ATWS DISTRIBUTION LIST

F Cherny
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NOTE TO: Distribution

As per 3/9 ATWS meeting, the attachment provides Minners, Easterling's, and Thadani's comments.

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Ashok C. Thadani

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R. Baer	T. Novak
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D. Bunch	H. Ornstein
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H. Krug	R. Vollmer
S. Levine	I. Wall
W. Minners	S. Weiss
R. Minogue	G. Wrobel

Comments by Frank Cherny on ATWS vs. Use of Faulted Stress Limit are attached at the end of this report.

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I. SAFETY GOALS

a. WASH-1270

10^{-3}/yr nationwide

Basis? arbitrary

1000 Reactors

May be fewer

10^{-6}/R yr all accidents

Old vs. new plants

local variations

10^{-7}/R yr for ATWS

1/10 assigned to ATWS

Basis? Arbitrary? Conservative?

Necessity

Regulate to the practical, possible or necessary?

Social question

Not limit but "goal"

Seems to have been successful

Demonstrable

Not demonstrated, but "aiming point"

Requires assumptions of independence which is difficult to prove

Same assumption as RSS

RSS has many low probabilistics which are justified as being products of several higher but more certain probabilities

b. WASH-1400

Core melt more probable (10^{-5})

Conservative because core melt criteria conservative

Uncertain whether all significant events included

Consequence less severe (10^{-1})

Uncertain consequence models

Risk appears low compared to other risks

Qualified by latent cancers and generic effects

Neglects site features (population, meteorology)

ATWS Risk

Large Contributor

50% BWR

20% PWR

Underestimated in RSS

BWR

Containment failure underestimated

Short term: Boiling instability

Vibrations, SLCS rate too small

Long term: Less certain Boiling Instability Problem

Probability 10^{-4} rather than 10^{-5}

PWR

Isolation valve operability neglected

For W

Q: S/V fail to reclose insignificant

Fail of turbine trip neglected

For B/W, CE

Pressures higher

Acceptance Criteria 3

Radiological

A) Part 100

Siting criteria

Doses used in conjunction with conservative model and 95% site meteorology, not to be combined with realistic models

1. Risk

Sensitive to site

Complicated

No accepted limit

2. Core Melt

More complicated

How much permissible

B) RCS Integrity

Central issue

Emergency Stresses

Within yield

RPV not limiting component

Faulted Stresses

Uncertain material properties

Complicated analyses

Still limited by equipment operability

C) Fuel Integrity

Intent: core coolability

Not central issue

Pressure: ill defined

Above RCS pressure and therefore not limiting

Enthalpy: not appropriate for slow transient

CHF: easily calculated; conservative

PCI neglected in WASH-1270 but discussed in Status Reports

D) Containment

Pressure: not a problem

Isolation: omitted

Steam quenching instability: unrecognized; leak tightness may not be necessary; Water source necessary

II. Arrival Rate of Anticipated Transients

WASH-1270: The annual rate of a anticipated transient of significance for ATWS is believed to be in the range of 0.1 to 0.5, and it appears prudent to assign unit annual probability for such occurrences.

WASH-1400: WASH-1400 suggests that based on US LWR experience the annual likelihood of anticipated transients is between five and twenty with a median value of ten.

However, likelihood of transients of significance for ATWS in PWRs is judged to be about three per year.

EPRI: (Vol. 4) Basis. Experience Data up to 1975

BWR: Transient Frequency Mean 9.33/yr.
PWR: " " " 10/yr.
PWR loss of feedwater transient frequency mean 2.4/yr

If data trend extended to forty years

BWR: Transient Frequency Mean 4.19/yr.
PWR: " " " 5.2/yr.
PWR loss of feedwater transient mean 1.16/yr.

Status Reports: Assumed transient frequency of once per year.

Conclusion: Actual transient frequency may vary from once per year to ten per year, and thus the status report assumption may have an error factor as high as ten in the non-conservative direction. Consideration of forty year plant lifetime and extension of experience trends would realistically suggest perhaps a factor of five error in the non-conservative direction.

III. Probability Estimation for Scram Systems

A. Data

1. Systems Data

EPRI gives the following summary of operating experience through 1975:

Units	Accumulated Reactor Years	Total Scram Demands* (Estimated)
U.S. Comm. Power	228	
Army	57	Total = 39,212
N.S. Savannah	10	
Foreign Comm. Power	673	
Navy	1252	75,120
Total	2220†	114,332

*Includes tests and partial tests.

†WASH-1270 used 1627 Reactor Years.

Two known failures have occurred, both early in reactor development, and there is some question of whether one or both should be "counted" in drawing inferences from these data. There are also questions as to whether all the estimated scram demands test the system to the same extent that an anticipated transient would and whether it's reasonable to assume no navy failures.

2. Component Data

Attachment B, page 10, lists BWR rod failures identified by EPRI from the Nuclear Safety Information Center. Among BWR's there have been six incidents of less than full insertion of one or more rods. The number of affected rods ranged from 1 to 96. Three of these rods failed to insert at least to position 02 and were counted by EPRI as failures.

For PWR's, EPRI considers only failures at newer reactors which have a different control rod design from older reactors. For the new design, there have been 2 failure incidents, each involving one rod failure.

It appears that the only failure data included in the EPRI summaries are mechanical failures. Calibration errors and failures of the electronic components of scram systems are not included. This is an area where more information is needed.

3. System and component failures have happened. Thus, there is more to work with than the no-alligators-in-the control-room situation.

B. Methods

1. There are two approaches which have been used to obtain estimates of the probability of scram system failure. One is based on system data alone, such as in n reactor-years of experience there have been f system failures, or in d system demands there have been f failures. This is the approach of WASH-1270. The other approach is to separate the system probability into its components, such as hardware, test and maintenance, and human error; estimate the probabilities associated with each component; and then add. This is the approach of WASH-1400 and followed by GE in their ATWS analyses. EPRI takes both approaches, then merges the results.

2. Some other differences: The assumption in WASH-1270 is that scram system unreliability, call it Q_{WS}, is constant across time and across reactors. In WASH-1400 it is assumed, I believe, that Q_{WS} varies among reactors, but not over time, according to a specified distribution. EPRI goes one step further and assumes that rod failure probability varies from rod to rod in one reactor at one time. GE, I believe, regards Q_{WS} as a constant, but uses WASH-1400 methods anyhow. All this makes comparison of results difficult. For example, the upper 95% confidence limit on Q_{WS}, by WASH-1270 methods, is a bound on the unknown, but industry-wide, value of Q_{WS}. If it is assumed that Q_{WS} varies among reactors, but not very much, then the WASH-1270 95% confidence limit can be regarded as an approximate 95% confidence limit on the average Q_{WS}. The upper limits of WASH-1400, in contrast, are to be interpreted as bounds for 95% of the reactor population. That is, in one case we have a statistical bound on a population average; in the other, a probabilistic bound on population individuals. The numerical value of these bounds should not be expected to be comparable nor should coincidence of the bounds be taken as confirmation or support of one analysis for the other.

3. While the system synthesis approach of WASH-1400, EPRI, and GE have the potential for providing more precise estimates than the WASH-1270 approach, that potential is not realized because of flaws in the methodology which make the results highly questionable. Attachments A and B describe in detail the quantitative errors which result from the "square root" method of probability estimation used in these three analyses.

4. The errors the square root approach leads to are that failure probabilities are understated, relative to the assumptions on which they are based. It may be that the assumptions in these analyses are quite conservative and that the analysis method just offsets this conservatism, but this cannot be relied on. Before we can put any trust in numerical results, we must be able to trust the methods used to obtain those results. Incorrect probability equations are as much of a problem or danger as incorrect heat transfer or fluid dynamic equations.

5. The issue is not one of approach - Bayesian vs non-Bayesian vs empirical Bayesian, or statistical vs risk analytic, etc. - but one of mathematical correctness. Given a problem, given some assumptions, how is the answer to be derived? The mathematical rules of probability provide the answer. Ad hoc approaches which violate those rules do not.

C. <u>Probability Estimates</u>

1. <u>From System Data</u>
Under the not necessarily conservative assumptions that the system failure rate has been constant across the accumulated 2220-reactor years and continues to be, and that 1 failure has occurred, the maximum likelihood estimate of the system failure rate, θ_{WS}, is $\hat{\theta}_{WS} = 1/2220 = .00045$. Upper confidence limits on θ at the 75, 95, and 99% levels, respectively, are .001, .002, and .003. Corresponding lower confidence limits are 1×10^{-4}, 2×10^{-5}, and 5×10^{-6}. Under the nonconservative assumption that the time between transients is exponentially distributed with rate μ and the assumption of monthly testing which would detect and correct failures, ATWS probability is approximately

(for small $\mu\theta$) equal to $\mu\theta/24$. The following table gives upper and lower confidence limits on ATWS probability, given $\mu = 10$.

ATWS Probability

Confidence Level	Lower Limit	Upper Limit
75%	4×10^{-5}	4×10^{-4}
95%	8×10^{-6}	8×10^{-4}
99%	2×10^{-6}	1×10^{-3}

If only central power stations are considered, all these estimates would double. If no failures are assumed, the upper 75, 95, and 99% confidence limits on ATWS probability become, 3×10^{-4}, 6×10^{-4}, and 8×10^{-4}, respectively, little different from the upper limits assuming 1 failure, and the lower limits all equal zero.

An alternative to assuming constant system failure rate is to assume a constant probability of failure on demand. Under the nonconservative assumption that there have been 114,332 independent demands, all of which had a probability of failure, P_{WS}, and only one failure, the maximum likelihood estimate of P_{WS} is $\hat{P}_{WS} = 1/114,332 \approx 10^{-5}$. Under the same assumption as above concerning the arrival rate of transients, ATWS probability equals $1 - \exp(-10 P_{WS})$. From confidence limits on P_{WS}, the following limits are obtained on ATWS probability:

ATWS Probability

Confidence Level	Lower Limit	Upper Limit
75%	3×10^{-5}	2×10^{-4}
95%	5×10^{-6}	4×10^{-4}
99%	9×10^{-7}	6×10^{-4}

If no failures are assumed, the upper limits become 1×10^{-4}, 3×10^{-4}, and 4×10^{-4}. Note that the estimates obtained in the constant failure probability case are about 1/2 times those in the constant failure rate case and the assumption of monthly testing. Test frequency plays no direct role in the constant failure probability case.

2. <u>From Component Data</u>
Attachment B provides estimates of multiple control rod failures, conditional on successful operation of the reactor protection logic. The estimates are based on simple models, chosen to fit the EPRI data. The models are not particularly conservative because they entail rather smooth, constrained variability.

To estimate the conditional system failure probability from the results in Attachment B, a definition of system failure is needed. For BWR's, EPRI and WASH-1400 define it as failure of three adjacent rods. GE says 5 adjacent rods. To estimate the probability of these events, the probability of adjacency, given that x rods fail, is needed. One approach is to assume all sets of x are equally likely and then count how many of those yield 3 or 5 adjacent rods. EPRI and GE take this approach. As a simpler, but no more arbitrary (in the absence of any data pertaining to the probability of adjacency) approach one might consider a function of the form,

$$\text{Prob(a adjacent failures} \mid \text{x failures)} = 1-\delta^{b(x-a)+1}, \quad x=a,\ a+1,\ \ldots$$
$$= 0 \quad , \quad x=0,\ 1,\ \ldots,\ a-1.$$

For the models in Attachment B, this leads to

$$\text{Prob(a adjacent failures} \mid \text{scram attempt)} = \text{Prob}(x \geq a)\left[1-\frac{(1-\rho)}{1-\rho\delta^{b}}\right],$$

where ρ is the bracketed term on page 12 of Attachment B, namely $\rho = p\Theta/[1-(1-p)\Theta]$.

Consider the case of a = 5. GE estimates that the conditional probability of adjacency, given 5 failures in a 177 rod core as 10^{-5}. Thus, to coincide with this, one obtains $\delta = 1 - 10^{-5}$. To choose b, we use the GE results that system failure occurs conservatively with probability of .05 when 55 rods fail. Equating .05 to the probability of 5 adjacent failures, given 55 failures, leads to b = 100. Using the maximum likelihood estimates of r, Θ, and p in Attachment B thus leads to

$$\widehat{\text{Prob}}(\text{system failure} \mid \text{scram attempt}) = 2 \times 10^{-7},$$

Using the upper 95% confidence limits on r, Θ, and p leads to

$$\text{Prob}_{95}(\text{system failure} \mid \text{scram attempt}) = 8 \times 10^{-6}.$$

These results are highly conjectural and would require considerably more study. For example, the fact that there may be some previously failed rods present has not been accounted for. However, these results are no more conjectural than previous analyses and are at least consistent with available data and with the rules of probability.

For PWR's, failure is defined by EPRI as failure of 3 or more rods. An upper, approximate 99% confidence limit on the probability of this event is, from Table 3 of Attachment B, 1.5×10^{-3}. Thus, the PWR data, which are much more limited than the BWR data, and the less stringent system failure definition, lead to a considerably higher bound than that of BWR's.

D. Conclusion

There is too little information available right now to make a reasonably precise assessment of system failure probability from available component data. This leaves only the systems data on which to base a quantitative assessment. Available data do not resolve the question of whether an ATWS fix should be electrical, mechanical, or both.

Attachments: (Available on request)

Attachment A - Failure Probability Calculations for GE ATWS Analysis

Attachment B - Estimation of Control Rod Failure Probabilities

IV. Equipment Reliability

Low probability single failures not required; however, if the mitigating system unreliability was significantly greater than 10^{-3} per demand, analyses assuming these failures were required. The data source essentially was WASH-1400 median values. For estimating population risks, selection of median values as compared to mean values may be nonconservative.

V. Assumptions and Input to Evaluation Model

Proposed ANS Industry Standard N661 on Anticipated Transients Without Trip on PWR Plants recommends the following on plant conditions and assumptions for evaluation of ATWS events.

The value used for each condition and assumption shall be selected by one of the following methods.

1. Selection of a conservative value as specified or defined by either the design basis FSAR analysis, or the technical specification limit.

2. Selection of the design operational value allowance for control band, but excluding any allowance for measurement uncertainty, for variables regulated either by automatic control systems or manually under administrative control.

3. Selection of either the measured or design value excluding any allowance for design margin or measurement uncertainty.

4. Selection of a calculated value not expected to be exceeded (that is, more adverse) during the preponderance (at least 95%) of plant lifetime. Justification of this probability argument need not consider allowance for calculational uncertainty or for random statistical fluctuations.

In general, vendors used values consistent with 2 and 3 above except for MTC where 4 above was used as basis.

The present staff ATWS model input requirements are consistent with the standard except in the following areas.

a) Instead of 95% moderator temperature coefficient (MTC) use 99% value. Transient frequency is high when MTC is high. See note from A.Thadani to S.Hanauer dated 3/8/77 for more details on this requirement.

b) Purging in Progress during ATWS event. This requirement was imposed because of the staff belief that while some plants have frequent purging, others may undergo continuous purging. In any case, this requirement does not appear to require any design modification.

c) Ten percent primary safety valve accumulation to open for water discharge. Although the standard implies use of three percent accumulation, discussions with valve manufacturers indicate the ten percent value to be more realistic. Vendors seem to agree with us after discussing the problem with valve vendors.

On the other hand, the staff has not required inclusion of:

i) over ½ GPM Steam Generator leakage

ii) coincident loss of offsite power

iii) uncertainty in operating parameters

iv) any conservatism beyond 0.9* Homogeneous Equilibrium Model for primary system water relief through relief and safety valves (data lacking but the staff judges the model to provide low estimates on water relief)

v) no operator error

vi) no operator action in the first ten minutes of an ATWS event

vii) Seismic Events

VI. Effect of ATWS fix on non-ATWS Accidents

BWR:

WASH-1400 Significant Core Melt Sequences

Category 3

$TW-\gamma$	1×10^{-5}
$TC-\gamma$	1×10^{-5}

Category 2

$TW-\gamma'$	3×10^{-6}

Category 1

$TW-\alpha$	2×10^{-7}
$TC-\alpha$	1×10^{-7}

$T \equiv$ Transient Event
$C \equiv$ Failure to Shutdown Reactor
$W \equiv$ Failure to Remove Residual Core Heat
$\gamma \equiv$ Containment Failure due to Overpressure-Release through Reactor Building
$\gamma' \equiv$ Containment Failure due to Overpressure-Release direct to atmosphere
$\alpha \equiv$ Containment Failure due to Steam Explosion in the Vessel

WASH-1400 suggests that two types of accidents i.e. TC (ATWS) and TW (failure to remove decay heat) are major contributors to core melt.

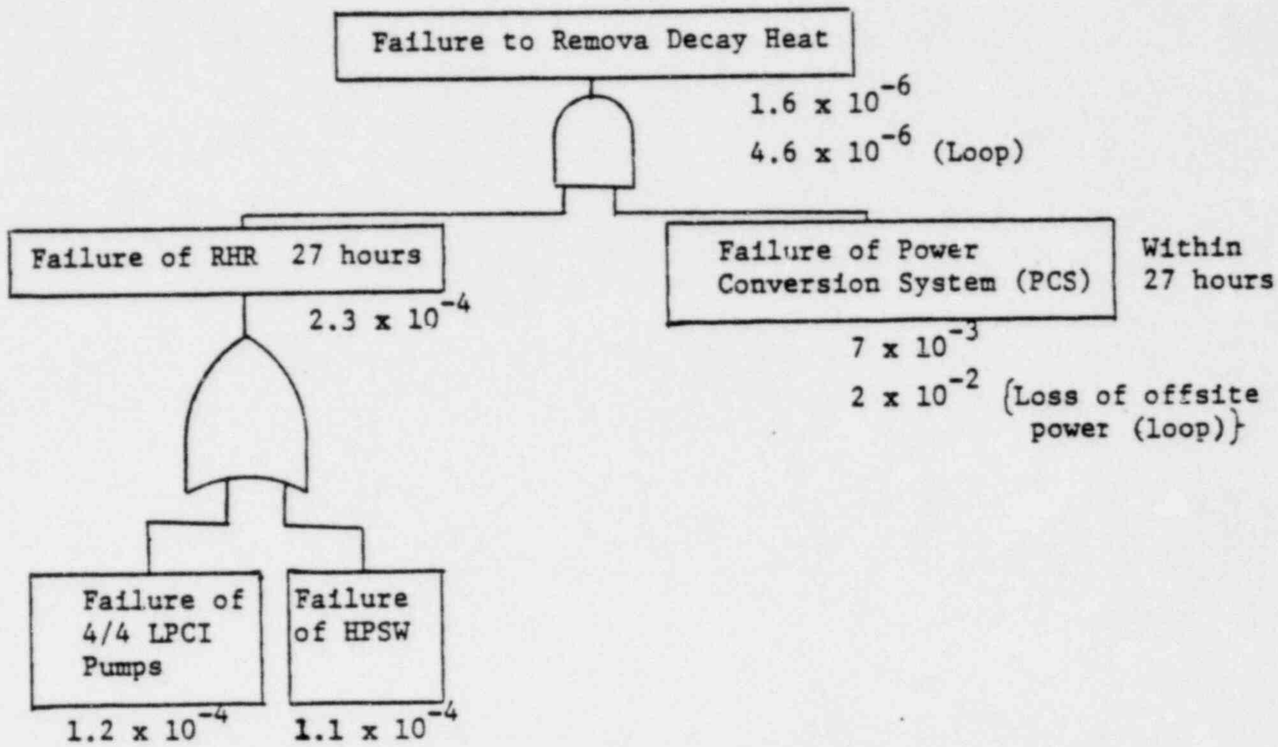
As discussed elsewhere (note from A. Thadani to S. Hanauer dated April 8, 1977).

The probability of unacceptable consequences due to ATWS is 10^{-4} per reactor year and one of the indicated design 'fix' was to add a high pressure special ATWS make up system (SAMS).

The failure probability of the decay heat removal system (W) is dominated by the failure probability of power conversion system to perform the function of transferring fission product decay heat to the environment. A careful consideration of this failure mode in the design of SAMS is expected to result in lower probability of core melt from TW sequences.

BWR

RESIDUAL HEAT REMOVAL SYSTEM



Note the probability of Core Melt due to Non-Loss of Offsite event $\sim 10 \times 1.6 \times 10^{-6} \sim 10^{-5}$ dominating over loop event. Success of Power Conversion System (PCS) depends on ability to

a) Operate one complete condensate feedwater piping i.e. condensate and feedwater pump.

b) Open one Isolation Valve and open bypass valve.

c) One condenser recirculating pump operable.

It is not clear in WASH-1400 what the individual contributions to PCS unavailability are but the contribution of a above would be expected to be reduced by availability of SAMS.

PWR:

WASH-1400 Significant Core Melt Sequences are:

1. Small LOCA followed by failure of containment and core cooling systems.

2. ATWS

3. Check Valve - Interfacing System LOCA

4. Transient followed by failure of main feed and auxiliary feedwater system.

Comments:

The Interfacing System LOCA probability has been reduced by recent NRC requirements.

It is not clear why transient followed by failure of main feed and auxiliary feed is assumed to result in core melt. In any case ATWS fix may help reduce this probability by cooling the reactor using high pressure injection system and pressurizer relief valves.

ATWS probability resulting in exceeding criteria is believed to be higher than the WASH-1400 estimate. Thus, if ATWS fix is provided the core melt probability may be controlled by small LOCA ($10^{-5} \sim 10^{-6}$).

Preliminary Considerations -

ATWS vs. Use of Faulted Stress Limit

The Faulted Limit permits primary membrane (average stress across vessel wall) stress levels considerably in excess of the yield strength of the material.

A. Reactor Vessel

Discontinuity regions of the vessel could be expected to deform plastically, i.e. they would not return to their original shape after load relaxation. Such regions would include intersections of primary coolant nozzles to vessel shell, flanges to shell, and head dome to vessel shell or to closure head flange. Regarding the latter item, the extent of the resulting permanent deformation could conceivably be great enough to prevent manual insertion of at least some of the control rods. The effect of vessel distortion on the position of the control rod blade passages relative to their design location would also have to be considered.

Another major consideration to be evaluated would be the behavior of the bolted closure head to vessel flange joint. At such high pressures leakage would probably be severe, i.e. the bolts are only torqued for normal operating pressure. What would be the effect of severe coolant leakage on such things as local fuel rod overheating etc.? Regarding the behavior of the vessel to head juncture and possible head distortion MEB currently has a contract with INEL to evaluate the effects of a pressure of 3750 psi on these areas of a B & W 171 FA reactor vessel.

In light of the ongoing ATWS re-evaluation, perhaps we should have INEL evaluate the effects of 4500-5000 psi instead. Structural integrity of instrumentation tube and control rod drive housing partial penetration welds would have to be evaluated. Section III of the ASME code only allows the use of such welds when the penetration nozzle is subjected to essentially zero piping reactions. Pressures which result in faulted limit stresses and consequent deformations of both vessel heads some of these welds may fail - consequences must be evaluated. Instrumentation probes and any control rods in core could be ejected.

B. Pumps and Valves

Active Valves

Valves of this type which have to function after the ATWS so that ECCS, CVCS etc. systems can be brought into play to shut down the plant would experience large permanent deformations. Their capability to function after exposure to such stress and strain levels could probably only be convincingly verified by test. Additionally, the comments below for inactive pumps and valves would apply.

Inactive Pumps and Valves

From a structural integrity point of view, these components, for the most part, can be likened to small stainless steel pressure vessels.

Permanent deformations in regions of discontinuity would be severe. The extent would have to be evaluated on a component by component basis.

Behavior of large bolted bonnets of these components, again, as for reactor vessel only torqued for 2500 psi, is open to question and would

have to be evaluated.

C. Pressurizers

Permanent deformations and their effects would have to be evaluated.

Areas of concern - bolted manway covers and pressurizer heater to bottom head welds (burst pressure of these welds could be exceeded).

D. Steam Generators

Effects of pressure per se on tubes would probably not be a problem. However, again, parts of this vessel would deform severely. The effects of such deformations on "tender" spots such as tube to tube sheet weld integrity would be difficult to evaluate with any degree of confidence.

Also bolted manway covers could be a problem.

E. Safety and Relief Valves

Valves could be expected to open and relieve because this occurs, for ATWS transients, at normal valve set pressure. Question here would be to what degree would they reclose after experiencing some deformation. From discussions with safety valve manufacturers, it would appear that from basic structural integrity point of view, the valves should be o.k.

F. Piping

Exposure to pressures which result in the stresses at the faulted limit would not be expected to result in any breach of piping from this effect alone. However, plastic deformation of other components

in the system, together with that in the piping would result in increase of nonpressure mechanical loads in various locations in the piping, such as at intersections with other large components, elbows etc.

The effects of the high pressure in combination with the other resulting mechanical loads on the dimensional stability of the piping would have to be evaluated. For instance, even though the pressure boundary of the piping may not be violated, dimensional distortion of piping at a critical location could impair flow of fluid from ECCS, CVCS etc. alternate shutdown systems.

General Remarks

The use of the Faulted limit allows for considerable inelastic deformation of the material. Permitting the use of this limit for pressure loading means that for the first time NRC would permit the entire reactor coolant pressure boundary to deform plastically for a postulated event. Thus far this stress limit has only been applied for relatively concentrated high load situations i.e. LOCA, LOCA plus SSE and always for events where the control rods are in. Although we say that we will accept the Faulted Limit for SSE alone; in general, the majority of components under just the SSE environment are not exposed to stress levels anywhere near the Faulted limit

In order to fully evaluate the consequences of going to the Faulted limit on pressure loading, all components would have to be analyzed inelastically. Then a complete system inelastic analysis would have to be performed to

determine the effects of the component deformations on the piping. NRC would have to review all such analyses. From the vendor's point of view, going this route would be expensive and would not necessarily result in an NRC approval that such high pressures would have no adverse impact on the public.

Also in performing these types of analyses, it is not always possible to accurately define the actual magnitude of the loads to be input to the analyses. Thus it is not clear at this time that with all of the unknowns and uncertainties involved in going to such a high stress limit, on such a gross basis, that safe shutdown of the plant could be demonstrated within the level of confidence NRC would require.

ITEM 4 - MAY 1 MCCRELESS MEMO

We interpret this question as asking the following:

"What is the probability of an ATWS-induced LOCA provided the staff fixes are implemented?"

More precisely, addressing the staff criterion recommended in NUREG-0460, which is most relevant to the probability of a LOCA, i.e., the recommendation to permit RCPB stress levels to reach ASME Code Service Limit C, we further characterize this ACRS question as "What is the probability of an ATWS-induced LOCA, providing the RCPB Service Limit C stress criterion is implemented?".

In response, we reply that we have not quantified this probability. However, based on engineering judgment we believe that considering the relatively low probability of ATWS as stated in the report, combined with an admittedly unquantified but, in our judgment, very low probability of RCPB failure utilizing the proposed Service Limit C criterion, that the overall probability of such a failure is on the order of 10^{-7} or less per reactor year (per scram demand?) and thus not worthy of further evaluation.

To put this in better perspective a few further remarks are worthy of note. The Service Limit C criterion, as has been noted in the report, essentially limits stresses to the yield strength of the material or below for primary membrane stresses. This is what the criterion, in theory, permits. However, analyses submitted by the vendors have shown that, in reality, there is one, or at the most, a few areas within the RCPB that reach Service Limit C "first", i.e., at the lowest pressure, and are thus controlling. Such areas are: reactor coolant pump bolts and casings, or valve bodies.

Analyses further show that considerable margin exists between calculated stress levels at these "controlling" pressures and those that would be permitted for other "noncontrolling" components if they would actually be exposed to pressures that the Service Limit C criterion would permit. More specifically, the analyses show that the majority of components when exposed to the pressures that the few limiting components can tolerate before reaching the Service Limit C stress level, would see stress levels somewhere between those of Service Limit Band C. Note that Service Limit B limits pressure stresses to roughly 10 percent less than Limit C for this type of application.

It's important to recognize that in conjunction with imposition of the Service Limit C criterion, the primary concern which we have had is with regard to shutdown system isolation valve operability. Service Limit C, while limiting primary membrane stresses to below the yield strength does permit bending stresses above the yield strength thus raising concerns over permanent deformation that active valve bodies and discs might receive from exposure to pressures permitted by the Service Limit C criterion. As stated in the report, we have taken the postiion that for those few valves where operability is a concern, assurance of operability must be provided by tests or analyses, approved by the staff, which demonstrate that any resulting deformations will have no adverse effect on operability.

By way of clarification, it should also be noted that analyses have shown that the reactor vessel is not the "weak link" in the "chain" of RCPB components. Generally, there is several hundred pounds of pressure, psi, of margin between that which the so-called "limiting" RCPB components can

take without being exposed to stresses in excess of Service Limit C, and that pressure which would stress the controlling location of the reactor vessel to that level. An independent analysis performed for us for a typical PWR reactor vessel has provided confirmation of this. ~~for us~~.

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Analyses further show that considerable margin exists between calculated stress levels at these "controlling" pressures and those that would be permitted for other "noncontrolling" components if they would actually be exposed to pressures that the Service Limit C criterion would permit. More specifically, the analyses show that the majority of components when exposed to the pressures that the few limiting components can tolerate before reaching the Service Limit C stress level, would see stress levels somewhere between those of Service Limit Band C. Note that Service Limit B limits pressure stresses to roughly 10 percent less than Limit C for this type of application.

It's important to recognize that in conjunction with imposition of the Service Limit C criterion, the primary concern which we have had is with regard to shutdown system isolation valve operability. Service Limit C, while limiting primary membrane stresses to below the yield strength does permit bending stresses above the yield strength thus raising concerns over permanent deformation that active valve bodies and discs might receive from exposure to pressures permitted by the Service Limit C criterion. As stated in the report, we have taken the postiion that for those few valves where operability is a concern, assurance of operability must be provided by tests or analyses, approved by the staff, which demonstrate that any resulting deformations will have no adverse effect on operability.

By way of clarification, it should also be noted that analyses have shown that the reactor vessel is not the "weak link" in the "chain" of RCPB components. Generally, there is several hundred pounds of pressure, psi, of margin between that which the so-called "limiting" RCPB components can

take without being exposed to stresses in excess of Service Limit C, and that pressure which would stress the controlling location of the reactor vessel to that level. An independent analysis performed for us for a typical PWR reactor vessel has provided confirmation of this for us.

ITEM 4 - MAY 1 MCCRELESS MEMO

We interpret this question as asking the following:

"What is the probability of an ATWS-induced LOCA provided the staff fixes are implemented?"

More precisely, addressing the staff criterion recommended in NUREG-0460, which is most relevant to the probability of a LOCA, i.e., the recommendation to permit RCPB stress levels to reach ASME Code Service Limit C, we further characterize this ACRS question as "What is the probability of an ATWS-induced LOCA, providing the RCPB Service Limit C stress criterion is implemented?".

In response, we reply that we have not quantified this probability. However, based on engineering judgment we believe that considering the relatively low probability of ATWS as stated in the report, combined with an admittedly unquantified but, in our judgment, very low probability of RCPB failure utilizing the proposed Service Limit C criterion, that the overall probability of such a failure is on the order of 10^{-7} or less per reactor year (per scram demand?) and thus not worthy of further evaluation.

To put this in better perspective a few further remarks are worthy of note. The Service Limit C criterion, as has been noted in the report, essentially limits stresses to the yield strength of the material or below for primary membrane stresses. This is what the criterion, in theory, permits. However, analyses submitted by the vendors have shown that, in reality, there is one, or at the most, a few areas within the RCPB that reach Service Limit C "first", i.e., at the lowest pressure, and are thus controlling. Such areas are: reactor coolant pump bolts and casings, or valve bodies.

Analyses further show that considerable margin exists between calculated stress levels at these "controlling" pressures and those that would be permitted for other "noncontrolling" components if they would actually be exposed to pressures that the Service Limit C criterion would permit. More specifically, the analyses show that the majority of components when exposed to the pressures that the few limiting components can tolerate before reaching the Service Limit C stress level, would see stress levels somewhere between those of Service Limit B and C. Note that Service Limit B limits pressure stresses to roughly 10 percent less than Limit C for this type of application.

It's important to recognize that in conjunction with imposition of the Service Limit C criterion, the primary concern which we have had is with regard to shutdown system isolation valve operability. Service Limit C, while limiting primary membrane stresses to below the yield strength does permit bending stresses above the yield strength thus raising concerns over permanent deformation that active valve bodies and discs might receive from exposure to pressures permitted by the Service Limit C criterion. As stated in the report, we have taken the postiion that for those few valves where operability is a concern, assurance of operability must be provided by tests or analyses, approved by the staff, which demonstrate that any resulting deformations will have no adverse effect on operability.

By way of clarification, it should also be noted that analyses have shown that the reactor vessel is not the "weak link" in the "chain" of RCPB components. Generally, there is several hundred pounds of pressure, psi, of margin between that which the so-called "limiting" RCPB components can

take without being exposed to stresses in excess of Service Limit C, and that pressure which would stress the controlling location of the reactor vessel to that level. An independent analysis performed for us for a typical PWR reactor vessel has provided confirmation of this for us.

698. Commentary on EPRI ATWS Report, Draft (11 pages)

699. September 15, 1971 Instances of Relay Failures in Reactor Protective Systems (6 pages)

700. Questions for Navy (1 page)

701. Handwritten Notes (6 pages)

702. Handwritten Notes (3 pages)

703. Handwritten Notes (16 pages)

704. Preliminary Draft, A Cursory Review of the Reactor Safety Study Draft with Respect to ATWS (3 pages)

705. Frequency of Transients, Handwritten Notes (3 pages)

706. Undated to Stephen Hanauer from Ashok Thadani (9 pages)

707. December 21, 1976 to Denny from Jim (4 pages)

708. Dominant PWR Core Melt Probabilities, ref: WASH-1400, Table V-3-14 (3 pages)

709. Dominant BWR Core Melt Probabilities ref: WASH-1400, Table V-3-16 (2 pages)

710. Handwritten Notes (7 pages)

711. Handwritten Notes (4 pages)

712. Handwritten Notes (2 pages)

713. Issue 8: Use of Pro[illegible]istic Assessment of Reliability (12 pages)

714. ATWT Events of WCAP 8330 (73 pages)

715. ATWT Moderator Coefficient (84 pages)

716. Handwritten Notes (1 page)

717. July, 1975 Combustion Engineering, Inc. Report CENPD-158, Supplement 2 (15 pages)

718. May 13, 1977 Routing and Transmittal Slip to F. Cherny et al., from Hal Ornstein, attaching Draft to W. R. Mikesell from Keith Wichman and Frank Hill (6 pages)

719. April 14, 1976 to D. Moeller from Robert E. Heineman Subject: SUPPLEMENT TO STATUS REPORT ON COMBUSTION ENGINEERING ANALYSES OF ANTICIPATED TRANSIENTS WITHOUT SCRAM (8 pages)

720. February 21, 1978 Routing and Transmittal Slip to D. Ross et al., from L. Beltracchi (2 pages)

721. December 13, 1977 Note to D. Ross from Ashok C. Thadani (2 pages)

722. July 22, 1975 to R. Schemel from Warren Minners Subject: COMMENTS ON PROPOSED REACTOR SAFETY POSITION ON PROMPT RELIEF TRIP FOR BWR(1 page)

723. Undated Routing and Transmittal Slip to D. Ross et al., from L. Beltracchi, attaching January 16, 1978 Letter to Dr. Beltracchi from W. Frisch (27 pages)

724. September 23, 1976 Memorandum for Ashok C. Thadani from William E. Vesely Subject: REVIEW OF EPRI REPORT "ATWS REAPPRAISAL" (EPRI NP-251) (3 pages)

725. November 9, 1976 Routing and Transmittal Slip to Ashok Thadani from Robert G. Eastlerling (8 pages)

726. Attachment 1, Analyses Performed for ATWS Sensitivity Study (10 pages)

727. Handwritten Notes (2 pages)

728. Handwritten Notes (9 pages)

729. Handwritten Notes (11 pages)

730. Handwritten Notes (13 pages)

731. Handwritten Notes, PWR ATWS Analysis (19 pages)

732. Handwritten Notes, PWR ATWS Analysis (13 pages)

733. Graphs (2 pages)

734. Draft, Probabilistic Modeling of ATWS (14 pages)

735. Handwritten Notes (2 pages)

736. Handwritten Note to T. Novak from Ashok (3 pages)

737. Handwritten Notes, PWR ATWS Analysis (9 pages)

738. August 1972, Fuel Cladding Integrity During Anticipated Transients Without Scram (38 pages)

DRAFT

COMMENTARY ON EPRI ATWS REPORT

General Comments

1. The philosophical viewpoint is shaky. If the nuclear community is, or can be, convinced, based on data and on an understanding of scram system design, fabrication, and operation, that the systems are OK, then let's state it that way. To give that conclusion an aura of precision and quantification by a Bayesian analysis is phony, a numerical game, not a communication of information.

2. The report proposes a model in terms of demand probabilities rather than in terms of time to failure, but then estimates ATWS probability under the latter approach, not the former. If they followed their model through correctly, their estimates would double. The confusion lies in apparently thinking a constant failure rate model is identical to a constant failure probability model. They are not identical. Jumping back and forth as they do yields an erroneous factor of 2.

3. The report proposes a Bayesian approach rather than a non-Bayesian one, but the~~n fails to distinguish them~~ authors apparently don't know the difference. What is presented as a classical (i.e., non-Bayesian) analysis is actually a Bayesian analysis.

4. The level of understanding evidenced by this report cause~~d~~s ~~one~~ me to shift my prior distribution on ATWS probability further to the right, than it used to be.

Specific Comments

Page	Comment
3	The last sentence of the first paragraph gives first, an estimate of F_T (probability of scram failure, on demand) and then an estimate of $F_T/2$, but without an explanation. The latter is appropriate only if F_T is dependent on the elapsed time since the last scram system test. The report is quite confused on this score, possibly ~~apparently~~ in an attempt to have the best of both worlds. F_T is assumed not dependent on elapsed time in order to rack up a large number of historical demands, then it is assumed to be time dependent in order to justify the factor of 1/2 in approximating unavailability.
5	Second paragraph. It's not the usefulness of statistical confidence bounds that is questionable. Rather it's the usefulness of past data to predict future performance.
6.	I'm surprised that the Bayesian approach only got them a factor of 2. A good Bayesian would have gotten at least an order of magnitude out of his prior.

<u>Page</u> <u>Comment</u>

6 The authors call the Bayesian approach "a more comprehensive statistical method." I would replace "comprehensive" by "arbitrary". The decision of what to do about ATWS ~~will~~ depends on many factors, one of which is historical experience. To force all of these factors to be quantitative and then to algebraically merge them imparts an unwarranted aura of precision to what in the end is a judgment.

18 All the authors have done is replace 0/N by F_T. Thus, they have <u>not</u> switched from a model in which failure probability is time dependent to one in which it is not. If they had, unavailibility on demand would be given by F_T, not $F_T/2$. ~~[illegible]~~ (More on this below.)

22 The Comment is confused. The failure probability is not exponentially distributed; the time to failure is. The data are not chi-squared distributed with 2r + 2 degrees of freedom. Rather, for time-censored data, and the exponential model, the expression for determining confidence limits can be reduced to an expression involving the cumulative chi-square distribution with 2r + 2 degrees of freedom. ~~[illegible]~~

22 The exponential model has a constant failure rate. Thus the reference to "continued degradation" is misleading.

Page	Comment
22	If the~~y~~ authors wish to model the scram system as a demand failure, then they will have to estimate unavailability by estimating F_T, not $F_T/2$. They cannot take credit for the fact that only 1/Nth of a year~~s~~ elapse~~s~~ between tests and that the system failure could be anywhere in that interval.
	To model in terms of demand, they have to assume that a failure, when it occurs, would have occurred on that demand, regardless of whether the previous demand was one day ago or one year ago. Thus, in essence, they must assume that the failure occurs immediately after the preceding demand and successful scram, so the factor of 1/2 is not justified. This discrepancy is not so important quantitatively as it is in showing the modeling and analysis fl~~o~~aws in the report.
25	What we are really after when we calculate a confidence limit is an indication (in this case) of how high the ATWS probability might be, consonant with experience to date. We don't require confidence in the result. In fact, there are many reasons not to be too confident - simplistic models for example. In summarizing past experience, there's no reason why one can't calculate a whole battery of confidence limits, 50%, 75%, 84%, 95%, 99.99%, or whatever.
25	Last paragraph, second sentence. I suggest replacing "systematically" by "arbitrarily."

Page	Comment
26	I don't know what a "fault tree prior" is. I suspect, though, that prior distributions are assumed for all component failure probabilities and then the resulting distribution of system failure probability is approximated a la Wash-1400. A fault tree is used to express system failure as a function of component failures.
39	The constant failure rate and the failure-on-demand (with a constant failure probability) models are not compatible. Thus, they both cannot be applicable as the second sentence of the first paragraph states.
40	Usual terminology is "binomial model", not binary. Is there a reason for the authors using the latter?
41	You may <u>not</u> treat P(r,p) as a probability density function. The authors claim they are following Wilks, P. ~~2~~369, but they are not. What they are actually doing is a Bayesian analysis with a uniform prior over the (0, 1) interval. Only then can treating P(r, p) as a probability density function of p, given r be justified. I find it strange that the authors would propose a Bayesian analysis yet not recognize one when they did it.
41	The correct expression for determining an upper 100 C_0% confidence limit on p is to solve

$$1 - C_0 = \sum_{X=0}^{r} \binom{n}{X} p^X (1 - p)^{n - X}$$

for p. For r = 0, the solution is

$$p^* = 1 - (1 - C_0)^{1/n}.$$

Page	Comment
	For large n, the quantitative difference is negligible, but the conceptual distinction still holds. Their formulation yields BAyesian posterior probability limits, given a uniform prior, not statistical confidence intervals.
43	With 0 failures in n trails, a failure probability of zero or arbitrarily close to zero can't be ruled out, at any confidence level. Yet, the authors do so, giving a lower bound of 3.2×10^{-7}.
49	Appendix B uses different notation from Appendix A, but no explanation for the change is given. This can happen when reports are written by committee. P becomes p, p becomes f, and n is inserted where before it wasn't. Moreover, it's not clear whether f is the occurrence of a failed state or the probability of that event. Equation (B-3) is incorrect. All that can be said is that the left side is proportional to the right. Their claim that this is the classical result is incorrect because they have the classical result incorrect.
50	The binomial distribution doesn't require the prior to be a beta distribution. That's merely a convenient choice. Haven't the authors read WASH-1400? Don't they know all our priors are log-normal?

Page	Comment
50	Numerical correspondence between (correct) classical results and certain Bayesian results doesn't mean we regard ATWS as a single toss of a (fair) coin. They're two different analyses for two different purposes. The authors have been misled down the path of Bayesian mysticism by an accident of prior.
52	The assertion of the last sentence of the second paragraph can easily be shown to be wrong. This is pointed out not because it affects the quantitative results of the report but because it reflects the level of understanding the authors brought to their task. I'm a good Bayesian so am trying to incorporate all information in making a decision on the acceptability of this report.
53	The first sentence of the last paragraph is mysticism.
55	Why is this appendix called numerical analysis? The appendix content is rough estimation of the number of historical scram system ~~tests~~ demands.

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Applied Statistics Group
Office of the Executive Director
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DRAFT

COMMENTS ON THE USE OF RECTIFIABILITY AS PROPOSED IN "ATWS A REAPPRAISAL PART I[1]"

The effective recipe in Reference [1] for statistical analysis involving rectified failures is given in the quotation from page 3 as "Hence, we argue that the lower limit to the scram failure rate is correctly calculated using zero failures." This recommendation of such an absolute device seems implicit elsewhere in Reference [1] and is made despite

a) the rather weak premise that "any common mode failure that is discovered <u>is not expected</u> to occur again (<u>certainly not with the same frequency</u>) since either redesign, test and maintenance, or other quality assurance methods will be <u>adjusted</u> to eliminate this failure mode. This rectification eliminates potential failure modes and produces a '<u>better than original</u> 'condition."

b) the explicit recognition that "One could argue that redesign might introduce a new mode of failure. The counter-argument is that we <u>must</u> in any event alter the design or QA methods to eliminate, or maintain surveillance for, known failure modes and careful FMEA <u>will tell us</u> whether the alteration has introduced any new mode."

[Emphasis added for (a) and (b)]

Even if one assumes that any discovered failure does not occur again and that no new modes are introduced, the proposed statistical analysis is worthless as demonstrated by the example below. The analysis will be worthless even if, in the process of removing discovered failures, some undiscovered modes are also removed.

The fallacy of eliminating failures for which explanations or presumed explanations have been found can be demonstrated by the following hypothetical example. Let us suppose there are 100 different ways in which a scram system can fail and that each way manifests itself independently of the others at a constant rate of one per 100,000 reactor years. If we had observed 10 failures in 10,000 reactor years and "corrected" each failure then the "principle of rectifiability" leads us to an "unbiased estimate" of zero for the failure rate of the scram system. The 95% upper confidence limit would be computed as 3/10,000 = .0003. It is perhaps idle to speculate what the actual failure rate of the system might be at this stage. Under the simplistic assumption that the 10 observed ways have been eliminated and the others are still unaffected the failure rate is 90/100,000 = .0009. If three failure modes were removed every time one was discovered, the failure rate would still be .0007.

The upper 95% upper confidence limit in this hypothetical case is very unsatisfactory and the unbiased estimate is grossly misleading. The numbers could have been contrived to paint an even worse picture and such a choice might be as plausible as any other. Blind

acceptance of such statistical procedures is in fact dangerous. The oversimplification used in this example does not detract from the basic logic used. The example clearly shows that selective elimination of failures from the data base and application of conventional statistical methods to the remaining data violate the very principles on which conventional statistical analysis is based.

The reliability improvement due to rectification may be treated by reliability growth models. Even this raises very difficult problems relating to appropriate models and possibly to the sufficiency of the data base. Another possibility of treating rectification is in the framework of selecting populations. This in turn requires rather rigid selection procedures and adequate samples.

D. Rubinstein

REFERENCE

[1] ATWS: A Reappraisal Part I, An Examination and Analysis of "WASH_1270, Technical Report on ATWS For Water-Cooled Power Reactors", dated June 1976.

OPERATING EXPERIENCES

REACTOR SAFETY

UNITED STATES ATOMIC ENERGY COMMISSION

ROE No. 71-16 September 15, 1971

INSTANCES OF RELAY FAILURES IN REACTOR PROTECTIVE SYSTEMS

1.0 Summary

Nine instances of RELAY FAILURES IN THE MONITORING AND REACTOR PROTECTIVE SYSTEMS OF REACTORS are described in this report. Although the significance of any individual relay failure depends on the particular circumstances, the proper operation of most of the control, protective, and engineered safety feature systems for nuclear power plants depends on the successful functioning of relays. Relay failures of the types illustrated by the occurrences reported here could, under different circumstances, have significant adverse effects on plant safety. None of these occurrences resulted in damage to the reactor nor in radiation exposure of any individual.

2.0 Circumstances

2.1 During a period of about two months during preoperational testing of a power reactor, there were four failures of relays in the Reactor Protection System or the Containment Isolation System. In each instance, the relays that had been designed to open when de-energized failed to drop out on loss of power. These relays were all of the same general type, and supplied by the same manufacturer. There are approximately 400 of them in this plant. After investigation, the manufacturer directed that all relays be inspected for misalignment and chipped or broken self-aligning tabs, which were thought to be the cause of the malfunctions.

A few days after the memorandum calling for this inspection was issued, a new relay, which had been installed to replace one of those mentioned in the previous paragraph, failed to drop out when its coil was de-energized. It was evident that this relay was not misaligned, and since there was no obvious reason for the failure, it was returned to the factory for more extensive tests.

Further investigation disclosed that the relay malfunction was due to heat bonding of the armature to the upper (contact end) pole piece. The bonding agent was a black chromate primer which

had been applied to the pole pieces as a corrosion inhibitor. Evidently, during production of some of these relays, the primer had not been fully cured. As a result of this lack of curing and the buildup of primer between laminations, the internal heat generation during initial extended energizing of the relays in the field for periods of approximately 12 hours caused the armature to bond to the pole piece.

Field instructions were issued stating that the chromate primer was to be mechanically removed from the pole piece faces. The field memorandum also specified additional testing of the relays in their finally installed condition, following removal of the primer. The memorandum was sent not only to the operator of the reactor involved in this instance, but also to operators of three other reactors that might have received similar relays from the manufacturer.

2.2 During preventive maintenance on the plant protective system at a research reactor, one of the scram relays was replaced because its contacts appeared to be dirty and pitted. During subsequent tests before reactor operation, when the manual scram button on the console was depressed, the control rods did not drop.

The trouble was traced to the newly installed relay. An internal short circuit was found that shunted the console scram buttons, thus nullifying the manual scram action. All relays at this facility had been tested on a relay checker prior to being placed into service, but the checking instrument did not include provision for testing for internal short-circuits. All new relays are now checked for this defect when they are received from a supplier.

2.3 While manually withdrawing a control rod at a power reactor, the operator observed that the rod continued to move out for several seconds after the selector switch was moved to another rod. The same occurrence was observed several times during tests of rod insertion and withdrawal. Current is supplied to the rod drive through the contacts of a relay in the rod control circuit.

The armature spring that opens the relay contacts was found to be weak. Tightening of the spring resolved the problem.

2.4 During a routine shutdown of a research reactor, all the rods were being driven in by operation of a gang switch. When the rods were about six inches from full insertion, shim rod #1 suddenly stopped moving. The operator, thinking that the rod had possibly jammed, turned the gang switch to the neutral position, and the stopped rod started moving out. The operator then returned the gang switch to the insert position and shim

rod #1 stopped moving outward but still did not insert. By rapidly turning the gang switch from neutral to insert, the operator was able to accomplish full insertion of all the rods, including shim rod #1. Subsequent investigation showed that a loose armature spring on a relay was the cause of this situation.

At this facility, the direction of rotation of each of the rod drive motors depends upon which of two relays in the rod drive circuit is energized.

The armature spring of the withdraw relay of shim rod #1 drive circuit had come loose, thus permitting the relay contacts to remain closed. With the gang switch in the insert position the rod drive motor stopped because it had two simultaneous signals, one to insert and the other to withdraw. When the gang switch was turned to its neutral position, removing the insert signal, the rod was free to withdraw. Evidently, when the operator rapidly turned the gang switch from neutral to insert, he energized the insert relay and the vibration of its contacts closing caused the loose armature spring of the withdraw relay to realign itself to open the withdraw relay contacts and permit the rod drive motor to energize in the insert direction. Replacement of the faulty withdraw relay corrected the problem.

In order to increase reliability, additional relays were inserted in series with the existing control relays. The springs on both relays in series would have to fail at the same time in order to cause the same situation to recur. Maintenance procedures were changed so that relays will be replaced, checked and visually inspected for proper operation every six months.

2.5 During the daily checkout of a pulsed research reactor, it was observed that the transient rod did not drop consistently upon a scram signal. A relay in the control power circuit to the air solenoid for the transient rod was found to stick in the energized position intermittently. With the relay stuck, the solenoid is energized, and air pressure is applied to the transient rod piston preventing dropping of the transient rod until the power supply is shut off. It could not be determined whether the cause of the intermittent failure was residual magnetism in the relay armature or sticking contacts. The relay was replaced, and the scram circuitry for the transient rod performed normally.

2.6 A pulsed research reactor had been operated for about 8 hours at a constant power level, with the pulse rod withdrawn and the pulse rod "UP" button depressed by hold-in power on its relay coil. The pulse rod had performed properly on scram tests prior to this run. At the conclusion of the run, when the manual scram bar was depressed, the "UP" button on the pulse rod failed to release. As a result the pulse rod did not drop, although the other two rods scrammed properly and shut the reactor down. The pulse rod dropped when the "DOWN" button was depressed but since the "UP" button was still depressed, the rod immediately withdrew upon release of the "DOWN" button. Repeated action of the "DOWN" button produced the same results. Finally, when the "DOWN" button was depressed for an estimated 15 to 20 seconds the "UP" button was released and the rod dropped.

The "UP" button on the pulse rod is held in by a holding coil. Current to the coil is supplied by the scram circuit and the current was apparently interrupted satisfactorily since the associated light operated correctly to indicate rod scram circuit action. The investigators concluded that the sticking of the "UP" button may have been caused either by mechanical binding of the switch or by a residual magnetic flux in the electromagnet core due to hysteresis. Since the button released some seconds after current collapse, it was judged that the second explanation was the most likely. The switch and coil assembly was disassembled, cleaned and reassembled and no obvious defect was observed. Further testing, after reinstallation of the assembly, failed to reproduce the effect previously observed. Operating procedures were issued to instruct the operator to turn off the Reactor Key or Reactor Power immediately should this situation recur.

A few weeks later, a similar situation did occur. The assembly was again removed and tested for residual magnetism. The coil and plunger were observed to exhibit some magnetism, which was eliminated by a degaussing coil. Operation was resumed, with the concurrence of the Reactor Safety Committee, after a replacement switch was installed. Repeated testing of the old switch on a mock-up test circuit failed to reproduce the effect following degaussing.

2.7 During checkout prior to startup of a research reactor, it was found that none of the automatic scram circuits could be reset. Investigation revealed that one of the six scram reset switches, all of which must be in the closed position in order to reset the scram system, was open, thus interrupting the common return

from all six of the automatic scram relays to the power supply for the rod magnets. The defective switch was replaced and the scram system was reset.

During the investigation of the cause of the inability of the scram system to reset, analysis of the circuit revealed that the magnetic and automatic shutdown circuit was not protected from a single failure in the scram relay power supply for the six automatic scram relays. One or more of these six relays must be energized in order to interrupt the current to the magnets in the control rod drives (by de-energizing the normally closed master scram relay in the circuit to the magnet power supply). These individual automatic scram relays would not be energized in the event of a failure of any component in the relay power supply, if such failure caused the power supply voltage to drop significantly. The normally-closed, series-wired contacts in the vital scram bus remain closed so long as the automatic shutdown relays are de-energized. To correct this condition, a new relay has been added to the circuit, which is energized by the scram relay power supply via a voltage divider network. One set of contacts on this new relay is used to interrupt the magnet power to the control rods in the event of a failure of the scram relay power supply or one of the scram reset switches. Another set of contacts is used for an indicator light which indicates that the relay is energized. A similar problem in the reactor protection-system at another research reactor was reported in ROE: 69-16, dated June 13, 1969.

2.8 At a sodium-cooled reactor a leak detector is provided to trip the auxiliary primary coolant system pump if sodium leakage is detected. During the performance of monthly tests of the leak detectors, the leak detector trip contacts in the safety chassis were jumpered to prevent tripping of the pump each time a detector was tested. During subsequent removal of the jumper, one end was inadvertently grounded resulting in high current. The control circuit fuse opened, was immediately replaced and the detector was returned to service.

A month later, during the monthly leak detector testing, the pump failed to trip when a leak was simulated by shorting across the detector output. Investigation disclosed that the leak detector relay contacts in the pump control circuit had been welded together, evidently as a result of the maintenance error the previous month.

2.9 The signals from the stack effluent sensors, at a power reactor, feed into indicators, equipped with containment isolation trip contacts, of the type sometimes referred to as meter-relays. In meter-relays, the indicator needle moves upscale until it contacts an internal trip contact, at which point the trip is initiated. The indicator cover is sealed by a sponge rubber gasket.

In this instance, the gasket evidently had age-hardened, dried out and cracked into pieces. One small piece of gasket material dropped into one of the indicator needle mechanisms and prevented it from going upscale (the containment isolation would still have been initiated, if required, by redundant trip units). The operator discovered the problem during the daily trip test of the system. Other meters of the same design were also found to have loose pieces of gasket. The gaskets on all the meters were replaced with a more resilient material of a guaranteed longer life.

Division of Reactor Licensing
U. S. Atomic Energy Commission

QUESTIONS FOR NAVY

1. Do Naval RPS Systems utilize equipment diversity?

2. What, if any, are the significant differences in the design of Naval and Commerical Scram systems?

3. What, if any, are the significant differences in the Naval and the Commercial Scram systems quality assurance programs?

4. How frequently is Scram system tested?
   Do the tests consist of partial tests?
   Do the tests include the CRDMs?
   How would the tests compare with the commercial tests?
   Do the operators perform the tests?

5. Would the tests detect CMFs in the electronics?

6. Would the tests detect CMFs in the CRDMs?

7. Has the testing contributed to any CMFs in the Scram system?

8. Have there been any CMFs or potential for CMFs in the Scram systems?

9. Is the EPRI use of Naval data appropriate?

EPRI Report NP-424
To: T. Novak
From: D. Solberg Date: 5-27-77

Transients Without Scram (ATWS) ✓
To: H. G. Rickover
From: W. B. Loewenstein Date: March 16, 1977

ATWS subject ✓
To: G. S. Lellouche
From: Ben C. Rusche Date: April 4, 1977

EPRI Report "ATWS Reappraisal" ✓
To: Ben C. Rusche
From: Gerald S. Lellouche Date: 10-7-76

EPRI Report ✓
To: G. S. Lellouche
From: Ben C. Rusche Date: Sep 21, 1976

EPRI ATWS Reappraisal ✓
To: Williams E. Vesely
From: Robert G. Easterling Date: Nov 8, 1976

Look on the back

EPRI Report ATWS Reappraisal
To: Ian B. Wall
From: R.E. Heineman Sep 17, 1976

EPRI Report
To: W. Minners
From: Ashok Thadani Sep 01, 1976

ATWS
To: R. Moore
From: Ashok Thadani Aug 26, 1976

EPRI
To: R. Easterling
From: Ashok C. Thadani 10-8-76

EPRI Report
To: A. Thadani
From: Stephen H. Hanauer Sep 15, 1976

IEEE-577
To: A. Thadani
From: S. H. HANAUER Date: 2-10-77 ✓

ATWS Results for BWR-4's ✓
To: Stephen H. HANAUER
From: HAL ORNSTEIN Date: Feb 8. 1977

EPRI Rept: ATWS: ✓
To: B. Rusche
From: G. S. Lellouche Date: 9-20-76

Electric Power Research Institute (EPRI) ✓
on ATWS To: A. Thadani
From: Thomas Novak Date: 2-14-77

EPRI meeting ✓ Date: May 12, 1977
To: Ashok C. Thadani
From: Robert G. Easterling

EPRI ATWS meeting summary ✓
To: T. M. Novak Date: June 2. 1977
From: A. C. Thadani

EPRI report "ATWS Reappraisal" NRR-1287 ✓
To: A. Thadani
From: Ben C. Rusche Date: Nov 17. 1976

EPRI letter
To: R. Easterling
From: S. H. HANAUER Date: 3-18-77

EPRI Report "ATWS Reappraisal"
To: Ian B. Wall
From: R.E. Heineman Date: Sep 17.

EPRI Report
To: S Hanauer
From: Ashok C. Thadani Date: Sep 1.

Reactor Protective Systems
To: Summary Date: Sep
From: U.S. Atomic Energy Commission

ATWS
To: L. Nichols, R. Moore
From: Ashok Thadani Date: August 2

ATWS for Water Cooled Power Reactor
To: Ashok Thadani
From: S. H. Hanauer Date: October 6

EPRI ATWS Report
To: R. Easterling
From: Ashok C. Thadani Date: 10-8-76 ✓

EPRI Report SAI/SR-126-PA
To: A. Thadani Date: Sep 15.
From: Stephen H. Hanauer

AIF/NRC meeting
To: B. Rusche ✓
From: G. S. Lellouche Date: August 24. 1[illegible]

EPRI - Report NP251
To: Ashok Thadani
From: David Rubinstein Da

EPRI Reappraisal of ATWS
To: T. Novak
From: Ben C. Rusche Date

EPRI Report "ATWS Reapp
To: Ian B.
From: R. E. Heineman Date:

EPRI Report
To: G. S. Lellouche
From: Ben C. Rusche Date:

EPRI ATWS
To: T. Novak
From: Ashok C. Thadani Date:

ATWS for Water-Cooled Power Rea
To: Gerald S. Lellouche
From: Ben C. Rusche

Log Normal Bounding
From: Robert G. Easterling
To: William E. Vesely Date: Feb

Gatlinburg Paper
To: Williams E. Vesely
From: Robert G. Easterling Date: Fe

(EPRI)
To: D. F. Ross, Jr. ✓

ATWS
To: B. C. Rische ✓
From: R. R. Fullwood Date: 11-24-76

ATWS
To: B. Rusche ✓
From: G. S. Lellouche Date: Sep 20, 1976

PWR risk due to ATWS
To: B. Rusche ✓
From: G. S. Lellouche Date: October 28. 1976

ATWS ✓
To: S. H. Hanauer
From: G. S. Lellouche Date: January 13. 1977

EPRI ATWS
To: Ashok Thadani ✓
From: Robert G. Easterling Date: 11-9-76

EPRI NP-251 (ATWS)
To: Tom. Novak ✓
From: Ben C. Rusche Date: 8-22-76

ATWS
To: B. Rusche ✓
From: G. S. Lellouche Date: Sep 3. 1976

EPRI Report "ATWS Reappraisal
To: Ashok C. Thadani ✓
From: William E Vesely Date: Sep 23. 1976

Institute Study of ATWS ✓
To: B. C. Rusche

Modification of Staff Position on Anticipated Transients without SCRAM (ATWS).
To: The Commissioners
From: Ben C. Rusche Date: Nov 18. 1975

Alternatives to staff Position Regarding Anticipated Transients without SCRAM (ATWS)
To: Lee V. Gossick
From: W. Kerr Date: October 17. 1975

ATWS Meeting with the ACRS Subcommittee
To: Thomas M. Novak
From: Ashok C. Thadani Date: Jan. 15. 1975

Water-Cooled Power Reports (WASH-1270).
To: Dr. Dade W. Moeller
From: Ben C. Rusche Date: April 12. 1976

NRC Status Report
To: Mr. Ivan Stuart
From: Robert E. Heineman Date: April 7. 1976

Recirculation Pump Trip To Help Limit The Consequences of An ATWS event in BWRs
To: Lee V. Gossick
From: Dade W. Moeller Date: March 12. 1976

ACRS ATWS Subcommittee meeting summary
To: D. F. Ross
From: Warren Minners Date: Jan 22. 1976

ACRS - ATWS Subcommittee Meeting
To: Thomas M. Novak
From: Ashok Thadani Date: Feb 24. 1977

ACRS Staff
To: Thomas Novak
From: Ashok Thadani Date: 2-15-77

Reports on BWR-ATWS
To: David Okrent
From: Ashok Thadani Date: Jan 21. 1977

ATW criteria of Class A plants.
To: Thomas M. Novak
From: Ashok Thadani Date: June 27.1974

ACRS Subcommittee meeting on ATWS
To: J. Hendrie
From: Victor Stello, Jr. Date: July 15, 1974

James H. Conran
OGC
From: Victor Stello, Jr. Date: Sep 4.1974

ATWS Position
To: William Kerr
From: Ben C. Rusche Date: Sep 12, 1975

ACRS consultant report (CT-0656)
To: D. Ross
From: F. Schroeder Date: 4-16-76

Advisory Committee on Reactor Safeguards
To: Raymond F. Fraley
From: D. F. Ross Date: April 12.1976

automa[illegible]
event to meet the [illegible]
and published their findings in December 1975 as the staff report.

This report lists staff concerns with the GE submittal. Subsequently, in accordance with the staff request (April 7, R. E. Heineman to I. Stuart), GE provided analyses (7/2/76) consistent with the staff status report guidelines. Further on 9/30/76 GE submitted a report on "BWR Scram System Reliability Analysis" to support their position that the staff status report requirements were excessive. The staff is reviewing the GE submittals and will require that any changes which may be required be incorporated into the design in a timely manner.

Black Fox ATWS Contention
To: S. Varga
From: Thomas Novak June 29. 1977

Black Fox ATWS Contention
To: Thomas Novak
From: L. Dow Davis May 23. 1977

Reactor Safety Study
To: Sir
From: Richard E. Webb, Date: Oct 31. 197

Turkey Pt. Unit #3

	3-5-73	Turbine trip due to control oil problem	1350 Mwt ~~Hot Shutdown~~
	4-3-73	Volt spike from #3A inverter plus chan. #I prot. circuit in trip mode	1750 Mwt
			" "
	4-4-73	3A inverter failure	" "
	5-11-73	Volt. transient on 240,000 volt. switchyard bus	1150 "
	6-23-73	Undervoltage caused by system disturbance	1830 "
Unit #4	6-23-73	" " " "	600 "

Surry #1

			Eff. full pow ~~?~~
	~~8-8-73~~	~~MNO Rx trip by turbine trip failure of EHC card~~	~~90%~~
	11-28-73	" " " " " turb. tripped due to A line differential lockout relay trip ("C" main transformer) lighting arrester defective	90%
~~Surry #2~~	~~8-8-73~~	~~[illegible]~~	
	1-4-73	Turb. trip - loss of auto-stop oil	74%
	2-6-73	" " hi level in 1B SG caused by controller failure	74%
	2-25-73	" " Hi SG level - instrumentation problem	75%
	3-27-73	" " maintenance personnel operating wrong controlling relays which caused turb. gen. output breaker to open	75%
	4-13-73	" " caused by "Line Differential" Lockout Relay Trip" - Generator repaired	85%
	4-21-73	" " caused by same as above - gen. volt reg. repaired	85%
	6-13-73	" " EHC oil leak	"
Surry #2	3-10-73	" " Faulty relay in general protection system	30
	4-10-73	" " " gen. volt. regulator	55
	4-21-73	Loss of gen. field	75
	4-22-73	Gen. time diff. lockout relay trip	"

Zion 1	12-9-73.	Trip - Generator cooling Failure	Cold Shutdown
	8-2-73:	Turbine runback - trip	Hot Shutdown ?
	9-17-73:	Turbine reverse power-trip	" "
	9-18-73:	Decreased vacuum - trip	" "
	10-18-73:	Operator error during bus switching	" "
Zion 2			
	12-24-73	Turbine trip, Rx power indicated over 10%	" "
Vermont Yankee			
	4-13-73	Turbine trip assoc. w. testing the turbine emergency governor	75% Power
	1-17-73	Low reactor water level scram ~~during~~ while attempting to calibrate LI-2-3-73A (Yarway)	100%
Yankee Rowe			
	5-3-73	Elec. disturbance on 69 KV line actuating low main coolant flow under-current of main coolant pump motors for loops 2 & 3 resulting in reactor scram & turb.-gen trip	575 MW
	5-23-73	Diff. operation in 115 KV yard (owl on the Bus) actuating ... (same as above)	583 MW
Turkey Point Unit 4			
	5-6-73	Turbine overspeed protection control system misoperation. Incorrect pressure switch setting found	unit shutd.
	10-23-73	Loss of field on feed drive M-G set due to loose connection in diode circuit	" "
Unit #3	2-4-73	Generator loss of field trip	1470 "
	2-7-73		1500 "
	2-28-73	System fault-undervolt. on both 4160 v busses	1820 MW

San Onofre 1-10-73 No 4 Volt Reg opened by being bumped 3.9x10^6 kw
(Turb was off-line at the time) — Not a loss of load

H.B Robinson - None

Quad Cities 1 (BWR) Up in power

✓ 9-12-73 Turb trip on high reactor water level. 780MWe
Feedwater "A" reg valve went wide open 800

10-30-73 Scram due to hi flux - caused by 9 turb. 780MWe
BV opening & CV closing. Problem caused
by removal of a jumper by a GE rep
and IM which they installed for grounding

Quad cities 2511

Quad cities 2 9-16-73 Scram APRM Hi Flux - Aux. oil pump 650
for 2A recirc. M-G set tripped due
to an Agastat coil failure causing
recirc. pump to fluctuate rapidly

Surveillance testing main work usually checks at lower level of power

✓ 9-29-73 Scram - MSIV 2-203-2D drifting close
following surveillance testing

10-12-73 Scram on low H_2O lvl. Low volt. on
essential service bus

✓ 10-23-73 Trip from high mois. sep. level. While
relatching a htr. string, emerg. drain vlve.
on MS dr. tk. failed to open

✓ 11-13-73 Scram - mois. sep. lvl. hi while latching
2B2 drain on FW htrs.

Quad cities #1

1/4/73 Scram - lo wtr. lvl. Hi diff press. across
cond. demineralizers

✓ 1/17/73 Scram - main stm line hi flow. Broken air
line to solenoid pilot on MSIV 1-203-1A
caused valve to close

✓ 3/10/73 Scram - turb gen. load m/m. Low EHC oil press.
spike occurred.

3/19/73 Scram - low H_2O lvl - elect. control failure

3/20/73 Scram - hi mois sep lvl. - two MS dr. tk. level
control vlv. controllers had defective diaphragms.

F

Quad Cities 1 (cont.)

3-26-73 Scram APRM Hi Flux. Bad megacycle bd. in EHC speed circuit caused unstable bypass & CV operation.

6-6-73 Scram - low wtr. lvl. - short capacitor in limiting network of lvl. master controller.

QC #2

1-4-73 Turb. trip - mois. sep. hi lvl. MS dr. tk. 2B emerg. dr. vlv 2-3311B stuck half open

1-7-73 Scram - 2A mois. sep. dr. tk. level hi turb trip. Energ. dr. vlvs open due to a drift of MSDT level xmitter.

1-30-73 Scram - hi wtr. lvl. - Spike when a FW reg. module cause FW pump runout

3-2-73 Scram - erroneous signal of ~~dis to~~ main stm line hi flow

3-11-73 Scram while shutting down. Manual hand wheel on FW reg vlv. backed out from vibration, vlv. would not close. Turb. trip for hi wtr. level in Rx.

5-17-73 Scram - low cond vac.

6-1-73 " - low Rx wtr. lvl. Fitting in air supply line to FW reg. vlv. broke at vlv., causing vlv. to close

Prairie Island - None (just starting up)

Point Beach (PWR)

#2 12-15-73 Malf. in EH Control sys shifted it from load control to speed cont. All CV shut causing 100% load rejection & overtemp. ΔT trip

Pilgrim (50-293) (BWR)

1-15-73 MSIV closure caused by surveillance test of main stm hi temp. switches

2-6-73 During test MSIV to 90% open pos. & then reopening, outbd. vlv. on "B" stm. line malfunction

Oyster Creek (BWR)

~~9-8-73 Scram - lo Rx wtr lvl - Plant shutdown~~ ~~lk~~ At you

11-23-73 Scram - ~~NMS~~ Neutron Mon. Syst. trip initiated by loss of continuous power supply

1-1-73 Scram - MSIV closure following loss of gen. field due to opening of 24 KV potential xformer cabinet

2-18-73 Scram - mats sep. hi lvl.

4-13-73 " " " " "

6-21-73 Turb. trip on hi. wtr lvl. caused by trip of "B" FW pump & subsequent rapid load reduction with recirc. flow

6-22-73 Rx high press scram caused by sluggish turb CV operation.

6-30-73 Turb-gen. trip due to relay operation on both 230 KV lines due to an electrical storm

Oconee #1 (PWR)

7-~~14~~-73 Malf. in speed controller on B ~~FW~~ pump resulted in loss of FW & hi press Rx trip

9-14-73 Generator trip due to lightning surge caused Rx trip

12-11-73 Transient in EHC caused turb gen trip which caused the Rx trip on hi press.

Nine Mile Pt (BWR)

7-8-73 Scram during MSIV closure testing

11-29-73 Scram on Rx lo wtr lvl. when operator experienced difficulty with FW pump line up

2-9-73 Gen trip due to loss of excitation

Monticello (BWR)

11-6-73 MSIV closure scram - inadvertent trip of a main stm. line hi flow sensor during inst surveillance

Millstone 1 (BWR) ← ~~No details in book, [illegible] trip [illegible]~~ for July - Dec

1-17-73 IRM Ch. #16 trip (noise on system)

3-14-73 Scram - while doing vessel hi press. Scram Functional test with a half scram, fuse blow on CRD scram solenoid Group II scram this group of rods. Resulting void collapse caused a low wtr. lvl. & full scram

Maine Yankee 1 (PWR)

5-19-73 Turb. & Rx trip from 74% power caused by de-energizing DC control power to Rx trip brkrs. TB-1 & TB-5.

3-17-73 Turb. & Rx trip from 73% power caused by malf. of turb. EH governor control power supply

LA Crosse (BWR)

7-2-73 Rx. Scram on low recirc. Flow when both Forced Circ. Pump tripped on low SI ΔP.

7-6-73 Scram on low recirc flow. 1A FCP had tripped from erroneous hi seal leak off temp. trip.

7-9-73 Scram Seal injection sys. caused both Forced Circ. Pumps to trip

7-13-73 Rx hi flux scram on Ch. #6 due to feeding too fast with Rx feed pump in auto & the stm flow too low to give proper pump control

6-25-73 Rx scram due to both forced circ. pumps tripping caused by control problem on seal inj. sys

Indian Point #2 (PWR) Scrams

6-26-73	Unit Aux. xformer Diff. due to incorrectly wired (dwg. error) current xformer
6-29-73	#22 Main xformer Diff. due to wiring error (dwg. error)
7-21-73	Low auto stop oil pres due to blockage orifice
7-23-73	underwolt. while changing faultly relay in auto. volt. reg
9-21-73	Loss of 6.9 Bus during transfer of aux from station supply to unit supply
10-11-73	Loss of Field

Humboldt Bay (BWR) - NONE

Ginna (PWR) No loss of load

Haddam Neck (PWR) No loss of load

Fort Calhoun (PWR) No loss of load

Dresden 3 (BWR)

Scrams

7-1-73	Low condenser vacuum
7-11-73	Gen load reject
7-18-73	Loss of gen. field excitation
8-25-73	EHC oil leak
9-7-73	Low Rx wtr. lvl. - malf. of master lvl. cont. for FW sy
10-21-73	Failure of mode switch contacts to properly make-up
11-27-73	Low Rx wtr. lvl. - malf. in FW control system
5-28-73	Low condenser vac. Vacuum was broken to slow turb. rapidly to minimize vibration following a turb trip
6-6-73	Hi Flux - "A" recirc. pump being returned to service

Dresden 2 (BWR)

9-29-73	APRM Hi-Hi caused by "2B" recirc. pump flow spike
10-19-73	MSIV closure due to drywell pneumatic supply valve being closed
11-14-73	Low Rx wtr. lvl. caused by failure to open FW reg. valve contrary to starting proced.
11-27-73	" " " " caused by feed pump trip
1-13-73	Low " " " caused by FW reg. vlv. problem
2-19-73	" " " " " " " " " "
4-29-73	Premature switching from "startup" to "run" mode caused a Group I isolation

Dresden 1

6/19/73	Scram due to low wtr. lvl. in drum during startup - FW valve failed to open against a differential pressure

Big Rock Point (BWR) NONE

Turkey Pt. Unit #3			
	3-5-73	Turbine trip due to control oil problem	1350 Mwt ~~Hot Shutdown~~
	4-3-73	Volt spike from #3A inverter plus chan. #I prot. circuit in trip mode	1750 Mwt
			" "
	4-4-73	3A inverter failure	" "
	5-11-73	Volt. transient on 240,000 volt. switchyard bus	1150 "
	6-23-73	Undervoltage caused by system disturbance	1830 "
Unit #4	6-23-73	" " " "	600 "
Surry #1			Eff. full pow ~~...~~?
	~~3-8-73~~	MNR ~~Rx trip by turbine trip failure of EHC card~~	~~30%~~
	11-28-73	" " " " " turb. tripped due to A line differential lockout relay trip (on main transformer) lighting arrester defective	90%
~~Surry #2~~	~~date~~	~~cause~~	
	1-4-73	Turb. trip - loss of auto-stop oil	74%
	2-6-73	" " hi. level in 1B SG caused by controller failure	74%
	2-25-73	" " Hi SG level - instrumentation problem	75%
	3-27-73	" " maintenance personnel operating wrong controlling relays which caused turb. gen. output breaker to open	75%
	4-13-73	" " caused by "Line Differential" Lockout Relay Trip" - Generator repaired	85%
	4-21-73	" " caused by same as above - gen. volt reg. repaired	85%
	6-13-73	" " EHC oil leak	"
Surry #2	3-10-73	" " Faulty relay in general protection system	30
	4-10-73	" " " gen. volt. regulator	55
	4-21-73	" " Loss of gen. field	73
	4-22-73	" Gen. time diff. lockout relay trip	"

VY	1
Quad Cities 1	3
" " 2	7
Pilgrim	1
Oyster Creek	6
Nine Mile Pt	2
Monticello	1
Millstone 1	
LaCrosse	1
Dresden 3	5
Dresden 2	1

Frequency of Transients

I PWR EXPERIENCE

LOSS OF OFFSITE POWER

GINNA (W)	10/21/73		
SAN ONOFRE (B&W)	6/7/73		
HADDAM NECK (W)	1/19/74		
"	7/15/69		
POINT BEACH (W)	10/13/73	?	Auxiliary only
"	1/4/71	?	"
"	2/5/71		
INDIAN POINT 1 (B&W)	12/26/70		
"	7/20/72		
INDIAN POINT 2 (W)	9/23/73		

TOTAL DEFINITE LOOP = 8 (Perhaps 10)

COMPLETE LOSS OF FEEDWATER

GINNA	7/14/72	Seal H_2O low ΔP
IPP#2	10/5/73	closure of all FCV

References

(1) ANSI - N18.2

2 (1) WCAP - 8330

3 (2) BAW - 10099

4 (3) CENPD - 158

5 & Millstone Letter

(4) WASH - 1400

(5) NEDO - 10349

NEDO - 20626

(6) (6) WASH - 1270

6 WASH - 1400

7 WASH - 1270

8 NEDO - 10349

9 NEDO - 20626

10. Pool temp data

2

Frequency Of Transients

Only 1973 Experience

BWR

Most Significant Transient
Loss Of Load

Plant	Incident	Date	No.
Vermont Yankee	Turbine Trip	4/13/73	1
	Low Level	1/17/73	1
Quad Cities 1 & 2			21
	Several transients occurred as a result of testing		
Pilgrim	"		2
Oyster Creek			7
Nine Mile Pt.	Some during testing		3
Monticello	MSIV closure		1
Millstone # 1	Testing		2
La Crosse			5
Dresden 3			9
Dresden 2			7

Total ≈ 79

Frequency Of Transients

Only 1973 Experience

PWR

Most Significant Transients

a) Complete Loss Of Feedwater

b) Loss of Offsite Power

c) Loss of Load

INCIDENT		PLANT	DATE	NUMB
a	Complete Loss of feedwater	IPP #2 (W)	10/5/73	1
b	Complete loss of Offsite Power	Ginna (W)	10/21/73	3
	"	San Onofre (B&W)	6/7/73	
		IPP # 2	9/23/73	
c	Loss of Load	Turkey Pt. 3 & 4		9
	"	Surry #1		8
	"	" #2		4
	"	Zion 2		1
	"	Yankee Rowe		2
	"	San Onofre		
	"	Oconee #1		1
	"	Maine Yankee		2
	"	IPP # 2		6

AB

P R E L I M I N A R Y   D R A F T

A CURSORY REVIEW OF THE REACTOR SAFETY STUDY DRAFT WITH RESPECT TO ATWS

1. Introduction

Since the publication of Technical Report of Anticipated Transients Without Scram for Water-Cooled Power Reactors (WASH 1270) in September 1973, some additional technical information has become available that bears on the ATWS problem.

No! [illegible] since [illegible] 1974

a. The light water reactor vendors have each published detailed analyses of the reliabilities of their protection systems including some common mode failure considerations.

b. The light water reactor vendors have each published detailed evaluations of the course of postulated ATWS events, thus extending the evaluations referenced in WASH 1270. These more recent evaluations have included improvements in the calculational models used, and more nearly consistent assumptions, initial conditions, and value of parameters.

c. The Reactor Safety Study draft issued in August 1974 (WASH 1400) included considerable detail in its consideration of both anticipated transients and scram failures.

In their publications, and in letters to the NRC staff, the light water reactor vendors have suggested forcefully that the staff position set forth in WASH 1270 is technically incorrect, and that this is easily and convincingly demonstrated by the analyses contained in the industry publications referred to above and in the Reactor Safety Study. This present review was undertaken

to consider a part of the question thus raised; to review the information contained in the Reactor Safety Study draft that is relevant to ATWS and to evaluate whether this information is, in fact,in technical contradiction to WASH 1270.

This review is not intended to be exhaustive and cannot be final. We are dealing with a draft of the Reactor Safety Study and are mindful of the Commission's admonition (39FR 30964) that the draft and comments on it from interested individuals and institutions must be thoroughly evaluated in its final form before it is to be used as a basis for licensing decisions. However, it seemed appropriate to take into account the new information contained in the study draft and to make this preliminary review in view of the considerable time yet remaining before a final study report can be issued and evaluated.

2. Logical Framework

Briefly describe (and attach all the considerations which must be taken into account).The reasoning in WASH 1270 is as follows: The frequency of an ATWS event is given by the product of the frequency of occurrence of anticipated transients multiplied by the conditional probability of failure to scram. Such events have a spectrum of consequences. An estimate was made of the frequency of ATWS events leading to consequences in excess of 10 CFR 100 guidelines.On the basis of this evaluation the staff concluded that a staff ? improvement in reactor shutdown system reliability should be implemented.

Our review of the Reactor Safety Study draft information as it affects ATWS therefore naturally questions the following topics:

a. The frequency of occurrence of anticipated transients with potentially severe consequences.

b. Reliability of reactor shutdown on the occurrence of such an anticipated transient.

c. The consequences of various possible postulated ATWS events turned up in Items a and b.

d. The safety objective for the frequency of severe accidents.

C. The limits that should not be exceeded following an ATWS event.

19th March

Wednesday

Frequency of Transients

BWR

Monticello SER

Table III

MSIV Closure	7/14/71
Low Condenser Vacuum	8/7/71
Improper Opening of Sensing Line	8/9/71
High flux	8/20/71
Low Water Level	9/5/71
High flux	9/28/71
High flux	2/11/72
Closure of Turbine stop Valves	2/26/72
High Pressure	3/3/72
Closure of Turbine Stop Valves	4/8/72
Hi Radiation (?)	4/21/72
Generator Trip	5/23/72

From this date on Info from Supplemental Report

Turbine Lockout	7/10/72
Generator "	7/21/72
Accidental Jarring of Steam flow Transmitter	11/6/72
Hi Reactor Water Level	5/26/73
Control Valve fast Closure	6/16/73
Low Condenser Vacuum	6/10/74
Generator Lockout	6/19/74

Total Incidents	1971	1972	1973	Aver
	6	9	2	6/.

TRANSIENTS

1973 TRANSIENTS

PWR

1. LOSS OF FEEDWATER
2. LOSS OF OFFSITE POWER
3. LOSS OF LOAD (TURBINE TRIP)

BWR.

TRANSIENTS REQUIRING SCRAM

1973

SIGNIFICANT POINTS

Appendix V

P 56, P 58 and P. 60

Loss of fw flow ≈ 3 times/year

Appendix V

P. 62

Prob of valves to fail to reclose ≈ 10^{-2} based on Operating Data. If no ATWS 'fix' provided the primary pressure would be far in excess of those experienced for the data base of hence probability of 10^{-2} is not truly reflective of this event.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Stephen H. Hanauer, Director, Office of Technical Advisor

For the past two weeks I have been reviewing the reactor safety study (RSS) draft report ATWS evaluation. ~~The RSS definition for core melt, however conservative, has been maintained.~~ With the assumption that the unreliability of the reactor protection system is about 10^{-4} per demand, I looked at the frequency of the significant transients, the reliability of certain significant systems and the sequence of events that may lead to core melt. In summary, the safety study may have overestimated ~~been too conservative in the estimation of~~ the significant transient occurrence rate and appears not to have considered some events that may lead to core melt. Using the RSS criteria for core melt, it appears that conservatively the probability of core melt from ATWS events may be about 10^{-5}, per reactor year and at 10^{-6}.
Attached are details of my evaluation.

Ashok Thadani
Reactor Systems Branch
Division of Technical Review

Enclosure:
Evaluation

cc: T. Novak, RSB:TR
W. Minners, RSB:TR

Steve.
As you can see this is in a predraft form. References are incomplete and I might revise the cover letter a little. I shall get a typewritten copy to you tomorrow pm. But in the mean time I hope this gives you the information you need.
Ashok. 3/19/75

Section 1. ANTICIPATED TRANSIENTS FREQUENCY

Anticipated Transients

Any deviations from normal operating conditions occurring with a frequency of one or more times during the service life of a plant are called "Anticipated Transients'. There are a number of anticipated transients, some of quite trivial nature and others that are more significant in terms of the demands imposed on plant equipment.

Anticipated transients are condition II events, Incidents of Moderate Frequency, as defined by ANSI[1].

PWR Limiting ATWS Events

In the case of transients with failure of scram system, only a limited number of transients need to be considered for the PWR plants. Transients such as complete loss of feedwater, loss of offsite power and turbine trip concurrent with failure of scram system result in the fuel heat imbalances and consequential overpressurization of the reactor coolant system.(2, 3, 4 & 5) It is possible that the reactor coolant system boundary may be compromised as a result of overpressurization and in the most unlikely case may result in core melt.

PWR Transient Occurrence Frequency

Complete loss of feedwater, loss of offsite power and to a lesser degree loss of load ATWS events could conservatively lead to a core melt. The Reactor Safety Study only evaluated loss of offsite power and loss of feedwater transients in ATWS events and assigned an occurrence rate of one per year. During the year of 1973, ten PWR plants experienced the severe anticipated transients at the following frequencies.

Complete loss of feedwater	1
Loss of offsite power	3
Loss of load (Turbine Trip)	33

Of these thirty seven occurrences about half of them occurred when the power level was pretty high (75% or higher). This then results in about 19 anticipated transients for ten reactors which may result in severe consequences if the scram system were to fail. It is not clear that turbine trip without scram action would result in unacceptable consequences for all PWR plants and therefore a more realistic number of transients resulting in severe consequences is probably about 10 per year for 10 plants.

Therefore assigning a value of one anticipated transient per reactor year [6, 7] appears to be appropriate. It should be pointed out that the frequency of occurrence of the most serious transient namely the complete loss of feedwater is about 10^{-1} per reactor year based on the 1973 data. But some analyses[5] have shown that loss of load may also yield primary pressure values in excess of the limits[7].

BWR Limiting ATWS Events

BWR systems (assuming recirculation pump trip is installed,) characteristics are such that a turbine trip concurrent with in failure of scram system causes the pool temperatures to reach very high values[8, 9]. Most anticipated transients, by virtue of the BWR plant design, cause turbine trip and therefore most anticipated transients have to be included in this evaluation. Only a few transients such as recirculation pump controller malfunctions concurrent with failure of scram system do not yield severe consequences.[8]

BWR Transient Occurrence Frequency

The reactor safety study assigned an occurrence rate of ten per year per reactor. The 1973 data for eleven BWR plants showed that 79 anticipated transients requiring scram action were experienced. About half of these transients were of little interest whereas the remaining 40 ~~may~~ could have resulted in severe consequences had the scram system not responded to demand. However, it was not clear what the power levels were prior to occurrence of these transients. Based on the PWR experience half the transients were assumed to have occurred at high power levels. Therefore a value of two anticipated transients per reactor year appears to be most appropriate for ATWS evaluation.

Section 3. ATWS EVENTS

3.1 ATWS Consequences

3.1.1 PWR Plants

The reactor safety study group issued a draft report[6] in August 1974 evaluating probabilities of radiological releases as results of various possible accidents. As a part of its study, the report evaluates the hazards presented by a possible occurrence of an ATWS event. The study correctly points our that the most severe anticipated transients without scram are those that cause an imbalance between heat generation and the heat removal capability. Any reduction in heat removal capability results in primary pressure increase and in some cases may exceed 4000 psi.[2, 3, 4] It is pointed out that if the scram system were to fail as a result of a common mode failure in the RPS logic, neither the turbine trip nor auxiliary feedwater pumps would be actuated and thus perhaps resulting in severe damage to the reactor coolant system. Transients that fall into this category are a) a complete loss of feedwater, b) loss of offsite power and to a lower extent c) loss of load. A ~~The most severe~~ path that might lead to severe consequences ~~(possibly even core melt)~~ is described below. (RSS states this path might lead to core melt).

T: Probability of occurrence of a severe anticipated transient per year.

K: Unreliability of the RPS to work on demand

Q: Probability that the valves may fail to fully reclose

E: Containment failure mode

It is assumed that a transient occurs with failure of the RPS to trip the reactor. The resulting imbalance between the heat generation and the heat removal causes primary pressure rise and the primary relief and safety valves open to limit the pressure rise. The safety valves are assumed not to fully reclose resulting in a small LOCA with failure of the RPS system (conservatively assumed to lead to core melt).

The probability of the occurrence of this sequence of events is developed as follows:

IMAGE EVALUATION
TEST TARGET (MT-3)

6″

The frequency of transient occurrence was discussed in Section 1. The value chosen for the unreliability of the RPS has been discussed in Section 2. Since ATWS events cause high primary pressures and discharge solid water through the pressurizer safety valves, consequently the valves are exposed to high reaction forces and conditions for which on test data are available. Opinions vary as to the likelihood of these valves to fail to reclose ranging from valves certainly failing to reclose fully to valves possibly failing to reclose. Therefore, a value of 10^{-1} is assigned to the probability of these valves failing to reclose fully with perhaps an uncertainty factor of 10. The probability of containment failure mode is essentially the same as that used by the safety study group. Therefore if a very conservative assumption is made that this sequence of events may result in core melt, the probability associated with this event tree is about 10^{-5}/reactor year as shown below

$T \approx 1/yr$

$K \approx 10^{-4}/\text{damand}$

$Q \approx 10^{-1}$

$E \approx 1.0$

Therefore $TKQE \approx 10^{-5}$/year and not 10^{-6} as indicated in the RSS.

3.1.2 BWR Plants

The reactor safety study assigns a failure probability of 4×10^{-7} per demand to the reactor shutdown mechanism (C). This value is arrived at as follows.

$$C = \text{RPS failure} + \text{Failure of recirculation pumps to trip and manual insertion of poison}$$

$$C = \left(1.3 \times 10^{-5}\right) \times \left(3 \times 10^{-2}\right)$$

$$= 4 \times 10^{-7}$$

The study assumes that the availability of the recirculation pump trip and manual actuation of the poison system such that the reactor is made subcritical in thirty eight minutes is a successful mode. However, studies [8, 9] have shown that extremely high suppression pool temperatures will be attained even with recirculation pump trip and manual actuation of the standby liquid control system. The effectiveness of the pool decreases as the pool temperature increases and in particular around 180°F the water vapor pressure increases significantly causing a highly reduced pool functionability. Further the data available[10] have shown that the vibratory oscillations occur around 180°F and may well result in compromising the integrity of the suppression pool. ~~Therefore~~ a value of 10^{-1} failure probability is assigned to this failure mode. Thus the failure of the suppression pool may well result in core melt. This sequence

of events is further developed below:

T: Probability of occurrence of anticipated transient per year

C: Unreliability of the RPS per demand

P: Probability of containment failure as a result of high pool temperatures

T = 2/yr (see section 1)

C = 10^{-4}/demand (see section 2)

P = 10^{-1}

Therefore, an estimate of core melt as a result of this sequence of events is about 2×10^{-5} and not 10^{-6} as indicated by the RSS

There is yet another sequence of events that might lead to a core melt that was apparently not considered by the reactor safety study. This sequence is discussed below

a) Assume the recirculation pumps trip and the poison system is actuated manually such that the reactor is made subcritical in about 35 minutes.

b) If the HPCI system fails, it is likely that the vessel level may not be maintained and core melt may occur. The probability of HPCI failure (U') is 10^{-1}. Therefore TC'U' $\approx 2 \times 10^{-5}$.

From these sequences it appears that the probability of core melt as a result of an ATWS event is about 10^{-5} and not 10^{-6} as noted in the safety study.

ATWS Limits

The light water reactor vendors have published detailed evaluations of the ATWS events.[1, 2, 3, 4, 5, 8, 9] These analysis have shown that if no plant modifications are made, an ATWS event may well result in exceeding WASH-1270[6] criteria. In particular the reactor coolant system pressures may well exceed 4000 psig and the reactor coolant boundary pressure may well be compromised. Just as the Reactor Safety Study made a conservative assumption that core melt occurs if the clad temperature exceeds 2300°F, it is our ~~opinion~~ that if WASH-1270 limits are exceeded core melt may occur with a probability in the range of 10^{-4} to 10^{-6} per reactor year. Admittedly these

evaluations have uncertainties associated with them but it seems to us that a responsible position would be to apply conservative limits to minimize the risk of any errors in the evaluations.

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12/21/76

Denny
Thought you'd be interested
in this little writeup
Jim Cermak

Some rather amusing observations, in which you may be interested, are obvious when the major contributors to risk (per WASH-1400) are listed in order of importance.

The WASH-1400 Final Report Tables 5-1 and 5-2 indicate the relative importance of the various accident sequences studied. Table 1 attached summarizes these results in terms of the percent contribution to the total core meltdown probability. The Table presents the results in a cross-matrix of initiators vs. subsequent failures.

Table 2 lists the public risks which we inferred from WASH-1400 data. The results are weighted in terms of both probability and fractional iodine release to the environment. This consequently is an approximate measure of the percent public risk contribution of the various sequences.

Comparison between Tables shows a shift in the ranking of controlling sequences reflecting that different meltdown scenarios have different consequences.

It is interesting to note that if there were no large break ECCS installed - the number for the probability of large LOCA and ECCS unavailable (meltdown) would increase by a factor of 30. The total meltdown probability (all events) would increase by a factor of 3. The public risk would increase by approximate 3% -- an apparently

negligible amount considering the cost savings achievable by deleting a large LOCA as a design basis event.

It is further presumed that if no large break ECCS were installed, there would be no cold leg connections to a low pressure ECCS outside the containment and hence - the check valve interfacing LOCA event would be absent. Since this event is dominant in terms of public risk (estimated at 53%), it can only be concluded from WASH-1400 that the addition of large LOCA ECCS for Surry has actually resulted in a factor of 2 increase in public risk.

This is not to suggest that deleting the large LOCA low head injection system would reduce risk; but to illustrate that in some cases the addition of safeguards can in fact result in increased public risk if the risk from new failure modes exceeds the risk averted. It is noted that the impact of the check valve interfacing LOCA can be greatly reduced by simple measures such as testing and a third check valve.

The results on Table 2 also indicate to us the total lack of impact of WASH-1400 on the licensing and regulatory process. If WASH-1400 is to be believed, attention should be devoted to the major contributors to risk; double check valve failure; total loss of AC power; small LOCA; and loss of main and auxiliary feedwater. Instead, large LOCA, ATWT, and upgrading qualification requirements of safety-related equipment have dominated the licensing scene.

Table 1 - Probability Weighted Contributions to Meltdown

Initiator \ Subsequent Failure	None	ECCS Inj/Recirc	Cont. Sys. Spray/Heat Removal	Elec Power	AFW	RPS	Total
Large LOCA		6%	0.02%	0.002%			6 %
Small LOCA		54%	4%	0.03%			58 %
Vessel Rupture	0.2%						0.2%
Check Valves Fail	8%						8%
Loss of Offsite Power					6.7%		6.7%
Loss of Feedwater					12 %	8%	20%
Total	8.2%	60%	4%	0.03%	18.7%	8%	

Table 2 - Probability and Consequence Weighted Contributions To Meltdown

Initiator \ Subsequent Failure	None	ECCS Inj/Recirc	Cont. Sys. Spray/Heat Removal	Elec Power	AFW	RPS	Total
Large LOCA		0.116 %	0.075%				0.2%
Small LOCA		0.815%	8.4%				9.2%
Vessel Rupture	0.004%						0.004%
Check Valves Fail	53.2%						53.2%
Loss of Offsite Power					37%		37 %
Loss of Feedwater					0.27%	0.18%	0.45%
Total	53.2%	0.93%	8.5%		37.3%	0.18%	

2

DOMINANT PWR CORE MELT PROBABILITIES

Ref: WASH-1400, Table V-3-14

3.6 $\times 10^{-5}$

Large LOCA ≈ 3.4×10^{-6}

Largest contributors

ADE - 2×10^{-6} (failure of ECC injection system)

AHE - 1×10^{-6} (failure of ECC recirc. system)

Small LOCA ≈ 6.9×10^{-6}

Largest contributors

S_1D - 3×10^{-6} (failure of ECC injection system)

S_1H - 3×10^{-6} (failure of ECC recirc. system)

Small Small LOCA ≈ 2.6×10^{-5}

Largest contributors

S_2D - 9.1×10^{-6} (failure of ECC injection system)

S_2H - 6×10^{-6} (failure of ECC recirc. system)

S_2C - 2×10^{-6} (failure of containment spray injection system)

Transients ≈ 1.6×10^{-5}

Largest contributors

TML - 6×10^{-6} (failure of secondary system relief valves & auxiliary feedwater system and power conversion system)

TMLB' - 3×10^{-6} (failure of secondary system relief valves and auxiliary feedwater system and power conversion system and failure to recover either onsite or offsite ele. power within about 1 to 3 hours following an initiating transient which is a loss of offsite AC power)

ATWS (total 4×10^{-6})

TKQ - 3×10^{-6} (RPS failure and primary system relief valves to reclose after opening)

TKMQ - 1×10^{-6} (RPS failure and primary system relief valves to reclose after opening and failure of the power conversion system)

Other $\approx 4.8 \times 10^{-6}$

Largest contributor

V - 4.4×10^{-6} (interfacing systems LOCA (check valve))

Conclusions

PWR's

For PWR's transient events represent about 28% of the total core melt probability with ATWS events representing 25% of the transient events. At first glance, it appears that significantly reducing the probability of core melt due to ATWS events will not have much of an impact on the overall core melt probability. However, improvements made to mitigate ATWS would reduce the probability of core melt from other transient events. If all of the status report fixes were made the best possible improvement in core melt probability would be a drop from 5.7×10^{-5} to 4.1×10^{-5}.

DOMINANT BWR CORE MELT PROBABILITIES

Ref: WASH-1400, Table V-3-16

Large LOCA ≈ 2.8×10^{-7}

Largest contributor

AE - 1.4×10^{-7} (failure of ECC cooling injection)

$\approx 1.2 \times 10^{-6}$

Small LOCA ≈ 3.2×10^{-7}

Largest contributor

S_1E - 1.5×10^{-7} (failure of ECC cooling injection)

Small Small LOCA ≈ 5.6×10^{-7}

Largest contributors

S_2I - 1.1×10^{-7} (failure of low pressure recirc. system)

S_2HI - 1.1×10^{-7} (failure of low pressure recirc. system/core spray recirc. system)

S_2J - 1×10^{-7} (failure of high pressure service water system)

Transients ≈ 2.[illegible] x 10^{-5}

Largest contributors

ATWS: ≈ TC-γ 1×10^{-5} (RPS failure)

TW 1.5×10^{-5} (failure to remove residual core heat (RHR))

Conclusions

BWR's

The probability of ~~risk of~~ core melt is ~~as~~ dominated by transient events - (ATWS events make up 35% of the transient event total) ~~which 35% of the risk is make up of ATWS.~~ The core melt probability from the combination of all LOCA's (large and small) represent 12% of that for ATWS, 4% of .at for all transient events.

"Status report fixes" short of fixing RHR could improve core melt probability from 3×10^{-5} to 2×10^{-5}. However, by also improving RHR, the core melt probability from all events would be lowered by an order of magnitude.

The ATWS status report fixes and the RHR fix do not appear to have an effect upon the LOCA sequences which leads to core melt.

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ISSUE 8: Use of Probabilistic Assessment of Reliability

Introduction

This issue was identified in a meeting of the Electrical, Instrumentation and Control Systems Branch held on September 10, 1976. In the attachment to the November 3, 1976 memorandum from the Director, NRR to the NRR Staff it was listed as Issue #8 and defined as follows:

"The development of probabilistic evaluation techniques has led the industry to propose designs based on probabilistic assessments of the reliability of and need for certain systems. Numerical reliability goals and methods of analysis have not yet been established by the NRC, which continues to rely on the single failure criterion and other design rules pending further development of the methods."

A meeting of all members of the Electrical, Instrumentation and Control Systems Branch was held on November 12, 1976 to discuss, clarify and redefine this issue as necessary in order to aid in developing a staff response. As a result, the issue was redefined by one or more concerned members of the Branch as follows:

a. "The development of probabilistic evaluation techniques has led the industry to propose designs based on probabilistic assessments of the reliability of analysis have not yet been established by the NRC which continues to rely on the single failure criterion and other design rules. This lack of acceptable reliability goals and methods has led to marked inconsistencies in the conduct of reviews by different branches. Some branches have accepted the goals, methods, component failure data and results while others continue to rely on the single failure criterion pending further development of the new methods."

b. "WASH-1270 goals are not attainable for present day nuclear power plants (see letter from H. C. Kouts to B. C. Rusche dated April 6, 1976). The Reactor Shutdown System of the Clinch River Breeder Reactor Plant (CRBRP) is purported to be designed to meet WASH-1270 goals by taking a Class A design approach. Considering the fact that the CRBRP design of the Reactor Shutdown System is the basis for excluding Core Disruptive Accidents (CDAs) from consideration as Design Basis Accidents (DBAs) such exclusion was not made on a sound technical basis. Resolution of this matter was suggested but not materialized (see letter from R. L. Tedesco to R. P. Denise dated August 17, 1976)."

The following sections will address each issue separately.

Summary of Issue

a. The industry is using probability techniques to support proposed designs of safety related equipment. The NRC has not established numerical reliability goals or analysis methods to be used by the staff. Inconsistencies exist between branches in the conduct of safety reviews.

b. The technical concern raised on the Clinch River reactor shutdown system is that a decision had been made to permit credit for system reliability to a degree which excludes core melt accidents from the design basis accident spectrum; because of expressed opinions that the required reliability cannot be attained, this decision is not technically sound.

Summary Response

a. The prime bases for licensing evaluations and decisions are the General Design Criteria, related guidance, and the engineering

judgment of the staff. In some cases engineering decisions are made using reliability information as a factor in safety evaluations. Within the total context of WASH-1270, including recognition that the stated reliability goal was not demonstrable, the staff believes that there is no conflict between the engineering implementation prescribed in WASH-1270 for reliable shutdown systems and the staff implementation on CRBRP. The staff also believes that the philosophical and technical basis for the decisions made thus far are sound and well known, and have been well ventilated. The staff is not relying on reliability goals or analyses to preclude core melt accidents from the design basis accident spectrum. ?

revised with responses to Part b.

Detailed Discussion

a. The prime baseline requirements for licensing nuclear power facilities are set forth in the General Design Criteria. These requirements are interpreted and implemented through Regulatory Guides, industry standards, and numerous other documents containing guidance as presented in the Standard Review Plans.

Statistical and reliability methods, in general, have not been established by the NRC as the means for assessing safety related equipment designs and processes proposed by the industry. In the vast majority of the engineering evaluations conducted by the NRC the bases for acceptance or rejection stem from the General Design Criteria. Our regulations require that the consequences of accidents be successfully

may not be completely applicable to ATWS

Since vendors do have an option to demonstrate that mitigating [illegible] have high reliability

mitigated despite the occurrence of a series of independent events. The intent of this approach is to provide a high degree of assurance that postulated accidents can be successfully mitigated. In those cases where an event could both initiate an accident and possibly cause a failure of a safety system or subsystem, additional design features are required to prevent such an occurrence.

not for ATWS
Where data are not available conservative estimates of reliability of systems are made

In some instances, statistical and reliability methods have been used to aid our engineering judgment. This has been done on those problems where well simulated experimental data or actual plant operating data were available from which applicable statistical information could be derived. In some instances, such as ATWS, conservative estimates of system avail. were obtained using statistical techniques. Examples are listed in Table 1. We will continue to endorse the use of such information in guiding and assisting our engineering decisions. Since the information available varies with the technology, the application of risk assessment varies, and to that degree could be termed inconsistent.

Probabilistic criteria have been used to define the severity of certain external hazards involving nearby industrial, military and transportation facilities against which nuclear power plants should be designed. In these cases considerable data bearing directly on a specific issue has (were) been available or well established mathematical models have been used with considerable conservatisms. This approach is intended to provide assurance that a consistent level of protection is provided against external hazards which might result in both initiating an accident and disabling the equipment provided to cope with that accident.

In summary, the prime bases for licensing evaluations and decisions are the General Design Criteria and related guidance, coupled with the engineering judgment of the staff. This engineering judgment can be based on many factors, including any available information with respect to the reliability of systems and components or the probability of occurrence of certain events.

This belongs with Part b

(3) A contrasting, and less conservative, viewpoint on reliability is discussed further in Section b, on Clinch River; however, to the extent it may be considered applicable to LWRs, it is discussed below:

(1) Our goal
(2) Engg judgement to meet goal
(3) WASH-1400 Risk
(4) Why WASH-1400 risks can't be blindly applied
(5) Rationale for CRBRP to meet 10^{-7}

i) From WASH-1400, the LWR calculations (using Peach Bottom, Units 2 and and Surry as base plants) show that the likelihood of exceeding 10 CFR 100 is of the order of 10^{-5} per year; any requirement to protect to a level of 10^{-7} per year could be construed as unnecessary or not needed; and,

ii) The probability of death from 100 LWRs is only 10^{-4} of all other societal risks, and a factor of 100 lessthan the next smallest contributor.

(4) While these conclusions may be acceptable from a research viewpoint, other important considerations include:

1. our licensing philosophy for requiring design basis events, irrespective of an initiating mechanism is an important part of and played a major role in reducing risk to the level calculated to be achieved in WASH-1400.

2. our regulatory policies have continuously evolved since design and construction of Peach Bottom Units 2 and 3 and Surry (the base plants of WASH-1400). We consider it inappropriate to base regulatory decisions only on the calculated risks for those two plants Therefore, we would not conclude that a plant needs to be designed only to the risk levels calculated to be achieved by those two plants.

3. as noted earlier, more work is needed before risk assessment methodology can be used routinely in licensing decisions. For

example, we need goals on overall acceptable risk, as well as allocation goals of risk fraction per individual contributor (such as ATWS).

Table 1

EXAMPLES WHERE PROBABILISTIC CONSIDERATIONS
WERE USED AS AN AID IN REACHING DECISIONS

1. We decided not to backfit rod sequence control on certain GE plants on the basis of very low combined probability of six low-probability independent events which had to occur in succession.
2. We decided that for ATWS the probability of exceeding our guidelines should be on the order of 10^{-7} per reactor year.
3. We concluded that the likelihood of a catastrophic reactor vessel failure was sufficiently low (10^{-6}-10^{-7} per service year) that ECCS would not have to be designed to cope with vessel failure.
4. We included in our Standard Review Plan Section 2.2.3 a probabilistic target for extreme man-made events.
5. We used some techniques of the RSS to study the marginal effect of ECCS equipment outage times (for test and maintenance) on the overall availability of ECCS to perform its function. Based on this we made some adjustments in the PWR STS.
6. The probability of a seismically-induced fire was estimated and found to be small compared to the overall probability of fire. This perspective was one factor considered for the draft Regulatory Guide on fire protection which requires most fire protection systems to be designed as seismic Class II rather than Class I.
7. As part of the work leading to the draft Regulatory Guide on pre-operational and surveillance testing of diesel-generators, alternative sequential testing schemes were evaluated. Diesel generators are also required to have high probability of successful starts and acceptance of load in the

Table 1 Continued

predicted time span.

8. On the Big Rock Point Nuclear Power Station, reliability methods were utilized as part of the basis for recommending an exception from the ECCS acceptance criteria (10 CFR 50.46), thereby not requiring this licensee to backfit a redundant emergency diesel generator for this facility.

9. The Auxiliary Feedwater Systems, used in PWRs, are required to provide diverse power sources based on the unreliability estimates of the RSS.

10. In the calculation of radioactive releases following a postulated LOCA, some aspects are treated on the basis of probabilistic consideration (permissible time for purge system operations are treated probabilistically).

11. Turbine missiles generated as a result of destructive overspeed subsequently hitting vital equipment such as primary piping or containment are treated probabilistically to assure that the probability of exceeding 10 CFR 100 doses are acdeptably low (10^{-6}-10^{-7} per year).

Appendix 8.1

Single Failure Discussion

Single Failure

Appendix A to 10 CFR Part 50 contains requirements for design to preclude loss of system functions due to single failures. There are about ten General Design Criteria which make specific reference to a requirement that the safety function be accomplished even in the presence of a single failure that is assumed to occur. Appendix A to part 50 contains a definition for single failures to guide the designer.

> "Single Failure. A single failure means an occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electric systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly) nor (2) a single failure of a passive component (assuming active components function properly), results in a loss of the capability of the system to perform its safety function.

Single failures of passive components in electric systems should be assumed in designing against a single failure. The conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are under development."

In using the guidelines provided in Appendix A it seems straightforward to arrive . judgment that says a system that is required to meet the single failure criteric will have to have redundancy of components. If pumps are r-quired, then at least two will be needed; if isolation valves are required, then at least two in each line will be required; there will need to be at least two independent sources of motive power - independent vital buses; and if a credible failure of one component can induce the failure of another redundant component, such as through flooding of compartments, then suitable protection must be provided.

An example of a specific area which has received considerable attention is the question of spurious action of valves in meeting single failure criteria. This topic was addressed in a recent ANSI Standard ANSI-N658, which included in the context of "active" failure the following definition.

> "Spurious action of a powered component originating within its actuation or control system shall be regarded as an active failure unless specific design features or operating restrictions preclude such spurious action."

We have not accepted this definition completely but have eliminated spurious actions as a design deficiency in some cases by application of the EI&CS Branch Technical Position Number 18. The most commonly used solution is to provide a lockout feature in the valve control system so that spurious action is not possible. For example, this approach is suited for designs such as isolation valves in the accumulator lines.

It might be argued that the single failure requirements in the General Design Criteria do not consider probabilities of failure. This is not generally true. The "active-passive" distinction is one way in which failure likelihood is already taken into account. During an actual design process, when a failure modes and effects analysis should be done, decisions must be made regarding the disposition or resolution of each single failure. These decisions require knowledge of the likelihood of failure of an item.

The Use of Failure Rate Estimates

Estimated failure rates for electrical gear from failures per demand for "active components" varies from $3x10^{-2}$ for the start of a diesel plant, to $1x10^{-4}$ for failure of a relay to energize. Failure rates of "passive components" are substantially less.

In some instances alternate methods may be used to cope with single failures. Examples include the motor-operated valves and the check valves in ECCS accumulator lines, not to mention the accumulators. To design away these single failure "points" would require redundant lines and valves from each accumulator to the primary pressure boundary. This redundancy would introduce complexities in fabrication, testing, hydraulic, and control. A preferred position is the electrical power lockout fea-ture for "active" valves and which includes acceptance of the rather small likelihood of failure of check valves and piping, which are relatively high reliability "passive" components. This and our testing and inspection requirements provide substantial assurance that these accepted "single failure points" are sufficiently unlikely to result in loss of safety function.

Thadani

The principal ATWT events of WCAP 8330 have been re-analyzed in order to reflect the following assumptions:

1. Moderator temperature coefficient valid for 99 percent of core life (-7 pcm/°F)
2. 10 percent pressure accumulation on the pressurizer safety valve setpoints when discharging water (normally, 3 percent accumulation is applied)
3. Failure of the turbine-driven auxiliary feedwater pump causing half of the total auxiliary feedwater capacity flowrate to be unavailable
4. Failure of a high head safety injection pump
5. Failure of a pressurizer safety valve to re-seat properly

The results of these ATWT re-analyses show the effects of the imposition of certai equipment failures, and a higher pressure accumulation on the pressurizer safety valves during water relief, in addition to a more positive moderator temperature coefficient. The reference case of each of the ATWT transients, listed below, is based upon the 99 percent moderator temperature coefficient.

1. Loss of feedwater
2. Loss of load
3. Station blackout
4. Rod withdrawal at power with turbine trip
5. Rod withdrawal from subcritical
6. Primary system depressurization

Each of these selected ATWT events results in high primary system pressure or large discharge of reactor coolant into the containment, or both. The homogeneous equilibrium subcooled water relief was assumed to be a function

of pressurizer pressure with a constant water enthalpy at the initial value.

1. <u>Loss of feedwater</u>

The loss of main feedwater ATWT generates the highest reactor coolant system pressures. The highest of these pressures occur at the discharges of the reactor coolant pumps 113 seconds into the transient, shortly after the pressurizer fills. The peak pressure, 2842 psia, is dependent upon the ra homogeneous equilibrium subcooled water relief through the pressurizer relief and safety valves. If the safety valves do not fully open to discharge the water until the pressure accumulates to 10 percent over the set pressure, then the peak pressure attained is higher, 2882 psia. The peak pressure is also dependent upon the heat-up and swell of the reactor coolant. Impairment of the heat sink leads to a greater coolant swell and higher peak pressure. If half of the auxiliary feedwater flow is not available due to a pump failure, then the peak pressure attained in the reactor coolant system is 3017 psia, at 117 sec. These pressures are higher than those reported in WCAP-8330, due mainly to the more positive moderator temperature coefficient and other conservative assumptions; but they are still within emergency stress limits.

In the long term, the plant may be shut down by one of the boration systems, such as the safety injection system, following an ATWT occurrence. Failure of a high head safety injection pump reduces the boron insertion rate; but does not significantly degrade the effectiveness of the safety injection system. A comparision of the reactivity transients in figures 5 and 14 demonstrates that operator-actuated safety injection can easily shut the plant down, even if a safety injection pump is not available. Failure of a pressurizer safety

valve to reseat properly, once opened during a loss of feedwater ATWT, actually aids boron insertion by depressurizing the reactor coolant system, allowing more borated safety injection water to enter the primary system.

2. Loss of Load

Generally, the loss of load is similar to the loss of feedwater ATWT; but less severe. The main difference is that the turbine is tripped immediately (the loss of load) rather than later in the transient when a steam generator low level signal is generated. Therefore, only a reference case and a case with 10 percent pressure accumulation were analyzed. The peak pressures of both transients are lower than the corresponding loss of feedwater transients.

3. Station Blackout ATWS

The station blackout ATWT generates low DNB ratios and high primary system pressures. It was shown, in an earlier analysis, that the 99 percent moderator temperature coefficient does not result in unacceptably low DNB ratios. Furthermore, any assumptions that produce higher primary system pressures would tend to also raise the DNB ratios. A failure in the auxiliary feedwater system does not affect the DNB ratio because the minimum DNB ratio occurs long before the auxiliary feedwater system delivers water to the steam generator. The station blackout cases presented here verify that primary system pressures are not excessive. In fact, the pressurizer safety valves pass only steam at the time of peak pressure. This means that the assumption of 10 percent pressure accumulation for water relief has no effect on the peak pressure. Table 1

and figures 28, 33 and 37 show that peak pressures are unchanged and very little coolant is discharged into the containment.

4. Rod Withdrawal at Power with Turbine Trip

The rod withdrawal at power with turbine trip produced the highest reactor coolant pressure of all the rod withdrawal at power cases presented in WCAP 8330. The peak pressure in Table 1 and in Figure 40 is higher than that of WCAP 8330 due to the more positive moderator temperature coefficient. The pressurizer does not fill during this ATWT, so 10 percent pressure accumulation for water relief is not applicable.

The rod withdrawal at 50 and 25 percent power were based upon the moderator temperature coefficients consistent at those powers and the full power moderator temperature coefficient of -7 pcm/°F. That is, the moderator temperature coefficients at these lower powers were more positive than -7 pcm/°F. These cases resulted in reasonable peak pressures and very low steam releases from the pressurizer.

5. Rod Withdrawal from Subcritical

The rod withdrawal from subcritical ATWT is based upon a zero power moderator temperature coefficient of 0 pcm/°F. The pressurizer fills at 317 seconds and a peak pressure of 2683 psia is attained 23 seconds later.

6. RCS Depressurization

This ATWT produces low DNB ratios; but like the blackout ATWT, the DNB ratios resulting from the -7 pcm/°F moderator temperature coefficient have been shown to be satisfactory. The main concern then becomes the reactor coolant release into the containment, and the effectiveness of the safety injection system in shutting down the plant and replacing the mass.

Mass and Energy Releases from the Reactor Coolant System

A plot of the total mass released into the containment during the first ten minutes of the loss of main feedwater ATWT is given on figure 66. By approximately 250 seconds, the pressurizer pressure has decreased to below the pressurizer relief valve setpoint, causing all valves to close and terminate coolant discharge. This is evident on the plot, as the total mass release curve levels off to a constant value.

Superimposed on this plot is another plot representative of the same ATWT event; but with one pressurizer safety valve stuck in the fully open position. This is the same assumption as in the RCS depressurization ATWT. Then the stuck safety valve during the loss of feedwater ATWT constitutes a depressurization ATWT on top of a loss of feedwater. Mass is discharged at higher temperature and pressure than the simple depressurization ATWT, and therefore envelops the most conservative amount of mass/energy release attributable to the depressurization ATWT.

Figure 66 is a good basis for the calculation of energy releases. The total energy released is merely the mass shown in figure 66 multiplied by the saturated water or steam enthalpy at 2250 psia. The time at which water relief begins is marked on the curve, "pzr fills."

Table 1

	Presented in figures:	Time Prz. Fills (Sec)	Peak Press at Pump Discharge (psia)	(sec)
1. Loss of main feedwater	1-7	82	2842	113
with 10% pressure accumulation	8-11	82	2882	109
with SI failure	12-14	82	2842	113
with aux. FW failure	19-21	84	3017	117
with safety valve failure	15-18	82	2842	113
2. Loss of load	22-24	89	2833	121
with 10% pressure accumulation	25-27	89	2874	114
3. Station Blackout	28-32	28	2593	13
with 10% pressure accumulation	33-36	28	2593	13
with aux FW failure	37-39	28	2593	13
4. Rod Withdrawal at Power; turbine trip	40-44	--	2658	25
with safety valve failure	45-48	52	2658	25
at 50% power	49-51	--	2425	79
at 25% power	52-54	--	2422	489
5. Rod Withdrawal from subcritical	55-58	317	2683	340
6. RCS depressurization	59-62	126	--	--
with SI failure	63-65	126	--	--

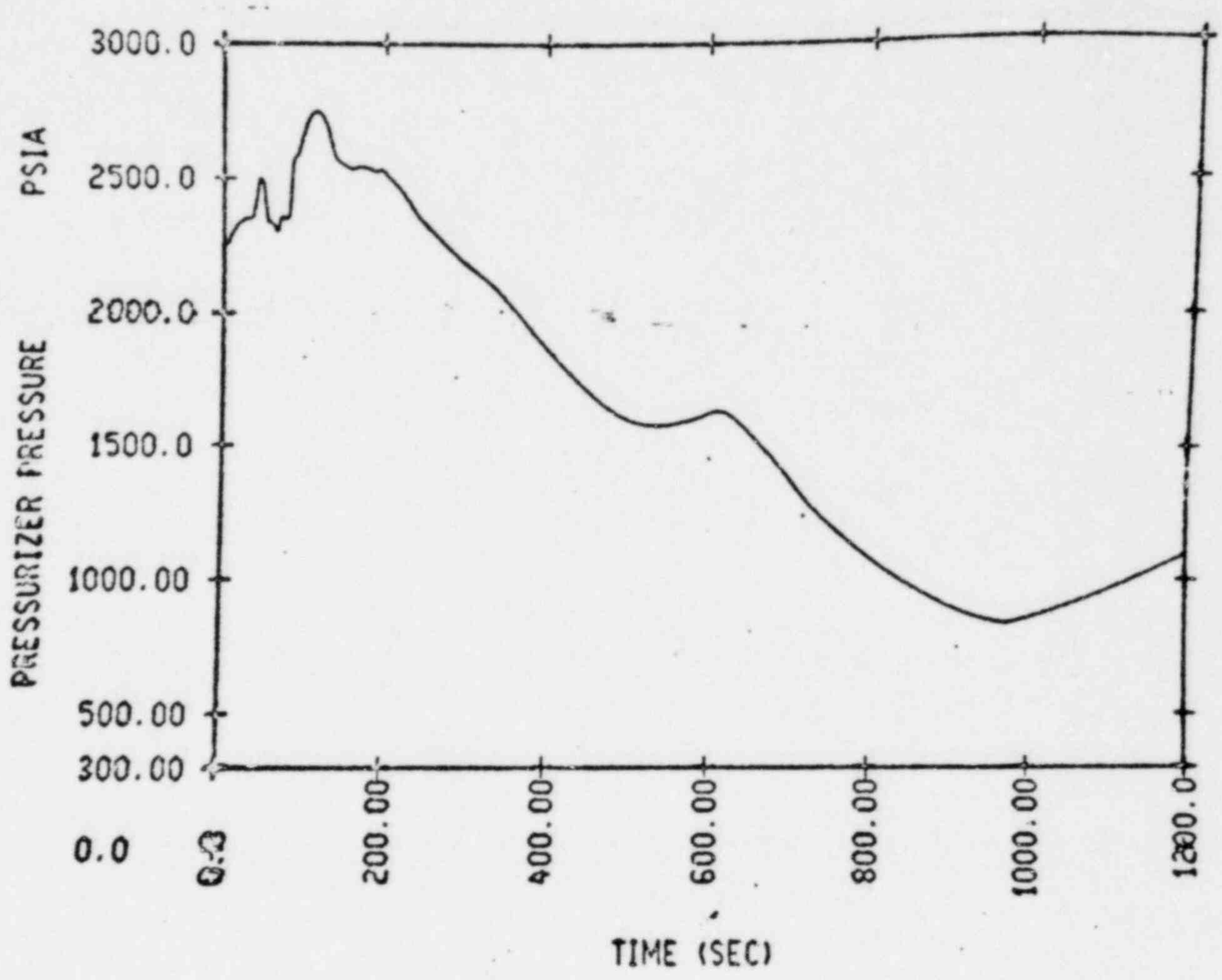


Figure 1. Loss of Feedwater ATWS

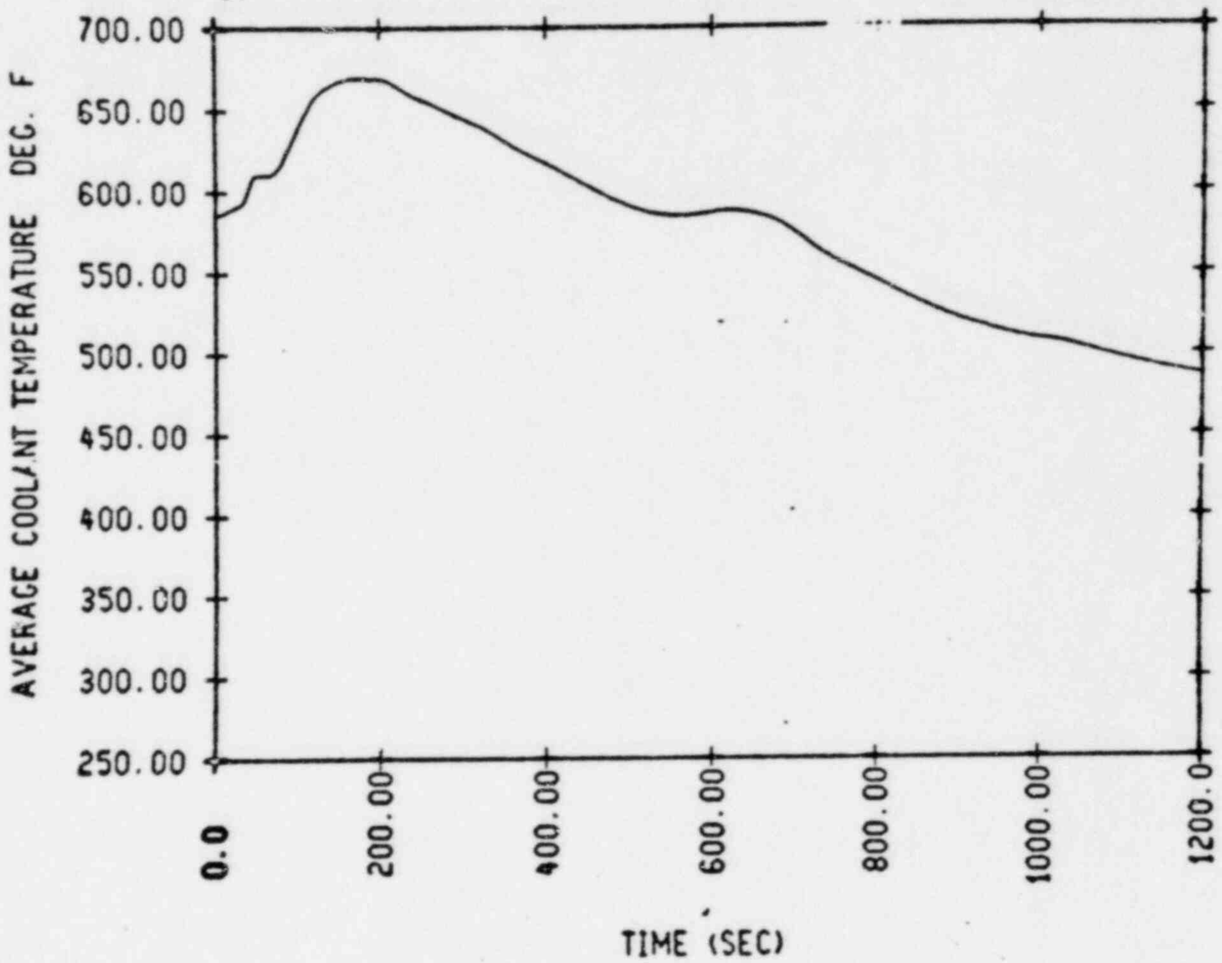


Figure 2. Loss of Feedwater ATWS

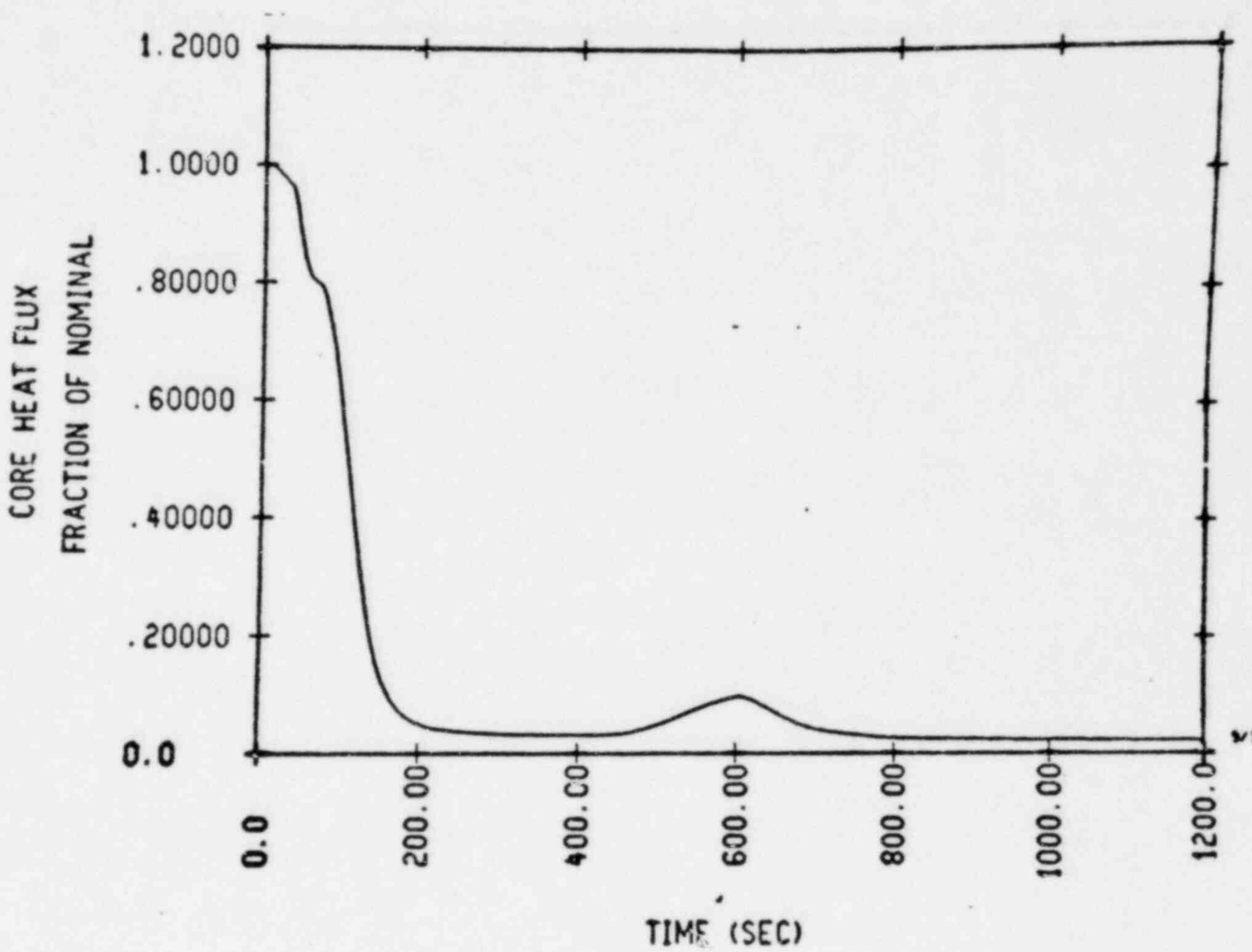


Figure 3. Loss of Feedwater ATWS

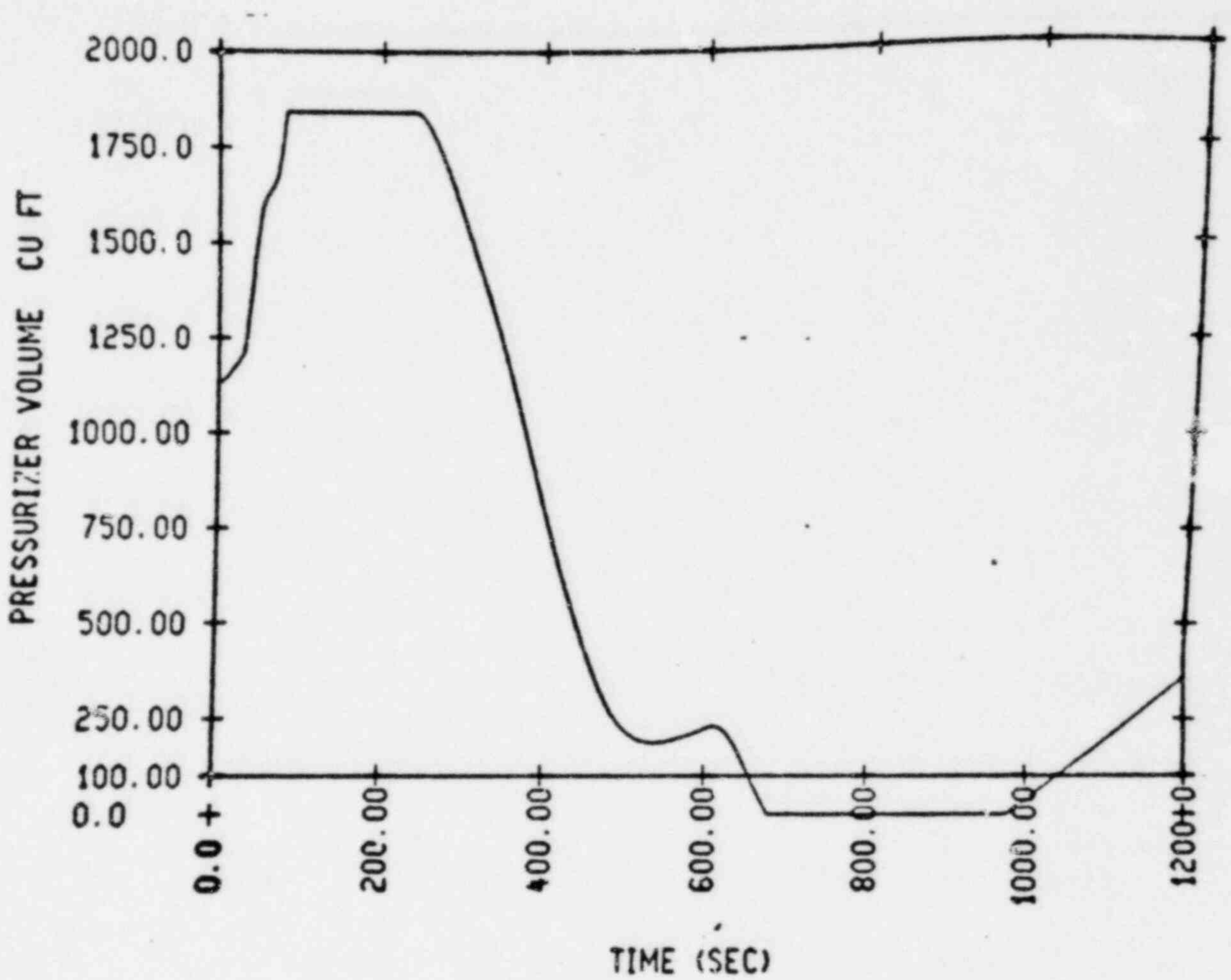


Figure 4. Loss of Feedwater ATWS

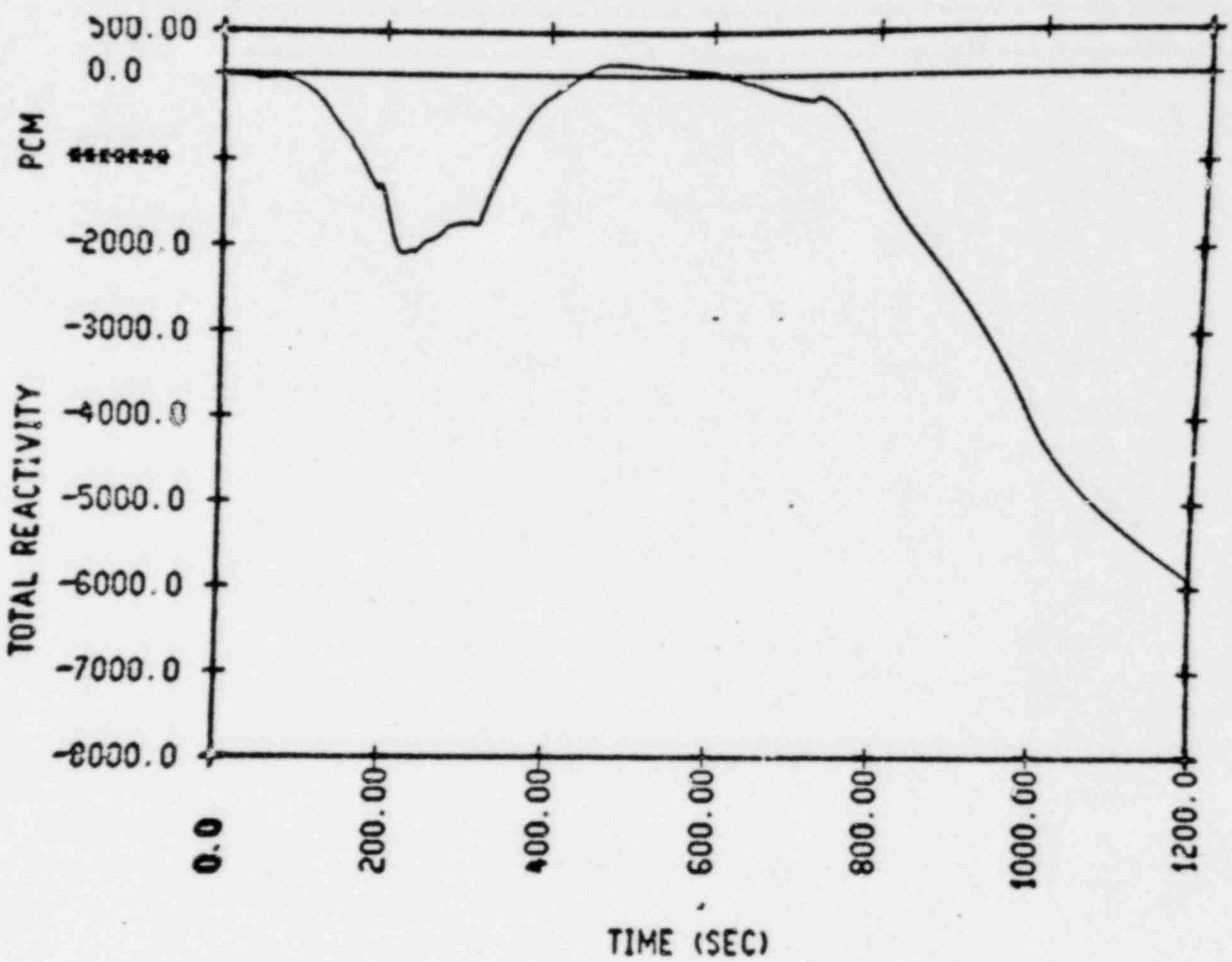


Figure 5. Loss of Feedwater ATWS

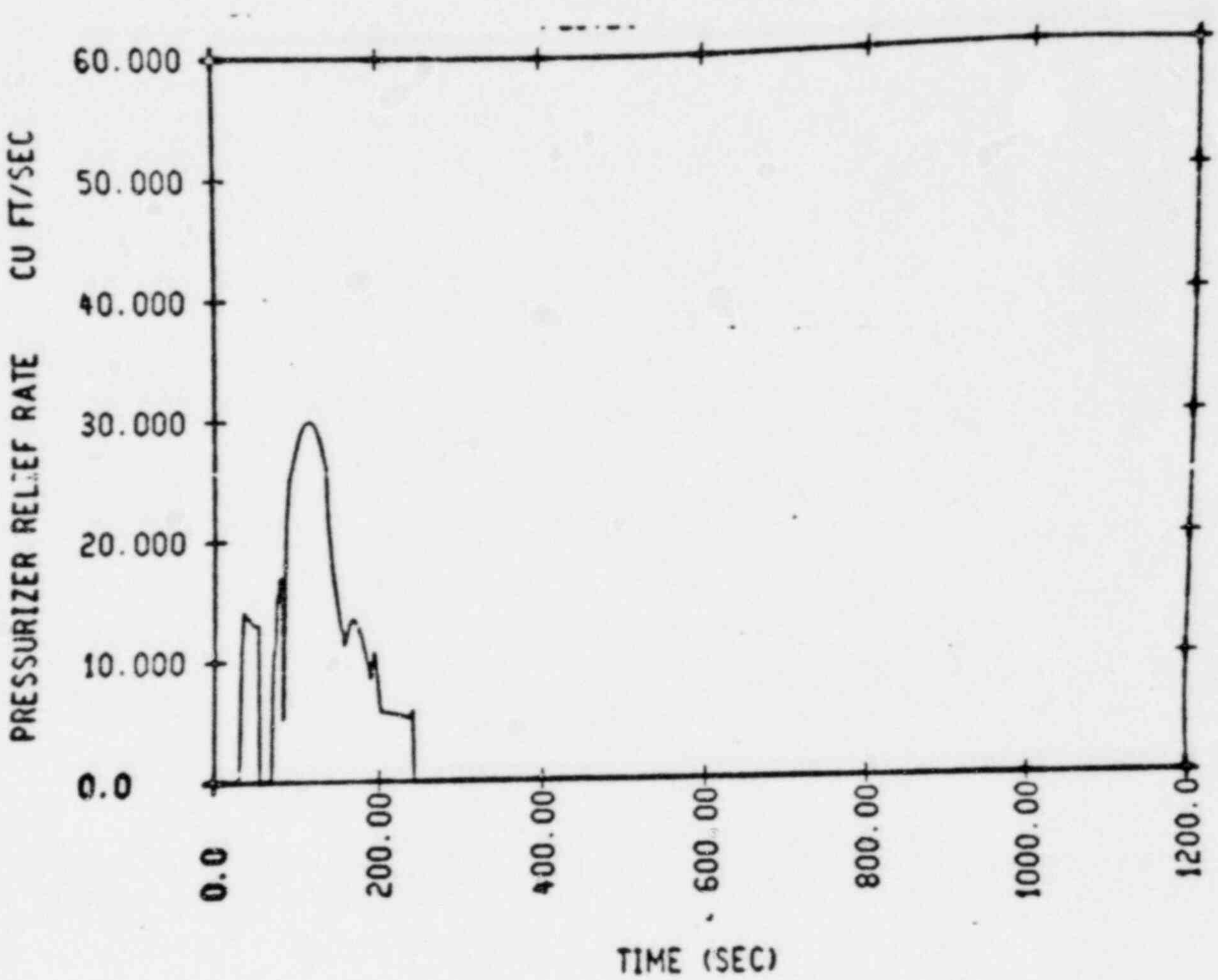


Figure 6. Loss of Feedwater ATWS

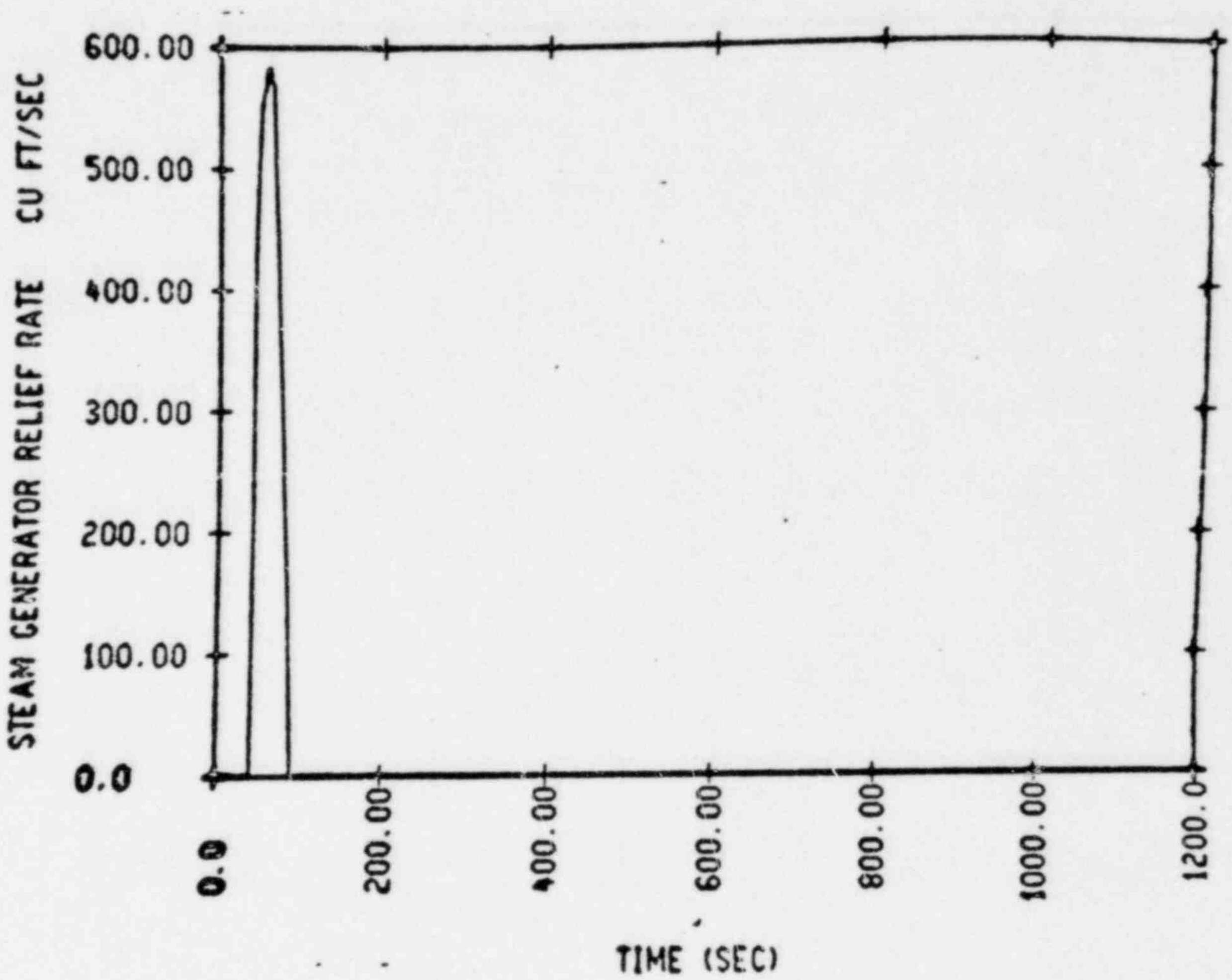


Figure 7. Loss of Feedwater ATWS

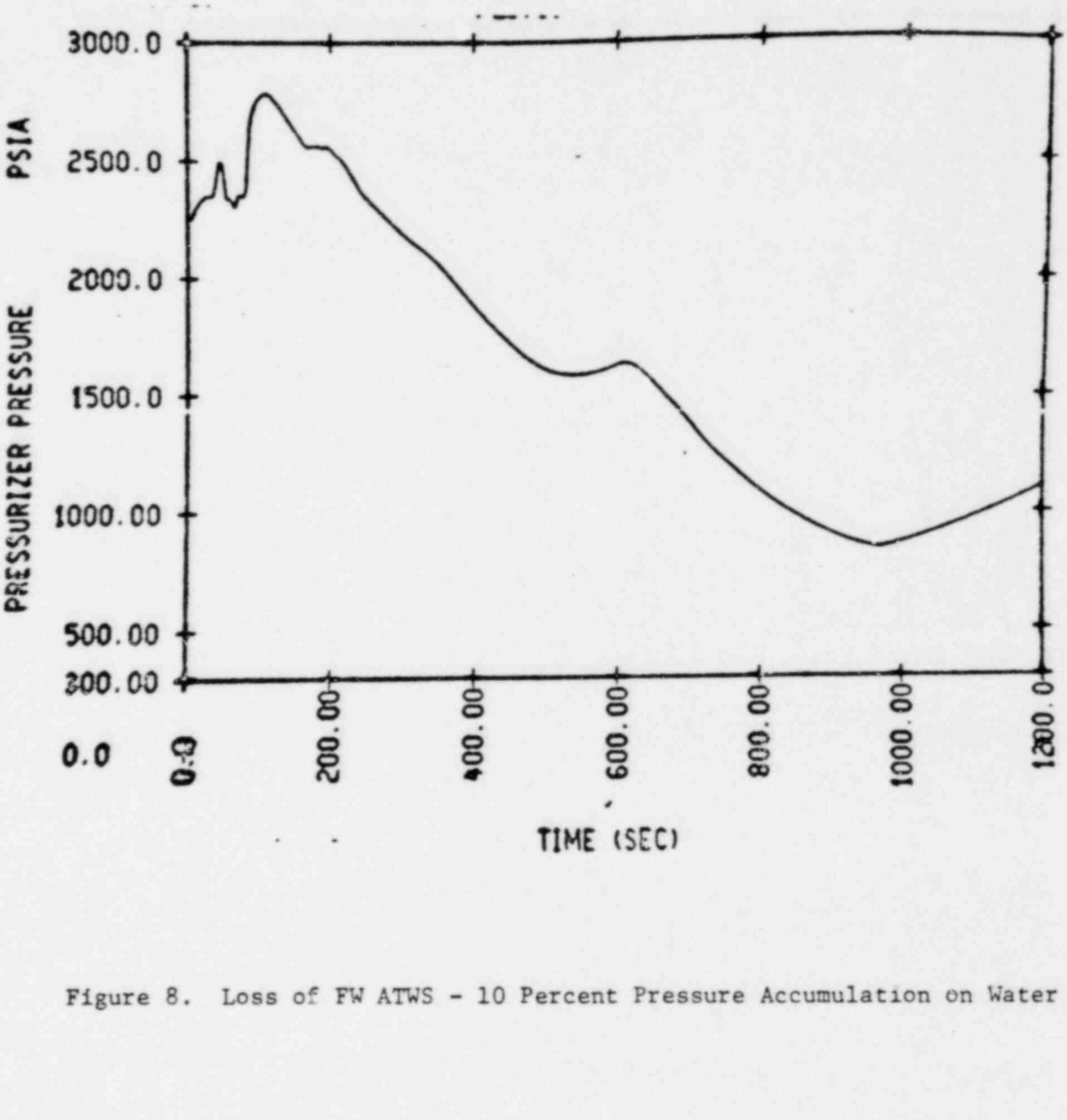


Figure 8. Loss of FW ATWS - 10 Percent Pressure Accumulation on Water Rel

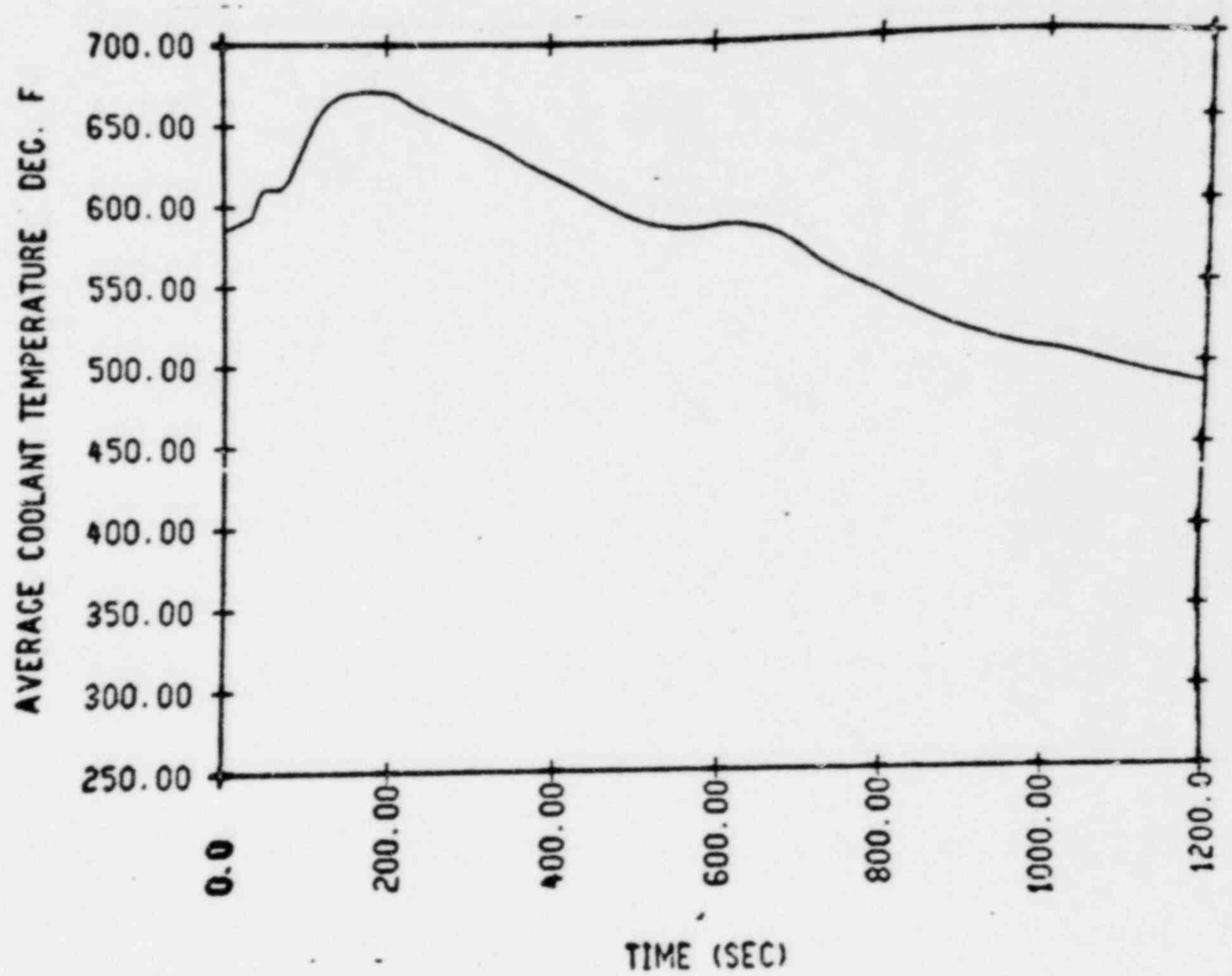


Figure 9. Loss of FW ATWS - 10 Percent Pressure Accumulation on Wa
Relief

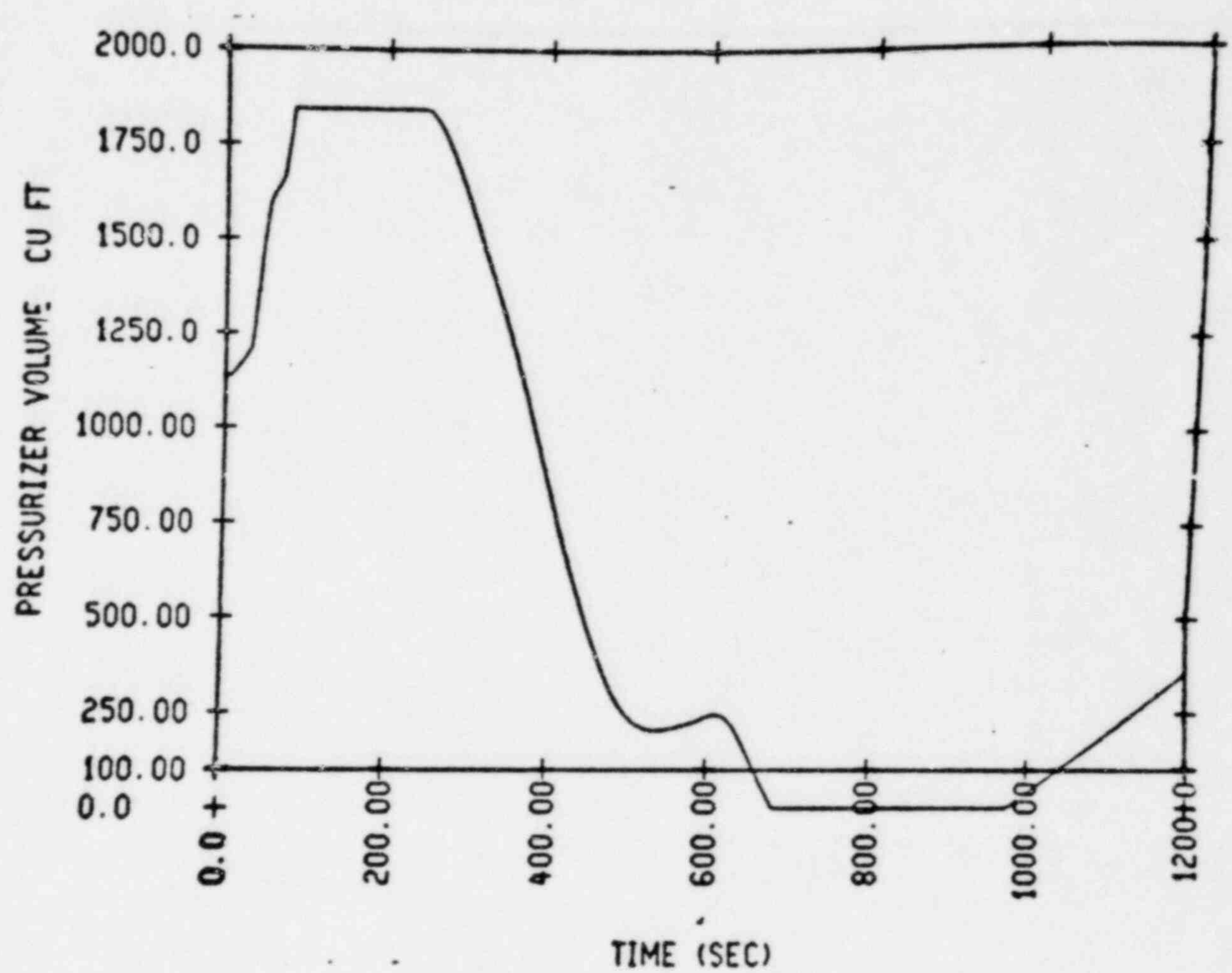


Figure 10. Loss of FW ATWS - 10 Percent Pressure Accumulation on Water Relief

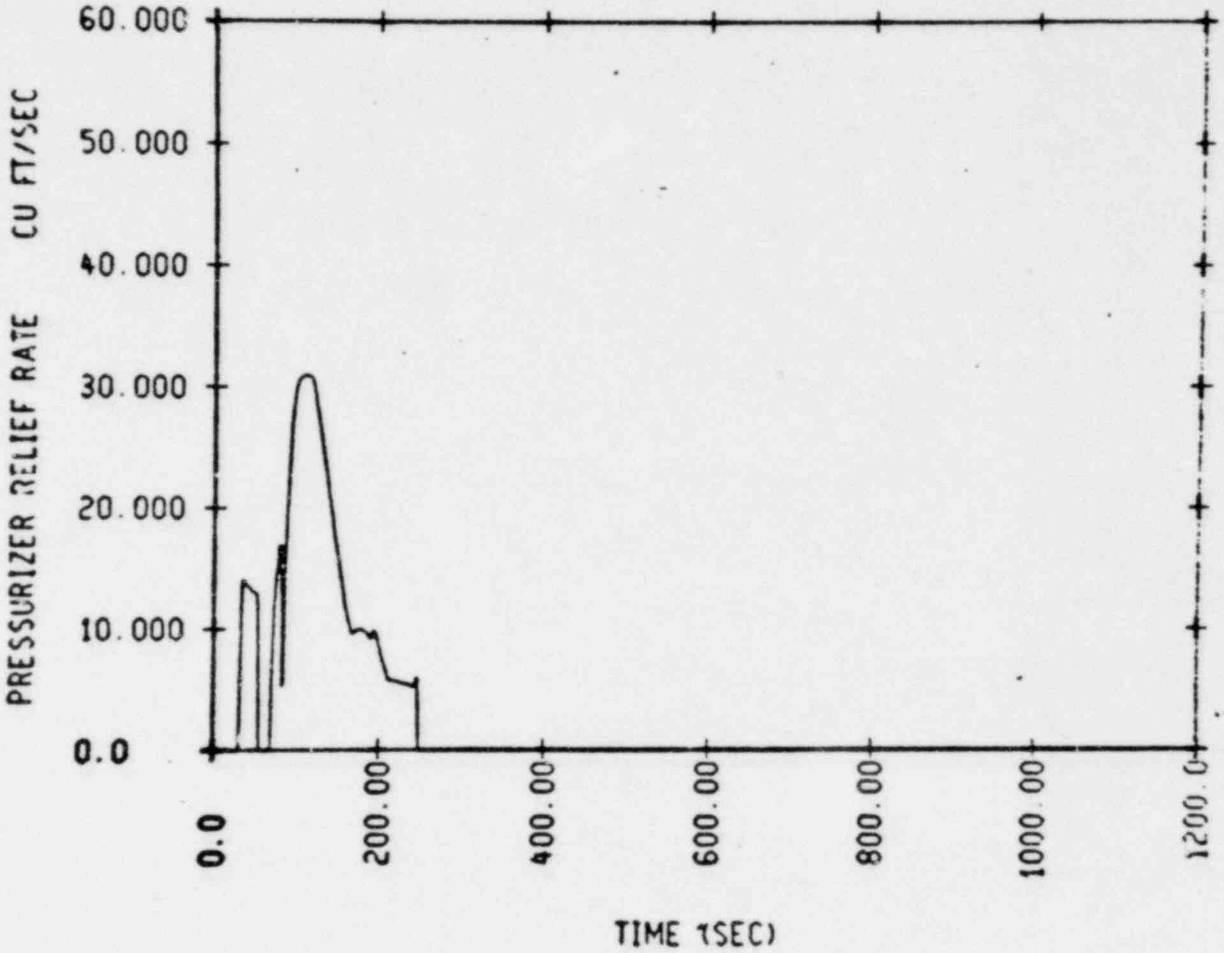


Figure 11. Loss of FW ATWS - 10 Percent Pressure Accumulation on Water Relief

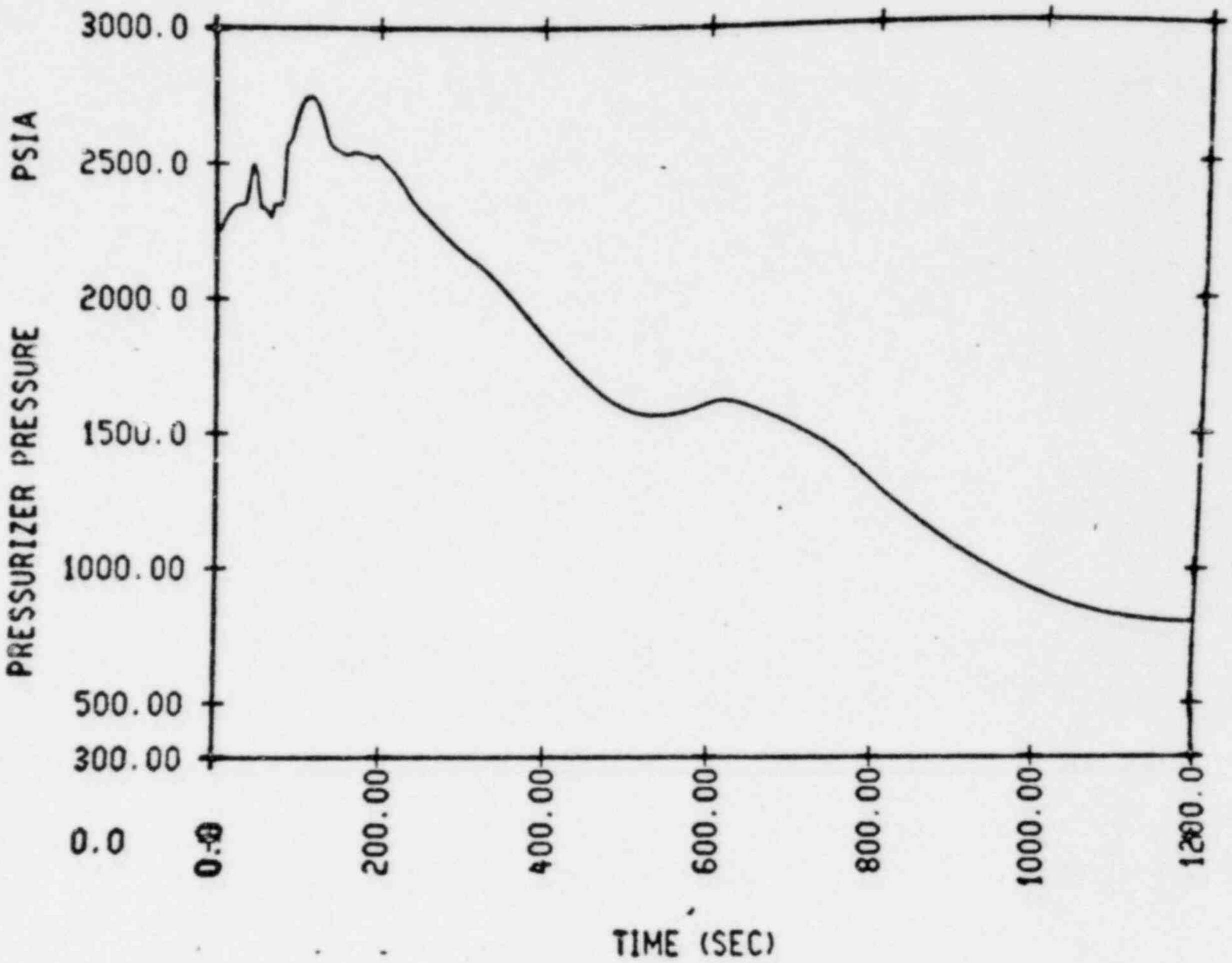


Figure 12. Loss of FW ATWS With a Single Failure in the SI System

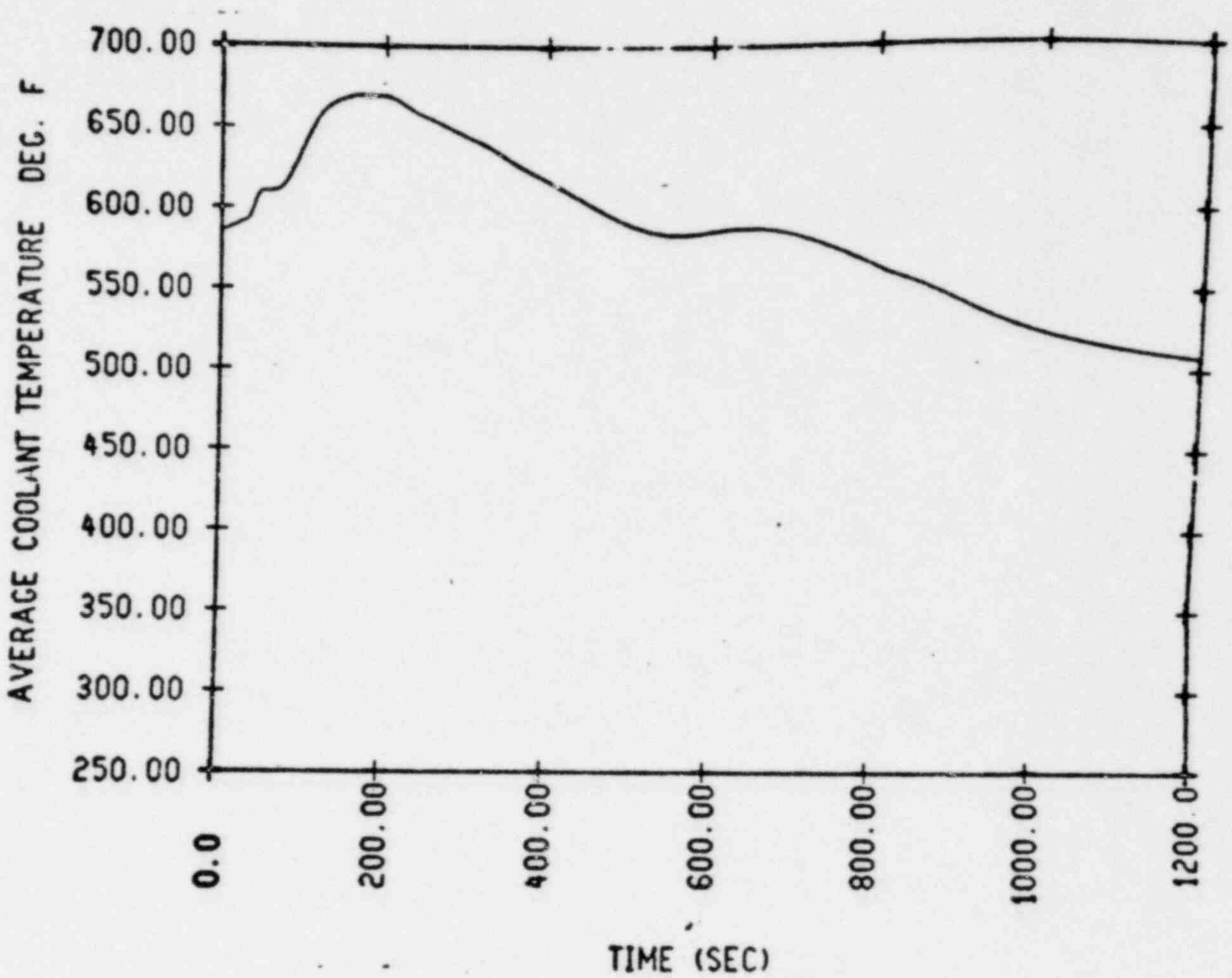


Figure 13. Loss of FW ATWS With a Single Failure in the SI System

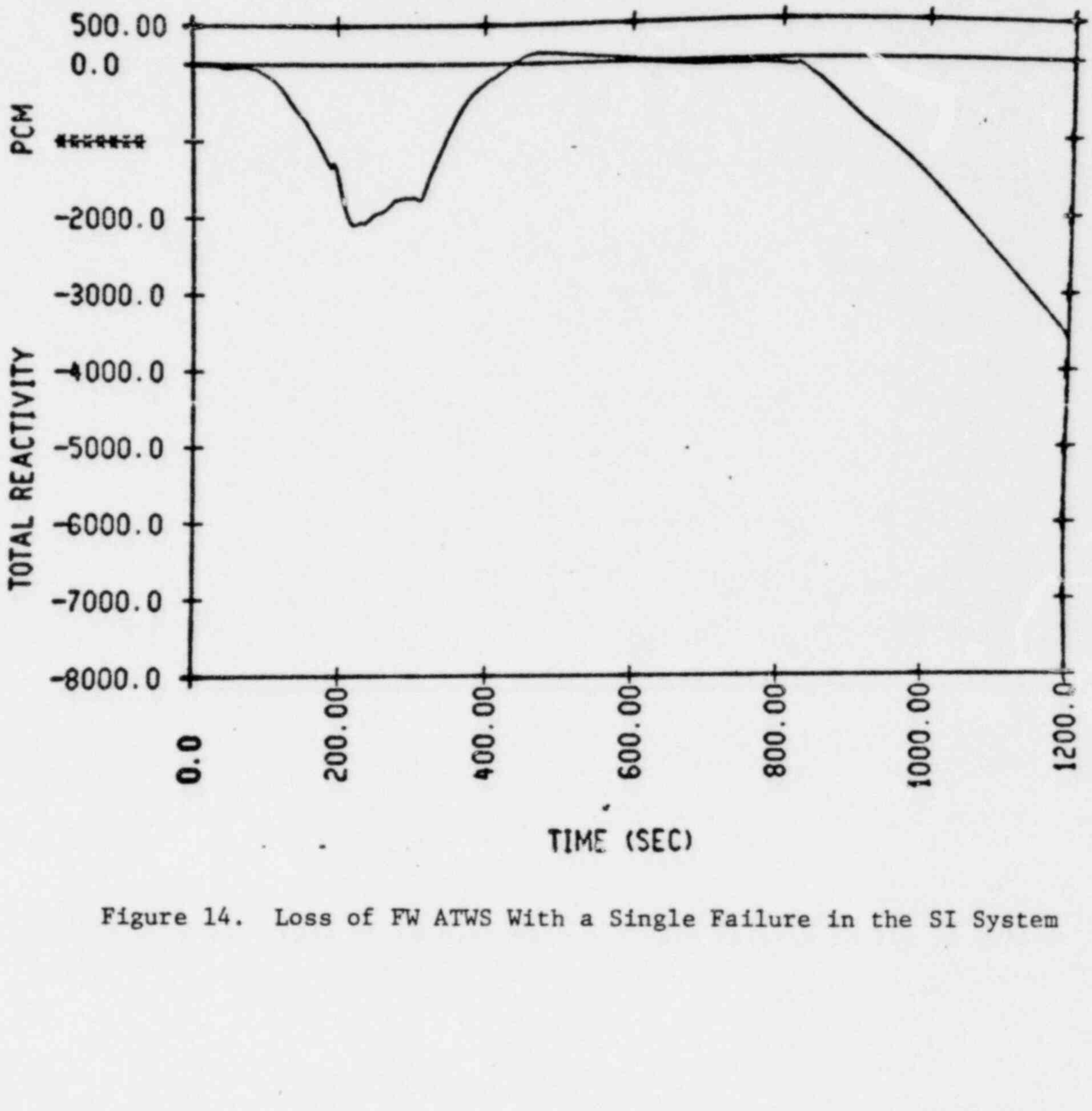


Figure 14. Loss of FW ATWS With a Single Failure in the SI System

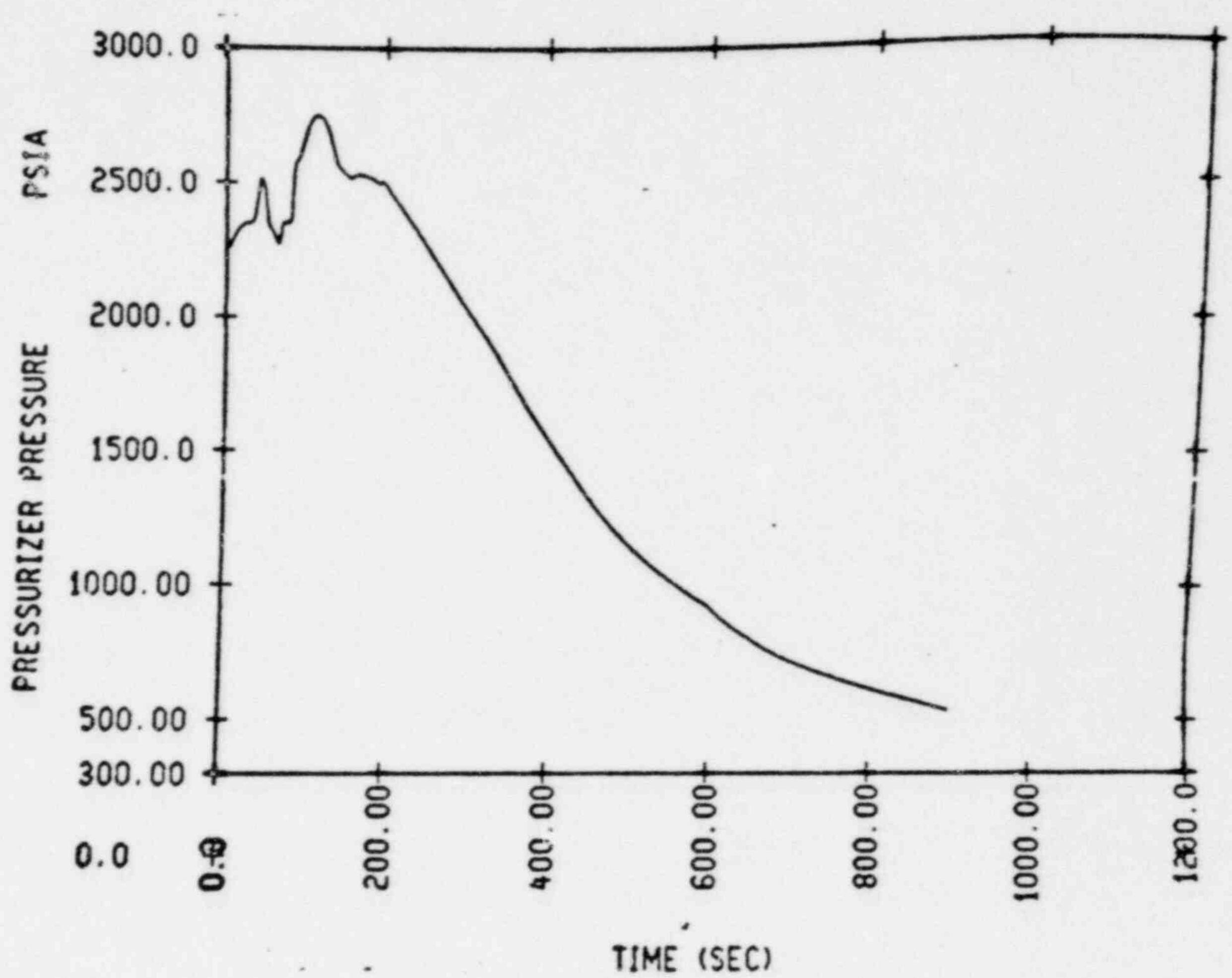


Figure 15. Loss of FW ATWS With a PZR Safety Valve that Fails to Re-Seat

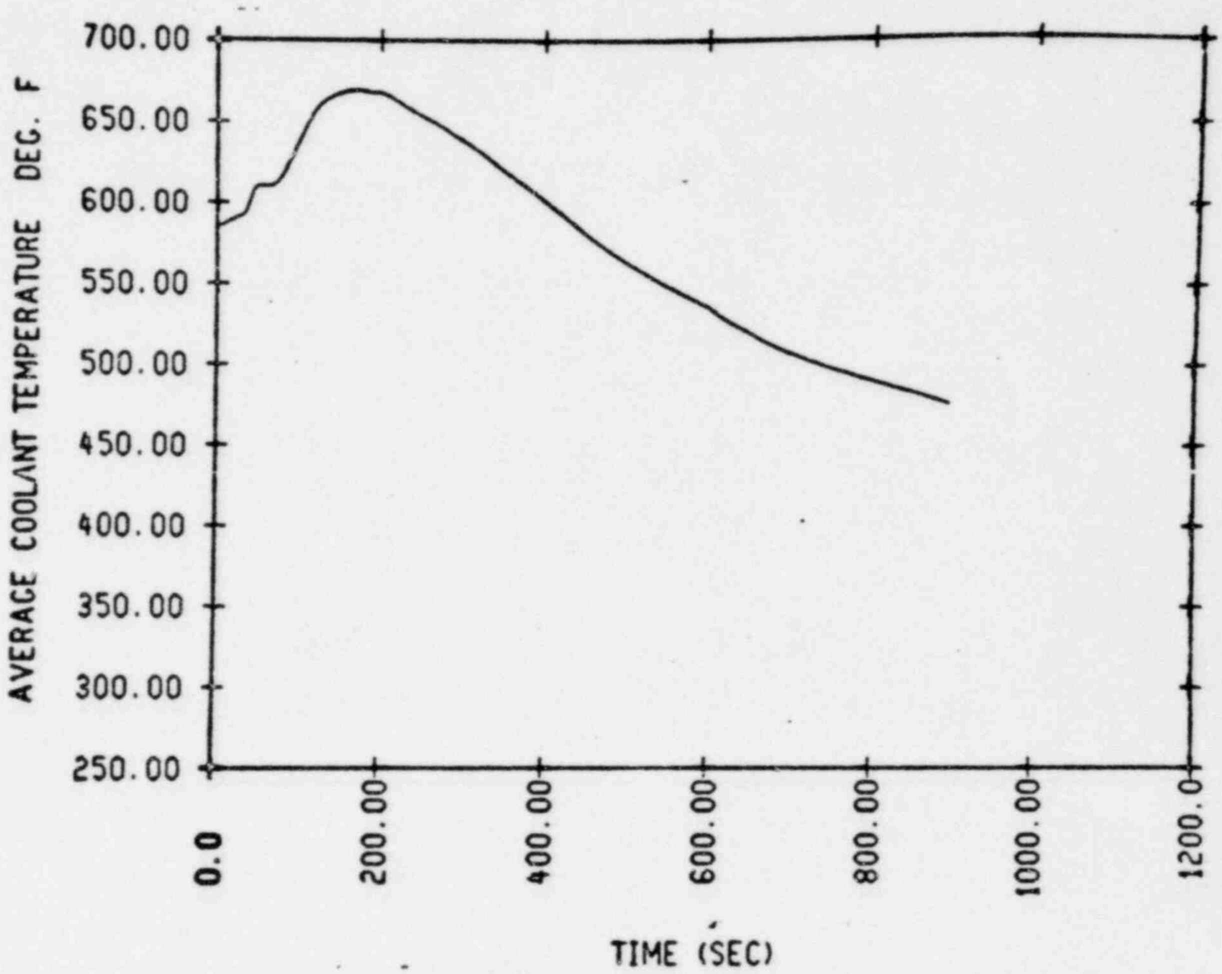


Figure 16. Loss of FW ATWS With a PZR Safety Valve that Fails to Re-Seat

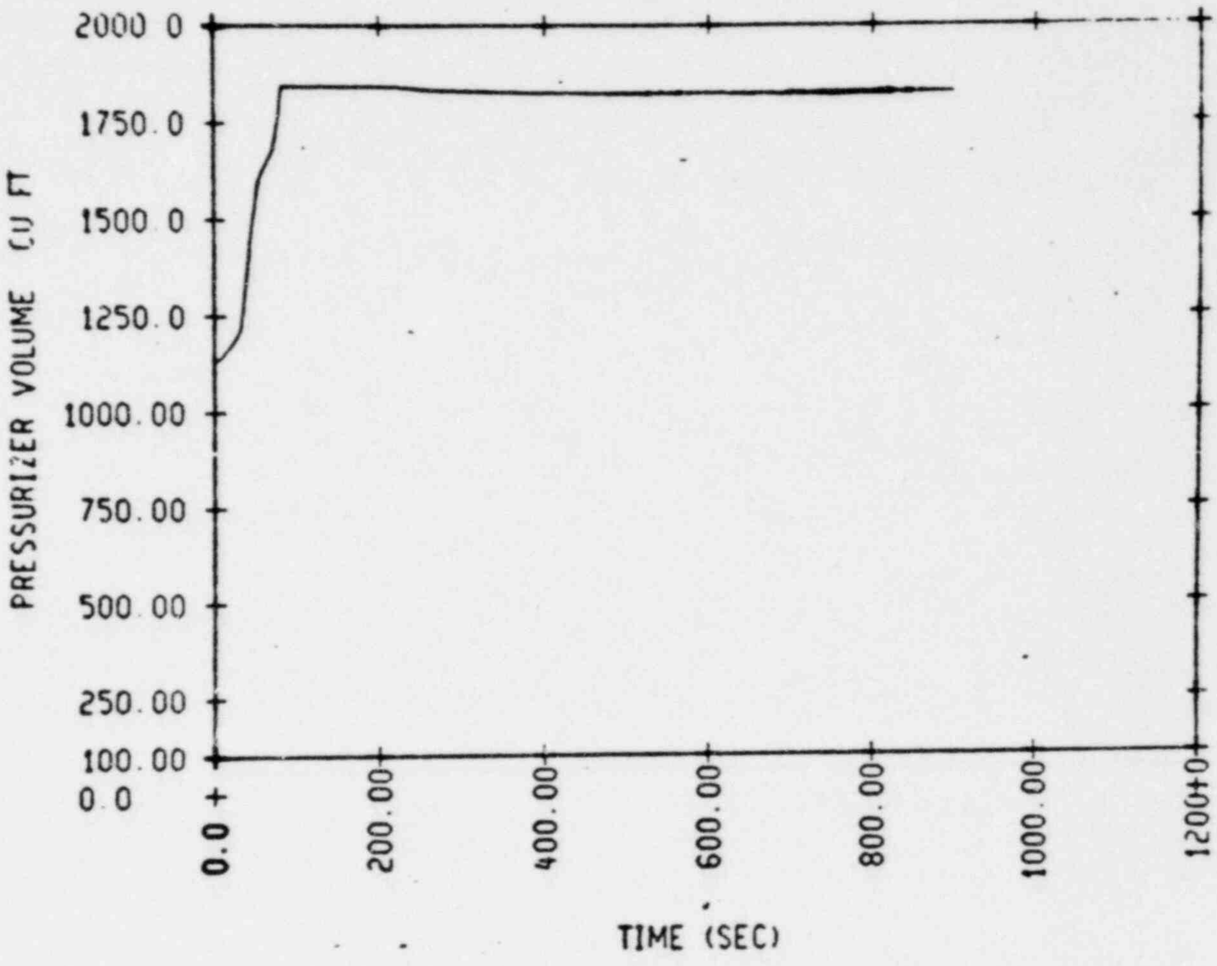


Figure 17. Loss of FW ATWS With a PZR Safety Valve that Fails to Re-Sea

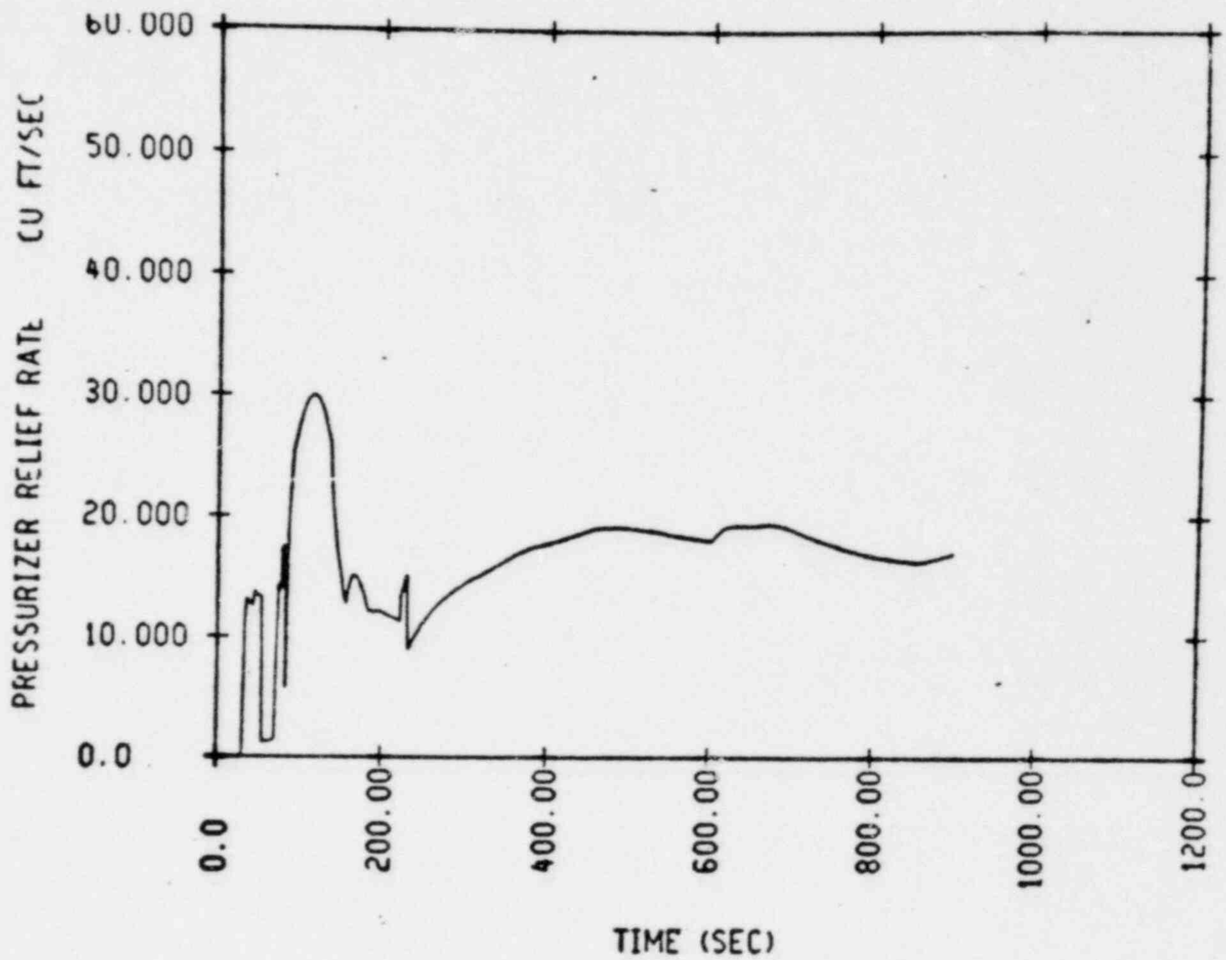


Figure 18. Loss of FW ATWS With a PZR Safety Valve that Fails to Re-Seat

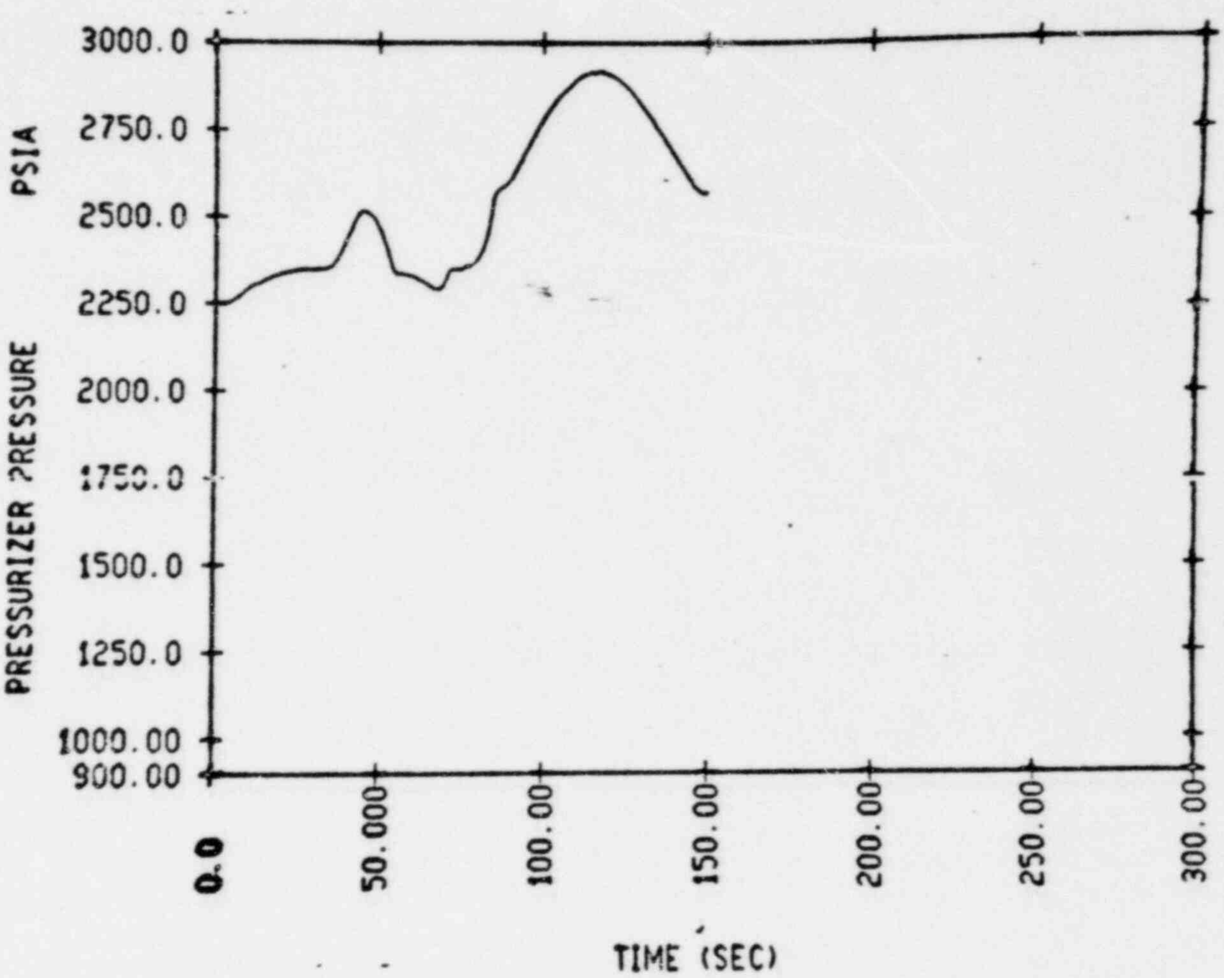


Figure 19. Loss of FW ATWS - Turbine Driven Aux. FW Pump Fails

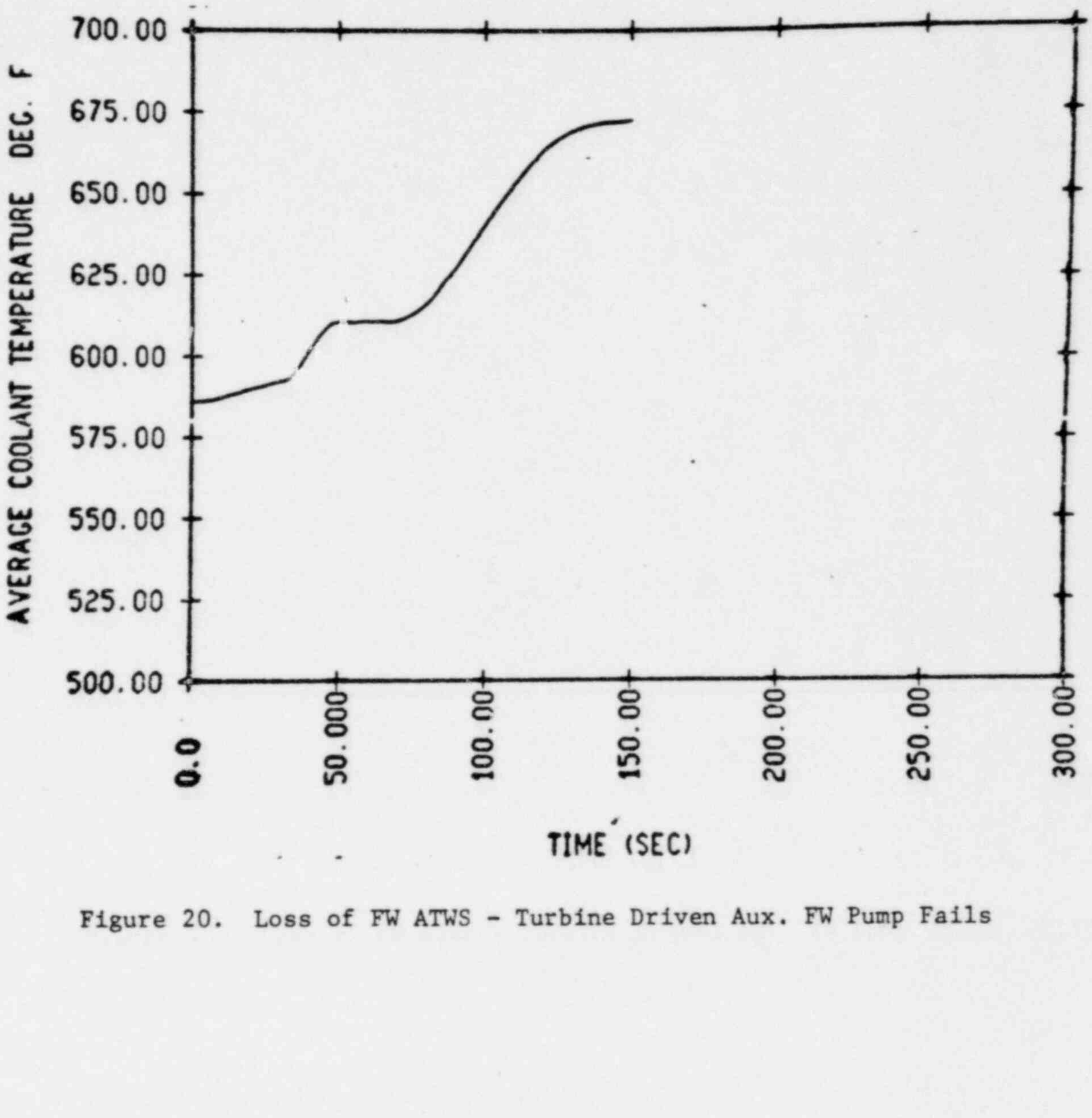


Figure 20. Loss of FW ATWS - Turbine Driven Aux. FW Pump Fails

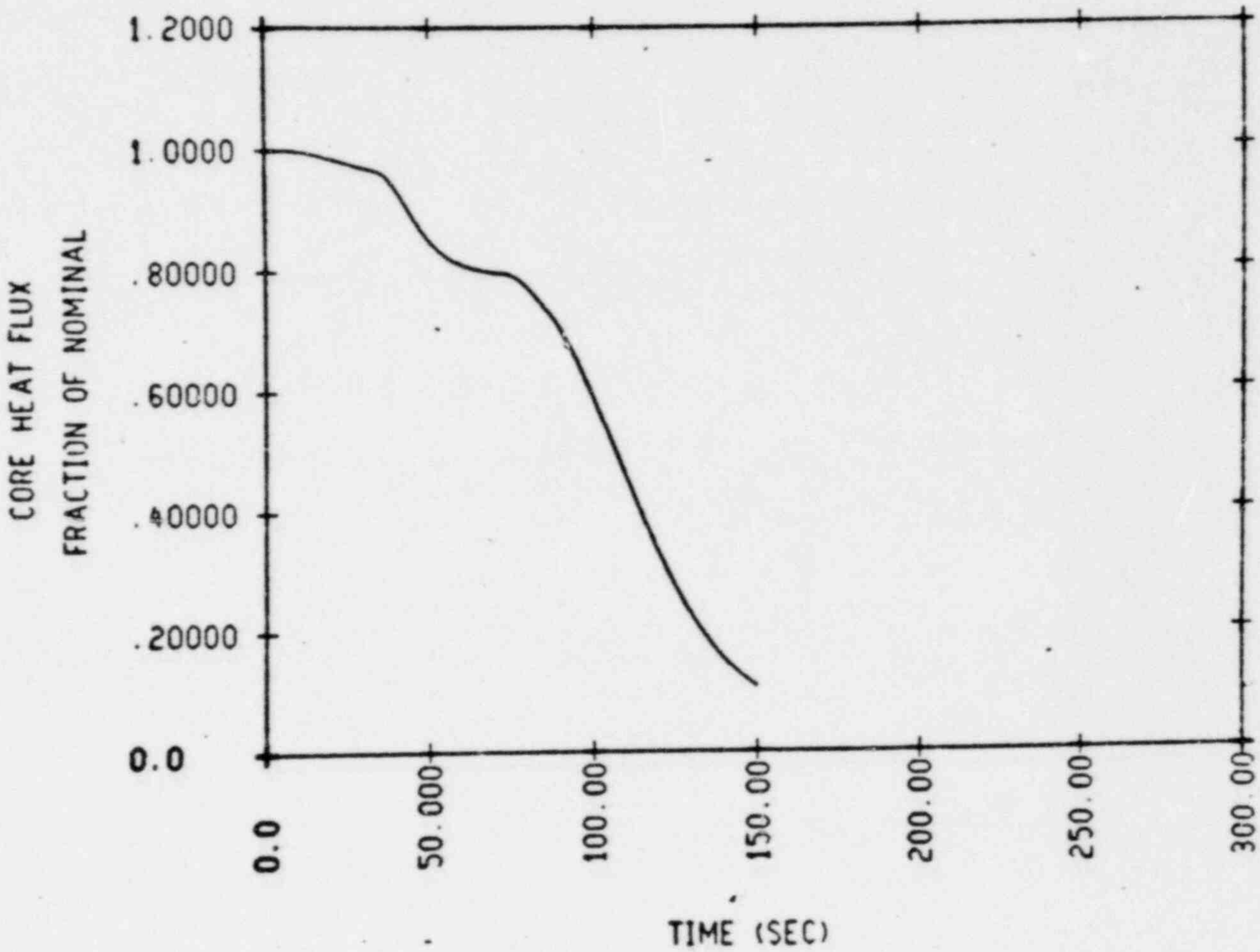


Figure 21. Loss of FW ATWS - Turbine Driven Aux. FW Pump Fails

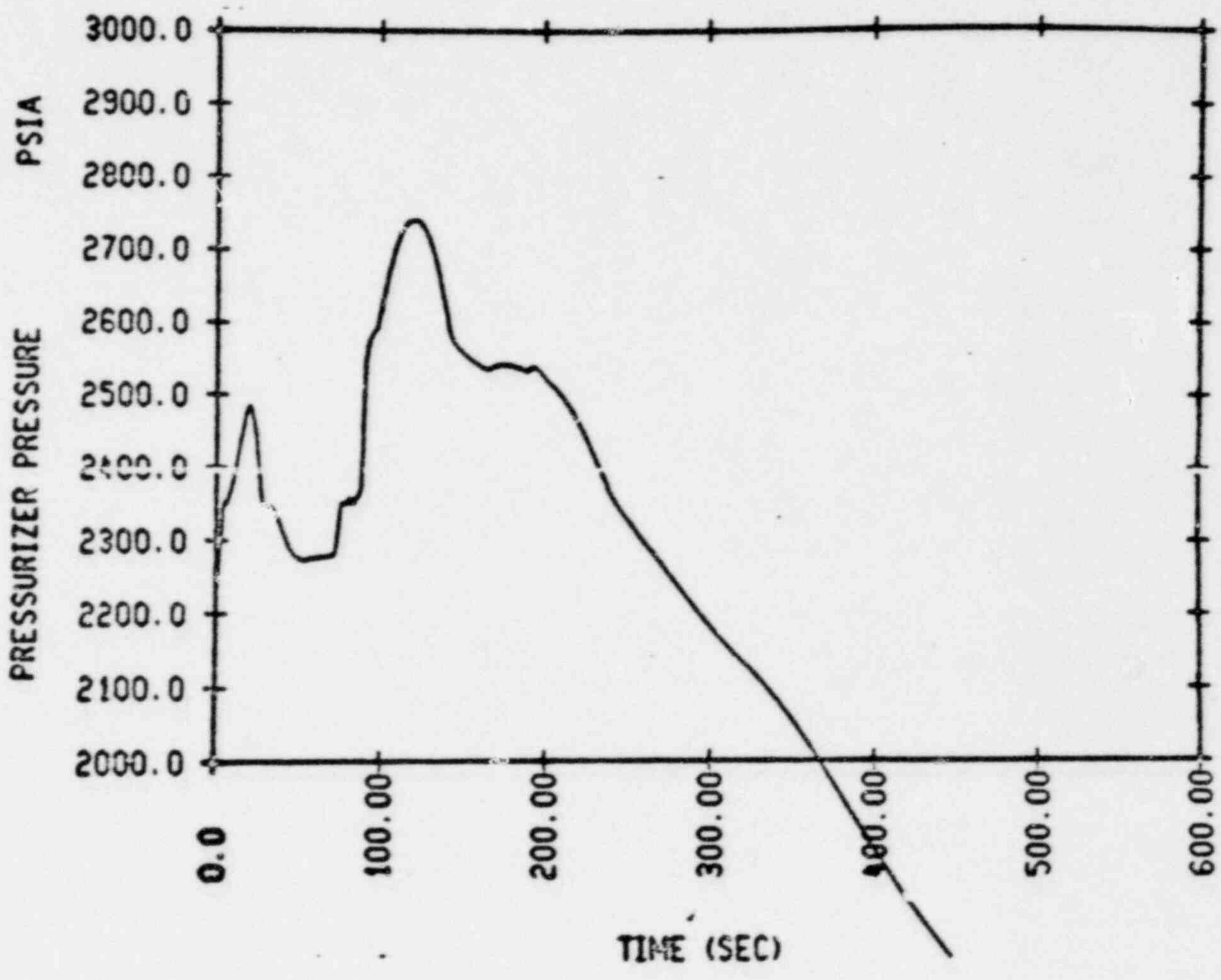


Figure 22. Loss of Load ATWS

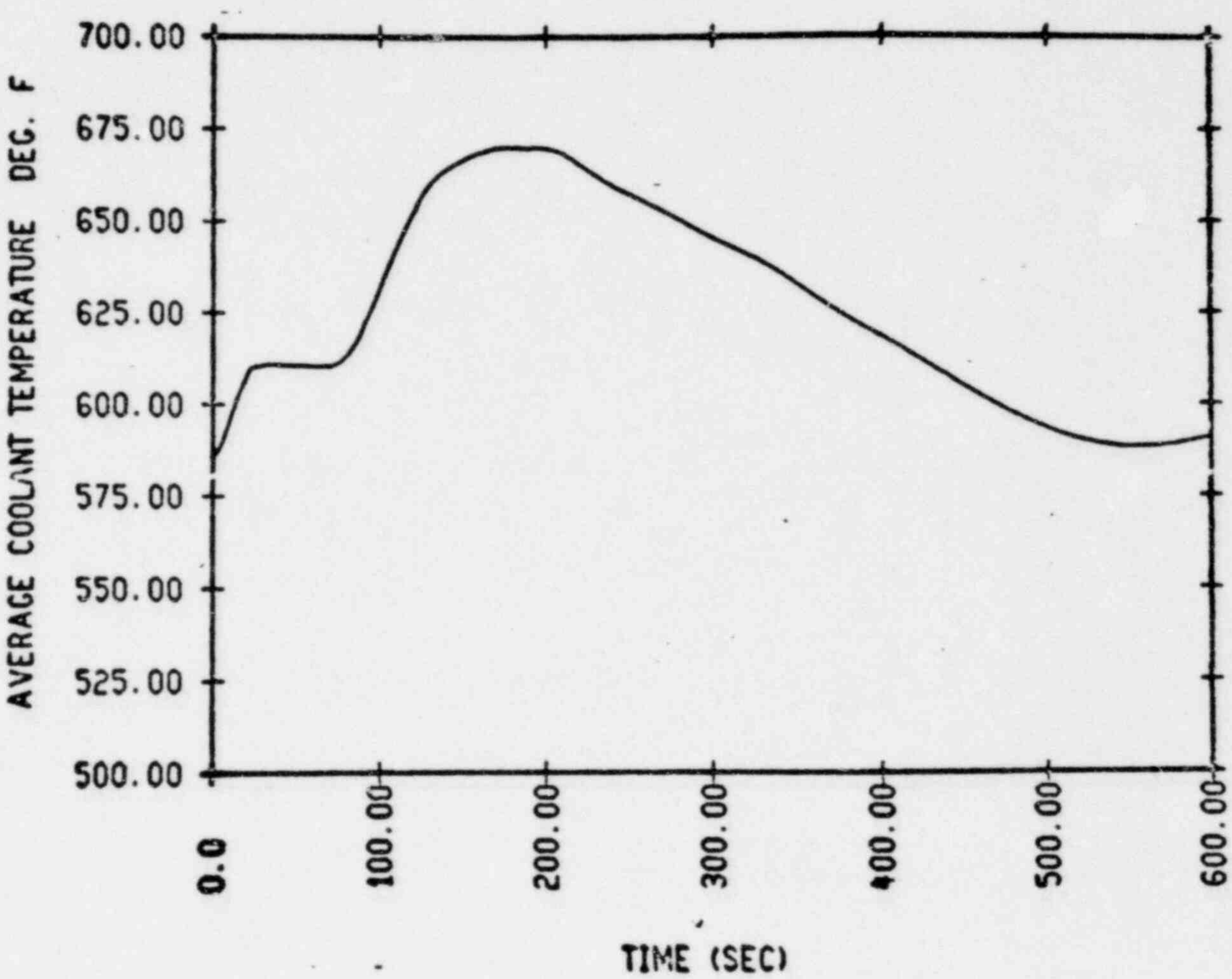


Figure 23. Loss of Load ATWS - Ref Case

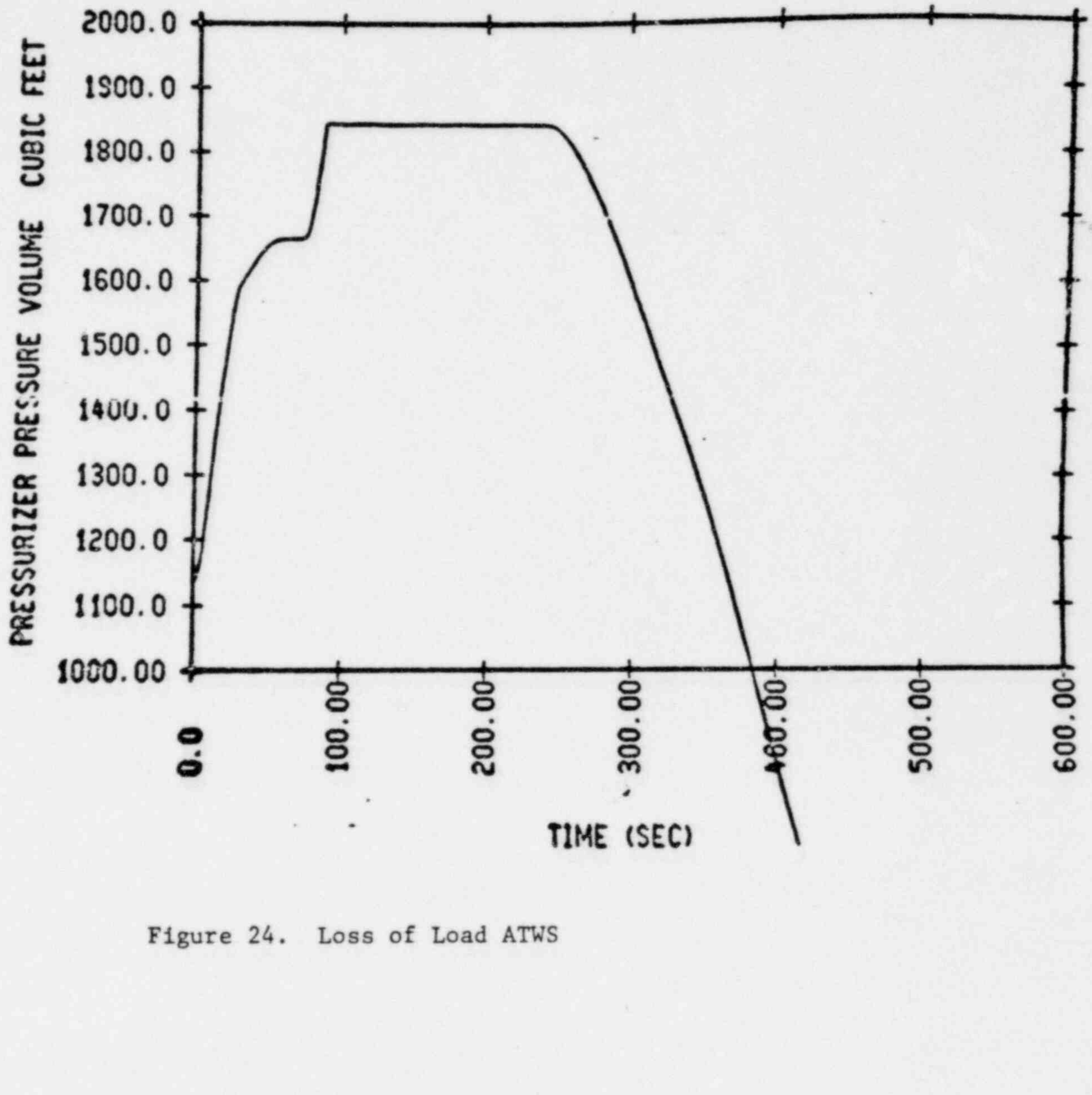


Figure 24. Loss of Load ATWS

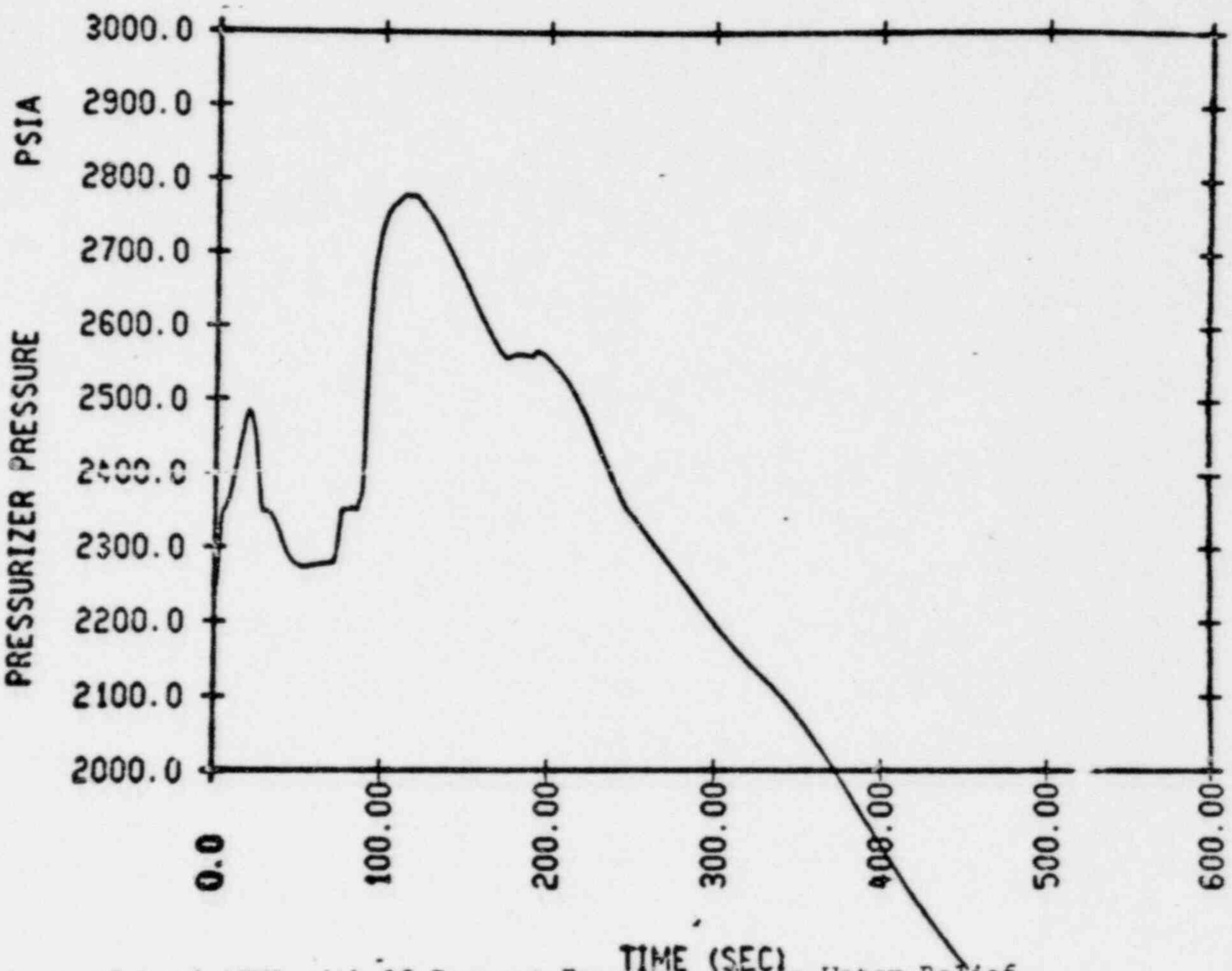


Figure 25. Loss of Load ATWS with 10 Percent Pressure ACC in Water Relief

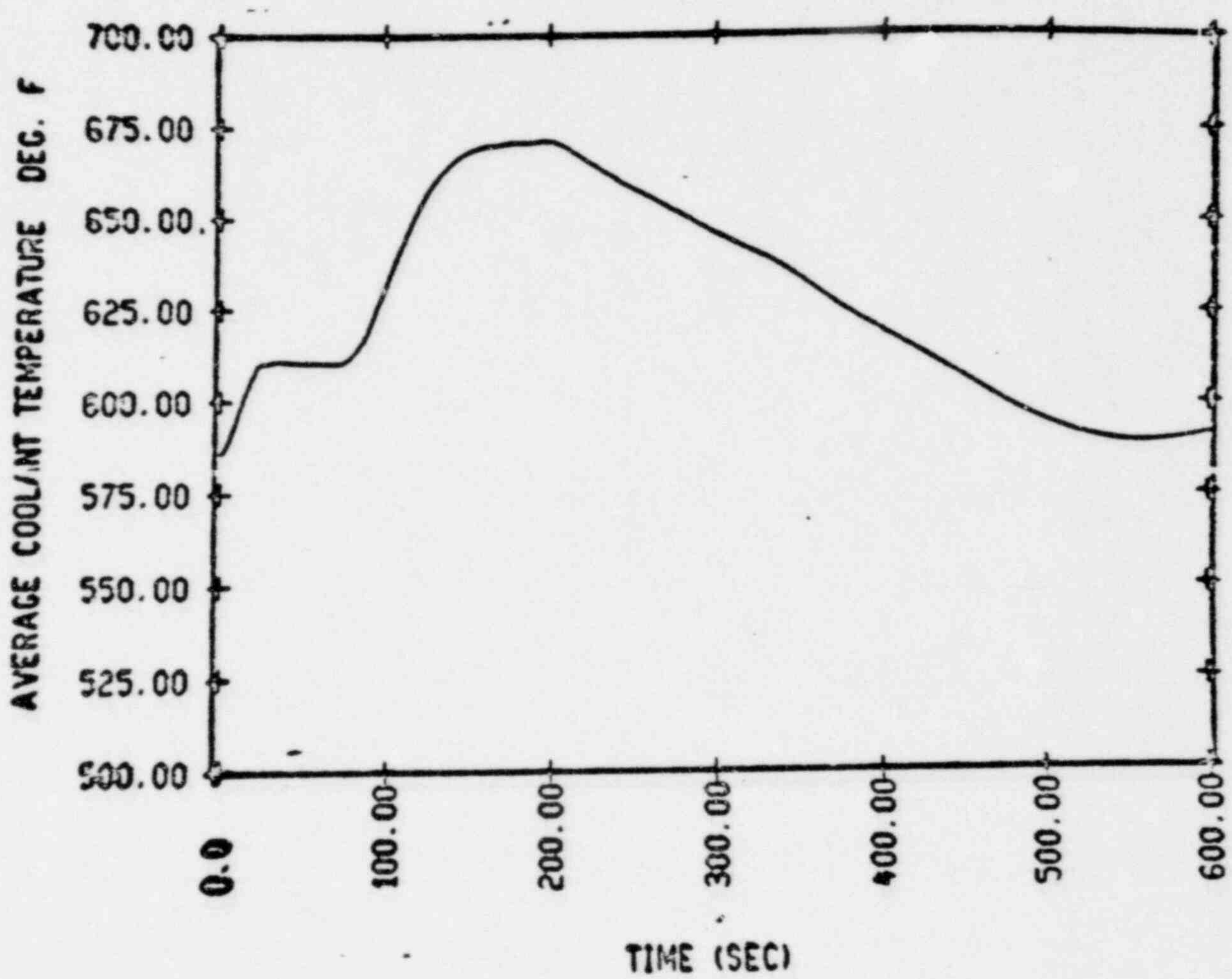
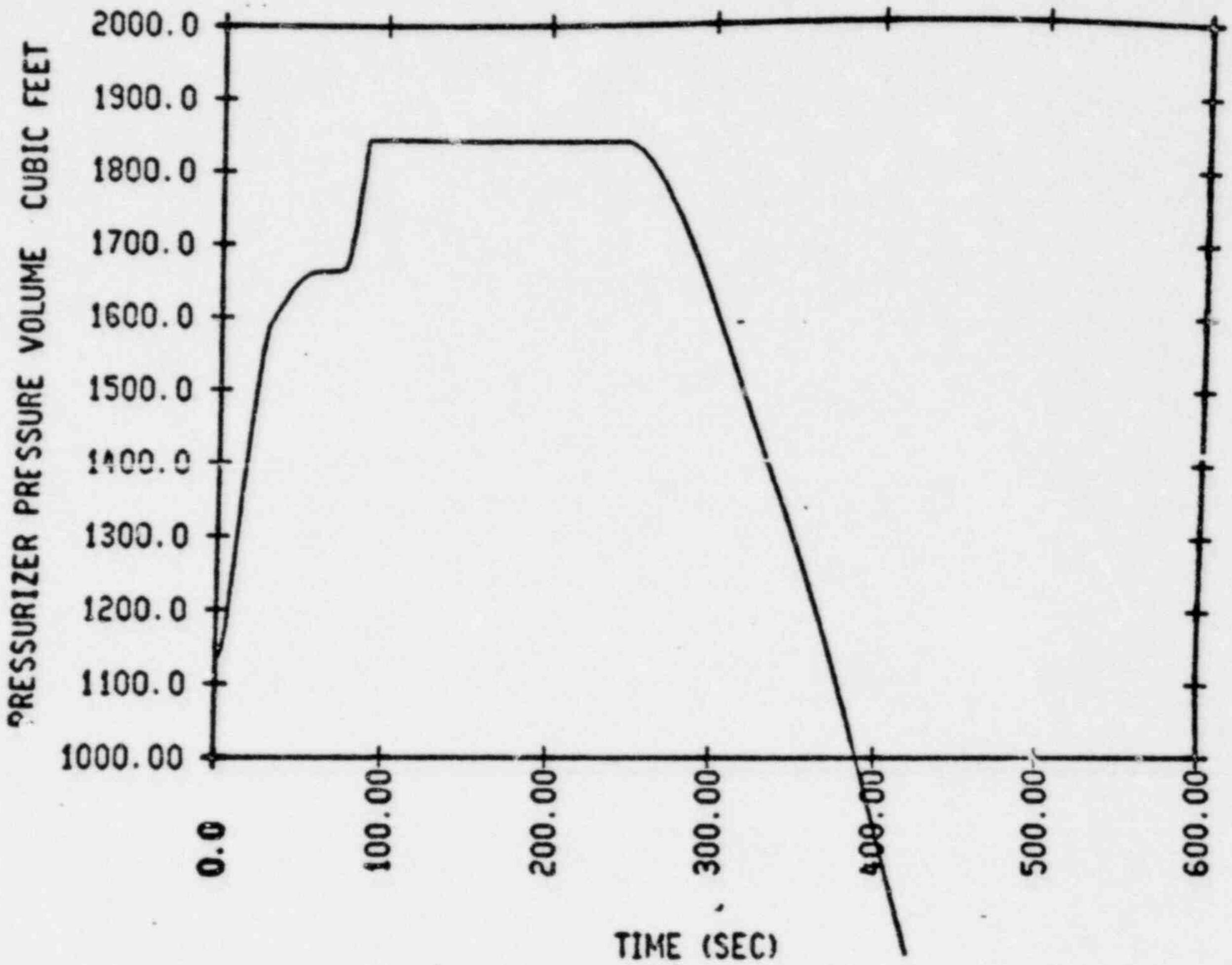


Figure 26. Loss of Load ATWS with 10 Percent Pressure ACC In Water Relief



Loss of Load ATWS With 10 Percent Pressure ACC in Water Relief

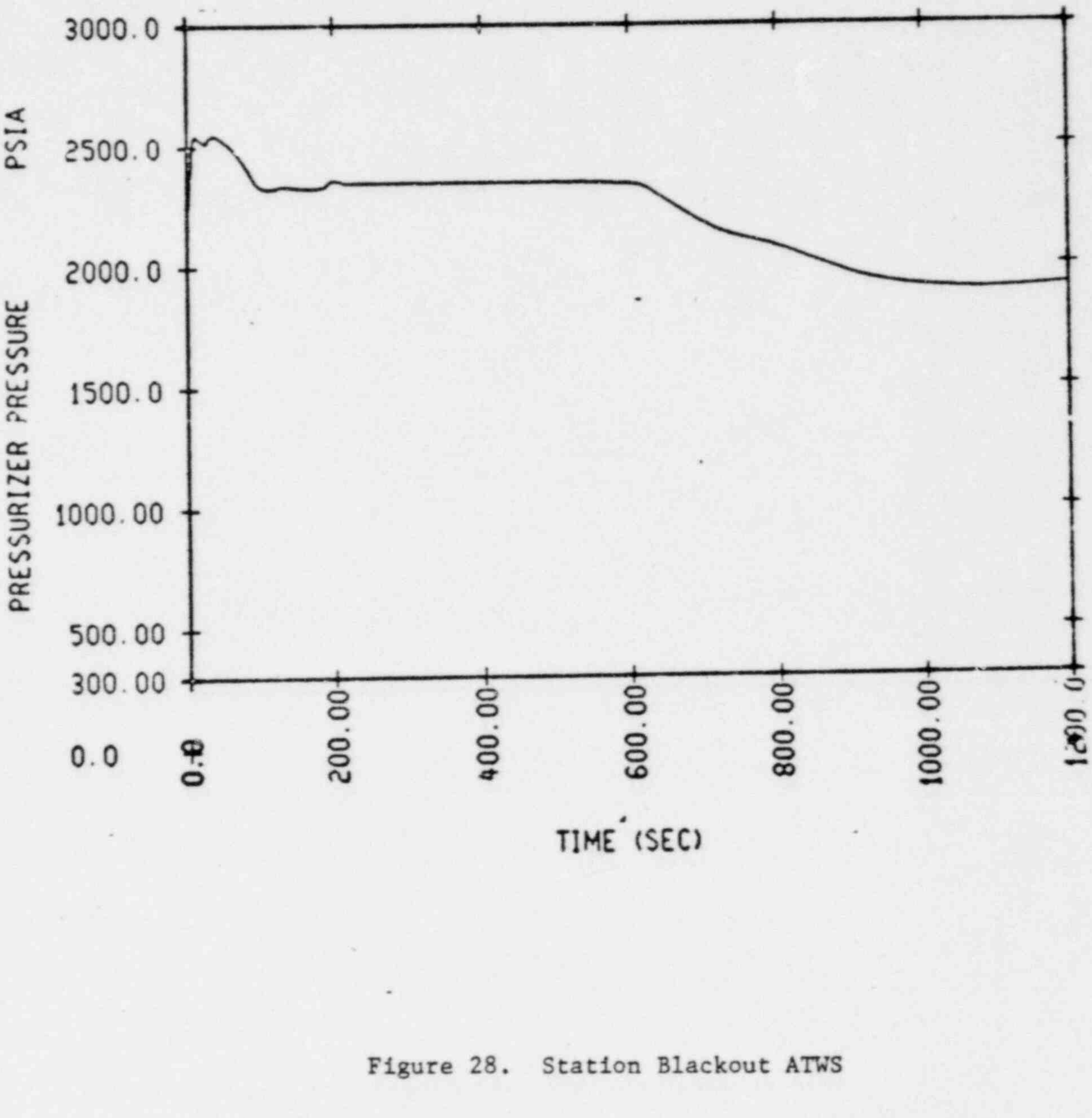


Figure 28. Station Blackout ATWS

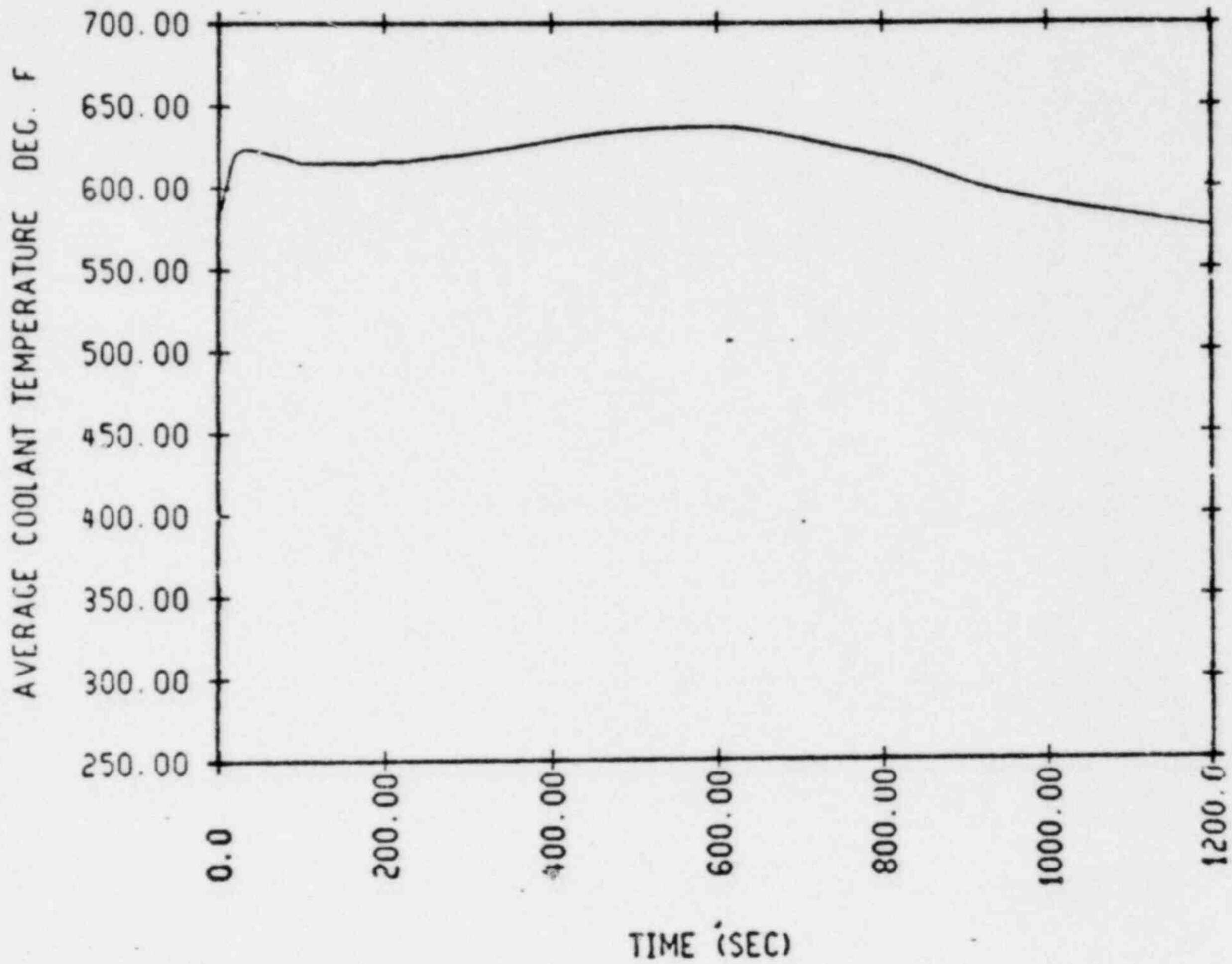


Figure 29. Station Blackout ATWS

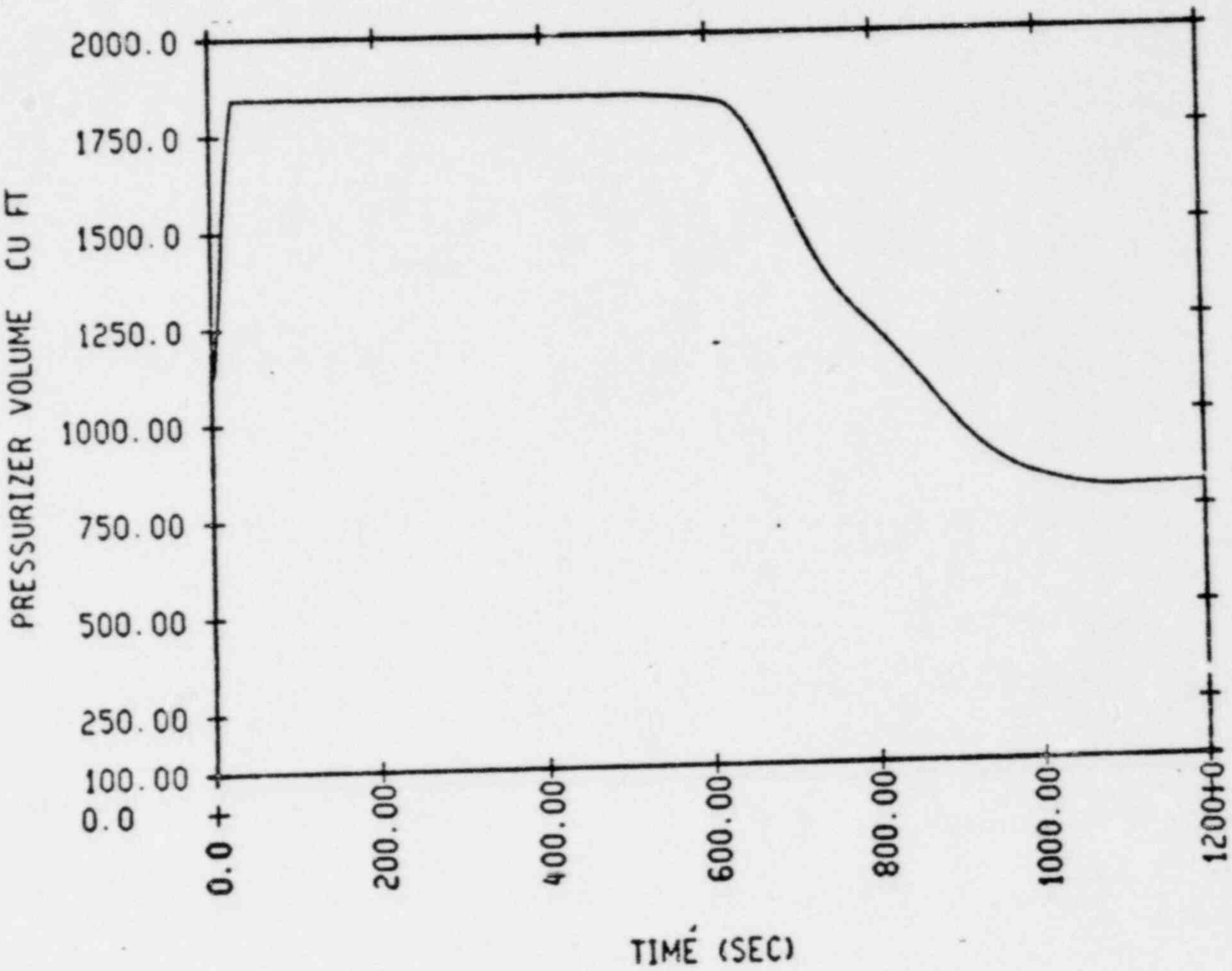


Figure ~~20~~ 30. Station Blackout ATWS

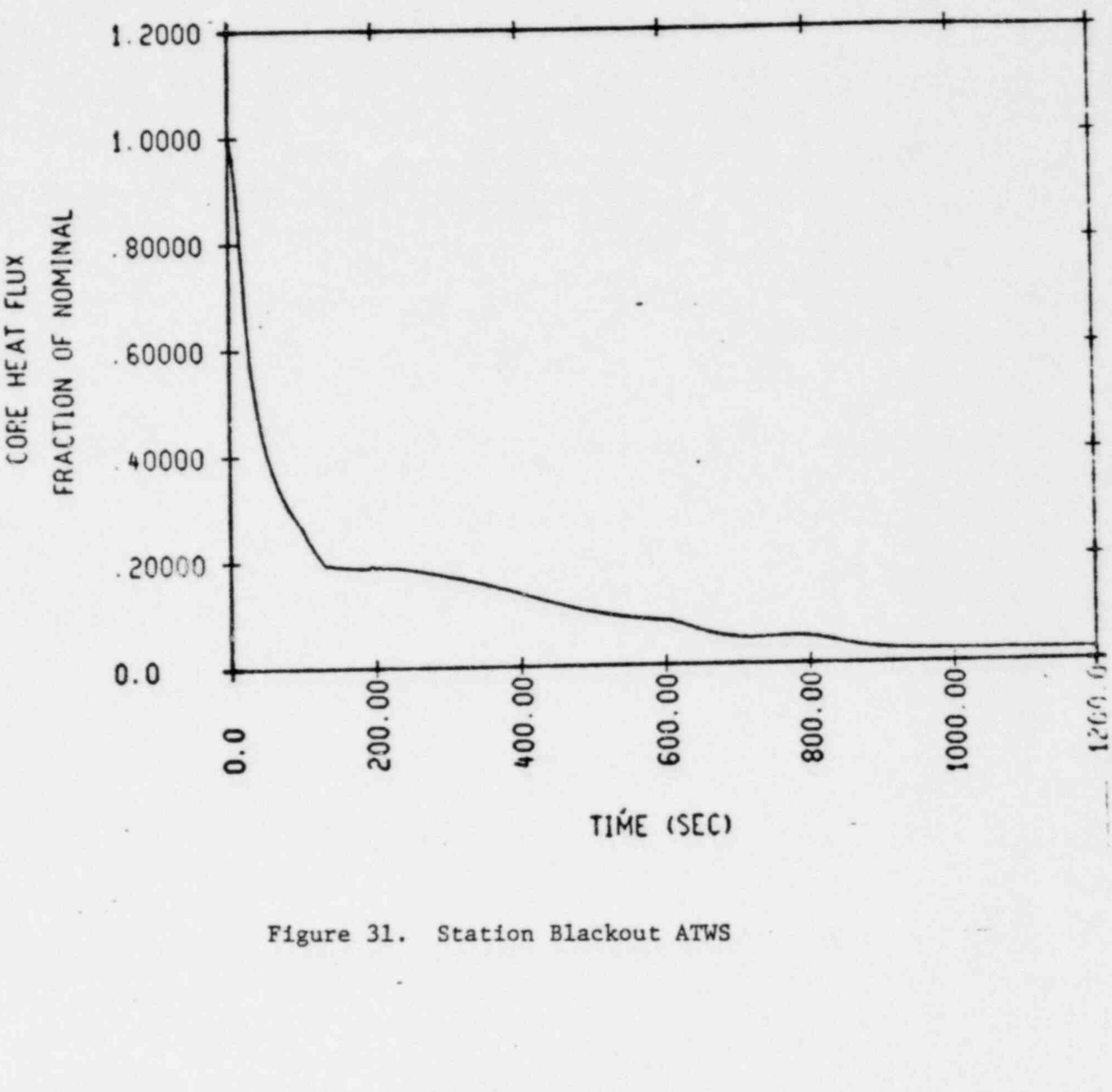


Figure 31. Station Blackout ATWS

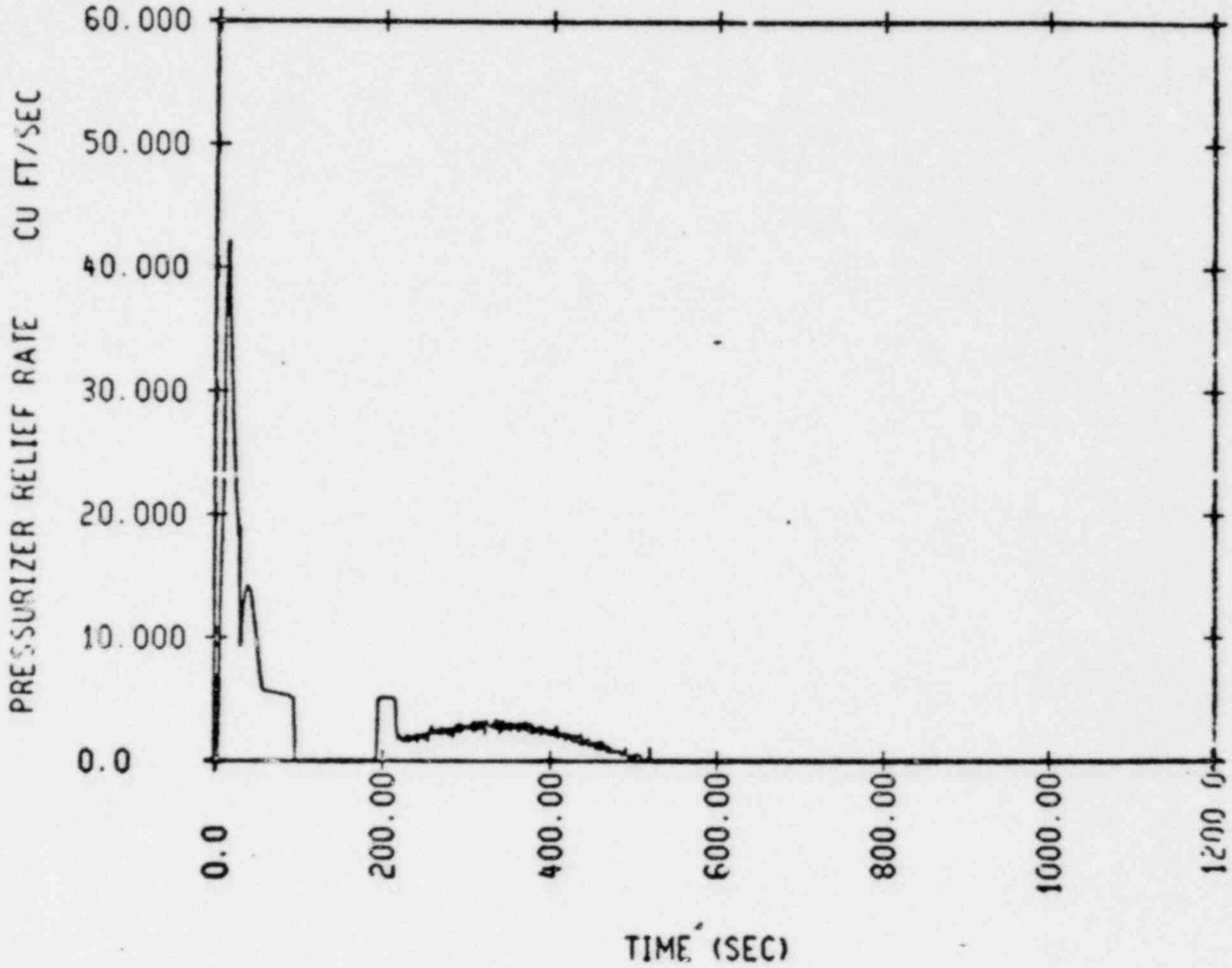


Figure 32. Station Blackout ATWS

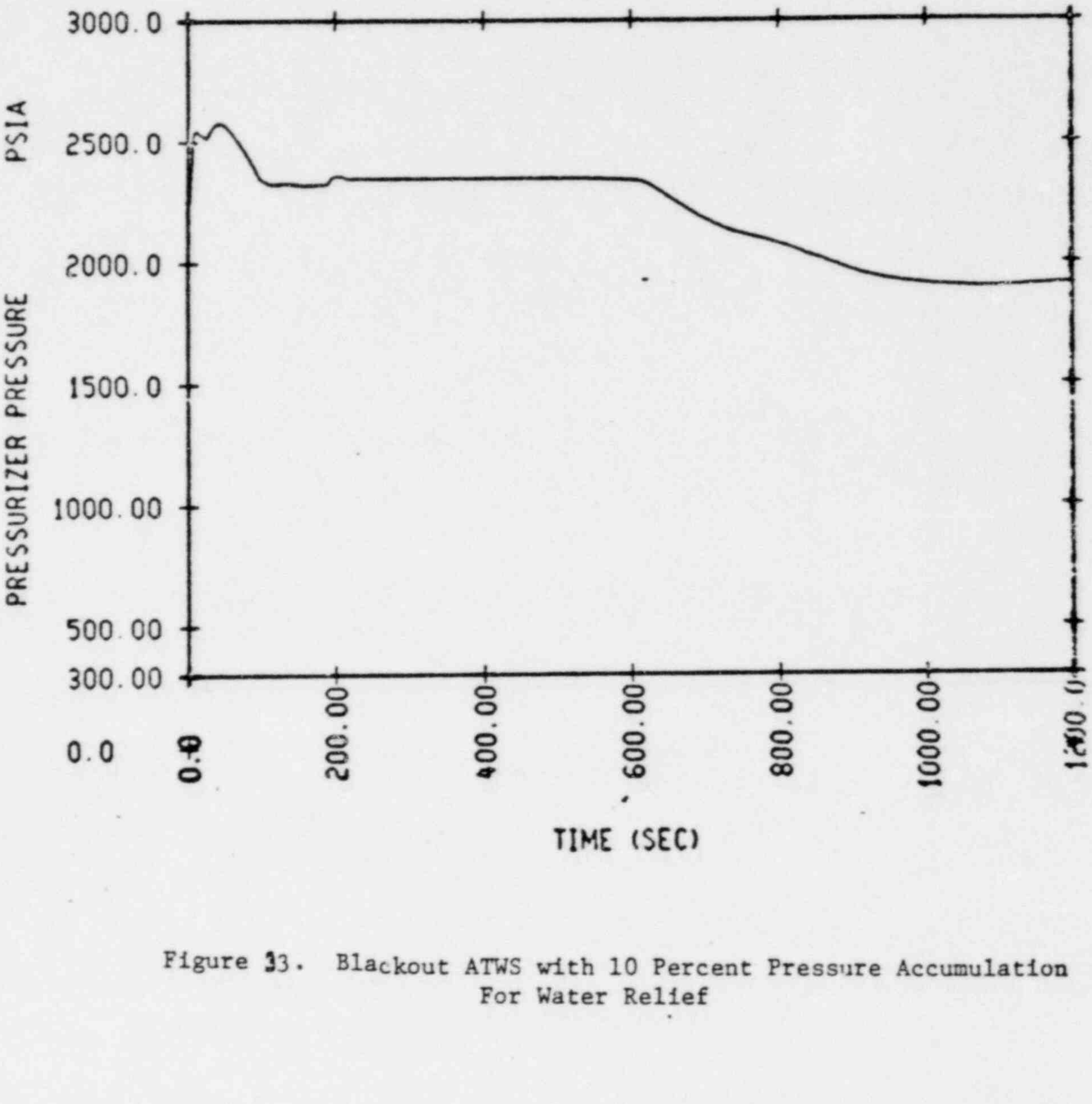


Figure 33. Blackout ATWS with 10 Percent Pressure Accumulation For Water Relief

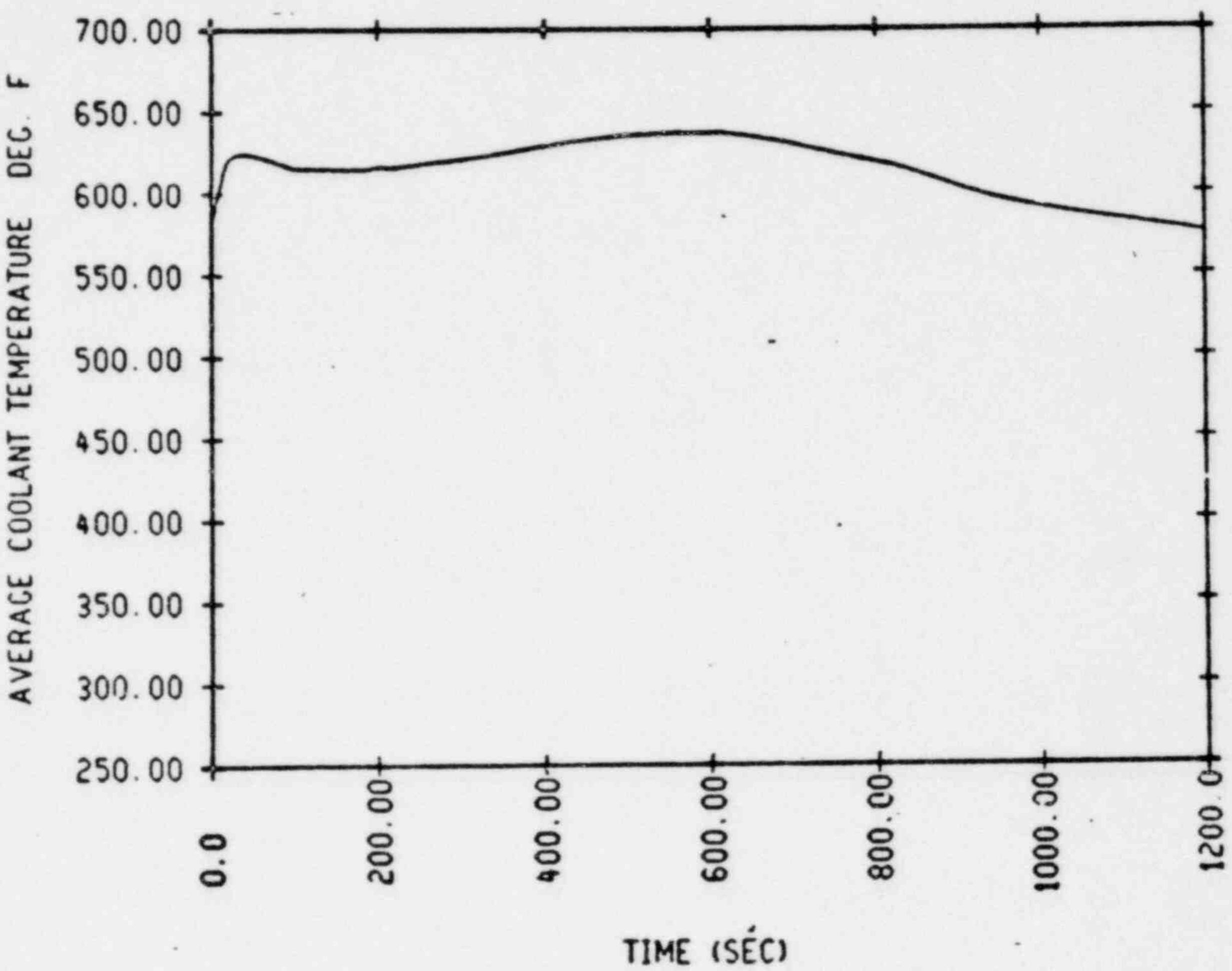


Figure 34. Blackout ATWS With 10 Percent Pressure Accumulation for Water Relief

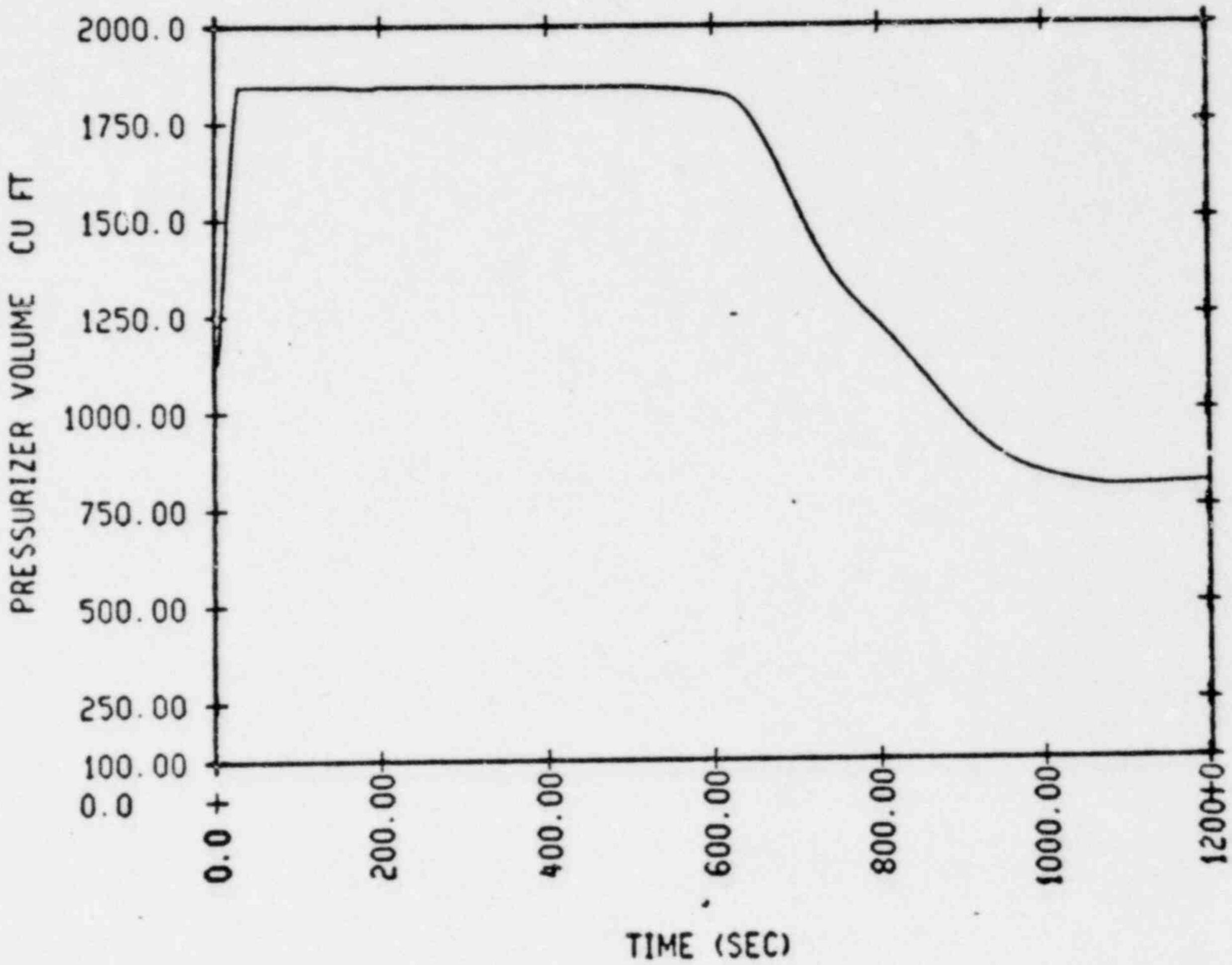


Figure 35. Blackout ATWS With 10 Percent Pressure Accumulation for Water Relief

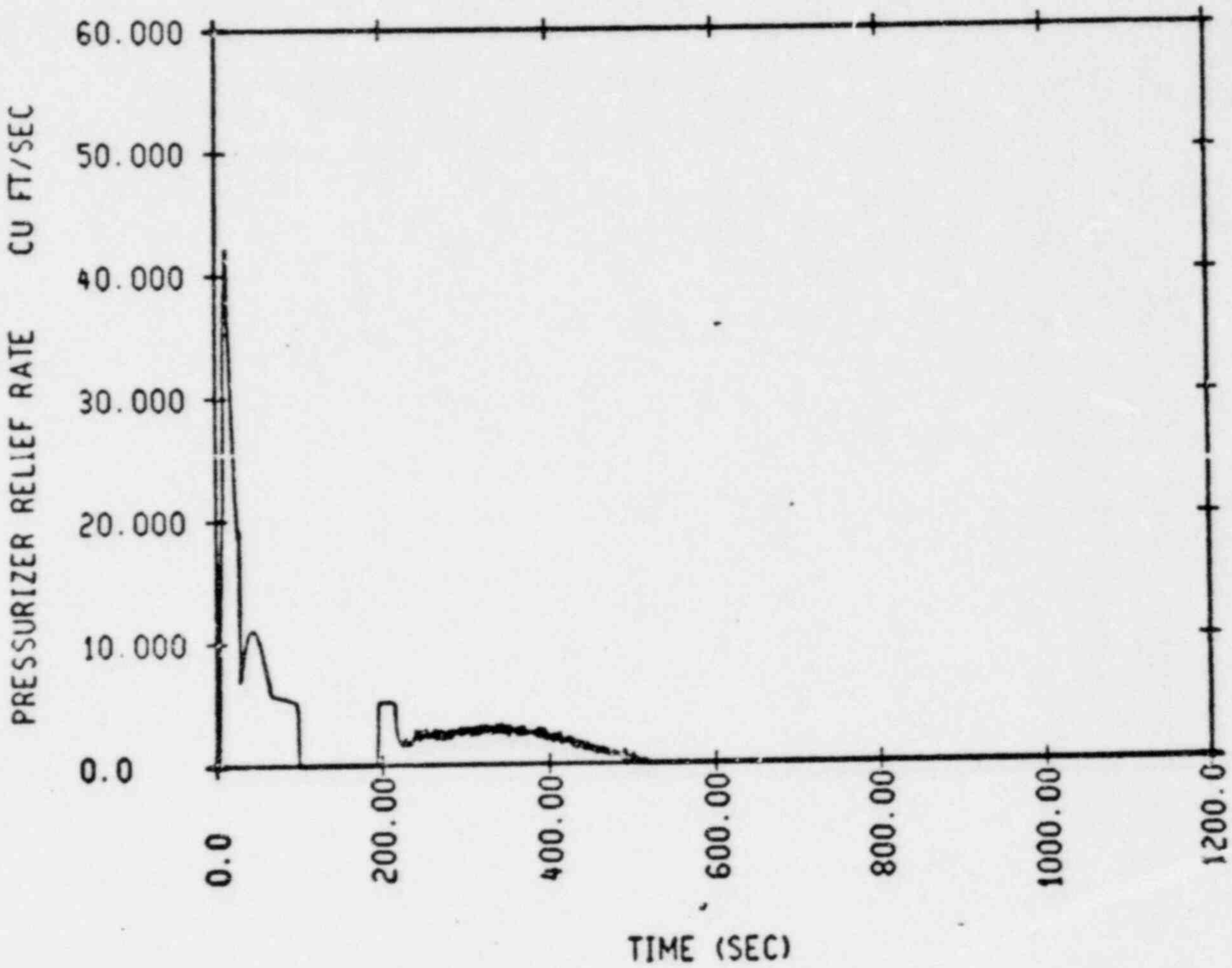


Figure 36. Blackout ATWS With 10 Percent Pressure Accumulation for Water Relief

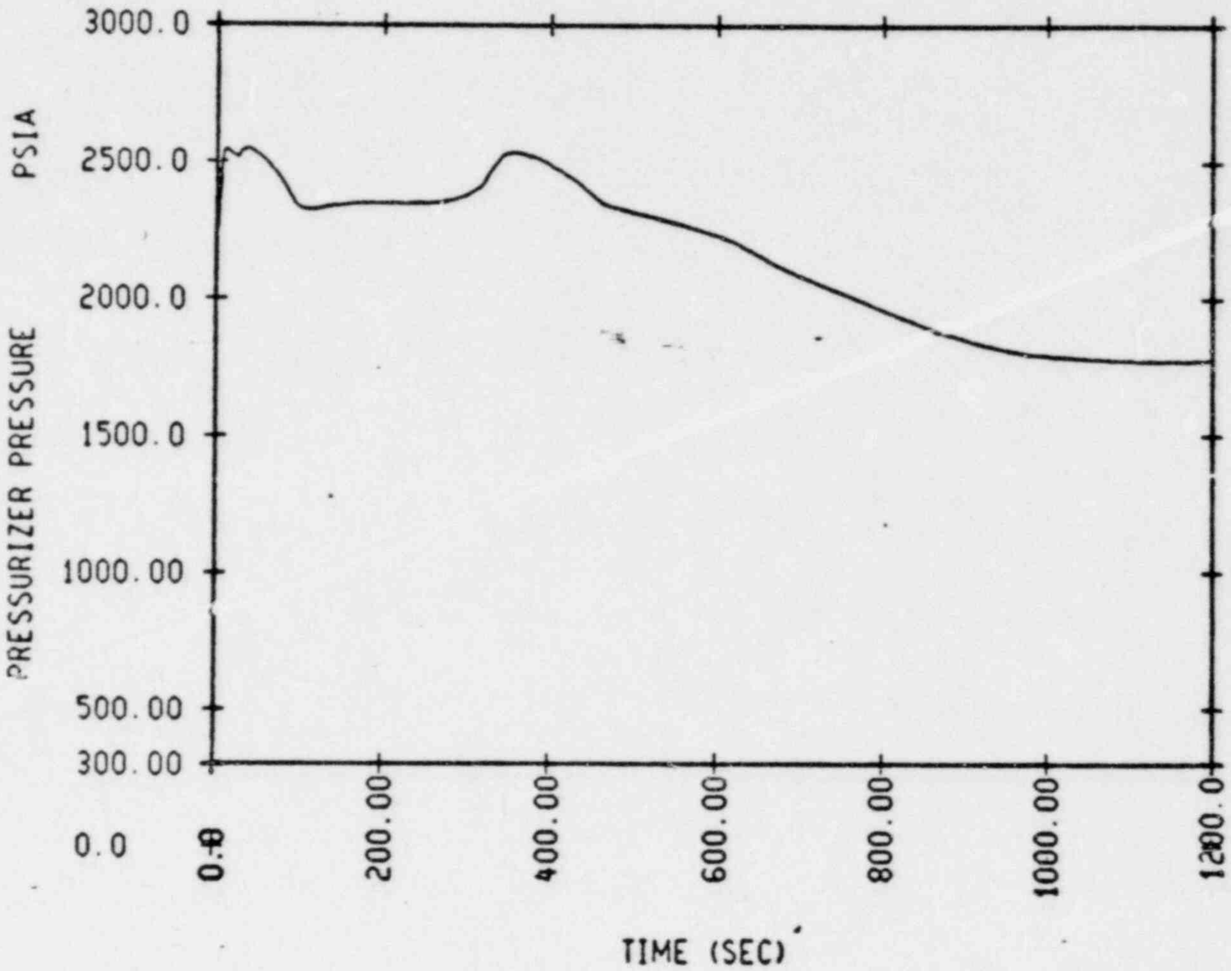


Figure 37. Blackout ATWS - Turbine-Driven Aux. FW Pump Fails

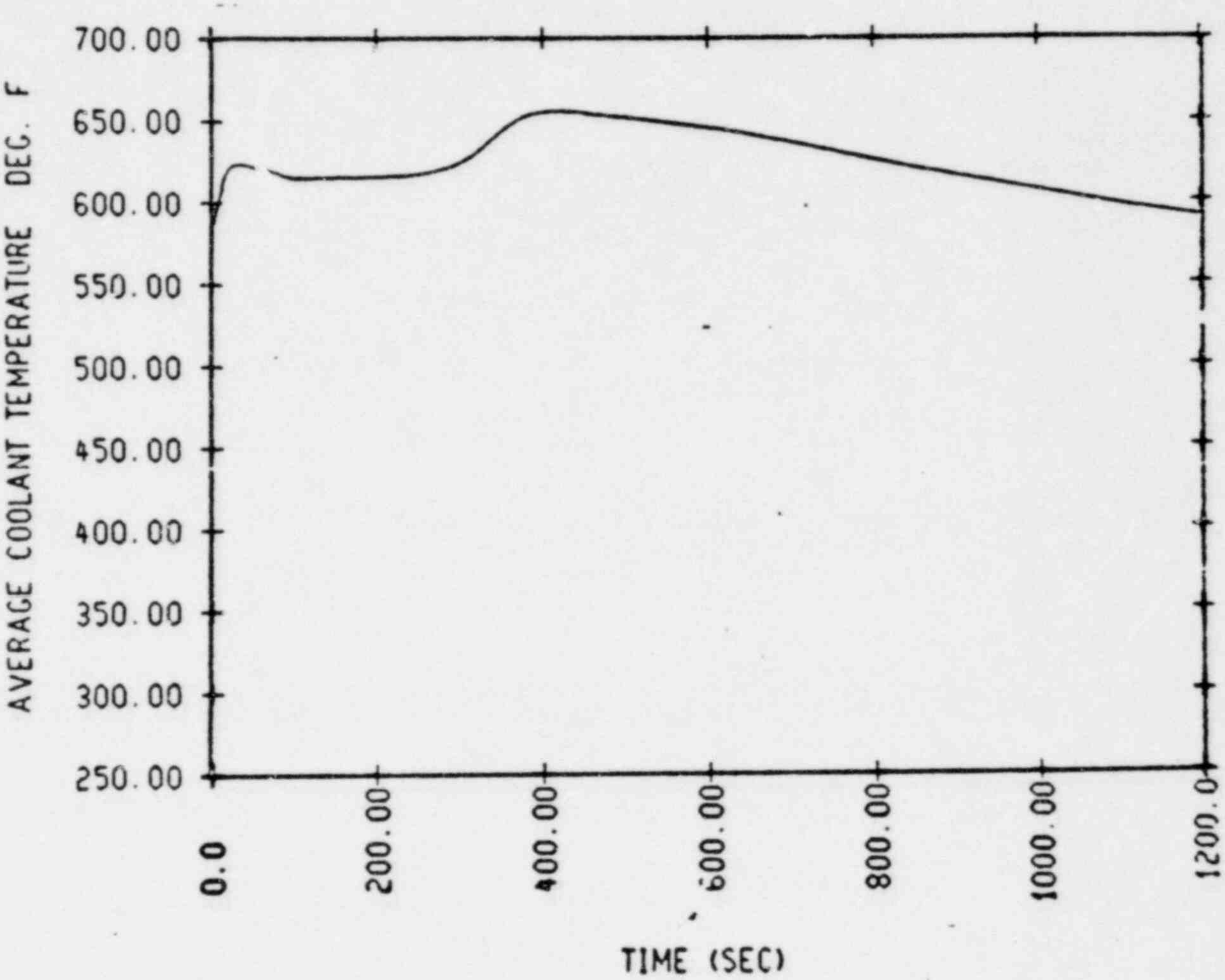


Figure 38. Blackout ATWS - Turbine-Driven Aux. FW Pump Fails

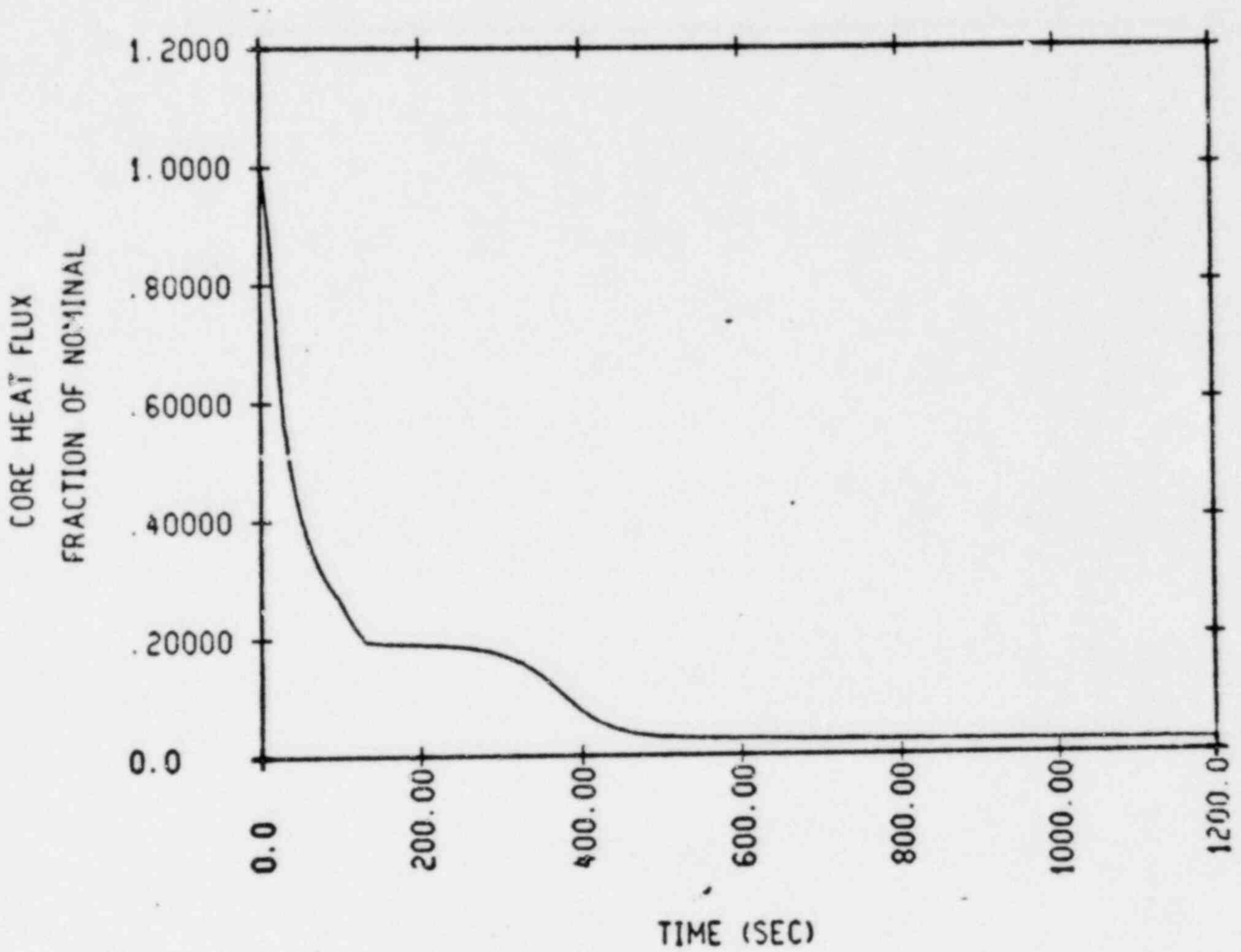


Figure 39. Blackout ATWS - Turbine-Driven Aux. FW Pump Fails

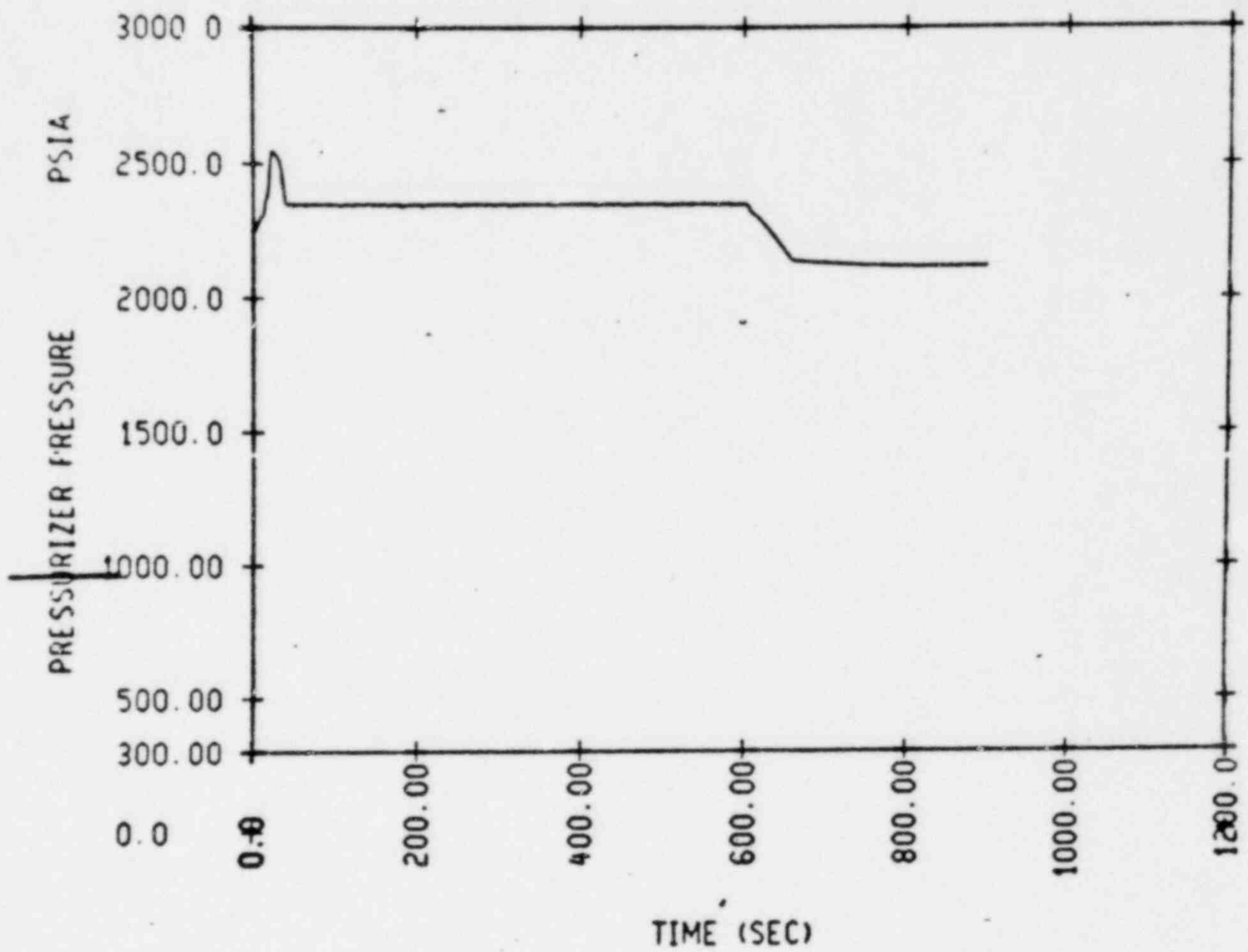


Figure 40. RWAP ATWS With Turbine Trip

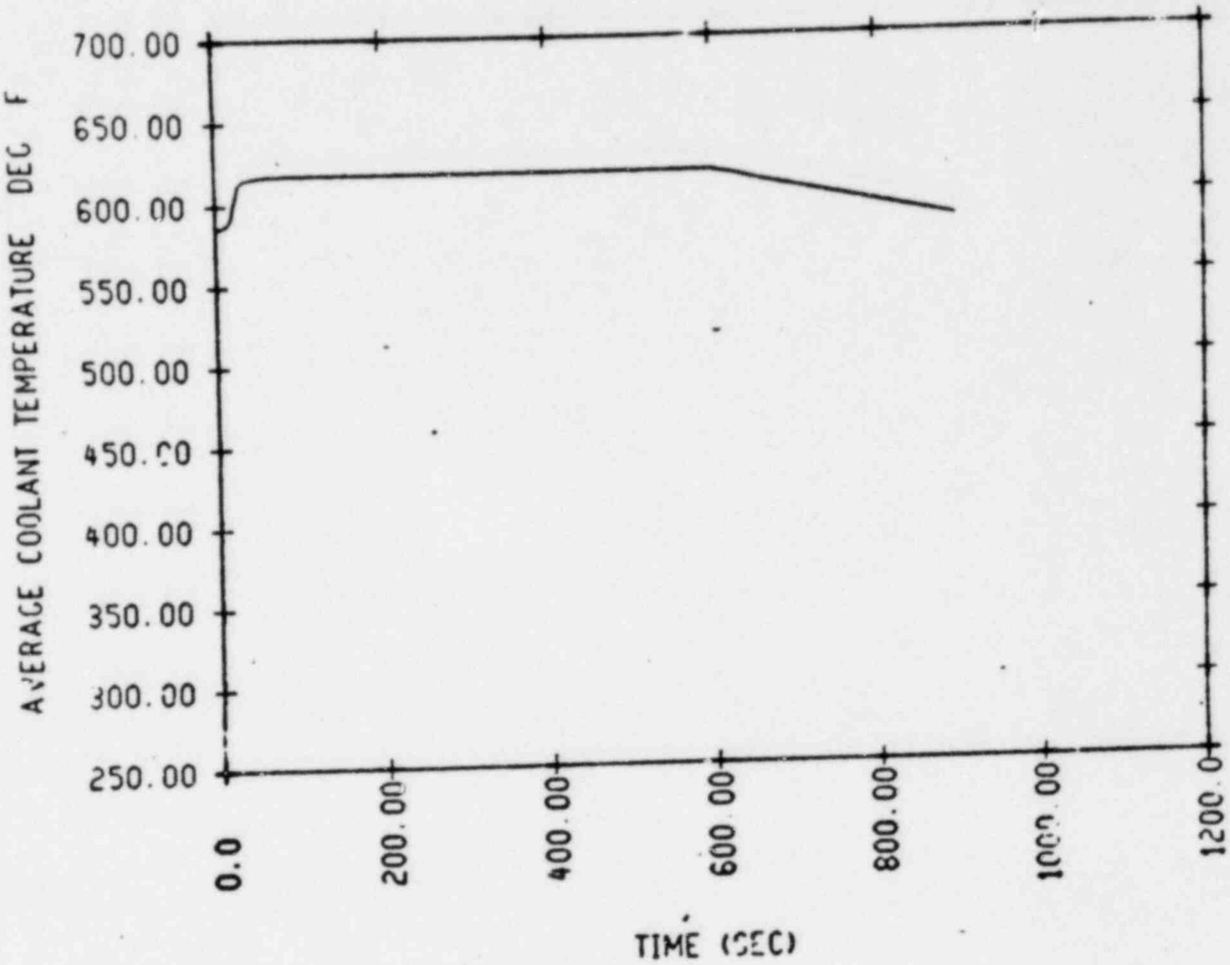


Figure 41. RWAP ATWS With Turbine Trip

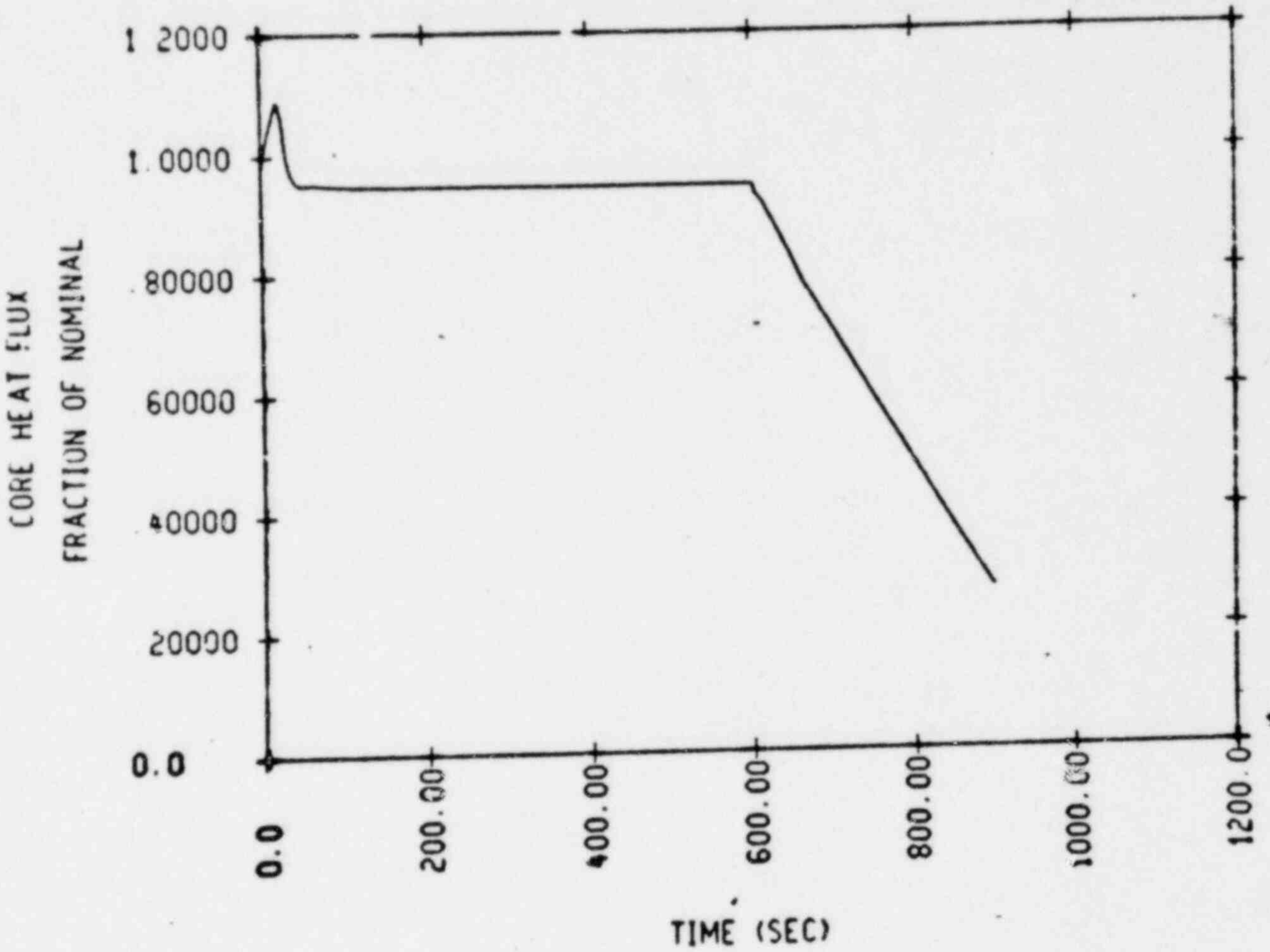


Figure 42. RWAP ATWS with Turbine Trip

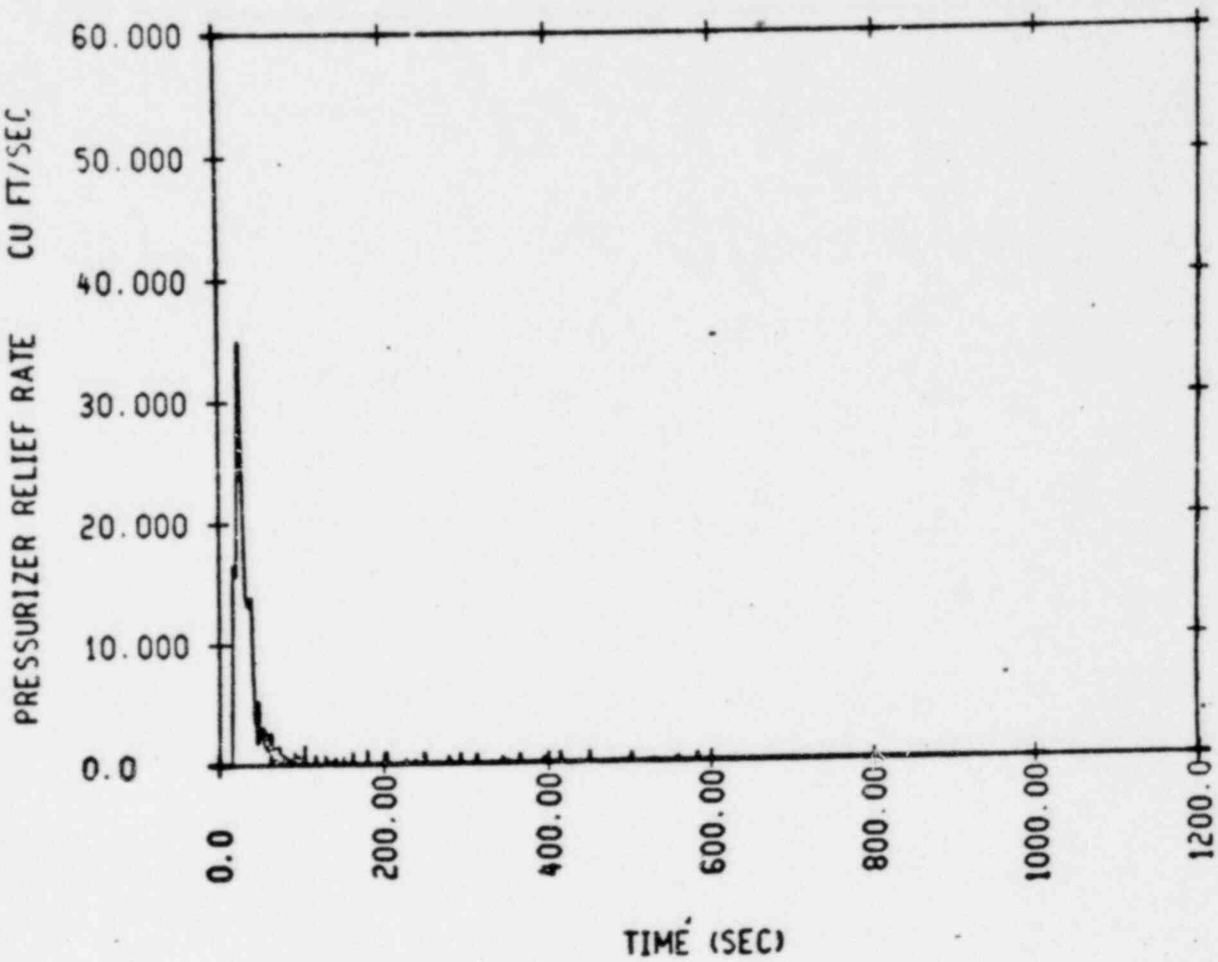


Figure 43. RWAP ATWS with Turbine Trip

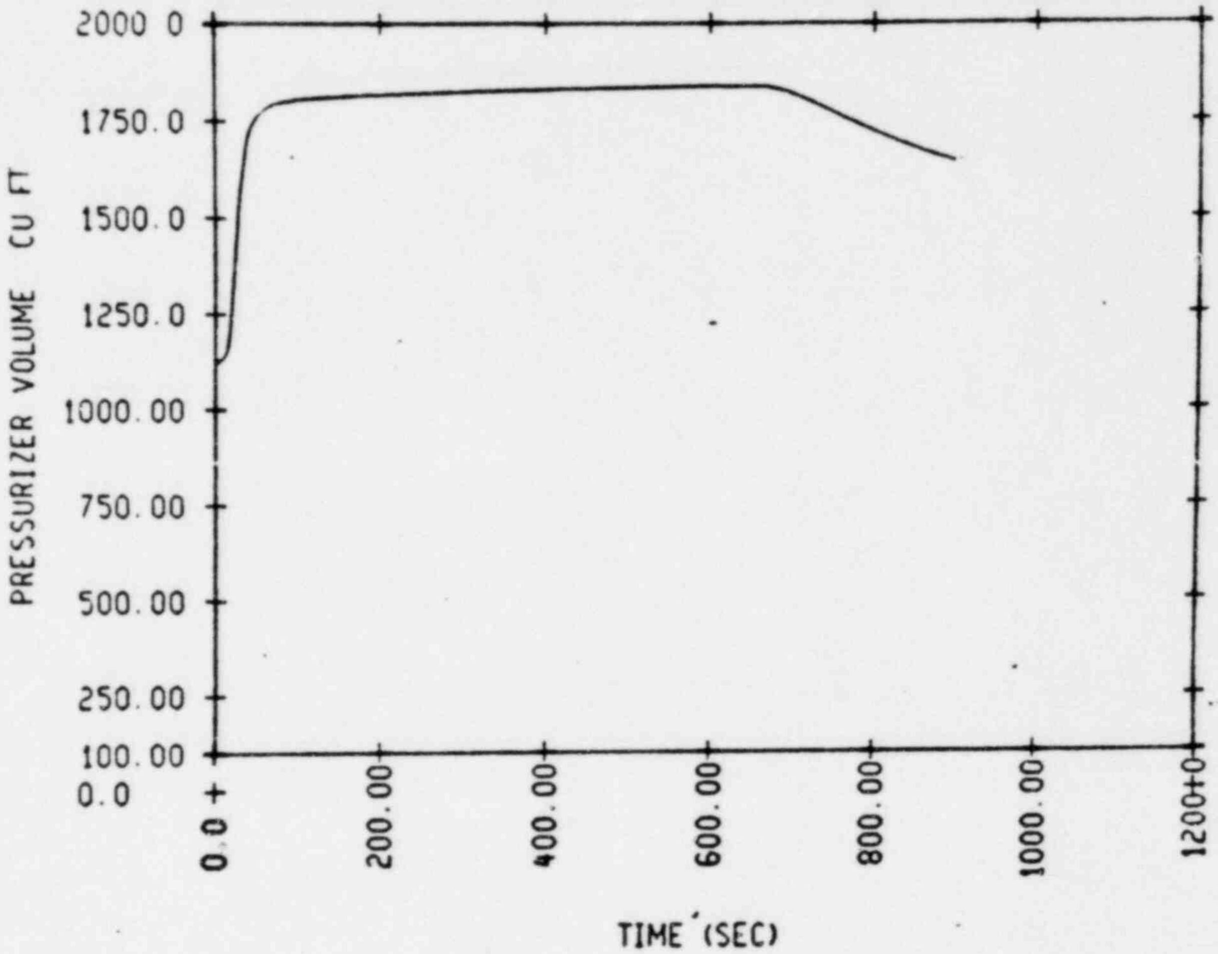


Figure 44. RWAP ATWS with Turbine Trip

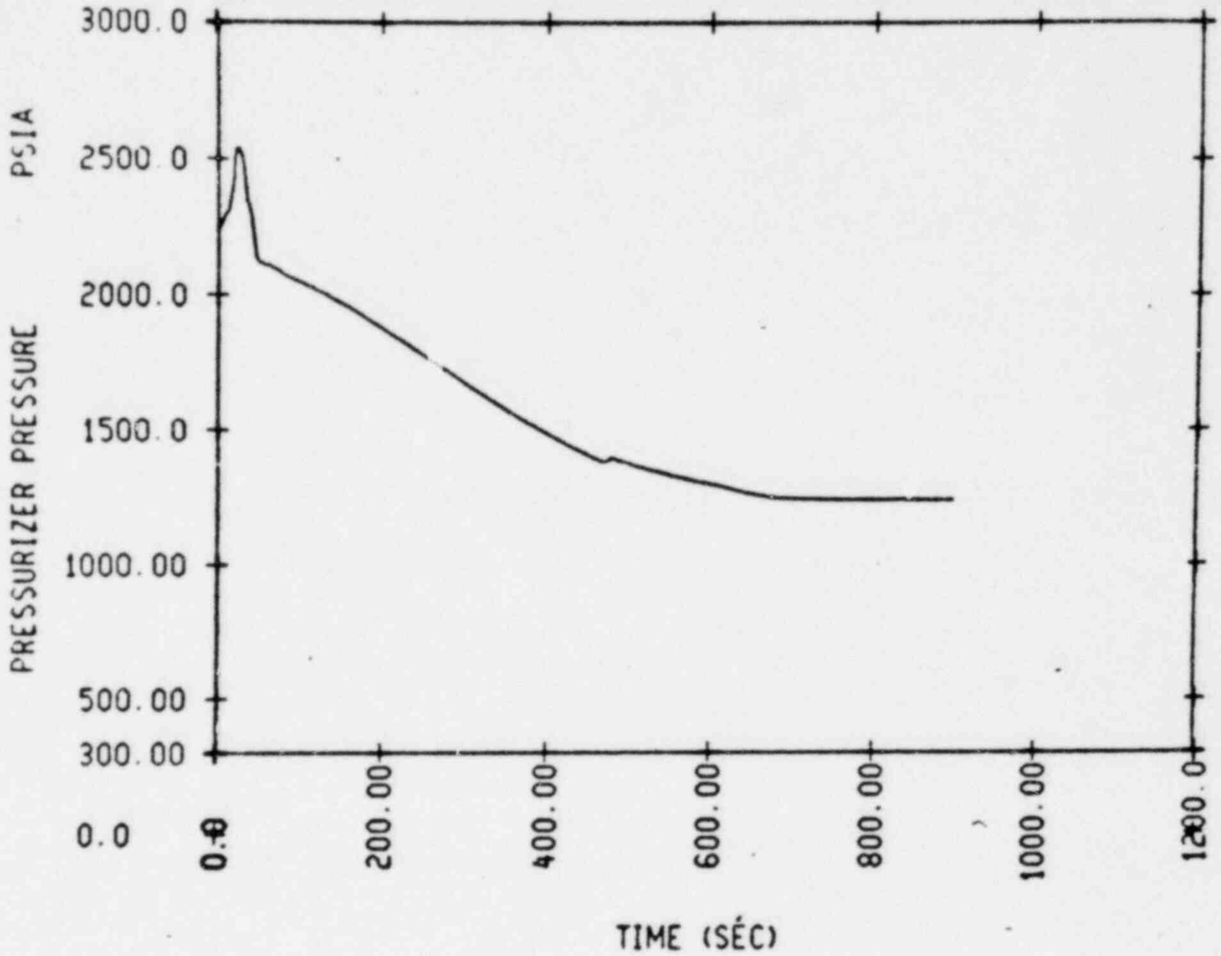


Figure 45. RWAP ATWS With Failure of Safety Valve to Re-Seat

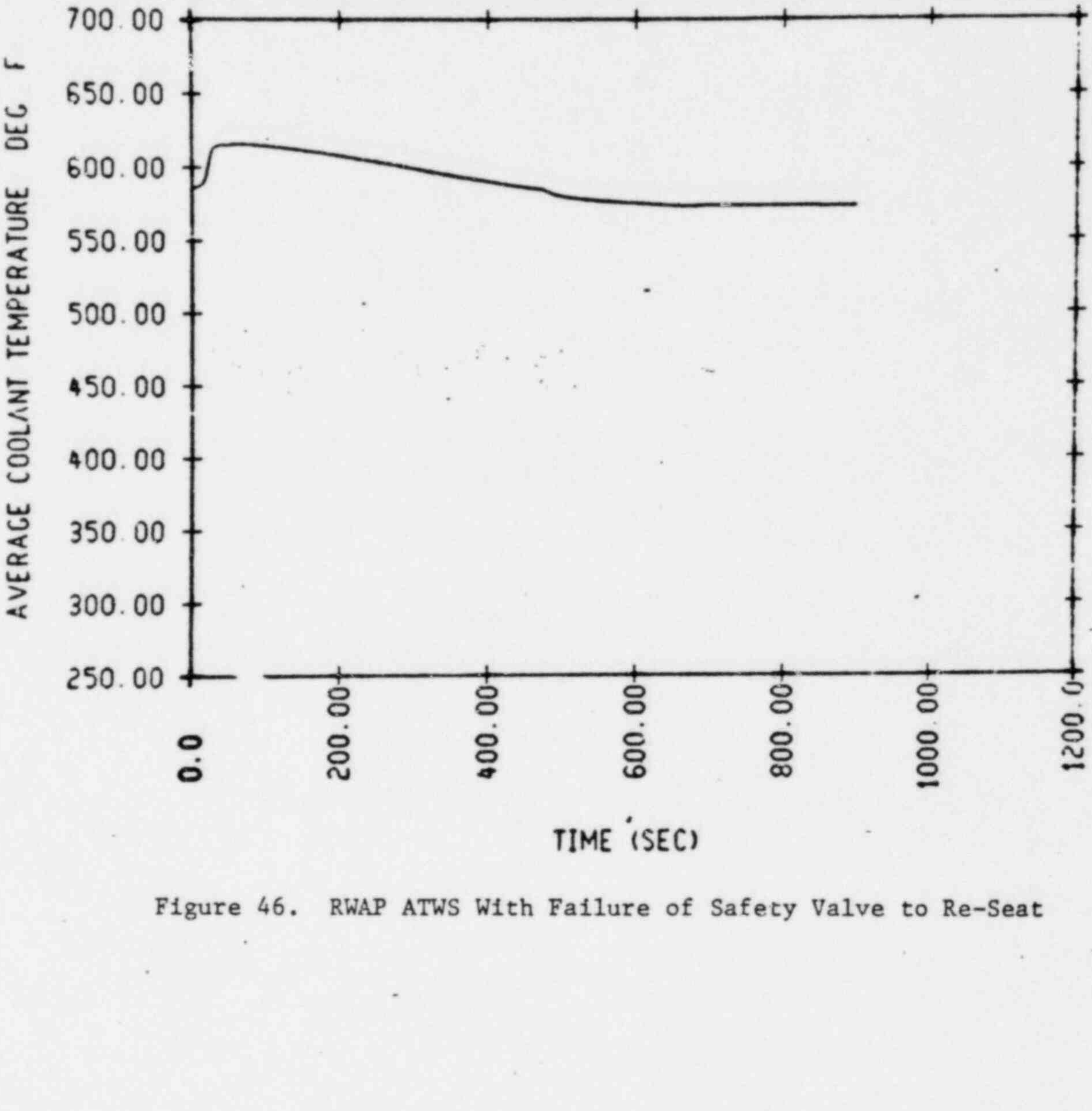


Figure 46. RWAP ATWS With Failure of Safety Valve to Re-Seat

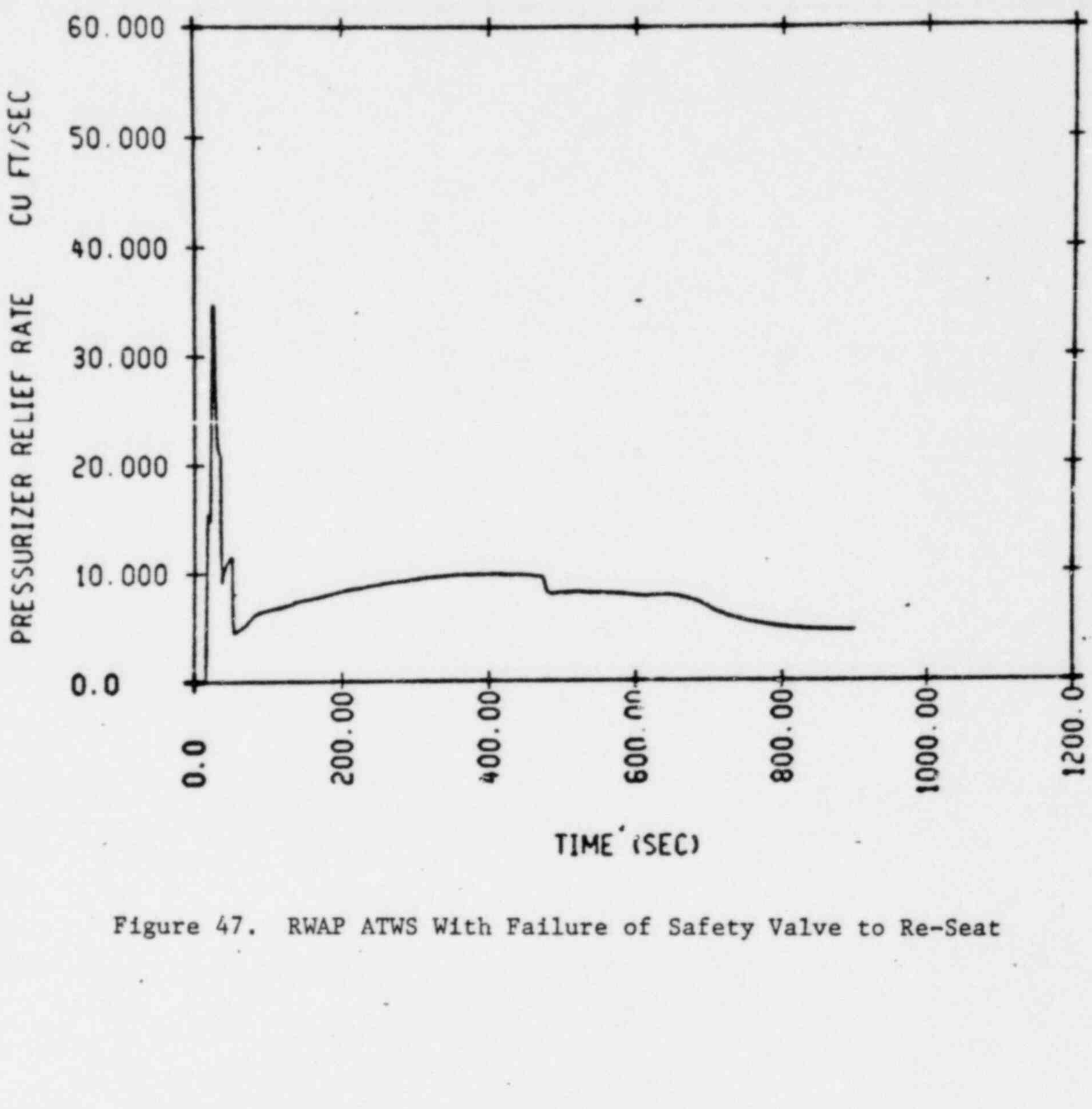


Figure 47. RWAP ATWS With Failure of Safety Valve to Re-Seat

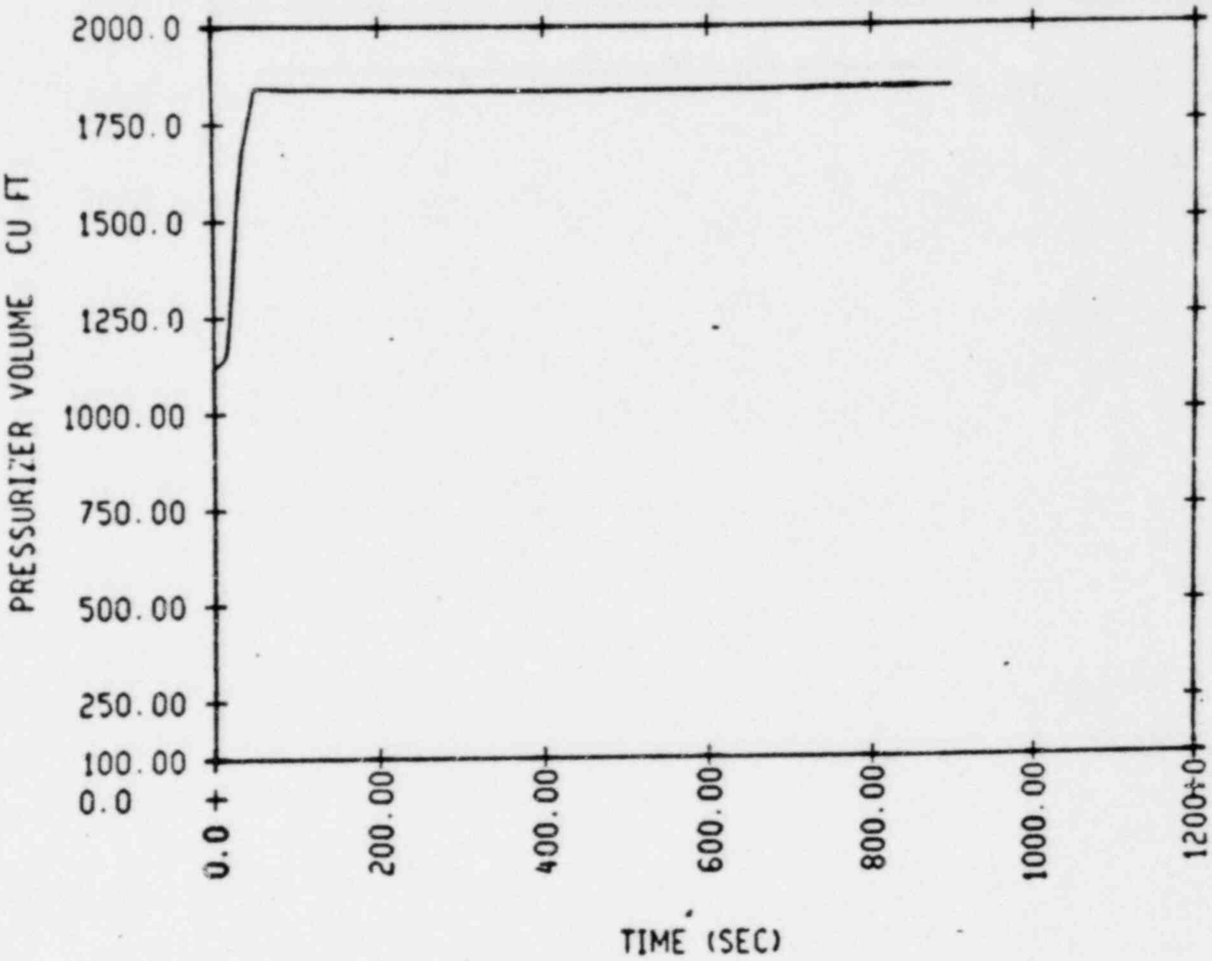


Figure 48. RWAP ATWS With Failure of Safety Valve to Re-Seat

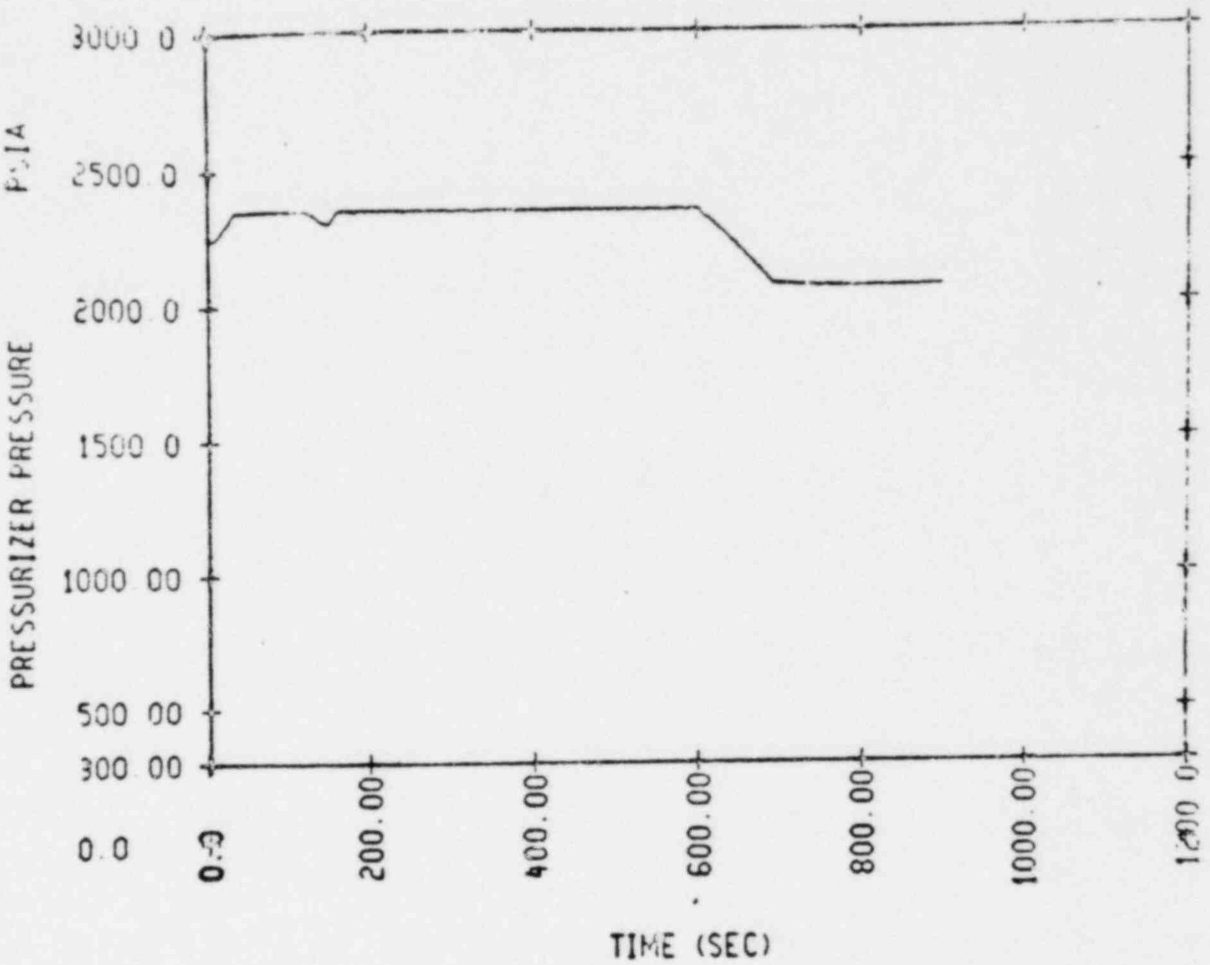


Figure 49. Rod withdrawal at 50% power

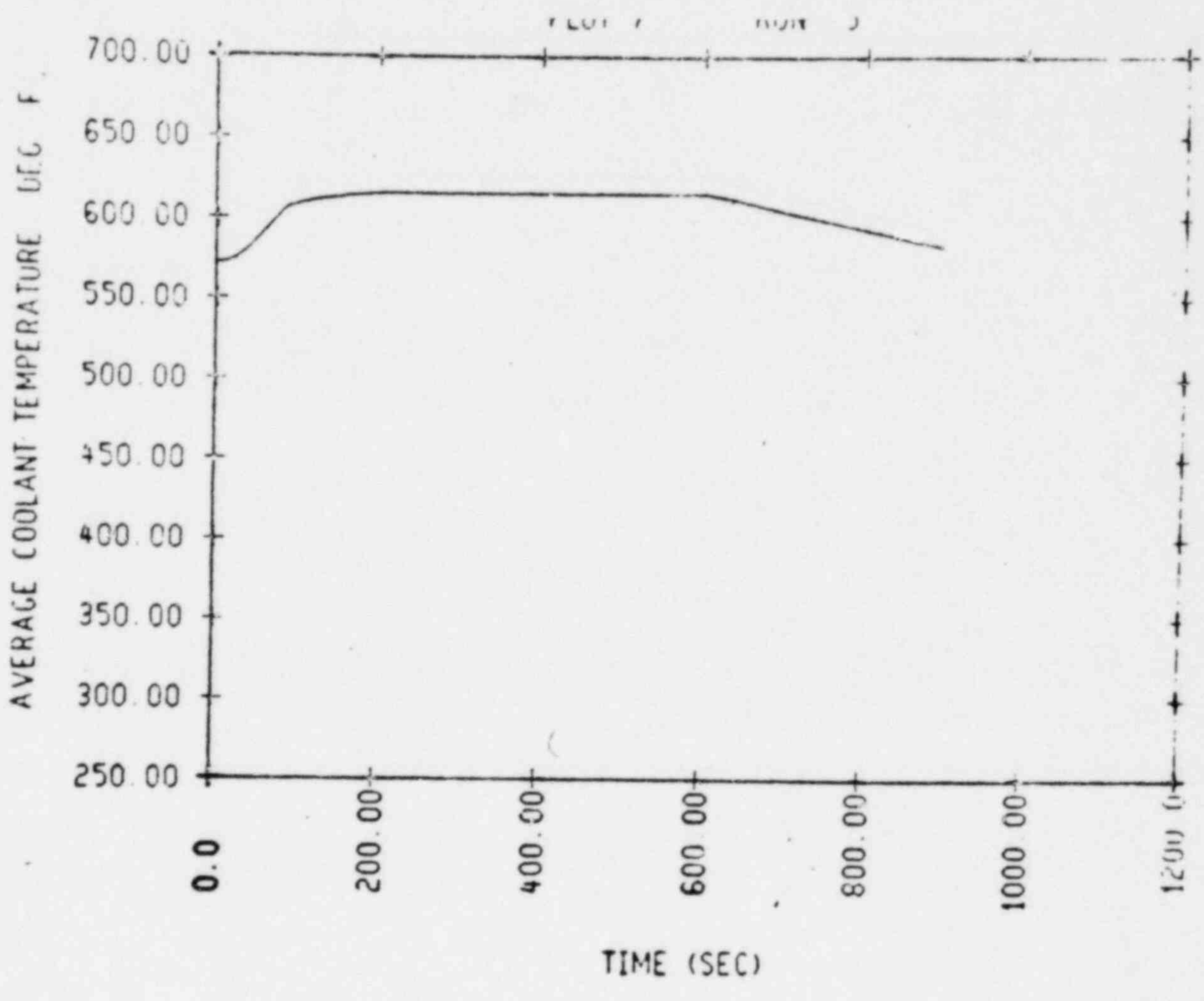


Figure 50. Rod withdrawal at 50% power

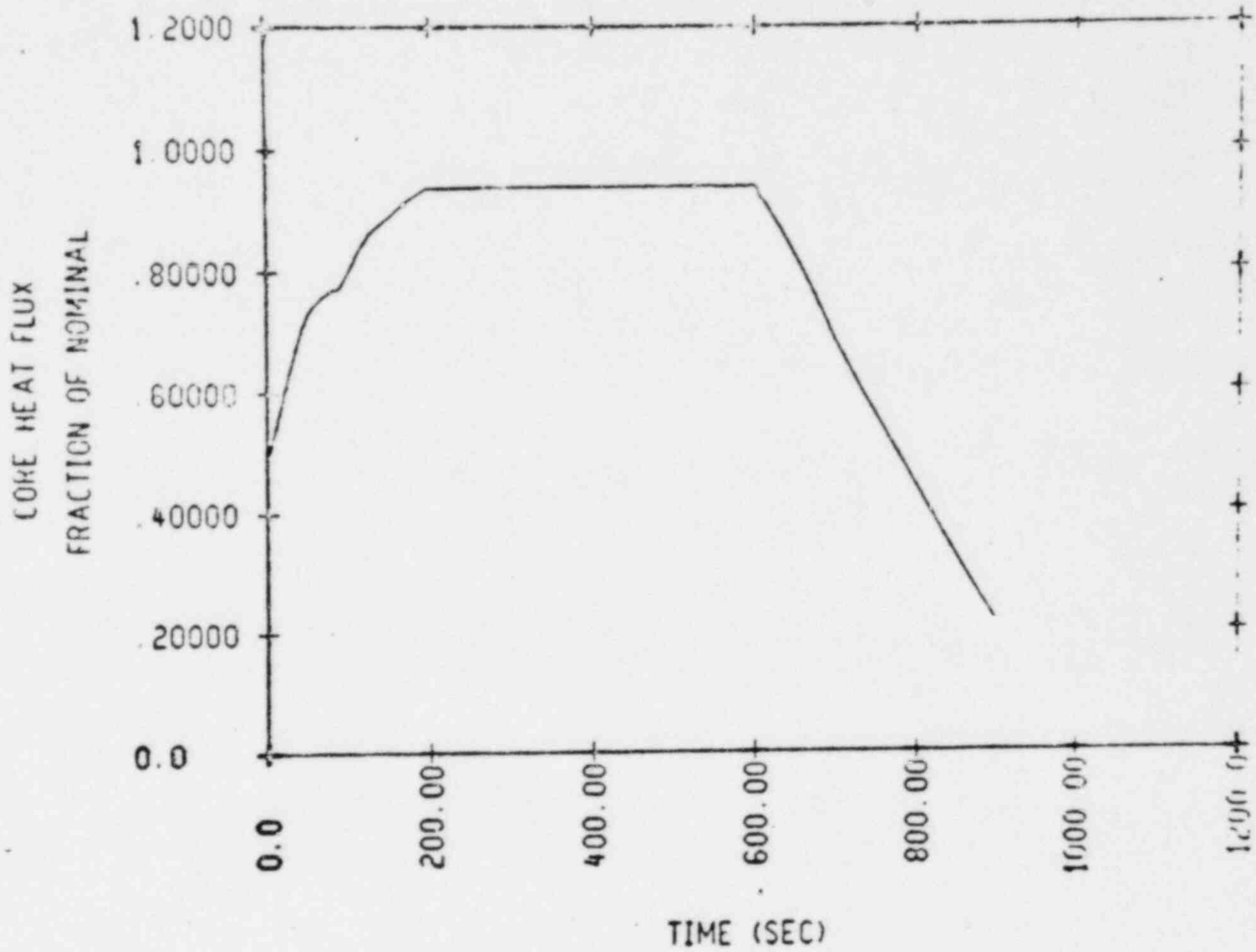


Figure 5]. Rod withdrawal at 50% power

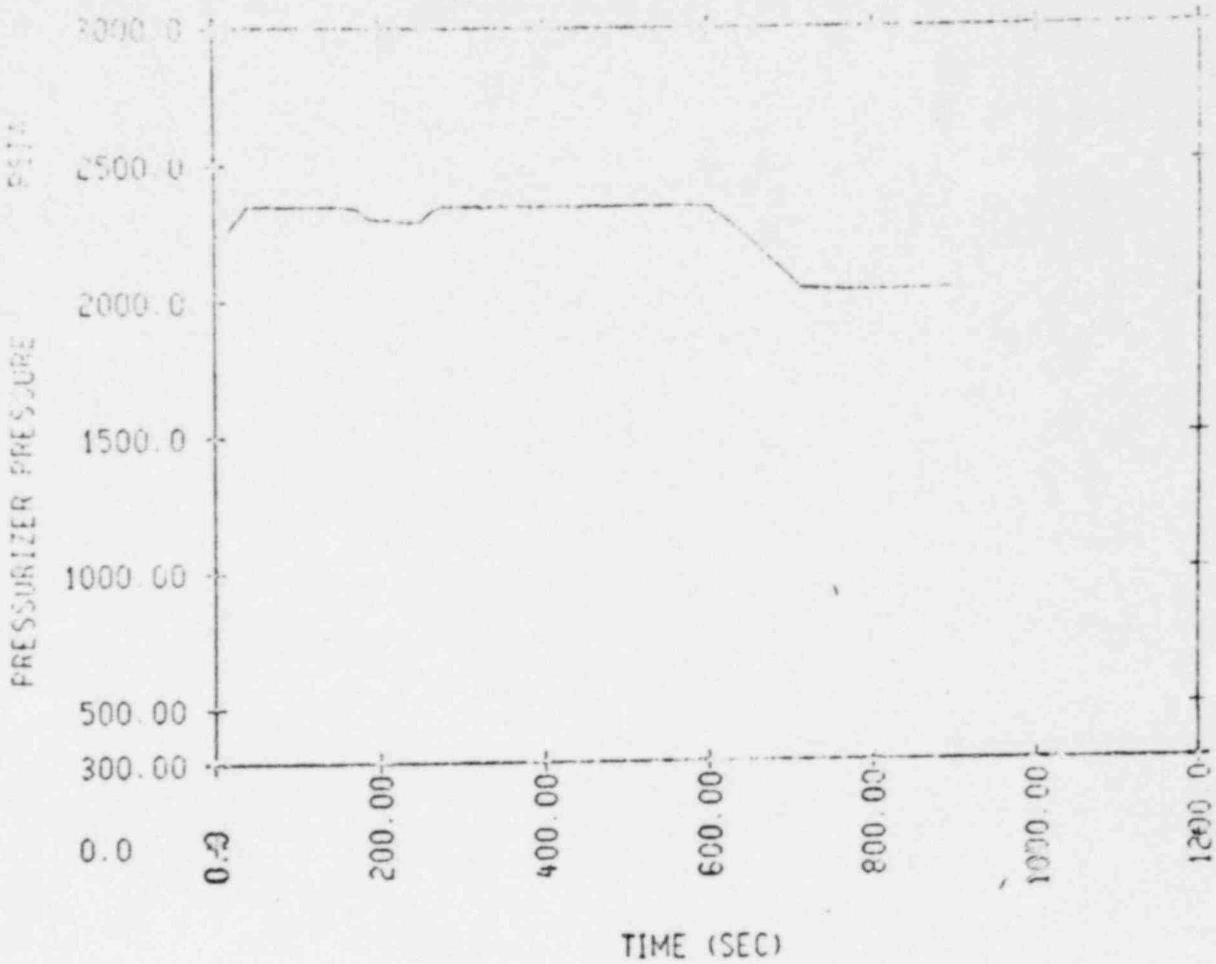


Figure 52. Rod withdrawal at 25% power

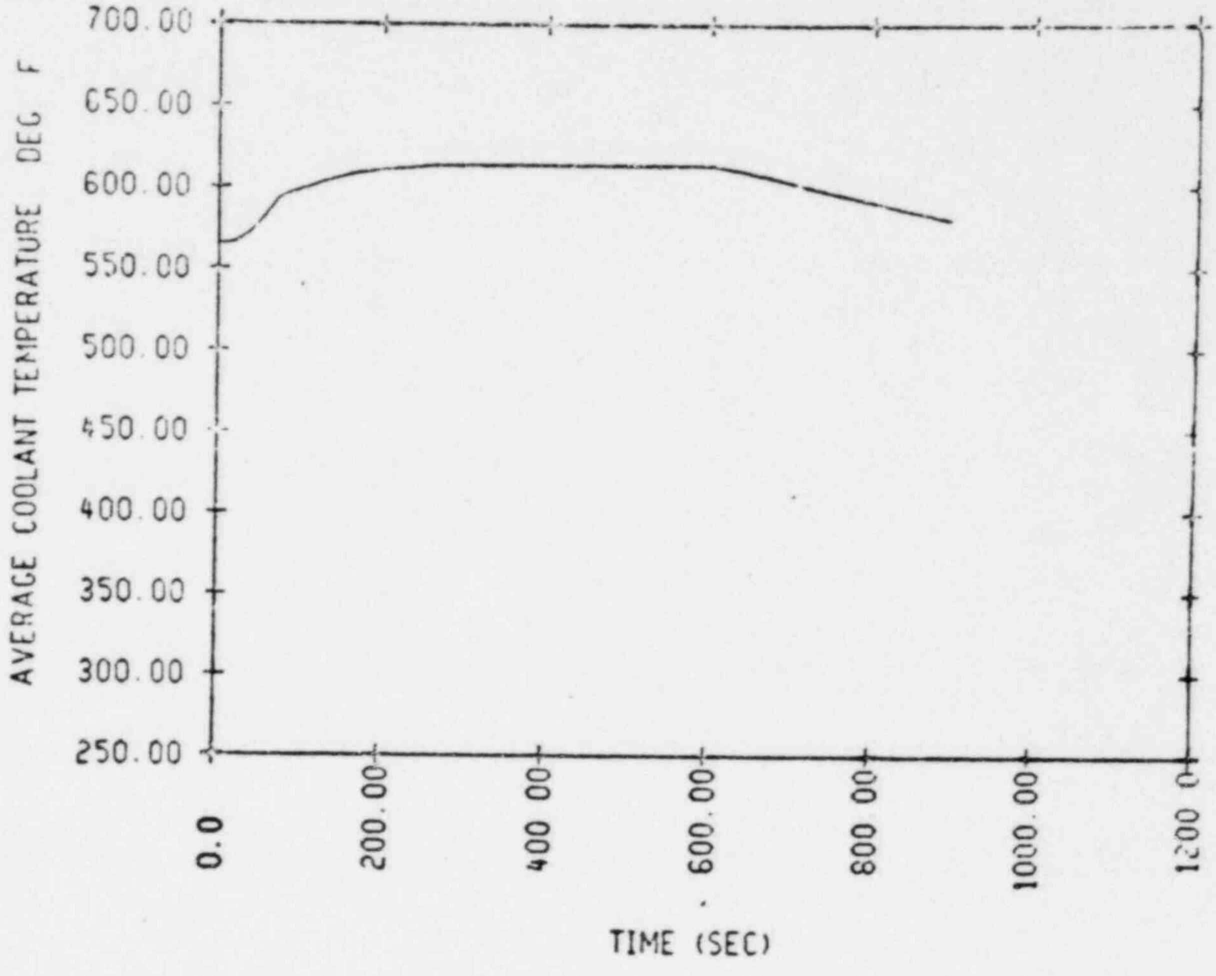


Figure 53. RWAP ATWS at 25% power

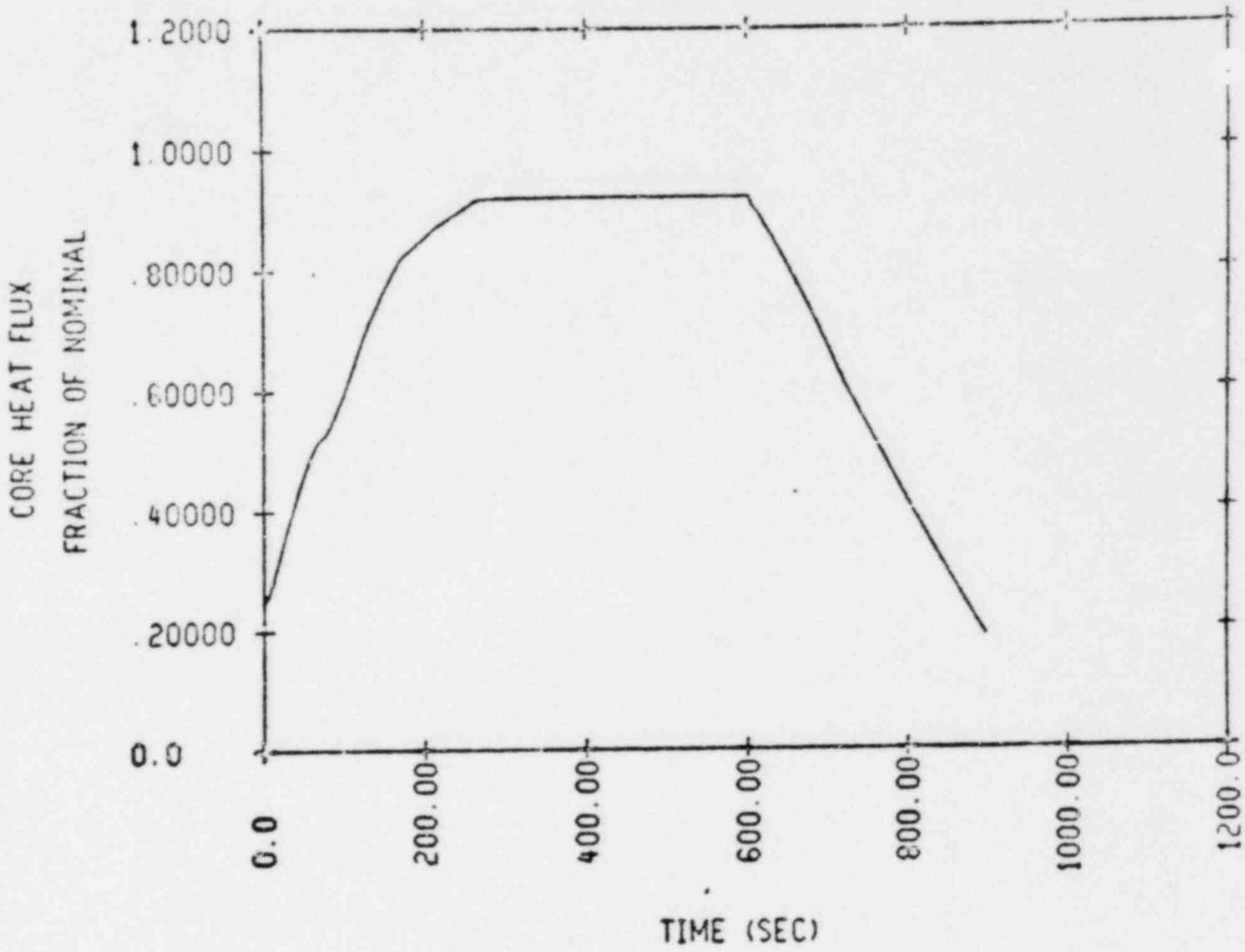


Figure 54. Rod Withdrawal at 25% power

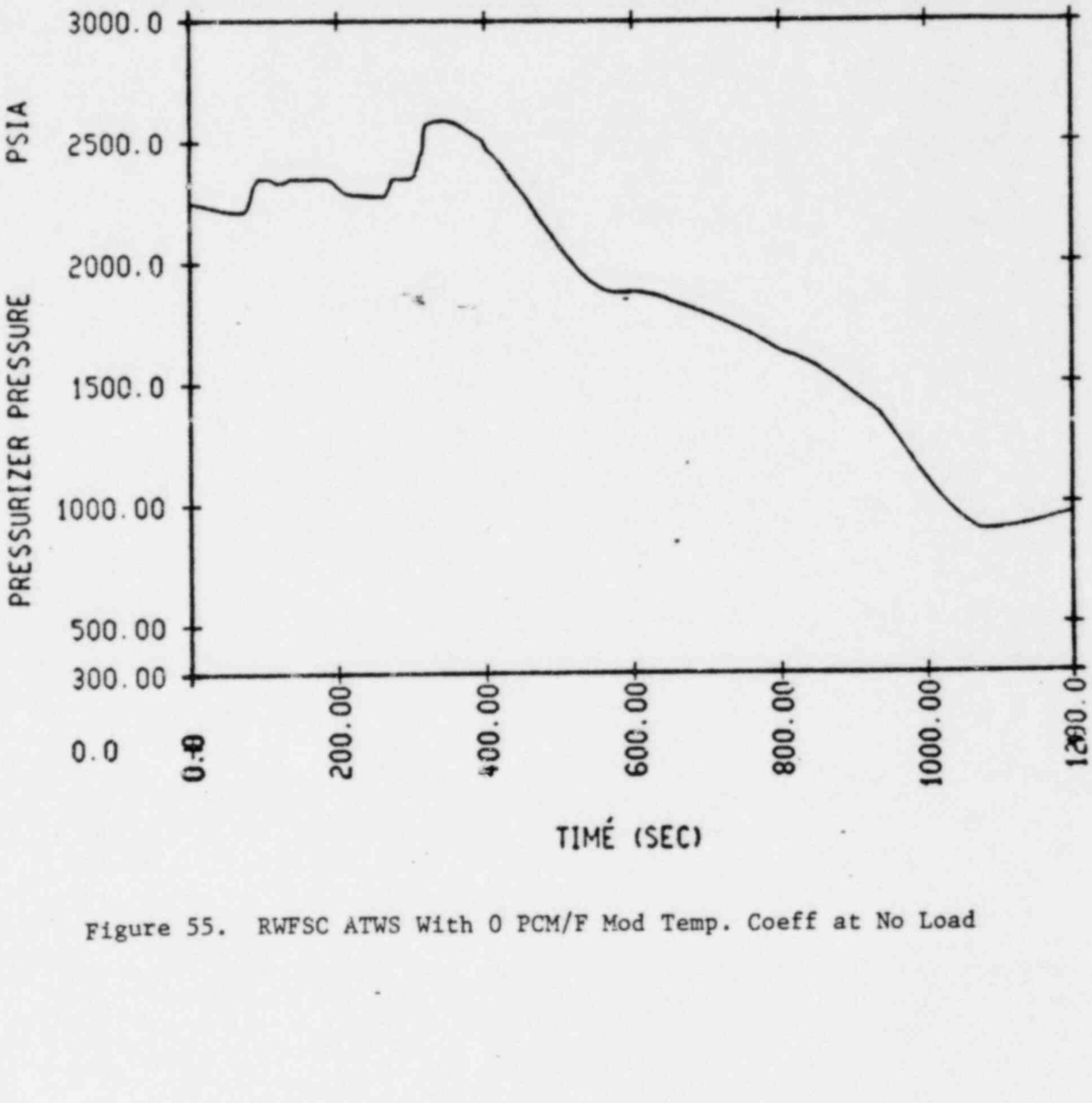


Figure 55. RWFSC ATWS With 0 PCM/F Mod Temp. Coeff at No Load

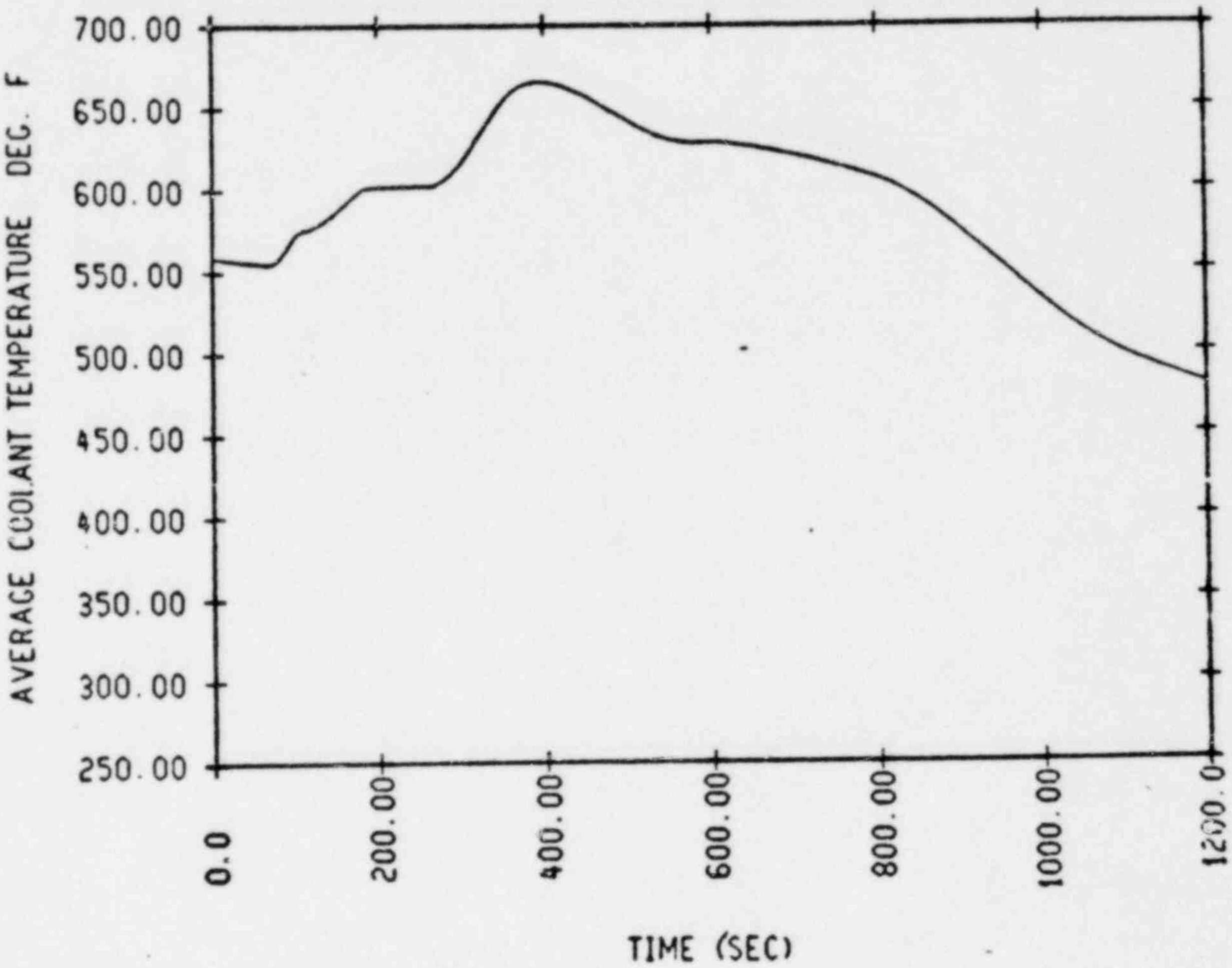


Figure 56. RWFSC ATWS With 0 PCM/F Mod Temp Coeff at No Load

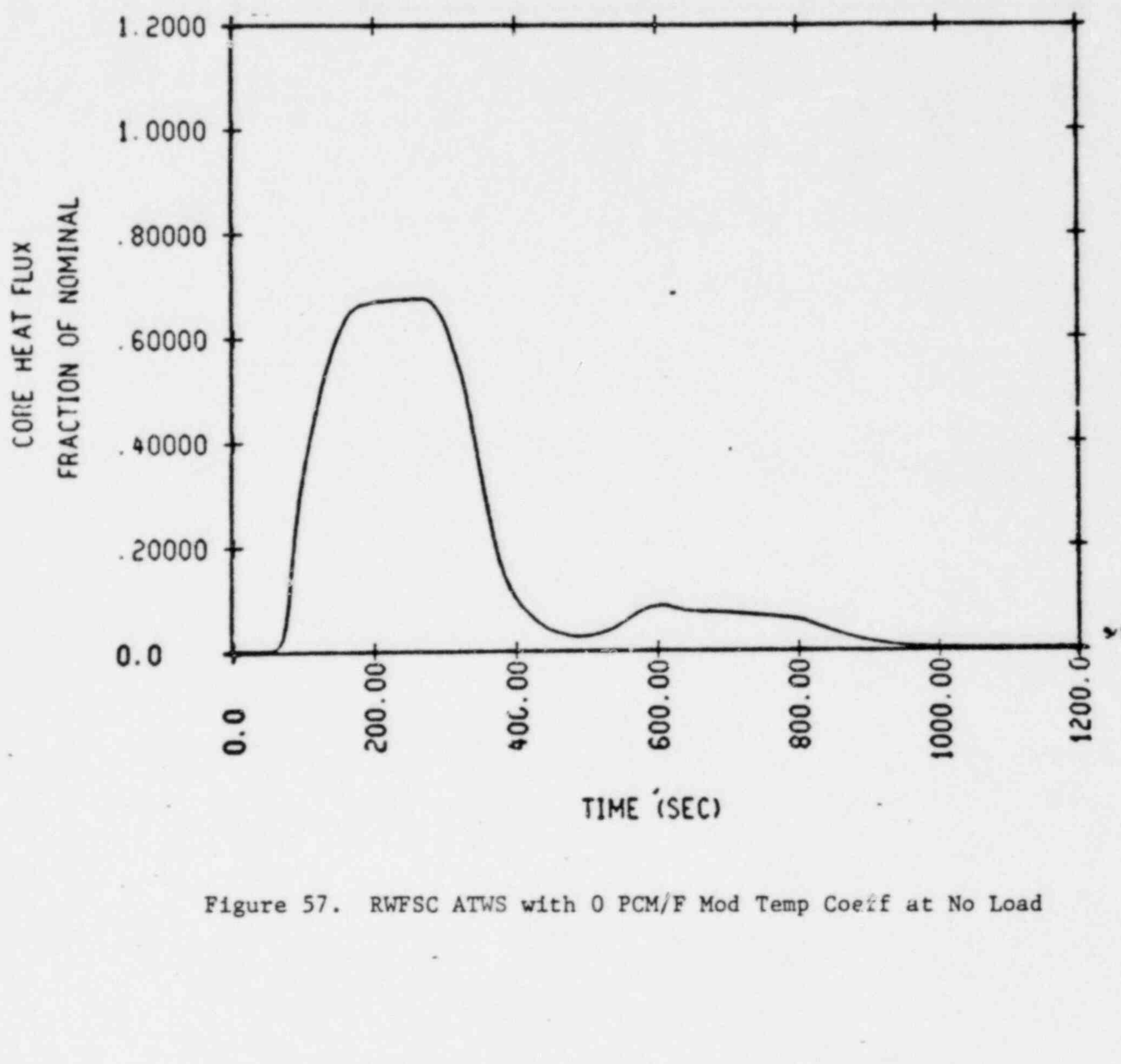


Figure 57. RWFSC ATWS with 0 PCM/F Mod Temp Coeff at No Load

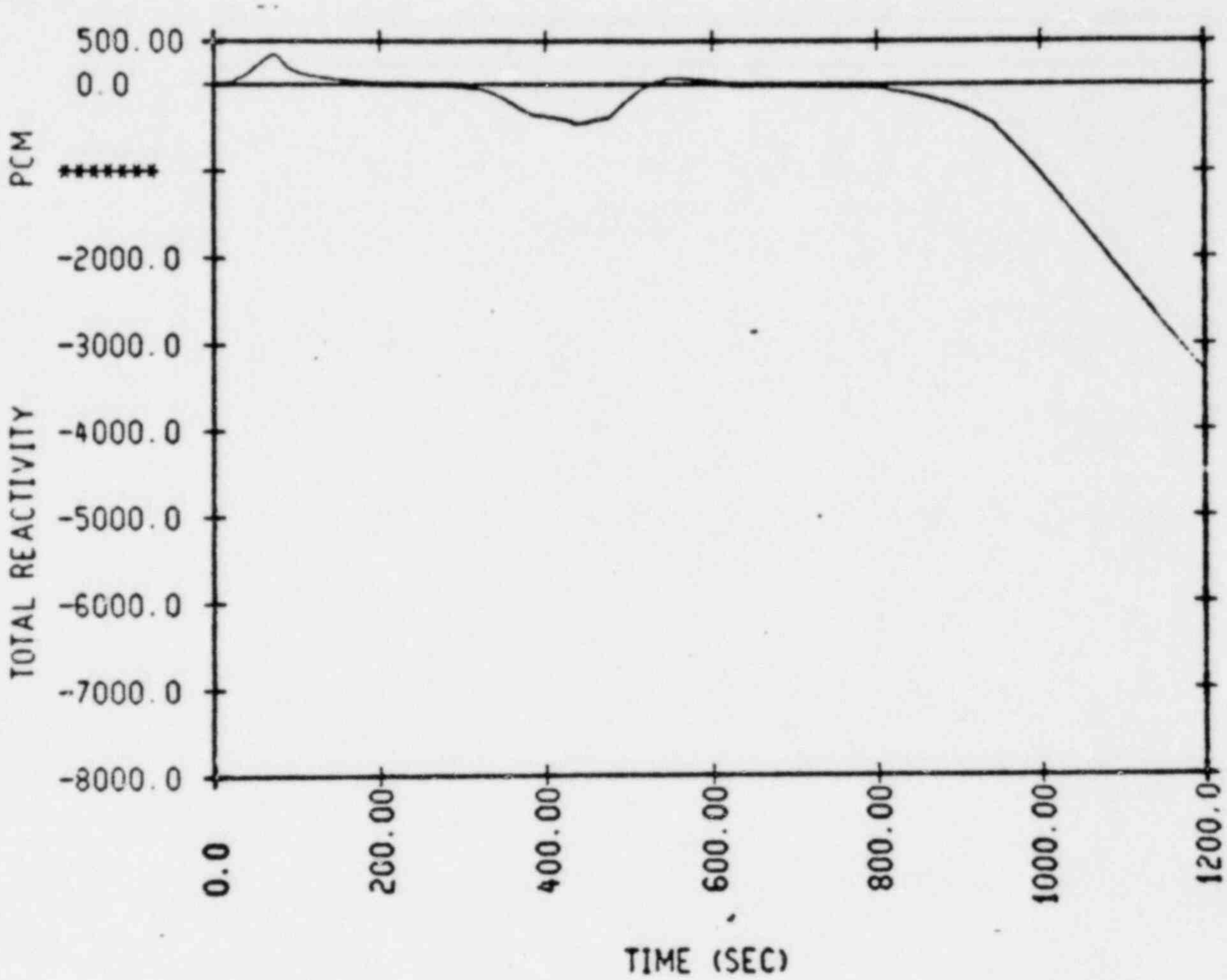


Figure 58. RWFSC ATWS with 0 PCM/F Mod Temp Coeff at No Load

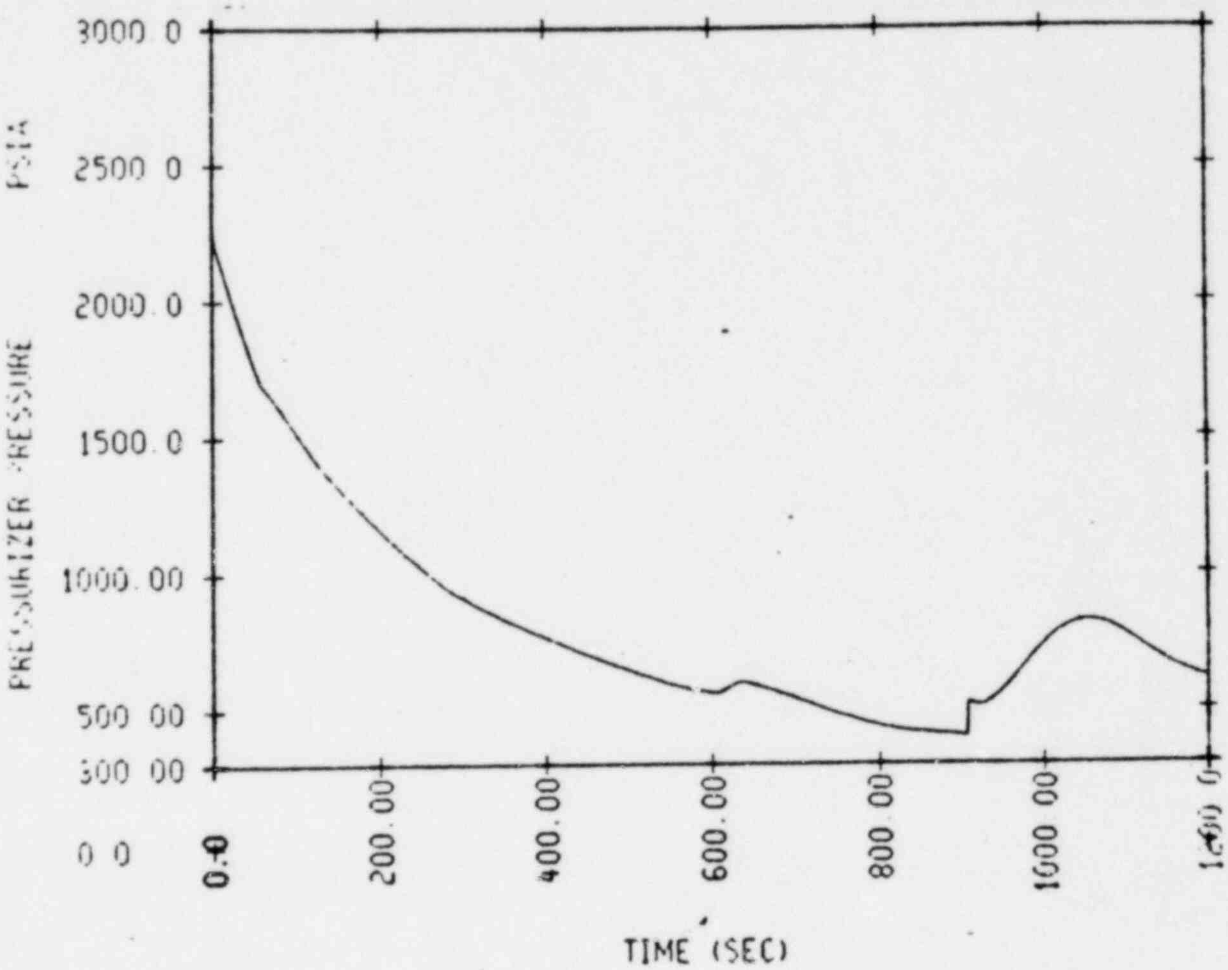


Figure 59. RCS Depressurization ATWS

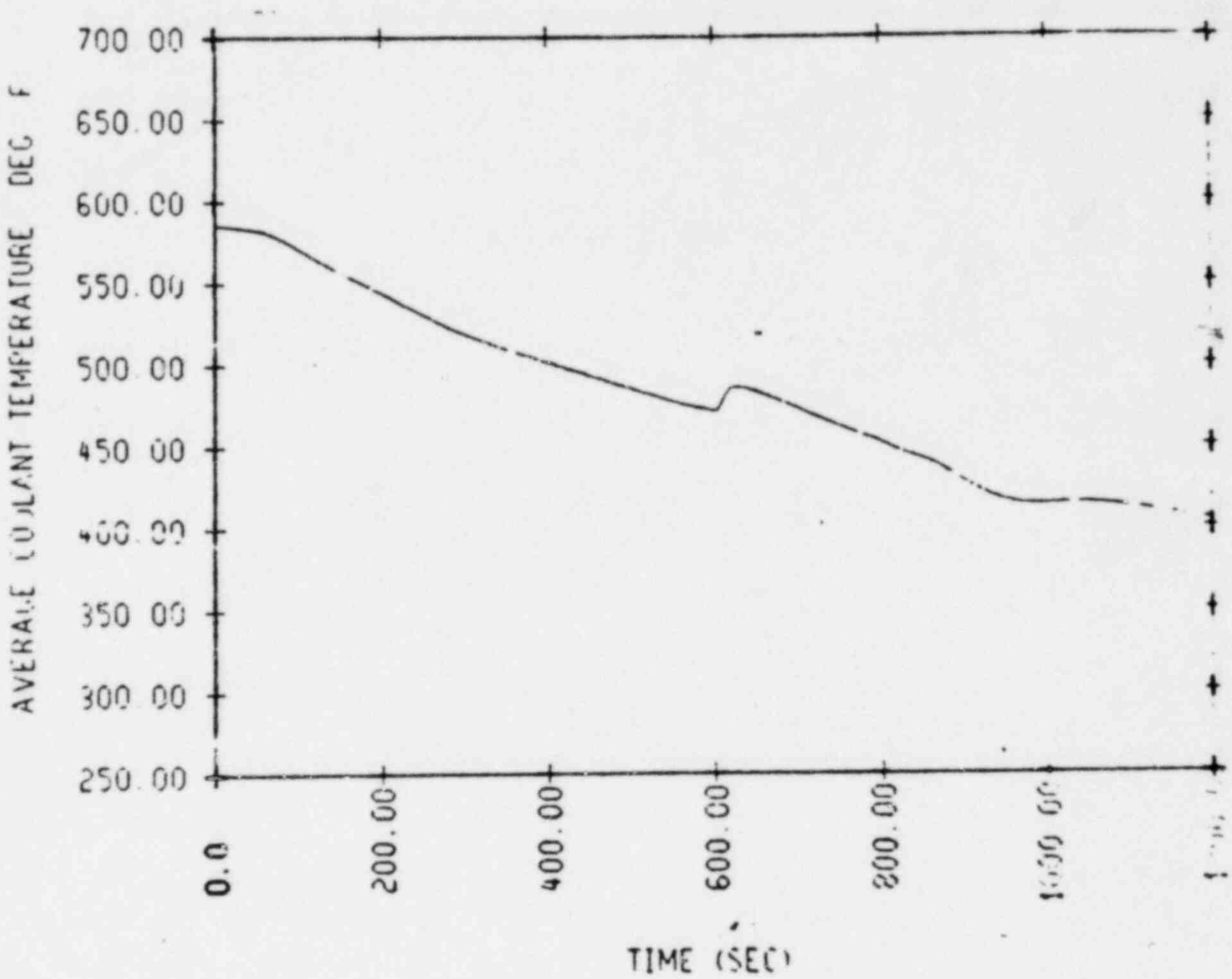


Figure 60. RCS Depressurization ATWS

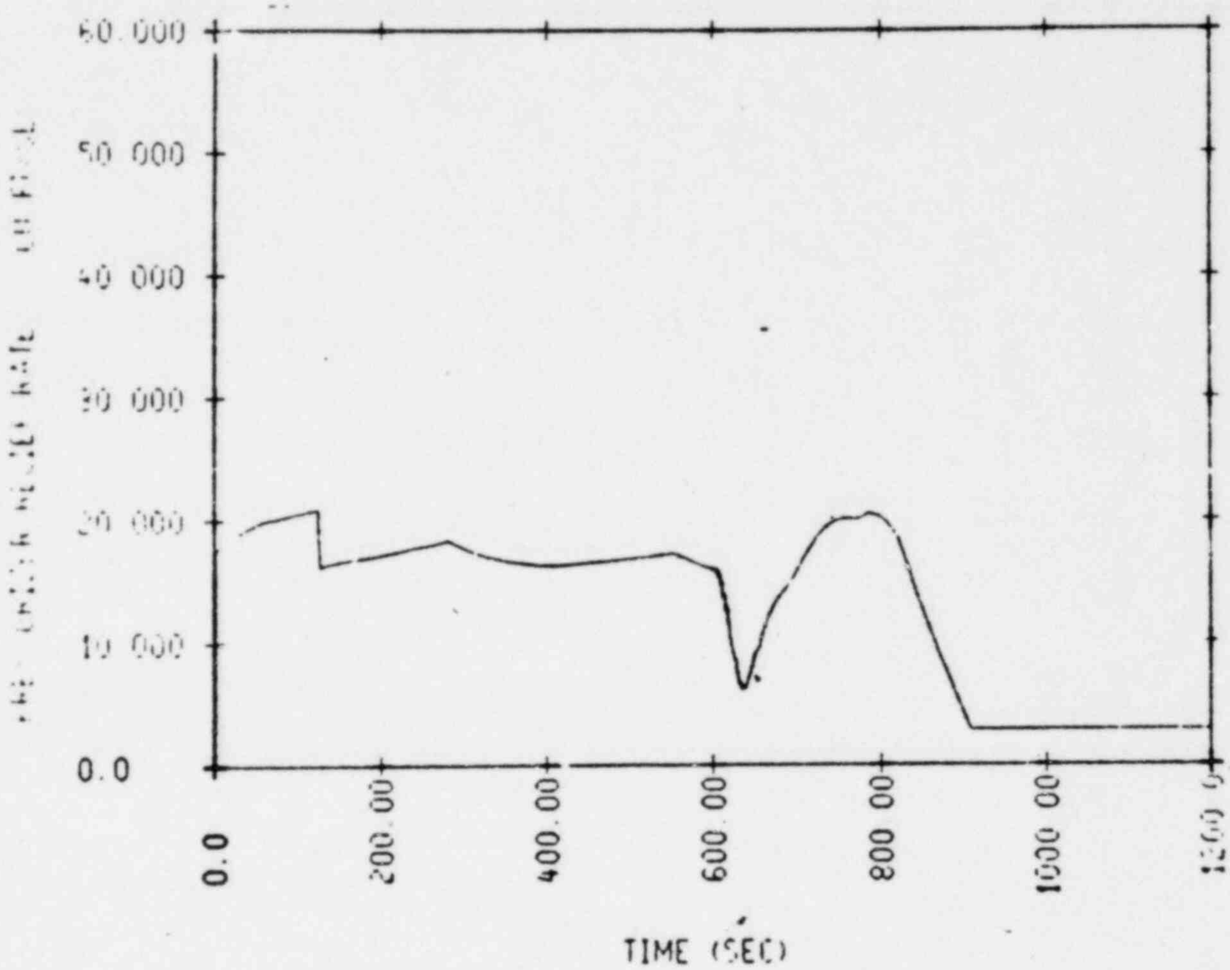


Figure 61. RCS Depressurization ATWS

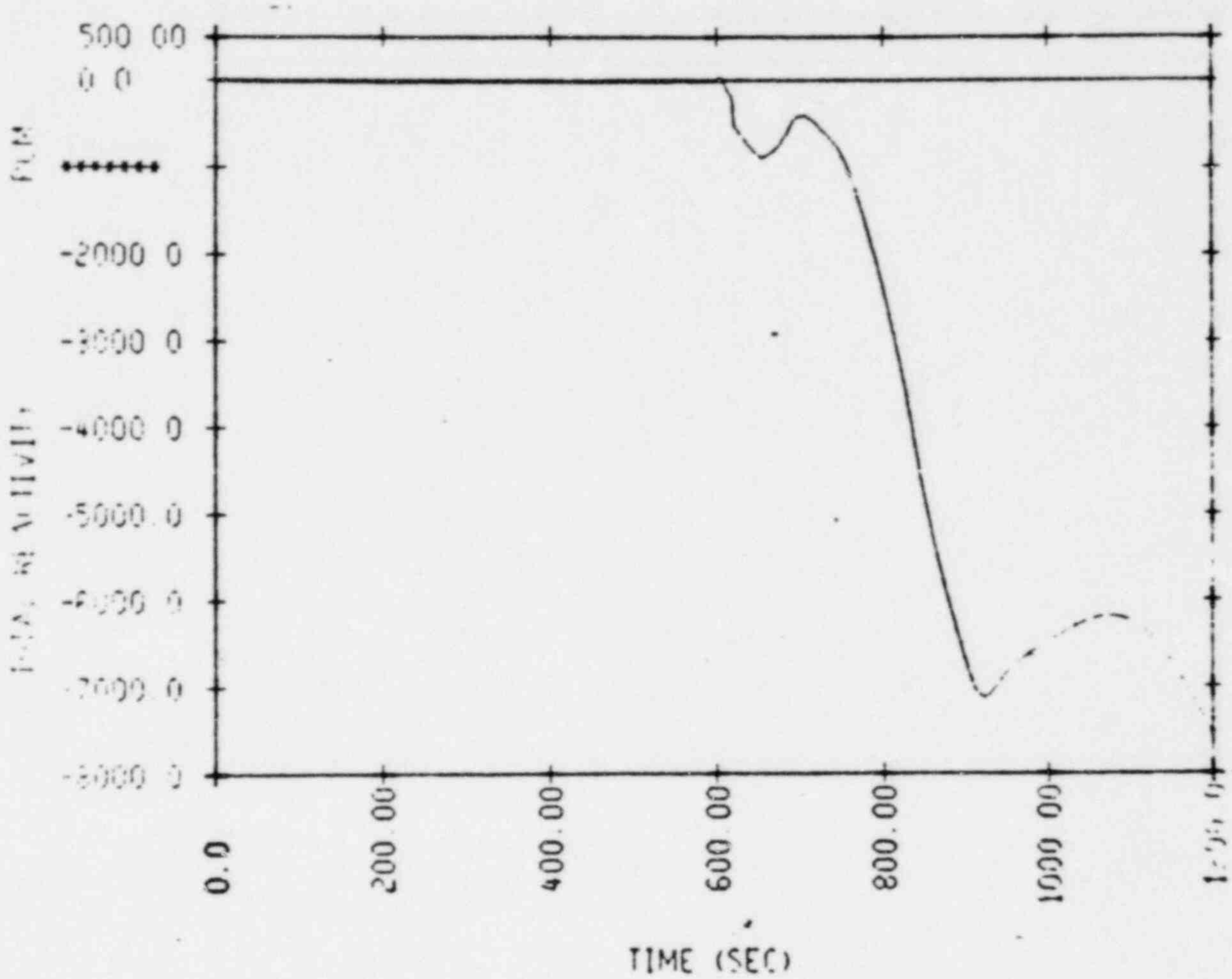


Figure 62. RCS Depressurization ATWS

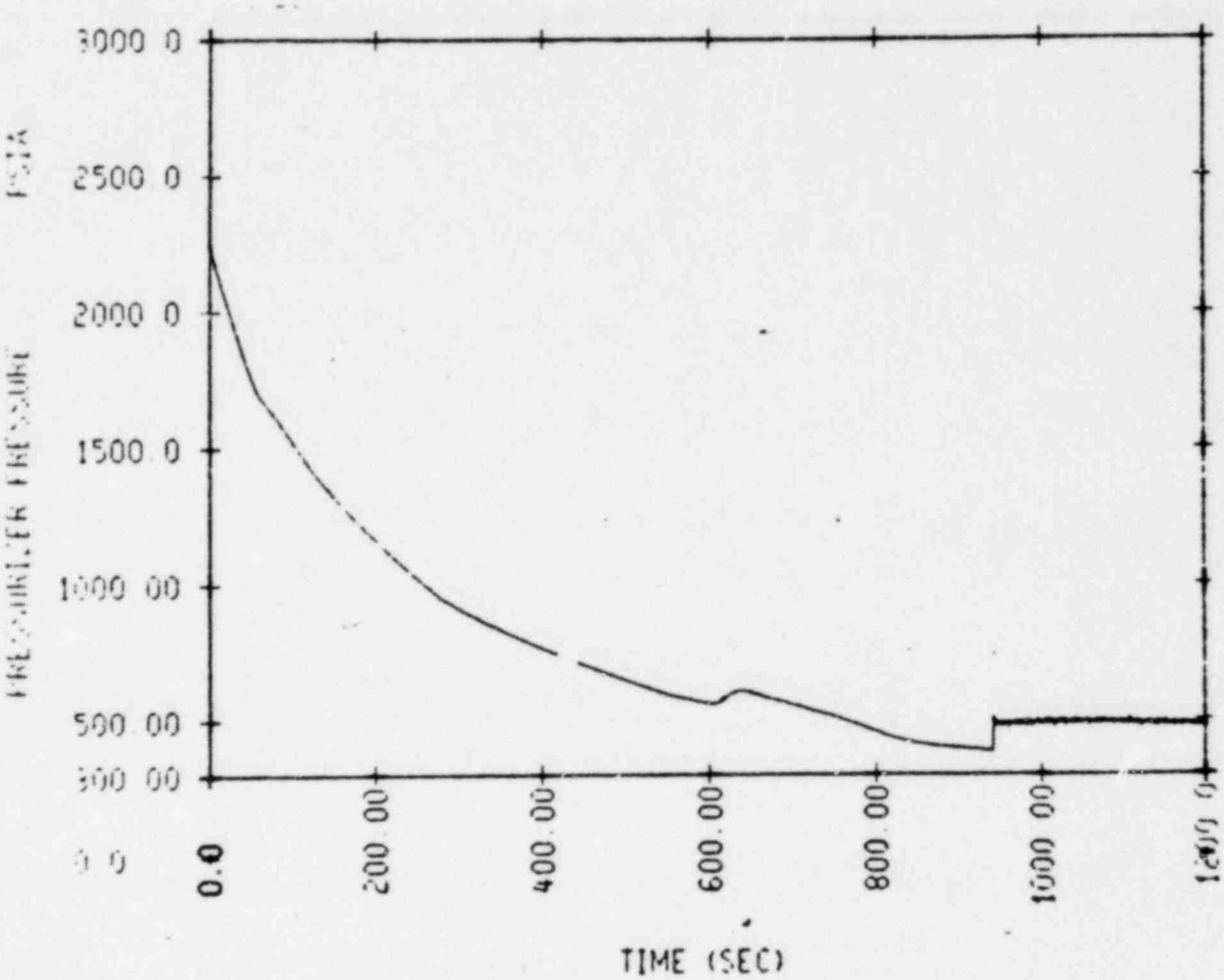


Figure 63. Depressurization ATWS With a Failure in the SI System

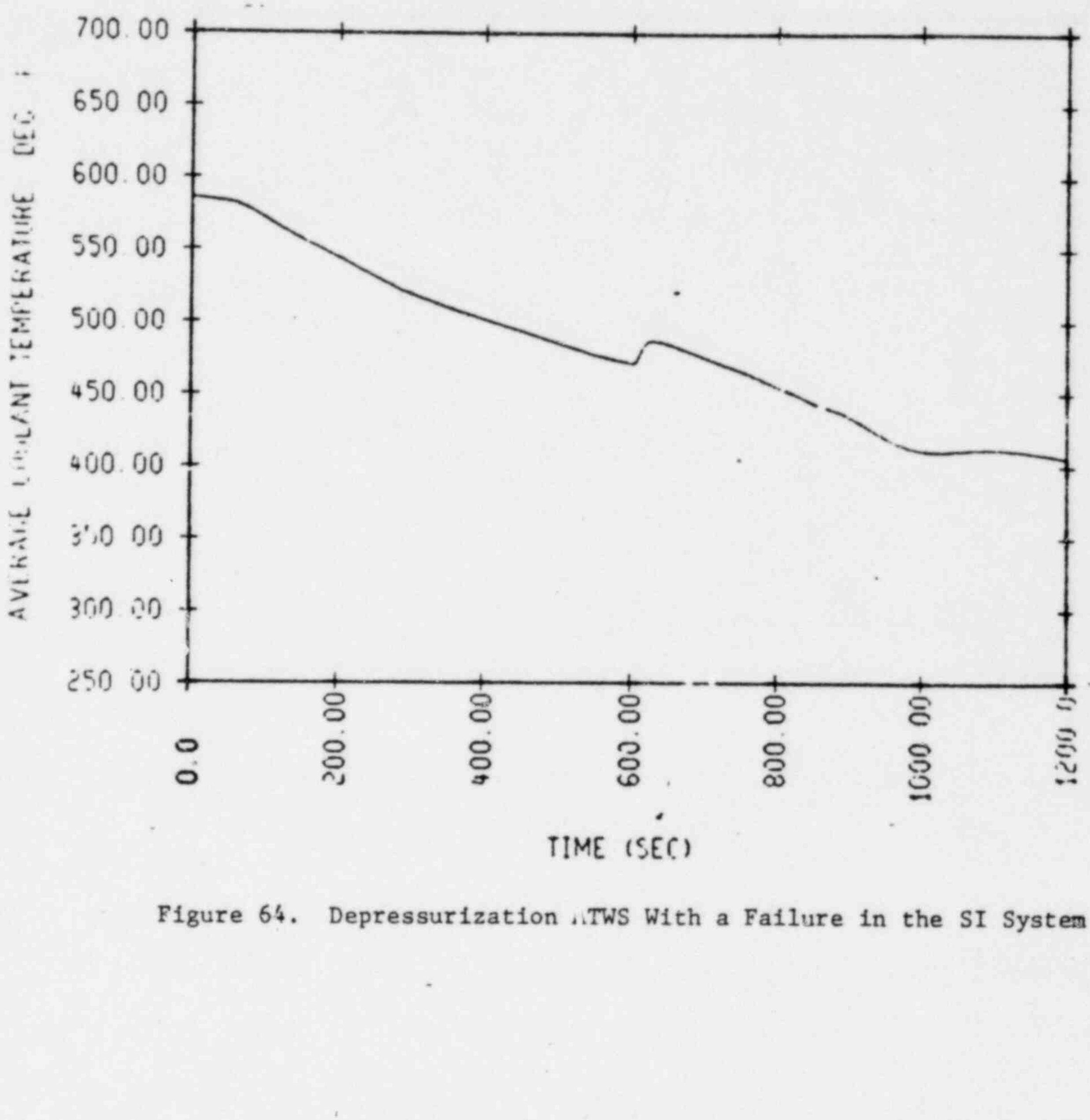


Figure 64. Depressurization ATWS With a Failure in the SI System

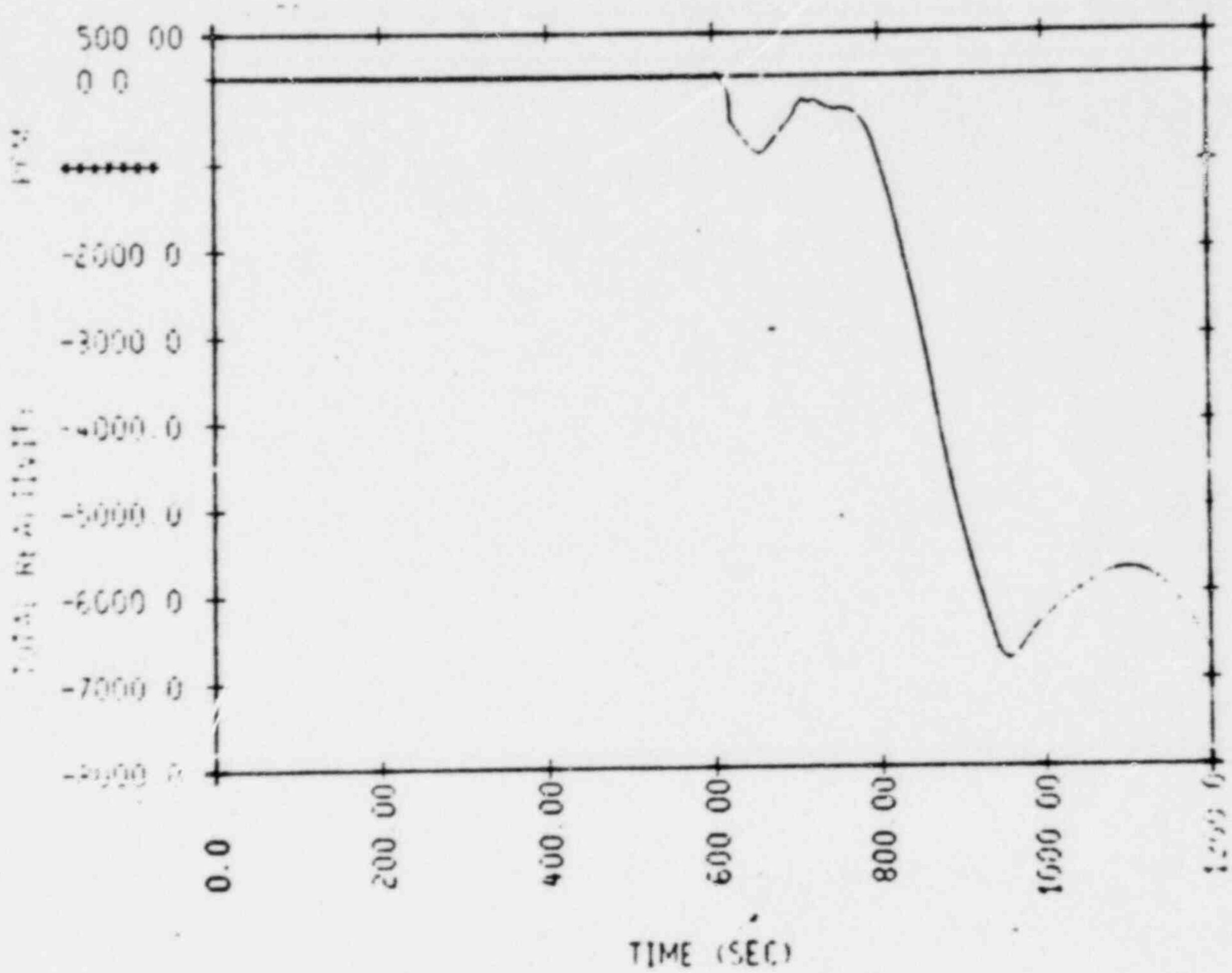
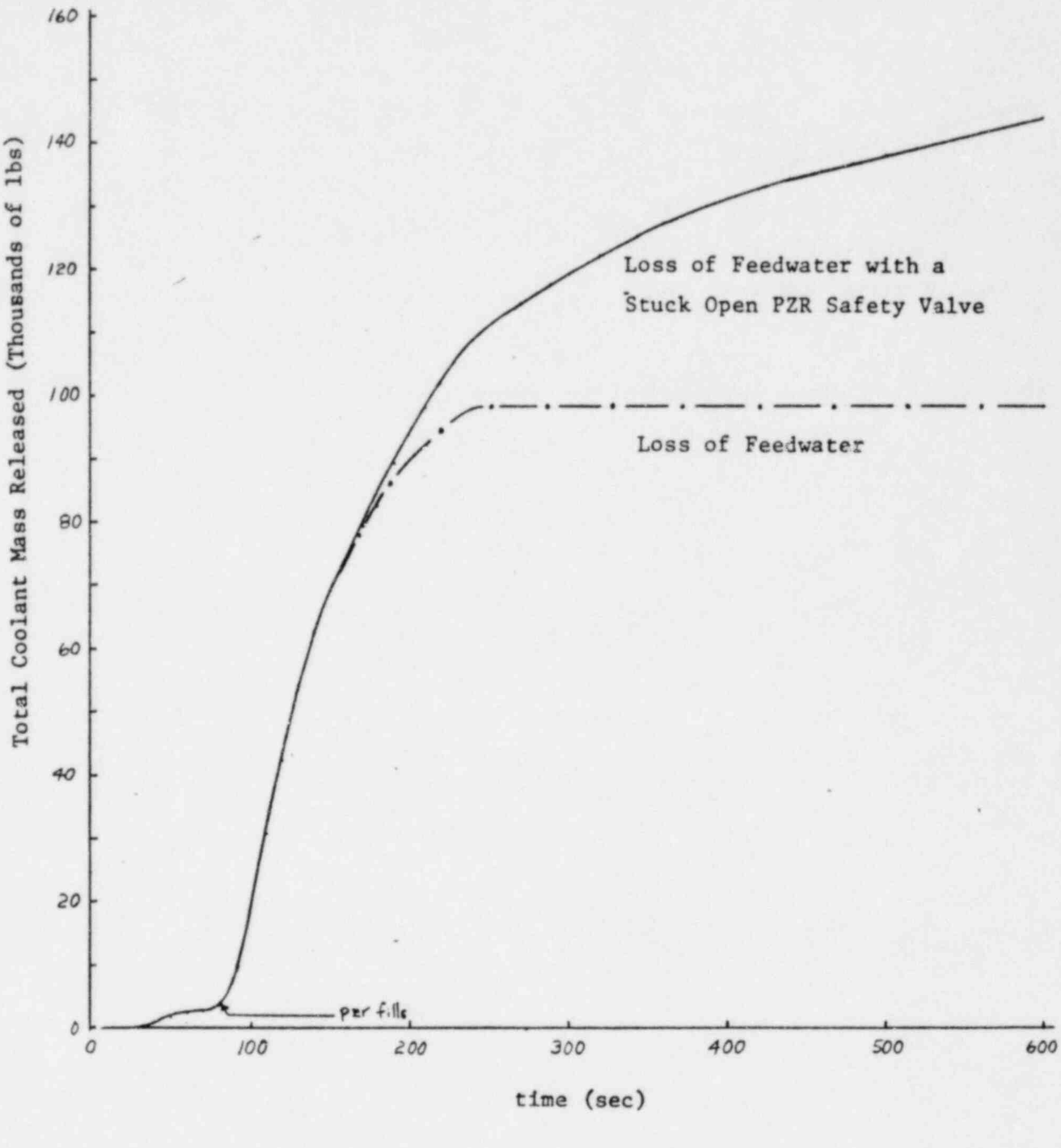


Figure 65. Depressurization ATWS With a Failure in the SI System

Figure 66. Total Mass Release



ATWT Moderator Coefficient

One of the most important parameters in the ATWT analysis is the moderator coefficient. Without scramming the control rods the major source of negative reactivity is the moderator heat-up. The PWR is designed to assure a negative temperature coefficient at operating power. The coefficient is typically less than -8 pcm/°F at beginning of cycle with equilibrium xenon, and decreases linearly to about -35 pcm/°F at the end of cycle.

One of the requirements in the ATWT status report was to show the effects of ATWT events with a moderator coefficient valid 99% of the time. This implies the coefficient may be more positive during only 1% of core life. Typically, the time between annual refuelings is about 7000 full power hours, which means that the time the coefficient can be more positive is only 70 hours, less than 3 days per cycle!

There are three major effects that determine the moderator coefficient:

1) core burnup,
2) reactor power level,
3) xenon concentration.

As stated earlier, the burnup effect of the coefficient varies from about -8 at beginning of cycle to about -35 at end of cycle. The use of a 99% coefficient (approximately 70 hours) means that burnup will have a small, but beneficial effect. The second term, reactor power, affects the coefficient in the following ways: 1) as power increases so does the average coolant temperature; reducing the coefficient and 2) as power increases, the average boron concentration decreases to compensate for the doppler feedback; also reducing the coefficient. The net effect is that by the time full power is obtained the coefficient is reduced to at least -3 pcm/°F.

The last term, xenon, is a neutron poison which is produced in the fuel and as it builds in, the boron concentration in the coolant is reduced to keep the core critical. The effect of equilibrium xenon is to reduce the moderator coefficient by -5 pcm/°F.

The least negative coefficient for current W fuel designs at beginning of cycle, full power, equilibrium xenon is then -3 plus -5 or -8 pcm/°F. The time to build in equilibrium xenon is about 3 days. Therefore, one definition of a 99% coefficient is -8 pcm/°F. This is the value used in WCAP-8330 except that it was defined to be valid 95% of the time. This was done so that the effects of shutdowns early in core life, when the xenon concentration and power vary, would be covered. If the same design philosophy is used to define a 99% coefficient, the value is -7 pcm/°F. This value can be met even with the following conservative variation in plant operation:

1) Two long shutdowns per month where the xenon concentration is reduced to zero,

2) Two short shutdowns per week of less than 12 hours,

3) Continuous load follow of 12 hours at full power ramping down over three hours to half power for six hours then ramping back-up to full power in three hours,

4) Power ramp rates following shutdown of at least 20%/hour, and

5) Initial power ramp rate following refueling of at least 3%/hour.

Single Failure

In safety analyses for SAR applications, single failures are assumed in order to ensure that the conclusions drawn are valid even when various systems or components do not operate. This has provided a conservative basis for the sizing and design of equipment. The probability of single failure, no matter how small, is not considered in a SAR analysis. For ATWT event, however, the probability of a single failure must be considered. This is because the acceptance of all ATWT events is based on a probability goal of 10^{-6} to 10^{-7} per year. This implies that if the reference case, without single failures, has a lower probability than the WASH-1270 limits of less than 10^{-6} to 10^{-7}, then additional failures need not be considered.

The safety goal defined in WASH-1270 is that the total probability of exceeding the limits (limited fuel damage and emergency stress limits) from all anticipated events should be on the order of 10^{-6} to 10^{-7}. This implies that the total probability of a severe ATWT is composed of at least three terms. The first of these terms is the probability of an event happening. The second term is the probability that the rods fail to scram given the event. And the third term is given the fact the event happened, and the rods failed, that the consequences exceed the limits. These probabilities are shown on the attached event tree (Figure 1).

The probability of not scramming has been defined to be 10^{-4} in WASH-1270. This number is the probability that a common mode failure (CMF) prevents a scram. This value was calculated in a very conservative manner in WASH-1270; as an upper bound with a very high confidence level. The value has also been accepted by the NRC Staff as a very conservative value*. The real value is significantly less than 10^{-4}. However, the value given in WASH-1270 will be used in order to be consistent with previous calculations.

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*ACRS testimony 12/11/75 by Warren Minners, page 21.

The probability of the event must also be considered. The value used in WASH-1270 was 1/year as a very conservative upper bound for all anticipated events. Data from WASH-1400 indicated that the probability of events that could lead to severe results is much smaller. For example, the probability of a loss of offsite power is on the order of 0.04/year and the probability of all other transients which could lead to severe ATWT results is ~0.04/year (see note on Figure 1). There are only five anticipated events that have a potential for severe results. If the combined probability of these events is used, then the probability of an anticipated event that could lead to severe results is bounded by 0.1/year, not 1.0/year. However, a value of 1.0/year will be used to be consistent with previous submittals.

The last term in the total probability of a severe ATWT is the probability that the results could exceed the limits. All of the potentially severe events result in a heatup of the reactor coolant system. This heatup will result in a power decrease due to the negative temperature reactivity coefficient. The rate at which the power decreases is directly dependent upon the magnitude of the temperature coefficient. The temperature coefficient decreases throughout core life such that the ATWT results get better (lower peak pressure, higher DNBR's). In fact, the results can only be severe or approach the limits for a very short time, early in core life. For this analysis, a coefficient valid 99% of the time is used. As shown in WCAP 8330 the results are acceptable with a 95% and it will be shown later in this report that the results are acceptable even when using a 99% coefficient. It it is conservatively assumed that a coefficient greater than the 99% value leads to severe results then the probability that the results may exceed the limits is less than 1%.

For the reference cases, the probability of <u>severe</u> ATWT conservatively represented by:

Event x CMF x temperature coefficient

$10^0 \times 10^{-4} \times 10^{-2} \approx 10^{-6}$

This means the reference cases already meet the safety goal of 10^{-6} to 10^{-7}. The effect of single failures is to add another branch to the event tree. The single failures that we were asked to evaluate are the loss of one power operated relief valve, a failure in the auxiliary feedwater system, and a failure in the safety injection system. A very conservative upper bound of the probability of the failure of all of these systems is 0.032. (See Figure 1)

When the total probabilities are calculated with and without single failures the results are:

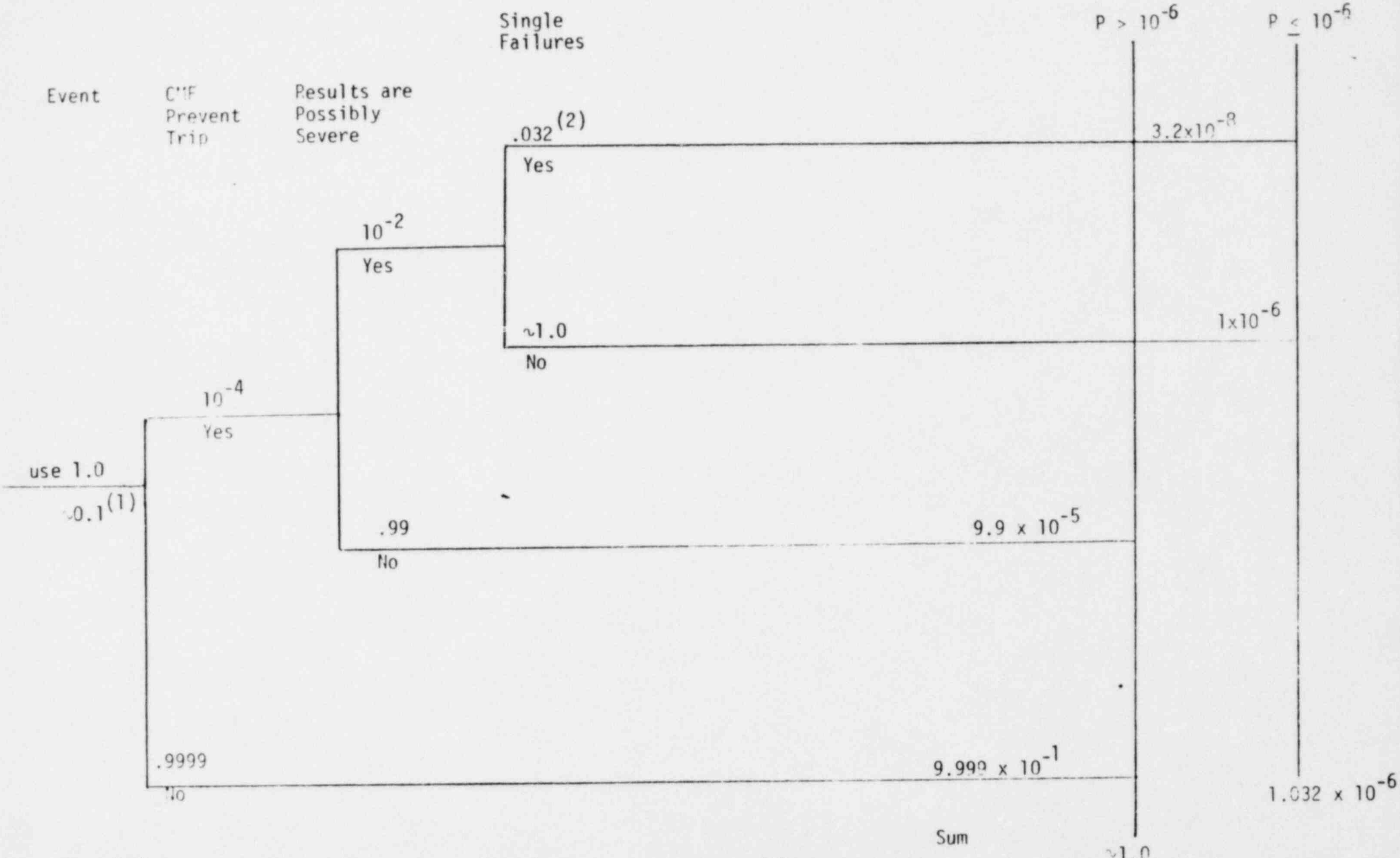
No single failure 1×10^{-6}
With single failure 1.032×10^{-6}

This means that if cases with single failures must show acceptable results then only a 3% gain in the safety goal for all ATWT events is obtained.

One of the problems of using probabilities is that very little data exists for the probabilities of a common mode failure and for the probability of the event itself. The numbers used in this argument are conservative and are supported by the existing data. Even if the probabilities are questioned, the simplified event tree does show the effect of single failure, only 3%, and also shows the importance of including the moderator coefficient in the total proability. It also shows that the reference cases already meet the safety goal.

The status report requests that analyses be presented with single failures. These analyses will be presented; but the results should not be judged against the criteria stated in WASH-1270. The addition of single failures makes the probability of severe ATWT smaller than the safety goal of 10^{-6} to 10^{-7}, for which the criteria in WASH-1270 were developed.

FIGURE 1



Note (1)

1) Loss of offsite power ~.04
2) Loss of Normal Feedwater ~.01
3) Loss of Load ~.01
4) Rod Withdrawal ~.01
5) ... .01

(2)

loss of 1 relief valve	~.020
loss of aux. feedwater	~.012
loss of safety injection	N/A
Total	~.032

ADDITIONAL TRANSIENT ANALYSIS

The status report requested that additional analyses be presented. These additional analyses were made using the same assumptions used in WCAP-8330 except:

1. A moderator coefficient valid 99% of the time is used. This value corresponds to -7pcm/°F at hot full power.

2. The water relief model used in WCAP-8330 assumed a constant fluid enthalpy throughout the transient. A more detailed treatment of water relief is included in this analysis, such that the calculated enthalpy in the pressurizer is used.

3. A new option in LØFTRAN is used such that pressure is calculated around the loop. This modification means that the 80 psi that was added in WCAP 8330 to account for the pressure difference between the RCS pressure and pump discharge pressure is no longer required. This change does not affect the pressurizer pressure calculation.

1.1 ROD WITHDRAWAL FROM SUBCRITICAL

The rod withdrawal from subcritical was reanalyzed. The only changes to the case presented in WCAP-8330 was the use of a moderator coefficient valid 99% of the time. The sensitivity studies done for WCAP-8330 are still valid to show the effects of coolant flow, amount of inserted rod worth, reactivity insertion rate, and initial steam generator water mass.

The reference case is a 4-loop plant with a model 51 steam generator. The rod withdrawal is simulated by withdrawing 1% $\Delta k/k$ at the maximum rod speed. The average coolant system temperature is shown on Figure 1 and core power on Figure 2.

To show the effect of a single failure, a power operated relief valve was assumed not to operate. The power and temperature transients are shown on Figures 3 and 4.

The key results of the transient are shown in Table 1. The peak system pressure for both of these cases are well below emergency stress limits. Because of the low core power and nominal core flow, the DNB ratio is very high throughout the transient.

TABLE 1

ROD WITHDRAWAL FROM SUBCRITICAL

Case	Peak power (fraction of nominal)
Reference	0.55
Assume One relief valve tailed	0.55

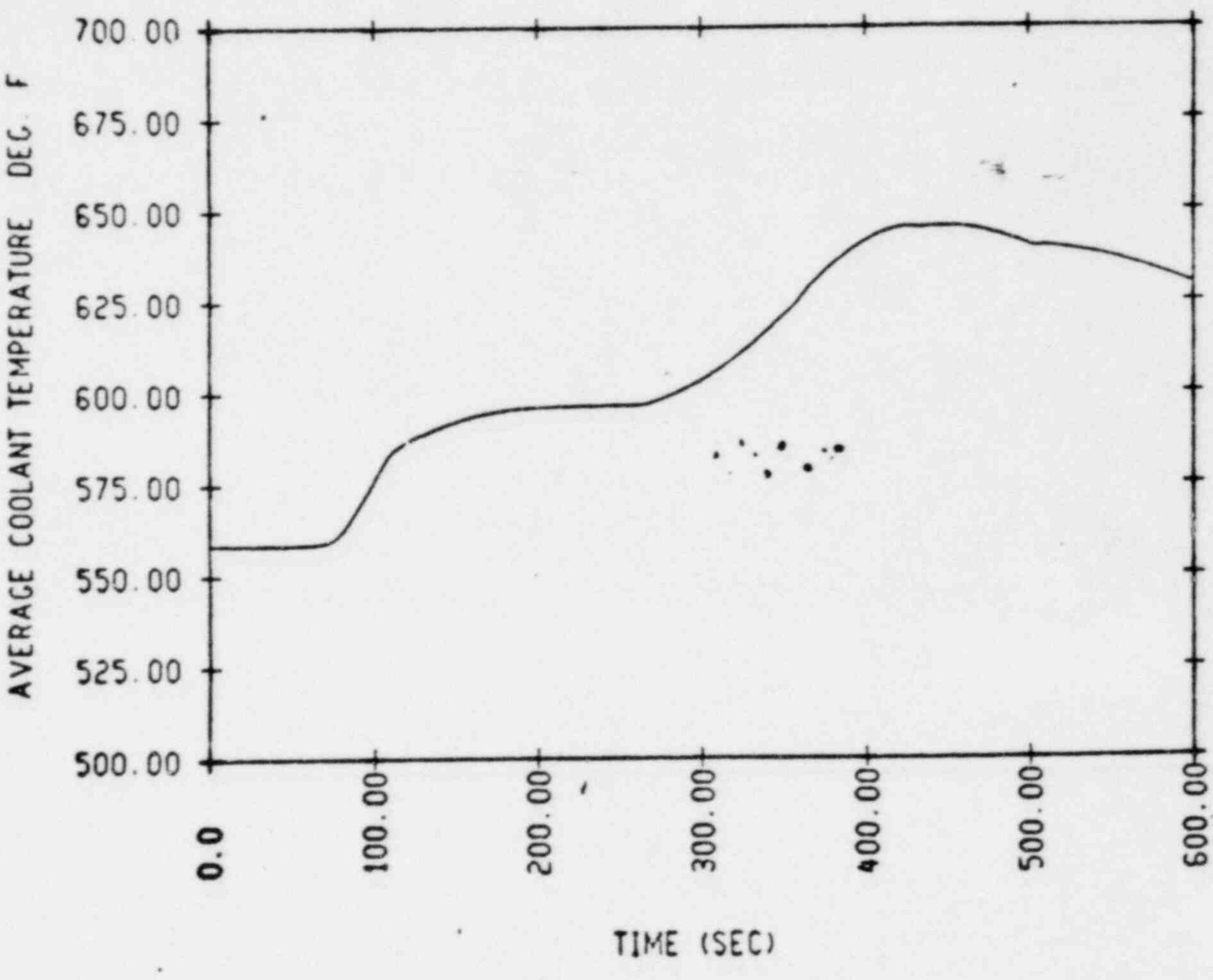


Figure 1. Rod Withdrawal from Subcritical Reference Case

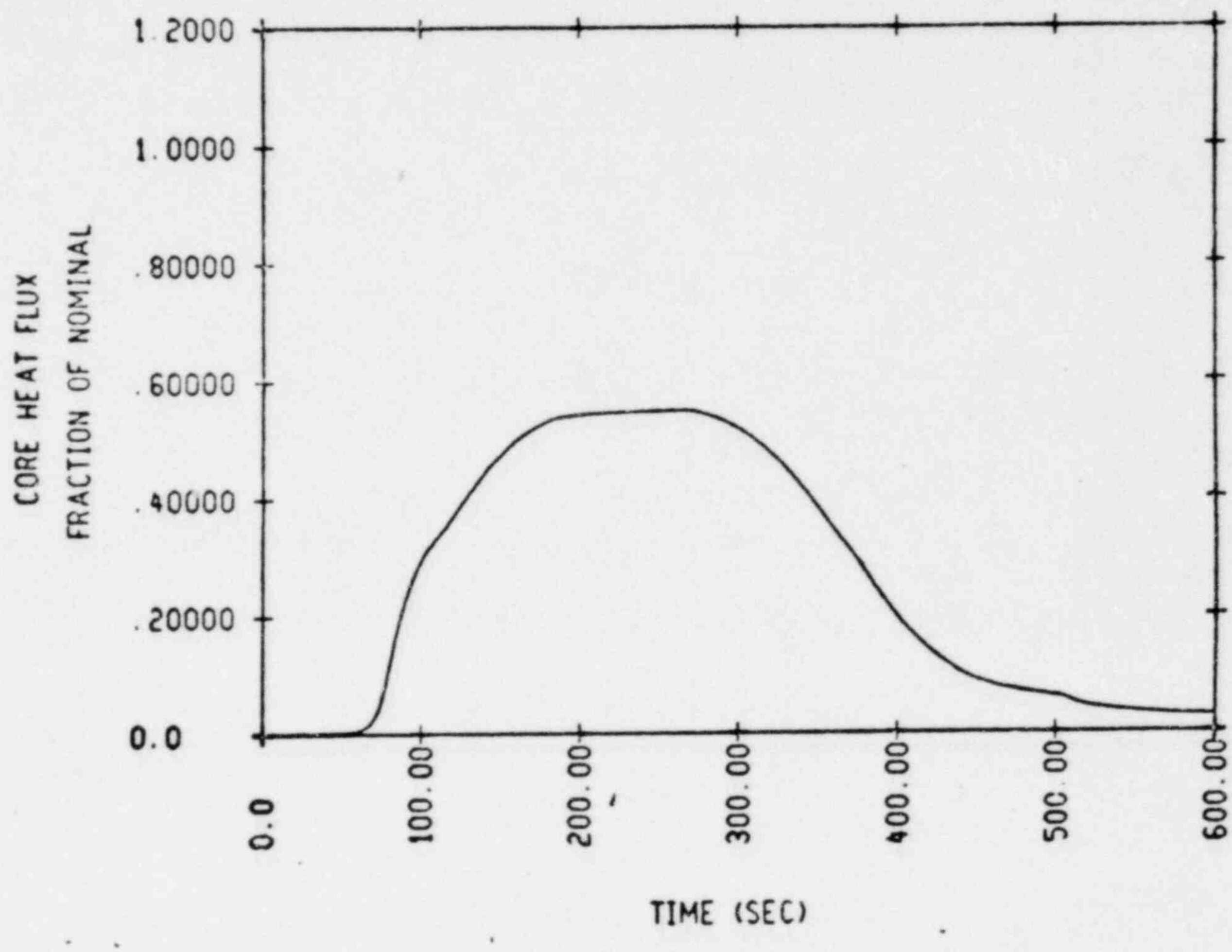


Figure 2 Rod Withdrawal from Subcritical Reference Case

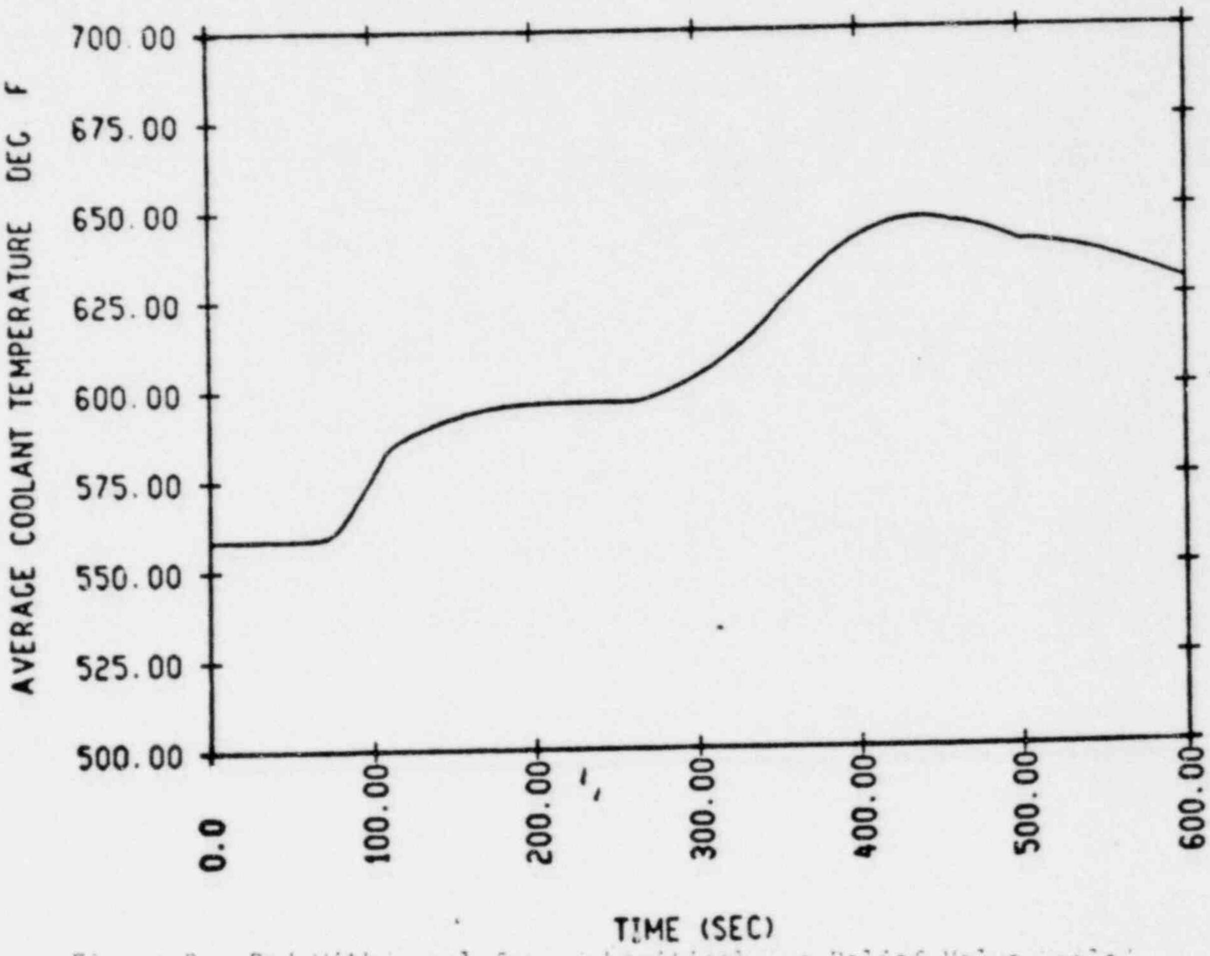


Figure 3. Rod Withdrawal from Subcritical (no Relief Valve Failure)

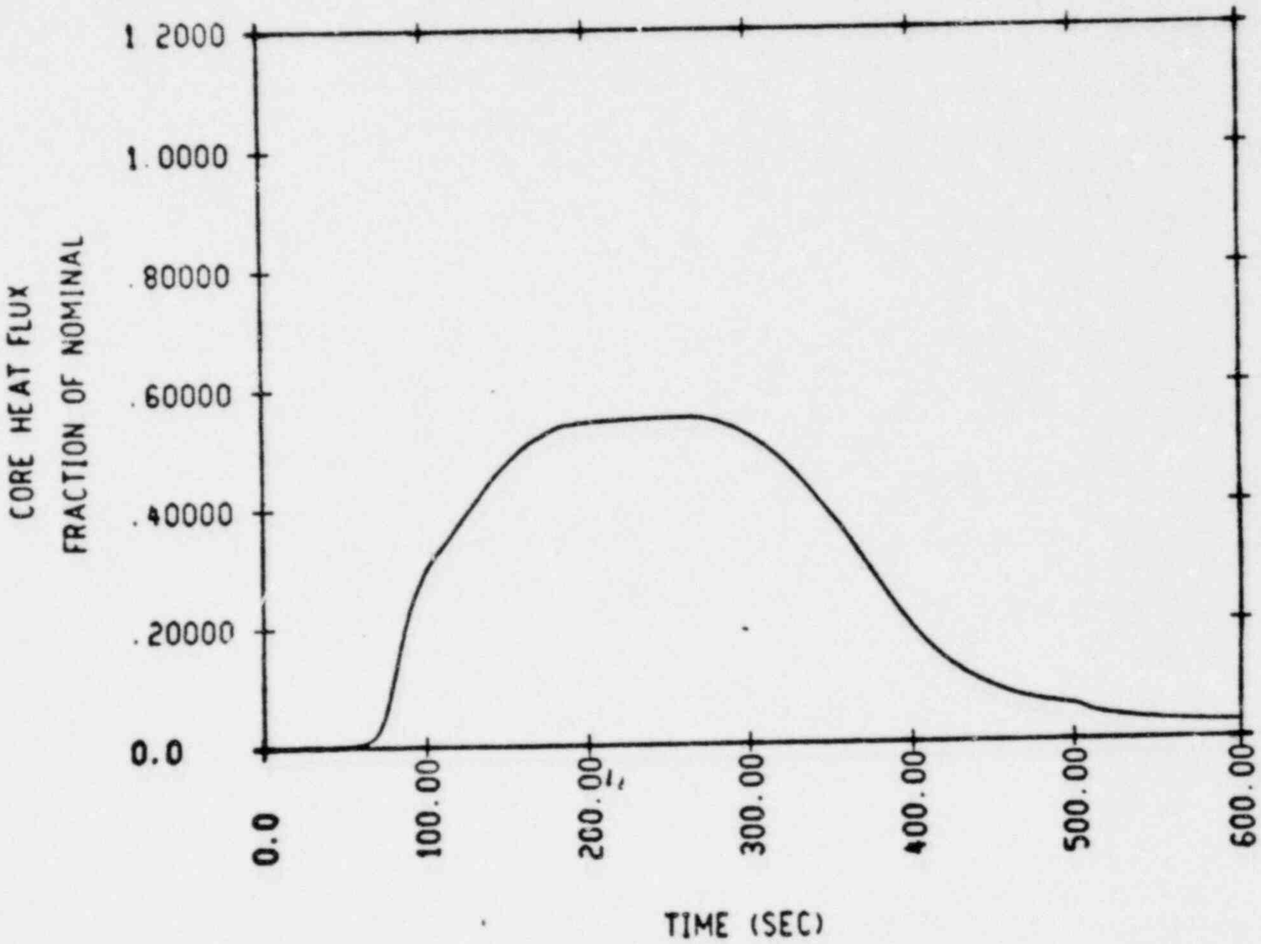


Figure 4: Rod Withdrawal from Subcritical One Relief Valve [illegible]

1.2 ROD WITHDRAWAL AT POWER

The rod withdrawal at power transient was reanalyzed with a moderator coefficient valid 99% of the time. The sensitivity studies presented in WCAP-8330 are still valid to show the effects of inserted worth, initial power, turbine trip, rate of reactivity insertion and initial steam generator mass.

Figures 5 through 7 show the transient response of the reference case with 99% coefficient. Figures 8 through 10 show the effect of an average temperature 8° higher than the reference and Figures 11 through 13 for an average temperature 20° lower than the reference case.

To show the effects of a single failure the reference case was reanalyzed assuming that one power operates relief valve failed. The pressure transient is shown in Figure 14. This single failure increases the minimum DNBR above the no single failure case.

The key results are summarized in Table 2. The peak system pressure for this transient are well below the emergency stress limits. The minimum DNBR's for all cases are greater than 1.30.

TABLE 2

ROD WITHDRAWAL AT POWER

Case	Peak System Pressure psia	Minimum DNBR
Reference	2401	1.48
15 x 15 fuel	2401	1.49
Tave + 8°F	2401	1.30
Tave - 20°F	2401	1.77
Assume 1 relief valve failed	2421	1.49

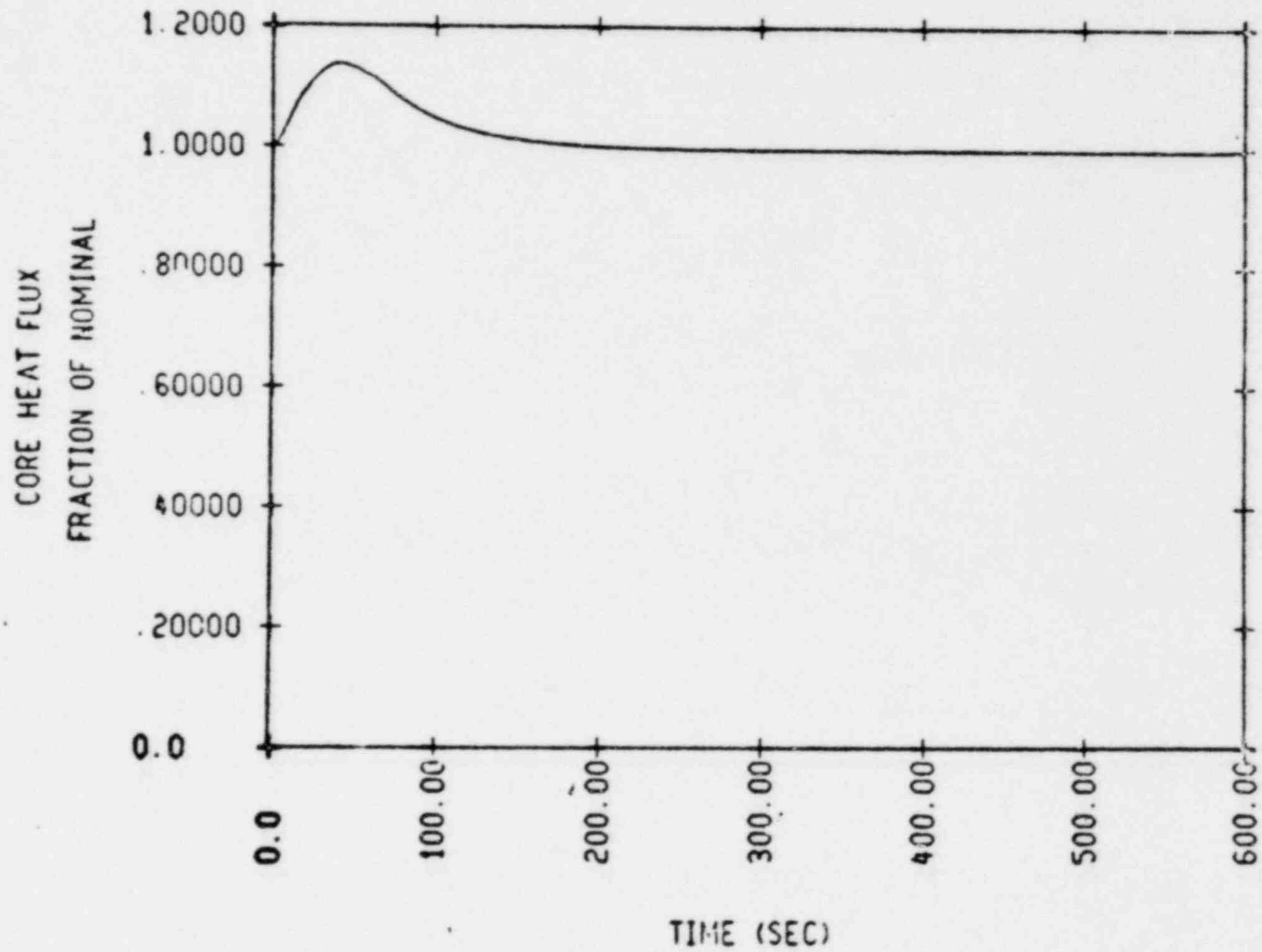


Figure 5. Rod Withdrawal at Power Reference Case

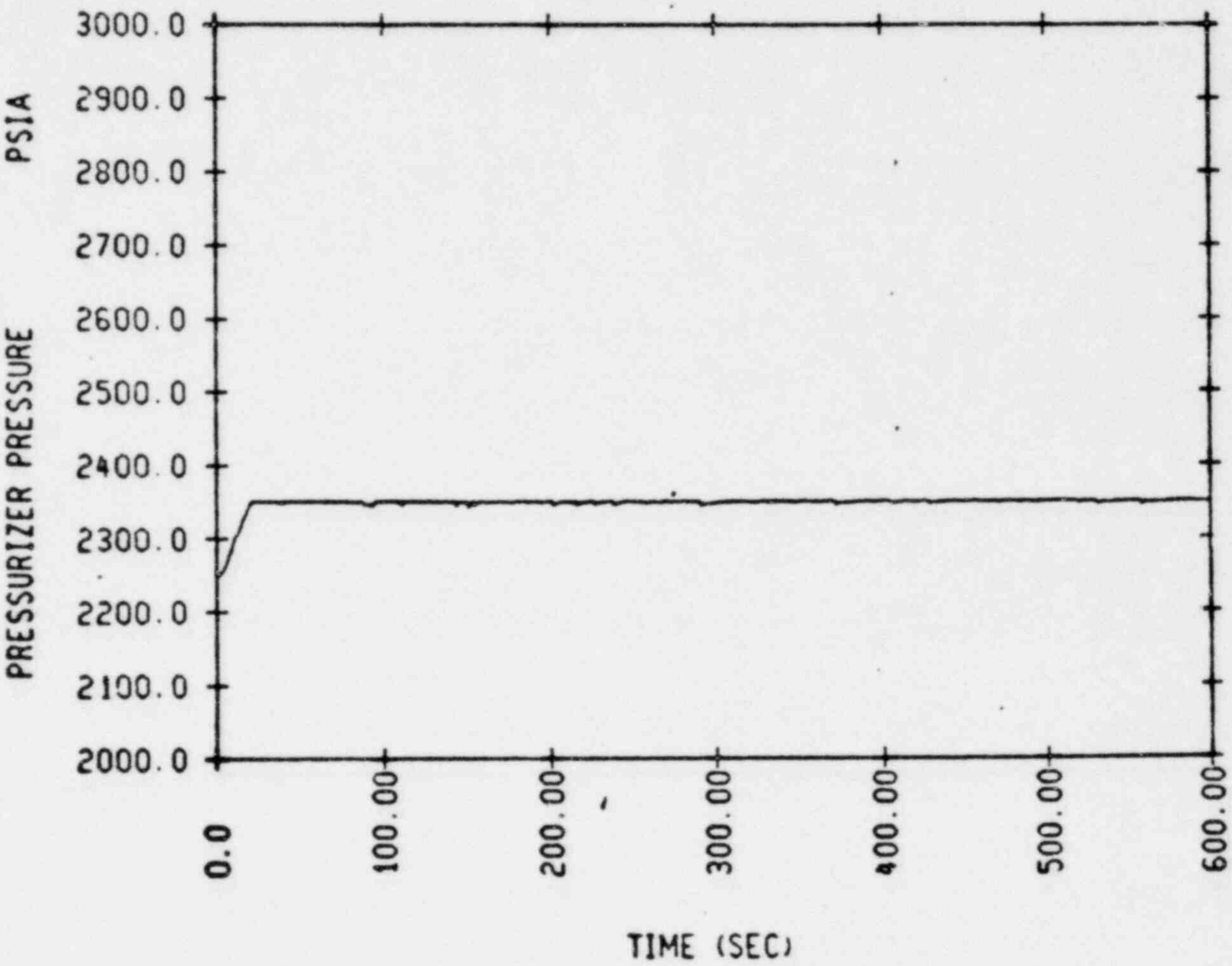


Figure 6. Rod Withdrawal at Power Reference Case

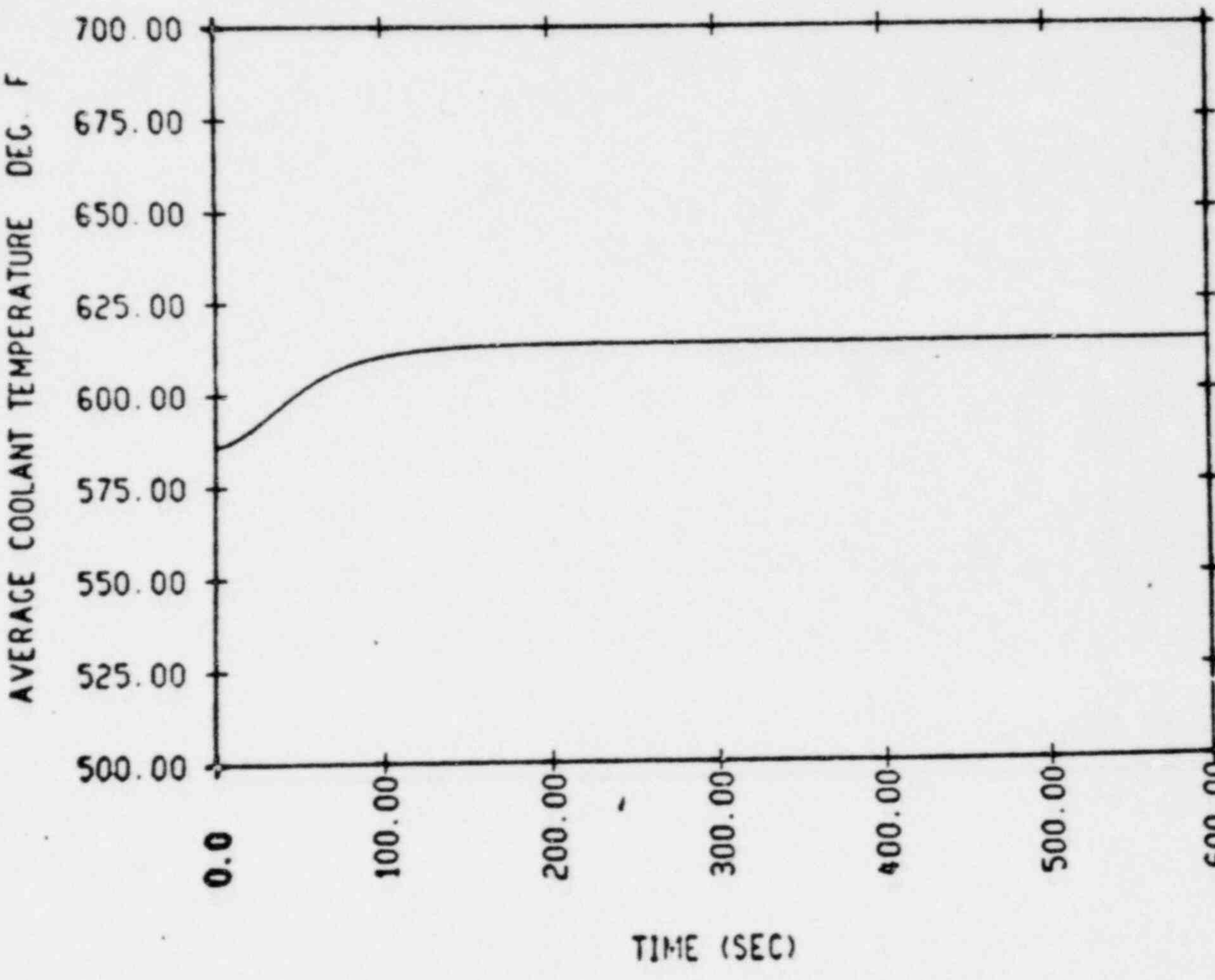


Figure 7. Rod Withdrawal at Power Reference Case

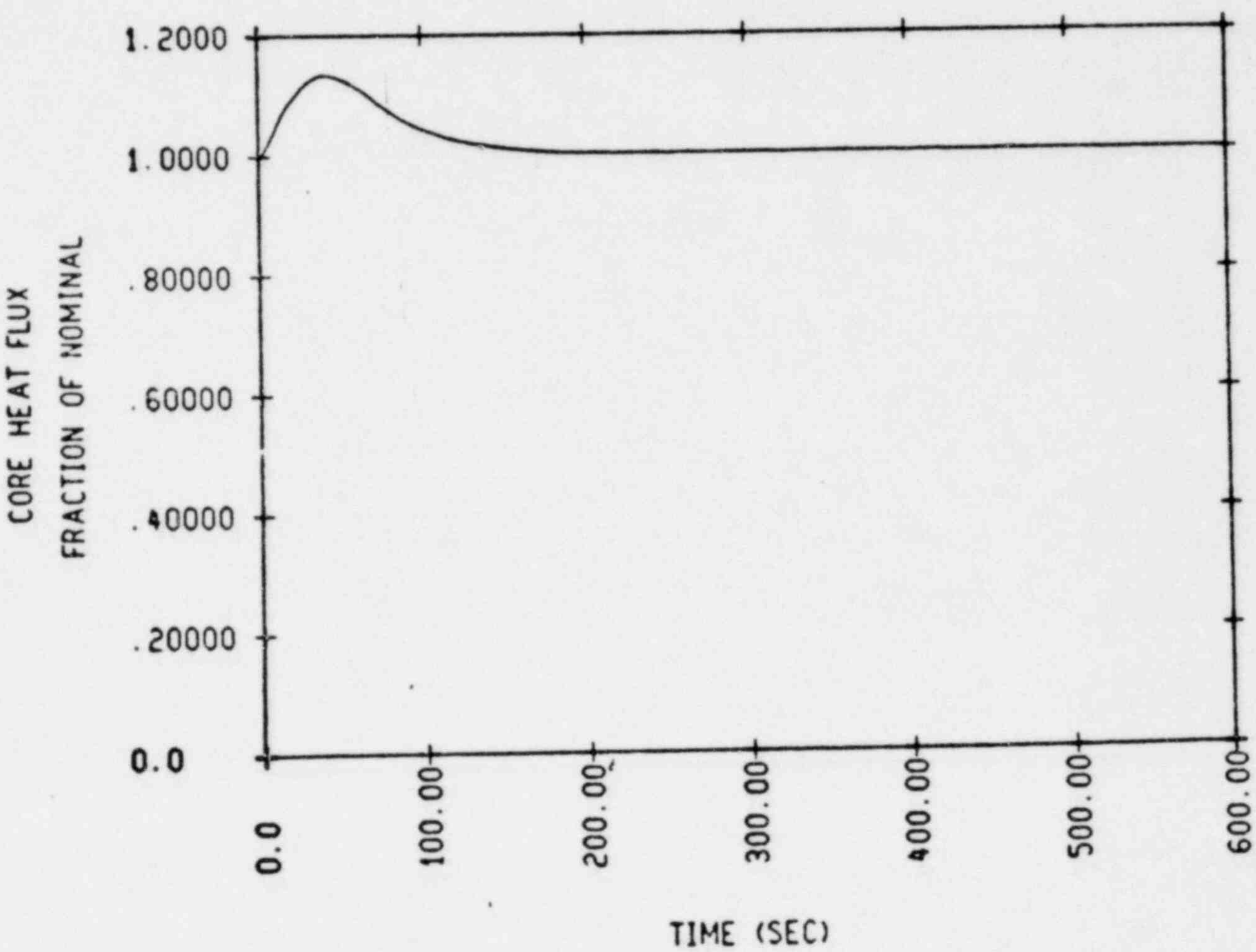


Figure 8. Rod Withdrawal at Power Average Temperature +8°F

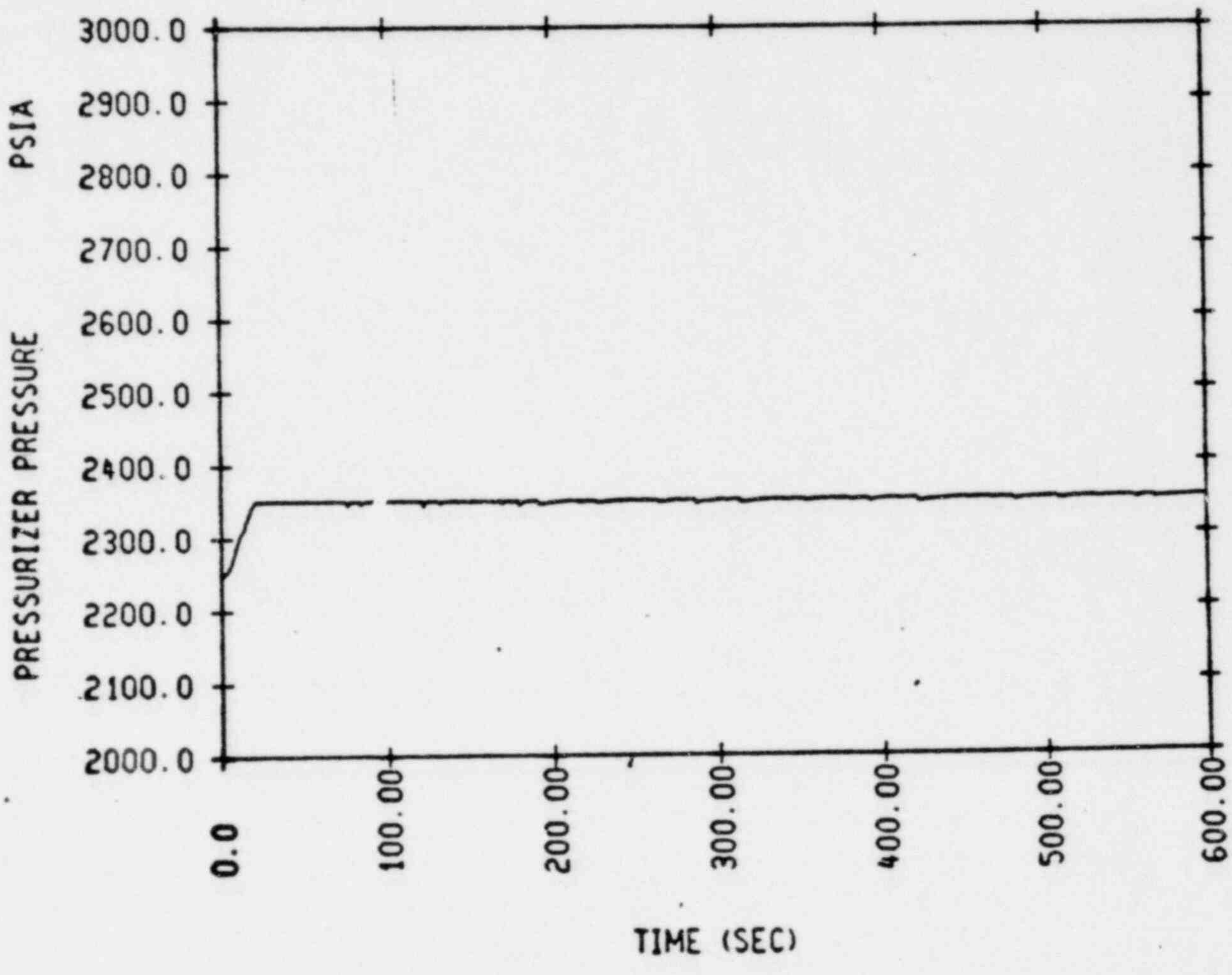


Figure 9. Rod Withdrawal at Power Average Temperature +8°F

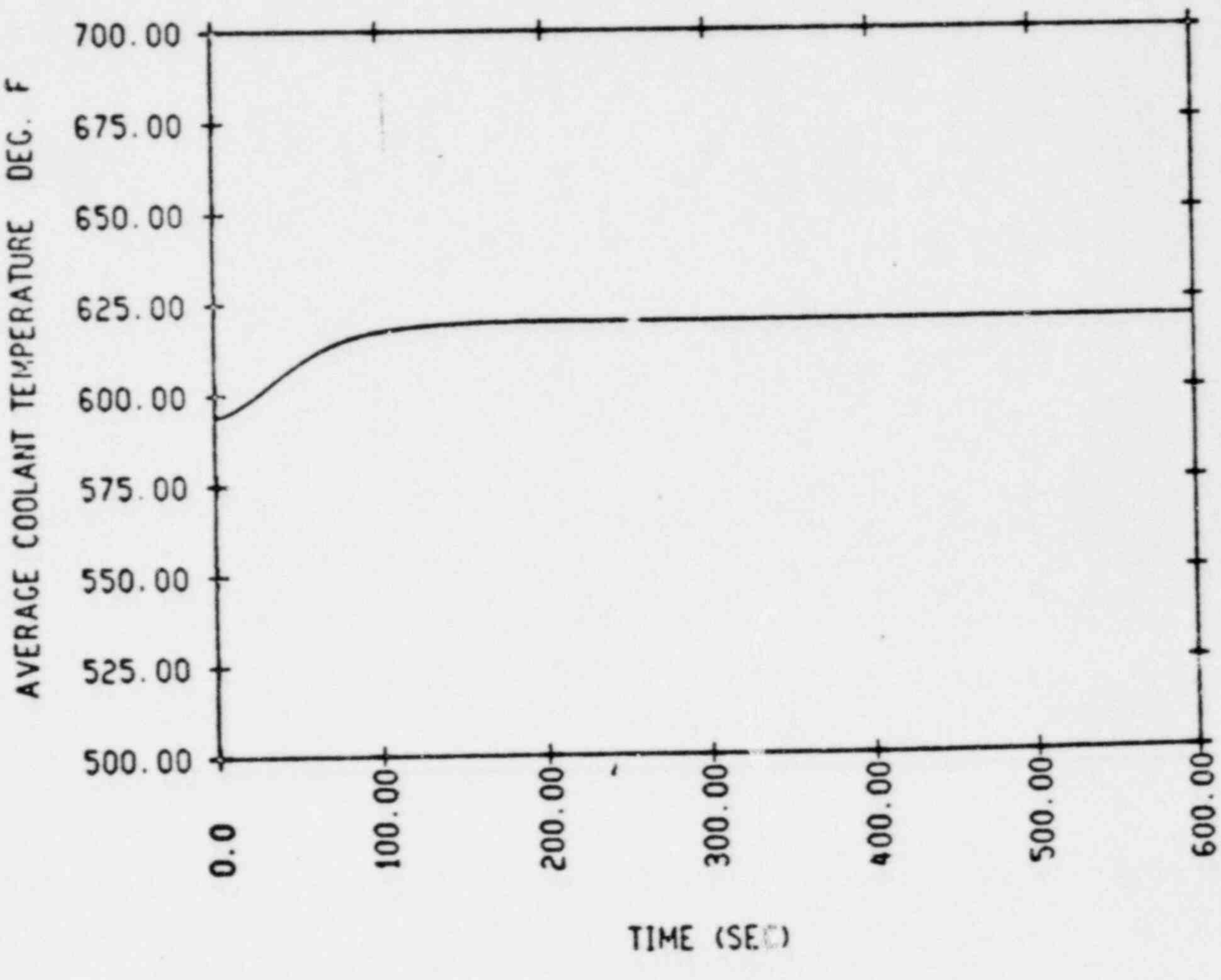


Figure 10. Rod Withdrawal at Power Average Temperature +8°F

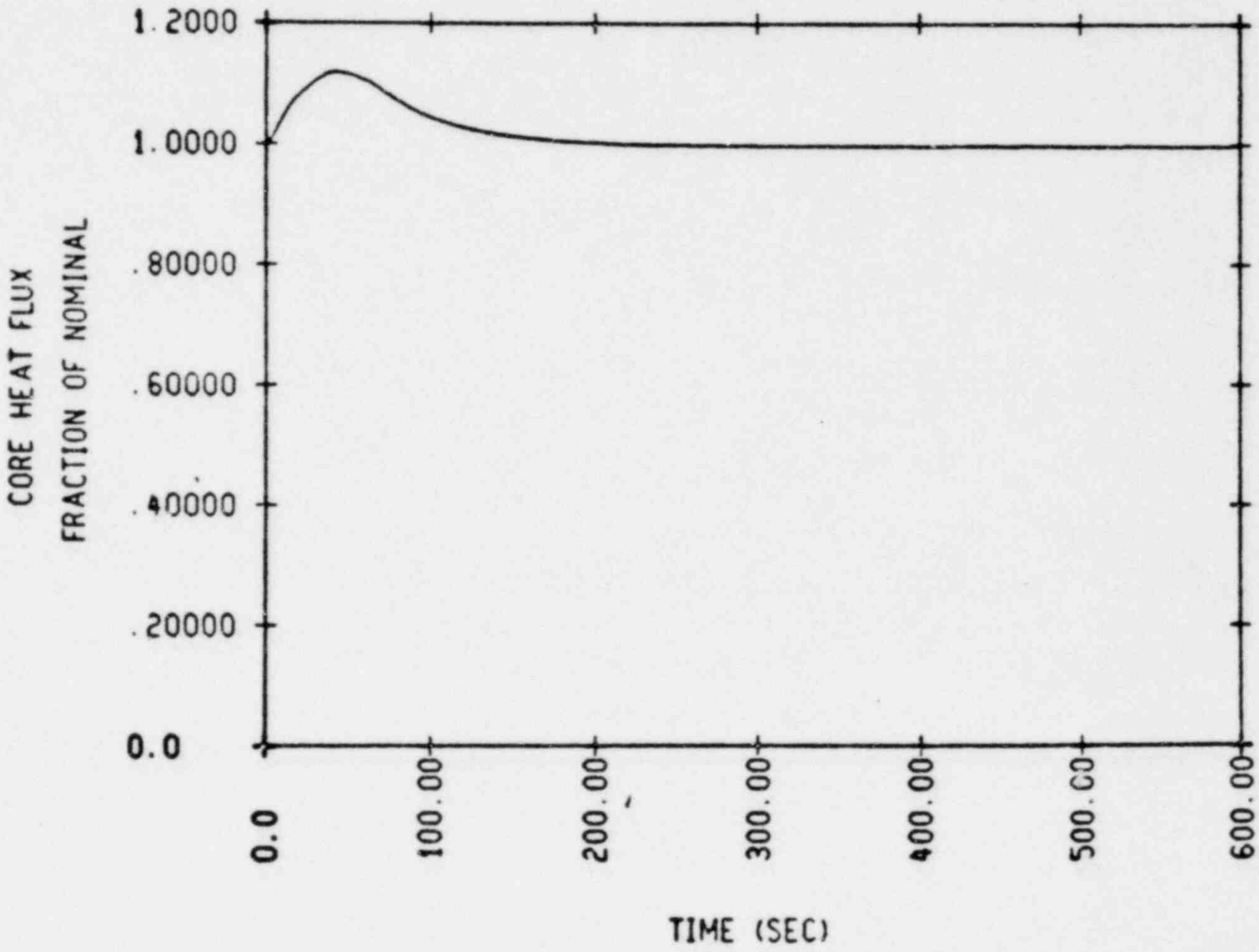


Figure 11. Rod Withdrawal at Power Average Temperature - 20°F

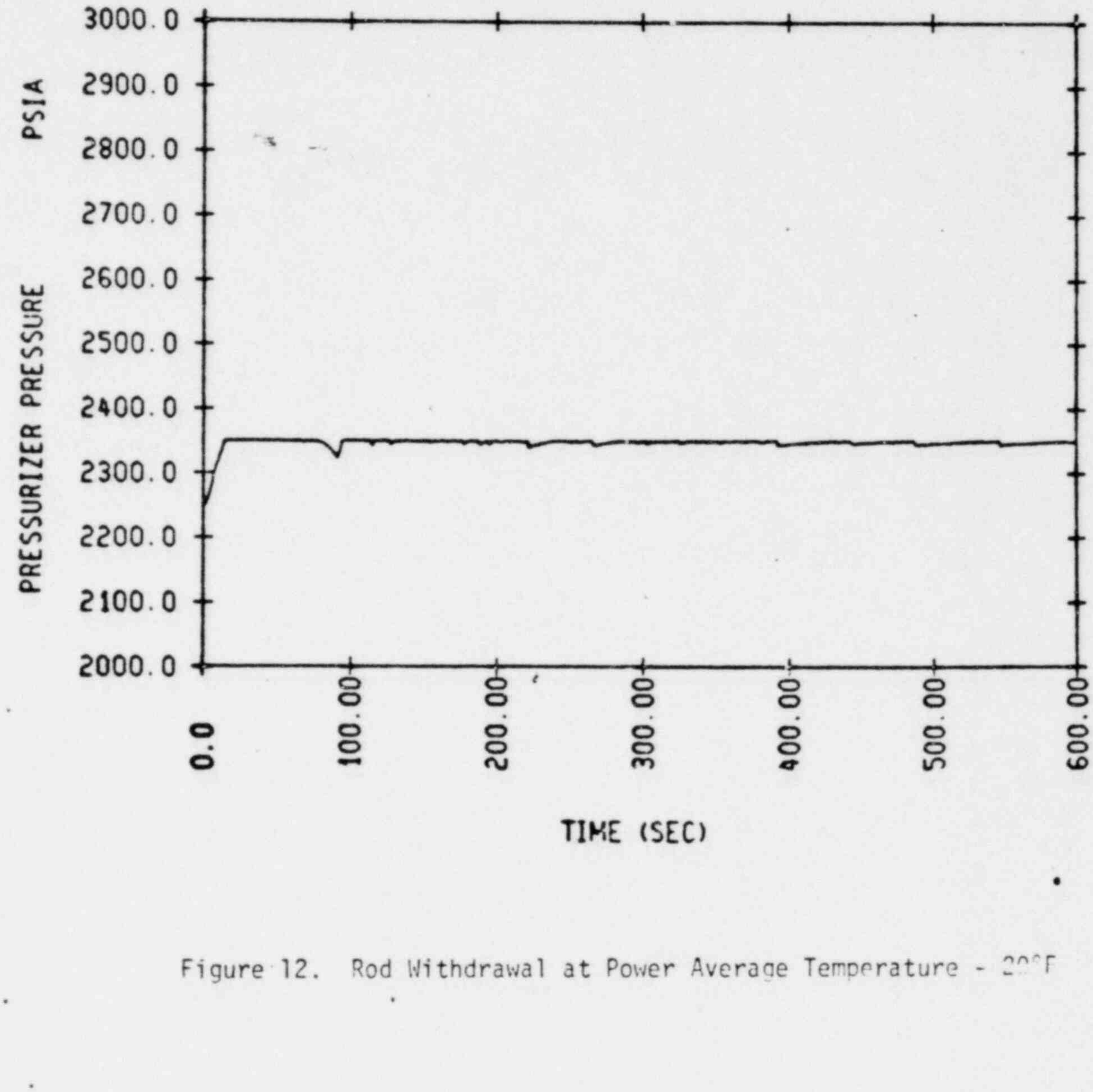


Figure 12. Rod Withdrawal at Power Average Temperature - 20°F

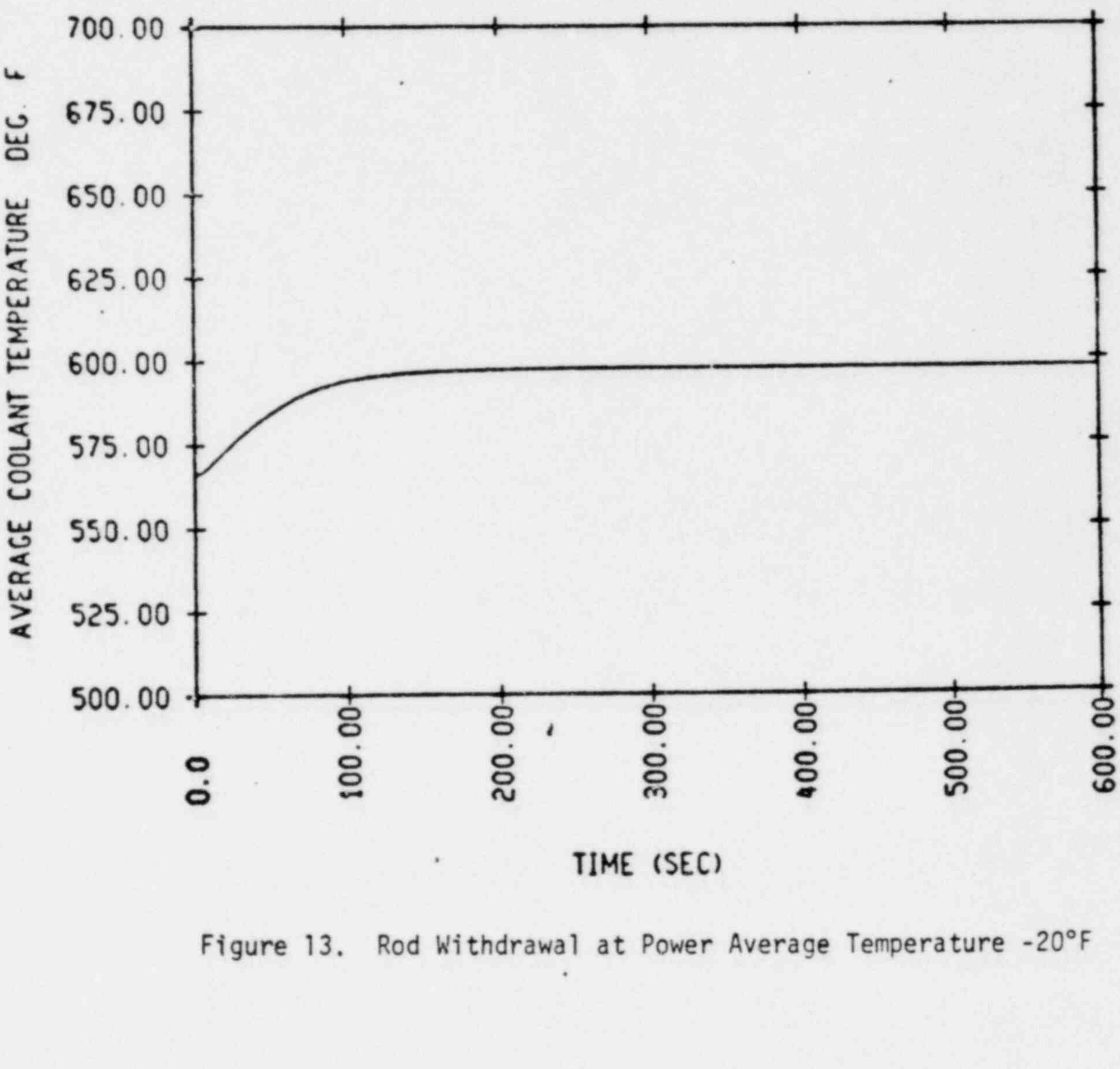


Figure 13. Rod Withdrawal at Power Average Temperature -20°F

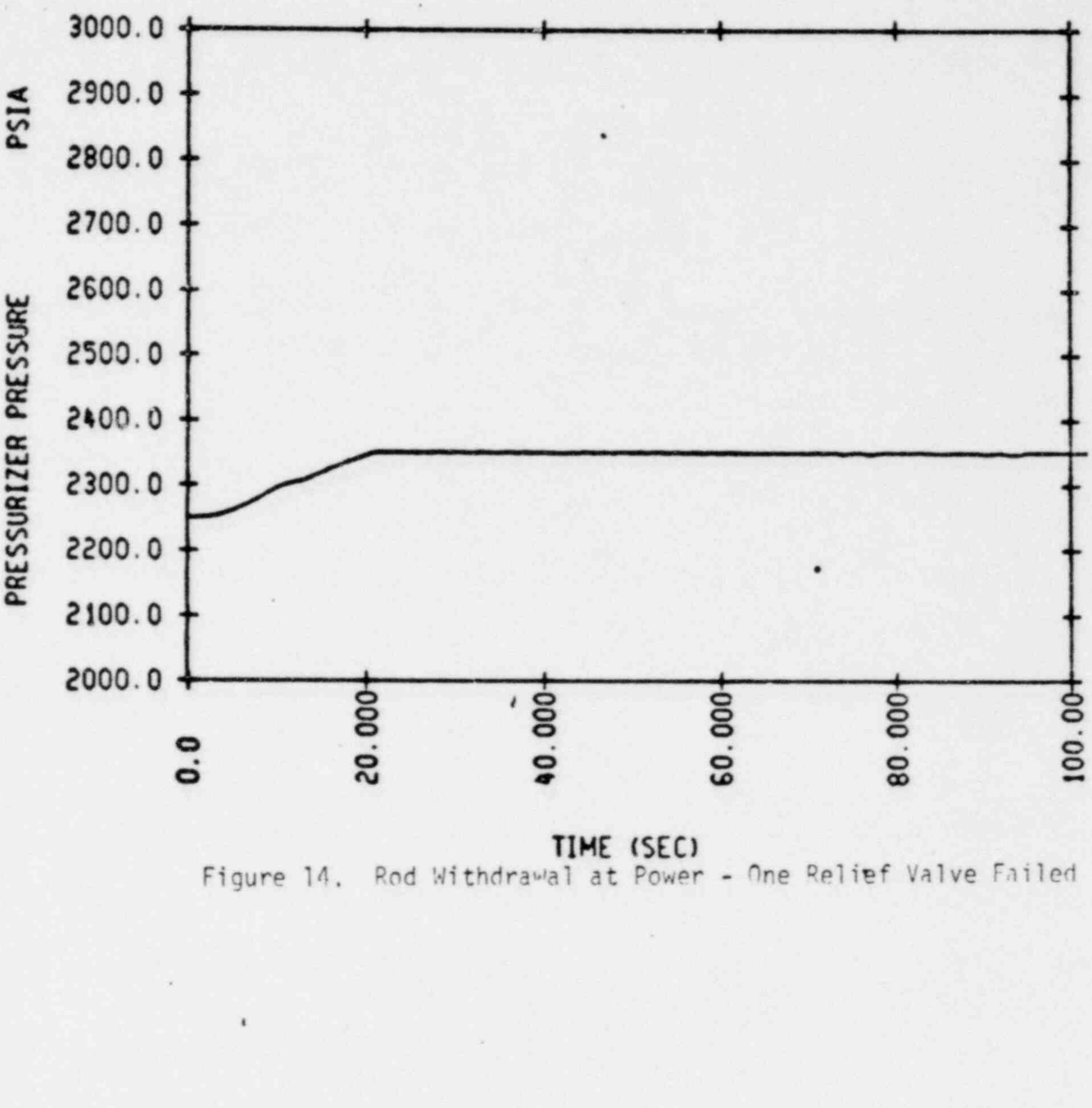


Figure 14. Rod Withdrawal at Power - One Relief Valve Failed

1.3 BORON DILUTION

The boron dilution was shown in WCAP-8330 to be less severe than the rod withdrawal at power. This conclusion was based on the integral amount of inserted reactivity due to the dilution. Thus the conclusion is not affected by the change to a 99% moderator coefficient and therefore not reanalyzed.

1.4 PARITAL LOSS OF FORCED REACTOR COOLANT FLOW

The partial loss of forced reactor coolant flow was shown in WCAP-8330 to be less severe than the loss of offsite power transient. This transient will be affected by the change in the moderator coefficient but the conclusion that it is less severe than the loss of offsite transient is still valid. For this reason, this case was not reanalyzed.

1.5 STARTUP OF AN INACTIVE LOOP

In WCAP-8330 the startup of an inactive loop was shown to be less severe than the rod withdrawal at power. While the change in the coefficient will affect the results, the rod withdrawal remains more severe. The conclusion in WCAP-8330 remains valid and, hence, not reanalyzed.

1.6 LOSS OF EXTERNAL ELECTRICAL LOAD AND/OR TURBINE GENERATOR TRIP

Only the reference case was reanalyzed for the loss of external electrical load transient with a moderator coefficient valid 99% of the time. This was done because of loss of external electrical load is less severe than the loss of normal feedwater which will be considered later.

Figures 15 through 17 show the system response to this transient. The peak system pressure obtained was 2680 psi, and the DNB ratio did not decrease below its initial value.

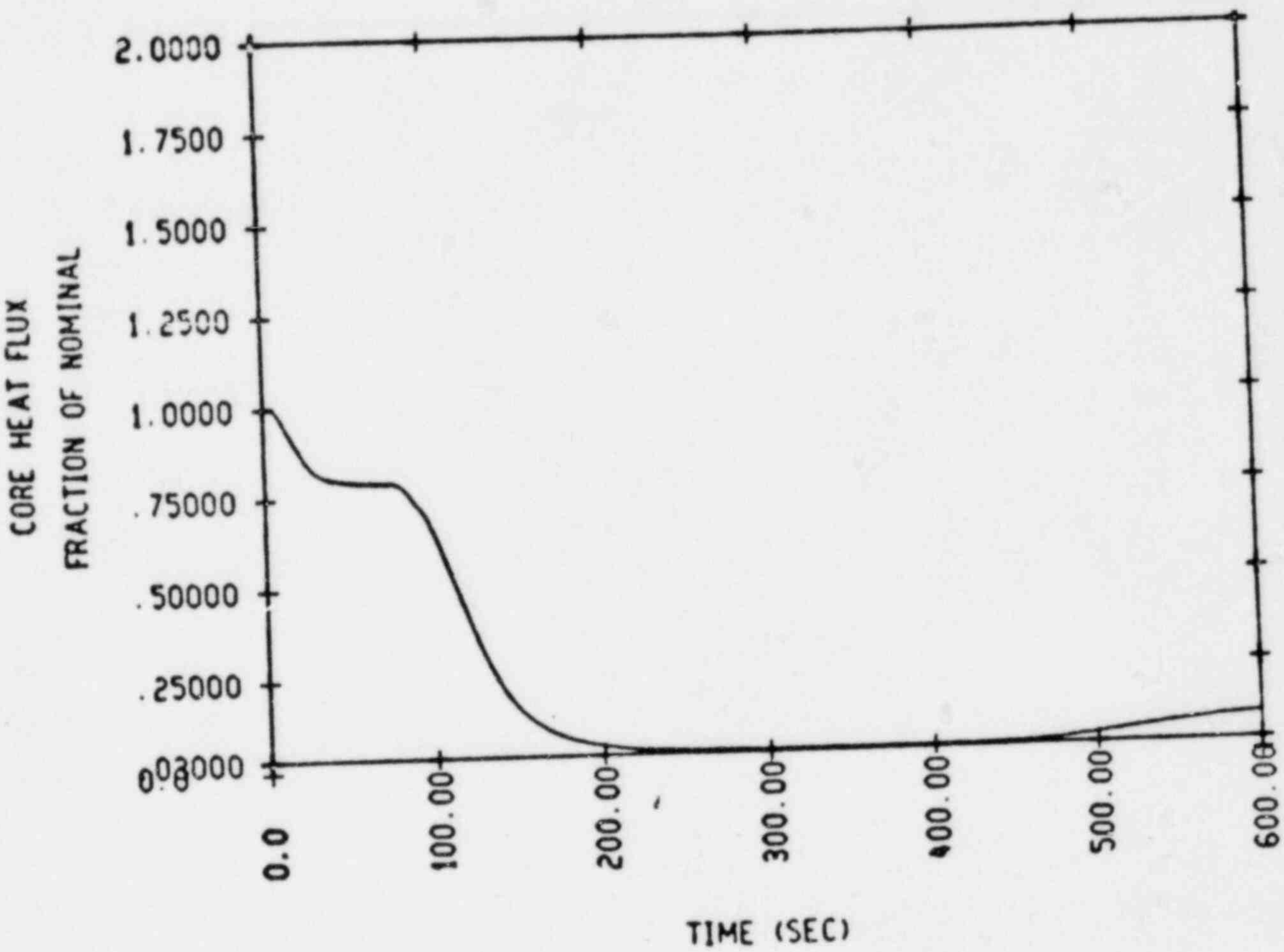


Figure 15. Loss of Load - Reference Case

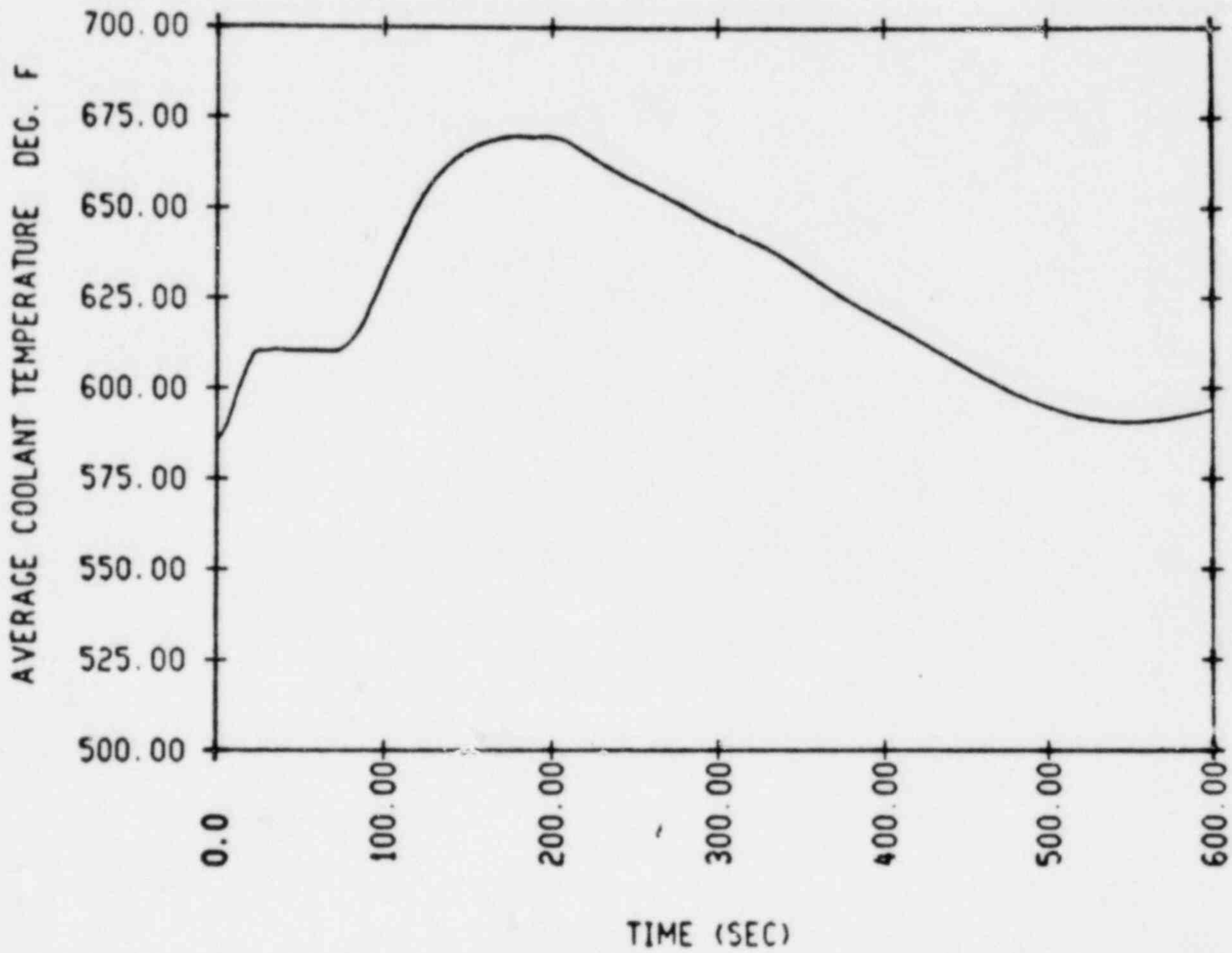


Figure 16. Loss of Load - Reference Case

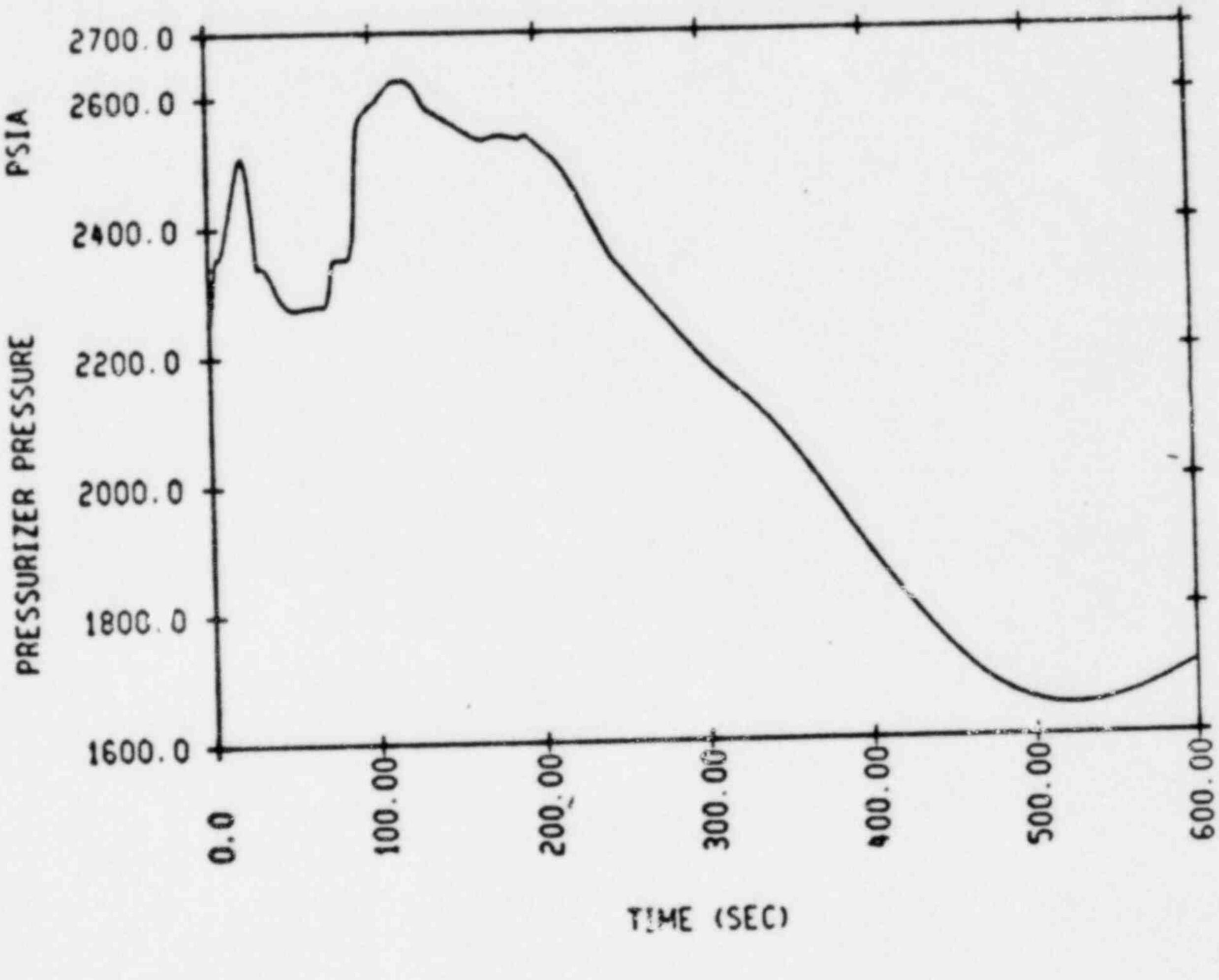


Figure 17. Loss of Load - Reference Case

1.7 LOSS OF NORMAL FEEDWATER

The loss of normal feedwater is the limiting transient with respect to peak system pressure. This transient was reanalyzed using a moderator coefficient valid 99% of the time and the pressurizer modelling discussed in previous submittals. All other parameters are those assumed in WCAP-8330. The sensitivity studies done in WCAP-8330, i.e., no turbine trip, no pressurizer spray, use of rod control, initial pressurizer level, and initial power, are still valid.

The system response to the reference case is shown in Figures 18 - 20. The response of the system for initial average temperature 8° higher than the reference are shown in Figures 21 - 23. For an initial temperature 20° less than the reference the system response is shown in Figures 24 - 26.

To show the effects of single failure only, the reference case was reanalyzed. If a power operated relief valve fails to open, the peak pressure will be somewhat higher because of the reduced relief capability. The system response to this case are shown in Figures 27 through 29. Another single failure, that was asked for, was the response given that a motor-driven auxiliary feedwater pump was assumed to fail. For this case, the pressures are again higher due to the reduction in the heat sink. Figures 30 through 32 show the system response.

Table 3 summarizes the key results of the loss of normal feedwater transient. The peak system pressure for this case are somewhat larger than the emergency stress limits quoted in WCAP-8330. These stress limits were calculated in a very conservative manner. The stress limits are currently being recalculated for limiting components using more realistic input. When these calculations are submitted, it is expected that the peak system pressure will be below the new emergency stress limits.

TABLE 3

LOSS OF NORMAL FEEDWATER

Case	Peak System Pressure, psia
Reference	2738
Tave + 8°F	2958
Tave - 20°F	2641
Assume 1 relief valve fails	2888
Assume 1 motor driven auxiliary feed pump fails	2811

ATWT Moderator Coefficient

One of the most important parameters in the ATWT analysis is the moderator coefficient. Without scramming the control rods the major source of negative reactivity is the moderator heat-up. The PWR is designed to assure a negative temperature coefficient at operating power. The coefficient is typically less than -8 pcm/°F at beginning of cycle with equilibrium xenon, and decreases linearly to about -35 pcm/°F at the end of cycle.

One of the requirements in the ATWT status report was to show the effects of ATWT events with a moderator coefficient valid 99% of the time. This implies the coefficient may be more positive during only 1% of core life. Typically, the time between annual refuelings is about 7000 full power hours, which means that the time the coefficient can be more positive is only 70 hours, less than 3 days per cycle!

There are three major effects that determine the moderator coefficient:

1) core burnup,
2) reactor power level,
3) xenon concentration.

As stated earlier, the burnup effect of the coefficient varies from about -8 at beginning of cycle to about -35, at end of cycle. The use of a 99% coefficient (approximately 70 hours) means that burnup will have a small, but beneficial effect. The second term, reactor power, affects the coefficient in the following ways: 1) as power increases so does the average coolant temperature; reducing the coefficient and 2) as power increases, the average boron concentration decreases to compensate for the doppler feedback; also reducing the coefficient. The net effect is that by the time full power is obtained the coefficient is reduced to at least -3 pcm/°F.

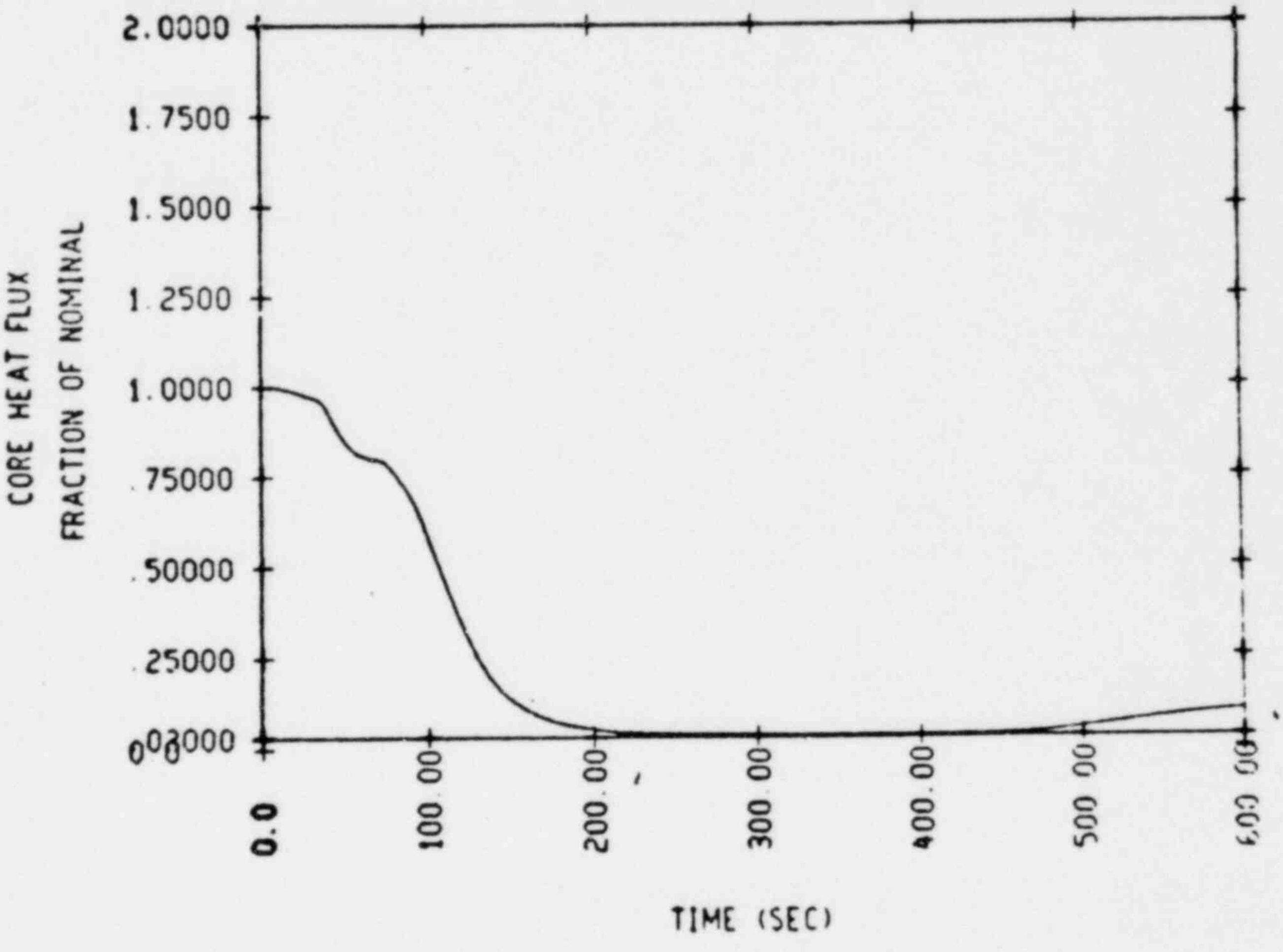


Figure 18. Loss of Normal Feedwater - Reference Case

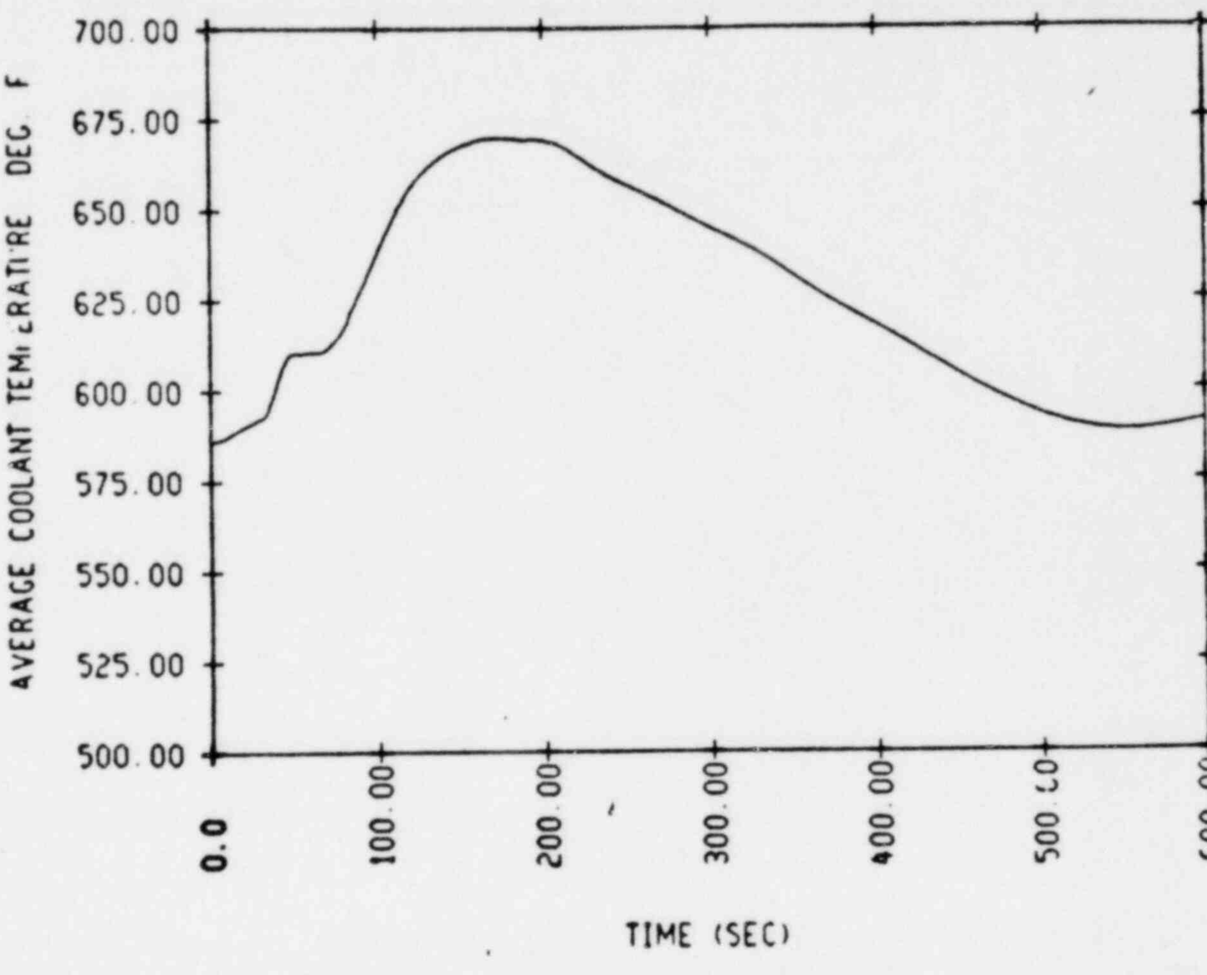


Figure 19. Loss of Normal Feedwater - Reference Case

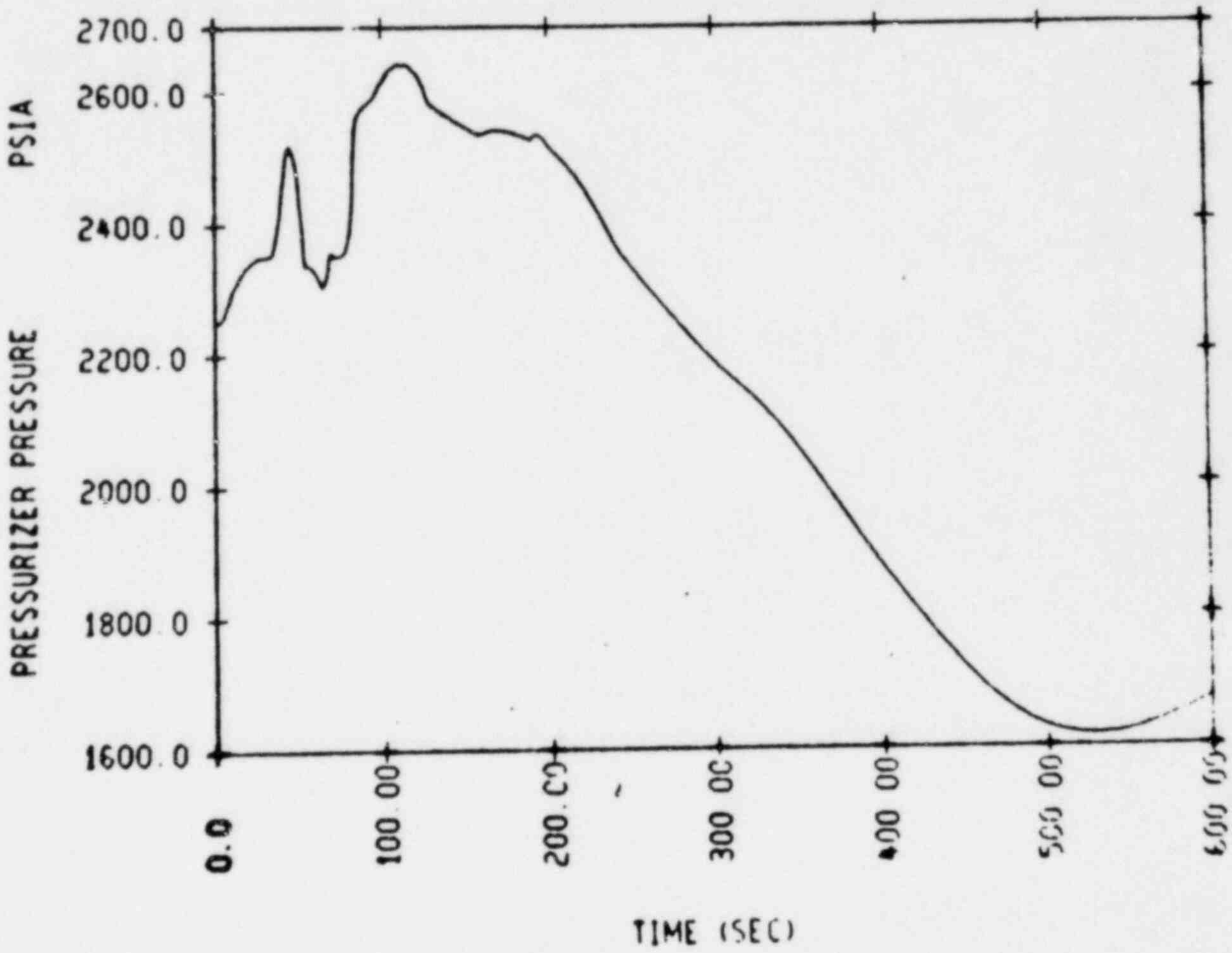


Figure 20. Loss of Normal Feedwater - Reference Case

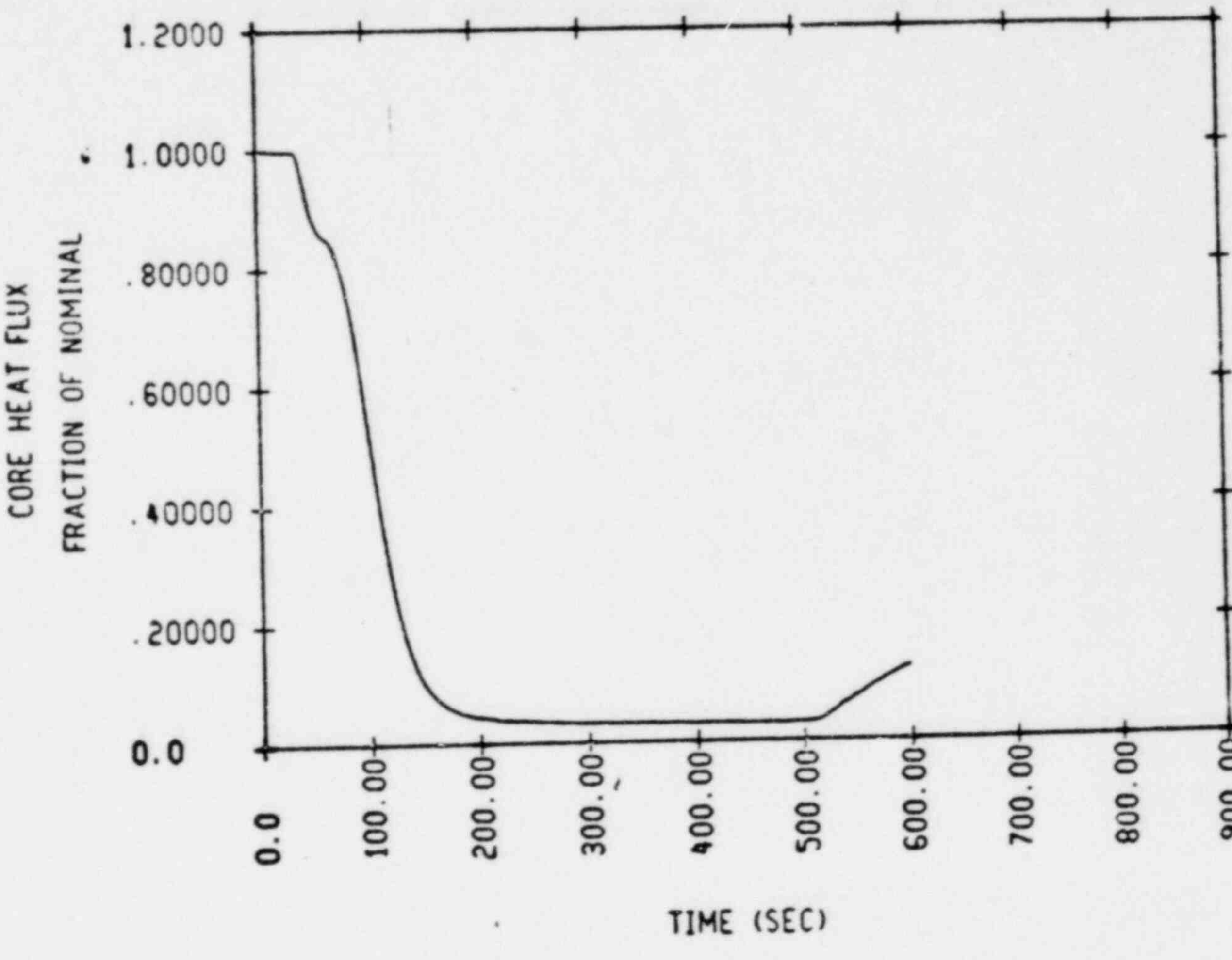


Figure 21. Loss of Normal Feedwater Average Temperature +8°F

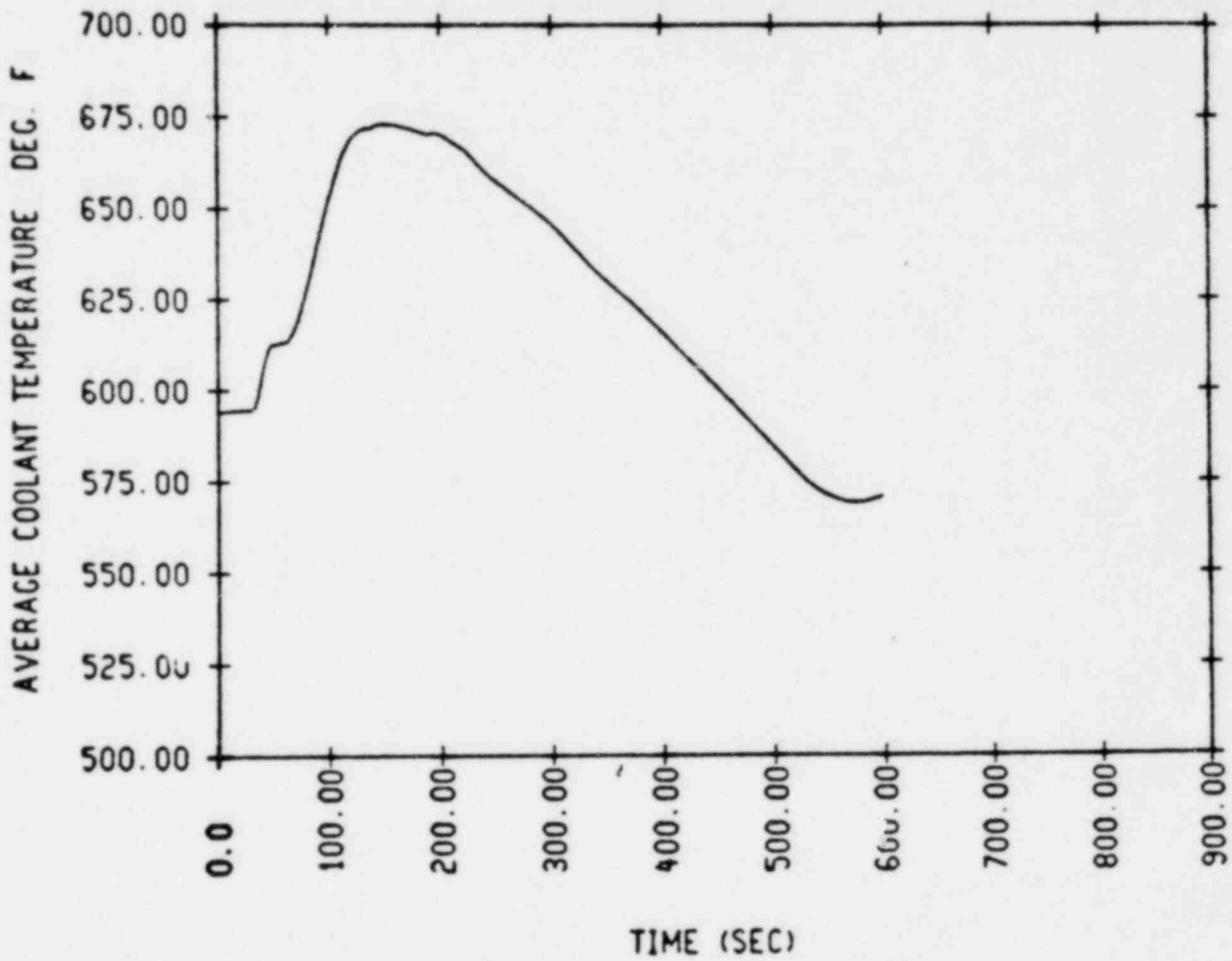


Figure 22, Loss of Normal Feedwater Average Temperature +8°F

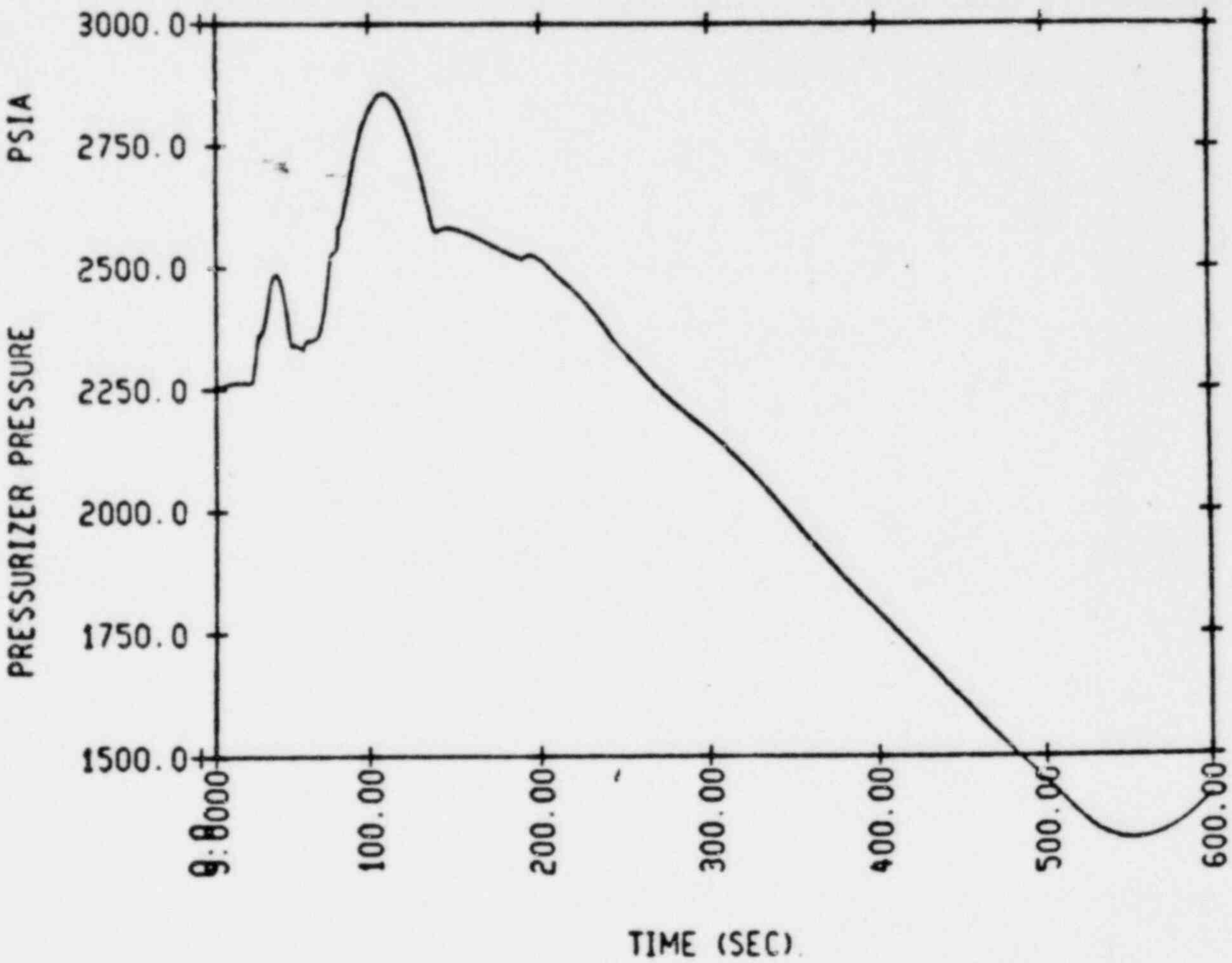


Figure 23. Loss of Normal Feedwater Average Temperature +8°F

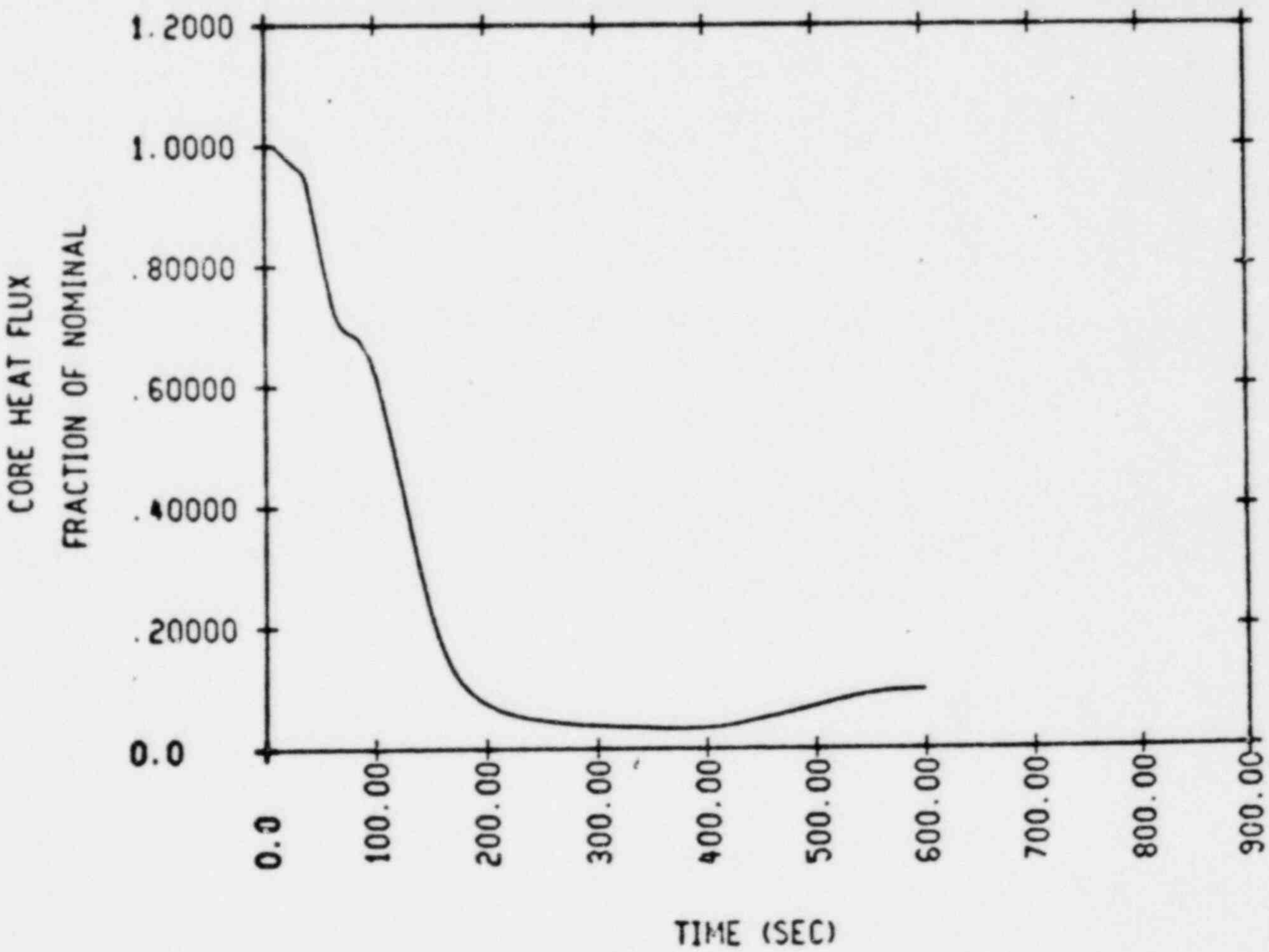


Figure 24. Loss of Normal Feedwater Average Temperature -20°F

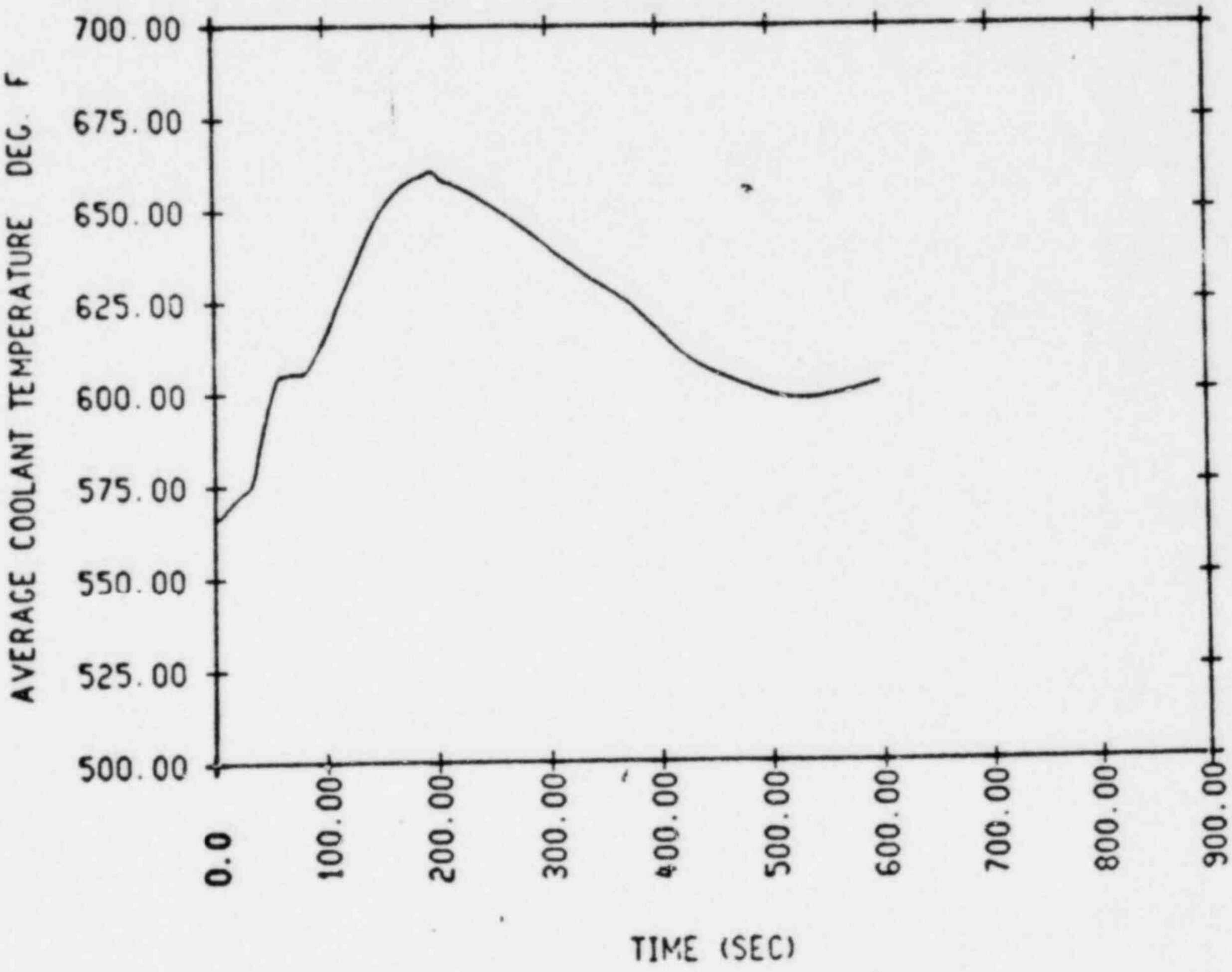


Figure 25. Loss of Normal Feedwater Average Temperature -20°F

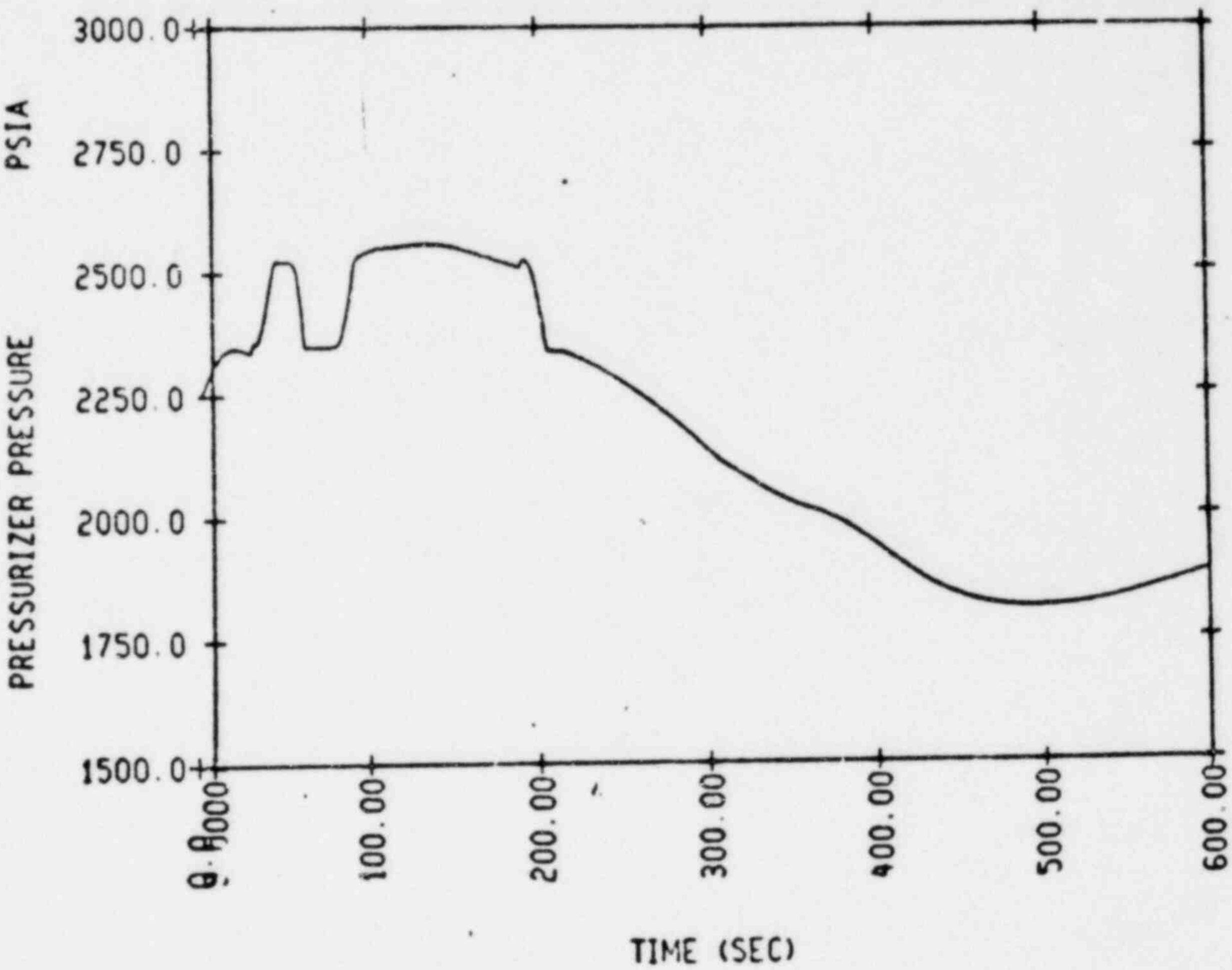


Figure 26. Loss of Normal Feedwater Average Temperature -20°F

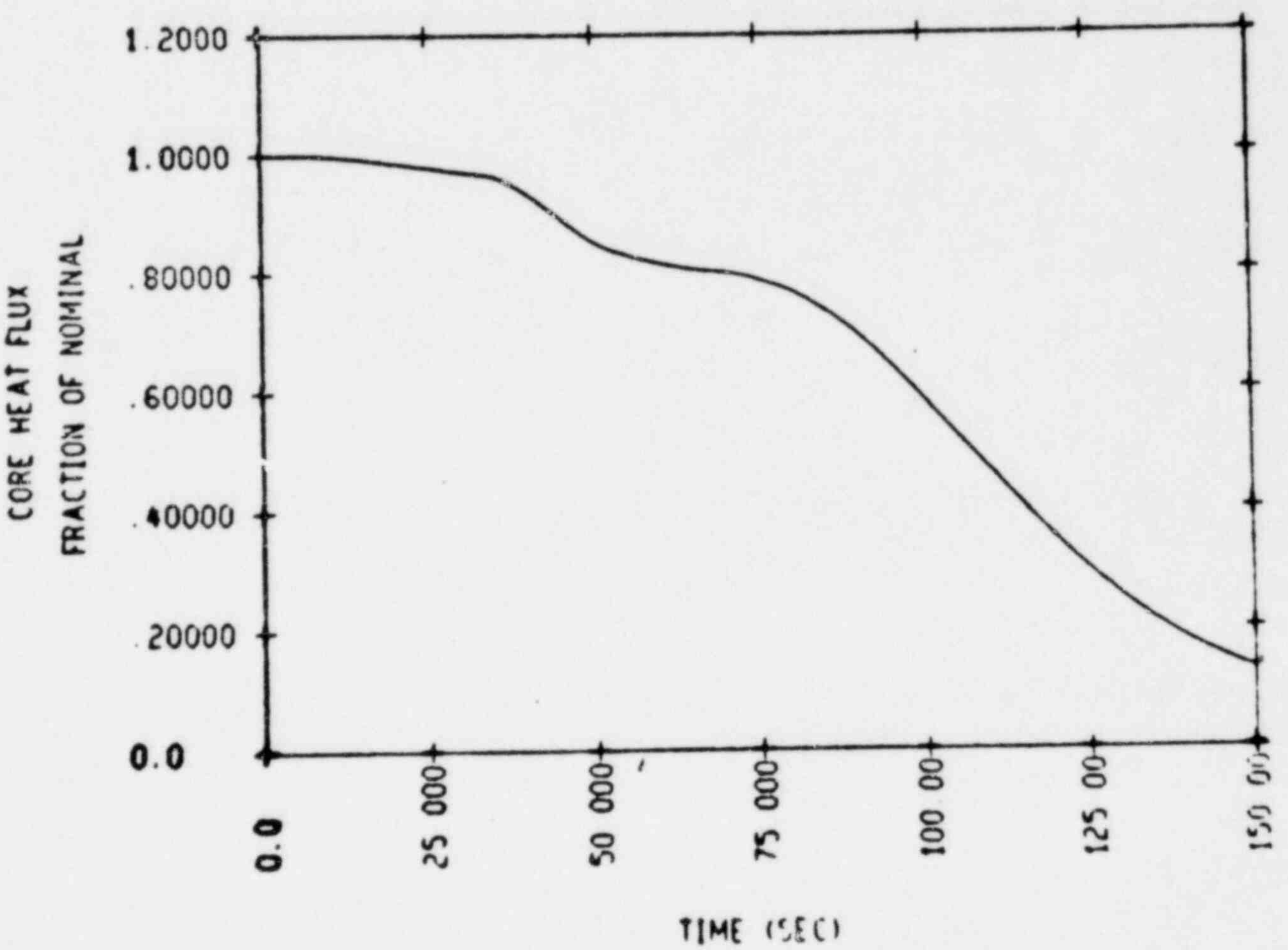


Figure 27. Loss of Normal Feedwater - One Relief Valve Failed

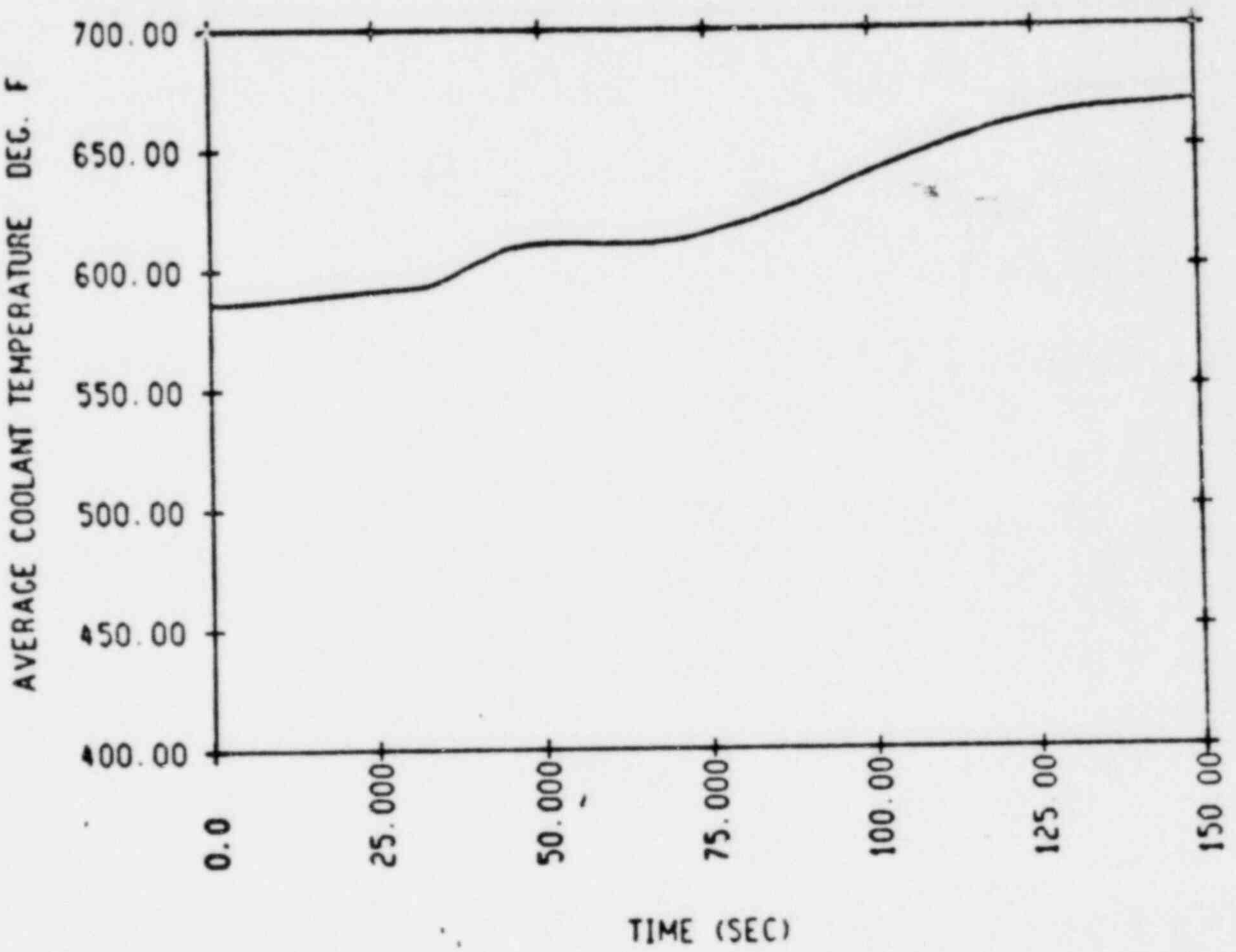


Figure 28. Loss of Normal Feedwater - One Relief Va.ve Failed

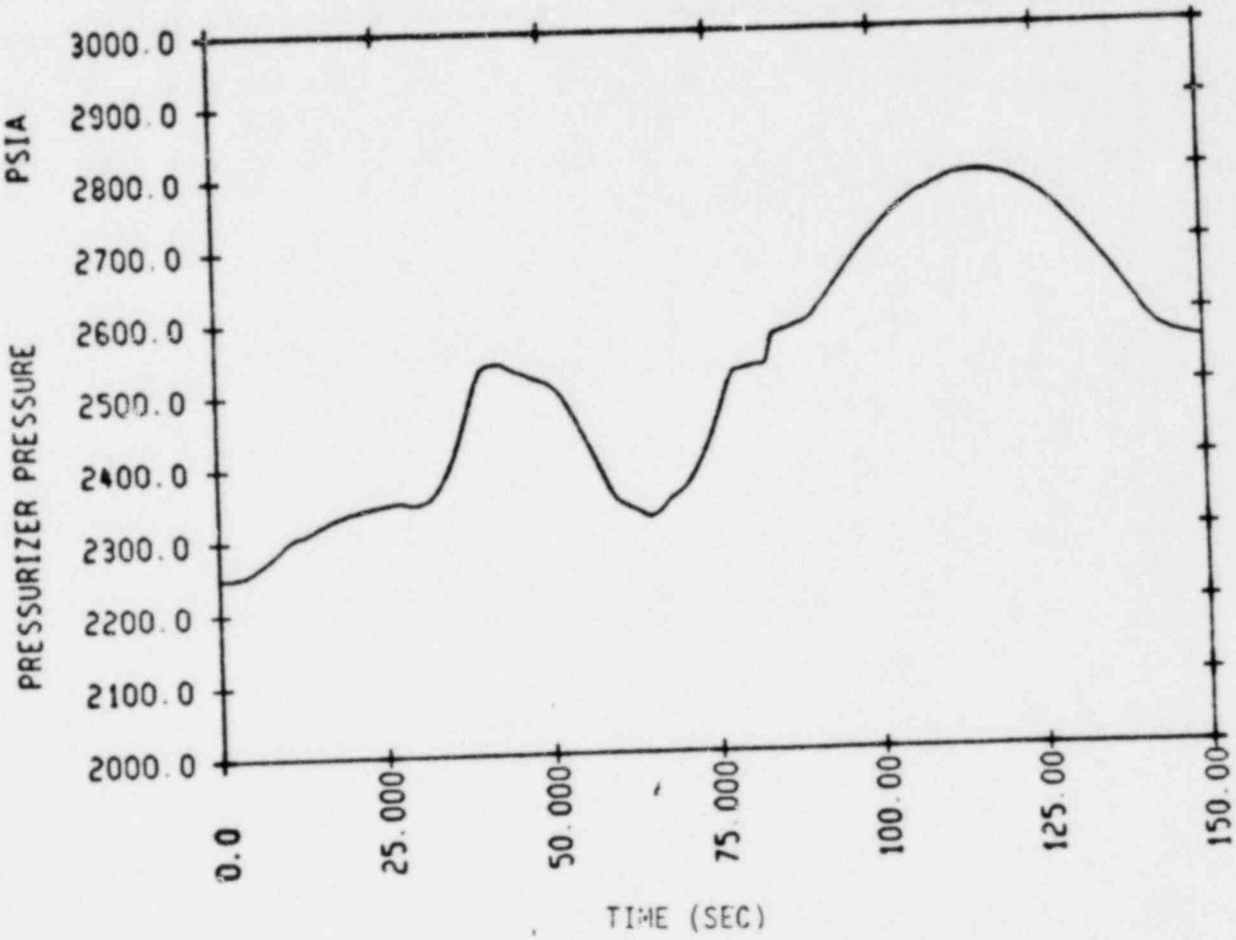


Figure 29. Loss of Normal Feedwater - One Relief Valve Failed

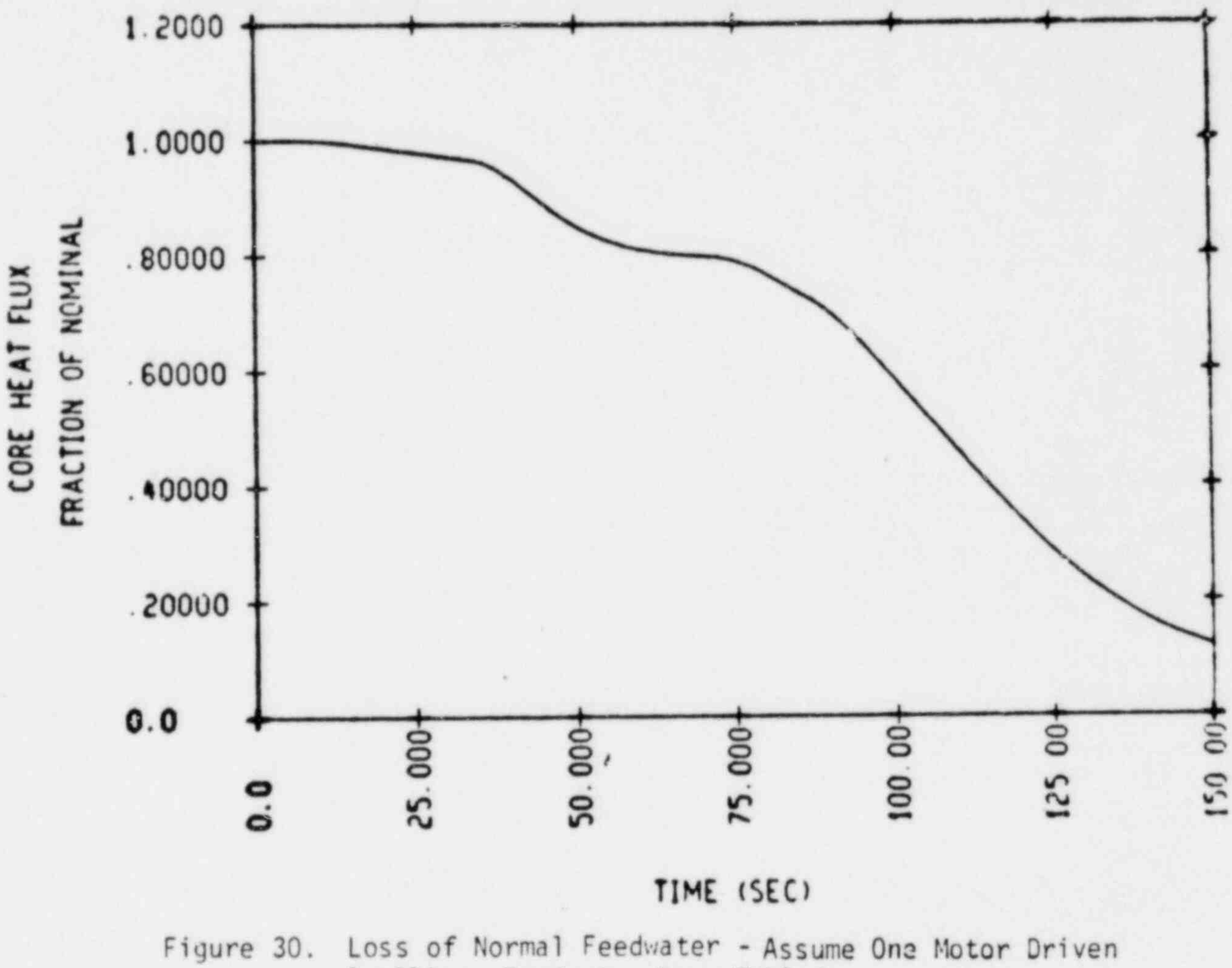


Figure 30. Loss of Normal Feedwater - Assume One Motor Driven Auxiliary Feedwater Pump Failed

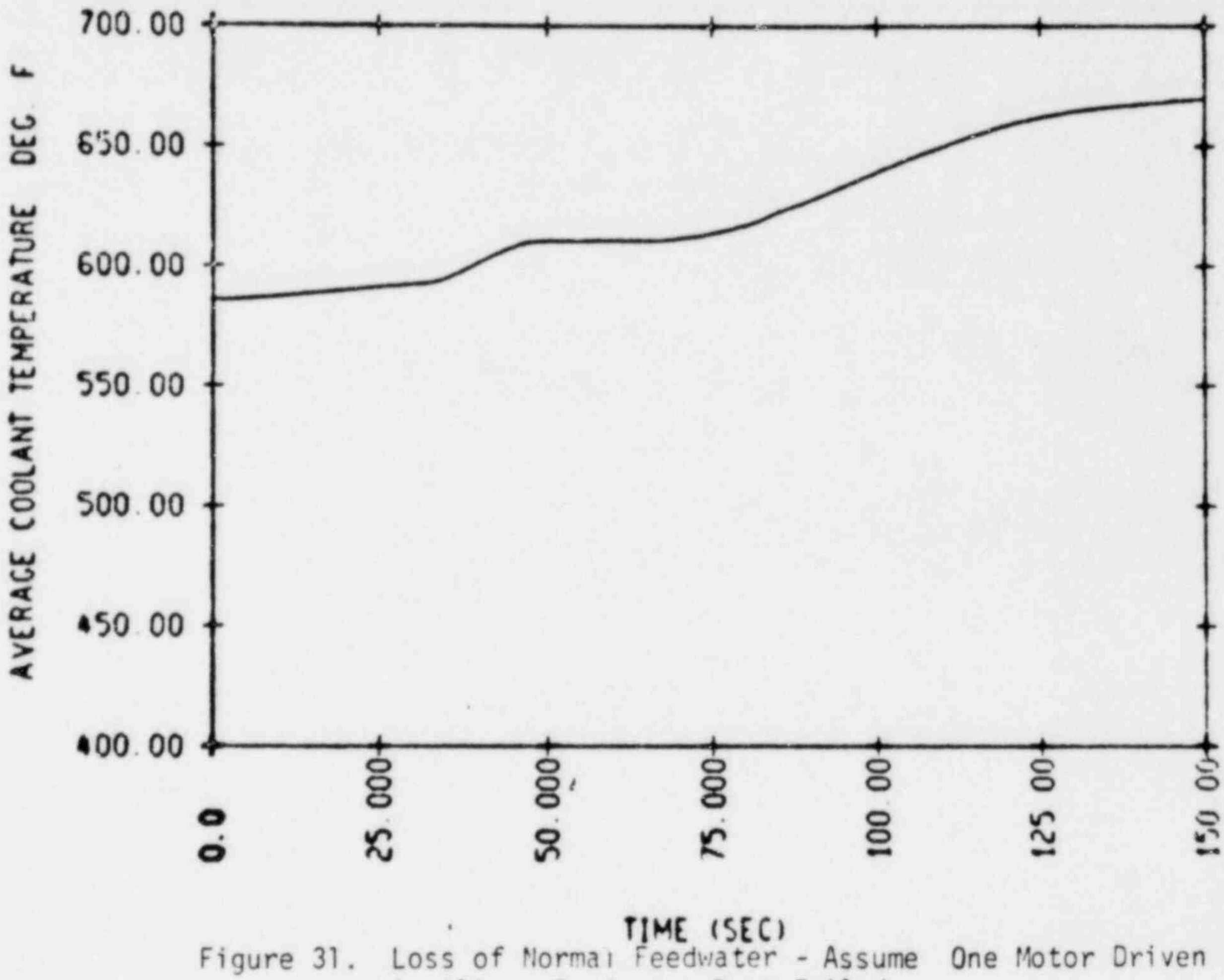


Figure 31. Loss of Normal Feedwater - Assume One Motor Driven Auxiliary Feedwater Pump Failed

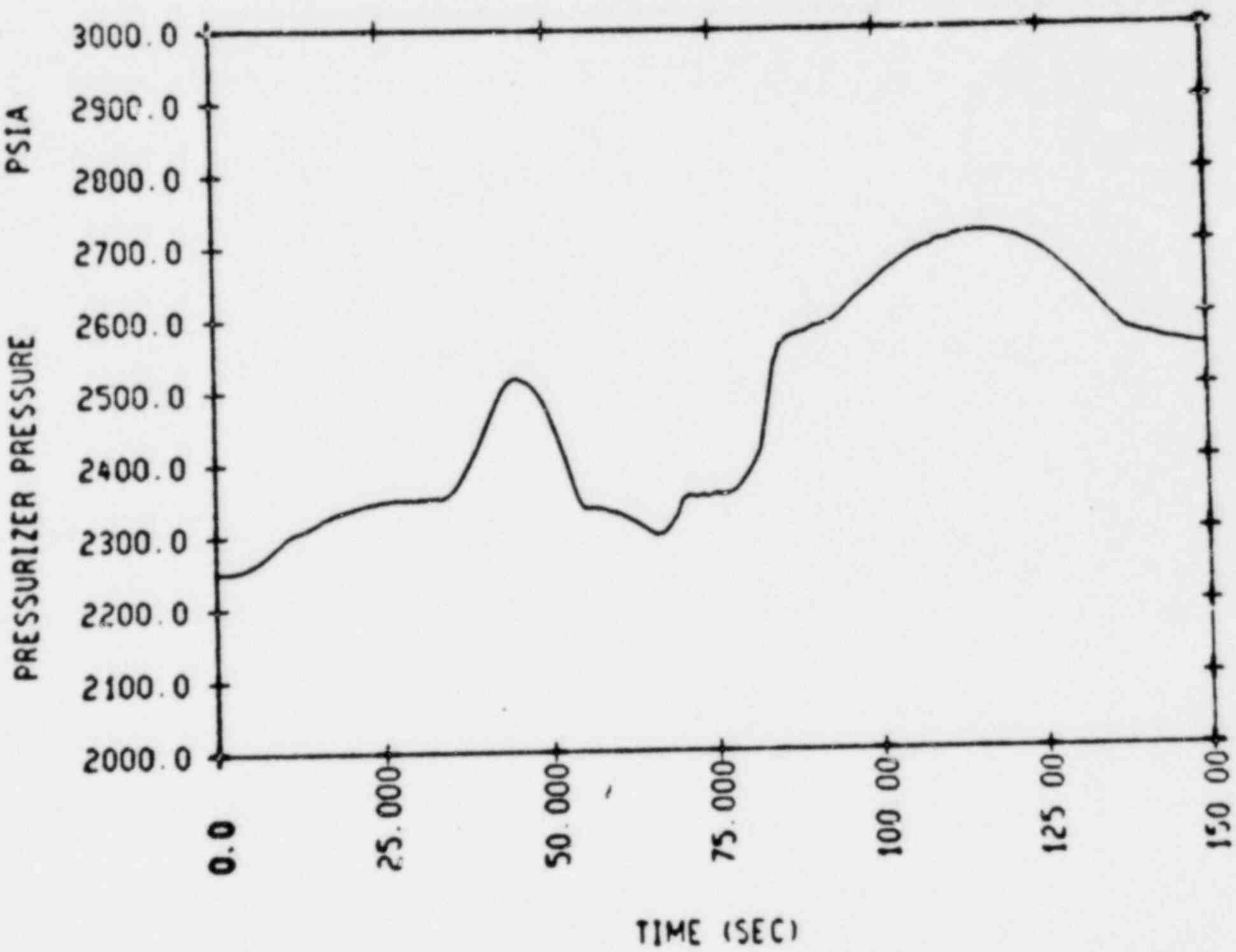


Figure 32. Loss of Normal Feedwater - Assume One Motor Driven Auxiliary Feedwater Pump Failed

1.8 LOSS OF OFFSITE POWER

The loss of offsite power is the limiting transient with regard to the approach to DNB. The reference case from WCAP-8330 was reanalyzed assuming a moderator coefficient valid 99% of the time. One of the consequences of a loss of offsite power without a reactor trip is a change in the axial flux shape due to the increased temperature. The axial shape was determined for these analyses from a three dimensional nuclear code that allows for feedback effects due to coolant temperature and fuel temperature.

The sensitivity studies presented in WCAP-8330 can be used to see the effects of: initial steam generator mass, initial power, flow coastdown rates, and automatic rod control.

The system response for the reference are shown in Figures 33 through 36. For the initial coolant varying from +8 to -20, the resulting DNBR's are shown in Figures 37 and 38 respectively. For the case of Tave +8°F the calculated radial peaking factor was used. For plants with 15 x 15 fuel, the corresponding maximum power and average temperature were used with the resulting minimum DNBR shown in Figure 39.

To show the effect of a single failure, a power operated relief valve was assumed to fail. The resulting pressure and DNBR are shown in Figures 40 and 41. Failures in the auxiliary feedwater system do not effect the minimum DNBR.

Key results are shown in Table 4 in WCAP-8330. All of the resulting DNBR's were greater than 1.30. For these analyses the reference case minimum DNBR is 1.296 which is so close to a value of 1.30 that the conclusions that no fuel damage is still valid.

TABLE 4

LOSS OF OFFSITE POWER

Case	Peak System Pressure Psi	Minimum DNBR
Reference	2663	1.296
Tave + 8°F	2654	1.219
Tave - 20°F	2651	1.879
15 x 15 fuel	2637	1.702
Assume 1 relief valve failed	2677	1.293

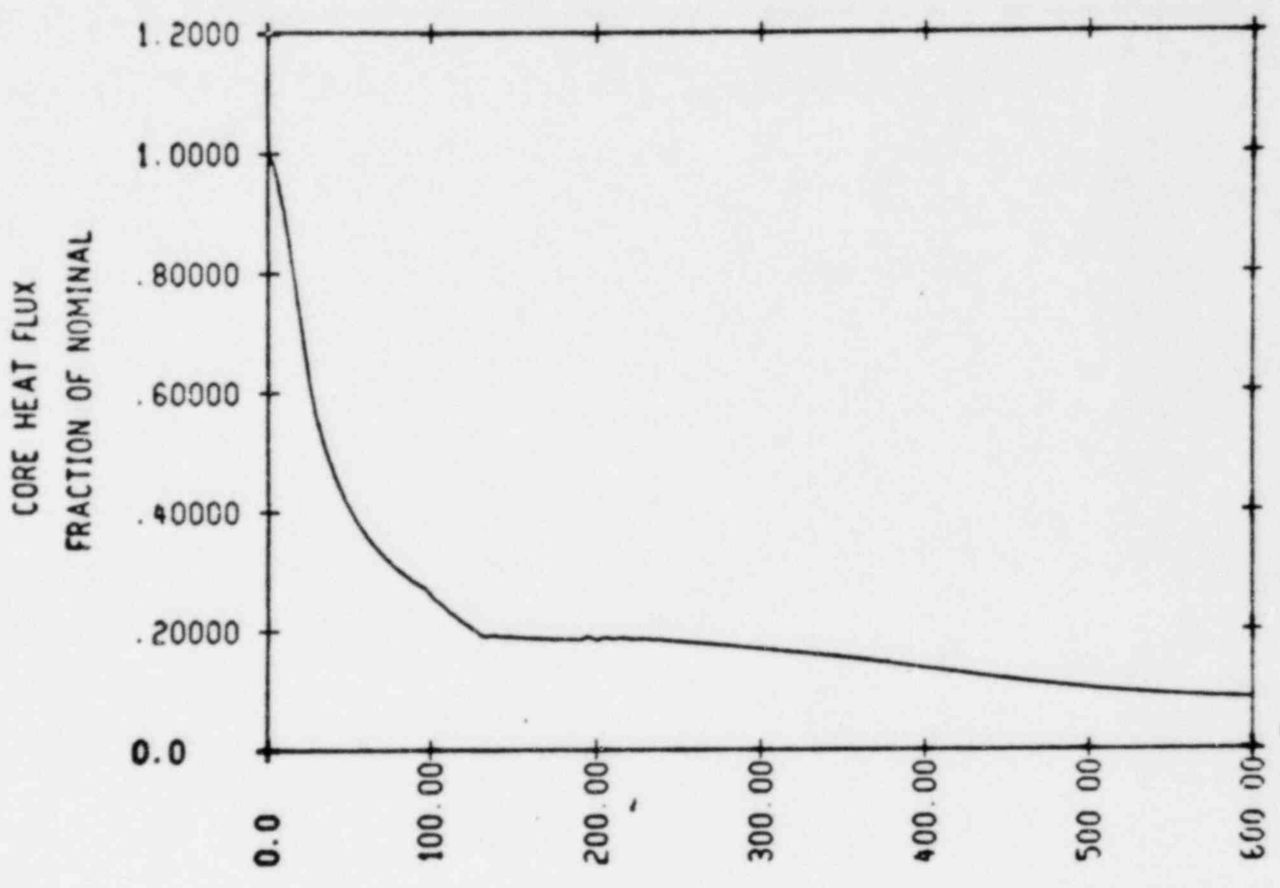


Figure 33. Loss of Offsite Power - Reference Case

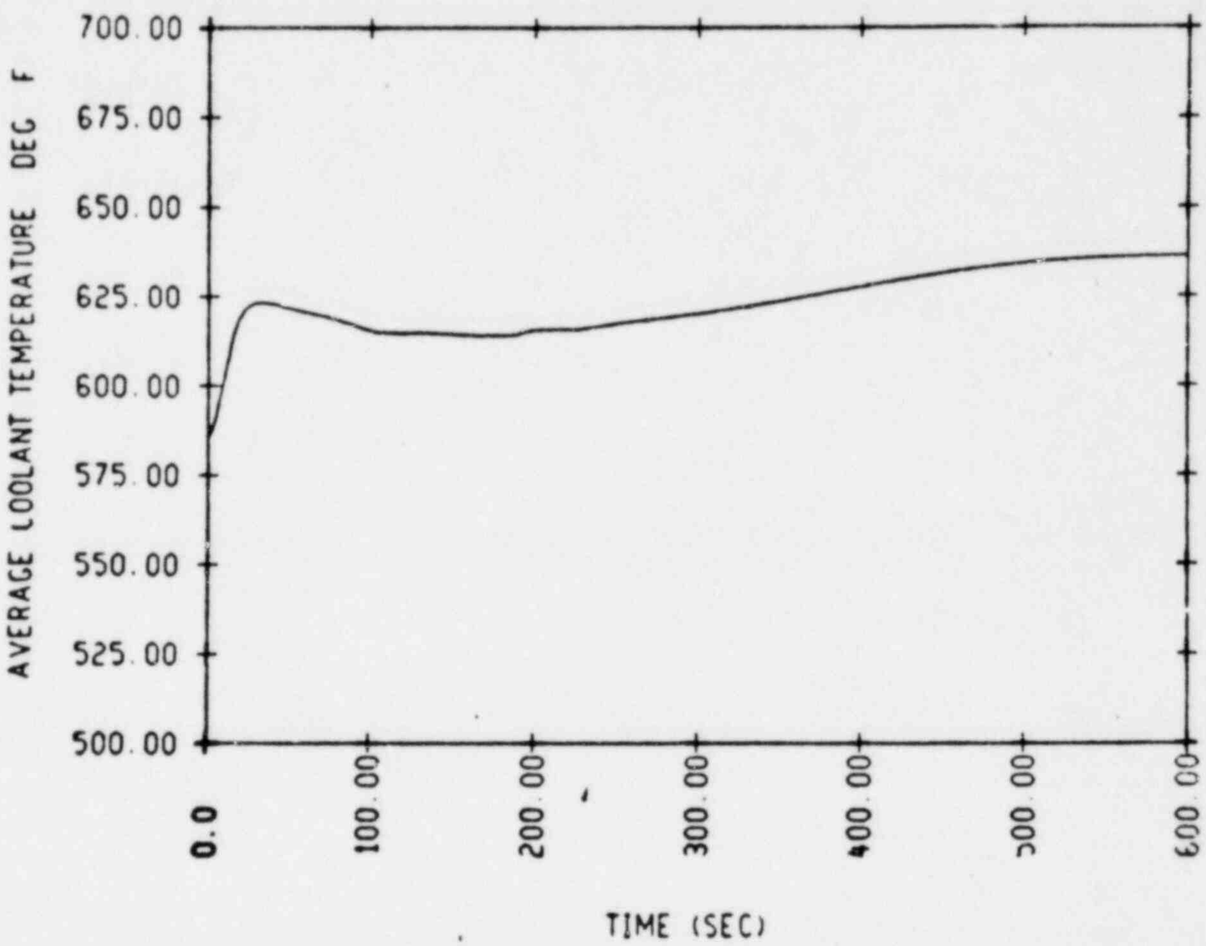


Figure 34. Loss of Offsite Power - Reference Case

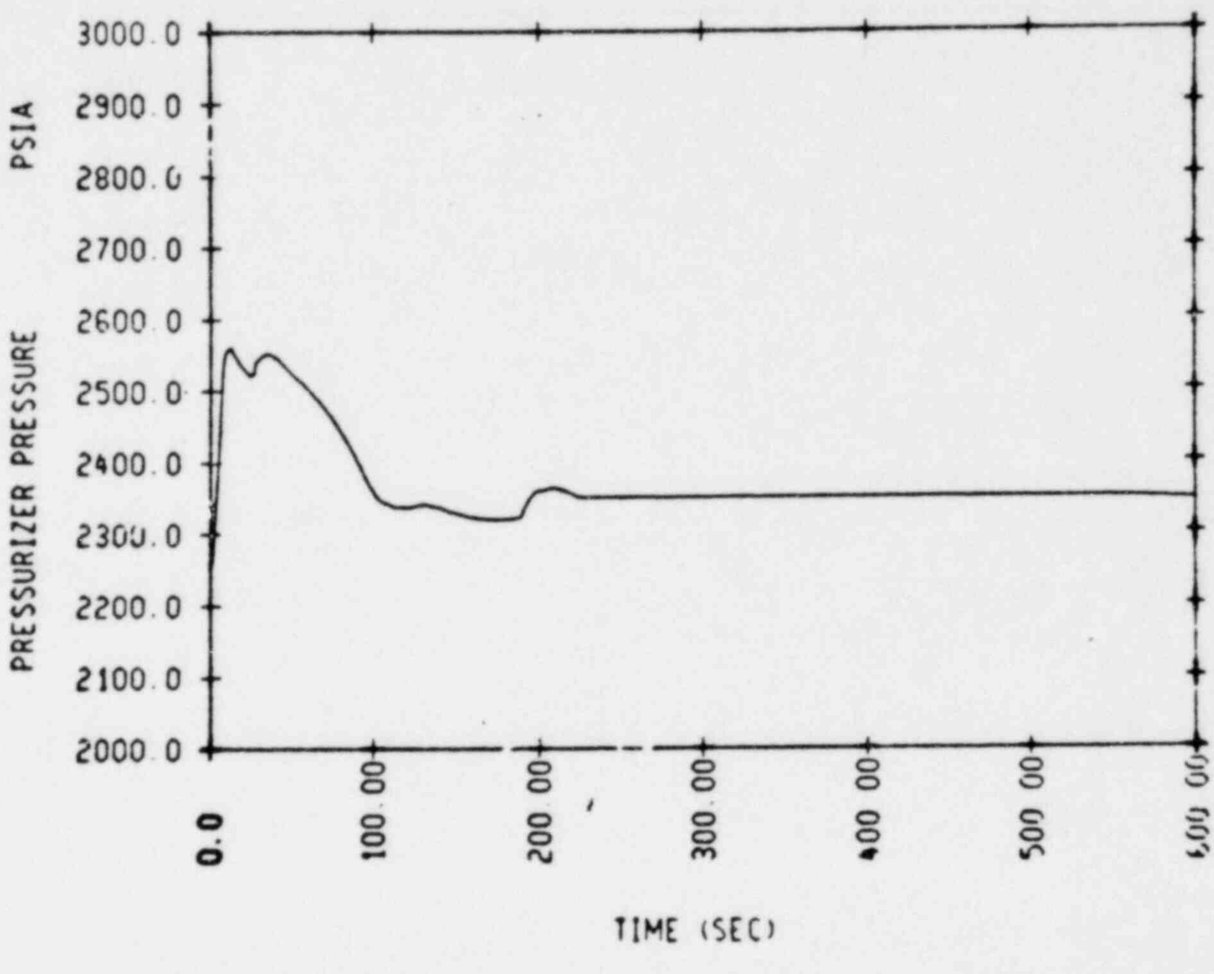


Figure 35. Loss of Offsite Power - Reference Case

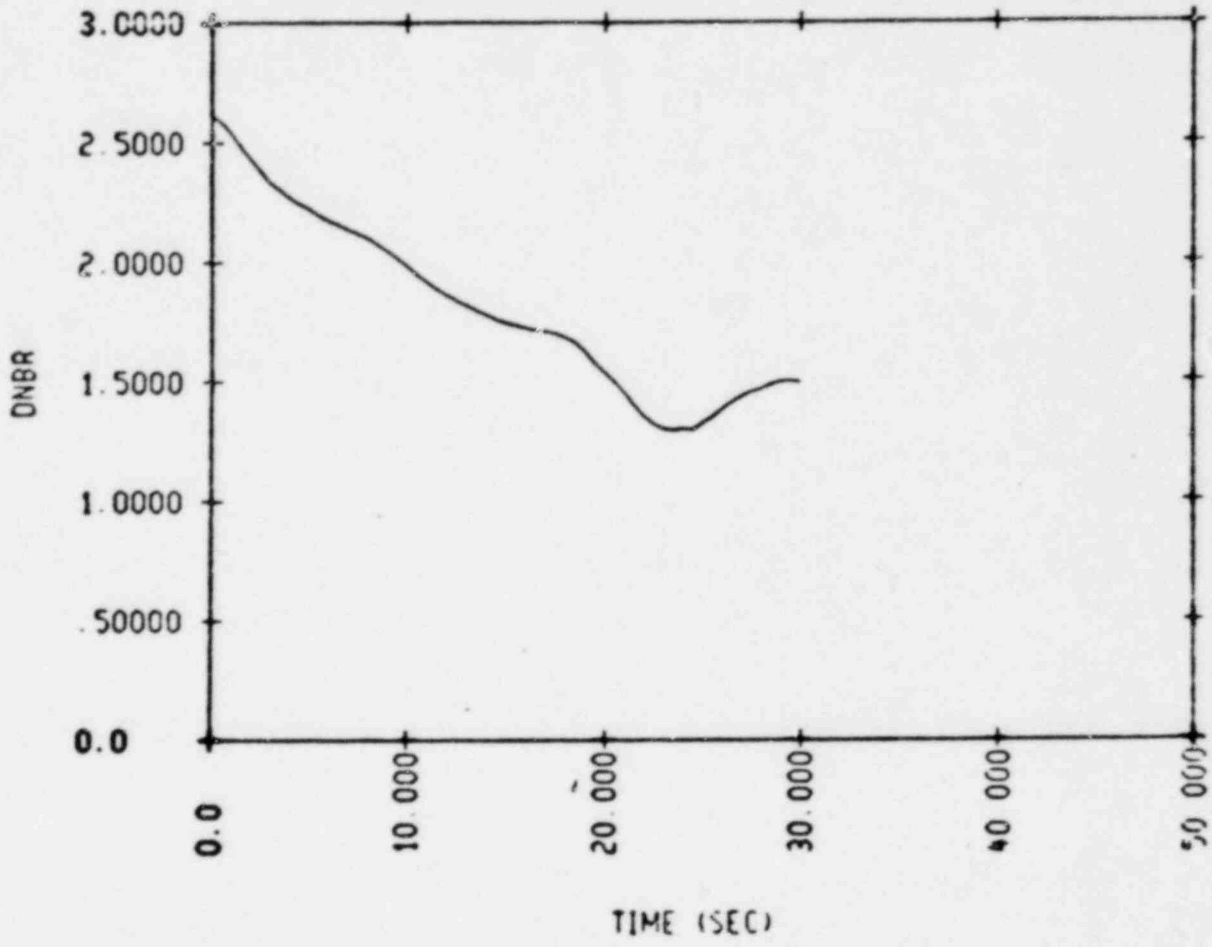


Figure 36. Loss of Offsite Power - Reference Case

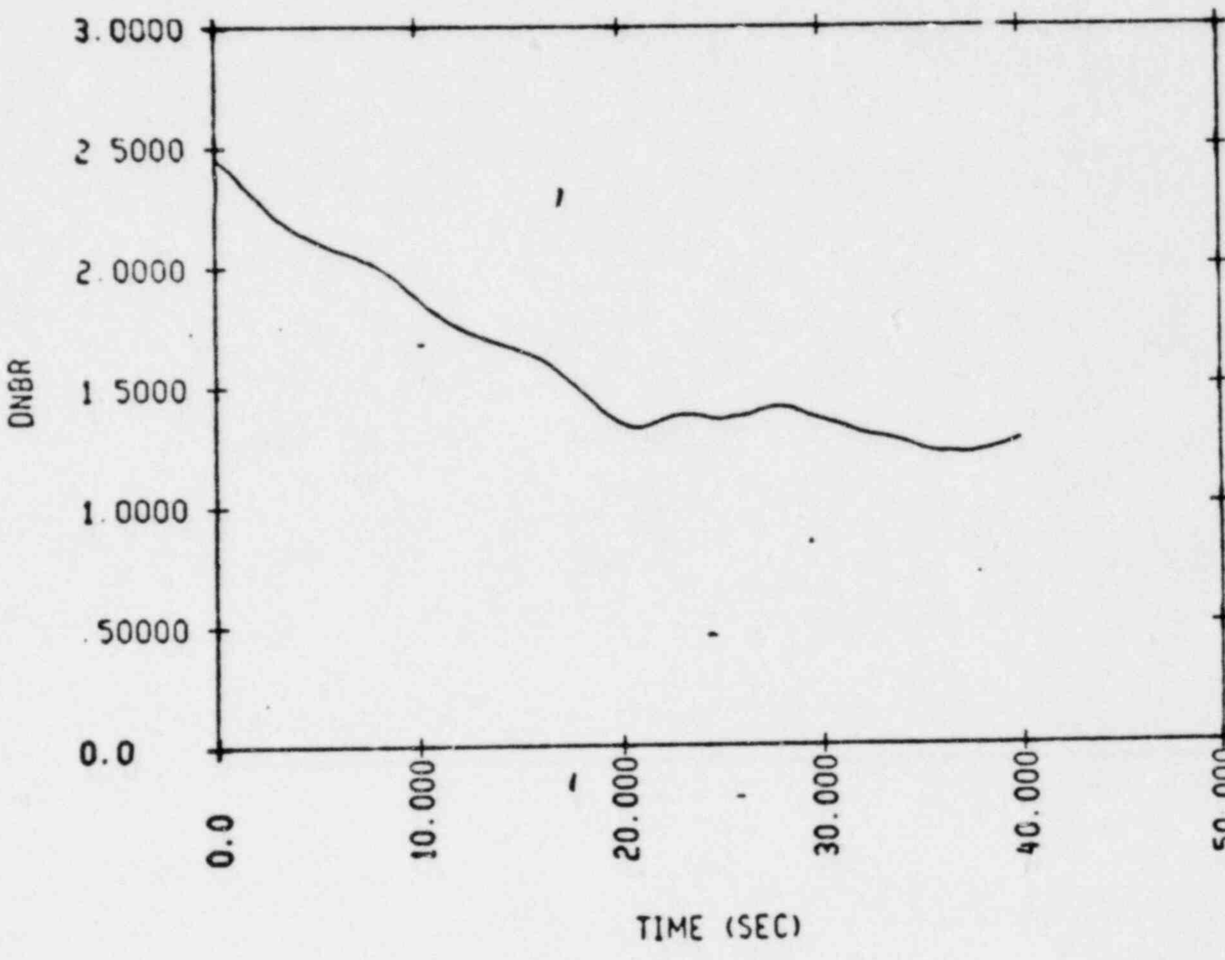


Figure 37. Loss of Offsite Power - Average Temperature +[illegible]°F

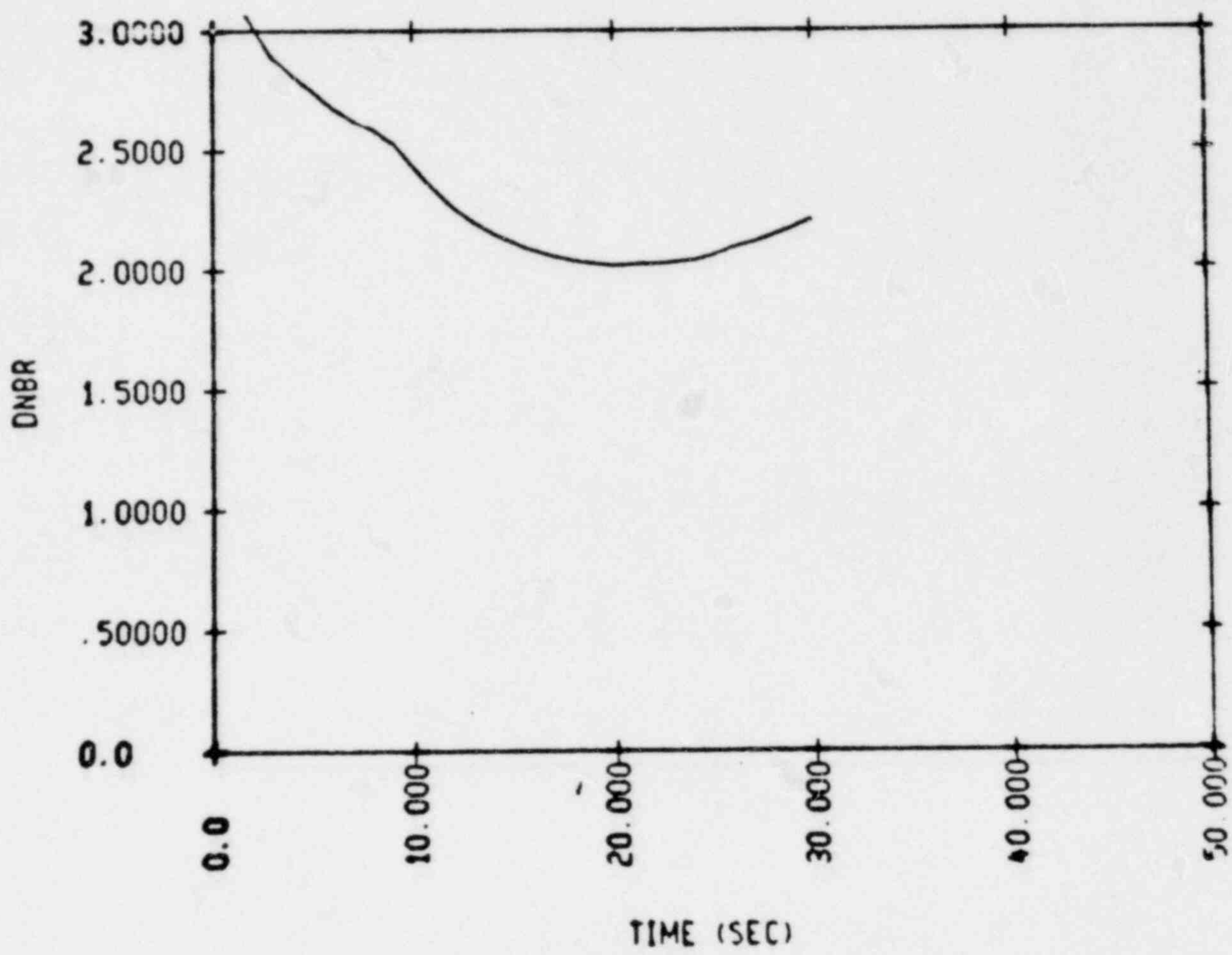


Figure 38. Loss of Offsite Power - Average Temperature - 20°F

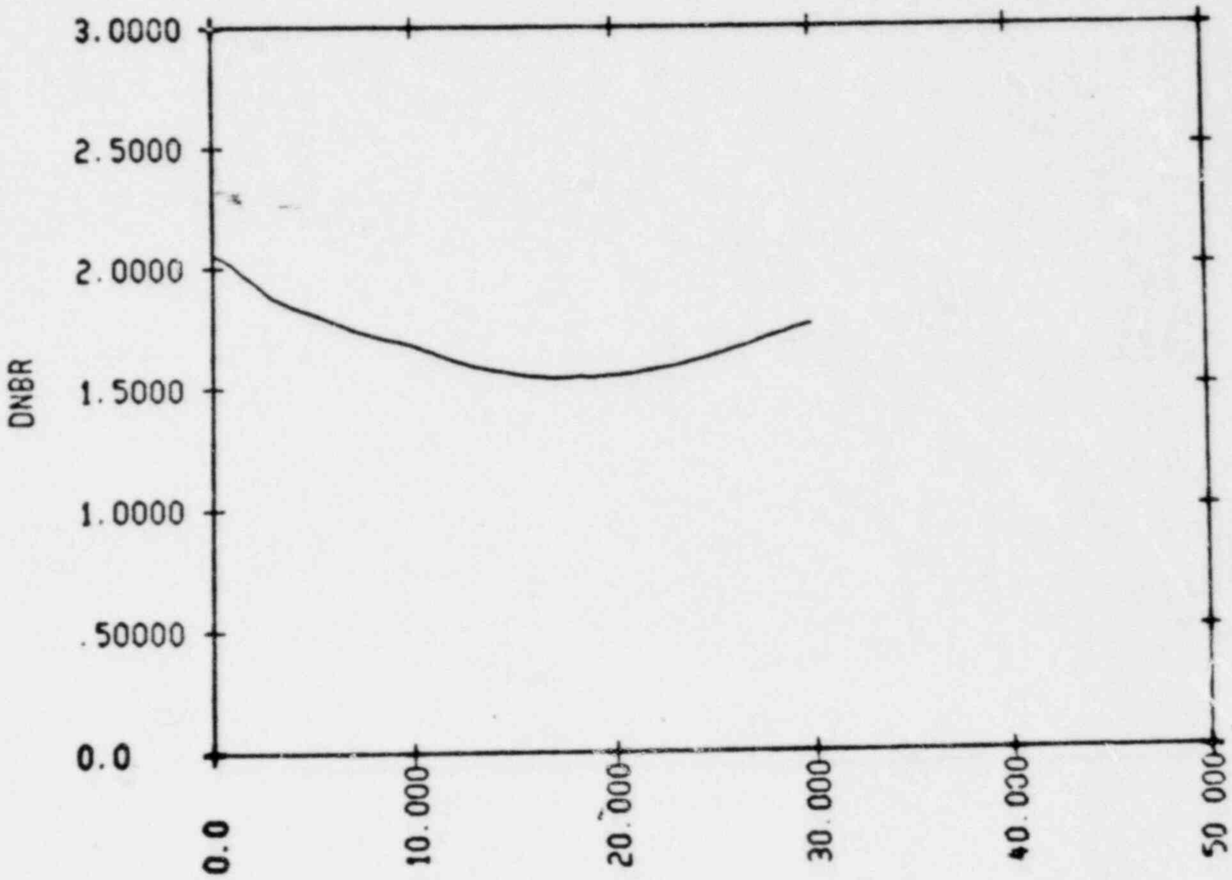


Figure 39. Loss of Offsite Power - 15 x 15

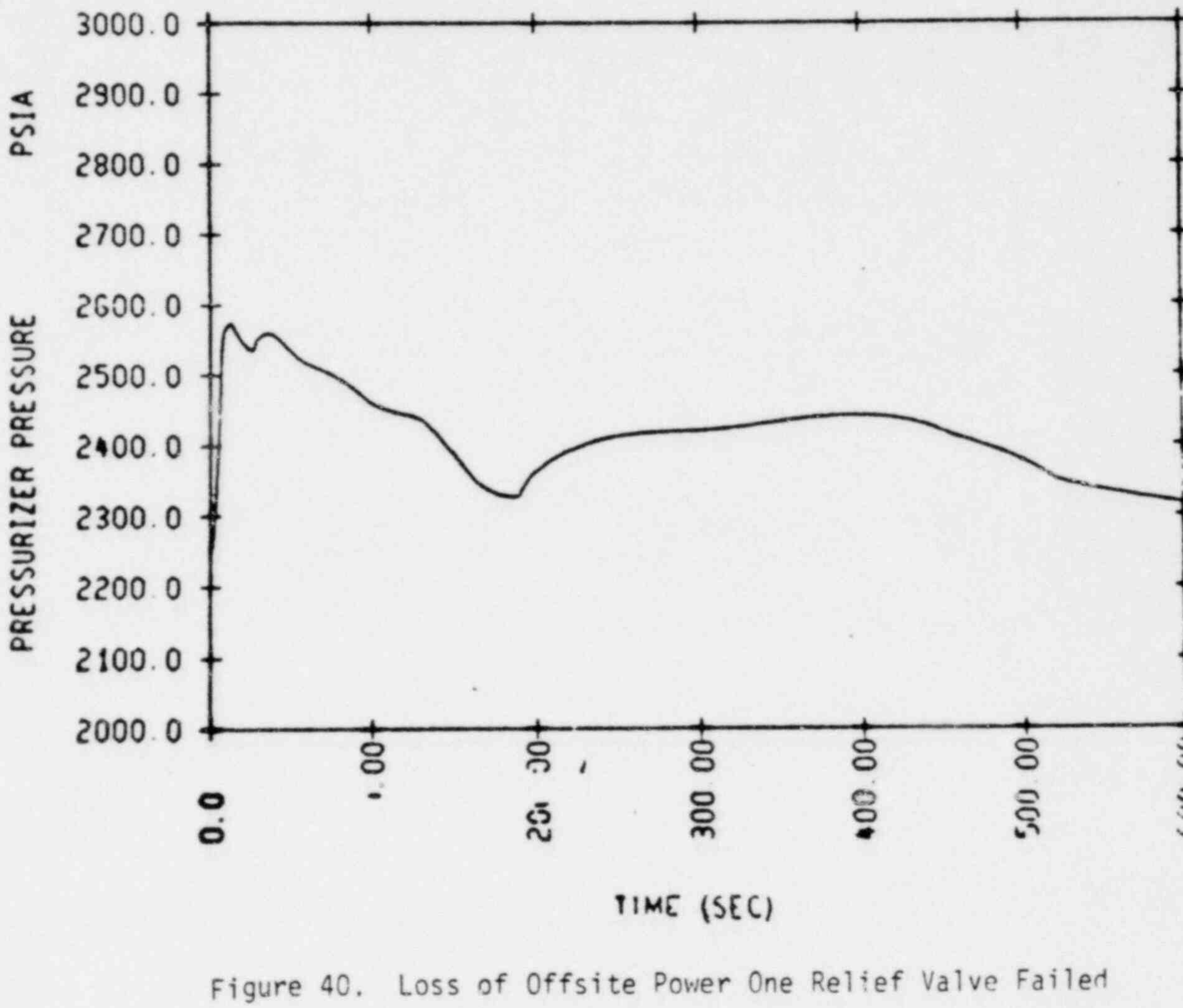


Figure 40. Loss of Offsite Power One Relief Valve Failed

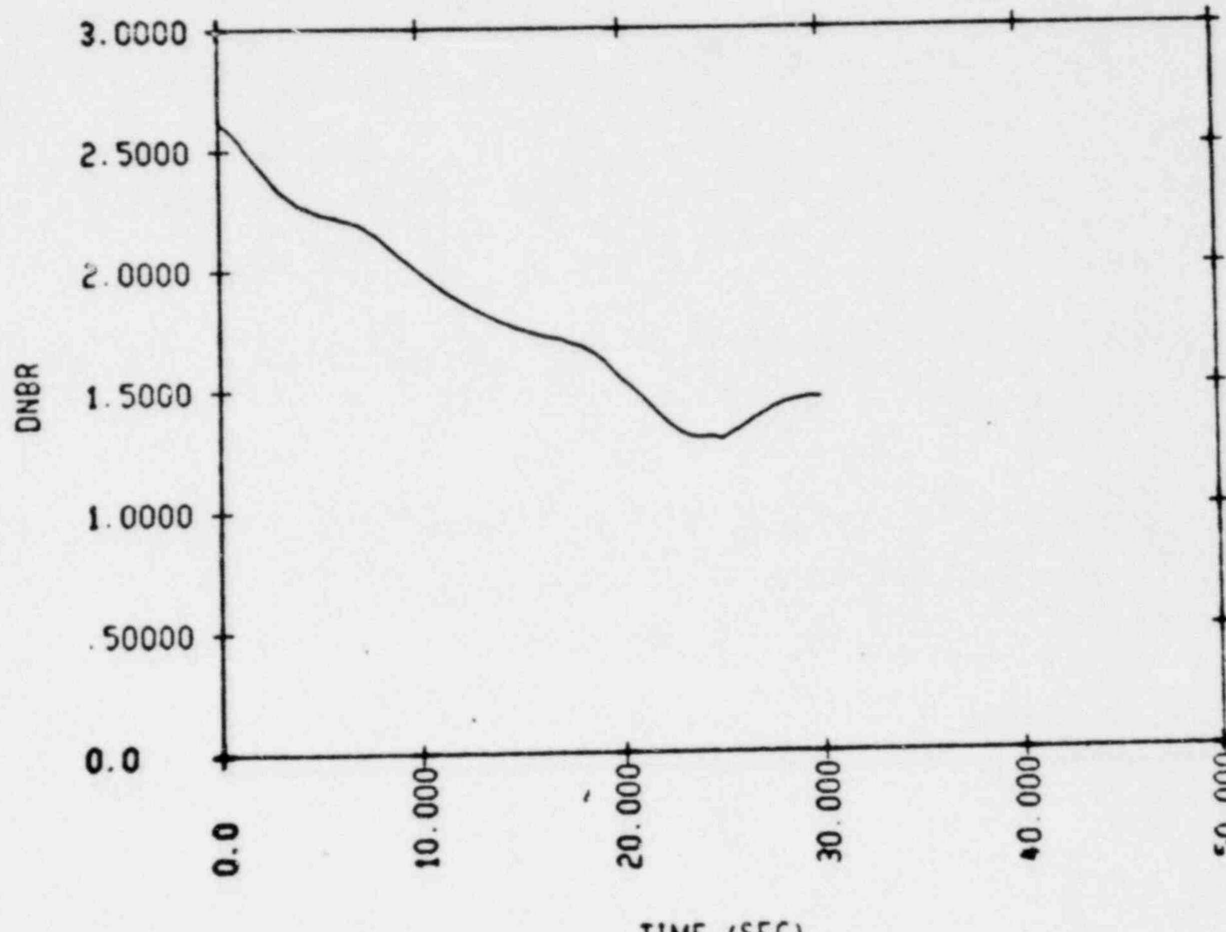


Figure 41. Loss of Offsite Power One Relief Valve Failed

1.9 EXCESSIVE LOAD INCREASE

As discussed in WCAP-8330, a reactor trip is not requi d for this transient. Therefore, the effect of using a 99% coeffici..t versus 95% will not change the conclusion.

1.10 ACCIDENTAL DEPRESSURIZATION OF THE REACTOR COOLANT SYSTEM

The accidental depressurization of the reactor coolant system was re-analyzed using a 99% moderator coefficient. The reference case in WCAP-8330 was for a 4 loop plant with 15 x 15 fuel. To be consistent with all of the other analysis presented, the reference case has been changed to a 4 loop plant with 17 x 17 fuel and the 15 x 15 case was treated as a sensitivity study. The sensitivity studies presented in WCAP-8330 should be used to judge the effects of initial power level and relief rates.

The system response for the reference case is shown in Figures 42 - 45. The effect of varying the initial average temperature by +8, -20 is shown in Figures 46 through 49. The effect of 15 x 15 fuel on the minimum DNBR is shown in the table. If automatic rod control is assumed to operate, the average temperature is somewhat higher and hence the minimum DNBR is lower. This is shown in Figures 50 and 51.

Table 5 summarizes the results. For all cases the minimum DNBR is greater than 1.30.

TABLE 5

ACCIDENTAL DEPRESSURIZATION OF THE REACTOR COOLANT SYSTEM

Case	Minimum DNBR
Reference	1.570
Tave + 8°F	1.496
Tave - 20°F	1.707
15 x 15 fuel	1.441
Automatic rod control	1.416

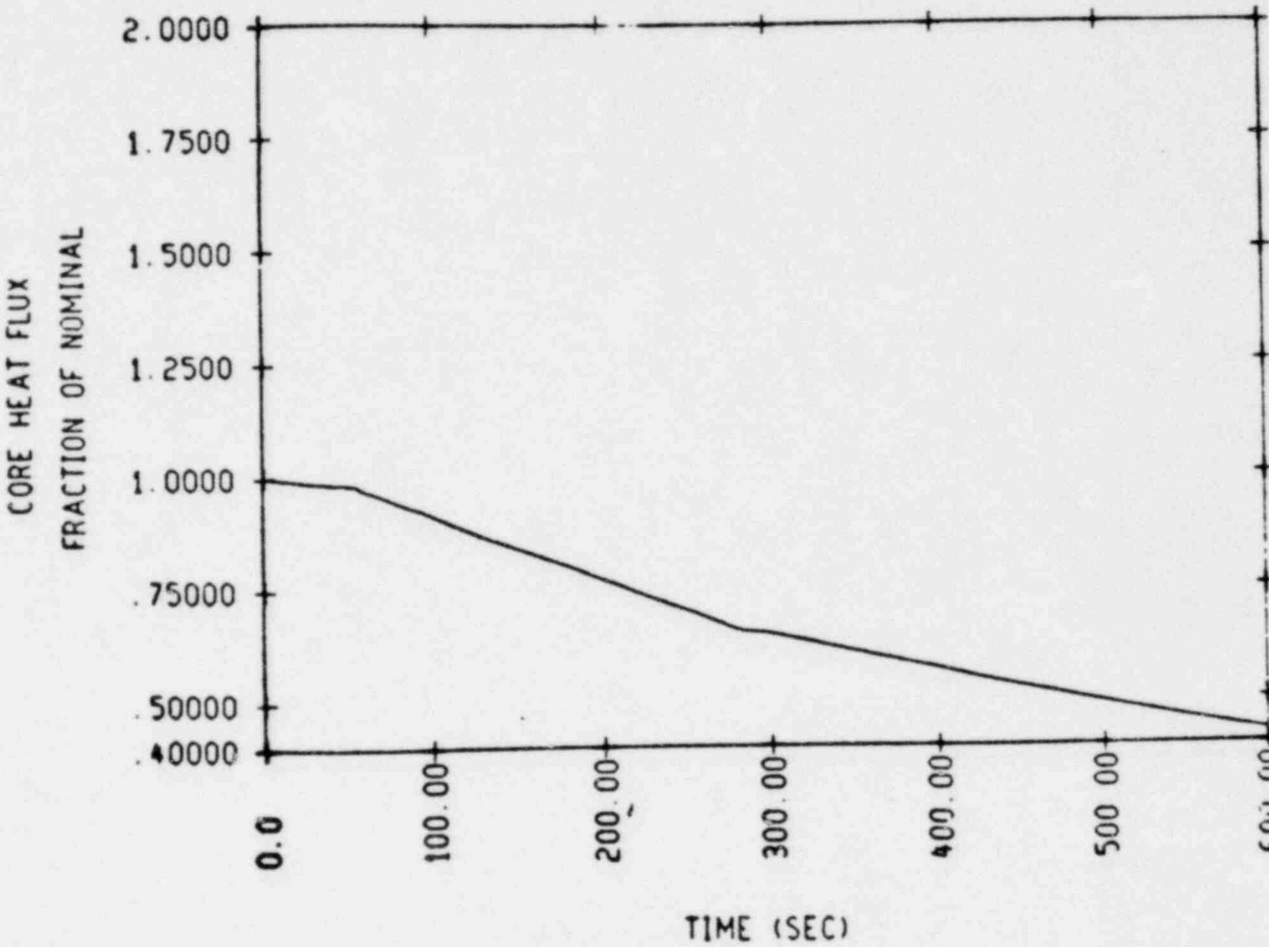


Figure 42. Depressurization - Reference Case

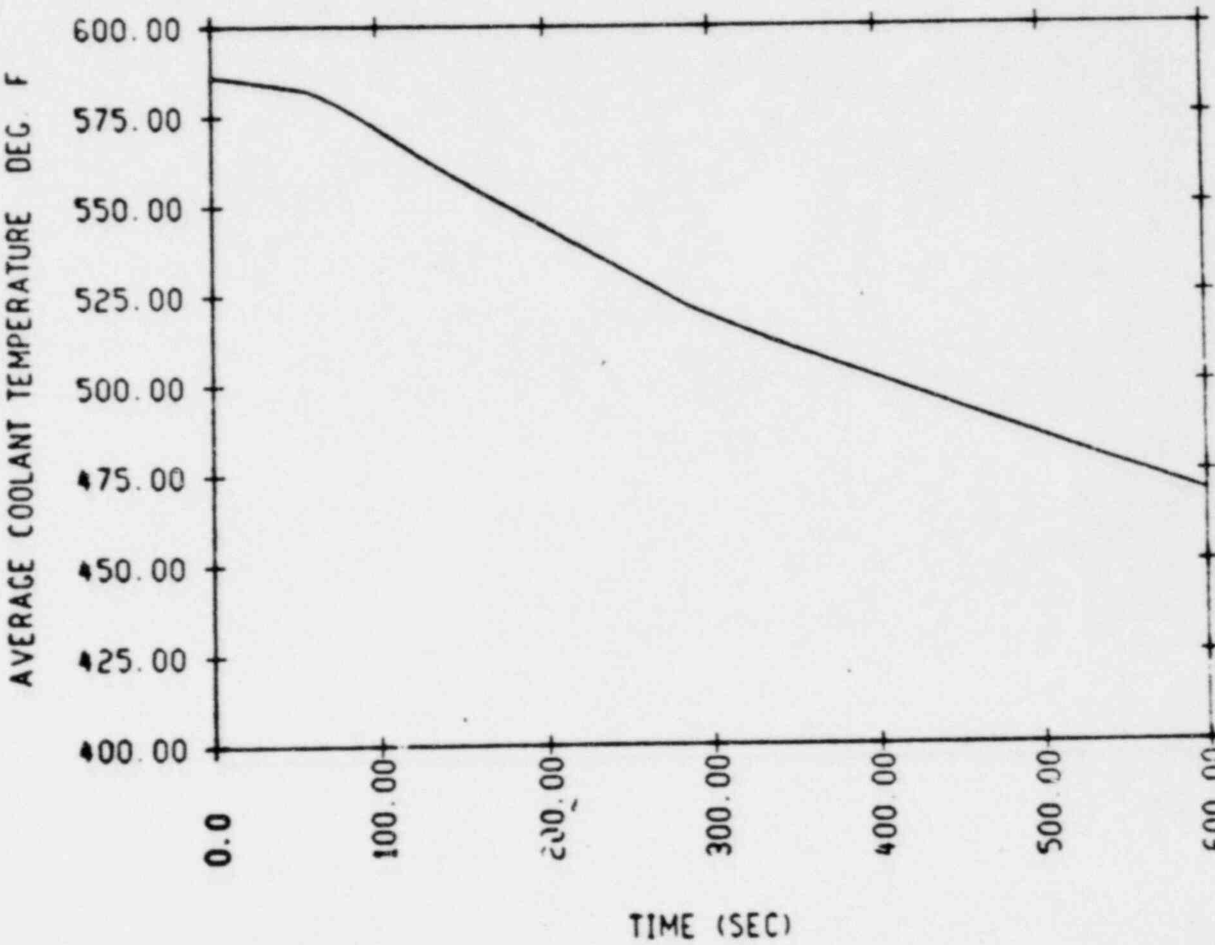


Figure 43. Depressurization - Reference Case

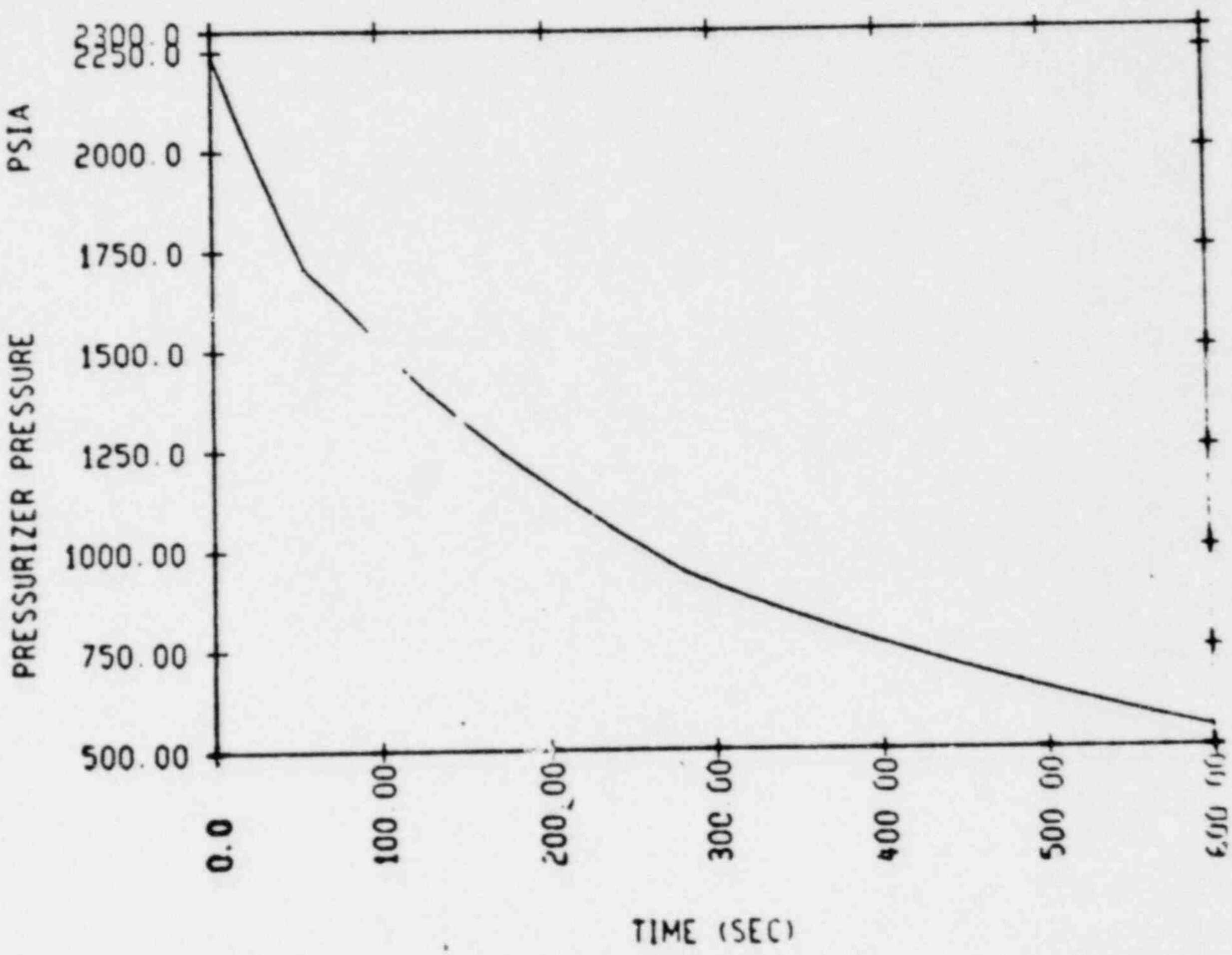


Figure 44. Depressurization - Reference Case

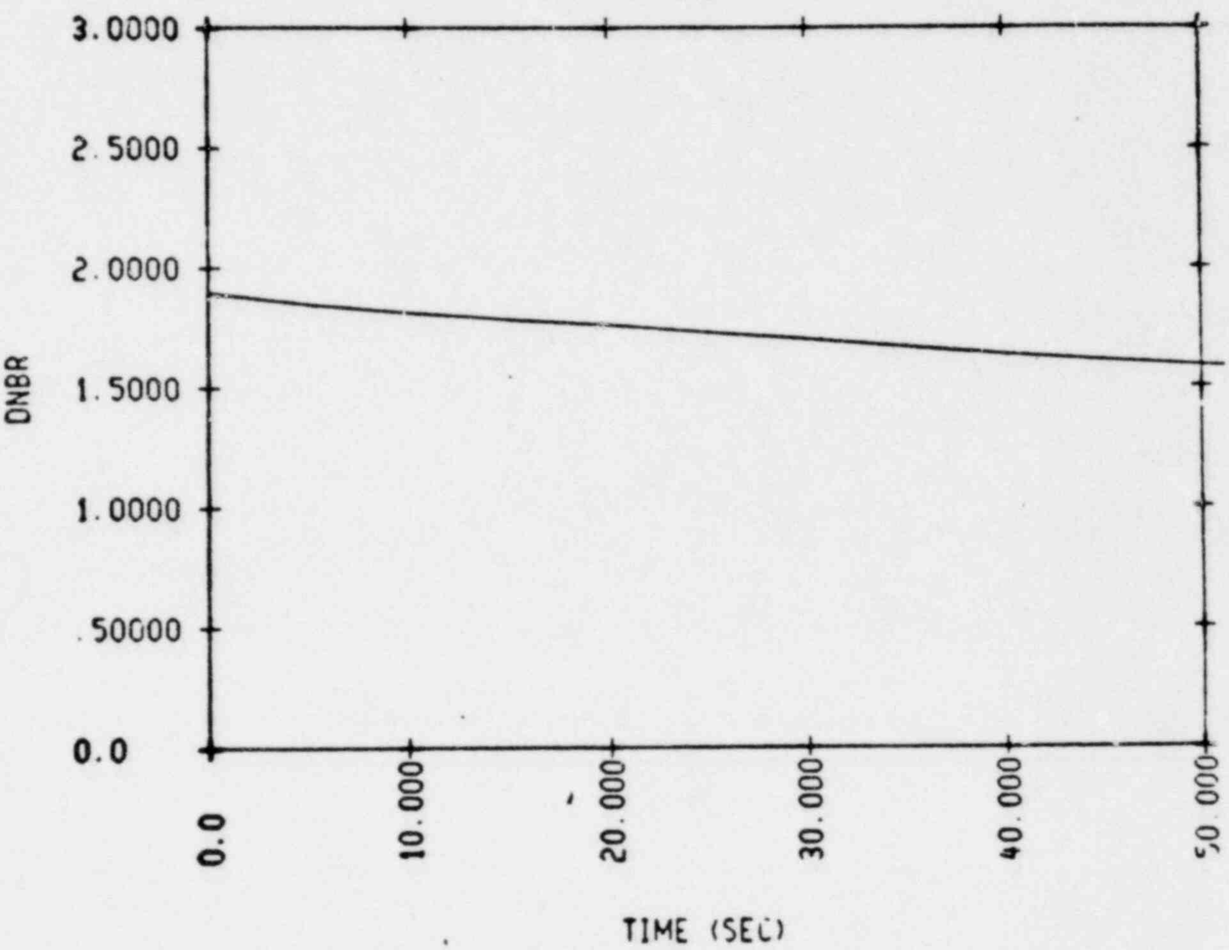


Figure 45. Depressurization - Reference

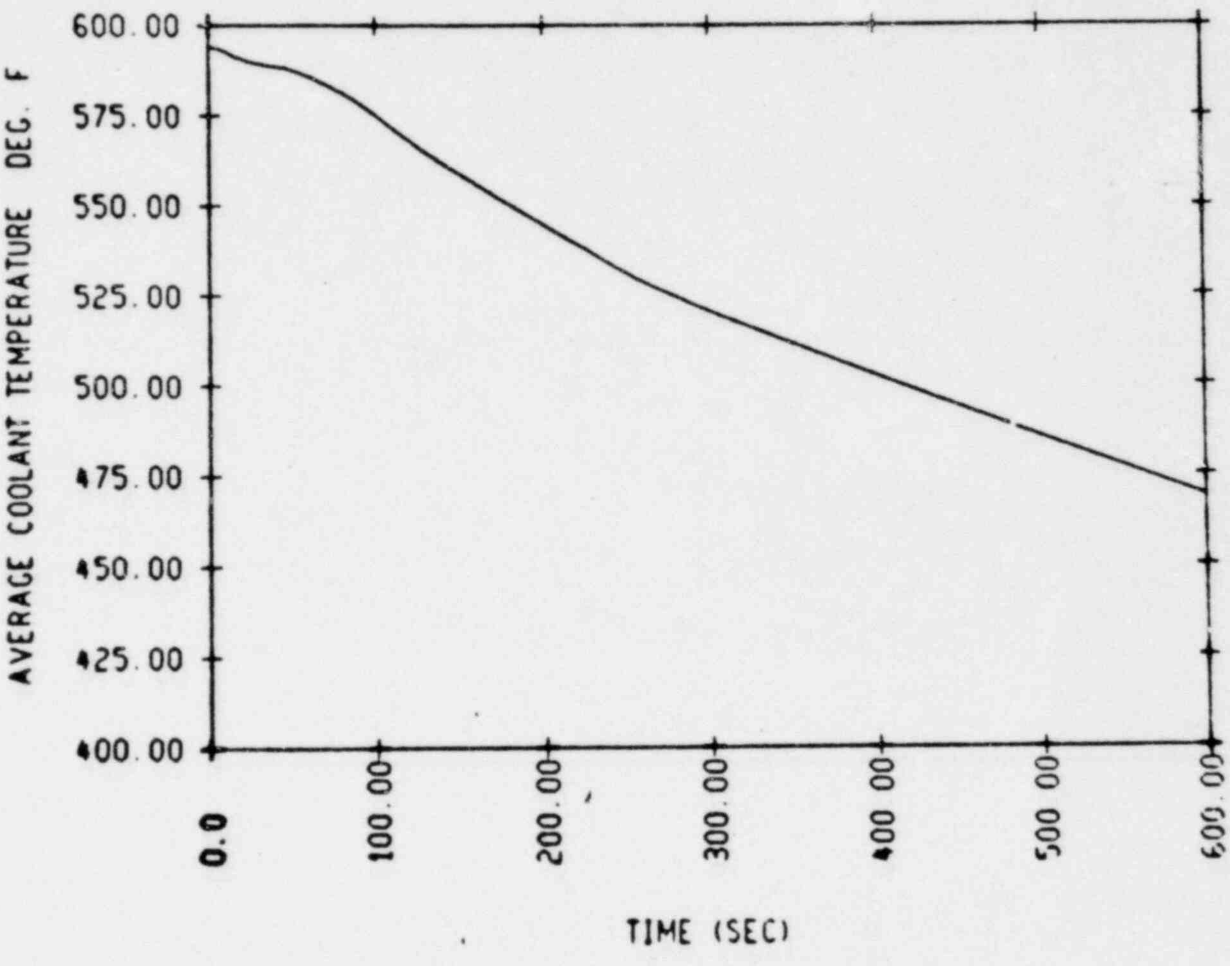


Figure 46. Depressurization - Average Temperature + 8°F

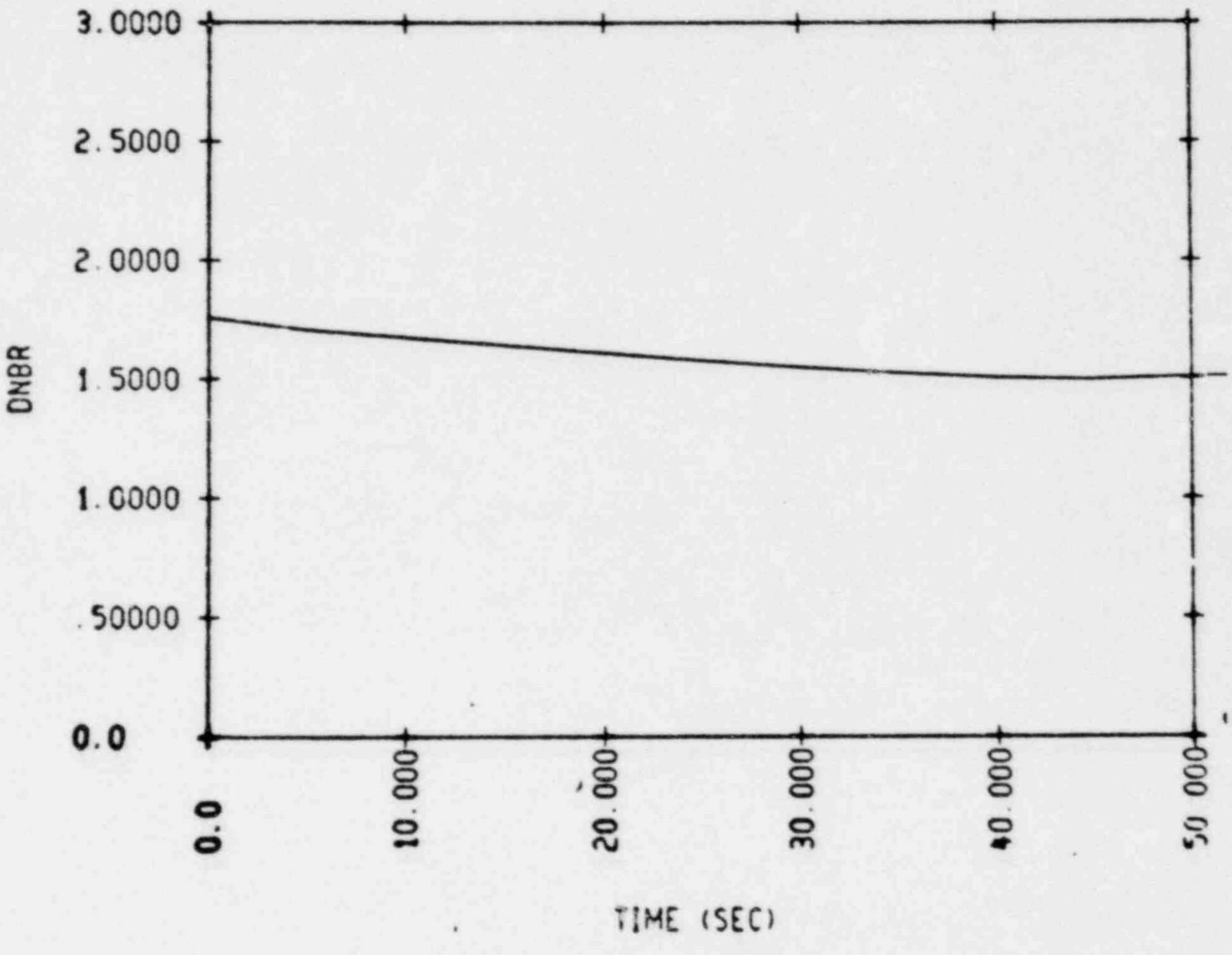


Figure 47. Depressurization - Average Temperature + 8°F

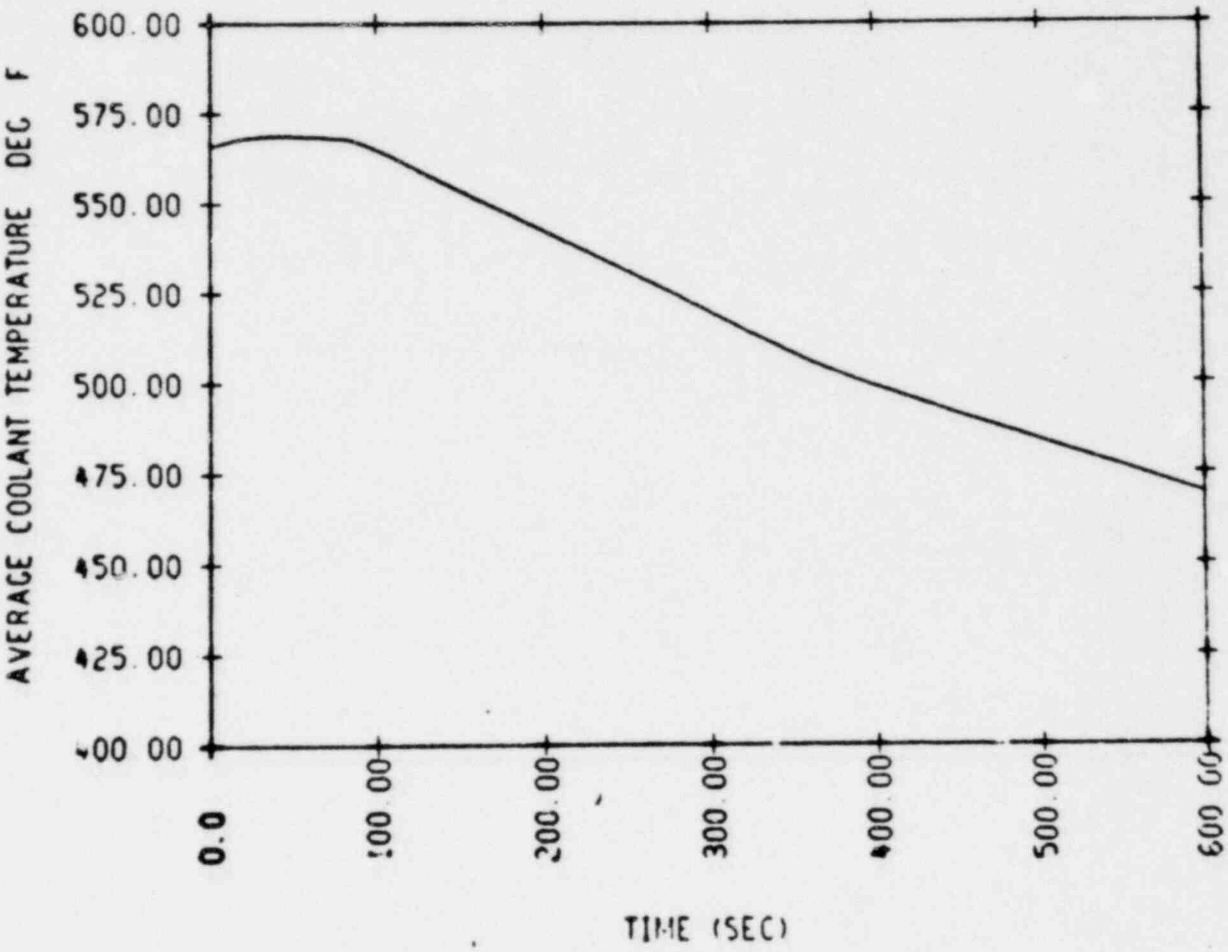


Figure 48. Depressurization - Average Temperature - 20°F

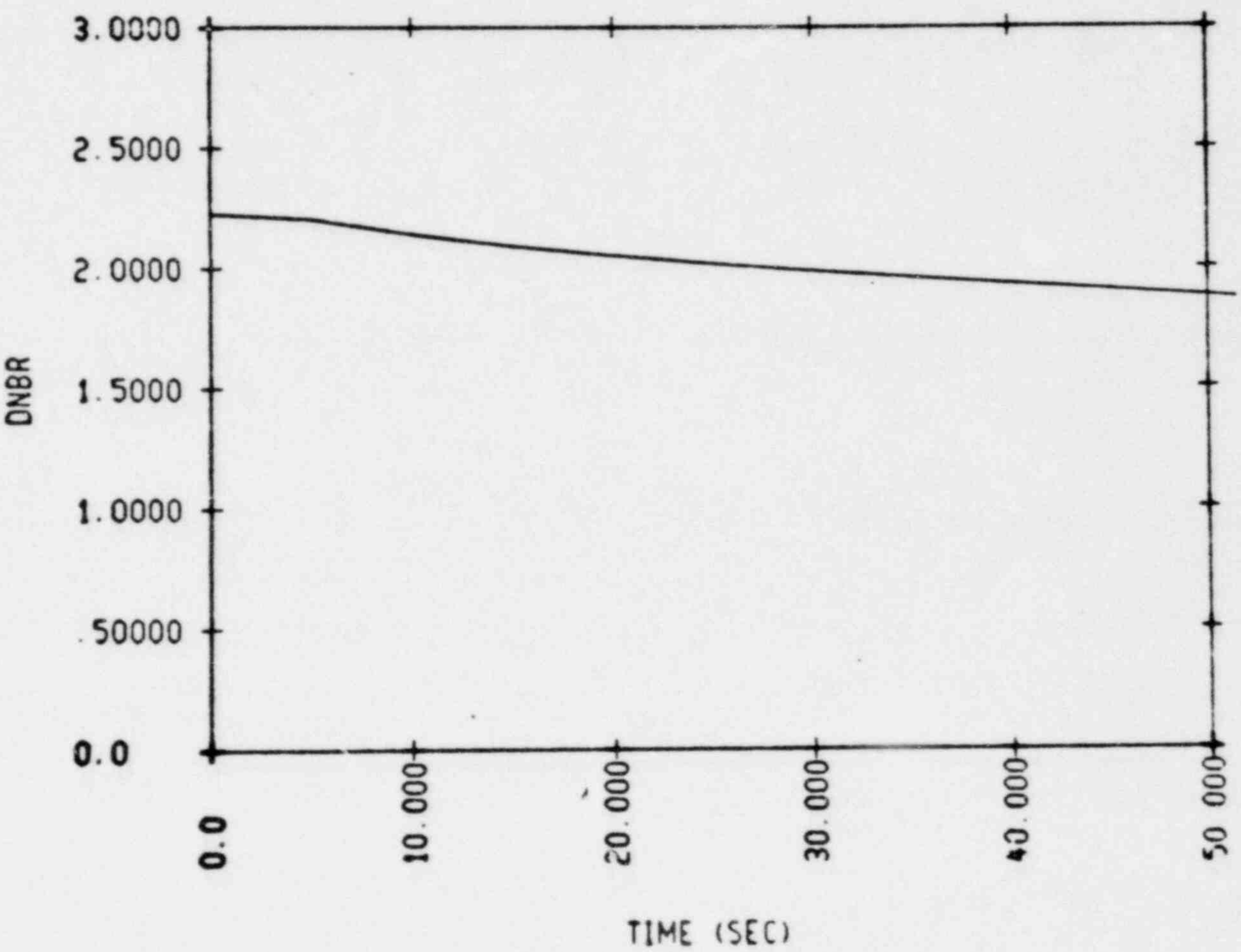


Figure 49. Depressurization - Average Temperature - 20°F

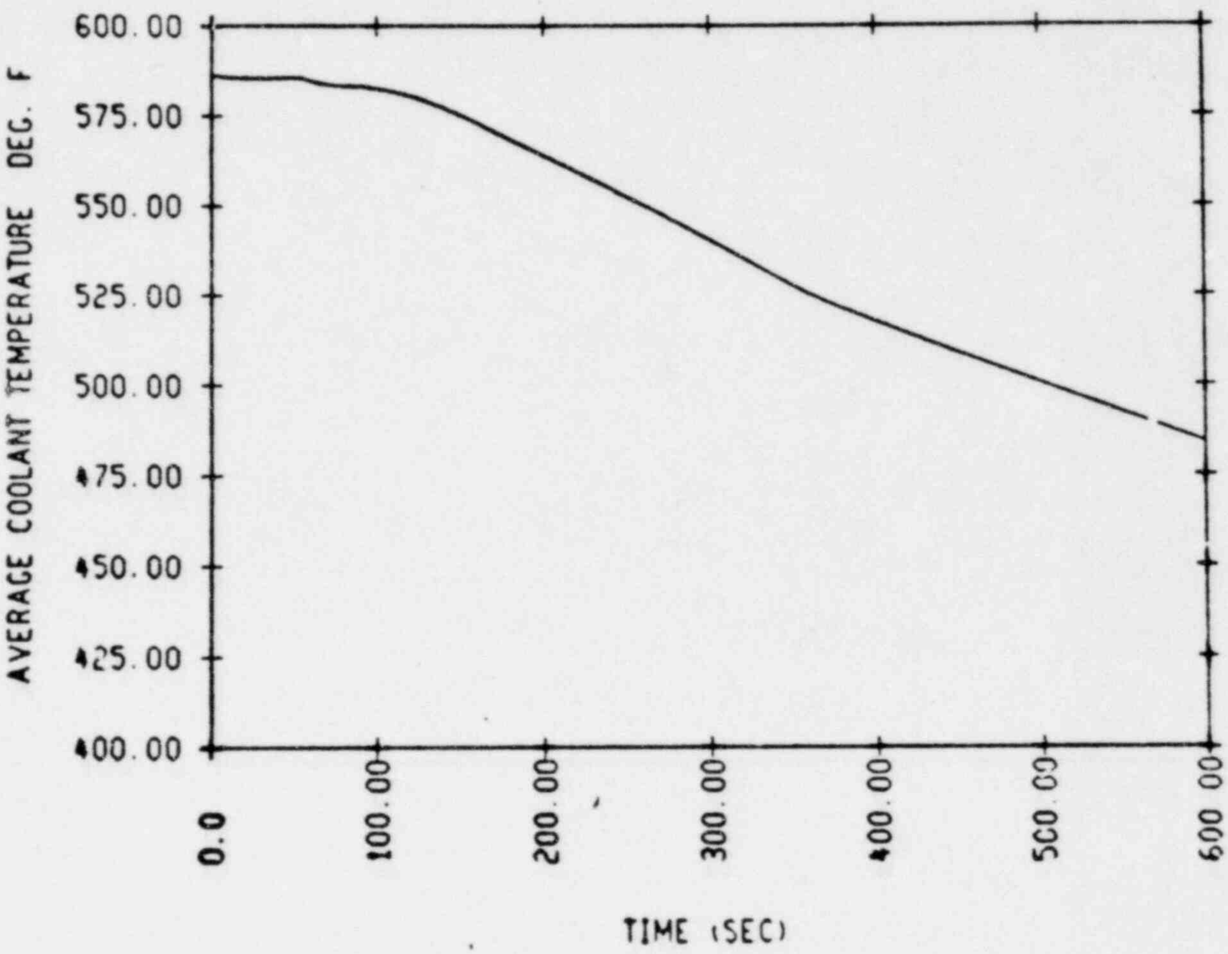


Figure 50. Depressurization - With Rod Control

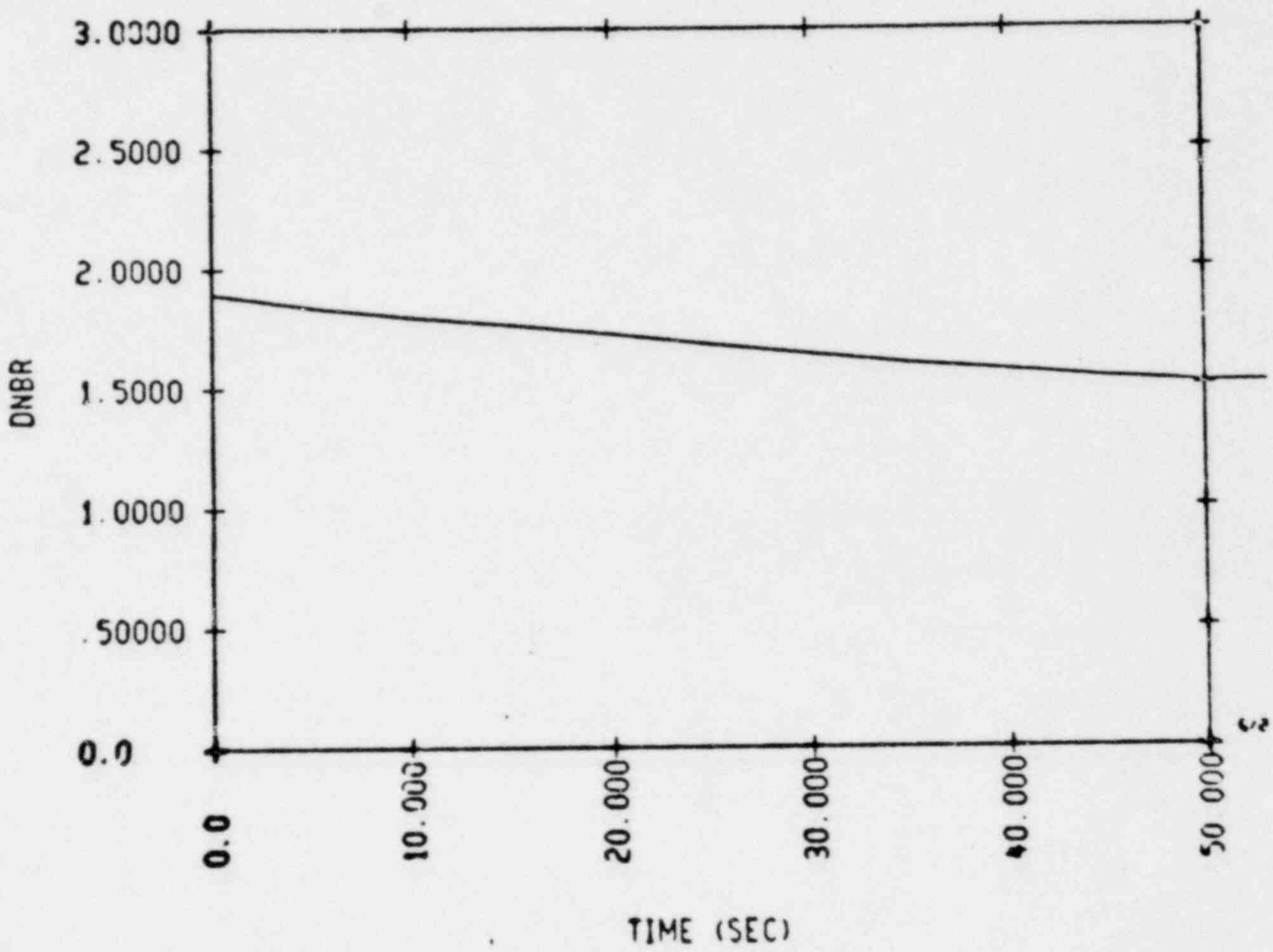


Figure 51. Depressurization - With Rod Control

1.11 ROD DROP

As discussed in WCAP-8330, a reactor trip is not needed for this transient.

2.0 Long Term Shutdown

A review of all the ATWT transients show that after ten minutes the plant is in a near equilibrium condition, i.e. system parameters are not varying greatly. The operator is assumed to take action at this time to bring the plant to a shutdown condition. For some of the ATWT events, the turbine may not trip, e.g. rod withdrawal and accidental depressurization. For these cases the operator would trip the turbine. Following the turbine trip, the operator would take action to add negative reactivity.

The operator may add negative reactivity in several ways, he can manually scram the control rods, or he may use the boration systems. The safety injection system can provide borated water from the refueling water storage tank and also insert the highly concentrated boric acid ($\sim$20,000 ppm) through the boron injection tank. The chemical volume control system can also provide boron to the core via the charging pumps. These pumps may be realigned to provide water from the refueling water storage tank with a boron concentration of about 2,000 ppm.

Once the core is subcritical (accomplished by scramming rods or boration systems) the operator can cool the plant down and/or remove decay heat. The decay heat is removed by dumping steam to the condensor or to the atmosphere, if the condensor is not available. If offsite power has been lost, the operator can control the steam dump manually. During this period of time the core is being borated, steam generator water level is being maintained by auxiliary feedwater and pressurizer water level is controlled by the charging and letdown flow.

Cooldown to the cold shutdown condition requires steam dump. Steam dumping continues throughout the cooldown until the plant is cool enough to permit the operation of the residual heat removal system (350^{o}F). The steam dumping, low temperature auxiliary feed, and relief valves pressurizer spray, and the charging of borated make-up water all contribute to the cooldown and depressurization of the reactor coolant

system, over a period of several hours, to the RHR connection conditions (P 400 psia, T 350^{0}F). At that time the RHR isolation valves are opened, steam dumping is terminated, and the cooldown continues via the RHR heat exchanger. The cooldown rate is limited to a rate of 50^{0}F/hr; but 100^{0}F/hr may be permitted if required.

To show the effects of these actions, the rod withdrawal at power and the loss of normal feedwater transients were reanalyzed assuming the operator started safety injection and steam dumping at 10 minutes.

Figures 1 - 4 show the effect of safety injection at 10 minutes. Note that in about 40 minutes for the rod withdrawal case, and 30 minutes for the loss of feedwater case, the temperature and pressure are both low enough for the operator to switch to the residual heat removal system.

The status report requested to see the effects on the shutdown capability if a pressurizer safety valve failed to reset. The loss of feedwater was reanalyzed assuming that a safety valve did not reset once it had opened. These results are shown in Figures 5 - 7. Note the effect of the open safety valve is to reduce the pressure faster thereby reducing the time needed to go to RHR.

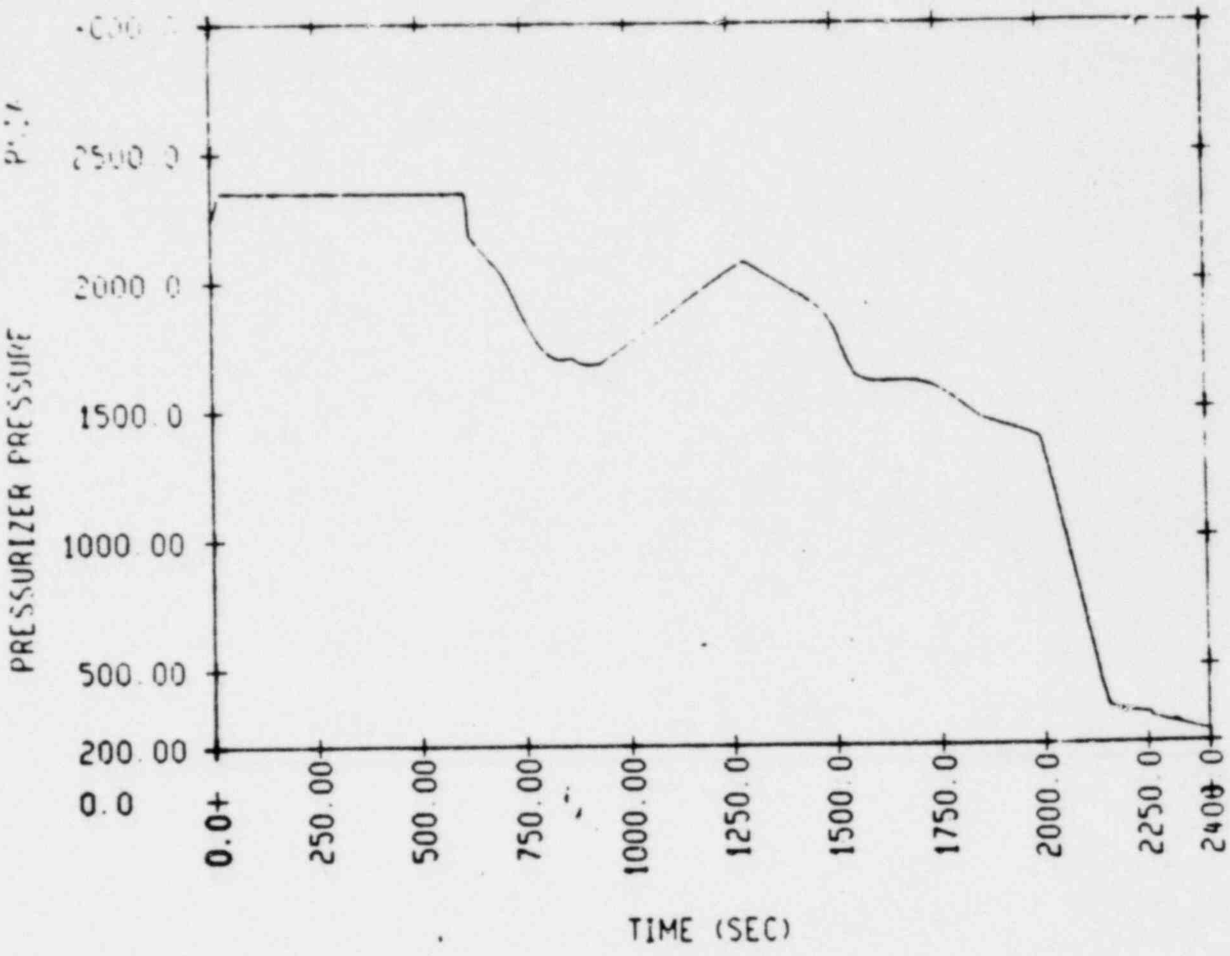


Figure 1 Cooldown Following a Rod Withdrawal at Power ATWS

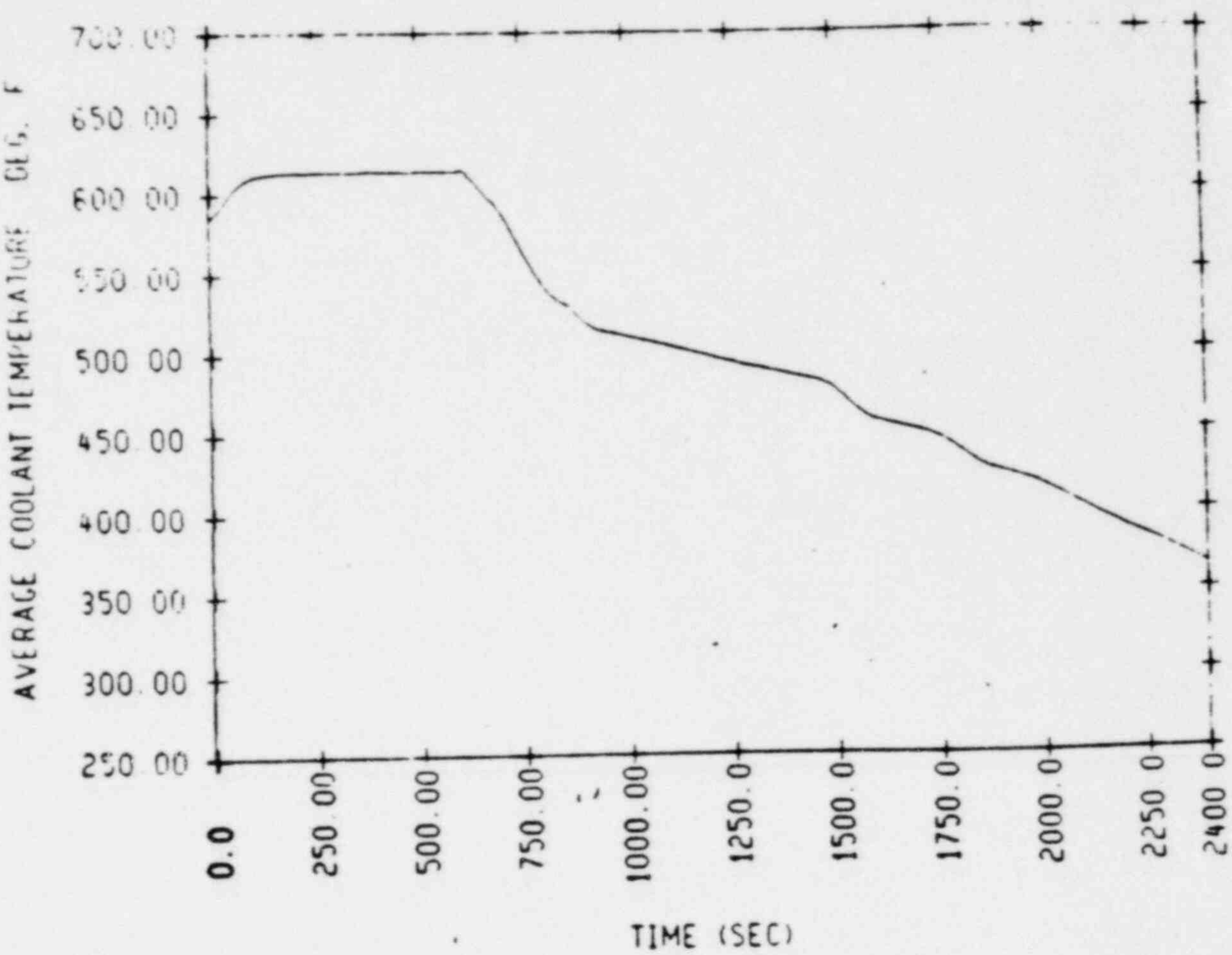


Figure 2 Cooldown Following a Rod Withdrawal at Power ATWS

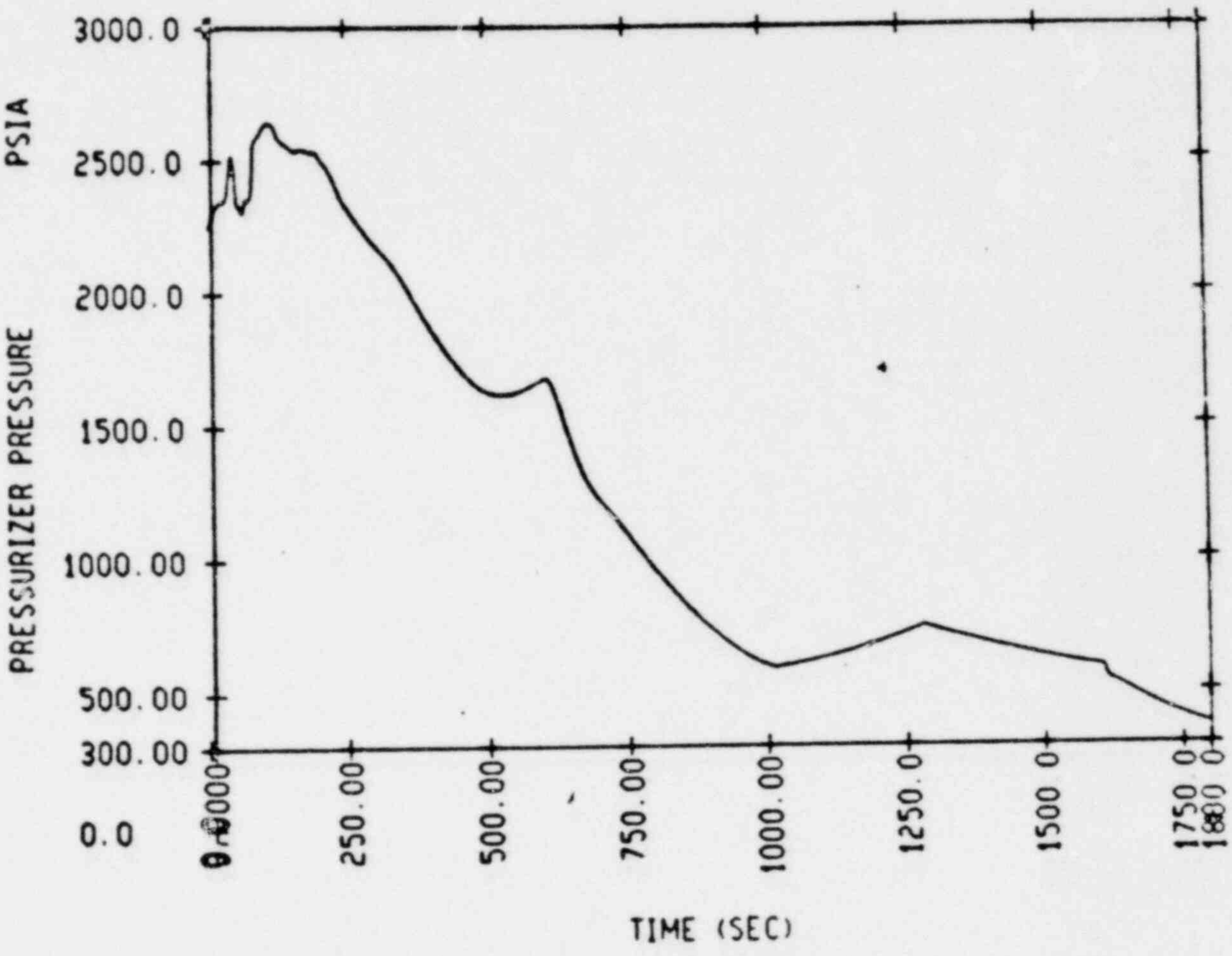


Figure 3 Cooldown Following a Loss of Feedwater ATWS

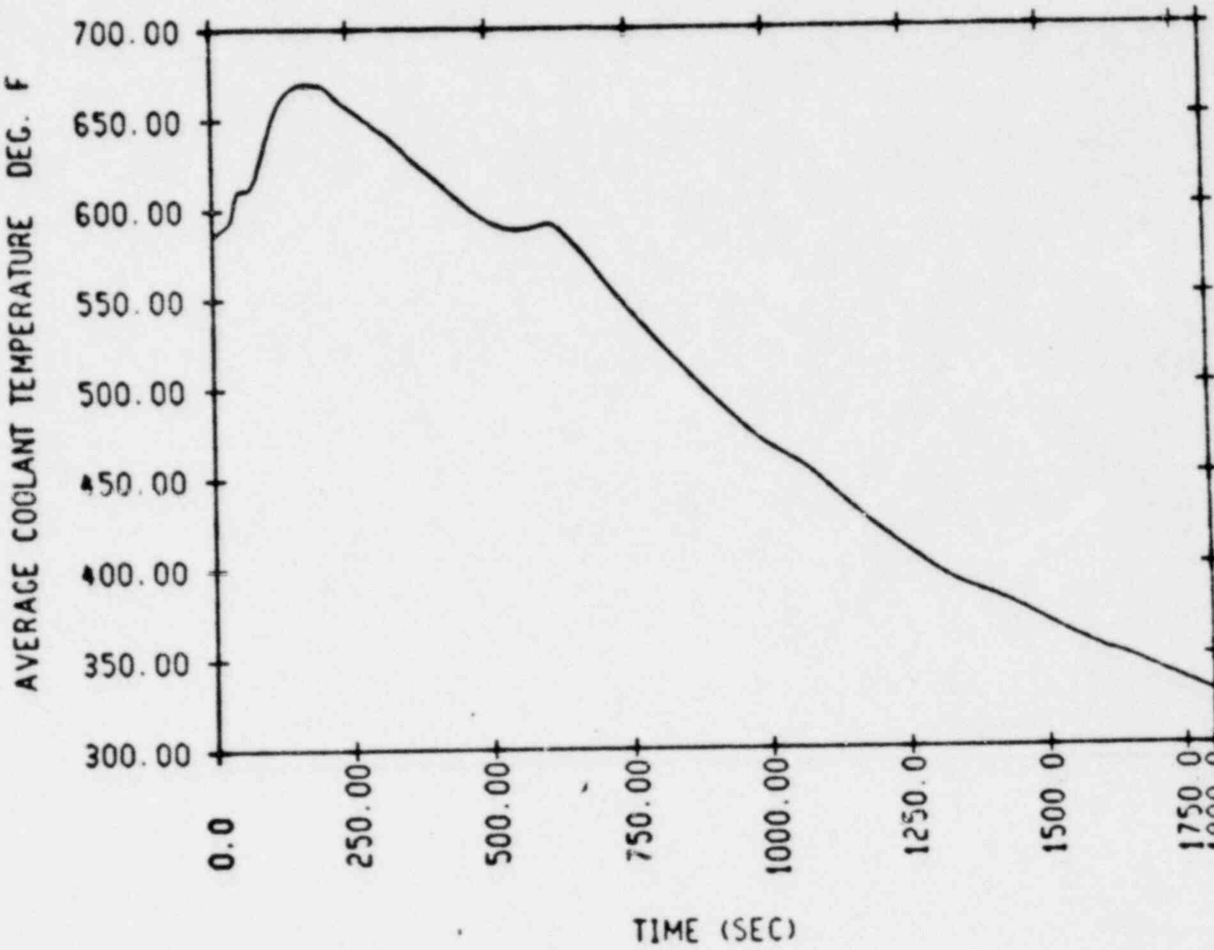


Figure 4 Cooldown Following a Loss of Feedwater ATWS

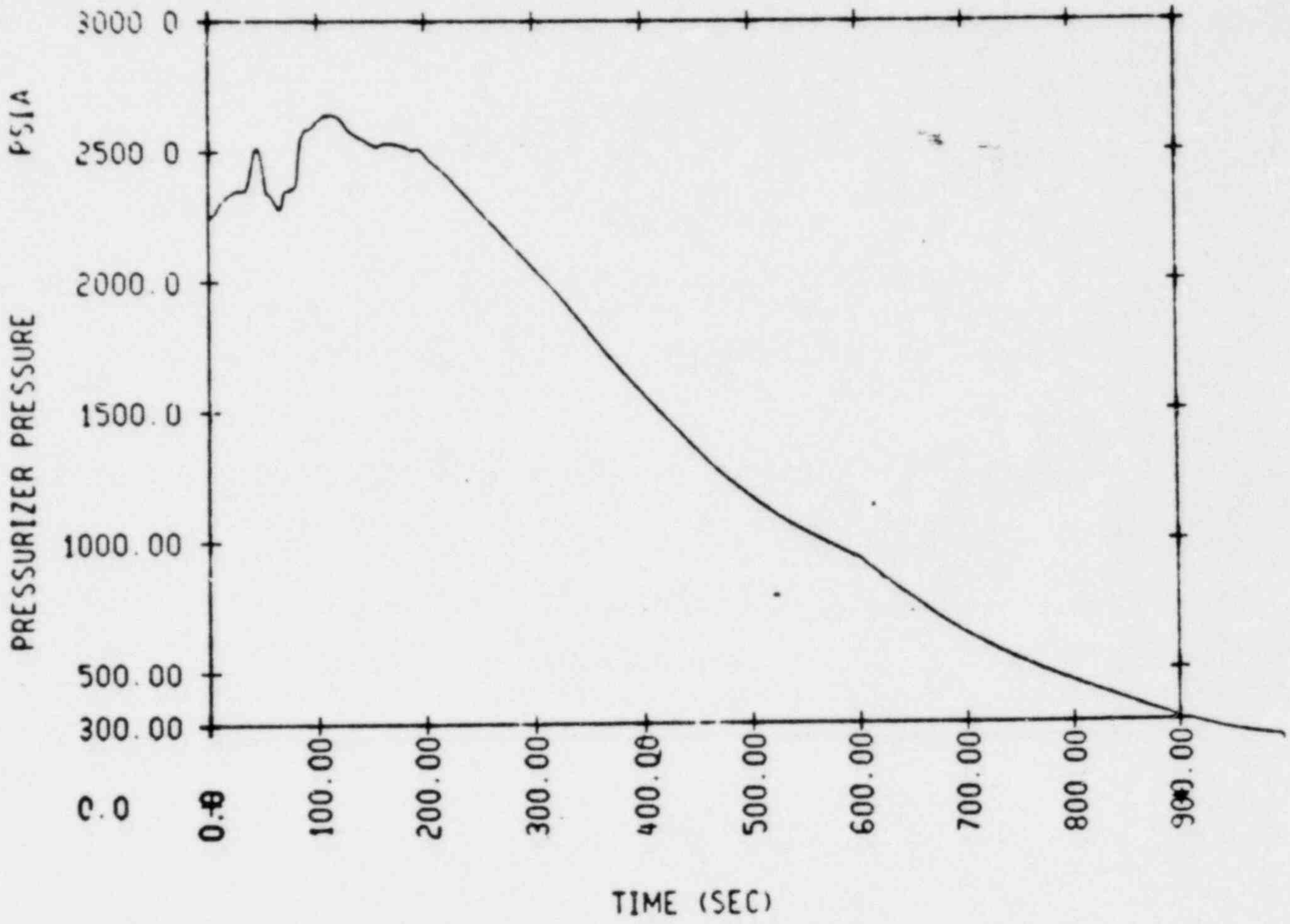


Figure 5 Cooldown Following a Loss of Feedwater ATWS - With a Stuck Open Safety Valve

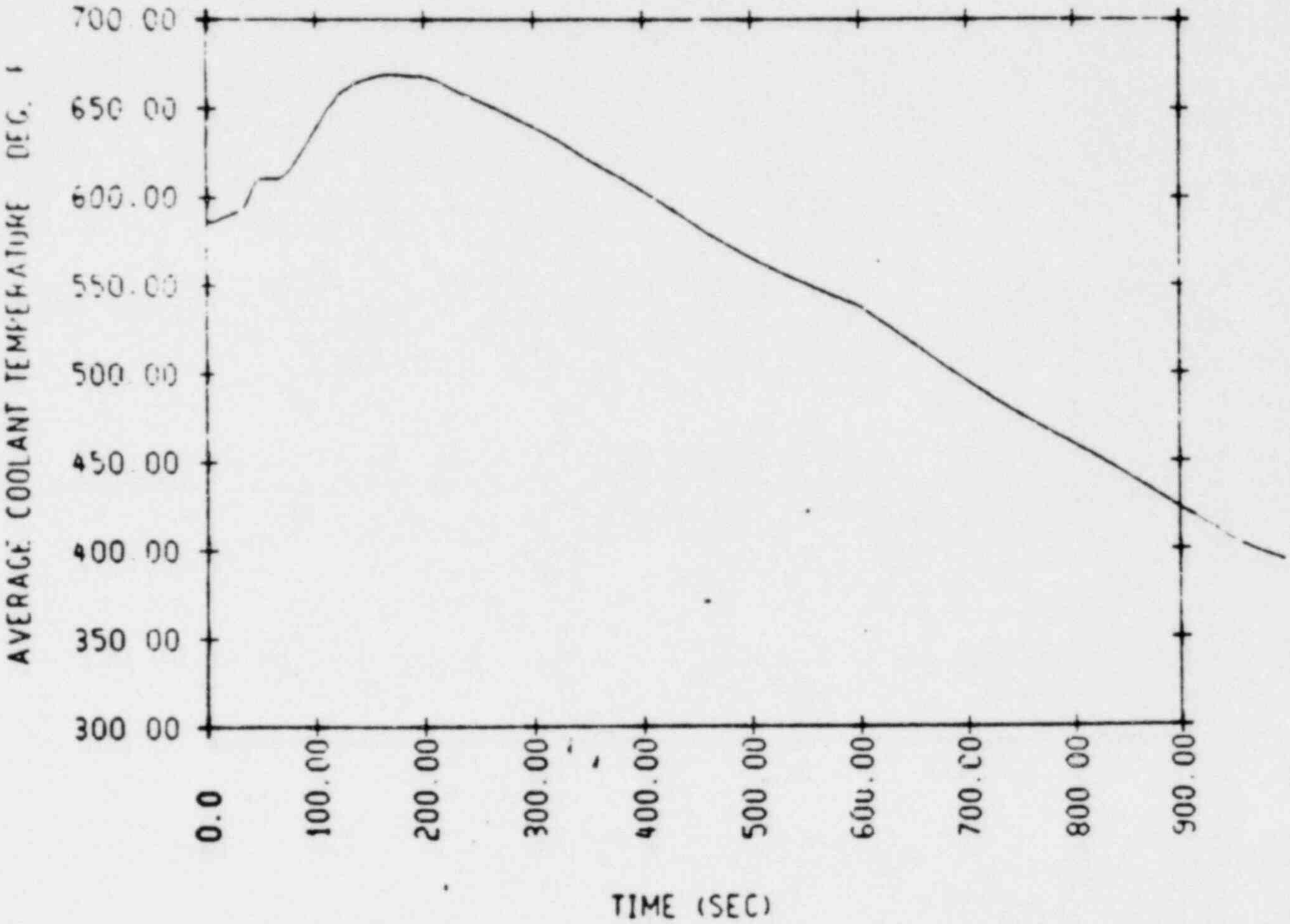


Figure 6 Cooldown Following a Loss of Feedwater
ATWS - With a Stuck Open Safety Valve

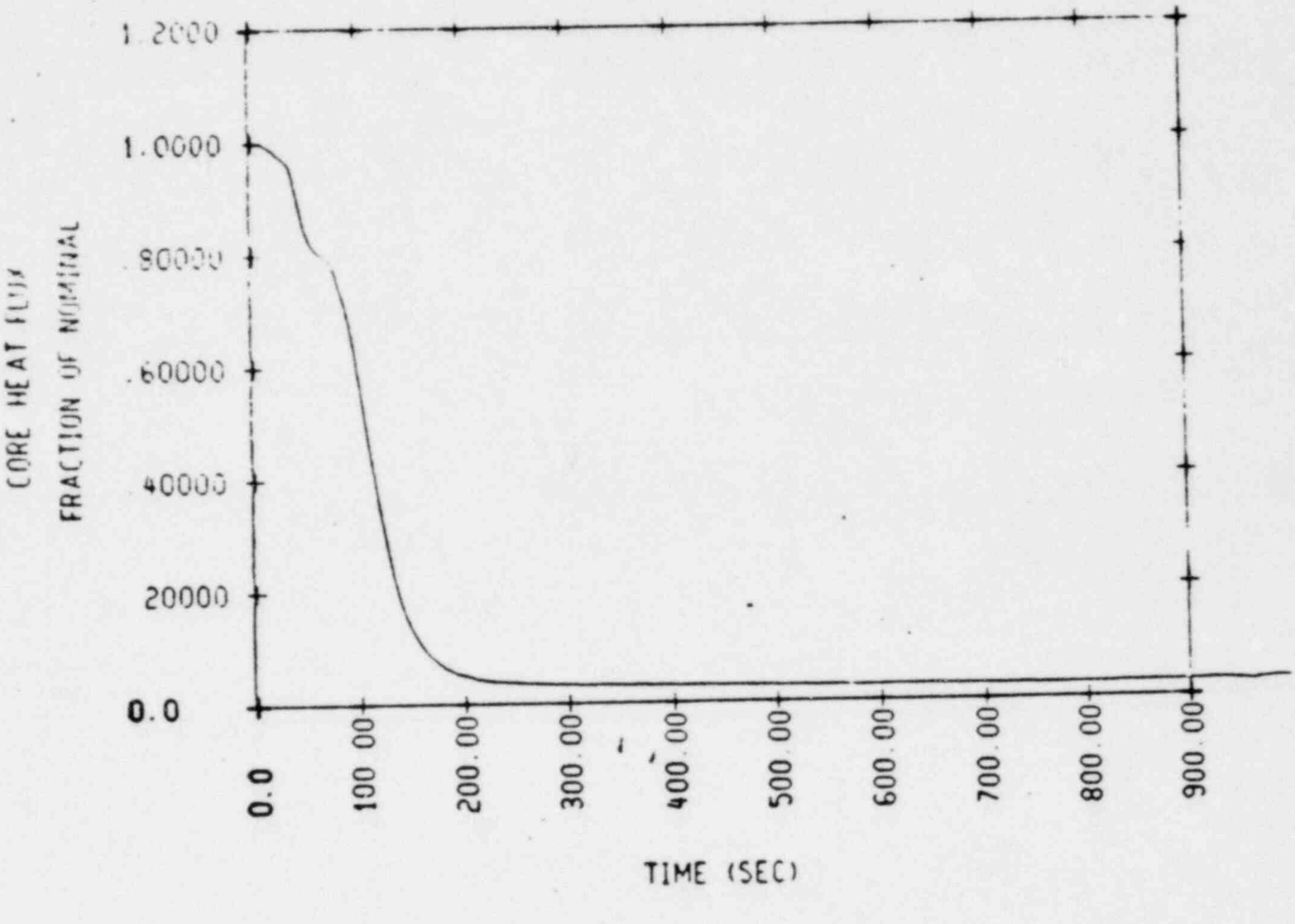


Figure 7 Cooldown Following a Loss of Feedwater ATWS - With a Stuck Open Safety Valve

W ATWS
To: R.E. Heineman
From: Denwood F. Ross, Jr.   Date: August 25. 1976

Comment and Questions on Westinghouse Replies to staff's ATWS Status Report
To: Thomas M. Novak.   From: Zoltan R. Rosztoczy   Aug 30. 76

Westinghouse in the Status Report
To: R. E. Heineman
From: C. Eicheldinger   Date: June 30. 1976