Westinghouse Electric Corporation
Power Systems
Box 355
Pittsburgh Pennsylvania 15230 0355

November 7, 1986

NS-NRC-86-3180

Mr. James Lyons, Chief
Technical & Operations Support Branch
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Document Control Desk

Attention: Carl Berlinger, Reactor Systems Branch Chief
Division of PWR Licensing-A

Dear Mr. Lyons:

Enclosed are:

1. Twenty (25) copies of WCAP-8745-P-A, "Design Bases for the Thermal Overpower Delta-T and Thermal Overtemperature Delta-T Trip Functions," (Proprietary).

2. Fifteen (15) copies of WCAP-8746-A, "Design Bases for the Thermal Overpower Delta-T and Thermal Overtemperature Delta-T Trip Functions," (Non-Proprietary).

Also enclosed are:

1. One (1) copy of the Application for Withholding, AW-86-108, (Non-Proprietary), with Proprietary Information Notice.

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

The enclosed approved versions have been prepared for reference in future Westinghouse licensing applications, and are being submitted in accordance with NUREG-0390.

This submittal contains Proprietary information of Westinghouse Electric Corporation. 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 sets forth the basis on which the information may be withheld from public disclosure by the Commission.

Two Rids
T007
1/25 WCAP-8745-P-A
T008
1/15 WCAP-8746-A

Correspondence with respect to the Affidavit or Application for Withholding should reference AW-86-108 and should be addressed to R. A. Wiesemann, Manager of Regulatory and Legislative Affairs, Westinghouse Electric Corporation, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

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

WMS/bek/1852n

Enclosures

Westinghouse Electric Corporation
Power Systems
Box 355
Pittsburgh Pennsylvania 15230-0355

November 7, 1986
AW-86-108

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

SUBJECT: Transmittal of Approved Versions (WCAP-8745-P-A/WCAP-8746-A) of Topical, "Design Bases for the Thermal Overpower Delta-T and Thermal Overtemperature Delta-T Trip Functions"

Reference: Westinghouse Letter No. NS-NRC-86-3180, Rahe to Lyons, dated November 7, 1986

Dear Mr. Denton:

The 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. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The affidavit previously provided to justify withholding proprietary information in this matter was submitted as AW-77-16 with letter NS-CE-1390 dated March 28, 1977, and is equally applicable to this material.

Accordingly, it is respectfully requested that the subject information which is proprietary to Westinghouse 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-86-108 and should be addressed to the undersigned.

Very truly yours,

Robert A. Wiesemann

Robert A. Wiesemann, Manager
Regulatory and Legislative Affairs

WMS/bek/1852n
Enclosure(s)

cc: E. C. Shomaker, Esq.
Office of the General Council, NRC

PROPRIETARY INFORMATION NOTICE

TRANSMITTED HEREWITH ARE PROPRIETARY AND/OR NON-PROPRIETARY VERSIONS OF DOCUMENTS FURNISHED TO THE NRC IN CONNECTION WITH REQUESTS FOR GENERIC AND/OR PLANT SPECIFIC REVIEW AND APPROVAL.

IN ORDER TO CONFORM TO THE REQUIREMENTS OF 10CFR 2.790 OF THE COMMISSION'S REGULATIONS CONCERNING THE PROTECTION OF PROPRIETARY INFORMATION SO SUBMITTED TO THE NRC, THE INFORMATION WHICH IS PROPRIETARY IN THE PROPRIETARY VERSIONS IS CONTAINED WITHIN BRACKETS AND WHERE THE PROPRIETARY INFORMATION HAS BEEN DELETED IN THE NON-PROPRIETARY VERSIONS ONLY THE BRACKETS REMAIN, THE INFORMATION THAT WAS CONTAINED WITHIN THE BRACKETS IN THE PROPRIETARY VERSIONS HAVING BEEN DELETED. THE JUSTIFICATION FOR CLAIMING THE INFORMATION SO DESIGNATED AS PROPRIETARY IS INDICATED IN BOTH VERSIONS BY MEANS OF LOWER CASE LETTERS (a) THROUGH (g) CONTAINED WITHIN PARENTHESES LOCATED AS A SUPERSCRIPT IMMEDIATELY FOLLOWING THE BRACKETS ENCLOSING EACH ITEM OF INFORMATION BEING IDENTIFIED AS PROPRIETARY OR IN THE MARGIN OPPOSITE SUCH INFORMATION. THESE LOWER CASE LETTERS REFER TO THE TYPES OF INFORMATION WESTINGHOUSE CUSTOMARILY HOLDS IN CONFIDENCE IDENTIFIED IN SECTIONS (4)(ii)(a) through (4)(ii)(g) OF THE AFFIDAVIT ACCOMPANYING THIS TRANSMITTAL PURSUANT TO 10CFR2.790(b)(1).

AW-77-16

# 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 30 day of March 1977.

Notary Public

(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 WCAP-8745, "Design Bases for the Thermal Overpower Delta T and Thermal Overtemperature Delta T Trip Functions" (Proprietary), transmitted by Westinghouse letter number NS-CE-1390, Eicheldinger to Stolz dated March 31, 1977. The letter and attachment are being submitted in response to the NRC request dated February 12, 1975.

This information and the methodology it supports, when fully developed, will enable Westinghouse to:

(a) Justify one part of the design basis for the fuel.

(b) Justify that Westinghouse can assure customers of safe and efficient operation.

(c) Assist its customers to obtain licenses.

(d) Meet warranties.

(e) Optimize performance while maintaining high level of fuel integrity.

Further, this information, when fully implemented, will have substantial commercial value as follows:

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

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

(c) Westinghouse will use the information to offer nuclear fuel and related services to potential customers.

Further 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, in the application of empirically derived models based on prior data, and in the use of proven analytical models to simulate behavior of the fuel in normal operation and under hypothetical transients.

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 data in the attached document to normalize or verify their own methods or models. 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 man-years of Westinghouse effort and the expenditure of a considerable sum of money. Because these data are generic in application, competitors of Westinghouse would require the investment of substantially the same amount of effort and expertise to duplicate this information that Westinghouse possesses and which was acquired over a period of more than fifteen years and by the investment of millions of dollars. Over the years, this has included the development of heat transfer codes, nuclear analysis codes, transient analysis codes, core and system simulation methods and an experimental data base to support them.

Further the deponent sayeth not.

WCAP-8746-A

WESTINGHOUSE CLASS 3

# DESIGN BASES FOR THE THERMAL OVERPOWER ΔT AND THERMAL OVERTEMPERATURE ΔT TRIP FUNCTIONS

| ORIGINAL VERSION | APPROVED VERSION |
|---|---|
| March 1977 | September 1986 |


S. L. Ellenberger
K. T. Chen
J. A. Fici
T. Morita
L. R. Scherpereel
K. D. Sheppard
G. G. Uram
D. J. VandeWalle


APPROVED: Robert Wiesemann APPROVED: [illegible]

C. Eicheldinger, Manager
Nuclear Safety

F. W. Kramer, Manager
NFD Engineering


Work performed under project DGRP-43501

WESTINGHOUSE ELECTRIC CORPORATION
Nuclear Energy Systems
P. O. Box 355
Pittsburgh, Pennsylvania 15230

WESTINGHOUSE CLASS 3

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| NRC Safety Evaluation Report | B |
| WCAP-8746-A Text | C |

WESTINGHOUSE CLASS 3

# SECTION A

# NRC ACCEPTANCE LETTER

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

APR 17 1986

Mr. E. P. Rahe, Jr., Manager
Nuclear Safety Department
Westinghouse Electric Corporation
Post Office Box 355
Pittsburgh, Pennsylvania 15230

Dear Mr. Rahe:

SUBJECT: ACCEPTANCE FOR REFERENCING OF LICENSING TOPICAL REPORT WCAP-8745(P)/8746(NP), "DESIGN BASIS FOR THE THERMAL OVERPOWER AND THERMAL OVERTEMPERATURE $\Delta$T TRIP FUNCTIONS"

We have completed our review of the subject topical report submitted by the Westinghouse Electric Corporation (Westinghouse) by letter dated March 31, 1977. We find the report to be acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and the associated NRC evaluation, which is enclosed. The evaluation defines the basis for acceptance of the report.

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

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

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

Sincerely,

Carl H Berlinger

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

Enclosure:
As stated

WESTINGHOUSE CLASS 3

# SECTION B

# NRC SAFETY EVALUATION REPORT

SAFETY EVALUATION REPORT

Topical Report Title: Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions

Topical Report Numbers: WCAP-8745

Topical Report Date: March 1977

## 1. INTRODUCTION

This report describes the bases for the overpower and overtemperature ΔT trip functions in Westinghouse reactors, and the analytical methods used to derive the limiting safety system settings for the trips. These trip functions are designed to provide primary protection against departure from nucleate boiling (DNB) (overtemperature ΔT), and fuel centerline melt (overpower ΔT) through excessive linear heat generation rates (LHGR) during postulated transients. Since ΔT, the coolant temperature difference between vessel outlet and inlet, is (to a good approximation) proportional to the core power, and since the core power level is an important determinant of both DNBR and LHGR, the indicated ΔT serves as a useful primary parameter for these trip functions. Other parameters such as the average coolant temperature, the pressurizer pressure and the axial power offset modify the ΔT trip setpoint and thereby account for the effects of pressure on DNB, and of the power shape on both DNB and LHGR. In addition, delays in signal propagation are accounted for with rate-lag or lead-lag compensation.

The overpower and overtemperature ΔT trip functions involve the Westinghouse (i) design bases and methods for evaluating fuel centerline temperature and DNBR, (ii) calculational methods for core power distribution using coupled core and systems transient codes, and (iii) and the application of the computer codes: THINC, THINC-IV, LOFTRAN, TWINKLE and PANDA. However, the review of these design and calculational methods, and computer codes is considered outside the scope of the present evaluation. This review focused, instead, on the applicability of the methods and the results to the derivation of the overtemperature and overpower setpoint limits.

## 2. Summary of Topical Report

Section 1 of the report presents a short background, specifies the primary purpose of the overtemperature and overpower ΔT trips, and provides a summary of the report. The protection system and methods for set point determination described in the report are stated to apply to Westinghouse plants that reference RESAR-3S and operate under the guidelines of constant axial offset control.

Section 2 of the report specifies the design bases for core protection during normal operation, operational transients, and postulated transients occurring with moderate frequency, and describes the functional form of the thermal overpower and overtemperature trips. The general design criteria are specified as (i) $UO_2$ melting temperature will not be exceeded for 95% of the fuel rods at the 95% confidence level, and (ii) at least a 95% probability that DNB will not occur at the limiting fuel rod at a 95% confidence level. These criteria are to be met by restricting the calculated fuel centerline temperature to less than 4700°F, and by limiting the minimum DNBR to 1.3. A third design limit, namely that the hot leg temperature be maintained below saturation temperature enables the vessel average inlet/outlet coolant temperature difference (ΔT) to be used as a measure of the core power.

The overpower and overtemperature trips are activated on a two-out-of three logic for three-loop plants, and on a two-out-of-four logic for two and four-loop plants. The indicated vessel ΔT is continuously compared with the setpoint for each channel, which is calculated by analog circuitry programmed to evaluate the four-term setpoint representation. The leading term in the expression for the overpower ΔT setpoint is an adjustable, preset value of coolant temperature rise that is independent of the process variables. The second term is dependent on the average coolant temperature, and applies rate-lag compensation for pipe and thermal time delays. The third term accounts for the effects of coolant density and heat capacity on the relationship between ΔT and core power. The last term reduces the ΔT setpoint to account for adverse power distribution effects, and is a function of the axial flux difference. For the overtemperature $\Delta T$ setpoint, the leading term is also a preset adjustable value of $\Delta T$ independent of the process variables. The second term accounts for the effect of temperature on the design limits, and is lead/lag compensated for instrumentation and piping delays. The third term accounts for the effects of pressure on the design limits. As with the overpower $\Delta T$ trip, the last term accounts for the effects of adverse power distributions, and is dependent on the axial flux difference.

Section 3 of the report presents the procedures for calculating the safety setpoints for the overpower $\Delta T$ trip. The calculation proceeds in four steps: (1) A trip setpoint independent of the power distribution, typically at 118% nominal power level, is selected, (2) power level and distribution during control bank and boration/dilution system malfunctions are evaluated using a static nuclear core model without feedback, (3) the limiting LHGR occurring during these transients is compared to the threshold for fuel centerline melt, and (4) if the threshold is exceeded, either the trip setpoint is appropriately lowered or, more frequently, a trip reset function $f(\Delta I)$ is determined such that highly skewed power distributions are eliminated, and the threshold for fuel centerline melt is not exceeded. The evolution of Westinghouse methodology for calculating core power distribution effects is described in Section 3.2. The basic method consists of calculating the envelope of maximum $F_Q$, as a function of axial offset, for expected and unexpected plant maneuvers. Originally all maneuvers that satisfied the control rod insertion limits were admitted, and an $f(\Delta I)$ function was generated based on the peaking factor analysis. In response to concerns regarding fuel densification, the $f(\Delta I)$ trip reset function was made appropriately more restrictive, the trip setpoint of 118% was sometimes reduced, and operating restrictions on part length control rods were introduced. Later, the constant axial off-set control (CAOC) method of plant operation was introduced in response to the requirements of the loss-of-coolant accident (LOCA) emergency core cooling. CAOC operation maintains the axial power distribution within a specified band, diminishes the adverse effects of xenon transients and serves to lower core peaking factors. Therefore, $f(\Delta I)$ trip setpoints established prior to the introduction of the added CAOC constraints are considered conservative under CAOC operation. This situation exists for some 14x14 and 15x15 fuel assembly plants. For 16x16 and 17x17 fuel assembly plants operating under CAOC, Westinghouse analyses have indicated that no f(ΔI) function is required to preclude fuel centerline melting during overpower transients because the thermal overpower limit of 118 percent of rated reactor power alone provides adequate protection against fuel melting.

Section 4 of the report presents the procedures for calculating the safety setpoints of the overtemperature ΔT trip. The effects of core-wide parameters such as thermal power level and vessel average temperature are separated from power distribution effects by determining the former with a reference chopped cosine shape and accounting for the latter through the f(ΔI) portion of the trip. Assuming the reference power distribution, limits of safe operation are defined in the space of thermal power level, coolant inlet temperature, and primary system pressure. These limits of safe operation are determined by the conditions that the vessel exit temperature be less than the saturation temperature, and that the minimum DNBR be above 1.3. To account for the effects of adverse axial power shapes, a set of "standard power distributions" (a set of limiting shapes having various values of axial offset), are generated using three-dimensional static nuclear calculations. For each power shape, the power level that gives a minimum DNBR of 1.3 is determined by iterative use of the THINC code. The procedure generates an envelope of allowable power vs. axial offset for a given pressure and inlet temperature. Two envelopes, one at the inlet temperature corresponding to 118% power and the second corresponding to 80% power, are generated. The envelopes consist of positive and negative deadband regions of zero slope (in allowable power vs. axial offset), and regions of positive and negative slope. The widths of the deadband and the slopes are utilized to generate ΔT trip reset as a function of the axial offset, and hence determine f(ΔI).

Section 5 of the report describes analyses of anticipated transients with a coupled-core-system transient model used by Westinghouse to verify that the methodology of standard power distributions described in Section 4 is applicable under current plant operating procedures. The coupled-core-system model is a combination of the lumped-parameter single-loop system code, LOFTRAN, and the three-dimensional spatial neutron kinetics code, TWINKLE. DNB evaluations were performed with the THINC and equivalent codes. DNBR was calculated using the axial power distribution predicted with LOFTRAN/TWINKLE, a control-bank-position-dependent value of $F^N_{\Delta H}$, and the coolant conditions present at the moment. Five DNB-related transients were analyzed with the LOFTRAN/TWINKLE model: (1) uncontrolled bank withdrawal at power, (2) step increase in steam flow caused by equipment malfunction, (3) inadvertent opening of turbine throttle valve, (4) uncontrolled boron dilution at power with manual rod control, and (5) uncontrolled boration/dilutions with automatic rod control. Worst pre-accident core conditions (i.e., power level, control bank position and xenon distribution) to be used in the LOFTRAN/TWINKLE analyses were determined by analyzing a complete set of initial conditions with the static nuclear model. Sensitivity to such variables as bank worth, bank withdrawal speed, automatic versus manual rod control, moderator feedback and Doppler feedback were analyzed with the LOFTRAN/TWINKLE model to identify limiting transients and conservative assumptions.

The adequacy of the standard power shape methodology can be established if the f(ΔI) functions generated using this methodology can be shown to be conservative when compared with the results of the LOFTRAN/TWINKLE analyses. Section 6 of the report presents these comparisons which demonstrate that the f(ΔI) trip reset function generated using the standard shape methodology is conservative with respect to the results of the LOFTRAN/TWINKLE analyses.

## 3. Summary of Technical Evaluation

The evaluation of WCAP-8745 was based on an assessment of the general methodology presented, the scope and applicability of the methods discussed, uncertainties in the trip function design bases, and verification of the standard power shape methodology with the LOFTRAN/TWINKLE model. The following sections address each of these concerns.

### 3.1 General Methodology

The design bases and criteria for the overpower and overtemperature ΔT trip have been clearly defined and are consistent with Westinghouse general safety limits pertaining to maximum fuel temperature and minimum DNBR. The threshold for fuel centerline melt has been correlated with a limiting value of kw/ft. The correlation includes the effects of burnup, flow rate, power distribution asymmetry and initial fill gas pressure level, and is based on an approved PAD analysis and is therefore acceptable. The minimum DNBR of 1.3 assumed in the analyses is an acceptable thermal safety limit. The functional forms of the trip setpoints appropriately account for effects such as coolant density and pressure variation, adverse core power distribution and instrumentation and piping delays (in addition to the variations in core power level), and for monitoring LHGR and DNBR.

### 3.2 Scope and Applicability

Although Section 1 of the topical report specifies its applicability to Westinghouse plants that reference RESAR-3S and operate under CAOC, Westinghouse has indicated that they consider WCAP-8745 applicable to all Westinghouse plants that employ overpower and overtemperature $\Delta T$ trip for core protection. Westinghouse has stated that new methods and technology developed after the submittal of WCAP-8745 are described in separate topical reports, and do not invalidate the conclusions of WCAP-8745. As examples of such new methods, Westinghouse has cited changes in DNB analysis methodology (Improved Thermal Design Procedure and WRB-1 and WRB-2 correlations), fuel design (Optimized Fuel Assembly), and plant operating procedure (Relaxed Axial Offset Control), and referenced topical reports describing these changes. While we agree that the basic design philosophy described in WCAP-8745 is not invalidated by changes in DNB analysis methodology, fuel design, and plant operating procedure, the application of this methodology must account for changes in system design and operation. The adequacy of the standard power shapes in establishing the core DNB protection system must be evaluated whenever changes are introduced that could potentially effect the core power distribution.

### 3.3 Uncertainties in Trip Function Design Bases

In response to a request for information regarding uncertainties in the trip function design bases, Westinghouse has provided the error allowances included for bistable error, signal linearity and reproducibility, calorimetric error, error in the $T_{avg}$ measurement and error in the pressure measurement. Uncertainty in flow is accounted for by the use of a minimum technical specification flow in the analysis. The error allowances are arithmetically summed to obtain a total error allowance. Currently Westinghouse has introduced a method of statistically combining error allowances, and has verified the conservatism of the old error allowance methodology by several plant specific statistical setpoint calculations. The statistical method has been reviewed and approved by the NRC staff. Since the error allowance methodology of WCAP-8745 has been demonstrated to be conservative with respect to the statistical method, we find it acceptable.

### 3.4 Verification of the Standard Shape Methodology with LOFTRAN/TWINKEL

In support of the setpoint methodology, Westinghouse has provided the core axial offset, peak-to-average power, and shape of the standard power shapes used in the standard shape methodology. The adequacy of the standard power shape methodology was demonstrated by establishing that the f(ΔI) functions used in this methodology are conservative (for the prediction of DNBR) when compared with the results of the LOFTRAN/TWINKLE analyses. In the comparison, five DNB-related transients were chosen after sensitivity to bank worth, bank withdrawal speed, control rod operation mode, and moderator and Doppler feedback were analyzed to identify the limiting transients and conservative initial conditions. The $F_{\Delta H}$ versus rod position function used in the DNB analysis had been demonstrated to be conservative for 50 different rod insertions in nearly 30 different plants. The power shapes used in the DNB verification covered the entire cycle life. We therefore conclude that the comparison between the standard power shapes methodology and the LOFTRAN/TWINKLE analyses is sufficiently comprehensive in the choice of transients studied and in the applicability of the results to different Westinghouse core designs studied at sufficient points in the cycle life.

## 4. Recommendation

We have reviewed the Westinghouse design bases for the thermal overpower and overtemperature $\Delta T$ Trip functions described in WCAP-8745, and find them acceptable for referencing by Westinghouse in licensing documents for plants that operate under constant axial offset control.

## REFERENCES


1. F.E. Motley, et al., "New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids," WCAP-8762 (Proprietary), July 1976.

2. H. Chelemer, et al., "Improved Thermal Design Procedure," WCAP-8567 (Proprietary), July 1975.

3. J. Skaritka, et al., "Fuel Rod Bow Evaluation," WCAP-8691, Rev. 1 (Proprietary), July 1979.

4. S.L. Davidson, ed., "Reference Core Report 17x17 Optimized Fuel Assembly," WCAP-9500-A, May 1982.

5. R.W. Miller, et al., "Relaxation of Constant Axial Offset Control," WCAP-10216-P-A (Proprietary), June 1983.

6. S.L. Davidson, ed., "Reference Core Report Vantage 5 Fuel Assembly," WCAP-10444-P-A (Proprietary), September 1985.

WESTINGHOUSE CLASS 3

SECTION C

WCAP 8746-A TEXT

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# SECTION 1
# INTRODUCTION

## 1-1. BACKGROUND

In early 1975 the Nuclear Regulatory Commission (NRC) requested[1] Westinghouse to describe the analysis methods used to derive limiting safety-system settings for the thermal overpower $\Delta T$ and overtemperature $\Delta T$ trip functions. These trip functions provide primary protection against departure from nucleate boiling (DNB) and fuel centerline melting (excessive kw/ft) during postulated transients (ANSI N18.2 Condition II events) in Westinghouse reactors.

In this report responding to the NRC request, the protection systems and methods for setpoint determination apply to all Westinghouse plants which reference RESAR - 3S[2], and which are operating under the guidelines of constant axial offset control without part-length control rods (Mode A operation).

## 1-2. SUMMARY

Section 2 gives the design bases for core protection during postulated transients, and functionally describes the thermal overpower and overtemperature trips.

Section 3 gives the procedure for calculating safety settings for the thermal overpower trip. This is the trip designed to protect against fuel centerline melting (i.e., high kw/ft). Experience has shown that this function can be accomplished by ensuring that the gross core average thermal power does not exceed a prescribed limit (typically 118 percent of nominal). Thus safety settings for the thermal overpower trip are chosen to control the gross core thermal power within this limit.

Also covered in section 3 is the derivation of a compensating term introduced into the thermal overpower trip setpoint determination to account for important power distribution effects. The term is a function of the core axial flux difference $\Delta I$.

1. Letter from R. C. DeYoung, NRC Division of Reactor Licensing, to C. Eicheldinger, Westinghouse PWRSD, Nuclear Safety Department, dated February 12, 1975
2. RESAR-3S, "Reference Safety Analysis Report," Consolidated Version and Subsequent Amendments, July 1975, Docket No STN 50-545

Section 4 gives the procedure for calculating safety settings for the thermal overtemperature trip. This trip primarily provides core protection against DNB, which is a function of the core power level, coolant temperature, coolant pressure, and the core power distribution. As such, the thermal overtemperature trip has been constructed as a function of several measurable process variables which are closely related to these quantities. Safety settings for this trip are derived via a two-step process. In the first step, described in paragraphs 4-1 through 4-4, safety settings are derived which account for the effects of core power level, coolant temperature, and coolant pressure on DNB assuming a fixed reference core power distribution (e.g., a power-dependent value of $F_{\Delta H}$ and a fixed cosine axial power distribution). In the second step, in paragraphs 4-5 through 4-8, a compensating term is derived which accounts for core power distributions more severe with respect to DNB than the reference core power distribution. This compensating term, which is a function of $\Delta I$, is similar in nature to that employed in the thermal overpower trip, and is derived based on a set of standard non-symmetric axial power distributions.

To verify the appropriateness of the standard shape methodology employed in determining thermal overtemperature trip setpoints, a detailed evaluation of power distributions during postulated transients was conducted. This evaluation, based on a coupled core-system transient evaluation model, is discussed in Section 5 and evaluated in Section 6. All transients originally considered in the design basis for the overtemperature $\Delta T$ trip were analyzed, as well as several accidents not originally considered. [

]a,c

# SECTION 2

# CORE THERMAL OVERPOWER AND OVERTEMPERATURE PROTECTION

Fuel integrity is both an economic and a safety concern in the operation of a pressurized water reactor. In recognition of this, the Westinghouse design philosophy is to preclude all but a very limited amount of fuel damage during Condition I and II events. It is not possible to completely prevent all fuel-rod failures. The very small number of rod failures which may occur during Condition I and II events are within the capabilities of the plant cleanup system and are consistent with the design bases listed below. The specific concern of this report is with Condition II overpower and overtemperature events. The following sections will define the core protection design bases for such events, and describe the protection systems developed to meet these design bases.

## 2-1. DESIGN BASES FOR OVERPOWER AND OVERTEMPERATURE EVENTS

The core protection design bases for Condition II overpower and overtemperature events discussed below place limits on certain design variables. The thermal overpower and overtemperature protection systems are designed to maintain these design variables within their respective limits. The events to be discussed are:

- withdrawal of a rod cluster control assembly bank
- uncontrolled dilution and boration
- excessive plant cooldown due to a feedwater system malfunction
- excessive steam load increase
- excessive cooldown due to a turbine throttle valve malfunction

The thermal overpower and overtemperature trips provide primary protection against damage from these events.

### 2-2. Fuel Centerline Temperature Design Basis

The fuel centerline temperature design basis is: during normal operation, operational transients, and transient conditions arising from faults of moderate frequency (Condition I and II events), the Uranium Dioxide melting temperature shall not be exceeded for at least 95 percent of the limiting fuel rods at a 95 percent confidence level. If $UO_2$ melting is precluded, preservation of fuel geometry is assured, and the potentially adverse effects of molten fuel-cladding interactions are avoided. To achieve this, a calculated fuel centerline temperature limit of 4700°F has been selected. This is significantly below the actual $UO_2$ melting temperature to allow for fuel temperature calculational model and other uncertainties.[1]

### 2-3. Departure from Nucleate Boiling Design Basis

The departure from nucleate boiling (DNB) design basis is: during Condition I and II events, the probability that DNB will not occur on the limiting fuel rod(s) is at least 95 percent at a 95-percent confidence level. This basis is traditionally met by limiting the minimum departure from nucleate boiling ratio (DNBR) to 1.30. If DNB is precluded, adequate heat transfer is assured between the fuel cladding and the reactor coolant, and damage due to inadequate cooling is prevented.

### 2-4. Hot-Leg Boiling Limit

The hot-leg boiling limit is: during Condition I and II events, the hot-leg temperature shall be less than the saturation temperature. This limit is not a core protection limit. The protection system design uses the vessel average temperature difference ($\Delta T$) as a measure of the core power. To assure that $\Delta T$ is proportional to core power, hot-leg boiling must be precluded.

## 2-5. FUNCTIONAL DESCRIPTION OF THE THERMAL OVERPOWER AND OVERTEMPERATURE TRIPS

The core protection design bases described in paragraphs 2-1 through 2-4 impose limits on important design variables, which in turn are translatable into limits on related process variables. In particular, for a given coolant volumetric flow rate, these design bases can be related to the following process variables: core thermal power, coolant temperature, coolant pressure, and core axial neutron flux difference ($\Delta I$). The thermal overpower and overtemperature trips respond to unacceptable conditions (or combinations thereof) in some or all of these variables. A brief overview of these two trips is in order here, before the trips are described functionally in detail below. The thermal overpower trip is designed specifically to ensure operation within the fuel temperature design basis. Experience with Westinghouse PWRs has shown that this can be accomplished by controlling the gross core thermal power within a prescribed limit (typically 118 percent of nominal power). This is done through the overpower trip by correlating core thermal power with the temperature difference across the vessel ($\Delta T$). Since thermal power is not precisely proportional to $\Delta T$ because of the effects of changes in coolant density and heat capacity, a compensating term which is a function of

1. RESAR-3S, "Reference Safety Analysis Report," Consolidated Version and Subsequent Amendments, July, 1975, Docket No. STN 50-545.

vessel average temperature is factored into the overpower trip setting. Similarly, since the prescribed overpower limit may not be adequate for highly skewed axial power distributions, another compensating term (this one related to the axial flux difference) is factored into the overpower trip setting.

The thermal overtemperature trip is designed to ensure operation within the DNB design basis and the hot-leg boiling limit. Since both of these limits are functions of coolant temperature and pressure as well as core thermal power, the overtemperature trip is correlated with vessel $\Delta T$, vessel average temperature, and primary system pressure. A compensating term which is a function of $\Delta I$ is also factored into the overtemperature trip setting to offset the effect of core power distribution on DNB.

## 2-6. Thermal Overpower Trip Description

The thermal overpower protection function will trip the reactor when the compensated $\Delta T$ in any two channels exceeds the setpoint (three- and four-loop plants have one channel per loop; two-loop plants have two channels per loop). The setpoint for each channel is continuously calculated by analog circuitry programmed to evaluate according to the following equation:

$$\Delta T_{se\ point} = K_4 - K_5 \frac{\tau_3 s}{1 + \tau_3 s} T_{avg} - K_6 \left(T_{avg} - T^{\circ}_{avg}\right) - f(\Delta I)$$

where:

$\Delta T_{setpoint}$ = Overpower $\Delta T$ setpoint (percent of full-power $\Delta T$)

$K_4$ = A preset, manually adjustable bias (percent of full-power $\Delta T$)

$K_5$ = A constant that compensates for piping and thermal time delays (percent of full-power $\Delta T/^{\circ}F$)

$K_6$ = A constant that accounts for the effects of coolant density and heat capacity on the relationship between $\Delta T$ and thermal power (percent of full-power $\Delta T/^{\circ}F$)

$T^{\circ}_{avg}$ = Indicated* average reactor coolant temperature at full-power ($^{\circ}F$)

$T_{avg}$ = Average reactor-coolant temperature ($^{\circ}F$)

$\tau_3$ = Time constant (seconds)

$s$ = Laplace transform operator ($seconds^{-1}$)

* Calibration temperature for $\Delta T$ instrumentation, $\leq T_{avg}^{nom}$. Adjusted after plant startup test measurements.

f(ΔI) = A function of the neutron flux difference between upper and lower long ion chamber section (percent of full-power ΔT). Increases in ΔI beyond a predefined deadband result in a decrease in the trip setpoint.

The process inputs to the above equation are generated as follows. The temperature information for each channel (ΔT and $T_{avg}$) is obtained from independent pairs of resistance temperature devices in the hot and cold legs of each loop. The neutron flux information for each channel is supplied by a separate out-of-core detector. These detectors are long ion chambers mounted vertically outside the pressure vessel 90 degrees apart in plan. Each detector is 10 feet long, centered on the core horizontal midplane, and is divided into an upper and lower half. Where $P_t$ and $P_b$ represent the calibrated signals for a single channel from the top and bottom detectors respectively, ΔI is defined as $P_t - P_b$ and axial offset is defined as:

$$\frac{P_t - P_b}{P_t + P_b}$$

For a discussion of the time constant, $\tau_3$, in the above equation, refer to appendix A.

## 2-7. Thermal Overtemperature Trip Description

The thermal overtemperature protection function will trip the reactor when the compensated ΔT in any two channels exceeds the setpoint. The setpoint for each channel is continuously calculated according to the following equation:

$$\Delta T_{setpoint} = K_1 - K_2 \frac{1 + \tau_1 s}{1 + \tau_2 s} (T_{avg} - T_{avg}^{nom}) + K_3 (P - P^{nom}) - f(\Delta I)$$

where:

$\Delta T_{setpoint}$ = Overtemperature ΔT setpoint (percent of full-power ΔT)

$K_1$ = A preset, manually adjustable bias (percent of full-power ΔT)

$K_2$ = A constant based on the effect of temperature on the design limits (percent of full-power ΔT/°F)

$K_3$ = A constant based on the effect of pressure on the design limits (percent of full-power ΔT/psi)

$T_{avg}$ = Average reactor-coolant temperature (°F)

$T_{avg}^{nom}$ = Nominal average reactor-coolant temperature at full-power (°F)

$P$ = Pressurizer pressure (psig)

$P^{nom}$ = Nominal reactor-coolant system pressure (psig)

$\tau_1, \tau_2$ = Time constants (seconds)

$s$ = Laplace transform operator (seconds$^{-1}$)

$f(\Delta I)$ = A function of the neutron flux difference between upper and lower long ion chambers (percent of full-power $\Delta T$).

The source of temperature and neutron flux level information for the thermal overtemperature trip is identical to that for the overpower trip, and the resulting setpoint is compared to the same compensated $\Delta T$ signal from each channel. The required one pressurizer pressure signal per channel is obtained from separate sensors connected to pressure taps at the top of the pressurizer. For a discussion of the derivation of the time constants, $\tau_1$ and $\tau_2$ in the above equation, refer to appendix A.

Note that both the overtemperature and overpower trip equations given above are in units of percent of full power $\Delta T$. This is consistent with the manner in which the setpoints are entered into the protection system process instrumentation for all later-generation Westinghouse plants. In Plant Technical Specification documents, the equations may be written in units of degrees F obtained by multiplying by the indicated full-power $\Delta T$. In such a case, the constant terms $K_1$ to $K_6$ would be given in fractions of full-power $\Delta T$. The full-power $\Delta T$ is determined during plant startup test measurements.

# SECTION 3
# CALCULATIONAL BASIS FOR THERMAL OVERPOWER TRIP SETPOINTS

Historically, the purpose of the thermal overpower protection system in Westinghouse plants has been to prevent fuel center-line melting during Condition II transients. The overall approach taken to provide this protection is as follows: (1) A thermal overpower trip setpoint independent of power distribution is chosen. A value of 118 percent[1] power is typical. (2) Evaluate the power level and power distribution in the core during limiting transients (discussed below) through the use of static nuclear core models (no benefit for plant and core feedback is taken). (3) Compare the limiting kw/ft values in these transients to the values which would be equivalent to fuel centerline melting. (4) If the fuel centerline melt values are exceeded, either: (a) establish a lower (than 118 percent) power distribution–independent trip setpoint to preclude the high values of kw/ft; or (b) determine an appropriate flux difference trip reset function, f(ΔI), such that highly skewed power distributions, leading to high values of kw/ft, are eliminated. The normal procedure is to employ a flux difference trip reset function rather than to lower the 118 percent value. This section of the report discusses the application of design techniques, past and present, to the evaluation of the thermal overpower trip functions.

## 3-1. CORE POWER, COOLANT DENSITY, AND COOLANT HEAT-CAPACITY EFFECTS

As noted above, for most Westinghouse plants a thermal overpower trip setpoint equivalent to 118 percent of nominal power is typical, if we disregard the power distribution compensating term. Because thermal power is nearly proportional to the coolant temperature difference ΔT across the vessel (assuming a constant volumetric flow rate), the thermal overpower trip is designed to trip the reactor when the measured ΔT exceeds 118 percent of the nominal, full-power ΔT. Thermal power being not precisely proportional to ΔT due to coolant density and heat capacity effects, a compensating term is factored into the trip setpoint. The term is a function of the vessel average coolant temperature ($T_{avg}$). The derivation of the thermal overpower trip equation (without power distribution compensation) is detailed in appendix B.

1. The 118 percent value refered to here does not allow for instrumentation errors (see appendix B)

## 3-2. CORE POWER DISTRIBUTION EFFECTS

In the past, Westinghouse methods for establishing thermal overpower protection have been based on the so-called "flyspeck" approach. The basic method was to calculate the axial power distribution for both expected and unexpected plant maneuvers, including uncontrolled xenon transients. The philosophy behind this methodclogy has been discussed by McFarlane.[1] The "flyspeck", constructed by consideration of these maneuvers, shows the maximum expected value of $F_Q$ vs axial offset irrespective of the power level at which $F_Q$ was calculated. Appendix C describes determination of the f(ΔI) portion of the thermal overpower trip by means of the flyspeck.

This technique had been in use for several years when the fuel densification phenomenon was observed in Westinghouse plants. Because densification (and possible clad collapse) had the effect of increasing the local power on the fuel rods, a "power spike factor" was incorporated multiplicatively in the calculated values (see figure 3-1). The power spike factor, which increased with core height, had two effects on the flyspeck; (1) the $F_Q$ value at the deadband was increased; and (2) the slopes of the wings were increased. The direct effects of these changes were as follows. First, in some cases the $F_Q$ increase was so great that the trip setpoint of 118 percent had to either be reduced or operating restriction on the PL rods had to be enforced in the plant Technical Specifications, either through an insertion limit or total exclusion. This was particularly the case where clad collapse was predicted for the plant's operating cycle. Second, whenever either collapse or densification was predicted, the f(ΔI) trip reset function became more restrictive because of the greater slopes of the wings. This was particularly the case for the positive axial offset points, which were affected most by the power spikes. In time, all Westinghouse operating plants were analyzed to reset the f(ΔI) functions based on the amount of densification predicted, and Plant Technical Specifications were appropriately modified. All operating 14 x 14 and 15 x 15 fuel assembly design Westinghouse plants with overpower ΔT protection systems presently have their trip reset functions established by such methods.

In 1974, in response to revised acceptance criteria[2] for loss-of-coolant accident emergency core cooling analyses, Westinghouse introduced the constant axial-offset control method of operation in its plants. This method permitted the simultaneous satisfaction of meeting the low core peaking factors demanded by the revised criteria, and of maintaining good operational flexibiiity in Westinghouse plants. This method of operation was incorporated into all

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1. McFarlane, A. F., "Topical Report – Power Peaking Factors," WCAP-7912, Marcn, 1972.
2. "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," 10CRF50.46, January 4, 1974, as published in Federal Register 39FR1001.

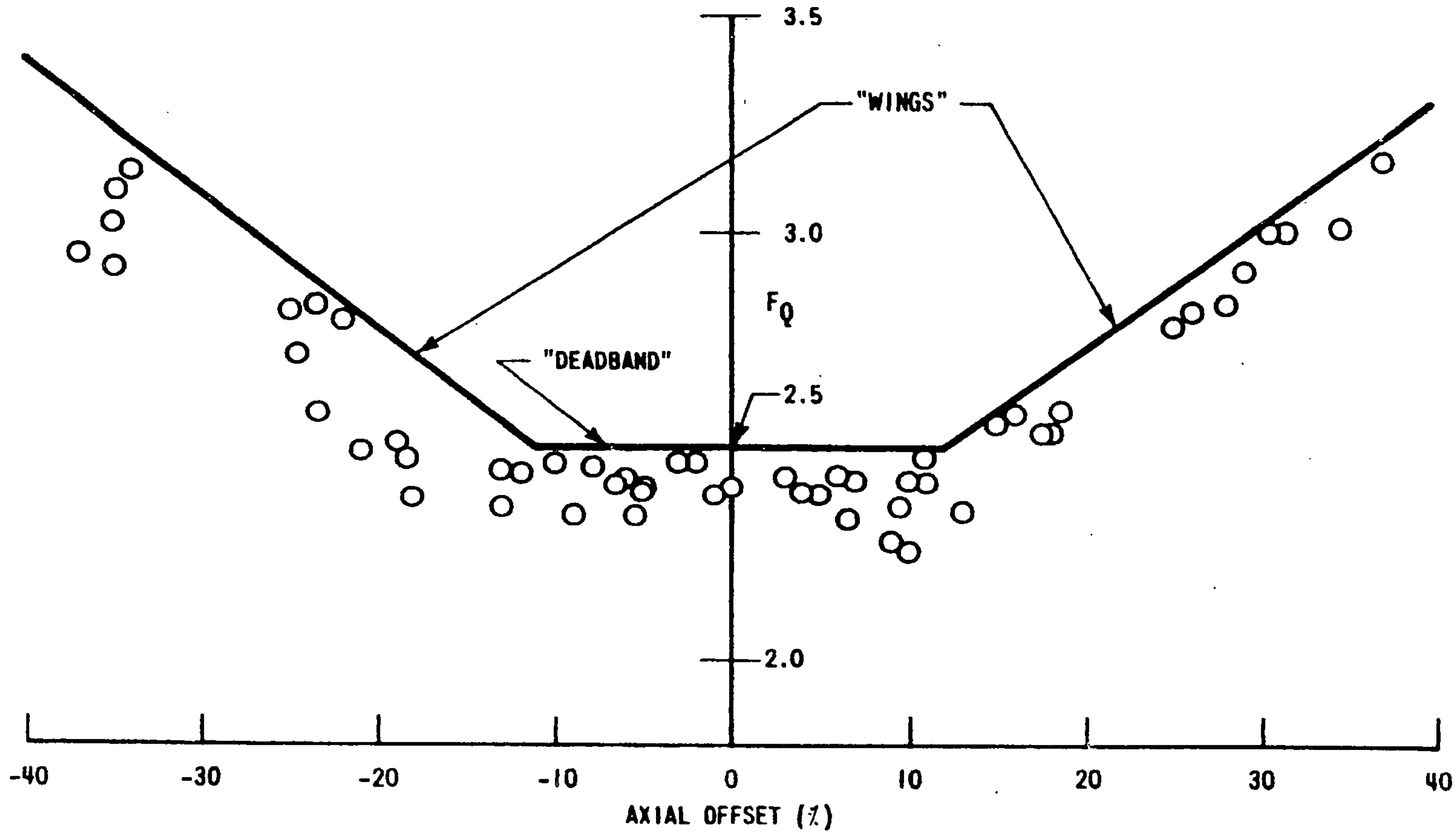


Figure 3-1. Example of $F_Q$ Flyspeck Showing "Wings" and "Deadband"

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Westinghouse plants subjected to the Final Acceptance Criteria; appropriate analytical methods were established for evaluating peaking factors. This methodology has been discussed thoroughly by Morita.[1]

One of the major features of the concept of constant axial offset reactor control is that a clear distinction can be drawn between normal operation and abnormal operation. Prior to the application of constant axial offset control, any reactor operation was assumed to be normal operation as long as the control rods remained above their specified insertion limit. Therefore, some relatively adverse power distributions could be created under postulated xenon transients. Such power distributions compromised the operating margins to some extent, because all power distributions which could be created during normal operation had to be considered as initial conditions for LOCA analysis. Under constant axial offset control, the axial power distribution during normal operation remains in a clearly defined band (characterized by axial flux difference) during all operations, including load follow, thus diminishing greatly any xenon transient effects. Operating instructions and alarms (provided for all plants) enable the application of constant axial offset control in a straight-forward manner.

Deviations from these operating restrictions could result from various abnormal operations, such as control bank malfunctions, erroneous boration/dilution, and/or operator errors, which must be analyzed in order to assure that no fuel damage could occur. It should be noted that reactor overpower conditions created by abnormal operations should be protected from fuel failures such as centerline melting. However, the LOCA requirement need not be met during these events because two independent failures or events are assumed not to occur simultaneously for reactor core design.

Morita[1] defines two categories of abnormal operation for study as potentially limiting during overpower events:

- Control bank malfunctions
- Boration/dilution system malfunctions (or operator errors)

He also gives details of the accident simulation. In short, a static one-dimensional core model similar to the one described in paragraphs 5-7 through 5-9 of this report was used to calculate the axial power profile, $P_z(z)$, during such abnormal operating conditions. It was assumed in the simulation that the core initially operates within the limits of the technical specifications; i.e., that $\Delta I$ is within the appropriate bounds as defined by constant axial offset control, and that the control rod banks are above their respective insertion limits. Control rod bank

1. Morita, T., et al, "Topical Report, Power Distribution Control and Load Following Procedures," WCAP-8403, September, 1974.

malfunctions are simulated by withdrawing a control bank while maintaining a constant boron concentration; and the simulation is terminated when core power reaches the overpower limit (118 percent of nominal). Boration/dilution transients are simulated by inserting or withdrawing control banks while maintaining a constant core power level. The nuclear peaking factor, $F_Q$, is then constructed via standard synthesis procedures, as follows:

$$F_Q = \text{Max}\ [P_z(Z)\ \times F_{XY}(Z) \times S(Z)] \times F_U^N \times F_Q^E$$

where:

| | | |
|---|---|---|
| $P_z(Z)$ | = | core average axial power distribution from the 1D calculation. |
| $F_{XY}(Z)$ | = | ratio of the peak power density to the plane-averaged power density at elevation Z. |
| $S(Z)$ | = | the densification power spike factor. |
| $F_U^N$ | = | 1.05 conservatism. |
| $F_Q^E$ | = | 1.03 engineering heat-flux hot-channel factor. |

The maximum power density during such abnormal events is then defined as: Max. Power Density (kw/ft) = Core Average Power Density (kw/ft) x $F_Q$ x Power Level.

The results of these calculations are then plotted in "flyspeck" format as shown for a typical plant in figures 3-2 and 3-3 in the control bank malfunction and boration/dilution malfunction cases, respectively. The results so analyzed are then compared to the fuel centerline melting criteria to determine the sufficiency of the overpower protection system and the need for an $f(\Delta I)$ protection function for overpower events.

The conclusions of such analyses to date have indicated the following:

- A thermal overpower limit of 118 percent of rated reactor power provides adequate protection against fuel melting.
- No $f(\Delta I)$ function is required to preclude fuel centerline melting during overpower incidents in 16 x 16 and 17 x 17 fuel assembly plants (all results are considerably less in these plants than the fuel centerline melting limit of ~ 23 kw/ft, regardless of the value of $\Delta I$).

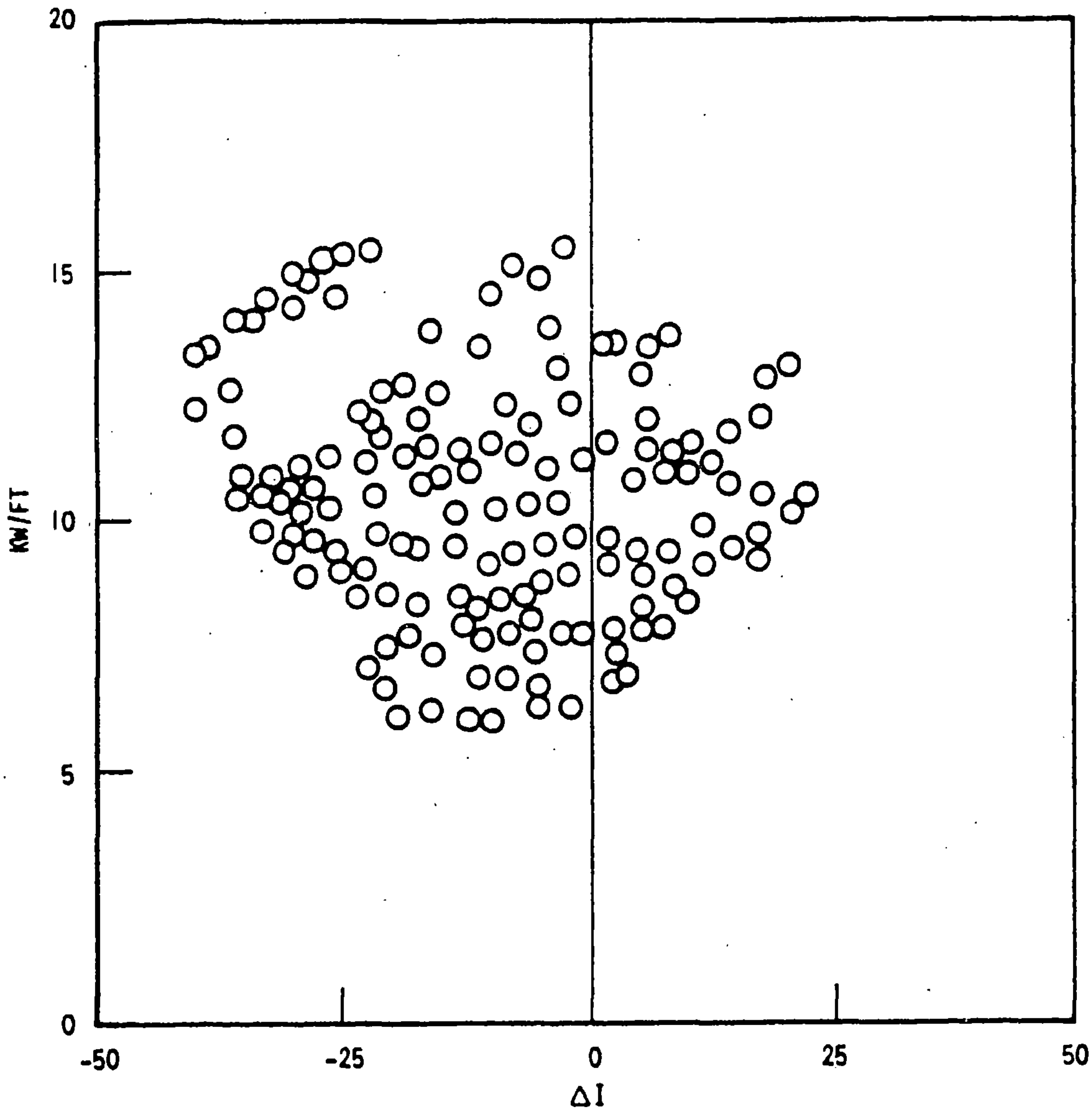


Figure 3-2. Flyspeck of Typical Control-Bank Malfunction

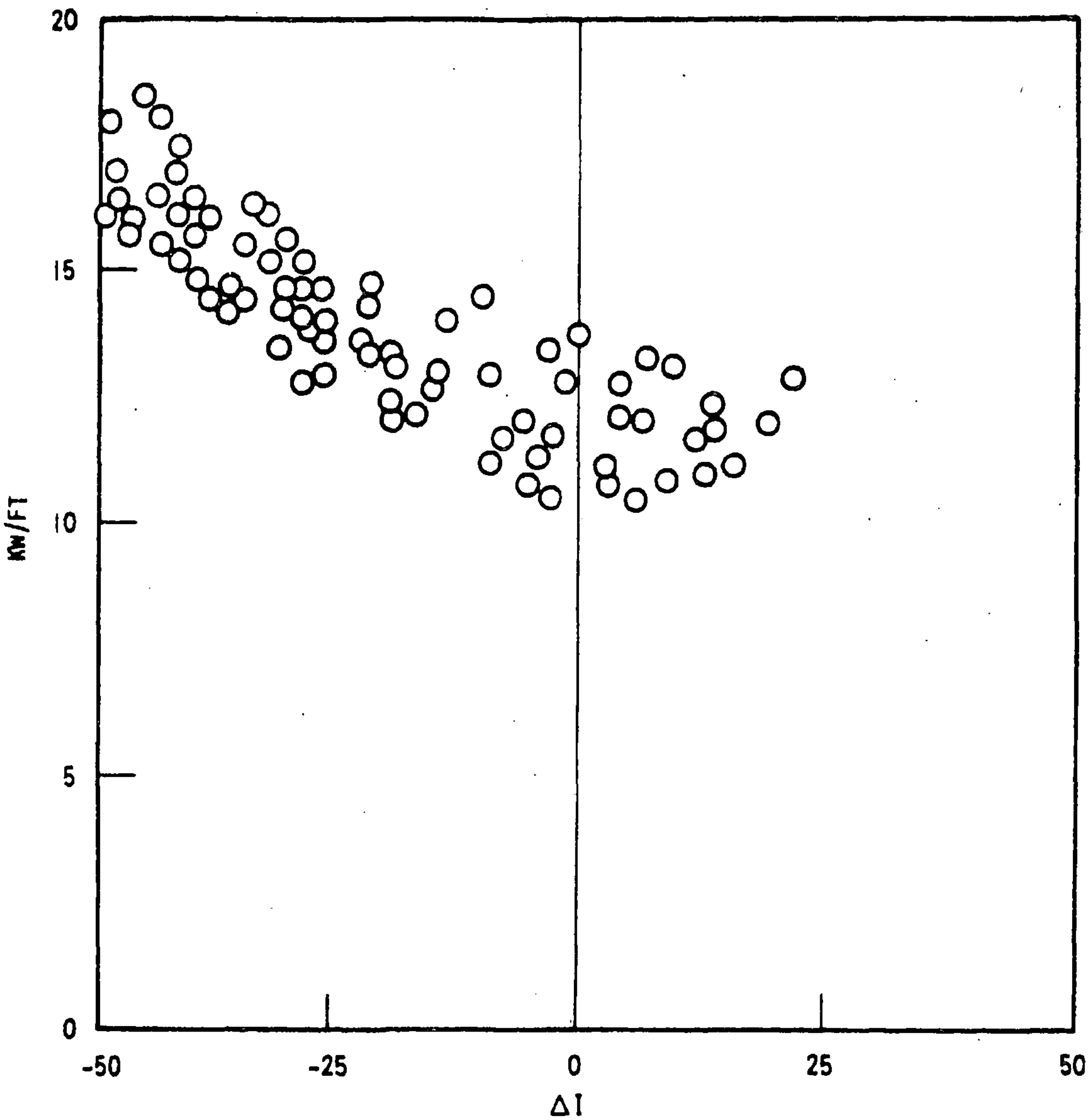


Figure 3-3. Typical Boration/Dilution Malfunction (or Operator-Error) Flyspeck

- The overpower $\Delta T$ trip $f(\Delta I)$ setpoints for plants which were operating prior to the initiation of constant axial offset control (14 x 14 and 15 x 15 fuel assembly plants) were established in a manner more conservative, as discussed above, than would be required to prevent fuel centerline melting (~ 21 kw/ft). Hence they have been left unaltered.

# SECTION 4

# CALCULATIONAL BASIS FOR THERMAL OVERTEMPERATURE TRIP SETPOINT

For each Westinghouse plant, a series of design calculations is performed to determine setpoints for the thermal overtemperature trip such that the design bases discussed in paragraphs 2-1 through 2-4 are met. As discussed in paragraph 2-5 in the design of the thermal overtemperature trip, power distribution effects are separated from the effects of core-wide parameters such as thermal power level and vessel average temperature. This separation is accomplished by first defining a reference core power distribution with which the effects of the core-wide parameters can be evaluated, and then accounting for power distributions different from the reference distribution through the $f(\Delta I)$ portion of the trip.

Paragraphs 4-1 through 4-4 describe the analyses which are performed to define regions and limits of safe operation in the space of thermal power level, coolant temperature, and coolant pressure, assuming the reference core power distribution. The procedure used to convert this information into protection system trip setpoints is detailed in appendix B.

Paragraphs 4-5 through 4-8 describe the methods used to address core power distribution effects in the thermal overtemperature trip. A relationship between power distributions more severe than the reference power distribution and the axial power imbalance is derived based on a set of "standard" nonsymmetric axial power distributions. The procedure by which this information is converted into a $\Delta T$ trip reset function, $f(\Delta I)$, is discussed in appendix C.

## 4-1. CORE POWER, COOLANT TEMPERATURE, AND COOLANT PRESSURE EFFECTS

To ensure that the DNB design basis is met, core DNB limits are determined for a range of reactor operating conditions. The core DNB limits represent the locus of points of core thermal power, primary system pressure, and coolant inlet temperature which satisfy certain criteria. This section covers the criteria, the assumptions made, the calculational method, and sample results.

### 4-2. Criteria and Constraints

The DNB design basis is that the probability that departure from nucleate boiling will not occur on the limiting fuel rod(s) is at least 95 percent at a 95-percent confidence level during normal operation and operational transients and any transient conditions arising from faults of moderate frequency (Condition I and II events). Historically this has been conservatively met by limiting the minimum departure from nucleate boiling ratio (DNBR) to 1.30.[1] Thus, the criterion on DNBR for the core DNB limits is that the minimum DNBR in the hot channel be not less than 1.30. Both the hot typical cell (bounded by four fuel rods) and the hot thimble coldwall cell (bounded by three fuel rods and a control rod guide thimble) must be analyzed, as either type of cell may be limiting.

The quality at the outlet of the heated length of the hot channel is presently limited to + 15 percent. This is the minimum upper range of applicability of the R grid DNB correlation[2] for the range of pressures (typically 1800-2400 psia) over which the core DNB limits are required. Limiting of the outlet quality in turn limits the local quality at the point of minimum DNBR to within the range of applicability of the DNB correlation for any location of minimum DNBR. As with the DNBR criterion, the exit quality criterion must be met by both the hot typical and hot thimble cells. Usually, this criterion is not limiting when compared to the DNBR limit and other constraints.

Certain constraints limit the range over which the core DNB limits must apply. In particular, the thermal overpower reactor trip places an upper constraint on the power range, the high and low pressurizer pressure reactor trips limit the pressure range, and the $T_{Hot\ Leg} < T_{Sat}$ requirement plus the opening of the steam generator safety valves place upper constraints on the temperature range. (See appendix B for a discussion of the inherent limit on temperature which is imposed by the steam generator safety valves.)

**4-3. Assumptions**

The following key assumptions are made in generating core DNB limits. These assumptions are also described by Salvatori.[1]

- The reference axial power distribution, a chopped cosine with a peak to average value, $F_Z^N$, of 1.55, is used. The effect of other axial power distributions will be discussed in paragraphs 4-5 through 4-8.
- The radial peaking factor, $F_{\Delta H}^N$, will obey the following equation at powers below 100 percent, provided the control-rod insertion limits are observed.

$$F_{\Delta H}^N (P) = F_{\Delta H\ Design}^N \ [1 + .2\ (1\text{-}P)]$$

1. Salvatori, R., "Reference Core Report 17 x 17," WCAP-8185, September, 1973.
2. Hill, K. W., F. E. Motley, F. F. Cadek, and A. H Wenzel, "Effect of 17 x 17 Fuel Assembly Geometry on DNB," WCAP-8297, March 1974.

where P is the fraction of rated core power.

At power levels above 100 percent, the value of $F^N_{\Delta H}$ is $F^N_{\Delta H\ Design}$, or, typically, 1.55.

- The coolant flow rate will be the Thermal Design Flow, in gallons per minute, which is usually about 5 percent less than the best-estimate flow.
- The maximum bypass flow is assumed; thus, only 95.5 percent of the Thermal Design Flow is available for core heat removal.
- The coolant flow into the hot assembly (the fuel assembly containing the hot rod) is reduced by 5 percent.

THINC evaluations apply the methods of Shefcheck[1] for THINC-1 analyses and Hochreiter[2] for THINC-IV analyses.

## 4-4. Calculational Method and Sample Results

A sample result of the calculation of the core DNB limits is shown in figure 4-1. This section describes how each of the line segments is generated. Limits are generated for at least four pressures spanning the range between the high and low pressure trips. The calculational method is the same for each pressure.

The locus of points where the hot leg temperature $T_{hot}$ is equal to the saturation temperature $T_{Sat}$ may be easily determined from an energy balance on the vessel:

$$h_{in} + {}^{Q}/_{\dot{m}} = h_f$$

where:

$Q$ = core power, Btu/hr

$\dot{m}$ = vessel flow rate, lbm/hr

$h_{in}$ = inlet enthalpy

$h_f$ = enthalpy of saturated liquid

1. Shefcheck, J., "Application of the THINC Program to PWR Design," WCAP-7838, January, 1972.
2. Hochreiter, L. E., and Chelemer, H., "Application of the THINC-IV Program to PWR Design, WCAP-8195, October, 1973.

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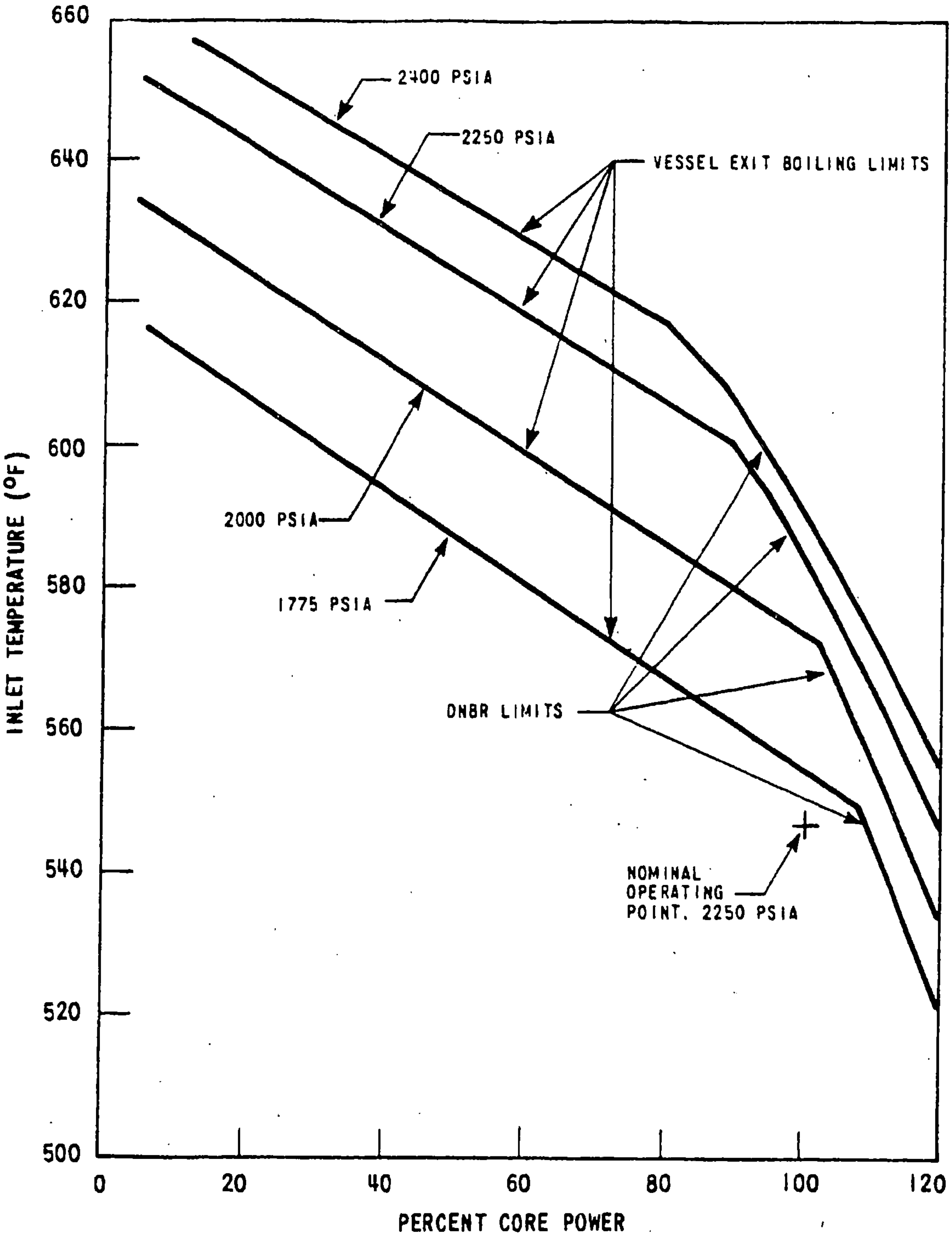


Figure 4-1. Typical Core-Thermal Limits for a 3-Loop Plant

This line is used to determine the lowest power at which DNBR limits and outlet quality limits are required. The method of calculating this line for setpoint determination is given in appendix B.

Allowable inlet temperatures, whether DNB or exit quality limited, are determined by the THINC computer code at 10 to 20 percent intervals of power from the high power limit to the intersection with the $T_{HOT} = T_{SAT}$ line. These calculations are performed for a range of pressures between the low and high pressure trips.

## 4-5. CORE POWER DISTRIBUTION EFFECTS

The core DNB limits discussed in paragraphs 4-1 through 4-4 account for core power, coolant temperature and pressure, and radial power distribution ($F_{\Delta H}^{N}$) effects, but (as noted) these limits assume a 1.55 $F_Z^N$ chopped cosine axial power distribution. The effect of the axial power distributions is accounted for by modifying the overtemperature $\Delta T$ setpoints with the $f(\Delta I)$ function. Paragraphs 4-6, 4-7, and 4-8 detail the method for determining the basis of this modification.

### 4-6. Criteria

The criteria for axial power distribution effects are the same as for the core DNB limits. That is, the minimum DNBR must not fall below 1.30, and the outlet quality of the hot channel must not exceed 15 percent. The outlet quality of the hot channel is governed by the peaking factor $F_{\Delta H}^{N}$ and the core power level, flow, and inlet temperature. [

] a,c Furthermore, as noted above, the use of the outlet quality is a bounding approach, as the quality at the point of minimum DNBR for any axial power distribution must always be less than the outlet quality. [

] a,c

### 4-7. Assumptions

All the assumptions given in paragraph 4-3 for the core DNB limits apply to the determination of axial power distribution effects, with the exception of the assumption of the 1.55 $F_Z^N$ cosine.

It is assumed that a lower-bound envelope of DNBR (or allowable power, allowable temperature, etc.) vs. Axial Offset can be developed by using a 1.55 $F_Z^N$ cosine and a set of standard axial power distributions. Axial Offset is defined as $\frac{P_T - P_B}{P_T + P_B}$ when $P_T$ is the integrated power in the top half of the core and $P_B$ is the integrated power in the bottom half of the core. It is commonly expressed as a percentage. These standard power distributions were the limiting shapes from a comprehensive set of three-dimensional nuclear calculations.

## 4-8. Calculational Method and Sample Results

a,c

The THINC computer code is used to determine (by iteration), for each power shape at each inlet temperature, the power level that gives a minimum DNBR of 1.30. Both typical and thimble cells are calculated. Sample results of this calculation are shown in figure 4-2. The method of converting this result into a ΔT trip reset function, f(ΔI), is discussed in appendix C.

a,c

Figure 4-2. Typical Axial-Offset DNB Limits for a 3-Loop Plant

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# SECTION 5
# VERIFICATION OF THE THERMAL OVERTEMPERATURE TRIP

The methods used by Westinghouse to determine thermal overtemperature trip setpoints are discussed in Section 4. This procedure employs a reference core power distribution (i.e., a power-dependent $F_{\Delta H}$ and a fixed cosine axial distribution) in determining the core thermal limits. In addition, a set of standard nonsymmetric axial power distributions is employed in determining the trip setpoint compensation necessary to account for power distributions more adverse with respect to DNB than the reference core power distribution. The standard shapes employed in that procedure are the most limiting power distributions with respect to DNB that would be expected at limiting thermal conditions during anticipated transients. They were chosen from a comprehensive study which used a static three-dimensional nuclear model. Because this study was conducted several years ago under plant operating procedures quite different from today's, a need for further assurance of the applicability of these shapes was apparent. This section presents the extensive study thus made of anticipated transients. The study employed a transient model which included a detailed core and system simulation. Paragraphs 5-1 through 5-5 cover the type of plant analyzed, the method of analysis (model used), and the accidents considered. Paragraphs 5-6 through 5-16 cover the choice of initial conditions and the assumptions made. Finally, paragraph 5-17 gives the method of interpretation of the results.

## 5-1. MODELS, EVENTS, AND PLANT CHARACTERISTICS ANALYZED

### 5-2. Coupled Core-System Evaluation Model

A coupled-core-system evaluation model was developed to better investigate the actual Nuclear Steam Supply System (NSSS) responses during selected DNB-related Condition II events. A model of this nature provides a more realistic evaluation of transient core power level, temperature, pressure, coolant flow rate, control-rod positions, and hence, core heat flux profile than models presently used for accident analyses in safety analysis reports. Since all of these parameters are required in the evaluation of the core minimum-DNB ratio, it follows that this calculational approach provides a better tool to determine those transient conditions which lead to limiting DNB ratios. Likewise, the results from this calculational model can be used to verify the adequacy of the standard methods used to derive the thermal overtemperature trip setpoints discussed in Section 4 of this report.

A number of important system responses are considered in such a calculational model. These responses include:

- The inherent primary system heatup or cooldown due to power mismatches between the primary and secondary systems of a pressurized water reactor NSSS
- The operation of the full-length rod control system and other control systems available in Westinghouse NSSS designs when the automatic function of those systems increases the severity of the transient
- The actuation of reactor protection trip functions other than the overtemperature $\Delta T$ and overpower $\Delta T$ trip functions

Explicit modeling of these system responses provides the necessary feedback mechanisms that affect the transient behavior of the reactor core.

The important reactor core conditions and transient responses pertinent to this methodology consider:

- Core burnup distributions and xenon distributions resulting from normal plant operation
- The limited inserted integral rod worth at the initial condition of the transient
- Nonconstant differential rod worth as a function of control rod position
- Realistic temperature reactivity feedback

Inclusion of these effects in the calculational model determine the transient core response to changing system parameters and changes to the core heat flux profile due to control rod movement and reactivity feedback.

An evaluation model of this nature represents a methodology more complex than some of the methods and models currently used in the determination of the core protection trip setpoints. Appendix D thoroughly describes the coupled core-system evaluation model used in this study. Briefly, this evaluation model, referred to as LOFTRAN/TWINKLE, is a combination of the LOFTRAN[1] code, a lumped-parameter single-loop system model used to study the transient response of a pressurized water reactor system, with the TWINKLE[2] code, a multi-dimensional spatial neutron kinetics code.

1. Burnett, T. W. T., McIntyre, C. J., and Buker, J. C., "LOFTRAN Code Description," WCAP-7907, June, 1972.
2. Risher, D. H., Jr., and Barry, R. F., "TWINKLE – A Multi-Dimensional Neutron Kinetics Computer Code," WCAP-8028-A, January, 1975.

## 5-3. DNB Evaluation Model

a,c

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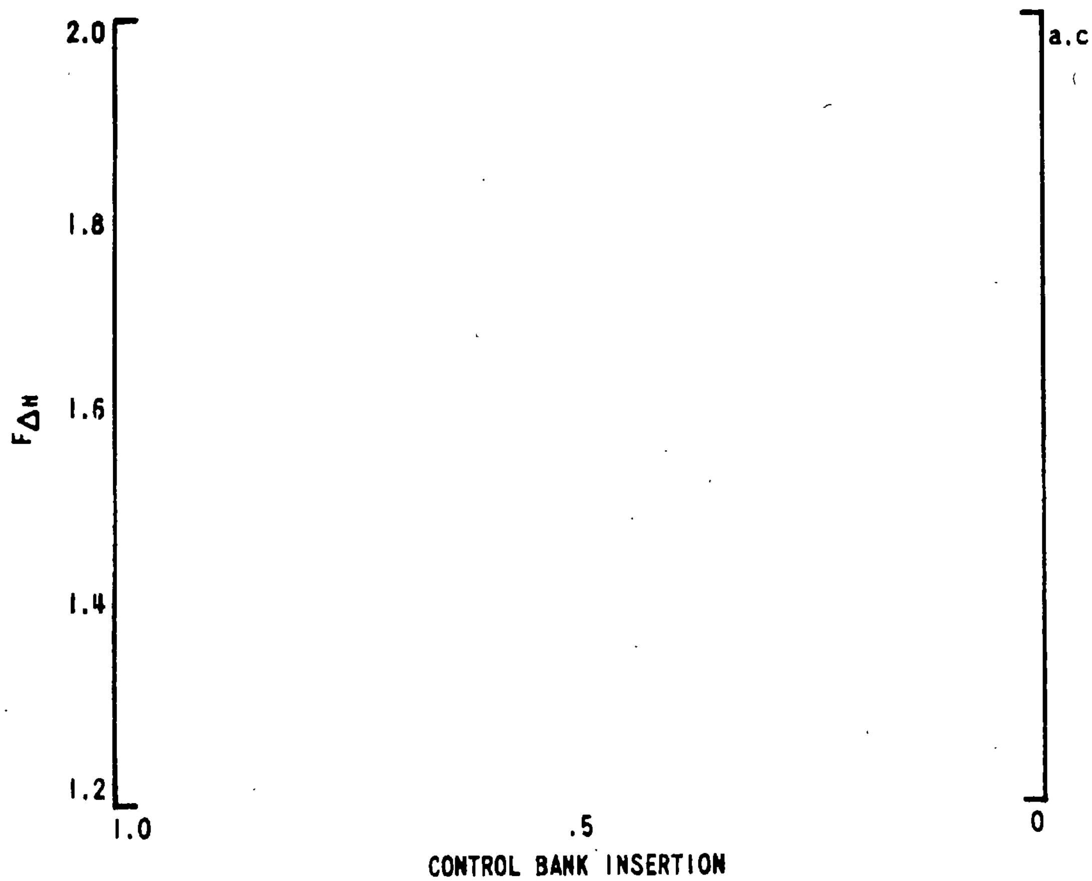


Figure 5-1. $F_{\Delta H}$ versus Rod-Position Function

### 5-4. Condition II Events Considered

The overtemperature $\Delta T$ trip function provides primary core protection during a number of DNB-related Condition II events. These events and the current methods used to evaluate the potential for DNB during these events are described in Chapter 15 of RESAR-3S.[1] This section describes the coupled core-system analysis of these selected postulated transients with the LOFTRAN/TWINKLE coupled core-system evaluation model.

In general, DNB-related transients can be classified as either primary system heatup or cool-down events. Two postulated transients are considered to be heatup events because the heat extraction from the steam generator lags behind the core power generation until the steam generator pressure reaches either the relief or safety valve setpoint:

- Uncontrolled Rod Cluster Control Assembly (RCCA) Bank Withdrawal at Power
- Uncontrolled Boron Dilution During Power Operation with Manual Rod Control

Both of these events are evaluated with the coupled core-system evaluation model.

Cooldown transients can be defined as transients which would cause a decrease in the reactor coolant inlet temperature and a corresponding increase in core power level due to the effects of the negative moderator temperature coefficient. Following are two such postulated transients:

- Excessive Steam Load Increase Event
- Excessive Heat Removal due to a Feedwater System Malfunction at Power

The feedwater system malfunction is very similar to, but less severe than, the excessive load increase event in terms of the resulting reduction in core inlet temperature. As such, only events of the excessive steam load increase type are evaluated with the coupled core-system evaluation model. There are, however, two types of excessive load increase events, each of which cause a rapid increase in the secondary-side steam flow, and both are considered in this analysis. The first event considers a 10 percent step load increase in steam flow caused by an equipment malfunction in the steam dump control system. The second event considers the inadvertent opening of the turbine throttle valve to its fully open position. This event, called a turbine control-valve malfunction, can produce an increase in the reactor coolant system load demand to approximately 105 percent power regardless of the initial power level.

A DNB-related transient, not categorized as either a heatup or cooldown event, is the uncontrolled boron dilution/boration with automatic rod control. In this event the core power level and the coolant average temperature are maintained at their initial values by the rod control system, but with changes in the axial power distribution. This event is also evaluated with the coupled core-system evaluation model.

1. RESAR-3S, "Reference Safety Analysis Report," Consolidated Version and Subsequent Amendments, July, 1975, Docket No. STN 50-545.

## 5-5. Plant Characteristics

The following nominal plant characteristics (typical of a three-loop Westinghouse Plant with a 12-foot core) were used in the accident analyses.

Core:

| | |
|---|---|
| Core Power | 2652 MWt |
| Fuel Array | 17 x 17 |
| Number of Assemblies | 157 |

Reactor Coolant System:

| | |
|---|---|
| Total Volume Including Pressurizer and Surge Line | 9600 $ft^3$ |
| Nominal Pressure | 2250 psia |
| Thermal Design Flow | 265500 gpm |
| Vessel Average Temperature | 576.2°F |
| Core Inlet Temperature | 542.5 °F |
| No-load Temperature | 547°F |

Pressurizer:

| | |
|---|---|
| Total Volume Including Surge Line | 1443 $ft^3$ |
| Heater Capacity | 1400 kw |
| Maximum Spray Rate | 63.2 lb/sec |
| Power-operated Relief Valve Steam Flow Capacity at 2350 psia | 2-210000 lb/hr (each) |
| Safety-Valve Steam Flow Capacity at 2500 psia | 3-345000 lb/hr (each) |
| Power-operated Relief Valve Opening Pressure | 2350 psia |
| Safety Valve, Start-Open - Full-Open Pressure | 2500-2575 psia |

Secondary System:

| | |
|---|---|
| Steam Generator Type | 51 |
| Steam Generator Design Pressure | 1100 psia |
| Nominal Steam Pressure | 790 psia |
| No-load Steam Pressure | 1020 psia |
| Nominal Steam Temperature | 516.8 °F |
| Nominal Steam Flow per Steam Generator | 1075 lb/sec |
| Nominal Fluid Mass per Steam Generator | 101,600 lbs |

## 5-6. INITIAL CONDITIONS AND ASSUMPTIONS

### 5-7. Initial Core Conditions

A comprehensive search was conducted to determine limiting core initial conditions for the analysis of Condition II events with the coupled core-system evaluation model. This search, described below, consisted of two steps:

- Determination of pre-accident core conditions
- Accident analysis with the use of a static core model for a wide range of initial reactor conditions

To adequately assess the appropriateness of the standard shapes, a very large number of core initial conditons should be studied so as to be reasonably sure that a complete set of core initial conditions have been considered. Because such an analysis is too expensive with the LOFTRAN/TWINKLE model, the static nuclear model (second of the above two steps) was used to reveal the worst pre-accident core conditions (i.e., power level, control-rod-position, xenon distribution). Detailed analysis of these fewer conditions by means of the LOFTRAN/TWINKLE model was then feasible.

5-8. **Determination of Pre-Accident Core Conditions** — The choice of pre-conditions for the accident analysis was performed with a modified version of the PANDA[1] code. This portion of the process is based on the premise that the axial xenon distribution in the core changes relatively slowly with respect to other variables during a Condition II transient. Hence, an approach was adopted which defines a process for selecting the most adverse xenon distributions which could occur. These worst xenon distributions would be those which give rise to the limiting axial power distributions during Condition II transients. It should be recalled that the progression of the axial power shape during the transient (not the initial power shape) is the important item of these events. Hence, the static nuclear analysis focuses on the core-related item which can best be defined during these transients, i.e., the axial distribution of xenon.

The method of core pre-condition selection for the accident analysis was accomplished via the static nuclear code in the following two basic steps.

[ ] a,c

1. Altomare, S., and Minton, G., "The PANDA Code," WCAP-7757-A, February, 1975.

[

] a,c

**5-9. Static Core Model Accident Simulation** – The following four accident simulations were carried out in the static core model, a modified version of PANDA, to provide guidance in the selection of limiting core initial conditions (i.e., xenon distributions, core power levels, and rod insertions) for evaluation with the coupled core-system transient model. All pre-accident core conditions determined as discussed above were evaluated in these simulations.

First Simulation:

- Dilution Accident

  This simulates uncontrolled dilution, either by system malfunction or operator error. The control rods are assumed to be in the manual mode of operation. [ ] a,c The calculation is terminated if the reactor power reaches the thermal overpower limit (118 percent of rated power), [ ] a,c

Second Simulation:

- Cooldown Accident

  This accident assumes a reduction of the inlet temperature of the coolant, due to a secondary plant transient such as a sudden excessive load increase, excessive feedwater flow, or a turbine throttle-valve opening. The control rods are assumed to stay at their original insertion (i.e., manual mode). [ ] a,c The calculation is terminated if the reactor power reaches the thermal overpower limit, [ ] a,c

Third Simulation:

- Full-Length (F/L) Rod Withdrawal

  This accident assumes uncontrolled F/L rod withdrawal either by system malfunction or operator error. The boron concentration is fixed. The F/L rods are withdrawn in [ ]$^{a,c}$ intervals of core height up to the fully withdrawn position. A reactor trip is assumed to occur if the reactor power reaches the thermal overpower limit. This type of calculation also simulates the excessive load increase with the control rods in the automatic mode.

Fourth Simulation:

- Boration/Dilution with F/L Rod in Automatic

  This uncontrolled boration/dilution accident assumes that the F/L rod is in the automatic mode and that reactor power is maintained at a constant level. Only full-power operation is considered. The F/L rod bank is moved in [ ]$^{a,c}$ intervals of core height. Cirticality is maintained by adjusting the boron concentration.

[ ]$^{a,c}$

The resulting values of DNBR were then plotted versus ΔI for each of the accidents. Figure 5-2 is an example of such a plot. Based on the information contained in such plots, the most adverse sets of core initial conditions were selected by determining the cases which produced the minimum DNB ratios in the transients simulated above. These initial conditions were then used as the starting point for the coupled core-system analysis.

Table 5-1 lists the number of different xenon distributions and initial power levels that were selected for evaluation with LOFTRAN/TWINKLE for each Condition II event. For each set of initial conditions, the static core model values for core burnup distribution, core xenon

1. Cooldown accident simulations are run at reduced temperatures.

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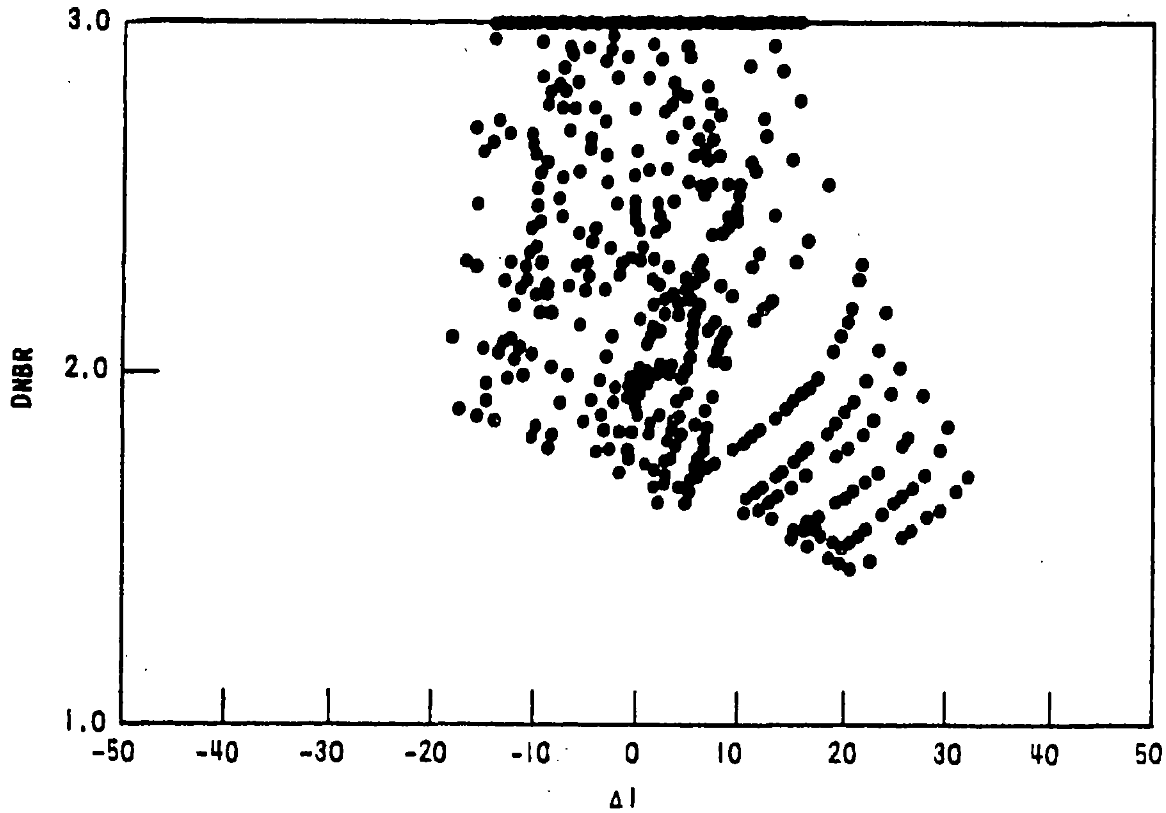


Figure 5-2. DNBR for Rod Malfunction

TABLE 5-1

INITIAL CORE CONDITIONS SELECTED FROM STATIC CORE MODEL RESULTS

| CONDITION II EVENT | CORE[a] LIFETIME | AXIAL XENON DISTRIBUTIONS | INITIAL[b] POWER LEVELS |
|---|---|---|---|
| Uncontrolled RCCA Bank Withdrawal at Power | [ ] a,c | [ ] a,c | [ ] a,c |
| Uncontrolled Boron Dilution with Manual Rod Control | | | |
| Uncontrolled Boration/Dilution with Automatic Rod Control | | | |
| Turbine Control Valve Malfunction | | | |
| Ten Percent Step Load Increase | | | |

[ ] a,c

distribution, initial power level, and initial rod position are input directly to the LOFTRAN/TWINKLE evaluation model. As noted previously, only operation without part-length rods was considered in the coupled core-system evaluation, although various control bands with widths up to [ ] $^{a,c}$ $\Delta I$ were considered.

### 5-10. Initial Thermal Conditions

As noted above, for each xenon distribution selected for further analysis, a number of Condition II events initiated from a number of thermal power levels were evaluated with the coupled core-system model. For each of the different cases, initial thermal conditions were obtained by adding the following errors to the steady-state values of thermal power, vessel average coolant temperature, and pressurizer pressure:

| | |
|---|---|
| Thermal Power | + 2 percent of nominal full-rated power |
| Vessel Average Coolant Temperature | + 6.5°F |
| Pressurizer Pressure | - 30 psi |

These conservative assumptions regarding core power level, vessel average coolant temperature, and reactor coolant system pressure account for allowable system instrumentation errors and control system deadbands. These conditions are all (conservatively) assumed to occur at the initiation of the transient, regardless of all other core-system conditions.

### 5-11. Assumptions

A number of parameter sensitivity studies was performed with the LOFTRAN/TWINKLE computer code as part of this extensive analysis of selected DNB-related condition II events. The results of these sensitivity studies were used to identify limiting transients and conservative analysis assumptions. This approach, which is applied in current safety-analysis procedures, provides important guidelines concerning which sets of core-system parameters should be considered in order to produce conservative results. At the same time this approach sets a realistic limit on number of required transient calculations so that attention can be focused on the variables of interest (i.e., core burnup, core xenon distribution and initial power level) in the extensive analysis with the coupled core-system model. To achieve limiting DNB conditions, credit was not taken for the overtemperature $\Delta T$ and overpower $\Delta T$ trip functions in this analysis. Table 5-2 lists those reactor trip functions and control systems that were assumed operable in this analysis.

Several core and system parameters definitely influence the progression of these Condition II events. Each of these parameters, considered separately in the sensitivity studies, is discussed separately (paragraphs 5-12 through 5-16).

TABLE 5-2

AVAILABLE REACTOR TRIP FUNCTIONS AND CONTROL SYSTEMS

| CONTROL SYSTEMS | REACTOR TRIP FUNCTIONS |
|---|---|
| Rod Control (if applicable) | High Neutron Flux (118 percent) |
| Feedwater Control | High and Low Pressure |
| Pressurizer Pressure Control | High Pressurizer Level |

**5-12. Effect of Bank Withdrawal Speed on Rod Bank Withdrawal Events** – The maximum rod speed considered was 72 steps/minute. This is a physical limit imposed on the control-rod drive mechanism by the rod control system. Slower rod withdrawal speeds (36 and 18 steps/minute) were considered in the study. The most conservative DNB results were obtained with [ ]a,c steps/minute. [

]a,c

**5-13. Effect of Automatic versus Manual Rod Control on Cooldown Events** – The automatic rod-control system was modeled with nominal values for deadband, lockup, minimum, proportional, and maximum rod-speed bands. In contrast, when manual control was considered, no operator intervention was assumed, so the control rods remained in their initial positions. a,c

**5-14. Effect of Moderator Feedback** – Unlike some calculational models where point kinetics is used to determine reactivity feedback, the exact numerical value of the moderator temperature reactivity coefficient in LOFTRAN/TWINKLE cannot be conveniently controlled through the use of input data. The value of this coefficient is characteristic of the fuel enrichment, fuel burnup, moderator temperature, and boron concentration that exist in the core. a,c

The changes in moderator feedback just described have two important features. First of all, these variations in best-estimate moderator feedback, although not extreme in nature, ensure conservative results. Secondly, the realistic transient responses of the coupled core-system model are not unrealistically disguised; the purpose of the model is not defeated. a,c

a,c

**5-15.** **Effect of Control-Rod Worth on Events Involving Rod Movement – [**

] a,c

**5-16.** **Effect of Doppler Feedback on All Transients – [**

] a,c

The above discussion of important parameters is summarized in table 5-3. For each of the five events extensively studied via the LOFTRAN/TWINKLE model, conservative analysis assumptions are listed. [

]a,c

## 5-17. METHOD OF INTERPRETATION OF RESULTS

The intent of this study was to determine the adequacy of the standard power shapes currently used to generate the power distribution compensating term in the thermal overtemperature trip. The method used to interpret the transient results in a straightforward comparison with the standard shapes is discussed below.

The procedures used to determine the core DNB limits are presented in paragraphs 4-1 through 4-4. These limits, which account for a wide variety of operating conditions, are calculated based on the reference hot rod power distribution (i.e., a 1.55 chopped cosine axial distribution and an $F_{\Delta H}$ which is a function of power level). The core DNB limits are then used in the determination of the overtemperature $\Delta T$ trip lines. As described in appendix B, these trip lines are generated without an $f(\Delta I)$ trip reset function. Given these procedures and the resultant core protection trip lines, it is by design that if a hot-rod power shape, comprised of a 1.55 $F_Z^N$ chopped cosine axial power distribution and a power dependent $F_{\Delta H}$, were to exist in the core during an anticipated transient, the reactor would be shut down before the minimum core DNB ratio fell below 1.30. It follows, therefore, that the adequacy of the overtemperature $\Delta T$ trip function can only be determined if expected transient hot rod power shapes are compared to the reference hot-rod power shape during DNB-related Condition II events. In effect, the adequacy of the methods used to generate the $f(\Delta I)$ trip reset function must be examined to ensure that the core is protected by this function against transient hot-rod power shapes more limiting than the referenced one.

Paragraphs 4-5 through 4-8 treat power distribution effects for overtemperature (DNB) protection and present the calculational procedures which address these effects. Through the use of the standard axial power distributions, lower-bound envelopes of allowable power versus axial offset are developed (see figure 4-2). As described in appendix C, this information is used in the determination of the overtemperature $\Delta T$ $f(\Delta I)$ trip reset function. These allowable power versus axial offset lower-bound envelopes can be inverted to determine the minimum power reduction versus axial offset required to maintain a minimum core DNB ratio equal to 1.30. This is illustrated in figure 5-3. In effect, this is the same information from which the $f(\Delta I)$ trip reset function is determined as discussed in appendix C.

## TABLE 5-3

## LIST OF LIMITING DNB-RELATED CONDITION II EVENTS AND CONSERVATIVE ANALYSIS ASSUMPTIONS

| CONDITION II EVENT | TYPE OF EVENT | ROD SPEED | MODERATOR FEEDBACK | ROD CONTROL |
|---|---|---|---|---|
| Uncontrolled RCCA Bank Withdrawal at Power | Heatup | [ ] a,c | [ ] a,c | [ ] a,c |
| Uncontrolled Boron Dilution with Manual Rod Control | Heatup | | | |
| Uncontrolled Boration/Dilution with Automatic Rod Control | N/A | | | |
| Turbine Control Valve Malfunction | Cooldown | | | |
| Ten Percent Step Load Increase | Cooldown | | | |

a,c

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Figure 5-3. ΔQ versus Axial-Offset Limit Lines

a,c

a,c

# SECTION 6
# RESULTS AND CONCLUSIONS

Section 5 presents the assumptions and initial conditions used in the analysis of DNB-related Condition II events with the coupled core-system model, and provides a method of interpretation of the results. This section presents the results of the calculations. Paragraphs 6-2 through 6-6 give a very general description of the transient conditions which produced the most-limiting DNB ratios. Paragraphs 6-7 and 6-8 illustrate a comparison of the calculated results with the standard power shapes and discuss the adequacy of the calculational bases used to determine the thermal overtemperature trip setpoints.

## 6-1. TRANSIENT RESULTS

The calculated results for all of the DNB-related Condition II events analyzed in this study are illustrated in a general fashion on figures 6-1 and 6-2. Plotted on figure 6-1 are the actual DNB ratios produced during the many transient simulations conducted for this study, versus the axial offset of the power shapes which produced these DNB ratios. It is important to note that this figure represents the results of many transients and also a wide range of core conditions (thermal power, coolant temperature, and coolant pressure) that exist during the transient calculation. Plotted on this figure are the results of transients initiated from Mode A operation with $\Delta I$ control bands of up to [ ] a,c Figure 6-1 represents a composite of all the limiting transient power shapes and the conditions at which these shapes exist that could be expected from postulated transients initiated during plant operation without part-length rods.

[ ] a,c

As illustrated on figure 6-1, certain transients can produce fairly low DNB ratios at large positive and negative axial offsets. The following paragraphs will discuss each transient analyzed in this study and identify which transients produced the lowest DNB ratio points plotted on figure 6-1.

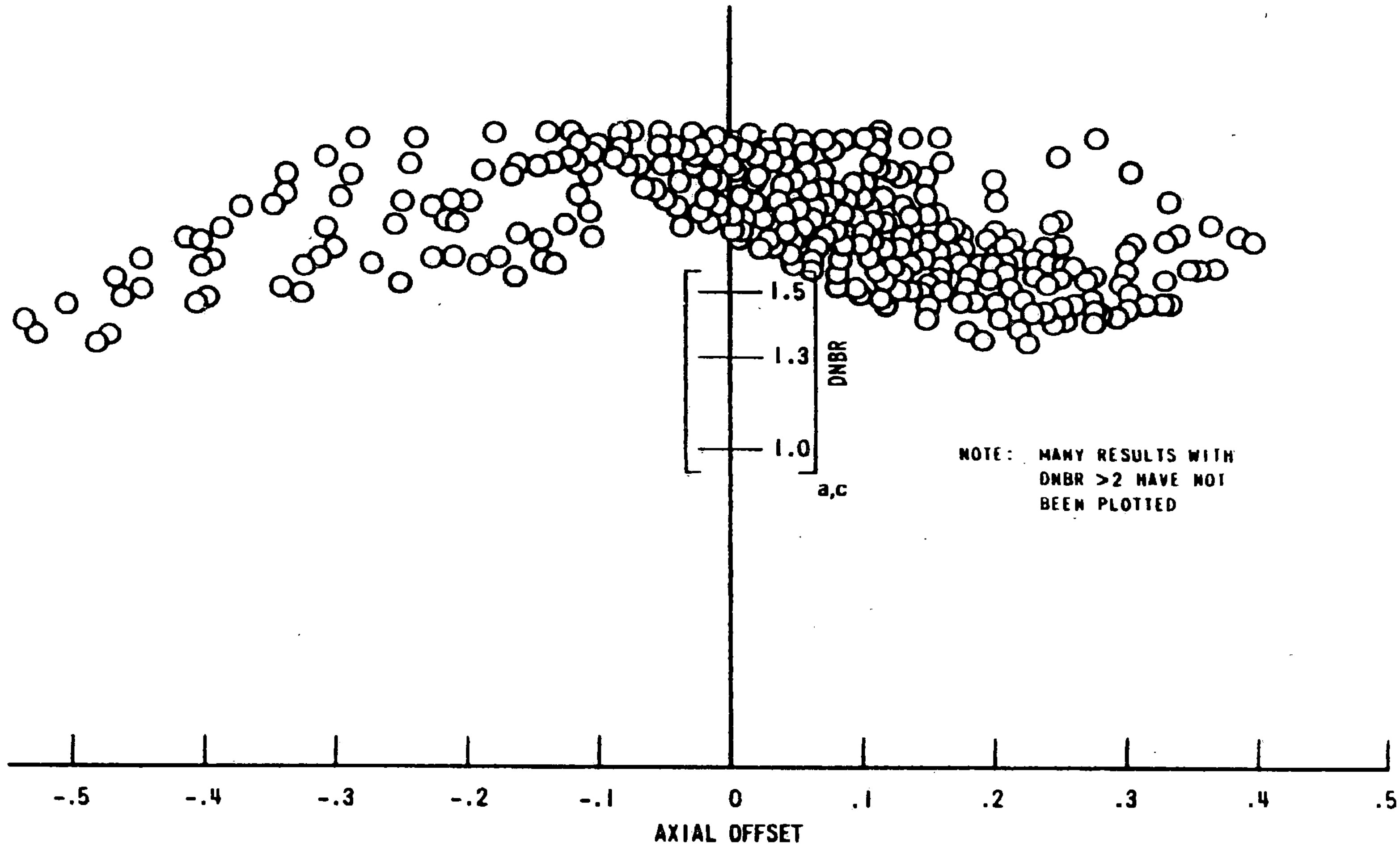


Figure 6-1. Summary of DNBR versus Axial Offset for All Transient Simulations

a,c

Figure 6-2. Summary of Minimum DNBR for All Transient Simulations

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## 6-2. Uncontrolled RCCA Bank Withdrawal At Power

This event results in a rapid increase in core power level and coolant temperature, and a positive shift (towards the top of the core) in power distribution. Thus many of the positive axial offset points on figure 6-1 are the result of rod withdrawal transients. The rod withdrawal transients which were found to yield the lowest DNB ratios were [

] a,c

## 6-3. Uncontrolled Boron Dilution With Manual Rod Control

This event results in a less-rapid increase in core power level than the rod withdrawal event due to dilution rate limitations, but has the potential for generating severe core power distributions because the control rods are stationary, thus allowing for a potential violation of the control-rod insertion limits. [

] a,c

## 6-4. Turbine Control-Valve Malfunction with Automatic Rod Control

Like the rod withdrawal event discussed above, this event results in a rapid increase in core thermal power level and a positive shift in core power distribution. However, this transient is accompanied by decreases in coolant inlet temperature. [

] a,c

6-5. Ten Percent Step Load Increase With Automatic Rod Control

The consequences of this event are nearly identical to that of a turbine control-valve malfunction from full power, except that a possibly greater increase in steam flow (ten percent rather than five) could make this event more severe. [

]a,c

6-6. Uncontrolled Boron Dilution/Boration With Automatic Rod Control

Significant changes in core power distribution occur during this event as the control rods are driven into or out of the core to maintain core power level and coolant average temperature. [

]a,c As the control rods move into or out of the core to compensate for the reduction or increase in boron concentration, severely skewed power distributions result. In addition, if a dilution is allowed to progress too long, the rod insertion limits will be violated, thus further aggravating the situation. [

]a,c

## 6-7. ADEQUACY OF THE STANDARD POWER SHAPES

Application of the method of interpretation detailed in paragraph 5-17 yields the coupled core-system transient results for Mode A operation as shown in figures 6-3 and 6-4. [

]a,c

a,c

Figure 6-3. Positive Axial-Offset Transient Results

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a,c

Figure 6-4. Negative Axial-Offset Transient Results

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a,c

## 6-8. CONCLUSIONS

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# APPENDIX A
# COMPENSATION FOR DYNAMIC EFFECTS

## A-1. $\Delta T$ and $T_{avg}$ SIGNAL DYNAMIC COMPENSATION

The dynamic terms in the thermal overpower and overtemperature trip equations compensate for inherent instrument delays and piping lags between the reactor core and the loop temperature sensors. Lead/lag and rate/lag compensations are required, in addition to noise filters, for four reasons:

- To offset RTD instrumentation time delays measured during plant startup tests.
- To offset piping lags including the RTD bypass-loop transport lag and bypass-pipe heat-capacity effects.
- To decrease the likelihood of an unnecessary reactor trip following a large load rejection.
- To ensure the protection system response is within the limits required for the accident analyses.

A simplified schematic diagram of one channel of the thermal overpower and overtemperature protection system is shown in figure A-1. Three different control functions will be described below:

- Lead/Lag function, $\dfrac{1 + \tau_1 S}{1 + \tau_2 S}$ :

  The lead/lag function provides a step of $\frac{\tau_1}{\tau_2}$ times the input step, then decays to the steady-state value. By way of examples, the inverse Laplace Transformation can be applied to this function with step and ramp inputs. The resulting output function expressed in the time domain is shown in figure A-2.

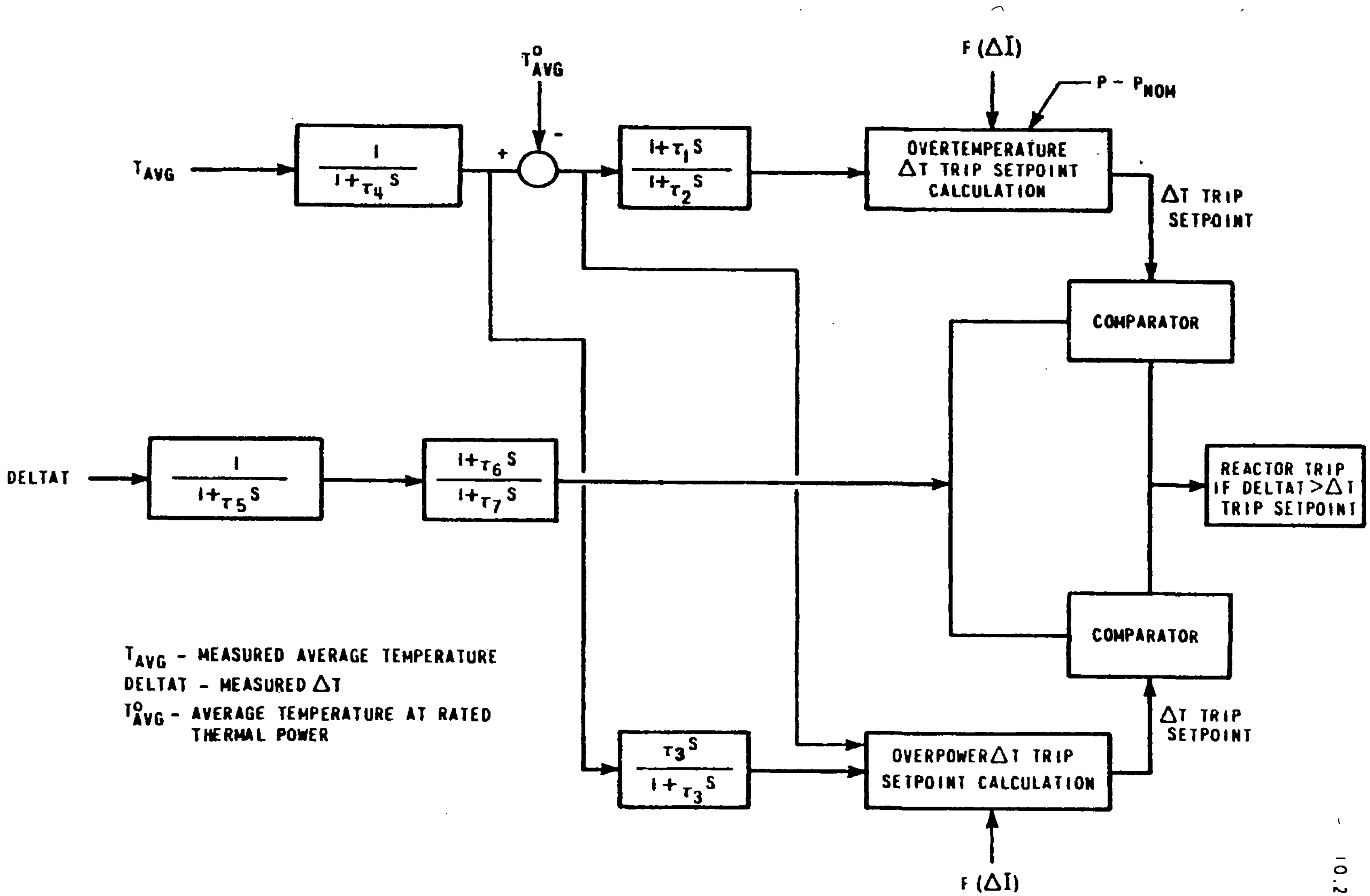


Figure A-1. Schematic Diagram of Overpower and Overtemperature $\Delta T$ Protection

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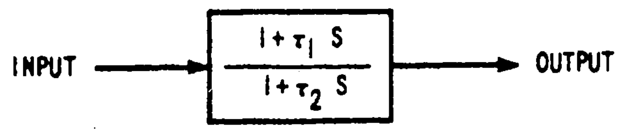
LEAD - LAG UNIT: $\frac{1+\tau_1 S}{1+\tau_2 S}$

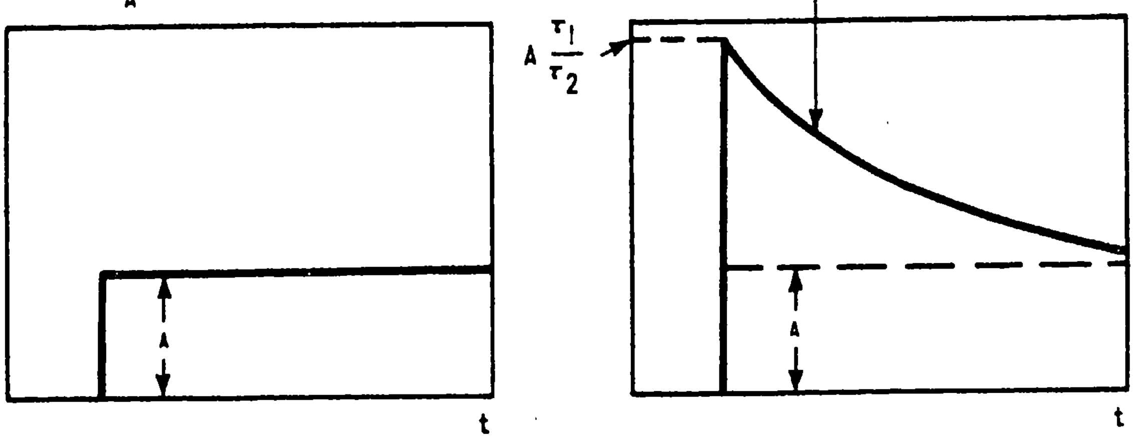
$\tau_1$: LEAD TIME CONSTANT

$\tau_2$: LAG TIME CONSTANT



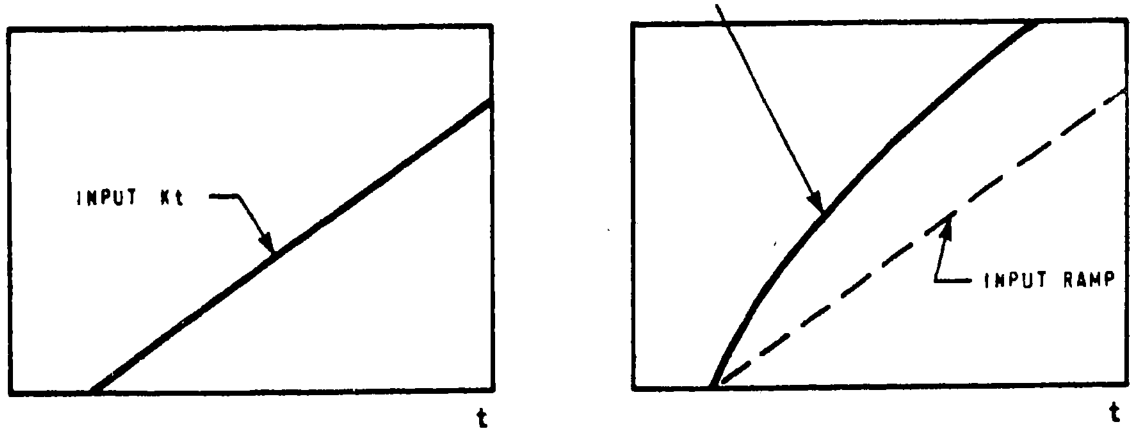

INPUT: STEP FUNCTION OF AMPLITUDE A

OUTPUT: $A\left[1-\left(1-\frac{\tau_1}{\tau_2}\right)e^{-\frac{t}{\tau_2}}\right]$




INPUT: RAMP INPUT FUNCTION OF RATE K

OUTPUT: $K\left[t+(\tau_1-\tau_2)\left(1-e^{\frac{t}{\tau_2}}\right)\right]$




NOTE: IN ALL PRACTICAL CASES CONSIDERED. $\tau_1 > \tau_2$ GIVES AN OVERSHOOTING RESPONSE TO A STEP INPUT AND A PREDOMINATELY LEAD RESPONSE TO A RAMP INPUT.

Figure A-2. Lead — Lag Unit Step and Ramp Response Characteristics

For an input, the relationship of the output signal $E_{out}$ to the input signal $E_{in}$ is:

$$E_{out} = E_{in} K \frac{1 + \tau_1 s}{1 + \tau_2 s}$$

The function (lead or lag) depends on the adjustments K, $\tau_1$ and $\tau_2$.

Several sensitivity studies have been performed by using different lead-time constants ($\tau_1$) and lag-time constants ($\tau_2$) with step and ramp inputs as shown in figures A-3 and A-4. The results show that the higher the lead-time constant, the faster the channel response; however, this must be done with caution since it will increase the noise in the signal. The $\Delta T$ signal lead/lag function is plant-dependent; the typical lead time ($\tau_6$) is 8 seconds and lag time ($\tau_7$) is 3 seconds. The addition of this $\Delta T$ lead/lag function may allow the $T_{avg}$ lead-time constant ($\tau_1$) in the thermal overtemperature trip to be reduced. The noise in the overtemperature $\Delta T$ setpoint will thus be decreased.

- Derivative function (Rate/Lag unit), $\frac{\tau s}{1 + \tau s}$ :

  The second function, the rate/lag unit, provides a step of unity times the input step, then decays to zero. As an example, the inverse Laplace Transformation can be applied to this function with step and ramp inputs. The resulting output function expressed in the time domain is shown on figure A-5. Sensitivity Studies using different $\tau$ values with step and ramp inputs are shown in figures A-6 and A-7. The results show that the higher the $\tau$ value the faster the channel response, at the cost, however, of more noise in the signal.

- Lag function (Noise filter), $\frac{1}{1 + \tau s}$ :

  The lag function provides a gradual change in output for a change in input. As an example, the inverse Laplace Transformation can be applied to this function with step and ramp inputs. The resulting output function expressed in the time domain is shown on figure A-8. Sensitivity studies in which various lag values are paired with step and ramp inputs are shown in figures A-9 and A-10. The lag-time-constant setpoints depend on the sensor (RTD) response time. For slower RDT response times, the $T_{avg}$ and $\Delta T$ filters, $\tau_4$ and $\tau_5$, may be set to zero. For an RTD with faster response, the filter may be set to 2 seconds for noise elimination.

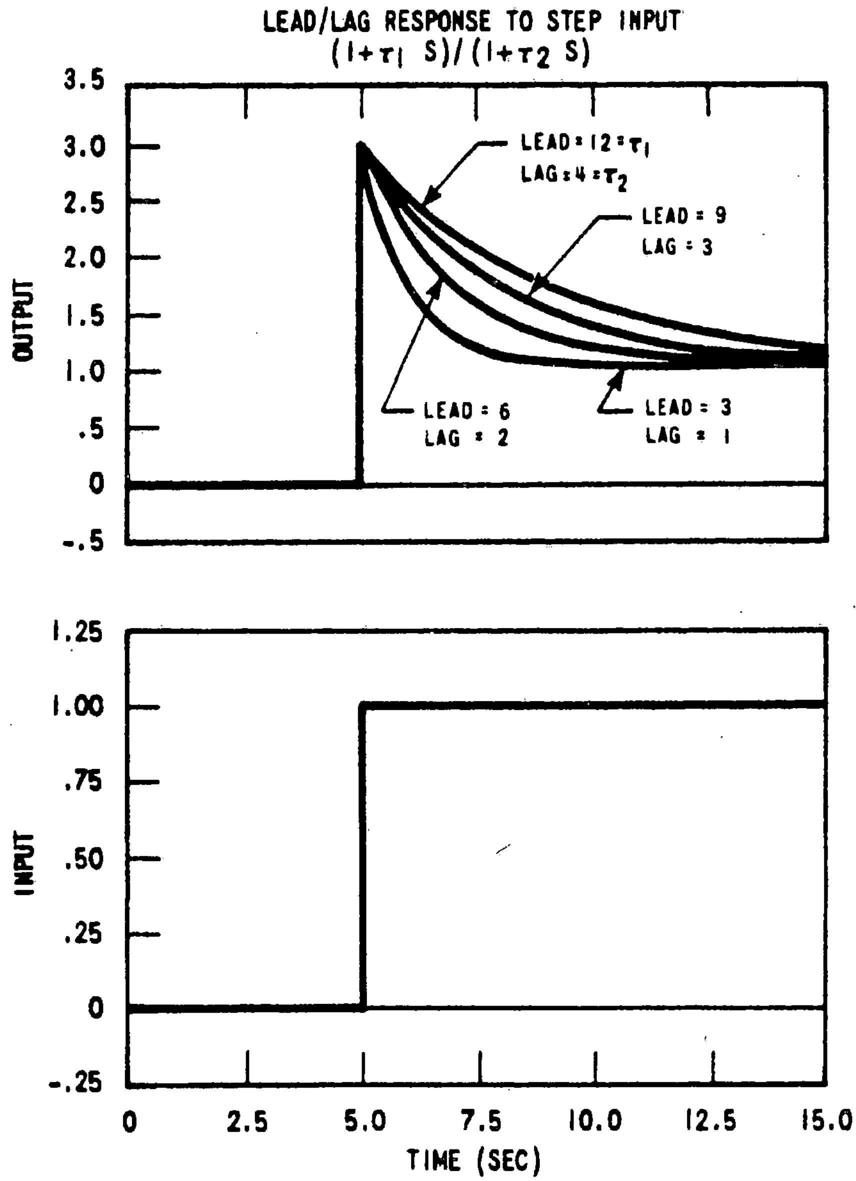



Figure A-3. Lead – Lag Unit Step Response at Various Values of $\tau_1$, and $\tau_2$

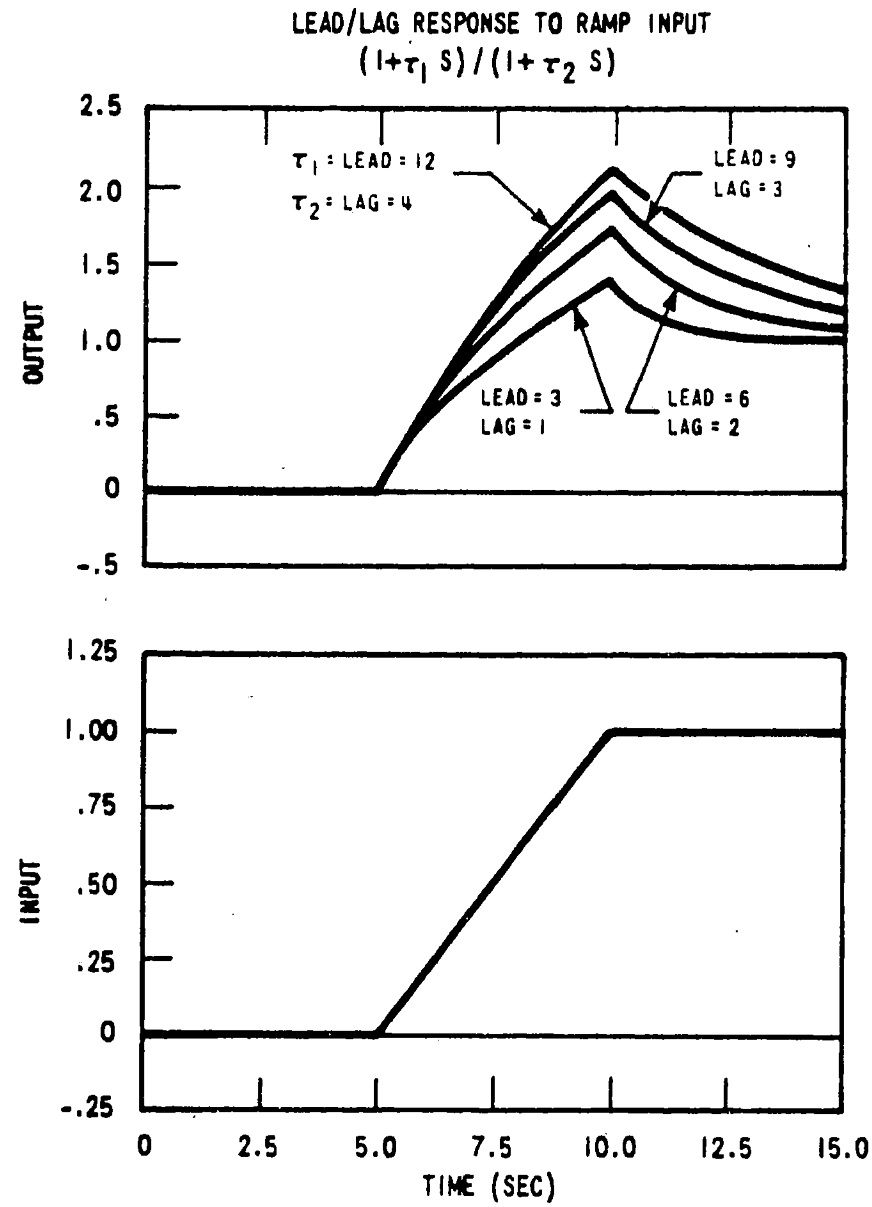
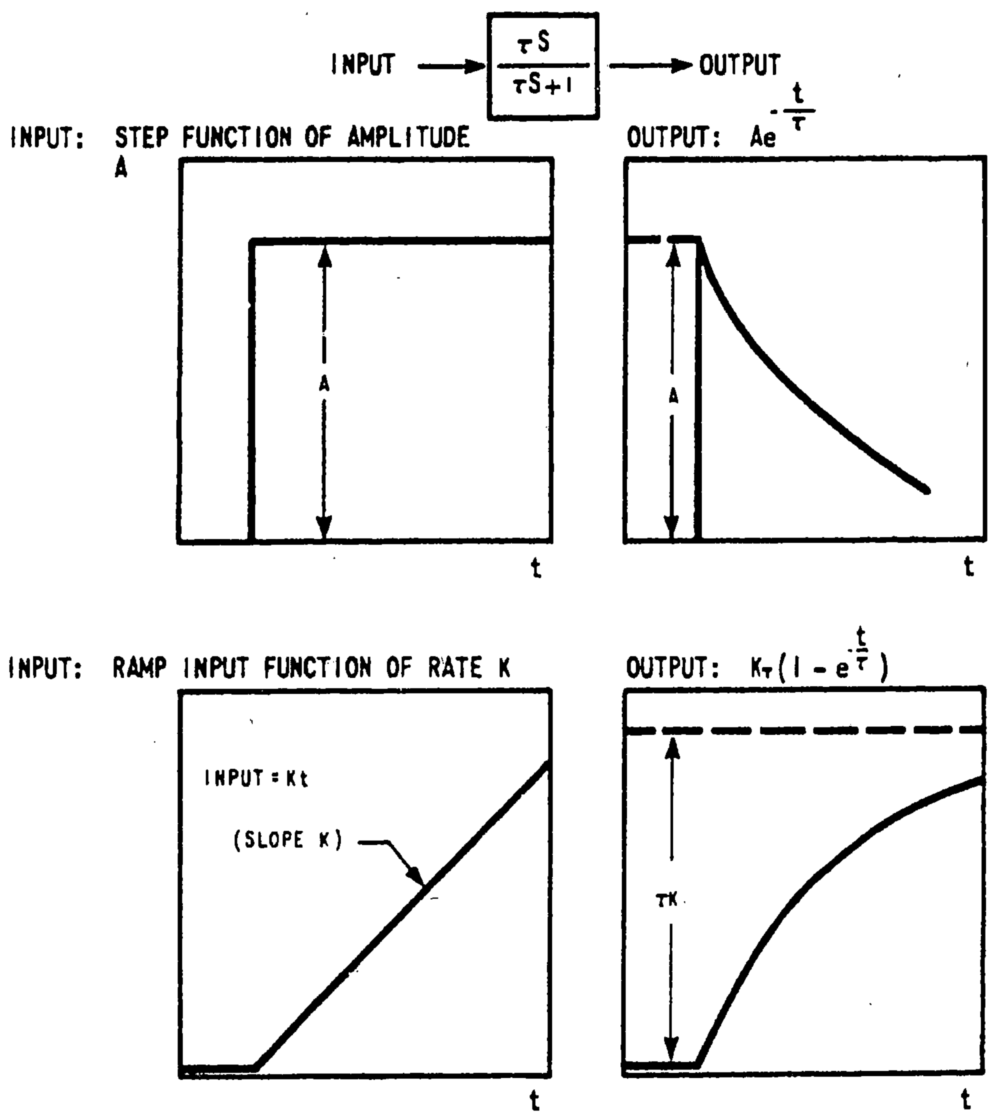



Figure A-4. Lead – Lag Unit Ramp Response at Various Values of $\tau_1$ and $\tau_2$

RATE-LAG UNIT: $\frac{\tau S}{1+\tau S}$




NOTE: THERE IS NO RESPONSE TO STATIC SIGNAL.

Figure A-5. Rate – Lag Unit Step and Ramp Response Characteristics

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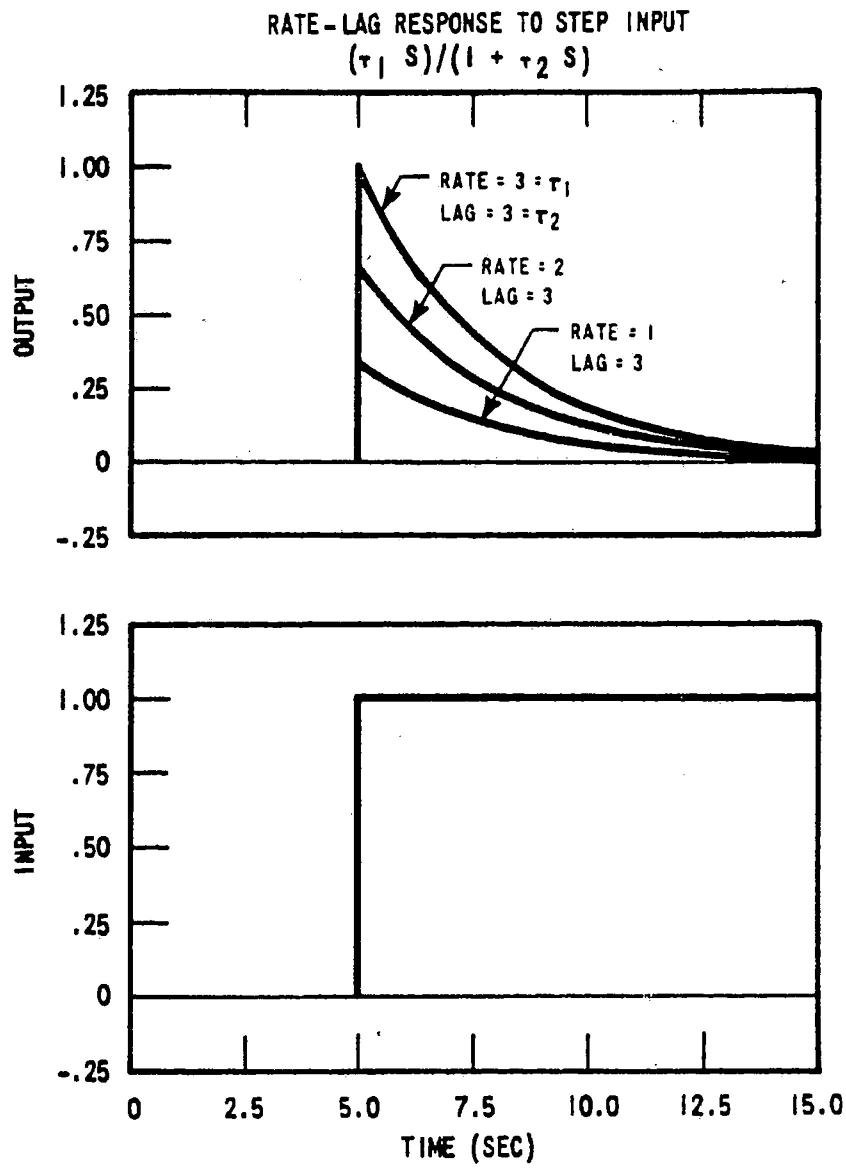


Figure A-6. Rate – Lag Unit Step Response at Various Values of $\tau_1$ and $\tau_2$

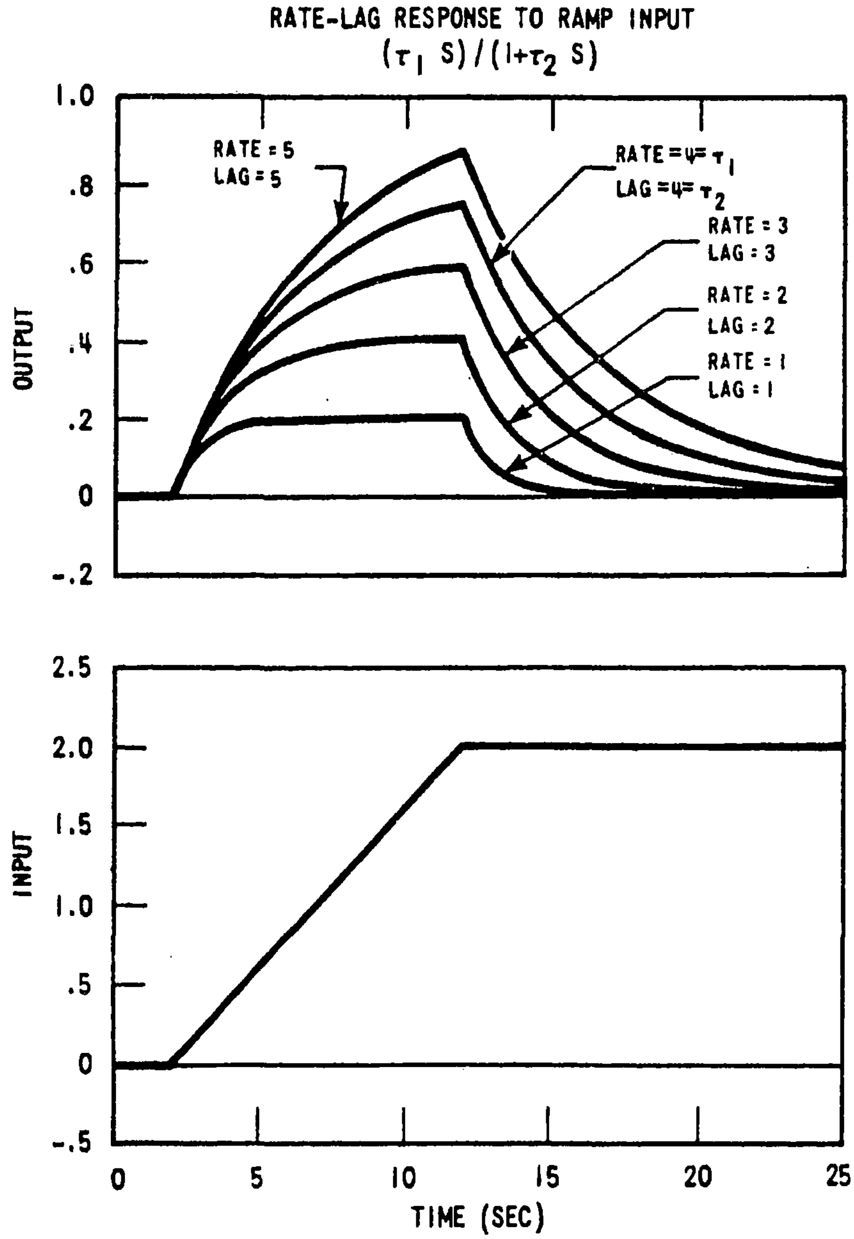


Figure A-7. Rate - Lag Unit Ramp Response at Various Values of $\tau_1$ and $\tau_2$

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LAG UNIT: $\dfrac{1}{1+\tau S}$

$\tau$: LAG TIME CONSTANT

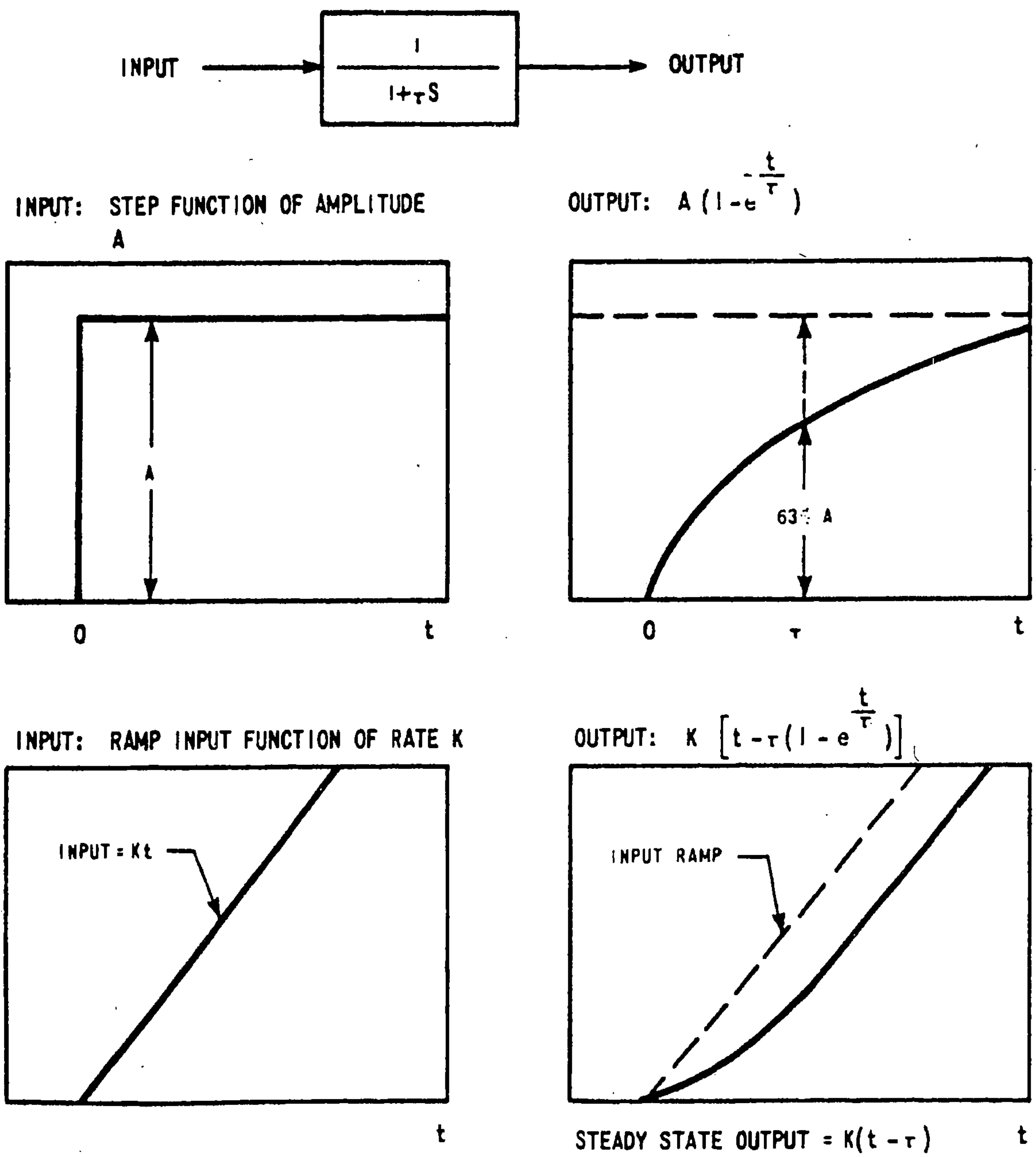


Figure A-8. Lag Unit Step and Ramp Response Characteristics

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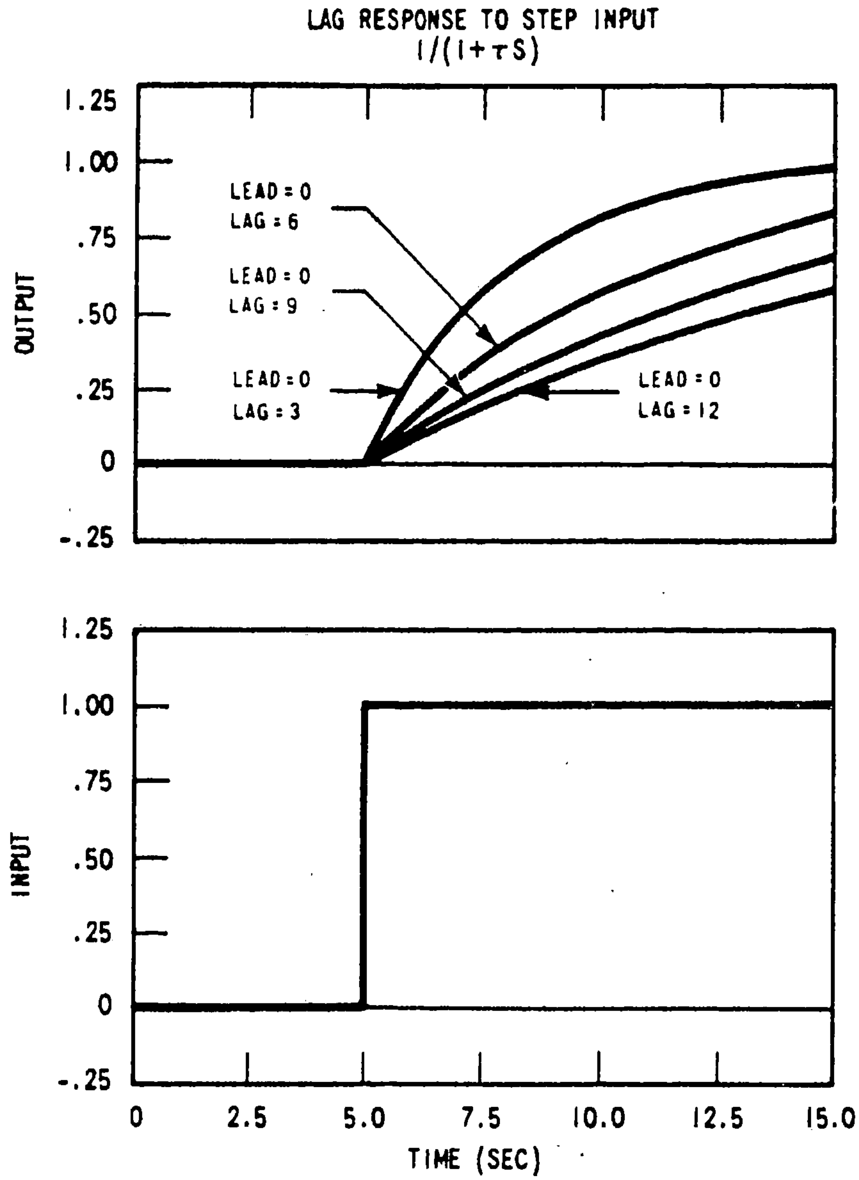


Figure A-9. Lag Unit Step Response at Various Values of $\tau$

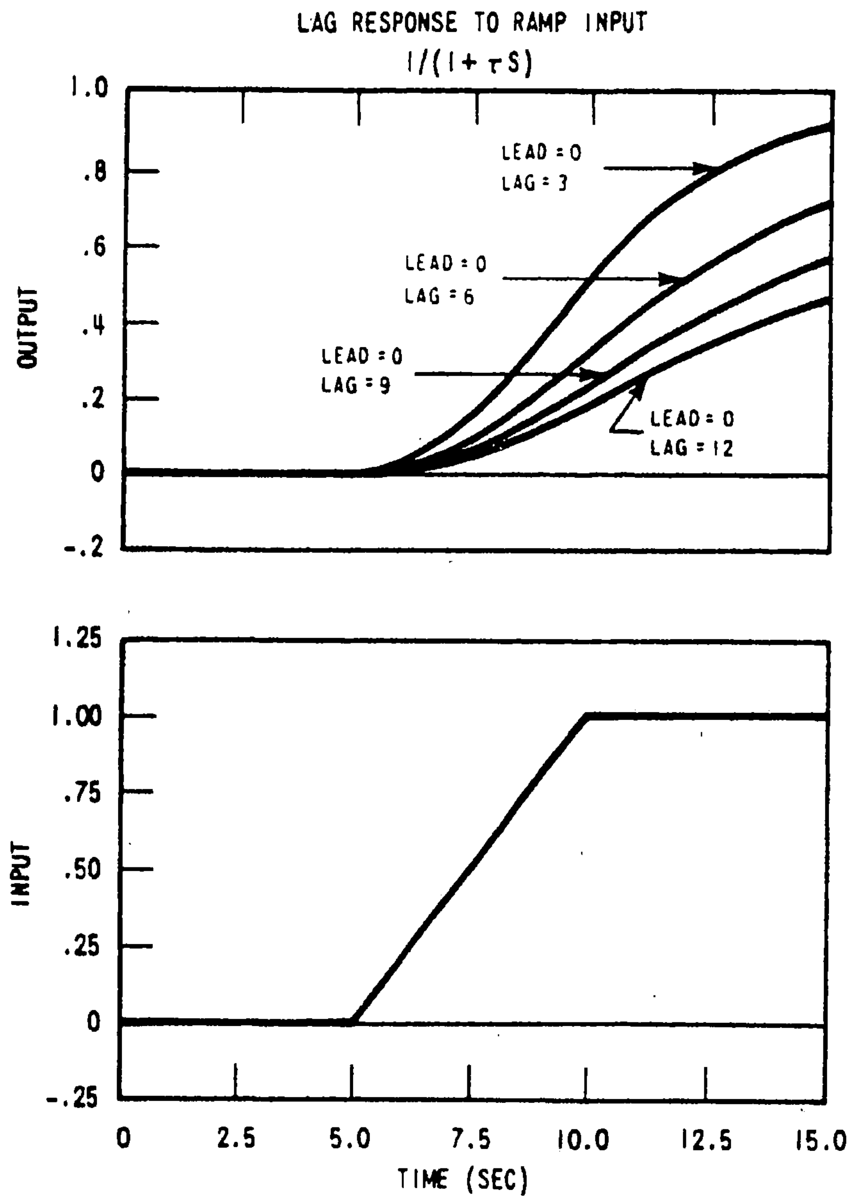


Figure A-10. Lag Unit Ramp Response at Various Values of $\tau$

## A-2. DETERMINATION OF OVERPOWER ΔT SETPOINT EQUATION DYNAMIC TERM

The overpower ΔT setpoint equation is:

1. Static equation:

$$\Delta T \text{ setpoint} = K_4 - K_6\ (T_{avg} - T^{\circ}_{avg}) - f(\Delta I)$$

2. Dynamic equation:

$$\Delta T \text{ setpoint} = K_4 - K_5 \left(\frac{\tau_3 s}{1 + \tau_3 s}\right) T_{avg} - K_6 (T_{avg} - T^{\circ}_{avg}) - f(\Delta I)$$

All of the equation's coefficients and parameters were defined in paragraph 2-6. The method to calculate the $K_4$, $K_6$, and $f(\Delta I)$ is illustrated in appendices B and C.

The dynamic term, $K_5 \left(\frac{\tau_3 s}{1 + \tau_3 s}\right) T_{avg}$, compensates for the piping lag and instrument delay. Detailed analyses and sensitivity studies based on a hybrid system simulation of the complete Nuclear Steam Supply System have shown that a $K_5$ of 0.02/°F and a $\tau_3$ of 10 seconds are the optimum selections for the Westinghouse pressurized water reactor system.

## A-3. DETERMINATION OF OVERTEMPERATURE ΔT SETPOINT EQUATION DYNAMIC TERM

The overtemperature ΔT setpoint equation is:

1. Static equation:

$$\Delta T_{setpoint} = K_1 - K_2 (T_{avg} - T^{nom}_{avg}) + K_3\ (P - P^{nom}) - f(\Delta I)$$

2. Dynamic equation:

$$\Delta T_{setpoint} = K_1 - K_2\ \frac{1 + \tau_1 s}{1 + \tau_2 s}\ (T_{avg} - T^{nom}_{avg}) + K_3\ (P - P^{nom}) - f(\Delta I)$$

All of the equation's coefficients and parameters are defined in paragraph 2-7. The method to calculate the $K_1$, $K_2$, $K_3$, and $f(\Delta I)$ is illustrated in appendices B and C.

The dynamic term, $K_2 \frac{1 + \tau_1 s}{1 + \tau_2 s} (T_{avg} - T^{nom}_{avg})$, compensates for the piping lag and instrument delay. In order to insure that the accident analysis limits are met, and to lessen the likelihood of spurious trip, $\tau_1$ and $\tau_2$ have been chosen based on a detailed digital simulation of the system. The time constants are plant-dependent, the typical lead time ($\tau_1$) is 30 seconds, and lag time ($\tau_2$) is 4 seconds. A higher lead-time constant may cause more noise in the signal and less operating margin.

# APPENDIX B

# CALCULATION OF THE OVERPOWER $\Delta T$ AND OVERTEMPERATURE $\Delta T$ SETPOINT EQUATION

The intent of the thermal overpower and overtemperature protection system is to define a region of permissible operation in terms of power, temperature, RCS pressure, and axial power shape; and to trip the reactor automatically when the limits of this region are approached.

The three boundaries defining this region of permissible operation are:

- the thermal overpower limit, which protects against excessive fuel-centerline temperature (see Section 3).
- the thermal overtemperature limits, which protect against DNB and hot leg boiling at pressures within the defined maximum and minimum pressure bounds (see Section 4).
- the locus of conditions where the steam generator safety valves open.

Ignoring the power shape effect, which is considered separately, the region can be interpreted through a two-dimensional plot of average core temperature ($T_{avg}$) and temperature difference across the vessel ($\Delta T$), as shown by the shaded region in figure B-1. It should be noted that the low-pressure trip is a bound which cannot be illustrated in terms of $\Delta T$ and $T_{avg}$, and therefore the overtemperature protection limit at the minimum pressure is the upper bound for that pressure.

The thermal overpower and overtemperature limits are bounded by the thermal overpower and overtemperature trip functions. This appendix describes how the trip functions are determined based on the core thermal limits discussed in paragraphs 4-1 through 4-4. Paragraph B-1 explains how the core thermal limits and the steam generator safety valve line are converted into the $\Delta T$ and $T_{avg}$ coordinate system. Paragraphs B-2 and B-3 describe how the overpower and overtemperature trip equations are determined. The equations define a maximum allowable $\Delta T$ for any combinations of conditions and for simplicity are referred to as the $\Delta T$ protection limit ($\Delta T_{pl}$).

Adjustments are made to the overpower $\Delta T$ and overtemperature $\Delta T$ protection limits based on appropriate error allowances to determine the final $\Delta T$ setpoints ($\Delta T_{sp}$). This is done by adjusting the $K_4$ term in the overpower trip function and the $K_1$ term for the

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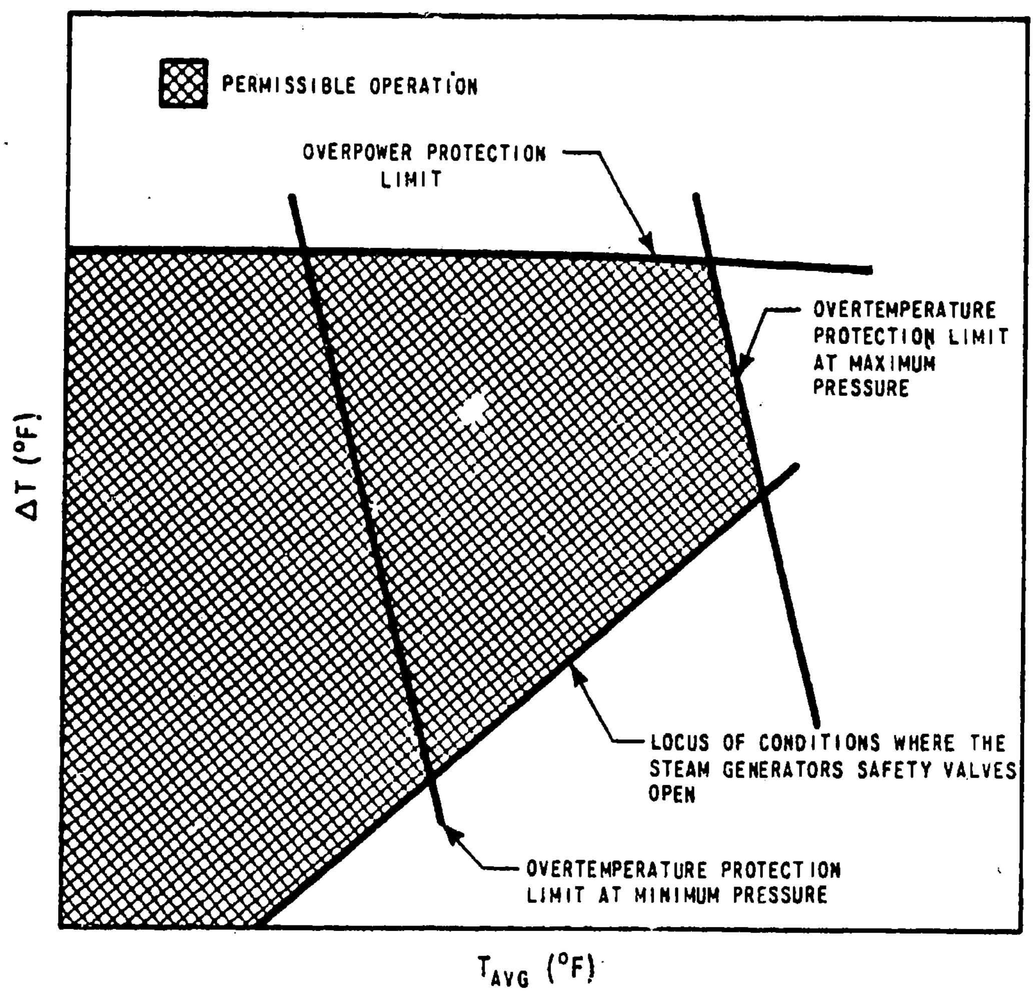


Figure B-1. Region of Permissible Operation as Defined by Core Protection System overtemperature trip function. Tables B-1 and B-2 illustrate a breakdown of errors for a typical plant.

The solid lines as illustrated in figure B-2 represent the overtemperature $\Delta T$ and overpower $\Delta T$ setpoint equations, and the locus of conditions where the steam generator safety valves open. The overpower $\Delta T$ and overtemperature $\Delta T$ protection limits, as computed in paragraphs B-2 and B-3 are represented by dashed lines.

Compensation for axial power shapes and dynamic effects is ignored in this section. Appendix C discusses axial offset penalties for overtemperature protection. The compensation for dynamic effects is discussed in appendix A.

## B-1. DETERMINATION OF THE CORE THERMAL LIMITS AND STEAM GENERATOR SAFETY VALVE LINE IN A $\Delta T$ AND $T_{avg}$ COORDINATE SYSTEM

Core thermal limits as developed in paragraphs 4-1 through 4-4 are presented in figure B-3. This figure illustrates the combination of thermal power and core inlet temperature at various pressures for which one or more thermal design limits would be met. Actual measured plant variables, $\Delta T$ and $T_{avg}$, are used in both the overpower and overtemperature trip functions. Figure B-3 is therefore converted into figure B-4, which plots the core thermal limits in units of $\Delta T$ (temperature difference across the vessel) as a function of $T_{avg}$ for several primary-system pressures.

An increase in plant temperature may result in the opening of the steam-generator safety valves. This imposes a physical limit on reactor power and temperature. The temperature drop from the steam generator primary to secondary is approximately proportional to power transferred. The maximum secondary temperature is approximately constant at the saturation temperature corresponding to the safety-valve pressure setting. Therefore, the primary temperature cannot rise above this saturation temperature plus the temperature drop across the steam generator. This temperature limit serves as one of the boundaries on power and temperature in addition to the bounds imposed by the thermal overpower and overtemperature trips, and the high and low pressure trips.

For a given steam-generator type at a known primary flow and fixed operating conditions, the inherent power and temperature limit can be calculated in terms of $T_{avg}$ and $\Delta T$. The locus of points in terms of the values of $\Delta T$ and $T_{avg}$ at which the steam generator safety valves open, called the "steam generator safety valve line", is illustrated in figure B-4.

The following procedure is used to convert the core thermal limits and the steam generator safety valve line to the $\Delta T$ and $T_{avg}$ coordinate system:

TABLE B-1

OVERPOWER ΔT SETPOINTS BREAKDOWN

The overpower ΔT protection-limit equation (including all errors) is typically:

$\Delta T_{pl}$ = 71.20 When $T_{avg}$ is less than 582.2.

$\Delta T_{pl}$ = 61.60 (1.15586 - .00111738 ($T_{avg}$ - 582.2)) When $T_{avg}$ is greater than 582.2

The maximum allowable ΔT at nominal pressure and RCS average temperature is given in percent of the full-power ΔT. To obtain the nominal setpoint, the following errors are subtracted. The errors are given in percent of full-power ΔT.

| | |
|---|---|
| Maximum allowable ΔT at nominal pressure and RCS average temperature | 115.59 |
| Error allowance for calibration and instrument channel errors | 7.22 |
| Allowing for the above errors, the nominal trip setpoint becomes | 108.37 |
| Difference between the above nominal trip setpoint and turbine runback setpoint | 3.00 |
| Runback setpoint | 105.37 |

The nominal overpower ΔT setpoint equation is:

$\Delta T_{sp}$ = 66.75 When $T_{avg}$ is less than 582.2

$\Delta T_{sp}$ = 61.60 (1.0837 - .00111738 ($T_{avg}$ - 582.2)) When $T_{avg}$ is greater than 582.2

## TABLE B-2

## OVERTEMPERATURE ΔT SETPOINTS BREAKDOWN

The following equation defines a typical overtemperature ΔT protection limit (including all errors):

$$\Delta T_{pl} = 61.6\ (1.21582 - .009476\ (T_{avg} - 582.2) + .00067119\ (P\text{-}2235^{*})$$

To obtain the nominal setpoint, the following errors are subtracted. The nominal setpoint and the errors are given in percent of full-power ΔT.

| | |
|---|---|
| Maximum allowable ΔT at nominal pressure and RCS average temperature | 121.58 |
| Error allowance for calibration and instrument channel errors | 9.57 |
| Allowing for the above errors, the nominal trip setpoint becomes | 112.01 |
| Difference between the above nominal trip setpoint and turbine runback setpoint | 3.00 |
| Runback setpoint | 109.01 |

The margin of 3 percent between the turbine runback setpoint and the trip setpoint ensures a turbine runback before the ΔT trip setpoint is reached. With less than design fouling of the steam generator tubes, and a lower-than-design full-power $T_{avg}$, the lower $T_{avg}$ would provide additional margin between the overtemperature ΔT setpoints and the operating ΔT.

The nominal overtemperature ΔT setpoint equation is:

$$\Delta T_{sp} = 61.6\ (1.12015 - .009476\ (T_{avg} - 582.2) + .00067119\ (P\text{-}2235^{*})$$

*psig

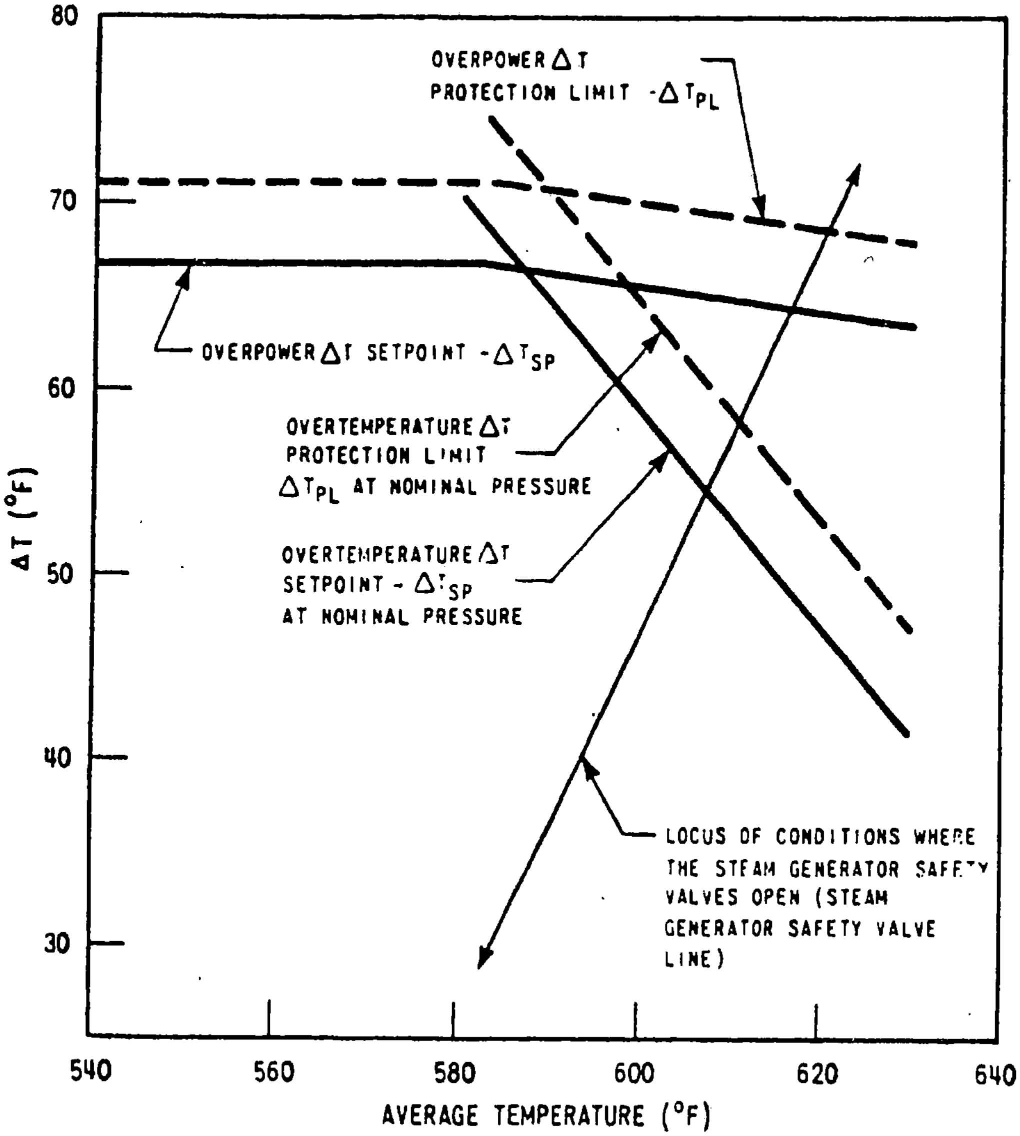


Figure B-2. ΔT Protection Limits versus ΔT Setpoints for the Overpower and Overtemperature Trips

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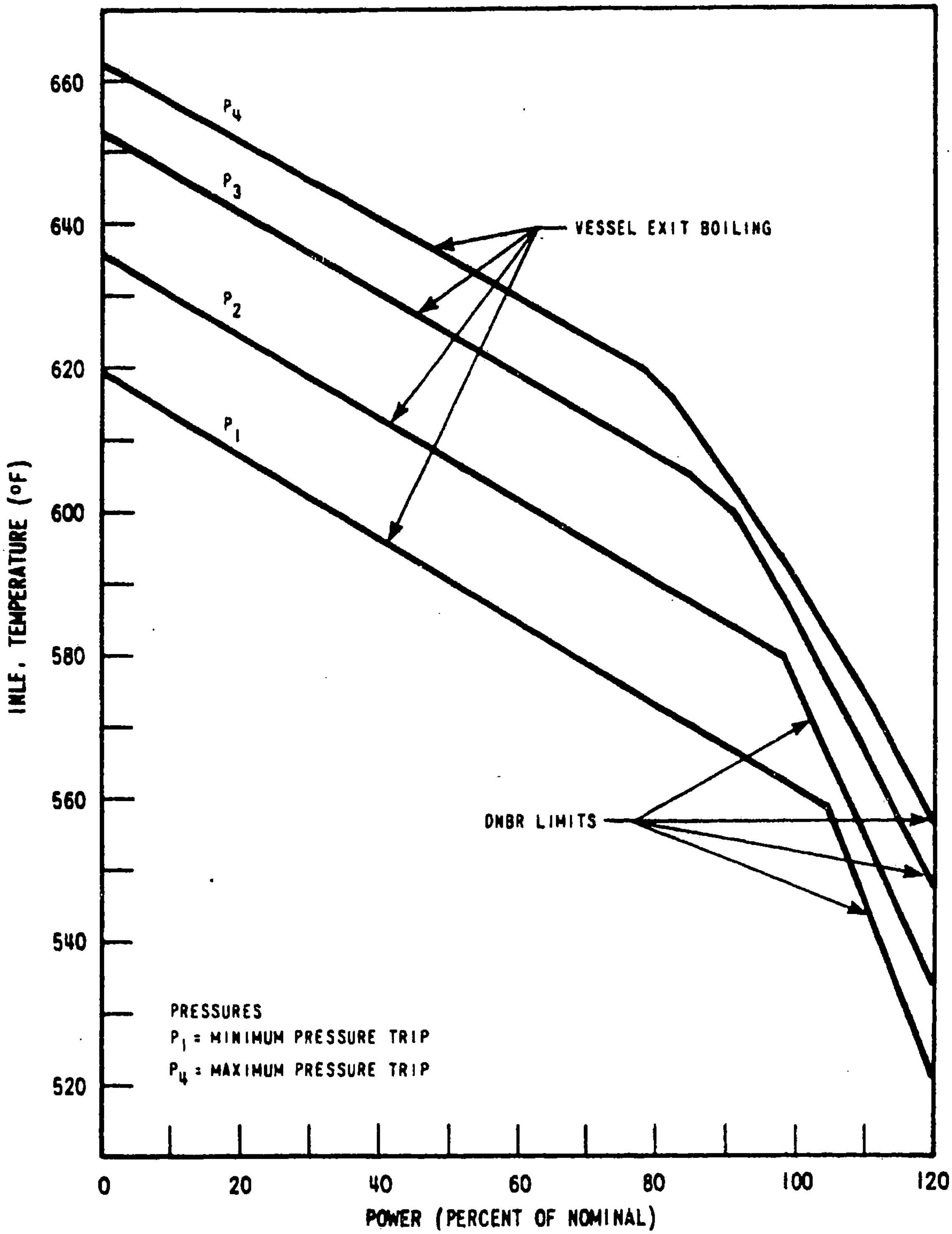


Figure B-3. Typical Thermal Core Limits: Inlet Temperature versus Power

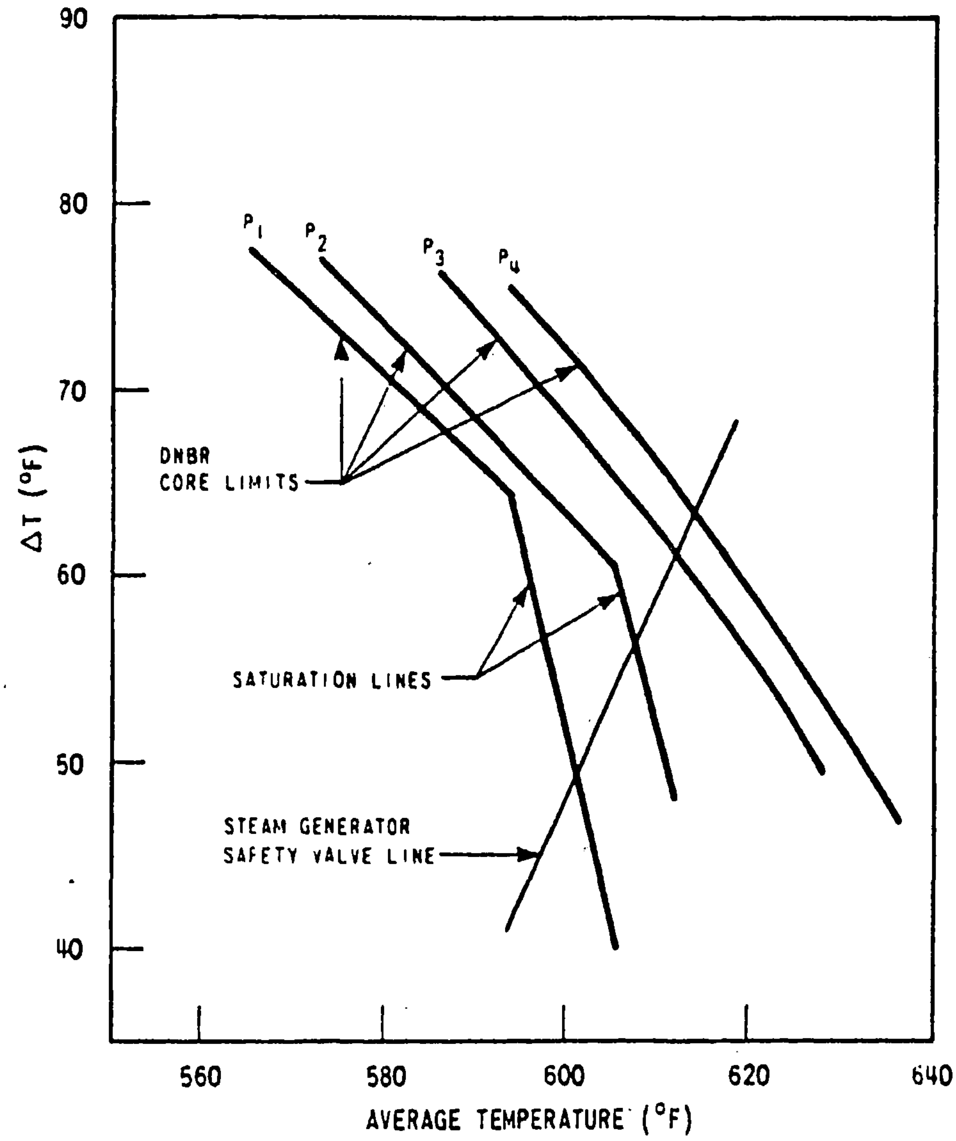


Figure B-4. Typical Thermal Core Limits in $\Delta T - T_{avg}$ Coordinate System

1. For each point on the thermal core limit lines, the following information is known:

   a. power as a fraction of the thermal design reactor power, (MWT);

   b. reactor coolant system pressure, (psia);

   c. thermal design flow (GPM); and

   d. core inlet temperature, $T_{in}$, (°F)

   From the information above, the inlet enthalpy is known and the vessel average enthalpy rise (and thus the exit enthalpy) can be calculated. From the exit enthalpy, the exit temperature ($T_{out}$) is known. $\Delta T$ and $T_{avg}$ for each point are therefore computed by the simple equations:

   - $\Delta T = T_{out} - T_{in}$; and
   - $T_{avg} = \dfrac{T_{out} + T_{in}}{2}$

2. For the hot-leg boiling condition, $\Delta T$ and $T_{avg}$ can be easily computed from the saturated temperature corresponding to a given pressure, from the relation

$$T_{sat} = \frac{\Delta T}{2} + T_{avg}$$

3. The steam generator safety-valve line can be computed from the fundamental long-mean-temperature-difference equation. The intersections of the core thermal limit lines and the steam generator safety-valve line are determined by taking points on the thermal-core-limit line for each pressure until the following equation is satisfied:

$$Q = UA \cdot \frac{T_{out} - T_{in}}{\ln\left[\dfrac{T_{out} - T_{sv}}{T_{in} - T_{sv}}\right]}$$

where:

| | | |
|---|---|---|
| Q | = | power, btu/hr |
| UA | = | overall heat transfer coefficient obtained from known performance of the steam generator at nominal $T_{avg}$ and power. |
| $T_{out}$ | = | hot-leg temperature of the reactor vessel (and steam generator inlet), °F. |

$T_{in}$ = cold leg temperature of the reactor vessel (and steam generator outlet), °F

$T_{sv}$ = saturated temperature corresponding to 103 percent of the steam generator shell design pressure (Safety-valve set pressure plus 3 percent accumulation).

A point on each core thermal limit line satisfies the condition where the steam generator safety valves open. These points then constitute a locus where the steam generator safety valves open, as illustrated in figure B-4.

The following paragraphs B-2 and B-3 explain how the overpower ΔT and overtemperature ΔT protection limits can be determined from these core limits and the steam generator safety-valve line.

## B-2. OVERPOWER ΔT SETPOINT CALCULATION

A thermal overpower limit to prevent fuel melting is normally 118 percent of nominal power, as discussed in section 3. The reactor is protected from exceeding this limit by the thermal overpower trip. Westinghouse reactors operate at a constant volumetric coolant flow rate with the result that the temperature rise through the reactor vessel (ΔT) is approximately proportional to power. There is, however, a slight dependence on pressure and reactor inlet temperature due to the changes in the density and heat capacity of the water. This is compensated for by a correction term in $T_{avg}$. The actual ΔT is continuously monitored and compared to a trip setpoint.

The overpower ΔT protection limit equation is determined based on the intersection points of the overpower limit, plotted as a function of $T_{avg}$ for fixed pressures, and the DNB core limits at corresponding pressures. The intersections are determined for various pressures ranging from the low-pressure trip to the high-pressure trip. As illustrated in figure B-5, the ΔT at the intersection point decreases as pressure increases. However, in consideration of the slight pressure dependence, a simplified first step is to remove any pressure dependence from the overpower ΔT limit equation. This could easily be effected by at least two approaches:

- An equation defining a line which passes through the overpower – DNB core limit intersections at the high- and low-pressure trips, or
- An equation defining a constant ΔT which corresponds to the ΔT computed at the overpower – DNB core limit intersection at the high-pressure trip.

Either method would prevent the overpower limit from being exceeded for all combinations of temperature and pressure. In practice, a two-equation concept was derived from the two approaches above, based on the nominal value of $T_{avg}$, to maximize operating margin. The setpoint is a constant ΔT if $T_{avg}$ is less than the nominal value; it is a diminishing ΔT if $T_{avg}$ is greater than nominal.

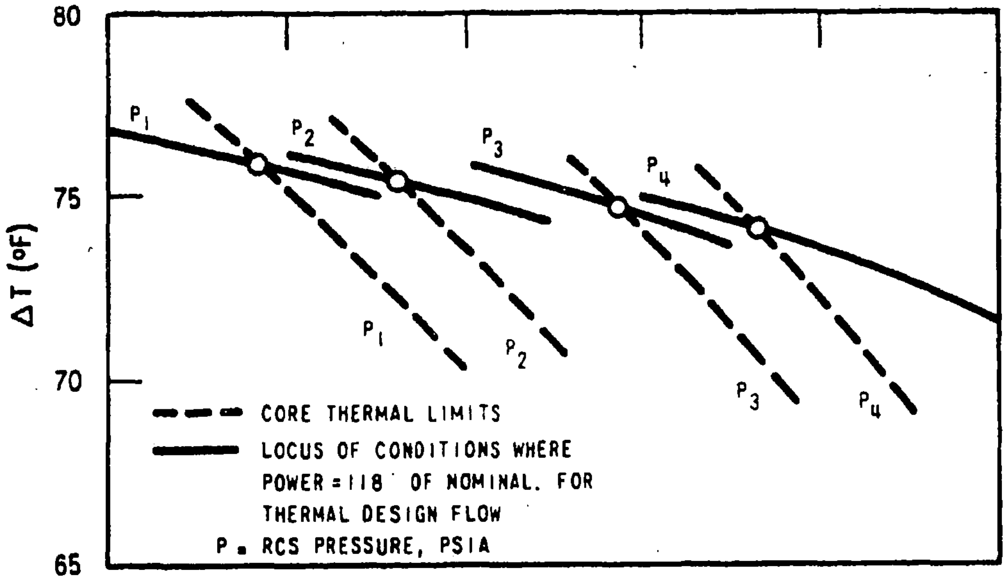


**Figure B-5.** Intersection of Thermal Core Limits and Constant Power at 118% of Nominal

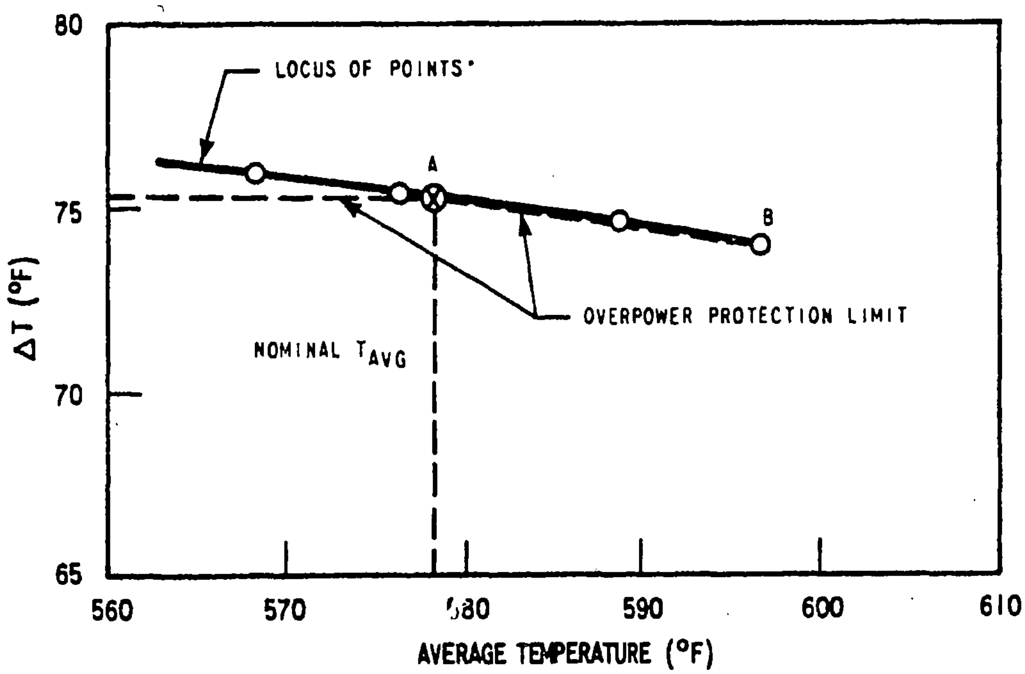


Figure B-6. Locus of Points* for a) Power at 118% of Nominal and b) DNBR = 1.3 at Full Thermal Design Flow

The overpower limit $\Delta T$ equations are thus:

$$\text{Below nominal } T_{avg}\text{: } \Delta T_{sp} = \Delta T_o K_4$$

$$\text{Above nominal } T_{avg}\text{: } \Delta T_{sp} = \Delta T_o [K_4 - K_6 (T_{avg} - T_{avg_{nom}})]$$

where:

$\Delta T_{sp}$ = Setpoint value of $\Delta T$, °F

$\Delta T_o$ = Indicated $\Delta T$ at nominal plant conditions, °F

$T_{avg}$ = Measured average temperature, °F

$T_{avg_{nom}}$ = Nominal average temperature at rated power, °F

$K_4$ = Preset manually adjusted bias

$K_6$ = A constant that compensates for the change in density, flow, and heat capacity of water with change in temperature

Following is the procedure for determining the overpower protection system setpoints:

1. The overpower limits are constructed in terms of $\Delta T$ and $T_{avg}$ for each pressure corresponding to the pressures represented by the DNB core limits (ranging from the low-pressure trip to the high-pressure trip). $\Delta T$ for 118 percent of rated thermal power (Mwt), at full thermal design reactor coolant flow, is computed as a function of $T_{avg}$ for each of the pressures, until an intersection with the DNB core limit line at the corresponding pressure is obtained. For overpower conditions beyond the intersection at each pressure (where $T_{avg}$ is greater than the $T_{avg}$ at the intersection point for a given pressure), protection is guaranteed by the thermal overtemperature trip. The loci of conditions at constant 118 percent of power are presented in figure B-5.

   Figure B-6 illustrates the locus of the intersection points which defines a locus of points at 118 percent of power and a DNBR of 1.30. $\Delta T$ is computed from this locus at nominal $T_{avg}$, as shown by point A of figure B-6.

2. The $\Delta T$ at point A ($\Delta T_A$), which becomes the constant overpower $\Delta T$ protection limit ($\Delta T_{pl}$) for $T_{avg}$ less than nominal, is written in terms of the indicated $\Delta T$ at nominal plant conditions ($\Delta T_o$):

$$\Delta T_{pl} = \frac{\Delta T_A}{\Delta T_o} \cdot \Delta T_o$$

Hence, when the measured $T_{avg}$ is less than $T_{avg_{nom}}$, the overpower $\Delta T$ protection limit equation is:

$$\Delta T_{pl} = \Delta T_o\ K_4$$

where: $K_4 = \dfrac{\Delta T_A}{\Delta T_o}$

3. For $T_{avg}$ greater than nominal, the overpower $\Delta T$ protection limit is represented by an equation which defines the line from point A to point B (refer to figure B-6) where B is the intersection point of the overpower limit with the DNB core limit at the high-pressure trip.

   The slope of this line is: $\dfrac{\Delta T_A - \Delta T_B}{T_{avg_A} - T_{avg_B}}$

The slope being always negative, a negative sign is preserved in the equation and the slope is expressed in the absolute value. In terms of the $\Delta T_o$, the equation becomes:

$$\Delta T_{pl} = \Delta T_o \left[\frac{\Delta T_A}{\Delta T_o} - \frac{\Delta T_A - \Delta T_B}{T_{avg_A} - T_{avg_B}} \frac{1}{\Delta T_o} (T_{avg_B} - T_{avg_A})\right]$$

Defining $K_6$ as follows:

$$K_6 = \frac{\Delta T_A - \Delta T_B}{T_{avg_A} - T_{avg_B}} \frac{1}{\Delta T_o}$$

The overpower $\Delta T$ protection limit equation for $T_{avg}$ greater than $T_{avg_{nom}}$ is expressed as:

$$\Delta T_{pl} = \Delta T_o\ [K_4 - K_6\ (T_{avg} - T_{avg_{nom}})]$$

Checks are made to ensure that all points on the overpower locus between points A and B are protected by this line. If not, step 3 is repeated with a new slope until protection is ensured.

These final equations represent the maximum allowable $\Delta T$ during operation. However, the final $\Delta T$ setpoint ($\Delta T_{sp}$) is determined by adjusting $K_4$ based on appropriate allowances for uncertainties and equipment and measurement errors. Final setpoint values for $\Delta T_o$ and $T_{avg_{nom}}$ in the overpower $\Delta T$ equation are determined based on actual plant startup test measurements.

## B-3. OVERTEMPERATURE $\Delta T$ SETPOINT

The thermal overtemperature trip protects the core against DNB and hot-leg boiling for any combination of power, pressure, and temperature. The overtemperature trip equation must be determined such that the core thermal limits will not be violated. Ideally the trip equation would be chosen to match the core limits at all temperatures, power levels, and pressures, while making appropriate allowances for uncertainties and equipment and measurement errors. However, to simplify protection system equipment requirements, the following equation was chosen to conservatively represent the core limits:

$$\Delta T = mT_{avg} + c \text{ Pressure} + d$$

Following is the procedure for determining the constants in this equation.

1. The core limits as discussed in paragraphs 4-1 through 4-4 are converted into a $\Delta T$ versus $T_{avg}$ coordinate system as described in paragraph B-1. The region for which overtemperature core protection must be provided is delimited by:

   a. the overpower $\Delta T$ protection line as computed in paragraph B-2

   b. the steam generator safety-valve line as computed in paragraph B-1

   c. the core thermal limit lines corresponding to the high- and low-pressure trips as computed in paragraph B-1

   Intersection points A, B, C and D as shown in figure B-7 provide the basis for calculation of the overtemperature $\Delta T$ equation. These four points define the intersections of:

   Point A: the 118 percent overpower line and the core limit line corresponding to the high-pressure trip

   Point B: the 118 percent overpower line and the core limit line corresponding to the low-pressure trip

   Point C: the steam generator safety-valve line and the core limit line corresponding to the high-pressure trip

   Point D: the steam generator safety-valve line and the core limit line corresponding to the low-pressure trip

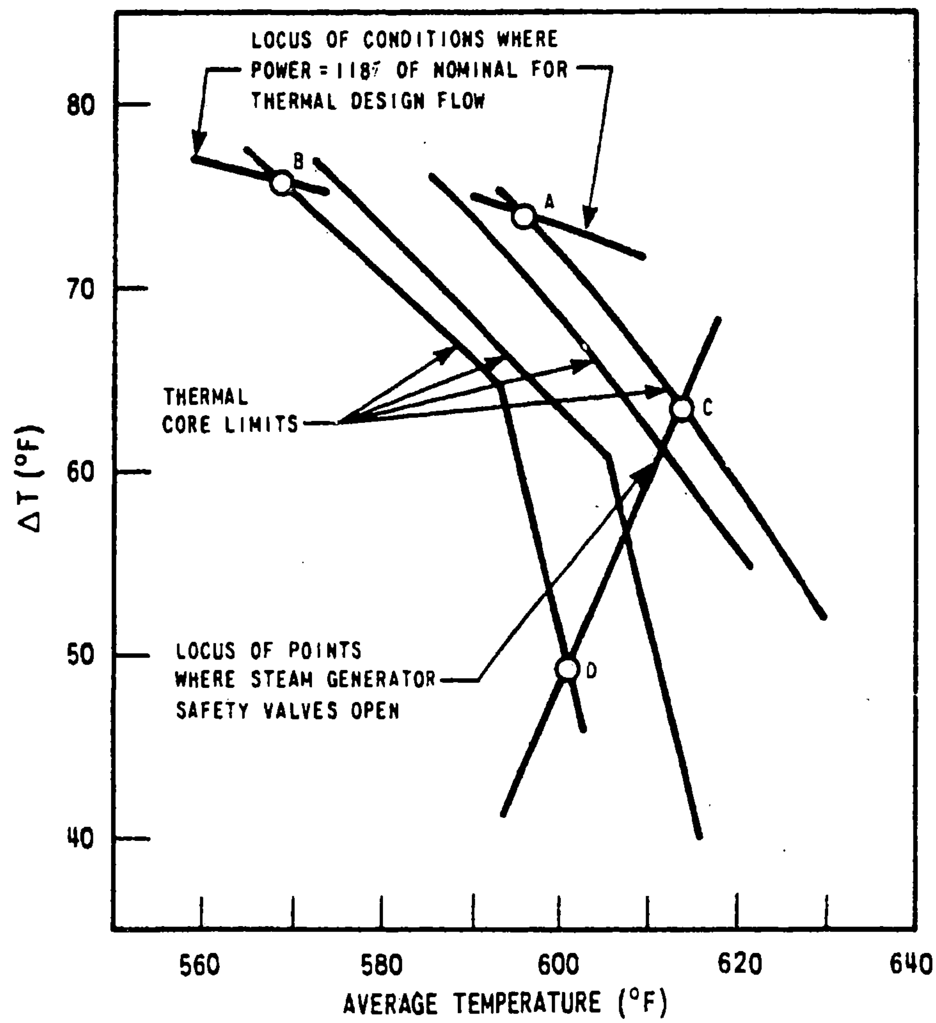


Figure B-7. Intersection Points Used to Determine Overtemperature ΔT Protection

2. The overtemperature $\Delta T$ protection limit equation, neglecting pressure, is a straight line in $\Delta T$-$T_{avg}$ coordinates. The equation is therefore:

$$\Delta T = K_1 - K_2 (T_{avg} - T_{avg_{nom}})$$

When the pressure effect is included, the overtemperature equation is a plane in three coordinates, $\Delta T$ - $T_{avg}$ - Pressure. The equation becomes:

$$\Delta T = K_1 - K_2 (T_{avg} - T_{avg_{nom}}) + K_3 (P - P_{nom})$$

The slopes and constants for four differing overtemperature $\Delta T$ protection limit equations can be determined from the four intersection points (A, B, C and D) by constructing:

a. A line parallel to AC which intercepts B;

b. A line parallel to AC which intercepts D;

c. A line parallel to BD which intercepts A; and

d. A line parallel to BD which intercepts C.

The constants, $K_1$, $K_2$, and $K_3$ for each of these equations can be determined by solving three simultaneous equations.

3. Each equation is tested for various pressures to ensure that all the core thermal limits are covered. Generally, two of the equations are found to provide protection over the entire range.

4. The final equation is selected based on maximum available operating margin. The values of $K_1$, $K_2$, and $K_3$ determined above are divided by $\Delta T_o$ and the overtemperature $\Delta T$ protection limit is:

$$\Delta T_{pl} = \Delta T_o \left[ K_1 - K_2 (T_{avg} - T_{avg_{nom}}) + K_3 (P - P_{nom}) \right]$$

where

$\Delta T_{pl}$ = The $\Delta T$ protection limit, °F

$\Delta T_o$ = Indicated $\Delta T$ at nominal plant conditions, °F

$T_{avg}$ = Measured average temperature, °F

$T_{avg_{nom}}$ = Nominal $T_{avg}$ at rated power, °F

| | | |
|---|---|---|
| $K_1$ | = | Preset manually adjustable bias |
| $K_2$ & $K_3$ | = | Preset manually adjustable gains |
| P | = | Measured RCS pressure, psig |
| $P_{nom}$ | = | Nominal RCS pressure at rated power, psig |

This equation represents the maximum allowable $\Delta T$ during operation. However, the final $\Delta T$ setpoint ($\Delta T_{sp}$) is determined by adjusting $K_1$, based on appropriate allowances for uncertainties and equipment and measurement errors. The final setpoint value for $\Delta T_o$ in the overtemperature $\Delta T$ equation is determined based on actual plant startup test measurements. The variables $T_{avg_{nom}}$ and $P_{nom}$ are fixed reference constants.

# APPENDIX C
# DETERMINATION OF F($\Delta$I) FUNCTIONS FOR OVERPOWER $\Delta$T AND OVERTEMPERATURE $\Delta$T TRIPS

## C-1. DETERMINATION OF OVERPOWER $\Delta$T f($\Delta$I) FUNCTION

The $\Delta$T trip reset function, f($\Delta$I), for the thermal overpower reactor trip is derived based on the "$F_Q$ flyspeck" as discussed in paragraph 3-2. This flyspeck defines the peak $F_Q$ which is conservatively expected to occur during Condition II events, versus the core axial offset. For plants with 17 x 17 and 16 x 16 fuel assembly cores, fuel melting may occur when the power at any point in the core exceeds a local power limit of about 23 kw/ft. For plants with 15 x 15 and 14 x 14 fuel assembly cores, the local power limit is somewhat lower. This limit is also evaluated for every cycle of every plant, because of its slight dependence upon the fuel cycle. The following procedure is used to determine the $\Delta$T trip reset function for the thermal overpower trip based on the $F_Q$ flyspeck. The procedure is illustrated in figure C-1.

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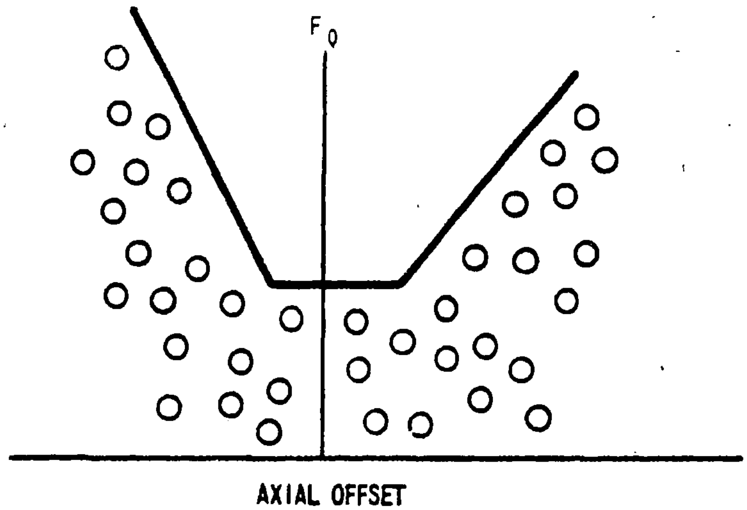
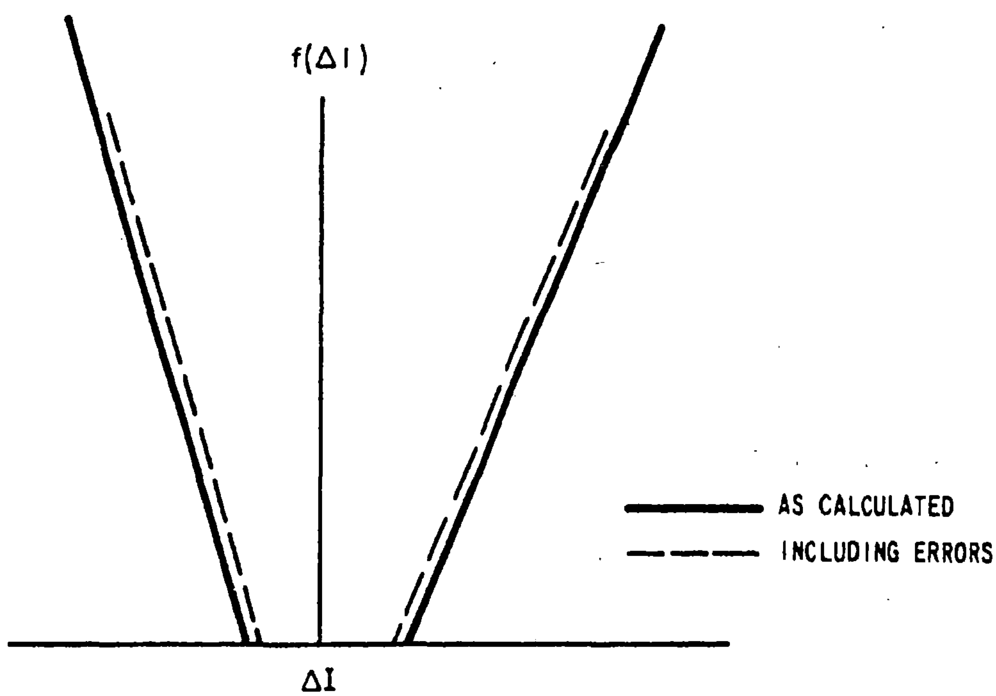



(A)




(B)

Figure C-1. Determination of Overpower ΔT f(ΔI) Function

Shown on figure C-1b is the trip reset function derived from the above equation and the reset function that would actually be incorporated into the thermal overpower trip circuitry. Because the uncertainty allowance in measurement of $\Delta I$ is 3 percent, the deadband has been reduced by 3 percent in $\Delta I$ on both the positive and negative sides, and the slopes of the wings have been maintained.

## C-2. DETERMINATION OF OVERTEMPERATURE $\Delta T$ $f(\Delta I)$ FUNCTION

The technique for determining the $\Delta T$ trip reset function, $f(\Delta I)$, for the thermal overtemperature trip function is described in the following paragraphs. Needed are: the core thermal limits from paragraphs 4-1 through 4-4, core power distribution information (the axial offset envelopes) from paragraphs 4-5 through 4-8, and the overtemperature and overpower $\Delta T$ protection limit equations from appendix B.

a,c

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a,c

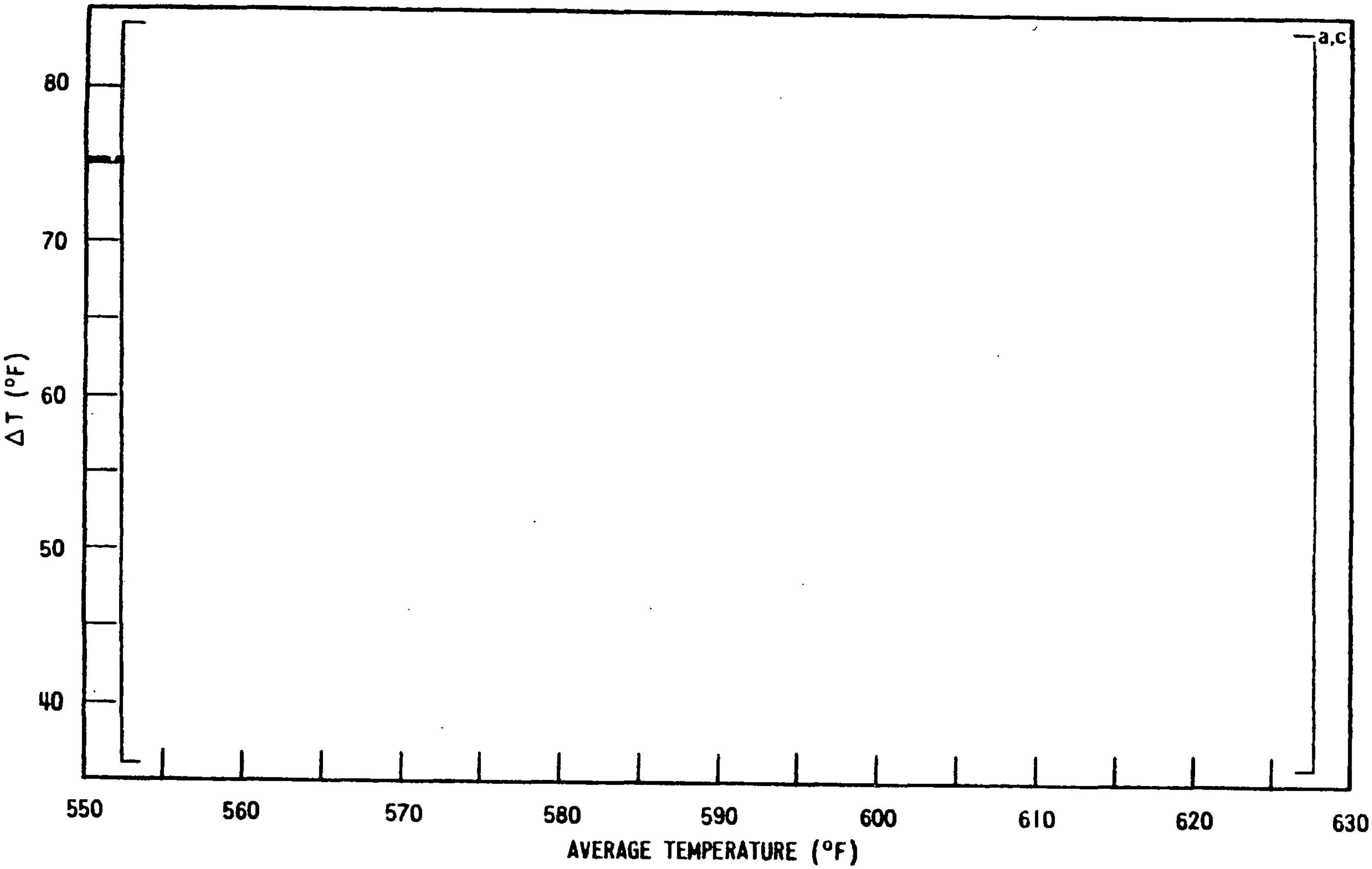


Figure C-2. Illustration For Developing ΔI Function

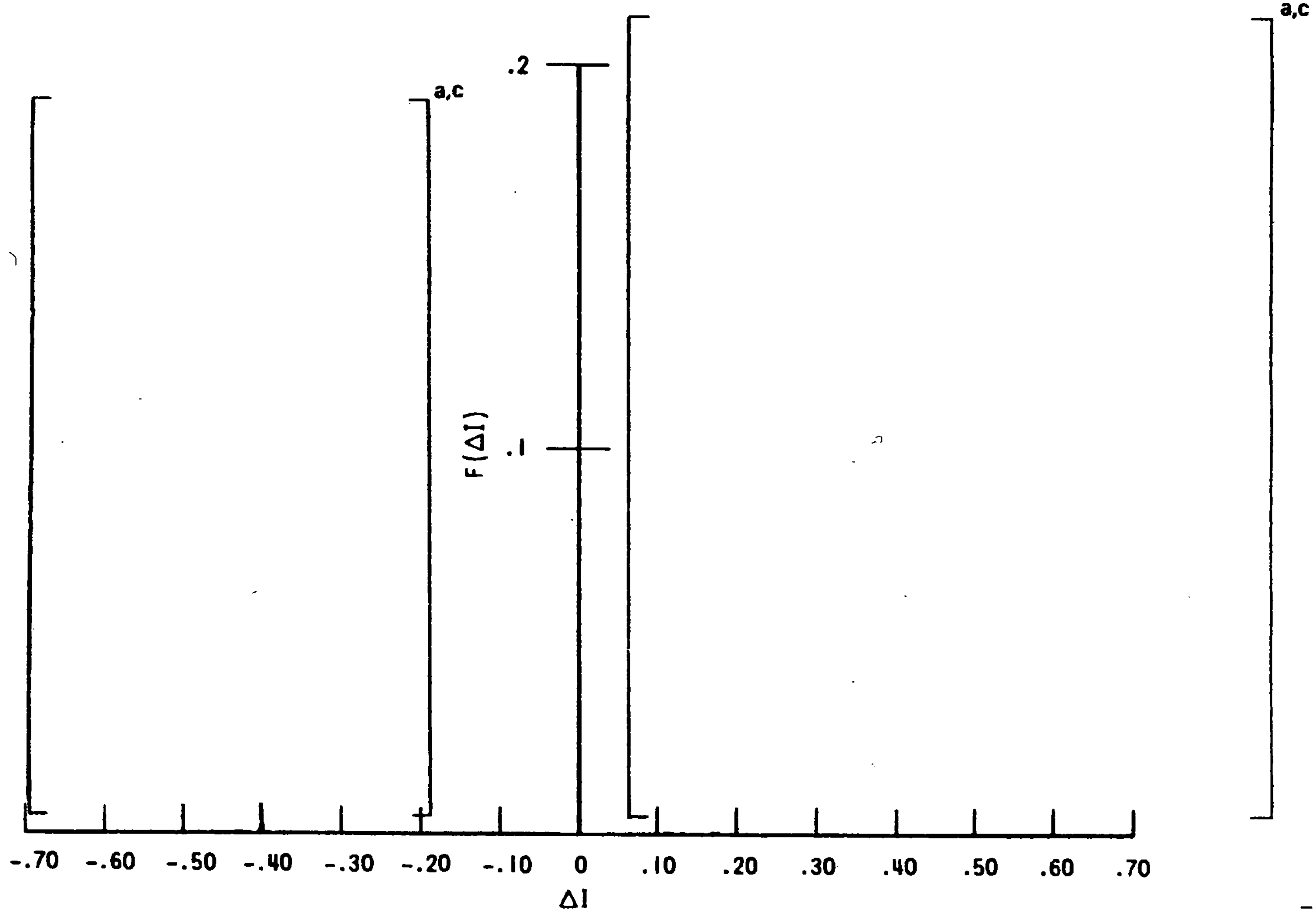


Figure C-3. Typical Overtemperature ΔT Reset Function

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# APPENDIX D
# COUPLED CORE-SYSTEM EVALUATION MODEL DESCRIPTION

It was recognized at the outset of the study discussed in Section 5 that two design computer codes already existed which together contained all the desired features discussed in paragraphs 5-1 through 5-5. These design codes are LOFTRAN,[1] a lumped-parameter single-loop system model used to study the transient response of a pressurized water reactor system, and TWINKLE,[2] a multi-dimensional spatial neutron kinetics code patterned after steady state codes presently used for reactor core design. A coupled core-system evaluátion model was conveniently established by merely mating these two design tools in a compatible fashion. Hereafter, this evaluation model will be referred to as LOFTRAN/TWINKLE.

## D-1. LOFTRAN/TWINKLE MODEL

A multi-loop system is simulated in LOFTRAN by a lumped-parameter single-loop model containing the reactor vessel, hot and cold leg piping, steam generator (tube and shell sides) and the pressurizer. 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. Reactor protection trip functions available in the code include reactor trips on neutron flux, overpower ΔT and overtemperature ΔT, high and low prcssure, low flow, and high pressurizer level. Control systems also simulated include rod control, steam dump, feedwater control, and pressurizer pressure control. The Safety Injection System, including the accumulators, is also modeled. Although LOFTRAN includes core reactivity feedback effects in the form of a point kinetics model, these calcualtions are obviously bypassed when LOFTRAN is linked to the TWINKLE code. WCAP-7907 gives supporting documentation for the LOFTRAN code.[1]

TWINKLE, a spatial neutron kinetics code, utilizes an implicit finite-difference method to solve the two-group transient neutron diffusion equations in one, two, and three dimensions. TWINKLE uses six delayed-neutron groups and contains a detailed multiregion fuel-clad-coolant heat-transfer model for calculating pointwise Doppler- and moderator-feedback effects. Aside from basic nuclear cross-section data and thermal-hydraulic parameters, TWINKLE accepts as

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1. Burnett, T. W. T., McIntyre, C. J., and Burker, J. C., "LOFTRAN Code Description," WCAP-7907, June, 1972.
2. Risher, D. H., Jr., and Barry, R. F., "TWINKLE – A Multi-Dimensional Neutron Kinetics Computer Code," WCAP-8028-A, January, 1975.

input basic driving functions such as inlet temperature, pressure, flow, boron concentration, control-rod motion, and others. WCAP-8028-A further describes the TWINKLE computer program.[1]

For this study, the LOFTRAN/TWINKLE coupled core-system model was developed to compute the normalized core average heat-flux profile in one-dimensional axial geometry. This model geometry is identical to the geometry used in the static nuclear calculations discussed in paragraphs 5-7 through 5-9. Consequently, the data transfer of core average axial burnup and xenon distributions was straightforward. The use of the normalized core average axial heat flux profile to construct the hot-rod power shape and the calculation of the core minimum DNB ratio will be discussed later in this section.

a,c

## D-2. MODEL VERIFICATION

Calculations were performed to verify that combining the LOFTRAN and TWINKLE programs into a coupled core-system model did not alter the results that would be obtained from the two reference programs executed in an independent fashion. Two differing Condition II events were run with the LOFTRAN/TWINKLE model:

- An Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power, and
- An Excessive Load Increase Event.

a,c

1. Risher, D. H., Jr., and Barry, R. F., "TWINKLE – A Multi-Dimensional Neutron Kinetics Computer Code," WCAP-8028-A, January, 1975.

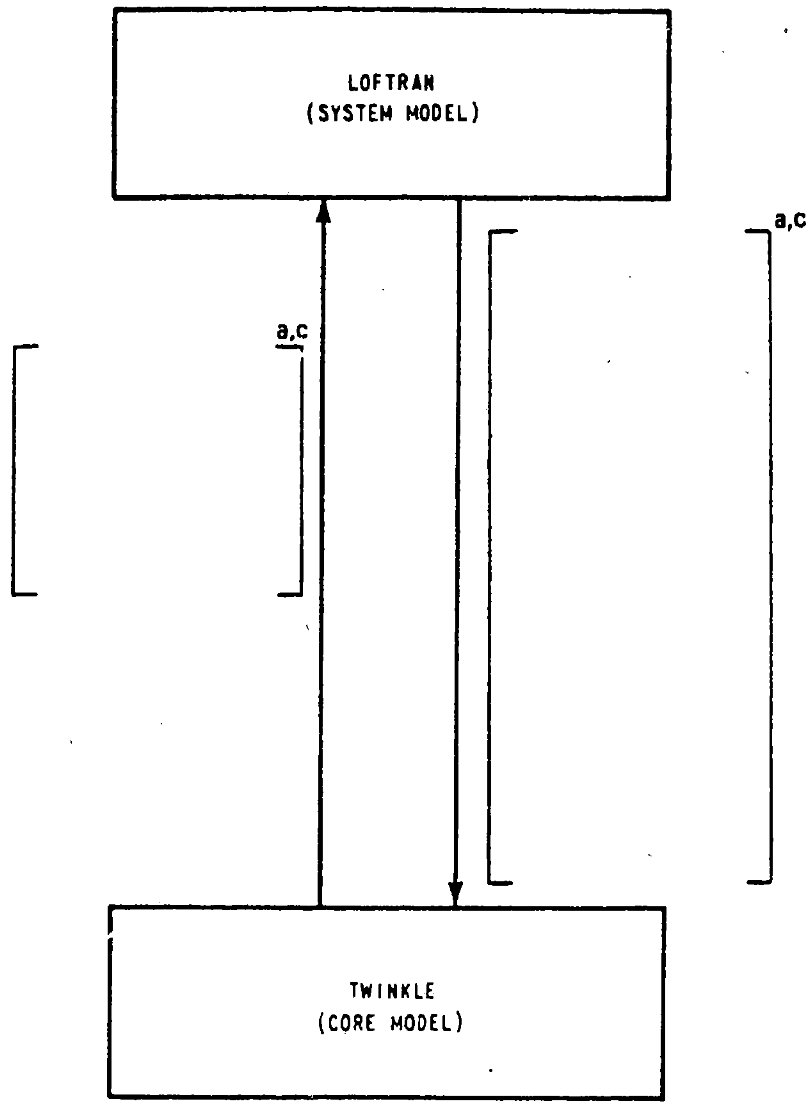


Figure D-1. Coupled Core-System Evaluation Model

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][a,c] In every comparison between the LOFTRAN/TWINKLE model and the reference programs, the agreement was excellent.

## D-3. DNBR CALCULATION

The results of the LOFTRAN/TWINKLE model are used to evaluate the minimum DNB ratio in the core during anticipated transients (Condition II events). These DNB ratios can then be used to illustrate the adequacy of standard methods used in the determination of the overtemperature $\Delta T$ core-protection trip setpoints. The remainder of this section describes the methods for evaluating the core minimum DNB ratio by considering the coupled core-system model response during DNB-related Condition II events.

[ ][a,c]

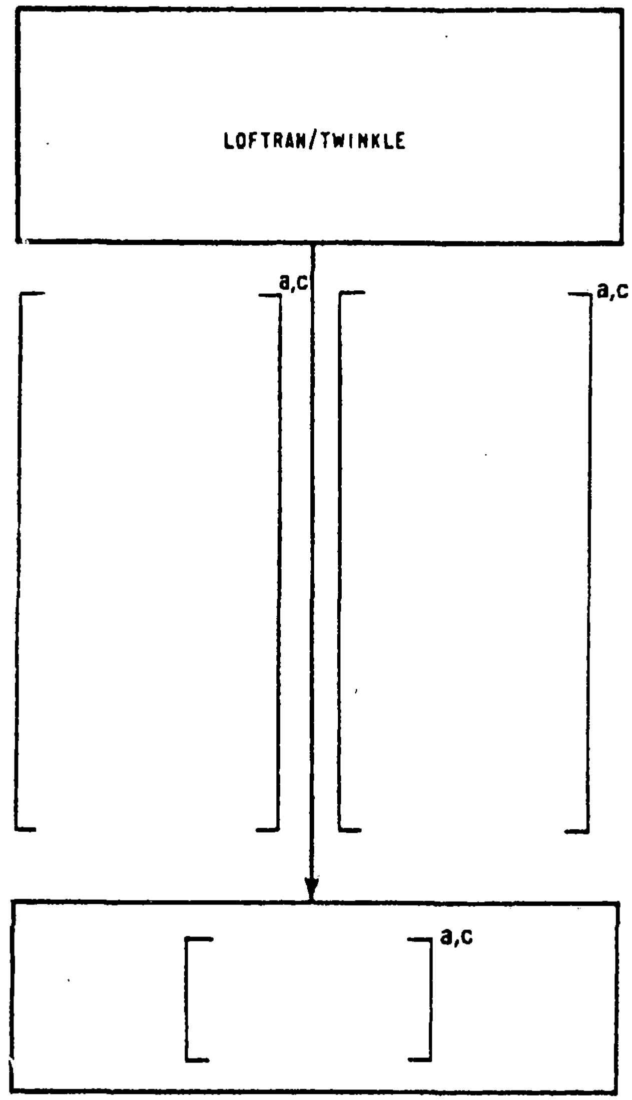


Figure D-2. Coupled Core-System DNBR Evaluation

Figure B-1. Region of Permissible Operation as Defined by Core Protection System



TABLE D-1

COUPLED CORE-SYSTEM EVALUATION MODEL
HOT ROD POWER SHAPES AND DNB RATIOS

a,c

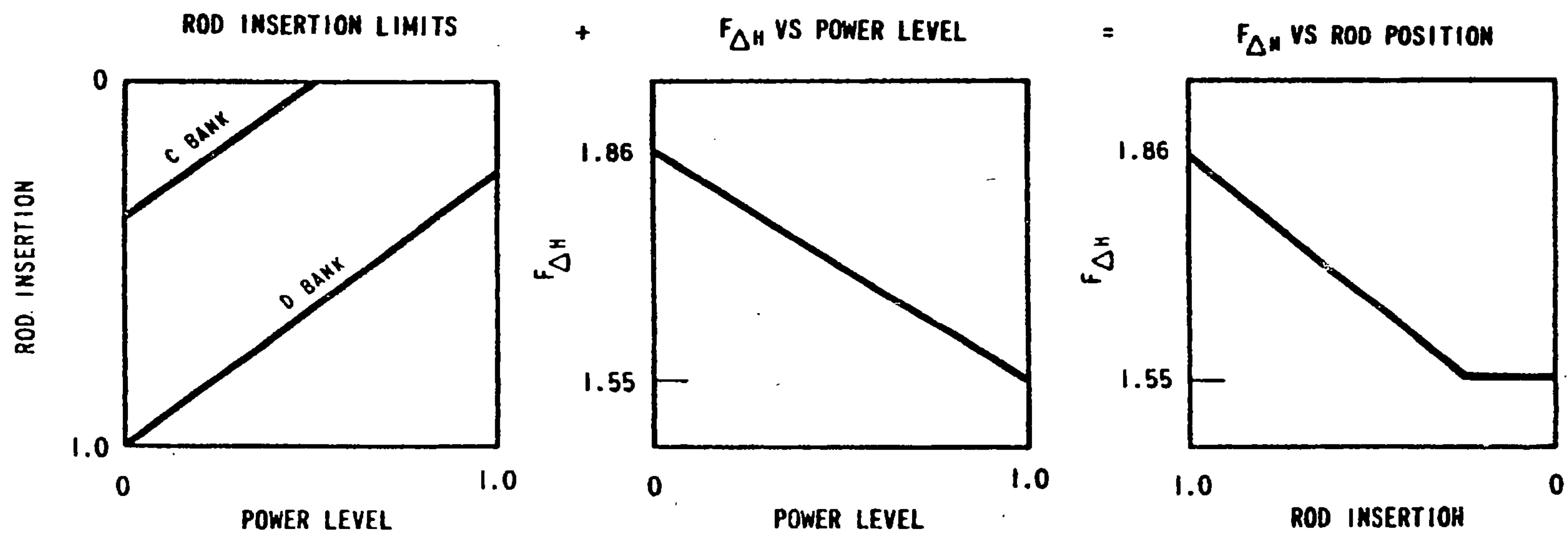


Figure D-3. $F_{\Delta H}$ as a Function of Control-Bank Position

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## D-4. TYPICAL COUPLED CORE-SYSTEM TRANSIENT RESULTS

Typical LOFTRAN/TWINKLE transient results are illustrated in figures D-4 through D-11. The results presented in these figures are very similar to those same transient results found in current Safety Analysis Reports. Any differences in transient core power level, pressure, and average temperature can be attributed to the coupling of core and system calculation models. By design, LOFTRAN/TWINKLE also provides additional information about the transient core power distribution. This is best illustrated by changes in the axial peaking factor ($F_Z$) and axial offset during the transient. These coupled core-system model results, although analyzed with conservative assumptions, provide a more realistic evaluation of NSSS responses during DNB-related Condition II events.

Figures D-4 through D-7 illustrate the results for a typical end-of-life rod withdrawal transient. This heatup event, initiated from an initial power level of 52 percent, shows the expected increases in reactor coolant system pressure, core power level, core inlet temperature, and core average temperature due to the primary-secondary power mismatch. Also expected for this event is the shifting of the core power distribution towards the top of the reactor core (positive axial offset) as the control rods are withdrawn. This power distribution shift is further influenced by the non-uniform axial burnup distribution which is present in the core at end of life.

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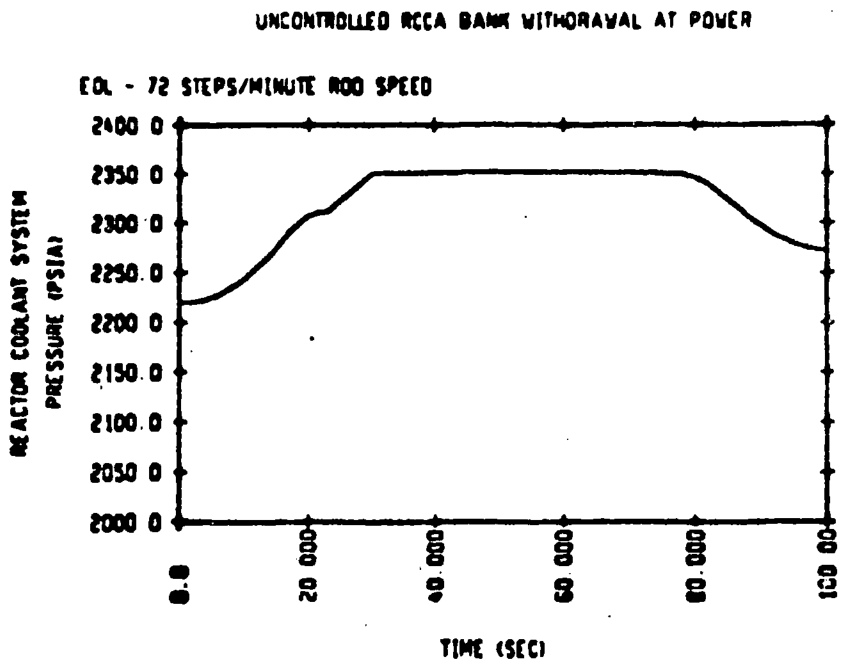


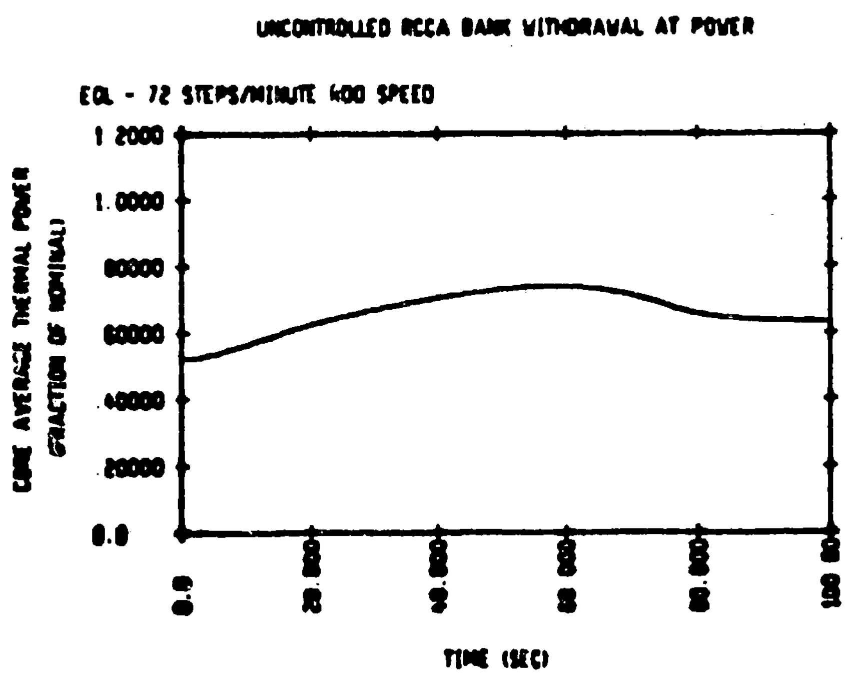
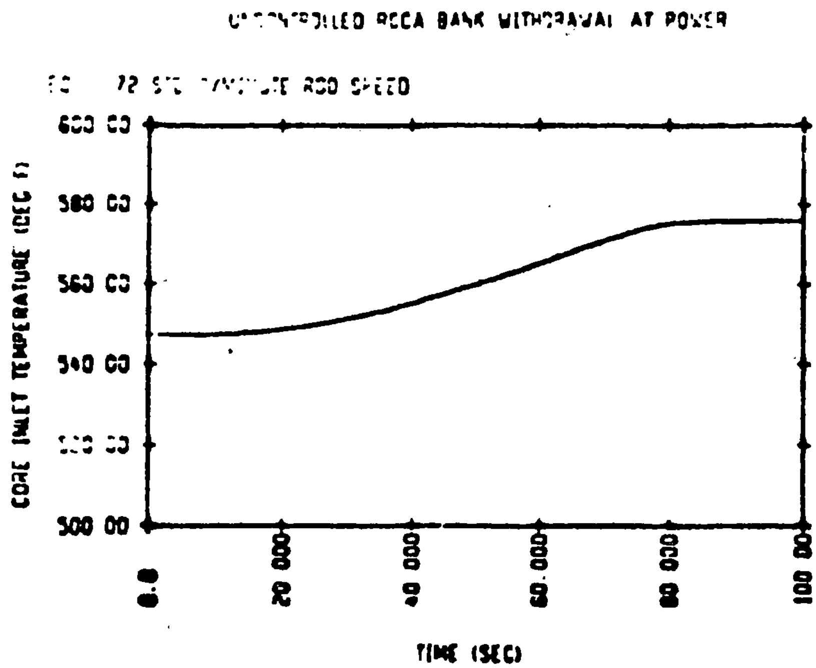




Figure D-4. Typical Coupled Core-System Transient Results (1 of 8)



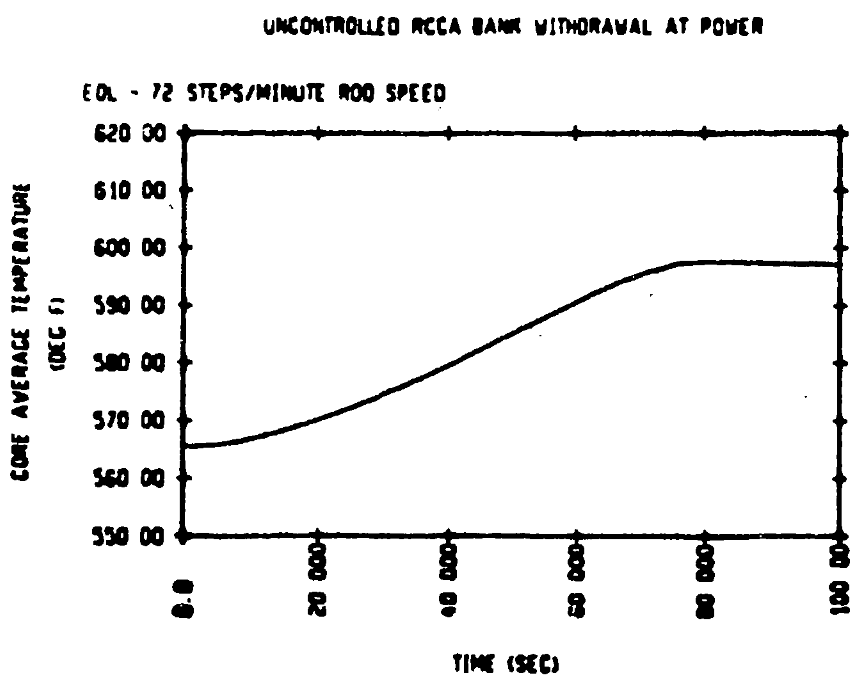
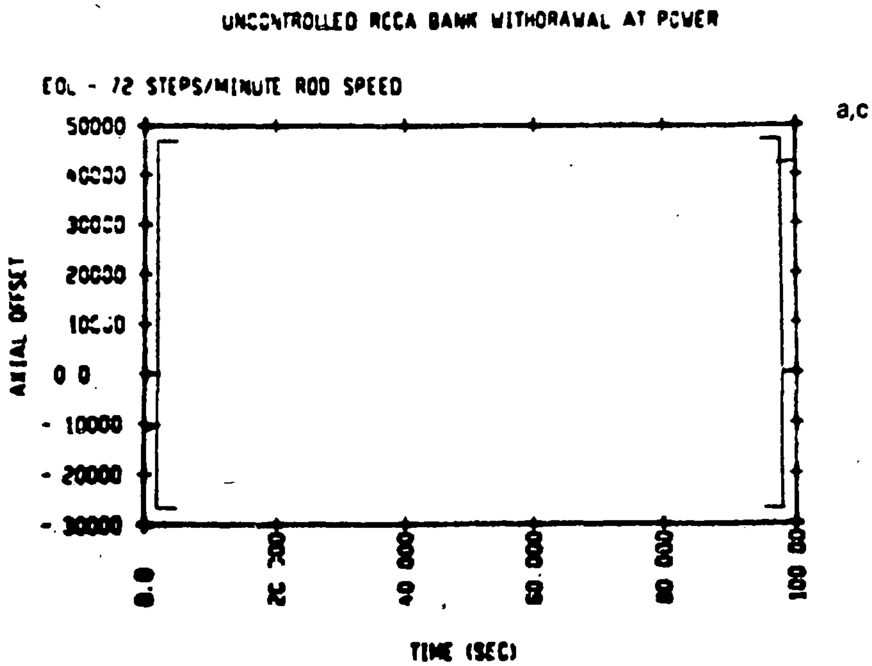




Figure D-5. Typical Coupled Core-System Transient Results (2 of 8)



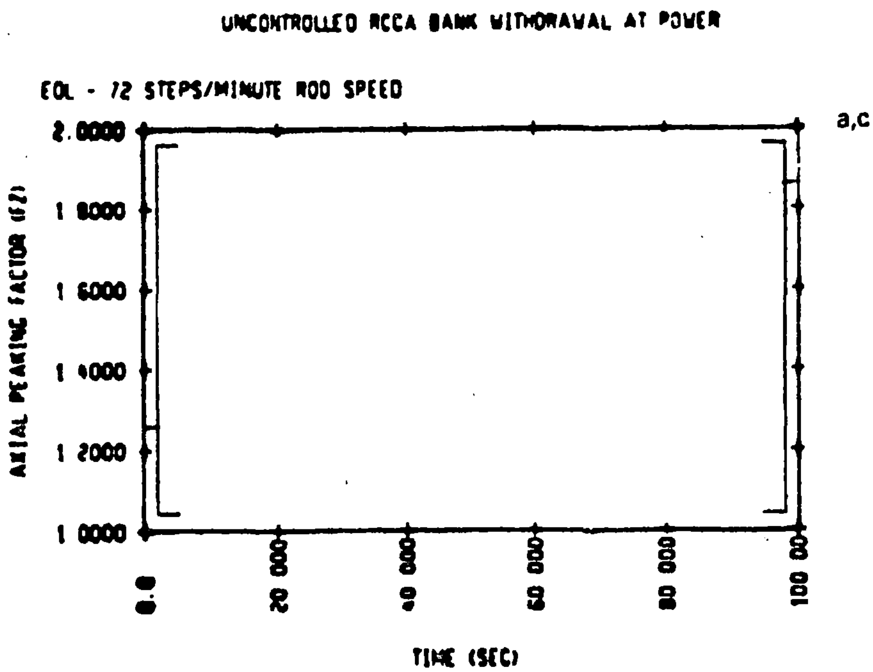




Figure D-6. Typical Coupled Core-System Transient Results (3 of 8)

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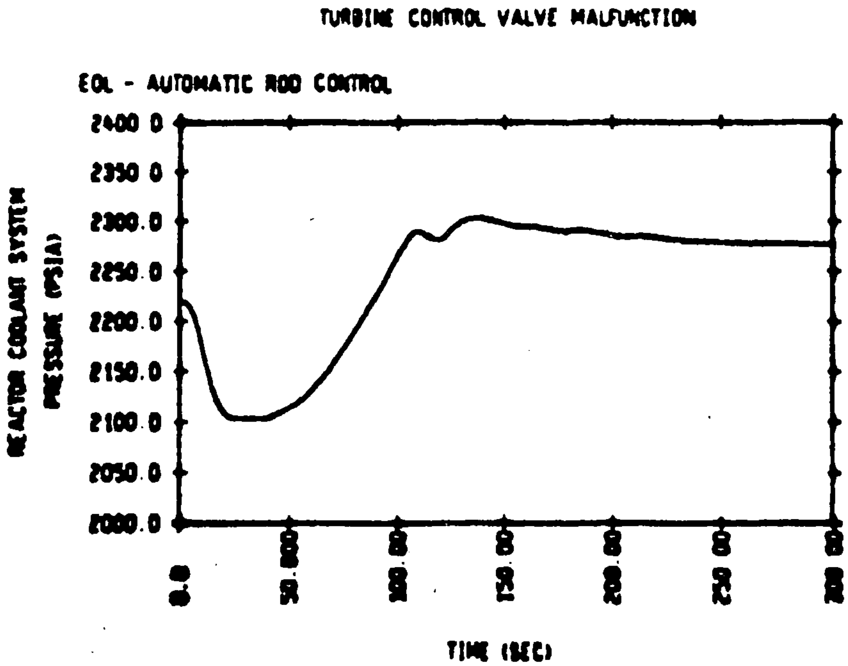
Figure D-7. Typical Coupled Core-System Transient Results (4 of 8)



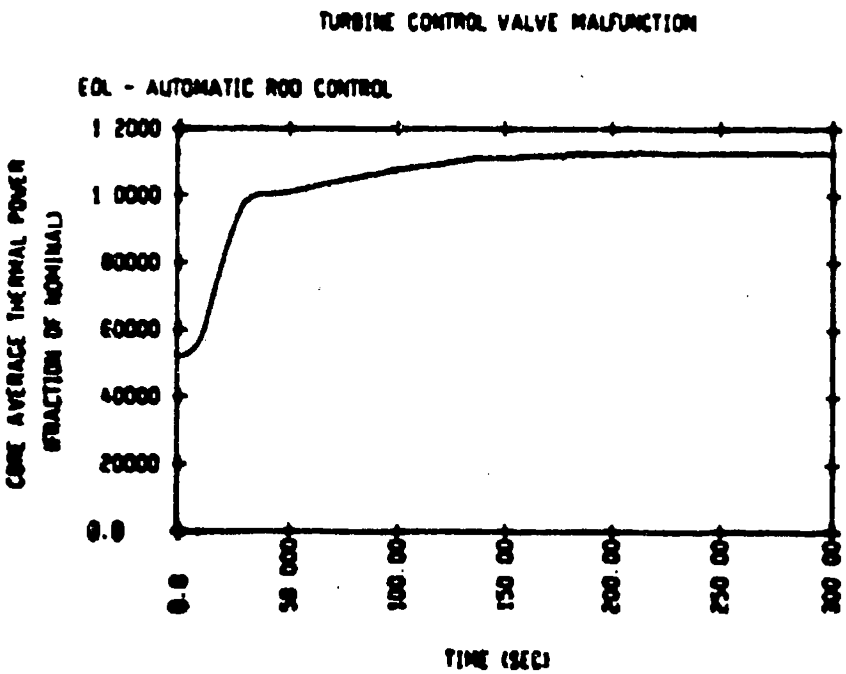
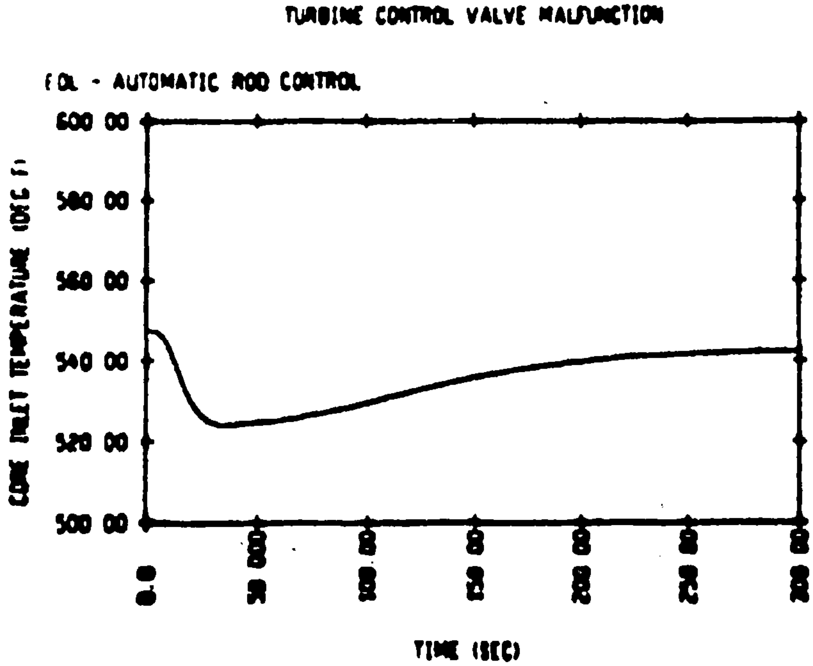




Figure D-8. Typical Coupled Core-System Transient Results (5 of 8)



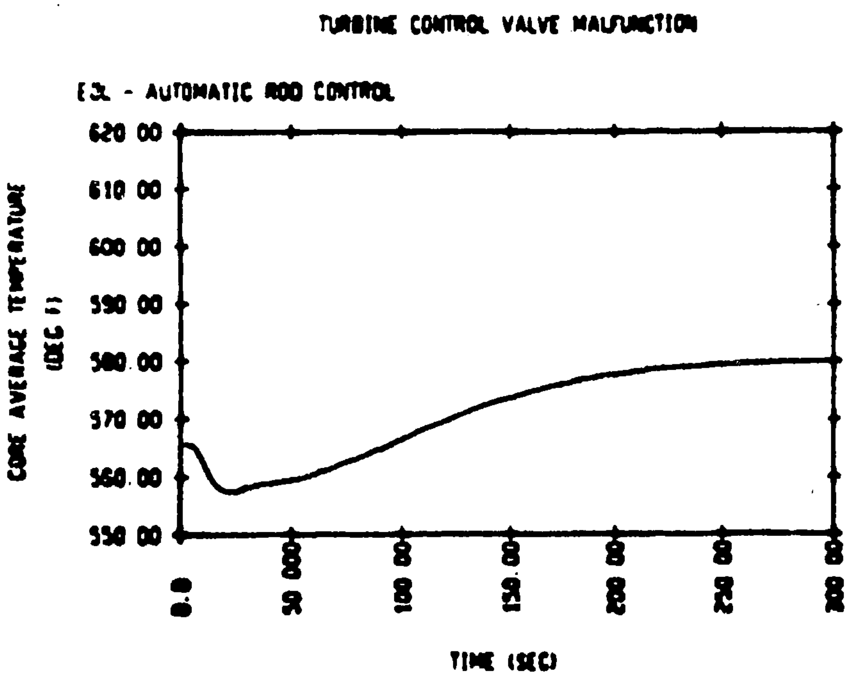
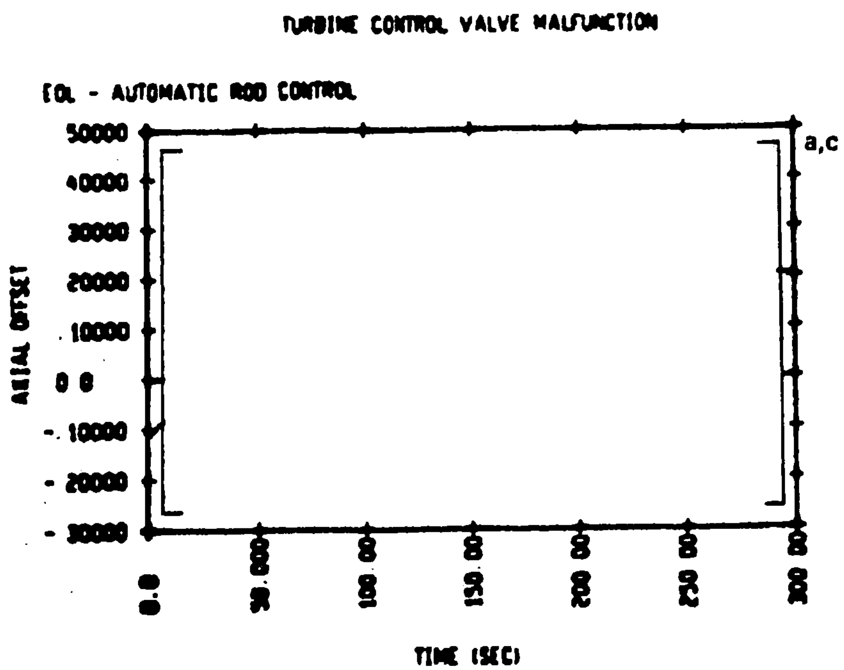




Figure D-9. Typical Coupled Core-System Transient Results (6 of 8)



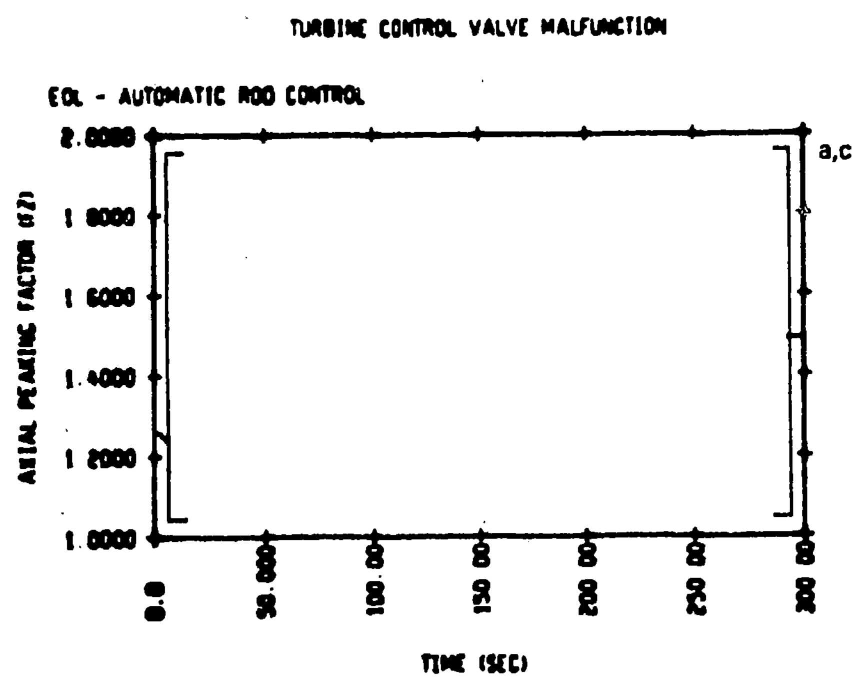




Figure D-10. Typical Coupled Core-System Transient Results (7 of 8)

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Figure D-11. Typical Coupled Core-System Transient Results (8 of 8)

The results presented in figures D-8 through D-11 are for a typical cooldown transient. The simulation of an end-of-life turbine control valve malfunction with automatic rod control is illustrated in these figures. The initial decreases in reactor coolant system pressure, core inlet temperature, and core average temperature are typical for this event. Likewise, a sudden increase in core power level is also observed due to moderator feedback effects. [

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