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September 13, 2001
L-01-112

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Subject: Beaver Valley Power Station, Unit No. 2
BV-2 Docket No. 50-412, License No. NPF-73
Response to a Request for Additional Information
In Support of LAR No. 168

This letter provides the FirstEnergy Nuclear Operating Company (FENOC) response to a NRC Request for Additional Information (RAI), dated August 2, 2001, pertaining to FENOC letter L-01-089, dated June 28, 2001. FENOC letter L-01-089 submitted License Amendment Request (LAR) No. 168 that proposed changes to the Beaver Valley Power Station (BVPS), Unit No. 2, to allow operation of the reactor core with a positive moderator temperature coefficient (PMTC) for NRC review and approval. The information provided by this letter consists of the following:

- additional justification that occurrence of the events analyzed in support of the submittal in conjunction with operations using a PMTC will not violate reactor safety limits,

- elaboration on why the events that were not reanalyzed are unaffected by operations using a PMTC,

- further clarification of the changes to the margin to trip analyses due to operations using a PMTC,

- discussion on how BVPS, Unit No. 2, will continue to comply with the Anticipated Transient Without Scram (ATWS) rule, and

- detail on the administrative controls to be put in place in accordance with a new commitment made in support of the LAR.

The FENOC responses to the RAI are provided in Attachment A of this letter.

A001

FENOC requests NRC approval of License Amendment Request No. 168 prior to the first entry into Mode 2 for BVPS, Unit No. 2, operating cycle 10.

This information does not change the evaluations or conclusions presented in FENOC letter L-01-089. If there are any questions concerning this matter, please contact Mr. Thomas S. Cosgrove, Manager Regulatory Affairs, at 724-682-5203.

Sincerely,

Lew W. Myers

Attachment

c: Mr. L. J. Burkhart, Project Manager
Mr. D. M. Kern, Sr. Resident Inspector
Mr. H. J. Miller, NRC Region I Administrator
Mr. D. A. Allard, Director BRP/DEP
Mr. L. E. Ryan (BRP/DEP)

Subject: **Beaver Valley Power Station, Unit No. 2**
BV-2 Docket No. 50-412, License No. NPF-73
Response to a Request for Additional Information
In Support of LAR No. 168

I, Lew W. Myers, being duly sworn, state that I am Senior Vice President of FirstEnergy Nuclear Operating Company (FENOC), that I am authorized to sign and file this submittal with the Nuclear Regulatory Commission on behalf of FENOC, and that the statements made and the matters set forth herein pertaining to FENOC are true and correct to the best of my knowledge and belief.

FirstEnergy Nuclear Operating Company

Lew W Myers

Lew W. Myers
Senior Vice President - FENOC

COMMONWEALTH OF PENNSYLVANIA

COUNTY OF BEAVER

Subscribed and sworn to me, a Notary Public, in and for the County and State above named, this 13 th day of September, 2001.

Tracey A Baczek

My Commission Expires:

Notarial Seal
Tracey A. Baczek, Notary Public
Shippingport Boro, Beaver County
My Commission Expires Aug. 16, 2005

Member, Pennsylvania Association of Notaries

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI)
POSITIVE MODERATOR TEMPERATURE COEFFICIENT (PMTC)
FOR BEAVER VALLEY POWER STATION UNIT NO. 2 (BVPS-2)
DATED JUNE 28, 2001
(LICENSE AMENDMENT REQUEST NO. 168)

NRC RAI Question 1

On page B-4, paragraph 2 lists Facility [sic] Safety Analysis Report (FSAR) analyses performed in support of the proposed PMTC. Please provide detailed quantitative and qualitative technical justification that the occurrence of these events in conjunction with operation of the plant with a PMTC will not cause reactor safety limits to be violated. Provide any technical evaluation reports by Westinghouse which demonstrate the ability to safely operate with a PMTC.

FirstEnergy Nuclear Operating Company (FENOC) Response

A +2 pcm/°F moderator temperature coefficient (MTC) was considered for all of the analyses as part of the recently completed revised thermal design procedure (RTDP) and 1.4% power uprating programs. The results of those analyses are summarized in the Tables 1 through 5. In all cases, all applicable acceptance criteria are met.

NRC RAI Question 2

On page B-5, paragraph 2 states that analyses other than those listed on page B-4 were not evaluated for the proposed PMTC. Please provide supportive analyses to demonstrate that all events not evaluated will be unaffected by operation of the reactor with a positive MTC.

FENOC Response

The events not listed on Page B-4 are analyzed with reactor core end-of-life (EOL) moderator density coefficients. For example, the steamline break transient is analyzed with a most positive EOL density coefficient to maximize the reactivity feedback due to the cooldown. Analysis of these events with a PMTC would yield less severe analysis results than those completed for the RTDP and 1.4% power uprating programs.

NRC RAI Question 3

On page B-5 a discussion of "Control Systems Margin to Trip Evaluation" is provided. Please provide the analyses performed to demonstrate the effects of a positive MTC on each of the events evaluated.

FENOC Response

The large-load rejection (50%) transient results confirmed that there was sufficient margin to the over-temperature delta temperature (OTΔT) trip setpoint. The analysis indicated acceptable results: a minimum margin of more than 22% to the OTΔT trip setpoint, and a peak pressurizer pressure of 2341 pounds per square inch absolute (psia), which remains below the pressurizer power operated relief valve (PORV) actuation setpoint. In addition, no other reactor trip or engineered safety features actuation system (ESFAS) setpoints were challenged. Therefore, the 50% load rejection can be accommodated with a PMTC without challenging any of the reactor trip setpoints.

The licensing basis for BVPS, Unit No. 2, is being revised to reflect that the full-load rejection is no longer considered.

The 10% step load increase from 90% nominal power results determined that the minimum steam line pressure was 610 pounds per square inch gauge (psig). This results in a margin of more than 110 psi to the low steam line pressure setpoint of 500 psig. Therefore, there is acceptable margin to the low steam line pressure ESFAS actuation setpoint with a PMTC.

The turbine trip without reactor trip from the P-9 setpoint transient results confirmed that it would not challenge the PORVs, even when assuming the second set of steam dump valves were unavailable. The peak pressurizer pressure reached 2311 psia, which is below the pressurizer PORV actuation setpoint, with manual rod control as the limiting case.

NRC RAI Question 4

On page B-6 a discussion on how BVPS-2 meets the Anticipated Transient Without Scram (ATWS) rule with a positive MTC is presented. Please provide quantitative and qualitative technical justification supporting the ability for BVPS-2 to comply with the ATWS rule when operating with a positive MTC. Include in this justification, information regarding the following factors for a PMTC: 1) Unfavorable exposure time and 2) ATWS core damage frequency. Additionally, please provide a copy of References 1 and 3 from your License Amendment Request (LAR) No. 168, dated June 28, 2001.

FENOC Response

As discussed on pages B-5 and B-6, the limiting concern for ATWS events is the potential for RCS overpressurization following a loss of normal feedwater (LONF) or loss of load (LOL) ATWS event. For reactor coolant system (RCS) overpressurization, the most limiting condition is the occurrence of these events from hot full power (HFP) initial conditions. The quantitative technical justification for this is provided in NS-TMA-2182 (LAR No. 168, Reference 1, a copy of which is provided as Attachment 1). The ATWS analyses in NS-TMA-2182 are the analytical basis for the Final ATWS Rule for Westinghouse pressurized water reactors (PWRs) as documented in SECY-83-293 (Reference 2). As documented and quantified in LAR No. 168, Reference 3 (a copy is provided as Attachment 2), an MTC less than or equal to -5.5 pcm/°F at HFP precludes the pressure limiting ATWS events from reaching an RCS pressure in excess of 3200 psig. By designing the core to maintain an MTC at HFP to a value less than or equal to -5.5 pcm/°F at all time during core life, RCS overpressurization (i.e., exceeding 3200 psig) is precluded. With RCS pressure maintained below 3200 psig, there is no unfavorable exposure time, and therefore, no subsequent ATWS related core damage.

NRC RAI Question 5

In your commitment list it is stated that "administrative controls" will be put in place to ensure the MTC at hot full power conditions will be less than or equal to -5.5 pcm/°F at all times during core life. Please provide a detailed description of the "administrative controls" to be implemented to verify this commitment is being met.

FENOC Response

The NRC approved WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," on May 28, 1985. This methodology is a systematic evaluation used to determine whether the reload parameters for each fuel cycle are bounded by the values contained in the reference safety analysis. For each reload cycle, the values of the key safety parameters are determined for the reload core during the nuclear, thermal and hydraulic, and fuel rod design processes. The MTC limit at HFP conditions of -5.5 pcm/°F at all times during core life will be added to the nuclear design process of the core reload to ensure that it is considered in the initial loading pattern development during the preliminary design phase of the core reload.

Summary of the Unit 2 Non-LOCA Analysis Results

Table 1

Event Name	**UFSAR Section**	**Minimum DNBR**	**Peak Primary Pressure**	**Peak Secondary Pressure**
Rod Withdrawal at Power	15.4.2	1.362	N/A (1)	1171 psia
Partial Loss of Flow (2)	15.3.1	1.790	2327.8 psia	920.6 psia
Loss of Load	15.2.2/15.2.3	1.67	2747.5 psia	1182.5 psia
Rod With. from Subcritical	15.4.1	Limit met (3)	N/A	N/A
RCS Depressurization	15.6.1	1.76	N/A	N/A
Complete Loss of Flow (2)	15.3.2	1.335	2414.2 psia	951.0 psia
Limits	---	1.33	2748.5 psia	1208.5 psia

(1) A generic Westinghouse evaluation addresses the peak pressures for the Rod Withdrawal at Power analyses.
(2) The analysis at full power with a zero moderator temperature coefficient (MTC) bounds the analysis at part power with a positive moderator temperature coefficient (PMTC).
(3) A minimum departure from nucleate boiling (DNB) ratio (DNBR) is not available. Transient statepoints are evaluated to determine whether or not the limit is met. This is repeated as part of each subsequent reload evaluation.

psia = pounds-per-square-inch absolute
RCS = reactor coolant system
UFSAR = Updated Final Safety Analysis Report

Table 2

Event Name	**UFSAR Section**	**Percentage of rods in DNB**	**Peak Primary Pressure**
Locked Rotor (2)	15.3.3	< 18%	2759.3 psia
Limits	---	18%	2997 psia (1)

(1) The peak Reactor Coolant System pressure reached during the transient is less than that which would cause the stresses to exceed the faulted condition stress limits.
(2) The analysis at full power with a zero MTC bounds the analysis at part power with a PMTC.

Summary of the Unit 2 Non-LOCA Analysis Results (continued)

Table 3

Event Name	**UFSAR Section**	**Peak Pressurizer Volume (ft^3)**
Loss of Normal Feed [(1)]	15.2.7	1454.
Loss of AC Power [(1)]	15.2.6	1009.
Limits	---	1457.9

[(1)] The analysis at full power with a zero MTC bounds the analysis at part power with a PMTC.

AC = alternating current

Table 4

Event Name	**UFSAR Section**	**Margin to Hot Leg Boiling (°F)**
Feedline Rupture	15.2.8	31.0
Limits	---	0.0

Table 5

Rod Ejection Case	**UFSAR Section**	**Maximum Fuel Stored Energy**
BOL-HZP	15.4.8	184.5
BOL-HFP	---	323.6
EOL-HZP	---	306.8
EOL-HFP	---	307.8
Limits	---	360 Btu/lb.

BOL = Beginning of core life
EOL = End of core life
HFP = Hot full power
HZP = Hot zero power

ATTACHMENT 1

Beaver Valley Power Station, Unit No. 2
Response to Request for Additional Information
License Amendment Request No. 168

LAR No. 168, Reference 1: NS-TMA-2182

Westinghouse Electric Corporation Power Systems

Box 355
Pittsburgh Pennsylvania 15230

December 30, 1979
NS-TMA-2182

Dr. Stephen H. Hanauer
Assistant Director for Plant Systems
Division of System Safety
U.S. Nuclear Regulatory Commission
1717 H Street NW
Mail Stop P-822
Washington, D.C. 20555

SUBJECT: ATWS SUBMITTAL

Dear Dr. Hanauer:

In continuing response to the staff's request dated February 15, 1979 enclosed please find twenty (20) copies of the following revised sections to our June 8, 1979 submittal on ATWS:

1. Introduction
2. ATWS Criteria
3. Plant Parameter Bases
4. Computer Models
5. Transient Analysis and Sensitivity Studies
6. Stress Limits
9. ATWS Mitigating Systems
11. Summary and Conclusions

Very truly yours,
WESTINGHOUSE ELECTRIC CORPORATION

T. M. Anderson, Manager
Nuclear Safety Department

PMS:kk
Enclosures

Accordingly, this report will provide the requested ATWS information as it applies to the group of Westinghouse plants which would be subject to the proposed Alternative 3 requirements. Transient analyses and sensitivity studies will be presented for the following ATWS events:

1. Loss of load
2. Loss of normal feedwater
3. Loss of offsite power
4. Accidental RCS depressurization
5. Rod withdrawal

Stress limits and radiological consequences will be addressed and a preliminary description of the ATWS hardware modifications in Westinghouse plants will be provided.

ANTICIPATED TRANSIENTS WITHOUT SCRAM
FOR WESTINGHOUSE PLANTS
DECEMBER, 1979

TABLE OF CONTENTS

1.0 INTRODUCTION

This report has been written in response to a request for information on Anticipated Transients Without Scram (ATWS) which was made in a letter from R. Mattson (Nuclear Regulatory Commission) to T. M. Anderson (Westinghouse Nuclear Technology Division), dated February 15, 1979. This request was made following the publication of NUREG-0460, Volume 3, and prior to the issuance of a proposed ATWS rule by the NRC.

Volume 3 of NUREG-0460 describes four proposed alternative solutions to the ATWS licensing issue for each of the four Nuclear Steam Supply System (NSSS) vendors. These solutions range from doing nothing (Alternative 1) to the installation of systems and equipment to mitigate the consequences of an ATWS (Alternative 4) with no preventative measures taken. Each solution specifies analytical, as well as hardware requirements. These will be addressed in the following section of this report.

The purpose of this report is to show that Westinghouse plants satisfactorily meet the proposed Alternative 3 and 4 ATWS criteria (Alternative 3 as defined in Volume 3 of NUREG 0460 is intended to apply to all plants with contruction permits dated before January, 1978, and Alternative 4 to all plants with construction permits after January, 1978) and to provide the NRC with a technical basis for its "early verification" approach.

Early verification, as defined in Volume 3 of NUREG-0460, is the documented assurance that a plant or group of plants complies with the applicable ATWS proposed requirements prior to, or during the course of, rule-making. The specific plant modifications necessary to meet the ATWS requirements, as determined during early verification, would then form the basis for a proposed ATWS rule. This process would permit the treatment of ATWS on a generic basis (by groups of similar plant designs).

2.0 ATWS CRITERIA

In Volume 3 of NUREG-0460, the NRC staff describes four alternative plant modifications that are proposed to be applied to each of the NSSS vendors. These alternatives represent the range of possibilities for resolving the ATWS concern. Each alternative is associated with some level of safety in the NRC staff's engineering judgment. This section will present these alternatives as they affect Westinghouse plants. (see Table 2-1).

Alternative 1 - No plant modifications

This alternative would require no modifications of any kind. Selection of this alternative acknowledges the industry position that ATWS is not a safety problem, and therefore no corrective action is required.

Alternative 2 - Modification to reduce susceptibility to common mode failures

Westinghouse is to "confirm the adequacy of actuation circuitry for ATWS mitigating equipment in the balance-of-plant designs", since "this circuitry already exists in a significant number of Westinghouse plants". This alternative has been proposed for early operating plants (pre-Dresden).

Alternative 3 - Modifications to reduce susceptibility to common mode electrical failures and to provide mitigation of most ATWS events

It is proposed that Westinghouse provide:

1. Confirmation or provision of diverse actuation circuitry for mitigating systems as described in Alternative 2.

2. Demonstration of the integrity of the primary coolant system boundary and functionability of valves needed for long-term cooling following conditions calculated for specified ATWS events. This

This alternative has been proposed in NUREG-0460, Volume 3, for all new plants with construction permits dated after January, 1978.

The hardware modifications needed to meet the proposed requirements for Westinghouse plants are identical for Alternatives 2, 3 and 4 and consist of the confirmation or provision of diverse actuation circuitry in the balance-of-plant to actuate auxiliary feedwater and trip the turbine. This has been designated by the NRC as AMSAC, ATWS Mitigating System Actuation Circuitry. (See Table 2-2.)

demonstration would include the postulated initiating events identified in Appendix IV of NUREG-0460, Section IV.2 and prescription five in Table 7 of Appendix VII of NUREG-0460 (95% MTC*, all other parameters at their nominal values, and no additional failures other than the scram system). This demonstration has been essentially completed for Westinghouse plants in the course of earlier generic ATWS reviews by the staff.

This alternative has been proposed in NUREG-0460, Volume 3, for all plants with construction permits dated before January, 1978. The purpose of this report is to re-state the adequacy of Westinghouse design with respect to Alternative 3.

Alternative 4 - Modifications to provide mitigation of ATWS events

It is proposed that Westinghouse provide:

1. Confirmation or provision of diverse actuation circuitry for mitigating systems, as described in Alternative 2.

2. Demonstration of the functionability of valves needed for long-term cooling following conditions calculated for specified ATWS events. All components must meet level C stress limits as defined by the ASME code. This demonstration would include the postulated initiating events described in Appendix IV of NUREG-0460, Section IV.2, and prescription four in Table 7 of Appendix VII of NUREG-0460 (99% MTC, all other parameters at their nominal values, with a single equipment failure in addition to the scram system).

* See Section 3.2.4 for discussion of the moderator temperature coefficient (MTC).

TABLE 2-1

WESTINGHOUSE PLANTS GROUPED BY THE ALTERNATIVES
OF NUREG-0460, VOLUME 3

ALTERNATIVE 2

Yankee Rowe Unit 1, Yankee Atomic Power Company
Haddam Neck Unit 1, Connecticut Yankee
San Onofre Unit 1, Southern California Edison

ALTERNATIVE 3

Zion Units 1 and 2, Comonwealth Edison
D. C. Cook Units 1 and 2, Indiana/Michigan Power
Diablo Canyon Units 1 and 2, Pacific Gas and Electric
Trojan Unit 1, Portland Gas and Electric
Sequoyah Units 1 and 2, Tennessee Valley Authority
Salem Units 1 and 2, Public Service Electric & Gas (N.J.)
Wm. B. McGuire Units 1 and 2, Duke Power Company
Catawba Units 1 and 2, Duke Power Company
Byron Units 1 and 2, Commonwealth Edison
Watts Bar Units 1 and 2, Tennessee Valley Authority
Comanche Peak Units 1 and 2, Texas Utilities
Braidwood Units 1 and 2, Commonwealth Edison
Surry Units 1 and 2, VEPCO
North Anna Units 1 and 2, VEPCO
Beaver Valley Units 1 and 2, Duquesne Light
J. M. Farley Units 1 and 2, Alabama Power Company
V. C. Summer Unit 1, South Carolina Electric & Gas
Prairie Island Units 1 and 2, Northern States
Kewaunee Unit 1, Wisconsin PSC
South Texas Units 1 and 2, Houston Lighting
A. W. Vogtle Units 1 and 2, Georgia Power Company

TABLE 2-1 (Continued)

WESTINGHOUSE PLANTS GROUPED BY THE ALTERNATIVES OF NUREG-0460, VOLUME 3

ALTERNATIVE 3 (Continued)

Millstone Unit 3, Northeast Nuclear
Seabrook Units 1 and 2, Public Service Company of N.H.
Wolf Creek Unit 1, Kansas Gas and Electric
Calloway Units 1 and 2, Union Electric Company
Tyrone Unit 1, Northern States
Sterling Unit 1, Rochester Gas and Electric
Indian Point Units 2 and 3, Commonwealth Edison, N.Y.
H. B. Robinson Unit 2, Carolina Power & Light Company
Turkey Point Units 3 and 4, Florida Power and Light Company
R. E. Ginna Unit 1, Rochester Gas and Electric
Point Beach Units 1 and 2, Wisconsin Electric Company

ALTERNATIVE 4

New England Units 1 and 2, New England Power Company
Marble Hill Units 1 and 2, Public Service Company of Indiana
Shearon Harris Units 1, 2, 3 and 4, Carolina Power and Light
Carroll County Units 1 and 2, Commonwealth Edison
Jamesport Units 1 and 2, Long Island Lighting
Haven Units 1 and 2, Wisconsin Energy Center
Wisconsin Units 3, 4, 5 and 6, Wisconsin Electric Company

TABLE 2-2

SUMMARY OF PROPOSED REQUIREMENTS FOR WESTINGHOUSE PLANTS

ALTERNATE PLANT MODIFICATIONS

1	2	3	4
Nothing	AMSAC*	AMSAC*	AMSAC* Analysis (99% MTC)

* ATWS Mitigating System Actuation Circuitry satisfying criteria in Appendix C, Volume 3, NUREG-0460.

3.0 ANALYTICAL BASIS

The ATWS analyses presented in WCAP-8330 (1974) were based upon the guidelines set forth in WASH-1270. The analyses presented herein are based upon guidelines specified by the NRC after 1974, and published in NUREG-0460 (1978) after the NRC review of WCAP-8330 and subsequent Westinghouse submittals.

At no time is automatic reactor scram or control rod insertion assumed. All other components, equipment, and systems are assumed to operate normally during the ATWS event provided that:

- Failure of the equipment, component, or system is not the cause of the transient being analyzed;

- The function of the equipment, component, or system is not disabled as a consequence of the transient being analyzed; and

- The probability of failure of the component, equipment, or system is reasonably small during the interval of the transient being analyzed.

Where an operating control band is associated with a parameter, the least favorable value within the band was chosen for each analysis. Instrument or calibration errors were not included. The initial plant power chosen was the least favorable power in the range 0 percent to 100 percent consistent with the nature of the transient being analyzed.

Various control and safety features within the system limit the consequences of a postulated ATWS event. These features fall into two general categories, normal control systems and standby systems. The normal control systems are assumed to be operating at the initiation of the ATWS event. Experience shows that such systems continue to operate reliably during plant transients, and these systems are assumed to continue operating normally for the relatively short times associated with the postulated ATWS events.

The standby features available to mitigate the consequences of plant transients have been designed to operate reliably upon demand, and are assumed to function as designed.

It should be noted that there is a difference between "pressurizer pressure" and "system pressure" as used in this report. When pressurizer pressure is given, it refers to the pressure in the pressurizer, whereas the system pressure is defined to be the pressure taken at the discharge of the reactor coolant pump, the maximum pressure in the reactor coolant system. The system pressure definition includes pump head and elevation head and will be higher than pressurizer pressure by as much as 100 psi.

3.1 METHODS

The computer codes used for the analysis of these ATWS events are basically the same as the codes used in 1974 for WCAP-8330. A detailed steam generator heat transfer simulation code has been added. The codes and models are discussed in Section 4.

3.2 ASSUMED PLANT PARAMETERS

3.2.1 PLANT DESCRIPTION

Table 3-1 provides a list of typical parameters for 2-, 3-, and 4-loop plants for various steam generator models. The table represents a composite of conservative parameters rather than a particular Westinghouse plant. Use of these typical parameters allows many plants to be bracketed by the reference case analyses. The reference plant is defined to be a 4 loop, 51 Series steam generator plant with the typical parameters shown in Table 3-1-a.

The NRC letter of February 15, 1979 requested values for a list of specific parameters which determine the plant behavior during an ATWS. Westinghouse has already supplied these in various previous submittals; but these parameters and values are listed in Table 3-2 for completeness.

3.2.2 REACTOR COOLANT FLOW

Reactor coolant flow is forced through the reactor core and loop piping by fixed speed centrifugal pumps. Flow is constant, depending only upon how many reactor coolant pumps are in operation. For calculational convenience in the ATWS analyses, the thermal-hydraulic design flow was assumed. This is conservative since design margins in core and loop pressure drops and in pump head ensure that measured flow, including allowance for measurement error, is at least equal to the design flow. Typically, coolant flow is 4 percent, or more, above thermal-hydraulic design.

During the transient, pump cavitation was assumed to occur when the cold leg temperature approached saturation (6°F was assumed in the analysis). Following cavitation, the flow was calculated using pump and pressure drop characteristics of the Reactor Coolant System. Cavitation of a single-stage centrifugal pump for high pressure fluid will cause some small reduction in flow. In all cases, the most adverse core and reactor coolant system conditions occur prior to cavitation.

3.2.3 LIQUID RELIEF DISCHARGE RATES

During some postulated ATWS events, the pressurizer fills with liquid due to expansion of the reactor coolant. An analytical model is used to predict the liquid relief rate for the power-operated relief valves and safety valves during these intervals. Homogeneous Equilibrium Model is used in these ATWS analyses as required by NUREG-0460, and as given in ANS Standard N.661.

A homogeneous equilibrium critical flow model applied at the nozzle of the valves predicts mass discharge rates through the valves as a function of upstream fluid temperature and pressure. For the typical downstream piping configuration, these homogeneous equilibrium valve discharge rates are independent of downstream choking phenomena.

A discharge coefficient of .975 was used and a conservative margin multiplier of 0.90 was applied, as per NRC requirements. The initial pressurizer water enthalpy (H_f at 2250 psia) is used to calculate the

H.E.M. relief rate, since the pressurizer fills at a higher pressure; therefore, this water enthalpy represents the maximum in subcooling. Also, 10 percent pressure accumulation was assumed for the spring-loaded pressurizer safety valves when relieving water, instead of the normal 3 percent for steam relief.

For the range of pressurizer fluid conditions encountered in ATWS, the homogeneous equilibrium critical flow calculation represents a lower bound to the prediction of mass discharge rates. This position is indicated by a review of applicable experimental data and by consideration of flow phenomena.

3.2.4 MODERATOR TEMPERATURE COEFFICIENT

An ATWS occurrence, which may lead to serious consequences, invariably results in an increase in the primary coolant temperature. Since the moderator temperature coefficient in the core is negative, this temperature increase results in an insertion of negative reactivity which terminates the transient. Because of the importance of the moderator temperature coefficient, detailed multidimensional calculations were performed.

3.2.4.1 Method of Calculation

The moderator density coefficient is used in the neutron kinetics equation instead of the moderator temperature coefficient. The density coefficient is easily derived from the temperature coefficient by using known reactor coolant system parameters, i.e., temperature and pressure. Three-dimensional diffusion theory was used to calculate the density coefficient because of the need to account for large enthalpy rises and possible boiling that could occur in the ATWS transients. The moderator density coefficient that is used is shown in Figure 3-1. These results are typical.

The boron concentration is the major factor affecting the density coefficients. The reason for this is that the density coefficient is the effect on reactivity of changes in the moderator density. For example, as the density decreases, moderation of neutrons by the water becomes less and the reactivity of the core becomes less. But also, as the density of the water decreases, the amount of boron/cm^3 decreases which increases reactivity. The trade-off between these two opposing effects results in the magnitude of the coefficient.

The change in the density coefficient between full power equilibrium xenon and no xenon is due to the resulting adjustment of boron concentration. The change in the density coefficient between BOL and EOL as shown in Figure 3-2 is due mainly to the boron concentration change. Large burnup accumulations ($\sim$ 10,000 MWD/MTU), after removing the effect of boron, cause only a slight increase in the density coefficient.

3.2.4.2 <u>Experimental Results</u>

During start-up of each core, there is a measurement of the moderator temperature coefficient. There have also been several other measurements, both at zero power and at power. The agreement between measured and predicted is quite good. Both the average difference between measured and prediction and the standard deviation of these values is less than 1 pcm/^{o}F.

3.2.4.3 <u>Definition of 95%/99% Value</u>

As discussed, the moderator coefficient is a strong function of boron concentration and somewhat weaker function of power level. The boron concentration changes during the core life because of burnup and to compensate for xenon concentration changes that would occur because of power changes such as load follow. To account for all of these effects and to determine what fraction of the time the coefficient would be more positive than a specific value the following conservative assumptions were made.

1. Continuous load follow throughout core life. This maximizes the xenon changes and require a higher boron concentration thereby making the coefficient more positive.

2. Slow start-up rate. A slow start-up rate maximizes the time to build-in xenon and thereby makes the coefficient more positive.

3. Short shutdowns. Two short shutdowns are assumed per month of operation. The shutdown time is chosen so that the xenon concentration peaks and thereby requires a larger boron concentration (more positive coefficient) when the plant returns to power.

4. Long shutdown. One long shutdown is assumed per month of operation such that all the xenon is removed from the core. This implies that the startup will require a higher boron concentration (more positive coefficient).

5. The Tech Spec value of a zero coefficient at zero power is used. Because of the higher average temperature at full power than at zero power the coefficient will be at least -3 pcm/°F at full power with no xenon.

6. with equilibirium xenon in the core the boron concentration will be reduced by about 270 ppm. This is equivalent to making the coefficient more negative by 5 pcm/°F.

These six effects are taken into account to determine what percentage of the time the coefficient is more positive or negative than a given value. The results of these calculations show that the coefficient will be more negative than -8 pcm/°F for 95% of the time, and more negative than -7 pcm/°F for 99% of the time that the core power is greater than 80% of nominal.

3.2.5 DOPPLER EFFECTS

The model for Doppler feedback used in the ATWS analyses contains two components. The first of these is the fuel temperature change that occurs because of power level changes. Figure 3-3 shows the integral of this term as a function of power level. The total defect is 1.22% $\Delta k/k$ which is typical of beginning of core life operation. The second term accounts for fuel temperature changes (including gap effects) because the moderator temperature changes. This is described in detail in the LOFTRAN report[2]. The value of the coefficients used in the ATWS analysis is -2 pcm/^{o}F.

3.2.6 INSERTED ROD WORTH

The inserted rod worth during normal operation will typically be less than 0.3%$\Delta\rho$ plus the power defect at BOL.

3.2.7 CORE PEAKING FACTORS

The peaking factors used to determine the minimum DNBR for the ATWS analyses were the same as those used in FSAR analyses except that the uncertainty associated with $F^N_{\Delta H}$ was not included. A value of 1.435 was used for $F^N_{\Delta H}$. Calculations indicate that 1.435 represents an upper bound to the radial hot channel power over the entire fuel cycle. Transient peaking factors were determined from multidimensional nuclear calculations using system statepoints. These analyses verified the conservatism of the DNBR calculations.

3.2.8 DECAY HEAT

For many of the postulated ATWS events, decay heat determines the equilibrium core thermal output that is approached after the fission power output ceases. The decay heat model used for the ATWS analyses contained in this report is based upon the ANS finite irradiation decay heat method described in ANS 5.1. This approach is conservative since the ANS finite irradiation decay heat method is based upon a minimum irradiation time of 8000 hours (about one year) in the newest core region, while ATWS thermal transients analyzed assume beginning of core life conditions (in order to predict the most severe transient). Thus the decay heat prediction based upon 8000 hours of operation overestimates the decay heat expected at beginning of life.

3.3 OPERABLE PLANT FEATURES

3.3.1 OPERATIONAL SYSTEMS

The following systems were assumed operational in the ATWS analyses.

3.3.2 PRESSURIZER PRESSURE CONTROL

The pressurizer control system is designed to maintain the pressurizer pressure at its nominal value, typically 2250 psia. If pressure increases, two separate, automatically controlled spray valves open to discharge water at cold leg temperatures into the steam space. The maximum design flows of the spray valves for the ATWS analyses are given in Table 3-1. If pressurizer pressure decreases, constant output and proportional heaters are actuated. The total heater capacity is also given in Table 3-1.

3.3.3 PRESSURIZER LEVEL CONTROL

Pressurizer level is also a controlled parameter. The water volume varies from 450 ft^3 at no load to 1080 ft^3 at full load for a 4-loop plant. Since pressurizer level control is relatively slow, its beneficial effect in maintaining level was neglected in the transient analyses.

3.3.4 FEEDWATER CONTROL

During normal plant operation, feedwater flow is automatically adjusted by a control valve that is controlled on the basis of feedwater flow, steam flow out of the steam generators, and steam generator water level.

3.3.5 TURBINE CONTROL

During normal plant operation, the steam flow to the turbine is dependent upon turbine demand and any changes in steam generator secondary side pressure are compensated for by automatic opening or closing of the turbine control valve. This valve is approximately 95 percent open at full power operation.

3.3.6 AUTOMATIC ROD CONTROL AND REACTOR COOLANT AVERAGE TEMPERATURE CONTROL

Automatic rod control was not assumed to be operational during the ATWS events, since one of the guidelines for these analyses was no trip or rod insertion. However, prior to the initiation of the ATWS event, it is assumed that the rod control system is operating normally, controlling the average temperature (i.e., the average temperature of the primary side). The average temperature is programmed to be controlled as a linear function of reactor power between zero and 100 percent load; however, a control deadband of $\pm$ 1-1/2°F is associated with the average temperature. The initial value of the average temperature for the ATWS analyses was taken to be the least favorable value within the control deadband for the assumed initial power.

3.3.7 STANDBY SYSTEMS

During normal operation, the following systems are ready to operate if called upon. The effects of these systems were included in the ATWS analyses.

3.3.8 TURBINE TRIP

A turbine trip is initiated by any reactor trip signal listed in Table 3-3, or directly by a high-high steam generator level. However, for the reference ATWS Loss of Feed analyses, turbine trip is assumed to occur after generation of an AMSAC signal. Turbine trip is part of the initiating sequence in the Loss of Load event, and results as a direct consequence of the Loss of Offsite Power event. Turbine trip is not assumed in any of the other transients.

3.3.9 PRESSURE RELIEVING DEVICES

If pressure continues to increase faster than the reducing effect of pressurizer spray, the pressurizer power-operated relief valves open. The setpoint of these valves is 2350 psia. The relieving capacities for these valves are given in Table 3-1. Two or more relief valves are available to reduce pressure. If pressure continues to increase beyond 2350 psia, the pressurizer is equipped with three spring-loaded safety valves, each with a set pressure of 2500 psia. Three percent pressure accumulation is assumed for steam relief, and ten percent pressure accumulation is assumed for water relief.

The steam flows listed in Table 3-1 are used in the ATWS analyses. For the transients which cause the pressurizer to fill and relieve water through the valves, the homogeneous equilibrium model with an 0.9 multiplier discussed in Paragraph 3.2.3 is used to determine the water relief rate as a function of pressure. The initial pressurizer water enthalpy is assumed to remain constant throughout the transient, for the purpose of calculating the water relief rate.

3.3.10 STEAM DUMP CONTROL

The steam dump is actuated following turbine trip to remove stored energy and core decay heat from the system without actuating the steam generator safety valves. A 40 percent steam dump capacity is assumed in the ATWS analyses.

3.3.11 AUXILIARY FEEDWATER SYSTEM

The auxiliary feedwater system is actuated on low-low water level in the steam generators, by loss of offsite power, by a safety injection signal, by a trip of all the main feedwater pumps, or by a manual start signal. For these analyses, actuation is assumed to occur upon generation of an AMSAC signal. The total auxiliary feedwater capacity for 2-, 3-, and 4-loop plants used in the ATWS analyses are given in Table 3-1. In each case, these flow rates represent a lower bound for the plants covered by the generic analyses, and therefore guarantee conservatism. After actuation of auxiliary feedwater, the 440°F water in the feedwater lines must be purged before the colder auxiliary feedwater enters the steam generator. The volume to be purged is dependent upon the plant and the number of loops. Typical purge volumes used in the ATWS analyses for 2-, 3-, and 4-loop plants are listed in Table 3-1.

3.3.12 SAFETY INJECTION SYSTEM

Safety injection is actuated by a manual signal from the operator, by a low pressurizer pressure signal or by a high containment pressure signal. If any of these signals are present, borated water is pumped into the Reactor Coolant System. The borated water increases the reactivity shutdown margin. Only the manual signal is assumed operative during the ATWS transient.

3.3.13 CHEMICAL AND VOLUME CONTROL SYSTEM

The chemical and volume control system provides for normal makeup for the reactor coolant system. However, it is also available to add borated water to the primary system by manual operator action. Credit is not taken for chemical and volume control system makeup during the first 600 seconds of the ATWS transients. However, this system provides an additional shutdown mode available to the operator.

TABLE 3-1-a

TYPICAL PARAMETERS FOR SERIES 51 STEAM GENERATOR PLANTS

Parameters	2-Loop	3-Loop	4-Loop
Core:			
Core power (MWt)	1650	2785	3423
Core length (ft)	12	12	12
Number of assemblies	121	157	193
Reactor Coolant System:			
Total volume (ft^3) including pressurizer and surge line	6230	9570	12,520
Nominal[a] pressure (psia)	2250	2250	2250
Nominal[a] flow (gpm)	178,000	278,400	354,000
Nominal[a] average temperature (^oF)	583.0	580.3	584.65
No-load temperature (^oF)	547	557	557
Nominal[a] reactor vessel inlet temperature (^oF)	551.9	546.6	552.3
Nominal[a] reactor vessel outlet temperature (^oF)	614.2	614.0	617.0
Pressurizer:			
Total volume of pressurizer and surge line (ft^3)	1021.3	1436.8	1843.7
Nominal[a] water volume (ft^3)	600	750	1080
Heater capacity (kw)	1000	1000	1800

TABLE 3-1-a (Continued)

TYPICAL PARAMETERS FOR SERIES 51 STEAM GENERATOR PLANTS

Parameters	2-Loop	3-Loop	4-Loop
Pressurizer (Continued)			
Maximum spray rate (lbs/sec)	51.9	75.0	87.2
Power-operated relief valve steam flow capacity (lbs/hr) (at 2350 psia)	2-210,000 (each)	2-210,000 (each)	2-210,000 (each)
Safety valve steam flow capacity (lbs/hr) (at 2500 psia)	2-325,000 (each)	3-345,000 (each)	3-420,000 (each)
Power-operated relief valve opening pressure (psia)	2350	2350	2350
Safety valve, start open to full open pressure (psia)	2500 to 2575	2500 to 2575	2500 to 2575
Secondary System:			
Steam generator (SG) type:	51	51	51
SG design pressure (psia)	1100	1200	1200
Nominal[a] steam pressure (psia)	750	850	910
No-load steam pressure (psia)	1020	1106	1106
Nominal[a] steam temperature (^{o}F)	510.8	525.2	533.3
Nominal[a] steam flow (lbs/sec)	996/SG	1142/SG	1046/SG
Nominal[a] SG secondary side fluid mass (lbs)	101,600/SG	105,600/SG	101,600/SG
Maximum steam moisture (%)	0.25	0.25	0.25

TABLE 3-1-a (Continued)

TYPICAL PARAMETERS FOR SERIES 51 STEAM GENERATOR PLANTS

Parameters	2-Loop	3-Loop	4-Loop
Secondary System (Continued)			
Nominal[a] feed temperature (^{o}F)	435.8	446.6	439.8
Nominal[a] feed enthalpy (Btu/lb)	414.8	426.6	419.6
Auxiliary feed flow capacity (gpm)	800	1400	1760
Auxiliary feed purge volume (ft^3)	261	500	667
Auxiliary feed water available (gal)	150,000	140,000	170,000
Auxiliary feed enthalpy (Btu/lb)	100	100	100

Note:

[a]Nominal refers to value at rated full power.

TABLE 3-1-b

TYPICAL PARAMETERS FOR MODEL D STEAM GENERATOR PLANTS

Parameters	3-Loop	4-Loop
Core:		
Core power (MWt)	2785	3427
Core length (ft)	12	12
Number of assemblies	157	193
Reactor Coolant System:		
Total volume (ft^3) including pressurizer and surge line	9570	11,939
Nominal[a] pressure (psia)	2250	2250
Nominal[a] flow (gpm)	282,000	377,600
Nominal[a] average temperature (^oF)	587.2	588.5
No-load temperature (^oF)	557	557
Nominal[a] reactor vessel inlet temperature (^oF)	554.3	558.3
Nominal[a] reactor vessel outlet temperature (^oF)	620.1	618.8
Pressurizer:		
Total volume of pressurizer and surge line (ft^3)	1436.8	1843.7
Nominal[a] water volume (ft^3)	750	1080
Heater capacity (kw)	1000	1800

TABLE 3-1-b (Continued)

TYPICAL PARAMETERS FOR MODEL D STEAM GENERATOR PLANTS

Parameters	3-Loop	4-Loop
Pressurizer (Continued)		
Maximum spray rate (lbs/sec)	75.0	87.2
Power-operated relief valve steam flow capacity (lbs/hr) (at 2350 psia)	2-210,000 (each)	2-210,000 (each)
Safety valve steam flow capacity (lbs/hr) (at 2500 psia)	3-345,000 (each)	3-420,000 (each)
Power-operated relief valve opening pressure (psia)	2350	2350
Safety valve, start open to full open pressure (psia)	2500 to 2575	2500 to 2575
Secondary System:		
Steam generator (SG) type:	D	D
SG design pressure (psia)	1200	1200
Nominal[a] steam pressure (psia)	850	910
No-load steam pressure (psia)	1106	1106
Nominal[a] steam temperature (^{o}F)	525.2	533.3
Nominal[a] steam flow (lbs/sec)	1142/SG	1047/SG
Nominal[a] SG secondary side fluid mass (lbs)	107,000/SG	107,000/SG
Maximum steam moisture (%)	0.25	0.25

TABLE 3-1-b (Continued)

TYPICAL PARAMETERS FOR MODEL D STEAM GENERATOR PLANTS

Parameters	3-Loop	4-Loop
Secondary System (Continued)		
Nominal[a] feed temperature (^{o}F)	446.6	439.8
Nominal[a] feed enthalpy (Btu/lb)	426.6	419.6
Auxiliary feed flow capacity (gpm)	1400	1760
Auxiliary feed purge volume (ft^3)	500	667
Auxiliary feed water available (gal)	140,000	170,000
Auxiliary feed enthalpy (Btu/lb)	100	100

Note:

[a]Nominal refers to value at rated full power.

TABLE 3-1-c

TYPICAL PARAMETERS FOR MODEL F STEAM GENERATOR PLANTS

Parameters	3-Loop	4-Loop
Core:		
Core power (MWt)	2785	3427
Core length (ft)	12	12
Number of assemblies	157	193
Reactor Coolant System:		
Total volume (ft^3) including pressurizer and surge line	9570	12,049
Nominal[a] pressure (psia)	2250	2250
Nominal[a] flow (gpm)	292,800	377,600
Nominal[a] average temperature (^oF)	587.2	591.5
No-load temperature (^oF)	557	557
Nominal[a] reactor vessel inlet temperature (^oF)	555.4	561.4
Nominal[a] reactor vessel outlet temperature (^oF)	619.0	621.7
Pressurizer:		
Total volume of pressurizer and surge line (ft^3)	1436.8	1843.7
Nominal[a] water volume (ft^3)	750	1080
Heater capacity (kw)	1000	1800

TABLE 3-1-c (Continued)

TYPICAL PARAMETERS FOR MODEL F STEAM GENERATOR PLANTS

Parameters	3-Loop	4-Loop
Pressurizer (Continued)		
Maximum spray rate (lbs/sec)	75.0	87.2
Power-operated relief valve steam flow capacity (lbs/hr) (at 2350 psia)	2-210,000 (each)	2-210,000 (each)
Safety valve steam flow capacity (lbs/hr) (at 2500 psia)	3-345,000 (each)	3-420,000 (each)
Power-operated relief valve opening pressure (psia)	2350	2350
Safety valve, start open to full open pressure (psia)	2500 to 2575	2500 to 2575
Secondary System:		
Steam generator (SG) type:	F	F
SG design pressure (psia)	1200	1200
Nominal[a] steam pressure (psia)	850	910
No-load steam pressure (psia)	1106	1106
Nominal[a] steam temperature (^{o}F)	525.2	533.3
Nominal[a] steam flow (lbs/sec)	1142/SG	1047/SG
Nominal[a] SG secondary side fluid mass (lbs)	109,000/SG	107,850/SG
Maximum steam moisture (%)	0.25	0.25

TABLE 3-1-c (Continued)

TYPICAL PARAMETERS FOR MODEL F STEAM GENERATOR PLANTS

Parameters	3-Loop	4-Loop
Secondary System (Continued)		
Nominal[a] feed temperature (°F)	446.6	439.8
Nominal[a] feed enthalpy (Btu/lb)	426.6	419.6
Auxiliary feed flow capacity (gpm)	1400	1760
Auxiliary feed purge volume (ft^3)	500	667
Auxiliary feed water available (gal)	140,000	170,000
Auxiliary feed enthalpy (Btu/lb)	100	100

Note:

[a]Nominal refers to value at rated full power.

TABLE 3-1-d

TYPICAL PARAMETERS FOR SERIES 44 STEAM GENERATOR PLANTS

Parameters	2-Loop	3-Loop	4-Loop
Core:			
Core power (MWt)	1520	2208	3025
Core length (ft)	12	12	12
Number of assemblies	121	157	193
Reactor Coolant System:			
Total volume (ft^3) including pressurizer and surge line	6230	9110	11,900
Nominal[a] pressure (psia)	2250	2250	2250
Nominal[a] flow (gpm)	178,000	268,500	358,800
Nominal[a] average temperature (^{o}F)	581.3	574.3	571.5
No-load temperature (^{o}F)	547	557	547
Nominal[a] reactor vessel inlet temperature (^{o}F)	552.4	546.1	542.6
Nominal[a] reactor vessel outlet temperature (^{o}F)	610.3	602.5	600.5
Pressurizer:			
Total volume of pressurizer and surge line (ft^3)	1021.3	1436.8	1843.7
Nominal[a] water volume (ft^3)	600	750	1080
Heater capacity (kw)	1000	1000	1800

TABLE 3-1-d (Continued)

TYPICAL PARAMETERS FOR SERIES 44 STEAM GENERATOR PLANTS

Parameters	2-Loop	3-Loop	4-Loop
Pressurizer (Continued)			
Maximum spray rate (lbs/sec)	42.6	75.0	87.2
Power-operated relief valve steam flow capacity (lbs/hr) (at 2350 psia)	2-179,000 (each)	2-179,000 (each)	2-179,000 (each)
Safety valve steam flow capacity (lbs/hr) (at 2500 psia)	2-288,000 (each)	3-288,000 (each)	3-408,000 (each)
Power-operated relief valve opening pressure (psia)	2350	2350	2350
Safety valve, start open to full open pressure (psia)	2500 to 2575	2500 to 2575	2500 to 2575
Secondary System:			
Steam generator (SG) type:	44	44	44
SG design pressure (psia)	1100	1100	1100
Nominal[a] steam pressure (psia)	750	850	910
No-load steam pressure (psia)	1020	1020	1020
Nominal[a] steam temperature (^{o}F)	510.8	525.2	533.3
Nominal[a] steam flow (lbs/sec)	918/SG	906/SG	924/SG
Nominal[a] SG secondary side fluid mass (lbs)	86,750/SG	86,750/SG	86,750/SG
Maximum steam moisture (%)	0.25	0.25	0.25

TABLE 3-1-d (Continued)

TYPICAL PARAMETERS FOR SERIES 44 STEAM GENERATOR PLANTS

Parameters	2-Loop	3-Loop	4-Loop
Secondary System (Continued)			
Nominal[a] feed temperature (^{o}F)	435.8	446.6	439.8
Nominal[a] feed enthalpy (Btu/lb)	414.8	426.6	419.6
Auxiliary feed flow capacity (gpm)	800	1400	1760
Auxiliary feed purge volume (ft^3)	261	500	667
Auxiliary feed water available (gal)	150,000	140,000	170,000
Auxiliary feed enthalpy (Btu/lb)	100	100	100

Note:

[a]Nominal refers to value at rated full power.

TABLE 3-2

INITIAL AND BOUNDARY CONDITIONS FOR PARAMETERS TO BE CHARACTERIZED AND JUSTIFIED IN ATWS ANALYSIS

As required by NRC letter of February 15, 1979

(1)	Reactivity coefficients	See Section 3.3.4
(2)	Core power distribution	Section 3.3.7
(3)	Core power	Table 3-1
(4)	Core coolant flow	Table 3-1
(5)	Core inlet temperature	Table 3-1
(6)	Control rod insertion	Section 3.3.6
(7)	Soluble boron concentration in the core	Section 3.3.4
(8)	Reactor decay heat function. Best estimate (no ANS & 20)	Section 3.3.8
(9)	Pressurizer water level	Section 3.4.3
(10)	Pressurizer pressure	Section 3.4.2 and Table 3-1
(11)	Pressurizer safety and relief valve flow rate (both steam and water)	Section 3.4.9
(12)	Steam generator temperature	Table 3-1
(13)	Steam generator pressure	Table 3-1
(14)	Steam generator steam flow rate	Table 3-1
(15)	Steam generator safety and relief valve flow rate	Sized to relieve more than nominal flow rate
(16)	Steam generator heat transfer coefficient	See Section 4 on Tranflo model
(17)	Steam generator secondary side water inventory	Table 3-1
(18)	Feedwater temperature	Section 3.4.11
(19)	Equipment performance	Section 3.4.7
(20)	Turbine bypass condition	Section 3.4.10

TABLE 3-2 (cont)

INITIAL AND BOUNDARY CONDITIONS FOR PARAMETERS TO BE CHARACTERIZED AND JUSTIFIED IN ATWS ANALYSIS (Continued)

As required by NRC letter of February 15, 1979

(21) Containment ambient conditions (pressure, temperature, including suppression pool temperature and level, etc.)	NA
(22) Fuel element gap size	Section 3.2.5
(23) Auxiliary feedwater: number of pumps available, number assumed in analysis	Section 3.4.11 and Table 3-1
(24) Auxiliary feedwater flow for each pump as a function of system pressure	Full capacity at all pressures
(25) Auxiliary feedwater temperature	Table 3-1
(26) Vessel water level	NA
(27) All water sources inventory	NA
(28) HPSI and any other high pressure makeup and poison system flow rate as a function of system pressure	NA
(29) HPCI(s) and any other high pressure makeup and poison system flow rate as a function of system pressure	NA
(30) Boron concentration of HPSI and/or other borated solution	NA
(31) Sodium pentaborate concentration of SLCS	NA
(32) Reactor coolant mass	500000 lbs
(33) Reactor coolant system volume	Table 3-1
(34) Core average temperature	Table 3-1
(35) Pressurizer water volume	Table 3-1
(36) Pressurizer total volume	Table 3-1
(37) Number of relief valves on pressurizer	Table 3-1

TABLE 3-2 (cont)

INITIAL AND BOUNDARY CONDITIONS FOR PARAMETERS TO BE CHARACTERIZED AND JUSTIFIED IN ATWS ANALYSIS (Continued)

As required by NRC letter of February 15, 1979

(38) Number of safety valves on pressurizer	Table 3-1
(39) Relief valve setpoint on pressurizer	Table 3-1
(40) Safety valve setpoint on pressurizer	Table 3-1
(41) Feedwater flow rate	Equal to steam flow rate, Table 3-1
(42) Number of S/V per steam generator on steam line	Typically 4 or 5 safety valves
(43) Number of steam bypass valves	NA
(44) Setpoint of steam bypass valves	NA
(45) Capacity of steam bypass valves	NA
(46) Number of atmospheric dump valves.	One per steam line
(47) Capacity of atmospheric dump valves	40 percent
(48) Steam flow rate including bypass flow	NA
(49) Valve closure times	NA
(50) Auxiliary feedwater pump start times	Within 60 seconds of an AMSAC signal
(51) Poison reactivity worth	NA
(52) Poison-water mixing efficiency	NA
(53) Signals and setpoints for all automatically actuated systems	Sections 3.4.8 and 3.4.11
(54) Containment pressure, temperature and volume	NA
(55) Steam generator tube leakage	Tech spec value adjusted for pressure
(56) Service water temperature and flow	NA
(57) Component cooling water temperature and flow	NA
(58) RHR heat exchanger performance	NA

TABLE 3-2 (cont)

INITIAL AND BOUNDARY CONDITIONS FOR PARAMETERS TO BE CHARACTERIZED AND JUSTIFIED IN ATWS ANALYSIS (Continued)

As required by NRC letter of February 15, 1979

(59) Fouling factors in heat exchangers	NA
(60) Power availability for pressurizer relief valves during loss of offsite power ATWS event	Air operated valves
(61) Core average void fraction	NA
(62) Heat transfer surface area	Typically 60000 ft^3 core heat transfer area for a 4 loop 12 ft. core

TABLE 3.3

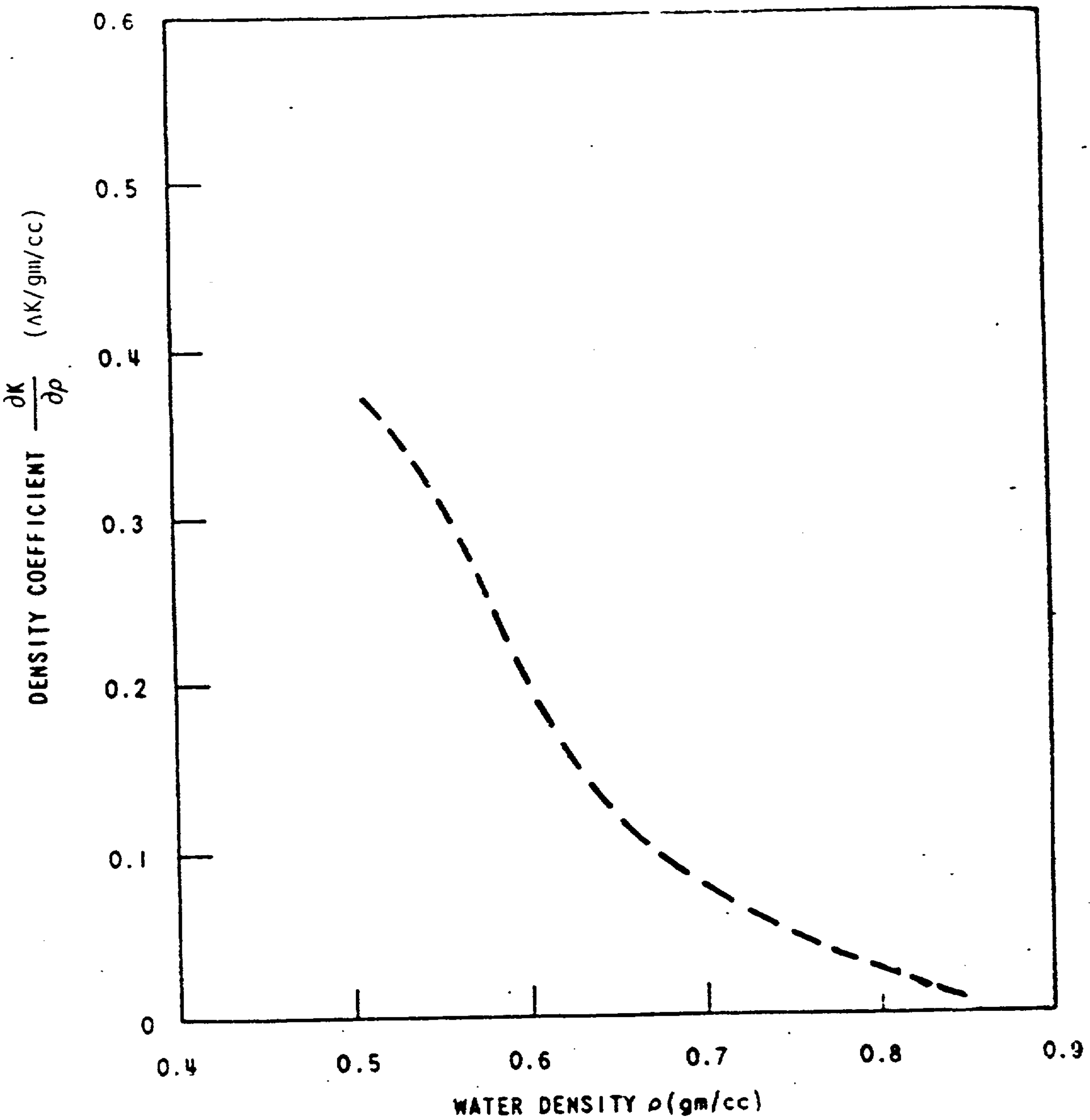
TRIP FUNCTIONS

Trip Function	Actuating Signals	Criteria	Degree of Protection
Trip reactor upon complete loss of reactor coolant flow	Undervoltage; RCP breaker position	ANS PWR ANS 4.1	No core damage
Trip reactor upon partial loss of reactor coolant flow	Frequency sensor	ANS PWR ANS 4.1	No core damage for frequency decreasing at rates below maximum credible rate (usually 4 Hz/sec)
Trip reactor upon RCS overpressurization	Pressure sensor	ANS PWR ANS 4.1	No core damage; no loss of function of any barrier to the escape of radio-active products
Trip reactor upon RCS depressurization	Pressure sensor	ANS PWR ANS 4.1	No core damage
Trip reactor upon approach to DNB (power operation)	Power Range High Neutron Flux; Over-temperature ΔT (temperature and pressure sensors, excore ion chambers)	ANS PWR ANS 4.1	No core damage
Trip reactor upon approach to kw/ft limit (power operation)	Power Range High Neutron Flux; Over-power ΔT (temperature sensors, excore ion chambers)	ANS PWR ANS 4.1	No core damage
Trip reactor upon turbine trip	Auto-stop oil pressure switches, turbine stop valve position sensors		No actuation of primary or secondary safety valves; limit severity of transient occurring with a relatively high frequency

TABLE 3-3 (Continued)

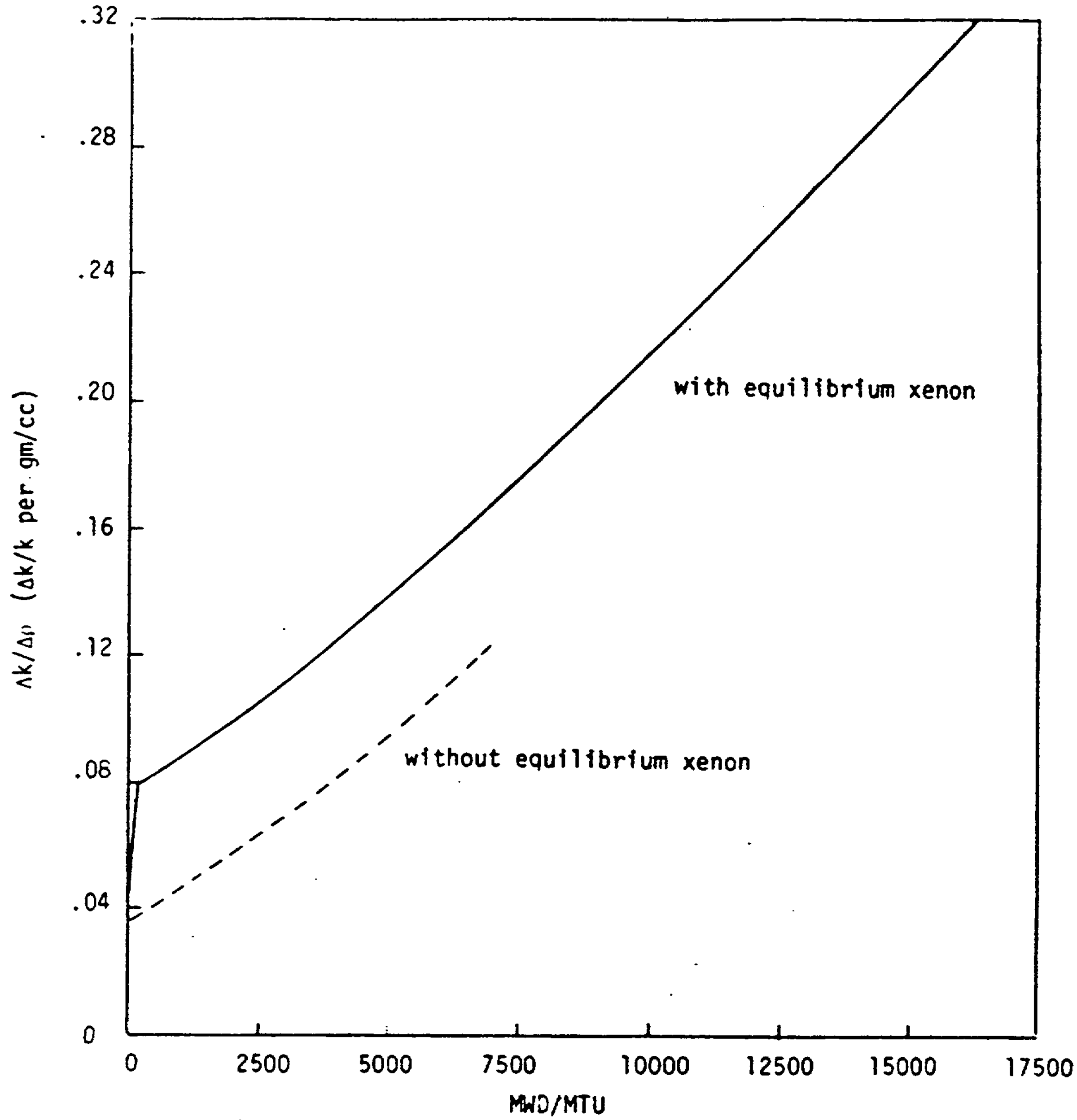
TRIP FUNCTIONS

Trip Function	Actuating Signals	Criteria	Degree of Protection
Trip reactor upon pressurizer high water level	Level sensors; differential pressure sensors		Prevent water solid RCS at power; no water relief through pressurizer relief or safety valves
Trip reactor upon loss of heat sink	Steam generator level sensors (actually differential pressure sensors); feedwater flow and steam flow sensors	ANS PWR ANS 4.1	No core damage; no loss of function of any barrier to the escape of radioactive products; no water relief through pressurizer relief or safety valves; minimizes required auxiliary feed pump sizes; maximizes time for operator action following feed pipe break; minimizes steam generator thermal shock for loss of feed or feed pipe break
Trip reactor on operator judgment	Control board button or switch	ANS PWR ANS 4.1	Back-up trip
Trip reactor on SIS actuation	SI signal	ANS PWR ANS 4.1	No core damage
Trip reactor upon rod ejection	Neutron Flux sensors	ANS PWR ANS 4.1	Minimize core damage
Trip reactor upon rod bank drop	Neutron Flux sensors	ANS PWR ANS 4.1	No core damage
Trip reactor on approach to DNB or kw/ft limit (startup operation)	Source and Intermediate range neutron flux sensors	ANS PWR ANS 4.1	No core damage



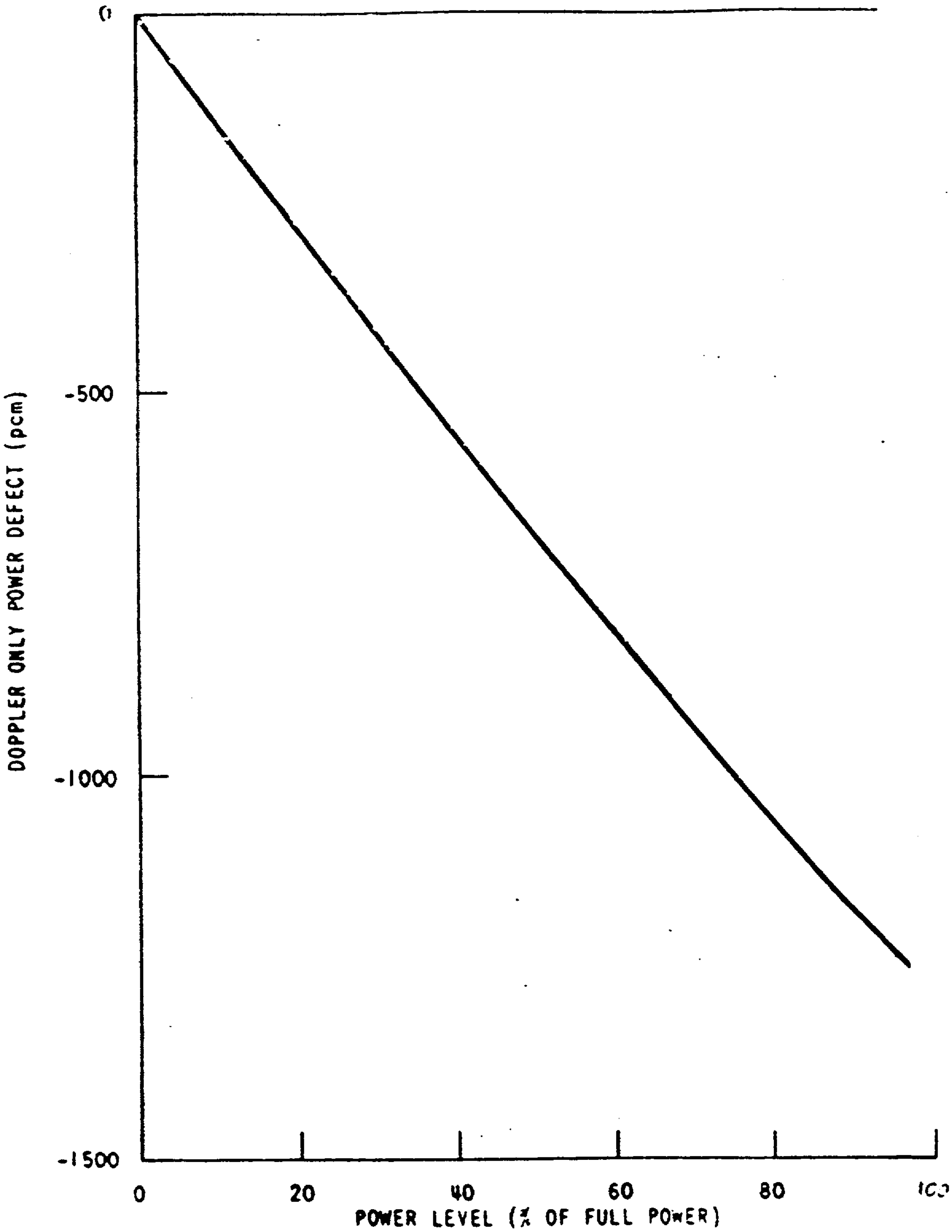
Density Coefficient as a Function of Water Density (900 ppm Boron)

FIGURE 3-1



Hot Full Power Temperature Coefficient for Typical 17 x 17 4-Loop Plant at the Critical Boron Concentration

FIGURE 3-2



Doppler Only Power Defect BOL
for Typical 4-Loop Plant

FIGURE 3-3

4.0 COMPUTER CODES USED FOR ATWS ANALYSIS

4.1 INTRODUCTION

Four computer codes were used in the ATWS analyses. These codes are LOFTRAN(2), FACTRAN(3), THINC-III(4), and TRANFLO(5). The important input, output and model assumptions for each code are given in the following sections. The data flow between codes is shown in Figure 4-1.

4.2 LOFTRAN

The systems code used in the ATWS analyses was LOFTRAN. The basic flow nodalization uses an explicit solution of the system equations. The core region and steam generator primary can be subdivided into many nodes to provide an accurate representation of heat transfer and flow in these regions. The core was represented by 15 nodes and the steam generator primary by 14 nodes in these analyses.

The calculated pressurizer pressure and calculated reactor coolant system pressure differ by the pressure drop in the surge line. This effect is explicitly accounted for in LOFTRAN calculations. The effects of loop pressure drops and elevation head are also explicitly accounted for in the system pressure calculated by LOFTRAN.

4.2.1 PRESSURIZER MODEL

An important consideration in the system modeling is the treatment of the pressurizer. LOFTRAN represents the pressurizer as two separate nodes, one to model the water region and one for the steam region. Mass transfer, but not heat transfer between the nodes, is modeled. It includes the effects of heaters, spray, steam condensation and valve relief.

4.2.2 CORE HYDRAULIC MODEL

A solution of the momentum equation including frictional losses, fluid inertia and density changes is used for transients that involve a flow coastdown (e.g., station blackout). The core is modeled as a single average channel with 15 axial nodes. Heat transfer from the fuel, fuel and coolant temperatures, and coolant density and flow are calculated in each node.

4.2.3 STEAM GENERATOR MODEL

The LOFTRAN steam generator model used for the ATWS analyses divides the primary side into 14 nodes. The primary side film coefficient was determined using the Dittus-Boelter correlation. The secondary side film coefficient is calculated as a function of heat transferred to the secondary and secondary side pressure, using the Jens-Lottes correlation.

The secondary side heat transfer coefficient is reduced as the water inventory in the secondary decreases below the volume needed to cover the tube bundle. The heat transfer correlation used in LOFTRAN is calculated and verified using the TRANFLO code discussed in Section 4.5.

4.2.4 LOFTRAN INPUT

The significant system parameters input to the LOFTRAN code are given in Table 3-1. Those parameters which are input to model system response to a specific transient are listed in the discussion of that transient.

4.2.5 LOFTRAN OUTPUT

LOFTRAN outputs a variety of parameters at time intervals specified by the user. The key parameters for the ATWS analyses that are of direct interest or are needed as input for FACTRAN and/or THINC-III are given below.

- Nuclear Power Vs. Time
- System Pressure Vs. Time
- Coolant Temperatures Vs. Time
- Coolant Flow Rate Vs. Time
- Pressurizer Water Volume Vs. Time
- Surge Rates Into the Pressurizer Vs. Time
- Flow Out of Pressurizer Relief & Safety Valves Vs. Time

4.3 FACTRAN

FACTRAN calculates the transient temperature distribution in a cross-section of a metal-clad UO_2 fuel rod and the heat flux at the surface of the rod, using as input the nuclear power and coolant conditions (pressure, flow, temperature) predicted by LOFTRAN.

The primary function of the FACTRAN code in ATWS analyses is to calculate core heat flux used by the THINC code to generate DNB ratios; no hot spot calculations are done.

The fuel rod is divided into a number of concentric rings. The number of rings required for the fuel itself is optional and specified in the input. In the ATWS analyses six fuel regions were used. Three more rings were added at the outside of the fuel: they represent, respectively, the gap, the clad, and the film. The transient heat conduction equations are written for each ring, in finite difference form, as a system of linear equations that are solved simultaneously. The coefficients of the system are calculated from the temperatures in each ring at time t, and the unknowns are the temperature and heat flux in each ring at time $t + \Delta t$.

4.3.1 FILM HEAT TRANSFER COEFFICIENT

FACTRAN has the capability of using different correlations to predict heat transfer coefficients before and after DNB is predicted. However, DNB is not predicted for any of the ATWS transients so the transition between correlations is not required. The heat transfer coefficients used to predict the heat flux for the THINC-III code for both the average and hot channel are the same as used in non-ATWS analyses.

Before DNB:

At each time step, the forced convection clad surface temperature (Dittus-Boelter correlation) and the local boiling surface temperature (Jens-Lottes correlation) are calculated, based on the heat flux at the previous time step. If the local boiling temperature is higher (forced convection regime), the film is considered as the last section in the system of concentric rings, and the outside boundary condition is the coolant temperature.

If the forced convection temperature is higher (local boiling regime) the clad is considered as the last section in the system, and the outside boundary condition is the local boiling temperature (clad surface temperature).

4.3.2 MATERIAL PROPERTIES

The thermal and mechanical properties of UO_2 and Zircaloy are built into the code in the form of data tables as functions of temperature. At each time step, the properties of the materials constituting each ring of the model are calculated at the ring average temperature.

4.3.3 GAP HEAT TRANSFER COEFFICIENT

The gap heat transfer coefficient is calculated based on the thermal expansion of the pellet; that is, the sum of the radial (one-dimensional) expansions of the rings. Each ring is assumed to expand freely. The cladding diameter is calculated based on thermal expansion and internal and external pressures.

If the outside radius of the expanded pellet is smaller than the inside radius of the expanded clad, there is no fuel-clad contact and the gap conductance is calculated on the basis of the thermal conductvity of the gas contained in the gap. If the pellet outside radius so calculated is larger than the clad inside radius (negative gap), the pellet and the clad are pictured as exerting upon each other a pressure sufficiently large to reduce the gap to zero by elastic deformation of both. This contact pressure determines the gap heat transfer coefficient.

4.3.4 FACTRAN INPUT

The significant fuel parameters needed for FACTRAN are given in Table 4-1.

4.3.5 FACTRAN OUTPUT

The FACTRAN output of interest consists of the following parameters:

- Heat Flux Vs. Time
- Fuel Temperatures Vs. Time
- Clad Temperatures Vs. Time
- Stored Energy in the Fuel Vs. Time

4.4 THINC-III

The THINC-III code is a detailed Thermal-Hydraulic simulation of the reactor core. In THINC-III, the region of the core being studied is considered to be made up of continuous channels divided axially into increments of equal length. At time T = 0, equations representing the conservation of mass, energy, and momentum within a length increment are written for each channel. Considering the static pressure at a given elevation to be uniform, these equations are solved simultaneously to give the changes in density, velocity, and static pressure along the length increment for each channel. This procedure is continued stepwise up the core by using the values at the top of one length step as input quantities for the next axial step. A total of 37 axial steps is used for the ATWS analyses. The core is divided into 5 radial channels, in the following manner.

Channel 1 = hot channel
Channel 2 = surrounding 8 unit cells
Channel 3 = remainder of hot assembly
Channel 4 = surrounding 8 assemblies
Channel 5 = remainder of core

Therefore, the core is divided into 185 nodes (5 radial x 37 axial) for the calculation of minimum DNBR in the ATWS analyses.

Basic assumptions in THINC-III are given below.

- The static pressure at any elevation is considered to be uniform throughout the channel array.

- Local boiling voids are taken as those computed by the modified Thom correlation.

- The flow is considered to be homogeneous. Correction factors for subcooled and bulk boiling are applied to the friction and momentum pressure drop terms in the force balance equation to account for vapor voids effects.

4.4.1 THINC-III INPUT

Typical input paramters for 17x17 fuel used by THINC-III to calculate the DNB ratio in the hot channel for these ATWS analyses are listed in Table 4-2.

4.4.2 THINC-III OUTPUT

The THINC-III output of primary concern for the ATWS analyses is DNB ratio as a function of time.

4.5 STEAM GENERATOR HEAT TRANSFER EQUATION

The TRANFLO code is a general purpose control volume code, developed to allow detailed modeling of thermo-hydraulic effects occurring inside mixed phase processing equipment, to model steam generator responses to transient conditions. The code modeling concept is one of using multiple independent fluid control volumes (nodes), each represented by mass and energy, interconnected by the appropriate flow paths (connectors) to allow mass and energy exchange. Masses and energies associated with the control volumes are assumed to exist homogeneously throughout the volumes, and flows and pressure drops associated with flow paths are applied one-dimensionally between center points of connected control volumes. Mass may be introduced or removed from the system by pre-defined flow leaks into or out of any control volume. Energy may also be added to the system through a specified heat source represented by a continuous pass tube bundle having hot compressed water as a working fluid.

The code computationally solves for system conditions by satisfying the mass and energy conservation equations for all control volumes and balancing the system pressure losses between volumes via the momentum equation. An implicit backward differencing technique which numerically integrates the conservation equations is used to continually adjust control volume masses and energies with time. All other thermodynamic parameters are determined from the adjusted mass and energy by assuming equilibrium throughout each control volume. The adjusted conditions are then used for successive calculations with limits on time step size being determined by a specified maximum fractional change in mass or energy for any node.

The effects of slip between steam and water phases are defined by the Armand correlation which is used to calculate quality of flows in flow connectors as well as determine frictional pressure losses in two phase flow regions. Pressure losses are modified to include two phase flow effects by using an effective fluid specific volume in the frictional and form loss pressure drop equations. The influence of gravity on phase separation is included in regions where phase velocities are between the critical steam separation velocity as defined by Davis and phase velocities for which the Armand correlation provides adequate representation of separation phenomenon.

Heat transfer from the tube bundle into control volumes is determined from the conditions of the working fluid, the tube wall characteristics, and the thermodynamic conditions of the control volumes. An independent selection of an appropriate heat transfer coefficient is made from among eight experimental correlations for all nodes enclosing a segment, or segments, of the tube bundle. The selection is based on the calculated temperature difference between the working fluid and the control volume and the local phase conditions in the control volume. Cooling of the working fluid as it flows through the tube bundle is included in the solution.

The heat transfer correlations used in TRANFLO are listed in Table 4-3 along with the region where they are applied. The conservation of mass, energy, and momentum equations are shown as well as the solution of the system equation.

Selection of the appropriate heat transfer correlation is made as follows:

1. Subcooled Water Regions

   If the calculations from previous time steps indicate a node contains only subcooled liquid, forced convection heat transfer is assumed initially and the Dittus-Boelter correlation is used to determine heat-flux. The heat flux is then used to determine the required tube wall temperature. If the wall temperature is less than the node saturation temperature the heat flux via DittusBoelter is acceptable; if not, local boiling is assumed and the heat flux predicted by the Thom correlation using the difference of the saturation temperature and the node temperature is found. This is compared to the critical heat flux predicted by MacBeth. If it is less than the critical value it is used; if not the larger value of heat flux predicted by using either the Westinghouse transition boiling correlation or Sandberg's stable film boiling (in subcooled liquid) correlation is used.

2. Saturated Mixtures

In saturated regions, if the void fraction is less than 0.9, a similar process to that used in subcooled regions is used. The Dougall and Rohsenow correlation is used for determining heat flux for stable film boiling. If the void fraction in the region is greater than 0.9, forced convection vaporization as predicted by the Schrock and Grossman correlation is assumed rather than stable film boiling. Critical heat flux is again calculated using MacBeth.

3. Superheated Steam

If a region contains only superheated steam, forced convection heat transfer is assumed and the Heineman correlation is used.

4.5.1 TRANFLO INPUT

The TRANFLO input is obtained from the transient results found in the output of the LOFTRAN run. The primary side flow and temperature into the steam generator tubes are input to TRANFLO as a function of time. The secondary side steam flow and enthalpy at the steam generator outlet nozzle and feedwater flow and enthalpy at the inlet nozzle from LOFTRAN are also input to TRANFLO as a function of time.

4.5.2 TRANFLO OUTPUT

The TRANFLO code is used to determine the heat transfer in the steam generator during ATWS event when the steam generator fluid level falls below the top of the U-tubes. The amount of heat transfered during the ATWS event is translated into a curve of steam generator overall UA verses steam generator fluid mass. This curve can then be used in place of the LOFTRAN steam generator heat transfer calculation.

4.5.3 LOFTRAN/TRANFLO COUPLING

Figure 4-2 shows the method in which the LOFTRAN and TRANFLO codes are used together for ATWS calculations. The initial LOFTRAN run assumes a representative curve of UA verses mass from which the steam generator inlet flow, temperature, pressure, steam flow and steam enthalply are input into TRANFLO. The TRANFLO code then calculates a new UA verses curve. This new heat transfer curve is then used in LOFTRAN. A check for convergence is made on the input to the TRANFLO code, i.e. steam generator inlet flow, temperature, and pressure. Studies have shown that the input to TRANFLO converges very quickly.

TABLE 4-1

TYPICAL FACTRAN INPUT

Input	Value
Clad Material	Zircaloy
Clad Outside Diameter	0.374
Clad Thickness	0.0225 in.
Fuel Pellet	0.3210 in.
Nuclear Power Vs. Time	Output from LOFTRAN
System Pressure Vs. Time	Output from LOFTRAN
Coolant Temperature Vs. Time	Output from LOFTRAN
Coolant Mass Flow Vs. Time	Output from LOFTRAN
Time of DNB	Output from THINC-III

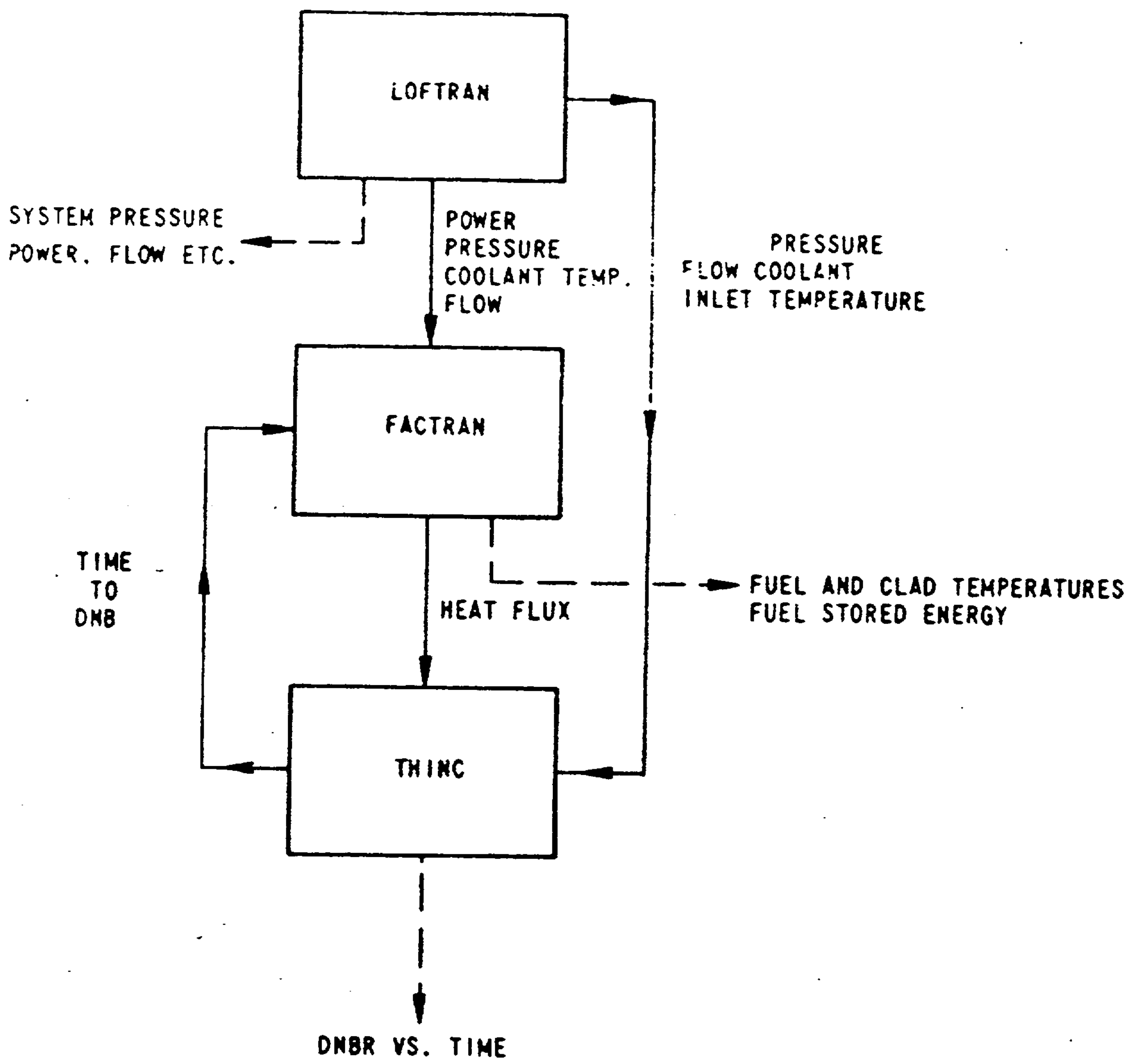
TABLE 4-2

THINC-III INPUT

Input	Value
Peaking Factors	$F_{\Delta H} = 1.435$ $F_Z = 1.55$
Average Heat Flux Vs. Time	Output from FACTRAN
Core Inlet Enthalpy Vs. Time	Output from LOFTRAN
Core Inlet Flow Vs. Time	Output from LOFTRAN
Core Pressure Vs. Time	Output from LOFTRAN

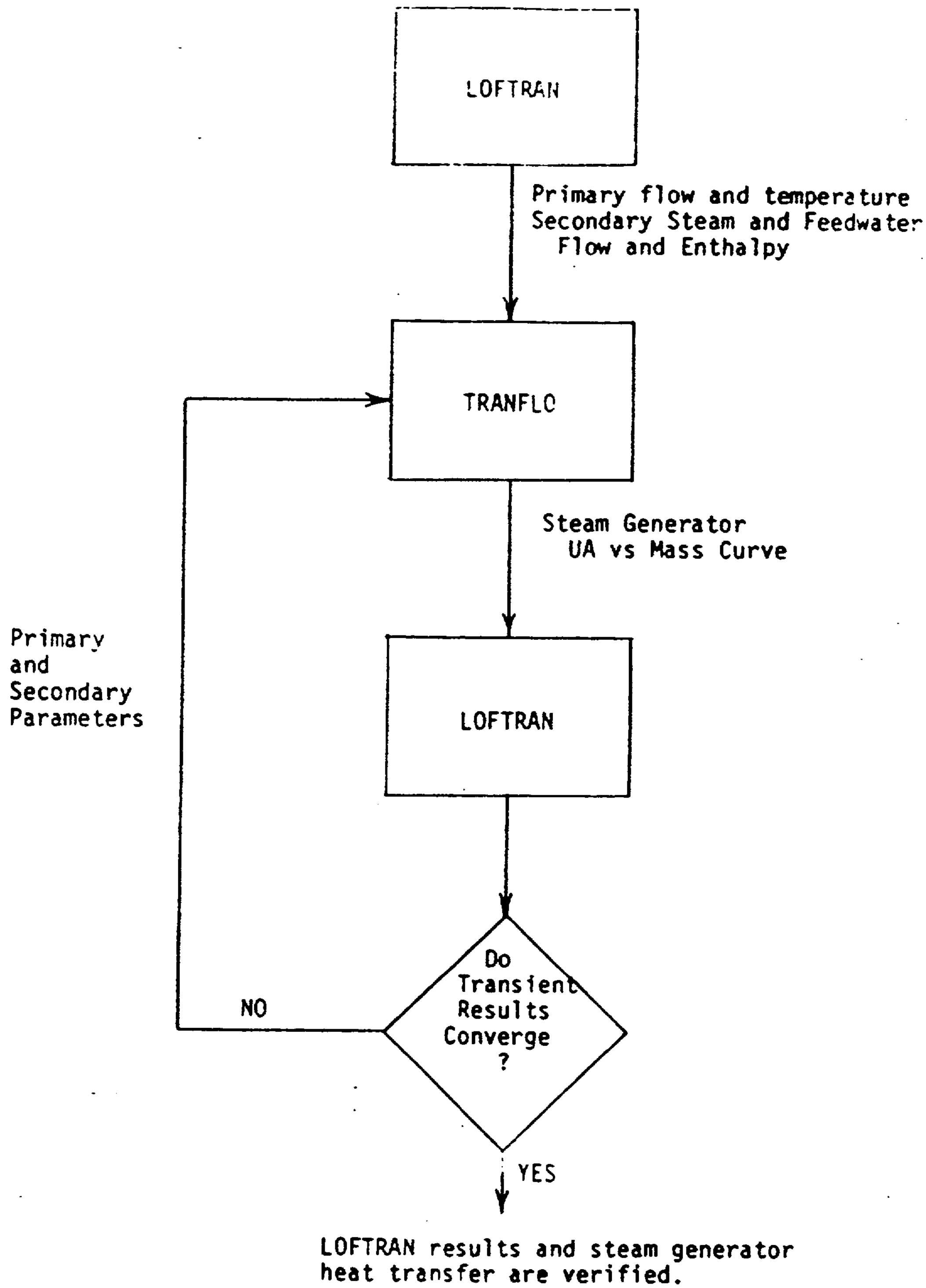
TABLE 4-3

FLUID CONDITIONS	CORRELATION	EQUATION	REFERENCE
Subcooled	Dittus-Boelter	$H_{fc}^{W} = \frac{0.023}{DE} P_r^{.4} R_e^{.8} K_w$	7
	Thom	$Q_{LB} = 193e^{P/630} (T_W - T_{SAT})^2$	8
	Sandberg, et.al.	$H_{FB} = 0.0193 R_e^{.8} P_r^{1.23} \left(\frac{V_w}{V_s} \frac{V_{sc}}{V_{SAT}}\right)^{.068} \frac{K_s}{DE}$	9
	MacBeth	$Q_{crit} = \frac{1.7583 h_{fg}}{DE^{0.1}} \left(\frac{G}{10^6}\right)^{.51} (1 - x)$	10
Saturated	Dougal-Rohsenow.	$H_{FB} = \frac{0.023 K_s}{DE} \left[\frac{DE\,G((1 - x) V_w + xV_s)}{V_s \mu_s}\right]^{.8} P_r^{.4}$	11
	Schrock-Grossman	$H_{FV} = \frac{2.5 H_{fc}^{W}}{\left[\left(\frac{\mu_w}{\mu_s}\right)^{.1} \left(\frac{V_w}{V_s}\right)^{.5} \left(\frac{1}{x} - 1\right)^{.9}\right]^{.75}}$	9
	MacBeth	(Same as Correlation for Subcooled)	12
Superheated	Heineman	$H_{fc}^{*} = \frac{0.0133}{DE} P_r^{.333} R_e^{.84} K_s$	13
Transition Boiling	Westinghouse Transition Boiling	Westinghouse Proprietary	14



Data Flow Between Computer Codes

FIGURE 4-1



Data Flow between LOFTRAN and TRANFLO

FIGURE 4-2

5.0 TRANSIENT ANALYSES AND SENSITIVITY STUDIES

ANSI N.18.2 Condition II transients have been evaluated on several occasions with the assumption that no reactor trip occurred. Detailed analyses have been done and presented to the staff in the past for the limiting ATWS event covering the majority of Westinghouse plants. This report updates the majority of these analyses using the multi-loop version of the LOFTRAN(2) code.

The analyses were performed using composite plant parameters to bound as many Westinghouse plants as possible, rather than using parameters for any specific plant. Sensitivity studies were performed for the limiting cases to demonstrate that the conclusions are valid for all plants covered by the generic approach. These analyses consider 2-, 3-, and 4-loop plant configurations with 51 and 44 series and Model D and F steam generators.

The majority of Westinghouse plants are considered Alternative 3, as discussed in Section 2.0. For this reason, the analyses presented in this report contain the assumptions for Alternate 3 plants, i.e. a moderator temperature coefficient valid for 95% of core life. Limiting transients are presented with a 99% value of the coefficient to show its effect for Alternative 4 plants. Single failures are presented as part of the sensitivity studies.

5.1 LOSS OF EXTERNAL ELECTRICAL LOAD AND/OR TURBINE GENERATOR TRIP WITHOUT REACTOR TRIP

5.1.1 IDENTIFICATION OF CAUSES AND TRANSIENT DESCRIPTION

A major load loss could result either from a loss of external electrical load or from a turbine/generator trip. In either case, unless a loss of ac power to the station auxiliaries also occurs, off-site power would be available for the combined operation of plant components, such as the reactor coolant pumps. In this section, the loss of load accident is analyzed assuming that the control rods fail to drop into the core following a turbine trip from full power, which would produce the maximum possible load loss.

For turbine trips, the reactor normally trips directly (unless below a specific power level that is related to the amount of steam dump capacity available) from a signal derived from the turbine auto-stop oil pressure (Westinghouse Turbine) or from closure of both turbine stop valves. The automatic steam dump system opens valves to pass off the excess generated steam, and therefore, reactor coolant temperatures and pressure do not significantly increase. If the turbine condenser were not available to receive steam through the steam dump system, the excess steam would be dumped into the atmosphere through the steam generator relief and safety valves. In addition, main feedwater flow might be lost if the turbine condenser were not available to run the turbine driven pumps but some feedwater flow would be supplied by the auxiliary feedwater system at a rate sufficient to remove the sensible heat of the fuel and coolant plus the residual heat produced in the reactor.

For a complete loss of external electrical load without subsequent turbine trip, no direct reactor trip signal would be generated. Plants designed with full load rejection capability would continue operation without a reactor trip, since the mismatch between core power and turbine load would be accommodated by sufficient steam dump capacity and primary pressure relief. The Reactor Control System would bring the reactor to a turbine/generator electric load of approximately five percent after a complete loss of external electrical load to match the power requirements of the plant auxiliaries. Plants designed with less than full load rejection capability that undergo a full load rejection might possibly have the reactor trip from the first four reactor protection system signals listed in the following paragraph. Plant startup tests, however, have demonstrated that Westinghouse plants with 40 percent steam dump capacity can generally ride through a complete loss of electric load even under the most adverse operating conditions[15].

If the steam dump valves fail to open following a large loss of load, or if the plant does not have full load rejection capability, the steam generator safety valves may lift since steam generator shell side pressure increases rapidly. If reactor core or primary system safety limits are approached, a reactor trip signal would be generated by the reactor trip signals which are listed below:

- Direct reactor trip on turbine trip.

- High pressurizer pressure reactor trip.

- High pressurizer water level reactor trip.

- Overtemperature ΔT reactor trip.

- Low feedwater flow reactor trip.

- Low-low steam generator water level reactor trip.

The most severe plant conditions that could result from a loss of load occur following a turbine trip from full power when the turbine trip is caused by a loss of condenser vacuum. Since the main feedwater pumps may be turbine driven with steam exhaust to the main condenser, loss of feedwater may also result from a loss of condenser vacuum. For this reason, the low feedwater flow reactor trip and the low-low steam generator water level trip are included in the above listing.

The pressurizer safety valves and steam generator safety valves are sized to protect the Reactor Coolant System and steam generator against overpressure for all load losses without assuming the operation of the steam dump system, pressurizer spray, pressurizer power-operated relief valves, steam generator power-operated relief valves, automatic rod control, or direct reactor trip on turbine trip. That is, the steam relief capacity of the pressurizer safety valves is selected to match the maximum pressurizer insurge following a turbine trip without credit for the items mentioned above. The steam generator safety valve relief capacity is sized to remove the steam flow at the Engineered Safeguards Design rating ($\sim$ 105 percent of the steam flow at rated power) from the steam generator without exceeding 110 percent of the steam system design pressure. The pressurizer safety valve capacity is sized for a complete loss of heat sink with the plant initially operating at the maximum calculated turbine load and with operation of the steam generator safety valves. The pressurizer safety valves are then able to maintain the Reactor Coolant System pressure to within 110 percent of the Reactor Coolant System design pressure without direct or immediate reactor trip action.

5.1.2 ANALYSIS OF EFFECTS AND CONSEQUENCES

Plant behavior is evaluated for a turbine trip and loss of main feedwater occurring from full power with the assumption that the control rods fail to drop into the core following generation of a reactor trip signal. The evaluation shows the effectiveness of Reactor Coolant System pressure-relief devices and the extent of any approach to core safety limits.

The loss of load transient is analyzed using the LOFTRAN digital computer code. The program computes pertinent plant variables including temperatures, pressures and power level.

The following assumptions are made in the analysis:

- Initial normal full power operation early in core life. Since the negative temperature coefficient of reactivity reduces core power as the coolant temperature rises, and the temperature coefficient becomes more negative with core life, the ATWS loss of load is less severe later in core life.

- Normal operation of the following control systems:

1. Pressurizer pressure control, including heaters, spray, and both the power-operated and the spring-loaded relief valves.

2. Turbine governor valves in impulse pressure control prior to trip, and valve closure on turbine trip.

- Loss of condenser vacuum at t = 0.

- No credit for automatic reactor trip.

- No credit for automatic control rod insertion as reactor coolant temperature rises.

- Main feedwater flow falls to zero in the first four seconds of the transient, with no main feed after that time.

- Auxiliary feedwater flow begins at 60 seconds, at a rate of 1760 gpm. The initiation signal would come from AMSAC.

- Auxiliary feedwater is injected into the feedwater pipe at a temperature of 130°F, upstream of the steam generator, such that the cooler water enters the steam generator after the volume in Table 3-1 is purged of main feedwater (440°F).

- Primary to secondary heat transfer area is reduced as the steam generator shell-side water inventory drops below the value necessary to wet the tubes.

The auxiliary feedwater initiation requirements described above would be provided by AMSAC (Alternative Mitigating Systems Actuation Circuitry). AMSAC is a diverse set of circuitry designed to initiate any mitigation systems required for an ATWS condition that would normally be provided by the reactor scram system. This assumes a complete common-cause failure of the total reactor scram system. The signal to initiate auxiliary feedwater from the AMSAC circuitry would provide the function in much less time than the 60 seconds assumed in this analysis.

5.1.3 RESULTS

5.1.3.1 51 Series Steam Generator

Figures 5.1-1 through 5.1-10 show the plant transient response for a loss of load without reactor trip for a 4-loop Alternative 3 plant with a 51 series steam generator. Sequence of events for this transient are shown in Table 5.1-1. The first peak in pressurizer pressure occurs when the steam generator safety valves lift, and the second, higher peak (maximum system pressure of 2974 psia) occurs after the pressurizer is filled with water due to a coolant volume surge resulting from a rapid reduction of steam generator heat transfer. Nuclear power decreases to a value of 68 percent due to negative reactivity feedback caused by moderator (coolant) heating. Further coolant heatup, caused by loss of steam generator heat transfer, decreases nuclear power further, starting at about 110 seconds.

The DNB ratio does not drop below its initial value during the transient.

At ten minutes into the transient, conditions are stabilized, with auxiliary feedwater providing heat removal capability and with an intact Reactor Coolant System and core. Thus, the operator could begin shutdown operations through rod insertion, actuation of the safety injection system, or through the BORATE or EMERGENCY BORATE modes of the Chemical and Volume Control System.

Transient results for 3-loop and 2-loop plants with 51 Series steam generators are similar to those presented for the 4-loop case. A peak reactor coolant system pressure of 2861 psia results for a 3-loop plant, and a peak pressure of 2753 psia results for a 2-loop plant configuration.

Since the loss of load transient is a limiting ATWS transient with respect to peak pressure, an analysis was made for the Alternative 4 assumption of a moderator temperature coefficient that is valid for over 99% of core life. The limiting plant configuration, 4-loop, was used for this analysis.

The transient results for a 4-loop Alternative 4 plant with a 51 Series steam generator are shown in figures 5.1-11 through 5.1-20. The results are similar to but more severe than the Alternative 3 results shown earlier. The peak reactor coolant system pressure in this case is 3084 psia.

5.1.3.2 Model D Steam Generator

The loss of load ATWS transient results for a Model D steam generator are similar to and less severe than the 51 Series steam generator. A 4-loop plant configuration with a Model D steam generator yields a peak reactor coolant system pressure of 2780 psia. The 3-loop Model D configuration results in a peak system pressures of 2785 psia.

5.1.3.3 Model F Steam Generator

The results of a loss of load ATWS transient for the Model F plant configuration follow those of the 51 Series steam generator closely.

The base 4-loop plant with a Model F produces a peak system pressure of 2902 psia for the loss of load ATWS. The peak system pressure for a 3-loop plant configuration is 2786 psia.

5.1.3.4 44 Series Steam Generator

The transient results of a loss of load ATWS for a 44 Series steam generator are similar to the 51 Series results. The peak reactor coolant system pressure for the 4-loop configuration is 2979 psia. A 3-loop plant produces a peak system pressure of 2839, and a 2-loop, 2753 psia.

5.1.4 SENSITIVITY STUDIES

The loss of load transient without reactor trip is analyzed with changes in certain assumptions and initial conditions. The evaluation is done to determine the effects of the reactor coolant system pressure due to each parameter change. The sensitivity studies were conducted on the base 4-loop plant with a Model 51 steam generator. The sensitivity study effects are presented as variations on this reference case, and are summarized in Table 5.1-2.

5.1.4.1 Effect of One Auxiliary Feedwater Pump Failing to Start (Single Failure)

The single failure effect of one auxiliary feedwater pump failing to start was studied. The pump chosen for the failure is that which delivers the greatest flow, typically the steam-driven pump. Failure of this pump effectively reduces auxiliary feedwater flow rate by one-half. The peak reactor coolant system pressure was increased by 64 psi due to the reduced flow rate.

5.1.4.2 Effect of One PORV Failing to Open (Single Failure)

The effect of one power-operated relief valve failing to open upon demand was determined. The failure produced a net increase of 166 psi over the 4-loop, 51 steam generator reference case.

5.1.4.3 Effect of Variations in Pressurizer Level

The initial value of the pressurizer level was varied by +10 and -10 percent to determine the effect on the peak reactor coolant system pressure. An increase in the level of 10 percent resulted in an increased peak pressure of 5 psi. Reducing the level by 10 percent resulted in a reduction of 17 psi on the peak pressure.

5.1.4.4 Effect of Variations in the Steam Generator Water Inventory

The steam generator initial water mass was varied to determine peak pressure effects. An increase of 10 percent in initial water mass yielded no difference in the peak pressure. Reducing the initial mass by 10 percent resulted in an increase of only 2 psi.

5.1.4.5 Effect of Variations in the Main Feedwater Enthalpy

Variations in the main feedwater enthalpy resulted in small changes in the peak system pressure. An increase of 10 percent in the enthalpy yielded an increase of 10 psi over the reference case. Decreasing the enthalpy by 10 percent decreased the peak pressure by 9 psi.

5.1.4.6 Effect of Variations in the Reactor Coolant System Volume

The reactor coolant system volume was increased by 10 percent i.e. each node volume was increased by 10 percent except for the pressurizer, resulting in an increase in the peak system pressure of 42 psi. A decrease of 10 percent in the reactor coolant system volume resulted in a decrease in the peak pressure of 44 psi.

5.1.4.7 Effect of Variations in the Auxiliary Feedwater Flow Rate

Increasing the auxiliary feedwater flow rate by 10 percent results in a decrease of 11 psi in the peak reactor coolant system pressure when compared to the reference case. Decreasing the flow rate by 10 percent increases the peak system pressure by 12 psi.

5.1.4.8 Effect of Variations in the Fuel UA

Variations in the fuel UA produce small effects on the peak reactor coolant system pressure. Increasing the fuel UA by 10 percent results in a decrease of 6 psi in the peak pressure. A 10 percent decrease in the fuel UA increased the peak pressure by 8 psi.

5.1.4.9 Effect of the Pressurizer Spray

Since the loss of load analysis assumes that the pressurizer spray is inoperable, the effect of proper operation was studied. Assuming the pressurizer spray system operates, the peak reactor coolant system pressure is decreased by 11 psi.

5.1.4.10 Effect of Variations in Reactor Power

The initial reactor power was increased by 2 percent, resulting in a peak pressure increase of 44 psi. Decreasing the initial reactor power resulted in a decrease of 41 psi in the peak pressure.

5.1.4.11 Effect of Auxiliary Feedwater Initiation Delay

A delay of 60 seconds over the reference case for the start of auxiliary feedwater was studied. This case assumes the auxiliary feedwater pumps start 120 seconds into the transient. The peak reactor coolant system pressure is increased by 134 psi due to the delay.

5.1.4.12 Effect of Variation in Steam Generator Design Pressure

The 4-loop, 51 Series reference plant is analyzed utilizing a steam generator design pressure of 1200 psia. A sensitivity study on the reference case with a design pressure of 1100 psia resulted in an increase in peak pressure of 151 psi.

5.1.5 CONCLUSIONS

During a loss of load with failure of rod insertion after a reactor trip signal generation, core safety limits are not exceeded since the DNB ratio does not go below its initial value and the peak reactor coolant pressure is limited to 2974 psia for the 4-loop 51 Series reference case. Further, plant conditions are stabilized at 10 minutes such that the operator can begin shutdown operations.

TABLE 5.1-1

SEQUENCE OF EVENTS FOR LOSS OF LOAD WITHOUT A REACTOR TRIP FOR THE REFERENCE CASE*

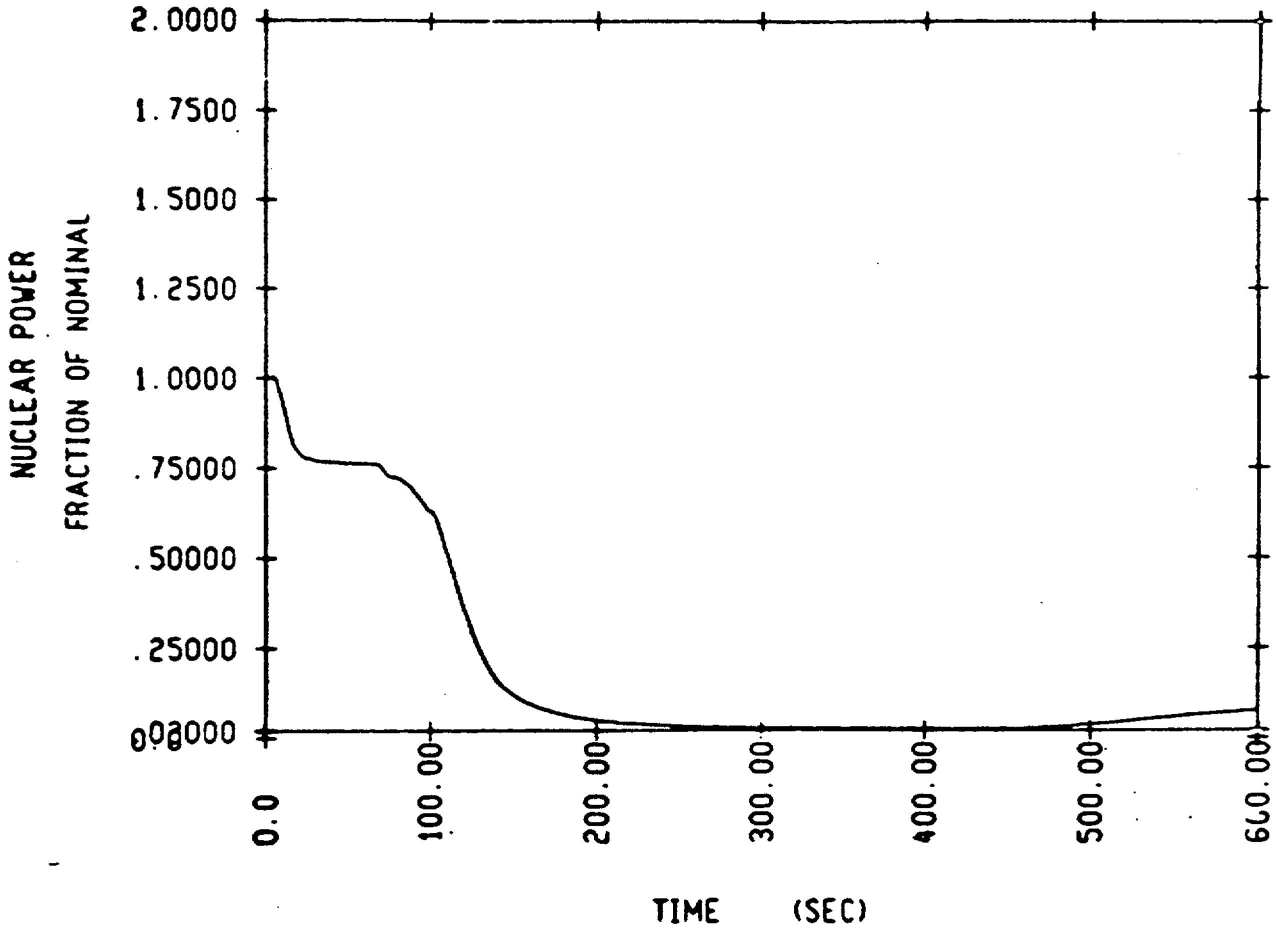
Event	Time (seconds)
Turbine trips	
Reactor trip signal generated on turbine trip	0
Pressurizer relief valves lift	5
High pressurizer pressure reactor trip setpoint reached	6.4
Overtemperature ΔT reactor trip setpoint reached	8.4
Steam generator safety valves lift	11
Auxiliary feed pumps begin delivering flow	60
Pressurizer safety valves lift and pressurizer fills with water	99
Maximum reactor coolant pressure (2974 psia) reached	120

* Reference case: 4-loop plant with a 51 Series steam generator, Alternative 3.

TABLE 5.1-2

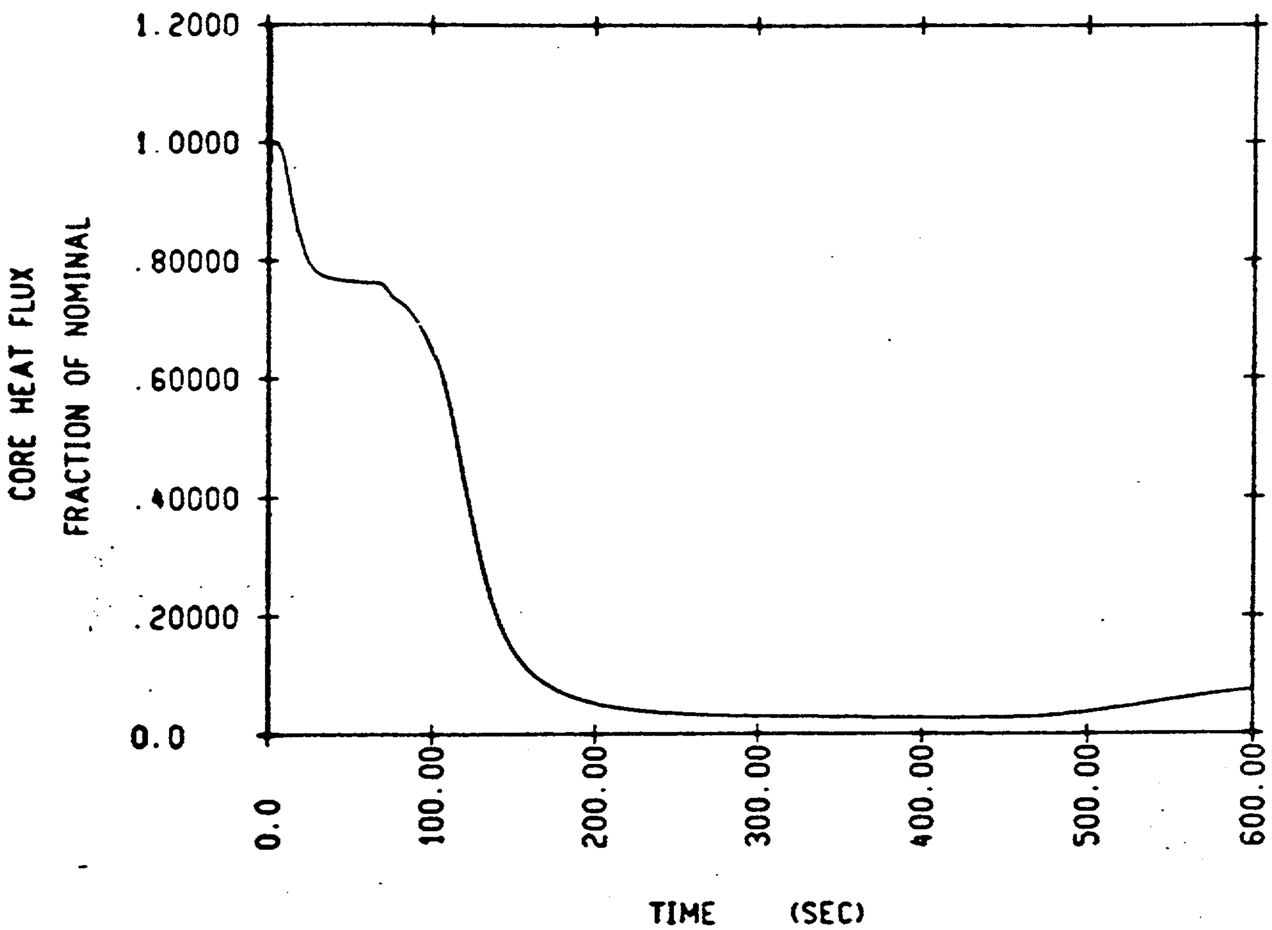
SUMMARY OF RESULTS FOR LOSS OF LOAD WITHOUT A REACTOR TRIP

Case	Change Relative to Reference Case Maximum Reactor Coolant System Pressure (psia)
Reference Case	2974.
One Half Auxiliary Feedwater Flow	+64
One PORV Fails to Open	+166
Pressurizer Water Level +10%	+5
Pressurizer Water Level -10%	-17
Steam Generator Water Mass +10%	+0
Steam Generator Water Mass -10%	+2
Main Feedwater Enthalpy +10%	+10
Main Feedwater Enthalpy -10%	-9
RCS Volume +10%	+42
RCS Volume -10%	-44
Auxiliary Feedwater Flow +10%	-11
Auxiliary Feedwater Flow -10%	+12
Fuel UA +10%	-6
Fuel UA -10%	+8
Pressurizer Spray On	-11
Reactor Power +2%	+44
Reactor Power -2%	-41
60 Second Auxiliary Feedwater Delay	+134
1100 psia Steam Generator Design Pressure	+151



LOSS OF LOAD ATWS
REFERENCE CASE
95% MTC

Figure 5.1-1



LOSS OF LOAD ATWS
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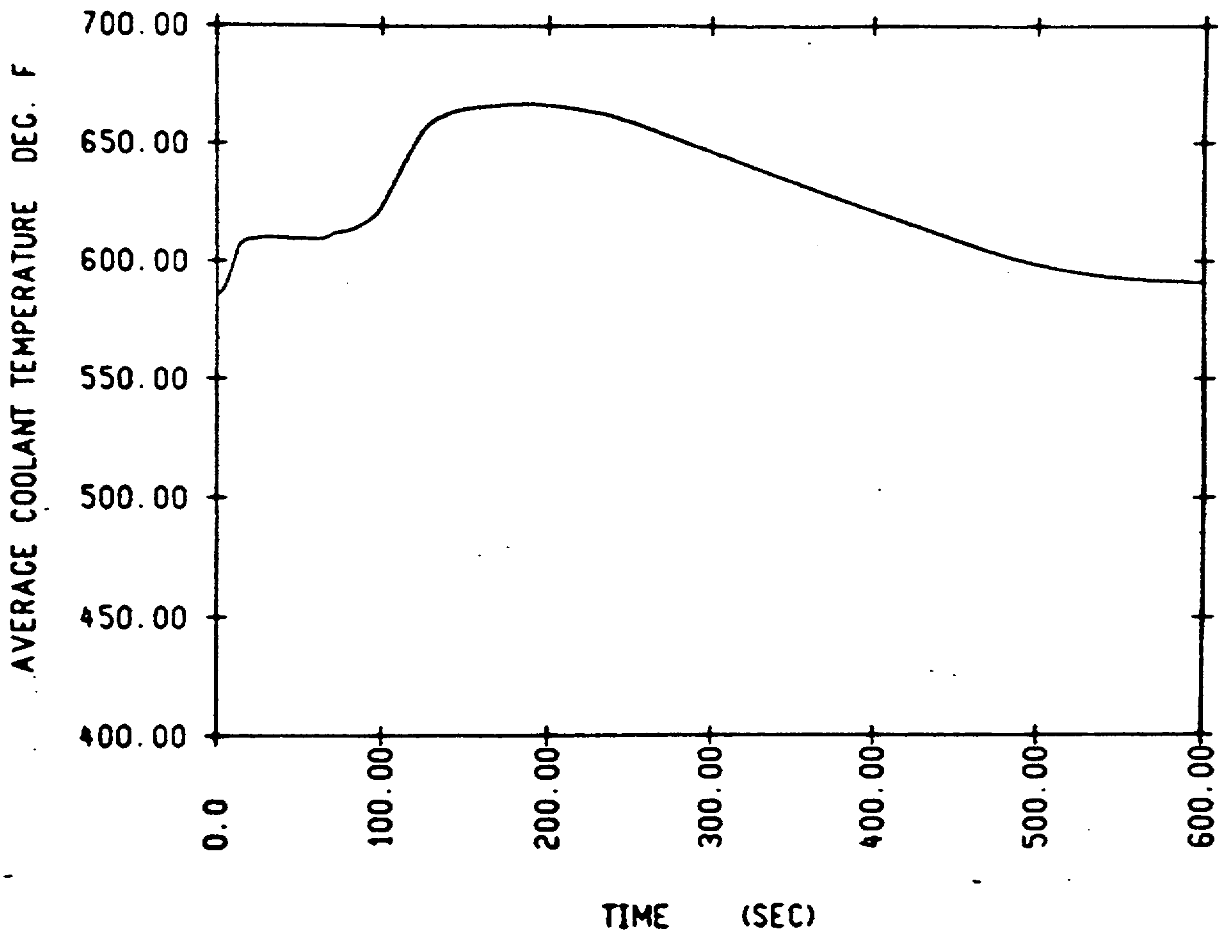
95% MTC

Figure 5.1-2



LOSS OF LOAD ATWS
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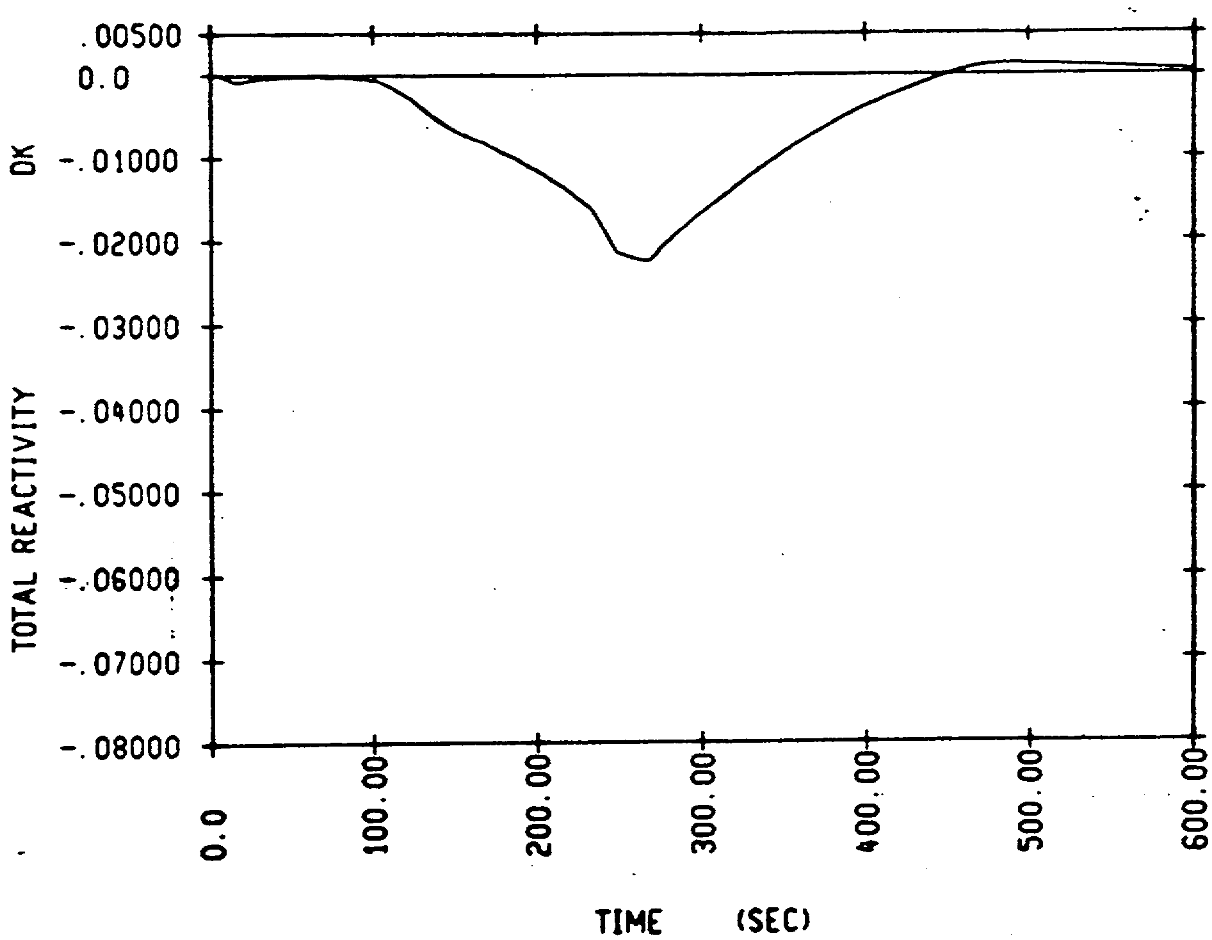
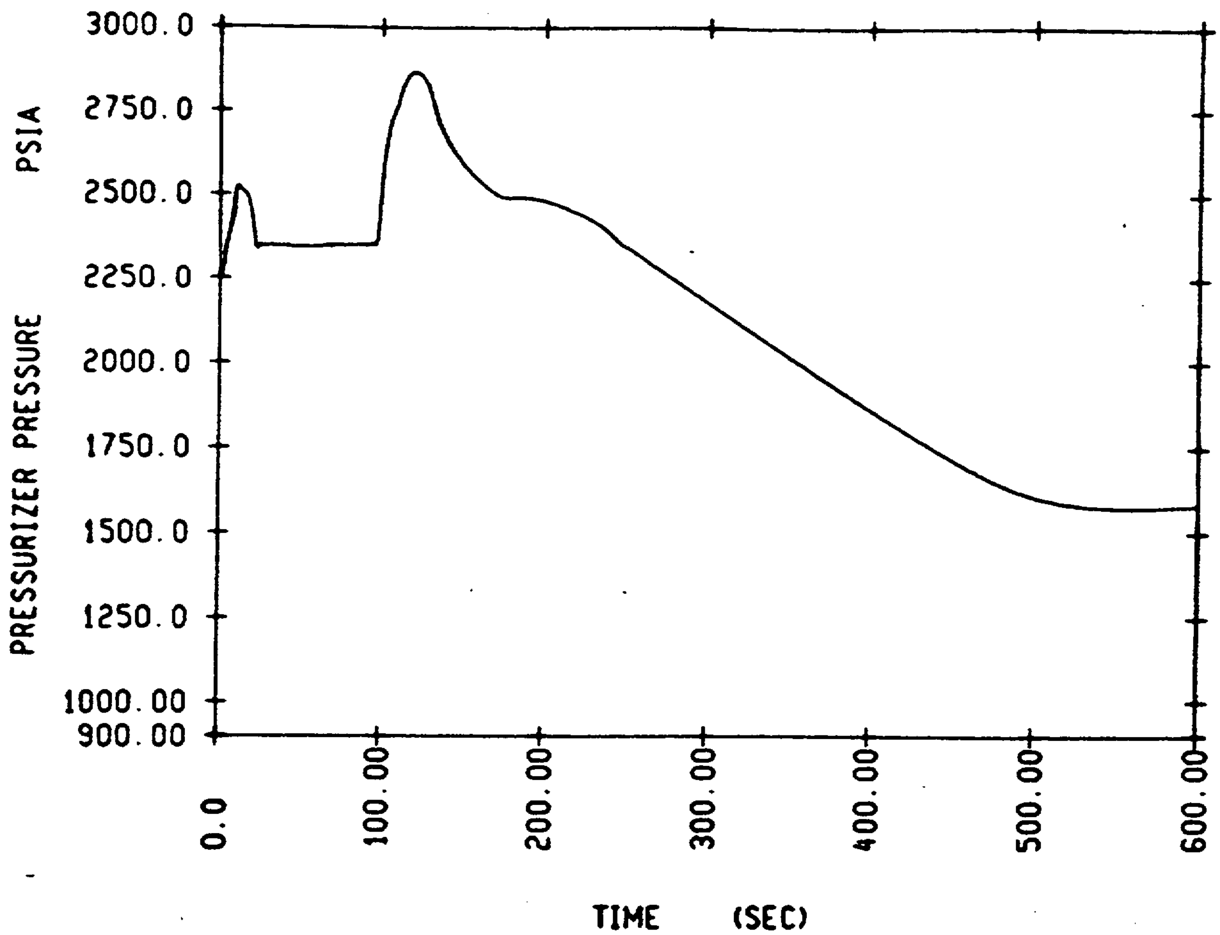
95% MTC

Figure 5.1-3



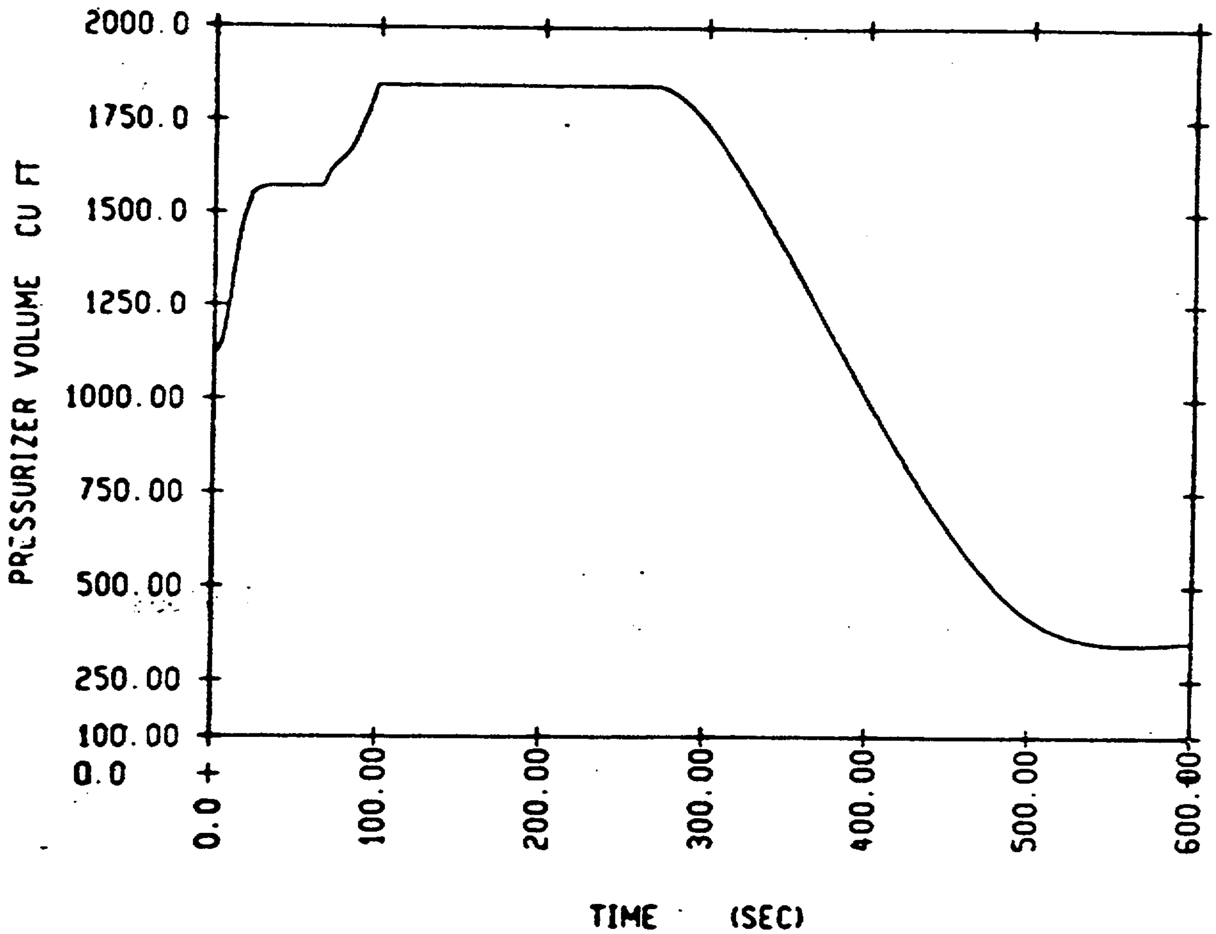
LOSS OF LOAD ATWS
REFERENCE CASE
95% MTC

Figure 5.1-4



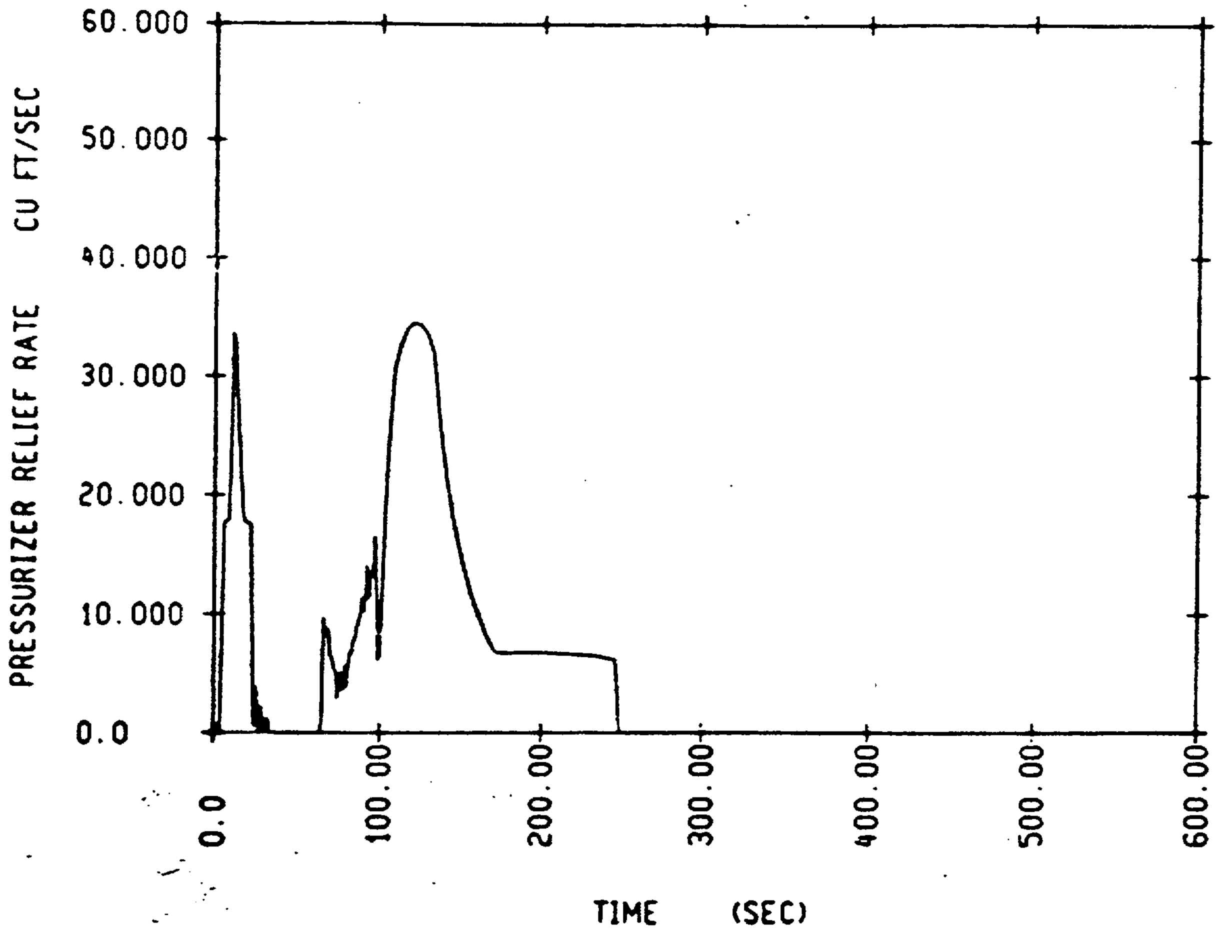
LOSS OF LOAD ATWS
REFERENCE CASE
95% MTC

Figure 5.1-5



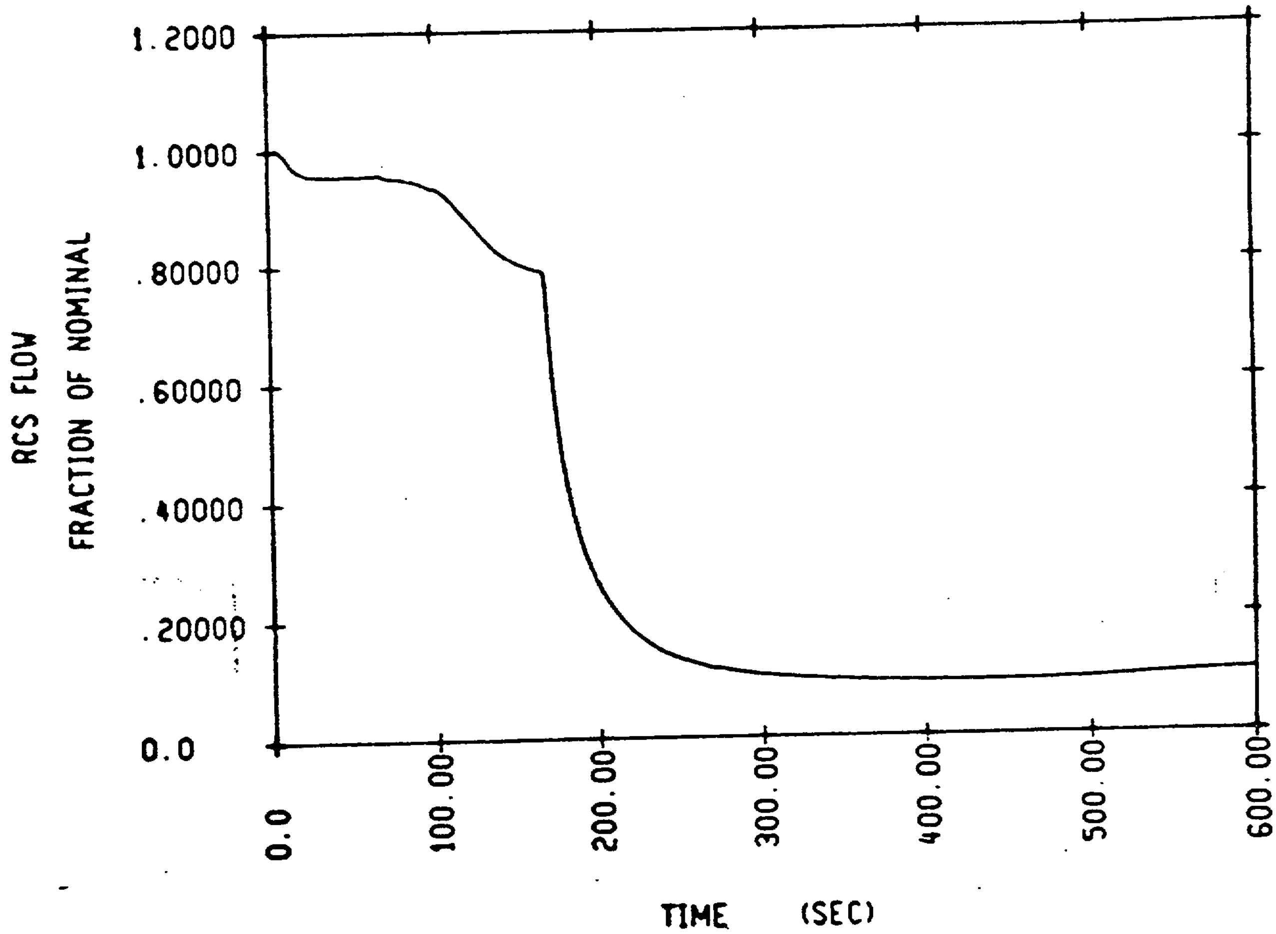
LOSS OF LOAD ATWS
REFERENCE CASE
95% MTC

Figure 5.1-6



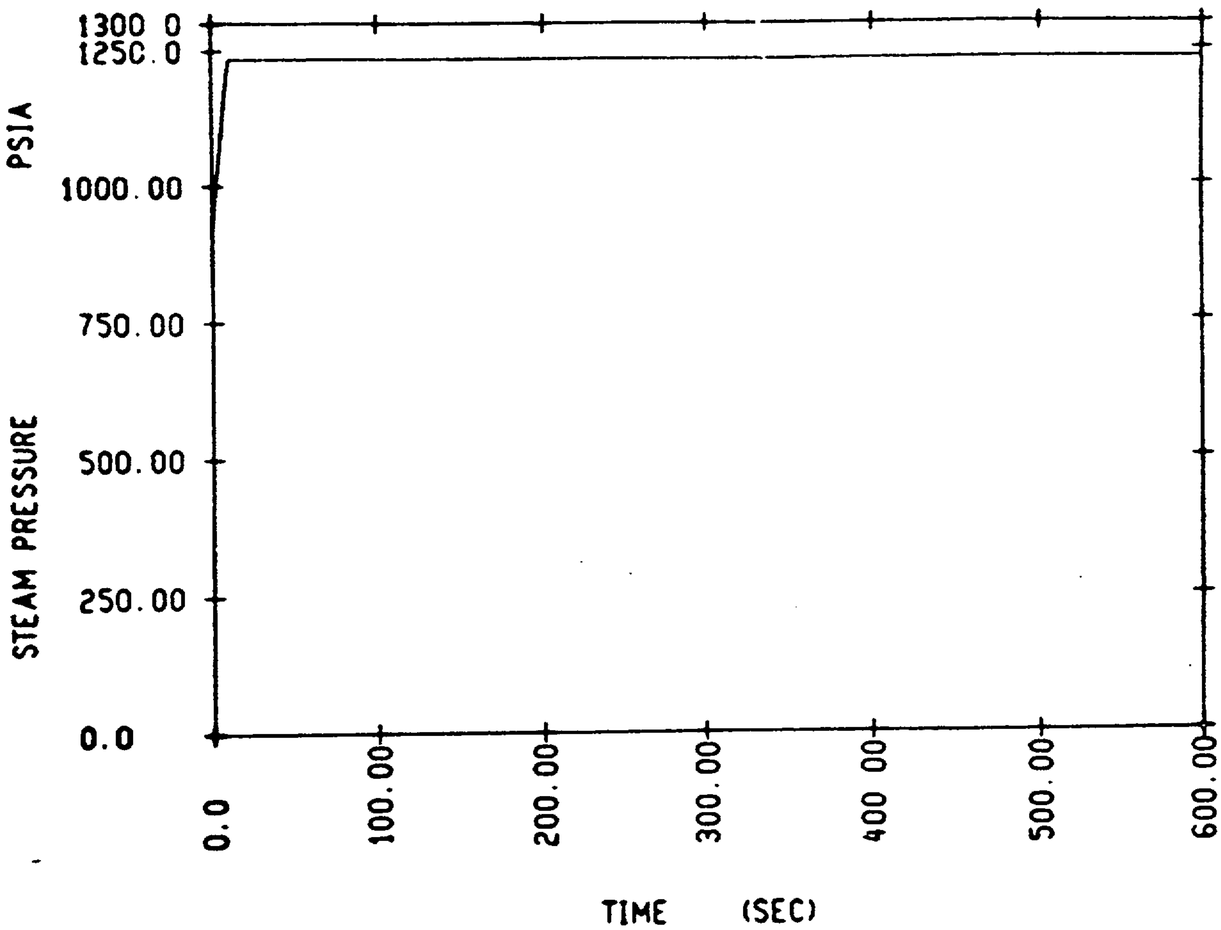
LOSS OF LOAD ATWS
REFERENCE CASE
95% MTC

Figure 5.1-7



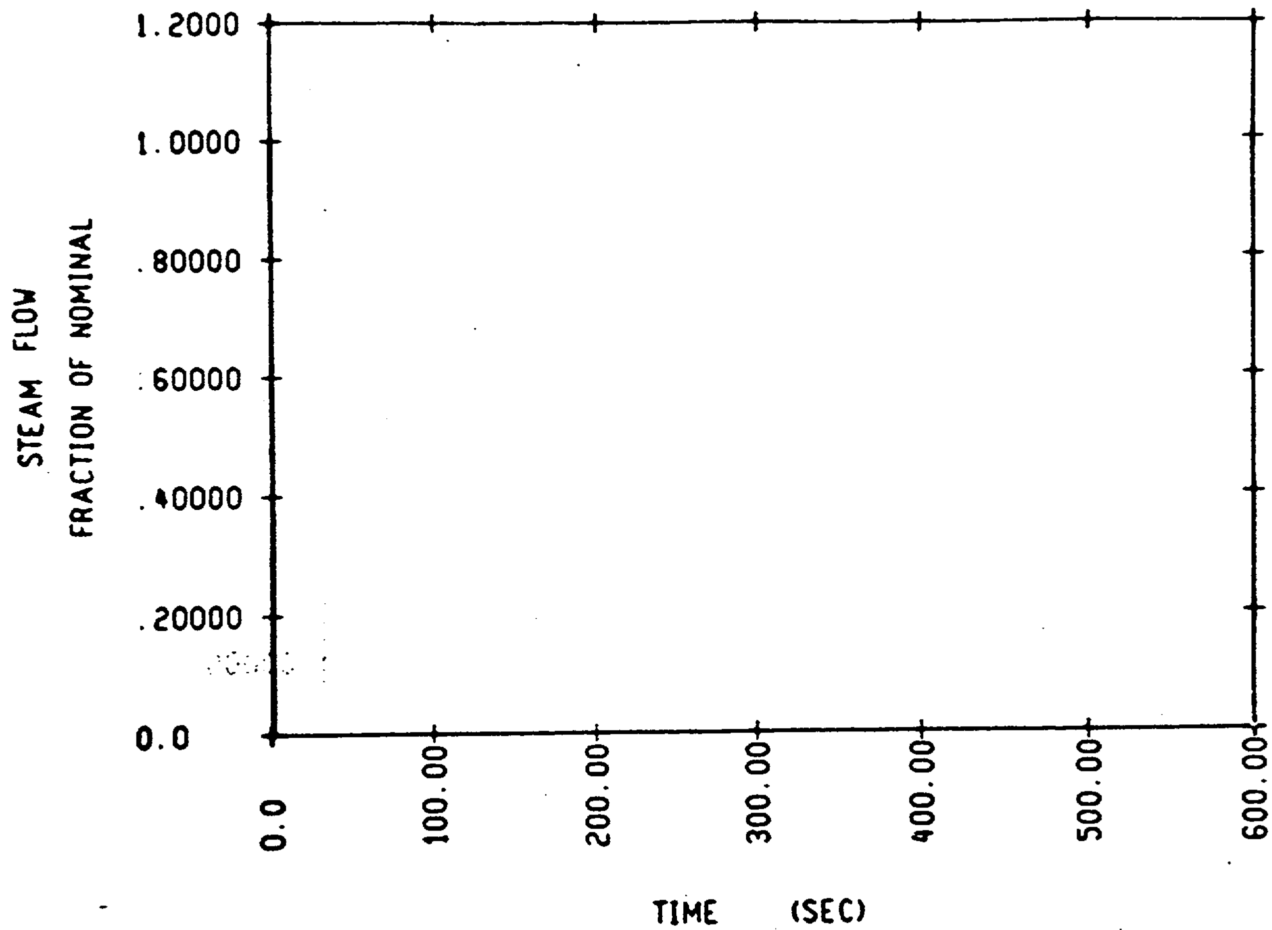
LOSS OF LOAD ATWS
REFERENCE CASE
95% MTC

Figure 5.1-8



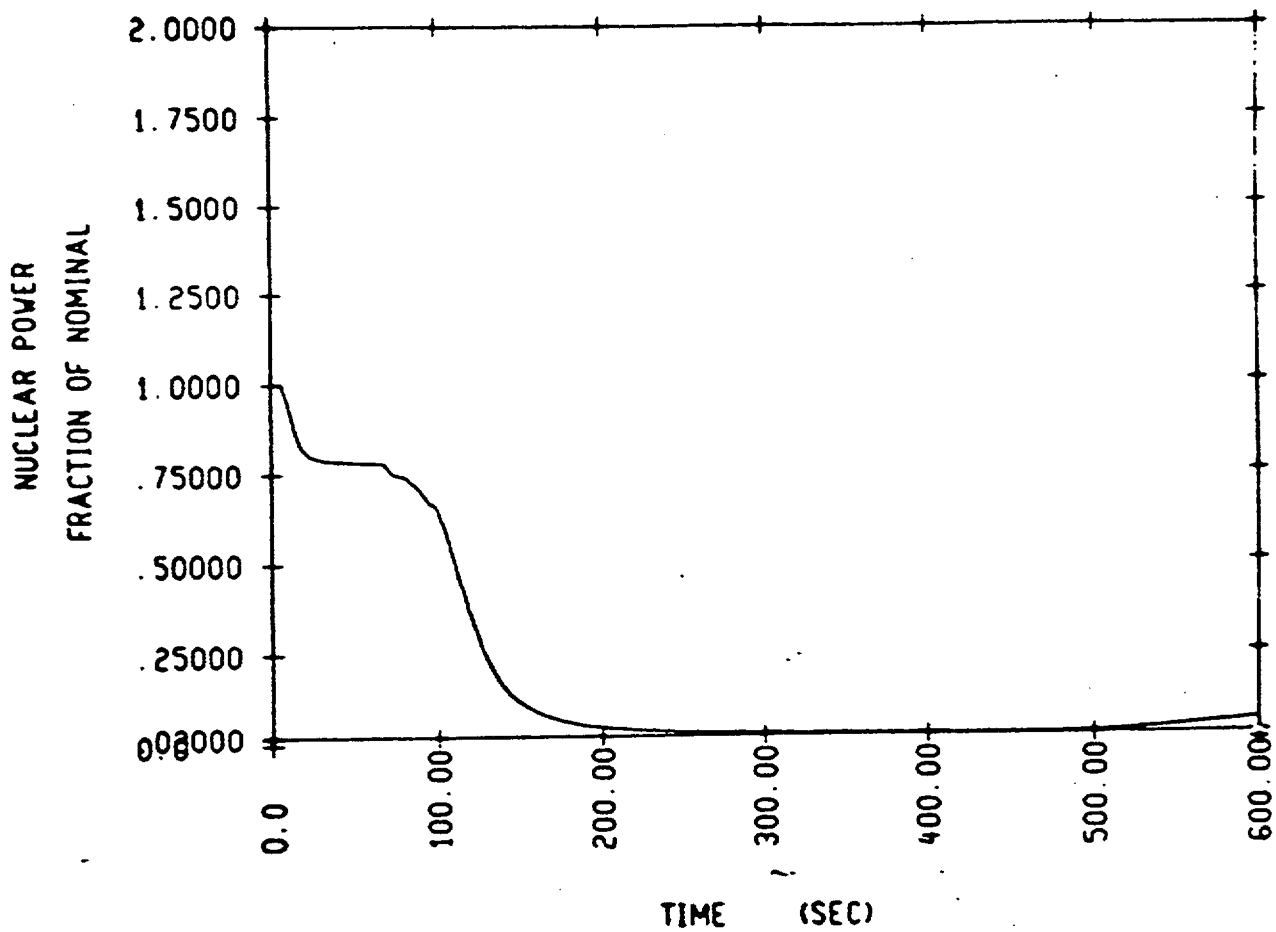
LOSS OF LOAD ATWS
REFERENCE CASE
95% MTC

Figure 5.1-9



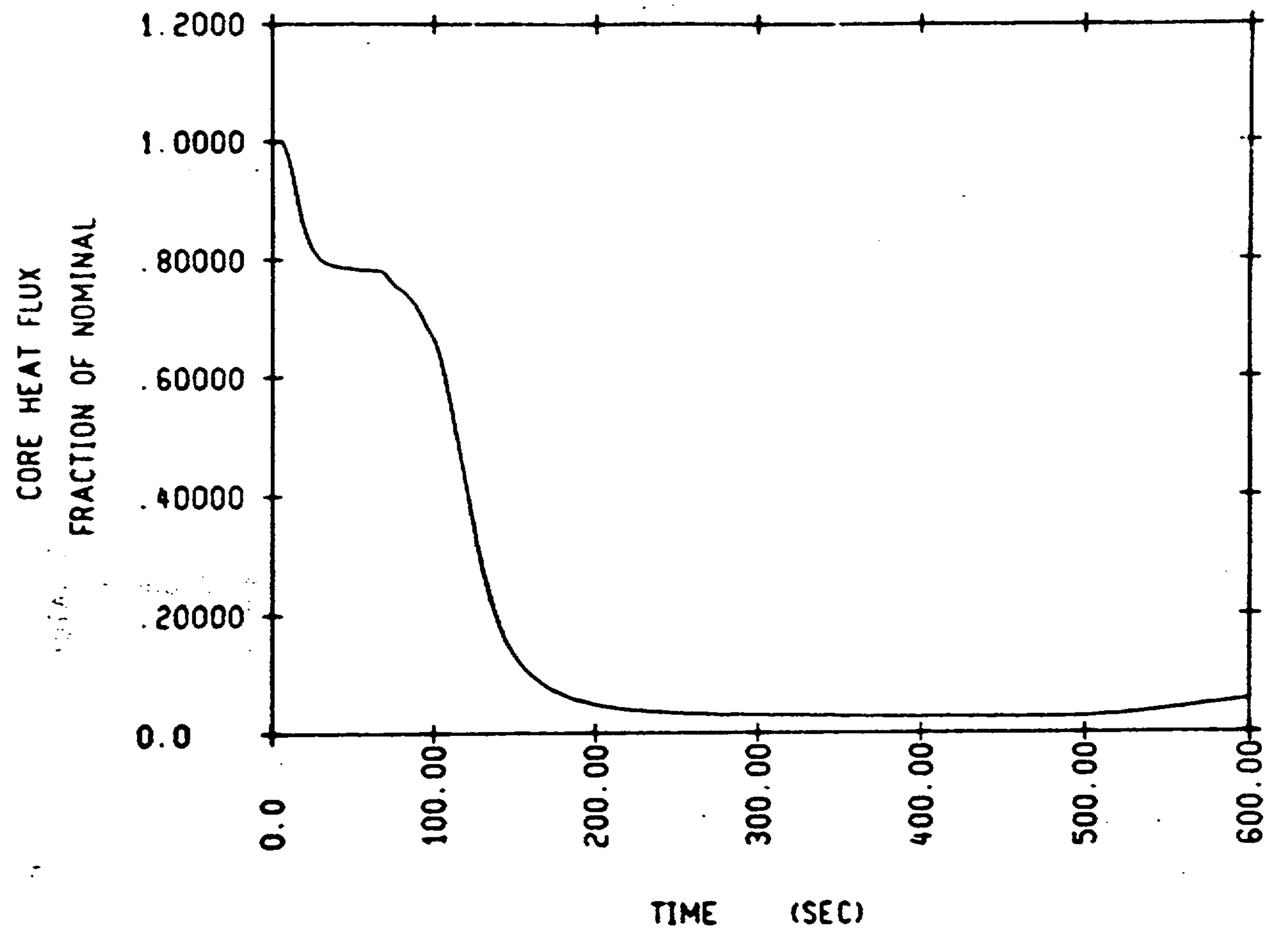
LOSS OF LOAD ATWS
REFERENCE CASE
95% MTC

Figure 5.1-10



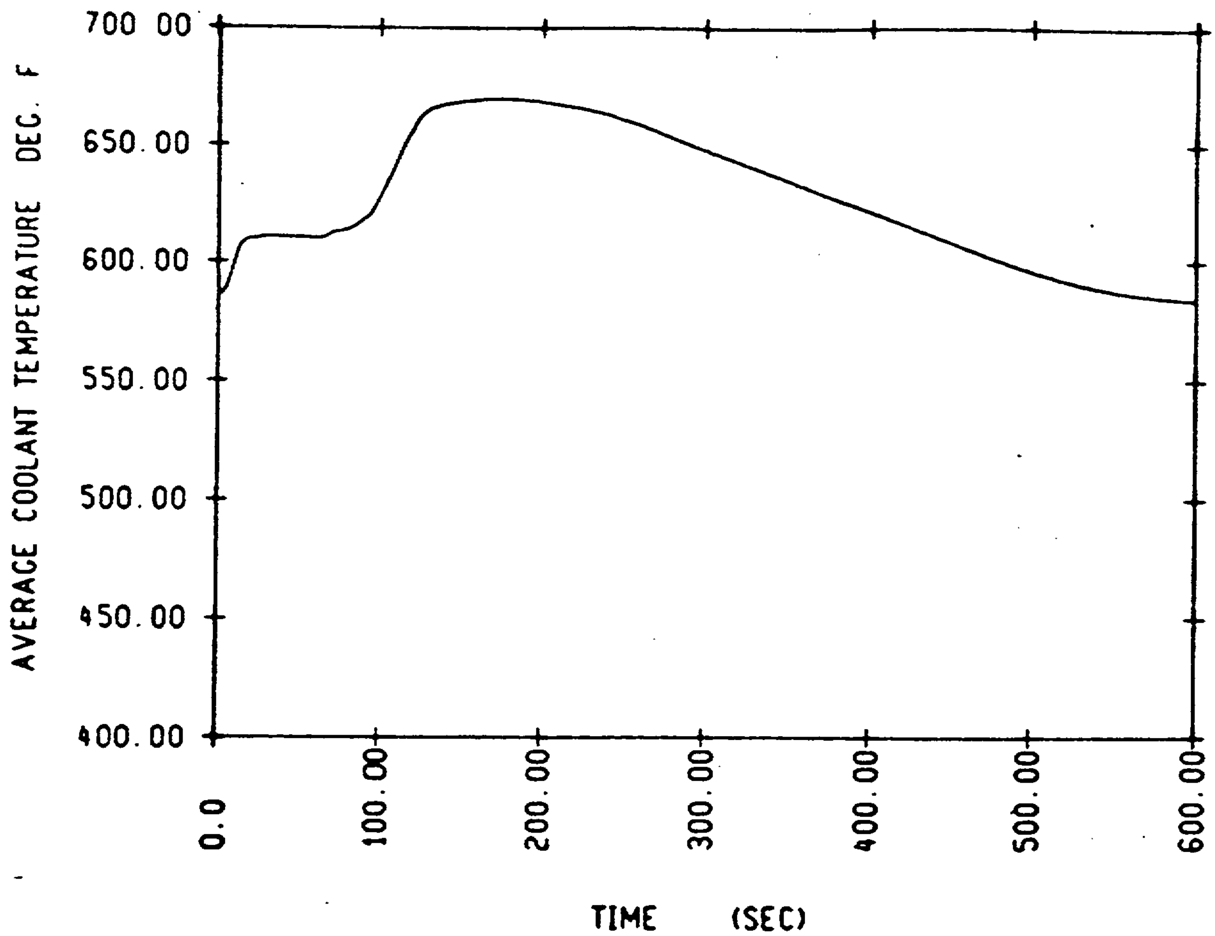
LOSS OF LOAD ATWS
REFERENCE CASE
99% MTC

Figure 5.1-11



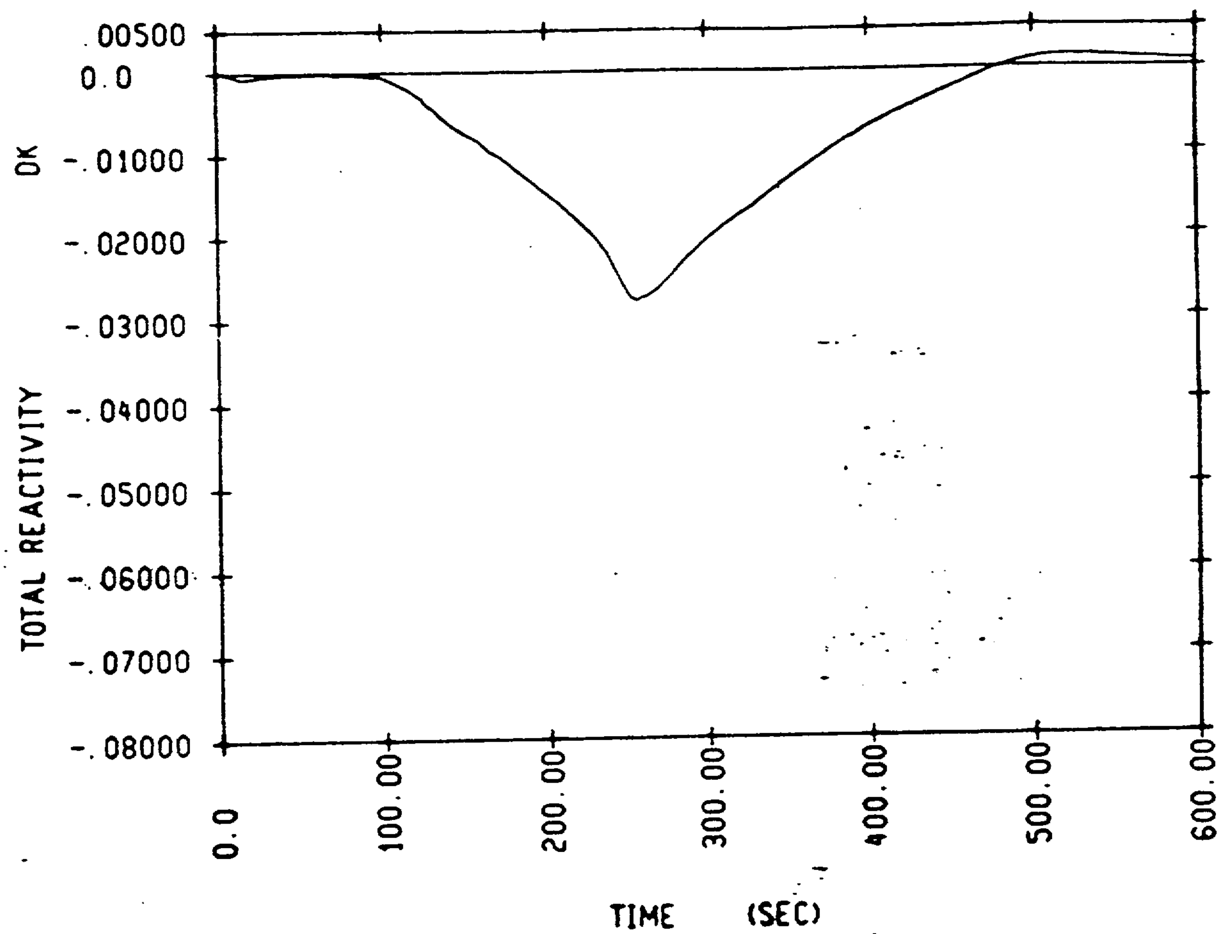
LOSS OF LOAD ATWS
REFERENCE CASE
99% MTC

Figure 5.1-12



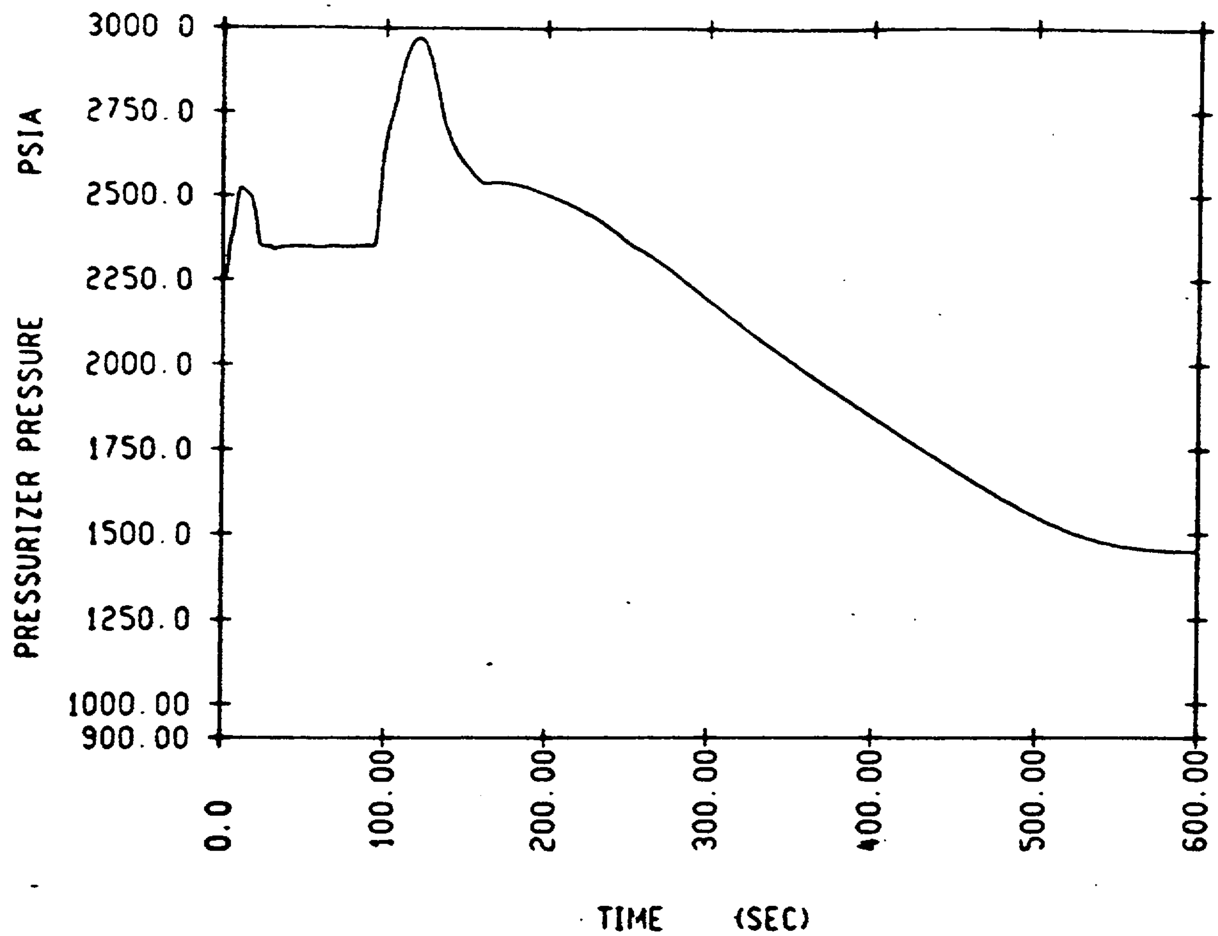
LOSS OF LOAD ATWS
REFERENCE CASE
99% MTC

Figure 5.1-13



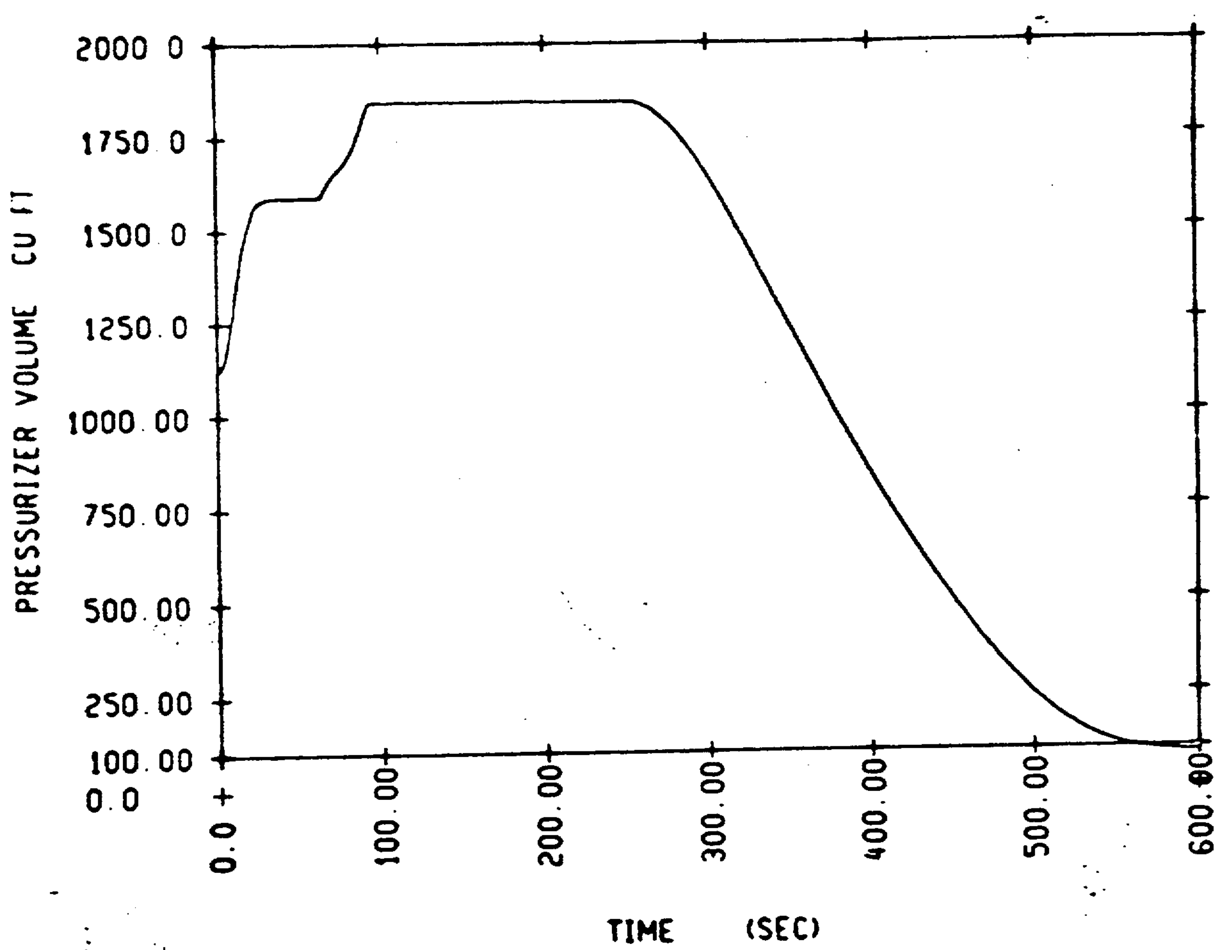
LOSS OF LOAD ATWS
REFERENCE CASE
99% MTC

Figure 5.1-14



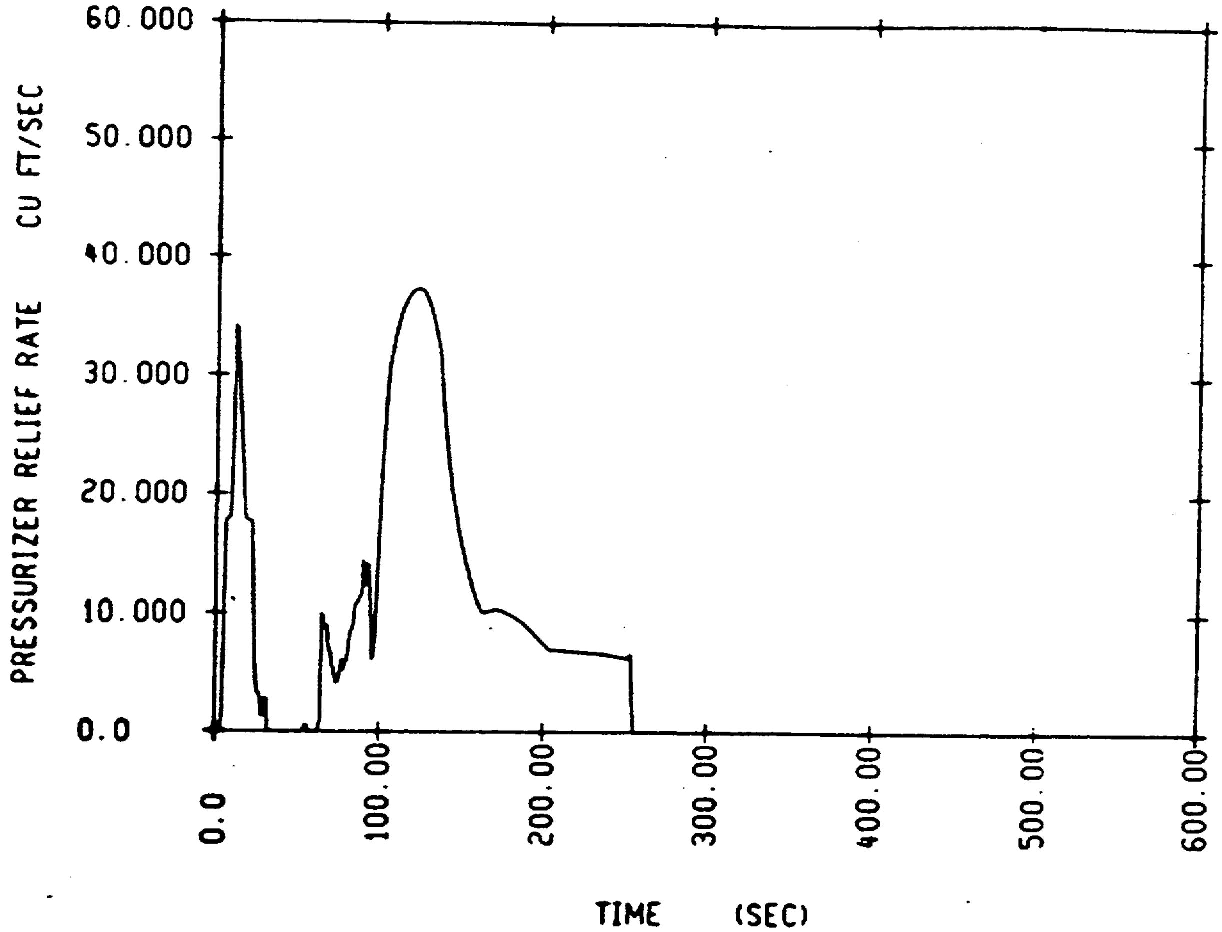
LOSS OF LOAD ATWS
REFERENCE CASE
99% MTC

Figure 5.1-15



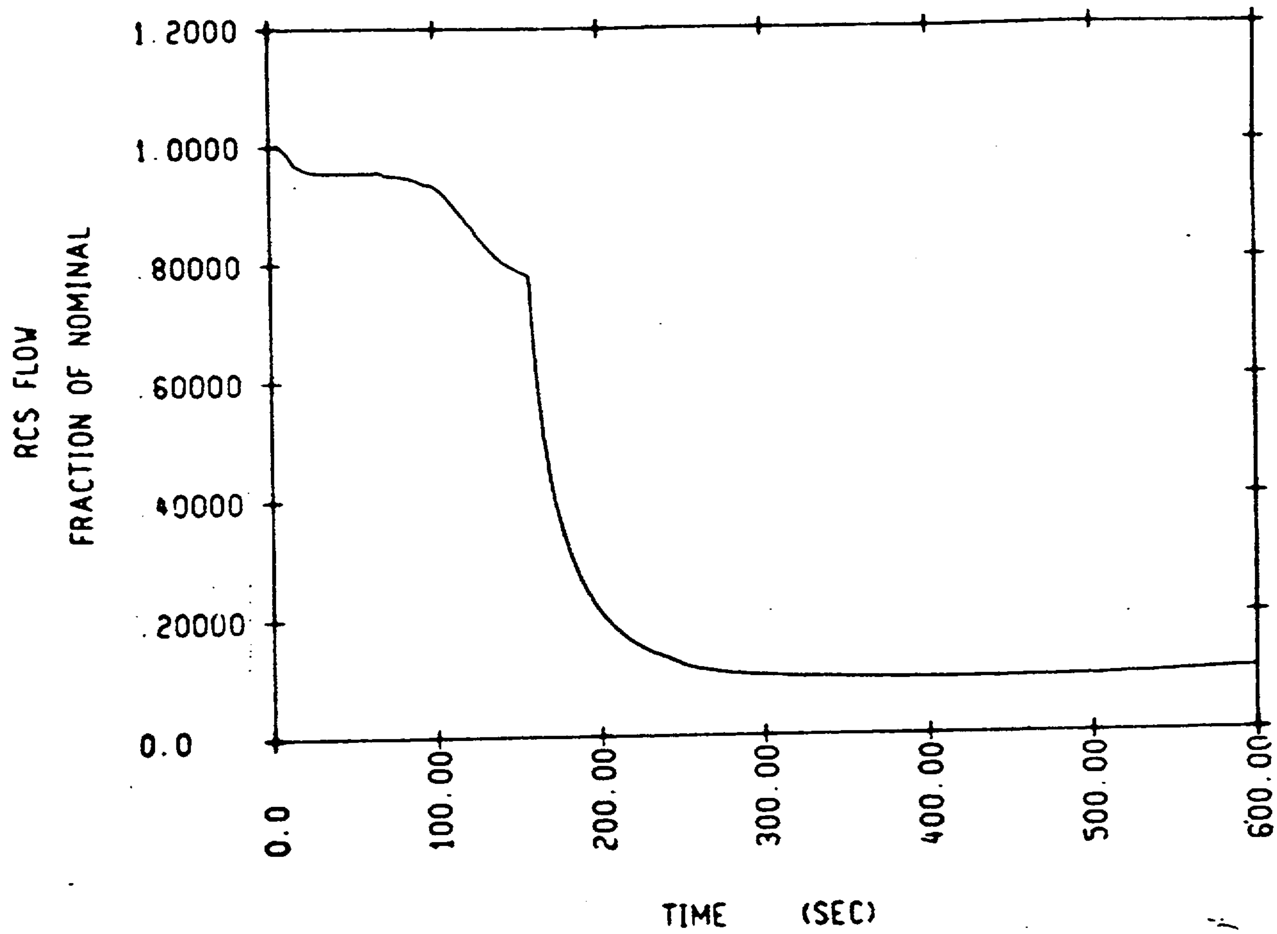
LOSS OF LOAD ATWS
REFERENCE CASE
99% MTC

Figure 5.1-16



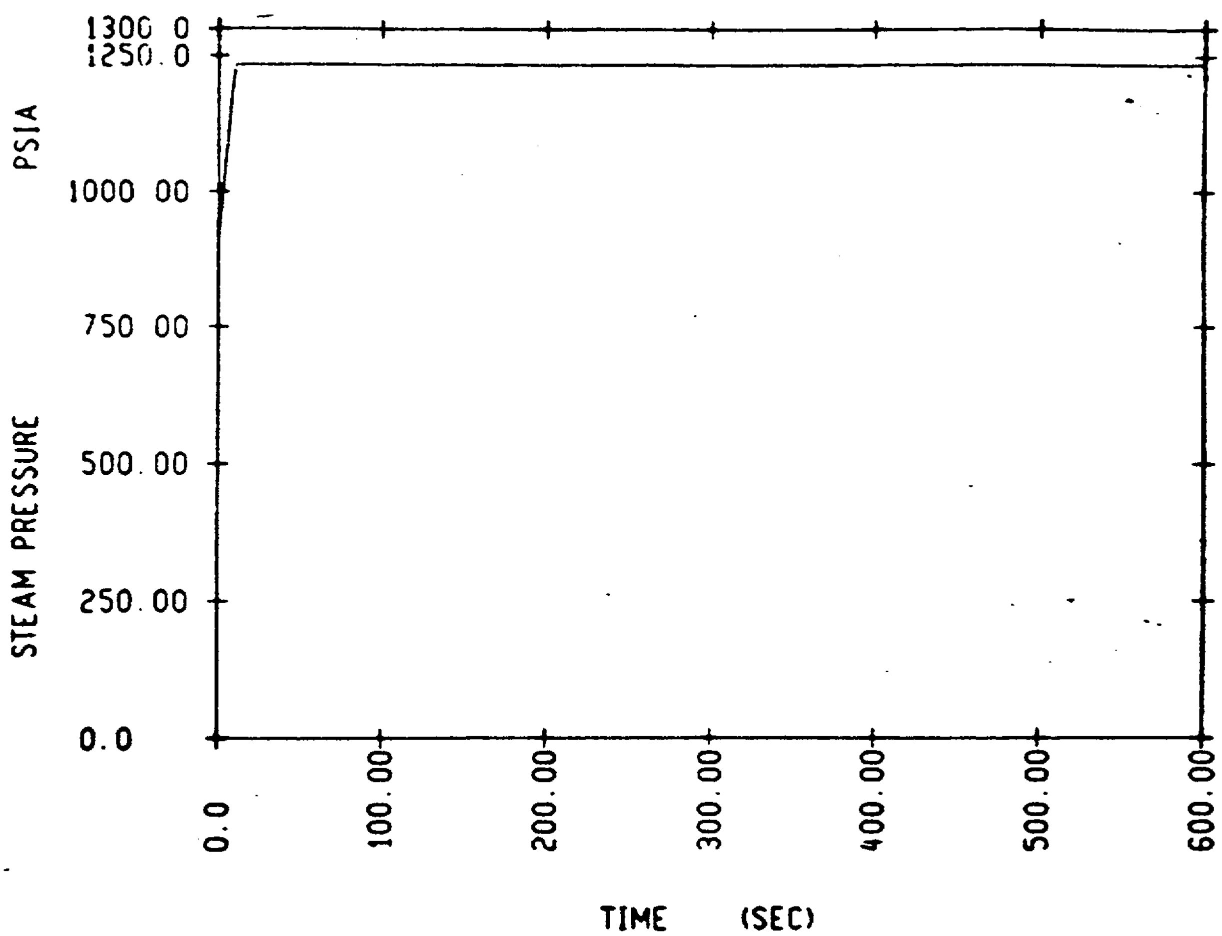
LOSS OF LOAD ATWS
REFERENCE CASE
99% MTC

Figure 5.1-17



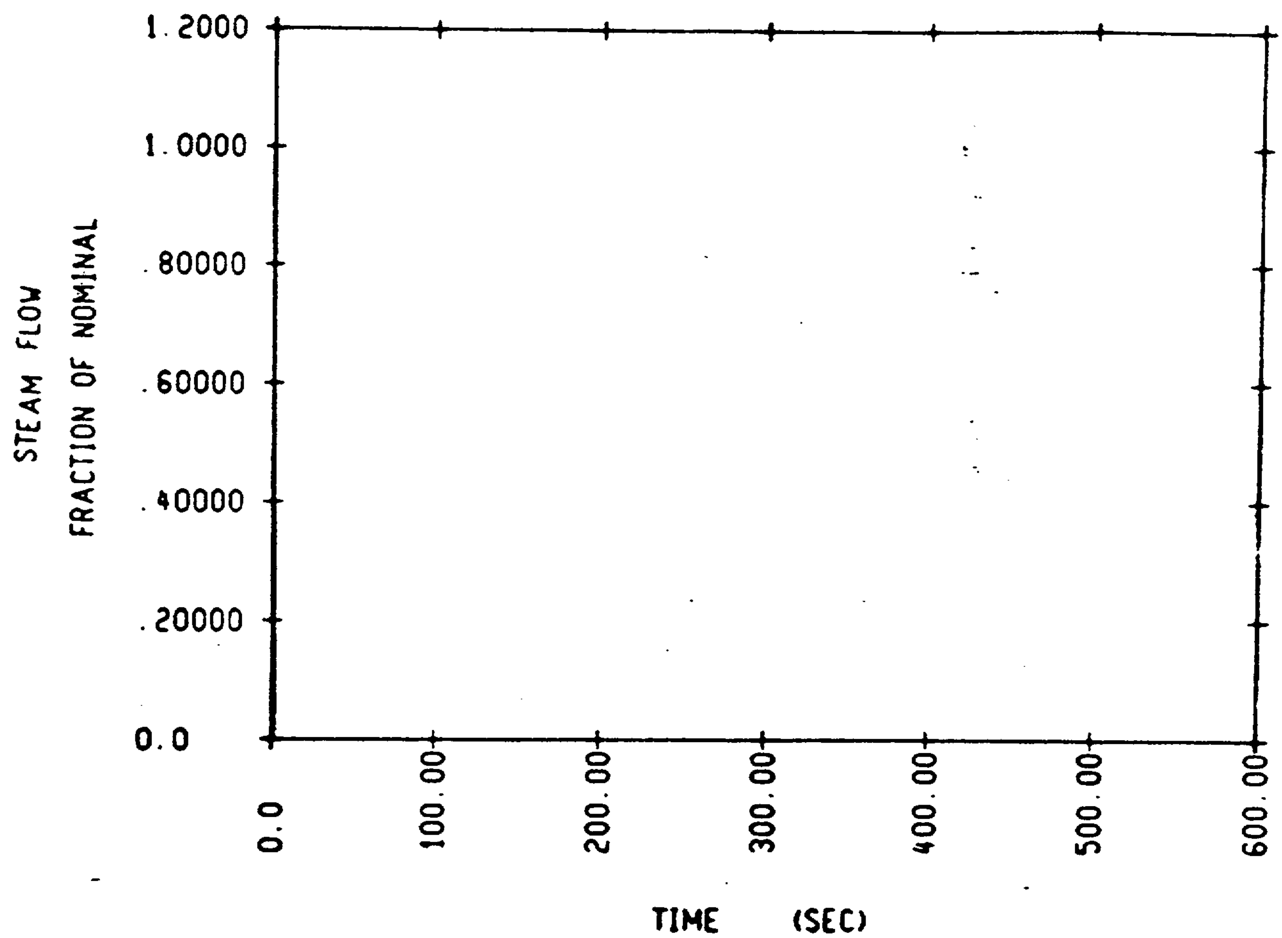
LOSS OF LOAD ATWS
REFERENCE CASE
99% MTC

Figure 5.1-18



LOSS OF LOAD ATWS
REFERENCE CASE
99% MTC

Figure 5.1-19



LOSS OF LOAD ATWS
REFERENCE CASE
99% MTC

Figure 5.1-20

5.2 COMPLETE LOSS OF NORMAL FEEDWATER WITHOUT REACTOR TRIP

5.2.1 IDENTIFICATION OF CAUSES AND TRANSIENT DESCRIPTION

Loss of normal feedwater could result from a malfunction in the feedwater condensate system or its control system from such causes as simultaneous trip of both condensate pumps, simultaneous trip of both main feedwater pumps (or closure of their discharge valves), or simultaneous closure of all feedwater control valves. The vast majority of these cases would cause only a partial loss of feedwater flow. The most likely cause of a complete loss of feedwater would be loss of offsite power which is evaluated in Section 5.3.

The loss of main feedwater produces a large imbalance in the heat source/sink relationship. When feedwater flow to the steam generators is terminated, the secondary system can no longer remove all of the heat that is generated in the reactor core. This heat buildup in the primary system is indicated by rising Reactor Coolant System temperature and pressure, and by increasing pressurizer water level, which is due to the insurge of expanding reactor coolant. Water level in the steam generators drops as the remaining water in the secondary system, unreplenished by main feedwater flow, is boiled off. When the steam generator water level falls to the point where the steam generator tubes are effectively exposed and primary-to-secondary system heat transfer is reduced, the reactor coolant temperature and pressure begin to increase at a greater rate. This greater rate of primary system temperature and pressure increase is maintained as the pressurizer fills and discharges water through the safety and relief valves. Reactivity feedback, due to the high primary system temperature, reduces core power. Eventually, the system pressure begins to decrease, and a steam space is again formed in the pressurizer.

For protection for loss of feedwater, the reactor would be tripped when any of the following conditions are reached.

- Steam/feedwater flow mismatch (low feedwater flow) and low steam generator water level
- Overtemperature ΔT reactor trip
- High pressuizer pressure
- High pressuizer level
- Steam generator low-low water level
- Low reactor coolant flow

None of these trips is assumed during the loss of feedwater ATWS.

5.2.2 ANALYSIS OF EFFECTS AND CONSEQUENCES

The analysis includes a moderator temperature coefficient that is valid for at least 95 percent of core life. This corresponds to the assumptions required for Alternative 3 transient analyses. A 99 percent value of the coefficient is required to be used for Alternative 4 transients. The effect of the 99 percent value is studied for the reference case, a 4-loop, 51 series steam generator plant configuration, for the loss of normal feedwater ATWS.

The following assumptions were made in the analysis:

- Initial normal full power operation early in core life. Since the negative temperature coefficient of reactivity reduces core power as the coolant temperature rises, and the temperature coefficient becomes more negative with core life, the ATWS loss of feed is less severe in later core life.

- Normal operation of the following control systems:

1. Both the power-operated and the spring-loaded relief valves are operable

2. Turbine governor valves in impulse control prior to trip, and valve closure on turbine trip

3. Steam dump to condenser at 40 percent of rated turbine flow following turbine trip

- Turbine trip 30 seconds after loss of feed. AMSAC circuitry will trip the turbine directly.

- No credit for automatic reactor trip

- No credit for automatic control rod insertion as reactor coolant temperature rises

- Main feedwater flow falls to zero in the first four seconds of the transient, with no main feed after that time.

- Auxiliary feedwater flow begins at 60 seconds, at a rate of 1760 gpm, with the initiation signal provided by AMSAC.

- Auxiliary feedwater is injected into the feedwater pipe at a temperature of 130°, 500 ft^3 upstream of the steam generators, such that the cooler water enters the steam generator after this volume is purged.

- Primary-to-secondary heat transfer area is reduced as the steam generator shell-side water inventory drops below the value neccesary to cover the tubes.

The assumptions described above include a turbine trip 30 seconds into the transient, and auxiliary feedwater available in 60 seconds, both initiated by AMSAC. AMSAC is a diverse set of circuitry that would initiate ATWS - required mitigation systems in the event of an unknown common- cause failure of the complete reactor scram system. Both turbine trip and auxiliary feedwater would be actuated by AMSAC in much less time than assumed in this analysis.

5.2.3 RESULTS

5.2.3.1 51 Series Steam Generator

The 4-loop, 51 Series steam generator plant configuration was analyzed for the loss of normal feedwater transient as described above. This plant configuration, assuming a 95 percent value of the moderator temperature coefficient, is considered the reference case.

The peak pressure in the Reactor Coolant System for the reference case was 2848 psia and occurred approximately 106 seconds after the termination of feedwater supply to the steam generators. The pressurizer reached a peak pressure of 2746 psia at the same time, while relieving 1189 lb/sec of water.

The chronology of events for this case is shown in Table 5.2-1 and plots are presented in figures 5.2.1 through 5.2.11. The gradual drop in flow rate, before pump cavitation occurs, is due to coolant expansion (density decreases). The volumetric flow rate, however, is relatively constant before the pump is assumed to cavitate.

The loss of normal feedwater ATWS reference case was also analyzed utilizing a moderator temperature coefficient that is valid for over 99 percent of core life. The transient results for this case are presented in Figures 5.2.12 through 5.2.22. The reactor coolant system pressure reached a peak of 2914 psia, higher than the reference case with a 95 percent value of the coefficient.

A 51 Series steam generator 3-loop and 2-loop plant configurations were analyzed for a loss of normal feedwater ATWS to determine their effect on the peak system pressure when compared to the reference 95% case. In both plant configurations, the transient trends are similar to the reference case. The 3-loop case results in a peak system pressure of 2783 psia, while the resulting peak pressure for the 2-loop case is 2753 psia.

5.2.3.2 Model D Steam Generator

The loss of normal feedwater ATWS was analyzed for 4-loop and 3-loop plants with a Model D steam generator and the assumptions described in Section 5.2.2. The resulting plant parameter transients are similar in nature to the loss of normal feedwater ATWS reference case. The 4-loop Model D plant configuration results in a peak reactor coolant system pressure of 2725 psia. The 3-loop Model D plant attained a peak pressure of 2735 psia.

5.2.3.3 Model F Steam Generator

The loss of normal Feedwater ATWS transient results for a Model F steam generator compare closely with the results of the reference case presented in this section. A 4-loop Model F plant configuration yields a peak reactor coolant system pressure of 2830 psia, while a 3-loop configuration attains a peak pressure of 2750 psia.

5.2.3.4 44 Series Steam Generator

A 44 Series steam generator plant configuration was analyzed for the loss of normal feedwater ATWS, with the assumptions described earlier in this section. The transients resulting from this model of steam generator are similar to the reference case. The peak reactor coolant system pressure for a 4-loop, 44 Series plant loss of normal feedwater ATWS is 2857 psia. The 3-loop plant configuration yields a 2717 psia peak pressure.

5.2.4 SENSITIVITY STUDIES

The loss of normal feedwater transient reference case is evaluated with changes in certain assumptions and initial conditions to determine their effect on the results of the transient. The results of these sensitivity studies are listed here as variations in pressure with respect to the reference case provided in this section.

5.2.4.1 Effect of One Auxiliary Feedwater Pump Failing (Single Failure)

The single failure criteria of assuming that the largest auxiliary feedwater pump fails to start upon receiving an AMSAC signal during a loss of normal feedwater ATWS is considered. This assumption effectively reduces the auxiliary feedwater available during the transient by one half. The effect of reduced auxiliary feedwater is a reduction is primary to secondary heat transfer, with a corresponding increase in the primary system temperature and pressure. In this case, the peak reactor coolant system pressure that is attained during the transient is increased by 31 psi over the reference case.

5.2.4.2 Effect of One PORV Failing to Open (Single Failure)

If one power-operated relief valve fails to open upon demand during a loss of normal feedwater transient, the mass and energy release is reduced, resulting in an increase in the primary system pressure. A sensitivity study on the reference case shows that under these conditions, the peak pressure reached during the transient is increased by 108 psia in the reactor coolant system.

5.2.4.3 Effect Of Variation In Initial Pressurizer Water Level

Variation of ±10 percent in initial pressurizer water level is considered. The maximum pressurizer pressure attained during a loss of feedwater ATWS, with the pressurizer water level at 10 percent above the nominal level, is increased by 4 psia over the base case. Another transient which is based on a lower pressurizer water level (10 percent below nominal level) produces a maximum pressurizer presssure that is lower by 5 psia. The higher initial water level means that the pressurizer fills to capacity earlier in the transient when the core power is still relatively high. A lower than normal water level delays the filling of the pressurizer, and provides more steam for volumetric relief through the relief valves, resulting in a lower pressurizer pressure.

5.2.4.4 Effect of Variation In Steam Generator Water Inventory

The initial steam generator water mass was varied to determine the effect on the loss of normal feedwater transient. An increase in the initial water mass of 10 percent does not affect the peak reactor coolant system pressure. The peak pressure increases by 2 psi when the initial mass is decreased by 10 percent.

5.2.4.5 Effect of Variation In Main Feedwater Enthalpy

A variation in the initial main feedwater enthalpy does appreciably not effect the peak reactor coolant system pressure because the assumed transient is a loss of main feedwater. The only effect is due to the difference in purge volume enthalpy when auxiliary feedwater is initiated. An increase of 10 percent in the main feedwater enthalpy produces a psi peak pressure increase. Likewise, a 10 percent decrease in enthalpy results in a 3 psi pressure decrease.

5.2.4.6 Effect of Variation in the Reactor Coolant System Volume

An increase in the reactor coolant system volume of 10 percent corresponds to an increase of 18 psi in the peak pressure over the reference case. A decrease in the volume of 10 percent yields a decrease in the peak pressure of 12 psi.

5.2.4.7 Effect of Variation In the Auxiliary Feedwater Flow Rate

As described in Section 5.2.4.1, a decrease in auxiliary feedwater flow will cause an increase in peak reactor coolant system pressures. A sensitivity study reducing the flow rate by 10 percent shows an increase of 3 psi in the peak pressure. Likewise, an increase of 10 percent in the available flow yields a decrease in the peak system pressure of 3 psi.

5.2.4.8 Effect of Variation In Fuel UA

Sensitivity studies with changes in the fuel UA show that there is little effect on the peak reactor coolant system pressure. An increase in fuel UA of 10 percent results in a decrease of 2 psi in the peak pressure, while a corresponding 10 percent decrease yields a 3 psi increase in the peak pressure.

5.2.4.9 Effect of the Pressurizer Spray

Assuming the pressurizer spray system in operable during a loss of normal feedwater transient tends to reduce the pressures attained during the transient. An analysis was done assuming proper operation of the pressurizer spray, resulting in a decrease in the peak system pressure of 6 psi.

5.2.4.10 Effect of Variation in Reactor Power

If the initial reactor power level is increased by 2 percent during a loss of normal feedwater ATWS, the resulting peak system pressure attained during the transient in increased by 23 psi when compared to the reference case. A decrease of 2 percent in initial reactor power results in a decrease of 15 psi in the peak pressure attained.

5.2.4.11 Effect of Delay In Auxiliary Feedwater Initiation

A delay in the initiation of the auxiliary feedwater system during a loss of normal feedwater ATWS affects the ability of the steam generators to remove excess heat. The reduced heat transfer ability causes the primary system pressures to be increased when compared to the reference case with no initiation delay. (Normal initiation time is 60 seconds after the generation of the AMSAC signal). A 60 delay, or auxiliary feedwater initiation within 120 seconds of the AMSAC signal, results in increasing the peak primary system pressure attained by 90 psi.

5.2.4.12 Effect of Variation in Steam Generator Design Pressure

The steam generator design pressure assumed in the reference 51 Series case is 1200 psia. An analysis was made to determine the effect of an 1100 psia design pressure. The net result is an increase in peak pressure of 136 psi.

5.2.4.13 Effect of Turbine Trip Delay

In the reference loss of normal feedwater ATWS, the turbine is assumed to be tripped within 30 seconds of generation of the AMSAC signal. If the turbine is assumed to be tripped with a 30 delay over the base case, this results in turbine trip within 60 seconds of the AMSAC signal. The peak reactor coolant pressure is increased by 57 psi for this case.

5.2.5 CONCLUSIONS

Table 5.2-2 summarizes the results for the loss of feedwater ATWS reference case and sensitivity studies. The DNB ratio increases above its initial value during the transient as pressure increases. The peak Reactor Coolant System pressure is about 2848 psia for the reference 95% MTC case. Thus, no core damage or impairment of Reactor Coolant System integrity would occur for the loss of feedwater ATWS.

TABLE 5.2-1

SEQUENCE OF EVENTS FOR LOSS OF FEEDWATER WITHOUT A REACTOR TRIP

REFERENCE CASE*

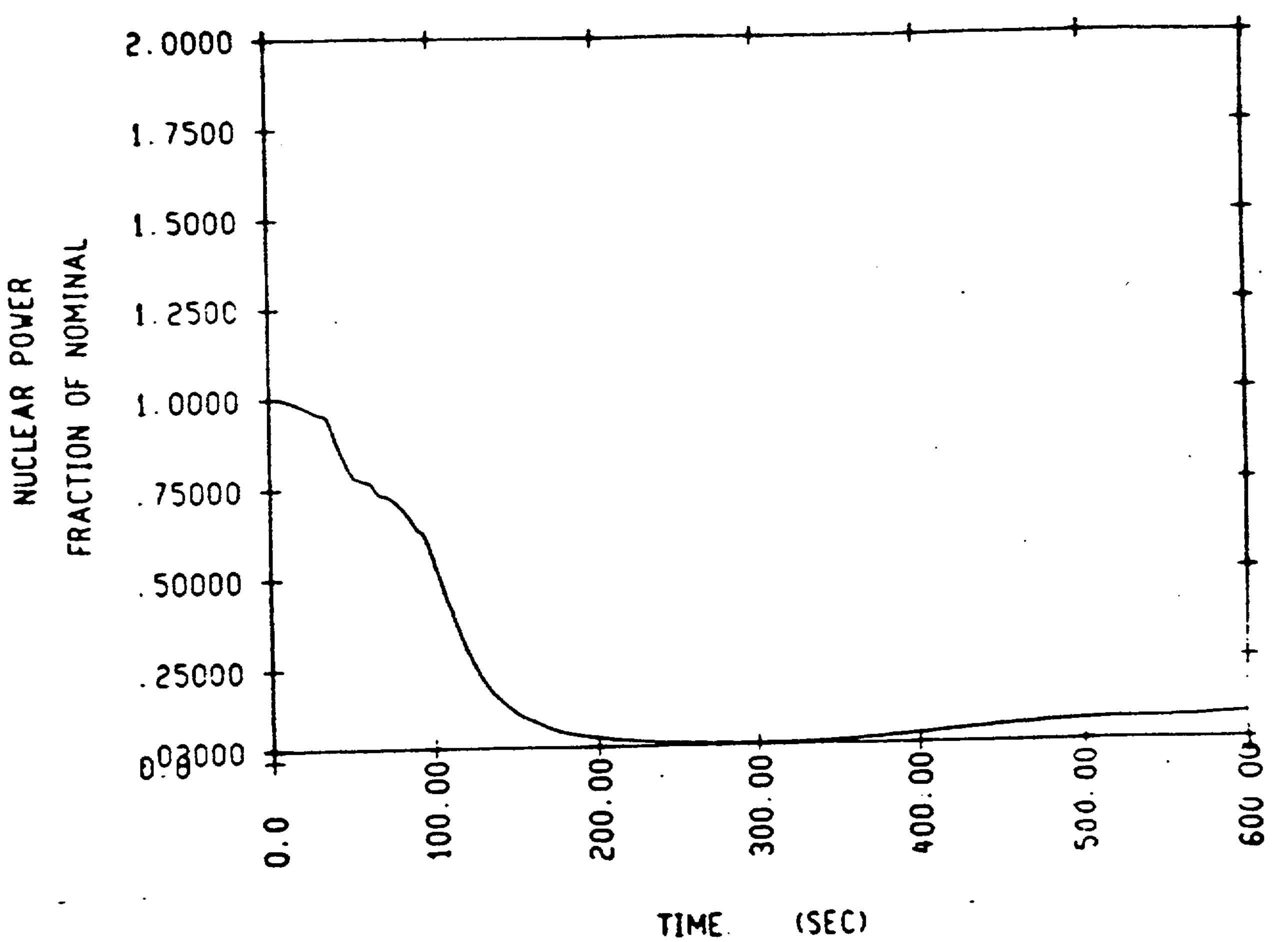
Event	Time (sec)
Main Feedwater Supply To All Steam Generators Is Terminated	0-4
Power-Operated Relief Valves on the Pressurizer Open and Release Steam	17
Turbine is Assumed to Trip	30
Reactor/Turbine Trip Signal: Overtemperature ΔT	36
Reactor/Turbine Trip Signal: High Pressurizer Pressure	40
Steam Generator Safety Valves Open and Hold Steam Pressure Constant	44
All Auxiliary Feedwater Pumps Are Assumed to Start	60
Pressurizer Fills With Water	90
Peak Reactor Coolant System Pressure Is Reached (2848 psia)	106

*Reference case: 4-loop plant with a 51 Series steam generator, 95% value of the moderator temperature coefficient.

TABLE 5.2-2

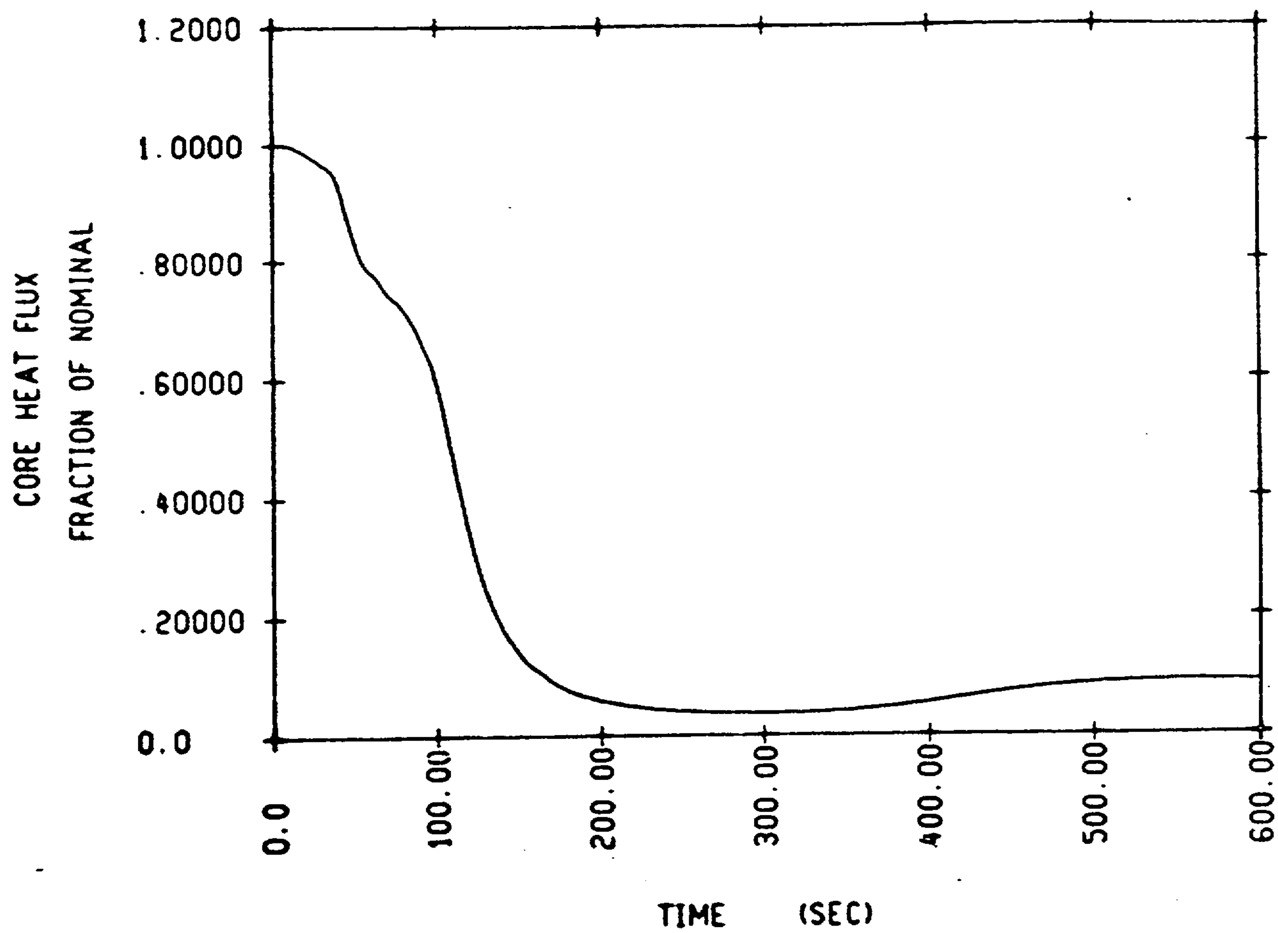
SUMMARY OF RESULTS FOR LOSS OF FEEDWATER WITHOUT A REACTOR TRIP

Case	Change Relative To Reference Case Maximum Reactor Coolant System Pressure (psia)
Reference Case	2931.
One Half Auxiliary Feedwater Flow	+31
One PORV Fails to Open	+108
Pressurizer Water Level +10%	+4
Pressurizer Water Level -10%	-5
Steam Generator Water Mass +10%	+0
Steam Generator Water Mass -10%	+2
Main Feedwater Enthalpy +10%	+3
Main Feedwater Enthalpy -10%	-3
RCS Volume +10%	+18
RCS Volume -10%	-12
Auxiliary Feedwater Flow +10%	-3
Auxiliary Feedwater Flow -10%	+3
Fuel UA +10%	-2
Fuel UA -10%	+3
Pressurizer Spray On	-6
Reactor Power +2%	+23
Reactor Power -2%	-15
60 Second Auxiliary Feedwater Delay	+108
1100 psia Steam Generator Design Pressure	+136
Turbine Trip at 60 Seconds	+57



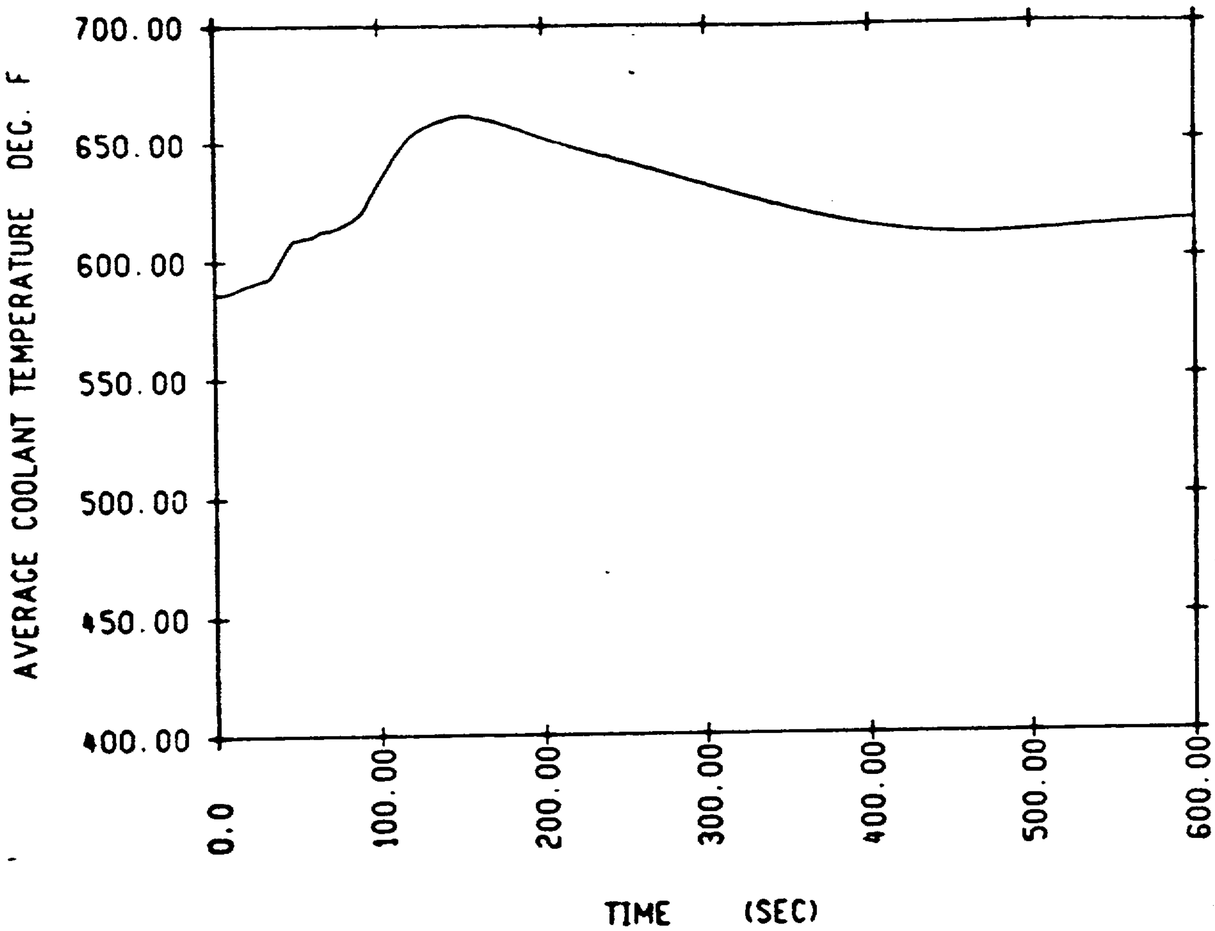
TOTAL LOSS OF NORMAL FEEDWATER
ATWS
REFERENCE CASE
95% MTC

Figure 5.2-1



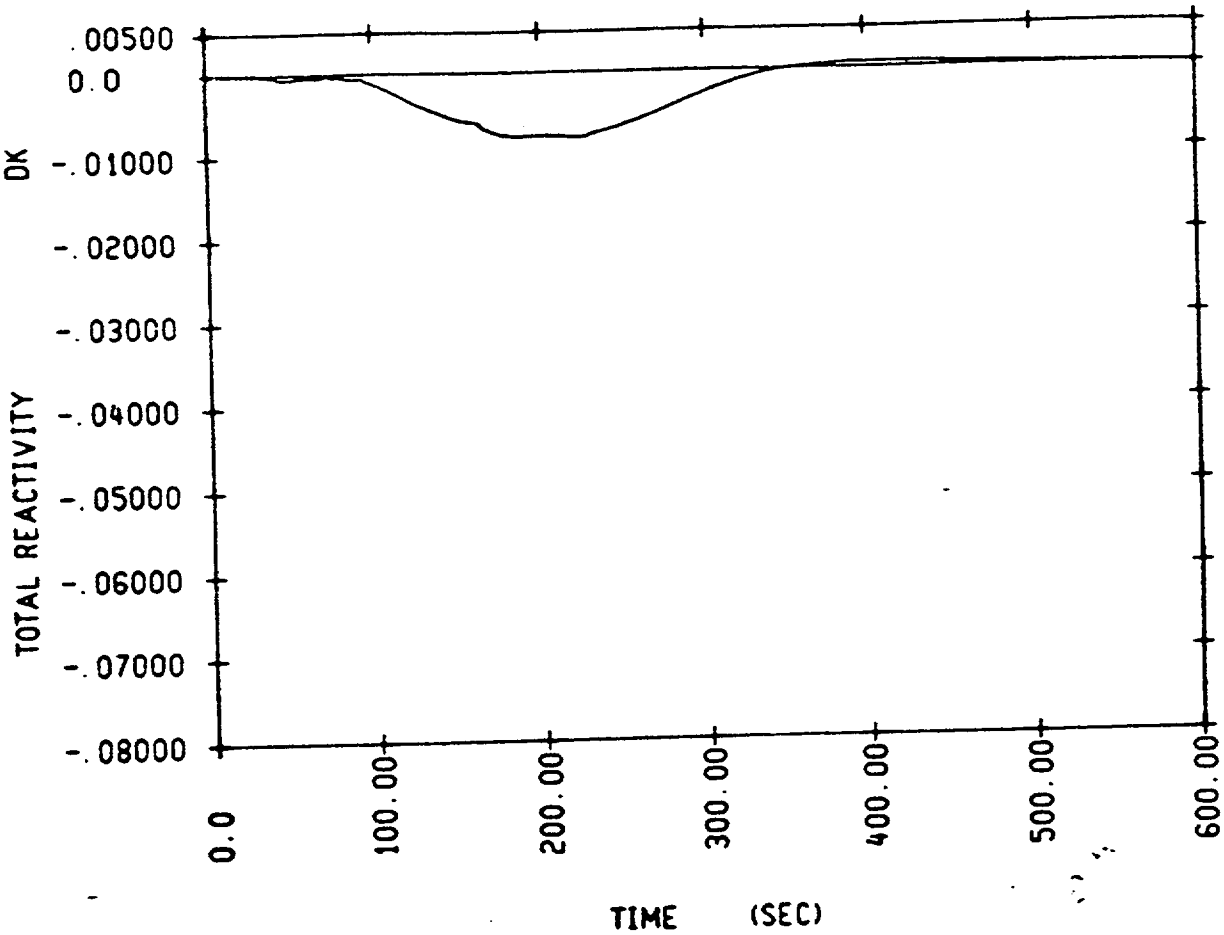
TOTAL LOSS OF NORMAL FEEDWATER
ATWS
REFERENCE CASE
95% MTC

Figure 5.2-2



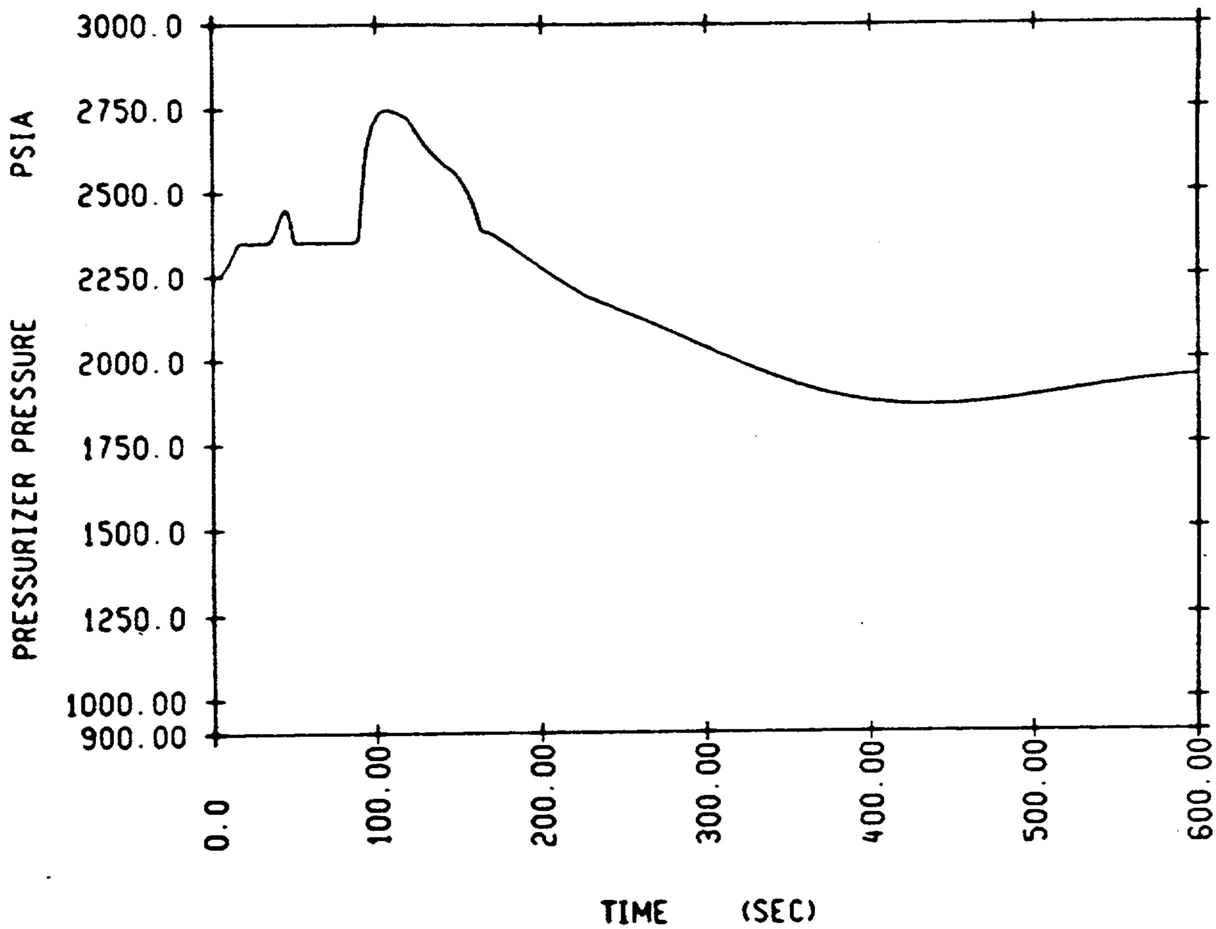
TOTAL LOSS OF NORMAL FEEDWATER
ATWS
REFERENCE CASE
95% MTC

Figure 5.2-3



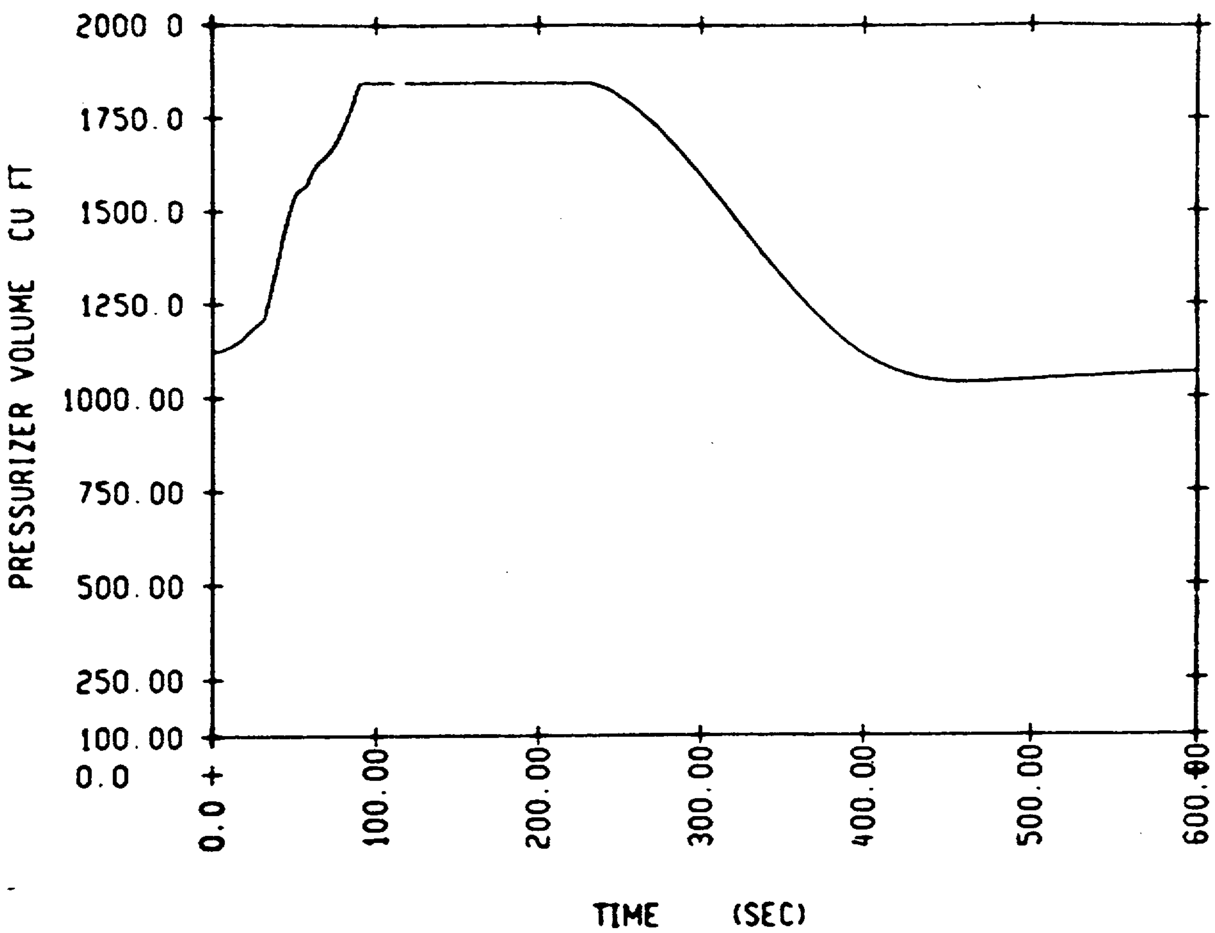
TOTAL LOSS OF NORMAL FEEDWATER
ATWS
REFERENCE CASE
95% MTC

Figure 5.2-4



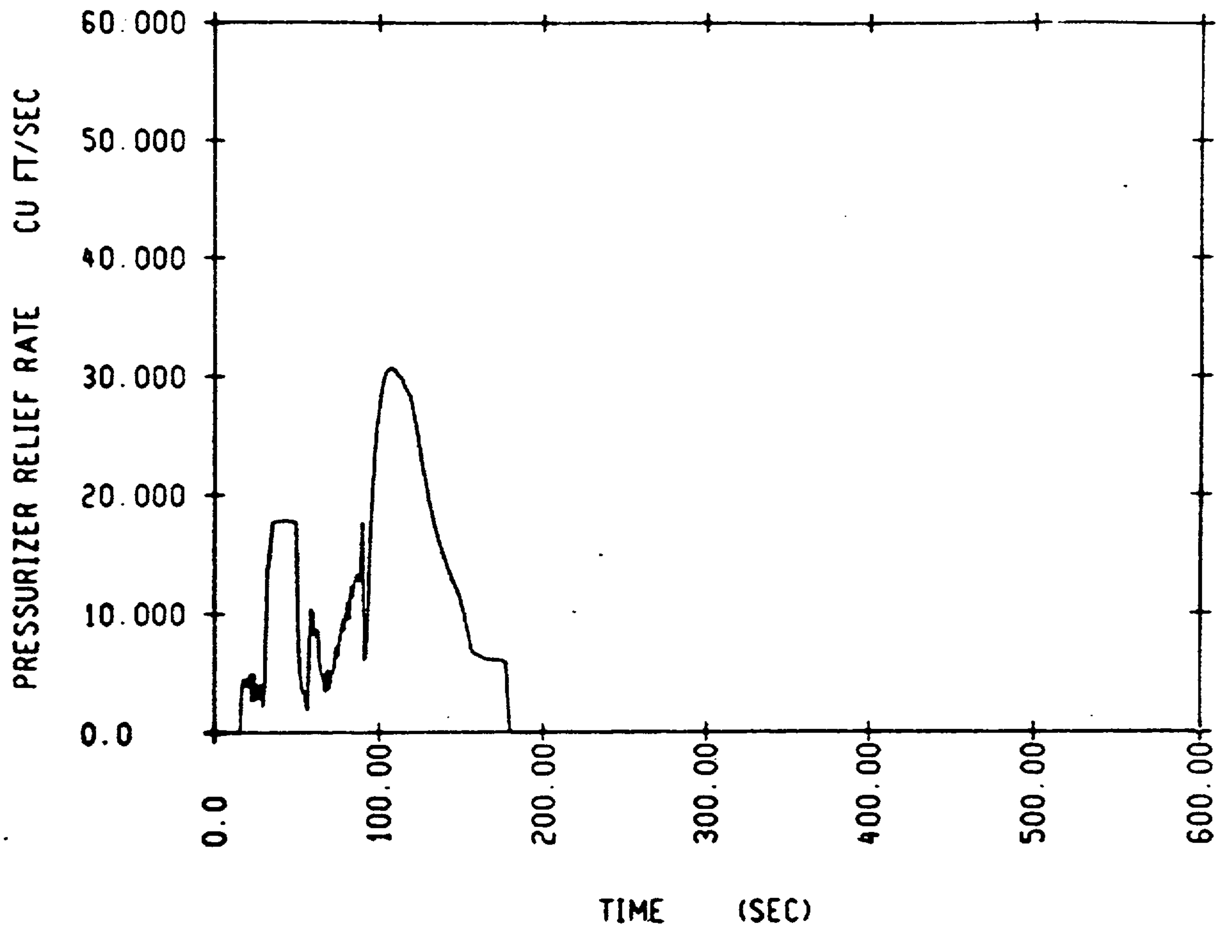
TOTAL LOSS OF NORMAL FEEDWATER
ATWS
REFERENCE CASE
95% MTC

Figure 5.2-5



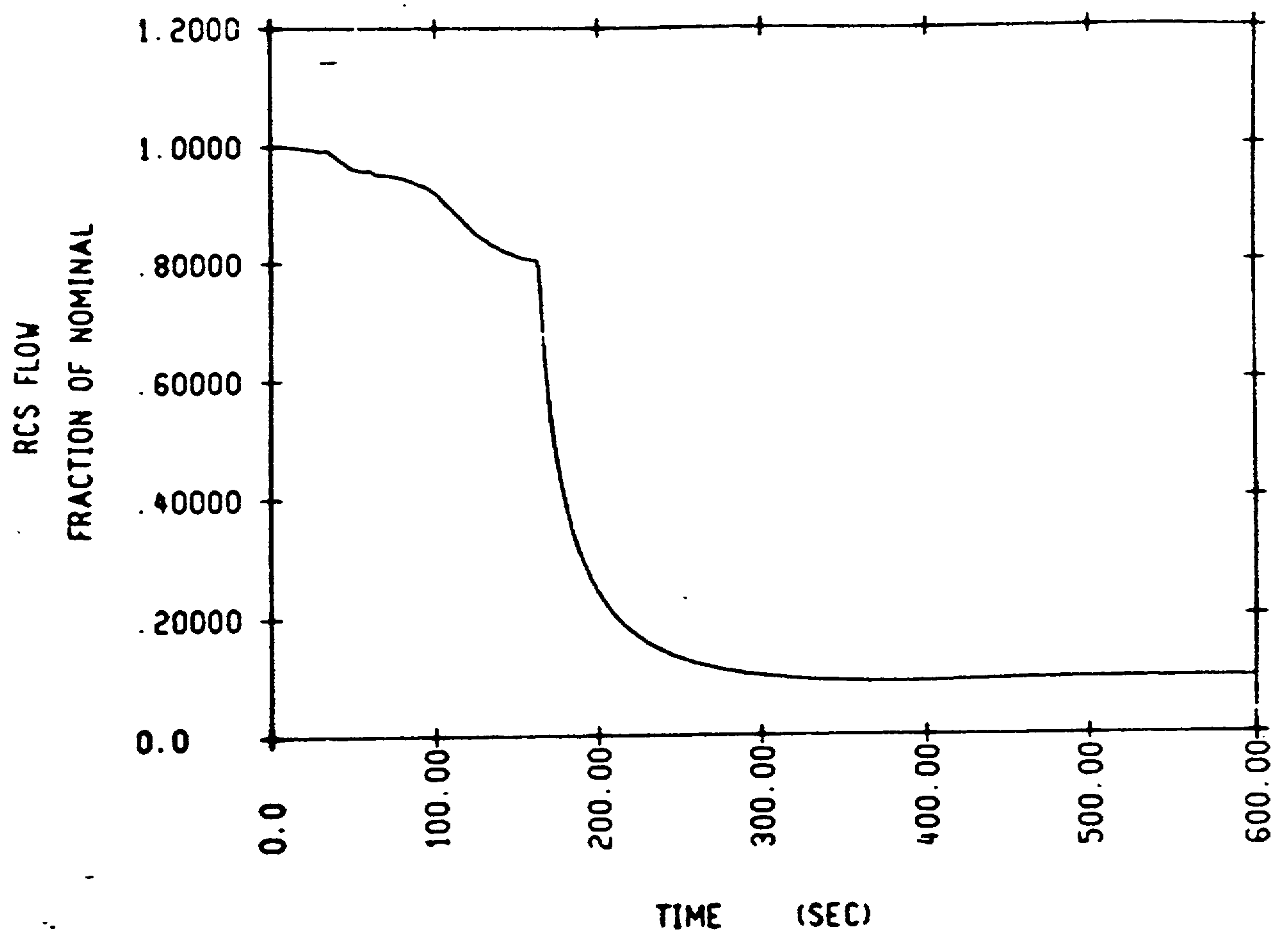
TOTAL LOSS OF NORMAL FEEDWATER
ATWS
REFERENCE CASE
95% MTC

Figure 5.2-6



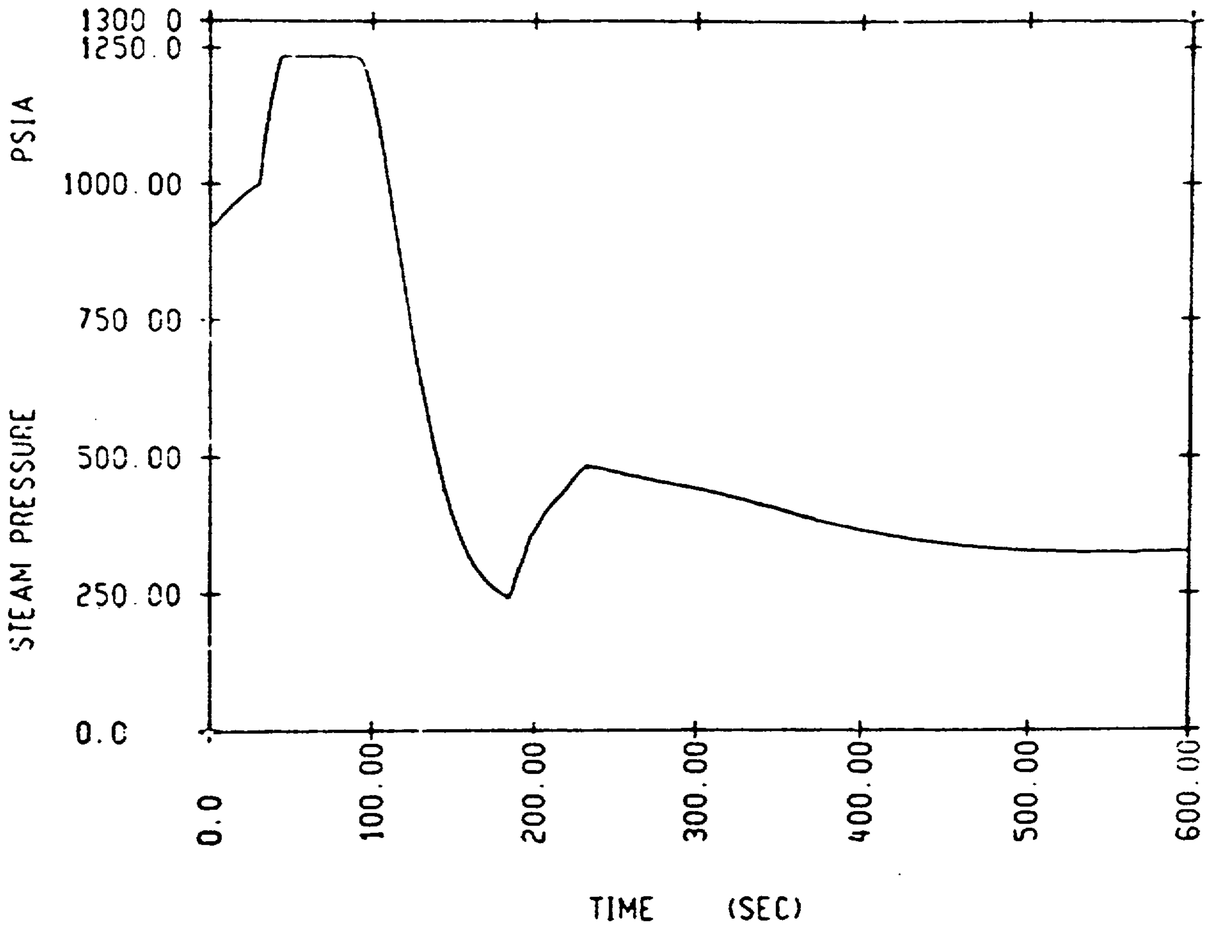
TOTAL LOSS OF NORMAL FEEDWATER
ATWS
REFERENCE CASE
95% MTC

Figure 5.2-7



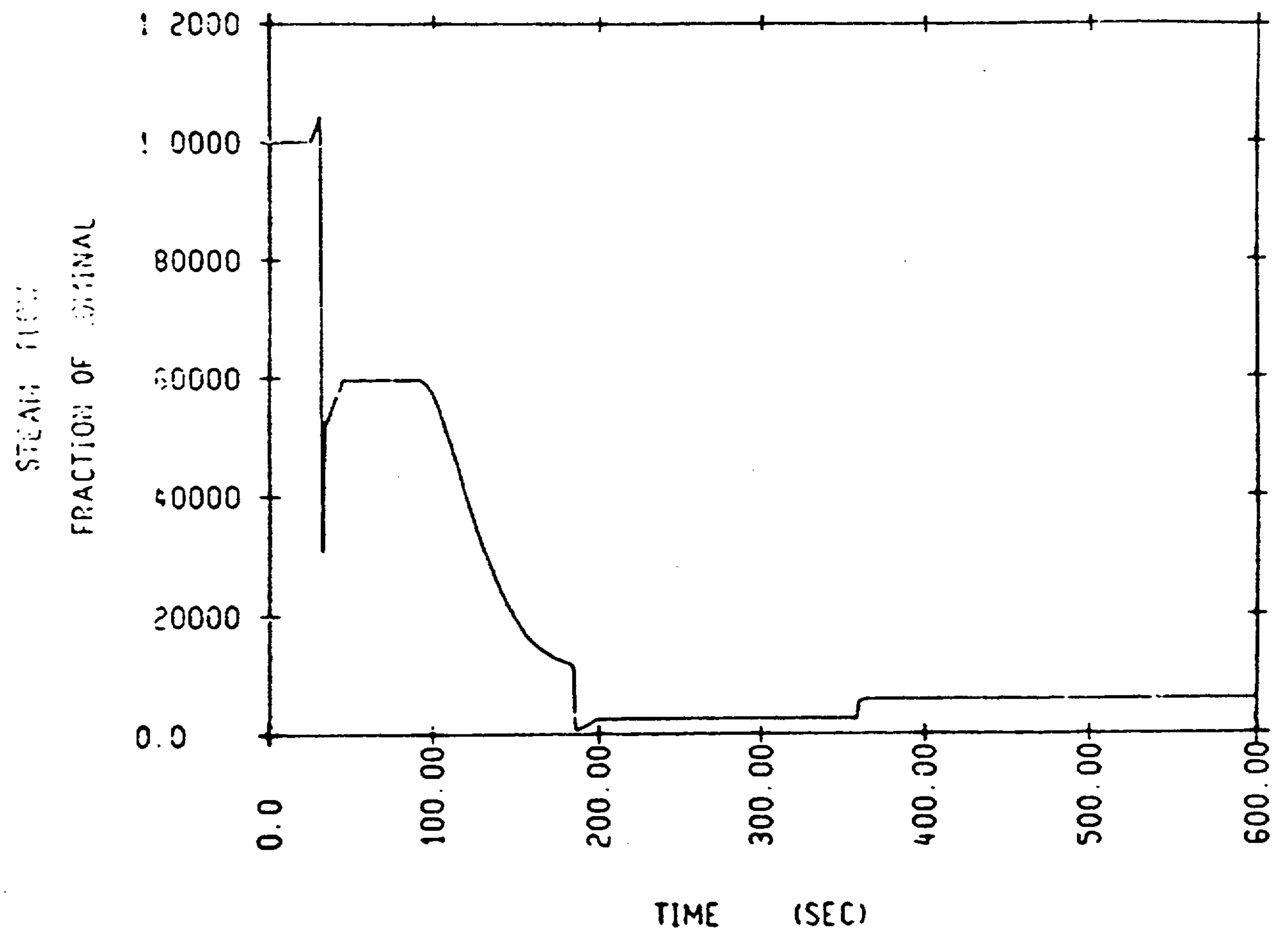
TOTAL LOSS OF NORMAL FEEDWATER
ATWS
REFERENCE CASE
95% MTC

Figure 5.2-8



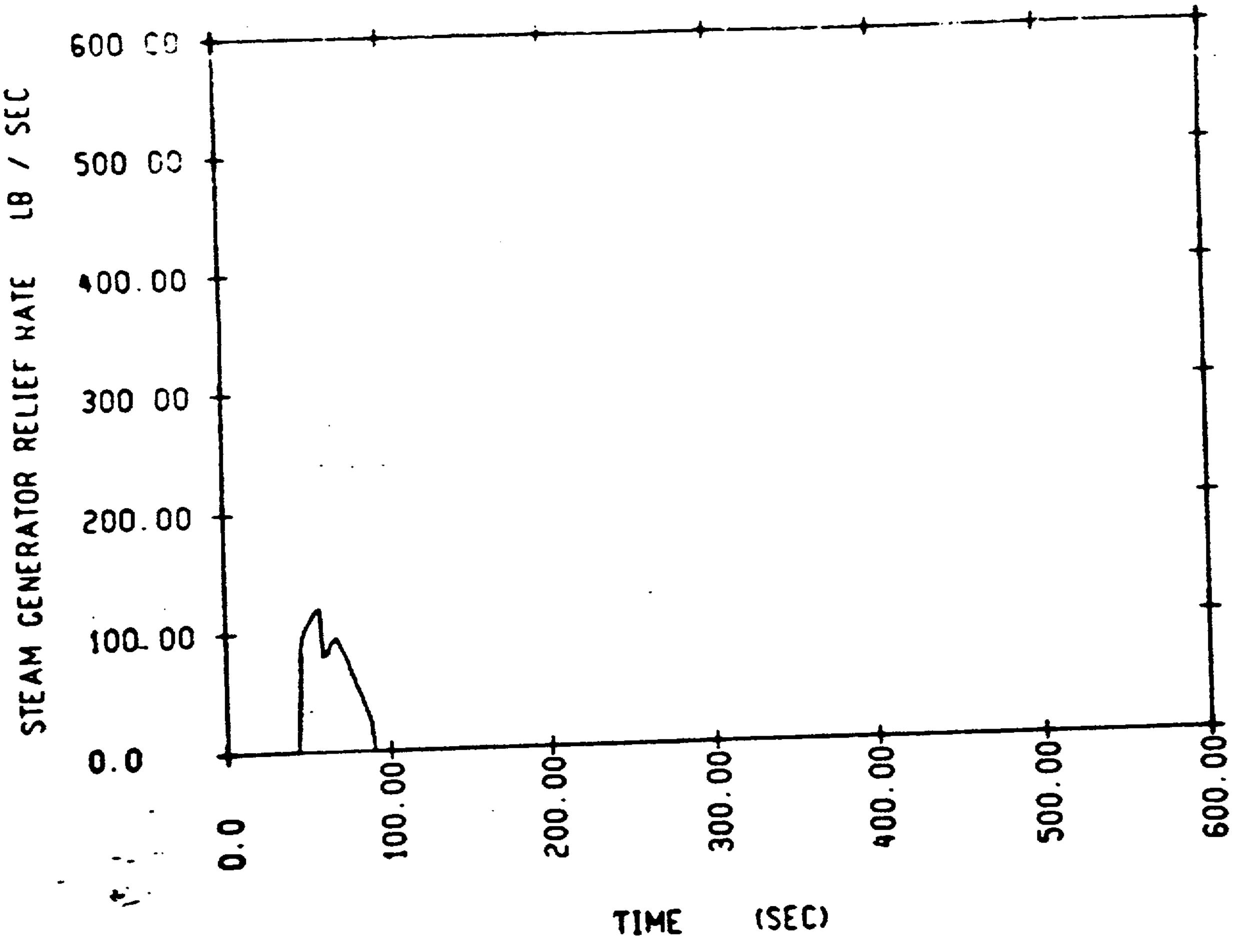
TOTAL LOSS OF NORMAL FEEDWATER
ATWS
REFERENCE CASE
95% MTC

Figure 5.2-9



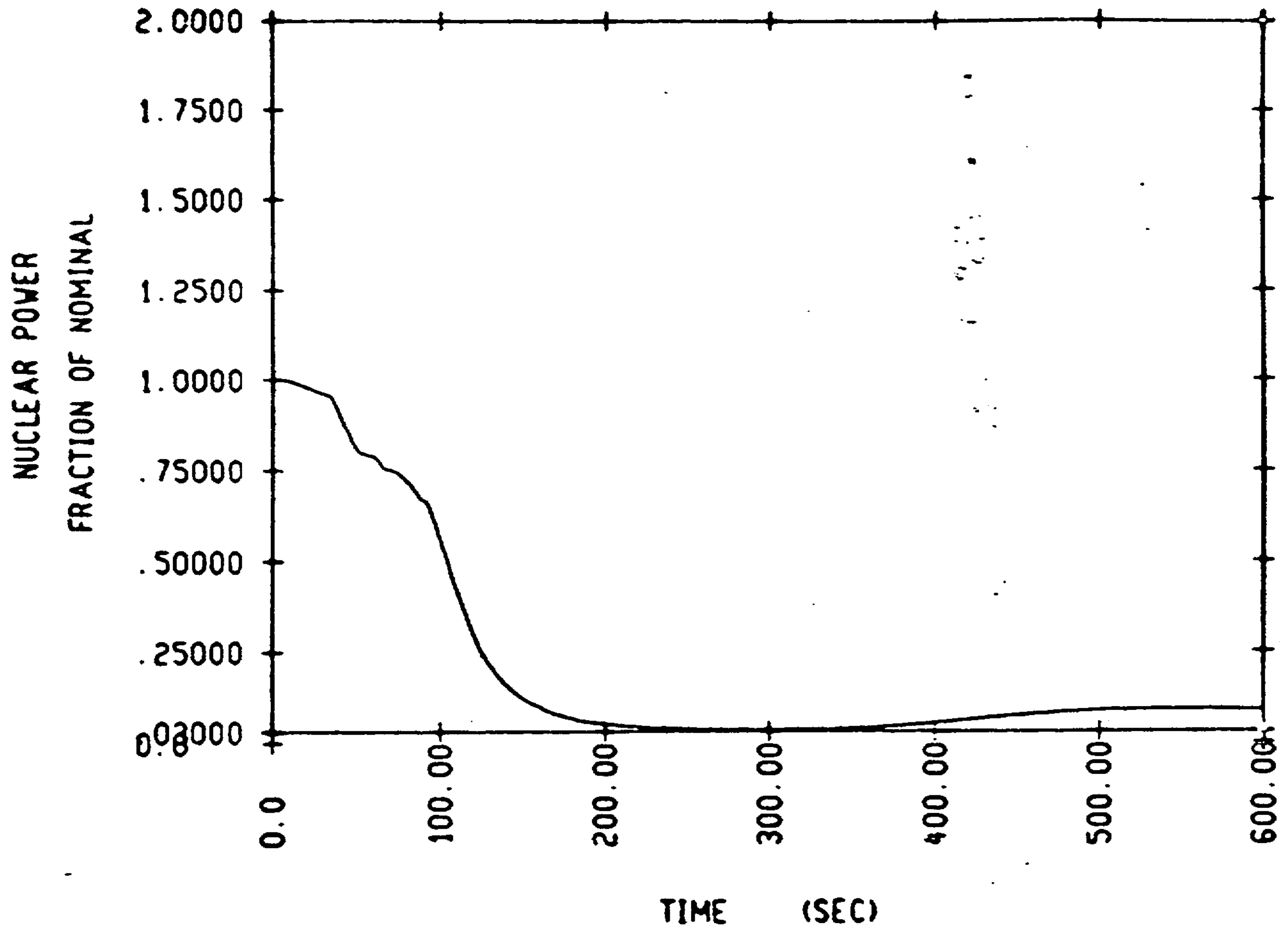
TOTAL LOSS OF NORMAL FEEDWATER
ATWS
REFERENCE CASE
95% MTC

Figure 5.2-10



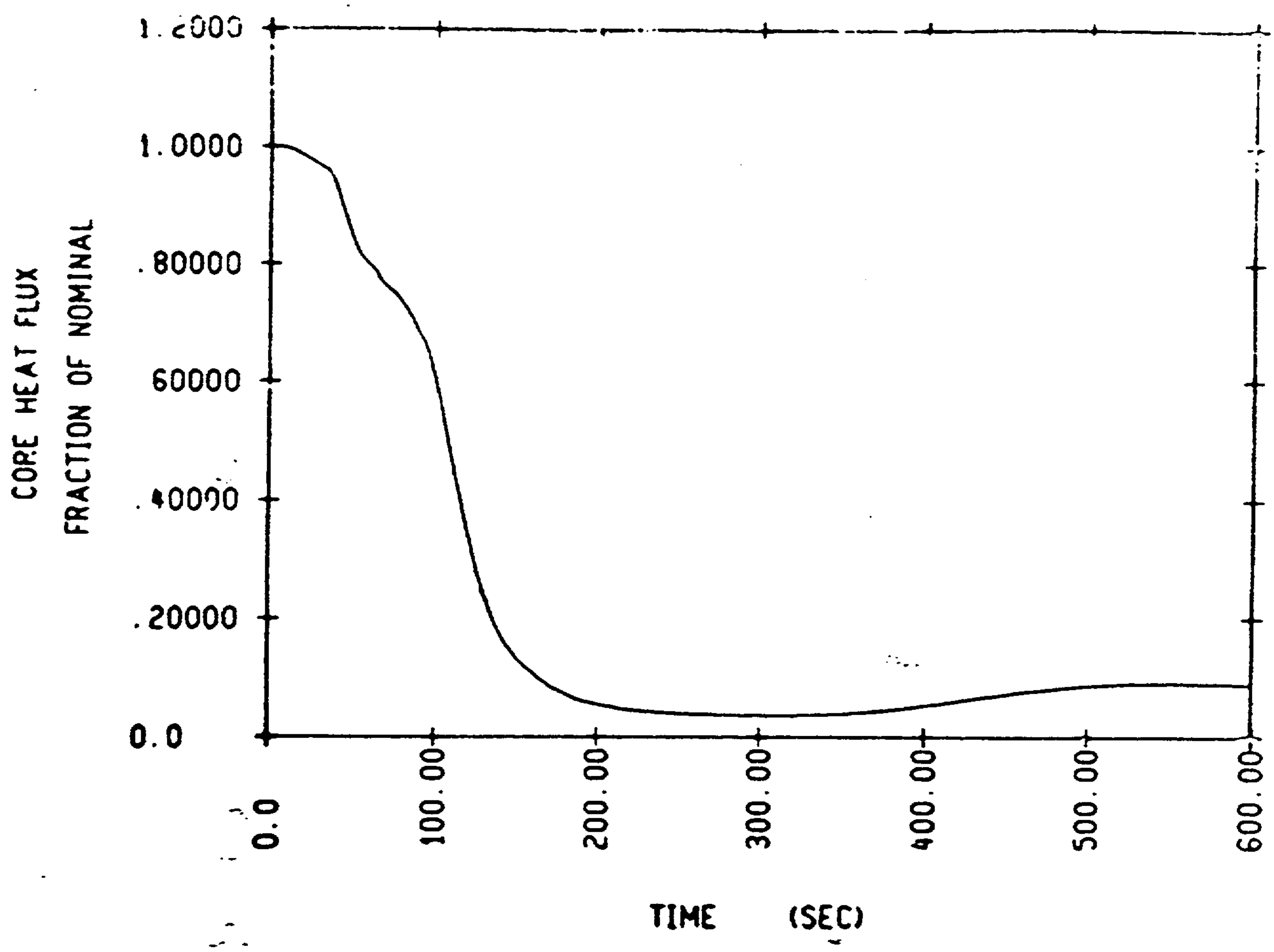
TOTAL LOSS OF NORMAL FEEDWATER
ATWS
REFERENCE CASE
95% MTC

Figure 5.2-11



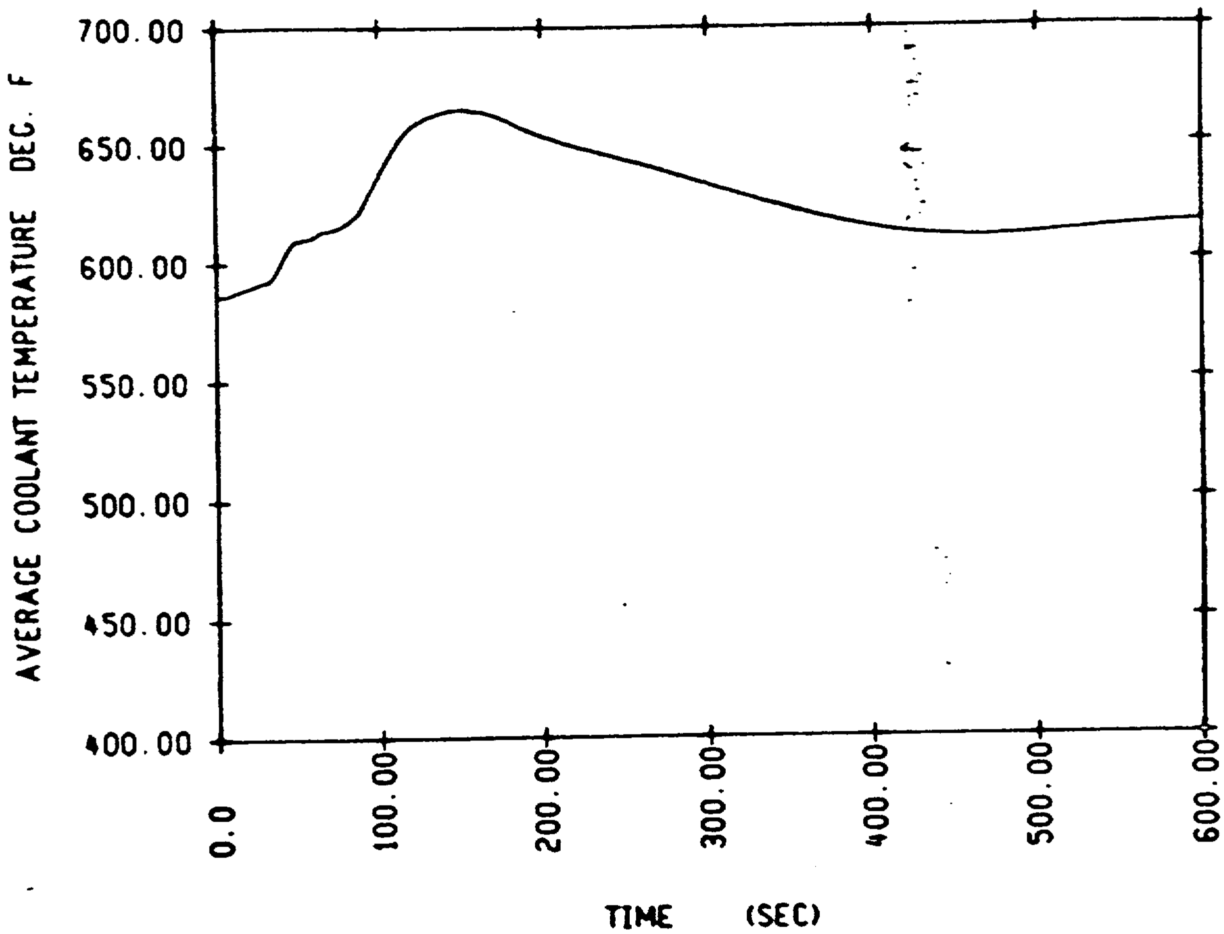
TOTAL LOSS OF NORMAL FEEDWATER
ATWS
REFERENCE CASE
99% MTC

Figure 5.2-12



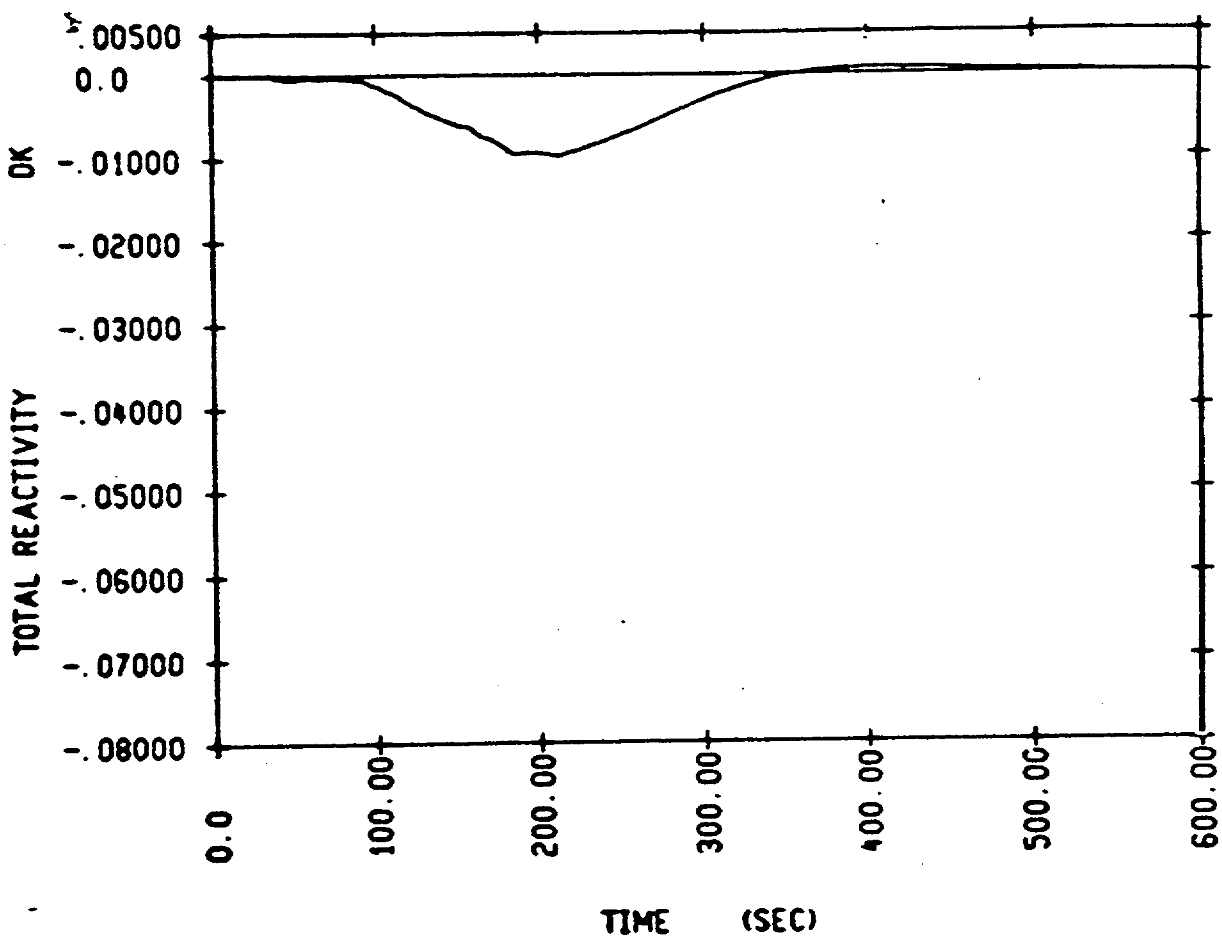
TOTAL LOSS OF NORMAL FEEDWATER
ATWS
REFERENCE CASE
99% MTC

Figure 5.2-13



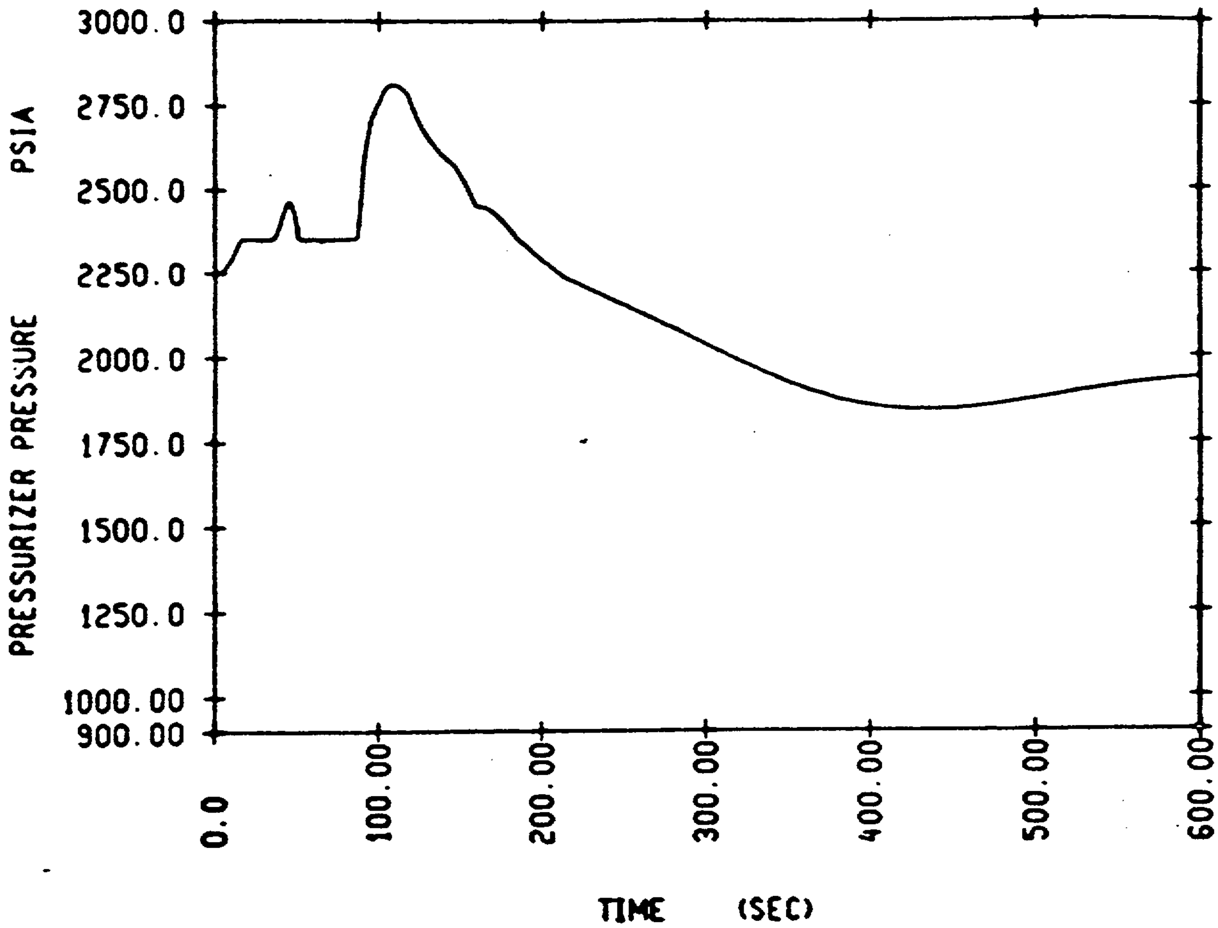
TOTAL LOSS OF NORMAL FEEDWATER
ATWS
REFERENCE CASE
99% MTC

Figure 5.2-14



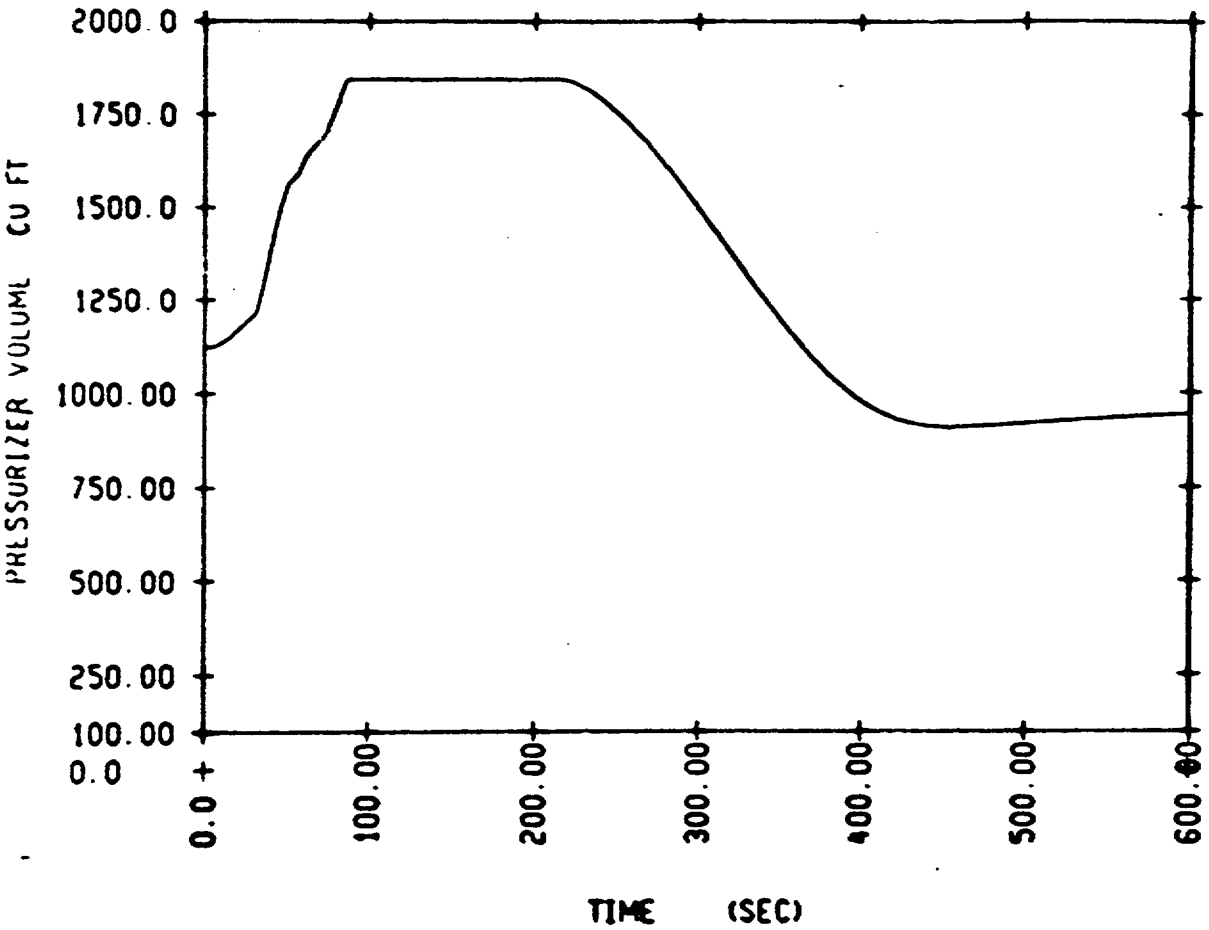
TOTAL LOSS OF NORMAL FEEDWATER
ATWS
REFERENCE CASE
99% MTC

Figure 5.2-15



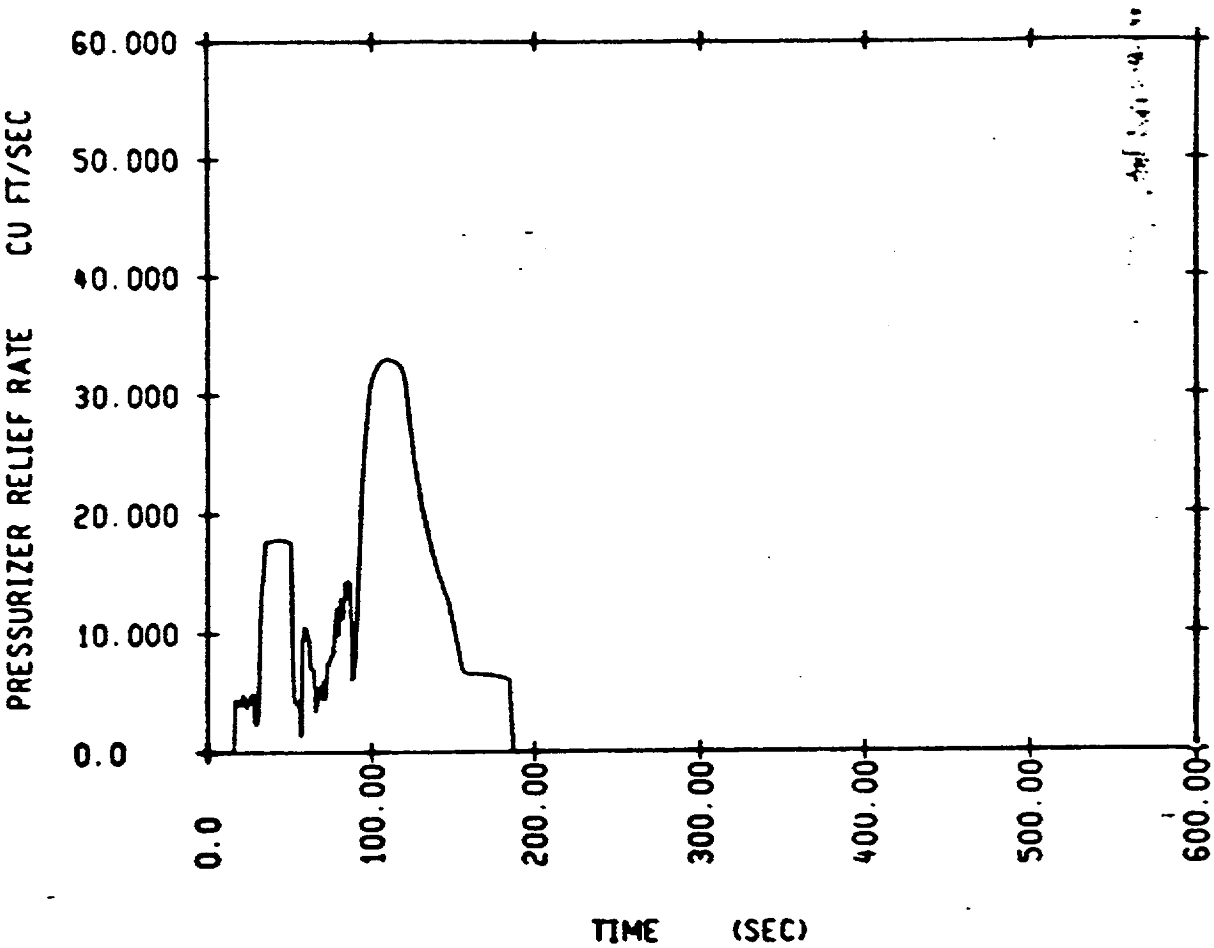
TOTAL LOSS OF NORMAL FEEDWATER
ATWS
REFERENCE CASE
99% MTC

Figure 5.2-16



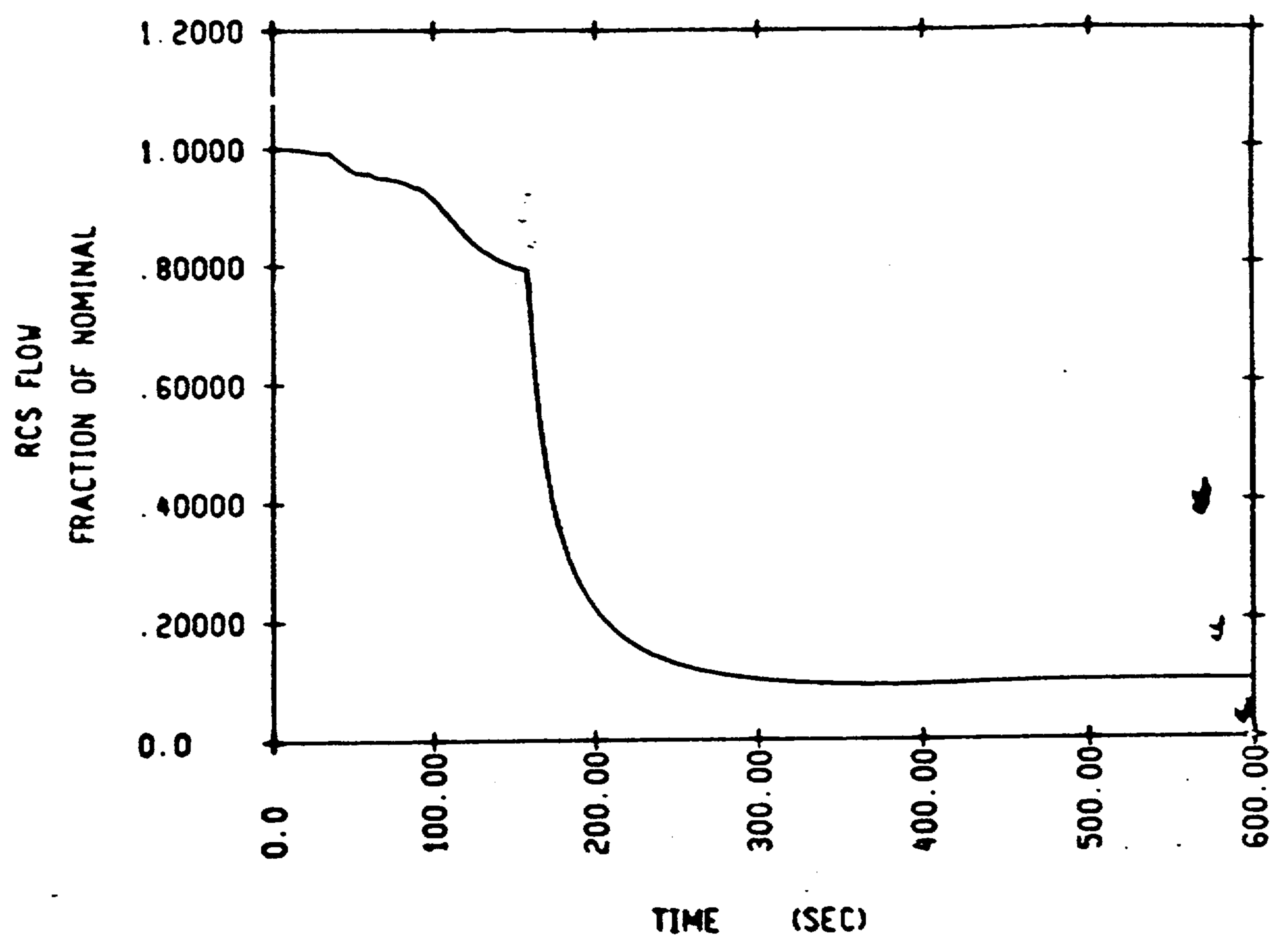
TOTAL LOSS OF NORMAL FEEDWATER
ATWS
REFERENCE CASE
99% MTC

Figure 5.2-17



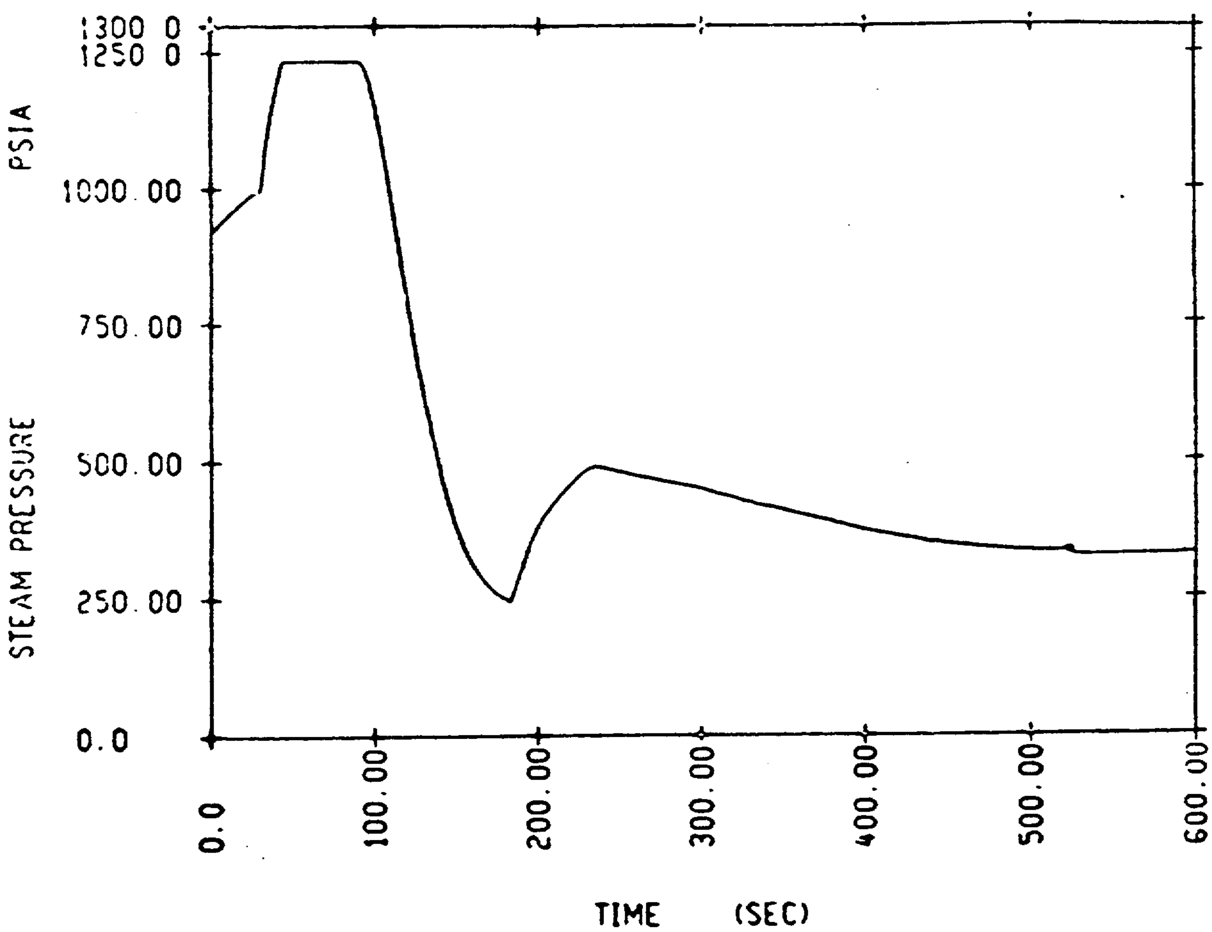
TOTAL LOSS OF NORMAL FEEDWATER
ATWS
REFERENCE CASE
99% MTC

Figure 5.2-18



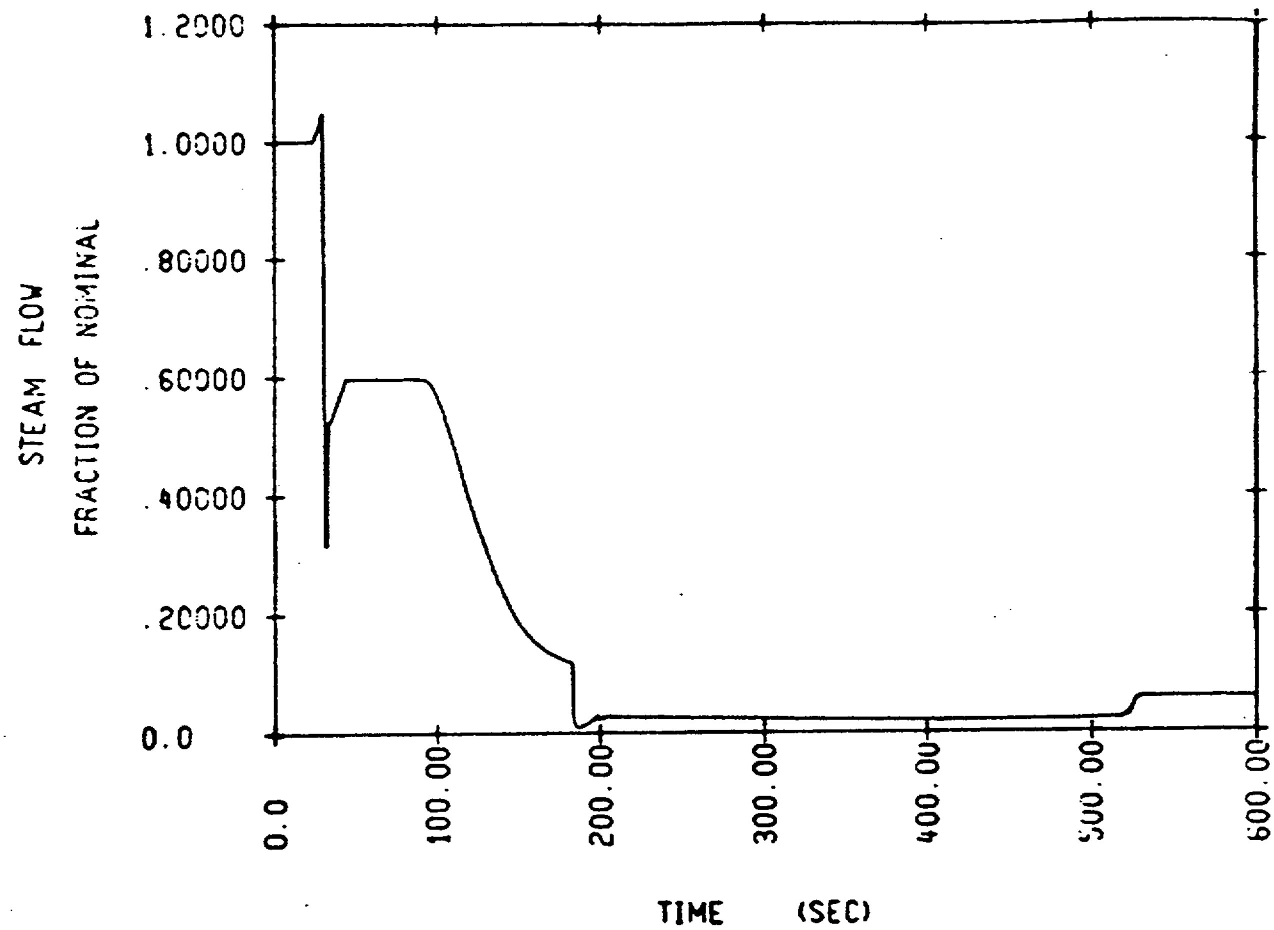
TOTAL LOSS OF NORMAL FEEDWATER
ATWS
REFERENCE CASE
99% MTC

Figure 5.2-19



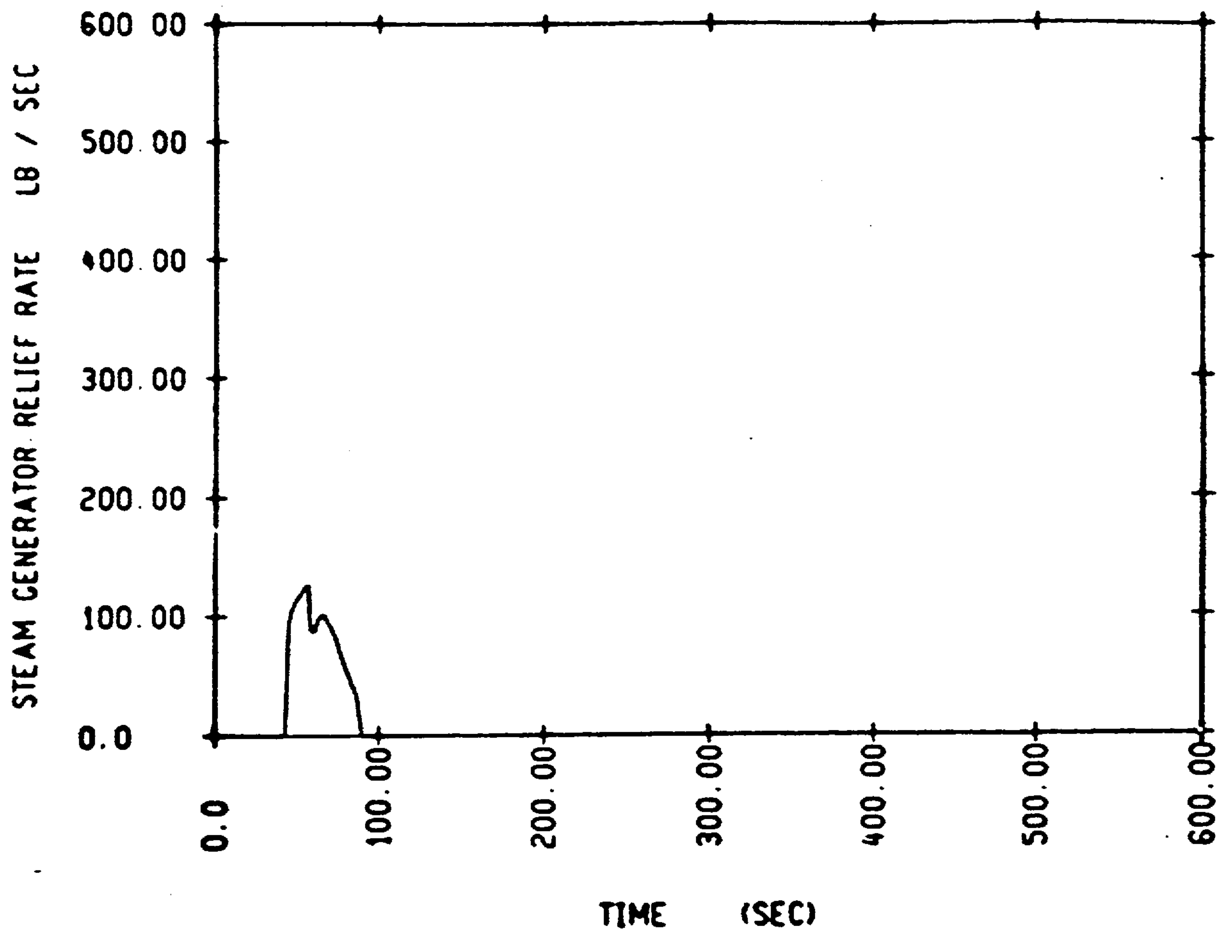
TOTAL LOSS OF NORMAL FEEDWATER
ATWS
REFERENCE CASE
99% MTC

Figure 5.2-20



TOTAL LOSS OF NORMAL FEEDWATER
ATWS
REFERENCE CASE
99% MTC

Figure 5.2-21



TOTAL LOSS OF NORMAL FEEDWATER
ATWS
REFERENCE CASE
99% MTC

Figure 5.2-22