Omaha Public Power District
Nuclear Analysis
Reload Core Analysis Methodology

# Transient and Accident Methods and Verification

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ABSTRACT

This document is a Topical Report describing Omaha Public Power District's reload core transient and accident methods for application to Fort Calhoun Station Unit No. 1.

The report addresses the District's transient and accident analysis methodology and its application to the analysis of reload cores. In addition, comparisons of results using the NSSS simulation code to results from experimental measurements and independent calculations are provided.

## Proprietary Data Clause

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# 1.0 INTRODUCTION AND SUMMARY

This report discusses the methodology the Omaha Public Power District utilizes to analyze transients and accidents for reload cores. In addition, the report discusses the District's verification of the CE NSSS simulator CESEC for Fort Calhoun Station transients. The purpose of this verification is to demonstrate the District's ability to properly utilize the CESEC code.

The District's transient and accident analysis methodology for reload cores is based upon the reanalysis of those Updated Safety Analysis Report (USAR), Chapter 14 events whose consequences may be adversely affected by changes in parameters associated with any reload core. The USAR Chapter 14 events which must be considered during a reload core analysis are discussed in Section 2.0. Section 3.0 discusses the transient analyses which determine certain parameters specified in the Technical Specifications. The District's transient analysis models are discussed in Section 4.0. The District's application of these transient analysis models to the various Chapter 14 events is discussed in Section 5.0. The verification of the NSSS simulator model used by the District is discussed in Section 6.0. References are provided in Section 7.0.

# 2.0 CHAPTER 14 EVENTS CONSIDERED IN THE RELOAD CORE ANALYSES

This section discusses the criteria utilized to determine if a Chapter 14 event need be considered in reload core analyses. Each event which is not formally considered in a reload core analysis is discussed and the reasons given for not normally including the event in the reload core analyses. The methodology applied to these events will not be discussed in this report.

## 2.1 Criteria

The criterion used to determine the events considered in reload core analyses is that changes in various neutronics parameters

2.1 Criteria (Continued)

adversely effect the safety analyses of these events. The core parameters considered are the pin peaking factors, $F_R$ and $F_{xy}$, the Moderator Temperature Coefficient (MTC), the Fuel Temperature Coefficient (FTC) or Doppler Coefficient, the boron concentration, the inverse boron worth, the neutron kinetics parameters, the CEA reactivity worth and the cooldown reactivity associated with a steam line break. If these parameters change such that the previously reported results for a Chapter 14 event are no longer conservative, then this event must be reanalyzed. If these parameters are conservative with respect to the values assumed in the referenced safety analyses, the criteria of 10 CFR 50.59 are met and this event is not reanalyzed. If a change in some of the parameters may cause the results of a safety analyses to be nonconservative, the event is reanalyzed. If the criteria for the event are still met, then the requirements of 10 CFR 50.59 are satisfied. The event is reported as being reanalyzed and that it has been determined that no unreviewed safety question exists for the event. In some cases it may be possible that an event is reanalyzed and it is determined that an unreviewed safety question exists. In these cases the analysis for these events are submitted. In addition, any safety analyses which are performed as a result of a change in the Technical Specifications are reported as part of the supporting documentation for a Facility License Change.

Criteria not directly associated with the reload core but which may be considered in a reload analysis are changes to plant systems which would take place during a refueling and would first be utilized during the operation of the subsequent core. In cases, where either physical modifications or modifications in operating procedures are made such that they do impact the safety analyses, the results of the revised safety analyses are reported in a reload core analysis. This methodology report does not consider the methodology that is required to analyze all events which could be affected by this criteria, rather, if

2.1 Criteria (Continued)

submittals are made which require analyses of events other than those discussed in this report, revisions to this methodology report will be made to incorporate the methodology used for those events.

## 2.2 USAR, Chapter 14, Safety Analysis Events Not Considered in Reload Core Analyses

This section discusses the USAR, Section 14, safety analyses which are not normally considered in a reload core analysis. The USAR section is discussed and the reasons for not including it in the scope of these analyses is discussed. Typically, the reasons for not analyzing these events are that the operating modes considered in the events are no longer allowable at Fort Calhoun Station, the event is not associated with any core parameters or the event is analyzed by a fuel vendor for the District.

### 2.2.1 Malpositioning of Part-Length CEAs

This event is not analyzed in the reload core analysis because the use of the part-length CEAs is prohibited by the Technical Specifications. In addition, the drop of a part-length CEA is less severe than the drop of a full-length CEA.

### 2.2.2 Idle-Loop Startup Incident

This event is not analyzed because part-loop operation is not permitted by the Fort Calhoun Technical Specifications.

## 2.2 USAR, Chapter 14, Safety Analysis Events Not Considered in Reload Core Analyses (Continued)

### 2.2.3 Turbine Generator Overspeed Incident

This event is an analysis of the consequences of a turbine wheel failure and is unrelated to any reload core changes.

### 2.2.4 Loss of Load

The loss of load to both generators is assessed to determine if:

A. The pressurizer safety valves limit the reactor coolant system pressure to a value below 110% of design pressure (2750 psia) in accordance with Section III of the ASME Boiler and Pressure Vessel Code, and sufficient thermal margin is maintained in the hot fuel assembly to assure that Departure from Nucleate Boiling (DNB) does not occur throughout the transient. This event is not analyzed with respect to the first criteria since the relief capacity of the pressurizer safety valves does not change and the initial energy contained in the reactor coolant system will not change unless power level is raised above 1500 MW or the reactor coolant system inlet temperature is significantly increased. Section 14.9 of the USAR reports that the DNBR for the loss of load transient never decreases below the initial value considered in the analysis. Therefore, it is concluded that any change in a parameter which could effect the DNBR for this event would much more significantly effect other events and that it is not necessary to analyze this event with respect to DNBR criteria.

## 2.2 USAR, Chapter 14, Safety Analysis Events Not Considered in Reload Core Analyses (Continued)

### 2.2.4 Loss of Load (Continued)

The loss of load to one steam generator is discussed in this methodology report as one of the asymmetric steam generator transients.

### 2.2.5 Malfunctions of the Feedwater System

The analyses which are reported in USAR, Section 14.10 Malfunctions of the Feedwater System, are the total loss of feedwater flow and the loss of feedwater heating. The results of the total loss of feedwater flow show that the minimum DNBR does not decrease below its initial steady state value and that no safety limits are approached during the event. Therefore, this event is not reanalyzed in a reload core analysis.

The loss of feedwater heating is the most adverse feedwater malfunction in terms of cooling on the RCS. This event, like the excess load event, is more limiting at EOC. This event has the same effect on the primary system as a small increase in turbine demand which is not matched by an increase in core power. As a result, the DNBR degradation associated with this event is less severe than that for the excess load where a large effective increase in turbine demand is analyzed. The excess load event analysis is reported elsewhere in this document.

### 2.2.6 Steam Generator Tube Rupture Incident

The steam generator tube rupture incident is analyzed to determine if the off site dose acceptance criteria of 10 CFR Part 100 is met. The analysis is a radio-

2.2 USAR, Chapter 14, Safety Analysis Events Not Considered in Reload Core Analyses (Continued)

2.2.6 Steam Generator Tube Rupture Incident

active material release analysis based upon 1% failed fuel within the core. It is not dependent upon any reload core analysis related parameters, therefore, it is not analyzed in the reload core analysis. In the future, the steam generator tube rupture incident analysis may be verified for high burnup fuel.

2.2.7 Loss of Coolant Accident

The loss of coolant accident as reported in USAR, Section 14.15, is analyzed for the District by ENC and CE. The large break analysis was performed by ENC, the small break analysis was performed by CE. The District confirms the assumptions used in these analyses are valid for each reload core. If reanalysis is required, the reanalysis is done by a nuclear fuel vendor. The District does not perform any loss of coolant accident analyses.

2.2.8 Containment Pressure Analysis

Containment pressure analysis is dependent upon the initial liquid mass and energy contained in the primary or secondary system. Since these parameters do not change when the core is refueled, the containment pressure analysis is not done in a reload core analysis.

2.2.9 Generation of Hydrogen in Containment

The generation of hydrogen in containment analysis is independent of any reload core parameters, therefore, the analysis is not performed during the course of a reload core analysis.

## 2.0 CHAPTER 14 EVENTS CONSIDERED IN THE RELOAD CORE ANALYSES (Continued)

### 2.2 USAR, Chapter 14, Safety Analysis Events Not Considered in Reload Core Analyses (Continued)

#### 2.2.10 Fuel Handling Accident

The fuel handling accident is a function of the isotopic inventory contained in the fuel pins. This is not normally considered in a reload core analysis, however, it may be necessary to reconsider this analyses for high burnup fuel.

#### 2.2.11 Gas Decay Tank Rupture

The gas decay tank rupture is independent of any parameters associated with refueling the core. Therefore, the analysis is not performed during a normal reload core analysis.

#### 2.2.12 Waste Liquid Incident

The waste liquid incident analysis is not affected by refueling the core. Therefore, the waste liquid incident analysis is not performed in the course of a normal reload core analysis.

### 2.3 USAR, Section 14, Events Considered in a Reload Core Analysis

The reload core analysis consists of analyzing several events which are considered in the USAR and two events which previously were not analyzed in the USAR. These events are analyzed in accordance with the criteria discussed in this report and to determine if an unreviewed safety question exists for a reload core. The USAR Chapter 14 events considered in a reload core analysis are the Control Element Assembly Withdrawal (CEAW) incident, the boron dilution incident, the Control Element Assembly (CEA) drop incident, the loss of coolant flow incident, the excess load incident, the steam line break accident, the CEA

2.3 USAR, Section 14, Events Considered in a Reload Core Analysis (Continued)

ejection accident and the seized rotor accident. In addition, analyses are performed for incidents resulting from the malfunction of one steam generator and for the RCS depressurization incident. The analysis for each of these events will be discussed in detail in Section 5.0 of this report.

## 3.0 TRANSIENT AND ACCIDENT ANALYSIS AND TECHNICAL SPECIFICATIONS

Results of transient and accident analyses are used in the Technical Specifications in two ways. The first way is that values from the Technical Specifications are included in the initial conditions of the transient analyses. These Technical Specifications guarantee that the various transient and accident analysis acceptance criteria will not be exceeded if the reactor is operated within the bounds of these Technical Specifications. Technical Specifications of this type include the limits on $F_R$, $F_{xy}$, the PDIL and the Moderator Temperature Coefficient.

The second type of values factored into the Technical Specifications are those that are determined by transient analysis. These parameters consist of the transient response term applied to the TM/LP equation, the minimum required shutdown margin, the linear heat rate LCO and the DNBR LCO. The transient response term applied to the TM/LP equation in the Technical Specifications is a result of the analysis of the RCS depressurization event or excess load event. The minimum required shutdown margin at hot shutdown conditions is determined by the steam line break event. This value is also confirmed for the boron dilution event. The minimum required shutdown margin for cold shutdown and refueling shutdown conditions is determined by the boron dilution event or the five percent subcriticality requirement for refueling. The values used in the linear heat rate LCO are typically determined by the loss of coolant accident. These values are also confirmed for the dropped CEA event. The LCO on DNBR margin is calculated based on results from the dropped CEA event, the loss of four pump flow analysis or the CEA withdrawal analysis.

## 4.0 TRANSIENT AND ACCIDENT ANALYSIS MODELS

The District utilizes the latest version of the CESEC code (CESEC-III and hereafter referred to as CESEC) in the simulation of plant response to non-LOCA initiating events. The District utilizes the CETOP and TORC computer codes for calculation of DNBR during these events.

### 4.1 Plant Simulation Model

The District utilizes the CESEC digital computer code, References 4-1 through 4-10, to provide the simulation of the Fort Calhoun Station nuclear steam supply system. The program calculates the plant response to non-LOCA initiating events for a wide range of operating conditions. The information presented in Reference 4-9 supercedes information provided in References 4-1 through 4-8. Additional information on the model is provided in Reference 4-10. The CESEC program, which numerically integrates one dimensional mass and energy conservation equations, assumes a node/flow-path network to model the NSSS. The primary system components considered in the code include the reactor vessel, the reactor core, the primary coolant loops, the pressurizer, the steam generators and the reactor coolant pumps. The secondary system components include the secondary side of the steam generators, the main steam system, the feedwater system and the various steam control valves. In addition, the program models some of the control and plant protection systems.

The code self initializes for any given, but constant, set of reactor power level, reactor coolant flow rate and steam generator power sharing. During the transient calculations, the time rate of change in the system pressure and enthalpy are obtained from solution of the conservation equations. These derivatives are then numerically integrated in time under the assumption of thermal equilibrium to give the system pressure and nodal enthalpies. The fluid states recognized by the code are subcooled and saturated; superheating is allowed in the pressurizer. Fluid in the reactor coolant system is assumed to

4.1 Plant Simulation Model (Continued)

be homogenous. Reference 4-9 provides a description of the CESEC code, including the major models, and the input, output and plot packages.

The pressurizer model is described in Reference 4-9 and further discussed in Reference 4-10. The District utilizes the wall heat transfer model to permit simulation of voiding in any node in which steam formation occurs. Voiding may occur in events such as a steam line break or steam generator tube rupture. Nodalization of the closure head, described in Reference 4-9 and further discussed in Reference 4-10, allows for the formation of a void in the upper head region when the pressurizer empties. Flow to the closure head is terminated in simulations of those events in which natural circulation occurs and in those events such as the steam line break where this action delays safety injection.

The capabilities and limitations of the CESEC code are discussed in References 4-9 and 4-10. The District's CESEC model of Fort Calhoun Station is valid for the transients discussed in Section 5 of this report, with the exception of the CEA Ejection Analysis and LOCA Analysis. The CESEC model is also valid for analysis of the loss of load, malfunctions of the feedwater system and the steam generator tube rupture incidents.

The CESEC code is maintained by CE on the CE computer system in Windsor, Connecticut. The District accesses the code through a time sharing system. CE maintains all documentation and quality assurance programs related to this code.

4.2 DNBR Analysis Models

The DNBR analysis is currently performed using either the TORC code, Reference 4-11, or both the TORC and CETOP codes, Reference 4-12. The TORC code is used as a benchmark for the CETOP

4.2 DNBR Analysis Models (Continued)

code model. TORC solves the conservation equations, as applied to a three-dimensional representation of the open lattice core, to determine the local coolant conditions at all points in the core. Lateral transfer of mass and energy between neighboring flow channels (open core effects) are accounted for in the calculation of local coolant conditions. These coolant conditions are then used with a Critical Heat Flux (CHF) correlation supplied as a code subroutine to determine the minimum value of DNBR for the reactor core. The CE-1 CHF correlation (References 4-13 and 4-14) is used for the Fort Calhoun reactor as approved in Reference 4-15. The Detailed TORC code is used directly in the seized rotor analysis.

The CETOP code has been developed to reduce the computer time needed for thermal hydraulic analyses while retaining all of the capabilities of the TORC design model. The CETOP model provides an additional simplification to the conservation equations due to the specific geometry of the model. A complete description of the CETOP code is contained in Reference 4-12 and a description of the District's application of the CETOP code is contained in Reference 4-16.

The fraction of inlet flow to the hot assembly in the CETOP model is adjusted such that the model yields appropriate MDNBR results when compared to the results of the TORC analysis for a specified range of operating conditions.

The CETOP code is used to calculate DNBR for all transient analyses discussed in Section 5 with the exception of the seized rotor analysis.

4.3 Application of Uncertainties

Uncertainties are taken into account either by deterministic or statistical methods. The deterministic method applies all un-

4.3 Application of Uncertainties (Continued)

certainties adversely and simultaneously when calculating the approach to a limit.

Uncertainties in DNBR calculations are taken into account by statistical methods. The statistical method takes into account the likelihood that the uncertainties will all be adverse. The statistical method is discussed in Reference 4-17. In this method the impact of component uncertainties on DNBR is assessed and the DNBR SAFDL is increased to include the effects of the uncertainties. Since the uncertainties are accomodated by the increased DNBR SAFDL in the statistical method, engineering factors are not applied to the DNBR analysis model. The statistical method of applying uncertainties is applied to the CEA withdrawal, CEA drop, loss of RCS flow, excess load, seized rotor and asymmetric steam generator event DNBR calculations.

## 5.0 TRANSIENT AND ACCIDENT ANALYSIS METHODS

This section addresses the evaluation of the various transients and accidents that are performed during a reload core analysis. Specific methods are described for each transient and accident. For each accident or transient the following material is described:

A. Definition of the Event - A brief description of the causes, consequences, and RPS trips involved in the incident.

B. Analysis Criteria - A brief description of the classification of the event and the Specified Acceptable Fuel Design Limit (SAFDL) or the offsite dose criteria which must be met.

C. Objectives of the Analysis - A brief description of the methods that are used to assure that the criteria of the analysis are met.

D. Key Parameters and Analysis Assumptions - A description of the key parameters and assumptions used in the analysis.

5.0 TRANSIENT AND ACCIDENT ANALYSIS METHODS (Continued)

E. Analysis Method - A description of the methodology employed by the District to analyze the event.

F. Analysis Results and 10 CFR 50.59 Criteria - The expected results of the analysis and a discussion of the methods used to determine if the event meets the criteria of 10 CFR 50.59.

G. Conservatism of Results - A description of the conservatism of the analysis.

The values of the trip setpoints and trip delay times used in these analyses are shown in Table 5.0-1.

## 5.1 CEA Withdrawal

### 5.1.1 Definition of the Event

A sequential CEA Group Withdrawal Event is assumed to occur as a result of a failure of the control element assembly drive mechanism control system or by operator error. The CEA Block System eliminates the possibility of an out of sequence bank withdrawal or single CEA withdrawal due to a single failure.

Any controlled or unplanned withdrawals of the CEA's results in a positive reactivity addition which causes the core power, core average heat flux and reactor coolant system temperature and pressure to rise and in turn decrease the DNB and Linear Heat Rate (LHR) margins. The pressure increase, if large enough, activates the pressurizer sprays which mitigate the pressure rise. In the presence of a positive Moderator Temperature Coefficient (MTC) of reactivity, the temperature increase results in an additional positive reactivity addition further decreasing the margin to the DNB and LHR limits.

Table 5.0-1

REACTOR PROTECTIVE SYSTEM TRIPS AND SAFETY INJECTION

| | | | Used in Analysis | |
|---|---|---|---|---|
| Trip | Setpoint | Uncertainty | Delay Time (Sec) | Setpoint |
| High Rate-of-Change of Power | 2.6 dec/min | ±0.5 dec/min | 0.4 | 2.1 dec/min |
| High Power Level | 107% | 5.0% | 0.4 | 112% |
| Variable High Power Level | 9.1% above set power level to a low of 19.1% | 0.9% | 0.4 | 10% above initial power level |
| Low Reactor Coolant Flow | 95% | ±2% | 0.65 | 93% |
| High Pressurizer Pressure | 2400 psia | ±22 psi | 0.9 | 2422 psia |
| Thermal Margin/Low Pressure(1) | 1750 psia | ±22 psi | 0.9 | 1728 psia |
| Low Steam Generator Pressure | 500 psia | ±22 psi | 0.9 | 478 psia |
| Low Steam Generator Water Level | 31.2% of narrow range span | ±10 in. (5.7% of narrow range span) | 0.9 | 25.5% of span |
| Containment Pressure High | 5 psig | ±0.4 psi | 0.1 | 5.4 psig |
| High Pressure Safety Injection | 1600 psia | ±22 psi | 12(2) | 1578 psia |

(1) Values represent the low limit of the thermal margin/low pressure trip. The setpoint of this trip is discussed in Reference 5-2.

(2) Pump start - loop valve opening time.

Withdrawal of the CEA's causes the axial power distribution to shift to the top of the core. The associated increase in the axial peak is partially compensated by the corresponding decrease in the integrated radial peaking factor. The magnitude of the 3-D peak change depends primarily on the initial CEA configuration and axial power distribution.

The withdrawal of the CEA's causes the neutron flux as measured by the excore detectors to be decalibrated due to CEA motion, i.e., rod shadowing effects. This decalibration of excore detectors, however, is partially compensated by neutron attenuation rising from moderator density changes (i.e., temperature shadowing effects).

As the core power and heat flux increase, a reactor trip on high power, variable high power, or Thermal Margin/Low Pressure may occur to terminate the event depending on the initial operating conditions and rate of reactivity addition. Other potential trips include the axial power distribution and high pressurizer pressure trips. If a trip occurs, the CEA's drop into the core and insert negative reactivity which quickly terminates further margin degradation. If no trip occurs and corrective action is not taken by the operators, the CEA's fully withdraw and the NSSS achieves a new steady state equilibrium with higher power, temperature, peak linear heat rate and lower hot channel DNBR value.

#### 5.1.2 Analysis Criteria

The CEA Withdrawal (CEAW) event is classified as an Anticipated Operational Occurrence (AOO) for which the

following criteria must be met:

a. The transient minimum DNBR is greater than the 95/95 confidence interval limit for the CE-1 correlation, and

b. the Peak Linear Heat Generation Rate (PLHGR) does not exceed 21 kw/ft.

### 5.1.3 Objectives of the Analysis

The objectives of the analysis performed for the "at power" CEAW event is to calculate the Required Overpower Margin (ROPM) which must be factored into the setpoint analysis.

The objective of the analysis for the hot zero power CEAW event is to demonstrate that the Variable High Power Trip (VHPT) is initiated in time to insure that the analysis criteria are met.

### 5.1.4 Key Parameters and Analysis Assumptions

The initial conditions assumed in the CEAW analysis are shown in Table 5.1-1. The reactor state parameters of primary importance in calculating the margin degradation are:

1. CEA withdrawal rate* (i.e., reactivity insertion rate),

2. Gap thermal conductivity (HGAP),

*NOTE: The term CEA withdrawal rate and CEA reactivity insertion rate are used interchangably in this report.

3. Initial power level,

4. Flux power level determined from the excore detector response during the transient,

5. The moderator temperature coefficient reactivity, and

6. Changes in the axial power distribution and planar and integrated radial peaking factor during the transient.

The excore responses for each initial power level analyzed are based on the CEA insertions allowed by the Power Dependent Insertion Limit (PDIL) at the selected power level, the changes in CEA position prior to trip, and the corresponding rod shadowing and temperature attenuation (shadowing) factors.

For the CEAW cases where combinations of parameters result in a reactor trip, the scram reactivity versus insertion characteristics are assumed to be those associated with the core average axial power distribution peaked at the bottom of the core. The scram reactivity versus insertion characteristics associated with this bottom peak shape minimize the amount of negative reactivity inserted during initial portions of the scram following a reactor trip.

All control systems except the pressurizer pressure control system and the pressurizer level control system are assumed to be in a manual mode. These are the most adverse operating modes for this event. The pressuri-

Table 5.1-1

Initial Conditions Assumed in CEAW Event Analysis

| Parameter | Units | Value |
|---|---|---|
| Initial Core Power Level | MWt | 1-1500* |
| Initial Core Inlet Coolant Temperature | °F | 532-545* |
| Moderator Temperature Coefficient | $X10^{-4}$ $\Delta\rho$/°F | Tech. Spec. Range |
| Initial RCS Pressure | psia | Minimum allowed by* Tech. Specs. |
| Fuel Temperature Coefficient | $X10^{-4}$ $\Delta\rho$/°F | Least Negative Predicted During A Cycle |
| Initial Core Mass Velocity | $10^6$ lbm/hr | Minimum allowed by* Tech. Specs. |
| Fuel Temp. Coeff. Uncertainty | % | -15.0 |
| Gap Thermal Conductivity | BTU/hr-$ft^2$ | [ ] |
| CEA Differential Worth | *$10^{-4}$ /inch | [ ] |
| CEA Withdrawal Speed | in/min | 46.0 |
| Radial Peaks | | Maximum Allowed by Tech. Specs. for a Given Initial Power Level |
| Scram Reactivity | % | Minimum Predicted During a Cycle |
| High Power Trip Analysis Setpoint | % of 1500 MWt | 112.0 |
| Variable High Power Trip Analysis Setpoint | % Above Initial Power Level | 10.0 |
| Temperature Shadowing Factor | % Power/°F | [ ] |

* For DNBR calculations, effects of uncertainties are combined statistically.

#### 5.1.4 Key Parameters and Analysis Assumptions (Continued)

zer pressure control system and pressurizer level control system are assumed to be in the automatic mode since the actuation of these systems minimizes a rise in the coolant system pressure. The net effect, is to delay a reactor trip until a high power trip is initiated. This allows the transient increases in power, heat flux and coolant temperature to proceed for a longer period of time. In addition, minimizing the pressure increase is conservative in the margin degradation calculations since increases in pressure would offset some of the DNB margin degradation caused by increases in the core heat flux and coolant temperatures.

#### 5.1.5 Analysis Methodology

The methodology used for analysis of the CEAW event is described in CEN-121(B)-P, Reference 5-1. The District does not perform all parametric analyses discussed in Reference 5-1 for Fort Calhoun Station. Rather, the District utilizes the analyses performed in Reference 5-1 to limit the number of analyses necessary for Fort Calhoun Station. Specifically, the District utilizes the result that [

] In addition, the result from Reference 5-1 that [

] when combined with [

] can be used to perform sensitivity analyses on the CEA withdrawal rate to achieve [

] is utilized.

5.1 CEA Withdrawal (Continued)

5.1.5 Analysis Methodology (Continued)

The rod shadowing factors for the Fort Calhoun Station full power case with Bank 4 inserted are the inverse of the rod shadowing factors used in Reference 5-1 (The rod shadowing factors for Fort Calhoun Station are such that the excore detectors see more flux when the rods are withdrawn than when they are inserted. Therefore, the [

] during a full power CEA withdrawal event). Because of this effect, it may be necessary to assume a [ ] in order to achieve [

]

The analysis at intermediate power levels is the same as documented in Reference 5-1.

The hot zero power CEAW event is analyzed assuming the variable high power trip is initiated at 29.1% (19.1% plus 10% uncertainty) of rated thermal power. In addition, the analysis assumes that the maximum CEA withdrawal rate is combined with the maximum differential rod worth. This case is analyzed using CESEC and the minimum DNBR is calculated using CETOP using the assumptions discussed in Reference 5-1.

The CEAW event analyzed to determine the closest approach to the fuel centerline melt SAFDL assumes those values of the CEAW rate and $H_{gap}$ discussed in Reference 5-1. This combination of CEAW rate and $H_{gap}$ was used to determine the PLHGR at all power levels.

5.1.6 Typical Analysis Results and 10 CFR 50.59 Criteria

The results of the analyses of the CEAW event for Fort Calhoun Station at full power and at intermediate power

5.0 TRANSIENT AND ACCIDENT ANALYSIS METHODS (Continued)

5.1 CEA Withdrawal (Continued)

5.1.6 Typical Analysis Results and 10 CFR 50.59 Criteria (Continued)

levels are expected to be similar to those presented in Reference 5-1. The results of the hot zero power CEA withdrawal analysis are expected to be similar to those discussed in the Cycle 8 reload submittal and the 1983 update of the USAR. The 10 CFR 50.59 criteria are met if the analysis for the full power and intermediate power level CEAW events shows that the required over-power margin for these events is less than the available overpower margin required by the current Technical Specification DNB and PLHGR LCO's. The 10 CFR 50.59 criteria is satisfied for the hot zero power CEAW event if the minimum DNBR is greater than that reported in the latest submitted analysis.

5.1.7 Conservatism of Results

Conservatism of the results of the CEAW incident analyses is discussed in Reference 5-1 for the full power, intermediate power level and hot zero power cases.

5.2 Boron Dilution Incident

5.2.1. Definition of Event

Boron dilution is a manual operation, conducted under strict procedural controls which specify permissible limits on the rate and magnitude of any required change in boron concentration. Boron concentration in the reactor coolant system can be decreased by either controlled addition of unborated makeup water with a corresponding removal of reactor coolant or by using

5.2 Boron Dilution Incident (Continued)

5.2.1. Definition of Event (Continued)

the deborating ion exchangers. To effect boron dilution the makeup controller mode selector of the chemical and volume control system (CVCS) must be set to "dilute" and then the demineralized water batch quantity selector set for the desired quantity. When the specific amount has been injected, the demineralized water control valve is shut automatically. An inadvertent boron dilution can occur only if there is a combination of operator error and a CVCS malfunction occurring at the same time. No RPS trips are assumed to terminate this incident.

5.2.2 Analysis Criteria

The boron dilution event is classified as an AOO for which the following criteria cannot be exceeded:

A. DNBR greater than the 95/95 confidence interval limit using the CE-1 correlation, and

B. The PLHGR less than 21 kw/ft.

5.2.3 Objectives of the Analysis

The DNBR and PLHGR criteria are met by showing that sufficient time exists for the operator to take corrective action to terminate the event prior to exceeding the SAFDLs. This is accomplished by calculating the time interval in which the minimum Technical Specification shutdown margin is lost. The acceptable time interval for the operator to take corrective actions before shutdown margin is lost are 15 minutes for Modes 2, 3 and 4 and 30 minutes in Mode 5.

#### 5.2.4 Key Parameters and Analysis Assumptions

The boron dilution event at power (Mode 1) is bounded by the faster reactivity insertion rate of the CEA withdrawal event and it lacks the local power peaking associated with the withdrawn CEA. For the boron dilution event in Modes 2 through 5, it is assumed that all three charging pumps are operating at their maximum capacity for a total charging rate of 120 gpm. For the dilution at hot standby (Mode 2) the event is assumed to be initiated at the Technical Specification hot shutdown margin requirement at 532°F. The reactor coolant system is 5,506 cubic feet.

The boron dilution incident at hot shutdown (Mode 3) is assumed to be initiated from the Technical Specification shutdown margin requirement at 210°F. The boron dilution incident cold shutdown (Mode 4) is initiated from the Technical Specification minimum shutdown margin requirement at 68°F. The analysis is conducted for two RCS volumes, one of 5,506 cubic feet and the other of 2,036 cubic feet, which corresponds to the volume for a refueling operation condition. The analysis for the lower volume cold shutdown condition assumes that shutdown groups A and B are withdrawn from the core and all regulating groups are inserted in the core with the exception of the most reactive rod which is assumed to be stuck in its fully withdrawn position. These assumptions are consistent with the Technical Specifications for cold shutdown conditions. The boron dilution event during refueling is analyzed assuming that reactor refueling has just been completed and the head is in place but the coolant volume is sufficient to only fill the reactor vessel to the bottom of the piping nozzles

5.2.4 Key Parameters and Analysis Assumptions (Continued)

(2,036 cubic feet) and the minimum permissible boron concentration allowed by Technical Specification for refueling exists. All CEA's are withdrawn from the core.

These assumptions represent shutdown conditions for the various modes wherein the core reactivity is greatest, the water volume and total boron content is at a minimum, and the rate of dilution is as large as possible. Hence, these conditions represent the minimum time to achieve inadvertent criticality in the event of an uncontrolled boron dilution.

### 5.2.5 Analysis Methods

The method used to calculate the dilution time to criticality from Modes 2 through 5 is through the use of the following equation:

$$\Delta t_{crit} = \tau_{BD} \cdot \ln \left[ \qquad \right]$$

Where $\tau_{BD}$ = boron dilution time constant, which is a function of RCS volume and temperature (sec)

$C_B$ = critical boron concentration (ppm)

SDM = shutdown margin (%$\Delta\rho$)

IBW = inverse boron worth (ppm/%$\Delta\rho$)

As can be seen from this equation, the dilution time to criticality is minimized with a greater critical boron concentration, a smaller inverse boron worth, or a smaller $\tau_{BD}$.

5.2 Boron Dilution Incident (Continued)

5.2.6 Analysis Results and 10 CFR 50.59 Criteria

The analysis results are similar to those reported in the Cycle 8 safety analysis report and in the 1983 update of the USAR. The criteria of 10 CFR 50.59 are satisfied if the Technical Specification requirements on shutdown margin and the refueling boron concentration is unchanged as a result of this analysis.

5.2.7 Conservatism of Results

Because of the procedures involved in the boron dilution and the numerous alarm indications available to the operator, the probability of a sustained or erroneous boron dilution is very low. There is usually a large interval between the calculated time and the time limit for the boron dilution at hot standby and hot shutdown modes. Therefore, the results show considerable margin to the limit. The calculated time to critical for the boron dilution at cold shutdown with the minimum RCS volume is reasonably close to the acceptance criteria; however, the event is analyzed with only shutdown groups A and B being fully withdrawn from the core. Cold shutdown is normally achieved with the shutdown groups A and B fully inserted in the core and, therefore, the core has a much lower $k_{eff}$ than assumed in the analysis. The boron dilution at refueling is conservative since it is improbable that more than a few CEA's will be removed at any one time during a refueling and the approach to critical following refueling is done under strict administrative control with only one bank of CEA's removed at a time. The analysis assumes that all CEA's are withdrawn from the core.

## 5.3 Control Element Assembly Drop Incident

### 5.3.1 Definition of Event

The control element assembly (CEA) drop incident is defined as the inadvertent release of a CEA causing it to drop into the reactor core. The CEA drive is of the rack and pinion type with the drive shaft running parallel to and driving the rack through a pinion gear and a set of bevel gears. The drive mechanism is equipped with a mechanical brake which maintains the position of the CEA. The CEA drop may occur due to an inadvertent interruption of power to the CEA drive magnetic clutch or an electrical or mechanical failure of the mechanical brake in the CEA drive mechanism when the CEA is being moved.

The full-length CEA drop event is classified as an AOO which does not require an RPS trip to provide protection against exceeding the SAFDLs. The CEA drop results in a redistribution of the core radial power distribution and an increase in the radial peaks which are not directly monitored by the RPS and which are not among those analyzed in determining the DNB and LHR LCOs and LSSSs. As such, initial steady state margin must be built into the Technical Specification LCOs to allow the reactor to "ride out" the event without exceeding the DNBR and LHR SAFDLs.

### 5.3.2 Analysis Criteria

The full-length CEA drop event is classified as an Anticipated Operational Occurrence for which the following criteria must be met: a) The transient minimum DNBR must be greater than or equal to the 95/95 confidence interval limit, using the CE-1 correlation, b) The Peak Linear Heat Rate (PLHR) must be less than or equal to 21 kw/ft.

# 5.0 TRANSIENT AND ACCIDENT ANALYSIS METHODS (Continued)

## 5.3 Control Element Assembly Drop Incident (Continued)

### 5.3.3 Objectives of the Analysis

The objective of the analysis is to determine the Required Overpower Margin (ROPM) which must be built into the LCOs to assure the DNBR and LHR SAFDLs are not exceeded for the CEA drop which produces the highest distortion in the hot channel power distribution. Since the ROPM is dependent upon initial power level, rod configuration and axial shape index, an analysis parametric in these variables is performed.

### 5.3.4 Key Parameters and Analysis Assumptions

Table 5.3.4-1 contains a list of the key parameters assumed in the full-length CEA drop analysis. Assumptions used in the analysis include:

1. [ ]

2. The rod block system is assumed to prevent any other rod motion during the transient.

3. The turbine admission valves are maintained at a constant position during the transient. This is because the turbine admission valve position is set manually at Fort Calhoun Station and, therefore, the turbine admission valves will not automatically open in response to a reduced electrical generation output.

Table 5.3.4-1

KEY PARAMETERS ASSUMED IN THE FULL LENGTH CEA DROP ANALYSIS

| Parameter | Units | Value |
|---|---|---|
| Initial Core Power Level | $MW_t$ | 750 to 1500† |
| Initial Core Inlet Temperature | $^oF$ | Maximum allowed† by Tech. Specs. |
| Initial RCS Pressure | psia | Minimum allowed† by Tech. Specs. |
| Initial Core Mass Flow Rate | $*10^6$ lbm/hr | Minimum allowed† by Tech. Specs. |
| Moderator Temperature Coefficient | $*10^{-4}\Delta\rho/^oF$ | Most negative allowed by Tech. Specs. |
| CEA Insertion | % Insertion | Maximum allowed by Tech. Specs. |
| Radial Peaking Distortion Factor | | Maximum value predicted during core life |
| Dropped CEA Worth | $\%\Delta\rho$ | [ ] |
| Core Average $H_{gap}$ | $BTU/hr\text{-}Ft^2\text{-}^oF$ | [ ] |
| Fuel Temperature Coefficient | $*10^{-4}\Delta\rho/^oF$ | [ |

†For DNBR calculations, the effects of uncertainties on these parameters are combined statistically.

5.3 Control Element Assembly Drop Incident (Continued)

### 5.3.5 Analysis Method

The analysis methods utilized by the District to analyze the CEA drop incident are discussed in Section 8 of Reference 5-2.

### 5.3.6 Analysis Results and 10 CFR 50.59 Criteria

Typical analysis results are contained in Section 8 of Reference 5-2 and in the 1983 update of the Fort Calhoun Station Unit No. 1 USAR. The criteria of 10 CFR 50.59 are met if the required overpower margin calculated for this incident is less than the overpower margin being maintained by the current Technical Specifications.

### 5.3.7 Conservatism of Results

The following areas of conservatism are included in the analysis.

1. The most negative moderator [ ] coefficients of reactivity are utilized because these coefficients produce the minimum RCS coolant temperature decrease.

2. The [ ] distortion factor at any time during core life is combined with the [ ] CEA worth at any time during core life.

3. The moderator temperature coefficient assumed in the analysis is the most negative value allowed by the Technical Specifications. The actual end of life value, including measurement uncertainty, is less negative.

5.3 Control Element Assembly Drop Incident (Continued)

5.3.7 Conservatism of Results (Continued)

4. [ ]

## 5.4 Four-Pump Loss of Flow Event

### 5.4.1 Definition of the Event

The four-pump loss of coolant flow event is initiated by the simultaneous loss of electrical power to all four reactor coolant pumps. The loss of AC power to reactor coolant pumps may result from either the complete loss of AC power to the plant, or the failuire of the fast transfer breakers to close after a loss of offsite power.

Reactor trip for the loss of coolant flow is initiated by a low coolant flow rate as determined by a reduction in the sum of the steam generator hot to cold leg pressure drop. This signal is compared to a setpoint which is a function of the number of reactor coolant pumps in operation (which current Technical Specifications require to be four). A reactor trip would be initiated when the flow rate drops to 93% of full flow (95% minus 2% uncertainty).

### 5.4.2 Analysis Criteria

The four-pump loss of flow event is classified as an AOO for which the transient minimum DNBR must be

5.4.2 Analysis Criteria (Continued)

greater than the 95/95 percent confidence interval limit using the CE-1 correlation.

5.4.3 Objectives of the Analysis

The objective of the analysis is to determine the required overpower margin that must be built into the DNB LCOs such that in conjunction with the low flow trip the DNBR SAFDL is not exceeded. Since the required overpower margin is dependent upon both axial shape index and the CEA rod configuration, an analysis parametric in these parameters is performed.

5.4.4 Key Parameters and Analysis Assumptions

The closest approach to the DNBR SAFDL occurs for a loss of flow event initiated from the full power conditions. Table 5.4.4-1 gives the key parameters used in this analysis. The flow coast down is calculated in the CESEC code.

5.4.5 Analysis Method

The analysis method used by the District to analyze the four-pump loss of coolant flow is discussed in Section 7 of Reference 5-2. The District utilizes the CESEC-TORC method to analyze axial power distributions characterized by both negative and positive shape indices. The STRIKIN-TORC method is not utilized by the District because of the high rotational energy of the pumps (N = 1185 rpm, I = 71,000 lb-ft$^2$/pump). The District also utilizes the [static reactivity insertion rate rather than the space time reactivity insertion rate.]

Table 5.4.4-1

KEY PARAMETERS ASSUMED IN THE LOSS OF COOLANT FLOW ANALYSIS

| Parameter | Units | Value |
|---|---|---|
| Initial Core Power Level | $MW_t$ | 1500† |
| Initial Core Inlet Temperature | $^oF$ | [ |
| Initial RCS Pressure | psia | |
| Initial Core Mass Flow Rate | $*10^6$ lbm/hr. | ] |
| Moderator Temperature Coefficient | $*10^{-4}\Delta\rho/^oF$ | Maximum allowed by Tech. Specs. |
| Fuel Temperature Coefficient | $*10^{-4}\Delta\rho/^oF$ | Least negative predicted during core life. |
| Low Flow Trip Delay Time | sec. | Maximum |
| CEA Drop Time | sec. | Maximum allowed by Tech. Specs. |
| Scram Reactivity Worth | $\Delta\rho$ | Minimum predicted during core lifetime |
| Scram Reactivity Curve | | Consistent with axial shape of interest |
| Core Average $H_{gap}$ | $BTU/hr\text{-}Ft^2\text{-}^oF$ | [ ] |

†For DNBR calculations, effects of uncertainties on these parameters were combined statistically.

5.4 Four-Pump Loss of Flow Event (Continued)

### 5.4.6 Analysis Results and 10 CFR 50.59 Criteria

Expected analysis results are presented in Section 7.1 of Reference 5-2. The main difference between these results and the results for Fort Calhoun Station is that the ROPM will be significantly reduced for Fort Calhoun Station. This is because of the higher rotational energy of the Fort Calhoun reactor coolant pumps.

The criteria of 10 CFR 50.59 are met if the required overpower margin calculated for the four-pump loss of coolant flow event is less than the overpower margin being maintained by the current Technical Specifications.

### 5.4.7 Conservatism of Results

The conservative nature of the DNBR ROPM values calculated for the four-pump loss of flow event is demonstrated by the following conservative assumptions.

1. Field measurements of the CEA magnetic clutch decay is more rapid than assumed in the safety analysis.

2. The available scram worth is higher than assumed in the safety analysis.

3. The MTC at full power is more negative than the value assumed in the safety analysis.

4. The actual CEA drop time to 90% inserted is faster than that assumed in the safety analysis.

5. The conservatism of the CETOP calculations is discussed in Section 7 of Reference 5-2.

## 5.5 Asymmetric Steam Generator Event

### 5.5.1 Definition of the Event

The asymmetric transients arising from a secondary system malfunction in one steam generator result in changes in core power distribution which are not inherently covered by the TM/LP or APD LSSS. Consequently, these events must be analyzed to determine the initial steady state thermal margin which is built into and maintained by the Technical Specification LCO such that assurance is provided that the DNBR and peak linear heat rate SAFDLs are not exceeded for these transients. The four events which effect the steam generator are:

1. Loss of load to one steam generator.

2. Loss of feedwater to one steam generator.

3. Excess feedwater to one steam generator.

4. Excess load to one steam generator.

The possible RPS trips which can occur to mitigate the consequences of these events include the low steam generator level, TM/LP, low steam generator pressure, and the asymmetric steam generator transient protection trip function (ASGTPTF). The particular trip which intervenes is dependent upon the event initiator and the initial operating conditions.

The ASGTPTF trip will be installed in the Fort Calhoun Station RPS prior to operation of Cycle 9 to reduce the margin requirements associated with these asymmetric events and to insure that these events do not become a limiting AOO for establishing initial margin which must

5.5 Asymmetric Steam Generator Event (Continued)

5.5.1 Definition of the Event (Continued)

be maintained by the LCO. A system description of the ASGTPTF is presented in Appendix B of Reference 5-2.

#### 5.5.2 Analysis Criteria

The asymmetric steam generator events are classified as AOOs for which the following criteria must be met, a) the transient minimum DNBR must be greater than or equal to the 95/95 confidence interval limit using the CE-1 correlation, and b) the peak linear heat must be less than or equal to 21 kw/ft.

#### 5.5.3 Objectives of the Analysis

The objectives of the analysis are to determine the required overpower margin that must be built into the LCO's such that in conjunction with the ASGTPTF the DNBR and PLHGR SAFDL's is not exceeded.

#### 5.5.4 Key Parameters and Analysis Assumptions

Section 7 of Reference 5-2 demonstrates that the loss of load to one steam generator (LL/1SG) is the limiting asymmetric steam generator transient for establishing initial steady state thermal margin which must be maintained by the Technical Specification LCO. Therefore, information is only provided for this asymmetric steam generator event. The key parameters used in the analysis of the LL/1SG event are given in Table 5.4.4-1. The charging pumps and proportional heater systems are assumed to be inoperable during the transient. This maximizes the pressure drop during the event. The tur-

Table 5.5.4-1

KEY PARAMETERS ASSUMED IN THE LL/1SG EVENT

| Parameter | Units | Value |
|---|---|---|
| Initial Core Power | $MW_t$ | 700 to 1500† |
| Initial Core Inlet Temperature | $^oF$ | Maximum allowed† by Tech. Specs. |
| Initial Reactor Coolant System Pressure | psia | Minimum allowed† by Tech. Specs. |
| Moderator Temperature Coefficient | $*10^{-4}\Delta\rho/^oF$ | |
| Fuel Temperature Coefficient | $*10^{-4}\Delta\rho/^oF$ | |
| Core Average $H_{gap}$ | $BTU/hr\text{-}ft^2\text{-}^oF$ | Maximum value predicted during core life. |
| Initial Core Mass Flow Rate | $*10^6$ lbm/hr. | Best estimate flow† |
| Scram Reactivity Worth | $\%\Delta\rho$ | Minimum predicted during core life. |

†For DNBR calculations, effects of uncertainties on these parameters were combined statistically.

5.5 Asymmetric Steam Generator Event (Continued)

5.5.4 Key Parameters and Analysis Assumptions (Continued)

bine admission valves are assumed to maintain a constant position throughout the event since the turbine control system at Fort Calhoun utilizes manual setting of the turbine admission valves.

### 5.5.5 Analysis Method

The method utilized by the District to analyze the LL/1SG is discussed in Section 7 of Reference 5-2.

### 5.5.6 Analysis Results and 10 CFR 50.59 Criteria

The results of the analysis for the LL/1SG event are discussed in Section 7 of Reference 5-2.

The results for Fort Calhoun Station are expected to be similar. The criteria of 10 CFR 50.59 are satisfied if the required overpower margin calculated for the LL/1SG event is less than the overpower margin being maintained by the current Technical Specifications.

## 5.6 Excess Load Incident

### 5.6.1 Definition of Event

An excess load transient is defined as any rapid increase in the steam generator steam flow other than a steam line break. Such a rapid increase in steam flow results in a power mismatch between the reactor core and the steam generator load demand. In addition, there is a decrease in the reactor coolant temperature and pressure. Under these conditions the negative moderator temperature coefficient reactivity causes an increase in core power.

5.0 TRANSIENT AND ACCIDENT ANALYSIS METHODS (Continued)

5.6 Excess Load Incident (Continued)

5.6.1 Definition of Event (Continued)

The rapid opening of the turbine admission valves or the steam dump bypass to the condenser causes an excess load event. Turbine valves are not sized to accomodate steam flow for powers much in excess of 1500 MWt. The steam dump valves and steam bypass valves to the condenser are sized to accomodate 33% and 5%, respectively, of the steam flow at 1500 MW. Therefore, the following load increase incidents are examined:

A. Rapid opening of the turbine control valves at power: The maximum increase in the steam flow due to the turbine control valves opening is limited by the turbine load limit control. The load limit control function is used to maintain load, so unless valve failure occurs, the control valves will remain where positioned.

B. Opening of all dump and bypass valves at power due to steam dump control interlock failure: The circuit between the steam dump controller and the dump valves is open when the turbine generator is on line. Accidental closing of the steam dump control interlock under full load conditions, according to the temperature program of the controller, causes full opening of the dump and bypass valves. Since the reactor coolant temperature decreases during the event, these valves will be closed again after the average reactor coolant temperature decreases to 535°F.

C. Opening of the dump and bypass valves at hot standby conditions due to low reference temperature setting in the steam dump controller: When

C. (Continued)

the plant is in hot standby conditions the dump valve controller is operative but does not act because the hot standby temperature is lower than the lowest value required to open the valves. At hot standby the reactor coolant temperature is 532°F, which is 8°F below the minimum temperature required to open the dump and bypass valves (540°F). The maximum error that can be introduced in the referenced temperature setting is limited to 70°F since a narrow range instrument is used for this purpose. Reducing the dump valve controller reference setting from 532° to 515° would result in a partial opening of the valves but as soon as the reactor coolant temperature dropped to 518°F the valves would again be completely closed.

D. Opening the dump and bypass valves at hot standby due to steam dump controller malfunction: The most severe incident at hot standby would occur in the event the steam dump valve controller yields an incorrect signal and causes the steam dump and bypass valves to open completely. This case is considered to be much less probable than case C above but represents the most limiting event under hot standby conditions.

The possible RPS trips that might be encountered during this event are:

1. Variable high power trip (VHPT).

## 5.0 TRANSIENT AND ACCIDENT ANALYSIS METHODS (Continued)

### 5.6 Excess Load Incident (Continued)

#### 5.6.1 Definition of Event (Continued)

D. (Continued)

2. TM/LP trip.

3. Low steam generator water level trip.

4. Low steam generator pressure trip.

The RPS trip initiated to mitigate the consequences of the event will depend upon the initial conditions and the rate of reactivity insertion due to moderator feedback effects.

#### 5.6.2 Analysis Criteria

The excess load event is classified as a AOO for which the following criteria must be met.

A. The transient minimum DNBR must be greater than or equal to the 95/95 confidence interval limit using the CE-1 correlation.

B. The peak linear heat rate (PLHR) must be less than or equal to 21 kw/ft.

#### 5.6.3 Objectives of the Analysis

The objectives of the analysis are to calculate a [ ] which, when incorporated in the TM/LP equation will ensure that the SAFDL's are exceeded for those excess load events which require a TM/LP trip for protection and to ensure that the DNBR and LHR SAFDL's are not exceeded for excess load events for which the TM/LP does not provide protection.

## 5.6 Excess Load Incident (Continued)

### 5.6.4 Key Parameters and Analysis Assumptions

As discussed in Section 5 of CENPD-199-P (Reference 5-2), sensitivity studies performed by CE have demonstrated that the maximum calculated [ ] for the excess load event occurs for the [ ] at hot full power conditions. District sensitivity studies show similar results. Therefore, only the hot full power case is analyzed. The key parameters used in the analysis of the excess load event are given in Table 5.6.4-1. The remaining assumptions are the same as those discussed in Reference 5-2.

### 5.6.5 Analysis Method

The steps used for determining the [ ] value and calculating the largest [ ] for all excess load events which rely on the TM/LP trip for DNBR protection are given in Section 5 of CENPD-199-P (Reference 5-2). The minimum transient DNBR value for excess load events protected by the Variable High Power Trip is calculated using the procedure discussed in the same Section.

The PLHR is calculated by obtaining the core average linear heat rate at time of peak core power and multiplying it by the appropriate peaking factors and associated uncertainties.

### 5.6.6 Analysis Results and 10 CFR 50.59 Criteria

The results of the excess load analysis are similar to those presented in Section 5 of CENPD-199-P (Reference

Table 5.6.4-1

KEY PARAMETERS ASSUMED IN THE EXCESS LOAD EVENT ANALYSIS

| Parameter | Units | Value |
|---|---|---|
| Initial Core Power | $MW_t$ | 1500† |
| Initial Core Inlet Temperature | $^oF$ At Power | Maximum allowed† by Tech. Specs. |
| Initial Reactor Coolant System Pressure | psia | Minimum allowed† by Tech. Specs. |
| Initial Core Mass Flow Rate | $*10^6$ lbm/hr. | Minimum allowed† by Tech. Specs. |
| CEA Drop Time | sec. | Maximum allowed by Tech. Specs |
| Scram Reactivity Worth | $\%\Delta\rho$ | Minimum predicted during core life. |
| Moderator Temperature Coefficient | $*10^{-4}\Delta\rho/^oF$ | Negative values up to the most negative value allowed by Tech. Specs. |

†For DNBR calculations, effects of uncertainties on these parameters were combined statistically.

#### 5.6.6 Analysis Results and 10 CFR 50.59 Criteria (Continued)

5-2). The criteria of 10 CFR 50.59 are met if the [pressure bias term] is less than or equal to the value used in the current TM/LP trip equation.

#### 5.6.7 Conservatism of Results

The following points demonstrate the conservatism of the overall results for the excess load event:

1. Field measurements demonstrate that the CEA magnetic clutch decay time is less than that assumed in the analysis.

2. The actual scram worths are higher than those in the analysis.

3. Where the most negative MTC is used, the value is more negative than that measured during plant operation.

4. The actual Doppler reactivity is more negative than assumed in the analysis.

5. [ ]

6. Field data demonstrates that the actual CEA drop time is less than that assumed in the analysis.

7. The conservatism of the [ ] is discussed in Section 5 of CENPD-199-P (Reference 5-2).

## 5.7 RCS Depressurization

### 5.7.1 Definition of Event

The RCS depressurization event is characterized by a rapid decrease in the primary system pressure caused by either the inadvertent opening of both power operated relief valves (PORVs) or the inadvertent opening of a single primary safety valve operating at rated thermal power. Following the initiation of the event, steam is discharged from the pressurizer steam space to the quench tank where it is condensed and stored. To compensate for the decreasing pressure the water in the pressurizer flashes to steam and the proportional heaters increase the heat added to the water in the pressurizer in an attempt to maintain pressure. During this time the pressurizer level also begins to decrease causing the letdown control valves to close and additional charging pumps to start so as to maintain level. As pressure continues to drop, the backup heaters energize to further assist in maintaining primary pressure. A reactor trip is initiated by the TM/LP trip to prevent exceeding the DNBR SAFDL.

### 5.7.2 Analysis Criteria

The RCS depressurization event is classified as an AOO for which the transient minimum DNBR must be greater than or equal to the 95/95 percent confidence interval limit using the CE-1 correlation.

### 5.7.3 Objectives of the Analysis

This event is classified as an AOO for which there must be sufficient margin built into the TM/LP trip such that the DNBR SAFDL is not exceeded. The objective of this analysis is to calculate a conservative [

] for incorporation into the TM/LP equation.

5.7 RCS Depressurization (Continued)

#### 5.7.4 Key Parameters and Analysis Assumptions

The key parameters for the RCS depressurization event analysis are given in Table 5.7.4-1. Additional assumptions are discussed in Section 5 of CENPD-199-P (Reference 5-2).

#### 5.7.5 Analysis Method

The methods used by the District to analyze the RCS depressurization event are contained in Section 5 of CENPD-199-P (Reference 5-2).

#### 5.7.6 Analysis Results and 10 CFR 50.59

Results of the RCS depressurization transient are discussed in Reference 5-2 and in the 1983 update of the Fort Calhoun Station Unit No. 1 USAR. The criteria of 10 CFR 50.59 are satisfied if the [ ] is less than or equal to the value used in the current TM/LP trip equation.

#### 5.7.7 Conservatism of Results

The conservatism of the calculated pressure bias term is obtained by using the combination of the following conservative key parameters:

1. Conservative scram reactivity characteristics are used in the analysis.

2. Conservatively slow RPS response times are used.

3. Conservatively high primary relief or safety valve areas are used.

Table 5.7.4-1

KEY PARAMETERS ASSUMED IN THE RCS DEPRESSURIZATION EVENT ANALYSIS

| Parameter | Units | Value |
|---|---|---|
| Initial Core Power Level | $MW_t$ | 1530 |
| Initial Core Inlet Temperature | $^oF$ | Maximum allowed by Tech. Specs. |
| Initial Reactor Coolant System Pressure | psia | Upper limit of normal operating range |
| Moderator Temperature Coefficient | $*10^{-4}\Delta\rho/^oF$ | Most negative allowed by Tech. Specs. |
| Fuel Temperature Coefficient | $*10^{-4}\Delta\rho/^oF$ | Most negative predicted during core life. |
| Core Average $H_{gap}$ | $BTU/hr.-Ft.^2-^oF$ | Minimum predicted during core life. |
| Total Trip Delay Time | sec. | 1.4 |

5.7 RCS Depressurization (Continued)

5.7.7 Conservatism of Results (Continued)

4. The RCS pressure is initially assumed to be in its upper limit as opposed to the normal operating pressure.

## 5.8 Main Steam Line Break Accident

### 5.8.1 Definition of the Event

A large break of a pipe in the main steam system causes a rapid depletion of steam generator inventory and an increased rate of heat extraction from the primary system. The resultant cooldown of the reactor coolant, in the presence of a negative moderator temperature coefficient of reactivity, will cause an increase in nuclear power and trip the reactor. A severe decrease in main steam pressure will also initiate reactor trip and cause the main steam isolation valves to close. If the steam line rupture occurs between the isolation valve and the steam generator outlet nozzle, blowdown of the affected steam generator will continue. (However, closure of the check valve in the ruptured steam line, as well as closure of the isolation valves in both steam lines, will terminate blowdown from the intact steam generator). The fastest blowdown, and therefore, the most rapid reactivity addition, occurs when the break is at a steam generator nozzle. This break location is assumed for the cases analyzed.

Both full power and no-load (hot standby) initial condition cases were considered for two-loop operation (i.e., four reactor coolant pumps).

#### 5.8.1 Definition of the Event (Continued)

Since the steam generators are designed to withstand reactor coolant system operating pressure on the tube side with atmospheric pressure on the shell side, the continued integrity of the reactor coolant system barrier is assured.

The most probable trip signals resulting from an MSLB include low steam generator pressure, high power, low steam generator water level, TM/LP, and high rate-of-change of power (for the no-load case).

#### 5.8.2 Analysis Criteria

The steam line break accident event is classified as a postulated accident for which the site boundary doses must be within the 10 CFR 100 criteria. Acceptable site boundary doses are demonstrated by showing that the critical heat flux is not exceeded.

#### 5.8.3 Objectives of the Analysis

The objectives of the analysis are to demonstrate that the margins to DNB for the reload core no-load two-loop and full-load two-loop main steam line break cases are greater than that for the Cycle 1 cases given in the original FSAR. This is accomplished by demonstrating that the return to power during the event for the reload core is less than the return to power calculated for Cycle 1.

#### 5.8.4 Key Parameter and Analysis Assumptions

The MSLB accident is assumed to start from steady state conditions with the initial power being 1530 MWt (102%)

5.8 Main Steam Line Break Accident

5.8.4 Key Parameter and Analysis Assumptions (Continued)

for the full power case and 1 MWt for the no-load case. The reactor coolant system cooldown causes the greatest positive reactivity insertion into the core when the Moderator Temperature Coefficient (MTC) is the most negative. For this reason the Technical Specification negative MTC limit corresponding to the end-of-cycle is assumed in the analysis. Since the reactivity change associated with moderator feedback varies significantly over the temperature range covered in the analysis, a curve of reactivity insertion versus temperature rather than a single value of MTC is assumed. This curve is derived on the basis that upon reactor trip the most reactive CEA is stuck in the fully withdrawn position thus yielding the most adverse combination of scram worth and reactivity insertion. Although no single value of MTC is assumed in the analysis, the moderator cooldown reactivity function is calculated assuming an initial MTC equal to the most negative Technical Specification limit.

Reactivity feedback effects from the variation of fuel temperture (i.e., Doppler) are included in the analysis. The most negative Doppler defect function, when used in conjunction with the decreasing fuel temperature causes the greatest positive reactivity insertion during the MSLB event. In addition to assuming the most negative Doppler defect function, an additional 15% uncertainty is assumed, i.e., a 1.15 multiplier. This multiplier conservatively increases the subcritical multiplication and results in a larger return-to-power.

The delayed neutron precursor fraction, $\beta$, assumed is the maximum absolute value including uncertainties for

5.8.4 Key Parameter and Analysis Assumptions (Continued)

end of cycle conditions. This is conservative since it also maximizes subcritical multiplication and thus, enhances the potential for a return-to-power.

The steam generator low pressure trip, which occurs at 478 psia (including a 22 psia uncertainty below the nominal trip setting of 500 psia), is the trip assumed in the analysis. No credit is taken for the high power trip which occurs at approximately the same time for the full power case. For the cases analyzed, it is assumed that the most reactive CEA is stuck in the fully withdrawn position. If all CEA's insert (no stuck CEA's), there is no return-to-critical and no power transient following trip.

The cold edge temperatures are used to calculate moderator reactivity insertion during the cooldown, thus maximizing the return-to-critical and return-to-power potentials.

The Emergency Operating Procedures are to be revised prior to the operation of Cycle 9 to eliminate the requirement of manually tripping the reactor coolant pumps (RCP's) as discussed in Reference 5-3. However, if tripping of the RCP's is required, the following discussion is applicable.

The MSLB case with the RCP's tripped is similar to the MSLB case with a loss of offsite power since the RCP's coastdown in both events. As discussed in Reference 5-4, the loss of offsite power delays safety injection due to the time delay for the emergency diesel genera-

5.8.4 Key Parameter and Analysis Assumptions (Continued)

tors to restore power to the safety injection pumps and causes a coastdown of the RCP's. The coastdown affects the degree of overcooling and increases the time for safety injection borated water to reach the core midplane. Because manual tripping of the RCP's results in a later coastdown of the RCP's and because safety injection is not delayed since offsite power is available (i.e., the diesel generator startup and pump loading delays are not present), the injected boron will arrive at the core midplane sooner for a MSLB with the RCP's tripped than for a MSLB with a loss of offsite power. Therefore, the reactivity effects of a MSLB with the RCP's tripped are less severe than for the MSLB with a loss of offsite power.

Reference 5-4 states that the MSLB case with a loss of offsite power results in the injected boron being dominant over the RCS cooldown and concludes that the reactivity effects of a MSLB accident would be reduced in severity with a concurrent loss of offsite power when compared to the same event with offsite power available and the RCP's operating. Because the reactivity effects of a MSLB with the RCP's tripped after SIAS are less severe than a MSLB with a concurrent loss of offsite power, it is concluded that the reactivity effects for the MSLB case with the RCP's tripped after SIAS are less severe than for a MSLB with offsite power available and RCP's operating.

The reactor coolant volumetric flow rate is assumed to be constant during the incident. The LCO flow rate (197,000 gpm) was used in order to obtain the most ad-

5.8.4 Key Parameter and Analysis Assumptions (Continued)

verse results. A lower flow rate increases the initial fuel and average primary coolant temperatures and consequently results in a higher steam generator pressure and a greater steam generator mass inventory. These effects cause a longer blowdown, a greater blowdown rate and a greater decrease in average primary coolant temperature. After MSIV closure the lower flow rate decreases the rate of reverse heat transfer from the intact steam generator, thereby increasing the heat extracted from the primary steam by the ruptured steam generator. The overall effect is that the potential for a return-to-power is maximized.

Maximum values for the heat transfer coefficient across the steam generator are used for the no-load initial condition case, while nominal values are used for the full-load initial condition. These heat transfer coefficients result in the most severe conditions during the incident because of the shape of the reactivity versus moderator temperature function and the difference in average moderator temperature for the maximum and minimum values of the steam generator heat transfer coefficients.

The fast cooldown following a MSLB results in a rapid shrinking of the reactor coolant. After the pressurizer is emptied, the reactor coolant pressure is assumed to be equal to the saturation pressure corresponding to the highest temperature in the system.

Safety injection actuation occurs at 1578 psia (i.e., 1600 psia minus the 22 psia uncertainty) after the

5.8.4 Key Parameter and Analysis Assumptions (Continued)

pressurizer empties. Additional time is required for pump acceleration, valve opening, and flushing of the unborated part of the safety injection piping along with the requirement that the RCS pressure decrease below the shutoff head of the safety injection pumps (1376 psia for high pressure safety injection (HPSI) pumps and 201 psia for low pressure safety injection pumps (LPSI) pumps). The analysis takes credit for one HPSI pump, one LPSI pump, and the safety injection tanks.

The boric acid is assumed to mix homogeneously with the reactor coolant at the points of injection into the cold legs. Slug flow is assumed for movement of the mixture through the piping, plena, and core. After the boron reaches the core midplane, the concentration within the core is assumed to increase as a step function after each loop transit interval.

The boron concentration of the safety injection water is assumed to be at the Technical Specification minimum limit. The values of the inverse boron worth are conservatively chosen to be large to minimize the negative reactivity insertion from safety injection.

Since the rate of temperature reduction in the reactor coolant system increases with rupture size and with steam pressure at the point of rupture, it is assumed that a circumferential rupture of a 26-inch (inside diameter) steam line occurs at the steam generator main steam line nozzle, with unrestricted blowdown. Critical flow is assumed at the point of rupture, and all of

5.8 Main Steam Line Break Accident (Continued)

5.8.4 Key Parameter and Analysis Assumptions (Continued)

the mass leaving the break is assumed to be in the steam phase. This assumption results in the maximum heat removal from the reactor coolant per pound of secondary water, since the latent heat of vaporization is included in the net heat removal. A single failure of the reverse flow check valve in the ruptured steam generator is assumed; so that the intact steam generator will have steam flow through the unaffected steam line and back through and out the ruptured line. Based on sensitivity analyses performed by the District, this is the most severe single failure for the steam line break event. The analysis credits a choke which is installed in each steam line immediately above the steam generator and assumes the steam flow from the intact steam generator is through a 50% area reduction choke installed in a 24 inch steam line. This flow will be terminated upon MSIV closure.

The feedwater flow at the start of the MSLB corresponds to the initial steady state operation. For the full load initial condition, it is automatically reduced in accordance with the program used in the valve controller. For the no load initial condition, feedwater flow is assumed to match energy input by the reactor coolant pumps and the 1 MWt core power. Feedwater isolation upon the receipt of a low steam generator pressure (at 478 psia) is credited for both the full load and no load cases. A valve closure time of 30 seconds was used.

5.8 Main Steam Line Break Accident (Continued)

5.8.5 Analysis Method

The analysis of the main steam line break accident is performed using CESEC which models neutron kinetics with fuel and moderator temperature feedback, the reactor protective system, the reactor coolant system, the steam generators and the main steam and feedwater systems.

5.8.6 Analysis Results and 10 CFR 50.59 Criteria

The results of the analysis for the Fort Calhoun steam line break event are discussed in Section 14.12 of the 1983 update of the Fort Calhoun Station Unit No. 1 USAR. The criteria of 10 CFR 50.59 are met if the calculated return-to-power is less than the return-to-power reported for the Cycle 1 analysis, using the current Technical Specification limit on shutdown margin and moderator temperature coefficient.

5.8.7 Conservatism of Results

Conservatism is added to the analysis by inclusion of uncertainties in moderator and fuel temperature coefficients of reactivity, by taking no credit for void reactivity feedback, by taking credit for only 1 HPSI pump and by taking no credit for the stuck CEA worth.

5.9 Seized Rotor Event

5.9.1 Definition of Event

The seized rotor event is assumed to be caused by a mechanical failure of a single reactor coolant pump. It

is assumed that the rotor shears instantaneously, leaving a low inertia impeller attached to a bent shaft. This latter combination comes to a halt immediately causing a sharp drop in the flow rate. The rapid reduction in core flow will initiate a reactor trip on low flow within the first few seconds of the transient.

#### 5.9.2 Analysis Criteria

A single reactor coolant pump shaft seisure is classified as a postulated accident for which the dose rates must be within 10 CFR 100 guidelines.

#### 5.9.3 Objective of the Analysis

The objective of the analysis is to demonstrate that the radiological releases are within a small fraction of 10 CFR 100 guidelines. This objective is met if it can be shown that less than 1% of the pins fail during the event.

#### 5.9.4 Key Parameters and Analysis Assumptions

The key parameters used in the analysis of the seized rotor event are given in Table 5.9.4-1. The seized rotor is conservatively assumed to result in a 0.1 second rampdown of the core flow from its initial value to the 3 pump value. For CETOP calculations, [

]

Table 5.9.4-1

KEY PARAMETERS ASSUMED IN THE SEIZED ROTOR ANALYSIS

| Parameter | Units | Value |
|---|---|---|
| Initial Core Power Level | $MW_t$ | 1500$^{+}$ |
| Initial Core Inlet Temperature | $^oF$ | Maximum allowed$^{+}$ by Tech. Specs. |
| Initial Reactor Coolant System Pressure | psia | Minimum allowed$^{+}$ by Tech. Specs. |
| Initial Core Mass Flow Rate | $*10^6$ lbm/hr. | Minimum allowed$^{+}$ by Tech. Specs. |
| Moderator Temperature Coefficient | $*10^{-4}\Delta\rho/^oF$ | Most positive allowed by Tech. Specs. |
| Fuel Temperature Coefficient | $*10^{-4}\Delta\rho/^oF$ | Least negative predicted during core life. |
| Core Average $H_{gap}$ | BTU/hr-Ft.$^2$-$^oF$ | Minimum predicted during core life. |
| CEA Drop Time | sec. | Maximum allowed by Tech. Specs. |
| Scram Reactivity Worth | $\%\Delta\rho$ | Minimum predicted during core life. |

$^{+}$Uncertainties on these parameters are combined statistically.

5.9 Seized Rotor Event (Continued)

5.9.5 Analysis Method

Two methods of analyzing the seized rotor event are discussed in this section. Section 5.9.5.1 discusses a method which does not require transient analysis input. Section 5.9.5.2 discusses a method which utilizes transient analysis input.

5.9.5.1 Analysis Method Without Transient Analysis Response Input

This method calculates the number of pin failures assuming that the core flow instantaneously decreases to the 3-pump flow rate. This method utilizes the TORC analysis with a 3-pump inlet flow distribution. The initial RCS pressure and core inlet temperature are used as input to TORC and the core average heat flux is conservatively assumed to remain at its initial value. The maximum value of $F_R T$ is combined with a conservatively flat power distribution. The TORC calculation [

] the number of pins that have failed is calculated.

5.9.5.2 Analysis Methods Using Transient Analysis

This method utilizes the CESEC code to calculate the transient response for the

5.9 Seized Rotor Event (Continued)

5.9.5 Analysis Method (Continued)

5.9.5.2 Analysis Methods Using Transient Analysis (Continued)

seized rotor event. The CETOP code is then used to determine the time of minimum DNBR. The TORC code utilizes the 3-pump inlet flow distribution, 3-pump core flow rate, and the RCS pressure, core inlet temperature and core heat flux calculated at the time of minimum DNBR by CESEC. The steps to determine the number of pin failures is then performed [ ] as discussed in Section 5.9.5.1.

### 5.9.6 Analysis Results and 10 CFR 50.59 Criteria

The results of the seized rotor analysis are contained in the 1983 update of the Fort Calhoun Station Unit No. 1 USAR. The criteria of 10 CFR 50.59 are met, if the number of pin failures is less than one percent.

### 5.9.7 Conservatism of Results

Conservatism in the calculated number of fuel pins predicted to experience DNBR is added through the use of the following assumptions:

1. The most positive MTC is assumed in the analysis. The actual MTC is more negative and would limit core power and heat flux rise.

5.9 Seized Rotor Event (Continued)

5.9.7 Conservatism of Results (Continued)

2. A relatively flat pin census is assumed in the analysis. A more peaked pin census distribution would lower the number of pins predicted to experience DNB.

3. For the case without transient analysis, no credit is taken for the pressure increase during the transient and calculating the minimum transient DNBR.

## 5.10 CEA Ejection Accident

### 5.10.1 Definition of Event

A CEA ejection accident is defined as a mechanical failure of a control rod mechanical pressure housing such that the coolant system pressure would eject the CEA and the drive shaft to a fully withdrawn position. The consequences of this mechanical failure is a rapid reactivity insertion which when combined with an adverse core power distribution potentially leads to localized fuel damage. The CEA ejection accident is the most rapid reactivity insertion that can be reasonably postulated. The resultant core and thermal power excursion is limited primarily by the Doppler reactivity effect of the increased fuel temperatures and is terminated by reactor trip of the remaining CEA's activated by the high power trip or variable high power trip.

5.10 CEA Ejection Accident (Continued)

5.10.2 Analysis Criteria

The CEA ejection event is classified as a postulated accident. The design and limiting criteria are:

1. The average fuel pellet enthalpy at the hot spot will be equal to or less than 280 calories/gram.

2. The peak reactor pressure during a portion of the transient will be less than the value that will cause stress to exceed the emergency conditions stress limits as defined in Section 3 of the ASME Boiler and Pressure Vessel Code.

3. Fuel melting will be limited to keep the offsite dose consequences well within the guidelines of 10 CFR 100.

These limiting criteria are taken from the NRC Regulatory Guide 1.77 "Assumptions Used for Evaluating a Control Rod Injection Accident for Pressurized Water Reactors".

5.10.3 Objectives of the Analysis

The objective of the analysis is to demonstrate that the total average enthalpy of the hottest fuel pellet for the hot full power and hot zero power cases is less than that reported for the reference cycle.

5.10.4 Analysis Method

The District utilizes the CEA Ejection Accident Analysis Methodology of our current fuel vendor, Exxon

5.10 CEA Ejection Accident (Continued)

5.10.4 Analysis Method (Continued)

Nuclear Corporation. This analysis methodology is documented in Reference 5-5. This methodology utilizes physics parameters, calculated by the District in accordance with the methods outlined in Reference 5-6. The power peaking factor, $F_q^T$, is defined as the post ejected 3-D fuel rod power peak.

5.10.5 Analysis Results and 10 CFR 50.59 Criteria

The results of the CEA Ejection Analysis are reported in Section 14.13 of the Fort Calhoun Unit 1 USAR. Criteria of 10 CFR 50.59 are satisfied if the total average enthalpy of the hottest fuel pellet is less than or equal to the values reported in the reference cycle.

5.10.6 Conservatism of Results

The major area of conservatism is the calculation method used to obtain the ejected CEA worth and the ejected radial peak. The ejected worth and the ejected radial peak are calculated without any credit for Doppler or Xenon feedback. In addition, the hot full power ejected worth and ejected peak are calculated assuming the no-load temperature of 532°F. The lower temperature is more adverse since this causes a power role to the core periphery which also happens to be the location of the ejected CEA. Also, the ejected worth is calculated assuming the CEA'S are fully inserted for hot full power case regardless of PDIL. Thus, the ejected worth is conservative.

### 5.11 Loss of Coolant Accident

The District does not perform the Loss of Coolant Accident Analysis. The large break loss of coolant analysis was performed by Exxon Nuclear Corporation (ENC) and the small break analysis was performed by Combustion Engineering. The large break analysis shows the closest approach to the Appendix K criteria for ECCS analysis. The District verifies that the physics input assumptions and the maximum rod burnup are within the bounds assumed in the ENC large break analysis.

## 6.0 TRANSIENT ANALYSIS CODE VERIFICATION

### 6.1 Introduction

The District utilizes the CESEC-III computer code to calculate the transient response of the NSSS during events discussed in this document. Combustion Engineering has provided overall verification of the CESEC-III code in Reference 6-1. The purpose of the work reported here is to demonstrate the District's ability to correctly utilize the CESEC-III code.

In order to demonstrate Omaha Public Power District's ability to correctly use the CESEC-III computer code, verification work has been performed by benchmarking both actual plant transient data and independent safety analyses previously accepted by the NRC. The plant transients which were benchmarked were the Turbine-Reactor trip and Four-Pump Loss of Coolant Flow events. The independent safety analyses which were benchmarked were the Dropped CEA, Main Steamline Break, and RCS Depressurization events. Each of the comparisons will be addressed below.

### 6.2 Comparison to Plant Data

A prerequisite for beginning performance of transient analyses is verification that the code will stabilize with the correct system parameters when simulating steady state operation. This step was performed following setup of the CESEC-III code and correct results were obtained.

For plant transient benchmarking, the type of transients that have occurred and both the quality and quantity of data existing for each is very limited. In nearly all cases, operators take actions which reduce the consequences of the event, introducing complicated perturbations in system response which cannot be easily modeled, because the actions taken and the time at which they are performed are not recorded. Strip chart recordings on an extremely compressed time scale are generally

6.2 Comparison to Plant Data (Continued)

the only form of data available. This compressed time scale (with graduations typically of 10 minutes) do not permit adequate comparisons to CESEC-III modeling in which seconds are of major concern. The only source of plant transient data in which system parameters were measured with high speed strip chart recorders and no operator action taken, was during the Cycle 1 startup testing. Good data existed for a nominal full power turbine-reactor trip and a 35% power total loss of RCS flow event. The CESEC-III computer code was set up to model Cycle 1 in a best estimate mode to permit accurate comparisons to the actual measured plant responses for both of the above cases. A summary of each of these comparisons follows.

6.2.1 Turbine-Reactor Trip

For the turbine-reactor trip case, the plant comparison data were obtained from the Cycle 1 startup testing performed May 10, 1974. The event was initiated from 97% of full power, all-rods-out, and equilibrium xenon. The plant response data used in the CESEC-III comparisons were obtained from vendor test recorders. No operator action was taken following the manual generator-turbine trip (which provided the RPS "loss of load" trip). Prior to the trip the main feedwater, the pressurizer pressure, and pressurizer level control systems were all in the automatic mode, and the letdown backpressure control valve was in the manual mode. With the exception of adjusting the letdown backpressure control valve at 20 seconds, no operator action was taken for 60 seconds following the trip.

Figures 1-1 through 1-7 show plots of the comparisons between the measured plant responses and the CESEC-III predicted responses. It should be noted that

6.2 Comparison to Plant Data (Continued)

6.2.1 Turbine-Reactor Trip (Continued)

this test was performed based on a rated power level of 1420 MWt rather than the current limit of 1500 MWt (the design power for which licensing was obtained in Cycle 6).

Figure 1-1 shows the nuclear power response following the turbine-reactor trip. The CESEC-III prediction follows the same power decay rate, however, the end-point residual power is slightly higher, i.e., conservative. It should be noted that trip delays included in the CESEC-III modeling prevent the immediate power drop observed in the plant data; again this is conservative. The pressurizer pressure response predicted by CESEC-III and shown in Figure 1-2 shows very good agreement with the plant response. The CESEC-III case was initiated 10 psia above the plant data and remained slightly above the plant response for the duration of the transient. The difference between the predicted and measured pressurizer pressures increased slightly due to the higher residual power after trip as shown in Figure 1-1. This difference between the predicted and measured pressurizer pressures at 60 seconds is only 19 psia, a value which is less than the pressure measurement uncertainty. Figure 1-3 shows the pressurizer level response. The comparison between the measured and predicted values shows excellent agreement. Figure 1-4 shows the RCS cold-leg and hot-leg temperature responses for each steam generator loop for the plant data and the CESEC-III predicted average cold-leg and hot-leg temperatures. The differences in the transient response of the two steam generator loops for the plant data

is attributable to the differences in the main feedwater flow rate rampdown after trip (see Figure 1-6). The CESEC-III responses lead the loop measurements because of the measurement delays associated with the response time of the RTDs (resistance temperature devices) providing the temperature signals. Figure 1-5 shows the measured and predicted steam generator pressure responses. These two plots show very good agreement with each other with only minor differences. The predicted pressure is slightly higher early in the event due to a combination of the greater heat residual as shown in Figure 1-1, a quicker turbine stop valve closure, and quicker steam dump-bypass operation assumed in the CESEC-III analysis. The latter two effects, which are shown in the steam flow of Figure 1-7, would show better agreement if the CESEC-III input were modified, however, the overall differences are small enough not to warrant the reanalysis.

In conclusion, the CESEC-III predicted parameters for the turbine-reactor trip show very good agreement with those measured in the Cycle 1 startup testing performed at nominal full power conditions.

### 6.2.2 Four-Pump Loss of Coolant Flow

For the four-pump loss of coolant flow case, the plant comparison data were obtained from the Cycle 1 startup test performed March 6, 1974. This event was initiated from 35% power by manually and simulataneously tripping all four reactor coolant pumps. At

the time of trip the pressurizer pressure, pressurizer level, main feedwater, and steam dump and bypass controllers were in the automatic mode. At approximately 20 seconds after the trip, the operators took manual control of feedwater in order to preclude overfeeding of the steam generators and too rapid of a cooldown for the following natural circulation test.

The behavior of the various RCS and secondary parameters that were measured and the CESEC-III predictions for the first 30 seconds following the RCP trips are shown in Figures 2-1 through 2-8. These comparisons show excellent agreement. The minor differences that exist are discussed below.

Figure 2-1 shows a plot of the measured total RCS flow versus time and that predicted by the CESEC-III code which incorporates explicit modeling of the reactor coolant pumps. These data show excellent agreement with the predicted flow being slightly conservative. Figures 2-2 and 2-3 show the pressurizer pressure and level response comparisons which also show excellent agreement. Figure 2-4 shows plots of core nuclear power versus time. As in the turbine-reactor trip case, CESEC-III shows a slightly higher residual power after trip. The predicted and measured steam generator pressure responses as plotted in Figure 2-5, also show very good agreement. The response of the hot-leg and cold-leg temperatures, as shown in Figure 2-6, is consistent with the data obtained from the turbine-reactor trip case. Again the delay associated with the RTD response causes the predicted tem-

6.2 Comparison to Plant Data (Continued)

6.2.2 Four-Pump Loss of Coolant Flow (Continued)

peratures to lead those that were measured. Figure 2-7 shows that the main feedwater input function used in CESEC-III was acceptable in terms of the actual feedwater system response. It should be noted that the operator action of assuming manual control of the main feedwater system at approximately 20 seconds had little effect on any of the other system parameters examined, and that following a several second reduction in flow the previous flow rate was reestablished. Figure 2-8 shows that turbine stop valve closure rate assumed in the CESEC-III analysis was quicker than the actual valve response. The figure also shows a steam flow rate mismatch between the two steam generators for the plant data. This is something one would not expect and raises the question of the validity of the measurement or its uncertainty for this steam generator steam rate flow, because the two corresponding feedwater flow rates (in Figure 2-7) are consistent.

In conclusion, the CESEC-III predicted parameters for the 35% power total loss of coolant flow show very good agreement with those measured during Cycle 1 startup testing.

## 6.3 Comparisons Between OPPD Analyses and Independent Analyses Previously Performed by the Fuel Vendors

Of the transients analyzed by OPPD for reload core licensing (using CE methodology) no plant data existed, so comparison of the limiting events to previous independent analyses performed by either Exxon Nuclear Company (ENC) or Combustion Engineer-

6.3 Comparisons Between OPPD Analyses and Independent Analyses Previously Performed by the Fuel Vendors (Continued)

ing (CE) was done. For the comparison cases, the assumptions used in the analyses were similar to those used by the District, i.e., the core physics parameters did not vary significantly between fuel cycles. The events chosen for comparison were:

(1) The Dropped CEA event is dependent upon the initial available overpower margin to prevent exceeding the SAFDL's. The goal of the analysis is to determine the DNBR required overpower margin (ROPM).

(2) The Hot Zero Power (HZP) Main Steamline Break which determines the minimum required shutdown margin.

(3) The Hot Full Power (HFP) Main Steamline Break which determines the most negative moderator temperature coefficient of reactivity allowed.

(4) The RCS Depressurization event which is used in the determination of the [ ]. The [ ] accounts for DNBR margin degradation in the thermal margin/low pressure (TM/LP) trip [ ]

6.3.1 Dropped CEA

The Cycle 8 Dropped CEA analysis performed by OPPD was compared to the previous analysis, contained in the Updated Safety Analysis Report (USAR). The USAR analysis was performed by ENC for Cycle 6. Table 1 summarizes the parameters and their values for Cycles

6.3 Comparisons Between OPPD Analyses and Independent Analyses Previously Performed by the Fuel Vendors (Continued)

6.3.1 Dropped CEA (Continued)

6 and 8. Plots of core power versus time for the OPPD (Cycle 8) and ENC (Cycle 6) analyses are found in Figure 3-1. The curves show a very similar prompt drop, to 69% versus 70%, respectively, and both cases show a return to a nominal 100% power. Both cases assumed that the turbine admission valves opened to their full open position in an attempt to maintain full load during the event (i.e., the turbine control system was placed in the load set mode which is not used at Fort Calhoun Station). The core heat flux plots are contained in Figure 3-2. Both are very similar, as was the case in the core power cases. Figure 3-3 contains plots of the coolant average temperature versus time. Both figures are in good agreement showing a drop in average coolant temperature to 567°F. Plots of the inlet and outlet temperatures for Cycle 8 are also included. Figure 3-4 shows plots of the pressurizer pressure versus time. The minimum pressures predicted at 160 seconds are 1957 psia and 1945 psia for Cycle 8 and Cycle 6, respectively. This difference is small enough to be less than the pressure measurement uncertainty.

In summary, the primary system responses between the ENC and OPPD analyses show excellent agreement with each other which is consistent with reload cores having similar core physics parameters.

6.3.2 Hot Zero Power Main Steamline Break

The hot zero power (HZP) Main Steamline Break, which is the basis for determination of the required shut-

down margin, was analyzed by OPPD for Cycle 8. The results of this analysis have been compared to those of ENC in their Cycle 6 analysis and to those obtained by CE in their Cycle 6 control grade auxiliary feedwater (AFW) system analysis. Table 2 shows comparisons of the pertinent input values for each of the analyses.

Figure 4-1 shows plots of core power for the Cycle 8 OPPD analysis and Cycle 6 ENC analysis, respectively. The maximum return-to-power is less for Cycle 8 than for Cycle 6 and occurs later due to the use of a higher shutdown margin. The Cycle 6 CE AFW analysis power is not included because there was no return-to-critical and no return-to-power. Figure 4-2 shows plots of the core average heat flux for OPPD, ENC and CE, respectively. Both the OPPD and CE analyses, which were performed using CESEC-III and CESEC-I, respectively, show a slight heat flux increase at approximately 12 seconds. This is due to subcritical multiplication. Otherwise, the heat flux curves within the specific analyses are essentially the same as the core power curves with a slight decay. Figure 4-3 shows the total reactivity versus time for each of the analyses. With very similar moderator cooldown curves, the peak reactivities occur chronologically with increasing shutdown margin as expected; i.e., for increased shutdown margin (CEAs) it takes longer to be offset by the positive moderator cooldown reactivity insertion.

6.3 Comparisons Between OPPD Analyses and Independent Analyses Previously Performed by the Fuel Vendors (Continued)

6.3.2 Hot Zero Power Main Steamline Break (Continued)

Figure 4-4 shows plots of RCS pressure versus time for Cycle 8 (OPPD) and Cycle 6 AFW (CE). Also included in Figure 4-4 is the Cycle 1 (CE) results. All three of these curves show excellent agreement. The Cycle 6 AFW (CE) analysis shows a lower endpoint pressure than the Cycle 1 (CE) and Cycle 8 (OPPD) analyses due to the assumption of auxiliary feedwater addition. The ENC data available did not include the RCS pressure response.

Figure 4-5 shows plots of the steam generator pressures for Cycle 8 (OPPD) and Cycle 6 AFW (CE), respectively. These plots show reasonable agreement between pressures and times. The increase in the intact steam generator's pressure is due to MSIV closure; i.e., failure of the reverse flow check valve on the intact steam generator was chosen as the most adverse single failure. Following dryout of the ruptured steam generator, the pressure drops to atmospheric. The times of dryout are slightly different due to the increased normal water level value used in the Cycle 8 analysis.

In summary, the HZP Main Steamline Break analysis for Cycle 8 shows trends similar to those in Cycle 6 as analyzed by both CE and ENC.

6.3.3 Hot Full Power Main Steamline Break

The hot full power (HFP) Main Steamline Break provides an acceptance criteria for the most negative

## 6.3 Comparisons Between OPPD Analyses and Independent Analyses Previously Performed by the Fuel Vendors (Continued)

### 6.3.3 Hot Full Power Main Steamline Break (Continued)

moderator temperature coefficient (MTC) of reactivity. If a return-to-critical occurs, the goal of the reload analysis is to show that the return-to-power is bounded by the most limiting case which, for the Fort Calhoun Station, is the Cycle 1 analysis. The Cycle 8 HFP analysis of this event was compared to the previous analyses performed by ENC in Cycle 6 and by CE in their Cycle 6 control grade AFW system analysis. Table 3 shows a comparison of the important input parameters for each of the analyses.

Figures 5-1, 5-2, and 5-3 show plots of core power, core average heat flux, and total reactivity for Cycle 8 (OPPD), Cycle 6 (ENC), and Cycle 6 AFW (CE). Within each cycle's analysis, the core average heat flux slightly lags the core power which peaks at a time several seconds after the peak reactivity is reached (for the return-to-critical cases). The return-to-power peaks occur at different times due to the different scram worths used, as explained for the shutdown margin in the HZP Steamline Break analysis section.

Figure 5-4 shows plots of the RCS pressure versus time for the Cycle 8, Cycle 6 AFW, and Cycle 1 analyses. These plots are very similar and show excellent agreement. Figures 5-A and 5-B show plots of the RCS temperatures for Cycle 8 and Cycle 6 AFW. Again good agreement exists to approximately 180 seconds. At this time, the Cycle 6 AFW analysis assumed runout flow from both AFW pumps to the rup-

6.3 Comparisons Between OPPD Analyses and Independent Analyses Previously Performed by the Fuel Vendors (Continued)

6.3.3 Hot Full Power Main Steamline Break (Continued)

tured steam generator which resumed the RCS cooldown. This additional cooldown caused by the AFW system is prevented from occurring in Cycle 8 by the logic of the newer safety grade AFW system.

Figure 5-6 shows plots of steam generator pressures versus time for Cycle 8 and Cycle 6 AFW (CE). These results are very similar except that the intact steam generator pressure, in the CE analysis, begins to drop after 180 seconds due to the AFW induced RCS cooldown.

### 6.3.4 RCS Depressurization

The RCS Depressurization analysis is performed to calculate a [ ] for the TM/LP trip which accounts for the DNBR margin degradation [

]

Because no figures from previous cycle analyses exist, comparison was made between the transient analysis training manual sample analysis and the figures generated by OPPD for Cycle 8. Pertinent input parameters are summarized in Table 4.

Figure 6-1 shows the plots of RCS pressure versus time for the initial case run without a trip which is used to determine the time manual trip is to be used.

## 6.3 Comparisons Between OPPD Analyses and Independent Analyses Previously Performed by the Fuel Vendors (Continued)

### 6.3.4 RCS Depressurization (Continued)

A manual trip is next simulated at the time of maximum margin degradation; i.e., at the time the maximum RCS Depressurization rate occurs. The maximum RCS Depressurization rate occurs in approximately the first 20 seconds and is constant. Therefore, the time at which a manual trip should occur is arbitrary but must be in the first 20 seconds. A trip time corresponding to a 100 psia drop is adequate to perform the analysis.

Figure 6-2 shows plots of core power versus time for the Cycle 8 analysis and CE's example. The core average heat flux curves are found in Figure 6-3. The RCS pressure versus time plots are shown in Figure 6-4. In the CE example, the initial pressure was 2300 psia, a value which corresponds to the maximum pressure before which the pressurizer sprays will be activated in a 2700 MW(th) class plant (whose normal RCS pressure is 2250 psia). In the Cycle 8 analysis, a value of 2172 psia was used for the initial RCS pressure, since the normal operating RCS pressure at Fort Calhoun is 2100 psia. The Fort Calhoun pressurizer sprays are fully closed at 2175 psia and fully open at 2225 psia.

The comparison of the figures show good agreement in the trends for the core power, core average heat flux, and RCS pressure. The [

]

### 6.4 Summary

Initial setup and operation of the CESEC-III code was performed by showing that the code stabilized for steady state plant operation. Benchmarking against Cycle 1 plant data for the Turbine-Reactor Trip and the Four-Pump Loss of Coolant Flow was performed and excellent agreement between the predicted and observed responses was obtained.

For transients in which plant data were not available, comparisons were performed between the OPPD Cycle 8 analyses of the limiting transients and the Cycle 6 anlayses of the fuel vendors (CE and ENC) and, in one case, the transient analysis training manual example. In all cases, these benchmarking comparisons showed very good agreement.

TABLE 1

COMPARISON OF PARAMETERS INCLUDING UNCERTAINTIES
USED IN THE CEA DROP ANALYSES FOR CYCLES 6 AND 8

| Parameter | Units | Cycle 6 | Cycle 8 |
|---|---|---|---|
| Initial Core Power Level | MWt | 102% of 1500 | 102% of 1500 |
| Core Inlet Temperature | °F | 547 | 547 |
| Pressurizer Pressure | psia | 2053 | 2053 |
| RCS Flow Rate | gpm | 190,000 | 197,000 |
| Moderator Temperature Coeff. | $10^{-4}$ $\Delta\rho$/°F | -2.3 | -2.7 |
| Doppler Coeff. Multiplier | | 1.20 | 1.15 |
| CEA Insertion at Full Power | % Insertion | 0.0 | 25.0 |
| Dropped CEA Worth | %$\Delta\rho$ | -0.34 | -0.28 |

TABLE 2

COMPARISON OF PARAMETERS INCLUDING UNCERTAINTIES
USED IN THE HZP MAIN STEAMLINE BREAK ANALYSIS FOR CYCLES 6 AND 8

| Parameter | Units | Cycle 6 | Cycle 6 AFW | Cycle 8 |
|---|---|---|---|---|
| Initial Core Power Level | MWt | 0.0 | 1.0 | 1.0 |
| Core Inlet Temperature | °F | 532 | 532 | 532 |
| Pressurizer Pressure | psia | 2053 | 2175 | 2172 |
| RCS Flow Rate | gpm | 190,000 | 190,000 | 197,000 |
| Effective Moderator Temperature Coefficient | $10^{-4}$ $\Delta\rho$/°F | -2.3 | -2.3 | -2.5 |
| Doppler Coeff. Multiplier | | 0.8 | 1.15 | 1.15 |
| Minimum CEA Scram Worth (Shutdown Margin) | % $\Delta\rho$ | -3.0 | -4.2 | -4.0 |
| Initial Steam Generator Pressure | psia | N/A | 900 | 895 |
| Initial Steam Generator Mass Inventory (Level) | % Narrow Range Scale | 63 | 63 | 70 |

TABLE 3

COMPARISON OF PARAMETERS INCLUDING UNCERTAINTIES
USED IN THE HFP MAIN STEAMLINE BREAK ANALYSES FOR CYCLES 6 AND 8

| Parameter | Units | Cycle 6 | Cycle 6 AFW | Cycle 8 |
|---|---|---|---|---|
| Initial Core Power Level | MWt | 102% of 1500 | 102% of 1500 | 102% of 1500 |
| Core Inlet Temperature | °F | 547 | 547 | 547 |
| Pressurizer Pressure | psia | 2078 | 2175 | 2172 |
| RCS Flow Rate | gpm | 190,000 | 190,000 | 197,000 |
| Moderator Temperature Coefficient | $10^{-4}$ $\Delta\rho$/°F | -2.3 | -2.3 | -2.5 |
| Doppler Coeff. Multiplier | | 0.8 | 1.15 | 1.15 |
| Minimum CEA Scram Worth | % $\Delta\rho$ | -5.81 | -5.81 | -6.68* |
| Initial Steam Generator Pressure | psia | N/A | 880.5 | 890 |
| Initial Steam Generator Mass Inventory (Level) | % Narrow Range Scale | 63 | 63 | 70 |

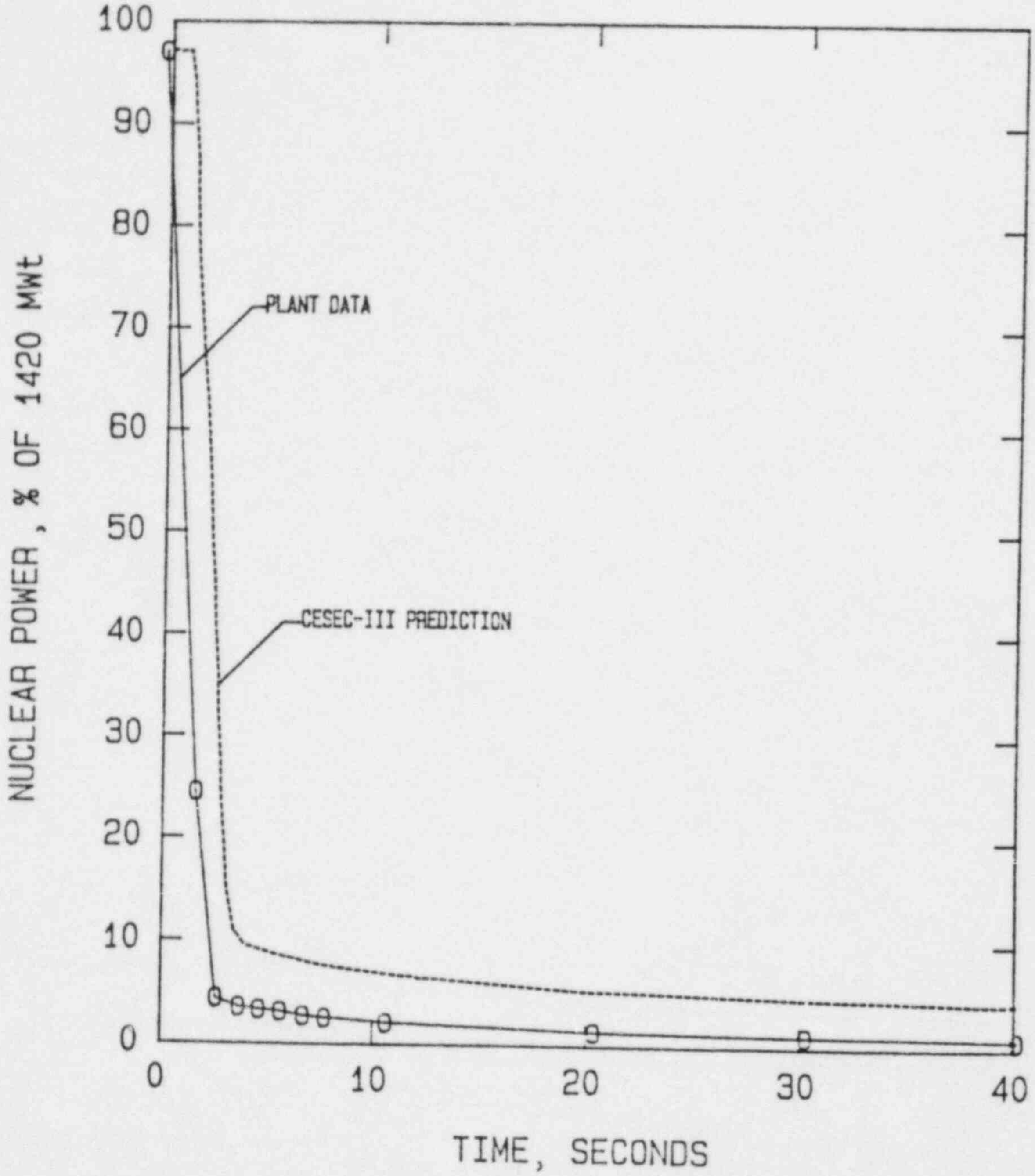
*Reduced to -6.57 to account for axial shape.

TABLE 4

COMPARISON OF PARAMETERS INCLUDING UNCERTAINTIES
USED IN THE RCS DEPRESSURIZATION ANALYSES FOR CYCLES 6 AND 8

| Parameter | Units | Example Case* | Cycle 8 |
|---|---|---|---|
| Initial Core Power Level | MWt | 102% of 1500 | 102% of 1500 |
| Core Inlet Temperature | °F | 547 | 547 |
| Pressurizer Pressure | psia | 2300 | 2172 |
| RCS Flow Rate | gpm | N/A | 209,796 |
| Moderator Temperature Coefficient | $10^{-4}$ $\Delta\rho$/°F | -2.5 | -2.7 |
| Doppler Coeff. Multiplier | | 1.15 | 1.15 |

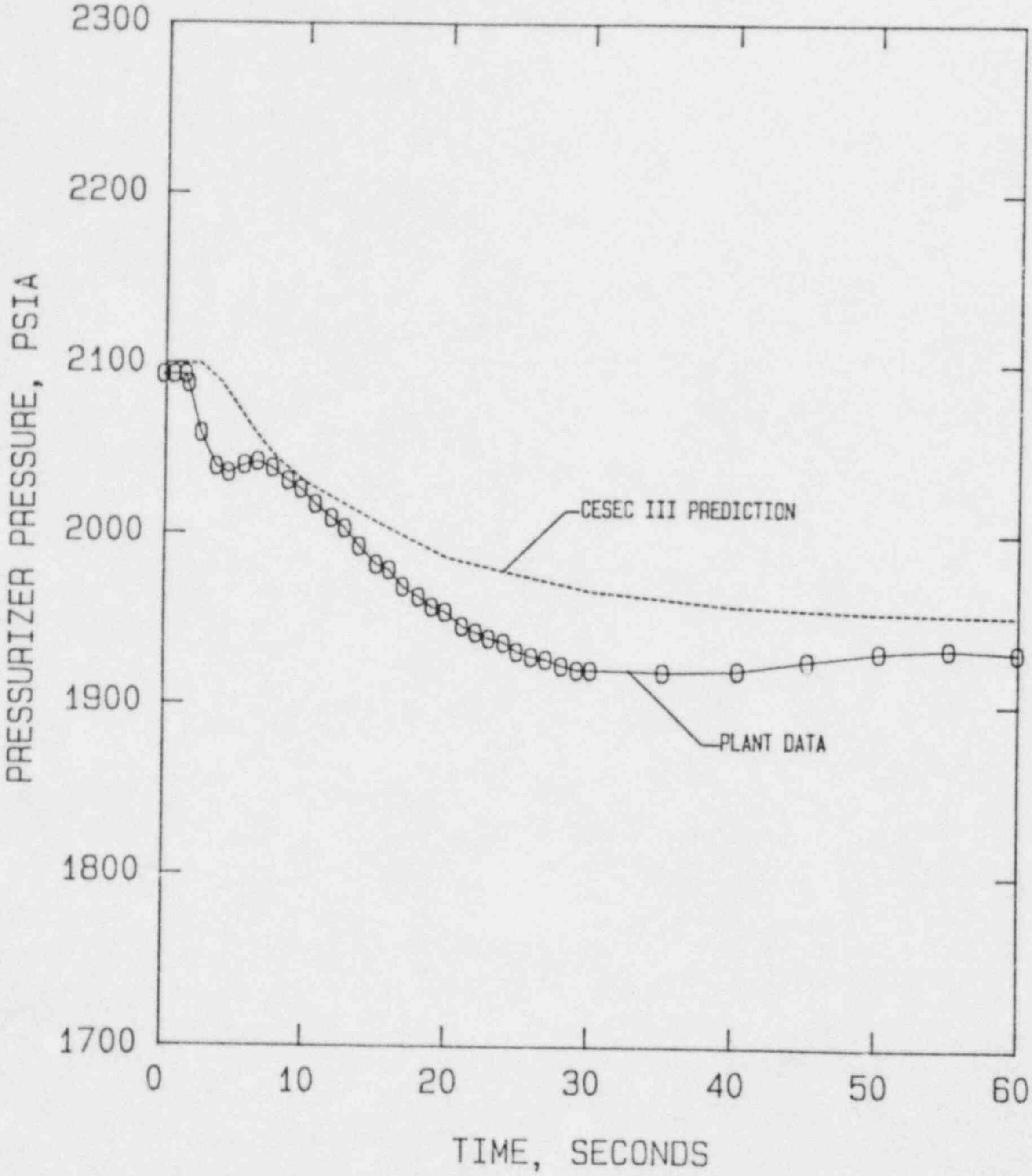
*Example case input data consistent with 2700 MWt plant operating characteristics.




NOTE :

CYCLE 1: (FULL POWER = 1420 MWt )
PLANT DATA: TEST PERFORMED MAY 10, 1974

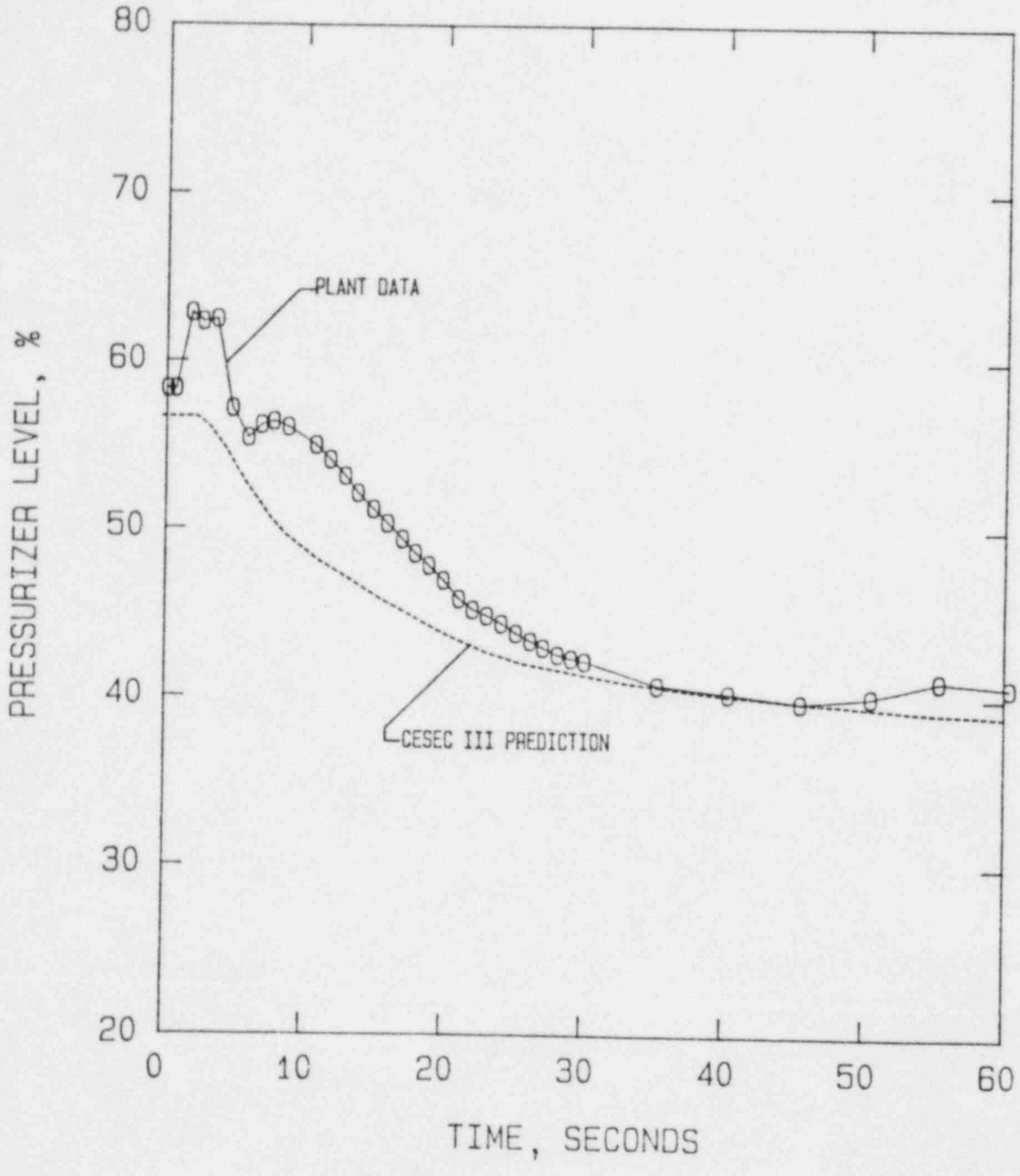
| | | |
|---|---|---|
| Full Power Turbine Trip<br>Nuclear Power vs Time | Omaha Public Power District<br>Fort Calhoun Station-Unit No. 1 | Figure<br>1-1 |




NOTE :

CYCLE 1 (FULL POWER = 1420 MWt)
PLANT DATA: TEST PERFORMED MAY 10, 1974

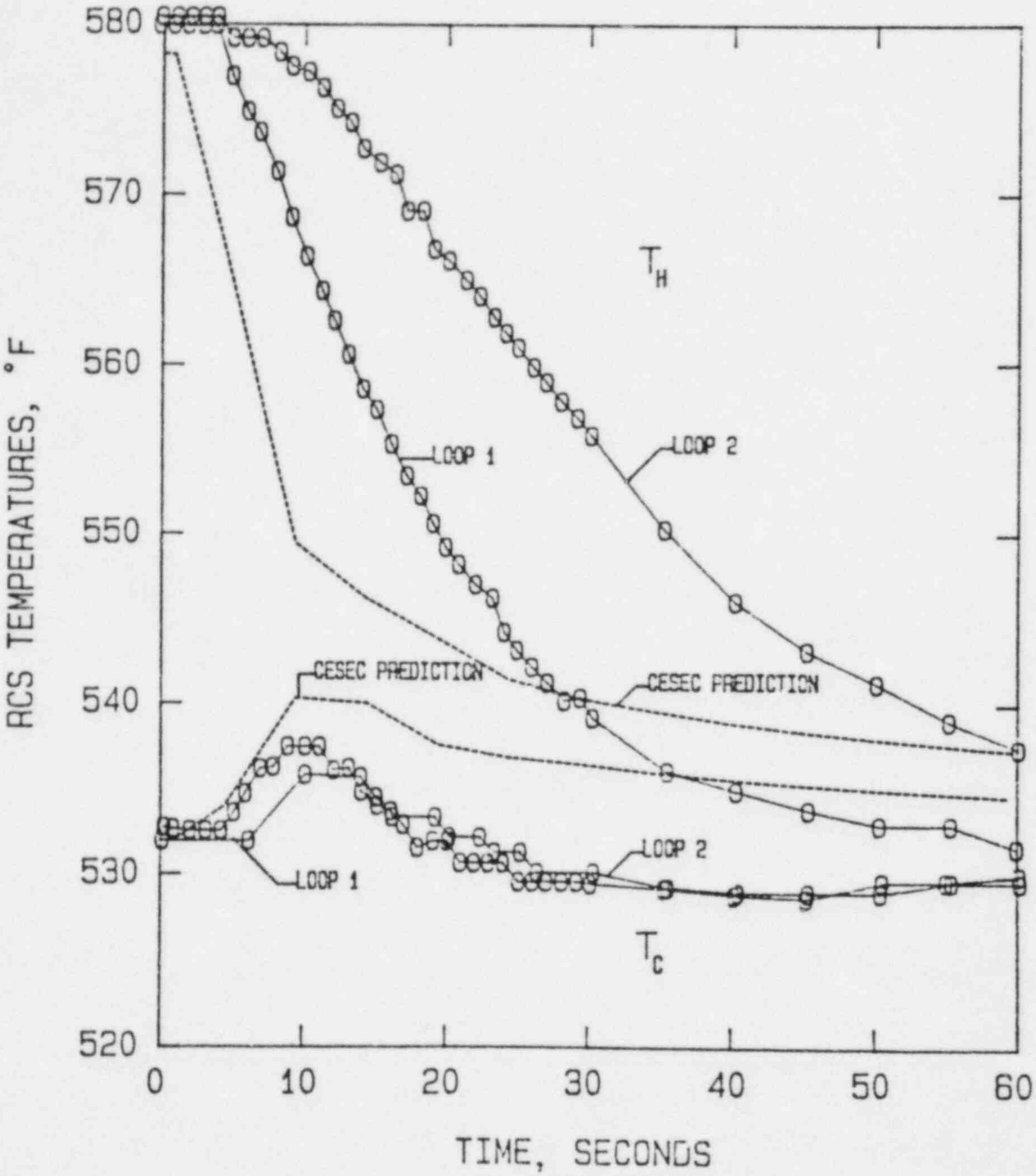
| Full Power Turbine Trip<br>Pressurizer Pressure vs Time | Omaha Public Power District<br>Fort Calhoun Station-Unit No. 1 | Figure<br>1-2 |
|---|---|---|




NOTE :

CYCLE 1 (FULL POWER = 1420 MWt)
PLANT DATA: TEST PERFORMED MAY 10, 1974

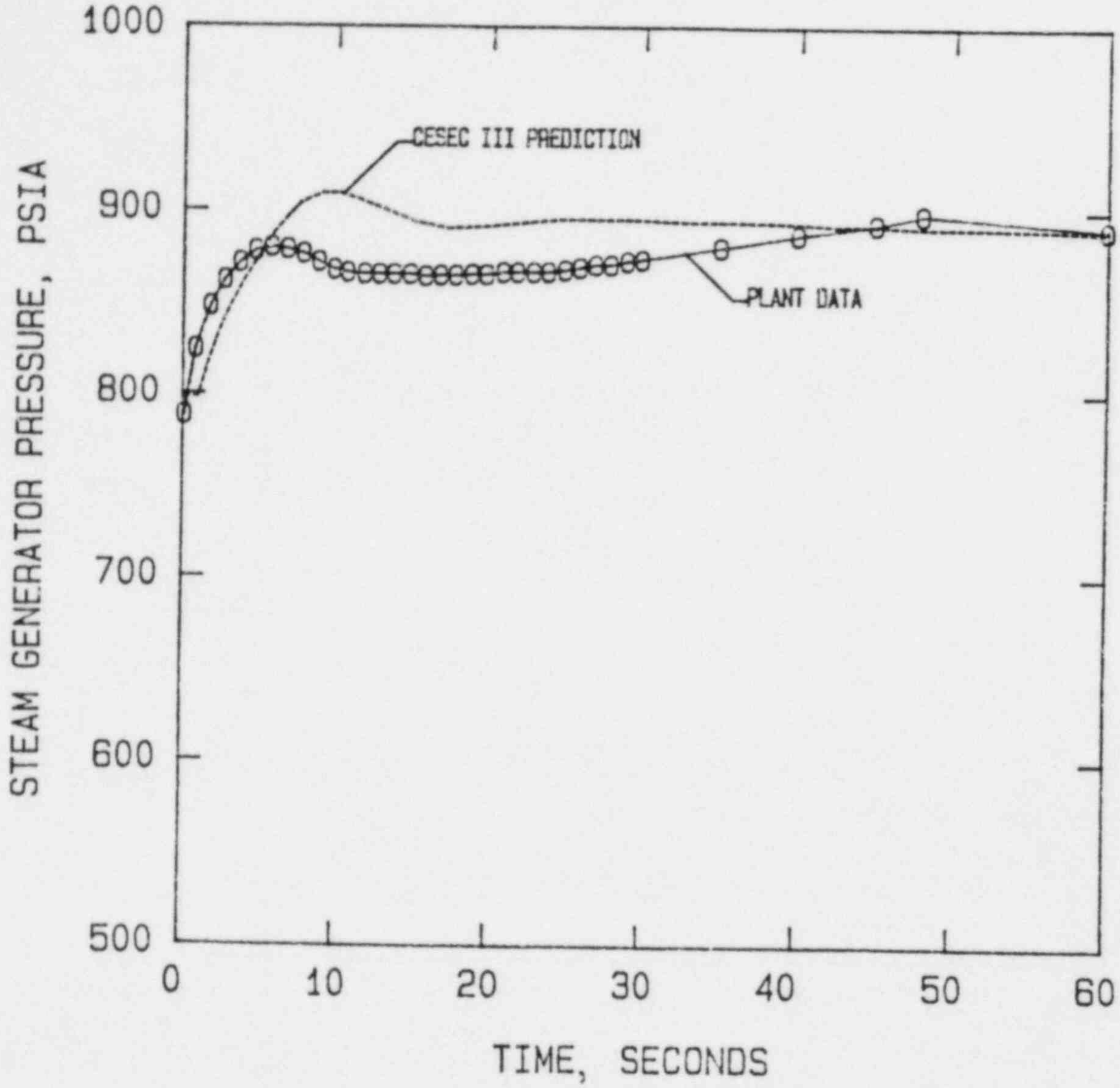
| Full Power Turbine Trip<br>Pressurizer Level vs Time | Omaha Public Power District<br>Fort Calhoun Station-Unit No. 1 | Figure<br>1-3 |
|---|---|---|




NOTE :

CYCLE 1 (FULL POWER = 1420 MWt)
PLANT DATA: TEST PERFORMED MAY 10, 1974

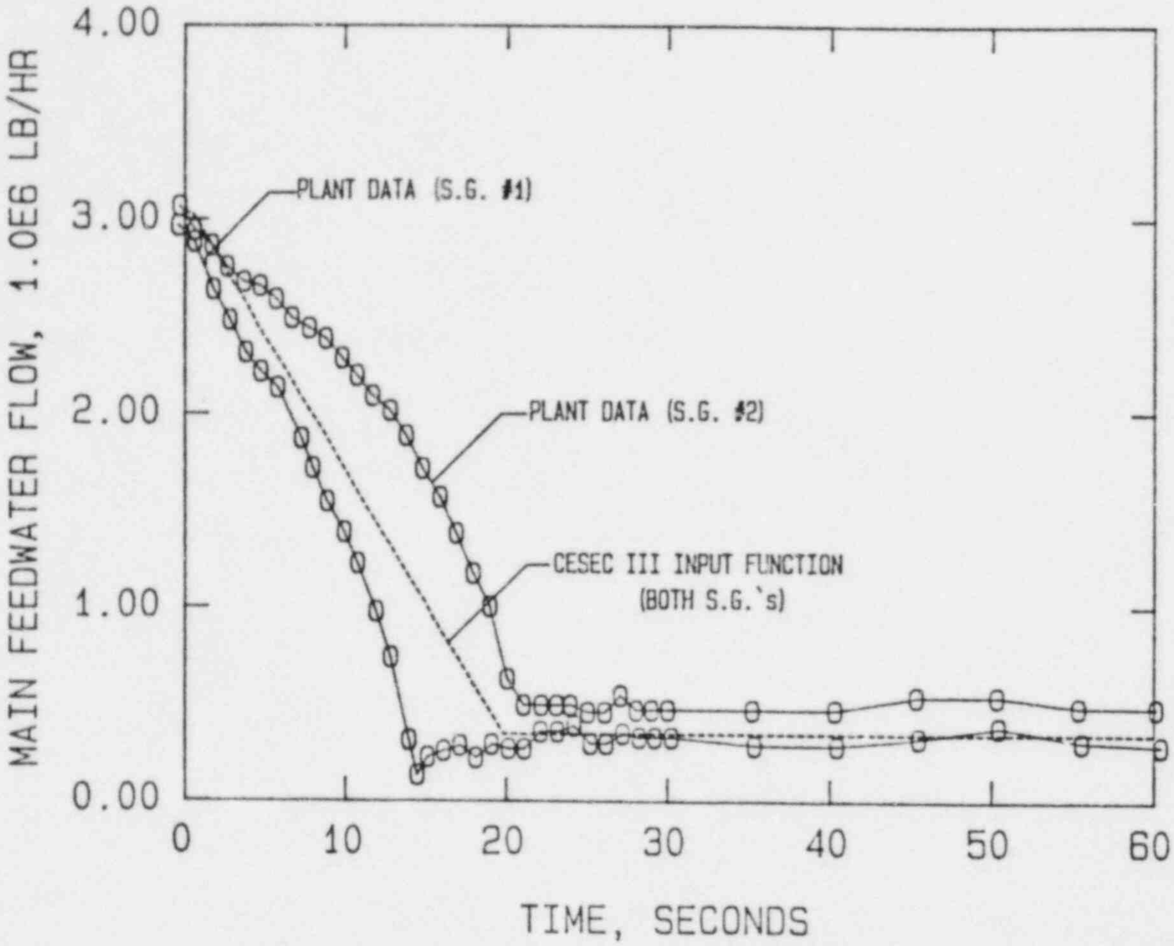
| Full Power Turbine Trip<br>RCS Temperatures vs Time | Omaha Public Power District<br>Fort Calhoun Station-Unit No. 1 | Figure<br>1-4 |
|---|---|---|




NOTE :

CYCLE 1 (FULL POWER = 1420 MWt)
PLANT DATA: TEST PERFORMED MAY 10, 1974

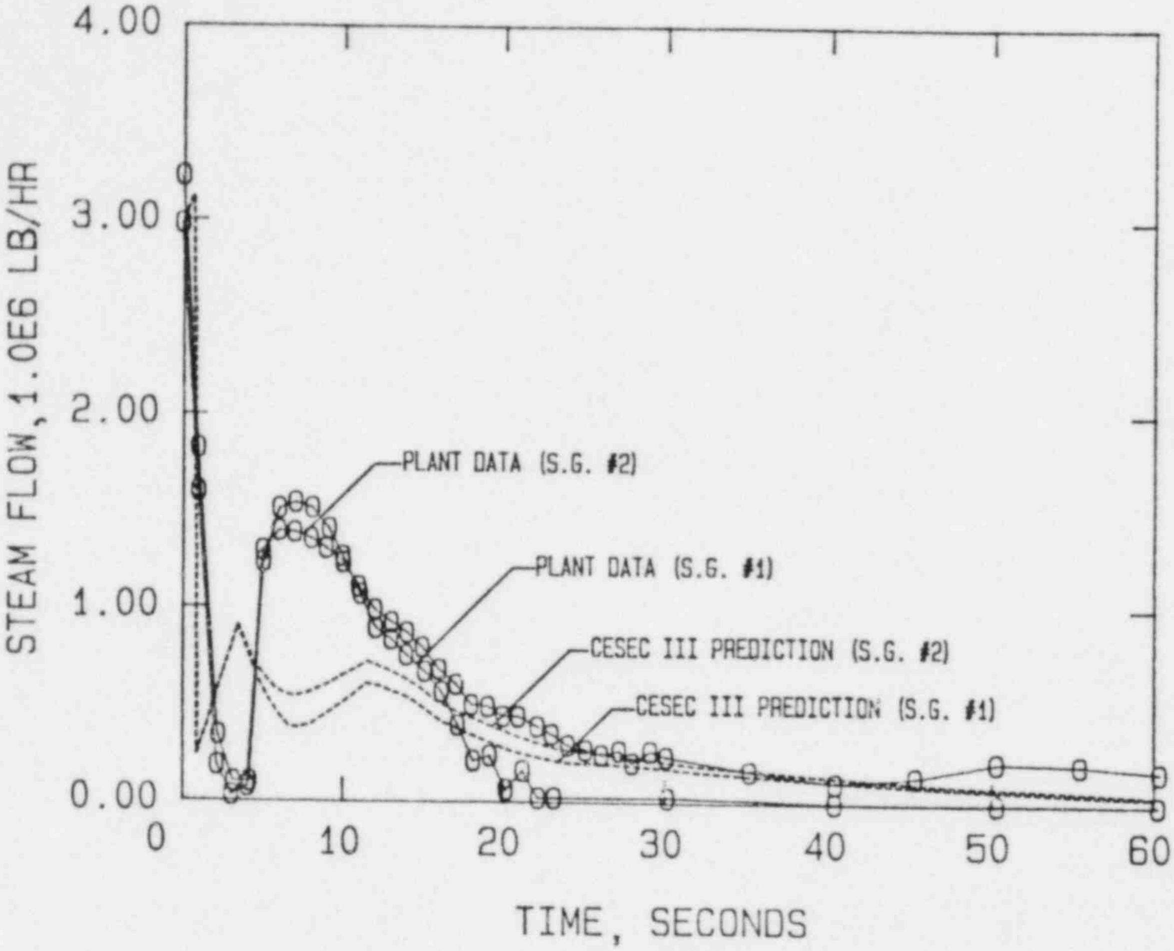
| Full Power Turbine Trip<br>Steam Generator Pressure vs Time | Omaha Public Power District<br>Fort Calhoun Station-Unit No. 1 | Figure<br>1-5 |
|---|---|---|




NOTE :

CYCLE 1 (FULL POWER = 1420 MWt)
PLANT DATA: TEST PERFORMED MAY 10, 1974

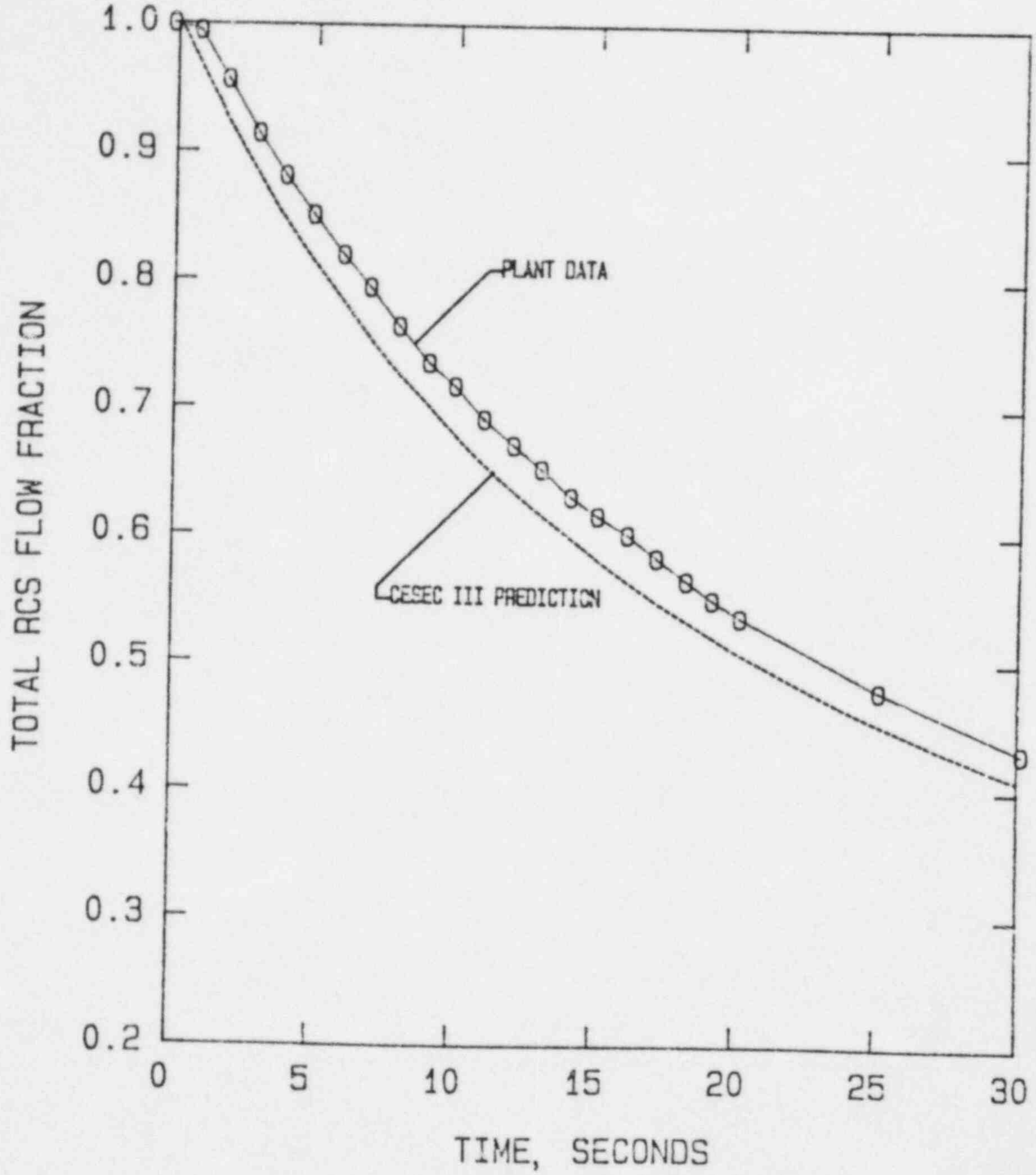
| | | |
|---|---|---|
| Full Power Turbine Trip<br>Main Feedwater Flow vs Time | Omaha Public Power District<br>Fort Calhoun Station-Unit No. 1 | Figure<br>1-6 |




NOTE :

CYCLE 1 (FULL POWER = 1420 MWt)
PLANT DATA: TEST PERFORMED MAY 10, 1974

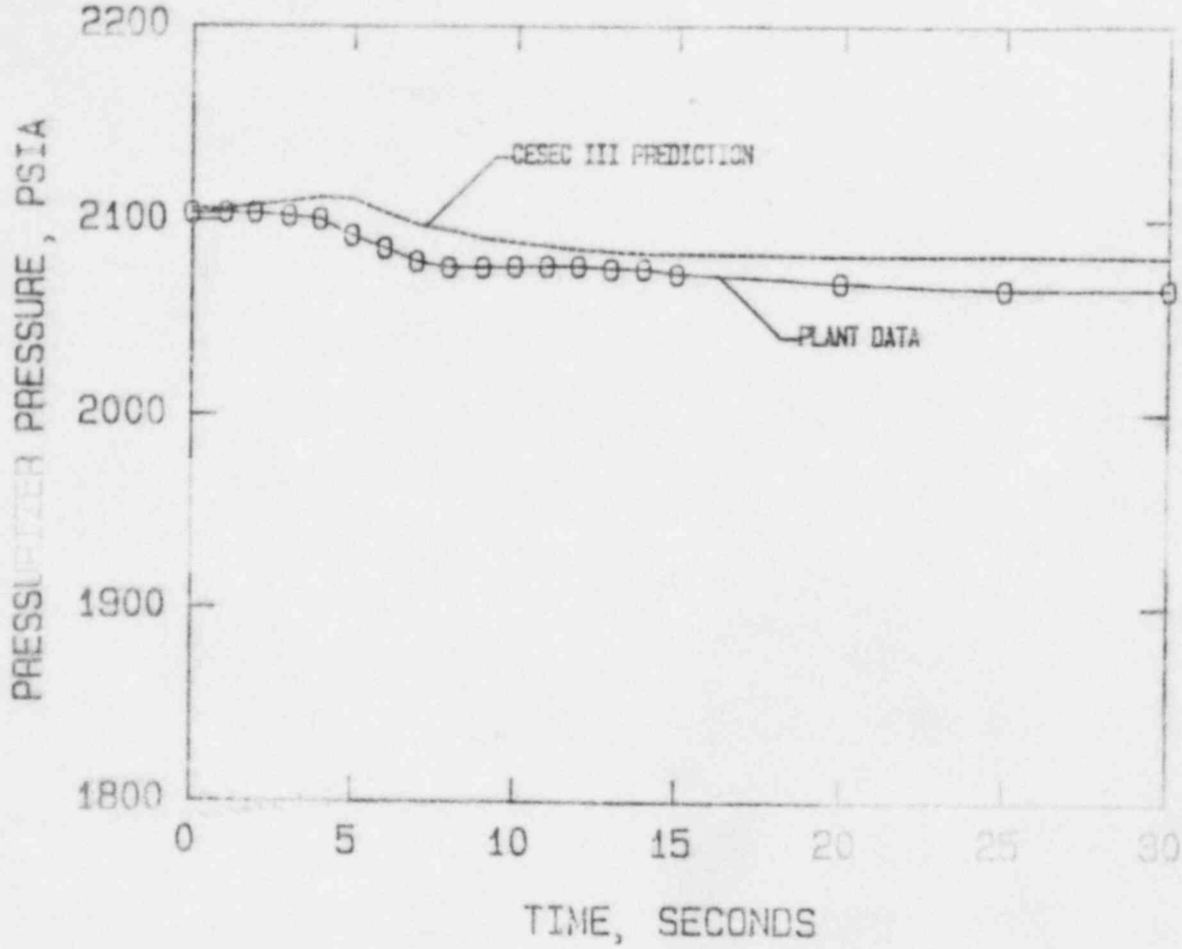
| Full Power Turbine Trip<br>Steam Flow vs Time | Omaha Public Power District<br>Fort Calhoun Station-Unit No. 1 | Figure<br>1-7 |
|---|---|---|




NOTE :

CYCLE 1 (FULL POWER = 1420 MWt)
INITIAL POWER = 35%
PLANT DATA: TEST PERFORMED MARCH 6, 1974

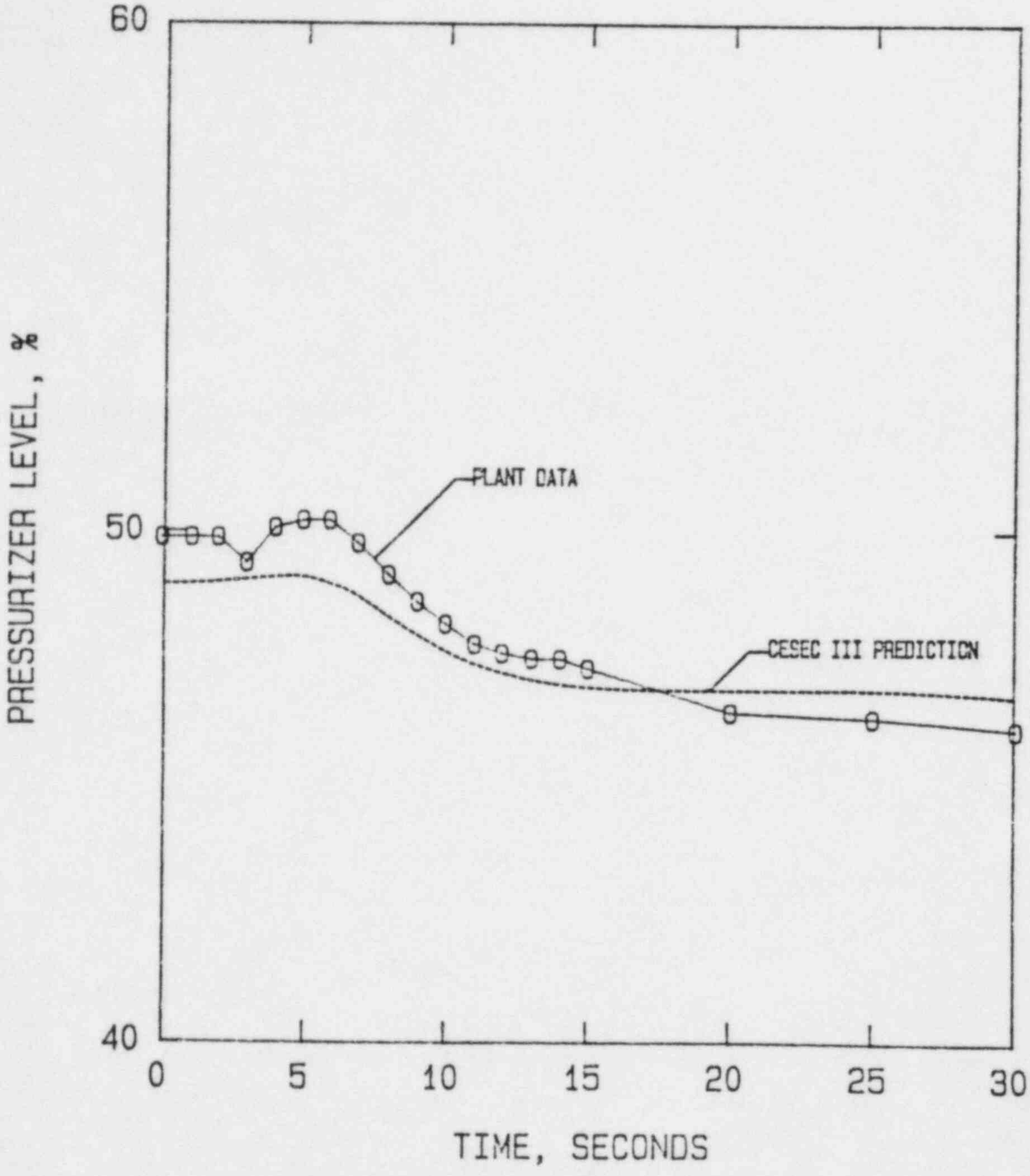
| 4 - Pump Loss Of Flow<br>Total RCS Flow vs Time | Omaha Public Power District<br>Fort Calhoun Station-Unit No. 1 | Figure<br>2-1 |
|---|---|---|




NOTE :

CYCLE 1 (FULL POWER = 1420 MWt)
INITIAL POWER = 35%
PLANT DATA: TEST PERFORMED MARCH 6, 1974

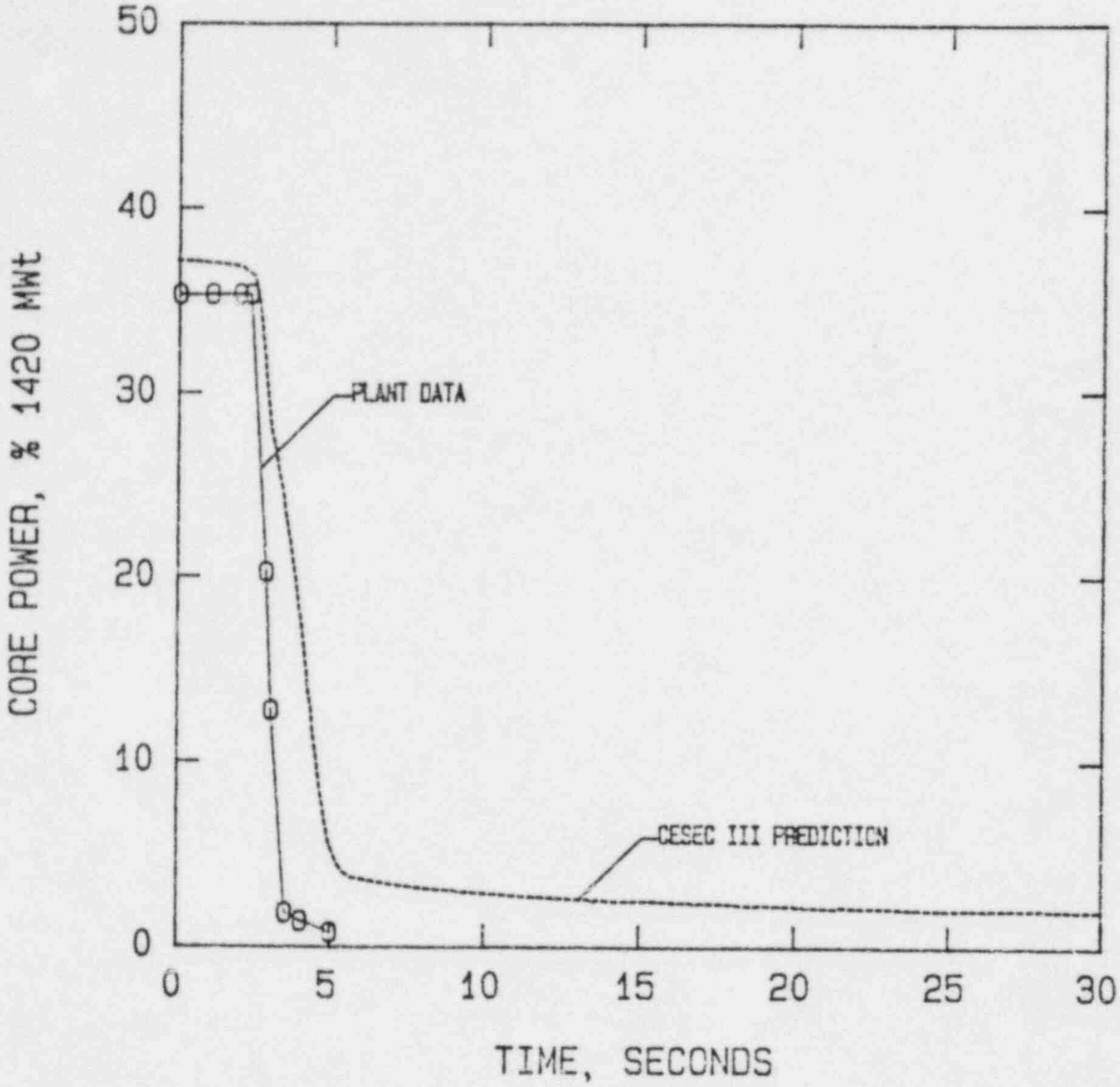
| 4 - Pump Loss Of Flow<br>Pressurizer Pressure vs Time | Omaha Public Power District<br>Fort Calhoun Station-Unit No. 1 | Figure<br>2-2 |
|---|---|---|




NOTE :

CYCLE 1 (FULL POWER = 1420 MWt)
INITIAL POWER = 35%
PLANT DATA: TEST PERFORMED MARCH 6, 1974

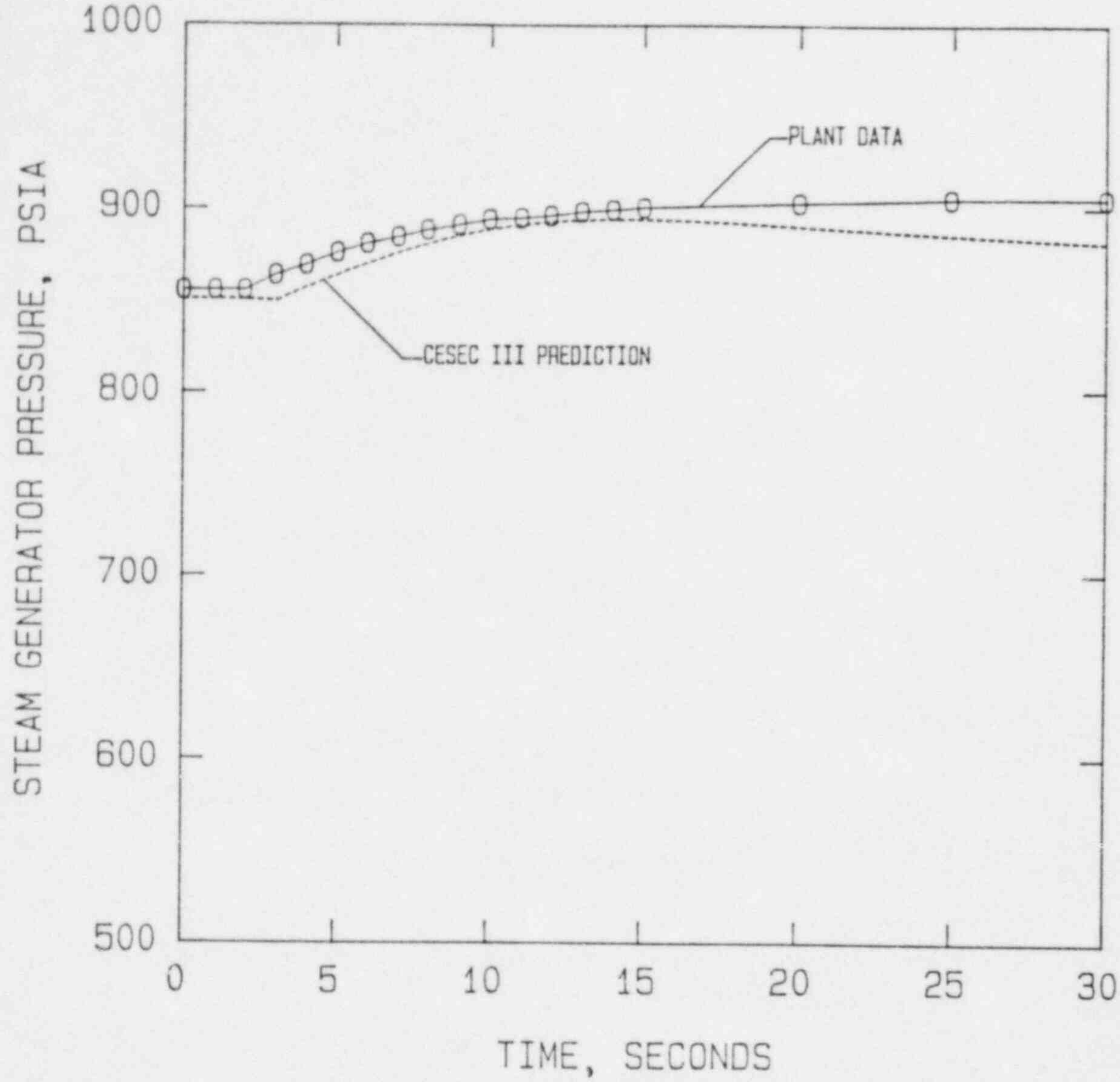
| 4 - Pump Loss Of Flow<br>Pressurizer Level vs Time | Omaha Public Power District<br>Fort Calhoun Station-Unit No. 1 | Figure<br>2-3 |
|---|---|---|




NOTE :

CYCLE 1 (FULL POWER = 1420 MWt)
PLANT DATA: TEST PERFORMED MARCH 6, 1974

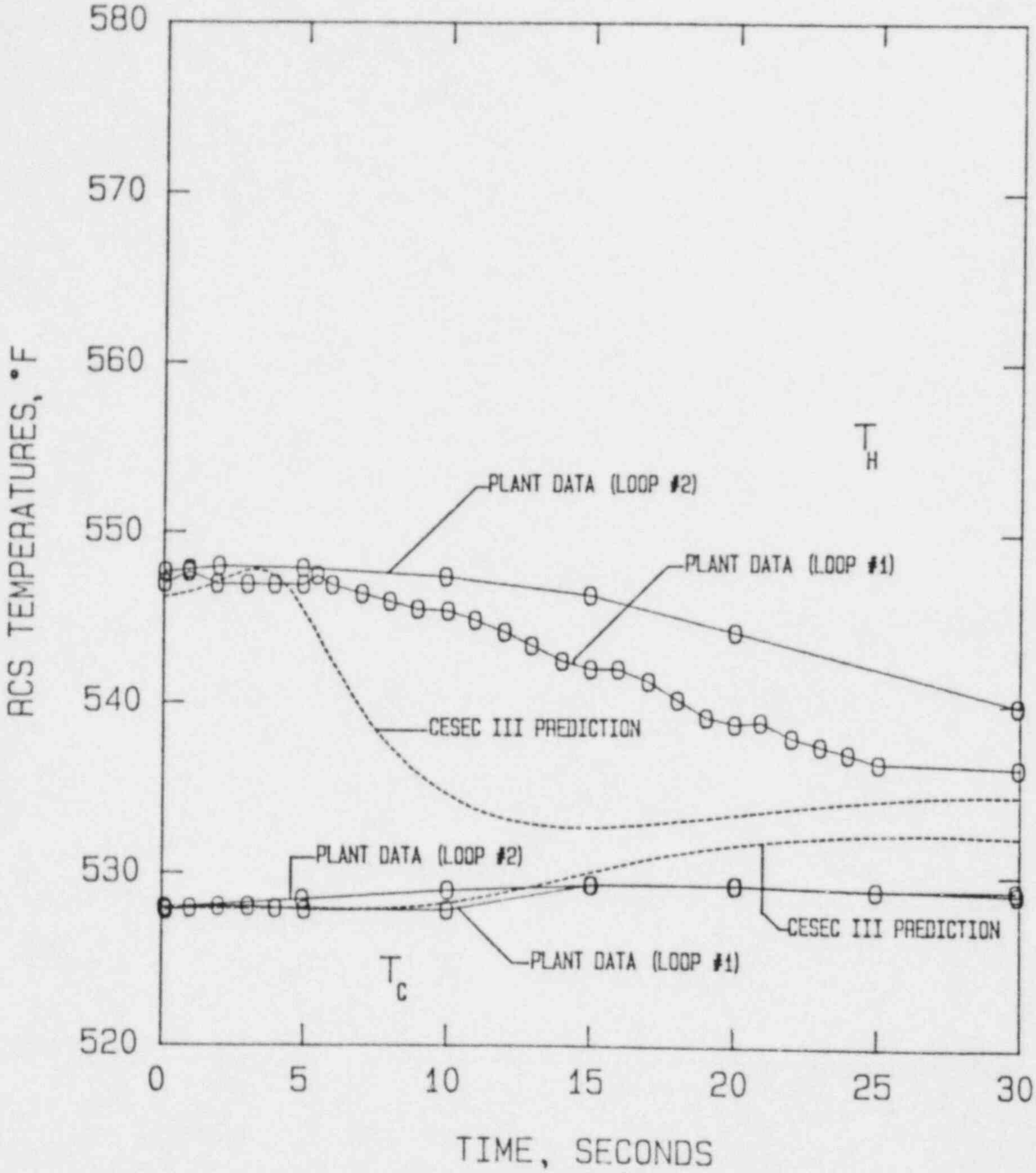
| | | |
|---|---|---|
| 4 - Pump Loss Of Flow<br>Core Power vs Time | Omaha Public Power District<br>Fort Calhoun Station-Unit No. 1 | Figure<br>2-4 |




NOTE :

CYCLE 1 (FULL POWER = 1420 MWt)
INITIAL POWER = 35%
PLANT DATA: TEST PERFORMED MARCH 6, 1974

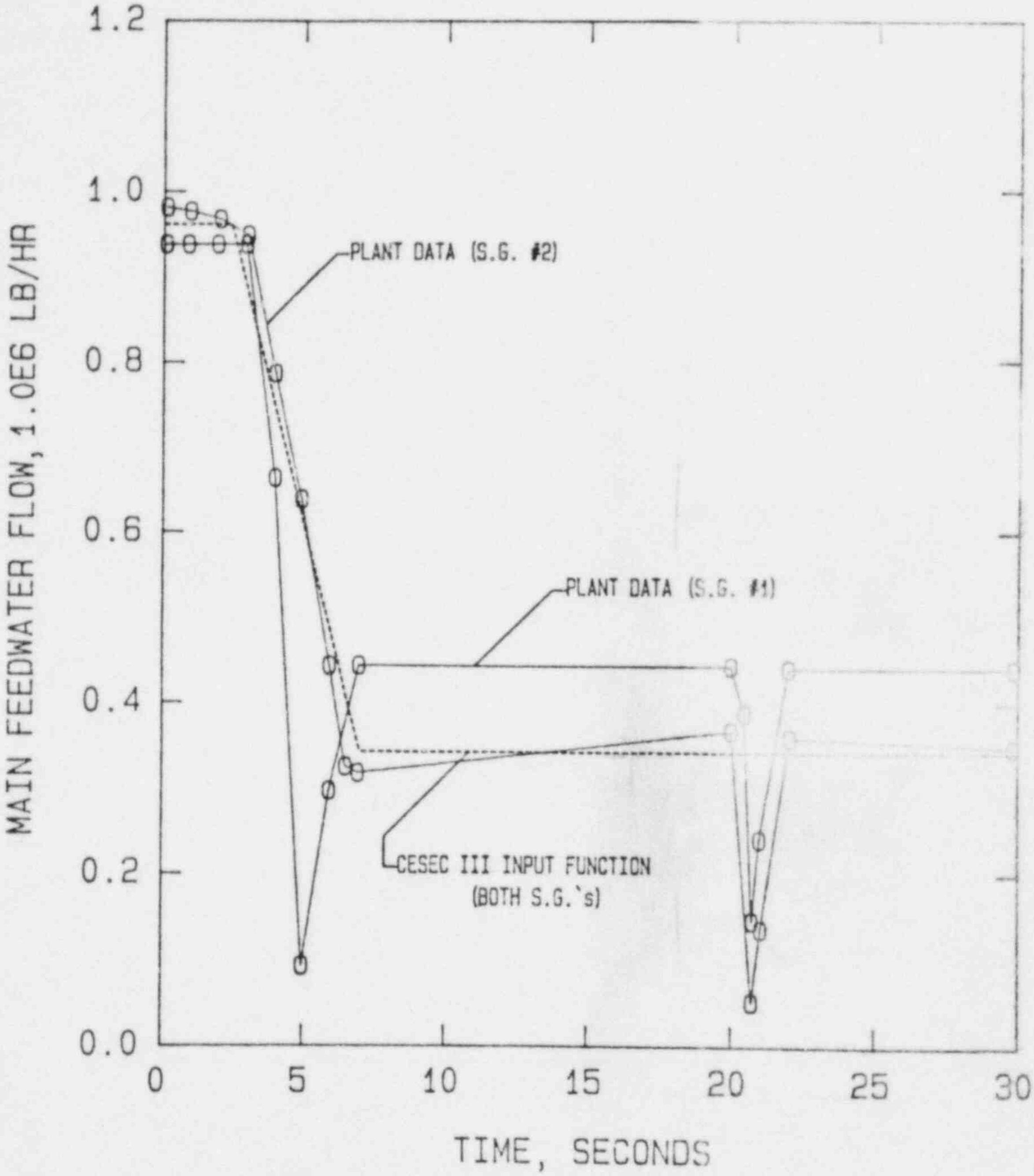
| 4 - Pump Loss Of Flow<br>Steam Generator Pressure vs Time | Omaha Public Power District<br>Fort Calhoun Station-Unit No. 1 | Figure<br>2-5 |
|---|---|---|




NOTE :

CYCLE 1 (FULL POWER = 1420 MWt)
INITIAL POWER = 35%
PLANT DATA: TEST PERFORMED MARCH 6, 1974

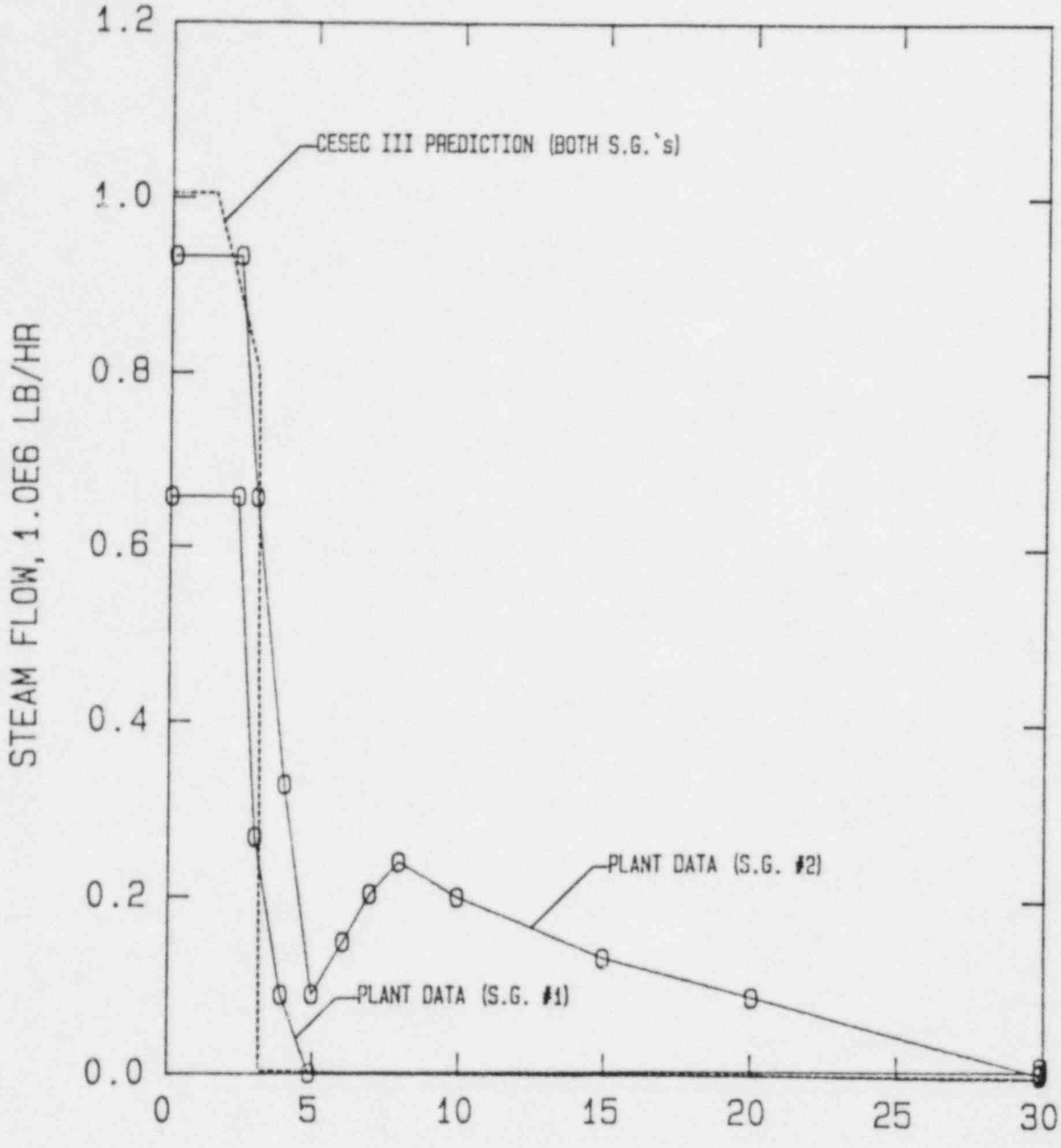
| 4 - Pump Loss Of Flow<br>RCS Temperatures vs Time | Omaha Public Power District<br>Fort Calhoun Station-Unit No. 1 | Figure<br>2-6 |
|---|---|---|




NOTE :

CYCLE 1 (FULL POWER = 1420 MWt)
INITIAL POWER = 35%
PLANT DATA: TEST PERFORMED MARCH 6, 1974

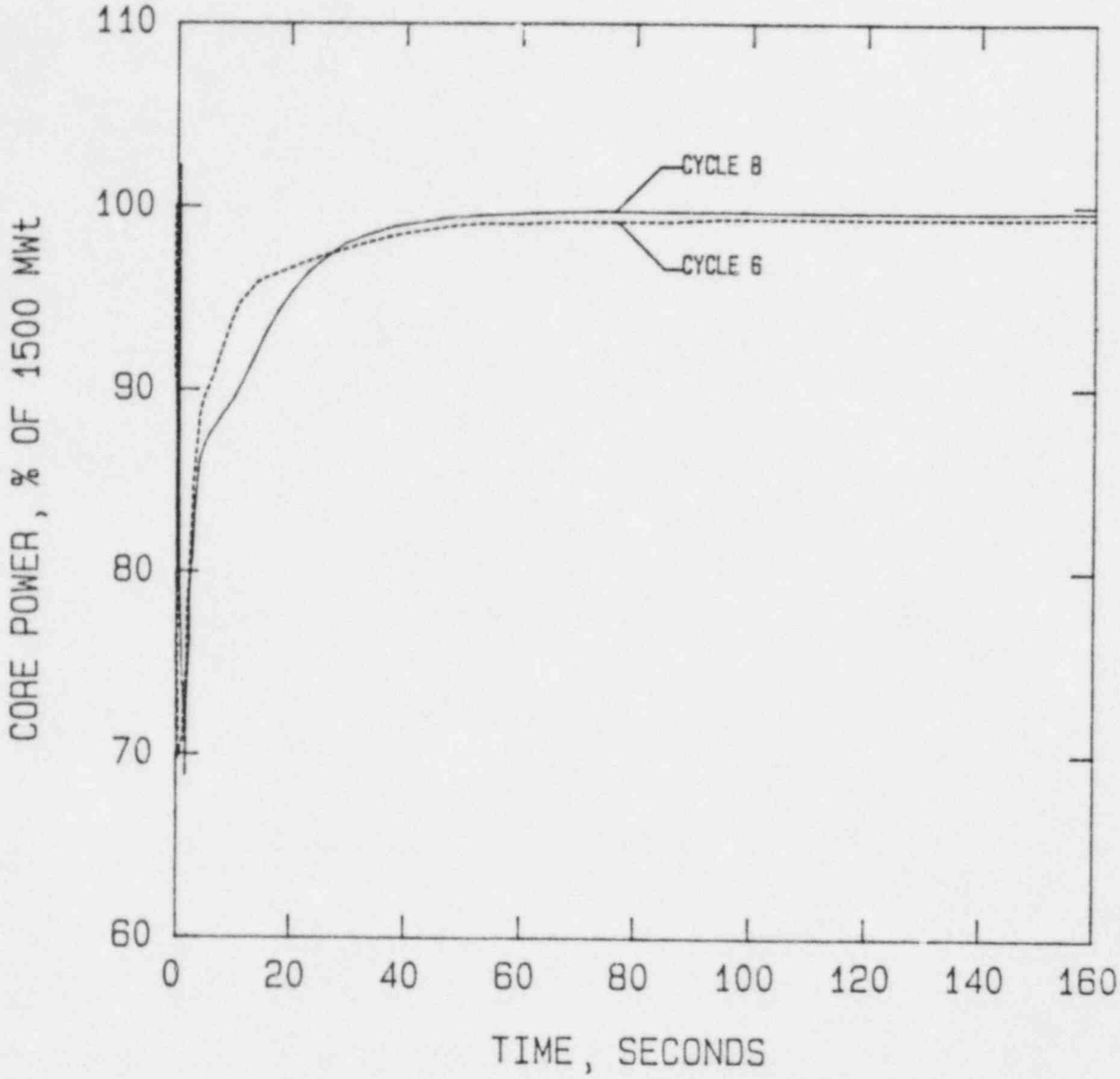
| 4 - Pump Loss Of Flow<br>Main Feedwater Flow vs Time | Omaha Public Power District<br>Fort Calhoun Station-Unit No. 1 | Figure<br>2-7 |
|---|---|---|




NOTE :

CYCLE 1 (FULL POWER = 1420 MWt)
INITIAL POWER = 35%
PLANT DATA: TEST PERFORMED MARCH 6, 1974

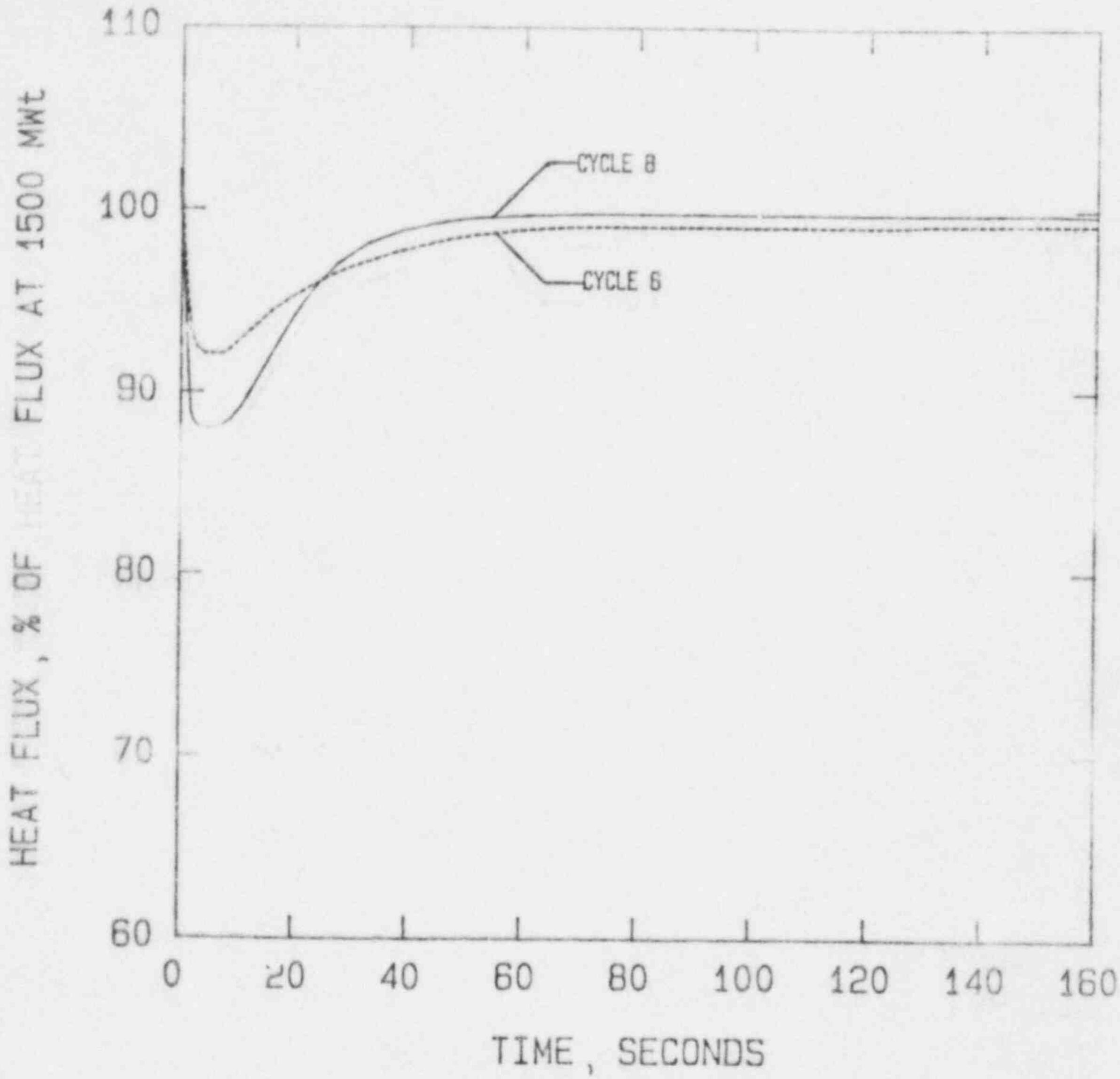
| 4 - Pump Loss Of Flow<br>Steam Flow vs Time | Omaha Public Power District<br>Fort Calhoun Station-Unit No. 1 | Figure<br>2-8 |
|---|---|---|




NOTE :

CYCLE 6: ENC ANALYSIS
CYCLE 8: OPPD ANALYSIS

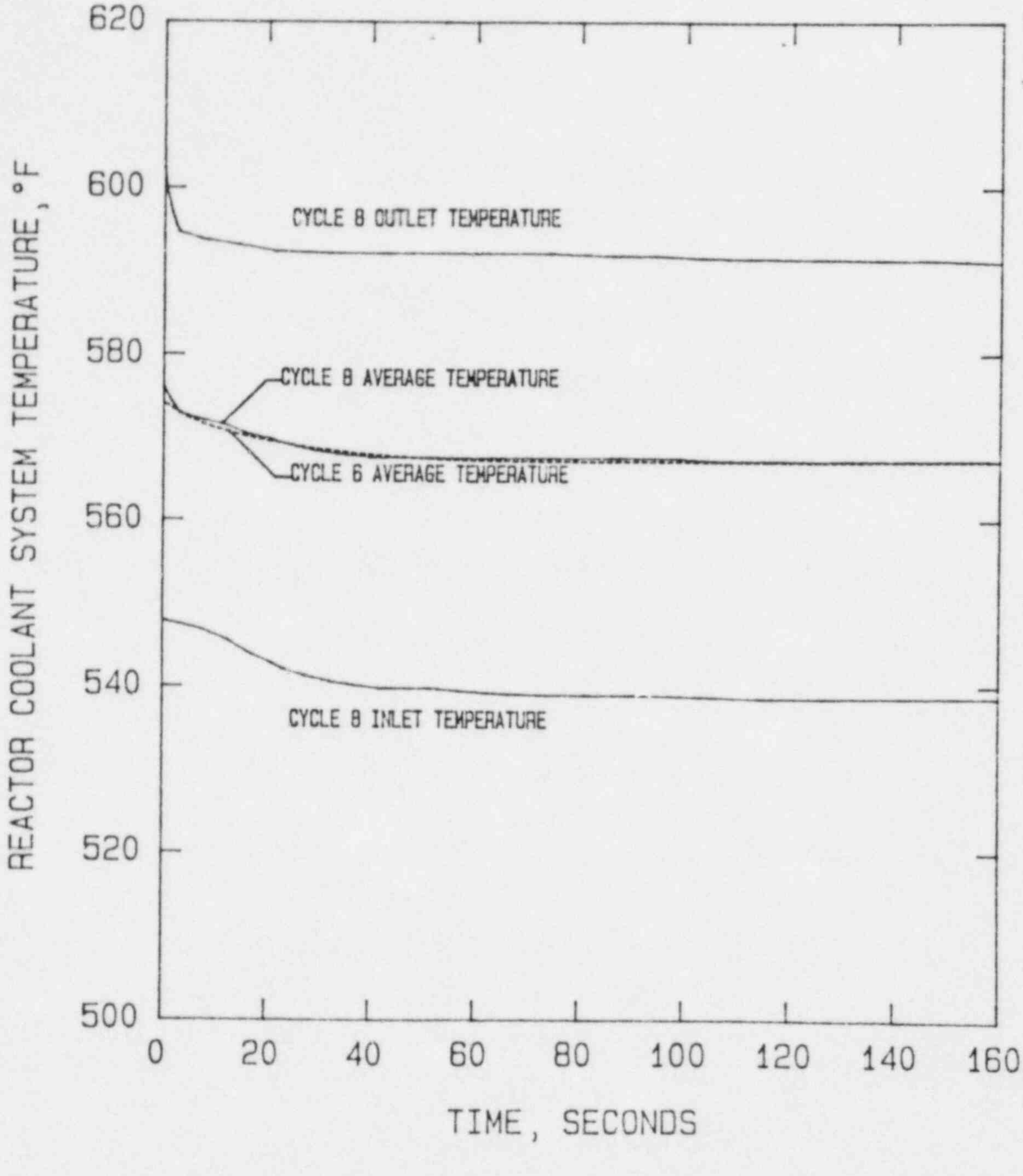
| | | |
|---|---|---|
| CEA Drop Incident<br>Core Power vs Time | Omaha Public Power District<br>Fort Calhoun Station-Unit No. 1 | Figure<br>3-1 |




NOTE :

CYCLE 6: ENC ANALYSIS
CYCLE 8: OPPD ANALYSIS

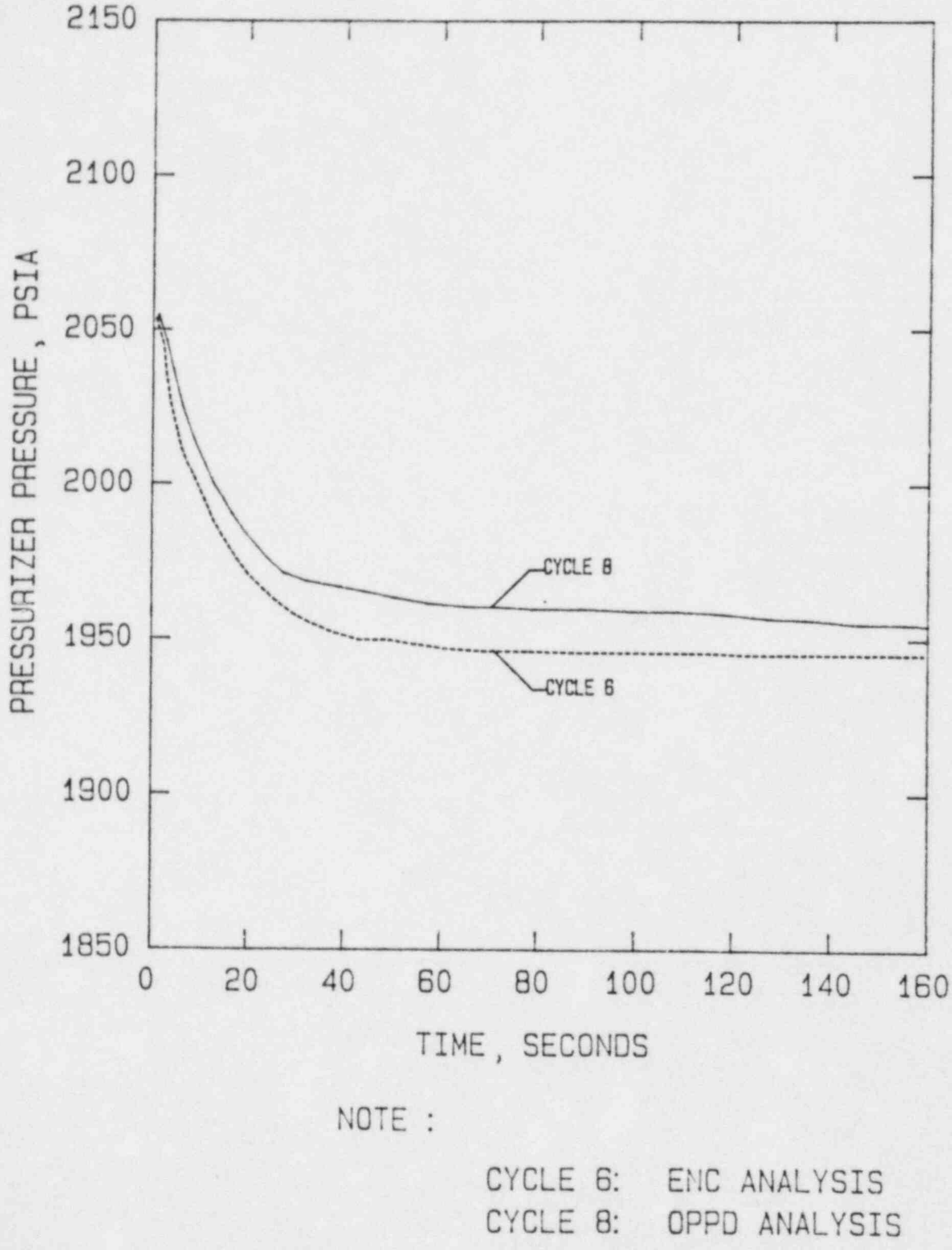
| | | |
|---|---|---|
| CEA Drop Incident<br>Core Average Heat Flux vs Time | Omaha Public Power District<br>Fort Calhoun Station-Unit No. 1 | Figure<br>3-2 |



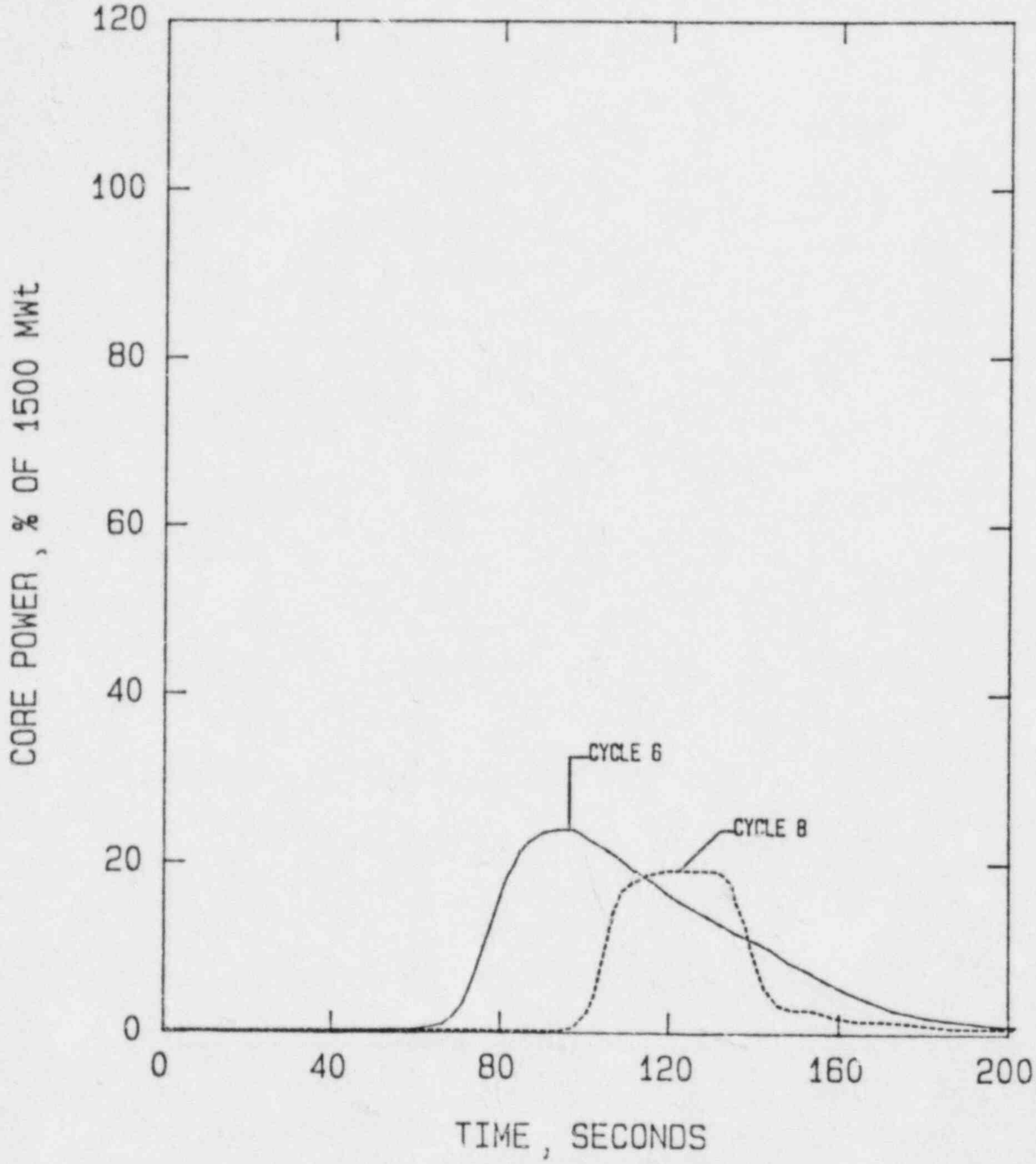

NOTE :

CYCLE 6: ENC ANALYSIS
CYCLE 8: OPPD ANALYSIS

| CEA Drop Incident<br>Coolant Temperature vs Time | Omaha Public Power District<br>Fort Calhoun Station-Unit No. 1 | Figure<br>3-3 |
|---|---|---|



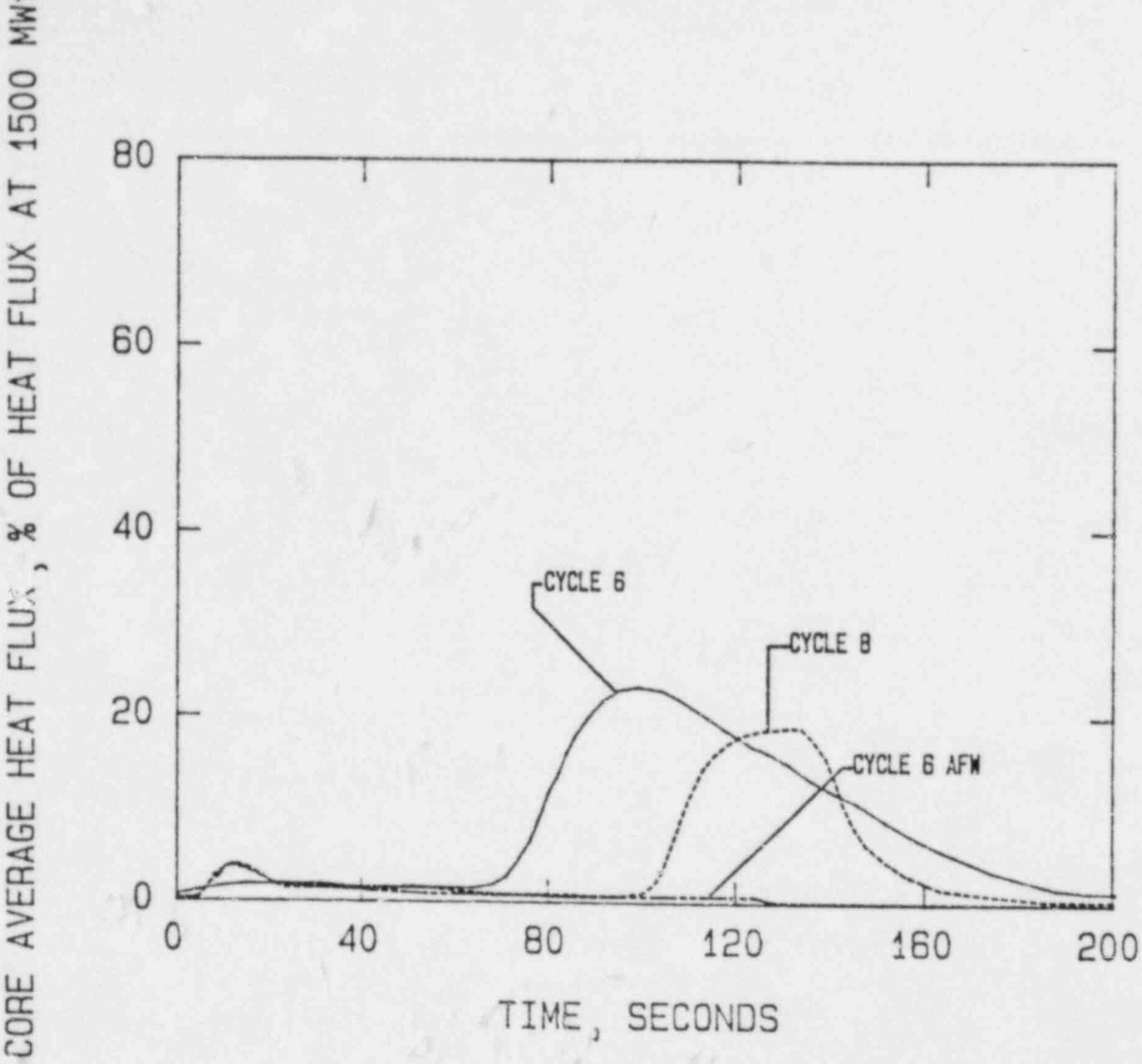

| CEA Drop Incident<br>Pressurizer Pressure vs Time | Omaha Public Power District<br>Fort Calhoun Station-Unit No. 1 | Figure<br>3-4 |
|---|---|---|




NOTE :

CYCLE 6: ENC ANALYSIS WITH SDM= 3.0% $\Delta\rho$
CYCLE 8: OPPD ANALYSIS WITH SDM= 4.0% $\Delta\rho$

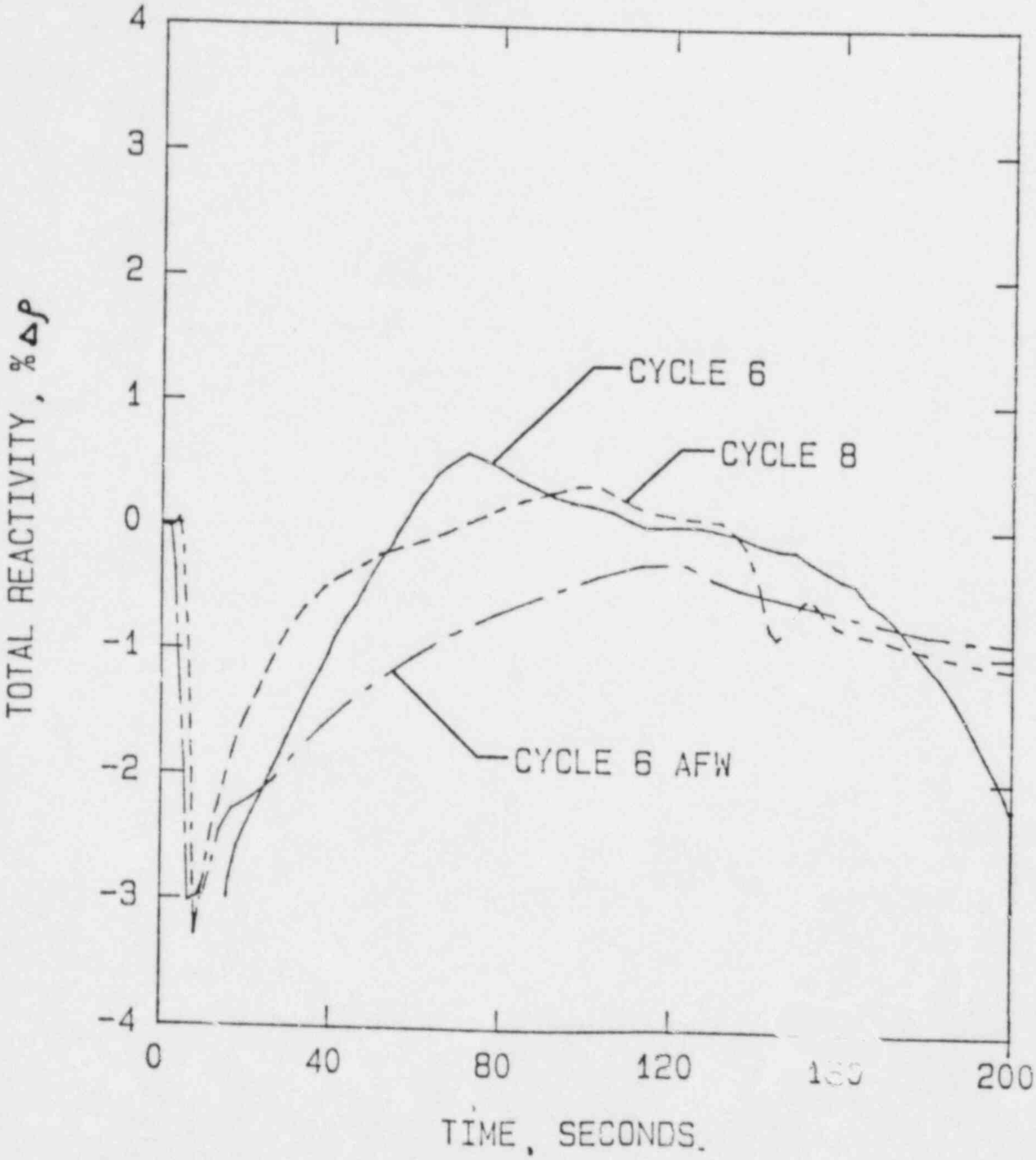
| Zero Power Steam Line Break Incident<br>Core Power vs Time | Omaha Public Power District<br>Fort Calhoun Station-Unit No. 1 | Figure<br>4-1 |
|---|---|---|




NOTE :

CYCLE 6: ENC ANALYSIS WITH SDM= 3.0% $\Delta\rho$
CYCLE 6 AFW: CE ANALYSIS WITH SDM= 4.2% $\Delta\rho$
CYCLE 8: OPPD ANALYSIS WITH SDM= 4.0% $\Delta\rho$

| Zero Power Steam Line Break Incident<br>Core Average Heat Flux vs Time | Omaha Public Power District<br>Fort Calhoun Station-Unit No. 1 | Figure<br>4-2 |
|---|---|---|



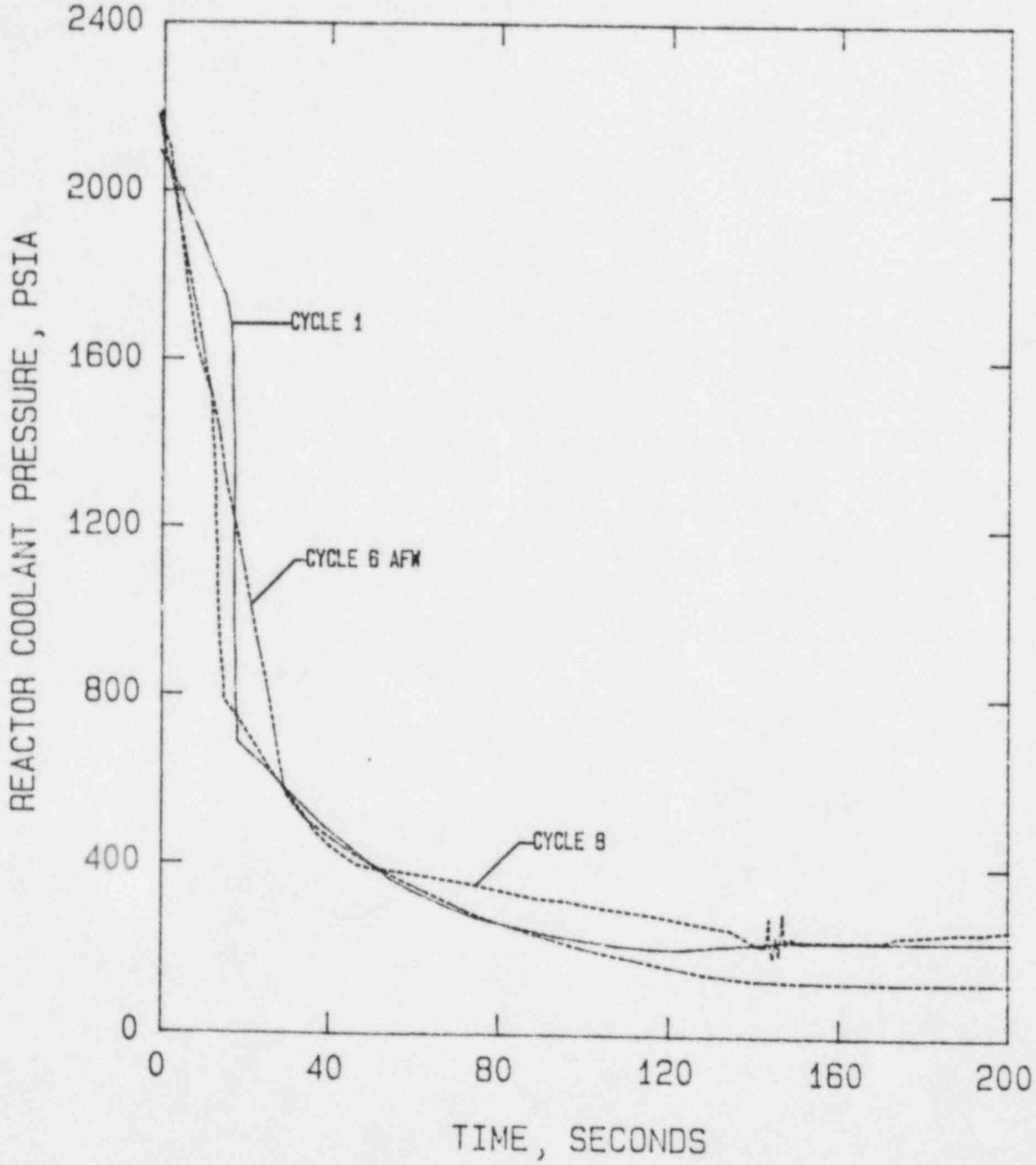

NOTE:

CYCLE 6: ENC ANALYSIS WITH SDM= 3.0% $\Delta\rho$

CYCLE 6 AFW: CE ANALYSIS WITH SDM= 4.2% $\Delta\rho$

CYCLE 8: OPPD ANALYSIS WITH SDM= 4.0% $\Delta\rho$

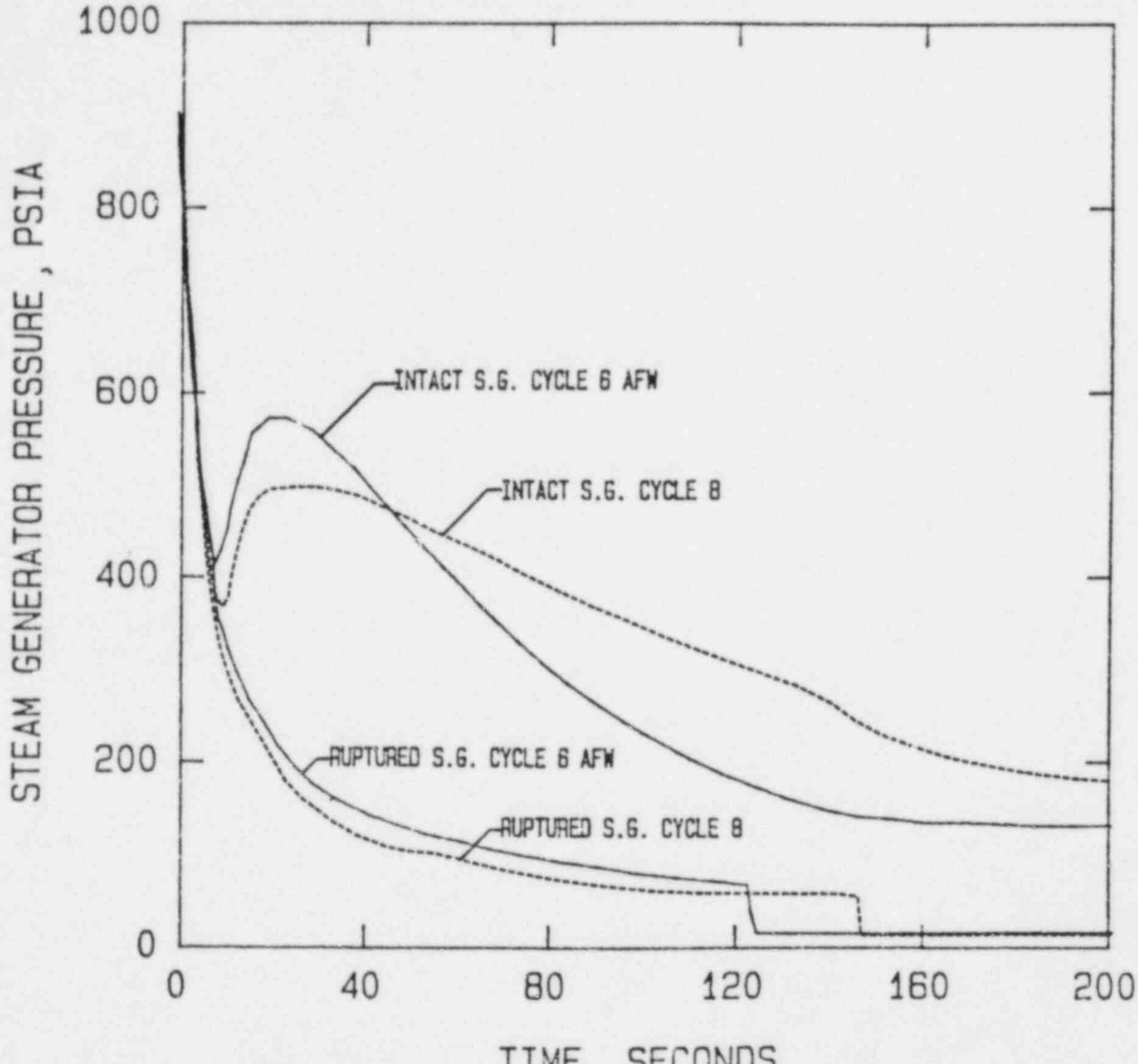
| | | |
|---|---|---|
| Zero Power Steam Line Break Incident<br>Total Reactivity vs Time | Omaha Public Power District<br>Fort Calhoun Station-Unit No. 1 | Figure<br>4-3 |




NOTE :

CYCLE 1: CE ANALYSIS
CYCLE 6 AFW: CE ANALYSIS
CYCLE 8: OPPD ANALYSIS

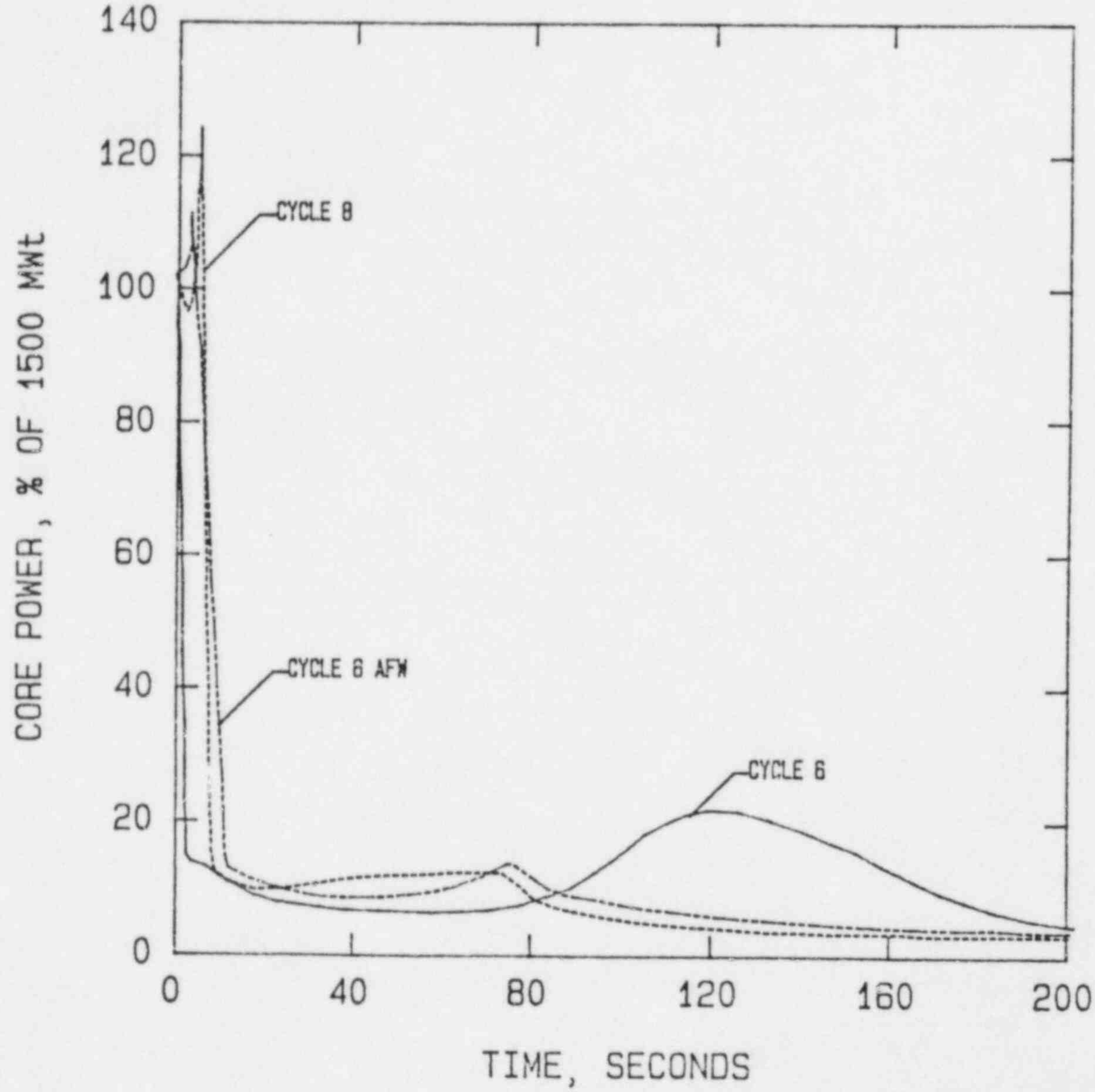
| | | |
|---|---|---|
| Zero Power Steam Line Break Incident<br>Coolant System Pressure vs Time | Omaha Public Power District<br>Fort Calhoun Station-Unit No. 1 | Figure<br>4-4 |




NOTE :

CYCLE 6 AFW: CE ANALYSIS
CYCLE 8: OPPD ANALYSIS

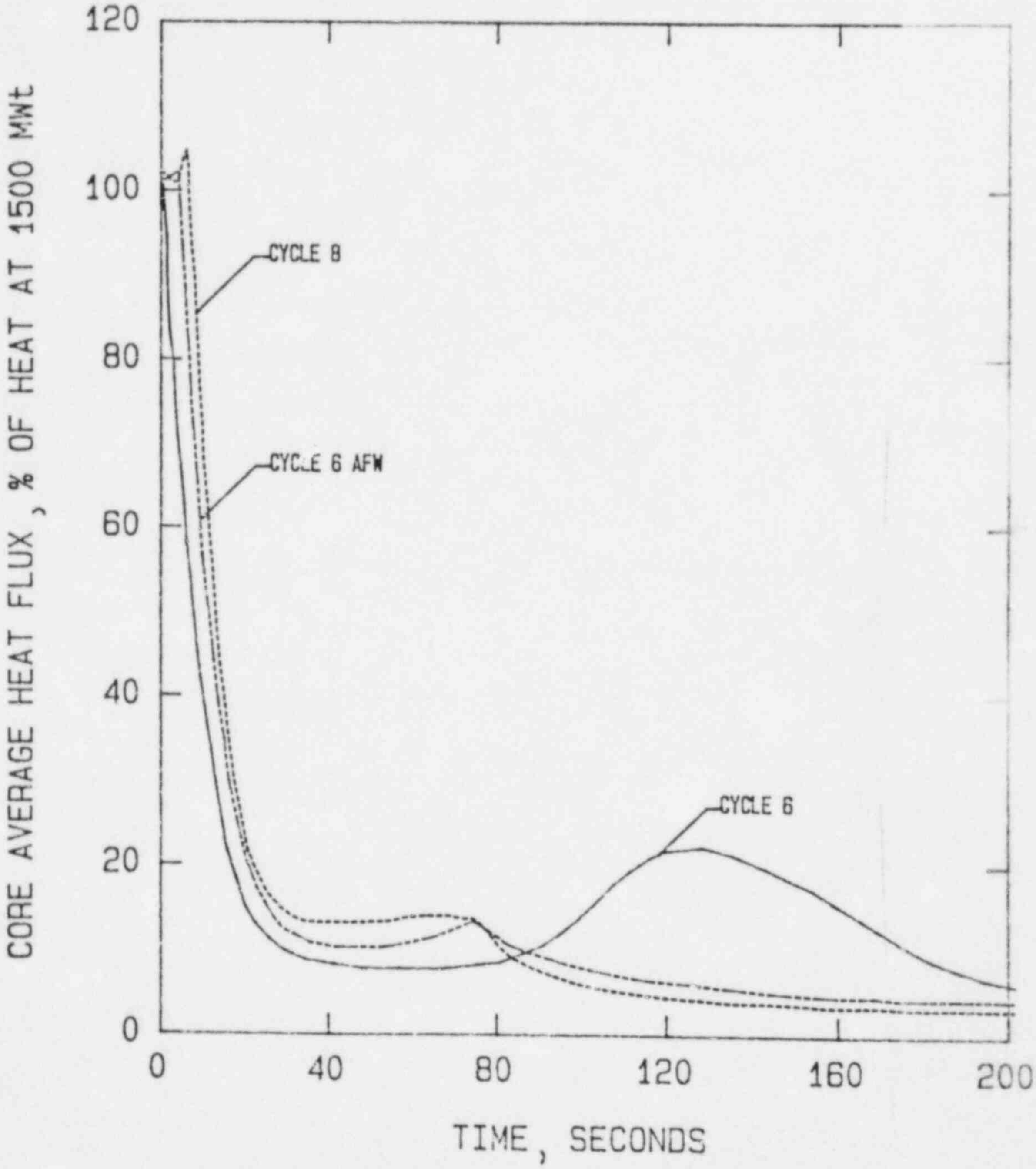
| Zero Power Steam Line Break Incident<br>Steam Generator Pressure vs Time | Omaha Public Power District<br>Fort Calhoun Station-Unit No. 1 | Figure<br>4-5 |
|---|---|---|




NOTE :

CYCLE 6: ENC ANALYSIS (CEA WORTH= 5.81%$\Delta\rho$)
CYCLE 6 AFW: CE ANALYSIS (CEA WORTH= 5.81%$\Delta\rho$)
CYCLE 8: OPPD ANALYSIS (CEA WORTH= 6.57%$\Delta\rho$)

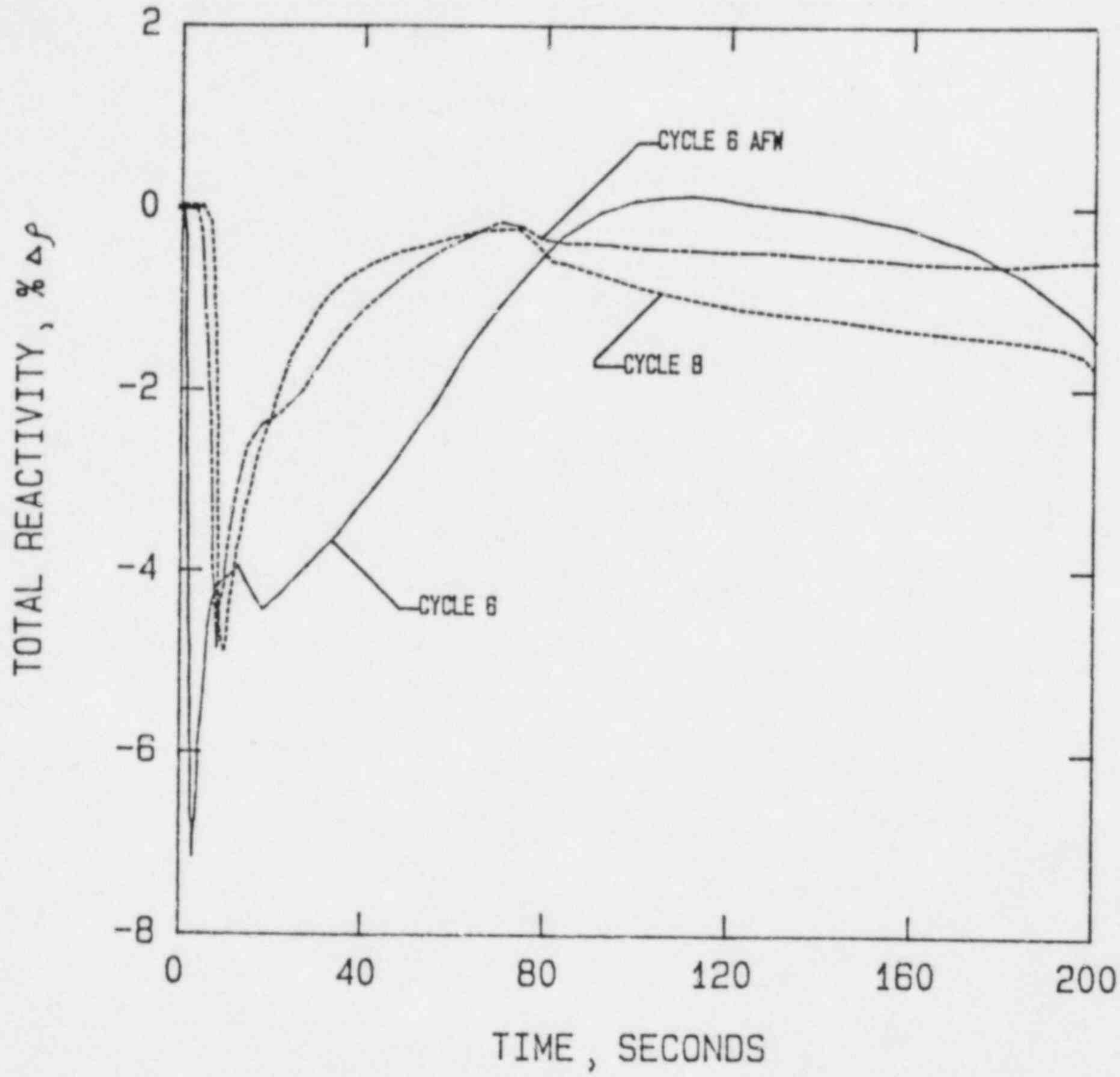
| | | |
|---|---|---|
| Full Power Steam Line Break Incident<br>Core Power vs Time | Omaha Public Power District<br>Fort Calhoun Station-Unit No. 1 | Figure<br>5-1 |




NOTE :

CYCLE 6: ENC ANALYSIS
CYCLE 6 AFW: CE ANALYSIS
CYCLE 8: OPPD ANALYSIS

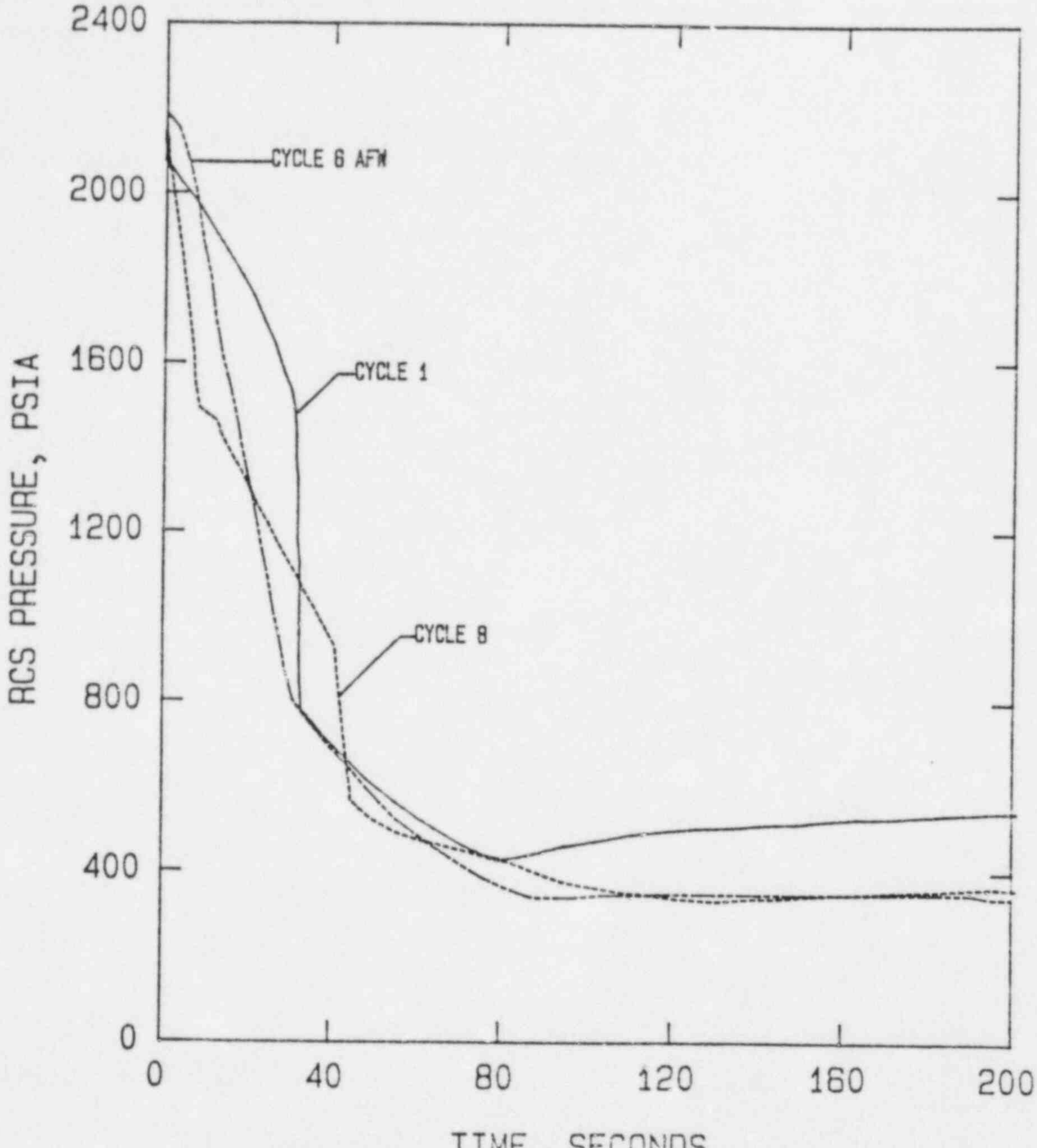
| Full Power Steam Line Break Incident<br>Core Average Heat Flux vs Time | Omaha Public Power District<br>Fort Calhoun Station-Unit No. 1 | Figure<br>5-2 |
|---|---|---|




NOTE :

CYCLE 6: ENC ANALYSIS (CEA WORTH= 5.81%Δρ)
CYCLE 5 AFW: CE ANALYSIS (CEA WORTH= 5.81%Δρ)
CYCLE 8: OPPD ANALYSIS (CEA WORTH= 6.57%Δρ)

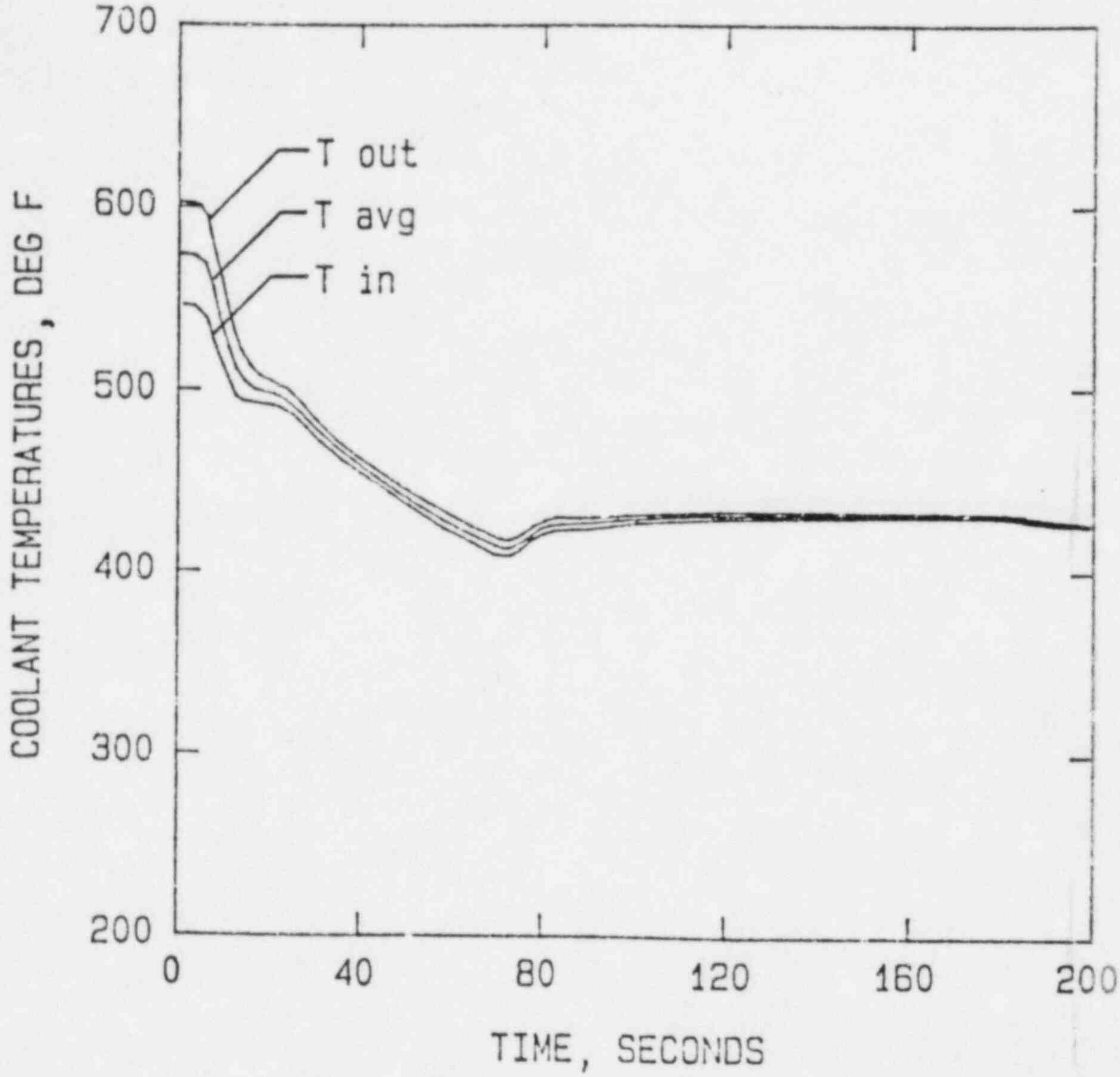
| | | |
|---|---|---|
| Full Power Steam Line Break Incident<br>Total Reactivity vs Time | Omana Public Power District<br>Fort Calhoun Station-Unit No. 1 | Figure<br>5-3 |




NOTE :

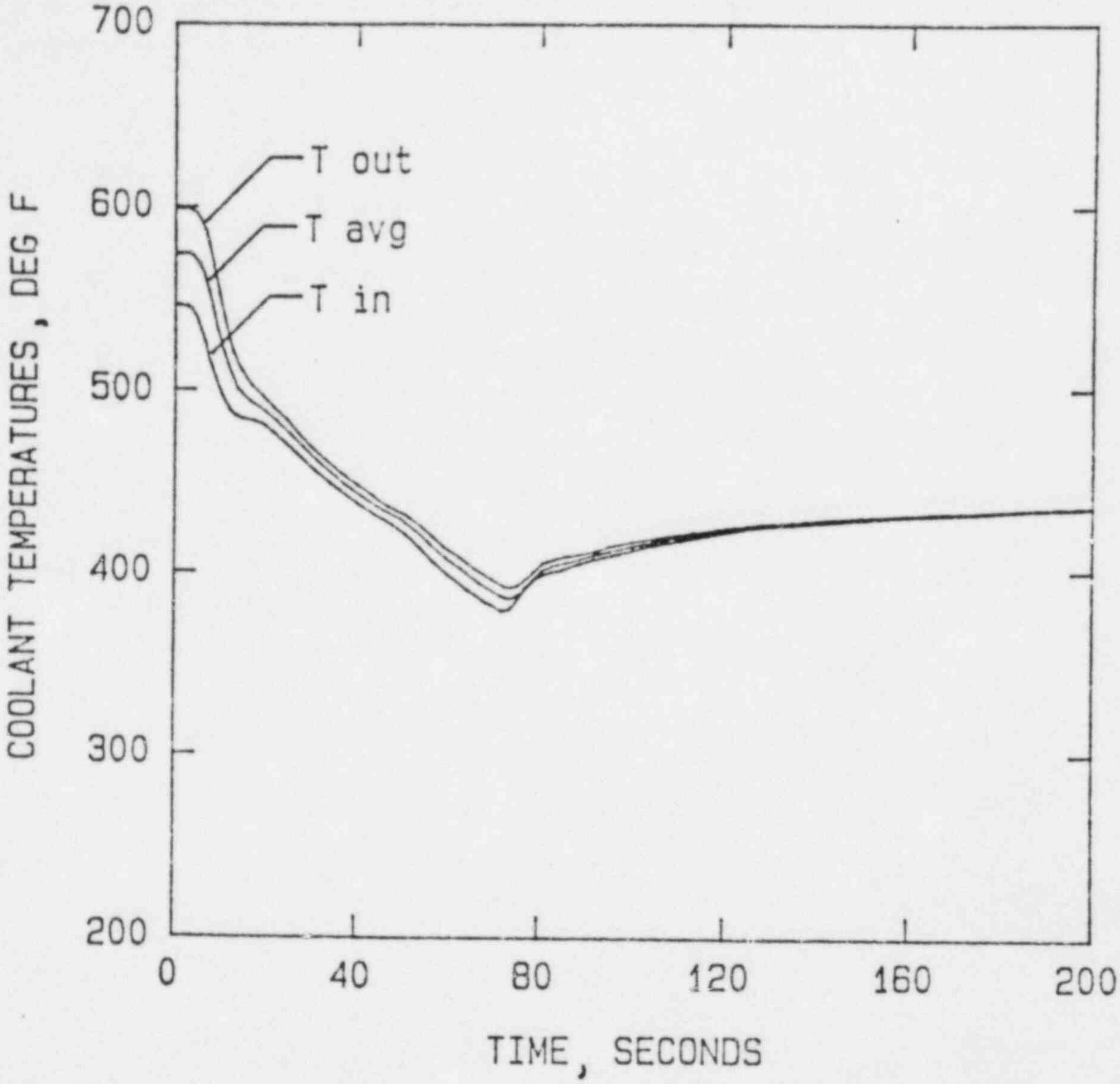
CYCLE 1: CE ANALYSIS
CYCLE 6 AFW: CE ANALYSIS
CYCLE 8: OPPD ANALYSIS

| Full Power Steam Line Break Incident<br>Coolant System Pressure vs Time | Omaha Public Power District<br>Fort Calhoun Station-Unit No. 1 | Figure<br>5-4 |
|---|---|---|




NOTE:
CYCLE 6 AFW: CE ANALYSIS

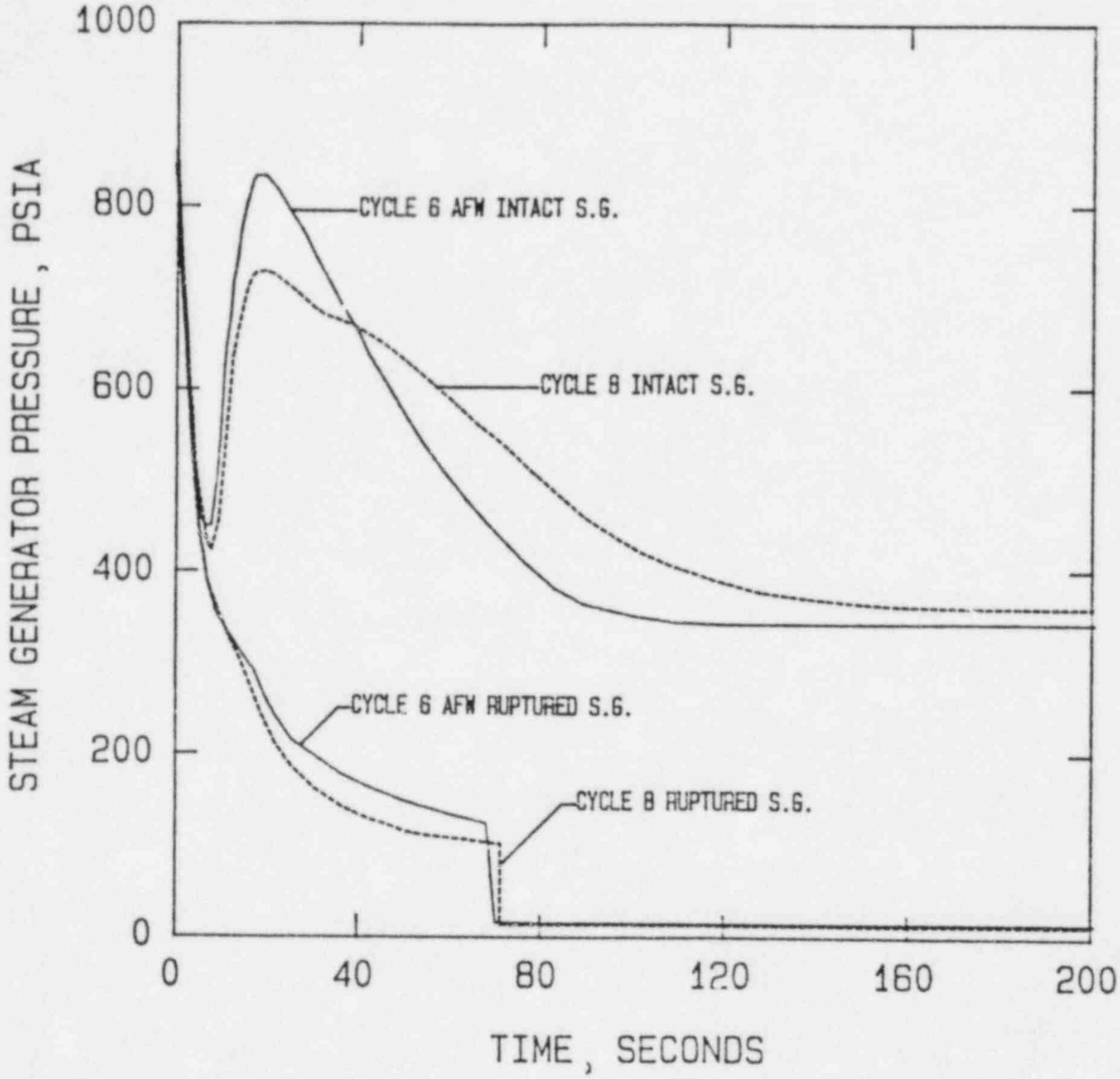
| | | |
|---|---|---|
| Full Power Steam Line Break Incident<br>Reactor Coolant Temperatures vs Time | Omaha Public Power District<br>Fort Calhoun Station-Unit No. 1 | Figure<br>5-6A |




NOTE:

CYCLE 8: OPPD ANALYSIS

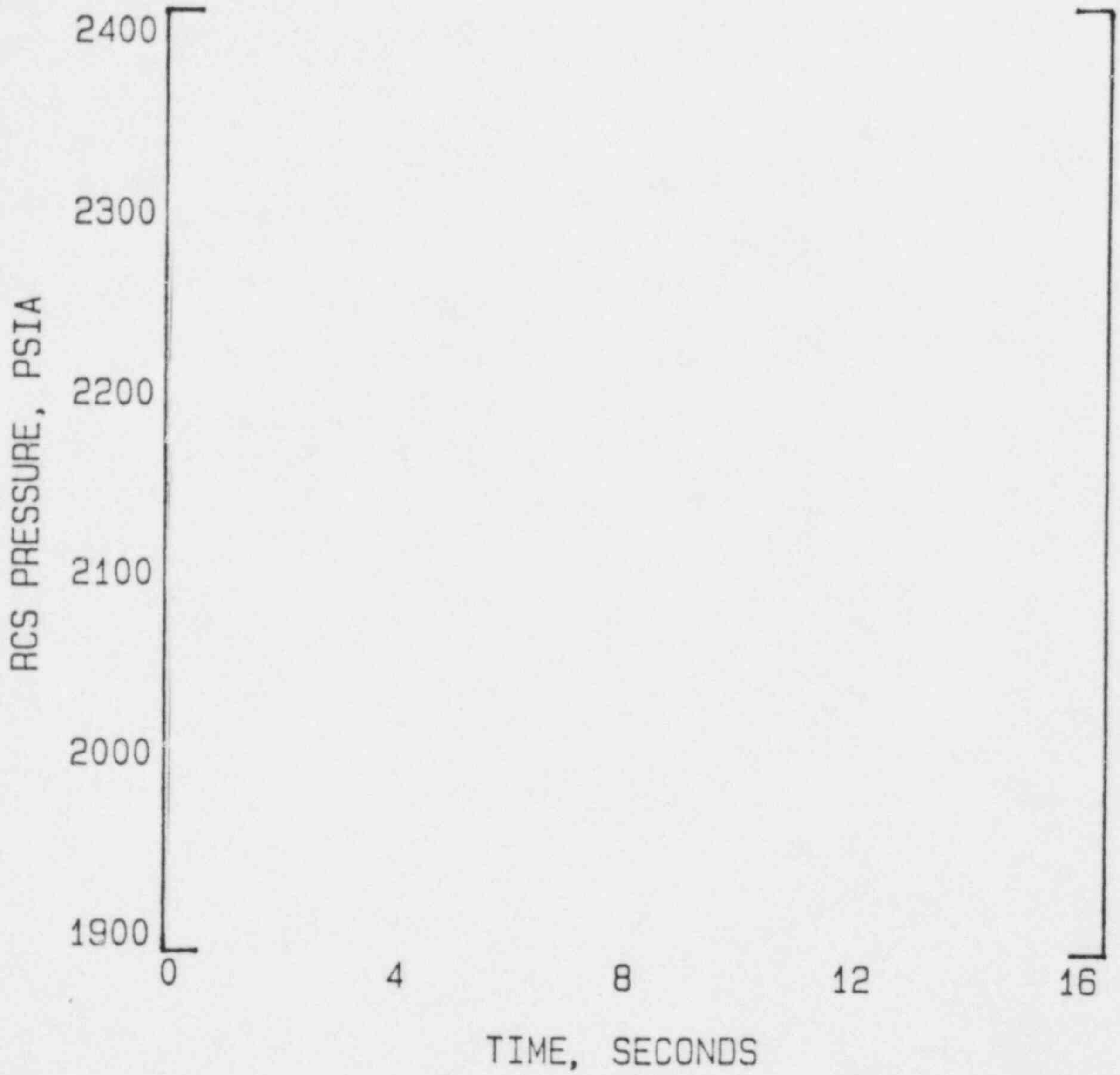
| Full Power Steam Line Break Incident<br>Reactor Coolant Temperatures vs Time | Omaha Public Power District<br>Fort Calhoun Station-Unit No. 1 | Figure<br>5-5B |
|---|---|---|




NOTE :

CYCLE 6 AFW: CE ANALYSIS
CYCLE 8: OPPD ANALYSIS

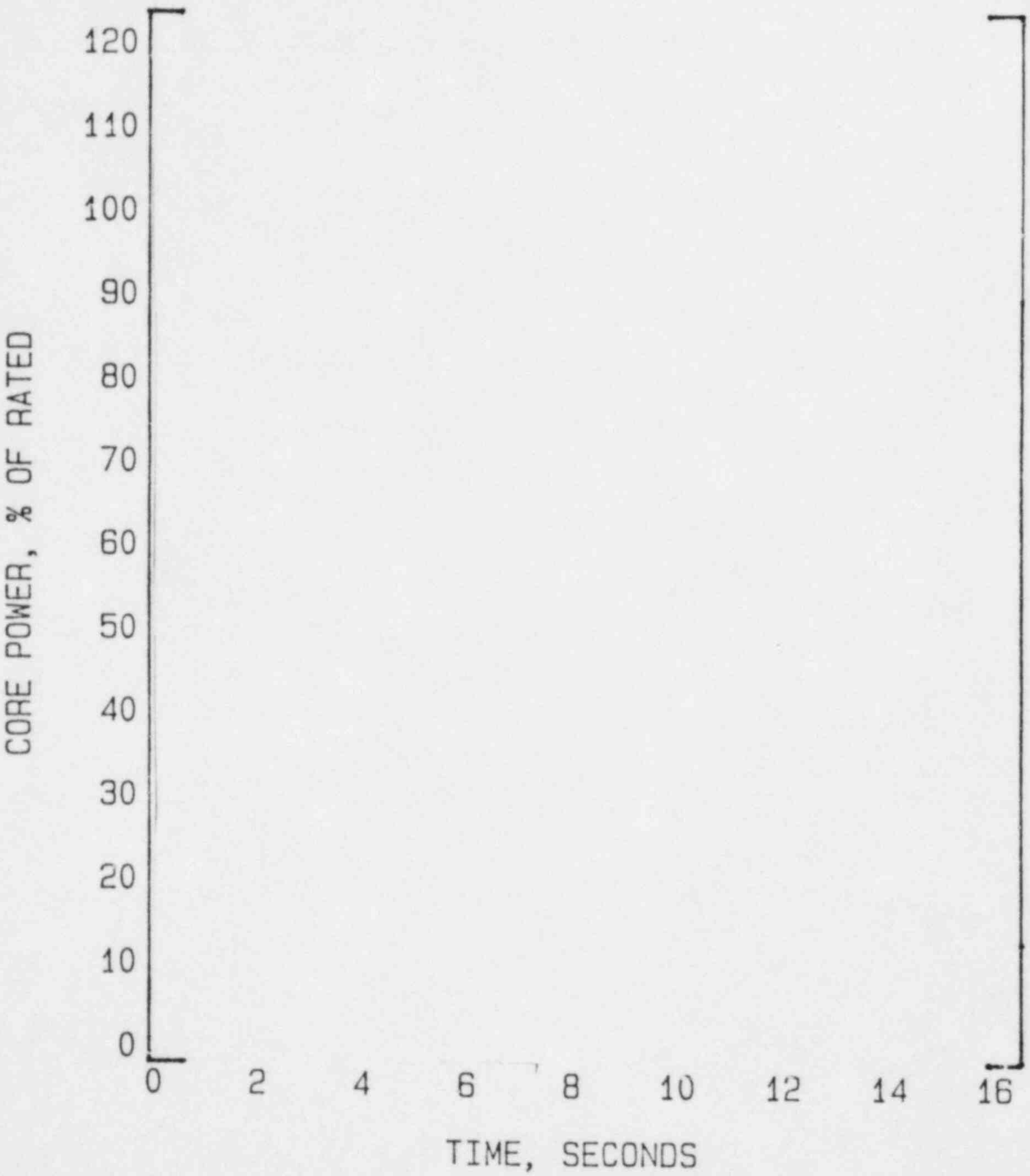
| Full Power Steam Line Break Incident<br>Steam Generator Pressure vs Time | Omaha Public Power District<br>Fort Calhoun Station-Unit No. 1 | Figure<br>5-6 |
|---|---|---|




NOTE :

EXAMPLE CASE: CE ANALYSIS FOR 2700 MWt UNIT
CYCLE 8: OPPD ANALYSIS

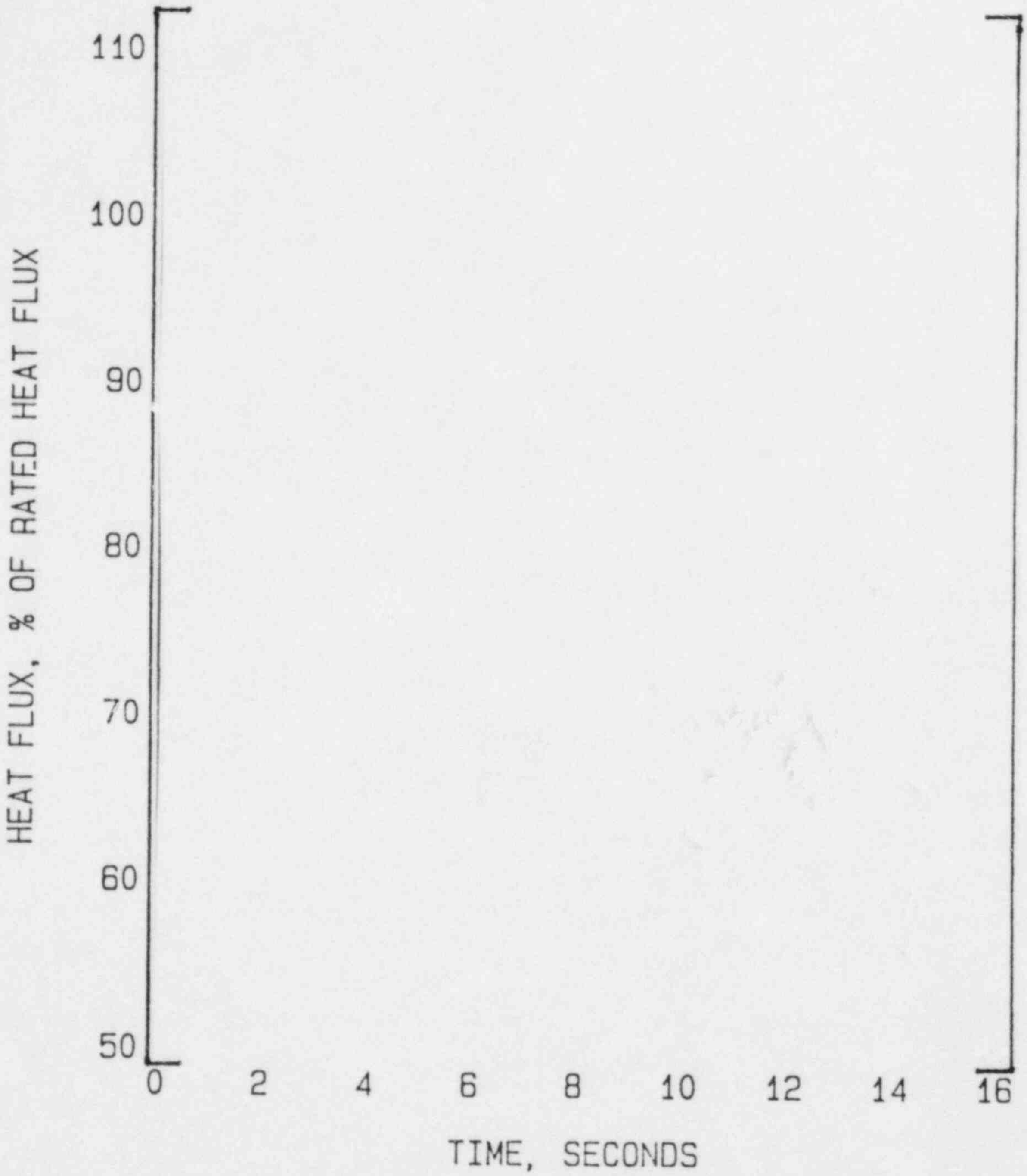
| | | |
|---|---|---|
| HCS Depressurization Event<br>RCS Pressure vs Time | Omana Public Power District<br>Fort Calhoun Station-Unit No. 1 | Figure<br>6-1 |




NOTE :

EXAMPLE CASE: CE ANALYSIS FOR 2700 MWt UNIT
CYCLE 8: OPPD ANALYSIS

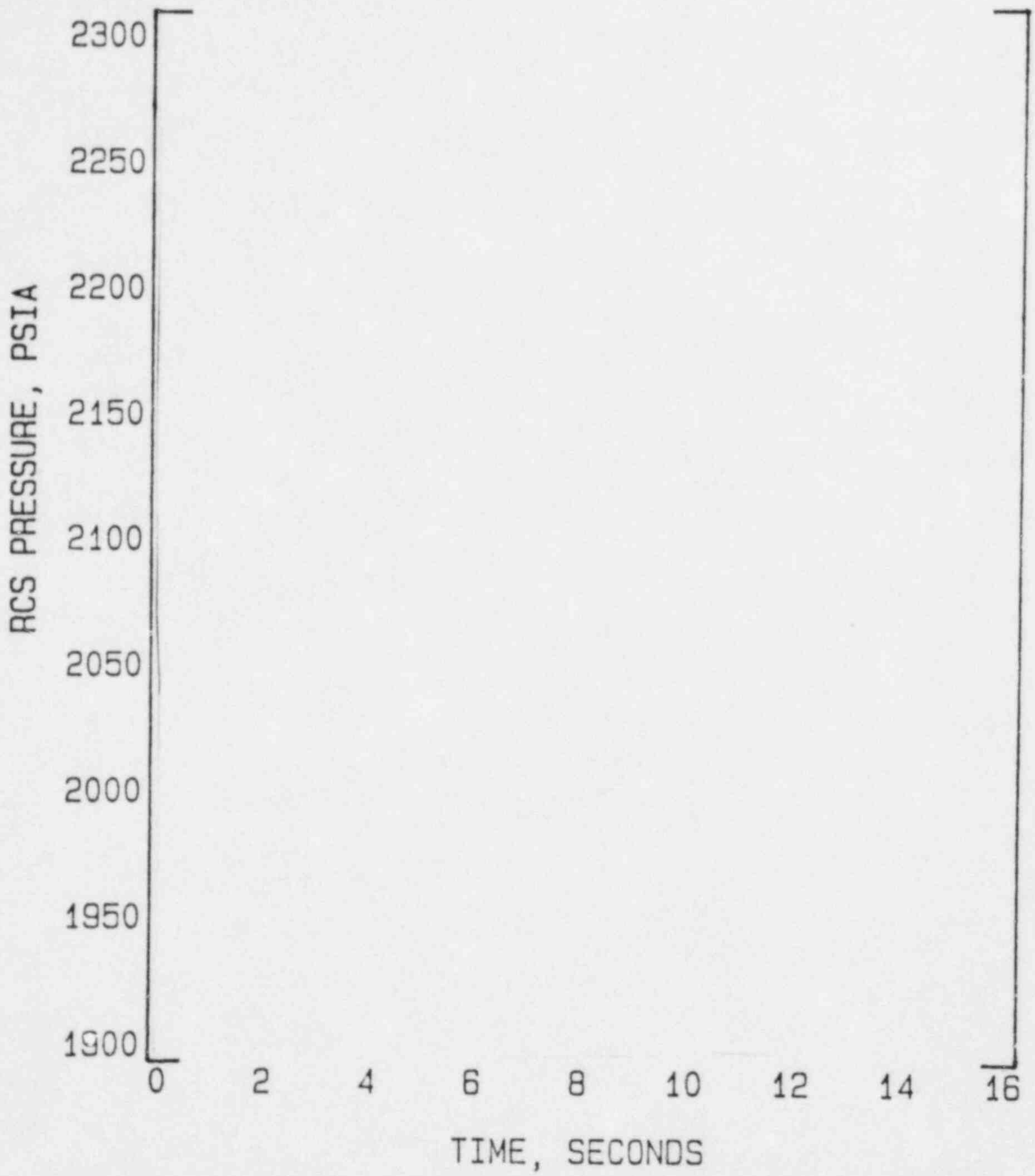
| | | |
|---|---|---|
| RCS Depressurization Incident<br>Core Power vs Time | Omaha Public Power District<br>Fort Calhoun Station-Unit No. 1 | Figure<br>6-2 |




NOTE :

EXAMPLE CASE: CE ANALYSIS FOR 2700 MWt UNIT
CYCLE 8: OPPD ANALYSIS

| RCS Depressurization Incident<br>Core Average Heat Flux vs Time | Omaha Public Power District<br>Fort Calhoun Station-Unit No. 1 | Figure<br>6-3 |
|---|---|---|




NOTE :

EXAMPLE CASE: CE ANALYSIS FOR 2700 MWt UNIT
CYCLE 8: OPPD ANALYSIS

| RCS Depressurization Incident<br>RCS Pressure vs Time | Omaha Public Power District<br>Fort Calhoun Station-Unit No. 1 | Figure<br>6-4 |
|---|---|---|

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