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September 2, 1982

Dr. W. Jensen
Reactor Systems Branch
Division of Systems Integration
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Completion of Task 3 of FIN A2121: Final Reviews of the TRAP-2, POWER TRAIN and CADDS Topical Reports

Dear Dr. Jensen:

The enclosed final reports, which document the ANL review of the TRAP-2, POWER TRAIN and CADDS topical reports, are submitted to fulfill Task 3 of FIN #A2121. In accordance with the telephone conversation between ANL and NRR of August 17, 1982, these documents include revisions to the draft reviews submitted August 2, 1982.

Sincerely,

P. B. Abramson

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PBA:kr

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SUGGESTED TOPICAL REPORT EVALUATION ON POWER TRAIN
PROVIDED TO NRR BY ANL

Report No. and Title: BAW-10149, Rev. 1, "POWER TRAIN, Hybrid Computer Simulation of a Babcock & Wilcox Nuclear Power Plant"

Originating Organization: Babcock & Wilcox Co.

I. Summary of Topical Report

Report BAW-10149, Rev. 1, describes the POWER TRAIN code used by Babcock & Wilcox to predict normal and transient operation of Babcock & Wilcox nuclear power plants (177-, 205-, and 145-fuel assembly plants). As is stated in the report, POWER TRAIN is a real-time, on-line hybrid computer simulation used to predict the performance and behavior of the major components in the nuclear steam system (NSS) for a wide range of plant conditions and operation. POWER TRAIN is designed to model as much of the power plant as is feasible, especially those components whose behavior is interrelated with that of others. The objectives of the simulation are:

1. Two-loop simulation.
2. Automatic setup and checkout capability.
3. Flexibility to change system parameters, especially control system gains and setpoints.
4. Operation of the simulation (as nearly as possible) by actuating pushbuttons on or off.

The scope of the simulation, together with the automatic setup and checkout requirements, necessitates implementation techniques employing digital and hybrid (digital-analog) methods. The simulation requires the resources of two digital computers (CDC 1700 and EAI-640) and two solid-state analog computers (EIA-680). Calculational speed requirements and consideration of future machine requirements result in the use of a hardware floating point array processor (AP-120B) that performs calculations for the CDC-1700, and thereby greatly increases the amount of the simulation that can be programmed on the digital computers.

Implementation of the major components on the various computers is as follows:

1. CDC-1700, AP-120B -- Reactor, control rod drive, pressurizer, primary system flows and temperatures, condensate system flows, feedwater and condensate system energy balance, and turbine extraction flows.

2. EAI-640 -- Control system (ICS), protection system, atmospheric dump valves, bypass valves, and steam relief valves.

3. EAI-680, Console 1 -- Steam generators and portion of steam line.

4. EAI-680, Console 2 -- Feedwater system flows, remainder of steam lines, feedwater control valves, feedwater pumps, and turbine-generator.

The analog computer portions of the simulation are set up and checked out automatically using COMANCHE, a program developed by Control Data Corporation for the CDC 1700 computer, COMANCHE is a system routine (resident on the CDC 1700) which sets potentiometers on the analog consoles and then performs a "static check" of the patchboards and analog components before performing production runs.

II. Summary of Regulatory Evaluation

In various publications, Babcock & Wilcox has stated that POWER TRAIN is used for transient analysis of the following events: (The numbers preceding the events refer to the Safety Analysis Report section in which the results of the transient analyses are reported.)

- 15.1.1 Decrease in Feedwater Temperature
- 15.1.2 Increase in Feedwater Flow
- 15.1.3 Increase in Steam Flow
- 15.2.2 Loss of External Load
- 15.2.3 Turbine Trip
- 15.2.6 Loss of Nonemergency AC Power to the Station Auxiliaries
- 15.2.7 Loss of Normal Feedwater Flow
- 15.8 Anticipated Transients Without Scram (for the above events)

In our evaluation we have applied the acceptance criteria presented in the above-numbered 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 RELAP4/MOD6 computer code.[1] Our conclusions regarding the use of the POWER TRAIN code as described by BAW-10149, Rev. 1, are stated in the Staff Position section of this report.

A. Review of Analytical Models

POWER TRAIN is a two-loop fixed nodalization code (with the exception of the steam generator where variable axial nodalization capability is available) which is essentially an analog simulator. The feedwater train is modeled in detail with simple one-node approximations for each component. The equations used are basically perturbations around a steady state utilizing steady state characteristics. The primary side consists of a one node core with bypass, a three region nonequilibrium pressurizer, a transport delay model for the piping, input tables for the pumps and a variably noded steam generator.

Initialization on the primary side consists of inputting temperatures in the core and the primary/tube metal heat transfer coefficients. On the secondary side, the steam generator is initialized by inputting heat transfer coefficients and nominal pressure gradients. Pressure distributions are input for the feedwater train. The mass/energy equations for the train are initialized through the use of the steady state characteristics.

A detailed evaluation of the major models in POWER TRAIN is presented next.

1. Core

a. Core Neutronics Model

The equations used for the neutronics model are the well-known point kinetics equations, with use of up to six groups of delayed neutrons permitted. The prompt jump approximation is made. The code therefore cannot be used near prompt critical. Reactivity feedbacks from both average fuel and average moderator temperature changes are calculated. Reactivity contributions from boron addition or dilution, and from control and safety rod movements are obtained from models that use boron-reactivity addition/dilution rate, normal rod velocity, initial rod insertion, maximum rod insertion, rod drive time constant, and reactivity vs. rod position data, as input. There is no boron transport model. No provision is made for the voids produced by subcooled boiling and the concomitant reactivity effect is therefore not modelled.

b. Decay Heat Model

Decay heat is provided for by a model that uses a one time constant asymptotic decay-heat power, a threshhold power below which the decay-heat model functions, and a minimum power, all input.

c. Heat Transfer Model/Thermal-Hydraulics

Conservation of energy equations for fuel, cladding, and coolant are solved, where each material is modeled as a single node. Heat transfer conductances between materials are input and held constant, except that the cladding-coolant heat transfer conductance is proportional to the coolant flowrate to the 0.8 power. All of the power is deposited in the fuel. Coolant specific heat capacity is an input function of both pressure and temperature and a constant bypass fraction is used. Uniform pressure is assumed and flow has to be input. Pump energy terms therefore cannot be properly accounted for in the energy equation.

2. Reactor Loop

The primary flow system is simulated as two loops connecting the reactor vessel with two steam generators. Flow vs. time data for each loop, data normally prepared by the PUMP code, are input. Reversed (negative) flow may be specified. Logic is provided for reversing node inlet and outlet locations during reversed flow situations. Only single-phase, incompressible flow is allowed. Temperature response between any two successive points in the primary system is modeled by a flow-dependent mixing constant, a flow-dependent transport delay, or a set of differential equations. Plenum models assume that all inlet flow mixes instantaneously with fluid already in each plenum and that the outlet temperature is equal to the bulk average temperature. Outlet temperatures from the hot and cold legs are calculated by a flow-dependent transport delay model. Primary system pressure is calculated by considering thermal expansion of the primary coolant and net mass addition (including makeup) to the primary system, excluding the pressurizer. The pressurizer pressure is related to the primary system pressure through the surge line pressure drop with the drop being treated quasistatically.

The nonequilibrium pressurizer is modeled with two liquid regions, upper and lower (the latter representing any coolant insurge), and a vapor region. All regions are at the same pressure but can be at different temperatures. The factor for mixing between upper and lower liquid regions is input. The energy and mass conservation equations are solved; included in the solution are energy transferred to the pressurizer wall (constant wall temperature is input), energy transferred across the steam-liquid interface, condensation, and evaporation. An equivalent coefficient for boiloff is obtained by calibration with the CADDS pressurizer model. The forcing function is surge rate, with sprays, heaters, pilot valves, and relief valves operating at setpoints provided as input. All spray flow is assumed to be heated to saturation by condensation of vapor on the spray. Spray flowrate is determined by spray valve position as determined from input stem velocity. The number of pumps operating also affect spray flowrate. Flow capacities of safety and power operated relief valves are input; the effects of pressure and quality cannot be taken into account. The water properties for the pressurizer are polynomial fits and spot checking within a limited operating range has shown that with the exception of the saturation liquid density most of the properties are accurate to well within a percent. The error in the saturation liquid density can be of an order of a few percent.

POWER TRAIN has options for either a once-through steam generator (OTSG, feedwater preheated to saturation by recirculated steam in a downcomer), or an integral economizer OTSG (feedwater heated to saturation by the tube bundle). As elsewhere on the primary side, only single-phase, incompressible, uniform pressure flow is considered. Conservation of energy equations for primary and secondary fluid are solved using the continuous space, discrete time method, where finite differencing is used for the time derivatives and continuous one-dimensional integration is used for the spatial derivatives. On the secondary side there are three regions, subcooled, boiling, and superheated, whose boundaries may change each time step depending on the local fluid state. Tube metal temperature is calculated by a one node radial conduction model. On the primary side the heat transfer coefficient is input while on the secondary side it is dependent on the flow regime. Due to the assumption of constant primary-to-metal heat transfer coefficient in the steam generator, at low flowrates calculated heat transfer can be, due to this one source, 20 percent too high. On the secondary side, in the superheated region, the heat transfer coefficient is assumed proportional to flow. In the nucleate boiling region the Thom correlation with a multiplier is used. In the subcooled region it is assumed to be a weighted sum of the Thom coefficient and a term proportional to the flow. The Thom multiplier is calibrated against the Alliance Research Center (ARC) 19 tube Loss of Feedwater Flow tests. The pressure drop across the unit is calculated using a pressure balance which includes the gravity head, neglects the kinetic energy and inertial terms, and assumes that shock and friction losses in each region are proportional to mass flow squared and region length. The pressure gradient so calculated is, however, only used to provide a pressure boundary condition for the feedwater train. Secondary side flow is found by solution of the conservation of mass and energy equations assuming that the SG is at the outlet pressure. The density in the boiling region is calculated using the Zuber drift flux model with the drift flux parameters obtained by calibrating against the ARC tests. For recirculating units (OTSGs) the one-node downcomer model uses the conservation of mass, energy, and momentum equations plus the following assumptions:

1. The aspiration steam maintains the downcomer at saturation conditions.
2. There is a distinct saturated fluid level.
3. Spatial momentum and transient fluid acceleration pressure differentials are negligible.
4. The time rate of change of the saturated liquid and vapor properties (specifically density and enthalpy) are negligible.

Downcomer friction is lumped into the coefficient of the orifice between downcomer lower-end and tube bundle inlet. Secondary side steam generator water properties are obtained from polynomial expressions which upon spot checking in a limited operating range have been found, with the exception of the superheated steam density, to be accurate to well within a percent. The superheated steam density can be off by several percent.

3. Steam and Feedwater Loops

Each of the two steam generators has one steam line leading to the turbine header. Steam line and turbine-extraction flow enthalpies are assumed to remain constant with time. This is justified on the basis that steam generator outlet enthalpy varies little during the transients POWER TRAIN analyzes, and turbine-extraction-flow enthalpies have been measured to vary little with power level. In the steam line model the conservation of mass and momentum equations are solved to obtain pressures and flowrates, with kinetic energy changes overlooked in the latter equation. Flows through the safety relief, atmospheric dump, turbine control, and bypass valves are assumed to be proportional to the pressure at each valve and to valve-opening areas; wide-open flowrate vs. pressure data are vendor-supplied.

The turbine model includes the moisture separator and reheaters. Dynamics of the model are approximated by applying time constants to steady-state heat balance data. The turbine control valve time constant is hardwired with a value of 1/23 sec. The heat balance data, as functions of throttle flow, second stage reheater outlet flow, or low pressure turbine inlet flow, plus the time constants, are input. In the generator model, net mechanical steam torque, determined by turbine steam throughput, is compared to electrical load torque, determined from generator load angle (as affected by the difference between generator and grid frequencies). Minor torque contributions are also included. Generator frequency is calculated from net torque. Grid frequency can be stepped if desired.

The feedwater system for each steam generator is modeled as a single train. Constant local feedwater densities are assumed, so that flows and pressures can be calculated by the conservation of momentum equation alone in the lines. Feedwater pumps utilize homologous curves, with pump speed modeled as a second-order system fit to manufacturer's data for the pump controller. Feedwater control, as in the actual Integrated Control System, is a complex function of such things as power demand, reactor power, "BTU limits", difference in reactor loop temperatures, and steam generator level.

Emergency feedwater capability is provided, with pump-startup delay time and maximum flow as input. A proportional-integral controller model acts to maintain a constant steam generator level.

Condensate flows and pressures are calculated in the same way as for feedwater; the dividing line between the two systems is at the feedwater pumps. Drain tank flows are calculated assuming that tank levels are constant and that flowrate time derivatives are simple first-order lags.

High-pressure heaters are, if operating, assumed to provide proper-temperature feedwater. One node steady-state mass and energy balances are solved for each heater, and the transient performances of each heater, feedwater outlet enthalpies, and drain flowrates, are simple first-order lags.

4. Integrated Control System (ICS)

The POWER TRAIN model is controlled by a digital simulation of an actual ICS. The model includes the major subsystems (a) unit load demand, (b) integrated master control, (c) steam generator feedwater control, and (d) reactor control subsystems. Demand signals for the control rod drive, feedwater, and turbine pressure models are driven by a unit load demand signal established by the model operator, who can initiate many of the trips and abnormal conditions present in an actual plant. The operator can also select either manual or automatic control for many controllers and final control elements of an actual plant. In performing Section 15 accident analyses, no credit can be taken for the ICS, because it is not of safety grade.

On the basis of the above understanding the following evaluation can be made.

5. Specific Limitations

For the events listed at the beginning of this section, Summary of Regulatory Evaluation, POWER TRAIN is sometimes used to provide input, regarding heat removal from the Reactor Coolant System, to second and even third codes such as CADDS and RADAR. This is done because the added code(s) model aspects of the reactor coolant system, particularly the reactor, more accurately than POWER TRAIN.

In the case of ATWS the water properties need to be extended to higher pressures.

6. General Limitations

The scaling requirements of the analog portion of the simulation result in the following limits on the magnitude and rate-of-change of the variables specified below (if these limits are exceeded, the code results are not valid):

- Fluid temperatures: 350 to 650°F
- Cladding and fuel temperatures: 350 to 3000°F
- Primary system pressure: 1500 to 3000 psia
- Primary flow rate: 5 to 200%
- Reactor power level: 3 to 120%
- Secondary fluid enthalpy: 300 to 1400 Btu/lb
- Secondary fluid density: 0 to 60 lb/ft^3
- Secondary flow rate: 0 to 115%
- Secondary pressure: 500 to 1500 psia
- Secondary pressure maximum rate-of-change: 100 psia/s
- Secondary flow rate maximum rate-of-change: 100%/s

In addition to the limits above, limitations on the maximum rate of change of the enthalpy increase on the secondary side of the steam generator simulation will result in a higher than normal heat removal rate for zero secondary flow conditions. Therefore, when the steam generator is dry, caution should be exercised in the interpretation of code results. It should always be kept in mind that this version of POWER TRAIN was developed as a power range (15-100%) analysis tool and is not suitable for analysis of low power (<15%) operation (e.g., emergency feedwater control studies) despite the fact that it will operate at low power levels.

Several other code limitations apply and are listed below. These limitations are necessary because of mathematical model assumptions, scaling limitations on the analog portion of the simulation, and the time step employed.

a. The pressurizer cannot go solid or completely empty. If either condition occurs, the code results from that point are invalid.

b. Two-phase conditions in the primary system are not modeled.

c. Operation of the secondary system which results in the introduction of saturated fluid conditions in the steam lines is not valid as the steam is assumed to be always superheated.

d. The code is not capable of analyzing system piping breaks in the primary system.

e. No emergency safety system features (such as high-pressure injection) are included in the code capability.

f. It is not valid for analysis of fast primary system reactivity excursions (e.g., rod ejections) but should be able to calculate transients with ramp rates on the order of a few cents/second. Multidimensional neutronic space time effects cannot be simulated.

g. The code uses a uniform pressure on the primary side and therefore cannot be utilized to calculate localized voiding.

h. Imperfect mixing in the steam generator cannot be described by the one dimensional thermal hyddraulics.

i. There are no natural convection heat transfer correlations. Low flow conditions in the steam generator could be in error.

j. The steam generator downcomer model with its quasistatic aspirator cannot describe dynamic aspirator flow effects.

B. Code Qualification

In support of their code qualification work B&W has submitted some information on comparisons with a limited set of measured data. These are summarized in the next section. As part of this evaluation ANL has also made audit calculations, using the Midland and Bellafonte plant models with RELAP4/MOD6 for an inadvertent feedwater flow increase and for a turbine trip.

1. B&W Comparisons

B&W has compared POWER TRAIN results with data obtained at the B&W Alliance Research Center using a 19-tube steam generator for a 55-65 percent steam-flow step-up. The calculated peak steam flow was 62.0 percent after an elapsed 5 s, vs. 63.0 percent after an elapsed 2 s actual. The POWER TRAIN outlet pressure leveled out 52 psi lower than the initial pressure, vs. 44 psi lower actual. More recent comparisons were made for feedwater steps to produce 100-20%, 100-10%, and 100-0% steam flow and recovery; the agreement between POWER TRAIN and measured data was very good. It should however be noted that some calibration was performed.

POWER TRAIN results have also been compared to data from 177-FA operating plants for (a) a power ramp from 100 to 15% full power at 20% per minute, and (b) a reactor coolant pump trip at 90% power. General agreement is good; explanations for discrepancies are provided in the report.

2. ANL Audit Calculations

ANL performed the following computations with RELAP4/MOD6 in support of this review. In general, however, the audit calculations do not permit conclusive remarks regarding code models and methods due to lack of information regarding the modeling assumptions in the B&W computations being audited.

a. Feedwater Flow Increase for the Bellefonte Plant

In the ANL calculation, the feedwater flowrate was ramped from 15% nominal to 100% nominal in one steam generator with the reactor assumed to be operating at 15% power. The feedwater flow ramp rate was obtained from the Bellefonte FSAR. The ANL RELAP4/MOD6 computational results differed substantially from those obtained by B&W with POWER TRAIN; this difference is evidenced by the fact that the B&W calculation indicated that the reactor went to a new equilibrium at ~62% of full power while the RELAP4/MOD6 computation went to a new equilibrium at 24% of full power. This difference is believed to be due to simplistic but conservative modeling of the primary to secondary heat transfer in POWER TRAIN, since RELAP4/MOD6 indicates that the overfed steam generator has low output flow quality rather than pure steam.

b. Midland Analyses

i) Main Feedwater (MFW) Overfeed

ANL performed a series of analyses of a 15% overfeed of MFW at full power for comparison to analysis in the Midland FSAR. The ANL analyses resulted in achievement of a new steady state without reaching any of the stated trip setpoints, while the Midland POWER TRAIN analyses showed trip reached at roughly 29 seconds on high power. This difference is believed to be due to the quasistatic aspirator model in POWER TRAIN.

ii) Turbine Trip

Sufficient information regarding the analysis assumptions was not available, so ANL performed only one analysis which confirmed the qualitative statements in the FSAR that this is a relatively mild transient. This analysis does not permit any conclusive statements regarding the POWER TRAIN modeling since the analytical assumptions were not completely defined to permit confirmatory auditing.

III. Staff Position

We have reviewed the methods and assumptions described in BAW-10149, Rev. 1, and have concluded, subject to the following conditions, that the POWER TRAIN code is an acceptable method for transient analysis of the specific events listed above under Summary of Regulatory Evaluation. However future analyses with the code should be accompanied by detailed review of the specific application.

For all use of POWER TRAIN the above "General Limitations" and "Specific Limitations" shall not be violated.

Because of the assumption of constant primary-to-metal heat transfer coefficient in the steam generator, POWER TRAIN shall not be used whenever excessive primary-to-secondary heat transfer, at the lower flowrates POWER TRAIN limitations allow, since it would be non-conservative.

The following items which ANL regards as probable typographical errors have not been resolved.

Page 2-42, Eqn. 2-196

W_{gw} should be W_{fw}; "1" in brackets should be "L".

Open Issue: In the B&W response, "1" was not replaced by "L".

Page 2-7, Eqns. 2-14 to 2-17

The units do not balance. On the left sides the units are Btu/sec and on the right sides typically % of rated power/sec. On the right side of Eqn. 2-16, the left term is in % of rated power per second and the right term is in Btu/sec.

Open Issue: In the B&W response, MC_p units were incorrectly changed to "100% of rated power/(°F/s);" % of rated power/(°F/s)" would be acceptable.

B&W shall provide formal assurance that the program statements are accurate.

References

1. S. R. Fisher et al., "RELAP4/MOD6, A Computer Program for Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems, User's Manual," CDAP TR 003, EG&G Idaho, Inc. (January, 1978).

SUGGESTED TOPICAL REPORT EVALUATION ON CADDS, PROVIDED TO NRR BY ANL

Report No. and Title: BAW-10098P, Rev. 1, "CADDS - Computer Application to Direct Digital Simulation of Transients in PWRs With or Without Scram."

Originating Organization: Babcock & Wilcox Company

I. Summary of Topical Report

Report BAW-10098P, Rev. 1, describes the CADDS code used by Babcock & Wilcox to analyze reactor transients, with or without reactor scram, in a pressurized water reactor. CADDS solves the time-dependent neutron kinetics equations in conjunction with a thermal-hydraulic solution for an average fuel pin during a transient. The code incorporates the major feedback mechanisms while accounting for some of the details of single phase, nucleate boiling, and film boiling heat transfer in the reactor core. Two-phase (steam-water) flow is represented by a slip flow model and a slip ratio of one is used to represent homogeneous ("fog") flow. The pressure along the coolant channel is assumed constant for each time step.

For a simultaneous solution of the entire primary system, the user couples the core model with a one loop simulation of the reactor coolant loop (hot leg, steam generator, cold leg, and reactor core) and the pressurizer. A region-averaged model of each portion of the loop is utilized in determining the temperature response in the loop for either variable or constant flow; this, in turn, contributes to the pressurizer model. The heat transfer from the primary to the secondary system is determined by either of two methods: (1) solving a set of thermal-hydraulic equations on both sides of the steam generator or (2) the user may simply input the steam generator heat transfer to the secondary side as a function of time.

A reactor core can be evaluated independently (separate from the primary system) by using as input reactor inlet enthalpy, reactor inlet flow, system pressure, and either rod and boron reactivity or heat generation as functions of time. The use of the loop option requires that only reactor inlet flow versus time and rod and boron reactivity (or heat generation) versus time be input.

The neutron kinetics model incorporates reactivity feedback effects due to moderator density and temperature changes and Doppler broadening. The fuel pin is represented by a detailed axial and radial sectionalization to provide better thermal evaluation for the neutronic feedback calculation.

The ability for the CADDS user to set reactor trip points allows one to follow accidents to completion with core shutdown. Using additional program options, the user can (1) include decay heat and (2) ascertain the cladding oxidation front penetration distance calculated from a metal-water reaction based on the parabolic rate law.

Thus, the CADDS program could be used to study typical anticipated transients such as those caused by flow variation.

II. Summary of Regulatory Evaluation

In various publications, Babcock & Wilcox has stated that CADDS is used for transient analyses of the following events: (The numbers preceding the events refer to the Safety Analysis Report section in which the results of the transient analyses are reported.)

15.1.1 Decrease in Feedwater Temperature

15.1.2 Increase in Feedwater Flow

15.2.6 Loss of Nonemergency AC Power to the Station Auxiliaries

15.2.7 Loss of Normal Feedwater Flow

15.3.1 Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions

15.3.3 Reactor Coolant Pump Rotor Seizure

15.4.1 Uncontrolled Control Rod Assembly Withdrawal From a Subcritical or Low Power Startup Condition

15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power

15.4.3 Control Rod Misoperation (System Malfunction or Operator Error)

15.4.4 Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature

15.4.6 Chemical and Volume Control System Malfunction That Results in a Decrease in Boron Concentration in the Reactor Coolant (PWR)

15.4.8 Spectrum of Rod Ejection Accidents

15.8 Anticipated Transients Without Scram

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 a small series of audit calculations using the RELAP4/MOD6 computer code.[1] Our conclusions regarding the use of the CADDS code as described by BAW-10098P, Rev. 1, are stated in the Staff Position section of this report.

A. Review of Analytical Models

In the following sections, many statements made earlier with regard to CADD, predecessor code to CADDS, are repeated. The two codes are essentially identical except that CADDS has secondary-side (S) modeling capability lacking in CADD.

CADDS is a one-loop fixed-nodalization code (with the exception of the core and the steam generator where variable axial nodalization capability is available) modeling a reactor core with a bypass, a steam generator, one hot leg and one cold leg node, and a nonequilibrium two-region pressurizer on the hot leg. At B&W's request, this review was restricted to the multinode Integral Economizer Once Through Steam Generator (IEOTSG) option and the OTSG model was not evaluated. Although CADDS has a pump model, B&W also requested that it not be reviewed. CADDS has no boron transport model, ECCS is not included, no provision was made for a hot channel DNB calculation, there are no bubble rise models, and except for the pressurizer there are no heat slabs so metal heat transfer cannot be accounted for. Critical flow is treated by using the HEM model or through input, and CADDS has a transport delay model for the piping.

The code has a partial initializer with the pressurizer initialized at saturation given the initial pressure and mass of water while the steam generator initialization depends upon the nodal option chosen. For the single node option, a tube side heat transfer coefficient is computed using initial heat load and average tube metal/coolant temperatures. In the case of the multinodal model, the primary side pressure, flow, inlet and outlet enthalpy, and the secondary side pressure, feedwater flow rate and enthalpy are input by the user. To obtain the thermal steady state the code adjusts the feedwater flowrate. The secondary side pressure distribution is calculated by solving the steady state momentum equation with input form loss coefficients. Input loss coefficients are also used for the other parts of the system where a pressure drop is involved. Since B&W requested that the review be restricted to the IEOTSG, the aspirator flow initialization is not discussed here.

A detailed evaluation of the major models in CADDS follows.

1. Core

a. Neutronics Model

In addition to a power input table option, CADDS provides a kinetics option using the well-known point kinetics equations. The code permits up to twelve delayed neutron groups, and reactivity feedback due to moderator temperature changes takes into account the axial variation of the temperature by the use of weighting coefficients. The reactivity feedback due to moderator density changes also accounts for axial density variations and differences from the average or hot fuel pin. The reactivity feedback due to fuel temperature changes (Doppler) accounts for both the axial and radial variation of the temperature of a fuel pin by using weighting procedures. The reactivity feedback model also features trip reactivities and reactivity insertions which are input by the user in tabular form as a function of time. No provision is made for the voids produced by subcooled boiling and the concomitant reactivity effect is therefore not modeled.

The method of solution of the CADDS point kinetics equations is conventional in that it is based upon the Westinghouse CHIC-KIN program. A procedure is used to control the time step to ensure the stability of the equations. However, the accuracy of the solution may be reduced if the input minimum time step is too large.

The point kinetics model used by CADDS is satisfactory with respect to the equations, method of solution, and reactivity feedback models.

b. Decay Heat Model

CADDS has, as an option, the inclusion of a six group decay heat model to the total power. The decay heat is based on infinite operation at the initial power of the reactivity transient. The equations used in this calculation are similar in form to those used for computing the delayed neutron precursors. The constants used in the decay heat computation are listed in Table B-1 in the report. The formalism used for the decay heat model is satisfactory. The Table B-1 constants are the same as those used for the CADD code, which has already been accepted.

c. Metal-Water Reaction

The Baker-Just equation is used for calculating metal-water reaction rates. Energy generated by the reaction is added to the outermost cladding node, after energy required to heat the vapor to the reaction temperature has first been subtracted. Cladding oxidation penetration due to the reaction is tracked.

d. Heat Transfer Model

The heat generation model assumes that the heat source is a separable function of space and time. The spatial variation, radial and axial, of the heat source can be input to CADDS and is invariant in time. The time dependent variation of the amplitude of the heat source can either be input to CADDS or computed from the point kinetics equations. Fixed fractions of the generated heat are deposited directly into the coolant and into the cladding. Multinode radial heat conduction through the fuel pin with an explicit gap conductance is considered but axial heat conduction is not. The material properties used in this calculation in general compare well with the data presented in the MATPRO[2] and NSM handbooks.[3] The UO_2 thermal conductivity is predicted to within a few percent of the data base up to 1500K. Beyond that temperature it is overpredicted by as much as twenty five percent but better agreement is obtained for higher porosity fuel. For the UO_2 specific heat, the error is a few percent up to the melting point, while the Zircaloy-4 thermal conductivity is within the two standard deviation limits of the least square fit given in Reference 2. The Zircaloy-4 specific heat agrees very well with the data up to 1300K and beyond that in the β-phase region is exactly that of the MATPRO model. For the Type 304 SS both the thermal conductivity and specific heat in the temperature range 0-900°C are predicted to within a few percent of the values given in the NSM handbook. The appropriate conditions associated with the heat transfer regions of conduction or convection (Colburn), nucleate boiling (Thom) or film boiling (Quinn's modified Sieder-Tate or Groeneveld) are applied at the surface of the cladding. Transition boiling is not considered. Superheat (Colburn) is considered as an extension of the film boiling region. Switching criteria for the various heat transfer regimes are based on the critical wall temperature for nucleate boiling, and the W-3 or B&W-2 critical heat flux correlations in conjunction with the GE CHF correlation for film boiling which are standard options. However, heat transfer coefficients can also be input by the user as a function of time. For each time step, the total pressure is assumed to be constant along each flow channel and equal to the reactor inlet pressure, unless the conservation of momentum option is used (see next section).

The CADDS program does not consider dimensional variations in the gap between the fuel pellets and cladding in the heat transfer expressions for the fuel pin. Rather, B&W reported that they input a constant but conservative value of gap conductance.

e. Thermal Hydraulics

Thermal-hydraulic conditions are calculated by simultaneous solution of the HEM equations of mass and energy conservation with the assumption of uniform pressure, as restricted to one-dimensional constant-area flow passage applications. Assumption of a single pressure removes the momentum equation, and implies that pump energy will not be properly accounted for in the energy equation.

When in single phase the flow is treated as incompressible. The slip option model is based on Thom data. The core bypass flow fraction is held constant during the transient. This constant fraction assumption is invalid whenever significant vapor is produced in the core or whenever transition to laminar flow is out of phase between core and bypass. Thermodynamic and transport fluid properties are those approved for CADD.

2. Reactor Loop

This single-loop model includes volumes to model the hot leg, steam generator primary side, cold leg, reactor core, and core bypass. Inlet, outlet, and average conditions are maintained for each volume so that transport delays and specific volumes may be determined. The specific volumes are used in determining surge flow to and from the pressurizer. Except for the pressurizer and local core channels, all volumes are assumed to be in the subcooled (single-phase) condition. The coolant flowrate is held constant around the loop at the value of the reactor-inlet flowrate during each time step. The surge line flow is calculated simultaneously with the system pressure, taking into account the surge line quasistatic pressure drop and is based on the system expansion or contraction assuming that the pressure time derivative is spatially independent. The pressurizer pressure is computed in tandem.

Card No. 10226, page 6-15, requires that the surge flowrate be reduced by the spray flowrate, but is stated to be optional. The code user should be directed that this option should be used whenever the spray option is used.

Critical flow in the surge line to the pressurizer is not considered because critical flow never occurs in the surge line for the transients analyzed with CADDS.

There are two steam generator models, single-node and multi-node. In the single-node model, two differential equations describing the average primary-side coolant enthalpy and the average tube temperature are solved. Heat demand on the secondary side must be input as a function of time; this heat demand is applied to the tube-temperature equation.

In the multi-node IEOTSG model, the one-dimensional conservation of mass, energy, and (secondary only) momentum, and fluid state equations are used on both primary and secondary sides. The steam outlet pressure on the secondary side must be input as a function of time.

Tube pressure drop is neglected with the result that the entire primary system, aside from the surge line and the pressurizer, is at a uniform pressure. The primary side flow is treated as incompressible. On the two phase shell side, thermal equilibrium is assumed and the same core slip correlation is used. Two phase friction factor multipliers appear to be Martinelli-Nelson. The single phase friction factor used is not recommended for low flow situations. Heat transfer across the tube is computed using a three node radial conduction approximation. On the primary side the regime is subcooled forced convection (Colburn) while on the secondary side the regimes are subcooled forced (Colburn), nucleate boiling (Thom), transition boiling (McDonough-Milich-King), stable film boiling (Hao-Morgan-Parker-Howard), and superheated forced convection (Hao, et al.). Insufficient data has been provided to justify the use of the Hao correlation. Switching criteria between the various flow regimes are, critical wall temperature and the following CHF correlations: BAWL for low flow and the Griffith countercurrent flow CHF for extremely low flow conditions. The switch between the stable film and the transition boiling regimes is accomplished by using the maximum heat flux.

In the nonequilibrium pressurizer model, the energy and mass conservation laws are solved for a 100% liquid and a 100% vapor region. Insurge flow is assumed to instantaneously equilibrate with the liquid region thus eliminating any potential fluid stratification. Included in the energy and mass balances are the energy transferred to the pressurizer walls, the energy transferred across the steam-liquid interface, condensation, and evaporation. The forcing function is the surge rate with safety valves, heaters, and sprays to hold the transient within acceptable limits. The program permits the pressurizer to completely fill with liquid and the liquid to escape out the safety valves. The heater model assigns a time constant to heater surface thermal response, while the spray flow is obtained from a table input activated by pressure. Since spray flow varies with time and is dependent upon pressure, for their initial run a constant value is used as the time behavior is otherwise unknown. B&W claims that experiments show the sprays are designed so that spray droplets reach saturation within a couple of feet. The safety valve model also includes a relief line pressure drop with an input loss coefficient. For steam relief, however, the flow is obtained from a table input. Constant film coefficients between pressurizer vapor and liquid and the vessel wall are assumed. The code has logic to prevent negative rates of either evaporation or condensation in the pressurizer. Factor-of-four damping factors are included that delay consequences for four time steps. However, the time steps are shorter than pressure-oscillation frequencies.

On the basis of the above understanding of the code, the following evaluation is made.

3. Specific Limitations

a. 15.1.1, 15.1.2, 15.2.6, and 15.2.7 Decrease in Feedwater Temperature, Increase in Feedwater Flow, Loss of Non-emergency AC Power, and Loss of Normal Feedwater Flow

For these events, there are two ways in which CADDS can be used:

(1) When the single-node steam generator model is employed, the heat demand is input as a function of time. (The input is obtained by modeling with a system code such as POWER TRAIN, or using a conservative approximation of the heat removal rate as a function of time.)

(2) If an IEOTSG plant is being analyzed, the CADDS multi-node steam generator model can be used, provided the steam outlet pressure on the secondary side is input as a function of time. (This pressure vs. time data is similarly obtained from modeling with a code such as POWER TRAIN or by using a conservative approximation.)

When POWER TRAIN or the like is used to provide CADDS input, it is probable that the CADDS calculations will produce different steam generator primary inlet conditions than the POWER TRAIN calculations. If the discrepancies are conservative, this is acceptable; otherwise iteration is necessary to remove the discrepancies.

b. 15.3.1 and 15.3.3 Loss of Forced Reactor Coolant Flow, and Reactor Coolant Pump Rotor Seizure

Because CADDS has no pump model, a code such as PUMP is normally used to provide CADDS with reactor flowrate following a loss of pumping power. Also, because CADDS does not consider axial core pressure gradients for core heat transfer calculations, a code such as RADAR is normally used for calculating DNBR during these events, using CADDS-calculated input.

The combination of CADDS with a code such as PUMP cannot demonstrate that a reactor is capable of being properly cooled by natural circulation after loss of all forced flow. This is because neither PUMP nor CADDS calculates either system "thermal head" or the coolant flow that this head induces.

c. 15.4.1, 15.4.2, 15.4.3, and 15.4.8 Uncontrolled Control Rod Assembly Withdrawal (at Startup) Uncontrolled Control Rod Assembly Withdrawal at Power, Control Rod Misoperation, and Spectrum of Rod Ejection Accidents

The CADDS neutron point-kinetics approach for handling this event is acceptable, subject to meeting the Standard Review Plan requirements for conservatism regarding power distribution and reactivity-coefficient weighting factors, etc.

d. 15.4.4 Startup of an Inactive Loop

With the CADDS single-loop capability it is possible to analyze the situation in which, with one pump running in each of two loops initially, the second pumps in each of the two loops are started simultaneously. However, CADDS cannot treat asymmetric pump startup events, in which pumping power is different in each of two loops, since CADDS represents the entire plant with one loop. Nevertheless, since single loop operation is not allowed, and since the other possible asymmetric situations involve smaller flow increases, temperature decreases, and reactivity increases than the symmetric case of simultaneously starting one pump in each loop, when one pump is already operating in each loop, CADDS is sufficient for analyzing this event because it will provide conservative results.

e. 15.4.6 Chemical and Volume Control System Malfunctions

BAW-10098P, Rev. 1, does not describe how this event can be analyzed using CADDS. Therefore, whenever CADDS is specified for analyzing this event, details of how the analysis was performed must be provided.

f. 15.8 Anticipated Transients Without Scram (for above listed events)

Whenever CADDS is used for analyzing any of the foregoing events, but without allowance for scram, the foregoing comments for each event remain applicable.

4. General Limitations

With respect to the above-listed events, the following general conditions apply to use of CADDS:

a. Because CADDS requires the primary loop (except for the reactor core and pressurizer) to be in the subcooled condition, CADDS should not be used whenever the hot leg temperature exceeds the saturation temperature. Furthermore, due to the assumption of a constant bypass fraction, CADDS should not be used for cases having two phase flow in the core without further specific justification.

b. As discussed above under 3b, CADDS does not consider axial core pressure gradients for core heat transfer calculations; therefore, for all events in which minimum DNBR is an issue, a code such as RADAR must be combined with CADDS for calculating DNBR.

c. Because CADDS has only single-loop capability, it cannot calculate reactor temperature maldistributions that occur whenever one of the actual two loops is asymmetric with the other. Examples of when asymmetry occurs are: decrease in feedwater temperature to only one steam generator (15.1.1), increase in feedwater flow to only one steam generator (15.1.2), loss of feedwater flow to only one steam generator (15.2.7), coolant flowrate variations different in one loop than in the other (15.3.1, 15.3.3, and 15.4.4), dilution/injection rates different in one loop than in the other (15.4.6). A different method must be used to show that such maldistributions are acceptable (e.g., bounded by acceptable symmetric events).

d. Reverse flow situations cannot be analyzed with CADDS, due to specific mathematical features in the code.

e. NYPFLO Option 3 (stabilized flow equations with numerical weighting factor) has not been demonstrated to be reliable. B&W has agreed not to use this option.

f. Natural convection flow cannot be calculated with CADDS.

g. Multidimensional neutronic space time effects cannot be simulated. Conservative usage of the point kinetics will have to be demonstrated in such cases where the space time effects are significant.

h. The code uses uniform pressure on the primary side and therefore cannot be utilized to calculate localized voiding.

i. Imperfect flow mixing cannot be analyzed by the one-dimensional thermal hydraulics.

j. Insufficient data has been provided for justifying the use of the Hao correlation in post CHF situations on the secondary side.

B. Code Qualification

In support of their code qualification work B&W has performed sensitivity studies with CADDS and submitted some information on comparisons with (limited) measured data. These are summarized in the next section. As part of this evaluation ANL has made audit calculations using the Bellefonte plant model with RELAP4/MOD6 for Loss of Feedwater, Control Rod Withdrawal, and Loss of Forced Reactor Coolant Flow transients.

1. B&W Studies

B&W has examined the effect of fuel-cladding gap conductance on peak system pressure, during a transient involving loss of feedwater without scram. Peak pressure increased by about 170 psi when gap conductance (constant during a transient) was increased from -30 to +20 percent of the "best" (steady-state) value; this increase in peak pressure with assumed gap conductance was essentially linear.

The effect of time step size on accuracy of kinetics solutions, at 0.5 s. after initiation of a 0.01 $\Delta k/k$-s. ramp rate, was calculated using four different methods including the CADDS method. All methods gave the same neutron density (to eight figures) when time steps of 10^{-7} s were used. With the CADDS method, one percent accuracy (the maximum considered acceptable by B&W) was obtained using time steps of 3×10^{-5} s.

The effect of heat transfer coefficient between liquid and vapor in the pressurizer was investigated. It was found that as long as the coefficient was in a reasonable range, no significant effect on results occurred.

CADDS results have been compared with measured data obtained after a turbine trip at 72% power at Oconee Unit I and after a unit generator trip at 96% power at Three Mile Island Unit I. Both transients resulted in a high reactor coolant pressure trip. In the Oconee case, CADDS calculated a peak reactor power level of 75% while the measured value was higher, 82%. However, CADDS calculated higher peak reactor coolant pressure and temperature. In the Three Mile Island case, CADDS calculated all three parameters to be very close to the measured data.

2. ANL Audit Calculations

(a) Loss of Feedwater

ANL obtained similar results to those shown by B&W in the Bellefonte FSAR inasmuch as the series of ANL RELAP4/MOD6 calculations obtained peak pressures within 10-20 psi of those reported by B&W and reactor trip times within 1 second of those by B&W. However, since complete details on the B&W modeling assumptions were not available, ANL is unable to reproduce the B&W computational assumptions and therefore cannot make conclusive remarks regarding the code validity.

(b) Control Rod Withdrawal

ANL analyzed the transient generated by the withdrawal of the most reactive single control rod group. ANL's series of calculations yielded much faster rates of flux increase than those reported by B&W and therefore reached the flux trip setpoint much earlier (the RELAP4/MOD6 calculations typically tripped at roughly 7.5 s while the B&W tripped at approximately 9 s). Therefore, the peak pressures obtained by ANL were substantially lower than that of B&W. Once again, however, since the detailed analysis assumptions were not available, ANL is unable to make conclusive remarks regarding code validity.

(c) Loss of Forced Reactor Coolant Flow

ANL analyzed a four pump coastdown transient for Bellefonte such as due to loss of offsite power (reactor scram at transient initiation). ANL obtained similar flow vs time and power to flow vs time behavior. This audit calculation provides little assurance regarding the validity of any models in CADDS other than the decay heat curve and the pump head/flow decay curve. As the FSAR calculation did not provide sufficient information regarding natural circulation conditions ANL did not audit that portion of the transient.

(d) Overcooling Transients

B&W intends to perform overcooling transient analysis with CADDS. ANL has not audited the code for that application.

III. Staff Position

We have reviewed the methods and assumptions described in BAW-10098P, Rev. 1, and have concluded, subject to the following conditions, that the CADDS code contains acceptable models and methods for transient analysis of the specific events listed above under Summary of Regulatory Evaluation. However, future analyses with the code should be accompanied by detailed review of the specific application.

B&W asked that we not review the CADDS multi-node steam generator option except for Integral Economizer Once Through (IEOTSG) units; therefore, the multi-node option should not be used to analyze ordinary OTSG units without further justification. Furthermore, even with IEOTSG units, the CADDS requirement for input of secondary side steam outlet pressure vs. time makes interaction with a system code such as POWER TRAIN necessary. Such interaction must be performed iteratively to obtain identical behavior of common parameters between CADDS and the system code calculations, unless discrepancies between code results are demonstrated to lead to conservative conclusions. Also, insufficient data has been provided to justify the use of the Mao correlation in post CHF conditions on the secondary side.

CADDS does not consider axial pressure gradients necessary for natural convection flow calculations. Also, the CADDS pump model was not reviewed. Therefore, for analyzing Loss of Forced Reactor Coolant Flow (15.3.1) and similar transient-flowrate events, input must be provided.

CADDS contains only a simple neutron point kinetics approach. Thus, when CADDS is used for analyzing Uncontrolled Control Rod Assembly Withdrawal at Startup (15.4.1), at Power (15.4.2), Control Rod Misoperation (15.4.3), and Spectrum of Rod Ejection Accidents (15.4.8), evidence must be provided that the Standard Review Plan requirements for conservatism regarding power distribution and reactivity-coefficient weighting factors, etc. are met.

Since BAW-10098P, Rev. 1, does not discuss boron dilution, when CADDS is used for analyzing Chemical and Volume Control System Malfunction (15.4.6), justification must be provided for the boron reactivity insertion rate used.

CADDS assumes subcooled conditions in the primary loop. Therefore, CADDS may not be used whenever the hot leg temperature exceeds the saturation temperature. Because CADDS assumes a constant ratio between core and bypass flowrates, CADDS may not be used whenever significant vapor is generated in the core or if either the core or bypass passes through a transition to laminar flow.

Whenever asymmetric conditions (different conditions in one primary loop than in the other) can occur with respect to any event CADDS is used to analyze, a different method must be used to show that consequences with asymmetric conditions are less severe than with the symmetric conditions of CADDS.

CADDS does not consider axial core pressure gradients for core heat transfer calculations; therefore a code such as RADAR must be used for calculating DNBR, for all events in which minimum DNBR is an issue.

Since CADDS is not designed to handle reversed flow situations, it may not be used whenever reversed flow occurs.

Because the reliability of NYPFLO Option 3 has not been demonstrated, it may not be used. The NYPFLO options, 1 and 2, are acceptable.

Card No. 10226, which requires that the surge flowrate be reduced by the spray flowrate must be used whenever the spray option is used.

When CADDS is used for Anticipated Transients Without Scram (15.8), all relevant conditions listed above are applicable.

The following items which ANL regards as probable typographical errors have not been resolved.

Page 2-2, Eqn. 2-2

The heat/work conversion factor J is used inconsistently throughout the report. The main-text nomenclature, Appendix A, states that the units of J are 0.185 (in.^{2}-Btu)/(ft^3-lb$_F$). If this were true, J would be in the numerator, not denominator, in Eqn. 2-2 and also in Eqn. 2-5 (page 2-5), and Eqn. 4-30 (page 4-8).

The Appendix C nomenclature, page "C-17", states that the units of J are ft-lbf/Btu. If this were true, J would be in the denominator, not numerator, in several equations on both pages C-6 and C-9.

Open Issue: In the eqn. for E_{SG} uder Eqn. 4-53, page 4-12, the right hand term should be $P^{N}_{sys} v_{CL}/J$.

Page 4-12, Eqns. 4-53 and 4-54

The HL subscripts should be CL (4 places).

Open Issue: Same as above open issue.

Page 5-3

The symbol Q is used inconsistently. Here and in Eqns. 4-57 to 4-59 it is energy per time interval. Elsewhere (e.g., Eqn. 3-7) it is energy per unit time.

Open Issue: The line preceding Eqn. 5-4, plus Q descriptions on page 5-3, continue to say "energy" instead of "energy rate" (e.g., Btu/cm^2 instead of Btu/cm^2-sec). Also, it is no longer clear in revised Eqn. 5-4 that the lb/mg multiplier is indeed a multiplier and not a divisor.

Page C-4, section 8

In the ΔP_{sj} equation, the G terms should be squared as at bottom of page C-13.

Open Issue: The B&W response introduced a new error: ρ should be $\bar{\rho}$ to be consistent with the definition at the top of the following page, C5.

Page 4-23, Eqns. 4-87 and 4-88

The L/L_0 and S/S_0 factors should be switched between the equations, for consistency with Eqns. 4-73, 4-75, 4-91, and 4-92.

Open Issue: B&W changed Eqns. 4-91 and 4-92 instead of Eqns. 4-87 and 4-88. As a result, Eqns. 4-87, -88, -91, and -92 are now inconsistent with Eqns. 4-73 and 4-75 which, since they appear first, should presumably set the nomenclature standard.

Page 4-28, Eqn. 4-120

The E_L^* on the right side comes from Eqn. 4-101, which does not have w_4 and w_5 terms in it. Therefore, the right-hand correction term of Eqn. 4-120 is incorrect. Explain.

Open Issue: In the B&W response the signs on the evaporation and condensation rate terms remain wrong in that they show evaporation to increase, and condensation to decrease, internal energy of the pressurizer liquid. (Correct signs are indicated in earlier Eqn. 4-63.)

Page 4-28, Eqn. 4-126

Δv_L and Δv_S are not defined. Verify that they are $(\Delta v_{fg})_{\text{liquid } T,P}$ and $(\Delta v_{fg})_{\text{steam } T,P}$.

Open Issue: Because of the other three open issues regarding the pressurizer model, which is extremely important because it is used for calibrating the POWER TRAIN pressurizer model, we reviewed the B&W response concerning Eqn. 4-126 and found we are unable to verify the derivation of Eqns. 4-124 to 4-126.

Page 4-26, Eqns. 4-105 and 4-106

Provide more-detailed derivations.

Open Issue: B&W has agreed that Eqn. B.11 in the derivation they provided has the wrong sign on the pressurizer evaporation rate. However, they have not agreed formally with us that this error carried into Eqn. 4-106 of the report. Also, we have found that both derivation Eqn. B.8 and corresponding report Eqn. 4-114 have sign errors resulting from using the wrong sign on pressurizer condensation rate.

References

1. "RELAP4/MOD6 A Computer Program for Transient Thermal Hydraulic Analysis of Nuclear Reactors and Related Systems - Users Manual" CDAP TR 003, EG&G Idaho, Inc. (January 1978).

2. D. L. Hagrman, G. A. Geymann and R. E. Mason, "MATPRO-Version 11 (Revision 1) A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior," NUREG/CR-0497 TREE-1280, Rev. 1, R3 and R4, EG&G Idaho Inc. (February, 1980).

3. J. C. Spanner, et al., "Nuclear System Materials Handbook," Vol. 1, Design Data, TID 2666, Hanford Engineering Development Laboratory (June, 1978).

SUGGESTED TOPICAL REPORT EVALUATION ON TRAP2, PROVIDED TO NRR BY ANL

Report No. and Title: BAW-10128, "TRAP2, FORTRAN Program for Digital Simulation of the Transient Behavior of the Once-Through Steam Generator and Associated Reactor Coolant System" plus steady state aspiration model, valve model, and non equilibrium pressurizer model described in 9 July 82 letter from J. H. Taylor (B&W) to W. L. Jensen (NRC).

Originating Organization: Babcock & Wilcox Co.

I. Summary of Topical Report

Report BAW-10128 describes the TRAP2 code used by Babcock & Wilcox to predict transient thermal-hydraulic behavior of a Babcock & Wilcox nuclear power plant during postulated Chapter 15 events. As is stated in the report, TRAP2 is an extension of the CRAFT1 primary system loss-of-coolant accident analysis code (its successor CRAFT2 has already been accepted) through the addition of a detailed steam generator model and by provision for substantial representation of secondary steam and feedwater piping. TRAP2 requires a user input control volume/flow path network model of both primary and secondary system elements identical in type to that utilized in CRAFT primary coolant system studies. The method and logic of CRAFT are applied to solve the equations of conservation of mass, energy, and momentum in both primary and secondary networks. TRAP2 relies upon CRAFT2 pressure and property search subroutines to determine the fluid state throughout the system.

In TRAP2, each steam generator is represented as N primary, secondary, and tube metal volumes. These control volumes are determined by dividing the steam generator into segments. User input to TRAP2 should be based on a detailed steady-state analysis to establish initial control volume pressures, enthalpies, heat transfer coefficients, and tube metal temperatures.

II. Summary of Regulatory Evaluation

In various publications, Babcock & Wilcox (B&W) has stated that TRAP2 is used for transient analysis of the following events: (The numbers preceding the events refer to the Safety Analysis Report section in which the results of the transient analyses are reported.)

- 15.1.2 Increase in Feedwater Flow (in conjunction with turbine trip/reactor trip)
- 15.1.5 Steam System Piping Failures Inside and Outside of Containment (PWR)
- 15.2.5 Steam Pressure Regulator Failure (Closed)
- 15.2.6 Loss of Nonemergency AC Power to the Station Auxiliaries
- 15.2.7 Loss of Normal Feedwater Flow
- 15.2.8 Feedwater System Pipe Breaks Inside and Outside Containment (PWR)
- 15.6.3 Steam Generator Tube Rupture
- 15.8 Anticipated Transients Without Scram
- General - Non-symmetric Secondary and/or Primary System Transients
- General - Events Leading to Two-Phase Conditions

In our evaluation we have applied the acceptance criteria presented in the above-numbered 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 RELAP4/MOD6 code.[1] Our conclusions regarding the use of the TRAP2 code as described by BAW-10128 are stated in the Staff Position section of this report.

A. Review of Analytical Models

TRAP2 is a variable nodalization code similar to those of the RELAP series. Plant models are built from arbitrary control volumes and flow paths. While TRAP2 does not have a general heat slab with two sided heat transfer, there are heat slabs for use specifically with the steam generator tubes and each control volume has a one sided heat transfer slab associated with it to model metal heat transfer. UA for these slabs is obtained either through input or is assumed proportional to volume mixture height. Phase separation can be simulated by the use of an equilibrium bubble rise model which assumes a uniform distribution of bubbles in the mixture region. B&W has indicated that the Redfield bubble rise velocity will not be used so the separation velocity is obtained through input. In the steam generator special provision is made so that layering will not occur by the transfer of liquid/vapor mass between upper and lower control volumes.

The code has a two-region non-equilibrium pressurizer model and the surge line is checked for critical flow. The ECCS (Emergency Core Cooling System) is simulated by flood tanks and fill flow tables (to model the high and low pressure injection system) where the tables are flow versus pressure. The flood tank model utilizes the ideal gas law and a general momentum equation. Control system trips on time, pressure, and level with time delays are available for reactor scram. Power trips appear not to be available.

Valves are modelled using tables of flow area while critical flow is calculated either using the orifice equation or the Moody correlation with a user specified discharge coefficient.

There is a feedwater path model and a boron transport model using donor cell techniques but no general transport delay model.

There is a single-phase pump model which uses four quadrant homologous curves and solves the pump speed equation accounting for windage/

bearing losses (manufacturer data) when coasting down. Electrical torque-speed curves are input.

The code initialization algorithm computes loss coefficients when the initial pressure distribution and flow rates are input. Steam generator initialization requires heat transfer rates at each volume to solve for the initial enthalpies. Initial tube metal temperatures or initial primary side heat transfer coefficients can then be calculated. Initial secondary side heat transfer coefficients are then inferred. Phase stratification is precluded during the initialization. Normalization of the transient heat transfer coefficients to steady state values is carried out through the use of multipliers. When the heat transfer regime changes the multiplier is switched to a value of 1. Upon reversion to the original steady state heat transfer regime the multiplier is switched back to the original value.

A detailed review of the major models in TRAP-2 is presented next.

1. <u>Core</u>

a. <u>Neutronics Model</u>

The well-known point kinetics equations are used, with up to six delayed neutron precursor groups allowed. Reactivity feedbacks from weighted channel-averaged fuel temperatures (Doppler broadening, proportional to the square root of the absolute temperature) and moderator densities are permitted. Reactivity contributions from control rod movement (input) and boron concentration (as calculated from input fill flow and the boron transport model) are included. There is no reactivity feedback from voids produced by subcooled boiling. The neutronics model used by TRAP2 is satisfactory with respect to the equations, method of solution, and reactivity feedback models.

b. Decay Heat Model

Coupled to the kinetics equations are decay heat equations that are similar in form to those used for computing the delayed neutron precursors. Up to 11 fission product groups can be specified. The formalism used for the decay heat model is satisfactory.

A power-specified option is available for which a table of normalized power (representing both fission and decay heat) vs. time is input. During a calculation a switch can be made from the kinetics model to the power-specified model.

c. Metal Water Reaction

The Baker and Just equation is used for calculating metal-water reaction rates. Rates are limited according to the amount of steam available for reaction. At temperatures below which the rates are negligibly small, the reaction-rate calculations are bypassed.

d. Heat Transfer Model

As stated in the report, control volumes may be provided to describe segments of the reactor core. Connecting these volumes are flow paths describing the hydraulic characteristics of the core and which also include an enthalpy transport model. In general, control volumes may have more than one inlet and outlet. Core flow path exit enthalpies are calculated from an energy balance that considers heat flux, inlet enthalpy, and flow path time constant. The core heat transfer is evaluated using the flow path temperature which therefore maximizes the transfer rate. Associated with each reactor core flow path is a corresponding section of the fuel pins. From the fuel pins, heat is transferred into the core water from the energy generated by fission and fission product decay. One dimensional radial heat conduction equations for fuel and cladding are solved, where each material is modeled as a single node in each flow path. For the heat source, one of the above-described power-specified and neutron-kinetics options is used. All of the heat is deposited in the fuel. The fuel-cladding heat conductance remains constant through each transient, whereas the cladding-water conductance is based on forced convection (proportional to flow raised to the 0.8 power), film boiling (Quinn's modified Seider-Tate with a minimum modifier quality of 0.3), or superheat modes (Seider-Tate), whichever is appropriate. The steam to liquid heat transfer coefficient is an input constant. If nucleate boiling occurs, the cladding surface temperature is calculated using the well-known Jens-Lottes equation. Switching criteria for the various heat transfer regimes are critical wall temperature for nucleate boiling and W-3 in conjunction with the Jansen-Levy correlation and a zero CHF for qualities >45% for film boiling. These are standard options. The user can also input a DNB heat flux.

e. Thermal Hydraulics

TRAP2 uses the one dimensional HEM equations for the fluid thermal hydraulics solving the mass and energy equations in control volumes and the momentum equation in flow paths. The one dimensional momentum equations include terms for inertia, friction, acceleration and gravity. Two phase friction factors used are the Martinelli-Nelson. Thermodynamic and transport fluid properties are obtained using the same tables already accepted for CRAFT.

In general, inlet fluid properties of a flow path are determined from upstream control volume mixture properties.

2. Reactor Loop

In the flow paths, the coolant circulating pumps are represented as active path elements. In pump flow paths, the time rate of change of momentum is related to the total pressure difference across the pump.

Three pressurizer alternatives have been used by B&W. First, a single control volume may be used that is treated the same way as any other TRAP2 node. This approach results in an equilibrium model that is used only in overcooling transients when the surge line flow is out of the pressurizer. Second, the pressurizer is divided into three vertical nodes of equal height; the bottom node is subcooled liquid, the top node is superheated steam, and the middle node is about half and half saturated liquid and steam at equilibrium. This option approximates a non-equilibrium pressurizer for overheating transients.

The third pressurizer alternative uses a two region non-equilibrium model option similar to that used in the CADDS and POWER TRAIN codes.

There is a bubble rise model in the 'liquid region' and a droplet model in the 'steam region.' The two regions simulated are a two phase mixture or a subcooled region and a stratified steam region. There is both interface mass and energy transport. Conservation of mass and energy are maintained using a specialized flowpath for the surge line which assumes that form losses dominate friction losses. Relief valve flow is calculated using the HEM isentropic expansion model or input tables. Spray flow is based on quasi-static pressure drop or is input with heat transfer to spray droplets explicitly modelled. The heater model uses a time constant approach.

Two leak-flow options are available, the first using an orifice equation for when leak flow is too rapid for flashing to occur. The second option provides the well-known Moody equation for the choked flow that accompanies flashing.

With TRAP2 the steam generators are modeled using fluid control volumes and flow paths. In addition, tube conduction is solved for, using a one node approximation with the one dimensional radial heat conduction equation. Forced convection, subcooled (proportional to flow raised to the 0.8 power) and superheated coolant (Dittus Boelter), and boiling (pressure fit of the Rohsenow pool boiling correlation) heat transfer modes are used as appropriate to local tube-wall and coolant temperature conditions. The use of a pool boiling correlation on the primary side for a tube forced convection situation may not be the optimum choice.

3. Secondary Loop

On the secondary side of the steam generator the same heat transfer regimes/coefficients are used as on the primary. Stratification effects on heat transfer are accounted for. One or two feedwater paths can be connected to one or two, respectively, steam generator secondary control volumes. The feedwater flowrate is determined using the one-dimensional conservation of momentum equation that considers the feedwater pump head and suction pressure. The latter plus the suction enthalpy are input vs. time. The feedwater pump speed is also input as a function of time.

Steady state aspiration is modeled with a user-specified flow path. A constant aspiration flowrate is user specified and applicable until an input time is reached when the aspiration flowrate becomes determined by momentum equation solution. The switch to use of the momentum equation can be instantaneous, or the aspiration flow can be ramped to zero before the momentum equation is used. The average mass flowrate, through the control volume upstream of the aspiration flow path, is a user-weighted function of the flowrates in the regular secondary flow paths upstream and downstream from the control volume.

An optional valve model can simulate feedwater control valves, relief and safety valves, and valved leak paths. Inputs include setpoints, signal time delays, and stroke times. Valve flowrate is proportional to valve opening area, which varies linearly with stem position.

On the basis of the above understanding the following evaluation can be made.

4. General Limitations

TRAP2, being similar to the RELAP series of codes, is a very flexible tool due to the arbitrary nodalization schemes possible. Approval of the models described in the topical report will not obviate justification of specific nodal applications of the code on a case by case basis. Limitations of the models are

a. The use of point kinetics which means that use for transients which involve 3-D space time effects such as local rod ejection transients would have to be justified on a conservative basis.

b. B&W has not justified the use of the code for SBLOCA and further modelling justification would have to be provided for such use.

c. For ATWS it will have to be verified that the water properties tables are appropriate for the high pressure region.

d. The two node fuel pin model neglects transient temperature profile effects. Rapid reactivity transients may be in error due to inaccuracies in the Doppler feedback calculation.

e. The heat transfer multiplier algorithm used with the steam generator heat transfer coefficients could have a potential to impact the transient results.

f. As no natural convection heat transfer correlations are implemented in the core, low flow conditions may be in error.

TRAP2 alone cannot be used to predict cross flow and/or non-uniform thermal mixing. In cases such as steam line break where the effect could be important, conservative assumptions concerning partial or no mixing plus possibly analysis with a 3-D space-time kinetics code, should be used. The 3-D analysis should also be used in all cases involving local power peaks such as a stuck control rod can produce.

B. Code Qualification

In support of their code qualification work B&W has performed sensitivity studies with TRAP2 and submitted some information on comparisons with limited measured data. These are summarized in the next section. As part of this evaluation ANL has also made audit calculations with RELAP4/MOD6, using the Midland Plant Model, for a Loss of Feedwater Flow transient and for a Steam Line Break transient.

1. B&W Studies

B&W performed noding and time step studies to show that TRAP2 solutions are convergent. The severe base case used was a double-ended rupture of a main steam line for a 205 FA B&W plant, using the 0.008 s time steps normally used, with the 10 axial-node steam generator representation. Two other calculations of 35 s duration were made using 0.0004 and 0.001 s time steps, respectively, and no differences of note were observed in the results from all three calculations. Another calculation was performed using far fewer nodes, for example only four axial steam generator nodes instead of ten. Again there was little difference between the results from the few-node calculation and the base-case calculation.

To indicate that lumping the core with the outlet (upper) plenum (as was done in the above base case) is adequate, another calculation of 65 s duration was performed in which the core and outlet plenum were assigned separate nodes. No differences of note were observed between the results from the lumped core/outlet-plenum calculation and the separated-node calculation.

To verify that omitting the upper head region (as was done in the above base case) is acceptable, another calculation of 65 s duration was conducted in which the upper head region was assigned a node. In this case depressurization was arrested, as expected, by flashing of the upper head coolant, thereby increasing minimum DNBR. For steam line break events, omitting the upper head region is thus conservative.

To show that using static conditions to calculate break flow (as was done in the above base case) is acceptable another calculation was performed in which the break area was increased by 40 percent to generate the higher break flowrates that proper use of stagnation conditions, rather than static conditions, would produce. The results obtained were no more severe than those from the base case.

To indicate that the two-node fuel pin conduction model (one fuel and one cladding node) is adequate for analyses of secondary system breaks, a steam line break event was analyzed with the CADDS code using both three and six radial fuel nodes. For each case, core heat demand was provided from TRAP2 results. No noteworthy differences were observed among the results from the one, three, and six radial fuel node results.

To verify that the transient modeling is capable of maintaining the steady-state level in the event of a negligible break, the above base case analysis was repeated with a very small break. This calculation showed system conditions to remain at steady-state values during the transient.

To show that the assumption of zero bubble rise velocity (as was used for the above base case) is conservative for a steam line break event, additional calculations using bubble rise velocities of 2.0 and 9.0 ft/s were conducted. Increasing bubble rise velocity was found to decrease primary system depressurization rate; consequently increasing bubble rise velocity reduces, or at worst does not increase, severity of event results.

To indicate that omission of primary and secondary metal slabs, as was done for the above base case, is acceptable for mass and energy release calculations, the base case analysis was repeated with primary and secondary metal slabs included. There was essentially no difference between results from the two comparable calculations.

A comparison was made with data measured during a reactor trip/turbine trip at Oconee Unit 1. Discrepancies between calculated and measured data were explained satisfactorily in a written response to one of our questions.

A comparison was made with data measured during a loss of feedwater event at Oconee Unit 1. Use of the non-equilibrium pressurizer option in the calculations gave good agreement between calculated and measured results.

2. <u>ANL Audit Calculations</u>

(a) <u>Steam Line Break</u>

ANL audited the 2 ft^2 split break steam line break which was the reported worst case SLB in the Midland FSAR. The RELAP4/6 case indicated substantially larger margin to recriticality (-18¢ vs -6¢) than those obtained by TRAP2 indicating that the Midland analyses were conservative. However, the FSAR did not supply sufficient details on the results of their analysis to permit resolution of the fine structure of the differences in these analyses.

(b) Loss of Feedwater Flow

ANL audited the Midland FSAR analysis of Total Loss of Main Feedwater Flow transient. Since the FSAR did not specify the details of their analytical assumptions (such as the secondary side safety and dump valve flows), ANL was unable to precisely confirm the Midland results. However for ANL's "best guess", the results were similar and Midland was conservative.

(c) Feedwater Line Break

B&W intends to perform Feedwater Line Break (FWLB) transient calculations with TRAP2. Although ANL did not perform audit calculations for a FWLB accident, ANL believes the models would be reasonably accurate for mild FWLB transients but may not be for breaks which could cause extreme thermal hydraulic conditions.

In summary, these audit calculations do not provide complete verification of the code or models and methods therein due to lack of information on the B&W analyses.

III. Staff Position

We have reviewed the methods and assumptions described in BAW-10128 and have concluded, subject to the following conditions, that the TRAP2 code is an acceptable method for transient analysis of the specific events listed above under Summary of Regulatory Evaluation. However future analyses with the code should be accompanied by detailed review of the specific application.

"Non-symmetric Secondary and/or Primary System Transients" and "Events Leading to Two-phase Conditions" are categories too general for us to accept. We require that use of TRAP2 be limited to the numbered specific events listed above under Summary of Regulatory Evaluation, unless justified for individual exceptions. This requirement is not intended to limit the use of TRAP2 for non-symmetric transients occurring within the numbered event categories.

ATWS analysis will require verification of the high pressure regime of the water properties tables.

Long term cooldown to natural convection levels will require separate justification.

Specific nodalization schemes will be approved on a case by case basis.

We have not reviewed TRAP2 capability for use of more than one core channel. TRAP2 is unable to treat effects of imperfect mixing in the reactor and/or local power peaks such as a stuck control rod can cause. We require that, whenever such situations occur, conservative mixing assumptions and/or three-dimensional space-time kinetics analyses be utilized, as applicable, to show that TRAP2 results are adequate.

Due to the lack of confirmatory calculations, it may be that for FWLB transients which produce extreme conditions use of TRAP2 could lead to results of insufficient accuracy.

The following items which ANL regards as probably typographical errors have not been resolved.

Page 1-24, Eqn. 1-58

The W-3 correlation is missing brackets and the G term is missing an X factor.

Open Issue: The B&W response introduced a new error: an opening bracket was omitted between the preceding asterisk and the first parenthesis of the G term.

Page 1-26, Eqn. 1-61, first part

Provide the basis for the constant 3716.

Open Issue: The B&W response was not consistent with the units given in Appendix A for K_{MW} (Mg^2/cm^4-s) and ρ_C (lb/ft^3). Further B&W responses should address also the units used in Eqns. 1-62 to 1-65.

Page 1-51, Eqn. 1-130

Provide a more-detailed derivation. (Is "path J" a control volume? Why isn't the boron mass in upstream volume K1 affected by flow into it, and in downstream volume K2 by flow out of it?)

Open Issue: No response for this has been received.

Page 1-52, Eqns. 1-134 and 1-135

Explain why two expressions for VDK are necessary. Provide a more-detailed derivation of Eqn. 1-135 (e.g., is BORONB initial boron concentration at beginning of time step or at beginning of transient?) Explain difference between DKV and VDK (Eqn. 1-136).

Open Issues: Revised Eqn. 1-135 was stated to give "change in boron reactivity due to change in boron concentration in the flow path." However, revised Eqn. 1-135 in actuality does not give either (a) change in boron reactivity since the beginning of the transient, or (b) change in boron reactivity during the last time step. Also, VDK (reactivity change due to density change) and DKB (reactivity change due to boron-concentration change) are redundant, in revised Eqn. 1-136, because density change causes concentration change.

References

1. S. R. Fischer, "RELAP4/MOD6, A Computer Program for Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems, User's Manual," CDAP TR 003, EG&G Idaho, Inc. (January, 1978).