XN-NF-74-5(NP) (A)
& SUPP. 1,2,3,4,5 and 6
REVISION 2

DESCRIPTION OF THE EXXON NUCLEAR PLANT TRANSIENT SIMULATION MODEL FOR PRESSURIZED WATER REACTORS (PTS-PWR)

OCTOBER 1986

RICHLAND, WA 99352

EXXON NUCLEAR COMPANY, INC.

8611040200 861020
PDR TOPRP EMVEXXN
B PDR

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

MAY 05 1986

Mr. G. N. Ward
Reload Licensing
Exxon Nuclear Company, Inc.
2101 Horn Rapids Road
P. O. Box 130
Richland, WA 99352

Dear Mr. Ward:

Subject: Acceptance for Referencing of Licensing Topical Report XN-NF-74-5(P), Revision 2, "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS-PWR)" and Supplements 1 through 6

We have completed our review of the subject topical report. We find this report is acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and the associated NRC evaluation which is enclosed. The evaluation defines the basis for acceptance of the report.

We do not intend to repeat our review of the matters described in the report and found acceptable when the report appears as a reference in license applications except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the report.

In accordance with procedures established in NUREG-0390, it is requested that Exxon publish accepted versions of this report, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions should incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted version shall include an -A (designating accepted) following the report identification symbol.

Should our criteria or regulations change such that our conclusions as to the acceptability of the report are invalidated, Exxon and/or the licensees referencing the topical report will be expected to revise and resubmit their respective documentation, or submit justification for the continued effective applicability of the topical report without revision of their respective documentation.

Charles E. Rossi

Charles E. Rossi, Assistant Director
for PWR Licensing - A

Enclosure:
As stated

TOPICAL REPORT EVALUATION
OF PTS-PWR2

Report Nos. and Title:

XN-NF-74-5, Rev. 2, "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS-PWR)" October 1983.

Originating Organization:

Exxon Nuclear Company, Inc.

I. BACKGROUND

On May 15, 1975, the Exxon Nuclear Company submitted topical report XN-74-05 Rev. 1 for staff review and approval. XN-74-05 Rev. 1, "Description of Exxon Nuclear Plant Transient for Pressurized Water Reactors" (PTSPWR), documents the analytical equations and assumptions used in the PTSPWR computer program. The PTSPWR computer program was designed to analyze FSAR Chapter 15 (as defined in NUREG-0800, "The Standard Review Plan") non-loss of coolant accident (LOCA) events. Since the initial submittal of XN-74-5 Rev. 1, the PTSPWR computer program has undergone extensive modification and validation. This is documented in XN-74-5(P) Rev. 2, Supplements 1 through 6 of Rev. 2 (see References 1 through 6) and in XN-NF-CC-38 Supplement 1 (Ref. 7). As a consequence of the modifications to the PTSPWR program, the code was renamed PTSPWR2.

This SER addresses the PTSPWR2 computer program and its application to FSAR Chapter 15 non-LOCA transient analysis. The staff's review has concluded that selected licensing calculations of Chapter 15 non-LOCA transient events are acceptable if analyzed using PTSPWR2 with appropriately conservative input/ methodology assumptions.

II. SUMMARY OF PTSPWR2 AND THE TOPICAL REPORTS

Topical report XN-NF-74-5(P) Rev. 2 describes the PTSPWR2 code and its applicability to calculate transient and accident events for both Westinghouse and Combustion Engineering (CE) designed nuclear steam supply systems (NSSS). The report documents the governing equations and their numerical solution for the various models. Report XN-74-5(P) Rev. 2, Supp. 1 details the model modifications required to simulate Combustion Engineering 2 x 4 plants.

Application of the code to Chapter 15 transients and accidents is discussed in XN-74-5(P) Rev. 2 Supplement 2. Supplement 3 provides responses to NRC questions and provides corrections to the original topical report. Additional updates are provided in Supplement 4. Benchmark comparisons performed by Exxon to qualify PTSPWR2 for Chapter 15 transients and accidents are detailed in

Supplement 5. An update to the steam generator heat transfer model and an additional benchmark calculation are contained in Supplement 6. Revisions and benchmarks to the pressurizer model are described in XN-NF-CC-38 Sup. 1.

PTSPWR2 is a fixed nodalization code with two primary loops, each modeling a U-tube steam generator. The primary system components modeled include the vessel plena, the reactor core, the primary coolant loops, the pressurizer, the steam generators, and the reactor coolant pumps. The secondary side components include the riser section of the steam generator, the downcomer, the steam separators, the steam lines with their associated systems, and the feed water system. The code also models plant control and protection systems. Point kinetics is used for modeling the neutronics. The thermal hydraulics model assumes one-dimensional HEM (homogeneous equilibrium model) flow with the additional assumptions of incompressibility and quasistatics in specific volumes. Mixing models and transport delay models are used in the plena and the loops. Water properties are taken from the 1967 ASME steam tables. Control system models are provided to describe plant protection and control systems. Boron transport through the coolant loops is also described. Component models include a two region non-equilibrium pressurizer, reactor coolant pumps, and a fuel rod model. An automatic steady state initialization procedure is available.

PTSPWR2 is not utilized to predict localized heat transfer phenomena such as hot channel DNBR. This is done by inputting core flows and inlet thermodynamic conditions from PTSPWR2 into a detailed core heat transfer model. The XCOBRA III B code (Ref. 8) has been approved for this purpose.

III. SUMMARY OF REGULATORY EVALUATION

A. Review Approach

This evaluation assesses the acceptability of PTSPWR2 to perform FSAR Chapter 15 licensing analyses for the following events found in the Standard Review Plan for both Westinghouse and CE plants.

SRP Section	
15.1.1.	Decrease in feedwater temperature
15.1.2.	Increase in feedwater flow
15.1.3.	Increase in main steam flow
15.1.4	Inadvertent Opening of Steam Generator Relief/Safety Valve
15.2.1.	Loss of external load
15.2.2.	Turbine trip
15.2.3.	Loss of condenser vacuum
15.4.4.	Closure of Main Steam Isolation Valve
15.4.5	Steam Regulator Failure
15.2.6.	Loss of normal AC power
15.3.1.	Total and partial loss of forced reactor coolant flow
15.3.3.	Reactor Coolant Pump Rotor Seizure
15.3.4	Reactor Coolant Pump Shaft Break
15.4.1	Uncontrolled control rod assembly withdrawal from sub-critical or low power condition
15.4.2.	Uncontrolled control rod assembly withdrawal at power
15.4.3.	Loop Simulation for control rod misoperation
15.4.4	Startup of an inactive loop
15.4.5	Flow controller malfunction
15.4.6.	Boron dilution at power
15.4.8.	Loop simulation for control rod ejection
15.5.1	Inadvertent ECCS Operation
15.5.2	Excessive charging flow
15.6.1	Inadvertent Pressurizer Relief Valve Opening

To support our evaluation, we have applied the acceptance criteria presented in the respective sections of the NRC Standard Review Plan (NUREG-0800, July 1981 Revisions). We have reviewed the applicant's supporting derivations and experimental data and have made audit calculations using the RELAP5 computer code. Our conclusions regarding the use of the PTSPWR2 code are stated in the Staff Position section of this report.

B. Review of Analytical Models

The fixed nodalization two loop thermal hydraulic node/flow path network used by PTSPWR2 is shown in Fig. 1. The reduced noding arrangement of PTSPWR2 causes some approximations. On the primary side, the pressurizer is connected directly to the upper plenum and no separate volume is provided for the relatively stagnant upper head region. In addition, no distinct volumes are provided for the downcomer and core bypass regions. While Fig. 1 depicts no pressurizer surge line, a model for pressure loss across the surge line was added during the course of this review. Sensible heat for primary and secondary system metal is not described. These simplifications limit the range of application of the code.

Other aspects of the model provide separate volumes for the core, upper plenum, lower plenum, inlet and outlet steam generator plena, U-tubes, and the cold and hot legs. The pressurizer is a special non-equilibrium volume. Additions are made to the nodalization for Combustion Engineering (CE) 2 x 4 plants to allow for dual cold legs per loop. On the secondary side, the steam generators are represented by volumes for the downcomer, tube region, upper plenum, and steam dome. Each steam line is represented by two volumes connecting the common header volume.

Core Neutronics Model

The point kinetics equations of the neutronics are used with six delayed neutron precursor groups. The prompt jump approximation is used for small reactivity changes. The default set of delayed neutron parameters programmed into the code corresponds to those utilized by the staff. Reactivity feedback from fuel temperature (Doppler) and moderator temperature and pressure changes are modeled. Reactivity contributions from control rod movement, as an input function of time and boron concentration can also be modeled. As a consequence of modeling the core as one axial node, the reactivity feedbacks from the moderator and fuel are computed from average temperatures with no spatial weighting. An attempt is made at refining spatial shapes by assuming that the moderator temperature profile follows the integrated power axial shape, thereby calculating an average temperature from that profile. Exxon presented

calculations which demonstrate that the moderator reactivities from this model differed by no more than a percent for both quasistatic and fast transients from those of a multiaxial node model using flux squared weighting. In the case of the Doppler feedback, the reactivity can be off by as much as five percent for quasistatic transients and as much as thirty percent for fast transients, depending upon the range of axial peaking factors used. The reactivity weighting procedures are sensitive to peaking in the respective temperature profiles. However, it should be noted that uncertainties in neutronics data are of the order of 20% and that the Exxon calculations show that effective Doppler temperature changes are underestimated by the PTSPWR2 model which produces higher power on heat ups. PWRPTS2 will not be used to evaluate rapid cooldown events (steam line breaks).

The staff finds the point kinetics model in PTSPWR2 satisfactory with respect to the equations and reactivity feedback model. The use of the prompt jump approximation should be restricted to step reactivity increases of less than \$0.20, for prompt neutron lifetimes greater than 10^{-3} seconds and less than \$0.40 for lifetimes greater than 10^{-4} seconds.

Decay heat is calculated using the model in Branch Position APCSB 9-2 of the NRC standard review plan. The decay constants are fits to the ANS 1971 standard data with additional margin to account for uncertainties. The contribution from heavy element decay is included. This model is acceptable to the staff.

Thermal Hydraulics

Only single phase flow is permitted within the core and the coolant loops, which places limits on the range of code application. The primary coolant loops are further assumed to be incompressible. This assumption maximizes the surge flow into the pressurizer from thermal expansion of the coolant. Within the coolant loops gravity heads, hydraulic losses and inertia are accounted for in the momentum equations which are solved to determine loop flow.

Coolant system pressure loss factors are initialized as constants and remain constant during the calculation. This implies that in the case of flow coastdowns or decreasing flows the effect of increasing friction factors will not be seen. Much of the reactor system pressure loss results from form losses which are little affected by changes in flow rate, however. The flow in each loop is solved separately. Pump heads are computed using the homologous pump model.

The loop pressure is determined by the pressurizer model. During this review Exxon modified the surge line model to include the line friction drop with a constant friction coefficient and a gravity head but no inertia term. However, no iteration is performed between this model and the pressure in the primary loop. The surge flow therefore does not see the line resistance and is based only upon the unconstrained expansion rate in the primary system over a time step. This approximation maximizes the pressure response to thermal expansion and contraction within the coolant loops.

The thermal energy balances are treated differently depending upon the section of the primary system being considered. In the legs a transport delay model is used. The transport delay model divides the hot leg into six fixed segments and the cold leg into 30 fixed segments. A front is tracked through the incompressible segments as the velocities change. Separate effects calculations were presented by Exxon which demonstrated that trends in energy and mass propagation are properly modeled. In the steam generator and vessel plena, homogenized single node energy balances are used. The one time constant mixing models assume a predominant flow direction in order to determine the plenum mixing time constant. The core coolant thermal behavior is treated quasi statically. The coolant enthalpy distribution for the ten axial nodes is computed on the assumption that the distribution follows the integrated power shape up the channel.

For the core pin to coolant heat transfer, the single phase Dittus Boelter correlation is used normalized to an input initial heat transfer coefficient.

Core heat conduction is calculated by a one dimensional radial heat transfer calculation using the Fourier conduction equation with a uniform power profile and temperature dependent thermal properties in the pellet but constant properties in the cladding. The pin is divided into four equal volume fuel radial nodes and one cladding node with the appropriate node center to node center conductances. The fuel/clad gap is treated by a constant gap conductance as all dimensions are assumed fixed during a transient. The radially averaged fuel temperature is used for the Doppler feedback. The fuel rod model is found acceptable for applications when core DNBR is not directly calculated by PTSPWR2.

The steam generator model was modified during the review. The present model assumes that the primary side of the tubes is always single phase liquid and the secondary side is always two phase and saturated. The heat transfer coefficients on the secondary side remain constant at the steady state values throughout the transient. This assumption is acceptable as long as the steam generator tubes remained covered. The heat transfer on the inner tube walls within the primary system is calculated using the flow dependent Dittus-Boelter correlation. This flow dependence allows PTSPWR2 to be used for analysis of flow coastdown transients as will be discussed further in the section on code audits.

Steam flow out of the steam generators into the steam line volumes is assumed to be saturated steam. Complete separation of steam is assumed so that only steam enters the steam lines and any liquid water remains in the steam generators. The steam line momentum balances include the momentum fluxes, the frictional drops and the inertia terms. Steam flow out of the various valves is calculated using the valve model discussed in the following section.

The code utilizes saturated thermodynamic properties and derivatives in the pressure range 100-3100 psi which are taken directly from the 1967 ASME steam tables. In the subcooled region the pressure dependence of the thermodynamic properties is neglected within the coolant loops. Below the pressure of 2200 psi, the 2200 psi subcooled properties are used; in the range from 2200-2300 psi the 2300 psi properties are used while from 2300 psi to 3100 psi the

3100 psi properties are used. Absolute errors by making such an approximation are of the order of ~ 1% which is acceptable.

For the pressurizer model the thermodynamic properties of water are utilized in comprehensive form with no approximations assumed. Both the superheated and subcooled region are treated in the form of two dimensional tables.

Components

(a) Pressurizer

Reactor system pressure is determined by the pressurizer model. A two region (vapor and liquid) non-equilibrium model was developed with each region being single phase and homogenized. The inter-region mass and energy transfer terms model the condensation and bulk flashing processes. Convective heat transfer is user determined between the two regions. The model provides a reasonable approximation for FSAR transients.

Boundary conditions include a heater model and a pressurizer spray model which is user controlled. The fraction of spray deposited in either region is input. Heat loss to the pressurizer boundary is not assumed which should produce pressure overestimates.

Based on the technically acceptable analytical equations and qualification work discussed later in this report the staff approves the use of the PTSPWR2 pressurizer model with a zero inter-region heat transfer coefficient which is conservative.

(b) Reactor Coolant Pumps

The reactor coolant pump model utilizes four quadrant homologous torque/head curves in single phase. The pump heat is input and is added to the core heat. The pump speed equation incorporates electrical torque (set to zero for coastdowns), hydraulic torque (using the homologous torque curves) and friction losses defined as a linear polynomial in the speed. Thus a coastdown including

assumed sheared shaft can be simulated. An option is also available for modelling the pump as a geometric loss (locked rotor).

(c) Valves

Both safety/relief valves and isolation valves are modelled allowing for hysteresis, different opening and closing rates and partially open positions. The Exxon critical flow correlation for steam flow out of relief and safety valves compares well with the Murdock-Bauman correlation typically used for pure steam flow. Liquid critical flow is not modeled.

Special Purpose Models

(a) Control and Safety Systems

The plant protection trip options include both Westinghouse and Combustion Engineering design logic. For Westinghouse plants the trip signals include high nuclear power, overtemperature ΔT and overpower ΔT. Additions made for Combustion Engineering plants include the asymmetric steam generator protection trip, the variable high power trip, the thermal margin/low pressure trip and the local power density trip. The coding has been modified so that each trip signal utilizes a unique scram delay time. The remaining control system models include pressurizer level and pressure controls, the feedwater controller, turbine pressure regulator, steam dump system, and a rod speed controller.

(b) Boron Transport

Transport from the point of injection to the lower plenum is determined by input as the model assumes injection in the lower plenum with an input time delay. A table of head versus flow is used to simulate the injection pump. Transport in the various plena and legs is the same as for the loop thermal hydraulics. As the concentration is tracked in terms of ppm, errors in coolant density do not translate into inaccuracies in the boron concentration. The model is therefore approved.

Initialization

Automatic initialization of the secondary system is carried out by adjusting feedwater flow rates and bundle exit qualities to be compatible with input data for T_{av}, dome pressures, steam generator water levels and feedwater enthalpies. Steam generator heat transfer coefficients and primary side temperatures are extracted from the design conditions of loop flow and power. Adjustments are made to coolant pump speed to be compatible with the loop flow resistances. The steam flow is initialized with the adjusted feedwater flow and steam line friction. The pressurizer is initialized with the input pressure. The outsurge is set equal to the adjusted spray flow rate.

Numerical Technique

The numerical techniques employed are based upon first order explicit schemes and are not unconditionally stable. Modifications were made to overcome instabilities during benchmark comparisons. These modifications restrict temperature changes during a single time step.

C. Code Qualification

Exxon has presented qualification work in the form of three separate categories: (1) audit calculations using RELAP5, (2) comparisons with system effects experiments and (3) a benchmark against real plant data. While each category has certain limitations, the overall combination of audit calculations, benchmarking against systems effects experiments and comparisons with real plant data provides justification for the overall acceptability of the various code models as applied to specific transients. The audit calculations were performed for a flow coastdown event (three pump coastdown) using an H. B. Robinson plant model (three loop Westinghouse). The comparisons against data took the form of analyses of three LOFT tests; L6-1 (loss of load) L6-2 (pump coastdown) and L6-3 (excess load). These comparisons therefore cover both heatup and cooldown events in addition to testing the models for the pump coastdown against measured data. The benchmark calculation of the Prairie

Island steam generator tube rupture incident provides a measure of scaling effects in going to real plant scale.

Audit Calculations

Results from a detailed RELAP5 model of the H. B. Robinson plant were presented by Exxon for a three pump coastdown event. Differences in initial conditions appear to be of the order of a percent or less whereas differences in boundary conditions are minimal. The audit calculation shows good agreement for the core power behavior and reasonable agreement for the loop flow coastdown. The differences in coolant temperature are only a few degrees (see Figures 2 and 3).

It can be seen from Fig. 4 that the PTSPWR2 pressurizer model is over-reactive. Both insurge and outsurge behavior are off by ~ 100 psi. This behavior can be attributed to the forcing of the inter-region heat transfer to zero, the lack of compressibility modeling of the subcooled water in the coolant loops and the lack of a primary system metal heat flow model in PTSPWR2. It should be concluded that the pressurizer model will systematically overpredict the extremes in the pressure behavior both during insurges and outsurges.

LOFT Experiments

(a) LOFT L6-1

The L6-1 loss of load test is a heatup event. During the first loop cycle time the steam flow rate and feedwater flow rate are well matched in the comparisons against data. While the initial conditions are offset slightly the hot leg flow is in reasonable agreement. The comparison for core power is within a couple of percent. During this initial period of the first loop cycle time both the hot leg and cold leg temperature are overpredicted. This is due in part to the overprediction of the steam generator pressure. Conversely on the downswing during the reopening of the steam valve the rate of decrease in cold leg and hot leg temperatures is overpredicted (see Figure 5). The pressurizer

model is overreactive; however, in this case the outsurge is markedly underpredicted while the insurge is reasonably well matched.

The excessive decrease in pressurizer pressure and reactor system temperature probably results from the lack of a primary system metal heat transfer model in PTSPWR2 whereas the LOFT test facility contains an untypically large metal mass for the amount of reactor coolant.

(b) LOFT L6-2

This system transient test includes a pump coastdown similar to the RELAP5 audit calculation. Both power and primary flow are reasonably predicted by PTSPWR2 until pump coastdown is complete.

Complications then arise which in part may be attributed to limitations in LOFT instrumentation or to deficiencies in the simulation, both in modelling and in specific input used. The boundary conditions of feedwater flow and steam flow are reasonably matched. For the first loop cycle time the loop hydraulics appear to be well simulated by the PTSPWR2 pump model and hydraulics model. Fig. 6 shows the cold leg temperature which is predicted to within 5°F of the test data. As with the RELAP5 H. B. Robinson audit calculation the pressurizer model demonstrates an oversensitivity both on insurges and outsurges.

(c) LOFT L6-3

This excess load test is a sample of the cooldown class of events and tests the code models in a range different from L6-1 and L6-2. It was initiated at LOFT by ramping the main steam flow control valve MFECV to full open. The valve was closed at 37 seconds. Reactor trip occurred at 15.6 seconds. Good matches are obtained for the boundary conditions of feedwater flow and steam line flow. There is also reasonable agreement on the power and the primary side hot leg flow. The steam generator dome pressure is well matched in the period when the MSFCV is open (see Figure 7). When the valve is closed PWR-PTS2 which does not predict heat transfer from the steam generator shell to the secondary coolant

under predicts steam pressure. Both the cold and hot leg temperature predictions are low. This difference again probably results from the lack of metal heat transfer in the model and the relatively large metal mass in the test. The LOFT L6-3 benchmark also shows over-reactive behavior both on upswings, and down-swings for the pressurizer model. The predictions of PTS-PWR2 are acceptable in the time of interest for DNBR calculations.

PWR Comparison

To resolve questions regarding the effect of scaling, comparisons against real plant data are helpful. Exxon has submitted such a comparison: a benchmark against the Prairie Island reactor trip of October 2, 1979. The reactor trip resulted from low reactor system pressure following a steam generator tube rupture. The loss of inventory resulted in depressurization, load runback and finally reactor scram. Turbine trip on low pressurizer pressure produced a loss of load transient and a challenge to the validity of the PTSPWR2 models. As is common for real plant experience the data are limited and provide some inconsistencies. There are specific deviations but the trend in data appears to be in qualitative agreement. Quantitative conclusions are difficult to draw. There are definite discrepancies in cold leg temperature comparisons. The loop T_{av}, the loop ΔT and the hot leg temperature, appear to be in good agreement (see Figure 8).

D. Limitations on PTSPWR2 for Transient and Accident Analysis

(a) The PTSPWR2 steam generator model does not account for the effect of steam generator tube uncovery on secondary system heat transfer. The code therefore cannot be used for analyses of large steam or feedwater line break events or loss of feedwater events for which the steam generator tubes might become uncovered without further justification and staff review.

(b) The relief and safety valve flow model is not designed to calculate flow rates when the inlet is liquid or two-phase. The code therefore cannot be used when the steam lines or pressurizer become flooded to the valve

inlets without further justification and staff review. Further justification should also be provided for the adequacy of steam generator or pressurizer pressure calculations if flooded conditions are calculated.

(c) Because PTSPWR2 requires the primary loop (except for the pressurizer) to be subcooled, use of the code is not acceptable whenever the core outlet temperature exceeds saturation. Since coolant loop saturation is likely to occur if the pressurizer becomes drained of liquid, the code should not be used if a zero pressurizer level is calculated without further justification and staff review.

(d) Staff review did not include the ability of PTSPWR2 to implicitly calculate heat transfer in the core hot channels; therefore, for all events in which minimum DNBR is an issue, an acceptable hot channel thermal hydraulic code such as COBRA III B must be combined with PTSPWR2 for calculating DNBR.

(e) Analyses utilizing PTSPWR2 should be made conservative by selecting conservative input. Exxon methology for selecting conservative input is described in XN-NF-84-73(P) Revision 2 "Exxon Nuclear Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," March 1986. This document is currently undergoing staff review for application to Westinghouse 3 and 4 loop PWRs. Approval is expected without significant modification. PTSPWR2 should be utilized for licensing actions only as described in XN-NF-84-73(P) Revision 2 or the latest approved version of that document.

(f) Since PTSPWR2 does not contain a space dependent core neutronics model, local power peaking must be calculated separately for events producing high local peaking. Acceptable methology is described in XN-CC-28 for control rod misoperation and XN-NF-78-44 for control rod ejection.

(g) The staff reviewed the adequacy of the PTSPWR2 for analysis of the transients and accidents listed in Section III.A for Westinghouse and Combustion Engineering pressurized water reactors with inverted U-tube steam

generators. Applications of the code for other reactor types will require additional justification and staff review.

IV. QUALITY ASSURANCE AUDIT

The NRC staff conducted audits of Exxon quality assurance procedures for computer code development on April 3-5, 1984 and again on July 8-12, 1985. Two nonconformances related to the PTSPWR2 code were identified. One item concerned the lack of proper sign off on a calculation. This was resolved and accepted by the NRC staff. The other item concerned engineering training. Exxon has instituted a training program which deals with PWR safety analysis. The staff finds the quality assurance procedures applied in the development of PTSPWR 2 acceptable.

V. STAFF POSITION

We have reviewed the methods and assumptions described in XN-NF-74-5 Rev. 2 including the 6 supplements and have concluded, subject to the limitations stated in Section III.D of this report that the PTSPWR2 code contains acceptable models and methods for transient analysis of the specific events listed in Section III.A. The adequacy of the results obtained with PTSPWR2 will be highly dependent on the adequacy of the plant parameters and input into the code. For this reason the staff requires that results obtained with the PTSPWR2 code be accompanied with appropriate descriptions and justifications for the code input. We require that topical report XN-NF-74-5(P) be reissued to include all revisions and corrections as described in the supplements or otherwise identified by Exxon during the review.

REFERENCES

(1) XN-74-5(P), Revision 2, Supplement 1, "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS-PWR): PTSPWR2 Modifications for St. Lucie Unit 1," Exxon Nuclear Company, Inc., Richland, WA, October 1983.

(2) XN-74-5(P), Revision 2, Supplement 2, "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS-PWR): Methodology and Application," Exxon Nuclear Company, Inc., Richland, WA, January 1984.

(3) XN-74-5(P), Revision 2, Supplement 3, "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS-PWR): Responses to NRC Questions," Exxon Nuclear Company, Inc., Richland, WA, September 1984.

(4) XN-74-5(P), Revision 2, Supplement 4, "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS-PWR): Code Updates ," Exxon Nuclear Company, Inc., Richland, WA, August 1984.

(5) XN-74-5(P), Revision 2, Supplement 5, "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS-PWR): Code Benchmarks," Exxon Nuclear Company, Inc., Richland, WA, August 1984.

(6) XN-74-5(P), Revision 2, Supplement 6, "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS-PWR): Revised PTSPWR2/RELAP5 Benchmark Analysis," Exxon Nuclear Company, Inc., Richland, WA, November 1984.

(7) XN-NF-CC-38, Supplement 1, "Users Manual for PTSPWR2 a FORTRAN Program for Simulation of Pressurized Water Reactor Plant Transients," Exxon Nuclear Company, Inc., Richland, WA, October 1979.

(8) XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company, Inc., Richland, WA, December 1984.

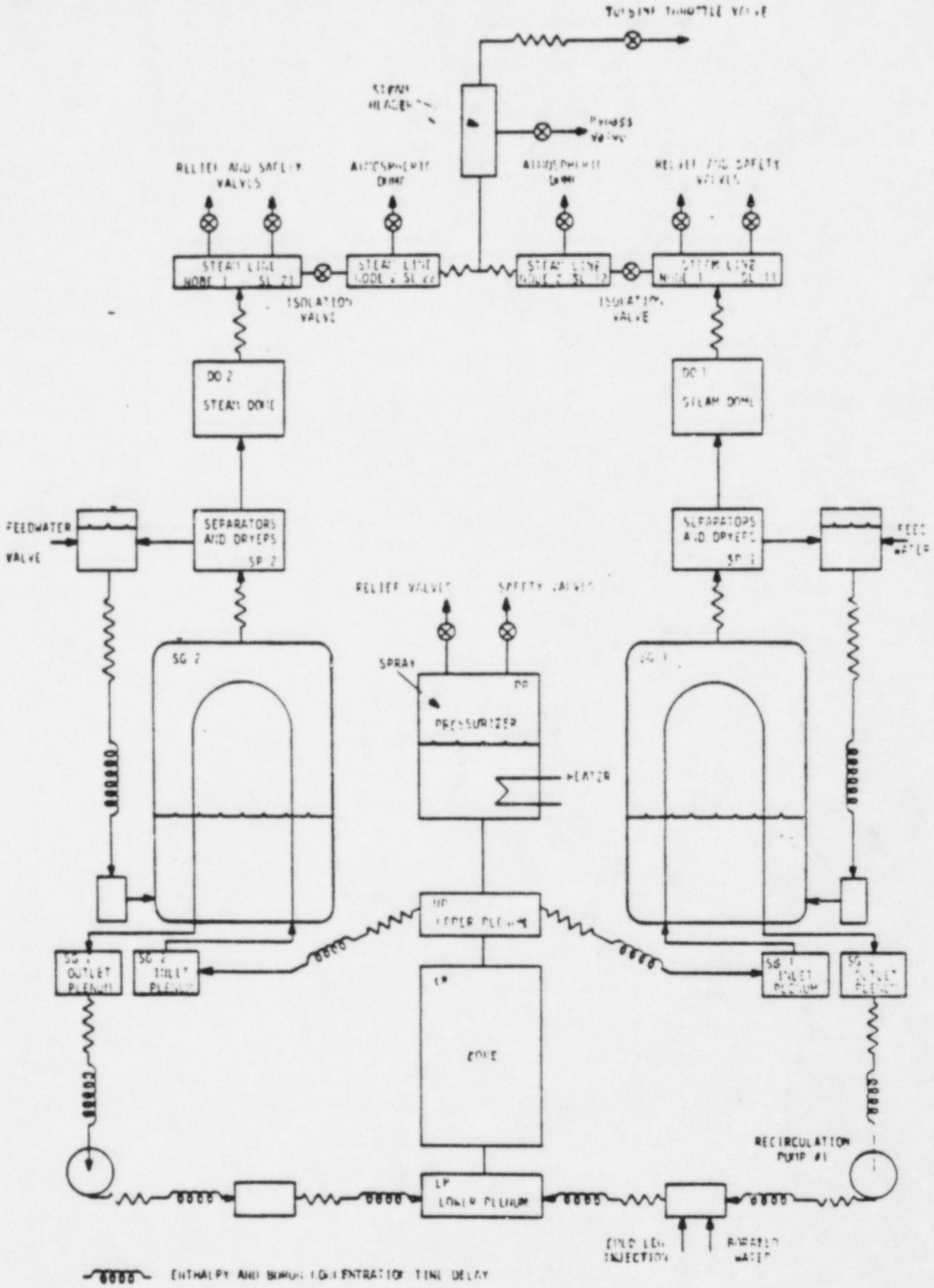


FIGURE 1 PWR MODEL BLOCK DIAGRAM

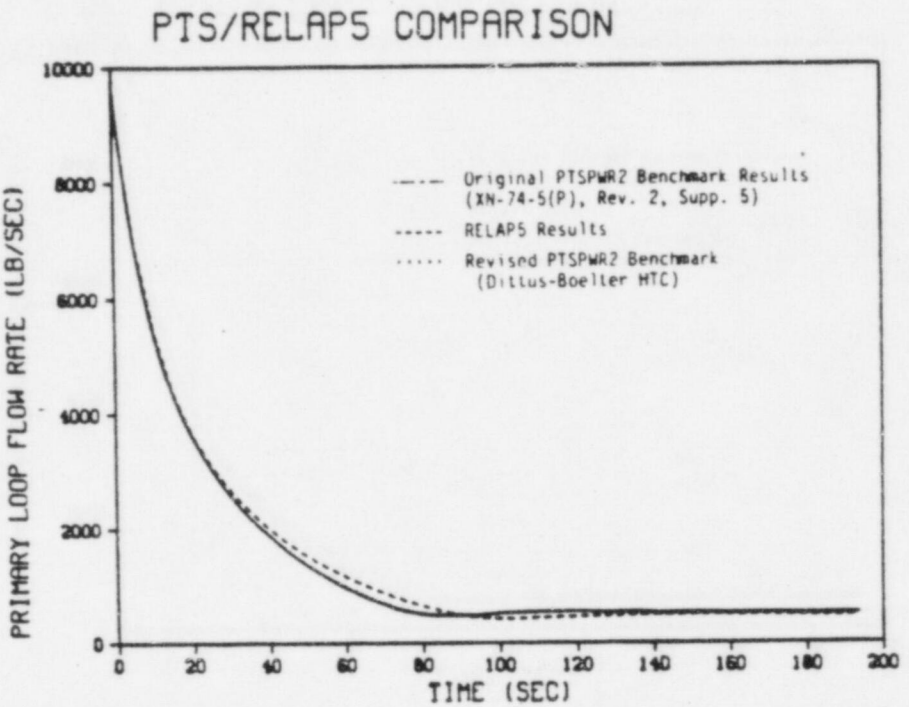


Figure 2: Flow Coastdown

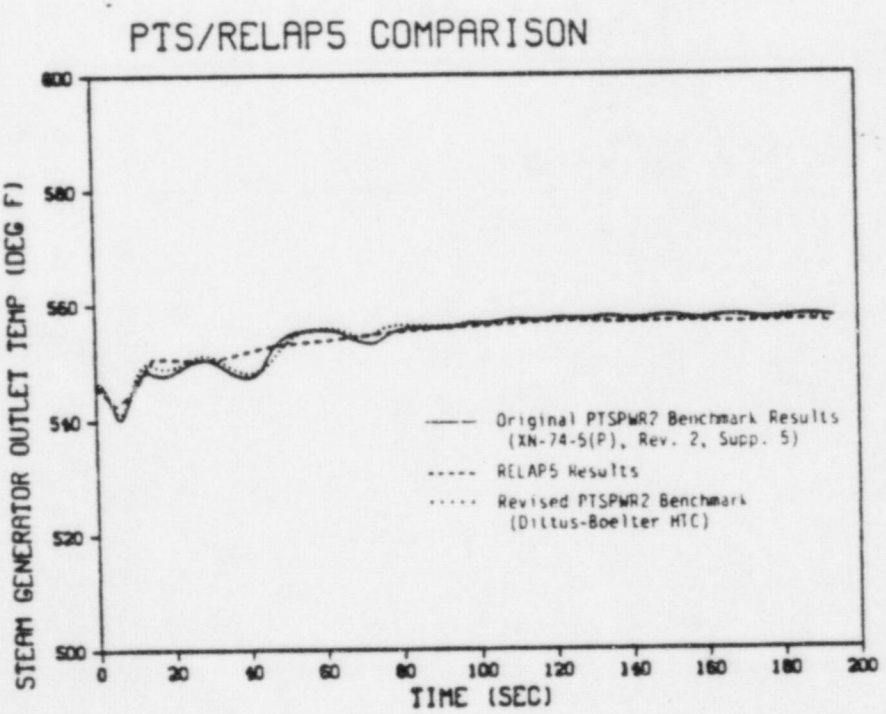


Figure 3: Cold Leg Temperature

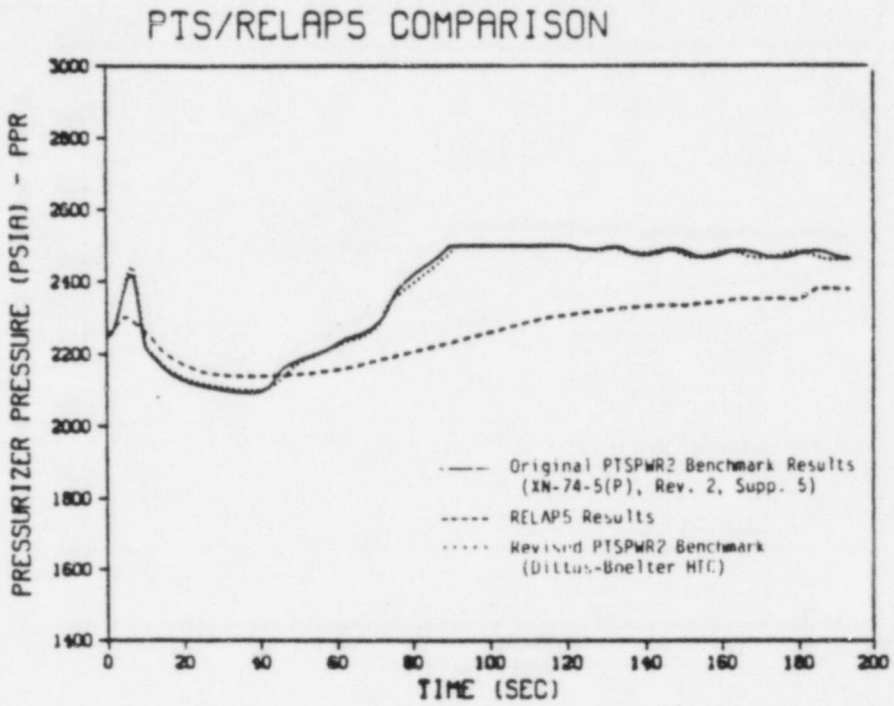


Figure 4: Pressurizer Pressure

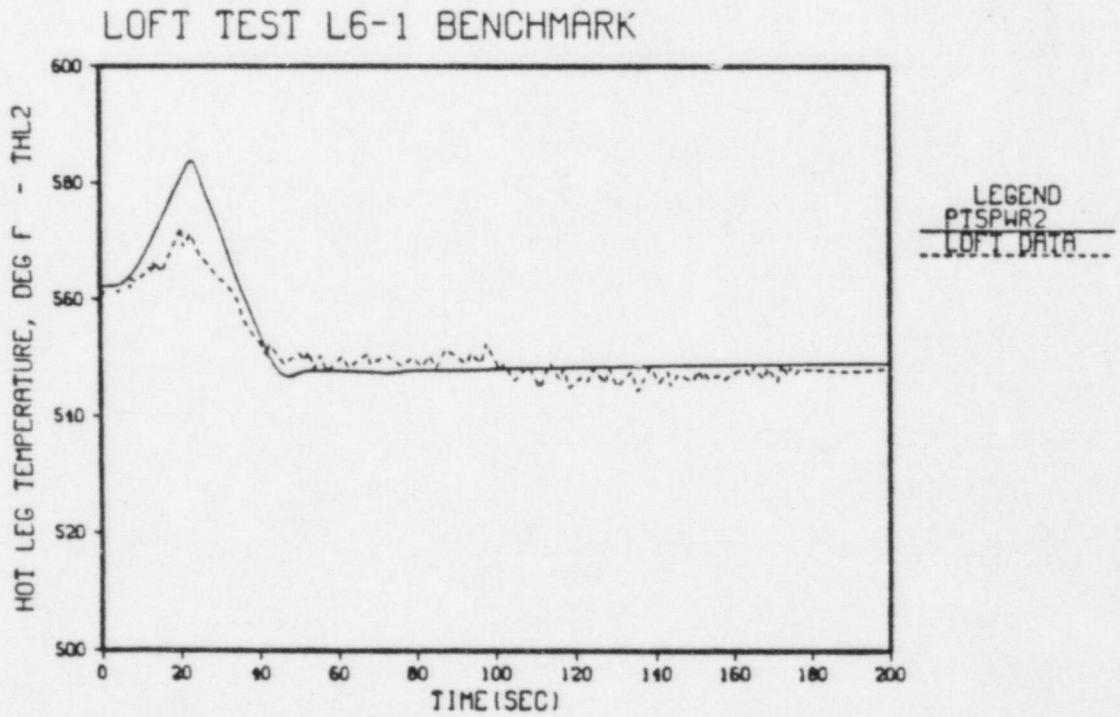


Figure 5: LOFT L6-1 Hot Leg Temperature

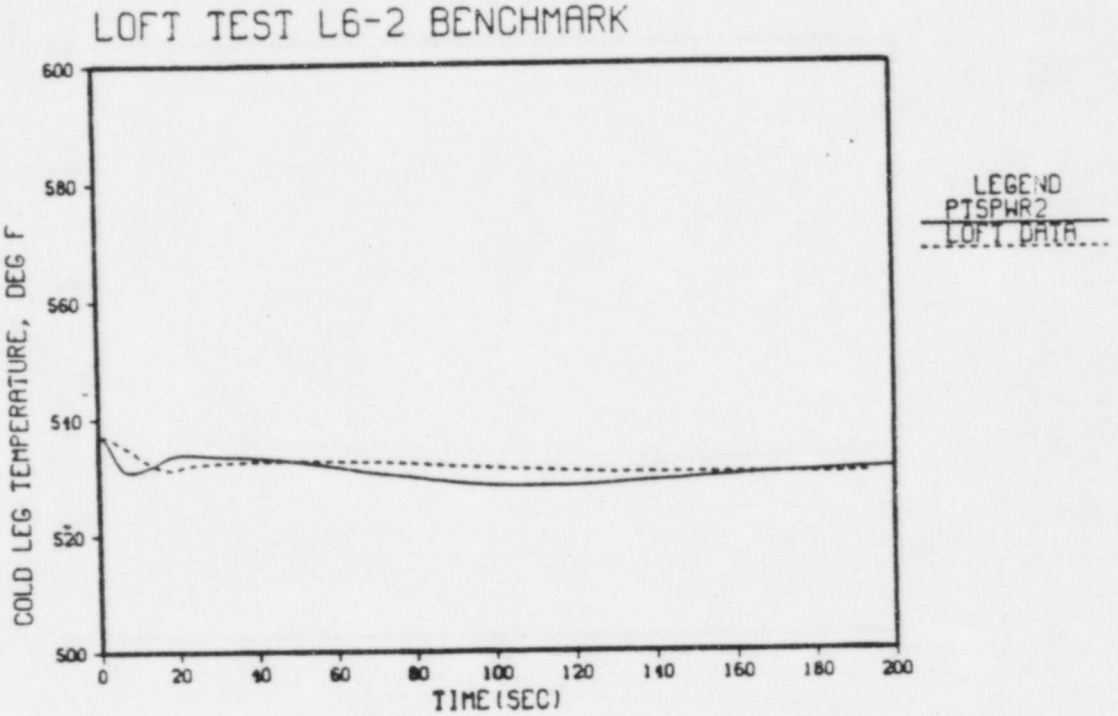


Figure 6: L6-2 Cold Leg Temperature

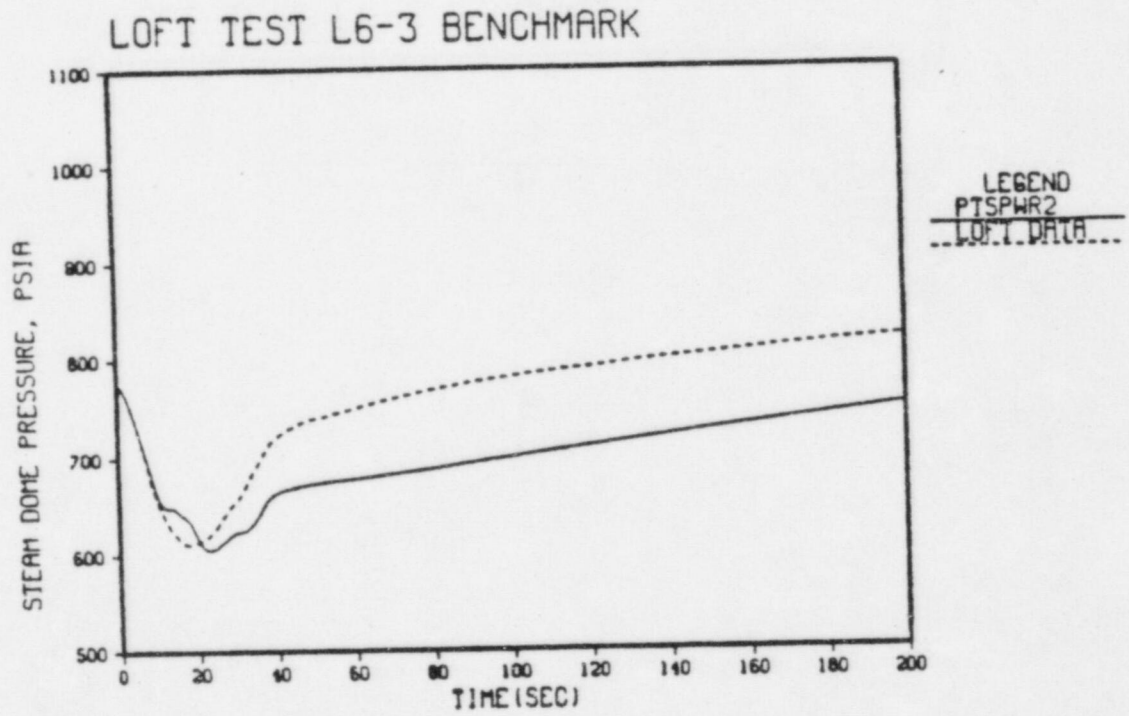


Figure 7: L6-3 Steam Pressure

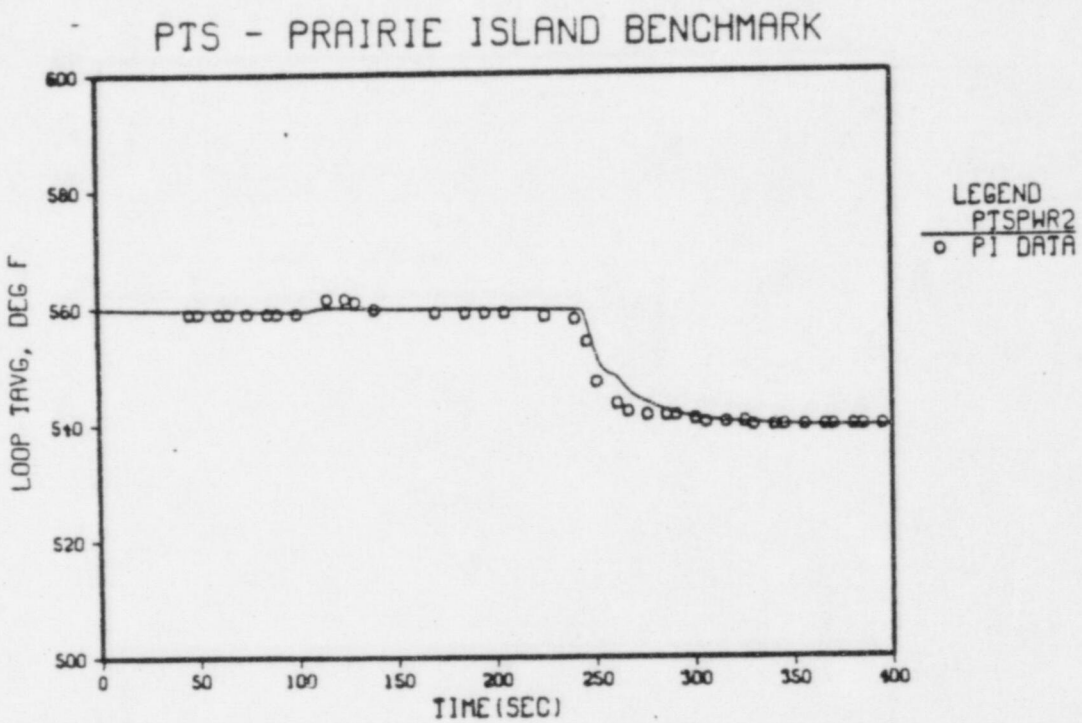


Figure 8: Prairie Island LOOP T_{avg}

XN-NF-74-5(NP)(A)
Revision 2
Issue Date 10/8/86

DESCRIPTION OF THE EXXON NUCLEAR PLANT TRANSIENT SIMULATION MODEL FOR PRESSURIZED WATER REACTORS (PTS-PWR)

Prepared by: David M. Turner
D. M. Turner
Model Development

Concur: G. C. Cooke 6-10-83
G. C. Cooke, Manager
Model Development

Concur: W V Kayser 8/25/83
W. V. Kayser, Manager
PWR Safety Analysis

Concur: for J. C. Chandler 9/20/83
J. C. Chandler
Reload Fuel Licensing

Approve:
R. B. Stout, Manager
Licensing & Safety Engineering

Approve:
G. A. Sofer, Manager
Fuel Engineering & Technical Services

EXXON NUCLEAR COMPANY, INC.

NUCLEAR REGULATORY COMMISSION REPORT 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 U.S. Nuclear Regulatory Commission as part of a technical contribution to facilitate safety analyses by licensees of the U.S. Nuclear Regulatory Commission 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 U.S. Nuclear Regulatory Commission in its review of this report, and under the terms of the respective agreements, by licensees or applicants before the U.S. Nuclear Regulatory Commission which are customers of Exxon Nuclear in their demonstration of compliance with the U.S. Nuclear Regulatory Commission's regulations.

Exxon Nuclear's warranties and representations concerning the subject matter of this document are those set forth in the agreement between Exxon Nuclear and the customer to which this document is issued. Accordingly, except as otherwise expressly provided in such agreement, neither Exxon Nuclear nor any person acting on its behalf:

A. Makes any warranty, or representation 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

	Page
1.0 INTRODUCTION AND SUMMARY	1
2.0 PRIMARY LOOP MODEL	6
2.1 CORE KINETICS MODEL	6
2.2 CORE THERMAL-HYDRAULICS	9
2.2.1 Fuel Temperatures	9
2.2.2 Average Channel Thermal-Hydraulics	14
2.3 PRIMARY LOOP HYDRODYNAMICS MODEL	16
2.3.1 Primary Loop Flow	16
2.3.2 Primary Loop Temperatures and Specific Volumes	20
2.3.3 Pressurizer Surge Flow	23
2.3.4 Primary Loop Boron Concentrations	24
2.4 PRESSURIZER MODEL	25
3.0 STEAM GENERATOR MODEL	33
3.1 STEAM GENERATOR HEAT TRANSFER MODEL	33
3.2 STEAM GENERATOR HYDRODYNAMICS MODEL	42
4.0 STEAM LINE MODEL	51
5.0 CONTROL AND SAFETY SYSTEM MODELS	57
5.1 TRIP SYSTEM LOGIC	57
5.2 VALVE FLOWS	58
5.3 PRESSURIZER LEVEL CONTROLS	60
5.4 PRESSURIZER PRESSURE CONTROL	60

TABLE OF CONTENTS (Continued)

		Page
5.5	FEEDWATER CONTROLLER.	61
5.6	TURBINE PRESSURE REGULATOR	61
5.7	STEAM DUMP CONTROLS.	62
5.8	ROD SPEED CONTROL SYSTEM	63
6.0	HOT CHANNEL ANALYSIS	74
7.0	NOMENCLATURE	78
8.0	CODE USAGE	107
8.1	BUILT-IN CHARACTERISTIC PROPERTIES	107
8.1.1	Water and Steam Tables	107
8.1.2	Pump Characteristics	107
8.1.3	Decay Energy Calculations.	107
8.2	PREPARATION OF INPUT.	109
8.3	STEADY STATE CALCULATIONS	110
8.4	RESTART OPTION.	112
8.5	TRANSIENT RELATED INPUT	113
8.5.1	Transient Specific Input	113
8.5.2	Additional Imposable Conditions	116
8.5.3	Automatic Reactor Control.	116
8.6	INTERPRETATION OF OUTPUT	117

TABLE OF CONTENTS (Continued)

	Page
9.0 REFERENCES	163
APPENDIX A SUBROUTINE CALLING SEQUENCES	A-1
APPENDIX B SUBROUTINE FUNCTIONAL DESCRIPTION	B-1
APPENDIX C SIMULATED PRESSURIZER TEST DATA	C-1

LIST OF TABLES

		Page
1.1	PTSPWR2 TRANSIENT ANALYSES.	4
5.1	TRIP SIGNALS INCORPORATED IN PTSPWR2.	64
8.1	AVAILABLE WATER AND STEAM TABLES IN DATA.	119
8.2	PTSPWR2 DEFAULT PARAMETERS.	120
8.3	TABULAR PARAMETER ARRAYS.	148
8.4	SUMMARY OF PARAMETERS NOT USUALLY ENTERED	149
8.5	TYPICAL TRANSIENT RUN TIMES	150
8.6	INTERPRETATION OF OUTPUT.	151
8.7	TABULAR OUTPUT.	160
8.8	SCRAM CODES	161

LIST OF FIGURES

		Page
1.1	PWR MODEL BLOCK DIAGRAM.	5
2.1	FUEL ROD THERMAL MODEL	30
2.2	AXIAL NODALIZATION OF CORE COOLANT CHANNEL	31
2.3	PRESSURIZER MODEL.	32
3.1	PWR STEAM GENERATOR NODALIZING SCHEME (LOOP 1)	49
3.2	STEAM GENERATOR HYDRODYNAMICS MODEL.	50
4.1	STEAM LINE NODALIZATION.	56
5.1	OVERTEMPERATURE ΔT TRIP LOGIC.	65
5.2	OVERPOWER ΔT TRIP LOGIC.	66
5.3	VALVE OPENING AND CLOSING CHARACTERISTICS.	67
5.4	PRESSURIZER LEVEL CONTROLLER	68
5.5	PRESSURIZER SPRAY FLOW CHARACTERISTICS	69
5.6	FEEDWATER CONTROLLER	70
5.7	TURBINE PRESSURE REGULATOR	71
5.8	STEAM DUMP CONTROLS.	72
5.9	ROD SPEED CONTROL SYSTEM	73
8.1	TYPICAL FLOW-HEAD PUMP CURVES.	162

ABSTRACT

The PTSPWR2 code is a digital computer program, written in FORTRAN language, which simulates the behavior of multiloop pressurized water reactors subjected to abnormal operating conditions, such as loss of electrical load, flow coastdown, dropped rod, etc. The model is based on the solution of the basic transient conservation equations for the primary and secondary coolant systems, of the transient conduction equation for the fuel rods, and of the point kinetics equation for the core neutronics. The program calculates fluid conditions such as flow, pressure, mass inventory and quality, heat flux in the core and reactor power, and reactivity during the transient and provides tabular printout of these parameters. The PTSPWR2 code is presently in operation on the CDC Cyber-175 computer.

ACKNOWLEDGEMENT

The plant transient simulation code described in this report is based on a model originally developed and coded for the PNL-1 hybrid computer at Battelle Northwest Laboratories, Richland, Washington, under a research and engineering contract between Exxon Nuclear Company, Inc., and Battelle Northwest Laboratories. Personnel involved in the conversion effort were J. D. Kahn and M. S. Foster of Exxon Nuclear Company and R. A. Burnett of Battelle Northwest.

1.0 INTRODUCTION AND SUMMARY

The supply of reload fuel for Pressurized Water Reactors (PWRs) could result in core parameters which are slightly different from those used in developing the information in plant FSARs and their amendments. Some changes in parameters would appear to cause plant response to various accidents to provide greater safety margins, while some would tend to reduce margins relative to design criteria and safety system performance limits. For this reason, it is necessary to have the capability to re-analyze the accidents with the new kinetics parameters so as to determine whether the plant technical specifications can continue to be met with the reload fuel.

This report describes the Plant Transient Simulator for PWRs (PTSPWR2) developed by Exxon Nuclear Company. The code is used to predict plant response to the transients considered in the Safety Analyses section of the Final Safety Analysis Reports for PWRs. Table 1.1 identifies some of the applicable transients.

Figure 1.1 is a block diagram of the PWR plant components that are contained in PTSPWR2. The components are grouped into three separate interfaced models: (1) primary loop, (2) steam generator, and (3) steam line. These models are described in Sections 2.0 through 4.0.

The primary loop is modeled in several parts. A point kinetics model is employed to determine reactivity changes during the transient.

The hydrodynamics model uses pressure drop data around the loop and steam generator thermal demand to predict loop flow, pressure, and inlet temperatures to the steam generator and reactor lower plenums. The pressurizer is modeled to include the effects of spray flow, heater output, and surge flow from the hot leg in determining system pressure.

The primary side has both counter-current and parallel flow sections. The secondary side includes a downcomer region where steam generator level and consequent feedwater demand is determined.

In addition to these models, various control and safety system components are included as necessary to analyze the desired transients. A separate hot channel model is also included to trace the departure from nucleate boiling (DNB) during transients. The control systems are described in Section 5.0 and the hot channel model is outlined in Section 6.0.

The present version of PTSPWR2 is written in FORTRAN IV and is readily available for use by Exxon Nuclear personnel and/or their customers. The output from the code includes both tabular information and plots of the major plant parameters in a format easily comparable with existing licensing information. The code usage with input and output descriptions is given in Section 8.0.

PTSPWR2 was originally submitted for NRC Staff review in support of the H.B. Robinson Unit 2 nuclear plant as the initial issue of XN-74-5. This report was revised and supplemented during subsequent applications. User-oriented information was added to the basic report, which was reissued for internal use.(1) During the review of the critical ENC reload for D.C. Cook Unit 2, the Reference 1 report was provided to the NRC Staff as the latest available description of PTSPWR2.

Table 1.1 PTSPWR2 Transient Analyses

NOTE:

Text sections in the body of the report describe the analysis of other transients which are included for document completeness. Only the transients in this table are identified for PTSPWRs licensing applicability.

2.0 PRIMARY LOOP MODEL

The primary loop is modeled in four parts. They are:

- Core Kinetics
- Core Thermal Hydraulics
- Primary Loop Hydrodynamics and Transport Delays
- Pressurizer

Each of these models is described in the following subsections.

2.1 CORE KINETICS MODEL

The transients analyzed by the PTSPWR2 code result in reactivity insertions of less than 1.0$. The core kinetics are adequately represented by a point kinetics model with 6 delay groups.

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:

$$\delta k = \delta k_D + \delta k_M + \delta k_B + \delta k_{PR} \qquad (2.2)$$

where:

δk_D = Doppler reactivity change ($)

$\delta k_D = \alpha_D (T_f - T_{fIC})$

δk_M = Moderator reactivity change ($)

$\delta k_M = \alpha_M (T_M - T_{MIC})$

δk_B = Shim boron reactivity change ($)

$\delta k_B = \alpha_B (B - B_{IC})$

δk_{PR} = Pressure reactivity change ($)

$\delta k_{PR} = \alpha_{PR} (P_{cr} - P_{crIC})$

δk_{CR} = Control rod reactivity change ($)

= RCCA control and scram reactivity

$\delta k_{CR} = \delta k_{CRC} + \delta k_{CRS}$

$\delta k_{CRS} = f\{t - (t_s + t_{sdc})\}$*

where:

t_s = time at which scram is initiated

t_{sdc} = time delay before control rods begin to move.

[
]

[

]

The heat from U_{235} fission product decay is computed by a relationship of the form of a summation of eleven decay equations. Decay heat from the actinides U_{239} and Np_{239} are accounted for by adding two additional decay equation terms to the eleven equations for fission product decay heat. The models used for the fission product and actinide decay heat generation are described in Reference 3.

2.2 CORE THERMAL-HYDRAULICS

The PTSPWR2 core thermal-hydraulics model calculates the following parameters:

- Core Average Fuel Temperature
- Core Average Coolant Temperature
- Core Exit Coolant Temperature

2.2.1 Fuel Temperatures

[

]

The transient temperature for node i, $T_{f,i}$, is given by the heat balance for the node i:

$$C_{f,i} \frac{dT_{f,i}}{dt} = Y_{i-1} (T_{f,i-1} - T_{f,i}) + Y_i (T_{f,i+1} - T_{f,i}) + Q_i \qquad (2.4)$$

where:

$C_{f,i}$ = thermal capacitance of fuel in node i per unit length of pin (Btu/ft/°F)

t = time (sec)

Y_i = thermal admittance of fuel in node i per unit length of pin (Btu/ft/sec/°F)

Q_i = heat generation rate in node i per unit length of pin (Btu/ft/sec)

where:

h_g = heat transfer coefficient between fuel and clad (BTU/sec/ft^2/°F)

A_g = effective fuel-clad gap heat transfer area per unit length of pin (ft^2/ft)

h_{cc} = convection coefficient between clad and coolant (from average channel thermal-hydraulics model, (BTU/sec/ft^2/°F)

A_h = active clad-coolant heat transfer area per unit length of pin (ft^2/ft)

$K_{f,i}$ = fuel thermal conductivity (BTU/sec/ft/°F)

$K_{f,i} = K_f \{T_{f,i}\}$

K_{cl} = cladding thermal conductivity (BTU/ft/°F)

The fuel capacitances are given by:

where:

$\rho_{f,i}$ = density of fuel in node i (lb/ft^3)

$C_{pf,i}$ = specific heat of fuel in node i (Btu/lb/°F)

$V_{f,i}$ = volume of fuel in node i per unit length of pin (ft^3/ft)

The thermal capacitance at the clad node is:

$$C_{cl} = \rho_{cl} \, CP_{cl} \, V_{cl} \qquad (2.12)$$

where:

ρ_{cl} = density of the clad (lb/ft^3)

CP_{cl} = specific heat of clad (Btu/lb/°F)

V_{cl} = volume of clad per unit length of pin (ft^3/ft)

For the fuel, temperature dependent values of specific heat and thermal conductivity were used to evaluate Y_i's, and $C_{f,i}$'s. Then the following equations and definitions were used to solve for the estimated fuel and cladding temperatures.

where:

T_{ca} = core coolant average temperature

The average fuel temperature supplied to the neutron kinetics model is given by: [ ] (2.17)

2.2.2 Average Channel Thermal-Hydraulics

The average channel thermal analysis yields the core exit and average coolant temperatures. [ ]

The total heat input to the primary coolant is given by:

[ ] (2.18)

The total enthalpy change over the length of the core is:

$$\Delta H_{core} = QT/W_{]pcr} \quad (Btu/lb) \tag{2.19}$$

The average enthalpy at nodal junction i is given by:

$$HC(i) = H_{c,in} + \Delta H_{core}\ INPF(i) \tag{2.20}$$

where:

$INPF(i)$ = integral under the axial power profile curve from the core inlet to nodal junction i

The integral is approximated by:

[

]

where PF(i) are the points on the axial power profile curve. The axial profile is entered as input data and integrated internally.

The moderator temperature is determined for each nodal junction node from the corresponding enthalpy and subcooled water properties table.

$$T_{c(i)} = TWSC \left\{ HC(i) \right\} \qquad (2.22)$$

The core average moderator temperature is given by:

(2.23)

2.3 PRIMARY LOOP HYDRODYNAMICS MODEL

The primary loop model calculates the following parameters:

1. Primary Loop Flow
2. Primary Loop Pressure
3. Coolant Temperature at the steam generator inlets
4. Coolant temperature at the reactor inlet
5. Boron concentration in the core

2.3.1 Primary Loop Flow

The following assumptions are made to model the primary system flow.

balance around each loop. The momentum balance is split into two sections:

Around each loop the net pressure drop must be zero, hence:

$$\Delta P_{pump} - RV_{head} + PL_{head} - \Delta PRV - \Delta PLP - \Delta PKERV - \Delta PKELP = 0 \qquad (2.26)$$

The terms for the reactor vessel are:

(2.26a)

$$\Delta PRV = K_{rv}\,\overline{V}_{cr}\,W_{1pcr}\,\left|W_{1pcr}\right| \qquad (2.26b)$$

$$\Delta PKERV = \frac{1}{144\,g_c}\sum_{RV}\left(\frac{L}{A}\right)\frac{d\left(W_{1pcr}\right)}{dt} \qquad (2.26c)$$

where:

$$\frac{d\left(W_{1pcr}\right)}{dt} = \frac{d\left(W_{\ell p1}\right)}{dt} + \frac{d\left(W_{\ell p2}\right)}{dt}$$

For the remainder of the loop, the equations are:

(2.26d)

where:

$$PL_{head} = \frac{Z_{c\ell}}{144\,V_{c\ell}} + \frac{Z_{sg}}{144\,\overline{V}_{sg}} - \frac{Z_{h\ell}}{144\,V_{h\ell}} \qquad (2.26e)$$

$$\Delta PLP = \left[K_{c\ell}\left(V_{c\ell}\right) + K_{sg}\left(\overline{V}_{sg}\right) + K_{h\ell}\left(V_{h\ell}\right)\right]\,W_{\ell p}\,\left|W_{\ell p}\right| \qquad (2.26f)$$

$$\Delta PKELP = \frac{1}{144\ g_c} \sum_{LP} \left(\frac{L}{A} \right) \frac{d\left(W_{2D}\right)}{dt} \qquad (2.26g)$$

Combining all the pressure drop components and solving for the flow derivative in loop 1 yields:

and for loop 2.

To simplify, the following variables are defined:

Using the new variables and substituting the flow derivative from the loop 2 equation, $\frac{d(W_{\ell p2})}{dt}$, into the loop 1 equation yields:

where

The loop 2 flow derivative is then given by:

(2.30)

Integration of the two flow derivative equations yields the loop flow at any time.

2.3.2 Primary Loop Temperatures and Specific Volumes

The temperature of liquid in the upper plenum is given by a first order linear differential equation representing a mixing volume.

$$T_{up} + \tau_{up} \frac{d(T_{up})}{dt} = T_{cro} \qquad (2.31)$$

where T_{cro} = core outlet temperature

τ_{up} = time constant for flow mixing in the upper plenum

$$\tau_{up} = \frac{CF_{up}}{W_{rv}(V_{up})}$$

CF_{up} = Volume of the upper plenum (ft^3)

V_{up} = Specific volume of fluid in the upper plenum (ft^3/lb)

$V_{up} = f\{T_{up}\}$

W_{rv} = Total flow through the vessel (lb/sec)

(2.32)

The mixed temperature in the steam generator inlet plenum is given by:

$$T_{lpi} + \tau_{pi} \frac{d(T_{lpi})}{dt} = T_{h\&ol} \qquad (2.33)$$

where $\tau_{pi} = \frac{CF_{slpi}}{W_{\&pl}(V_{lpi})}$

$V_{lpi} = f\{T_{lpi}\}$

CF_{slpi} = volume of the loop-one steam generator inlet plenum (ft^3)

The temperature of the primary fluid at the steam generator discharge, T_{1po}, is determined in the steam generator model (Section 3.1). The steam generator discharge plenum temperature is given by:

$$T_{c\ell 1i} + \tau_{po} \frac{T_{c\ell 1i}}{dt} = T_{1po} \qquad (2.34)$$

where $\tau_{po} = \frac{CF_{s1po}}{W_{\ell p1}(V_{1po})}$

$V_{1po} = f\{T_{1po}\}$

CF_{s1po} = Volume of the loop 1 steam generator outlet plenum (ft^3)

(2.35)

A similar set of equations are used to determine the temperature in the second primary loop.

(2.36)

The bulk lower plenum fluid temperature (and, thus, the core inlet temperature) is given by integrating:

$$\dot{T}_{cri} = \frac{VF_{sc}\ \{T_{ci(I)}\}\ W_{lpcr}\left(T_{lpi} - T_{ci(I)}\right)}{CF_{lp}} \qquad (2.37)$$

CF_{lp} = volume of lower plenum. (ft^3)

The hot leg and cold leg specific volumes are determined from the average temperatures.

$$V_{h\ell 1} = VF_{sc}\ \{T_{h\ell 1}\}$$

$$V_{c\ell 2} = VF_{sc}\ \{T_{c\ell 1}\}$$

2.3.3 Pressurizer Surge Flow

The surge flow into, or out of, the pressurizer is dependent on a mass balance on the primary loop.

A mass balance on the primary loop, neglecting compressibility of water, yields:

$$\frac{d(M_{p\ell})}{dt} = \sum W_{in} - \sum W_{out} \qquad (2.38)$$

where $\sum W_{in} = W_{c\ell i} + W_{h\ell i} + W_{mup}$

$W_{c\ell i}$ = cold leg safety injection flow

$W_{h\ell i}$ = hot leg safety injection flow

W_{mup} = make up water flow

$$\sum W_{out} = W_{uppr} + W_{leak} + W_{prsp}$$

W_{uppr} = pressurizer surge flow

W_{leak} = total primary system leakage flow

W_{prsp} = pressurizer spray flow

hence,

$$\frac{d(M_{p\ell})}{dt} = W_{c\ell i} + W_{h\ell i} + W_{mup} - W_{uppr} - W_{leak} - W_{prsp} \qquad (2.39)$$

Solving for the surge flow:

$$W_{uppr} = W_{c\ell i} + W_{h\ell i} + W_{mup} - W_{leak} - W_{prsp} - \frac{d(M_{p\ell})}{dt} \qquad (2.40)$$

The change in fluid mass stored in the primary system, $\frac{d(M_{p\ell})}{dt}$, is formed from:

(2.41)

2.3.4 Primary Loop Boron Concentrations

The propagation of boron throughout the primary loop is modeled using relationships similar to those employed for fluid temperature propagation (Equations 2.31 through 2.37). The boron injection with appropriate time delay takes place in the reactor vessel lower plenum. First order linear differential equations are used to describe the

volumetric mixing in the steam generator and reactor vessel components.

Therefore, boron transport and accumulation takes place throughout the entire primary loop, including the reactor core, where its reactivity effects are calculated.

2.4 PRESSURIZER MODEL

The new model introduces two empirical factors that may be changed from their nominal values should subsequent benchmarking so dictate.

Pressurizer Model Theory

The heat and mass flows into and out of the pressurizer are defined by Figure 2.3.

The governing equations include conservation of mass, conservation of energy, and appropriate state equations, being given by ASME steam tables for each phase. Additional relationships are used to model spray flow, relief valve flow, safety valve flow, and heating rate.

Mass Balance

(a) Liquid Part

(2.42)

(b) Vapor Part

(2.43)

where W_{epr} = Flash or boiling rate, lb/sec

W_{uppr} = Insurge or outsurge determined by primary loop balance, lb/sec

W_{prsp} = Spray flow, lb/sec

W_{prrv} = Relief valve flow, lb/sec

W_{prsv} = Safety valve flow, lb/sec

M_{spr} = Mass vapor in pressurizer

M_{wpr} = Mass liquid in pressurizer

W_{cond} = Rate of condensation in vapor region, lb/sec

Energy Balance

General Form $\dot{H} = \dot{M}h + M\dot{h} = \dot{Q} + V\dot{p} + \Sigma(Wh)_i$ (2.44)

$\dot{H}$ = Total rate of change of enthalpy

where $\dot{Q}$ = Heating rate influx

$\Sigma(Wh)_i$ = Energy inflows (outflows) due mass inflows (outflows) to a control volume

V = Volume of region

$\dot{p}$ = Change in pressure of system

h = Specific enthalpy

Application of energy equation:

(a) Liquid

(2.45)

(b) Vapor

(2.46)

where h_{spr} = Enthalpy of pressurizer steam

h_w = Enthalpy of pressurizer water

h_i = Enthalpy of surge liquid; for insurge h_i = enthalpy of subcooled liquid from reactor upper plenum; for outsurge $h_i = h_w$

V_w = Water volume

V_s = Steam volume

h_f = Saturated water enthalpy

h_g = Saturated steam enthalpy

$h_{c\ell 1}$ = Cold leg fluid enthalpy

$\dot{Q}_H$ = Heater power

and where (2.47)

Overall Volume Constraint

$$\dot{V} = v_{spr}\dot{M}_{spr} + M_{spr}\dot{v}_{spr} + v_w\dot{M}_{wpr} + M_{wpr}\dot{v}_w = 0. \quad (2.48)$$

where v_{spr} = Steam specific volume

v_w = Water specific volume

State Equations

(a) Liquid

$v_w = v_w(p, T_w)$ Specific volume, ft^3/lb (2.49)

$h_w = h_w(p, T_w)$ Specific enthalpy, Btu/lb (2.50)

(b) Vapor

$v_{spr} = v_{spr}(p, T_s)$ Specific volume, ft^3/lb (2.51)

$h_{spr} = h_{spr}(p, T_s)$ Specific enthalpy, Btu/lb (2.52)

If the steam region is saturated, then the following additional relation applies:

(2.53)

If the steam region is superheated, then

(2.54)

If the water is at saturation, then

(2.55)

If the water region is subcooled, then

(2.56)

Equations 2.42 through 2.56 are solved simultaneously for each time step by an iterative process.

3.0 STEAM GENERATOR MODEL

Two models were developed for the steam generators.

The primary coolant has both parallel and counter-flow sections.

(3.1)

A separate hydrodynamics model was developed to establish secondary flow pressure and water level in the steam generator. Each of these models are described for one steam generator. Similar equations are utilized to describe conditions in the second loop.

3.1 STEAM GENERATOR HEAT TRANSFER MODEL

Energy balances on primary fluid nodes results in the following equations for node temperature change.

(3.2)

3.2 STEAM GENERATOR HYDRODYNAMICS MODEL

The steam generator hydrodynamics model is developed from the schematic shown in Figure 3.2. The steam generator secondary side, upper plenum, and steam dome are compressible nodes, the remainder of the loop is liquid and assumed incompressible.

A mass and energy balance are performed on each compressible node to determine the pressure derivative in the node. A momentum balance is made between each node around the recirculation loop to determine the secondary fluid flow through the steam generator.

The following equations summarize the calculations for steam generator number 1. Similar equations are used for steam generator number 2.

Steam Dome

(3.25)

(3.26)

(3.27)

(3.28)

(3.29)

(3.30)

(3.31)

(3.32)

4.0 STEAM LINE MODEL

The steam line model of PTSPWR2 solves the equations for the mass, energy and momentum balances. The basic equations for a compressible node, j, are:

[

(4.1)

(4.2)

(4.3)

(4.4)

]

[where:

(4.5)

(4.6)]

The gravitational head is neglected in the steam line nodes. The steam line is nodalized as shown in Figure 4.1. Equations (4.1) thru (4.4) are applied to each node. The following is a summary of the equations for steam line number 1 and the steam header. Similar equations are used for the second steam line.

[

(4.7)

(4.8)

(4.9)

(4.10)

]

5.0 CONTROL AND SAFETY SYSTEM MODELS

The PTSPWR2 code simulates those plant control and safety systems components necessary to adequately model a broad spectrum of transients. The following systems are currently modeled in PTSPWR2.

- Trip Signal Logic
- Pressurizer and Steam System Relief and Safety Valves
- Pressurizer Level Control
- Pressurizer Pressure Control
- Feedwater Controls
- Turbine Pressure Regulator Controls
- Steam Dump System
- Rod Speed Control

Each application of PTSPWR2 to a specific reactor will require review of the control and safety system to determine if components should be added to the model or their functional form changed.

5.1 TRIP SYSTEM LOGIC

The signals which can be used to generate a reactor trip are listed in Table 5.1. When one of the above trip signals is received, calculation of negative reactivity due to control rod insertion begins. The reactivity feedback has the functional form:

[(5.1)]

where:

t_s = time the trip signal is received

t_{dsc} = Instrumentation time delay before control rods begin to move.

The time delay used in each accident analysis is based on conservatively selected values for maximum instrumentation delay.

The control reactivity vs time for a particular case is developed by establishing the reactivity vs rod position in the core from neutronics calculations.

5.2 VALVE FLOWS

The pressurizer relief and safety valves are actuated by the pressurizer pressure exceeding their respective set points.

The relief valve model allows for hysteresis in the opening and closing characteristics.

Relief valves open if: $P_{pr} \geq P_{prrv1}$

Relief valves close if: $P_{pr} \leq P_{prrv2}$

The relief valves can also have different opening and closing rates specified as depicted in Figure 5.3(a). The valve flow with the valve

(5.2)

The fully open normalized valve flow is modified for valve position.

(5.3)

The relief valve flow is then given by:

(5.4)

The steam generator relief valve flows are calculated in a similar manner. Different flow coefficients and opening and closing characteristics are specified.

The spring loaded safety valves for the pressurizer and steam generators are modeled in a manner similar to the relief valves except the opening and closing characteristics are different as shown in Figure 5.3(b). The flow coefficients are determined such that the total flows from each set of safety valves are lumped together. The safety valve flows are also modified for valve position as in Equations (5.2 and 5.3).

The steam line isolation valve flow coefficient as a function of time after trip is modeled as shown in Figure 5.3(c). The isolation valve flow is calculated as discussed in Section 4.0.

The turbine stop valves are tripped manually. When the valves have been tripped, the valve position as a function of time is as shown in Figure 5.3(d).

The closing characteristics are again used.

[(5.5)]

and the turbine flow is given by:

[(5.6)]

where W_{tot} is the flow demand from the pressure regulator discussed later in Section 5.6.

5.3 PRESSURIZER LEVEL CONTROLS

Functionally, the pressurizer level control adjusts the primary loop makeup water flow to maintain the pressurizer water level within the desired range. A block diagram of the control circuit used is shown in Figure 5.4(a). The programmed pressurizer level is as shown in Figure 5.4(b). The no-load and full-load average temperatures and levels are obtained from plant design calculations or operating data.

5.4 PRESSURIZER PRESSURE CONTROL

The pressurizer pressure is controlled by activating either the heating elements in the bottom of the pressurizer to increase the pressure or spraying water from the primary loop cold leg into the top of the pressurizer to reduce the pressure.

The heater output is either off or on dependent on the pressure and water volume in the pressurizer.

The spray flow characteristics as a function of pressurizer pressure are shown in Figure 5.5.

5.5 FEEDWATER CONTROLLER

Functionally, the feedwater controller adjusts the feedwater flow to maintain the steam generator water level within the desired range. A schematic of the PTSPWR2 3-element feedwater controller is shown in Figure 5.6. The inputs to the controller are steam flow, feedwater flow, steam generator level, and the desired setpoints. The level setpoint is a function of load. The generated level setpoint deviation is combined with a steam/feedwater flow mismatch to give the required feedwater valve position. The amount of feedwater flow is a function of the valve position.

5.6 TURBINE PRESSURE REGULATOR

The pressure regulator in the PTSPWR2 code is shown in Figure 5.7. The pressure regulator thus adjusts turbine flow to try to maintain turbine throttle pressure at a preset value.

A proportional plus integral controller is used for the pressure regulator. The control signal is formed by:

[(5.7)

(5.8)]

where:

[ ]

KP_{pr} = proportional gain

KI_{pr} = integral gain

IC_{pr} = initial condition

$IC_{pr} = W_{tbo}/W_{tbmax}$

The controller output is then converted to total flow demand.

$$W_{tot} = C_{pr} * W_{tbmax} \quad (5.9)$$

The turbine flow is then determined by Equation 5.6.

5.7 STEAM DUMP CONTROLS

The PTSPWR2 code has two functional modes of steam dump operation: 1) loss of load steam dump control and 2) turbine trip steam dump control. Diagrams of the steam dump systems are shown in Figure 5.8.

The loss of load steam dump controller allows bypass to the condenser and/or atmospheric dump following a load rejection incident. The inputs into the model are the primary coolant average temperature and a reference temperature. Bypass valve position and flow is modulated as a function of the non-linear gain associated with the mismatch of the two signals.

5.8 ROD SPEED CONTROL SYSTEM

The rod speed control system allows simulation of automatic reactor control. In the model, the rod speed (reactivity insertion) is a function of the reactor power, coolant average temperature, and pressurizer pressure. These signals are compared to nominal and/or preset valves in the logic train, the output of which determines the rod speed and position and consequently the amount of reactivity addition or deletion required to compensate for the error signal. A schematic of the rod speed program is shown in Figure 5.3.

TABLE 5.1

TRIP SIGNALS INCORPORATED IN PTSPWR2

High Power

High Pressurizer Pressure

Low Pressurizer Pressure

High Pressurizer Water Level

Low Primary Coolant Flow

Low-Low Steam Generator Level

Turbine Stop Valve Closure

Isolation Valve Closure

Overtemperature ΔT*

Overpower ΔT*

---

* The overtemperature and overpower ΔT trip setpoints are functional setpoints, varying as a function of primary loop conditions. Their logic train schematics are depicted in Figures 5.1 and 5.2.

6.0 HOT CHANNEL ANALYSIS

To determine the possibility of departure from nucleate boiling (DNB) occurring on the fuel rods during a transient, a hot channel model is included as a supplement to PTSPWR2.

In addition to the axial profile, the peaking factors for heat flux and enthalpy rise must be specified.

PF_ϕ = Total heat flux peaking factor

PF_ϕ = PFR x PFL x PFE

Where

PFR = Radial peaking factor

PFL = Local peaking factor

PFE = Engineering peaking factor.

PF_ϕ can contain other components as required for a particular application.

PF(ENTH) = Total enthalpy rise peaking factor

PF(ENTH) = PFW x PFIW x PFR x PFL

Where

PFW = Flow maldistribution factor

PFIW = Inlet flow maldistribution factor

(6.1)

Where

The enthalpy rise in the hot channel is given by

(6.2)

The enthalpy in each node is then given by

(6.3)

The thermodynamic quality of the hot channel node is

$$X_{hc(i)} = (H_{hc(i)} - HF_c)/HFG_c \qquad (6.4)$$

Core average heat flux is given by

$$QOA = \frac{QT * 3600}{A_H} \qquad (BTU/hr/ft^2) \qquad (6.5)$$

Where A_H = total clad heat transfer area (ft^2)

The hot channel nodal heat flux is then given by

$$PHI(i) = QOA * PF(i) * PF_\phi \tag{6.6}$$

The mass flow rate in the hot channel is given by

$$G_{hc} = W_{chc} * 3600/A_{ceff} \ (lb/hr\text{-}ft^2) \tag{6.7}$$

The DNB heat flux at each axial core node is determined by using either the W-3[2] correlation or the Exxon Nuclear DNB[4] correlation. The W-3 correlation is formulated as follows:

$$QDNB(i) = \left\{(2.022 - 0.0004302\ P_{cr}) + (0.1722 - 0.0000984\ P_{cr})\right.$$

$$\left. * \left[EXP(18.177 - 0.004129\ P_{cr}) * X_{hc(i)}\right]\right\}$$

$$*\left[G_{hc} * \left(0.1484 - 1.596\ X_{hc(i)} + 0.1729\ X_{hc(i)}\left|X_{hc(i)}\right|\right) + 1037000.\right]$$

$$*\left[1.157 - 0.869\ X_{hc(i)}\right] * \left[0.2664 + 0.8357\ EXP(-3.151\ D_e)\right]$$

$$*\left[0.8258 + 0.000794\ (HF_{cr} - H_{cin})\right] \tag{6.8}$$

The ratio of actual to DNB heat flux is then given by

$$DNBR(i) = QDNB(i)/PHI(i) * F \tag{6.9}$$

where

F = Correction for nonuniform channel heating

And the minimum ratio in the core is:

MDNBR = Minimum of DNBR(i)

The code user also has the option of using the XNB correlation when applicable. A description of the correlation and the data used for its benchmarking is found in Reference 4.

7.0 NOMENCLATURE

Symbol	Description	Units
A	General term for area	ft^2
AA_{dosl}	Flow acceleration factor between steam generator dome and steam line	ft^{-4}
A_B	Boron reactivity coefficient correction factor	\$/ppm
A_{cssg1}	Secondary coolant cross sectional area in steam generator 1	ft^2
$A_{c,eff}$	Effective core flow area for heat transfer in DNB calculation	ft^2
A_g	Effective fuel-clad gap heat transfer area per unit length	ft^2/ft
A_h	Active clad-coolant heat transfer area per unit length of pin	ft^2/ft
A_H	Total clad heat transfer area	ft^2
A_j	Cross sectional area flow in node j	ft^2
A_k	Cross sectional area for flow in node k	ft^2
A_{mt}	Mean heat transfer area through steam generator tube wall	ft^2
A_{pi}	Heat transfer area of inner tube surface in steam generator tube walls	ft^2
A_o	Heat transfer area on secondary side of steam generator tubes	ft^2
A_{tdsg1}	Fluid cross sectional area of steam generator 1 external to steam separators (area at top of downcomer)	ft^2
A_{1ps1}	Heat transfer area in the parallel flow section of steam generator 1 exposed to subcooled water on the secondary side	ft^2

7.0 NOMENCLATURE (Continued)

Symbol	Description	Units
A_{lps12}	Heat transfer area in the parallel flow section of steam generator 1 exposed to boiling water on the secondary side	ft^2
A_{lps23}	Heat transfer area in steam generator 1 exposed to steam on the secondary side	ft^2
A_{lps34}	Heat transfer area in the counter-flow section of steam generator 1 exposed to boiling water on the secondary side	ft^2
A_{lps40}	Heat transfer area in the counter-flow section of steam generator 1 exposed to subcooled water on the secondary side	ft^2
B	Boron concentration	ppm
B_{IC}	Initial boron concentration	ppm
C_{cl}	Thermal capacitance per unit length of clad	Btu/ft^oF
CF_{dol}	Volume of steam dome in steam generator 1	ft^3
CF_j	Total volume in node j	ft^3
CF_{lp}	Volume of reactor vessel lower plenum	ft^3
CF_{off}	Pressurizer heater turns off if water volume is less than this amount	ft^3
CF_{on}	Pressurizer heater turns on if water volume is greater than this value	ft^3
$CF_{p,TOT}$	Total primary fluid volume in the steam generator tubes	ft^3
CF_{sgup}	Volume of steam generator upper plenum	ft^3

7.0 NOMENCLATURE (Continued)

Symbol	Description	Units
CF_{sg1T}	Volume of the active secondary side of steam generator 1	ft^3
CF_{sh}	Volume of steam header node in steam line	ft^3
CF_{sl1}	Volume of first node in steam line 1	ft^3
CF_{sl2}	Volume of node 2 in steam line 1	ft^3
CF_{spr}	Volume of steam in pressurizer	ft^3
CF_{slpi}	Liquid volume in the inlet plenum of steam generator 1	ft^3
CF_{slpo}	Liquid volume in the outlet plenum of steam generator 1	ft^3
CF_{up}	Total volume (including dead space) of liquid in reactor vessel upper plenum	ft^3
CF_{upl}	Volume of upper plenum in steam generator 1	ft^3
CF_{wpr}	Volume of water in the pressurizer	ft^3
CF_{lpi1}	Volume of primary coolant node in steam generator 1 in parallel-flow region surrounded by subcooled liquid on the secondary side	ft^3
CF_{lp12}	Volume of primary coolant node in steam generator 1 in parallel-flow region surrounded by boiling liquid on the secondary side	ft^3
CF_{lp23}	Volume of primary coolant node in steam generator 1 in the region surrounded by steam on the secondary side	ft^3
CF_{lp34}	Volume of primary coolant node in steam generator 1 in the counter-flow region surrounded by boiling liquid on the secondary side	ft^3
CF_{lp4o}	Volume of primary coolant node in steam generator 1 in the counter-flow region surrounded by subcooled water on the secondary side	ft^3

7.0 NOMENCLATURE (Continued)

Symbol	Description	Units
CF_{1sB}	Volume of boiling region on secondary side of steam generator 1	ft^3
CF_{1sNB}	Volume of non-boiling region on secondary side of steam generator 1	ft^3
CF_{1sST}	Volume of steam region in secondary side of steam generator 1	ft^3
C_i	Power equivalent of concentration of isotopes in ith delayed neutron group	Mw
C_{iv}	Relative position of isolation valve	---
C_{lc}	Normalized feedwater flow demand	---
CP_{cl}	Specific heat of clad	Btu/lb/^oF
CP_f	Specific heat of fuel	Btu/lb/^oF
C_{pr}	Normalized steam flow demand	---
$CP_{wtb\{T\}}$	Specific heat of primary coolant as a function of temperature	Btu/lb/^oF
CP_{1pi1}	Specific heat of primary coolant in steam generator 1 in region between inlet and first node	Btu/lb/^oF
CP_{1p12}	Specific heat of primary coolant in steam generator 1 in volume between nodes *1* and *2*	Btu/lb/^oF
CP_{1p23}	Specific heat of primary coolant in steam generator 1 in region between nodes *2* and *3*	Btu/lb/^oF
CP_{1p34}	Specific heat of primary coolant in steam generator 1 in region between nodes *3* and *4*	Btu/lb/^oF
CP_{1p4o}	Specific heat of primary coolant in steam generator 1 in the region between node *4* and outlet	Btu/lb/^oF
CW_{prrv}	Flow coefficient for pressurizer relief valve	ft^2
$C_{f,1}$	Thermal capacitance of fuel node *1*	Btu/lb/^oF
$C_{f,2}$	Thermal capacitance of fuel node *2*	Btu/lb/^oF

7.0 NOMENCLATURE (Continued)

Symbol	Description	Units
$C_{f,3}$	Thermal capacitance of fuel node 3	Btu/ft/^{o}F
$C_{f,4}$	Thermal capacitance of fuel node 4	Btu/ft/^{o}F
D_{ei}	Equivalent steam generator tube diameter in modified Dittus-Boelter correlation	ft
D_{h}	Hydraulic diameter of core coolant channels used in modified Dittus-Boelter correlation	ft
$DHDP_{dl}$	Change of enthalpy with respect to pressure for the mixed fluid in dome of steam generator 1	Btu in^2/lb^2
$DHDP_{sh}$	Change of enthalpy with respect to pressure for fluid in steam header node	Btu in^2/lb^2
$DHDP_{upl}$	Change of enthalpy with respect to pressure for fluid in upper plenum of steam generator 1	Btu in^2/lb^2
$DHDP_{1I}$	Change of enthalpy with respect to pressure in node I of steam line 1	Btu in^2/lb^2
$DHDP_{12}$	Change of enthalpy with respect to pressure for fluid in node 2 of steam line 1	Btu in^2/lb^2
$DHDV_{dl}$	Change of specific enthalpy with respect to specific volume for the transition of liquid to gas, i.e., $(\partial h_{fg}/\partial v_{fg})$, at the pressure in the dome of steam generator 1	Btu/ft^3
$DHDV_{gl}$	(h_{fg}/v_{fg}) at the pressure in steam generator 1	Btu/ft^3
$DHDV_{sh}$	(h_{fg}/v_{fg}) at the pressure in steam header	Btu/ft^3
$DHDV_{upl}$	(h_{fg}/v_{fg}) at the pressure in the upper plenum of steam generator 1	Btu/ft^3
$DHDV_{1I}$	(h_{fg}/v_{fg}) at the pressure of steam node I in steam line 2	Btu/ft^3
$DHDV_{12}$	(h_{fg}/v_{fg}) at the pressure of steam in node 2 in steam line 1	Btu/ft^3
$DHFDPT\{P_{sgl}\}$	Change in specific enthalpy of saturated liquid water with respect to pressure as a function of pressure in steam generator 1	Btu in^2/lb^2

7.0 NOMENCLATURE (Continued)

Symbol	Description	Units
$DHFP_{g1}$	Change in specific enthalpy of saturated water with respect to pressure in steam generator 1	$Btu/in^2/lb^2$
$DHGP_{g2}$	Change in specific enthalpy of saturated steam with respect to pressure in steam generator 1	$Btu/in^2/lb^2$
DLTMN	Minimum control signal used in pressure regulator control	lb/in^2
DLTPMX	Maximum control signal used in pressure regulator control	lb/in^2
DNBR(i)	Rates of actual to DNB heat flux at node i	
d_o	Outer diameter of steam generator tubes used in calculation of heat transfer area	ft
DT_{1Mi1}	Mean temperature difference for heat transfer in steam generator 1 between inlet and node 1	^oF
DT_{1M12}	Mean temperature difference for heat transfer in steam generator 1 between nodes 1 and 2	^oF
DT_{1M23}	Mean temperature difference for heat transfer in steam generator 1 between nodes 2 and 3	^oF
DT_{1M34}	Mean temperature difference for heat transfer in steam generator 2 between nodes 3 and 4	^oF
DT_{1M4o}	Mean temperature difference for heat transfer in steam generator 2 between nodes 4 and outlet	^oF
DVP_{rv}	Fraction open for pressurizer relief valve	---
DVP_{sv}	Fraction open for pressurizer safety valve	---
DVP_{Tsv}	Fraction open for turbine stop valve	---
E_{lc}	Error signal in feedwater flow control	---
E_{pr}	Error signal in turbine throttle pressure control	---
f	Fouling factor for steam generator tubes	$hr\ ft^2\ ^oF/BTU$
F	Correction factor for non-uniform channel heating in calculation of DNBR	---

7.0 NOMENCLATURE (Continued)

Symbol	Description	Units
FDELAY	"Flow" delay for time required for enthalpy change (e.g.) to traverse large volume	---
f(Δq)	Overtemperature and overpower ΔT input signal as a function of axial offset	---
g_c	Gravitational constant	lbm ft/lbf/sec^2
G_{hc}	Hot channel mass flow rate	lb/hr/ft^2
h	Specific enthalpy (general term)	Btu/lb
H_{bd1}	Specific enthalpy in the secondary side of steam generator 1 at the bottom of the downcomer	Btu/lb
h_{cc}	Clad-coolant heat transfer coefficient	Btu/sec/^{o}F/ft^2
HC(i)	Average specific enthalpy at nodal junction i in core	Btu/lb
$H_{c,in}$	Average specific enthalpy of fluid entering core	Btu/lb
H_{cl1}	Specific enthalpy of liquid in loop 1 cold leg	Btu/lb
H_{co}	Specific enthalpy of fluid leaving core	Btu/lb
H_{do1}	Specific enthalpy of fluid in the dome of steam generator 1	Btu/lb
HF_c	Enthalpy of saturated liquid water at core average pressure	Btu/lb
HFG_c	Enthalpy change associated with change of phase from liquid to gas at the core average pressure	Btu/lb
$HFGP\{P_{up1}\}$	$(\partial h_{fg}/\partial p)$ as a function of pressure in the upper plenum of steam generator 1	Btu in^2/lb^2
$HFG\{P_{pr}\}$	Enthalpy change associated with change of phase from liquid to gas as a function of pressurizer pressure, P_{pr}	Btu/lb
$HFGP\{P_{sl12}\}$	$(\partial h_{fg}/\partial p)$ as a function of pressure at node 2 in steam line 1	Btu in^2/lb^2
HFG_{pr}	Enthalpy change associated with change of phase from liquid to gas at pressurizer pressure	Btu/lb

7.0 NOMENCLATURE (Continued)

Symbol	Description	Units
$HF\{P_{pr}\}$	Enthalpy of saturated liquid water at pressurizer pressure	Btu/lb
$HFP\{P_{sl12}\}$	$(\partial h_f/\partial p)$ for saturated water as a function of pressure in steamline 1, node 2	Btu in^2/lb^2
$HFP\{P_{up1}\}$	$(\partial h_f/\partial p)$ for saturated water as a function of pressure in the upper plenum of steam generator 2	Btu in^2/lb^2
HF_{pr}	Enthalpy of saturated water at pressure of pressurizer	Btu/lb
HF_{sg1}	Enthalpy of saturated water in steam generator 1 on the secondary side	---
HFW1	Enthalpy of feedwater to steam generator 1	Btu/lb
h_g	Heat transfer coefficient across fuel-clad gap	Btu/sec/ft^2/°F
HG_{pr}	Enthalpy of saturated steam in the pressurizer	Btu/lb
$H_{hc(i)}$	Enthalpy of fluid in node i of hot channel	Btu/lb
h_i	Heat transfer coefficient at inner (primary) tube surface in steam generator	Btu/sec/ft^2/°F
$\dot{H}_j$	General term for change in specific enthalpy with respect to time at node j	Btu/lb/sec
H_{lp}	Specific enthalpy of fluid entering the hot channel from the lower plenum	Btu/lb
h_o	Heat transfer coefficient at outer surface of steam generator tubes	Btu/sec/ft^2/°F
$h_{cc,IC}$	Initial steady-state heat transfer coefficient between clad and coolant	Btu/ft^2/sec/°F
$h_{i,IC}$	Steady-state (initial) heat transfer coefficient at outer surface of steam generator tubes	Btu/sec/ft^2/°F
$H_{out,i}$	Specific enthalpy at junction i	Btu/lb

7.0 NOMENCLATURE (Continued)

Symbol	Description	Units
RVHEAD	Potential head of vessel water	psia
H_{sgup1}	Specific enthalpy of fluid in upper plenum, steam generator 1	Btu/lb
H_{sg1}	Average enthalpy of fluid in secondary side of steam generator 1	Btu/lb
H_{sh}	Enthalpy of mixed fluid in steam header node	Btu/lb
H_{sl1}	Enthalpy of fluid in steamline 1, node 1	Btu/lb
H_{sl2}	Enthalpy in steamline 1, node 2	Btu/lb
H_{sl22}	Enthalpy in steamline 2, node 2	Btu/lb
H_{spr}	Enthalpy of steam in the pressurizer	Btu/lb
H_{td1}	Enthalpy of mixed fluid in top of downcomer, steam generator 1 secondary	Btu/lb
H_{up}	Enthalpy of fluid in upper plenum of reactor vessel	Btu/lb
H_{wpr}	Enthalpy of water in the pressurizer	Btu/lb
H_{89sg1}	Enthalpy of steam (quality = 0.89) in steam generator 1	Btu/lb
IC_{fc}	Initial control signal for feedwater controller	---
IC_{pr}	Initial control signal for turbine throttle pressure control	---
INPF(i)	Integrated axial power profile curve from core inlet to nodal junction i	---
Iv_{trip}	Time at which isolation valve trips	---
k	Reactivity of core	$
K_{bdsg1}	Frictional loss coefficient from bottom of downcomer to active region of steam generator 1	psia $sec^2/lb/ft^3$

7.0 NOMENCLATURE (Contined)

Symbol	Description	Units
K_{bdtd}	Frictional loss coefficient from top to bottom of downcomer	psia $sec^2/lb/ft^3$
K_{cl}	Thermal conductivity of fuel clad	Btu/sec/ft/^oF
K_{cl}	Combined momentum and friction loss coefficient for flow in cold leg	psi $sec^2/lb/ft^3$
K_{dosl1}	Momentum and friction loss coefficient for flow from the steam generator vessel to the first node of steamline 1	psi $sec^2/lb/ft^3$
$K_{f,12}$	Thermal conductivity of fuel in the region between nodes 1 and 2	Btu/sec/ft/^oF
$K_{f,23}$	Thermal conductivity of fuel in the region between nodes 2 and 3	Btu/sec/ft/^oF
$K_{f,4}$	Thermal conductivity of fuel in nodal volume 4	Btu/sec/ft/^oF
K_{hl}	Combined momentum and friction loss coefficient for flow in hot leg	psi $sec^2/lb/ft^3$
KI	Integral gain for feedwater flow controller	1/sec
KI_{pr}	Integral gain for turbine throttle pressure controller	1/sec
K_{jk}	General term for combined momentum and friction loss for flow between nodes j and k	psi $sec^2/lb/ft^3$
K_{ca}	Thermal conductivity of primary coolant in the core	Btu/sec/ft/^oF
K_p	Thermal conductivity of primary coolant in steam generator tubes	Btu/sec/ft/^oF
KP_{lc}	Proportional gain in feedwater flow controller	---
KP_{pr}	Proportional gain in turbine throttle pressure controller	---
KQ	Conversion factor used in calculation of volumetric heat generation rate	Btu/MW/sec/ft^3

7.0 NOMENCLATURE (Continued)

Symbol	Description	Units
K_{rv}	Combined momentum and friction loss coefficient for flow through the reactor vessel	psi $sec^2/lb/ft^3$
K_{sg}	Combined momentum and friction loss coefficient for flow through the primary side of the steam generator	psi $sec^2/lb/ft^3$
K_{sgupl}	Combined loss coefficient for flow between active region and upper plenum in steam generator 1	psi $sec^2/lb/ft^3$
K_{shtb}	Combined loss coefficient for steam flow between steam header and turbine	psi $sec^2/lb/ft^3$
$K_{s\ell sh1}$	Momentum and friction loss coefficient for flow from second node to steam header in steamline 1	psi $sec^2/lb/ft^3$
$K_{s\ell sh2}$	Momentum and friction loss coefficient for flow from second node to steam header in steamline 2	psi $sec^2/lb/ft^3$
K_t	Thermal conductivity in steam generator tubes	Btu/sec/ft/^oF
K_{upspl}	Combined loss coefficient for flow between upper plenum and separators in steam generator 2	psi $sec^2/lb/ft^3$
$K_{s\ell I I2}$	Momentum and friction loss coefficient for flow between nodes *I* and *2* in steamline	psi $sec^2/lb/ft^3$
K1 - K7	Gain coefficients for various control systems	
$K\Delta P_2$	Loss coefficient for flow in primary loop 2	psi sec^2/lb^2
$K\Delta PRV$	Loss coefficient for flow in reactor vessel	psi sec^2/lb^2
ℓ	Prompt neutron lifetime	sec
L	General symbol for length	ft
LA_{bdsgl}	Length to area ratio for entrance region of steam generator 1	ft^{-1}
LA_{bdtdl}	Length to area ratio for downcomer region of steam generator 1	ft^{-1}
$LA_{dos\ell l}$	Length to area ratio in the exit region of steam dome in steam generator 1	1/ft

7.0 NOMENCLATURE (Continued)

Symbol	Description	Units
LAivl	Length to area ratio in region of isolation valve in steam line 1	1/ft
$LALP_1$	Inertance coefficient for flow from reactor vessel exit to reactor vessel entrance through loop 1	$lbm/psi/sec^2$
LAM1	Combined coefficient for flow in loop 1	$sec^2/lbm/psi$
LAM2	Combined coefficient for flow in loop 2	$sec^2/lbm/psi$
LARV	Inertance coefficient for flow through reactor vessel	$lbm/psi/sec^2$
LA_{szsh1}	Length to area ratio in the region of node 1 to steam header in steamline 1	1 ft
LA_{szsh2}	Length to area ratio in the region of node 2 to steam header in loop 2	1/ft
LB_1	Length of boiling region in steam generator 1	ft
L_{cr}	Length of active core region	ft
LEV_{pct}	Steam generator level expressed as percent of span	
LEVPR	Water level in the pressurizer	ft
LEV_{sg}	Water level in either steam generator (secondary)	ft
LEV_{sg1}	Water level in steam generator 1	ft
LNB_1	Height of non-boiling (subcooled) region in steam generator 1	ft
L_{prfz}	Maximum programmed level used in pressurized level control	ft
L_{prnz}	Minimum programed level used in pressurizer level control	ft
LST_1	Height of steam region in steam generator 1	ft
LTOP	Total height in steam generator secondary side	ft
M_{do1}	Mass of fluid in steam dome of steam generator 1	lb

7.0 NOMENCLATURE (Continued)

Symbol	Description	Units
MDNBR	Minimum calculated DNBR (i)	---
M_j	General term for mass of fluid in node j	lb
M_{pl}	Mass of fluid in primary loop	lb
M_{sg}	Mass of fluid in active region steam generator 1	lb
M_{shl}	Mass of fluid in steam header	lb
M_{sl1}	Mass of fluid in steam line 1, node 1	lb
M_{sl2}	Mass of fluid in steam line 1, node 2	lb
M_{spr}	Mass of steam in pressurizer	lb
M_{td1}	Mass in node at top of downcomer, steam generator 1	lb
M_{up1}	Mass in upper plenum, steam generator 1	lb
M_{wpr}	Mass of liquid water in pressurizer	lb
N	Genral term for ratio of pump speed to pump speed at rated flow	---
N_{fp}	Number of fuel pins	---
N_{PR}	Prandtl number	---
$N_{pump\,1}$	Ratio of pump speed in loop 1 to pump speed at rated flow	---
$N_{pump\,2}$	Ratio of pump speed in loop 2 to pump speed at rated flow	---
N_{RE}	Reynolds number	---
N_{TUBES}	Number of tubes in steam generator	---
NVF_{rv}	Normalized relief valve flow from the pressurizer	---
NVF_{Tsv}	Normalized flow through turbine stop valve	---

7.0 NOMENCLATURE (Continued)

Symbol	Description	Units
p	General term for pressure	psi
P	Thermal power of core	MWt
P^0	Reference power for rod speed controler	MWt
P_{cr}	Pressure in the core	psia
$P_{cr,IC}$	Pressure in core under initial steady-state conditions	psia
P_{dol}	Pressure in steam dome for steam generator 1	psia
PF_{cr}	Fraction of core power generating heat in fuel	---
PFE	Engineering peaking factor for hot channel analysis	---
PF(ENTH)	Total enthalpy rise peaking factor for hot channel analysis	---
PFIW	Inlet flow maldistribution factor for hot channel analysis	---
PFL	Local power peaking factor for hot channel analysis	---
PFR	Radial power peaking factor for hot channel analysis	---
PFW	Flow maldistribution factor for hot channel analysis	---
PFWD	Ratio of nominal hot channel flow to core average flow for hot channel anslysis	---
PF_ϕ	Total heat flux peaking factor for hot channel flow	---
PHI(i)	Hot channel heat flux at nodal junction i	$Btu/sec/ft^2$
P_j	General term for pressure at node j	psi
PLHEAD	Potential head of water in primary loop	psia
P_{pr}	Average pressure in the pressurizer	psia

7.0 NOMENCLATURE (Continued)

Symbol	Description	Units
P_{prrv1}	Setpoint for pressurizer relief valve opening	psia
P_{prrv2}	Setpoint for pressurizer relief valve close	psia
P_{prset}	Reference pressure for overtemperature ΔT trip and rod speed controllers	psia
P_{prsp}	Setpoint for pressurizer heater turn on depending on pressure	psia
P_{sg1}	Average pressure in secondary side steam generator 1	psia
P_{sh}	Pressure in steam header	psia
P_{sl1}	Pressure at node 1, steamline 1	psia
P_{sl2}	Pressure at node 2, steamline 1	psia
P_{s22}	Pressure at node 2, steamline 2	psia
P_{tt}	Pressure at the turbine throttle	psia
P_{ttsp}	Setpoint for pressure at the turbine throttle	psia
P_{up1}	Pressure in the upper plenum of steam generator 1	psia
PV	Generalized pressure-volume term	---
QDNB(i)	Calculated heat flux for unitial DNB at hot channel node i	Btu/ft^2/sec
Q_{do1}	Heat flow in dome of steam generator 1	Btu/sec
Q_{heater}	Heat output for pressurizer heater	kW
Q	General term for heat input to node	Btu/sec
QOA	Core average heat flux for hot channel analysis	Btu/ft^2/hr
Q_{pr}	Pressurizer heater output	Btu/sec
Q_{sgB1}	Heat transferred from primary coolant to secondary in boiling region of steam generator 1	Btu/sec

7.0 NOMENCLATURE (Continued)

Symbol	Description	Units
Q_{sgNB1}	Heat transferred from primary coolant to secondary in non-boiling region of steam generator 1	Btu/sec
Q_{sg1T}	Total heat transferred from primary coolant to secondary in steam generator 1	Btu/sec
Q_{sh}	Net heat deposited in steam header node	Btu/sec
Q_{sl1}	Net heat deposited in node 1, steamline 1	Btu/sec
Q_{sl2}	Net heat deposited in node 2, steamline 1	Btu/sec
QT	Total heat input to primary coolant in core	Btu/sec
Q_{up1}	Net rate of heat deposition in upper plenum of steam generator 1	Btu/sec
Q_v	General heat term for pressurizer	Btu/sec
Q_{lpsi1}	Net rate of heat transfer from primary to secondary between inlet and node 1, steam generator 1	Btu/sec
Q_{lps12}	Net rate of heat transfer from primary to secondary between nodes 1 and 2, steam generator 1	Btu/sec
Q_{lps23}	Net rate of heat transfer from primary to secondary nodes 2 and 3 in steam generator 1	Btu/sec
Q_{lps34}	Net rate of heat transfer from primary to secondary between nodes 3 and 4 in steam generator 1	Btu/sec
Q_{lps4o}	Net rate of heat transfer from primary to secondary between node 4 and outlet of steam generator 1	Btu/sec
R_{cl}	Average cladding radius	in
RCCA	Rod-cluster control assembly	
$R_{cl,i}$	Inner radius of cladding	in
$R_{cl,o}$	Outer radius of cladding	in
$R_{f,o}$	Outside radius of fuel pellet	in

7.0 NOMENCLATURE (Continued)

Symbol	Description	Units
$R_{f,1}$	Radius of first (innermost) fuel node	in
$R_{f,2}$	Outside radius of second fuel node	in
$R_{f,3}$	Outside radius of third fuel node	in
r_i	Inside radius of tubes in heat exchanger	ft
$R_{n,1}$	Radius of center of first fuel node	in
$R_{n,2}$	Radius of center of second fuel node	in
$R_{n,3}$	Radius of center of third fuel node	in
$R_{n,4}$	Rodius of center of fourth fuel node	in
r_o	Outer radius of steam generator tubes	ft
S	Laplacian Operator	---
t	General term for time	sec
T	General temperature symbol	^oF
T_{AVG}	Primary loop average temperature	^oF
T^o_{AVG}	Primary loop average temperature during nominal operation	^oF
T_{bd1}	Temperature at bottom of downcomer in steam generator 1	^oF
T_{ca}	Average temperature of the core coolant	^oF
$Tc(i)$	Moderator temperature at node	^oF
TC_{iv}	Time required to close isolation valve	sec
T_{cl}	Temperature of fuel clad	^oF
$T_{c\ell 1i}$	Temperature at entrance of loop 1 cold leg	^oF
$T_{c\ell 1o}$	Temperature at discharge end of loop 1 cold leg	^oF

7.0 NOMENCLATURE (Continued)

Symbol	Description	Units
T_{cri}	Mixed liquid temperature in the lower plenum (core inlet temperature)	^{o}F
T_{cro}	Average core outlet temperature	^{o}F
$td_{c\ell min}$	Minimum time delay in cold leg	sec
$td_{h\ell min}$	Minimum time delay in hot leg	sec
TD_{iv}	Time delay before isolation valve starts to close	sec
t_{drvc}	Time required to close relief valve	sec
t_{drvo}	Time required to open relief valve	sec
td_{sc}	Time delay on scram signal before scram actually starts	sec
td_{sv}	Time required to open safety valves	sec
T_{f}	Average fuel temperature	^{o}F
$T_{f,IC}$	Initial (steady-state) average fuel temperature	^{o}F
TF_{sg1}	Temperature of boiling water at pressure in steam generator 1	^{o}F
$T_{f,1}$	Fuel centerline temperature	^{o}F
$T_{f,2}$	Average temperature in second fuel node	^{o}F
$T_{f,3}$	Average temperature in third fuel node	^{o}F
$T_{f,4}$	Average temperature in fourth fuel node	^{o}F
$T_{h\ell o1}$	Temperature of hot leg discharge for loop 1	^{o}F
$T_{h\ell 1}$	Average temperature in hot leg of loop 1	^{o}F
$T_{h\ell 2}$	Average temperature in hot leg of loop 2	^{o}F
T_{lpi}	Inlet temperature to reactor vessel lower plenum inlet	^{o}F
T_{M}	Moderator temperature	^{o}F

7.0 NOMENCLATURE (Continued)

Symbol	Description	Units
$T_{M,IC}$	Moderator temperature under initial steady-state conditions	^{o}F
T_{NL}	Average primary loop hot leg and cold leg temperature at no load	^{o}F
T_{REF}	Reference temperature for load reject controller	^{o}F
t_s	Time scram signal is received	---
t_{sdc}	Time delay before control rods begin to move	sec
T_{sglpa}	Average temperature for primary loop flow	^{o}F
T_{svTRIP}	Time turbine stop valve trip signal is initiated	sec
T_{td1}	Temperature of saturated water at top of downcomer in steam generator 1	^{o}F
T_{Tsv}	Time required to close turbine stop valve	sec
T_{up}	Temperature of fluid in the upper plenum of the reactor vessel	^{o}F
T_{wpr}	Pressurizer water temperature	^{o}F
$TWSC\{HC(i)\}$	Temperature of subcooled water as a function of enthalpy	^{o}F
T_{1pi}	Temperature of primary fluid at inlet of steam generator 1	^{o}F
T_{1po}	Temperature of primary fluid at outlet of steam generator 1	^{o}F
T_{1p1}	Temperature of primary loop fluid at node 1 steam generator 1	^{o}F
T_{1p2}	Temperature of fluid in primary loop at node 2 of steam generator 1	^{o}F
T_{1p3}	Temperature of fluid in primary loop at node 3 steam generator 1	^{o}F

7.0 NOMENCLATURE (Continued)

Symbol	Description	Units
T_{lp4}	Temperature of fluid at node 4 in primary loop steam generator 1	^{o}F
U_A	Overall thermal conductance from primary to secondary fluid in steam generator	Btu/sec/^{o}F
U_{BOIL}	Overall thermal conductance from primary to secondary fluid in the boiling secondary region of the steam generator	Btu/sec/^{o}F
U_{STEAM}	Overall thermal conductance from primary to secondary in the steam secondary region of the steam generator	Btu/sec/^{o}F
U_{SUBC}	Overall thermal conductance from primary to secondary fluid in the subcooled (liquid) region of the steam generator	Btu/sec^{o}F
V	General term for specific volume	ft^3/lb
V_{bd1}	Specific volume at the bottom of the downcomer for steam generator 1	ft^3/lb
V_{c1}	Volume of fuel clad per unit length	ft^3/ft
$V_{c\ell}$	Specific volume of fluid in the cold leg	ft^3/lb
$\overline{V}_{cr}$	Average specific volume of fluid in the core	ft^3/lb
V_{do1}	Specific volume of fluid in the steam dome of steam generator 1	ft^3/lb
$VFGP\{P_{s\ell 12}\}$	Variation of change in specific volume on vaporization of liquid water with respect to pressure at the pressure of node 2, steamline 1	$ft^3\ in^2/lb^2$
$V_{f,i}$	Volume of one fuel node per unit length	ft^3/ft
VF_{sc}	Specific volume of subcooled liquid water as a function of temperature	ft^3/lb
VF_{sg1}	Specific volume of liquid in boiling region steam generator 1	ft^3/lb

7.0 NOMENCLATURE (Continued)

Symbol	Description	Units
V_{hl}	Specific volume of water in the primary hot leg loop	ft^3/lb
V_j	Specific volume of water at node j	ft^3/lb
V_{jk}	Specific volume of fluid in region between nodes j and k	ft^3/lb
V_{lc}	Calculated control signal for feedwater flow controller	---
V_{lcsp}	Setpoint for control signal for feedwater flow controller	---
V_{pr}	Calculated control signal for turbine throttle pressure controller	---
V_{prsp}	Setpoint for control signal for turbine throttle pressure controller	---
$\bar{V}_{sg}$	Average specific volume of fluid in primary side of steam generator	ft^3/lb
V_{sgl}	Specific volume of mixed fluid in steam generator 1 secondary	ft^3/lb
V_{sh}	Specific volume of fluid in steam header	ft^3/lb
V_{sl1I}	Specific volume of fluid in node *I* of steamline 1	ft^3/lb
V_{sl12}	Specific volume of fluid in node *2* of steamline 1	ft^3/lb
V_{sl22}	Specific volume of fluid in node *2* of steamline 2	ft^3/lb
V_{td1}	Specific volume at top of downcomer in steam generator 1	ft^3/lb
V_{up}	Specific volume of fluid in upper plenum of reactor vessel	ft^3/lb
V_{wpr}	Specific volume of liquid water in the pressurizer	ft^3/lb
V_{upl}	Specific volume of fluid in the upper plenum of steam generator 1	ft^3/lb

7.0 NOMENCLATURE (Continued)

Symbol	Description	Units
V_{1pi}	Specific volume of fluid in steam generator 1 primary at the inlet	ft^3/lb
V_{1pI}	Specific volume of primary loop fluid in steam generator 1 in the region between inlet and node 1	ft^3/lb
V_{1po}	Specific volume in primary loop at discharge of steam generator 1	ft^3/lb
V_{1pI2}	Specific volume of primary loop fluid in steam generator 1 for the region between nodes *I* and *2*	ft^3/lb
V_{1p23}	Specific volume of primary loop fluid in steam generator 1 between nodes *2* and *3*	ft^3/lb
V_{1p34}	Specific volume of primary loop fluid in steam generator 1 between nodes *3* and *4*	ft^3/lb
V_{1p4o}	Specific volume of primary loop fluid in steam generator 1 between nodes *4* and outlet	ft^3/lb
V_{1sB}	Specific volume of secondary fluid in the boiling region of steam generator	ft^3/lb
V_{1sNB}	Specific volume of secondary fluid in the subcooled region of steam generator 1	ft^3/lb
V_{1sST}	Specific volume of secondary fluid in the steam region of steam generator 1	ft^3/lb
V_{1sO}	Specific volume of actual mixed fluid in upper plenum of steam generator 1 when quality $\geq$ 0.89	ft^3/lb
V_{89sg1}	Specific volume of mixed fluid in upper plenum of steam generator 1 based on quality of 0.82	ft^3/lb
W	General term for flow	lb/sec
W_{bdsg1}	Flow from bottom of downcomer into steam generator 1	lb/sec
W_{chc}	Hot channel flow	lb/sec

7.0 NOMENCLATURE (Continued)

Symbol	Description	Units
$W_{c\ell i}$	Cold leg safety injection flow	lb/sec
W_{crup}	Flow from core to upper plenum	lb/sec
W_{d}	General term for rated pump flow	lb/sec
W_{dbp}	Bypass steam flow to condenser	lb/sec
$W_{dos\ell 1}$	Steam flow from dome to steamline 1	lb/sec
W_{dosv}	Genrreral term for steam flow from dome of steam generator	lb/sec
W_{dsh}	Atmospheric steam dump flow from steam header	lb/sec
$W_{ds\ell 1}$	Atmospheric steam dump flow from steamline 1	lb/sec
W_{epr}	Steam flashing rate in pressurizer	lb/sec
W_{fw}	General term for feedwater flow	lb/sec
$W_{fw,max}$	Maximum feedwater flow allowed	lb/sec
$W_{fw,IC}$	Initial steady-state feedwater flow	lb/sec
$W_{fw,1}$	Feedwater flow for steam generator 1	lb/sec
WH	General term for enthalpy flux	Btu/sec
$W_{h\ell i}$	Hot leg safety injection flow	lb/sec
W_{in}	General term for total flow into volume	lb/sec
W_{iv1}	Flow through isolation valve in steamline 1	lb/sec
W_{jk}	General term for flow from node j to node k	lb/sec
W_{leak}	Total primary system leakage flow	lb/sec
$W_{\ell p}$	General term for flow in loop	lb/sec
W_{lpcr}	Flow in primary loop from lower plenum to core	lb/sec

7.0 NOMENCLATURE (Continued)

Symbol	Description	Units
$W_{lpcr,IC}$	Initial steady-state flow from lower plenum to core	lb/sec
$W_{\ell p1}$	Total flow in primary loop 1	lb/sec
$W_{\ell p1,max}$	Maximum flow corresponding to minimum time delay in primary loop 1	lb/sec
$W_{\ell p2}$	Total flow in primary loop 2	lb/sec
W_{mup}	Make up water flow	lb/sec
W_{out}	General term for flow out of volume	lb/sec
$W_{out,i}$	General term for flow from volume i	lb/sec
W_{pi}	Primary flow into steam generator	lb/sec
W_{po}	Primary loop flow from outlet of steam generator	lb/sec
W_{prrvm}	Relief valve flow if valve is wide open	lb/sec
W_{prrv}	Flow from pressurizer relief valve	lb/sec
W_{prsp}	Spray flow into pressurizer	lb/sec
W_{prsv}	Safety valve flow from pressurizer	lb/sec
WRC1	Recirculation flow in steam generator 1 secondary	lb/sec
W_{rv}	Total flow through reactor vessel	lb/sec
W_{rv1}	Flow from relief valve in steamline 1	lb/sec
W_{s}	Total steam flow from steam generator	lb/sec
W_{sd}	Steam dump flow	lb/sec
$W_{sd,max}$	Steam dump maximum flow	lb/sec
$W_{sda,max}$	Steam dump maximum flow for group A valves	lb/sec

7.0 NOMENCLATURE (Continued)

Symbol	Description	Units
$W_{sdb,max}$	Steam dump maximum flow for group B valves	lb/sec
W_{sgup1}	Flow from steam generator 1 secondary to upper plenum	lb/sec
W_{sish1}	Flow into steam header from steamline 1	lb/sec
W_{sish2}	Flow into steam header from steamline 2	lb/sec
W_{sv1a}	Flow through safety valve set "A" in steamline 1	lb/sec
W_{sv1b}	Flow through safety valve set "B" in steamline 1	lb/sec
W_{sv1c}	Flow through safety valve set "C" in steamline 1	lb/sec
W_{sv1d}	Flow through safety valve sed "D" in steamline 1	lb/sec
W_s^o	Total (normal) operating steam flow from both steam generators	lb/sec
W_{tb}	Turbine flow	lb/sec
$W_{tb,max}$	Maximum turbine flow	lb/sec
$W_{tb,IC}$	Initial (steady-state) flow through turbine	lb/sec
W_{tot}	Turbine flow demand from turbine throttle pressure controller	lb/sec
W_{updo1}	Flow from upper plenum to dome in secondary of steam generator 1	lb/sec
W_{uppr}	Flow from reactor upper plenum into pressurizer	lb/sec
W_{upsp1}	Flow of mixed fluid from upper plenum to separators on secondary sides of steam generator 1	lb/sec
W_{uptd1}	Liquid flow from upper plenum to top of downcomer in steam generator 1	lb/sec

7.0 NOMENCLATURE (Continued)

Symbol	Description	Units
X_{dol}	Quality in dome of steam generator 1	---
$X_{hc(i)}$	Quality at nodal junction i of hot channel	---
X_{sgupl}	Quality of fluid in upper plenum steam generator 1	---
X_{sgl}	Quality in steam generator 1	---
X_{sh}	Quality in steam header	---
X_{sl1}	Quality in steamline 1, node *1*	---
X_{sl2}	Quality in steamline 1, node *2*	---
X_{upl}	Quality in upper plenum of steam generator 1	---
Y_1	Thermal admittance for fuel node *1*	Btu/sec/^{o}F
Y_2	Thermal admittance for fuel node *2*	Btu/sec/^{o}F
Y_3	Thermal admittance for fuel node *3*	Btu/sec/^{o}F
Y_4	Thermal admittance for fuel node *4*	Btu/sec/^{o}F
Z_{bdsg}	Difference in elevation between active region and bottom of steam generator downcomer annulus	ft
Z_{bdsgl}	Z_{bdsg} for steam generator 1	ft
Z_{cl}	Cold leg elevation difference	ft
Z_{hl}	Hot leg elevation difference	ft
Z_{rv}	Elevation of core mid-plane above recirculation pump discharge	ft
Z_{sg}	Elevation of steam generator mid-plane above recirculation pump discharge	ft
Z_{sgup}	Elevation of upper plenum from active region steam generator	ft
Z_{updo}	Elevation of upper plenum referenced to dome	

7.0 NOMENCLATURE (Continued)

Symbol	Description	Units
Z_{updol}	Z_{updo} for steam generator 1	ft
Z_{LEVEL}	Water level in downcomer above reference	ft
α_B	Boron reactivity coefficient	$/ppm
α_D	Doppler reactivity coefficient	$/°F
α_M	Moderator reactivity coefficient	$/°F
$\alpha_{M,IC}$	Moderator reactivity coefficient at initial conditions	$/°F
α_{PR}	Pressure reactivity coefficient	$/psi
β_{eff}	Effective delayed neutron fraction	---
β_i	Delayed neutron fraction for ith delayed neutron group	---
ΔH_{core}	Total enthalpy change for primary coolant through the core	Btu/lb
ΔH_{hc}	Enthalpy rise in hot channel	Btu/lb
δk	Total reactivity change from steady-state operation	$
δk_B	Boron shim reactivity change	$
δk_{CR}	Total reactivity change from control rod motion and RCCA control model	$
δk_{CRC}	Change in reactivity from RCCA control model	$
δk_{CRS}	Reactivity change from control rod scram motion	$
δk_D	Doppler reactivity change	$
δk_M	Moderator reactivity change	$
δk_{PR}	Pressure reactivity change	$
ΔP_{acc}	Recoverable acceleration head for fluid flow	psia

7.0 NOMENCLATURE (Continued)

Symbol		Units
ΔP_{fr}	Friction and form loss of pressure	psi
ΔPH2	Calculated term for pressure change due to differential density in primary loop	psi
ΔPKELP	Acceleration head in primary loop	psi
ΔPKERV	Acceleration head in reactor vessel	psi
ΔPLP	Friction and momentum losses around the primary loop	psi
ΔP_{pump}	Primary pump head	psi
ΔPRV	Friction and momentum losses across the reactor vessel	psi
Δt	Time increment	sec
ΔT	Auctioneered primary loop delta temperature	^oF
ΔT^o	Delta temperature generated by overtemperature and overpower trip logic	^oF
$(\Delta TM1)_{i1}$	Mean temperature difference for heat exchange in steam generator 1 in the region between the inlet and node *1*	^oF
Δx	Axial length of a coolant channel node	ft
λ_i	Decay constant for isotopes in *i*th delay group	1/sec
μ	Pressurizer spray flow condensation factor	---
ρ_{cl}	Density of clad	lb/ft^3
ρ_f	Fuel density	lb/ft^3
ρ_w	Density of water	lb/ft^3
$\tau_{f,1}$	Time constant calculated for heat flow in fuel node *1*	sec
$\tau_{f,2}$	Time constant calculated for heat flow in fuel node *2*	sec

7.0 NOMENCLATURE (Continued)

Symbol	Description	Units
$\tau_{f,3}$	Time constant calculated for heat flow in fuel node 3	sec
$\tau_{f,4}$	Time constant calculated for heat flow in fuel node 4	sec
τ_{lp}	Time constant for flow mixing in the reactor vessel lower plenum	sec
τ_{pi}	Time constant for flow mixing in steam generator inlet plenum	sec
τ_{po}	Time constant for flow mixing in the steam generator outlet plenum	sec
τ_{up}	Time constant for flow mixing in the upper plenum of the reactor vessel	sec
$\tau_1 - \tau_{99}$	Time constants in various controllers	sec

8.0 CODE USAGE

8.1 BUILT-IN CHARACTERISTIC PROPERTIES

While most of the constants used by PTSPWR2 code may be changed by input values (see Section 8.2), some are so nearly characteristic that they may be changed only by altering the individual subroutines. In this section a description of these characteristic properties is given.

8.1.1 Water and Steam Tables

The constants which are most infrequently changed are the characteristic properties of water and steam. These are entered in DATA, a Fortran BLOCK DATA subprogram, in tabular form, the parameters being a function of either pressure, temperature or enthalpy. Interpolation is done by routine FG4. Table 8.1 lists the specific water and steam properties available in DATA.

8.1.2 Pump Characteristics

The table is located in subprogram DATA. Typical pump curves applicable to nuclear power plant analysis are depected in Figure 8.1. The default pump curve in PTSPWR2 is the Westinghouse pump curve designated by $\overline{W}$.

8.1.3 Decay Energy Calculations

The decay energy calculations used by PTSPWR2 are the same as those described in Reference 3. The decay calculations include contributions from both fission product decay (eleven decay

groups) and actinide decay of U_{239} and Np_{239}. The only input parameter necessary is the ratio of U_{238} captures per fissions (UDUFF) which is typically a function of burnup.

8.2 PREPARATION OF INPUT

The prospective user of this code should first determine that the model (as described in Sections 1 through 6) and the built-in characteristics (Section 8.1) are suitable for the application at hand.

Modification of most parameters is simply accomplished by the use of input cards. The format of the cards falls into one of the following two groups.

1) Single parameter entry

   The parameter name is given in columns 1 - 6 of the card; the value of the parameter is given in columns 11 - 20. The Fortran format is (A6, 4X, E10.0).

2) Tabular parameter entry

   The parameter array name is given columns 1 - 6 of the first card (A6). The array of values is contained on the following card(s) in (8E10.0) format.

In addition, the first card in every run may be a title card (format 12A6.0). If the first card contains a recognized parameter name in columns 1 - 6, the run title defaults to all blanks.

A listing of all single-parameter inputs with their descriptions and default values is given in Table 8.2. A listing of tabular parameter inputs with their descriptions is given in Table 8.3. The user is encouraged to use these tables as the main references* for all input modifications. A summary of parameters not usually entered is given in Table 8.4.

---

* Note: All default values, including built-in characteristic properties are specific to the H. B. Robinson Unit 2 PWR, a 3-Loop Westinghouse NSSS.

8.3 STEADY STATE CALCULATIONS

To obtain a stable steady state condition prior to transient initiation, extreme care must be exercised in entering initial values.

To facilitate the initialization of steady state conditions, the PTSPWR2 code has a subroutine STEADY which balances the mass and energy equations around the loops and iterates until a steady state is achieved before commencement of the transient.

STEADY may be exercised by including an input parameter card which sets

STSTSW = 1.0 (Call STEADY.)

The use of this option will generate a listing of input parameter values and new initial values as readjusted by STEADY. This listing precedes the graphical output for the transient. This option is advised for all transient

calculations initialized from a normal operating condition. For special cases where the user wishes to initiate a transient event from a non-steady state condition, this option is not advised.

8.4 RESTART OPTION

For certain extremely long-term transients (loss of feedwater flow, for example), the user may wish to examine the transient response in short time segments during the transient to ensure that all control systems are functioning properly or that the proper input has been entered which would give an expected response. To enable such examination, yet ensure transient continuation, a restart option has been made available in PTSPWR2. To exercise the option, the following parameters must be entered .

SAVER = 1.0	Create a restart tape
RWRITE = x.xx	Time interval for saving restart variables on tape (This value should be about 10 sec for long runs)

The user is also cautioned to call for a tape mount before exercising this option.

On the restart job, only the following parameters need to be entered in addition to the title card.

RSTART = 1.0	Restart from tape file
RTIME = x.xx	Time in transient at which restart is to begin.

8.5 TRANSIENT RELATED INPUT

8.5.1 Transient Specific Input

The PTSPWR2 code is capable of simulating those transients listed in Table 1.1. The appropriate input for proper selection of each particular transient to be analyzed is given below. In all cases, the appropriate set of kinetics parameters must be defined. Typical run times for each transient are given in Table 8.5.

Uncontrolled Rod Withdrawals

RCCA Drop

Chemical and Volume Control System Malfunctions

The input for a chemical and volume control system malfunction is the same as for an uncontrolled rod withdrawal, except that the rate of reactivity insertion | | and the maximum value of reactivity insertion | | will be a function of the boron dilution rate.

Loss of Coolant Flow

|

or

|

|

Locked Rotor

|

Loss of Electrical Load

|

|

|

Excessive Load Increase

|

|

|

Loss of AC Power

Loss of Feedwater

Reduction in Feedwater Enthalpy

or

or

8.5.2 Additional Imposable Conditions

The user may desire to impose certain conditions on each transient event to determine the effects of an inoperable component on system response or to insure conservatism in transient responses. An example of this would be not allowing the effects of pressurizer spray or relief valve actuation during a loss of external load transient. This is easily done by setting the maximum flow rates equal to zero on input. A selection of commonly used conditions follows.

8.5.3 Automatic Reactor Control

An additional feature incorporated into PTSPWR2 is the automatic reactor control discussed in Section 5.4. This may be used with any transient initiated from a non-zero power condition. In addition to the numerous control variables needed to describe the automatic reactor control system, the following parameters must be set for its use during a transient.

8.6 INTERPRETATION OF OUTPUT

The output always begins with an ECHO listing of the input cards, followed by a summary of changes affected by the input cards.

If subroutine STEADY is called for (STSTSW = 1.0), a listing of initial values and values as reinitialized by STEADY will follow. This option is particularly useful in assuring steady-state stability for a particular application.

Immediately following is a summary of the output. This summary gives the maximum (or minimum) values for significant parameters during the transient and the time of their occurrance (i.e., peak system pressure, MDNBR, etc. Also included is the time of trip and trip signal for the transient.

If PNTPSW is 1.0 (default value is 1.0), printer plots are given which provide a gross indication of plant behavior during the transient.

The tabular output from the simulation is given as variable vs time. The time between print frames normally produces one page of printout with 11 variables per page. The time frame sequencing may be adjusted, however, by changing the time step size (DELTAT), the number of lines of printout per page (PGSIZE), or the total number of lines of tabular

printout per set of variables (PLINES). If no tabular printout is desired, PRNTSW may be set equal to 0.0. A list of output variables with their description is given in Table 8.6. In addition to these output variables, the code also prints out pertinent quantities nodalized axially in the core. These output arrays are described in Table 8.7.

The computer time required for various sections of code is also printed out.

TABLE 8.5

TYPICAL TRANSIENT RUN TIMES

Transient	TFINAL (sec)
Uncontrolled Rod Withdrawal	[ ]
Fast Withdrawal	
Slow Withdrawal	
RCCA Drop	
Chemical and Volume Control System Malfunction	
Loss of Coolant Flow	
Locked Rotor	
Startup of Inactive Loop	
Loss of External Load	
Excessive Load Increase	
Loss of AC Power	
Loss of Feedwater	
Reduction in Feedwater Enthalpy	

TABLE 8.6

INTERPRETATION OF OUTPUT

Output Variable	Description	Units
CFSPR	Volume of steam in pressurizer	ft^3
CFWPR	Volume of water in pressurizer	ft^3
CIV1,2	Valve position for isolation valve in steam line 1, 2	-
CSV	Valve position for turbine stop valve	-
DELHCR	Core enthalpy rise	BTU/lb
DELHHC	Hot channel enthalpy rise	BTU/lb
DELT1,2	Hot leg - cold leg delta temperature, Loop 1, 2	°F
DELTSP	Overtemperature ΔT setpoint	°F
DK	Total change in reactivity	$
DKBOR	Change in reactivity due to boron addition	$
DKCTL	Change in reactivity due to scram	$
DKDOP	Change in reactivity due to Doppler feedback	$
DKMAN	Change in reactivity due to rod withdrawal or insertion	$
DKMOD	Change in reactivity due to moderator temperature feedback	$
DKPRES	Change in reactivity due to pressure	$
DKRCCA	Reactivity change due to movement of automatic RCCA rods	$

TABLE 8.6 (Continued)

Output Variable	Description	Units
DNBW3	Minimum DNB ratio	-
DOSL(1,2)D	$\frac{d(WDOSL1,2)}{dt}$	lb/sec^2
DTPSP	Overpower ΔT setpoint	°F
GHC	Hot channel mass velocity	$lb/hr\text{-}ft^2$
HBD1,2	Enthalpy at bottom of downcomer, steam generator 1, 2	BTU/lb
HCRUP	Enthalpy from core to upper plenum	BTU/lb
HFW1,2	Feedwater Enthalpy, Loop 1, 2	BTU/lb
HGUP(1,2)D	$\frac{d(HSGUP1,2)}{dt}$	BTU/lb-sec
HSGUP1,2	Enthalpy from active region to upper plenum in steam generator 1, 2	BTU/lb
HTD1,2	Enthalpy at top of downcomer in steam generator 1, 2	BTU/lb
HRVUP	Enthalpy in reactor vessel upper plenum	BTU/lb
HUP1,2	Enthalpy in upper plenum of steam generator 1, 2	BTU/lb
HWPR	Enthalpy of water in pressurizer	BTU/lb
HWPRD	$\frac{d(HWPR)}{dt}$	BTU/lb-sec
LB1,2	Length of boiling zone in steam generator 1, 2	ft
LEVG1,2D	$\frac{d(LEVSG1,2)}{dt}$	ft/sec

TABLE 8.6 (Continued)

Output Variable	Description	Units
LEVSG1,2	Level in steam generators 1, 2	ft
LEVPR	Level in pressurizer	ft
LNB1,2	Length of non-boiling zone in steam generator 1, 2	ft
MDO1,2	Mass in dome of steam generator 1, 2	lb
MDO1,2D	$\frac{d(MDO1,2)}{dt}$	lb/sec
MSG1,2	Mass in active region of steam generator 1, 2	lb
MSG1,2D	$\frac{d(MSG1,2)}{dt}$	lb/sec
MSH	Mass in steam line header node	lb
MSHD	$\frac{d(MSH)}{dt}$	lb/sec
MSLij	Mass in steam line i node j	lb
MSLijD	$\frac{d(MSLij)}{dt}$	lb/sec
MSPR	Mass of steam in pressurizer	lb
MSPRD	$\frac{d(MSPR)}{dt}$	lb/sec
MUP1,2	Mass in upper plenum of steam generator 1, 2	lb
MUP1,2D	$\frac{d(MUP1,2)}{dt}$	lb/sec

TABLE 8.6 (Continued)

Output Variable	Description	Units
MWPR	Mass of water in pressurizer	lb
MWPRD	$\frac{d(MWPR)}{dt}$	lb/sec
NPUMP1,2	Normalized pump speed for reactor coolant Pump 1, 2	-
PBD1,2	Pressure at bottom of downcomer in steam generator 1, 2	psia
PDO1,2	Pressure in dome of steam gener or 1, 2	psia
PDO(1,2)D	$\frac{d(PDO1,2)}{dt}$	psia/sec
PL	Power level	Mwt
PPR	Pressurizer Pressure	psia
PPRD	$\frac{d(PPR)}{dt}$	psia/sec
PSG1,2	Pressure in active region of steam generator 1, 2	psia
PSG(1,2)D	$\frac{d(PSG1,2)}{dt}$	psia/sec
PSH	Pressure in steam line steam header node	psia
PSHD	$\frac{d(PSH)}{dt}$	psia/sec
PSLij	Pressure in steam line i node j	psia
PSLijD	$\frac{d(PSLij)}{dt}$	psia/sec

TABLE 8.6 (Continued)

Output Variable	Description	Units
PTT	Turbine throttle pressure	psia
PUP1,2	Pressure in upper plenum of steam generator 1, 2	psia
PUP(1,2)D	$\frac{d(PUP1,2)}{dt}$	psia/sec
QOA	Core average heat flux	BTU/lb-ft^2
QPR	Pressurizer heater output	BTU/sec
QSG(1,2)T	Total heat transferred in steam generator 1, 2	BTU/sec
QT	Total heat output of reactor	BTU/sec
RSPEED	Control rod speed	in/sec
SCCODE	Scram code, indicates signal which initiated scram. Scram codes are given in Table 8.8	
SCTIME	Time of scram initiation. A negative number indicates scram not initiated and gives the appropriate scram delay time	sec
SLSH(1,2)D	$\frac{d(WSLSH1,2)}{dt}$	lb/sec^2
TAVG1,2	Hot leg - cold leg average temperature, Loop 1, 2	°F
TCA	Core average temperature	°F
TCID	$\frac{d(TCI)}{dt}$	°F/sec
TCLAD	Cladding temperature	°F

TABLE 8.6 (Continued)

Output Variable	Description	Units
TCL1,2	Cold leg temperature, Loop 1, 2	°F
TFAVG	Fuel average temperature	°F
TFi	Fuel temperature for fuel node i	°F
THL1,2	Hot leg temperature, Loop 1, 2	°F
TLPI	Lower plenum inlet temperature	°F
TRVUP1,2	Reactor vessel upper plenum temperature to Hot leg 1, 2	°F
TSDAVG	Overall average primary coolant temperature	°F
TSDREF	Reference T_{AVG} for control systems	°F
TSG(1,2)PA	Average primary coolant temperature in steam generator 1, 2	°F
TSG(1,2)PI	Inlet plenum primary coolant temperature for steam generator 1, 2	°F
TSG(1,2)PO	Outlet plenum primary coolant temperature for steam generator 1, 2	°F
TWPR	Temperature of water in pressurizer	°F
TiPj	Primary coolant temperature at point j in steam generator i	°F
TiPjD	$\frac{d(T_iP_j)}{dt}$	°F/sec
TiPI	Primary coolant temperature at inlet of steam generator tubes in Loop i	°F
TiPID	$\frac{d(T_iPI)}{dt}$	°F/sec

TABLE 8.6 (Continued)

Output Variable	Description	Units
T$_i$PO	Primary coolant temperature at exit of steam generator tubes in Loop i	°F
T$_i$POD	$\frac{d(T_iPO)}{dt}$	°F/sec
UPSP(1,2)D	$\frac{d(WUPSP1,2)}{dt}$	lb/sec^2
WBDG(1,2)D	$\frac{d(WBDSG1,2)}{dt}$	lb/sec^2
WBDSG1,2	Flow at bottom of downcomer in steam generator 1, 2	lb/sec
WCHC	Hot channel flow	lb/sec
WDBP	Condenser Bypass flow	lb/sec
WDOSLT	Total flow from dome to steam line (steam generator 1 and 2)	lb/sec
WDOSL1,2	Flow from dome to steam line steam generator 1, 2	lb/sec
WDSH	(Not used)	
WDSL1,2	Atmospheric steam dump flow from steam line 1, 2	lb/sec
WEPR	Flashing rate in pressurizer	lb/sec
WFWT	Total feedwater flow	lb/sec
WFW1,2	Feedwater flow for Loop 1, 2	lb/sec
WIV1,2	Flow through isolation valve in steam line 1, 2	lb/sec
WIV(1,2)D	$\frac{d(WIV1,2)}{dt}$	lb/sec^2

TABLE 8.6 (Continued)

Output Variable	Description	Units
WLPCR	Flow from lower plenum to core	lb/sec
WLP1,2	Flow to lower plenum from cold leg 1, 2	lb/sec
WLP(1,2)D	$\frac{d(WLP1,2)}{dt}$	lb/sec^2
WMUP	Make up water flow	lb/sec
WPPRV	Pressurizer relief valve flow	lb/sec
WPRSP	Pressurizer spray flow	lb/sec
WPRSV	Pressurizer safety valve flow	lb/sec
WRV1,2	Flow through relief valves on steam line 1, 2	lb/sec
WSGUP1,2	Flow from active region to upper plenum in steam generator 1, 2	lb/sec
WSLSH1,2	Flow from steamline 1, 2 to steam header node	lb/sec
WSV1,2	Flow through safety valves in steam lines 1, 2	lb/sec
WTB	Flow to turbine	lb/sec
WUPDO1,2	Steam flow from upper plenum to dome in steam generator 1, 2	lb/sec
WUPPR	Pressurizer surge flow	lb/sec
WUPSP1,2	Flow from upper plenum to separators in steam generator 1, 2	lb/sec
WUPTD1,2	Flow from upper plenum to top of downcomer in steam generator 1, 2	lb/sec

TABLE 8.6 (Continued)

Output Variable	Description	Units
W3NODE	Axial node at which MDNBR occurs	-
XDO1,2	Quality in dome in steam generator 1, 2	-
XSGUP1,2	Quality from active region to upper plenum in steam generator 1, 2	-
XSG1,2	Quality in active region in steam generator 1, 2	-
XUP1,2	Quality in upper plenum in steam generator 1, 2	-

TABLE 8.7

TABULAR OUTPUT

(With Reference to Core Variables)

Output Variable	Description	Units
TCI(i)*	Coolant temperature at node i	°F
HHCI(i)	Hot channel enthalpy at node i	BTU/lb
PHI(i)	Hot channel heat flux at node i	BTU/lb-ft^2
QDNB(i)	DNB heat flux at node i	BTU/lb-ft^2
DNBR(i)	DNBR at node i	-
XHCI(i)	Hot channel quality at node i	-

* In all cases, i = 0 refers to the core inlet condition.

TABLE 8.8

SCRAM CODES

SCCODE	Trip Signal
1	High power
2	Low level in steam generator 1
3	Low level in steam generator 2
4	Low flow in Loop 1
5	Low flow in Loop 2
6	High pressurizer level
7	High pressurizer pressure
8	Low pressurizer pressure
9	Closure of main steam line isolation valve in steam line 1
10	Closure of main steam line isolation valve in steam line 2
11	Turbine trip
12	Overtemperature ΔT
13	Overpower ΔT

9.0 REFERENCES

1. Kahn, J.D. and Foster, M.S., XN-CC-38, and Supplement 1, "Users Manual for PTSPWR2 - A Fortran Program for Simulation of Pressurized Water Reactor Plant Transients," December 1976.

2. Tong. L.S., "Critical Heat Fluxes in Rod Bundles," p. 31-46 of Two-Phase Flow and Heat Transfer in Rod Bundles, AEEW-R358 (1964).

3. Skogen, F.B., XN-75-43, "Core Physics Methods and Data Used as Input to LOCA Analysis," August 1975.

4. Macduff, R.B., XN-NF-621(P)(A), Rev. 1, "Exxon Nuclear DNB Correlation for PWR Fuel Designs," September 1983.

5. Turner, D.M., XN-NF-82-91(P), "H.B. Robinson Loss of Feedwater Transient at 1955 MWt - Model Description and Results," November 1982.

APPENDIX A

SUBROUTINE CALLING SEQUENCES

A general picture of the overall probram/subroutine calling sequence is given below. The main routine is named PTSPWR.

Subroutine	Calls
PTSPWR	ECHO
	DNB
	INIT
	SETUP
	MRUN
	PLTFIL
	PLOT
	PPLOT
	PRINT2
	PTSPLT
	TAPOUT
CTHERM	FG4
	LIN1
CTRL1	CVALVE
	LIN1
	NORLIM
	PIC

Subroutine	Calls
	REACON
	RINT
	TRLIM
CUTUP	DASHEN
	PLOT
	SYMBOL
	WHERE
DASHEN	PLOT
DNB	FG4
DRAW	NUMBER
	PLOT
ECHO	DOY
	ENDPCF

ENTRY (CTHINT, FUNCTS, PLDLAY, PLINT, BORINT, BORDLY, SGDLAY, UPSUB1, UPSUB2, VANDCP, WEPRSB, INT2A, INT2B, INT1A, INT2A)

FG4

LIN1

RINT

Subroutine	Calls
	RLIN1
	RTD
	RTDMIX
HOTROD	FG4
	LIN1
	XPAND
INIT	FG4
LEDGEN	NUMBER
	PLOT
	SYMBOL
MRUN	DATE
	NUMBER
	SYMBOL
	TOD
PGRAPH	DXDY
	STRIP
PIC	NORLIM

Subroutine	Calls
PKIN	FG4
	LIN1
	RINT
PLOOP	FG4
	LIN1
	LIN1A
	WUPCAL
PLTFIL	DNB
PPLOT	LEGEND
	PGRAPH
	TITLE
PRESSI	CVALVE
	FG4
	LIN1
	WEPRSB
PRINTZ	DATE
	TOD

Subroutine	Calls
PTSPLT	CUTUP
	DRAW
	DXDY
	FACTOR
	GRID
	LEDGEN
	NAMEIT
	PLOT
	PLOTS
	STEPZZ
	SYMBOL
	WHERE
REACON	LIN1
	RINT
RLIN1	LIN1
SETUP	CENTER
	FG4

Subroutine	Calls
SGREC	CVALVE
	DHPCAL
	FG4
	FUNCTS
	HCALC
	LIN1
	RINT
	SSQRT
	UPSUB1
	UPSUB2
	XCALC
STEADY	FUNCTS
	MRUN
	NORLIM
	PLOOP
	SGREC
	ZHEAT
STMLIN	CVALVE
	DHPCAL
	FUNCTS
	HCALC
	LIN1
	RINT

Subroutine	Calls
	SSQRT
	SVFLOW
	XCALC
XCALC	NORLIM
ZHEAT	DTSQRT
	FG4
	LGMEAN
	NORLIM
	RINT
	VANDCP
ZPRINT	INT1A
	INT1B
	INT2A
	INT2B

APPENDIX B

SUBROUTINE FUNCTIONAL DESCRIPTION

A brief description of the function of the more important subroutines is given below. This description, along with the calling order given in Appendix A, should facilitate planning for any desired modifications.

PTSPWR:

The main routine

APPENDIX C

SIMULATED PRESSURIZER TEST DATA

Previously unreported pressure and level data from the THLF are used for comparison with the updated PTS version. The original version projection is also illustrated for comparison. Figure C.1 shows several traces. The time history for pressure is shown with "O's" and represents actual pressure data recorded by visual observation of a pressure gauge on the THLF during the transient. The volume of the pressurizer, the initial level, initial pressure, initial mass, and other pertinent initial conditions and related set points were obtained and input to the original version and the updated version. [

]

XN-74-5(NP)(A)
Supplement 1
Revision 2
Issue Date 10/8/86

DESCRIPTION OF THE EXXON NUCLEAR PLANT TRANSIENT SIMULATION MODEL FOR PRESSURIZED WATER REACTORS (PTS-PWR):

PTSPWR2 MODIFICATIONS FOR ST. LUCIE UNIT 1

Prepared by: William T. Nutt 6/24/83
W. T. Nutt
PWR Safety Analysis

Concur: G. C. Cooke 6/27/83
G. C. Cooke, Manager
Model Development

Concur: W. V. Kayser 6/29/83
W. V. Kayser, Manager
PWR Safety Analysis

Approve: J. C. Chandler 7/6/83
J. C. Chandler
Reload Fuel Licensing

Approve: R. B. Stout
R. B. Stout, Manager
Licensing & Safety Engineering

Approve:
G. A. Sofer, Manager
Fuel Engineering & Technical Services

EXXON NUCLEAR COMPANY, INC.

NUCLEAR REGULATORY COMMISSION REPORT 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 U.S. Nuclear Regulatory Commission as part of a technical contribution to facilitate safety analyses by licensees of the U.S. Nuclear Regulatory Commission 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 U.S. Nuclear Regulatory Commission in its review of this report, and under the terms of the respective agreements, by licensees or applicants before the U.S. Nuclear Regulatory Commission which are customers of Exxon Nuclear in their demonstration of compliance with the U.S. Nuclear Regulatory Commission's regulations.

Exxon Nuclear's warranties and representations concerning the subject matter of this document are those set forth in the agreement between Exxon Nuclear and the customer to which this document is issued. Accordingly, except as otherwise expressly provided in such agreement, neither Exxon Nuclear nor any person acting on its behalf:

A. Makes any warranty, or representation 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 AND SUMMARY	1
	1.1 PUMP TRANSIENT MODELING CHANGES	1
	1.2 PRIMARY LOOP DYNAMIC MODELING	1
	1.3 THERMAL CALCULATIONS	2
	1.4 CRITICAL HEAT FLUX CORRELATION	2
	1.5 SCRAM CURVES	2
	1.6 PLANT PROTECTION SYSTEM	2
	1.7 PRESSURE CONTROL SYSTEM	2
	1.8 STEAM LINE BREAK	3
2.0	PUMP TRANSIENT PERFORMANCE	4
	2.1 PUMP DYNAMICS	4
	2.2 IN SITU DYNAMIC MODELING	6
	2.3 IMBALANCED LOOP PUMP COASTDOWNS	7
3.0	PRIMARY LOOP HYDRODYNAMICS FOR C.E. 2x4 PLANTS	11
4.0	THERMAL CALCULATIONS	18
5.0	CRITICAL HEAT FLUX CORRELATION	19
6.0	SCRAM CURVES	20
7.0	PLANT PROTECTION SYSTEM	24
8.0	PRESSURIZER PRESSURE CONTROL SYSTEM	30

TABLE OF CONTENTS (Cont.)

Section		Page
9.0	STEAM LINE BREAK	33
	9.1 FEEDWATER FLOW	33
	9.2 STEAM LINE BREAK FLOW	34
	9.3 STEAM GENERATOR BLOWDOWN	34
	9.4 REACTIVITY MODELS	36
10.0	REFERENCES	42

LIST OF TABLES

Table		Page
7.1	Plant Protection System Trip Functions	26

LIST OF FIGURES

Figure		Page
2.1	Single-Phase Four-Quadrant Head Curves for the MOD-1 Semiscale Pump	9
2.2	Single-Phase Four-Quadrant Torque Curves for the MOD-1 Semiscale Pump	10
3.1	Combustion Engineering 2x4 Plant Configuration	16
3.2	Head Curve for St. Lucie Unit 1 Primary Coolant Pumps	17
6.1	Control Rod Shutdown Worth for St. Lucie-1 as a Function of Rod Insertion	21
6.2	Control Rod Insertion for St. Lucie-1 as a Function of Time from Holding Coil Release	22
6.3	Control Rod Shutdown Worth for St. Lucie-1 as a Function of Time from Holding Coil Release	23
7.1	Axial Shape Index Correction Function for the TM/LP Trip in St. Lucie-1	27
7.2	Power Correction Function for the TM/LP Trip in St. Lucie-1	28
7.3	Local Power Density Trip Function for St. Lucie-1	29
8.1	Controller Function Used in PTSPWR2	32
9.1	Moody Flow Factor	38
9.2	Fit to High Pressure Safety Injection Pump Flow for St. Lucie-1	39
9.3	Moderator Density Feedback for St. Lucie-1 with Uniform Core Temperature	40
9.4	Doppler Feedback for a Return to Power in a Steamline Break	41

1.0 INTRODUCTION AND SUMMARY

Several modifications to the Exxon Nuclear proprietary Pressurized Water Reactor (PWR) Simulation code, PTSPWR2[1,27], were implemented in order to model St. Lucie Unit 1. St. Lucie-1 is a Combustion Engineering (CE) 2x4 PWR, two hot legs and four cold legs, with an advanced CE analog reactor protection system (RPS). The modifications, discussed in the succeeding Sections 2 through 9, describe these changes. A discussion of the motivation for each change and its impact is described in Subsections 1.1 through 1.8.

The changes required to modify PTSPWR2 for use with St. Lucie-1, while extensive, were implemented in order to: (1) describe the analog trip system, (2) model the operation of the control system, (3) model the dynamics of imbalanced loops, (4) include the logic for a 2x4 plant, and (5) model a steam line rupture in a CE plant. With the exception of steam line rupture modeling, the changes were not substantial.

1.1 PUMP TRANSIENT MODELING CHANGES

Section 2.0 describes the dynamic model for pump coastdown used in the St. Lucie-1 analysis. Prior PTSPWR2 models had tabular input for pump coastdowns based on plant data. The present procedure provides a model which better describes four-pump coastdown during off-normal events[3] such as a transition to natural circulation following a steam line rupture.

1.2 PRIMARY LOOP DYNAMIC MODELING

Section 3.0 presents a derivation of the dynamic response equations for loop flow in a 2x4 plant. The approach described in the Section is considered to be more a realistic method for obtaining total flow response in the simulation of a single pump rotor seizure.

1.3 THERMAL CALCULATIONS

An extension of the PTSPWR2 model was also included in the St. Lucie-1 analysis to provide consistency between the thermal and hydraulic behavior.

1.4 CRITICAL HEAT FLUX CORRELATION

The critical heat flux correlation was changed to the Exxon Nuclear proprietary critical heat flux correlation, XNB[4].

1.5 SCRAM CURVES

Section 6.0 describes the calculation of the scram curve based on the axial shape. This modification provides a more realistic model of the implications of the limiting conditions of operation (LCOs), including the Technical Specification limit on scram speed. The model is less conservative than the prior PTSPWR2 models, which were all worst case curves.

1.6 PLANT PROTECTION SYSTEM

Section 7.0 describes the plant protection system (PPS) changes incorporated in PTSPWR2 specifically for St. Lucie-1. These differ from prior analyses since the St. Lucie-1 PPS differs from Westinghouse plants and even earlier CE analog plants.

1.7 PRESSURE CONTROL SYSTEM

The changes to the pressurizer pressure control system described in Section 8.0 were included to allow for easier steady-state balancing, to allow for control band overlaps, and to stop the heaters when the pressurizer water level fell below the heaters. The impact of these changes on prior PTSPWR2 analyses is negligible.

1.8 STEAM LINE BREAK

Section 9.0 describes the changes made to PTSPWR2 to model the feedwater flow, the break flow, and the heat removal during blowdown of the steam generators. The changes to steam line break flow and to heat removal were substantial in nature and removed considerable conservatism from the St. Lucie-1 steam line break simulation which had been present in prior analyses.

2.0 PUMP TRANSIENT PERFORMANCE

The hydraulic head produced by a reactor coolant pump can be described as a function of flow and shaft speed for steady state operation. In describing the transient behavior of the pump, two other characteristics become important: the pump torque requirements and the pump inertia. Combining these three dynamic properties (head, torque and inertia) with the dynamic properties of the loop provides a physical model for pump transient behavior.

2.1 PUMP DYNAMICS

The pressure increase produced by a pump depends on the flow through the pump and on the impeller speed. A typical four-quadrant pressure map for a centrifugal pump is shown in Figure 2.1. The lack of symmetry between positive and negative speeds shows the effect of vane and volute shapes. Similarly, Figure 2.2 shows a typical four-quadrant torque map for the same pump.

The basic equation of motion for the pump shaft is.

$$\tau = \frac{d}{dt}(I\omega)$$

where τ is the torque applied to the shaft, ω is the angular frequency of rotation of the shaft, and I is the moment of inertia of the rotating components of the pump. Torque can be subdivided into several subcategories:

$$\tau = \tau_{motor} - \tau_{pumping} - \tau_{friction} - \tau_{windage} \qquad (2.2)$$

where the motor torque just offsets the sum of all the other torques at steady state operation. The pumping torque is the torque shown in Figure 2.2. The windage torque is a motor loss and the friction torque summarizes the other mechanical losses.

Following a trip of the motor, there exists a large torque imbalance which results in an immediate decrease in the pump speed,

$$\frac{dN}{dt} = -\frac{60}{2\pi}\,\frac{1}{RPM_{RATED}}\,\frac{1}{I}\,[\,\Sigma\ \text{Torques}] \qquad (2.3)$$

where N is the pump shaft speed normalized to rated shaft speed and the conversion factors are included to change angular frequency to normalized speed. These conversion factors can be absorbed in an effective moment of inertia to simplify the form.

If one were provided with flow and shaft speed traces for pump coastdowns which were prototypic of the desired coastdowns in the sense of flow-to-rpm ratios, one could adjust I' in Equation 2.4 along with the parameters of the friction torque model to provide the best fit to the data[3]. In fact, of the three state variables; flow, shaft speed and pressure, any two may be used to adjust the model since the head curve, Figure 2.1, can be used to calculate the third variable. An alternative technique which may be employed when these data are not available is discussed in the following subsection.

2.2 IN SITU DYNAMIC MODELING

Data packages for plants are not often filled with the type of detail required to directly tune the parameters in Equation 2.4. It is still possible to adjust the parameters based on N-loop pump coastdown data. This is done by utilizing one additional relationship:

$$\frac{1}{144 g_c}\left(\frac{L}{A}\right)\frac{dW}{dt} = \Delta P_{pump} + \Delta P_{thermal\ head} - kVW|W| \qquad (2.5)$$

where W is the loop flow for either loop, V is the average specific volume, ΔP_{pump} is the pump head, $\Delta P_{thermal\ head}$ is the static pressure difference, and the last term is the total dynamic pressure drop. For simplicity, the N loops are assumed to be identical, reducing the need for a subscript on flow and for dynamic balancing of dissimilar loops. All loop flows are identical and the core flow is just N times the loop flow. Thus, Equation 2.5 provides a valid basis for loop dynamics for this symmetrical case which is typical of data available from most plants.

Equations 2.4 and 2.5, which incorporate the head curve, can be solved together to allow for parameter adjustment. Generally, this is what can be done using a PTSPWR2 simulation of the loss-of-coolant-flow event. Equation 2.5 is explicitly contained in PTSPWR2 and only Equation 2.4 need be implemented.

To adjust the parameters in Equation 2.4 using PTSPWR2 with the pump dynamics incorporated, one should recognize that the initial part of the coastdown is highly sensitive to I' and the latter part is dominated by friction. Thus, I' is adjusted by running a PTSPWR2 loss-of-coolant-flow simulation for 5 to 10 seconds and establishing the best choice of I'. A

coastdown to a stop allows for the identification of the onset of increased friction in the pump and an adjustment to the torque for speeds below that value causes the pump model to cease shaft rotation at the appropriate time.

2.3 IMBALANCED LOOP PUMP COASTDOWNS

In the preceding subsection, all N loops were assumed to be identical and a simple method for determining the parameters of the pump coastdown model based on limited plant data was presented. The resultant model is sufficient to describe the base event, the loss-of-coolant-flow, and to allow for a meaningful transition to natural circulation at low pump speeds. It does, however, still lack the unambiguous definition of the parameters which resulted from the analysis in which both RPM and flow were provided. This becomes obvious in asymmetrical transients such as a rotor seizure followed by a trip of the remaining pumps. These transients would only be accurately modeled if the inertances and dynamic pressure drops which are a part of PTSPWR2 were properly adjusted, since errors in identifying the appropriate pressure drops and/or inertances could be compensated for, to a great extent, by adjusting the pump dynamic parameters.

There are two types of data which can prove to be useful: (1) pressure drops across different components at steady state; and (2) pump coastdowns for less than N pumps. The first type of information helps to remove the ambiguity in the component pressure drops and is used to set the K-factors for the primary loop components. The off-normal pump coastdowns, less than N pumps operating, are sensitive to all of the parameters and, along with the N-pump coastdowns, provide the best source of dynamic hydraulic

data. In fact, the effective component inertances can be adjusted (loop-to-vessel ratios) to provide a fit to both N-pump and off-normal coastdowns. This is best done using PTSPWR2 for 5 to 10 second transient runs.

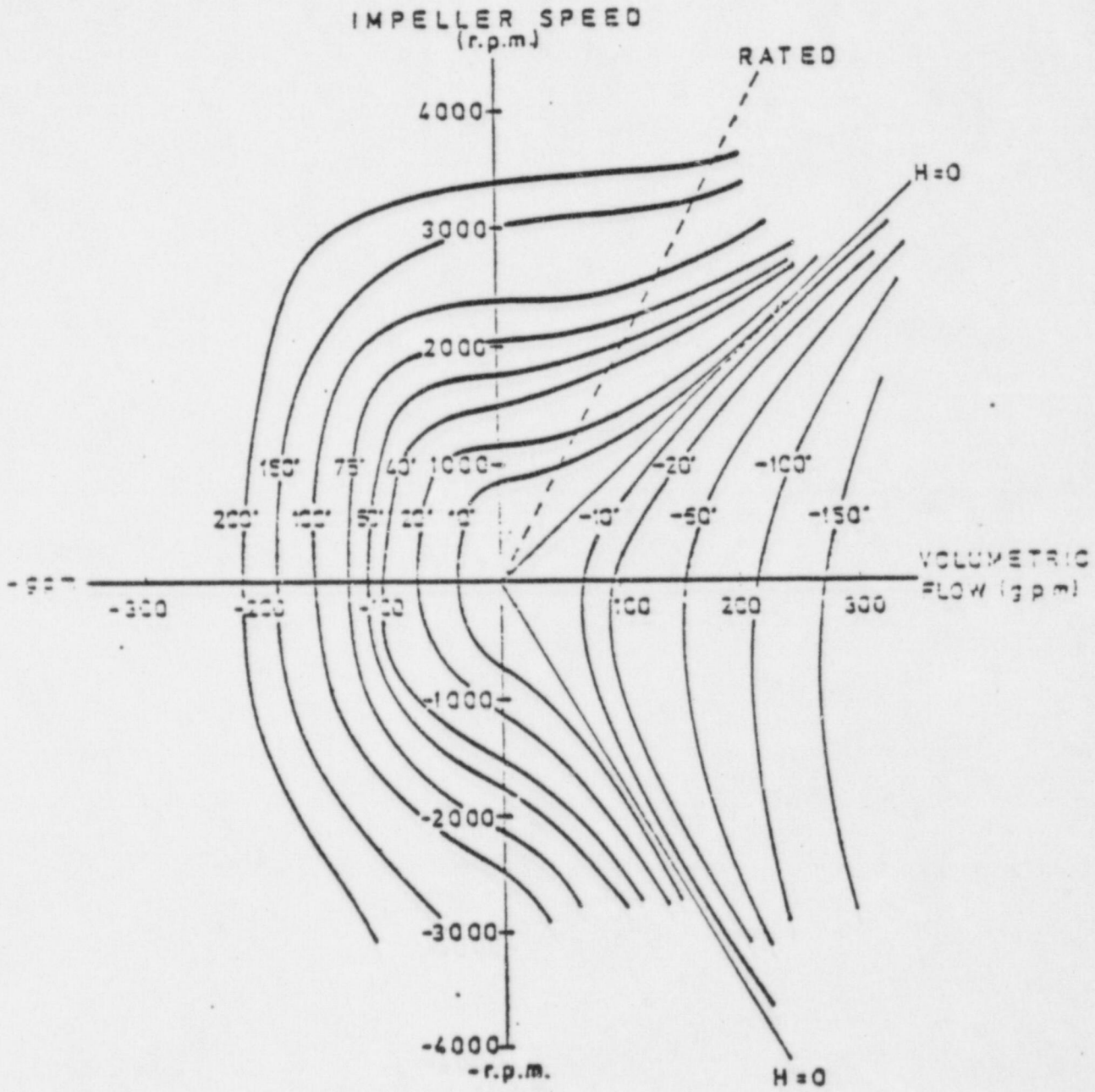


Figure 2.1 SINGLE-PHASE FOUR-QUADRANT HEAD CURVES FOR THE MOD-1 SEMISCALE PUMP

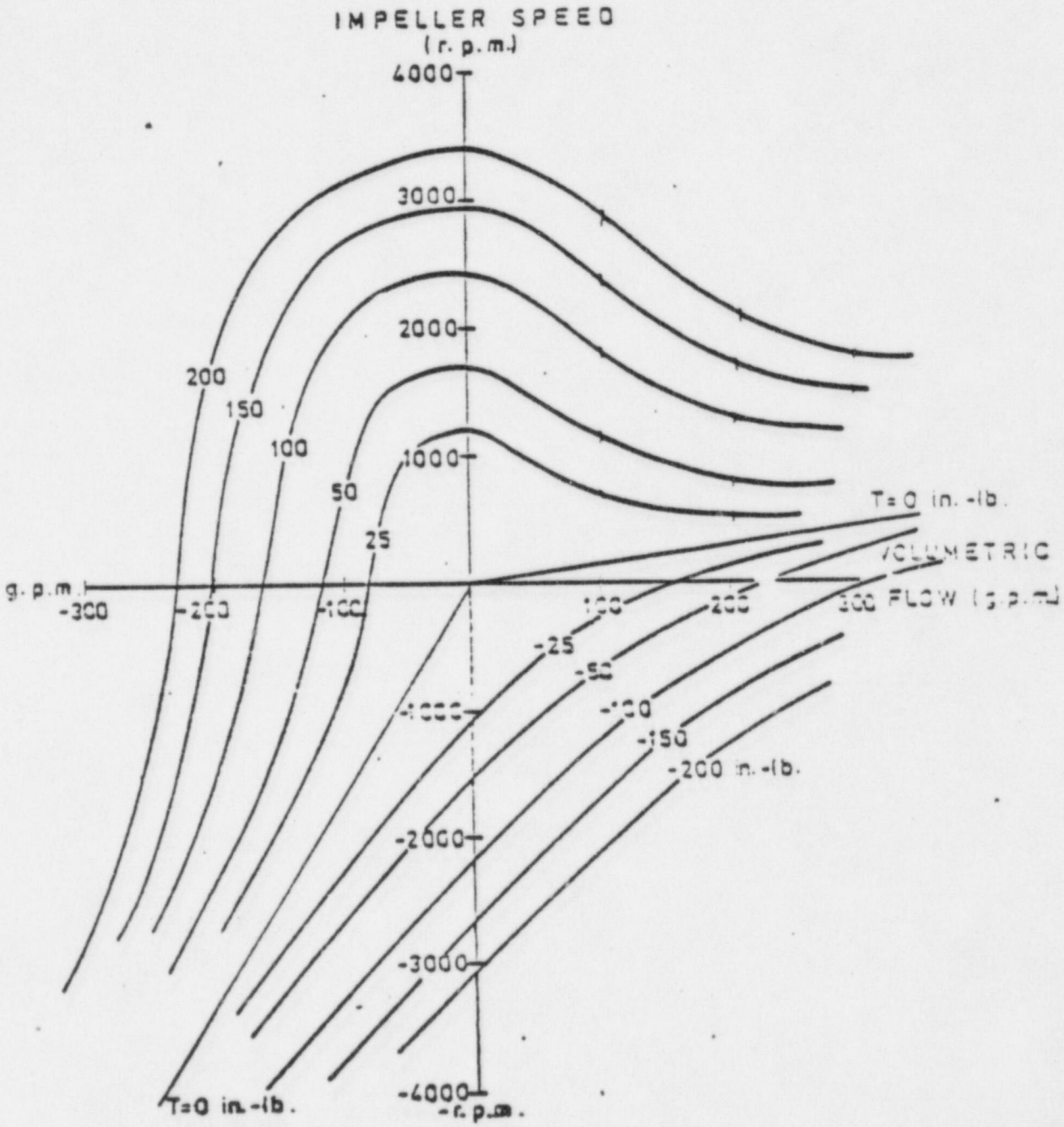


Figure 2.2 SINGLE-PHASE FOUR-QUADRANT TORQUE CURVES FOR THE MOD-1 SEMISCALE PUMP

3.0 PRIMARY LOOP HYDRODYNAMICS FOR CE 2x4 PLANTS

The configuration of a Combustion Engineering (CE) plant, shown in Figure 3.1, provides two hot legs and four cold legs. The transient flow response for this configuration is complicated by possible asymmetries in initiating events such as a single pump seizure. The following analysis rederives the dynamic response equations for primary coolant flows.

drop around the loop must be zero,

For the j^{th} loop,

[redacted]

[redacted]

Equations 3.14 through 3.27 along with the definition given in Equations 3.11 and 3.12 provide flow derivatives for all the primary coolant flows.

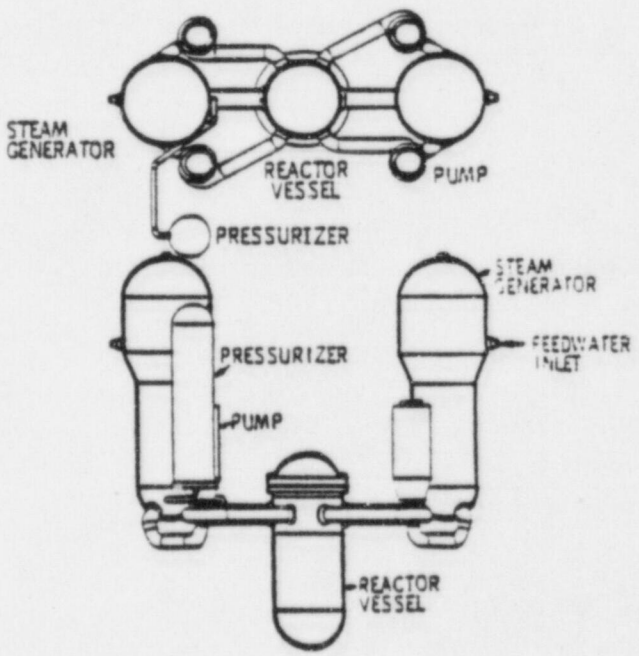


Figure 3.1 Combustion Engineering 2x4 Plant Configuration

4.0 THERMAL CALCULATIONS

The thermal calculations used in the Combustion Engineering version of PTSPWR2 differ only in that they take into account the dual cold legs in each loop. This required parallel calculations of the cold leg thermal quantities and a redefinition of the hot leg to cold leg delta-T. The thermal mixing equations at the junctions were also made dependent on the thermal and hydraulic conditions in all four cold legs.

5.0 CRITICAL HEAT FLUX CORRELATION

The Exxon Nuclear proprietary critical heat flux (CHF) correlation[4] was employed in the departure from nucleate boiling ratio (DNBR) calculations for pressures greater than 1450 psia. For pressures below this value, the MacBeth CHF correlation was modified to account for the increase in the latent heat of vaporization in the same way that the Barnett CHF correlation[5] was modified. The MacBeth CHF correlation was chosen for application to the low pressure cases because emptying of the pressurizer results in the operator tripping the main coolant pumps and much of the data for the MacBeth CHF correlation is at low flows.

The hot channel model remained unchanged with only the CHF correlation being replaced.

6.0 SCRAM CURVES

Reactivity inserted in the core as a result of an insertion of control rods varies with rod position and with the axial shape. Figure 6.1 shows this behavior for three axial shapes characterized by the axial shape index, ASI.

$$ASI = \frac{\text{(Power in lower half - Power in upper half)}}{\text{(Power in lower half + Power in upper half)}} \quad (6.1)$$

The rod position as a function of time following a scram is shown in Figure 6.2. This curve is representative of a typical rod position trace following release of the control element assembly (CEA) with the time scale adjusted so that the 90% insertion time is in agreement with Technical Specification requirements.

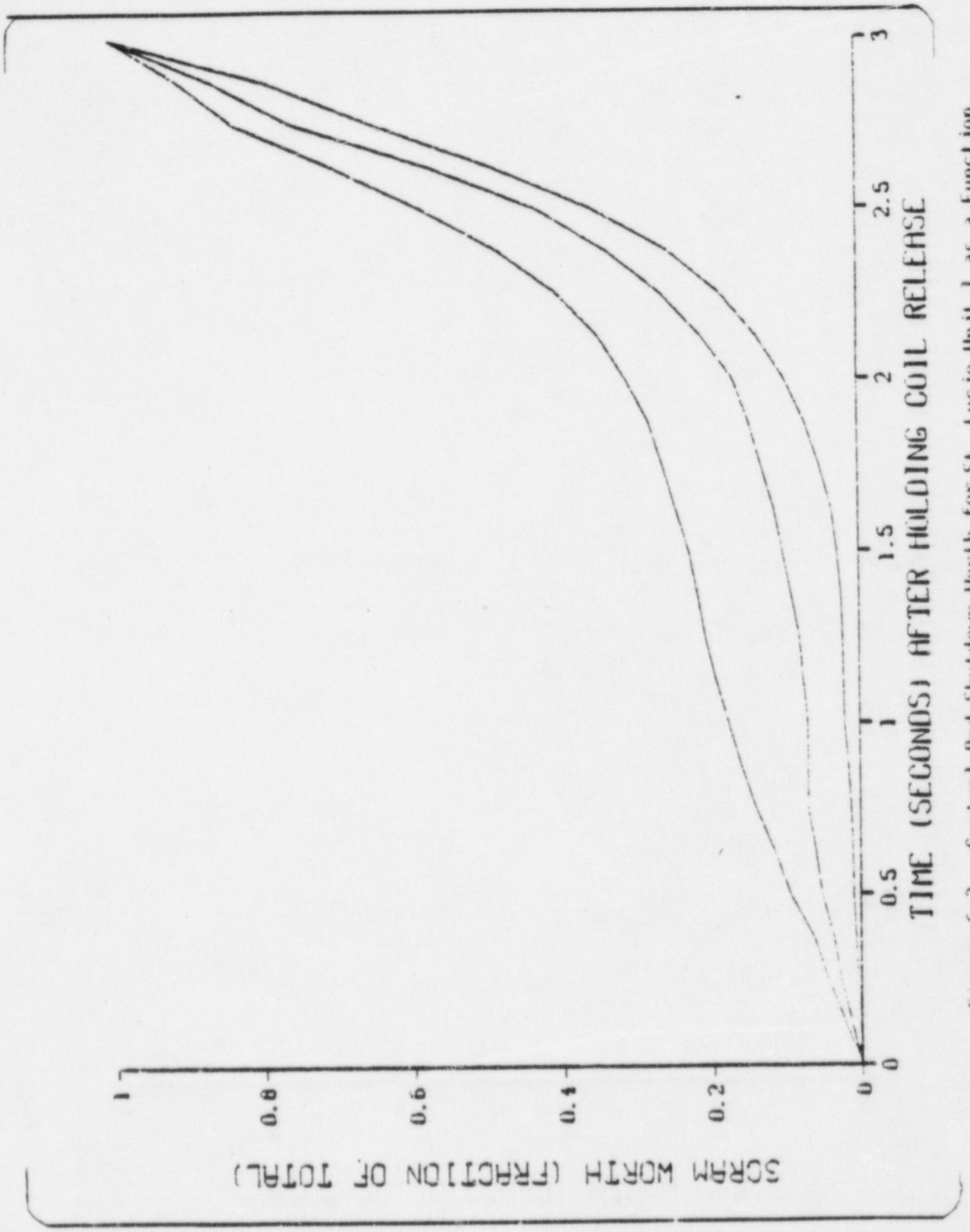


Figure 6.3 Control Rod Shutdown Worth for St. Lucie Unit 1 as a Function of Time from Holding Coil Release

7.0 PLANT PROTECTION SYSTEM

Changes were required in PTSPWR2 to incorporate the Plant Protection System (PPS) logic for a Combustion Engineering plant. Table 7.1 lists the eight trip functions implemented in PTSPWR2 to model St. Lucie Unit 1 and typical values of the trip setpoints, uncertainties and delays. Of the trip functions implemented in PTSPWR2, only three differ significantly from the functions already contained in PTSPWR2: Asymmetric Steam Generator Protection Trip (ASGPT), Thermal Margin/Low Pressure (TM/LP) trip, and the local power density trip (LPD).

The power used in the variable high power trip (VHPT), the TM/LP and the LPD trip is the maximum of neutron flux power and the thermal power, calculated as follows:

$$Q_{Auction} = MAX\ (Q_{NF}, Q_{TH}) \qquad (7.1)$$

Where Q_{NF} is the power based on the neutron flux and Q_{TH} is defined by,

$$Q_{TH} = 2700\ \Delta T\ (1.584 \times 10^{-2} + 2.238 \times 10^{-5}\ \Delta T + 2.363 \times 10^{-5}\ (T_{CL} - 427.5)\ ). \qquad (7.2)$$

Where ΔT is the average hot leg to cold leg difference, T_{CL} is the average primary cold leg temperature and 2700 is the rated power in megawatts of the plant for which this algorithm was used, SL-1.

To account for measurement errors in temperature and the time delay in the response of the RTDs used to measure the temperatures, $2^{o}F$ was subtracted from the cold leg temperature, $4^{o}F$ was subtracted from the average temperature difference, and both signals were passed through a first-order lag equation

with a time constant of 12 seconds. The power error, including a transient allowance, was included in $Q_{Auction}$ by subtracting 5% of the rated power, or 135 MWt, from $Q_{Auction}$.

The ASGPT was incorporated using the absolute value of the difference between steam generator pressures and providing a trip at,

$$|P_{SG1} - P_{SG2}| \geq 166 \text{ psi} \qquad (7.3)$$

Where $P_{SG1,2}$ represent the dome pressures in steam generators #1 and #2.

The TM/LP function is a calculated pressure at which the reactor trips. The following functional form was used:

$$P_{VAR} = 2061.0 \times A1(ASIp) * QR1(Q_{Auction}) + 15.85\ T_{CL} - 9076.66 \qquad (7.4)$$

Where A1 and QR1 are shown in Figures 7.1 and 7.2 and ASIp is the measured ASI, including a 0.06 offset to account for measurement error. The bias term includes an allowance of 126.66 psi to account for transient measurement errors in the pressure sensors.

The LPD trip function is shown in Figure 7.3 where the abscissa is the peripheral, or measured, ASI and the trip is compared to $Q_{Auction}$.

Table 7.1 Plant Protection System Trip Functions

Function	Value	Uncertainty	Time Delay* (seconds)
Variable High Power	107%	5% of rated	0.9
S.G. Level	37% of Span	5.5% of Span	1.4
Low Flow	95% of Design	2% of Design	1.15
Low S.G. Pressure	600 psia	22 psi	1.4
Antisymmetric S.G. Pressure	135 psi	31 psi	1.4
Pressurizer Pressure			
High	2400 psia	22 psi	1.4
Low	1600 psia	22 psi	1.4
TM/LP	Calculated	126.66 psi	1.4
Local Power Density	Calculated	5% of rated power	0.9

*Includes 0.5 seconds for the field to decay in the CEA holding coils.

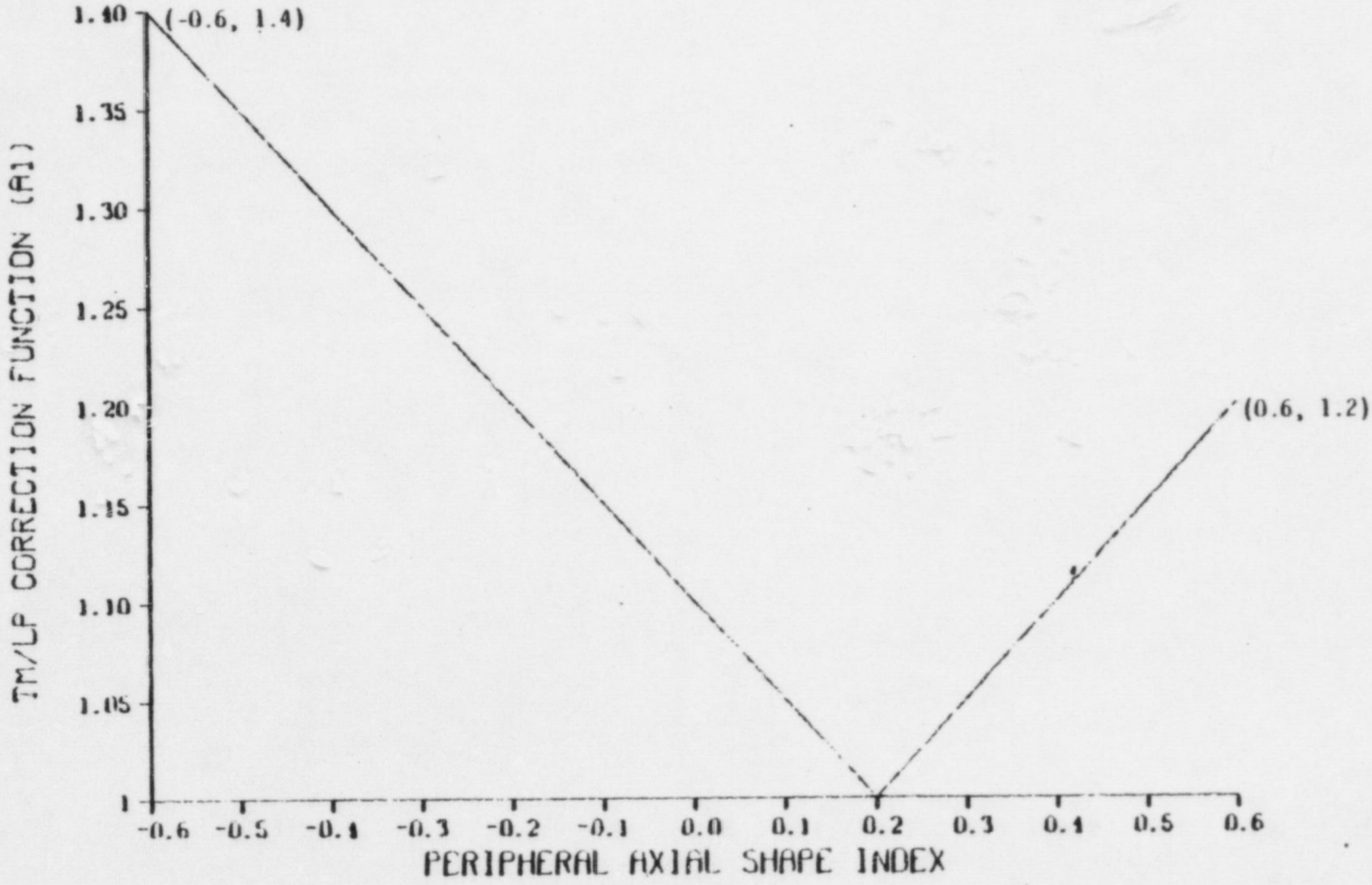


Figure 7.1 Axial Shape Index Correction Function for the TM/LP Trip in St. Lucie Unit 1

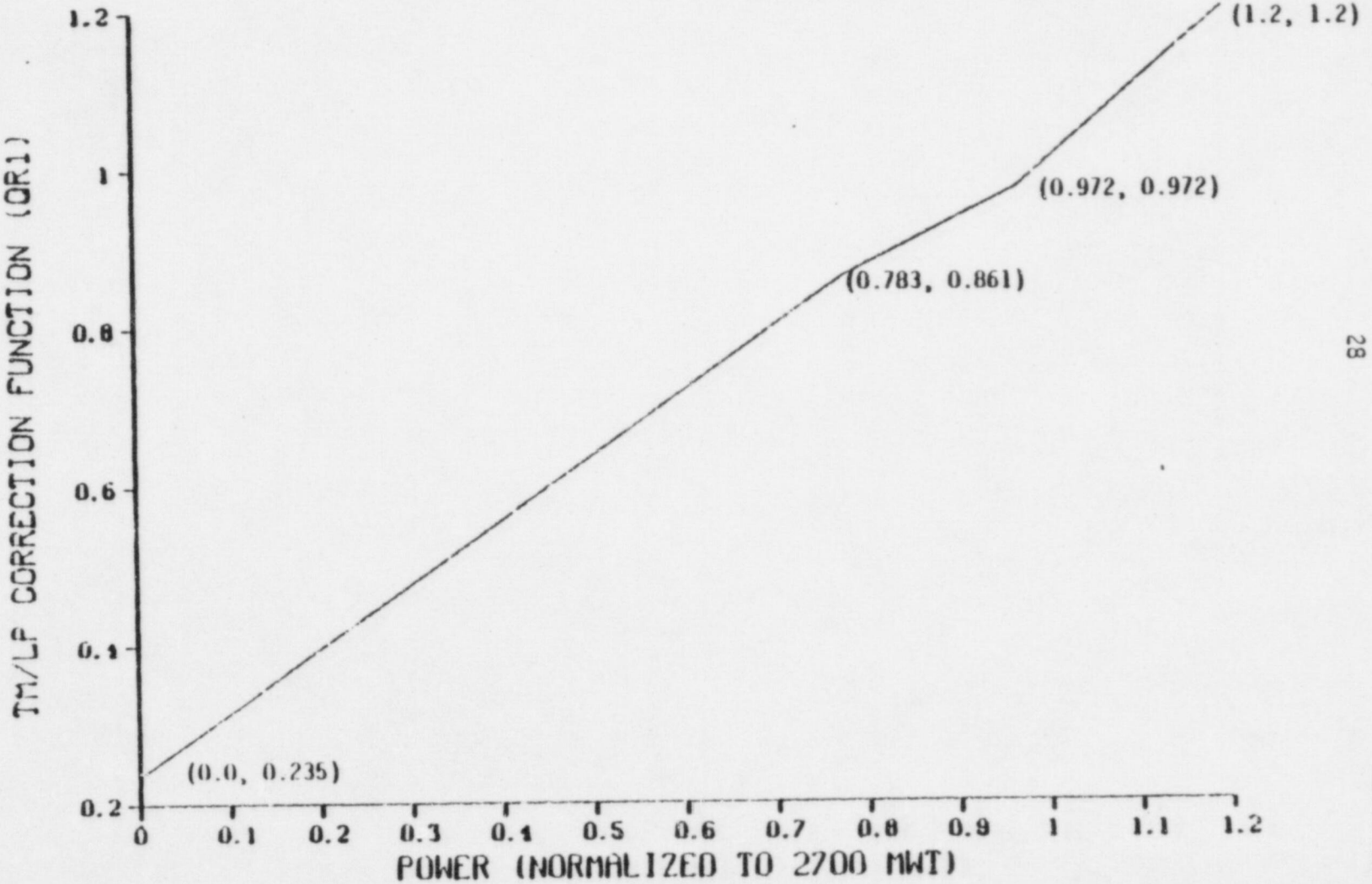


Figure 7.2 Power Correction Function for the TM/LP Trip in St. Lucie Unit 1

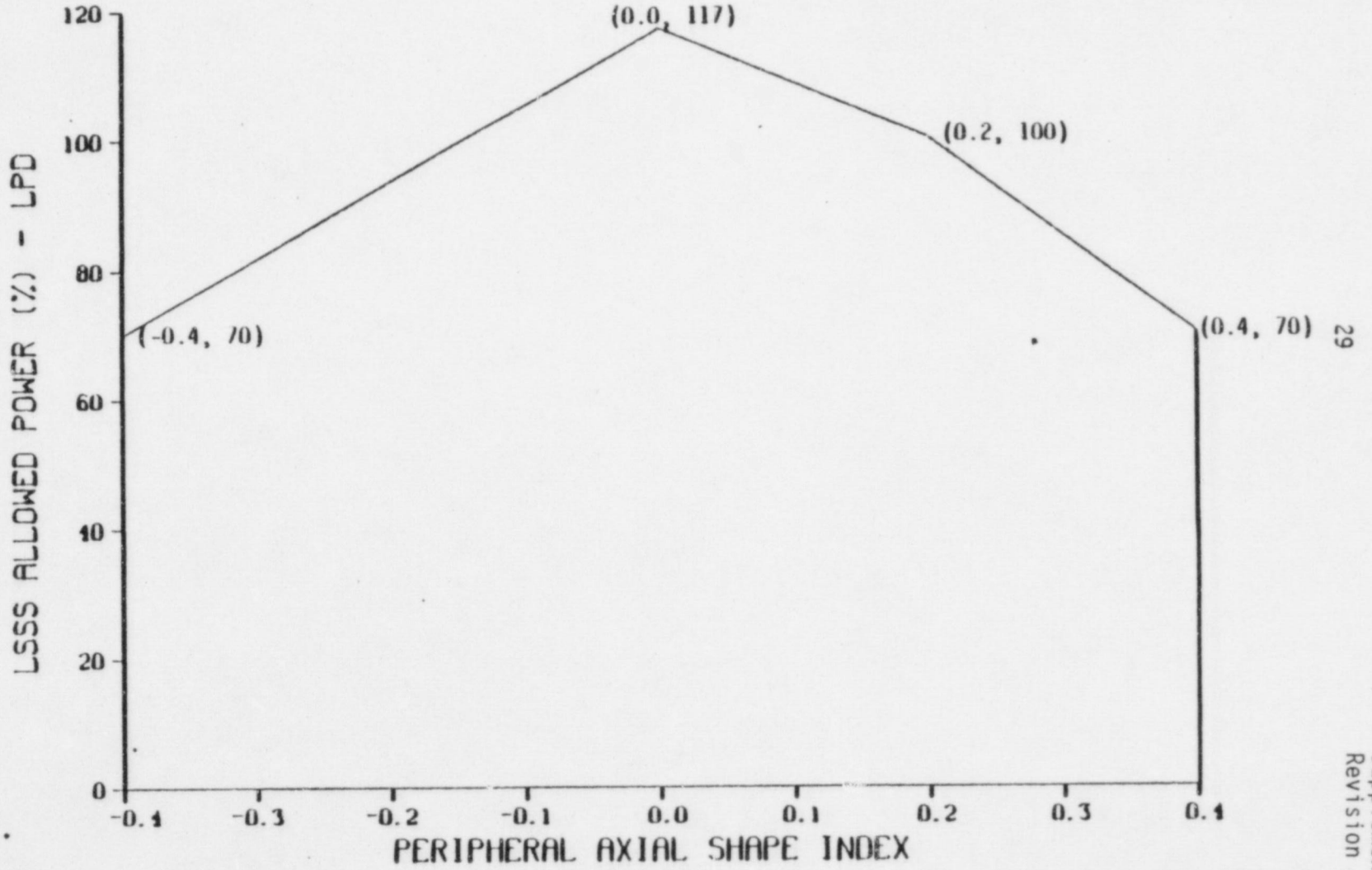


Figure 7.3 Local Power Density Trip Function for St. Lucie Unit 1

8.0 PRESSURIZER PRESSURE CONTROL SYSTEM

The pressurizer control model was modified in several ways: [

[

]

The backup heaters were modeled so that they came on when the linear controller was fully saturated. This is expressed by,

[

]

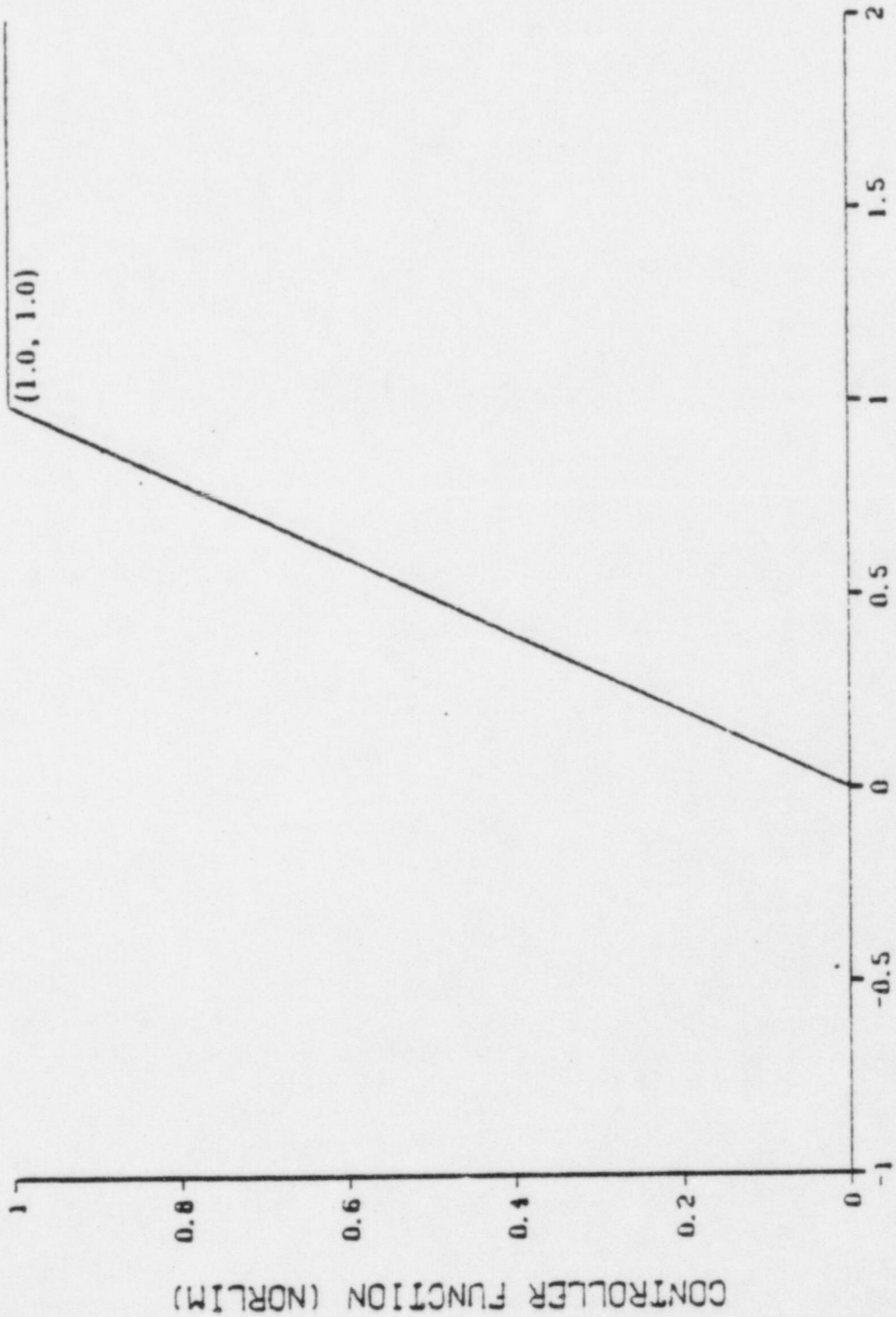


Figure 8.1 Controller Function Used in PTSPWR2

9.0 STEAM LINE BREAK

The steam line break model in PTSPWR2 was changed to model the action of feedwater flow, both normal and auxiliary, and to use the Moody choke flow curve[6] to model steam flow from the break. The model for steam flow into the steam dome during a blowdown was modified to conserve energy during the depressurization. Boron injection flow rates were based on a fit to the measured pump performance and moderator reactivity was calculated as a function of moderator density.

9.1 FEEDWATER FLOW

At the initiation of the simulation of a steam line break for two-loop operation at hot zero power (HZP) the feedwater flow was set to 16 lbm/second. The power being removed by this feedwater flow is approximately 13 MWt, which represents the pump work required to circulate the primary coolant.

Following the break, the flow was allowed to decrease linearly to zero over 60 seconds. Three minutes after the closure of the main steam isolation valves (MSIVs) auxiliary feedwater was initiated to both steam generators. The rates were set to represent run-out flow, 253.6 lbm/second, into the ruptured steam generator since its pressure had fallen to nearly atmospheric. The intact steam generator, isolated by the MSIVs, was at a much higher pressure and the assumed flow into this steam generator was therefore lower, 100 lbm/second.

The effect of the auxiliary feedwater flow initiation was to introduce a second cooldown transient corresponding to losses in the broken steam generator loop of around 214 MWt.

9.4 REACTIVITY MODELS

Boron injection was based on the performance of a single high pressure safety injection pump (HPSI) delivering borated water from the refueling water storage tank (RWST) at a concentration of 1720 ppm. The performance of the pump was fit to the actual test data for the three HPSI pumps (see Figure 9.2). The enthalpy and mass of the injected water was included in the calculation. Boron reactivity was calculated based on the average boron concentration in the core. This concentration was calculated by mixing the injected HPSI water with the cold-leg water and then propagating the mixed cold-leg water through the piping and the lower plenum and then to the core.

Moderator feedback was based on the average moderator density in the core using the curve in Figure 9.3. The density was obtained by adding a table look-up for the density of subcooled water as a function of temperature and pressure.

Doppler reactivity feedback was calculated using a linear function of fuel temperature when the reactor was subcritical,

$$\Delta K_{DOPPLER} = \alpha_{DOPPLER} (T_{FUEL} - T_{FUEL,0}), \qquad (9.11)$$

where T_{FUEL} is the average fuel temperature and the Doppler temperature coefficient was selected at a conservative value. When the reactor became critical, the Doppler Reactivity feedback was calculated as a function of power using the curve in Figure 9.4.

Figure 9.1 Moody Flow Factor

Figure 9.2 Fit to High Pressure Safety Injection Pump Flow for St. Lucie Unit 1

10.0 REFERENCES

1. "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTSPWR)," XN-NF-74-5(P), Rev. 2, Exxon Nuclear Company, Inc., Richland, Washington 99352, July 1983.

2. "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTSPWR): Appendix A," XN-NF-74-5(P), Rev. 2, Exxon Nuclear Company, Inc., Richland, Washington 99352, July 1983.

3. Nutt, W.T., Addition, S.L. and Parziale, E.A., "Application of Optional Estimation Techniques to FFTF Decay Heat Removal Analysis," Proceedings of ANS/ENS International Meeting on Fast Reactor Safety Technology, Seattle, Washington, August 1979.

4. Exxon Nuclear DNB Correlation for PWR Fuel Design," XN-NF-621(P), Rev. 1, Exxon Nuclear Company, Inc., Richland, Washington 99352, April 1982.

5. Hughes, E.D., "A Correlation of Rod Bundle Critical Heat Flux for Water in the Pressure Range 150 to 725 psia," IN-1412.

6. Moody, F.J., "ASME Transactions," p. 134, February 1965.

XN-74-5(NP)(A)
Revision 2
Supplement 2
Issue Date 10/8/86

DESCRIPTION OF THE EXXON NUCLEAR PLANT TRANSIENT SIMULATION MODEL FOR PRESSURIZED WATER REACTORS (PTS-PWR)

SUPPLEMENT 2

METHODOLOGY AND APPLICATIONS

Prepared by: _____
F. T. Adams
PWR Safety Analysis

Prepared by: _____
W. V. Kayser, Manager
PWR Safety Analysis

Prepared by: G. C. Cooke 1-27-84
G. C. Cooke, Manager
Model Development

Concur: J. C. Chandler 1/30/84
J. C. Chandler, Lead Engineer
Reload Fuel Licensing

Approve: _____
R. B. Stout, Manager
Licensing & Safety Engineering

Approve: _____
G. A. Sofer, Manager
Fuel Engineering & Technical Services

gf

EXXON NUCLEAR COMPANY, INC.

NUCLEAR REGULATORY COMMISSION REPORT 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 U.S. Nuclear Regulatory Commission as part of a technical contribution to facilitate safety analyses by licensees of the U.S. Nuclear Regulatory Commission 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 U.S. Nuclear Regulatory Commission in its review of this report, and under the terms of the respective agreements, by licensees or applicants before the U.S. Nuclear Regulatory Commission which are customers of Exxon Nuclear in their demonstration of compliance with the U.S. Nuclear Regulatory Commission's regulations.

Exxon Nuclear's warranties and representations concerning the subject matter of this document are those set forth in the agreement between Exxon Nuclear and the customer to which this document is issued. Accordingly, except as otherwise expressly provided in such agreement, neither Exxon Nuclear nor any person acting on its behalf:

A. Makes any warranty, or representation 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	SCOPE AND PURPOSE OF PTSPWR2 APPLICATIONS	2
3.0	GENERAL METHODS AND INPUTS	10
	3.1 DNBR EVALUATION	10
	3.2 REACTOR PROTECTION SYSTEM PARAMETERS	12
	3.3 GENERAL ASPECTS OF TRANSIENT EVALUATION	13
4.0	EVALUATION OF SPECIFIC TRANSIENTS	21
	4.1 EVENTS WHERE A MALFUNCTION CAUSES AN INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM	21
	4.1.1 Feedwater Malfunctions Leading to Increased Heat Removal and Excess Load Transients	22
	4.1.2 Secondary Safety or Relief Valve Opening (Small Steamline Break)	23
	4.1.3 Large Steamline Break	23
	4.2 MALFUNCTIONS THAT RESULT IN A DECREASE OF HEAT REMOVAL BY THE SECONDARY SYSTEM	24
	4.2.1 Loss of External Load, Turbine Trip, Loss of Condenser Vacuum and Steam Pressure Regulatory Failure Events	25
	4.2.2 Loss of AC Power	26
	4.2.3 Loss of Normal Feedwater	27
	4.2.4 Feedwater Pipe Break	27

TABLE OF CONTENTS (Cont.)

Section		Page
4.3	LOSS OF FLOW EVENTS	27
4.3.1	Flow Coastdown	28
4.3.2	Seized Pump Rotor	29
4.4	REACTIVITY AND POWER DISTRIBUTION ANOMOLIES	30
4.4.1	Uncontrolled Control Rod Bank Withdrawal from a Subcritical or Low Power Condition	31
4.4.2	Uncontrolled Control Rod Bank Withdrawal at Power	31
4.4.3	Control Rod Misoperation	33
4.4.4	Inactive Loop Startup	34
4.4.5	Boron Dilution	34
4.4.6	Fuel Assembly Misloading	35
4.4.7	Rod Ejection Transients	35
4.5	TRANSIENTS INVOLVING INCREASED PRIMARY COOLANT INVENTORY	35
4.6	TRANSIENTS INVOLVING DECREASED PRIMARY COOLANT INVENTORY	35
4.6.1	Inadvertent Opening of a Pressurizer Relief or Safety Valve	36
4.6.2	Steam Generator Tube Rupture Event	36
4.6.3	LOCA	36
4.7	RADIOLOGICAL RELEASES FROM A SUBSYSTEM OR COMPONENT	36
4.8	ANTICIPATED TRANSIENTS WITHOUT SCRAM	36

TABLE OF CONTENTS (Cont.)

Section		Page
5.0	REFERENCES	37
APPENDIX A	- PTSPWR2/XCOBRA-IIIC INTERFACE	A-1
APPENDIX B	- UTILIZATION OF AND INPUTS FOR PTSPWR2	B-1
B.1	SPECIFICATION OF CONTROL SYSTEM CONFIGURATION	B-1
B.2	SIMULATION OF POSTULATED EVENTS	B-3
B.3	PTSPWR2 ACCOMMODATION TO PWR NSSS DESIGN DISSIMILARITIES	B-7
B.4	PLANT SPECIFIC DATA SOURCES	B-8

LIST OF TABLES

Table		Page
2.1	Plant System Changes Evaluated with PTSPWR2	6
2.2	Description of Required Standard Review Plan Chapter 15 PWR Safety Analyses and Listing of ENC Codes or Methods Used in Analyses	7
3.1	Core Parameter Uncertainties	16
3.2	DNB Parameters	17
3.3	Reactor Protection System Parameters	18
3.4	Plant Condition Parameters	19
B.1	PTSPWR2 Generic Input Parameters	B-12
B.2	CPFUEL vs. Fuel Temperature: Typical Fuel Specific Heat	B-13
B.3	KFUEL vs. Fuel Temperature: Typical Fuel Thermal Conductivity	B-14
B.4	XNSCRT vs. Scram Time: Typical Scram Worth Insertion Curve	B-15

1.0 INTRODUCTION

PTSPWR2 is a computer code to evaluate plant transient events in pressurized water reactors. It has been accepted by the NRC for plant specific application. Models and input to the PTSPWR2 simulation code are described in XN-74-5(P), Revision 2.[1] Modifications to the PTSPWR2 code required to simulate the St. Lucie Unit 1 reactor, a Combustion Engineering 2x4 PWR, are described in Supplement 1[2] to that document. These documents are currently under NRC review. The NRC Staff has further requested that Exxon Nuclear Company (ENC) provide a detailed description of the applications methodology for evaluation of plant transients using the PTSPWR2 code. This methodology document will describe how the PTSPWR2 code is applied to plant transients such that conservative results are assured.

2.0 SCOPE AND PURPOSE OF PTSPWR2 APPLICATIONS

The PTSPWR2 applications that are covered by this methodology document are for situations where a comprehensive safety evaluation of the plant has already been performed and documented in the plant FSAR. The PTSPWR2 code is used by ENC to evaluate the safety margin impact of a change to ENC reload fuel and the safety margin impact of certain plant configuration or operation changes that may be coincident with the change to ENC reload fuel. The nature of these plant configuration or operation changes is indicated in Table 2.1.

Generally, the fuel or other plant changes considered by ENC are sufficiently limited in scope that most aspects of the FSAR safety evaluation such as functional design and reliability of reactor protection systems, plausible initiating events and event sequences, guidelines for operator intervention, etc. remain intact and need not be reevaluated by ENC. On the other hand, the fuel or other changes considered by ENC invariably do impact DNB margins and may involve changes to allowable power peaking. The changes are usually small but nevertheless do require evaluation to ensure conservatism.

The plant transient events analyzed by PTSPWR2 are primarily those for which major system parameter changes and adverse consequences are limited by the reactor protection systems. Most of these transients involve potential loss of DNB margin. Other transients such as LOCA, rod ejection events, fuel misloading events, etc., which are typically more severe, are analyzed with other ENC computer codes.

Table 2.2 categorizes and lists transient events that are considered in PWR safety analyses. This table follows the NRC Standard Review Plan

(SRP)[3] and indicates those transients for which PTSPWR2 is applied. In general, it is applied to those transients in which single phase conditions exist in the primary coolant loops, the secondary liquid levels remain below the steam separators, and pressurizer packing does not occur. For long term cooling transients such as the loss of feedwater event, where time periods of interest are large compared to transport times for the primary loop, ENC's SLOTRAX code[4] rather than PTSPWR2 is usually applied.

The events in Table 2.2 are considered as either anticipated operational occurrences (AOOs) or postulated accidents (PAs) depending upon the probability or frequency of occurrence. The postulated accidents are delineated in Table 2.2 consistent with their classification in the SRP and FSARs. For the events of moderate frequency, i.e., AOOs, the two principal criteria addressed in ENC PTSPWR2 analyses are:

- The specified acceptable fuel design limits (SAFDLs) must not be exceeded.
- System pressures must not exceed 110% of design values to assure that the pressure boundary will not be breached during normal operation or AOOs.

The principal SAFDL of interest is the 95/95 DNBR limit based on an acceptable DNB correlation. The other SAFDL of interest is the avoidance of fuel centerline melt. For postulated accidents, the principal acceptance criteria are that calculated doses at the site boundary shall be within the guidelines specified by 10 CFR Part 100 and that core damage be limited so that the core remains coolable. In addition, specific criteria are normally specified for each accident.

PTSPWR2 is used to analyze the seized pump rotor and the main steamline break accidents. Both of these accidents require evaluation of DNBR and comparison against the applicable 95/95 DNBR limit values. When the limit value is exceeded, ENC calculates the number of rods in the core expected to experience DNB and these rods are conservatively assumed to fail. This evaluation is performed to ensure that the extent of core damage is limited and that 10 CFR 100 requirements for these accidents are satisfied.

Evaluation of DNB margins or DNBRs is a major aspect in applications of PTSPWR2 to both AOOs and PAs. The base method of DNBR evaluation is the XCOBRA-IIIC[5,6] PWR core thermal-hydraulics model which has been accepted by the NRC. With this model, ENC performs detailed core and subchannel analyses and implements the W-3 CHF correlation or ENC's XNB CHF correlation.[7] With XCOBRA-IIIC pertinent details of the core configuration, such as assembly-by-assembly fuel geometry, assembly-by-assembly hydraulic characteristics, and the core power peaking and distribution, are taken into account.

When calculating the thermal margin, the principal role of PTSPWR2 is to establish the changes that occur in core boundary conditions during transient events. The core boundary conditions relevant to DNB are core pressure, total core flow, core inlet temperature and fuel rod surface heat flux. Further details of the PTSPWR2/XCOBRA-IIIC interface are given in Appendix A.

Conservatism in the application of PTSPWR2 is accomplished at three levels in the PTSPWR2 transient evaluation methodology. These are:

1. Conservatism in the DNBR calculation by conservatively treating uncertainties in reactor core parameters and by using bounding power peaking factors. The reactor core parameters to which

uncertainties are applied are core power, core flow, core pressure and core inlet temperature.

2. Conservatism in the treatment of the reactor protection system functions. This includes a conservative account of reactor trip function uncertainties and trip delays, and a conservative treatment of reactor scram, both in terms of scram worth and timing. Also included in this category are the use of conservative pressurizer safety valve settings and relief capacity.

3. Evaluation of transients in a conservative manner over a range of plant operating conditions so as to maximize the challenge to the reactor protection systems.

Conservatism at each of these three levels provides comprehensive assurance of overall conservatism in PTSPWR2 applications.

Table 2.1 Plant System Changes Evaluated with PTSPWR2

- Fuel assembly flow resistance changed due to reload fuel design.
- Neutron kinetics feedback coefficients outside the range considered in previous safety analyses.
- Plant operation at a higher power level.
- Plant operation on a significantly different T_{ave} program than previously analyzed.
- Plant operation with steam generator tube plugging significantly greater than previously considered.
- Different event sequence or failure mode previously not considered for FSAR event.
- Revised Reactor Protection System setpoints or logic.

Table 2.2 Description of Required Standard Review Plan Chapter 15 PWR Safety Analyses and Listing of ENC Codes or Methods Used in Analyses

			SRP Number
SECTION 1 - Increase in Heat Removal by the Secondary System			
Decrease in Feedwater Temperature	AOO	PTSPWR2	15.1.1
Increase in Feedwater Flow	AOO	PTSPWR2	15.1.2
Increase in Steam Flow	AOO	PTSPWR2	15.1.3
Inadvertent Opening of Steam Generator Relief/Safety Valve	AOO	PTSPWR2	15.1.4
Steamline Break Inside/Outside Containment	PA	PTSPWR2/ New Methodology*	15.1.5
Radiological Consequences of Steamline Break Outside of Containment	PA	ENC Radiological Method (Ref.8)	15.1.5 Appendix A
SECTION 2 - Decrease in Heat Removal by Secondary System			
Loss of External Load	AOO	PTSPWR2	15.2.1
Turbine Trip	AOO	PTSPWR2	15.2.2
Loss of Condenser Vacuum	AOO	PTSPWR2	15.2.3
Steam Pressure Regulatory Failure	AOO	PTSPWR2	15.2.5
Loss of AC Power to Station Auxiliaries	AOO	PTSPWR2	15.2.6
Loss of Feedwater Flow	AOO	SLOTRAX/PTSPWR2	15.2.7
Feedwater System Pipe Breaks Inside/Outside Containment	PA	Generally covered by FSAR	15.2.8

*A new steamline break accident methodology is being developed by ENC for future applications. This new methodology does not involve PTSPWR2.

Table 2.2 (Continued)

			SRP Number
SECTION 3 - Decrease in Reactor Coolant Flow Rate			
Loss of Forced Coolant Flow	AOO	PTSPWR2	15.3.1
Primary Pump Rotor Seizure	PA	PTSPWR2	15.3.3
Primary Pump Shaft Break	PA	PTSPWR2	15.3.4
SECTION 4 - Reactivity and Power Distribution Anomalies			
Uncontrolled Rod Assembly Withdrawal at Low Power	AOO	PTSPWR2	15.4.1
Uncontrolled Rod Assembly Withdrawal at Power	AOO	PTSPWR2	15.4.2
Control Rod Misoperation	AOO	PTSPWR2/XCOBRA-IIIC	15.4.3
Startup of Inactive Loop	AOO	Generally covered by FSAR	15.4.4
System Malfunction Causing Decrease in Boron Concentration	AOO	PTSPWR2	15.4.6
Fuel Misloading Events	AOO	ENC Neutronic Analyses	15.4.7
Spectrum of Rod Ejection Accidents	PA	ENC Rod Ejection Methods (Ref.9)	15.4.8
Radiological Consequences of a Control Rod Ejection Accident	PA	ENC Radiological Methods (Ref.8)	15.4.8 Appendix A
SECTION 5 - Increase in Reactor Coolant Inventory			
Inadvertent Operation of High Pressure ECCS or Malfunction of CVCS	AOO	PTSPWR2	15.5.2

Table 2.2 (Continued)

			SRP Number
SECTION 6 - Decrease in Reactor Coolant Inventory			
Inadvertent Opening of PORV	AOO	PTSPWR2	15.6.1
Radiological Consequences of Failure of Small Line Carrying Primary Coolant Outside Containment	PA	Generally covered by FSAR	15.6.2
Radiological Consequences of Steam Generator Tube Failure	PA	Generally covered by FSAR	15.6.3
Spectrum of LOCAs	PA	ENC LOCA Methodology (Ref. 10)	15.6.5
Radiological Consequences of DBA LOCA Including Containment Leakage Contribution	PA	ENC Radiological Method (Ref. 8)	15.6.5 Appendix A
Radiological Consequences of DBA LOCA: Leakage from Engineered Safety Feature Components Outside Containment	PA	ENC Radiological Method (Ref. 8)	15.6.5 Appendix B
SECTION 7 - Radiological Release from a Subsystem or Component			
Radiological Consequences of Fuel Damaging Accident	PA	ENC Radiological Method (Ref. 8)	15.7.4
Radiological Consequences of Spent Fuel Cask Drop Accident	PA	ENC Radiological Method (Ref. 8)	15.7.5
SECTION 8 - Anticipated Transients Without Scram			
Anticipated Transient Without Scram	PA	Generally covered by FSAR	15.8

3.0 GENERAL METHODS AND INPUTS

This section covers general methods and inputs in applications of PTSPWR2. The section is organized to address the three levels at which conservatisms are introduced to ensure valid results. Subsection 3.3, dealing with transient evaluation, is general in nature. The specific evaluation of each transient for which PTSPWR2 is applicable is detailed in Section 4.0.

3.1 DNBR EVALUATION

DNBR is evaluated based on core coolant and heat flux boundary conditions from PTSPWR2. Calculation of DNBR is done with XCOBRA-IIIC or with PTSPWR2 as discussed in Appendix A. When DNBR is evaluated during a transient, individual uncertainties for reactor power, core pressure, core inlet temperature, and core flow are taken into account.

ENC uses the conservative uncertainty values for these parameters established in the applicable FSAR. As an example, typical values for three of the uncertainties in Westinghouse plants are 2% power uncertainty, 30 psi pressure uncertainty and 4°F core temperature uncertainty. With respect to establishing a conservatively low core flow for DNBR evaluation, ENC sometimes departs from the FSAR.

Reactor trip functions used in prior safety analyses are generally identified in the FSAR. Where applicable, these trip functions take into account one or more of the four core parameter uncertainties being discussed as well as other factors such as instrument drift. Table 3.1 identifies which of the core parameter uncertainties are accounted for in the various reactor trip functions in Westinghouse and Combustion Engineering (C.E.) plants

The simultaneous imposition of all four core parameter uncertainties,

, is considered to provide substantial conservative margin in the evaluation of DNBRs for transient events.

The second area in which conservatisms are introduced into DNBR evaluation is in core power peaking. Radial power peaking is assumed to be at Technical Specification limits, including allowance for measurement and engineering uncertainty factors. The axial power peaking factor used in evaluation of DNBR is a value equal to or greater than the axial power peaking allowed, or implied by linear heat generation rate (LHGR) limits, in the Technical Specifications. The Technical Specification axial power peaking or LHGR limits are generally based upon LOCA considerations. The axial power peaking margin to limits in ENC PTSPWR2 analyses is generally considered sufficient to bound any small power peaking changes that might occur in rod

withdrawal transients. In some ENC analyses, a spectrum of possible axial power shapes is generated by neutronic analysis of xenon oscillations and the most limiting of these is used in the DNBR evaluation.

Table 3.2 summarizes the DNBR calculation parameter uncertainties just discussed. Conservatism in power peaking and conservatism in core thermal-hydraulic parameters ensures conservative DNBR evaluation for every set of core boundary conditions provided by PTSPWR2 in the safety analysis of events to which this code is applied.

3.2 REACTOR PROTECTION SYSTEM PARAMETERS

The reactor protection system parameters of interest are:

Reactor trip setpoints and setpoint functions

Uncertainties in trip setpoints and setpoint functions

Safety valve relief capacity

Safety valve setpoints and uncertainties

Scram delays

Normalized scram reactivity versus time

Scram worth

These parameters are important since they have a direct bearing on the ultimate magnitude of possible perturbations in reactor conditions such as reactor power, pressure, etc. in transient events analyzed with PTSPWR2.

In ENC analyses the reactor protection system setpoints and functions are based on the plant Technical Specifications or other plant documents that specify these settings. Conservative trip uncertainties and trip delays are established by the plant FSAR or Technical Specifications. For Westinghouse plants, ENC uses a generic normalized scram reactivity

versus time curve that assumes a bounding rod drop time, without delays, of 1.8 seconds. This function is based on ENC neutronic calculations and is given in Appendix B. When rod drop time is greater than 1.8 seconds, this curve can be conservatively employed if the difference in rod drop times is added to the scram delay. Alternatively in this case, the FSAR scram curve can be used. For C.E. plants, either the FSAR scram insertion curve or a conservative ENC generated scram curve(s) is used. Scram worth is based on ENC neutronic calculations for each plant application. These calculations assume that the highest worth rod does not insert. Additional conservatism is introduced by reducing the calculated scram worth by | | prior to input in PTSPWR2 analyses.

Table 3.3 summarizes the bases for the reactor protection system parameters used in ENC PTSPWR2 analyses. The conservative evaluation of each of these individual parameters ensures that the potential extremes of perturbations in reactor conditions are bounded in ENC PTSPWR2 calculations that involve one or more of the reactor protection systems.

3.3 GENERAL ASPECTS OF TRANSIENT EVALUATION

Generally ENC uses the existing plant FSAR to determine the bases behind prior analysis results and the reasons why certain transients tend to be limiting. For example, it is important to understand the conditions, and the transient scenarios that might produce these conditions, in which the reactor protection systems provide the least margin to DNBR limits. With this understanding, the effect of changes in fuel design or other plant parameters that ENC is addressing can then be properly evaluated in PTSPWR2

safety analyses. This understanding also enables ENC to change or modify assumptions made in the original analysis if they are no longer applicable or are unnecessarily conservative. Section 4.0 summarizes much of this knowledge based upon the safety analyses conducted to date by ENC for Westinghouse and C.E. plants.

For most of the transients to which PTSPWR2 is applied, the reactor protection systems (RPSs) act to limit the consequences of the event. For many of these transients, the computed trip functions are the ones that provide reactor protection. The computed trip functions are the over-temperature ΔT and the overpower ΔT trip functions in Westinghouse plants and the thermal margin/low pressure (TM/LP) and local power density (LPD) trip functions in C.E. plants. These computed trips are designed to protect SAFDLs and limit vessel pressurization independent of the exact nature of the transient event. For example, in Westinghouse reactors the overtemperature ΔT trip function and in C.E. reactors the TM/LP trip function have been designed to protect the DNBR SAFDL, and they are based upon steady state thermal-hydraulic analyses with parameters that conservatively bound a spectrum of transient events. Thus, PTSPWR2 analyses to evaluate safety margins can also be viewed as analyses to verify the adequacy of the RPSs to prevent violation of SAFDLs and vessel pressure limits. With the view that transient analyses should serve to verify the bases for the computed reactor trips, it is sufficient if PTSPWR2 analyses address the potential rates of change in system parameters and combinations of extreme conditions that the computed reactor trips may experience. Accuracy of PTSPWR2 in predicting the

exact values of all system parameters during the postulated transient is not important when this approach is adopted since the computed reactor trips are designed to function adequately over a complete spectrum of perturbed conditions.

An important aspect of PTSPWR2 transient evaluations is consideration of the potential range of conditions encountered in reactor operation. For plant transient analyses using PTSPWR2, these conditions envelope operation as allowed in the plant Technical Specifications from hot zero power to hot full power and from beginning to end of cycle operation. PTSPWR2 input parameters related to operating conditions are detailed in Table 3.4. Among these, the moderator temperature coefficient is one of the more significant parameters related to plant power response since it varies greatly from beginning-of-cycle (BOC) to end-of-cycle (EOC) and can be a major source of reactivity insertion. Generally, BOC and EOC reactor kinetics parameters are not mixed in a PTSPWR2 evaluation of a single transient. Rather, if the results of the transient are expected to be dependent on BOC or EOC kinetics, the transient is run first for BOC and then for EOC.

Consideration of the potential extremes in reactor condition parameters in PTSPWR2 analyses ensures applicability of PTSPWR2 analyses to the full spectrum of reactor operations. Further considerations in the evaluation of transient events tend to be event specific and are detailed in Section 4.0.

Table 3.1 Core Parameter Uncertainties

Reactor Trip Function	Core Parameter Uncertainties Included in Trip Function
High Nuclear Flux or Variable High Power	Power
Low Flow	Flow
Overtemperature ΔT or Thermal Margin/Low Pressure (TM/LP)	Power, pressure, temperature
High or Low Pressurizer Pressure	Pressure
Overpower ΔT or Local Power Density (LPD)	Power, temperature
All other cases, including no trip	

Table 3.2 DNB Parameters

Parameter	Source/ENC Basis	Comment
Reactor Power Uncertainty	FSAR	Usually 2%
Core Pressure Uncertainty	FSAR	Usually ± 30 psi for W plants
Core Inlet Temperature Uncertainty	FSAR	Usually ± 4°F for W plants
Core Flow	FSAR or plant measurement with allowance for measurement uncertainty	For a few transients where it is important to maximize primary to secondary heat transfer (e.g., steam line break accident), maximum flow is considered.
Initial Core Radial Peaking	Tech. Specs.	-
Radial Peaking Uncertainty	Tech. Specs.	Includes power distribution measurement uncertainty and engineering factor.
Axial Peaking	Axial peaking in conjunction with radial peaking must bound Tech Spec allowed total peaking	1.55 center peaked axial is used for Westinghouse plants or demonstrated equivalent. C.E. plants used bounding axial based on neutronic analysis of xenon oscillations

Table 3.3 Reactor Protection System Parameters

Parameter	Source/ENC Basis	Comment
RPS Settings and Functions, including Safety Valve Setpoints	Tech. Specs. or other utility documentation	Some information can be found in FSAR.
Trip and Safety Valve Setpoint Uncertainties	FSAR, utility documentation	-
Safety Valve Capacity	FSAR	-
Scram Delay	FSAR, Tech. Specs., utility documentation	-
Normalized Scram Reactivity versus Time	Appendix B for W plants; alternatively, FSAR for W plants, FSAR or ENC neutronic analysis for C.E. plants; FSAR for C.E. plants	ENC calculated generic curve for W plants is given in Appendix B. 18
Scram Worth with One Withdrawn Control Rod	ENC neutronics analyses	[redacted] uncertainty is applied.

Table 3.4 Plant Condition Parameters

Parameter	Source/ENC Basis	Comment
Moderator Temp. Coefficient - BOC	Tech. Spec.	-
Moderator Temp. Coefficient - EOC	FSAR or ENC neutronic analysis	Generally is bounding of plant/ cycle specific operation by [redacted]
Moderator Pressure Coefficient	ENC analysis	Always consistent with moderator temperature coefficient
Doppler Coefficient	ENC neutronic analyses	Minimum magnitude at BOC; maximum magnitude at EOC. [redacted] uncertainty is added to PTSPWR2 analyses to ensure bounding calculations
Shutdown Margin	Tech. Spec.	-
Neutron Lifetime	ENC neutronic analyses	Should be conservatively low and bound plant and cycle specific ENC neutronic analyses
Beff	ENC neutronic analyses	Should be conservatively low and bound plant and cycle specific ENC neutronic analyses in transients where moderator or doppler feedback is controlling. Should be high when trip and reactor scram is controlling.
System Temperature at Hot Zero Power	FSAR or utility documentation of plant operation & control systems	-
Rated Power and Primary & Secondary Temperature at Rated Power	FSAR, Tech. Specs., utility documentation	-

Table 3.4 Plant Condition Parameters (Cont.)

Parameter	Source/ENC Basis	Comment
Pressurizer Level	Utility documentation of plant operation and control systems	Dead band and measurement uncertainties are considered in establishing a conservatively high pressurizer level in the loss of load transient.
S.G. Level	Utility documentation of plant operation and control systems	-
Feedwater Flow Rate	FSAR/utility measurements	-

4.0 EVALUATION OF SPECIFIC TRANSIENTS

This section considers application of PTSPWR2 to specific transients. These transients are categorized in accordance with the NRC's Standard Review Plan.

4.1 EVENTS WHERE A MALFUNCTION CAUSES AN INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

All five of the Standard Review Plan transients in this category have been analyzed with PTSPWR2. These transients are:

Reduced feedwater temperature

Increased feedwater flow

Increased steam flow (excess load)

Secondary safety or relief valve opening (small steamline break)

Main steamline break

The unplanned increase in heat removal by the secondary system results in a decrease in primary coolant temperature. At the end of cycle when there is a large negative moderator temperature feedback, the decrease in primary temperature can result in significant increases in core power and decreased margin to DNB. At either beginning of cycle or end of cycle, decreases in the primary coolant temperature can also result in power increases due to the automatic reactor power control system when it is in operation, since this system works to maintain primary coolant temperature at a level consistent with the power demand.

All the transients identified above except the main steamline break are considered AOOs. ENC generally analyzes only the most limiting of these

AOOs on the basis of previous analyses documented in the plant FSAR.

The main steamline break transient is a postulated accident. The system response in this accident is independent of fuel type; however, this accident has been analyzed in all prior licensing applications using PTSPWR2 since the occurrence of DNB depends upon fuel design and since core loading can influence power peaking about the withdrawn control rod in this accident.

4.1.1 Feedwater Malfunctions Leading to Increased Heat Removal and Excess Load Transients

These events are usually initiated from normal full power operating conditions. A bounding end-of-cycle moderator temperature coefficient is used to ensure that power increases are not underestimated for either manual or automatic reactor power control. In these transients, primary system pressure decreases in response to the primary side cooldown. The adverse effects of decreasing pressure and increasing power on DNBR generally will outweigh the favorable effect of reduced coolant temperature. Hence, thermal margin will generally decrease in these events. In contrast to steamline or feed line breaks, these transients are viewed as relatively minor perturbations of the secondary system. This, in combination with transport delays, tends to make these transients slow in comparison to other AOOs and postulated accidents (PAs).

Because coolant temperature decreases in these feedwater malfunctions or excess load transients, reactor trip often occurs on a high flux (or variable high power trip in some CE plants); however, primary pressure decreases potentially may activate the overtemperature ΔT trip in

Westinghouse plants or the TM/LP trip in C.E. plants. Occasionally a steady state is reached which lies within the RPS envelope and no trip occurs.

Conservatism in the ENC analyses is assessed on the basis that PTSPWR2 is conservative relative to the FSAR with respect to primary side cooldown. Where this is not possible, due to lack of FSAR information, the PTSPWR2 power increases are compared to FSAR power increases. To further ensure maximum DNBR impact, pressurizer pressure control via the pressurizer heater is not considered.

4.1.2 Secondary Safety or Relief Valve Opening (Small Steamline Break)

This event is considered to occur from hot standby conditions when the steam generator inventory is largest. This maximizes the reactor cooldown. The objective is generally to show that reactor cooldown with a bounding end-of-cycle moderator temperature feedback is insufficient to overcome the minimum allowed shutdown margin and thus the reactor does not return to power. Actually, some return to power would be an acceptable outcome provided the DNB SAFDL is met. In applications to date, reactor coolant pumps are assumed to continue to operate. Conservatism of the PTSPWR2 analysis is verified by comparing the ENC calculated cooldown against FSAR results for comparable conditions.

4.1.3 Large Steamline Breaks

PTSPWR2 has been applied to the analysis of large steamline breaks for PWRs. The basis for conservatism has been to compare PTSPWR2 results for core power, pressure and temperature to FSAR results for comparable conditions. Certain aspects of this accident such as steam void

formation in the primary coolant system and asymmetrical core temperatures are not found in other transients to which PTSPWR2 is applied, and are not handled explicitly in PTSPWR2. In view of these factors, it is important to establish acceptability of the FSAR analysis prior to making comparisons of PTSPWR2 results for system response to the FSAR results. The justification for comparison of PTSPWR2 results for ENC fuel to the FSAR results for comparable conditions is that plant system response to a steamline break is insensitive to fuel design since ENC reload fuel always closely approximates other fuel in the core in terms of geometry and overall hydraulic losses. Valid comparisons require no change in EOC moderator temperature coefficient or shutdown margin. Based upon the above considerations, the applicability of PTSPWR2 to steamline breaks must be assessed on a plant specific basis. Further, ENC analyses with PTSPWR2 generally consider only the most limiting steamline breaks in terms of maximum return to power. The accident may be initiated from hot zero power or full power operation. Steamline breaks outside containment normally are not considered.

4.2 MALFUNCTIONS THAT RESULT IN A DECREASE OF HEAT REMOVAL BY THE SECONDARY SYSTEM

Events in this category that lead to a decrease of secondary heat removal include:

Loss of external load

Turbine trip

Loss of condenser vacuum

Steam pressure regulator failure

Loss of AC power

Loss of normal feedwater flow

Feedwater pipe break.

These events can result in a rapid increase in primary coolant temperature and primary pressurization due to thermal expansion of the primary coolant. Thermal margin degradation seldom occurs in spite of the increased coolant temperature since the pressurization tends to dominate. Since moderator temperature coefficients are never large positive numbers, reactivity insertion and power increases in these transients are generally small and the thermal margin degradation is usually less than in rod withdrawal transients where increases in reactor power produce the primary coolant temperature increase. Thermal margin degradation in these transients is further mitigated by the increase in primary system pressure. While PTSPWR2 has been applied to a determination of thermal margin degradation for these events, it is usually applied to reconfirm that primary pressures do not exceed vessel pressure integrity criteria and that pressurizer packing does not occur. Since fuel design is not a significant factor in plant system response, application of PTSPWR2 to this category of events is primarily of interest when the plant system configuration or operating conditions are changed and not when the only change is the reload fuel design.

4.2.1 Loss of External Load, Turbine Trip, Loss of Condenser Vacuum and Steam Pressure Regulator Failure Events

PTSPWR2, while considered applicable to all of these events, is only applied to the most pressure limiting of these events, which in applications to date has been termed the loss of load event. For calculation of the peak vessel pressurization using PTSPWR2, the exact initiating event is not considered important. What is important in PTSPWR2 applications is that

steam flow is brought rapidly to zero (within 1 to 2 seconds) and that the primary side pressure response without direct reactor scram from the initiating event (worst single failure) is calculated. Reactor scram occurs on the pressurizer high pressure trip function.

To ensure conservatism, ENC assumes 102% power operation of the plant, a bounding beginning-of-cycle moderator temperature coefficient, a small doppler feedback coefficient, and other applicable conservatisms detailed in Tables 3.3 and 3.4. Pressurizer pressure control (spray) and relief valves are assumed not to function. It is very important to avoid overestimating the pressurizer safety valve capacity. To bound operation of the plant, the initial pressurizer level is set on the high side to provide a conservative pressurization transient. ENC uses an isentropic pressurizer model (Reference 1) in PTSPWR2 to conservatively bound pressurizer pressure response to the primary coolant expansion that characterizes this event.

Adequacy of PTSPWR2 results is assessed against FSAR predictions of primary system heatup and rate of pressure response. In all PTSPWR2 analyses of Westinghouse plants to date, ENC has found that the pressurizer safety relief capacity far exceeds what is needed to accommodate core heatup with a 100% loss of load. Hence, vessel pressure criteria have never been threatened. A similar conclusion has been reached on C.E. plants analyzed to date; however, the safety relief valve capacity margin is not as large as in Westinghouse plants.

4.2.2 Loss of AC Power

The loss of AC power event has more impact on DNB margin than other events in this category due to the pump coastdown and the corresponding

loss in primary coolant flow. However, because of event timing and the probable reactor trip on low coolant flow versus a later trip on high pressurizer pressure, this event is not as pressure limiting as the loss of load transient. Thus, application of PTSPWR2 to this transient has been related to its thermal margin degradation aspects and, because of its similarity to the pump coastdown event, it is treated as being in the loss of flow event category (see Section 4.3).

4.2.3 Loss of Normal Feedwater

PTSPWR2 has been applied to examine or confirm the short term (1-2 min.) consequences of the loss of normal feedwater flow event such as the degradation in steam generator inventory and the timing of reactor trip on low steam generator water level. The long term simulation of loss of normal feedwater transients is performed with ENC's SLOTRAX computer code.

4.2.4 Feedwater Pipe Break

PTSPWR2 has not been applied to PWR feedwater pipe break accidents since DNB margin is not an issue for this event and since the other consequences are independent of the fuel or other plant changes which would be considered by ENC. If PTSPWR2 were to be applied to this accident, it would require as a minimum validation that primary-to-secondary heat transfer is conservatively low for the low steam generator inventories that characterize this event.

4.3 LOSS OF FLOW EVENTS

The events in this category include:

Simultaneous coastdown of all primary coolant pumps

Loss of AC power

Primary pump rotor seizure

Primary pump shaft break.

In all applications of PTSPWR2, FSAR analyses have indicated that the loss of AC power transient is either the same as or is bounded by the pump coastdown transient insofar as thermal margin is concerned. Thus, while applicable to the loss of AC power transient, PTSPWR2 calculations have not distinguished this as a limiting event requiring separate analysis from the flow coastdown transient. Likewise, FSAR analyses have indicated that the seized rotor accident is more limiting to DNB margin than the primary pump shaft break accident, since the initial flow degradation is greater for the seized rotor event. Hence, PTSPWR2 applications are not performed for primary pump shaft break accident.

4.3.1 Flow Coastdown

The flow coastdown event for PWRs is always one of the most limiting AOOs since no credit is taken for direct reactor scram on the initiating malfunction (e.g., low frequency or voltage). Rather, reactor scram initiates upon detection of reduced coolant flow, by which time significant thermal margin degradation has already occurred.

The flow coastdown transient is typically quite rapid with MDNBR occurring within 2-4 seconds after initiation of the transient. Because of the rapid nature of this transient, steam generator heat transfer characteristics and secondary side occurrences such as turbine isolation have no impact on thermal margin. Changes in core power and primary system pressure, until the time of MDNBR, are also typically small compared to other

AOOs. Prior to reactor trip, the major system parameter changes, aside from flow reduction, are an increase in core exit temperatures due to longer residence of the coolant in the core and a decrease in steam generator (primary) exit temperatures due to a longer residence of coolant in the steam generator. The analysis is performed with bounding beginning of cycle moderator temperature coefficients to maximize any power increase due to increased core coolant temperature. The flow coastdown is developed to be consistent with the FSAR or applicable plant coastdown data. Where plant coastdown data is available, a pump model (Reference 2) is available for use after proper benchmarking.

4.3.2 Seized Pump Rotor

The seized pump rotor accident is in many respects similar to the pump coastdown transient except that reactor coolant flow decreases more rapidly. In the time between the reactor trip initiation and reactor scram, the coolant flow decrease is greater than in the pump coastdown transient and therefore thermal margins are reduced to a greater extent. Simultaneous pump coastdown of the remaining reactor coolant pumps, assumed coincident with the loss of AC power, has been considered in some recent applications of PTSPWR2. This further contributes to thermal margin degradation. As with the flow coastdown transient, conservatism is introduced by using a bounding BOC moderator coefficient, bounding scram characteristics, along with conservative FSAR trip delays and trip uncertainties as detailed in Table 3.3.

In this accident, like the steamline break accident, penetration of DNB limits is often calculated to occur especially for two-loop Westinghouse plants. In this case, the extent of fuel damage in the core is

evaluated by summing the probability of DNBR occurrence for all rods in the core that fall below the 95/95 DNBR safety limit value. This evaluation is based on typical core power distributions which are conservatively modified to place the peak rod at maximum allowable power peaking limits.

4.4 REACTIVITY AND POWER DISTRIBUTION ANOMOLIES

Events in this category include:

An uncontrolled control rod bank withdrawal from a subcritical or low power condition

An uncontrolled control rod bank withdrawal at power

Dropped rod (control rod misoperation)

Inactive loop startup

Boron dilution transients

Fuel assembly misloading

Rod ejection analysis

The spectrum of transients in this category is large and entails a variety of plant responses. Generally, these transients all involve unplanned or inadvertent increases in reactor power or reactor power peaking. For the rod withdrawal, boron dilution and rod ejection events, some degree of reactor coolant temperature increase is also involved. Any of these events have the potential for approaching applicable thermal margin limits and therefore cannot be categorically written off. In the case of AOO events in this category, the reactor and DNBR limits on the fuel are usually protected by either a thermal margin trip (i.e., the overtemperature ΔT trip in Westinghouse reactors, the TM/LP trip in Combustion Engineering reactors) or

a high nuclear flux reactor trip. For the dropped rod event, reactor trip is usually not calculated to occur.

4.4.1 Uncontrolled Control Rod Bank Withdrawal from a Subcritical or Low Power Condition

This event is characterized by a very rapid increase in reactor power. Shutdown mechanisms for this event are doppler feedback and reactor scram. The reactor scram may be due to low range high nuclear flux or high flux rate. However, based on FSAR analyses, credit for the latter is usually not taken. This transient is sufficiently rapid that total energy deposition in the fuel is limited so that fuel temperatures and clad surface heat fluxes do not normally exceed those experienced in normal 100% power operation. For this reason, moderator feedback also has only a limited role. Conservatism in PTSPWR2 analyses is assured by using conservatively low doppler feedback and conservative scram characteristics. Conservative BOC moderator feedback characteristics are also used.

4.4.2 Uncontrolled Control Rod Bank Withdrawal at Power

FSAR analyses generally consider a spectrum of reactivity insertion rates from very slow rates to very fast rates that clearly bound the situation where maximum worth rods are withdrawn at the maximum withdrawal rate. At very high reactivity insertion rates, primary coolant temperature increases are limited and reactor scram occurs because of high nuclear flux. Secondary side characteristics are not significant because of the short times involved. Because of the lag between power increases and heat flux increases, fast rod withdrawal transients that trigger the high nuclear flux trip tend to be less limiting with respect to DNBR and more limiting with respect to peak LHGR as reactivity insertion rate increases.

For slow rod withdrawal transients, primary coolant temperature heatup occurs and, consequently, the operation of the secondary side becomes a major factor. In slow rod withdrawal transients, high coolant temperatures sufficient to threaten thermal margin are possible with powers well below the high flux trip setpoint. The principal reactor protection is from the thermal margin trip system that senses core coolant temperature, primary system pressure and reactor power, and will trip the reactor if core coolant temperature at the measured power is high enough to threaten DNB limits. In Westinghouse plants, the measured power is based on the hot-to-cold leg temperature difference. In some C.E. plants, this is further augmented by a nuclear flux power measurement. The thermal margin trip system provides comprehensive protection of DNB for slow rod withdrawal transients. However, as reactivity insertion rates increase from slow towards fast, transport and instrumentation delays can reduce the effectiveness of the thermal margin trip system even though some lead-lag compensation may be incorporated in the trip functions. The most limiting reactivity insertion rates thus are generally found in the intermediate range usually at the point where reactor trip would occur simultaneously on thermal margin and high nuclear flux. It is also possible in this region of intermediate insertion rates that a trip will occur due to high pressure in the pressurizer. The PTSPWR2 transient calculation can be viewed as an analysis which is performed to verify that the heatup rates assumed in developing the thermal margin trip are adequate.

To ensure conservatism in the PTSPWR2 analyses, it is important to avoid overestimating primary to secondary heat transfer since

this would lead to larger hot-to-cold leg temperature differences, high sensed ΔT power and premature reactor trip. Adequacy of PTSPWR2 analyses in this regard can be checked internally and can be judged against FSAR results for plant system response. It is also important not to underestimate transport and instrumentation delays. The analyses are typically performed for both BOC and EOC kinetic parameters to determine which results in more limiting DNB performance.

While thermal margin trip functions provide compensation for changes in reactor pressure, it is nevertheless important not to overestimate primary system pressure increases both with respect to giving conservative MDNBR results when the reactor trips on high nuclear flux and to avoid premature trip on pressurizer pressure. Hence, PTSPWR2 analyses are made with the pressurizer pressure control (spray) functional.

4.4.3 Control Rod Misoperation

PTSPWR2 has been applied to two transients of this type, namely single rod withdrawal events and rod drop events. The single rod withdrawal event is much like the control rod bank withdrawal transients just discussed except that the maximum rate of reactivity insertion and the net reactivity inserted is less. DNB evaluations require consideration of the magnitude of peaking increases in the vicinity of the withdrawn rod. FSARs have not identified this as a limiting transient.

On the other hand, the single rod drop event, particularly with the reactor under automatic control, is very often a DNBR limiting event for PWRs, since the reactor can return to full power operation with a

perturbed power distribution. The MDNBR is often evaluated with steady-state neutronic and thermal-hydraulic analyses without the use of PTSPWR2. When PTSPWR2 is used for this event, it is used to show the system response characteristics both with and without turbine runback, and with and without automatic reactor control, and to examine the potential for the control system to produce power overshoots through inadequate control system damping. Errors in one core power detection due to the perturbed (skewed) power distribution is also considered in determining system response. In applications of PTSPWR2, care is taken to properly represent the plant's automatic control system. The plant response predicted by PTSPWR2 is confirmed against FSAR predictions. Even without automatic control, moderator feedback at the end of the cycle can be sufficiently large to return the reactor to near full power. A conservatively bounding large EOC negative moderator temperature coefficient is used to not underestimate the maximum return to power.

4.4.4 Inactive Loop Startup

PTSPWR2 has been used to assess the general nature of this transient. Conservatism was introduced by neglecting the delay for cold idle loop water to reach the reactor vessel. Plant power restrictions for idle loop operation were such that this transient was clearly non-limiting. If PTSPWR2 is applied to the inactive loop startup event under more limiting conditions, further consideration will be required to handle the effects of asymmetric coolant temperatures in the core.

4.4.5 Boron Dilution

The thermal margin impact of boron dilution transients is identified in FSARs as being bounded by rod withdrawal events and hence has

not been analyzed with PTSPWR2. The code is considered applicable to these transients following considerations detailed under rod withdrawal events to ensure conservative results.

4.4.6 Fuel Assembly Misloading

PTSPWR2 is not applied to the fuel assembly misloading event, which is analyzed with steady-state neutronic and thermal-hydraulic codes.

4.4.7 Rod Ejection Transients

ENC's rod ejection methodology[9] and generic analysis which has been approved by the NRC is used to analyze this event for margin to peak enthalpy deposition criteria. This methodology does not involve PTSPWR2. If necessary, PTSPWR2 can be used to evaluate primary system pressurization for this event. This would require inputting the core energy deposition from the R-Z neutronic analysis of this event.

4.5 TRANSIENTS INVOLVING INCREASED PRIMARY COOLANT INVENTORY

Transients in this category involve coolant inventory addition by either the high pressure safety injection system or by the chemical volume control system. These events are identified in the FSAR as non-limiting and therefore they have not been analyzed using PTSPWR2.

4.6 TRANSIENTS INVOLVING DECREASED PRIMARY COOLANT INVENTORY

Transients in this category include:

Inadvertent opening of a pressurizer relief or safety valve

The steam generator tube rupture event

LOCA.

4.6.1 Inadvertent Opening of a Pressurizer Relief or Safety Valve

This transient has been evaluated with PTSPWR2 to confirm the adequacy of the thermal margin trip system since it is a very rapid RCS depressurization event. In PTSPWR2, pressurizer surge line pressure losses are neglected which is conservative. Because of scram delays, DNBRs for this event are sensitive to depressurization rate. Pressurizer inventory is maximized to give the highest rate of depressurization. This transient is insensitive to reactor kinetics parameters and transport delays in the primary loop.

4.6.2 Steam Generator Tube Rupture Event

Adverse consequences of this event do not generally threaten DNB because of reactor scram on either the low pressurizer pressure trip or the thermal margin trip. PTSPWR2 is presently being benchmarked to the Prairie Island Unit 1 steam tube rupture event. This actual transient did not lead to primary side voiding. PTSPWR2 is not considered applicable to this transient when significant primary loop voiding occurs.

4.6.3 LOCA

PTSPWR2 is not involved in PWR LOCA analyses.

4.7 RADIOLOGICAL RELEASES FROM A SUBSYSTEM OR COMPONENT

PTSPWR2 is not used to evaluate subsystem or component radiological releases.

4.8 ANTICIPATED TRANSIENTS WITHOUT SCRAM

PTSPWR2 is not used for ATWS analyses.

5.0 REFERENCES

(1) XN-74-5(P), Revision 2, "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS-PWR)," Exxon Nuclear Company, Inc., Richland, WA, October 1983.

(2) XN-74-5(P), Revision 2, Supplement 1, "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS-PWR): PTSPWR2 Modifications for St. Lucie Unit 1," Exxon Nuclear Company, Inc., Richland, WA, October 1983.

(3) NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," United States Nuclear Regulatory Commission, Washington, DC, July 1981.

(4) XN-NF-82-91(P), "H.B. Robinson Loss of Feedwater Transient at 1955 MWt: Model Description and Results," Exxon Nuclear Company, Richland, WA, November 1982; and XN-NF-83-49(P), "H.B. Robinson Unit 2 Loss of Normal Feedwater Analysis with Reduced, Asymmetric Auxiliary Feedwater Delivery," Exxon Nuclear Company, Richland, WA, July 1983.

(5) XN-75-21(P), Revision 2, "XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady-State and Transient Core Operation," Exxon Nuclear Company, Richland, WA, September 1982.

(6) XN-NF-82-21(A), Revision 1, "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company, Richland, WA, September 1983.

(7) XN-NF-621(A), Revision 1, "Exxon Nuclear DNB Correlation for PWR Fuel Designs," Exxon Nuclear Company, Richland, WA, September 1983.

(8) XN-NF-719(P), "Assessment of Potential Radiological Consequences for High Exposure Fuel," Exxon Nuclear Company, Richland, WA, August 1983.

(9) XN-NF-78-44(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," Exxon Nuclear Company, Richland, WA, October 1983.

(10) XN-NF-82-20(P), Rev. 1, August 1982; Supplement 1, March 1982; and Supplement 2, March 1982, "Exxon Nuclear Company Evaluation Model EXEM/PWR ECCS Model Updates," Exxon Nuclear Company, Richland, WA.

APPENDIX A

PTSPWR2/XCOBRA-IIIC INTERFACE

There are two possible configurations for the interface between PTSPWR2 and XCOBRA-IIIC. The first is the direct approach of simply applying the core boundary conditions from the PTSPWR2 calculation to the XCOBRA-IIIC code and then letting XCOBRA-IIIC compute MDNBR. Conservatism in the MDNBR evaluation is introduced by modifying these boundary conditions to allow for the uncertainties on reactor power, primary coolant temperature, primary system pressure, and core flow

The simultaneous imposition of uncertainties for all four of the core parameters follows conventional industry practice and is considered to provide substantial conservative margin in the evaluation of MDNBRs for transient events.

The second configuration for the interface between PTSPWR2 and XCOBRA-IIIC is slightly more conservative but is often used to save calculation time. In this configuration, the hot-channel model capability of PTSPWR2 is used to conservatively evaluate MDNBR. In this case, XCOBRA-IIIC is used to model the DNBR at the time the transient is initiated. As in the first approach, the uncertainties in the core parameters, which are not accounted for in the reactor trip function for the event, are accounted for in the XCOBRA-IIIC calculation as discussed above. For the purposes of the hot channel MDNBR calculation, a heat flux peaking factor, an enthalpy rise peaking factor, a flow factor, and an MDNBR penalty factor are provided as input to the PTSPWR2

code. The heat flux peaking factor corresponds to the maximum allowable pin radial peaking factor including allowances for nuclear measurement uncertainties and engineering uncertainties (these are also included in all XCOBRA-IIIC analyses). The enthalpy rise factor is based on the DNB limiting subchannel enthalpy rise calculated by XCOBRA-IIIC versus the core average enthalpy rise. The flow factor is based on the limiting subchannel mass-velocity calculated by XCOBRA-IIIC versus the core average mass-velocity. The direct DNB penalty factor allows for slight differences in axial nodalization between PTSPWR2 and XCOBRA-IIIC and the lack of a Tong factor evaluation in PTSPWR2. It is important that the DNBR calculated by PTSPWR2 at the initiation of the transient bound the corresponding DNBR calculated by XCOBRA-IIIC to the extent that it will account for the adverse effects of changes in core flow distribution that will occur during transient events. The transient is analyzed using the PTSPWR2 code and a value for DNBR is established for each point in the transient. The lowest value of DNBR calculated for the transient is the MDNBR for the transient. In the case of AOOs, this MDNBR (or that from the direct approach), along with added rod bow penalties, must show margin to the 95/95 safety limit of the applicable CHF correlation (W-3 or XNB).

The extra conservatism of this second approach results from the necessity of choosing the hot channel flow factor in a way that bounds potential effects of changes in core flow distribution that may occur during the transient. Such flow factors are calculated in a straightforward manner with steady-state XCOBRA-IIIC calculations that consider perturbations in power, pressure, inlet temperature and flow that are large enough in magnitude

code. The heat flux peaking factor corresponds to the maximum allowable pin radial peaking factor including allowances for nuclear measurement uncertainties and engineering uncertainties (these are also included in all XCOBRA-IIIC analyses). The enthalpy rise factor is based on the DNB limiting subchannel enthalpy rise calculated by XCOBRA-IIIC versus the core average enthalpy rise. The flow factor is based on the limiting subchannel mass-velocity calculated by XCOBRA-IIIC versus the core average mass-velocity. The direct DNB penalty factor allows for slight differences in axial nodalization between PTSPWR2 and XCOBRA-IIIC and the lack of a Tong factor evaluation in PTSPWR2. It is important that the DNBR calculated by PTSPWR2 at the initiation of the transient bound the corresponding DNBR calculated by XCOBRA-IIIC to the extent that it will account for the adverse effects of changes in core flow distribution that will occur during transient events. The transient is analyzed using the PTSPWR2 code and a value for DNBR is established for each point in the transient. The lowest value of DNBR calculated for the transient is the MDNBR for the transient. In the case of AOOs, this MDNBR (or that from the direct approach), along with added rod bow penalties, must show margin to the 95/95 safety limit of the applicable CHF correlation (W-3 or XNB).

The extra conservatism of this second approach results from the necessity of choosing the hot channel flow factor in a way that bounds potential effects of changes in core flow distribution that may occur during the transient. Such flow factors are calculated in a straightforward manner with steady-state XCOBRA-IIIC calculations that consider perturbations in power, pressure, inlet temperature and flow that are large enough in magnitude

to give DNBRs below the MDNBRs calculated for the transients of interest. Several combinations of power, pressure, flow and inlet temperature are examined to give the lowest most conservative flow factor.

One final aspect of the XCOBRA-IIIC/PTSPWR2 interface to be noted is that the PTSPWR2 fuel thermal response model is used to establish the delay between heat flux and power during transient events. Thus, in transient calculations performed with XCOBRA-IIIC (the direct approach), transient core heat flux is input from PTSPWR2 into XCOBRA-IIIC rather than transient power. The delay between heat flux and power is one aspect of plant system transient response that can be influenced by fuel design. In rapid transients, the increased clad thickness of many ENC designs versus NSSS fuel designs will lead to an improvement in MDNBR.

XN-74-5(NP)(A)
Revision 2
Supplement 3

Issue Date 10/8/86

DESCRIPTION OF THE EXXON NUCLEAR PLANT TRANSIENT SIMULATION MODEL FOR PRESSURIZED WATER REACTORS (PTS-PWR): RESPONSES TO NRC QUESTIONS

Prepared by: D. M. Turner
D. M. Turner
Model Development

Concur: G. C. Cooke 9-27-84
G. C. Cooke, Manager
Model Development

Concur: W. V. Kayser 9/28/84
W. V. Kayser, Manager
PWR Safety Analysis

Concur: R. A. Copeland for 28 Sep 84
J. C. Chandler, Lead Engineer
Reload Fuel Licensing

Approve: R. B. Stout 28 Sep 84
R. B. Stout, Manager
Licensing & Safety Engineering

Approve: R. B. Stout for
G. A. Sofer, Manager
Fuel Engineering & Technical Services

gf

EXXON NUCLEAR COMPANY, INC.

NUCLEAR REGULATORY COMMISSION REPORT 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 U.S. Nuclear Regulatory Commission as part of a technical contribution to facilitate safety analyses by licensees of the U.S. Nuclear Regulatory Commission 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 U.S. Nuclear Regulatory Commission in its review of this report, and under the terms of the respective agreements, by licensees or applicants before the U.S. Nuclear Regulatory Commission which are customers of Exxon Nuclear in their demonstration of compliance with the U.S. Nuclear Regulatory Commission's regulations.

Exxon Nuclear's warranties and representations concerning the subject matter of this document are those set forth in the agreement between Exxon Nuclear and the customer to which this document is issued. Accordingly, except as otherwise expressly provided in such agreement, neither Exxon Nuclear nor any person acting on its behalf:

A. Makes any warranty, or representation 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

Item	Page
1.0 INTRODUCTION	1
QUESTION 440.1(a)	2
QUESTION 440.1(b)	3
QUESTION 440.1(c)	5
QUESTION 440.1(d)	6
QUESTION 440.1(e)	7
QUESTION 440.1(f)	8
QUESTION 440.2(a)	9
QUESTION 440.2(b)	10
QUESTION 440.2(c)	12
QUESTION 440.2(d)	13
QUESTION 440.2(e)	14
QUESTION 440.3(a)	15
QUESTION 440.3(b)	16
QUESTION 440.3(c)	18
QUESTION 440.3(d)	19
QUESTION 440.3(e)	24
QUESTION 440.3(f)	25
QUESTION 440.4(a)	26
QUESTION 440.4(b)	27
QUESTION 440.4(c)	28
QUESTION 440.4(d)	29
QUESTION 440.5(a)	30
QUESTION 440.5(b)	31
QUESTION 440.5(c)	32
QUESTION 440.5(d)	33

TABLE OF CONTENTS (Cont.)

Item	Page
QUESTION 440.5(e)	34
QUESTION 440.5(f)	35
QUESTION 440.5(g)	36
QUESTION 440.5(h)	37
QUESTION 440.5(i)	38
QUESTION 440.5(j)	41
QUESTION 440.5(k)	42
QUESTION 440.5(l)	43
QUESTION 440.5(m)	45
QUESTION 440.5(n)	46
QUESTION 440.5(o)	47
QUESTION 440.5(p)	49
QUESTION 440.6(a)	50
QUESTION 440.6(b)	51
QUESTION 440.6(c)	53
QUESTION 440.7(a)	54
QUESTION 440.7(b)	55
QUESTION 440.7(c)	57
QUESTION 440.7(d)	58
QUESTION 440.8	59
QUESTION 440.9(a)	60
QUESTION 440.9(b)	62
QUESTION 440.9(c)	63
QUESTION 440.9(d)	64
QUESTION 440.9(e)	65
QUESTION 440.10	66
QUESTION 440.11	67
QUESTION 440.12	68

TABLE OF CONTENTS (Cont.)

Item	Page
QUESTION 440.13	69
QUESTION 440.14	70
QUESTION 440.15	71
QUESTION 440.16	72
QUESTION 440.17	79
QUESTION 440.18	80
QUESTION 440.19(a)	81
QUESTION 440.19(b)	87
QUESTION 440.19(c)	89
QUESTION 440.19(d)	90
QUESTION 440.20(a) & (b)	91
QUESTION 440.20(c)	92
QUESTION 440.20(d)	93
QUESTION 440.20(e)	94
QUESTION 440.21	95
QUESTION 440.22	96
QUESTION 440.23	98
QUESTION 440.24	99
QUESTION 440.25	100
QUESTION 440.26(a)	101
QUESTION 440.26(b)	102
QUESTION 440.26(c)	103
QUESTION 440.26(d)	104
QUESTION 440.26(e)	105
QUESTION 440.26(f)	106
QUESTION 440.26(g)	107

TABLE OF CONTENTS (Cont.)

Item	Page
QUESTION 440.26(h)	108
QUESTION 440.26(i)	109
QUESTION 440.27	110

1.0 INTRODUCTION AND SUMMARY

This report provides Exxon Nuclear Company's (ENC's) responses to the NRC questions on our topical report XN-74-5(P), Rev. 2. This ENC document provides a description of the models contained in PTSPWR2, a digital computer program used to simulate the behavior of multi-loop pressurizer water reactors subjected to abnormal operating conditions. The code is used to predict plant transient responses to the transients considered in the Safety Analyses section of the Final Safety Analysis Reports for PWRs. Supplement 3, along with other supplements (1, 4 and 5) and the basic document [XN-74-5(P), Rev. 2] provide the basis for the application of the PTSPWR2 code to PWR plant transient analyses performed by ENC.

Question 440.1(a)

Describe in detail how metal heat transfer is modeled in PTSPWR2.

Response

Question 440.1(b)

Describe in detail the boron transport model used in PTSPWR2.

Response

The boron transport model in PTSPWR2 consists of a safety injection time delay model, reactor vessel and steam generator plena mixing models, and cold and hot leg pipe transport models. This transport model is integral within an overall injection model that consists of a safety injection (SI) pump and a SI system volume that is modeled to discharge directly into the reactor vessel lower plenum.

Subsequent to the boron transport delay time, the borated SI flow is effectively injected into the lower plenum and transported through the primary system using the same models described for the enthalpy transport in

Subsection 2.3.4 of XN-NF-74-5(P), Rev. 2.

Question 440.1(c)

Justify modeling the pressurizer surgeline directly from the reactor vessel upper plenum rather than to the appropriate hot leg. Discuss the conditions and events for which this model is not valid.

Response

The PTSPWR2 methodology requires that state variables be evaluated in the control volumes (nodes). Therefore, the temperature of the surge flow into the pressurizer is taken to be the upper plenum temperature. The only other choice would be the steam generator inlet plenum temperature, which is downstream of the surge line connection. Because of the proximity of the surge line connection to the reactor vessel hot leg nozzle, connection of the surge line to the upper plenum node is the appropriate choice.

The validity of this model comes into question when hot leg temperatures differ appreciably between loops. In particular, the model comes into question for (1) rapid assymetric outsurge events when fluid from the pressurizer could have a further impact on hot leg temperature differences, and (2) slow assymetric insurge events that are sufficiently long that mixing of the insurge flow with fluid already in the pressurizer could impact system pressure.

Question 440.1(d)

It is our understanding that a flow resistance model has been incorporated into PTSPWR2. Provide details of this model.

Response

Calculation of primary system pressure used in PTSPWR2 has been modified to add pressurizer surge line pressure drop to the pressure at the bottom of the pressurizer. The surge line pressure drop is calculated by

$$\Delta P_{s\ell} = -K_{surge} W^2_{uppr} V_{up}$$

for flow into the pressurizer and

$$\Delta P_{s\ell} = K_{surge} W^2_{uppr} V_{\ell pr}$$

for flow out of the pressurizer.

Where,

K_{surge} = equivalent loss coefficient for surge line, psi-sec^2/lbm-ft^3
$V_{\ell pr}$ = specific volume of water in pressurizer, ft^3/lbm
V_{up} = specific volume of water in RV upper plenum, ft^3/lbm
W_{uppr} = surge line flow rate, lbm/sec

A gravity head is also calculated using the height of water in the pressurizer as a basis. The pressurizer pressure, gravity head, surge line, pressure drop, and core to surge line pressure drop are used to calculate the core pressure used for the PTSPWR2 DNB calculations.

Question 440.1(e)

Provide further details of the steam generator model used in PTSPWR2. Provide validation of its acceptability.

Response

The steam generator model has two major aspects. The first is the primary-to-secondary heat transfer that is done on a nodal basis along the U-tubes. The second aspect is the secondary liquid level, recirculation flow rate, and secondary fluid subcooling at the bottom of the U-tubes.

Validation of this model is being established by the benchmarking of PTSPWR2 against LOFT transient tests, the Prairie Island steam generator tube rupture event, and a RELAP5 calculation for the H.B. Robinson 3-pump coastdown transient.

Question 440.1(f)

Provide the details of the numerical scheme and the time step selection algorithms used in PTSPWR2.

Response

The time step size is determined by user input, the value selected being determined by past experience with the transient being simulated, and code calculation stability in general.

Basically, the PTSPWR2 code uses explicit methods for solving the appropriate differential equations.

Question 440.2(a)

Demonstrate that the prompt term in the power equation is consistent with the prompt term in the precursor equation as calculated by the point kinetics model (e.g., Eqn. 2.1).

Response

One way of demonstrating the consistency of the two prompt terms of the reactor kinetics equations and the consistency of the two equations themselves is to perform a steady-state prediction with them.

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

$$\frac{dC_i}{dt} = \frac{\beta_i P}{\ell} - \lambda_i C_i \quad ; \; i = 1, 6 \qquad (2)$$

where δk = reactivity in dollars ($).

At steady state, all the dCi/dt terms are zero. Therefore, summing the precursor equation (2) over all 6 groups gives,

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

(3)

Since at steady state, $\dot{P}=0$ and $\delta k=0$, eq. (1) gives

(4)

Thus, the same relationship is derived from both equations at steady state, indicating the consistency of the set of equations.

Question 440.2(b)

Present the derivation for the prompt jump approximation equation (Eqn. 2.3).

Response

Consider a step input of reactivity, δk. For reactivity insertions much less than $1.00, the transient terms in the power solution die out rapidly and the stable reactor period "ω" quickly attained. Since,

(1)

where

The dominant term in the solution of this equation is:

(2)

Since, $\dot{P} = \omega P$, substitution into

(3)

yields

(4)

For small reactivities (and consequently, a small ω), Eq. (4) is approximately

A term-by-term comparison with Eq. (3) indicates that the first and second terms in the right-hand side of Eq. (3) are opposite in sign and nearly equal in magnitude for slow changes. The presence of the δk term in Eq. (4) is offset by the ω in denominator of the last term.

If Eq. (3) is replaced by,

(5)

the Prompt-Jump approximation is defined. The $\dot{P}$ in Eq. (3) is dropped by assuming that

which is true when the right-hand sides of Eq. (3) are opposite in sign and nearly equal, as shown previously.

Rearrangement of Eq. (6) yields Eq. (2-3)

(7)

(1) Glasstone, S. and Sesonske, A., Nuclear Reactor Engineering, pp. 230-251, Van Nostrand Reinhold Co., Inc., New York City, NY, 1967.

(2) Keepin, G.R., Physics of Nuclear Kinetics, pp. 187-194, Addison-Wesley Co., Inc., Palo Alto, CA, 1965.

Question 440.2(c)

The equation for Rf3 appears incorrect. Please correct and confirm that this error has not been implemented in the code.

Response

The correct expression as implemented in PTSPWR2 for R_{f3} is:

Question 440.2(d)

Demonstrate the acceptability of modeling the core as a one axial node approximation. In particular, address core heat transfer and reactivity feedback.

Response

Question 440.2(e)

Justify the quasistatic approach for modeling the coolant enthalpy distribution.

Response

The total enthalpy change over the core length is calculated for each time step in a transient manner. The core average moderator temperature corresponds to the core average moderator enthalpy, the calculations being detailed in Section 2.2.2 of XN-74-5(P). This calculation is considered valid subject to the restrictions noted in the response to Question 440.2(d).

Question 440.3(a)

Define "f", as used in the pump equation, and provide justification for its functional form.

Response

The "f" function used for calculation of the pump head consists of a table containing pump head as a function of normalized flow rate. The tabular values are changed to represent the performance of the pumps used for each plant that is analyzed. In future applications, a new pump performance model using homologous curves described in XN-74-5(P), Rev. 2, Supplement 4 will be used in lieu of the "f" function.

Question 440.3(b)

Provide the derivation for the gravity heads used in Equations (2.26e), (2.27) and (2.28).

Response

The gravity heads in Eqs. (2.26e), (2.27) and (2.28) have been updated (see response to Question 440.19(a)). The updated primary loop flow model which replaces that described in Section 2.3.1 of XN-NF-74-5(P), Rev. 2, provides an accurate calculation of gravity heads for natural circulation flow when PTSPWR2 is used on a stand-alone basis.

The reactor vessel potential head defined by Eq. 2.26a in XN-NF-74-5(P), Rev. 2, has been redefined by the following expression:

where

The primary loop potential head defined by Eqn. 2.26e has been modified as follows:

where,

The potential (thermal) head terms appearing in Eqns. 2.27, 2.28, 2.29 and 2.30 of XN-NF-74-5(P), Revision 2, have been modified to reflect the modifications described above.

Question 440.3(c)

Describe how the friction loss coefficients are determined.

Response

Question 440.3(d)

Define the variable name FDELAY used in Equation 2.32 and provide details of the equation used for the cold leg as well.

Response

In order to illustrate the pipe transport model characteristics, boundary conditions were forced on the calculating subroutine (RTD) and Figures 440.3(d)-1 to 440.3(d)-3 were generated. For Figure 440.3(d)-1, full positive flow from the left end to the right end of the pipe was assumed. The transport delay at full flow was 5.5 seconds. A positive ramp temperature input was forced on the left end of the pipe for times between 5 and 15 seconds, as indicated by the curve with the square symbols. From 15 to 25 seconds, a negative ramp was input. The response at the right end of the pipe to the forced temperature boundary condition is shown by the curve labeled "Right". With no flow variation, the input and output curves are shifted by the 5.5 second transport time. The magnitude of the peak temperature was not altered.

The effect of lowering the flow through the pipe is shown in Figure 440.3(d)-2. At 15 seconds the flow was halved and consequently the effective transport delay time was increased from 5.5 to 11.0 seconds. The slope change in the response curve (Right) at 15 seconds reflects the step decrease in flow.

The effect of flow reversal is presented in Figure 440.3(d)-3. For the first 15 seconds full positive flow existed. At 15 seconds, a step decrease to full negative flow was then forced on the system. For the first 15 seconds, the boundary condition temperature at the left end of the pipe was used to calculate the temperature at the right end (Right). For times greater than 15 seconds, the right end ramp decrease was forced and the left end (Left) became responsive. This figure illustrates that the wave fronts propagating down the pipe reverse direction and exit when the flow reverses with no magnitude alteration.

Question 440.3(e)

Derive Equation 2.37.

Response

Define the following parameters by:

CF_{lp} = Volume of reactor vessel lower plenum

Cp_{lp} = Specific heat of fluid in lower plenum

M_{lp} = Mass of fluid in lower plenum

T_{lpi} = Avg. temperature of Loop #1 and Loop #2 cold leg discharges (Eq. (2-36))

$T_{ci(1)}$ = Temperature of fluid in lower plenum $\equiv$ core inlet temperature $\equiv T_{ci}(1)$

$VF_{sc}(T_{ci(1)})$ = Specific volume of lower plenum fluid

W_{lpcr} = Total flow rate from lower plenum to core

$\dot{T}_{cri}$ = Rate of change of lower plenum fluid temperature $\equiv \dot{T}_{ci}(1)$

If incompressible flow is assumed for each time step, an energy balance for the lower plenum yields

	(1)

This equation assumes:

	(2)

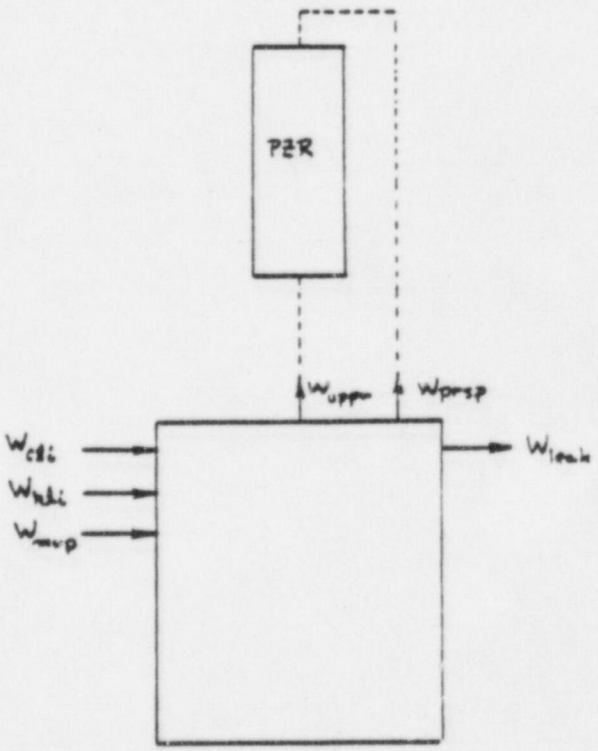
Eq. (1) becomes upon rearrangement

Question 440.3(f)

Illustrate the control volume for which the mass balance Equation 2.38 is written and identify the different flow terms.

Response

The control volume consists of the entire primary loop volume exclusive of the pressurizer. The cold leg safety injection (W_{CLi}), hot leg injection (W_{hLi}), and makeup water (W_{mup}) flows enter through the control volume boundary. The pressurizer spray (W_{prsp}), primary system leakage (W_{leak}) and pressurizer surge (W_{uppr}) flows exit through the boundary. A negative surge flow results when the pressurizer is emptying.



Question 440.4(a)

Define h_f as used in Equation 2.5 for pressurizer outsurge.

Response

Eq. 2.5 of the initial draft document does not appear in the formally issued version of XN-NF-74-5(P), Rev. 2.

Question 440.4(b)

With regards to pressurizer insurge analysis:

(i) Explain why, although the analytical equations give an indeterminate answer, the numerical algorithm described gives a unique solution.

(ii) Justify the neglection of bulk condensation interregion mass transfer in the mass equations, Eqs. 2.1 and 2.2.

Response

Both parts of this question deal with the pressurizer model description in the initial draft of XN-74-5(P), Rev. 2.

More specifically,

(i) The pressurizer model as presented in the formally issued version of XN-74-5(P), Rev. 2, does not have to deal with this indeterminate answer problem.

(ii) Bulk condensation and interregion mass transfer are not neglected in the latest PTSPWR2 model presented in the formally issued version of the report.

Question 440.4(c)

Discuss the pressurizer outsurge model in more detail; in particular, the calculation of the flashing rate.

Response:

A detailed discussion of the Pressurizer Model Theory is presented on Pages 26-29 of the formally issued version of XN-74-5(P), Rev. 2.

Question 440.4(d)

Provide the equation for a steam filled (no liquid) empty pressurizer model.

Response

Question 440.5(a)

Document the criterion used to differentiate between the steam, boiling, and non-boiling regions of the steam generator model.

Response

Question 440.5(b)

Clarify and summarize the primary side equations used for the steam generator model in the case of reverse flow (Eqs. (3.3), (3.5), (3.7), (3.9), (3.11)...).

Response:

Basically the same form for the energy equations are used for forward and reverse flow. For forward flow,

$$M_i \overline{C}p_i \frac{dT_i}{dt} = W_i \overline{Cp}_i (T_{i-1} - T_i) - Q_i \qquad (1)$$

where

$\overline{C}p_i$ = Avg. specific heat for node "i" between points "i-1" and "i"

M_i = Mass in node "i"

Q_i = Heat removal rate from node "i" to S.G. secondary side

T_i = Temperature at interface between nodes "i" and "i-1"
= Average temperature in node "i"

T_{i-1} = Temperature in upstream node "i-1"

W_i = Flow rate in S.G. tubes, positive value

For reverse flow, the T_{i-1} term becomes T_i, T_i becomes T_{i+1}, and W_i becomes a negative value in the right-hand side of Eq. (1). T_{lp_i}, T_{lp1}, T_{lp2}, and T_{lp3} are all calculated in this manner and T_{lpo} becomes forced by the S.G. outlet plenum temperature during reverse flow.

Question 440.5(c)

Derive the expression for $(\Delta TMI)_{i1}$ used in Equation 3.2.

Response

The expression ΔTMI is an approximation for the log mean temperature difference (LMTD) where

$$LMTD \equiv \frac{\Delta T_2 - \Delta T_1}{\ln\left(\Delta T_2 / \Delta T_1\right)}$$

As a numerical example of this approximation, the initial steam generator inlet temperature (T_{HL}), outlet temperature (T_{CL}), and secondary side temperature (T_{sec}) for the H.B. Robinson Unit 2 PTSPWR2 benchmark (XN-74-5(P), Rev. 2, Supp. 5) are used to calculate the LMTD and the approximated value ΔTMI.

Question 440.5(d)

Justify the quasistatic approach which neglects the steam generator tube heat capacity.

Response

Question 440.5(e)

Document the physical interpretation of Equation 3.6 for the U-bend region. Discuss the modeling when one or more regions disappear or reappears.

Response

The steam generator model basically consists of one (if no steam region exists) or two (if a steam region exists) moving interfaces. Eq. (3.6) basically states that if no steam region exists, the location of point #2 and point #3 coincide, the location being effectively at the top of the U-bend.

If the steam region disappears, points #2 and #3 coincide at the top of the U-bend. If the boiling region disappears (subcooled liquid only), no steam is generated and steam flows in the control volumes above the active region may actually reverse because of condensation.

Question 440.5(f)

Is U_{steam} always set to zero?

Response

| |

Question 440.5(g)

Define the temperatures used in Equation 3.13.

Response

T_{1pi1}	=	Average temperature in steam generator tubes between tube inlet and point #1
T_{1p12}	=	Avg. temperature between points #1 and #2
T_{1p23}	=	Avg. temperature between points #2 and #3
T_{1p34}	=	Avg. temperature between points #3 and #4
T_{1p4o}	=	Avg. temperature between point #4 and tube outlet
T_{sg1pa}	=	Volume weighted average temperature of water in steam generator tubes (primary side)

Question 440.5(h)

On the secondary side of the steam generator model:

(i) Explain why the specific volume in Equation 3.14 is evaluated at subcooled conditions when the nonboiling region is assumed to be at saturation; and

(ii) Justify the modulus signs in Equation 3.14 for the nonboiling length.

Response

i)

ii) The saturated fluid enthalpy, HF_{sg1}, is a function of only one state variable, P_{sg1} in this case. Therefore

$$\frac{d\,HF_{sg1}}{dt} = \frac{dHF}{dp}\,\frac{d\,p_{sg1}}{dt} = \frac{dHF}{dp}\,\dot{p}_{sg1}$$

The energy balance represented by Eq. (3.14) is straightforward and no sign ambiguity should exist, as should be the case for the definitions presented for V_{1sNG}, CF_{1sNB}, and Q_{sgNB1}. However, Eq. (3.15) contains a typographical error. The absolute value signs occurring in the numerator should be changed to brackets, resulting in no modulus sign ambiguity. PTSPWR2 coding is correct.

Question 440.5(i)

Present the derivation for Eqn. (3.19) for the steam region, justify the expression used for V_{1SST}, and define CF_{sg1T}.

Response

Question 440.5( )

Define A_{CS} used in Eqn. (3.18) and A_{mt} used in Eqn. (3.21).

Response

1) $A_{CS} = A_{CSSG1}$ = Secondary coolant cross-sectional area in steam generator 1.

2) Amt = Mean heat transfer area through steam generator tube wall
= π * Avg. Diameter * Height of region.

Question 440.5(k)

Justify the numerical value used for the fouling factor f.

Response

Question 440.5(1)

Derive the gravity head terms in Eqns. (3.32) and (3.36) and state which of the two equations is programmed in the code.

Response

The following discussion refers to Figure 440.5(1)-1, which is a generic schematic of the steam generator secondary side modelling.

Question 440.5(m)

How is W_{uptd1} determined?

Response

Question 440.5(n)

Justify the use of Eqn. (3.52) for the top of the downcomer when the level changes.

Question 440.5(o)

Derive Eqn. (3.55) which determines the pressure at the bottom of the downcomer.

Response

Consider the following sketch:

A momentum balance for the downcomer region yields:

A similar balance for the nonboiling region yields:

A mass flow balance between the two regions gives

where,

Differentiation of the two right-hand side groupings in Eq. (3) gives

Rearranging Eq. (5) and solving for P_{bdl} yields

Question 440.5(p)

Provide the equation for FDELAY.

Response

The transport delay in the downcomer is calculated by the same methods as described in the answer to Question 440.3(d).

Question 440.6(a)

Equations (4.3) and (4.4) are believed to contain typographical errors. Please confirm and state that these errors have not been incorporated into the program.

Response

Eq. (4.3) should read

This typographical error was not implemented in the PTSPWR2 code. There does not appear to be a typographical error in Eq. (4.4) if the acceleration head is correctly defined (see response to Question 440.6(b)).

Question 440.6(b)

Justify the modulus signs in Eqn. (4.6) for the recoverable acceleration head and the flow acceleration term in Eqn. (4.21).

Response

1) Eq. (4.6) should read

Therefore, the code calculates an acceleration pressure drop whenever the downstream node area A_k is less than the upstream node area A_j.

(2) The modulus φ definition is given incorrectly in Eq. 4.21 of XN-74-5(P), Rev. 2. The actual definition used in the PTSPWR2 code is

As steam enters the steam line from the dome, an acceleration occurs causing a resultant pressure decrease.

For negative flows, the negative modulus multiplied by the negative flow in Eq. (7) results in a calculated $P_{do1} > P_{sl_{11}}$ as expected since the equivalent steam dome flow area is

sign. Since $AA_{dos\,l_1} > 0.0$, Eq. (1)

greater than the steam line area. Therefore, the modulus sign φ is correct.

Question 440.6(c)

Is the steam line model limited to saturation conditions only?

Response:

Question 440.7(a)

Equation (5.1) is believed to contain a typographical error. Please confirm and state that this error has not been incorporated into the program.

Response:

Eq. (5.1) should read

$$\left| \qquad \right| \qquad (5\text{-}1)$$

This typographical error has not been implemented in the code.

Question 440.7(b)

Compare the valve flow model against the Murdock and Bauman correlation for single phase steam flow and against the Moody correlation for two phase mixture flow.

Response

For the purposes of the comparison, the pressurizer safety relief and steam generator safety relief valves for H.B. Robinson Unit 2 were evaluated. The comparisons were performed at the valve setpoints, and mass fluxes were calculated using the PTSPWR2 models, the Murdock-Bauman correlation, and the Moody correlation, respectively. Results are shown in the following table.

Valve(s)	Pressure (psia)	$G_{calc.}$ (lb/sec-ft^2)	G_{M-B} (lb/sec-ft^2)	G_{Moody} (lb/sec-ft^2)
Pressurizer safety (3 valves, total)	2500		6258	5890
Steam Generator Safety A	1100		2371	2293
Steam Generator Safety B	1125		2429	2348
Steam Generator Safety C	1140		2464	2381
Steam Generator Safety D	1155		2498	2414

The Murdock-Bauman correlation used to determine G_{M-B} was

$$G_{M-B} = 45.25 \left(\frac{\overline{P}_1}{v_1} \right)^{.5}$$

which is Equation 8 in

Murdock, J.W. and Bauman, J.M., "The Critical Flow Function for Superheated Steam," ASME TRANS., Vol. 86, Sept. 1964, pp. 507-515.

The mass flux calculated using the expression cited above is proportional to the square root of the ratio of the inlet pressure $\overline{P}_1$ and the inlet steam specific volume v_1, the same dependence that appears in Eq. 5.2 of XN-NF-74-5(P), Rev. 2. Therefore, the calculated flows will match up well with the Murdock-Bauman predictions throughout the pressure ranges of interest as illustrated in the table above.

Question 440.7(c)

Justify the valve flow modifier, Eqn. (5.3).

Response

The valve flow modifier presented in Eq. (5.3) of XN-74-5(P), Rev. 2 is used in modeling primary and secondary relief valve flows.

In general, the effective valve flow area does not linearly vary with position, as shown in Figure 3 of Reference 1, the normalized area having a higher value than the normalized position. Equation (5.3) of XN-74-5(P), Rev. 2 conservatively overpredicts both the normalized flow area shown in Figure 3 of Reference 1 and the normalized position value.

(1) Singh, Autar, "A Model for Predicting the Performance of Spring Loaded Safety Valves," (EPRI/PWR Safety and Relief Valve Testing Program), ASME 82-WA/NE-12.

Question 440.7(d)

Describe the model which converts rod speed into reactivity insertion/deletion.

Response

The rod speed is integrated over each time step to obtain rod position versus time.

Question 440.8

Equation 6.8 is believed to contain a typographical error. Please confirm and state that this error has not been incorporated into the program.

Response:

The third term in the expression should be

$$[\ EXP\ ((18.177 - 0.004129\ P_{cr}) * X_{hc(i)})\ ]$$

This error was not incorporated in the PTSPWR2 code.

Question 440.9(a)

Provide references for the thermodynamic and transport water properties used and compare against the ASME steam table data (or an equivalent reference).

Response

The steam and water properties used in PTSPWR2 are in tabular form and were obtained directly from the 1967 ASME steam tables. Therefore, a one-to-one correspondence will result from a comparison with steam table data.

The following tables in PTSPWR2 contain saturation properties as a function of pressure:

[redacted]

The values used in these tables were either obtained from the 1967 ASME Steam Tables or calculated from values presented in that reference.

The following tables in PTSPWR2 contain subcooled water properties as a function of either temperature or enthalpy:

Table	Description	Temperature or Enthalpy Range

As an example, the subcooled water specific volume table (VFSC) is compared to the 1967 ASME Steam Table values:

T (°F)	f (VFSC) (ft^3/lbm)	f (1967 ASME) (ft^3/lbm)	P (psia)
50		.01590	2200
100		.01602	2200
150		.01623	2200
200		.01652	2200
250		.01687	2200
300		.01730	2200
350		.01781	2200
400		.01842	2200
450		.01916	2200
500		.02010	2200
550		.02135	2200
600		.02320	2200
650		.02655	2300
700		.03628	3100

Question 440.9(b)

Provide references for the default pump curve.

Response

The default head table formerly used was for H.B. Robinson Unit 2. The data used to generate that table may be obtained from the FSAR.

Question 440.9(c)

Submit a copy of Reference 3 and compare the decay heat model described therein with the 1971 ANS standards.

Response

A copy of XN-75-43(P) has been submitted. A detailed comparison of decay heat predictions made with the model to the ANS Standard model is presented in Table 2.2 of:

NUREG-075/056, WREM: Water Reactor Evaluation Model, Revision 1, May 1975, pp. 2-4 to 2-9.

Question 440.9(d)

Provide a reference for the default delayed neutron parameters used and compare against the RELAP4-EM set.

Response

The neutronics parameters used in PTSPWR2 are those used in the WREM-based ENC Evaluation Model. The default values for the delayed neutron group parameters are presented in:

"Core Physics Methods and Data Used as Input to LOCA Analysis," XN-75-43(A), Exxon Nuclear Company, Richland, WA 99352, August 1975.

Question 440.9(e)

Describe in detail the initialization procedure used in the steam line and the pressurizer.

Response

Question 440.10

It is our understanding that the boron transport model uses the equations of the thermal model, Equations (2.31) - (2.37). What variable is used in place of temperature?

Response

Boron concentration in ppm.

Question 440.11

It is our understanding that boron injection is assumed to occur in the lower plenum of the reactor vessel. Justify the conservatism of this model and address the reasons for neglecting injection through the legs.

Response

Boron injection into the cold legs versus the lower plenum is not neglected. It is accounted for by including a conservative transport delay time in the input overall boron injection delay time. Further details are given in response to Question 440.1(b).

Question 440.12

It is not clear if wall heat is modeled in the pressurizer. Document if wall heat is calculated and justify its use or neglect.

Response

Question 440.13

Provide validation of PTSPWR2 with appropriate integral experiment and operating reactor data.

Response

The requested validation is being provided in the PTSPWR2 benchmark report along with a comparison of PTSPWR2 and RELAP5 results for the H.B. Robinson three-pump coastdown transient.

Question 440.14

Provide justification for the simplified steam generator model used in PTSPWR2. Provide details and justification of the initialization procedures.

Response

From an empirical standpoint, the simplified steam generator model is justified on the basis of good benchmarking against integral test and plant operating data and against a RELAP5 calculation.

Question 440.15

The neutronics equation is missing the source term. How are events analyzed from subcritical conditions?

Response

Question 440.16

Justify the acceptability of a one node core model, which implies that the reactivity weighting factor (power^2) is not important.

Response

Acceptability of the one node core model is demonstrated by some example cases. The first three cases assume a simplified axial profile and calculate the effective moderator (Example 1) and doppler (Examples 2 and 3) temperature changes in the core with and without flux squared weighting. The fourth case (Example 4) considers a rod withdrawal transient from subcritical conditions in which a very rapid power increase occurs. The last case (Example 5) considers an end-of-cycle excess load transient. This is a cooldown event where moderator and doppler feedback effects are of comparable magnitude.

Example 1

Question 440.17

Justify not modeling the core bypass and the downcomer bypass regions in PTSPWR2. Also justify modeling the upper plenum and upper head as one node.

Response

(1) Core and downcomer bypass regions can be important in severe depressurization events since the temperatures of these regions will impact void formation in the primary system. For example, downcomer bypass to the upper head will control upper head temperatures and hence upper head void formation.

(2)

Question 440.18

The Staff has received a draft copy of a benchmark calculation of PTSPWR2 with a Prairie Island steam generator tube rupture event. Due to insufficient plant data for benchmark purpose, excessive engineering manipulation of input parameters were required. Perform the same event analysis using the licensing assumptions and demonstrate the acceptability of the code. Overlay the results with the benchmark analysis and continue the analysis sufficiently through the period of upper-head voiding to demonstrate code acceptability.

Response

Since the submittal of the draft copy of the Prairie Island benchmark results, the PTSPWR2 analysis was extended to 400 seconds, including roughly 160 seconds of post-scram predictions at the NRC's request. The results of this extended analysis are presented in the benchmark documentation.

The objectives of this benchmark were to provide information on primary side temperature responses, pressurizer modeling, and system control models pertaining to a realistic transient for a full-sized plant. As noted in the NRC's request to extend the benchmark time, the comparison is insufficient during the load reduction phase of the transient prior to scram to provide the necessary information about the models. However, during and after reactor scram, sufficient plant data was generated to provide a good benchmark basis to demonstrate the model responses without engineering manipulation of input parameters.

Question 440.19(a)

Provide details of the pump model and describe how the pump energy is included in the hydraulics.

Response:

(1) A pump coastdown model for PTSPWR2 was described in XN-74-5(P), Rev. 2, Supplement 4. Further detailed user information is provided here.

The following definitions are used to define the torque tables:

- N = pump speed, rpm
- N_D = design pump speed, rpm
- Q = pump volumetric flow rate, gpm
- Q_D = design pump volumetric flow rate, gpm
- α = normalized pump speed (N/N_D), dimensionless = N_{pump1}, N_{pump2}
- β = normalized pump torque (τ/τ_D), dimensionless
- ν = normalized volumetric flow rate (Q/Q_D), dimensionless
- τ = pump torque, lb_f-ft
- τ_D = design pump torque, lb_f-ft

The homologous torque curve tables used for positive pump speeds and positive loop flow rates are PBANTB and PBVNTB. Table PBANTB, for which β/α^2 is input as a function of ν/α, is used when $0.0 \leq \nu/\alpha \leq 1.0$. Table PBVNTB, for which β/ν^2 is input as a function of α/ν, is used when $1.0 < \nu/\alpha$.

Tables PBADTB and PBVDTB are used when negative loop flow conditions exist. Table PBADTB, for which β/α^2 is input as a function of ν/α, is used when $-1.0 \leq \nu/\alpha < 0.0$. Table PBVDTB, for which β/ν^2 is input as a function of α/ν, is used when $\nu/\alpha < -1.0$.

The pump speed during the transient is determined from the following expression:

where,

The pump torque T is defined by:

$$T = \left(\frac{\beta}{\alpha^2}\right)\alpha^2 \, T_D \, \frac{\rho_{cl}}{\rho_D}$$

when the PBANTB or PBADTB table is applicable and

$$T = \left(\frac{\beta}{\nu^2}\right)\nu^2 \, T_D \, \frac{\rho_{cl}}{\rho_D}$$

when the PBVNTB or PBVDTB table is applicable.

where:

ρ_{cl} = density of fluid in cold leg, lbm/ft^3

ρ_D = design pump fluid density, lbm/ft^3

During a pump coastdown, $T_{motor} = 0.0$. The pump friction/windage torque is defined by

where,

-

The pump heads are now also determined from input homologous curves. The following definitions are used to define the head tables:

h = normalized pump head (H/H_D), dimensionless

H = pump head, lb_f-ft/lbm (ft)

H_D = design head, lb_f-ft/lbm (ft)

The homologous head curve tables used for positive pump speeds and positive loop flow rates are PHANTB and PHVNTB. Table PHANTB, for which h/α^2 is input as a function of ν/α, is used when $0.0 \leq \nu/\alpha \leq 1.0$. Table PHVNTB, for which h/ν^2 is input as a function of α/ν, is used when $1.0 < \nu/\alpha$.

Tables PHADTB and PHVDTB are used when negative loop flow conditions exist. Table PHADTB, for which h/α^2 is input as a function of ν/α, is used when $-1.0 \leq \nu/\alpha < 0.0$. Table PBVDTB, for which h/ν^2 is input as a function of α/ν, is used when $\nu/\alpha < -1.0$.

The pump pressure rise or drop is calculated from

$$P_{pump} = (h/\alpha^2)\alpha^2 \rho_{ef} H_D \ (1/144)$$

when the PHANTB or PHADTB tables are applicable, and

$$P_{pump} = (h/\ ^2)\nu^2 \rho_{ef} H_D \ (1/144)$$

when the PHVNTB or PHVDTB tables are used.

The above pump pressure expressions replace Eqn. 2.26d of XN-NF-74-5(P), Rev. 2, and are now used to calculate the pump heads cited in Eqs. 2.27, 2.28, 2.29, and 2.30. The discussion of pump characteristics presented in Section 8.1.2 is no longer applicable to the PTSPWR2 code.

(2) Pump heat is now included in the PTSPWR2 analyses.

Additional comment:

The pump model is a principal constituent of the Primary Loop Flow model addressed in Section 2.3.1 of XN-NF-74-5(P) Revision 2. The pump coastdown model of XN-74-5(P) Rev. 2, Supplement 4, that has been generically incorporated into PTSPWR2 as described above is thus an important update to the Primary Loop Flow model of the base document. A second update of the Primary Loop Flow model is the modification of thermal heads discussed in response to Question 440.3(b)

In order to implement the Primary Loop Flow model updates, the definitions of some input parameters had to be modified and some new input parameters created.

The following system geometry dimensions have been redefined:

Parameter	Description	Units

Question 440.19(b):

Provide details of the models for the motor torque, friction torque, and windage torque.

Response:

Motor torque is adjusted at steady state to balance the other components.

Thus in terms of XN-74-5(P) Rev. 2 Supplement 1 notation, the steady state model becomes $\frac{d\omega}{dt} = 0.$

When the pump is tripped, $\tau_{motor} = 0$ and the equation in Subsection 2.1 of XN-NF-74-5(P), Rev. 2, Supp. 1 becomes

Thus, in the end, the St. Lucie analysis model was

and

Question 440.19(c):

List the parameters adjusted in the modeling of a pump coastdown and the data used to adjust the above parameters. Justify the use of SEMISCALE pump data for CE pumps.

Response:

The only parameter adjusted to match the pump coastdown curve was the effective moment of inertia. It was adjusted to match the coastdown tests performed at St. Lucie Unit 1. The adjustment was performed for the first 5 seconds (crucial to the loss-of-coolant flow event) and found to provide a reasonably accurate model of the coastdown over the full 200 seconds of a normal coastdown.

The only pump data used for the analysis were the torque and head curves for the St. Lucie Unit 1 pumps.

Question 440.19(d):

Describe and justify which torque is adjusted to stop shaft rotation.

Response:

No adjustments were made to the torque to stop shaft rotation. No evidence in the St. Lucie coastdown data ENC has indicates the need for either a dramatic increase in friction at low rpm or a correction for the inapplicability of the affinity relationships.

Question 440.20(a) and (b):

Derive Equation (3.5) and show how it is consistent with Equation (2.26e) in the main report. Define the variable $V_{sg,j}$.

Response:

Equation 3.5 is just the equivalent mass per unit area (pressure/g_c) differential from the exit of the reactor vessel to the inlet of the vessel. The steam generator is divided into an ascending side,

and a descending side

The average specific volume for the steam generator is denoted by $V_{sg,j}$ in the j^{th} loop.

The above formulation was only used in the ENC analysis of St. Lucie Cycle 6. A new primary loop flow model described in XN-NF-74-5(P) Rev. 2 Supp. 4 provides an equivalent formulation for these pressure head terms that replace the formulation of Equation 2-26e in the main report.

Question 440.20(c):

Justify the functional form used in Eq. (3.8) and provide a figure showing the elevation differences used.

Response:

The functional form in Eq. (3.8) is the same as that in Equation 2.26d of the main report, except W_D is the hot leg design flow (=cold leg flow times two).

The elevation definitions used in the current generic PTSPWR2 primary loop model are presented in the response to Question 440.19(a).

Question 440.20(d):

Provide a reference for the pump head curve shown in Fig. 3.2.

Response:

The pump head curve in Figure 3.2 was calculated from data obtained from the following:

Question 440.20(e):

Define the variables δ_j and λ_j used in Equation (3.14).

Response:

These terms are defined in Equations 3.24 and 3.25 in the same section of XN-NF-74-5(P), Rev. 2, Supp.1.

Question 440.21:

In Section 4.0, the hot to cold leg delta-T is redefined. Document the new definition.

Response:

It is equal to the identification of a separate hot leg to cold leg delta-T for each cold leg. In a Westinghouse PWR there is a single cold leg for each hot leg.

Question 440.22

Section 5.0 references the use of a modified MacBeth CHF correlation for pressures below 1450 psia. Document and justify the use of this correlation. Provide the thermal-hydraulic conditions for which the correlation is valid, the data base for the modification, and the 95/95 SAFDL for this correlation.

Response

The following form of the Modified MacBeth Correlation[1,2] was used for the St. Lucie steam line break analysis:

[redacted]

where,

[redacted]

This correlation was originally developed by MacBeth[1] using 1000 psia data. The correlation predicted 97% of the results within ±12%. Barnett[2] recommended the inclusion of the factor (h_{fg}/649.4) in the A term of his correlation to allow the applicable pressure ranges of the correlation to be extended from 600 psia to 1450 psia.

(1) R. V. MacBeth, "Burn-out Analysis: Part 5 - Examination of Published World Data for Rod Bundles," AEEW-R 358, 1964.

(2) P. G. Barnett, "A Correlation of Burnout Data for Uniformly Heated Annuli and Its Use for Predicting Burnout in Uniformly Heated Rod Bundles," AEEW-R 463, 1966.

Question 440.23:

Provide references for Equations (7.2) - (7.4) and Fig. 7.3, which describe various trips.

Response:

The source of this material is the FSAR for St. Lucie Unit 1, the stretch power application, and the Technical Specifications for St. Lucie Unit 1.

Question 440.24:

Explain how Eq. (8.2) used in the initialization of the pressurizer is consistent with Eqs. (2.45) - (2.46) of the new pressurizer model described in the main report.

Response

Equation 8.2 is obtained from Equations (2.45) and (2.46) by setting the transient terms in Equations (2.45) and (2.46) to zero, i.e.,

and by adding the two equations. This results in

(1)

(2)

(3)

Equations (1) and (3) yield

(4)

Equations (2) and (4) yield

Question 440.25:

Section 9.0 implies that PTSPWR2 does not conserve energy in the steam dome during depressurization events. Please justify and clarify the acceptability of this model.

Response:

Question 440.26(a):

Equation (5.2) of the main report is not consistent with Equation (9.1) of Supplement 1 for the break flow. Reconcile the differences.

Response:

Question 440.26(b):

What are the units used in Eqn. (9.2) for the break flow?

Response:

W_{DOSL} - lbm/sec

A_{SLB} - ft^2

F_m - lbm/ft^2-sec

Question 440.26(c):

Provide the value of the quality used for the Moody flow factor of Fig. 9.1 and justify the choice.

Response:

Question 440.26(d):

Why is dome pressure and not steam line pressure used to calculate steam line break flow?

Response:

Question 440.26(e):

Equation (9.4) is not the same as Equation (3.30) in the main report. Please reconcile the differences.

Response:

Question 440.26(f):

Provide a figure to illustrate Equation (9.5).

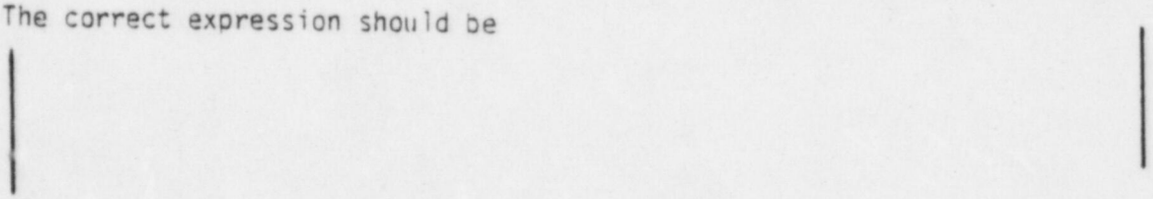
Response:

Question 440.26(g):

Equation (9.8) is believed to contain typographical errors. Please confirm and document that these errors have not been incorporated into the program.

Response:

The correct expression should be



This typographical error was not incorporated in PTSPWR2.

Question 440.26(h):

Explain why the steam generator model of the main report requires the modifications of the supplement to calculate the mass in the dome.

Response:

Question 440.26(i):

Provide the equations for the boron transport model.

Response:

The boron transport model in PTSPWR2 was not changed for the St. Lucie analysis. This model has been outlined in the response to Question 440.1(b).

Question 440.27:

We have reviewed the subject topical report which describes the methodology of implementing PTSPWR2 for licensing calculations. We find that the report contains no specific details regarding the biases in the choice of initial and boundary conditions. The report consists of vague generalizations from which we are unable to conclude that the analyzed events are either conservatively or non-conservatively calculated.

PTSPWR2 is not a "best-estimate" computer program for calculation of many of the Chapter 15 events. Consequently we require demonstration that the means by which the code is applied will bound the consequences of the postulated event. It is our understanding that ENC is in the process of redeveloping its methodology for implementing PTSPWR2 for licensing calculations. We require that the subject topical report be resubmitted to include the most recent methodology of implementation and its associated justification for each event to be analyzed.

Response:

The copy of XN-74-5(P) Revision 2 Supplement 2 submitted in a January 31 submittal was an advance copy that comments and concerns from the NRC could be incorporated into prior to formal issuance. As required in the above question, this supplement has been completely redone. The redone version of the methodology was discussed informally with the NRC during June 1984. This redone version will be formally submitted as a separate report, XN-NF-84-73(P). Supplement 2 of XN-74-5(P), Revision 2, will not be used.

XN-74-5(NP)(A)
Revision 2
Supplement 4
Issue Date 10/8/86

DESCRIPTION OF THE EXXON NUCLEAR PLANT TRANSIENT SIMULATION MODEL FOR PRESSURIZED WATER REACTORS (PTS-PWR): CODE UPDATES

Prepared by: D. M. Turner 5-27-84
D. M. Turner
Model Development

Concur: G. C. Cooke 5-24-84
G. C. Cooke, Manager
Model Development

Concur: W. V. Kayser 8/24/84
W. V. Kayser, Manager
PWR Safety Analysis

Concur: J. C. Chandler 8/24/84
J. C. Chandler
Reload Fuel Licensing

Approve: R. B. Stout
R. B. Stout, Manager
Licensing & Safety Engineering

Approve: G. A. Sofer
G. A. Sofer, Manager
Fuel Engineering & Technical Services

naa

EXXON NUCLEAR COMPANY, INC.

NUCLEAR REGULATORY COMMISSION REPORT 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 U.S. Nuclear Regulatory Commission as part of a technical contribution to facilitate safety analyses by licensees of the U.S. Nuclear Regulatory Commission 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 U.S. Nuclear Regulatory Commission in its review of this report, and under the terms of the respective agreements, by licensees or applicants before the U.S. Nuclear Regulatory Commission which are customers of Exxon Nuclear in their demonstration of compliance with the U.S. Nuclear Regulatory Commission's regulations.

Exxon Nuclear's warranties and representations concerning the subject matter of this document are those set forth in the agreement between Exxon Nuclear and the customer to which this document is issued. Accordingly, except as otherwise expressly provided in such agreement, neither Exxon Nuclear nor any person acting on its behalf:

A. Makes any warranty, or representation 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.

PTSPWR2 UPDATES

This supplement to XN-NF-74-5(P), Rev. 2, documents recent updates to the PTSPWR2 code. In general, the purposes of these updates are to improve code models in order to obtain more realistic predictions and to increase code application flexibility.

The model changes and additions incorporated in these updates occur in the following general areas:

I. Core Model

II. Primary Loop Flow Model

III. Pressurizer Model

IV. Steam Generator Model

V. Trip System Logic Model

VI. Code Output

A section of this report is devoted to the changes made in each of the general areas listed above. A listing of the input modifications for the updated code is presented in Section VI.

I. CORE MODEL

Calculation of primary system pressure used in PTSPWR2 has been modified to add pressurizer surge line pressure drop to the pressure at the bottom of the pressurizer. The surge line pressure drop is calculated by

$$\Delta P_{s\ell} = -K_{surge} W^2_{uppr} V_{up}$$

for flow into the pressurizer and

$$\Delta P_{s\ell} = K_{surge} W^2_{uppr} V_{\ell pr}$$

for flow out of the pressurizer.

Where,

K_{surge} = equivalent loss coefficient for surge line, psi-sec^2/lbm-ft^3
V_{lpr} = specific volume of water in pressurizer, ft^3/lbm
V_{up} = specific volume of water in RV upper plenum, ft^3/lbm
W_{uppr} = surge line flow rate, lbm/sec

Pump heat is now included in the PTSPWR2 analyses.

The code user can now specify whether the W3 or XNB DNB correlation is to be used for the hot channel analysis.

II. PRIMARY LOOP FLOW MODEL

The following constituents of the Primary Loop Flow model described in Section 2.3.1 of XN-NF-74-5(P), Rev. 2 have been modified:

(1) The primary loop and reactor vessel thermal heads

(2) The primary loop pump model

The reactor vessel potential head defined by Eq. 2.26a in XN-NF-74-5(P), Rev. 2, has been redefined by the following expression:

The primary loop potential head defined by Eq. 2.26e has been modified as follows:

where,

The potential (thermal) head terms appearing in Eqs. 2.27, 2.28, 2 29 and 2.30 of XN-NF-74-5 have been modified to reflect the modifications described above. These updates will improve the accuracy of natural circulation flow calculations.

A pump coastdown model for PTSPWR2 was described in XN-74-5(P), Rev. 2, Supplement 1. Further user information is provided here.

The following definitions are used to define the torque tables:

N = pump speed, rpm

N_D = design pump speed, rpm

Q = pump volumetric flow rate, gpm

Q_D = design pump volumetric flow rate, gpm

α = normalized pump speed (N/N_D), dimensionless = N_{pump1}, N_{pump2}

β = normalized pump torque (τ/τ_D), dimensionless

ν = normalized volumetric flow rate (Q/Q_D), dimensionless

τ = pump torque, lb_f-ft

τ_D = design pump torque, lb_f-ft

The homologous torque curve tables used for positive pump speeds and positive loop flow rates are PBANTB and PBVNTB. Table PBANTB, for which β/α^2 is input as a function of ν/α, is used when $0.0 \leq \nu/\alpha \leq 1.0$. Table PBVNTB, for which β/ν^2 is input as a function of α/ν, is used when $1.0 < \nu/\alpha$.

Tables PBADTB and PBVDTB are used when negative loop flow conditions exist. Table PBADTB, for which β/α^2 is input as a function of ν/α, is used when $-1.0 \leq \nu/\alpha < 0.0$. Table PBVDTB, for which β/ν^2 is input as a function of α/ν, is used when $\nu/\alpha < -1.0$.

The pump speed during the transient is determined from the following expression:

where,

The pump torque T is defined by:

wnen the BPANTB or PBADTB tables are applicable, and

when the PBVNTB or PBVDTB tables are applicable.

Where,

The pump friction/windage torque is defined by

where,

The pump heads are now also determined from input homologous curves. The following definitions are used to define the head tables:

h = normalized pump head (H/H_D), dimensionless
H = pump head, lb_f-ft/lbm (ft)
H_D = design pump head, lb_f-ft/lbm (ft)

The homologous head curve tables used for positive pump speeds and loop flow rates are PHANTB and PHVNTB. Table PHANTB, for which h/α^2 is input as a function of ν/α, is used when $0.0 \leq \nu/\alpha \leq 1.0$. Table PHVNTB, for which h/ν^2 is input as a function of α/ν, is used when $1.0 < \nu/\alpha$.

Tables PHADTB and PHVDTB are used when negative loop flow conditions exist. Table PHADTB, for which h/α^2 is input as a function of ν/α, is used when $-1.0 \leq \nu/\alpha < 0.0$. Table PBVDTB, for which h/ν^2 is input as a function of α/ν, is used when $\nu/\alpha < -1.0$.

The pump pressure rise or drop is calculated from

when the PHANTB or PHADTB tables are applicable, and

when the PHVNTB or PHVDTB tables are applicable.

The above pump pressure expressions replace Eq. 2.26d of XN-NF-74-5(P), Rev. 2, and are now used to calculate the pump heads cited in Eqs. 2.27, 2.28, 2.29 and 2.30. The discussion of pump characteristics presented in Section 8.1.2 is no longer applicable to the PTSPWR2 code.

III. PRESSURIZER MODEL

Eq. 2.47 of XN-NF-74-5(P), Rev. 2, has been replaced by

A new pressurizer initialization technique has been incorporated into the PTSPWR2 code.

IV. STEAM GENERATOR MODEL

The heat transfer model for the primary side of the steam generators described in Section 3.1 of XN-NF-74-5(P), Rev. 2 has been improved.

The node specific heats are now calculated from

where,

This expression is now used to calculate the specific heats appearing in Eqs. 3.2, 3.3, 3.4, 3.8, 3.10, 3.12 and 3.13 of XN-NF-74-5(P), Rev. 2.

V. TRIP SYSTEM LOGIC MODEL

Additions to the trip system logic of PTSPWR2 have been made to improve code versatility.

VI. Code Output

The plot tape generated by the code (TAPE 16) now consists of 401 data points for each of the variables described in the output description section (Table 8.6) of XN-74-5(P), Rev. 2.

VII. NEW INPUT

In order to implement the code model changes discussed in previous sections of this presentation, the definitions of some input parameters had to be modified and some new input parameters created. The following system geometry dimensions have been redefined:

XN-74-5(NP)(A)
Revision 2
Supplement 5

Issue Date 10/8/86

DESCRIPTION OF THE EXXON NUCLEAR PLANT TRANSIENT SIMULATION MODEL FOR PRESSURIZED WATER REACTORS (PTS-PWR): CODE BENCHMARKS

Prepared by: D. M. Turner
D. M. Turner
Model Development

Concur: G. C. Cooke 8/31/84
G. C. Cooke, Manager
Model Development

Concur: W. V. Kayser 8/31/84
W. V. Kayser, Manager
PWR Safety Analysis

Concur: J. C. Chandler 8/31/84
J. C. Chandler, Manager
Reload Fuel Licensing

Approve: R. B. Stout 31 Aug 84
R. B. Stout, Manager
Licensing & Safety Engineering

Approve: George A. Sofer 31 AUG 84
G. A. Sofer, Manager
Fuel Engineering & Technical Services

naa EXXON NUCLEAR COMPANY, INC.

NUCLEAR REGULATORY COMMISSION REPORT 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 U.S. Nuclear Regulatory Commission as part of a technical contribution to facilitate safety analyses by licensees of the U.S. Nuclear Regulatory Commission 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 U.S. Nuclear Regulatory Commission in its review of this report, and under the terms of the respective agreements, by licensees or applicants before the U.S. Nuclear Regulatory Commission which are customers of Exxon Nuclear in their demonstration of compliance with the U.S. Nuclear Regulatory Commission's regulations.

Exxon Nuclear's warranties and representations concerning the subject matter of this document are those set forth in the agreement between Exxon Nuclear and the customer to which this document is issued. Accordingly, except as otherwise expressly provided in such agreement, neither Exxon Nuclear nor any person acting on its behalf:

A. Makes any warranty, or representation 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 AND SUMMARY		1
	1.1	LOFT COMPARISON	1
	1.2	PRAIRIE ISLAND COMPARISON	2
	1.3	COMPARISON TO RELAP5	3
2.0	TRANSIENT DESCRIPTIONS		6
	2.1	LOFT TESTS	6
		2.1.1 LOFT Test L6-1	7
		2.1.2 LOFT Test L6-2	7
		2.1.3 LOFT Test L6-3	7
	2.2	PRAIRIE ISLAND GENERATOR TUBE RUPTURE TRANSIENT	8
	2.3	RELAP5 LOSS OF FLOW BENCHMARK TRANSIENT	9
3.0	BENCHMARK CASES - INPUT		23
	3.1	LOFT TESTS	23
		3.1.1 LOFT Test L6-1 - Input	23
		3.1.2 LOFT Test L6-2 - Input	24
		3.1.3 LOFT Test L6-3 - Input	24
	3.2	PRAIRIE ISLAND STEAM GENERATOR TUBE RUPTURE EVENT - INPUT	25
		3.2.1 Leak Rate	26
		3.2.2 Initial Pressurizer Level	27
		3.2.3 Heaters, Charging Pumps, Letdown	27

TABLE OF CONTENTS (Cont.)

Section		Page
3.2.4	Load Demand, Turbine Demand, and Feedwater Enthalpy Ramp Rates	28
3.2.5	Steam Dump Controls	29
3.3	PTSPWR2/RELAP5 CODE COMPARISONS - INPUT	29
4.0	RESULTS - PTSPWR2 PREDICTIONS/DATA COMPARISONS	32
4.1	LOFT TEST DATA COMPARISONS	32
4.1.1	LOFT Test L6-1 Loss-of-Load Transient	32
4.1.2	LOFT Test L6-2 Loss of Primary Flow Transient	38
4.1.3	LOFT Test L6-3 Excessive Load Increase Transient	41
4.2	PRAIRIE ISLAND STEAM GENERATOR TUBE RUPTURE DATA COMPARISONS	82
4.3	PTSPWR2/RELAP5 COMPARISON FOR THE H.B. ROBINSON UNIT 2 THREE-PUMP COASTDOWN SIMULATION	99
5.0	REFERENCES	122

LIST OF FIGURES

Figure		Page
2.3.1	PTSPWR2 Schematic	21
2.3.2	RELAP5 Schematic	22
4.1.1-1	L6-1 Power Level	47
4.1.1-2	L6-1 Hot Leg Flow	48
4.1.1-3	L6-1 Steam Line Flow	49
4.1.1-4	L6-1 Steam Dome Pressure	50
4.1.1-5	L6-1 Steam Generator Level	51
4.1.1-6	L6-1 Feedwater Flow	52
4.1.1-7	L6-1 Steam Generator Downcomer Temperature	53
4.1.1-8	L6-1 Cold Leg Temperature	54
4.1.1-9	L6-1 Hot Leg Temperature	55
4.1.1-10	L6-1 Pressurizer Level	56
4.1.1-11	L6-1 Pressurizer Pressure	57
4.1.2-1	L6-2 Power Level	58
4.1.2-2	L6-2 Normalized Pump Speed	59
4.1.2-3	L6-2 Hot Leg Flow	60
4.1.2-4	L6-2 Steam Line Flow	61
4.1.2-5	L6-2 Steam Dome Pressure	62
4.1.2-6	L6-2 Steam Generator Level	63
4.1.2-7	L6-2 Feedwater Flow	64
4.1.2-8	L6-2 Steam Generator Downcomer Temperature	65

LIST OF FIGURES (Cont.)

Figure		Page
4.1.2-9	L6-2 Hot Leg Temperature	66
4.1.2-10	L6-2 Cold Leg Temperature	67
4.1.2-11	L6-2 Pressurizer Level	68
4.1.2-12	L6-2 Pressurizer Pressure	69
4.1.3-1	L6-3 Power Level	70
4.1.3-2	L6-3 Hot Leg Flow	71
4.1.3-3	L6-3 Steam Line Flow	72
4.1.3-4	L6-3 Feedwater Flow	73
4.1.3-5	L6-3 Steam Generator Level	74
4.1.3-6	L6-3 Steam Generator Downcomer Temperature	75
4.1.3-7	L6-3 Steam Dome Pressure	76
4.1.3-8	L6-3 HPSI Flow	77
4.1.3-9	L6-3 Cold Leg Temperature	78
4.1.3-10	L6-3 Hot Leg Temperature	79
4.1.3-11	L6-3 Pressurizer Level	80
4.1.3-12	L6-3 Pressurizer Pressure	81
4.2-1	Prairie Island Power	88
4.2-2	Prairie Island Loop T_{avg}	89
4.2-3	Prairie Island Hot Leg Temperature	90
4.2-4	Prairie Island Cold Leg Temperature	91
4.2-5	Prairie Island Loop Delta T	92

LIST OF FIGURES (Cont.)

Figure		Page
4.2-6	Prairie Island Pressurizer Level	93
4.2-7	Prairie Island Pressurizer Pressure	94
4.2-8	Prairie Island Steam Generator Level	95
4.2-9	Prairie Island Steam Line Flow	96
4.2-10	Prairie Island Steam Generator Pressure	97
4.2-11	Prairie Island Feedwater Flow	98
4.3-1	PTS/RELAP5 Comparison Core Power	106
4.3-2	PTS/RELAP5 Comparison RCS Pump Speed	107
4.3-3	PTS/RELAP5 Comparison Primary Loop Flow Rate	108
4.3-4	PTS/RELAP5 Comparison Steam Generator Pressure	109
4.3-5	PTS/RELAP5 Comparison Steam Generator Level	110
4.3-6	PTS/RELAP5 Comparison Steam Generator Downcomer Temperature	111
4.3-7	PTS/RELAP5 Comparison Core Outlet Temperature	112
4.3-8	PTS/RELAP5 Comparison Reactor Vessel Outlet Temperature	113
4.3-9	PTS/RELAP5 Comparison Hot Leg Temperature	114
4.3-10	PTS/RELAP5 Comparison Steam Generator Inlet Temperature	115
4.3-11	PTS/RELAP5 Comparison Steam Generator Outlet Temperature	116
4.3-12	PTS/RELAP5 Comparison Cold Leg Temperature	117

LIST OF FIGURES (Cont.)

Figure		Page
4.3-13	PTS/RELAP5 Comparison Reactor Vessel Inlet Temperature	118
4.3-14	PTS/RELAP5 Comparison Core Inlet Temperature	119
4.3-15	PTS/RELAP5 Comparison Pressurizer Level	120
4.3-16	PTS/RELAP5 Comparison Pressurizer Pressure	121

List of Tables

Table		Page
2.1.1	Range of Initial Conditions for LOFT Tests L6-1, L6-2, and L6-3	12
2.1.2	General Subsystem Component Characteristics and Trip Setpoints	13
2.1.3	LOFT L6-1 Event Sequence	14
2.1.4	LOFT L6-2 Event Sequence	15
2.1.5	LOFT L6-3 Event Sequence	16
2.2.1	Initial Conditions for Prairie Island Transient	17
2.2.2	Prairie Island Event Sequence	18
2.3.1	Initial Conditions Summary Comparison Between RELAP5 and PTSPWR2	19
2.3.2	Three Pump Coastdown Event Sequence	20

1.0 INTRODUCTION AND SUMMARY

The Exxon Nuclear Company (ENC) Plant Transient Simulator computer code PTSPWR2 is used by ENC for safety analysis of pressurized water reactors. References 1 and 2 contain descriptions of the code and its application to PWR transients, respectively. As part of the generic review process currently being undertaken by the U.S. Nuclear Regulatory Commission (NRC), benchmark comparisons of calculated primary and secondary system performances against available data were requested by NRC. The benchmark cases selected by ENC consist of three LOFT facility transients, the October 2, 1979 Prairie Island Unit 1 steam generator tube rupture incident, and a comparison to the RELAP5 code prediction of a three-pump coastdown incident for H.B. Robinson Unit 2. This report presents the results of these PTSPWR2 benchmarks.

1.1 LOFT COMPARISON

The following three LOFT Series tests were used as benchmark cases:

LOFT L6-1 - Loss of Load

LOFT L6-2 - Loss of Flow

LOFT L6-3 - Excessive Load Increase

These tests, which represent plant transients that are normally analyzed for licensing submittals, provide meaningful information for plant transient code evaluation. Test L6-1 provides an exercise of the pressurizer in-surge model while L6-3 tests the out-surge model. Both tests also provide exercises for the steam generator heat transfer modelling, L6-1 being characterized by a decrease in primary to secondary heat transfer and L6-3 being characterized by an increase. The primary loop hydrodynamics model is exercised by the L6-2 test transient data. These experiments have been

previously analyzed with codes such as RETRAN and RELAP5.

Specific comparisons of PTSPWR2 to LOFT experimental data show good agreement with respect to the following important features:

- Good tracking of cold and hot leg temperatures in response to steam generator secondary temperature (pressure) change and core power level (scram) change.
- Close matching of flow coastdown response.
- Good tracking of pressurizer level changes induced by changes in primary loop temperatures and good pressure response during pressurizer outsurge. For pressurizer insurge, the PTSPWR2 non-equilibrium model without the interphase heat transfer option being excercised tends to overpredict pressure response.
- PTSPWR2 also provided good predictions of changes in secondary conditions as reflected by steam generator level and pressure.

1.2 PRAIRIE ISLAND COMPARISON

The Prairie Island steam generator tube rupture induced transient was selected on the basis that it was a significant primary system loss-of-inventory and depressurization event that occurred in an actual full-sized nuclear plant. Quality data for a number of primary and secondary system plant parameters were collected for 200 seconds before and after reactor trip. Analysis of this transient exercises the primary side thermal model, the

pressurizer model, the steam generator heat transfer model, the secondary side thermal-hydraulics model, and some plant control models. This event has been previously analyzed with such codes as RETRAN and DYNODE-P.

Specific comparisons of the PTSPWR2 predictions to the Prairie Island data are highlighted with respect to the following important features:

- Good tracking of the primary hot leg temperature and loop ΔT during the load reduction and post-scram periods.
- Good tracking of the pressurizer pressure response.
- Close matching of steam generator parameters throughout the transient.
- Accurate feedwater control response.

The results of the PTSPWR2/Prairie Island benchmark indicate that the PTSPWR2 code can be successfully used to analyze licensing basis events for full sized plants. In general, accurate code predictions of the data both before and after scram resulted from the PTSPWR2 analysis of the Prairie Island steam generator tube rupture induced transient.

1.3 COMPARISON TO RELAP5

The H.B. Robinson three-pump coastdown transient analyzed with RELAP5 and PTSPWR2 exercises all the primary loop thermal-hydraulic models which have a bearing on departure from nucleate boiling (DNB) and transition to natural circulation. The steam generator secondary side thermal-hydraulic models are also tested by this transient. The PTSPWR2 and RELAP5 codes were both initialized at steady-state conditions of full power and nominal flow rates and temperatures. The transient simulation was initiated by tripping all three main reactor coolant system (RCS) pumps. After pump trip, the two

ccdes calculated the plant control system responses while using similar boundary conditions such as feedwater enthalpy (temperature), valve closing times, and setpoints.

Comparisons of important parameters predicted by the two codes were made with special attention being given to those parameters having a direct effect on DNB and loop flow calculations. The PTSPWR2 results for these parameters are in good agreement with values predicted by RELAP5. Specifically, comparisons of the PTSPWR2 predictions to the RELAP5 predictions are highlighted by the following significant features.

- Accurate matching of the initial conditions for important plant parameters.
- Excellent tracking of RELAP5 core power transient by PTSPWR2.
- Close agreement of transient loop flow rates.
- Good tracking of hot and cold leg temperatures and reactor core inlet and outlet temperatures.
- Virtually identical predictions of reactor scram time.

Comparison of the code predictions show good agreement for the parameter values affecting DNB calculation, especially at the time of occurrence of the minimum departure from nucleate boiling ratio (DNBR). Plant parameter values predicted by the codes for times up to 200 seconds after RCS pump trips are in good agreement.

It is concluded from the PTSPWR2/RELAP5 benchmark that the PTSPWR2 code as currently used to perform safety analysis calculations in support of licensing provides parameter values for DNB and primary loop flow that are in good agreement with RELAP5 values. The good overall agreement of all

parameter values especially to the DNB and flow calculation parameters between the codes indicates the suitability of the application of the PTSPWR2 code to licensing basis calculations.

The benchmark cases selected for the PTSPWR2 code qualification effort represent plant transients that are normally analyzed for licensing submittals and provide meaningful information pertinent to transient code evaluation. Specifically, comparisons with the test data cases and the RELAP5 simulation provide information on the adequacy of the primary loop flow dynamics, pressurizer, core thermal, core kinetics, primary loop energy transport, steam generator heat transfer, secondary side thermal-hydraulics, relief and safety valve, and plant controller models used in the PTSPWR2 code.

In general, the PTSPWR2 predictions agreed well with the data and the RELAP5 code predictions, even for a scaled-down LOFT system for which metal stored heat effects are more significant than for a full-sized commercial power plant. The apparent discrepancies between PTSPWR2 predictions and the data bases are explainable and do not appear to reflect on the adequacy of the models incorporated in the PTSPWR2 code. Therefore, since meaningful tests have been adequately simulated by the PTSPWR2 code, strong evidence has been presented for qualification of the code for licensing basis analyses.

2.0 TRANSIENT DESCRIPTIONS

This section of the report consists of three subsections. The first subsection contains descriptions of the three LOFT transients analyzed, the second contains a description of the Prairie Island steam generator tube rupture transient and the third contains a description of the postulated H.B. Robinson Unit 2 three-pump coastdown event.

2.1 LOFT TESTS

A general description of the LOFT facility and tests can be found in Reference 3. Reference 4 contains specific test and instrument information. The data plots for selected parameters were digitized and replotted for comparison with the PTSPWR2 code predictions.

Table 2.1.1 presents the range of initial conditions for the three LOFT tests analyzed. Although the initial values for the presented parameters varied slightly from test to test, they were in general within the test specification limits and within the relatively narrow ranges presented in the table.

Table 2.1.2 presents some of the important subsystem characteristics and setpoints. The LOFT system contains many active subsystems such as the feedwater control system, high pressure injection system (HPIS), pressurizer pressure control system and the main steam flow control valve (MSFCV) system, all of which were active at least once during the three LOFT transients being considered. Since most of these systems are representative of systems associated with PWRs, successful benchmarking of test results verifies the PTSPWR2 modelling of those components.

2.1.1 LOFT Test L6-1

This Loss-of-Load test was initiated by closing the main steam flow control valve (MSFCV) while the power level was steady at roughly 37 MWT (~75% of Rated). The MSFCV valve closed in 11.6 seconds and the reactor scrammed on high primary coolant system (PCS) pressure at 21.8 seconds. The MSFCV, which also serves as a steam generator pressure relief valve, reopened at 22.2 seconds when the pressure exceeded the set-point. Closure occurred at 40.6 seconds. A second opening and third closing of the valve occurred at 91.2 and 104.4 seconds, respectively. A tabular listing of the event sequence is found in Table 2.1.3.

2.1.2 LOFT Test L6-2

This Loss of Primary Flow test was initiated by a trip of the primary coolant pumps while the reactor was operating at 37 MWT. The MSFCV closure and reactor trip, triggered by a low primary flow indication, occurred roughly 2 seconds after pump trip. At 18 seconds the pumps were decoupled from the motor generator sets and coasted down by inertia. Natural circulation conditions were first indicated at 23 seconds and the primary pressure continued to recover. After 100 seconds, the pressure had recovered to values near the initial condition value, and by 200 seconds the pressurizer liquid level was back in its operating band. Table 2.1.5 presents the sequence of events for Test L6-2 for the first 200 seconds after transient initiation.

2.1.3 LOFT Test L6-3

This Load Increase test was initiated at a power level of 37 MWT by ramping the MSFCV open from its steady-state initial value. At 15.6

seconds, a primary loop low pressure signal resulted in a reactor scram, and at 17.8 seconds, the MSFCV began to close. Between 26 and 27 seconds after transient initiation, the high pressure injection system (HPIS) pumps turned on, and at 36.2 seconds the MSFCV finished closing. The HPIS pumps were manually shut off at a point roughly 49 seconds into the transient. The sequence of events for Test L6-3 may be found in Table 2.1.5.

2.2 PRAIRIE ISLAND STEAM GENERATOR TUBE RUPTURE TRANSIENT

The steam generator tube rupture event used for this benchmark occurred at the Prairie Island plant on October 2, 1979[5,6]. The transient was initiated by a small leak in one of the two steam generators, resulting in depressurization of the primary system from 2250 to 1915 psia, at which time the reactor tripped on receiving a low primary pressure signal. Although the steam generator tube rupture was calculated to have occurred roughly 240 seconds before scram initiation[6], the data used as the basis for this benchmark[5] were not available for the first 50 seconds following the rupture. The PTSPWR2 predictions for this transient were initiated (TIME = 0.0 seconds) 240 seconds before scram which was approximately 50 seconds before the first plant data points. Table 2.2.1 presents the initial conditions used for the PTSPWR2 simulation and Table 2.2.2 presents the sequence of events referenced to the PTSPWR2 transient initiation time (240 seconds before scram).

Data from Reference 5 indicates that approximately 160 seconds before reactor trip time, a 15% reduction in turbine flow was initiated. Roughly 15 seconds later a 10% reduction in reactor power level was

initiated. This combination of load and power level reductions resulted in additional plant responses being superimposed upon those resulting from the leak. These responses were primarily the adjustments in primary and secondary system temperatures, secondary presssure and secondary flows required to achieve a new equilibrium configuration at the reduced load condition.

The loss of primary coolant through the tube rupture caused a depressurization from 2250 psia to 1915 psia over a period of roughly four minutes at which time the reactor tripped. Although the exact leak rate as a function of time is unknown, estimates of its value have been made by comparing plant data for pressurizer level against calculations[5,6].

The primary side pressure dropped below 1845 psia approximately 8 seconds after scram, initiating startup of the high pressure injection system (HPIS) pumps.

2.3 RELAP5 LOSS OF FLOW BENCHMARK TRANSIENT

The transient simulated by the PTSPWR2 and RELAP5[7] codes was initiated by a three-pump coastdown of the RCS pumps. A simulation of the H.B. Robinson Unit 2 (Carolina Power and Light) plant was used for this comparison.

Figures 2.3.1 and 2.3.2, respectively, show the PTSPWR2 and RELAP5 nodalization diagrams. Plant initial conditions were set to nominal 100% power values on the primary coolant side. The secondary coolant parameter values were selected to initialize the remaining primary side values to nominal conditions. The initial conditions used for the two codes are summarized in Table 2.3.1.

The close comparison between the initial conditions is indicative of the accuracy of the representation of the plant's characteristics by the two codes.

After initial steady-state conditions were attained, the pump coastdown was initiated at time zero. Table 2.3.2 presents the sequence of events for the transient analyzed. After the pump trips, the pump coastdowns were calculated in both codes by the controlling torque-inertia equations, which use the input pump friction, pump inertia, pump torque curves, and fluid inertia to determine the time-dependent pump speeds.

The loss of reactor core flow resulted in core and plant primary system heatup due to the power/flow imbalance before scram. This primary system heatup forced a pressurizer insurge and resultant primary system pressure increase.

Roughly 1.9 seconds after pump trip, a reactor scram signal resulted from low primary flow conditions (flow less than ~85% of initial

value). The 2.2-second scram delay time used in both codes initiated a reactor scram and subsequent closing of the turbine stop valve and feedwater control valves at approximately 4.1 seconds. After scram, the decrease in hot leg temperature resulting from the rapid decrease in core heat generation produced a significant drop in pressurizer level due to the resultant pressurizer outsurge. Level recovery eventually resulted from hot leg temperature increase produced by the low natural circulation flow rate through the core and cold leg temperature increase resulting from the increased steam generator secondary side temperature following isolation. The heatup was terminated when the power production in the reactor core decreased to a level comparable to the degraded steam generator heat removal capabilities. The simulation was continued for 200 seconds to verify comparability between PTSPWR2 and RELAP5 for natural circulation flow conditions.

Table 2.1.1 Range of Initial Conditions for LOFT Tests L6-1, L6-2, and L6-3

PRIMARY LOOP	
Mass flow	1010 - 1070 lb/sec
Hot leg pressure	2130 - 2170 psia
Cold leg temperature	533 - 539°F
Hot leg temeprature	558 - 569°F
REACTOR VESSEL	
Power level	36 - 38 MWT
PRESSURIZER	
Liquid volume	20 - 25 ft^3
Liquid temperature	645 - 650°F
Pressure	2115 - 2185 psia
Liquid level	3.65 - 4.25 ft
STEAM GENERATOR SECONDARY SIDE	
Liquid level	.45 - .88 ft
Liquid temperature	513 - 517°F
Pressure	765 - 798 psia
Mass flow	43 - 47 lb/sec

Table 2.1.2 General Subsystem Component Characteristics and Trip Setpoints

PRESSURIZER

1.	Relief valve opens*	2422 psia
2.	Relief valve closes*	2402 psia
3.	Relief valve maximum flow rate*	2.0 lb/sec
4.	Safety valve setpoint*	2562 psia
5.	Safety valve maximum flow rate*	6.3 lb/sec
6.	Spray flow on setpoint	2222 psia
7.	Spray flow off setpoint	2199 psia
8.	Spray maximum flow rate	1.1 lb/sec
9.	Heaters on setpoint	2154 psia
10.	Heaters off setpoint	2184 psia
11.	Heater capacity	12 kw
12.	Backup heaters on setpoint	2148 psia
13.	Backup heaters off setpoint	2164 psia
14.	Backup heater capacity	36 kw

REACTOR

1.	High power scram*	51.5 MWt
2.	Manual scram	Operator action

PRIMARY LOOP

1.	Low pressure scram	2059 psia
2.	High pressure scram	2282 psia
3.	High hot leg temperature scram*	590°F
4	Low flow scram	953 lb/sec

STEAM GENERATOR

!.	MSFCV starts opening**	1033 psia
2.	MSFCV stops opening	1012 psia
3.	MSFCV starts closing	943 psia
4.	MSFCV stops closing	953 psia
5.	Low liquid level scram*	111 in.
6.	Feedwater pump trip	Indirect trip on scram

* Not exercised during tests L6-1, L6-2, L6-3.

** The main steam flow control valve (MSFCV) controls steam flow during steady-state operation and transient tests. It also serves as a pressure relief valve (Test L6-1).

Table 2.1.3 LOFT L6-1 Event Sequence

Event	Time After Transient Initiation, Sec	
	PTSPWR2	Plant Data
MSFCV closing initiated		0.0
Pressurizer heater off		6.1
Pressurizer spray on		9.1
MSFCV closed		11.6
Reactor scrammed		21.8
Maximum PCS pressure reached		22.0
MSFCV opened		22.2
$Q_{SG} > Q_{core}$ (pressurizer level at maximum)		26.5
Pressurizer spray off		30.4
Pressurizer heaters on		32.5
MSFCV closed		40.6
MSFCV opened		91.2
MSFCV closed		104.4

Table 2.1.4 LOFT L6-2 Event Sequence

Event	Time After Transient Initiation, Sec PTSPWR2	Plant Data
Pumps tripped		0.0
MSFCV closing initiated		1.8
Reactor scrammed		2.0
Pressurizer backup heater on		6.0
MSFCV closed		13.4
PCP coastdown complete**		18.0
Natural circulation established		~23.0
$Q_{SG} < Q_{core}$ (pressurizer level at minimum)		29.0
Pressurizer backup heaters off		97.2

** Time at which motor-generator sets were disconnected.

Table 2.1.5 LOFT L6-3 Event Sequence

Event	Time After Transient Initiation, Sec PTSPWR2	Plant Data
MSFCV started to open		0.0
Feedwater flow increased		1.4
Pressurizer backup heaters on		10.2
Maximum reactor power reached during experiment		15.6
Reactor scrammed		15.6
Feedwater flow terminated		16.6
MSFCV started to close		17.8
HPIS Pump A on		26.4
HPIS Pump B on		26.6
Minimum PCS pressure reached		26.8
$Q_{SG} < Q_{core}$ (pressurizer level at minimum)		33.0
MSFCV closed		36.2
HPIS Pump A off		48.6
HPIS Pump B off		50.0
Pressurizer backup heaters off		105.4
Pressurizer cycling heaters off		154.9

Table 2.2.1 Initial Conditions For Prairie Island Transient

Parameter	PTSPWR2	Plant Data
Power		1650 MWt
Core Flow		18900 lb/s
Hot Leg Temperature		591°F
Cold Leg Temperature		527°F
T_{AVE}		559°F
Loop ΔT		64°F
Pressurizer Pressure		2250 psia
Pressurizer Level		10 Ft
Feedwater Flow Rate*		944 lbm/s
Steam Generator Pressure		715 psia

*Per Generator

Table 2.2.2 Prairie Island Event Sequence

Event	Time After Transient* Initiation (sec) PTSPWR2	Plant Data
Tube break occurrence		0.*
Start of plant data		40.
Load reduction initiated		75.
Power reduction initiated		90.
Load reduction completed		90.
Reduced power (90%) attained		120.
Second charging pump started		171.
Third charging pump started		231.
Reactor trip (pressure <1915 psia)		240.
Turbine stop valve closed		240.
HPIS flow started (pressure<1845 psia)		248.
Broken loop RCS pump shut off		351.
Termination of computer run		—

* Assumed for the purposes of this comparison since the exact rupture time is unknown.

TABLE 2.3.1

INITIAL CONDITIONS SUMMARY COMPARISON BETWEEN RELAP5 AND PTSPWR2

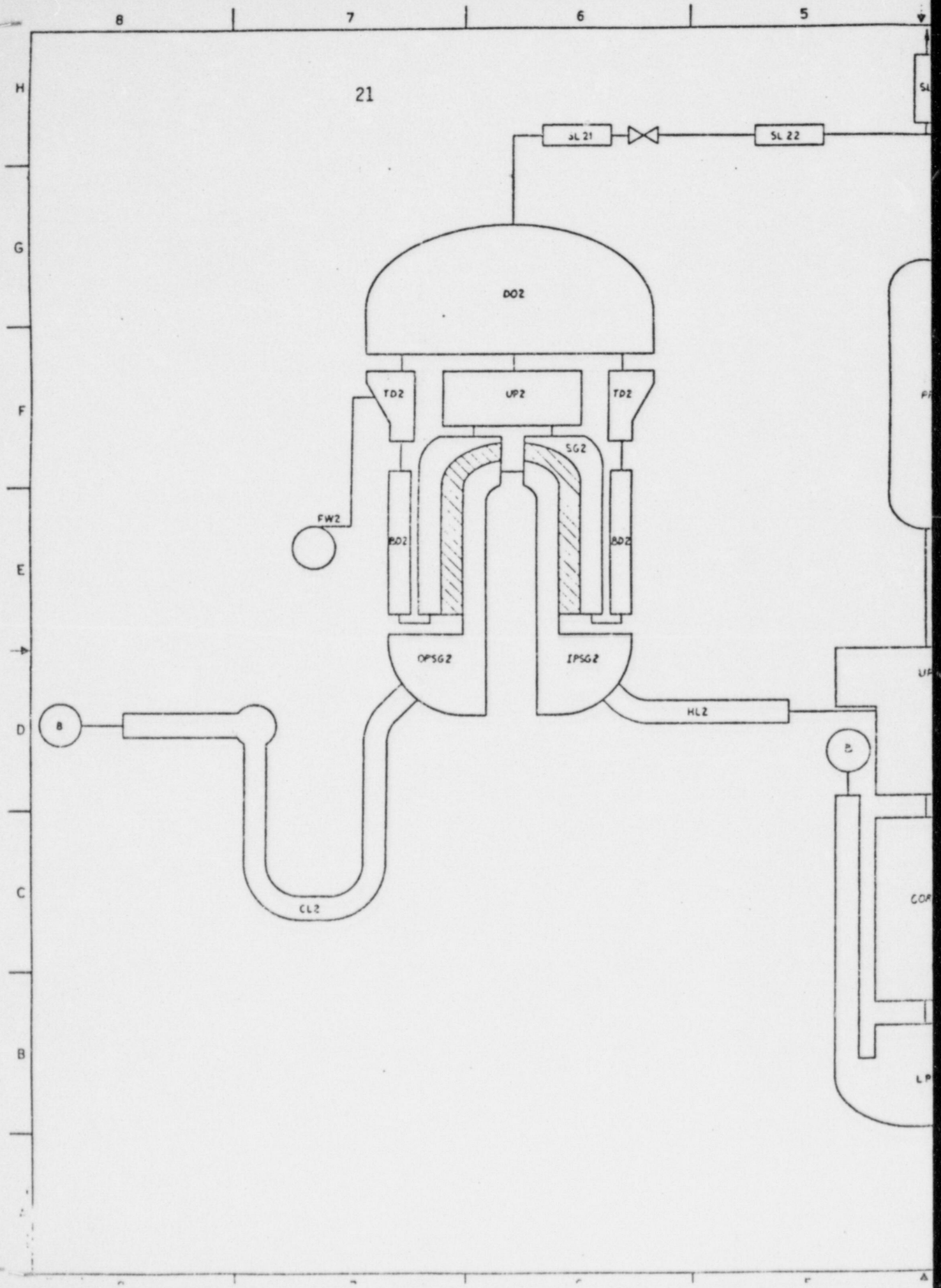
Parameter	Value PTSPWR2	RELAP5	Units	% Difference
T_{AVE}		575.2	°F	
Loop ΔT		57.67	°F	
Pressurizer Pressure		2250.0	psia	
Pump Speed		1190.0	rpm	
Core Flow		28522.0	lbm/s	
Core Power		2300.0	MWt	
Feed Flow *		935.0	lbm/s	
Steam Generator Pressure		748.2	psia	

*Per generator

TABLE 2.3.2.

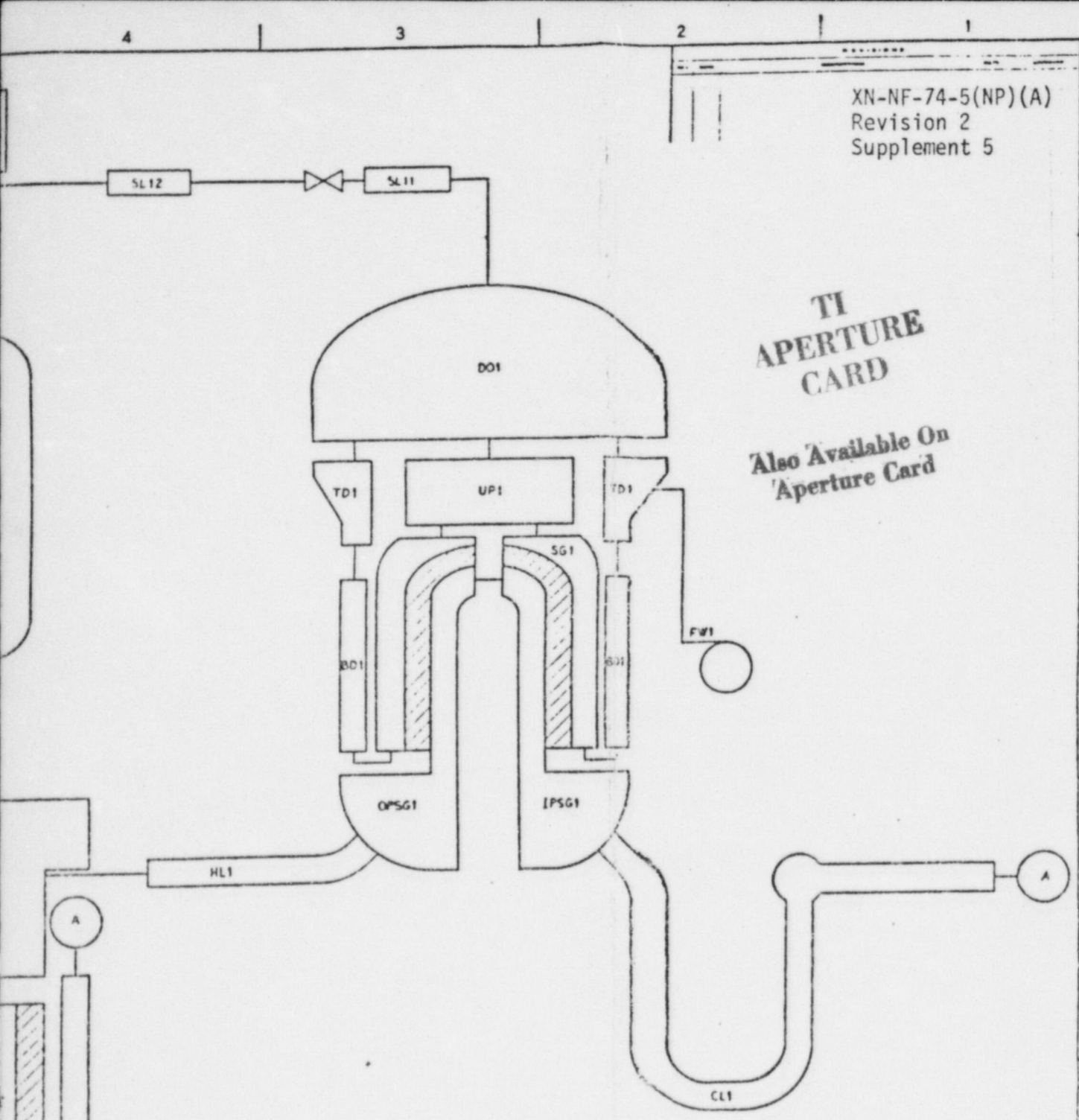
THREE PUMP COASTDOWN EVENT SEQUENCE

Event	Time After Transient Initiation (sec)
Pump trip (3 pumps)	
Scram signal (low loop flow)	
Reactor scrammed	
Turbine stop valve closing initiated	
Feedwater control valve closing initiated	
Turbine stop valve closed	
Run termination	


8
7
6
5
H
G
F
E
D
C
B
21
SL 21
SL 22
DO2
TD2
UP2
TD2
SG2
FW2
BD2
BD2
OPSG2
IPSG2
HL2
CL2
8

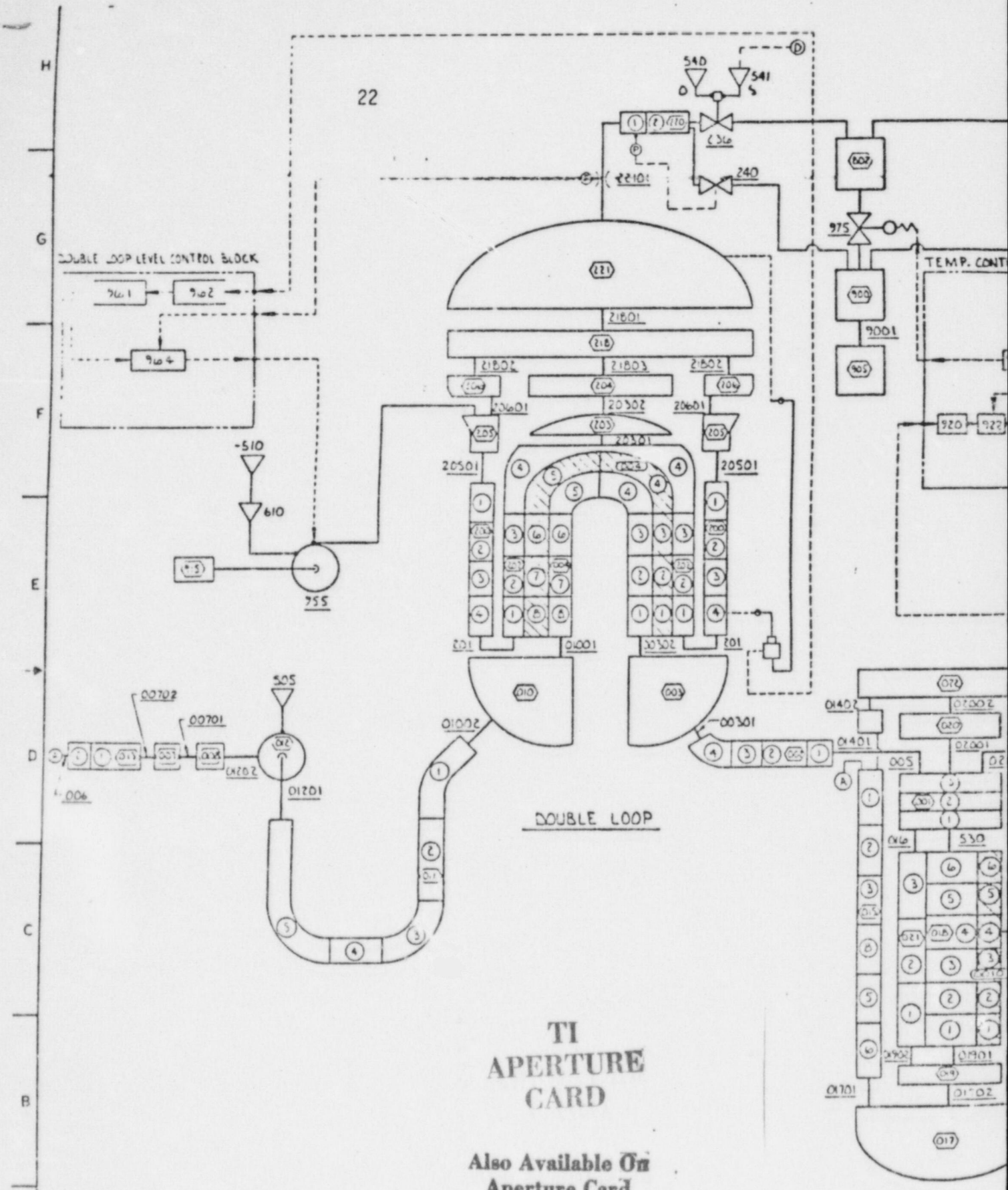
TI
APERTURE
CARD

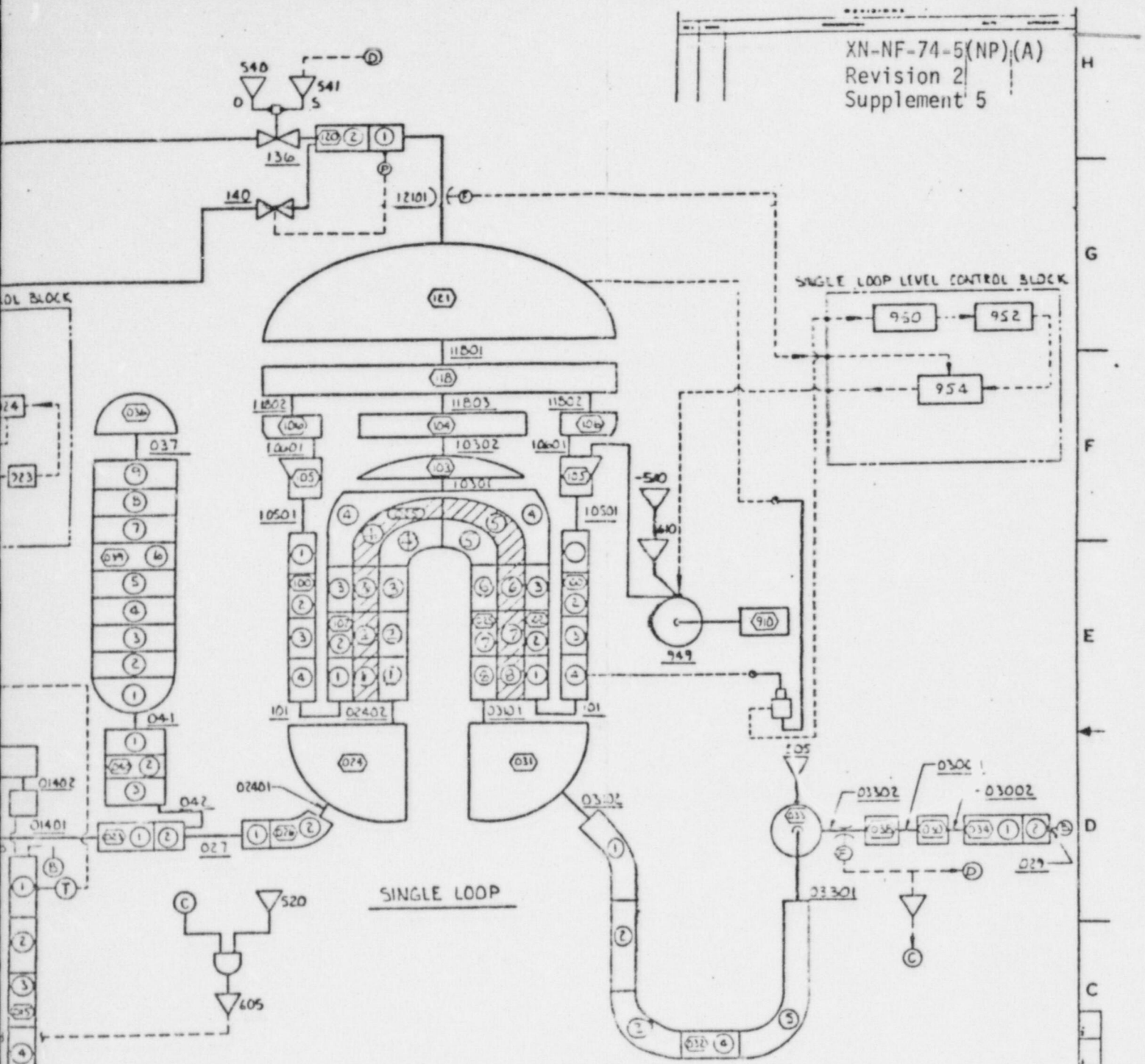
Also Available On
Aperture Card



8611040200-01

Figure 2.3.1 PTSPWR2 Schematic


22
DOUBLE LOOP LEVEL CONTROL BLOCK
DOUBLE LOOP
TEMP. CONT
TI
APERTURE
CARD
Also Available On
Aperture Card



8611040200-02

Figure 2.3.2 RELAP5 Schematic

3.0 BENCHMARK CASES - INPUT

This section presents a general discussion of the PTSPWR2 code input and modifications used to perform the benchmark calculations. The code modifications were necessary in order to model the special circumstances and plant specific controls associated with each of the cases.

3.1 LOFT TESTS

A general input deck containing geometry, system control, component performance, and common initial condition data was established for use for all three LOFT test simulations. The initial conditions were obtained from the LOFT post-test data report.(4) Some minor code modifications were necessary in order to simulate specific transient conditions which occurred during the tests, such as the manual HPIS pump shutoff during LOFT Test L6-3. Since the LOFT facility blowdown loop was blocked off during these tests, a one-loop simulation was performed with the PTSPWR2 code.

The following three subsections contain information related to specific changes that were required for each LOFT test simulation.

3.1.1 LOFT Test L6-1 - Input

For this transient, the turbine stop valve model in PTSPWR2 was used to simulate the initial closure of the main steam flow control valve

(MSFCV). An 11.6 second closing time was input to correspond to the value obtained from the test data[4].

The closing and opening times for the relief valves were set to 11.6 seconds and opening and closing setpoints of 1033 and 950 psia were input. These values correspond to MSFCV setpoints obtained from Reference 8.

3.1.2 LOFT Test L6-2 - Input

Homologous pump torque and head curves were input to the code in order to simulate the pump coastdown.

As was the case for the L6-1 simulation, an 11.6 second MSFCV closing time was assumed.

3.1.3 LOFT Test L6-3 - Input

For this simulation, an 87% increase in turbine flow rate over an 8 second period was programmed to simulate the steam flow rate ramp at the MSFCV during the test.

HPIS pump flow/pressure data was input to facilitate code predictions of the safety injection flow rates.

3.2 PRAIRIE ISLAND STEAM GENERATOR TUBE RUPTURE EVENT - INPUT

The bulk of the PTSPWR2 code input was taken directly from an established input deck used by ENC for licensing reloads for Prairie Island. Nuclear parameters were adjusted consistent with best estimates based on a cycle exposure of approximately 3600 MWD/MT and a critical boron concentration of approximately 740 ppm corresponding to the October 2, 1979 incident.

The data indicate slight asymmetric behavior prior to the reactor trip. After the trip, somewhat more asymmetry existed primarily due to the leak.

Steady-state operating conditions were obtained from References 9 and 10. Some of the initial values used for the PTSPWR2 code steady-state initialization differed slightly from the first recorded data points. During the initial portion of the event, with the exception of the pressurizer pressure and level, the system parameters essentially remained at their initial steady-state values or within controller steady-state operating dead bands. The pressurizer pressure and level steadily decreased due to primary system mass loss through the ruptured steam generator tube.

The primary temperatures and pressures were set at their nominally specified operating values.

In order to perform the Prairie Island calculations, certain assumptions had to be made and some iterative calculations had to be performed to establish reasonable estimates of certain input parameters. These are discussed in the following subsections.

3.2.1 Leak Rate

Since neither the time at which the leak began nor the initial leak rate are known, iterative calculations were performed to establish reasonable values for these parameters by comparing calculated and actual plant parameters (particularly pressurizer level).

3.2.2 Initial Pressurizer Level

The plant pressurizer consists of a right circular cylinder with hemispherical heads with electrical heaters located in the lower head region.

The initial pressurizer level was taken as 32.6% of the span between pressurizer level taps plus the equivalent distance between the lower tap and the bottom of the right circular cylinder equivalent, or a total of 10 ft.

3.2.3 Heaters, Charging Pumps, Letdown

There are 192.3 kW of proportional heater capacity and 807.7 kW of backup heater capacity in the Prairie Island Unit 1 pressurizer. The proportional heaters come on at 2250 psia and are full on at 2235 psia. The backup heaters are full on at 2225 psia.

The heaters are assumed to be fully off at a pressurizer level of 6.5275 ft. based on the above level definition.

There are three charging pumps with a total assumed pumping capacity of 22 lbm/sec. Based on event descriptions in References 5 and 6, the second pump is turned on at 171 seconds after initiation and the third pump is turned on at 231 seconds after leak initiation.

3.2.4 Load Demand, Turbine Demand, and Feedwater Enthalpy Ramp Rates

The 10% load demand reduction during this transient resulted in a 15% reduction in steam flow and a 10% reduction in core power. In order to ultimately achieve a new lower equilibrium operating point, a reduction in feedwater enthalpy would have had to occur. However, no feedwater enthalpy data were available.

The feedwater enthalpy reduction was necessary to match the steady average loop temperature response following the load reduction.

These three parameters were all ramped beginning at 90 seconds after leak initiation. This results in a time lag between the calculated and the published data for steam flow. The reported plant data appears to be shifted in time an unreasonable amount relative to core power since the reported large lag between core power and steam flow would cause more significant temperature and pressure increases than shown in the data and these effects would be displaced in time relative to the data.

The rod speed control system in PTSPWR2 normally uses loop average temperature, turbine flow, and reactor power as controller inputs.

3.2.5 Steam Dump Controls

After reactor scram and subsequent turbine stop valve closure, the turbine bypass valves and atmospheric dump valves were activated to mitigate the secondary side pressure rise and resultant primary loop average temperature increase. Depending on the control model used, the valves can be controlled by primary temperature, secondary side pressure, or a combination of these two parameters. Some code analyses of this transient have apparently keyed to control of the secondary side pressure.(6)

3.3 PTSPWR2/RELAP5 CODE COMPARISONS - INPUT

This section presents a general discussion of the PTSPWR2(1) and RELAP5(7) code inputs. The RELAP5 and PTSPWR2 inputs were developed from the existing Carolina Power and Light (CP&L) data used in their H. B. Robinson RETRAN model and from other supporting data supplied to ENC by CP&L(11).

The two input models were cross checked for consistency, especially in relationship to the parameters affecting the pump coastdown transient. In particular, the pump homologous curves, loop pressure drops, pressurizer data, and core inlet flow characteristics were made to be as consistent as

possible. Additionally, each input parameter was checked to be consistent within the constraint of the two code models and, where deviations appeared, the input data for the two codes were made to be consistent with each other.

Applicable nodalization diagrams are shown in Figures 2.3-1 and 2.3-2.

The major boundary conditions used for this analysis were feedwater flow and enthalpy, turbine inlet pressure, and reactivity insertion rate. In RELAP5, the feedwater flow rate was held constant until reactor trip, at which time the flow was linearly ramped to zero over a 25 second period. After initialization, the PTSPWR2 feedwater controller determined the flow rate until the turbine stop valve began to close. At that time, the valve closing time input to PTSPWR2 resulted in a feedwater flow ramp similar to the RELAP5 input ramp. Both codes used a constant value of 418.14 BTU/lb for the feedwater enthalpy throughout the transient.

The initial value of the turbine pressure setpoint was adjusted to initialize the PTSPWR2-calculated steam generator pressure to correspond to the RELAP5 initial value. The turbine stop valve closing time was set at 0.15 seconds, which determined the valve closing rate following reactor scram.

The reactor kinetics input data for both codes are consistent. The moderator and Doppler coefficients were set to conservative BOC values, resulting in a positive moderator coefficient and low Doppler. These values were chosen to achieve the maximum power feedback for this transient (heatup). Scram delay was set at 2.2 seconds after activation of the low flow trip, the low flow trip setpoint value being roughly equal to 85% of the initial loop flow value. The input rod reactivity insertion rate values for the two codes were equal.

All of the safety and relief valves in the NSSS are modelled by PTSPWR2. However, in order to achieve a maximum primary system pressure rise, the pressurizer safety valves were defeated by making the setpoint 2500 psia. In both codes, the steam generator relief valves were modelled so that they would reset if the pressure dropped below the closing pressure. Automatic pressurizer pressure control was not exercised either in PTSPWR2 or in RELAP5.

4.0 RESULTS - PTSPWR2 PREDICTIONS/DATA COMPARISONS

This section contains a discussion of the PTSPWR2 predictions for the LOFT tests, the Prairie Island steam generator tube rupture event, and for the H.B. Robinson 3-pump coastdown (RELAP5) analysis. Presentations of the comparisons and interpretations of the results are presented in Sections 4.1, 4.2, and 4.3, respectively.

4.1 LOFT TEST DATA COMPARISONS

Each of the three LOFT test benchmarks are discussed separately in the following subsections. Figures 4.1.1-1 to 4.1.1-11, 4.1.2-1 to 4.1.2-12, and 4.1.3-1 to 4.1.3-12 are comparison plots for tests L6-1, L6-2, and L6-3, respectively. The experimental data for the LOFT tests is from Reference 4.

4.1.1 LOFT Test L6-1 - Loss-of-Load Transient

LOFT Test L6-1 was initiated by a closure of the main steam flow control valve (MSFCV) while the reactor power was at 37 MWT. The valve closed over a period of 11.6 seconds and the reactor scrammed at 21.8 seconds due to high primary coolant system (PCS) pressure. After initial closure, the MSFCV, acting as a secondary coolant system (SCS) pressure relief valve, cycled two more times during the first 200 seconds of the test.

Figure 4.1.1-1 indicates the good agreement between predicted and actual reactor power level. The initial offset between the curves resulted from the use of the nominal steady-state power level listed in Reference 4 for code initialization. This value was roughly one

megawatt greater than the zero time value obtained from the actual test power level figure presented in that reference and reproduced in Figure 4.1.1-1. Roughly 10 seconds into the transient, the PCS temperatures began to increase because of the closure of the MSFCV. At that time core reactivity effects resulted in a reactor power decrease until scram occurred at 21.8 seconds. The ramp down in power was well predicted by PTSPWR2, especially when the initial offset is taken into consideration.

The PCS loop flow rate data displayed in Figure 4.1.1-2 is well predicted by the code. The initial flow rate value used for the simulation was obtained from the listed initial values appearing in Reference 4. The initial decrease in flow rate during the first 20 seconds resulted from PCS heatup during the MSFCV closure.

The turbine stop valve model in PTSPWR2 was modified to better simulate the linear closing characteristic of the LOFT MSFCV which is not representative of those installed in commercial nuclear power plant steamlines. Subsequent opening of the LOFT MSFCV was modeled with the PTSPWR2 steam generator relief valve model.

The LOFT steam flow rate data presented in Figure 4.1.1-3 indicates the possibility of a MSFCV leak after closure. Similar evidence of steam leakage is obtained from the valve position and condenser inlet flow rate data presented for Tests L6-2 and L6-3[4].

The initial rise, fall, and second rise in the steam generator pressure shown in Figure 4.1.1-4 resulted from the closing, opening, and reclosing of the MSFCV during the first 45 seconds. The opening/closing pressure values used for the simulation were obtained from Reference 3.

The steam generator level predictions found in Figure 4.1.1-5 follow the trend of the data. The rise in level occurring at 90 seconds resulted from steam generator relief valve actuation, which was not predicted by the simulation for reasons presented in the preceding paragraph. The long term level prediction was quite good, as can be ascertained from the figure.

Figure 4.1.1-6 shows that the predicted feedwater valve flow rates closely matched the data during the transient. The LOFT steam generator downcomer temperature data presented in Figure 4.1.1-7 were obtained from a sensor located near the bottom of the downcomer that exhibited a general sluggish response to the termination of feedwater flow and openings and closings of the MSFCV.

The predicted and measured temperatures during the last 100 seconds of the transient are in good agreement.

The cold leg and hot leg temperature rise and decrease during the first 40 seconds after transient initiation shown in Figures 4.1.1-8 and 4.1.1-9 resulted from the effects of steamline flow variations on SCS pressure and temperature.

The initial rise in predicted pressurizer pressure shown in Figure 4.1.1-11 resulted from the pressurizer insurge induced by the rise in PCS temperatures. Actuation of the pressurizer spray 9.1 seconds into the transient mitigated the pressure rise until the occurrence of reactor scram at 21.8 seconds.

In summary, the PTSPWR2 code predictions of the LOFT L6-1 test data are in generally good agreement with the test data. Further, the variation of PTSPWR2 predictions from the LOFT test data are explainable.

4.1.2 LOFT Test L6-2 Loss of Primary Flow Transient

LOFT Test L6-2 was initiated by tripping the power to the primary coolant pump (PCP) while the reactor was operating at 37 MWT. A low primary flow indication triggered a reactor trip and MSFCV closure occurred approximately two seconds after PCP trip. Following pump coastdown, a transition to natural circulation flow conditions occurred in the PCS.

During the pump coastdown period, roughly the first 25 seconds, good agreement between the primary loop flow prediction and the experimental data exists (Figure 4.1.2-3).

Steam generator related parameter comparisons are presented in Figures 4.1.2-4 to 4.1.2-8. As was the case for Test L6-1, evidence of a fluid loss is presented in Figure 4.1.2-4.

Figure 4.1.2-5 shows the initial steam generator rise resulting from the isolation of the steam generator during the first 14 seconds of the transient.

The initial rise in predicted hot leg temperature shown in Figure 4.1.2-9 resulted from a slight power/flow mismatch in the PCS during the early portion of the transient. The primary loop flow was slightly underpredicted during the coastdown and the hot leg temperatures are quite sensitive to a power/flow mismatch during a pump coastdown. Slight power/flow mismatches during the 30 second pump coastdown period when the flow and power decay rapidly are significant and can easily result in loop temperature discrepancies such as those shown in Figure 4.1.2-9. The hot leg temperature comparison during the last 60 seconds of the test when the cold leg temperature tracked the SCS saturation temperature was quite good. The cold leg data and predictions both indicate the relatively small variation in cold leg temperature associated with this LOFT flow coastdown test. The increase in secondary size pressure and temperature in this transient is limited by the power decrease after scram (2.0 seconds) and the decreased primary flow in the steam generator.

In summary, the primary loop hydraulics models are confirmed by the LOFT L6-2 benchmark. Primary loop pump speeds and flows predicted by PTSPWR2 during pump coastdown were in close agreement with the experimentally determined values. No flow data was available after the pump coastdowns, but the code predicted reasonable values for the natural circulation flow rates. Slight mismatches in power/flow during pump coastdown resulted in some small hot leg temperature data/prediction discrepancies.

4.1.3 LOFT Test L6-3 Excessive Load Increase Transient

LOFT Test L6-3 was initiated at a power level of 37 MWT by ramping the MSFCV to a full open condition. Low primary system pressure triggered reactor scram 15.6 seconds after the transient initiation and 2

seconds later the MSFCV began to close. The feedwater flow was terminated at approximately 17 seconds and the MSFCV closed at 36 seconds. High pressure injection system (HPIS) operation took place in the time period from 26 to 49 seconds after experiment initiation.

The reactor power level comparison prior to scram (Figure 4.1.3-1) indicated good prediction/data agreement. The 15% power level increase was induced by core reactivity effects resulting from the decreasing core inlet temperatures.

The hot leg flow data and prediction presented in Figure 4.1.3-2 both indicate an increase in loop flow rate resulting from the increased fluid densities associated with overall loop temperature decreases. The slight increase and decrease in predicted flow rate at 26 and 49 seconds, respectively, reflect the HPIS flow contribution.

The LOFT feedwater flow data recorder appeared to peg high from 5 to 15 seconds after the initiation of the steam load ramp, as shown

in Figure 4.1.3-4. It is therefore difficult to determine the amount of code underprediction during that period.

The HPIS flows predicted for this transient matched the data well, as shown in Figure 4.1.3-8. Since two HPIS pumps were operating, the total flow rate was double the values presented in the figure. Hence, there is good agreement between PTSPWR2 and the LOFT test result for HPIS flow.

The pressurizer level predictions (Figure 4.1.3-11) showed the same trend as the PCS loop temperature predictions.

Termination of HPIS flow at 49 seconds resulted in the level rise rate decrease at that time.

The pressurizer pressure comparison is shown in Figure 4.1.3-12.

In summary, the significant discrepancies between the predicted parameter values and the data for LOFT Test L6-3 occurred in the steam generator pressure, primary loop temperature, and pressurizer pressure comparisons.

The cold leg temperature response shown by the data is suspect during the first 10 seconds when the secondary sink temperature is significantly decreasing.

In conclusion, the LOFT L6-3 test demonstrates PWR system response to major perturbations of secondary side conditions. The PTSPWR2

predictions are reasonable for this event and the major differences between the LOFT L6-3 test results and PTSPWR2 predictions are explainable.

4.2 PRAIRIE ISLAND STEAM GENERATOR TUBE RUPTURE DATA COMPARISONS

The Prairie Island transient is characterized by the following four significant events:

- A steam generator tube rupture resulting in a primary system depressurization
- A load and power reduction
- A reactor trip due to low pressurizer pressure
- Operation of the steam turbine bypass and atmospheric steam dump systems to control primary temperatures and steam generator pressure.

The reactor power level comparison is presented in Figure 4.2-1. The ramp down of power resulting from the actuation of the automatic rod control system is well predicted by the code. The plant steam flow data indicated a load reduction of 15% over a 15 second period and the power data indicated a 10% reduction in power generation over a ten second period.

Following the load/power reduction, the power level stabilized and was again well predicted by the PTSPWR2 code.

The primary loop average temperature prediction presented in Figure 4.2-2 tracks the plant data well during the entire transient. Following reactor scram, the transient loop temperatures were strongly influenced by the steam bypass flow rate and duration (see Figure 4.2-10).

Figure 4.2-2 confirms the validity of the bypass controller modelling in PTSPWR2.

Hot and cold leg temperature comparisons are presented in Figures 4.2-3 and 4.2-4, respectively.

The rise in cold leg temperature results from the increased secondary side pressure caused by the decreasing steam flow.

Following reactor scram, the hot leg temperature decreases due to the power decay and the cold leg temperature rises because of the isolation of the steam generators and the subsequent increase in secondary side temperature. The loop temperature difference decreases rapidly as indicated in Figure 4.2-5.

In summary, the loop average temperatures, hot leg temperatures, and the loop hot-to-cold leg temperature differences following scram are

all well predicted by PTSPWR2.

The pressurizer level prediction shown in Figure 4.2-6 accurately tracks the data prior to scram since the initial primary-to-secondary leak rate used in PTSPWR2 was established to ensure a reasonable fit to the data. For a short period following scram, the pressurizer liquid level fell below the location of the bottom pressure tap, resulting in constant level value of 2.5 ft. shown by the data. The code predictions indicate the pressurizer may have emptied shortly after scram due to the drop in loop average temperature combined with the loss in primary inventory.

Prior to reactor trip, the pressurizer pressure (Figure 4.2-7) accurately tracks the plant data.

The steam generator level prediction shown in Figure 4.2-8 accurately predicts the data during the pre-scram period.

The actual liquid level dropped below the lower pressure tap shortly after scram, resulting in invalid comparisons for the duration of the transient.

Figure 4.2-9 presents the steamline flow predictions for this transient.

The long term steam flow prediction accurately tracks the steam flow data and maintains the primary loop average temperature at the prescribed setpoint, attesting to the adequacy of the overall energy balance model used in PTSPWR2.

The steam generator pressure data presented in Figure 4.2-10 is in general well predicted by the code. The rise time and magnitude immediately following scram are strong functions of the dump and bypass valve capacities, controller gains and time constants, and controller operation scenario which are not accurately known.

The transient feedwater flow rate prediction found in Figure 4.2-11 closely follows the data trends.

The drop in steam generator level during the load reduction and subsequent feedwater controller action resulted in the dips and rises after 90 seconds.

In summary, the PTSPWR2 code in general accurately predicted the plant data obtained during the Prairie Island steam generator tube rupture transient. In particular, the comparison indicates the validity of the PTSPWR2 primary and secondary energy balances, pressurizer pressure model, steam generator heat transfer model, the automatic rod speed control model and the turbine-bypass control model.

4.3 PTSPWR2/RELAP5 COMPARISON FOR H.B. ROBINSON UNIT 2 THREE-PUMP COASTDOWN SIMULATION

The H.B. Robinson Unit 2 flow coastdown simulated by the RELAP5 and PTSPWR2 codes was initiated by a three pump coastdown of the RCS pumps.

Following reactor scram the primary loops experienced a transition to natural circulation flow conditions.

Once the feedwater flow was terminated (29.1 seconds), the only two control systems that could be activated were the pressurizer safety valve (setpoint = 2500 psia) and the steam generator relief valves (setpoint = 1100 psia). The pressurizer relief valves, spray flow, and heaters were deactivated for both simulations.

Figures 4.3-1 to 4.3-16 present the comparisons of the PTSPWR2 and RELAP5 code predictions. Code initialization and tracking information for the more important system parameters can be obtained from these plots.

Figure 4.3-1 indicates the excellent agreement of the transient core power level predicted by the two codes. The figure indicates a strong similarity between the core kinetics decay heat, and reactivity insertion models utilized by the codes. Over the long term, the PTSPWR2 code conservatively overpredicted the RELAP5 results slightly.

Figures 4.3-2 and 4.3-3 illustrate the generally good agreement of the pump coastdown speeds and primary loop flow rates respectively.

Steam generator parameters of interest are found in Figures 4.3-4 to 4.3-6.

Figures 4.3-7 to 4.3-14 trace the predicted fluid temperatures around the primary loops from core outlet back to core inlet. In Figure 4.3-7 the core outlet temperatures from the PTSPWR2 simulation are compared to the temperatures in the RELAP5 upper plenum node 101.

During the first 25 seconds when the core flows closely agree, temperature predictions are in excellent agreement, indicating the validity of the PTSPWR2 core thermal model. The initial peak in temperature (Figure 4.3-7 to 4.3-10) results from a power/flow mismatch during the pump coastdown before reactor scram. Immediately after scram the reactor power decays much more rapidly than the loop flow, creating a sharp decline in core outlet temperature.

Figures 4.3-8 to 4.3-10 illustrate the fluid temperature propagation from the reactor vessel outlet to the inlet of the steam generator inlet plenum. The RELAP5 node numbers are indicated on each plot. PTSPWR2 cold leg temperatures were obtained by averaging the reactor vessel outlet and steam generator inlet temperatures. Near the end of the 200 second simulation time, the temperatures calculated by the two codes are virtually identical,

indicating the excellent agreement of the core power and loop flows predicted by the two codes.

The RELAP5 data shown in Figures 4.3-7 to 4.3-10 were plotted at the rate of one data point for every 5 seconds of transient time compared to the PTSPWR2 rate of one point for every 0.5 seconds. Since the first peak in the RELAP5 core outlet temperature occurred at approximately, five seconds after transient initiation at the second plot point, the height of that peak corresponds to the PTSPWR2 peak. However, as the peaks in Figures 4.3-8 to 4.3-10 shift to the right away from the five second edit, they appear truncated compared to the PTSPWR2 peak. This truncation is therefore induced by the fact that ten PTSPWR2 data points are plotted for every RELAP5 point.

Figures 4.3-11 to 4.3-14 track the primary loop fluid temperatures from the steam generator outlet to the core inlet. The initial decrease in steam generator outlet temperature (Figure 4.3-11) reflects the approach of the tube outlet temperature to the secondary side temperature as the flow decreases. The rise occurring at 5 seconds follows the increase in secondary side (Figure 4.3-4) saturation temperature. The plateau occurring at 10 seconds results from the limitations of the secondary side temperature rise by the steep decrease in primary side inlet temperature (Figure 4.3-10). Subsequent peaks and valleys are in general induced by secondary side pressure fluctuations produced by openings and closings of the steam generator relief valves. The substantial rise in the PTSPWR2 steam generator outlet temperature (Figure 4.3-11) occurring at approximately 40 seconds reflects the rise in steam generator inlet temperature (Figure 4.3-10). During the final 100 seconds of the simulation, the steam generator outlet temperatures

predicted by both codes closely approached the saturation temperature of the secondary side, as would be expected for flow conditions in a steam generator designed for flow rates roughly twenty times greater than natural circulation values.

Figures 4.3-12 to 4.3-14 illustrate the fluid temperature propagation through the cold leg and reactor vessel inlet plenum. The temperature peaks and valleys shown in Figure 4.3-12 are somewhat muted because of the spatial averaging used to define the cold leg temperature for the PTSPWR2 simulation.

The actual dip in the RELAP5 core inlet temperature occurring at 12 seconds is not well represented in Figure 4.3-14 because of the five second edits mentioned previously.

Near the end of the run when quais-static natural circulation conditions are established, the core inlet temperature predicted by the two codes are in excellent agreement.

In summary, the generally excellent agreement of the loop temperatures predicted by the PTSPWR2 and RELAP5 codes attests to the validity of the PTSPWR2 code's core thermal model, primary loop thermal transport model, plena mixing models, and steam generator heat transfer model.

Pressurizer parameters are presented in Figures 4.3-15 and 4.3-16. The initial rise in pressurizer level shown in Figure 4.3-15 is induced by the temperature spike occurring in the hot leg and reactor vessel upper plenum (Figures 4.2-9 and 4.2-8, respectively).

During the last 80 seconds the loop temperatures predicted by both codes are similar and the calculated pressurizer levels are in good agreement.

The pressurizer pressure trend (Figure 4.3-16) closely resembles the level trend. The RELAP5 pressurizer is known to underpredict pressure increases during insurges[13,14].

In summary, the PTSPWR2/ RELAP5 benchmark results attest to the applicability of the PTSPWR2 pump coastdown, natural circulation, loop and plena energy transport, steam generator heat transfer reactor core thermal, and reactor core power generation models to plant transient analyses. In general, the relatively minor differences in predictions are explainable. The core inlet temperature predicitons produced by the two codes are in good

agreement, especially during the first 10 seconds when MDNBR is expected to occur and near the end of the transient when natural circulation conditions are fully established.

5.0 REFERENCES

1. D.M. Turner, "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS-PWR)," XN-74-5(P), Rev. 2, Exxon Nuclear Co., Richland, WA, October 1983.

2. (a) W.T. Nutt, "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS-PWR): PTSPWR Modifications for St. Lucie Unit 1," XN-74-5(P), Rev. 2, Supplement 1, Exxon Nuclear Company, Richland, WA, October 1983.

   (b) D.M. Turner, G.C. Cooke, "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS-PWR): Responses to NRC Questions," XN-74-5(P), Rev. 2, Supplement 3, Exxon Nuclear Company, Richland, WA, to be issued.

   (c) D.M. Turner, "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS-PWR): Code Updates," XN-74-5(P), Rev. 2, Supplement 4, Exxon Nuclear Company, Richland, WA, August 1984.

3. D.L. Reeder, "LOFT System and Test Description (5.5-ft Nuclear Core 1 LOCES)," NUREG/CR-0247 TREE-1208, EG&G Idaho, Inc., Idaho Falls, ID, July 1978.

4. D.L. Batt and J.M. Carpenter, "Experiment Data Report for LOFT Anticipated Transient Experiments L6-1, L6-2, and L6-3," NUREG/CR-1797 EGG-2067, EG&G Idaho, Inc., Idaho Falls, ID, December 1980.

5. "Prairie Island Nuclear Generating Plant License Event Report," RO79-027, Update, Northern States Power Company, Minneapolis, Minnesota, February 1980.

6. "Using RETRAN-02 and DYNODE-P to analyze Steam Generator Tube Breaks, NSAC-47, Nuclear Safety Analysis Center, May 1982.

7. V.H. Ransom, et al., "RELAP5/MOD1 Code Manual," NUREC/CR-1826, EGG-2070, U.S. Nuclear Regulatory Commission Report, March 1982.

8. C.L. Nalezny, "Summary of Nuclear Regulatory Commission's LOFT Program Experiments," NUREG/CR-3214 EGG-2248, EG&G Idaho, Inc., Idaho Falls, ID, July 1983.

9. "Prairie Island 1 Reactor Data Package," NSPNAD-8311P, Northern States Power Company, August 1983.

10. "Precautions, Limitations, and Set Points for Nuclear Steam Supply Systems, Prairie Island Nuclear Generating Plant Units 1 and 2," Revision 6, Westinghouse Electric Corporation, Pittsburgh, PA, March 1976.

11. Letter from T.M. Dresser to T.J. Helbling, "Principal ECCS and Plant Transient Analysis Parameters for H.B. Robinson Unit 2, NF-84-1666, File No. FG-1111.08, Exxon Nuclear Co., Richland, WA, March 14, 1984.

12. H. Ollikkala and S.R. Behling, "Posttest Analysis of LOFT Anticipated Transient Experiments L6-1, L6-2, L6-3, and L6-5," EGG-LOFT-6159, EG&G Idaho, Inc., Idaho Falls, ID, January 1983.

13. H.R. Saedi, "Insurge Pressure Response and Heat Transfer for PWR Pressurizer," Master's Thesis, Massachusetts Institute of Technology, Dept. of Mechanical Engineering, November 1982.

14. J.C. Lin, et al., "RELAP5/MOD2 Pressurizer Modeling," EGG-M-10984, EG&G Idaho, Co., Idaho Falls, ID, April 1984.

15. N.F. Fausz and T.W. Patten, "XCOBRA-IIIC, A Computer Code to Determine the Distribution of Coolant During Steady State and Transient Core Operation," XN-NF-75-21, Rev. 2, Exxon Nuclear Co., Richland, WA, July 1982.

XN-74-5(NP)(A)
Revision 2
Supplement 6
Issue Date 10/8/86

DESCRIPTION OF THE EXXON NUCLEAR PLANT TRANSIENT SIMULATION MODEL FOR PRESSURIZED WATER REACTORS (PTS-PWR): REVISED PTSPWR2/RELAP5 BENCHMARK ANALYSIS

Prepared by: D. M. Turner
D. M. Turner
Model Development

Concur: G. C. Cooke 11/21/84
G. C. Cooke, Manager
Model Development

Concur: W. V. Kayser 11/21/84
W. V. Kayser, Manager
PWR Safety Analysis

Concur: J. C. Chandler 11/21/84
J. C. Chandler
Reload Fuel Licensing

Approve: R. B. Stout 11/26/84
R. B. Stout, Manager
Licensing & Safety Engineering

Approve: L. E. Hansen for 11/26/84
G. A. Sofer, Manager
Fuel Engineering & Technical Services

naa

EXXON NUCLEAR COMPANY, INC.

NUCLEAR REGULATORY COMMISSION REPORT 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 U.S. Nuclear Regulatory Commission as part of a technical contribution to facilitate safety analyses by licensees of the U.S. Nuclear Regulatory Commission 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 U.S. Nuclear Regulatory Commission in its review of this report, and under the terms of the respective agreements, by licensees or applicants before the U.S. Nuclear Regulatory Commission which are customers of Exxon Nuclear in their demonstration of compliance with the U.S. Nuclear Regulatory Commission's regulations.

Exxon Nuclear's warranties and representations concerning the subject matter of this document are those set forth in the agreement between Exxon Nuclear and the customer to which this document is issued. Accordingly, except as otherwise expressly provided in such agreement, neither Exxon Nuclear nor any person acting on its behalf:

A. Makes any warranty, or representation 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 AND SUMMARY	1
2.0	DESCRIPTION OF CODE MODIFICATION	3
3.0	RESULTS	5
4.0	REFERENCES	24

LIST OF FIGURES

Figure		Page
3.1	Revised PTS/RELAP5 Benchmark - Core Power	8
3.2	Revised PTS/RELAP5 Benchmark - RCS Pump Speed	9
3.3	Revised PTS/RELAP5 Benchmark - Primary Loop Flow Rate	10
3.4	Revised PTS/RELAP5 Benchmark - Steam Generator Pressure	11
3.5	Revised PTS/RELAP5 Benchmark - Steam Generator Level	12
3.6	Revised PTS/RELAP5 Benchmark - Steam Generator Downcomer Temperature	13
3.7	Revised PTS/RELAP5 Benchmark - Core Outlet Temperature	14
3.8	Revised PTS/RELAP5 Benchmark - Reactor Vessel Outlet Temperature	15
3.9	Revised PTS/RELAP5 Benchmark - Hot Leg Temperature	16
3.10	Revised PTS/RELAP5 Benchmark - Steam Generator Inlet Temperature	17
3.11	Revised PTS/RELAP5 Benchmark - Steam Generator Outlet Temperature	18
3.12	Revised PTS/RELAP5 Benchmark - Cold Leg Temperature	19
3.13	Revised PTS/RELAP5 Benchmark - Reactor Vessel Inlet Temperature	20
3.14	Revised PTS/RELAP5 Benchmark - Core Inlet Temperature	21
3.15	Revised PTS/RELAP5 Benchmark - Pressurizer Level	22
3.16	Revised PTS/RELAP5 Benchmark - Pressurizer Pressure	23

1.0 INTRODUCTION AND SUMMARY

The Exxon Nuclear Company (ENC) plant transient simulator computer code PTSPWR2 is used by ENC for safety analysis of pressurized water reactors. References 1 through 5 contain descriptions of the code and its application to PWR transients. In particular, Reference 5 contains benchmark comparisons of calculated primary and secondary system performances against available data.

This supplement contains the results of a revised PTSPWR2/RELAP5 comparison benchmark calculation generated by a new version of PTSPWR2 that uses a flow dependent heat transfer coefficient based on the Dittus-Boelter correlation for the primary side surface of the steam generator tubes. The results of the revised analysis are compared to the results of the original PTSPWR2/RELAP5 benchmark of the H.B. Robinson Unit 2 three-pump coastdown calculation.

2.0 DESCRIPTION OF CODE MODIFICATION

The following form of the Dittus-Boelter heat transfer coefficient correlation[6] was incorporated into the PTSPWR2 steam generator model to account for primary loop flow dependence:

$$h_{DB_i} = 0.023 \frac{K_i}{3600\ D_{id}} (N_{RE_i})^{0.8} (N_{PR_i})^{0.4} \qquad (2.1)$$

where,

D_{id} = steam generator tube inside diameter, ft.

h_{DB_i} = Dittus-Boelter heat transfer coefficient for i^{th} steam generator node, BTU/sec-ft^2-^{o}F

K_i = thermal conductivity of water, in i^{th} node, BTU/hr-ft-^{o}F

N_{PR_i} = Prandtl number of water in i^{th} primary side node, dimensionless

N_{RE_i} = Reynolds number of water in i^{th} primary side node, dimensionless

The thermal conductivity of water, Prandtl number, and Reynolds number for each steam generator node are all evaluated using the mixed (outlet) temperature for that node.

where,

The overall heat transfer coefficient U_{T_i} is used in conjunction with the overall primary-secondary temperature difference and tube outside diameter D_{od} to determine the primary-secondary heat transfer for each steam generator node.

The overall steam generator heat transfer coefficient during the transient can therefore be calculated for each time step by

where,

The overall heat transfer coefficient U_{T_i} is used in conjunction with the overall primary-secondary temperature differences and tube outside diameter D_{od} to determine the primary-secondary heat transfer for each node during a transient analysis.

3.0 RESULTS

The results of the comparison of the revised PTSPWR2 code predictions to the original benchmark predictions[5] are presented in Figures 3.1 through 3.16. In these figures, the solid curves represent the original PTSPWR2 code predictions[5], the dotted curves represent the revised PTSPWR2 code predictions, and the dashed curves represent the RELAP5 code predictions[5].

Figures 3.12 through 3.14 illustrate the effects of the steam generator outlet temperature differences on the cold leg, reactor vessel inlet, and core inlet temperature comparisons.

Figures 3.15 and 3.16 present the pressurizer level and pressure predictions, respectively.

4.0 REFERENCES

1. D.M. Turner, "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS-PWR)," XN-74-5(P), Rev. 2, Exxon Nuclear Co., Richland, WA, October 1983.

2. W.T. Nutt, "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS-PWR): PTSPWR Modifications for St. Lucie Unit 1," XN-74-5(P), Rev. 2, Supplement 1, Exxon Nuclear Company, Richland, WA, October 1983.

3. D.M. Turner, G.C. Cooke, "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS-PWR): Responses to NRC Questions," XN-74-5(P), Rev. 2, Supplement 3, Exxon Nuclear Company, Richland, WA, September 1984.

4. D.M. Turner, G.C. Cooke, "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS-PWR): Code Updates," XN-74-5(P), Rev. 2, Supplement 4, Exxon Nuclear Company, Richland, WA, August 1984.

5. D.M. Turner, G.C. Cooke, "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS-PWR): Code Benchmarks," XN-74-5(P), Rev. 2, Supplement 5, Exxon Nuclear Company, Richland, WA, August 1984.

6. W.H. McAdams, "Heat Transmission," McGraw-Hill, New York, 1954, p. 219.

XN-74-5(NP)(A)
Revision 2

Issue Date: 10/9/86

DESCRIPTION OF THE EXXON NUCLEAR PLANT TRANSIENT SIMULATION MODEL FOR PRESSURIZED WATER REACTORS (PTS-PWR)

Distribution

NRC/LA Nielsen (19)

Document Control (5)