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CENPD 282-NP-A

Volume 4

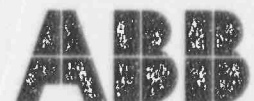
and

Responses to Questions

Technical Manual
for the
CENTS CODE

ABB Combustion Engineering Nuclear Fuel

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CENTS Code Verification for the Non-LOCA Analysis of the Westinghouse-Type Plants

December, 1992

ABB-CE

**E.J. Schulz
F.D. Lawrence**

KAERI

**H.Y. Yoon
W.K. In
H.C. Kim
W.J. Lee
G.S. Auh
C.C. Lee**

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UNITED STATES
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

February 24, 1995

Mr. S. A. Toelle
Manager, Nuclear Licensing
ABB Combustion Engineering
1000 Prospect Hill Road
Windsor, CT 06095-0500

Dear Mr. Toelle:

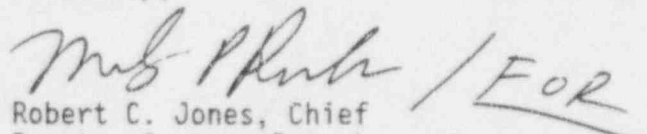
SUBJECT: ACCEPTANCE FOR REFERENCING OF LICENSING TOPICAL REPORT CE-NPD 282-P,
VOL. 4, "TECHNICAL MANUAL FOR THE CENTS CODE" (TAC NO. M85911)

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed topical report CE-NPD 282-P, Vol. 4, submitted by ABB Combustion Engineering by letter dated February 17, 1993. The report is acceptable for referencing in license applications to the extent specified and under the limitations stated in the enclosed NRC safety evaluation (SE) and the Brookhaven National Laboratory (BNL) report attached to the SE. The BNL report also includes the previous evaluation of Volumes 1-3 for completeness.

The staff will not repeat its review of the matters described in CE-NPD 282-P and found acceptable when the report appears as a reference in license applications, except to ensure that the material presented applies to the specific plant involved. NRC acceptance applies only to the matters described in the report. In accordance with procedures established in NUREG-0390, the NRC requests that ABB Combustion Engineering publish accepted versions of the report, proprietary and nonproprietary, within 3 months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed SE between the title page and the abstract and an -A (designating accepted) should follow the report identification symbol.

If the NRC's criteria or regulations change so that its conclusions concerning the acceptability of the report are invalidated, ABB Combustion Engineering and/or the applicants referencing the topical report will be expected to revise and resubmit their respective documentation, or submit justification for the continued applicability of the topical report without revision of their respective documentation.

Sincerely,


Robert C. Jones, Chief
Reactor Systems Branch
Division of Systems Safety and Analysis

Enclosure:
CE-NPD 282-P, Vol. 4, Safety Evaluation



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENCLOSURE

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATING TO TOPICAL REPORT CE-NPD 282-P, VOL. 4
"TECHNICAL MANUAL FOR THE CENTS CODE"
ABB COMBUSTION ENGINEERING

1. INTRODUCTION

In a letter of February 17, 1993 (Ref. 1), from S. A. Toelle to Document Control Desk (NRC), ABB Combustion Engineering (CE) submitted topical report CE-NPD 282-P, Vol. 4, "Technical Manual for the CENTS Code," for NRC review. The topical report presents comparisons of CENTS predictions with plant transient data and with predictions made with the RELAP5/MOD3 code for a Westinghouse plant. The staff had previously approved the CENTS code for use by CE to calculate the transient behavior of a CE-designed pressurized water reactor (PWR) under normal and abnormal conditions including accidents (Ref. 2).

The NRC staff was supported in this review by its consultant, Brookhaven National Laboratory. The staff has reviewed and adopted the findings recommended in the consultant's technical evaluation report (TER), which is attached (Ref. 3).

2. EVALUATION

The attached TER provides the evaluation for Volume 4. The previous evaluation of Volumes 1, 2 and 3 are also included for completeness.

3. CONCLUSIONS

The staff has reviewed topical report CE-NPD 282-P, Vol. 4, and the supporting documentation sent in response to its request for additional information (Ref. 4). On the basis of this review, the staff concludes that CE-NPD 282-P, Vol. 4, is

acceptable for referencing in licensing actions with respect to the calculation of non-LOCA (loss-of-coolant-accident) transient behavior in PWRs designed by Westinghouse, subject to the limitations stated in the attached TER.

4. REFERENCES

- (1) Letter from S. A. Toelle (CE) to Document Control Desk (NRC), "Submittal of Volume 4 of CENTS Code Topical Report," February 17, 1993.
- (2) Letter from M. J. Virgilio (NRC) to S. A. Toelle (CE), "Acceptance for Referencing of Licensing Topical Report CE-NPD 282-P, "Technical Manual for the CENTS Code," March 17, 1994.
- (3) Letter from J. F. Carew (BNL) to L. I. Kopp (NRC), "Review of Volume-4 of the ABB-CE CENTS Transient Analysis Methodology Topical Report CE-NPD-282-P (FIN A-2589/Task-8/TAC-M85911)," December 30, 1994.
- (4) Letter from J. F. Carew (BNL) to L. I. Kopp (NRC), "Review of Volume-4 of the ABB-CE CENTS Transient Analysis Methodology Topical Report CE-NPD-282-P (FIN A-2589/Task-8/TAC-M85911)," June 30, 1994.

TECHNICAL EVALUATION REPORT

Topical Report Title: Technical Manual for the CENTS Code
Topical Report Number: CE-NPD 282-P, Volumes 1-4
Report Issue Date: Volumes 1-3, October 1991
Volume 4, December 1992
Originating Organization: ABB Combustion Engineering Nuclear Power

1.0 INTRODUCTION

ABB Combustion Engineering (ABB-CE) and the Omaha Public Power District (OPPD) have submitted in References-1 and 2 the four volumes of the CENTS Technical Manual Topical Report CE-NPD 282-P for NRC review and approval. CENTS is a new ABB-CE computer code for the simulation of PWR transient behavior under normal and abnormal conditions. CENTS provides an interactive capability for simulating the standard NSSS components, and may be used to determine the transient thermal-hydraulic conditions in the primary and secondary systems and the transient core power. CENTS is intended for prediction of plant behavior for conditions ranging from normal plant operation to operational and licensing transients.

CENTS is a best-estimate code designed to provide a realistic simulation of the neutronics, thermal-hydraulics and plant systems response during transient conditions. The CENTS models are based on PWR design codes. The primary system models are based on the ABB-CE design version of CEFLASH-4AS (Reference-3) and the secondary system models are based on the Long Term Cooling (LTC) computer code (Reference-4). The point reactor kinetics and decay heat models are taken from FLASH (Reference-5).

The CENTS modeling of the reactor core, primary and secondary systems, and control systems is presented in Volume-1. A detailed description of the CENTS input is given in Volume-2. Comparisons of CENTS to startup measurements, operational transient measurements and to the CESEC (Reference-6) and CEFLASH-4AS codes are presented in Volumes 2 and 3 for ABB-CE type plants. Similar comparisons are presented in Volume-4 as validation for CENTS application to Westinghouse plants.

The purpose of this review was to evaluate the acceptability of CENTS for performing PWR licensing analyses. This involved the evaluation of both the CENTS methodology and the completeness and accuracy of the CENTS benchmarking. The CENTS methodology and benchmarking are summarized in Section-2, and the evaluation of the important technical issues raised during this review is presented in Section-3. The technical position is given in Section-4.

2.0 SUMMARY OF THE TOPICAL REPORT

2.1 CENTS Methodology

2.1.1 Reactor Core

In the CENTS methodology, the transient core power is determined by either a point kinetics or three-dimensional neutronics model. In the point kinetics model a user-specified axial power distribution is employed. The kinetics and decay heat calculation used in the point kinetics model is based on the treatment of FLASH (Reference-5) using six delayed groups. The fission decay power calculation uses eleven fission product groups together with fission product decay constants and yields, assuming steady-state operation at the initial power level. The decay

heat is calculated using the ANS/ANSI-5.1 decay heat curve, including the fission product capture contribution and without uncertainties.

The core heat transfer to the coolant is determined using a finite difference form of the heat transfer equation in cylindrical geometry for a finite axial height including the fuel pellet, gap, clad and coolant. Temperature dependent conduction and heat capacities are used. Heat transfer coefficients are provided for both forced convection and pool boiling, and from subcooled through critical heat flux conditions. The coolant axial temperature distribution is determined using a closed channel heat balance.

2.1.2 Primary System

The CENTS model employs a flow path network together with a control volume representation of the primary system components to solve the fluid conservation equations. The primary system representation includes models for the inner-vessel, upper head, hot-leg, pressurizer, steam generator (3 nodes), coolant pumps, cold-leg, and the annulus and lower plenum. Each Reactor Coolant System (RCS) loop is modeled explicitly. The thermal-hydraulic model solves a one-dimensional conservation equation for each of the following: liquid mass, mixture mass, mixture energy, steam enthalpy and mixture momentum. The numerical solution is carried out using a linearized discretized form of these equations together with an iterative calculation of the nodal pressure and enthalpy.

The heat rate includes contributions from the fuel, pumps, steam generator, pressurizer heater and control element assembly (CEA). Correlations are provided for condensation and

critical flow. The primary system wall temperatures are calculated by integrating the nodal heat conduction equations.

2.1.3 Secondary System

The CENTS secondary system includes control volumes for the steam generator downcomer, evaporator/riser and steamdome, and an additional volume for the main steamline header. Safety valves, atmospheric dump valves, MSIVs, and steamline/feedline check valves are included. The Steam Generator (SG) model maintains the conservation equations using a node-flowpath representation. The flows between internal SG control volumes are determined using empirical correlations. The SG model also includes calculations of the nuclide concentrations, heat losses, and indicated level. CENTS provides a simplified SG model in which the feedwater flow specified in the control system is input directly to the SG, as well as a detailed model in which the feedwater flow network is represented.

2.1.4 Control Systems

The CENTS control system model simulates the operation of the reactor protection, control rod regulation, pressurizer level and pressure, feedwater, turbine and safety injection control systems. CENTS employs a set of generic modules to perform the standard arithmetic, integro-differential and logic transforms used in these control systems. The model provides an extensive set of Reactor Protection System (RPS) trips, including the power rate-of-change and the overpower and overtemperature ΔT trips. The pressurizer level control system model determines an error signal based on the programmed level and adjusts the charging/letdown flow

rates and/or the pressurizer heaters. The turbine control model typically generates a turbine trip with a core trip, feedwater trip, loss of load, or high SG level. The control rod regulating model may be used to maintain criticality, satisfy power demand, and trip the reactor. A typical control rod regulator determines an error signal from the core average and reference temperatures, and the core power and turbine demand mismatch. The control rod speed and reactivity worth are input.

2.2 CENTS Benchmarking

In order to qualify the CENTS coding and models for PWR transient analysis, ABB-CE has compared the CENTS predictions to startup measurements, operating transients and to calculations made with the ABB-CE CESEC and CEFLASH-4AS for ABB-CE type plants, and to RELAP5/MOD3 (Reference-7) for Westinghouse type plants.

As an initial test of the coding, initialization procedure, numerical stability and conservation laws, a "steady-state" transient calculation (with the controllers disabled) was performed. After fifteen minutes, the CENTS calculation indicated the changes in the important system variables were very small. Comparison of CENTS with CESEC and plant flow measurements for a four pump coastdown startup test indicated good agreement with the CESEC and measured RCS flow. CENTS calculations of a plant overcooling transient and the St. Lucie-1 natural circulation plant cooldown transient (Reference-8) were also carried out. Calculation-to-measurement comparisons of the pressure, and hot and cold leg temperatures for those transients indicated generally good agreement.

In Volume-3 of CE-NPD 282-P, additional benchmarking is presented for two ABB-CE PWRs. Startup measurement comparisons as well as comparisons for a set of representative CESEC licensing calculations are presented. Detailed CENTS/CESEC comparisons of the sequence-of-events, and primary and secondary system parameters indicate that the agreement is generally good, and consistent with the differences in the modeling and accuracy of these codes.

In Volume-4 of CE-NPD 282-P, benchmarking comparisons are presented for a three-loop 2775 MW_t Westinghouse plant. CENTS predictions are compared to plant measurement data for loss of AC power and loss of electrical load transients, from rated power conditions. CENTS/RELAP5 comparisons are also presented for a set of typical plant transient licensing analyses. These comparisons indicate generally good agreement between CENTS and the benchmark data.

3.0 SUMMARY OF THE TECHNICAL EVALUATION

The CENTS Technical Manual Topical Report CE-NPD 282-P provides a detailed description of the CENTS methodology and the benchmarking qualification via comparisons to plant measurements and to the NRC approved CESEC and CEFLASH-4AS codes. The review of CENTS focused on the approximations and assumptions implicit in the CENTS methodology, and the completeness and accuracy of the benchmarking of this methodology. Several important technical issues were raised during the initial review which required additional information and clarification from ABB-CE. This information was requested in Reference-9 and was provided

in the ABB-CE response included in References 2, 10 and 11. This evaluation is based on the material presented in the topical report and in References 10 and 11, and on discussions with ABB-CE and the NRC staff at a meeting in Rockville, Maryland on January 28, 1993. The evaluation of the major issues raised during this review are summarized in the following.

3.1 CENTS Modeling

3.1.1 Comparison of CENTS, CESEC and CEFLASH-4AS Models

The CENTS models are improved versions of the models included in the NRC approved CESEC and CEFLASH-4AS codes. The CEFLASH-4AS models have been adapted to the specific intended non-LOCA licensing analyses and the CESEC-III models have been updated and improved by including more detail for certain components.

The CENTS neutronics model includes a three-dimensional capability however, since no benchmarking of this capability has been provided in the topical report, the present review includes only the point kinetics neutronics model. The neutronics input to the CENTS kinetics model is determined in the same manner as for CESEC. As indicated in Response 25 (Reference-10), the determination of the effective delayed neutron fraction for the point kinetics model accounts for the reduced worth of the delayed neutrons.

The CENTS primary system calculation uses a non-equilibrium, non-homogeneous model with five conservation equations (mixture and liquid mass, mixture energy, steam enthalpy, and mixture momentum), while CEFLASH-4AS uses a non-homogeneous, equilibrium model with three mixture conservation equations. CESEC-III uses an equilibrium, homogeneous model with nodal mass and energy equations and a momentum equation for each coolant pump loop. The

CENTS primary system nodalization is similar to that of CEFLASH-4AS. CENTS solves the conservation and core heat transfer equations implicitly providing improved stability for large time steps.

Explicit models for determining (1) the nodal solute concentrations (e.g., boron, xenon, iodine and hydrogen) and (2) heat loss to the containment have been added to CENTS. CENTS includes an improved model of the upper head which allows a leakage path from the annulus. CENTS provides a multinode steam generator model while CESEC-III employs a single node steam generator model. The CENTS steam generator model has been benchmarked against plant secondary side measurements for a range of transients including a turbine trip, loss of load, pump coastdown and overcooling transient.

The flow mixing model is important for the steamline break analysis. The CENTS mixing model employs an enthalpy tilt factor to the hot legs. The inlet, outlet and flow imbalance factors used in determining the enthalpy tilt are described in Response-27 (Reference-10). CESEC uses experimentally based constants to calculate the mixing in the lower plenum, upper plena and in the upper head. In Figures 19.1-19.3 the CENTS and CESEC calculated temperatures for the affected and unaffected loops are compared for a steamline break event and indicate good agreement.

3.1.2 Neutronics Modeling

When point kinetics is used to calculate the core neutronics, the time-dependent radial and axial power distributions are calculated outside of CENTS. These calculations are performed as part of the local DNBR and fuel limits analysis and are performed with NRC approved design

codes such as ROCS/MC. The DNBR and fuel limits analyses use the system response data calculated by CENTS (average heat flux and RCS pressure, temperature and flow) together with conservative radial and axial power distributions. The DNBR and local limits analyses are performed with NRC approved methods outside of CENTS. This approach is the same as used with CESEC. CENTS provides a separate DNBR calculation for determining overall trends in thermal margin. However, this DNBR calculation is not approved and should not be used for safety related or licensing analyses.

The CENTS procedure for determining the boron reactivity employs a precalculated table of reactivity as a function of boron concentration. This reactivity table assumes a constant moderator density. This approximation introduces no error in transients where the boron concentration does not change. In Response-7 (Reference-10), ABB-CE indicates that the steamline break is the only Standard Review Plan event which might be affected by this approximation. In this case, the CENTS reactivity table underestimates the negative boron reactivity insertion which makes the steamline break analysis conservative.

3.1.3 Numerical Methods

The CENTS primary system nodalization is similar in concept to the approach used in CESEC and uses approximately the same number of nodes. The CENTS secondary side nodalization is more detailed than CESEC. CENTS uses up to ten radial nodes to describe the fuel rod. ABB-CE has performed sensitivity analyses which indicate that increasing the number of radial nodes in the fuel rod from six to ten has very little effect on the NSSS response

(Response-14, Reference-10). ABB-CE will use multiple nodes in the pellet to insure an accurate transient response. (Response-31, Reference-10).

ABB-CE has also performed detailed time step sensitivity calculations to demonstrate the numerical convergence of the CENTS solution. Results of these calculations indicate that the necessary time step depends on the specific transient and acceptable values are given for the steamline break and CEA withdrawal transients in Appendix A of Reference 10.

CENTS determines the primary and secondary side pressures using an iterative procedure. For equilibrium conditions the primary system pressure is converged to within 0.5 psia or better. For non-equilibrium conditions, the primary system pressure is converged to within 0.2 psia and the enthalpy is converged to within 0.2 Btu/lbm. CENTS converges the secondary side pressure to within 5.0×10^{-4} psia. For non-LOCA transients voiding does not occur and a void iteration is not required.

3.1.4 Modifications to CENTS

During the preparation of Volume-4 of CENPD 282-P, ABB-CE has modified CENTS in order to make exact benchmarking comparisons between CENTS and RELAP5/MOD3. In the modeling of the steamline break accident, ABB-CE conservatively neglects the moisture carryover from the steam generators. However, in order to allow an exact comparison between CENTS and RELAP5/MOD3, the CENTS moisture carryover option was set so that CENTS calculated moisture carryover during the transient. In addition, the CENTS model was modified to include a bubble-rise model in the steam generator downcomer and to predict the moisture carryover based on the average quality of the steam dome region.

It is noteworthy that ABB-CE has recently modified the neutronics solution to allow the inclusion of a fixed source term, and to improve the kinetics numerical solution for transients involving large reactivity insertions. These modifications included (1) an improved algorithm for selecting the neutronics time steps (2) a more accurate solution to the precursor equations and (3) an explicit convergence test. These solution improvements were included in Volume-4 and in the responses of Supplements 1 and 2, but were not included in Volumes 1-3 of the CENTS topical report. However, ABB-CE has recalculated the benchmarks/analyses of Volumes 1-3 and determined that the effect of the modifications on these, relatively slow, transients is negligible.

These CENTS modifications provide a more accurate and detailed solution and are considered acceptable.

3.1.5 Modeling of Westinghouse Plants

The initial application and supporting benchmarking for the CENTS analysis of ABB-CE plants is presented in Volumes 1-3 of CE-NPD 282-P. In Volume-4 and Supplement 2-P, ABB-CE provides the benchmarking and basis of the application of CENTS to Westinghouse (W) plants. The benchmarking comparisons of Volume-4 for the W plants indicate agreement that is comparable (or superior) to that provided in Volumes 1-3 for the ABB-CE plants. In addition, ABB-CE has indicated (Response-2, Reference-11) that the same level of detail employed in modeling the ABB-CE plants will be retained in the nodalization and assignment of flowpaths for the system components of W plants.

The design and operational differences between the ABB-CE and W plants are accommodated by the input selection for the control and protection system modules. The dependence on fuel and control rod design are included in the core and reactivity input. The reactor trip and control system logic are described using the CENTS generic controller module, and the CENTS tabular input is used to incorporate the differences in system parameters (flows, setpoints, etc.).

3.2 CENTS Benchmarking

3.2.1 Selection of Benchmarking Transients

As part of the CENTS qualification for the analysis of non-LOCA design basis transients and for performing safety related analyses, ABB-CE has analyzed a series of design basis events. The events analyzed have been selected to provide the most severe design basis events. In Response-3 (Reference-10), ABB-CE indicates that the selected steamline break (SLB) event provides the most rapid and severe NSSS response for a secondary side heat removal transient. The feedline break event provides the most rapid and severe overpressurization of the secondary side heatup transients, and allows the evaluation of affected loop versus unaffected loop asymmetries. The seized rotor event selected provides characteristic DNB limits evaluations for loss of flow transients. The CEA withdrawal and the CEA drop events are the only reactivity and power distribution anomaly events that will be analyzed with CENTS. The CEA withdrawal event provides the most severe power and heat flux transient for this event classification. The comparisons for the CEA withdrawal from subcritical and from hot-zero-power provided in

Appendices B and C, respectively, indicate good agreement in the predicted power and heat flux transients relative to the predictions with the approved CESEC methodology.

The benchmark comparisons provided in the topical report generally indicate good agreement relative to the CESEC calculations and plant measurements. CENTS-to-Benchmark differences which were larger than expected are described in the following.

3.2.2 Steamline Break Analysis

The Plant-A steamline break analysis presented indicates a relatively large difference between the CENTS and CESEC predictions of the cold leg temperature. In Response-19 (Reference-10), it is indicated that this difference is due to an inconsistency in the mixing model input between the CENTS and CESEC calculations. In Figures 19.1-19.3 of Reference-10, ABB-CE has provided a comparison of CESEC and an updated CENTS calculation in which the mixing models are consistent. These comparisons indicate good agreement for both the affected and unaffected loop temperatures.

In the Plant-B steamline break analysis, the CENTS safety injection occurs earlier than in the CESEC prediction. This difference in the safety injection timing is due to a more detailed upperhead model in CENTS. In Response-21 (Reference-10), ABB-CE provides detailed calculations that show that the reduced time to the safety injection setpoint is due to a faster depressurization which results primarily from the more detailed CENTS upperhead modeling.

3.2.3 Steam Generator Tube Rupture

In the Plant-A steam generator tube rupture analysis, after the initial coolant system depressurization, the CENTS break flow decreases while the CESEC break flow increases. This decreased break flow in CENTS (relative to CESEC) is due to a reduced RCS temperature and pressure. In Response-29 (Reference-10), ABB-CE indicates that the lower RCS pressure is due to a more detailed modeling of the vessel upper head region.

CENTS predicts a lower RCS pressure, pressurizer pressure and break flow in the Plant-B steam generator tube rupture (with loss of AC power) event. After the reactor trip, the RCS pressure is determined by the temperature of the coolant in the upper head. In a manner similar to Plant-A, the more detailed and accurate CENTS upper head modeling results in a lower RCS temperature, pressure and break flow (Response-22, Reference-10).

3.3 CENTS Applications

The CENTS models and solution methodology provides a realistic best estimate calculation rather than a conservative or bounding approach. It is intended that the conservatism required in licensing analyses will generally be provided by the selection of transient-specific initial conditions and plant performance data. The initial conditions are typically taken as the worst-case conditions allowed by the Technical Specifications resulting in the most severe transient results. The CENTS neutronics input is calculated to provide a conservative transient prediction. Redundant plant equipment is assumed to be out of service if allowed by the Technical Specifications. The plant performance parameters such as the RPS response times,

safety/relief valve flow capacities, coolant pump flywheel inertia and HPSI/LPSI flows are taken to be conservative relative to actual best estimate values (Response-1, Reference-10).

CENTS is intended for the analysis of the design basis licensing events. The benchmarking provided in Volumes 1-3 of CENPD 282-P includes no severe accident comparisons and only one small break LOCA comparison. In Response-3 (Reference-10), ABB-CE indicates that CENTS will not be used for performing LOCA or severe accident licensing analyses.

CENTS includes a three-dimensional coupled neutronic/thermal-hydraulic calculational capability. No benchmarking of this capability was provided in Volumes 1-3 of CENPD 282-P. Consequently, the licensing applications of CENTS are limited to the point kinetics model.

The benchmarking comparisons provided in Volumes 1-3 of CENPD 282-P do not include large rapid power transients with strong local reactivity effects typical of the control element assembly (CEA) ejection transient. In Response-3 (Reference-10), ABB-CE indicates that the CEA ejection licensing analyses will be performed with the NRC approved methods of Reference-12.

The CENTS model requires input data that is determined using relatively complex computer programs and methods. Typical examples include the calculation of core reactivity and fuel rod gap conductance. In addition, several of the modeling and input parameters are determined by adjusting the CENTS model to match the predictions of detailed design codes. Parameters of this type include the steam generator recirculation flow, evaporator region mass, nodal pressure drops and the flow mixing factors for the lower plenum.

In performing the CENTS/RELAP5 code-to-code benchmarking, representative (rather than actual plant-specific) code input data may be used to estimate the CENTS prediction uncertainty. Consequently, the ABB-CE Volume-4 benchmarking for W plants employed representative model and input parameters. However, in licensing applications of CENTS, NRC approved codes and methods must be used to determine the required model and input parameters.

4.0 TECHNICAL POSITION

The Topical Report CE-NPD 282-P and supporting documentation provided in References 2 and 10 have been reviewed in detail. Based on this review, it is concluded that the CENTS code is acceptable for performing reload licensing analyses for ABB-CE and Westinghouse PWRs subject to the conditions stated in Section-3 of this evaluation and summarized in the following.

1) CENTS DNBR Analysis

The CENTS DNBR calculation for determining overall trends in thermal margin should not be used for licensing analyses. The DNBR licensing analyses should be performed with the presently approved ABB-CE DNBR methods (Section-3.1.2).

2) LOCA and Severe Accident Assessments

Adequate benchmarking of the CENTS LOCA and severe accident capabilities has not been provided. Consequently, CENTS should not be used for performing LOCA licensing analyses or severe accident assessments (Section-3.3).

3) Three-Dimensional Core Neutronics

Benchmarking for the CENTS three-dimensional core neutronics capability has not been provided and, consequently, licensing applications of CENTS must use the point kinetics model (Section-3.3).

4) Rod Ejection Analysis

Benchmarking for the rod ejection transient has not been provided and, consequently, CENTS is not approved for performing rod ejection licensing analyses (Section-3.3).

REFERENCES

1. "Submittal of Volumes 1 and 2 of the CENTS Code Topical Report," Letter, W.G. Gates (OPPD) to NRC, dated April 19, 1991. "Submittal of Volume 3 of the CENTS Code Topical Report," Letter, S.A. Toelle (ABB-CE) to John T. Larkins (NRC), dated November 4, 1991.
2. "Submittal of Volume-4 of the CENTS Code Topical Report," Letter, S.A. Toelle (ABB-CE) to U.S. Nuclear Regulatory Commission, dated February 17, 1993.
3. "CE FLASH-4AS, A Computer Program for the Reactor Blowdown Analysis of the Small Break Loss of Coolant Accident," CENPD-138, Supplement 1 (Non-proprietary), August 1974.
4. "Response of Combustion Engineering Nuclear Steam Supply System to Transients and Accidents," CEN-128, Vol. 1 (Non-proprietary), April 1980.
5. Margolis, S.G. and Redfield, J.A., "FLASH - A Program for Digital Simulation of the Loss of Coolant Accident," WAPD-TM-534, May 1966.
6. "CESEC Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," Enclosure 1-P to LD-82-001, Letter, A.E. Scherer to D.G. Eisenhut, December 1981.
7. "RELAP5/MOD3 Code Manual, Vol. I, II, III, IV, V," NUREG/CR-5535, EGG 2596, EG&G, June, 1990.
8. "Analysis and Evaluation of St. Lucie Unit 1 Natural Circulation Cooldown," INPO-2, NSAC-16, December 1980.
9. "Request for Additional Information on the CENTS Code Topical Report," Letter, R. C. Jones (NRC) to S.A. Toelle (ABB-CE), dated December 7, 1992.
10. "Response to NRC Request for Additional Information on the CENTS Code Topical Report," CENPD 282-P, Supplement 1-P, June 1993.
11. "Response to NRC Request for Additional Information on Volume-4 of CENTS Topical Report," CENPD 282-P, Supplement 2-P, September 1994.
12. "C-E Method for Control Element Assembly Ejection Analysis," CENPD-190-A, January, 1976.

ABSTRACT

CENTS is an interactive, faster-than-real-time computer code for simulation of the Nuclear Steam Supply System and related systems. It calculates the behavior of a PWR for normal and abnormal conditions including accidents. It is a flexible tool for PWR analysis which gives the user complete control over the simulation through convenient input and output options.

Volumes 1 and 2 of CE-NPD 282-P describe the CENTS models, the input and output variables, and the data base and data dictionary. For the ABB-CE type plants, Volume 2 also presents several comparisons of plant behavior predicted by CENTS to plant data or to the results predicted by CEFLASH-4AS for the small break loss of coolant accident. Volume 3 of CE-NPD 282-P presents a more comprehensive set of comparison cases. CENTS predictions are compared to plant test data and to predictions made with CESEC, the NRC-approved NSSS simulation code used by ABB-CE for Non-LOCA safety analysis. These comparisons demonstrate that CENTS provides accurate simulations for the problems considered.

This report presents comparisons of CENTS predictions to plant transient data and to predictions made with RELAP5/MOD3 for a Westinghouse plant. The good agreement provides validation of the CENTS models and coding for the use of the CENTS code in licensing safety analyses of Non-LOCA design basis events for Westinghouse type plants.

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1.0 Introduction

CENTS provides a digital simulation of a Nuclear Steam Supply System (NSSS) for a wide range of operating conditions. CENTS is a highly flexible analytical tool which models major plant components for both the primary and secondary systems as well as the control and protection systems. It calculates the transient behavior of a PWR for normal and abnormal conditions including accidents. CENTS determines the core power and heat transfer throughout the NSSS. It also computes the thermal and hydraulic behavior of the reactor coolant in the primary and secondary systems. It includes the primary and secondary control systems and the balance-of-plant fluid systems.

CENTS is designed to support engineering, operations, and training. It supports evaluation of plant behavior for accidents, for operator actions, for design, or for scoping studies. It may be used for optimization, procedure preparation or evaluation, and training. It simulates a wide range of variations in the plant state from steady state conditions to severe accidents. It provides a full range of interactions between the analyst, the reactor control systems and the NSSS. It also allows analysis of multiple failures and the effects of operator intervention or mistakes.

CENTS has been used for several years to perform best estimate analyses of Combustion Engineering and Westinghouse plants. The CENTS models have been used as the basis for several nuclear plant full scope simulators. Extensive testing and verification was performed during those activities.

The specific aim of this report is to show that CENTS can provide appropriate predictions of plant response for the non-LOCA design basis events for the Westinghouse type plants. A set of benchmark cases were run which tested the

CENTS models for Westinghouse type plant. CENTS predictions were compared to plant transient data and to predictions made with RELAP5/MOD3, the NRC-developed NSSS simulation code used for safety analyses.

For the comparison to real plant data, complete loss of AC power accident and loss of electrical load test occurred in Plant C were simulated with the CENTS code. For the comparison to RELAP5/MOD3, the following Non-LOCA design basis events were analyzed with RELAP5/MOD3 and CENTS for Plant C.

- Seized Rotor Accident
- Feedline Break Accident
- Steam Generator Tube Rupture Accident
- Steamline Break Accident

Each of the above four events was chosen as a most extreme case of flow transients, RCS heatup transients, RCS inventory loss transients and RCS cooldown transients.

The agreement of the CENTS predictions with plant data and with the predictions of the RELAP5/MOD3 code provides validation of the CENTS models and coding. These results support use of the code to perform licensing analyses of non-LOCA transients for Westinghouse type plants.

2.0 Comparisons for Plant C, 2775 Mwt

2.1 Discussion

Verification of CENTS included comparison of plant behavior as predicted by CENTS both to measured data obtained during plant operation and to the results of accident simulations typical of those performed in support of Plant C licensing.

2.1.1 Plant Description

Plant C is one of two virtually identical units on the same site. The Nuclear Steam Supply Systems (NSSS) for both units were supplied by Westinghouse.

The initial license was to operate each of the facilities at a core thermal power output of 2,775 Mwt. Site parameters and the major systems and components including the engineered safety features and the containment structures were evaluated for operation at a core power level of 2775 Mwt. In addition, the plant Design Basis Events were evaluated at a core power level of 2775 MWt.

Plant Arrangement

Each containment structure houses a NSSS, consisting of a reactor, steam generators, reactor coolant pumps, a pressurizer, and some of the reactor auxiliaries which do not normally require access during power operation.

The turbine building houses main turbine unit (one high pressure element and three low pressure element), condensers, feedwater heaters, condensate and feed pumps, turbine auxiliaries and certain of the switchgear assemblies.

The auxiliary building houses the waste treatment facilities, engineered safety feature components, heating and ventilating system components, the emergency diesel generators, switchgear, laboratories, offices, laundry, control room, spent fuel pool and new fuel storage facilities. Fuel transfer to and from the containment is through a fuel transfer tube.

Reactor

The reactor is a pressurized light water cooled and moderated type fueled by slightly enriched uranium dioxide. The uranium dioxide is in the form of pellets and is contained in Zircalloy-4 tubes fitted with welded end caps. These fuel rods are arranged into fuel assemblies each consisting of 264 fuel rods arranged on a 17 rod square matrix. Space is left in the fuel rod array to allow for the installation of 25 guide tubes. The assembly is fitted with end fittings and spacer grids to maintain fuel rod alignment and to provide structural support. The end fittings are also drilled with flow holes to provide for the flow of cooling water past the fuel tubes.

The reactor is controlled by a combination of chemical shim and solid absorber. The solid absorber is silver-indium-cadmium alloy in the form of pellets contained in stainless steel tubes to prevent the rods from coming in direct contact with the coolant. Twenty-four tubes of absorber form a single Rod Cluster Control Assembly (RCCA). The twenty-four tubes are connected together at the tops by a yoke which is in turn connected to the Control Rod Drive Mechanism (CRDM) extension shaft. Each RCCA is aligned and is inserted into a guide tube in the fuel assembly.

Chemical shim is provided by boric acid dissolved in the coolant water. The concentration of boric acid is maintained and controlled as required by the Chemical and Volume Control System (CVCS).

Reactor Coolant System

The reactor coolant system consists of three closed heat transfer loops in parallel with the reactor vessel. Each loop, moving outward from the core exit, contains one hot leg, one steam generator, one coolant pump suction cold leg, one reactor coolant pump to circulate the coolant, and one discharge cold leg returning the coolant to the reactor vessel. A pressurizer vessel is connected to one of the coolant hot legs. The coolant system is designed to operate at a power level of 2775 MWt and to produce steam at 6.65 MPa.

Reactor coolant system pressure is maintained by 1400 KW of electrical heater elements in the lower region of the pressurizer and by pressurizer spray nozzles in the upper steam region of the pressurizer. Over-pressure protection is provided by power operated relief valves and spring-loaded safety valves connected to the pressurizer. Safety and relief valve discharge is released under water in the pressurizer relief tank where the steam discharged is condensed.

The three steam generators are vertical shell and U-tube design. Each steam generator produces approximately 517 Kg/sec of steam at rated power. Steam is generated on the shell side of the steam generator and flows upward through moisture separators. Steam outlet moisture content is less than 0.25%.

The reactor coolant is circulated by three electric motor driven, single suction, centrifugal pumps. Each pump is equipped with a nonreverse mechanism to prevent reverse rotation of any pump that has power removed.

The CENTS modelling of the Plant C NSSS is shown in Figure 2.1.1

Engineered Safety Features

Separate engineered safety features systems for each unit are included to localize, control, mitigate and terminate postulated accidents which could potentially release radioactive fission products from the fuel rods.

The engineered safety features systems include the high pressure safety injection system, the low pressure safety injection system, the accumulators and the auxiliary feedwater system.

For each unit, three accumulators are provided, each connected to one of the three cold legs. Each accumulator has a volume of 41 cubic meters containing 28 cubic meters of borated water at refueling boron concentration and 13 cubic meters of nitrogen at 4.65 MPa. In the event of a Loss of Coolant Accident, the borated water is forced into the RCS by the expansion of the nitrogen. The water from the accumulators adequately cools the entire core. In addition, borated water is injected into the same nozzles by two low pressure and three high pressure safety injection pumps taking suction from the refueling water storage tank. For maximum reliability, the design capacity from the combined operation of one high pressure and one low pressure pump provides adequate injection flow for any Loss of Coolant Accident. In the event of an accident at least one high pressure and one low pressure pump will receive power from the emergency power sources even if normal power is lost and one of the emergency diesel generators fails to start.

The auxiliary feedwater system consists of three pumps (two motor driven and one steam driven) which are capable of cooling the reactor coolant system in the event that normal feedwater is lost.

Reactor Protection System

Reactor parameters are maintained within acceptable limits by the inherent self-controlling characteristics of the reactor, by rod cluster control assembly positioning, by the boron content of the reactor coolant and by operating procedures. The function of the reactor protection system is to initiate reactor shutdown when any reactor parameter approaches the preset limits for safe operation.

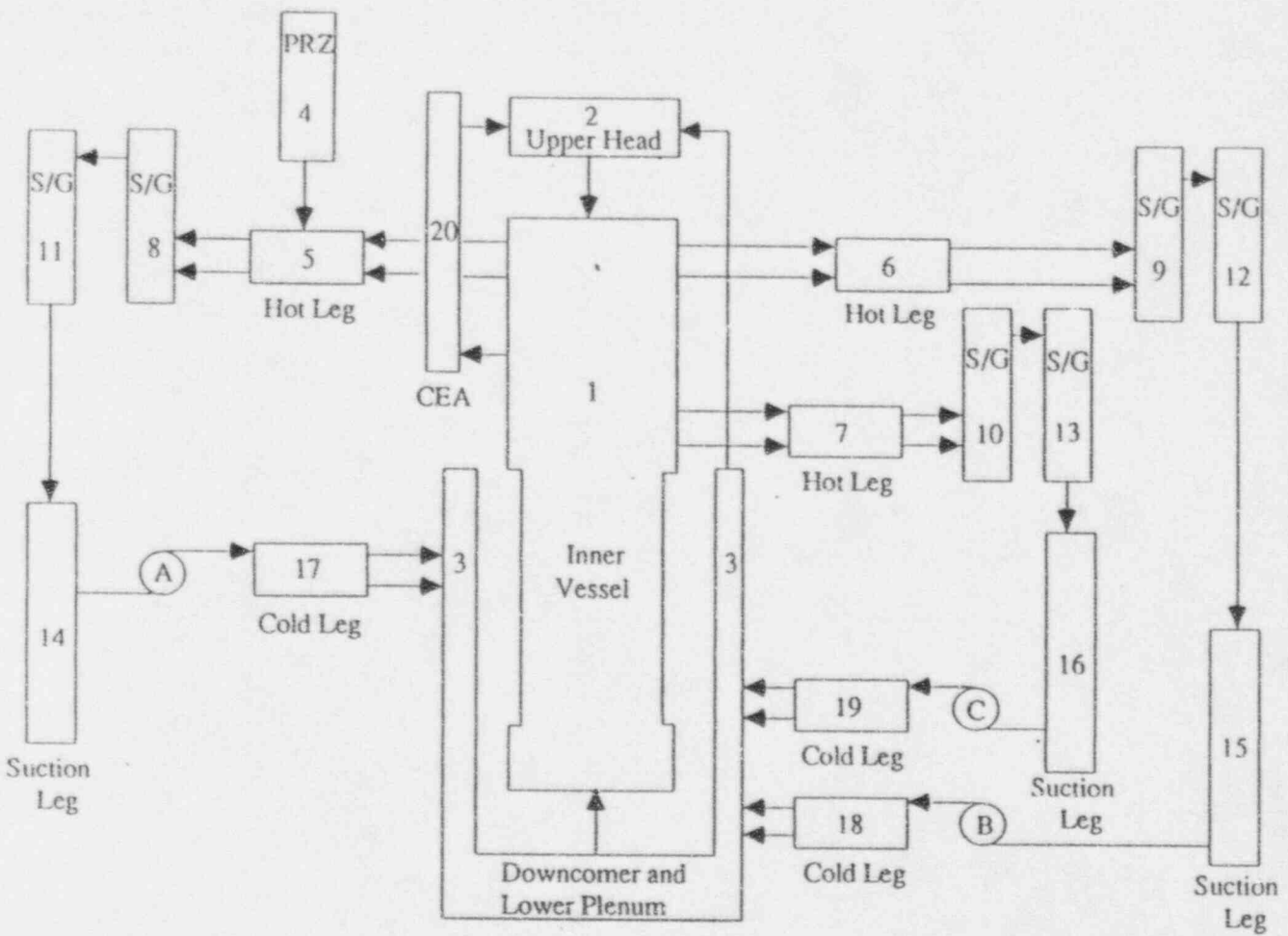
The reactor protection system is designed to provide redundant (one out of two, two out of three or two out of four) instrumentation channels for each protective function and one out of two logic train circuits. If any one of these two channels receives coincident signals, the power to the magnetic jack control rod drive mechanisms is interrupted, allowing the control rods to drop into the core to shut down the reactor. The protective system is completely independent of, and separate from, the control systems.

Operating Restrictions

Normal plant operation is restricted to the parameter limits included in the Technical Specifications. The limits are imposed to ensure that plant operation remains in compliance with the limits assumed in the safety analysis.

The Technical Specifications include restrictions such as the minimum number of safety injection pumps which must be operable, the slowest allowed response times of the containment isolation features, and restrictions on important process parameters such as reactor coolant system pressure and temperature and maximum allowed RCCA insertion.

Figure 2.1.1



CENTS Model of a Westinghouse Type Three-Loop Pressurized Water Reactor

2.2 Comparisons to Plant Data

The ability of a NSSS simulation computer code to predict the results of actual plant transients is important as it demonstrates the soundness of the basic models and indicates that the code correctly predicts plant transient response.

Two plant transients of Plant C were selected for comparison to the predictions of the CENTS code. The basis for the choice of the transients is that sufficient data was taken during the transients to perform a meaningful comparison and that the transients exercised major models of the CENTS code.

In performing the comparisons, the input data to the CENTS model was taken as best estimate for the conditions at the time of the transient.

2.2.1 Complete Loss of AC Power from 100% Initial Power

Discussion of Transient

The main transformer was damaged due to typhoon such that the complete loss of AC power occurred. Simultaneously, DC power was automatically provided to the safety-related equipment. The complete loss of AC power was immediately followed by a turbine trip and a reactor trip.

Initial Conditions

The conditions at the time of the transient are seen in Table 2.2.1.A. An input deck for CENTS was prepared to initiate the case at these conditions.

Results

Table 2.2.1.B shows the sequence of events from both the plant transient data and the CENTS simulation of the event. Because of the loss of AC power, the pressurizer pressure control system was the only control system which acted during the event. To model this, all of the control system except the pressurizer pressure control system were manually turned off in the CENTS simulation. Figures 2.2.1.A through 2.2.1.E portray comparisons of important parameters.

The predicted pressurizer pressure reached the minimum at the same time for the measured data, 68 seconds. The biggest difference in the pressurizer pressure is approximately 0.2 MPa, which is within the pressure measurement uncertainty. CENTS predicts the maximum steam generator pressure of 8.3 MPa at 10 seconds which caused the S/G safety valve open. The steam generator pressure also agrees within the acceptable range. The RCS average temperature also shows the agreement within 2°K, which is within the accuracy of the measurement. The other results show good agreement.

Table 2.2.1.A

Initial Conditions

Complete Loss of AC Power from 100% Initial Power

<u>Parameter</u>	<u>Value</u>
Core Power	100% of 2775 MWt
Core Inlet Temperature	564.3°K
Core Outlet Temperature	618.°K
Pressurizer Pressure	15.5 MPa
Pressurizer Level	6.7 Meter
Steam Generator Pressure	7.0 MPa
Steam Generator Level	12.7 Meter
Reactor Regulating System	Automatic
Pressurizer Pressure and Level Control	Automatic
Steam Generator Level Control	Automatic
Steam Dump and Bypass System	Automatic

Table 2.2.1.B

Sequence of Events

Complete Loss of AC Power from 100% Initial Power

Time(Sec)		Value	
<u>CENTS</u>	<u>Plant Data</u>	<u>Event</u>	<u>CENTS</u> <u>Plant Data</u>
[	]	Main Transformer damage	[
		Turbine trip	
		Reactor trip	
		RCP coastdown	
		Minimum pressurizer pressure, MPa	

2.2.2 Loss of Electrical Load Test from 100% Initial Power

Discussion of Test

The objectives of this test are to demonstrate the ability of the plant to sustain a 95% load-loss at 100% power without reactor trip and turbine trip and to evaluate the system response to the transient. However, the plant was tripped by a pressurizer pressure low signal which was caused by an over-response of the pressurizer Power Operated Relief Valve (PORV). Three of the four steam dump bypass valve banks opened within 5 seconds after the power circuit breaker (PCB) opened. The fourth bank opened 8 seconds after PCB opened. The PORV also opened due to the increase of the pressurizer pressure. The opening of PORV decreases the pressurizer pressure and in the event caused the reactor trip by the lead/lag compensated low pressurizer pressure signal, approximately 10 seconds after the PCB opened.

Initial Conditions

The conditions at the time of the test are seen in Table 2.2.2.A. An input deck for CENTS was prepared to initiate the case at these conditions.

Results

Table 2.2.2.B shows the sequence of events from both the plant test data and the CENTS simulation of the event. All of the control systems were automatically controlled in the CENTS simulation. Figures 2.2.2.A through 2.2.2.F portray comparisons of important parameters.

CENTS predicted the maximum pressurizer pressure at 6 seconds after PCB open while the test data shows the maximum pressurizer pressure at 8 seconds after PCB open (Fig. 2.2.2.A). However, overall comparison of the pressurizer pressure shows good agreement. CENTS result shows the reactor and turbine trip 9.7

seconds after the PCB opened but the test data shows trips 10 seconds after the PCB opened. The comparisons of other important parameters (Fig. 2.2.2.B through Fig. 2.2.2.F) show good agreement with those of the test data.

Table 2.2.2.A

Initial Conditions

Loss of Electrical Load Test from 100% Initial Power

<u>Parameter</u>	<u>Value</u>
Power	100% of 2775 MWt
Core Inlet Temperature	564.3°K
Core Outlet Temperature	618.°K
Pressurizer Pressure	15.6 MPa
Pressurizer Level	6.8 Meter
Steam Generator Pressure	6.8 MPa
Steam Generator Level	12.7 Meter
Pressurizer Pressure and Level Control	Automatic
Steam Generator Level Control	Automatic
Steam Dump and Bypass System	Automatic

Table 2.2.2.B

Sequence of Events

Loss of Electrical Load Test from 100% Initial Power

Time(Sec)		Value	
<u>CENTS</u>	<u>Test</u>	<u>Event</u>	<u>CENTS</u> <u>Test</u>
[	]	PCB open	[
		Maximum pressurizer pressure, MPa	
		Reactor & turbine trip	
		Maximum hot leg temperature, °K	
		Maximum steam header pressure, MPa	
		Minimum pressurizer pressure, MPa	

2.3 Comparisons to RELAP5/MOD3 Predictions

A set of benchmark cases were run which compared CENTS predictions to those made with RELAP5/MOD3, the NRC-developed code used for non-LOCA and LOCA safety analyses. RELAP5/MOD3 is described in Reference 1, which was supported by the NRC.

A set of RELAP5/MOD3 results for the Plant C unit were examined to find representative cases for comparison to the plant response as predicted by CENTS. The basis for the case selection was to challenge the CENTS models in order to identify any significant differences in the results of the two codes. Therefore the most severe design basis events were selected for the comparison.

2.3.1 Feed Line Break

Discussion of Event

A feedwater system pipe break may produce a total loss of normal feedwater and a blowdown of one steam generator. If normal sources of AC electrical power were lost, there would also be a simultaneous loss of primary coolant flow, turbine load, pressurizer pressure and level control and steam bypass control. The result of these events would be a rapid decrease in the heat transfer capability of both steam generators and eventually the complete loss of the heat transfer capability of one steam generator.

The NSSS is protected during this transient by the pressurizer safety valves and the following reactor trips;

- Low steam generator level

- Overtemperature delta-T
- High pressurizer pressure
- Safety injection signals from any of the following;
 - o Two out of three low steamline pressure
 - o Two out of three high containment pressure

Depending on the initial conditions, any one of these trips may terminate the transient. The NSSS is also protected by main steam isolation valves (MSIVs), feedwater line check valves, steam generator safety valves and the auxiliary feedwater system which serves to protect the integrity of the secondary heat sink following reactor trip.

The NRC criterion for this event is that the peak RCS pressure must be less than 120% of RCS design pressure.

For Westinghouse type plants, the most limiting feedline rupture is a double-ended rupture of the largest feedline. Thus, a single Feed Line Break case was simulated using the CENTS and RELAP5/MOD3 codes for a double-ended rupture of the largest feedline. The case assumes that a large break (0.13 Meter^2) occurs in the feedline to one of the steam generators, downstream of the feedwater check valve.

Table 2.3.1.A lists the important assumptions for this case. These assumptions were used in setting up the case data for CENTS and for RELAP5/MOD3.

Analysis Methods

The CENTS code includes the models necessary to incorporate the feedwater system pipe break methodology presented in Chapter 15 of FSAR of Westinghouse three loop plant. The case analyzed assumes a double-ended rupture of the largest feedwater pipe at full power. Major assumptions made in this analysis are;

- a) No credit for the pressurizer power-operated relief valves or pressurizer spray
- b) No credit for the high pressurizer pressure reactor trip
- c) Main feedwater to all steam generators stops at the time the break occurs (or, all main feedwater spills out through the break).
- d) The reactor trip occurs on steam generator low-low level.
- e) The auxiliary feedwater system is actuated by the low-low steam generator water level signal. The auxiliary feedwater system supplies a total of 0.028 Meter³/sec to the two unaffected steam generators, including allowance for possible spillage through the main feedwater line break.
- f) A 60-second delay following the low-low level signal to allow time for startup of the standby diesel generators and the auxiliary feed pumps.
- g) No credit for charging or letdown.
- h) Steam generator heat transfer area decreases as the shell-side liquid inventory decreases.

Receipt of a low-low steam generator water level signal in at least one steam generator starts the motor-driven auxiliary feedwater pumps, which in turn initiate auxiliary feedwater flow to the steam generators. Similarly, receipt of a low steamline pressure signal in at least one steamline initiates a steamline isolation signal which closes all main steamline isolation valves. This signal also gives a safety injection signal which initiates flow of cold borated water into the RCS. The amount of safety injection flow is a function of RCS pressure.

Results

Table 2.3.1.B provides a comparison of the sequence of events for the Feed Line Break Event. Figures 2.3.1.A through 2.3.1.Y provide comparisons of important parameters as calculated by the CENTS and the RELAP5/MOD3 codes.

The major concerns of this event are: short term RCS cooldown until the affected steam generator is empty, RCS heatup after the steam generator is empty, MSIV closing time, and the long term cooling capability of the two intact steam generators by auxiliary feedwater flow. During the period of event until steam generator empty for the broken loop, the RCS temperature calculated by CENTS is a little higher than that calculated by RELAP5/MOD3 due to the different liquid mass in the affected steam generator. The rapid decrease in liquid mass in the affected steam generator results in lower heat removal in CENTS than RELAP5/MOD3. Except this difference, the other parameters concerning the RCS heatup and long term cooling of two intact steam generators show good agreement between the two codes.

Table 2.3.1.A

**Important Assumptions
Feed Line Break**

<u>Parameter</u>	<u>Value</u>
Break Size	0.13 Meter ²
Core Power	2775 MWt
Core Inlet Temperature	565.17°K
Pressurizer Pressure	15.526 MPa
Steam Generator Pressure	6.805 MPa
Steam Generator Level	50 % of Level Span
Scram Rod Worth	4000 pcm
All Control Systems	Manual Mode
Loss of Offsite Power reactor trip signal is generated.	Power is lost 2 seconds after
Auxiliary Feed Pump Delay	60 seconds
MSIV Signal Delay	7 seconds

Table 2.3.1.B

Sequence of Events
Feed Line Break

Time(Sec)		Event	Value	
<u>CENTS</u>	<u>RELAP5</u>		<u>CENTS</u>	<u>RELAP5</u>
		Feed line break, Meter ²		
		Steam generator low-low level trip signal is generated, %		
		Rods begin to drop		
		Reactor Coolant Pumps begin to coast down		
		Turbine trip and main feedwater is terminated		
		Auxiliary feedwater is delivered, Meter ³ /sec		
		Low steamline setpoint is reached in affected s/g, MPa		
		Main steamline isolation		

Table 2.3.1.B (Continued)

Time(Sec)		Event	Value	
<u>CENTS</u>	<u>RELAP5</u>		<u>CENTS</u>	<u>RELAP5</u>
[	]	HPSI is delivered to each cold leg	[	]
		Pressurizer safety valves open, MPa		
		RCS temperature begins to decrease due to auxiliary feedwater flow		

2.3.2 Seized Rotor

Discussion of Event

A single reactor coolant pump rotor seizure can be caused by seizure of the upper or lower thrust-journal bearings.

Following seizure of a reactor coolant pump shaft, the core flow rate rapidly decreases to the value which occurs with only two of the reactor coolant pumps in operation. The reduction in core flow with the associated increase in core coolant inlet temperature will reduce the margin to the DNB safety limit and increase the system pressure.

For Plant C the event is terminated by the Low Reactor Coolant Flow reactor trip.

A single Seized Rotor event case was simulated using the CENTS and RELAP5/MOD3 codes. The case assumes that a reactor coolant pump stops instantaneously at the initiation of event.

Table 2.3.2.A lists of the important assumptions for this case. These assumptions were used in setting up the case data for CENTS and for RELAP5/MOD3.

Results

Table 2.3.2.B provides a comparison of the sequence of events for the Seized Rotor event. Figures 2.3.2.A through 2.3.2.J provide comparisons of important parameters as calculated by the CENTS and the RELAP5/MOD3 codes.

Figure 2.3.2.B shows a small difference in pressurizer pressure between the two sets of results. The other system parameters show good agreement especially, for loop mass flow rate and RCS temperatures.

Table 2.3.2.A

**Important Assumptions
Seized Rotor**

<u>Parameter</u>	<u>Value</u>
Core Power	2775 MWt
Scram Worth	4000 pcm
Initial Core Inlet Temperature	565.17°K
Pressurizer Pressure	15.526 MPa
Steam Generator Pressure	6.805 MPa
RCCA Drop Time	3.3 seconds
Pressurizer Pressure Control System	Manual
Turbine Bypass System	Inoperable
Loss of Offsite Power	Power is lost 1 second after a reactor trip signal is generated.

Table 2.3.2.B

Sequence of Events
Seized Rotor

Time(Sec)		Event	Value	
<u>CENTS</u>	<u>RELAP5</u>		<u>CENTS</u>	<u>RELAP5</u>
[	]	Seizure of a single reactor coolant pump shaft	[	]
		Low reactor coolant flow reactor trip condition, Fraction of initial		
		Rods begin to drop		
		Loss of offsite power; Cooldown of the remaining reactor coolant pumps		
		Turbine stop valve is instantaneously closed		
		Peak primary system pressure, MPa		
		Main steam safety valves begin to open, MPa		

2.3.3 Steam Generator Tube Rupture

Discussion of Event

The Steam Generator Tube Rupture accident is a penetration of the barrier between the reactor coolant system and the main steam system which results from the failure of a steam generator U-tube. Integrity of the barrier between the RCS and main steam system is significant from a radiological release standpoint. The radioactivity from the leaking steam generator mixes with the shell-side water in the affected steam generator. A fraction of the radioactive inventory which leaks into the affected steam generator is subsequently released to atmosphere.

A Steam Generator Tube Rupture event causes a depressurization of the RCS. A reactor trip is generated by either the over-temperature delta-T trip or the Low Pressurizer Pressure trip. For this analysis, a reactor trip is assumed to occur when the Pressurizer Low Pressure reaches the trip setpoint. This is the latest time at which a reactor trip would occur.

A single Steam Generator Tube Rupture case was simulated using the CENTS and RELAP5/MOD3 codes. Table 2.3.3.A lists of the important assumptions for this case. These assumptions were used in setting up the case data for CENTS and for RELAP5/MOD3.

Results

Table 2.3.3.B provides a comparison of the sequence of events for the Steam Generator Tube Rupture Event. Figures 2.3.3.A through 2.3.3.N provide comparisons of important parameters as calculated by the CENTS and the RELAP5/MOD3 codes.

Figure 2.3.3.B traces the pressurizer pressure calculated by CENTS and by RELAP5/MOD3. The rate of depressurization calculated by CENTS is slightly slower than that calculated by RELAP5/MOD3 due to the fact that the break flow predicted by RELAP5/MOD3 is slightly higher than that of CENTS. For the calculation of critical flow, CENTS uses the Henry-Fauske model however RELAP5/MOD3 uses an equation derived from the Bernoulli equation which slightly overpredicts the critical flow in most cases. Since, reactor trip is by low pressurizer pressure in this case, this overprediction of CENTS for the pressurizer pressure resulted in a delay in reactor trip time.

Except the effects of different trip time, the comparisons show good agreement.

Table 2.3.3.A

**Important Assumptions
Steam Generator Tube Rupture**

<u>Parameter</u>	<u>Value</u>
Core Power	2775 MWt
Scram Worth	4000 pcm
Core Inlet Temperature	565.17°K
Pressurizer Pressure	15.526 MPa
Steam Generator Pressure	6.805 MPa
Scram Delay	2 seconds
RCCA Drop Time	3.3 seconds
Pressurizer Level Control System	Manual
Pressurizer Pressure Control System	Manual
Turbine Bypass System	Inoperable
Loss of Offsite Power	Power is lost 2 seconds after a reactor trip signal is generated

Table 2.3.3.B

Sequence of Events
Steam Generator Tube Rupture

Time(Sec)		Event	Value	
<u>CENTS</u>	<u>RELAP5</u>		<u>CENTS</u>	<u>RELAP5</u>
		Double ended rupture of a steam generator tube, cm ²		
		Pressurizer low pressure trip setpoint is reached, MPa		
		Rods begin to drop		
		Turbine stop valve is instantaneously closed		
		Coast down of reactor coolant pumps		
		Steam generator safety valves open, MPa		
		Safety injection actuation		
		Auxiliary feedwater pumps begin to deliver the flow		

2.3.4 Steam Line Break

Discussion of Event

A postulated rupture of the main steam line is analyzed in accordance with Section 15.1.5 of the Standard Review Plan, Reference 2. The analysis is performed to demonstrate that sufficient sources of negative reactivity are available to offset the insertion of positive reactivity added during the transient by the rapid cooldown of the moderator.

A single Steam Line Break case was simulated using the RELAP5/MOD3 and the CENTS codes. The case assumes that a double ended guillotine break occurs in the main steam line inside the containment building from zero power initial conditions. This case does not assume a loss of AC power so that the reactor coolant pumps continue to operate throughout the event.

Table 2.3.4.A contains a listing of the important assumptions for this case. These assumptions were used in setting up the case data for CENTS and for RELAP5/MOD3.

The cooldown of the reactor coolant system continues until the affected steam generator empties. The Steam Line Break case is run to the time at which the core is subcritical and negative reactivity is being added.

Analysis Methods

A number of analysis assumptions affect the calculation of the maximum reactivity feedback. The CENTS code includes several options to ensure that the simulation of a steam line break event provides conservative results. These options in CENTS enable the code to simulate all of the reactivity effects discussed in Chapter 15 of

the FSAR for Westinghouse three loop plant. Important assumptions used in both the CENTS and RELAP5/MOD3 calculations include:

- a) End of life shutdown margin at no load, equilibrium xenon conditions, and the most reactive RCCA stuck in its fully withdrawn position.
- b) A negative moderator coefficient corresponding to the end of life core condition.
- c) Minimum capability for injection of high concentration boric acid (2000 ppm) solution corresponding to the most restrictive single failure in safety injection system. The emergency core cooling system (ECCS) consists of three systems: 1) the passive accumulators, 2) the low head safety injection (residual heat removal) system, 3) the high head safety injection (charging) system. Only the high head safety injection system is considered for this analysis.
- d) After the generation of the safety injection signal (appropriate delays for instrumentation, logic, and signal transport included), the appropriate valves begin to operate and the high head safety injection pump starts. In 12 seconds, the valves are assumed to be in their final position and the pump is assumed to be at full speed.
- e) Since the steam generators are provided with integral flow restrictors with a 0.13 square meter throat area, any rupture with a break area greater than 0.13 square meter, regardless of location, would have a similar effect on the NSSS as the 0.13 square meter break.
- f) The MSIVs (main steam line isolation valves) fully close within 10 seconds.

Results

Table 2.3.4.B provides a comparison of the sequence of events for the Steam Line Break Event without Loss of AC power. Figure 2.3.4.A through 2.3.4.P provide comparisons of important parameters calculated by the CENTS and the RELAP5/MOD3 codes.

As shown in Figures 2.3.4.E and 2.3.4.F, RELAP5/MOD3 results show that the break flow stops at around 130 seconds even though the liquid mass of the affected steam generator is still present due to the stagnant region in moisture separator. In contrast, CENTS results show that the break flow exists until the affected steam generator is totally empty. Due to this difference in break flow, the cold leg temperature of the affected loop predicted by CENTS increases more slowly than that of RELAP5/MOD3 at about 100 seconds when the neutron power starts increasing.

The hot leg temperature of the affected loop predicted by CENTS in Figure 2.3.4.H increases for a while in the beginning of event due to the outsurge flow from the pressurizer. However, the hot leg temperature calculated by RELAP5/MOD3 does not increase due to the fact that the RELAP5/MOD3 has several control volumes for the hot leg and the compared hot leg temperature was taken from a node connected just after the core outlet nozzle without pressurizer surge line connection.

Except for the differences described above, the results show good agreement.

Table 2.3.4.A

**Important Assumptions
Steam Line Break**

<u>Parameter</u>	<u>Value</u>
Break Size	0.13 Meter ²
Core Power	1 Wt
Shutdown Margin	1770 pcm
Core Burnup	End of Cycle
Core Inlet Temperature	565.17°K
Pressurizer Pressure	15.526 MPa
Steam Generator Pressure	6.632 MPa
Steam Generator Level	50 %
All Control Systems	Inoperable
Loss of Offsite Power	Offsite Power is not lost during the event

Table 2.3.4.B

Sequence of Events

Steam Line Break

Time(Sec)		Event	Value	
<u>CENTS</u>	<u>RELAP5</u>		<u>CENTS</u>	<u>RELAP5</u>
[	]	Main steam line break, Meter ²	[	]
		Low steam line pressure signal actuats SI system, MPa		
		Main steam isolation valves are closed		
		Auxiliary feedwater pumps begin to deliver flow		
		HPSI system begins to deliver borated water		

Comparison of CENTS to Plant C Measured Data
Complete Loss of AC Power from 100% Initial Power

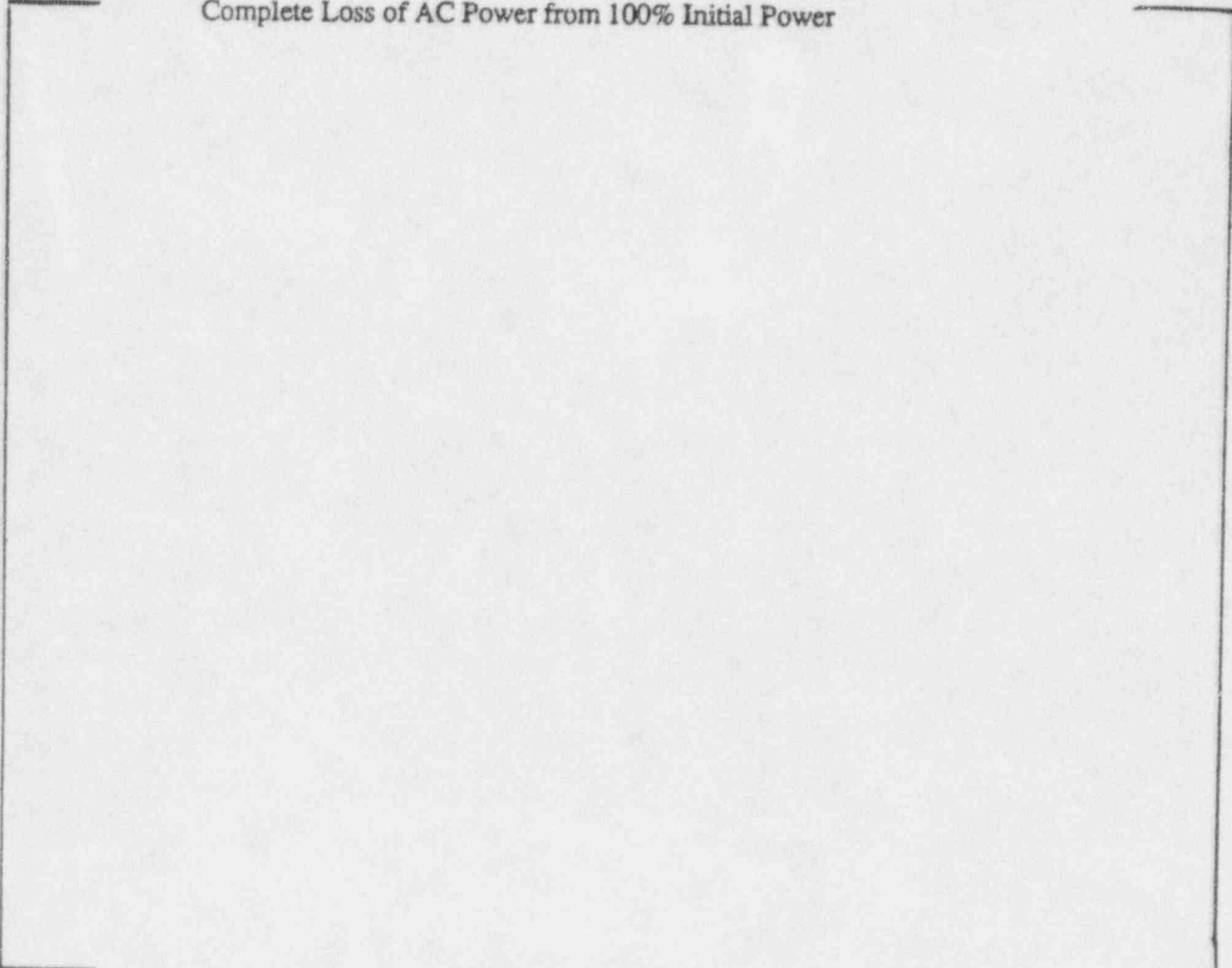


Figure 2.2.1.A Pressurizer Pressure

Comparison of CENTS to Plant C Measured Data
Complete Loss of AC Power from 100% Initial Power

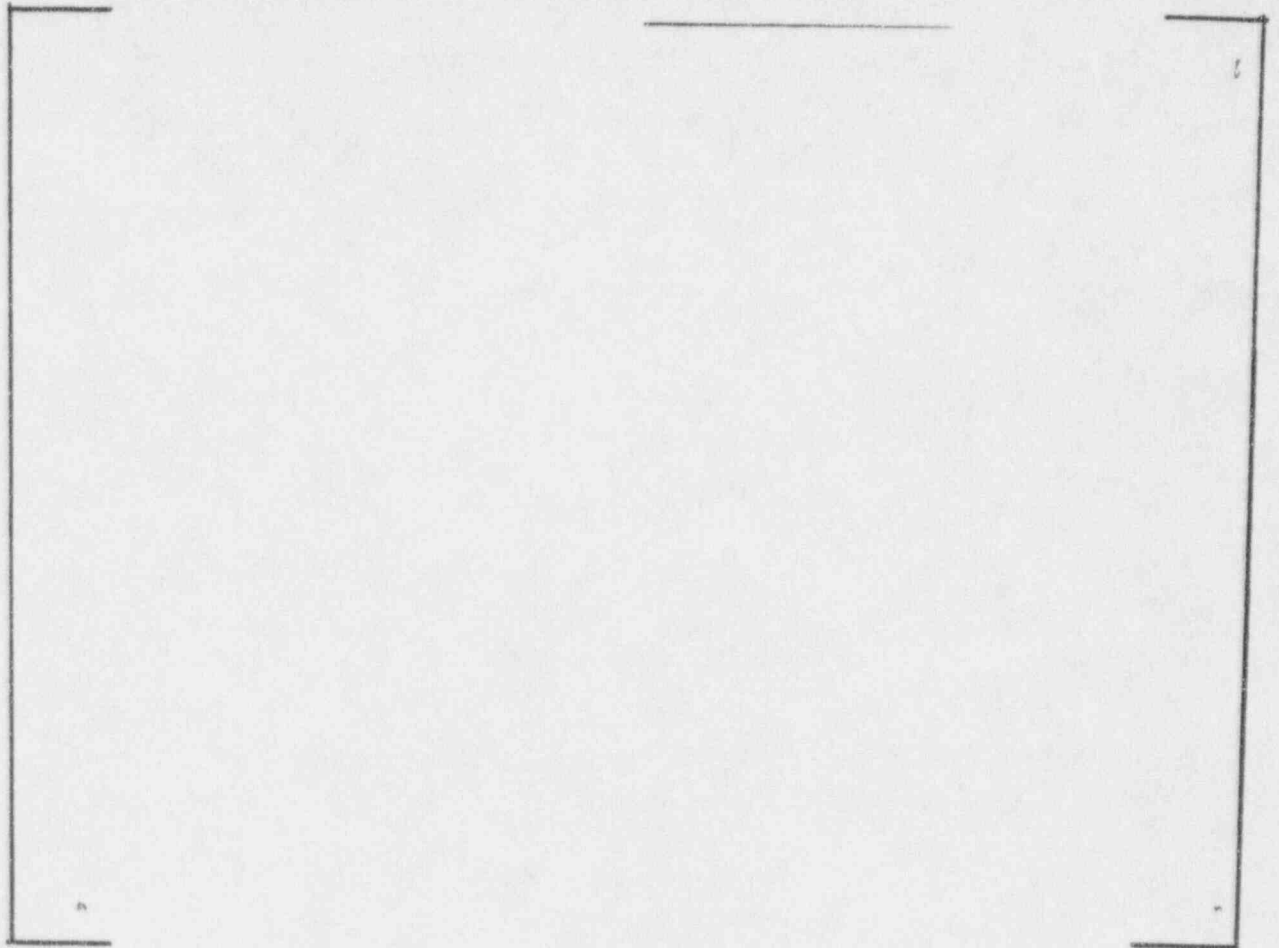


Figure 2.2.1.B Steam Generator Pressure

Comparison of CENTS to Plant C Measured Data
Complete Loss of AC Power from 100% Initial Power

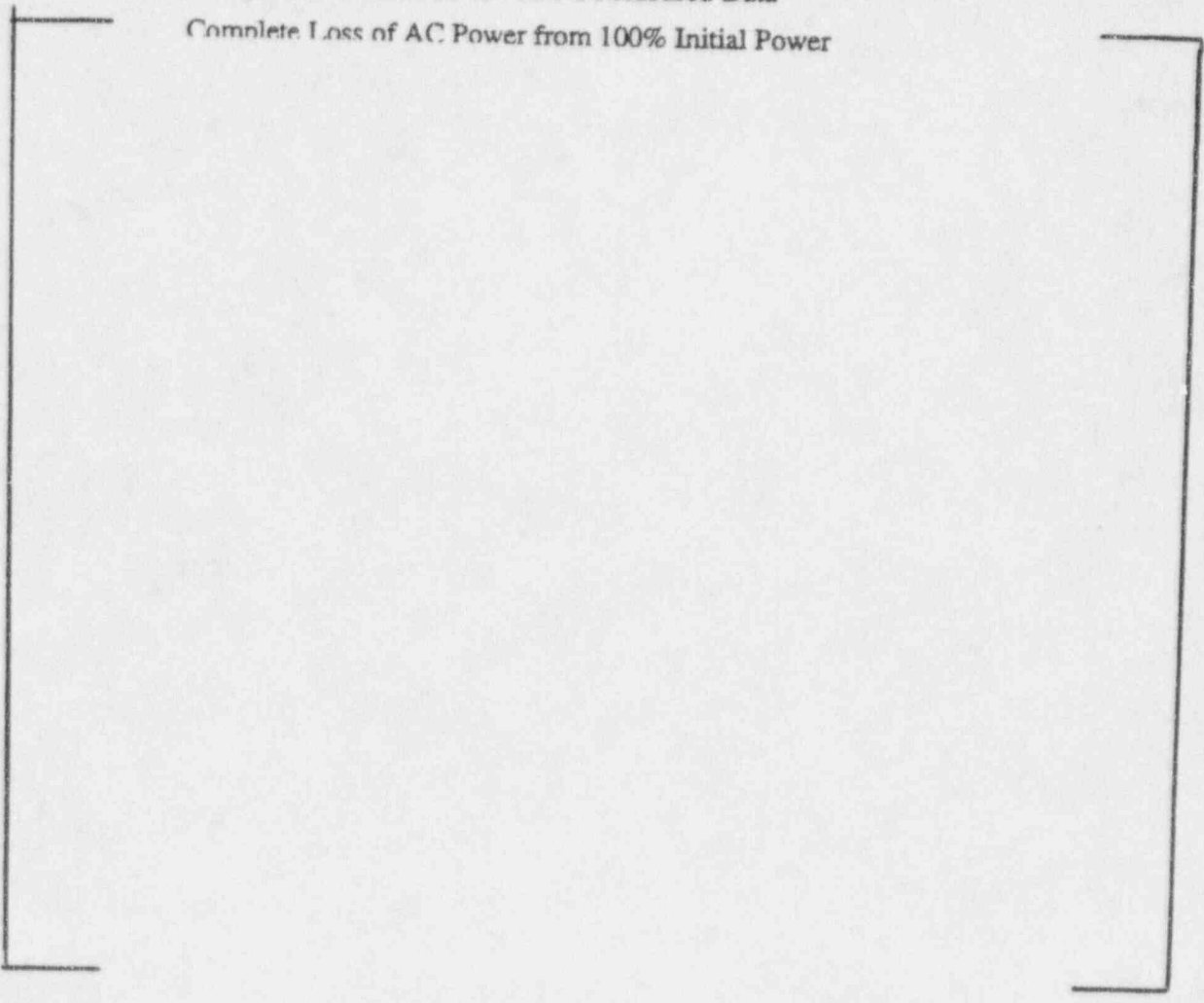


Figure 2.2.1.C RCS Average Temperature

Comparison of CENTS to Plant C Measured Data
Complete Loss of AC Power from 100% Initial Power

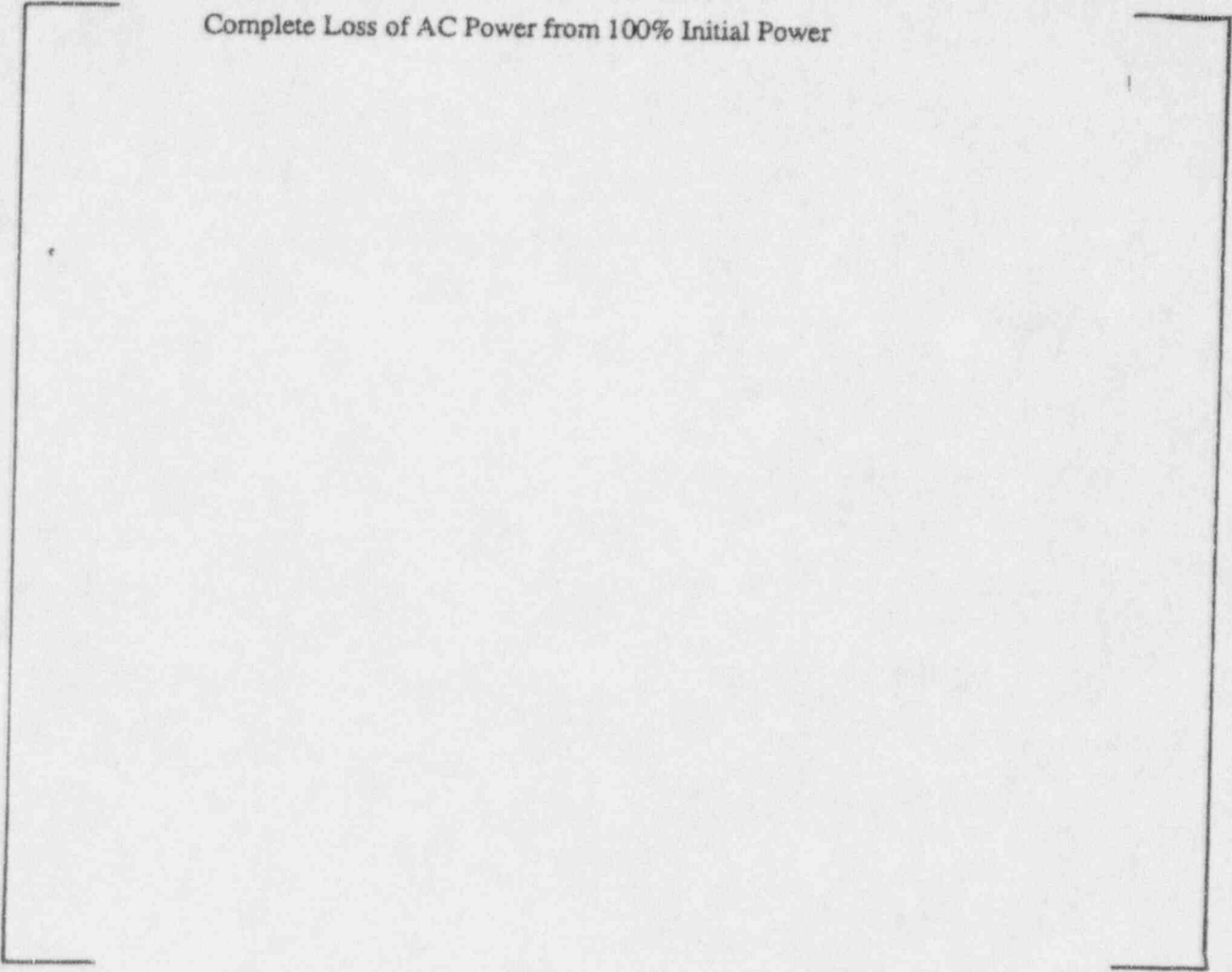


Figure 2.2.1.D RCS Loop Flow

Comparison of CENTS to Plant C Measured Data
Complete Loss of AC Power from 100% Initial Power

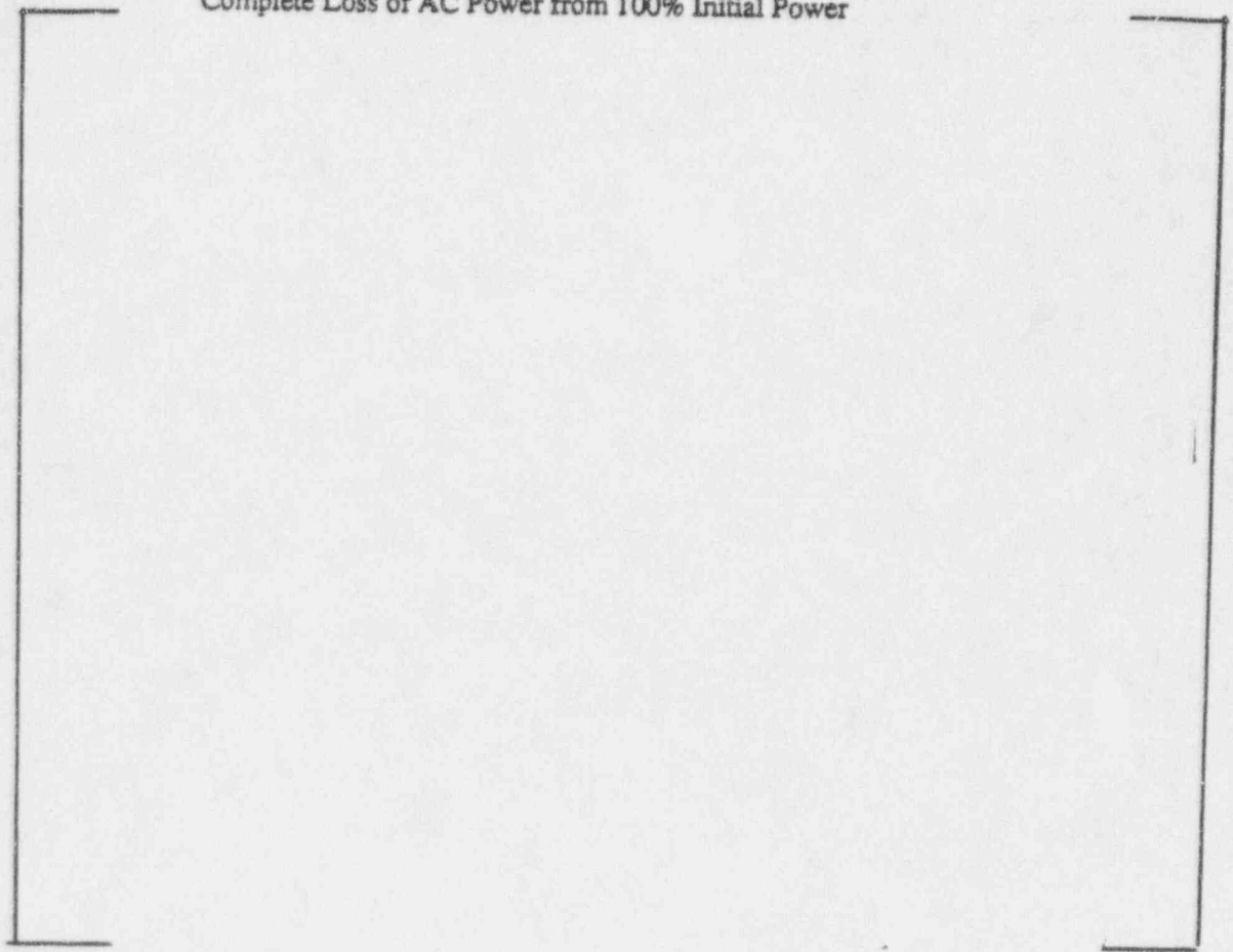


Figure 2.2.1.E Steam Generator Water Level

Comparison of CENTS to Plant C Measured Data
Loss of Electrical Load Test from 100% Initial Power

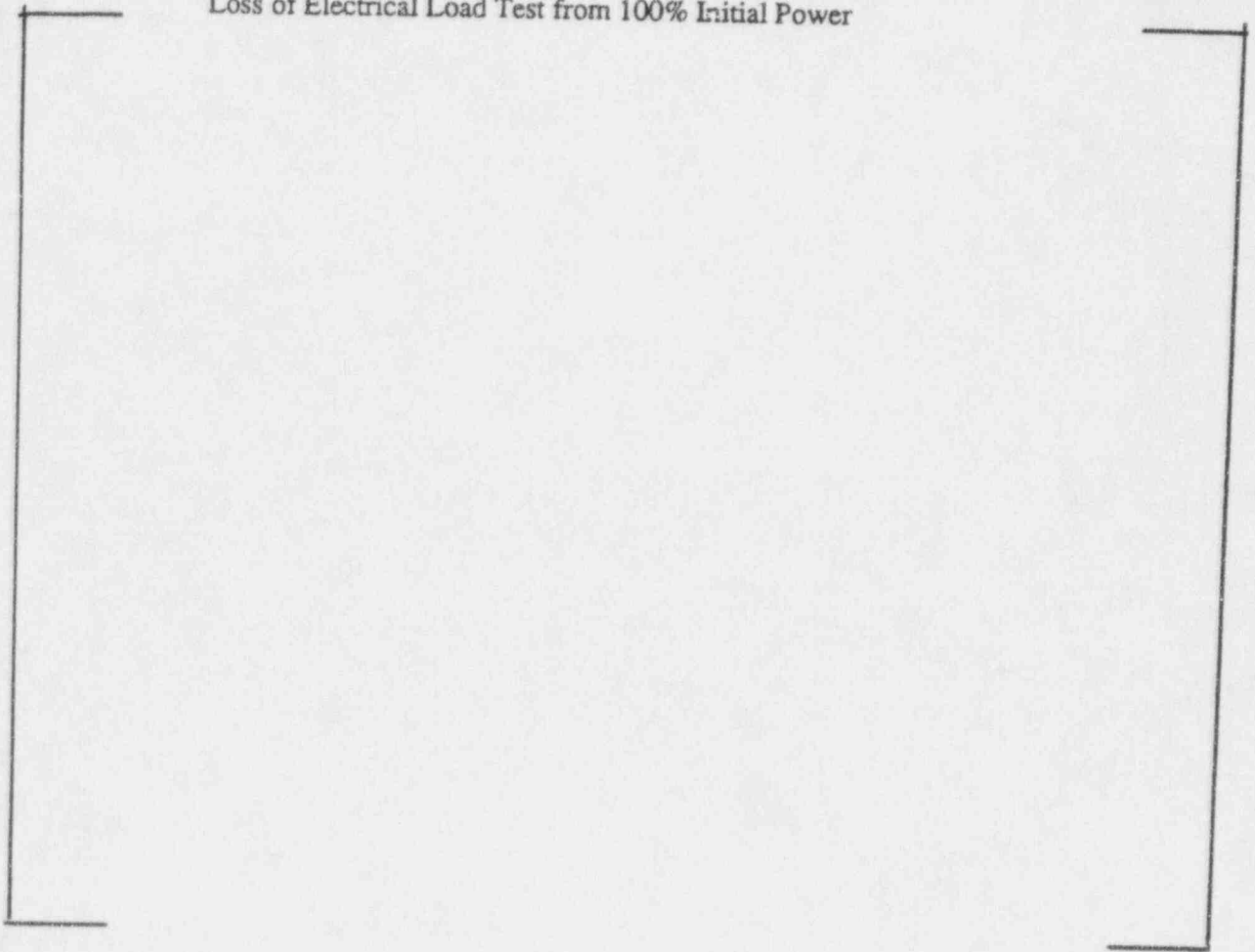


Figure 2.2.2.A Pressurizer Pressure

Comparison of CENTS to Plant C Measured Data
Loss of Electrical Load Test from 100% Initial Power

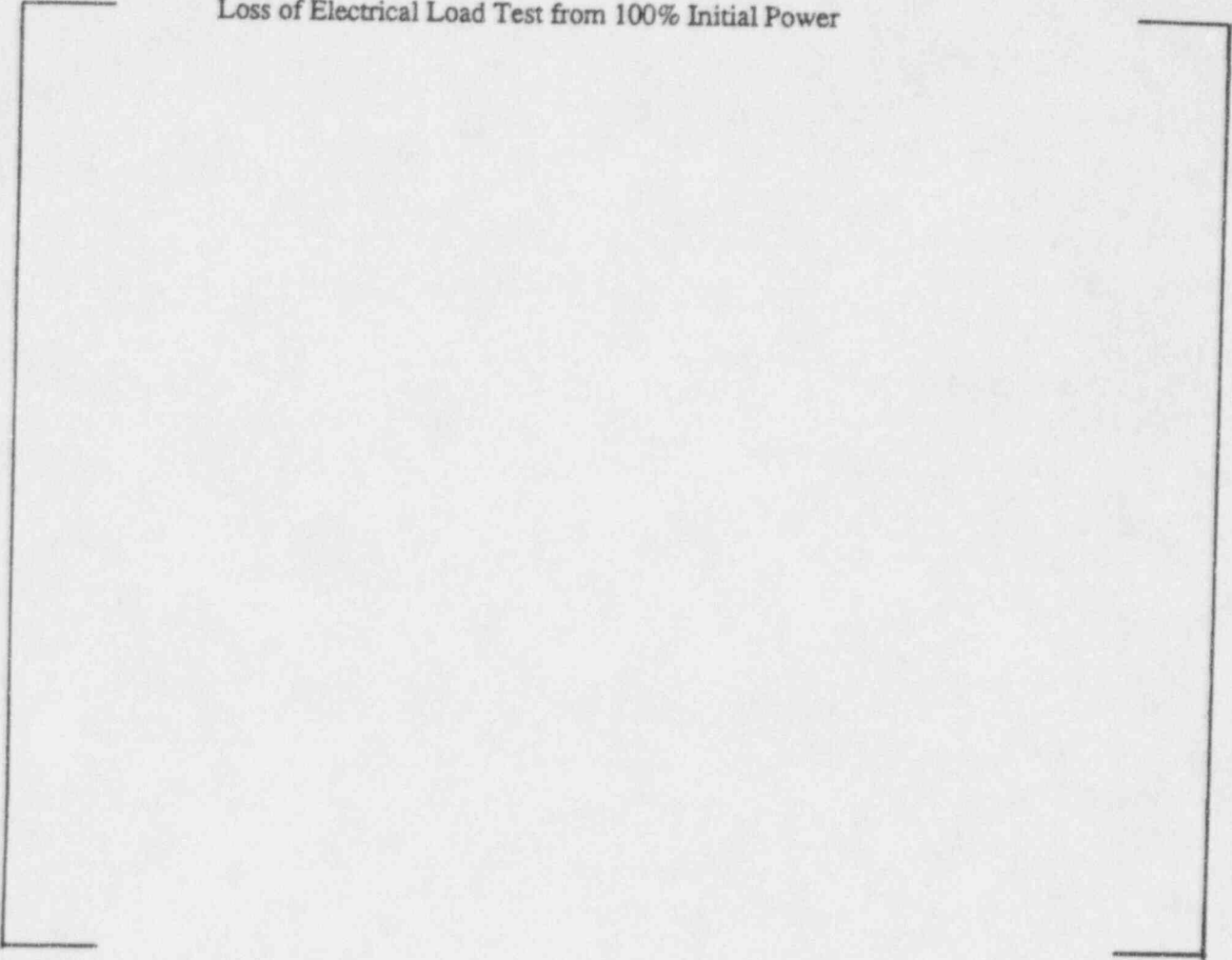


Figure 2.2.2.B Steam Header Pressure

Comparison of CENTS to Plant C Measured Data
Loss of Electrical Load Test from 100% Initial Power

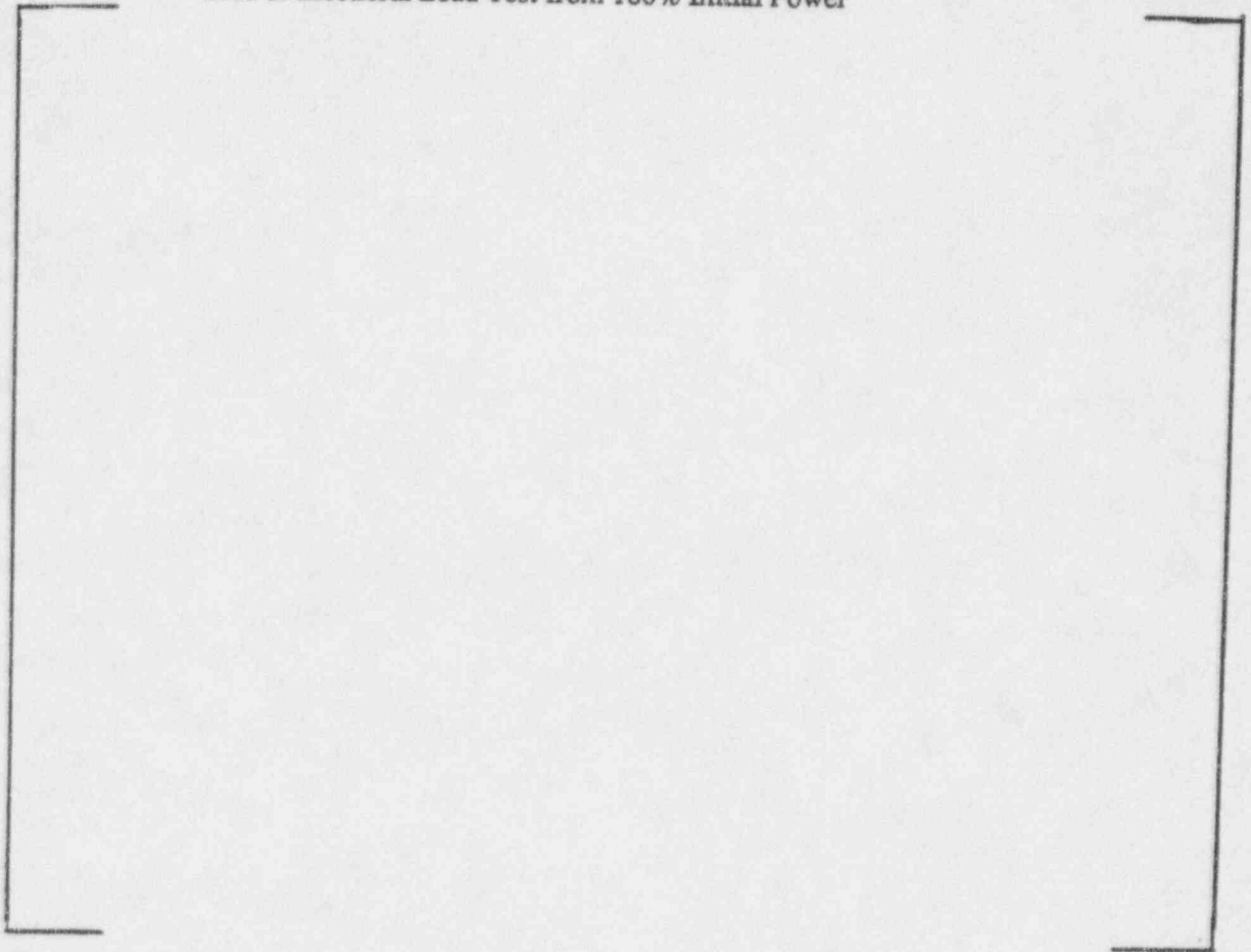


Figure 2.2.2.C RCS Average Temperature

Comparison of CENTS to Plant C Measured Data
Loss of Electrical Load Test from 100% Initial Power

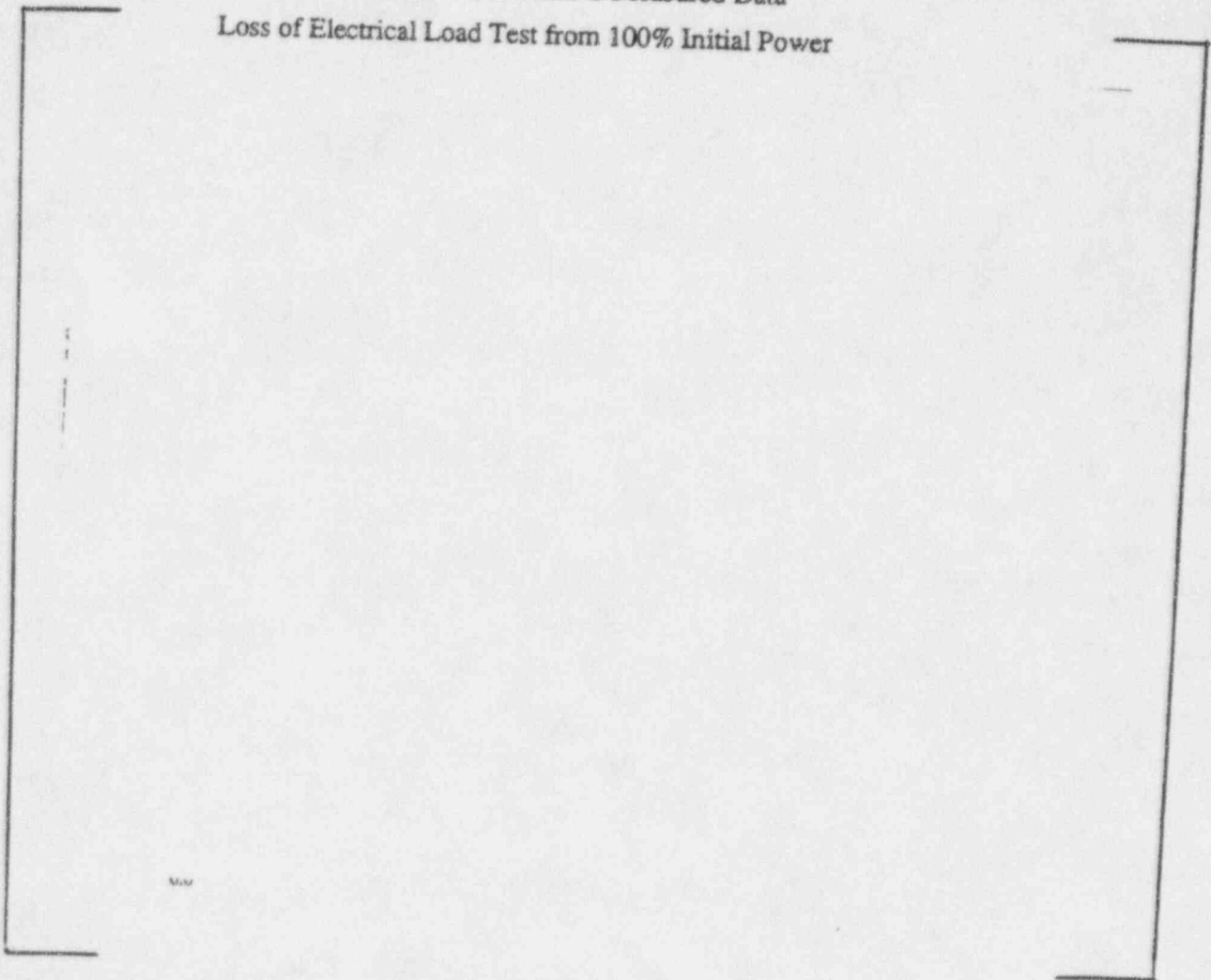


Figure 2.2.2.D Pressurizer Water Level

Comparison of CENTS to Plant C Measured Data
Loss of Electrical Load Test from 100% Initial Power

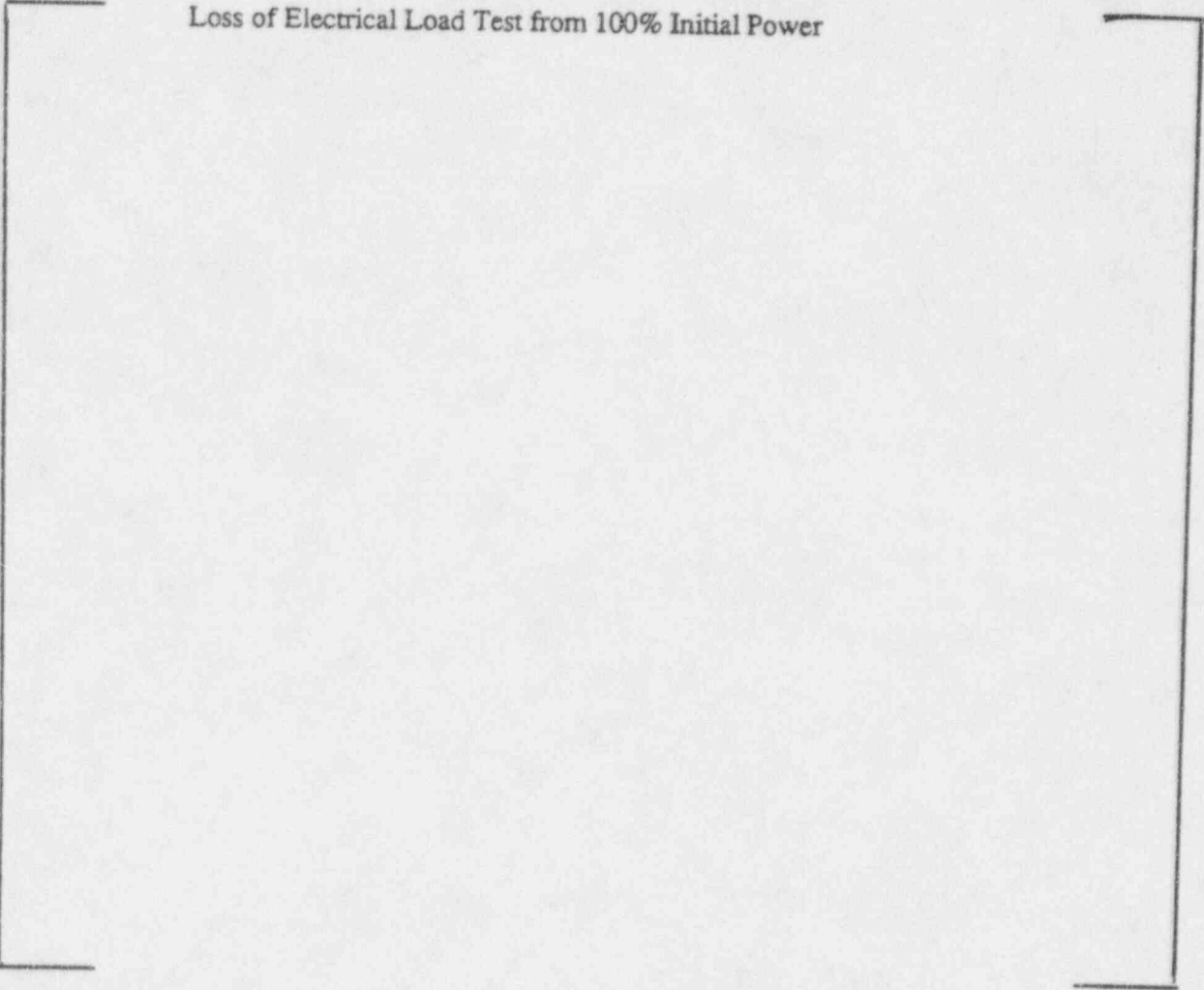


Figure 2.2.2.E Steam Generator Water Level

Comparison of CENTS to Plant C Measured Data
Loss of Electrical Load Test from 100% Initial Power

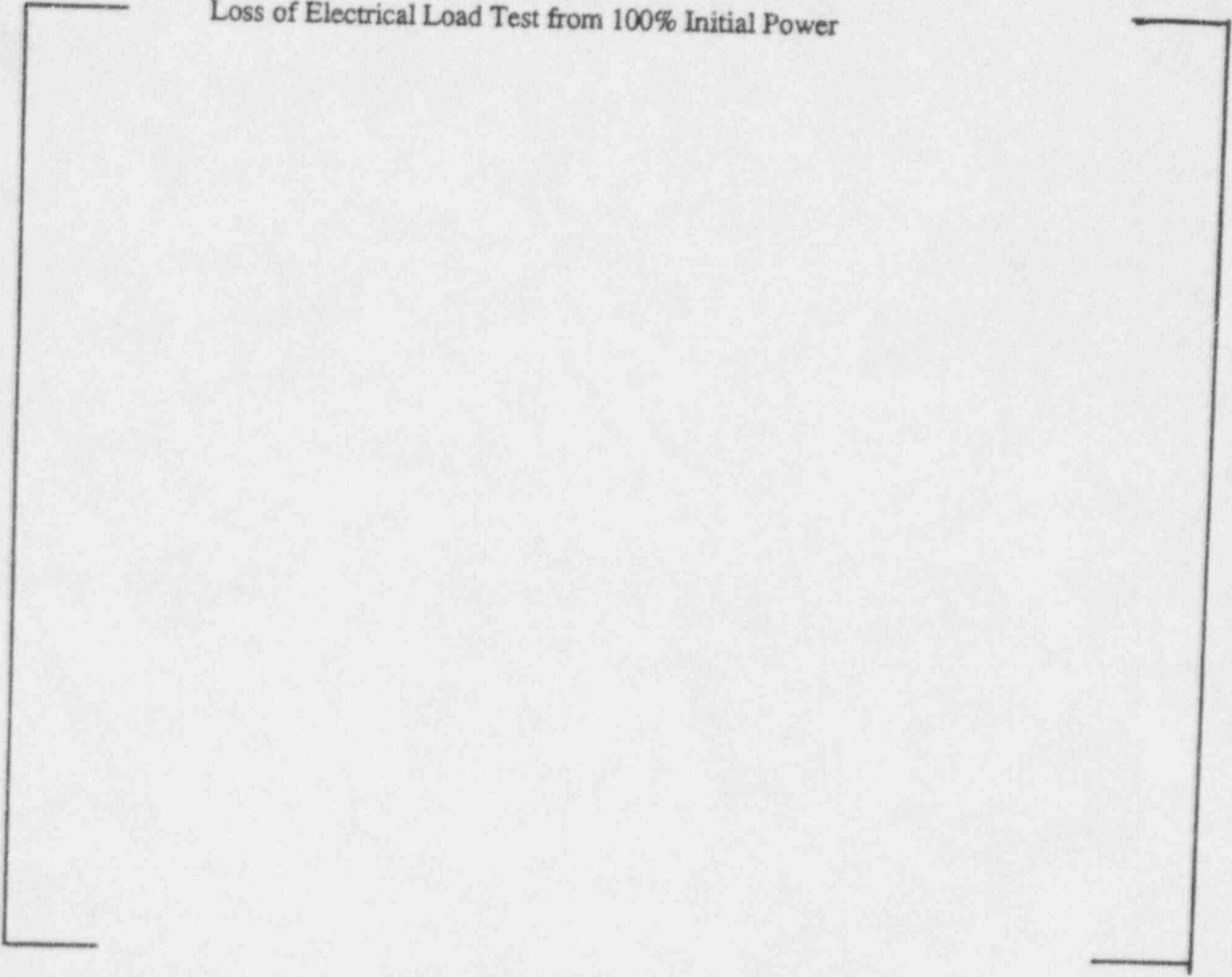


Figure 2.2.2.F Hot Leg Temperature

Comparison of CENTS to RELAP5/MOD3
Feedline Break Event for Plant C

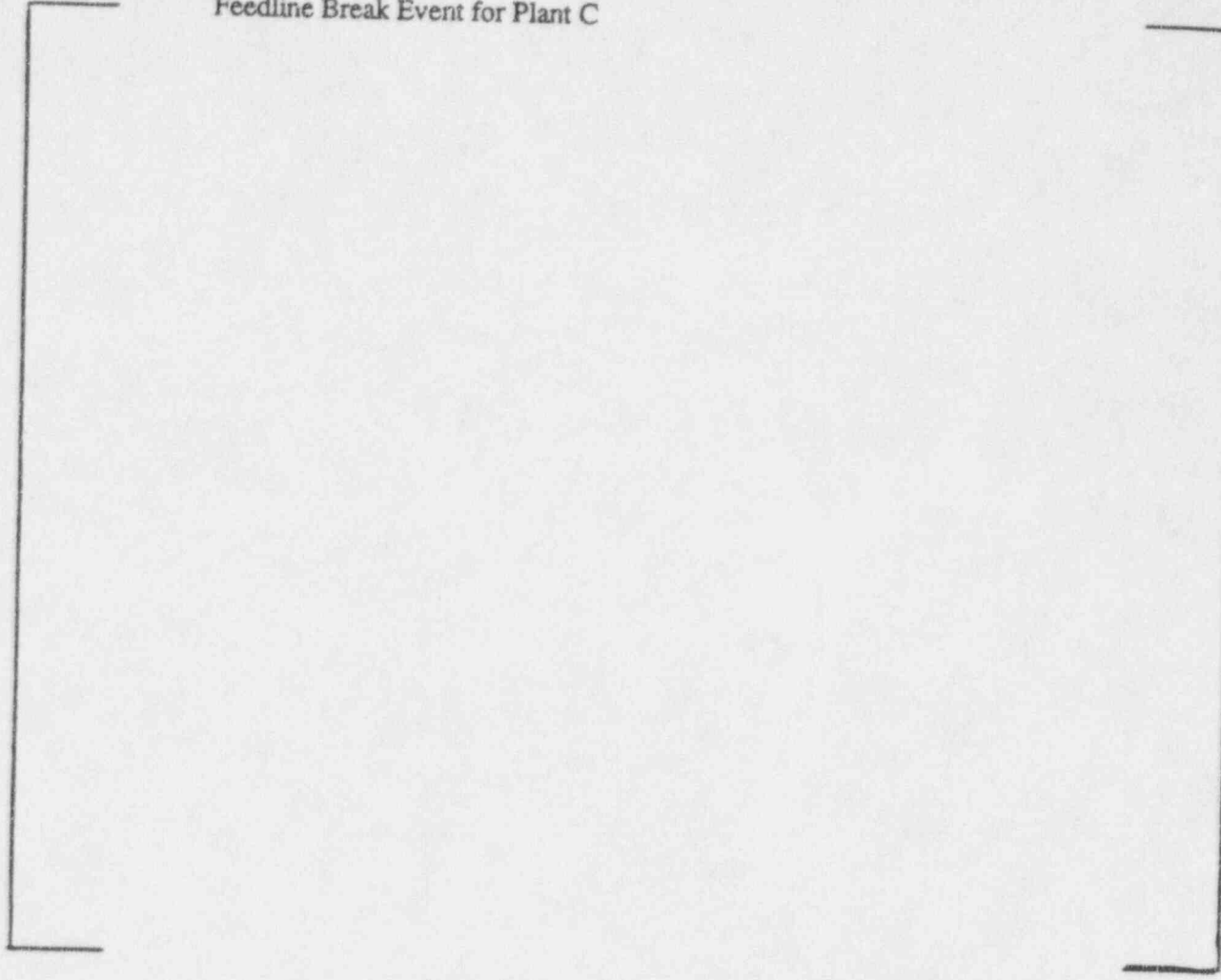


Figure 2.3.1.A Normalized Core Power (180 sec)

Comparison of CENTS to RELAP5/MOD3
Feedline Break Event for Plant C

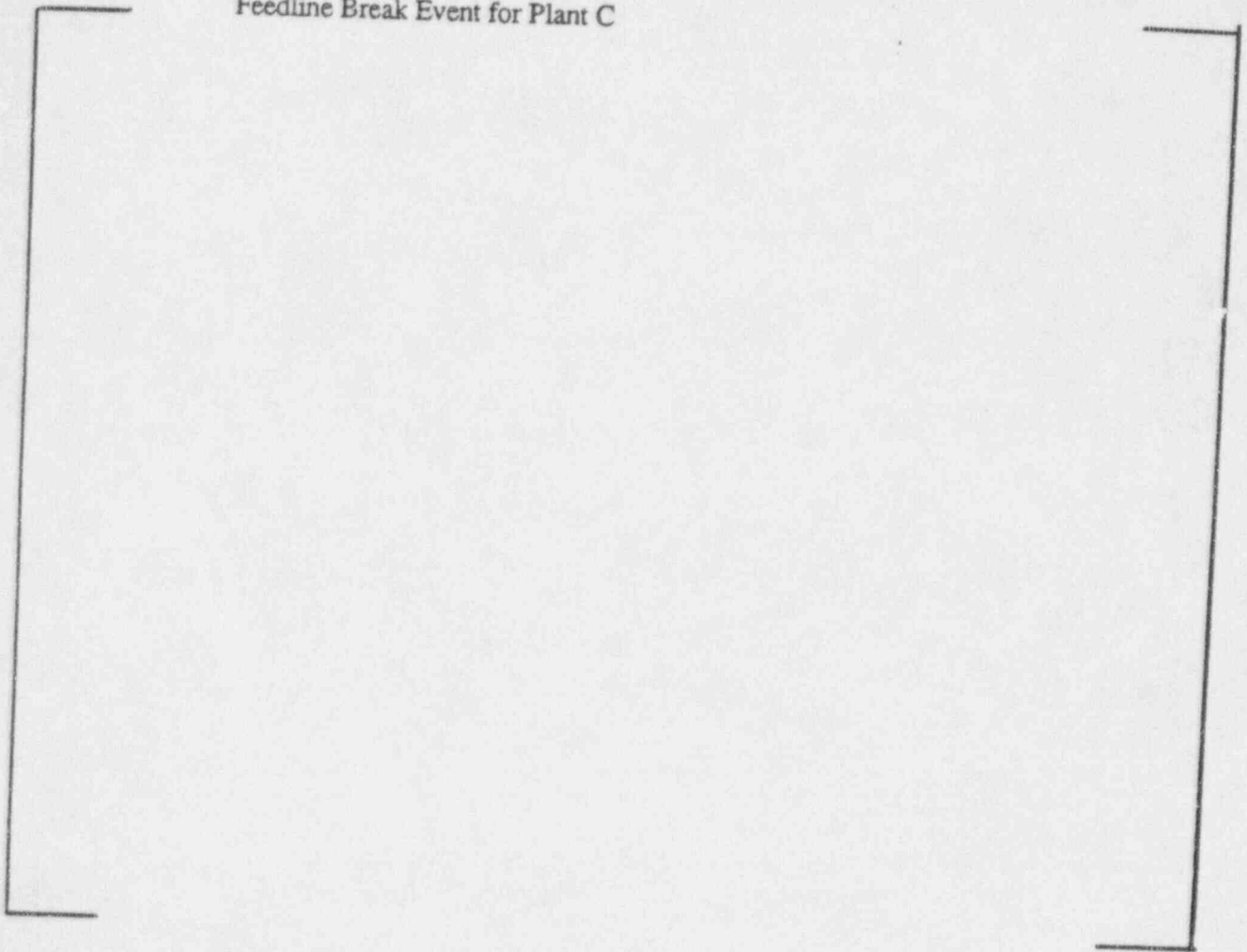


Figure 2.3.1.B Normalized Core Power (3000 sec)

Comparison of CENTS to RELAP5/MOD3
Feedline Break Event for Plant C

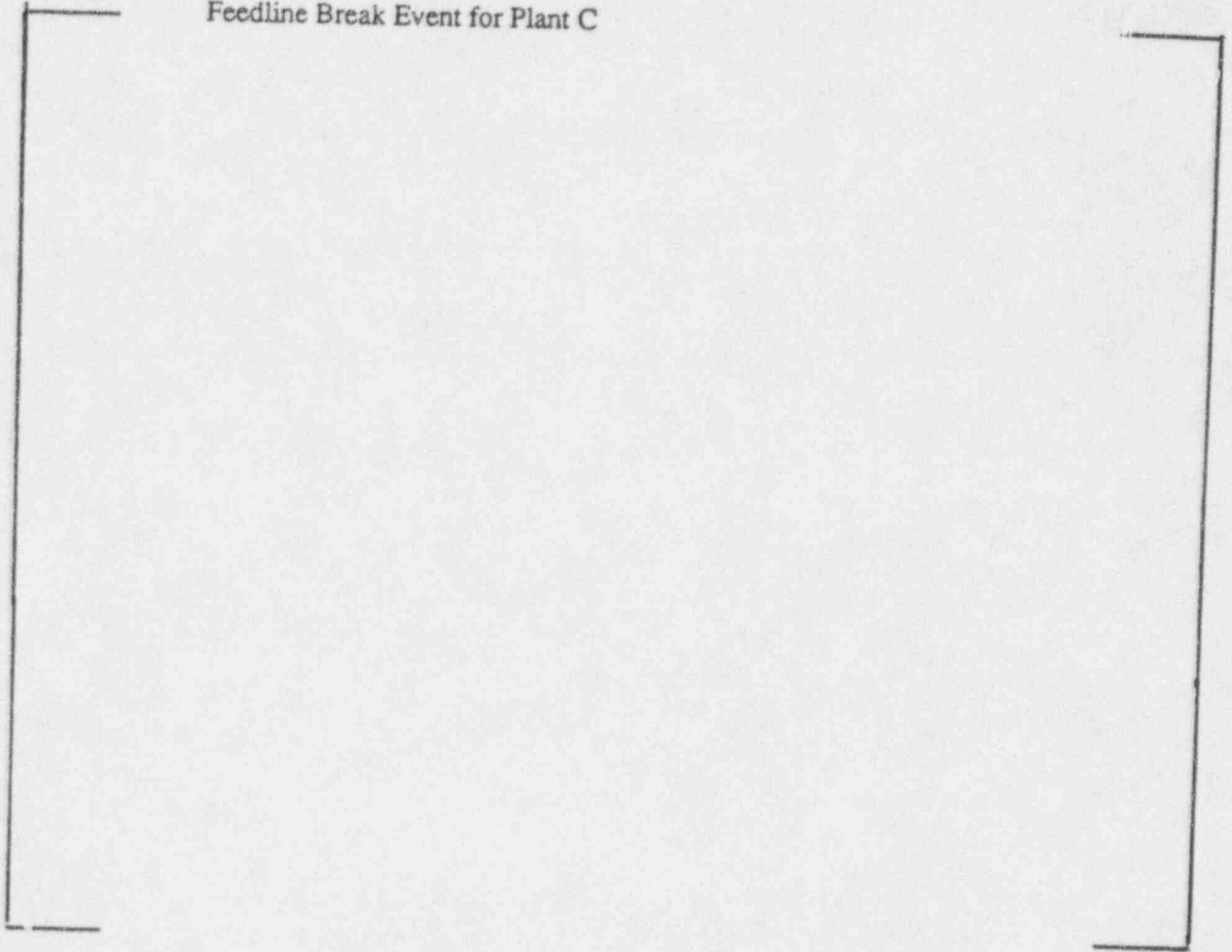


Figure 2.3.1.C Pressurizer Pressure (180 sec)

Comparison of CENTS to RELAP5/MOD3
Feedline Break Event for Plant C

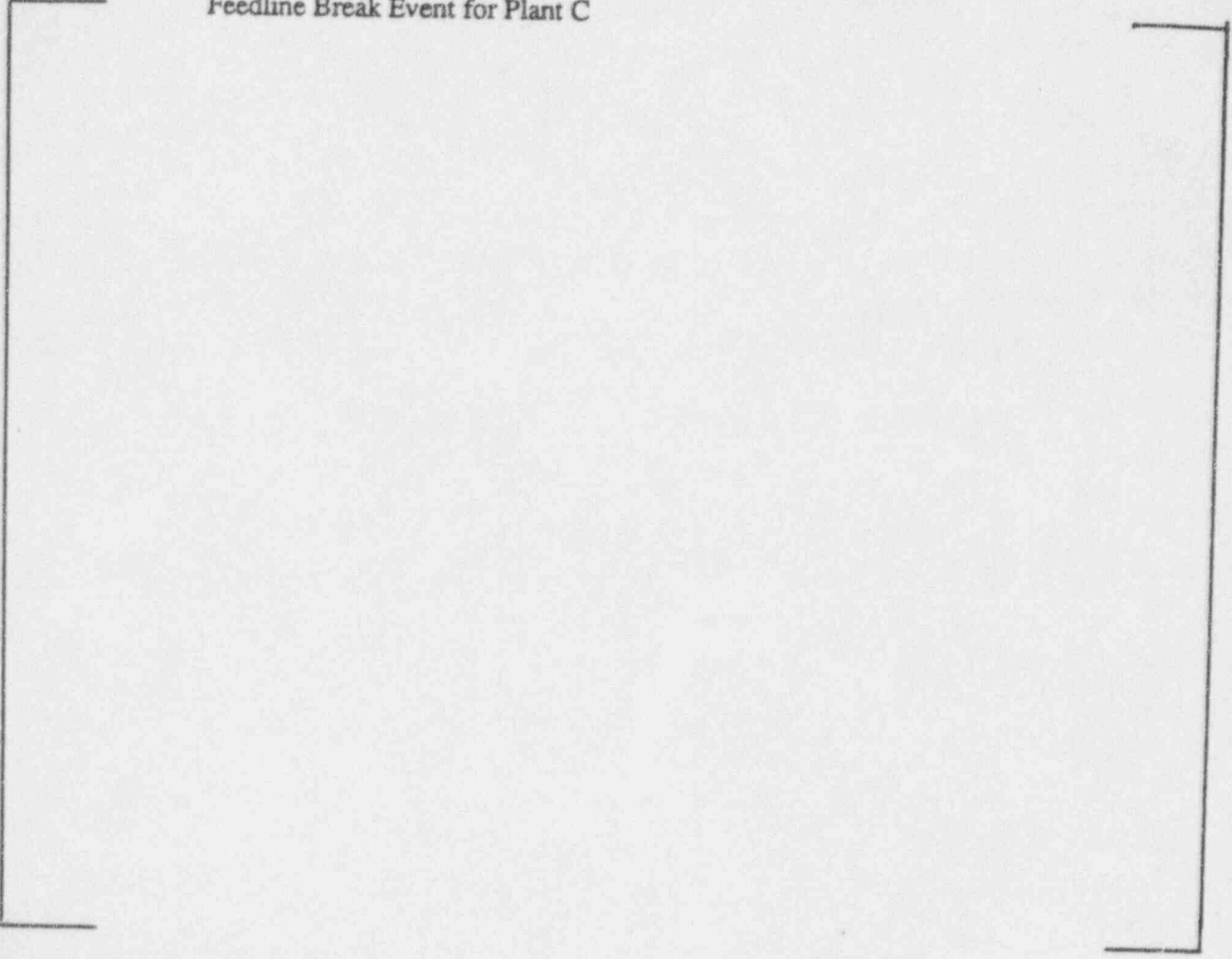


Figure 2.3.1.D Pressurizer Pressure (3000 sec)

Comparison of CENTS to RELAP5/MOD3
Feedline Break Event for Plant C

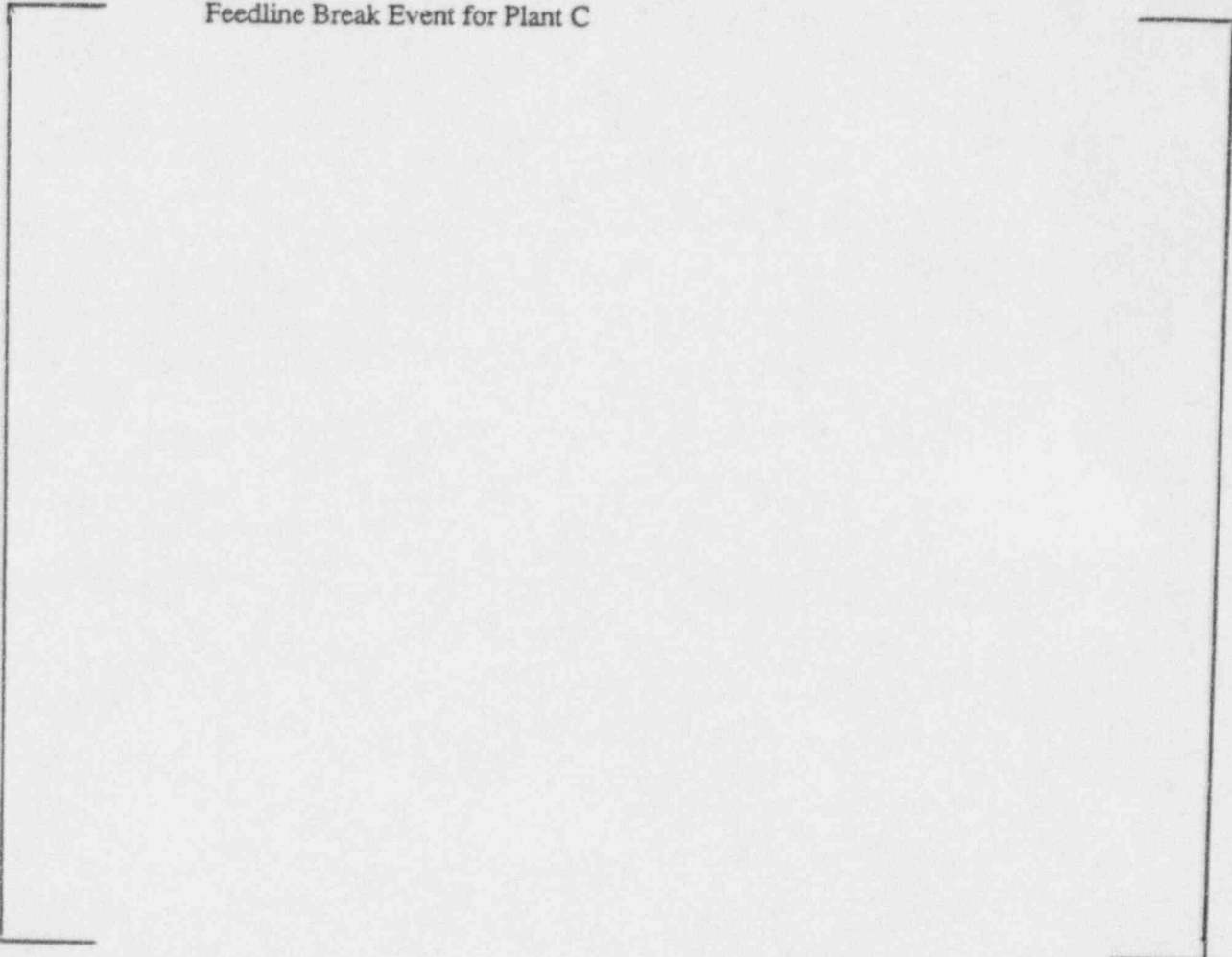


Figure 2.3.1.E Steam Generator Pressure (180 sec)

Comparison of CENTS to RELAP5/MOD3
Feedline Break Event for Plant C

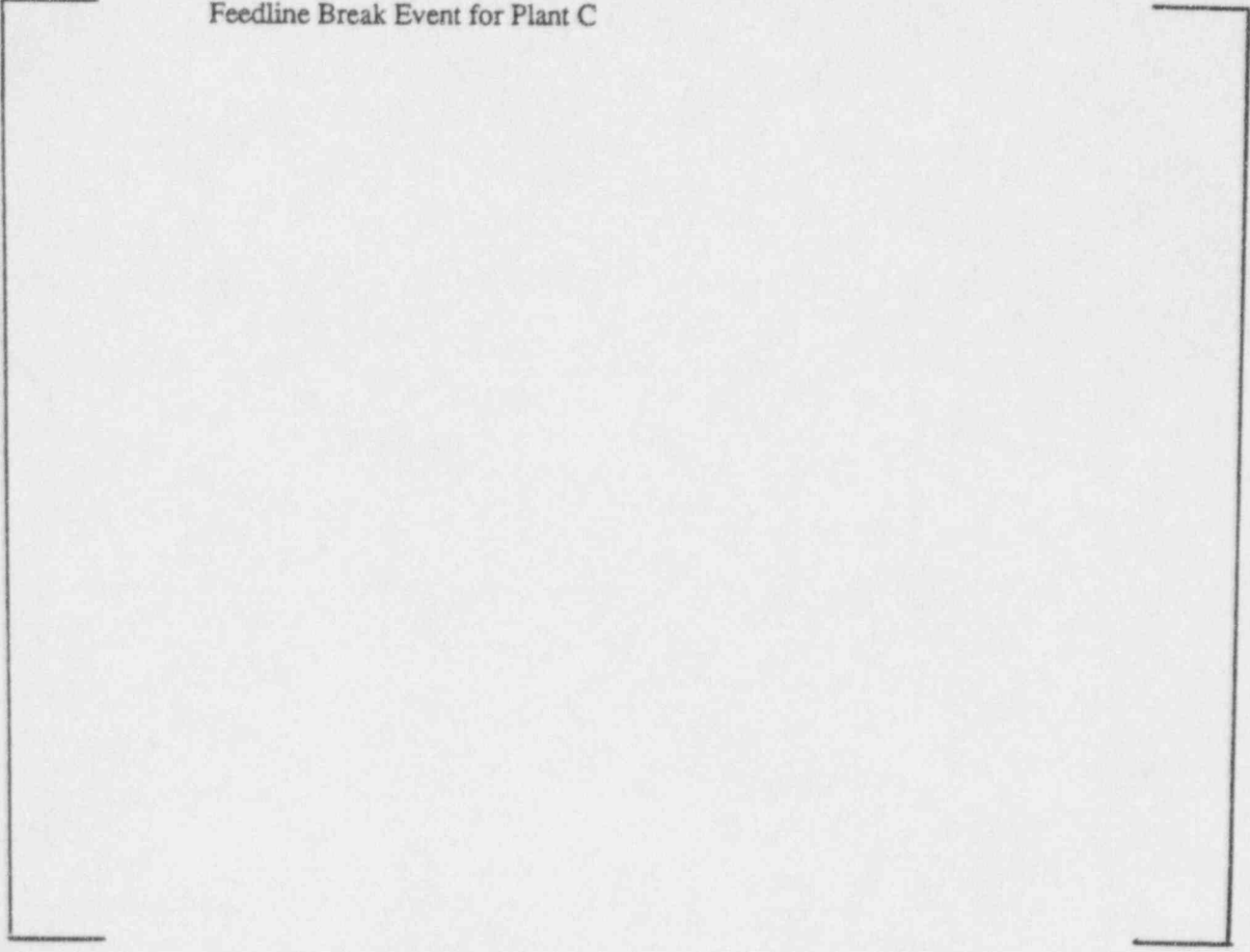


Figure 2.3.1.F Steam Generator Pressure (3000 sec)

Comparison of CENTS to RELAP5/MOD3
Feedline Break Event for Plant C

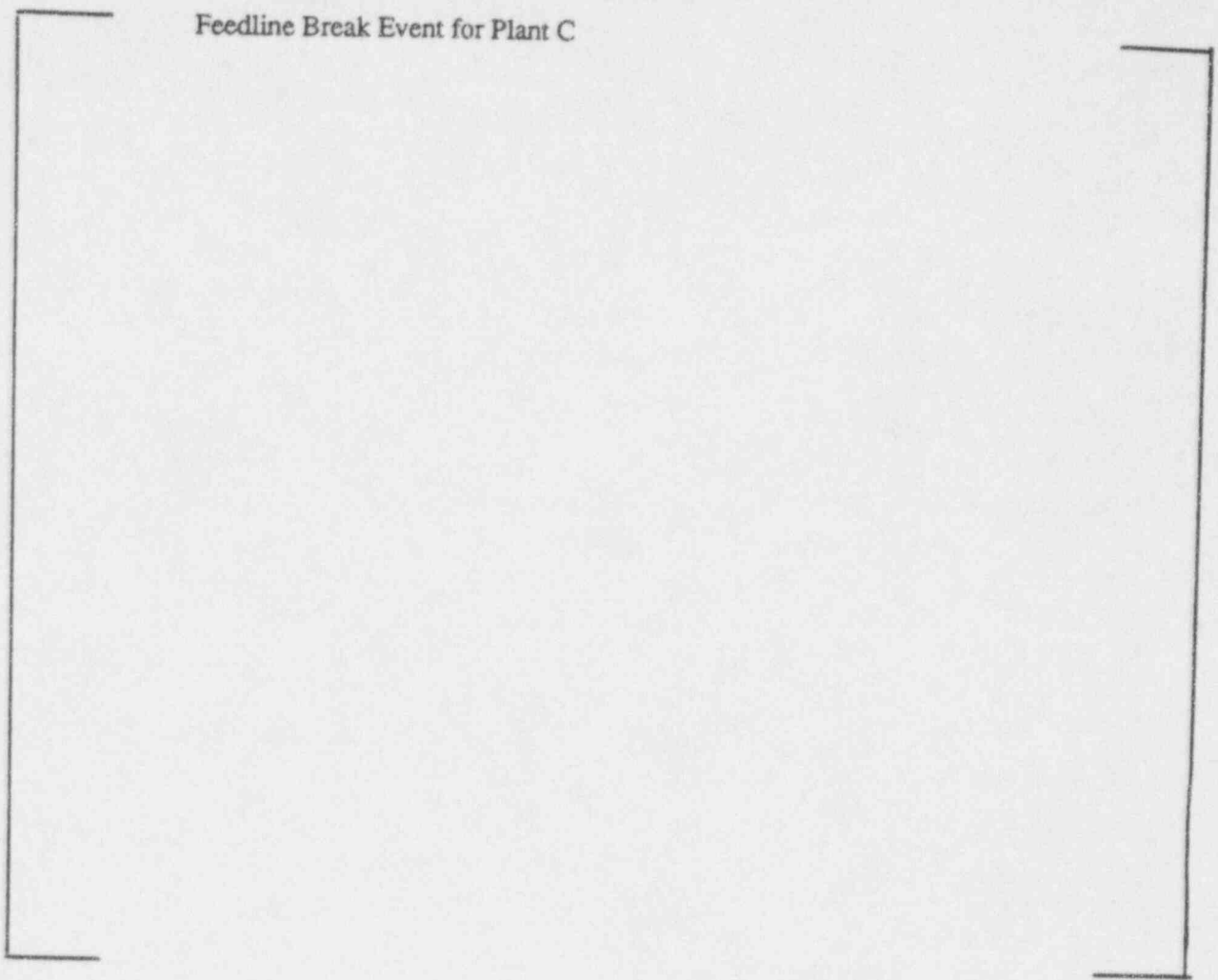


Figure 2.3.1.G Hot Leg Temperature - Affected Loop (180 sec)

Comparison of CENTS to RELAP5/MOD3
Feedline Break Event for Plant C

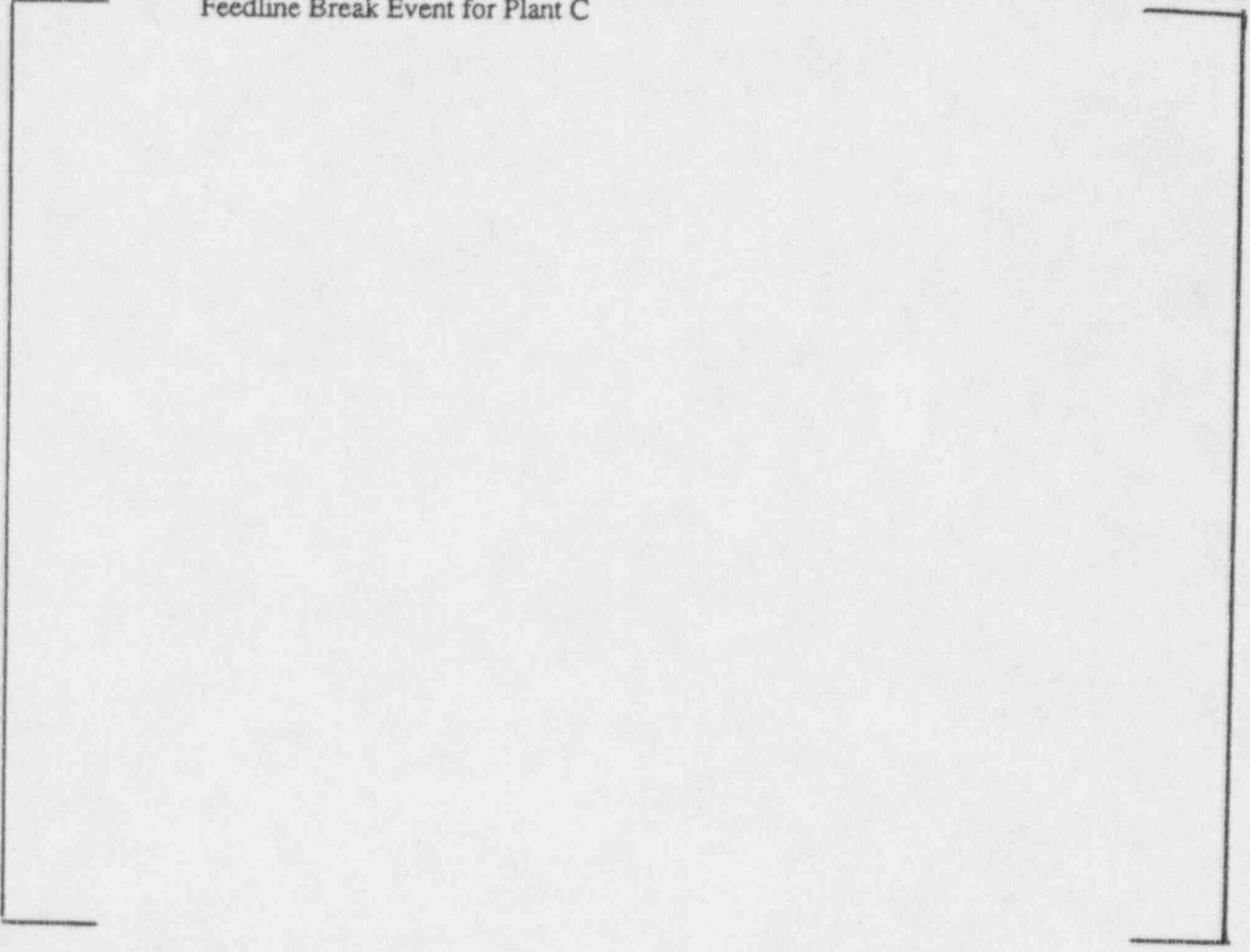


Figure 2.3.1.H Hot Leg Temperature - Affected Loop (3000 sec)

Comparison of CENTS to RELAP5/MOD3
Feedline Break Event for Plant C

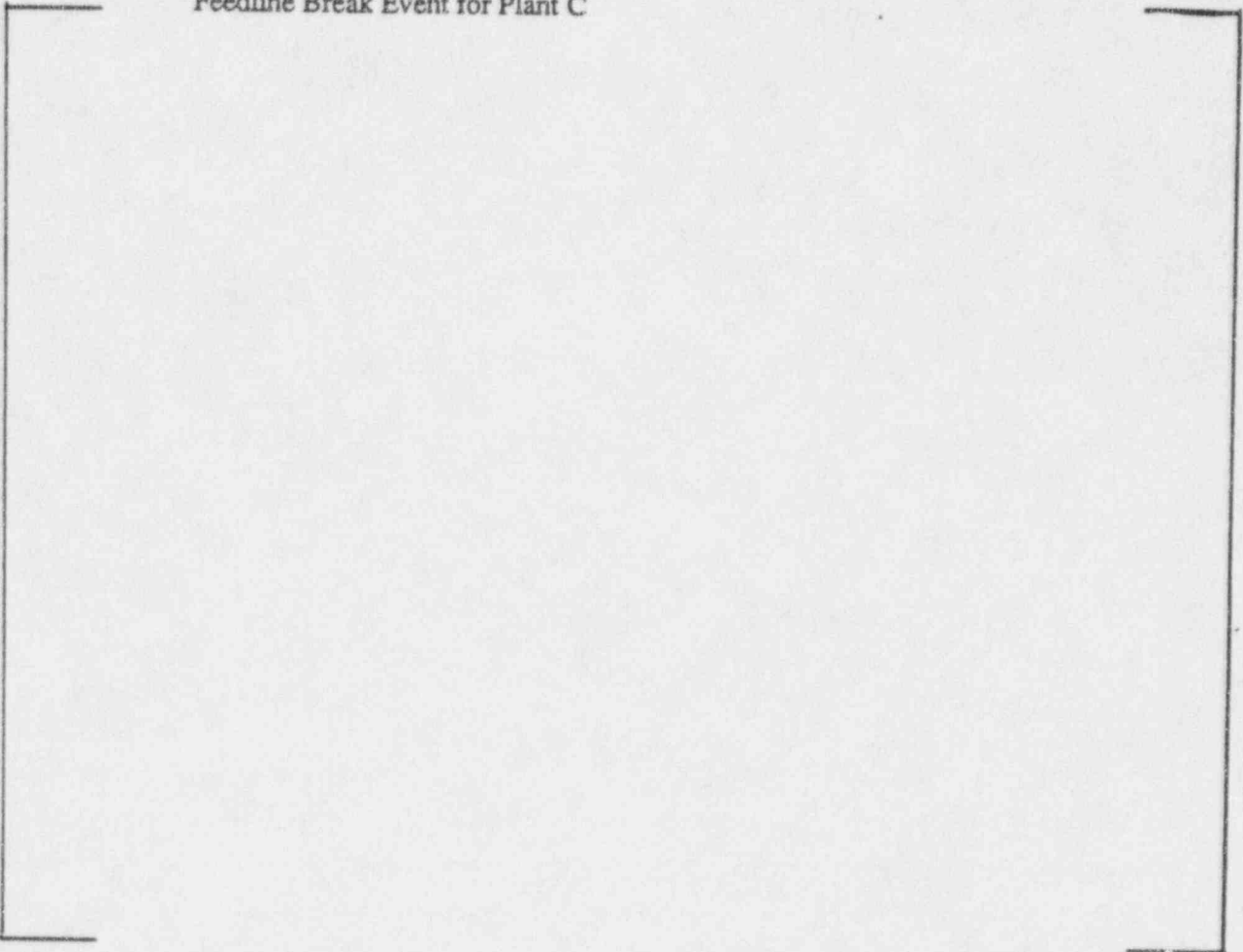


Figure 2.3.1.I Hot Leg Temperature - Intact Loop (180 sec)

Comparison of CENTS to RELAP5/MOD3
Feedline Break Event for Plant C

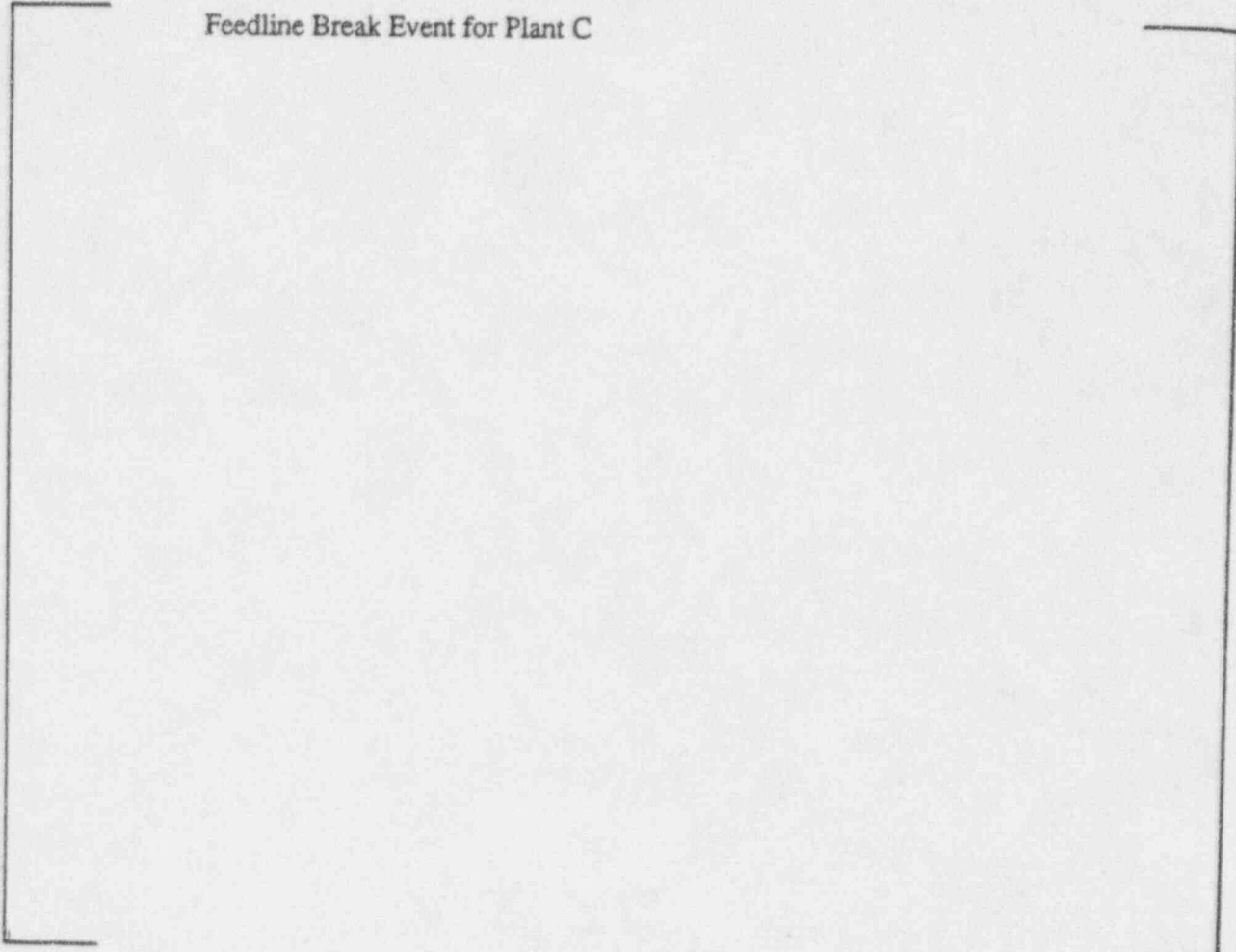


Figure 2.3.1.J Hot Leg Temperature - Intact Loop (3000 sec)

Comparison of CENTS to RELAP5/MOD3
Feedline Break Event for Plant C

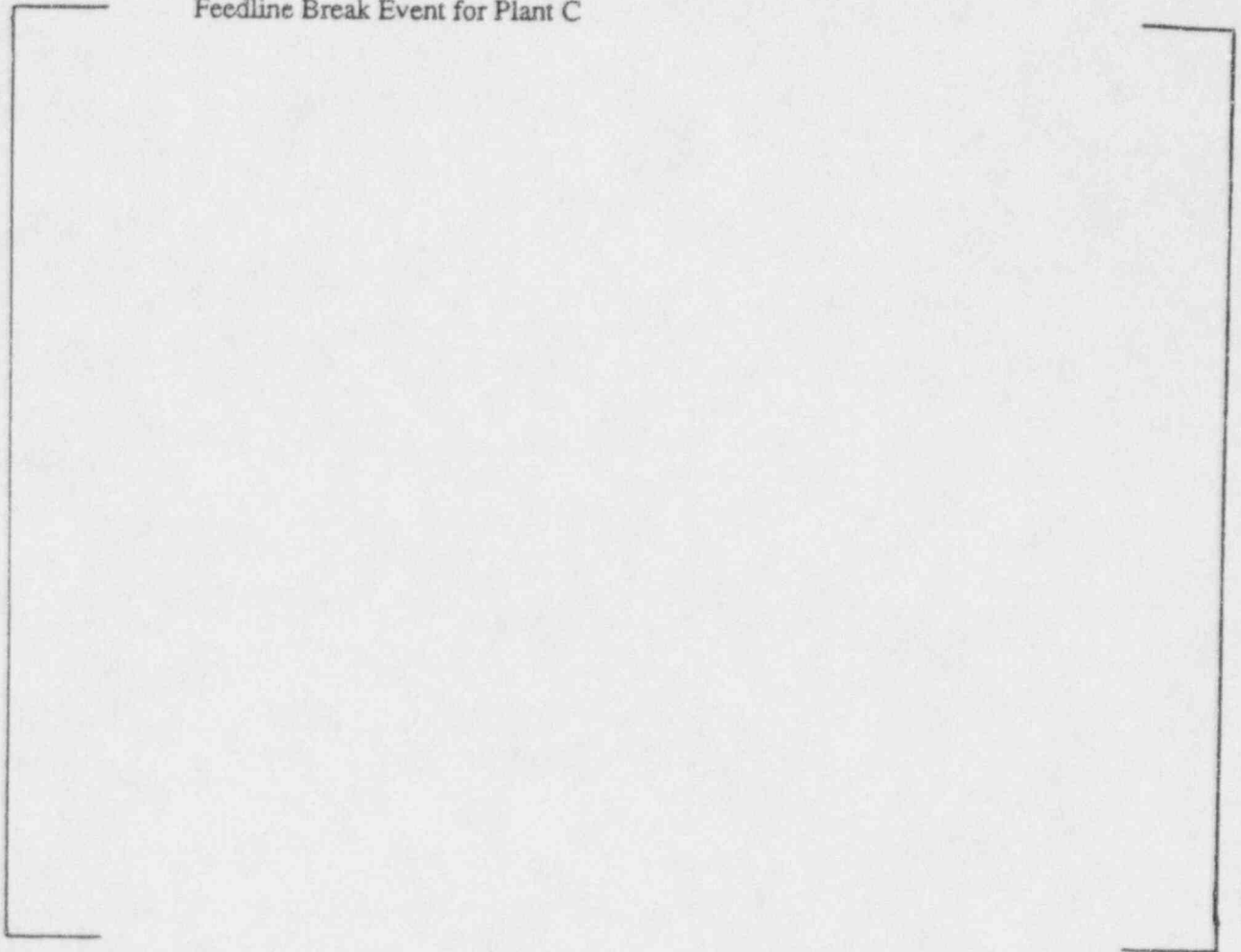


Figure 2.3.1.K Cold Leg Temperature - Affected Loop (180 sec)

Comparison of CENTS to RELAP5/MOD3
Feedline Break Event for Plant C

Figure 2.3.1.L Cold Leg Temperature - Affected Loop (3000 sec)

Comparison of CENTS to RELAP5/MOD3
Feedline Break Event for Plant C

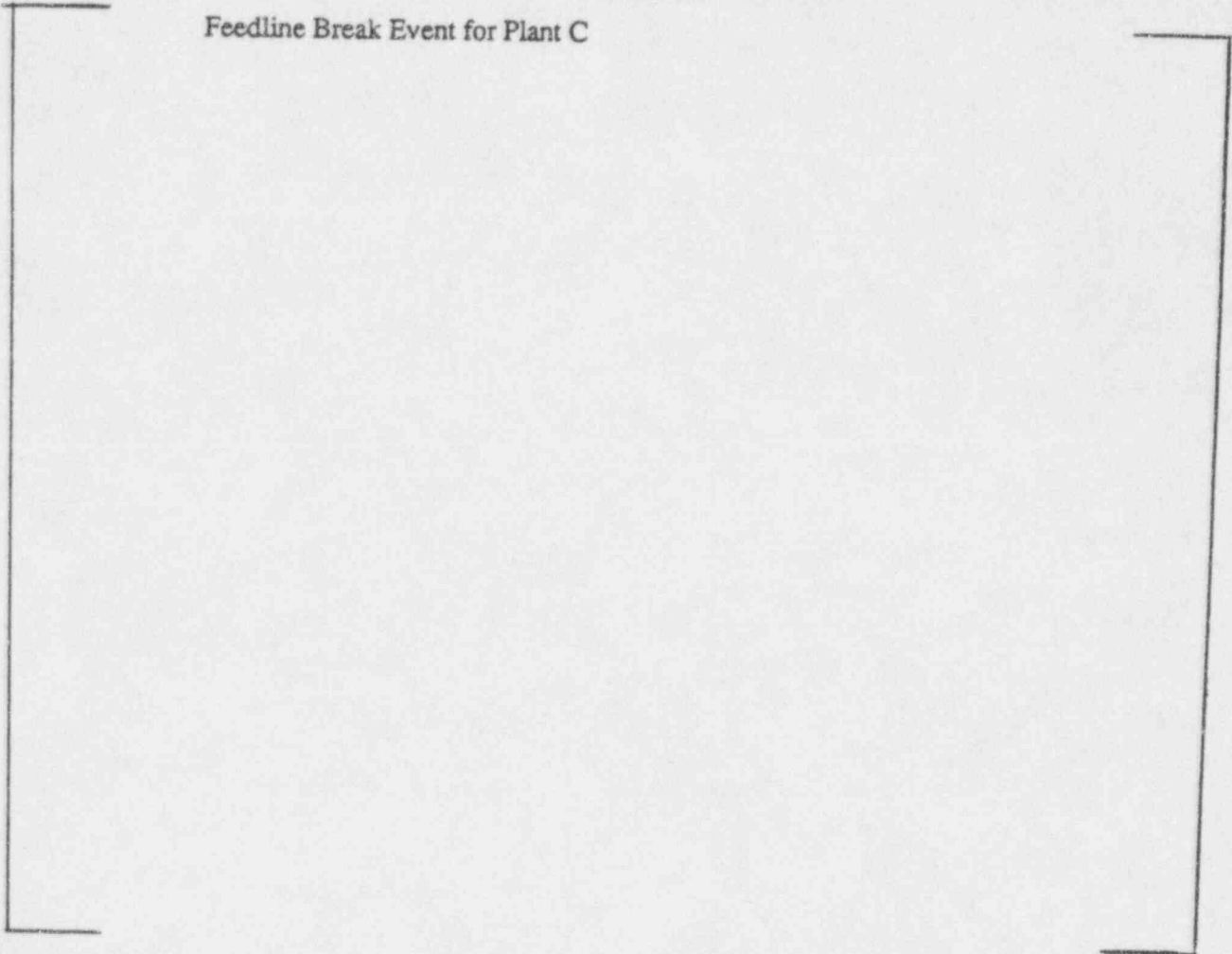
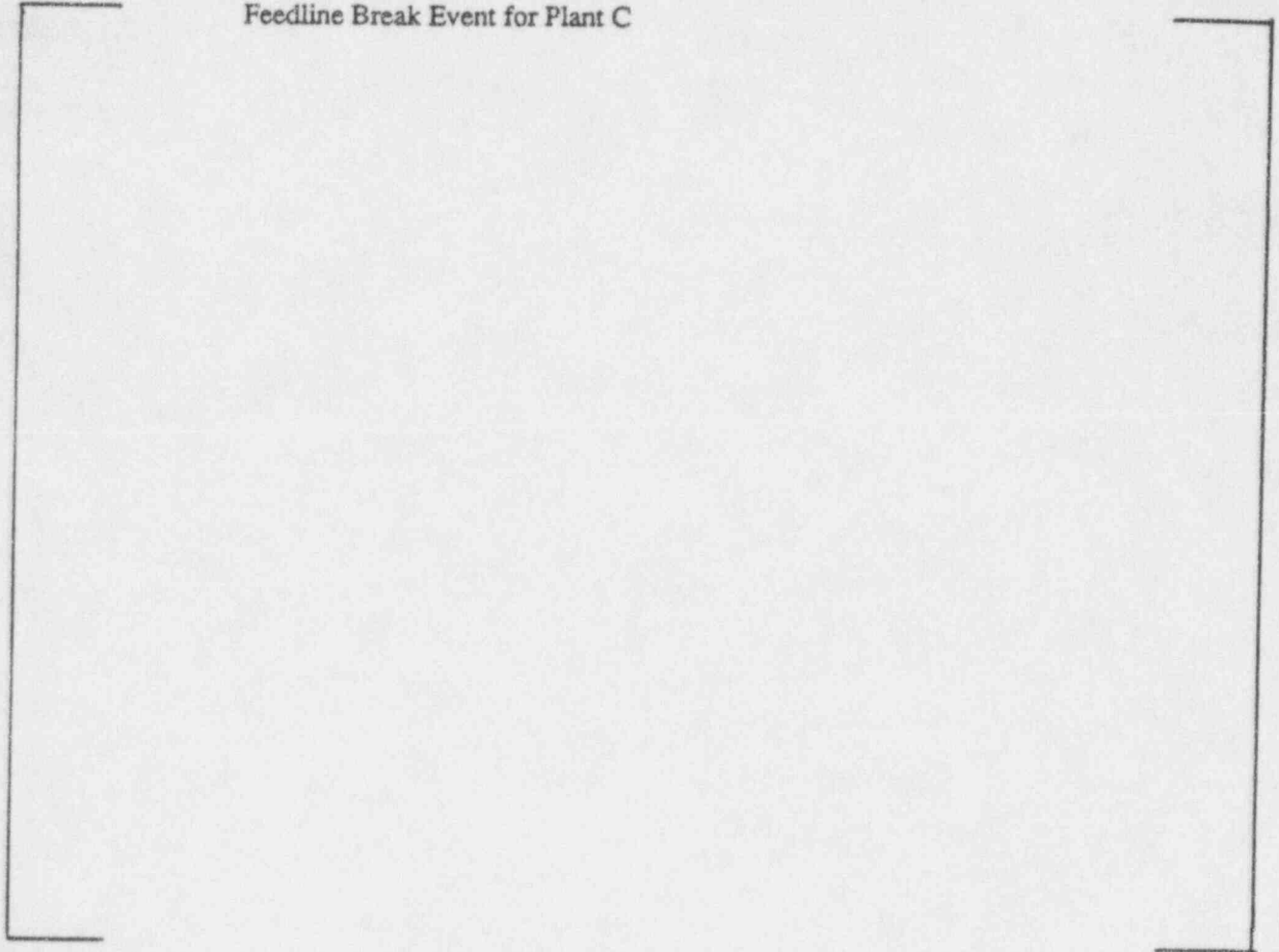


Figure 2.3.1.M Cold Leg Temperature - Intact Loop (180 sec)

Comparison of CENTS to RELAP5/MOD3
Feedline Break Event for Plant C



2

Figure 2.3.1.N Cold Leg Temperature - Intact Loop (3000 sec)

Comparison of CENTS to RELAP5/MOD3
Feedline Break Event for Plant C

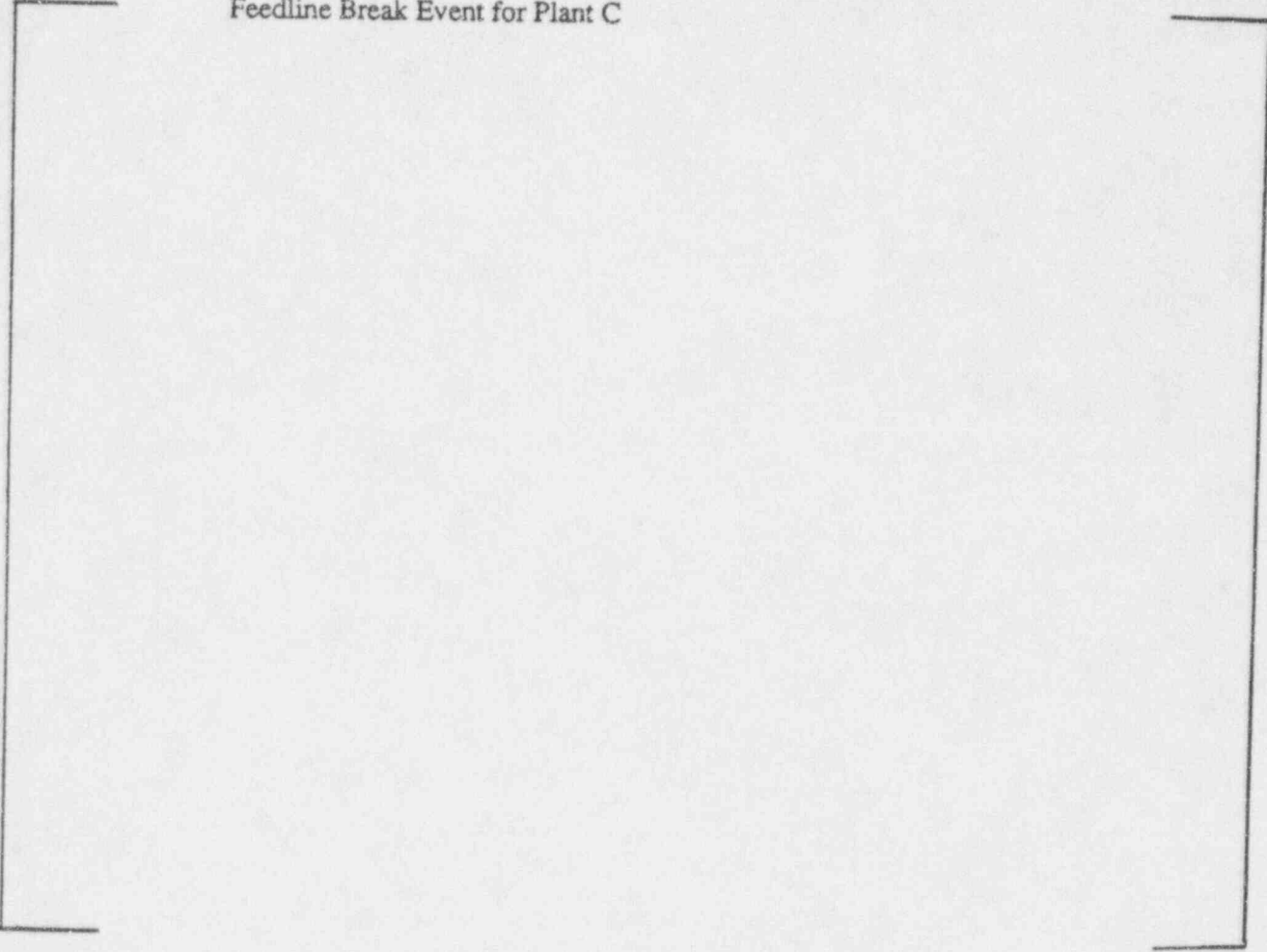


Figure 2.3.1.O Steam Generator Steam Flow (180 sec)

Comparison of CENTS to RELAP5/MOD3
Feedline Break Event for Plant C

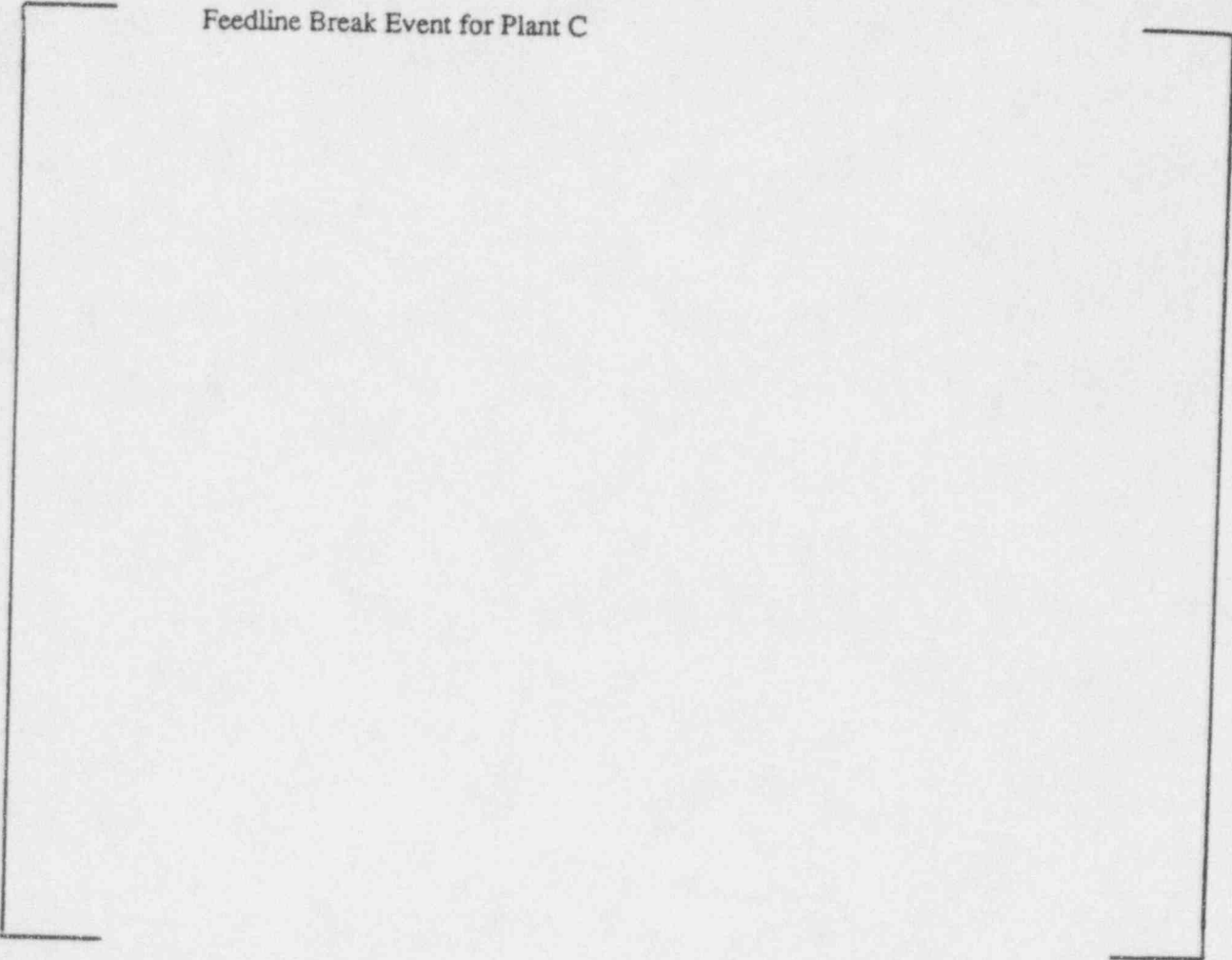


Figure 2.3.1.P Steam Generator Steam Flow (3000 sec)

Comparison of CENTS to RELAP5/MOD3
Feedline Break Event for Plant C

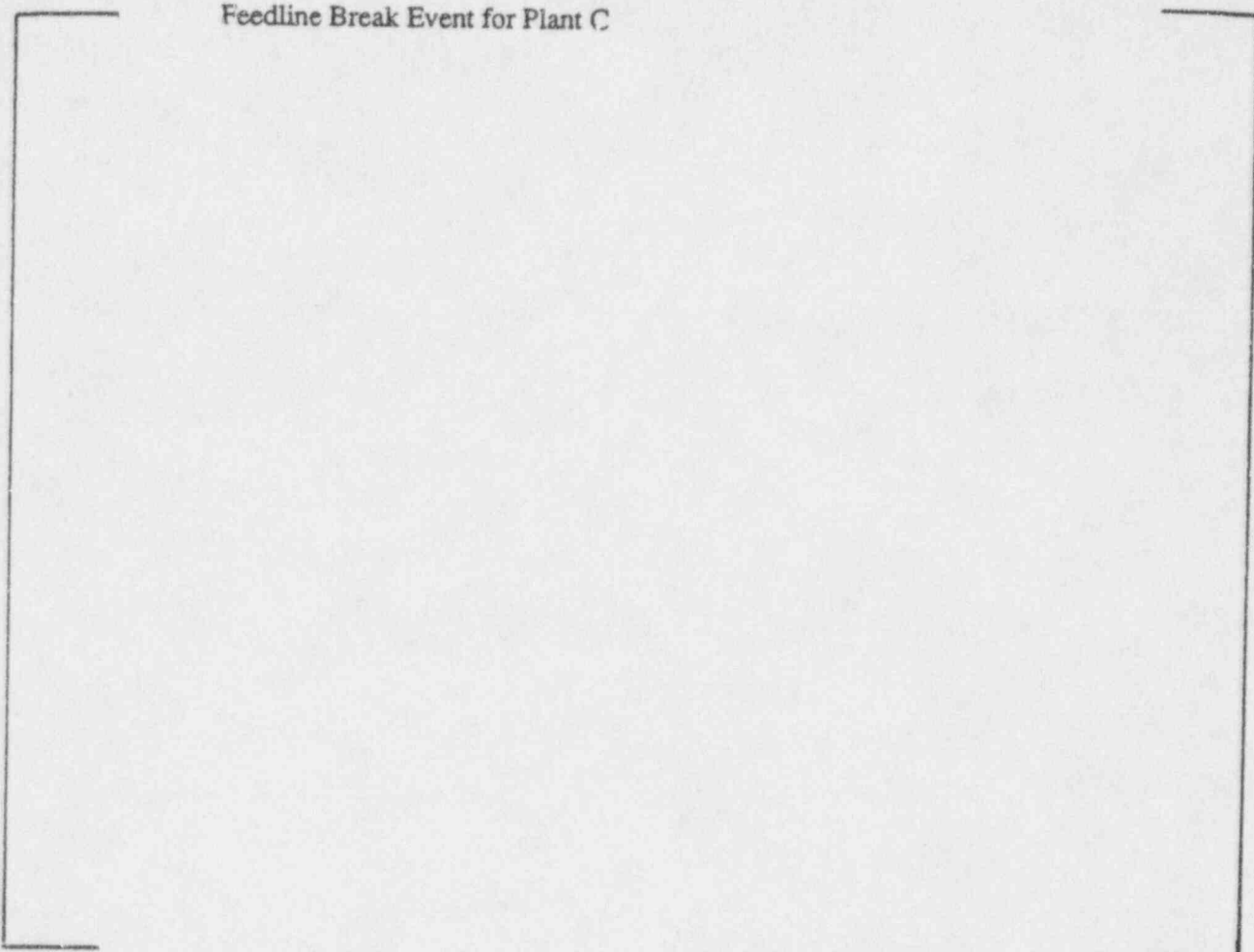


Figure 2.3.1.Q Break Flow (180 sec)

Comparison of CENTS to RELAP5/MOD3
Feedline Break Event for Plant C

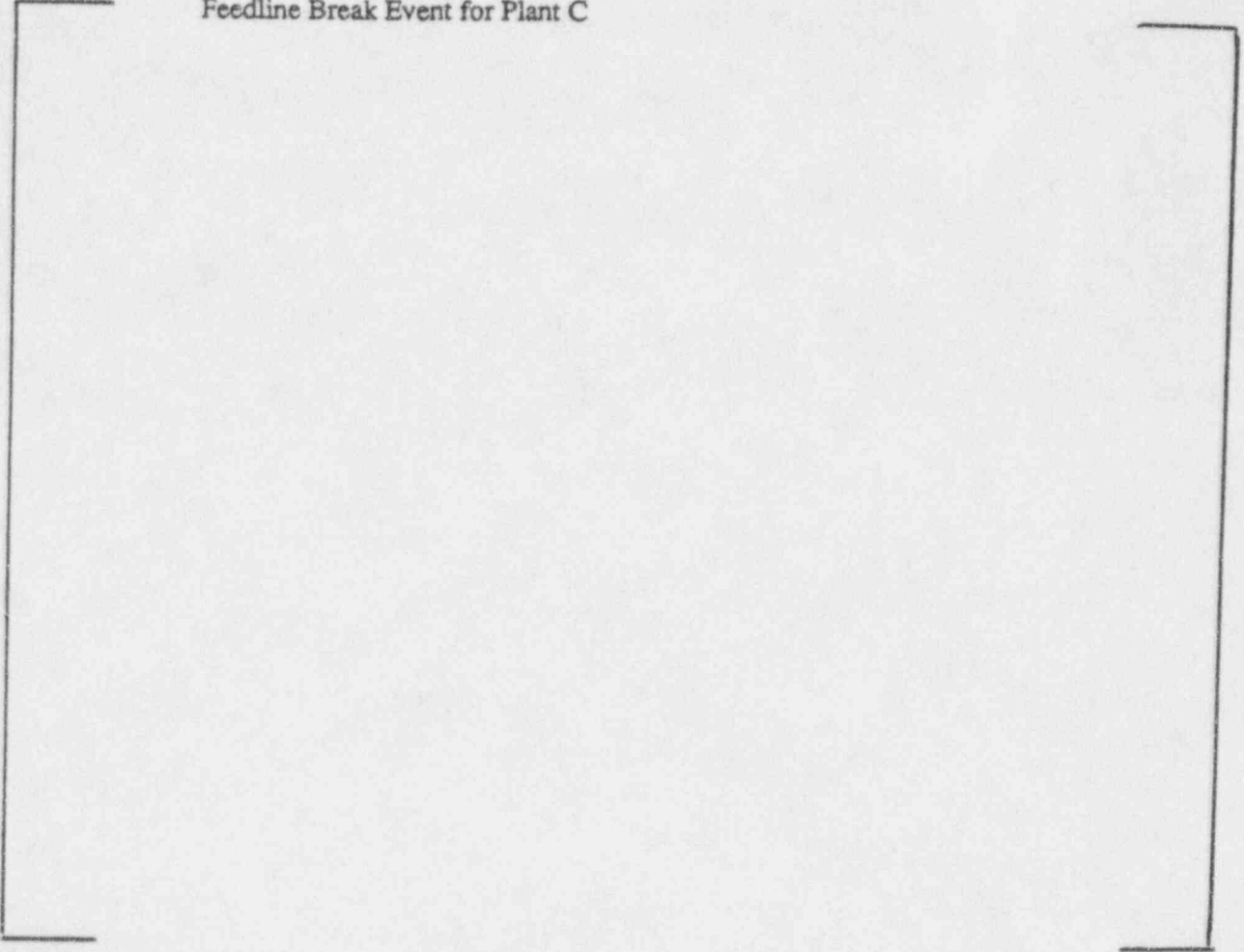


Figure 2.3.1.R Integrated Break Flow (3000 sec)

Comparison of CENTS to RELAP5/MOD3
Feedline Break Event for Plant C

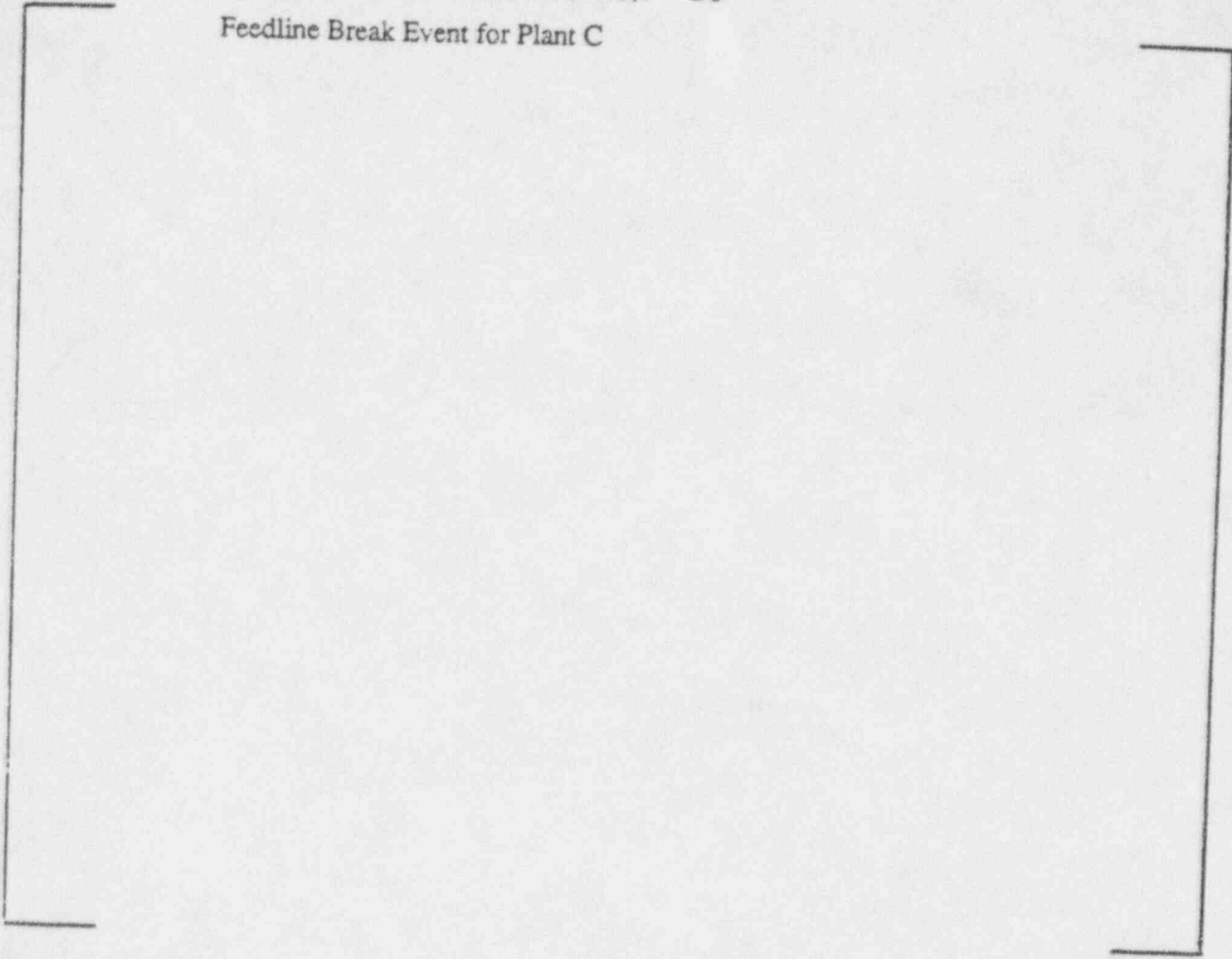


Figure 2.3.1.S Total HPSI Flow (3000 sec)

Comparison of CENTS to RELAP5/MOD3
Feedline Break Event for Plant C

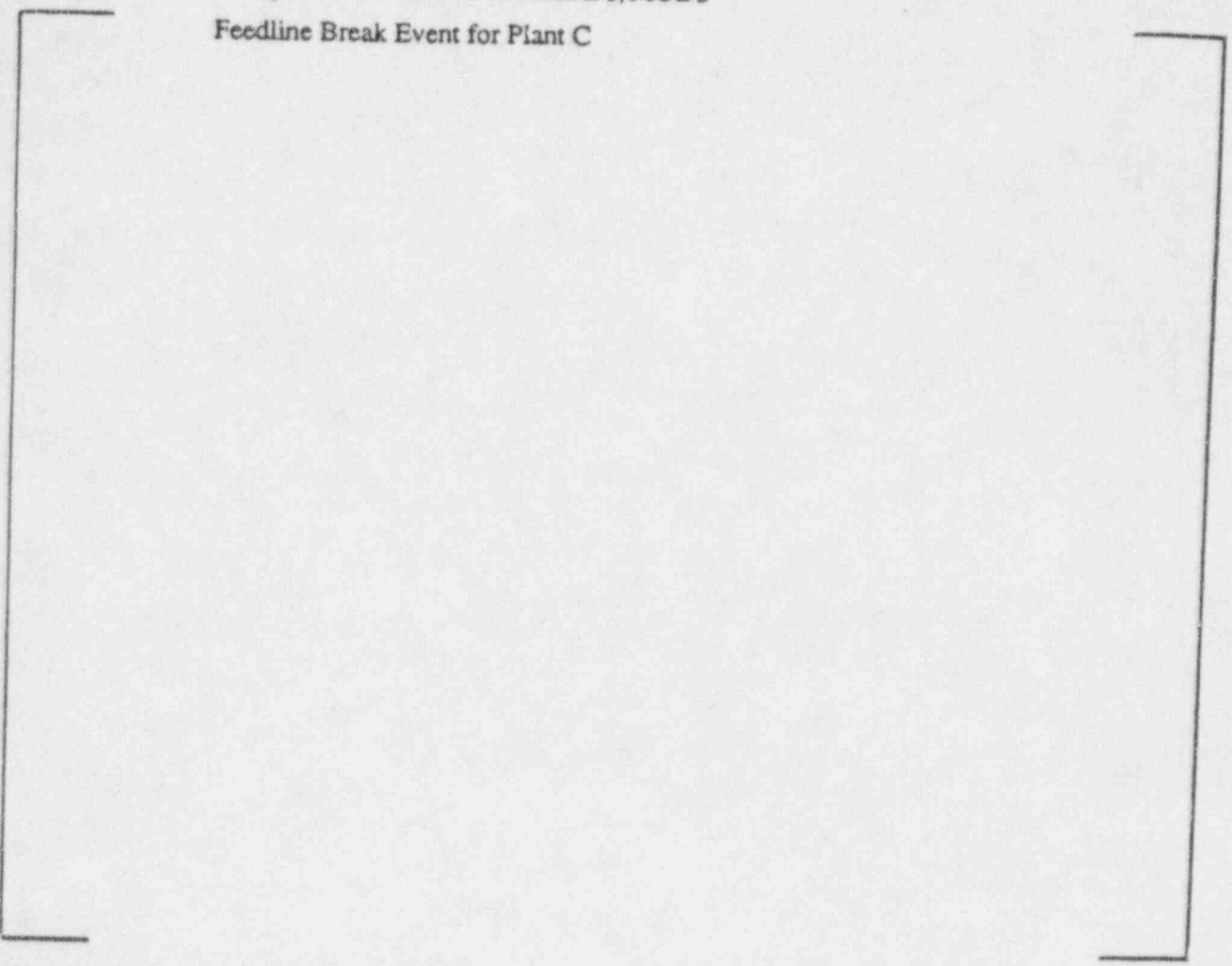


Figure 2.3.1.T Pressurizer Water Level (3000 sec)

Comparison of CENTS to RELAP5/MOD3
Feedline Break Event for Plant C

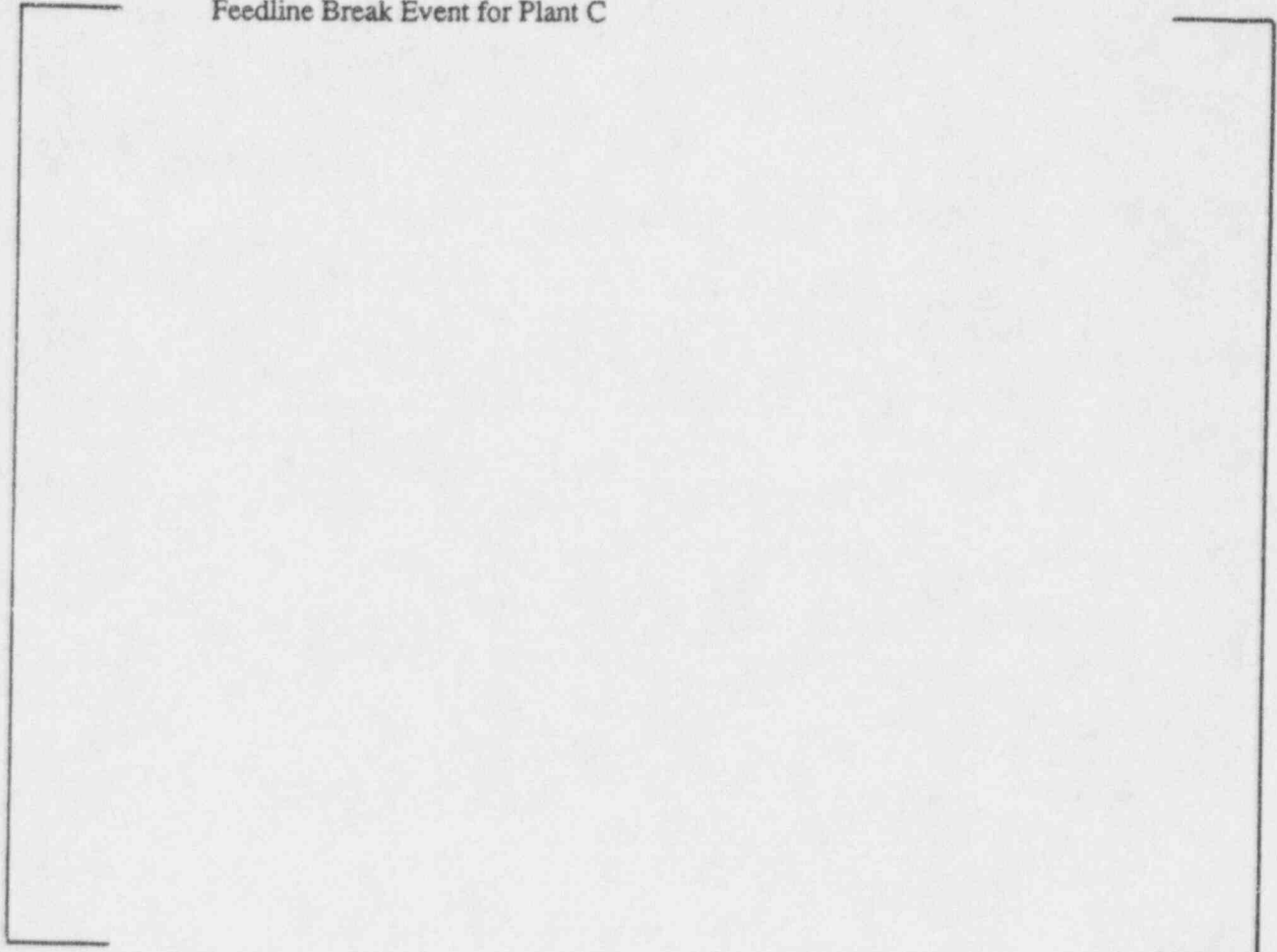


Figure 2.3.1.U Fuel Average Temperature (180 sec)

Comparison of CENTS to RELAP5/MOD3
Feedline Break Event for Plant C

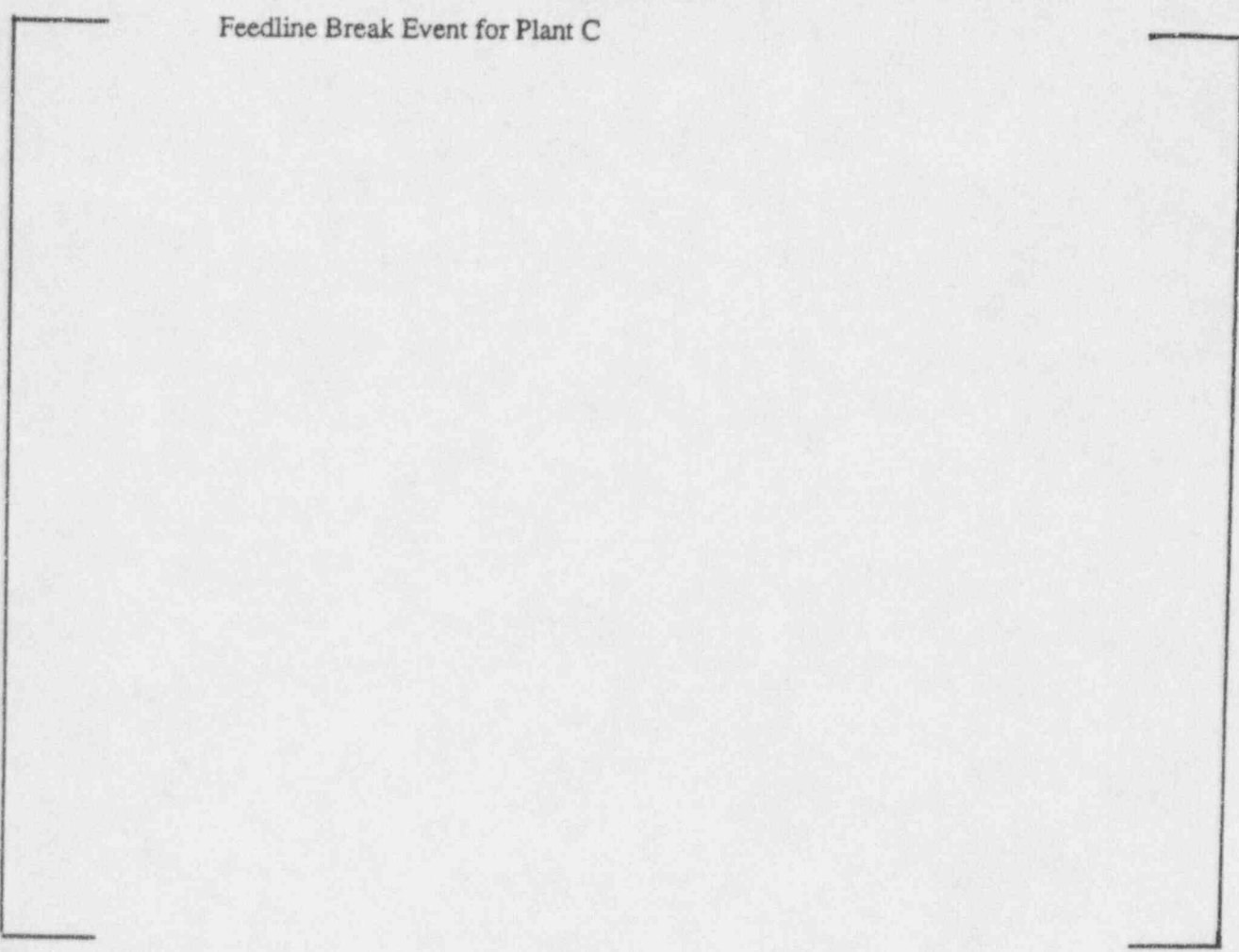


Figure 2.3.1.V Fuel Average Temperature (3000 sec)

Comparison of CENTS to RELAP5/MOD3
Feedline Break Event for Plant C

2

Figure 2.3.1.W Steam Generator Liquid Mass (3000 sec)

Comparison of CENTS to RELAP5/MOD3
Feedline Break Event for Plant C

TIME (SECONDS) $\times 10^2$

Figure 2.3.1.X Pressurizer SV Integrated Flow (3000 sec)

Comparison of CENTS to RELAP5/MOD3
Seized Rotor Event for Plant C

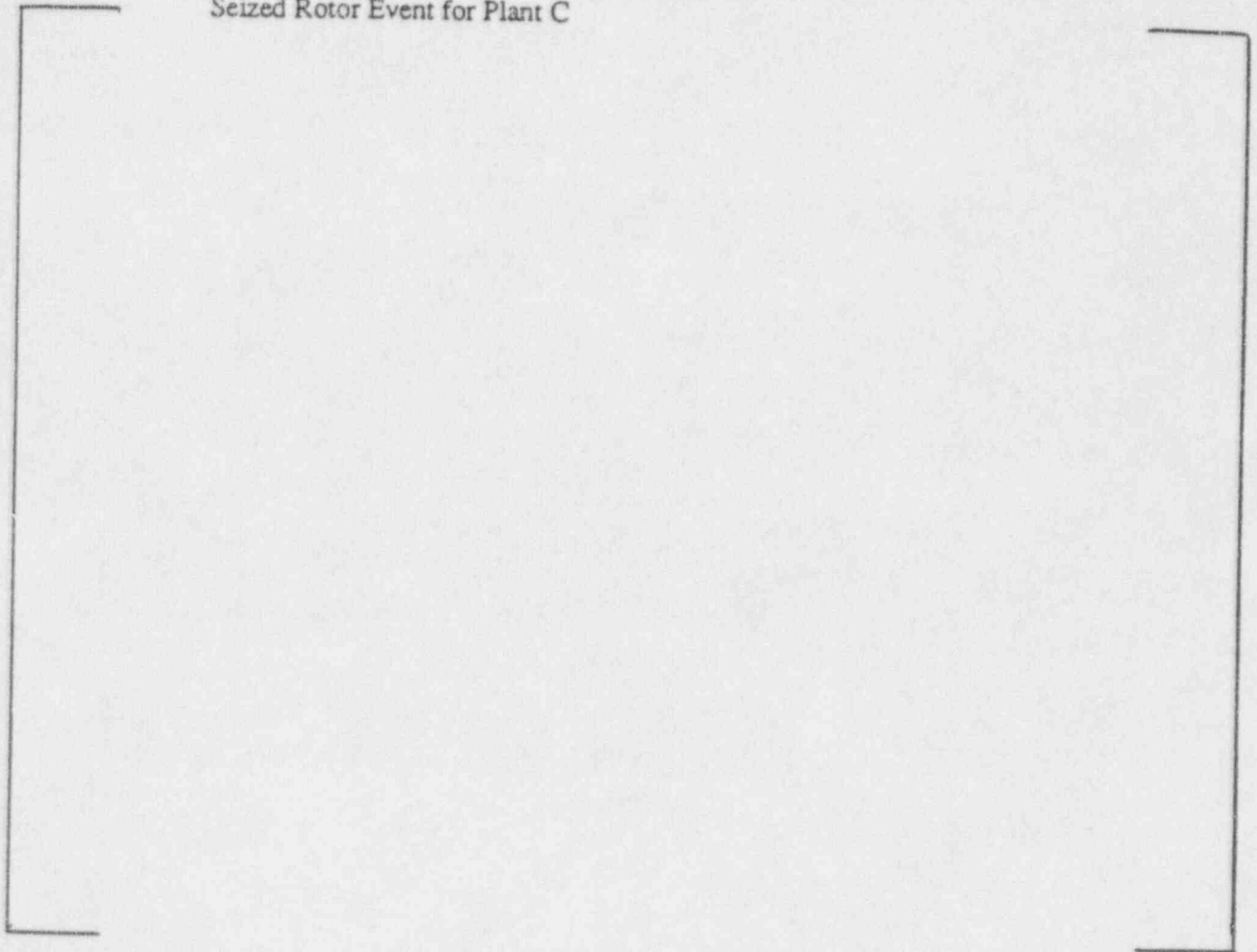


Figure 2.3.2.A Normalized Core Power

Comparison of CENTS to RELAP5/MOD3
Seized Rotor Event for Plant C

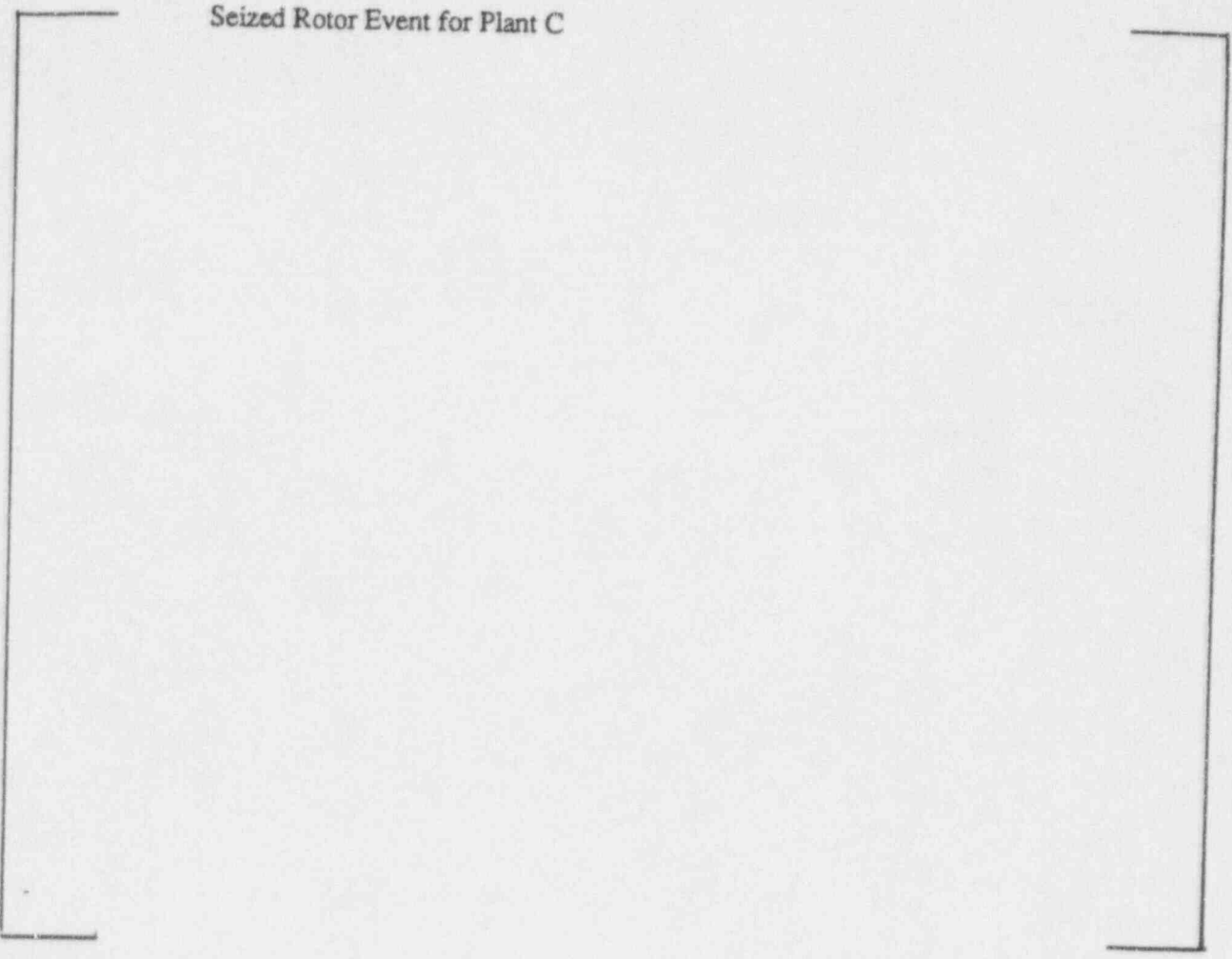


Figure 2.3.2.B Pressurizer Pressure

Comparison of CENTS to RELAP5/MOD3
Seized Rotor Event for Plant C

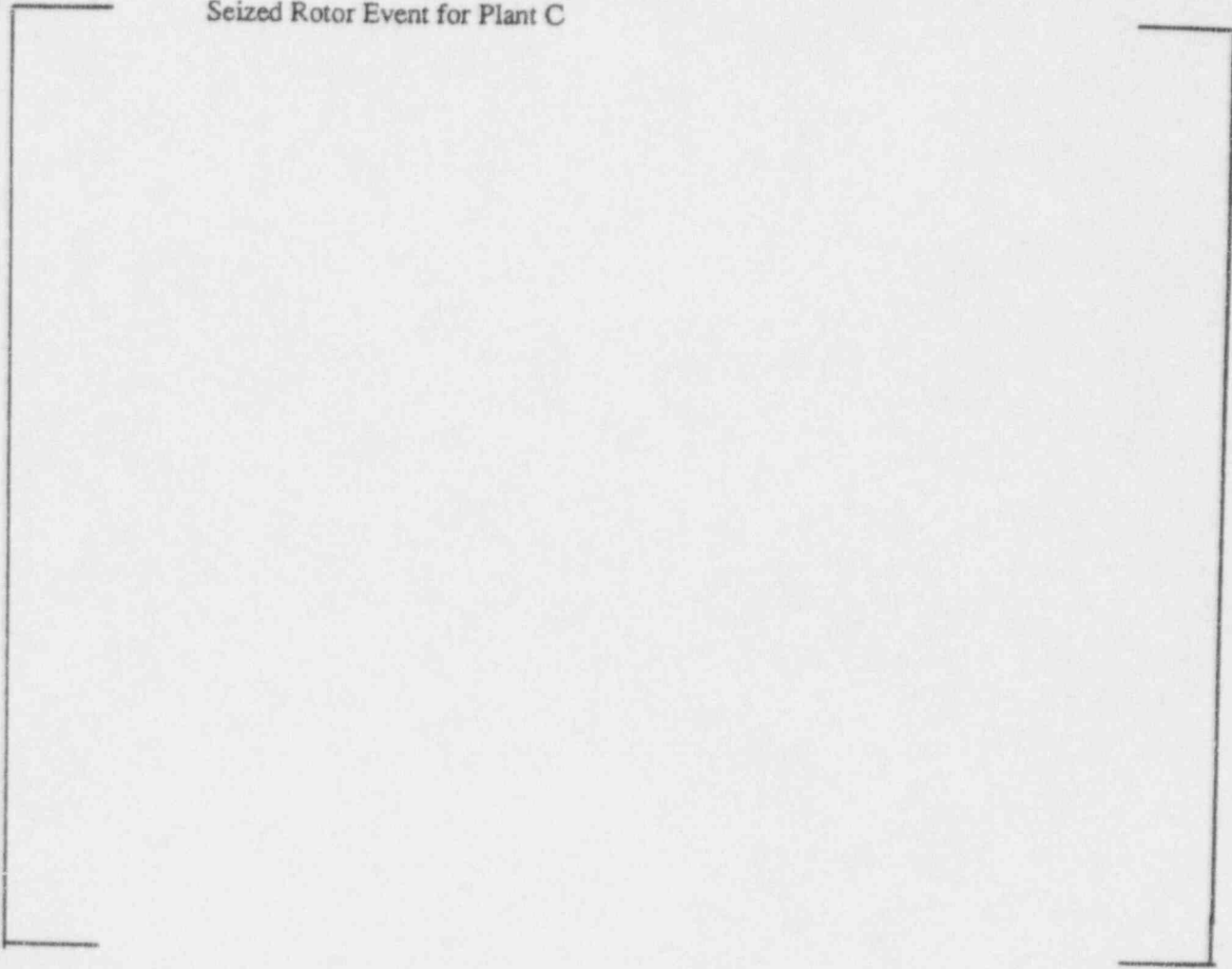


Figure 2.3.2.C Pressurizer Surge Flow

Comparison of CENTS to RELAP5/MOD3
Seized Rotor Event for Plant C

Figure 2.3.2.D RCS Loop Flow

Comparison of CENTS to RELAP5/MOD3
Seized Rotor Event for Plant C

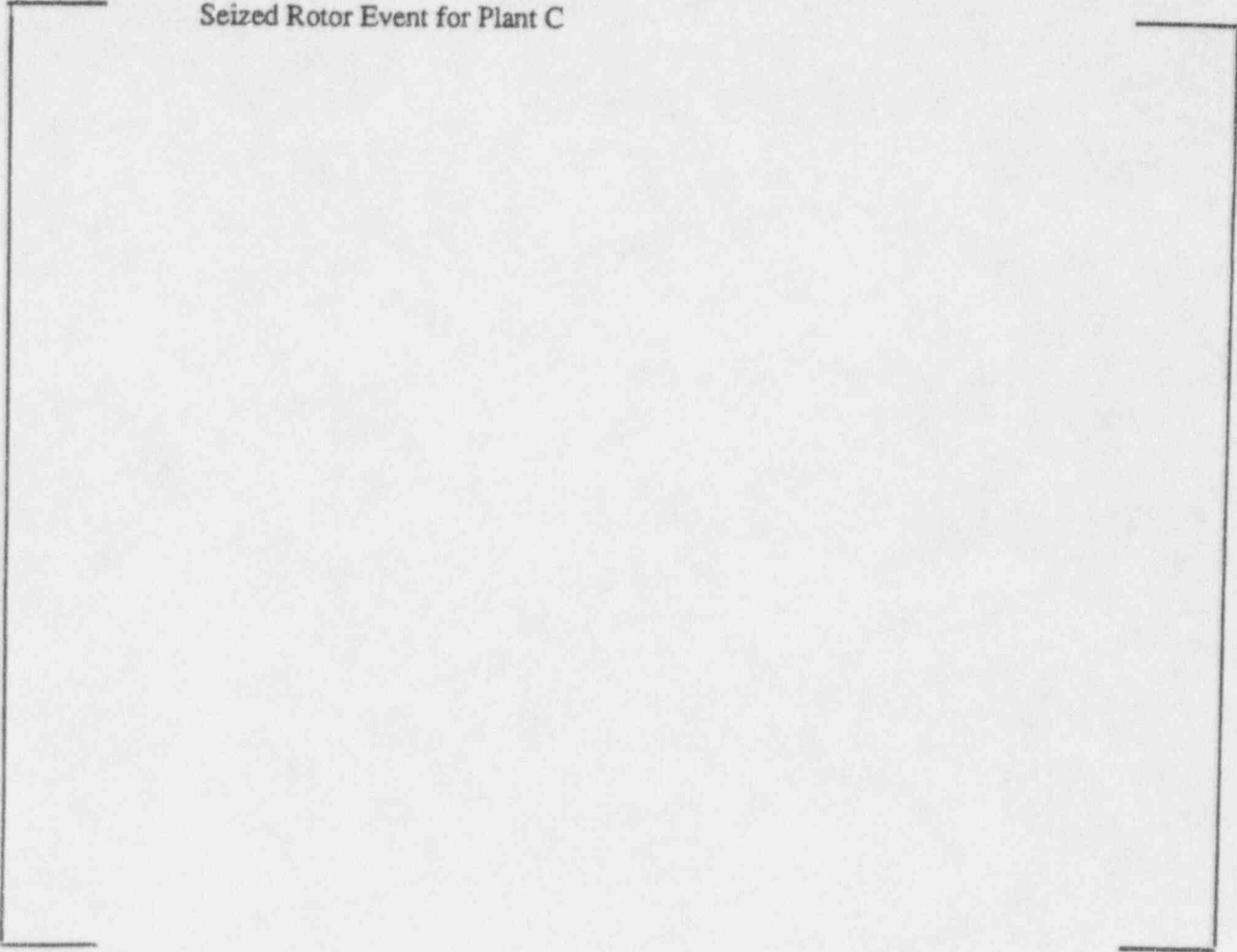


Figure 2.3.2.E Cold Leg Temperature - Intact Loop

Comparison of CENTS to RELAP5/MOD3
Seized Rotor Event for Plant C

Figure 2.3.2.F Core Exit Temperature

Comparison of CENTS to RELAP5/MOD3
Seized Rotor Event for Plant C

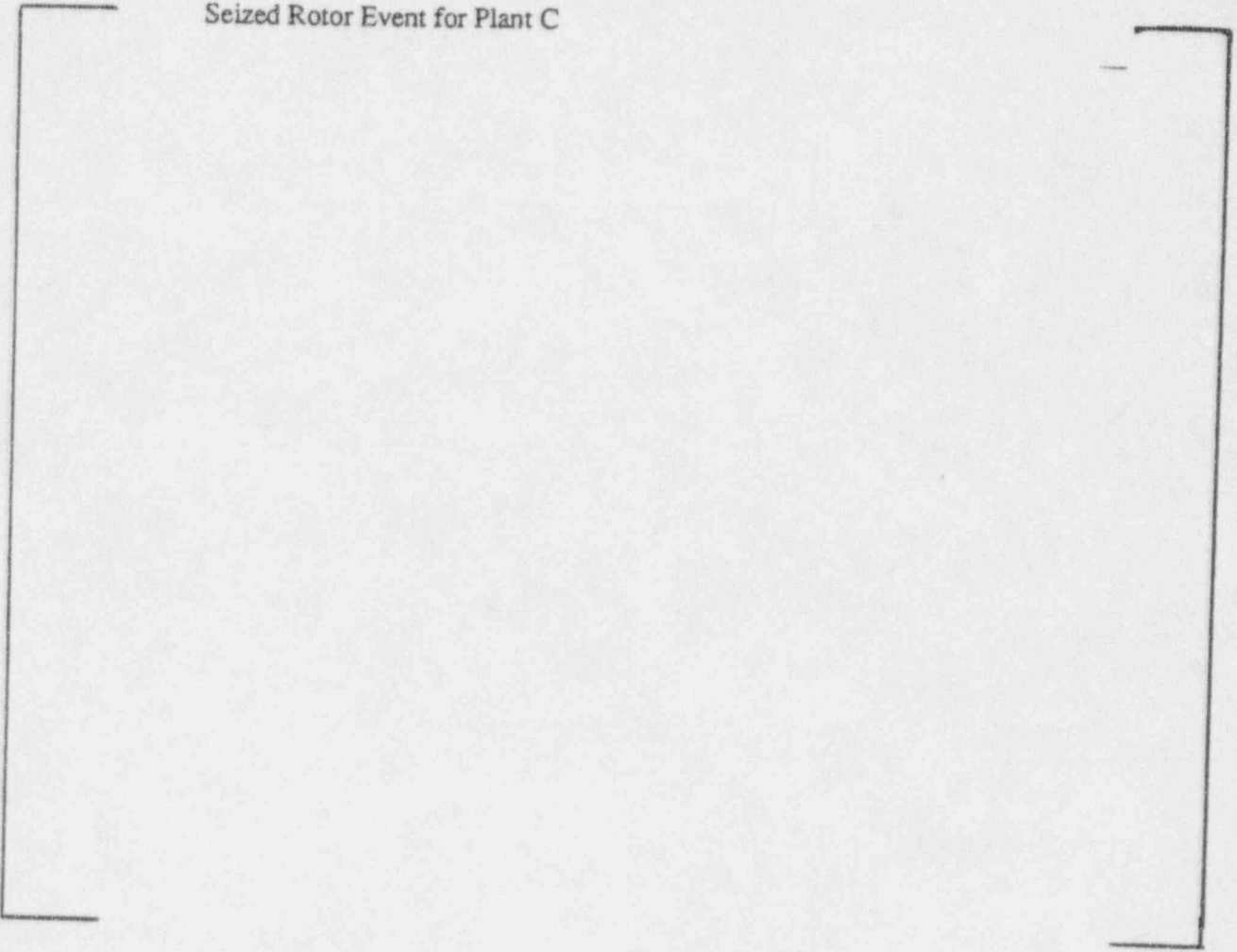


Figure 2.3.2.G Steam Generator Pressure

Comparison of CENTS to RELAP5/MOD3
Seized Rotor Event for Plant C

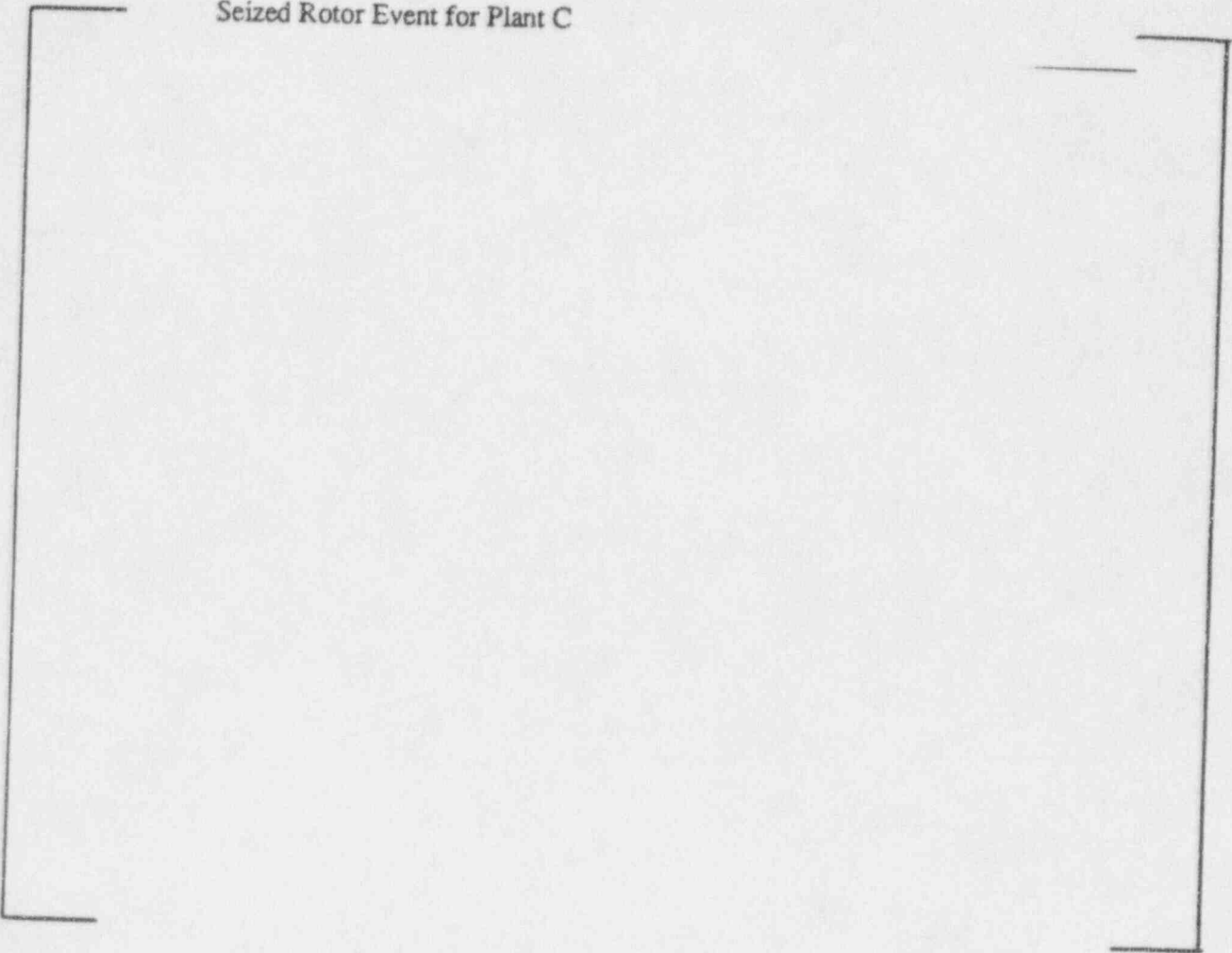


Figure 2.3.2.H Steam Generator Steam Flow

Comparison of CENTS to RELAP5/MOD3
Seized Rotor Event for Plant C

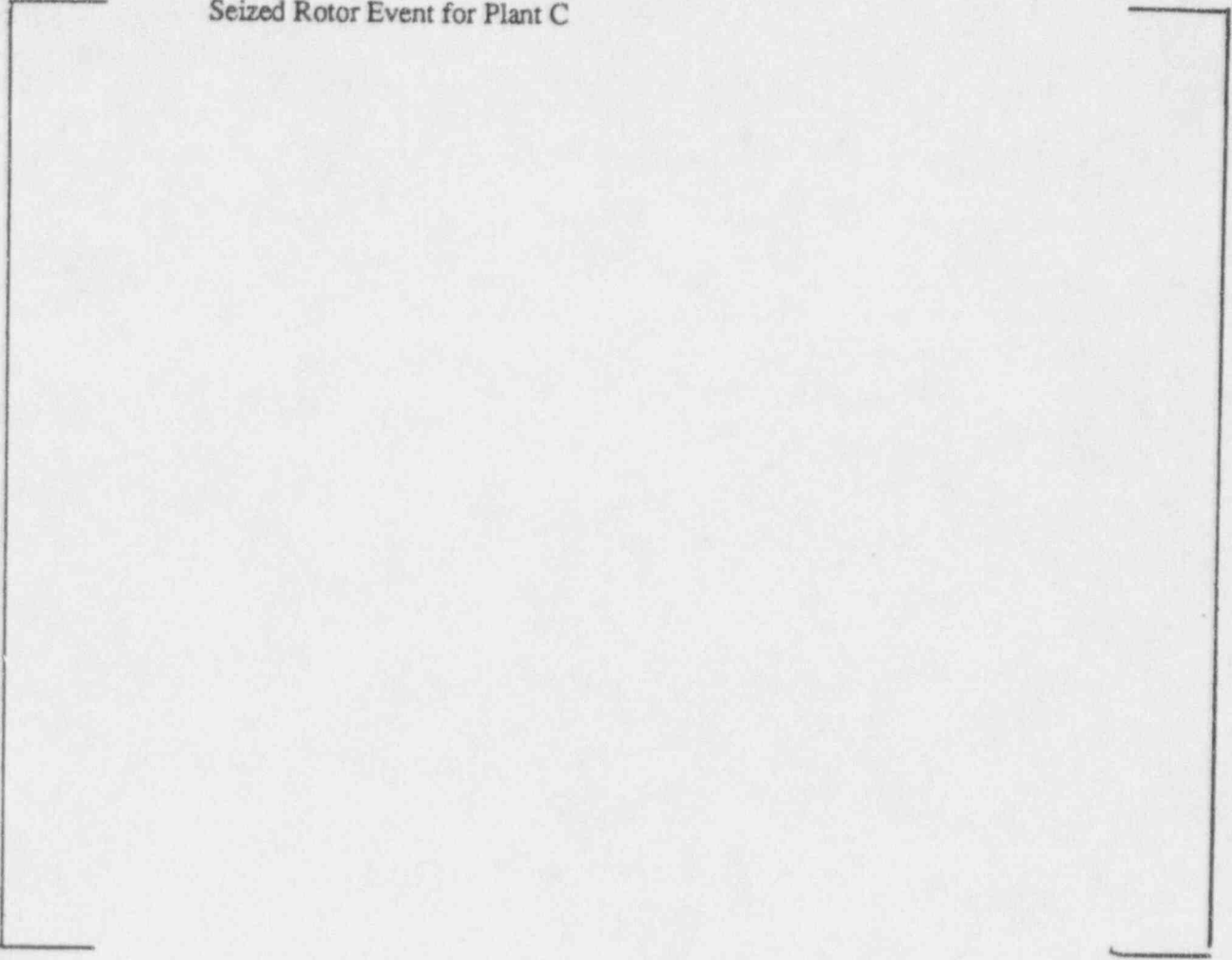


Figure 2.3.2.I Fuel Average Temperature

Comparison of CENTS to RELAP5/MOD3
Seized Rotor Event for Plant C

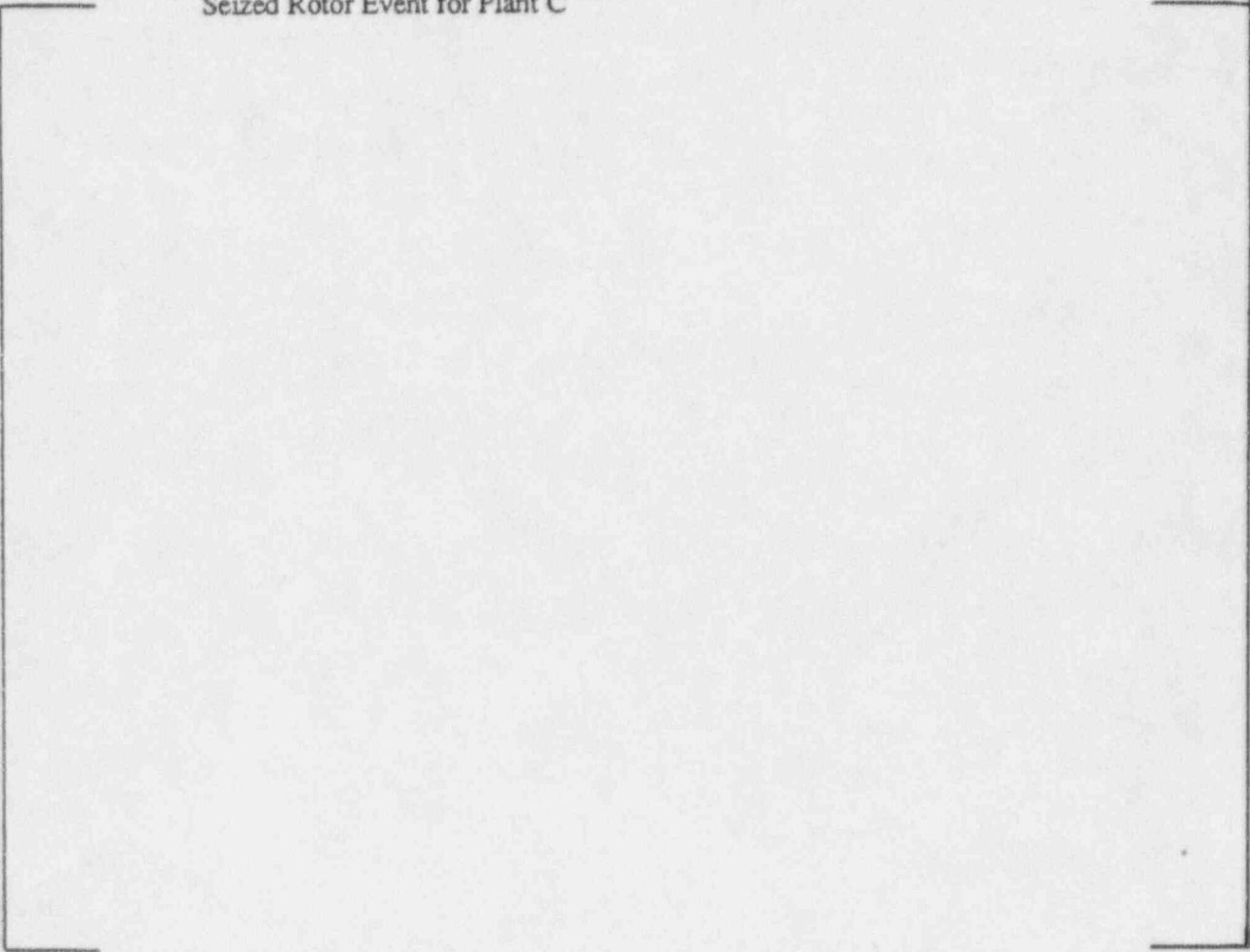


Figure 2.3.2.J Pressurizer Water Level

Comparison of CENTS to RELAP5/MOD3
Steam Generator Tube Rupture Event for Plant C

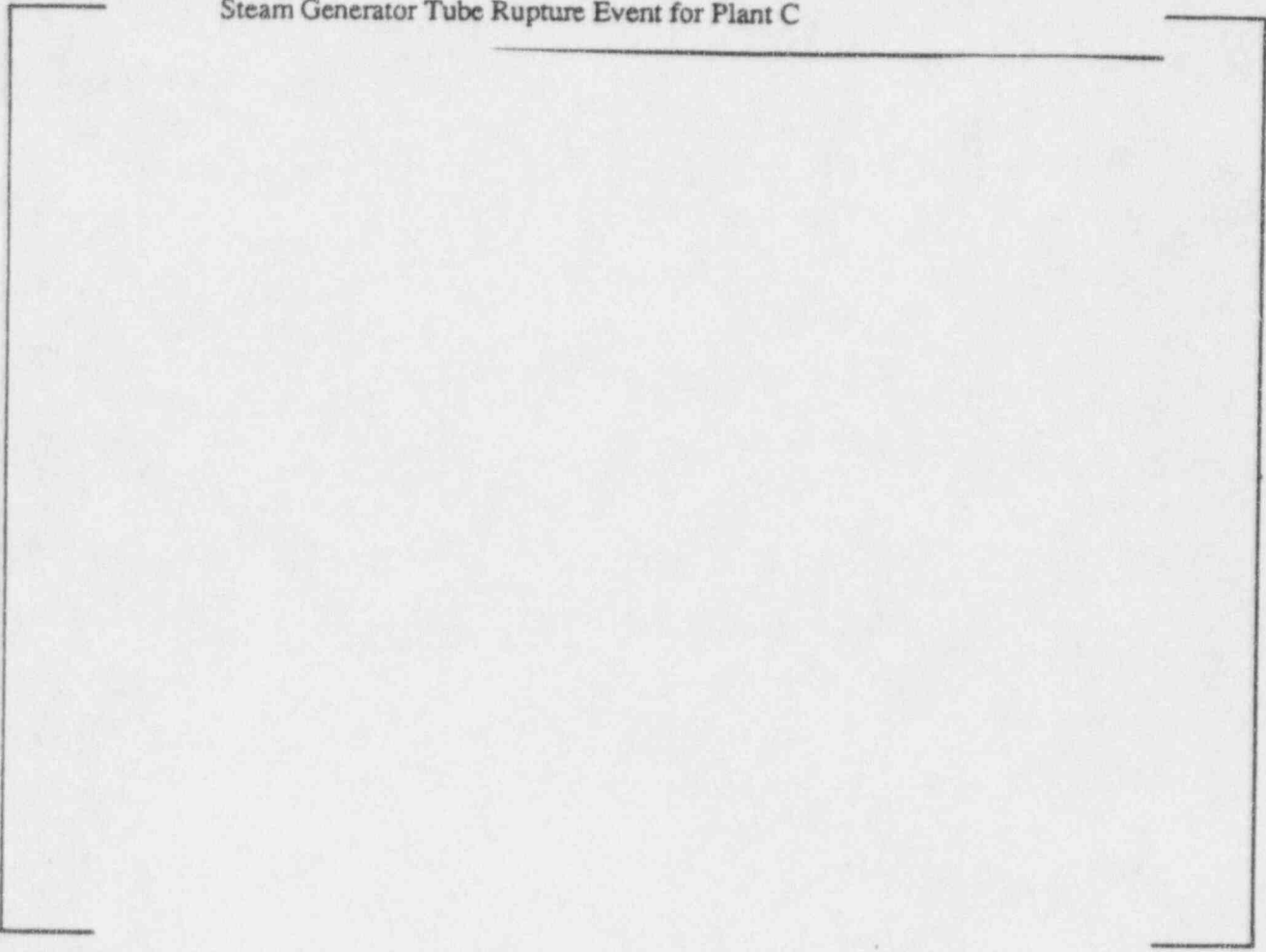


Figure 2.3.3.A Normalized Core Power

Comparison of CENTS to RELAP5/MOD3
Steam Generator Tube Rupture Event for Plant C

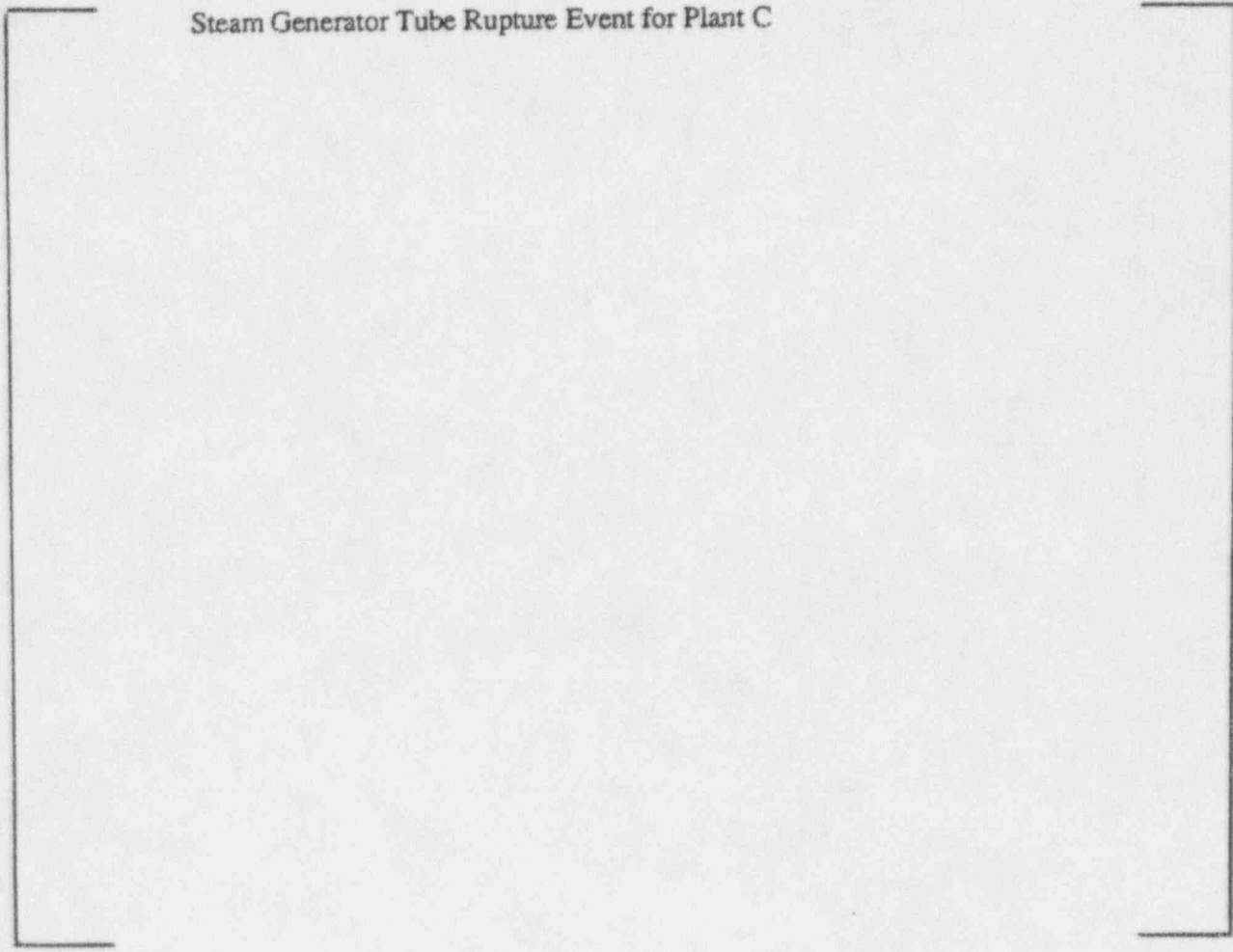


Figure 2.3.3.B Pressurizer Pressure

Comparison of CENTS to RELAP5/MOD3

Steam Generator Tube Rupture Event for Plant C

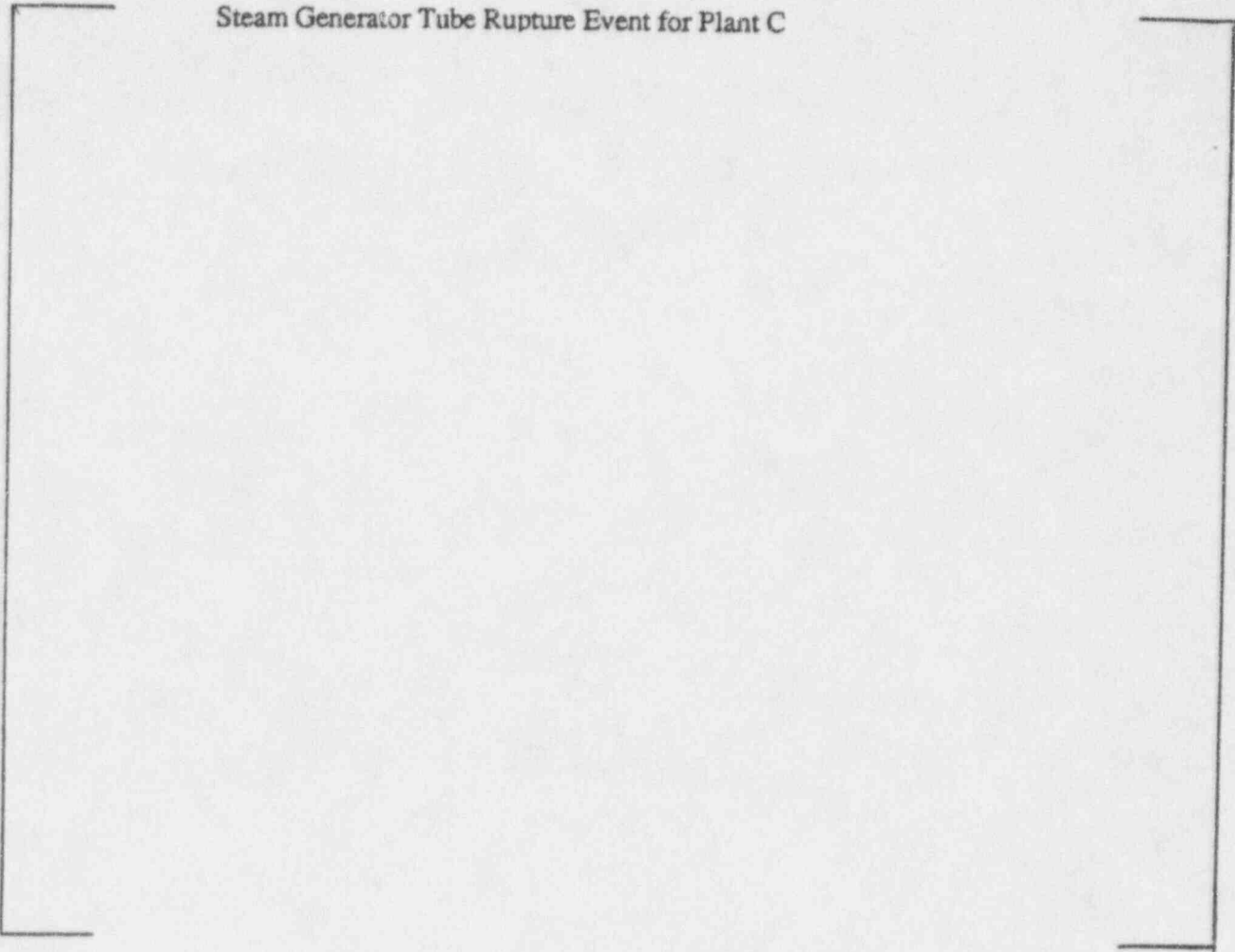


Figure 2.3.3.C Steam Generator Pressure

Comparison of CENTS to RELAP5/MOD3
Steam Generator Tube Rupture Event for Plant C

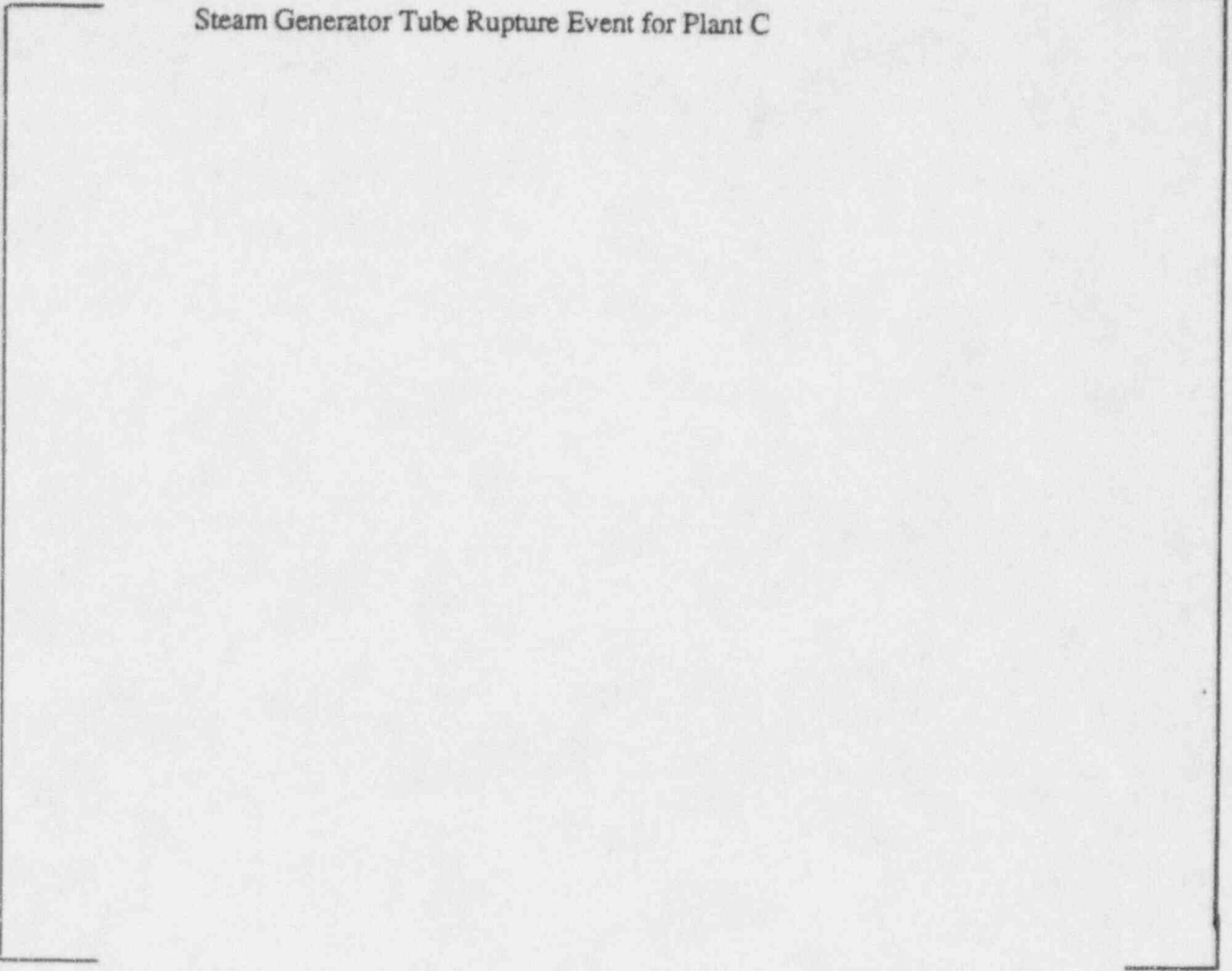


Figure 2.3.3.D Hot Leg Temperature - Affected Loop

Comparison of CENTS to RELAP5/MOD3
Steam Generator Tube Rupture Event for Plant C

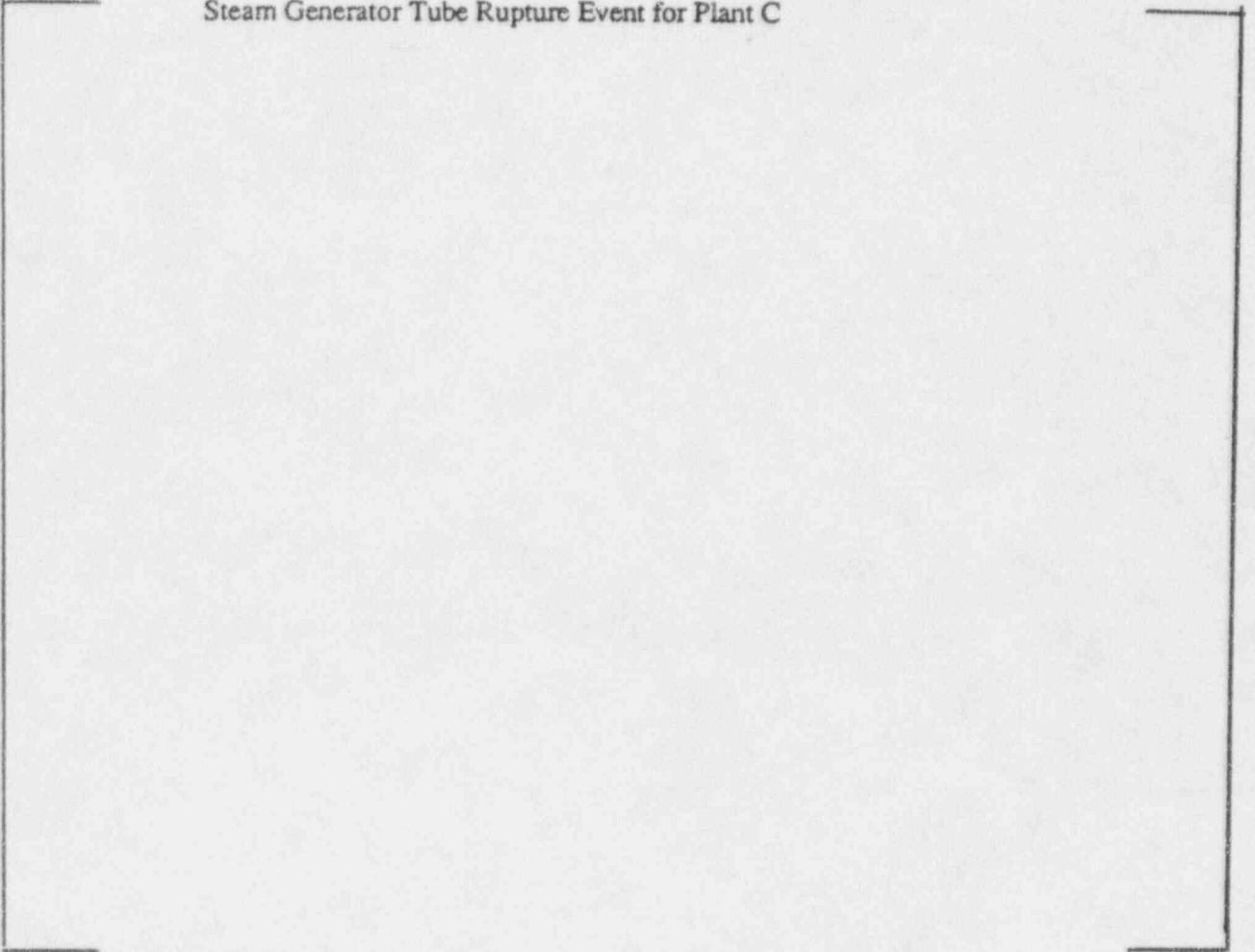


Figure 2.3.3.E Hot Leg Temperature - Intact Loop

Comparison of CENTS to RELAP5/MOD3
Steam Generator Tube Rupture Event for Plant C

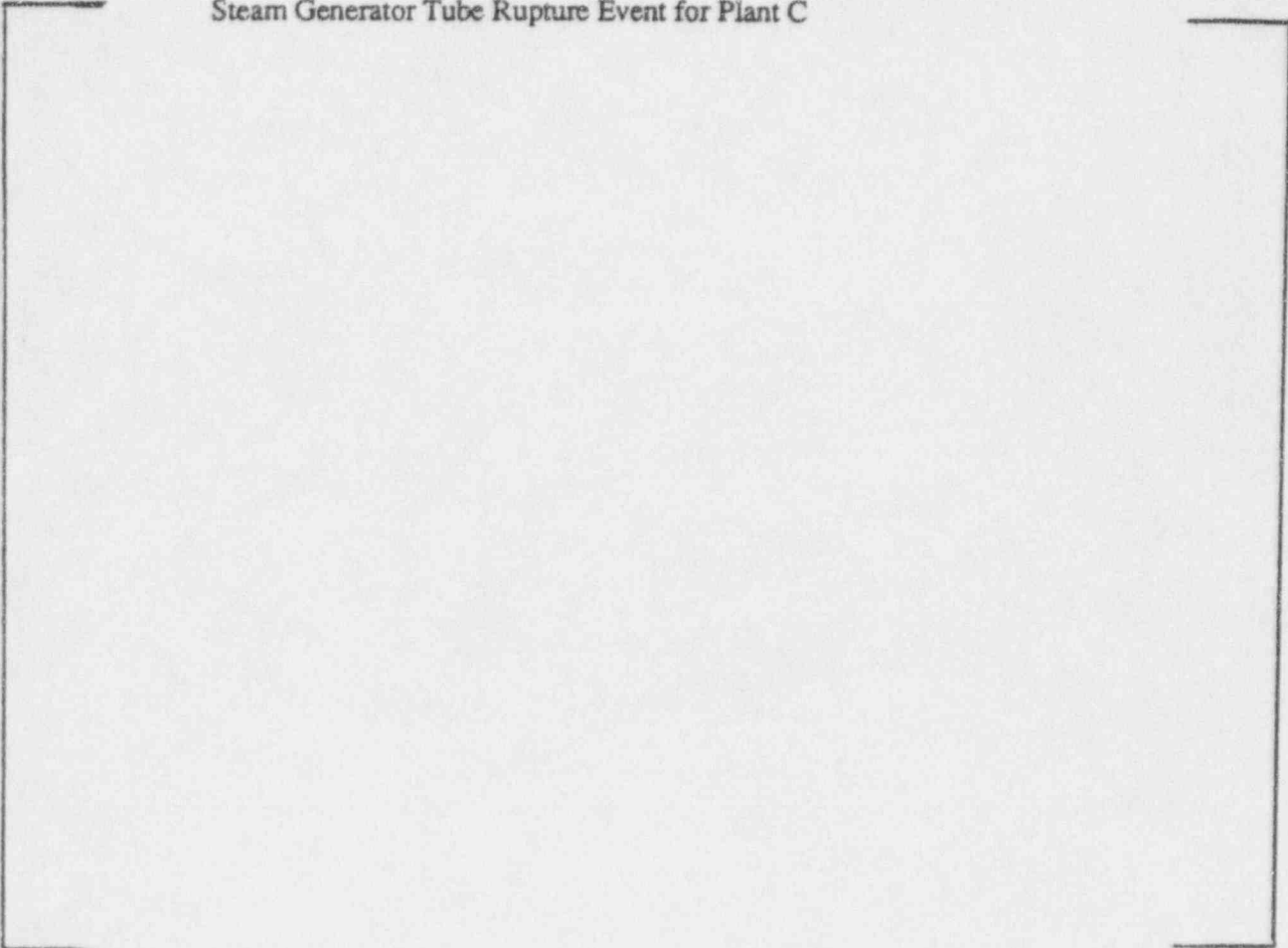


Figure 2.3.3.F Cold Leg Temperature - Affected Loop

Comparison of CENTS to RELAP5/MOD3
Steam Generator Tube Rupture Event for Plant C

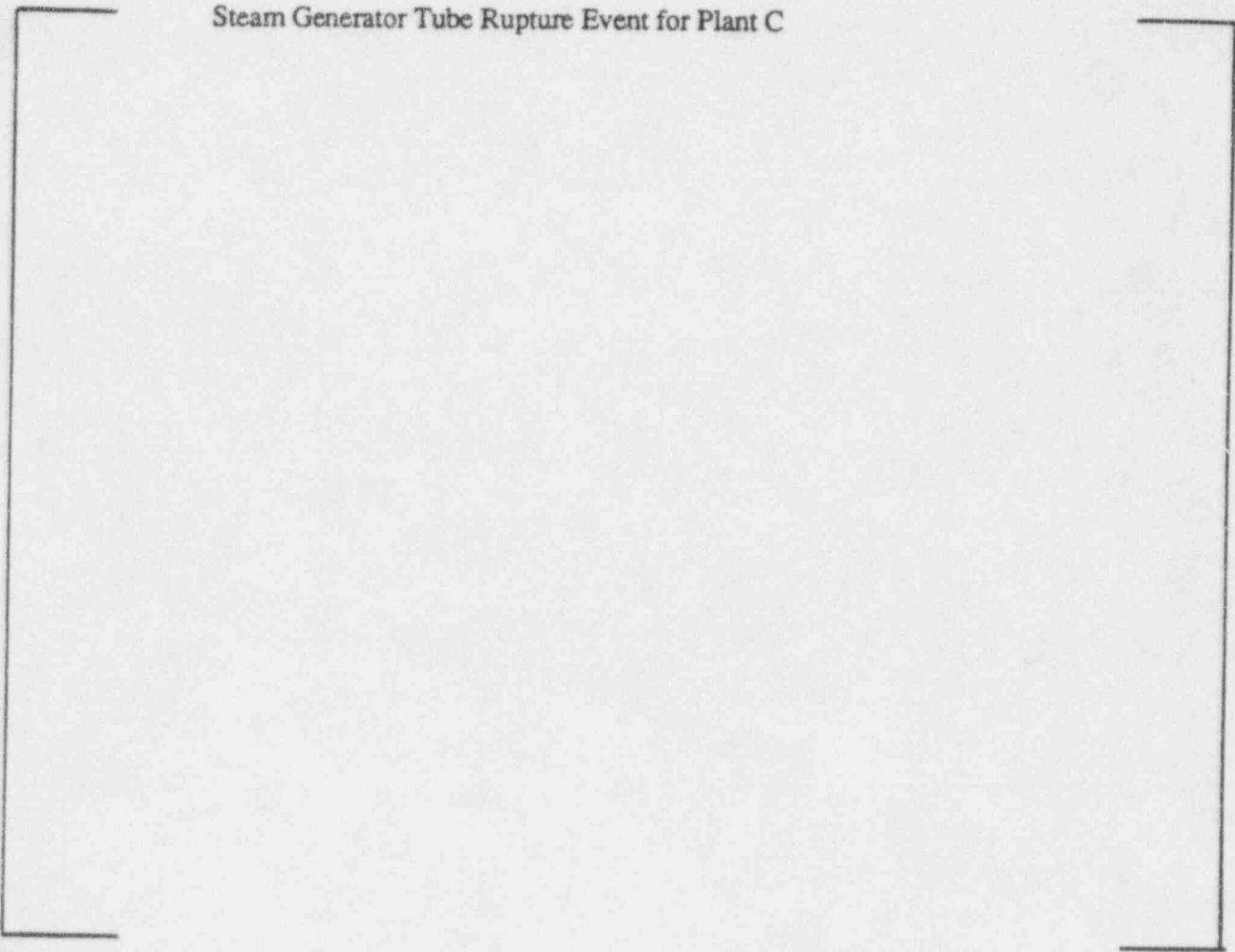


Figure 2.3.3.G Cold Leg Temperature - Intact Loop

Comparison of CENTS to RELAP5/MOD3

Steam Generator Tube Rupture Event for Plant C

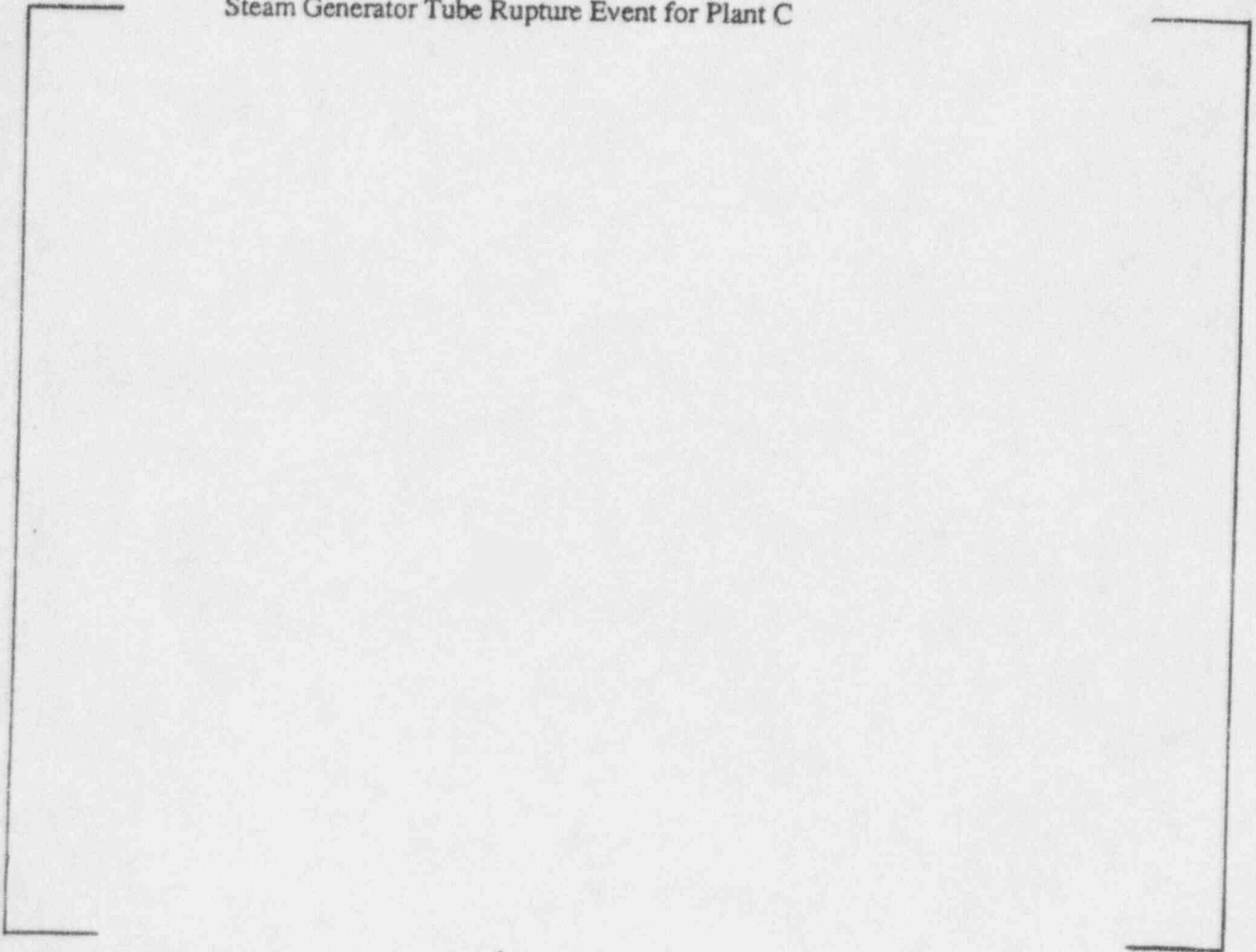


Figure 2.3.3.H Pressurizer Surge Flow

Comparison of CENTS to RELAP5/MOD3
Steam Generator Tube Rupture Event for Plant C

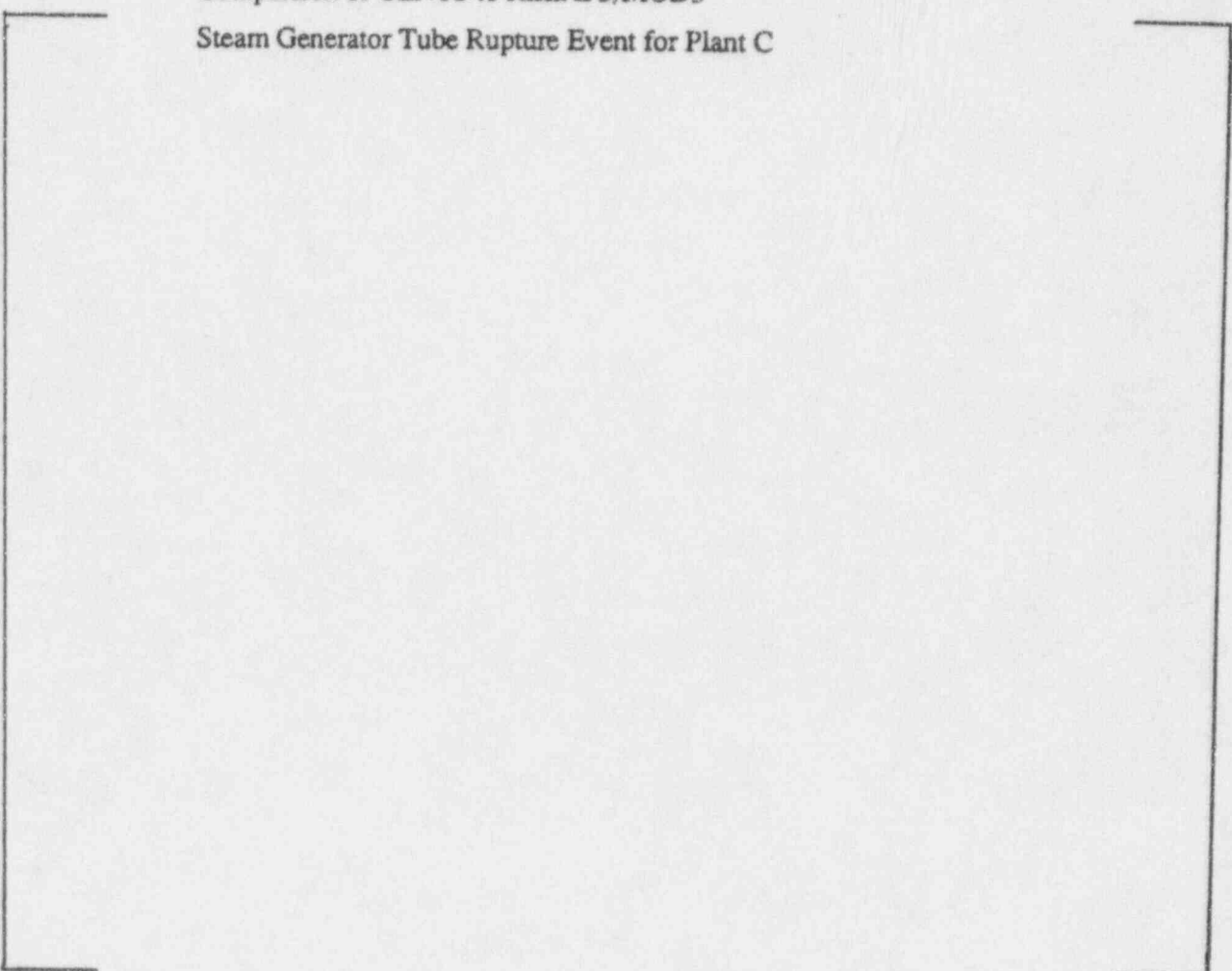


Figure 2.3.3.I Steam Generator Steam Flow

Comparison of CENTS to RELAP5/MOD3
Steam Generator Tube Rupture Event for Plant C

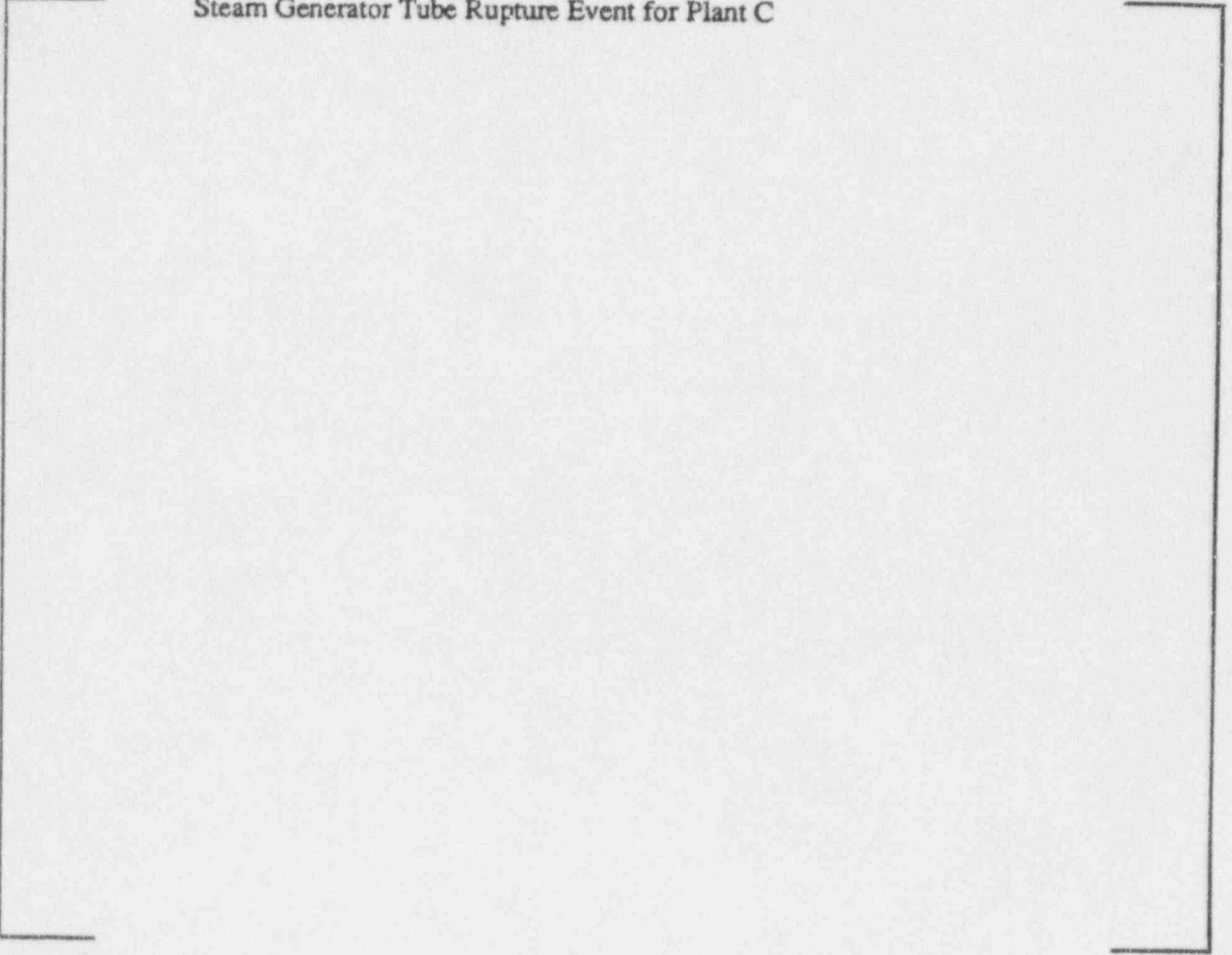


Figure 2.3.3.J Break Flow

Comparison of CENTS to RELAP5/MOD3
Steam Generator Tube Rupture Event for Plant C

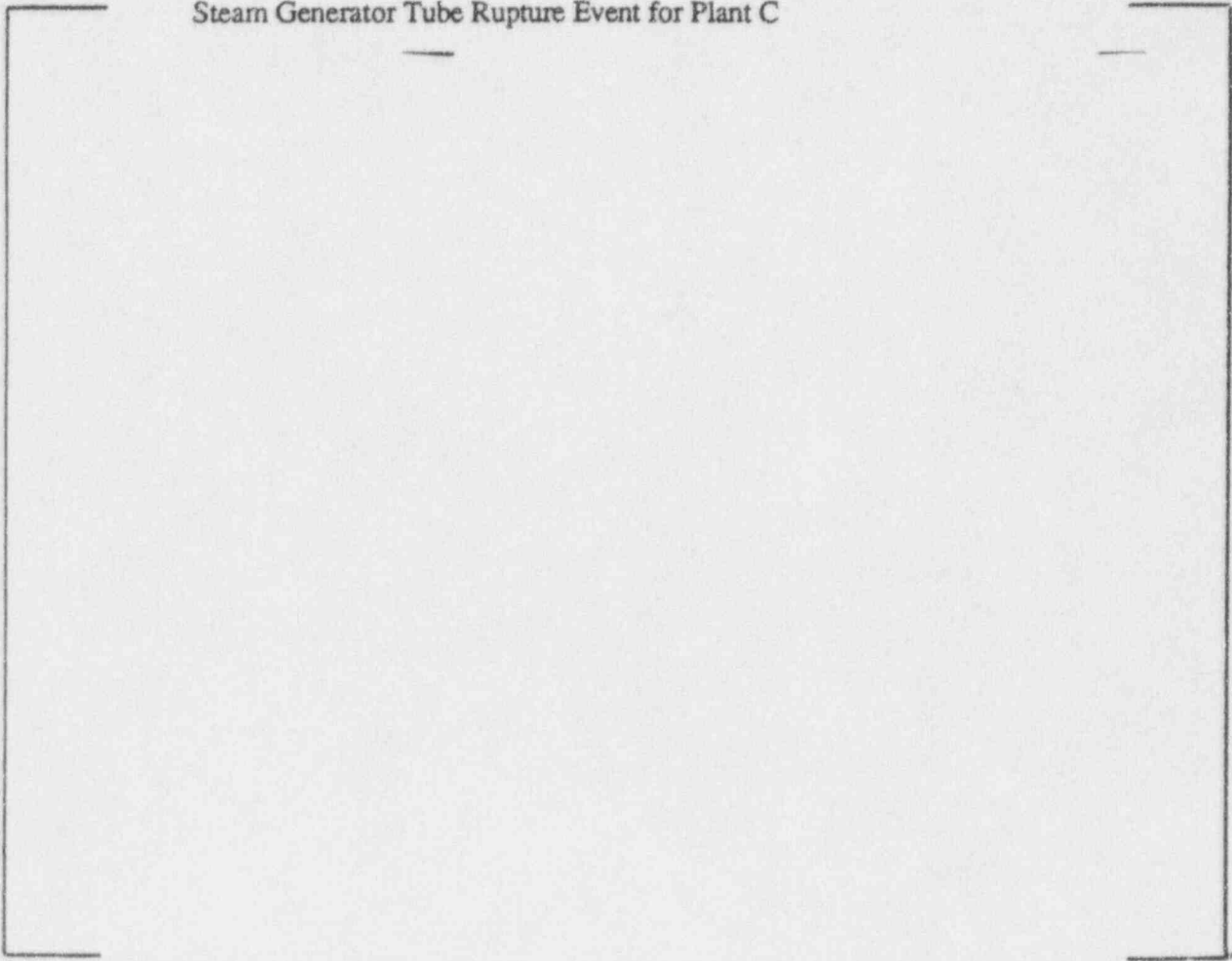


Figure 2.3.3.K Total HPSI Flow

Comparison of CENTS to RELAP5/MOD3
Steam Generator Tube Rupture Event for Plant C

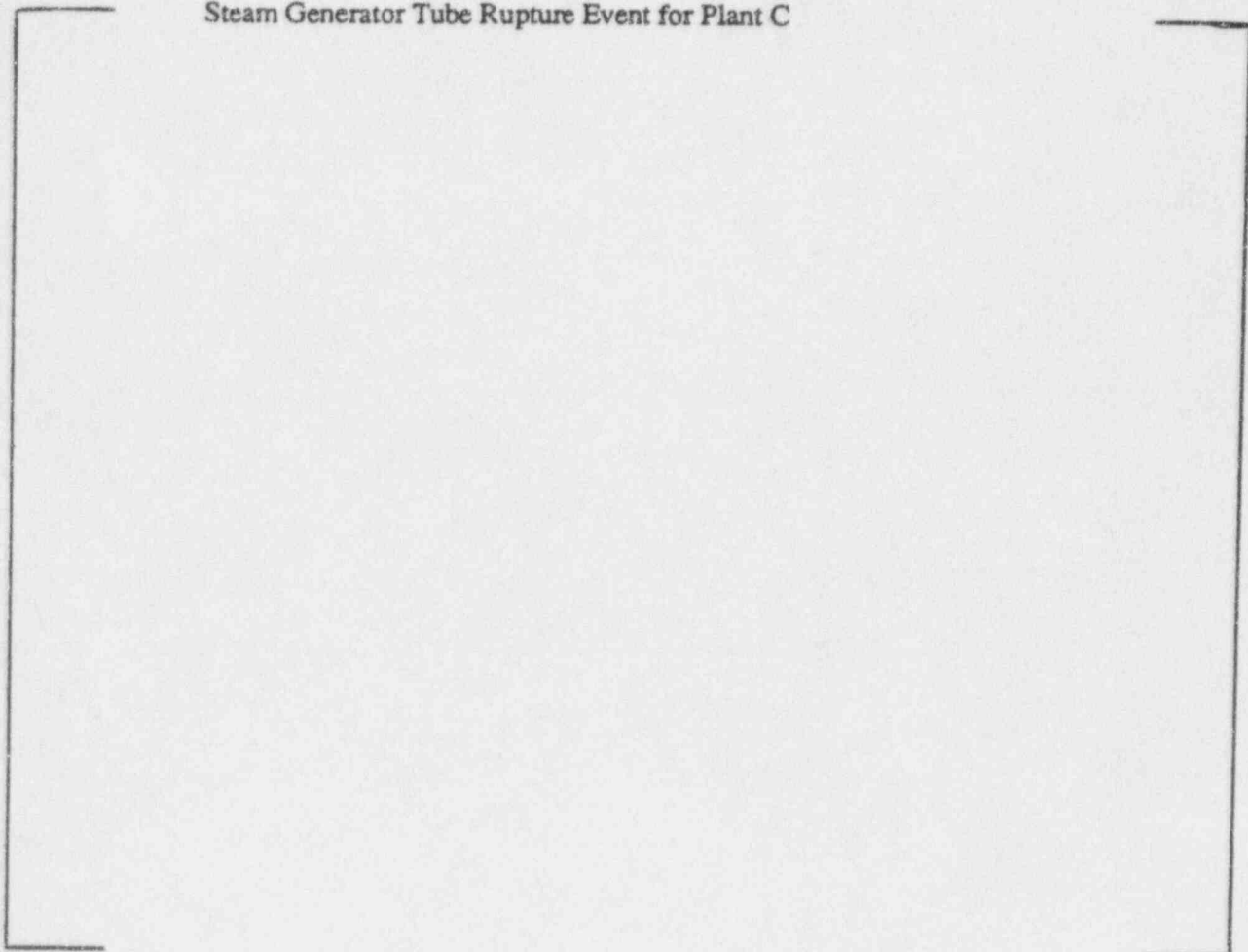


Figure 2.3.3.L Pressurizer Water Level

Comparison of CENTS to RELAP5/MOD3
Steam Generator Tube Rupture Event for Plant C

Figure 2.3.3.M Fuel Average Temperature

Comparison of CENTS to RELAP5/MOD3
Steam Generator Tube Rupture Event for Plant C

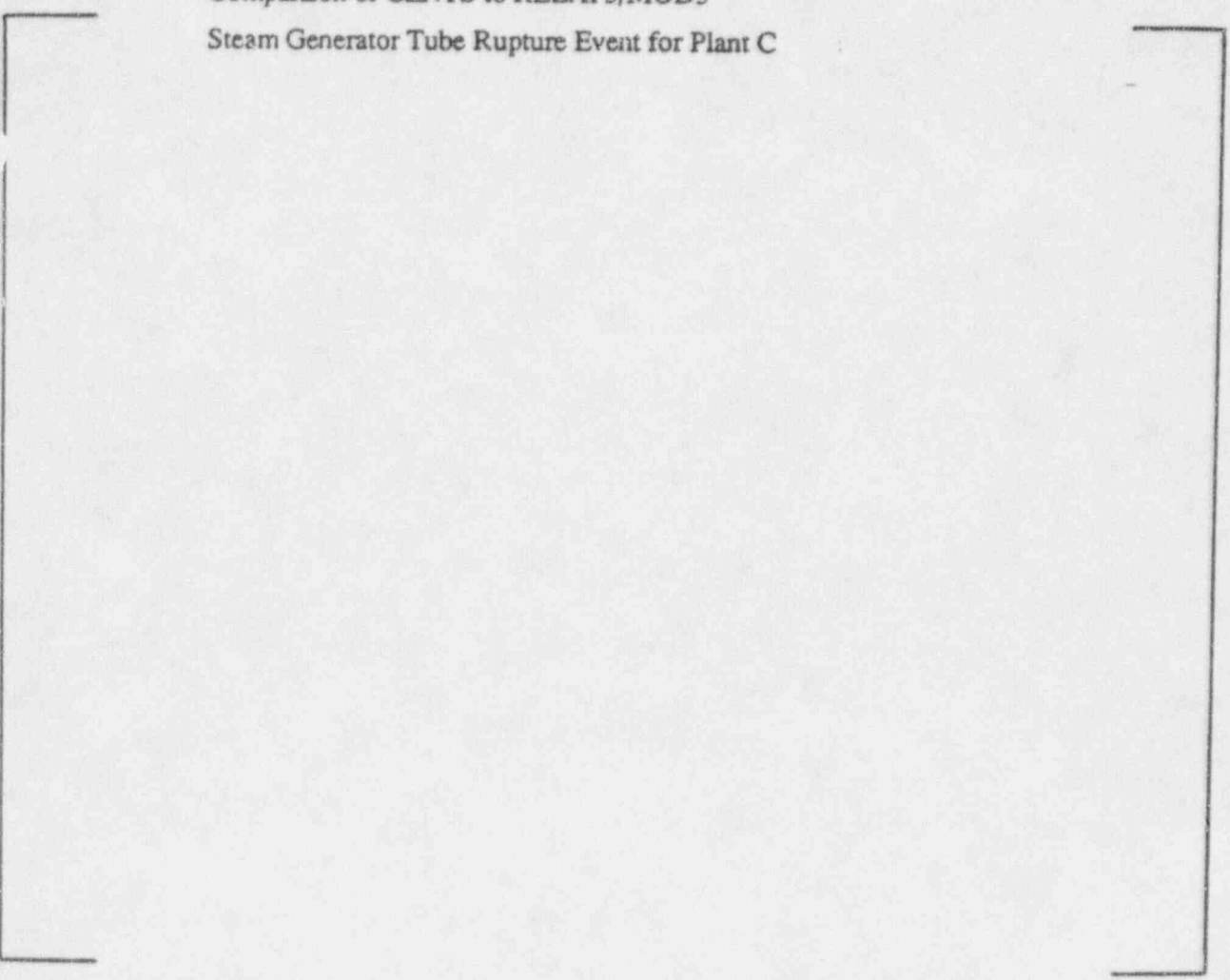


Figure 2.3.3.N Steam Generator Liquid Mass

Comparison of CENTS to RELAP5/MOD3
Steam Line Break Event for Plant C

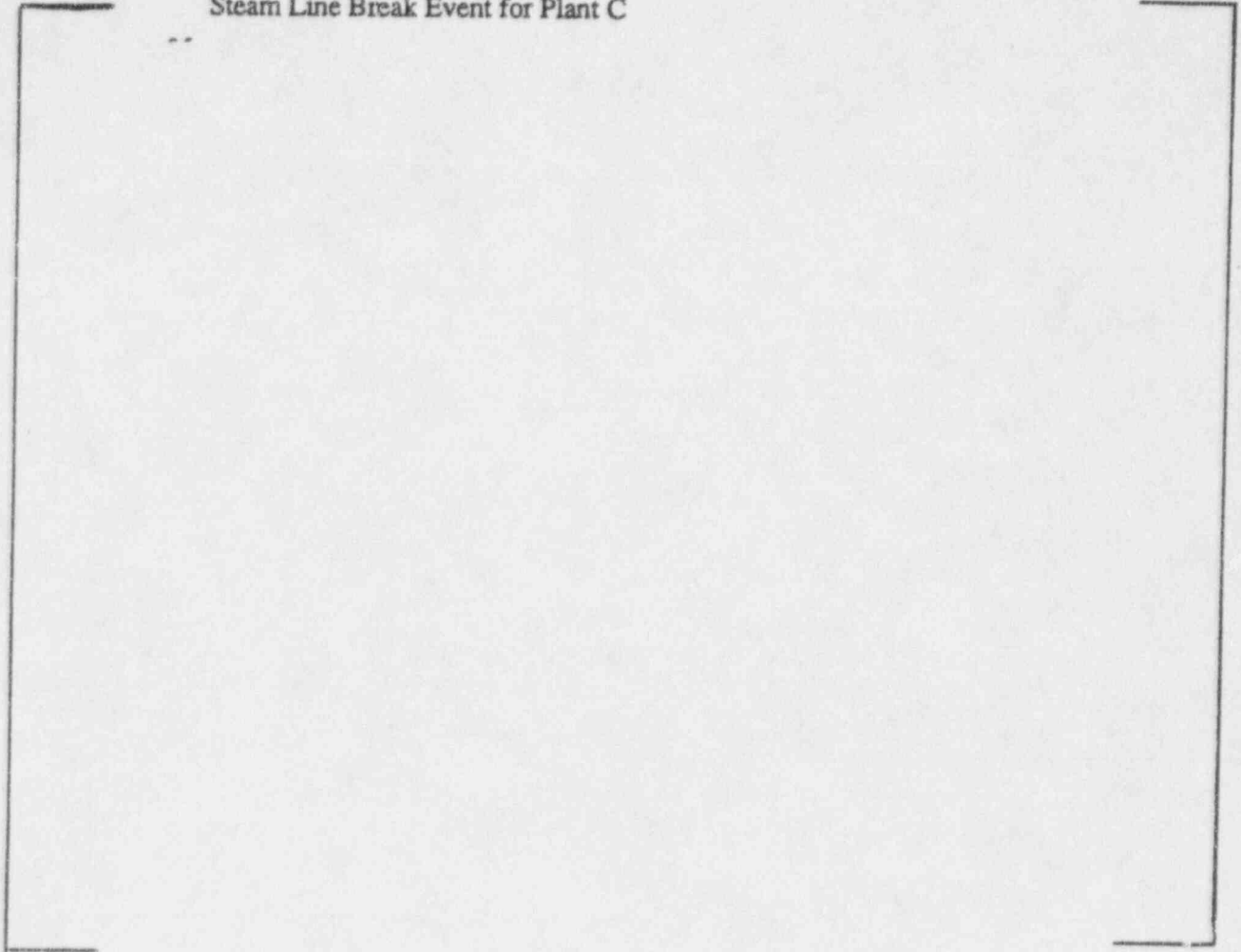


Figure 2.3.4.A Normalized Core Power

Comparison of CENTS to RELAP5/MOD3
Steam Line Break Event for Plant C

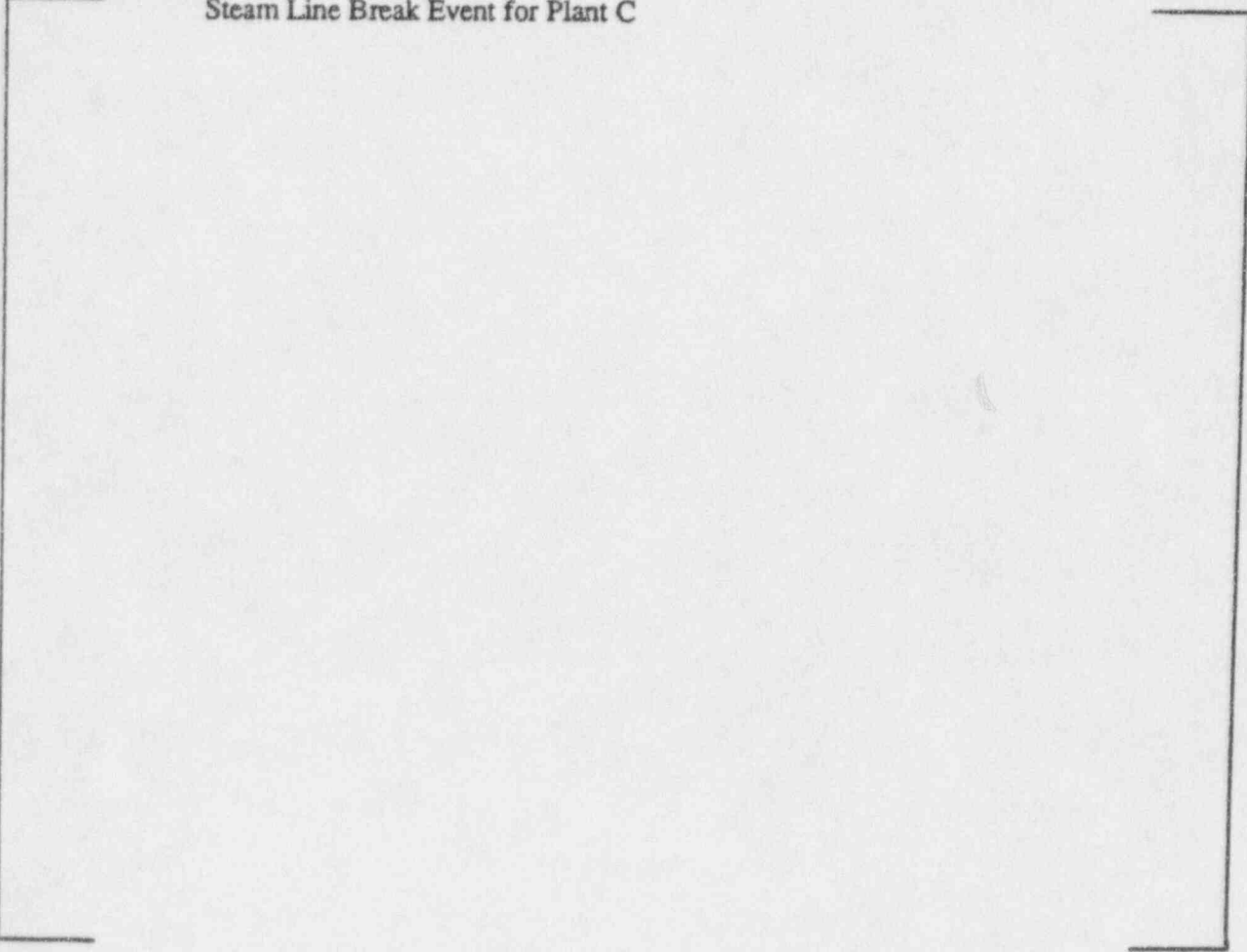


Figure 2.3.4.B Fuel Average Surface Heat Flux

Comparison of CENTS to RELAP5/MOD3
Steam Line Break Event for Plant C

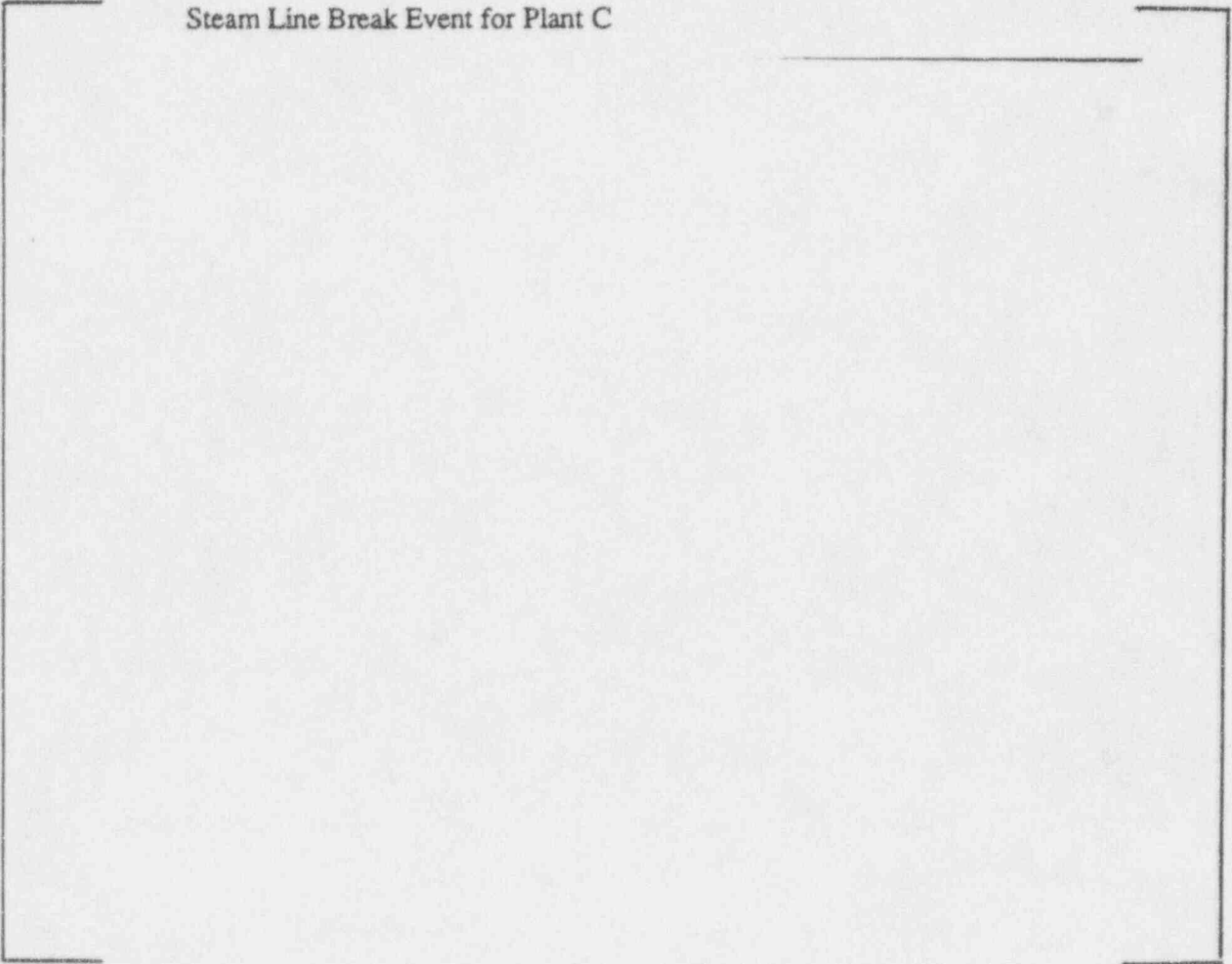


Figure 2.3.4.C Pressurizer Pressure

Comparison of CENTS to RELAP5/MOD3
Steam Line Break Event for Plant C

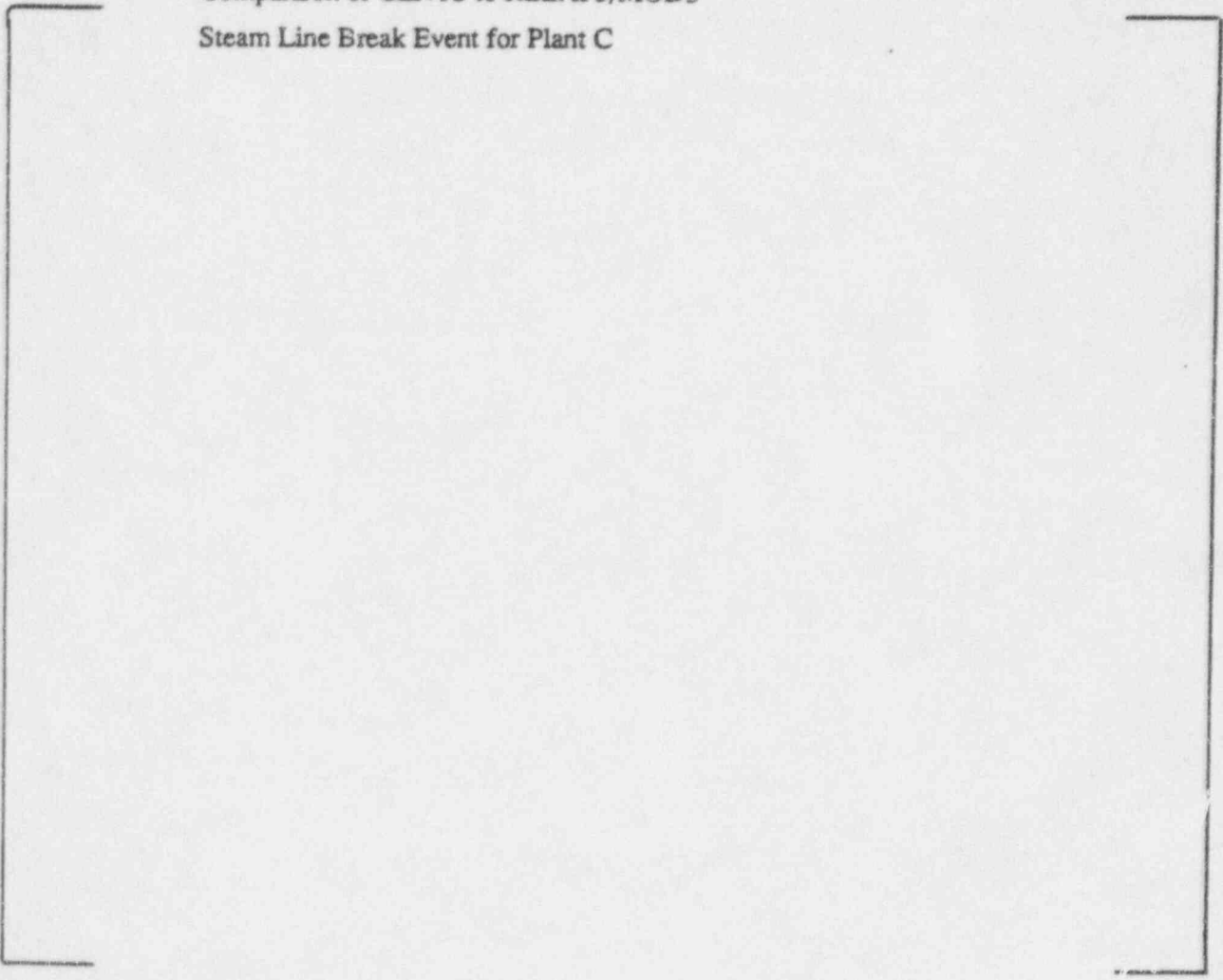


Figure 2.3.4.D Steam Generator Pressure

Comparison of CENTS to RELAP5/MOD3
Steam Line Break Event for Plant C

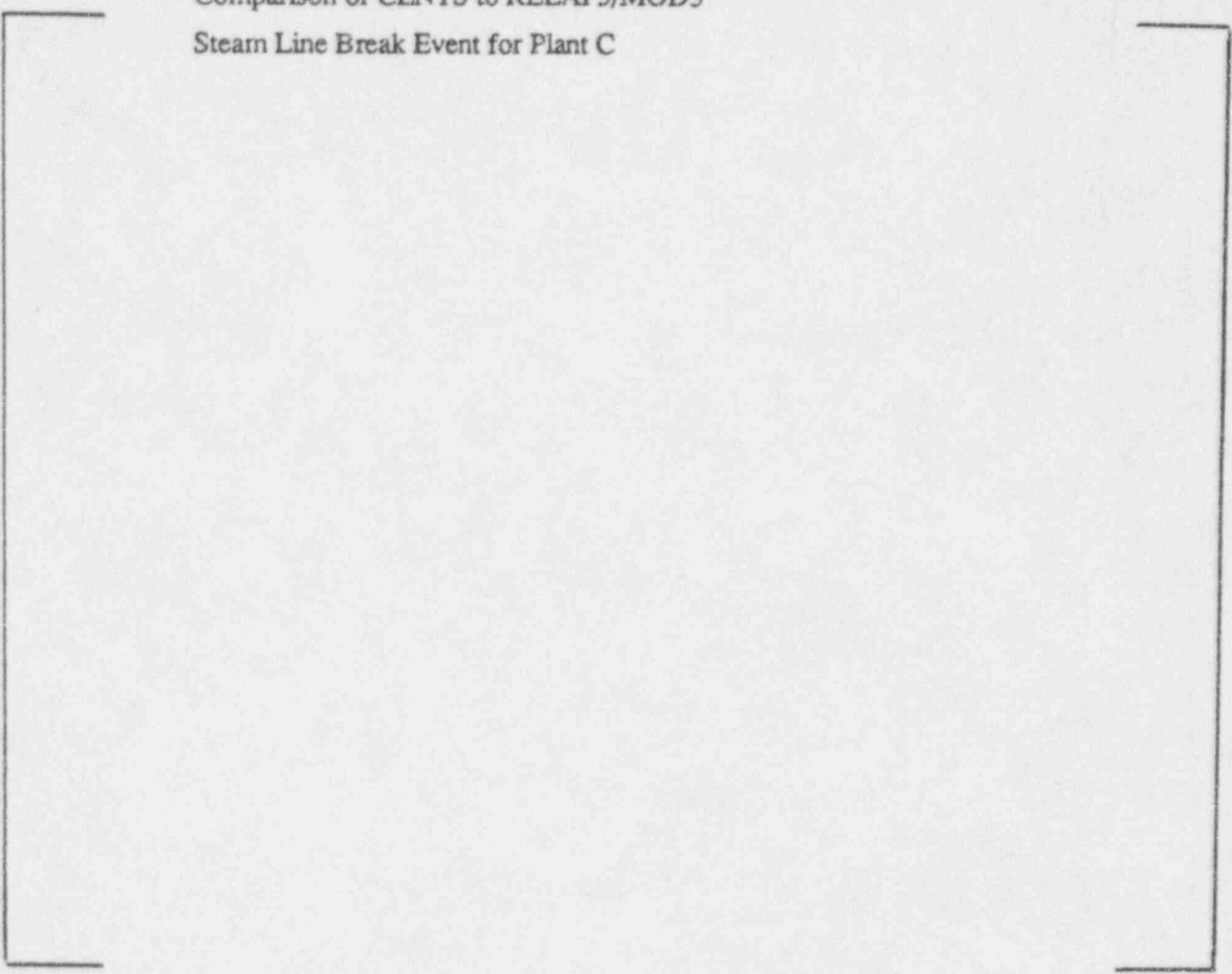


Figure 2.3.4.E Steam Generator Steam Flow

Comparison of CENTS to RELAP5/MOD3
Steam Line Break Event for Plant C

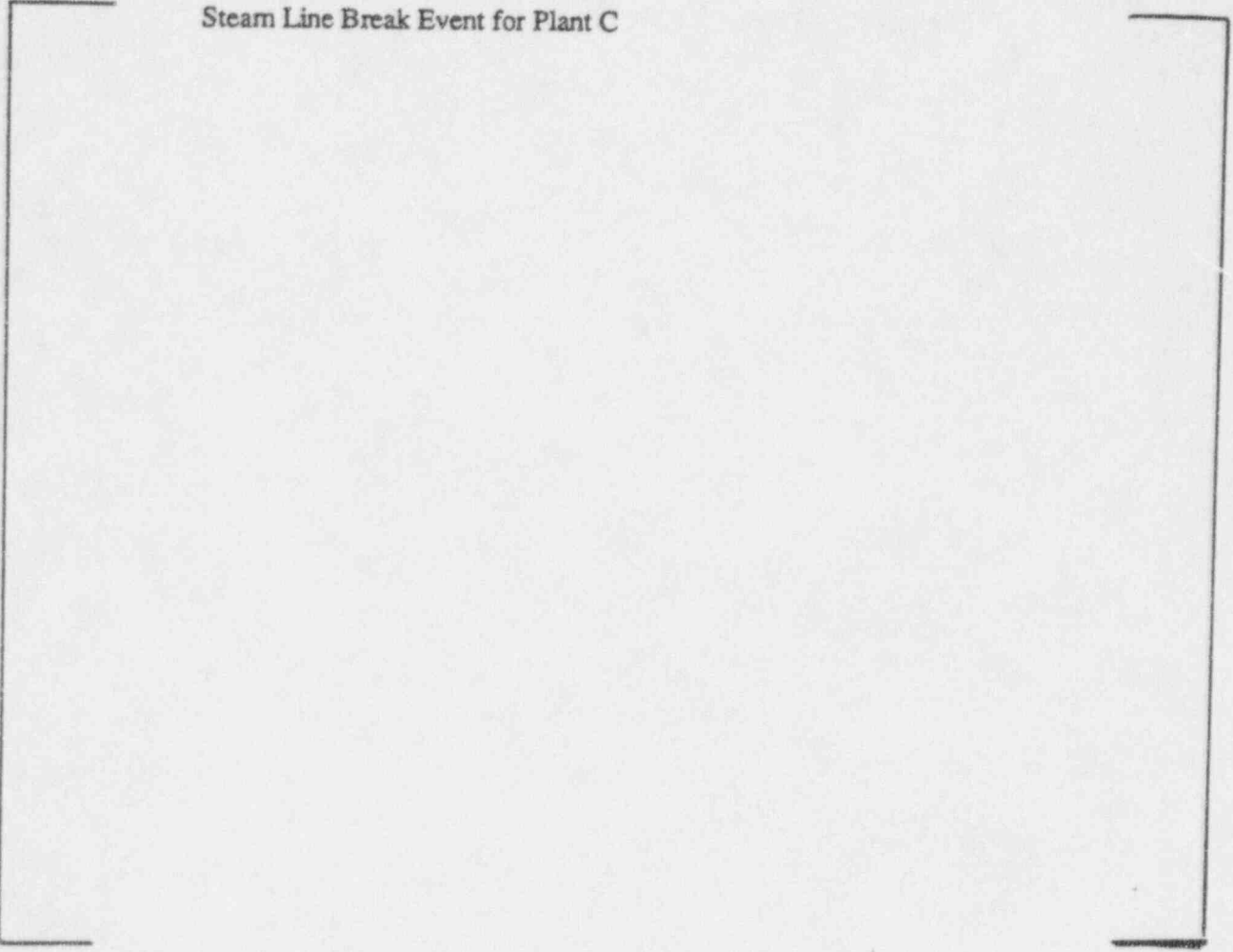


Figure 2.3.4.F Steam Generator Liquid Mass

Comparison of CENTS to RELAP5/MOD3
Steam Line Break Event for Plant C

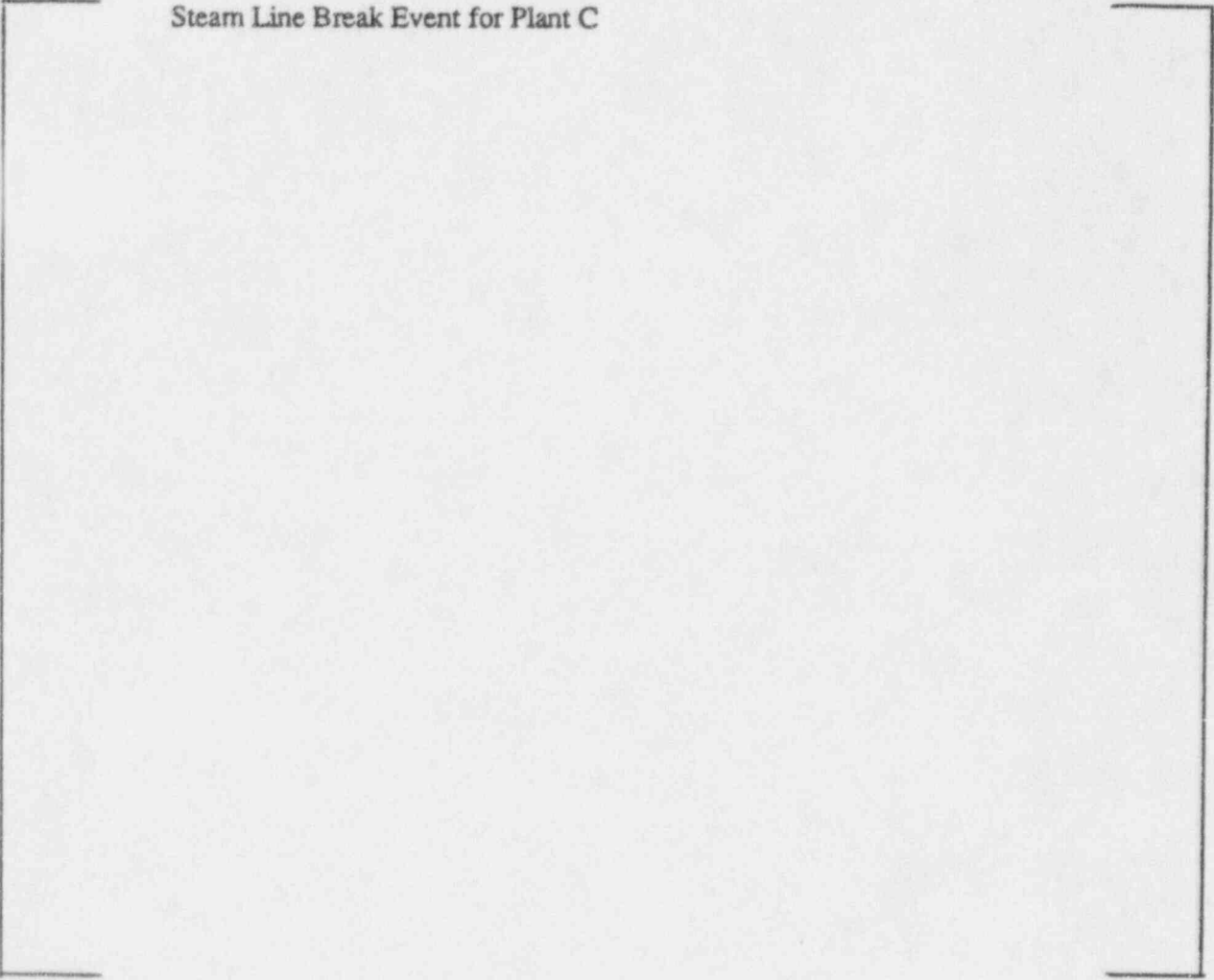


Figure 2.3.4.G Break Flow Quality

Comparison of CENTS to RELAP5/MOD3
Steam Line Break Event for Plant C

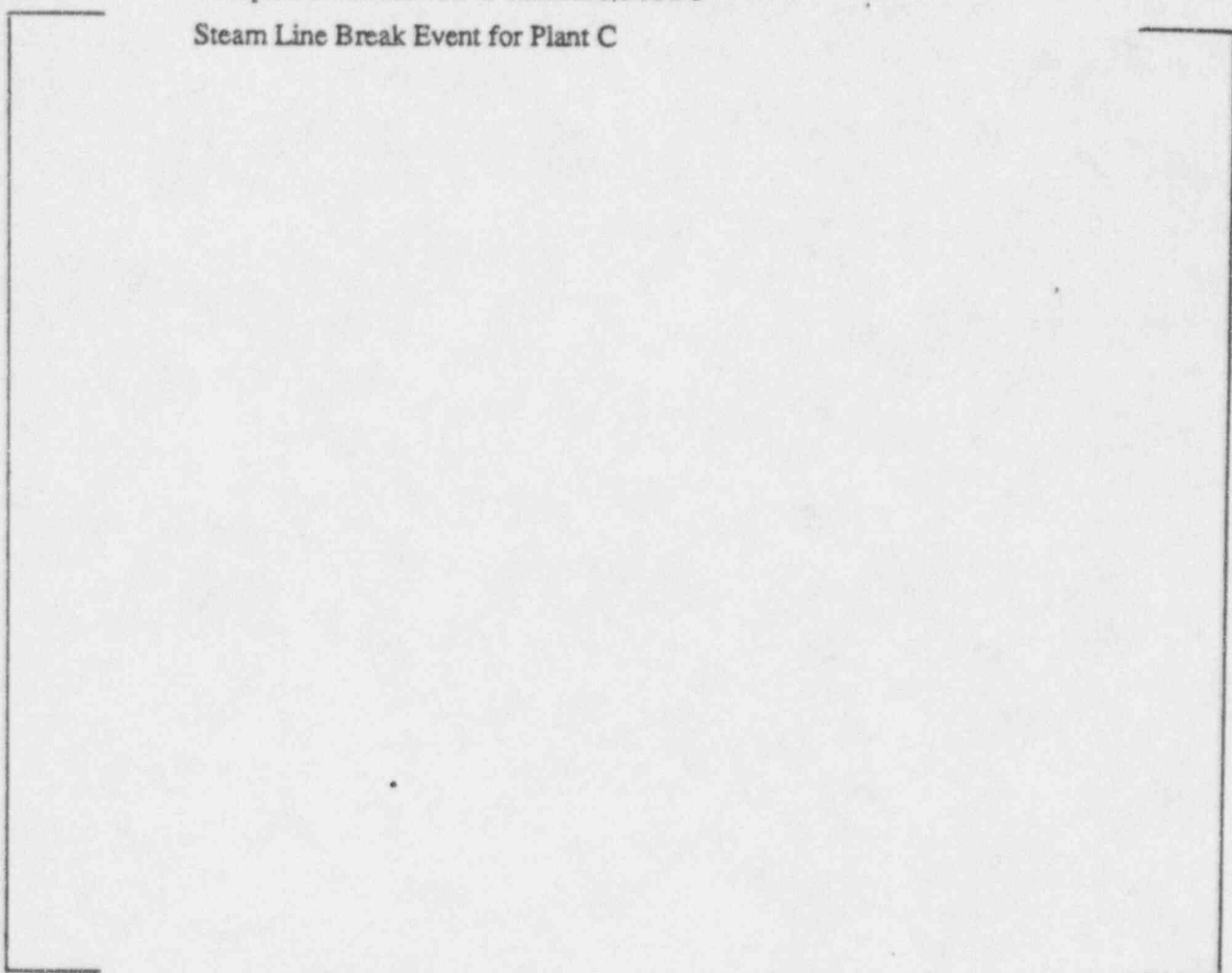


Figure 2.3.4.H Hot Leg Temperature - Affected Loop

Comparison of CENTS to RELAP5/MOD3
Steam Line Break Event for Plant C

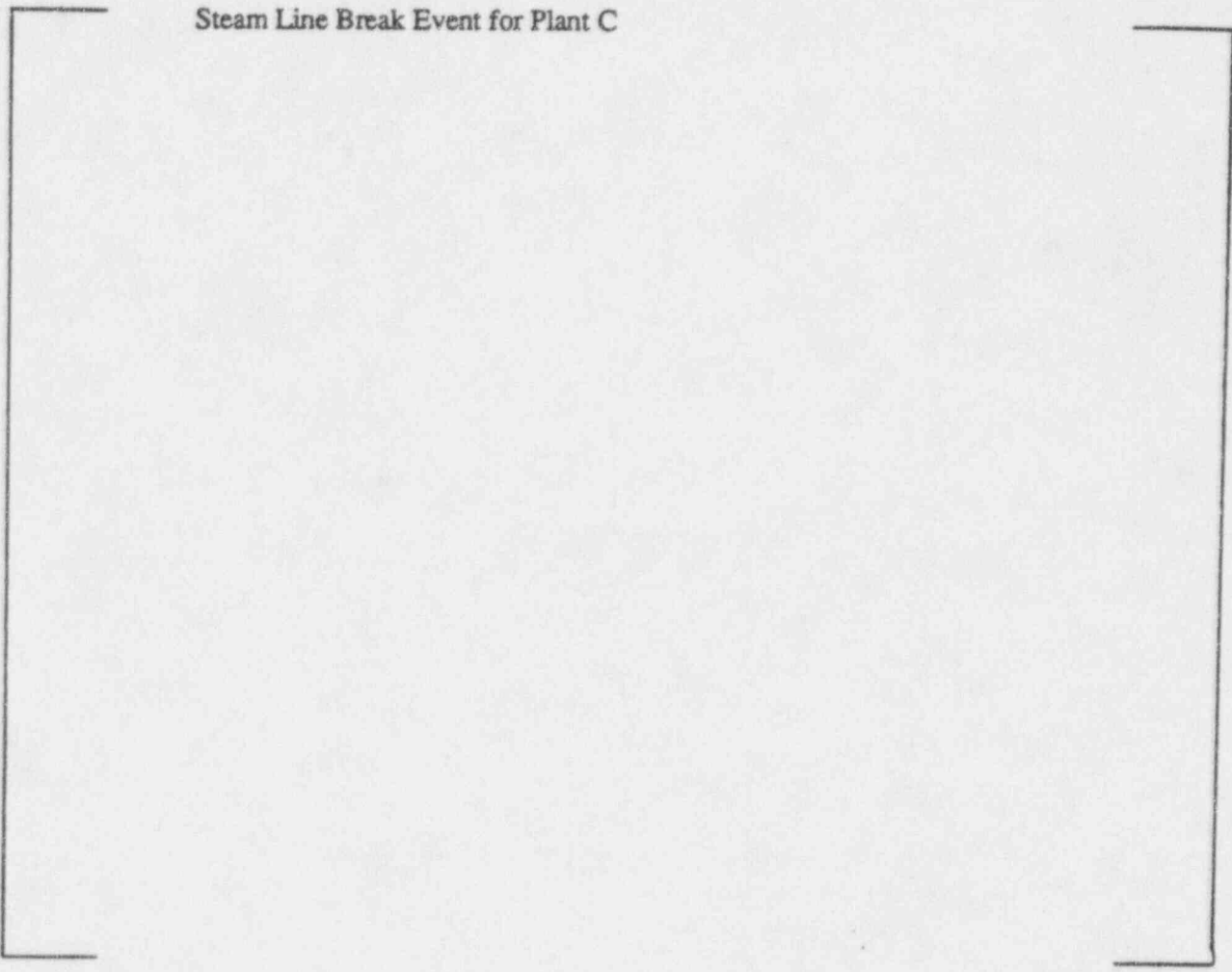


Figure 2.3.4.I Hot Leg Temperature - Intact Loop

Comparison of CENTS to RELAP5/MOD3
Steam Line Break Event for Plant C

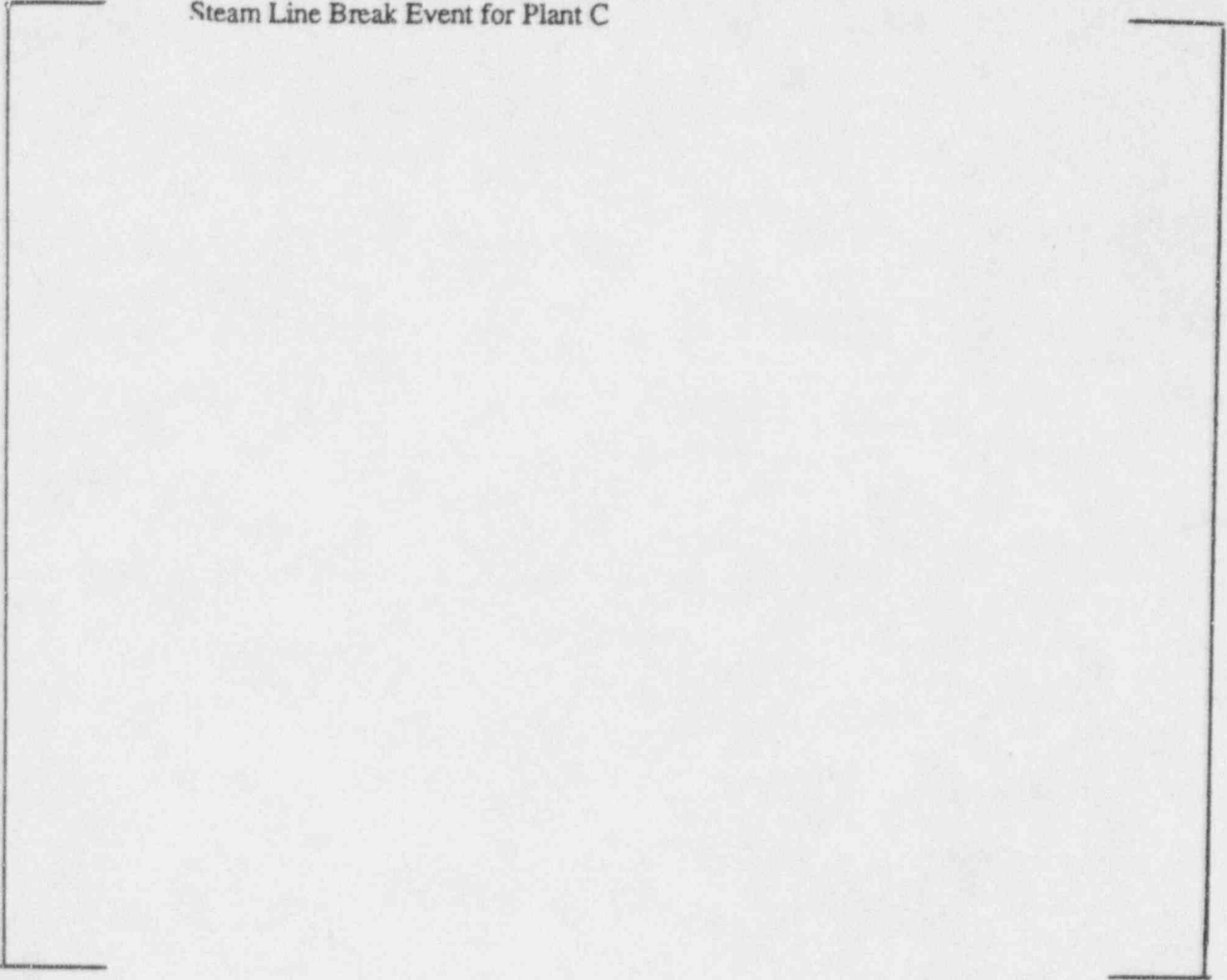


Figure 2.3.4.J Cold Leg Temperature - Affected Loop

Comparison of CENTS to RELAP5/MOD3
Steam Line Break Event for Plant C

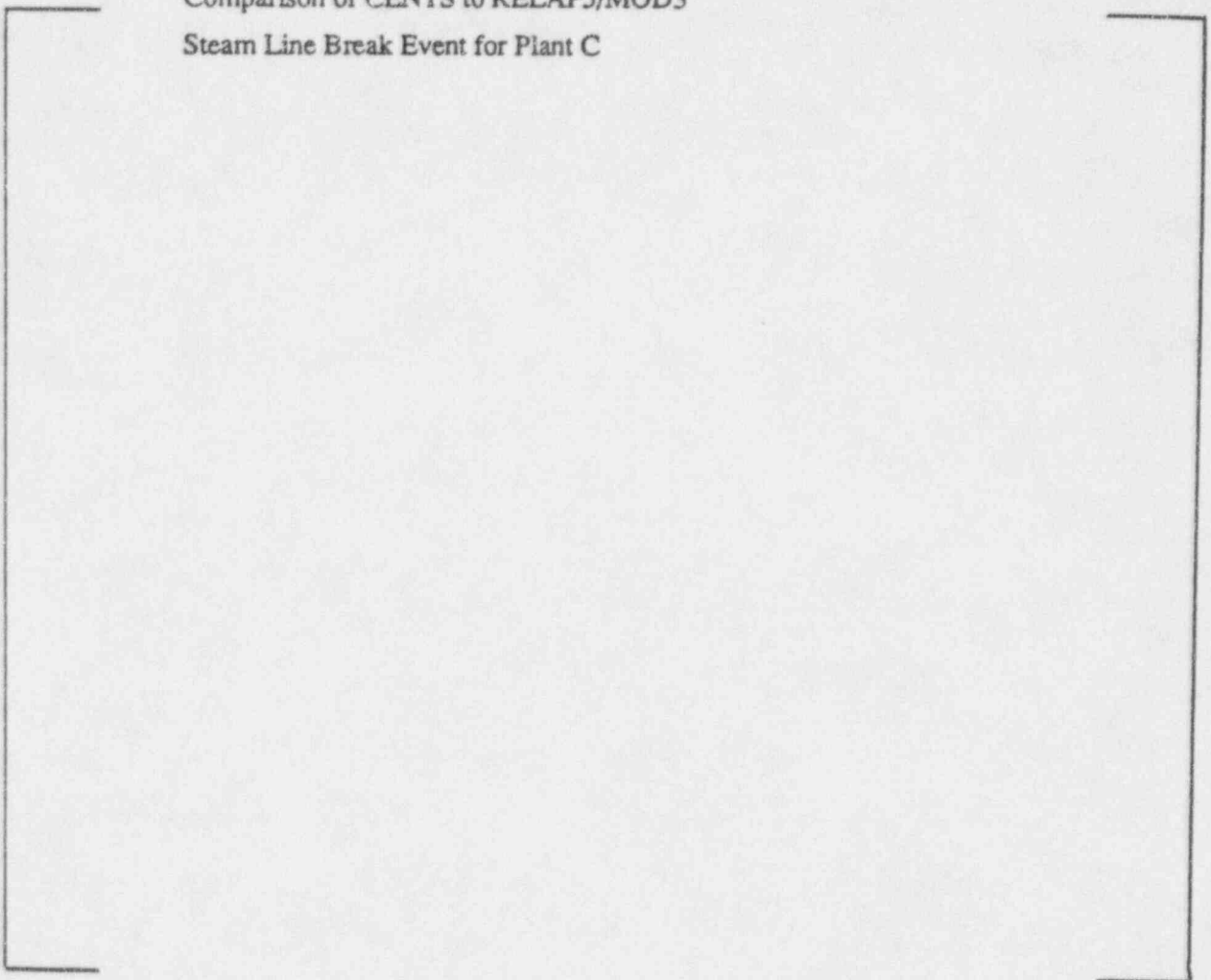


Figure 2.3.4.K Cold Leg Temperature - Intact Loop

Comparison of CENTS to RELAP5/MOD3
Steam Line Break Event for Plant C

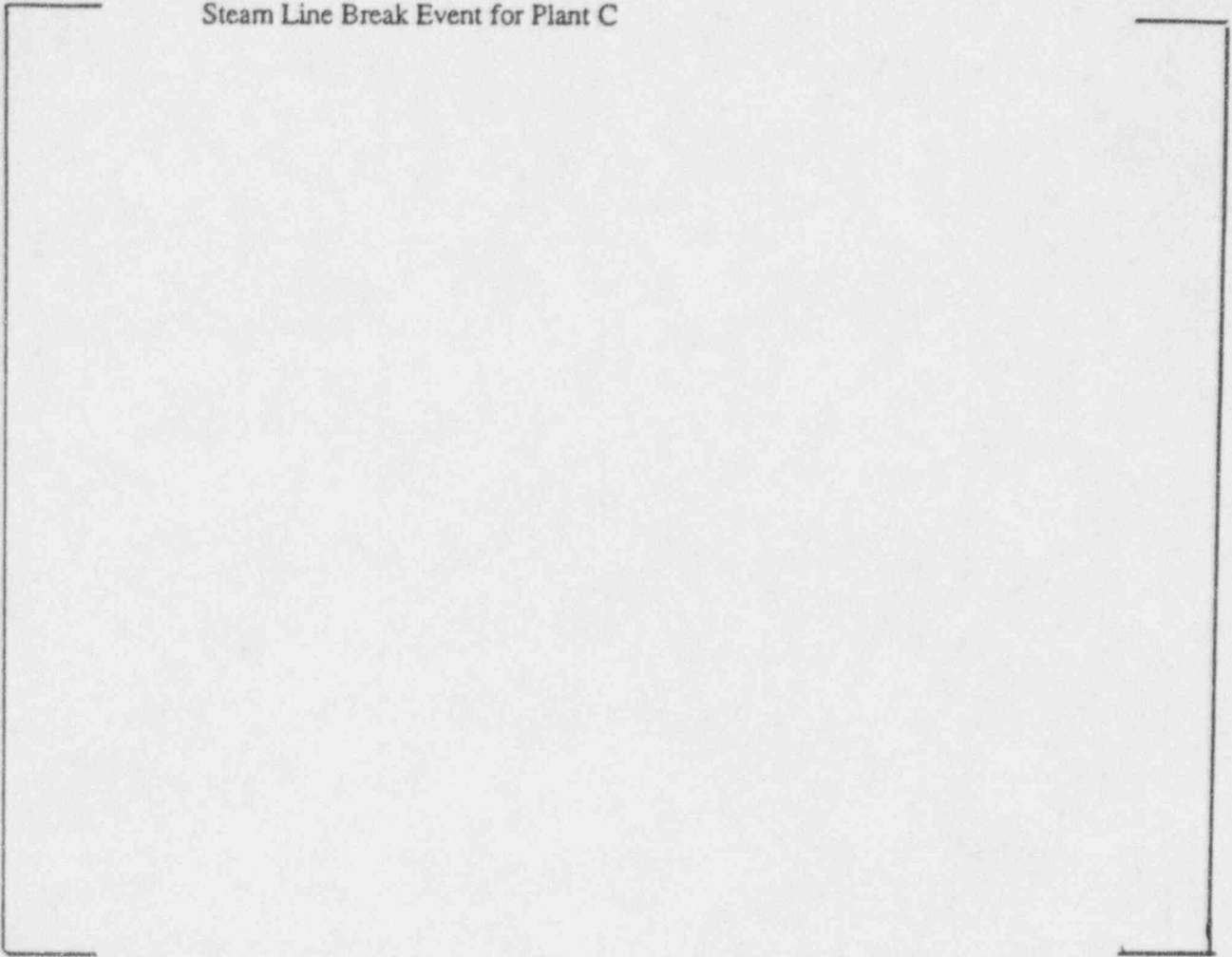


Figure 2.3.4.L Fuel Average Temperature

Comparison of CENTS to RELAP5/MOD3
Steam Line Break Event for Plant C

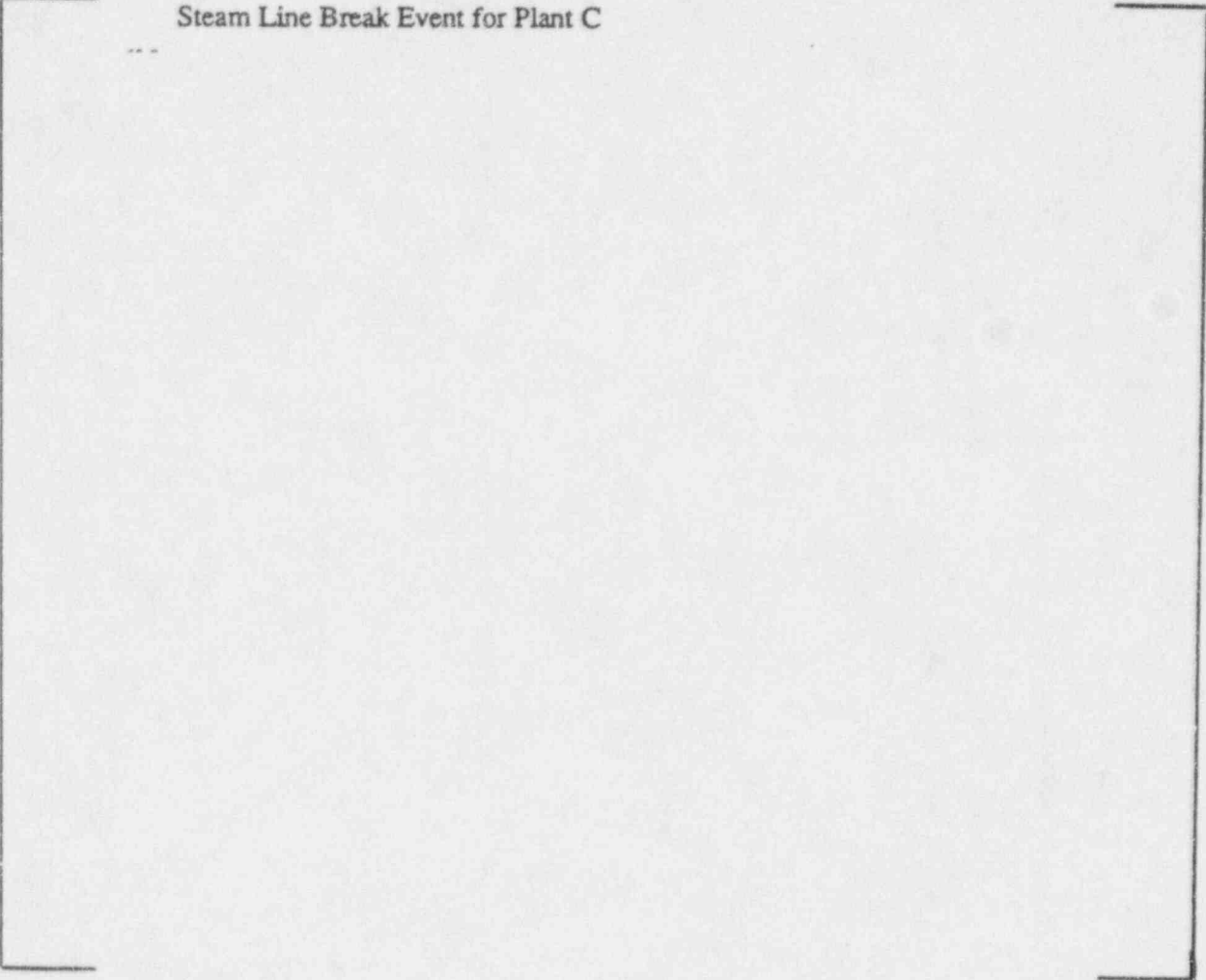


Figure 2.3.4.M Loop Mass Flow

Comparison of CENTS to RELAP5/MOD3
Steam Line Break Event for Plant C

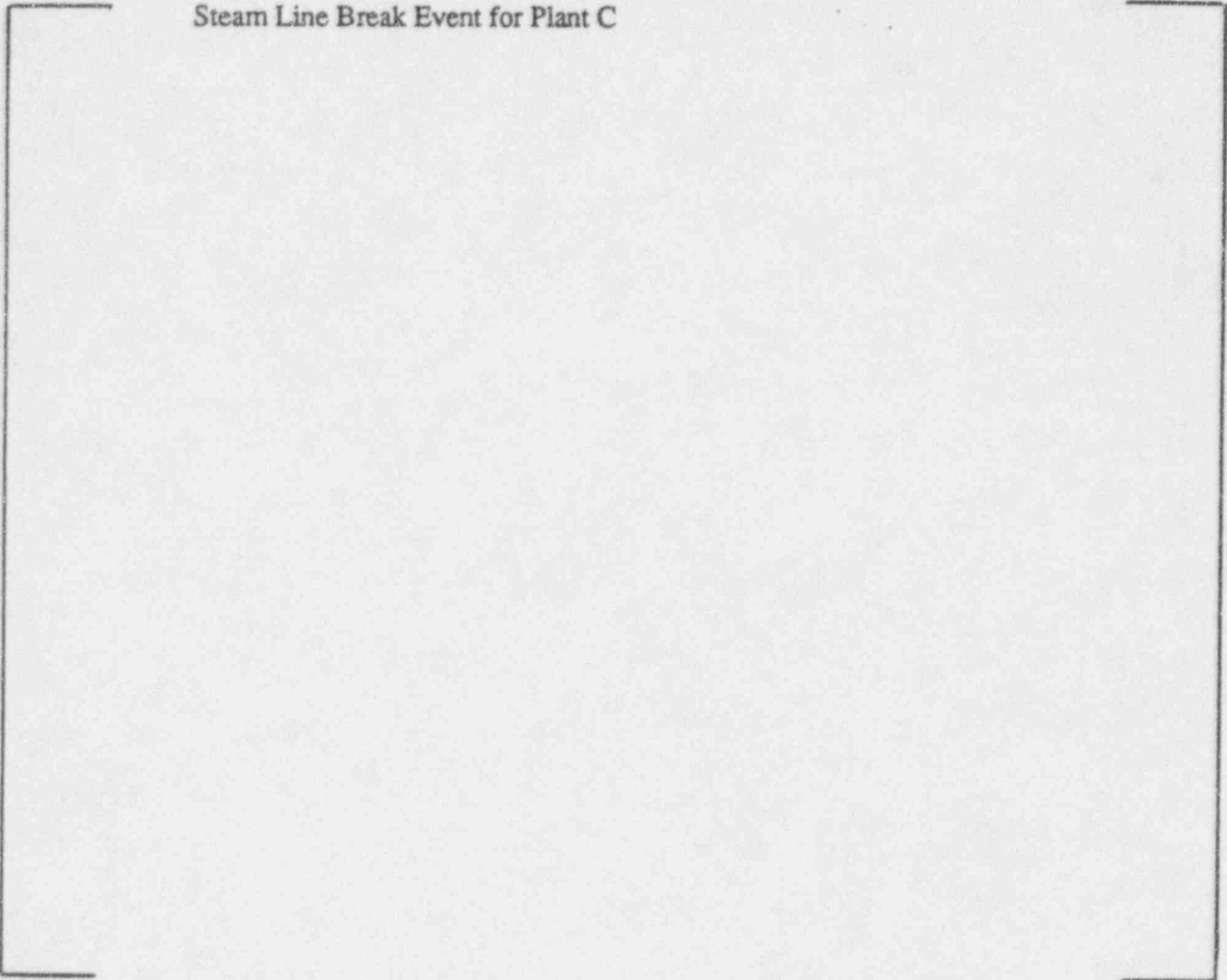


Figure 2.3.4.N Total HPSI Flow

Comparison of CENTS to RELAP5/MOD3
Steam Line Break Event for Plant C

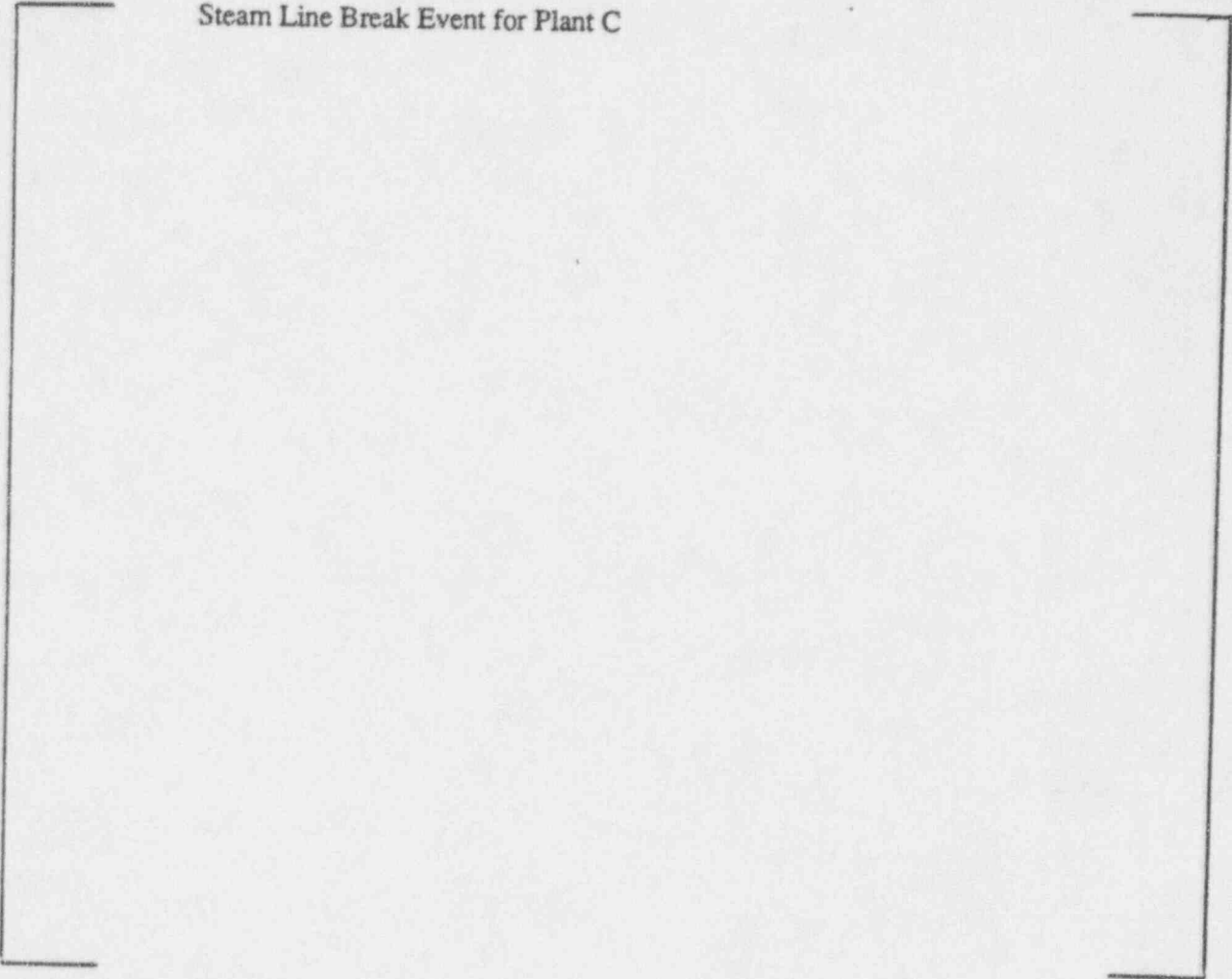


Figure 2.3.4.O Total Reactivity

Comparison of CENTS to RELAP5/MOD3
Steam Line Break Event for Plant C

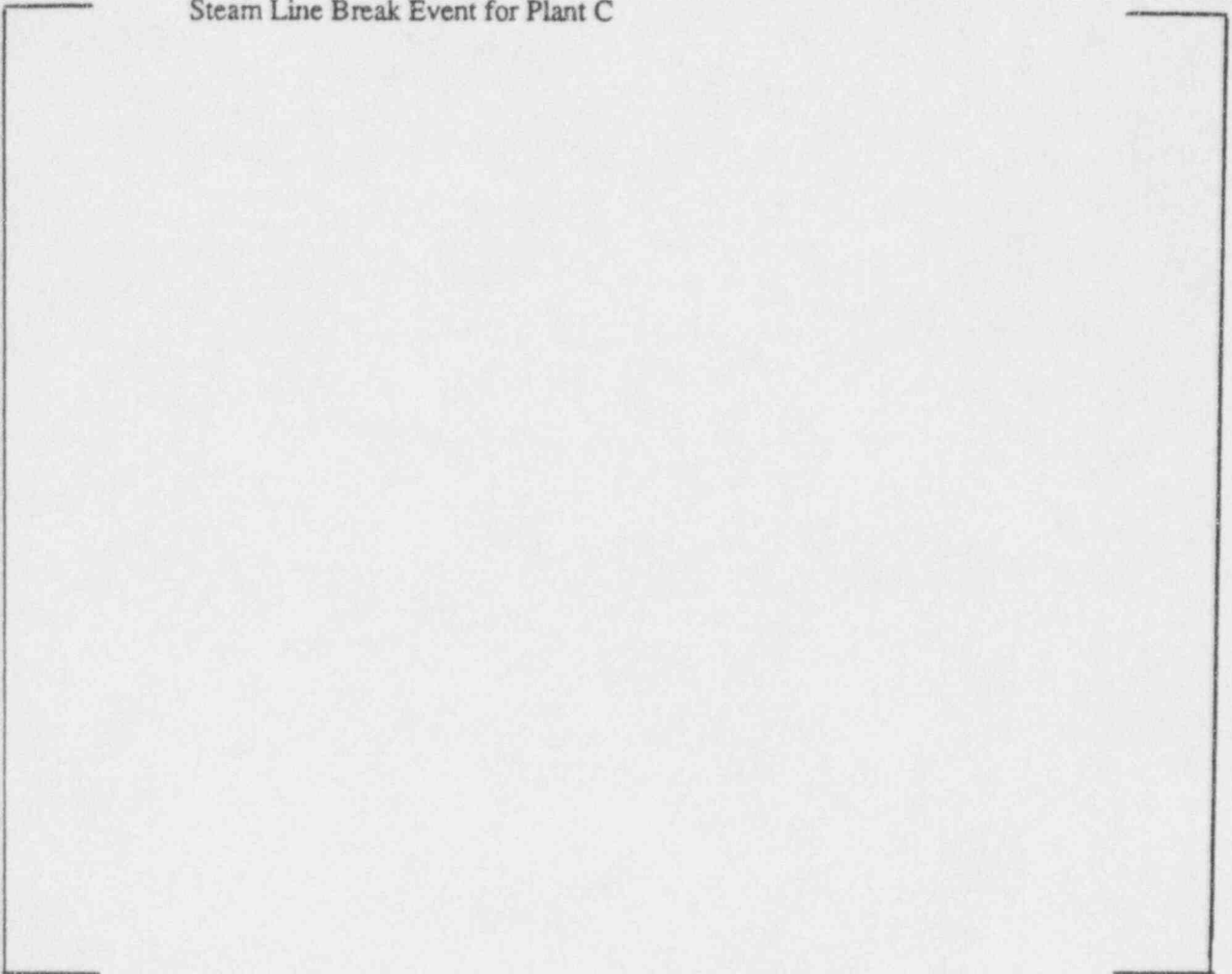


Figure 2.3.4.P Boron Reactivity

3.0 Conclusions

This volume presents a comprehensive set of benchmark cases for the CENTS computer code. The cases demonstrate that the CENTS models can accurately predict Westinghouse type PWR plant response to upset conditions. In addition, various options were tested which force the CENTS code to provide conservative predictions of plant response. The verification effort supports the following conclusions:

1. CENTS has a numerically stable solution methodology with a proper conservation of mass, momentum, and energy.
2. CENTS reproduces measured plant behavior for a range of different events. Deviations from plant behavior are generally within the uncertainty of the measurement.
3. CENTS satisfactorily reproduces the plant behavior as predicted by RELAP5/MOD3, the NRC-approved code used for wide range of safety analyses. Differences between the predictions of CENTS and the RELAP5/MOD3 code can be generally ascribed to differences in the details of the models used in RELAP5/MOD3.
4. CENTS is basically a best estimate code. Appropriate conservatism of licensing analyses of non-LOCA design basis events is introduced primarily through code input. In a few instances, code options are available which enable the analyst to choose a conservative, rather than a best-estimate, model of a physical process.

CENTS is shown here to be capable of predicting system response for Westinghouse type PWR non-LOCA design basis events for a range of operating conditions. Thus, CENTS can be effectively used as a predictive tool for licensing analyses of non-LOCA events.

4.0 References

1. "RELAP5/MOD3 Code Manual, Vol. I,II,III,IV,V", NUREG/CR-5535, EGG-2596, EG&G, June, 1990.
2. "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants", NUREG-0800, U.S. Nuclear Regulatory Commission, July 1981.

Response to NRC
Request for Additional Information
on Volume 4 of CENTS Topical Report

September 1994

Korea Atomic Energy Research Institute

ABB Combustion Engineering Nuclear Operations
Windsor, CT

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Question 1 :

Discuss and provide the basis for any changes made to the CENTS code (relative to the version employed in the Volume-3 "ABB-CE plant benchmarking) in order to perform the Westinghouse plant analyses of Volume-4.

Response :

A number changes have been made to the CENTS code since the Volume-3 ABB-CE plant benchmarking was completed. Although these changes were made during preparation of Volume-4, none of the changes to CENTS are specific to W type plants. ABB-CE will maintain a single code version which is applicable to both ABB-CE plants and to W type plants.

The following code changes have been made:

- 1) Many in-line comments have been added and program statements have been indented for readability.
 - 2) A few cases of 'dead code' and stray variables were eliminated.
 - 3) Options were added to provide improved formatting of CENTS output.
 - 4) A few specific fixes were made to avoid errors such as divide-by-zero, inconsistent subroutine calls, etc.
 - 5) The numerical solution to the point kinetics solution was improved. These changes were made to improve the accuracy for cases initiated at subcritical conditions and for cases with very high rates of reactivity insertion.
 - 6) A single set of changes to the CENTS models were made as a direct result of the RELAP5/MOD3 benchmarking. As discussed in the response to Question 11 below, both CENTS and RELAP5/MOD3 calculated moisture carryover for the steam line break event. The comparison disclosed that the CENTS model was somewhat inaccurate. Two model changes were made which affect the CENTS moisture carryover calculation. [
-] These model changes only affect SLB cases for which the moisture carryover option has been selected. The changes do not affect 'licensing cases' for which the CENTS option for no moisture carryover is selected.
- 7) As discussed in the Supplement-1 responses, the CENTS fuel rod radial nodalization was improved so that up to eight radial nodes may now be defined for the fuel pellet.

The CENTS version that was used for the work presented in the Supplement-1 responses to questions included all of these changes. The work presented in Volume-4 used an intermediate version which did not include the new fuel rod radial nodalization (i.e., change number 7). However, all of the Volume-4 CENTS cases have since been rerun with the current version. A 3 radial node model was defined for the fuel pin (one node for the pellet) for consistency with the earlier cases. As expected, the results were nearly identical.

Question 2 :

Have any changes been made in the level of detail in the plant/core modeling and numerical description used in the Chapter-2 benchmark calculations, relative to that used in Volume-3? If so, provide the basis for these changes.

Response :

Except as discussed in the response to Question 1, no changes were made in the level of detail in the plant/core modeling and numerical descriptions used in the Chapter-2 benchmark calculations. The same approach for representing the components with nodes, flow paths and component models was used. Differences in the number of loops and in the details of the auxiliary systems between a C-E and W plant were accommodated through appropriate modification of the input.

Figure 1 shows the nodalization used for the Westinghouse type plant. Figure 2 shows the nodalization used for the ABB-CE type plant. As shown in these figures, equivalent nodalization was used.

Question 3 :

To what specific W licensing analyses will CENTS be applied? Discuss the adequacy of the Chapter-2 benchmarking for representing these transients.

Response :

CENTS will be used for licensing analysis of all non-LOCA design basis events. It will not be used for LOCA or severe accident analysis.

The list of design basis events presented in the FSAR for a Westinghouse type plant is nearly identical to that for a ABB-CE type plant. The events for which CENTS would be used are :

- Excess Heat Removal by the Secondary Side
- Steam Line Break
- Loss of Secondary Load
- Loss of Offsite Power
- Loss of Normal Feedwater Flow
- Feed Line Break
- Partial and Complete Loss of RCS Flow
- Seized RCP Rotor
- Uncontrolled Rod Bank Withdrawal from zero power, low power, and full power initial conditions
- Inadvertent Boron Dilution
- Inadvertent Actuation of the ECCS
- Steam Generator Tube Rupture

In addition to the analyses of the above events, CENTS may be used for safety related analyses of events which are not presented in detail in the plant FSAR. For example, parametric analyses at various initial conditions may be performed in order to determine or verify RPS setpoints. CENTS may also be used to determine the effect of plant modifications or changes to plant operation on the results of the safety analyses.

The basis for the case selection for the code-to-code benchmarking comparisons presented in Volume-4 was to challenge the CENTS models in order to identify any significant differences in the results of CENTS and RELAP5/MOD3. That is, the cases were selected so as to represent the most severe transients for the heat-up transients, flow transients, RCS inventory loss transients, and cooldown transients. Thus, the cases which were selected are sufficient to validate the use of CENTS for the analysis of design basis events for Westinghouse type plants.

Question 4 :

In the Chapter-2 benchmarking, have all codes used to determine input for the CENTS model and the correlations used in the analysis been approved for application to W plants?

Question 5 :

How were the reactivity inputs determined for the Chapter-2 benchmarking analyses?

Question 12 :

Were the same codes and procedures used to determine the input for the Volume-4 W plant model, as were used in the Volume-3 CE plant model? If not, provide the bases for these differences.

Response :

The following response addresses the issues presented in Questions 4, 5, and 12.

The input for the CENTS plant model of a W type plant used essentially the same procedures as were used in the Volume-3 CE plant model.

Most of the inputs required by CENTS are not the direct output of other computer codes. The input is prepared by a series of hand calculations which define nodal volume and height, flow path connections, flow path resistances, etc. Control and protection systems and reactor trip functions are represented by series of simple modules (adders, multipliers, leads, lags, etc) that represent the function of these systems. The reactor core is represented by a simple model of an average fuel rod. Auxiliary systems are represented by tabular input to standard component models.

Primary sources are used as the basis for most of these calculations. Data is taken from plant drawings, system descriptions, etc. In some cases, data is taken from the plant FSAR.

The following CENTS inputs are based on the results of other computer codes :

[

]

Question 6 :

How were the flow mixing and the spatial weighting of the moderator temperature reactivity feedback determined in Chapter-2 RELAP5/MOD3 and CENTS steamline break analysis?

Response :

The nodalization of the available RELAP5/MOD3 deck results in complete mixing of the RCS fluid in the reactor vessel downcomer. In order to perform a meaningful comparison of the two computer codes, the CENTS flow mixing and spatial weighting terms were chosen to also produce perfect mixing.

In practice, the steamline break analysis will not assume perfect mixing. Conservative mixing factors will be justified case by case.

Note that the CENTS flow mixing and spatial weighting algorithms were checked as part of the Volume-3 benchmarking.

Question 7 :

How will the CENTS flow mixing factors used in the steamline break analysis be determined for the intended W plant design?

Response :

The most conservative approach is to assume that no mixing occurs in reactor vessel. This results in the greatest cooldown of the core cold edge. This assumption has been used for the SLB analyses for the plant chosen for the Volume-4 benchmarking. These analyses, using the NLOOP code, have been accepted by the Korean regulatory authority.

This conservative approach is possible because margin exists to the regulatory limits for Plant C.

Flow mixing tests have been performed for W type plants. It is expected that, in most cases, mixing data will be available through the plant owner for which a SLB analysis is performed. Alternately, a detailed flow model of the reactor vessel could provide equivalent data. Thus, ABB-CE does not expect to need to use the overly conservative assumption which has been used for Plant C.

Question 8 :

To what Westinghouse plant designs will CENTS be applied? Discuss the adequacy of the Volume-4 benchmarking comparisons, for the 2775 MW three-loop plant, for representing the intended Westinghouse plant designs.

Response :

CENTS will be used for the analysis of 2, 3, and 4 loop W type plants.

The benchmarking performed in Volumes 3 and 4 is sufficient to show that the CENTS models are adequate to represent a variety of RCS configurations. The models in CENTS are formulated in a manner that is not dependent on the number of loops. The underlying methods are based on generic node/flowpath models as is done for other codes, such as RELAP5/MOD3, which are not specific to a particular plant geometry.

Question 9 :

What are the design and operational differences between the ABB-CE and W plants that have a significant effect on the validity of the CENTS modeling approximations? Discuss the adequacy of the CENTS modeling for accommodating these differences.

Response :

There are no design or operational differences between the ABB-CE and W type plants that have a significant effect on the validity of the CENTS modeling. Differences in the plant systems can be accommodated through the use of appropriate nodalization and by using appropriate input to represent the specific control and protection systems.

Some of the differences between the two types of plants which were considered in reaching this conclusion include the following :

The number of loops and number of cold legs per loop are different between ABB-CE and W type plants. This difference is accommodated by CENTS general node/flowpath modeling.

Fuel assemblies for W type plants may use either a 14x14, 15x15, 16x16, or 17x17 array of fuel rods. Fuel assemblies for ABB-CE type plants are either 14x14 or 16x16 arrays. In addition, there are other significant differences in fuel assembly design. These differences are accommodated by choosing the proper input to the CENTS core model.

Most W type plants employ Ag-In-Cd scram and control rods. ABB-CE plants use B4C pellets in thicker rods. The ABB-CE scram rods do not insert as quickly as the thinner rods. Total scram worth is typically higher for ABB-CE plant. Differences in scram worth and scram reactivity vs time are accommodated by tabular input data.

The logic for several of the reactor trips and control systems are different. CENTS represents the different controller logic using plant specific input to the generic controller module.

The differences in the design of NSSS components such as the steam generators, reactor coolant pumps, reactor vessel, and the pressurizer are accommodated by the generality of the CENTS node/flowpath models.

There are differences in safety injection pump shutoff head and flow, charging pump flow, auxiliary feedwater pump flow, etc. These differences are accommodated by tabular input data.

Question 10 :

Discuss and provide the basis for any changes that have been made to RELAP5/MOD3 (relative to Reference-1). What is the effect of these changes on the Chapter-2 benchmarking comparisons.

Response :

A single change was made to the RELAP5/MOD3 code. A large mass truncation error was accumulated in the secondary side for the SLB case. The error limits for mass truncation are defined in the RELAP5/MOD3 code itself as opposed to being defined by input. The RELAP5/MOD3 code was modified so that the error limits were reduced by a factor of ten.

This change produced much better comparison between the RELAP5/MOD3 and CENTS results for the SLB case. The RELAP5/MOD3 results using the old limits were obviously incorrect. The accumulated error using the revised code was negligible.

Question 11 :

Have any adjustments been made to either the CENTS or RELAP5/MOD3 coding or plant modeling to improve the agreement of the Chapter-2 benchmark comparisons? If so, discuss the effect of these changes on the comparisons.

Response :

As discussed in the responses to Questions 1 and 10, a few changes were made to RELAP5/MOD3 and to CENTS in order to correct deficiencies discovered during the benchmarking. However, the coding changes are not case specific. Neither the coding nor the plant modeling was 'tuned' case-by-case to improve the results of the benchmarking.

The steam line break case which was used for the benchmarking was defined in a way that accommodated limitations of the available RELAP5/MOD3 model. The CENTS moisture carryover option was set so that CENTS calculated moisture carryover during the event. This was necessary because RELAP5/MOD3 does not have an option to defeat the moisture carryover calculation. Similarly, the available RELAP5/MOD3 model has only one node for the vessel inlet plenum so that perfect mixing occurred in the reactor vessel. In order to perform a meaningful code-to-code comparison it was necessary to set the CENTS mixing factors to produce perfect mixing.

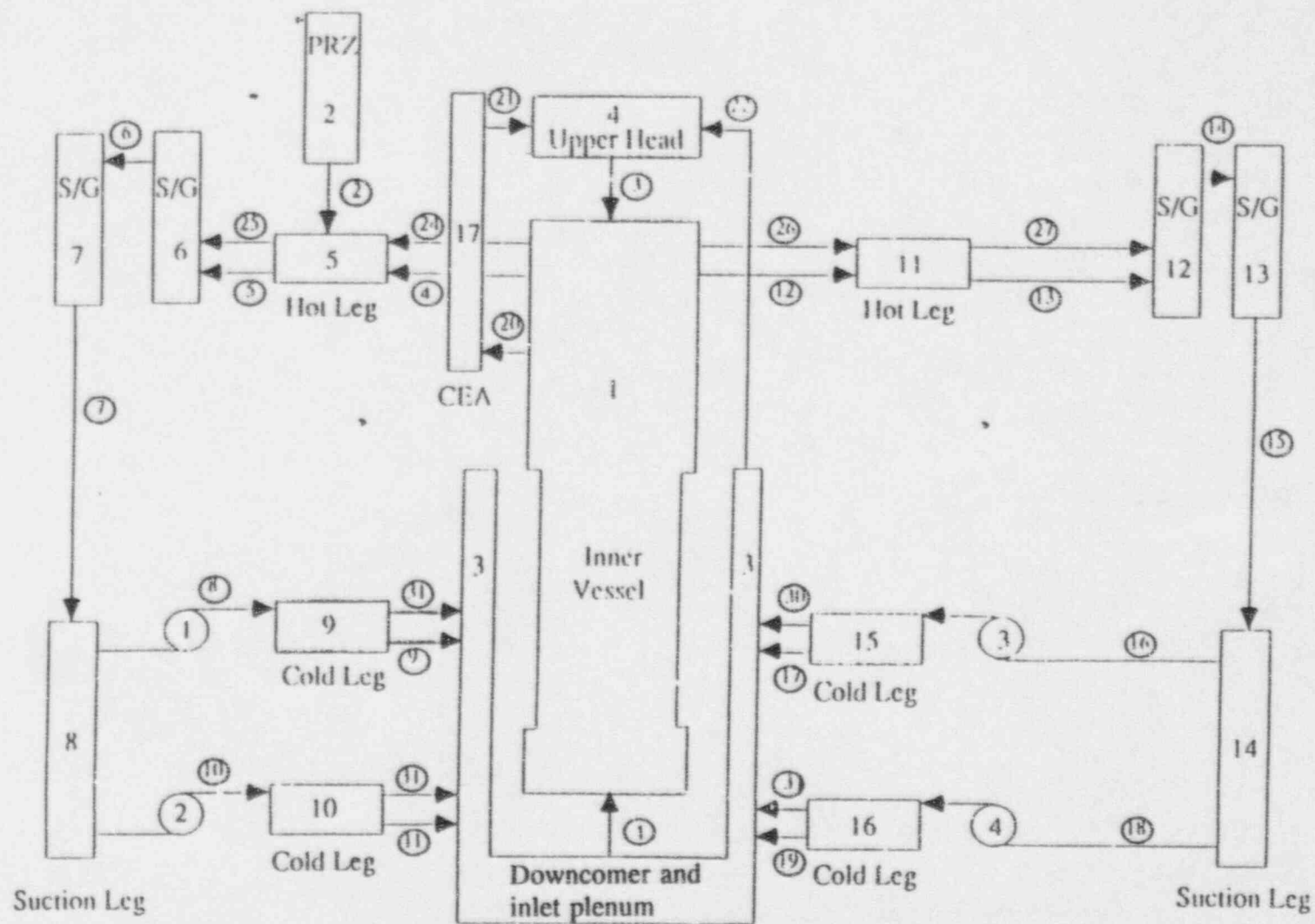


Figure 2. CENTS Model of a ABB-CE Type Two-Loop PWR



December 20, 1994
LD-94-075

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Response to Request for Additional Information on the CENTS Code
Topical Report

Dear Mr. Kopp:

In a telephone conversation on November 25, 1994, Mr. John Carew of Brookhaven National Laboratory (BNL) requested that ABB Combustion Engineering respond to four additional questions about the CENTS Code point kinetics solution. The responses to these questions are provided in Enclosure I.

Although only Volume 4 of the CENTS Code Topical Report is currently under review by the NRC, the responses provided in Enclosure I are generic and also apply to Volumes 1 through 3 of the CENTS Topical Report, already approved by the NRC. It should be noted that the description of the point kinetic solution in these approved Volumes are more general than described in the Enclosure and do need to be revised. If required, however, the responses in Enclosure I will be included as part of the approved version of Volume 4 with a note indicating that they apply to all volumes of the CENTS Code Topical Report.

We anticipate that these responses will allow the NRC reviewers at BNL to complete their review of the Topical Report. If there are any more questions on the CENTS Topical Report, please do not hesitate to contact me or Mr. Mario Robles of my staff at (203) 285-5215.

Very truly yours,

S. A. Toelle
Manager
Nuclear Licensing

mr:kpr

Enclosures: 1

ABB Combustion Engineering Nuclear Power

Document Control Desk
December 20, 1994

LD-94-075
Page 2

cc: J. Carew (BNL)
R. C. Jones (NRC)
L. I. Kopp (NRC)

Question 1 :

Are any revisions to previously submitted material required due to the changes made to the point kinetics solution described in Response 1 of Supplement 2?

Response :

No changes are required. The motivation for making the change to the CENTS neutronics solution was to bring the code into conformance with the description in Volume 1. Equation 3.7 of section 3.1.1 of Volume 1 includes the contribution from a fixed neutron source. However, the original CENTS solution did not include this contribution. The current model conforms to the model described in section 3.1.1.

The other changes to the neutronics solution were made as a result of testing for high reactivity insertions. These changes affected specific details of the numerical solution. These changes are described in response 3 below.

Although specific details of the numerical methods were changed, the basic solution strategy was not changed. The current documentation correctly describes the equations which are solved.

Question 2 :

Which of the benchmark cases described in Volumes 1-4 and supplements 1 and 2 used the original CENTS point kinetics solution? Do these cases remain valid?

Response :

The benchmark cases described in Section 8 of Volume 2 and in Volume 3 used the original CENTS point kinetics solution. The benchmark cases in Supplement 1, Volume 4, and Supplement 2 used the revised code.

In particular, the revised code was used for the comparisons to CESEC for the CEA Withdrawal event from subcritical and from hot zero power conditions given in Appendices B and C of Supplement 1.

The revised code was also used for the timing study described in Appendix A of Supplement 1. Figures A17 and A18 show the affect of time step size on the results of the CEA withdrawal event from subcritical and from hot zero power conditions.

The earlier work which used the original CENTS numerical algorithm for the solution of the point kinetics equations did not involve high reactivity insertions and remain valid.

The Volume 3 comparison cases have been rerun with the current version of CENTS. The changes to the numerical methods used to solve the neutronics equations had no appreciable affect on the results.

Question 3 :

Describe the changes which were made to the point kinetics solution.

Response :

Definition of Terms

The standard point kinetics equations are

$$1) \quad \frac{d}{dt}n(t) = \frac{\rho - \bar{\beta}}{l^*}n(t) + \sum_{i=1}^6 \lambda_i C_i(t) + S_{neutron}$$

and

$$2) \quad \frac{d}{dt}C_i(t) = \frac{\beta_i}{l^*}n(t) - \lambda_i C_i(t)$$

where

- $n(t)$ is the neutron number density
- ρ is the total reactivity
- l^* is the prompt neutron lifetime
- β_i is the effective delayed neutron fraction for the i 'th delayed neutron source group
- $\bar{\beta}$ is the total beta fraction
- $C_i(t)$ is the concentration of the i 'th delayed neutron precursor group
- λ_i is the decay constant of the i 'th delayed neutron source group
- $S_{neutron}$ is the effective neutron source term

The equations are normalized by defining the normalized precursor concentration

$$3) \quad \chi_i(t) = \frac{l^* \lambda_i C_i(t)}{\beta_i n(0)}$$

the normalized source term

$$4) \quad S = \frac{S_{neutron}}{l^* n(0)}$$

and the fractional core power.

$$5) \quad P(t) = \frac{n(t)}{n(0)}$$

The resulting equations are

$$6) \quad \frac{d}{dt}P(t) = \frac{\rho - \beta}{l^*}P(t) + \frac{1}{l^*} \sum_{i=1}^6 \beta_i \chi_i(t) + \frac{S}{l^*}$$

and

$$7) \quad \frac{d}{dt}\chi_i(t) = \lambda_i(P(t) - \chi_i(t))$$

With this normalization, $P(t) = 1.0$ at the initial power and $\chi_i(t) = P(t)$ at steady state.

Solution

CENTS uses a central difference approximation to solve the kinetics equations. The central difference formula is

$$8) \quad y(\Delta t) = y(0) + \left(\frac{\dot{y}(0) + \dot{y}(\Delta t)}{2} \right) \Delta t$$

This approximation neglects terms of the Taylor expansion which are of order Δt^3 or higher.

Applying the central difference approximation to (6) gives

$$9) \quad P(\Delta t) = P(0) + \left[\frac{\rho - \beta}{l^*} P(0) + \frac{1}{l^*} \sum_{i=1}^6 \beta_i \chi_i(0) + \frac{\rho - \beta}{l^*} P(\Delta t) + \frac{1}{l^*} \sum_{i=1}^6 \beta_i \chi_i(\Delta t) + \frac{2S}{l^*} \right] \frac{\Delta t}{2}$$

Solving for $P(\Delta t)$ yields:

$$10) \quad P(\Delta t) = \frac{\left(1 + \frac{\rho - \beta}{2l^*} \Delta t \right) P(0) + \frac{\Delta t}{l^*} \sum_{i=1}^6 \beta_i \frac{\chi_i(0) + \chi_i(\Delta t)}{2} + \frac{S \Delta t}{l^*}}{1 - \frac{\rho - \beta}{2l^*} \Delta t}$$

Implementation

The point kinetics equations are integrated over a time interval T_{interval} which is the timestep being used by the other CENTS algorithms. Reactivity is constant over this interval. CENTS integrates the equations by dividing the T_{interval} into smaller intervals (Δt) and solving (10) for each subinterval.

The original CENTS algorithm was :

- Step 1 The neutronics timestep is user input. Typical values were in the range of .01 to .05 seconds.
- Step 2 If the denominator of Equation (10) is close to zero, a constant parameter analytical solution is used in the place of the trapazoidal integration. In practice, such high reactivity insertions were not reached.
- Step 3 For each subinterval (Δt) :

3a Approximate $\chi_i(t+\Delta t) = \chi_i(t) + \lambda_i(P(t) - \chi_i(t))\Delta t$.

- 3b Calculate $P(t)$ using equation 10, except that no contribution from a neutron source is included :

$$P(t+\Delta t) = \frac{\left(1 + \frac{\rho - \beta}{2l^*} \Delta t\right) P(t) + \frac{\Delta t}{l^*} \sum_{i=1}^6 B_i \frac{\chi_i(t) + \chi_i(t+\Delta t)}{2}}{1 - \frac{\rho - \beta}{2l^*} \Delta t}$$

The current CENTS algorithm is :

- Step 1 Choose the neutronics timestep Δt . Δt is limited to the range $.0005 \leq \Delta t \leq .05$
- Step 2 If necessary, reduce Δt to ensure that the denominator of Equation 10 is not zero or negative.
- Step 3 For each subinterval (Δt) :

3a Approximate $\chi_i^0(t+\Delta t) = \chi_i(t) + \lambda_i(P(t) - \chi_i(t))\Delta t$.

This is a first-order approximation.

3b Calculate $P(t)$ using equation 10 :

$$P(t+\Delta t) = \frac{\left(1 + \frac{\rho - \beta}{2I^*} \Delta t\right) P(t) + \frac{\Delta t}{I^*} \sum_{i=1}^6 B_i \frac{\chi_i(t) + \chi_i^0(t+\Delta t)}{2} + \frac{S \Delta t}{I^*}}{1 - \frac{\rho - \beta}{2I^*} \Delta t}$$

3c Recalculate

$$\chi_i(t+\Delta t) = \chi_i(t) + \lambda_i \left(\frac{P(t) + P(t+\Delta t)}{2} - \frac{\chi_i(t) + \chi_i^0(t+\Delta t)}{2} \right) \Delta t .$$

This is a second order approximation and is the value which is saved to be used as the starting value for the next timestep.

3d If power has changed by more than 2% during the current kinetics timestep, set $\Delta t = 0.9 \Delta t \left(\frac{.02}{\text{Power Change}} \right)$ and restart the calculation at step 1. Note that this constraint is much more restrictive than Step 2.

Comparison of New and Old Solutions

The new algorithm differs from the previous solution in four ways:

- 1) The contribution from a fixed source is now included.
- 2) The neutronics timestep size is calculated by the program rather than being chosen by the user. The logic ensures that the timesteps are small enough for good accuracy. The lower limit on the allowed timestep size prevents inaccuracy due to round-off errors.
- 3) The central difference formula is used consistently. The switch to an alternate numerical solution was eliminated.
- 4) Step 3c was added. As shown below in the response to question 4, the correction 3c increases the accuracy of the solution by about an order of magnitude for a given neutronics timestep size.

Question 4 :

Describe any testing which has been performed for the revised algorithm.

Response :

As discussed in the response to Question 2, the comparisons to CESEC provided in Supplement 1 used the new algorithm. In addition, separate testing of the new solution was performed in order to assess the accuracy and to choose the constants which are used in steps 1 and 3d above.

A separate driver program was written to test the revised algorithm against the original algorithm and against the CESEC Runge-Kutta scheme. A step change in reactivity was modeled. The effect of temperature feedbacks was not included, nor was the contribution of decay power.

Description of Test Case

The kinetics equations were integrated over a one second interval using time steps of 0.2 seconds subdivided into kinetics time steps in the range of 10^{-6} to 0.05 seconds. Typical kinetics parameters were used and are listed in Table 4.1. A fixed source was not modeled. The case assumed a relatively large reactivity insertion of

$\Delta\rho = 0.95 \bar{\beta}$. In response to this reactivity insertion, the power increased by a factor of more than 3000 during the one second interval.

Steps 1 and 3d of the new algorithm were disabled for the test in order to determine the effect of time step size on accuracy.

Although CENTS uses single precision arithmetic, the cases were repeated using double precision arithmetic. This made it possible to see the effect of arithmetic round-off errors.

The Runge-Kutta solution used double precision arithmetic. The error criterion for the solution was set at 1.0×10^{-6} .

Results of Test

Tables 4.2 and 4.3 and Figures 4.1 and 4.2 below give the results of the study.

Using double precision arithmetic, both solutions converged to the same value of power (3812.67) as timestep size was reduced. The covered value agreed with the value determined by the Runge-Kutta solution. The new CENTS solution converged much more rapidly than the original CENTS solution. Table 4.3 and Figure 4.1 show that the error over the interval decreases linearly with the time step size using the old solution and that the error decreases with the square of the time step size using the new solution.

Figure 4.2 plots the error for timesteps less than 0.001 sec. Using single precision arithmetic, both the old and the new solution show a significant effect due to accumulated round-off errors when time step size is decreased below about .00005 sec.

Finally, a case was run with steps 1 and 3d above enabled. This is a direct test of of the current CENTS algorithm. The calculated value of power (3813.04) was accurate to 0.016% relative error.

TABLE 4.1
KINETICS PARAMETERS FOR TESTING

<u>Group</u>	β_i ——	λ_i
1	0.00017	0.013
2	0.00119	0.031
3	0.00105	0.124
4	0.00219	0.328
5	0.00085	1.406
6	0.00021	3.784

$$\bar{\beta} = \sum_{i=1}^6 \beta_i = 0.00566$$

TABLE 4.2
CALCULATED POWER VS $\Delta T_{\text{kinetics}}$

$\Delta t_{\text{kinetics}}$	<u>Double Precision</u>		<u>Single Precision</u>	
	<u>New</u>	<u>Old</u>	<u>New</u>	<u>Old</u>
1.00E-06	3813.67	3813.62	3795.10	3795.05
5.00E-06	3813.67	3813.41	3807.88	3807.62
1.00E-05	3813.67	3813.16	3807.69	3807.19
5.00E-05	3813.67	3811.12	3813.59	3811.04
1.00E-04	3813.67	3808.57	3813.59	3808.49
5.00E-04	3813.64	3788.27	3813.57	3788.19
1.00E-03	3813.56	3763.10	3813.54	3763.09
5.00E-03	3810.92	3569.77	3810.90	3569.76
1.00E-02	3802.63	3346.77	3802.62	3346.76
2.00E-02	3769.39	2954.80	3769.38	2954.80
4.00E-02	3638.39	2340.30	3638.38	2340.30
5.00E-02	3543.30	2097.62	3543.30	2097.62

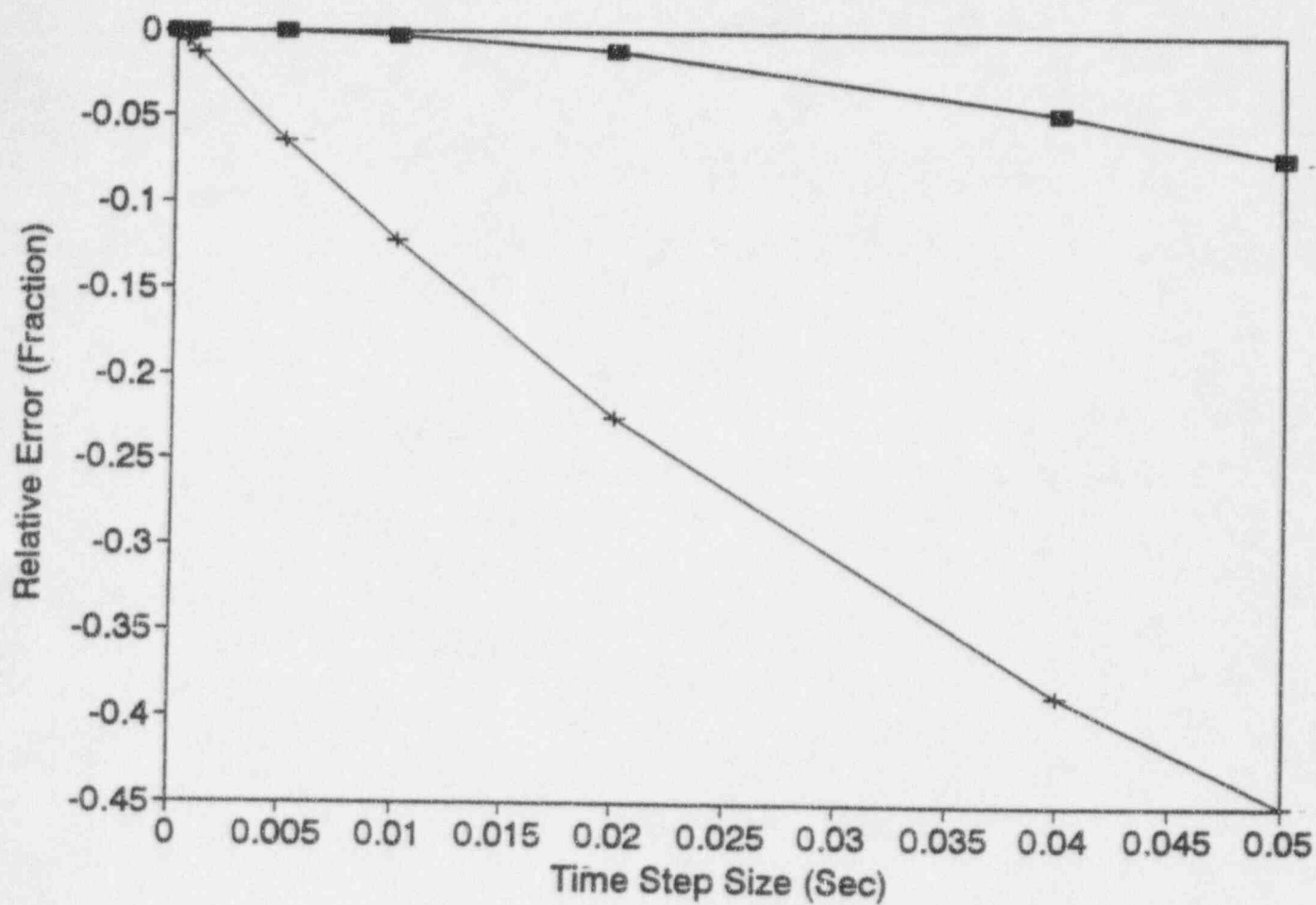
Runge Kutta result : 3813.67

Current CENTS algorithm : 3813.04

TABLE 4.3
RELATIVE ERROR VS $\Delta T_{\text{kinetics}}$

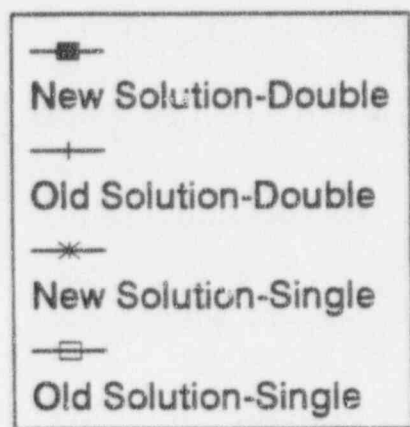
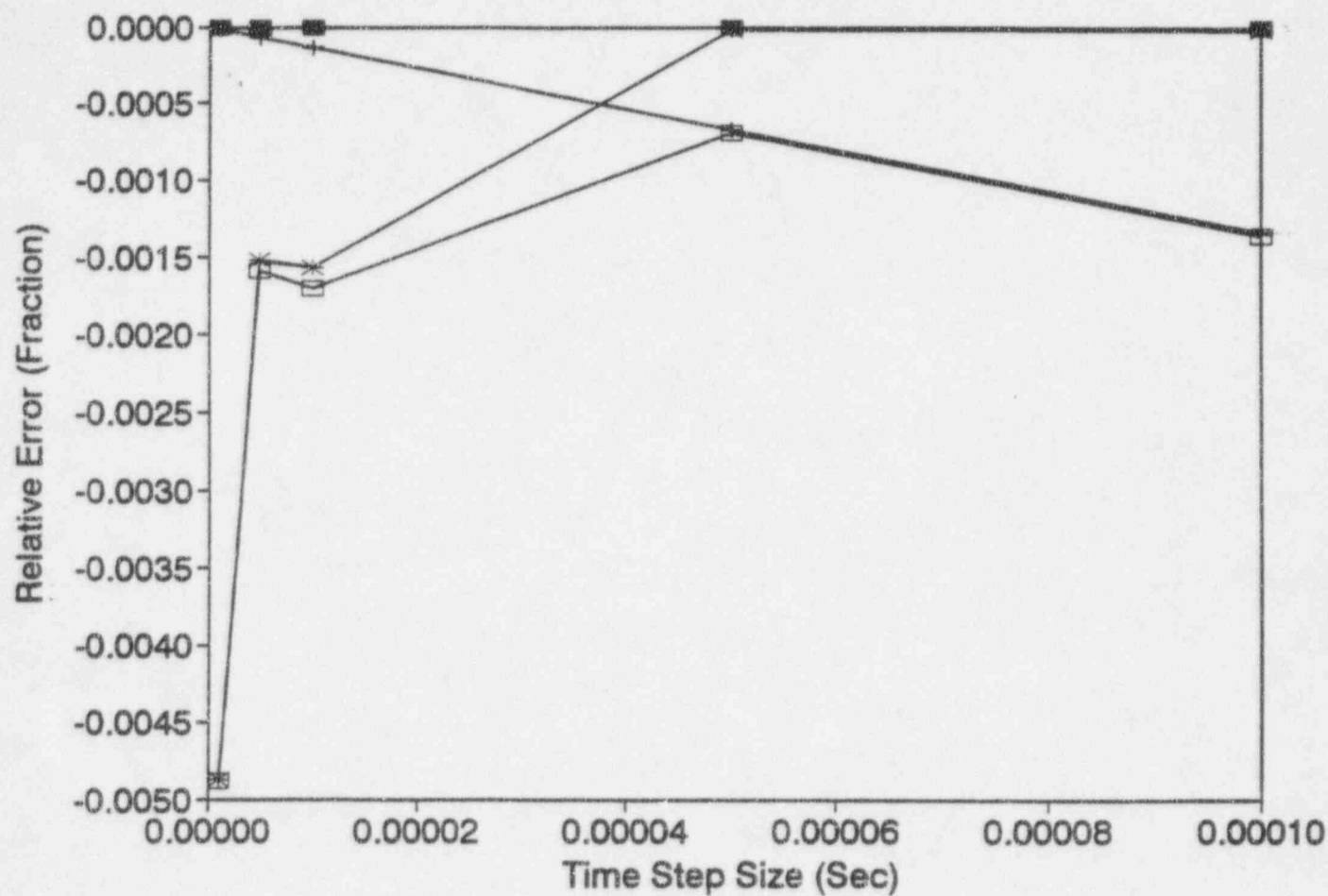
$\Delta t_{\text{kinetics}}$	<u>Double Precision</u>		<u>Single Precision</u>	
	<u>New</u>	<u>Old</u>	<u>New</u>	<u>Old</u>
1.00E-06	0.0%	-0.0013%	-0.4869%	-0.4882%
5.00E-06	0.0%	-0.0068%	-0.1518%	-0.1586%
1.00E-05	0.0%	-0.0134%	-0.1568%	-0.1699%
5.00E-05	0.0%	-0.0669%	-0.0021%	-0.0690%
1.00E-04	0.0%	-0.1337%	-0.0021%	-0.1358%
5.00E-04	-0.0008%	-0.6660%	-0.0026%	-0.6681%
1.00E-03	-0.0029%	-1.3260%	-0.0034%	-1.3263%
5.00E-03	-0.0721%	-6.3954%	-0.0726%	-6.3957%
1.00E-02	-0.2895%	-12.2428%	-0.2897%	-12.2431%
2.00E-02	-1.1611%	-22.5208%	-1.1613%	-22.5208%
4.00E-02	-4.5961%	-38.6339%	-4.5964%	-38.6339%
5.00E-02	-7.0895%	-44.9973%	-7.0895%	-44.9973%

FIGURE 4.1
RELATIVE ERROR VS $\Delta T_{\text{kinetics}}$
 $\Delta T_{\text{kinetics}} \geq .0001$



—■— New Solution —+— Old Solution

FIGURE 4.2
RELATIVE ERROR VS $\Delta T_{\text{kinetics}}$
 $\Delta T_{\text{kinetics}} \leq .0001$



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ABB

ABB Corporation
Corporate Headquarters
500 Franklin Road
P.O. Box 100
Baldwin, New York 11513
Telephone: (516) 634-4111
Fax: (516) 634-4111

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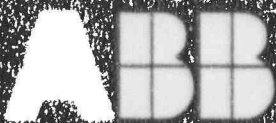


ABB Combustion Engineering Nuclear Fuel
Combustion Engineering, Inc.
1000 Prospect Hill Road
Post Office Box 500
Windsor, Connecticut 06095-0500
Telephone: (203) 285-8679
Fax: (203) 285-4117