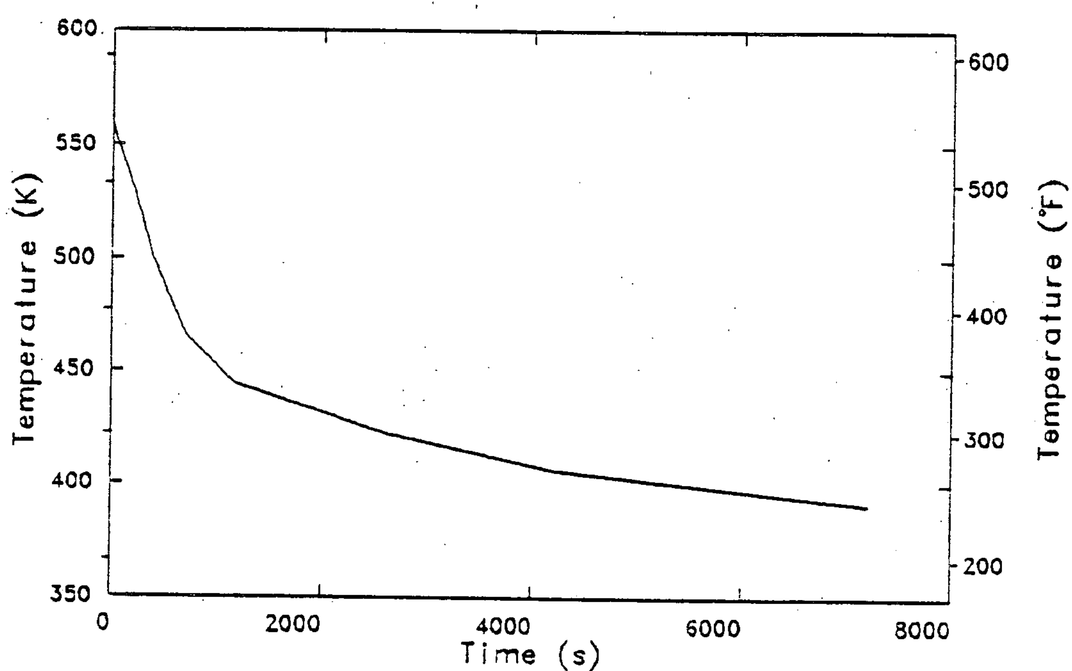

Figure A-254. Sequence 9-44 reactor vessel downcomer fluid temperature.

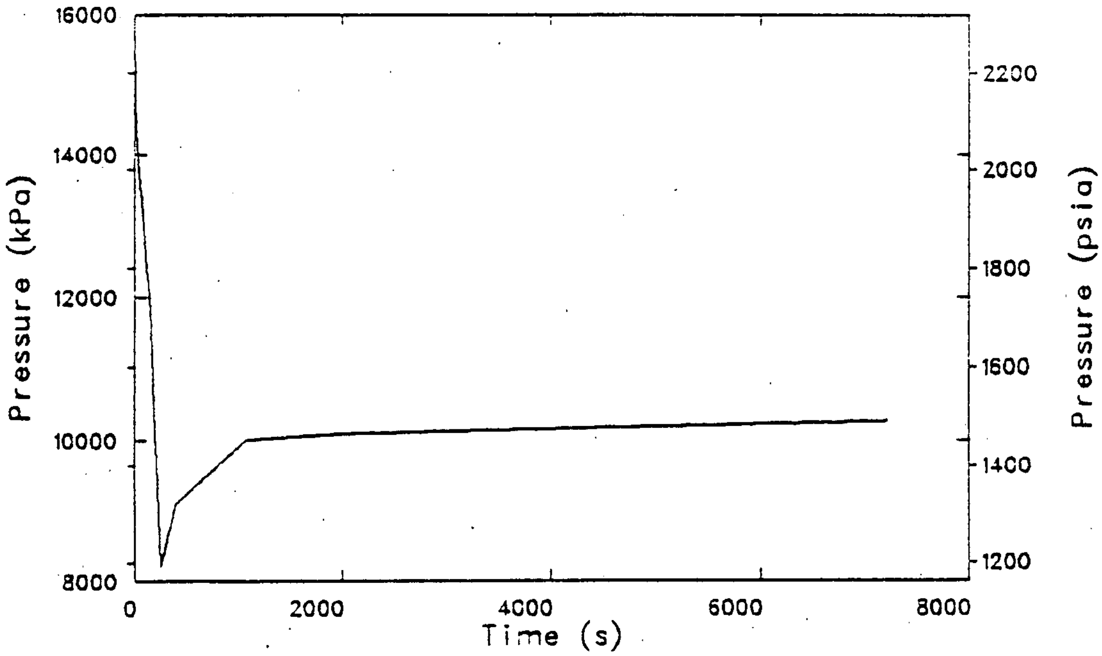

Figure A-255. Sequence 9-45 reactor vessel downcomer fluid pressure.

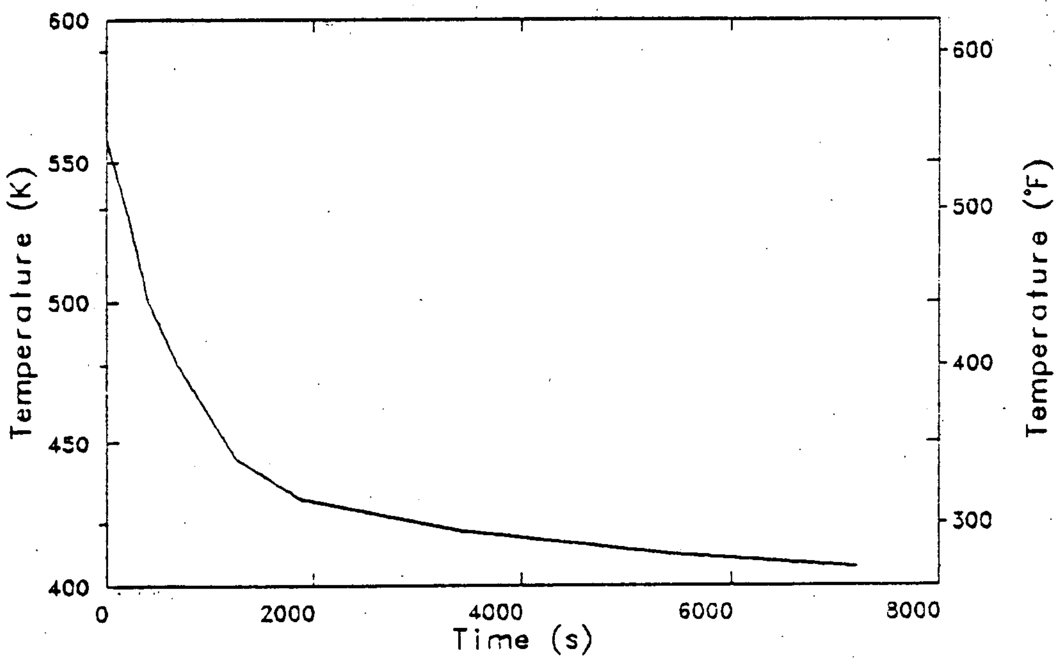

Figure A-256. Sequence 9-45 reactor vessel downcomer fluid temperature.

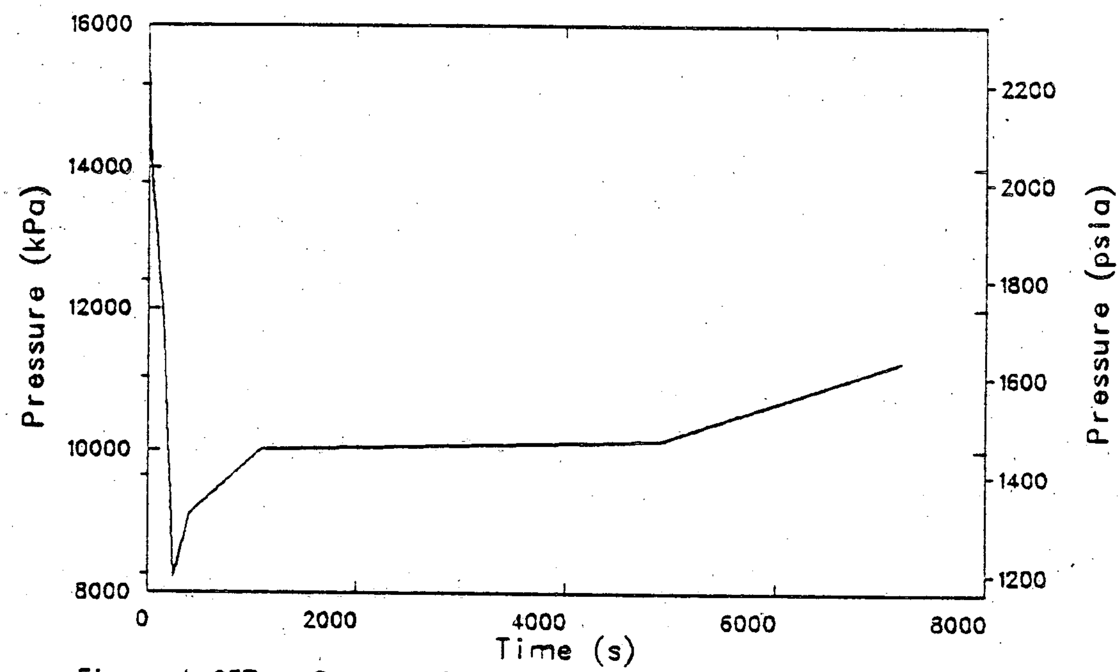

Figure A-257. Sequence 9-46 reactor vessel downcomer fluid pressure.

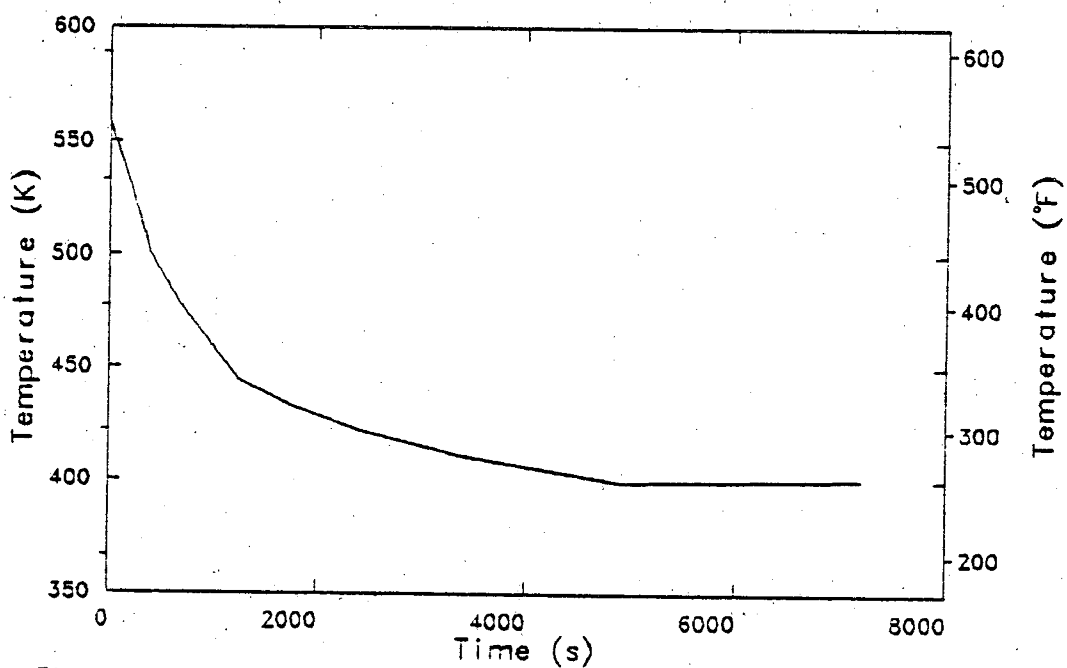

Figure A-258. Sequence 9-46 reactor vessel downcomer fluid temperature.

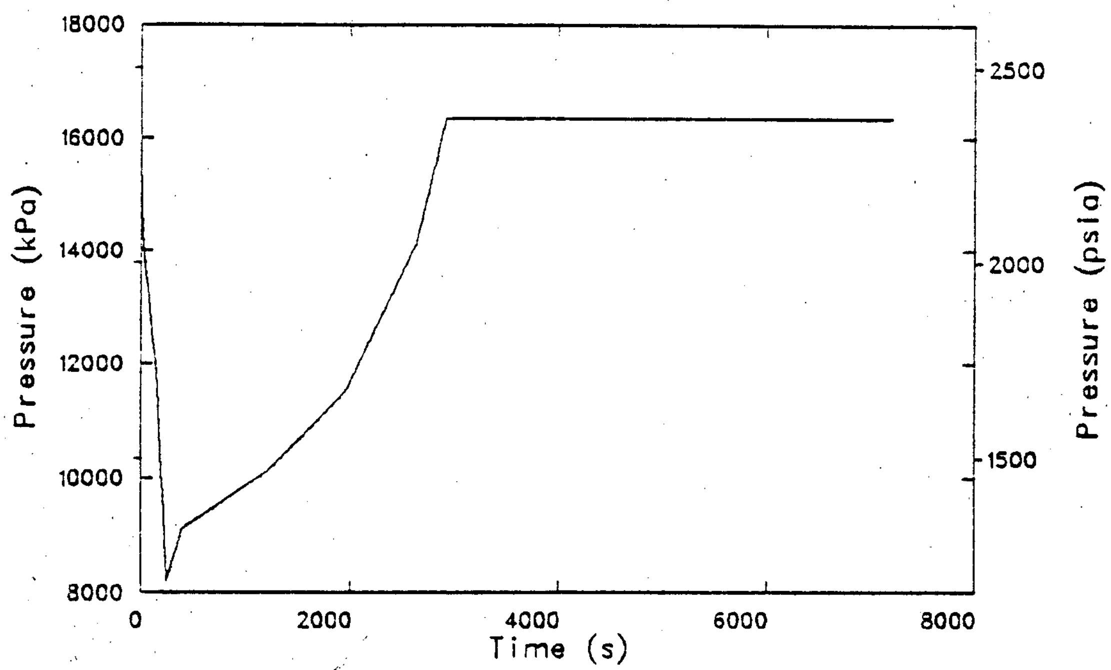

Figure A-259. Sequence 9-47 reactor vessel downcomer fluid pressure.

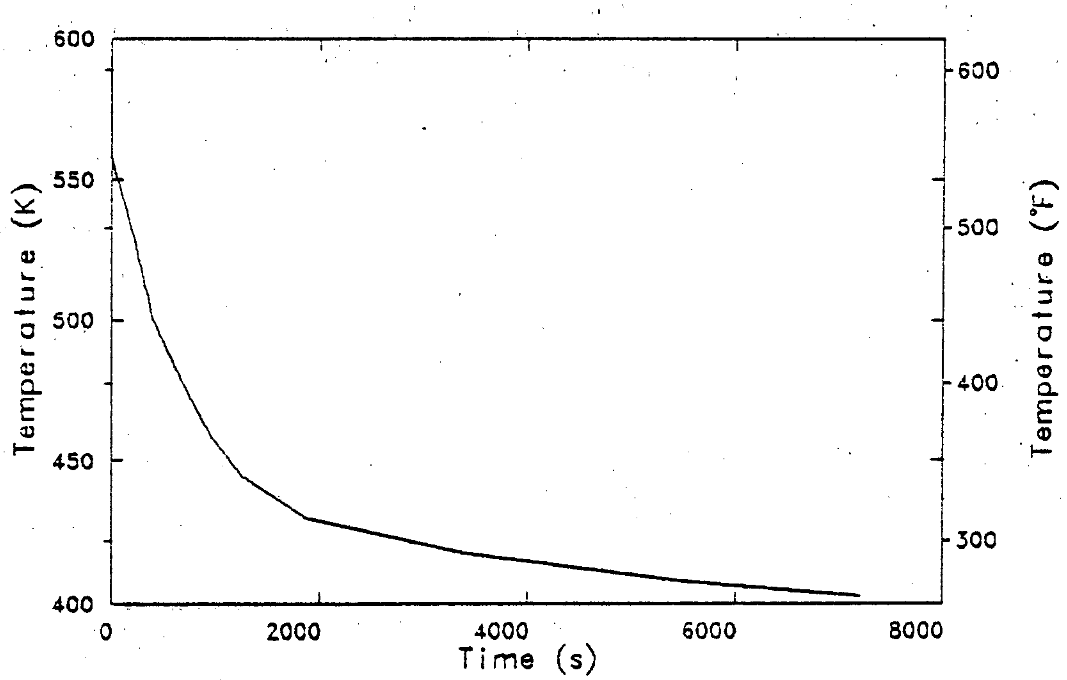

Figure A-260. Sequence 9-47 reactor vessel downcomer fluid temperature.

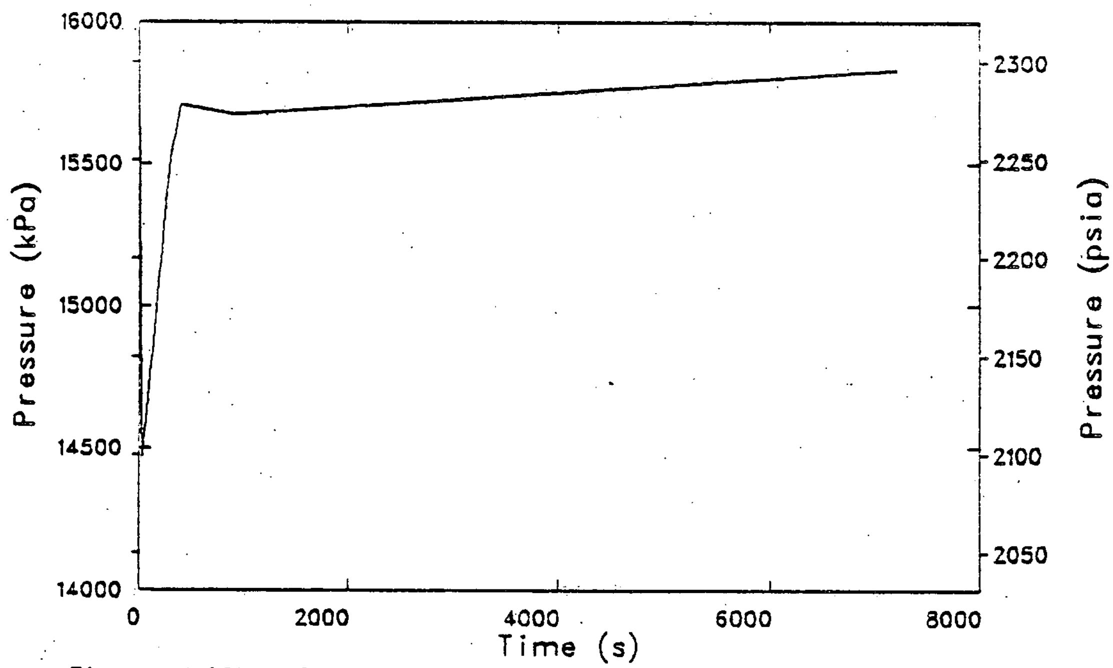

Figure A-261. Sequence 9-48 reactor vessel downcomer fluid pressure.

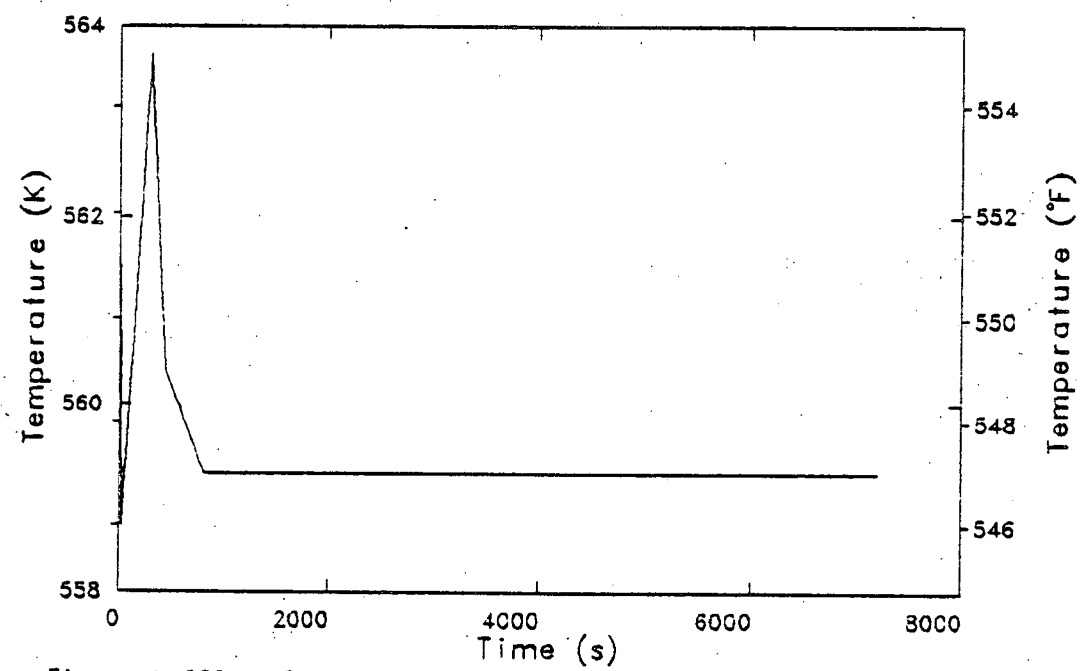

Figure A-262. Sequence 9-48 reactor vessel downcomer fluid temperature.

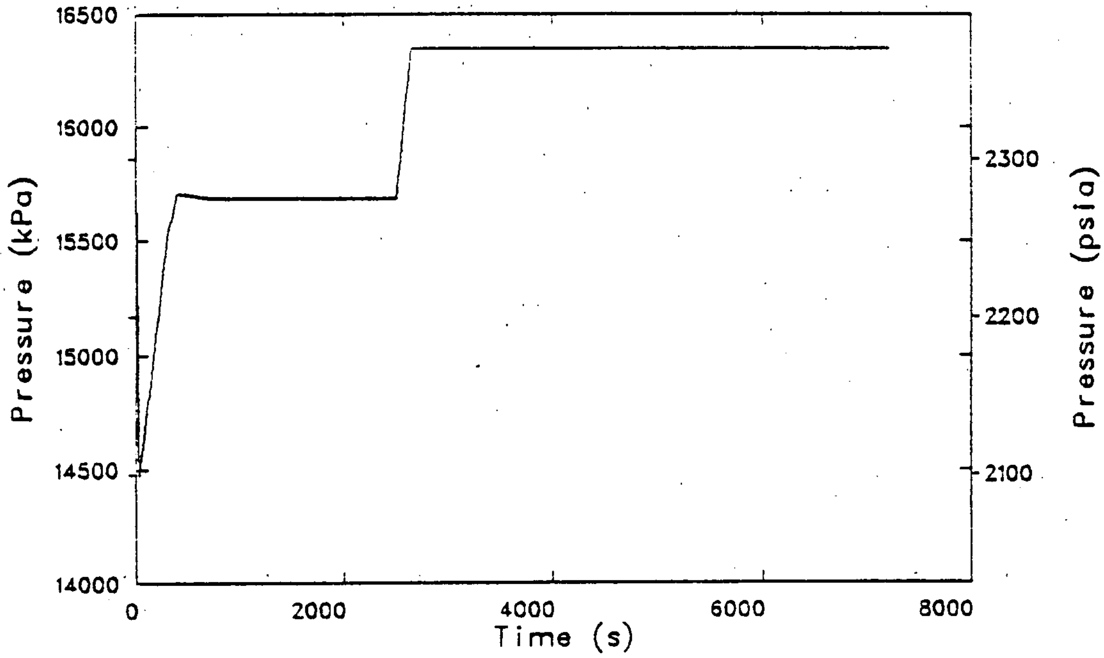

Figure A-263. Sequence 9-49 reactor vessel downcomer fluid pressure.

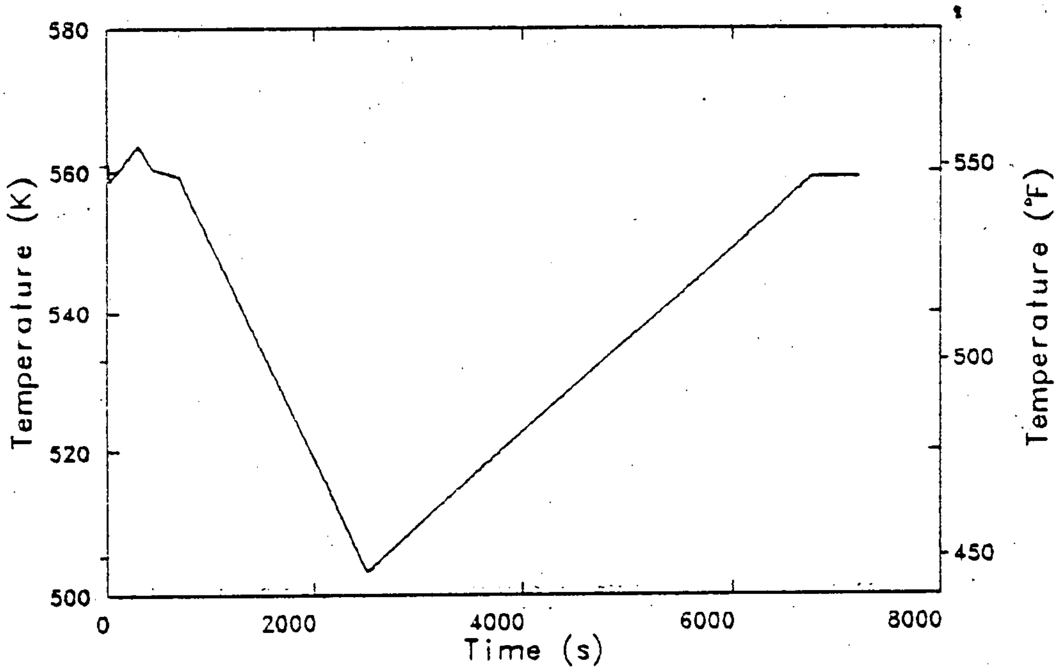

Figure A-264. Sequence 9-49 reactor vessel downcomer fluid temperature.

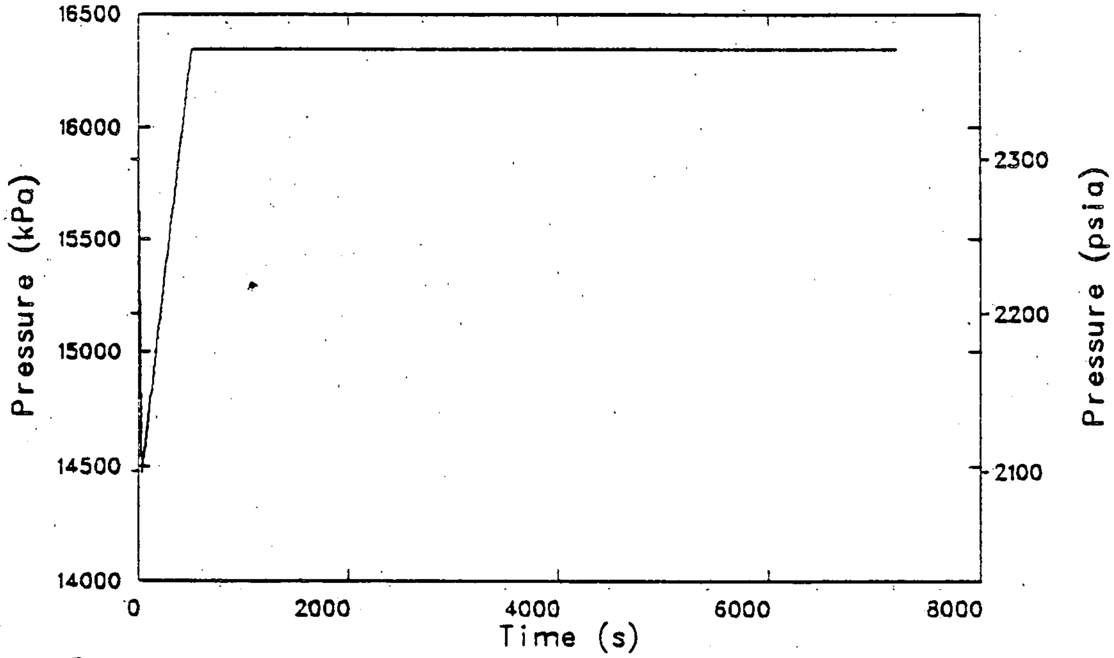

Figure A-265. Sequence 9-50 reactor vessel downcomer fluid pressure.

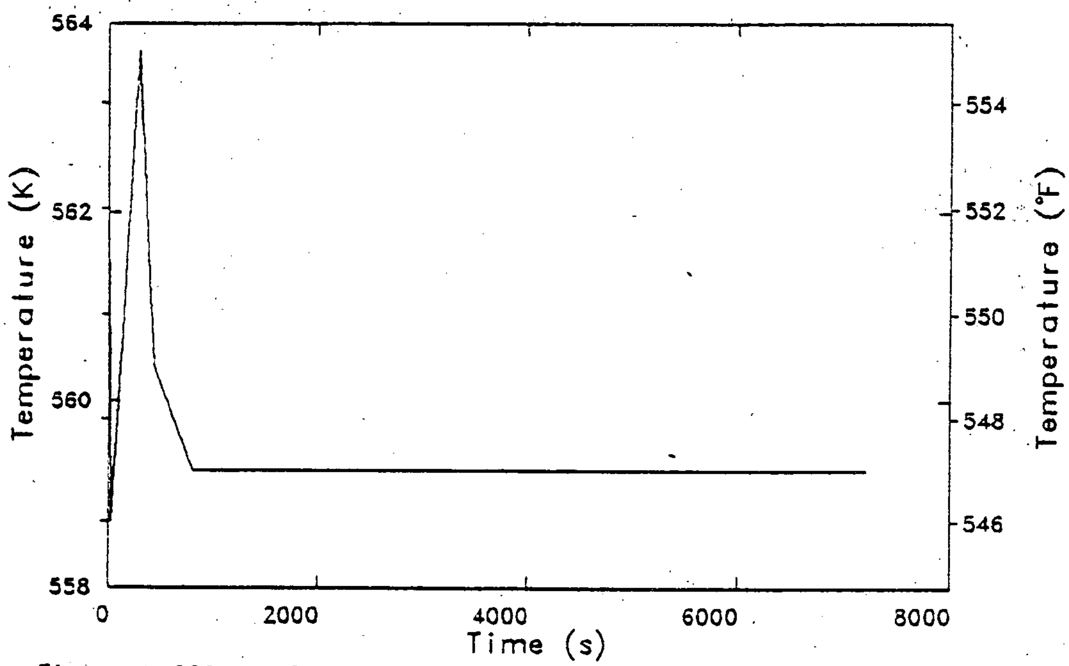

Figure A-266. Sequence 9-50 reactor vessel downcomer fluid temperature.

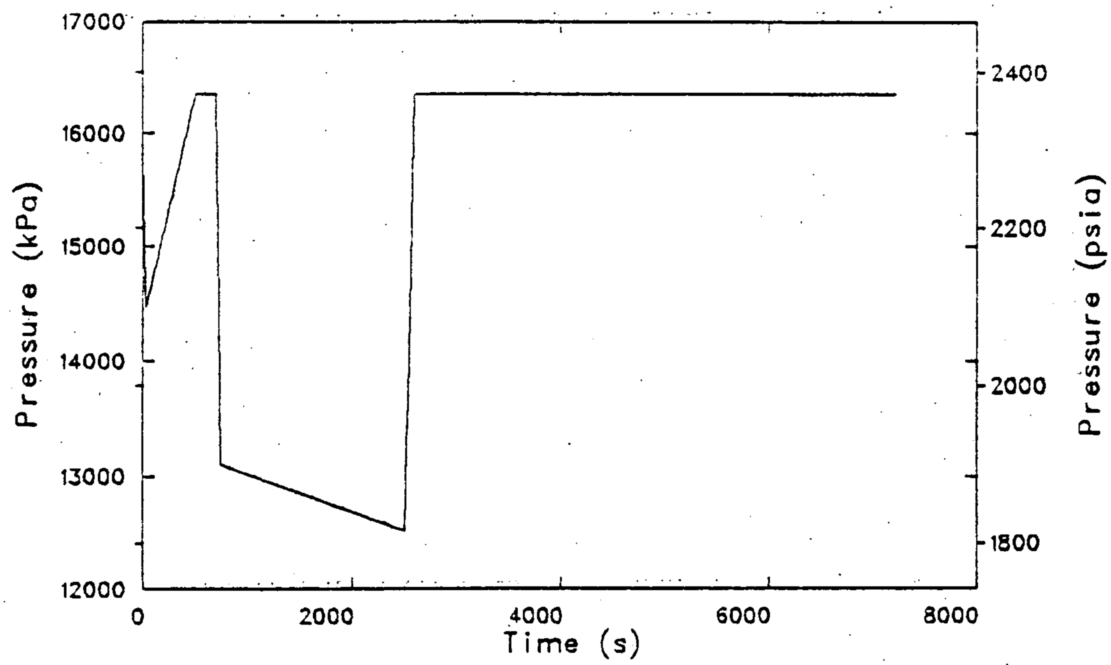

Figure A-267. Sequence 9-51 reactor vessel downcomer fluid pressure.

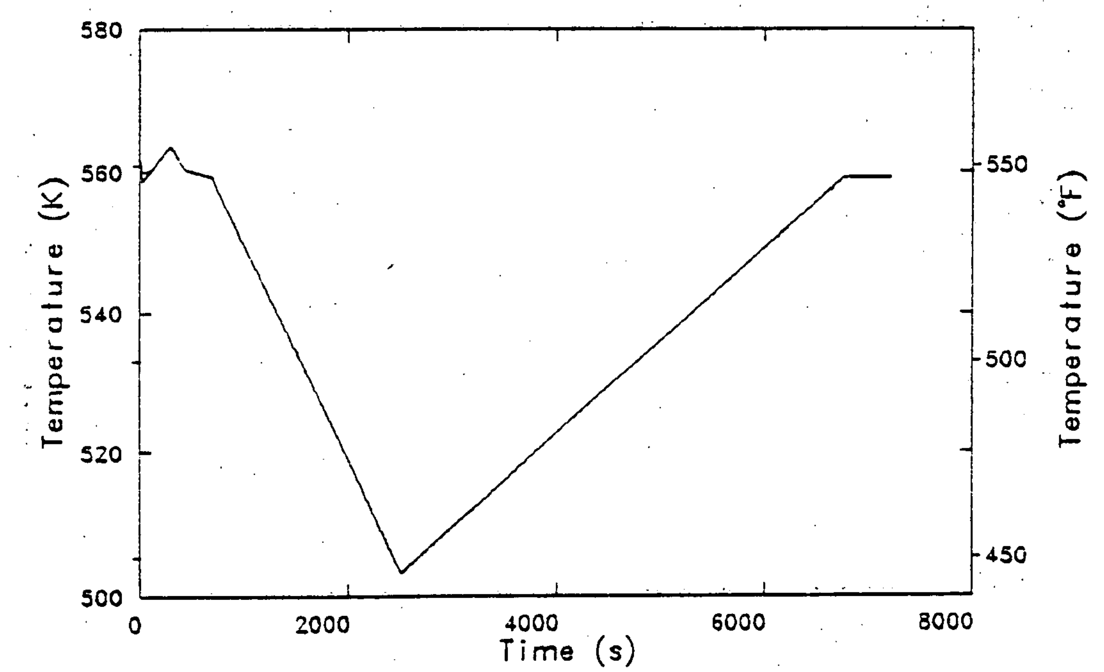

Figure A-268. Sequence 9-51 reactor vessel downcomer fluid temperature.

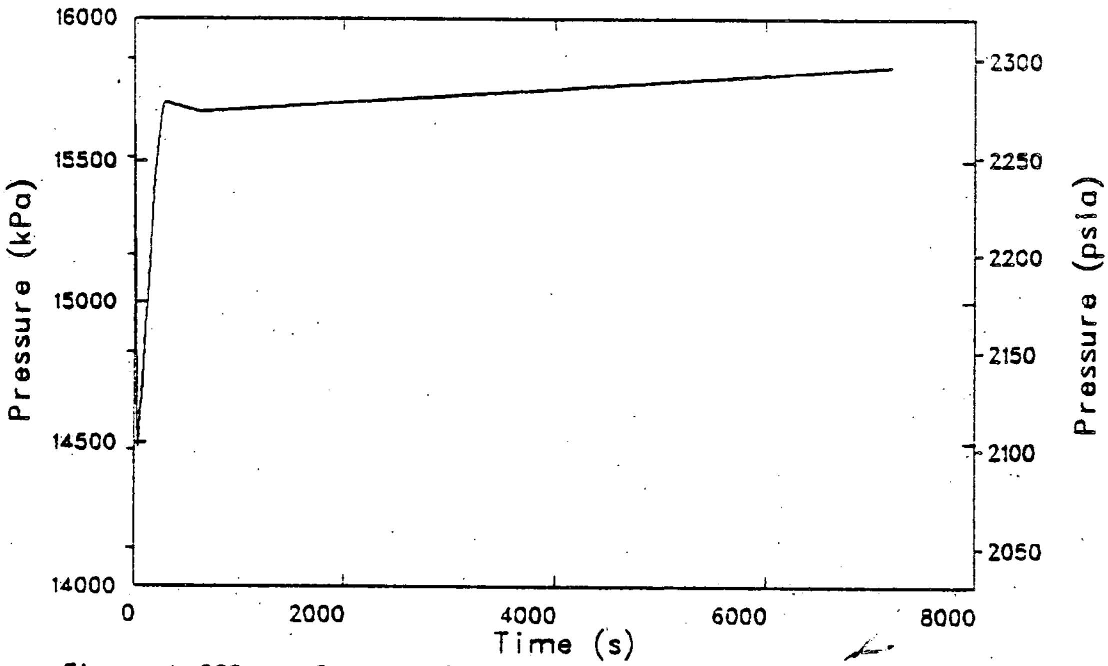

Figure A-269. Sequence 9-52 reactor vessel downcomer fluid pressure.

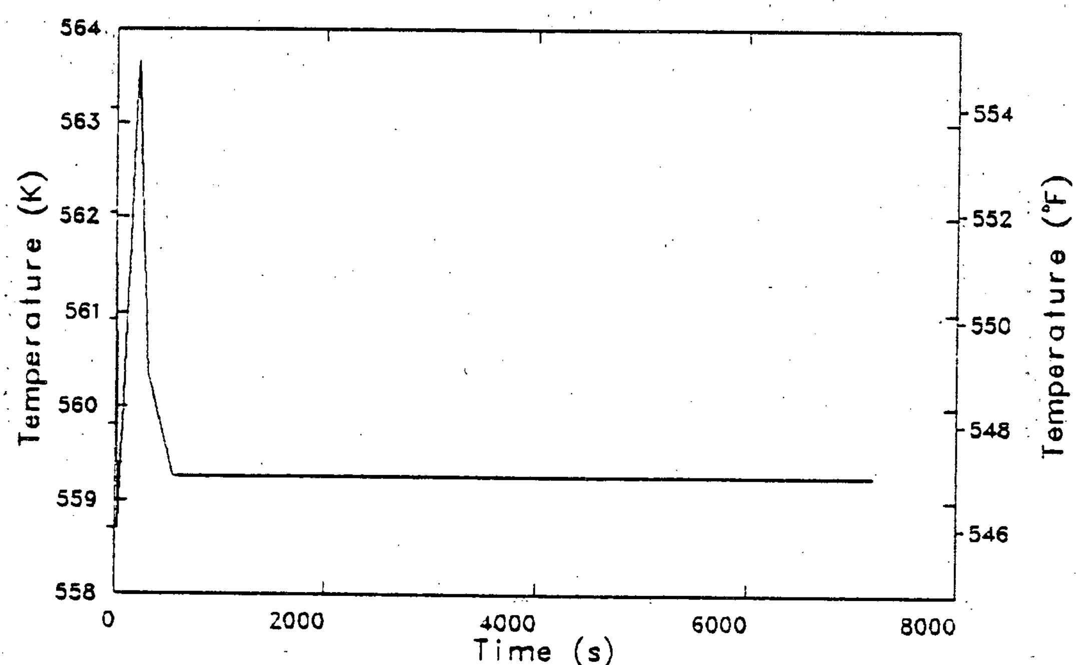

Figure A-270. Sequence 9-52 reactor vessel downcomer fluid temperature.

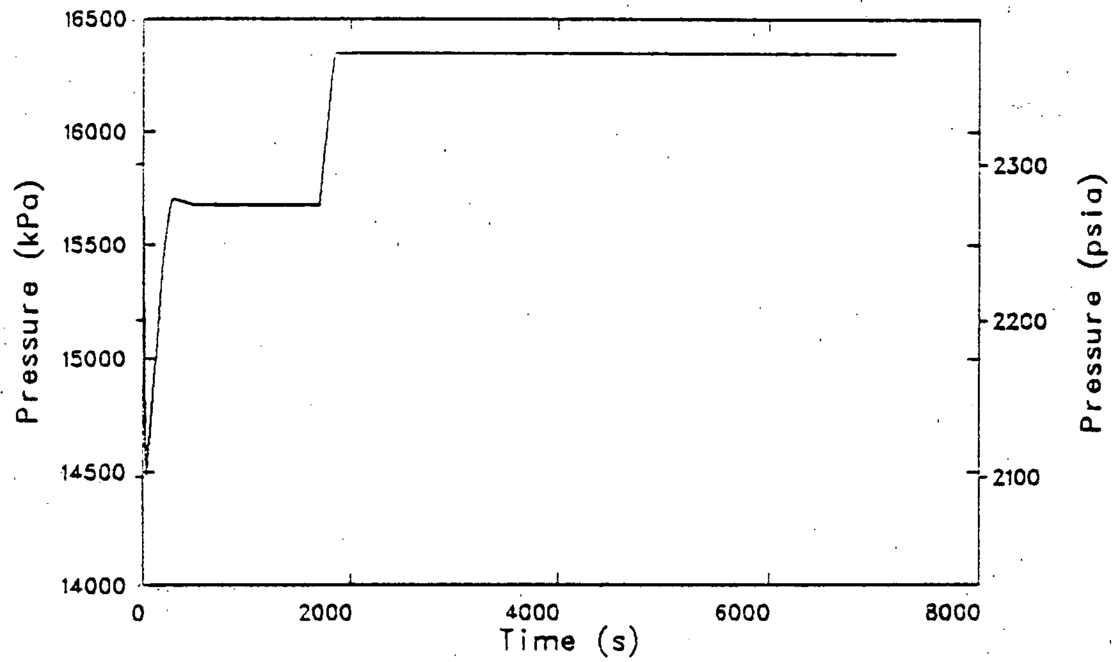

Figure A-271. Sequence 9-53 reactor vessel downcomer fluid pressure.

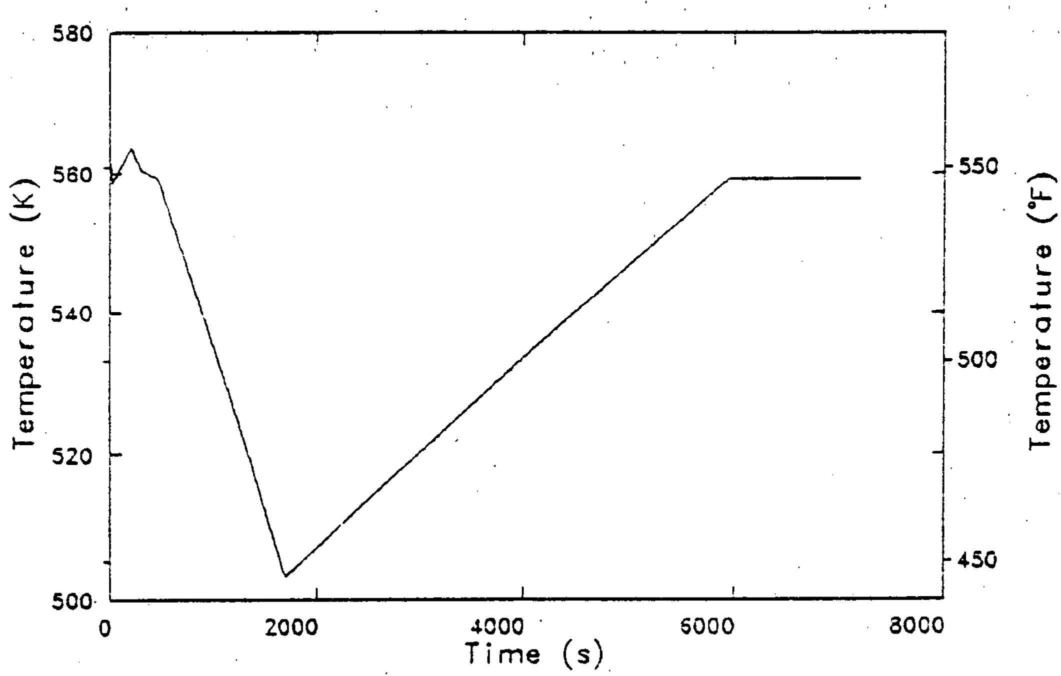

Figure A-272. Sequence 9-53 reactor vessel downcomer fluid temperature.

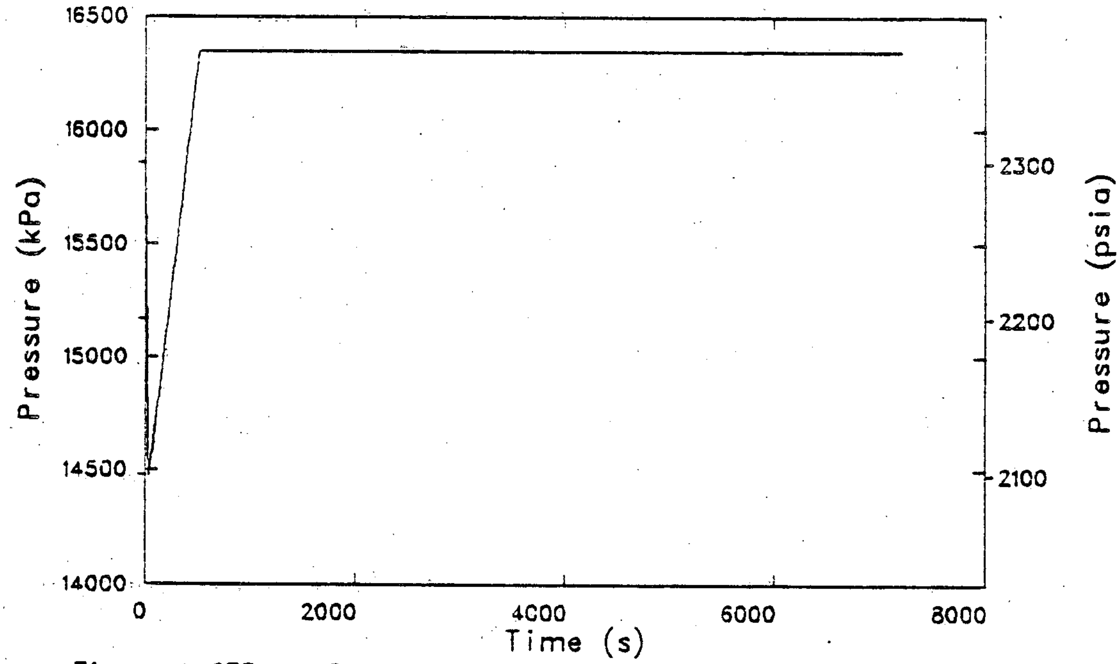

Figure A-273. Sequence 9-54 reactor vessel downcomer fluid pressure.

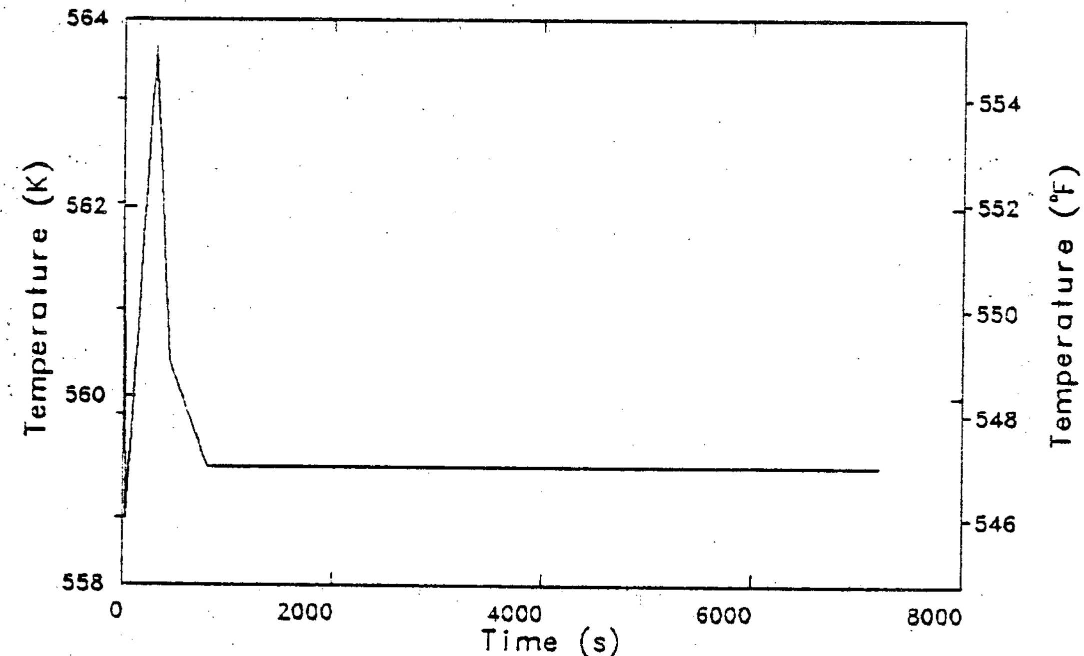

Figure A-274. Sequence 9-54 reactor vessel downcomer fluid temperature.

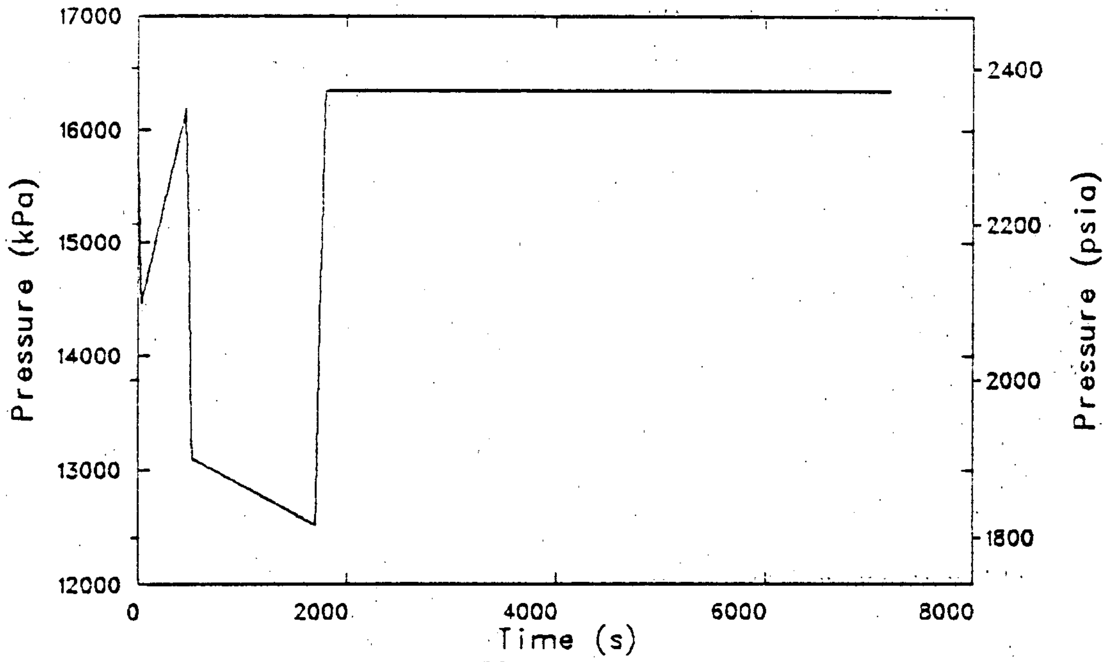

Figure A-275. Sequence 9-55 reactor vessel downcomer fluid pressure.

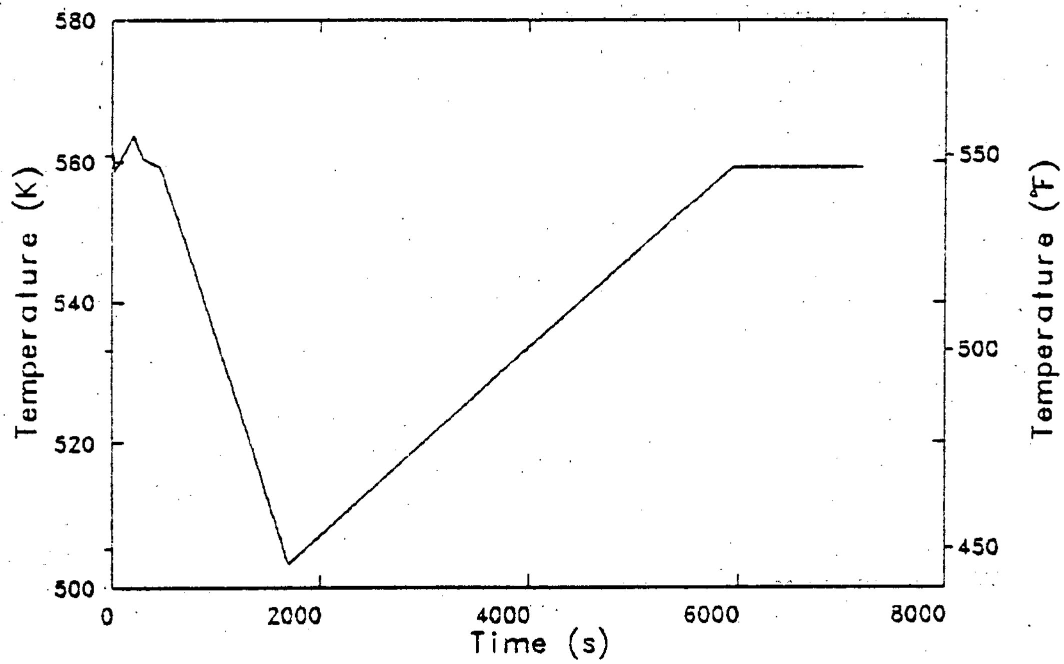

Figure A-276. Sequence 9-55 reactor vessel downcomer fluid temperature.

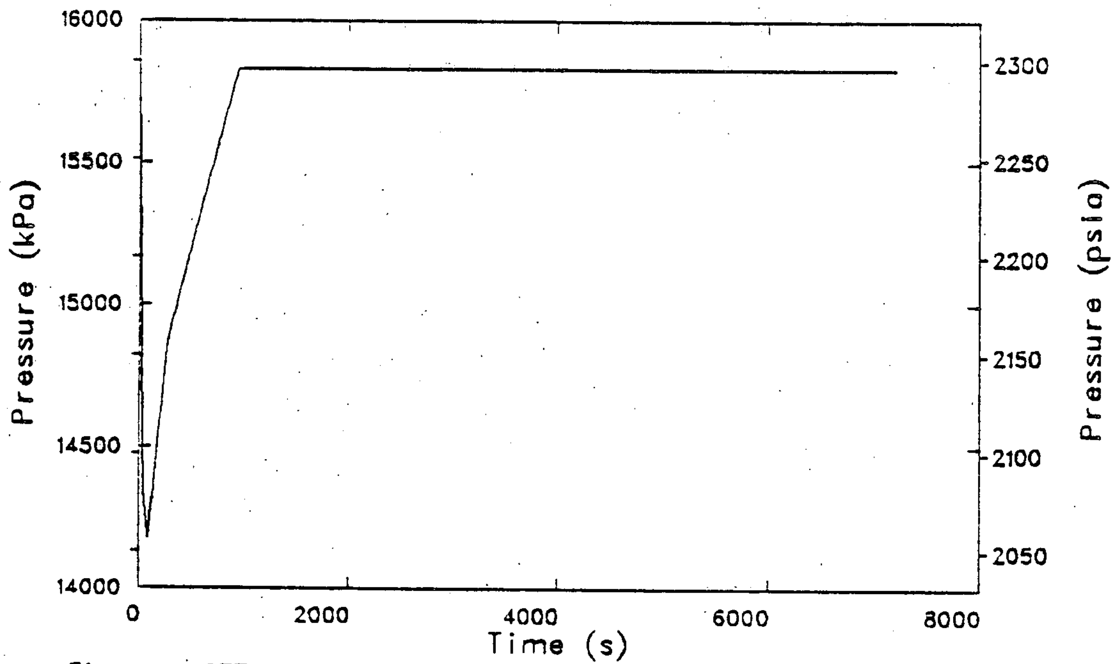

Figure A-277. Sequence 9-56 reactor vessel downcomer fluid pressure.

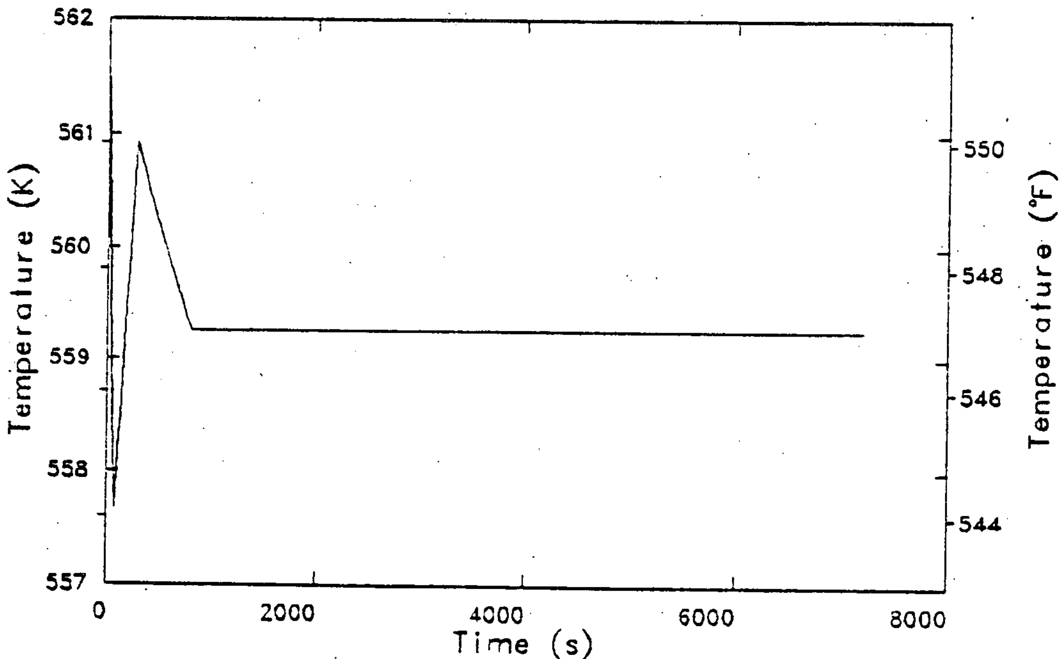

Figure A-278. Sequence 9-56 reactor vessel downcomer fluid temperature.

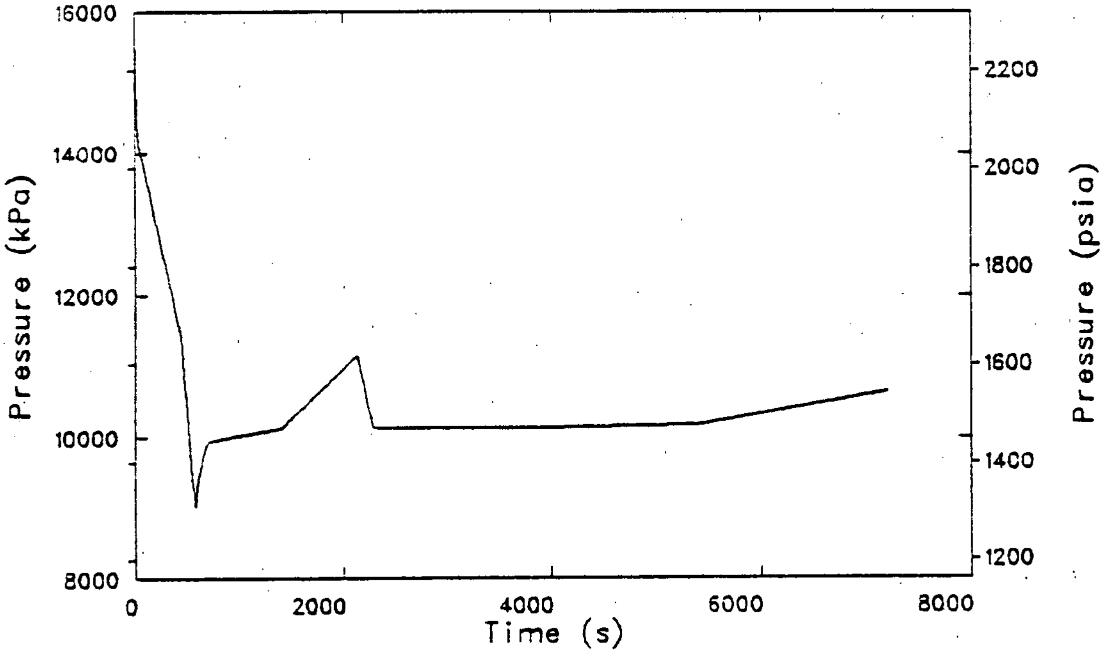

Figure A-279. Sequence 9-57 reactor vessel downcomer fluid pressure.

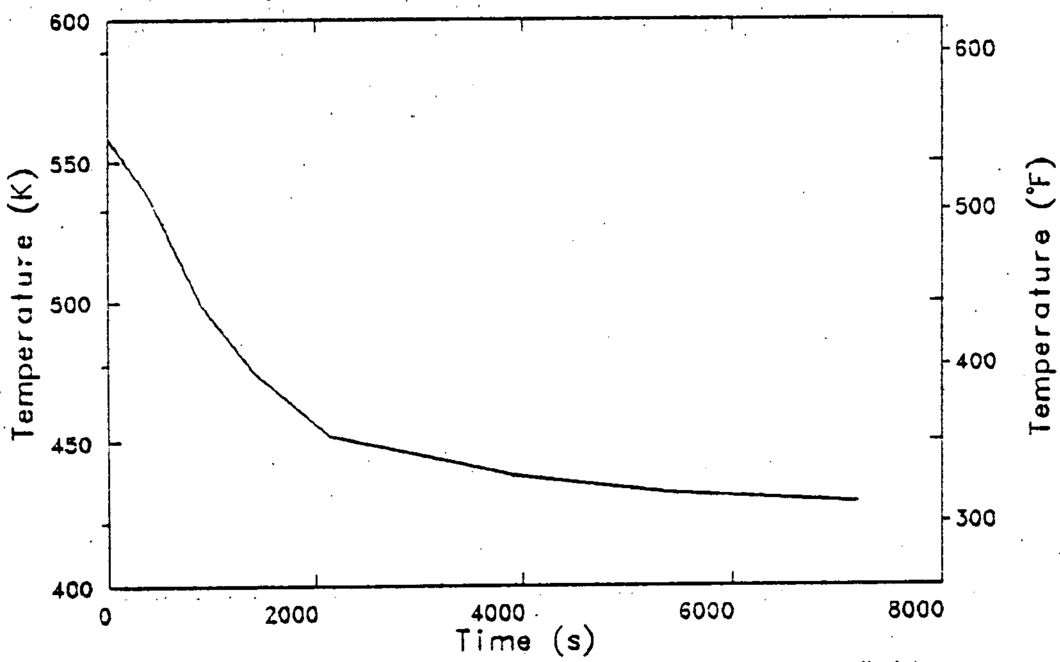

Figure A-280. Sequence 9-57 reactor vessel downcomer fluid temperature.

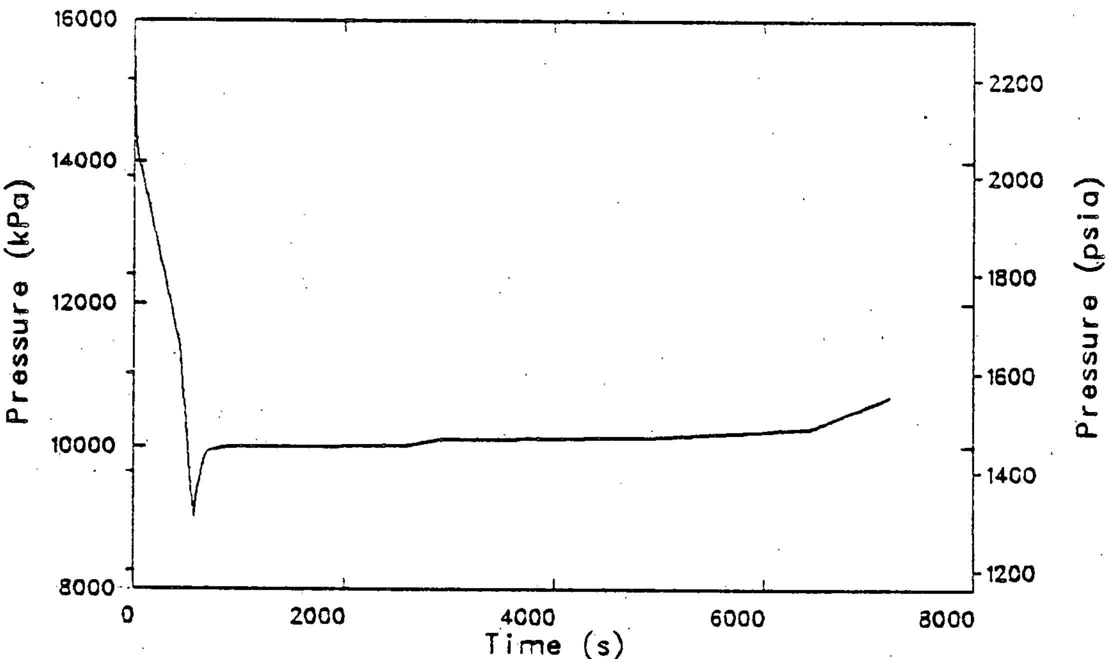

Figure A-281. Sequence 9-58 reactor vessel downcomer fluid pressure.

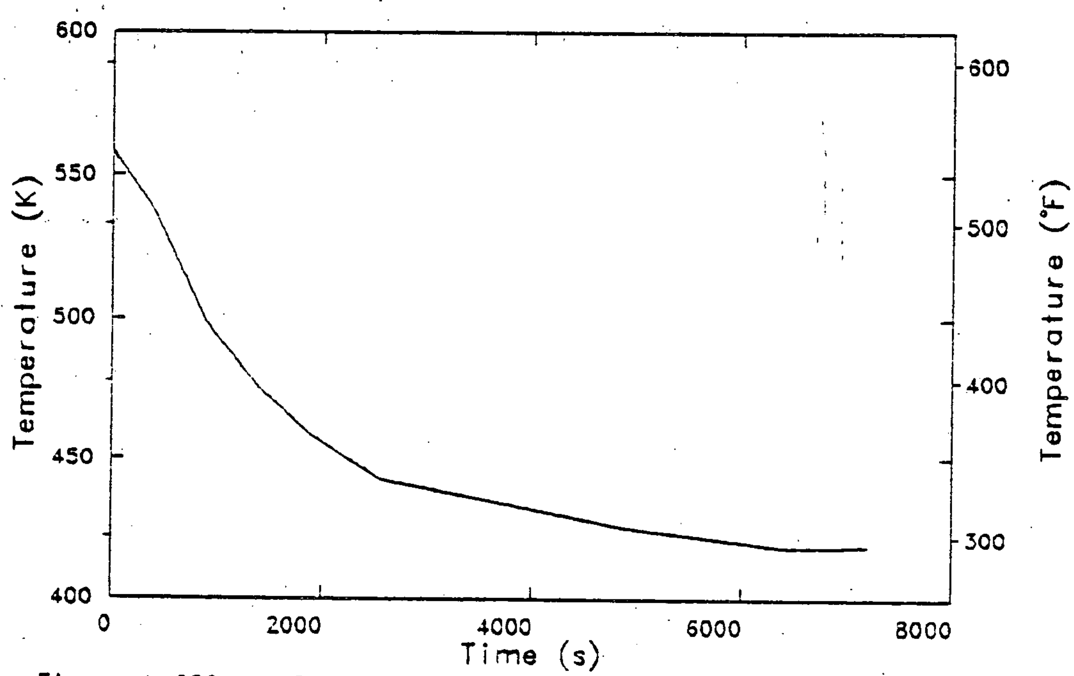

Figure A-282. Sequence 9-58 reactor vessel downcomer fluid temperature.

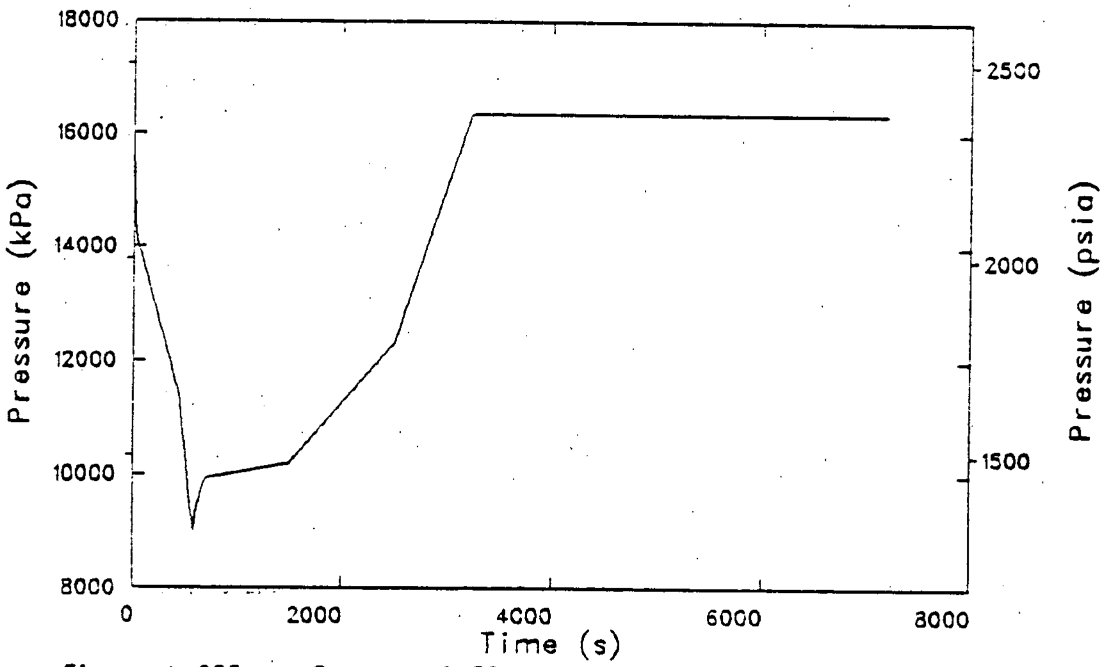

Figure A-283. Sequence 9-59 reactor vessel downcomer fluid pressure.

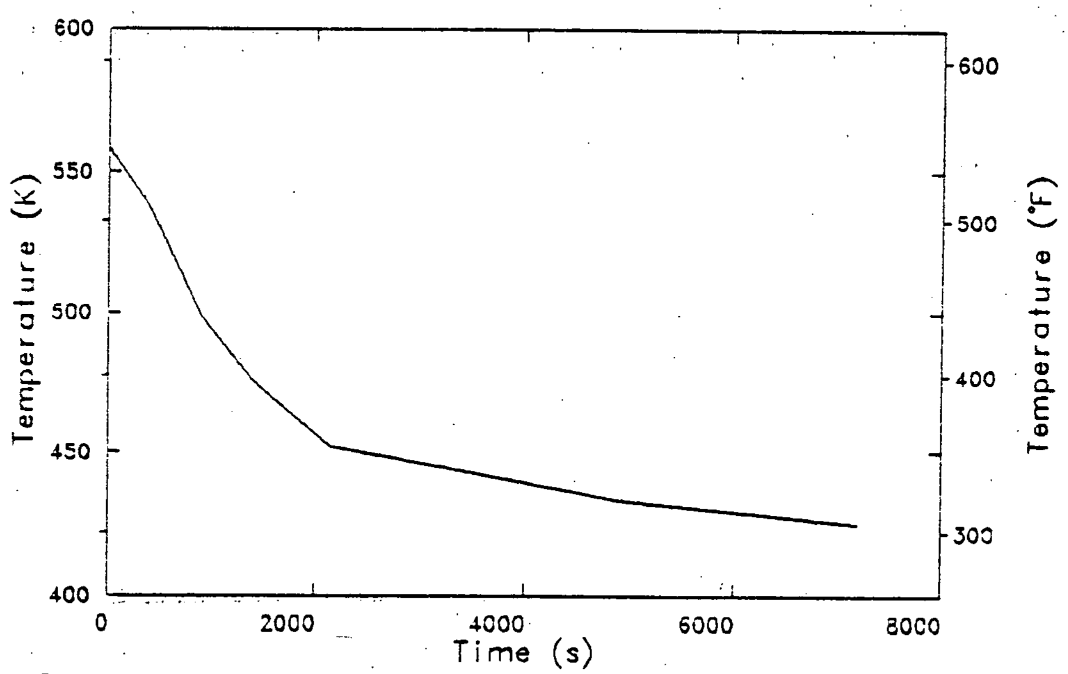

Figure A-284. Sequence 9-59 reactor vessel downcomer fluid temperature.

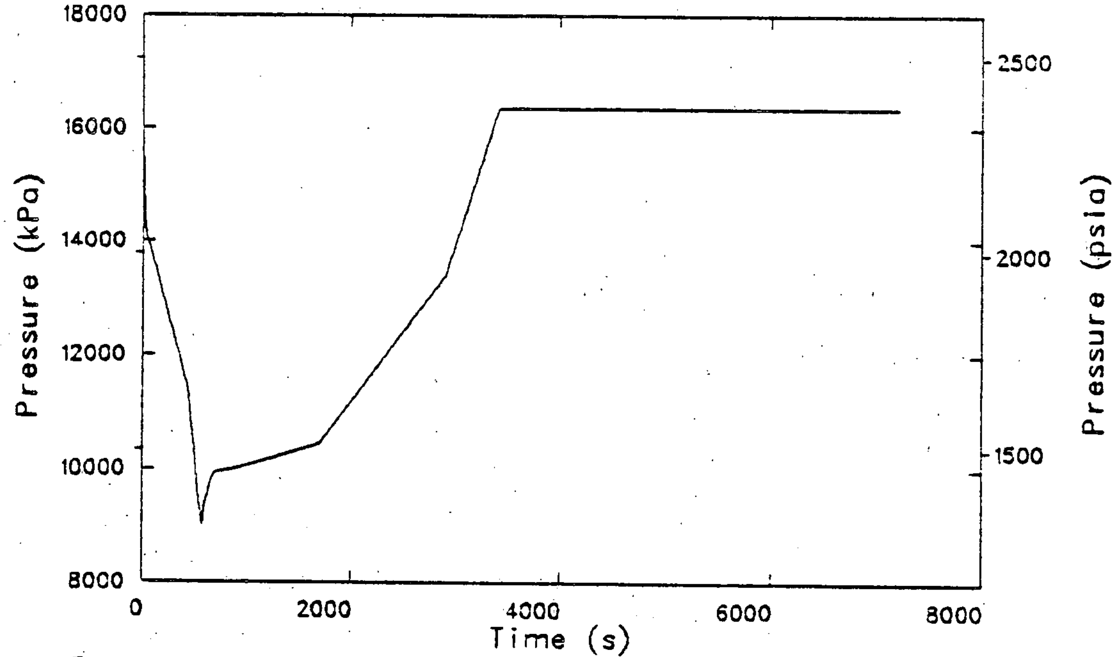

Figure A-285. Sequence 9-60 reactor vessel downcomer fluid pressure.

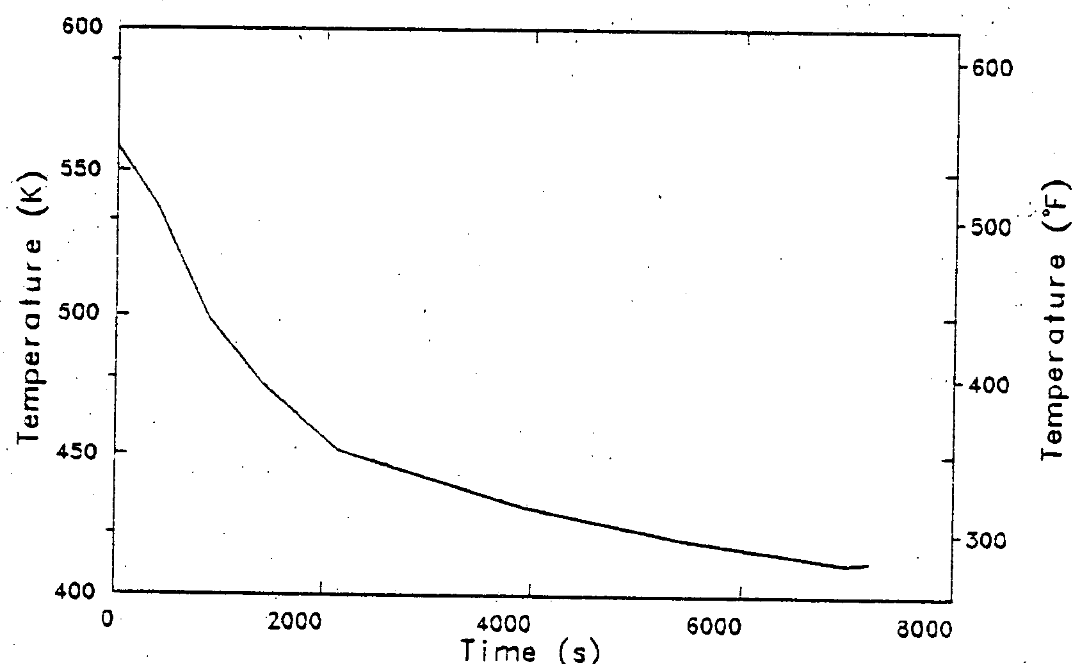

Figure A-286. Sequence 9-60 reactor vessel downcomer fluid temperature.

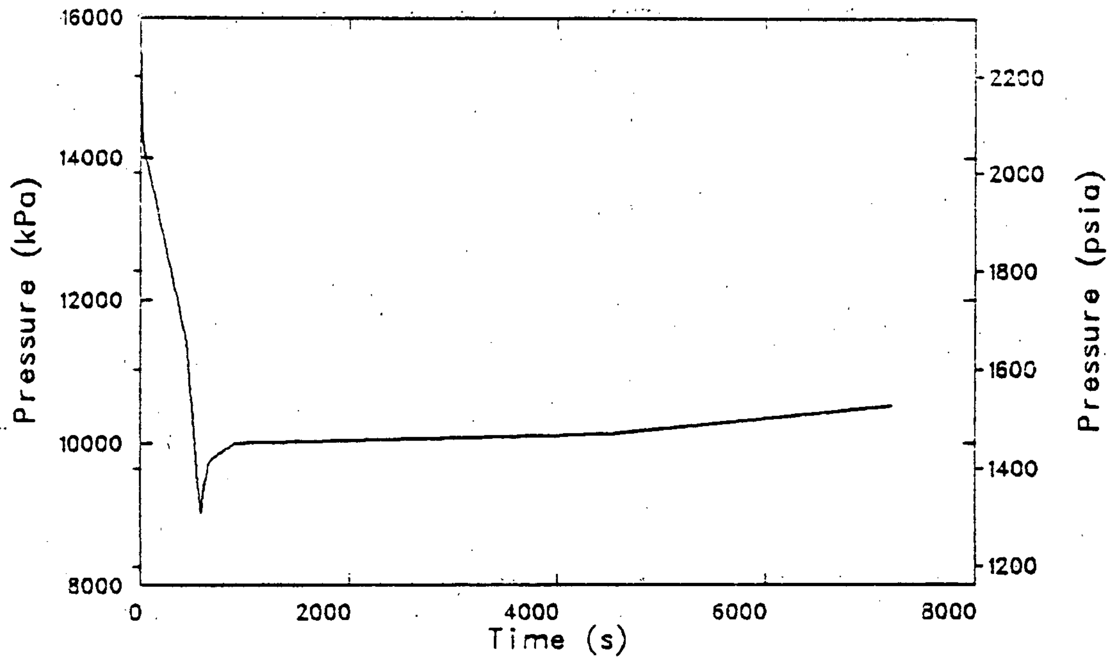

Figure A-287. Sequence 9-61 reactor vessel downcomer fluid pressure.

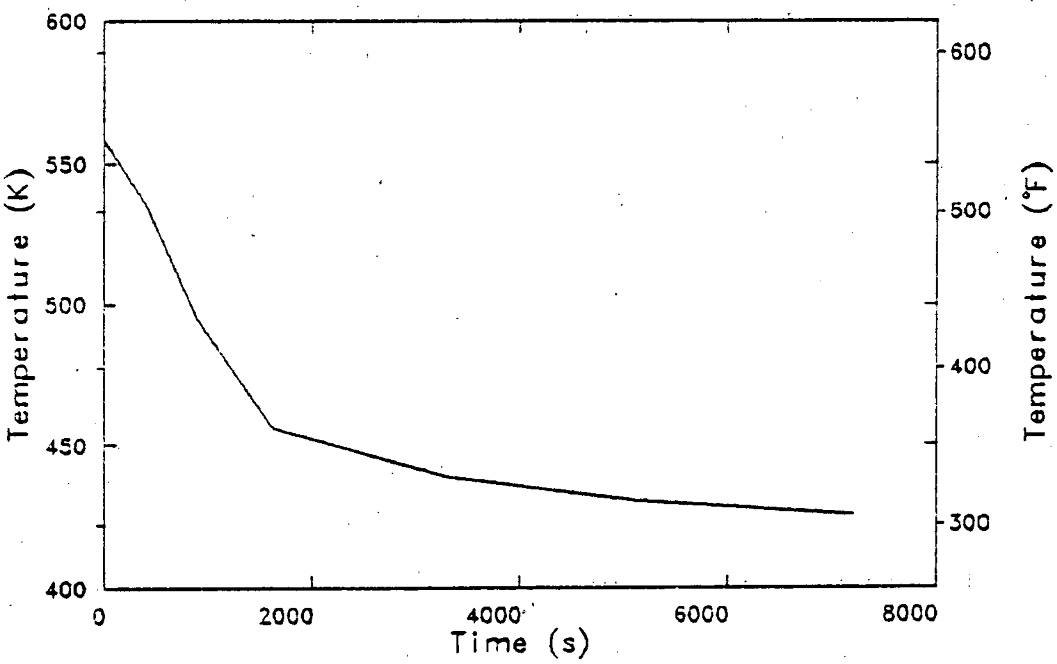

Figure A-288. Sequence 9-61 reactor vessel downcomer fluid temperature.

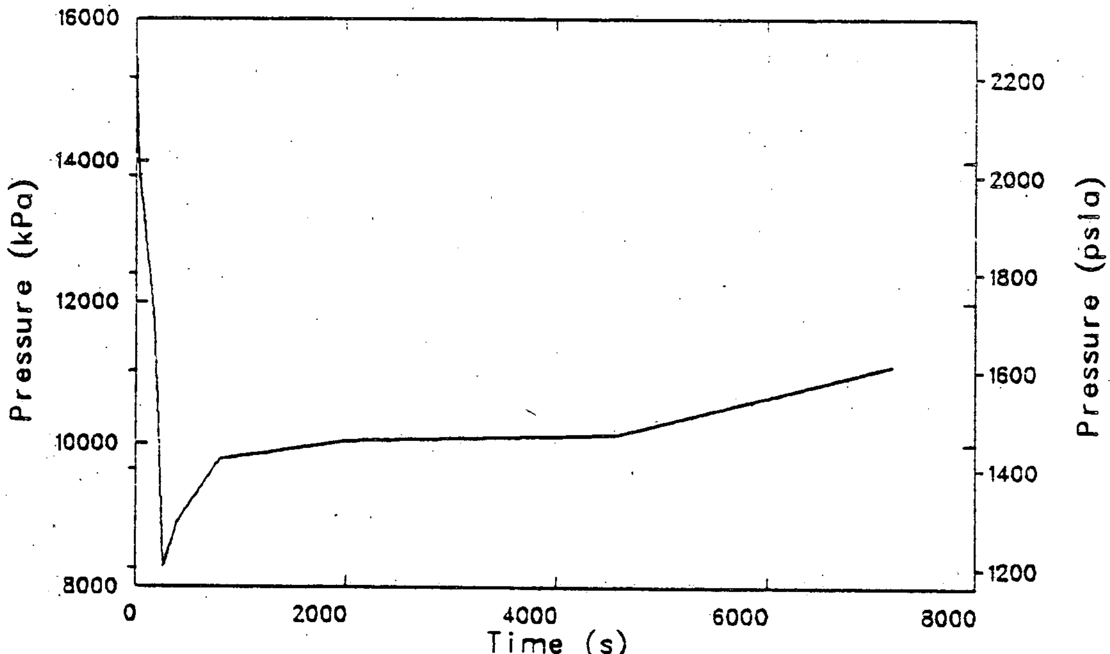

Figure A-289. Sequence 9-62 reactor vessel downcomer fluid pressure.

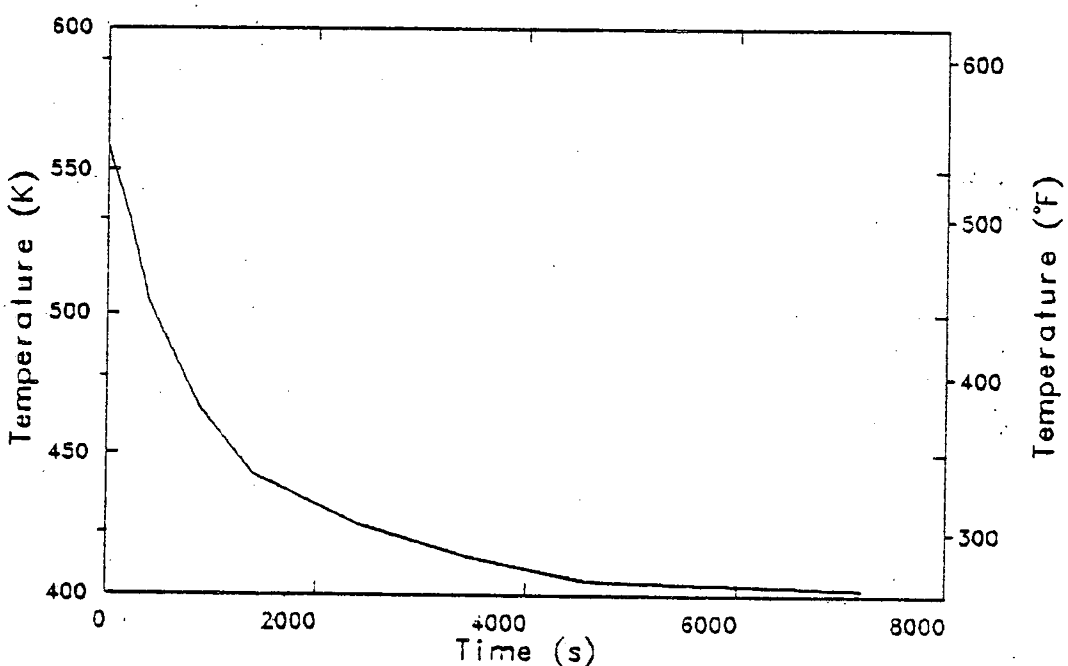

Figure A-290. Sequance 9-62 reactor vessel downcomer fluid temperature.

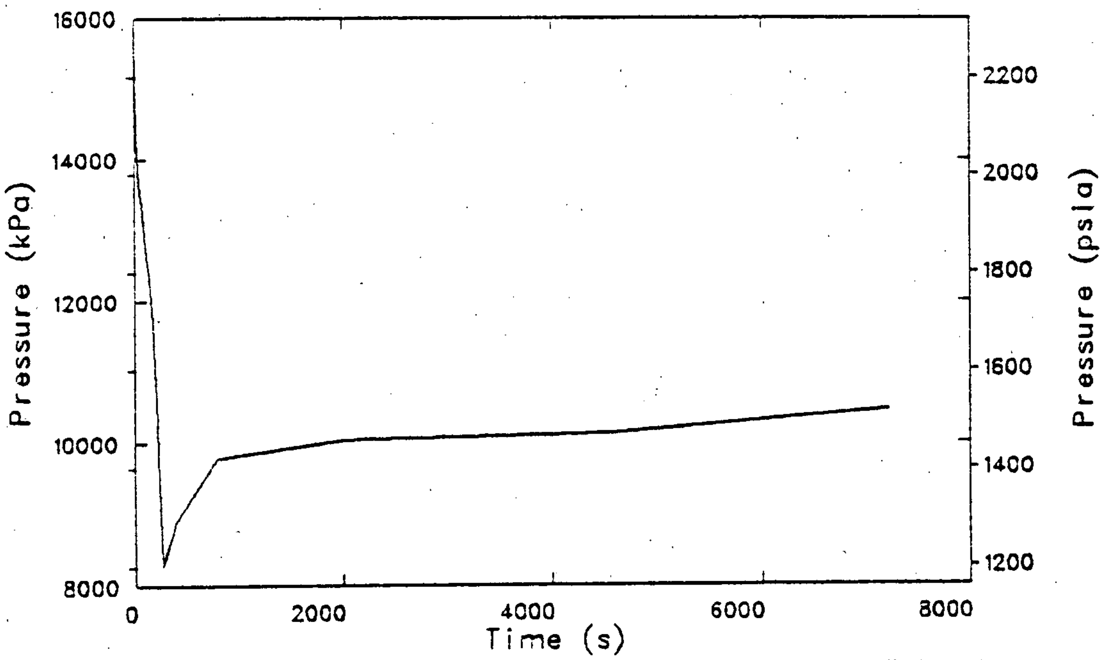

Figure A-291. Sequence 9-63 reactor vessel downcomer fluid pressure.

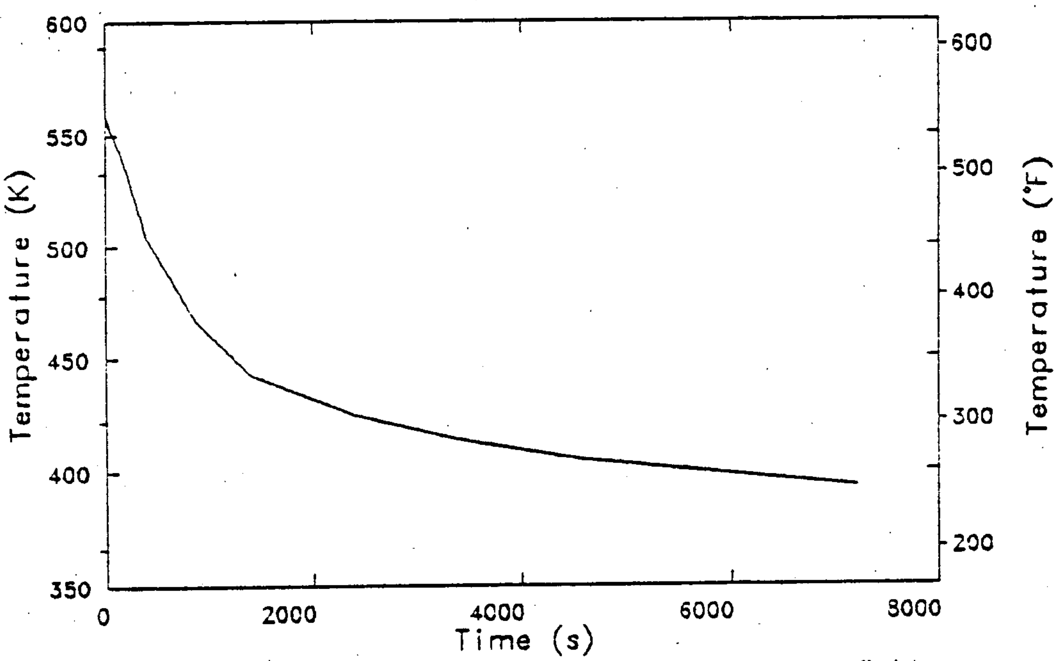

Figure A-292. Sequence 9-63 reactor vessel downcomer fluid temperature.

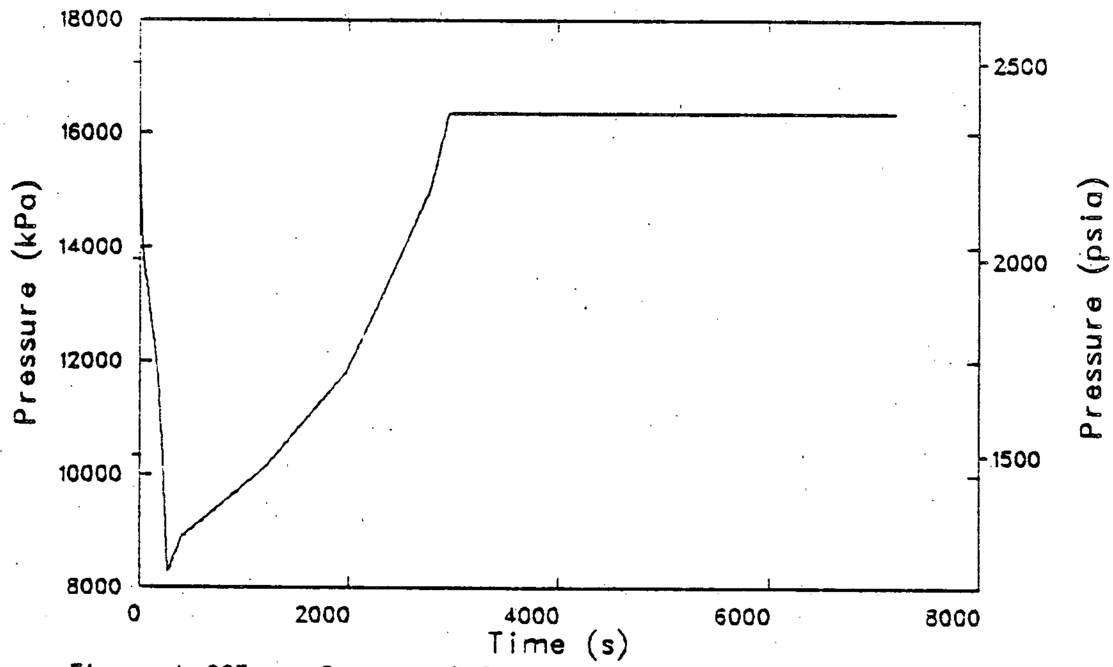

Figure A-293. Sequence 9-64 reactor vessel downcomer fluid pressure.

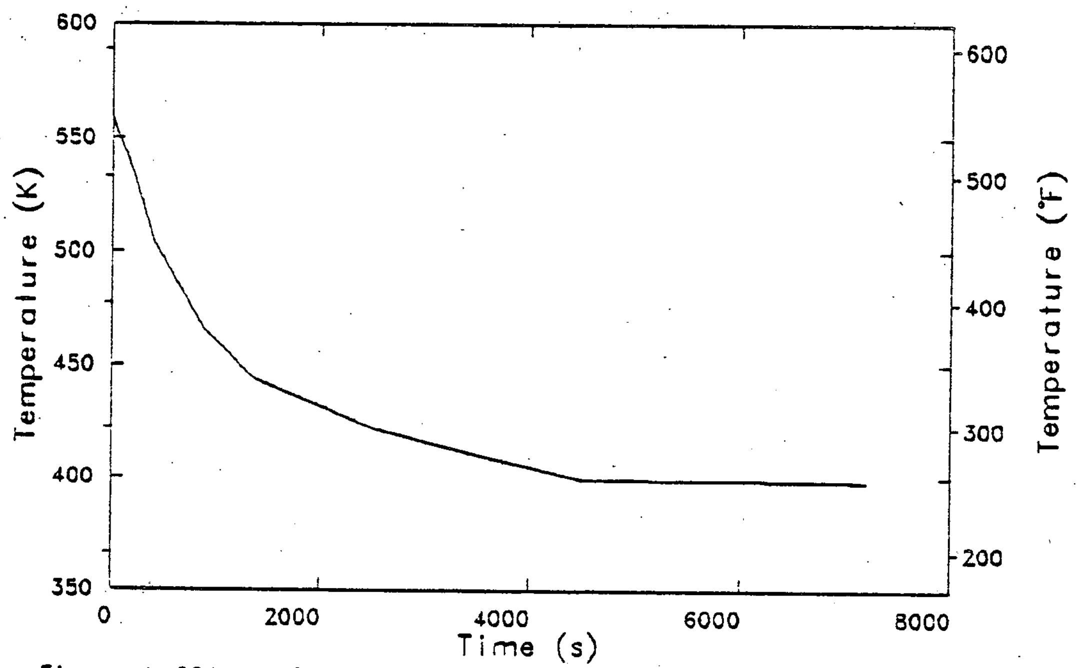

Figure A-294. Sequence 9-64 reactor vessel downcomer fluid temperature.

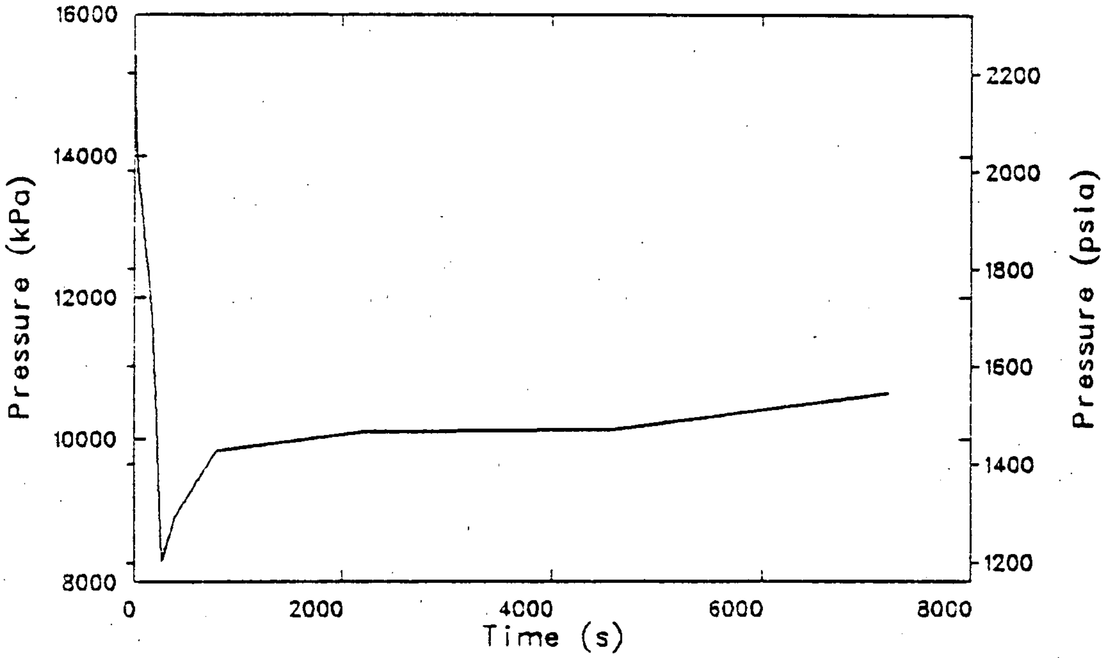

Figure A-295. Sequence 9-65 reactor vessel downcomer fluid pressure.

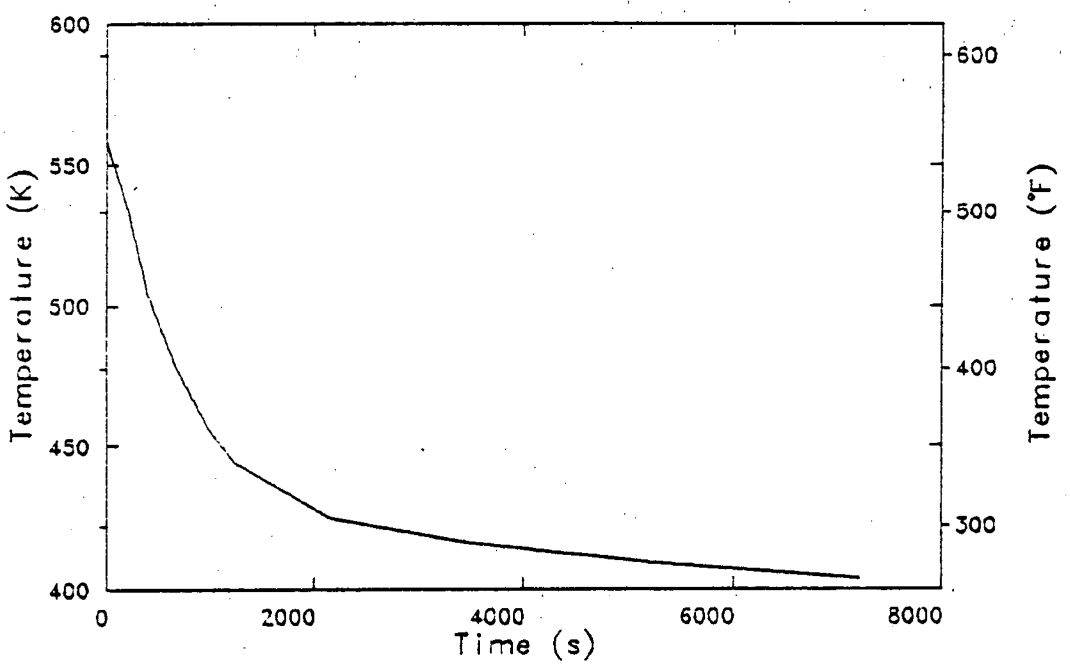

Figure A-296. Sequence 9-65 reactor vessel downcomer fluid temperature.

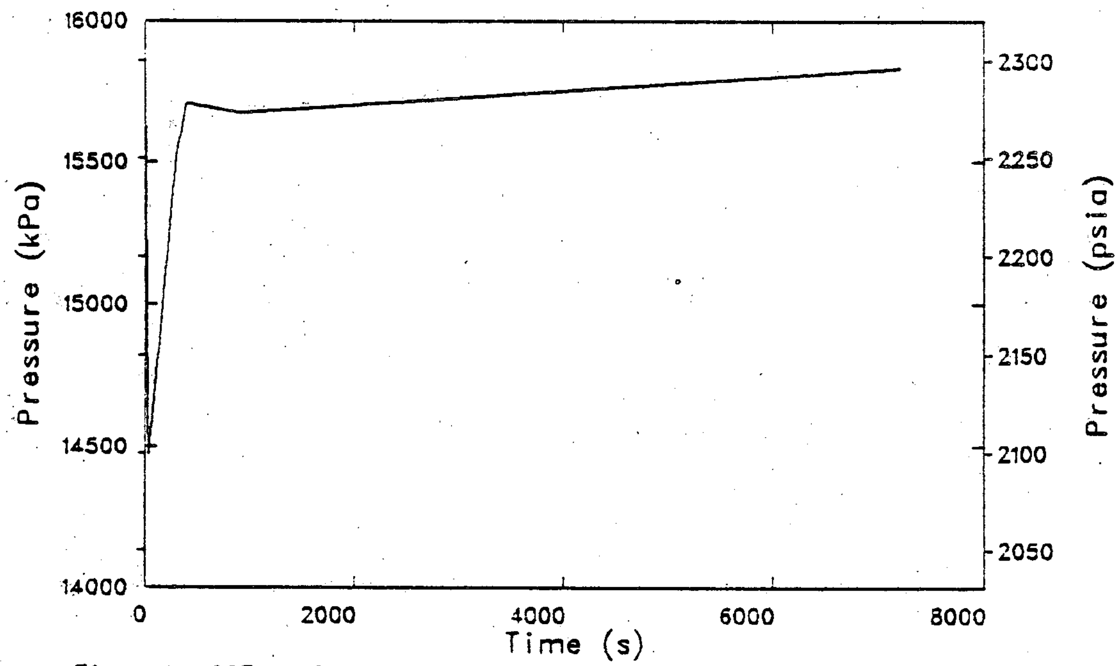

Figure A-297. Sequence 9-66 reactor vessel downcomer fluid pressure.

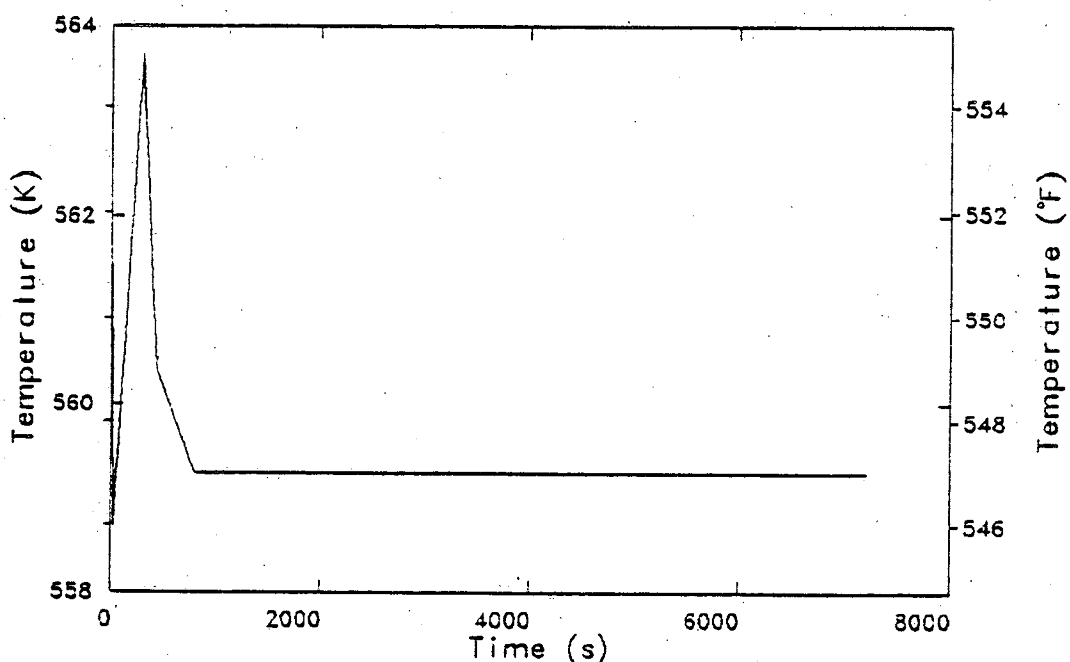

Figure A-298. Sequence 9-66 reactor vessel downcomer fluid temperature.

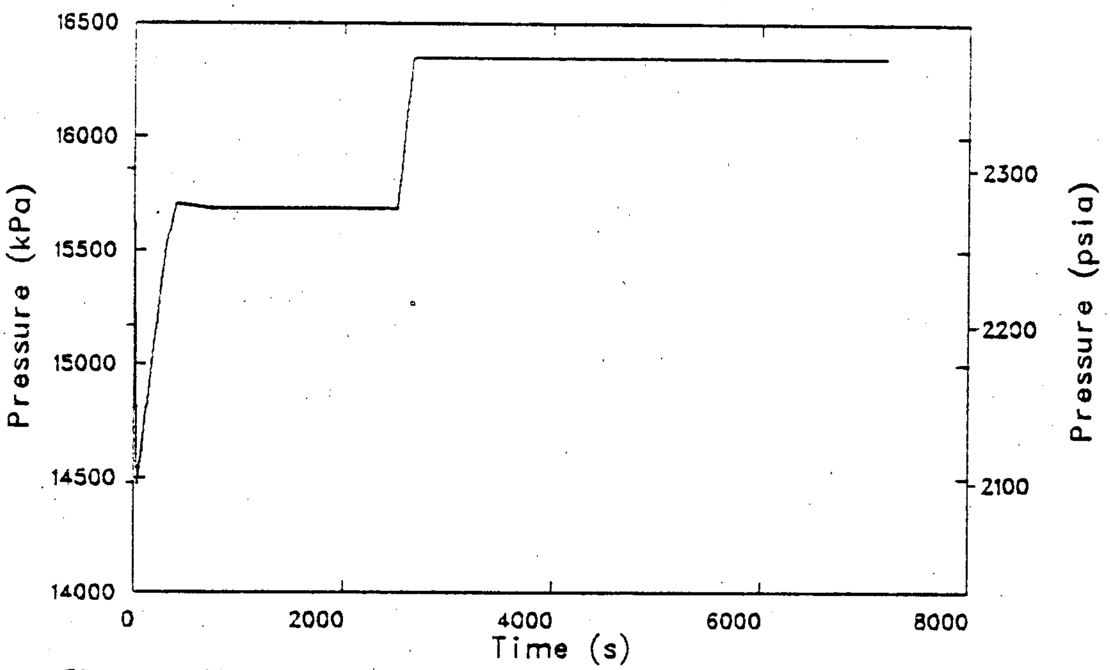

Figure A-299. Sequence 9-67 reactor vessel downcomer fluid pressure.

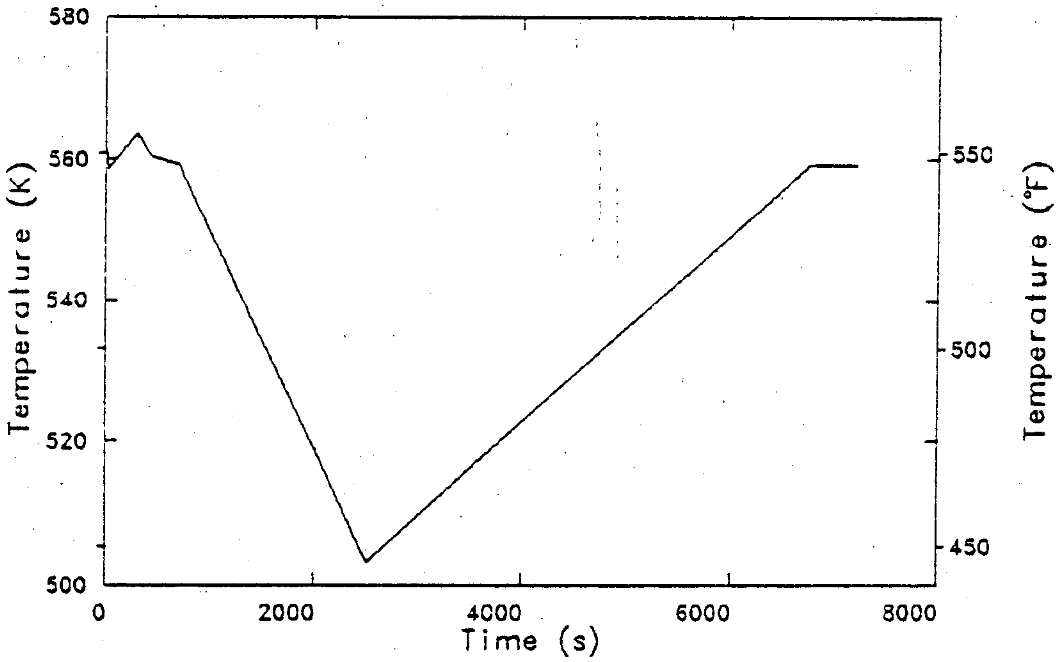

Figure A-300. Sequence 9-67 reactor vessel downcomer fluid temperature.

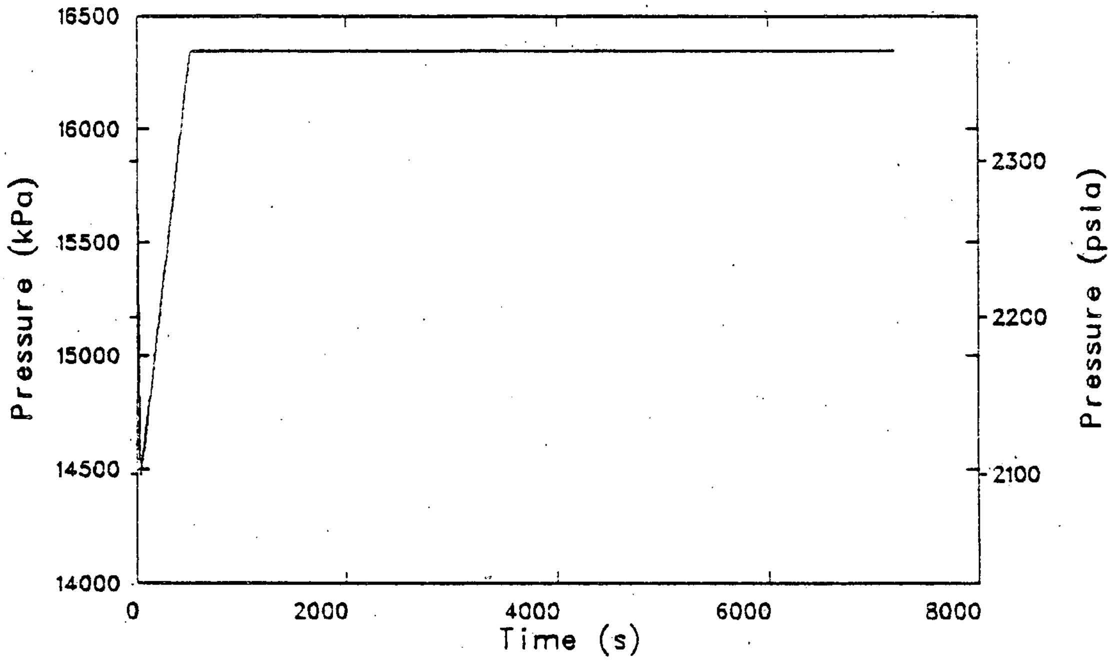

Figure A-301. Sequence 9-68 reactor vessel downcomer fluid pressure.

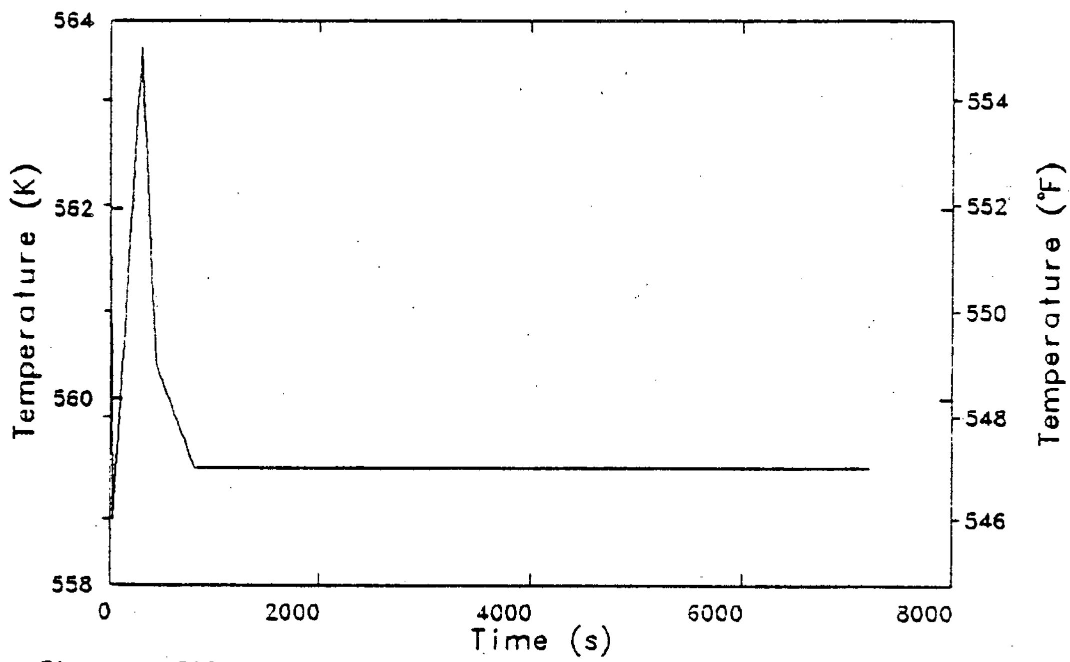

Figure A-302. Sequence 9-68 reactor vessel downcomer fluid temperature.

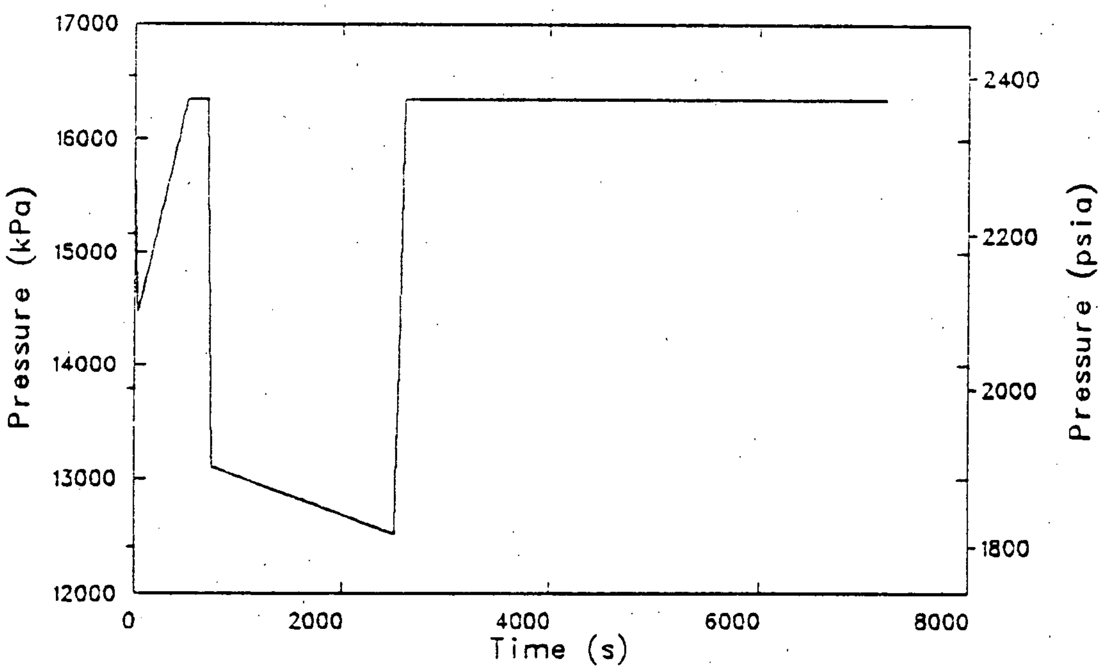

Figure A-303. Sequence 9-69 reactor vessel downcomer fluid pressure.

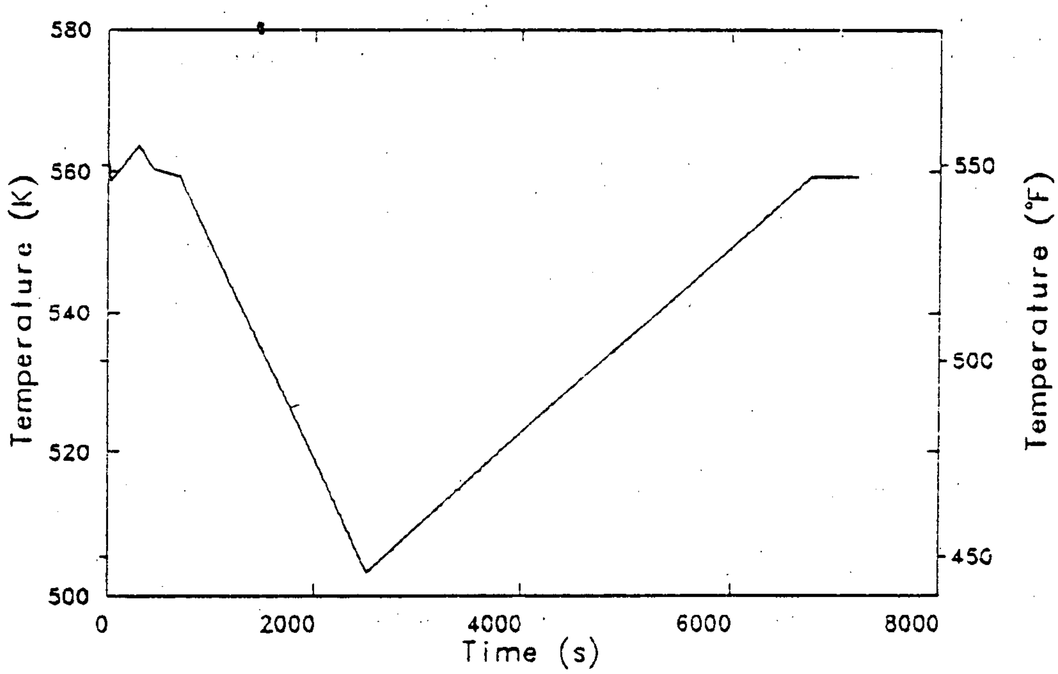

Figure A-304. Sequence 9-69 reactor vessel downcomer fluid temperature.

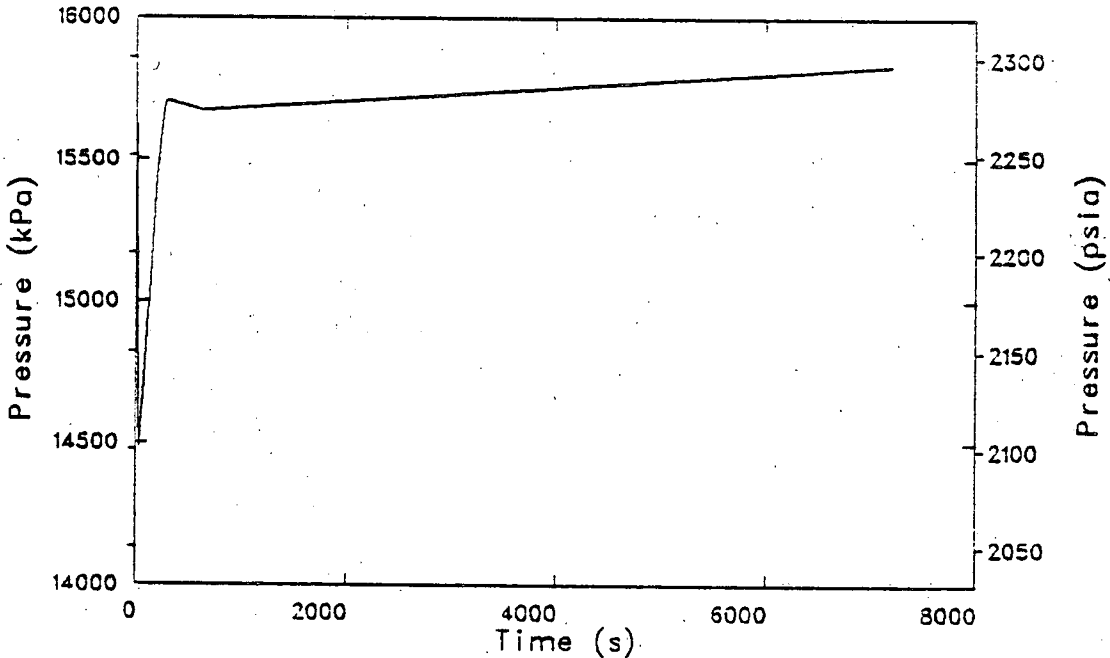

Figure A-305. Sequence 9-70 reactor vessel downcomer fluid pressure.

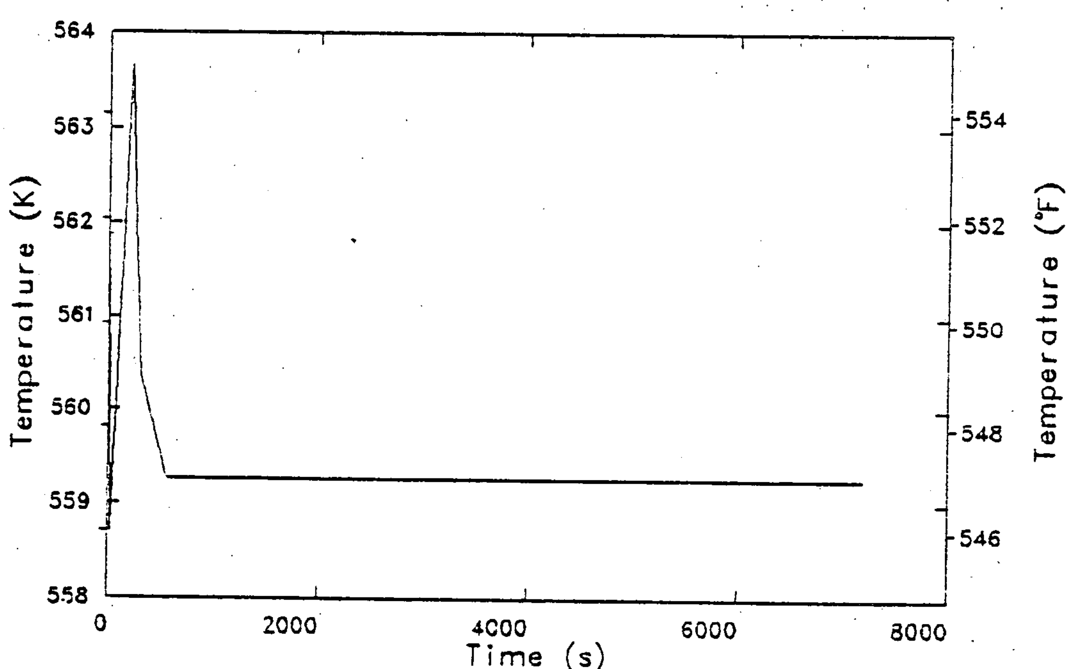

Figure A-306. Sequenca 9-70 reactor vessel downcomer fluid temperature.

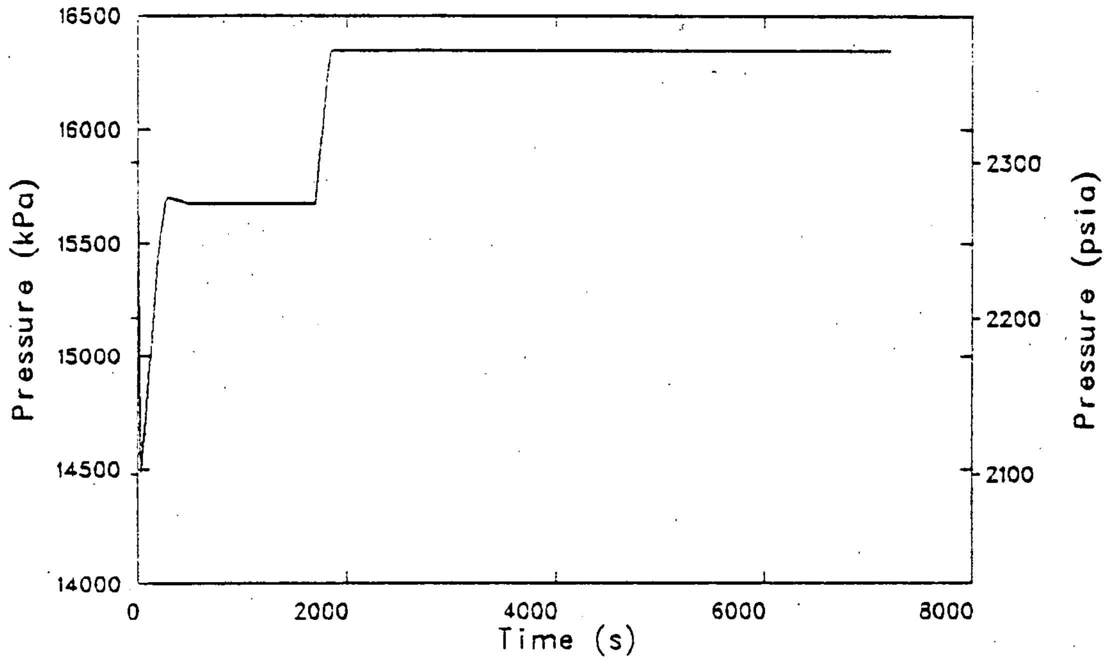

Figure A-307. Sequence 9-71 reactor vessel downcomer fluid pressure.

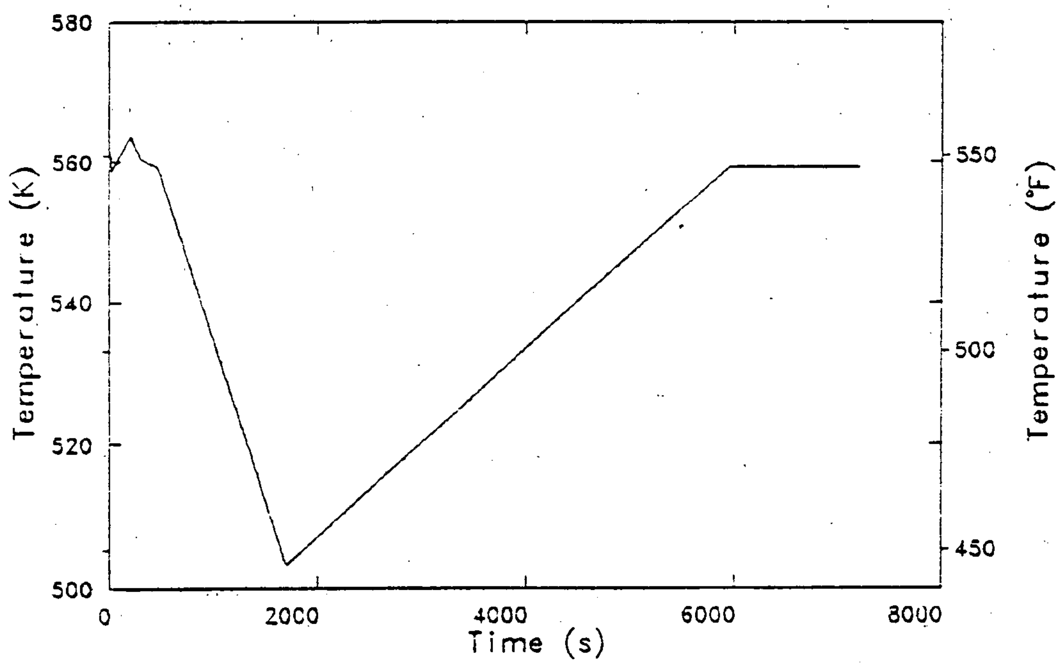

Figure A-308. Sequence 9-71 reactor vessel downcomer fluid temperature.

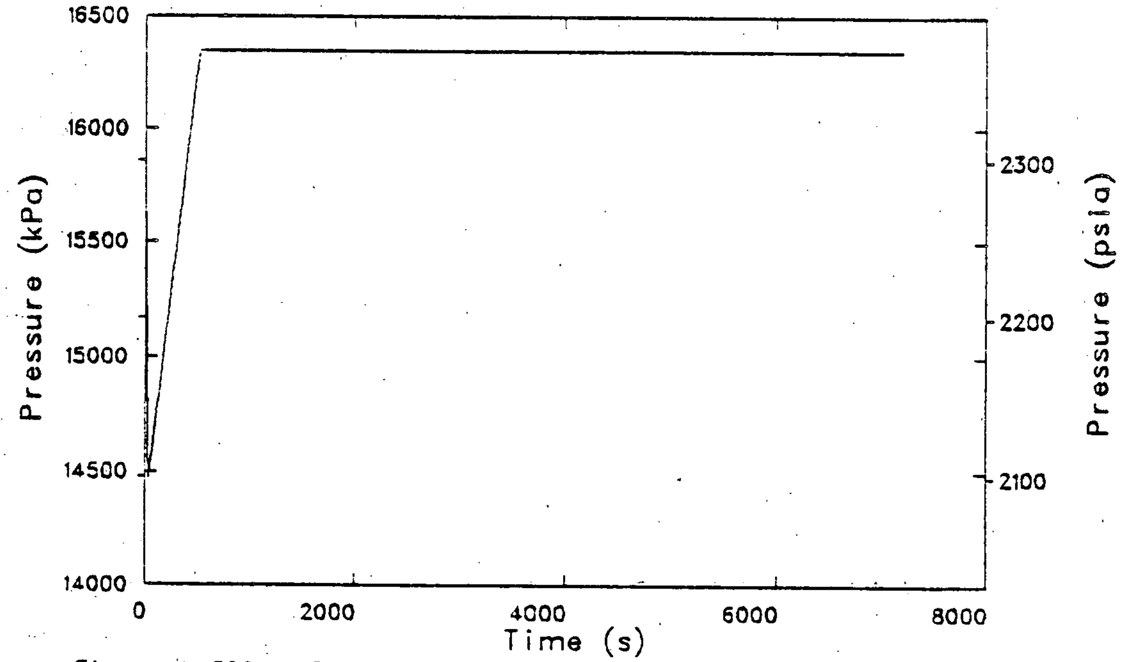

Figure A-309. Sequence 9-72 reactor vessel downcomer fluid pressure.

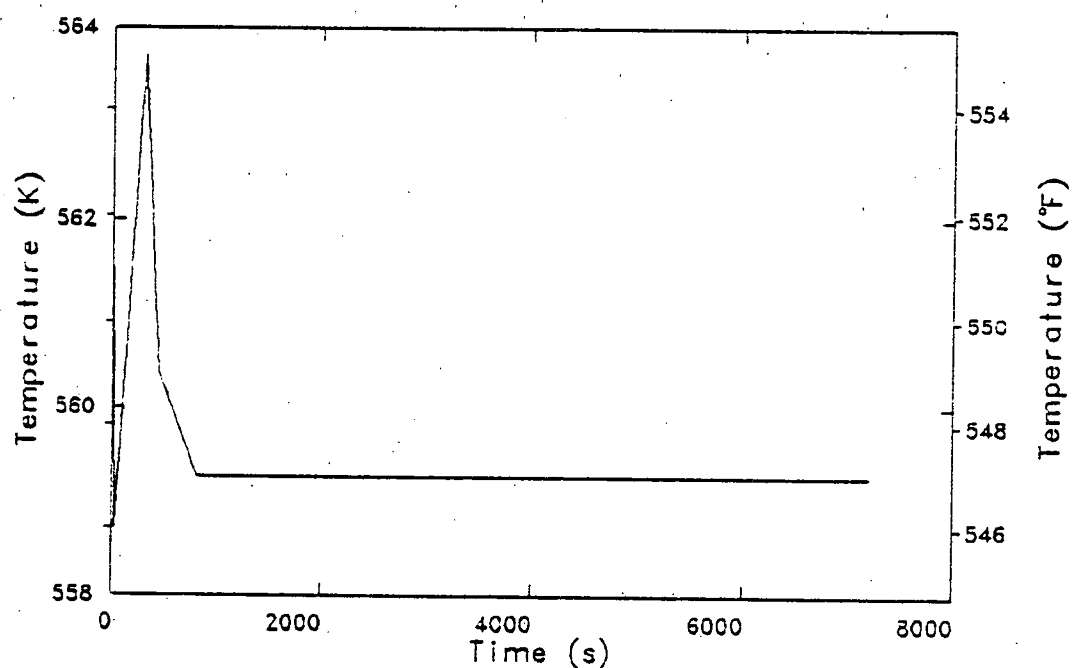

Figure A-310. Sequence 9-72 reactor vessel downcomer fluid temperature.

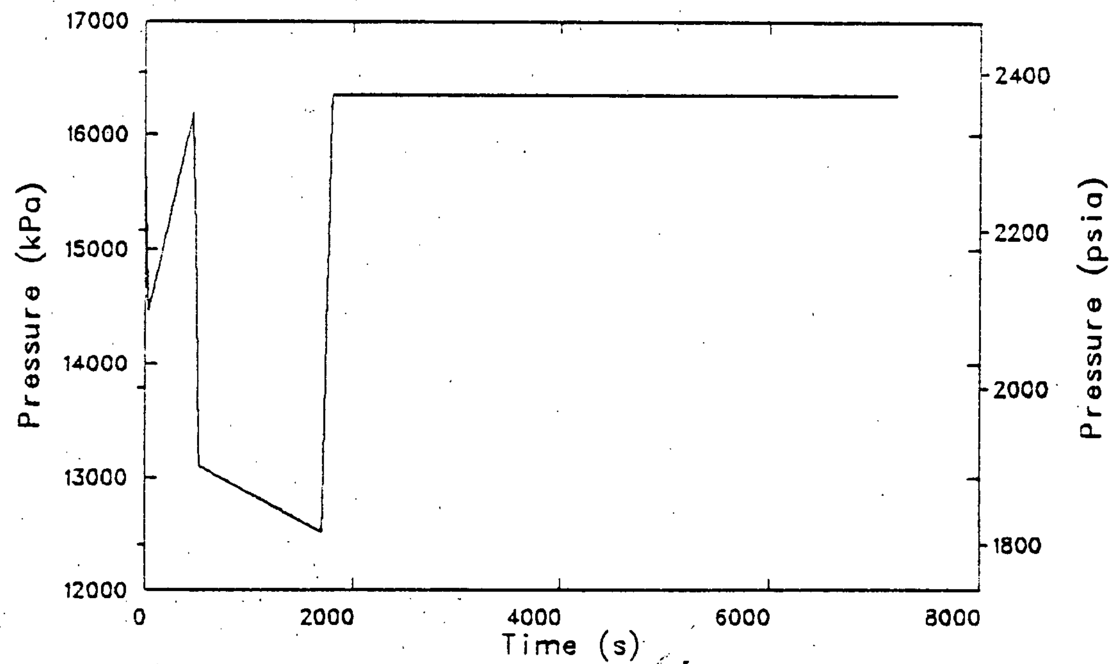

Figure A-311. Sequence 9-73 reactor vessel downcomer fluid pressure.

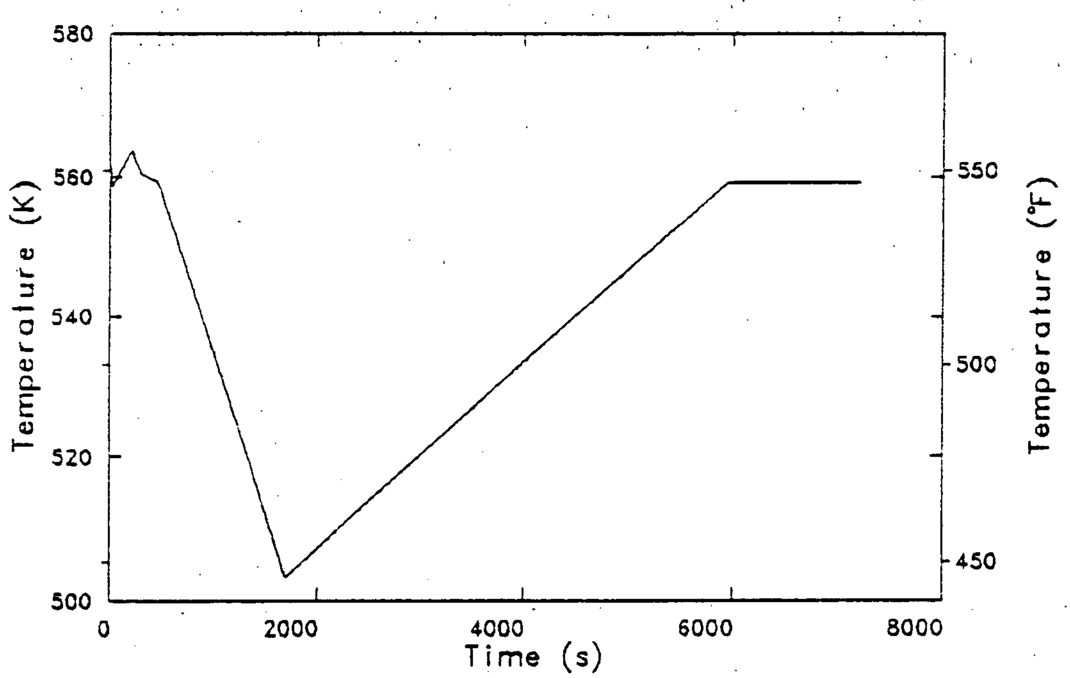

Figure A-312. Sequence 9-73 reactor vessel downcomer fluid temperature.

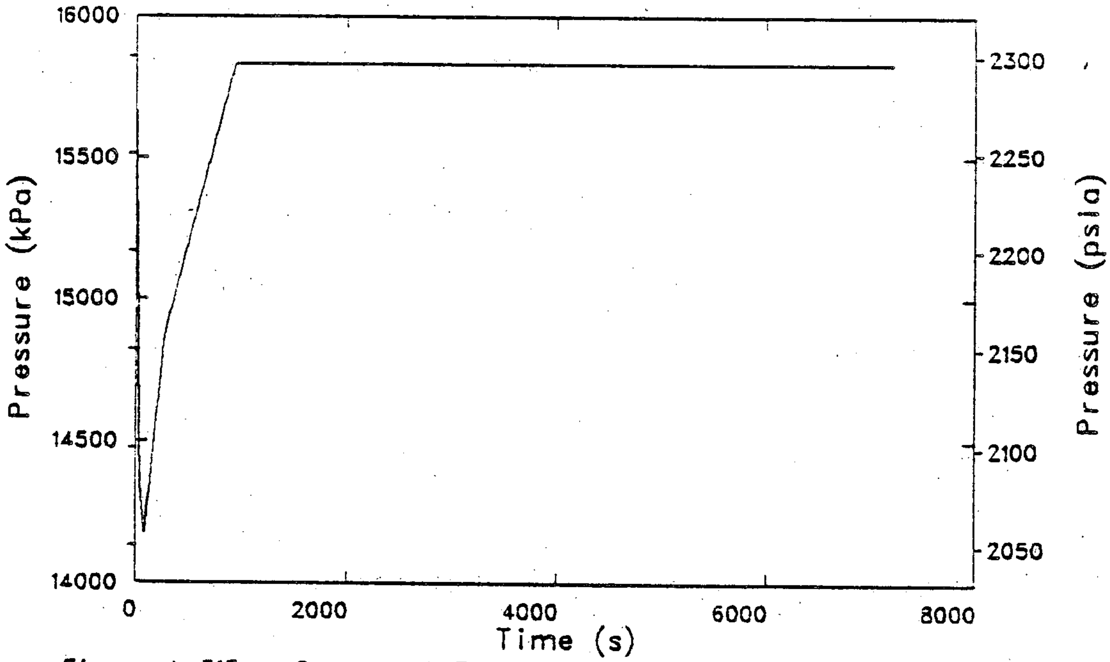

Figure A-313. Sequence 9-74 reactor vessel downcomer fluid pressure.

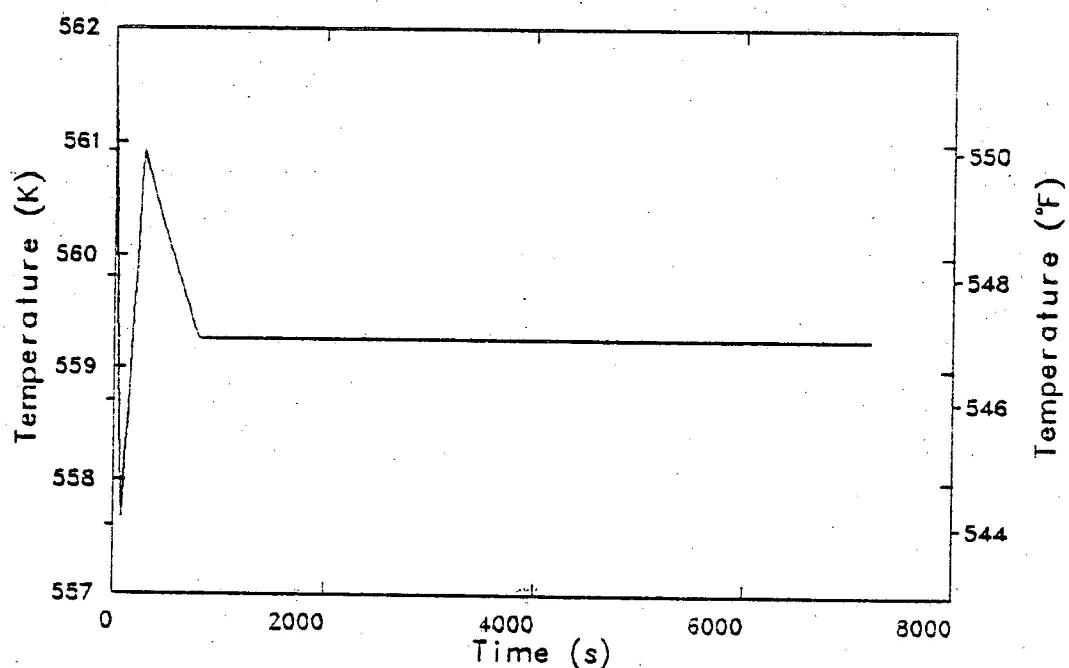

Figure A-314. Sequence 9-74 reactor vessel downcomer fluid temperature.

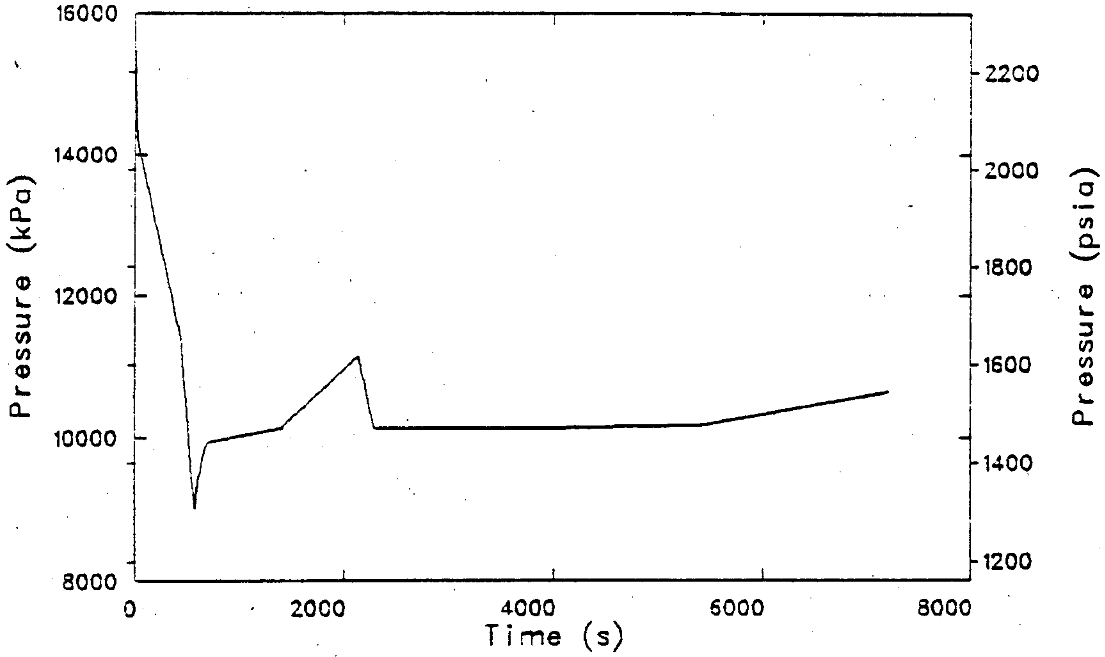

Figure A-315. Sequence 9-75 reactor vessel downcomer fluid pressure.

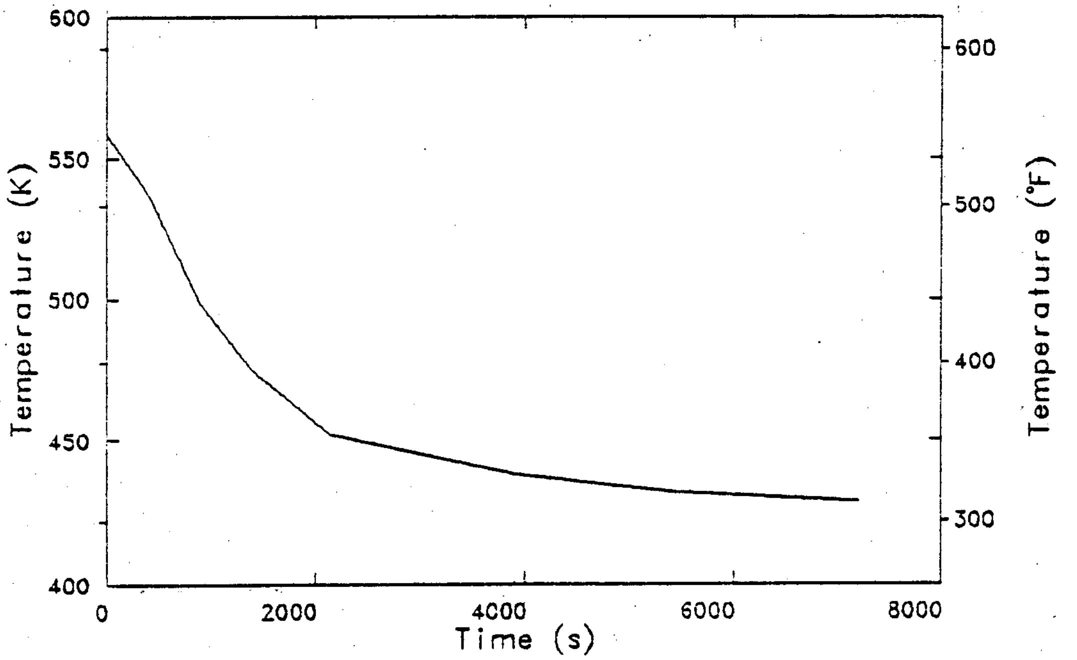

Figure A-316. Sequence 9-75 reactor vessel downcomer fluid temperature.

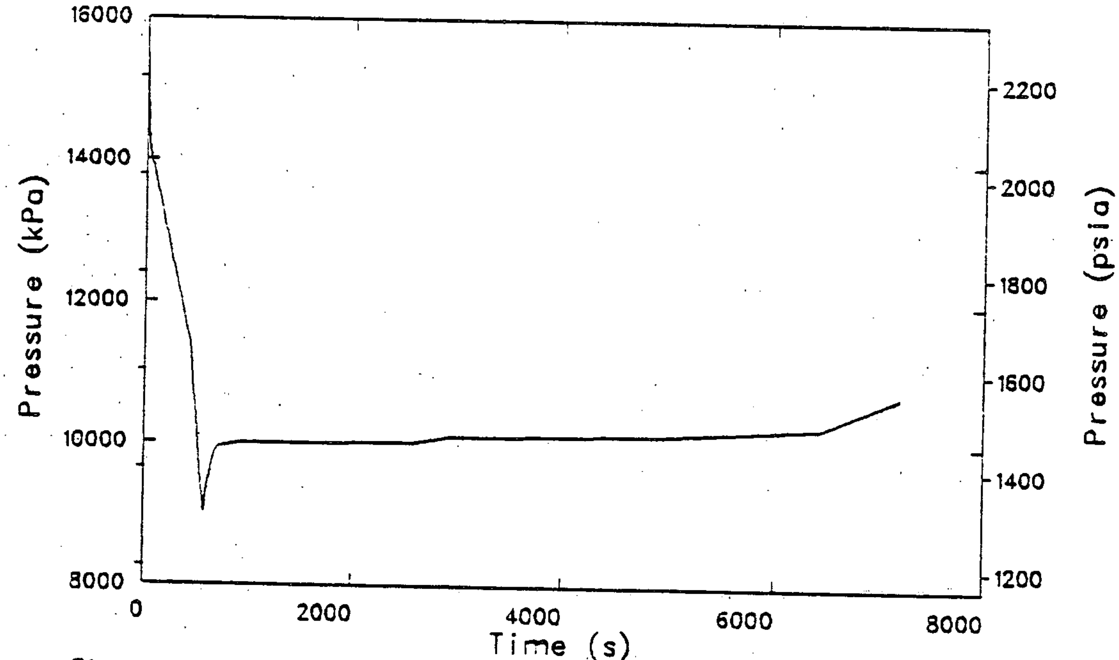

Figure A-317. Sequence 9-76 reactor vessel downcomer fluid pressure.

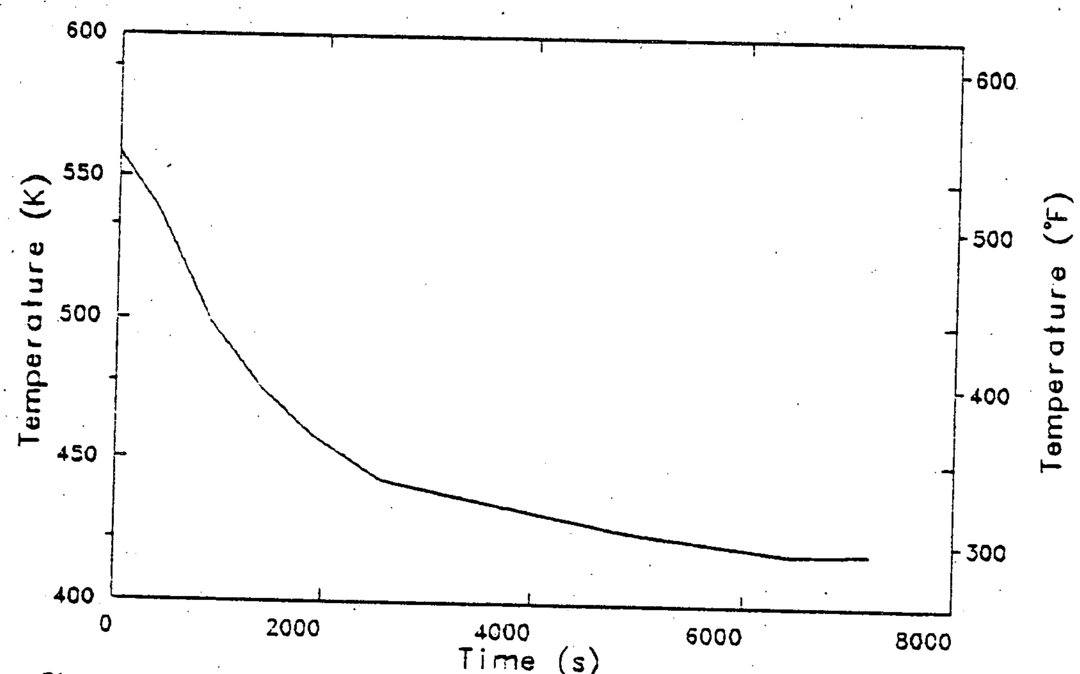

Figure A-318. Sequence 9-76 reactor vessel downcomer fluid temperature.

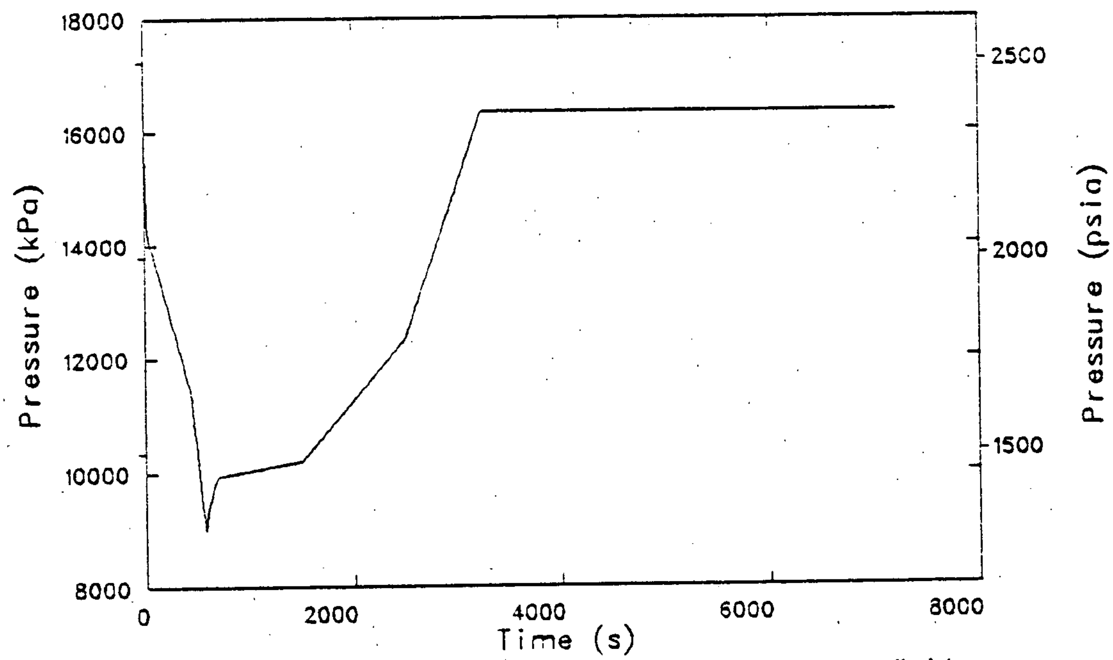

Figure A-319. Sequence 9-77 reactor vessel downcomer fluid pressure.

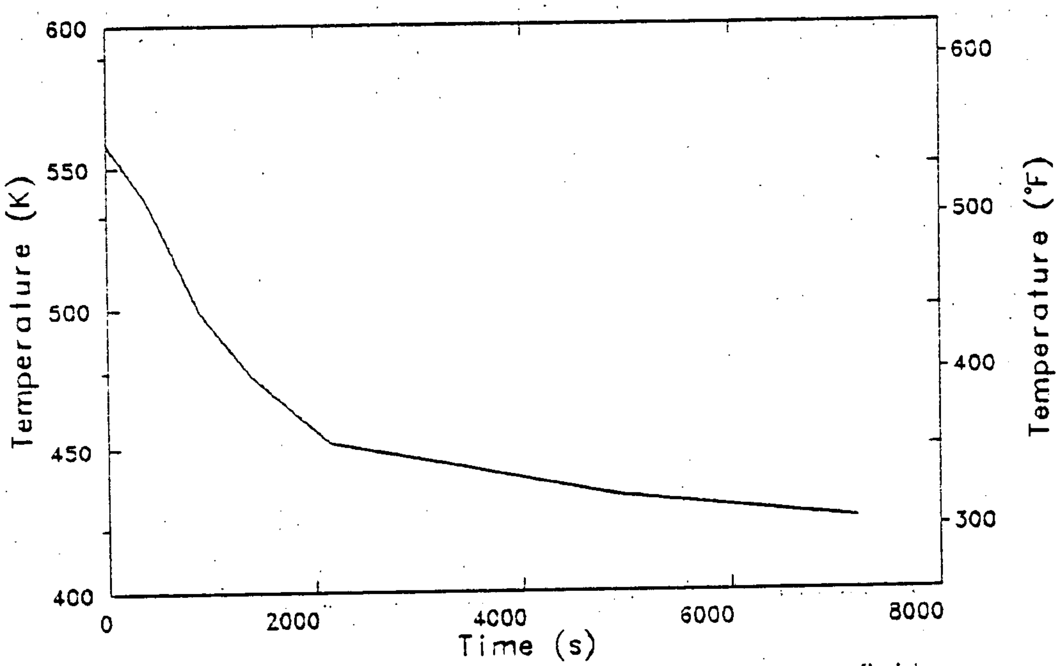

Figure A-320. Sequence 9-77 reactor vessel downcomer fluid temperature.

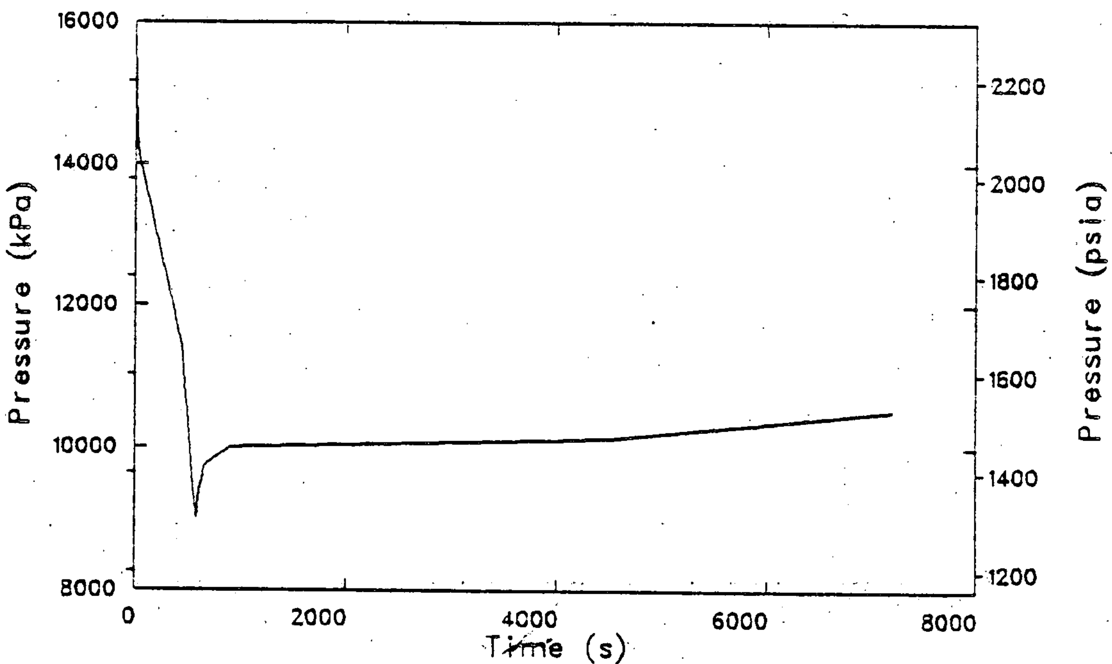

Figure A-321. Sequence 9-78 reactor vessel downcomer fluid pressure.

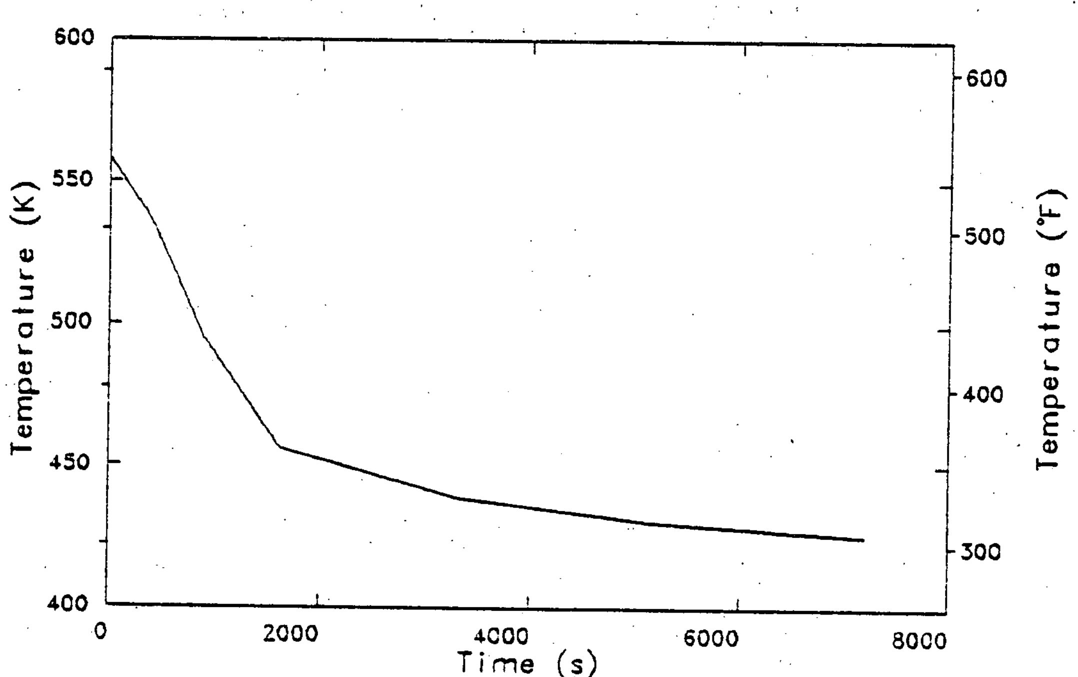

Figure A-322. Sequence 9-78 reactor vessel downcomer fluid temperature.

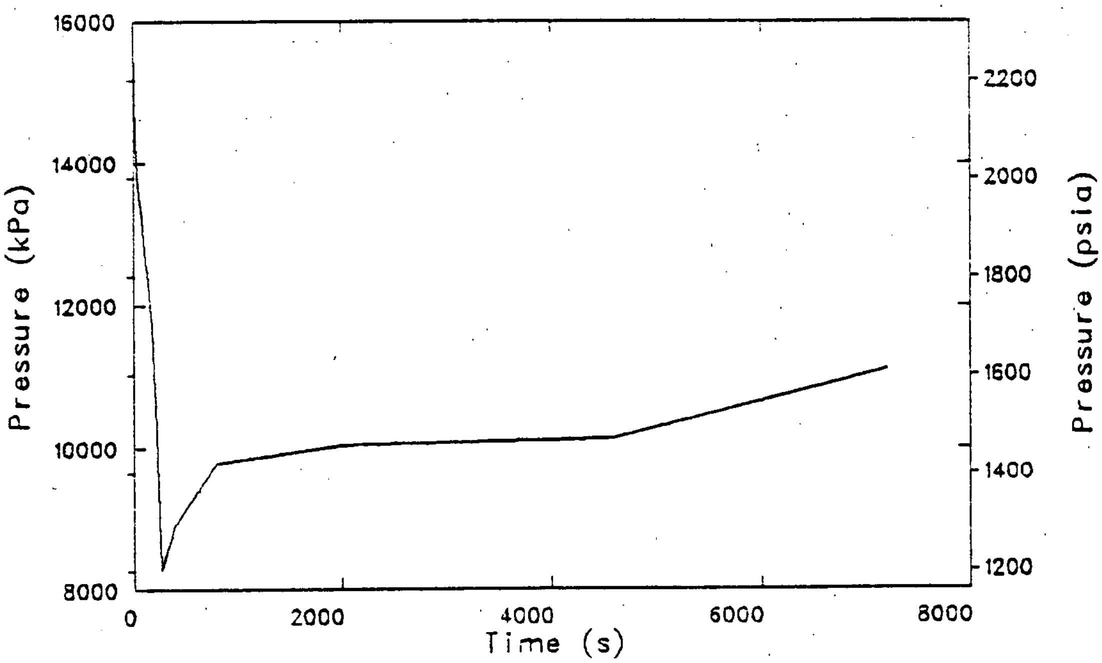

Figure A-323. Sequence 9-79 reactor vessel downcomer fluid pressure.

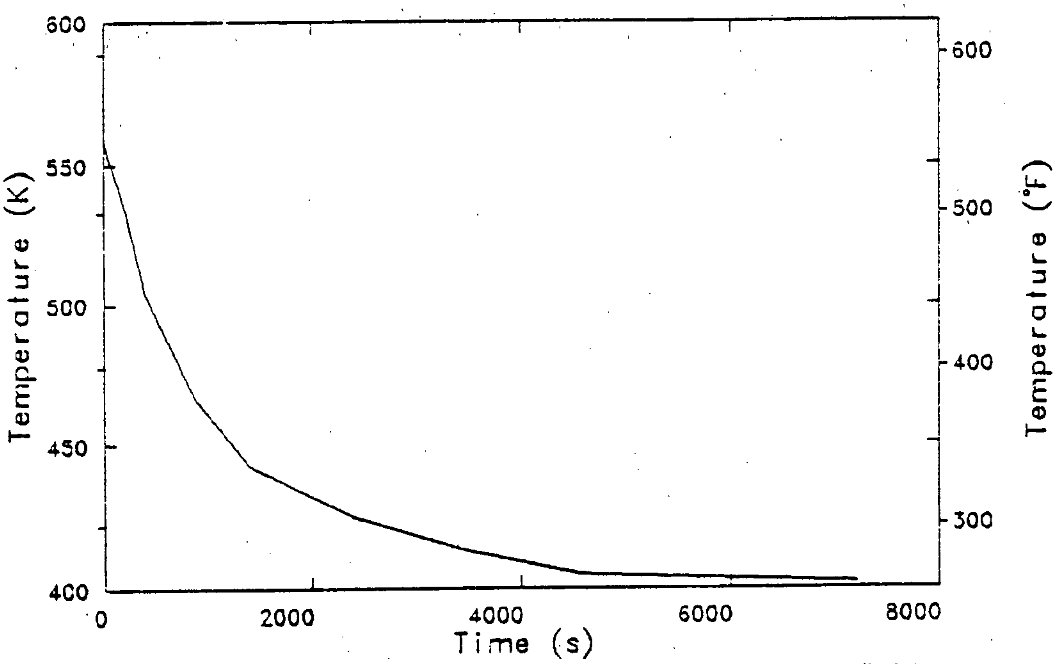

Figure A-324. Sequence 9-79 reactor vessel downcomer fluid temperature.

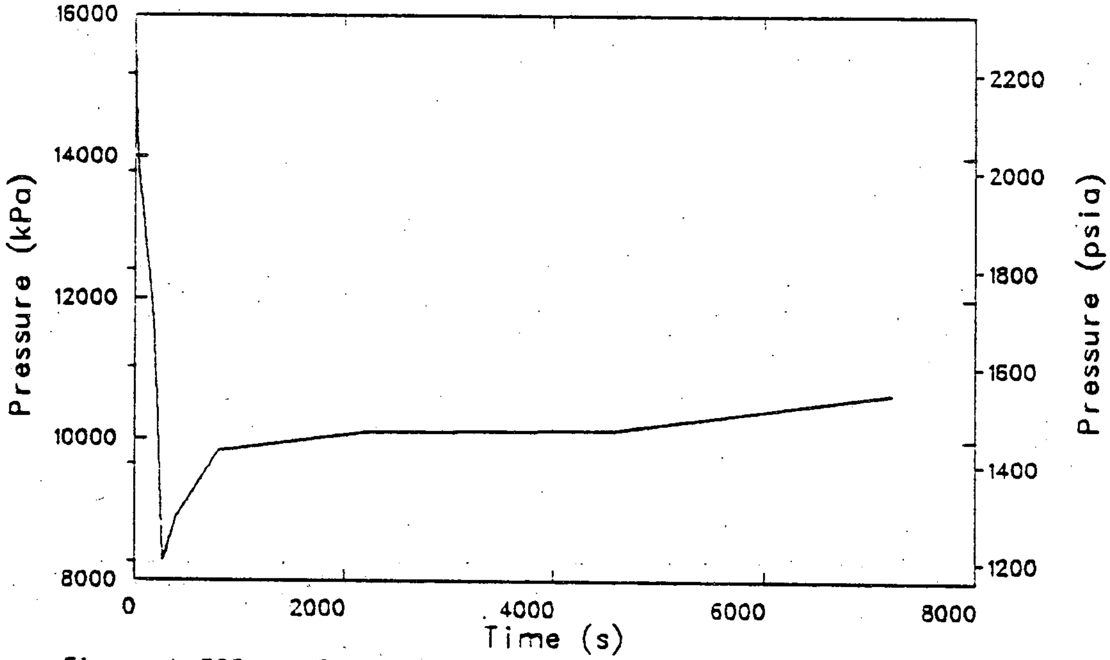

Figure A-325. Sequence 9-80 reactor vessel downcomer fluid pressure.

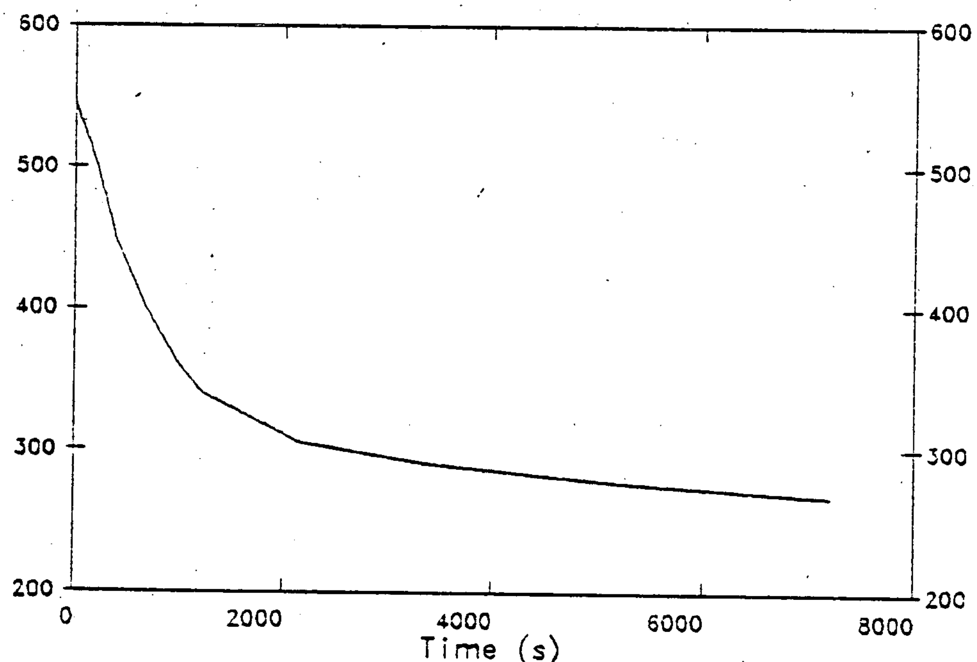

Figure A-326. Sequence 9-80 reactor vessel downcomer fluid temperature.

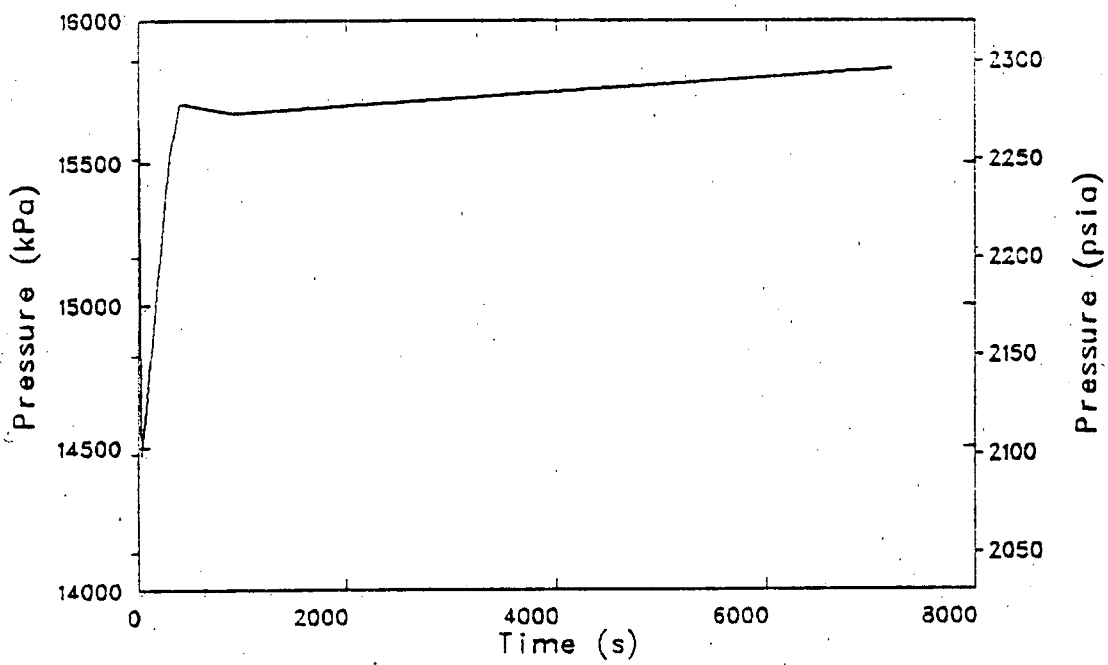

Figure A-327. Sequence 9-81 reactor vessel downcomer fluid pressure.

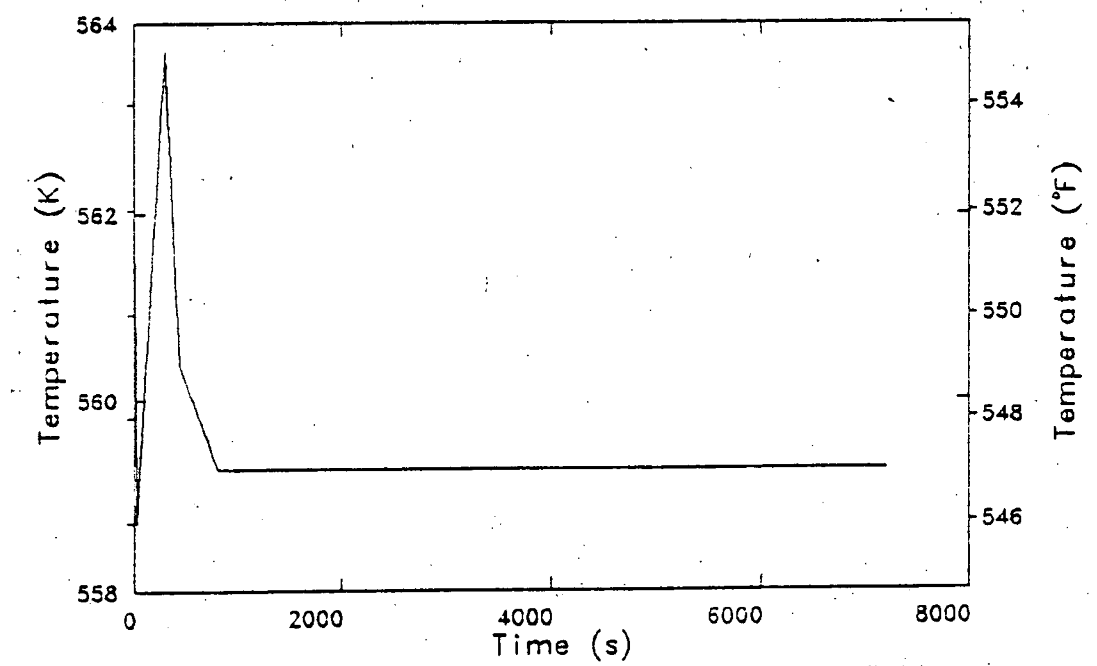

Figure A-328. Sequence 9-81 reactor vessel downcomer fluid temperature.

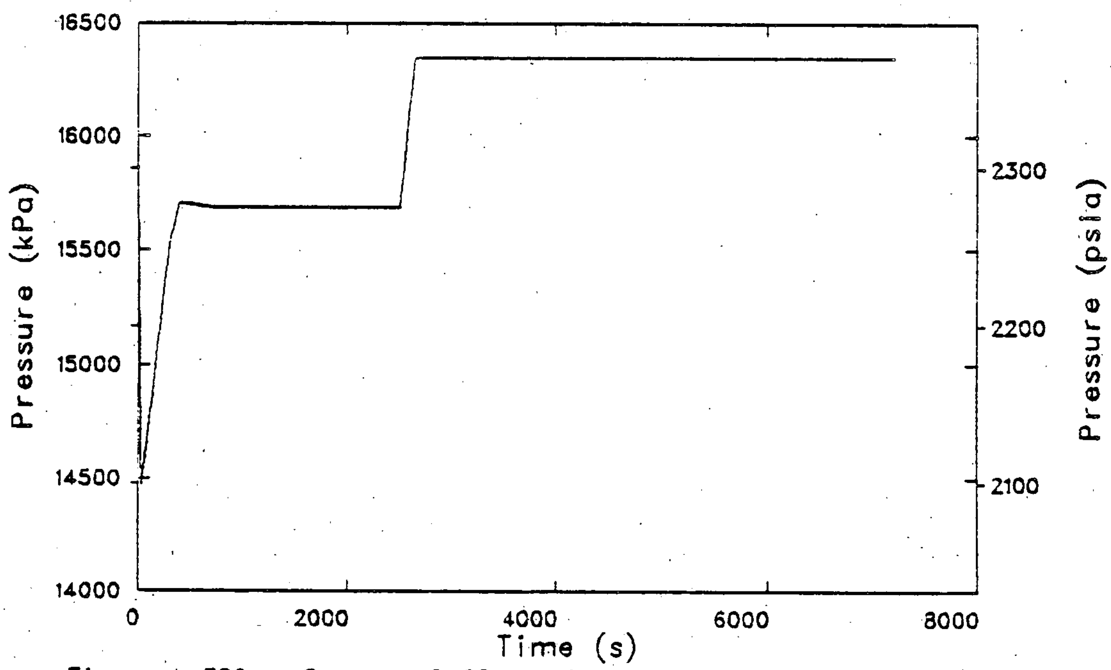

Figure A-329. Sequence 9-82 reactor vessel downcomer fluid pressure.

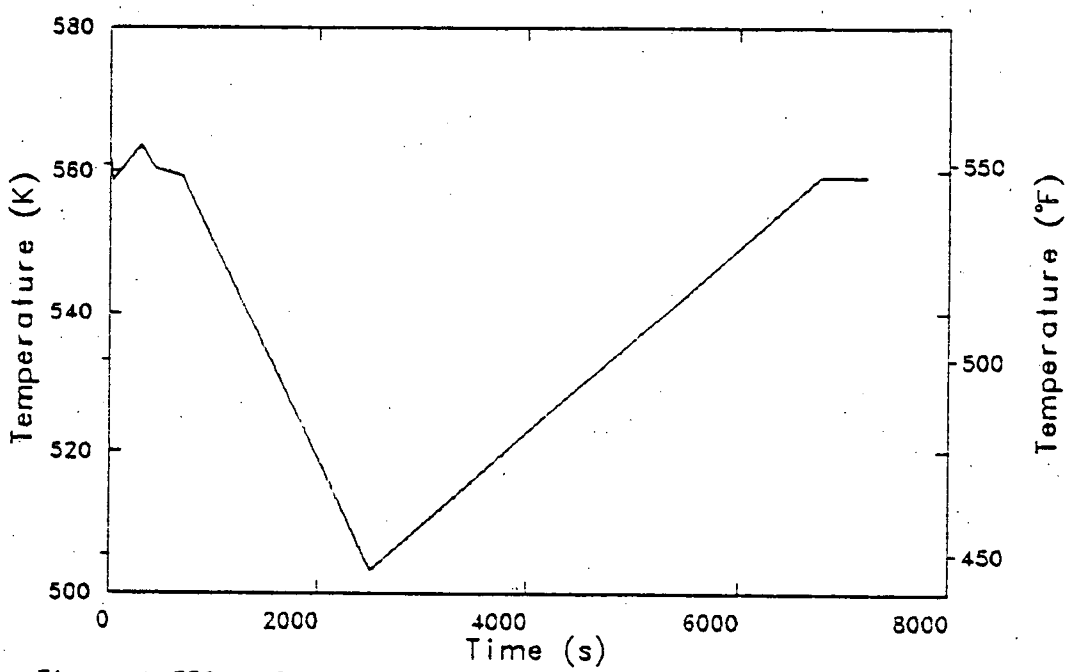

Figure A-330. Sequence 9-82 reactor vessel downcomer fluid temperature.

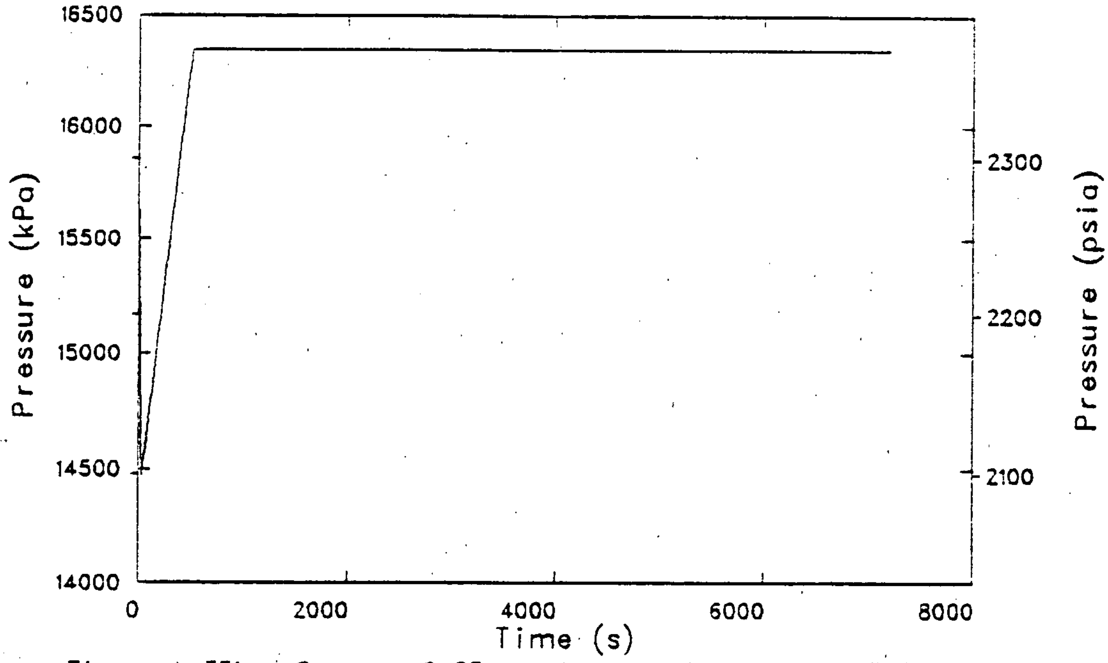

Figure A-331. Sequence 9-83 reactor vessel downcomer fluid pressure.

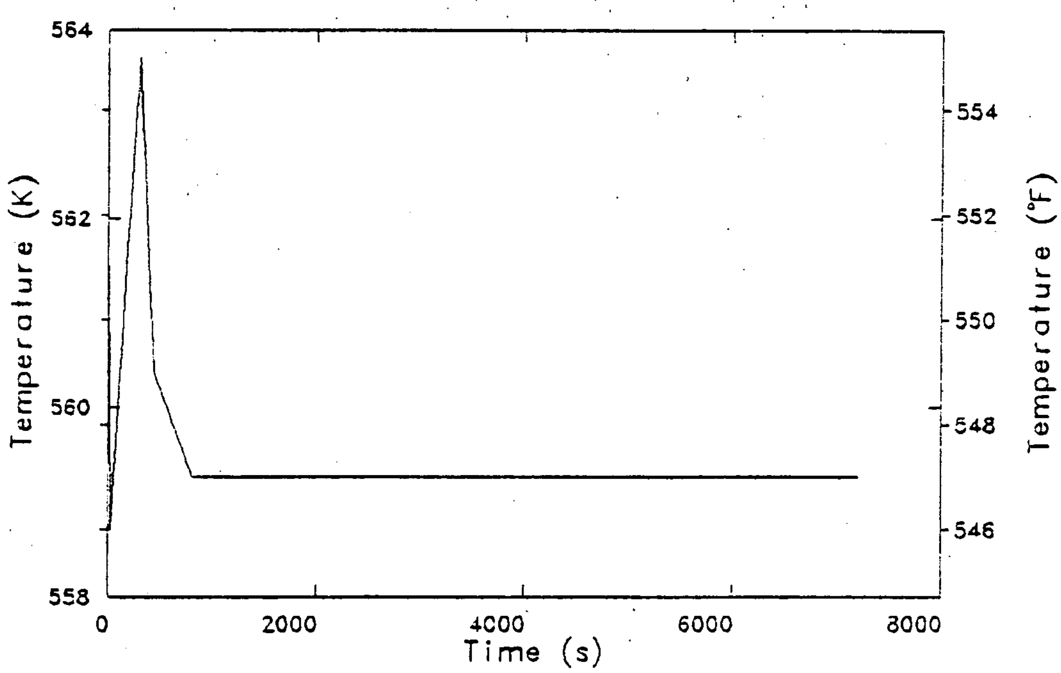

Figure A-332. Sequence 9-83 reactor vessel downcomer fluid temperature.

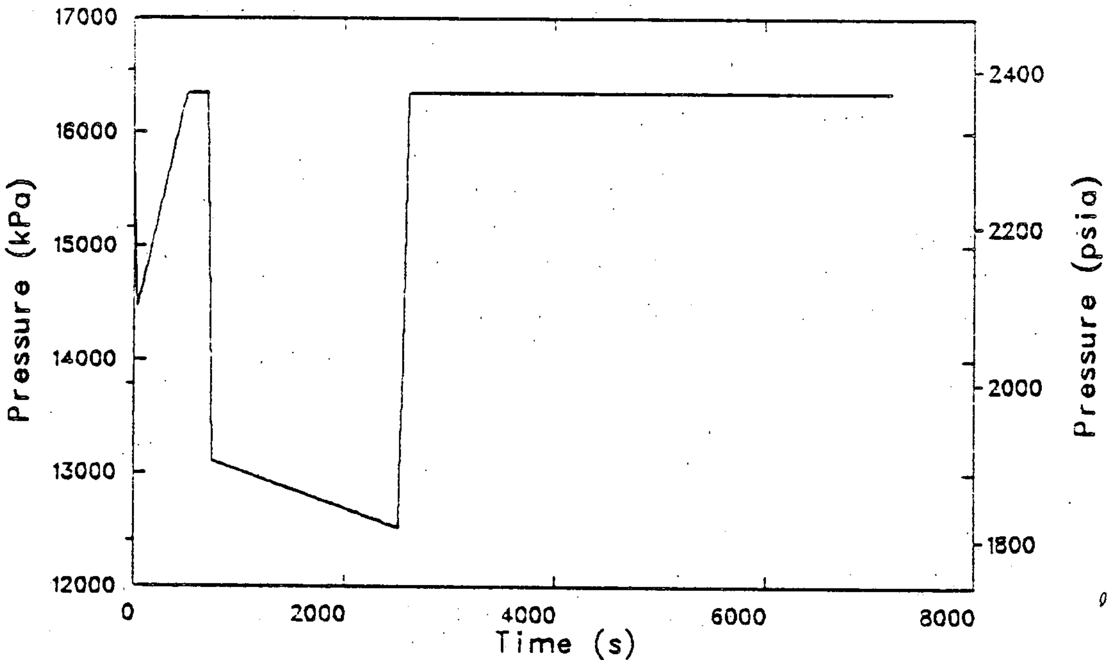

Figure A-333. Sequence 9-84 reactor vessel downcomer fluid pressure.

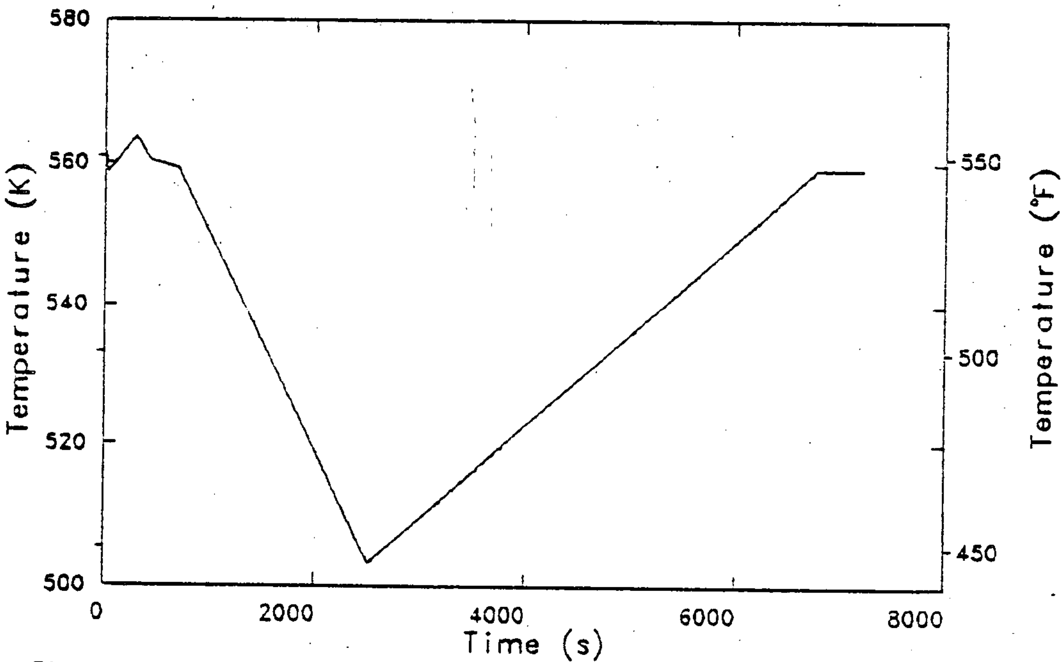

Figure A-334. Sequence 9-84 reactor vessel downcomer fluid temperature.

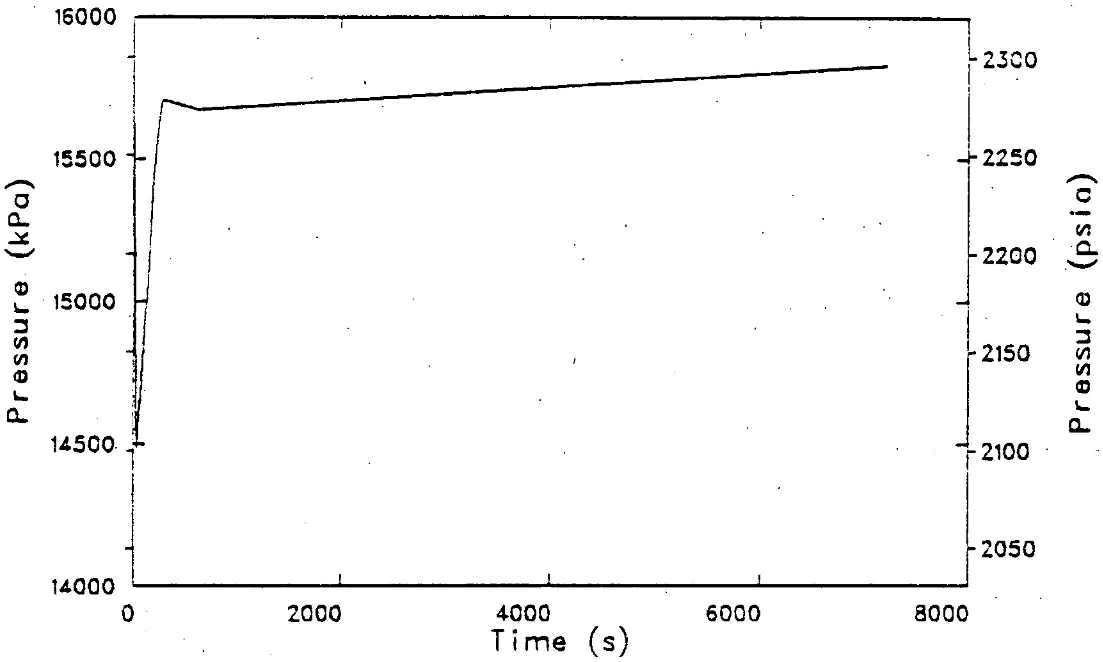

Figure A-335. Sequence 9-85 reactor vessel downcomer fluid pressure.

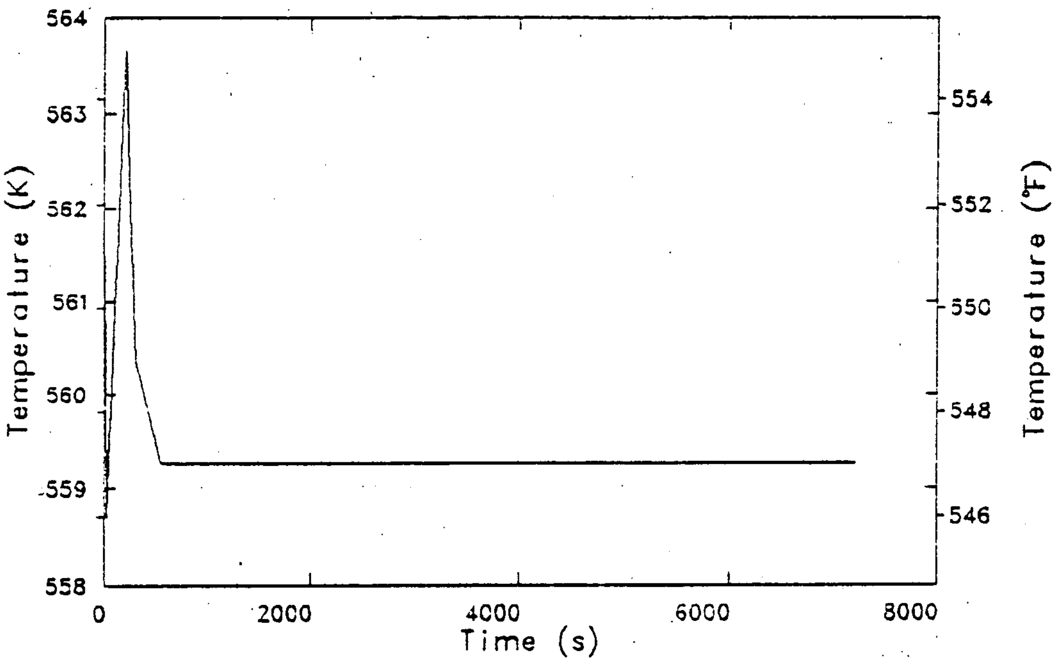

Figure A-336. Sequence 9-85 reactor vessel downcomer fluid temperature.

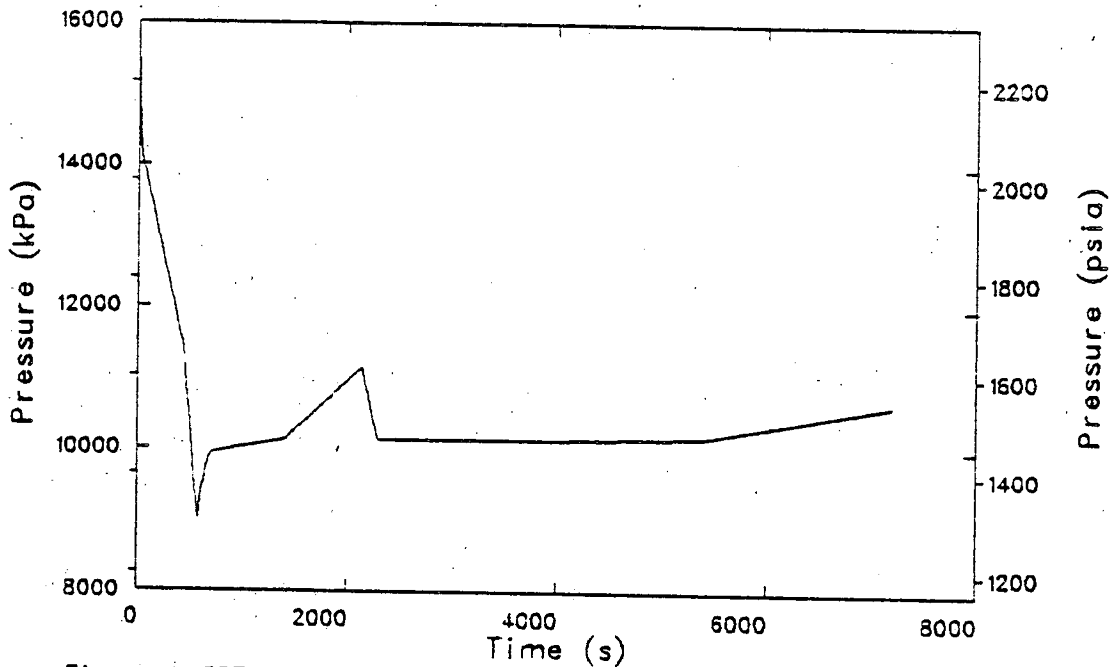

Figure A-337. Sequence 9-86 reactor vessel downcomer fluid pressure.

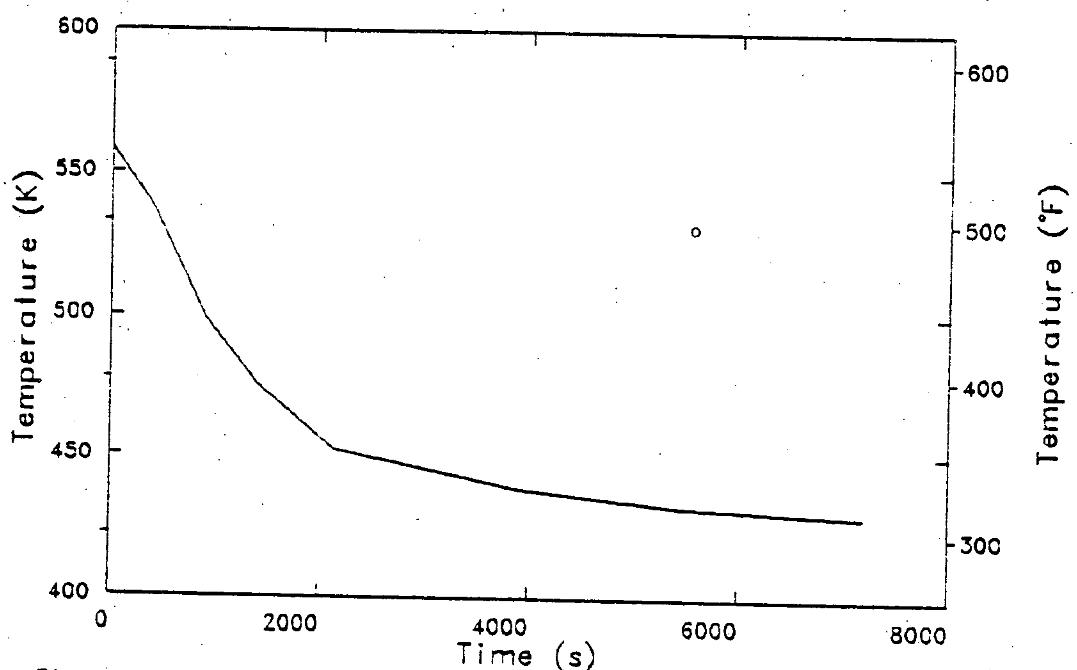

Figure A-338. Sequence 9-86 reactor vessel downcomer fluid temperature.

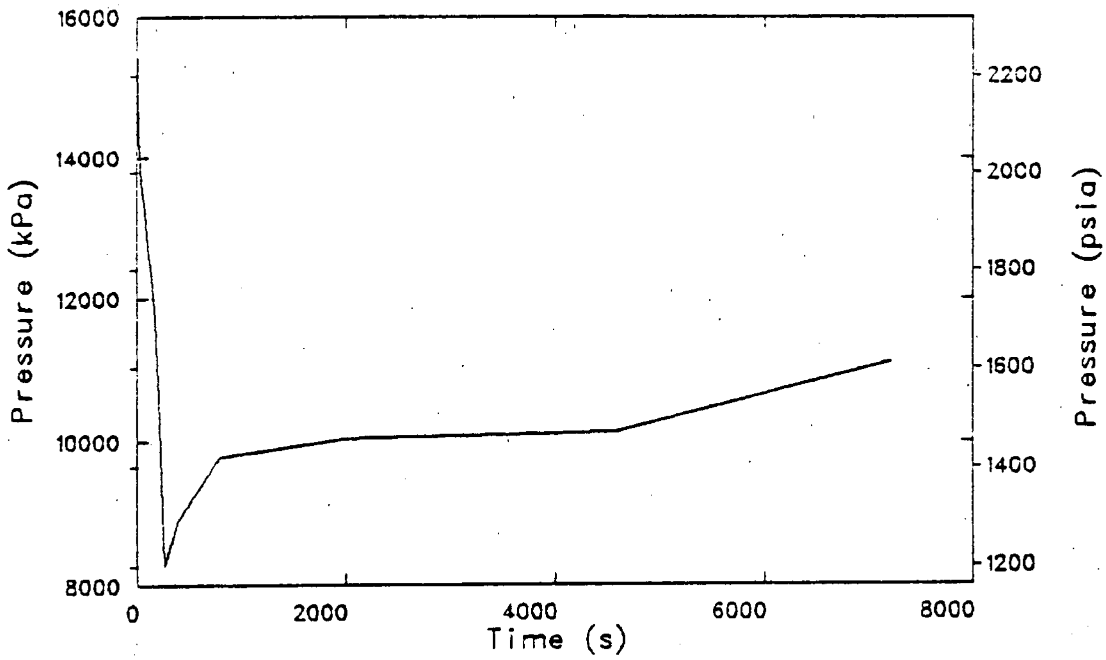

Figure A-339. Sequence 9-87 reactor vessel downcomer fluid pressure.

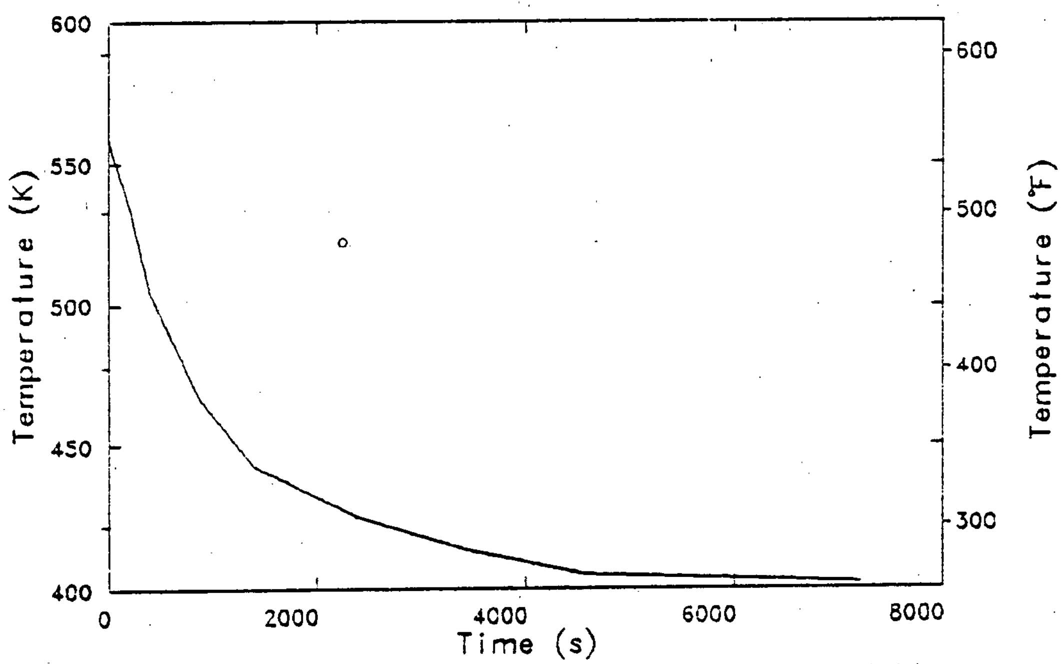

Figure A-340. Sequence 9-87 reactor vessel downcomer fluid temperature.

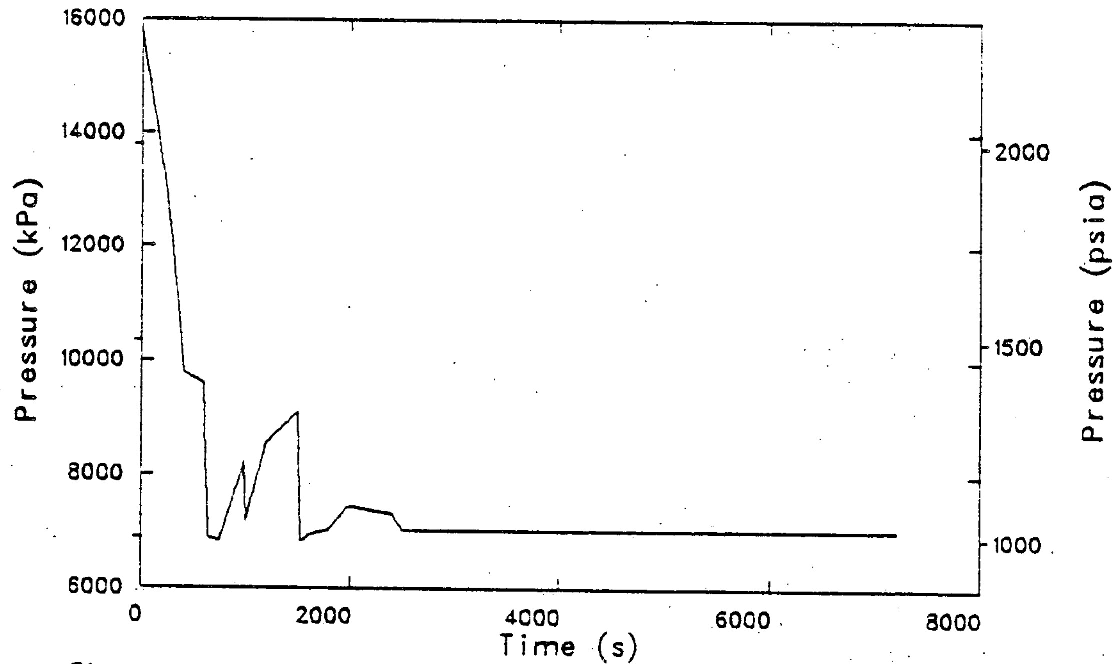

Figure A-341. Sequence 10-1 reactor vessel downcomer fluid pressure.

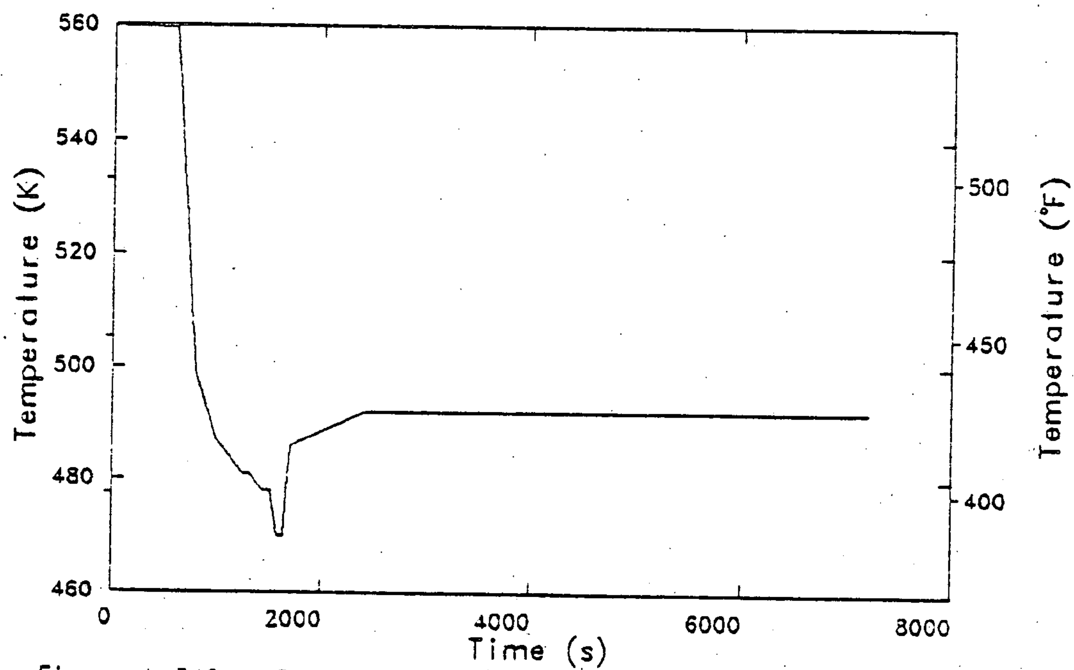

Figure A-342. Sequence 10-1 reactor vessel downcomer fluid temperature.

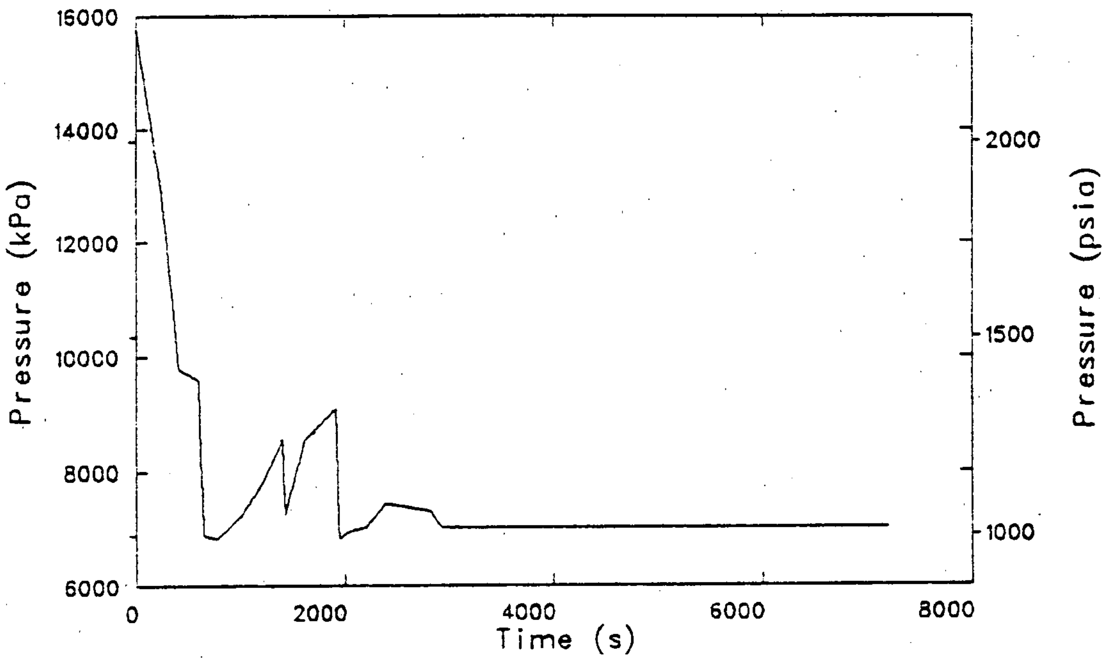

Figure A-343. Sequence 10-2 reactor vessel downcomer fluid pressure.

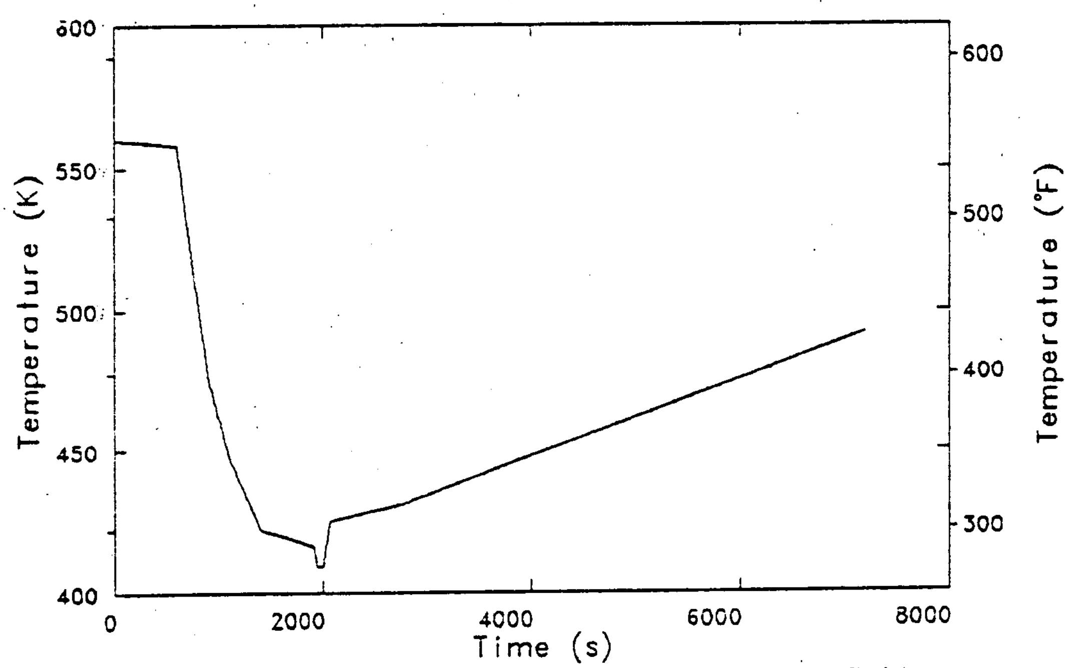

Figure A-344. Sequence 10-2 reactor vessel downcomer fluid temperature.

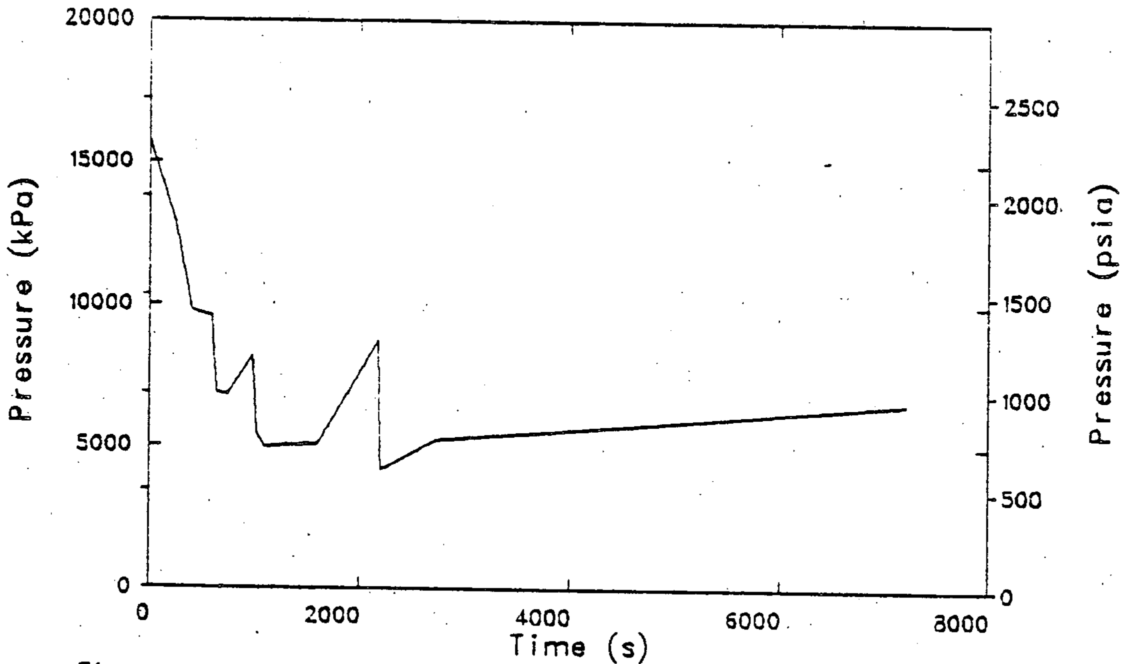

Figure A-345. Sequence 10-3 reactor vessel downcomer fluid pressure.

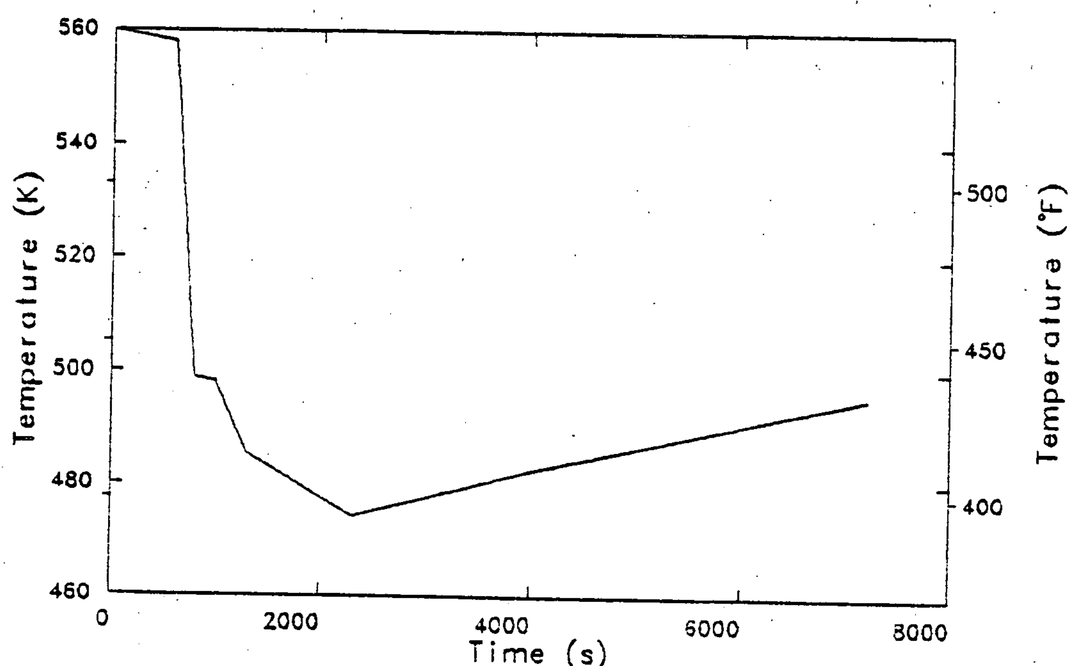

Figure A-346. Sequence 10-3 reactor vessel downcomer fluid temperature.

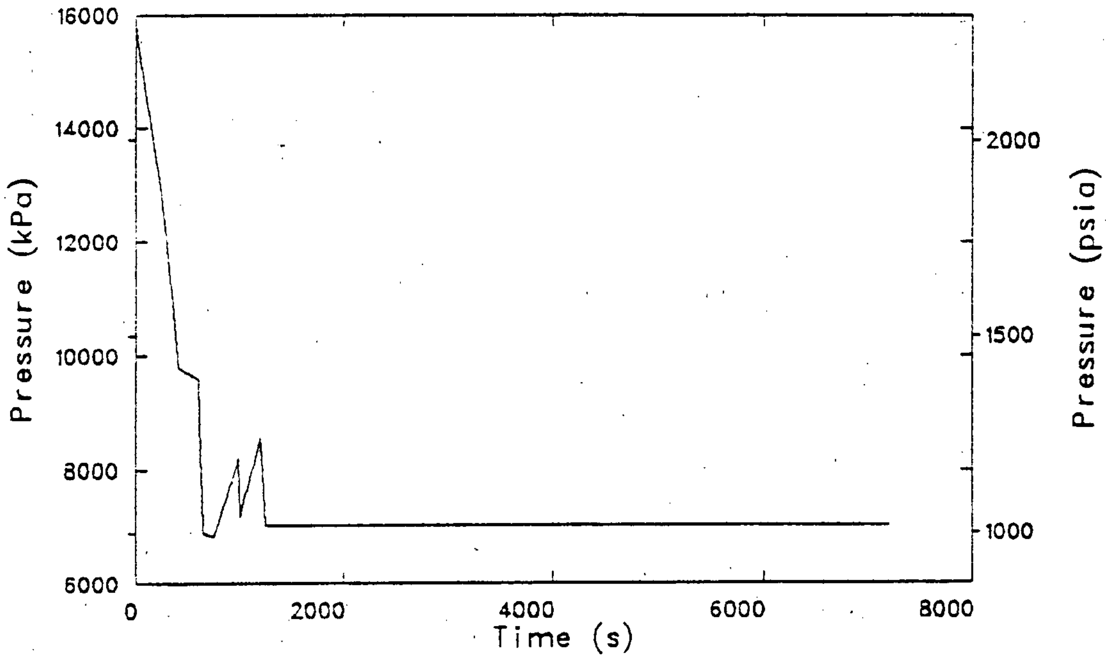

Figure A-347. Sequence 10-4 reactor vessel downcomer fluid pressure.

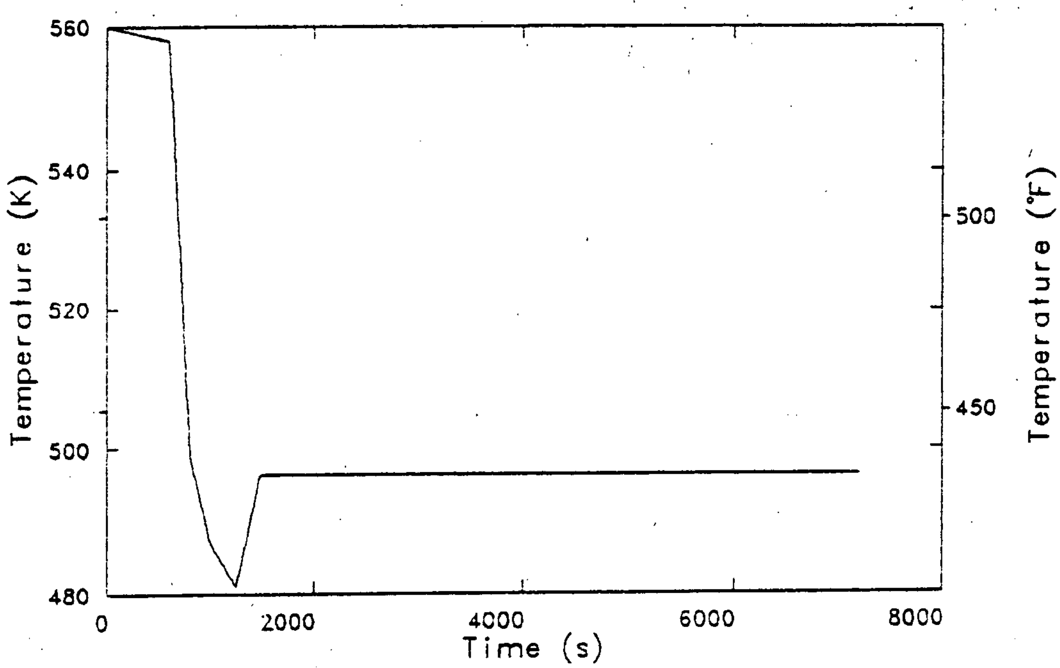

Figure A-348. Sequence 10-4 reactor vessel downcomer fluid temperature.

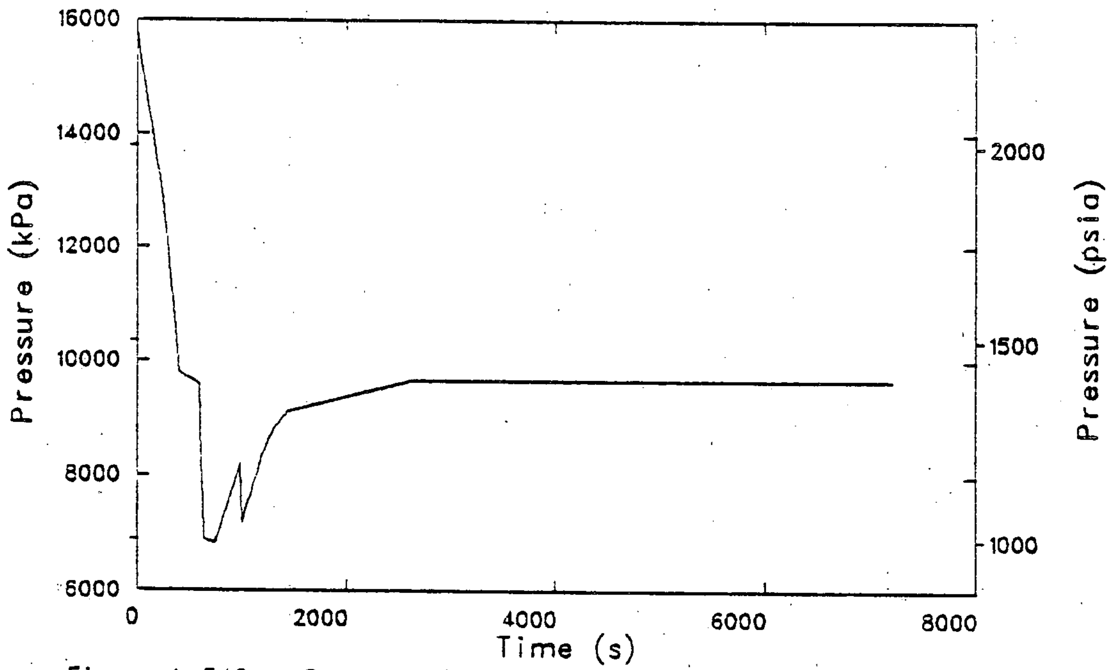

Figure A-349. Sequence 10-5 reactor vessel downcomer fluid pressure.

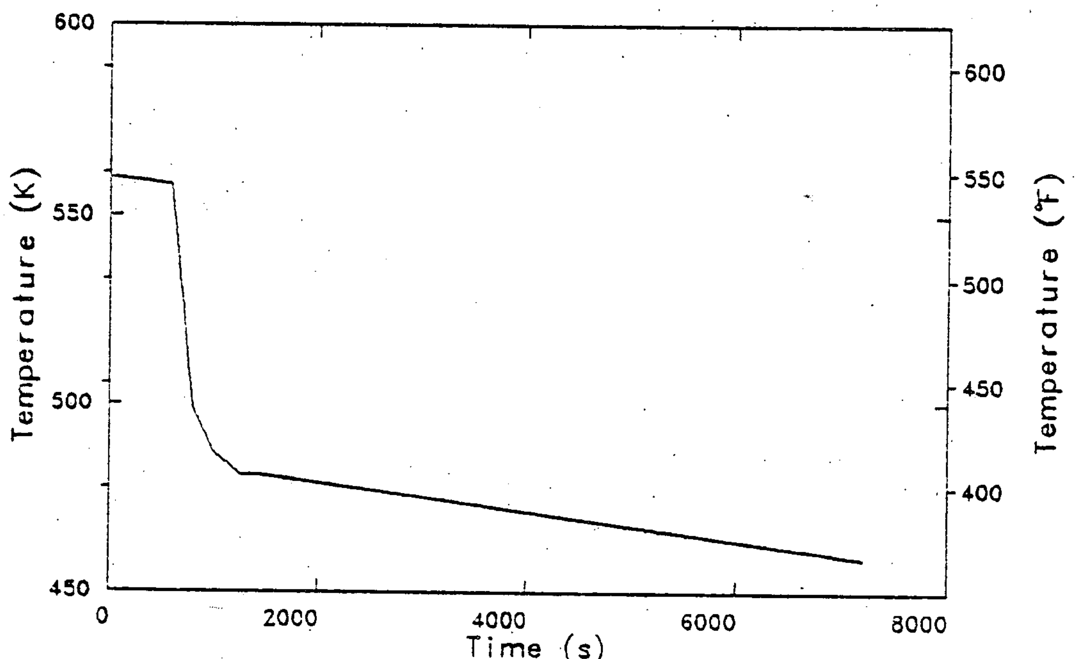

Figure A-350. Sequence 10-5 reactor vessel downcomer fluid temperature.

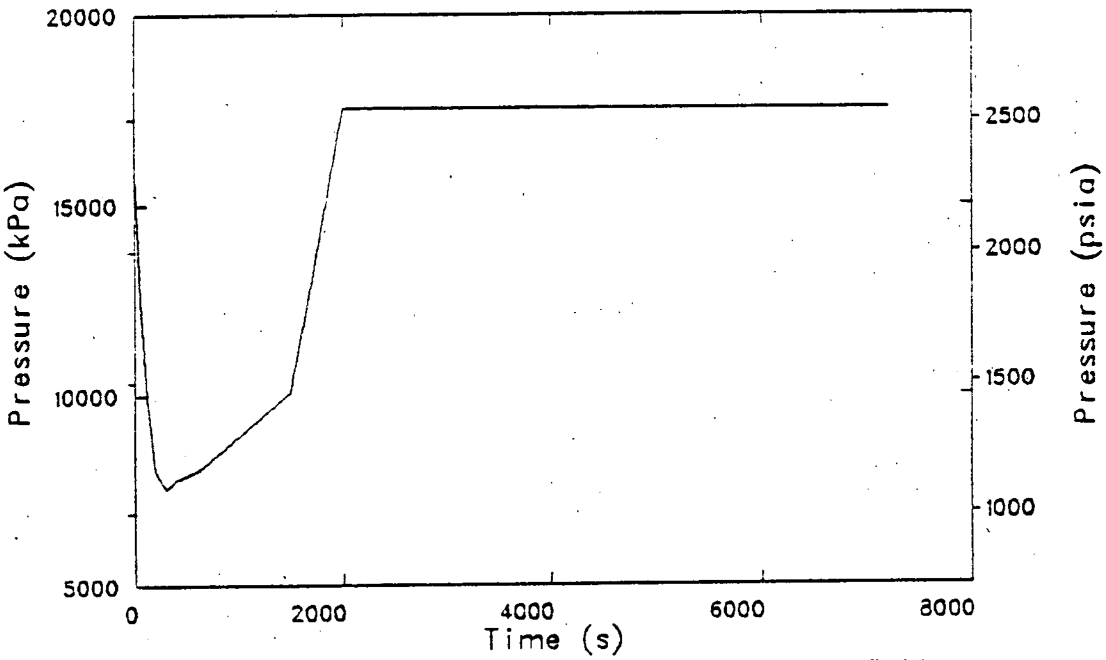

Figure A-351. Sequence 11-1 reactor vessel downcomer fluid pressure.

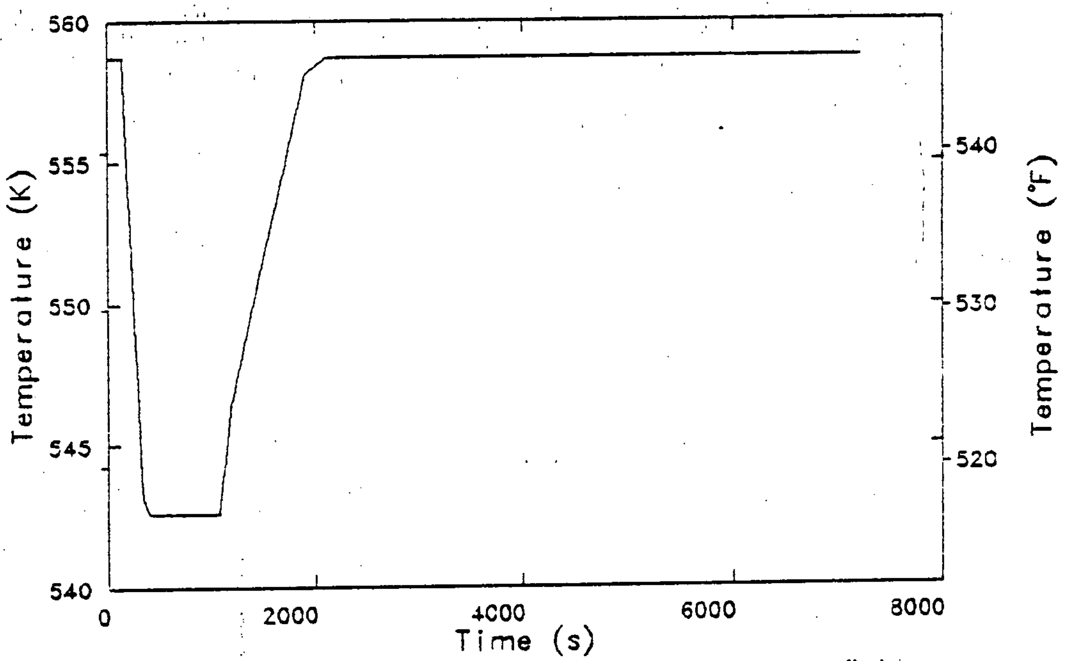

Figure A-352. Sequence 11-1 reactor vessel downcomer fluid temperature.

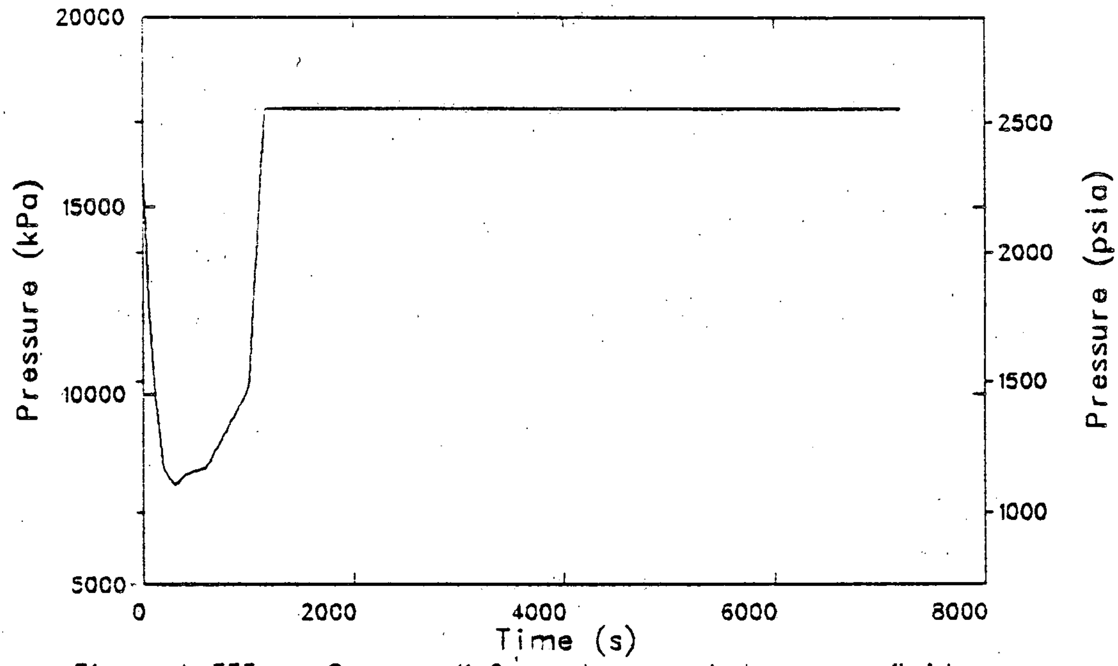

Figure A-353. Sequence 11-2 reactor vessel downcomer fluid pressure.

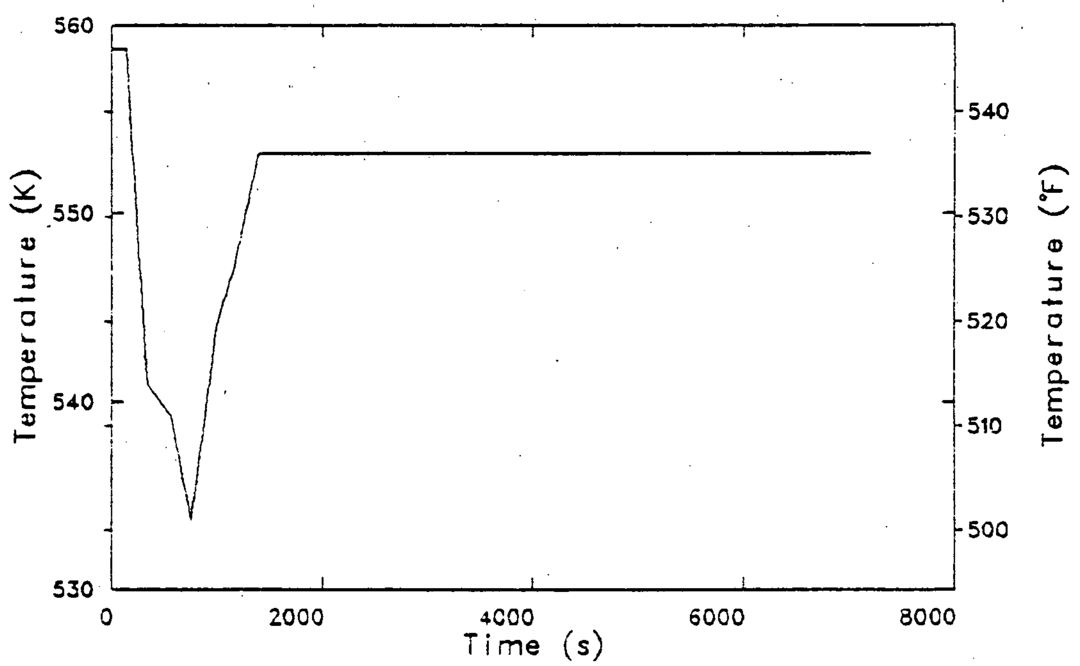

Figure A-354. Sequence 11-2 reactor vessel downcomer fluid temperature.

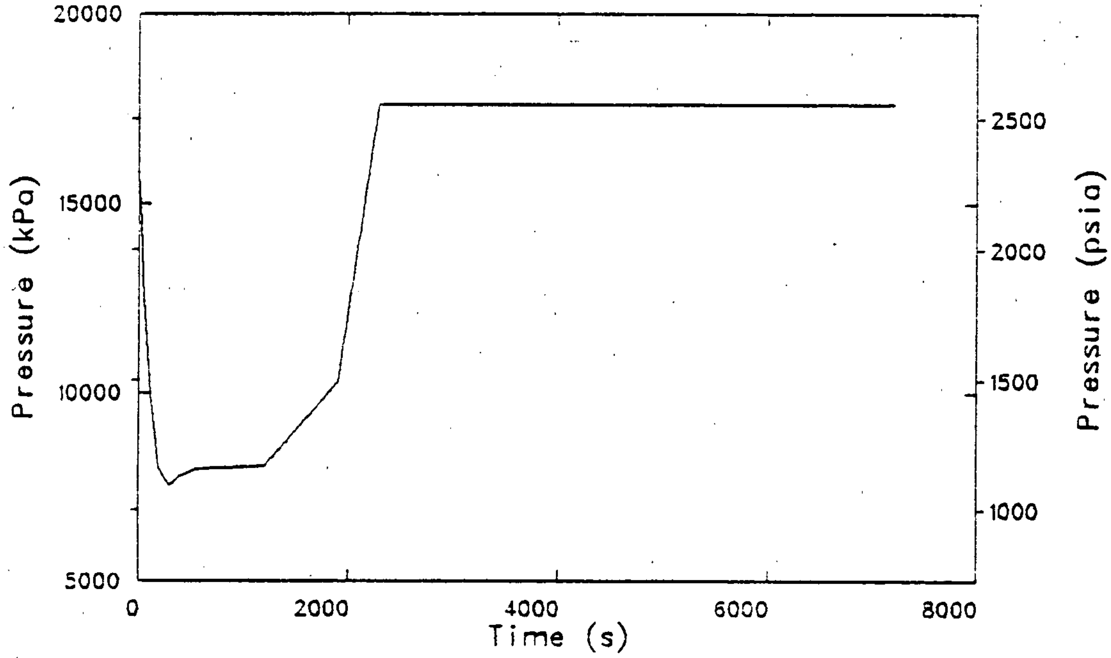

Figure A-355. Sequence 11-3 reactor vessel downcomer fluid pressure.

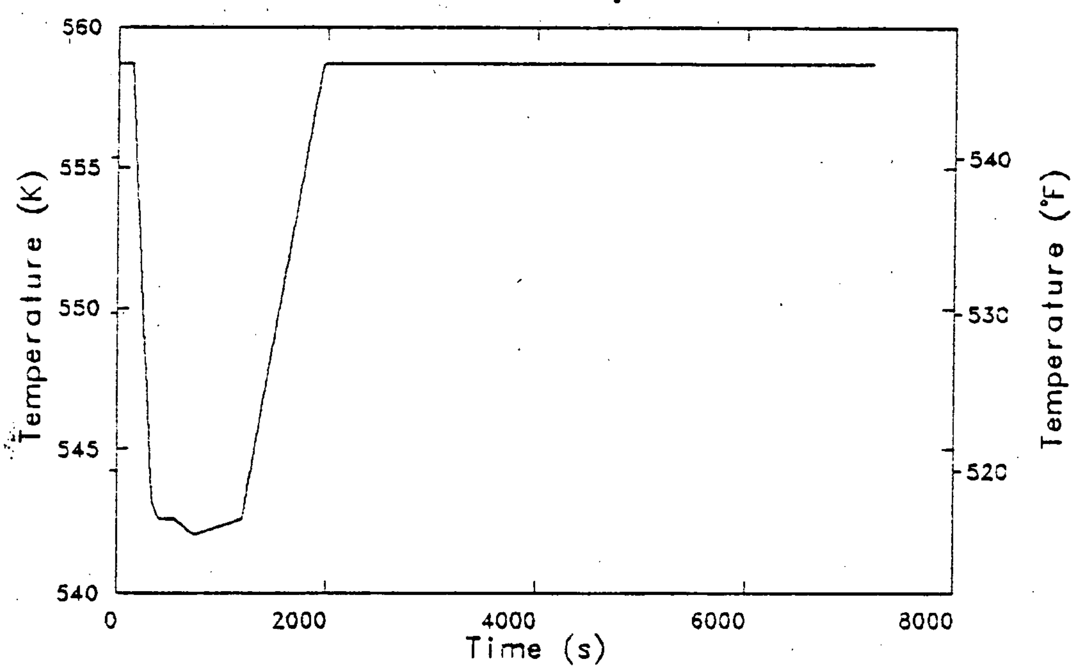

Figure A-356. Sequence 11-3 reactor vessel downcomer fluid temperature.

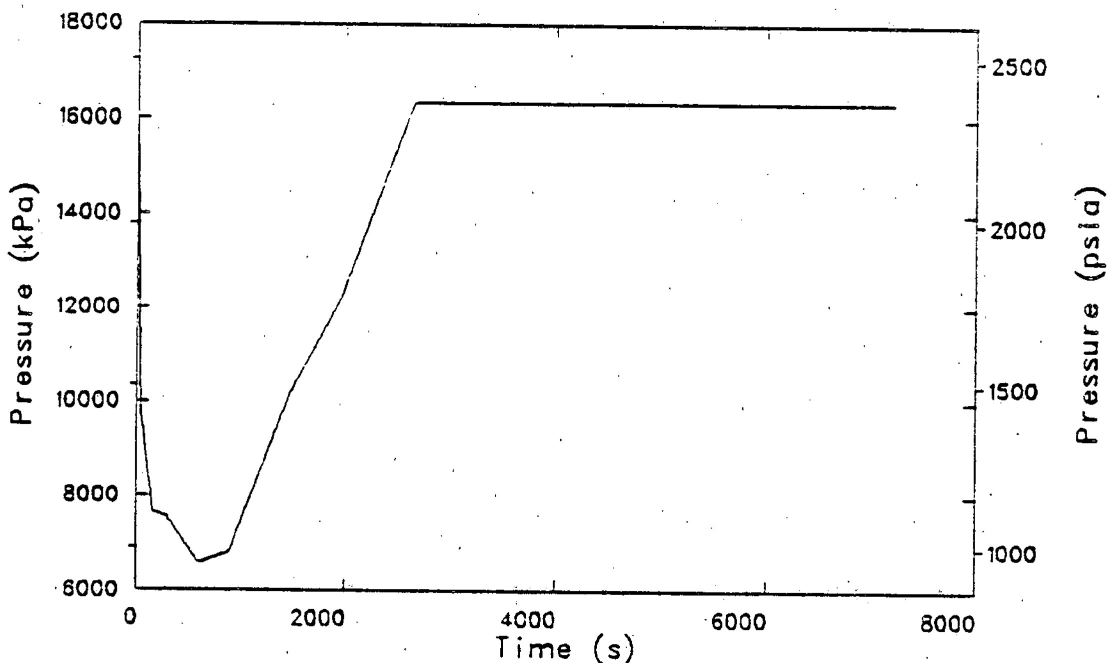

Figure A-359. Sequence 12-1 reactor vessel downcomer fluid pressure.

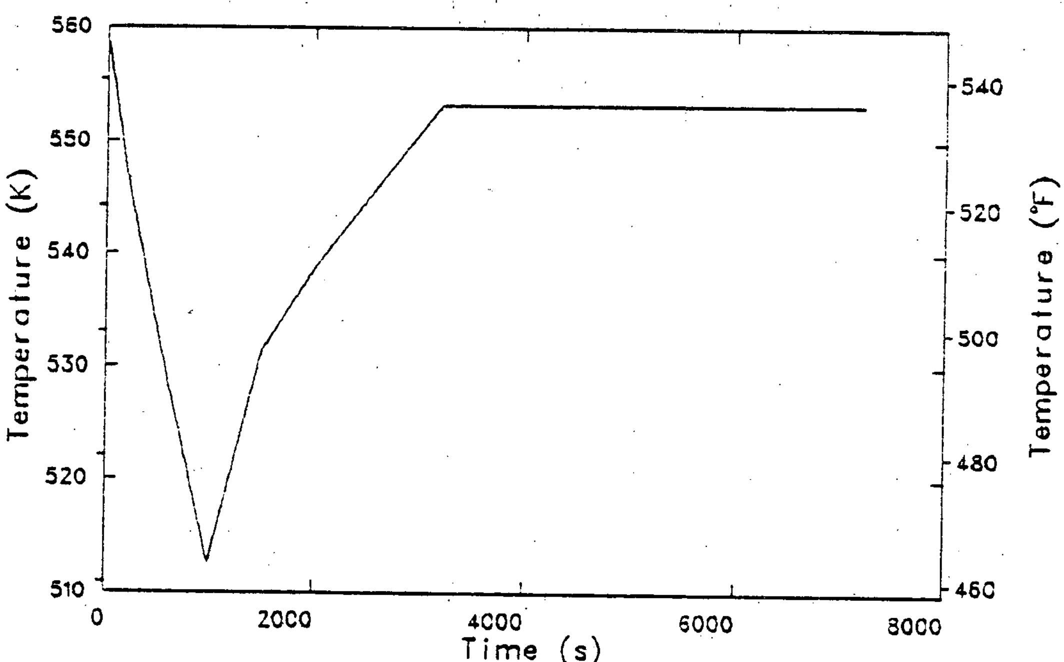

Figure A-360. Sequence 12-1 reactor vessel downcomer fluid temperature.

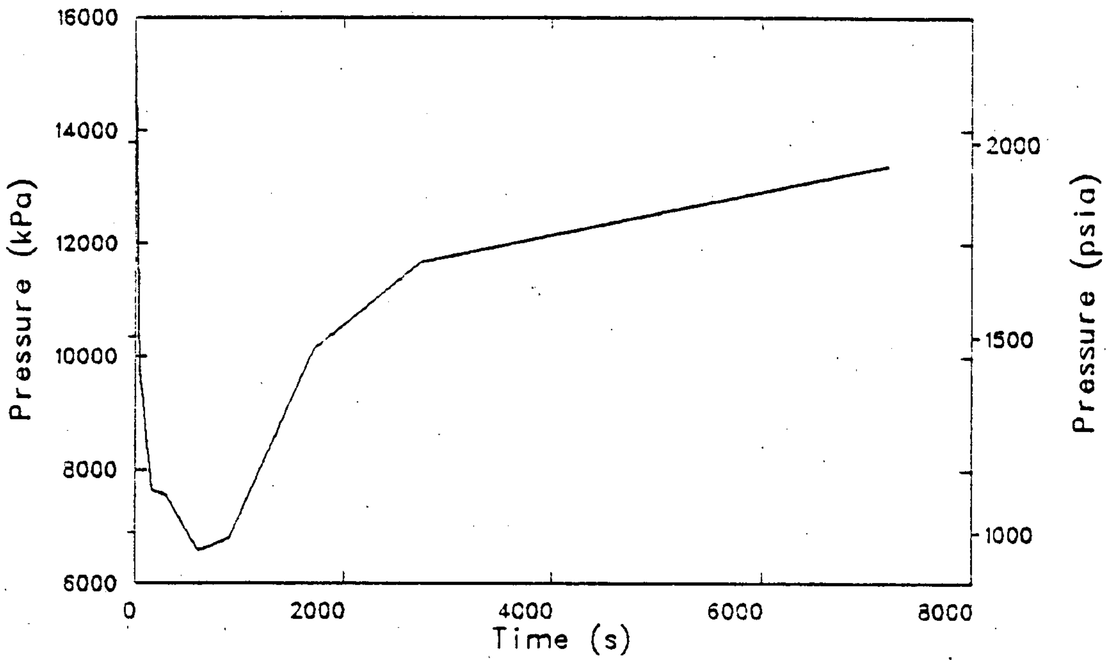

Figure A-361. Sequence 12-2 reactor vessel downcomer fluid pressure.

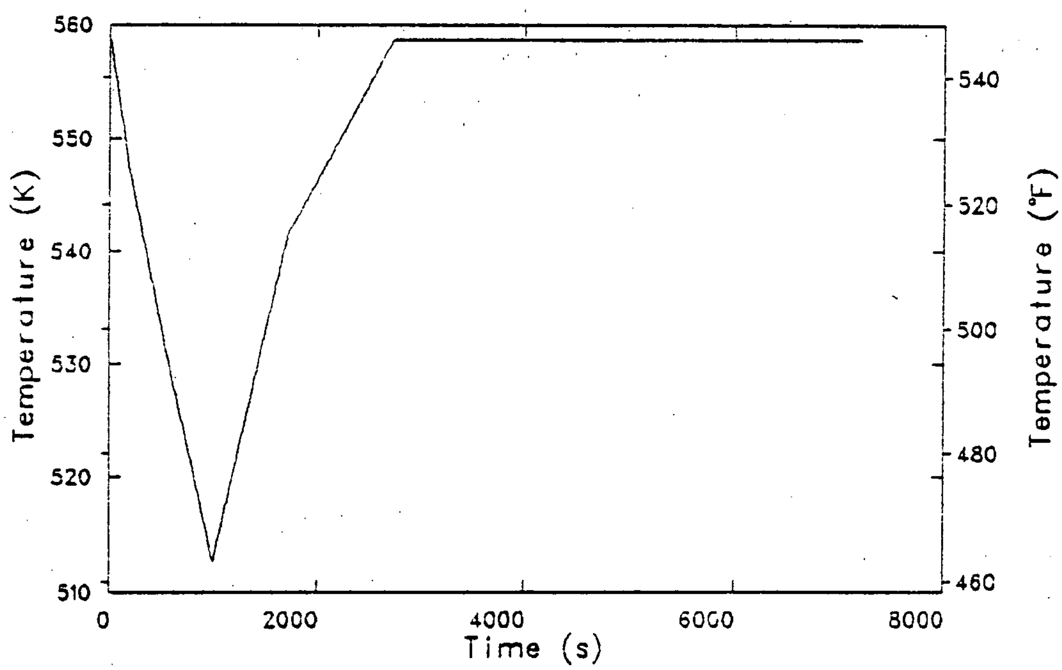

Figure A-362. Sequence 12-2 reactor vessel downcomer fluid temperature.

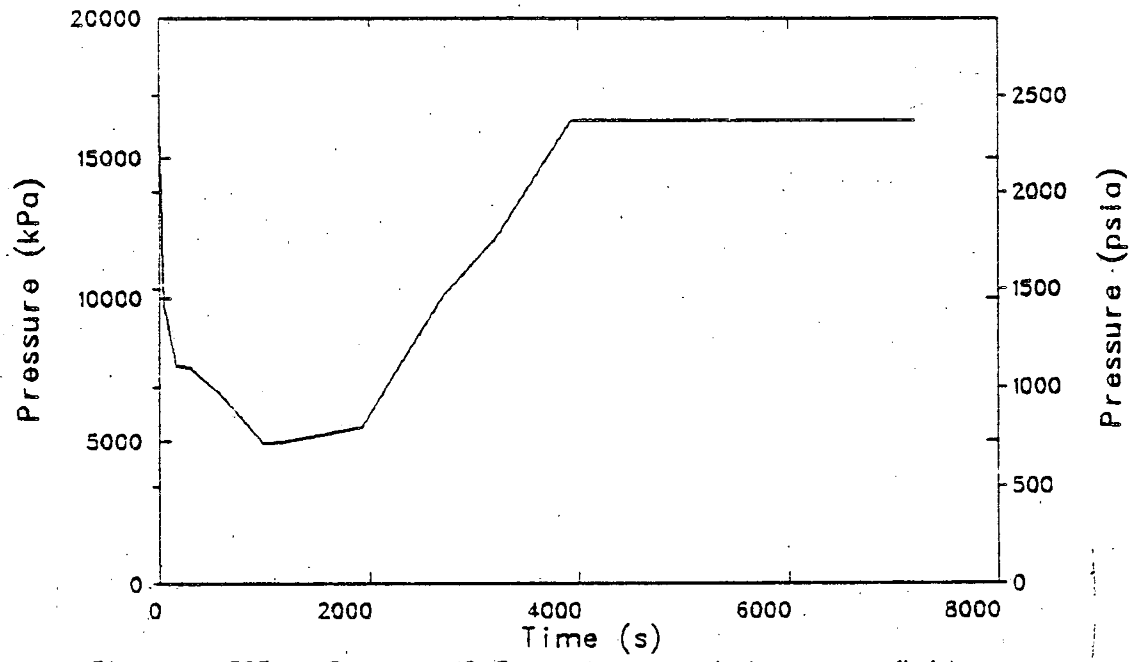

Figure A-363. Sequence 12-3 reactor vessel downcomer fluid pressure.

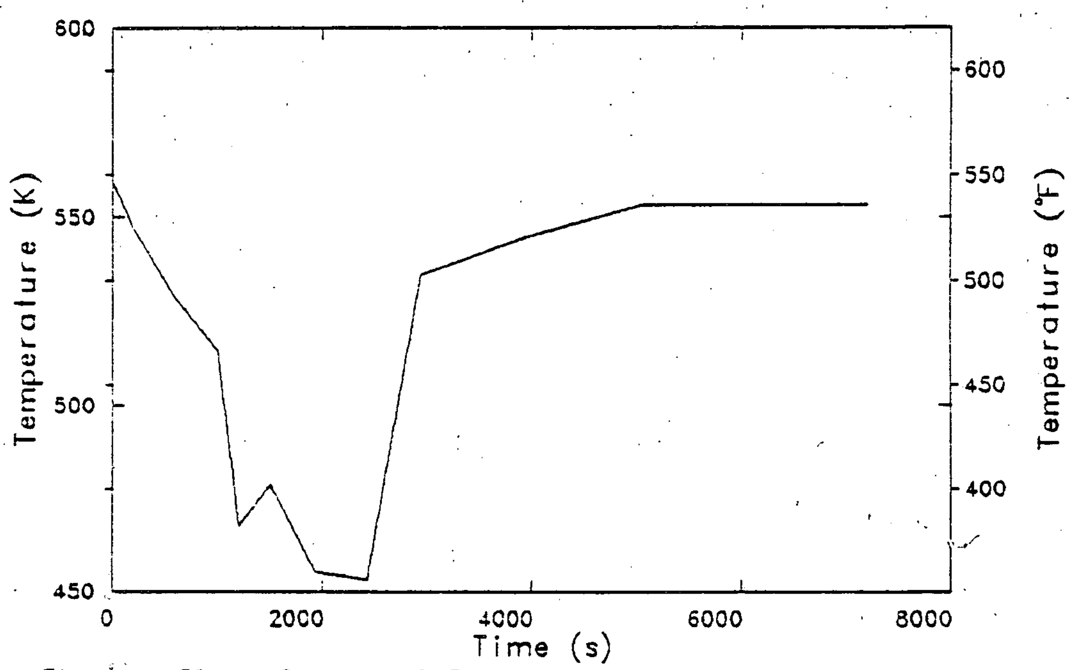

Figure A-364. Sequence 12-3 reactor vessel downcomer fluid temperature.

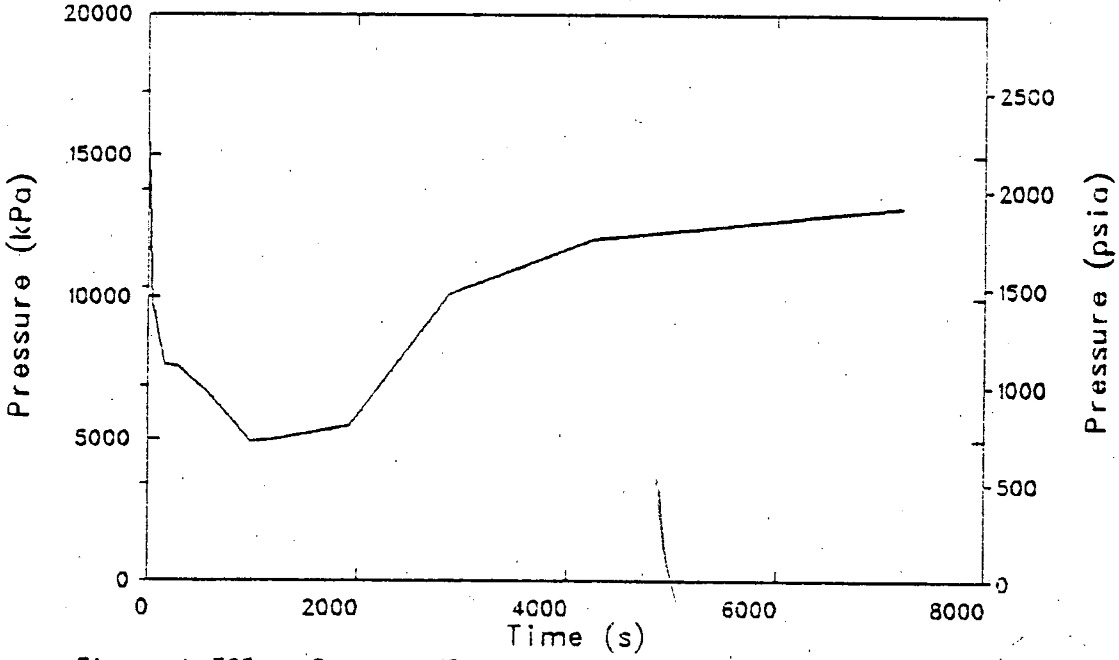

Figure A-365. Sequence 12-4 reactor vessel downcomer fluid pressure.

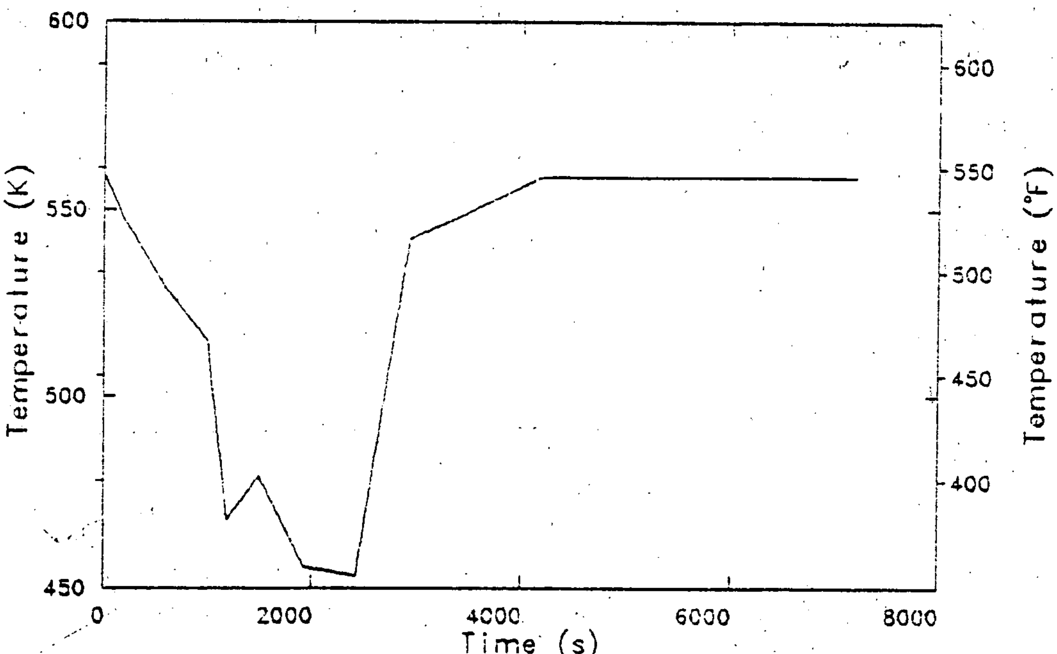

Figure A-366. Sequence 12-4 reactor vessel downcomer fluid temperature.