XN-NF-79-71 (NP)

REVISION 1

EXXON NUCLEAR PLANT TRANSIENT MODEL FOR JET PUMP BOILING WATER REACTORS

AUGUST 1980

RICHLAND, WA 99352

EXXON NUCLEAR COMPANY, Inc.

8009260467

XN-NF-79-71(NP)
Revision 1

Issue Date : 08/14/80

EXXON NUCLEAR PLANT TRANSIENT MODEL FOR JET PUMP BOILING WATER REACTORS

Prepared by

R. H. Kelley

Approved : TCK J N Morgan 5-29-80
J. N. Morgan, Manager
Licensing & Safety Engineering
Date

Approved : George A Sofer 5-29-80
G. A. Sofer, Manager
Nuclear Fuels Engineering
Date

/mb

EXXON NUCLEAR COMPANY, Inc.

NUCLEAR REGULATORY COMMISSION DISCLAIMER

IMPORTANT NOTICE REGARDING CONTENTS AND USE OF THIS DOCUMENT

PLEASE READ CAREFULLY

This technical report was derived through research and development programs sponsored by Exxon Nuclear Company, Inc. It is being submitted by Exxon Nuclear to the USNRC as part of a technical contribution to facilitate safety analyses by licensees of the USNRC which utilize Exxon Nuclear-fabricated reload fuel or other technical services provided by Exxon Nuclear for light water power reactors and it is true and correct to the best of Exxon Nuclear's knowledge, information, and belief. The information contained herein may be used by the USNRC in its review of this report, and by licensees or applicants before the USNRC which are customers of Exxon Nuclear in their demonstration of compliance with the USNRC's regulations.

Without derogating from the foregoing, neither Exxon Nuclear nor any person acting on its behalf:

A. Makes any warranty, express or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this document, or that the use of any information, apparatus, method, or process disclosed in this document will not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of, any information, apparatus, method, or process disclosed in this document.

TABLE OF CONTENTS

Section		Page
1.0	INTRODUCTION	1
2.0	SYSTEM MODELS	4
	2.1 CORE KINETICS MODEL	4
	2.2 CORE THERMAL-HYDRAULICS	5
	2.3 RECIRCULATION LOOP	8
	2.4 STEAM LINE MODEL	12
	2.5 CONTROL AND SAFETY SYSTEM MODELS	13
	2.6 CONTROL SYSTEMS	15
	2.7 HOT CHANNEL MODEL	17
3.0	NOMENCLATURE	29
4.0	CODE CAPABILITIES	39
	4.1 SOURCES OF INPUT	39
	4.2 VARIETY OF OUTPUTS	40
	4.3 APPLICATION TO NON-STANDARD DESIGNS	41
5.0	VERIFICATION PROGRAM	42
	5.1 PARAMETRIC BENCHMARKS	42
	5.2 SYSTEMS BENCHMARK	43
	5.3 INTEGRAL PLANT TEST BENCHMARK	43
6.0	REFERENCES	52

LIST OF TABLES

Table No.		Page
5.3.1	Transient Test Initial Conditions	48
5.3.2	PTSBWR3 Verification Benchmark (Turbine Trip #1) .	49
5.3.3	PTSBWR3 Verification Benchmark (Turbine Trip #2) .	50
5.3.4	PTSBWR3 Verification Benchmark (Turbine Trip #3) .	51

LIST OF FIGURES

Figure No.		Page
1.1	PTSBWR3 Schematic for Jet-Pump Plants	3
2.1	Fuel Nodalization	18
2.2	BWR Core Nodalization	19
2.3	Recirculating Flow Loop	20
2.4	Jet Pump Schematic	21
2.5	Recirculating Loop Enthalpy Time Delay	22
2.6	Valve Characteristics	23
2.7	BWR/4 Generation Feedwater Controller	24
2.8	BWR/6 Generation Feedwater Controller	25
2.9	Pressure Regulator Model	26
2.10	Recirculating Flow Controller (Part 1)	27
2.11	Recirculating Flow Controller (Part 2)	28
5.3.1	Peach Bottom TT3 Power Excursion	46
5.3.2	Peach Bottom TT3 Pressure Rise	47

1.0 INTRODUCTION

This report describes the PTSBWR3 code developed by Exxon Nuclear Company (ENC) for the simulation of jet pump BWRs during abnormal conditions. This model was derived from ENC's previously developed simulation code for non jet-pump BWRs[1]. The non-jet-pump models have been used in the analysis of several nuclear reactor plants[2,3,4,5] to provide licensing information in support of ENC fuel reloads. The major modifications included in PTSBWR3 to model jet pump BWRs are:

(a) A revision of the recirculation loop model to include the jet pump system and its interaction with other plant systems.

(b) An expansion of the control system models to include the control logic of current generation jet pump BWRs.

(c) An improved steam line model capable of predicting the wave phenomena noted during pressurization tests at Peach Bottom Unit 2 in April 1977. This improvement was made to ensure accurate prediction of the core void collapse and subsequent reactor power increase as a result of a turbine isolation event or other perturbance which results in a rapid pressure increase.

(d) A transient fuel/clad gap conduction model to account for variation in the fuel/clad gap heat transfer characteristics during abnormal operating conditions.

(e) Other minor modifications which provide consistency between the transient evaluation methodology and the upgraded methodology of other functional disciplines developed in the ENC Jet Pump BWR Program.

The mathematical models of the primary coolant pressure boundary are essentially the same as those developed for the non-jet pump PTSBWR2 code[1]. A synopsis of the principal models is provided in Section 2. Figure 1.1 provides a block diagram of the basic models incorporated in the PTSBWR3 code.

An alphabetical listing of all symbols and abbreviations used throughout the system descriptions is provided in Section 3.0.

Additional code characteristics, major input and output features of the code, and the current status of the ENC verification program for PTSBWR3 are discussed in Sections 4.0 and 5.0 of this report.

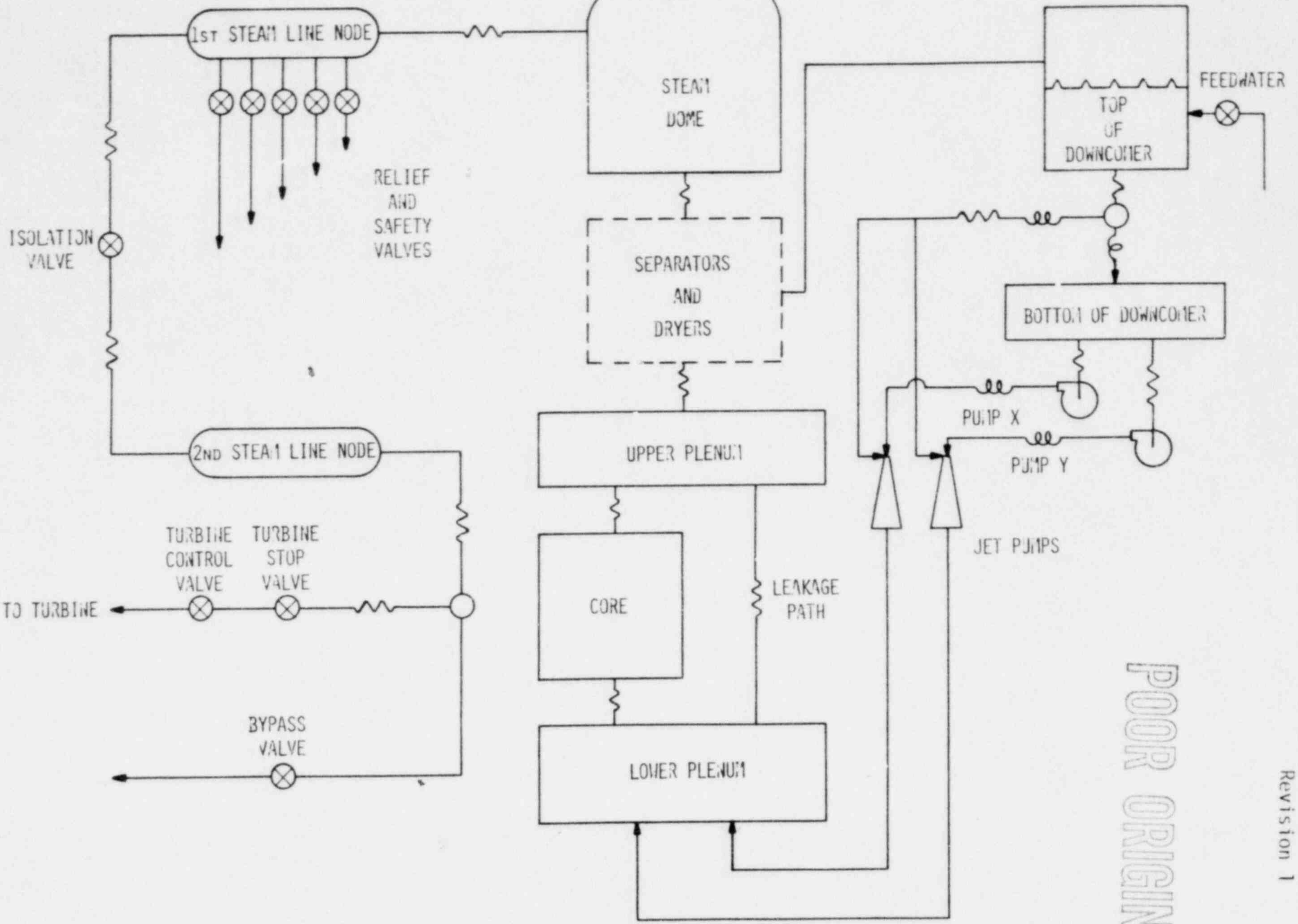


Figure 1.1 PTSBWR3 SCHEMATIC FOR JET-PUMP PLANTS

2.0 SYSTEM MODELS

2.1 CORE KINETICS MODEL

The variation of the average core neutron power is represented by a point kinetics model with six delayed neutron groups. The basic equation for power is:

$$\dot{P} = \frac{\left[\delta k\ (1 - \beta_{eff}) - 1\right]}{\ell / \beta_{eff}}\ P + \sum_{i=1}^{6} \lambda_i\ C_i \qquad (2.1)$$

where:

$$\beta_{eff} = \sum_{i=1}^{6} \beta_i$$

$$\frac{dC_i}{dt} = \frac{\beta_i\ P}{\ell} - \lambda_i\ C_i\ .$$

The time-dependent feedbacks and control reactivity are then represented as a sum of small changes from the unit value of k prior to the initiation of the transient.

The reactivity is given by:

$$k = 1 + \delta k_D + \delta k_V + \delta k_{CR} = 1 + \delta k \qquad (2.2)$$

where:

δk_D = Doppler reactivity change

δk_V = Moderator reactivity change (i.e., void)

δk_{CR} = Control rod scram reactivity change

The Doppler reactivity is assumed to be a function of fuel temperature and moderator void fraction, while the void reactivity is represented as a function of void only. Both variables' functional relationships are provided as code inputs. The changes in Doppler and void reactivity are calculated based on a comparison of current and initial conditions. The control reactivity due to control rod insertion is specified as tabular values at discrete elapsed time intervals following a scram initiation signal and a period of signal delay (which includes deenergization of the pilot scram valve solenoids).

2.2 CORE THERMAL HYDRAULICS

The basic models involved are fuel temperature, void fraction, and coolant enthalpy calculations. These models are identical to those previously reported(1) with the exception of those noted in Section 1.

2.2.1 Fuel Temperatures

The core average fuel temperature is calculated with a one-dimensional fuel pin thermal model. The fuel pin is divided into four equal volume fuel nodes and one cladding node as shown in Figure 2.1. The transient temperature for node T_f is given by the heat balance for the node:

(2.3)

Y_i, or the thermal admittance, represents the inverse of the resistance to heat flow across each node as determined from the overall heat transfer coefficient and the geometry of each node. The fuel and clad thermal properties and the fuel/clad gap and film convective heat transfer coefficients are available or provided as code input.

The core average fuel/clad gap heat transfer coefficient is dependent upon conditions determined from the exposure history of the fuel and the reactor power distribution. The initial value is determined from methodologies and models documented elsewhere[6]. During abnormal operating conditions, differential fuel and clad expansion/contraction characteristics and the gap gas conductivity temperature dependence would lead to changing gap resistance to heat flow. This variation of the gap heat transfer capability is calculated by models described in reference 7. Essentially, the fuel to clad gap size is updated based upon the reported thermal expansion characteristics of fuel[8] and zircaloy[9], and the gap heat transfer conductance is modified based upon the hot gap gas thermal conductivity variation:

where the above relationship was derived from data reported by Olander[10].

The heat generation rate for each node is derived from the results of the integration of the power from the kinetics model. Appropriate conversion to deposited energy and separation of fuel and moderator generated energy is provided. The total energy change for each megawatt change in neutron power is assumed to be 948.06 BTU/sec for each cubic foot of fuel.

The subsequent solution of the resulting five temperature equations of five unknowns (four fuel node temperatures and clad temperature) is used to provide an average fuel temperature for the kinetics model and a clad surface temperature for the void fraction model. The core average coolant temperature is calculated as the saturated steam temperature for the core pressure, computed elsewhere.

2.2.2 Void Fraction and Coolant Temperature

The core void calculation is based on the Zuber-Finley bulk void correlation with flow quality being derived from the Levy subcooled void correlation. The details and justification of these correlations are described elsewhere(11).

The core model is nodalized axially into regions of equal volumes. The net heat flow to the active coolant flow is the heat generated in the coolant plus heat transferred through the clad (see Fuel Temperature Model) less the heat gained by the core leakage. This heat flow is added to each node in accordance with the axial power distribution:

$$Q_{c,i} = Q_n P_{F,i} / n \qquad (2.4)$$

The energy balance for each node determines the exit enthalpy:

(2.5)

From which the thermal equilibrium quality is determined for each node boundary:

$$X_{c,i} = \frac{H_{c,i} - H_{f,cr}}{H_{fg,cr}} \tag{2.6}$$

The flow quality is determined from the Levy subcooled void correlation and the bulk void fraction from the Zuber-Findlay model.[6] The average core void fraction used in the kinetics model is the weighted average of the nodal boundary valves.

2.3 RECIRCULATION LOOP

The recirculation system is nodalized as shown in Figure 2.3. The nodes in the system are characterized as either compressible (two-phase or vapor) or incompressible. The core and upper plenum nodes are changed from compressible to incompressible nodes based upon the calculated quality. Compressible nodes are considered primarily as volumes connected by flow resistances. The incompressible nodes are considered a point in the circuit connected by flow resistances and enthalpy delays. The general equations for compressible and incompressible nodes have been developed elsewhere in detail[11] and will only be summarized here. The general equations are:

- o Conservation of Mass
- o Conservation of Energy
- o Conservation of Momentum
- o Equation of State

2.3.1 Conservation of Mass

The conservation of mass equation for each node is independent of compressibility or incompressibility:

$$\dot{M}_i = \sum_{i=1}^{M} W_{ini} - \sum_{i=1}^{n} W_{outi} \tag{2.7}$$

2.3.2 Conservation of Energy

The energy balance for a constant volume compressible node can be written as:

$$\dot{M}_i h_i + M_i \dot{h}_i - V_i \dot{P}_i = \sum_{i=1}^{m} W_{ini}\, h_{ini} - \sum_{i=1}^{n} W_{outi}\, h_{outi} \tag{2.8}$$

For an incompressible node w'thout energy addition, the energy balance is:

$$h_{i,j} + \frac{V_i}{W_i V_i} - \frac{dh_{ij}}{dt} = h_{i,j+1} \tag{2.9}$$

where the j subscript signifies time.

The treatment of energy addition is discussed in Section 2.2.2.

2.3.3 Conservation of Momentum

The general equation for momentum balance across any node is given as:

$$\dot{W}_i = \left(\frac{gc}{L/A} \cdot \Delta P_i \right) \Delta t \qquad (2.10)$$

with the incompressible model ignoring the acceleration pressure term.

2.3.4 Equation of State

The equations of state are represented as tabular relationships represented as saturated liquid and steam properties. For compressible nodes an additional useful relationship can be derived:

$$\dot{H}_i = \left(\frac{\partial H}{\partial p} \right)_v \dot{P}_i - \left(\frac{\partial H}{\partial v} \right)_p \frac{v_i}{M_i} \dot{M}_i \qquad (2.11)$$

2.3.5 Jet Pump Modeling

The solution to the above equations progresses assuming consistent initial conditions (Pressures, Flow rates, etc.) are provided as input. Special treatment of the downcomer accounts for its variable cross section and exchange of volume with the steam dome. Also, the dynamics of jet pumps, recirculating lines, and recirculating pump, though based primarily on the incompressible equations above, are of special interest here.

The jet-pump as modeled in PTSBWR3 is shown schematically in Figure 2.4. The fluid flow equations used to define the jet pump performance are given below:

- Jet-Pump Suction

At the jet-pump suction, the enthalpy and pressure must be defined. The enthalpy is determined from the time delay between the top and bottom of the downcomer.

The suction pressure is given by:

(2.12)

- Jet-Pump Flow Calculations

Neglecting inertial effects in the jet-pump suction, the jet pressure (P_{jx}) is given by:

(2.13)

With the suction and jet pressures known, a solution to a momentum balance around the drive loop yields the drive flow derivative:

(2.14)

The throat pressure (P_{thx}) is given by applying a momentum balance in the region of the throat (see Figure 2.4) :

(2.15)

The suction flow (W_{sx}) is defined by a mass balance on the jet pump:

$$W_{sx} = W_x - W_{dx}$$

With both the lower plenum pressure (P_{lp}) and the throat pressure (P_{thr}) known, a solution to the momentum balance on the diffuser gives the total flow in loop X (W_x):

(2.16)

Similar equations are employed to define the flow of loop Y.

For the downcomer flow path flow variable and flow reversible transport delays are calculated through a given volume with no mixing assumed except at flow junctions. Essentially, each volume is assigned a time constant which refelcts enthalpy delay through a given volume at a given flow rate. The logic diagram for the enthalpy delay is depicted in Figure 2.5.

2.4 STEAMLINE MODEL

The steam line model of PTSBWR3 solves the equations for the mass, energy and momentum balances. The basic equations for a compressible node are the mass, energy, momentum, and state equations developed in

Section 2. These equations were solved over two steamline nodes as represented in Figure 1.1. Rapid changes in the steam conditions require relatively smaller calculational time steps to adequately represent the wave propagation phenomena. This is generally solved without requiring the entire code to utilize the restrictive time division by integrating successively in the steamline model with a smaller time division than used elsewhere.

The calculation of flows at interfaces to the steam lines (turbine, valves) is discussed in Section 2.5. The flow at the junction between the steam lines include a characterization of the main steam isolation valve (MSIV) including valve position, closing time, and signal delay. The signal to close the MSIV is provided as a safety system setting input for the appropriate system parameter (vessel water level, pressure)

2.5 CONTROL AND SAFETY SYSTEM MODELS

2.5.1 Trip System Logic

The PTSBWR3 model employs logic to provide a signal indicating that a system variable has exceeded a specified safety setting. Once this signal is actuated, the delay between the trip signal and control rod-motion is simulated in time. Then, the control reactivity versus time for a particular case is developed from the reactivity versus distance relationship developed from neutronics calculations.

The following SCRAM signals are explicit to PTSBWR3:

<u>Signal</u>

High Neutron Flux

High Vessel Pressure

Low Water Level

Low, Low Water Level

Turbine Trip

Isolation Valve Trip

Generator Trip

2.5.2 <u>Valve Flow</u>

The relief, combination safety/relief, and or safety valves are actuated by pressure reaching their safety setpoints as specified as input. The basic model includes a finite delay time, opening time, and a closing time if applicable. Valve flow is given as:

$$W_v = N_v C_v \left[\frac{288\ g_c\ P_{sv}}{V_g \left\{P_{sv}\right\}}\right]^{1/2} \qquad (2.17)$$

where the flow coefficient (C_v) is characteristic of the valve flow capacity.

The isolation valve flow is calculated similiarly as discussed in Section 2.4.

The turbine stop and control valve are tripped manually by specifying closing rate, after which the flow is

$$W_{tb} = C_{w,tsv} \, Min(W_{tot}, W_{tb,max}) \qquad (2.18)$$

Figure 2.6 depicts the valve characteristics for each type.

2.6 CONTROL SYSTEMS

PTSBWR2 divides the control system of a Jet Pump BWR into three basic components:

- o Feedwater Flow Control
- o Pressure Regulator (to include Bypass)
- o Recirculating Coolant Flow Control

Each component is described below. Additionally, each controller's characteristics can be observed in a test mode, independent of the balance of the total system. In this way the performance of each control function can be compared with Plant Test Data.

2.6.1 Feedwater Controller

Functionally, the feedwater controller adjusts the feedwater flow rate to maintain vessel water level and the balance between steam flow and feed flow. The inputs to the controller are the steam flow, feedwater flow and water level (Z_{level}). The output is the adjusted feedwater flow.

The PTSBWR3 provides the option of simulating two generations of feedwater control for modern generation BWR's. These are shown as Laplace transfer functions in Figures 2.7 and 2.8 for BWR/4 and BWR/6 type plants, respectively. BWR/5 and hybrid plants must be evaluated separately to determine which model is applicable.

2.6.2 Pressure Regulator

Functionally, the pressure regulator adjusts turbine and bypass flow to maintain turbine throttle pressure at a desired setpoint. The control model is shown in Figure 2.9.

Essentially, the system produces an error signal by comparing sensed pressure with a pressure setpoint. This error signal must be conditioned by the lead/lag characteristics of the valves in addition to the controller transfer function to determine the new valve position and hence flow rate.

2.6.3 Recirculating Flow Controller

Automatic recirculating flow control is possible with many modern generation BWR's. The principle function of this control system is to maintain the proper flow required for the reactor power change needed to eliminate the difference between required and current load. Specific plant applications require verification that automatic flow control is a mode of operation.

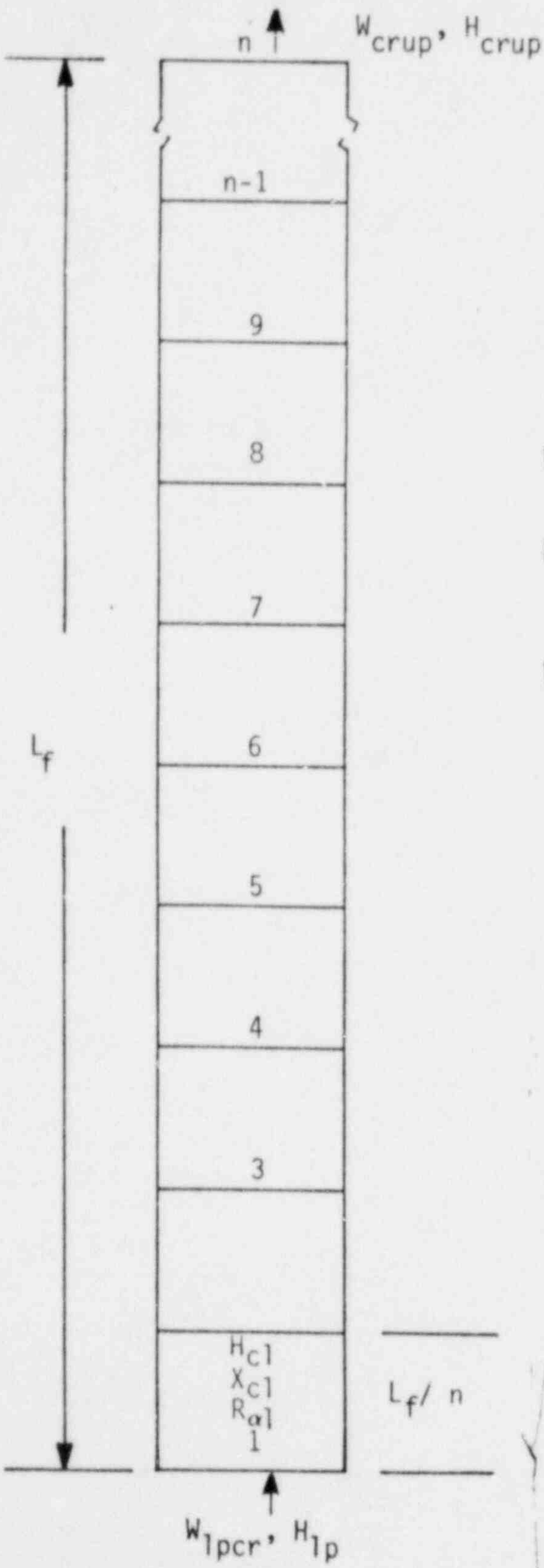
Figure 2.10 depicts the control block diagram for the recirculating flow system. PTSBWR3 allows independent testing of the recirculating flow controller to compare with plant performance or performance criteria.

Both automatic and manual flow control modes of operation may be simulated at the option of the user. Normally, manual flow adjustments by plant operators during transient operation cannot be predicted, but the evaluation of the impact of various sequences of operator actions can be performed provided that potential scenarios are available.

2.7 HOT CHANNEL MODEL

PTSBWR3 includes separate models to monitor the thermal performance of a single fuel assembly within the core and may be used in the determination of the limiting assembly thermal margin (MCPR). A detailed description of this model is presented in Reference (1).

FIGURE 2.1 FUEL NODALIZATION



W_{crup} = flow from core to upper plenum (lb/sec)

H_{crup} = enthalpy of (mixed) fluid flowing from core to upper plenum (Btu/lb)

W_{1pcr} = flow from lower plenum to core (lb/sec)

H_{1p} = enthalpy of water in lower plenum (Btu/lb)

L_f = length of active fuel (in.)

H_{c1} = enthalpy of (mixed) fluid at core node 1 (Btu/lb)

X_{c1} = quality at core node 1

$R_{\alpha 1}$ = void fraction at core node 1

FIGURE 2.2 BWR CORE NODALIZATION

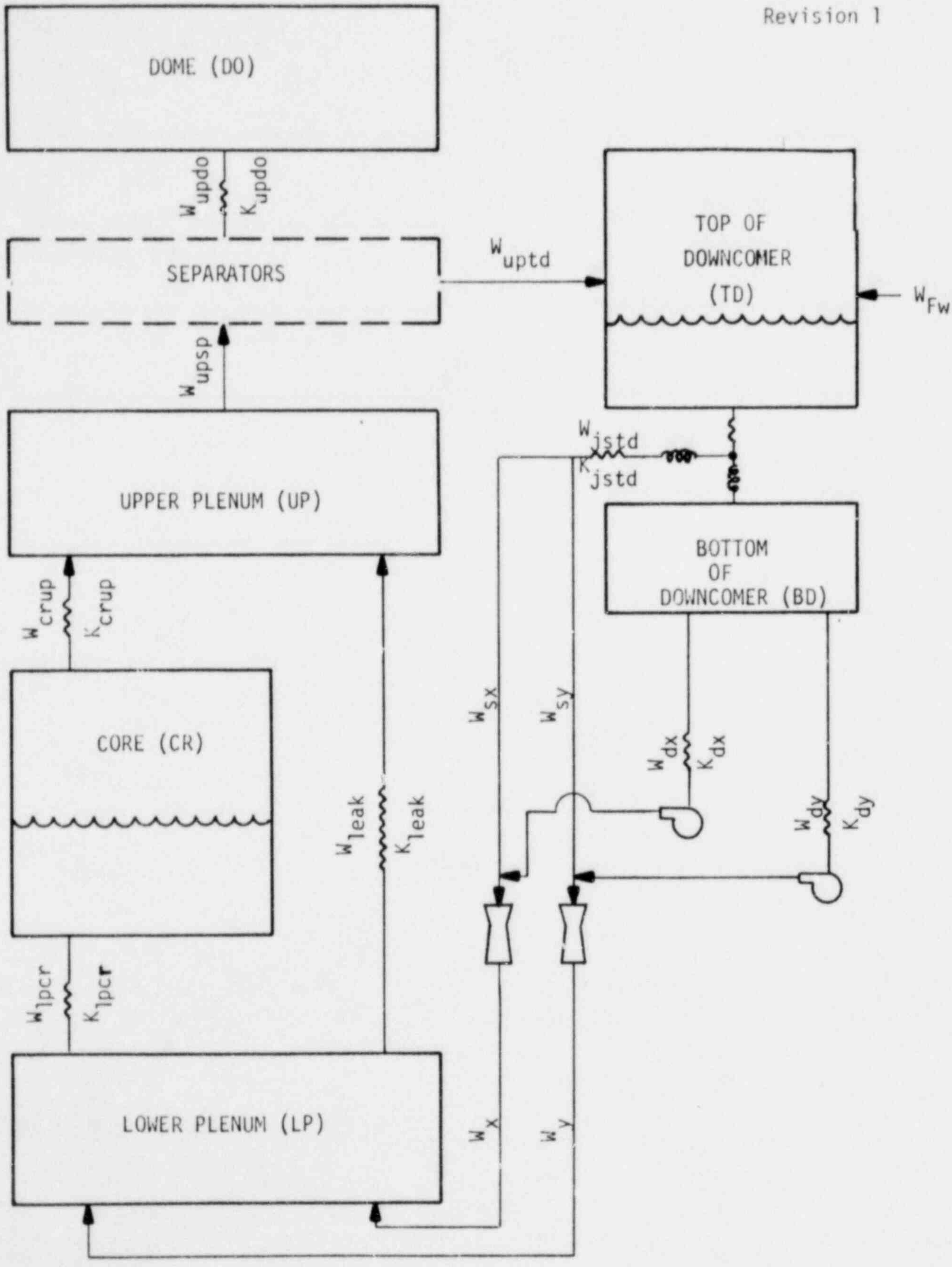


Figure 2.3 Recirculating Flow Loop

JET-PUMP SCHEMATIC

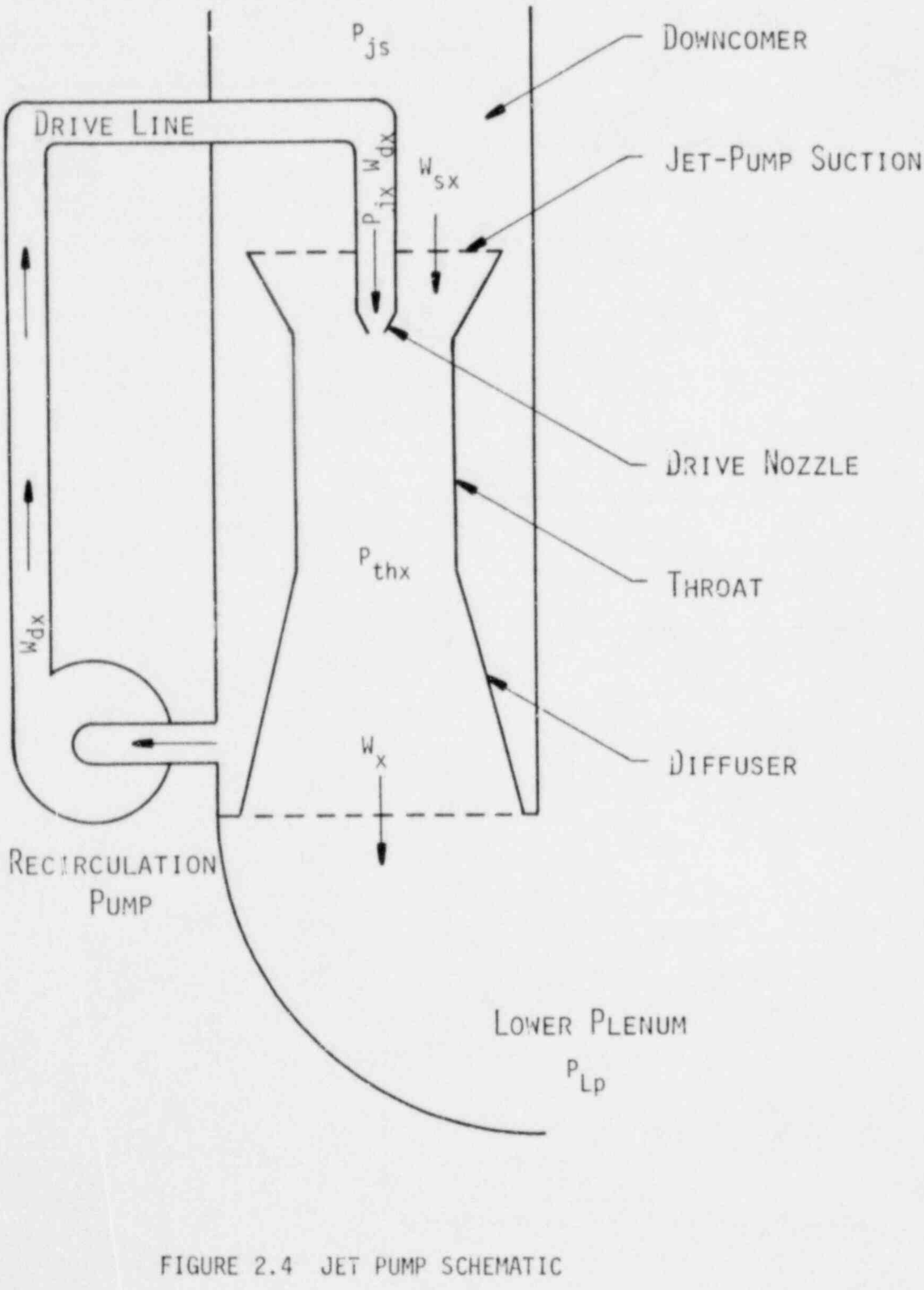


FIGURE 2.4 JET PUMP SCHEMATIC

Figure 2.5 Recirculating Loop Enthalpy Time Delay

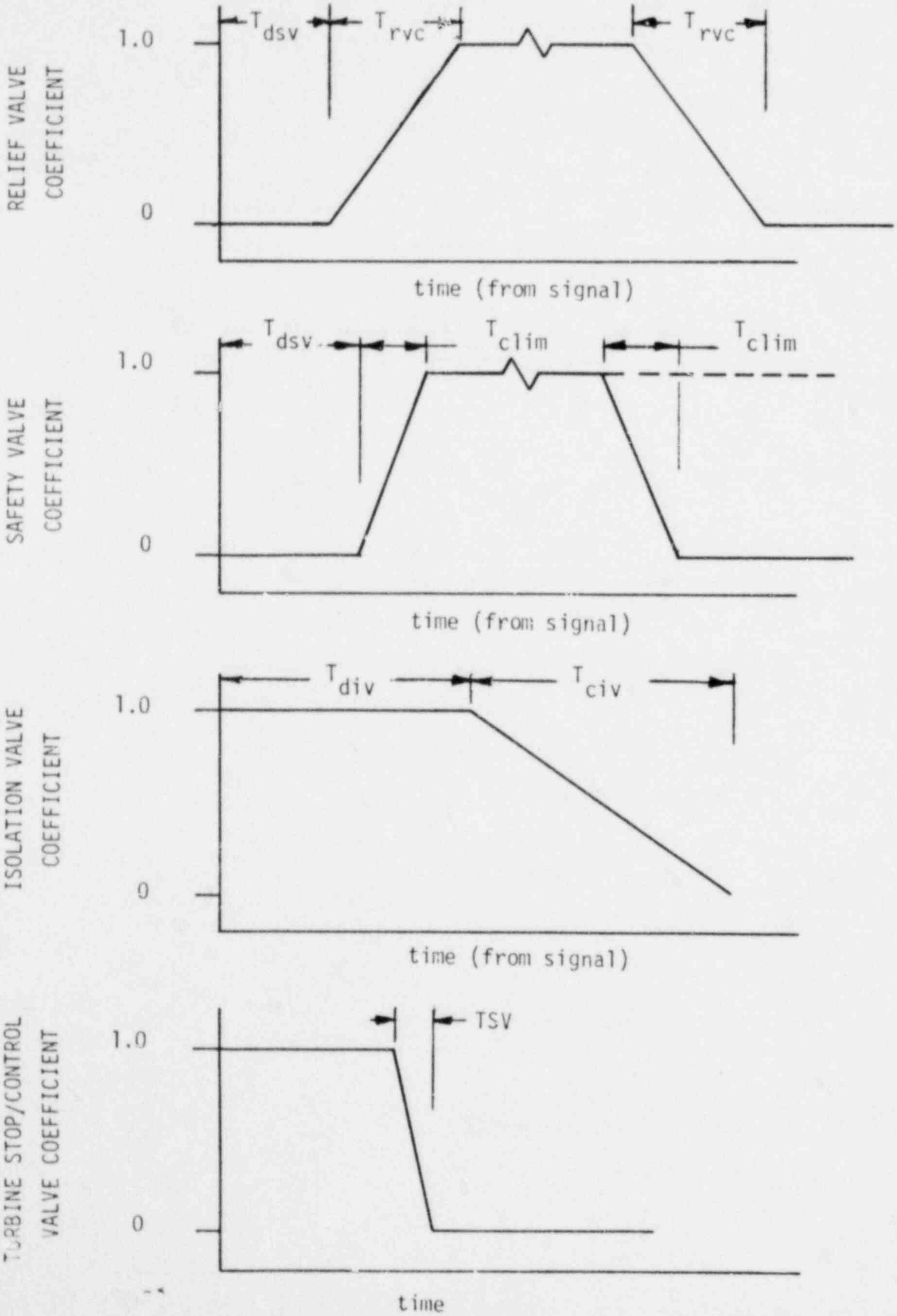


Figure 2.6 Valve Characteristics

Figure 2.7 BWR/4 Generation Feedwater Controller

Figure 2.8 BWR/6 Generation Feedwater Controller

Figure 2.9 Pressure Regulator Model

Figure 2.10 Recirculating Flow Controller (Part 1)

Figure 2.11 Recirculating Flow Controller (Part 2)

3.0 NOMENCLATURE

A_{LP}	- Lower Plenum Gain (1)
A_{noz}	- Cross sectional area of jet pump nozzle (ft^2)
A_{RFC1}	- Actuator Gain
A_{RFC2}	- Fluid Clutch Gain
A_{sc}	- Cross sectional area for jet pump suction (ft^2)
A_{th}	- Cross sectional area of jet pump throat (ft^2)
A_1	- Steam flow measurement gain
A_2	- Feedwater flow measurement gain
A_3	- Water level measurement gain
$C_{C1(C1,C2,C3,C4)}$	- Thermal capacity of Fuel nodes (BTU $°F^{-1}$) 1 through 4 and clad
$C_{f,i}$	- Thermal capacitance of fuel in node i (BTU $°F^{-1}$)
C_i	- Power equivalent of concentration of isotopes in ith delay group (Mw)
C_v	- Valve(s) Flow Coefficient
$C_{w,tsv}$	- Normalized flow coefficient reflecting position of turbine stop valve
FPUMP	- Function for generating pressure drop (psi) across pump from flow rate through pump and normalized pump speed
g_c	- Gravitational constant
H_i (or h_i)	- General term for enthalpy (BTU lb^{-1})
$\dot{H}_i$ (or $\dot{h}_i$)	- General term for time derivative of enthalpy (BTu $lb^{-1} sec^{-1}$)

$H_{c,i}$	- Enthalpy of nodal boundary i (BTU lb^{-1})
$H_{i,j}$	- Enthalpy in node i at time j (BTU/lbm)
H_{crup}	- Enthalpy of mixed fluid flowing from active core region to upper plenum (BTU lb^{-1})
$H_{f,cr}$	- Enthalpy of saturated water (liquid) in the core (BTU lb^{-1})
$H_{fg,cr}$	- Enthalpy change associated with change of phase from liquid to vapor (BTU lb^{-1}) at core pressure
h_{ini}	- Enthalpy of stream entering node i (BTU/lbm)
h_{out}	- Enthalpy of stream leaving node i (BTU/lbm)
HUPTD	- Upper Plenum to Top of Downcomer Enthalpy (BTU/lbm)
HFW	- Feedwater Enthalpy (BTU/lbm)
HSX, HSY	- Jet Pump Suction Enthalpy (BTU/lbm)
HBD	- Bottom of downcomer Enthalpy (BTU/lbm)
HDX, HDY	- Jet Pump Drive Flow Enthalpy (BTU/lbm
HX, HY	- Recirculating Loop Flow Enthalpy (to Lower Plenum) (BTU/lbm)
HLP	- Lower Plenum Enthalpy (BTU/lbm)
k	- Total reactivity of core (\$)
K_{crup}	- Combined loss coefficient from core to upper plenum (psia ft^{-3} sec^2 lb^{-1})
K_d	- Combined loss coefficient for drive (psia ft^{-3} sec^2 lb^{-1}
K_{dif}	- Combined loss coefficient of jet pump diffuser flow (psia ft^{-3} sec^2 lb^{-1})
K_{dx}	- Combined loss coefficient for the recirculating drive line flow path through pump X (psia ft^{-3} sec^2 lb^{-1})

K_{dy}	- Combined loss coefficient for the recirculating drive line flow path through pump Y (psia ft^{-3} sec^2 lb^{-1})
Kg	- Hot gap gas thermal conductivity (Btu hr^{-1} ft^{-2} °F^{-1})
Kgo	- Kg at time = o
K_{jsbd}	- Combined loss coefficient for jet pump suction flow (psia ft^{-3} sec^2 lb^{-1})
K_{jstd}	- Combined loss coefficient for recirculating flow from top of downcomer to jet pump suction flow (psia ft^{-3} sec^2 lb^{-1})
K_{leak}	- Combined loss coefficient for core leakage flow (psia ft^{-3} sec^2 lb^{-1})
K_{lpcr}	- Combined loss coefficient for flow from lower plenum to core (psia ft^{-3} sec^2 lb^{-1})
K_{LS}	- Feedwater gain
K_{PTS}	- Comparator
K_r	- Pressure Control Proportion gain
K_{rp}	- Control valve gain
K_{updo}	- Combined loss ciefficient from upper plenum to dome (psia ft^{-3} sec^2 lb^{-1})
K_2	- Feedwater valve gain
K_{FC1}	- Feedwater flow control proportional gain
K_{FC2}	- Feedwater flow control reset
K_{LC1}	- Level control proportional gain
K_{LC2}	- Level control reset
K_{LS1}	- Set Point Adjustor Gain
K_{RFC2}	- Master Flow Control Gain
K_{RFC3}	- Master Load Control Reset
K_{RFC4}	- Speed Control Gain
K_{RFC5}	- Speed Control Reset
L/A	- General term for inertance (ft^{-1})

LA_{bdjs}	- Length to area ratio from bottom of downcomer to jet pump suction (ft^{-1})
LA_{dif}	- Length to area ratio of the jet pump at diffuser (ft^{-1})
LA_{dl}	- Length to area ratio for driveline (ft^{-1})
LA_{jstd}	- Length to area ratio from downcomer to jet pump suction (ft^{-1})
LDEM	- Load Demand Error
LDERR	- Adjusted Load Demand Error
L_f	- Length of active core (in)
LVSPAN	- Level control span (feet)
M_i	- General term for mass (lbm) at node i
$\dot{M}_i$	- General term for timed derivative of mass (lbm/sec) at node i
n	- Total number of core axial nodal boundaries
NPUMPX	- Normalized Pump Speed (X)
NPUMPY	- Normalized Pump Speed (Y)
N_v	- Normalized position (0-1) of a valve
P	- Reactor neutron power level (Mw)
$\dot{P}$	- Time derivative of neutron power (Mw/sec)
P_i	- General term for pressure at node i (psia)
$\dot{P}_i$	- General term for pressure derivative at node i (psia/sec)
$\tilde{\dot{P}}_{cr}$	- Time derivative of pressure in core (psia sec^{-1})
$P_{f,i}$	- Axial power factor, node i
P_{jx}	- Jet pump pressure (psia)
P_{js}	- Jet pump suction pressure (psia)

P_{sv}	- Pressure at first steamline node (psia)
P_{td}	- Pressure at top of downcomer (psia)
P_{thx}	- Jet pump throat pressure (psia)
PTSENS	- Sensed Setpoint (psia)
PTT	- Sensed Pressure (psia)
PTTSP	- Pressure Setpoint (psia)
Q	- General term for energy (BTU/sec)
$Q_{c,i}$	- Heat flow to coolant in active core, node i (BTU sec^{-1})
Q_i	- Heat generation rate in fuel node i (BTU sec^{-1})
Q_N	- Net heat available for heating coolant in active core (BTU sec^{-1})
RATWFW	- Rated feedwater flow (lbm sec^{-1})
RATWMS	- Rated steam flow (lbm sec^{-1})
R_{ci}	- Inner radius of clad (in)
R_{co}	- Outer radius of clad (in)
R_f	- Fuel pellet radius (in)
R_{fo}	- R_f
R_{f1}	- Fuel radius, node 1 (in)
R_{f2}	- Fuel radius, node 2 (in)
R_{f3}	- Fuel radius, node 3 (in)
S	- Laplace Transform Variable
t	- Time (sec)

T_{ca}	- Core coolant average temperature (°F)
T_{civ}	- Isolation valve closing time (sec) after delay
T_{clim}	- Safety/relief valve opening/closing times (sec)
T_{div}	- Isolation valve closing delay (sec)
T_{dsv}	- Safety/relief valve activated on delay time (sec)
$T_{f,i}$	- Fuel temperature, node i (°F)
T_{gap}	- Hot fuel/clad gap temperature (°F)
T_{gapo}	- T_{gap} at time = o
T_{rvc}	- Relief valve closing time (sec)
T_{rvo}	- Relief valve opening time (sec)
T_{sv}	- Control/stop valve closing time (sec)
TF1(TF2,TF3,TF4,Tcl)	- Fuel temperatures of 4 fuel and clad nodes (°F)
V_i	- General term for Volume (ft^3) at node i
$v_{c,i}$	- Specific volume of fluid in core at nodal boudary i (Ft^3/lbm)
v_{cr}	- Specific volume of liquid in core from average core density (ft^3 lb^{-1})
V_{cr}	- Total volume of fluid in core (ft^3)
v_f	- Specific volume of saturated liquid at average conditions (ft^3 lb^{-1})
v_g	- Specific volume of saturated vapor at average conditins (ft^3 lb^{-1})
$v_{f,td}$	- Specific volume of saturated liquid at top of downcomer (ft^3 lb^{-1})
WBV	- Turbine bypass flow (lbm sec^{-1})

W_{crup}	- Flow from active core region to upper plenum (lb sec^{-1})
W_{dx}	- Jet pump drive for pump X (lbm/sec)
W_{dy}	- Jet pump drive for pump Y (lbm/sec)
W_{fw}	- Flow rate of feedwater (lb sec^{-1})
WFW	- Feedwater flow (lbm sec^{-1})
W_{ini}	- Mass flow entering a node (lbm/sec)
$\dot{W}_i$	- Time derivative of mass flow rate (lbm sec^{-2}) of node i
W_i	- General term for mass flow rate at node i (lbm/sec)
W_{jstd}	- Flow rate from top of downcomer to jet pump suction (lbm sec^{-1})
W_{leak}	- Leakage flow through core (lb sec^{-1})
W_{lpcr}	- Flow of subcooled water from lower plenum to active core (lb sec^{-1})
WIV	- Isolation valve steam flow (lbm/sec)
W_{outi}	- Mass flow leaving a node (lbm/sec)
W_{rc}	- Recirculation rate into lower plenum (lb sec^{-1})
W_{sx}	- Jet pump suction flow for pump X (lbm sec^{-1})
W_{sy}	- Jet pump suction flow for pump Y (lbm sec^{-1})
W_{tb}	- Flow rate through trubine stop valve (lb sec^{-1})
WTB	- Turbine Steam Flow (lbm/sec)
WTB	- Turbine header flow (lbm/sec)
$W_{tb,max}$	- Maximum flow rate to turbine (lb sec^{-1})

WTOTMAX	- Maximum total steam flow (lbm/sec)
WTBMAX	- Maximum Turbine Steam Flow (lbm/sec)
W_{tot}	- Total flow rate demand from turbine pressure regulator model (lb sec^{-1})
W_{updo}	- Flow rate, upper plenum to dome (lb sec^{-1})
W_v	- General term for valve mass flow rate (lbm/sec)
W_{upsp}	- Flow from upper plenum to separator (lb sec^{-1})
W_{uptd}	- Flow to downcomer annulus from upper plenum (lb sec^{-1})
W_X	- Flow rate through X pump (lb sec^{-1})
W_Y	- Flow rate through Y pump (lb sec^{-1})
$X_{c,i}$	- Thermodynamic quality in core, node i
Y_i(also, Y1 through Y5)	- Thermal admittance of fuel in node i (BTU sec^{-1} °F^{-1})
Z_{max}	- Upper level control limit (feet)
Z_{min}	- Lower level control limit (feet)
ZLEVEL	- Vessel water level (feet)
ZLEVSP	- Water level setpoint (feet)
Z_{dif}	- Jet pump diffuser length (ft)
Z_{jstd}	- Distance between top of downcomer to jet pump suction (ft)
Z_{level}	- Water level to reference height at top of down comer (ft)

β_{eff}	- Effective delayed neutron fraction
β_i	- Delayed neutron fraction for ith delay group
δ_k	- Total reactivity change (\$)
δk_{CR}	- Control (scram) reactivity (also, k_c) (\$)
δk_D	- Doppler reactivity feedback (\$)
δk_V	- Void reactivity feedback (\$)
ΔP_i	- Total pressure drop (psid) across node i
λ_i	- Decay constant for isotopes in ith delay group (sec^{-1})
ℓ	- Prompt neutron lifetime (sec)
$\frac{\partial H}{\partial P})_v$	- Partial derivative of enthalpy with respect to pressure at constant specific volume
$\frac{\partial H}{\partial V})_p$	- Partial derivative of enthalpy with respect to specific volume at constant pressure
τ_{ac}	- Downcomer Time Delay (sec)
τ_{dx}, τ_{dy}	- Drive Line Time Delay (sec)
τ_{sx}, τ_{sy}	- Suction Time Delay (sec)
τ_x, τ_y	- Jet Pump Time Delay (sec)
τ_F	- Feedwater flow measurement time constant (sec)
τ_{LP}	- Lower Plenum Time Constant (sec)
τ_{pcs}	- Set Point Adjuster Time (sec)
τ_{ps}	- Sensor Time Constant (sec)
τ_s	- Steam flow measurement time constant (sec)
τ_1	- Water level measurement time constant (sec)
τ_1	- Valve lead (sec)

τ_2	- Valve lag (sec)
τ_3	- Valve lag (sec)
τ_3	- Feedwater valve lead (sec)
τ_4	- Feedwater valve lag (sec)
τ_{rc1}	- Actuator Time Constant (sec)
τ_{rc2}	- Fluid Clutch Time Constant (sec)

4.0 CODE CAPABILITIES

PTSBWR3 can evaluate the general system response for a number of abnormal conditions. The important assumptions are:

1) The core responses to the transient generally as a whole thermally and hydraulically, i.e., there is not a great change in the relative energy and flow distributions within the core.

2) The rate change of flow is limited, i.e., PTSBWR3 cannot evaluate large breaks in the steam or recirculating lines.

As such, PTSBWR3 is adequate to evaluate incidents such as:

- Turbine or generator trip with or without condenser bypass.
- Recirculating Pump Trip or loss of pump power.
- Seizure of one recirculating pump
- Inadvertent valve(s) closures or openings.
- Loss of feedwater heating
- Malfunction of feedwater, recirculating, or steam control systems.

as well as other hypothetical situations.

4.1 SOURCES OF INPUT

The sources of input for the PTSBWR3 code are:

- Plant Component Information
- Fuel Specific Parameters
- Neutronics Parameters
- Initial Operating Conditions

The bulk of input by quantity comes directly from plant drawings and functional descriptive documents. These inputs include areas, volumes, rated conditions, design specifications, lengths, radii, and other geometric data.

The Plant Technical Specifications generally provide the principle reference for safety system settings and minimum performance for safety systems (safety/relief valves, bypass, etc.).

Most input parameters concerning the core are based on the specific nature of the fuel type(s) resident in the core. A physical description of the fuel would include fuel pellet diameter, clad dimensions, active fuel length, and design power distribution.

The balance of parameters are based on calculations such as the kinetic characteristics discussed in Section 2.1. Also important as input are the assumed initial nodal boundary mass flow rates and nodal pressures. The code automatically determines the nodal hydraulic characteristics and the remainder of the heat balance terms (Feedwater enthalpy, etc.)

4.2 VARIETY OF OUTPUTS

The PTSBWR3 code determines the following:

- Critical parameters (lower plenum pressure, power, fuel rod heat flux, reactor coolant flow, and coolant enthalpy) from which the margins of plant safety can be derived.
- Systems performances (valve flows, value positions, pressures, and timing) to describe anticipated behavior.

- Heat balance parameters (steam flow, feedwater flow, feedwater enthalpy, steam enthalpy) to include segmented heat balances (steamlines, steam dome, plenums, etc.).
- Parameter derivatives which, in conjunction with heat balance parameters verify the initial conditions for transient simulation. This precludes an undesired transient feedback not attributable to the intended simulated incident.

Specifically, PTSBWR3 includes an option to exercise control system models independent from the balance of code models. This allows verification and differentiation of control system performance. Additionally the optimization of control system settings for transient protection can be achieved.

PTSBWR3 employs a plotting routine to graphically display selected parameters for convenience of interpretation of results.

4.3 APPLICATION TO NON-STANDARD DESIGNS

It is recognized that, while many similiarities exist between Jet Pump BWR's, flexibility must be maintained to account for changes in design philosophy of reactors from one generation to another. Among others, the PTSBWR3 code has accounted for differences involving:

- Control System Designs
- Pressure Relief and/or Safety Valve Arrangements.
- Levels of Safety System Performance
- Safety Systems (HPSI, Emergency Condensers) available.

Similiarly, future design changes or backfitting can be included.

5.0 VERIFICATION PROGRAM

The verification of PTSBWR3 includes comparison of the individual subprograms and methods against available data and theory, as well as comparison of the overall code results against reactor plant transient measurements. The PTSBWR core hydraulic and fuel temperature models are consistent with ENC's standard methodologies(13,14). Special features of PTSBWR3 allow individual plant system responses (control systems etc.) to be compared to plant tests. Finally, the integral response of all interacting systems as calculated by PTSBWR3 can be compared to actual plant tests.

Plant measurement data is normally available in plant startup report documentation or open literature experimental results. Documentation of integral tests, such as Peach Bottom pressurization tests (April, 1977) are also available in the open literature. ENC conducts hydraulic test programs on critical fuel parameters for which comparisons with code predictions are possible.

5.1 PARAMETRIC BENCHMARKS

As PTSBWR3 models the plant coolant pressure boundary, the calculation of fuel and core coolant conditions provide numerous parameters for benchmarking. Other codes(13,14) used in analyzing ENC fuel exist which solve similar problems of smaller scope but in greater detail. The verification of the following parameters can be accomplished in this manner:

- Core Average Void Fraction
- Core Average Fuel Temperature

- Core Average Exit Quality and Void Fraction
- Upper Plenum Average Enthalpy (mixed)

Results of these comparisons indicate that the PTSBWR3 calculational models perform well in predicting nodal average parameter values.

5.2 SYSTEMS BENCHMARK

PTSBWR3 provides an option to separate the control systems (feedwater flow, recirculating coolant flow, and pressure regulation) from their interaction with the balance of the code models. Input functions can be supplied to each control model and the output observed and compared to plant data such as that acquired from plant startup tests. In this manner, the controller performance simulated can be benchmarked against actual plant performance.

The performance of other systems are also monitored by the code. Included are individual valve flows which can be compared to vendor or plant test measurements. The calculated valve performance is compared to verify the valve model.

5.3 INTEGRAL PLANT TEST BENCHMARK

During April 1977, a series of special turbine trip tests were performed at the Peach Bottom Unit 2 (BWR/4 Jet Pump) for plant performance and model qualification data. A more critical test of plant performance and best data for qualification of analytical methods is attained when nuclear power plants are operated at or near design basis conditions. These series of special tests were planned and conducted by Philadelphia Electric, General Electric and EPRI. The special tests consisted of three turbine trip tests near and less than full core flow and varying power levels up to 69-percent of rated power. Special data acquisition and instrumentation were designed to make these tests provide the most accurate measurements possible. The details of the test is given in EPRI Report #NP-564, (June 1978) titled "Transient and Stability Tests at Peach Bottom Atomic Power Station Unit 2 at End of Cycle 2."

5.3.1 PTSBWR3 Input (Peach Bottom Unit 2)

The bases for plant specific input for Peach Bottom Unit 2 are EPRI Report #NP-563[15] and NP-564[16]. Table 6.3.1 depicts critical initial values determined by the code as compared to the reported test values for test number 3 (TT3). Good agreement was also noted for test TT1 and TT2.

5.3.2 Results

All three pressurization tests were simulated with PTSBWR3 to provide a verification of the jet pump, steamline, and other model changes to PTSBWR2. A summary of comparisons of PTSBWR3 code predictions of TT1, TT2 and TT3 results are given in Tables 6.3.2, 6.3.3 and 6.3.4. Figures 6.3.1 and 6.3.2 give a more detailed comparison of predicted power rise and vessel pressure variation for the TT3 case.

All these results show favorable comparison between predicted and measured results. The PTSBWR3 code will be verified against other plant transient tests as the test conditions and results are available.

COMPARISONS OF PREDICTED REACTOR POWER AGAINST TEST DATA

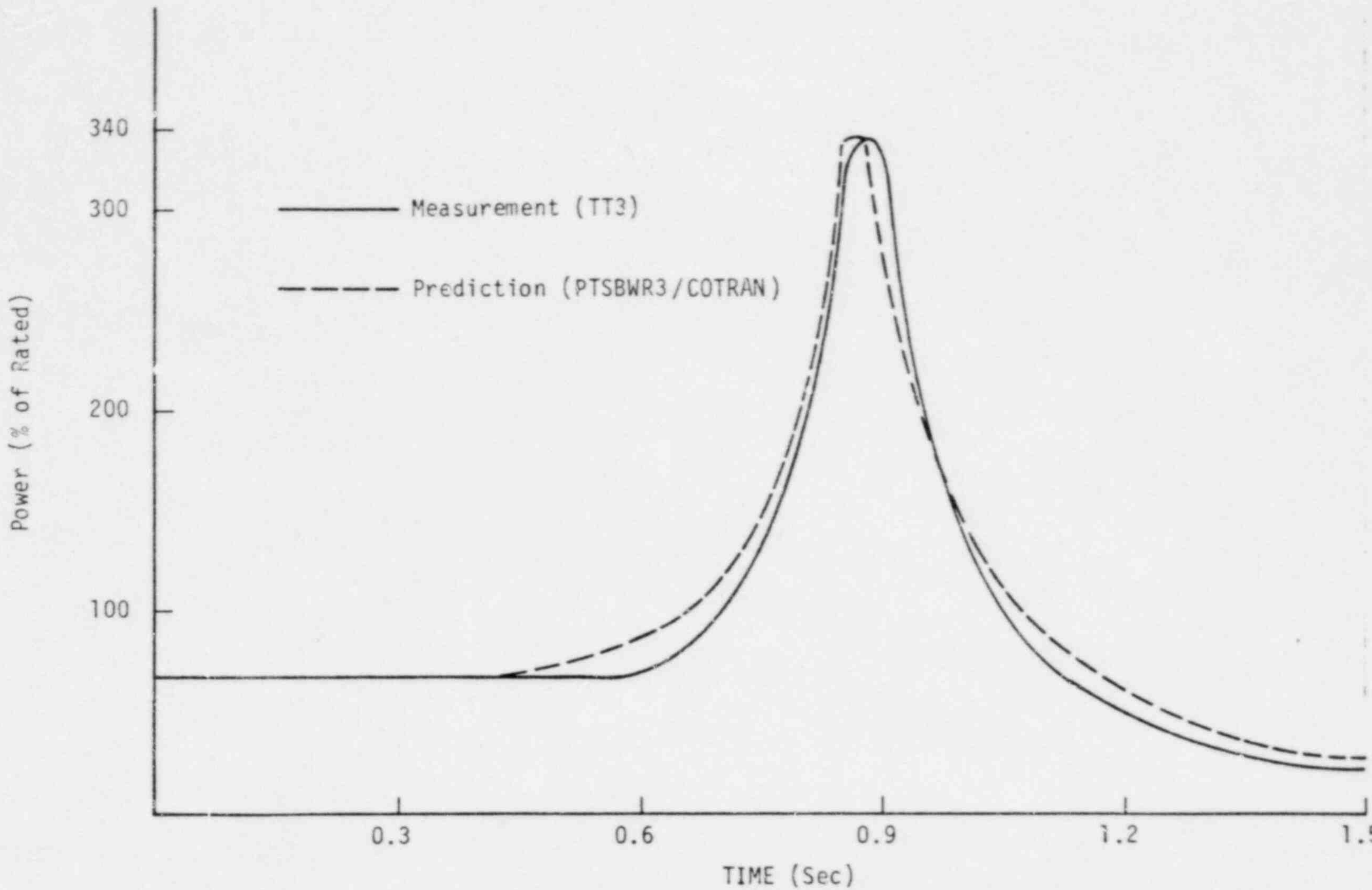


Figure 5.3.1 Peach Bottom TT3 Power Excursion

COMPARISON OF REACTOR PRESSURE
PREDICTION WITH TEST DATA
(PEACH BOTTOM BENCHMARKING)

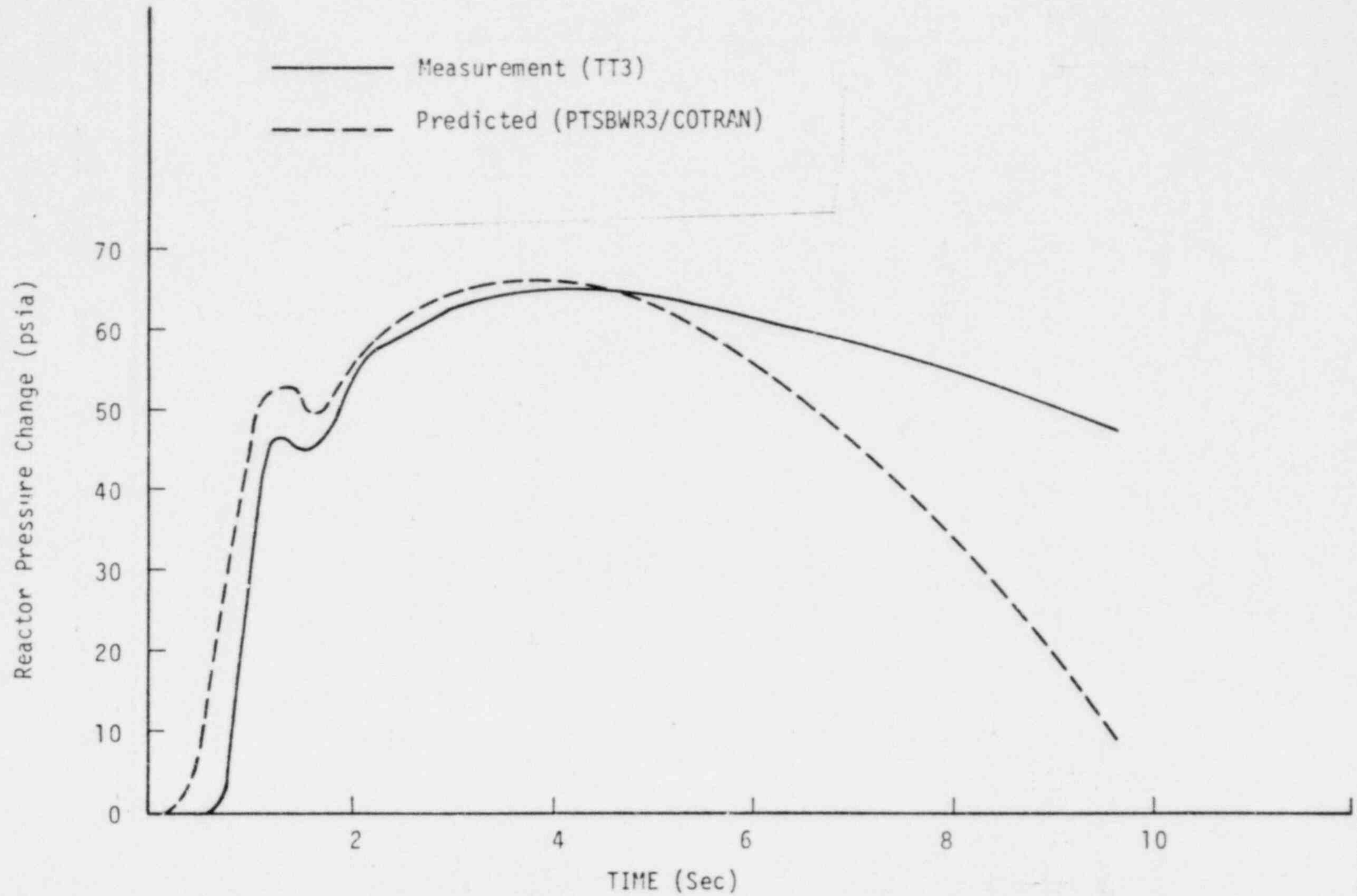


Figure 5.3.2 Peach Bottom TT3 Pressure Rise

TABLE 5.3.1

TRANSIENT TEST INITIAL CONDITIONS

Item	PTSBWR3	Test TT3
Reactor Power (MW)	2275	2275
Total Recirculating Flow (Mlb/hr)	101.9	101.9
Steam Dome Pressure (psia)	992.8	986.6
Upper Plenum Pressure (psia)	1005	993.0
Core Pressure (psig)	1014.4	1005.0
Lower Plenum Pressure (psig)	1026.3	Not Given
Turbine Emission Pressure (psig)	970.3	970.0
Core Inlet Enthalpy (Btu/lbm)	521.92	523.6
FW Enthalpy (Btu/lbm)	297.03	Not Given
FW Flow (Mlb/hr)	8.86	8.86
Steam Flow (Mlb/hr)	8.86	Not Given
Core Leakage (Lbm/sec)	2293	Not Given
Power Trip (MWt)	2535.6	2535.6
Bypass Valve Capacity (Lbm/hr)	976	Not Given

TABLE 5.3.2

PTSBWR3 VERIFICATION BENCHMARK
(Turbine Trip #1)

Item	Reported	Predicted
Initial Core Power (MWt),	1562	1562
Initial Core Flow (10^6 lbm/hr)	101.3	101.3
Initial Core Pressure (psia)	1005	1005
Initial Core Inlet Enthalpy (Btu/lbm)	528.4	526.3
Peak Average Power Rise (% of rated)	239	230
Maximum Core Pressure (Δpsia)	37.3	35.4
Maximum Dome Pressure Rise (Δpsia)	39.7	31.7
Maximum Change in Reactor Water Level (in)*	-28	-26.1

*Wide range water level measurement.

TABLE 5.3.3

PTSBWR3 VERIFICATION BENCHMARK
(Turbine Trip #2)

Item	Reported	Predicted
Initial Core Power (MWt)	2030	2030
Initial Core Flow (10^6 lbm/hr)	82.9	82.9
Initial Core Pressure (psia)	995	995
Initial Core Enthalpy (Btu/lbm)	519.8	517.7
Peak Average Power (% of rated)	280	290
Maximum Core Pressure Rise (Δpsia)	53.2	58.2
Maximum Dome Pressure Rise (Δpsia)	61.7	60.8
Maximum Change in Reactor Water Level (in)*	-43	-35.4

*Wide range water level measurement.

TABLE 5.3.4

PTSBWR3 VERIFICATION BENCHMARK
(Turbine Trip #3)

Item	Reported	Predicted
Initial Core Power (MWt)	2275	2275
Initial Core Flow (10^6 lbm/hr)	101.9	101.9
Initial Core Pressure (psia)	1005	1005
Initial Core Inlet Enthalpy (Btu/lbm)	523.6	521.9
Peak Average Power (% of rated)	339	340
Maximum Core Pressure Rise (Δpsia)	79	71.8
Maximum Dome Pressure Rise (Δpsia)	74.4	76.7
Maximum Change in Reactor Water Level (in)*	-38	-30.6

*Wide range water level measurement.

6.0 REFERENCES

1) Kahn, J. D. and Foster, M. S., "PTSBWR2 Plant Transient Simulation Code for Boiling Water Reactors," XN-74-6, Rev. 4, March 1977.

2) Kahn, J. D., "Analysis of Turbine Trip Without Bypass Transient for KRB," XN-NF-77-17, Rev. 1, July 1977.

3) Kahn, J. D., "Plant Transient Analysis for the Humboldt Bay Nuclear Power Plant," XN-76-10, Rev. 1, December 15, 1975.

4) Kahn, J. D., "Additional Plant Transient Analyses of the Oyster Creek BWR with Exxon Nuclear Fuel Assemblies," XN-75-41, September 1975.

5) Kelley, R. H., "Plant Transient Analysis for the Dresden 1 Reactor," XN-NF-77-46 (+ Supplement 1), October 1977.

6) Galbraith, K. P., "GAPEX: A Computer Program for Predicting Pellet-to-Cladding Heat Transfer Coefficients," XN-73-25, August 13, 1973.

7) Steves, L. H., et al., "HUXY: A Generalized Multirod Heatup Code," XN-CC-33(A), Rev. 1, November 1975.

8) Conway, J. B., et al., "The Thermal Expansion and Heat Capacity of UO_2 to 2200°C," Transactions of the American Nuclear Society, Volume 6, June 1963.

9) Scott, D. B., "Physical and Mechanical Properties of Zircaloy-2 and Zircaloy-4," WCAP-3629-41, May 1965.

10) Olander, D. R., "Fundamental Aspects of Nuclear Reactor Fuel Elements," U. S. Energy Research and Development Administration, TID-26711-P1 (1976).

11) Patten, T. W., "Methodology for Calculation of Pressure Drop in Power Fuel Assemblies," XN-NF-79-59, October 1979.

12) "PTSBWR-Plant Transient Simulation Code for Boiling Water Reactors," XN-74-6, Rev. 0, March 31, 1974.

13) Galbraith, K. P., "GAPEX: A Computer Code for Predicting Pellet-to-Clad Heat Transfer Coefficients," XN-75-24, August 31, 1975.

14) Patten, T. W., "XCOBRA Code User's Manual," XN-NF-CC-43, July 1979.

15) Larsen, N. H., "Core Design and Operating Data for Cycle 1 and 2 of Peach Bottom 2," EPRI-NP-563, June 1978.

16) Carmichael, L. A. and Niemi, R. O., "Transients and Stability Tests at Peach Bottom Atomic Power Station Unit 2 at End of Cycle 2," EPRI-NP-564, June 1978.

XN-NF-79-71(NP)

Revision 1

08/14/80

EXXON NUCLEAR PLANT TRANSIENT MODEL FOR JET PUMP BOILING WATER REACTORS

Distribution

GC Cooke
RH Kelley
GF Owsley

USNRC/GF Owsley (20)

Document Control (5)