Enclosure

SAFETY EVALUATION REPORT

"CESEC DIGITAL SIMULATION OF A COMBUSTION ENGINEERING NUCLEAR STEAM SUPPLY SYSTEM" (TAC No.:01142)

I BACKGROUND

Combustion Engineering submitted CENPD-107 on June 30, 1976. CENPD-107 documents the analytical equations and assumptions used in the CESEC computer program. CESEC is a simplified thermal-hydraulic transient computer program developed for analyzing FSAR Chapter 15 transient and accident events. Following the submittal in 1976, the CESEC computer program underwent extensive modifications which were required to model ATWS, steam line break, and operating reactor events. These modifications were documented in 6 supplements to CENPD-107. As the code was upgraded to calculate increasingly complex hydraulic behaviors, the code identifier was accordingly changed (i.e., CESEC, CESEC-I, CESEC-ATWS, CESEC-II and CESEC-III). The latest version of CESEC is designated CESEC-III and was submitted on the System 80, St. Lucie-2, Waterford-3, and Arkansas-2 dockets, as referenced below.*

This SER addresses the CESEC-III computer program. As part of this review, the acceptability of previous analyses performed on older versions of the code was assessed. The staff's review has concluded that licensing calculations of non-severe transients such as a decrease in feedwater temperature, increase in feedwater flow, turbine trip, loss of normal AC power, reactor pump shaft seizure, CEA withdrawal and rod ejection accidents would not be altered if reanalyzed with CESEC-III. The events which could show significant differences in results are severe depressurization events and events which lead to voiding in the reactor vessel upper head (e.g., steam line break, steam generator tube rupture, and letdown line break events). For plants licensed with older versions of CESEC, the more severe transient and accident events have been reanalyzed on CESEC-III. The results of these analyses were in compliance with present regulatory requirements.

This review did not address the use of CESEC-III for analyzing anticipated transient without SCRAM (ATWS) events. To model ATWS pressurization events, CESEC-III would require a more detailed steam generator model, similar to that developed in CESEC-ATWS, an older but specific version of the code.

The following is the staff's evaluation of CESEC-III.

*References: 1- Letter from A.E. Scherer (CE, CESSAR Docket) to D.G. Eisenhut (NRC) "CESEC," January 6, 1982 (LD-82-001)

2- St. Lucie 2 FSAR, Section 15D.11, Amendment No. 3.

3- Letter from L.V. Maurin (LP&L, Waterford-3 Docket) to G. Knighton (NRC, "CESEC," February 2, 1983 (W3P83-0247)

4- Letter from J.R. Marshall (AP&L, Arkansas-2 Docket) to J. Miller (NRC), "CESEC," December 27, 1983 (2CAN128310).

II SUMMARY OF CESEC-III AND THE TOPICAL REPORT

CESEC-III, a thermal-hydraulic computer program, was developed for analysis of FSAR Chapter 15 transient and accident events, as identified in NUREG-0800, the "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition" (SRP). The code is a fixed node program designed to calculate the thermal-hydraulics of a two-by-four loop nuclear steam supply system (NSSS) designed by Combustion Engineering (CE). The code solves the thermal-hydraulic mass, momentum (limited) and energy equations using homogeneous-equilibrium assumptions. Point kinetics are used for neutronic calculations. The primary system components modeled include the reactor vessel, reactor core, primary coolant legs, pressurizer, steam generators and reactor coolant pumps (as shown in Figure 1). The secondary components modeled are shown in Figure 2. These include the shell of the steam generators, the feedwater system, the main steam system and control valves. The code also calculates some of the control and plant protection systems.

The topical report documents the equations and assumptions comprising CESEC-III. The major differences between CESEC-I, CESEC-II, and CESEC-III are also summarized. Validation of CESEC-III with operating reactor transients and approved thermalhydraulic computer codes, such as CEFLASH and COAST, are also presented. Finally, the report describes the input, output and plot package for program execution and analysis.

The following is the staff's evaluation of CESEC-III, its analytical models, code validation, analytical audits, audit of the quality assurance procedures applied in the development and use of CESEC-III, and the staff's conclusions.

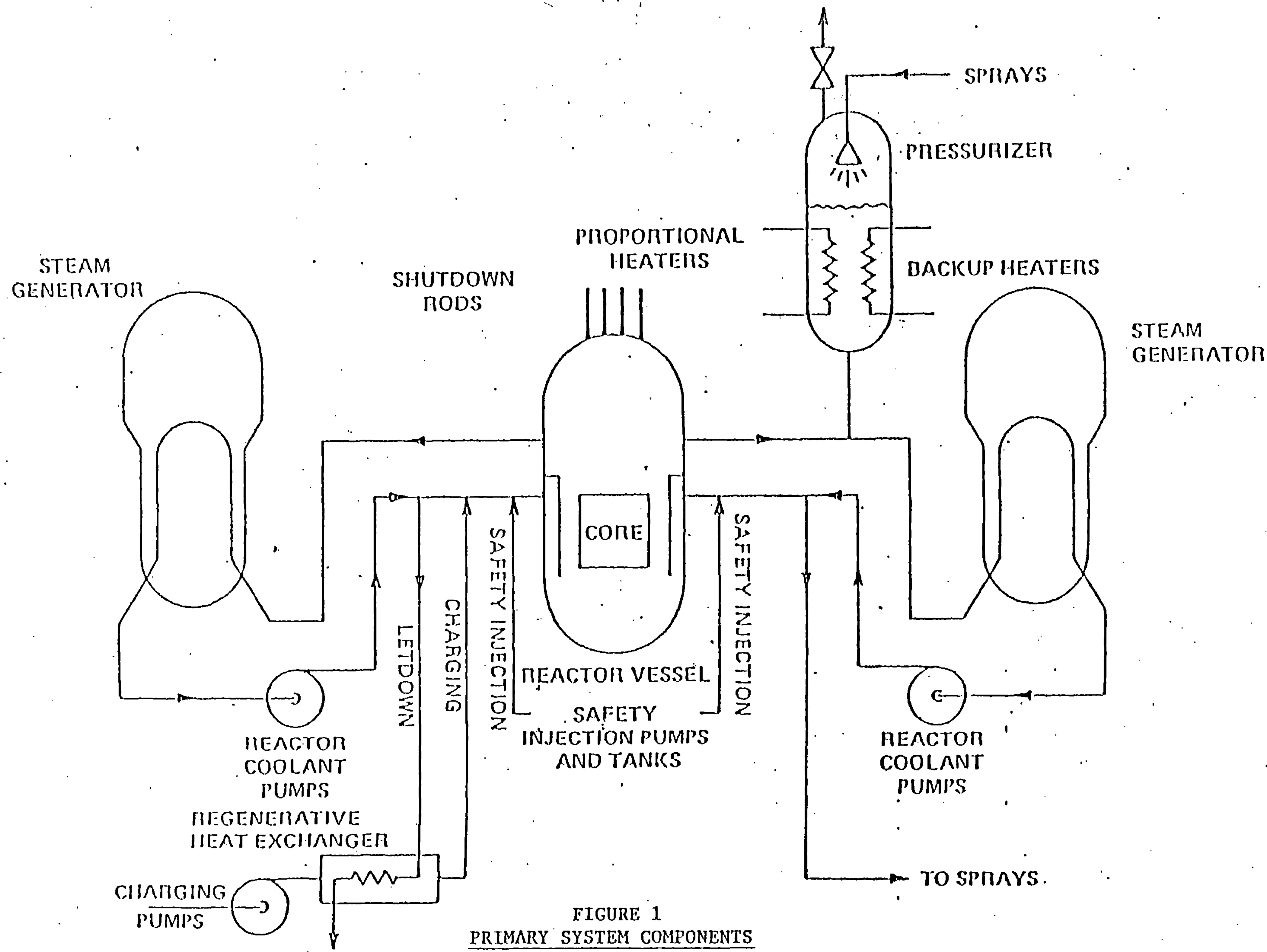

FIGURE 1
PRIMARY SYSTEM COMPONENTS

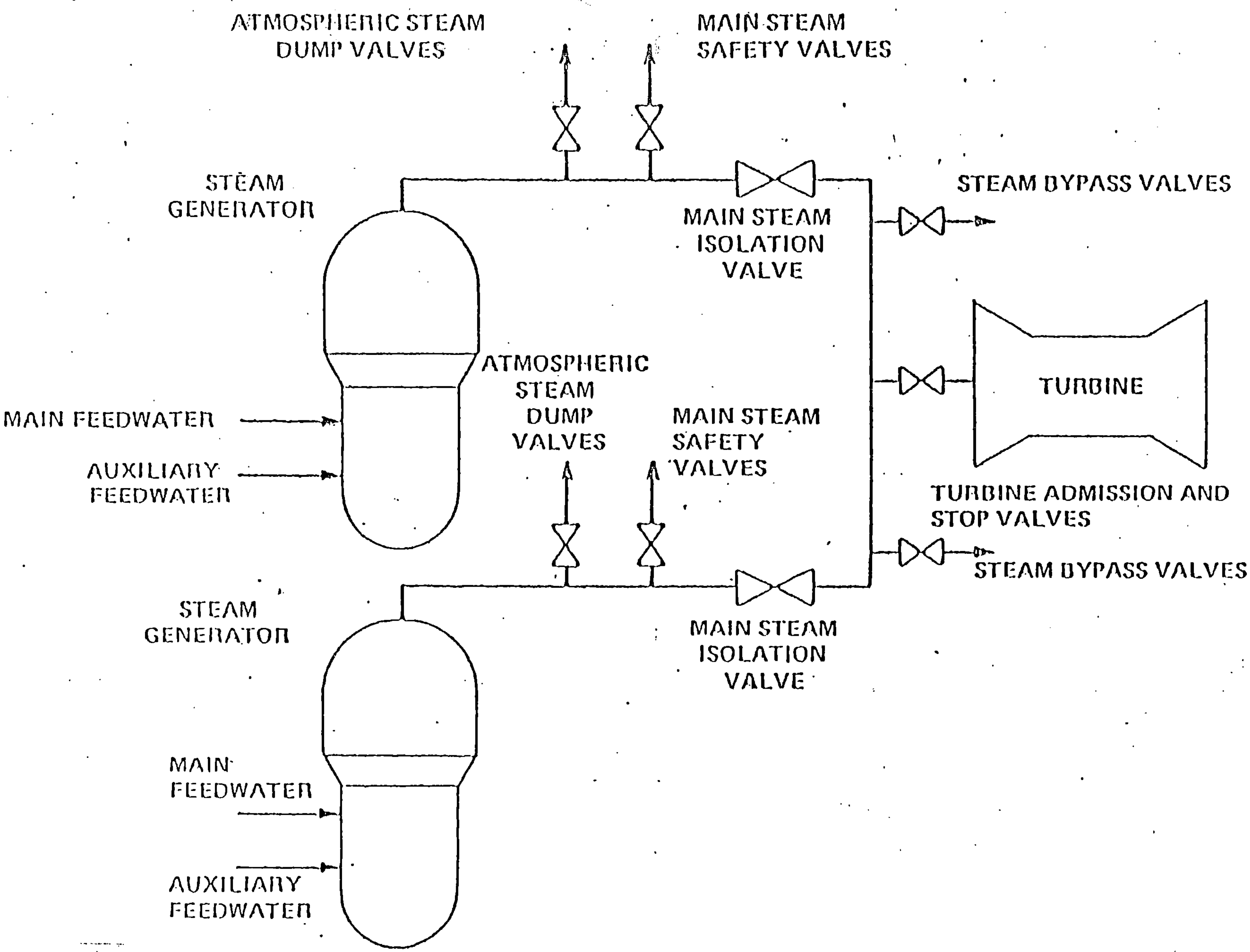

FIGURE 2

SECONDARY SYSTEM COMPONENTS

III CESEC-III APPLICATION

This report assesses the acceptability of CESEC-III to perform FSAR Chapter 15 licensing analyses for the following events:

1. Decrease in feedwater temperature (SRP Section 15.1.1)

2. Increase in feedwater flow (SRP Section 15.1.2)

3. Increase in main steam flow (SRP Section 15.1.3)

4. Inadvertent opening of a steam generator atmospheric dump valve (SRP Section 15.1.4)

5. Steam system piping failures (SRP Section 15.1.5)

6. Loss of external load (SRP Section 15.2.2)

7. Turbine Trip (SRP Section 15.2.3)

8. Loss of Condenser Vacuum (SRP Section 15.2.5)

9. Loss of Normal AC Power (SRP Section 15.2.6)

10. Loss of Normal Feedwater Flow (SRP Section 15.2.7)

11. Feedwater System Pipe Breaks (SRP Section 15.2.8)

12. Total and Partial Loss of Forced Reactor Coolant Flow (SRP Section 15.3.1)

13. Reactor Pump Shaft Seizure (SRP Section 15.3.3)

14. Uncontrolled CEA Withdrawal (SRP Section 15.4.1 and 15.4.2)

15. CEA Misoperation (SRP Section 15.4.3)

16. Boron Dilution (SRP Section 15.4.6)

17. Spectrum of Rod Ejection Accidents (SRP Section 15.4.8)

18. Volume Control System Malfunction (SRP Section 15.5.2)

19. Break of a Letdown Line (SRP Section 15.6.2)

20. Steam Generator Tube Rupture (SRP Section 15.6.3)

The acceptability of CESEC-III to model anticipated transients without Scram (ATWS) events was not reviewed. CE has an approved version of CESEC (SESEC-ATWS) for calculating the pressurization response to ATWS events. For CESEC-III to calculate ATWS events, the steam generator model would require upgrading. Consequently, the code, as documented in the January 6, 1982 transmittal (LD-82-001) from A. E. Scherer to D. G. Eisenhut, is not approved for ATWS analysis.

The staff concludes that CESEC-III is acceptable to perform licensing calculations of the above events. This approval is conditional upon an appropriate methodology of implementation which provides conservative results. This includes appropriate input and in specific, steam and feedwater line break methodologies. These methodologies are under staff review and documented in Appendices 15B and 15C of the System 80 CESSAR. The feedwater line break methodology has been approved and an SER for the steam line break methodology is nearing completion. CESEC-III is not a best-estimate computer program for all events. However, with appropriate input and boundary conditions, the CESEC-III computer program will generate conservative results.

The following describes the analytical models in CESEC-III.

IV REVIEW OF THE CESEC-III ANALYTICAL MODELS

The fixed nodalization developed in CESEC-III is shown in Fig. 3. The reactor vessel is symmetrically split with nodes (volumes) representing the downcomer, inlet plenum, core, outlet plenum, and vessel head. The vessel head is common to both loops. The purpose of the symmetric division is to model asymmetric thermal-hydraulics in the reactor vessel. This is achieved by junctions which transfer a specified fraction of flow between the parallel vessel components. The specified fractions, known as mixing factors, are input to the code. This review does not address the selection of mixing factors. That is being reviewed as part of the methodology of implementation (i.e., steam line break methodology).

Each hot and cold leg of the primary piping is modeled as a single node. Each steam generator is modeled as two nodes representing the tube section and two nodes for the inlet and outlet plena. The pressurizer is modeled as one node composed of two regions to simulate non-equilibrium conditions. The surge line, which connects the pressurizer to the hot leg of the reactor coolant system is not modeled as a separate node. Its response is calculated as a separate routine in the pressurizer model. The shell of each steam generator is modeled as a single node. A flow path has been incorporated between the primary and secondary sides of the steam generator to simulate steam generator tube rupture events. The code can model the stored energy in the reactor coolant system walls, approximate the heat transfer at the steam generator tubes, and model the energy generated in the core. The following is a brief description of these models.

1. Field Equations

(a) Neutronics Model

CESEC-III calculates core neutronics using the point kinetics model. This model permits up to six groups of delayed neutron precursors. Gamma heating of the moderator is modeled as a constant fraction of the total power. The reactivity feedback from changes in the fuel temperature (Doppler) is calculated using a single-pin/single-axial-node model. The Doppler reactivity is calculated for each radial node within the fuel pin and then volume averaged. The code computes the moderator feedback using either the core average moderator temperature or the core average moderator density.

For steam line break events, the code calculates the moderator reactivity feedback using the cold edge temperature. As defined by CE, the cold edge temperature is the coolant temperature at the core center plane along the core radial edge adjacent to the broken steam generator cold legs. This temperature subtracts out the inlet plenum mixing effects. The calculation of the cold edge temperature is based on several approximations. It assumes steady state symmetrically split core flow, mirror asymmetric planar enthalpy distribution, neglects core bypass flow and does not

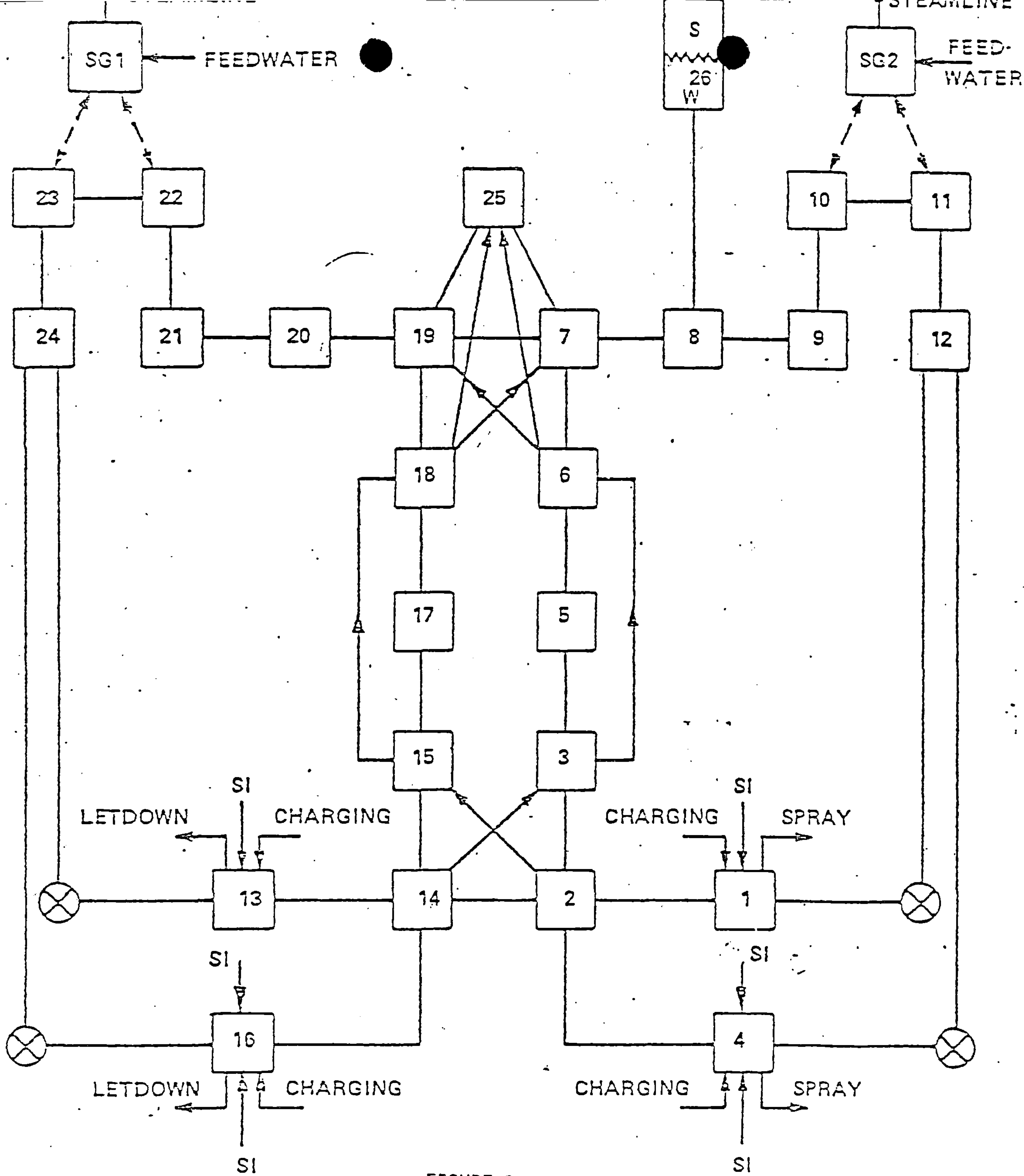

FIGURE 3

<u>CESEC Nodal Scheme</u>

subtract density driven cross flow (separate from the asymmetric inlet plenum mixing flows) in the downcomer. As an option, an additional refinement can be made to account for 3 dimensional (3-D) neutronic effects. The 3-D reactivity feedback is determined by coupling other neutronic and thermal-hydraulic codes to correlate the core radial temperature tilt, core flow and the core average power to flow ratio. The temperature tilt is defined to be the difference between the hot edge and cold edge temperatures at the core inlet. Since this refinement is an example of a specific code application, this procedure was not examined as part of the generic code review.

Reactivity contributions from control rod movement (input as a function of time and signal delay) and boron concentration (as calculated by the boron transport model) are modeled in the code. The code does not model subcooled voids. Reactor trip can occur from several input signals, which include high power, high or low pressurizer pressure, low coolant flow, low steam generator pressure, low steam generator level and manual initiation.

The kinetics equation is coupled to the 1971 ANS Standard for decay heat. Up to 11 fission product groups can be specified. The 11 group decay constants used are from the ANS Standards for infinite reactor operation.

The staff finds the applicant's model for neutronics and decay heat acceptable. The modeling of cold-edge temperature and reactivity control is also acceptable.

(b) Thermal-Hydraulics

The thermal-hydraulics for the reactor coolant system (RCS), excluding the pressurizer and its associated surge line, are calculated using homogeneous equilibrium model (HEM) assumptions. CESEC-III calculates a single RCS pressure, from which the thermodynamic states and fluid properties are derived. This assumption decouples the momentum equations from the mass and energy equations. The code does not solve individual junction momentum equations per se. The RCS flows are driven by the calculated pump and gravity heads using separate loop "mechanical energy" equations (there are no vector effects). The equation for mechanical energy balance assumes the flow throughout the loop changes uniformly and is composed of a pump model, gravity terms, geometric and D'Arcy friction pressure losses. The code also calculates two-phase flow resistance. The energy input from the reactor coolant pumps, when modeled, are input as added core heat. Otherwise, they are neglected. Flow from the upper plenum (upstream half) to the reactor vessel head is a user specified input fraction of the upper plenum flow. The flow from the reactor vessel head to the upper plenum is symmetrically distributed between the two plena (split vessel model). The plena mixing flows, which simulate asymmetric vessel thermal-hydraulics, are a fixed user input fraction of flow diverted between the two halves of the vessel. The core bypass flows are also modeled as fixed fractions of the split core flow. The code imposes a fixed pressure drop across the divided core.

This review finds the simplified model of the system hydraulics acceptable. This conclusion is based upon code verification submitted by CE and audit analyses, to be described later. The applicability of the code is limited to transients which do not result in two-phase fluid conditions in the cold legs of the reactor coolant system. For events leading to two-phase conditions in the cold legs, the applicant will need to submit further justification of the acceptability of the results.

(c) Thermal Conduction Model

CESEC-III has the capability to model heat capacities in the primary system walls, the steam generator tubes, and the reactor vessel upper head internals. Energy conduction from the component metal walls is solved using the Fourier equation. The wall energy distribution is calculated using 13 radial nodes which include two nodes in the cladding. Constant nodal properties are used and the distribution is solved explicitly. The thermal capacity of the control element assembly (CEA) shroud is modeled by assuming instantaneous equilibrium with the fluid.

Heat conduction in the steam generator tubes is modeled as a heat slab which assumes rapid equilibrium (tube time constant ~ 1 second) with the primary fluid. Conduction through the tubes is modelled quasistatically with the tube resistance determined by the system initialization procedure.

The core heat generation model is a constant dimension single pin with a single axial node and a single coolant node. The pin is modeled as three equivolume radial nodes with the cladding and gap homogenized into the third node. The homogenization procedure for the third node is essentially quasistatic. To account for the energy distribution in this node, the clad temperature is back calculated from the heat flux to the coolant. The back calculated temperature is then used in the calculation of the energy balance for the third node. Equivalent conductivities are used in the lumped parameter heat conduction calculation with a fixed fraction of the neutronic power being deposited in each pin node as well as in the single coolant node. To couple this model with the thermal-hydraulic model of a split core, an average core enthalpy is used to calculate the temperature for the single coolant node. The heat transferred to the coolant is then symmetrically split between the two core halves.

The staff has concluded that the simplified assumptions used in CESEC-III can be acceptably applied in licensing and scoping safety analyses. Our conclusions are based upon the technical derivation of the governing equations and independent audit analyses, to be described later.

2. Material Properties

The thermodynamic properties used in CESEC-III are based on the McClintock/Silvestri formulation. The saturation properties, temperature, enthalpy, and specific volume agree with the ASME steam tables to within tenths of a percent for a wide range of conditions. This is acceptable.

For subcooled and superheated regions, the agreement in specific heat is within two of percent. The thermodynamic derivatives are similar to those applied in CEFLASH-4AS for pressures above 550 psia. For pressures below 550 psia, the thermodynamic properties are obtained from equations developed by the Brookhaven National Laboratory as incorporated into the THOR computer program (BNL-NUREG-50534, July, 1976). This is acceptable.

Transport properties were obtained from the McClintock/Silvestri formulation and the 1967 ASME steam tables. The staff's examination shows that in general the values compare well with the ASME steam tables, except for the temperature range of 80°-200°F where it appears that the viscosity is off by ~ 20%. This inaccuracy will not affect the analyses of the transients for which the code is applied. There may be a mismatch in formulation at 600°F and 1200 psi where both the thermal conductivity and viscosity errors increase. As this is quite isolated, it is not expected to influence the results of the analysis.

3. Heat Transfer Correlations

CESEC-III models the primary to secondary heat transfer using the Dittus-Boelter forced convection model for subcooled primary system conditions and the Akets, Deans and Crosser Correlation for two-phase flow with condensation. The secondary system heat transfer is modeled using the C-E modified Rohsenow correlation for pool boiling. During reverse (secondary to primary) heat transfer regimes, CESEC-III applies the Dittus-Boelter correlation on the primary side during subcooled forced convection conditions and Hoeld's modification to the Chen correlation during two-phase conditions. The secondary system heat transfer is modeled using the McAdams single-phase free convection correlation. The criterion for switchover between the various models is based on bulk saturation temperature and the direction of heat transfer. In addition, the code can model the heat transfer resistance as a function of specified variables, such as secondary system liquid inventory. This option is applied to feedwater line break events.

The Dittus-Boelter correlation, as applied by CE, does not differentiate between modes of heating versus cooling. This limitation is considered second order. Similarly, CE's use of the McAdams correlation neglects the low Grashof number region. This also has a second order contribution to the total heat transfer. The application of both the Dittus-Boelter and the McAdams correlations are standard and acceptable. The McAdams correlation has previously been approved for CEFLASH-4AS and is acceptable for CESEC-III.

CE extended their application of the Akets correlation to a Reynolds number region where Akets recommended the use of the Eckert correlation. This results in a calculated heat transfer coefficient which underpredicts data. CE's modification to the Rohsenow pool boiling correlation has been found to be within 10 percent of the mean of appropriate data in the pressure range of 300-900 psia and a heat flux range of 3×10^4 thru 5.2×10^4 Btu/hr-ft^2. In the pressure range of 800-1200 psia and heat flux range of 2.7×10^4 thru 8.6×10^4 Btu/hr-ft^2, the deviation from the mean is on the order of 30%, within the range of data uncertainty.

The staff has uncovered a sign error in CE's modeling of the Chen correlation, as modified by Hoeld. This error affects the boiling induced microconvection component of the heat transfer coefficient for events which lead to primary system two-phase conditions and reverse heat transfer. CE has corrected this error and has demonstrated on the CESSAR docket (steam line break events) that this error does not alter the conclusions of the analyses.

Only single phase liquid forced convection is modeled in the core. An approximation to the Dittus-Boelter is used. CE contends that this approximation is good to a few percent in the pressure range of one atmosphere to 3200 psia and in the temperature range from 200°F to a temperature just below saturation. CE's justification appears to be valid except in the temperature range of 700°F where the error jumps to several tens of percent. The temperature range is outside the expected range for operational transient analysis. The inaccuracies of the model are compensated by the conservative application of the code.

The staff concludes that the simplified assumptions applied in CESEC-III for modeling heat transfer is not a best-estimate of expected physics but an approximation to it. For mild transients, its application is acceptable. For severe transients, such as steam line breaks and feedwater line breaks, the code could have limitations in both the conservative and non-conservative directions. However, the methodology of application of the code has been shown to result in an acceptably conservative solution. This is verified by the staff's audit, described in a later section.

4. Friction Correlation

CESEC-III applies the D'Arcy/Moody model for calculating single-phase fluid frictional flow resistance. During two-phase conditions, CE applies a combination of two-phase friction multiplier correlations, which include the Thom correlation for pressures above 250 psia and the Martinelli-Nelson correlation for pressures below 250 psia. These models have been previously approved for application in CEFLASH-4AS and are appropriate for application in CESEC-III. The two-phase form losses are computed using homogeneous equilibrium assumptions. This is acceptable.

5. Critical Flow

The CESEC-III options for critical fluid flow of steam are the Murdock-Bauman and the CRITCO correlations. Correlations available for modeling two-phase and subcooled fluid flow are: Henry-Fauske/ Moody combination, Henry-Fauske, Moody and the homogeneous equilibrium model (HEM). These models are applied as a function of the system stagnation enthalpy and pressure. The gravity heads are neglected since they are negligible relative to the stagnation pressure. The Murdock-Bauman and the Henry-Fauske/Moody options were previously approved for use in CEFLASH-4AS and are acceptable for use in CESEC-III. The HEM correlation is generally considered a best-estimate two-phase fluid flow model. This typically results in lower calculated flow rates than either the Henry-Fauske or the Moody models. The CRITCO correlation was developed for two-phase flow. CE has demonstrated that CRITCO is valid for single-phase steam flow, as it predicts within a couple of percent of the D'Arcy formulation for the pressure range between 100 and 1000 psia.

The user of CESEC-III can select any of the flow correlations listed above. In addition, the user can input a quality dependent discharge coefficient and a time dependent leak flow area. CESEC-III has an added option for simulating a leak, such as a 1 gpm tube leakage used in Chapter 15 licensing evaluations. This option is applied for dosage calculations and is not factored into the primary system mass and energy calculation.

The critical flow models used in CESEC-III are acceptable. Each model is recognized as being conservative or nonconservative, depending upon the analyzed event. The applicant must clearly identify within the submittal which model is applied and justify the selection.

6. Component Models

(a) Reactor Coolant Pumps

The pump model in CESEC-III is similar to the one used in RELAP4 and CEFLASH-4AS. The code applies the homologous curves and a two-phase degradation multiplier (the default uses the SEMISCALE data approved by NRC for use in CEFLASH-4AS). The equations for pump speed account for electrical torque (a function of speed), hydraulic torque and friction windage losses (proportional to the speed squared). Options are available for inputting pump speed as a function of time; for modelling the pump as a geometric loss (locked rotor); and for simulating a sheared shaft. The staff finds the pump model in CESEC-III acceptable.

(b) Pressurizer and Surgeline Models

The pressurizer model calculates non-equilibrium thermal-hydraulic conditions. The model separates the steam and liquid phases into two regions. Each region calculates a bulk mass and energy balance. The pressurizer sprays are assumed to instantaneously mix with the steam region. The pressurizer heaters act on the liquid region to generate steam. The only energy transfer between the steam and liquid interface occurs by mass transfer of saturated liquid condensing from the steam dome to the liquid region or transfer of saturated steam bubbles from the liquid region to the steam region. The pressurizer also models heat transfer from the pressurizer walls and energy release through the safety/relief valves.

CESEC-III calculates one net mass and energy balance for the entire NSSS, while constraining to a constant system volume. The mass from the pressurizer surge line is incorporated into the pressurizer node. The equations for frictional pressure drop, geometric losses, elevation heads, and inertial (L/A) terms are all incorporated in the equations of mass and energy conservation. The mass and energy conservation equations are solved simultaneously.

The thermodynamic states are calculated by predicting the trend of the event and not through iterative steam table lookups. The trend of the event is evaluated by the governing differential equations using partial derivatives of the fluid state. From the predicted state, CESEC-III calculates the error associated with the updated calculation. For small errors, the code continues to the next time step. For the great majority of the time-step calculations, this process results in very small errors. For rapid changes in system conditions, the magnitude of the calculated error may be beyond the acceptance limit (typically less than 1% for licensing calculations). For large errors, the code iterates to a solution until the calculated error is within the acceptance criteria and then proceeds with smaller time steps.

CESEC-III calculates two thermo-dynamic properties for the system. One for the pressurizer and the other for the primary system. The primary system thermodynamic properties (outside the pressurizer) are computed from the pressurizer pressure plus the differential developed across the surge line. The thermodynamic properties in the pressurizer are determined from the pressurizer pressure.

For conditions where the pressurizer is empty and voids develop in the reactor vessel upper head, the primary system pressure is calculated as the hottest coolant temperature in the primary system. This is typically the saturation temperature in the upper head. The user can maximize the coolant temperature in the upper head by modeling zero mixing between the cooler primary system water and the upper head inventory. CESEC-III does not apply the pressurizer non-equilibrium model in the upper head. The limitation of not modeling nonequilibrium conditions in the upper head is addressed below in Section (h).

The staff finds the CESEC-III non-equilibrium pressurizer model acceptable. This approval of the thermal-hydraulic model for system pressure and thermodynamic states is based on technically acceptable analytical equations and audit calculations, which are addressed later in this report.

(c) Safety Injection System

CESEC-III models the safety injection systems and the reactivity feedback associated with their boron concentration. The safety injection pumps are modeled as fill tables of flow versus primary system pressure. This model is typical of most thermal hydraulic codes. The safety injection tanks (accumulators) assume an adiabatic expansion of the fill gas and accounts for pressure differentials due to geometric losses and elevation heads. When modeling boron reactivity, the user specifies the volume of pipe between the Boric Acid Tank (BAT) and the reactor coolant system boundary (i.e., cold legs). Applying the flow rate of the safety injection pumps, the code calculates a delay for boron injection insurge, which is the time required to purge the ECCS lines as the ECCS aligns to the BAT.

The staff finds the CESEC-III model for the safety injection systems acceptable.

(d) Charging/Letdown System

The letdown and charging flows are modeled as a function of the pressurizer water level. The charging and letdown fluid temperatures are set equal to the cold leg temperature. For the letdown line break event, the code assumes single phase break flow of saturated (liquid) enthalpy which corresponds to the enthalpy at the exit of the regenerative heat exchanger (RHX). This leads to a conservatively high break flow calculation. The empirical formula used by CE for the RHX heat transfer coefficient has been found to give letdown outlet temperatures which agree with data to within a few tenths of a percent.

The staff finds the model of the Charging/Letdown system acceptable.

(e) Steam Generator Secondary Model

The steam generator secondary side is approximated as a single node. The secondary is maintained at saturated conditions and the liquid and steam phases are assumed to be ideally separated. The method for calculating mass and energy balance is identical to the method used for the primary system. CESEC-III does not mechanistically model in detail the thermal-hydraulics of the secondary side. The steam generator heat transfer resistance is obtained from the steady-state conditions and set as a constant throughout the analyzed event. Heat transfer degradation during tube uncovery is modeled as a reduction of heat transfer area. The reduction of heat transfer area is modeled as a function of secondary liquid inventory. There is a user controlled variable which "triggers" the initiation of heat transfer degradation. Once a specified threshold inventory is obtained, the heat transfer area is set to zero. From plant data, a model for level indication was developed. The model for level indication consists of steady-state data which is a function of load, flow, and liquid inventory. In addition, a trip signal on level and inventory can be user specified.

CESEC-III has several means to model the steam generators; none of which can be classified as best-estimate for severe pressurization and depressurization events. Consequently, the applicant must clearly identify and justify the assumptions made in the analysis.

(f) Steam Lines

CESEC-III models two steam lines, one for each steam generator. Each line is assumed to be at constant enthalpy and is represented by a quasi-static momentum equation (single secondary pressure). The code can account for friction pressure losses in the lines for break flow calculations. Compressibility effects are not modeled. The flow through the turbine valve is user controlled. The flow is presumed choked using the CRITCO critical flow correlation. The pressure is solved by applying the Newton-Raphson iteration technique to the set of nonlinear equations.

(g) Feedwater System

The feedwater system is modeled as either a table of time-dependent flows or by an automatic flow controller which is driven as a function of the downcomer water level. The feedwater flow can also be set to match the steam flow. The feedwater enthalpy is input as either a function of time or power demand. The auxiliary feedwater enthalpy is input as a function of time while the flow can be specified as a function of time or controlled by the level controller.

(h) Reactor Vessel Upper Head

The reactor vessel upper head governs the primary system pressure after pressurizer emptying. Heat slabs representing the upper head stored energy can be modeled to minimize system depressurization. The upper head is modeled as a homogeneous-equilibrium node. No phase separation model is used. The user can specify only liquid exiting the head, thus simulating complete bubble separation. In addition, the user can adjust the degree of upper head/upper plenum fluid mixing to simulate associated bypass flow of lower temperature coolant with the upper head inventory.

For depressurization events, as occurs during a steam line break event, the coolant in the reactor vessel upper head becomes two-phase. In addition, the pressurizer is emptied of liquid inventory. It is the upper head saturation pressure which then governs the primary system pressure response during the remaining transient until the upper head becomes water solid. CE has determined that minimizing the depressurization rate during a steam line break event is conservative because it limits the boron injected by the safety injection system. To minimize the depressurization rate, the code user selects a code option which prevents mixing of colder coolant (e.g., core and downcomer bypass flows) with the hotter upper head coolant. This maximizes the upper head saturation temperature throughout the period of upper head inventory depletion. This depletion results from coolant shrinkage as the primary system undergoes a severe cooldown.

CESEC-III does not use a nonequilibrium model in the upper head as applied in the pressurizer (see Section 6b). Consequently, the recovery period may not be conservatively calculated when cold primary coolant enters the upper head and instantaneously mixes with the hot steam. For Chapter 15 licensing evaluations, this period of the transient typically follows the time at which the minimum DNBR (point of interest) occurs. The staff, therefore, finds this model acceptable. However, should future licensing evaluations require an accurate or conservative evaluation of the system pressure response during the period of upper head inventory recovery, further justification and code validation would be required.

Due to the sensitivity of primary system depressurization to upper head mixing, the staff requires justification of the modeling assumptions used for all analyses in which upper head voiding occurs.

7. Special Models

(a) Enthalpy Transport

Enthalpy transport is a numerical means to transmit energy from one node to another. Rather than transmitting average nodal properties (i.e., temperature and enthalpy) from a volume containing a heat source (i.e., core), the enthalpy transport model calculates the exit fluid conditions and transfers those conditions into the downstream volume. This model is applied in the core and steam generators. This model provides more realistic outlet conditions and thereby a more realistic transient response. The staff finds this model acceptable.

(b) Boron Transport

CESEC-III can model Boron addition and Boron dilution events. Boron addition is supplied by the safety injection systems. If selected, the charging/letdown contribution to these events can be neglected. The model can account for delays in startup of the SI pumps, diesel generators, and sweepout of the ECC lines as the system aligns with the Boric Acid Tank (BAT). The Boron concentration is solved using a transport model which assumes instantaneous intranodal mixing or can be input as a function of time.

The staff finds the boron transport model acceptable when the reactivity changes slowly relative to the transport time of boron from the SI system. Application of this model should assure that the rapid transport of boron would not significantly alter the results of the event and that the results of the calculation are conservative.

8. Initialization

The initialization procedure calculates an effective steam generator tube heat transfer resistance to balance core energy with energy removed by the steam generators. Inputs to the initialization procedure include cold leg temperature, core power, reactor coolant system pressure and flows, and feedwater enthalpy. The code iterates on the steam generator and header pressures until steady-state conditions are obtained. The staff finds this procedure acceptable.

9. Numerical Technique

CESEC-III evaluates the point kinetics equations using the Runge-Kutta/Merson method which is a modification of the standard fourth order Runge-Kutta method of solution. This projects an estimated error in the reactivity and power calculation to limit the number of iterations.

The mass and energy equations are written in difference form using node/flow path techniques and the concept of a donor cell. However, the solution technique was not developed to calculate reverse core flow. Reverse flow in the loops can be calculated. The code solves the differential equations by substituting the conservation of mass equations and equations of state into the energy equations. This eliminates most of the temporal derivatives in the equations. Linearization is then accomplished by using a fractional time step procedure and splitting the variables. An explicit forward scheme is used. The surge line momentum equation closes the set of equations. CE uses an iterative procedure to calculate the surge line pressure drop by using a predictor step to solve the nonlinear set of equations. The next calculation solves the pump flow explicitly.

The steam generator and feedwater system conditions are used to advance the source and coefficient matrices for the mass and energy equations. During the numerical solution, the code checks for violation of the Second Law of thermodynamics in the steam generator before the enthalpy transport correction is made in the thermal/hydraulic coefficient matrix. If the Second Law is violated, the outlet enthalpy is set equal to the nodal average enthalpy and the heat transfer is recomputed to satisfy the enthalpy transport model. If the direction of heat transfer differs from that originally calculated with the steam generator model, the heat transfer is set to zero, otherwise the updated values are used.

CESEC-III has an automatic thermal/hydraulic time step adjuster based on the loop cycle time. CE exercises caution when selecting the size of the time step. Improper selection of the time step size can lead to large errors in conserving mass and energy. For scoping studies, accuracy is compromised for running time. For licensing analyses, errors are limited to less than 1%. This has been found acceptable through the audit calculations described later.

V CODE QUALIFICATION

CE submitted five comparative analyses to qualify the acceptability of CESEC-III for FSAR Chapter 15 licensing calculations. These included pump model comparisons with COAST, small break LOCA comparison with CEFLASH-4AS, plant data comparisons with a hot-zero power-four pump coastdown test, turbine trip test and a natural circulation cooldown event. The following is a brief description of these qualification runs.

The CESEC-III reactor coolant pump model was validated by comparing analyses to a previously approved pump model (COAST) and operational startup data. Both comparisons showed excellent agreement. Since the application of the CESEC-III computer program is not intended for events leading to two-phase fluid conditions at the pump, the staff finds the pump model in CESEC-III adequately validated for its intended use.

To validate CESEC-III for calculating pressurization and depressurization events, CE, in cooperation with one of its customers, performed a turbine trip test at 98% of rated core power. As a consequence of an atmospheric dump valve failing to close, the data obtained from this test drifted out of instrumentation range required for detailed code input. Consequently, CE had to modify the data in a manner consistent with sound engineering practices. The need for data modification was verified by the Argonne National Laboratory, under contract by the NRC for this review. Applying the available data and interpretation thereof, CESEC-III was demonstrated to satisfactorily predict the major trends of the test. The staff finds this assessment acceptable.

To demonstrate CESEC-III's capability to predict steam formation in the reactor vessel upper head and during natural circulation cooling, the code was assessed with plant data of a natural circulation cooldown event. Excellent agreement was obtained when comparing core inlet and core outlet coolant temperatures. Voiding in the reactor vessel upper head was calculated and validated by the pressurizer level which increased at a rate four times greater than could be attributed to the addition of primary system inventory. As with the turbine trip test, the data obtained from the natural circulation cooldown event was missing accurate and detailed data required for code input. Consequently, engineering extrapolation was required.

The staff recognizes that it is not cost effective to instrument operating reactors such that they provide sufficient and very accurate data for code validation. However, code comparisons with plant data when applying sound engineering practices are very desirable and can uncover code deficiencies (e.g., the St. Lucie 1 natural circulation cooldown event led to the development of CESEC-III).

To further substantiate CESEC-III's capability to calculate depressurization events, CE compared the code to an analysis performed on CEFLASH-4AS. CEFLASH-4AS is an approved LOCA code which more rigorously calculates the thermal-hydraulics of the primary system. The calculation was of a postulated leak in the cold leg. This simulated a letdown line break event. The size of the break was twice the cross-sectional area of the letdown line.

The two codes predicted very similar results. Differences between the results were primarily attributed to differences in modeling of the secondary system (i.e., turbine admission valve), which CEFLASH-4AS has only limited capability. Excluding the feedback of the admission valve, the maximum deviation between the system pressures predicted by the two codes remained within 60 psia. The deviation in primary system voiding was consistent with the difference in break flow and corresponding pressure history. Overall, the calculated trends were similar for the two codes. Deviations between the code predictions were understood.

The staff finds the validation of CESEC-III adequate and acceptable for use in calculating the Chapter 15 events outlined in Section III. Our conclusions are based on the acceptability of comparisons with previously approved codes, plant operational data, and the audit calculations described next.

VI ANALYTICAL STAFF AUDITS

The staff performed audits of CESEC-III using the RELAP5 computer program (an advanced thermal-hydraulic computer program developed by the Office of Nuclear Regulatory Research at NRC). These audit calculations were of the System 80 steam and feedwater line break events. Details of these audits are documented in Appendices H and G of Revision 1 to the System 80 SER. Figures 1, 2, 3 and 4 illustrate the good agreement between the two codes and the conservative trend predicted by CESEC-III. The audit calculation attempted to model the events with similar boundary and initial conditions as applied by CE. The feedwater line break audit confirmed the acceptability of CESEC-III to model severe pressurization events and the steam line break audit confirmed the acceptability of CESEC-III to model severe depressurization events.

The conclusions of these audits confirmed the acceptability of CESEC-III to calculate the thermal-hydraulic responses of the reactor coolant system with the boundary conditions employed. To predict conservative consequences, the applicant applies conservative constraints on the boundary conditions. The staff's audits confirmed the code's acceptability when applying these constraints.

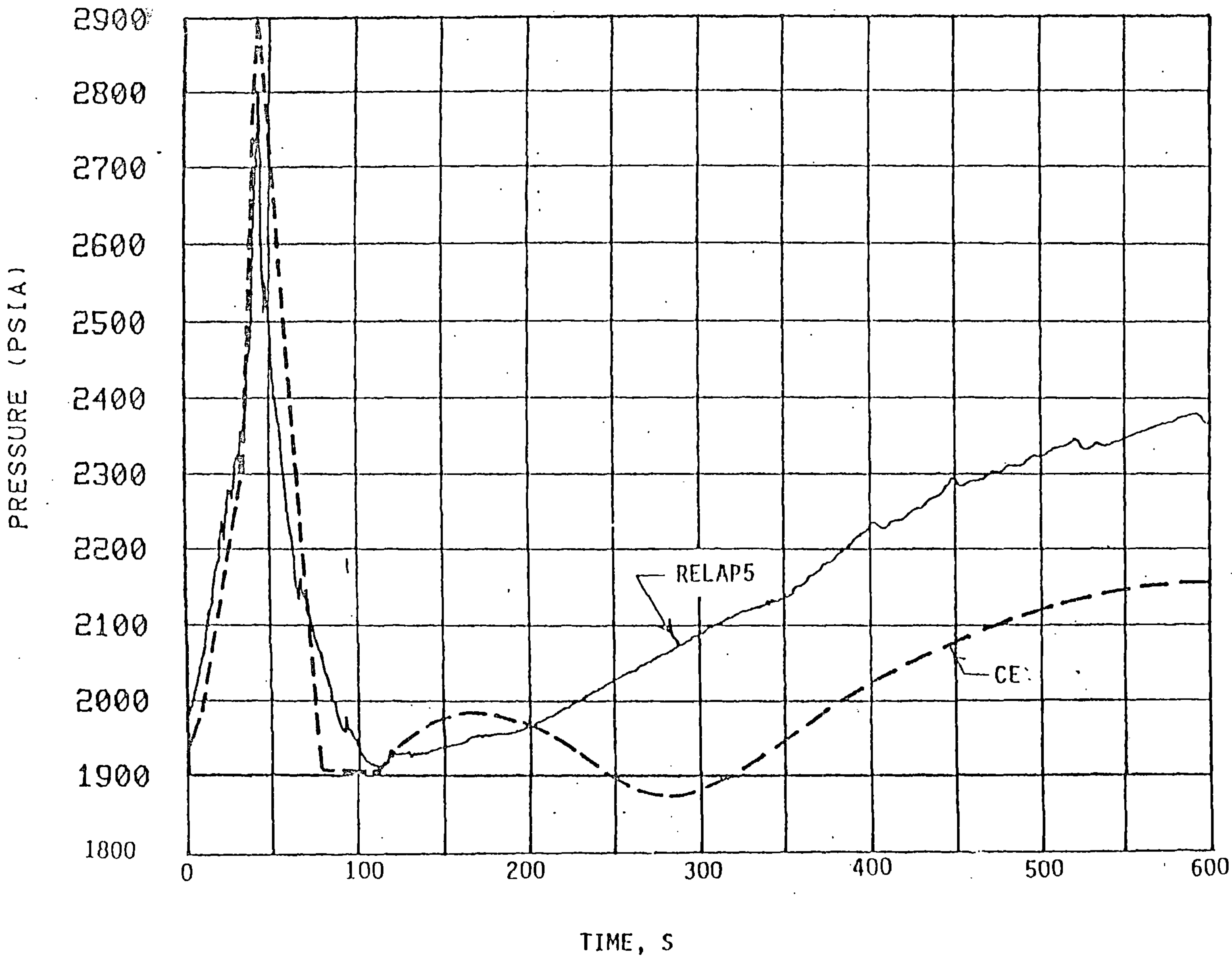

FIGURE 4

PRIMARY SYSTEM PEAK PRESSURE AS A FUNCTION OF TIME

FOR A POSTULATED FEEDWATER LINE BREAK

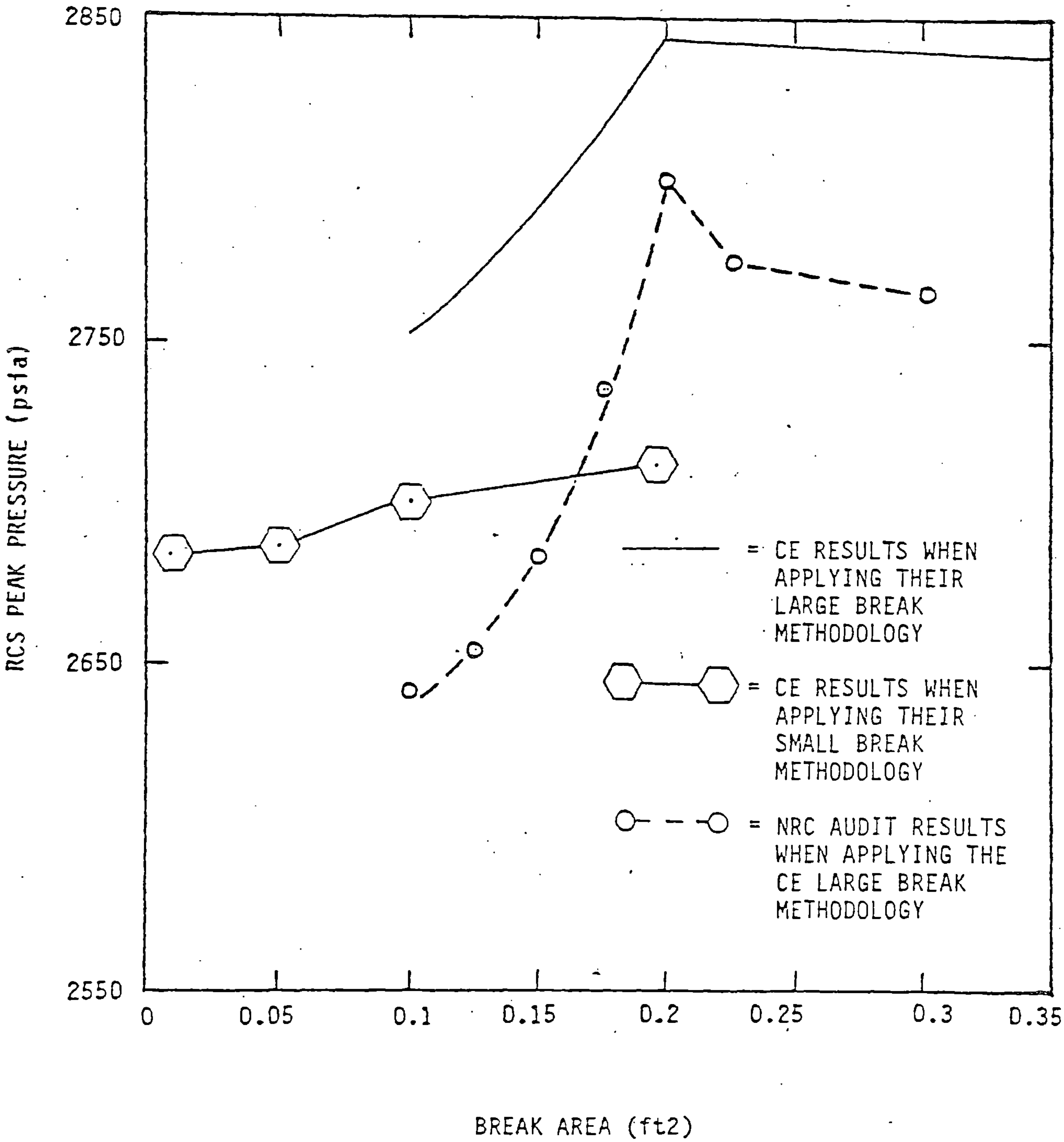

FIGURE 5

PEAK RCS PRESSURE AS A FUNCTION OF FEEDWATER LINE BREAK SIZE

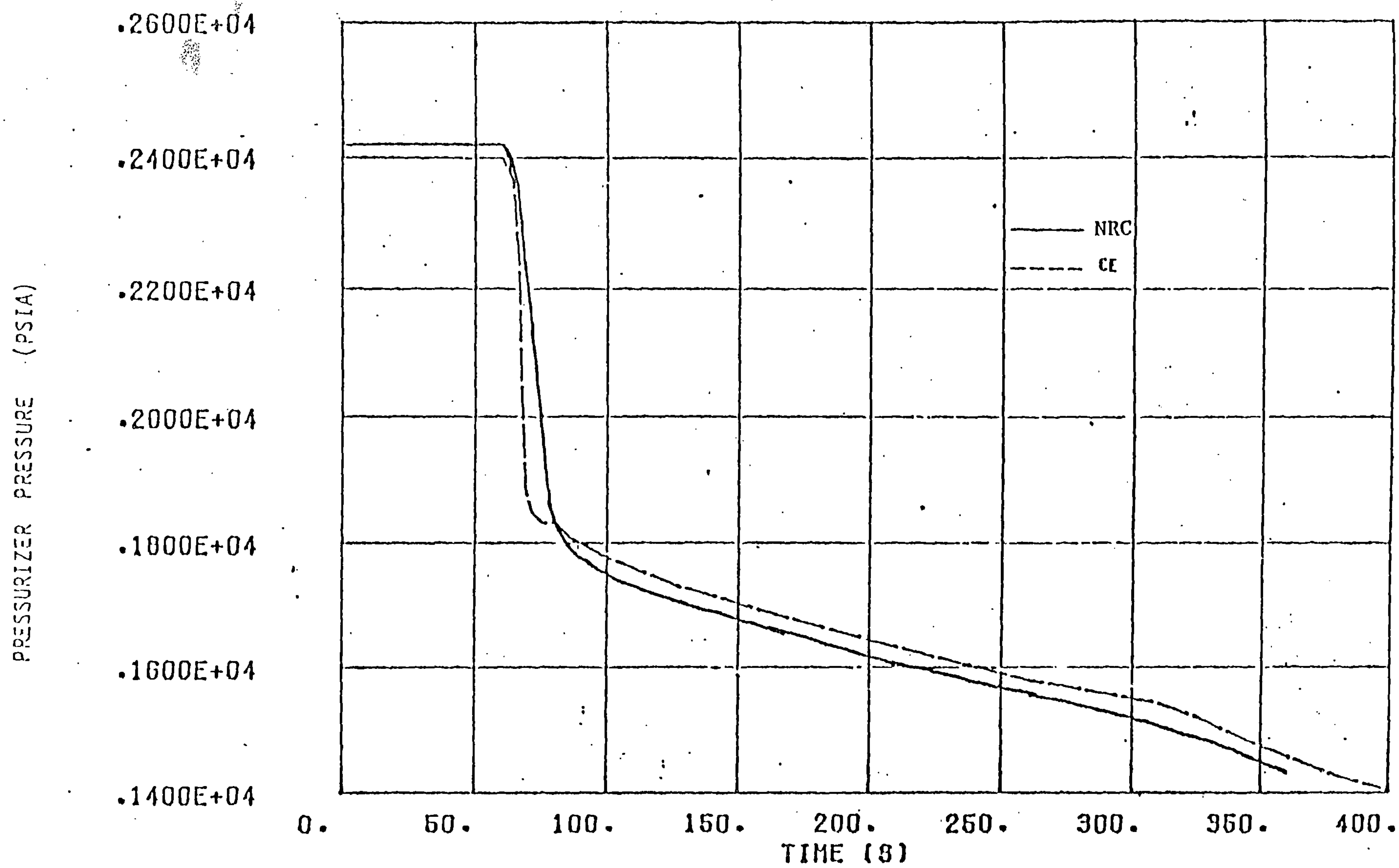

FIGURE 6

PRESSURIZER PRESSURE AS A FUNCTION OF TIME FOR A POSTULATED STEAM LINE BREAK

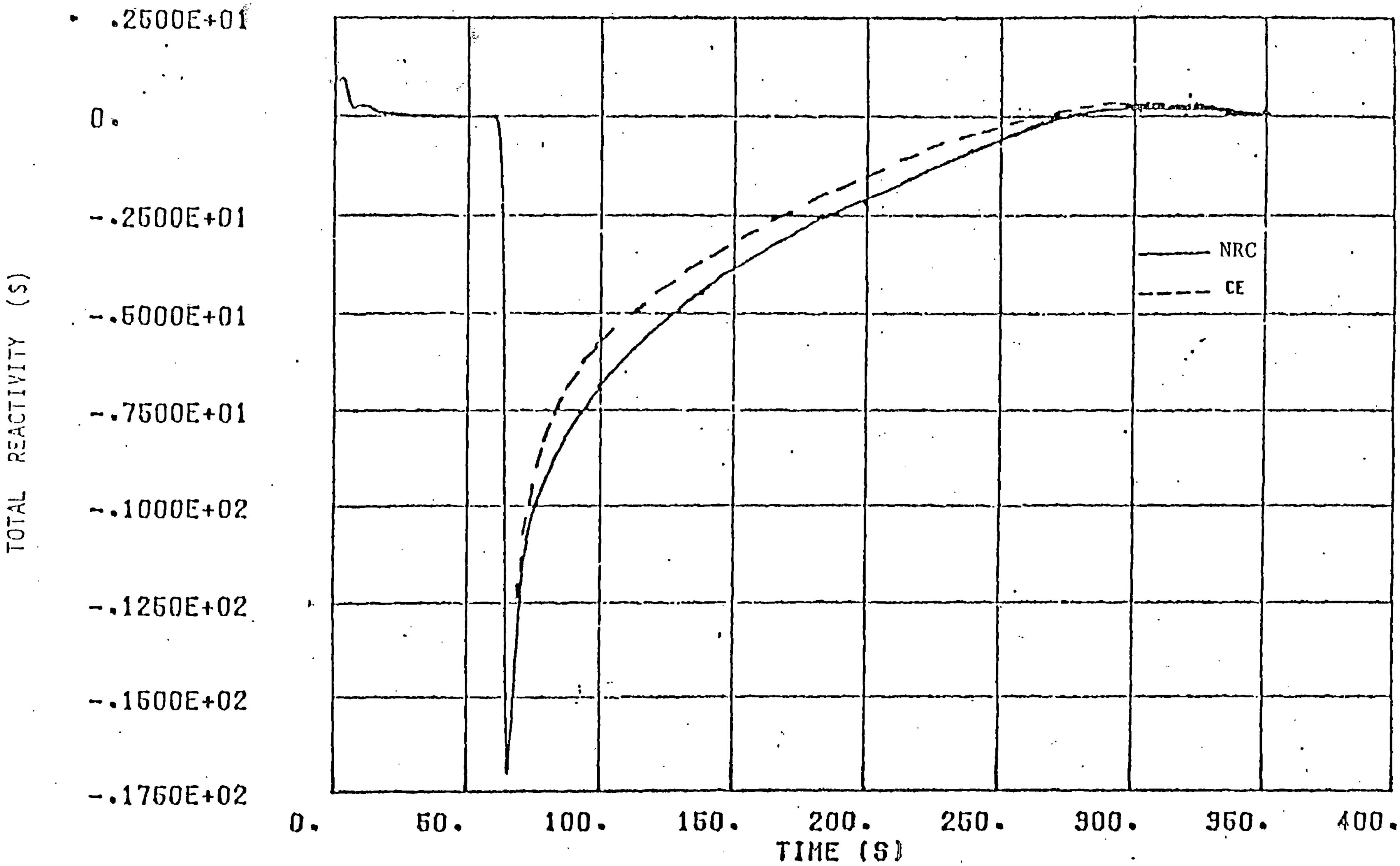

FIGURE 7

TOTAL REACTIVITY AS A FUNCTION OF TIME FOR A POSTULATED STEAM LINE BREAK

VII QUALITY ASSURANCE AUDIT

On April 28, 1981, the Region IV Office of NRC conducted an inspection of the Combustion Engineering quality assurance procedures used to develop CESEC. The audit examined the user's manuals, the theoretical manuals, and the verification analyses for several versions of the CESEC computer program. Conclusions of this inspection are as follows:

a. There were no nonconformance, unresolved, or follow-up items identified.

b. The computer program CESEC provides a digital simulation of the NSSS and is used in the safety analysis performed by CE. The program is designed to facilitate the analysis of abnormal or accident operational conditions via the study of postulated transients.

c. The CE Topical Report CENPD-107 and its supplements were found to provide a detailed description of the mathematical models, empirical data, assumptions, and applicable references.

d. The training manual (CESEC-II User Course) and the information contained in the topical report, and its supplements, were found to satisfy the "Users Manual" requirements.

e. The QA of Design Procedure 5.2 (Design Analysis), Revision 3, dated April 2, 1979, was found to impose the requirements of the objectives of the control of computer programs.

Further details of this inspection are documented in the inspection report number 99900401/81-01, dated May 13, 1981. Based on the findings of the Region IV inspection, the staff finds the quality assurance procedures applied in the development and use of CESEC acceptable.

VIII CONCLUSIONS

CESEC-III is the latest version of the CESEC series of thermal-hydraulic computer codes developed by Combustion Engineering for FSAR Chapter 15 analysis. The staff has reviewed the acceptability of the analyses performed by older versions of the code and has concluded that the majority of these analyses would not be significantly altered if reanalyzed on CESEC-III. Those events with significant deviations in results (when applying similar assumptions) have been generically reanalyzed on CESEC-III (with acceptable but different assumptions) and were found in conformance with present regulatory requirements. These events include the steam line break event, the letdown line break event and the steam generator tube rupture event. Most of these events have been reanalyzed and documented in CEN-199, "Effects of Vessel Head Voiding During Transients and Accidents in C-E NSSS's," March, 1982.

The review of CESEC-III was limited to FSAR Chapter 15 transients, as defined in NUREG-0800, "Standard Review Plan," and listed in Section III of this SER. CESEC-III was not assessed for analyzing anticipated transient without SCRAM (ATWS) events. For these events the applicant intends to use the CESEC-ATWS version of the code, which has been previously approved for licensing application.

The staff concludes that CESEC-III is an acceptable computer program for use in licensing applications for calculating FSAR Chapter 15 events. This approval is conditional upon an acceptable methodology of implementation. CESEC-III is composed of simplifying thermal-hydraulic assumptions which may not render best-estimate results for all analyzed events. An example is the modeling of the steam generator heat transfer response during a postulated feedwater line break. Imposing conservative boundary conditions on the model, as documented in Chapter 15B of the System 80 FSAR, will provide a conservative and acceptable licensing analysis. For mild transients, CESEC-III, with best estimate input assumptions, calculates expected transient responses.

CESEC-III is not approved for events which lead to two-phase coolant conditions in the cold legs of the reactor coolant system, nor for events leading to two-phase stratified flow in the coolant loops. Should the applicant intend to apply this code for such events, further justification is required. In addition, further justification is required if the code is applied to resolve licensing concerns for recovery events where upper head inventory is replenished.

Conditional upon the limitations stipulated in this report, the staff concludes that CESEC-III is an acceptable computer program for analyzing transient and accident events. This safety evaluation report may be referenced in future submittals as finding the CESEC-III computer program acceptable when implemented in accordance with the conclusions of this report. The staff requires that all licensing analyses (ATWS events excluded) performed following issuance of this SER be analyzed on CESEC-III.

IX. ACKNOWLEDGEMENT

This Safety Evaluation Report was prepared by Jack Guttman.