WESTINGHOUSE PROPRIETARY CLASS 3

WCAP-7907-A

LOFTRAN
CODE DESCRIPTION

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WASHINGTON, D. C. 20555

JUL 29 1983

Mr. E. P. Rahe, Jr., Manager
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Westinghouse Electric Corporation
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Dear Mr. Rahe:

Subject: Acceptance for Referencing of Licensing Topical Reports: WCAP-7907(P)/(NP) "LOFTRAN Code Description"; WCAP-7909(P)/(NP), as Superseded by WCAP-8843(P)/WCAP-8844(NP), "MARVEL - A Digital Computer Code for Transient Analysis of a Multiloop PWR System"

We have completed our review of the subject topical reports submitted by the Westinghouse Electric Corporation as follows: WCAP-7907(P)/(NP), submitted October 11, 1972 by letter NS-SL-453; WCAP-7909(P)/(NP) submitted October 28, 1977 by letter NS-CE-1585, as superseded by WCAP-8843(P)/WCAP-8844(NP), submitted December 5, 1977 by letter NS-CE-1627. We find these reports are acceptable for referencing by Westinghouse in license applications for pressurized water reactors to the extent specified and under the limitations delineated in the reports and the associated (NRC) evaluations which are enclosed. The evaluations define the basis for acceptance of the reports.

We do not intend to repeat our review of the matters described in the reports and found acceptable when the reports appear 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 reports.

In accordance with established procedures (NUREG-0390), it is requested that Westinghouse publish accepted versions of these reports, proprietary and nonproprietary, within three months of receipt of this letter. The accepted versions should incorporate this letter and the enclosed evaluations, as appropriate, between the title page and the abstract. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

WESTINGHOUSE

TOPICAL REPORT EVALUATION

LOFTRAN

CODE DESCRIPTION

WCAP-7907

Prepared By:

U. S. Nuclear Regulatory Commission

Division of Systems Integration, NRR

CONTENTS

Section 1: Summary

The mathematical models described in Section 5 of the LOFTRAN topical report adequately address the physical phenomena important in the accidents to which LOFTRAN has been applied (see Section 4.) The LOFTRAN code, when combined with appropriately conservative input assumptions and with the appropriately conservative use of related computer codes (e.g., detailed thermal-hydraulic codes that calculate DNBR) is capable of producing conservative results.

The LOFTRAN thermal-hydraulic model is best suited for use in transients in which the primary coolant flow system remains subcooled. The use of LOFTRAN for accidents involving a limited amount of primary coolant voiding, in the vessel upper head, is acceptable. LOFTRAN is acceptable for flow conditions which may be adequately described with the homogeneous, equilibrium flow model.

The pressurizer can be located in any loop, the only restriction being that reverse flow is not allowed in the loop with the pressurizer. This is a limitation for LOFTRAN, but does not impact on licensee safety analysis evaluations. LOFTRAN can initialize with reverse flow in one or more loops, although the core flow, and loop flow in the pressurizer loop must be positive. Flow reversal transients can be modeled with LOFTRAN.

The staff finds LOFTRAN to be an acceptable computer program for the analysis of Chapter 15 SAR transients and accidents, as identified in Section 4 of this evaluation.

Section 2: Introduction

This report documents the staff review of the Westinghouse computer program LOFTRAN, as used for transients analysis of the Westinghouse multi-loop pressurized water reactor (PWR) system. The scope of this review is limited to an examination of the mathematical models employed and to efforts conducted by Westinghouse to verify the computations of LOFTRAN. The primary goal of this review is to assess whether the code, as applied to analysis of specific accidents (i.e., those found in Chapter 15 of a Final Safety Analysis Report (FSAR)), can be used to produce conservative licensing evaluations. This review does not address other computer codes used in conjunction with LOFTRAN to perform safety analyses.

This review is based on the original description of LOFTRAN, WCAP-7907 (Reference 1), and responses to questions concerning WCAP-7907 (Reference 2). In addition, verification analyses were submitted (Reference 3) and reviewed. A technical meeting was held with Westinghouse to resolve all areas of concern (Reference 4) identified during the review of the LOFTRAN related information. All areas of concern were adequately addressed at this meeting, with the exception of additional verification studies regarding upper plenum voiding and major system effects resulting from such voiding. It was agreed that a LOFTRAN analysis of the January 25, 1982 Ginna steam generator tube rupture event would be analyzed in support of this review effort. This analysis has been submitted (Reference 5) and is discussed in Section 6.

A current review of a similar Westinghouse computer program (MARVEL) was conducted and provided additional insight for this review. Westinghouse

has stated that LOFTRAN has replaced MARVEL for all current safety analyses related work.

For the purpose of this evaluation, WCAP-7907 (Reference 1) is considered to be the description of the one-loop version of LOFTRAN. Reference 2 is considered to be the description of the 4-loop version of LOFTRAN.

Section 3: History of Development of LOFTRAN and MARVEL

The history of the LOFTRAN computer program should not be viewed separately from the history of the MARVEL computer program due to the similarity of the two codes. Development of LOFTRAN was begun in the late 1960's for application to the one-loop Zorita plant. LOFTRAN was subsequently applied to the analysis of two-loop plants by representing the two steam generators as a single composite steam generator. The use of LOFTRAN for asymmetric transients (such as steam line break in which one steam generator is typically affected much more than the other(s)) required special care to conservatively represent the transient behavior using only one composite steam generator loop.

Around 1970, a computer program named BLKOUT (based on the LOFTRAN code) was written for use in extended system transients (e.g., station blackout) for which long time steps were necessary for economical computation. The BLKOUT computer program incorporated a new feature, two simulation loops. The two-loop feature proved useful, so the task was initiated of modifying BLKOUT to handle the short time steps necessary for fast transients. The results were first called BLKOUT-5, but were later changed to MARVEL. By 1972 MARVEL was in use for Chapter 15 analysis of multi-loop plant transients involving asymmetric steam generator loop behavior (e.g., steamline rupture, and startup of an inactive reactor coolant loop). LOFTRAN remained in use for transients in which symmetric loop behavior could be assumed.

In 1976, LOFTRAN was modified to explicitly simulate four steam generator loops. After development of 4-loop-LOFTRAN, the use of MARVEL was generally discontinued. It was used for several years for the analysis of steam/water

release to containment after a steam line break since the first version of LOFTRAN-4 did not include the MARVEL water entrainment model. Later versions of LOFTRAN-4 have included the MARVEL water entrainment model. Presently, MARVEL is no longer used for safety analysis and Westinghouse does not support MARVEL for current safety applications.

Section 4: Applications of LOFTRAN

LOFTRAN is used to simulate plant response to many of the postulated events reported in Chapter 15 of PSARs and FSARs, to simulate anticipated transients without scram, for equipment sizing studies, and to define mass/energy releases for containment pressure analysis. The Chapter 15 events analyzed with LOFTRAN are:

- Feedwater System Malfunction
- Excessive Increase in Steam Flow
- Inadvertent Opening a Steam Generator Relief or Safety Valve
- Steamline Break
- Loss of External Load
- Loss of Offsite Power
- Loss of Normal Feedwater
- Feedwater Line Rupture
- Loss of Forced Reactor Coolant Flow
- Locked Pump Rotor
- Rod Withdrawal at Power
- Rod Drop
- Startup of an Inactive Pump
- Inadvertent ECCS Actuation
- Inadvertent Opening of a Pressurizer Relief or Safety Valve

This review is limited to the use of LOFTRAN for the licensee safety analyses of the Chapter 15 events listed above, and for a steam generator tube rupture (as identified by Westinghouse in Reference 4). These safety analyses evaluations include reactivity events, changes in secondary heat removal capacity, loss of

flow transients, and accidental depressurization.

Some of the LOFTRAN-analyzed accidents involve excessive cooling which lowers the average core moderator temperature, thereby causing a power excursion. The main parameter of interest for these accidents is the departure from nucleate boiling ratio (DNBR) associated with the power excursion. The calculation of the DNBR may be performed with other computer programs using (as input) the LOFTRAN-generated system variables; such as primary coolant pressure, normalized total core power, and core inlet enthalpy. The DNBR calculation typically involves a neutronics code to calculate axial and radial power shapes, and a detailed thermal-hydraulic code to calculate the coolant and fuel rod conditions including DNBR in the hot channel. These other codes are not considered in this review. The DNBR model used in LOFTRAN is considered in the review.

A few of the excessive cooling events, such as an inactive loop startup and feedwater system malfunction, are (typically) mild transients which are effectively terminated by reactor trip. The purpose of these analyses is to show that a reactor trip will occur before the increasing reactor power causes an undesirably low DNBR. The steam line rupture is a more severe accident because the primary coolant temperature continues to decrease rapidly even after the reactor is tripped (the worst case initial condition is hot shutdown with the reactor already tripped). If there are no stuck control rod assemblies (CRAs), the steam line break is of little consequence to the primary coolant system. If the most reactive CRA is stuck out of the core, then (with end-of-cycle moderator temperature feedback) the core returns to criticality when the average core moderator temperature has decreased by

about 70°F. The power level after criticality is low, but the primary coolant pressure is also low, and the power peaking factors are high, so the DNB ratio must be analyzed in detail. The core is returned to subcriticality by the Safety Injection System which injects borated water into the primary coolant.

Some of the LOFTRAN-analyzed accidents are under-cooling rather than over-cooling accidents. The major parameter of interest, primary coolant pressure, is calculated directly. The LOFTRAN analysis is performed to show that an excessive pressure is not reached and that successful cooling of the primary coolant system is established by startup of the auxiliary feedwater system.

Section 5: Review of LOFTRAN Models

LOFTRAN simulates a multi-loop system by modeling the reactor core and vessel, hot and cold leg piping, steam generator (tube and shell sides), pressurizer, and reactor coolant pumps, with up to four reactor coolant loops. The pressurizer model includes the effects of pressurizer heaters, spray, and relief and safety valve operation. The reactor core model employs a lumped fuel heat transfer model with point neutron kinetics and includes the reactivity effects of variations in moderator density, fuel temperature (Doppler), boron concentration, and control rod insertion and withdrawal. The secondary side of the steam generator model utilizes a homogeneous, saturated mixture for the thermal transients and a water level correlation for indication and control. The turbine, condenser, and feedwater heaters are not simulated; instead, steam demand and feedwater flow are input to LOFTRAN, or controlled by specifying certain control options. The reactor protection system is simulated to include reactor trips on neutron flux, overpower and overtemperature reactor coolant delta-T, high and low pressure, low flow, and high pressurizer level. Control Systems can be simulated, and include rod control, steam dump, feedwater control, and pressurizer pressure control. The Safety Injection System including the accumulators is also modeled. Reactor coolant pump operation is simulated including the effects of pump coastdown and pump startup, and flow reversal is allowed.

LOFTRAN is able to calculate the voiding expected during steam line breaks because, when the pressurizer drains, voiding in the primary coolant system

is restricted to the upper head of the reactor vessel, a hydraulically stagnant region which receives only a small fraction of the main coolant flow. Phase separation is assumed to occur in this region by virtue of the assumption that outflow is single phase liquid. A thick vessel wall metal model is available for the upper head, but it is generally not used, unless it is conservative to do so. Even the largest steam line breaks do not cause all the water to be boiled from the upper head.

The LOFTRAN core kinetics model consists of a single lumped fuel heat transfer model, a point neutron kinetics model, and a decay heat model.

The fuel heat transfer model uses variable axial nodes (user specified) and one radial node, with a parabolic axial power distribution. A fuel specific heat which varies with fuel temperature is used, and a small fraction of the heat is generated directly in the coolant. The overall fuel to coolant heat transfer coefficient (UA) is a parabolic fit to values input by the user. The values input are either maximum or minimum heat transfer values depending on the conservative direction for the transient of interest and are obtained from limiting values predicted from detailed Westinghouse fuel rod design codes. This model may be used for predicting average core heat transfer if the transient is not a very rapid core power transient such as the control rod ejection and control rod withdrawal from subcritical accidents. In transients where a deatiled knowledge of the heat transfer or fuel temperature is important, such as for detailed DNBR evaluations, the LOFTRAN thermal power prediction is not used. Instead, the nuclear power versus time is transferred to a more detailed transient fuel pellet heat transfer model (such as FACTRAN) for calculation of hot and average channel heat flux.

The DNBR calculated in LOFTRAN is not based on physically or empirically based equations. Instead, it is represented as an expansion about the design limit value. This expansion is in terms of the partial derivatives of DNBR with respect to power, with respect to flow, and with respect to pressure. The partial derivatives are supplied by the user as input. Multiple data sets are allowed, to account for non-linearity. LOFTRAN calculates the changes in power, flow, and pressure.

The point neutron kinetics model in LOFTRAN uses six delayed neutron groups and employs an implicit finite difference solution technique for stability. A source term and the prompt neutron cycle time are included in the equation. The model takes into account reactivity changes due to changes in moderator temperatures, the Doppler effect, boron concentration, control rod position, and input values of reactivity versus time. Moderator density and boron worth coefficients, variable rod worth versus position, and an integral Doppler defect versus power with a correction for water temperature change are input by the user. A scram reactivity curve versus time is also user input.

The code contains a facility for using a core quadrant weighted density, water temperature, and boron concentration in determining the reactivity feedback in order to predict the course of transients with large loop temperature and core power distribution asymmetries (such as in the steam-line break accident). The weighting factors must be supplied by the user. The accuracy of the point kinetics model employed by LOFTRAN is therefore dependent on the reactivity and feedback coefficients input by the user, and how representative they are for the accident or transient being evaluated.

The decay heat used in LOFTRAN is computed from a five-group precursor model similar to the delayed neutron precursors. The default value matches the ANS +20% curve; however, the total value used can be scaled up or down by the user. For some transients (e.g., steam line break) decay heat is a benefit and may be conservatively removed in the analysis.

The reactor coolant loop model employs a nodal technique with the number of nodes (actually control volumes) specified by the user. The pressurizer can be located in any loop, the only restriction being that reverse flow is not allowed in the loop with the pressurizer. The homogeneous, equilibrium flow model (HEM) is used. Hence, its validity with steam in the active flow paths is limited to those cases which might be adequately represented by a homogeneous mixture of steam and water with no slip. LOFTRAN will initialize with reverse flow in one or more loops, although the flow in the core and the loop with the pressurizer must be positive. Boron transport is modeled in LOFTRAN. However, because of the homogeneous slug flow models employed, the transport time delay is strictly correct only if the volumes and flow rates are such that an exact volume replacement occurs in one time step. This is most likely to be important for a fast transient under a low flow condition.

The momentum equation is solved for flow, including effects of friction pressure losses, elevation (density) heads, pump head, and fluid momentum. Reactor coolant pump homologous curves are input by the user, and LOFTRAN computes pump head and torque. The pump speed equation includes the effect of pump motor torque, hydraulic torque on the impeller, pump windage friction, and pump rotating inertia. The flow model is used for pump coastdowns, locked rotor, and natural circulation flow calculations. The code is capable of

calculating transient flow reversals.

Two phase pump degradation is not modeled in LOFTRAN. Since LOFTRAN employs the homogeneous equilibrium flow model to calculate the flow, and since experimental pump data indicates no degradation for void fractions of less then 0.1, the omission of this model is acceptable.

The mixing between loops which occurs in the inlet and outlet plenums of the reactor vessel without any net flow between loops is not simulated mechanically Instead LOFTRAN has a provision for mixing in the inlet plenum and in the outlet plenum. The amount of mixing is determined by coefficients which allow the user to input any desired degree of mixing between loops. For steamline break analyses, the input mixing coefficients are based on the fraction of mixing measured in the Indian Point One-Seventh Reactor Vessel Model Test. The oefficients used are conservative because they lead to a colder inlet temperature in the affected loop and thus to a more severe reactivity excursion.

The pressurizer model computes the mass and energy balances in a two (water and steam) region pressurizer. Since the water level may change during a transient, a variable volume model is used. Each region is assumed to be uniform (perfect mixing). Condensation or superheat is allowed in the steam region and evaporation or subcooling is allowed in the water region. Water drops are assumed to be uniformly distributed in the steam region and fall at a constant rate. Steam bubbles are assumed to be uniformly distributed in the water region and rise with a constant velocity. While this model is not a true non-equilibrium model and pressurizers do not appear to behave in thermal equilibrium, especially during insurges, comparisons of LOFTRAN to data and to other computer programs for conditions of pressurizer insurge demonstrate that this model is reasonable, and therefore is acceptable.

The model includes the effects of heaters, spray, and relief and safety valves, with their appropriate control systems. Safety analysis calculations are conservatively performed assuming no pressure control if such control would improve the results, or with full control if this is the conservative direction. Safety and relief valve flow rates for steam relief are obtained from values specified by the manufacturer using ASME code methods and are input by the user. For water relief, the valve area is generally input, and the user selects the appropriate critical flow model.

On the primary side, the steam generator model contains multiple control volumes which represent the tube region. The secondary side is represented as a one node model with a saturated mixture of steam and water. This model cannot account for thermal stratification on the secondary side. While thermal stratification is important for the recovery phase of a steam generator tube rupture event, the SAR Chapter 15 analysis assumptions are not concerned with this phase (following isolation of the faulted steam generator). The LOFTRAN secondary side model is therefore acceptable for SAR analyses. The feedwater and steam flow are determined based on demand according to the user input option selected. Heat transfer is computed using a Log Mean Temperature Difference type representation with the overall heat transfer (UA) determined using primary flow, heat flux, secondary side pressure and mass to compute primary and secondary side film resistance and tube resistance, and the heat transfer area. The overall UA is initialized by the code to match the nominal input conditions provided by the user. The nominal conditions are obtained from plant design thermal-hydraulic conditions and from known steam generator performance obtained from actual plant experience. A steam generator water level correlation is provided; however, this is normally used for information only. Reactor trip and auxiliary feedwater start on steam generator water level are based on a user-input value of

an equivalent secondary side mass. This value is based on a more detailed steam generator model which computes steam generator water mass at the reactor trip level setpoint. In addition to steam flow demand by the turbine, steam relief through safety valves, the steam dump system and through pipe breaks can be simulated. Feed flow versus time can be input or set equal to steam flow within the code, or various feedwater line breaks can be simulated.

The Moody correlation, with f(L/D) set to zero, is used to compute break flow. Steam and feedline isolation may also be simulated, including the effects of a failure to isolate one or more loops. Auxiliary feedwater flow is simulated as a constant flow versus time after actuation and is assumed to be injected in a slug flow model through a user-specified purge volume, which represents the feedwater line hot fluid volume. The user may control the fraction injected into each loop. Several options are provided for the user to take into account the effect of the degradation in heat transfer surface area caused by uncovering the steam generator tubes in a loss of secondary water inventory event. The appropriate input is selected to be conservative for the transient of interest, and in conjunction with prior results from more detailed steam generator models.

LOFTRAN simulates the safety-injection system consisting of a pump, boron injection tank, and associated piping. The pump head versus flow characteristics are input, and the system uses a homogeneous slug-flow model to inject the safety-injection flow into the cold legs at a cold leg location specified by the user. The pump section is assumed to come from the refueling water storage tank, and the volume, fluid enthalpy, and fluid boron concentration may be specified in each connecting pipe section. In addition, an accumulator may be specified for any loop or for upper head injection. A separate emergency

boration system model is also available and may be either a pumped system connected around the reactor coolant pump, or a passive system, depending on elevation heads and loop pressure drops, connected between the pressurizer and the cold leg. Both systems employ a homogeneous slug-flow model and inject into the cold leg between the steam generator and reactor coolant pump.

Control systems simulated in LOFTRAN are automatic rod control, steam dump control, and pressurizer pressure control via pressurizer heaters and spray. A digital simulation of each control system is provided including linear and nonlinear gain units, auctioneering, lead-lag compensation units, filters, PID controllers, dead bands, and simulation of the time responses of the sensor inputs. The protection system include reactor and turbine trips, safety-injection actuation, and steam and feedline valve actuation. Protection system error allowances and time response are simulated by inputting protection setpoints plus appropriate error allowances and actuation delay times. These values are determined independently of the LOFTRAN code from the specific system design characteristics and are verified on the actual plant at time of start-up.

While the major control systems, such as rod control and steam-dump control, are modeled in LOFTRAN, these systems do not figure heavily in the analysis of most Chapter 15 transients because they are normally defined to be non-safety systems. Non-safety control system action may act to mitigate the consequences, but safety analyses do not take credit for this mitigation; therefore, Chapter 15 analyses tend to ignore control systems effects. A typical assumption would be that the controllers remain at the pre-accident demand levels. Assumptions concerning control system action are selected to be conservative and

the non-safety control system should not respond to the accident in a manner to worsen overall system response. The modeling of non-safety control systems was not part of this review.

The reactor protection system is part of the LOFTRAN simulation, and is of importance to Chapter 15 safety analyses. The equations necessary to describe the response and actuation logic of the protection system are straightforward. The protective system actions assumed by the Chapter 15 analyses become a definition of minimum required protection system performance. Instrument accuracy, response time, and setpoints must be maintained by the licensee such that if the accident did occur the protective action would happen as soon or sooner than assumed in the Chapter 15 analysis.

Section 6: Verification and Quality Assurance

The LOFTRAN computer program verification performed by Westinghouse includes the following fourteen transients (Reference 4), and consists of comparison of LOFTRAN results to actual plant data and to other similar thermal-hydraulic programs:

(1) Reactor trip from 100% power at H. B. Robinson 2

(2) Reactor trip from 75% power at Mihama (Japan)

(3) 100% load reduction at Mihama

(4) 65% to 40% load step at Mihama

(5) 40% to 65% load step at Mihama

(6) 72% to 28% load step at North Anna 1

(7) 97% to 54% load step at North Anna 1

(8) Reactor trip from 96% power at North Anna 1

(9) Comparison of LOFTRAN to BLKOUT and MARVEL for a loss of normal feedwater transient

(10) Comparison of LOFTRAN to MARVEL for a steam line rupture transient

(11) Comparison of LOFTRAN to WFLASH and MARVEL for a 412 Standard Plant Cooldown Transient

(12) Comparison of LOFTRAN to WFLASH and MARVEL for a 412 Standard Plant Heatup Transient

(13) Comparison of LOFTRAN to WFLASH for ANS benchmark problem

(14) Comparison of LOFTRAN (1-loop) to LOFTRAN (4-loop) for loss of load transients at Comanche Peak.

LOFTRAN has also been reviewed by the staff for application to the Anticipated Transients Without Scram (ATWS) issue (Reference 6). The acceptability of LOFTRAN for ATWS, and the comparisons of LOFTRAN with RELAP3B results, has been accounted for in this review.

LOFTRAN is also used by Westinghouse to analyze steam generator tube rupture (SGTR) events. A LOFTRAN analysis for the R. E. Ginna SGTR events of January 25, 1982 has been submitted (Reference 5) as part of the LOFTRAN verification effort. The comparison of LOFTRAN to available plant data demonstrates the ability of LOFTRAN to analyze a SGTR event, at least prior to the failure of the PORV to close at Ginna. It is noted that the limitation of no reverse flow in the pressurizer loop was significant for the Ginna event. However, the Westinghouse evaluation of the analytical results, which considered the limitation of LOFTRAN, are judged to be adequate.

The data comparisons and computer program results comparisons provided by Westinghouse demonstrate the ability of LOFTRAN to analyze the types of events for which it has been used in licensee safety analyses. This is judged to be adequate verfication for LOFTRAN.

At the request of NRR, (Reference 7), Region IV performed an inspection of the quality assurance program used by Westinghouse for the development and use of LOFTRAN. The results of this inspection (Reference 8) found the Westinghouse quality assurance procedures as applied to the development of LOFTRAN to be acceptable.

Section 7: References

1. WCAP-7907, "LOFTRAN Code Description," Westinghouse, October 1972.

2. NS-TMA-1802, Letter from T. M. Anderson (W) to J. F. Stolz (NRC), Response to May 10, 1978 request for additional information on WCAP-7907, dated May 26, 1978.

3. NS-EPR-2536, Letter from E. P. Rahe, Jr., (W) to J. R. Miller (NRC), Forward Slides from December 15, 1981 meeting on LOFTRAN and MARVEL, dated January 19, 1982.

4. NS-EPR-2648, Letter from E. P. Rahe, Jr., (W) to C. O. Thomas (NRC), Summary of NRC/ORNL/Westinghouse Technical Review Meeting of July 13-14, 1982, dated August 27, 1982.

5. Docket 50-244, Letter from J. E. Maier (RGE) to D. M. Crutchfield (NRC), "Response to Safety Evaluation Report - NUREG-0916, Steam Generator Tube Rupture Incident, R. E. Ginna Nuclear Power Plant," dated November 22, 1982.

6. NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors," USNRC, April 1978.

7. Memorandum for U. Potapovs (NRC-IV) from T. P. Speis (NRC), "Request for Inspection," dated May 4, 1981.

8. Inspection Report No. 99900404/81-02, from U. Potapovs (NRC-IV) to R. J. Slemer (W), dated June 29, 1981.

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ABSTRACT

LOFTRAN is a digital computer code written in the FORTRAN language which was developed to simulate transient behavior in a multi-loop pressurized water reactor system. The code simulates a multi-loop system by modeling the reactor core and vessel, hot and cold leg piping, steam generator (tube and shell sides), pressurizer, and reactor coolant pumps, with up to four reactor coolant loops. The pressurizer model includes the effects of pressurizer heaters, spray, and relief and safety valve operation. The reactor core model employs a lumped fuel heat transfer model with point neutron kinetics and includes the reactivity effects of variations in moderator density, fuel temperature (Doppler), boron concentration and control rod insertion and withdrawal. The secondary side of the steam generator is represented by a single volume model (water and steam phase) with a homogenous, saturated mixture for thermal transients. The turbine, condenser, and feedwater heaters are not simulated; instead steam demand and feedwater flow and enthalpy are input to the code or controlled by specifying certain control options. The reactor protection system is simulated including reactor trips on DNB and overpower protection, overpower and overtemperature ΔT, high neutron flux, high and low pressurizer pressure, high pressurizer level, low reactor coolant flow, low steam generator level, and safety injection actuation. The engineered safeguards features simulated are feedwater and steamline isolation, auxiliary feed, and an emergency core cooling system with high and low head safety injection and accumulators. The control systems simulated are rod control, steam dump, and pressurizer pressure control. Reactor coolant pump operation is simulated including the effects of pump coastdown and pump startup with loop flow reversal allowed.

The LOFTRAN code has been used for many years by Westinghouse for accident evaluations for Safety Analysis Reports, and for control system performance and equipment sizing studies. The principal use of the code has been for accident evaluations for intact circuit (non-LOCA) faults including reactivity events, changes in secondary heat removal capacity, loss of flow transients, and accidental depressurization.

SECTION 1

SUMMARY DESCRIPTION OF PRINCIPAL LOFTRAN MODELS

1.1 INTRODUCTION

The principal models used in LOFTRAN are those associated with modeling the reactor core, reactor coolant flow, the pressurizer, steam generators, and control and protection systems. A more detailed description of the equations used, numerical solution techniques, approximations and correlations, and model limitations is found in later sections of this manual.

1.2 REACTOR CORE MODEL

The LOFTRAN core kinetics model consists of a lumped fuel heat transfer model, a point neutron kinetics model, and a decay heat model. In addition, the code can calculate DNBR during a transient.

1.2.1 Fuel Heat Transfer

The fuel heat transfer model uses up to [ ]$^{+a,c}$ axial nodes (user specified) and up to four radial nodes (one per loop), with a fixed parabolic axial power distribution of [ ]$^{+a,c}$ peak to average value. A fuel specific heat which varies with fuel temperature is used, and a fixed 2.6 percent of the heat is assumed generated directly in the coolant. The overall fuel to coolant heat transfer coefficient (UA) is [ ]$^{+a,c}$ The values input are usually either maximum or minimum heat transfer values depending on the conservative direction for the transient of interest, and are obtained from values predicted from the detailed Westinghouse fuel rod design codes. This method accounts for the effect of fuel-clad gap width variation with fuel temperature. This model is adequate for predicting average core power response for all except the most rapid core power transients such as the RCCA ejection and RCCA bank withdrawal from subcritical faults where the LOFTRAN code is not used. In addition, for transients where specific values of the

heat flux or fuel temperatures are important, the LOFTRAN nuclear power vs. time is transferred to a more detailed transient fuel pellet heat transfer model (such as the FACTRAN code) for calculation of hot and average channel heat flux.

1.2.2 Neutron Kinetics Model

The point neutron kinetics model in LOFTRAN uses six delayed neutron groups, and employs an implicit finite difference solution technique for stability. A source term and the prompt neutron lifetime are included in the equation. The model takes into account reactivity changes due to changes in moderator temperature, the Doppler effect, boron concentration, control rod position, and input values of reactivity vs. time. [

] +a,c

In addition, the code contains a facility for using a core quadrant weighted density, water temperature, and boron concentration to determine the reactivity feedback in order to conservatively predict the course of transients with large loop temperature and core power distribution asymmetries such as in the steamline break accident. The weighting factors must be supplied by the user.

[

] +a,c

1.2.3 Decay Heat Model

The decay heat used in LOFTRAN is calculated from a five-group precursor model in a manner similar to the delayed neutron precursors. The default value closely follows (within ± 1 percent after 1 sec) the ANS (1971) + 20 percent curve for finite irradiation (end of equilibrium cycle) plus the actinide contribution. The total value used can be scaled up or down by the user, or different constants can be input. For some transients (e.g., steamline break) decay heat is a benefit and may be conservatively removed in the analysis.

1.2.4 DNBR Evaluation Model

LOFTRAN has the facility for calculating the value of DNBR during a transient using a simple calculational model. The model employs user-input values of the minimum DNBR with respect to changes from nominal in the core average power, average coolant temperature, flow, and pressure. Experience has shown that this model is sufficiently accurate over the range from the nominal to the limit DNBR; nevertheless analysis procedures require limiting minimum DNBR's to be checked with a more detailed calculation of the type performed by a code such as THINC. The LOFTRAN model is not used to calculate DNBR for loss of flow faults, or for faults where asymmetric power distributions are important such as the steamline break or dropped RCCA faults. LOFTRAN may be used with FACTRAN and THINC for prediction of minimum DNBR for loss of flow faults, and with single or multi-channel THINC-type codes for steamline break DNBR evaluations.

1.3 REACTOR COOLANT LOOP MODEL

1.3.1 Reactor Coolant Loops

The reactor coolant loop model employs a nodal technique with the number of nodes (actually control volumes) specified by the user.

The code will handle up to [ ]$^{+a,c}$ core sections, [ ]$^{+a,c}$ hot leg sections per loop, [ ]$^{+a,c}$ steam generator tube sections per loop, and [ ]$^{+a,c}$ cold leg sections per loop.

Generally, a typical analysis employs about one half of the number of allowable sections in each component. The pressurizer can be located in any loop, the only restriction being that reverse flow is not allowed in the loop with the pressurizer. A homogeneous-equilibrium slug flow model is used, thus the code will handle void generation, but the steam and water phase are always in equilibrium and there is no slip. This model is entirely adequate for cases with moderate void generation and under pumped flow conditions.

[

]+a,c The fluid equations solved are those for conservation of mass and energy; the momentum equation is only solved to determine overall loop flow rate with the change in flow vs. time assumed to be uniform around the loop. The code will initialize with reverse flow in one or more loops, although the flow in the core and the loop with the pressurizer must be positive. Boron transport is handled; however, due to the homogeneous slug flow models employed, the transport time delay is strictly correct only if the volumes and flow rates are such that an exact volume replacement occurs in one time step. This is not likely to be important except for a fast transient with safety-injection under a low flow condition where the boron concentration could vary significantly around the loop.

1.3.2 Reactor Coolant Flow

The basic equation of motion is solved for flow, including effects of friction pressure losses, elevation (density) heads, pump head, and fluid momentum. Reactor coolant pump homologous curves are input by the user and the code computes pump head and torque. The pump speed equation includes the effect of pump motor torque, hydraulic torque on the impeller, pump windage and friction, and pump rotating inertia. The flow model is used for pump coastdowns, locked rotor, and natural circulation flow calculations. The code is capable of calculating transient flow reversals due to a RCP start-up or shutdown in one or more loops. The equations solved by this model are straightforward and the results have been found to be adequate in comparison with actual flow coastdown measurements.

1.3.3 Reactor Vessel Mixing

Reactor vessel mixing in the inlet and outlet plena is simulated by the code based on user input. The accuracy of this model therefore depends essentially on the accuracy and appropriateness of the user input rather than the code itself; however, only a few transients result in large inlet temperature asymmetries (for example the steamline break accident) and would be very sensitive to this input.

1.4 PRESSURIZER MODEL

The pressurizer model computes the mass and energy balances in a two (water and steam) region pressurizer. Since the water level may change during a transient, a variable control volume model is used. Each region is assumed to be uniform (perfect mixing). Condensation or superheat is allowed in the steam region, and evaporation or subcooling in the water region. Water drops are assumed uniformly distributed in the steam region and fall at a constant rate, while steam bubbles are uniformly distributed in the water region and rise with a constant velocity. The model includes the effects of heaters, spray, and relief and safety valves, with their appropriate control systems. Safety analysis calculations are usually performed conservatively assuming no pressure control if such control would improve the results or with full control if this is the conservative direction.

Relief valve flow rates for steam relief as a function of pressure are obtained from values provided by the valve manufacturer using ASME code methods and are input to the LOFTRAN code. For water relief, the valve area is generally input and the user selects the appropriate critical flow model. For ATWS, the homogenous-equilibrium model (HEM) is used, which in LOFTRAN is represented by a built-in table of values of critical mass flow rate vs. water enthalpy and pressure, taken from ANS Standard N661. In addition, to account for the effects of a possible non-uniform mixture in the pressurizer water

region due to the rapid insurge of fluid from the RCS hot leg, LOFTRAN includes a facility for specifying the water enthalpy in the table at the more conservative initial value.

1.5 STEAM GENERATOR MODEL

On the primary side, the steam generator model contains multiple (up to 16) tube sections. The secondary side is a one section model with a saturated mixture of steam and water. Multiple tube sections are used on the primary side in order to simulate a Log-Mean Temperature Difference (LMTD) type response. The overall UA is initialized by the code to match the nominal input conditions provided by the user. The nominal conditions are obtained from the plant design thermal-hydraulic conditions and the known steam generator performance from actual plant experience. The code uses the primary mass flow rate, heat flux, and secondary side pressure to compute changes in the heat transfer coefficient due to changes which can affect the primary and secondary side film resistance.

A steam generator water level correlation is provided for simulating SG level indication. Safety system actuation on steam generator water level is not based on the level correlation; instead it is based on a user-input value of an equivalent secondary side mass. This value is conservatively chosen by the user based on a prior calculation with a much more detailed steam generator model which computes steam generator water mass at the level setpoint.

Steam and feedwater flow are determined based on the user option selected. Although the steam turbine is not explicitly modeled in LOFTRAN, the effect of the turbine control system is simulated by assuming constant steam demand prior to turbine trip. As steam pressure rises, steam flow will remain constant due to closure of the turbine throttle control valves. As steam pressure falls, the opening of the TTCV is simulated by inputting the design excess valve capacity. Once the TTCV is fully open steam flow decreases proportionally with steam pressure. Changes in steam demand may be simulated

by inputting steam flow vs. time. In addition to steam flow demanded by the turbine, LOFTRAN calculates steam relief through safety valves, the steam dump system, and through pipe breaks in various locations. Feed flow vs. time can be input, set equal to steam flow within the code, or various breaks can be simulated. The Moody correlation with f(L/D)=0 is used to compute break flow. Break quality vs. mass can be input by the user. Steam and feedline isolation are simulated, and steamline check valves can also be specified. Auxiliary feedwater flow is simulated as a constant flow vs. time after actuation and is assumed to be injected in a slug flow model through a user-specified purge volume. The user may control the fraction injected to each steam generator. Steam generator tube breaks are simulated with the break flow calculated using the Zaloudek correlation.

Several options are provided for the code to take into account the effect of the degradation in heat transfer surface area caused by uncovering the SG tubes due to loss of water inventory. The user may input a secondary side water volume below which the tubes start to uncover, and the code will reduce heat transfer area linearly with the further reduction in water volume. The user determines the appropriate input (high or low value) depending upon which value is conservative for the transient of interest. LOFTRAN also has available a built-in correlation which calculates SG riser quality and reduces heat transfer with water volume once a user-input value of quality is exceeded. This value is obtained by the user from a much more detailed steam generator model. Alternatively, the UA calculated with the detailed model may be input directly as a function of steam generator water mass. Except for a few faults involving long term effects of a loss of feedwater fault or feed and steamline breaks, this model is not important since the reactor trips on the low SG level trip before tube uncovery occurs.

1.6 ENGINEERED SAFEGUARDS SIMULATION

The LOFTRAN code simulates a safety-injection system consisting of a pump, boron injection tank and associated piping, and accumulators. The pump head vs. flow characteristics are input and the system uses a homogenous slug-flow

model to inject the SI flow into the cold legs at a location specified by the user. The pump suction is assumed to come from the refueling water storage tank (RWST), and the volume, initial fluid enthalpy and boron concentration may be specified in each connecting pipe section. In addition, an accumulator may be specified for any loop or for upper head injection. A separate emergency boration system model is also available, and may be either a pumped system connected around the reactor coolant pump, or a passive system, depending for flow on elevation heads and loop pressure drops, connected between the pressurizer and the cold leg. Both systems employ a homogeneous slug flow model and inject into the cold leg between the steam generator and reactor coolant pump. Feed and steamline isolation is simulated, with the option to fail to isolate one or more loops.

1.7 CONTROL AND PROTECTION SYSTEMS SIMULATION

Control systems simulated are automatic rod control, steam dump control, and pressurizer pressure control via pressurizer heaters, spray, and relief valves. A complete digital simulation of each control system is provided including linear and non-linear gain units, auctioneering, lead-lag compensation units, filters, PID controllers, dead bands, and simulation of the time responses of the sensor inputs. The protection systems simulated include reactor and turbine trips, safety injection actuation, and steam and feedline isolation. The reactor trips simulated and the actuators for safety injection and steam and feedline isolation are discussed in Section 6. Failure of one or more protection channels may be simulated. Protection system inaccuracies and time response are simulated by inputting protection setpoints plus appropriate error allowances, and actuation delay times. These values are determined independently of the LOFTRAN code from the specific system design characteristics, and are verified on the actual plant at time of startup.

SECTION 2
PROGRAM DESCRIPTION

2.1 PROGRAM ARCHITECTURE

The LOFTRAN code is a Library of Transient routines, which together simulate operation of a pressurized water reactor. The program has been set up in a modular fashion to facilitate continuing code evolution.

Several subroutines are used for internal control of code execution, loading overlays, calling transient subroutines in the appropriate sequence, etc. Separate subroutines (or modules) are set up to perform specific functions, e.g., neutron kinetics, pressurizer, steam generators, input. Data is passed between the various modules through Blank Common. A number of auxiliary routines are also provided for use throughout the code, again with separate routines for specific functions such as slug flow computations, water properties, critical flow calculations, etc.

2.2 SUBROUTINE CALLING SEQUENCE

The general sequence of computations in a LOFTRAN run is summarized below and in Figure 2.2-1. Subroutines referenced in this description are described in detail in succeeding sections of this manual.

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[+a,c]

+3,c

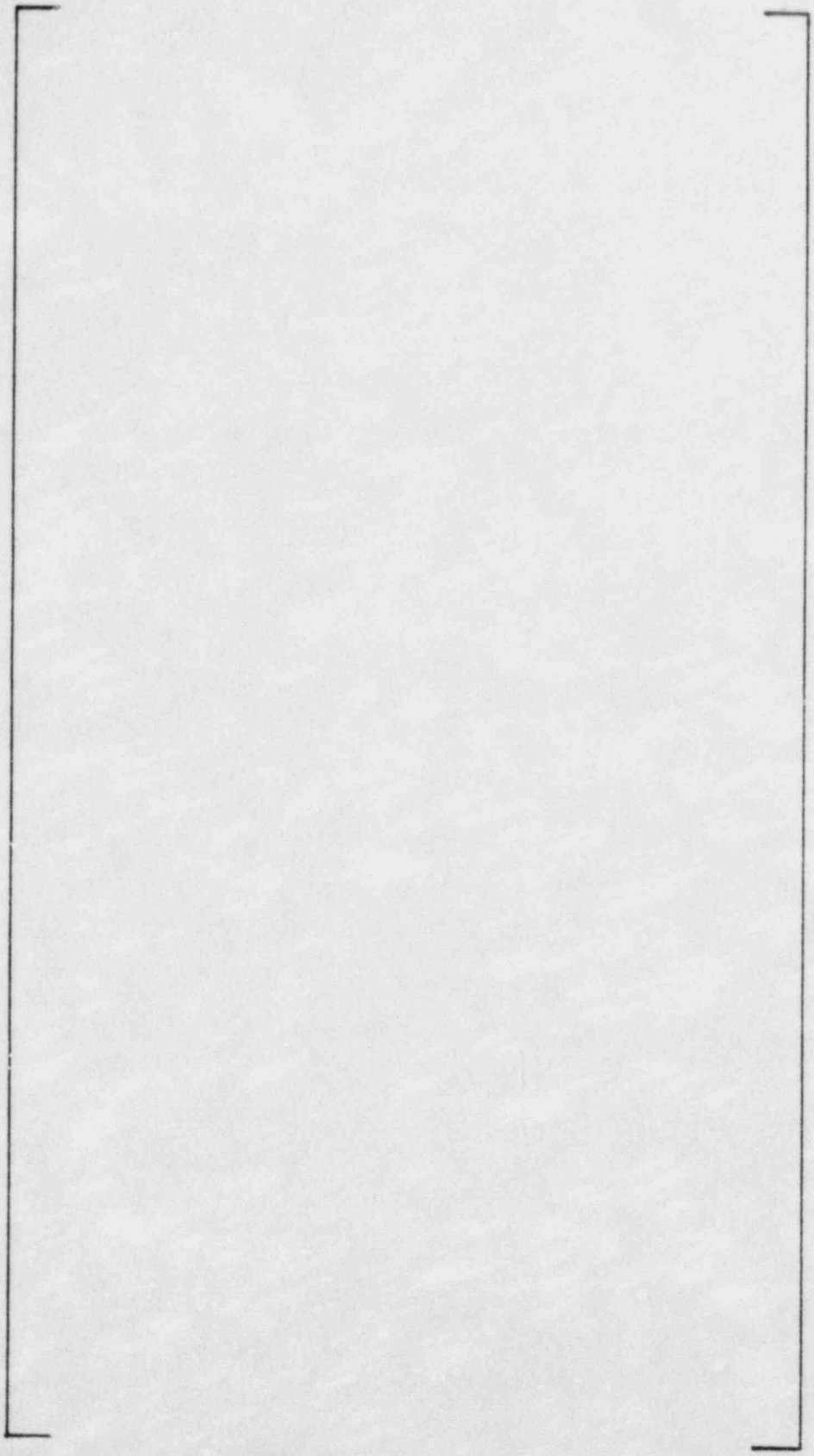

Figure 2.2-1. Block Diagram of LOFTRAN Solution Sequence

2.3 INTERNAL CODE CONTROL ROUTINES

Several subroutines are used for internal control of program execution. These routines are described below.

2.3.1 LOFTRAN

This subroutine is the main overlay (0,0) and is used to load the primary overlays in sequence. The subroutine also allocates space for the Blank Common block, which must be available to all the primary overlays.

2.3.2 INPUT

Controls execution of primary overlay (2,0), calling subroutines to read input, print the input summary, and initialize the system parameters.

2.3.3 LOFT

Controls execution of primary overlay (1,0) which performs all transient calculations and provides the standard format output. Also calls for reinitialization to asymmetric loop conditions if required.

2.4 SUBROUTINE FUNCTION SUMMARY

The subroutines in LOFTRAN are listed below, with a brief description of the function performed by each routine. A more detailed discussion of the models used in these subroutines may be found in later sections of this manual.

2.4.1 Primary Thermal - Hydraulic

1.	FCALC	reactor coolant flow
2.	CTRAN2	reactor vessel inlet mixing, core coolant
3.	PLTRAN	reactor vessel outlet mixing, upper vessel coolant
4.	HLTRAN	hot leg coolant, surgeline flow
5.	STMGEN	primary coolant in steam generator, cold leg
6.	PRESSR	pressurizer
7.	REVFLO	hot leg, steam generator, cold leg coolant during reverse flow

2.4.2 Steam Generator, Secondary Side

1.	SG2	Steam generator secondary side
2.	WTRVOL	Steam generator heat transfer area
3.	LEVEL	Steam generator water level

2.4.3 Core Kinetics

1.	NEUTRN	neutron kinetics, point kinetics model
2.	REACT	reactivity computation
3.	DKMOD	moderator reactivity
4.	QCALC	core heat flux

2.4.4 Control and Protection

1.	CONTRL	control and protection system
2.	SAFETY	safety injection system, EBS
3.	L106	control rod drive
4.	BCONC	boron and N-16 concentration

2.4.5 Auxiliary Thermal-Hydraulic

1.	SLUG	slug flow (mass and energy)
2.	MIXER	fluid flow through a plenum
3.	BSLUG	slug flow (boron or N-16)
4.	CRIT	Moody correlation
5.	ZALODEK	Zaloudek correlation
6.	RELIEF	ANS subcooled critical flow correlation
7.	HOMOSAT	ANS saturated critical flow correlation

2.4.6 Fluid Properties

1.	PROPV	water and steam properties: temperature,
2.	CALCSV	enthalpy, density, entropy, saturation
3.	TEMPH	properties
4.	ENTROP	

2.4.7 Input, Output, Initialization

1.	INPUT1	fluid systems, T-H design data
2.	INPUT2	initial conditions, forcing functions
	IN2OUT	
3.	INPUT3	nuclear kinetics parameters
4.	STSTVE	symmetric initialization
5.	FLOWIN	asymmetric initialization
6.	SUMMAT	summarization of output parameters

7.	UPRINT	output, standard format
8.	DOM, DOM2	output summary, user selected variables
9.	HINSER	initial inlet enthalpy

2.4.8 Controller Routines

1.	LAG	calculate lag-type response to input
2.	RATE	determine rate response to input
3.	LEAD	determine lead response to input
4.	PID	PID device
5.	VELLIM	rate limiter
6.	PROP	non-linear gain device
7.	MULT	multiplies or divides two input signals
8.	ADD	adds or subtracts two input signals
9.	CONV	time step conversion
10.	SCRAM	scram if trip setpoint reached

2.4.9 Numerical Manipulation

1.	QUAD	polynomial curve fit
2.	TERP	linear interpolation
3.	SINTEG	integration routine
4.	TBL	water table interpolation
5.	ACCUM1	adds elements in arrays
6.	ACCUM2	multiplies elements in arrays
7.	DVCKF	checking for infinite or indefinite result
8.	SCAN	put an array in ascending order

SECTION 3
PRIMARY THERMAL-HYDRAULIC ROUTINES

This group of routines is used to compute the thermal-hydraulic conditions throughout the primary side of the reactor coolant loops. The basic fluid system model is shown in Figure 3.0-1, with the LOFTRAN numbering nomenclature identified. The number of nodes used by the code may be specified by the user. Reactor coolant flow, enthalpy, and pressure are computed, with the following general assumptions:

1. Uniform pressure - in computing fluid density, spatial pressure gradients within the reactor coolant loops are neglected.

2. Homogeneous flow - fluid flow is assumed to be homogeneous, except in the reactor vessel upper plenum (PLTRAN).

3. Compresson and expansion are assumed to occur reversibly. (Where this occurs without heat or mass-energy transfer, the process is by definition isentropic.)

4. Control volume method - the fixed geometry method is used (finite difference approach), except in the pressurizer.

5. Reverse flow - not allowed in the core region or the loop with the pressurizer. All other loops will allow reverse flow.

+a,c

Figure 3.3-1 LOFTRAN Primary Flow Diagram

3.1 FCALC

Function: Compute reactor coolant flow into reactor vessel.

Three methods of flow computation are available:

1. Hyperbolic flow coastdown
2. Arbitrary function of time interpolated from table
3. Solution of momentum, pump kinetics equations

3.1.1 Hyperbolic flow coastdown:

$$F(LP) = F_o(LP) \text{ for } t \leq TPC(LP)$$

$$F(LP) = F_o(LP) \left[\frac{1. - CF}{1. + AF(t - TPC(LP))} + CF \right] \quad \text{for } t > TPC(LP)$$

where: $F(LP)$ = fraction of nominal loop volumetric flow
$F_o(LP)$ = initial loop flow fraction
AF, CF are constants input by the user
$TPC(LP)$ = time of start of flow coastdown in loop
t = time, seconds

3.1.2 Input Function:

F(LP) is obtained by linear interpolation from table input by user.

3.1.3 Solution of Momentum, Pump Kinetics Equations

The basic equation of motion is solved for flow, including effects of friction pressure losses, elevation (density) head, pump head, and momentum. Reactor coolant pump homologous curves are used to compute pump head and torque, and the momentum equation is solved for pump speed. This method of flow computation may be used for reactor coolant pump coastdown, locked rotor, and natural circulation flow calculations.

3.1.3.1 Solution of Fluid Momentum Equations:

The basic equation of motion, $f = m\frac{dv}{dt}$, is solved with the following assumptions:

1. The average flow between t and t + Δt equals the flow at t + Δt/2.

2. There is no change in density during the time step. The change in flow is uniform around the loop.

3. There is no change in elevation head during the time step.

The following equations are used to solve for flow:

$$f = m\frac{dv}{dt} \tag{1}$$

For one dimensional flow, equation (1) can be rewritten in terms of fluid properties for a control volume of fluid:

$$p = \frac{m}{A}\frac{dv}{dt} = \frac{V}{A^2 g_c}\frac{dw}{dt} \tag{2}$$

where: m = mass in control volume, lbm
A = area of control volume, ft^2
V = volume of control volume, ft^3
w = mass flow rate, lbm/sec
v = fluid velocity, ft/sec
g_c = proportionality constant, 32.174 lbm-ft/lbf-sec^2
p = net pressure applied to control volume, psf

Also, for a closed loop (such as a reactor coolant loop), the sum of the pressure drops around the loop must equal zero:

$$\oint PH + \oint BH - \oint FH - \oint MJ = 0 \tag{3}$$

where: PH = Pump head
BH = Buoyancy (elevation) head
FH = Frictional pressure drop
MH = Head due to fluid acceleration

The frictional pressure drop for each LOFTRAN section, j:

$$\Delta P_j = K_j W_j^2 / \rho_j \quad (4)$$

where: K_j = friction loss coefficient for jth section
W_j = mass flow for jth section, lbm/sec
ρ_j = fluid density in jth section, lbm/ft^3
ΔP_j = frictional pressure drop in jth section, psf

Note that when the pump kinetics model is used, the input frictional pressure drop terms will be rescaled during initialization to match the pump head computed from the homologous curves.

Buoyancy, or elevation head:

$$\text{Buoyancy} = \rho \, dZ$$

$$\oint_{BH} = \Delta P_{CORE} + \Delta P_{RVO} + \Delta P_{SGP} + \Delta P_{SGT} \quad (5)$$

$$\Delta P_{CORE} = Z_{CORE} (\rho_{RVI} - \rho_{CORE}) \quad (6)$$

$$\Delta P_{RVO} = Z_{RVO} (\rho_{RVI} - \rho_{RVO}) \quad (7)$$

$$\Delta P_{SGP} = Z_{SGP} (\rho_{SGO} - \rho_{SGI}) \quad (8)$$

$$\Delta P_{SGT} = Z_{SGT} [\bar{\rho}_2 - \bar{\rho}_1] \quad (9)$$

where:

ΔP_{CORE} = core pressure drop, psf

ΔP_{RVO} = pressure drop through upper core plenum, psf

ΔP_{SGP} = pressure drop through SG inlet plenum, psf

ΔP_{SGT} = pressure drop through steam generator tubes, psf

Z_{CORE} = core active fuel length, ft

Z_{RVO} = height from top of core to hot leg centerline, ft

Z_{SGP} = height from hot leg centerline to bottom of SG tube sheet, ft

Z_{SGI} = height of average SG tube from bottom of tube sheet, ft

ρ_{RVI} = density of fluid at reactor vessel inlet, lbm/ft^3

ρ_{CORE} = average density of fluid in core, lbm/ft^3

ρ_{RVO} = density of fluid at reactor vessel outlet, lbm/ft^3

ρ_{SGO} = density of fluid at primary outlet of steam generator, lbm/ft^3

ρ_{SGI} = density of fluid at primary inlet of steam generator, lbm/ft^3

$\bar{\rho}_1$ = density average 1st half of tube sections, lbm/ft^3

$\bar{\rho}_2$ = density average 2nd half of tube sections, lbm/ft^3

Pump head is computed from the homologous curves as described in the following section (3.1.3.2).

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3.1.3.2 Pump Kinetics

Transient operating conditions for the reactor coolant pump are derived from homologous curves for head and hydraulic torque, and from the pump motor speed-torque characteristics.

The pump head is interpolated from the homologous curves based on pump speed and flow, shown in Figure 3.1-1, and denormalized based on the homologous curve reference parameters.

The hydraulic torque on the pump is interpolated from the homologous curves as shown in Figure 3.1-2, and denormalized based on reference parameters.

The torque supplied by the pump motor is interpolated from a normalized speed-torque curve shown in Figure 3.1-3. The curve is denormalized to match the hydraulic torque and windage terms at t=0, providing initial steady-state operation at the input initial pump speed. Voltage to the motor is assumed constant at the initial value until time of coastdown (input TPC), and 0 following TPC. For pump startup, voltage is 0 before pump start (input TPS) and nominal after TPS.

The equation of motion is solved for the pump to give transient pump speed:

$$\frac{dS}{dt} = (T_m - T_h - WIND*S^2 - FRICT*S^{.5})/(PUMPI/g_c) \qquad (12)$$

where T_m = torque supplied by pump motor, ft-lb

T_h = hydraulic torque on impeller, ft-lb

WIND = pump windage term, ft-lb-sec^2

FRICT = pump friction term, ft-lb-sec$^{.5}$

PUMPI = pump rotating inertia, lbm-ft^2

g_c = proportionality constant, 32.174 lbm-ft/lbf-sec^2

S = pump speed, radians/sec

$\frac{dS}{dt}$ = transient pump speed, radians/sec^2

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Figure 3.1-1. Pump Homologous Curve Head – Speed – Flow

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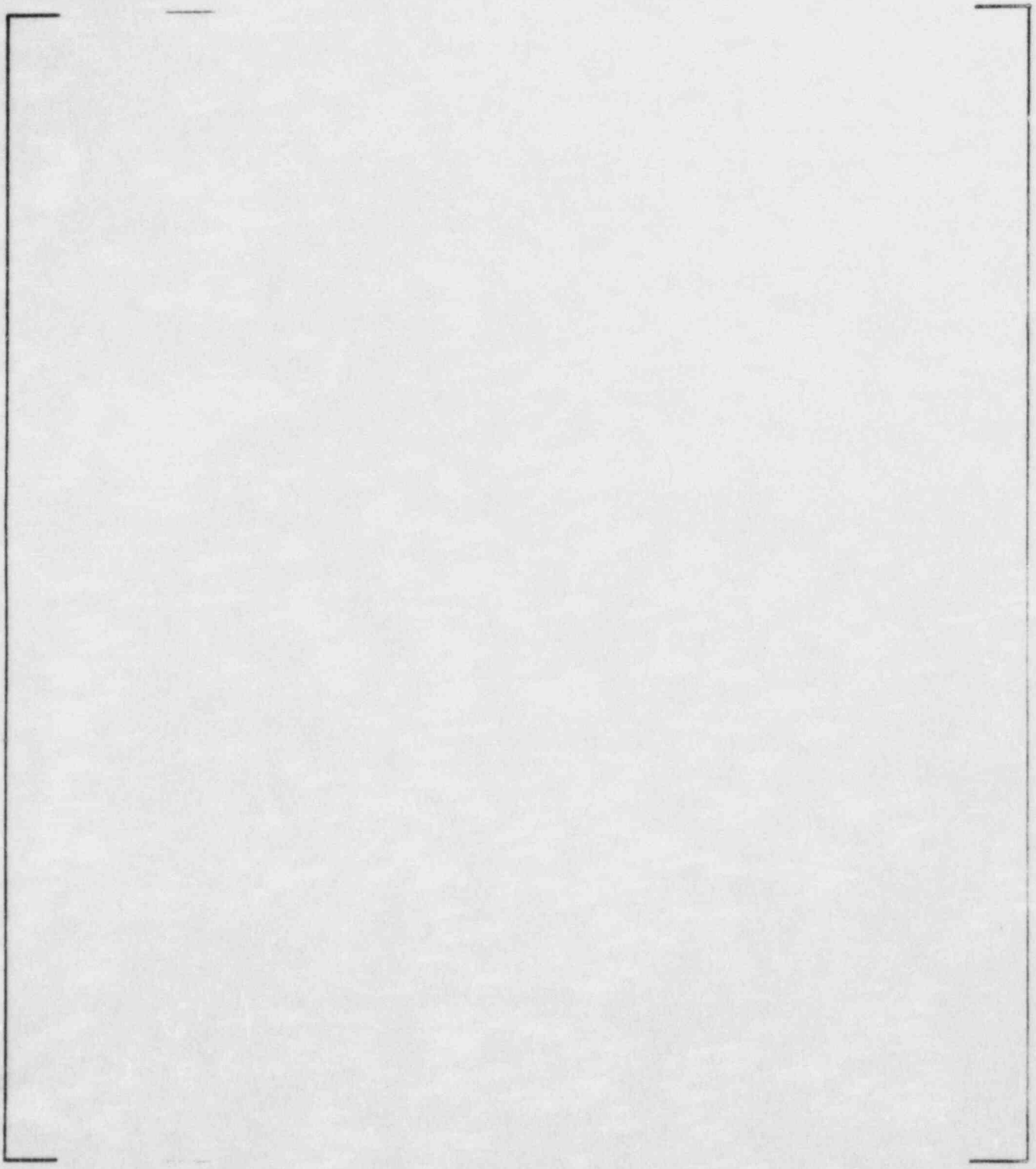
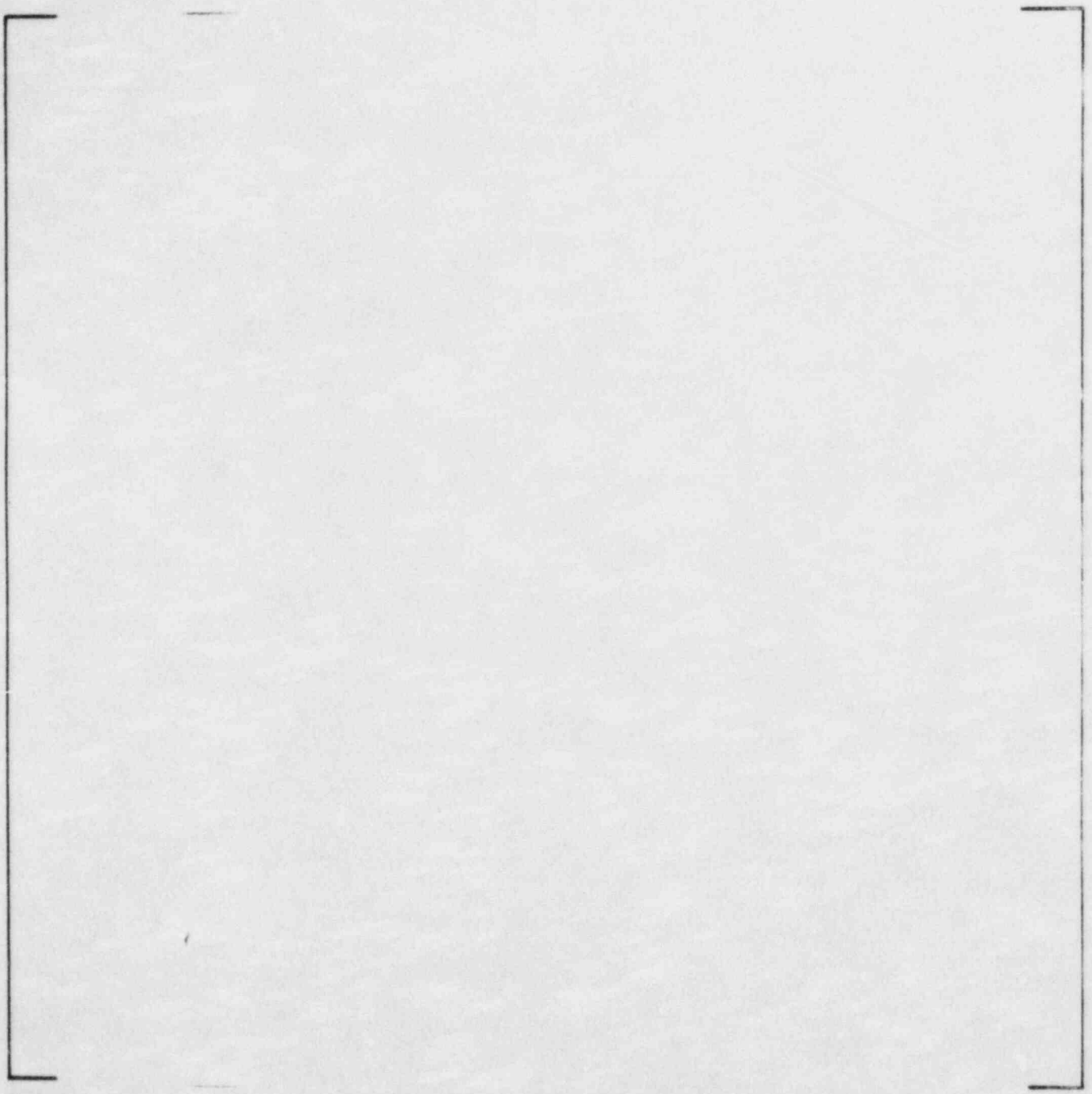


Figure 3.1-2. Pump Homologous Curve Torque – Speed – Flow



+a,(

Figure 3.1-3. Pump Motor Speed-Torque Curve

3.2 CTRAN2

Function:

1. Compute reactor vessel inlet enthalpy for each loop
2. Compute reactor vessel inlet plenum mixing
3. Compute flow and fluid conditions for inlet plenum and core sections.

Flow paths for this subroutine are shown in Figure 3.2-1.

Comments:

1. Inlet Enthalpy

   a. Volumetric flow into vessel has been previously computed by FCALC. Average inlet enthalpy is computed by checking the cold leg coolant volume that will move into the vessel during the present time step.

   b. The cold leg outlet enthalpy should equal the reactor vessel inlet enthalpy. This prediction is generally within 0.1 BTU/lbm.

   c. Error between h_{RVI} and h_{CLO} is corrected on next time step, so mass and energy are conserved. (h_{RVI} = reactor inlet enthalpy, BTU/lbm and h_{CLO} = cold leg outlet enthalpy, BTU/lbm).

   d. The error appraches zero as the cold leg transit time approaches an integral number of time steps.

   e. Safety injection flow is added to the cold leg flow at the safety injection nozzle location specified by input FCLSIS.

2. Reactor Vessel Inlet Plenum (RVI)

   a. RVI divided into radial sections, one per loop

b. Coolant flows into RVI from cold legs

c. Flow and enthalpy into each RVI section based on input mixing coefficient (AMXI)

d. Mixing between loops occurs between cold leg outlet and RVI, as shown in Figure 3.2-1.

e. Equal volumetric flow into each radial section of RVI

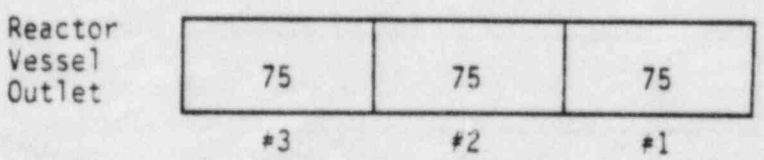
3. Core Coolant

a. core divided into radial sections, one per loop
b. up to 40 axial sections allowed
c. homogeneous slug flow axially through core
d. no flow between radial sections
e. heat flux is previously computed by QCALC
f. fixed fraction of bypass flow around core

3.2.1 Reactor Vessel Inlet Enthalpy

The cold legs are divided into equal-volume sections. For example, assume the cold leg is divided into 3 sections each consisting of 75 ft^3.

$$VCL = 225\ ft^3$$



If the amount of fluid entering the reactor vessel is less than or equal to the amount in section 3, then the enthalpy entering the reactor vessel inlet equals the enthalpy in section 3. In other words,

$$H_{CL} = H_3$$

If the volume of fluid entering is 150 ft^3, then the inlet enthalpy to the reactor vessel equals the average enthalpy of sections 2 and 3.

$$H_{CL} = \frac{M_2 H_2 + M_3 H_3}{M_2 + M_3}$$

If the volume of fluid entering is 100 ft^3, hen

$$H_{CL} = \frac{75\ H_3\ M_3 + 25\ H_2\ M_2}{75\ M_3 + 25\ M_2}$$

The mass entering the reactor vessel equals the mass of the cold leg flow plus the mass of the safety injection. The safety injection flow and energy are output from subroutine SAFETY. The outlet enthalpy is calculated as

$$H_{in} = \frac{H_{CL}\ M_{CL} + H_{SI}\ M_{SI} + \frac{1}{J} V\Delta P}{M_{CL} + M_{SI}}$$

where H_1, H_2, H_3 = the enthalpies of each cold leg section, BTU/lbm

M_1, M_2, M_3 = the mass of each cold leg section, lbm

H_{CL} = enthalpy entering the reactor vessel from the cold leg, BTU/lbm

M_{CL} = mass entering the reactor vessel from the cold leg, lbm

H_{SI} = enthalpy entering the reactor vessel from the safety injection system, BTU/lbm

M_{SI} = mass entering the reactor vessel from the safety injection system, lbm

H_{in} = enthalpy of the fluid entering the reactor vessel, BTU/lbm

$\frac{1}{J} V\Delta P$ = energy term due to compression, BTU

3.2.2 Reactor Vessel Inlet Mixing

A fixed fraction (FDEAD) of the flow from each cold leg is subtracted off and sent to the dead volume. The remaining flow into the reactor vessel is then split to provide equal volumetric flow to each radial section of the vessel inlet plenum. Flow to any loops with reverse flow is also accounted for.

The inlet mixing coefficient, AMXI, is related to the fraction of flow through a reactor vessel inlet nozzle which will flow through the radial core section nearest that inlet nozzle. For the case of symmetric flow (i.e., flow in all loops is equal), AMXI is defined as:

$$AMXI = \frac{NLOOP^* f_{mi} - 1}{NLOOP - 1} \qquad (1)$$

where: NLOOP = total number of reactor coolant loops

f_{mi} = fraction of flow from an inlet nozzle which flows up the corresponding core channel

AMXI = 0 represents perfect mixing

AMXI = 1 represents no mixing

For the more general case of different loop flows, mixing and flow redistribution will occur in the inlet plenum. In LOFTRAN, this mixing and redistribution is computed between the cold leg outlets and the inlet plenum volume such that equal volumetric flow enters each radial section of the inlet plenum, as shown in Figure 3.2-1.

The general equation used for computation of flow to the inlet plenum is:

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[ ] +a,c

Note that the volumetric flow to each radial section of the vessel is QV/NLOOP. Flow into any loops with reverse flow is computed at ρ_{avg}.

In a more detailed formulation, flow to the vessel inlet plenum is computed as follows:

[ ] +a,c

Enthalpy and boron concentration of the flow into the inlet plenum is computed with the same formulation as equation (3). The last term on the right on equation (3) represents an averaged property of the mixed portion of the flow; this averaged property is used for flow into any loops with reverse flow.

Fluid conditions in the inlet plenum region are computed using subroutine MIXER.

3.2.3 Heated Core Region

The heated core region is divided into radial flow channels of equal volume, one for each loop. Each radial channel may be divided into up to 40 axial sections of equal volume. Heat input to each section is computed in QCALC, and subroutine SLUG is used to compute flow and properties in each heated section.

A fixed flow fraction (FBO zero) bypasses the heated core region. No cross-flow is computed between the radial core channels.

+a,c

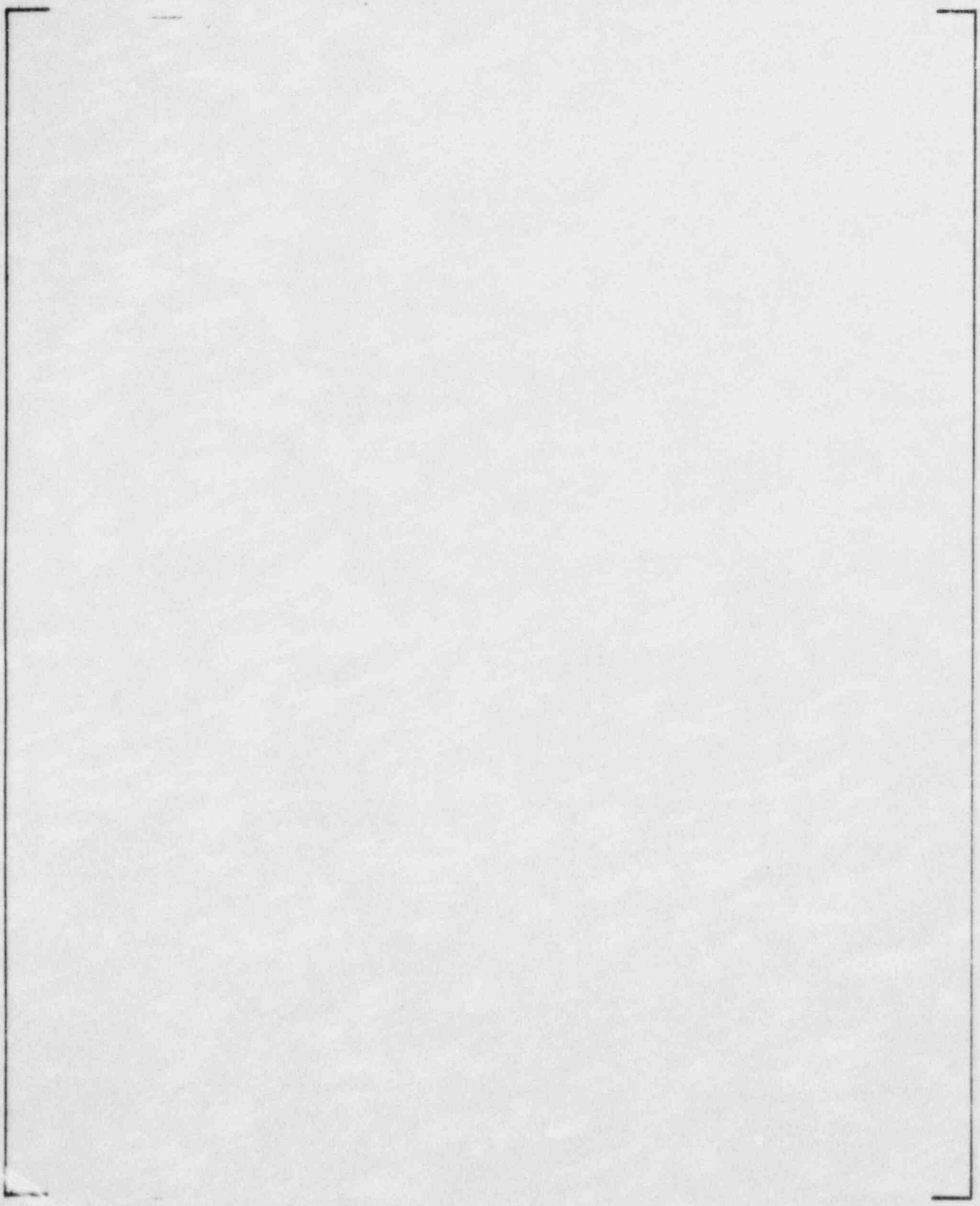


Figure 3.2-1. CTRAN2 Flow Paths

3.3 PLTRAN

Function:

1. Compute reactor vessel outlet mixing

2. Compute fluid conditions in upper core plenum, using multi-lump model for active and inactive regions

3. Compute fluid conditions in reactor vessel upper head region (dead volume)

Flow paths for this subroutine are shown in Figure 3.3-1.

3.3.1 Dead Volume

1. Because the dead volume is relatively stagnant (normal time constant of 3 to 10 minutes), it has a significant effect on steam break accidents and other abnormal cooldown transients. (Dead volume flashes and acts as a second pressurizer on steam breaks.)

2. A fixed fraction of vessel inlet flow (FDEAD) is diverted to the dead volume.

3. Perfect mixing is assumed in the vessel head.

4. No pressure gradient between upper head and inlet region.

5. Flow exists from the dead volume into the upper plenum mixing region.

6. Outflow from the dead volume is never less than 90 percent of inflow. (This simulates very slow refilling of head after boil off.)

3.3.2 Upper Plenum and Outlet Mixing

1. Active core flow and core bypass flow are homogenized in the upper core (inactive fuel rod region).

2. A fixed fraction of core outflow (AMXO) passes through a bypass volume (slug flow) directly to the hot leg nozzle. This flow fraction controls vessel outlet mixing.

3. The remaining water, plus a fixed fraction of the steam in core outflow, plus the flow out of the dead volume undergo perfect mixing in a mixing region of the plenum. Flow from loops with reverse flow also mixes in this region.

4. The remaining steam does not condense, but undergoes slug flow through the remaining plenum volume (forcing out water).

5. A fixed 2 percent of water leaving core passes through the inactive plenum region.

6. Steam separation is imperfect (some condensation occurs with slug flow, see section on SLUG).

The outlet mixing coefficient, AMXO, is related to the fraction of flow from a radial flow channel which flows into the corresponding hot leg. For the case of symmetric flow (i.e., flow in all loops is equal), AMXO is defined as

$$AMXO = \frac{NLOOP^* f_{mo} - 1}{NLOOP - 1}$$

where: NLOOP = total number of reactor coolant loops

f_{mo} = fraction of flow from a core channel which flows to the corresponding hot leg.

AMXO = 0 represents perfect mixing
= 1 represents no mixing

(Note: Consideration should be given to relative transport delay times through mixing and bypass volumes when choosing volume VPLBP.)

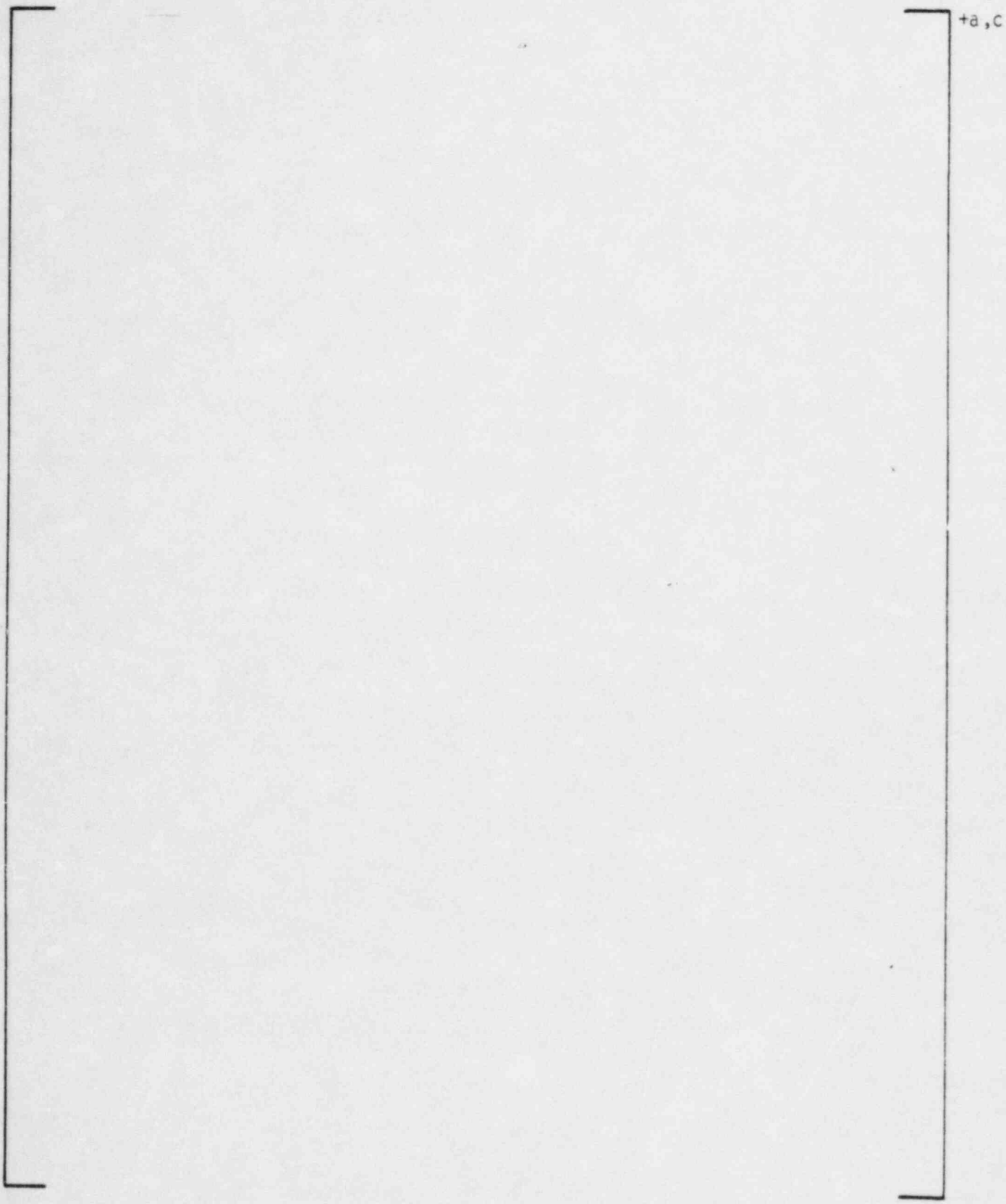
The total flow out of the reactor vessel is

$$W_{out} = W_{VPLS} + W_{PLMX} + \sum_{i=1}^{NLOOP} W_{PLBP}$$

where: W_{VPLS} = flow out of non-mixing plenum region, lbm/sec
W_{PLMX} = flow out of mixing region of plenum, lbm/sec
W_{PLBP} = flow out of plenum bypass, lbm/sec

The flow to each loop with forward flow is determined in HLTRAN. Flow from loops with reverse flow is determined in REVFLO. Reverse flow enters the plenum mixing region and undergoes perfect mixing.

6932-7



+a,c

Figure 3.3-1. PLTRAN Flow Paths

3.4 HLTRAN

Function:

1. Compute transient fluid conditions through hot leg piping.

2. Compute flow from reactor vessel into hot leg to achieve flow match for non-pressurizer loops.

3. Compute surge rate (insurge or outsurge) to pressurizer to achieve flow match in pressurizer loop (flow into vessel compared to flow out of cold leg).

4. Controls iteration through steam generator primary and through cold leg.

Comments:

1. Flow to non-pressurizer loops is computed first. The flow from the reactor vessel into each hot leg is determined such that flow out the cold leg equals the flow predicted from CTRAN2 to within 10 percent of volume error permitted for pressurizer convergence (0.1 x VPRCON). The flow to each hot leg is composed of W_{PLBP} plus any additional flow from W_{PLMX} and W_{VPLS} required to achieve flow balance. (See PLTRAN for variable definition).

2. Any remaining flow out of reactor vessel (from PLTRAN) is sent to the pressurizer loop. The pressurizer insurge or outsurge is determined such that flow out of the cold leg equals the flow predicted from CTRAN2 to within 0.1 x VPRCON.

3. Forward flow is always required in the pressurizer loop.

4. Homogeneous slug flow (see SLUG) is assumed in the hot leg piping.

5. Flow is in the same direction, either forward or reverse, through all sections of a loop (hot leg, steam generator, cold leg).

6. HLTRAN is used only for loops with forward flow. See section on REVFLO for reverse flow calculations.

7. Relief through a safety valve is allowed using QSVL(PSVL) input. Valve can be placed on either hot leg or cold leg of pressurizer loop.

3.5 STMGEN

Function:

Compute fluid conditions through steam generator primary side and through cold leg piping.

Comments or assumptions:

1. Perfect mixing is assumed in steam generator inlet and outlet plena. (See MIXER)

2. Homogeneous slug flow through the steam generator tubes and cold leg. (See SLUG)

3. Pressurizer spray flow is deducted from last cold leg section on the pressurizer loop.

4. Tube rupture option is available (flow through ruptured tube is deducted from either inlet or outlet header).

5. A water relief safety valve can be located on steam generator outlet header.

6. Use of multiple tube sections causes LMTD type response, rather than linearized UA.

7. Heat flux through tube regions is computed in SG2.

8. STMGEN used only for loops with forward flow. See REVFLO for reverse flow calculations.

3.6 REVFLO

Function:

Compute fluid conditions in loops with reverse flow for cold leg piping, steam generator primary, and hot leg piping.

Comments:

1. Homogeneous slug flow through cold leg piping, steam generator tubes, and hot leg piping. (See SLUG)

2. Perfect mixing in steam generator inlet and outlet plena. (See MIXER)

3. Flow into cold leg computed in CTRAN2.

4. Flow out of hot leg goes to reactor vessel upper plenum perfect mixing volume. (See PLTRAN)

3.7 PRESSR

Function:

1. Compute mass and energy balances in a two region pressurizer.

2. Compute surge line ΔP.

3. Determine pressure to give system volume balance.

4. Includes effects of heaters, spray, flashing, steam condensation and relief and safety valves.

5. Determines flow through safety valves.

Comments:

1. Water region and steam region are each assumed uniform (perfect mixing).

2. Surge line volume treated as part of the pressurizer water region.

3. On insurge, surge enthalpy is taken as instantaneous hot leg enthalpy; on outsurge, it is pressurizer water and/or steam enthalpy.

4. Several options are available for computing safety and relief valve relief rates.

5. Heaters:

   a. Proportional plus on-off control, with independent setpoints.

   b. No rate or reset controller.

   c. All heaters on for high water volume.

d. All heaters off for low water volume (heater decay energy added to steam phase).

e. Input heater time constant (time lag between electric power input to heaters and heat transfer).

f. A negative minimum heater capacity is recommended for simulation of heat losses.

6. Spray:

a. Spray enthalpy is taken as instantaneous cold leg enthalpy.

b. Spray time constant simulated as

$$\frac{e^{-\Delta t S}}{1 + (\tau - \Delta t)S}, \quad \Delta t = \text{time step}$$

The spray time constant is the amount of time required to open the pressurizer spray valve.

7. Steam condensation:

a. Steam specific volume is computed for both the quality and superheated regimes (as a function of the enthalpy computed from mass and energy balance), so condensation rate as used here applies to the rate at which condensed fluid entrained in the steam is added to the water phase.

b. Water drops are assumed to be uniformly distributed and falling at a constant velocity. (No spatial inventory of water drops).

c. Instantaneous condensation is simulated by an input time constant less than one time step.

d. Complete separation is simulated by a very large input time constant for condensation.

e. No steam condensation on walls is simulated.

8. Flashing:

a. Water specific volume is computed for both the subcooled and saturated regimes (as a function of enthalpy computed from mass and energy balance), so evaporation rate as used here applies to the rate at which evaporated fluid entrained in the water is added to the steam phase.

b. Steam drops are assumed to be uniformly distributed and rising at a constant velocity. (No spatial inventory of steam drops).

c. Instantaneous flashing is simulated by an input time constant less than one time step.

d. No water evaporation on the walls is simulated.

9. Pressure convergence:

a. Iteration through the entire reactor coolant system and pressurizer continues until fluid volume is within VPRCON ft^3 (input) of system volume.

This means that the amount of fluid that is contained in the total RCS is compared to the volume of the RCS. If the fluid volume is greater, for instance, than the RCS volume, then a higher pressure must be used to squeeze the fluid into the RCS volume.

b. Maximum of 50 iterations (run - not job - terminated).

c. Noise is anticipated for VPRCON greater than 0.01 percent of system volume.

d. Convergence is also satisfied if pressure correction is less than 0.00005P.

10. Although the calculations for the pressurizer spray and heater control system are performed in subroutine PRESSR, a detailed description is included in the CONTRL section of this manual, along with the other control systems simulated.

Equations:

Pressure:

A surge into or out of the pressurizer is computed in HLTRAN. The surge will cause a change in pressurizer pressure:

$$P_{pr} = P_r + K \dot{W}_{su} |Q_{su}| \qquad (1)$$

where: P_{pr} = pressurizer pressure, psia
P_r = RCS pressure, psia
K = surge line friction loss coefficient
$\dot{W}_{su}$ = surge mass flow rate, lbm/sec
Q_{su} = surge volumetric flow rate, ft^3/sec

Conservation of Mass:

[+a,c]

[ ] +a,c

Conservation of Energy:

[ ] +a,c

[+a,c]

Flashing and Condensation:

[+a,c]

[+a,c]

Volume Convergence:

[+a,c]

[+a,c]

SECTION 4

STEAM GENERATOR

4.1 SG2

Function:

1. Compute steam and feedwater flow, including break flow.
2. Compute heat flux between primary and secondary.
3. Compute fluid conditions in secondary side of steam generator.

Comments:

1. The basic mass - energy model is a one lump model valid for a saturated mixture of steam and water.

2. Steam pressure and enthalpy are computed as saturation properties of temperature (or steam enthalpy vs. time may be input).

3. The entire tube metal heat capacity is assumed to be at the secondary steam temperature.

4. Relief through safety valves is simulated.

5. Numerous options for steam and feedwater flow are available, including computation of flow through pipe breaks.

6. Option available for tube rupture break flow computed using ZALODEK correlation subroutine.

7. A preheater model option is available.

4.1.1 Primary to Secondary Heat Transfer

The primary side of the steam generator tube region is divided into a number of sections. Heat flux from each section is computed as:

$$Q_j = UA(T_{pj} - T_s) \quad (1)$$

where: Q_j = heat flux in section j, BTU/sec
UA = overall heat transfer coefficient, BTU/sec-°F
T_{pj} = primary side temperature in section j, °F
T_s = steam temperature, °F

Several options are available for computation of UA, determined by input MODEUA. The formulation recommended for general use is MODEUA = 3, where UA is computed as a function of primary flow, heat flux, and secondary side pressure and mass, initialized to match nominal input conditions. The calculations for MODEUA = 3 are described below.

Equations: (MODEUA = 3)

[+a,c]

[+a,c]

VSTB may be input as a constant (via VSTUBE), or computed internally based on steam generator geometry and fluid properties. Refer to section on subroutine WTRVOL for description of internal computation of VSTB.

R/R_{nom} is computed by summing the tube metal resistance, primary film resistance, and secondary film resistance:

[+a,c]

+a,c

+a,c

+a,c

[+a,c

] +a,c

4.1.2 Steam Generator Fluid Conditions

The principle assumption is that the secondary fluid is always at saturated conditions. Thus, if the fluid temperature is known, pressure, enthalpy, and specific volume can be computed as saturation properties of water.

Equations:

Conservation of Energy:

$$\frac{d(Mu)}{dt} = q_{sg} + w_{fw}h_{fw} - w_s h_s \qquad (1)$$

Conservation of Mass:

$$\frac{dM}{dt} = w_{fw} - w_s \qquad (2)$$

where: M = steam generator fluid mass, lbm

q_{sg} = heat flux into steam generator, BTU/sec

w_{fw}, w_s = feedwater, steam flow, lbm/sec

h_{fw}, h_s = feedwater, steam flow enthalpy, BTU/lbm

u = steam generator fluid average internal energy, BTU/lbm

[+a,c

] +a,c

+a,c

+a,c

4.1.3 Preheat Model

The preheater secondary side is a multi-node, fixed geometry control volume model. Homogeneous slug flow (see SLUG) is assumed. Heat transfer

correlations for forced convection and for boiling heat transfer are included. Up to ten nodes may be used for accurate transient response.

Equations:

Heat transfer is computed similar to the method described in Section 4.1.1:

$$Q_j = UA_j \, (T_{pj} - T_{sj}) \qquad (1)$$

where: T_{sj} = secondary side temperature in section j
T_{pj} = primary side temperature in section j
UA_j = overall heat transfer coefficient for section j

For MODEUA = 3, UA is computed as in equation (2) of Section 4.1.1, and R/R_{nom} as in equation (5) of Section 4.1.1.

+a,c

[+a,c]

4.1.4 Steam Flow

Steam flow from each steam generator is computed from the steamline pressure drop to the common steam header:

$$WS_j = [(PS_j - PH)\ \rho_j/K]^{1/2}$$

where: WS_j = steam flow from loop j, lbm/sec
PS_j = steam pressure in loop j, psia
PH = pressure in steam header, psia
ρ_j = steam density, lbm/ft^3
K = pipe friction loss coefficient

The header pressure is computed at each time step so that the total steam flow from all loops equals the input value.

4.1.5 Feedline Flashing

The feedline flashing calculation is based on a single node, homogeneous, isolated volume. Flowrate and pressure are computed at each time step using an iterative procedure. First, a new feedline pressure, PF, is assumed. The SLUG routine is then used to compute the flowrate, WF, out of the feedline flashing volume. This flowrate is then used in the pressure loss equation to determine the feedline pressure:

$$PF_1 = PS + K\ \frac{WF^2}{\rho} \qquad (1)$$

where: PS = steam generator pressure
ρ = feedwater density
K = feedline pressure loss coefficient

The iteration is continued until the assumed pressure PF, and the calculated pressure, PF_1, agree.

4.2 WTRVOL

Function:

Computes secondary side steam generator water volume at which primary to secondary heat transfer will begin to decrease.

Comments:

1. The results calculated by WTRVOL must be verified by use of a more detailed steam generator model. The user is cautioned to be certain that the input and methods used by WTRVOL are valid for the range of each transient where this option is used.

2. Calculations are based on an input riser exit quality assuming a linear enthalpy rise with elevation in the tube bundle.

3. Steam generator geometry data is determined by the input SGTYPE.

4. Refer to Section 4.1.1 for use of VSTB. (Input VSTUBE)

5. See Figure 4.2-1 for diagram of model.

Equations:

Secondary side water volume at which heat transfer begins to decrease is computed as:

$$VSTB = WVPR + WVDC + WVBL + WVR \quad (1)$$

where:
- WVPR = volume of water in subcooled (preheat) region
- WVDC = volume of water in downcomer
- WVBL = volume of water in boiling region
- WVR = volume of water in riser

The required water volumes in the individual sections are computed as follows:

$$WVPR = WLPR * A_{PR} \quad (2)$$

$$WVDC = C_1 * WLDC + C_2 \quad (3)$$

$$WVBL = (1-F)(1-\alpha_B)\ V_B \quad (4)$$

$$WVR = (1 - \alpha_R)\ V_R \quad (5)$$

$$F = (h_f - HTS)/(HSV-HTS) \quad (6)$$

$$WLPR = F * D_T \quad (7)$$

$$WLDC = WLPR + (1 - F)(1 - \alpha_B)\ D_T + (1 - \alpha_R)\ D_R \quad (8)$$

$$WRW = WS(1 - X)/X \quad (9)$$

$$HWV = h_f + X\ h_{fg} \quad (10)$$

$$HTS = [h_f * WRW + HFW * WFW]/[WRW + WFW] \quad (11)$$

where:

X = riser exit quality at which heat transfer area begins to decrease (input as negative VSTUBL)

α_R = riser void fraction (computed from X based on Armand void fraction correlation)

α_B = boiling region void fraction (computed from X, Armand)

WS = steam flow (lbm/sec)

WRW = recirculation water flow (lbm/sec)

WFW = feedwater flow (lbm/sec)

HFW = feedwater enthalpy (BTU/lbm)

HSV = enthalpy at riser exit (BTU/lbm)

HTS = enthalpy at tube sheet (BTU/lbm)

WLDC = height of water in downcomer (ft)

WLPR = height of subcooled water in bundle region (ft)

F = fraction of tube height covered by subcooled water

D_T = height from tube sheet to top of tubes (ft)

D_R = height of riser region (ft)

V_B = volume of tube bundle region (ft^3)

V_R = volume of riser region (ft^3)

A_{PR} = cross sectional area in subcooled (preheat) region (ft^2)

C_1, C_2 = geometry constants

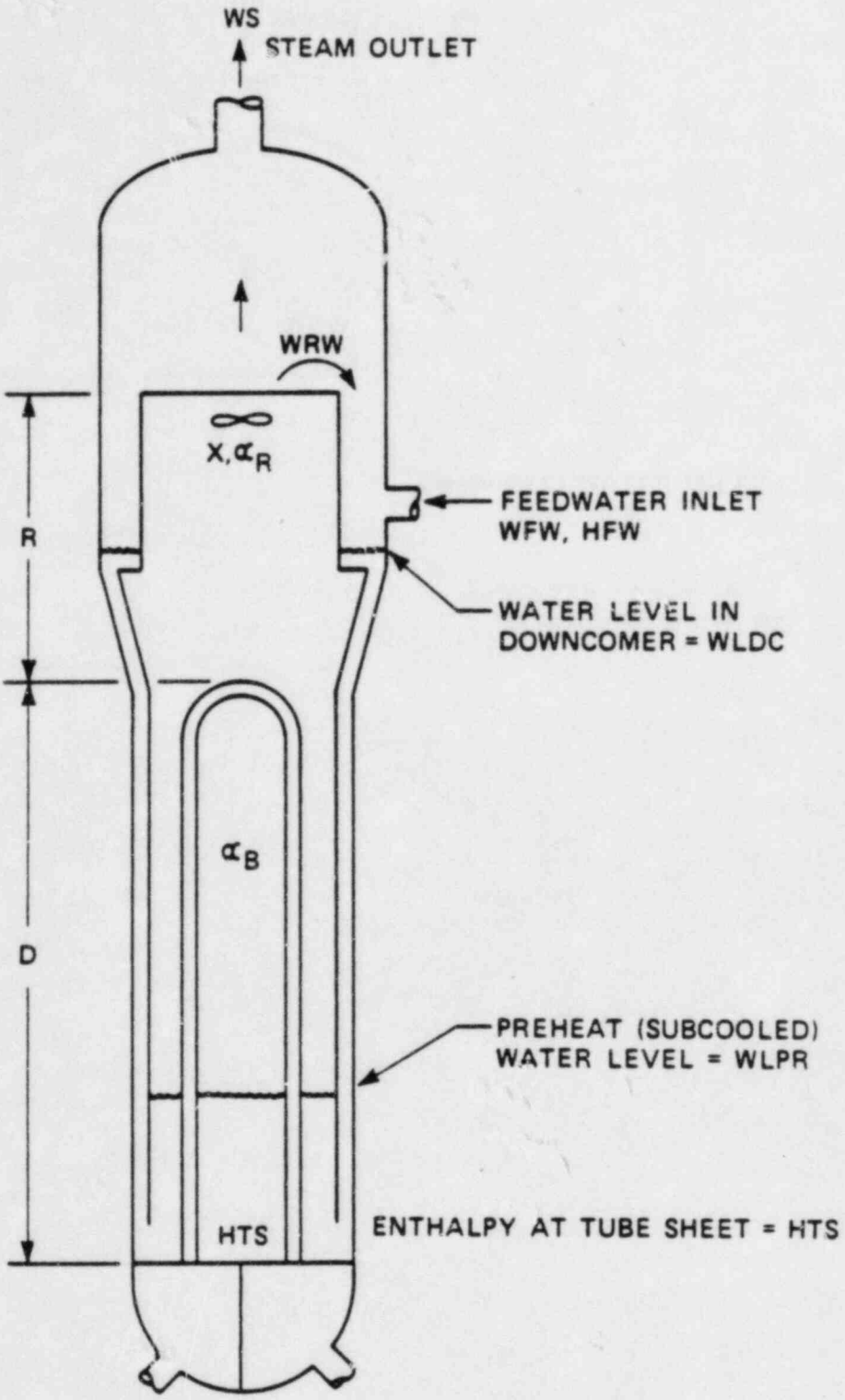


Figure 4.2-1. VSTUBE Model

4.3 LEVEL

Function:

Calculates steam generator water level.

Comments:

1. Normal water level is assumed to be a function of boiling power.

2. Boiling power is total power transferred to the shell side less the power required to heat incoming feedwater to saturation.

3. $\frac{\partial L}{\partial M}$ is assumed to be constant above 20 percent power and varies linearly from 0 percent to 20 percent power.

4. Appropriate constants for different steam generators are stored in the program (determined by input SGTYPE). These constants were selected such that in steady-state, the level agrees with calculations from more detailed steam generator codes.

Equations:

In order to compensate for the downcomer holdup time (about 10 seconds), the transient calculation of preheat power is

$$\text{Preheat power} = \frac{W_{fw}}{(1 + \tau S)^2}\left(h_f - \frac{h_{fw}}{(1 + \tau S)^2}\right)$$

where:

τ = 5 seconds
W_{fw} = is feedwater flow, lbm/sec
h_f = enthalpy of saturated liquid, BTU/lbm

h_{fw} = enthalpy of feedwater, BTU/lbm
s = Laplacian operator

In steady-state, preheat power = $W_{fw} (h_f - h_{fw})$

The water level is calculated from the difference between actual liquid inventory and liquid inventory at normal water level

$$level = NWL + \frac{\partial L}{\partial M} (M - M_{NWL})$$

where:

NWL = height of water in the downcomer, ft
M_{NWL} = mass in normal water level per steam generator, lbm
M = steam generator mass at time t, lbm
$\frac{\partial L}{\partial M}$ = rate of level change with mass, ft/lbm

SECTION 5

NEUTRON AND FUEL KINETICS

Sections 5.1 through 5.3 describe the point kinetics transient nuclear power model used by LOFTRAN, and 5.4 describes the one lump fuel model used with the point kinetics option to compute fuel to water heat flux.

5.1 NEUTRN

Function:

1. Computes instantaneous neutron power.
2. Computes instantaneous decay heat power.
3. Computes expected change in fuel temperature during the present time step (for Doppler feedback).

Comments:

1. Standard point kinetics equations are used.

2. Six delayed neutron groups are used.

3. Five decay heat groups are used. Default values are provided to match ANS finite model +20 percent.

4. LOFTRAN can use, as a maximum, ten nuclear time steps per thermal and hydraulic time step. (Reactivity is checked twice per nuclear time step).

5. Linear extrapolation of coolant density is used (for reactivity calculation).

6. Single lump fuel model is used for the calculation of change in average fuel temperature (uses a variable fuel time constant calculated in QCALC).

7. Source term and prompt neutron cycle time are included in the equation.

8. Input core heat flux, reactivity or nuclear power as a table QNT(TN), depending on the option selected on the master control card.

9. Several elements of the Control and Protection system are programmed in NEUTRN, including the power mismatch channel of the rod control system, and the overpower nuclear trip. For consistency, these are discussed in CONTRL.

Equations:

The basic point kinetics equations are:

$$\frac{dN}{dt} = \frac{\delta K - \beta}{\ell^*} N + \sum_{i=1}^{6} \lambda_i C_i + S \qquad (1)$$

$$\frac{dC_i}{dt} = \frac{\beta_i}{\ell^*} N - \lambda_i C_i \qquad (2)$$

$$\sum_{i=1}^{6} \beta_i = \beta \qquad (3)$$

where:
- N = neutron population
- t = time
- δK = excess reactivity
- β_i = effective ith group delayed neutron yield, fraction
- ℓ^* = mean prompt neutron lifetime
- λ_i = delayed neutron precursor decay constant for ith group
- C_i = delayed neutron precursor population for ith group
- S = source term

+a,c

[ ] +a,c

Decay heat is computed from a five group precursor model similar to equation (2). For decay heat the solution is expressed as:

[ ] +a,c

Total nuclear power is computed as the sum of neutron power and decay heat:

$$q(t + \Delta t) = N(t + \Delta t) + D_i(t + \Delta t) \qquad (11)$$

Again, note that LOFTRAN may use up to 10 nuclear time steps per thermal-hydraulic time step. The average nuclear power for a thermal-hydraulic time step is computed by Simpson's one third rule.

5.2 REACT

Function:

Compute reactivity as a function of fuel temperature, coolant density, boron concentration, rod position and time after scram.

Comments:

1. Input moderator density and boron worth coefficients are fit to a quadratic (see Section 11.1) which are then integrated by DKMOD.

2. Core space-average density and mass averaged boron are used, weighted between core channels by an input weighting function.

3. The fuel Doppler effect is simulated as a power defect (integral of the power coefficient) rather than as a fuel temperature effect.

4. Additional reactivity can be input.

5. Variable rod worth can be input (using AROD found in namelist RECOCD).

6. Linear interpolation of scram reactivity insertion as a function of time after trip. Here, the rods should be fully inserted at TDKS(10) (variable found in namelist RECOCD).

Equations:

Total reactivity is computed as

$$\delta K = \delta K_m (\bar{\rho}_{cw} \bar{C}_b) + \delta k_q + \alpha_f T_{CW} + \delta k_R + \delta K(t) + \delta K_s(t) - \delta K_o \quad (1)$$

where

δK = the change in reactivity (equals the sum of elements contributing to reactivity minus the initial reactivity)

δK_m = moderator reactivity contribution (see DKMOD)

$\bar{\rho}_{cw}$ = weighted average core fluid density, gm/cc

$\bar{C}_b$ = weighted average core boron concentration, ppm

δk_q = power defect (Doppler contribution only) =

$$\int_o^q \alpha_q \, dq$$

The Doppler only power defect may be computed by three different options, determined by input IDOP in namelist RECOCD. These options are further discussed on the following page.

$\alpha_f{}^T{}_{CW}$ = Doppler effect correction for change in fuel temperature at constant power due to coolant temperature change. (Alternately, this term may also be used if desired to simulate a constant moderator temperature coefficient).

δk_R = control rod reactivity contribution

$$\int_{R^\circ}^R \alpha(R) \, dR$$

$\delta K(t)$ = input reactivity contribution

+a,c

Three options are available for computation of the Doppler Power defect used in equation (1):

[+a,c]

[ ] +a,c

5.3 DKMOD

Function:

Perform three-dimensional integration along density and boron planes to determine change in reactivity caused by change in moderator condition.

Comments:

This function computes the moderator reactivity as

$$\delta K_m = \delta K\,(\bar{\rho}_{cw}, \bar{C}_B) = \int_{\rho_{ref}}^{\bar{\rho}_{cw}} \alpha_\rho\,(\rho, \bar{C}_B)\,d\rho + \int_0^{\bar{C}_B} \alpha_B\,(\rho_{ref}, C)\,dC$$

where

$\bar{\rho}_{cw}$ = average moderator density

$\bar{C}_B$ = average boron concentration in core, ppm

ρ_{ref} = reference moderator density for which boron worth is input

Quadratic polynomials are fit to the input density and boron coefficients and these quadratics are then integrated.

5.4 QCALC

Function: Compute reactor core fuel rod temperature and heat flux.
Compute RCS thick metal temperature and heat flux.

Comments:

1. Up to 40 axial nodes can be used.

2. Only one radial fuel node is simulated in each core channel.

3. The routine assumes that the overall thermal conductivity from fuel to coolant is a function of local radially averaged fuel temperature and this function is the same during a transient as in the steady-state.

4. C_p (of UO_2) is assumed to be a function of local $\bar{T}_f$.

5. The clad temperature is assumed to be equal to the coolant temperature (actually, it lags by one time step).

6. Parabolic axial power distribution with peak to average power = 1.5.

7. Local axial coolant temperature used for heat transfer is estimated by linear extrapolation in time.

8. It is assumed that 97.4 percent of nuclear power is generated in the fuel, 2.6 percent is generated in the coolant.

9. This program also computes the fuel time constant (and rate of change of the fuel time constant with fuel temperature) for use by neutron kinetics program in estimating change in Doppler reactivity.

10. One radial fuel node is not at all accurate for rod ejection. Start-up accidents are questionable. For other transients, one radial fuel lump should be generally adequate.

Equations:

The general differential equation for heat transfer from the fuel (one radial node) to the coolant can be written as:

$$m_f C_{pf} \frac{dT_f}{dt} = kq - UA(T_f - T_{cw}) \qquad (1)$$

where:
- m_f = fuel mass, lbm
- C_{pf} = fuel specific heat, BTU/lbm-°F
- T_f = fuel temperature, °F
- k = fraction of heat generated in fuel (=.974)
- q = total nuclear power, BTU/hr
- UA = heat transfer coefficient, BTU/hr-°F
- T_{cw} = coolant temperature, °F

The general solution to this equation, in finite difference form, is:

[ ]+a,c

+a,c

+a,c

Heat transfer between reactor coolant system water and thick metal is computed based on a one node heat transfer model of the same general form shown in equation (1). Heat transfer is computed for each LOFTRAN node as:

+a,c

SECTION 6
CONTROL AND PROTECTION

This section contains descriptions of the control and protection features which are simulated in LOFTRAN. Several of the control features described in Section 6.1, CONTRL, are actually programmed in other LOFTRAN subroutines, but are documented here for ease of reference. For example, the pressurizer spray and heater control is performed in PRESSR, and rod control is performed in NEUTRN.

Note that rod control and steam dump work only when MODEST $\geq$ 11, where a turbine load signal is generated. A trip will not change the actual steam flow, but will change the turbine load signal.

6.1 CONTRL

6.1.1 Rod Control System

Refer to diagram in Figure 6.1-1

The rod control system function is to provide automatic control of the RCC assemblies during power operation of the reactor. The rod control system is capable of restoring the reactor coolant average temperature to within the programmed temperature deadband, following changes in load. The basic reactor control system in LOFTRAN consists of two channels, which are temperature ($T_{AVG} - T_{REF}$), and power mismatch ($Q_N - Q_{TU}$).

The T_{AVG} signal (indicated T_{AVG}) is the auctioneered T_{AVG} (instantaneous average of hot leg and cold leg temperature), minus an input temperature error (TAVGER), with a first order lag (filter). Lead/lag compensation (with second-order lag) is then applied to the T_{AVG} signal. This signal is compared with a reference temperature (T_{REF}) signal which is programmed as a function of the turbine power. The turbine power is calculated from a first order lag on turbine steam flow. In the power mismatch channel, the deviation

between Q_{TU} and Q_N is passed through a rate/lag unit (impulse or high-pass filter), a non-linear gain unit and a variable gain unit. The power mismatch provides fast response to load perturbations and stability for the high gain temperature control loop.

The total error signal sent to the rod speed program is the sum of the temperature error and power mismatch error. The rod speed program creates a rod speed profile which is a function of the total error and input parameters for minimum rod speed, rod speed proportional band and maximum rod speed (the speed in LOFTRAN is given in steps/min). A positive error causes rod insertion and a negative error signal causes rod withdrawal.

6.1.2 Steam Dump Control

Refer to diagram in Figure 6.1-2.

Steam dump simulated in LOFTRAN is used to:

1. Permit the nuclear plant to accept large load rejections without incurring a reactor trip.

2. Remove stored energy and residual heat following a reactor trip without actuation of the steam generator safety valves with the plant at equilibrium no-load conditions.

3. Permits control of the steam pressure at no-load conditions.

The first two functions are controlled by T_{AVG} and the third by steam pressure. The T_{AVG} control mode is commonly termed as automatic control, with pressure control being considered as manual.

The lead/lag compensation for the T_{AVG} signal is to compensate for the lags in the plant thermal response and in valve positioning. For a sudden load decrease, T_{REF} is immediately decreased and T_{AVG} tends to increase, thus creating an error signal (TERRS1). T_{AVG} and T_{REF} signals are the same as

those used in the rod control system. LOFTRAN uses four banks of steam dump valves which are governed by the error signal. The valves begin to open when the error is greater than the deadband and stops when the error exceeds the proportional band. The valves are assumed to modulate open until the signal out of the proportional controller exceeds the trip open setpoint. The modulate time is assumed to be 20 seconds and the trip open time is 3 seconds.

For pressure control the turbine power (Q_{TU}) is lag compensated and programmed (proportional gain unit) to generate a steam pressure program setpoint. This setpoint pressure is subtracted from the steam pressure, causing a pressure error signal. This error is sent through a PID unit. The signal output is then converted from a pressure to an equivalent temperature signal which regulates the valve opening. Only the first two banks are used for pressure control on steam dump.

6.1.3 Feedwater Control

This system is currently not in use in LOFTRAN.

6.1.4 Pressure Control System

Refer to diagram in Figure 6.1-3.

The pressure control system maintains the reactor coolant system pressure at a predetermined value by employing heaters, spray and power relief valves. The heaters increase the pressurizer water temperature and increase pressure by boiling water. Spray located at the top of the pressurizer operates to condense steam and decrease pressure. Power relief valves have a larger effect than pressurizer spray in reducing pressure by relieving steam.

Proportional control, not rate and reset, is simulated. The pressure error controls proportional spray, proportional heaters and on-off heaters. The pressure error is the difference between instantaneous and nominal, not initial, pressurizer pressure. If an initial pressure different than nominal is input, the pressure control will act to restore nominal pressure.

A negative minimum proportional heater capacity may be input to simulate pressurizer heat losses. The on-off heaters are actuated by high pressurizer water volume as well as by low pressure. Heat from the heaters is added to the pressurizer water, or pressurizer steam if the pressurizer is nearly empty.

Proportional spray demand is multiplied by the square of relative reactor coolant flow to simulate the reduction in spray driving head during loss of flow. Spray enters the pressurizer at the enthalpy of fluid in the last cold leg section at the beginning of the current time step. It is added to the pressurizer steam, or to pressurizer water if the pressurizer is more than 90 percent full.

Pressurizer safety and relief valves are simulated by input tables. These tables specify the relief rate for water and steam as a function of pressure. If the pressurizer is full, the water relief table is used and water discharged is deducted from pressurizer water inventory. Otherwise, the steam relief table is used and the pressurizer steam inventory debited accordingly.

6.1.5 Protection System

Various protection systems are used as a safeguard to the NSSS system. These protection systems include reactor and turbine trips, safety injection actuation, and steamline isolation.

The reactor trips include:

1. low reactor coolant loop flow
2. nuclear overpower
3. high pressurizer level
4. high pressurizer pressure
5. low pressurizer pressure
6. overtemperature ΔT
7. overpower ΔT
8. overtemperature (N-16)
9. overpower (N-16)
10. low DNBR (N-16)
11. high kw/ft (N-16)

12. low steam generator level
13. safety injection
14. turbine trip
15. manual

The delta-T trip functions and compensation are shown in Figure 6.1-4; the N-16 trip functions are shown in Figure 6.1-5 (TBX Protection System) and Figure 6.1-6 (Integrated Protection System).

Safety injection is actuated by:

1. low pressurizer pressure
2. low pressurizer pressure and level
3. low steam pressure (lead/lag)
4. manual

Steamline isolation is actuated by:

1. low steam pressure (lead/lag)
2. high steam pressure rate (rate/lag)
3. manual

Turbine trips are caused by:

1. reactor trip
2. high steam generator level
3. manual

For each of the above trips, the time of trip signal actuation is computed if the signal crosses the setpoint (in the direction to cause trip) during the last time step.

For trips which have logic made up of inputs from different loops (e.g. overtemperature and N-16 trips) an input option allows trip conditions from any loop to be disregarded. This option can be used to simulate a bypassed or failed protection channel.

6.1.6 DNB Calculation

The DNB calculation performed by LOFTRAN is a partial derivative approximation of the DNB core limit lines. Care should be taken to choose the partial derivatives over the same parameter range as the transient to be analyzed.

+a,c

+a,c

Figure 6.1-1. Rod Control System

+a,c

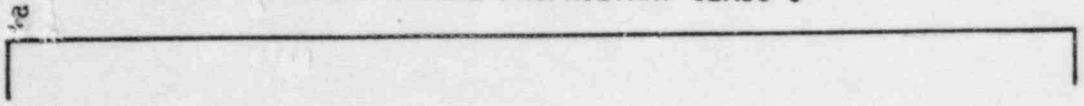
Figure 6.1-2. Steam Dump Control

Figure 6.1-3. Pressure Control System

Figure 6.1-4. ΔT Trip Functions

Figure 6.15 N 16 Overtemperature, Overpower Trip Functions (TBX Protection System) GAIN (12) = 1

2,a,c



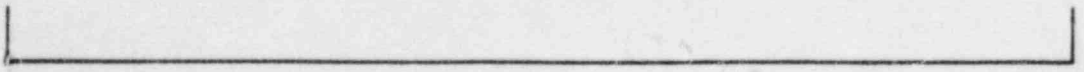


Figure 6.1 6. N 16 Low DNBR, High KW/FT Trip Functions (Integrated Protection System) GAIN (12) ≠ 1

6.2 SAFETY

Function: Compute safety injection flow from SIS, EBS and accumulator systems.

6.2.1 Safety Injection System (SIS)

Comments:

1. The safety injection system nodalization model is shown in Figure 6.2-1.

2. Volumes, initial boron concentrations and enthalpies are inputs.

3. Pump head vs. flow characteristics are input, and a time factor can be input to simulate pump start-up.

4. Homogeneous slug flow is assumed through all sections, however for typical BIT volumes and SI flowrates, mixing will occur in the BIT (see SLUG).

5. SI flow enters the cold leg at the safety injection nozzle location specified by input FCLSIS.

6.2.2 Emergency Boration System (EBS)

Comments:

1. Two EBS models are available:
   The EBS-I model is shown in Figure 6.2-2.
   The EBS-II model is shown in Figure 6.2-3.

2. Pressure in EBS-II tank (BIT) is assumed equal to pressure at top of pressurizer (connection line between pressurizer and BIT is not explicitly modeled).

3. EBS-II flow vs. pressure drop is input. Pressure drop is computed as difference between BIT pressure and RCS pressure at first cold leg section. EBS-I flow is input as a constant volumetric flow.

4. Slug flow is assumed through EBS lines.

5. EBS flow enters the cold legs at the safety injection nozzle location specified by input FCLSIS.

6.2.3 Accumulators

Comments:

1. One accumulator is provided for each loop, and one accumulator is provided for UHI (upper head injection).

2. Injection flow from the loop accumulators enters the cold leg at the safety injection nozzle location specified by input FCLSIS. Injection flow from the UHI accumulator enters the upper head, split between the Dead Volume (J=80) and the Outlet Plenum mixing region (J=98).

3. The perfect gas law is assumed to hold for the gas volume in the accumulators. For very long input thermal time constants, expansion is adiabatic and reversible (i.e. isentropic). For short input thermal time constants, expansion is isothermal. After many thermal time constants, the gas temperature returns to its initial value.

Equations:

Injection flow is computed from the friction loss equation:

$$F = \left(\frac{P_g - P_{RCS}}{K} \right)^{.5} \qquad (1)$$

where: F = accumulator flow, ft^3/sec
P_g = accumulator pressure, psia
P_{RCS} = reactor coolant system pressure, psia
K = pipe friction loss coefficient

The gas temperature in the accumulator is computed based on the input thermal time constant; the expansion can be isothermal or isentropic depending on the input:

$$T_g(t) = T_{go} \text{ for } \tau \leq .01\ DT \tag{2}$$

$$T_g(t) = T_g(t-\Delta t)\ e^{-DT/\tau} + T_{go}\ (1. - e^{-DT/\tau}) \tag{3}$$

$$\text{for } \tau > .01\ DT$$

where: T_{go} = ambient temperature, °R
T_g = gas temperature, °R
τ = thermal time constant, sec
DT = time step.

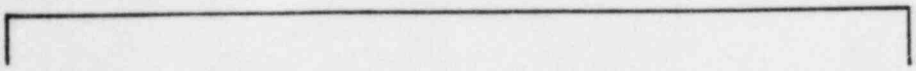
Gas pressure at the beginning of the time step is computed from the perfect gas law,

$$PV = RT \tag{4}$$

and pressure at the end of the time step is computed from:

$$PV^{\gamma} = \text{constant} \tag{5}$$

+a,c



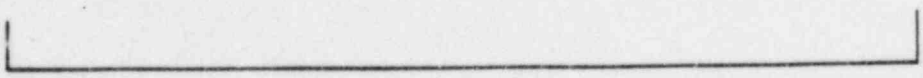


Figure 6.2-1. Safety Injection System

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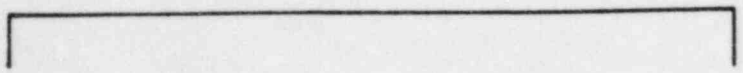


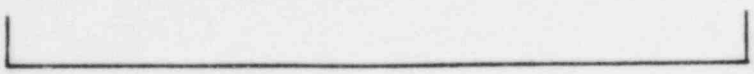


Figure 6.2-2. Emergency Boration System (EBS-1)

6970-8

+a,c

Figure 6.2-3. Emergency Boration System (EBS-II)

6.3 BCONC

Function: Computes space-time dependent boron concentration in the reactor coolant system.

Comments:

1. Boron concentration is assumed to be uniform in each section and to move from one section to the next as swept by local flow.

2. All sections are treated as slug flow (see BSLUG) except reactor vessel inlet plenum and dead volume, which are mixing regions.

3. Pressurizer is included in boron calculations, with option to initialize pressurizer at different concentration than RCS loops.

4. Core boron concentration transient may be input.

Equations:

Boron is conserved through the following equation

$$W_{out}C_{bout} = M^{\circ}C_b^{\circ} - M'C_b' + W_{in}C_{bin} \qquad (1)$$

where:
- W_{out} = integrated flow out of node, lbm.
- C_{bout} = boron concentration of node, ppm.
- W_{in} = integrated flow into node, lbm.
- C_{bin} = boron concentration into node, ppm.
- M° = mass in the node at time = t - Δt
- C_b° = boron concentration in the node at time = t - Δt
- M' = mass in the section at time = t
- C_b' = boron concentration at time = t

The equation for the reactor vessel inlet plenum is:

[ ] +a,c

For the dead volume:

[ ] +a,c

All other RCS sections use a slug flow formulation for boron transport, see BSLUG for equations.

6.4 L106

Function: Subroutine L106 simulates the rod drive programmer and L106 rod drive system.

Comments:

1. Determines rod position (in bank steps) at time t, and error signal between t - $\Delta\hat{t}$ and t which will allow the rods to move based upon the error.

2. Simulates 2 subgroups of control rods, 1/2 step apart.

3. Once a cycle has started, this routine will complete even if the error signal goes to zero.

4. There are provisions for rod stops.

5. Regardless of speed error signal, first step occurs after fixed time delay. Subsequent cycles are dependent on speed error signal.

SECTION 7
FLUID PROPERTIES

7.1 PROPV

Function:

1. Reduces property functions of enthalpy and pressure to functions of enthalpy only.

2. Computes saturation properties

Comments:

1. Functions of pressure and enthalpy are reduced to functions of only enthalpy each time the pressure is changed. Since many densities are calculated each time pressure is changed, this reduction results in significant computer time saving.

2. PROPV provides data for use in CALCSV and TEMPH, which is transmitted via the first line of Blank Common.

7.2 CALCSV

Function: Compute specific volume of water or steam as a function of enthalpy and pressure.

Comments:

1. Range: 0.2 to 10000 psia, 50°F to 1500°F

2. Accuracy: Error is less than 0.2 percent in density.

3. PROPV must have been previously called at the desired pressure.

7.3 TEMPH

Function: Computes a variety of water/steam properties.

Comments:

1. Temperature, pressure and enthalpy tables and saturation properties are stored in TEMPH. Various properties can be returned by the function TEMPH by adding various constants to the argument; e.g. TEMPH (-30000. -x) yields h_g (T). Other values of the argument H are:

0 > H > -10,000	- enthalpy as a function of temperature and pressure
-10,000 > H > -20,000	- h_f is a function of saturation temperature
-20,000 > H > -30,000	- Saturation pressure as a function of temperature
-30,000 > H > -40,000	- h_g as a function of temperature
-40,000 > H > -50,000	- v_f as a function of temperature
-50,000 > H	- $v_g P$ as a function of temperature
0 < H < 10,000	- Temperature as a function of enthalpy and pressure
10,000 < H < 20,000	- Saturation temperature as a function of h_f
20,000 < H	- Saturation temperature as a function of pressure

[ ] +a,c

7.4 ENTROP

Function: Computes entropy of saturated steam and water.

SECTION 8
AUXILIARY THERMAL AND HYDRAULIC ROUTINES

This group of subroutines is used throughout LOFTRAN to perform routine flow computations, such as slug or mixing flow, and critical flow correlations.

8.1 SLUG

Function:

Given initial fluid conditions and average input to a control volume (flow, enthalpy, and heat) over some time interval, compute final fluid conditions based on a single phase fluid with a uniform velocity profile and which moves as a slug.

Equations:

+a,c

+a,c

[+a ,c]

8.2 MIXER

Function:

Given initial conditions and average input (flow, enthalpy, etc.) over some time interval, compute final conditions and average exit conditions based on perfect mixing.

Comment:

For a perfect mixing model, the response of plenum enthalpy to a step change in inlet is:

$$h(t) = h_o + (h_{in} - h_o)\, e^{-\Delta t/\tau} \qquad (1)$$

The model used assumes that $\Delta t/\tau$ is given by $\frac{1}{M_o}\int_{t-\Delta t}^{t} W_{in}(t)\, dt$.

As with SLUG, the final mass is computed as a state property of enthalpy and pressure. The average flow out of the plenum is computed from the conservation of mass and the average exit enthalpy is determined to conserve energy.

[ ]+a,c

Equations:

The final enthalpy is calculated as

[ ]+a,c

[+a,c]

8.3 BSLUG

Function:

Compute boron (or anything else) concentration in a control volume given average input of flow and boron concentration over some time interval, based on slug flow.

Equations:

[+a,c]

Boron concentration in the flow out of the node is computed to conserve boron:

$$W_{out} C_{bout} = M^{\circ} C_b^{\circ} - M' C_b' + W_{in} C_{bin} \qquad (3)$$

where:

W_{out} = integrated flow out of node, lbm.
C_{bout} = boron concentration of node, ppm.
W_{in} = integrated flow into node, lbm.
C_{bin} = boron concentration into node, ppm.
M° = mass in the node at time t - Δt.
C_b° = boron concentration in the node at time t - Δt.
M' = mass in the section at time t.
C_b' = boron concentration at time t.

8.4 CRIT

Function:

Computes critical flow rate (lbm/sec) for saturated water based on Moody correlation for $f\frac{L}{D} = 0$.

8.5 ZALODEK

Function:

Computes critical flow (lbm/sec-ft^2) for subcooled water based on Zaloudek correlation. Uses orifice equation to compute flowrate when ΔP is too low for critical flow.

8.6 RELIEF

Computes critical flow (lbm/sec-ft^2) for subcooled water based on ANS homogeneous equilibrium tables for pressures of 1800 to 3100 psia.

8.7 HOMOSAT

Function:

Computes critical flow (lbm/sec-ft^2) for saturated water based on ANS homogeneous equilibrium tables.

SECTION 9
INPUT, OUTPUT, INITIALIZATION

9.1 INPUT1

Function:

1. Read in desired modes of code operation, power level, and flow.
2. Read in nominal fluid system and thermal-hydraulic data.
3. Set default values for any data not read in.
4. Print out summary of input.

Comments:

This subroutine reads in data in namelist MASTER (or master control card) and namelist FLSTHD.

9.2 INPUT2

Function:

1. Read in data for initial operating conditions, control and protection setpoints, and transient forcing functions.

2. Set default values for any data not read in.

Comments:

This subroutine reads in data in namelist OPCOND.

9.3 IN2OUT

Function:

Writes out summary of data associated with INPUT2.

9.4 INPUT3

Function:

1. Read in physics data and code control data.
2. Set default values for any data not read in.
3. Print summary of input data.

Comments:

This subroutine reads in data in namelist RECOCD.

9.5 STSTVE

Function:

1. Initializes volumes, areas, and miscellaneous data for the RCS nodal structure in each run.

2. Initializes steam generator secondary fluid conditions and UA.

3. Initializes reactor coolant system fluid conditions (mass, enthalpy, and flow).

Comments:

1. The plant is assumed to be in nuclear and thermal equilibrium at t = 0.

2. This subroutine initializes all RCS loops at equal conditions for flow and enthalpy. If non-symmetric initial conditions are required (by input), subroutine FLOWIN is called to complete initialization.

9.6 FLOWIN

Function:

Completes code initialization for cases requiring non-symmetric loop flows (e.g. operation with one loop out of service).

Comments:

Subroutine FLOWIN completes initialization by directly calling the transient subroutines QCALC, SG2, FCALC, CTRAN2, PLTRAN, HLTRAN and STMGEN. Nuclear power and RCS pressure are retained at the initial input values.

9.7 SUMMAT

Function:

1. Computes desired output and other miscellaneous variables.
2. Correct mass and energy errors caused by small flow errors.
3. Determines maximum and minimum values attained by certain variables.

9.8 UPRINT

Function:

1. Print (with format) pre-determined variables.
2. Update variables for next time step.

Comments:

1. A short printout (single line), consisting of the most useful variables are printed after each time step.

2. After a fixed number of time steps (determined by the integer DTPRIN - found in namelist RECOCD) a long printout is dumped. This printout consists of most major variables and fluid conditions throughout the reactor coolant system.

3. In general, "rate" variables (flow, power, enthalpy, etc.) are to be interpreted as average values existing during the previous time step and time step printed out. "State" properties (mass, temperature, pressure, etc.) are to be interpreted as conditions existing at the end of the time step.

9.9 DOM, DOM2

Function:

1. Subroutine DOM stores selected output variables at each time step for a summary output.

2. DOM2 prints the output summary at the end of each run in a condensed table format.

Comments:

1. These subroutines allow the user to obtain printed or plotted output of nearly any variable used in LOFTRAN.

9.10 HINSER

Function:

Compute initial core inlet enthalpy from power, flow, pressure, and T_{avg}.

Comments:

This function subprogram computes the inlet enthalpy to the core by iteration. The input groups use it during initialization, supplying T_{avg}.

power, volumetric flow and pressure. Density effects and variations in C_p are included.

Also computed are T_{HOT}, T_{COLD}, ΔT, h_{HOT} and cold leg density.

This routine operates in the compressed water region only. It requires prior calling of PROPV and the availability of CALCSV and TEMPH.

Equations:

T_{in} is first estimated as

$$T_{in} = TAVG - 3500 * \frac{POWER}{FLOW}$$

The following energy and state equations are then solved.

$$h_{in} = f_n (T_{in}, P)$$

$$v_{in} = f_n (h_{in}, P)$$

$$W = K_1 F/v_{in}$$

$$\Delta h = K_2 q/w$$

$$h_{out} = h_{in} + \Delta h$$

$$T_{out} = f_n (h_{out}, P)$$

$$\bar{T} = \frac{1}{2} (T_{in} + T_{out})$$

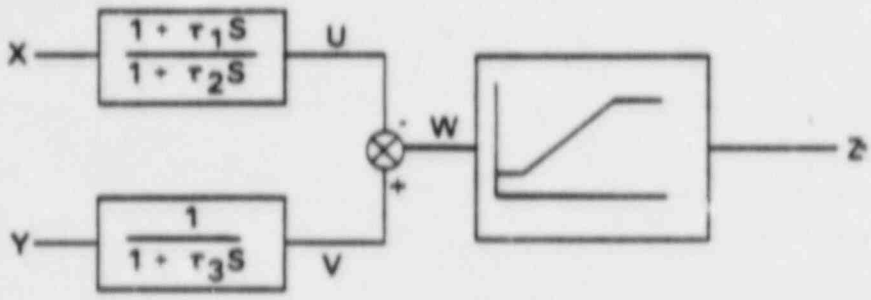
where K_1 and K_2 are appropriate conversion factors.

If $\bar{T}$ differs from the input TAVG by more than .001 °F, a new T_{in} is estimated and the process is repeated (up to a maximum of 10 iterations).

SECTION 10
CONTROLLER ROUTINES

This section covers the simulation of individual components in a control system, such a lead/lag units, proportional controllers, summers, etc. Refer to the CONTRL section for discussion of the way these controllers are connected in LOFTRAN. Some general features of the controller routines are described below.

1. These subroutines compute the output response of various controllers to specified input signals. For example, consider the following block diagram:



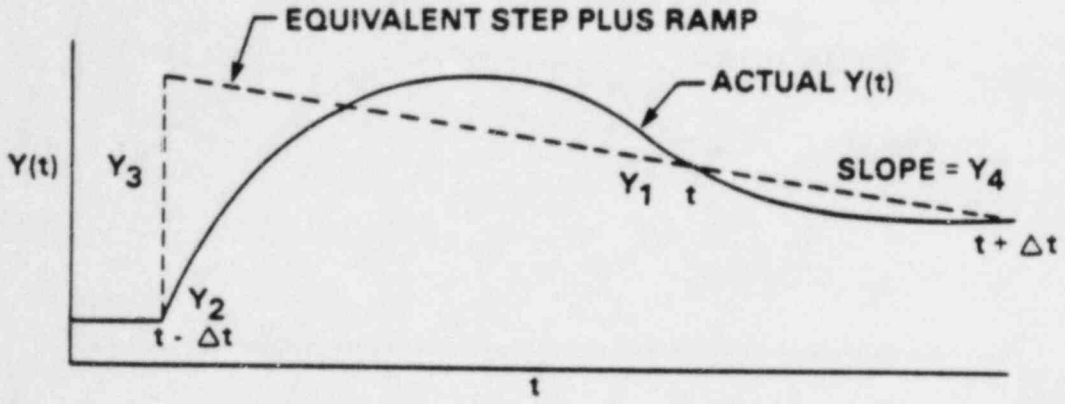
Given the input signals X and Y, the output Z would be determined by the following FORTRAN instructions:

```
CALL LEAD (X, U, T1, T2, ...)
CALL LAG (Y, V, T3, ...)
CALL ADD (U, V, W, ...)
CALL PROP (W, Z, (gains), ...)
```

2. [+a,c]

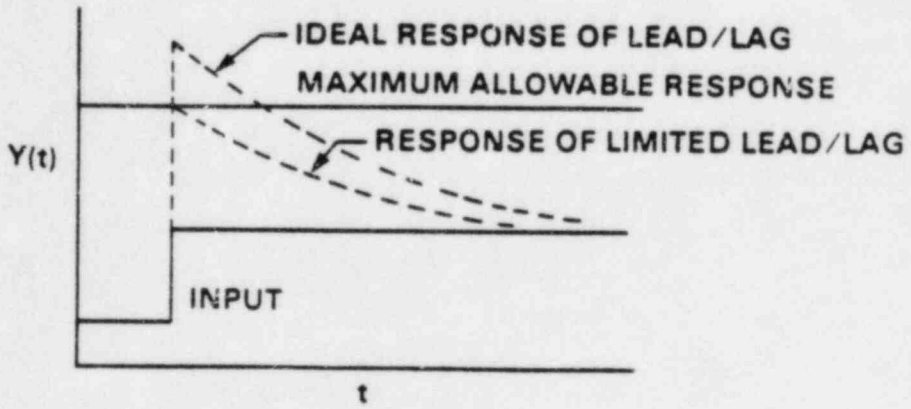
3. For dynamic controllers (such as LAG), the solution to the differential equation is programmed explicitly rather than a numerical solution to the differential equation. In each case, the input is assumed to be a step plus a ramp. The calculated output is exact to the extent that the input fits a step plus a ramp.

To illustrate,



The equivalent step plus ramp equals Y_1 at time t, and has the same integral between t - Δt and t as the curve being fit. The height of the step, then, is $2\ Y_3 - Y_1 - Y_2$, and the slope of the ramp is $2\ (Y_1 - Y_3)/\Delta t$

4. The various controllers have provision for high and low limits, but (in general) these limits cannot be input to LOFTRAN. Time dependent controllers (LAG, LEAD, RATE, and PID) are feedback limited; i.e., if the output signal would otherwise be outside the limits, a portion of the output is fed back as an input. Hence the controller does not saturate. As an example of this behavior for LEAD (lead/lag unit):



10.1 LAG

This subroutine computes the output response of a first-order lag; i.e.,

$$Y(S) = \frac{1}{1 + \tau S} X(S)$$

The equations are given below. As with all signals for controller routines, the input is assumed to be adequately fit by a step plus a ramp.

+a,c

10.2 RATE

This subroutine computes the output response of an impulse (rate/lag) circuit; i.e.,

$$Y(S) = \frac{\tau S}{1 + \tau S} X(S), \text{ or}$$

$$Y(S) = 1 - \frac{1}{1 + \tau S} X(S)$$

Subroutine LAG is used by RATE.

10.3 LEAD

This subroutine computes the output response of a lead/lag circuit; i.e.,

$$Y(S) = \frac{1 + \tau_1 S}{1 + \tau_2 S} X(S), \text{ or}$$

$$Y(S) = \left[\frac{1}{1 + \tau_2 S} + \frac{\tau_1}{\tau_2}\left(1 - \frac{1}{1 + \tau_2 S}\right)\right] X(S)$$

Subroutine LAG is used by LEAD.

10.4 PID

This subroutine computes the output response of a Proportional-Integral-Derivative (or proportional + rate + reset) controller; i.e.,

$$Y(S) = X(S)\left[K\left(1 + \frac{1}{\tau_1 S}\right) + \frac{\tau_2 S}{1 + 0.1\,\tau_2 S}\right]$$

where K is the proportional gain (note that K acts upon the integral, but not the derivative term).

Subroutine RATE is used by PID.

In steady-state, all input terms to a PID controller sum to zero. For constant initialization, (steady-state conditions are always assumed at t = 0). The integral term is determined to produce the desired output, i.e.,

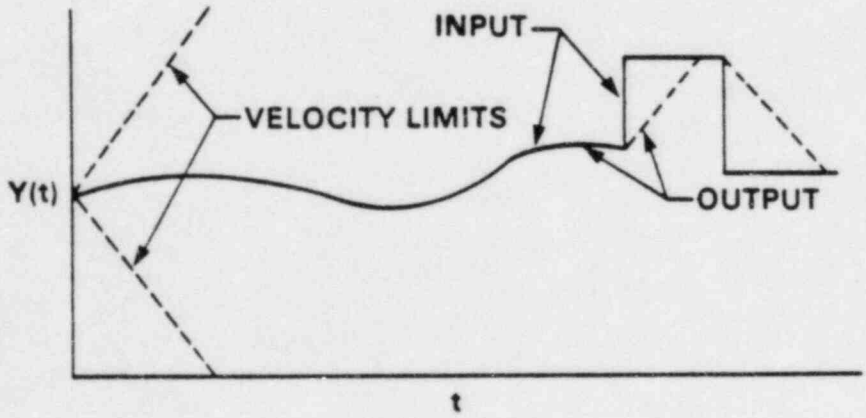
$$\int_{-\infty}^{0} X(t)\,dt = \frac{Y(0)}{K}, \text{ and}$$

$$X(0) = 0$$

High and low limits are implemented by feeding back a portion of the output to the integral term to simulate anti-reset windup. This mode of limiting will shift the controller setpoint, such that when the output ceases to be limited, it is controlling about a new setpoint. The setpoint will be reset to the normal one on the integral time constant, τ_1.

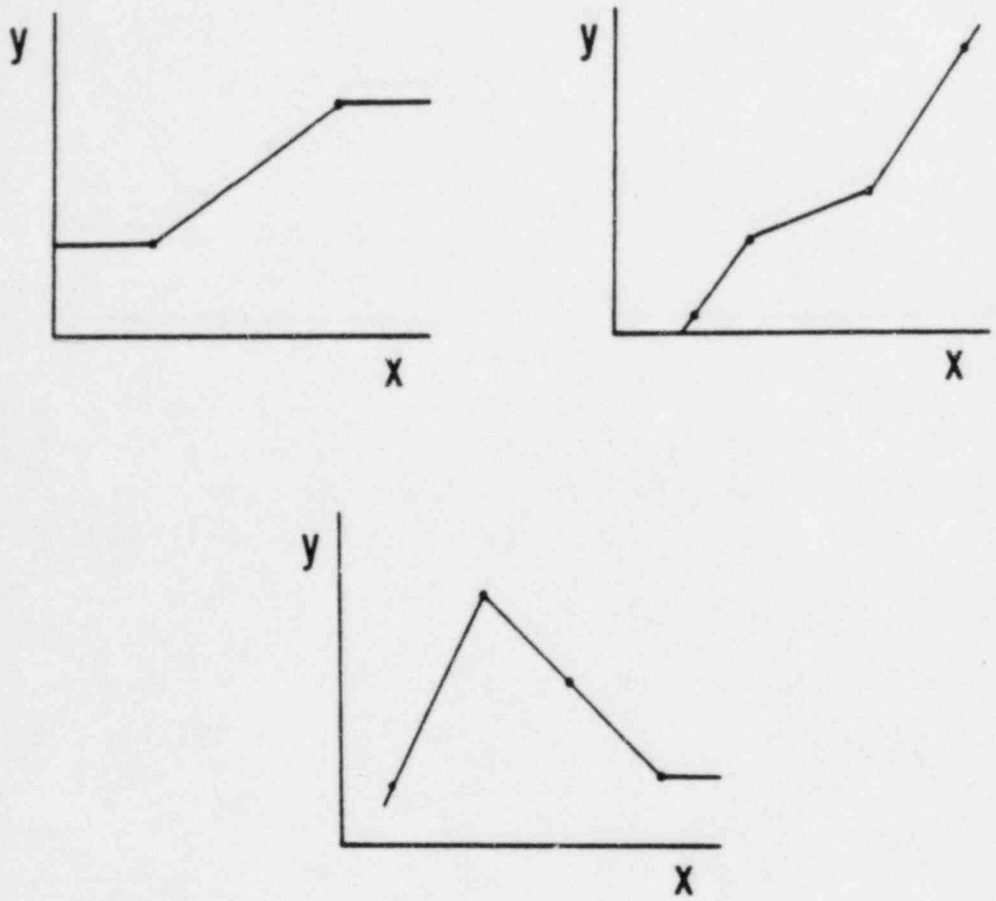
10.5 VELLIM

This subroutine computes the response of a unity gain, velocity limited device, such as a control valve. If the input is changing at less than the velocity limit, the output will follow the input exactly. However, the output can change no faster than its velocity limit. An illustration is shown below:



10.6 PROP

This subroutine computes the output response of a proportional or non-linear controller (or programmer). No dynamic terms are simulated. The program, $y = f(x)$ is input as tables of points, with linear interpolation assumed between the entries. Examples of programs permissible are:



For input x beyond the range of the y(x) tables, y is determined by either extrapolation or as the endpoint value. The only limitation is that y(x) must be single-valued.

10.7 MULT

This subroutine can:

1. multiply two input signals;
2. multiply an input signal by a constant; or
3. divide an input signal by a second input signal.

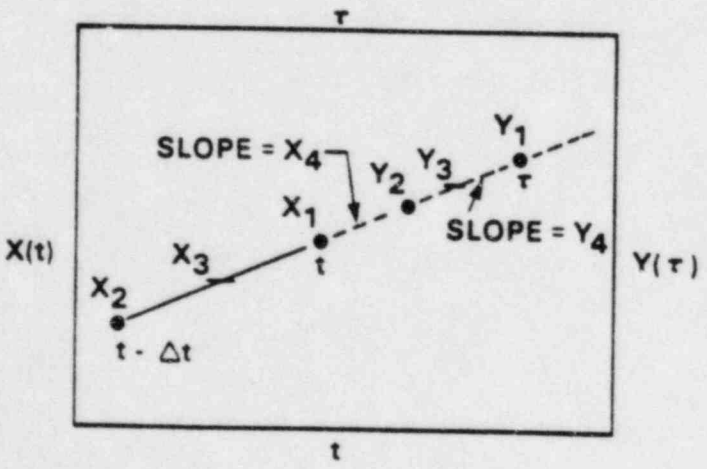
10.8 ADD

This subroutine can:

1. add two input signals;
2. add a constant to an input signal; or
3. subtract an input signal from a second input signal.

10.9 CONV

As explained in Section 10.0, each signal variable is described by four parameters identifying its value at time t, at time t - Δt, etc. This subroutine converts these four parameters to a new time, t, or time step, Δt. (Control system output signals are computed at the end of a time step and their effect is computed during the next time step.) Conversion may be either forward or backward in time (or no change). Conversion is best explained by illustration.



10.10 SCRAM

This subroutine compares with its trip setpoints (at both the beginning and end of a time step), and determines the time of trip (if it occurs during that time step) by linear interpolation. The subroutine also prints out a trip message.

SECTION 11
NUMERICAL MANIPULATION ROUTINES

11.1 QUAD

Function:

Quadratic fitting routine

Comments:

Given the data points (X_1, Y_1), (X_2, Y_2) and (X_3, Y_3) this subroutine determines the coefficients of the quadratic, $Y = A + BX + CX^2$, which passes through all the points.

11.2 TERP

Function:

Linear interpolation routine.

Comments:

1. Given an input array (X_i, Y_i) this function subprogram linearly interpolates the corresponding value of Y for a given X.

2. All elements in the X array must be in ascending order, but discontinuities are permissible.

3. There also exists an option for linear extrapolation, a constant value, or zero if a value on the X-axis is outside the limits of (X_i, Y_i) which was to be interpolated. Extrapolation exists for both ends of the table. (0 - constant, 1 - extrapolate, -1 - zero).

11.3 SINTEG

Function:

Determines the integral of a known function between two points.

Comments:

1. Given an input array (X_i, Y_i) this function subprogram determines the integral of the function Y(X) from X_1 to X_2.

2. The function is assumed to vary linearly between the known points, $Y_i = f(X_i)$ where $X_1 \leq X_i \leq X_2$.

3. For values of X outside the range of the input table Y(X) is assumed constant at its end-point value.

4. Discontinuities are acceptable.

5. The average value of Y(X) can also be calculated.

11.4 TBL

Function:

Quadratic interpolation routine for water tables

Comments:

Given an input array (X_i, Y_i) this function subprogram interpolates quadratically the corresponding value of Y for a given X. This function is used only for interpolation of the water tables.

11.5 ACCUM1

Function:

This routine sums all the elements in a single array.

11.6 ACCUM2

Function:

This routine sums all the products of the element in two arrays.

11.7 DVCKF

Function:

Checks a divide instruction for infinite or indefinite results.

Comments:

A zero result is returned if the denominator is zero or near zero.

11.8 SCAN

Function:

SCAN checks the order of an array to ensure that it progresses in ascending order.

APPENDIX

This Appendix contains the cover letters from the three major supplementary information packages provided to the NRC during the review of WCAP-7907. These letters are references 2, 3, and 4 of the Topical Report Evaluation which is attached at the beginning of this WCAP. The entire letters are substantial in size and are not included here. However, much of the information provided in the letters has been incorporated in the body of this report.

W ghouse Electric Corporation Power Systems

Box 355
Pittsburgh Pennsylvania 15230

May 26, 1978

NS-TMA-1802

Mr. John F. Stolz, Chief
Light Water Reactors Branch No. 1
Division of Projects Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Reference: Letter from J. F. Stolz to T. M. Anderson, May 10, 1978.
Subject: Review of WCAP-7907

Dear Mr. Stolz:

Enclosed are:

1. Twenty-five (25) copies of supplemental information on WCAP-7907 (Proprietary).
2. Twenty (20) copies of supplemental information WCAP-7907 (Non-Proprietary).

WCAP-7907 is entitled, "LOFTRAN Code Description".

The information transmitted in this response is additional information as requested by the above reference.

This submittal contains proprietary information. In conformance with the requirements of 10CFR Section 2.790, as amended, of the Commission's regulations, we are enclosing with this submittal, an application for withholding from public disclosure and an affidavit. The affidavit identifies the information sought to be withheld and sets forth the basis on which the information may be withheld from public disclosure by the Commission.

We expect that the non-proprietary version of this report, WCAP-7907 will be placed in the Public Document Room and identified as a Westinghouse Topical Report.

Correspondence with respect to the Westinghouse affidavit or application for withholding should be addressed to: R. A. Wiesemann, Manager, Licensing Programs, Westinghouse Electric Corporation, P. O. Box 355, Pittsburgh, Pennsylvania 15230.

Very truly yours,

T. M. Anderson, Manager
Nuclear Safety Department

LAC:kln
Enclosure

Westinghouse Electric Corporation
Water Reactor Divisions

Nuclear Technology Division
Box 355
Pittsburgh Pennsylvania 15230

January 19, 1982
NS-EPR-2536

Mr. James R. Miller, Chief
Special Projects Branch
Division of Project Management
U.S. Nuclear Regulatory Commission
Phillips Building
7920 Norfolk Avenue
Bethesda, Maryland 20014

Attention: T. P. Speis

Subject: Transmittal of Slides from the December 15, 1981 Westinghouse Meeting on the LOFTRAN (WCAP-7907) and MARVEL (WCAP-7909) Codes

Dear Mr. Miller:

Enclosed are:

1. Five (5) copies of slides on LOFTRAN and MARVEL (Proprietary)

2. Ten (10) copies of slides on LOFTRAN and MARVEL (Non-Proprietary)

Copies of the slides presented in the subject meeting were requested by Mr. Edward D. Throm, Reactor Systems Branch. Subject material is tentatively identified for future submittals for NRC review.

Also enclosed are:

1. One (1) copy of Application for Withholding (Non-Proprietary)

2. One (1) copy of original Affidavit (Non-Proprietary)

This submittal contains proprietary information of Westinghouse Electric Corporation. In conformance with the requirements of 10CFR2.790, as amended, of the Commission's regulations, we are enclosing with this submittal an application for withholding from public disclosure and an affidavit. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission.

Correspondence with respect to the affidavit or application for withholding should reference AW-82-1 and should be addressed to R. A. Wiesemann, Manager of Regulatory and Legislative Affairs, Westinghouse Electric Corporation, P.O. Box 355, Pittsburgh, Pennsylvania 15230.

Very truly yours,

E. P. Rahe, Jr., Manager
Nuclear Safety Department

SM/kk
Enclosures

Westinghouse
Electric Corporation

Water Reactor
Divisions

Nuclear Technology Division

Box 355
Pittsburgh Pennsylvania 15230

August 27, 1982

Dr. Cecil O. Thomas, Chief
Special Projects Branch
Division of Project Management
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Bethesda, Maryland 20014

Attention: Dr. B. Sheron, RSB

Dear Dr. Thomas:

Enclosed are:

1) Twenty-five (25) copies of NRC/ORNL/Westinghouse Technical Review of the LOFTRAN and MARVEL Safety Analysis Codes, August 1982, (Proprietary).

2) Fifteen (15) copies of NRC/ORNL/Westinghouse Technical Review of the LOFTRAN and MARVEL Safety Analysis Codes, August 1982, (Non-Proprietary).

Also enclosed are:

1) One (1) copy of Application for Withholding AW-82-52.

2) One (1) copy of Affidavit.

This submittal is in response to concerns addressed in NRC letter, Check to Anderson, July 8, 1981 and applies to approval of WCAP's 7907, 7909 and 8843.

Representatives from NRC, ORNL, and Westinghouse met on July 13 and July 14 of this year to answer any remaining questions pertaining to the NRC and ORNL review of the LOFTRAN and MARVEL safety analysis codes. The purpose of this letter is to transmit the minutes of the meeting and to express my sincere thanks for your participation in what turned out to be a very productive meeting towards completing the review of these two codes.

A preliminary copy of the minutes containing all of the technical input for LOFTRAN had been transmitted to ORNL on July 30, 1982. Absent from those minutes were the responses for LOFTRAN questions 1-6, which had already been provided, and also those portions pertaining to MARVEL. Attached is the finalized version which differs from the preliminary copy of the minutes only in that these sections have been included. Additionally those parts, which have been scored in the margin for your convenience, have been modified to reflect ORNL comments on the previously transmitted preliminary version of the minutes.

This final copy of the meeting minutes completes the Westinghouse response to the NRC and ORNL questions and the input required to complete the review of LOFTRAN and MARVEL. It is anticipated that Westinghouse will issue a final WCAP revision pending your acceptance.

Correspondence with respect to proprietary aspects of this submittal should reference AW-82-52 and should be addressed to R. A. Wiesemann, Manager, Regulatory and Legislative Affairs, Westinghouse Electric Corporation, P.O. Box 355, Pittsburgh, Pennsylvania 15230.

Please contact the undersigned regarding any questions or comments.

Very truly yours,

WESTINGHOUSE ELECTRIC CORPORATION

E. P. Rahe, Jr., Manager
Nuclear Safety Department

S. T. Maher/ds

Enclosures

cc: R. S. Stone ORNL
F. H. Clark ORNL
R. M. Harrington ORNL
E. Throm RSB
J. Guttman RSB