Westinghouse Electric Corporation
Water Reactor Divisions
Box 355
Pittsburgh Pennsylvania 15230

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

August 27, 1982
AW-82-52

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

SUBJECT: NRC/ORNL/Westinghouse Technical Review of the LOFTRAN and MARVEL Safety Analysis Codes, August 1982

REF: Westinghouse Letter NS-EPR-2648, Rahe to Thomas, dated August 27, 1982

Dear Dr. Thomas:

This application for withholding is submitted by Westinghouse Electric Corporation ("Westinghouse") pursuant to the provisions of paragraph (b)(1) of Section 2.790 of the Commission's regulations. Withholding from public disclosure is requested with respect to the subject information which is further identified in the affidavit accompanying this application.

The proprietary material for which withholding is being requested is of the same technical type as that proprietary material previously submitted with application for withholding AW-76-31 and was accompanied by an affidavit signed by the owner of the proprietary information, Westinghouse Electric Corporation.

The affidavit AW-76-31 submitted to justify the previous material is equally applicable to this material, a copy of which is attached.

Accordingly, it is respectfully requested that the subject information which is proprietary to Westinghouse and which is further identified in the affidavit be withheld from public disclosure in accordance with 10CFR Section 2.790 of the Commission's regulations.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference AW-82-52 and should be addressed to the undersigned.

Very truly yours,

Robert A. Wiesemann

Robert A. Wiesemann, Manager
Regulatory & Legislative Affairs

/bek
Enclosure

cc: E. C. Shomaker, Esq.
Office of the Executive Legal Director, NRC

8209140299 820827
PDR TOPRP EMVWEST
C PDR

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Robert A. Wiesemann, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Corporation ("Westinghouse") and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

Robert A. Wiesemann

Robert A. Wiesemann, Manager
Licensing Programs

Sworn to and subscribed before me this 15 day of July 1976.

Notary Public

RE ... IC

... COUNTY
MY COMMISSION EXPIRES APR. 15, 1978

(1) I am Manager, Licensing Programs, in the Pressurized Water Reactor Systems Division, of Westinghouse Electric Corporation and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing or rule-making proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Water Reactor Divisions.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.790 of the Commission's regulations and in conjunction with the Westinghouse application for withholding accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse Nuclear Energy Systems in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(g) It is not the property of Westinghouse, but must be treated as proprietary by Westinghouse according to agreements with the owner.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information which is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition in those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.790, it is to be received in confidence by the Commission.

(iv) The information is not available in public sources to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in the attachment to Westinghouse letter No. NS-CE-1142, Eicheldinger to Eisenhut dated July 27, 1976 concerning reproductions of viewgraphs used in the Westinghouse presentation to the NRC during the meeting on July 27, 1976 on the subject of Westinghouse Reload Safety Evaluation Methodology.

This information enables Westinghouse to:

(a) Justify the design for the reload core

(b) Assist its customers to obtain licenses

(c) Meet contractual requirements

(d) Provide greater flexibility to customers assuring them of safe and reliable operation.

Further, this information has substantial commercial value as follows:

(a) Westinghouse sells the use of the information to its customers for purposes of meeting NRC requirements for licensing documentation.

(b) Westinghouse uses the information to perform and justify analyses which are sold to customers.

(c) Westinghouse uses the information to sell nuclear fuel and related services to its customers.

Public disclosure of this information is likely to cause substantial harm to the competitive position of Westinghouse in selling nuclear fuel and related services.

Westinghouse retains a marketing advantage by virtue of the knowledge, experience and competence it has gained through long involvement and considerable investment in all aspects of the nuclear power generation industry. In particular Westinghouse has developed a unique understanding of the factors and parameters which are variable in the process of design of nuclear fuel and which do affect the in service performance of the fuel and its suitability for the purpose for which it was provided.

In all cases that purpose is to generate energy in a safe and efficient manner while enabling the operating nuclear generating station to meet all regulatory requirements affected by the core loading of nuclear fuel. Confidence in being able to accomplish this comes from the exercise of judgement based on experience.

Thus, the essence of the competitive advantage in this field lies in an understanding of which analyses should be performed and in the methods and models used to perform these analyses. A substantial part of this competitive advantage will be lost if the competitors of Westinghouse are able to use the results of the Westinghouse experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions. Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design and licensing of a similar product.

This information is a product of Westinghouse design technology. As such, it is broadly applicable to the sale and licensing of fuel in pressurized water reactors. The development of this information is the result of many years of Westinghouse effort and the expenditure of a considerable sum of money. In order for competitors of Westinghouse to duplicate this process

NS-EPR-2648

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 8343.

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.

Two Rids
T007 · Class II Prop Portion
1/28
T008 Class 3 Non Prop Portion
1/15

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

EMRahe Jr

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

Westinghouse Electric Corporation
Water Reactor Divisions
Box 355
Pittsburgh Pennsylvania 15230

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

August 27, 1982
AW-82-52

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

SUBJECT: NRC/ORNL/Westinghouse Technical Review of the LOFTRAN and MARVEL Safety Analysis Codes, August 1982

REF: Westinghouse Letter NS-EPR-2648, Rahe to Thomas, dated August 27, 1982

Dear Dr. Thomas:

This application for withholding is submitted by Westinghouse Electric Corporation ("Westinghouse") pursuant to the provisions of paragraph (b)(1) of Section 2.790 of the Commission's regulations. Withholding from public disclosure is requested with respect to the subject information which is further identified in the affidavit accompanying this application.

The proprietary material for which withholding is being requested is of the same technical type as that proprietary material previously submitted with application for withholding AW-76-31 and was accompanied by an affidavit signed by the owner of the proprietary information, Westinghouse Electric Corporation.

The affidavit AW-76-31 submitted to justify the previous material is equally applicable to this material, a copy of which is attached.

Accordingly, it is respectfully requested that the subject information which is proprietary to Westinghouse and which is further identified in the affidavit be withheld from public disclosure in accordance with 10CFR Section 2.790 of the Commission's regulations.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference AW-82-52 and should be addressed to the undersigned.

Very truly yours,

Robert A. Wiesemann

Robert A. Wiesemann, Manager
Regulatory & Legislative Affairs

/bek
Enclosure

cc: E. C. Shomaker, Esq.
Office of the Executive Legal Director, NRC

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Robert A. Wiesemann, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Corporation ("Westinghouse") and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

Robert A. Wiesemann

Robert A. Wiesemann, Manager
Licensing Programs

Sworn to and subscribed
before me this 5 day
of July 1976.

Notary Public

MY COMMISSION EXPIRES APR. 15, 1978

(1) I am Manager, Licensing Programs, in the Pressurized Water Reactor Systems Division, of Westinghouse Electric Corporation and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing or rulemaking proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Water Reactor Divisions.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.790 of the Commission's regulations and in conjunction with the Westinghouse application for withholding accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse Nuclear Energy Systems in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(g) It is not the property of Westinghouse, but must be treated as proprietary by Westinghouse according to agreements with the owner.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information which is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition in those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.790, it is to be received in confidence by the Commission.

(iv) The information is not available in public sources to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in the attachment to Westinghouse letter No. NS-CE-1142, Eicheldinger to Eisenhut dated July 27, 1976 concerning reproductions of viewgraphs used in the Westinghouse presentation to the NRC during the meeting on July 27, 1976 on the subject of Westinghouse Reload Safety Evaluation Methodology.

This information enables Westinghouse to:

(a) Justify the design for the reload core

(b) Assist its customers to obtain licenses

(c) Meet contractual requirements

(d) Provide greater flexibility to customers assuring them of safe and reliable operation.

Further, this information has substantial commercial value as follows:

(a) Westinghouse sells the use of the information to its customers for purposes of meeting NRC requirements for licensing documentation.

(b) Westinghouse uses the information to perform and justify analyses which are sold to customers.

(c) Westinghouse uses the information to sell nuclear fuel and related services to its customers.

Public disclosure of this information is likely to cause substantial harm to the competitive position of Westinghouse in selling nuclear fuel and related services.

Westinghouse retains a marketing advantage by virtue of the knowledge, experience and competence it has gained through long involvement and considerable investment in all aspects of the nuclear power generation industry. In particular Westinghouse has developed a unique understanding of the factors and parameters which are variable in the process of design of nuclear fuel and which do affect the in service performance of the fuel and its suitability for the purpose for which it was provided.

In all cases that purpose is to generate energy in a safe and efficient manner while enabling the operating nuclear generating station to meet all regulatory requirements affected by the core loading of nuclear fuel. Confidence in being able to accomplish this comes from the exercise of judgement based on experience.

Thus, the essence of the competitive advantage in this field lies in an understanding of which analyses should be performed and in the methods and models used to perform these analyses. A substantial part of this competitive advantage will be lost if the competitors of Westinghouse are able to use the results of the Westinghouse experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions. Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design and licensing of a similar product.

This information is a product of Westinghouse design technology. As such, it is broadly applicabl to the sale and licensing of fuel in pressurized water reactors. The development of this information is the result of many years of Westinghouse effort and the expenditure of a considerable sum of money. In order for competitors of Westinghouse to duplicate this process

WESTINGHOUSE PROPRIETARY CLASS 3

LOFTRAN/MARVEL TECHNICAL REVIEW
MEETING MINUTES

NRC - ORNL - WESTINGHOUSE

Place:

Monroeville Nuclear Center, July 13 & 14, 1982

Attendees/Contributors

NRC/RSB - Jack Guttmann
ORNL - Francis Clark
Bob Stone
Mike Harrington

W - Sam Miranda
Steve Maher
Gerard Elia
Toby Burnett
Dennis Richardson
Glenn Lang
Larry Campbell
Jim Little

Contents

*LOFTRAN and MARVEL questions submitted to Westinghouse in letter from P. S. Check to T. M. Anderson, July 8, 1981.

Discussion of Objectives

Following a discussion of the objectives the following was agreed upon:

- The goal of the meeting was for Westinghouse to supply sufficient information to enable ORNL to write a Technical Evaluation Report to the NRC leading to the production of the Safety Evaluation Report on the safety analysis codes.

- It was agreed in the interests of expediency that Westinghouse would be willing to support as much effort as would be necessary to address all the concerns of the NRC and ORNL at this meeting to limit the need for further review/interface prior to completion of the Technical Evaluation Report.

- All reponses to NRC/ORNL questions were discussed during the meeting, these discussions will be covered in the meeting minutes and agreed upon before concluding the meeting. It was agreed that these meeting minutes would constitute Westinghouse response to the NRC/ORNL questions.

Introduction by D. C. Richardson

Discussion of Agenda

- Tailor agenda to maximize material discussed as being directly applicable to verify the adequacy of the use of LOFTRAN and MARVEL in safety analyses.

- Westinghouse verified for LOFTRAN that: with the exception of the transition to the 4 loop code there were no changes which modified the analytical basis, just supplements to the original code. It was then decided to narrow the discussion to 3 safety analysis codes, hereafter described as:

  LOFTRAN 1 (One Loop LOFTRAN)
  LOFTRAN 2 (Four Loop LOFTRAN)
  MARVEL

- Westinghouse discussed the basis for safety analyses; analytical methodology is described in the SAR and in our methodology for performing safety analyses (Safety Analysis Standards). The safety analysis code is part of this methodology but the assumptions for the analyses are designed to ensure that the code is used in a conservative manner. Westinghouse contends that there is ample material to verify that LOFTRAN and MARVEL adequately model plant transients and that they are appropriate codes for use in safety analyses when coupled with appropriate safety analysis assumptions.

- Westinghouse discussed the development, background, and origins of the LOFTRAN and MARVEL codes and their common bases.

- Westinghouse discussed the comparison of LOFTRAN 1, MARVEL, and LOFTRAN 2 to plant transients and other safety analysis codes including RELAP-5.

- Westinghouse discussed the differences between LOFTRAN 1 and LOFTRAN 2. Except for the modification to upgrade LOFTRAN 1 to a multiloop code (LOFTRAN 2), the bases of LOFTRAN 1 and LOFTRAN 2 are practically identical; this has been verified in detail by comparing the results of the two codes. The multi-loop code is used exclusively now, but both are still acceptable for use in safety analyses.

- Because of the similarity in the basic assumptions of LOFTRAN 1 and LOFTRAN 2, the responses presented within will be applicable to both codes (referred to as just LOFTRAN) unless otherwise specified.

- LOFTRAN is the primary code for use in our current safety analysis methodology. It was agreed that the discussion should start with

LOFTRAN, because of its greater use in current safety analysis. MARVEL is no longer used in our current analysis methodology and because of the similarity in the analytical basis, since it was developed from LOFTRAN, it is expected that practically all LOFTRAN responses would be applicable to MARVEL.

Review of LOFTRAN NRC/ORNL Questions

Questions 1-6

The response to these questions had been previously generated and are included in the following pages. These questions were discussed during the meeting.

It was agreed that sufficient information had been provided to address the concerns.[1]

1. Even though it was agreed at the meeting that sufficient information had been provided to enable a proper evaluation of the concerns, it was clear to all participants that the final acceptability of LOFTRAN would be addressed in the ORNL report.

Q-1: What is the intended use of LOFTRAN, with respect to licensing analyses?

A-1: The LOFTRAN program is used for studies of transient response of a pressurized water reactor system to specified perturbations in process parameters. LOFTRAN simulates a multi-loop system containing reactor vessel, hot and cold leg piping, steam generator (tube and shell sides) and the pressurizer. The pressurizer heaters, spray, relief and safety valves are also considered in the program. Point model neutron kinetics, and reactivity effects of the moderator, fuel, boron and rods are included. The secondary side of the steam generator utilizes a homogeneous, saturated mixture for the thermal transients and a water level correlation for indication and control. 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 are also simulated including rod control, steam dump, feedwater control and pressurizer pressure control. The Safety Injection System including the accumulators are also modeled.

LOFTRAN is a versatile program which is suited to both accident evaluation and control studies as well as parameter sizing.

LOFTRAN is used to simulate plant response to many of the postulated events reported in Chapter 15 of PSAR's and FSAR's, to simulate anticipated transients without scram, and 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

Q-2: What state variables are solved for at what locations?

Q-3: What is the geometry (i.e., physical components and their spatial relations) of LOFTRAN? Figure 3-1 from WCAP-7907 and Figure 3.0-1 from WCAP-7907 Supplement are insufficient and insufficiently explained.

Q-4: How are state variables coupled between components? What is the order of solution? (WCAP-7907 Supplement 2-1, 2 is inadequate.) What iterative procedures, if any, are employed? What convergence criteria? Provide flow diagrams as appropriate.

A-2
A-3:
A-4

LOFTRAN is a digital computer code 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 model utilizes a homogenous, 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 the code or controlled by specifying certain control options. The reactor protection system is simulated including reactor trips DNB and overpower protection, high neutron flux, high and low pressure, high pressurizer level, low reactor coolant flow, low steam generator level, and safety injection actuation. The engineered safeguards features simulated are feedwater and steam line isolation, auxiliary feed, and an emergency core cooling system with high and low heat safety injection and accumulators. The control systems simulated are rod control, steam dump, feedwater control, and pressurizer pressure control. LOFTRAN also has the facility for calculating the transient value of DNBR, based on input developed from the core thermal-hydraulic limits. Reactor coolant pump operation is simulated including the effects of pump cooldown and pump startup with flow reversal allowed.

Applications of the Computer Code

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. Most of the faults studied do not place severe requirements on the modeling capability of the code; however, fairly complex transients can be studied with the code including the steam line break, feedwater line break, steam generator tube rupture, and ATWS events. The code has also been used to analyze and investigate safety implications of actual plant faults which have occurred in the U.S. A list of the faults for the UK PWR in which the LOFTRAN code is expected to be used is shown in Table 1. In addition, the code is expected to be used in modeling studies of control system behavior to assist in optimization of control setpoints and gains for normal operation including load swings and load follow, for demonstrating the adequacy of the sizing criteria for steam dump and primary and secondary pressure relief systems for load rejections, and for generation of design transients for component mechanical evaluation.

DESCRIPTION OF PRINCIPAL MODELS

The principal models used in LOFTRAN are those associated with modelling the reactor core, reactor coolant flow, the pressuriser and steam generator. A detailed description of the equations used, numerical solution techniques, approximations and correlations, and model limitations is found in the LOFTRAN code manual, WCAP-7907.

The description and discussion of the models and their usage presented in this section are intended only to supplement the more detailed discussion in the above reference.

Reactor Core Model

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

Fuel Heat Transfer Model

The fuel heat transfer model uses up to 40 axial nodes (user specified) and one radial node, with a parabolic axial power distribution of 1.5 peak to average value. A fuel specific heat which varies with fuel temperature is used, and 2.6% 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 usually 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 the detailed Westinghouse fuel rod design codes. This model is adequate for predicting average core heat transfer for all except the most rapid core power transients such a the RCCA ejection and RCCA withdrawal from subcritical accidents. In addition, in transients where a detailed knowledge of the heat transfer or fuel temperature are important, such as for transient DNBR evaluations, the LOFTRAN code thermal power prediction is not used in the THINC DNBR code; instead the nuclear power versus time is transferred to a more detailed transient fuel pellet heat transfer model (FACTRAN) for calculation of hot and average channel heat flux.

Neutron Kinetics Model

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

In addition, 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 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. The accuracy of any point kinetics model such as that employed by LOFTRAN is therefore dependent on how representative are the reactivity and feedback coefficients input by the user.

Decay Heat Model

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-break) decay heat is a benefit and may be conservatively removed in the analysis.

Reactor Coolant Loop Model

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 40 steam generator tube sections, and 8 cold leg sections. Generally, a typical analysis employs about 1/2 of the number of allowable sections in each component. The pressuriser can be located in any loop, the only restriction being that reverse flow is not allowed in the loop with the pressuriser. A homogeneous 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

(Continued)

a,c

The code will initialize with reverse flow in one or more loops, although the flow in the core and the loop with the pressuriser must be positive. Boron transport is handled, however due to the homogenous 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 under a low flow condition.

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. The equations solved by this model are straightforward, and the results have been proven to be adequate by comparison with flow coastdown measurements which are required to be performed in every US PWR during pre-startup testing.

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 asymmetrics (for example the steambreak accident) and would be very sensitive to this input.

Pressuriser Model

The pressuriser model computes the mass and energy balances in a two (water and steam) region pressuriser. 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, whilst 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. Safety and relief valve flow rates for steam relief are obtained from values specified by the manufacturer using ASME code methods and are input to the code. For water relief, the valve area is generally input and the user selects the appropriate critical flow model.

Steam Generator Model

On the primary side, the steam geneator model contains multiple (up to 16) tube sections. the secondary side is a one section model with a saturated mixture of steam and water. 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 UA determined using primary fow, heat flux and secondary side pressure and mass to compute primary and secondary side film resistance and tube resistance and 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 the plant design thermal-hydraulic conditions and the known steam generator performance from actual plant experience. A steam generator water level correlation is provided, however this is normally used for information only. The reactor trip and auxiliary feedwater start on steam generator water level is based on a user-input value of an equivalent secondary side mass; this value is chosen by the user based on a much more detailed steam generator model which computes steam generator water mass at the reactor trip level setpoint. In addition to steam flow demanded by the turbine, steam relief through safety valves, the steam dump system and through pipe breaks is simulated. 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 ,

$$\frac{fL}{D} = 0,$$

is used to compute break flow. Steam and feedline isolation are also simulated, including the effects of a failure to isolate one or more loops. 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

(Continued)

injected to 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 due to a loss of water inventory. The user determines the appropriate input depending on which direction of the input is conservative for the transient of interest, and in conjunction with prior results from much more detailed steam generator models. .

Engineered Safeguards Simulation

The LOFTRAN code simulates a safety-injection system consisting of a pump, boron injection tank, and associated piping. 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 CL location specified by the user. The pump section is assumed to come from the refuelling 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 pressuriser and the cold leg. Both systems employ a homogenous slug flow model and inject into the cold leg between the steam generator and reactor coolant pump.

Control and Protection Simulation

Control systems simulated are automatic rod control, steam dump control, feedwater control (not currently available) and pressuriser pressure control via pressuriser heaters and spray. A complete digital simulation of each control sytem 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 include reactor and turbine trips, safety injection actuation, and steam and feedline actuation. The reactor trips simulated and the actuators for safety injection and steam and feedline isolation are given in Table 2. 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 independantly of the LOFTRAN code from the specific system design characteristics, and in the US are verified on the actual plant at time of startup.

LOFTRAN Code Solution Sequence

The general sequence of calculations performed in a LOFTRAN run is shown in the block diagram presented in Figure 1.

WESTINGHOUSE PROPRIETARY CLASS 3

FIGURE 1

BLOCK DIAGRAM OF LOFTRAN SOLUTION SEQUENCE

a,c

TABLE 1

INITIATING EVENTS FROM PWR FAULT SCHEDULE CAPABLE OF BEING ANALYSED WITH LOFTRAN

I Fault Category No. 1 - Spurious Trip

Reactor Trip from Spurious Guardline Operation.

Spurious drop of all rods.

II Fault Category No. 2 - Increase in Heat Removal by the Secondary System

Feed System Fault leading to a Decrease in Feed Temperature.

Feed System Fault leading to Excessive Flow in one Steam Generator.

Feed System Fault leading to Excessive Flow in all Steam Generators.

Accidental Depressurisation of the Main Steam System.

Main Steamline Break outside containment.

Main Steamline Break inside containment.

III Fault Categroty No. 3 - Decrease in Heat Removal by the Secondary System.

Loss of Main Feed to one Steam Generator.

Loss of Main Feed to all Steam Generators.

Double Turbine or Condenser Trip.

Grid Underfrequency

Spurious Closure of one MSIV.

Spurious Closure of all MSIV's.

Main Feedline Break.

WESTINGHOUSE PROPRIETARY CLASS 3

TABLE 1 (Continued)

IV Fault Category No. 4 - Decrease in Reactor Coolant System Flow Rate.

Partial Loss of Reactor Coolant Flow.

Complete Loss of Reactor Coolant Flow.

Reactor Coolant Pump Locked Rotor.

Reactor Coolant Pump Shaft Break.

V Fault Category No. 5 - Reactivity Faults

Uncontrolled RCCA Bank Withdrawal of Power.

Uncontrolled Single RCCA Withdrawal. (RCS Transient)

Multiple Dropped RCCA's. (RCS Transient)

Dropped RCCA Bank. (RCS Transient).

Boron Dilution at Power, Manual Control.

Boron Dilution at Power, Automatic Control.

VI Fault Category No. 6 - Increase in Reactor Coolant Inventory.

Excessive Charging Flow at Power.

VII Fault Category No. 7 - Decrease in Reactor Coolant Inventory.

Steam Generator Tube Rupture.

Stuck Open Pressuriser PORV/SRV.

TABLE 2

REACTOR PROTECTION SYSTEMS
SIMULATED BY LOFTRAN

I Reactor Trip is Actuated by:

1. Low Flow.
2. Nuclear Power.
3. High Pressuriser Level.
4. High Pressuriser Pressure.
5. Low Pressuriser Pressure.
6. Overtemperature Delta-T.
7. Overpower Delta-T.
8. Low DNBR (N-16).
9. High Kw/ft (N-16).
10. Low Steam Generator Level (Mass).
11. Safety Injection.
12. Turbine Trip.
13. Manual.

II Safety Injection is Actuated by:

1. Low Pressuriser Pressure.
2. Low Pressuriser Pressure and Level.
3. Low Steam Pressure (lead/lag).
4. Manual.

III Steam Line Isolation is Actuated by:

1. Low Steam Pressure (lead/lag).
2. High Steam Pressure Rate (rate/lag).
3. Manual.

TABLE 2 (Continued)

IV Feedline Isolation is Actuated by:

1. Safety Injection.

2. Manual.

V Auxiliary Feedwater Start Actuated by:

1. Safety Injection.

2. Low Steam Generator Level (Mass).

3. Manual.

VI Turbine Trip is Actuated by:

1. Reactor Trip.

2. High Steam Generator Level.

3. Manual.

INPUT/OUTPUT

GENERAL

Automatic preparation of input for LOFTRAN minimizes input preparation time. In the extreme, representative results can be obtained by inputing two numbers plus a transient index. On the other hand, more than 500 numbers can be input to the present model.

All input data not manually input are assumed to be non-critical parameters which can be adequately scaled from current PWR design values. It is the user's responsibility to know which parameters are important for his usage and to verify their values.

Input for the first data deck consists of two title cards, a master control card, and three groups of input data (3 NAME LIST groups), in the following order:

1. Title 1 - title of study

2. Master control card (MASTER)

3. Title 2 - title of run

4. Fluid Systems, thermal and hydraulics data, and nominal design parameters (FLSTHD)

5. Initial operating conditions, control system set points, perturbation (ØPCØND)

6. Nuclear Data, reactivity coefficients, code control (RECØCD)

A-2, 3, 4
(Continued)

For subsequent runs Title 1 is not read in, and a stacking control card first in that data deck, specifies which of the other groups are to be read in.

Title cards 1 & 2 are standard. That is, any alphanumeric information can be placed on them for desired identification. (72 column field)

The master control card contains nominal power and flow, and the transient modes to be used. This card may be input as either format data or NAME LIST. A more complete description of it is on the suceeding pages.

Items 4 thru 6 are input by use of three NAME LIST statements (see FORTRAN MANUAL). The definition of variable names is also attached.

Four integers of two digits each (4I2) on the stacking control card specify what input cards to read on subsequent data decks. (One integer for the master control card and one integer for each of the NAME LIST groups.) Meanings of the integer values are:

0 - Do not read or rescale (all variables retain their value from the previous run and omit the namelist data card).

1 - All values including initial conditions are unchanged unless changed by namelist. The value of a variable at time zero as input by a table will not override the initial conditions of the variable if they differ. In the first run initial conditions are calculated by the program. Sucessive runs use the previous run for the initial conditions unless other initial conditions are input. (Note that if power level is changed, these initial conditions will be erroneous).

Values which changed, with power level for instance, need to be recalculated. This can be done in two ways. The new number can be input or set certain numbers equal to minus one(1). This will allow the variables in ØPCØND to be recalculated. The variables which can be rescaled are MSSGO, VPWO, ISO, TAVGO, HFWO, and HSO. The other way is to use a 2 on the stacking control card to rescale those variables.

Generally, 1 should be used and input initial conditions to be changed. (Easier than inputting all ØPCØND data not changed).

2 - All previous namelist values are void for this run. Input all variables needed (similar to first run).

Note: Steam and feed enthalpies are not calculated by the program during a run but remain the same at time zero unless input as a table. Therefore, heat load is directly proportional to steam flow because of constant enthalpy during the run. If enthalpies are input as a table, a heat balance must be made to determine the proper ratio of final to initial steam flow to obtain desired final load.

Input groups read on subsequent runs are in the same sequence as in first data deck. Title 2 is always read.

Example 1

0 0 1 2
(Title of run)

$ØPCØND . . . . .$
$RECØCD . . . . . $

All input variable retain their previous value except:

1. Variables input in ØPCØND and RECØCD groups

2. RECØCD variables not input are rescaled.

Example 2

1 1 1 1
(master control card, or namelist master
(Title of run)
$FLSTHD . . . $
$ØPCØND . . . $
$RECØCD . . . $

All variables on the master control card, plus those specified in the NAME LIST groups, take new values. No rescaling is done.

A blank stacking control card causes the same action as (0 1 1 1).

A sample input is shown on the next page.

MASTER CONTROL CARD

Format - 1X, F7.2, F8.1, 17X, 13I3 (or NAME LIST "MASTER")

QHNØM - nominal stretch rating, MWt

QFNØM - nominal (design) flow, GPM

ITRAN*	- transient index*	Col. 34-36
MØDEQN	- nuclear power mode	Col. 37-39
MØDEPR	- pressurizer mode	Col. 40-42
MØDEFL	- flow (reactor coolant) mode	Col. 43-45
MØDEHL	- hot leg mode	Col. 46-48
MØDESG	- steam generator mode	Col. 49-51
MØDEST	- steam flow mode	Col. 52-54
MØDEFW	- feedwater mode	Col. 55-57
MØDEUA	- Stm. gen. UA selection	Col. 58-60
MØDEØT*	- Output format selection	Col. 61-63
MØDEPH	- Preheater selection	Col. 64-66
MPAPER	- output option	Col. 67-69

LOFTRAN Modes of Operation

MØDEQN (core power) (Col. 37-39)

0: internal computation of reactivity, nuclear power, and core heat flux
1: as 0, plus input reactivity
2: input reactivity only, (no feedback)
3: input nuclear power for $t > t_{scram}$ (q_n = 0 for $t < t_{scram}$),
4: input core heat flux for $t > t_{scram}$, fraction of nominal
5: input $C_b(t - t_{SISA})$, QNT (TN) table, fraction of nominal
6: input nuclear power as a function of time (starts at t - 0.), fraction of nominal
7: input core heat flux as a function of time (starts at t = 0.), fraction of nominal

All these variables are input via the QNT(TN) table.

MØDEPR (Pressurizer) (Col. 40-42)

0: normal, (pressure interation for volume balance)
1: constant pressure
2: ΔP_{surge} line = 0 for t > 50 sec.
3: input mass relief rates (LBM/Sec) in Tables QSVS and QSVW
4: input mass relief rates (LBM/Sec) for steam relief in QSVS, and safety/relief valve flow area (ft^2) including any desired multipliers or discharge coefficients in Table QSVW. This area is multiplied by critical mass velocity in LBM/Sec ft^2 obtained from an intrinsic table as a function of pressure and enthalpy. These rates correspond to the homogeneous equilibrium subcooled flowrates specified by ANS-N611
5: QSVS, PSVS is safety/relief valve flow area (ft^2) vs. pressurizer

pressure for both water and steam relief. ANS homogeneous subcooled tables are used for subcooled flow and the Moody Correlation is used for saturated steam and water flow.
QSVW, PSVW is safety/relief valve flow area (ft^2) of any failed valves, i.e., once these valves open, they will not close.
Total flow from pressurizer valves is defined as that calculated from QSVS table + that from QSVW.
This mode also has the option of using multipliers on these relief rates. XSVS (steam) and XSVW (water) may be input and multiply relief rates directly. These multipliers default to 1.

6: same as 5 except homogeneous saturated tables are used for saturated water relief instead of the moody correlation.
7: as 5, except QSVW table is set to zero.
8: as 6, except QSVW table is set to zero.
Caution .... options 7 and 8 do not permit the stacking of several cases unless all relief tables are re-input.

MØDEFL (Reactor coolant flow) (Col. 43-45)

[ ]a,c

MØDEHL (Hot leg -- surge line) (Col. 46-48)

0: normal iteration for flow match

MØDESG (steam generator) (Col. 49-51)

0: no tube rupture
1: tube rupture at inlet header
2: tube rupture at outlet header.

MØDEST (steam flow) (Col. 52-54)

0: $w_s = W_s^o$ for $t < t_{scram}$,
$W_s = 0$ for $t > t_{scram}$
1: as 0, but proportional steam dump control for $t < t_{scram}$ (includes pressure effect on dump)
2: as 1, but pressure effect on load for $t < t_{scram}$
3: steam break at $t = 0$, constant flow area (Use WSDO in FLSTHD)
4: as 3, less turbine flow after trip (awkward to use: 7 is a preferable mode)
5: as 3, less closure of isolation values
6: combination of 4 & 5
7: as 3, plus turbine flow (with pressure effect on load) for $t < t_{scram}$
8: blank
9: blank
10: blank
11:+ input steam flow (WST(TWS) and WSDMPT (TWSDMP) tables as a function of time, with pressure effect on load. (input as a fraction of initial steam flow) Steam enthalpy interpolated from HST (THS) table as a function of time.
12:+ As 11, but without pressure effect on load.
13:+ As 11, but input as a fraction of nominal steam flow.
14:+ As 13, but without pressure effect on load.

+ Used to actuate automatic steam dump.

(Continued)

MØDEFW (Feedwater flow) (Col. 55-57)

0: $w_{fw} = w_s^0$ for $t < t_{scram}$, $w_{fw} = 0$ for $t > t_{scram}$
1: as 0, except on-off T_{avg} control of feedwater for $t > t^{*}_{scram}$
2: as 0, except $w_{fw} = w_s^{nom}$ for $t > t_{scram}$
3: as 0, except max. valve closure rate (20 sec. for <u>full</u> stroke) for $t > t_{scram}$
4: $W_{fw} \rightarrow 0$ at max. valve closure rate for $t > 0$ (loss of feedwater accident)
5: $W_{fw} = 1.12\ w_s^{nom}$ for $t > 0$
6: as 5, but max. valve closure for $t > t_{scram}$
7: $W_{fw} = w_s$ before trip, on-off T_{avg} control after trip (20 sec. stroke time)
8: blank
9: blank
10: blank
11: input feedwater flow (WFWT(TWFW)table), fraction of initial flow as a function of time. Feedwater enthalpy interpolated from HFWT(THFW) table as a function of time.
12: auto feed control if MØDEFW=12 (see Figure 3-8). Feedwater enthalpy as in mode 11 above.
13: internal calculation of feed enthalpy during transient

MØDEUA (steam generator UA) (Col. 58-60)

a,c

MØDEØT (printout) (Col. 61-63)

MØDEPH (preheat option) (Col. 64-66)

0: non preheat steam generator
1: pre-heat steam generator

MPAPER (short printout) (Col. 67-69)

0: long printout on paper, short printout on film
1: short printout on paper, long printout on film
Defaults to 1

FLSTHD INPUT DATA

Fluid Systems Thermal and Hydraulic Data.
Nominal Design Parameters

These input data are read into INPUT1, under the NAME LIST name FLSTHD. A sample output format for this data shown on the following page. For all these parameters, internally scaled values are used if not input.

Input data in this section are:

1. Reactor coolant fluid volumes
2. Flow fractions, auxiliary system volumes
3. T_{avg} program and nominal full power fluid conditions
4. Fuel and clad characteristics
5. Secondary system parameters
6. Safety valve characteristics
7. Reactor coolant pump parameters
8. Reactor coolant loop pressure drops at design full load
9. Flow areas and dimensions

8.2.1 Basic Reactor Coolant System Fluid Volumes

a,c

a,c

The following parameters are computed and printed out with the above group:

a,c

8.2.2 Flow Fractions, Auxiliary Volumes, and Miscellaneous Primary Constants

a,c

The following parameter is computed and printed out with the above group:

a,c

A-2, 3, 4

(Continued)

a,c

8.2.3 T_{avg} Program and Reactor Coolant Fluid Conditions at Nominal Full Power

[ ]a,c

The following parameters are computed and printed out with the above group:

[ ]a,c

A-2, 3, 4

(Continued)

a,c

8.2.4 Fuel and Clad Characteristics

This section included data for overall UA (fuel-to-coolant, Mass Fuel, and clad MC_p)

a,c

3.2.5 Secondary System Parameters and Nominal Design Conditions

a,c

A-2, 3, 4

(Continued)

a, c

Pre-heat Steam Generator Data

a, c

A-2, 3, 4

(Continued)

a,c

The following parameters are computed and printed out with the above group.

a,c

8.2.6 Pressurizer Safety Valve Characteristics (Steam and Water Relief), and Loop Safety Valve. (Note: the following description is only good for MODEPR of 0, 1, or 2. See MODEPR description for further information.)

a,c

8.2.7 Reactor Coolant Pump Parameters

a,c

(Continued)

a,c

8.2.8 Reactor Coolant Loop Steady-State Frictional Pressure Drops At Design Full Flow

a,c

a,c

8.2.9 Fluid Flow Areas and Dimensions Used in Computing Momentum

a,c

A-2, 3, 4

(Continued)

a,c

3.3 ØPCØND INPUT DATA

Initial Operating Conditions, Control System Set Points, Perturbation Data

These data are read by INPUT2 under NAMELIST ØPCØND and printed out by IN2ØUT. A sample of the IN2ØUT output is shown on the following two pages. All data (except WST and TWS) are internally scaled from the FLSTHD parameters unless input by the user.

1. Initial Primary Plant Conditions
2. Secondary Plant Initial Conditions
3. Reactor Protection System Set Points
4. Overpower-Overtemperature Protection Limits and DNB Correlation
5. Pressurizer Pressure Control System Set Points
6. Input Heat Flux, Nuclear Power, or Reactivity
7. Input Feedwater, Steam Flow and Steam Generator Conditions
8. Steam Generator UA vs. Time
9. Input Reactor Coolant Flow
10. Safety Injection System Parameters
11. Miscellaneous Baloney
12. Lag Time Constants for Tavg and Delta-T Signal
13. Overtemperature and Overpower Trip Set Point
14. TAU and GAIN arrays

Initial Primary Plant Conditions

a,c

The following variables are also computed and printed out with the above group:

a,c

(Continued)

SECONDARY PLANT INITIAL CONDITIONS

a,c

The following variables are computed and printed out with the above group:

a,c

A-2, 3, 4

(Continued)

Reactor Protection System Set Points and Delay Times

a,c

A-2, 3, 4

(Continued)

a,c

(Continued)

a,c

a,c

8.3.5 Pressurizer Pressure Control Set Points (Heaters & Spray)

a,c

a,c

8.3.6 Input Heat Flux, Nuclear Power, or Reactivity

Table, QNT (TN) (12 points permitted)

a,c

(Continued)

Input Feedwater, Steam Flow and Steam Generator Conditions

a,c

If no values are entered for these tables, they are set equal to the appropriate initial value.

(Continued)

Input Steam Generator UA

a,c

Input Reactor Coolant Flow

a,c

a,c

The following input data refers to the accumulators.

a,c

A-2, 3, 4

(Continued)

a,c

(Continued)

Lag Time Constants for TAVG and Delta-T Signals

a,c

Overtemperature and Overpower Trip Set Point

a,c

The description of the GAIN's and TAU's are described on the following pages.

a,c

TAU and Gain Arrays

Time constants and gains used in the control and protection system are input via the TAU and GAIN arrays. Entries are defined below.

VARIABLE DEFINITION NUMBER IF NOT INPUT

a,c

A-2, 3, 4

(Continued)

VARIABLE DEFINITION NUMBER IF NOT INPUT

a,c

A-2, 3, 4

(Continued)

VARIABLE	DEFINITION	NUMBER IF NOT INPUT

a,c

(Continued)

VARIABLE	DEFINITION	NUMBER IF NOT INPUT

a,c

A-2, 3, 4

(Continued)

VARIABLE	DEFINITION	NUMBER IF NOT INPUT

a,c

A-2, 3, 4

(Continued)

VARIABLE	DEFINITION	NUMBER IF NOT INPUT

a,c

A-2, 3, 4

(Continued)

VARIABLE	DEFINITION	NUMBER IF NOT INPUT

a,c

A-2, 3, 4

(Continued)

VARIABLE	DEFINITION	NUMBER IF NOT INPUT

a,c

A-2, 3, 4

(Continued)

VARIABLE	DEFINITION	NUMBER IF NOT INPUT

a,c

A-2, 3, 4

(Continued)

RECØCD INPUT DATA

Reactivity Coefficients, Code Control Data, and Neutron Kinetics Parameters

These data read by INPUT3 under NAMELIST RECØCD. A sample of the values obtained by scaling (except TMAX) is shown on the following two pages. Note that this is the only input group not printed out with a FORMAT specification.

Subdivisions of this section are:

1. Reactivity coefficients

2. Code control data

3. Neutron kinetics parameters

Reactivity Coefficients

Refer to REACT (section 3.3.2) or DKMØD (section 3.3.3) for the way in which these coefficients are used.

The units for all coefficients are δk/unit change

a,c

A-2, 3, 4

(Continued)

a,c

a,c

8.4.2 Code Control Data

a,c

A-2, 3, 4

(Continued)

a,c

8.4.3 Netron Kinetics Parameters

a,c

Q-5: What data must be input? What guidance is provided to assure that appropriate data is used for licensing calculations?

A-5: See attached table of LOFTRAN input variables. Input is made via three name lists. LOFTRAN defaults to the values listed after each input variable and enclosed by parentheses, if that variable is not specifically input.

Guidelines for the determination of code inputs, including LOFTRAN, are issued for each accident reported in Chapter 15. These guidelines are used by all analysts in order to guarantee conservative and consistent input regardless of plant or analyst. A record of the calculations used to determine code inputs, as well as code results, is kept on file, which is available for audit by the NRC and customers.

Q-6: What are the assumptions used to develop the analytical models? What errors and limitations are imposed by these assumptions?

The claim of "conservatism", wherever made, should be supported with an explanation of what conservatism means in that context and why the assumption or result is conservative.

A-6: The assumptions used to develop the analytical models are summarized for each LOFTRAN model sub-routine described in Section 3 of WCAP-7907. The same section of WCAP-7907 contains comments on these models, which would provide the reader with an idea of nature and extent of any errors or limitations. In general, the errors are small and are usually attributable to programming refinements such as curve fit approximation of the steam tables. Limitations are apparent from the model descriptions. The major limitation is that LOFTRAN is not equipped (nor intended) to simulate LOCA transients.

In all cases, "conservative", when used in safety analyses, means that a value of a given parameter is chosen such that the consequences of the event under analysis are aggravated or more severe than expected. In some cases, different values of the same parameter will be conservative for different events. For example, a high auxiliary feedwater flow rate would be conservative for an accident which cools down the RCS, such as a steam line rupture, while a low auxiliary feedwater flow rate would aggravate the consequences of a loss of feedwater accident which heats up the RCS.

Question 7

- Westinghouse clarified that LOFTRAN is a system's code and that there are obvious limitations on its simplified model of the core. The computation of DNBR within LOFTRAN is a correlation meant to produce conservative values of DNBR based on Core Limits. Should one wish to more accurately model DNBR or to verify the LOFTRAN computation of DNBR, more accurate thermal/hydraulic codes can be used. This is also true for fuel temperatures. The following parameters can be produced by LOFTRAN to feed into FACTRAN and THINC for more detailed calculations:

    Average Core Heat Flux
    RCS Pressure
    Core Average Temperature
    Primary Flow
    Core Inlet Temperature
    Density

- Also discussed was the use of LOFTRAN in the SAR safety analyses. The FSAR text describes some of the more important bounding assumptions used for safety analyses. Excruciating detail for the assumptions to be used for each safety analysis can be found in the Safety Analysis Standards, which were available for examination at the meeting. It was clarified that the discussion of LOFTRAN should be limited to the appropriateness of its use in safety analyses.

- It was requested that Westinghouse provide a list of parameters output by LOFTRAN for inclusion in the SAR. The following page contains a table of computer code output parameters presented in the typical SAR.

- Westinghouse states that any upgrades to the multi-loop LOFTRAN code since its inception have just been to add additional features and

WESTINGHOUSE PROPRIETARY CLASS

LOFTRAN RESULTS PRESENTED IN A TYPICAL FSAR

	FEEDWATER MALFUNCTION 15.1.2	EXCESS LOAD 15.1.3	STEAMLINE DE-PRESSURIZATION 15.1.4	MAIN STEAM-LINE RUPTURE 15.1.5	LOSS OF LOAD 15.2.3	LOSS OF FEED 15.2.6	FEEDLINE BREAK 15.2.8	PARTIAL LOSS OF FLOW 15.3.1	COMPLETE LOSS OF FLOW 15.3.2	LOCKED ROTOR 15.3.3	ROD WITH DRAWL@POWER 15.4.2	DROPPED ROD 15.4.3	STARTUP OF AN INACTIVE LOOP 15.4.4	INADVERTANT ECCS 15.5.1	RCS DEPRES-SURIZATION 15.6.1
NUCLEAR POWER	X	X			X		X	X	X	X	X	X	X	X	X
THERMAL POWER	X		X	X			X				X	X	X		
TEMPERATURE (Avg., T_{hot}, T_{cold}, ΔT)	X	X	X	X	X	X	X				X	X	X	X	X
DNBR	X	X			X			THINC3 X	THINC3 X		X	X	THINC3 X	X	X
PRESSURIZER WATER VOLUME		X	X	X	X	X	X				X			X	
STEAM FLOW			X	X										X	
REACTIVITY			X	X			X								
PRESSURE (PRESSURIZER & VARIOUS POINTS IN RCS)	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
CORE BORON CONCENTRATION			X	X											
STEAM PRESSURE						X	X								
PRIMARY SYSTEM FLOW RATE				X				X	X	X			X		
FEEDWATER BREAK FLOW							X								
PRESSURIZER RELIEF RATE							X								

have not been modifications to the bases of the codes. The upgrades are documented via our internal QA procedure for computer codes. Included in that QA procedure is the use of standard cases to compare the results of the upgraded version with the previous version of the code. The transients included in these standard cases are:

- RCS Heatup with Offsite Power Available
- Steamline Break
- Excessive Load Increase
- Startup of an Inactive RCS loop
- Loss of Main Feedwater with a Concurrent Loss of Offsite Power

These problems are intended to properly exercise LOFTRAN and reveal any coding errors.

Additional test transients are used to specifically exercise any new features in LOFTRAN as necessary.

- It was agreed that sufficient information has been provided to address this question.[1]

- The response to this question is applicable to both LOFTRAN and MARVEL, since both codes are/were used to report transient analysis results in SAR's.

Question 8

- This question deals with Control Systems. Control system actuation is not assumed in the SAR. Westinghouse makes no claim that LOFTRAN or Marvel adequately models control systems for the purpose of this review.

- It was agreed that sufficient information had been provided to address the concerns.[1]

Question 9

Westinghouse typically provides input into the startup testing performed by the utility to verify either directly or indirectly the basic assumptions of the SAR and to verify the applicability of our models; however we do not usually receive detailed data following plant testing. There is much information in the previously supplied verification package (NS-EPR-2536).

For this transient in particular, Westinghouse performed analyses meant to bound the actual plant transient to verify that applicable safety criteria were met; and the results have been sent to the utility. This information is available from the utility.

- It was agreed upon that sufficient information had been provided to address the concerns.[1]

Question 10

- The concerns expressed b[illegible]/ORNL were related to the way in which LOFTRAN handles primary side voiding. Westinghouse explained that the pressurizer may be water solid, steam, or water and steam, not necessarily at equilibrium conditions.

- It was stated that LOFTRAN has a complete pump model for flow calculations and accurately predicts flow and pressure drops throughout the RCS.

- Westinghouse stated that the method of flow calculation and the convergence procedure were based on the conservation of Volume, Mass, and Energy.

- NRC/ORNL explained that their concern centered on LOFTRAN ability to handle voiding in the reactor vessel upper head. The St. Lucie incident concerned an extensive RCS cooldown which considerably voided the upper head. This voiding caused pressure hangups by acting like a second pressurizer.

- Westinghouse stated that there is a thick metal model available for heat transport in LOFTRAN but that it would not be used unless it were conservative to do so. Westinghouse confirmed that a small amount of voiding may occur in the upper head during very adverse main steamline rupture transients, but that LOFTRAN has been studied by Westinghouse to observe any pressure hangup during the steamline break.

- Westinghouse also stated that LOFTRAN is used for a steam generator tube rupture event. Questions were raised concerning the adequacy of LOFTRAN for this long term application. Westinghouse confirmed that large scale voiding probably would not occur during the tube rupture, as evidenced by the 1979 Prairie Island tube rupture event (documented in SG 79-11-030) which showed no significant upper head voiding. In addition, LOFTRAN adequately handles relatively smaller amounts of voiding based on experience with steamline break transients. Furthermore, preliminary analyses of the Ginna event using LOFTRAN indicate that the effects of upper head voiding or major systems parameters (i.e., RCS pressure, pressurizer level, etc.) are adequately simulated.

- Westinghouse agreed that this will be an open item while information is collected to properly respond to this question.

To summarize the discussion on this question, which is the only identified open item in the LOFTRAN/MARVEL review:

Westinghouse explained the ability of LOFTRAN to adequately model the pressurizer during water solid, steam, or two-phase conditions. LOFTRAN has been verified to adequately model a small amount of voiding in the upper head during an extreme condition such as the steamline break transient.

The response in the upper head to the Ginna tube rupture is similar to that of the St. Lucie event. A Westinghouse evaluation of the Ginna tube rupture event will be completed soon and will contain comparisons between the plant transient and LOFTRAN predictions. We believe that LOFTRAN can adequately model the plant response to a tube rupture event even when there may be significant voiding in the upper head. This is supported by preliminary results of the analysis for the Ginna tube rupture event. Applicable final results of this evaluation will be made available for NRC/ORNL review by the end of November. This tube rupture information, when supplied, will complete Westinghouse response to all NRC/ORNL LOFTRAN/MARVEL review questions.

Question 11

Control system actuation is not assumed in the SAR. See also the response to question 8.

Question 12

LOFTRAN is not used for any transients where saturation conditions might be reached in the RCP. Should such conditions be suspected to occur, the use of LOFTRAN would be questioned and justified or rejected.

- It was agreed that sufficient information had been provided to address this concern.[1]

QUESTION 13

- The NRC explained that the concern which this question meant to address was whether LOFTRAN adequately models mixing in the Reactor Vessel.

a,c

-

-

- For background information, Westinghouse described some of the tests performed. Different configurations for 2, 3, and 4 loops were run with one loop colder than the rest. Contour maps of the temperature distribution form the basis for determining bounding mixing fractions.

- It was agreed that sufficient information had been provided to address this concern.[1]

Question 14

This concern is addressed in the response to Question 10.

- It was agreed that sufficient information had been provided to address this concern.[1]

Question 15

This concern is addressed in the response to Question 10.

- It was agreed that sufficient information had been provided to address this concern.[1]

Question 16

Westinghouse stated that this is a limitation but not a serious one. LOFTRAN cannot model reverse flow in the loop with the pressurizer, and this condition is not assumed in any safety analyses.

- It was agreed that sufficient information had been provided to address this concern.[1]

Question 17

The other methods for computing UA are more simplified versions of MODEUA=3 which is the one described in the text. The functions of the other options are simplified subsets of MODEUA=3 and need not be addressed further in the text.

- It was agreed that sufficient information had been provided to address this concern.[1]

Question 18

This question had been addressed in December, 1981 NRC-ORNL-Westinghouse meetings, and further explanation was provided at the meeting of July 13 and 14.

- It was agreed that sufficient information had been provided to address this concern.[1]

Question 19

The functions in question deal with simple calculations; however the following is offered to supplement WCAP-7907 to further describe these functions:

PROPV, CALCSV, TEMPH - are steam-water property functions which rely on some tabular data.

CRIT, RELIEF, HOMOSAT - are functions which rely on some tabular data. CRIT supplies MOODY critical flow and is used to calculate steam break flow. Relief and HOMOSAT are pressurizer water relief models based upon Homogeneous Equilibrium subcooled and staturated models respectively and are available for selection by the user depending upon the purposes of each particular analysis. These functions are the same standard functions used in other Westinghouse applications.

ZALODEK - Computes critical flowrate based on the Zaloudek correlation.

- It was agreed that sufficient information had been provided to address the concerns.[1]

Question 20

The equations for neutronics calculations are numerically stable for any size time step; however Westinghouse is aware that LOFTRAN has limitations using point kinetics which restricts its use for very fast reactivity excursions. LOFTRAN is not used for applications where fast neutronics transients occur. Fast neutronics transients such as Rod Ejection or a Rod Withdrawal from Subcritical Conditions are not analyzed using either LOFTRAN or MARVEL.

- It was agreed that sufficient information had been provided to address this concern.[1]

Question 21

The verification material in the previously supplied verification package (NS-EPR-2536) presents comparisons of LOFTRAN and MARVEL results with plant transints and other safety analysis codes. These were discussed and all plots were properly identified.

- It was agreed that sufficient information had been provided to address this concern.[1]

Question 22

Control system actuation is not assumed in the SAR. See also the response to question 8.

Question 23

LOFTRAN is able to model the release of either steam or water out of the pressurizer relief valves. This had also been discussed during the NRC/ORNL/Westinghouse meeting in December, 1981.

- It was agreed that sufficient information had been provided to address this concern.[1]

Question 24

Control system actuation is not assumed in the SAR. See also the response to question 8.

Question 25

The bubble rise model was discussed. Basically, the assumption is that steam bubbles are homogeneously dispersed in the liquid (i.e. there is no gradient in the bubble density).

- It was agreed that sufficient information had been provided to address this concern.[1]

Question 26

The hardwired functions of the computer changed when LOFTRAN was moved from the IBM 7094 on which it was originally developed to the CDC-7600 where it resides now. The CDC gave a fatal error stop when dividing by zero whereas the IBM had set the result of a division by zero equal to zero. The funtion DVCKF was called where appropriate to yield the same performance with the CDC as LOFTRAN did with the IBM. It was agreed that there are phsyical situations where it would be appropriate to use this function.

DVCKF is not used as an error trap and there are no consequences, as the function is presently used, which violate the accuracy of the LOFTRAN code.

- It was agreed that sufficient information had been provided to addresss this concern.[1]

Question 27

The NRC/ORNL explained that the concern rested on LOFTRAN's ability to model postulated physical situation where two separated phases may exist in the thermo-hydraulic loop.

The ability of LOFTRAN to model conditions where voiding may occur in the RCS was discussed in detail in questions 10 and 12.

LOFTRAN is currently not used under conditions where steam moving with water might occur in any part of the active loop(s). Should such conditions be suspected to occur, the results would not necessarily be invalid; but would be questioned on a case by case basis.

The use of LOFTRAN to analyze the Accidental Depressurization transient, as reported in Safety Analysis Reports, is appropriate since the objective of the analysis is to determine whether DNB conditions are reached in the short-term (before the automatic reactor trip). LOFTRAN is not used for longer-term analyses of this type, where two-phase conditions in the RCS active flow regions may occur.

- It was agreed that sufficient information had been provided to address this concern.[1]

Question 28

The NRC has conducted detailed reviews of LOFTRAN and MARVEL with respect to Quality Assurance and found the Westinghouse Quality Assurance procedures to be acceptable. Audits of LOFTRAN and MARVEL are performed by the NRC periodically to confirm the application of those procedures. The most recent QA audit was completed in 1981.

The Quality Assurance procedures concerning the development and verification of our computer codes lies in the Westinghouse Quality Control Standard-2, WCAP-8370, Revision 9A, "Westinghouse Water Reactor Division Quality Assurance Plant." A proprietary description which has been reviewed by the NRC is contained within WCAP-9550, "NSSS WRD Policies and Procedures."

To briefly summarize, changes to the code consist of updates to the previous versions which are kept to ensure traceability. Codes are protected such that changes can be made only by the group overseeing the use of all configurated codes based upon recommendations by the assigned cognizant group which are backed up by documentation controlled by our QA procedure.

Review of MARVEL NRC/ORNL Questions

Development of the Marvel Code

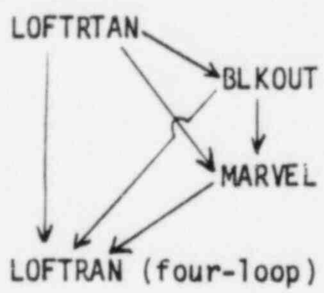
The MARVEL code and the four-loop LOFTRAN code both had their origins from the one-loop LOFTRAN code developed in the mid 60's to satisfy a need for a transient code which could handle a fast transient with substantial non-linearities over a period of seconds. The original LOFTRAN code was developed specifically for the analysis of a Reactor Coolant Pump Locked Rotor transient in the one-loop ZORITA plant and was developed from analytical procedures used in hand calculations performed at the time. It was developed solely on an as-needed basis to improve both the accuracy and the time required for transient analyses.

Following this there was a need for a code able to simulate long-term assymetrical transients in a multi-loop PWR. Using the one-loop LOFTRAN as a basis, the two-loop code BLKOUT was developed which was designed to simulate slow, long-term transients. LOFTRAN was still used for fast transients.

Recognizing the need for a multi-loop code able to model assymetric transients, the two-loop MARVEL code was written using the same analytical bases as the original one-loop LOFTRAN code, with BLKOUT's ability to model assymetric transients. MARVEL also had additional benefits such as the ability to model entrainment in the steam blowing down through a ruptured steam line. However, MARVEL did have one major disadvantage compared to LOFTRAN; its designer used a very unorthodox programming technique which made updates or further development difficult.

As operating plant experience increased, and nuclear regulatory activities increased, the need to model more assymetric transients under various postulated conditions increased. Additional abilities were required for the MARVEL code. MARVEL was difficult to modify for any new applications or major improvements. The need existed for a single code which would be able to handle separate events and be used exclusively for the analysis of all types of transients in a Westinghouse PWR. The decision was made to update the previous one-loop LOFTRAN into a four-loop code, which would take advantage of the modular programming architecture of LOFTRAN (making it readily adaptable for improvements or specific applications). The four-loop LOFTRAN code maintained the ability to model fast transients of the original LOFTRAN, the ability to model the long-term assymetric transients of BLKOUT, and the ability of MARVEL to model entrainment.

LOFTRAN is the principle systems transient code used today in the analysis of non-LOCA transients presented in Chapter 15 of the SAR. Because of their parallel development, the analytical bases, uses, and the calculational results of the LOFTRAN and MARVEL codes; they are nearly identical, and the calculational results have been verified by comparative studies. The only major differences between the two codes are the number of loops modeled and the programming architecture.



Background and Similarities Between LOFTRAN AND MARVEL

As discussed during the initial phase of the meeting, the LOFTRAN and MARVEL codes have similar origins and therefore have common bases. Since LOFTRAN and MARVEL are very similar codes, both having been developed from the same basic principles, and since the LOFTRAN and MARVEL questions are also similar, it was noted that practically all the responses provided for the LOFTRAN questions would also be applicable to MARVEL, thereby eliminating the need for specific responses to the MARVEL questions. Additionally, the similarity between the two codes has been backed by comparative material between LOFTRAN and MARVEL verifying that close agreement exists in the calculational results of the two codes. Any additional concerns raised by the NRC or ORNL concerning MARVEL during the meeting would be addressed in the meeting minutes.

- It was verified by Westinghouse that the previous discussion of the bases of the LOFTRAN code also apply to the MARVEL code.

- The interaction between MARVEL with FACTRAN and THINC and the requirement for the use of TRANFLO, where accurate steam generator calculations under transient conditions are required, are the same as that of LOFTRAN, therefore the discussion material from LOFTRAN is applicable.

- The verification method for the MARVEL code with respect to other codes and actual plant transients is similar to that of LOFTRAN; and the comparative basis between LOFTRAN and MARVEL only adds to the data base used for the verification of both codes.

- The same primary flow and pressure convergence procedure is used in LOFTRAN and MARVEL. The conservation of mass, volume, and energy are observed at all times. Westinghouse has noted that the MARVEL code may not calculate the pressure properly when the pressurizer is water-solid. Should such conditions occur, the results become suspect and the MARVEL code output includes a warning to the user.

- The slug flow model and the pressurizer surge line flowrate convergence scheme performed within MARVEL are the same as those for LOFTRAN. This process was reviewed and discussed for LOFTRAN, and that discussion is also applicable to MARVEL.

- ORNL raised some concerns with respect to a discrepancy in some results presented in a typical FSAR for the steamline break transient. These were discussed in detail and it was concluded that even if there were some discrepancy in the results, that it was not a code effect and therefore was not directly applicable to the discussion of the MARVEL code itself.

- In addition to the comparative studies between LOFTRAN and MARVEL presented in the verification package, for further information it was suggested that FSAR results for similar plants using the two different codes be examined. Representative transient results from the Beaver Valley Unit #1 FSAR using MARVEL and from the Beaver Valley Unit #2 FSAR using LOFTRAN were compared and similar transient effects and parameter values were favorably depicted by both codes.

Summary

LOFTRAN and MARVEL are two separate computer codes; however because of their origins and the similarity of the analytical bases, practically all of the responses to the LOFTRAN questions are applicable to MARVEL. These codes have the same bases, are used for the same transients, and give similar calculational results (verification material); therefore, in a pragmatic sense these codes can be discussed in the same light for the approval of the codes. The only difference in the abilities of the two codes is the pressure calculation in the MARVEL code following pressurizer fillup; this limitation on MARVEL is noted by Westinghouse and does not affect the results of any safety analyses, since the MARVEL code was not used to analyze any transients which fill the pressurizer.