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EXXON NUCLEAR COMPANY'S

PLANT TRANSIENT CODE FOR THE EVALUATION OF ABNORMAL TRANSIENTS FOR JET PUMP BOILING WATER REACTORS

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1.0 INTRODUCTION AND SUMMARY

The development program at Exxon Nuclear Company (ENC) of plant simulation models for jet pump boiling water reactors includes integration of several component submodels into a single calculational unit. The advantages gained by coupling complex codes include simplification of the evaluation process and better accounting of feedback between components of a system. This document describes the integration of ENC's plant system model, PTSBWR3, with ENC's detailed model for core kinetics and thermal hydraulics, COTRAN. Integration of these submodels is accomplished in ENC's COTRANSA (Coupled TRANsient System Analysis) computer code.

The starting point of this code is the plant transient simulation (PTS) model developed for boiling water reactors (BWR) and adapted for jet pump driven recirculation flow designs (PTSBWR3)[1]. This basic simulation model entails relatively simple core hydraulic and kinetic submodels suitable for evaluation of the reactor core without spatial detail. For many abnormal reactor incidents, the core generally responds with a substantial degree of uniformity. Therefore, such a model is often an adequate evaluation tool. However, the transient test data taken at Peach Bottom[2] and the attention in smaller accidents and transients since TMI have intensified the need for greater core information detail during the transient. This is accomplished in COTRANSA by applying boundary conditions derived from the PTS system model results at any point in the transient to the more detailed core model, COTRAN[3].

The detailed core response can then be used in a recalculation of dynamic core coefficients required for plant system response. This process is continued in an iterative fashion until convergence to a consistent solution. In cases where the detailed core calculations strongly influence the plant system response such as in a severe core pressurization transient, the COTRANSA code provides for an accurate combined calculation, with automated linkages between the submodels in a single computer code.

The COTRANSA computer code has been formed by replacement of the core hydraulics and kinetics submodels in PTSBWR3 with the spatially dependent equivalent submodels of COTRAN. This document reviews the principal submodels within COTRANSA and describes the interface between them. Detailed description of the submodels has been provided in References 1 and 4. Minor updates of the principal submodels as they are implemented in the COTRANSA code are also highlighted.

Documentation of verification and qualification work on COTRANSA completed to date is included in the final section of this report. More extensive qualification work has been performed on PTSBWR3 and COTRAN, separately, which are the principal components of COTRANSA. The consistency of the integral solutions of COTRANSA with results from PSBWR3 and COTRAN substantiate previous conclusions of consistency between these codes derived from execution of the component submodels. Further verification of this integral model will be performed in the future.

2.0 COTRANSA SUBMODELS

The submodels in the COTRANSA code are:

- Core Neutronics
- Core Thermal Hydraulics
- Recirculating Loop
- Steam Lines
- Safety System and Valves
- Control Systems

The first two submodels form the detailed core calculation and in essence are the same as in COTRAN. The last four submodels form the balance of plant model as in PTSBWR3. Each of the above submodels are discussed further below.

2.1 CORE NEUTRONICS

The basic components of the core neutronics submodel are:

- Space and time dependent neutron diffusion equation
- Core-Reflector interface
- Void and Doppler Feedback
- Radioactive Decay Heat

The detailed description of the neutronics submodel is contained in Reference 3. A summary account of each component of the core neutronics submodel is provided in the following subsections.

2.1.1 Space and Time Dependent Neutron Diffusion Equation

The fundamental equation solved in the neutronics submodel of COTRANSA is the one-group, space and time dependent neutron diffusion equation with no external sources:

$$\frac{1}{v(\vec{r},t)}\ \frac{d\phi(\vec{r},t)}{dt} = \left[(1-\beta)\nu\Sigma_f(\vec{r},t)-\Sigma_{A\to a}(\vec{r},t)\right]\phi(\vec{r},t) + \nabla\cdot D(\vec{r},t)\nabla\phi(\vec{r},t) \quad (2\text{-}1)$$

$$+ \sum_{\ell} \lambda_{\ell} C_{\ell}(\vec{r},t)$$

The detailed solution scheme is discussed in Reference 3.

2.1.2 Core-Reflector Interface

COTRANSA employs a simplified technique to account for the effects of the core-reflector interface. In finite difference form, the net leakage (L) into a node i can be expressed as:

$$L_i = \sum_j \frac{\sqrt{D_i\ D_j}\ A_{ij}}{d_{ij}} (\phi_j - \phi_i) \qquad (2\text{-}2)$$

The technique used in COTRANSA if node j is the reflector, is to assume that

$$\phi_j = 0$$

and to adjust the value of the reflector diffusion coefficient, D_j, until accurate flux distributions are obtained when compared to a more sophisticated static calculation.(3)

2.1.3 Void and Doppler Feedback

COTRANSA requires two sets of two group, macroscopic cross sections for each fuel type in the problem. These cross section sets describe the material in its entirely uncontrolled and completely controlled states [ ]. Linear interpolation is utilized to determine the cross sections for each fuel node at a given control density and void fraction. Doppler feedback is input to the two group cross sections as discussed in Reference 3.

2.1.4 Radioactive Decay Heat

During transient conditions the radioactive decay heat from fission products varies from the steady state value. To account for this effect, the fission products are grouped into eleven decay groups as shown in Table 2.1.1. The steady state effective concentration (G) of each group (j) is determined at each node (i) by:

$$G_{jo} = \frac{\gamma_j \quad P_i}{\lambda_j} \qquad (2\text{-}4)$$

A straightforward finite difference technique is utilized to update the nodal concentration of each fission product group by:

$$G_j\ (t + \Delta t) = (\gamma_j P_i - \lambda_j G_j(t))\ \Delta t + G_j(t) \qquad (2\text{-}5)$$

The nodal power density distribution is determined during the transient conditions then as:

$$P_i = 0.93\ \ (\kappa\ \Sigma_{fi}\ \phi_i) + \sum_j G_j(t)\ \lambda_j \qquad (2\text{-}6)$$

where $\sum_j G_{jo}\ \lambda_j = 0.07$ at initial conditions.

2.1.5 Core Neutronic Interfaces

The core neutronics calculation is coupled to the core thermal hydraulic model with a calculational hierarchy as follows:

(1) the core thermal-hydraulic submodel determines the axial distribution for void fraction and fuel temperature.

(2) the core neutronics submodel determines reactivity changes from step 1 and determines the axial flux profile and power density. This calculation provides the source term for the fuel conduction model. In addition, the core neutronics submodel determines the gamma heating in each hydraulic node based on a constant direct moderator heating fraction.

2.2 CORE THERMAL-HYDRAULICS

The basic components of the core thermal-hydraulics submodel are:

- o Fuel conductive heat transfer
- o One dimensional flow equations
- o Void distribution

The detailed description of the core thermal-hydraulics submodel is contained in Reference 3. A summary account of each component of the core thermal-hydraulic submodel is provided below.

2.2.1 Fuel Conductive Heat Transfer

The conductive heat transfer model used in COTRANSA calculates the internal temperature distribution of the fuel rod and the clad surface temperatures that are required for determining surface heat flux to the adjacent fluid channel. This method is detailed in Reference 4 and involves the Method of Weighted Residuals for the radial coordinate and finite differences in time for the axial coordinate. It includes an option for axial conduction, and accounts for temperature dependent fuel thermal conductivity.

The fundamental heat conduction equation for the fuel interior is:

$$\frac{\rho c \partial T}{\partial t} = \nabla \cdot \ (K \nabla T) + q''' \tag{2-7}$$

which is solved in cylindrical coordinates by the method of orthogonal collocation with the conductivity integral [5]. In this application, the radial positions are taken to be orthogonal polynomials as defined by Finlayson[6].

The time derivative term is evaluated by first forward difference approximation. The result is a heat conduction equation at each interior noda. position.

The boundary condition at the fuel surface is handled by a lumped resistance technique. The equation is:

$$\frac{-Ko}{R} \ \frac{\partial \theta}{\partial r} = H_g \ (T_N - T_{N+1}) \tag{2-8}$$

where the radial derivative is approximated in the same manner as in the fuel interior.

The transient energy balance for the lumped clad is:

$$\rho c \frac{\partial T_{N+1}}{\partial t} = \frac{Hg}{Yc}\left(\frac{r_N}{r_{N+1}}\right)\left(T_N - T_{N+1}\right) - \frac{Hs}{Yc}\left(T_{N+1} - \tilde{T}_F\right) + Kc\,\frac{\partial^2 \tilde{T}_{N+1}}{\partial x^2} \qquad (2\text{-}9)$$

which is solved implicitly in time with an explicit axial conduction term and with a truncated Taylor series approximation for the implicit temperature.

The detailed solution scheme is discussed in Reference 3. The source term (q''') for the fuel interior equation is provided by the neutronics submodel as described in Section 2.1. The calculation of clad heat flux is discussed in the following subsection.

2.2.2 One Dimensional Flow Equations

The mass, energy and momentum balance laws which form the basis of COTRANSA core hydraulics calculations are formulated in terms of an Eulerian control volume, V, which is bounded by a fixed surface A. This surface may include solid interfaces, such as a fuel rod or structural wall, and fluid boundaries; but all solid material is outside V and comprises the fuel thermal model in subsection 2.2.1. The fluid in V is a single component, two phase mixture of liquid and vapor in thermodynamic equilibrium.

When the definitions and assumptions detailed in Reference 3 are applied, the integral balance equations for mass, energy and momentum are reduced to:

Mass

$$\frac{\partial}{\partial t}\int_V \rho dV + \int_F \rho(\vec{u}\cdot\vec{n})dA = 0 \tag{2-10}$$

Energy

$$\frac{\partial}{\partial t}\int_V \rho h dV + \int_F \rho h(\vec{u}\cdot\vec{n})dA = -\int_F K(\nabla\vec{T}\cdot\vec{n})dA + \int_W H(T_W - T_F)dA \tag{2-11}$$

where

$\rho h = \rho i + p$ (The temporal derivative of pressure can be ignored for low-speed flow).

Momentum

$$\frac{\partial}{\partial t}\int_V \rho\vec{u}dV + \int_F \rho\vec{u}(\vec{u}\cdot\vec{n})dA = \int_V \rho\vec{g}dV - \int_F p\vec{n}dA + \int_F (\vec{\vec{\pi}}\cdot\vec{n})dA - \int_W p\vec{n}dA + \int_W (\vec{\vec{\pi}}\cdot\vec{n})dA \tag{2-12}$$

The implicit solution scheme, discussed in Reference 3, includes options for two-phase slip models, void-quality relations and two-phase friction multipliers.

The fuel is interfaced with the fluid thermal-hydraulics by means of a surface heat transfer correlation which uses Dittus-Boelter[7] for 1-phase and Thom[8] for 2-phase flows.

The COTRANSA solution scheme employs the reference pressure approach - that is to say, the local fluid density is assumed to be a function of the local enthalpy and a spatially uniform reference pressure. This assumption is valid since spatial pressure variations in problems of interest are small compared to the system pressure.

2.2.3 Void Distribution

COTRANSA allows the option of calculating nodal voids by homogeneous[9], constant slip[9], or Zuber-Findlay[10] correlations. Levy's subcooled quality correlation[10] is used as a basis for these calculations.

The core model is divided into up to 72 axial nodes, and the Zuber-Findlay, homogeneous or slip correlations may be applied within the implicit solution scheme.

2.2.4 Core Thermal Hydraulics Interfaces

The fuel temperature and coolant void information are important input to the neutronics submodel. The detailed core hydraulics calculation is coupled to the recirculation loop submodel with calculational hierarchies as follows:

- The recirculation loop submodel determines active core inlet flow rate and enthalpy and core exit pressure boundary conditions to the core hydraulic submodel.
- The core hydraulic submodel determines core exit flow rate and enthalpy and active core entrance pressure parameters to the recirculation loop submodel.
- The final boundary condition required of the core hydraulics submodel is the clad heat flux provided by the fuel conductive heat transfer calculation.

2.3 RECIRCULATION LOOP

The recirculation system is nodalized as shown in Figure 2.3.1 except that, as previously described, the core model is further subdivided. The nodes in the system (minus the core) are characterized as either compressible or incompressible. The upper plenum may be either, compressible or incompressible depending upon calculated quality. Compressible nodes are characterized primarily as volumes with inertial qualities and interconnected by flow resistances. The incompressible nodes are considered as a point in the circuit connected by flow resistances, enthalpy delays, and appropriate momentum losses. The constitutive equations for mass, energy, and momentum are summarized below for the recirculation system (minus core):

2.3.1 Conservation of Mass

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

$$\dot{M}_i = \sum_{i=1}^{M} W_{ini} - \sum_{i=1}^{n} W_{outi} \tag{2-13}$$

2.3.2 Conservation of Energy

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

$$\dot{M}_i h_i + M_i \dot{h}_i - V_i \dot{P}_i = \sum_{i=1}^{m} W_{ini}\, h_{ini} - \sum_{i=1}^{n} W_{outi}\, h_{outi} \tag{2-14}$$

For an incompressible node without energy addition, the energy balance is:

$$h_{i,j} + \frac{V_i}{W_i v_i} \cdot \frac{dh_{ij}}{dt} = h_{i,j+1} \qquad (2\text{-}15)$$

where the j subscript signifies time.

2.3.3 Conservation of Momentum

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

$$\dot{W}_i = \left(\frac{gc}{L/A} \cdot \Delta P_i \right) \Delta t \qquad (2\text{-}16)$$

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, the following relationship is also used.

$$\dot{H}_i = \left(\frac{\partial H}{\partial p} \right)_v \dot{P}_i - \left(\frac{\partial H}{\partial v} \right)_p \frac{v_i}{M_i} \dot{M}_i \qquad (2\text{-}17)$$

2.3.5 Core Interfacing

The system models solve for the active core inlet flow, inlet fluid enthalpy, and exit core pressure. Concurrently, the core model provides the system model with exit active core flow, exit fluid enthalpy, and core entrance pressure. Thus, the hydraulics of the loop are closed. The core model includes only the active fuel region so that other flow resistance and momentum factors associated with the core support plate, core orificing, bypass region, and other hardware is accounted for in the balance of the recirculation loop model. Thus, the modeling considers a point just at the boundary of the core inlet and accounts for flow length and effective flow area back to the previous nodal point, the lower plenum. Also, the flow resistance between the two nodes includes the pressure drop across the core orificing. Similar modeling of the upper plenum places its boundary at the top of the active fuel region. Definition of input parameters common to the PTSBWR3 and COTRANSA models are accordingly modified.

2.3.6 Jet Pump Modeling

The jet pumps are modeled in terms of mass, energy and momentum balance equations. Special consideration of the dynamic nature of the jet pumps in the downcomer region and the drive line system was highlighted in Reference 1 and is not repeated here. Also, the calculation of enthalpy transport is unchanged from the previously documented description of the PTSBWR3 model.

2.4 STEAM LINES

The steam line model of COTRANSA solves the equations for the mass, energy, and momentum balance of compressible nodes. The steam line volumes on either side of the main steam isolation valves may be subdivided into up to five equal volumes of equal momentum at the option of the user. This allows flexibility in modeling the steam lines. Past experience shows that the best comparison between calculations and measured data occurs when all nodes were approximately equal. The time rate of change in steam conditions can be large for some transients. This necessitates independent and smaller calculational time steps for the steam line calculational model to adequately track the propagation of hydraulic disturbances down the lines. The global time step for the balance of the COTRANSA calculation is sufficiently small to ensure good communication of information between the steam line and the vessel. 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 includes 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 SAFETY SYSTEMS AND VALVES

2.5.1 Safety Systems

The COTRANSA model employs logic to provide a parameter which indicates that a system variable had exceeded a specified safety setting. Once this occurs, the signal instrument delay is simulated in time before the safety action begins. Subsequently, the characteristic response of the safety system response is modeled such as valve stroke time or control rod position versus time.

The following safety systems are modeled:

Signal	Action
High neutron flux	Scram
High vessel pressure	Scram/Pump trip
Low water level	Scram
Low, Low water level	Vessel isolation
Turbine trip	Scram/Pump trip
Isolation valve trip	Scram
Generator trip	Scram/Pump trip
High steam line pressure	Open safety/Relief/Bypass valves
Reduction of steam pressure	Close safety/Relief/Bypass valves
Loss of offsite power	Pump trips, generator trip etc.

Options are available in COTRANSA to allow the inactivation of any safety system or the simulation of inadvertent actuation of any system as desired by the model user.

2.5.2 Valve Flow

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

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

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

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

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

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

Valve characteristics are equivalent to those documented previously[1].

2.6 CONTROL SYSTEMS

COTRANSA employs three automatic control systems designed to predict the boundary conditions of the model flow at the feedwater sparger, the turbine inlet nozzle and the recirculating pump speed. The three models are:

- o Feedwater Flow Control
- o Pressure Regulation (turbine control and bypass valve position)
- o Recirculating Flow Control

Detailed transfer functions for each system are shown in Reference 1. COTRANSA retains the capability of exercising each control system separate from the integrated model to allow verification of control response characteristics against test data.

Table 2.1.1 Radioactive Decay Constants

Group	γ_j	$\lambda_j(\text{sec}^{-1})$
1	.00299	1.772
2	.00825	.5774
3	.01550	6.743×10^{-2}
4	.01935	6.214×10^{-3}
5	.01165	4.739×10^{-4}
6	.00645	4.810×10^{-5}
7	.00231	5.344×10^{-6}
8	.00164	5.726×10^{-7}
9	.00085	1.036×10^{-7}
10	.00043	2.959×10^{-8}
11	.00057	7.585×10^{-10}
Sum	.0700	

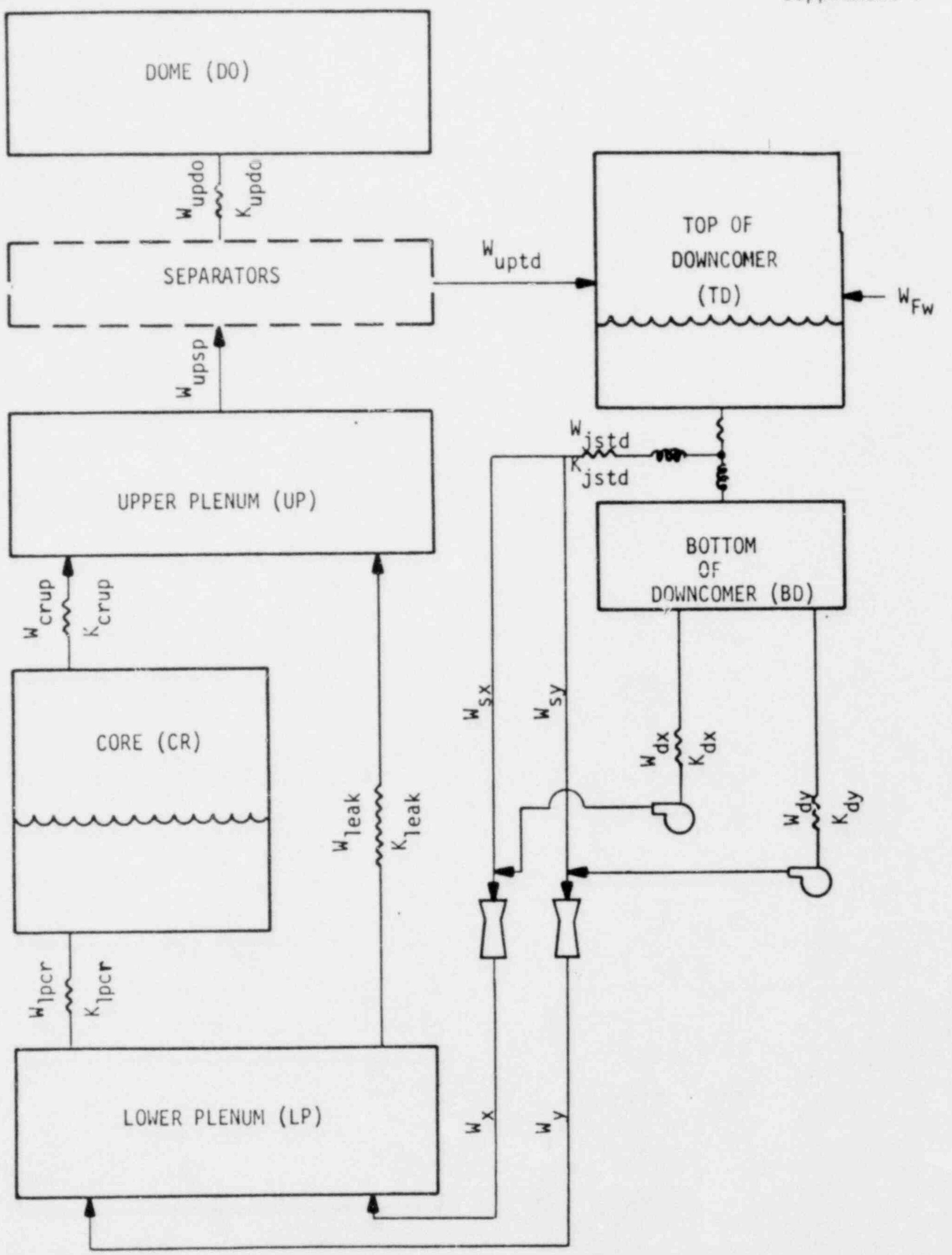

Figure 2.3.1 Recirculating Flow Loop

3.0 NOMENCLATURE

A General term for area

A_{ij} Interfacial area between core nodes i and j

$C_{\ell}(\vec{r},t)$ Precursor concentration for delayed neutron group ℓ

C_v Valve(s) Flow Coefficient

$C_{w,tsv}$ Normalized Flow Coefficient reflecting position of turbine stop valve

c Heat capacity

$D(\vec{r},t)$ One group neutron diffusion coefficient

D_1 Fast neutron diffusion coefficient

D_2 Thermal neutron diffusion coefficient

D_i, D_j One group neutron diffusion coefficient for node i (or j)

d_{ij} Centroid to centroid distance between nodes i and j

e Sum of fluid internal thermal and kinetic energies

$\vec{f}$ Sum of body forces acting on fluid

$G_j(t)$ Concentration of fission product group j at time t

$\vec{g}$ Gravitational force vector

gc Gravitational constant

$G_{j\circ}$ Steady state concentration of fission product group j (at t=o)

H, h General term for fluid enthalpy

h_i Enthalpy of fluid in node i

H_c Fuel clad gap conductance

h_{ini} Enthalpy of stream entering node i

h_{outi} Enthalpy of stream leaving node i

$\dot{h}_i$ General term for time derivative of enthalpy

$h_{i,j}$	Enthalpy of fluid in node i at time j
$h_{i,\ j+1}$	Enthalpy of fluid in node i at time j+1
H_c	Fuel/clad gap conductance
H_g	$\dfrac{1}{\frac{1}{H_c} + Y_c/K_c}$
Hs	Clad surface to fluid heat transfer coefficient
i	Internal energy
Kc	Clad thermal conductivity
$K(K_o)$	Thermal conductivity (at reference time = 0)
K_{crup}	Combined loss coefficient from core to upper plenum
K_{dx}, K_{dy}	Combined loss coefficient for the recirculating drive line flow path through pump, x,y
K_{jsbd}	Combined loss coefficient from jet pump suction to bottom of downcomer
K_{jstd}	Combined loss coefficient from top of downcomer to jet pump suction.
K_{leak}	Combined loss coefficient for core leakage path
K_{lpcr}	Combined loss coefficient from lower plenum to core
K_{updo}	Combined loss coefficient from upper plenum to vessel dome
L_i	Net neutron leakage into node i
L/A	Inertance of a node (length/area)
M_i	General term for mass in node i
$\dot{M}_i$	General term for time derivative of mass within node i
N_v	Normalized position (0-1) of a valve
$\vec{n}$	Unit outward normal vector
P, p	General terms for pressure (hydro static component)
$\dot{P}_i$	General term for pressure derivative in time at node i

P_i	Power density of node i (Section 2.1) General term for pressure at node i (Section 2.3)
P_{sv}	Pressure at steam line valve(s)
q'''	Volumetric heat generation rate
$\vec{q}$	Heat flux vector
R	Fuel pellet radius
r	Coordinate for radial direction
r_N	Ratio of radius of fuel node N to the total fuel pellet radius
$\vec{\vec{T}}$	Surface stress tensor
T_F	Temperature of fluid
T_w	Temperature of solid wall boundary
$\tilde{T}_F$	Fluid temperature at previous time step iteration
T_N	Temperature at fuel radial node N
t	Time
T	General term for temperature
$\vec{u}$	Fluid velocity vector
V, V_i	General term for volume at node i
$v(\vec{r},t)$	One group neutron velocity at position $\vec{r}$ and time t
v_g	Specific volume of saturated vapor at average conditions in node i
v_i	Specific volume of fluid at node i
W	General term for mass flow rate (mass/sec)
W_i	General term for mass flow rate at node i
W_{ini}	Mass flow entering a node i
W_{outi}	Mass flow leaving a node i
W_{crup}	Flow rate from active core region to upper plenum
W_{dx}, W_{dy}	Jet pump drive line flow rate for pump x, y

W_{jstd}	Flow rate from top of downcomer to jet pump suction
W_{leak}	Core leakage flow rate
W_{lpcr}	Flow rate of subcooled water from lower plenum to active core
W_{fw}	Feedwater flow rate
W_{sx}, W_{sy}	Jet pump suction flow rate for recirculating pump x,y
W_{tb}	Flow rate through turbine stop and control valves
$W_{tb,\ max}$	Maximum possible turbine steam flow rate
W_{tot}	Total flow rate demand from turbine pressure regulator
W_{upsp}	Flow rate from upper plenum to separator
W_{updo}	Flow rate from upper plenum to dome (or separator to dome)
W_{uptd}	Flow rate from upper plenum to top of downcomer (or from separator to downcomer)
W_v	Valve(s) mass flow rate
W_x, W_y	Flow rate from jet pump diffusers due to pump x, y
x	Core axial coordinate
Yc	Clad thickness

GREEK SYMBOLS

ρ	Density
β	Total delayed neutron fraction
ϕ_i, ϕ_j	One group neutron flux at node i (or j)
$\phi(\vec{r},t)$	One group neutron flux at position $\vec{r}$ at time t

ν	Average number of neutrons released per fission
λ_ℓ	Decay constant for delayed neutron group ℓ
∇	Gradient
κ	Recoverable energy per fission
$\bar{\phi}_1$	Average fast neutron flux
$\bar{\phi}_2$	Average thermal neutron flux
γ_j	Yield term for fission product group j
λ_j	Decay constant of fission product group j
$\Sigma_f(\vec{r},t)$	One group fission cross section at position $\vec{r}$ at time t
$\Sigma_a(\vec{r},t)$	One group absorption cross section at position $\vec{r}$ at time t
Σ_{fi}	One group fission cross section at node i
$\Sigma_{s_\ell}(1\rightarrow 2)$	Slowing down cross section from fast to thermal neutron groups
Σa_1	Fast neutron(s) absorption cross section
Σa_2	Thermal neutron(s) absorption cross section
Σf_1	Fast neutron(s) fission cross section
Σf_2	Thermal neutron(s) fission cross section
θ	Integral of conductivity over range of reference temperatures to fuel nodal temperature divided by K_0
$\vec{\vec{\pi}}$	Viscous stress tensor

4.0 CODE APPLICATION

4.1 TYPES OF TRANSIENTS

COTRANSA can evaluate the plant system response for a range of abnormal conditions. The COTRANSA code accounts for the effects on plant system response to time dependant axial variations in core radial average parameters. Assumptions are made in COTRANSA as to which regions of the primary system have, or potentially have, two-phase flow conditions. Certain assumptions are also made with respect to the ranges of pressure and flow. Due to these assumptions, certain transients such as large break LOCAs are not properly handled by COTRANSA. COTRANSA 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 incidents. It is important to note that with some of these transients, the time variation of axial power distribution and the interaction between the detailed core neutronic and thermal-hydraulic response with the plant system response, is not significant. In these instances, other methods of analysis, such as PTSBWR3, will provide an accurate calculation of plant and core response.

4.2 SOURCES OF INPUT

The sources of input for the COTRANSA code are:

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

The bulk of plant component 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 principal 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 composition.

The neutronics inputs include cross section tables such as those calculated using XTGBWR[3]. With XTGBWR, the procedure utilized is as follows:

(1) The initial conditions of the problem to be solved by COTRANSA are simulated by an XTGBWR calculation.

(2) The converged three-dimensional fast and thermal flux distributions are used to flux weight the two group XTGBWR cross sections to produce averaged two group COTRANSA cross sections. Usually one set of COTRANSA cross sections is generated for each axial level in the XTGBWR calculations so

that axial exposure, void history, and Xenon distributions are adequately modeled.

Parameters for delayed neutron fractions, fast and thermal velocities, and the derivatives of cross sections with fuel temperature are determined by exposure and volume weighting the fuel type dependent parameters as calculated by ENC with the XFYRE[3] code.

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.).

4.3 VARIETY OF OUTPUTS

The COTRANSA code determines the following:

- o Critical parameters (lower plenum pressure, power, fuel rod heat flux, reactor coolant flow, and coolant enthalpy) from which the plant thermal margins can be derived.
- o Primary system response (valve flows, valve positions, pressures, and timing) to describe anticipated behavior.
- o Heat balance parameters (steam flow, feedwater flow, feedwater enthalpy, steam enthalpy) to include segmented heat balances (steam lines, steam dome, plenums, etc.).
- o Parameter derivatives which, in conjunction with heat balance parameters verify the initial conditions for transient simulation.

Specifically, COTRANSA 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 evaluated.

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

4.4 JET PUMP PLANT PRIMARY SYSTEM DESIGN VARIATIONS

The COTRANSA code is capable of handling plant system differences involving:

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

5.0 CODE TESTING AND VERIFICATION

Exxon Nuclear Company's code verification program for COTRANSA is divided into three benchmark areas:

- o Parametric
- o Systems
- o Integral Plant Test

Work to date in these three areas is discussed in the following subsections.

5.1 PARAMETRIC BENCHMARKS

As COTRANSA models the plant coolant pressure boundary, the calculation of fuel and core coolant conditions provide numerous parameters for benchmarking. Other codes[11,12] 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:

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

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

5.2 SYSTEMS BENCHMARK

COTRANSA 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 is 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 the 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 are given in EPRI Report #NP-564, (June 1978) titled "<u>Transient and Stability Tests at Peach Bottom Atomic Power Station Unit 2 at End of Cycle 2</u>"[2].

5.3.1 COTRANSA Input (Peach Bottom Unit 2)

The bases for plant specific input for Peach Bottom Unit 2 are EPRI Report #NP-563[13] and NP-564[2]. Table 5.3.1 depicts critical initial values determined by the code as compared to the reported test values for test number 3 (TT3).

5.3.2 Results

A summary comparison of important parameters calculated by COTRANSA with measured data for TT3 is shown in Table 5.3.2. Figure 5.3.1 shows a prediction of the average core power compared to reported measurements in EPRI report #NP-564. Figure 5.3.2 depicts vessel pressure rise and recovery.

Table 5.3.1 Transient Test Initial Conditions

Item	COTRANSA	Test TT3
Reactor Power (MW)	2275	2275
Total Recirculating Flow (Mlb/hr)	101.9	101.9
Steam Dome Pressure (psia)	986.6	986.6
Upper Plenum Pressure (psia)	993.	993.
Core Pressure (psia)	998.5	1005.0
Lower Plenum Pressure (psia)	1017.0	Not given
Turbine Emission Pressure (psia)	970.	970.
Core Inlet Enthalpy (Btu/lbm)	523.6	523.6
FW Enthalpy (Btu/lbm)	331.	Not given
FW Flow (Mlb/hr)	8.86	8.86
Steam Flow (Mlb/hr)	8.86	Not given
Core Leakage (Lbm/sec)	2793	Not given
Power Trip (MWt)	2535.6	2535.6
Bypass Valve Capacity (% of rated steam flow)	33%	Not given

Table 5.3.2 COTRANSA Qualification 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	998.5
Initial Core Inlet Enthalpy (Btu/lbm)	523.6	523.6
Peak Average Power	339%	330%
Maximum Core Pressure Rise (psid)	79.0	77.5
Maximum Dome Pressure Rise (psid)	74.4	79.9
Maximum Change in Reactor Water Level (in)	-38.	-35.6

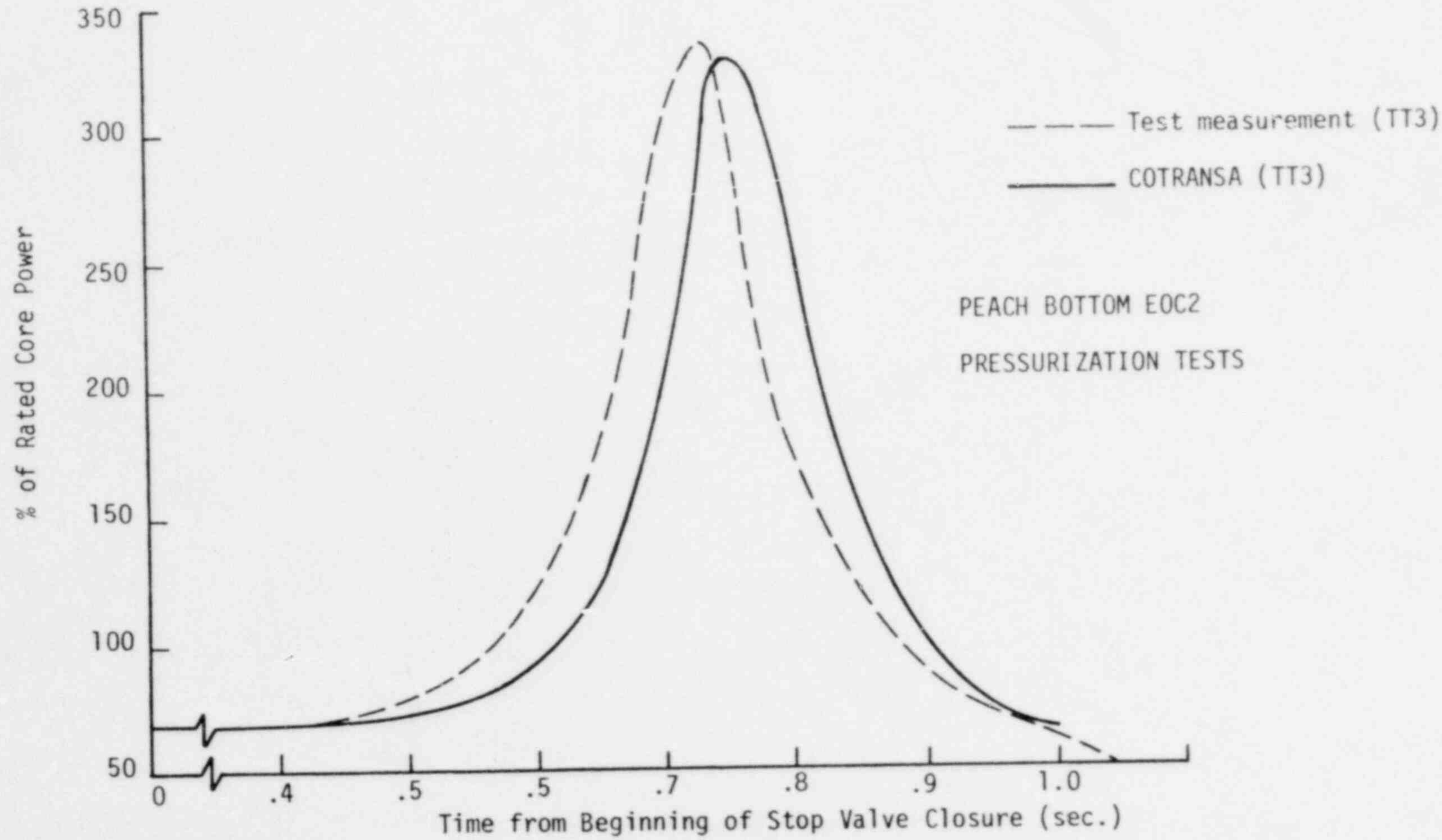

Figure 5.3.1 Average Core Power (TT3)

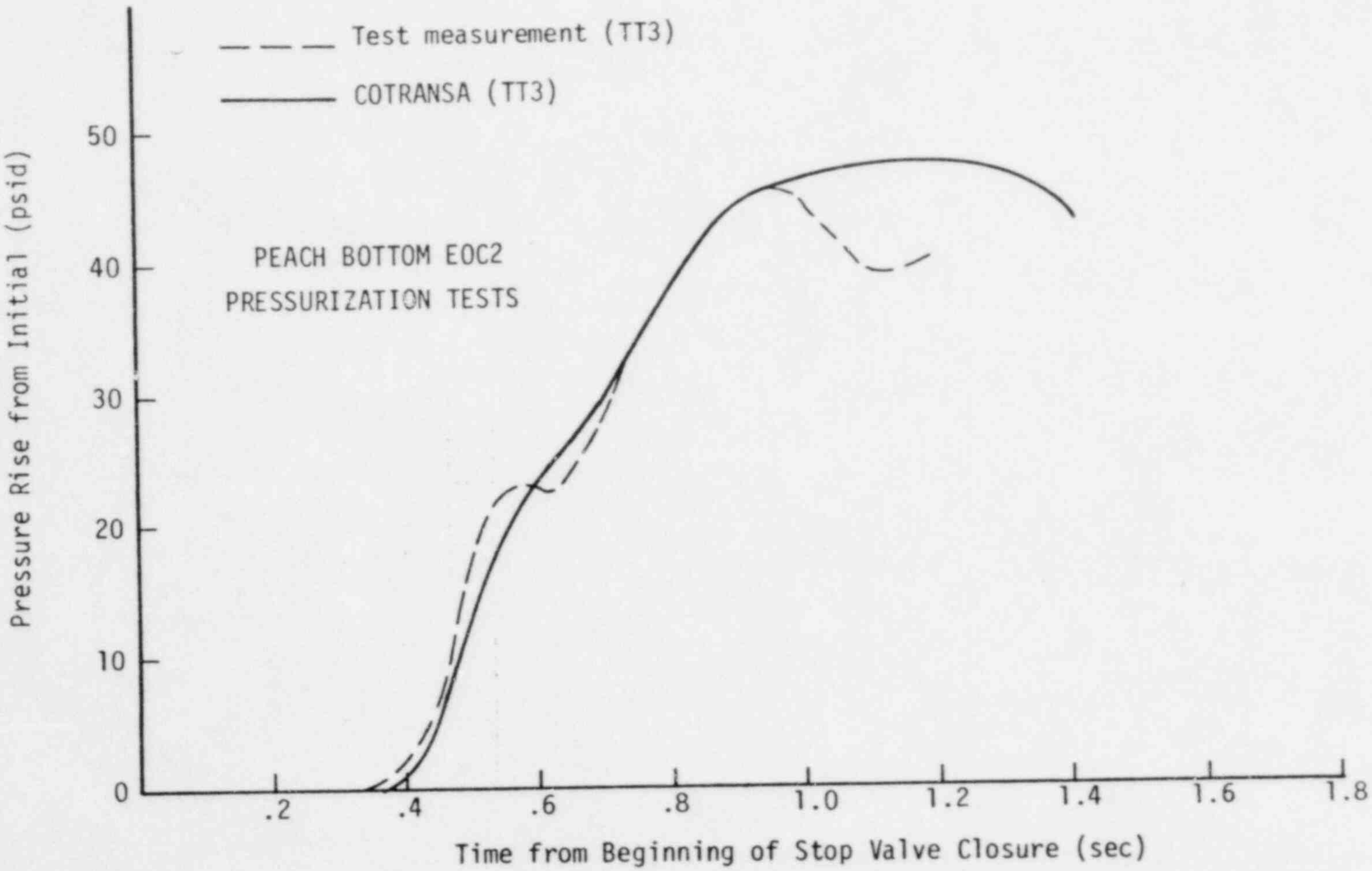

Figure 5.3.2 Dome Pressure (TT3)

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FOR JET PUMP BOILING WATER REACTORS

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