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EDD L9-3
March 11, 1982

LOFT EXPERIMENT DEFINITION DOCUMENT

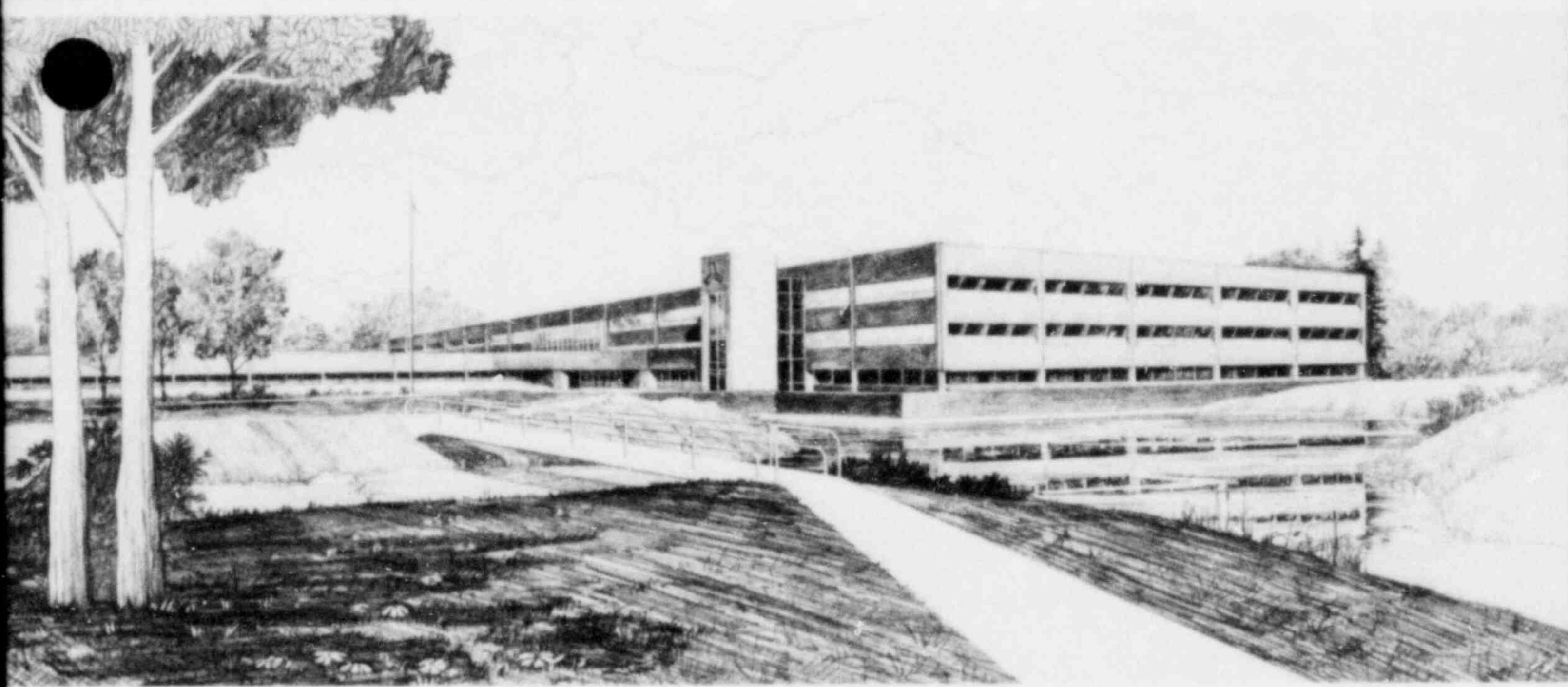
ANTICIPATED TRANSIENT TEST SERIES

NUCLEAR TEST L9-3

NRC Research and/or Technical Assistance Report

P. Kuan

U.S. Department of Energy
Idaho Operations Office • Idaho National Engineering Laboratory



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By

P. Kuan

March 11, 1982

LOFT EXPERIMENT DEFINITION DOCUMENT

ANTICIPATED TRANSIENTS WITH MULTIPLE FAILURES

NUCLEAR TEST L9-3

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FOREWORD

This document defines the objectives, system configuration, initial conditions, measurement requirements, and scenario for the Loss-of-Fluid Test (LOFT) Experiment L9-3, Anticipated Transient Without Scram: Loss of Feedwater. The information provided herein is intended to provide guidance to the preparation of (1) Experiment Operating Specification (EOS), (2) Instrument and Data Acquisition Requirements (IDAR), (3) Experiment Prediction (EP), and (4) Experiment Safety Analysis (ESA). Plant modifications and the installation of measuring instruments should also proceed on the basis of this document.

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ACRONYMS AND NOMENCLATURE

ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
BST	Blowdown Suppression Tank
CFR	Code of Federal Regulations
ECCS	Emergency Core Cooling System
EFPH	Effective-Full-Power-Hours
EOS	Experiment Operating Specification
EP	Experiment Prediction
ESA	Experiment Safety Analysis
HPIS	High Pressure Injecton System
IDAR	Instrument and Data Acquisition Requirements
k_{eff}	Effective Neutron Multiplication factor
LOCA	Loss-of-Coolant Accident
LOFT	Loss-of-Fluid Test
MPa	10^6 Pascals
MTC	Moderator Temperature Coefficient
NRC	Nuclear Regulatory Commission
PORV	Power Operated Relief Valve
PRD	Pressure Reduction and Decontamination
psia	Pounds per Square Inch, Absolute
PWR	Pressurized Water Reactor

LOFT EXPERIMENT DEFINITION DOCUMENT NUCLEAR TEST L9-3

1. INTRODUCTION

Anticipated Transients Without Scram (ATWS) for light water reactors is an unresolved safety issue of the U.S. Nuclear Regulatory Commission (NRC). The significance of ATWS in reactor safety is that some ATWS events can result in high system pressures which can potentially lead to fuel damage and the release of a large amount of fission products.

Despite differences of opinion within the nuclear industry, the regulatory staff of the NRC has consistently held that "... the likelihood of severe consequences arising from an ATWS event is acceptably small, but that the future likelihood of severe ATWS consequences could become unacceptably large and measures should be taken to diminish such consequences."(2) To address the ATWS issue, the NRC in September 1980 published a proposed ATWS rule(1) to amend the Code of Federal Regulations (10CFR50). After a year of public meetings and comments, the NRC will soon issue a revised ATWS rule (November 1981). The Code of Federal Regulations may then be amended and the regulation is expected to include measures both to reduce the likelihood of ATWS events and to mitigate the consequences of an ATWS once it has occurred.(2) The planning of the L9-3 experiment is based on NRC's regulatory position contained in SECY-80-409[1] as well as those expected to be addressed in the forthcoming revision to the ATWS rule.

In evaluating ATWS accidents the NRC lists ten initiating events for pressurized water reactors (PWRs), which are expected to occur one or more times during the life of a nuclear power unit.[3] These events can be classified into four categories, i.e., (a) reactivity related accidents (rod withdrawal, boron dilution, inactive primary loop startup, load increase, excessive cooldown), (b) degradation of reactor heat transfer (loss of primary flow, loss of electrical load, loss of normal electrical power), (c) degradation of reactor heat sink (loss of normal feedwater), and (d) primary system depressurization caused by accidental opening of a pressurizer relief valve. The L9-3 experiment is intended to simulate the important physical conditions following a loss of feedwater without scram transient hypothesized for future commercial PWRs conforming to the acceptance criteria proposed by the NRC.[1]

Upon loss of feedwater to the steam generators in a PWR power plant, the heat transfer from the primary to the secondary system is degraded with the decrease in steam generator secondary inventory. Normally the reactor will trip (insert control rods to shutdown the reactor, or scram) on a signal of low feedwater flow or low steam generator level. In the absence of a scram the steam generator secondary will soon boil dry and most of the heat produced by the reactor core will be dissipated in the primary fluid, raising its temperature. The expansion of the primary fluid associated with its temperature rise at first compresses the vapor space of the pressurizer, forcing the relief and safety valves to open. Subsequently the pressurizer will be filled with liquid water and the system pressure will continue to rise to a maximum when the volumetric relief flow rate equals the volumetric expansion rate of the primary fluid at constant pressure. It is this maximum pressure that constitutes one of the main safety concerns of ATWS events.

According to PWR vendors calculations,[4] a loss of feedwater ATWS yields one of the highest primary pressures among the initiating events mentioned earlier (the others being loss of load or rod withdrawal at zero power). With the exception of Westinghouse plants, such an accident in presently operating large commercial PWRs early in their life will result in maximum primary stresses exceeding the "Level C Service Limit" as defined in Article NCA-2000 of Section III of the ASME Boiler and Pressure Vessel Code. Such stresses may cause large deformations in areas of structural discontinuity (Level C Service Limit) or damages to components that may breach the integrity of the primary coolant pressure boundary (much higher than Level C Service Limit), leading to a loss-of-coolant accident (LOCA). The NRC's proposed regulation will limit the maximum primary stress to less than the "Level C Service Limit" in all components except in the steam generator tubes whose integrity may be evaluated based on a conservative assessment of tests and likely condition of the tubes over their design life. The LOFT L9-3 experiment is designed to achieve a peak primary pressure that will result in stress levels slightly below the "Level C Service Limit" in commercial PWRs. The L9-3 peak pressure will, therefore, be representative of the maximum expected pressure in commercial reactors that will be allowed under future rulemaking.

Another major concern of a loss of feedwater without scram accident is the long-term shutdown capabilities of PWR systems after the initial peak pressure has passed. According to the NRC staff, after evaluating the PWR vendors submittals in response to an NRC request on ATWS analysis, long-term shutdown has not been adequately addressed by the vendors.[2] One reason is that the transient computer codes[5,6,7] used by the PWR vendors are no longer applicable when significant voids appear in the primary system after coolant loss from the power operated relief valves (PORV)/safety valves; another is that the vendors have not clearly delineated the recovery procedures and the corresponding mitigating systems. The L9-3 experiment will explore a way to depressurize the primary system by timely latching open the PORV and by using the auxiliary feedwater system for additional heat removal. In addition, high concentration boron solution will be injected into the system to permanently shut down the reactor. This will be at least a first step in bringing the reactor to a stabl· cold shutdown condition after a loss of feedwater ATWS.

2. EXPERIMENT OBJECTIVES

To address issues relating to system response and plant recovery procedures following a loss-of-feedwater ATWS in a commercial nuclear power plant, the following programmatic objectives have been established for the L9-3 experiment:

1. Provide experimental data for benchmarking PWR vendors' ATWS computer codes as requred by the NRC proposed ATWS rule (USNRC SECY-80-409).

2. Evaluate alternate methods of achieving long term shutdown (without the insertion of control rods) following an ATWS event, to address concerns defined in the proposed NRC staff rule (Federal Register Vol. 46, No. 226).

To support the above programmatic objectives, several specific test objectives for the L9-3 experiment have been defined. In establishing these test objectives, it is realized that results from the LOFT experiment may not necessarily be directly applicable to the larger commercial plants. However, by application of the codes to LOFT results, it is expected that an assessment of the capabilities of the codes to predict important system response characteristics during an ATWS event in a commercial pressurized water reactor can be obtained. Therefore, the test specific objectives for L9-3 are:

1. To achieve a maximum primary system pressure that is several measuring standard errors above the code safety valve opening pressure setpoint but below 110% of the setpoint pressure.

2. To determine the transient reactor power by using available neutron flux instrumentation and measured core thermal-hydraulic parameters to assess the applicability of the point kinetics model used in predicting transient reactor power.

3. To determine the steam generator secondary dryout behavior and its effect on the primary system response characteristics.

4. To determine the two-phase and subcooled flow characteristics of the experimental pressurizer PORV and safety valve at high pressures ($\geq$ 17 MPa (2500 psia)).

In support of the above objectives, the test should provide data on at least the following parameters with a time resolution of 1 second or less in the first 250 seconds of the transient:

1. Pressurizer pressure, level, and temperature

2. Primary system average fluid temperature, core average fluid temperature, and cladding temperatures

3. Steam generator level and pressure

4. Transient reactor fission power derived from neutron flux measurements

5. Pressurizer PORV and safety valve flow rates

6. An estimate of the primary system leakage rate in addition to PORV and safety valve flow rates.

The recovery procedure consists of manually latching open the PORV to depressurize the primary system and manually initiating high pressure injection with high concentration boron solution. The auxiliary feedwater system will be used to regulate the cooldown rate of the primary system. Specifically, the objective of the recovery procedure is to answer the question:

With a present design of the high pressure injection system and the auxiliary feedwater system (the Trojan PWR), can the reactor be brought to a safe cold shutdown condition without inserting the control rods following a loss of main feedwater transient?

3. SYSTEM CONFIGURATION

The system configuration for the L9-3 experiment is shown in Figure 1. During most of the transient, especially during the phase when the experiment is to provide benchmark data for computer code assessment, the inactive broken loop fluid has negligible effects on the system response. For this reason and consistent with plant modification constraints, the steam generator and primary coolant pump simulators in the broken loop shall be isolated from the system; the broken loop hot leg shall terminate at flange FL-19 and the broken loop cold leg shall terminate at isolation valve CV-P-138-2. Other explanations on the system configuration are given below.

1. The reflood assist bypass valves CV-P-138-70 and CV-P-138-71 shall be closed during the test.

2. The pressurizer spray bypass valve V-4031-IVE shall be open during the test and the pressurizer spray control valve CV-P-139-5-1 shall open and close in response to normal pressurizer pressure control signals. Full flow through the spray control valve should be 0.57 kg/s (12 gpm, power scaled to a Westinghouse PWR).

3. The pressurizer cycling and backup heaters shall be operative.

4. The primary coolant pump injection system shall be isolated.

5. The primary coolant pumps PC-P-1 and PC-P-2 shall be operative.

6. The plant PORV isolation valve CV-P-139-18 shall be open and the plant PORV CV-P-139-5-4 shall be inactive and closed.

7. The experimental PORV and safety valves shall be simulated by a single valve, CV-P-139-87, with a double actuator such that the

first position corresponds to the PORV and the second position corresponds to the PORV and the safety valves combined. (The installation of a single valve in lieu of two or more valves in parallel, as in a commercial PWR, is due to the limitation of plant space.) The PORV setpoints are: Open, 16.20 MPa, (2350 psia); close, 16.00 MPa (2320 psia). The safety valve setpoints are: Open, 17.24 MPa, (2500 psia); close, 16.46 MPa, (2388 psia).

8. The plant safety valves RV-200 and RV-201 lifting setpoint shall be 19.3 MPa (2800 psia) and be designed in such a way as to give the maximum relief flow consistent with the plant configuration. A discussion of the setpoint is provided in Section 7.1.

9. The main steam isolation valve CV-P4-11 shall be open.

10. The steam control valve CV-P4-10 shall open and close according to requirements given in Section 6.

11. The main feedwater shall be shut off at the initiation of the test, and the auxiliary feedwater system shall be operative in accordance with requirements set in Section 6.

12. The primary coolant purification system shall be isolated during the test.

13. Only the high pressure injection system (HPIS) of the emergency core cooling system (ECCS) will be used in the test. The HPIS should be able to inject 1000 kg (2200 lb) of 7000 parts per million (ppm) boron (by weight) solution (temperature ≈300 K, 80°F) at a rate of 0.38 kg/s (6 gpm) up to a primary system pressure of 17.2 MPa (2500 psia). The injection is into the downcomer of the reactor vessel.

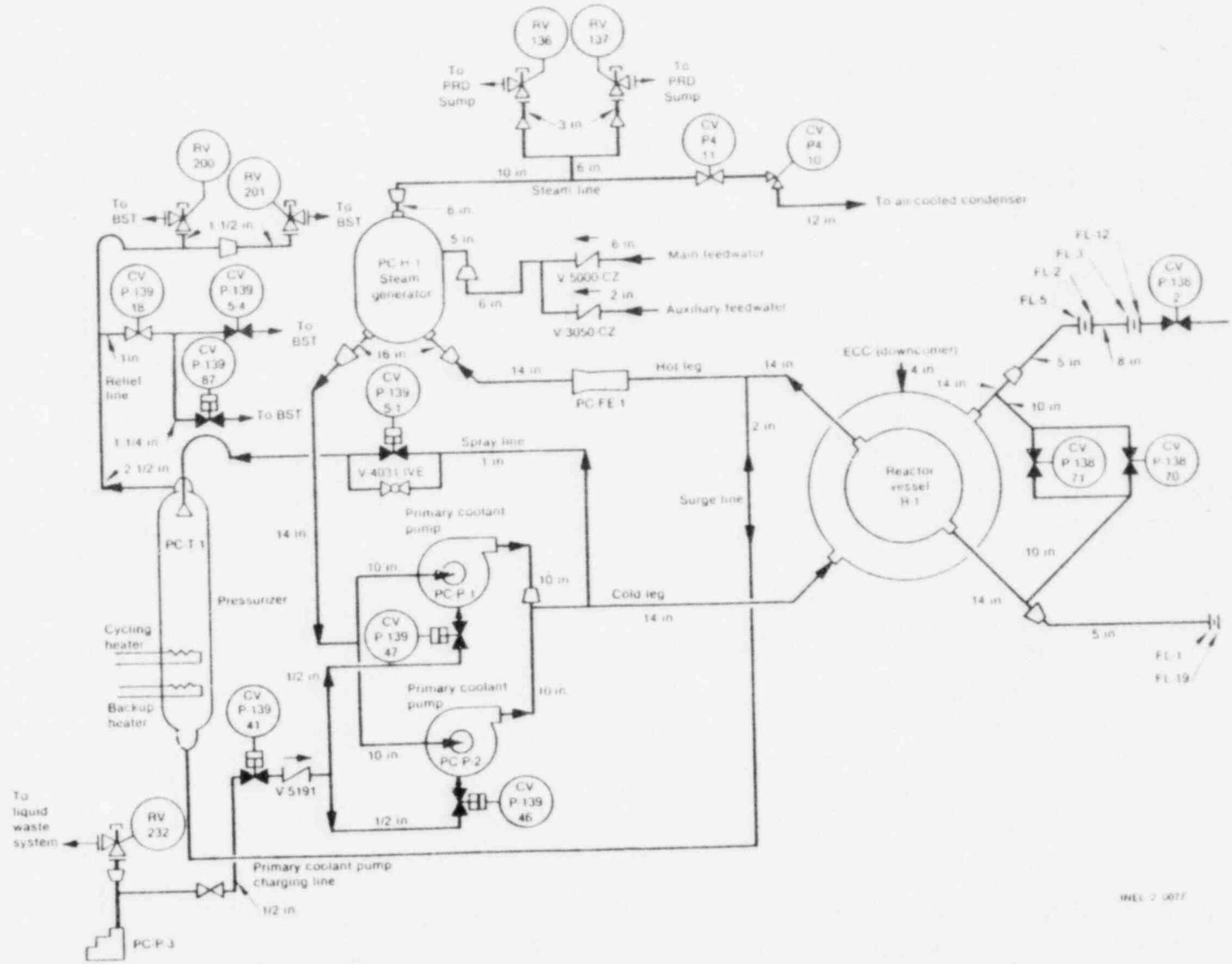

Figure 1. LOFT L9-3 test system configuration

4. INITIAL CONDITIONS

Normal operating conditions in commercial PWRs vary from plant to plant, so exact typicality considerations have little meaning. Instead, the L9-3 Test initial conditions are designed to approximate commercial PWR operating conditions and at the same time to be consistent with the safety analysis of the LOFT plant itself without compromising experiment objectives.

The L9-3 experiment initial conditions are defined as:

1.	Reactor power	49.5 ± 0.5 MW
2.	Average primary temperature	569.3 ± 1 K (565 ± 2°F)
3.	Core delta T (T_{hot} - T_{cold})	21.1 ± 1 K (38 ± 2°F)
4.	Pressurizer pressure	14.95 ± 0.1 MPa (2169 ± 15 psia)
5.	Pressurizer level	1.168 ± 0.05 m (46 ± 2 inches)
6.	Primary coolant pumps speed	Consistent with reactor power and core delta T
7.	Control rod position	1.37 ± 0.1 m (54.0 ± 0.5 inches)
8.	Main feedwater temperature	Consistent with primary conditions
9.	Main feedwater flow	Consistent with primary conditions
10.	Main steam flow	Consistent with primary conditions
11.	Steam pressure	Consistent with primary conditions
12.	Steam generator level	3.2 ± 0.1 m (126 ± 4 inches) above top of tube sheet
13.	Boron concentration	Enough to keep reactor critical

5. MEASUREMENT REQUIREMENTS

The following are important measurements to be obtained during the L9-3 experiment:

1. Pressurizer pressure, level, and temperature,

2. Hot leg fluid temperature (T_{hot}) and cold leg fluid temperature (T_{cold}), core average fluid temperature, cladding temperatures,

3. Steam generator level and pressure; steam flow rate,

4. Reactor neutron fluxes,

5. Pressurizer experimental PORV and safety valve flow rate and its upstream density.

In addition to the above, the following should be ascertained before the test, monitored during the test or after the test:

1. Pressurizer spray and pressurizer spray bypass flows,

2. Primary coolant pumps speed and primary system loop flow rate,

3. Auxiliary feedwater flow rate,

4. ECC injection flow rate,

5. Pretest monitoring of primary system leakage rate, and

6. Blowdown suppression tank (BST) inventory.

The important parameters of the test are shown in the planning calculation (Appendix).

6. SEQUENCE OF EVENTS

Prior to the initiation of the L9-3 Test, the LOFT reactor should be operated continuously near 50 MW such that the decay heat level at 1000 seconds after a scram from 50 MW should be at least 800 kW. This corresponds to an operating history of over 40 effective-full-power-hours (EFPH). After satisfying the requirement on decay heat build-up, the reactor should be brought to the initial conditions specified in Section 4 as soon as practicable. During this period of pre-test operation, the plant leakage rate should be carefully monitored. Experimental data recording should start at least 1 minute prior to experiment initiation.

To initiate the test, the main feedwater to the steam generator shall be shut off at the fastest possible rate. The normal scram system (reactor protection system) should be inhibited such that the control rods will remain at 1.37 m (54.0 inches) above the full-in position throughout the test. The primary coolant pumps shall be kept running at constant speed as at the initiation of the test. The pressurizer sprays and heaters should respond to plant conditions as during normal operation. The experimental PORV and safety valve should operate as designed for the test.

With the steam control valve left open at its initial position, the steam generator is expected to dry out about 100 seconds after termination of feedwater flow. The steam control valve should be closed upon indication of steam generator dryout (steam pressure below 4.14 MPa (600 psia)). The last action is designed to protect the steam generator tubes from overpressure but with little extra perturbation on the primary transient.

The maximum primary pressure is expected to occur within about 150 seconds after both the experimental PORV and safety valves are open and when they are discharging subcooled water from the pressurizer.

Subsequently the valves will cycle near their respective setpoints. No manual actions should be taken until at least 10 minutes into the transient (measured from the initiating time).

The recovery procedure starts at 10 minutes into the transient. The operator actions are: (1) latching open the PORV, (2) initiating auxiliary feedwater, and (3) initiating high pressure injection. Subsequently the PORV should be closed when the pressurizer pressure drops to 15.0 MPa (2175 psia). The auxiliary feedwater (temperature 294 K, 70°F) flow rate should be used to control primary cooldown until the average primary coolant temperature drops to 583 K (590°F) and then it should be used to regulate the average primary temperature at around 588 K (600°F), or be shut off to maintain minimum cooldown of the primary system. The steam generator secondary pressure should be controlled above 4.14 MPa (600 psia). The high pressure injection should be kept on until test termination (flow rate 0.38 kg/s, 6 gpm). The test will be complete when 10 minutes has elapsed since the average primary fluid temperature first drops to 583 K (590°F) after the initiation of the recovery procedures.

The expected test transient history is shown in the Appendix at the end of this document.

7. DISCUSSION

7.1 Maximum Pressure Planning

According to NRC's proposed regulation, the maximum calculated pressure for an early-in-life commercial PWR during an hypothesized ATWS accident should be below a pressure that will cause a Service Level C Limit stress to occur in the primary system except in the steam generator tubes (see Introduction). This pressure corresponds approximately to 120% of the design pressure which is usually 17.24 MPa (2500 psia). In order to provide benchmark data on the most severe pressure transient resulting from an ATWS for code assessment, it is therefore necessary to achieve a peak pressure slightly below 20.69 MPa (3000 psia) during the experiment.

The LOFT plant at the present has a design pressure of 17.24 MPa (2500 psia), a typical design pressure for commercial PWRs. If the maximum pressure during the L9-3 experiment does indeed cause Level C stress in the primary system (corresponding to a pressure above 18.96 MPa (2750 psia)), extensive requalification of the plant may have to be performed after the test and damaged components, if any, have to be replaced. Cost considerations, therefore, set an upper limit on the pressure to be achieved during the test.

The lower limit on the maximum pressure to be achieved during the L9-3 experiment is constrained by the experimental objective of obtaining benchmark data for code assessments. Measurement uncertainties, plant test conditions, and model sensitivities probably will result in maximum pressure uncertainties of 1.4 MPa (200 psia). In order to evaluate the capabilities of the codes to predict a peak pressure, the planned peak pressure should therefore be approximately 1.4 MPa (200 psia) above the safety valve setpoint, or the design pressure.

Based on the above discussion, the planning effort has been directed toward achieving a maximum pressure of 18.6 MPa (2700 psia), a pressure that will not require extensive plant requalification and at the same time will provide some challenge to model calculations.

Although at a pressure of 18.6 MPa (2700 psia) the LOFT plant is not expected to be damaged, the plant must be protected against higher pressures during the test. The two plant safety valves (RV 200 and RV 201) will be installed for this purpose. These two valves are designed to give the maximum relief capacity consistent with the current plant configuration such that when they open at any time during the transient, further pressure rise is limited to a few tenths of MPa (a few tens of psi). If these valves would open, the experiment, of course, would be terminated. In order to prevent an early abort of the test, the opening set-point of these valves is planned to be set at 19.3 MPa (2800 psia) pressurizer pressure.

7.2 Parameters Affecting the Value of the Maximum Pressure

The most important parameters affecting the maximum pressure attained during a loss of feedwater ATWS are (a) initial power to coolant volume ratio, (b) moderator temperature coefficient, (c) steam generator inventory at the time when feedwater is lost, (d) the timing of turbine trip (closing of the steam control valve for LOFT), (e) initial pressurizer vapor volume, and (f) total relief capacity of the pressurizer. These are discussed individually in the following subsections.

7.2.1 Power to Volume Ratio

The initial reactor power to coolant volume ratio affects the pressure transient in two ways through the rise in the coolant temperature after the heat sink is degraded or lost. First, a rise in the coolant, or moderator, temperature will cause a decrease in reactivity and, hence, in reactor power; the higher the rise in temperature, the more the decrease in reactor power. A high power to volume ratio will cause a higher initial temperature rise rate, but a faster feedback so that the eventual temperature rise rate will be alleviated to some extent. (It should be reiterated here that the peak pressure occurs when the primary coolant expansion rate at constant pressure, which is a function of the temperature rise rate, balances the relief rate.) Secondly, for the same power, a small coolant volume will have a higher temperature rise, but since the coolant expansion coefficient (volume increase per unit of temperature) increases with temperature, this will tend to increase the maximum pressure.

The coolant "volume" that is being discussed in this subsection is more appropriately defined as the effective volume of the coolant that is actually heated up during the transient. Therefore the pressurizer and dead-ended volumes in the system should be excluded in the consideration. The effective volume of LOFT is 5.4 m^3 (190 ft^3) and its initial power is 50 MW. The Trojan plant, a Westinghouse PWR, has an effective volume of 303 M^3 (10,700 ft^3) and a power of about 3400 MW. The power to volume ratios of LOFT and Trojan are respectively 9.3 MW/m^3 (0.26 MW/ft^3) and 11.2 MW/m^3 (0.32 MW/ft^3). The lower ratio for LOFT will tend to give a lower peak pressure during a loss of feedwater ATWS.

7.2.2 Moderator Temperature Coefficient

What is usually referred to as the moderator temperature coefficient (MTC) is actually a combined coefficient of several factors. The moderator in a PWR is a dilute aqueous boric acid solution. The boron in the solution is used to reduce the excess reactivity in the reactor core and the water molecules to moderate fission neutrons to thermal energies for further fission. As the temperature of the moderator rises, it expands and the moderating capability is reduced. This will (a) cause more leakage of neutrons from the reactor core and (b) less fission due to the reduction in fission cross-section. On the other hand the reduction in boron density due to the expansion will decrease the neutron absorption rate, preserving more neutrons for further fission. These combined effects in general make the MTC negative in PWRs, except possibly at the very beginning of their operating life. The MTC decreases (becomes more negative) with operating life of a PWR as the boron concentration is reduced.

The MTC is usually expressed as a change in k_{eff} per degree change in the moderator temperature. From a reactor power transient point of view it is more appropriately expressed in dollars (k_{eff} divided by the delayed neutron fraction) per degree temperature. For generic Westinghouse PWRs early in their operating life, the MTC is about -\$0.02/K (-\$0.012/°F). For LOFT at the time of the L9-3 Test, it is estimated to be -\$0.069/K (-\$0.039/°F). The more negative LOFT MTC will give a faster power decrease for the same rise in temperature. Consequently if all other parameters were scaled to Westinghouse PWRs, the peak pressure during the L9-3 Test would be less than those experienced by Westinghouse PWRs early in their operating life.

7.2.3 Steam Generator Secondary Inventory

The initial steam generator secondary inventory determines the power level of the reactor when the steam generator is boiled dry completely. Following steam generator dryout, practically all the heat generated in the reactor core will be dissipated in the primary coolant, raising its temperature. The larger the initial steam generator inventory, the longer will be the dryout time and consequently the lower will be the power level at the time of dryout due to moderator temperature-reactivity effects. Therefore the maximum pressure during the transient will be lower for a reactor system having a larger initial steam generator inventory. For reactors of different powers, the ratio of the steam generator inventory to the power is the relevant parameter.

For 4-loop Westinghouse PWRs, the steam generator inventory is equivalent to the steam produced in about 100 full-power-seconds; for LOFT, it is 75 full-power-seconds. Assuming other conditions being typical, the LOFT reactor will have a higher maximum pressure during a loss of feedwater ATWS than 4-loop Westinghouse PWRs.

7.2.4 Timing of Turbine Trip

In a Westinghouse PWR, a turbine trip (closure of all turbine steam admission valves) will be initiated on receiving a reactor trip signal. In an ATWS accident, the failure to trip the reactor is usually attributed to the failure to cut off electrical power to the reactor control rods or to mechanical blockage in the reactor core such that the control rods cannot be inserted. The contribution to the probability of failure to scram from signal failures is negligibly small because of the reliability of electronic systems and the degree of redundancy built into the systems. Upon loss of feedwater to the steam generators, a reactor trip signal may be generated from (a) steam flow-feedwater flow mismatch coincident with low steam generator water level, (b) low-low steam generator water level, (c) pressurizer high pressure, or (d) pressurizer high water level. In an ATWS, a turbine trip is, therefore, likely to occur before the steam generators boil dry.

A turbine trip before steam generator dryout will have the effect of temporarily raising the primary temperature, reducing the reactor power because of the moderator temperature rise. The steam generated following turbine trip will be dumped to the atmosphere through relief valves in a commercial PWR. Due to the early reduction in the reactor power, the integrated energy dissipation in the primary coolant will be less than it would be otherwise without a turbine trip before steam generator dryout. Therefore the earlier the turbine trip, the less will be the maximum pressure achieved during an ATWS.

In the L9-3 Test, the steam control valve will be closed when the steam generator secondary dries out (pressure drops below 4.14 MPa (600 psia)). This is not a typical action in a commercial PWR, but is designed to achieve a primary pressure higher than the pressurizer safety valve setpoint and to protect the integrity of the steam generator tubes.

Without such a late steam control valve closure, the primary pressure will peak only slightly above the safety valve setpoint due to the large negative LOFT MTC.

7.2.5 Pressurizer Vapor Volume

During a loss of feedwater ATWS, the maximum pressure is expected to occur after the pressurizer is totally filled with liquid water from the expansion of the primary coolant. The time required to achieve this depends on the ratio of the initial pressurizer vapor volume to the effective primary coolant volume; the higher the ratio, the longer the time interval, assuming the same power transient. The reactor power is expected to be a monotonically decreasing function of time, so the peak pressure will be lower when it occurs later in the transient, since lower reactor power means lower coolant expansion rate. The relief capacity is assumed to increase with pressure.

Pressurizer level settings vary from plant to plant in commercial PWRs. For example, the Zion plant has the same pressurizer volume as the Trojan plant (51 m^3 or 1800 ft^3), but the Zion operating pressurizer vapor volume is about 18 m^3 (650 ft^3)[10] while that of Trojan is 34 m^3 (1200 ft^3), according to the Trojan Final Safety Analysis Report.[11] The LOFT L9-3 Test initial pressurizer vapor volume is approximately 0.28 m^3 (10 ft^3). The ratios of pressurizer vapor volume to effective primary coolant volume are 5.3% for LOFT, 5.6% for Trojan, and 10.7% for Zion. The reason for setting a low ratio for LOFT is again intended to increase the maximum pressure during the transient to counteract the overwhelming effect of a too negative MTC. The setting is also consistent with safety requirements for the operation of LOFT.

7.2.6 Primary Relief Capacity

The primary relief capacity is probably the most important parameter which can be adjusted in LOFT to give the desired maximum pressure during a loss of feedwater ATWS. The total relief capacity in most commercial PWRs consists of relief flow from the PORVs and the code safety valves installed on the pressurizer. The higher the relief capacity, the lower will be the maximum pressure during an ATWS, assuming the maximum pressure exceeds the opening setpoints of the valves. If the volumetric relief flow rate is lower than the primary coolant expansion rate at constant pressure when the valves first open, the pressure will keep increasing until the relief flow rate equals the expansion rate.

The L9-3 Test PORV is sized to relieve 0.66 kg/s (5250 lb/hr) of saturated steam at 16.2 MPa (2350 psia) upstream pressure, a value consistent with the minimum PORV capacity of Westinghouse PWRs on a power scaled basis (1.33×10^{-2} kg/s-MW, or 105 lb/hr-MW). In the initial pressure rise during the transient, the PORV will be open at 16.2 MPa (2350 psia), but the pressure will keep rising until the safety valve opening setpoint (17.2 MPa, or 2500 psia) is reached. Further rise in pressure will depend on the relief capacity designed for the safety valve.

In order to achieve a maximum pressure of 18.6 MPa (2700 psia, see Section 7.1), planning calculations show that the safety valve relief capacity should be set at 1.26 kg/s (10,000 lb/hr) saturated steam at 17.2 MPa (2500 psia) upstream pressure. Incidentally, this relief capacity approximately corresponds to the relief capacity of two out of three safety valves in Westinghouse PWRs on a power scaled basis.

7.3 Auxiliary Feedwater

In a Westinghouse PWR, when the steam generator feedwater pumps are tripped, the auxiliary feedwater pumps will automatically start and deliver low temperature feedwater to the steam generators. The auxiliary feedwater system will also automatically operate when a low-low steam generator water level is detected. In general, part of the auxiliary feedwater is delivered by a steam turbine driven pump, the rest by an electrically driven pump(s). Trojan plant has two electrically driven auxiliary feedwater pumps, each capable of delivering 60 kg/s (960 gpm) 294 K (70°F) water. The LOFT auxiliary feedwater system is capable of delivering 1.0 kg/s (16 gpm) 294 K (70°F) water. An attempt should be made to use the full capacity to simulate approximately that of one electrically driven auxiliary feedwater pump at the Trojan plant on a power scaled basis. Actual plant temperature and pressure condition at the time of recovery may require lesser feedwater flow rate to sucessfully regain plant control.

Due to the large negative MTC in LOFT, the startup of the auxiliary feedwater will be delayed until the pressure peak has passed. In fact, the auxiliary feedwater will be delayed until the recovery portion of the test due to temperature typicality considerations. (See Section 7.4.2)

7.4 Recovery Procedure

The concept of a recovery from an ATWS is to bring the reactor to a stable cold shutdown condition indefinitely without ever inserting the control rods. To do this, high concentration boric acid solution has to be injected into the reactor core. In the Trojan plant, upon initiation of high pressure injection, 3.4 m^3 (120 ft^3) of 21,000 ppm by weight boron solution will be injected into the primary system against a system pressure below about 18 MPa (2600 psia) followed by an almost unlimited amount of 2000 ppm by weight boron solution. During the early operating life of a commercial PWR, the capacity of the 21,000 ppm boron may not be capable of assuring reactor subcriticality at cold shutdown temperatures less than 370 K (≈200°F) wihtout control rod insertion. So a large amount of 2000 ppm boron injection may be required over a long period of time. If during this time the primary temperature decreases so fast that the positive reactivity addition from the moderator temperature feedback exceeds the negative reactivity addition from the boron solution, the reactor will become critical again and the subsequent recovery will be considerably more complicated. The L9-4 Test recovery is designed to both inject boron solution and control the primary system cooldown rate.

7.4.1 High Pressure Injection Boric Acid Solution

Owing to the limitation of the LOFT facility and the short period allowed for test planning and preparation, the LOFT HPIS cannot be modified to simulate that of commercial PWRs in the L9-3 Test. Instead, 7000 ppm boron solution will be used in the test. This will assure that the reactor will be kept sufficiently subcritical during the recovery such that recriticality will not occur.

Another typicality of the L9-3 Test HPIS is that the injection is into the reactor vessel downcomer instead of the cold leg. The reason is that the cold leg injection train of the HPIS is required for pre-test reactor control with low concentration boron solution. This difference in the injection location in the L9-3 Test and is commercial PWR is not expected to be significant.

7.4.2 Control of the Auxiliary Feedwater Flow

The initiation of auxiliary feedwater at 10 minutes into the test is not considered as part of operator actions. In a commercial PWR, the auxiliary feedwater will come on quite early after the loss of main feedwater to the steam generators as mentioned earlier (Section 7.3). If a loss of feedwater ATWS were to occur at a commercial PWR early in its operating life, even with early auxiliary feedwater initiation, the primary coolant temperature would approach its saturation temperature in a short interval due to the slow decrease in reactor power. On the other hand, the large negative LOFT MTC will drive down the reactor power quite fast, giving a much slower temperature rise. The delay of the initiation of the auxiliary feedwater in the L9-3 Test is designed to achieve temperature typicality to a commercial PWR in its early operating life, just as the relief valve is sized to achieve pressure typicality (Section 7.2.6).

Later in the recovery, the auxiliary feedwater will be turned on and off to control the primary coolant temperature around 590 K (≈600°F) or to achieve a minimum cooldown rate by turning it off until test termination. These actions are properly considered as operator actions.

7.4.3 Termination of the Test

The termination criterion as discussed in Section 6 will assure the acquisition of the necessary information for assessing the interaction characteristics of the HPIS fluid with the primary coolant. It will also give valuable information on the controllability of the reactor system toward a cold shutdown without inserting the control rods.

REFERENCES

1. "Proposed Rulemaking to Amend 10 CFR Part 50 Concerning Anticipated Transients Without Scram (ATWS) Events," USNRC SECY-80-409, September 4, 1980.

2. "Anticipated Transients Without Scram for Light Water Reactors," NUREG 0460, Vol. 4, March 1980.

3. "Anticipated Transients Without Scram for Light Water Reactors," NUREG-0460, Vol. 2, Appendix IV, April 1978.

4. "Anticipated Transients Without Scram for Light Water Reactors," NUREG-0460, Vol. 2, Appendices XIV, XV, and XVII, April 1978.

5. "CADDS - Computer Applications to Direct Digital Simulation of Transients in PWRs With or Without Scram," Babcock & Wilcox Co. Report BAW-10098, Rev. 1, February 1978.

6. CENPD-107P, "CESEC - Digital Simulation of a Combustion Engineering Nuclear Supply System," April 1974 and Suppl. 1P, April 1974, Suppl. 1 Amendent 1P, November 1975, Suppl. 3, August 1975, Suppl. 4-P, December 1975, Suppl 5-P, June 1976 on "ATWS Model Modifications to CESEC."

7. "LOFTRAN Code Description," Westinghouse Electric Corporation Report WCAP-7878, Rev. 1, January 1977.

8. "Proposed Rulemaking to Amend 10 CFR Part 50 Concerning Anticipated Transients Without Scram (ATWS) Events," USNRC SECY-80-409, Enclosure E, September 4, 1980.

9. "Anticipated Transients Without Scram for Light Water Reactors," NUREG-0460, Vol. 4, Appendix B, March 1980.

10. J. E. Koske, Private Communication.

11. "Trojan Nuclear Plant Final Safety Analysis Report," Portland General Electric Company, USAEC Docket No. 50-344, February, 1973.

APPENDIX

PLANNING CALCULATION

The figures shown in this appendix are obtained from a RELAP5 calculation using a LOFT model derived from the one used in the LOFT L9-1 Test Experiment Prediction. The calculation should not be considered as an experiment prediction since the model has not incorporated the improvements made since the L9-1 Test, such as ambient heat losses and an upgraded steam generator model. The calculation also does not follow precisely the parameters defined in this document. The figures should be used as a visual aid to the understanding of the test and as a reference to design measuring instrument ranges.

LOFT L9-3 TEST: ATWS, LOSS OF FEEDWATER

INITIAL CONDITIONS

Average Primary Temperature.................565 F
Pressurizer Pressure.......................2168 psia
Core Delta T...............................38 F
Pressurizer Level..........................46 in
Reactor Power..............................50.0 MW
Operating History..........................40 efph

Auxiliary Feedwater at 600 seconds, 16 gpm, 70 F

OPERATOR ACTIONS AT 600 SECONDS

Latch open PORV to depressurize to 2175 psia and then close it

Initiate safety injection, 6 gpm, 70 F, 3000 ppm boron

EXPERIMENTAL RELIEF VALVE CHARACTERISTICS

PORV: Open, 2350 psia; Close, 2320 psia;
Saturated Steam Flow at 2350 psia: 5250 lb/hr

Safety: Open, 2500 psia; Close, 2470 psia;
Saturated Steam Flow at 2500 psia: 10,000 lb/hr

Steam Generator Secondary Pressure subsequently controlled between 400 and 450 psia

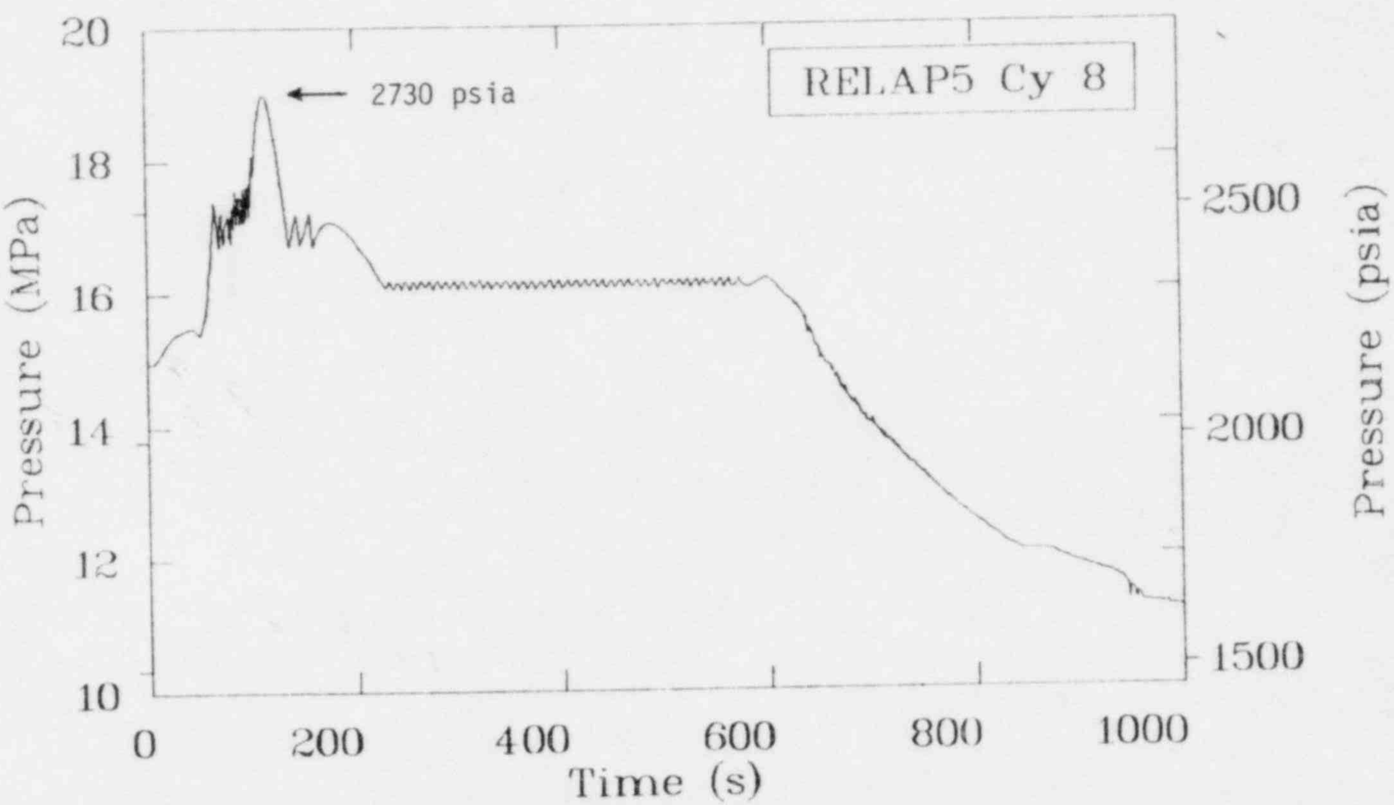

Figure 1 – Pressurizer Pressure
L9-3 ATWS – Loss of Feedwater

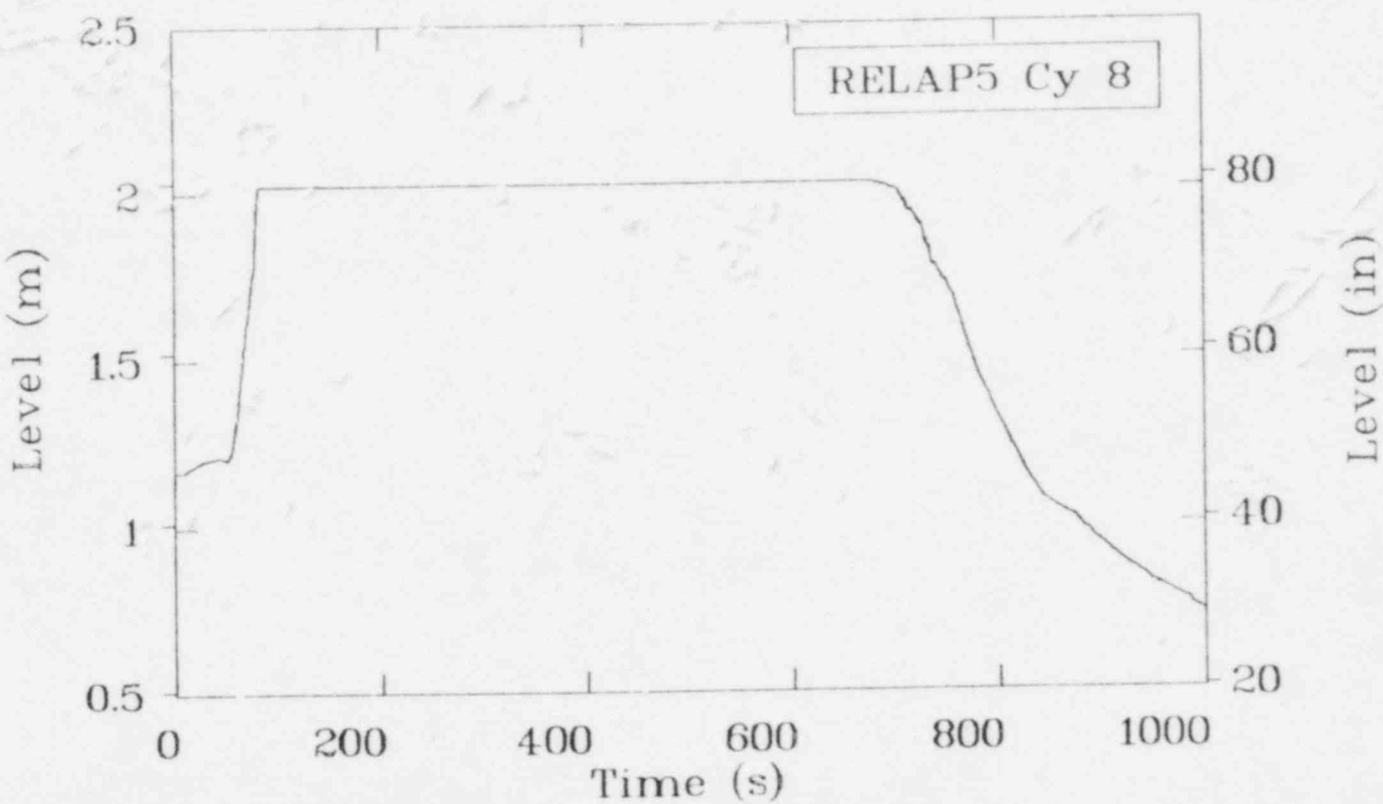

Figure 2 – Pressurizer Level
L9–3 ATWS – Loss of Feedwater

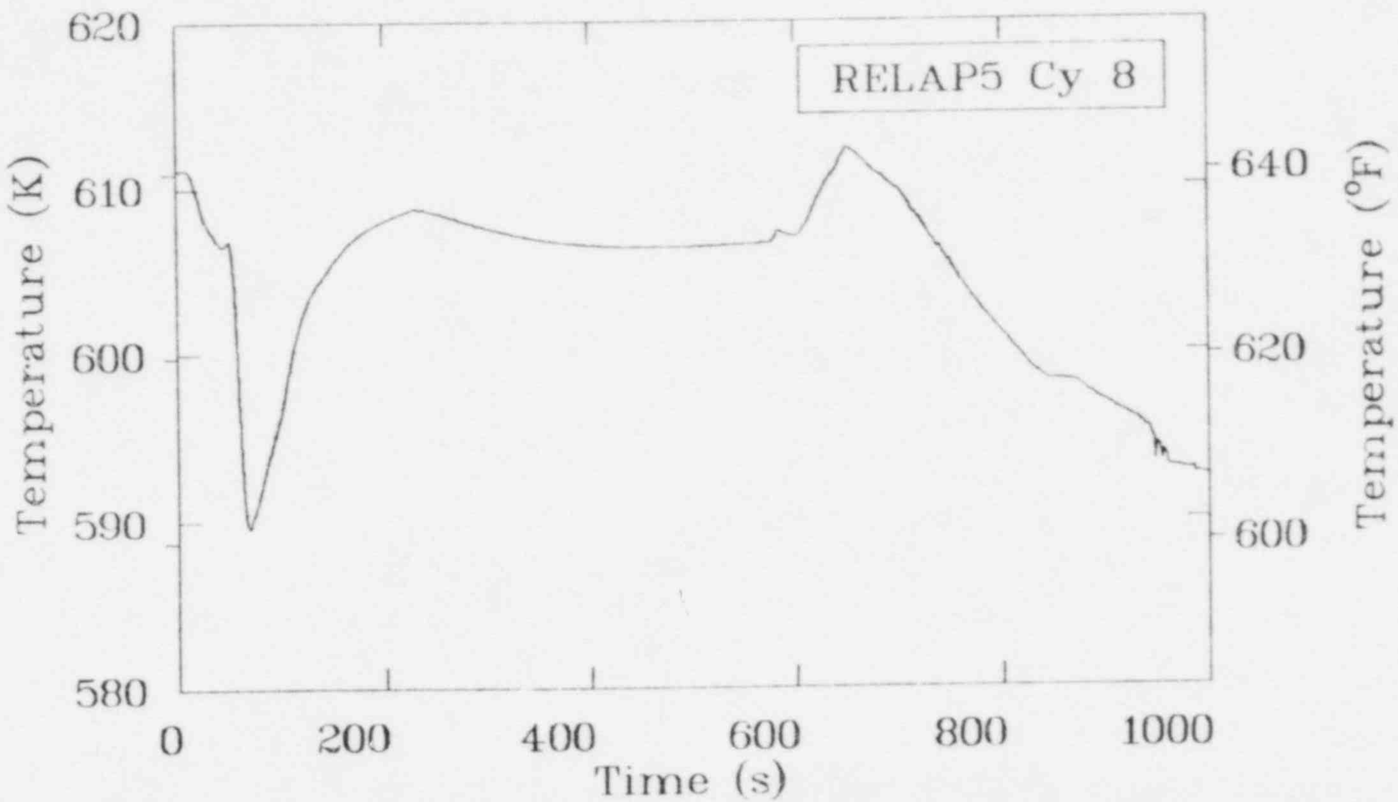

Figure 3 – Lower Pressurizer Temperature
L9–3 ATWS – Loss of Feedwater

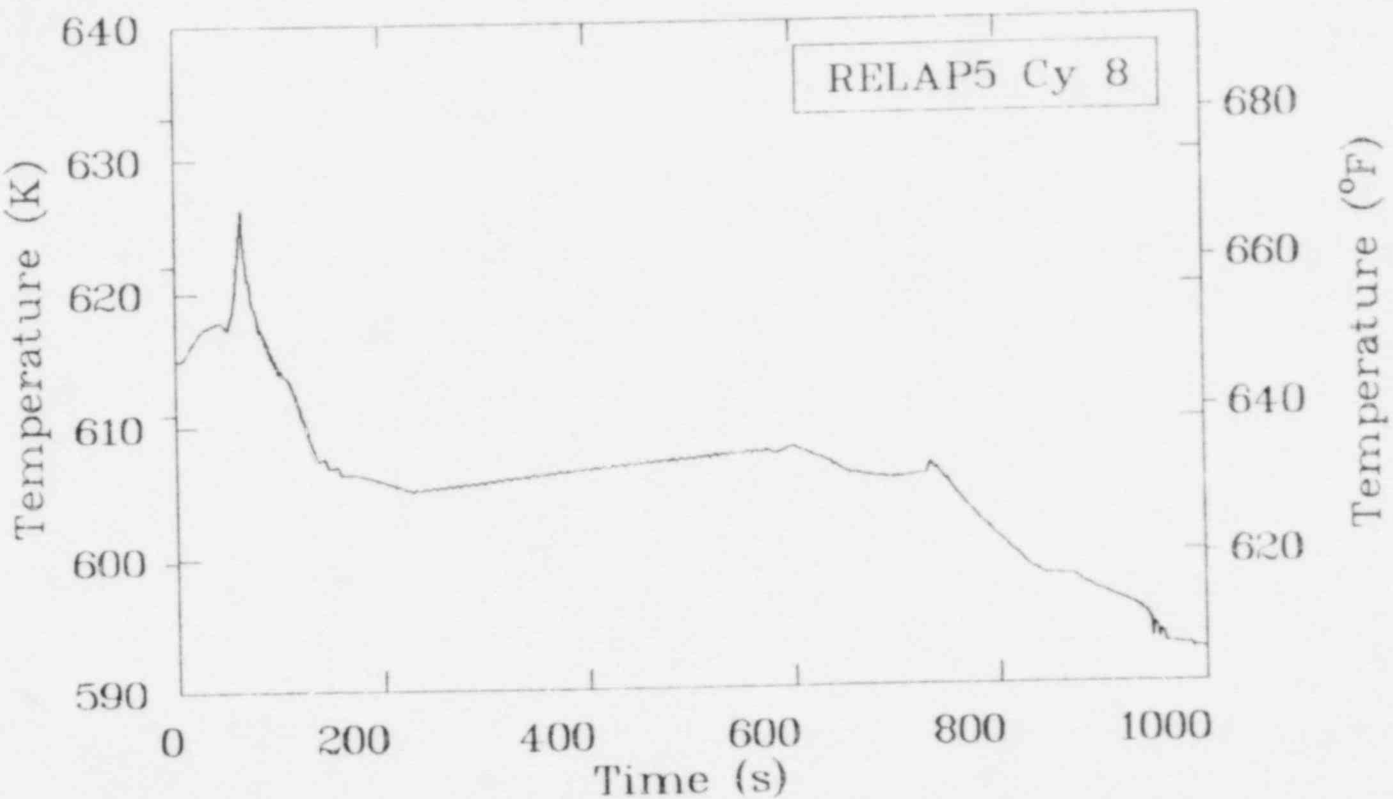

Figure 4 – Mid Pressurizer Temperature
L9-3 ATWS – Loss of Feedwater

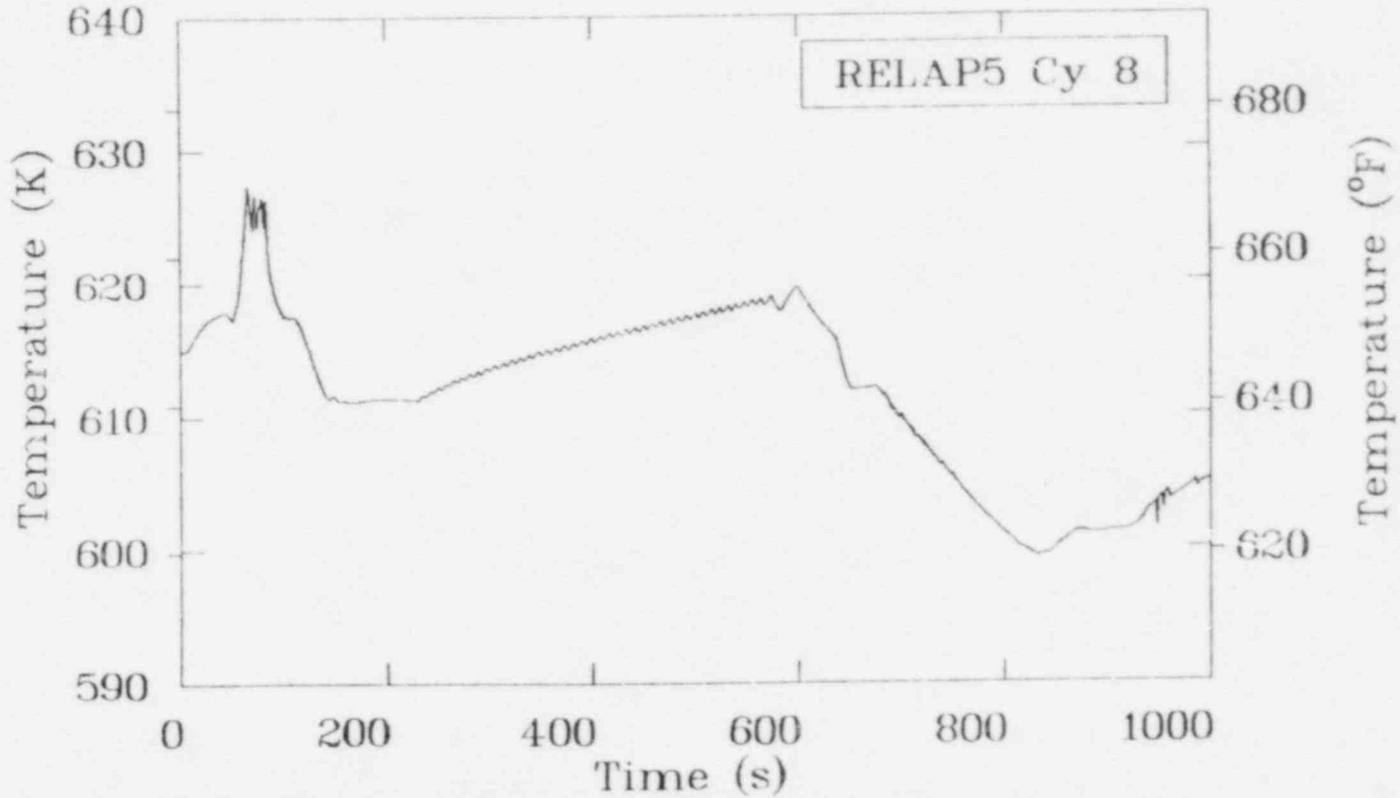

Figure 5 – Upper Pressurizer Temperature
L9–3 ATWS – Loss of Feedwater

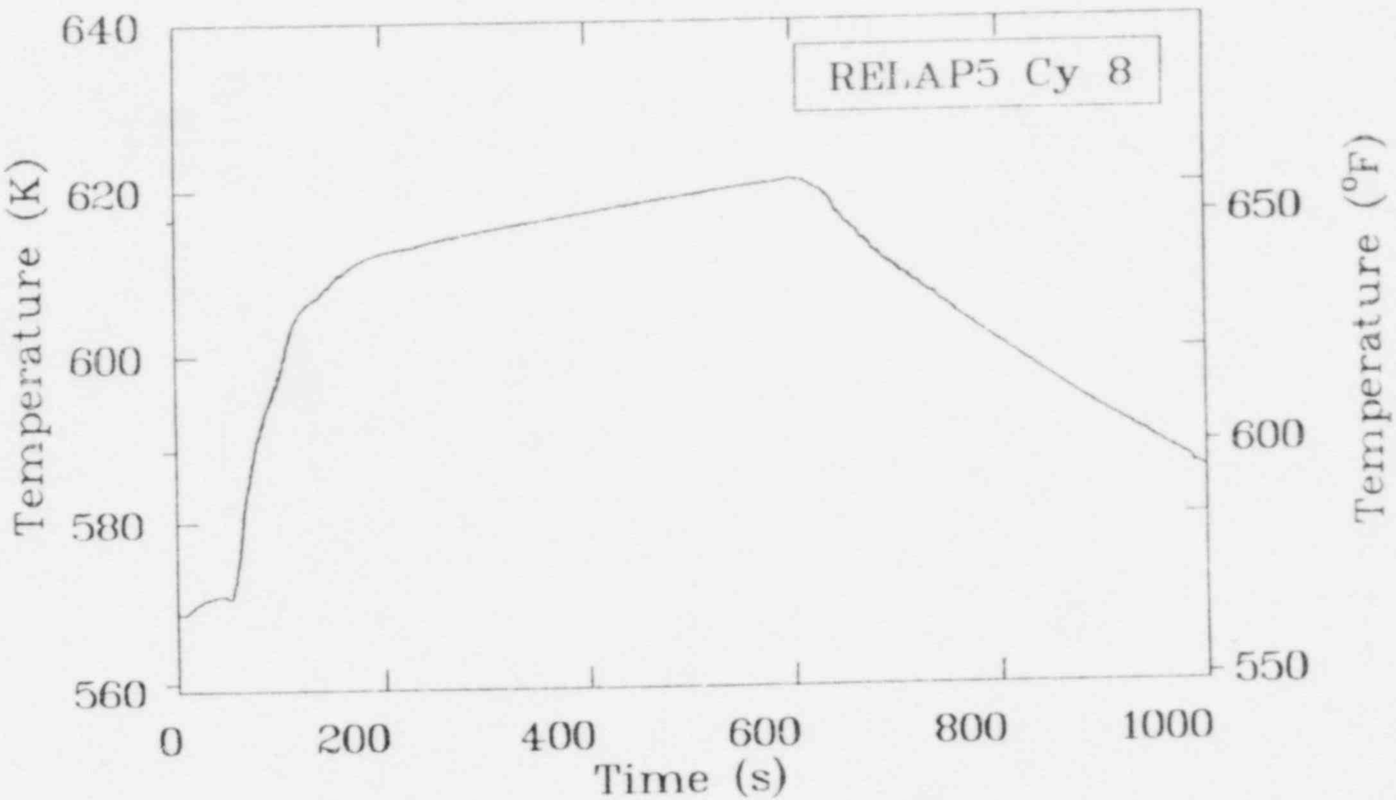

Figure 6 – Average Primary Temperature
L9-3 ATWS – Loss of Feedwater

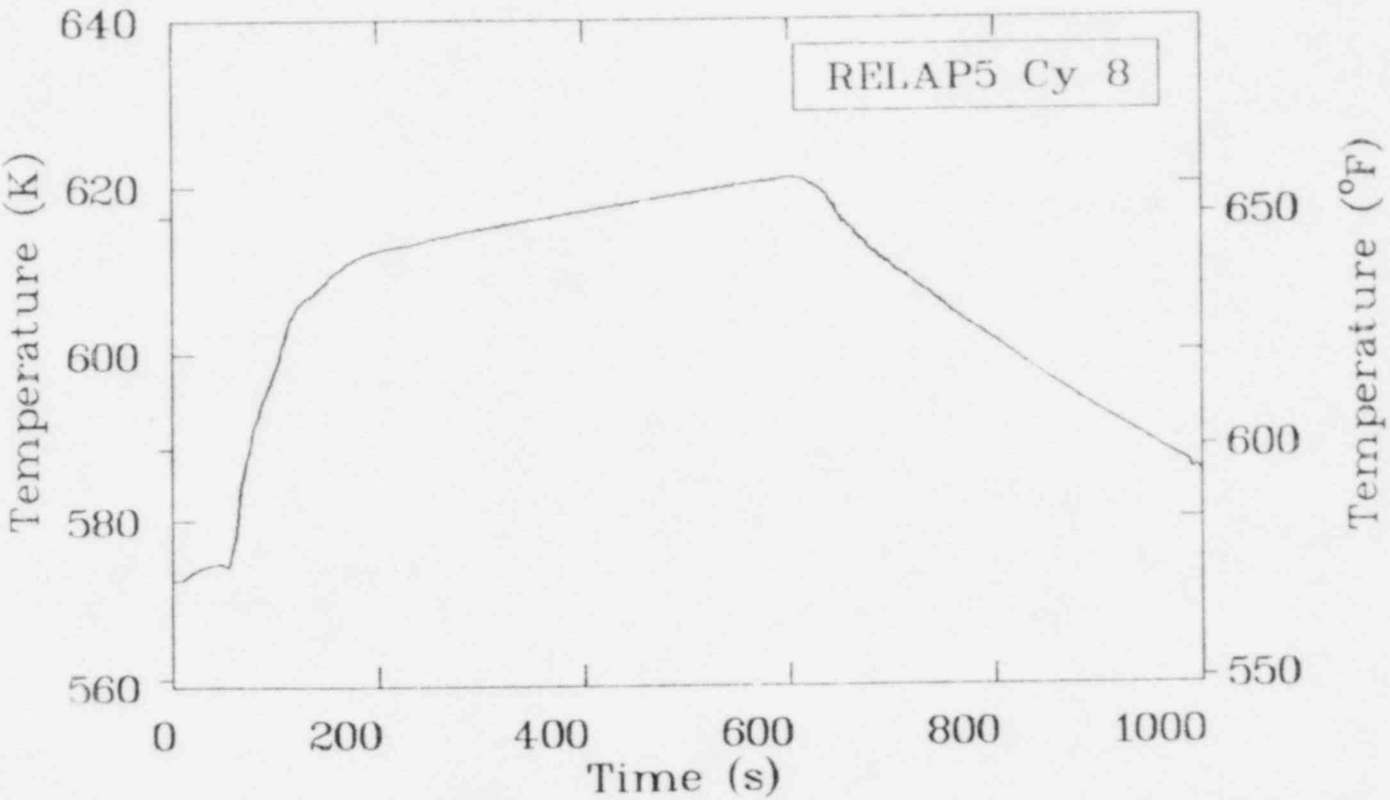

Figure 7 – Average Core Temperature
L9–3 ATWS – Loss of Feedwater

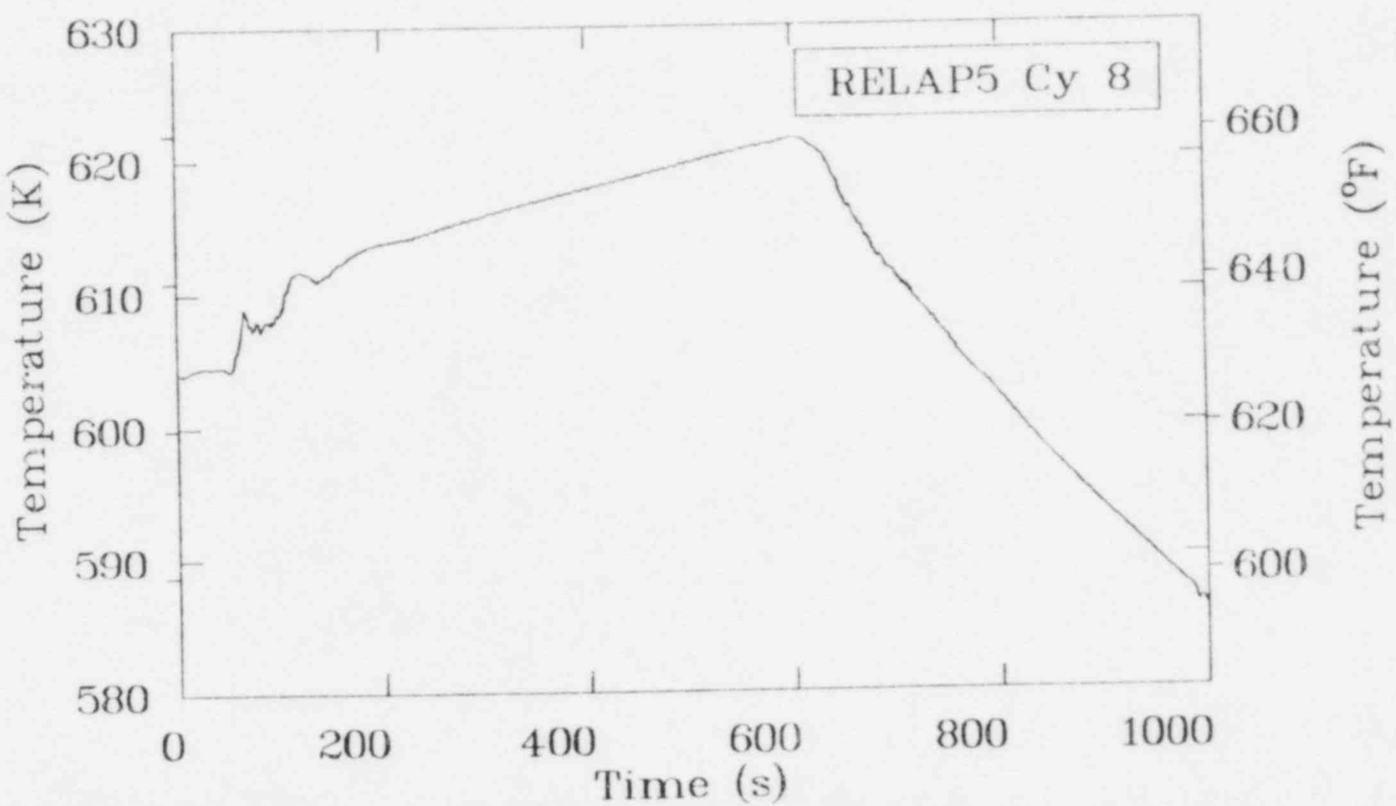

Figure 8 – Lower Clad Temperature
L9–3 ATWS – Loss of Feedwater

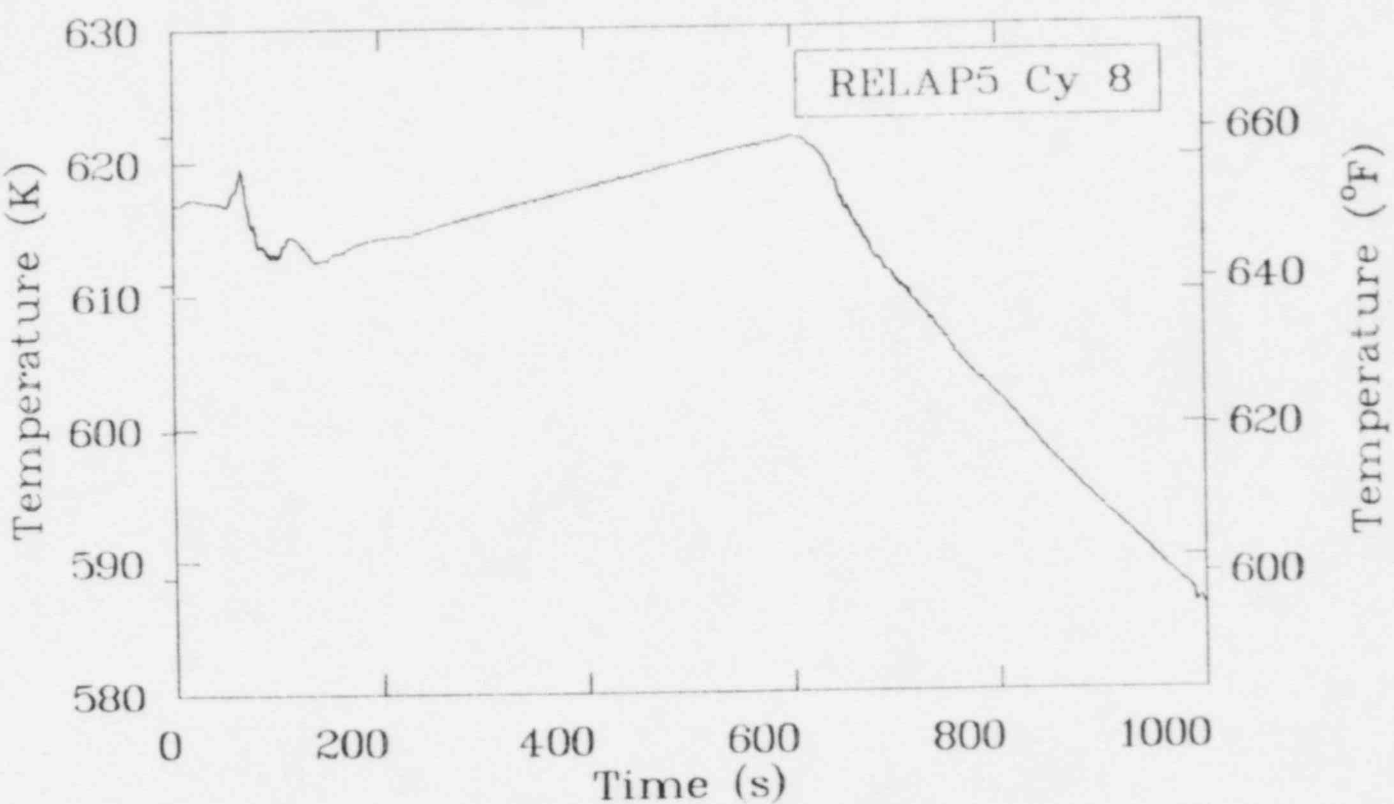

Figure 9 – Middle Clad Temperature
L9–3 ATWS – Loss of Feedwater

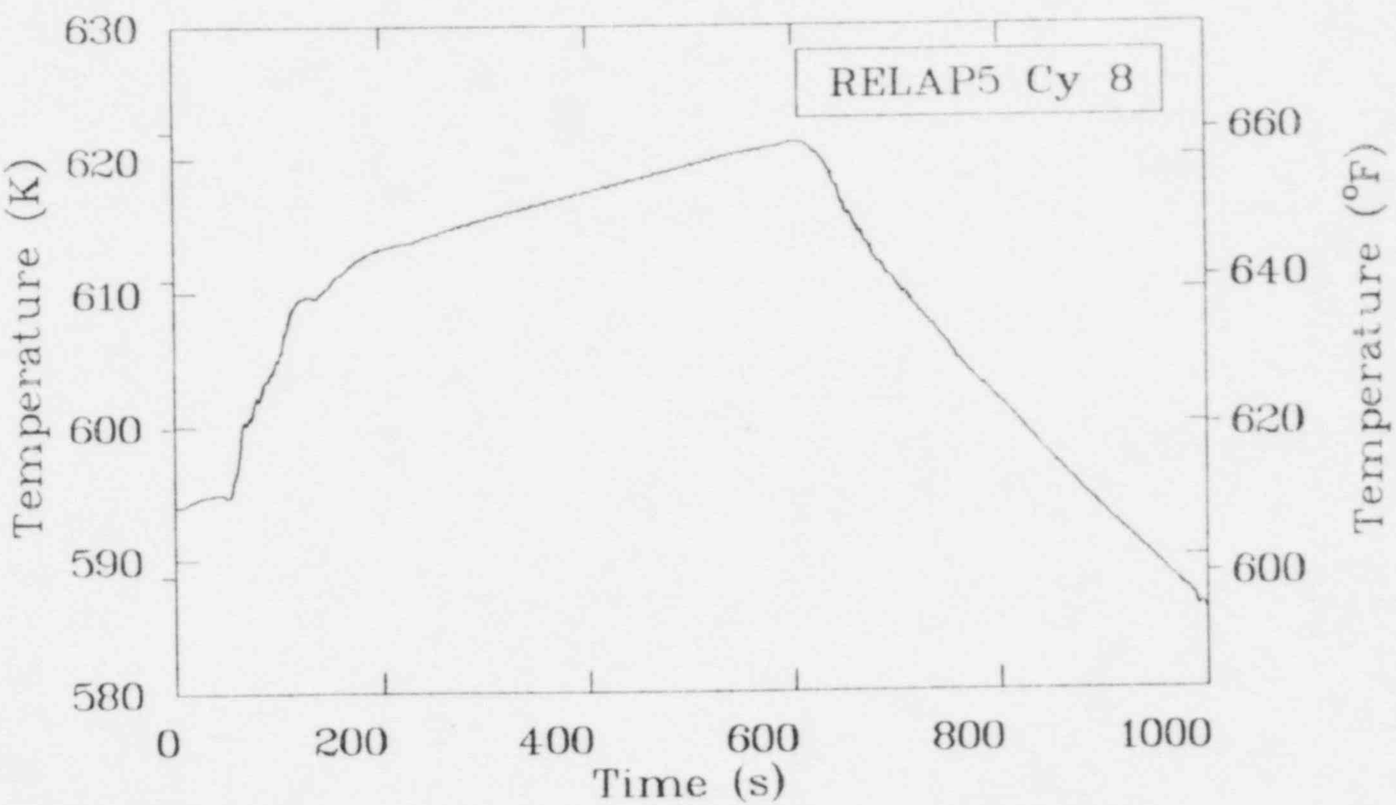

Figure 10 – Upper Clad Temperature
L9-3 ATWS – Loss of Feedwater

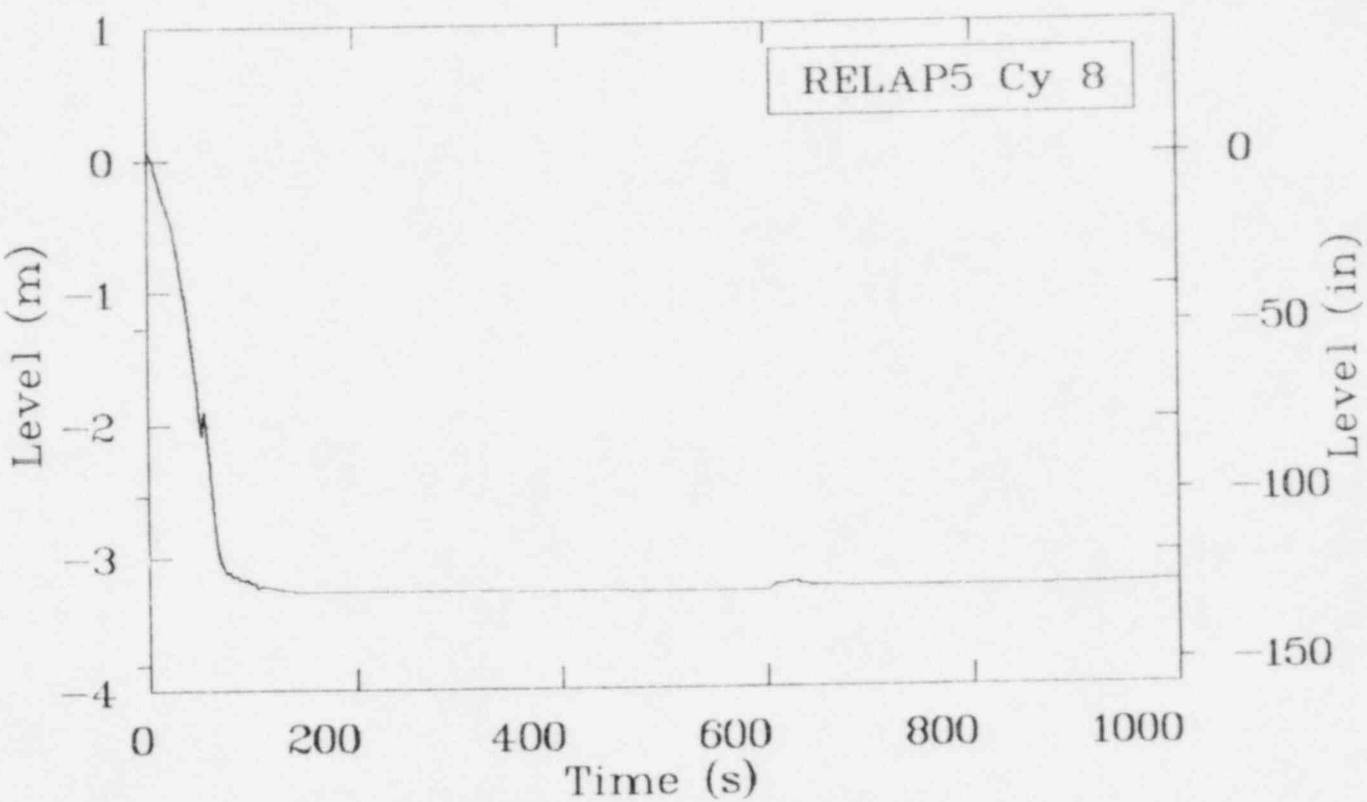

Figure 11 – Steam Generator Level
L9–3 ATWS – Loss of Feedwater

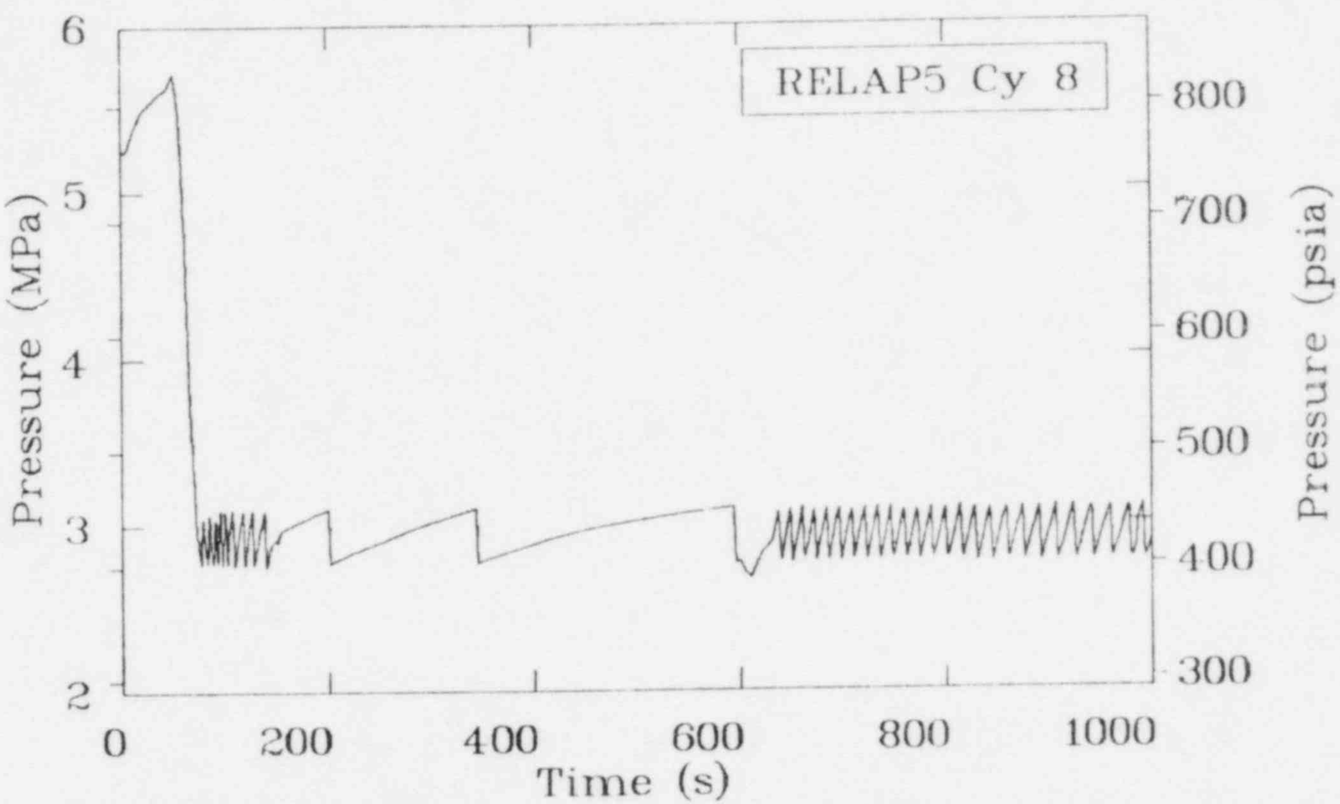

Figure 12 – Steam Generator Pressure
L9–3 ATWS – Loss of Feedwater

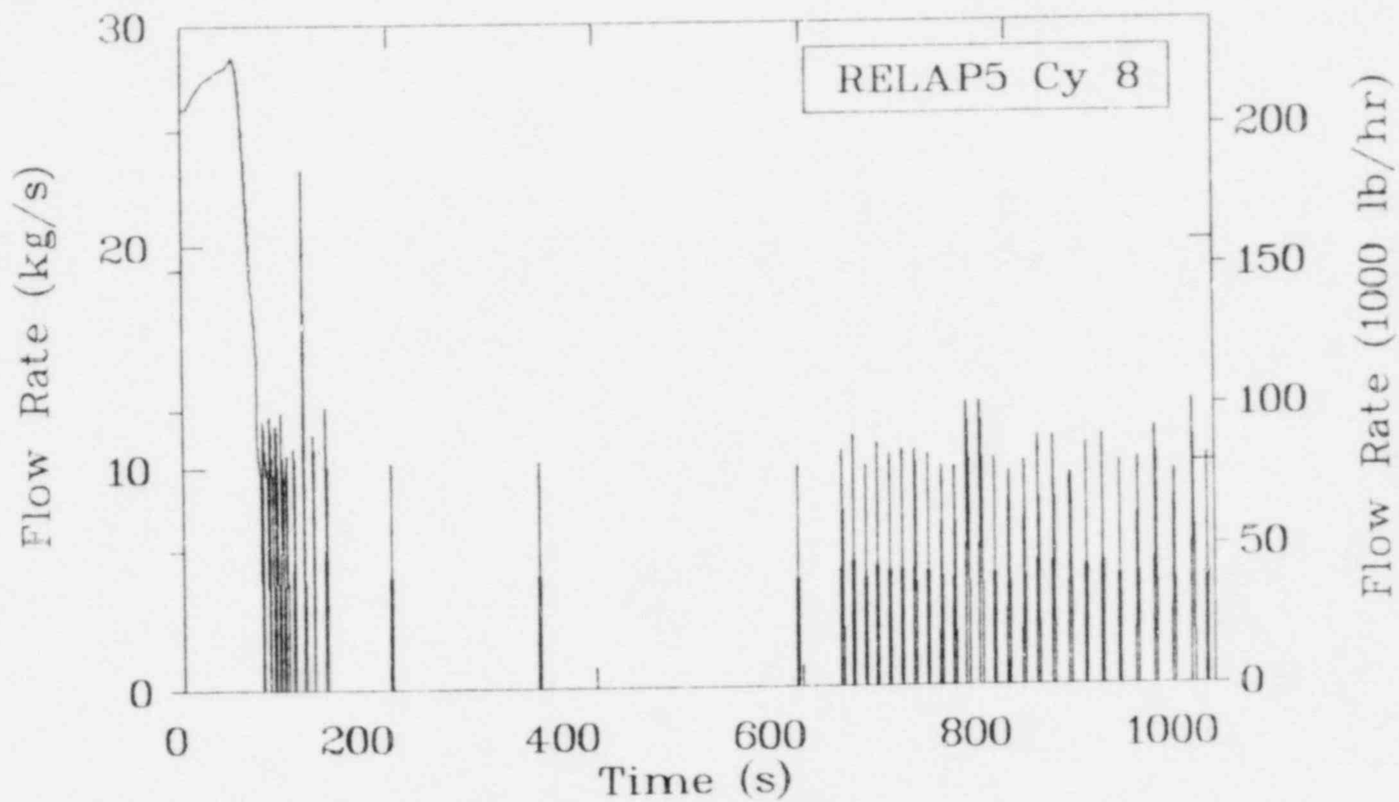

Figure 13 – Steam Generator Steam Flow
L9–3 ATWS – Loss of Feedwater

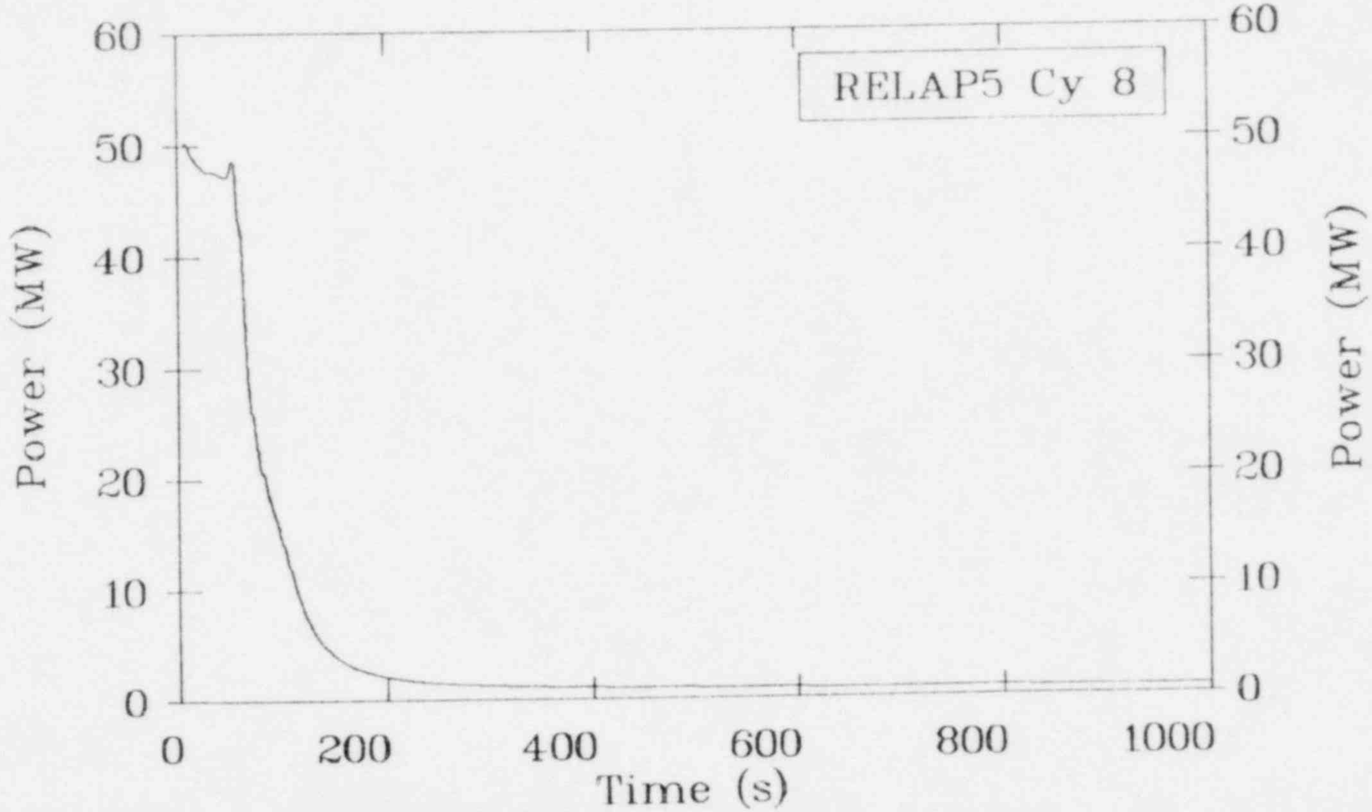

Figure 14 – Total Reactor Power
L9-3 ATWS – Loss of Feedwater

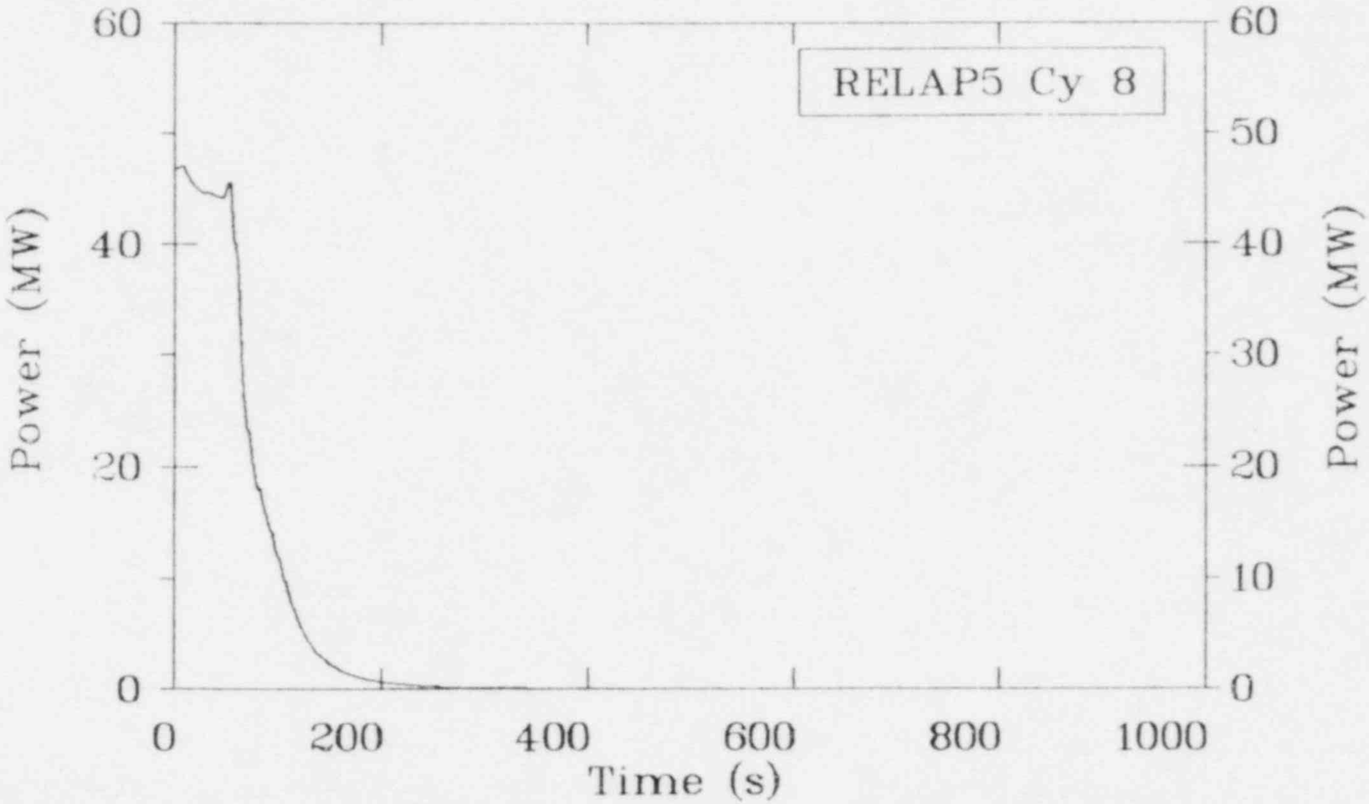

Figure 15 - Reactor Fission Power
L9-3 ATWS - Loss of Feedwater

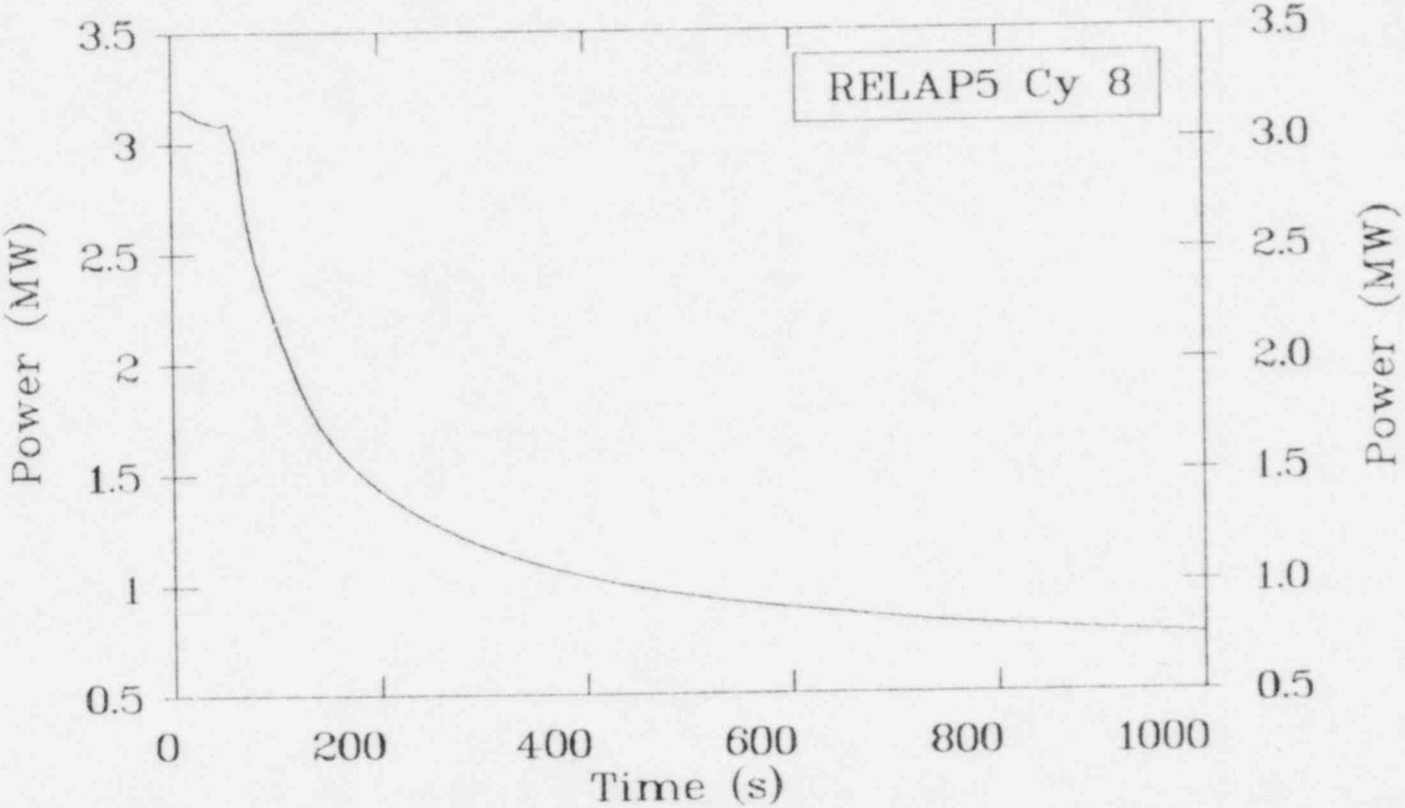

Figure 16 – Reactor Decay Power
L9–3 ATWS – Loss of Feedwater

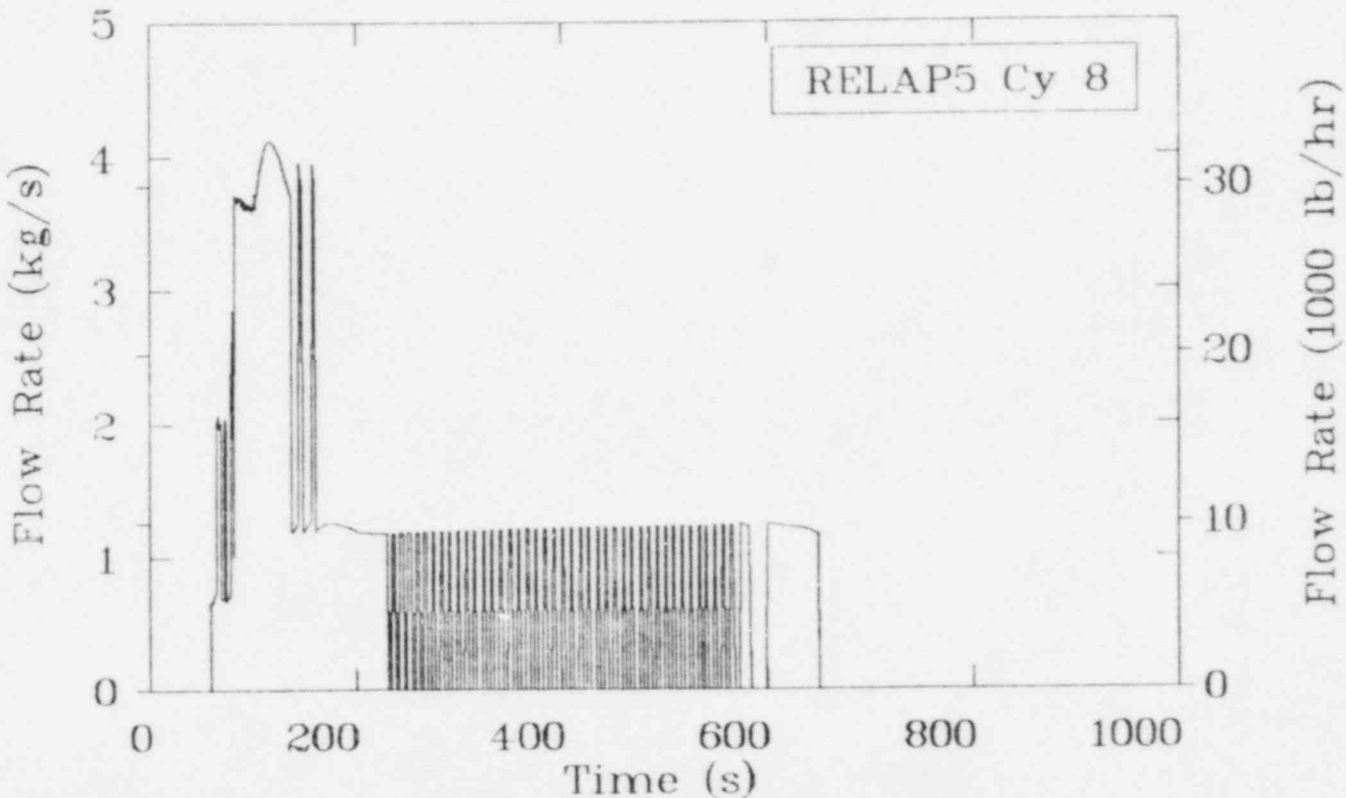

Figure 17 – Pressurizer Relief Flow
L9–3 ATWS – Loss of Feedwater

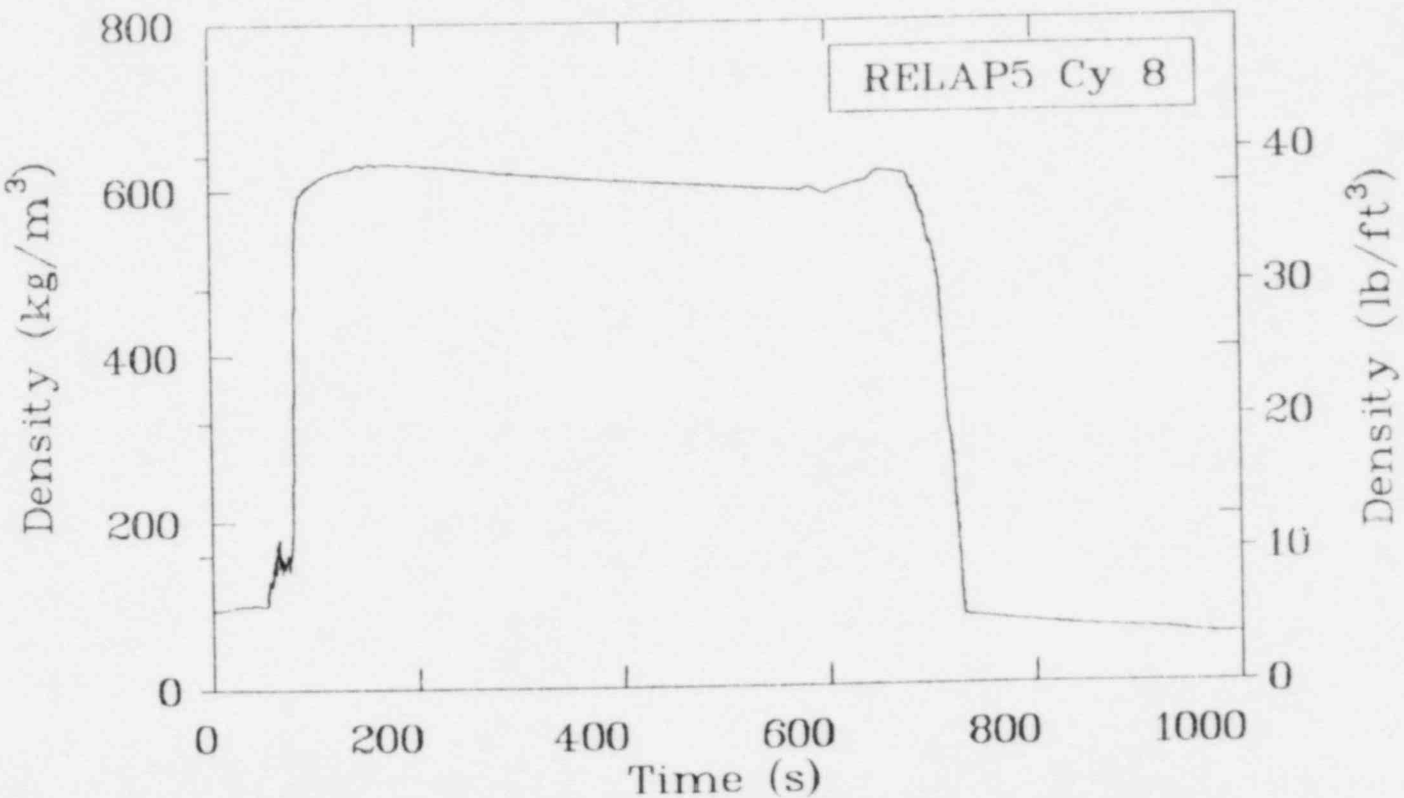

Figure 18 – Upper Pressurizer Density
L9–3 ATWS – Loss of Feedwater