Omaha Public Power District
Nuclear Analysis
Reload Core Analysis Methodology

Transient and Accident Methods and Verification

OPPD-NA-8303-NP
Rev. 03

March 1991

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Table 2

TRANSIENT AND ACCIDENT METHODS AND VERIFICATION

OPPD-NA-8303-NP
Rev. 03

Page	Section	Change
All	All	Added reference to CENTS. Changed "The District" or "District" to "OPPD"
iii, iv	All	Renumbered pages to indicate correct page numbers of sections.
v	All	Renumbered pages to agree with tables in document. Changed title of Table 5.1-1.
vi, vii	All	Renumbered figures to be consistant with other OPPD methodology topicals. Renumbered pages to reflect actual figure page number.
viii		Added Revision 03 to list.
1	1.0	Added information on CENTS as computer code.
2	2.2.1	Revised to indicate present rather than future tense.
5		Changed LOCA analyst from CE to W.
6	3.0	Defined TM/LP as Thermal margin/low pressure trip
7	3.0	Deleted Excess Load event from TM/LP transient term. Added Excess Load to DNB LCO events.
7	4.0	Added reference to CENTS and indicated use of CENTS as plant simulation code.
8 - 10	4.1	Additional CENTS Description
15	5.1.3E	Added "of" between "coefficient " and "reactivity"
17	Table 5.1-1	Changed "Initial Conditions" to "Key Parameters"
20		Reclassified boron dilution event.
21	5.2.5	Added multiplier "m" to account for density changes. Changed "T_{BD}" to "Dilution time constant"

Table 2 (Continued)

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32	5.6.1	Added information on Excess Load as DNB margin event. Transferred margin requirement from LSSS to LCO.
33, 34	5.6.1	Added information on ΔT power calculator.
34,35	5.6.3	Changed objective from bias term to ROPM value. Added objective on validating VHPT.
	5.6.4	Added information on ROPM and reference to Reference 5-2. Added and changed key parameters to reflect calculation of ROPM.
35	5.6.5	Changed method to calculate ROPM term rather than bias term.
36	Table 5.6.4-1	Changed/added key parameters.
37	5.6.6	Changed 10 CFR 50.59 criteria to reflect ROPM results from bias term.
43	5.8.4	Changed "197,000" to "196,000" for LCO flow rate to be used.
50	5.10.2/.4	Revised analysis criteria for W methods.
51	5.11	Revised LOCA to indicate W methods.
59-72,77-78	6.0	Added information on CENTS verification and use. Revised figure numbers to be consistent with other OPPD methodologies.
79	Ref. 2-1	Added reference for W LOCA methods.
Ref.s 4-11,12 and 13		Added references for CENTS..
80	Ref. 5-5	Changed reference from CE to W for CEA Ejection methodologies.
	Ref. 5-7	Added reference for W LOCA methods.
	Ref. 5-8	Corrected typographical error.

Table 2 (Continued)

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80	Refs. 6-3,4	Added references to CENTS.
81	Figure 5-1	Added figure as referenced in Section 5.6.5
82-125	Figures	Added CENTS information and renumbered figures.

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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OMAHA PUBLIC POWER DISTRICT

RELOAD CORE ANALYSIS METHODOLOGY

TRANSIENT AND ACCIDENT METHODS AND VERIFICATION

REVISION	DATE
00	September 1983
01	November 1986
02	April 1988
03	March 1991

Omaha Public Power District
Reload Core Analysis Methodology
Transient and Accident Methods and Verification

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 OPPD's verification of the Combustion Engineering System Excursion Code (CESEC) and the Combustion Engineering Nuclear Transient Simulation code (CENTS) for Fort Calhoun Station transients. The purpose of this verification is to demonstrate OPPD's ability to properly utilize the CESEC and CENTS codes.

OPPD'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. OPPD's transient analysis models are discussed in Section 4.0. OPPD'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 OPPD 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 adversely affect 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

2.1 Criteria (Continued)

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 analyses 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 that 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 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 OPPD.

2.2.1 Malpositioning of Group N CEAs (formerly Part-Length CEAs)

This event is not analyzed in the reload core analysis because the Group N part-length CEAs were replaced with full length CEAs in Cycle 11. The use of Group N during power operations is prohibited by the Technical Specifications. The drop of a Group N CEA is considered in the full length CEA analysis.

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

2.2.2 Idle-Loop Startup Event

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

2.2.3 Turbine Generator Overspeed Event

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 Event

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

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.

Steam generator tube plugging performed during a refueling outage has the potential for altering the heat transfer characteristics assumed in Section 14.9.1 of the USAR. Section 5.12 of this document addresses the methodology to be employed should the steam generator tube plugging exceed or be expected to exceed the current USAR analysis assumptions.

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

2.2.4 Loss Of Load Event (Continued)

B. 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 in Section 5.6 in this document.

Steam generator tube plugging performed during a refueling outage has the potential for degrading the heat transfer characteristics assumed in Section 14.10.1 of the USAR for the Loss of Feedwater Flow Event. Section 5.13 addresses the methodology to be employed should steam generator tube plugging exceed or be expected to exceed the assumptions of the current Loss of Feedwater Flow Event. Reduced heat transfer for the Loss of Feedwater Heating Event does not require reanalysis, since it is an overcooling event and the increase in plugged tubes reduces the consequences of the event.

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

2.2.6 <u>Steam Generator Tube Rupture Accident</u>

The steam generator tube rupture accident is analyzed to determine if the offsite dose acceptance criteria of 10 CFR Part 100 is met. The analysis is a radioactive 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 accident analysis may be verified for high burnup fuel and/or a change in heat transfer characteristics for an increase in the number of plugged tubes in the generators.

2.2.7 <u>Loss of Coolant Accident</u>

The loss of coolant accident as reported in USAR, Section 14.15, is analyzed for OPPD by W. The large and small break analyses were performed by W using NRC approved Methods. A summary of the methods used by W for the large and small break LOCA analyses is provided in Reference 2-1. OPPD 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. OPPD does not perform any loss of coolant accident analyses.

2.2.8 <u>Containment Pressure Analysis</u>

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 <u>Generation of Hydrogen in Containment</u>

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.

2.2.10 <u>Fuel Handling Accident</u>

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

2.2.11 Gas Decay Tank Rupture

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

2.2.12 Waste Liquid Event

The waste liquid event analysis is not affected by refueling the core. Therefore, the waste liquid event 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 would exist for a reload core. The USAR Chapter 14 events considered in a reload core analysis are the Control Element Assembly Withdrawal (CEAW) event, the boron dilution event, the Control Element Assembly (CEA) drop event, the loss of coolant flow event, the excess load event, the steam line break accident, the CEA ejection accident, RCS depressurization event and the seized rotor accident. In addition, an analysis is performed for incidents resulting from the malfunction of one steam generator. 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 Thermal Margin/ Low Pressure (TM/LP) trip 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. The minimum required shutdown margin at hot shutdown conditions is determined by the steam line break accident. 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 analysis, the loss of four pump flow analysis, the excess load analysis or the CEA withdrawal analysis.

4.0 TRANSIENT AND ACCIDENT ANALYSIS MODELS

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

4.1 Plant Simulation Model

OPPD utilizes the CESEC and CENTS digital computer codes, References 4-1, 4-2, and 4-11, to provide the simulation of the Fort Calhoun Station nuclear steam supply system. Both codes calculate the plant response to non-LOCA initiating events for a wide range of operating conditions. Additional information on the CESEC model is provided in Reference 4-3. 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.

CESEC 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 be homogenous. Reference 4-1 provides a description of the

4.1 Plant Simulation Model (Continued)

CESEC code, including the major models, and the input, output and plot packages.

The pressurizer model is described in Reference 4-1 and further discussed in Reference 4-2. OPPD 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 4-1 and further discussed in Reference 4-2, 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 4-1 and 4-2. OPPD's CESEC model of Fort Calhoun Station is valid as indicated in Reference 4-3 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. OPPD accesses the code through a time sharing system. CE maintains all documentation and quality assurance programs related to this code.

The CENTS primary system model is based on the design version of the CEFLASH-4AS code (Ref. 4-12). The thermal hydraulic response is modeled by a node and flowpath network. Nodes enclose control volumes which represent fluid mass and energy. Flowpaths connecting nodes represent fluid momentum and have no volume. Nodes are provided to model primary system components such as the inner vessel, upper head, hot and cold legs, pressurizer, steam generator (separate nodes for hot and cold sides of tubes), reactor coolant pump suction legs, reactor vessel downcomer, and the control element assembly guide tubes. The secondary side is represented by three nodes for each steam generator (downcomer, evaporator, and steam dome) and one node for the main steam line header. The secondary system model also models the secondary safety valves, atmospheric and condenser dump valves, main steam isolation valves, turbine bypass and admission valves, and main and auxiliary feedwater.

The RCS thermal hydraulic model is formulated with five one dimensional conservation

equations. Conservation of mixture (liquid and steam) mass, liquid mass, mixture energy, steam energy, and mixture momentum are all considered. Mass and energy for liquid and steam are calculated for each node, while mass flowrate is calculated for each flowpath. Transient thermal hydraulic response is calculated by integrating the five conservation equations. Pressure in each node is calculated after solution of the conservation equations.

CENTS also models phase separation within a node into a separate steam region and a liquid or two-phase region consisting of a continuous liquid phase with dispersed bubbles. Nodes with phase separation provide a discrete two phase mixture level in the node. And where appropriate, the fluid level impacts on heat transfer rates and on the quality of fluid mixture exiting through flowpaths connected to the node.

CENTS provides a full range of thermodynamic fluid states for all primary nodes. Nodes with homogeneous or fully mixed fluid are at equilibrium. The possible states for non-homogeneous or phase-separated nodes with separate two phase mixture and steam regions are (a) saturated liquid with saturated steam (equilibrium), (b) subcooled liquid with saturated steam, (c) saturated liquid with superheated steam, and (d) subcooled liquid with superheated steam.

Core power is calculated by CENTS using either a point kinetics model or a three dimensional core neutronics model. In the point kinetics model, the calculated power is distributed axially according to a user input axial power shape. The three dimensional model calculates a detailed power distribution with local power levels and fuel operating conditions for each fuel assembly.

CENTS features a flexible, modular method to handle control systems for the core, primary system, secondary system, and the reactor protective system. This method has been used to model the reactor protective system and other control systems for Fort Calhoun Station.

The capabilities and limitations of the CENTS code are discussed in References 4-11 and 4-13. OPPD's CENTS 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 CENTS model is also valid for analysis of the loss of load, malfunctions of the feedwater system and the steam generator tube rupture incidents.

4.0 TRANSIENT AND ACCIDENT ANALYSIS MODELS (Continued)

4.1 Plant Simulation Model (Continued)

CENTS is maintained by OPPD on an OPPD workstation computer. 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 4-4, or both the TORC and CETOP codes, Reference 4-5. The TORC code is used as a benchmark for the CETOP 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 4-6 and 4-7) is used for the Fort Calhoun reactor as approved in Reference 4-8. 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 4-5 and a description of OPPD's application of the CETOP code is contained in Reference 4-9.

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 uncertainties 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 4-10. In this method the

4.3 Application of Uncertainties (Continued)

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 accommodated 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.

E. Analysis Method - A description of the methodology employed by OPPD 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.

Table 5.0-1

REACTOR PROTECTIVE SYSTEM TRIPS AND SAFETY INJECTION

Trip	Setpoint	Uncertainty	Used in Analysis Delay Time (sec)	Used in Analysis 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
Steam Generator Differential Pressure	135 psid	±40 psi	0.9	175 psid
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 5-3.

(2) Pump start – loop valve opening time.

5.0 TRANSIENT AND ACCIDENT ANALYSIS METHODS (Continued)

5.1 CEA Withdrawal Event

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 CEAs 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.

Withdrawal of the CEAs 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 CEAs 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

5.0 TRANSIENT AND ACCIDENT ANALYSIS METHODS (Continued)

5.1 CEA Withdrawal Event (Continued)

5.1.1 Definition of the Event (Continued)

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 CEAs 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 CEAs 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 22 kw/ft (Reference 5-1).

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:

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

B. Gap thermal conductivity (Hgap),

C. Initial power level,

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

E. The moderator temperature coefficient of reactivity, and

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

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

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.

5.0 TRANSIENT AND ACCIDENT ANALYSIS METHODS (Continued)

5.1 CEA Withdrawal Event (Continued)

5.1.4 Key Parameters and Analysis Assumptions (Continued)

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 pressurizer 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 5-2. OPPD does not perform all parametric analyses discussed in Reference 5-2 for Fort Calhoun Station. Rather, OPPD utilizes the analyses performed in Reference 5-2 to limit the number of analyses necessary for Fort Calhoun Station. Specifically, OPPD utilizes the result that [

] In addition, the result from Reference 5-2 that [
] when combined with [
] can be used to perform sensitivity analyses on the CEA withdrawal rate to achieve [
] is utilized.

5.0 TRANSIENT AND ACCIDENT ANALYSIS METHODS (Continued)

Table 5.1-1
Key Parameters Assumed in CEAW Event Analysis

Parameter	Units	Value
Initial Core Power	MWt	1 (HZP) /1530 (HFP)*
Initial Core Inlet Coolant	°F	532 (HZP)*
Temperature		Maximum allowed by Tech. Specs.
Moderator Temperature Coefficient	$x10^{-4} \Delta\rho/°F$	Tech. Spec. Range
Initial RCS Pressure	psia	Minimum allowed byTech. Specs.*
Fuel Temperature Coefficient	$x10^{-4} \Delta\rho/°F$	Least Negative Predicted During a Cycle
Initial Core Mass Velocity	$x10^{6}$ lbm/hr	Minimum allowed byTech. Specs.*
Fuel Temp. Coeff. Uncertainty	%	-15.0
Gap Thermal Conductivity	BTU/hr-ft^2	[ ]
CEA Differential Worth	$x10^{-4}$ /inch	[ ]
CEA Withdrawal Speed	in/min	46.0
Radial Peaks		Maximum Allowed byTech Spec. 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.

5.1.5 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 5-2 (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 5-2.

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 5-2.
The CEAW event analyzed to determine the closest approach to the fuel centerline melt SAFDL assumes those values of the CEAW rate and Hgap discussed in Reference 5-2. This combination of CEAW rate and Hgap 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 levels are expected to be similar to those presented in Reference 5-2. 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 overpower margin for these events is less than the available overpower margin required by the current Technical Specification DNB and PLHGR LCOs. 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 5-2 for the full power, intermediate power level and hot zero power cases.

5.2 Boron Dilution Event

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 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 22 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 shutdown margin requirement at an RCS temperature between 515 °F and 545 °F. The minimum volume of the reactor coolant system is conservatively assumed to be 5,506 cubic feet.

The boron dilution event between hot and cold shutdown (Modes 3 and 4) is assumed to be initiated from the Technical Specification shutdown margin requirement at an RCS temperature between 210 °F and 515 °F. The boron dilution event at cold shutdown (Mode 4) is initiated from the Technical Specification minimum shutdown margin requirement at an RCS temperature between 68 °F and 210 °F. The analysis is conducted for two RCS volumes, one of 5,506 cubic feet and the other of 2,036 cubic feet, which is the volume conservatively assumed to be 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 (2,036 cubic feet) and the minimum permissible boron concentration allowed by Technical Specification for refueling exists. All CEAs 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:

$$t_{crit} = m * \tau_{BD} * \ln((CBC + SDM * IBW) / CBC)$$

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

CBC = critical boron concentration (ppm)

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

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

m = multiplier to conservatively account for density effects due to the temperature difference between the makeup water and the reactor coolant water. The multiplier is the ratio of the specific volume of the makeup water to the specific volume of the reactor coolant water at the highest temperature in the range for the mode of operation of interest. Makeup water is assumed to be at room temperature (68 °F).

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 dilution time constant.

5.2.6 Analysis Results and 10 CFR 50.59 Criteria

The analysis results are similar to those reported in the Cycle 11 safety analysis report and in the 1987 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 are 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

5.0 TRANSIENT AND ACCIDENT ANALYSIS METHODS (Continued)

5.2 Boron Dilution Event (Continued)

5.2.7 Conservatism of Results (Continued)

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 CEAs 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 CEAs removed at a time. The analysis assumes that all CEAs are withdrawn from the core.

5.3 Control Element Assembly Drop Event

5.3.1 Definition of Event

The control element assembly (CEA) drop event 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 Control Element Assembly Drop Event (Continued)

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, and

B. The Peak Linear Heat Rate (PLHR) must be less than or equal to 22 kw/ft.

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:

A. The charging pumps and proportional heater systems are assumed to be inoperable during the transient. This maximizes the pressure drop during the event.

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

C. 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.

5.0 TRANSIENT AND ACCIDENT ANALYSIS METHODS (Continued)

5.3 Control Element Assembly Drop Event (Continued)

5.3.5 Analysis Method

The analysis methods utilized by OPPD to analyze the CEA drop event are discussed in Section 8 of Reference 5-3.

5.3.6 Analysis Results and 10 CFR 50.59 Criteria

Typical analysis results are contained in Section 8 of Reference 5-3 and in the 1987 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:

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

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

C. 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.

D. The manual mode of the pressurizer pressure and level control systems are assumed in the analysis. If the AUTO mode of operation is assumed, the RCS pressure would be maintained at a higher value, thereby lowering the DNBR margin required for this event.

Table 5.3.4-1

KEY PARAMETERS ASSUMED IN THE FULL LENGTH CEA DROP ANALYSIS

Parameter	Units	Value
Initial Core Power	MWt	1500*
Initial Core Inlet Temperature	°F	Maximum allowed* by Tech. Specs.
Initial RCS Pressure	psia	Minimum allowed* by Tech. Specs.
Initial Core Mass Flow Rate	$x10^6$lbm/hr	Minimum allowed* by Tech. Specs.
Moderator Temperature Coefficient	$x10^{-4}\Delta\rho/°F$	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	$\%\Delta\rho$	Minimum value predicted during core life for the CEA producing the maximum distortion factor
Core Average Hgap	BTU/hr-ft²-°F	Maximum value predicted during core life
Fuel Temperature Coefficient	$x10^{-4}\Delta\rho/°F$	Most negative value predicted during core life

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

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 failure 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 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 Four-Pump Loss of Flow Event (Continued)

5.4.5 Analysis Method

The analysis method used by OPPD to analyze the four-pump loss of coolant flow is discussed in Section 7 of Reference 5-3. OPPD 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 OPPD because of the high rotational energy of the pumps (N = 1185 rpm, I = 71,000 lb-ft²/pump). OPPD also utilizes the [

]

5.4.6 Analysis Results and 10 CFR 50.59 Criteria

Expected analysis results are presented in Section 7.1 of Reference 5-3. 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

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

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

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

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

E. The conservatism of the CETOP calculations is discussed in Section 7 of Reference 5-3.

Table 5.4.4-1

KEY PARAMETERS ASSUMED IN THE LOSS OF COOLANT FLOW ANALYSIS

Parameter	Units	Value
Initial Core Power	MWt	1500*
Initial Core Inlet Temperature	°F	Maximum allowed* By Tech. Specs.
Initial RCS Pressure	psia	Minimum allowed* by Tech. Specs.
Initial Core Mass Flow Rate	$x10^6$ lbm/hr	Minimum allowed* by Tech. Specs.
Moderator Temperature Coefficient	$x10^{-4}$ $\Delta\rho$/°F	Maximum allowed by Tech. Specs.
Fuel Temperature Coefficient	$x10^{-4}$ $\Delta\rho$/°F	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 Hgap	BTU/hr-ft^2-°F	Minimum predicted during core lifetime

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

5.0 TRANSIENT AND ACCIDENT ANALYSIS METHODS (Continued)

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:

A. Loss of load to one steam generator.

B. Loss of feedwater to one steam generator.

C. Excess feedwater to one steam generator.

D. 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 was 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 be maintained by the LCO. A system description of the ASGTPTF is presented in Appendix B of Reference 5-3.

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 22 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 LCOs such that in conjunction with the ASGTPTF the DNBR and PLHGR SAFDLs is not exceeded.

5.5.4 Key Parameters and Analysis Assumptions

Section 7 of Reference 5-3 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 5.5.4-1. The charging pumps and proportional heater systems are assumed to be inoperable during the transient. This maximizes the pressure drop during the event. The turbine 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 OPPD to analyze the LL/1SG is discussed in Section 7 of Reference 5-3.

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 5-3.

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.

Table 5.5.4-1
KEY PARAMETERS ASSUMED IN THE LL/1SG EVENT

Parameter	Units	Value
Initial Core Power	MWt	1530*
Initial Core Inlet Temperature	°F	Maximum allowed* by Tech. Specs.
Initial Reactor Coolant System Pressure	psia	Minimum allowed* by Tech. Specs.
Moderator Temperature Coefficient	$x10^{-4}\Delta\rho/°F$	Most negative allowed by Tech. Specs.
Fuel Temperature Coefficient	$x10^{-4}\Delta\rho/°F$	Most negative predicted during core life.
Core Average H_{gap}	$BTU/hr\text{-}ft^2\text{-}°F$	Maximum value predicted during core life.
Initial Core Mass Flow Rate	$x10^6 lbm/hr$	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.

5.0 TRANSIENT AND ACCIDENT ANALYSIS METHODS (Continued)

5.6 Excess Load Event

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 and heat flux. This results in a decrease in DNB margin and an increase in LHR. In Cycle 14 the excess load event was reclassified from a [ ] event to a ROPM event. It should be noted that margin requirements based on explicit AOO transient analyses are absolute quantities. The use of one method versus the other method does not result in a net gain in margin. It only transfers the margin requirement from the LSSS to the LCO or vice versa. There is no gain in both LSSS and LCO space.

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 accommodate steam flow for powers much in excess of 1500 MWt. The steam dump valves and steam bypass valves to the condenser are sized to accommodate 33% and 5%, respectively, of the steam flow at 1500 MW. Therefore, the following load increase events 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.

5.6 Excess Load Event (Continued)

5.6.1 Definition of Event (Continued)

C. Opening of the dump and bypass valves at hot standby conditions due to low reference temperature setting in the steam dump controller: When 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 17°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 event 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).
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. The TM/LP and VHP trips use the auctioneered (higher) value of measured power from either the ex-core neutron flux power detectors or the ΔT power calculator. The ex-core power detectors may become decalibrated due to the temperature shadowing. The ΔT power input responds slowly due to the relatively long time constants assumed for the Resistance

5.6.1 Definition of Event (Continued)

Temperature Detectors (RTDs) in the hot and cold legs of the reactor coolant system.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 internal limit using the CE-1 correlation.

B. The peak linear heat rate (PLHR) must be less than or equal to 22 kw/ft.

5.6.3 Objectives of the Analysis

An objective of the analysis is to calculate a ROPM value which is factored into the setpoint analysis to ensure that the DNBR and LHR SAFDLs are not exceeded for excess load events for which the TM/LP does not provide protection.

The other objective of the analysis for the limiting excess load event is to demonstrate that the Variable High Power Trip (VHPT) is initiated in time to insure the analysis criteria are met.

5.6.4 Key Parameters and Analysis Assumptions

Reference 5-2 discusses a similar sensitivity study performed by CE consistant with Reference 5-3 to demonstrate that the maximum calculated ROPM for the excess load event occurs for the [ ] at hot full power conditions. OPPD 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 5-3. The reactor state parameters of primary importance in calculating the margin degradation are:

A. Axial Power Distributions,

B. Initial Core Inlet Temperature,

5.6 Excess Load Event (Continued)

5.6.4 Key Parameters and Analysis Assumptions (Continued)

C. Initial power level,

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

E. The moderator temperature coefficient of reactivity, and

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

5.6.5 Analysis Method

The steps used for determining the [ ] value and calculating the largest ROPM for all excess load events which rely on the TM/LP trip for DNBR protection are given in Section 5 of Reference 5-3.

The method of providing DNB protection is to build DNB margin into the DNB LCOs. The excess load event is protected by the RPS and sufficient initial margin which is maintained by the LCOs. The DNB ROPM is calculated and compared to the DNB ROPMs of the other AOOs to determine the limiting ROPM which should be incorporated into the setpoint LCO calculations.

This method ensures that sufficient margin is maintained by the LCOs in conjunction with the VHPT. Hence, the DNB and LHR ROPMs will be calculated based relative to the VHPT setpoint. To determine a conservatively large margin degradation the initial conditions are set such that a VHPT setpoint is reached when both the ΔT power calculator and the power from the ex-core detectors are equal. This is accomplished by determining the [ ] value as described in Section 5 of Reference 5-3.

To futher ensure that the most limiting case has been analyzed, the following key assumptions are made Since the ex-core detectors may become decalibrated due to the [ ], the [ ] factor is used to attenuate the ex-core detector response during the cooldown event. This results in a [ ] trip and [ ] ROPMs.

Figure 5-1 provides a block diagram of the methods used to calculate the DNB and LHR ROPMs for the excess load event.

Table 5.6.4-1

KEY PARAMETERS ASSUMED IN THE EXCESS LOAD EVENT ANALYSIS

Parameter	Units	Value
Initial Core Power	MWt	1530*
Initial Core Inlet Temperature	°F	Maximum allowed* by Tech. Specs.
Initial Reactor Coolant System Pressure	psia	Minimum allowed* by Tech. Specs.
Initial Core Mass Flow Rate	$x10^6$ lbm/hr	Minimum allowed* by Tech. Specs.
Axial Shape Index	asiu	Most Negative allowed by DNB LCO Tent
RTD Delay Time	sec	Minimum Hot Leg Maximum Cold Leg
Moderator Temperature Coefficient	$x10^{-4} \Delta\rho/°F$	Negative values up to the most negative value allowed by Tech.Specs.
Radial Peaks		Maximum Allowed byTech Spec. 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 Setpoint	% Above Initial Power Level	10.0
Temperature Shadowing Factor	% Power/°F	[ ]

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

5.6.5 Analysis Method (Continued)

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 Reference 5-2. The criteria of 10 CFR 50.59 are met if the ROPM calculated for this event is less than or equal to the overpower margin being maintained by the current Technical Specifications.

5.6.7 Conservatism of Results

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

A. The VHPT is not assumed until the right hand ΔT Power calculator and the left hand ΔT Power calculator values are both above the setpoint.

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

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

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

E. The scram reactivity curve associated with the strong positive axial power distribution is conservative with respect to the actual power distributions observed in the reactor.

F. The pressurizer level control is assumed to be in the manual mode (pressurizer heaters off). This maximizes the RCS pressure decrease which maximizes the ROPM for the event.

5.0 TRANSIENT AND ACCIDENT ANALYSIS METHODS (Continued)

5.7 RCS Depressurization Event

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.

5.7.4 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 Reference 5-3.

Table 5.7.4-1

KEY PARAMETERS ASSUMED IN THE RCS DEPRESSURIZATION EVENT ANALYSIS

Parameter	Units	Value
Initial Core Power	MWt	1530*
Initial Core Inlet Temperature	°F	Maximum allowed by* Tech. Specs.
Initial Reactor Coolant System Pressure	psia	Upper limit of normal operating range
Moderator Temperature Coefficient	$x10^{-4}\Delta\rho/°F$	Most negative allowed by Tech. Specs.
Fuel Temperature Coefficient	$x10^{-4}\Delta\rho/°F$	Most negative predicted during core life.
Core Average Hgap	BTU/hr-ft²-°F	Minimum predicted during core life.
Total Trip Delay Time	sec	1.4

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

5.7 RCS Depressurization Event (Continued)

5.7.5 Analysis Method

The methods used by OPPD to analyze the RCS depressurization event are contained in Section 5 of Reference 5-3.

5.7.6 Analysis Results and 10 CFR 50.59

Results of the RCS depressurization transient are discussed in Reference 5-3 and in the 1984 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:

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

B. Conservatively slow RPS response times are used.

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

D. 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).

5.8.1 Definition of the Event (Continued)

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).

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%) 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 temperature (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, b, assumed is the maximum absolute value including uncertainties for 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 CEAs insert (no stuck CEAs), 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 incorporate the Trip 2/Leave 2 RCP operating strategy as indicated in Reference 5-8. For a steam line break,

the Trip 2/Leave 2 strategy will result in tripping two RCPs (at 1350 psia). If the event was misdiagnosed as a LOCA, all four RCPs would be tripped. As discussed below, for a main steam line break the consequences of Trip 2/Leave 2 is bounded by the loss of offsite power and the loss of offsite power case is bounded by tripping no RCPs. Consequently, the limiting main steam line break accident occurs with all RCPs operating.

The MSLB case with the RCPs tripped is similar to the MSLB case with a loss of offsite power since the RCPs 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 generators to restore power to the safety injection pumps and causes a coastdown of the RCPs.

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 RCPs results in a later coastdown of the RCPs 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 RCPs tripped than for a MSLB with a loss of offsite power. Therefore, the reactivity effects of a MSLB with the RCPs 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 RCPs operating. Because the reactivity effects of a MSLB with the RCPs 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 RCPs tripped utilizing Trip 2/Leave 2 at 1350 psia are less severe than for a MSLB with offsite power available and RCPs operating. The reactor coolant volumetric flow rate is assumed to be constant during the incident. The LCO flow rate (196,000 gpm) was used in order to obtain the most adverse results.

5.0 TRANSIENT AND ACCIDENT ANALYSIS METHODS (Continued)

5.8 Main Steam Line Break Accident (Continued)

5.8.4 Key Parameter and Analysis Assumptions (Continued)

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 fullload 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 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.

5.8.4 Key Parameter and Analysis Assumptions (Continued)

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 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 OPPD, 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.

5.8.5 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 Main Steam Line Break Accident (Continued)

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, by assuming all RCPs operate instead of manually tripping two pumps and by taking no credit for the stuck CEA worth.

5.9 Seized Rotor Accident

5.9.1 Definition of Event

The seized rotor accident 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 seizure 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, [

].

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 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

Table 5.9.4-1
KEY PARAMETERS ASSUMED IN THE SEIZED ROTOR ANALYSIS

Parameter	Units	Value
Initial Core Power	MWt	1530*
Initial Core Inlet Temperature	°F	Maximum allowed* by Tech. Specs.
Initial Reactor Coolant System Pressure	psia	Minimum allowed* by Tech. Specs.
Moderator Temperature Coefficient	$x10^{-4} \Delta\rho/°F$	Most negative allowed by Tech. Specs.
Fuel Temperature Coefficient	$x10^{-4} \Delta\rho/°F$	Most negative predicted during core life.
Core Average H_{gap}	BTU/hr-ft²-°F	Maximum value predicted during core life.
Initial Core Mass Flow Rate	$x10^6$ lbm/hr	Best estimate flow*
CEA Drop Time	sec	Maximum value allowed by Tech. Specs..
Scram Reactivity Worth	$\%\Delta\rho$	Minimum predicted during core life.

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

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 Section 14.6.2 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:

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

B. 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.

C. 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 CEAs activated by the high power trip or variable high power trip.

5.0 TRANSIENT AND ACCIDENT ANALYSIS METHODS (Continued)

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:

A. Fuel cladding and enthalpy thresholds (Reference 5-5) are:

Clad Damage Threshold
Average Pellet Enthalpy (at hot spot) $\leq$ 200 cal/gram

Centerline Melting Threshold
Total Centerline Enthalpy $\leq$ 250 cal/gram

Fully Molten Centerline Threshold Total Centerline Enthalpy $\leq$ 310 cal/gram

B. 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. This objection is achieved if the peak RCS pressure does not exceed 2750 psia.

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

5.10.3 Objectives of the Analysis

The objective of the analysis is to demonstrate that fuel failures are less than those reported in Section 14.3.4.1 of the Fort Calhoun Station Unit No. 1 USAR of that site boundary doses are within the 10 CFR 100 limits.

5.10.4 Analysis Method

OPPD utilizes the CEA Ejection Accident Analysis of our current fuel vendor, Westinghouse. This analysis methodology is documented in Reference 5-5 and is performed by Westinghouse. This methodology utilizes physics parameters, calculated by OPPD in accordance with the methods outlined in Reference 5-6.

5.0 TRANSIENT AND ACCIDENT ANALYSIS METHODS (Continued)

5.10 CEA Ejection Accident (Continued)

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 Station Unit No. 1 USAR. Criteria of 10 CFR 50.59 are satisfied if fuel failures are less than those assumed for input to the Radiological Consequences portion of the analysis.

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 roll to the core periphery which also happens to be the location of the ejected CEA. Also, the ejected worth is calculated assuming the CEAs are fully inserted for hot full power case regardless of PDIL. Thus, the ejected worth is conservative.

5.11 Loss of Coolant Accident

OPPD does not perform the Loss of Coolant Accident Analysis. The large and small break loss of coolant analyses were performed by Westinghouse. The large break and small break topicals are mentioned in Reference 5-7. OPPD verifies that the physics input assumptions and the maximum rod burnup are within the bounds assumed in the W large break analysis.

5.12 Loss of Load to Both Steam Generators Event

5.12.1 Definition of the Event

A total loss of load to both steam generators usually results from a turbine trip due to a loss of external electrical load or to abnormal variations in electrical network frequencies. Other possible causes include the simultaneous closure of all turbine stop valves or main steam isolation valves. All initiating mechanisms result in a corresponding reduction in heat removal from the reactor coolant system due to the loss of secondary steam flow. Although a Reactor Protective System trip signal would normally result from a turbine trip, no credit is taken in the analysis of this event for the turbine trip signal.

5.12.2 Analysis Criteria

The loss of load to both steam generators event is classified as an Anticipated Operational Occurrence (AOO) for which the following criteria must be met:

A. The peak RCS pressure does not exceed 2750 psia (110% of design pressure).

B. The transient minimum DNBR is greater than the 95/95 confidence interval limit for the CE-1 correction limit.

C. The Peak Linear Heat Generation Rate (PLHGR) does not exceed 22 kw/ft.

Criteria B. and C. are not of major concern since DNBR increases during the event and the PLHGR margin required is much less limiting than other AOOs. Therefore, criterion A. is the main concern in analyzing this event. The loss of load to both steam generators event is the limiting AOO event with respect to peak RCS pressure.

5.12.3 Objectives of the Analysis

The objective of the analysis is to demonstrate, for modifications to the plant which potentially degrade RCS heat removal capability (including steam generator plugging) that the peak RCS pressure stays within 110% of the design pressure in accordance with Section III of the ASME Pressure Vessel Code. This objective is achieved if the peak RCS pressure does not exceed 2750 psia.

5.12.4 Key Parameters and Analysis Assumptions

The key parameters used in the loss of load (to both steam generators) event are given in Table 5.12.4-1. Assumptions used in the analysis to maximize heat up of the RCS and consequently peak RCS pressure include:

A. The event is initiated by a sudden closure of the turbine stop valves without a simultaneous reactor trip.

B. No credit is taken for operation of the PORVs, pressurizer sprays, and the turbine steam dump and bypass system, i.e., the pressurizer pressure control system is assumed to be in MANUAL.

Table 5.12.4-1

KEY PARAMETERS ASSUMED IN THE LOSS OF LOAD TO BOTH STEAM GENERATORS ANALYSIS

Parameter	Units	Value
Initial Core Power	MWt	1530
Initial Core Inlet Temperature	°F	Maximum allowed by Tech. Specs.
Initial RCS Pressure	psia	Minimum allowed by Tech. Specs
Initial Steam Generator Pressure	psia	Minimum value corresponding to core inlet temperature operating range.
Initial Core Mass Flow Rate	$x10^6$ lbm/hr	Minimum allowed by Tech. Specs.
Moderator Temperature Coefficient	$x10^{-4} \Delta\rho/°F$	Most positive allowed by Tech. Specs.
Fuel Temperature Coefficient	$x10^{-4} \Delta\rho/°F$	Least negative predicted during core life.
Fuel Temperature Coefficient Multiplier		0.85
CEA Drop Time	sec	Maximum allowed by Tech. Specs.
Scram Reactivity Worth	$\%\Delta\rho$	Minimum predicted during core lifetime
Scram Reactivity Curve		Consistent with most positive axial shape (bottom peaked) allowed by Tech. Specs.
Core Average Hgap	BTU/hr-ft^2-°F	Maximum predicted during core lifetime.
Kinetics Parameters		EOC parameters (minimum absolute β).
RPS Response Time	sec	1.4

5.0 TRANSIENT AND ACCIDENT ANALYSIS METHODS (Continued)

5.12 Loss of Load to Both Steam Generators Event (Continued)

5.12.4 Key Parameters and Analysis Assumptions (Continued)

C. The rod block system is assumed to prevent rod motion (other than scram) during the transient.

D. Maximum charging flow and zero letdown flow are assumed.

E. Termination of the event occurs as a result of a high pressurizer pressure trip.

5.12.5 Analysis Method

The analysis methods utilized by OPPD to analyze the loss of load to both steam generators event consists of simulating the event using the CESEC computer code, utilizing the analysis assumptions listed in Section 5.12.4 (above) as input, and extracting the peak RCS pressure for comparison with the 2750 psia upper limit.

5.12.6 Analysis Results and 10 CFR 50.59 Criteria

The results of the loss of load to both steam generators are contained in the Fort Calhoun Station Unit No. 1 USAR. The criteria of 10 CFR 50.59 are met, if the peak RCS pressure is less than 110% of design pressure in accordance with Section III of the ASME Pressure Vessel Code. This objective is achieved if the RCS pressure does not exceed 2750 psia.

5.12.7 Conservatism of Results

The following areas of conservatism are included in the analysis to obtain a conservatively high peak RCS pressure:

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

B. The actual scram worths are greater than those assumed in the analysis.

C. The actual MTC is more negative during power operation than assumed in the analysis.

D. The steam dump and bypass system and the pressurizer pressure control system (PORVs and sprays) are operated in the AUTO mode rather than MANUAL as assumed in the analysis.

5.12.7 Conservatism of Results (Continued)

E. Actual secondary pressure is higher which results in earlier secondary safety valve opening and earlier alleviation of the primary system temperature and pressure rises.

F. The maximum pressurizer safety valve capacities are assumed to be 90% of the ASME rated values.

G. A one percent pressure uncertainty is applied to the primary and secondary safety valve setpoints, i.e., a 1.01 multiplier.

5.13 Loss of Feedwater Flow Event

5.13.1 Definition of Event

A total loss of main feedwater flow event is defined as a loss of feedwater flow when operating at power without a corresponding reduction in steam flow from the steam generators. The most likely causes for this event are the loss of all feedwater or condensate pumps or the inadvertent closure of either the main feedwater regulating valves or the feedwater isolation valves due to a feedwater controller malfunction or manual positioning by the operator. The result of this mismatch in which turbine demand remains at 100%, is a reduction of the steam generator liquid inventories and a degrading RCS heat removal capability. As the heat removal capability is lost, through decreasing steam generator inventories (i.e., levels) the RCS temperatures and pressure increase. Normally the event would be terminated by a reactor trip on low steam generator level. Since no credit is taken in the analysis for the steam generator low level trip, a high pressurizer trip eventually results.

Automatic actuation of the auxiliary feedwater (AFW) system will also eventually occur (after reactor trip) if either main feedwater is not restored or manual actuation of the AFW system is not performed by the operator. The AFW system actuation ensures the maintenance of a secondary heat sink.

5.13.2 Analysis Criteria

The loss of feedwater flow event is classified as an Anticipated Operational Occurrence (AOO) for which the following criteria must be met:

5.0 TRANSIENT AND ACCIDENT ANALYSIS METHODS (Continued)

5.13 Loss of Feedwater Flow Event (Continued)

5.13.2 Analysis Criteria (Continued)

A. The peak RCS pressure does not exceed 2750 psia (110% of design pressure).

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

C. The Peak Linear Heat Generation Rate (PLHGR) does not exceed 22 kw/ft.

Criteria B. and C. are not of major concern because DNBR does not decrease below the initial steady state value and the PLHGR margin required is much less limiting than other AOOs. Therefore, only criterion A. requires reevaluation should plant modifications (such as steam generator tube plugging) be made which result in degraded secondary heat transfer capability beyond that of this event. For Fort Calhoun Station, this event is bounded by the loss of load incident.

5.13.3 Objectives of the Analysis

The objective of this analysis is to demonstrate, for plant modifications which potentially degrade RCS heat removal capability (including steam generator tube plugging), that the peak RCS pressure stays within 110% of the design pressure in accordance with Section III of the ASME Pressure Vessel Code. This objective is achieved if the peak RCS pressure does not exceed 2750 psia.

5.13.4 Key Parameters and Analysis Assumptions

The key parameters used in the loss of feedwater flow event are given in Table 5.13.4-1. Assumptions in the analysis to maximize heat up of the RCS and consequently the peak RCS pressure include:

A. The event is initiated by an instantaneous loss of main feedwater. No credit is taken for the low steam generator level trip.

B. The steam dump and bypass system is assumed to be in MANUAL (i.e., inoperative).

C. The pressurizer pressure control system is in MANUAL (i.e., PORVs and sprays are inoperable).

5.13.4 Key Parameters and Analysis Assumptions (Continued)

D. The pressurizer level control system is in MANUAL with maximum charging and zero letdown flows.

E. The rod block system is assumed to prevent rod motion (other than scram) during the transient.

5.13.5 Analysis Method

The analysis methods used by OPPD to analyze a loss of main feedwater flow event consists of using the CESEC computer code to simulate the event, utilizing the analysis assumption listed in Section 5.13.4 (above) as input, and extracting the peak RCS pressure for comparison with the 2750 psia upper limit.

5.13.6 Analysis Results and 10 CFR 50.59 Criteria

The results of the loss of feedwater flow are contained in the Fort Calhoun Station Unit No. 1 USAR. The criteria of 10 CFR 50.59 are met, if the peak RCS pressure is less than 110% of the design pressure in accordance with Section III of the ASME Pressure Vessel Code. This objective is achieved if the peak RCS pressure does not exceed 2750 psia.

5.13.7 Conservatisms of Results

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

B. The actual scram worths are greater than those assumed in the analysis.

C. The actual MTC is more negative during power operation than assumed in the analysis.

D. The steam dump and bypass system and the pressurizer pressure control system (PORVs and sprays) are operated in the AUTO mode rather than the MANUAL mode as assumed in the analysis.

E. Actual secondary pressure is higher which results in earlier secondary safety valve opening and earlier alleviation of the primary system temperature and pressure rises.

F. No credit is taken for a steam generator low level trip.

Table 5.13.4-1

KEY PARAMETERS ASSUMED IN THE LOSS OF FEEDWATER FLOW ANALYSIS

Parameter	Units	Value
Initial Core Power	MWt	1530
Initial Core Inlet Temperature	°F	Maximum allowed by Tech. Specs.
Initial RCS Pressure	psia	Minimum allowed by Tech. Specs
Initial Steam Generator Pressure	psia	Minimum value corresponding to core inlet temperature operating range.
Initial Core Mass Flow Rate	$x10^6$ lbm/hr	Minimum allowed by Tech. Specs.
Moderator Temperature Coefficient	$x10^{-4} \Delta\rho/°F$	Most positive allowed by Tech. Specs.
Fuel Temperature Coefficient	$x10^{-4} \Delta\rho/°F$	Least negative predicted during core life.
Fuel Temperature Coefficient Multiplier		0.85
CEA Drop Time	sec	Maximum allowed by Tech. Specs.
Scram Reactivity Worth	$\% \Delta\rho$	Minimum predicted during core lifetime
Scram Reactivity Curve		Consistent with most positive axial shape (bottom peaked) allowed by Tech. Specs.
Core Average Hgap	BTU/hr-ft^2-°F	Maximum predicted during core lifetime
Kinetics Parameters		EOC parameters (minimum absolute β).
RPS Response Time	sec	1.4

6.0 TRANSIENT ANALYSIS CODE VERIFICATION

6.1 Introduction

OPPD currently uses the CESEC-III computer code to calculate the transient response of the NSSS during events discussed in this document. OPPD has also benchmarked and justified the use of the alternate transient thermal hydraulic code CENTS. The CENTS code has advantages over CESEC-III in both modelling and flexibility of use. CENTS is a state-of-the-art code that can be used for best estimate, as well as, safety analyses. This allows CENTS to be used for evaluating safety and operability issues, as well as for reload safety analyses. Another incentive for replacing CESEC-III is that CENTS can model the as-built facility at Fort Calhoun more accurately than CESEC-III. An example of this increased ability is shown in the main steam safety valve models. CESEC-III requires a constant cross sectional flow area for the valve discharge, which means that the power operated safety valves can not be modelled as they have a smaller cross sectional area than the other main steam safeties. CENTS is more flexible and all five of the main steam safety valves can be accurately modelled. Combustion Engineering (CE) also plans to replace CESEC-III with CENTS for performing reload transient analyses. Once CENTS has been licensed, CE will provide only limited support for CESEC-III.

The verification of both CESEC-III and CENTS are documented in this chapter since CESEC-III is currently licensed as a safety analysis code, and will continue to be used for licensing purposes until CENTS is licensed for safety related applications.

Combustion Engineering has provided overall verification of the CESEC-III code in References 6-1 and 6-2, and CENTS in References 6-3 and 6-4. The purpose of the work documented in this chapter is to demonstrate OPPD's ability to correctly utilize both the CESEC-III and CENTS codes.

In order to demonstrate Omaha Public Power District's ability to correctly use both computer codes, verification work has been performed by benchmarking against 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 using CESEC-III were the Dropped CEA, Main Steamline Break, and RCS Depressurization events. Those transients benchmarked using CENTS were the Dropped CEA and RCS Depressurization events. These are two of the events that are typically analyzed on a cycle specific basis. The Main Steamline Break was not benchmarked using CENTS since the three-dimensional core neutronics model for

6.1 Introduction (Continued)

CENTS was not available during the benchmark effort. Since the MSLB event is not analyzed each cycle, CENTS will not be used in a licensing application for MSLB until a future benchmark is performed. Comparisons for the Dropped CEA and RCS Depressurization events are 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 for both the CESEC-III and CENTS codes 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 always recorded. Strip chart recordings on an extremely compressed time scale are generally the only form of data available. This compressed time scale (with graduations typically of 10 minutes) does not permit adequate comparisons to CESEC-III or CENTS modeling in which seconds are of major concern. Cycle 1 startup testing is the only source of plant transient data in which system parameters were measured with high speed strip chart recorders and no operator action taken. Good data exists for a nominal full power turbine-reactor trip and a 35% power total loss of RCS flow event.

The CESEC-III and CENTS computer codes were both 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 comparisons were obtained from vendor

6.2 Comparison to Plant Data (Continued)

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 6-1 through 6-9 show plots of the comparisons between the measured plant responses and the predicted CESEC-III and CENTS responses. It should be noted that this test was performed based on a rated power level of 1420 MWt rather than the current limit of 1500 MWt (the design power extension for which licensing was obtained in Cycle 6).

Figure 6-1 shows the nuclear power response following the turbine-reactor trip. The CESEC-III and CENTS predictions follow the same power decay rate. However, the endpoint residual power for both code predictions is slightly higher, which is 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. Trip delays can also be added to CENTS, but were not for this case since it was intended as a best estimate analysis. The pressurizer pressure response predicted by both codes are shown in Figure 6-2. The figure shows very good agreement of the codes' predictions 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 6-1. The CENTS case was initiated about 12 psia below the plant data and followed the plant response closely after the first 10 seconds. This difference between pressurizer pressure predicted by CESEC-III and measured pressurizer pressures at 60 seconds is only 19 psia, a value which is less than the pressure measurement uncertainty. The difference between the CENTS prediction and the actual pressurizer pressure at 60 seconds is less than 3 psia. Figures 6-3 and 6-4 show the RCS cold-leg and hot-leg temperature responses, respectively, for each steam generator loop for the plant data and the CESEC-III and CENTS predicted average

6.2 Comparison to Plant Data (Continued)

cold leg and hot leg temperatures. The differences in the transient response of the two steam generator loops for the plant data are attributable to the differences in the main feedwater flow rate rampdown after trip (see Figure 6-7). Both the CESEC-III and CENTS predictions lead the actual loop measurements because of the measurement delays associated with the response time of the resistance temperature devices (RTDs) providing the temperature signals. Figures 6-5 and 6-6 show the measured and predicted steam generator pressure responses for each steam generator. These plots show good agreement of the predictions from CENTS and CESEC-III with the Cycle 1 test data with only minor differences. The pressure predicted by both CESEC-III and CENTS is higher than the measured pressure early in the event due to a combination of the greater heat residual as shown in Figure 6-1, a quicker turbine stop valve closure, and quicker steam dump-bypass operation assumed in the CESEC-III and CENTS analyses. Figure 6-7 shows the feedwater flow during and after feedwater rampdown. The feedwater rampdown function is an input to both CESEC-III and CENTS. Figures 6-8 and 6-9 show the steam flow from both steam generators during the event. The CENTS predictions follow the measured steam flow closely. The discrepancies between the CESEC-III and CENTS predictions and the measured data are due mainly to a quicker turbine stop valve closure and quicker steam dump and bypass valve operation assumed in the CESEC-III and CENTS analyses.

In conclusion, both the CESEC-III and CENTS 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 simultaneously 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

6.2 Comparison to Plant Data (Continued)

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 6-10 through 6-17. These comparisons show excellent agreement. The minor differences that exist are discussed below. Currently, CENTS does not have the capability of being initialized at powers as low as 35% which would have been necessary to simulate this event. This feature is being added to the code and will be available in the next release. However, the RCS flow coastdown after all four pumps are tripped is relatively independent of the power level. Therefore, a CENTS case was run to predict the RCS flow coastdown. The predicted flow is plotted in Figure 6-10.

Figure 6-10 shows a plot of the measured total RCS flow versus time and that predicted by the CESEC-III and CENTS codes. Both codes incorporate explicit modeling of the reactor coolant pumps. This data shows excellent agreement with the predicted flow while being slightly conservative.

Figures 6-11 and 6-12 show the pressurizer pressure and level response comparisons which also show excellent agreement. Figure 6-13 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 6-14, also show very good agreement. The response of the hot-leg and cold-leg temperatures, as shown in Figure 6-15, is consistent with the data obtained from the turbine-reactor trip case. Again the delay associated with the RTD response causes the predicted temperatures to lead those that were measured. Figure 6-16 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

6.2 Comparison to Plant Data (Continued)

parameters examined, and that following a several second reduction in flow the previous flow rate was reestablished. Figure 6-17 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 6-16) are consistent.

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

6.3 Comparisons Between OPPD Analyses and Independent Analyses Previously Performed by the Fuel Vendors for Verification of CESEC-III

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 Advanced Nuclear Fuels, formerly Exxon Nuclear Company (ENC), or Combustion Engineering (CE) was done. Since Exxon Nuclear Company performed some analyses in this section (used for comparison) prior to becoming ANF, all references to this company will be to ENC. For the comparison cases, the assumptions used in the analyses were similar to those used by OPPD, 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 SAFDLs. 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.

6.3 Comparisons Between OPPD Analyses and Independent Analyses Previously Performed by the Fuel Vendors for Verification of CESEC-III (Continued)

(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 6.3-1 summarizes the parameters and their values for Cycles 6 and 8. Plots of core power versus time for the OPPD (Cycle 8) and ENC (Cycle 6) analyses are found in Figure 6-18. 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 6-19. Both are very similar, as was the case in the core power cases. Figure 6-20 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 6-21 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 shutdown margin, was analyzed by OPPD for

6.3 Comparisons Between OPPD Analyses and Independent Analyses Previously Performed by the Fuel Vendors for Verification of CESEC-III (Continued)

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 6.3-2 shows comparisons of the pertinent input values for each of the analyses.

Figure 6-22 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 6-23 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 6-24 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.

Figure 6-25 shows plots of RCS pressure versus time for Cycle 8 (OPPD) and Cycle 6 AFW (CE). Also included in Figure 6-25 is the Cycle 1 (CE) results. All three of these curves show excellent agreement. The Cycle 6 AFW (CE) analysis shows a lower end point 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 6-26 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

6.3 Comparisons Between OPPD Analyses and Independent Analyses Previously Performed by the Fuel Vendors for Verification of CESEC-III (Continued)

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 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 6.3-3 shows a comparison of the important input parameters for each of the analyses.

Figures 6-27, 6-28, and 6-29 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 6-30 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 6-31 and 6-32 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 ruptured steam

6.3 <u>Comparisons Between OPPD Analyses and Independent Analyses Previously Performed by the Fuel Vendors for Verification of CESEC-III</u> (Continued)

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 6-33 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 <u>RCS Depressurization</u>

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 6.3-4.

Figure 6-34 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.

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 6-35 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 Comparisons Between OPPD Analyses and Independent Analyses Previously Performed by the Fuel Vendors for Verification of CESEC-III (Continued)

6-36. The RCS pressure versus time plots are shown in Figure 6-37. 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 MWt 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 Comparisons Between OPPD Analyses and Independent Analyses Previously Performed with CESEC-III for Verification of CENTS

Of the transients analyzed by OPPD for reload core licensing (using CE methodology) no plant data existed, so a comparison of limiting events to previous analyses performed by OPPD using CESEC-III was done. The events chosen for comparison were:

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

(2) 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 [

]

OPPD has been using CESEC-III to perform reload safety analyses since 1982 and has successfully completed analyses for six cycles. Therefore, the predictions from CESEC-III provide a reliable baseline against which CENTS can be benchmarked.

6.4 Comparisons Between OPPD Analyses and Independent Analyses Previously Performed with CESEC-III for Verification of CENTS (Continued)

6.4.1 Dropped CEA

The Cycle 12 Dropped CEA analysis performed by OPPD using CENTS was compared to the previous analyses performed by OPPD using CESEC-III. The Dropped CEA analyses performed for Cycle 12 using CESEC-III was used for comparison. Table 6.4-1 summarizes the parameters and their values used for the Cycle 12 analysis. The response of various system parameters to the event are shown in Figures 6-38 to 6-41. All of the figures show good agreement between the CESEC-III and CENTS predictions.

Figures 6-38 and 6-39 show the response of core power and heat flux as predicted by the CESEC-III and CENTS codes. The predicted responses from both codes are similar, with CENTS predicting a higher final power and heat flux, which is conservative. As described in the analysis methodology in Section 5.3, both cases (CENTS and CESEC-III) assumed that the turbine admission valves were placed in manual during the event in order to simulate the actual mode of operation of the turbine control system at Fort Calhoun Station. Figure 6-40 contains plots of the RCS hot and cold leg coolant temperatures versus time. The response predicted by CESEC-III and CENTS are in good agreement. Figure 6-41 shows plots of the pressurizer pressure versus time. The pressures predicted at 200 seconds are 2003 psia and 2006 psia by the CESEC-III and CENTS codes, respectively. This difference is small enough to be less than the pressure measurement uncertainty.

In summary, the primary system responses between the CESEC-III and CENTS predictions show excellent agreement with each other.

6.4.2 RCS Depressurization

The Cycle 12 RCS Depressurization event was used as a basis for this comparison. This event was analyzed previously using CESEC-III. An analysis was performed using CENTS in order to compare the predictions from the two codes. Table 6.4-2 summarizes the parameters and their values used for the Cycle 12 analysis.

6.4 Comparisons Between OPPD Analyses and Independent Analyses Previously Performed with CESEC-III for Verification of CENTS (Continued)

The CESEC-III code does not simulate a thermal margin/low pressure (TM/LP) trip, so a TM/LP trip must be simulated using a manual trip. CENTS, however, does simulate a TM/LP trip (by virtue of CENTS control system modelling language), so the TM/LP trip was credited in the CENTS simulation of the event.

A manual trip is simulated in CESEC-III 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 6-42 shows plots of core power versus time for the CENTS and CESEC-III analyses. As can be seen in the figure, the two codes gave very similar predictions for core power versus time, except for the time of trip. However, as stated previously, the trip in CENTS was based on the actual TM/LP control system, while the time of trip in the CESEC-III case was chosen arbitrarily.

Figure 6-43 shows the RCS pressure response from the CESEC-III and CENTS analyses. The results show very good agreement post-trip.

The pressurizer pressure versus time plots are shown in Figure 6-44. In both the CESEC-III and CENTS analyses, a value of 2172 psia (2150 psia + 22 psia measurement uncertainty) was used for the initial RCS pressure. The CENTS analysis shows an initial depressurization rate slightly higher than that of CESEC-III. However, after about 3 seconds both cases depressurized at approximately equal rates.

The comparison of the figures show good agreement in the trends for the core power and pressurizer pressure.

6.5 Summary

The CESEC-III computer code was developed for the analysis of FSAR transient and accident events for the two-by-four loop Combustion Engineering plants. The CENTS code was developed for the simulation of the transient response for a PWR during normal and abnormal conditions including accidents. It is a flexible nodal code that can be used for resolving safety and operability issues, as well as analyzing FSAR transient and accident events. OPPD's engineering staff was trained for the proper use of CESEC-III and CENTS and a close working relationship has been maintained with both of the CE development teams. To ensure accurate prediction capabilities, both CESEC-III and CENTS plant models were developed, inputs tested, and the output verified and validated against both plant data and other SAR analyses.

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 both CENTS and CESEC-III.

Verification of CESEC-III for transients for which plant data was not available was accomplished by performing comparisons between the OPPD Cycle 8 analyses of the limiting transients and the Cycle 6 analyses by 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.

Verification of CENTS for transients for which plant data was not available was accomplished by performing comparisons between OPPD Cycle 12 CESEC-III analyses of limiting transients and analyses of the same events using CENTS. In all cases, these benchmarking comparisons showed very good agreement.

OPPD continues to maintain a quality assured model for CESEC and plans to QA the basedeck for CENTS before it is used for any licensing analyses. OPPD will also continue to provide an update basedeck for each cycle, contained in an Engineering Analysis (EA).

TABLE 6.3-1

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} °F/$\Delta\rho$	-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 6.3-2

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 6.[illegible]-3

COMPARISON OF PARAMETERS INCLUDING UNCERTAINTIES
USED IN THE HFP MAIN STEAMLINE BREAK ANALYSIS 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	[illegible]17	547	547
Pressurizer Pressure	psia	[illegible]73	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		0.3	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 6.3-4

COMPARISON OF PARAMETERS INCLUDING UNCERTAINTIES USED IN THE RCS DEPRESSURIZATION ANALYSES FOR CYCLE 8 AND EXAMPLE CASE

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.

TABLE 6.4-1

COMPARISON OF PARAMETERS INCLUDING UNCERTAINTIES
USED IN THE CEA DROP ANALYSES FOR CYCLE 12

Parameter	Units	CESEC-III	CENTS
Initial Core Power Level	MWt	100% of 1500	100% of 1500
Core Inlet Temperature	°F	545	545
Pressurizer Pressure	psia	2075	2075
RCS Flow Rate	gpm	208,280	208,280
Moderator Temperature Coeff.	10^{-4} $\Delta\rho$/°F	-2.7	-2.7
Doppler Coeff. Multiplier		1.15	1.15
CEA Insertion at Full Power	% Insertion	25.0	25.0
Dropped CEA Worth	%$\Delta\rho$	-0.23	-0.23

TABLE 6.4-2

COMPARISON OF PARAMETERS INCLUDING UNCERTAINTIES
USED IN THE RCS DEPRESSURIZATION ANALYSES FOR CYCLE 12

Parameter	Units	CENTS	CESEC-III
Initial Core Power Level	MWt	102% of 1500	102% of 1500
Core Inlet Temperature	°F	547	547
Pressurizer Pressure	psia	2172	2172
RCS Flow Rate	gpm	211,000	211,000
Moderator Temperature Coefficient	10^{-4} $\Delta\rho/°F$	-2.7	-2.7
Doppler Coeff. Multiplier		1.15	1.15
TM/LP Trip Unit Coefficients	α β γ	29.73 18.44 -11350	N/A
	A1(Y)*	$-0.35294Y + 1.08824$, $Y \leq .25$ $0.57143\,Y + 0.875$, $Y > .25$	N/A
	PF(B)**	1.0, $B \geq 100\%$ $-.008B + 1.8$, $50\% < B < 100\%$ 1.4, $B \leq 50\%$	N/A

*Y = internal ASI
**B = core power

7.0 REFERENCES

Section 2 References

2-1 Westinghouse Proprierary Document, "Report on the Analysis Methods and Evaluation Models to be Employed in the Large Break and Small Break LOCA Analyses for Fort Calhoun Unit 1 ", February 1991.

Section 4 References

4-1 CESEC, Digital Simulation of a Combustion Engineering Nuclear Steam Supply System, December, 1981, transmitted as Enclosure 1-P to LD-82-001, January 6, 1982.

4-2 CEN-234(C)-P, Louisiana Power and Light Company, Waterford Unit 3, Docket 50-382, Response to Questions on CESEC, December, 1982.

4-3 Letter from A. E. Scherer (CE) to F. J. Miraglia (NRC), "Applicability of CESEC-III to the Fort Calhoun Station," February 27, 1987.

4-4 CENPD-161-P, TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core," July, 1975.

4-5 CEN-191(B)-P, "CETOP-D Code Structure and Modeling Methods for Calvert Cliffs 1 and 2," December, 1981.

4-6 CENPD-162-P-A, "CE Critical Heat Flux, Critical Heat Flux Correlation for CE Fuel Assemblies with Standard Spacer Grids Part 1 Uniform Axial Power Distributions," September, 1976.

4-7 CENPD-207-P, "CE Critical Heat Flux, Critical Heat Flux Correlation for CE Fuel Assemblies with Standard Spacer Grids Part 2 Non-uniform Axial Power Distributions," June, 1978.

4-8 Letter from E. G. Tourigny (NRC) to W. C. Jones (OPPD) dated March 15, 1983.

4-9 OPPD-NA-8301, Rev. 03, "Reload Core Analysis Overview", April, 1988.

4-10 CEN-257(0)-P, "Statistical Combination of Uncertainties", November, 1983.

4-11 CE-NPD-282-P, "Technical Manual for the CENTS Code" Volumes 1 and 2, February 1991.

4-12 T. A. Porsching, et al., "FLASH-4: A Fully Implicit FORTRAN IV Program for the Digital Simulation of Transients in a Reactor Plan", WAPD-TM-840, March 1969.

4-13 CE-NPD-282-P, "Technical Manual for the CENTS Code" Volume 3, April 1991.

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5-1 CEN-347(0)-P, Rev. 01, "Omaha Batch M Reload Fuel Design Report", January, 1987.

5-2 CEN-121(B)-P, "CEAW, Method of Analyzing Sequential Control Element Assembly Group Withdrawal Event for Analog Protected Systems", November, 1979.

5-3 CENPD-199-P, Revision 1-P-A, "CE Setpoint Methodology", January 1986.

5-4 Fort Calhoun SER on Automatic Initiation of Auxiliary Feedwater, contained in the letter to W. C. Jones from Robert A. Clark, dated February 20, 1981.

5-5 Westinghouse Proprietary Document, "Control Element Assembly Ejection Accident Methodology Summary Report for Fort Calhoun Unit 1", April 1991.

5-6 OPPD-NA-8302, Rev. 02, "Nuclear Design Methods and Verifications", April, 1988.

5-7 Westinghouse Proprietary Document, "Report on the Analysis Methods and Evaluation Models to be Employed in the Large Break and Small Break LOCA Analyses for Fort Calhoun Unit 1 ", February 1991.

5-8 Fort Calhoun SER on Generic Letter 86-06 (TMI Action Item II.K.3.5, "Automatic Trip Reactor Coolant Pumps During Loss-of-Coolant Accident"), contained in the letter to R. L. Andrews (OPPD) from Anthony Bournia (NRC), dated March 25, 1988.

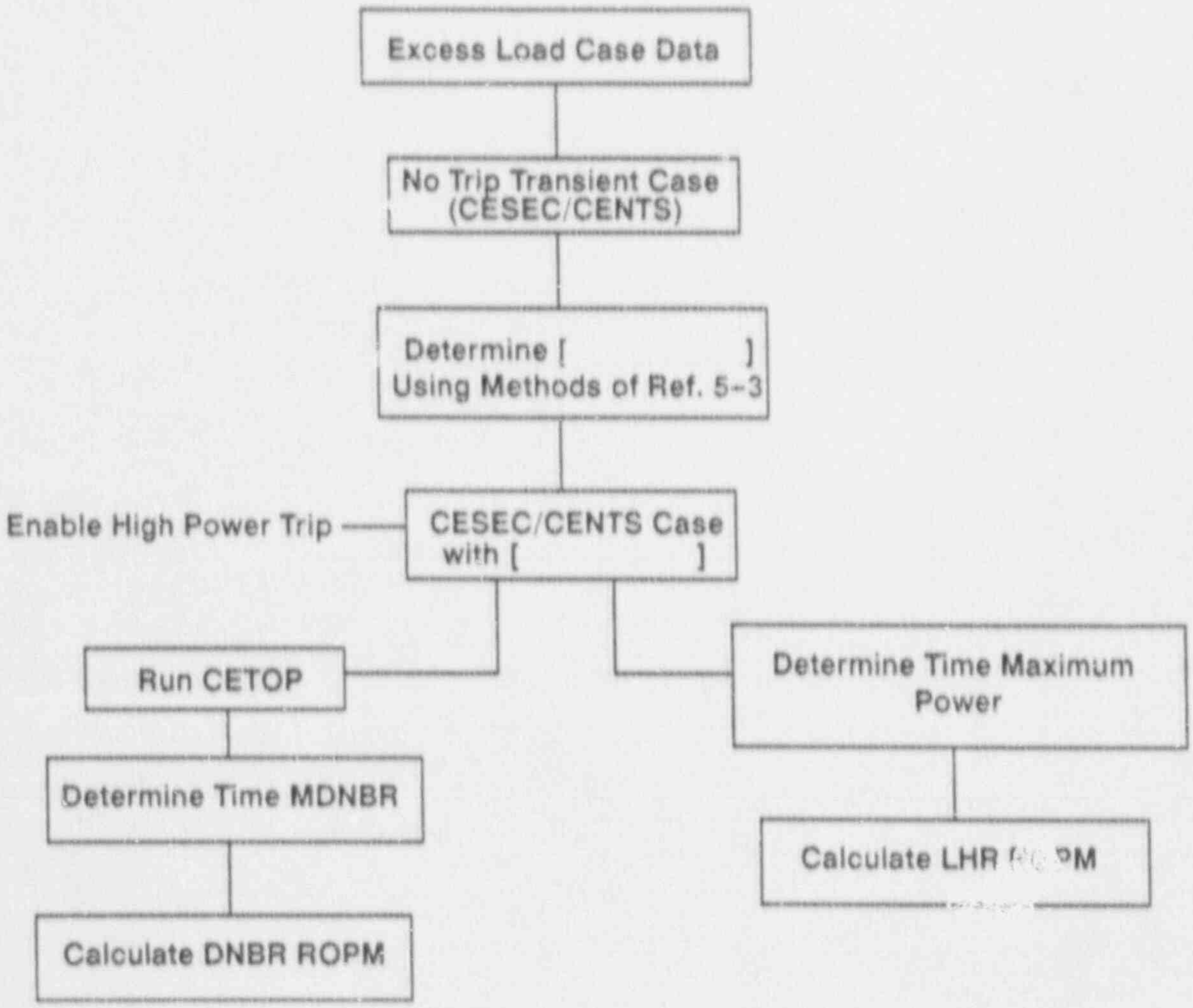
Section 6 References

6-1 "CESEC - Digital Simulation of a CE NSSS", Enclosure 1-P to LD-82-001, January 6, 1982.

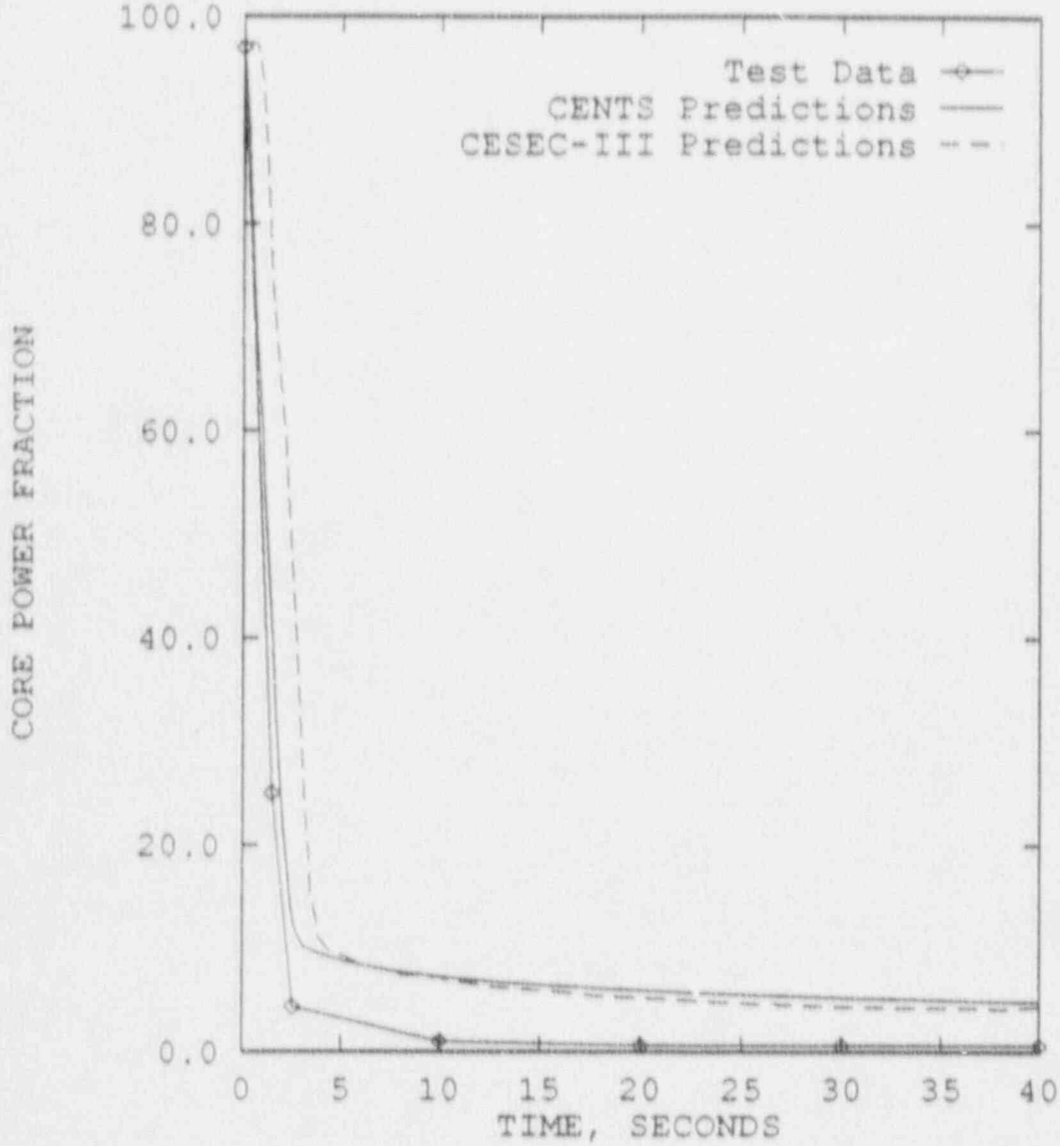
6-2 Letter from A. E. Scherer to F. J. Miraglia, LD-87-013 dated February 27, 1987.

6-3 CE-NPD-282-P, "Technical Manual for the CENTS Code" Volumes 1 and 2, February 1991.

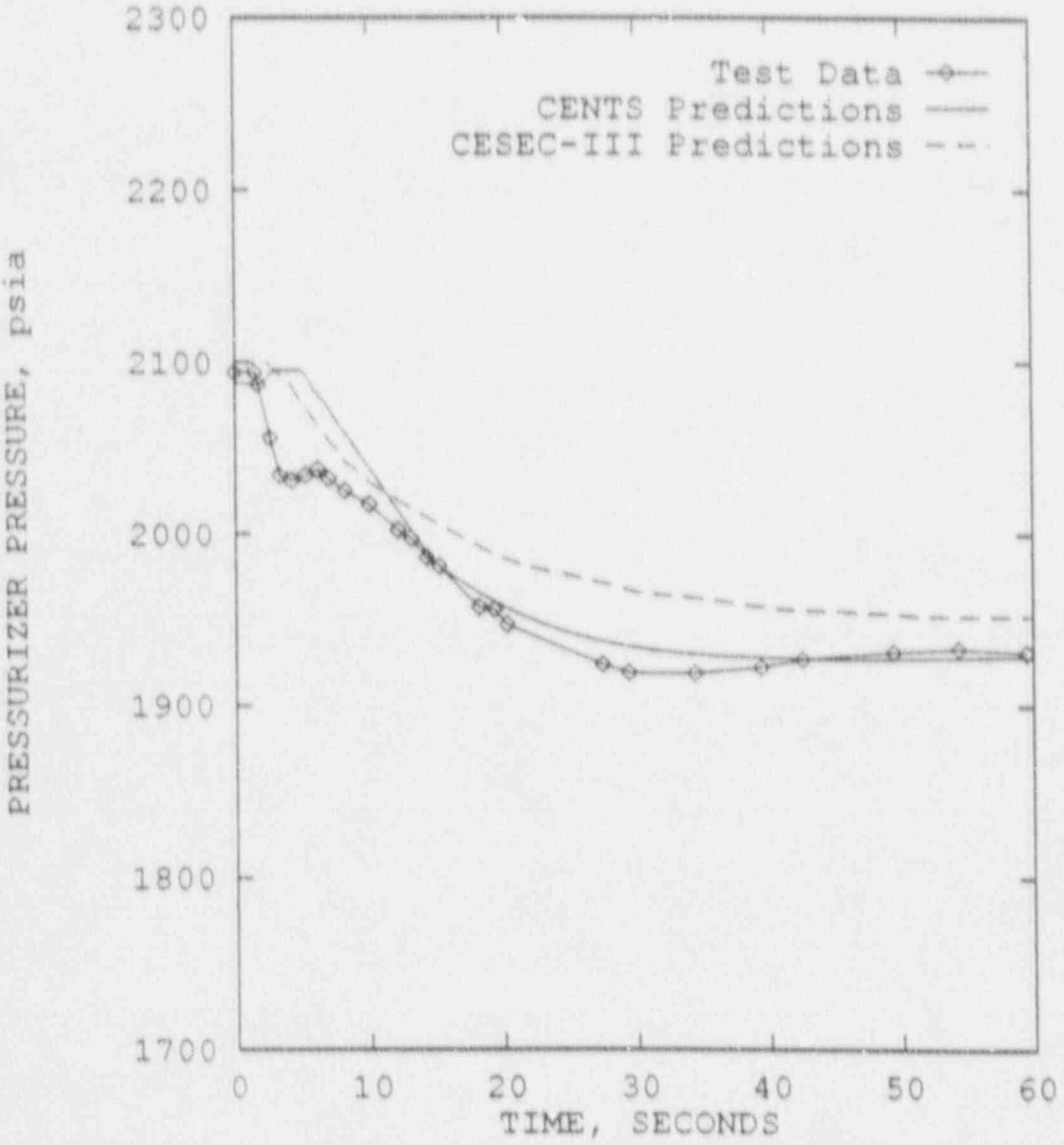
6-4 CE-NPD-282-P, "Technical Manual for the CENTS Code" Volume 3, April 1991.



Excess Load Event ROPM Methods	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 5-1



Full Power Turbine Trip Core Power vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-1



Full Power Turbine Trip Pressurizer Pressure vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-2

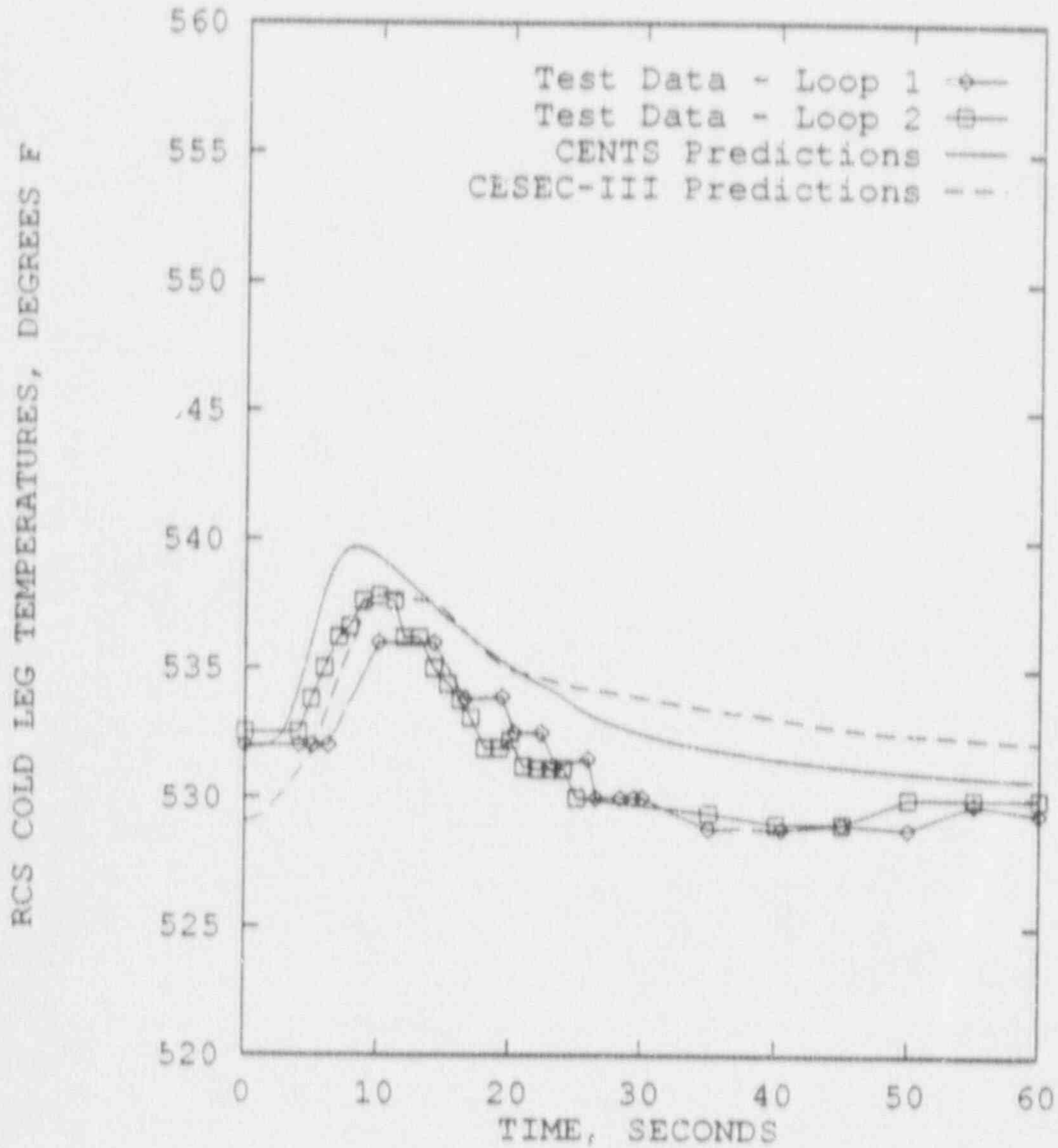
The ECT/RPC inspections revealed 104 tubes with circumferential cracks at the expansion transition. The macrocracks, as defined by ECT/RPC, consisted of several discontinuous microcracks that were separated by small ligaments of sound material. The discontinuous nature of the array of microcracks was confirmed by the UT and examination of the removed tube specimens. As measured by UT, the macrocracks ranged in circumference from 84 degrees to 329 degrees and ranged in depth up to 100-percent throughwall.

All tubes with crack indications were staked and plugged. In addition, the licensee evaluated the residual strength of the cracked tubes to assess their capability to sustain normal operating and postulated accident loadings before their removal from service. This structural evaluation considered the profiles for each crack obtained from the UT examination. This evaluation revealed one cracked tube which failed to meet the ASME Code, Section III, NB-3225 and Appendix F stress limits for postulated accident conditions. (Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," states that margins should be consistent with the stress limits in Section III of the code.) Based on these findings, the staff concludes that the integrity of the subject tube was not ensured under postulated accident conditions.

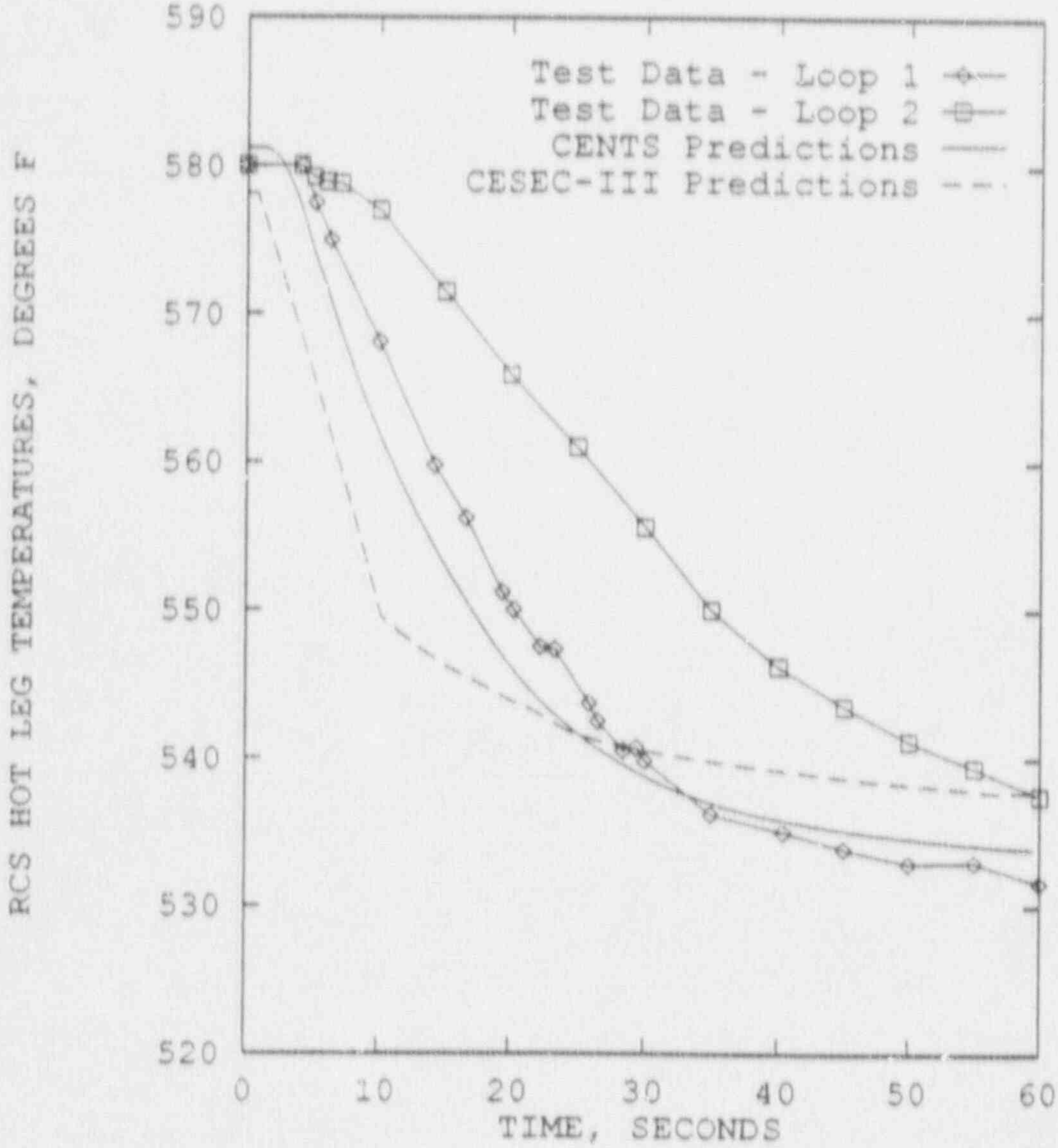
The staff has recently identified service induced, circumferential SCC, such as at Millstone Unit 2, to be a source of significant degradation to tubes in PWR steam generators. Such cracking is particularly noteworthy because it is generally not detectable with conventional bobbin probes used routinely for inservice inspection. Such cracking is generally only detectable through the use of specialized probes, such as the RPC probe.

Most circumferential cracking has been observed at tube expansion transitions at or near the top of the tubesheet. In addition to Millstone Unit 2, circumferential cracking at the expansion transition has recently been identified at one other Combustion Engineering (CE) plant (Maine Yankee), at three plants with Westinghouse Model 51 steam generators (North Anna Unit 1, Trojan Unit 1, and Sequoyah Unit 1), and at one plant with Westinghouse Model D steam generators (McGuire Unit 1). Tubes in the affected CE and Westinghouse Model 51 steam generators were explosively expanded against the tubesheet. Tubes in the McGuire Model D steam generators were expanded against the tubesheet by mechanical rolling.

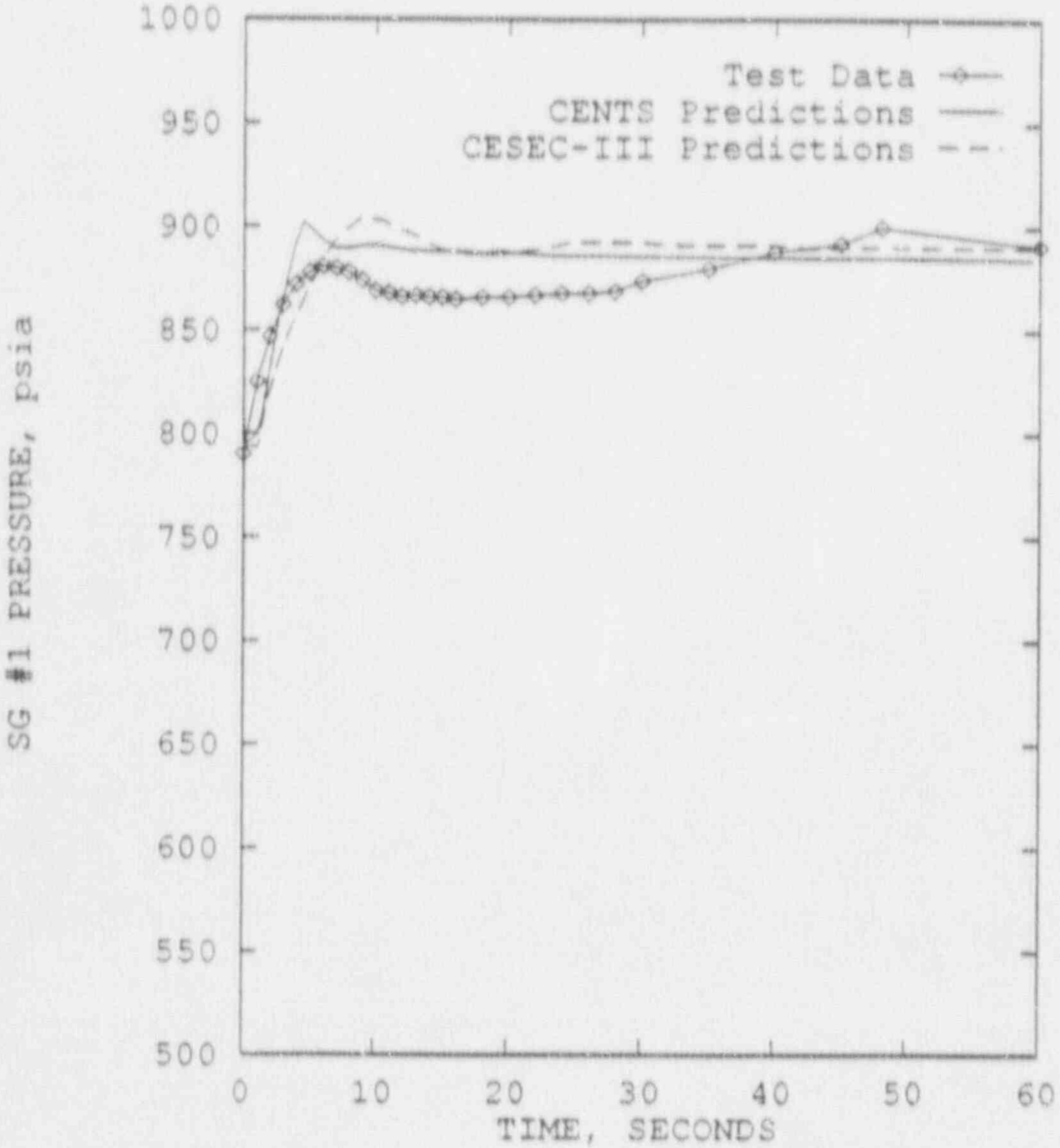
In addition to being found at the expansion transition location, widespread circumferential SCC has been observed at drilled-hole support plate locations at Palisades (CE steam generators). Isolated instances of circumferential SCC have been reported at the uppermost support plate of a pre-replacement Westinghouse Model 44 steam generator of Indian Point Unit 3 and at a row 1 U-bend of a Model 51 steam generator at Zion Unit 1. The circumferential SCC at Palisades and Indian Point Unit 3 appears to be associated with significant denting at the support plates.



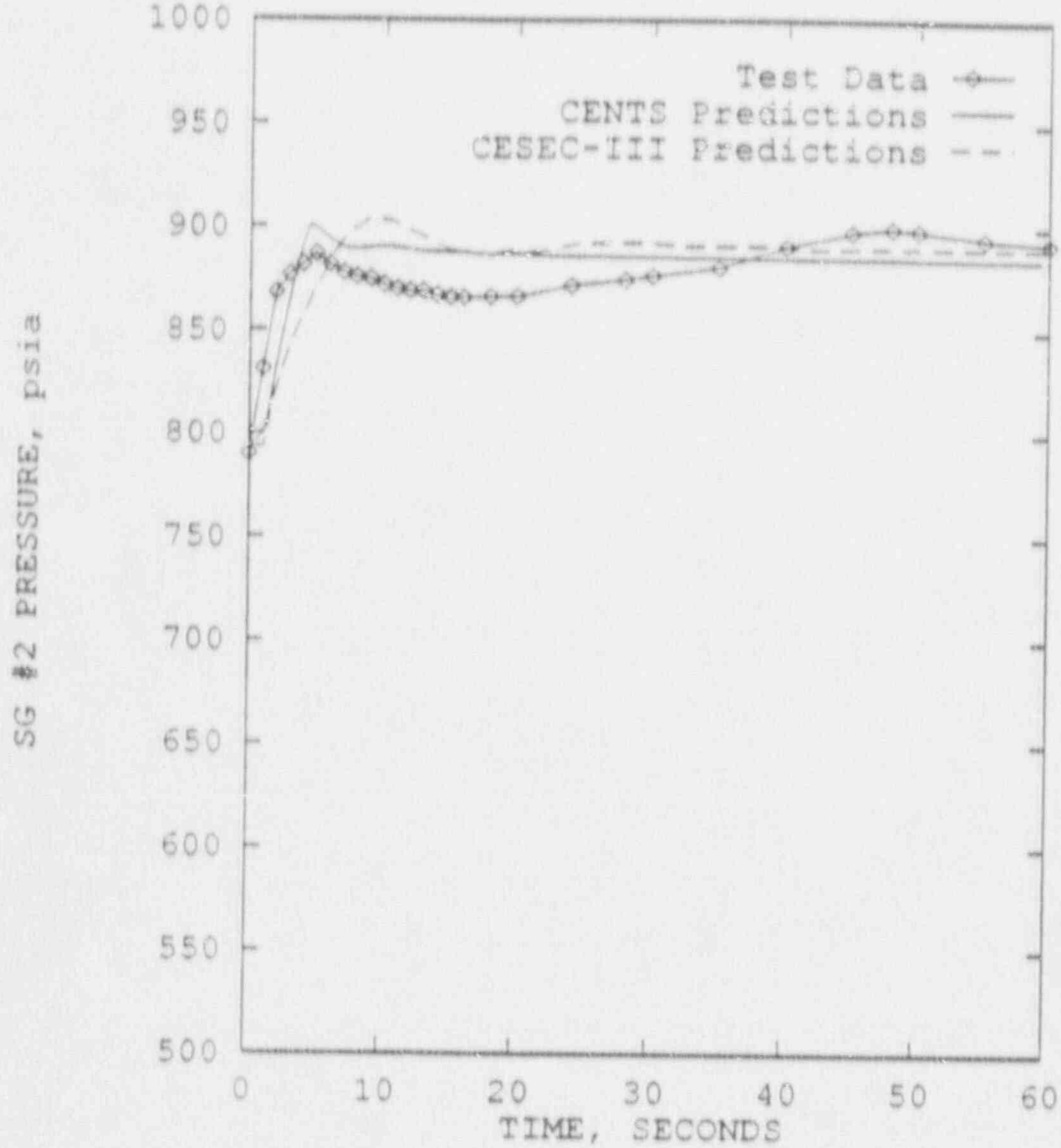
Full Power Turbine Trip RCS Cold Leg Temperatures	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-3



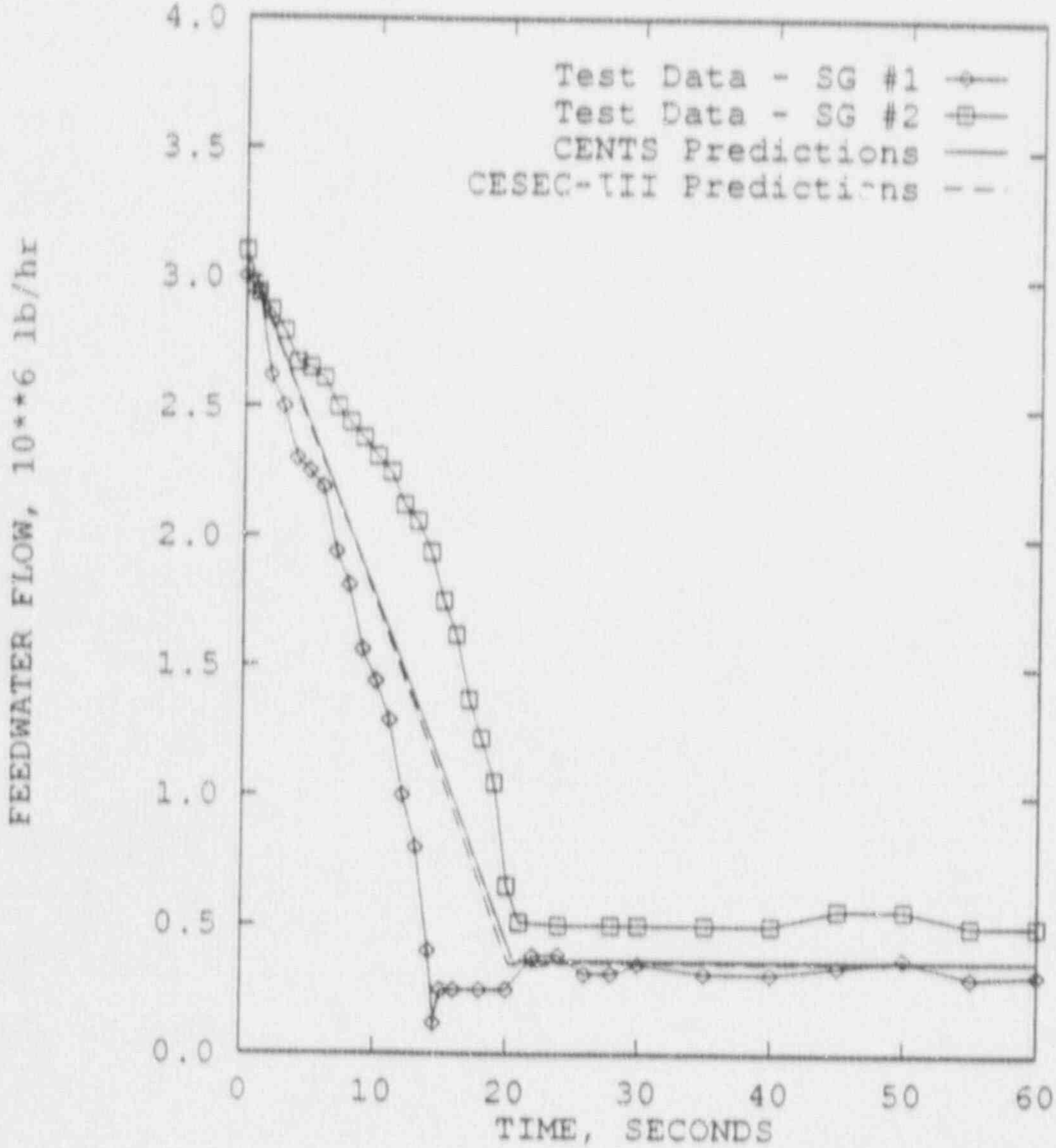
Full Power Turbine Trip RCS Hot Leg Temperatures	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-4



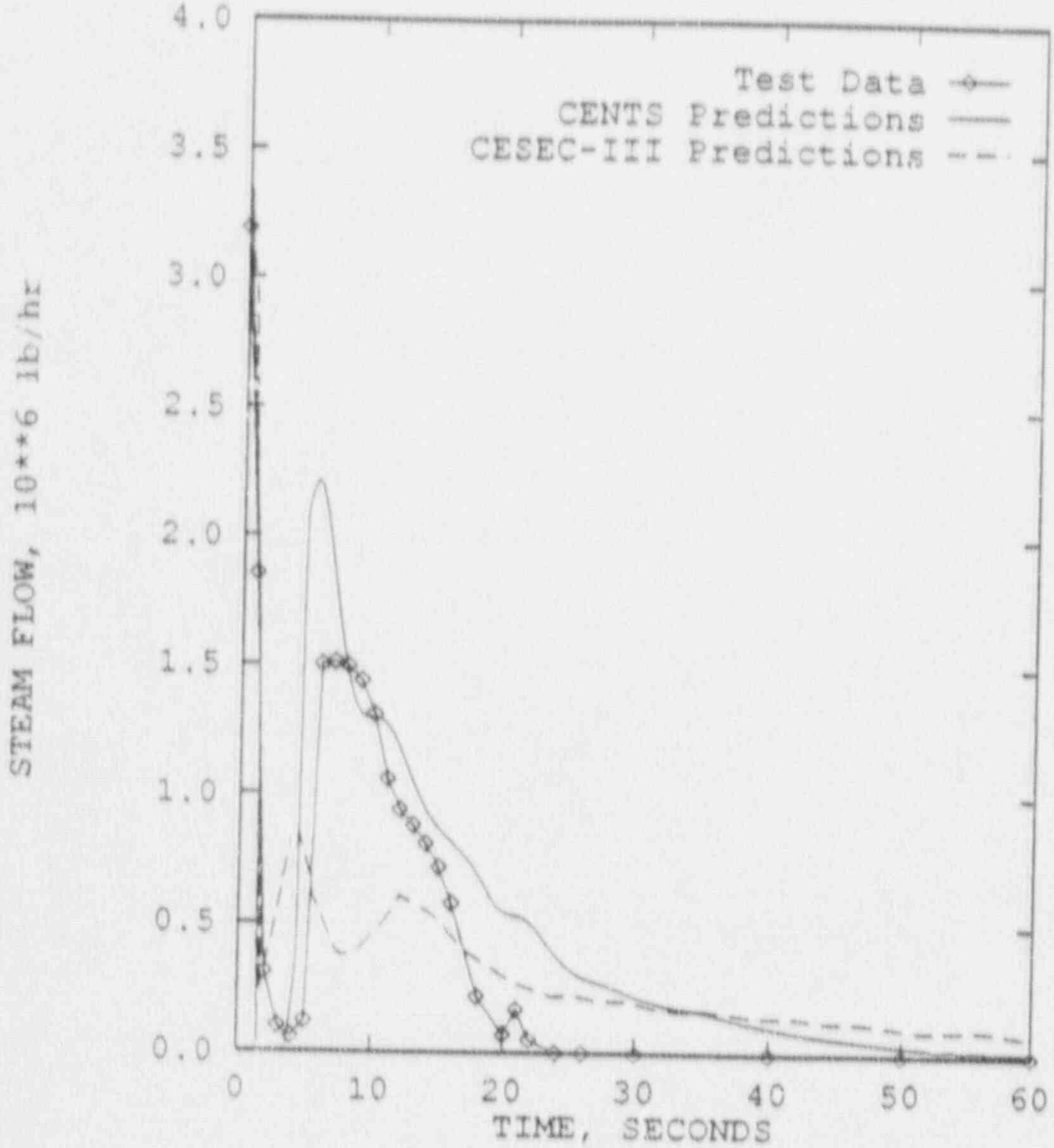
Full Power Turbine Trip Steam Generator #1 Pressure	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-5



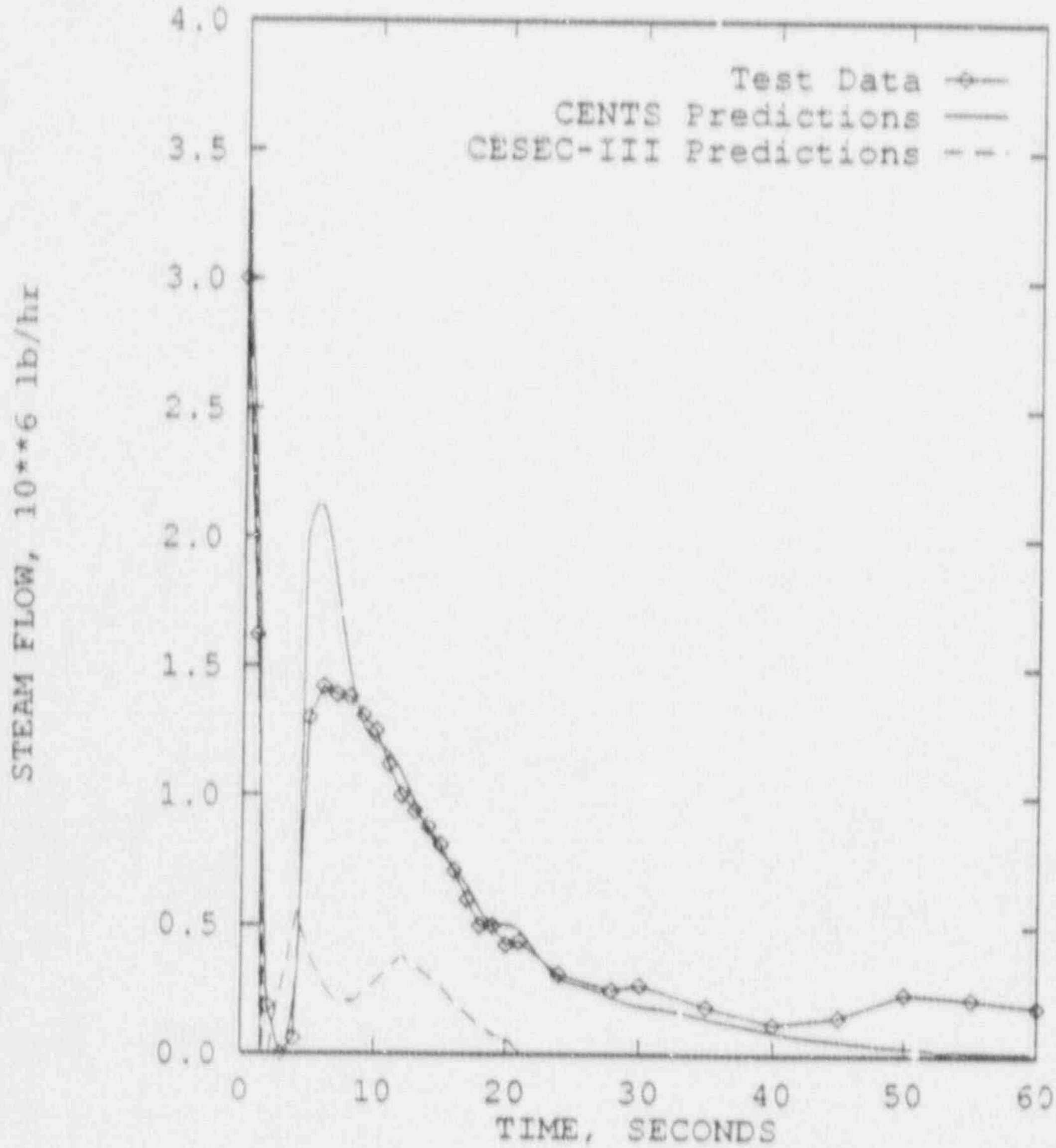
Full Power Turbine Trip Steam Generator #2 Pressure	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-6



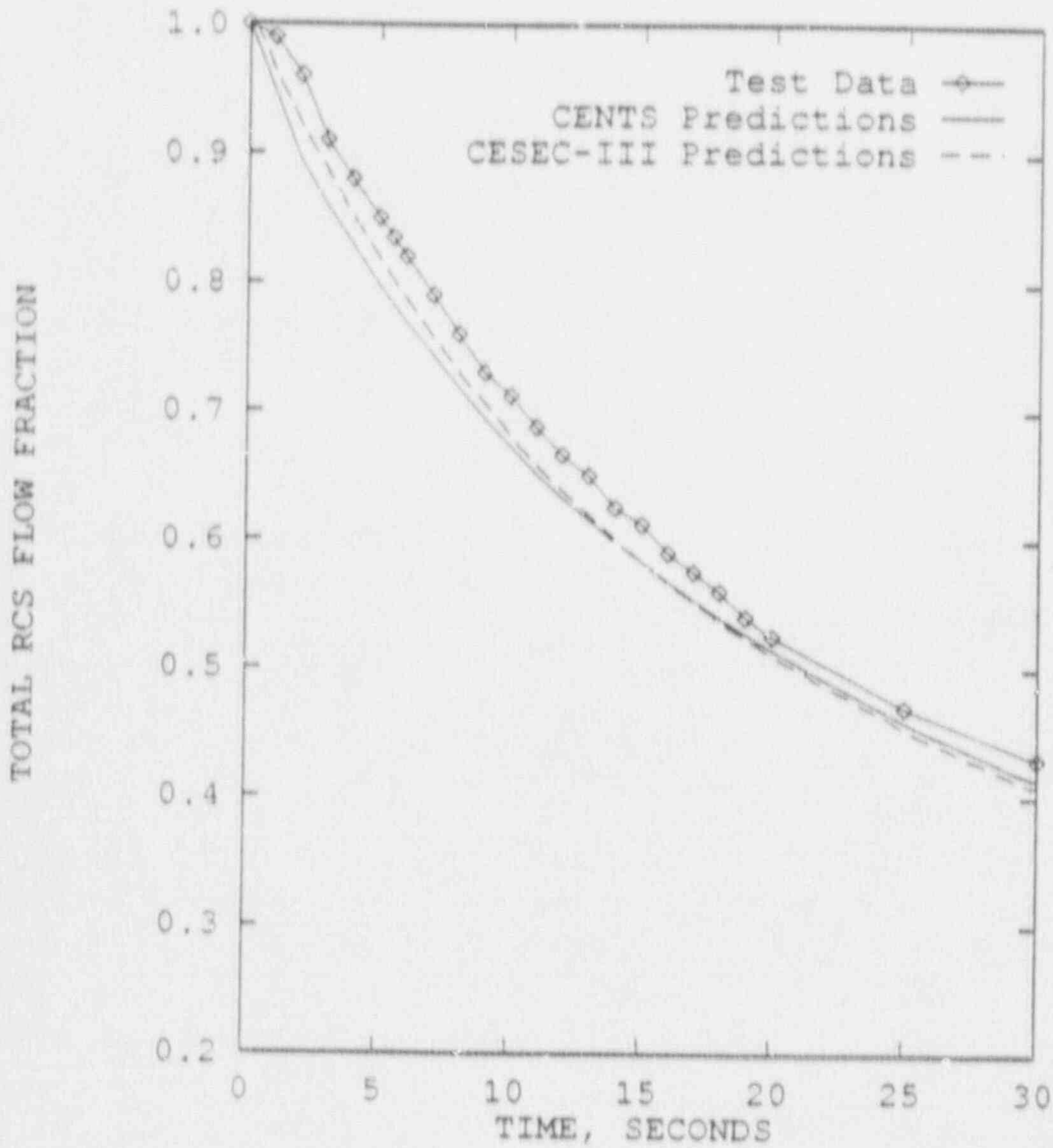
Full Power Turbine Trip Feedwater Flow vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-7



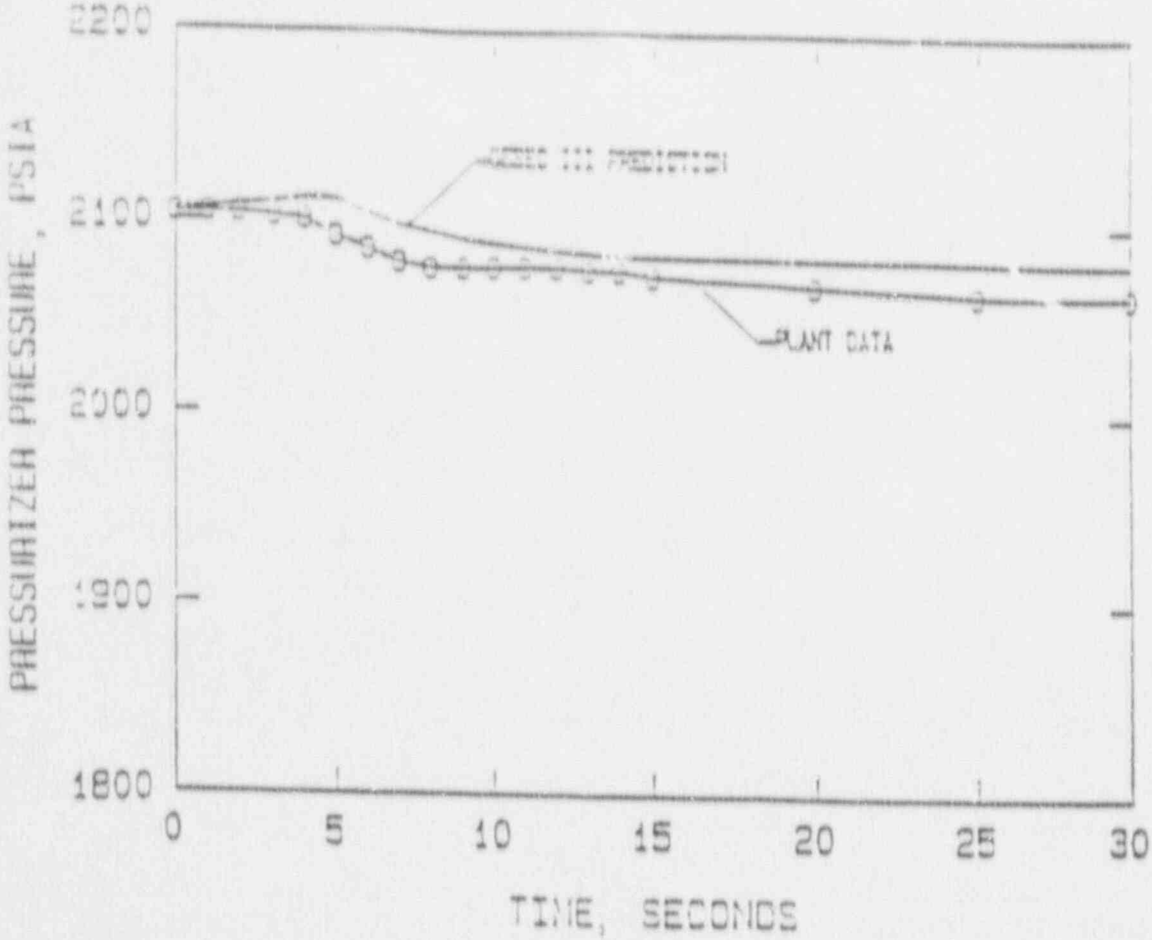
Full Power Turbine Trip Steam Generator #1 Steam Flow	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-8



Full Power Turbine Trip Steam Generator #2 Steam Flow	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-9



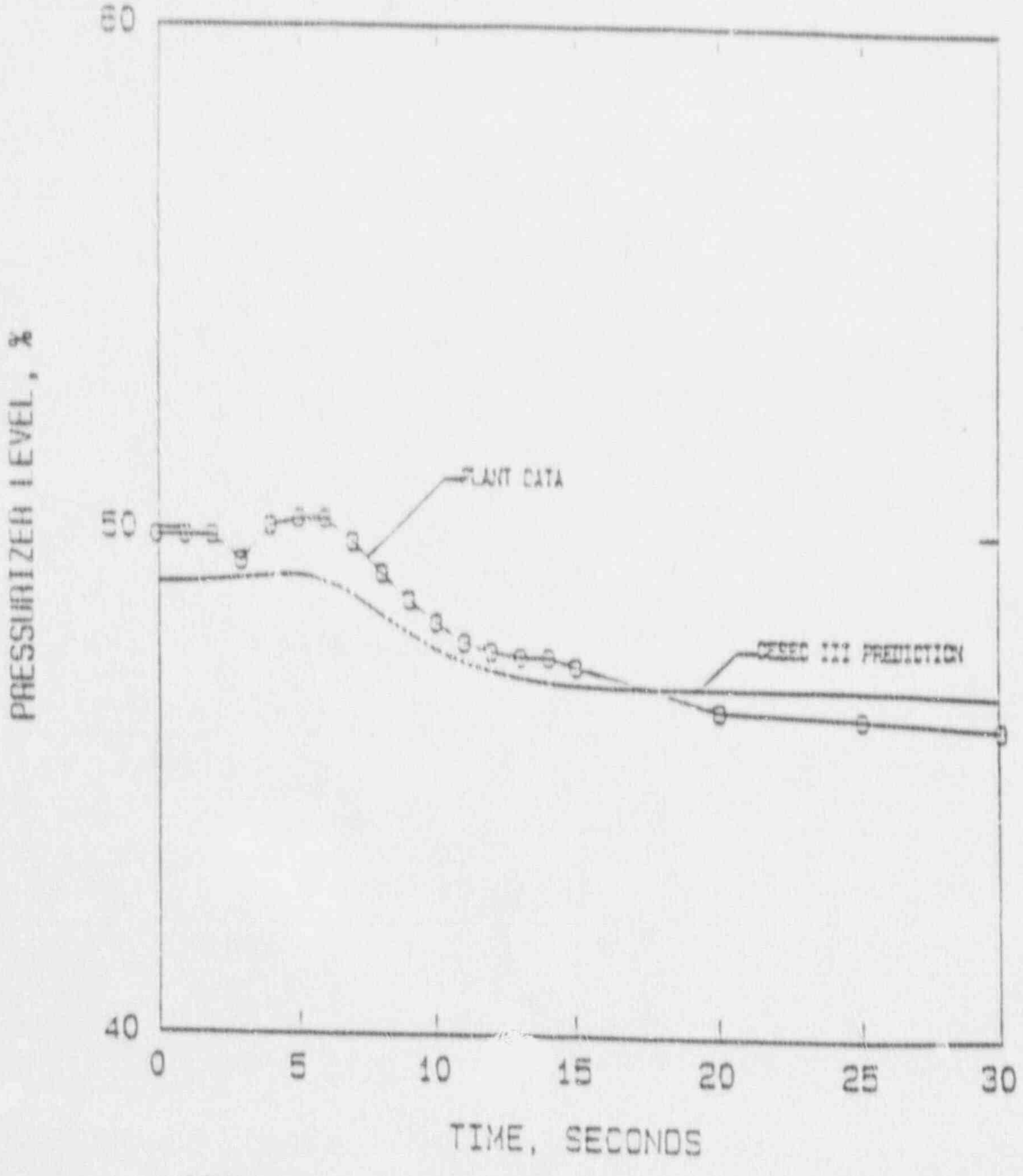
Four Pump Loss of Flow Total RCS Flow Fraction vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-10



NOTE :

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

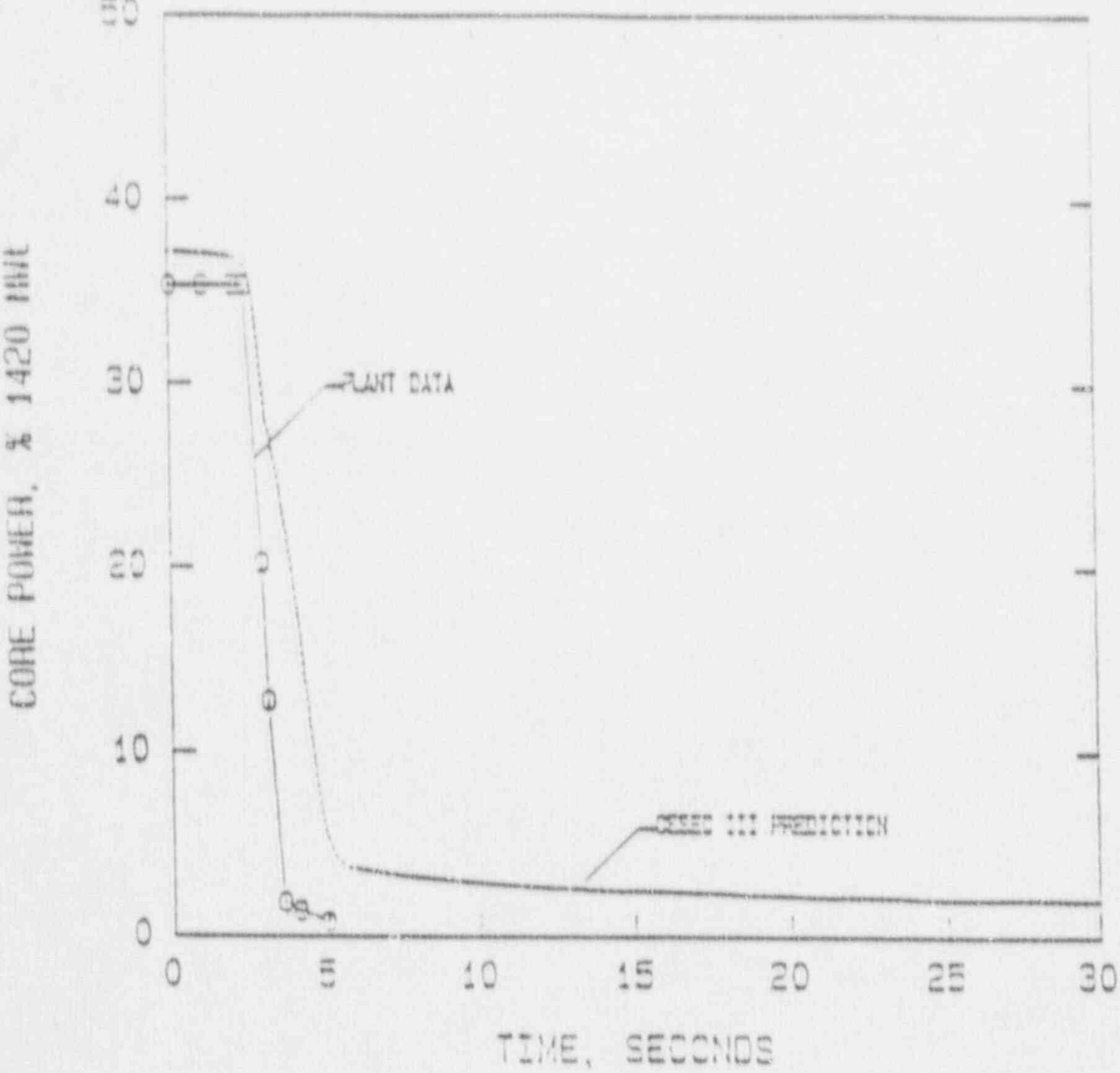
Four Pump Loss of Flow Pressurizer Pressure vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-11



NOTE :

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

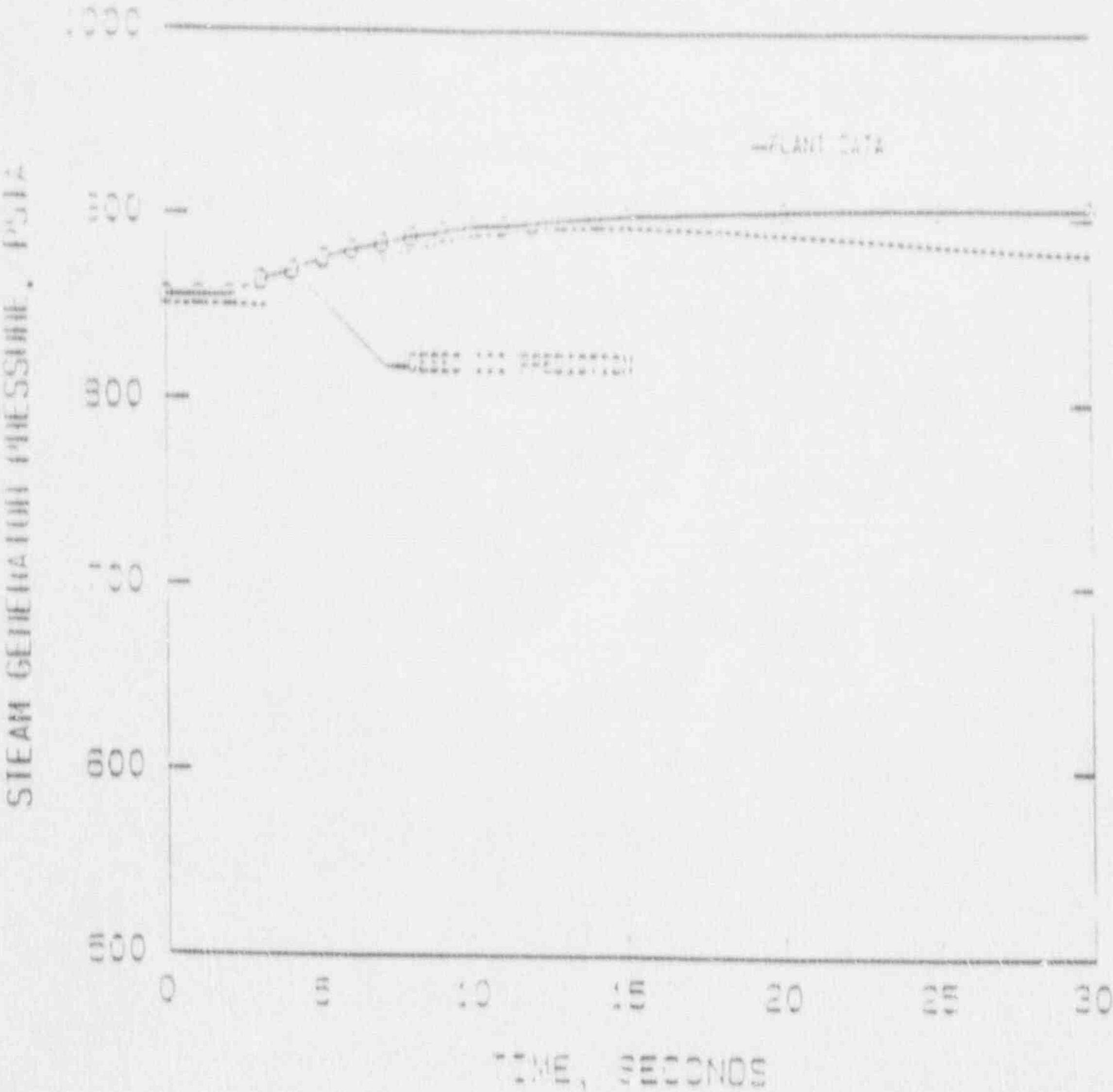
Four Pump Loss of Flow Pressurizer Level vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-12



NOTE :

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

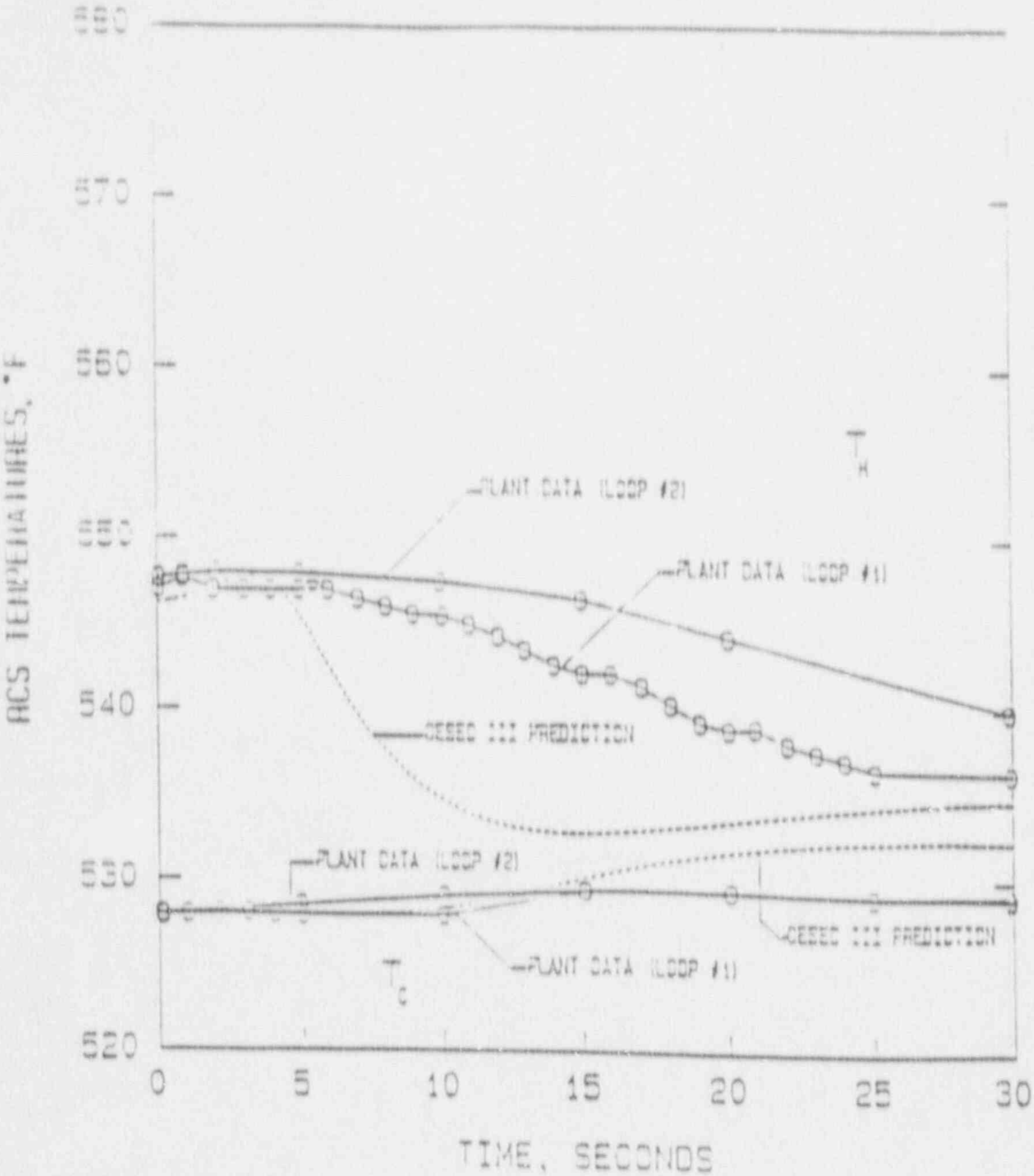
Four Pump Loss of Flow Core Power vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-13



NOTE

CYCLE 1 (FULL POWER = 1420 MWt)
INITIAL POWER = 35%
PLANT DATA: TEST PERFORMED MARCH 8, 1974

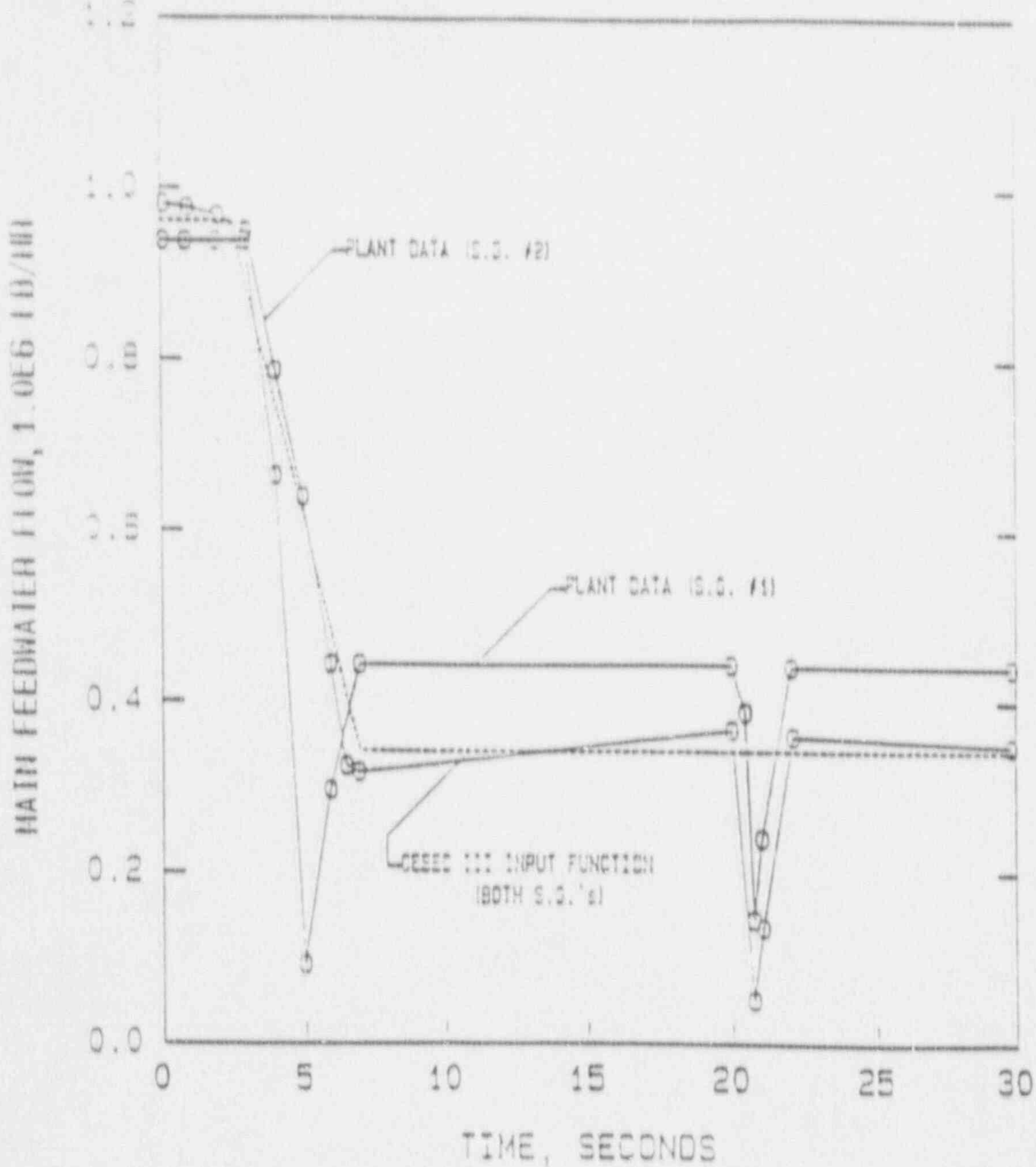
Four Pump Loss of Flow Steam Generator Pressure vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-14



NOTE :

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

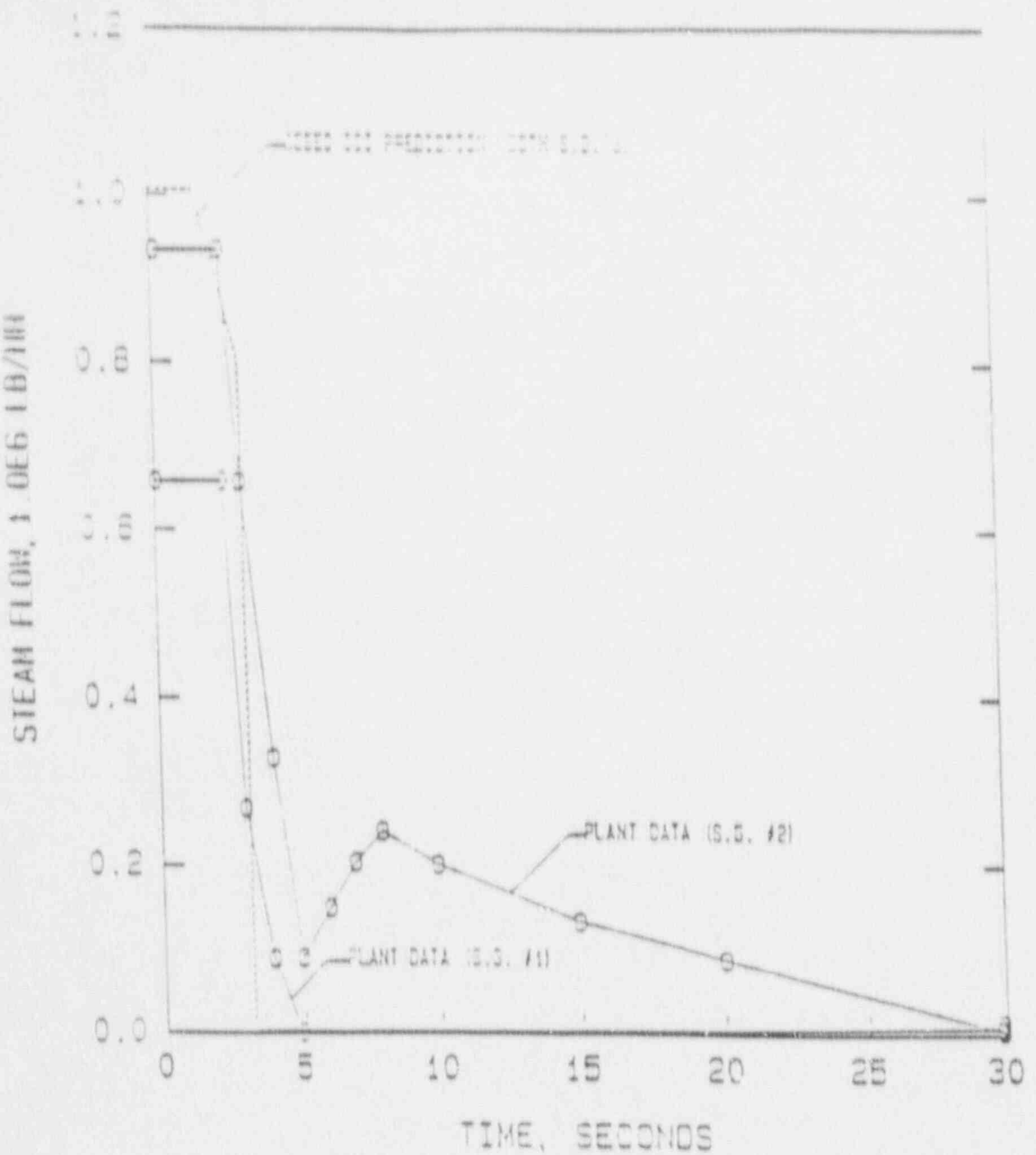
Four Pump Loss of Flow RCS Temperatures vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-15



NOTE :

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

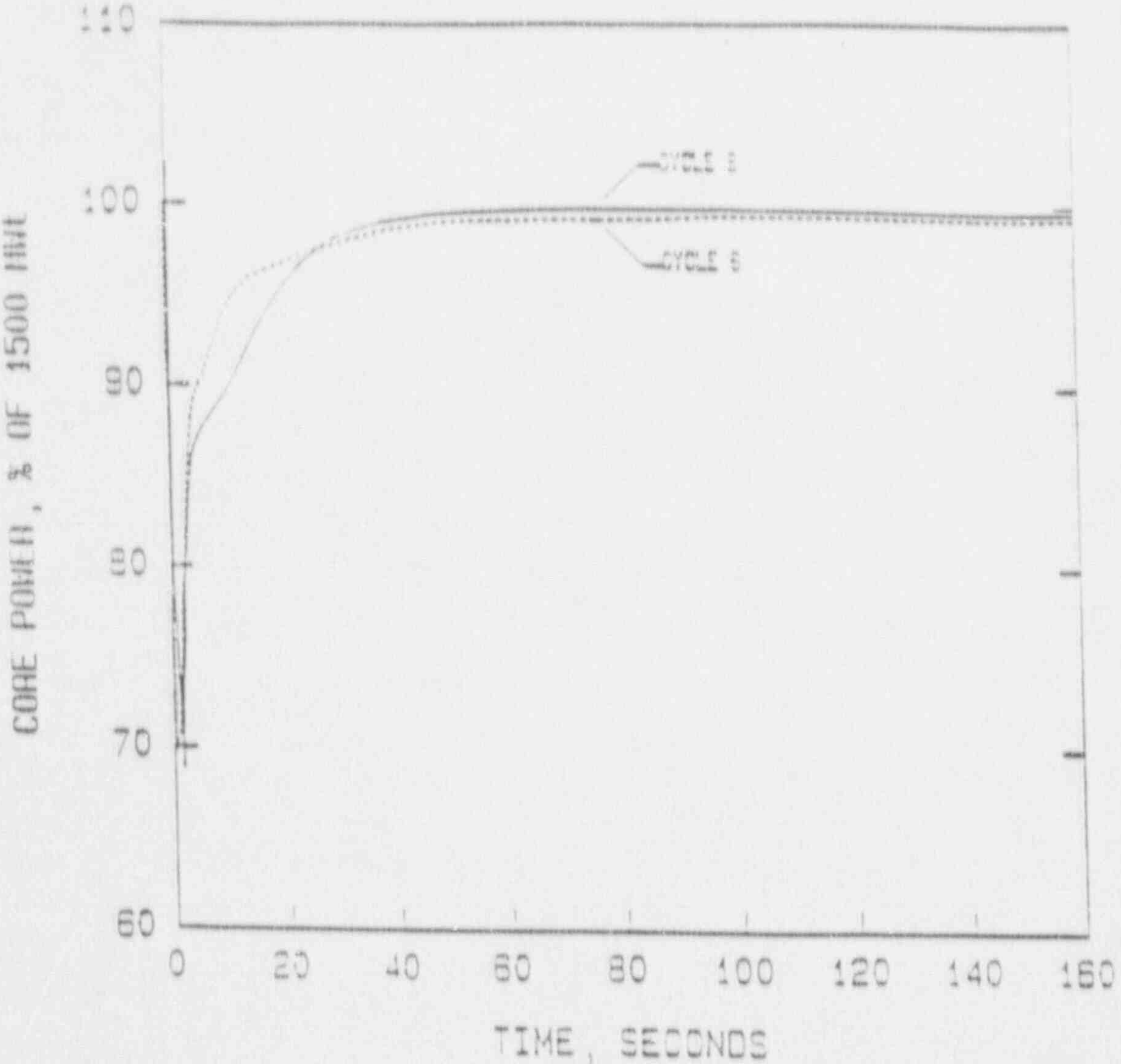
Four Pump Loss of Flow Main Feedwater Flow vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-16



NOTE :

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

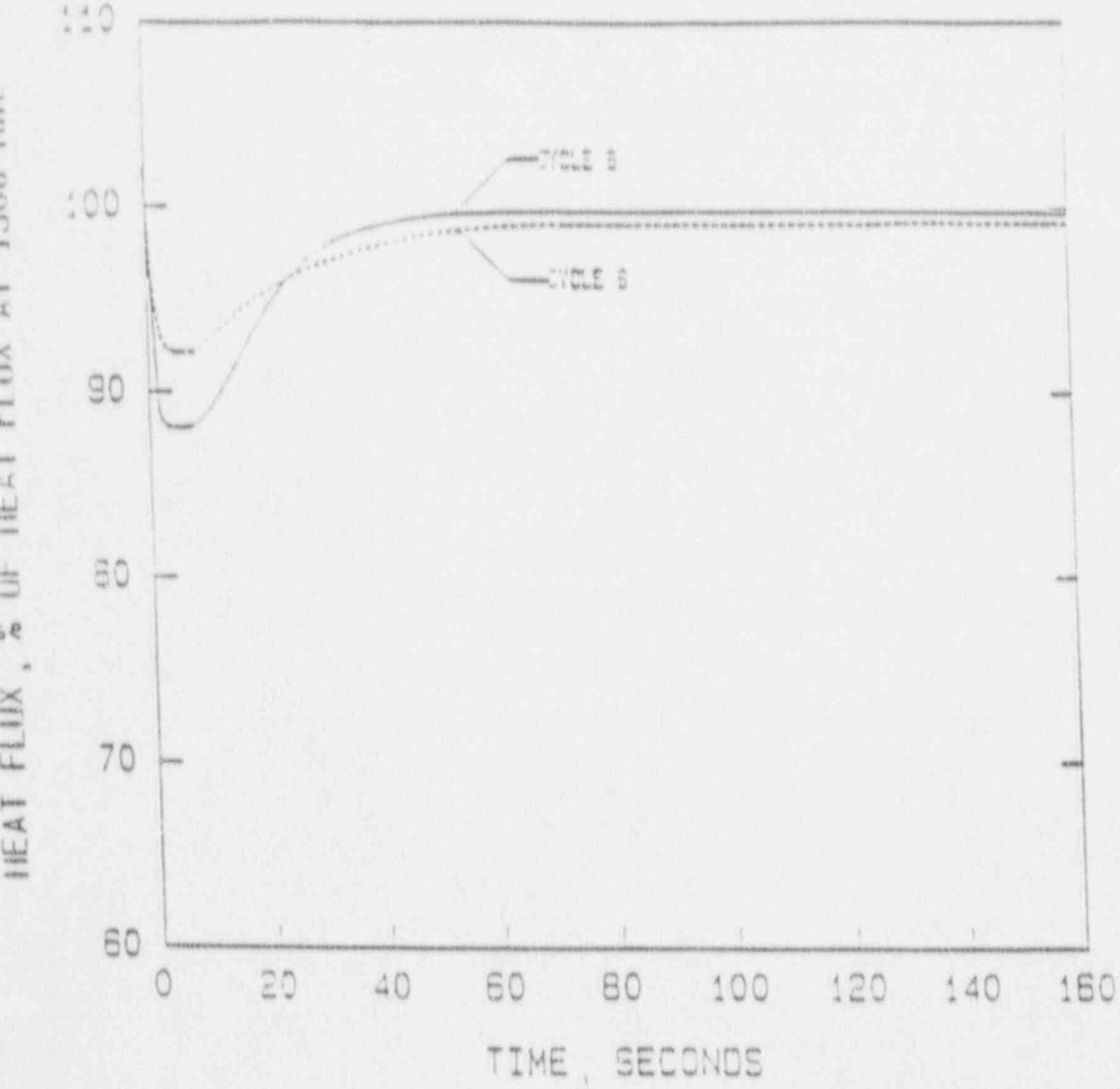
Four Pump Loss of Flow Steam Flow vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-17



NOTE :

CYCLE 6: ENC ANALYSIS
CYCLE 8: OPPD ANALYSIS

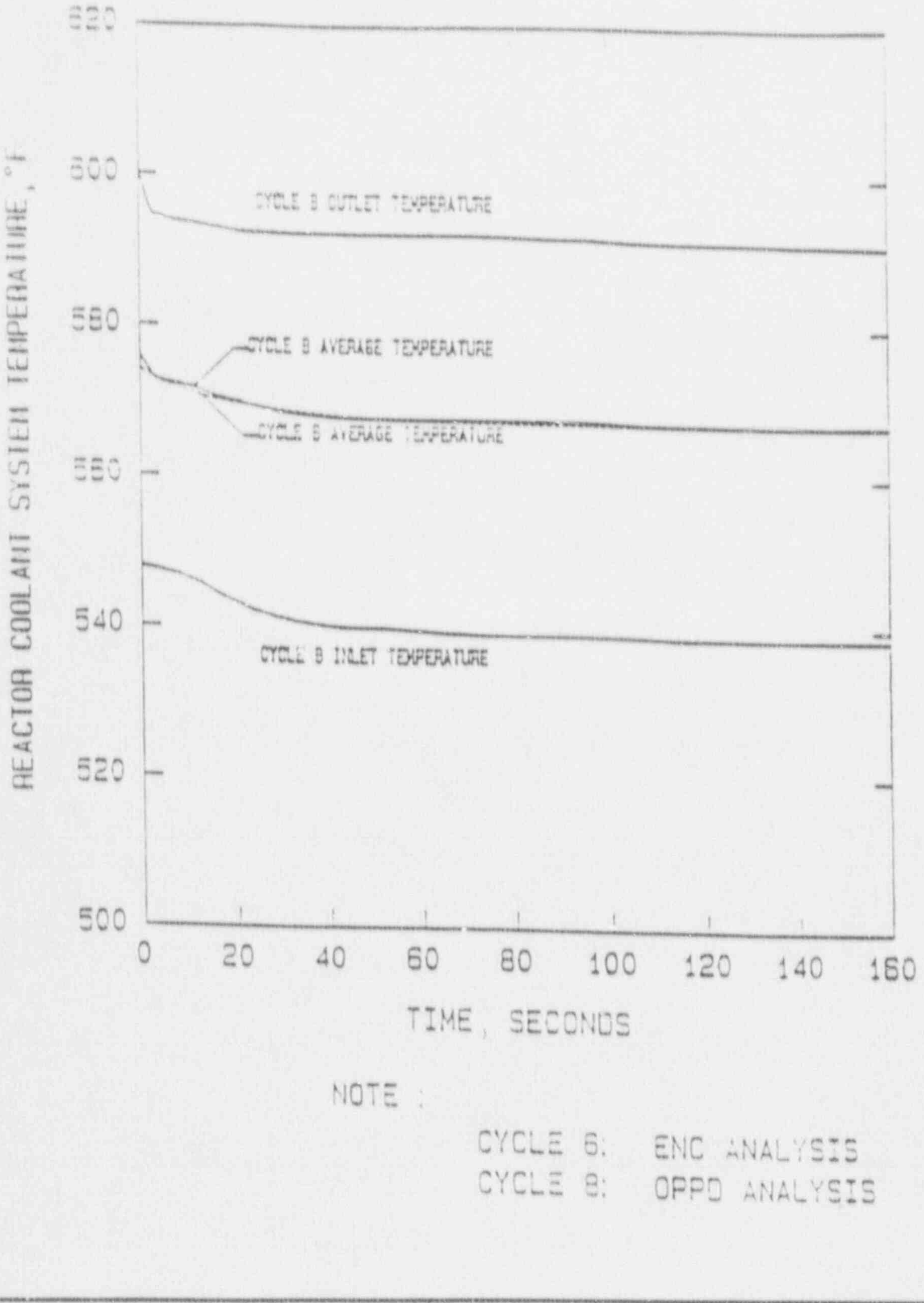
CEA Drop Incident Core Power vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-18



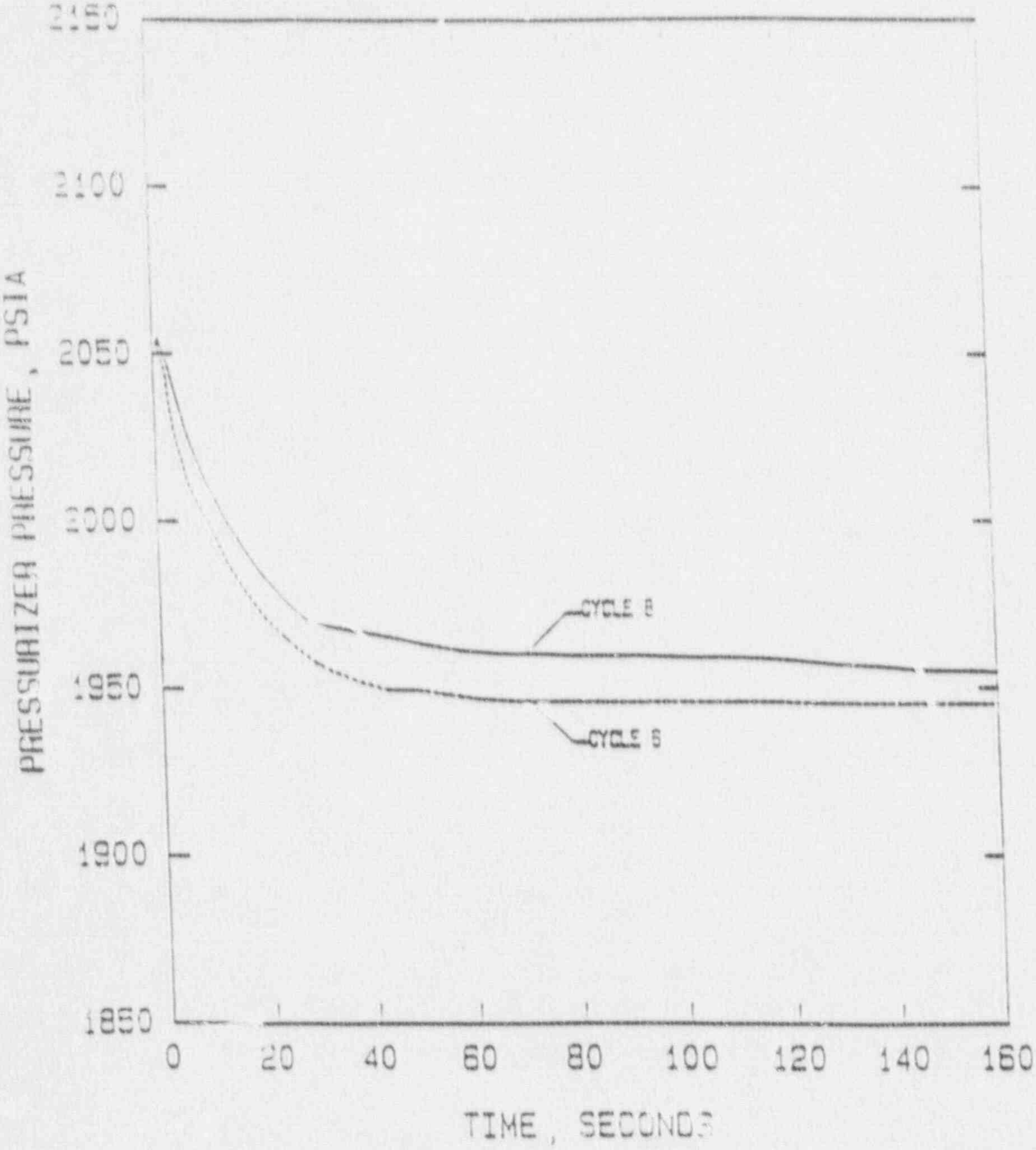
NOTE :

CYCLE 6: ENC ANALYSIS
CYCLE 8: OPPD ANALYSIS

CEA Drop Incident Core Average Heat Flux vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-19



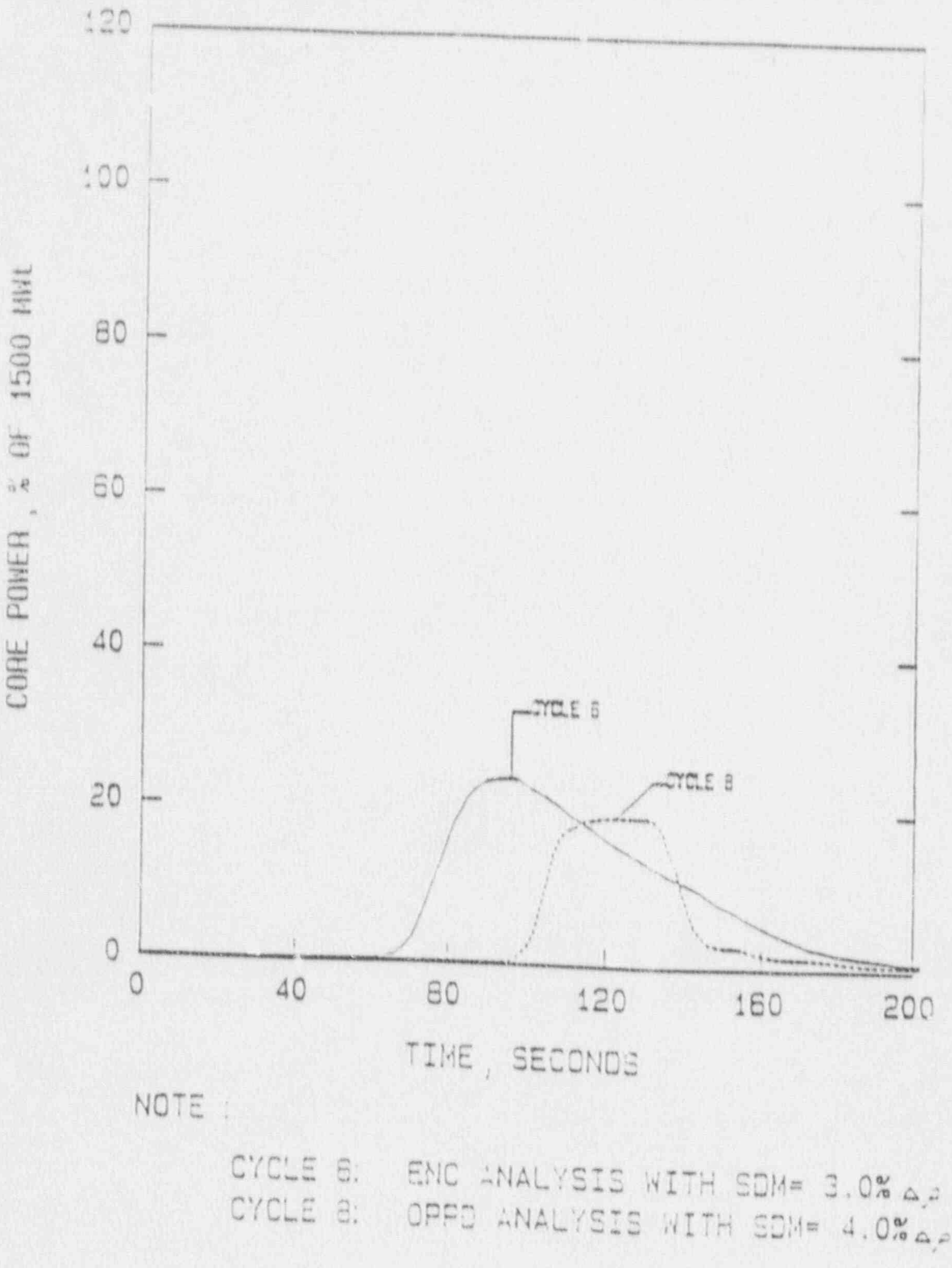
CEA Drop Incident Coolant Temperature vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-20



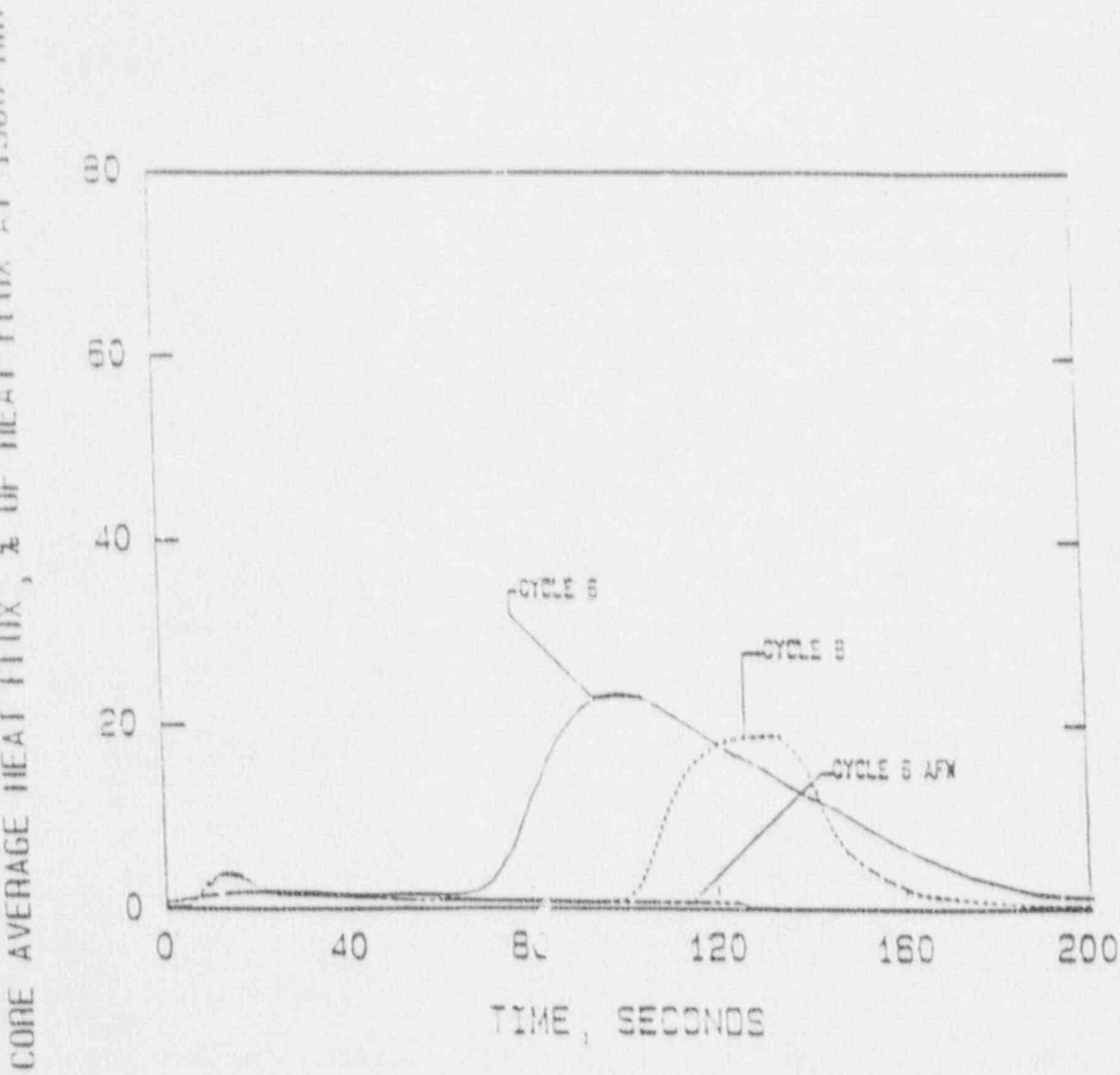
NOTE :

CYCLE 6: ENC ANALYSIS
CYCLE 8: OPPD ANALYSIS

CEA Drop Incident Pressurizer Pressure vs Time	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 6-21



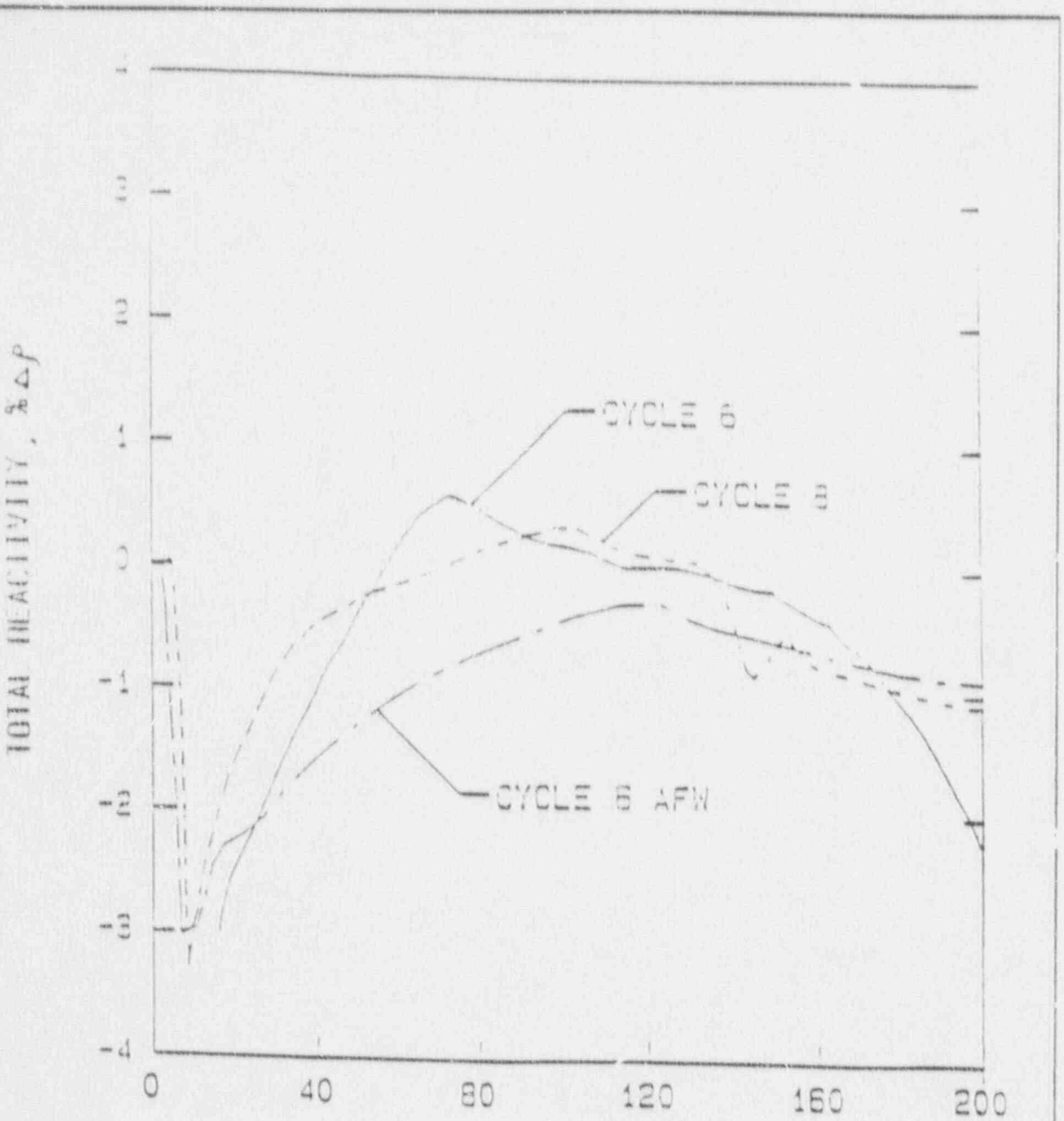
Zero Power MSLB Accident Core Power vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-22



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 MSLB Accident Core Average Heat Flux vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-23



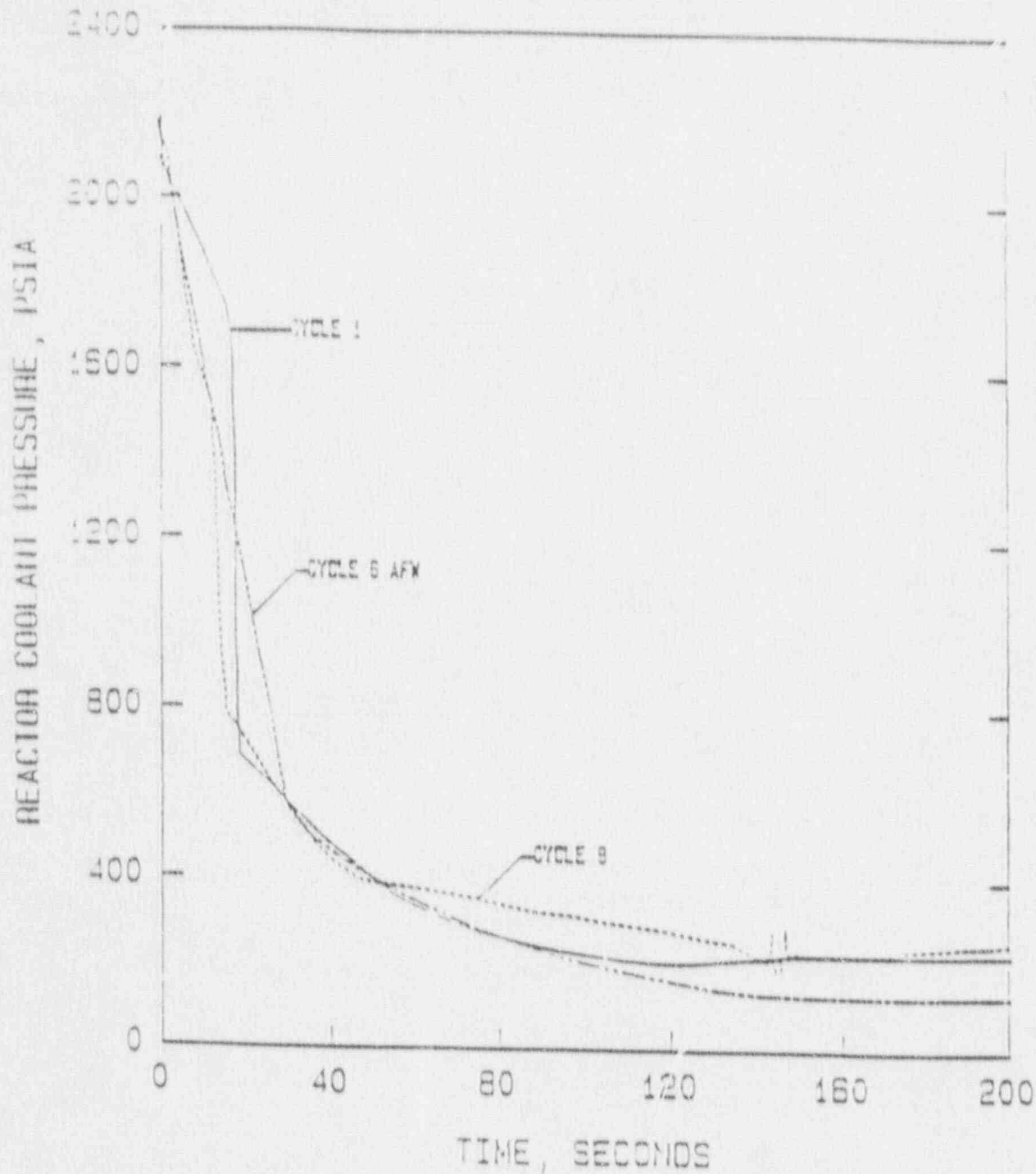
NOTE:

CYCLE 6: ENC ANALYSIS WITH SDM= 3.0% Δρ

CYCLE 6 AFW: CE ANALYSIS WITH SDM= 4.2% Δρ

CYCLE 8: OPPD ANALYSIS WITH SDM= 4.0% Δρ

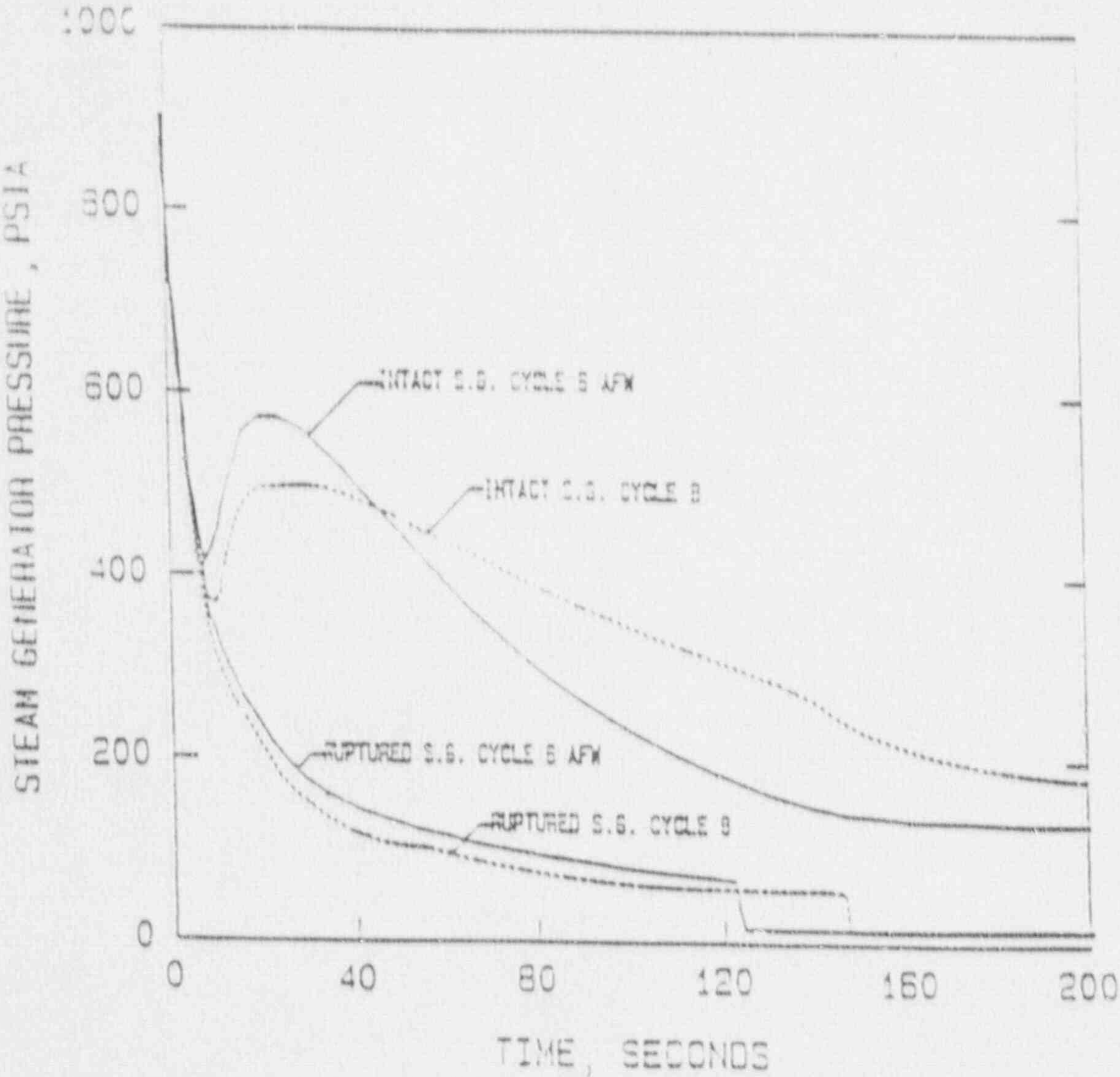
Zero Power MSLB Accident Total Reactivity vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-24



NOTE:

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

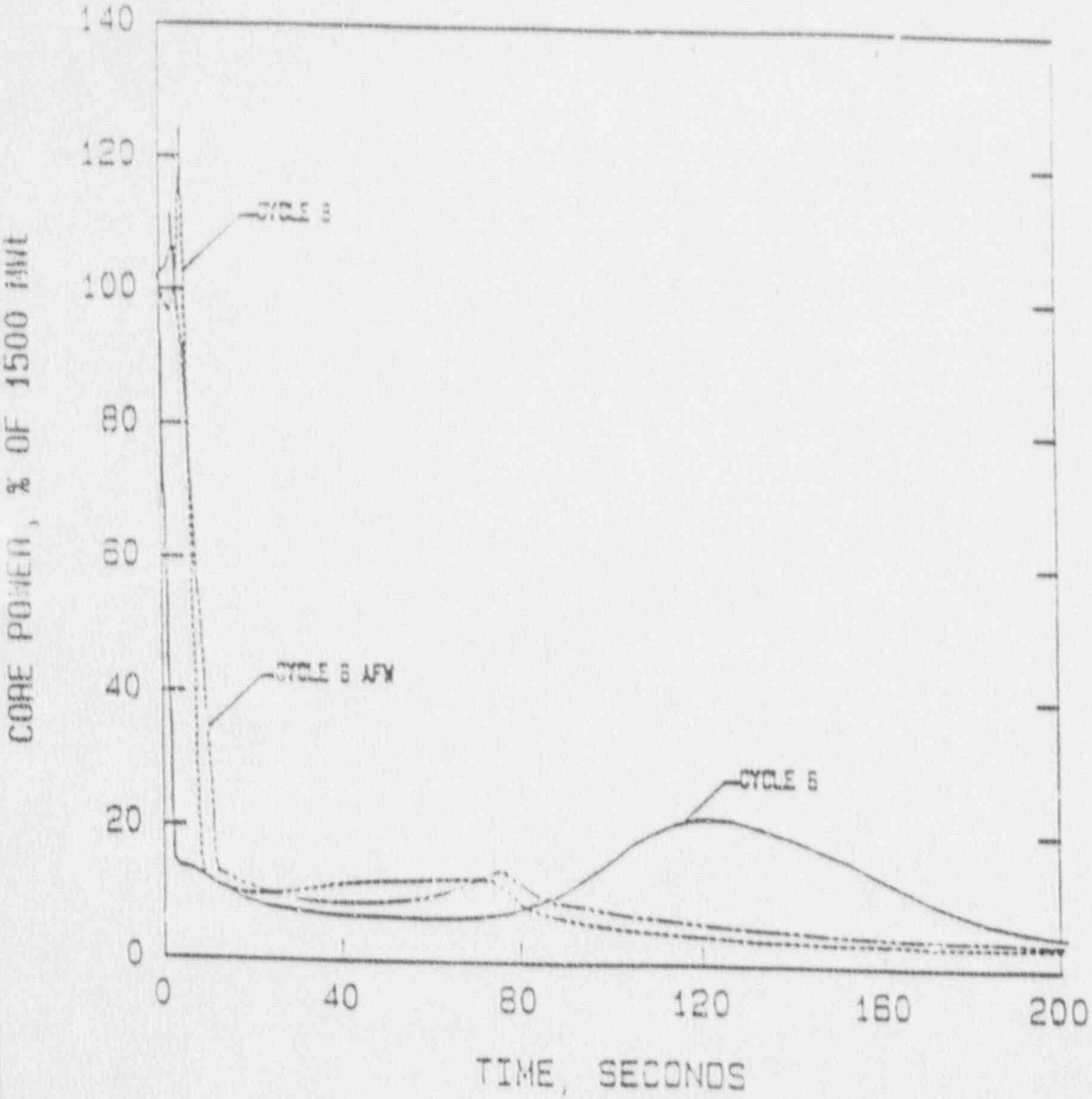
Zero Power MSLB Accident Coolant System Pressure vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-25



NOTE :

CYCLE 6 AFW: CE ANALYSIS
CYCLE 8: OPPD ANALYSIS

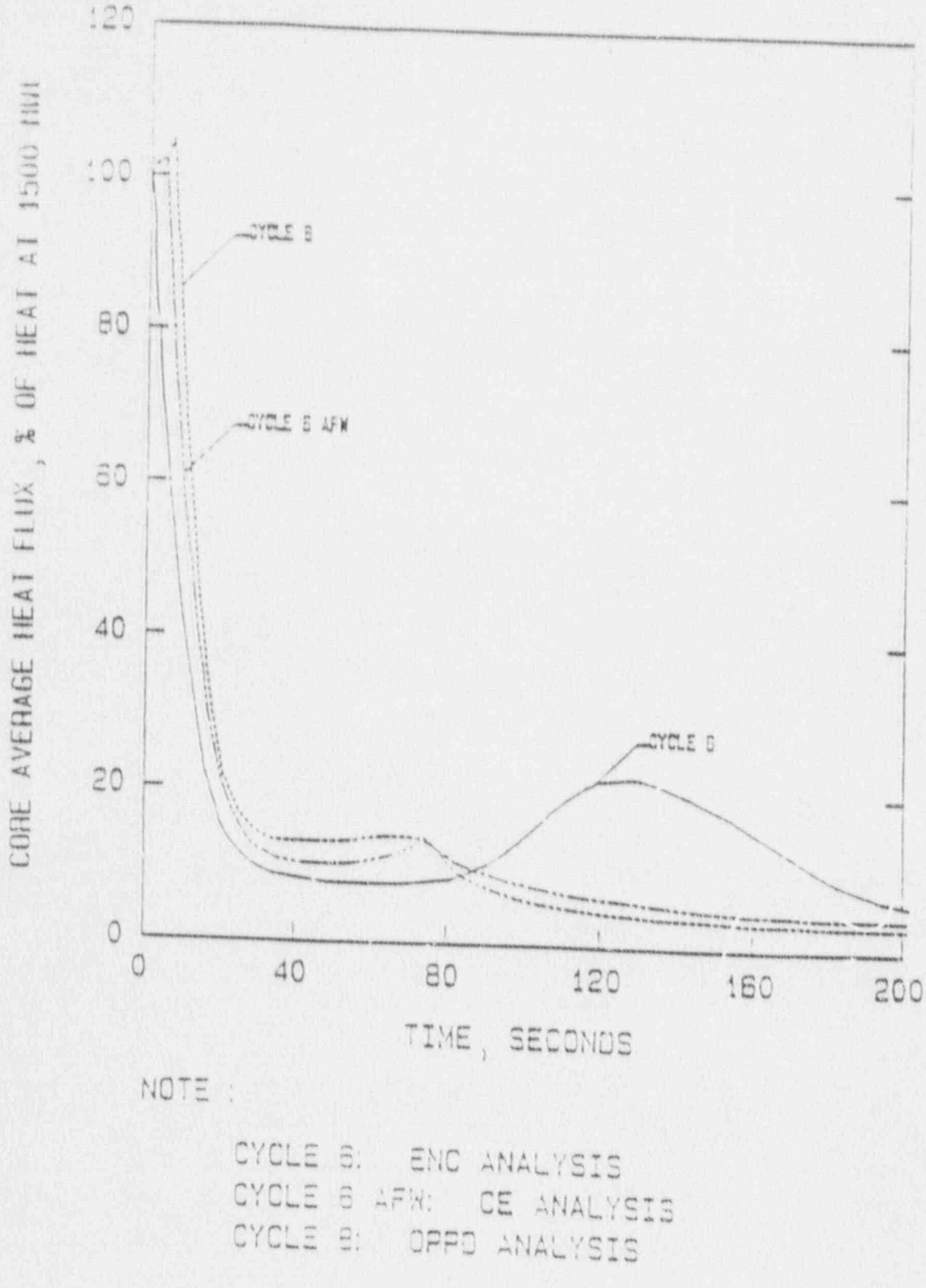
Zero Power MSLB Accident Steam Generator Pressure vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-26



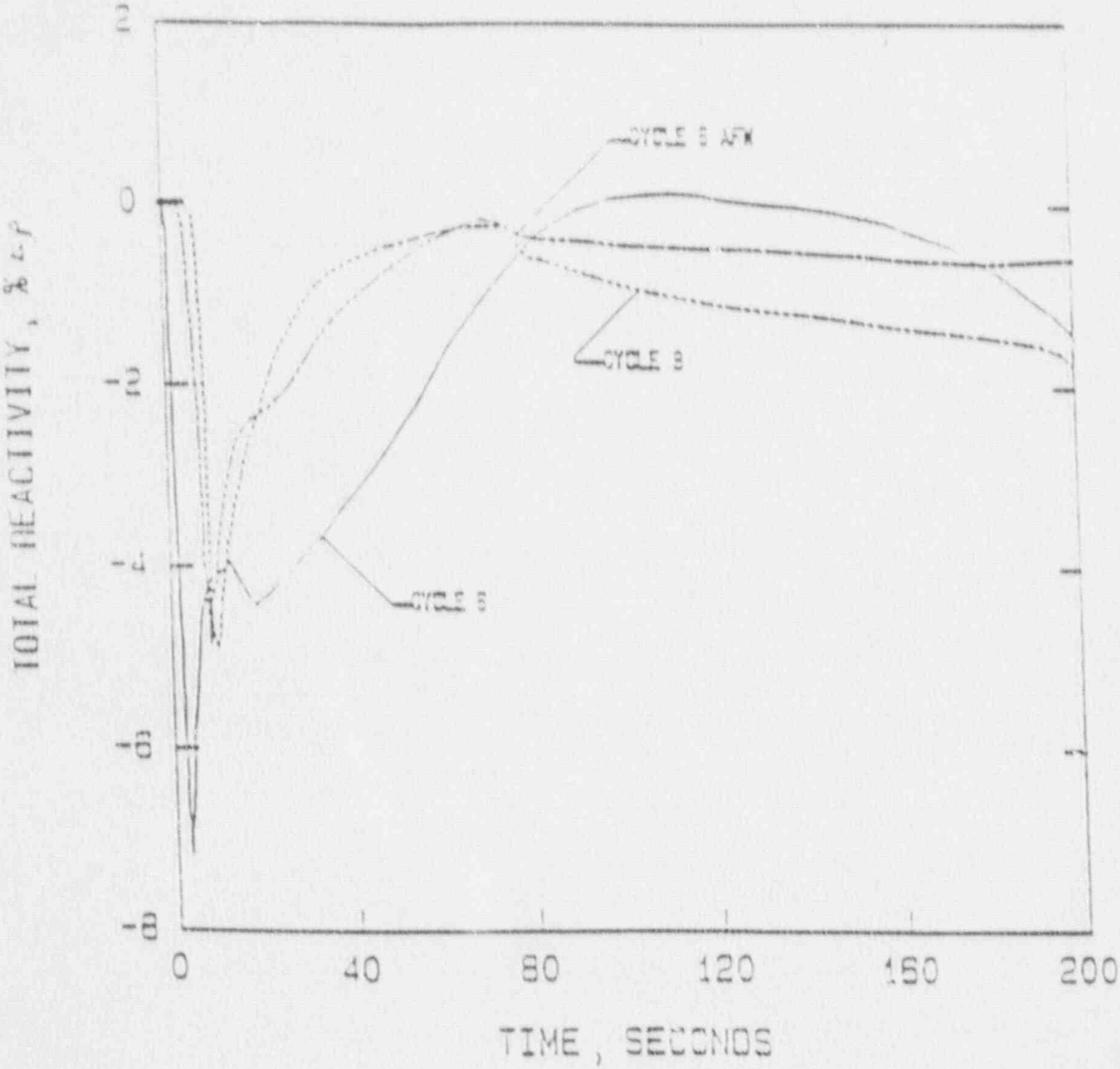
NOTE :

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 MSLB Accident Core Power vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-27



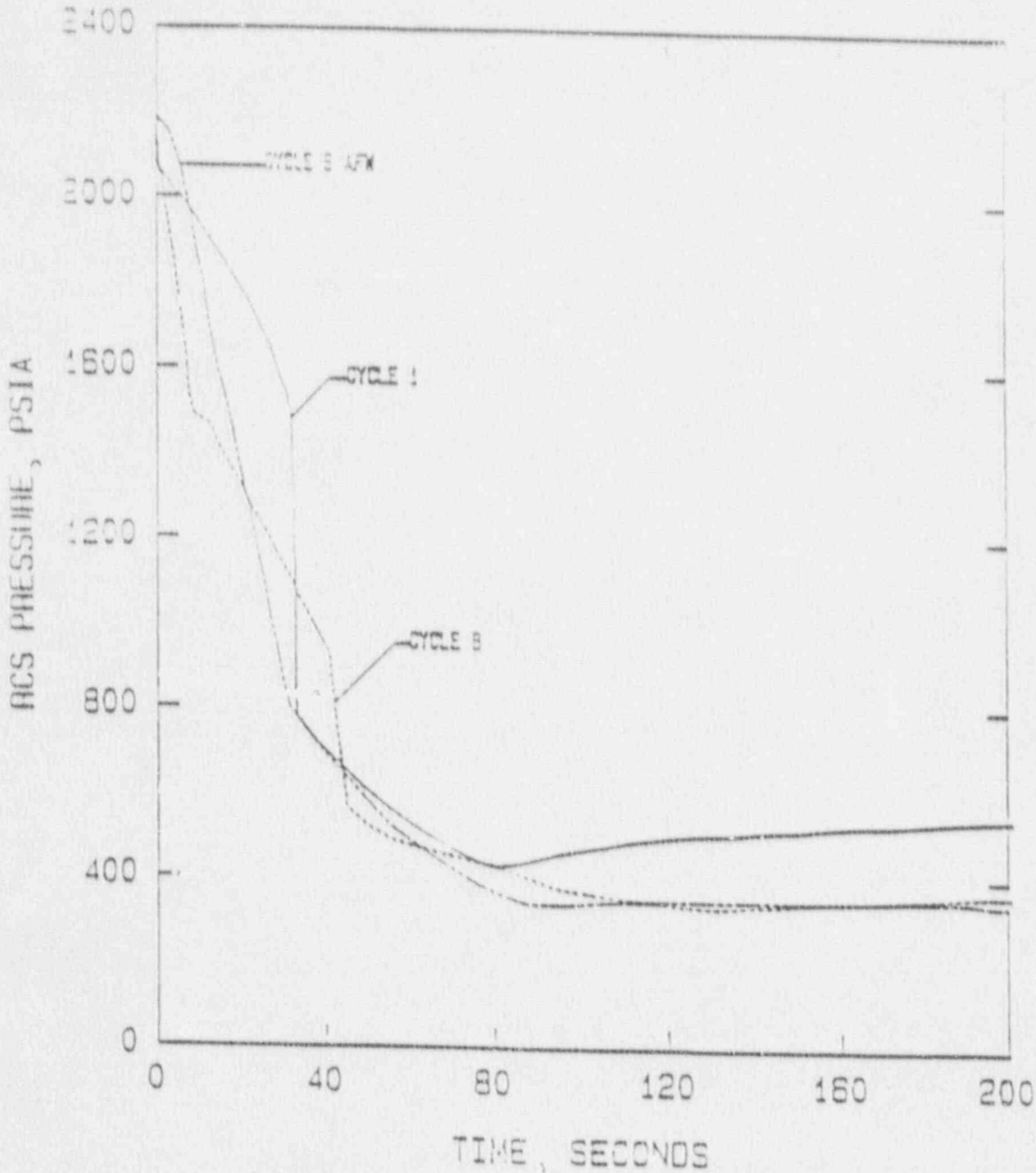
Full Power MSLB Accident Core Average Heat Flux vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-28



NOTE :

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

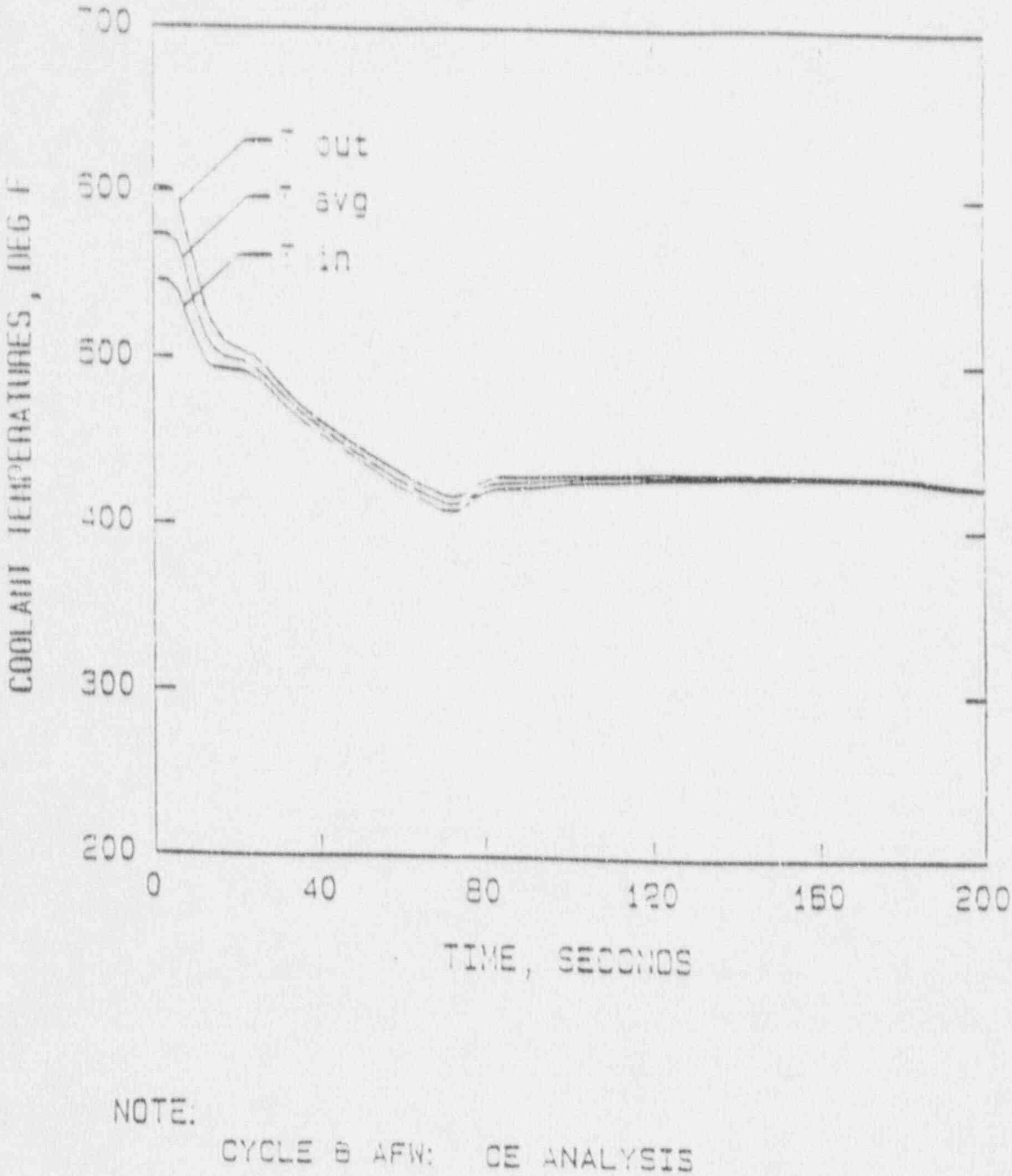
Full Power MSLB Accident Total Reactivity vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-29



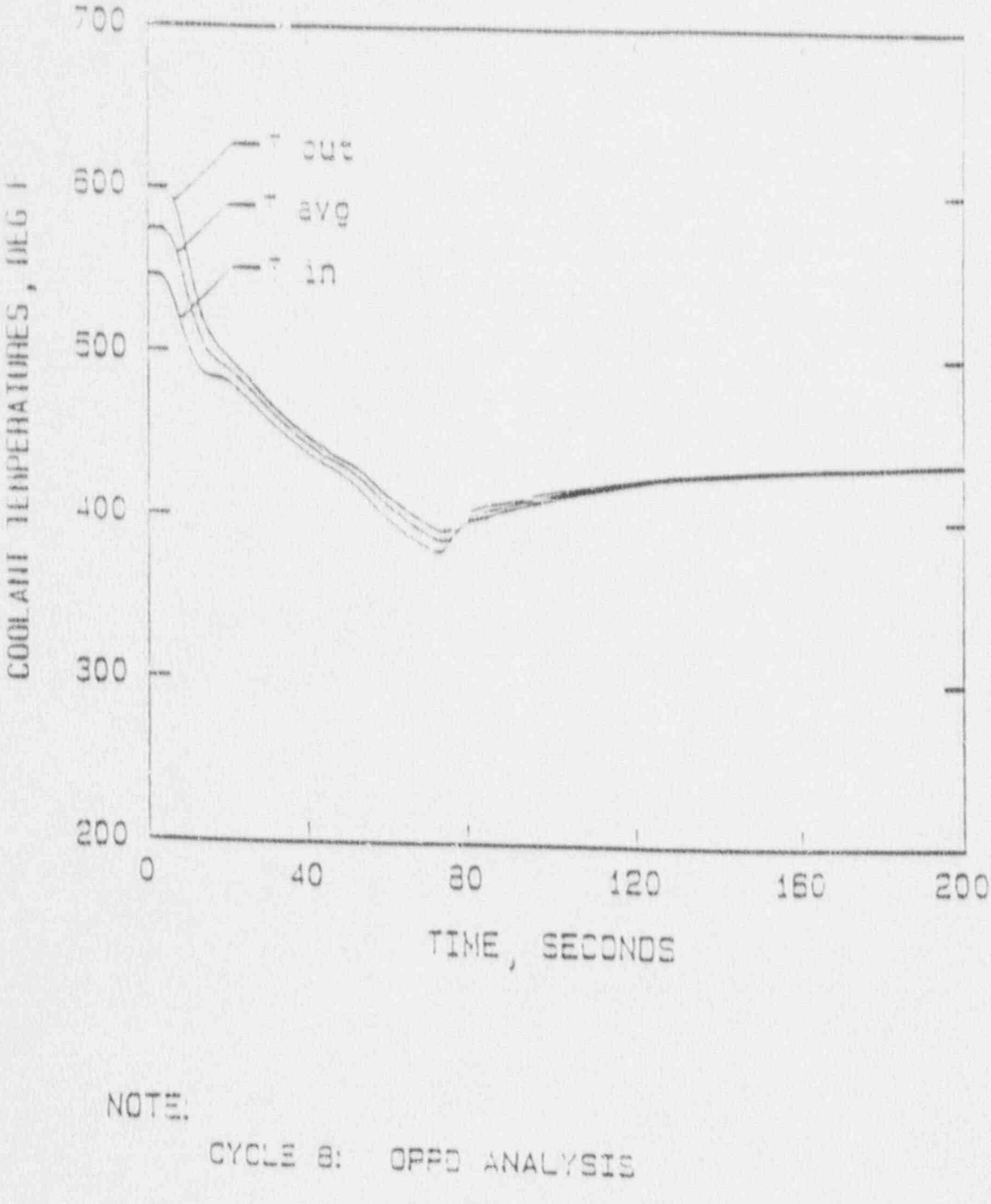
NOTE :

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

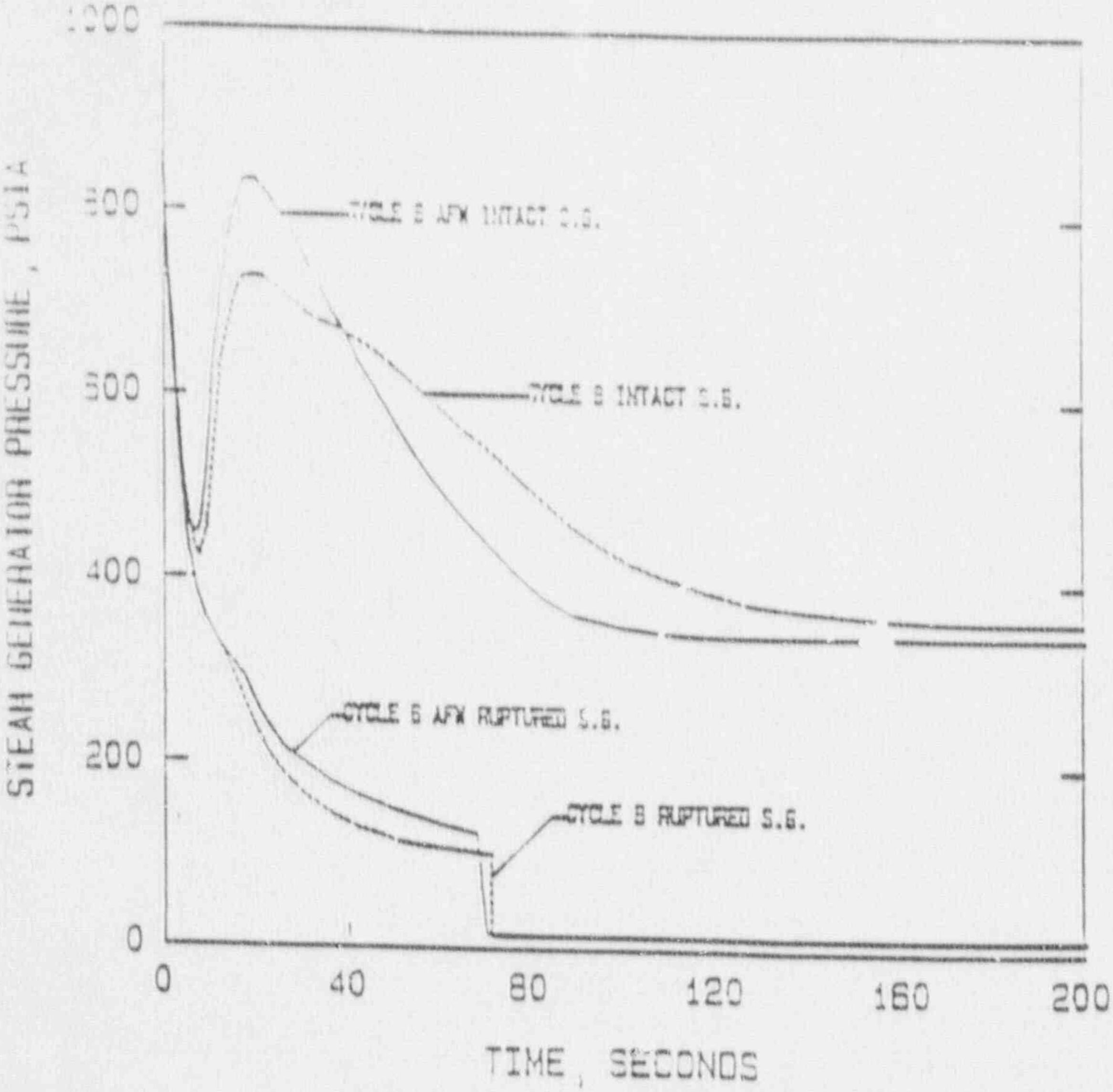
Full Power MSLB Accident Coolant System Pressure vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-30



Full Power MSLB Accident RCS Coolant Temperatures vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-31



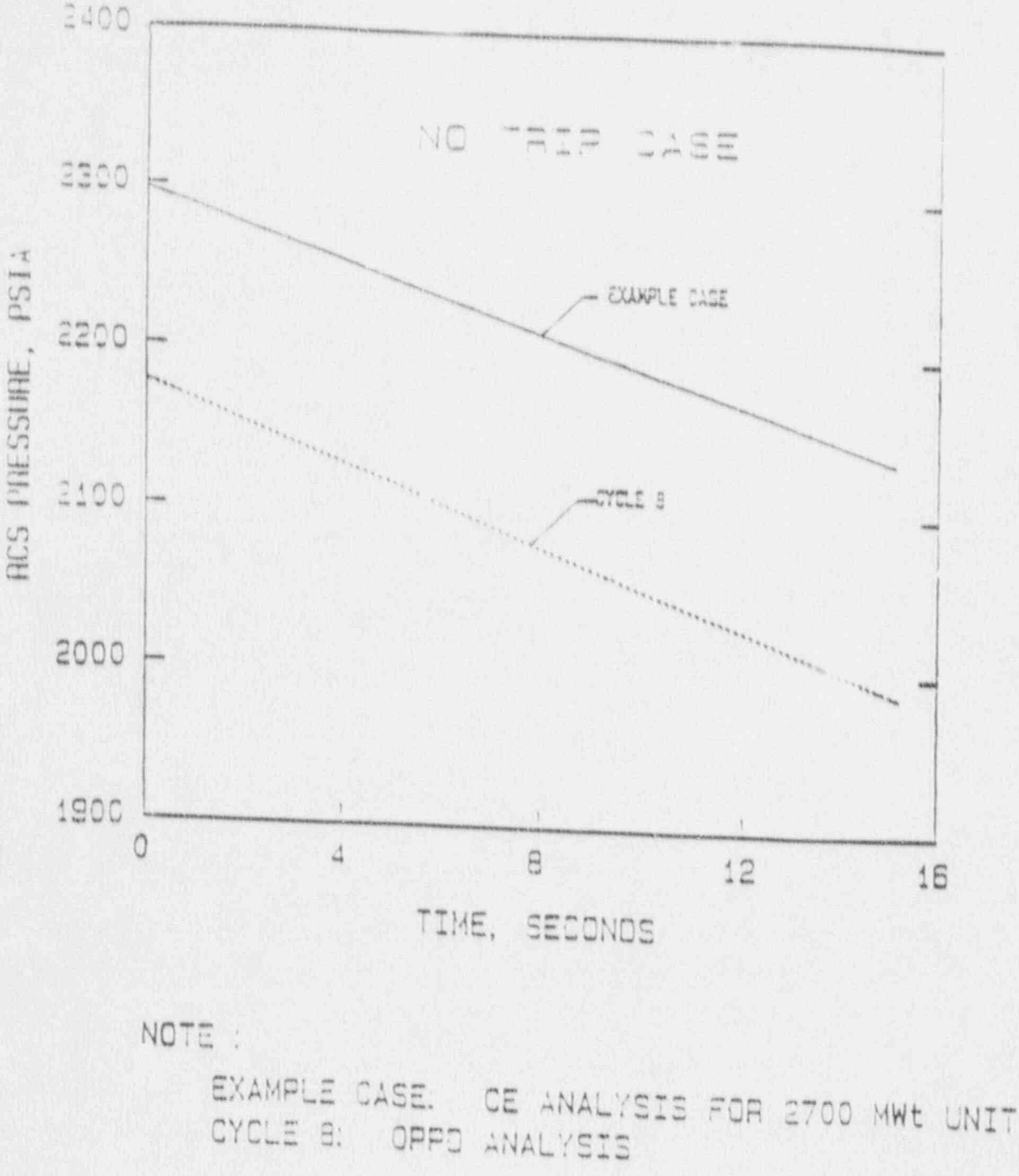
Full Power MSLB Accident RCS Coolant Temperatures vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-32



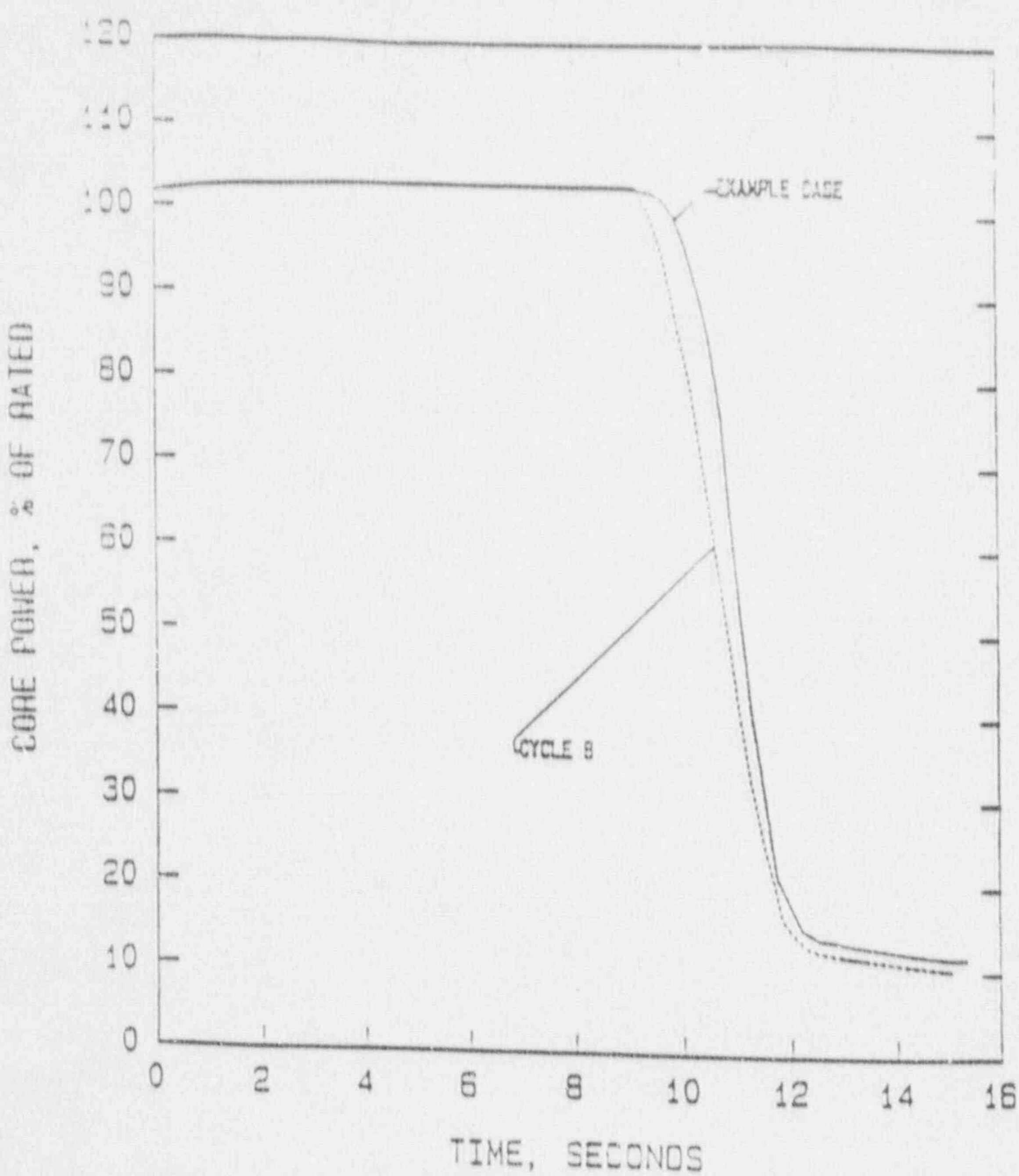
NOTE :

CYCLE 6 AFW: CE ANALYSIS
CYCLE 8: OPPD ANALYSIS

Full Power MSLB Accident Steam Generator Pressure vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-33



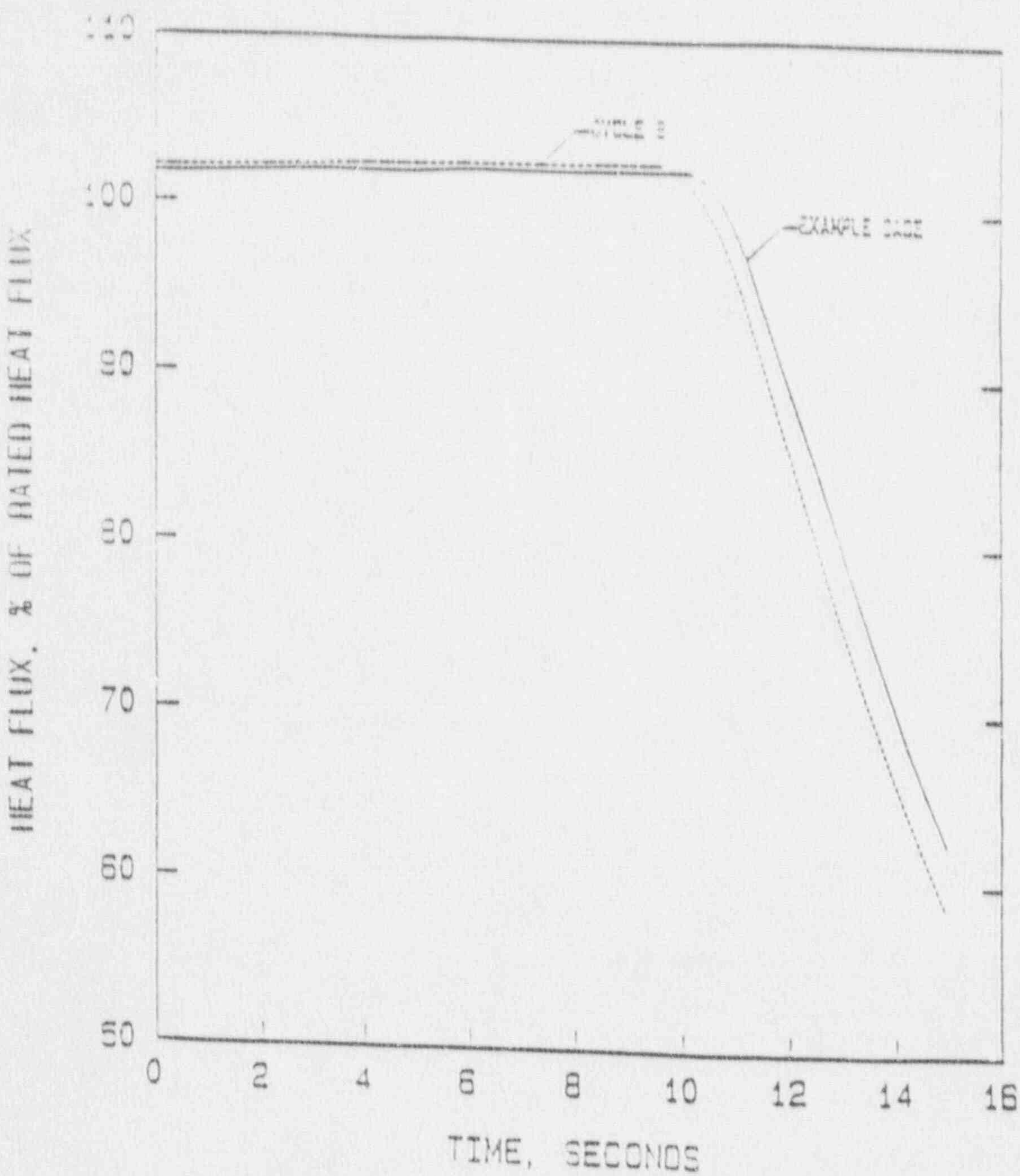
RCS Depressurization Incident RCS Pressure vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-34



NOTE :

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

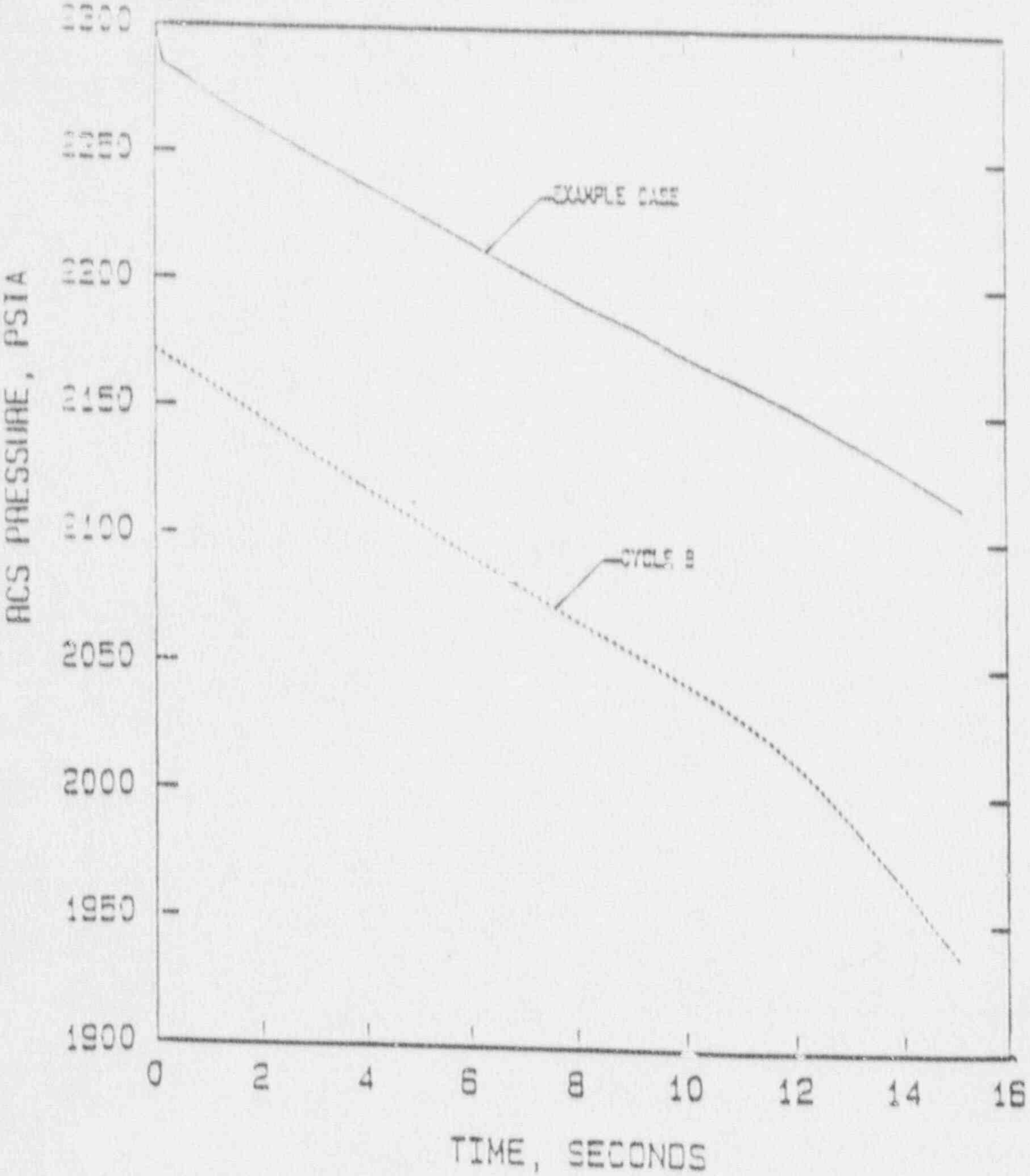
RCS Depressurization Incident Core Power vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-35



NOTE:

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

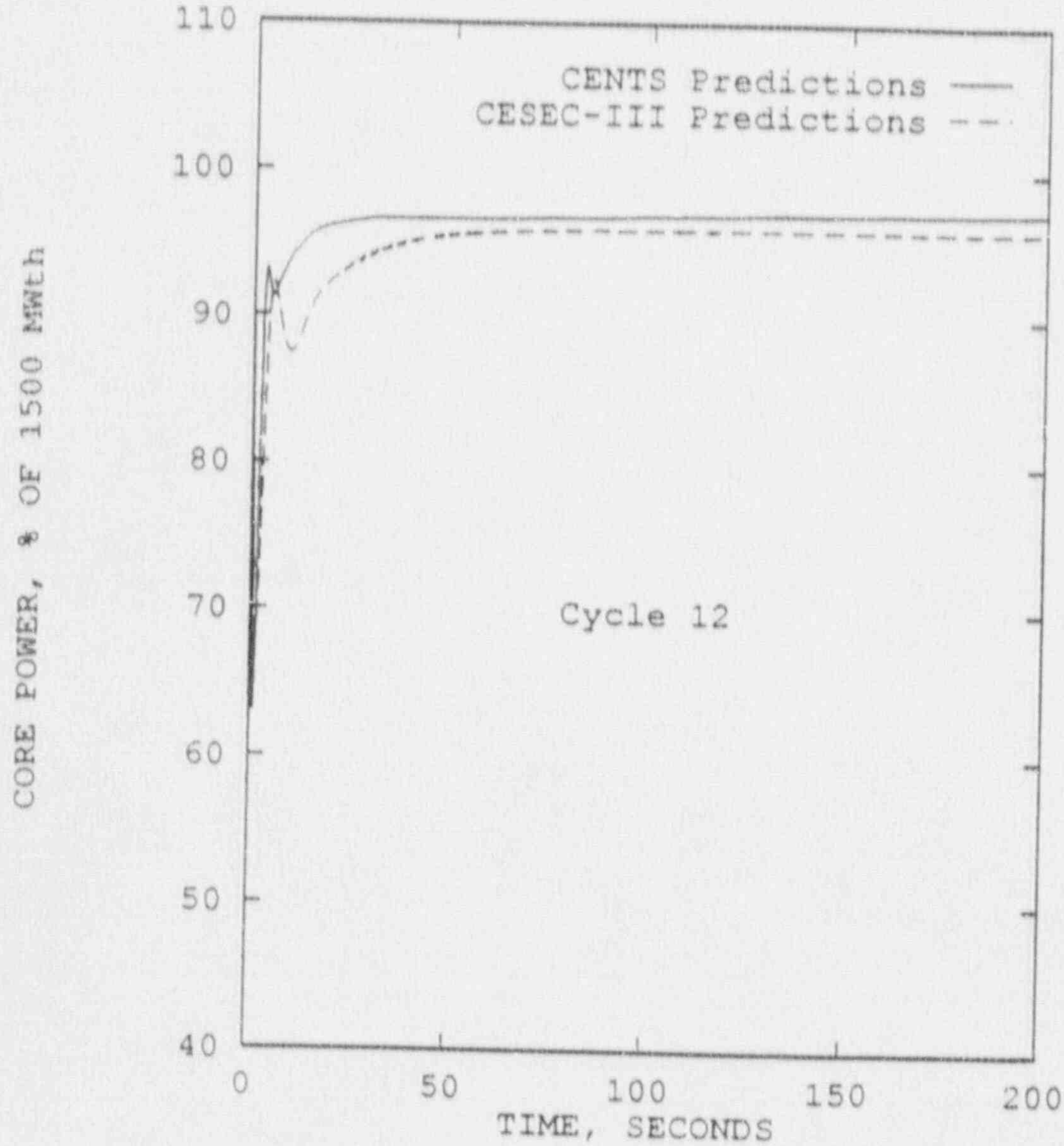
RCS Depressurization Incident Core Average Heat Flux vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-36



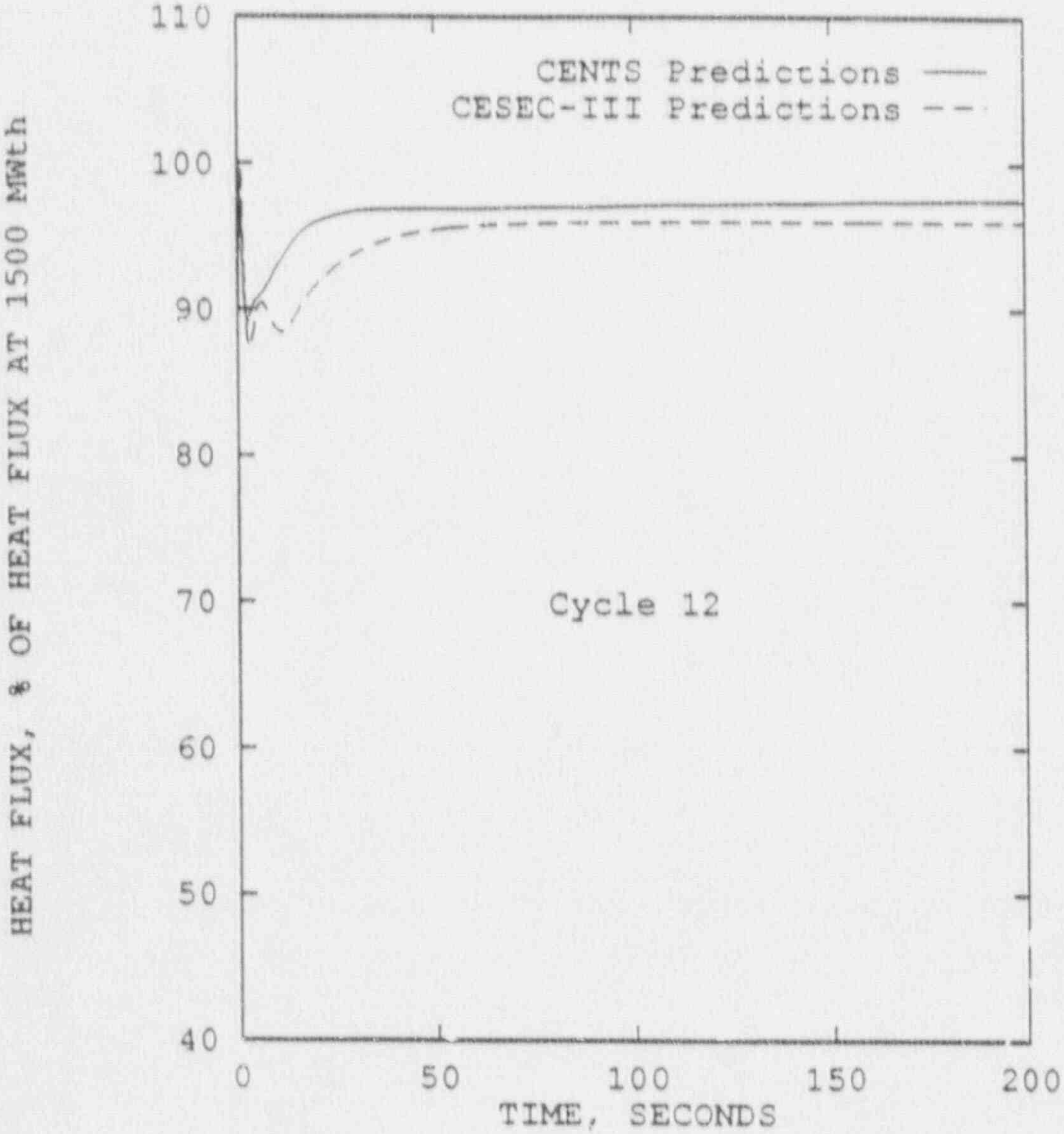
NOTE :

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

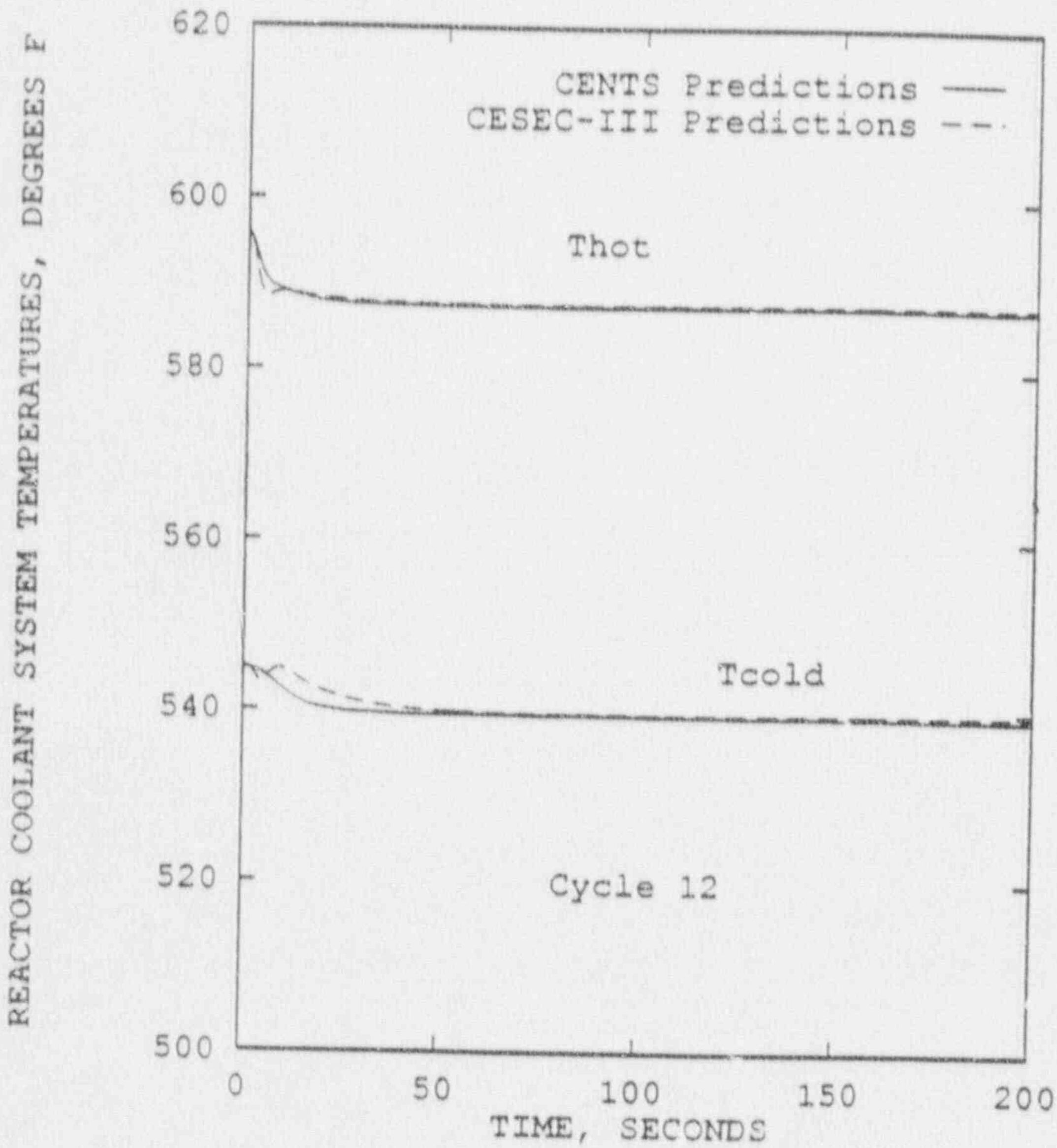
RCS Depressurization Incident RCS Pressure vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-37



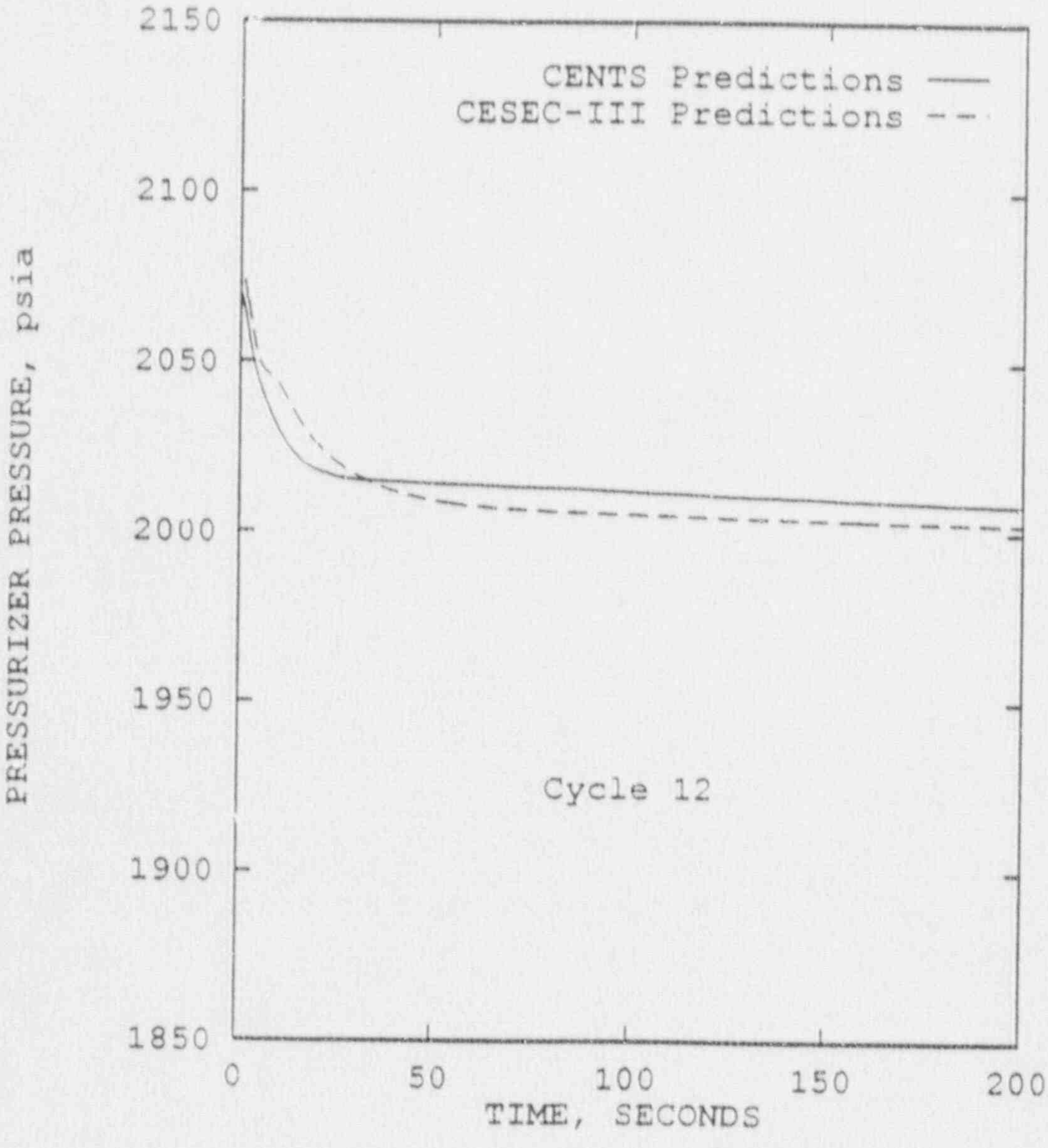
Dropped CEA Incident Core Power vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-38



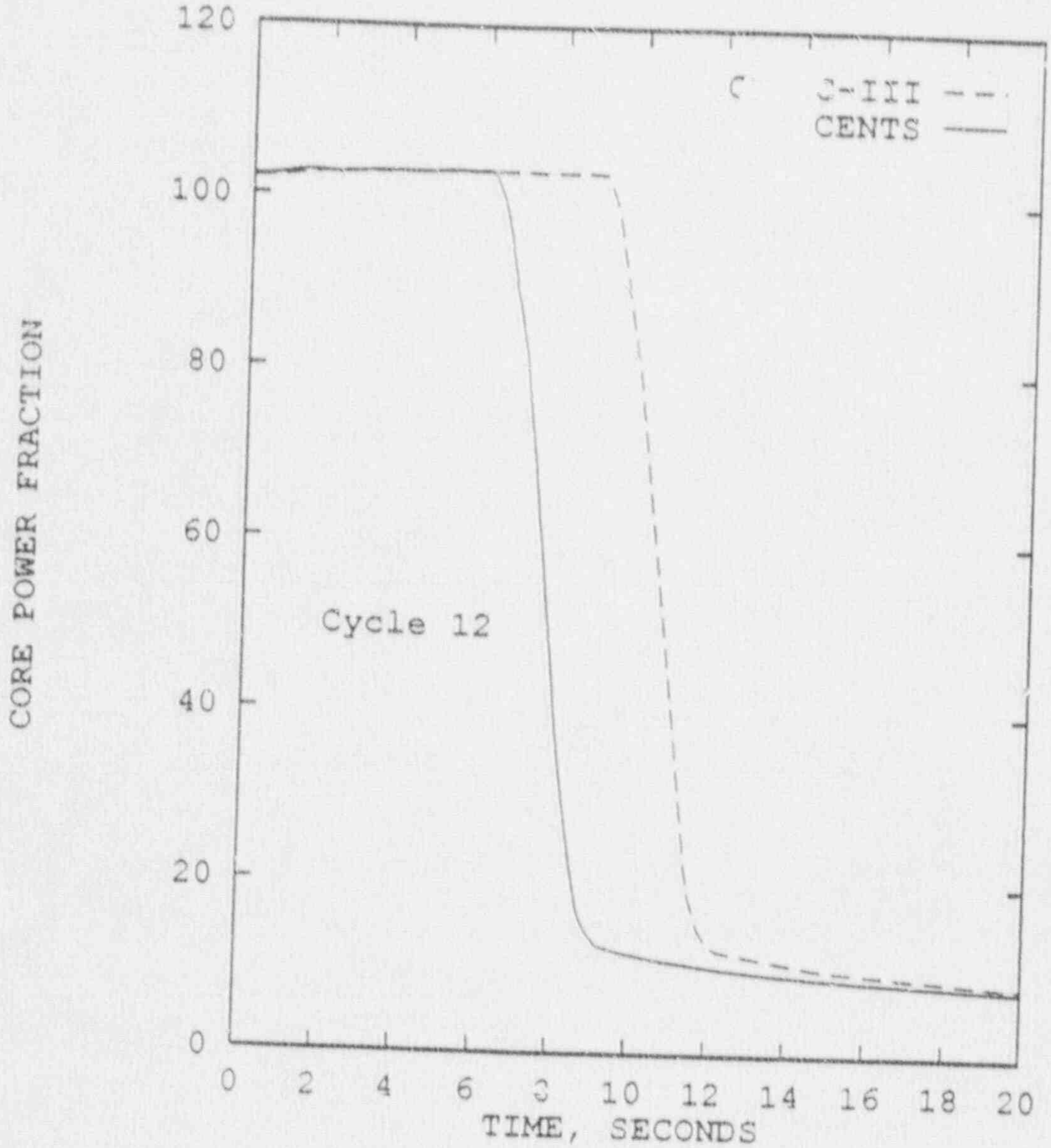
Dropped CEA Incident Core Average Heat Flux vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-39



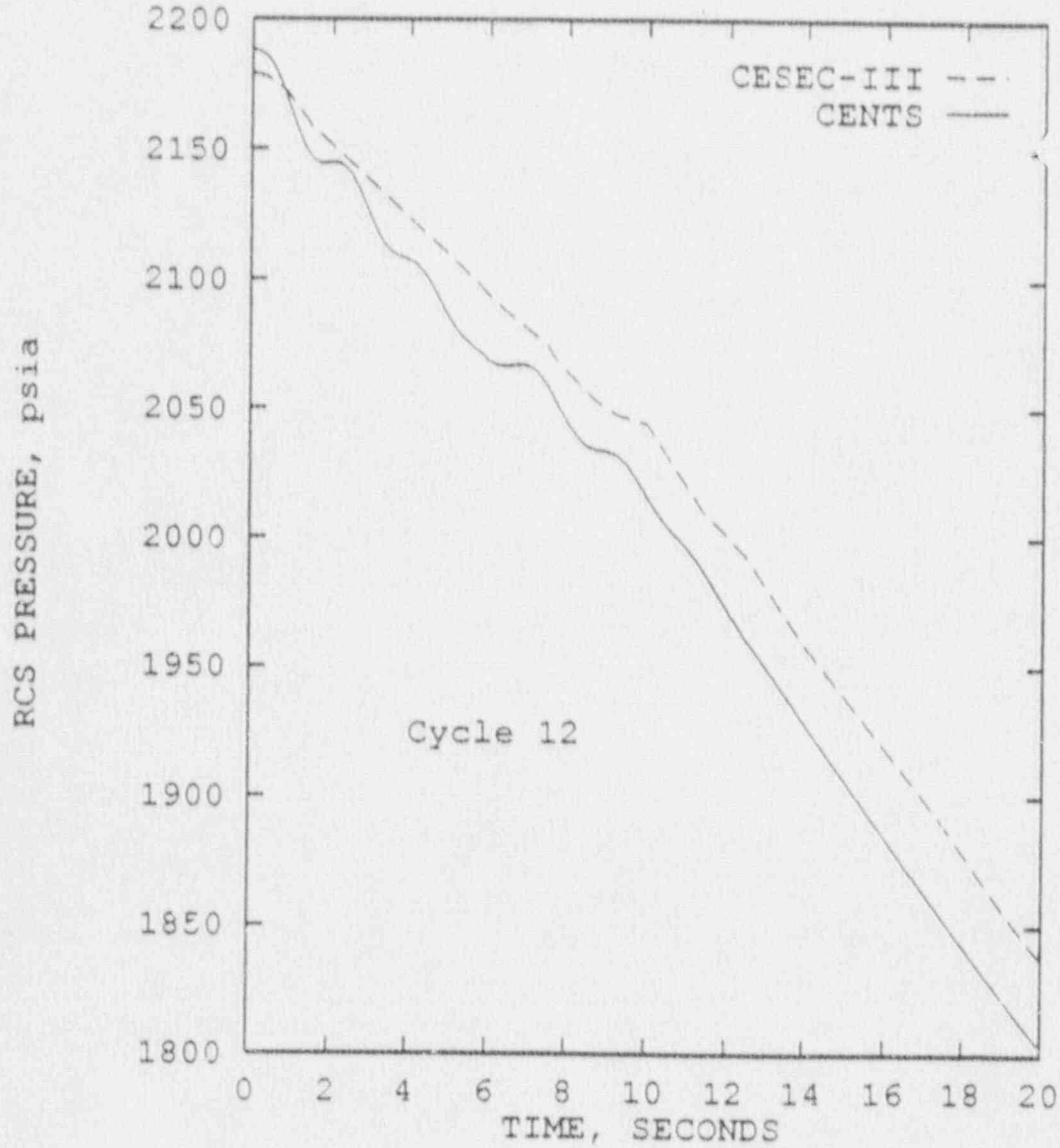
Dropped CEA Incident RCS Temperatures vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-40



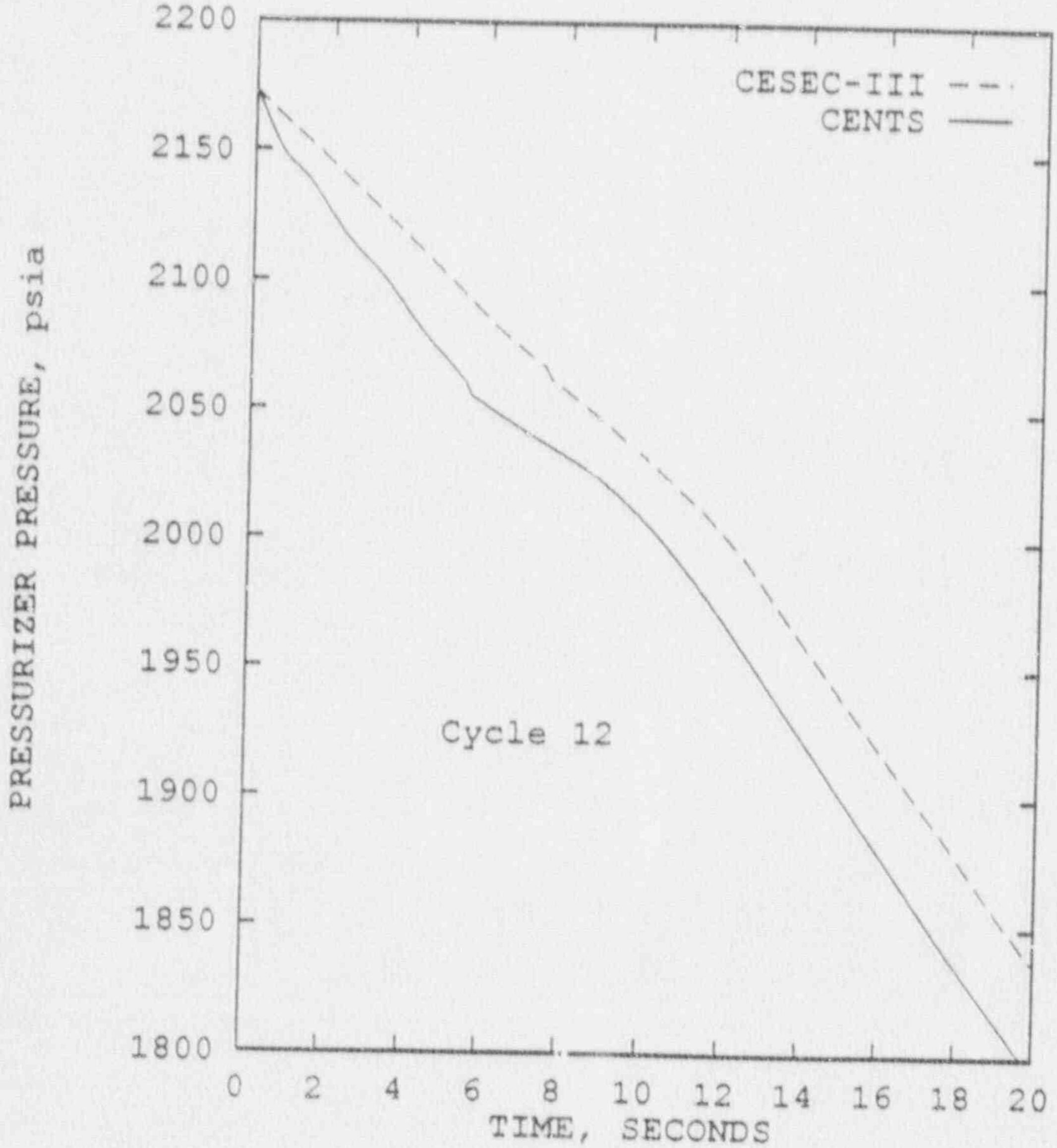
Dropped CEA Incident Pressurizer Pressure vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-41



RCS Depressurization Event Core Power vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-42



RCS Depressurization Event RCS Pressure vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-43



RCS Depressurization Event Pressurizer Pressure vs Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-44