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November 2, 1980

Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

ATTENTION: Mr. R. A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit No. 1, Docket No. 50-317
Amendment to Operating License DPR-53
Fifth Cycle License Application
Responses to NRC Staff Questions

Gentlemen:

Enclosed are our reponses to ques ions posed by NRC staff on the subject application.

Very truly yours,

BALTIMORE GAS AND ELECTRIC COMPANY

A. E. Lundvall, Jr.
Vice President - Supply

AEL/WJL/mit

Copy To: J. A. Biddison, Esquire (w/out Encl.)
G. F. Trowbridge, Esquire (w/out Encl.)
Messrs. E. L. Conner, Jr., NRC
P. W. Kruse, CE

Enclosure 1 (40 Copies)

Enclosure (Calvert Cliffs Unit 1, Cycle 5, NRC Reload Questions Response - Answers on CESEC Model Used in SLB Analysis) - Proprietary Copies #000001 - 000040, 20 Non-Proprietary Copies

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AFFIDAVIT PURSUANT

TO 10 CFR 2.790

Combustion Engineering, Inc.)
State of Connecticut)
County of Hartford) SS.:

I, A. E. Scherer depose and say that I am the Director, Nuclear Licensing of Combustion Engineering, Inc., duly authorized to make this affidavit, and have reviewed or caused to have reviewed the information which is identified as proprietary and referenced in the paragraph immediately below. I am submitting this affidavit in conformance with the provisions of 10 CFR 2.790 of the Commission's regulations and in conjunction with the application of Baltimore Gas and Electric Company, for withholding this information.

The information for which proprietary treatment is sought is contained in the following document:

Calvert Cliffs Unit 1, Cycle 5, NRC Reload Question Responses (Answers on CESEC Model Used in SLB Analysis)

This document has been appropriately designated as proprietary.

I have personal knowledge of the criteria and procedures utilized by Combustion Engineering in designating information as a trade secret, privileged or as confidential commercial or financial information.

Pursuant to the provisions of paragraph (b) (4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure, included in the above referenced document, should be withheld.

1. The information sought to be withheld from public disclosure are selected input data and results from the analysis of a Steam Line Break event, which is owned and has been held in confidence by Combustion Engineering.

2. The information consists of test data or other similar data concerning a process, method or component, the application of which results in a substantial competitive advantage to Combustion Engineering.

3. The information is of a type customarily held in confidence by Combustion Engineering and not customarily disclosed to the public. Combustion Engineering has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The details of the aforementioned system were provided to the Nuclear Regulatory Commission via letter DP-537 from F.M. Stern to Frank Schroeder dated December 2, 1974. This system was applied in determining that the subject documents herein are proprietary.

4. The information is being transmitted to the Commission in confidence under the provisions of 10 CFR 2.790 with the understanding that it is to be received in confidence by the Commission.

5. The information, to the best of my knowledge and belief, is not available in public sources, and any disclosure to third parties has been made pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence.

6. Public disclosure of the information is likely to cause substantial harm to the competitive position of Combustion Engineering because:

a. A similar product is manufactured and sold by major pressurized water reactors competitors of Combustion Engineering.

b. Development of this information by C-E required thousands of man-hours of effort and tens of thousands of dollars. To the best of my knowledge and belief a competitor would have to undergo similar expense in generating equivalent information.

c. In order to acquire such information, a competitor would also require considerable time and inconvenience related to development of methodologies and determination of input parameters for a Steam Line Break analysis.

d. The information required significant effort and expense to obtain the licensing approvals necessary for application of the information. Avoidance of this expense would decrease a competitor's cost in applying the information and marketing the product to which the information is applicable.

e. The information consists of selected input data and results from analyses of Calvert Cliffs, Unit 1 Cycle 5 Steam Line Break event, the application of which provides a competitive economic advantage. The availability of such information to competitors would enable them to modify their product to better compete with Combustion Engineering, take marketing or other actions to improve their product's position or impair the position of Combustion Engineering's product, and avoid developing similar data and analyses in support of their processes, methods or apparatus.

f. In pricing Combustion Engineering's products and services, significant research, development, engineering, analytical, manufacturing, licensing, quality assurance and other costs and expenses must be included. The ability of Combustion Engineering's competitors to utilize such information without similar expenditure of resources may enable them to sell at prices reflecting significantly lower costs.

g. Use of the information by competitors in the international marketplace would increase their ability to market nuclear steam supply systems by reducing the costs associated with their technology development. In addition, disclosure would have an adverse economic impact on Combustion Engineering's potential for obtaining or maintaining foreign licensees.

Further the deponent sayeth not.

A. E. Scherer

Director

Nuclear Licensing

Sworn to before me

this 3RD day of November, 1980

Susan M. Vilcu

Notary Public

ENCLOSURE 1

Question 1

Why was the code CESEC-SLB needed to simulate the Steam Line Break (SLB) event for Cycle 5?

Answer

The SLB event analyses for Cycle 5 included the effects of manually tripping the Reactor Coolant Pumps (RCPs) on Safety Injection Actuation Signal (SIAS) due to low pressurizer pressure and the automatic initiation of Auxiliary Feedwater (AFW) flow on low steam generator water level signal. The inclusion of these effects in the Cycle 5 analyses required the use of the latest version of CESEC (which has been referred to in Appendix C of Reference 1 as CESEC-SLB) for the following reasons:

1. The manual trip of RCPs results in a drastic reduction in flow. The reduced flow causes increased temperature tilt at the reactor vessel inlet which, due to incomplete mixing of the coolant in the vessel inlet plenum produces a much more severe radial temperature asymmetry in the core. The temperature asymmetry experienced during an SLB event with RCP trip requires the use of RCS coolant node scheme which is capable of representing incomplete mixing in the reactor vessel.

2. The trip of the RCPs also affects the Reactor Coolant System (RCS) pressure. The RCS pressure determines the magnitude of Safety Injection flow via the High Pressure Safety Injection (HPSI) pumps, and, thus the total negative reactivity added due to boron injected. Due to reduced flow through the reactor vessel closure head, a model which explicitly represents the reactor vessel closure head was required to more accurately predict the pressure variation during the event.

3. The RCP trip also required a more accurate prediction of the way boron injected via the HPSI pumps is distributed in the RCS loop to provide negative reactivity. Hence, an improved modelling of the boron transport in the primary coolant and of the safety injection system was required.

4. The RCP trip required a flow model which is able to explicitly calculate the time dependent reactor coolant mass flow rate.

5. The automatic initiation of AFW required modification of the primary to secondary heat transfer model in orde to calculate the RCS cooldown after AFW initiation to a potentially dry steam generator.

Hence, the version of CESEC which includes the above mentioned model improvements was used to analyze the SLB event for Cycle 5.

References to Question #1: Unit 1, Cycle 5 License Submittal

Question 2

Provide a description of the overall conservatism inherent in the CESEC-SLB code. List all conservative assumptions in the codes and inputs for each parameter.

Answer

The conservatism in the Steam Line Break analyses exists in mainly the input data rather than the CESEC code. The only inherent conservatism in this CESEC version is that the heat transfer area is calculated assuming that all tubes are covered until the mass in the steam generator is equal to 5000 lbm. This assumption is conservative for assessing the potential for a return to power, since it increases the heat transfer rate between the primary and secondary and thus, produces the maximum cooldown of the RCS.

The conservatisms in the key input data are given below.

Parameter	Value	Justification
Power Level	2754 Mwt	This is the maximum allowed thermal power including uncertainty. The maximum power level results in maximum coolant average temperature. The maximum coolant average temperature increases the moderator reactivity inserted during the cooldown by increasing the total change in the coolant temperatures. Also, the maximum power level increases the decay heat.
Core Inlet Temperature	550° F	This is the maximum core coolant inlet temperature including uncertainty. The maximum inlet temperature results in a higher initial steam generator pressure, which increases the blowdown rate from both steam generators.
Break Area	6.35 ft^2	The analysis assumes the largest break area of 6.35 ft^2. This results in the fastest blowdown and thus the most rapid cooldown of the RCS and the greatest rate of temperature reduction in the reactor core region. This leads to a maximum positive reactivity insertion and the greatest potential for a return to power.
Moderator Cooldown Curve	See Figure 1 & 2	The moderator cooldown curve corresponds to an effective MTC of -2.2×10^{-4} $\Delta\rho/^{\circ}F$ and is calculated assuming that a Control Element Assembly is stuck in the fully withdrawn position during the reactor scram.

Parameter	Value	Justification
Doppler Coefficient Doppler Multiplier	EOC 1.15	The EOC Doppler coefficient in combination with the decreasing fuel temperatures causes the greatest positive reactivity addition, due to fuel temperature change, during the event. The Doppler uncertainty of 15% assumed in the analyses is in a sense that enhances the Doppler feedback.
Scram Worth-HFP HZP	7.15% $\Delta\rho$ 4.3% $\Delta\rho$	This is the minimum available scram worth at EOC and was calculated allowing for the stuck rod which produced the limiting moderator cooldown curve.
Inverse Boron Worth HFP HZP	105 PPM/% $\Delta\rho$ 100 PPM/% $\Delta\rho$	These correspond to the minimum boron reactivity worths for the boron injected via the High Pressure Safety Injection Pumps and minimize the negative activity added by Safety Injection.
High Pressure Safety Injection a) Number of Pumps b) Time Delay to Start Pumps c) Volume to be Swept Out Prior to Boron Injected Enters RCS Cold Legs	 1 30 sec 76 ft^3	The analyses conservatively assumed that only one HPSI pump is operable. In addition, the maximum Technical Specification time delay to start the pumps is also assumed. The maximum volume to be swept out prior to when boron injection enters the core is also assumed. These assumptions are conservative, since they delay the time at which boron injected via the HPSI pumps enters RCS cold legs.
Main Feedwater Flow	5% Full Power Flowrate	Maintaining the main feedwater flow to the ruptured steam generator increases the mass released during the blowdown, lengthens the blowdown, and aggrevates the cooldown. These are maximum value allowed by Technical Specifications.
Time to Rampdown Main Feed After Trip	20 sec	
Feedwater Isolation after MSIS	80 sec	
Auxiliary Feedwater Flow	350 lbm/sec	This value is conservatively calculated assuming that both auxiliary feedwater pumps are functional. The value corresponds to the pump run-out value due to reduced back pressure. In addition, the auxiliary feedwater flow is fed only to the damaged steam generator. These conservative assumptions produce the maximum cooldown of the RCS and thus enhance the potential for Return-To-Power after initiation of auxiliary feedwater flow.
β Fraction	.0060	The maximum EOC β fraction is used in the analyses. This causes the fastest approach to Return-To-Power due to subcritical multiplication.
Initial Steam Generator Pressure	853 psia	The value is the maximum initial steam generator pressure for the initial power, the initial core coolant temperature and mass flow rate used. This value is conservative because it increases the rate of blowdown of the steam generator.

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Parameter	Value	Justification
Initial RCS Pressure	2300 psia	This is the maximum initial pressure allowed. The use of the maximum pressure delays the time of Safety Injection Actuation Signal and thus the amount of negative reactivity contributed by Safety Injection.
Main Steam Isolation Valves Closure Time after Main Steam Isolation Signal	6.9 seconds	This is the maximum time to close the MSIV's. The maximum time prolongs the blowdown from the unaffected Steam Generator.

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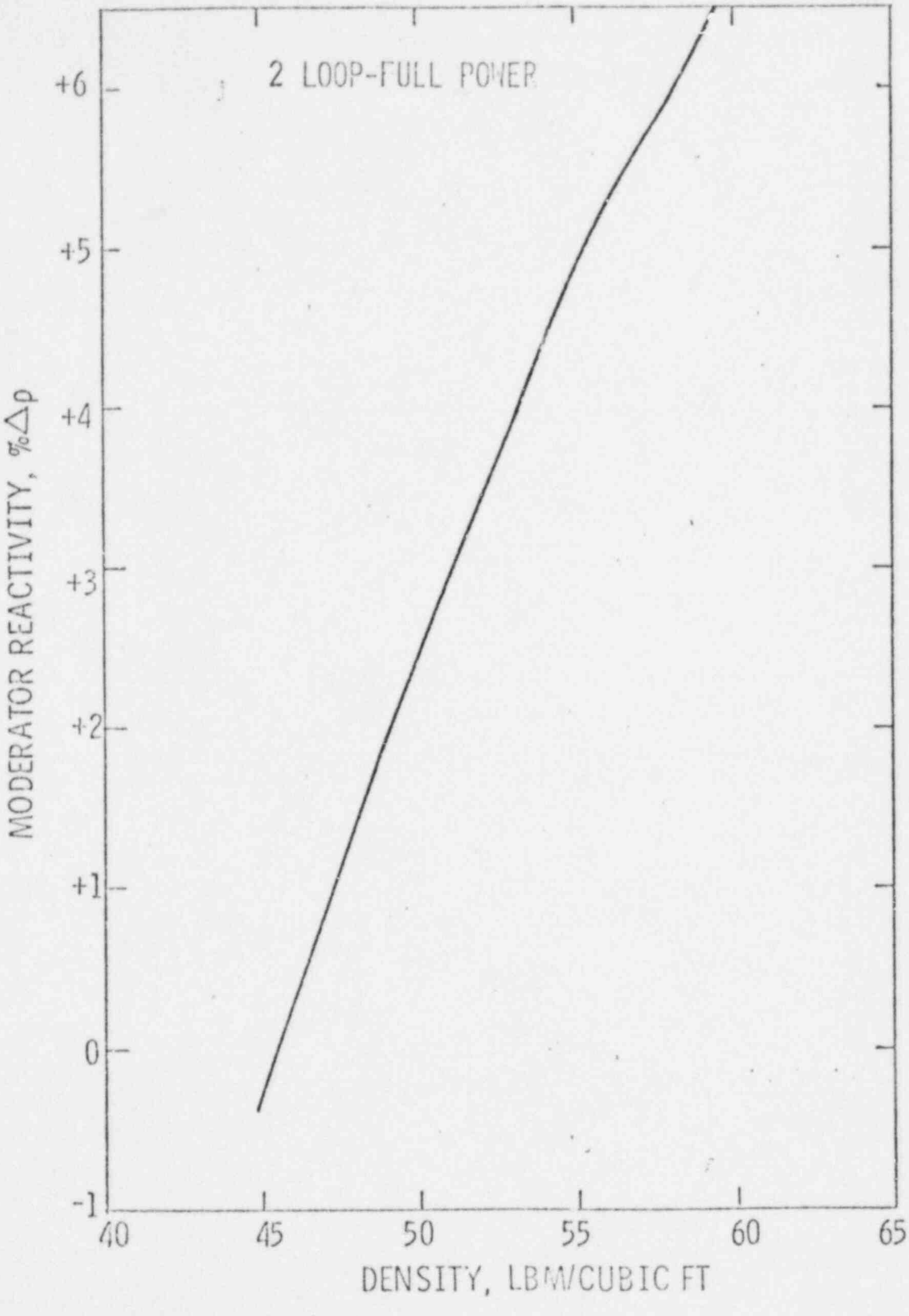
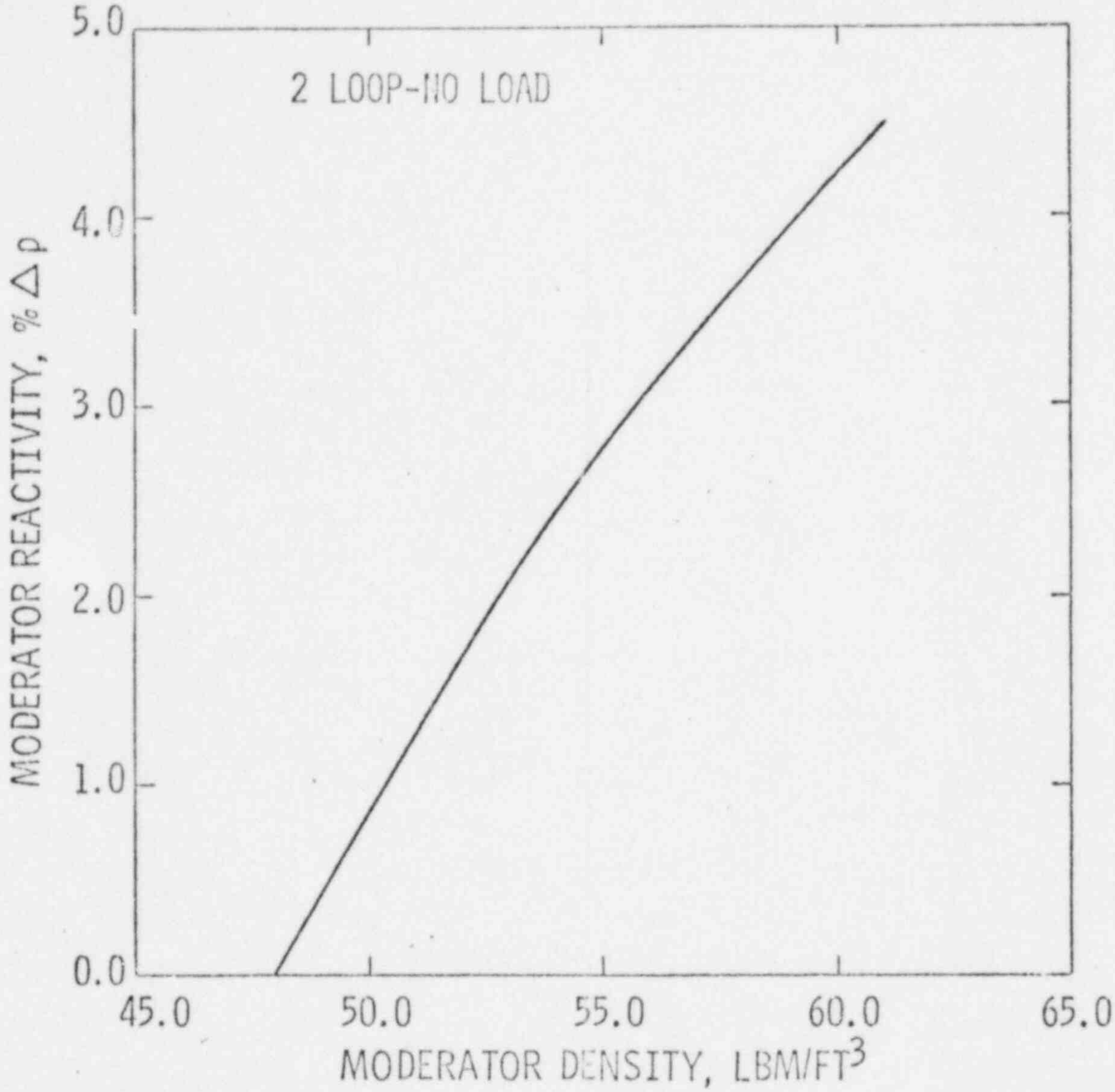


BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT MODERATOR REACTIVITY FEEDBACK vs MODERATOR DENSITY	1



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE RUPTURE EVENT MODERATOR REACTIVITY vs MODERATOR DENSITY	Figure 2

Question 3

Provide a more detailed description of the code CESEC-SLB, including pertinent equations.

Answer

The following provides a general description and equations used in the Steam Line Break analysis for Cycle 5. Models pertinent to the Steam Line Break analysis are specifically provided. Some models, such as flow from the Safety Injection Tanks are described for completeness but were not credited for in the Cycle 5 analysis.

SUMMARY DESCRIPTION OF THE VERSION OF THE CESEC CODE USED
FOR THE CALVERT CLIFFS UNIT 1, CYCLE 5, STEAM LINE BREAK ANALYSIS

Introduction

The CESEC digital computer program (1 to 8) provides for the simulation of a Combustion Engineering Nuclear Steam Supply System (NSSS). The program ... the plant response for non-LOCA (loss of coolant accident) ini- ...nts for a wide range of operating conditions.

...ogram, which numerically integrates the one-dimensional conser- vat... ...ions, assumes a node flow-path network to model the NSSS. The primary system components considered in the code include the reactor vessel, the reactor core, the primary coolant loops, the pressurizer, the steam generators, and the reactor coolant pumps (see Figure 1). The secondary system components, shown on Figure 2, include the secondary side of the steam generators, the Main Steam System, the Feedwater System, and the various steam control valves. In addition, the program models some of the control and plant protection systems.

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

In the subsections which follow, a description of the major models which comprise the version of the CESEC code which was used for the Steam Line Break analysis for the Calvert Cliffs Unit 1, Cycle 5 safety analysis is given.

Primary Coolant Thermal-Hydraulic Model

The CESEC code uses a node/flow-path type network to model the Reactor Coolant System. The conservation of mass and energy equations are solved for control volumes or nodes. Uniform pressure is assumed around the reactor coolant loops for the thermal-hydraulic solution. The conservation of momentum equation is solved independent of the conservation of mass and energy equations to obtain the pump flows for the thermal-hydraulic model (see section on Flow Model).

The Reactor Coolant System, consisting of two reactor coolant loops, the reactor vessel, the reactor vessel closure head and the pressurizer was divided into 27 nodes of constant volume for this analysis. The nodal scheme given in Figures 3A and 3B was chosen to appropriately simulate the RCS component volumes and, thus, provide an adequate description of the spatial variation of the coolant properties. As seen from Figures 3A and 3B, nodes are specified which represent one-half the reactor vessel inlet downcome section, the lower plenum, the core region, the bypass flow, and the upper plenum. These two symmetrical loops are linked by the cross flow at the reactor vessel inlet and outlet sections and by the flow mixing within the reactor vessel lower and upper plenums. The mixing factors are specified based on test data. No cross flow is assumed between the parallel regions in the core.

CESEC solves the conservation of mass and energy equations (see Figure 4) to obtain the time derivative of the pressurizer pressure, the internodal flows, the rate of vaporization or water enthalpy time derivative of the pressurizer water regions, and the rate of condensation or steam enthalpy time derivative of the pressurizer steam region. Computation of these parameters allows for the calculation of the RCS pressure time derivative, the time derivatives of the nodal enthalpies, the nodal specific volumes, and the nodal masses.

Closure Head Node

During the rapid contraction of the primary coolant which takes place as a result of a steam line break, the pressurizer empties and voids begin to form in the RCS. Since flow through the closure head is only a small fraction of the RCS flow, the temperatures in the closure head remain high and voiding first occurs there. To some extent, the closure head itself then begins to perform the function of a pressurizer. Therefore, the reactor vessel closure head region is explicitly modeled in this CESEC version to more accurately predict the RCS pressure. The coolant flow from the core outlet nodes to the vessel head node is specified by a user input fraction. It is assumed that the vessel head fluid returning into the outlet nodes is evenly distributed between the two loops.

Pressurizer

The CESEC pressurizer model assumes steam and liquid regions to exist in one of the eight thermal-hydraulic states shown in Figure 5. The model considers such components as sprays, heaters, and relief/safety valves. The Pressurizer Level Control System which controls charging flow and letdown flow by means of pressurizer level setpoints, is also modeled.

The mass and energy transport between the two fluid regions is assumed to occur as a result of liquid vaporization and/or steam condensation. The spray flow which enters the pressurizer is assumed to condense the steam if it is in the saturation state. That is, when the steam region is at saturation, the spray droplets are assumed to reach saturation temperature and will result in bulk condensation of the steam. However, when the steam region is superheated, the spray droplets are assumed to evaporate into the steam region.

The code models two spray operating modes, continuous and proportional. The continuous mode spray is a user input constant flow which is added continuously to the pressurizer. The proportional mode spray flow originates at the pump discharge in the RCS loop and is linked to the pressurizer as shown on Figure 3. The spray flow for the proportional mode is controlled automatically by two pressure setpoints which turn the spray on and off, respectively. Within these two setpoints, the spray flow increases linearly with the pressurizer pressure.

The code also models two types of heaters located near the bottom of the pressurizer: (1) the proportional heaters which are controlled by the Pressurizer Pressure Control System to generate heat at a rate which decreases linearly with increasing pressure between two pressure setpoints and (2) the backup heaters which turn on and off at two pressure setpoints.

In addition, the backup heaters are also controlled by the measured deviation of the pressurizer liquid level from the programmed level. The addition of heat from heaters to the fluid is accounted for in the conservation of energy equation.

Flow Model

The flow model in this CESEC version calculates the mass flow rate (lbm/sec) at the pump outlet for each reactor coolant system steam generator loop. The model includes explicit simulations of the reactor coolant pumps and of the effects of natural circulation flow. The calculation is based on a solution of the one-dimensional momentum equation for each RCS loop. The loops are divided into a number of nodes whose densities, temperatures, and flows are obtained from the CESEC thermohydraulics model.* The flow model utilizes this nodalization of the loop to calculate the sum of the various forces around the loop. The forces acting on the fluid volume consist of (1) gravitational forces due to density and elevation changes around the loop, (2) forces due to wall friction and geometric changes in the flow path, and (3) forces due to the RCS pumps. The one-dimensional momentum equation for each loop, is written as follows:

$$\frac{dw}{dt} = \frac{g\sum_{i=1}^{n} \rho_i h_i - \sum_{i=1}^{n} w_i^2 \left[\frac{\emptyset_i f_i R_{fric.i}}{\rho_{is}} + \frac{R_{geo,i}}{\rho_{in}}\right] + \Delta P_{pump}}{\sum_{i=1}^{n} (L_i/A_i)} \qquad (1)$$

*The average of the properties and the flows from parallel nodes are used for nodes representing the reactor vessel.

where w = mass flow rate at the pump, lbm/sec

w_i = mass flow rate of i^{th} node, lbm/sec

ρ_{in} = average fluid density of i^{th} node, lbm/ft^3

ρ_{is} = single-phase fluid density of i^{th} node, lbm/ft^3

$h_i = h_{in,\,i} - h_{out,i}$ = the elevation difference across the i^{th} node, ft

$\emptyset_i$ = Thom two-phase multiplier for the i^{th} node

f_i = Darcy friction factor for the i^{th} node

L_i = effective flow path length for the i^{th} node, ft

A_i = effective cross sectional flow area of i^{th} node, ft^2

and $R_{fric,\,i} = \dfrac{L_i/D_{e,i}}{2A_i^2}$

$$R_{geo,i} = \frac{K_{g,i}}{2A_i^2}$$

where $D_{e,i}$ = effective diameter of i^{th} node, ft

$K_{g,i}$ = flow loss coefficient for geometric changes in the flow path, dimensionless

The first term on the right hand side of Equation 1 represents the net pressure change around the loop due to the gravitational force acting on each fluid node. The second term represents the total pressure change around the loop due to the frictional loss and the geometric changes in the flow path. The Darcy friction factor, f, which is a function of the Reynold's number, Re, is determined by the following correlations:

$$f = 64/Re \qquad Re < 1250$$

$$f = (-0.000004)Re+0.056 \qquad 1250 \leq Re < 600 \qquad (2)$$

$$f = 0.184/Re^{0.2}) \qquad Re \geq 6000$$

The third term in Equation 1 represents the pressure difference ΔP_{pump}, across the RCS pump, which is calculated by the dynamic pump model described in the following paragraphs.

The pump model calculates the pressure difference across the pump for use in the conservation of momentum equation. The pressure difference, or pump head, is dependent on the pump speed and flow. The speed is calculated throughout the transient, thus, giving a pump head dependent on transient flow conditions. The pump speed is determined from the following equation:

$$\frac{d\omega}{dt} = (T_{el} - T_h - T_{f,W})\frac{g_c}{I} \qquad (3)$$

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where ω = angular velocity of the rotating assembly

t = time

T_{el} = electrically induced torque acting on the motor rotor

T_h = hydraulic torque exerted on the fluid by the pump impeller

$T_{f,W}$ = torque exerted on the rotating assembly due to bearing friction and windage losses

g_c = gravitational constant

I = moment of inertia of the rotating assembly

The three torques (T_{el}, T_h, and $T_{f,W}$) are calculated as follows:

a) The hydraulic torque is calculated from the following equations:

$$T_h = \begin{cases} (\beta/\alpha^2)(\alpha^2)(T_R)(\rho/\rho_R) & \text{for } \left|\nu/\alpha\right| \leq 1.0 \\ (\beta/\nu^2)(\nu^2)(T_R)(\rho/\rho_R) & \text{for } \left|\nu/\alpha\right| > 1.0 \end{cases} \qquad (4)$$

where

β = Ratio of the hydraulic torque to the rated hydraulic torques, $\beta \equiv T_h/T_R$

α = Ratio of the pump speed to the rated pump speed, $\alpha \equiv \omega/\omega_r$

ν = Ratio of the volumetric flow rate to the rated volumetric flow rate, $\nu \equiv Q/Q_r$

ρ = Density of coolant, lbm/ft^3

ρ_R = Density corresponding to pump rated conditions, lbm/ft^3

The values of β/α^2 and β/ν^2 as a functions of ν/α and α/ν, respectively, are determined from the single phase homologous pump curves.

b) The friction and windage torque is calculated from the following equation:

$$T_{fW} = \alpha|\alpha| T_{fWI} \qquad (5)$$

where T_{fWI} is the input friction and windage torque at rated speed.

c) Electrical torque is found by interpolating in an input table of speed vs. electrical torque using the pump speed from the previous time step.

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With all quantities known in Equation 3, the rate of change of pump speed can be calculated. The pump speed is then changed by the product of this rate with the time step size for the next time step.

The pump head, H, is calculated from the following equation:

$$H = \begin{cases} (\alpha^2)(H_R) \quad (h/\alpha^2) & \text{for } |v/\alpha| \leq 1.0 \\ (v^2)(H_R) \quad (h/v^2) & \text{for } |v/\alpha| > 1.0 \end{cases} \tag{6}$$

where:

H = Pump head, ft of water

H_R = Rated pump head, ft of water

h = Ratio of the pump head to the rated pump head, $h \equiv H/H_R$

The value of h/α^2 and h/v^2 as a function of v/α and α/v, respectively, are determined from the single phase homologous pump curves.

The pressure difference across the pump is then calculated from:

$$\Delta P_{pump} = H\rho g \tag{7}$$

Reactor Kinetics

The energy source in the CESEC code is from fission in the fuel. The core is represented by a cylindrical fuel rod located in an average coolant channel. This fission energy consists of two parts, the instantaneous fission power and the decay power released by the fission products. The instantaneous power is determined by solving the standard point kinetics neutron equations with six delayed neutron groups while the decay power is calculated from an 11 fission product group decay heat model.

The total reactivity in the point kinetics equation is calculated as the sum of the control rods, moderator, fuel temperature (Doppler), and boron contributions. The code also has an explicit function of time simulating the control rod reactivity insertion. A table of rod reactivity versus time after initiation of scram is user input. The moderator feedback effects considered include the moderator density or the moderator temperature. The moderator and Doppler reactivity feedback terms are calculated at each time step by interpolation of user input tables. The boron reactivity effect includes the contribution from the Safety Injection System and the letdown and charging portions of the Chemical and Volume Control System.

The kinetics equation is solved numerically by a fourth order Runge-Kutta/Merson method for the power generation at each time step.

Heat Transfer Within the Core

The CESEC core heat transfer model represents a fuel rod at core average conditions. The cylindrical configuration models the fuel, gap and clad. The fuel rod is divided into three equal-volume radial nodes (see Figure 6). The third radial node is assumed to contain the outer portion of the fuel, the gap, and the clad. The radial energy equation (see Figure 6) is formulated for each node with the nodal properties (e.g., specific heat and thermal conductivity of fuel and clad) determined by temperature dependent correlations. The input parameters required by the model include the fractions of power generated within the fuel, the clad, and the moderator, respectively, and the gap conductance which is assumed to be a constant. Within the fuel region, a uniform power distribution is assumed by the code.

The heat transfer at the clad - coolant interface is assumed to be given by the following correlation for all fluid conditions (Reference 6):

$$h = 0.148\ (1+0.01T - 0.00001T^2)\ \frac{V^{0.8}}{D^{0.2}} \tag{8}$$

where T = fluid temperature

V = fluid velocity

D = channel hydraulic diameter

Initially, the steady state fuel temperature distribution is determined by a scheme which solves the radial energy equation iteratively based on the initial reactor power output, the gap conductance, and the initial coolant condition. The radial energy equation is solved numerically at each time step by a fourth order Runge-Kutta/Merson method.

Charging and Letdown

The CESEC code provides a model for calculating the charging and letdown flows. The contributions from the charging and letdown flows are included in the conservation of mass and energy equations for the corresponding RCS nodes. Included in the model is a Pressurizer Level Control System which determines the deviation between the measured pressurizer water level and the programmed level. The programmed level is given by an input table as a function of either power or average RCS temperature. The algorithm by which the measured level is calculated is described in Reference 5. The charging flow is provided by a set of constant speed charging pumps, with the charging flow rate automatically controlled by switching each pump on or off at two input level deviation setpoints. The letdown flow control is provided either by a set of letdown control and backpressure valves, with the flow rate either controlled by the opening or the closing of each set of valves at two level deviation setpoints, or by a linear letdown flow control model.

The charging and letdown fluid temperatures are user input. In addition, the letdown fluid temperature can be selected to be that corresponding to the steam generator outlet temperature. The boron concentration from the letdown and charging portion of the Chemical and Volume Control System (CVCS) is only accounted for in CESEC when the Safety Injection System is activated. However, the user can optionally turn off the letdown and charging systems and take no credit for the boron reactivity contribution from the letdown and charging systems. The calculation of the boron concentration in the reactor coolant is described in the Safety Injection System section.

Reactor Protective System Trips

The reactor is shutdown by the insertion of the control element assemblies (CEAs) following the generation of a trip signal. A trip signal is initiated when a certain system parameter reaches a value which exceeds the corresponding user input trip setpoints. The delay time between the initiation of the trip signal and the start of CEA motion is accounted for in CESEC. The CEA motion is represented by an input rod worth versus time table. The following trips are programmed in the CESEC code:

1. high power trip,

2. high pressurizer pressure trip,

3. low pressurizer pressure trip,

4. low coolant flow trip,

5. low steam generator pressure trip,

6. low steam generator level trip, and

7. manual trip.

To generate the trip signal on the low steam generator water level, the steam generator water level is determined from a set of steady state input data and the transient inventory in the steam generator. The set of steady state curves relates steam generator water level to secondary water mass and power level. This data is then used in a table look-up routine to obtain the steam generator water level for the purpose of determining the trip signal.

Safety Injection System

The borated safety injection water from the high and low pressure safety injection pumps is injected into each cold leg node downstream of the reactor coolant pumps. The borated injection flow rates versus pressure are specified by input tables. Once the safety injection flow reaches the cold leg node, it is assumed to mix homogeneously with the reactor coolant in that node. The boron is transported through the RCS by solving at each time step the continuity equation for each coolant node for the boron concentration:

$$M \frac{dC}{dt} = W_{in} C_{in} - W_{out} C \tag{9}$$

where

- C is the boron concentration
- C_{in} is the inlet boron concentration
- W_{in} is the inlet flow rate
- W_{out} is the outlet flow rate
- M is the mass inventory in the node

The boron concentration for the reactor core node is used to calculate the reactivity contribution due to boron via an input reciprocal boron worth.

A time delay is input to CESEC to account for the time required to start the diesel generator and/or to bring the safety injection pumps to full speed. An additional time delay is calculated to account for the time required for the unborated water in the safety injection line (from the outlet of the safety injection pumps to the injection nozzles) to be swept out before borated water from the refueling water tanks enters the cold legs.

CESEC also solves an orifice equation to determine the rate of safety injection flow from the safety injection tanks into the RCS as a function of time. The input parameters are the initial nitrogen pressure, volume of water, volume of gas, flow coefficient, flow area, water specific volume, and elevation head of the safety injection tanks. In addition to the nitrogren pressure within the safety injection tank, the static head of fluid within the safety injection piping is considered when calculating the instantaneous pressure difference across the orifice. The nitrogen expansion process is assumed to be isentropic.

In computing the safety injection flow rate by means of an orifice equation, the code takes into account the effect of piping friction, turning losses, and expansion/contraction losses through the use of a single equivalent loss coefficient which is based on the minimum cross-sectional flow area. The instantaneous liquid discharge rate at time t is given by

POOR ORIGINAL

$$W(t) = A\left(\frac{288\ g_c \Delta P(t)}{Kv}\right)^{1/2} \tag{10}$$

$$\Delta P(t) = P_G(t) + P_E - P_{RCS}(t) \tag{11}$$

where W is the mass flow rate in lbm/sec
A is the flow area of safety injection tank line in ft^2
K is the friction loss coefficient for the flow area
v is the specific volume of liquid in ft^3/lbm

$P_G(t)$ = nitrogen pressure at time t

P_E = elevation head

$P_{RCS}(t)$ = RCS pressure at time t

If ΔP is less than or equal to zero, the code sets this variable equal to zero in order that no liquid mass be ejected from the tank for this condition.

The instantaneous liquid volume V in the tank at time t is then

$$V(t) = V(t-\Delta t) - W(t)\cdot\Delta t\cdot v \tag{12}$$

where Δt is the time step interval.

The instantaneous gas volume V_G in the tank at time t is given by

$$V_G(t) = V_G(0) + V(0) - V(t) \tag{13}$$

where $V_G(0)$ and $V(0)$ are the initial gas volume and liquid volume, respectively.

The instantaneous gas pressure in the tank at time t is given by:

$$P_G(t) = P_G(0)\left(\frac{V_G(0)}{V_G(t)}\right)^{1.4} \tag{14}$$

Critical Flow Model

For steam escaping from the ruptured steam line, the mass flow rate is calculated in CESEC from the following empirical critical flow correlation[4]:

$$\frac{W}{A} = 1977.6 \times \frac{P}{h-185.0} \qquad (15)$$

where:

W is the mass flow rate, lbm/sec

A is the effective flow area, ft^2

P is the steam pressure, Psia

h is the steam enthalpy, Btu/lbm

Steam Generator Model

The CESEC steam generator model performs a detailed computation of the overall heat transfer coefficient for each steam generator. The heat transfer correlation used in the primary side for calculating the film resistance is the same as for the core. The secondary side heat transfer mechanism is pool boiling. The boiling resistance is calculated using the modified Rohsenow pool boiling correlation[6].

$$h_{sec} = K_R \left(\frac{q}{A}\right)^{2/3} \qquad (16)$$

where

q = heat rate

A = heat transfer area

K_R = a pressure dependent coefficient given by a C-E proprietary correlation. (Reference 6)

In the CESEC version used for this analysis, the heat transfer between the primary and secondary sides of the steam generator is calculated using algorithms based upon a polynomial spatial variation of the primary side temperature. The polynomial algorithm selects the mid-point temperatures of the nodes simulating the steam generator tubes and the inlet and outlet plenum to obtain the spatial variation of the primary side temperature along the steam generator tubes. The difference between this temperature variation and the secondary side temperature is integrated between the node mid-points and divided by the distance between these points to obtain the average primary-to-secondary temperature difference. This temperature difference is then used to calculate the steam generator heat transfer rate on a node-center-to-node-center basis at each time step.

The overall heat transfer coefficient is determined from the film resistance of the primary and secondary sides, the tube wall resistance, and the tube fouling resistance. The tube wall/fouling resistance is determined initially by the design full power condition and is assumed to be constant thereafter during the transient. These heat transfer coefficients are calculated on a node-center-to-node-center basis rather than for the steam generator as a whole.

For low heat flux predictions, the secondary-side heat transfer coefficient is limited to a constant input value, rather than being allowed to go to zero with the heat flux as in Equation 16. This minimum value is also used for the secondary-side heat transfer coefficient for conditions of reverse (secondary-to-primary) heat transfer. The primary-side coefficients are calculated using the same correlation (Equation 8) for forward and reverse heat transfer.

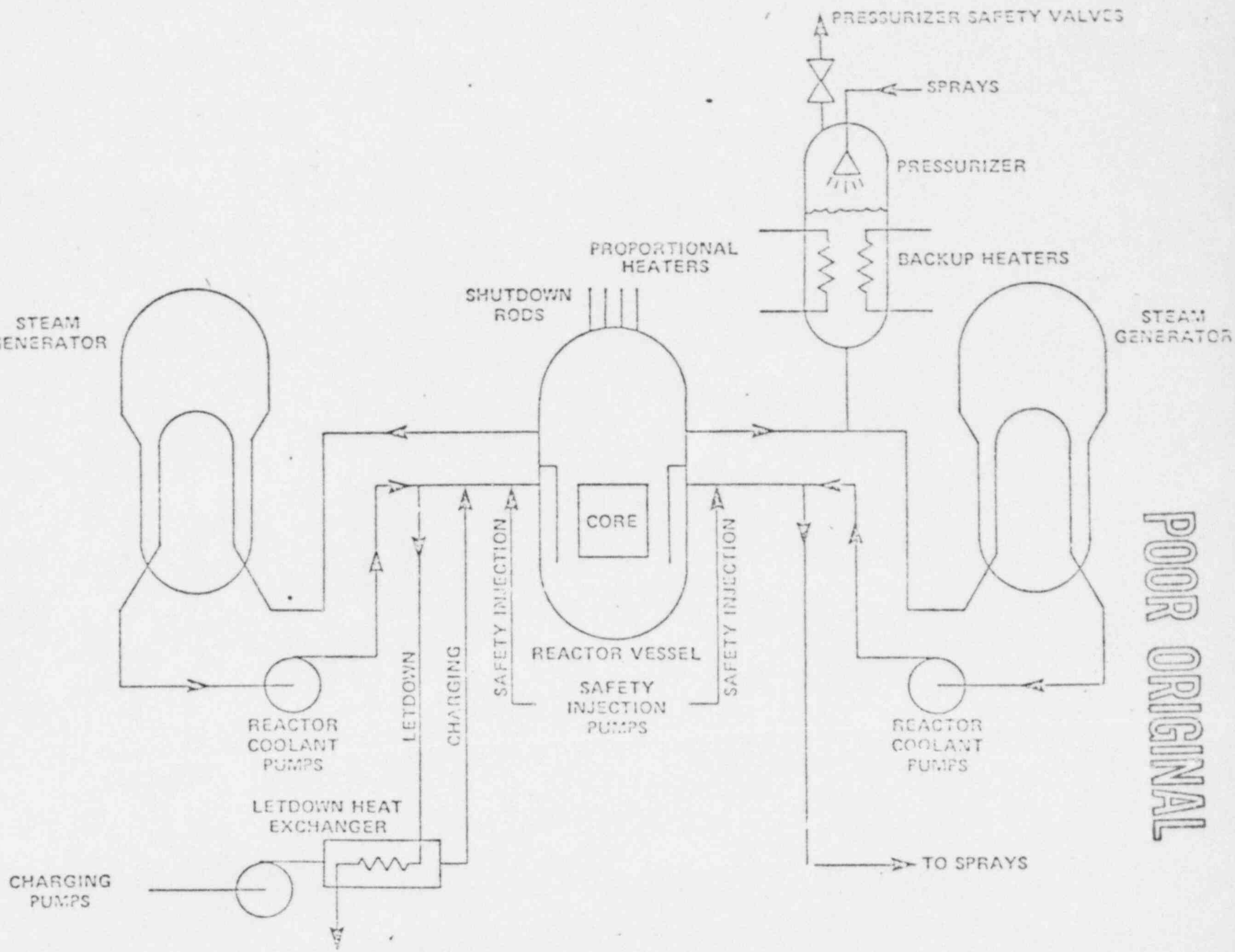
The secondary side of each steam generator is represented by a control volume. The control volume consists of saturated liquid and steam. The fluid properties and mass inventory are determined by solving the conservation of mass and energy equations shown on Figure 7. The initial conditions of the secondary side of the steam generator are determined by iterating on the secondary pressure, given the initial heat flux and power demand as specified by the user.

To avoid singularities in the solution of the conservation equations shown in Figure 7, the liquid inventory in each steam generator is limited to a minimum of 2500 lbm. Thus, steam is allowed to escape through the ruptured steam line with a critical flow velocity as long as the steam pressure is above atmospheric pressure and the steam generator liquid inventory is greater than 2500 lbm. Smooth transition from normal heat transfer conditions to a condition which does not reduce the liquid inventory below the minimum of 2500 lbm and also maintains the saturated conditions assumed by the model for the secondary side is achieved by a rampdown of the product of the overall heat transfer coefficient with the heat transfer area, UA. For steam generator inventories greater than 5000 lbm, the total heat transfer area is used together with a primary-to-secondary heat transfer coefficient calculated using the correlations given above. For liquid inventories less than 2500 lbm the product of heat transfer coefficient and heat transfer area is assumed to be just sufficient to raise the enthalpy of any incoming feedwater to that of the saturated liquid. For liquid inventories between 2500 lbm and 5000 lbm the product UA is scaled linearly between values calculated for inventories greater than 5000 lbm and inventories less than 2500 lbm.

The feedwater flow is optionally determined in CESEC by the following three methods: 1) matching the steam flow, 2) input table of flow rate versus time, 3) automatic feedwater control on the steam generator water level. The initial feedwater flow is assumed to match that corresponding to the power level at time zero. The flow during an event is calculated according to the user option selected. The feedwater isolation valves are programmed to close at a specified rate of closure following the main steam isolation signal which is actuated on low steam generator pressure. The feedwater enthalpy can be specified by input tables of enthalpy as a function of either power level or time. Auxiliary feedwater flow and enthalpy are modeled using the input table option given above.

The path of the steam flow from the secondary side of the steam generator is illustrated on Figure 2. Downstream of the main steam isolation valve, the main steam lines from each steam generator are connected together at a common steam header. At the initial steady state, the steam flow in each steam line is determined consistent with the reactor coolant flow rate in each steam generator loop. During the transient calculation, the steam flow is determined by the turbine power demand, the operating of the secondary valves, and the break flow rate. The steam flow through each valve is assumed to be choked. Thus, a critical flow correlation (Equation 15) for steam is used to calculate the flow rate.

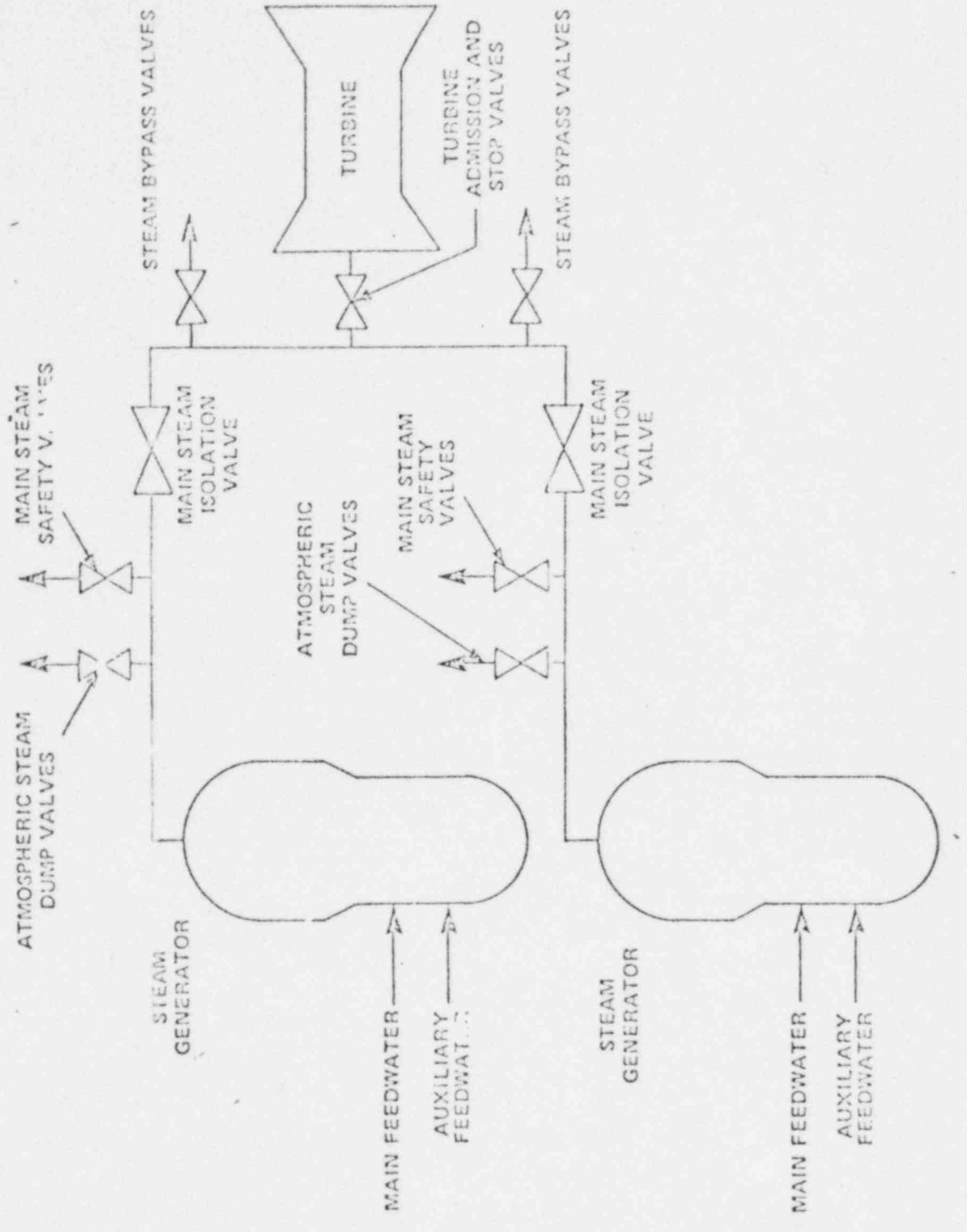
The code simulates two main steam isolation valves, one for each main steam line. These valves are normally open and do not affect steam generator operation unless the steam generator pressure drops below a specified setpoint. Once this occurs, the MSIVs begin to close after an input delay time. As the MSIVs close, steam flow to the turbine and other downstream components terminates.



BALTIMORE GAS AND ELECTRIC
CALVERT CLIFFS NUCLEAR POWER PLANT

THE PRIMARY SYSTEM AND CVCS COMPONENTS

FIGURE 1



BALTIMORE GAS AND ELECTRIC
CALVERT CLIFFS NUCLEAR POWER PLANT

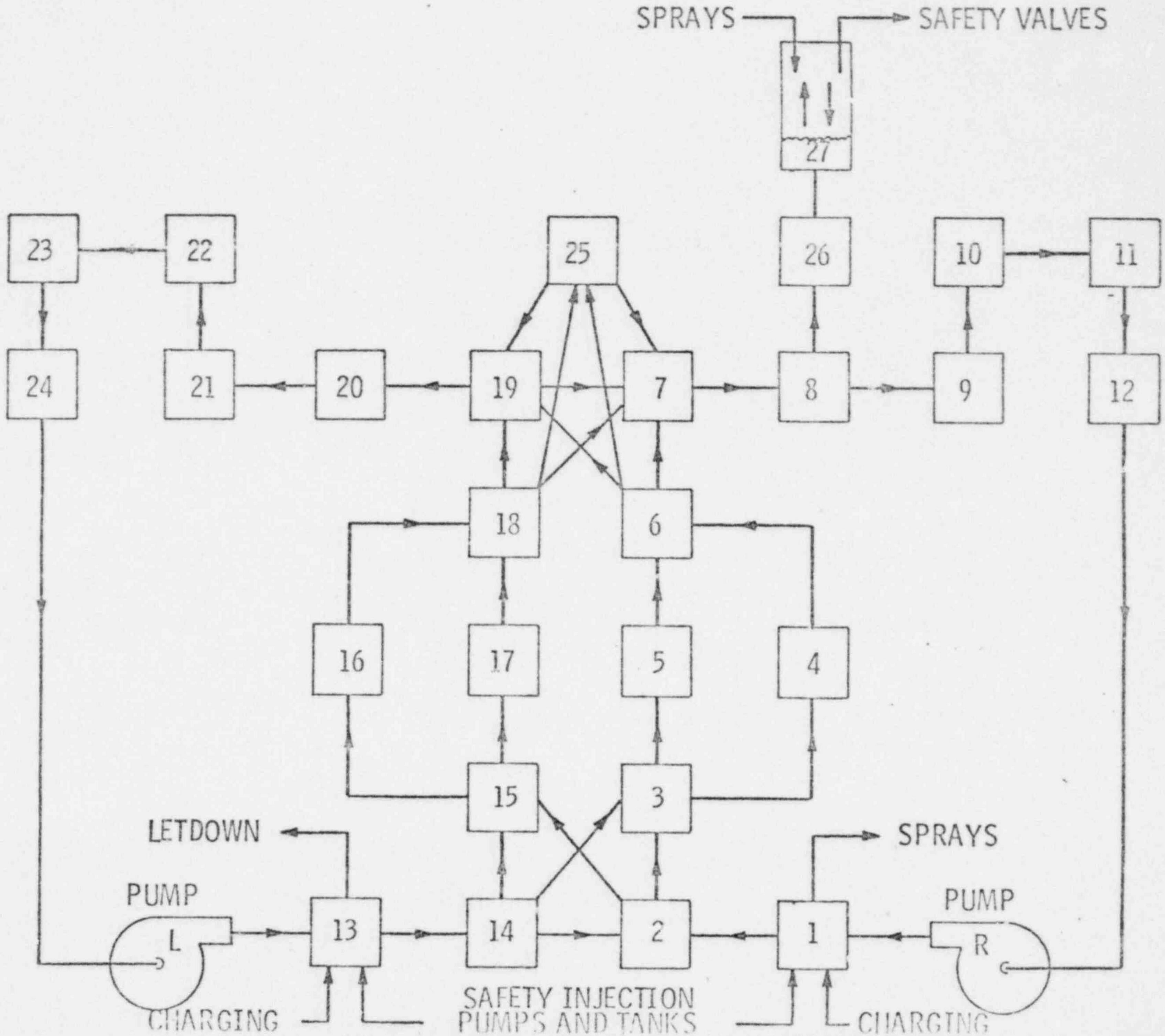
THE SECONDARY SYSTEM COMPONENTS

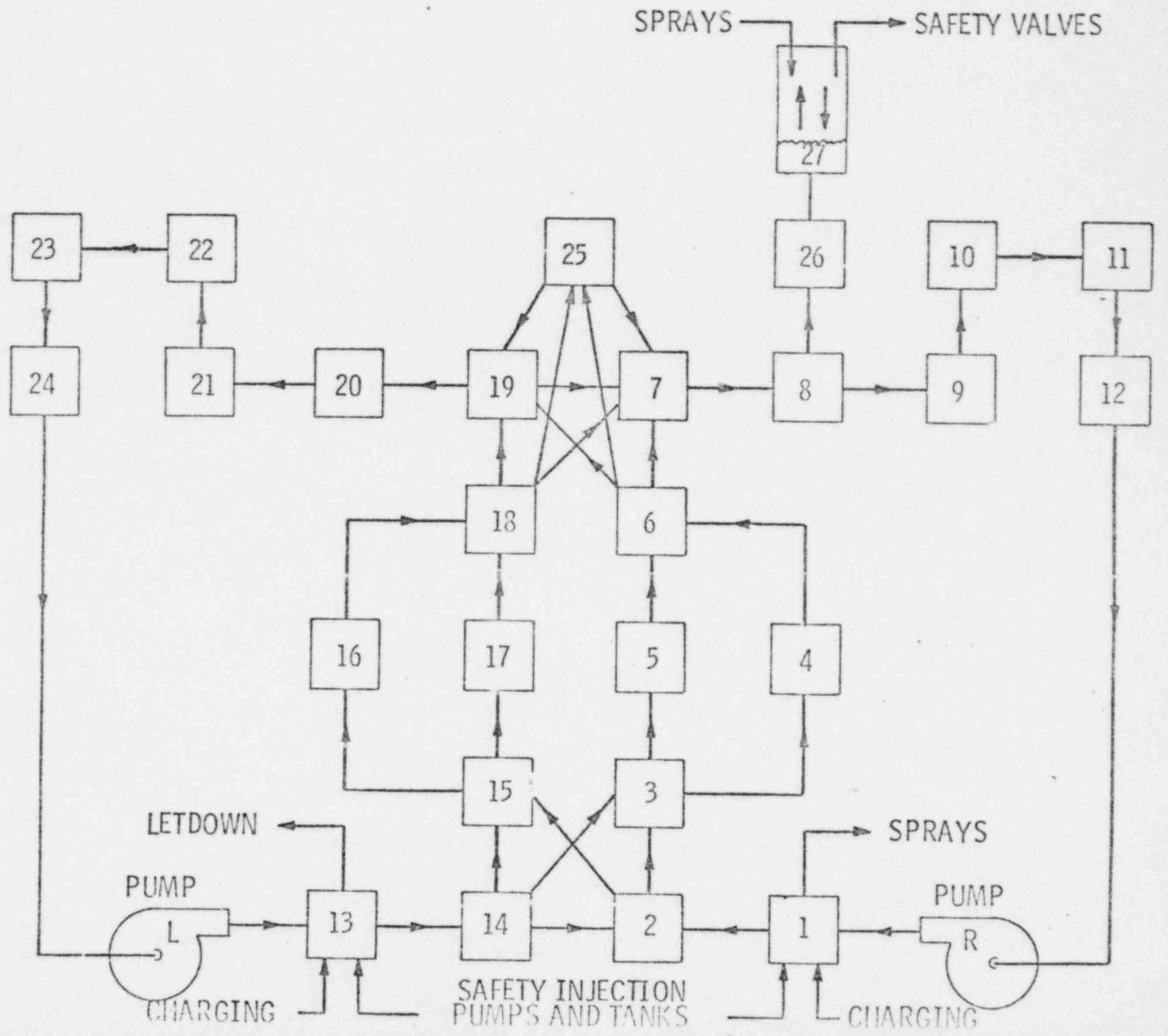
FIGURE 2

BALTIMORE GAS & ELECTRIC CO.
Calvert Cliffs Nuclear Power Plant

SCHEMATIC DIAGRAM OF PRIMARY COOLANT NODES

Figure 3A





BALTIMORE GAS & ELECTRIC CO.
Calvert Cliffs Nuclear Power Plant

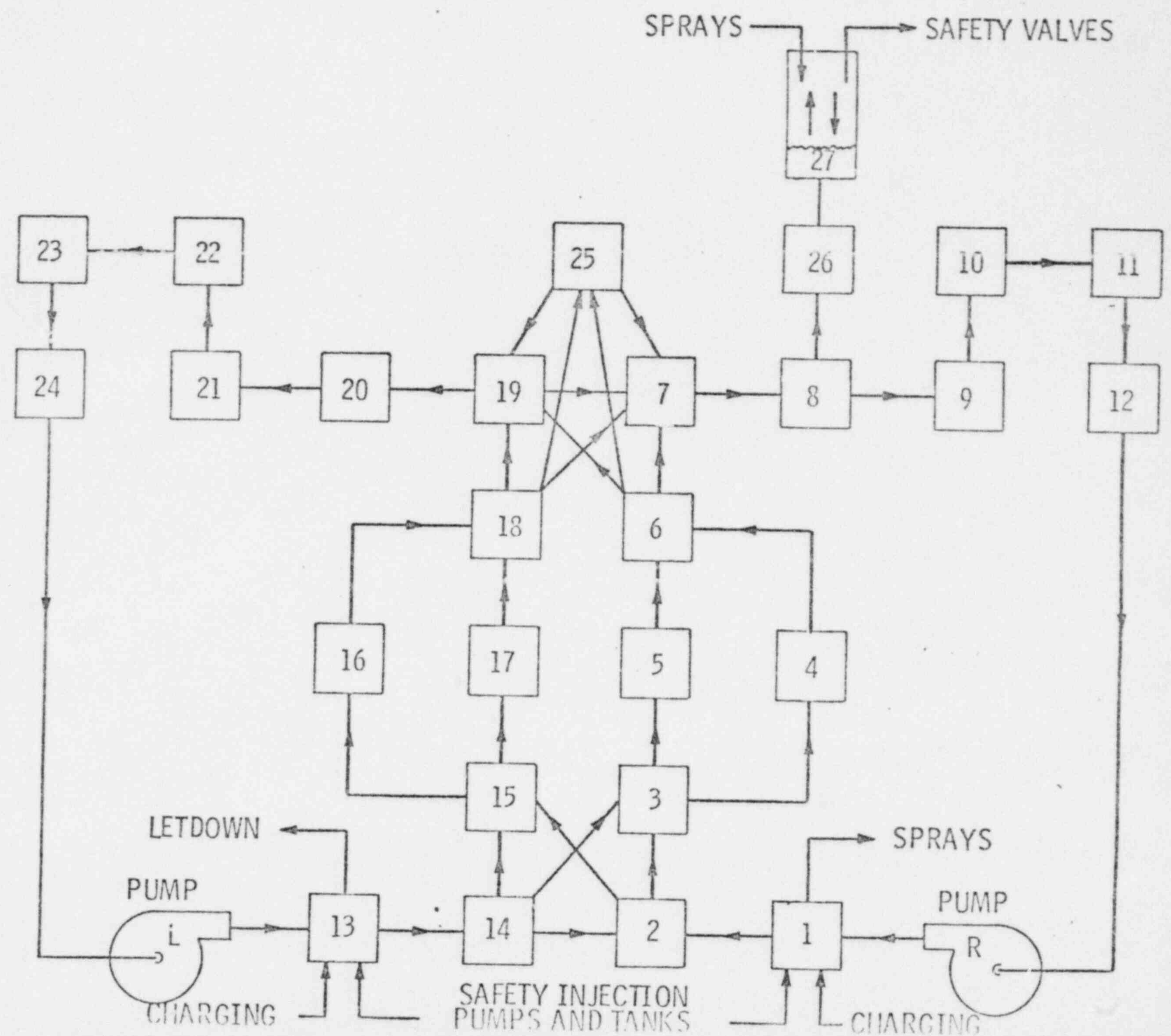
SCHEMATIC DIAGRAM OF PRIMARY COOLANT NODES

Figure 3A

BALTIMORE GAS & ELECTRIC CO.
Calvert Cliffs
Nuclear Power Plant

SCHEMATIC DIAGRAM OF PRIMARY COOLANT NODES

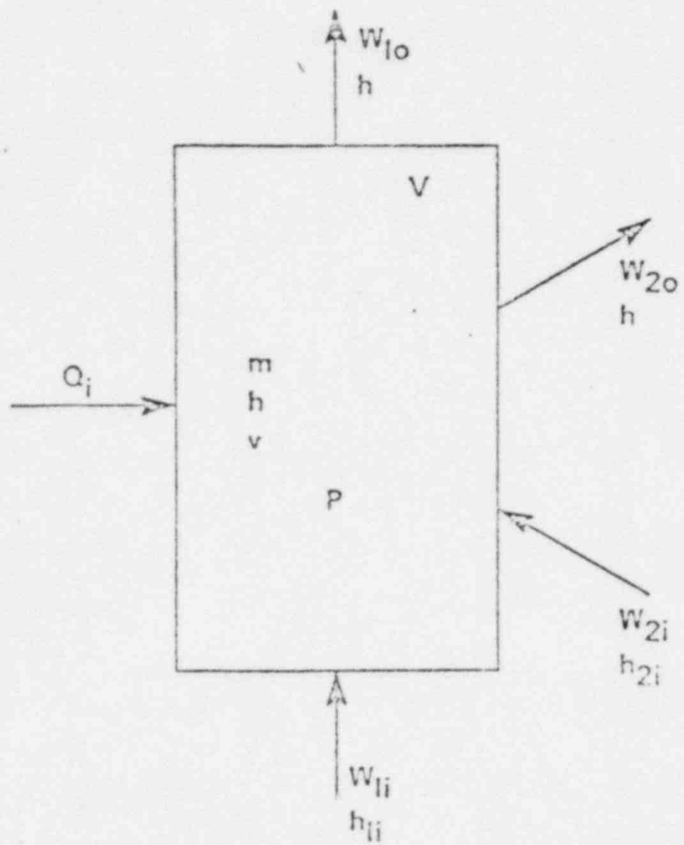
Figure 3A



POOR ORIGINAL

NODE	PHYSICAL DESCRIPTION
1	COLD LEG
2	UPSTREAM HALF OF INLET PLENUM (BEFORE FLOW MIXING)
3	DOWNSTREAM HALF OF INLET PLENUM (AFTER FLOW MIXING)
4	BYPASS FLOW
5	CORE
6	UPSTREAM HALF OF OUTLET PLENUM
7	DOWNSTREAM HALF OF OUTLET PLENUM
8	HOT LEG
9	STEAM GENERATOR INLET PLENUM
10	UPSTREAM HALF OF STEAM GENERATOR TUBES
11	DOWNSTREAM HALF OF STEAM GENERATOR TUBES
12	STEAM GENERATOR OUTLET PLENUM
13	SAME AS 1 IN OTHER STEAM GENERATOR LOOP
14	SAME AS 2 IN OTHER STEAM GENERATOR LOOP
15	SAME AS 3 IN OTHER STEAM GENERATOR LOOP
16	SAME AS 4 IN OTHER STEAM GENERATOR LOOP
17	SAME AS 5 IN OTHER STEAM GENERATOR LOOP
18	SAME AS 6 IN OTHER STEAM GENERATOR LOOP
19	SAME AS 7 IN OTHER STEAM GENERATOR LOOP
20	SAME AS 8 IN OTHER STEAM GENERATOR LOOP
21	SAME AS 9 IN OTHER STEAM GENERATOR LOOP
22	SAME AS 10 IN OTHER STEAM GENERATOR LOOP
23	SAME AS 11 IN OTHER STEAM GENERATOR LOOP
24	SAME AS 12 IN OTHER STEAM GENERATOR LOOP
25	REACTOR VESSEL CLOSURE HEAD
26	SURGE LINE
27	PRESSURIZER

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	PHYSICAL DESCRIPTION OF PRIMARY COOLANT NODES	Figure 3B



ENERGY EQUATION

$$\frac{d}{dt}(m\,h) - V\frac{dP}{dt} = Q + \sum_j W_{ji}\,h_{ji} - \sum_j W_{jo}\,h$$

MASS BALANCE

$$\frac{dm}{dt} = \sum_j W_{ji} - \sum_j W_{jo}$$

CONSTANT VOLUME

$$\frac{d\,(m\,v)}{dt} = 0.0$$

m = NODE TOTAL MASS

h = NODE AVERAGE ENTHALPY

V = NODE TOTAL VOLUME

v = NODE SPECIFIC VOLUME

Q = NODE HEAT RATE

W_{ji} = NODE INLET FLOWS, i DENOTES INLET

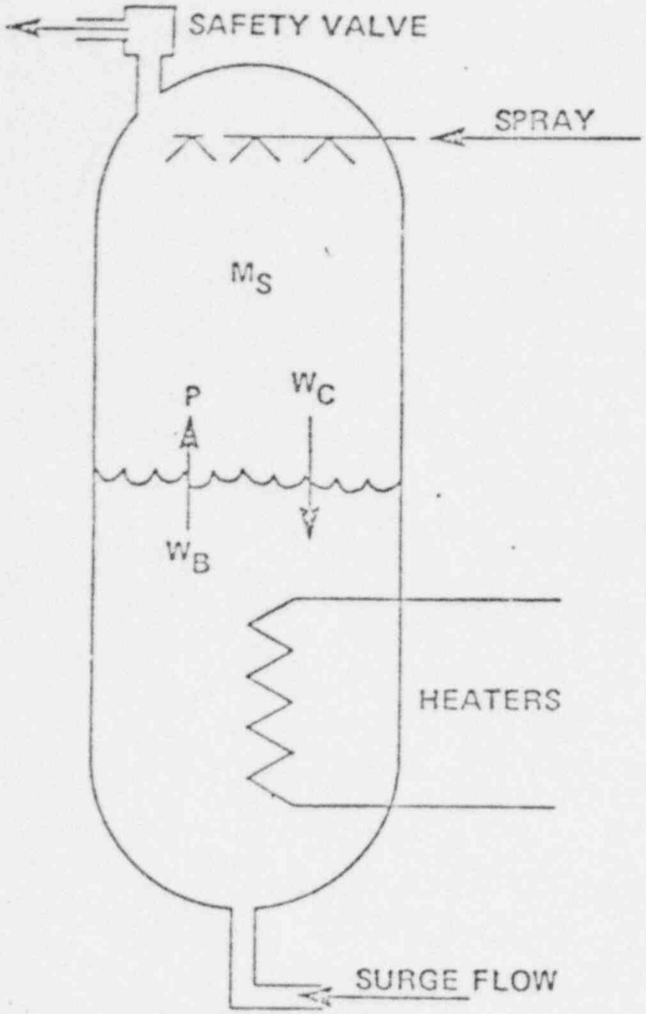
W_{jo} = NODE EXIT FLOWS, o DENOTES EXIT

h_{ji} = NODE INLET FLOW ENTHALPIES

P = COOLANT PRESSURE

BALTIMORE GAS AND ELECTRIC
CALVERT CLIFFS NUCLEAR POWER PLANT

GENERAL COOLANT NODE
EQUATIONS
FIGURE 4



PRESSURIZER

TWO VOLUMES (STEAM AND WATER)

CONSTANT VOLUME CONSTRAINT

$$M_S \, v_S + M_W \, v_W = V_P = \text{CONSTANT}$$

M_S = TOTAL MASS OF STEAM

v_S = SPECIFIC VOLUME OF STEAM

M_W = TOTAL MASS OF WATER

v_W = SPECIFIC VOLUME OF WATER

V_P = TOTAL VOLUME OF PRESSURIZER

INTERFACE FLOWS

W_B = BOIL NG FLOW RATE

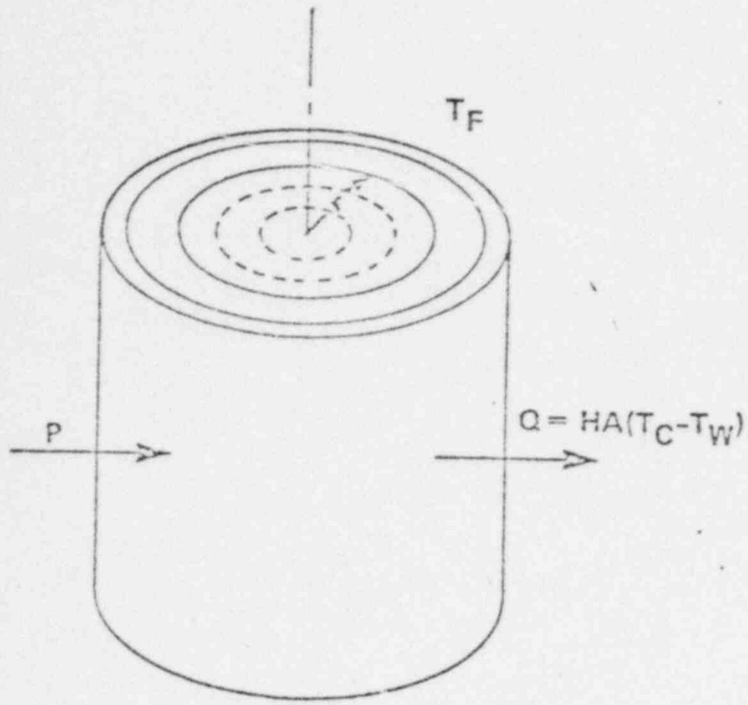
W_C = CONDENSATION FLOW RATE

CONDITIONS	WATER	STEAM
1	SUBCOOLED	SUPERHEATED
2	SATURATED	SUPERHEATED
3	SUBCOOLED	SATURATED STEAM
4	SATURATED	SATURATED STEAM
5	NONE	SUPERHEATED
6	NONE	SATURATED STEAM
7	SUBCOOLED	NONE
8	SATURATED	NONE

BALTIMORE GAS AND ELECTRIC
CALVERT CLIFFS NUCLEAR POWER PLANT

PRESSURIZER MODEL

FIGURE 5



HEAT FLOW EQUATION

$$pC\frac{\partial T_F}{\partial t} = K\left[\frac{\partial^2 T_F}{\partial r^2} + \frac{1}{r}\frac{\partial T_F}{\partial r}\right] + \frac{\partial K}{\partial r}\cdot\frac{\partial T_F}{\partial r} + q(r,P)$$

p	=	FUEL DENSITY
C	=	FUEL SPECIFIC HEAT
T	=	FUEL TEMPERATURE AT RADIUS r
r	=	RADIUS
K	=	FUEL THERMAL CONDUCTIVITY
t	=	TIME
q(r,P)	=	SPECIFIC HEAT GENERATION DEPENDING ON RADIUS AND THE REACTOR KINETICS POWER

SPACIAL FINITE DIFFERENCE METHODS REDUCE THE ABOVE EQUATION TO:

$$MC\frac{dT_{F_n}}{dt} = C_1 P + C_2\left(T_{F_{n-1}} - T_{F_n}\right) - C_3\left(T_{F_{n+1}} - T_{F_n}\right)$$

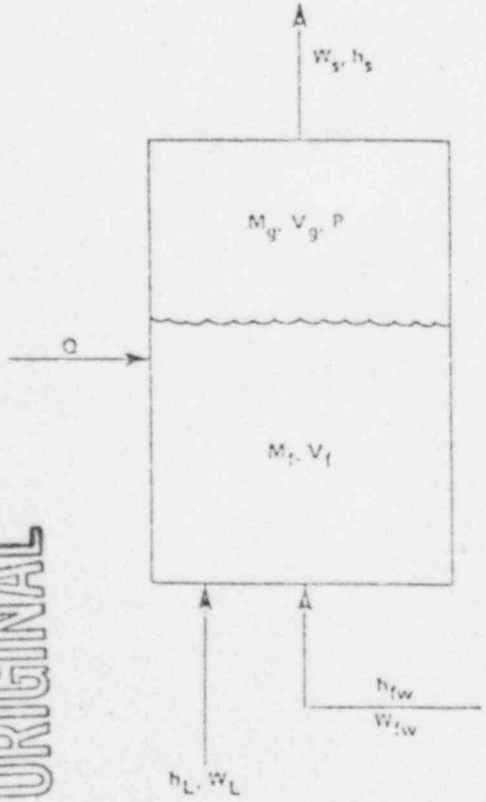
M	=	MASS OF THE NODE OF INTEREST
C_1	=	HEAT GENERATION FRACTION IN THE NODE
C_2	=	EFFECTIVE THERMAL CONDUCTANCE BETWEEN NODES
C_3	=	EFFECTIVE THERMAL CONDUCTANCE (HA FOR EDGE NODE)
T_{F_n}	=	AVERAGE NODE TEMPERATURE
$T_{F_{n-1}}$	=	AVERAGE NODE TEMPERATURE FOR NODE NEARER ℄
$T_{F_{n+1}}$	=	AVERAGE NODE TEMPERATURE FOR NODE AWAY FROM ℄

POOR ORIGINAL

BALTIMORE GAS AND ELECTRIC
CALVERT CLIFFS NUCLEAR POWER PLANT

FUEL ELEMENT TEMPERATURE
EQUATIONS

FIGURE 6



STEAM GENERATOR
SHELL SIDE

MASS BALANCE

$$\frac{dM_g}{dt} + \frac{dM_f}{dt} = W_{fw} + W_L - W_S$$

VOLUME BALANCE

$$\frac{d(M_g V_g)}{dt} + \frac{d(M_f V_f)}{dt} = \frac{dV}{dt} = 0$$

ENERGY BALANCE

$$\frac{d(M_f h_f)}{dt} + \frac{d(M_g h_g)}{dt} = Q + 144\ \frac{V}{J}\ \frac{dP}{dt} + W_{fw}h_{fw} + W_L h_L - W_s h_s$$

STATE: SATURATION

- P - STEAM GENERATOR SECONDARY PRESSURE
- M_g - TOTAL STEAM MASS IN STEAM GENERATOR
- V_g - SPECIFIC VOLUME OF STEAM
- M_f - TOTAL WATER MASS IN STEAM GENERATOR
- V_f - SPECIFIC VOLUME OF WATER
- W_{fw} - MASS FLOW RATE OF FEEDWATER
- h_{fw} - ENTHALPY OF FEEDWATER
- W_s - MASS FLOW RATE OF STEAM LEAVING STEAM GENERATOR
- h_s - ENTHALPY OF STEAM FLOW
- W_L - MASS FLOW RATE OF WATER LEAKING OUT OF STEAM GENERATOR TUBES
- h_L - ENTHALPY OF LEAK FLOW
- Q - HEAT RATE

BALTIMORE GAS AND ELECTRIC
CALVERT CLIFFS NUCLEAR POWER PLANT

CONSERVATION EQUATIONS FOR THE
STEAM GENERATOR SHELL SIDE

FIGURE 7

REFERENCES FOR QUESTION 3

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2. CENPD-107, Supplement 1, "ATWS Model Modifications to CESEC," C-E Proprietary Report (September, 1974).

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8. CENPD-107, Supplement 6, "CESEC - Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," C-E Non-Proprietary Report (August, 1978).