VEP-NFE-2-A
DECEMBER, 1984

Vepco

VEPCO EVALUATION OF THE CONTROL ROD EJECTION TRANSIENT

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OCTOBER 1983

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R. M. Berryman
Director, Nuclear Fuel Engineering

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

September 26, 1984

Mr. W. L. Stewart, Vice President
Virginia Electric and Power Company
Richmond, Virginia 23261

Dear Mr. Stewart:

Subject: Acceptance for Referencing of Licensing Topical Report VEP-NFE-2 (P/NP), "Vepco Evaluation of the Control Rod Ejection Transient"

We have completed our review of the subject topical report submitted November 23, 1983 by Virginia Electric and Power Company letter No. 657. We find that this report describes an acceptable method for analyzing the control rod ejection accident in Vepco's Surry and North Anna reactors and this method may be referenced in future Vepco 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 bases 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 Vepco 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, Vepco will be expected to revise

and resubmit the respective documentation, or submit justification for the continued effective applicability of the topical report without revision of the respective documentation.

Sincerely,

Cecil O. Thomas

Cecil O. Thomas, Chief
Standardization and Special
Projects Branch
Division of Licensing

Enclosure:
As stated

ENCLOSURE

CORE PERFORMANCE BRANCH SAFETY EVALUATION OF VEPCO EVALUATION OF THE CONTROL ROD EJECTION TRANSIENT VEP-NFE-2

Enclosure

CORE PERFORMANCE BRANCH SAFETY EVALUATION OF VEPCO EVALUATION OF THE CONTROL ROD EJECTION TRANSIENT VEP-NFE-2

1.0 SUMMARY OF TOPICAL REPORT

This report describes the methods developed by the Virginia Electric and Power Company (Vepco) for the analysis of a postulated control rod ejection transient in the North Anna and/or Surry Nuclear Power Stations. A description of the rod ejection transient and a discussion of the acceptance criteria which must be met to assure the safe operation of the plant in the event of such a transient is also presented.

The RETRAN computer program is used for the analysis which is performed in two parts. First, a point kinetics analysis is used to calculate the average core nuclear power history. Next, a hot spot thermal-hydraulic calculation is used to determine the hot spot enthalpy and temperature transients from which the amount of fuel damage and radiological consequences are assessed. Detailed descriptions of the calculational model and the techniques employed in the analysis are presented.

The results of sensitivity studies used to quantify the effect of uncertainties in important core parameters and modeling assumptions on the model's predictions are shown. Also presented are comparisons of the results of the fuel vendor (Westinghouse) methodology with the Vepco methodology as well as comparisons of point kinetics results with three-dimensional space-time kinetics model results.

2.0 STAFF EVALUATION

We have reviewed the subject report, including the mathematical models and analytical procedures and methods. The RETRAN computer program is the principal calculational tool. A point kinetics analysis is used

to calculate the average core nuclear power during a rod ejection transient. The hot spot (hottest fuel pin) enthalpy and temperature transients are determined from a hot spot thermal-hydraulic calculation. This is the usual procedure used by the nuclear industry to analyze the spatially dependent transient with a point kinetics model and has been found to be acceptable and usually conservative.

Although the RETRAN program is used to analyze the rod ejection transient, the report emphasizes the implementation of the RETRAN models since the detailed code description is available in a separate document. Therefore, we did not review the RETRAN program, per se, but rather the qualification of its use in determining the consequences of a rod ejection transient.

The staff position, as well as that of most of the reactor vendors and licensees, has been to limit the average fuel pellet enthalpy at the hot spot following a rod ejection transient to 280 calories per gram (cal/gm). This was based primarily on the results of the SPERT tests which showed that, in general, fuel failure consequences for UO_2 have been insignificant below 300 cal/gm for both irradiated and unirradiated fuel rods as far as rapid fragmentation and dispersal of fuel and cladding into the coolant are concerned. In this report, Vepco has chosen more stringent design limits. The limiting fuel failure criterion has been reduced to 225 cal/gm for unirradiated rods and 200 cal/gm for irradiated rods. Since this is a conservative revision, the staff finds these criteria acceptable.

Vepco proposes a clad temperature limitation of 2700°F as the temperature above which clad embrittlement may be expected. Although this is several hundred degrees above the maximum clad temperature limitation proposed in the ECCS criteria, the staff feels this is adequate in view of the relatively short time at temperature and the highly localized effect of this transient. The staff has no limiting temperature criterion for rod ejection transients.

The neutronics model including the reactivity insertion, neutron kinetics parameters, fuel and moderator temperature feedback, and reactor trip assumptions described in the report are in conformance with the recommendations of USNRC Regulatory Guide 1.77 and are acceptable. Since RETRAN uses a point kinetics neutronics model, the effect of locally peaked core flux shapes due to the rod ejection is not included in the Doppler reactivity calculation. Therefore, a power weighting factor (PWF) is used to modify the Doppler reactivity versus core average fuel temperature data in order to better approximate the local reactivity effects. The TWINKLE computer program is used by Vepco to justify the use of a PWF. Since TWINKLE is a three-dimensional spatial kinetics code, it accounts for the highly localized reactivity effects of a control rod ejection accident more realistically than a point kinetics calculation and is, therefore, an acceptable method for verifying the PWF. For cases initiated from both hot zero power and hot full power initial conditions, the RETRAN point kinetics predictions are shown to be conservative compared to the TWINKLE three-dimensional predictions and, therefore, the described RETRAN model is acceptable.

The radiological effects of a rod ejection transient have been addressed generically in the Surry 1 and 2 and the North Anna 1 and 2 Updated Final Safety Analysis Reports as well as in the Westinghouse "Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods" (WCAP-7588). In this latter report, a departure from nucleate boiling (DNB) analysis was performed using the transient THINC-III code for a worst-case three-dimensional rod ejection transient. The results indicated that less than 10% of the fuel rods enter DNB. Since one of the NRC acceptance criteria for this transient requires the assumption that all of the fuel rods which experience DNB release their entire gap inventory of fission products to the coolant, the generic result indicates that less than 10% of the core will release fission products. The source term for the radiological release calculations for these and other Westinghouse designed plants is, therefore, based on 10% of the fuel experiencing clad failure and releasing their

entire gap activity. This is in conformance with Regulatory Guide 1.77 and, hence, acceptable.

Since there is no fuel dispersal into the coolant, the pressure surge during the transient may be calculated based on conventional heat transfer methods. Generic pressure surge calculations for the most severe excess addition of energy to the coolant indicate that any system overpressurization due to a rod ejection transient will meet the NRC criterion of being less than that pressure which would cause stresses to exceed the Service Limit C as defined in Section III of the ASME Boiler and Pressure Vessel Code.

Vepco has analyzed the rod ejection transient using Westinghouse codes and techniques and compared their results to the results obtained by Westinghouse. We have reviewed these comparisons and concur that Vepco can adequately replicate Westinghouse results. We have also reviewed comparisons between results obtained with the NRC approved Westinghouse methodology with those obtained with the Vepco methodology presented in the topical report using RETRAN. We concur that these comparisons demonstrate the acceptability of the Vepco methodology. A comparison is also presented between the Vepco calculated results using both Westinghouse and Vepco methodologies and results obtained by Vepco using the Westinghouse three-dimensional space-time kinetics code, TWINKLE. The staff finds that this comparison adequately demonstrates the conservatism of the Vepco rod ejection calculational method.

3.0 EVALUATION PROCEDURE

The staff has reviewed the report within the guidelines provided by Sections 4.3 and 15.4.8 of the Standard Review Plan (NUREG-75/087) and by Regulatory Guide 1.77. Part of our review was based on our familiarity with and comparison of similar analyses for control rod

ejection transients provided in topical reports by other PWR vendors. The staff also had the benefit of a meeting with VEPCO concerning the report.

4.0 REGULATORY POSITION

The subject report (VEP-NFE-2) provides an acceptable method for analyzing the control rod ejection event for Vepco's Surry and North Anna Nuclear Power Stations. This methodology may be referenced in future Vepco license applications.

CLASSIFICATION/DISCLAIMER

The data, information, analytical techniques, and conclusions in this report have been prepared solely for use by the Virginia Electric and Power Company (the Company), and they may not be appropriate for use in situations other than those for which they were specifically prepared. The Company therefore makes no claim or warranty whatsoever, express or implied, as to their accuracy, usefulness, or applicability. In particular, THE COMPANY MAKES NO WARRANTY OF MERCHANTABILITY OR FITNESS FOR A PARTICULAR PURPOSE, NOR SHALL ANY WARRANTY BE DEEMED TO ARISE FROM COURSE OF DEALING OR USAGE OF TRADE, with respect to this report or any of the data, information, analytical techniques, or conclusions in it. By making this report available, the Company does not authorize its use by others, and any such use is expressly forbidden except with the prior written approval of the Company. Any such written approval shall itself be deemed to incorporate the disclaimers of liability and disclaimers of warranties provided herein. In no event shall the Company be liable, under any legal theory whatsoever (whether contract, tort, warranty, or strict or absolute liability), for any property damage, mental or physical injury or death, loss of use of property, or other damage resulting from or arising out of the use, authorized or unauthorized, of this report or the data, information, and analytical techniques, or conclusions in it.

ABSTRACT

This report describes the methods developed by the Virginia Electric and Power Company (Vepco) for the analysis of the postulated control rod ejection transient. The principle calculational tool used in this development has been the RETRAN transient thermal-hydraulics code; a point kinetics core physics option is used for predicting the system response of a plant undergoing a rod ejection event. The local fuel temperature response is calculated using a separate RETRAN model of the core hot spot. Comparison with the results from vendor methodologies are included to demonstrate the acceptability of the Vepco methodology for use in the reload core safety analysis and licensing process.

ACKNOWLEDGEMENTS

The authors would like to express their thanks to Messrs. N. A. Smith, N. P. Wolfhope, R. W. Cross and S. M. Bowman for their technical assistance in the development and preparation of this report. The authors would also like to express their appreciation to a number of people who reviewed and provided comments on this report.

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

SECTION 1 - INTRODUCTION

1.1 Purpose and Organization of the Report

This report documents the methodology developed by the Virginia Electric and Power Company (Vepco) for the analysis of the postulated control rod ejection transient for the North Anna and Surry Nuclear Power Stations. The intent of this methodology is to provide Vepco with the capability of performing the licensing analysis required for the Condition IV rod ejection transient addressed in the Final Safety Analysis Report.

The purpose of this report is to provide an acceptable reference for safety and licensing analysis of the rod ejection transient for a reload core. This analysis is performed in order to demonstrate that an occurrence of the transient for the core loading in question will neither interfere with the core cooling capability nor result in fuel damage which will lead to an unacceptable radiation release within the guidelines of 10 CFR Part 100, "Reactor Site Criteria." The approach adopted for verification of this transient analysis method is to compare results obtained with the Vepco methodology to results obtained by Vepco using an accepted vendor methodology. These results are demonstrated to be conservative by comparison to Vepco results based on a three-dimensional space-kinetics model and a detailed hot spot thermal hydraulic model.

The Introduction Section of this report presents a statement as to the purpose of the report, a description of the rod ejection transient, and a discussion of the acceptance criteria which must be met by an analysis

to insure the safe operation of the plant in the event such a transient occurs.

Section 2 provides detailed descriptions of the calculational model used and the methods by which it is employed to analyze the transient in a conservative manner. The licensing analysis is performed in two parts:

1. a point kinetics analysis to calculate the average core nuclear power history, and
2. a hot spot thermal-hydraulic calculation to determine the hot spot enthalpy and temperature transients from which the amount of fuel damage and radiological consequences of the accident may be assessed.

Both of the analyses are performed with the RETRAN transient thermal-hydraulic analysis code, but with different code models for each part. Description of the two RETRAN models, henceforth referred to as the RETRAN Single Loop Model for the average core nuclear power history calculation, and the RETRAN Hot Spot Model for the hot spot transient calculation, are provided in Section 2.

The actual techniques employed in the analysis are described in the latter part of Section 2, including the additional concerns of providing core physics parameters for the RETRAN models for the specific core loading to be analyzed, and the investigation of the system overpressure and radiological concerns.

The use of point reactor kinetics instead of spatial kinetics in the core average power history analysis allows for application of a weighting factor to the Doppler reactivity feedback model used in the

point kinetics analysis. This weighting factor, henceforth referred to as the Power Weighting Factor, provides a more accurate, although conservative, estimate of the actual Doppler reactivity feedback to be expected during the transient. A discussion of the derivation and use of this weighting factor is also presented in Section 2.

The results of a series of sensitivity studies performed with the RETRAN models used for the rod ejection study are provided in Section 3. These sensitivity studies are used to quantify the impact of uncertainties in important core parameters and modeling assumptions on the models' predictions.

Comparisons of the results of a vendor methodology for standard FSAR and reload core licensing analyses with those of the Vepco developed methodology are provided in Section 4 to further quantify the acceptability of the Vepco approach. In addition, a comparison of power history calculations between the point kinetics RETRAN model and a three-dimensional space-time kinetics model is presented to demonstrate the conservatism of the point kinetics approach.

The report's conclusions and references are provided in Sections 5 and 6 respectively.

.2 Description of the Transient

The rod ejection transient is a postulated Condition IV event initiated by the mechanical failure of a control rod mechanism pressure housing. Following such a failure, the action of the coolant pressure is assumed to result in a rapid ejection of a rod cluster control assembly and drive shaft from the core to a fully withdrawn position within a time interval on the order of 0.1 seconds. This in turn leads to a fast reactivity insertion and may cause a severe asymmetric core power distribution, possibly leading to fuel rod damage.

Reactor protection for the transient is provided by negative reactivity feedback effects and by reactor trips on high neutron flux levels. Trip setpoints for the Surry and North Anna nuclear units are 118% of full power (including setpoint and instrument errors) for reactor operations at or above 10% full power, and 35% of full power (including setpoint and instrument errors) for reactor operations below 10% full power. After an appropriate delay, the rod cluster control assemblies assumed available (less the ejected rod) are assumed to drop into the core. However, before such rod movement is initiated, the large power surge of the core is turned around by the negative Doppler reactivity feedback resulting from the quick rise in the core's fuel temperature, particularly in the vicinity of the ejected rod. This proves to be the dominant effect in limiting the consequences of the transient.

In general, the core loading for each reload cycle is designed so as to limit the amount of reactivity insertion which would result from the ejection of a rod cluster control assembly from any of the locations

here one would be inserted during normal power operation. The consequence of such a rod ejection event would be a very rapid increase in core average power level, with an accompanying core pressure surge and a particularly severe temperature transient in the vicinity of the ejected rod. This localized temperature transient may be severe enough to cause some of the fuel to experience DNB (Departure from Nucleate Boiling) and localized fuel damage. The degree of potential damage will be mainly governed by the worth of the ejected rod.

Additional details on the postulated transient's description, available protection mechanisms, and consequences may be found in the appropriate Updated FSARs (UFSARs) for the Surry and North Anna Nuclear Power Stations (Refs. 1 and 2.)

.3 Acceptance Criteria

As detailed in USNRC Regulatory Guide 1.77 (Ref. 3), the acceptance criteria to be used in evaluating the results of a rod ejection transient analysis are:

1. Reactivity excursions will not result in a radial average fuel enthalpy greater than 280 cal/g (504 Btu/lb) at any axial location in any fuel rod.
2. Maximum reactor pressure during any portion of the assumed transient will be less than the value that will cause stresses to exceed the Emergency Condition stress limits as defined in Section III of the ASME Boiler and Pressure Vessel Code.
3. Offsite dose consequences will be well within the guidelines of 10 CFR Part 100, "Reactor Site Criteria."

It should be noted that the 280 cal/g criteria applies to both irradiated and unirradiated fuel in providing a conservative maximum limit to limit fuel damage from the prompt pin burst phenomenon and not impair significantly the core cooling capabilities.

Additional in-house design limits for acceptability of the transient analysis for a reload have been imposed by Vepco. These limits are:

1. peak clad temperature less than or equal to 2700 °F,
2. fuel centerline melting less than or equal to 10% at the hot spot,
3. average hot spot fuel enthalpy less than 225 cal/gm (405 Btu/lb) for unirradiated fuel, and

4. average hot spot fuel enthalpy less than 200 cal/gm (360 Btu/lb) for irradiated fuel.

The peak clad temperature limit of 2700 °F reflects a conservative estimate of the temperature at which clad embrittlement may occur. These additional criteria reflect more severe limits than those delineated in Regulatory Guide 1.77, and are consistent with the limits applied in the design of both the Surry and North Anna Nuclear Power Stations.

The predicted results of an analysis which fall within the acceptability limits of these criteria demonstrates that the consequences of a rod ejection transient for the specific core reload design will be acceptable to the safety of the general public and will maintain the integrity of the core cooling capability.

SECTION 2 - METHODOLOGY

2.1 Calculational Model -- The RETRAN Code

The principal calculational tool used by Vepco for the rod ejection transient analysis is the RETRAN computer code. The RETRAN code has been developed by Energy Incorporated (EI) under the auspices of the Electric Power Research Institute (EPRI) as a generalized and versatile computer code for the analysis of light water reactor systems. The theory and numerical algorithms, programming details, user's input manual, and examples of its applications to Nuclear Steam Supply Systems (NSSS) are thoroughly documented for the RETRAN code in Volumes 1 through 4 of Ref. 4.

The RETRAN computer code is based upon the RELAP4/003 Update 85 code which was released by the United State Nuclear Regulatory Commission as part of the Water Reactor Evaluation Model (WREM), (Ref. 5).

The RETRAN fluid differential and state equations used to represent homogeneous equilibrium flow in one dimension are described in Ref. 4. The representations used in previous RELAP codes for control volumes and junctions are also used in RETRAN and allow the analyst to model a system in as much detail as desired. The modeling flexibility of the code is important and will be described in more detail in Sections 2.1.1 and 2.1.2 below.

The equation systems, which describe the flow conditions within the channels, are obtained from the local fluid conservation equations of mass, momentum and energy by use of mathematical integral-averaging

techniques. Forms of the momentum equation are available for both compressible and incompressible flow.

The heat conduction representation capabilities of RETRAN have been increased over previous RELAP versions. The principal augmentation to RETRAN is the capability to more accurately calculate two-sided heat transfer. The appropriate heat transfer correlation is selected based on thermodynamic conditions in each of two flow streams, on either side of a heat conducting solid. Consequently, representations of the heat transfer processes occurring in the steam generator, for example, are more accurate than previously possible.

Reactor kinetics are represented in RETRAN using a point kinetics model with reactivity feedback. The reactivity feedback can be represented by constant coefficients or in tabular form and accounts for explicit control actions (e.g., rod scram) and changes in fuel temperature, moderator temperature and density, and soluble boron concentration.

The system component models utilized in RETRAN include a pump model that describes the interaction between the centrifugal pump and the primary system fluid, and valve models that represent either simple valves, check valves or inertial valves. The flexibility of the valve representation and their configuration is important in allowing a wide variety of options to the user for the modeling of system dynamics. Several representations of heat exhangers can be modeled by the code. These include the previously discusssed two-sided heat transfer and one-sided heat transfer in conjunction with user specified boundary conditions. A non-equilibrium pressurizer can be modeled in which the

thermodynamic state solutions of the liquid and vapor regions of the pressurizer are determined from a distinct mass and energy balance for each region.

As in RELAP, a variety of trip functions can be modeled in the RETRAN code to represent various reactor protection system actions. A refinement of the RETRAN code over the RELAP4 code is the addition of a reactor control system modeling capability. Consequently, the dynamics of linear and non-linear control systems are represented with RETRAN models of the more common analog computer elements. This additional capability is necessary for both best-estimate and licensing analysis, since the response of various control and protection systems may have a significant effect on the overall system response.

The analysis of the rod ejection transient has been performed by Vepco with the RETRAN-02 version of the code. The second major code version of RETRAN, RETRAN-02 was released to allow for additional capability in modeling certain BWR transients, small break LOCAs, balance of plant modeling and anticipated transients without scrams. Volume 1 of Ref. 4 provides a detailed summary of the changes and new capabilities available with RETRAN-02.

One of these new capabilities is a one-dimensional kinetics model which allows a spatial kinetics modeling of the core in the axial direction (that is, along the axis of fluid flow through the core) to be used in place of the simpler point kinetics model. Since this is the dimension through which the rod is ejected, the one-dimensional kinetics model appears to offer the advantage of a more accurate modeling of the rod

ejection event in place of the point kinetics model. However, since the primary power redistribution effect which contributes to the negative rectivity feedback in the transient is in the radial plane, no significant advantage over using the point kinetics model is obtained by using the one-dimensional model. On the other hand, the simplicity of use and the flexibility of the point kinetics model compared to the one-dimensional kinetics model make the former the preferential choice for modeling this transient.

The licensing analysis for the rod ejection event as performed by Vepco is divided into two parts--a point kinetics analysis to calculate a conservative core average power history, and a prediction of the enthalpy and temperature transients in the hot spot of the core based on that power history. The core average power history calculation is performed using the point kinetics option of the RETRAN-02 code employing the standard Vepco Single Loop Model. This model, as described below in Section 2.1.1, is further documented in Ref. 6 for performing other non-LOCA transient analyses as well as best estimate analyses for plant operational support.

The core average power history as calculated with the RETRAN Single Loop Model is weighted by the maximum total power peaking factor for the transient to conservatively estimate the power history of the hot spot. This data is input to a detailed fuel heat transfer model of the hot spot location, denoted as the RETRAN Hot Spot Model, to determine the hot spot temperature and enthalpy history as a function of time following rod ejection. Results from this analysis are used to ascertain

the extent of the fuel damage, if any, and the radiological consequences of the transient.

These two RETRAN models will be described in detail below, followed by a discussion of their application to the actual transient analysis.

1.1 RETRAN Single Loop Model

For the rod ejection transient, the system thermal-hydraulic response of all reactor coolant loops is essentially identical. Hence, a single loop representation of the NSSS suffices. Furthermore, the quickness of the core response during the transient is such that the major changes in core parameters have all taken place in the time interval of about 10 seconds, approximately the time for the coolant to make one complete pass through the primary coolant loops. Therefore, little impact on the NSSS outside of the core is expected during the rod ejection transient. Essentially, only a modeling of the reactor core need be performed in order to predict the core average power history with sufficient conservatism for input to the Hot Spot Model. However, the ready availability and extensive documentation of the Vepco RETRAN Single Loop Model make it an ideal candidate for use in performing the first part of the transient analysis. Therefore, this model is used as part of the standard Vepco methodology for the rod ejection transient.

Vepco RETRAN Single Loop Models for either the Surry or North Anna Nuclear Power Stations are similarly constructed. Both stations consist of two identical operational nuclear units. All four units are Westinghouse designed three coolant loop pressurized water reactors with core thermal ratings of 2441 Mwt for the Surry units and 2775 Mwt for the North Anna units. The three similar heat transfer loops are connected in parallel to the reactor vessel with each loop containing a centrifugal pump, loop stop valves and a steam generator. The system includes a pressurizer and the associated control system and

instrumentation necessary for operational control and protection.

The reactor vessel encloses the reactor core consisting of 157 fuel assemblies with each Surry assembly having 204 fuel rods and 21 thimble tubes arranged in a 15 x 15 array while each North Anna assembly has 264 fuel rods and 25 thimble tubes arranged in a 17 x 17 array. The fuel for both stations consists of slightly enriched uranium dioxide fuel pellets contained within a Zircaloy-4 cladding. General thermal and hydraulic design parameters for the reactor systems are listed in Table 2-1.

The RETRAN thermal hydraulic model is formulated by representing individual portions of the hydraulic system as nodes or control volumes. Control volumes are specified by the thermodynamic state of the fluid within the volume and basic geometric data such as volume, flow area, equivalent diameter and elevation. The flow paths connecting the volumes or boundary conditions associated with a volume are designated as junctions. Junctions are described by specifying the flow, flow area, elevation, effective geometric inertia, form loss coefficient and flow equation specification for that particular flow path. Thermal interactions with system metal in the NSSS are modeled with heat conductors. Heat conductors may represent heat transfer from passive sources such as the metal of the reactor coolant system piping or the steam generator tubes. In addition, the internal generation of heat in the core may be represented by active heat conductors designated as powered conductors. Heat conductors are primarily specified by providing the heat transfer area, volume, hydraulic diameter, heated equivalent diameter and channel length of the particular part of the system being

modeled. Temperature dependent material properties (specific heat, thermal conductivity and linear thermal expansion coefficient) are also input. In general, the basic NSSS model is formulated with the code capabilities discussed above. An extensive research effort was conducted to determine the appropriate input required for the models of the Surry and North Anna units. Information was obtained from plant drawings, the Final Safety Analysis Reports, Vepco internal operating documents, equipment technical manuals and specific information requested from the NSSS vendor. Specific control capabilities and constitutive models of system components will be discussed in the following paragraphs.

The Vepco RETRAN Single Loop Models of the North Anna and Surry nuclear units represent the three actual reactor coolant loops as one loop. The resulting geometry is provided in Figure 2-1 and consists of 19 volumes, 29 junctions and 7 heat conductors. While the specific model input for the Surry and North Anna plants is different, the basic model description is the same for the single loop models of both plants. The reactor vessel includes representation of the downcomer, upper and lower plenums, core bypass and reactor core. The steam generator is represented by four volumes on the primary side, one volume on the secondary side and four heat conductors representing the tubes. Single volumes represent the hot leg piping, steam generator inlet plenum, pump suction piping, reactor coolant pump, cold leg piping, pressurizer, and pressurizer surge line. Primary system boundary conditions are specified with junctions representing the pressurizer relief and safety valves. Junctions representing the feedwater inlet, steam outlet, atmospheric steam relief and steam line safety valves provide secondary system

boundary conditions, Specific aspects of the basic model will be discussed below.

All control volumes in the model are homogeneous with the exceptions of volumes 17 and 19, the pressurizer and secondary side of the steam generator, which contain two-phase mixtures. Volumes modeling the loop piping use the RETRAN "temperature transport delay" option to represent fluid temperature change movements in the loop as a front, (that is, fluid entering a pipe does not mix with the fluid present but instead displaces it.) All junctions specify single-stream, compressible flow except for junction 21, the pressurizer surge line connection to the loop, for which incompressible flow with no momentum flux is specified. Extended Henry (subcooled) and Moody (saturated) choking is assigned for all junctions. All junctions use Baroczy two-phase multipliers with Fanning friction to define the wall friction except for the four junctions on the secondary side of the steam generator (volume 19) which use a homogenous two-phase multiplier with Fanning friction. All heat conductors use the Dougall Rohsenow heat transfer correlation to describe post-DNB heat transfer. Tables 2-2 and 2-3 summarize the volume and junction descriptions.

The RETRAN code contains several system component models which are used in the Vepco Single Loop Model. These include pump models which describe the interaction between the centrifugal pump and the primary system fluid. These models calculate pump behavior through the use of empirically developed pump characteristic curves which uniquely define the head and torque response of the pump as functions of volumetric flow

and pump speed. RETRAN includes "built-in" pump characteristics which are representative of pumps supplied by the major reactor coolant pump manufacturers. These curves may be modified, as appropriate, by the user to more realistically represent a specific pump design. Although the built-in data are not appreciably different from Vepco's plant-specific curves, Vepco's Single Loop Models incorporate the specific head versus flow response for first quadrant operation found in the units' UFSARs (Refs. 1 and 2).

The Single Loop Model incorporates the RETRAN pressurizer model which defines two separate thermodynamic regions that are not required to be in thermal equilibrium. A non-equilibrium capability is particularly necessary when the transient involves a surge of subcooled liquid into the pressurizer. In addition, the Single Loop Model represents the effects of subcooled spray, electrical immersion heaters, liquid droplet rainout and vapor rise in the pressurizer.

The reactor systems trip logic is modeled to the detail required for a specific analysis. RETRAN trip functions are used to model 1) protective functions, such as the overtemperature delta-T trip, which result in reactor scram, 2) control system bistable element logic, such as coincidence trips which model "majority" logic and 3) general problem control (e.g., problem termination, etc.)

The protective function trips modeled in the standard Single Loop Model include:

1. High flux
2. Overtemperature delta-T

3. Overpower delta-T
4. Low/high pressurizer pressure
5. High pressurizer level
6. Low coolant flow
7. Loss of power to the reactor coolant pumps

The Single Loop Model also incorporates the RETRAN control system capability to model the following NSSS control and protection features:

1. Overtemperature delta-T setpoint
2. Overpower delta-T setpoint
3. Pressure controller
4. Lead/lag compensation of the low pressure trip signal.

Table 2-4 presents a summary of the trips in the standard Single Loop Models.

The core power response is determined by the point kinetics model in conjunction with explicit reactivity forcing functions and thermal feedback effects from moderator and fuel in the three core regions. The point kinetics model specified for the Single Loop Model incorporates one prompt neutron group, six delayed neutron groups with decay heat represented by 11 gamma emitters, and the important radioactive actinides U-239 and Np-239. Explicit reactivity forcing functions represent scram and reactivity insertion due to control rod withdrawal in the Single Loop Model as the particular analysis requires. Constant temperature coefficients or reactivity tables as a function of temperature (fuel), density (moderator) or power represent feedback effects. Core power is distributed axially among the three core

onductors approximating a symmetric cosine shape. Three core materials regions are used to represent uranium dioxide fuel pellets, the helium filled gap and the Zircaloy cladding. Direct moderator heating is appropriately accounted for in the model. The transient fuel and clad temperatures are calculated based on temperature-dependent thermal properties, which are input in tabular form. Additional details on the point kinetics model as specifically used to analyze the rod ejection transient are provided in Section 2.2.2.

FIGURE 2-1

VEPCO RETRAN. SINGLE LOOP MODEL

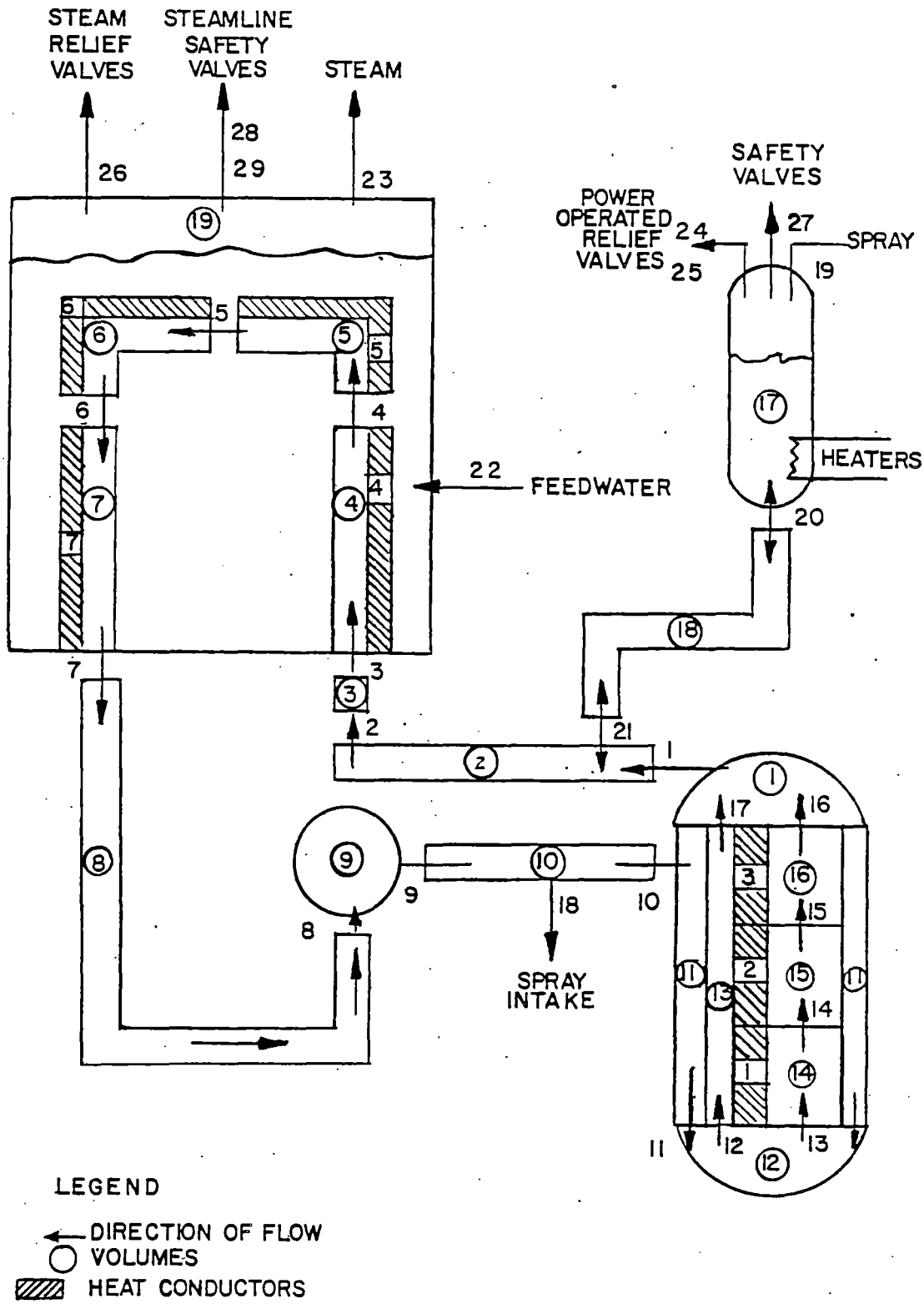


TABLE 2-1

THERMAL-HYDRAULIC DESIGN PARAMETERS

SURRY PLANT:

Total core heat output, Mwt	2441
Heat generated in fuel, %	97.4
System operating pressure, psi	2250
Total coolant flow rate, gpm	265500
Coolant temperatures, °F (@ 100% power)	
Nominal inlet	543
Average rise in core	65.3
Average rise in vessel	62.6
Core average	577.0
Vessel average	574.3
Average linear power density, Kw/ft	6.2

NORTH ANNA PLANT:

Total core heat output, Mwt	2775
Heat generated in fuel, %	97.4
System operating pressure, psi	2250
Total coolant flow rate, gpm	278400
Coolant temperatures, °F (@ 100% power)	
Nominal inlet	549.5
Average rise in core	69.4
Average rise in vessel	66.6
Core average	586.1
Vessel average	582.8
Average linear power density, Kw/ft	5.4

TABLE 2-2

SINGLE LOOP MODEL CONTROL VOLUME DESCRIPTION

Volume ID	Description	Mixture Type	Temperature Transport Delay
1	Vessel upper plenum	H	No
2	Reactor hot leg	H	Yes
3	S/G inlet plenum	H	No
4	S/G tube volume 1	H	No
5	S/G tube volume 2	H	No
6	S/G tube volume 3	H	No
7	S/G tube volume 4	H	No
8	Pump suction piping	H	Yes
9	Reactor coolant pump	H	No
10	Reactor cold leg	H	Yes
11	Downcomer	H	Yes
12	Vessel lower plenum	H	No
13	Core bypass	H	Yes
14	Core section 1	H	No
15	Core section 2	H	No
16	Core section 3	H	No
17	Pressurizer	N	No
18	Pressurizer surge line	H	Yes
19	S/G secondary side	T	No

Abbreviations:
S/G - steam generator
H - homogeneous equilibrium
N - two-phase non-equilibrium
T - two-phase equilibrium

TABLE 2-3

SINGLE LOOP MODEL JUNCTION DESCRIPTION

Junction ID	Description	Type	Two-Phase Fanning Friction Multiplier	Valve Index	H/V
1	Vessel outlet nozzle	Normal	Baroczy	No	V
2	Hot leg outlet	Normal	Baroczy	Yes	H
3	S/G inlet plenum	Normal	Baroczy	No	H
4	S/G tubes	Normal	Baroczy	No	H
5	S/G tubes	Normal	Baroczy	No	V
6	S/G tubes	Normal	Baroczy	No	H
7	S/G-pump suction	Normal	Baroczy	No	H
8	Pump intake	Normal	Baroczy	No	H
9	Pump discharge	Normal	Baroczy	No	V
10	Vessel inlet nozzle	Normal	Baroczy	No	V
11	Downcomer outlet	Normal	Baroczy	No	H
12	Bypass inlet	Normal	Baroczy	No	H
13	Lower plenum - core	Normal	Baroczy	No	H
14	Core internal	Normal	Baroczy	No	H
15	Core internal	Normal	Baroczy	No	H
16	Core - upper plenum	Normal	Baroczy	No	H
17	Bypass outlet	Normal	Baroczy	No	H
18	Cold leg spray intake	Fill	Baroczy	No	V
19	Przr. spray	Spray	Baroczy	No	H
20	Przr. - surge line	Normal	Baroczy	No	H
21	Surge line - hot leg	Normal	Baroczy	No	H
22	Feedwater fill	Fill	Baroczy	No	V
23	S/G outlet	Fill	Homog.	Yes	H
24	PORV 1	Fill	Baroczy	No	H
25	PORV 2	Fill	Baroczy	No	H

TABLE 2-3 (cont.)

SINGLE LOOP MODEL JUNCTION DESCRIPTION

Junction ID	Description	Type	Two-Phase Fanning Friction Multiplier	Valve Index	H/V
26	S/G atm. steam relief	Fill	Homog.	No	H
27	Przr. safety valve	Fill	Baroczy	No	H
28	Steamline safety valve 1	Fill	Homog.	No	H
29	Steamline safety valve 2	Fill	Homog.	No	V

Notes:
All junctions have single-stream compressible flow except junction 21 which is incompressible flow.

Abbreviations:
PORV - power operated relief valve
atm. - atmospheric
S/G - steam generator
Przr. - pressurizer
Homog. - homogeneous
V - vertically distributed junction area
H - horizontally distributed junction area

TABLE 2-4

SINGLE LOOP MODEL TRIP DESCRIPTION

Trip ID	Cause of Trip Activation	Trip Action
1	End of transient time	End calculation
2	High flux (normalized power)	Scram
3	Overtemperature delta-T	Scram
4	Overpower delta-T	Scram
5	High pressurizer pressure	Scram
6	Low pressurizer pressure	Scram
7	High pressurizer level	Scram
8	Low coolant flow	Scram
9	User specified time *	Close loop isolation valves
10	Low backup heater setpoint	Turn pressurizer heaters on
11	High backup heater setpoint	Turn pressurizer heaters off
12	User specified time *	Shut off reactor coolant pumps
13	Transient time = 0 sec	Trip initialization
14	User specified time *	Uncontrolled rod withdrawal
15	User specified time *	Scram
16	High pressurizer pressure	Open PORV # 1
17	Low pressurizer pressure	Close PORV # 1
18	High spray setpoint	Open PORV # 2
19	Low spray setpoint	Close PORV # 2
20	High S/G pressure	Open atm. steam relief valve
21	Low S/G pressure	Close atm. steam relief valve
22	High S/G pressure	Open S/G safety valves
23	Low S/G pressure	Close S/G safety valves
24	High pressurizer pressure	Open pressurizer safety valves
25	Low pressurizer pressure	Close pressurizer safety valves

TABLE 2-4 (cont.)

SINGLE LOOP MODEL TRIP DESCRIPTION

Trip ID	Cause of Trip Activation	Trip Action
26	User specified time *	Turbine trip
27	Low power	End calculation
28	Low-low steam generator mass	Scram
29	Low-low steam generator mass	Auxiliary feedwater on
30	Scram	Turbine trip

Notes:
* Not applicable for most transients.

Abbreviations:
PORV - power operated relief valve
atm. - atmospheric
S/G - steam generator

1.2 RETRAN Hot Spot Model

With a conservative prediction of the core average power history during a postulated rod ejection event obtained from the RETRAN Single Loop Model, a RETRAN Hot Spot Model is used to predict the thermal-hydraulic response of the fuel for the hot spot location of the core. The Hot Spot Model describes a segment of a single fuel rod at the location where the peak core power occurs during the transient. Using the core average power history as a basis for determining a conservative peak power history for the hot spot location, this information is input to the Hot Spot Model as a driving function. The resulting fuel and clad temperatures and enthalpy history for the hot spot as predicted by the model are then used to ensure that the fuel melt and clad embrittlement limits have not been exceeded for the transient.

The Vepco Hot Spot Model consists of 2 control volumes, 2 flow junctions and a single heat conductor. The volumes are stacked one on top of the other as shown in Figure 2-2. The cross sectional area of each volume equals that of a single channel. The lower volume models a one foot high hot channel section of the core surrounding the one foot high heat conductor. Upstream from this hot volume is a one-foot high unheated, time-dependent volume which serves as a reservoir for the flow from the hot channel. The hot channel's volume, flow area and hydraulic diameter are based on the nominal unit cell dimensions for either the Surry or North Anna nuclear cores, depending on which unit is undergoing analysis. Both control volumes are homogenous. Pressure and enthalpy are specified for both volumes for the initial pre-transient core average

onditions for the appropriate unit and initial power level.

Both junctions specify single-stream, compressible flow and use an isoenthalpic model for choking, (should the model predict choked flow at the junction.) The junctions have a horizontally distributed junction area and assume no two phase flow wall friction multipliers. Junction 1, which feeds into the bottom of the hot volume, is a fill junction which supplies flow to the hot volume and has the same flow area. Junction 2, which connects the hot volume to the time-dependent volume, is a normal junction.

Fluid properties tables (dynamic viscosity, specific heat and thermal conductivity) are specified for the coolant based on a pressure of 2250 psia which is the nominal operating pressure for both the Surry and North Anna cores. These tables are used by the control system models to calculate post-DNB heat transfer as discussed below. The fluid properties are derived from Ref. 7. Material properties for the fuel, fuel-clad gap, and Zircaloy clad along with an enthalpy table for the fuel as a function of temperature are built into the model. These property tables are essentially the same as those provided in the RETRAN Single Loop Model with the exception of covering a higher temperature range and allowing for changes which occur upon melting of the fuel. The material properties are derived from Ref. 8.

Three core materials regions are used to represent the fuel pellet, the helium filled gap and the zircaloy cladding. The fuel pellet consists of 10 concentric mesh spacings of equal radial size, the gap of a single mesh spacing and the cladding of 3 mesh spacings of equal size. A

parabolic power distribution is assumed through the pellet (Ref. 2, Section 15.4.6.2.1.2) based on a fuel enrichment which is conservative (high) for the fuel in the reload core loading under analysis.

The heat transfer correlations used for the surface of the heat conductor are modeled with the RETRAN code's control system. Initially the Thom correlation for nucleate boiling (Ref. 4, Vol. 1) is used. However, at 0.1 seconds into the transient, departure from nucleate boiling (DNB) is assumed to occur and a trip control in the model switches the heat transfer correlation to the Bishop-Sandberg-Tong film boiling heat transfer correlation, (Ref. 10.) These two heat transfer correlations are summarized in Table 2-5 while Figures 2-3 and 2-4 present an outline of the control block strategy used to model the correlations. Additional control blocks are used to calculate a fuel average temperature and enthalpy for the hot spot as described in Table 2-6.

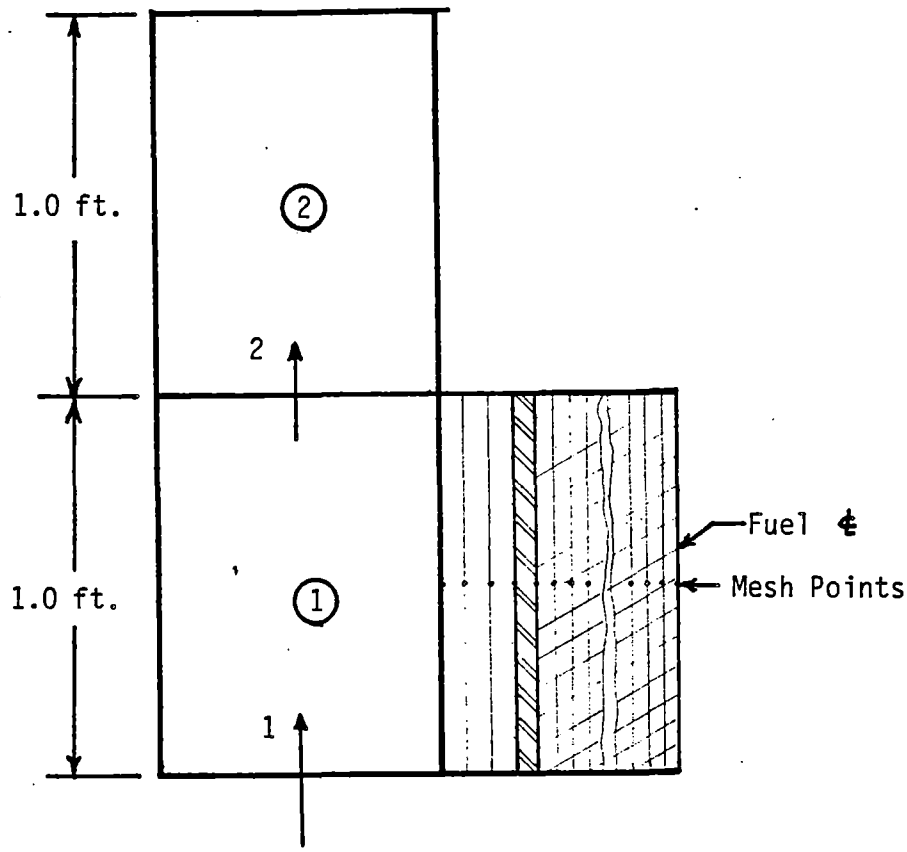
Initially the values of the gap heat transfer coefficient and nucleate boiling heat transfer coefficient are adjusted to yield the desired steady state temperature distribution throughout the fuel pin. At 0.1 seconds into the transient the value of the hot spot gap heat transfer coefficient is changed to a very large value (10000 Btu/ft^2-hr-°F) to model a closure of the gap at the hot spot location, (Ref. 17.)

The RETRAN option to calculate the heat generated from an exothermic Zircaloy-water reaction between the coolant and the cladding is included in the model. Since this reaction only occurs in the presence of steam, the inlet enthalpy of the fill junction emptying into the hot control

olume is deliberately ramped to 1000 Btu/lb over a 0.2 second interval to induce steam quality in the volume. (Additional details of this metal-water reaction option are presented in Vol. 1 of Ref. 4.)

FIGURE 2-2

VEPCO RETRAN HOT SPOT MODEL



Legend: Junction (flow direction shown)

Control Volume

Heat Conductor--Fuel

Gap

Clad

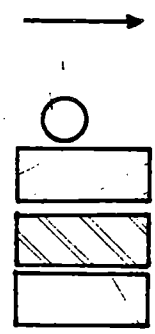


TABLE 2-5

HOT SPOT MODEL HEAT TRANSFER CORRELATIONS

I. Thom Subcooled Boiling Correlation

(Transient time 0.0 to 0.1 seconds.)

$hnb = qw/(Tw - Tc)$

hnb = nucleate boiling heat transfer coefficient, Btu/ft^2-hr-°F

qw = wall heat flux, Btu/ft^2-hr

Tw = wall temperature, °F

where $Tw = Tsat + 0.072*\exp(-P/1260)*qw^{1/2}$

Tsat = coolant saturation temperature, °F

P = coolant pressure, psia

Tc = coolant temperature, °F

TABLE 2-5 (cont.)

HOT SPOT MODEL HEAT TRANSFER CORRELATIONS

II. Bishop-Sandberg-Tong Film Boiling Correlation

(Transient time > 0.1 seconds.)

$$(hD/k)_f = 0.0193(DG/\mu)_f^{0.8} (Cp\mu/k)_f^{1.23} (\rho_g/\rho_b)^{0.68} (\rho_g/\rho_l)^{0.068}$$

h = film boiling heat transfer coefficient, Btu/ft^2-hr-°F

D = equivalent diameter, ft

k = thermal conductivity, Btu/ft-hr-°F

Cp = specific heat, Btu/lb-°F

μ = viscosity, lb/ft-hr

G = mass velocity, lb/ft^2-hr

ρ_g = saturated steam density, lb/ft^3

ρ_l = saturated liquid density, lb/ft^3

ρ_b = bulk density, lb/ft^3

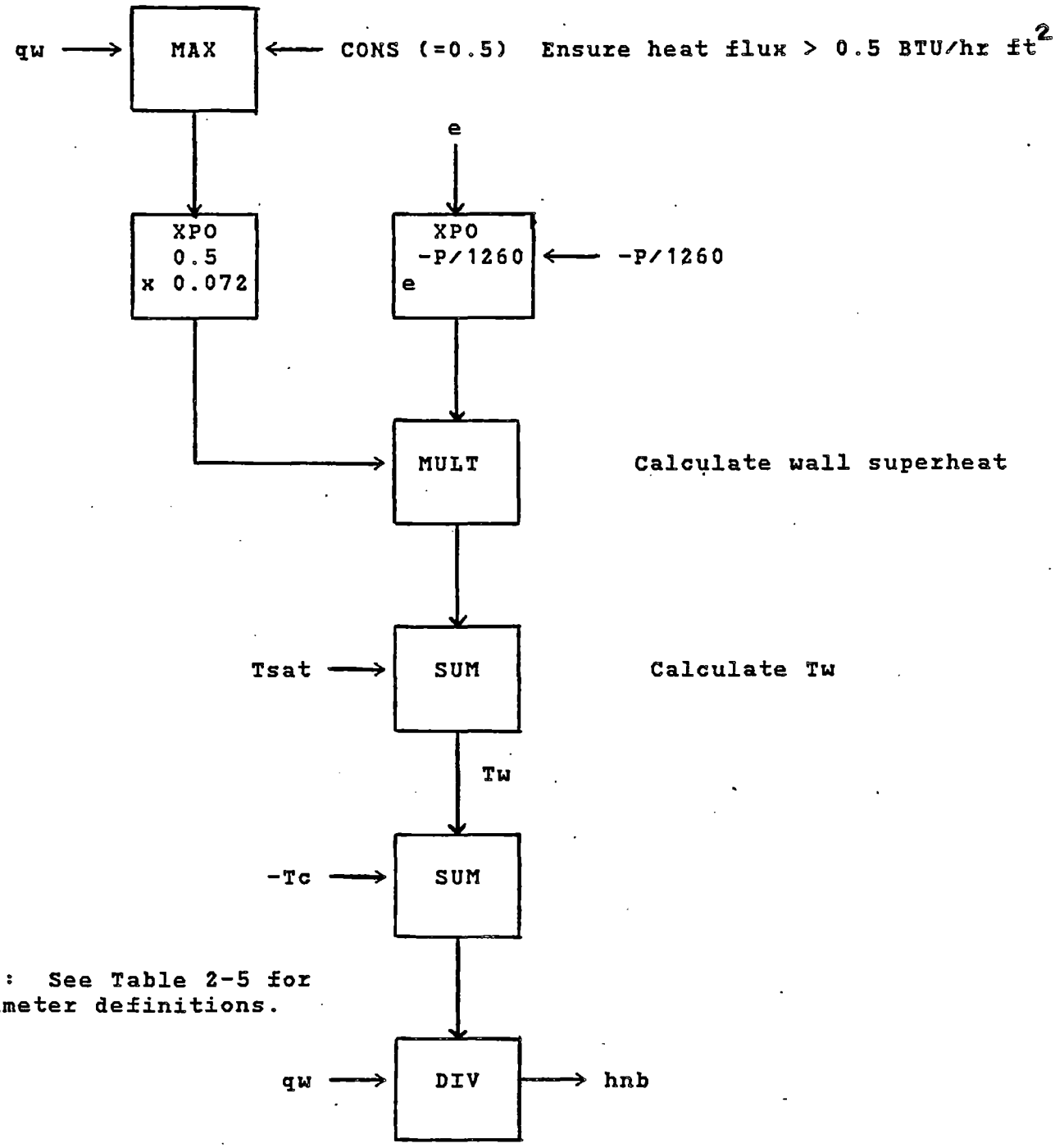
Subscript f refers to properties at the film temperature, Tfilm, which is defined as 0.5(Tw + Tsat) where

Tw = wall temperature, °F

Tsat = coolant saturation temperature, °F

FIGURE 2-3

RETRAN CONTROL MODEL FOR THOM CORRELATION



Note: See Table 2-5 for parameter definitions.

FIGURE 2-4

RETRAN CONTROL MODEL FOR BISHOP-SANDBERG-TONG CORRELATION

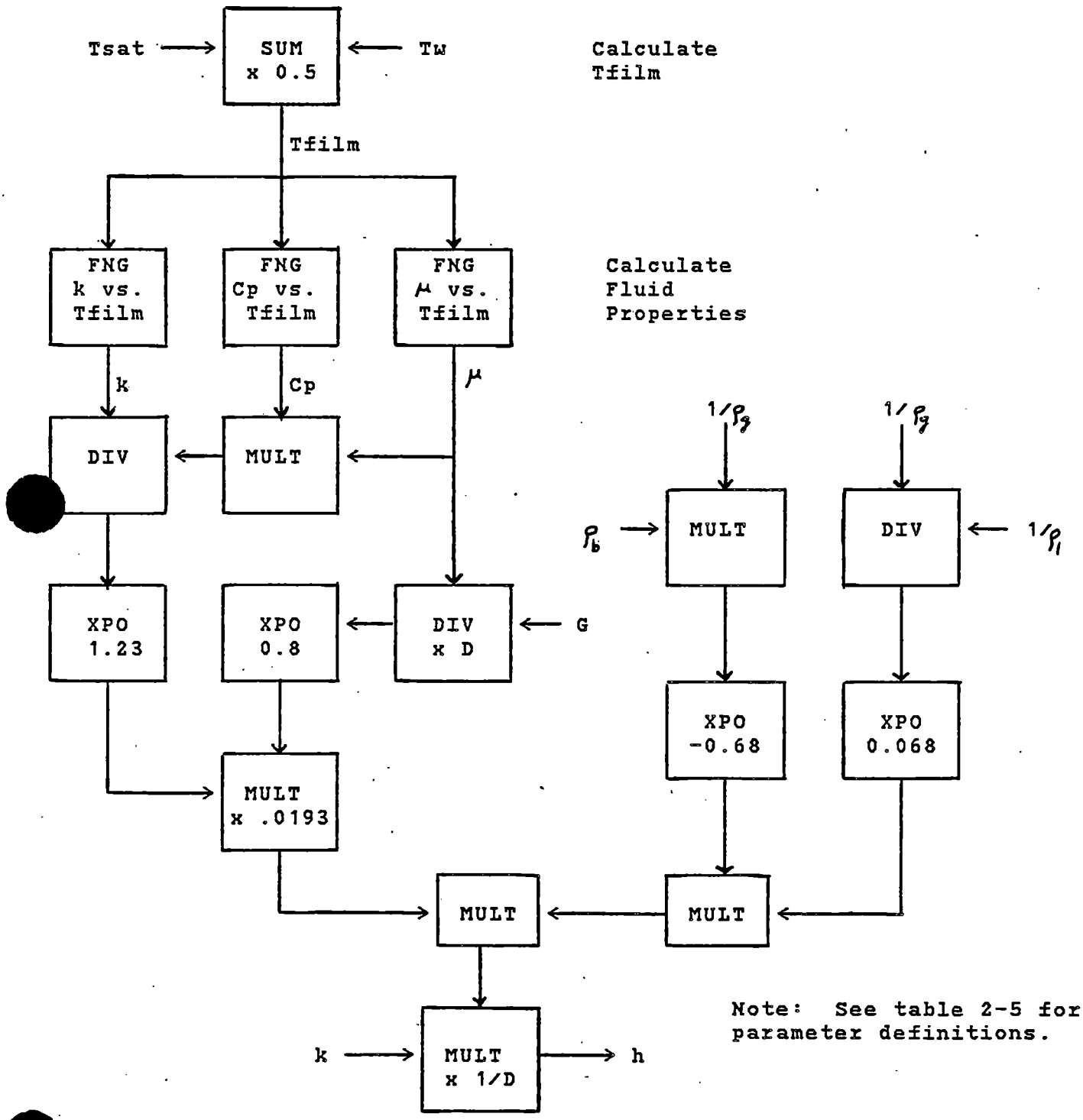


TABLE 2-6

HOT SPOT AVERAGE FUEL TEMPERATURE AND ENTHALPY

I. Average Fuel Temperature

Assume the fuel pellet is modeled in the radial dimension by n equally spaced concentric rings bounded by n+1 mesh points (nodes). Let the temperature at node i be given by Ti, where i = 1,2,...,n+1. (Therefore, T1 is the fuel centerline temperature and Tn+1 is the pellet surface temperature.) The area weighted average fuel temperature T is given by

$$T = (1/n^2)\left[(1/2)(T1) + \sum_{i=2}^{n}(Ti)(2i - 2) + (1/2)(Tn+1)(2n - 1)\right]$$

For the RETRAN Hot Spot Model, n = 10. Therefore,

$$T = (1/100)\left[(1/2)(T1) + \sum_{i=2}^{10}(Ti)(2i - 2) + (19/2)(T11)\right]$$

II. Average Fuel Enthalpy

From a table of enthalpy versus fuel temperature, the value of the fuel enthalpy at node i, hi, is found for each corresponding value of Ti. The same area weighting as given above for T is then used to derive the average fuel enthalpy h. That is,

$$h = (1/100)\left[(1/2)(h1) + \sum_{i=2}^{10}(hi)(2i - 2) + (19/2)(h11)\right]$$

.2 Calculational Technique

The analysis of the rod ejection transient by Vepco may be grouped into three major phases as follows:

1. Generation of physics data from steady state physics calculational methods for use by the RETRAN models,
2. An analysis of the reactor coolant system response to the transient using the RETRAN Single Loop Model with the point kinetics option to predict the core average power history, and
3. A thermal-hydraulic analysis of the core hot spot location using the RETRAN Hot Spot Model.

All three phases incorporate assumptions in their analytical techniques to provide a conservative prediction of the results of a rod ejection transient.

2.2.1 Steady State Physics Analysis

In order to model the core physics conditions for a specific reload design with the RETRAN Single Loop Model, the following core physics parameters are required:

1. ejected rod worth,
2. maximum total power peaking factor (Fq) before and after the rod is ejected,
3. Doppler power defect,
4. delayed neutron fraction,
5. moderator temperature coefficient,
6. total trip worth less the most reactive stuck rod, and
7. prompt neutron generation time.

The first four parameters are the most critical in determining the severity of the transient. The transient is minimally sensitive to the remaining three parameters as will be shown in Section 3.

The maximum total power peaking factor is not used directly in the point kinetics analysis performed with the RETRAN Single Loop Model, but is important in formulating a reasonable description of the Doppler reactivity feedback to be expected during the transient (see Section 2.2.3 below), and is of critical importance in predicting the thermal-hydraulic response of the hot spot.

The use of the term "rod" refers to a single rod cluster control assembly (RCCA); that is, a RCCA consists of the entire cluster of individual rodlets which move as a single unit within a single assembly.

For Westinghouse plants, a rod insertion limit for the various control rod banks is defined as a function of core power level. This limit is set such that operation of the plant with the control banks inserted above their respective limits insures that the core maintains an adequate shutdown margin and acceptable power distribution. A search is performed with the physics codes to determine the location of the highest worth ejected rod with the banks at their appropriate insertion limits for the desired core conditions. For example at hot zero power (HZP) core conditions only the D and C banks will typically be inserted into the core while at hot full power (HFP) only the D bank will be inserted. (See Refs. 1 and 2 for a description of the control rod bank nomenclature and geometry for Vepco's nuclear units.) Rod worths are calculated with frozen feedback as described below.

The parameters mentioned above are calculated for each reload core using Vepco's steady state physics codes as described in Refs. 11, 12 and 13. Due to the low sensitivity of the transient to the prompt neutron lifetime (see 3.2.1.4), a generic value is used in the RETRAN point kinetics analysis although this value is checked with the calculated reload value to assure that no large deviation has occurred. Generic values of the minimum total trip reactivity are also used in the RETRAN point kinetics model, but these values are always compared to the minimum available reload calculated values of the total trip reactivity to ensure that the generic values are conservative.

The remaining parameters when used by the RETRAN models are modified by their appropriate nuclear reliability factors in a direction which is

conservative for the rod ejection transient. For example, the ejected rod worth is increased by 10%, which leads to a higher core reactivity insertion and thus produces a more severe transient than would otherwise be expected. These nuclear reliability factors are documented in Ref. 14.

Additional conservatism is insured by calculating all physics parameters at steady state conditions using the "adiabatic assumption." The adiabatic assumption asserts that any fuel damage which might occur during the transient takes place in a small time interval immediately following the ejection of the rod and before the thermal-hydraulic feedback effects of the core become important. This freezing of the core's feedback leads to larger values of the total power peaking factor and ejected rod worth than would otherwise be expected in the transient. An analytical assessment of this effect is presented in Section 4.4 of this report where a comparison is presented between steady state peaking factors calculated without thermal-hydraulic feedback and transient peaking factors calculated with thermal-hydraulic feedback using a three-dimensional kinetics code.

For each core reload analysis, key steady state physics parameters are used in determining the severity of each transient. Values of these parameters derived for the reload core are compared to those used in a reference safety analysis. When any reload parameter is not bounded by the value used in the reference analysis, the affected transient(s) must be reevaluated based on the new key parameter values.

Additional details on the use of the Vepco physics design codes in

analyzing the safety of a reload cycle and in their application to the rod ejection transient in particular are provided in Ref. 15.

2.2 Core Average Transient Analysis

The core average transient analysis is performed using the RETRAN Single Loop Model with the point kinetics physics option. Core power is distributed among three equal size control volumes stacked axially one upon the other. Each contains a single heat conductor. Core power is distributed axially among the core volumes with an approximate cosine shape; that is, 25% of the power and reactivity feedback effects occur in the upper and lower core volumes while the remaining 50% occurs in the middle core volume. The fuel rod is modeled with two radial concentric fuel regions, a single gap region, and a single clad region.

Initial reactivity insertion due to the ejection of the control rod is implemented by modeling a linearly increasing reactivity insertion into the core from zero to the total worth of the ejected rod over a time interval of 0.1 seconds, (Ref. 17.) The value used for the total integral worth of the ejected rod is the value provided by the physics design codes increased by the Nuclear Reliability Factor of 10% for conservatism.

Negative reactivity insertion from reactor trip is initiated by the high power trips of the RETRAN Single Loop Model. For HFP cases the trip setpoint is 118% of full power (including setpoint and instrument errors) while for HZP cases the trip setpoint is 35% of full power (again including setpoint and instrument errors). A 0.5 second trip delay is assumed between the time the trip setpoint is reached and the time trip reactivity insertion begins, (Refs. 1 and 2). A conservative estimate of the total trip reactivity integral rod worth is provided for

the appropriate core condition. This is typically 4000 pcm for a HFP condition and 2000 pcm for a HZP condition. (A "pcm" is equal to a percent mille of reactivity.) This total integral trip worth is multiplied by a conservative curve of normalized trip worth versus time to provide an estimate of the actual negative reactivity inserted at a particular time following trip, (see Figure 2-5).

Reactivity feedback effects due to a change in the coolant density are modeled by a moderator temperature coefficient. Reactivity feedback effects due to a change in the fuel temperature are modeled by a table of Doppler defect versus fuel temperature. Both the moderator temperature coefficient and Doppler defect values are provided from the physics design codes and are modified by the appropriate Nuclear Reliability Factors in a conservative direction, (i.e., +3 pcm/°F for the moderator temperature coefficient and less 10% for the Doppler defect.) In addition the Doppler defect values are multiplied by a Power Weighting Factor (PWF) to more accurately model the additional feedback effect obtained from actual three-dimensional geometry versus the point kinetics geometry of the RETRAN code. A description of the derivation and use of the PWF is provided in Section 2.2.3 below.

All reactivity parameters input to the RETRAN code are in units of dollars ($). Hence, the values obtained from the physics design codes are converted to dollars by dividing by the value of the delayed neutron fraction. The delayed neutron fraction is provided by the physics design codes for the appropriate core conditions and modified by a Nuclear Reliability Factor in a conservative direction.

The value of the gap heat transfer coefficient (HGAP) used in the fuel pin model is held constant throughout the point kinetics calculation. Since the correlation between core average fuel temperature and power level (necessary to construct a table of Doppler defect values as a function of the fuel temperature) is dependent on HGAP, the problem arises of an appropriate value of HGAP to use. Steady state calculations were performed with the RETRAN Single Loop Model at various core power levels while varying the value of HGAP. Correlations were derived between power level and core average fuel temperatures as a function of HGAP. Heat transfer from the fuel increases or decreases with increasing or decreasing values of HGAP, respectively. Lower values of HGAP result in a higher relative core average fuel temperature for a given power level. Since higher fuel temperatures result in greater negative Doppler reactivity feedback, a high value of HGAP appears to be conservative for the rod ejection transient. This effect is examined in Section 3 where the RETRAN point kinetics model is shown to be only moderately sensitive to the value of HGAP used. A value of 1000 Btu/ft^2-hr-°F is used for HGAP in the point kinetics calculation.

In addition to a prediction of the core average power history for the transient, the RETRAN point kinetics model also calculates the normalized core energy release. This parameter is obtained by integrating the normalized power as a function of time using an integrator control block.

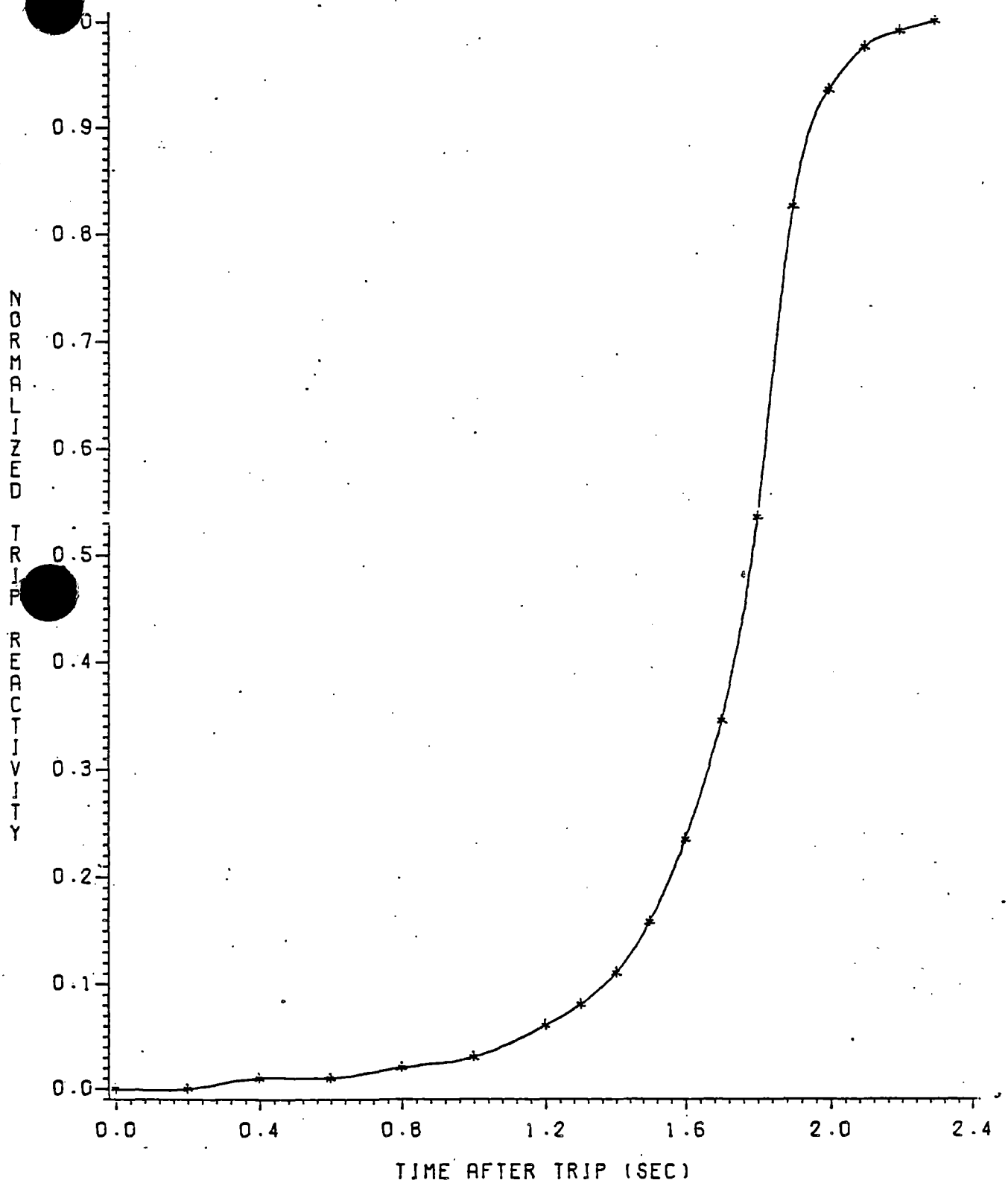
The transient is analyzed at four different core conditions for a given reload cycle. These conditions are:

1. Beginning-of-life (BOL), hot zero power (HZP)
2. Beginning-of-life (BOL), hot full power (HFP)
3. End-of-life (EOL), hot zero power (HZP)
4. End-of-life (EOL), hot full power (HFP)

These conditions are the current licensing basis for the Surry and North Anna nuclear units, (Refs. 1 and 2.)

FIGURE 2-5

NORMALIZED TRIP REACTIVITY CURVE



2.3 Power Weighting Factor

The RETRAN point kinetics model calculates the Doppler reactivity feedback during the transient from a table of negative reactivity insertion versus core average fuel temperature. This table is constructed from the results of Doppler reactivity calculations performed with a steady state diffusion code such as PDQ07. The table provides a reasonable set of values for the core during a pre-transient core condition; that is, with the ejected rod still inserted into the core.

Upon the ejection of the rod, an extreme skewness results in the radial flux and temperature shapes about the position of the ejected rod. This condition violates the basic assumption behind the applicability of the point kinetics approach--i.e., that the spatial flux shape throughout the core does not vary appreciably with time. Use of the typical point kinetics models produces results for this transient that can be very conservative. Accurate modeling of the core's actual reactivity feedback for such a case requires that the spatial kinetics effects, especially in the radial direction be included. However, the spatial kinetics approach has the disadvantage of requiring considerably greater computer resources than the point kinetics approach.

An acceptable solution to simulating the radial kinetics effects with the point kinetics model may be found in the application of a Power Weighting Factor (PWF) to the Doppler reactivity feedback table used by the point kinetics model. To understand the concept behind the use of a PWF, consider the case where one actually knows the Doppler temperature

coefficient for the post-ejected rod core condition. Let this value be denoted as DTC_OUT. This is the value of the Doppler reactivity feedback which should actually be reflected in the point kinetics Doppler reactivity table. Likewise, assume the value of the Doppler temperature coefficient for the core with the ejected rod still inserted, (i.e., the pre-transient value), is known. Let this value be denoted as DTC_IN.

In general, a Doppler reactivity coefficient is the change in core reactivity, DELRHO, resulting from some unit change in the core average fuel temperature, DELTF. That is,

DTC = DELRHO/DELTF

For the case of the ejected rod being fully withdrawn from the core, this equation may be written as:

DTC_OUT = DELRHO_OUT/DELTF_OUT

Likewise, for the pre-transient case, we have:

DTC_IN = DELRHO_IN/DELTF_IN

Since the unit changes in fuel temperature for the two cases are normally equal, (i.e., DELTF_OUT equals DELTF_IN), the ratio of the Doppler temperature coefficients for the two different rodded configuration may be written as:

DTC_OUT/DTC_IN = DELRHO_OUT/DELRHO_IN

The Power Weighting Factor is defined as:

PWF = DELRHO_OUT/DELRHO_IN

where the assumption is made that the change in core average fuel temperatures over which the resulting changes in reactivities are calculated is the same for both cases.

Given a method for calculating DELRHO_OUT and DELRHO_IN, and thereby the PWF, for various rod ejection event conditions, it would be advantageous to find a correlation between the PWF and some other parameter of the rod ejection transient. If such a correlation exists, the value of the PWF for a particular rod ejection case may be determined based on the corresponding value of the parameter to which the PWF is correlated. Since the value of DELRHO_IN, or more correctly, the Doppler reactivity table for the pre-transient condition, is readily available from steady state diffusion code calculations, the value of the PWF may be applied to the table to more correctly predict the true Doppler reactivity feedback to be used by point kinetics in modeling the transient. This approach has the advantage that DELRHO_OUT need not be explicitly calculated for each rod ejection analysis.

Two-group neutron perturbation theory was used to calculate the value of the PWF for various post-ejected rod core conditions for both Surry and North Anna core loadings. Changes in the core average reactivity due to a perturbation of the fuel temperature were calculated for the various radial fuel temperature distributions. Values of nuclear macroscopic cross sections and normal and adjoint flux shapes required for the calculations were provided by radial full core, coarse mesh calculations

erformed by the PD207 One-Zone Model, (Ref. 12.) Details on the use of neutron perturbation theory and its application to calculating point kinetics parameters are provided in Ref. 16.

The resulting values of PWFs were then correlated with the maximum values of the post-ejection radial power peaking factors (Fxy) as calculated with the two-dimensional PD207 Discrete Model, (Ref. 11.) The resulting least squares fit to the points is presented in Figure 2-6. The value of the weighting factor increases with increasing Fxy due to the stronger feedback effect. The value of Fxy is in turn strongly correlated with the worth of the ejected rod. This curve is used to derive the value of the PWF to be used for a specific rod ejection analysis. The Doppler defect curve used in the point kinetics RETRAN odel is then weighted by this value of the PWF to provide a more reasonable estimate of the Doppler reactivity feedback effect during the transient.

Since the value of the post-ejection Fq is routinely provided for the transient, (suitably increased by a Nuclear Reliability Factor,NRF), the value of the post-ejection Fxy used to derive the appropriate PWF to be used is calculated by:

$$Fxy = Fq/(NRF \times 1.55)$$

where NRF removes the additional conservatism of the Nuclear Reliability Factor (1.075 for the Vepco models) and the factor of 1.55 is a generic value of the axial peaking factor for a typical cosine axial power distribution shape.

ome additional points of clarification in the use of the PWF follow.

No benefit is taken for negative Doppler reactivity feedback effects which result from the change in the axial flux shape during the ejection of the rod. Hence, the application of the PWF is conservative from this standpoint.

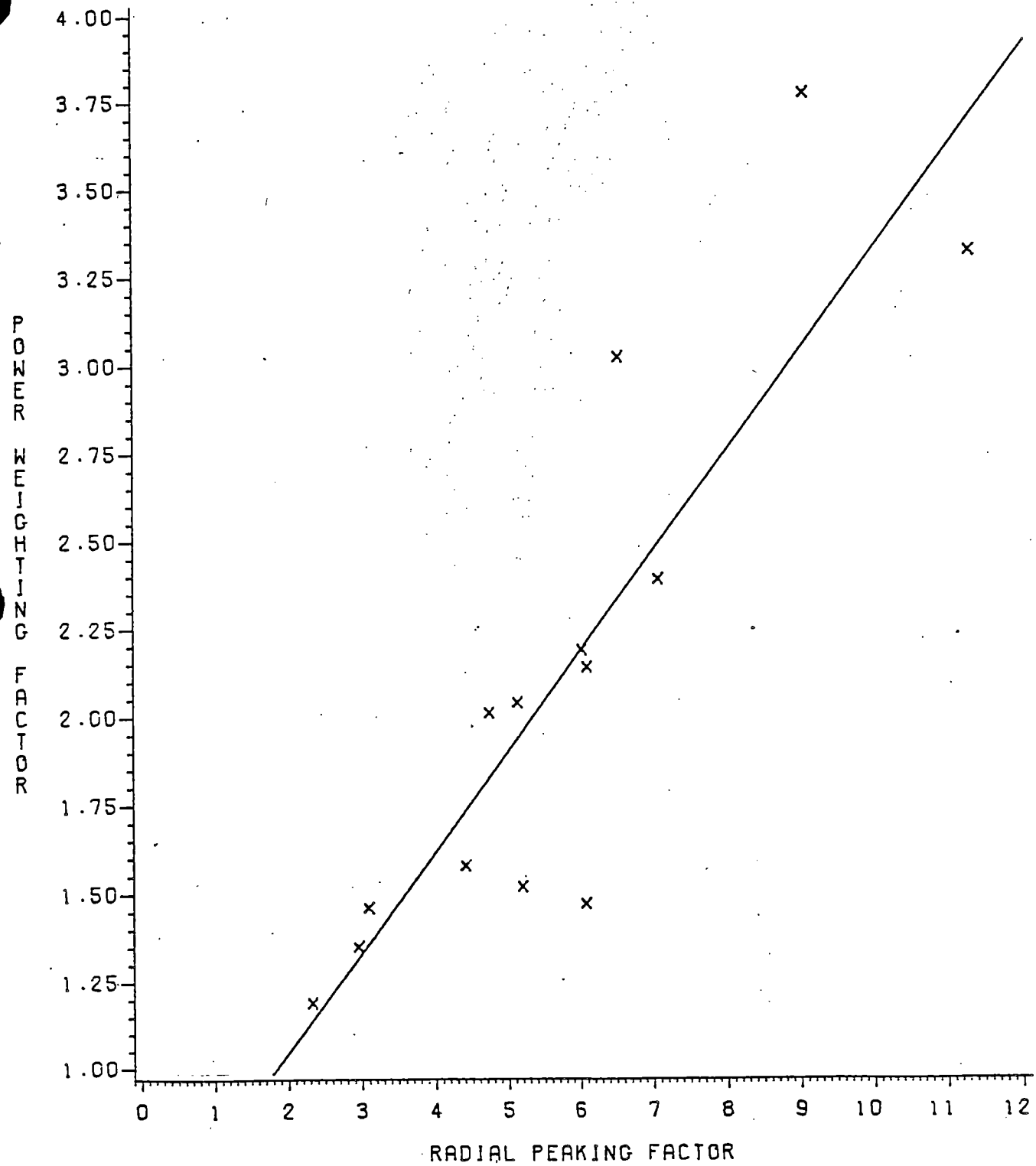
For a HFP case, the worth of the ejected rod is relatively low. Hence, the value of the post-ejection radial peak power is also relatively low and typically yields a value of the PWF near unity. As will be demonstrated in the sensitvity studies of Section 3, such a low value of PWF has little effect on the transient results. Therefore, the PWF is of less importance for a HFP case.

For a HZP case, the importance of the PWF is increased. The relatively high ejected rod worth usually encountered results in a large value of Fxy and hence a PWF value which has a more significant impact on limiting the initial power excursion following the rod ejection.

The application of a PWF derived from perturbation theory to the point kinetics model is an improvement in the prediction of what the actual Doppler reactivity feedback would be for the transient; however, best estimate analysis of the Doppler reactivity feedback is not implied. To demonstrate that the use of the PWF produces a result that is still conservative relative to a more accurate prediction of the transient, comparisons with results from a three-dimensional space-time kinetics model are provided in Section 4.4.2. These comparisons will not only verify the conservatism of the use of the PWF, but of the point kinetics model in general for the rod ejection analysis of Vepco nuclear units.

FIGURE 2-6

POWER WEIGHITNG FACTOR



2.2.4 Hot Spot Transient Analysis

As described in Section 2.1.2, in order to calculate the thermal-hydraulic response of the hot spot core location to the ejection of the rod, the power history of the hot spot is required. Ideally this would be modeled by multiplying the total power peaking factor (Fq) history of the hot spot by the core average power history calculated with the point kinetics RETRAN model.

In place of the hot spot Fq history, the pre- and post-ejection values of Fq as calculated by the physics design codes are available. Since these values have been calculated using the "adiabatic assumption", they will be higher (i.e., conservative) compared to the actual values of Fq expected to occur throughout the initial part of the transient. A conservative hot spot Fq history is constructed by assuming that initially the value of Fq is the steady state pre-ejection value provided by the physics codes. This value is then linearly increased to the post-ejection value of Fq over a time interval of 0.1 seconds and held there for the remainder of the transient. (This is the same time interval assumed for the complete ejection of the rod from the core.) This Fq power history is then multiplied by the core average power history to provide the hot channel power history for the Hot Spot model.

Upon reactor trip, the insertion of the scram banks into the core will cause additional perturbations to the core power distributions both radially and axially. It is possible for the value of Fq at a later time in the transient to actually exceed the post-ejection value of Fq which is assumed to occur at 0.1 seconds into the transient. However, because

of the relatively low value of the core average power at such a time, the resulting hot channel power will be appreciably lower than during the first part of the transient when the core average power is at a maximum. In addition, the location of the hot channel in the core will change during the transient whereas the hot spot analysis assumes a single location throughout the transient. An actual plot of the hot channel power during the transient as predicted by a three-dimensional space-time kinetics model is provided in Section 4.4.2 and shows the power in the hot channel to be steadily decreasing following the early peak due to the ejection of the rod.

In summary, the power history driving function input to the RETRAN Hot Spot Model consists of two components: a conservatively predicted core average power history and a conservative total peaking factor power history.

At 0.1 seconds into the transient, the hot channel is forced into DNB by switching from a subcooled nucleate boiling surface heat transfer coefficient, (Thom correlation, Ref 4. Vol. 1), to a film boiling surface heat transfer coefficient (Bishop-Sandberg-Tong correlation, Ref. 10.) Conservatisms applied to the latter correlation are the use of a safety factor to reduce the calculated heat flux and the assumption of a constant bulk coolant density.

The large increase in core power level is expected to lead to a rapid expansion of the fuel pellet. This reduces the pellet-clad gap size, resulting in an increase in the gap heat transfer coefficent. At 0.1 seconds into the transient, the value of HGAP in the Hot Spot Model is

increased from its initial value to a relatively high value of 10000 Btu/ft^2-hr-°F to reflect this closure of the gap.

From the hot channel calculation, values of the fuel temperatures at the boundaries of each of the 10 fuel pellet regions, the clad temperatures at the inner and outer clad surfaces and the pellet average enthalpy are obtained as a function of time. These parameters are used to assess the fuel response against the acceptance criteria presented in Section 1.3.

Typically, if the maximum amount of fuel melt at the hot spot is less than 10%, the pellet would be expected to maintain its configuration and, therefore, no unacceptable radiological consequences or core damage are expected to occur. An upper bound on the percentage of fuel melt is derived from the ratio of the cross sectional area of the portion of the pellet where melting occurs to the total pellet cross sectional area. The cross sectional area for melting is determined by noting those radial concentric fuel nodes whose temperature at some point in the transient exceeds the assumed temperature for fuel melt. At BOL case this melting temperature is assumed to be 4900 °F while for an EOL case a temperature of 4800 °F is assumed. Table 2-7 presents the table used to compute the maximum fraction of fuel melt for the RETRAN Hot Spot Model.

TABLE 2-7

HOT CHANNEL FUEL MELT FRACTION TABLE

Highest Numbered Node n for which Tn < Tmelt	Maximum melt fraction (%)
1	0
2	1
3	4
4	9
5	16

Notes:

Tn = fuel temperature of node n

Tmelt = fuel melting temperature
= 4900 °F for BOL case
= 4800 °F for EOL case

RETRAN Hot Spot Model contains 11 fuel nodes, (10 concentric fuel rings.)

4.2.5 System Overpressure Analysis

Included in the acceptance criteria is the limitation that the maximum reactor pressure during the transient will be less than the value that will cause the stress to exceed conservatively defined stress limits. This problem is resolved by a pressure stress calculation and has been addressed generically for the North Anna and Surry nuclear units as described in Refs. 1, 2 and 17. Since it was concluded that no violations will occur for Vepco nuclear units due to the rod ejection transient, the system overpressure analysis is not performed as part of the Vepco methodology.

2.6 Radiological Concerns

As with the case of the system overpressure analysis, the radiological concerns for the rod ejection transient for the Surry and North Anna nuclear units have been addressed generically, (Refs. 1, 2 and 17.) In assessing the fission product release, it is assumed that all of the rods which experience DNB release their entire gap inventory of fission products to the coolant. Vepco's additional acceptance criteria that the temperature for clad embrittlement will not be exceeded and that the fuel pellet configuration will be maintained guarantees that the condition of the fuel rod losing its integrity from entering DNB is a very conservative assumption.

The nature of the rod ejection event causes a large hot channel factor to occur only in a very localized region of the core, i.e., in the immediate vicinity of the ejected rod. Fuel census analysis as provided in Ref. 17 showed that for the worst case investigated, the average fuel rod failed to reach DNB, and even for those fuel rods reaching DNB no excessive release of fission products was to be expected.

In summary, the localized nature of the event coupled with the small number of rods expected to reach DNB and the large conservatisms inherent in the analysis ensures that meeting the Vepco imposed limits of allowable fuel melt and clad embrittlement temperature for the hot channel assures that the radiological limits for the event as specified in Regulatory Guide 1,77 (Ref. 3) will be met.

SECTION 3 - SENSITIVITY STUDIES

3.1 Introduction

A sensitivity study was performed for both the RETRAN Single Loop Model and RETRAN Hot Spot Model to quantify the impact of uncertainties in core parameters and modeling assumptions on the models' predictions for the rod ejection transient. This section provides a summary of the study's results.

The study is divided in three parts: (1) neutronics parameter sensitivities for the point kinetics calculation, (2) thermal hydraulic sensitivities for the point kinetics calculation, and (3) thermal hydraulic sensitivities for the hot spot calculation. The first two parts were performed with the RETRAN Single Loop Model for a Surry plant. For these two parts, the four bounding cycle conditions were analyzed for each sensitivity, (i.e., BOL HZP, BOL HFP, EOL HZP and EOL HFP.) The sensitivities for the hot spot calculation were analyzed with the RETRAN Hot Spot Model for typical HZP and HFP hot spot power histories only at BOL, since for this calculation, with the exception of the assumed temperature for fuel melt, the cycle lifetime is not explicitly reflected in the input.

3.2 Sensitivity Study Results

3.2.1 Point Kinetics Neutronics Parameters

Sensitivity studies were performed for the point kinetics calculation to assess the impact of various neutronics parameters on the predicted core average power history. For the rod ejection event, the core average power history may be divided for sensitivity analysis purposes into two major time domains. The first of these domains runs from the beginning of the transient until approximately 0.5 seconds into the transient. During this time the core average power rises rapidly, obtains its maximum value, and is turned around by the negative reactivity insertion due to Doppler feedback. The transient results are particularly sensitive to the core average power histories in this first domain, as a significant increase in the power value will lead to higher transient hot spot temperatures and thus produce more severe transient results.

The second domain, starting at approximately 0.5 seconds into the transient, models the reactor shutdown with the insertion of the control rod banks and the effects of moderator reactivity feedback. During this time interval, the core average power level is significantly lower than during the first domain.

Table 3-1 presents a summary of the results of the sensitivity study for the neutronics parameters for the four different cycle conditions. The peak normalized core power level (where unity is equal to full power) is presented for each case along with the total energy released during the first five seconds of the transient. This latter quantity is of

importance for HZP cases in demonstrating the sensitivity of a parameter. This is due to the narrowness of the initial core average power peak which occurs during the first time domain for HZP cases. What is in reality a small impact on the transient results may show a difference in the peak normalized power between the nominal and sensitivity cases of several full power levels. Hence, the total energy release (which is the integral of the core power history) at some later point in the transient is used as a more representative quantity for demonstrating the magnitude of the sensitivity for HZP cases. The total energy release is also useful in assessing the sensitivity of a parameter whose effects are not noticable until the second time domain, (e.g., trip worth.)

The first entry in the table for each cycle condition is for the nominal case with no perturbation to any of the neutronics parameters. Comparison of the sensitivity values for the peak normalized power and energy release for each case with those of the nominal case provides a concise summary of the relative sensitivity of each case.

3.2.1.1 Doppler Reactivity Feedback

The first sensitivity investigated is that of the Doppler reactivity feedback. This is modeled in the RETRAN point kinetics calculation through the input of a table of Doppler defect as a function of fuel temperature. Since the Doppler defect values in the table are multiplied by an appropriate Power Weighting Factor (PWF), comparisons are shown between the PWF used for the nominal case and a PWF of unity.

For the two HZP cases, the necessity of the use of the PWF becomes obvious. Without the PWF the total energy release is tripled for the BOL case and nearly doubled for the EOL case. Figure 3-1 presents plots of core average power histories for the two HZP cases with and without the PWF applied. Corresponding plots of the total energy releases are presented in Figure 3-2. For comparison, additional curves are plotted with the Doppler reactivity feedback curve used in the nominal cases decreased by 10% (typical of the uncertainty assumed in a Doppler feedback calculation by steady state physics models.) The 10% uncertainty in the Doppler defect shows little sensitivity compared to that resulting from deletion of the PWF.

For the HFP cases as summarized in Table 3-1, reducing the PWF to unity, although it increases the peak normalized power and total energy release, shows a much lower sensitivity than the HZP cases. This is as would be expected, since for a HFP case the ejected rod worth, and, therefore, the resulting power peaking and flux redistribution, are smaller than for a HZP case. Hence, PWF values which are near unity have little impact on the transient predictions. Figure 3-3 presents the normalized power history comparisons for the two HFP cases.

3.2.1.2 Moderator Reactivity Feedback

Moderator reactivity feedback effects in the RETRAN point kinetics calculation are modeled by inputing a moderator temperature coefficient (MTC). The assumed uncertainty in the value of the MTC calculated with Yepco steady state physics codes is presently +3 pcm/°F. The value for

The MTC input to the nominal cases was increased by this uncertainty for the sensitivity study. The result was a minor increase in the core average power prediction as would be expected for the insertion of an additional positive reactivity feedback effect. Figure 3-4 presents a comparison of the normalized power histories for the two BOL cases. The effect of the MTC is shown to be most prominent early on in the transient where the greatest change in core power level, and therefore, the greatest change in moderator temperature, occurs. A lag occurs between the time of peak power and the onset of the effect of the moderator feedback since the moderator temperature responds more slowly to changes in core power than the fuel temperature.

3.2.1.3 Delayed Neutron Fraction

The delayed neutron fraction is used in estimating the dollar worth of the reactivity effects in the solution of the point kinetics equation. The value of the delayed neutron fraction decreases with core burnup.

Decreasing the delayed neutron fraction value by 5%, (the assumed design uncertainty), represents a reduction in the percentage of delayed neutrons in the core and thereby implies a core with faster response to changing conditions. In addition, the dollar worth of the two dominant reactivity effects for the transient, the ejected rod worth and the Doppler reactivity feedback, are both modified by perturbing the value of the delayed neutron fraction. In effect, reducing the value of the delayed neutron fraction by 5% increases the dollar worth of both parameters by 5%. As described in Section 3.2.1.1 above, the transient

shows little sensitivity to an increase in the Doppler reactivity feedback of the magnitude of 5%. However, as documented in Section 3.2.1.5 below, a 5% increase in the ejected rod worth is significant. The ejected rod worth is therefore the stronger of the two competing reactivity effects. This results in a higher predicted core average power with decreasing delayed neutron fraction. As is demonstrated in a comparison of BOL and EOL cases with similar core parameters, the EOL cases with significantly lower delayed neutron fractions show higher predicted core average powers. The same effect appears in the sensitivity results which show slightly higher predicted powers than the nominal cases, (Table 3-1). Figure 3-5 presents a comparison of the power histories for the two BOL cases.

3.2.1.4 Prompt Neutron Generation Time

An increase in the mean value of the prompt neutron lifetime is expected to slow the rate of initial increase in core average power during the transient since, on the average, each prompt neutron will now survive longer in the core before it is absorbed. As shown in Table 3-1, increasing the value of this parameter by 5E-6 seconds (from a nominal value of 18E-6 seconds) typically decreases the predicted peak normalized power although the total energy release shows little sensitivity.

Likewise, a decrease in the mean value of the prompt neutron lifetime is expected to hasten the rate of initial increase in the core average power during the transient, with each neutron being absorbed sooner. As

shown in Table 3-1, decreasing the prompt neutron lifetime by 5E-6 (from a nominal value of 18E-6 seconds) increases the predicted peak normalized power while the total energy release again shows little sensitivity. The effects of these changes is more pronounced for the HZP cases than the HFP cases.

Since the assumed design uncertainty in this parameter is only 5%, the transient can be expected to show little sensitivity to even large variations in the value used by the point kinetics model. Therefore, a generic value of 18E-6 seconds is used for all analyses.

3.2.1.5 Ejected Rod Worth

The assumed design uncertainty for rod worth is 10%. Increasing the ejected rod worth by 10% shows the greatest impact on the power history of all the sensitivities investigated. As expected, the increased reactivity insertion due to the withdrawal of a greater rod worth from the core results in a higher peak normalized power. Figure 3-6 presents the power history comparisons for the BOL cases.

3.2.1.6 Rod Ejection Time

In the nominal cases, the rod is ejected by modeling a linear reactivity insertion into the core over a time interval of 0.1 seconds. Two sensitivity cases were investigated for both a shorter rod ejection time (0.05 seconds) and a longer rod ejection time (0.2 seconds.) Little impact resulted from either sensitivity. Typically, for the shorter rod ejection time, the peak core average power occurred earlier in the

transient (see Figure 3-7 for a BOL comparison), while for the longer rod ejection time the peak power occurred correspondingly later in the transient, (Figure 3-8). The lower peak powers resulting for the longer transient time for the HFP cases is a result of the negative feedback effects having more time to mitigate the effects of the positive reactivity insertion from the ejection of the rod. No significant impact on the integrated energy release was observed.

3.2.1.7 Trip Delay Time

A trip delay time of 0.5 seconds is assumed in the nominal cases. Sensitivity studies were performed by increasing this value to 1.5 seconds. Since the peak normalized power for the rod ejection transient typically occurs before the scram banks begin movement, increasing the trip delay time has no impact on the peak normalized power as shown in Table 3-1. However, the delay in negative reactivity insertion slows the rate of decrease in the core power level leading to a greater energy release throughout most of the period through which the scram banks are inserting. Figure 3-9 presents the BOL power history comparisons for this sensitivity.

3.2.1.8 Trip Worth

As with the ejected rod worth, the assumed design uncertainty of 10% was applied to the trip worth for the sensitivity study. In this case, the worth of the trip banks were decreased by 10% in order to cause an increase in the core power level. As with the "trip delay time"

sensitivity study, the power history predictions of the sensitivity cases were identical to those of the nominal cases during the first time domain. As shown in Table 3-1, the sensitivity of changing the trip worth is minor. Figure 3-10 presents comparisons of the power histories for the BOL cases.

3.2.1.9 Initial Zero Power Level

For the nominal zero power cases, an initial normalized core power level of 1.0E-9 was assumed. This power level was increased to a value of 1.0E-3 (i.e., 0.1% full power) for the sensitivity study. The transient showed little sensitivity to this change as shown in Table 3-1. Figure 3-11 presents the power history comparisons for the two HZP cases. For the sensitivity case, the peak normalized power is reached earlier than for the nominal case. This would be as expected since the sensitivity case is initially starting at a higher power level.

3.2.1.10 Time Step Size

The time step sizes used in obtaining the time-dependent numerical solutions to the equations in the RETRAN model are chosen according to the expected rate of change of core conditions during the transient. Thus, early in the transient where the most rapid changes in core power and temperatures are taking place, the time step size is chosen relatively small compared to later in the transient where the rate of change of core parameters is slower. Ideally, the smaller the time step size, the greater the confidence in the accuracy of the solution.

The sensitivity of the time step size was investigated by decreasing it by a factor of approximately 0.2 over the nominal value used throughout the transient. The results as presented in Table 3-1 show little sensitivity to this magnitude of reduction.

3.2.1.11 Beta Yield Fractions

The RETRAN point kinetics model has a built-in standard set of beta yield fractions for the reactor kinetics calculations. The sensitivity of this parameter was checked by inputting a set of beta yield fractions representative of Surry reloads in the RETRAN model. The results, as shown in Table 3-1, show little sensitivity. Figure 3-12 presents a comparison of the power histories for the BOL cases.

TABLE 3-1

POINT KINETICS NEUTRONICS SENSITIVITY STUDY

I. BOL HZP Studies:

Parameter	Sensitivity	Peak Norm. Power	Energy Release (t=5 sec)
Nominal case values	Not applicable	51.1	2.23
Power weighting factor (PWF)	Changed from 2.4 to 1.0	127.	6.69
Doppler reactivity feedback	Decreased 10% (PWF=2.4)	56.5	2.52
Moderator reactivity feedback	Increased +3 pcm/°F	51.6	2.34
Delayed neutron fraction	Decreased 5%	58.8	2.30
Prompt neutron generation time	Increased 5.0E-6 sec	41.5	2.23
Prompt neutron generation time	Decreased 5.0E-6 sec	73.5	2.24
Ejected rod worth	Increased 10%	81.8	2.61
Rod ejection time	Changed to 0.05 sec	52.5	2.24
Rod ejection time	Changed to 0.2 sec	53.2	2.23
Trip delay time	Increased to 1.5 sec	51.1	2.59
Trip worth	Decreased 10%	51.1	2.26
Initial zero power level	Changed to 1.0E-3	52.2	2.24
Time step size	Multiplied by 0.2	49.9	2.22
Beta yield fractions	Surry Reload Values	51.3	2.28

Notes:

Peak Norm. Power = peak normalized power occurring during transient, (1.0 is equivalent to full power level.)

Energy Release (t=5 sec) = total core energy release up to a transient time of 5.0 seconds, (units of full-power-seconds.)

TABLE 3-1 (cont.)

POINT KINETICS NEUTRONICS SENSITIVITY STUDY

II. BOL HFP Studies:

Parameter	Sensitivity	Peak Norm. Power	Energy Release (t=5 sec)
Nominal case values	Not applicable	2.02	3.50
Power weighting factor	Changed from 1.2 to 1.0	2.04	3.63
Moderator reactivity feedback	Increased +3 pcm/°F	2.02	3.58
Delayed neutron fraction	Decreased 5%	2.13	3.52
Prompt neutron generation time	Increased 5.0E-6 sec	2.02	3.50
Prompt neutron generation time	Decreased 5.0E-6 sec	2.02	3.50
Ejected rod worth	Increased 10%	2.25	3.67
Rod ejection time	Changed to 0.05 sec	2.04	3.49
Rod ejection time	Changed to 0.2 sec	1.94	3.52
Trip delay time	Increased to 1.5 sec	2.02	4.80
Trip worth	Decreased 10%	2.02	3.57
Time step size	Multiplied by 0.2	2.02	3.49
Beta yield fractions	Surry Reload Values	2.03	3.51

Notes:

Peak Norm. Power = peak normalized power occurring during transient, (1.0 is equivalent to full power level.)

Energy Release (t=5 sec) = total core energy release up to a transient time of 5.0 seconds, (units of full-power-seconds.)

TABLE 3-1 (cont.)

POINT KINETICS NEUTRONICS SENSITIVITY STUDY

III. EOL HZP Studies:

Parameter	Sensitivity	Peak Norm. Power	Energy Release (t=5 sec)
Nominal case values	Not applicable	80.2	1.87
Power weighting factor (PWF)	Changed from 2.3 to 1.0	139.	3.65
Doppler reactivity feedback	Decreased 10% (PWF=2.3)	87.2	2.02
Moderator reactivity feedback	Increased +3 pcm/°F	81.2	1.91
Delayed neutron fraction	Decreased 5%	87.0	1.91
Prompt neutron generation time	Increased 5.0E-6 sec	62.7	1.87
Prompt neutron generation time	Decreased 5.0E-6 sec	110.4	1.88
Ejected rod worth	Increased 10%	116.	2.14
Rod ejection time	Changed to 0.05 sec	78.4	1.87
Rod ejection time	Changed to 0.2 sec	80.1	1.87
Trip delay time	Increased to 1.5 sec	80.2	2.06
Trip worth	Decreased 10%	80.2	1.89
Initial zero power level	Changed to 1.0E-3	75.9	1.88
Time step size	Multiplied by 0.2	77.9	1.86
Beta yield fractions	Surry Reload Values	80.3	1.89

Notes:

Peak Norm. Power = peak normalized power occurring during transient, (1.0 is equivalent to full power level.)

Energy Release (t=5 sec) = total core energy release up to a transient time of 5.0 seconds, (units of full-power-seconds.)

TABLE 3-1 (cont.)

POINT KINETICS NEUTRONICS SENSITIVITY STUDY

IV. EOL HFP Studies:

Parameter	Sensitivity	Peak Norm. Power	Energy Release (t=5 sec)
Nominal case values	Not applicable	2.60	3.02
Power weighting factor	Changed from 1.2 to 1.0	2.63	3.05
Moderator reactivity feedback	Increased +3 pcm/°F	2.60	3.03
Delayed neutron fraction	Decreased 5%	2.79	2.99
Prompt neutron generation time	Increased 5.0E-6 sec	2.57	3.02
Prompt neutron generation time	Decreased 5.0E-6 sec	2.61	3.01
Ejected rod worth	Increased 10%	3.04	3.10
Rod ejection time	Changed to 0.05 sec	2.67	3.00
Rod ejection time	Changed to 0.2 sec	2.43	3.03
Trip delay time	Increased to 1.5 sec	2.60	4.06
Trip worth	Decreased 10%	2.60	3.07
Time step size	Multiplied by 0.2	2.59	3.00
Beta yield fractions	Surry Reload Values	2.60	3.01

Notes:

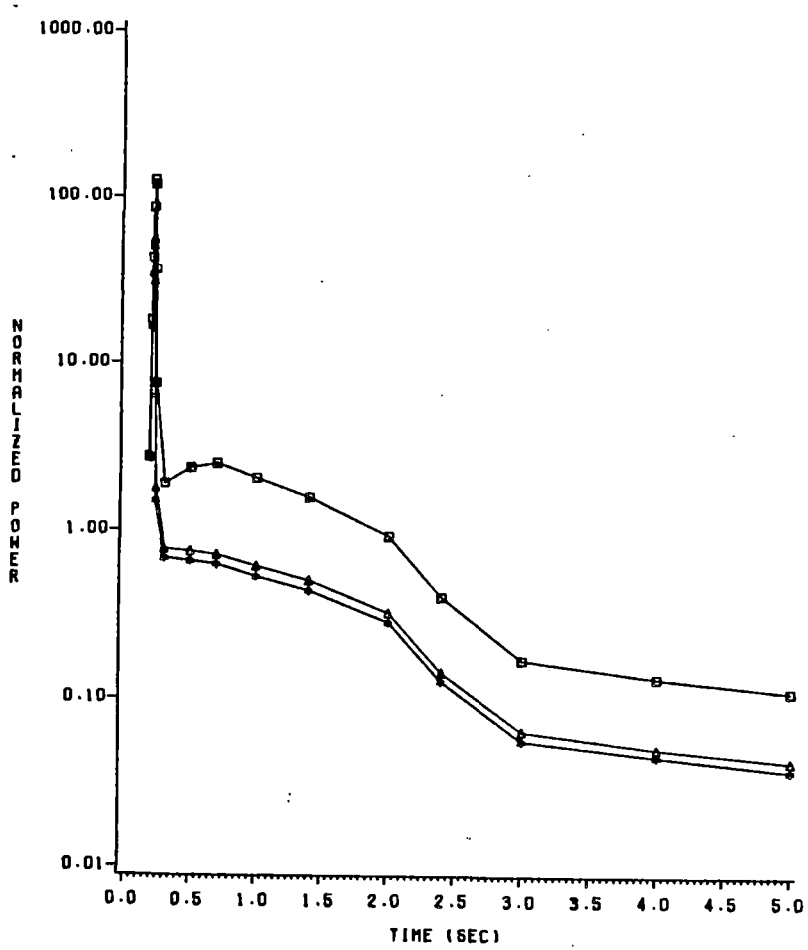
Peak Norm. Power = peak normalized power occurring during transient, (1.0 is equivalent to full power level.)

Energy Release (t=5 sec) = total core energy release up to a transient time of 5.0 seconds, (units of full-power-seconds.)

FIGURE 3-1

SENSITIVITY STUDY - HZP DOPPLER REACTIVITY FEEDBACK (ENERGY RELEASE)

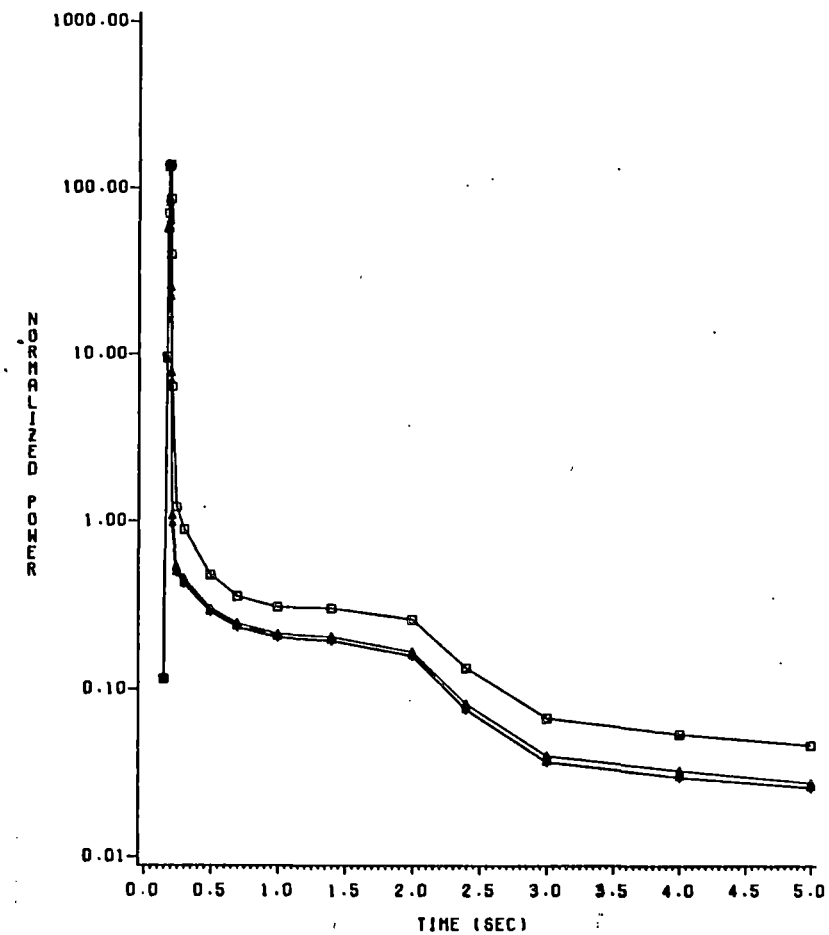
BEGINNING OF LIFE



STAR: NOMINAL CASE (PWF=2.4)
SQUARE: SENSITIVITY CASE (PWF=1.0)
TRIANGLE: NOMINAL CASE (LESS 10%)

FIGURE 3-1A

END OF LIFE



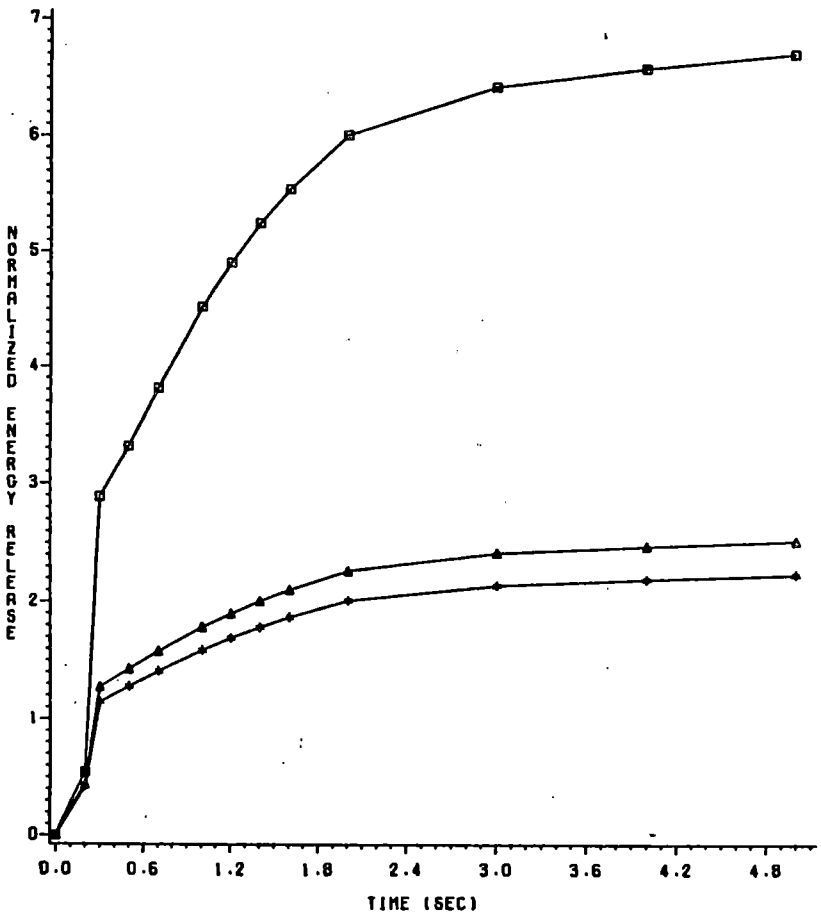
STAR: NOMINAL CASE (PWF=2.3)
SQUARE: SENSITIVITY CASE (PWF=1.0)
TRIANGLE: NOMINAL CASE (LESS 10%)

FIGURE 3-1B

FIGURE 3-2

SENSITIVITY STUDY - HZP DOPPLER REACTIVITY FEEDBACK (ENERGY RELEASE)

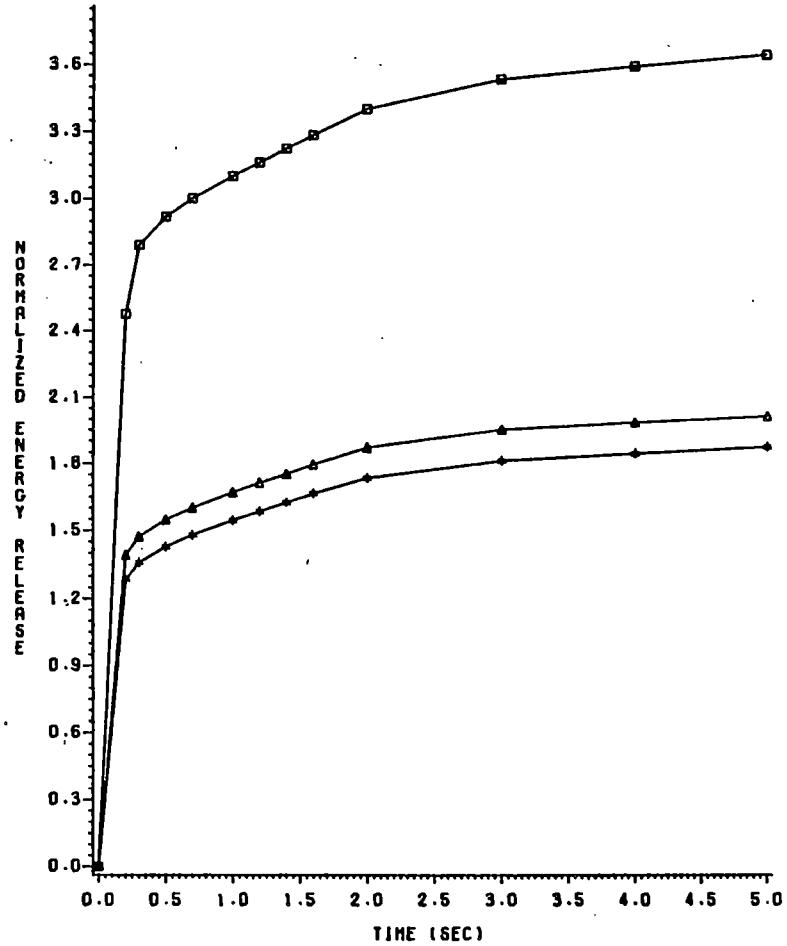
BEGINNING OF LIFE



STAR: NOMINAL CASE (PWF=2.4)
SQUARE: SENSITIVITY CASE (PWF=1.0)
TRIANGLE: NOMINAL CASE (LESS 10%)

FIGURE 3-2A

END OF LIFE



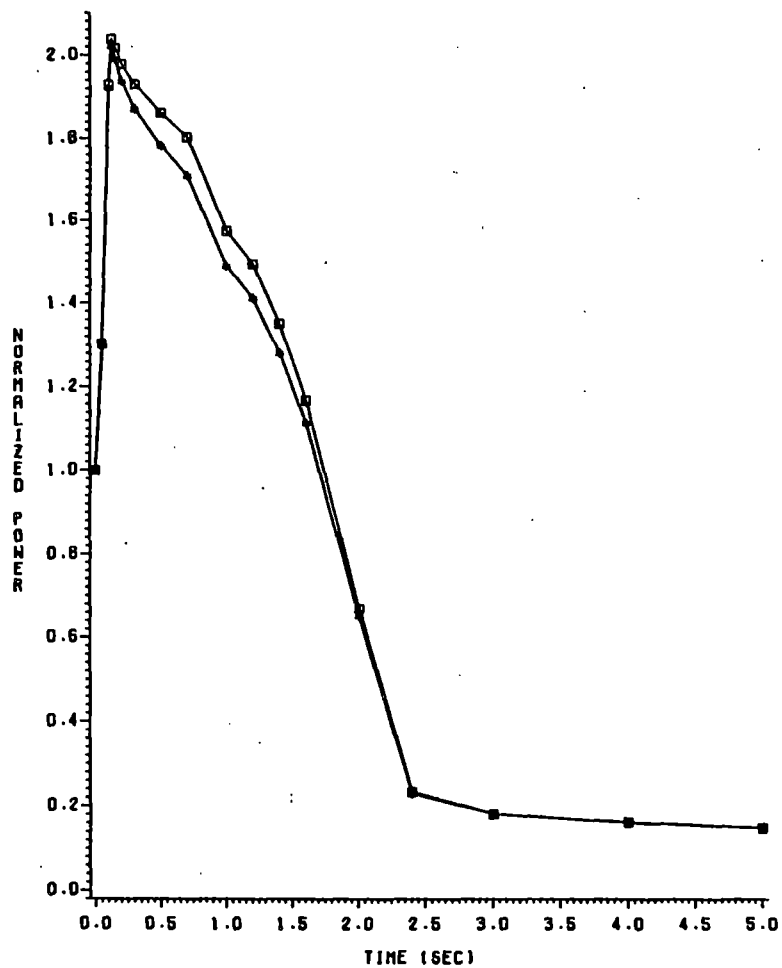
STAR: NOMINAL CASE (PWF=2.3)
SQUARE: SENSITIVITY CASE (PWF=1.0)
TRIANGLE: NOMINAL CASE (LESS 10%)

FIGURE 3-2B

FIGURE 3-3

SENSITIVITY STUDY - HFP DOPPLER REACTIVITY FEEDBACK

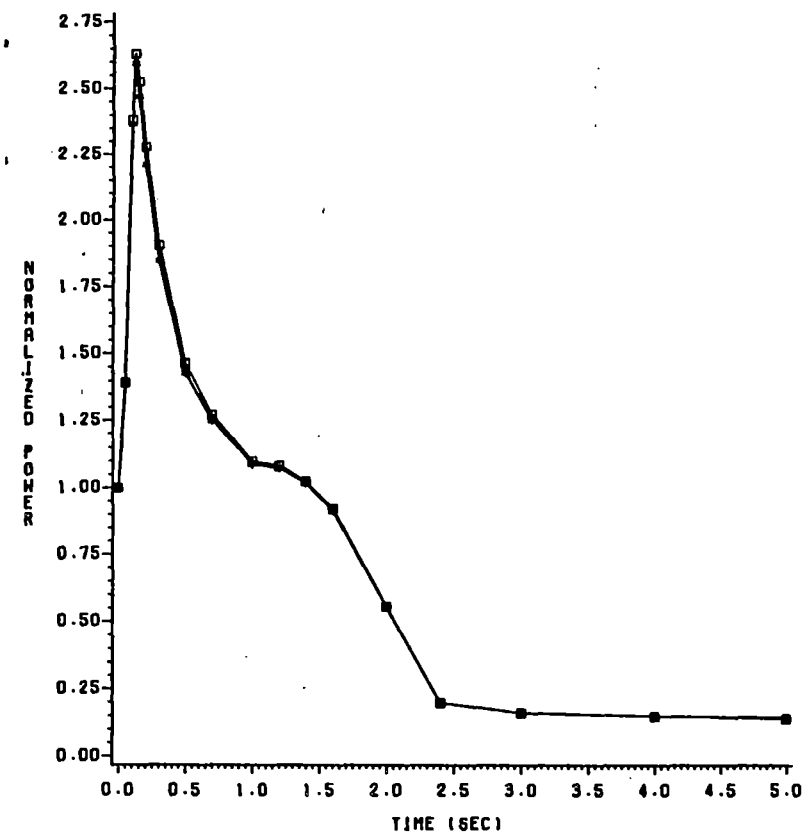
BEGINNING OF LIFE



STAR: NOMINAL CASE (PWF=1.2)
SQUARE: SENSITIVITY CASE (PWF=1.0)

FIGURE 3-3A

END OF LIFE



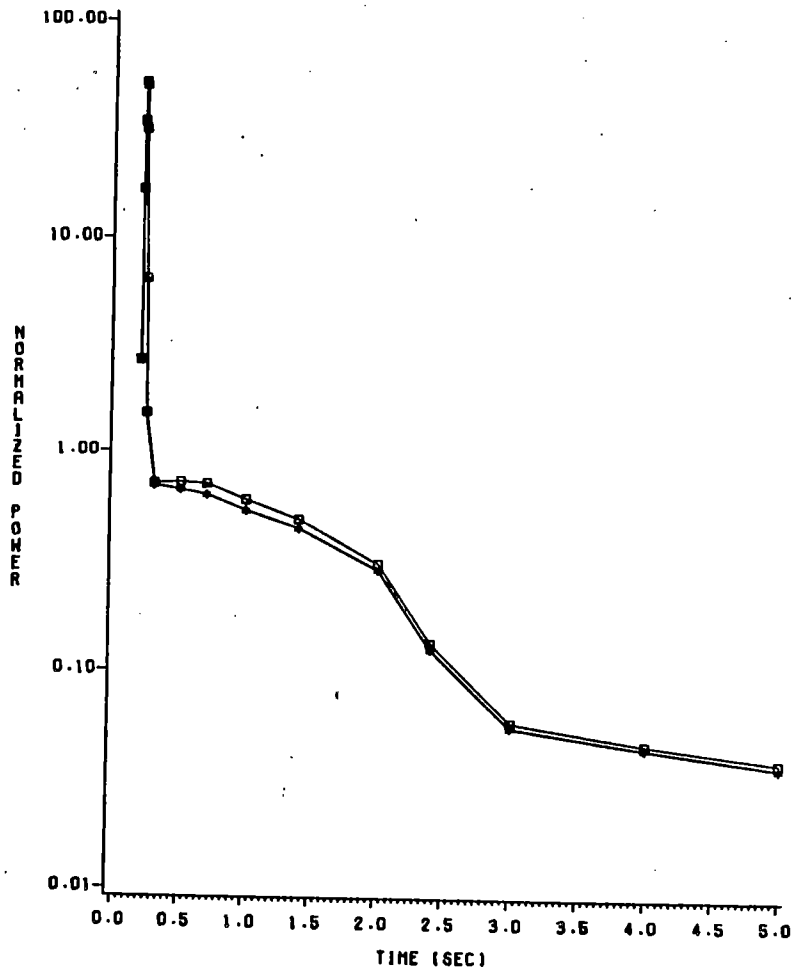
STAR: NOMINAL CASE (PWF=1.2)
SQUARE: SENSITIVITY CASE (PWF=1.0)

FIGURE 3-3B

FI[illegible] 3-4

SENSITIVITY STUDY - MODERATOR REACTIVITY FEEDBACK

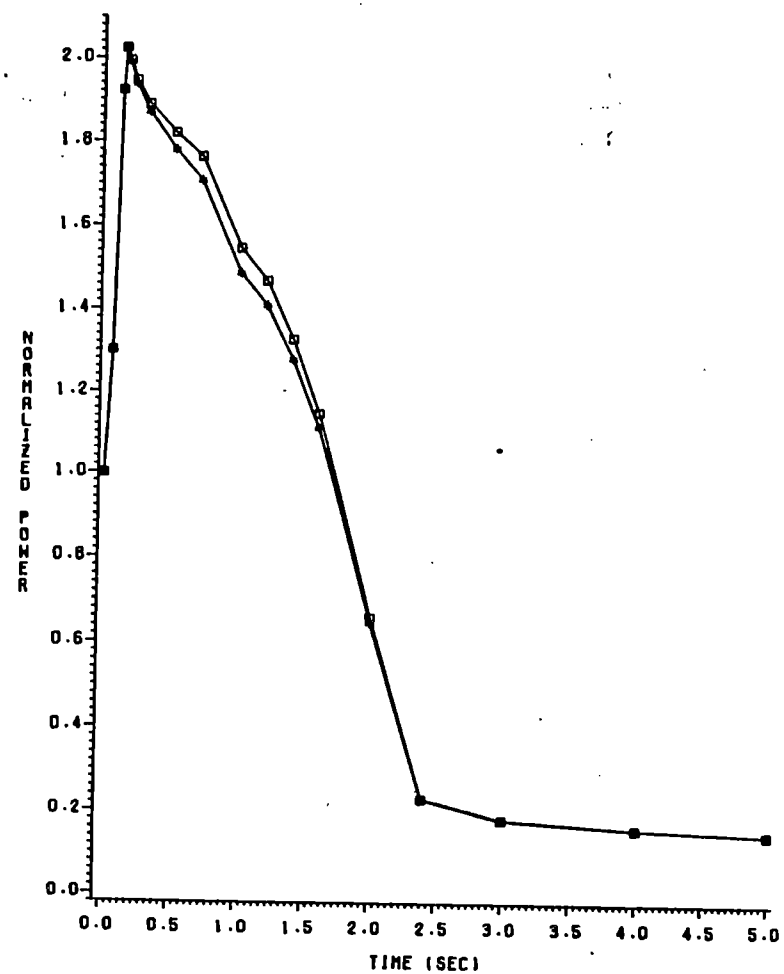
BEGINNING OF LIFE, HOT ZERO POWER



STAR: NOMINAL CASE
SQUARE: SENSITIVITY CASE (+ 3 PCM/DEG F)

FIGURE 3-4A

BEGINNING OF LIFE, HOT FULL POWER



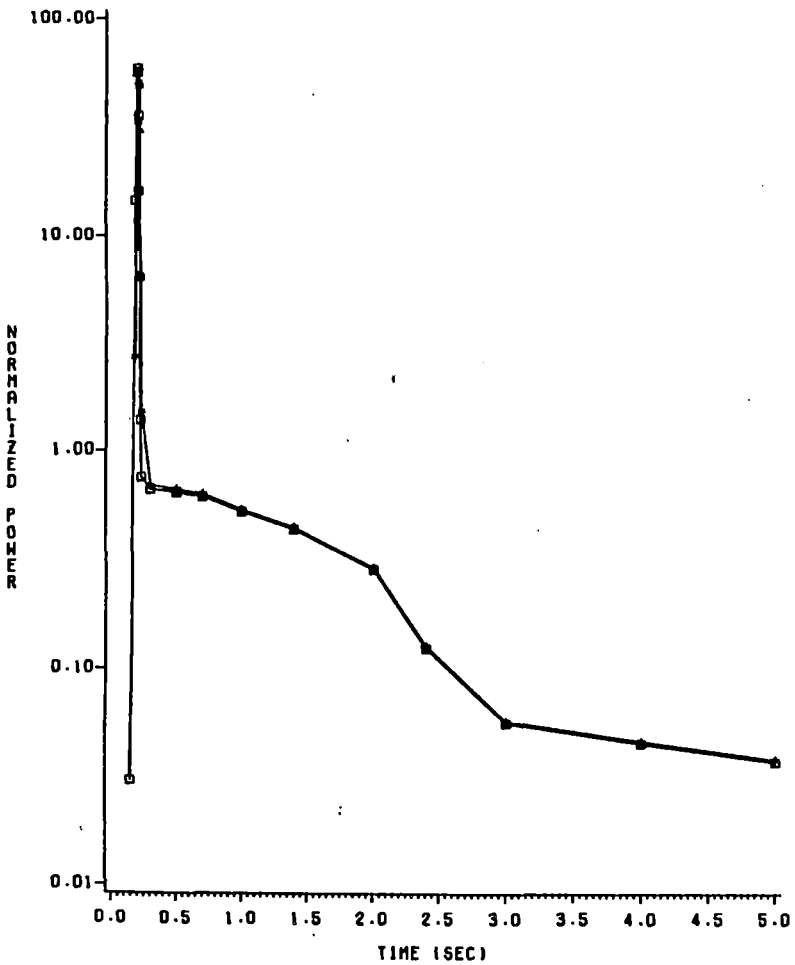
STAR: NOMINAL CASE
SQUARE: SENSITIVITY CASE (+ 3 PCM/DEG F)

FIGURE 3-4B

FIGURE 3-5

SENSITIVITY STUDY - DELAYED NEUTRON FRACTION

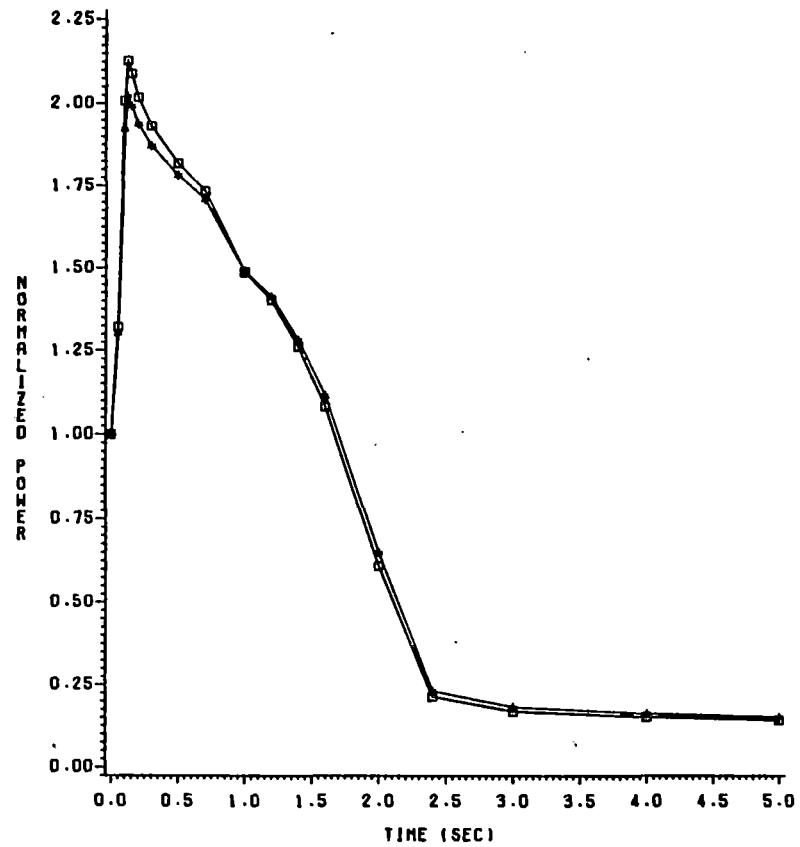
BEGINNING OF LIFE, HOT ZERO POWER



STAR: NOMINAL CASE
SQUARE: SENSITIVITY CASE (LESS 5%)

FIGURE 3-5A

BEGINNING OF LIFE, HOT FULL POWER



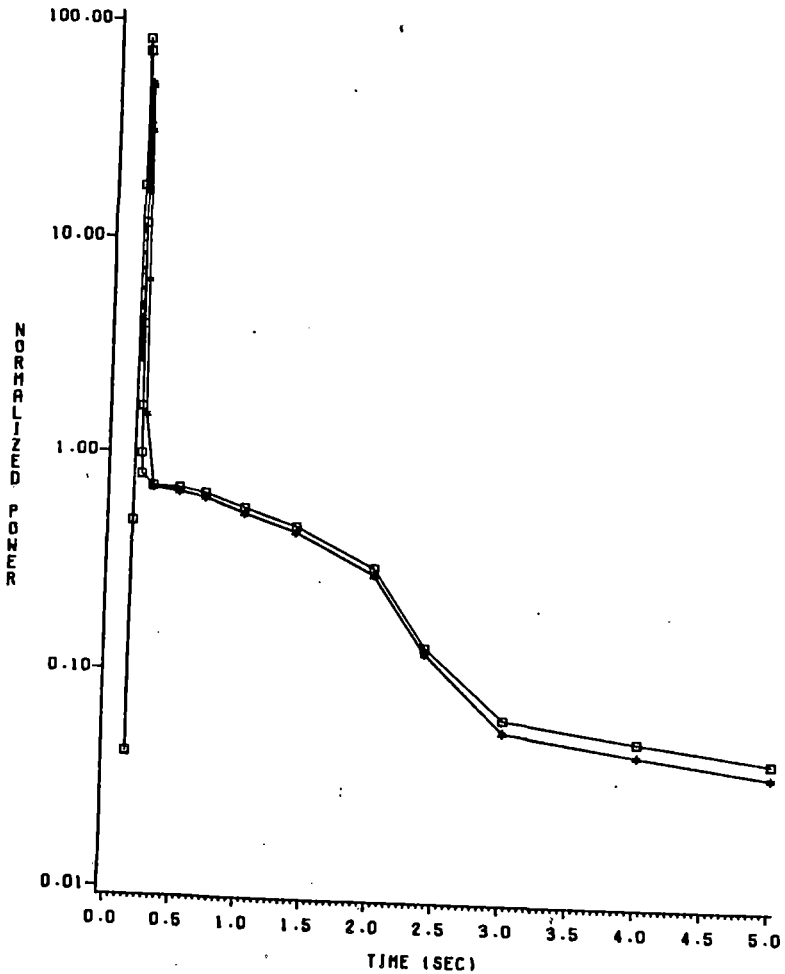
STAR: NOMINAL CASE
SQUARE: SENSITIVITY CASE (LESS 5%)

FIGURE 3-5B

FIGURE 3-6

SENSITIVITY STUDY - EJECTED ROD WORTH

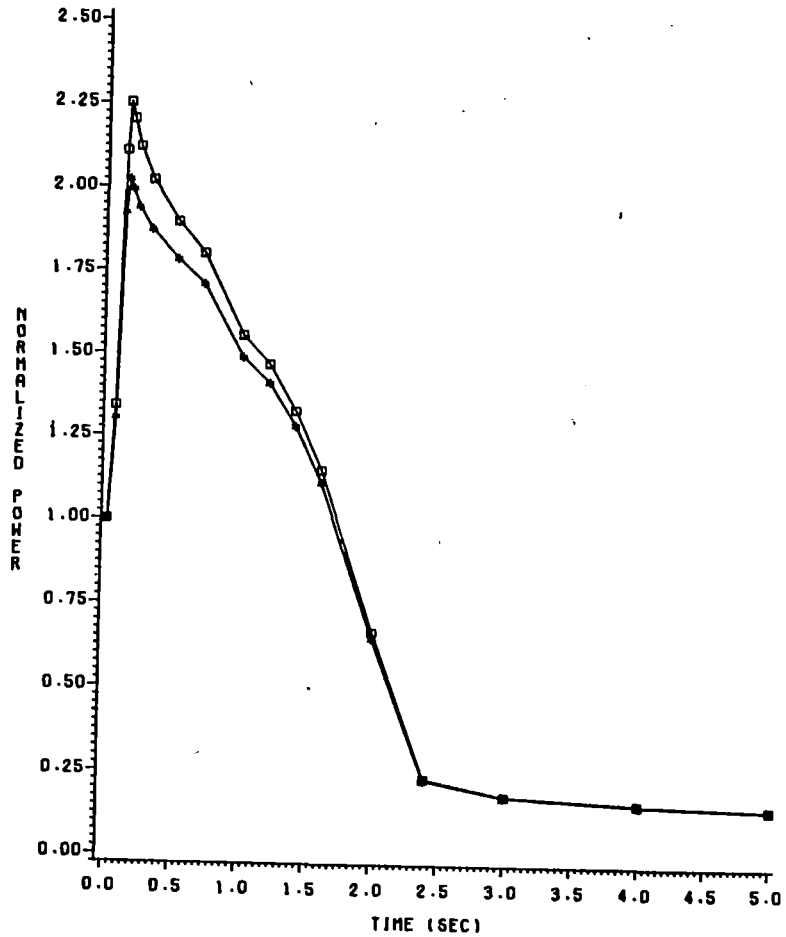
BEGINNING OF LIFE, HOT ZERO POWER



STAR: NOMINAL CASE
SQUARE: SENSITIVITY CASE (PLUS 10%)

FIGURE 3-6A

BEGINNING OF LIFE, HOT FULL POWER



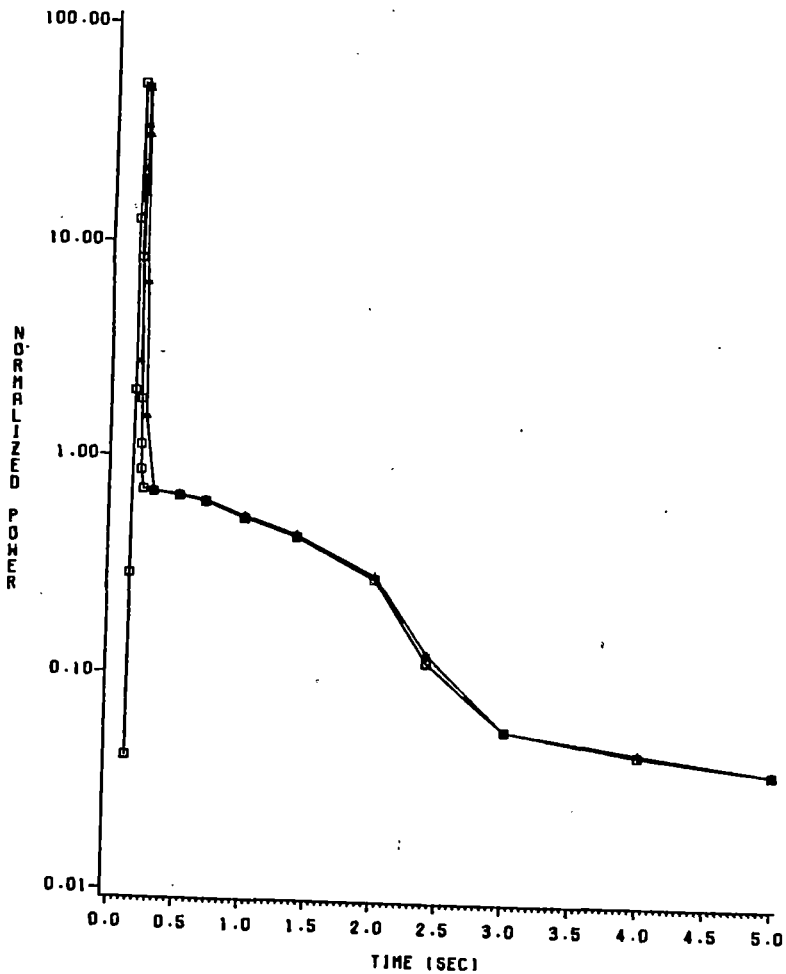
STAR: NOMINAL CASE
SQUARE: SENSITIVITY CASE (PLUS 10%)

FIGURE 3-6B

FIGURE 3-7

SENSITIVITY STUDY - DECREASED TIME OF EJECTION

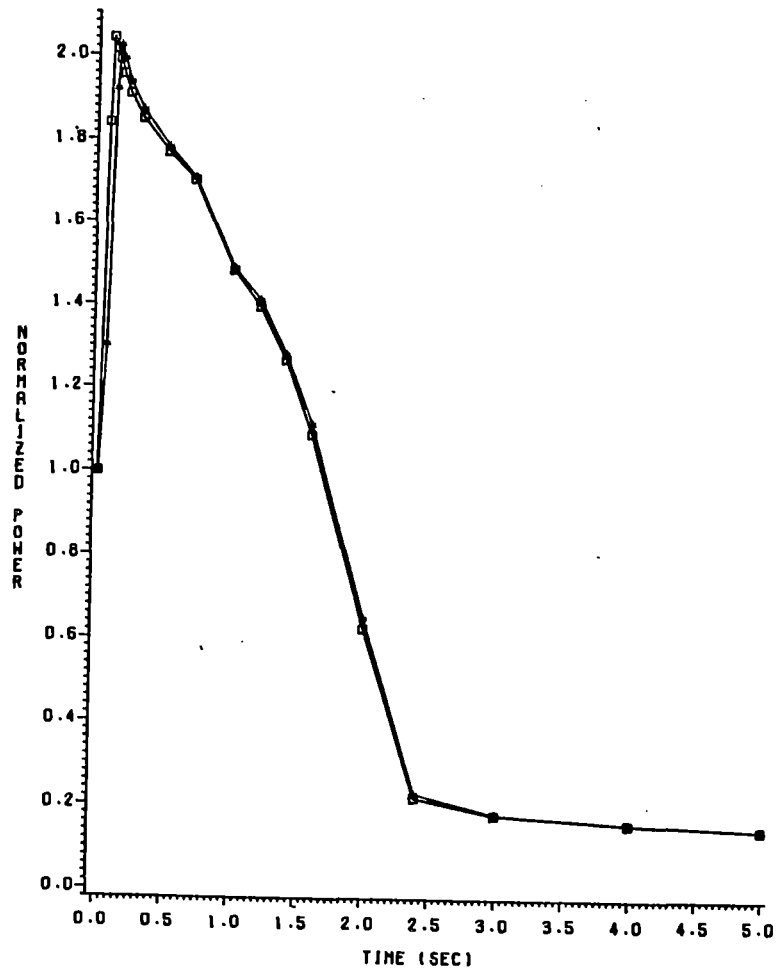
BEGINNING OF LIFE, HOT ZERO POWER



STAR: NOMINAL CASE (0.1 SEC)
SQUARE: SENSITIVITY STUDY (0.05 SEC)

FIGURE 3-7A

BEGINNING OF LIFE, HOT FULL POWER



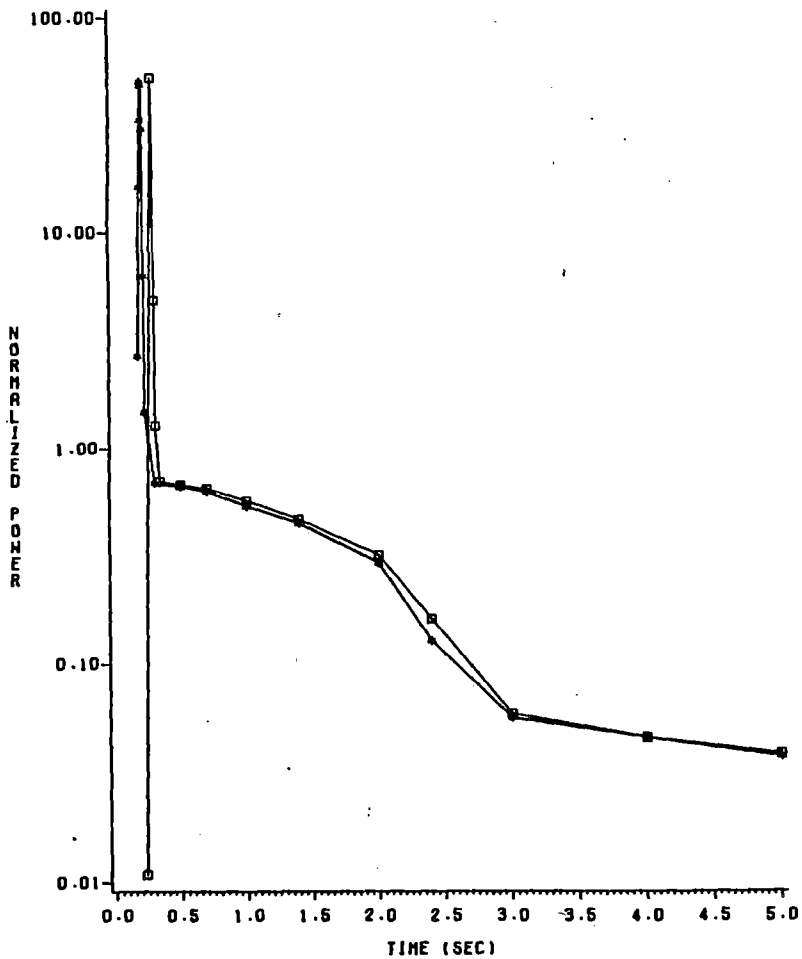
STAR: NOMINAL CASE (0.1 SEC)
SQUARE: SENSITIVITY CASE (0.05 SEC)

FIGURE 3-7B

FIGURE 3-8

SENSITIVITY STUDY - INCREASED TIME OF EJECTION

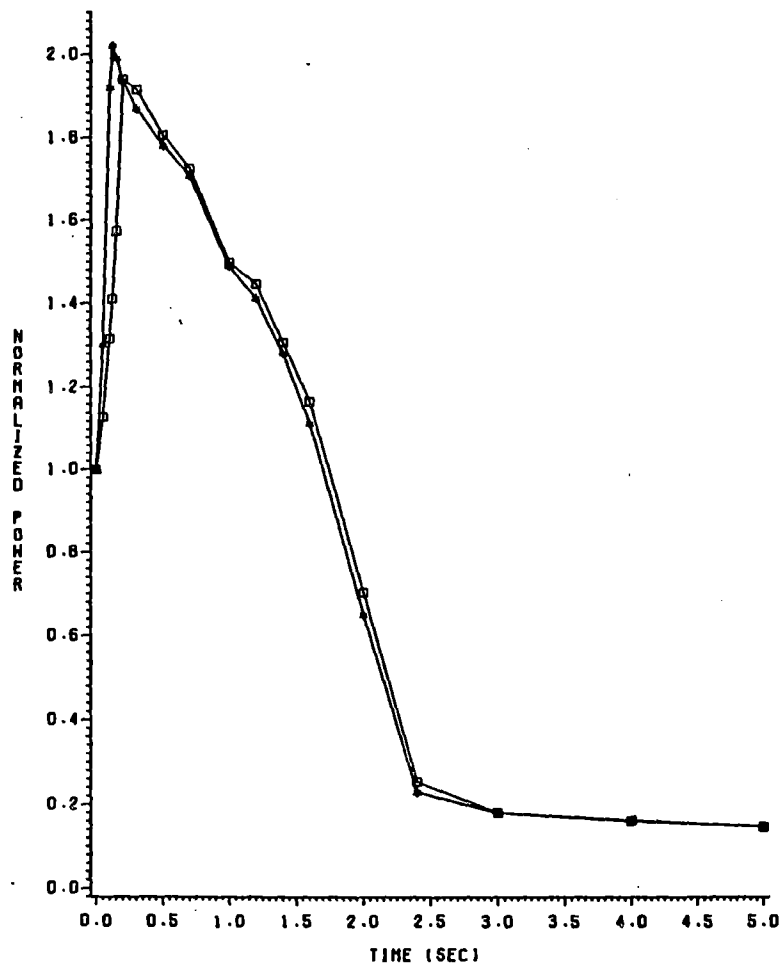
BEGINNING OF LIFE, HOT ZERO POWER



STAR: NOMINAL CASE (0.1 SEC)
SQUARE: SENSITIVITY CASE (0.2 SEC)

FIGURE 3-8A

BEGINNING OF LIFE, HOT FULL POWER



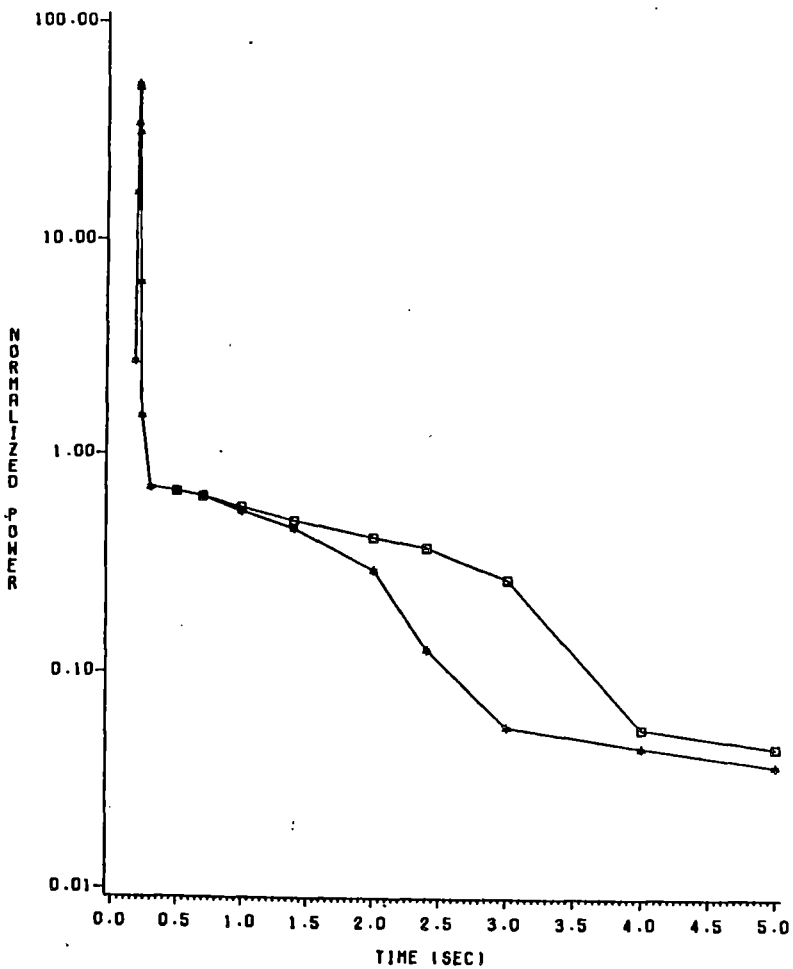
STAR: NOMINAL CASE (0.1 SEC)
SQUARE: SENSITIVITY CASE (0.2 SEC)

FIGURE 3-8B

FIGURE 3-9

SENSITIVITY STUDY - TRIP DELAY TIME

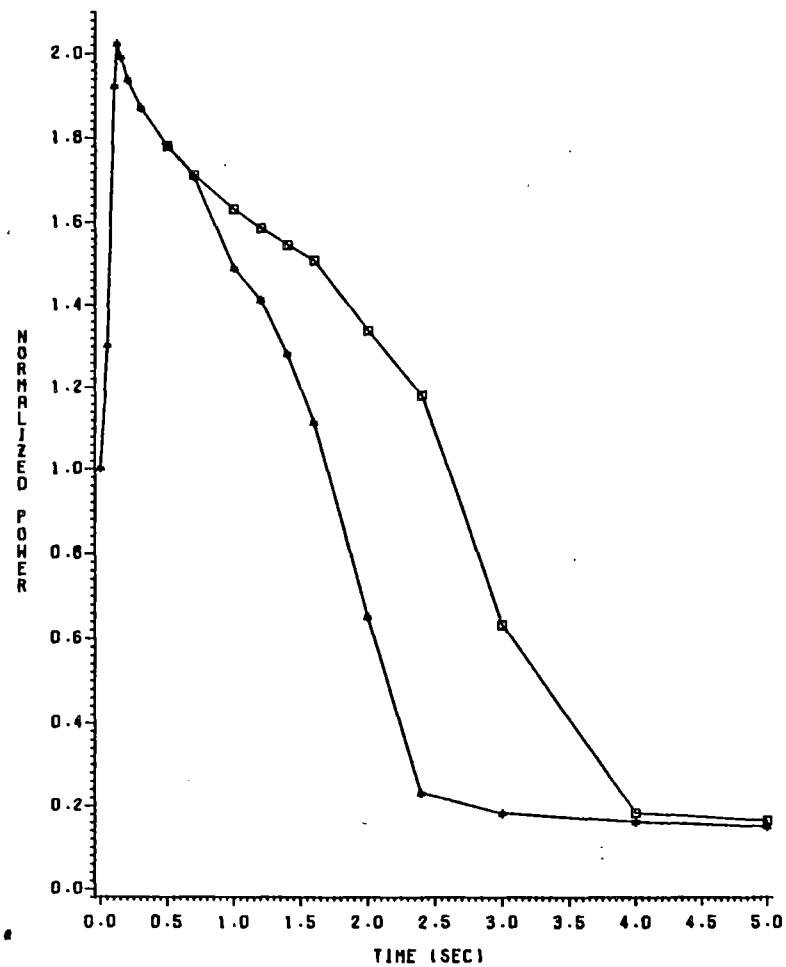
BEGINNING OF LIFE, HOT ZERO POWER



STAR: NOMINAL CASE (0.5 SEC)
SQUARE: SENSITIVITY CASE (1.5 SEC)

FIGURE 3-9A

BEGINNING OF LIFE, HOT FULL POWER



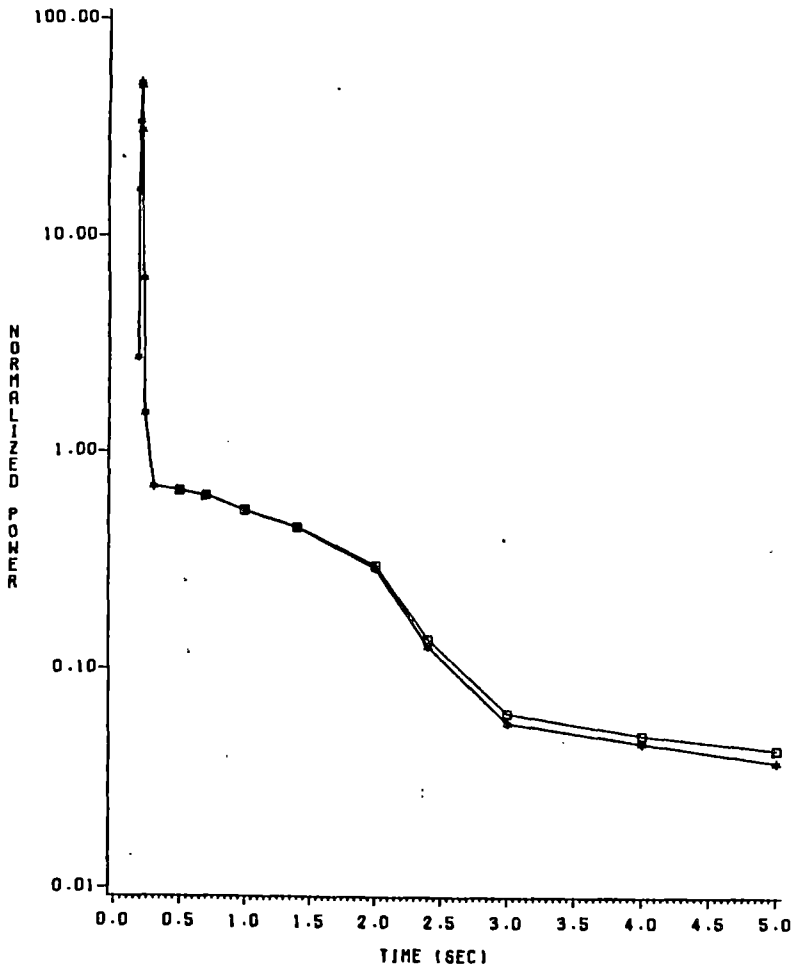
STAR: NOMINAL CASE (0.5 SEC)
SQUARE: SENSITIVITY CASE (1.5 SEC)

FIGURE 3-9B

FIGURE 3-10

SENSITIVITY STUDY - TRIP WORTH

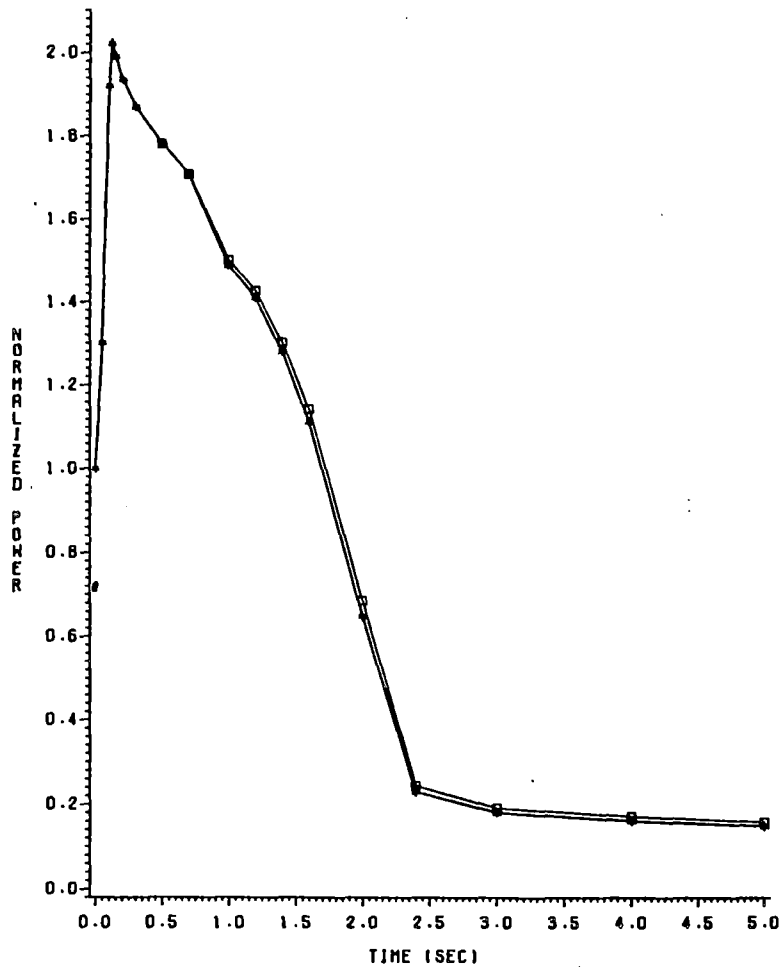
BEGINNING OF LIFE, HOT ZERO POWER



STAR: NOMINAL CASE
SQUARE: SENSITIVITY CASE (LESS 10%)

FIGURE 3-10A

BEGINNING OF LIFE, HOT FULL POWER



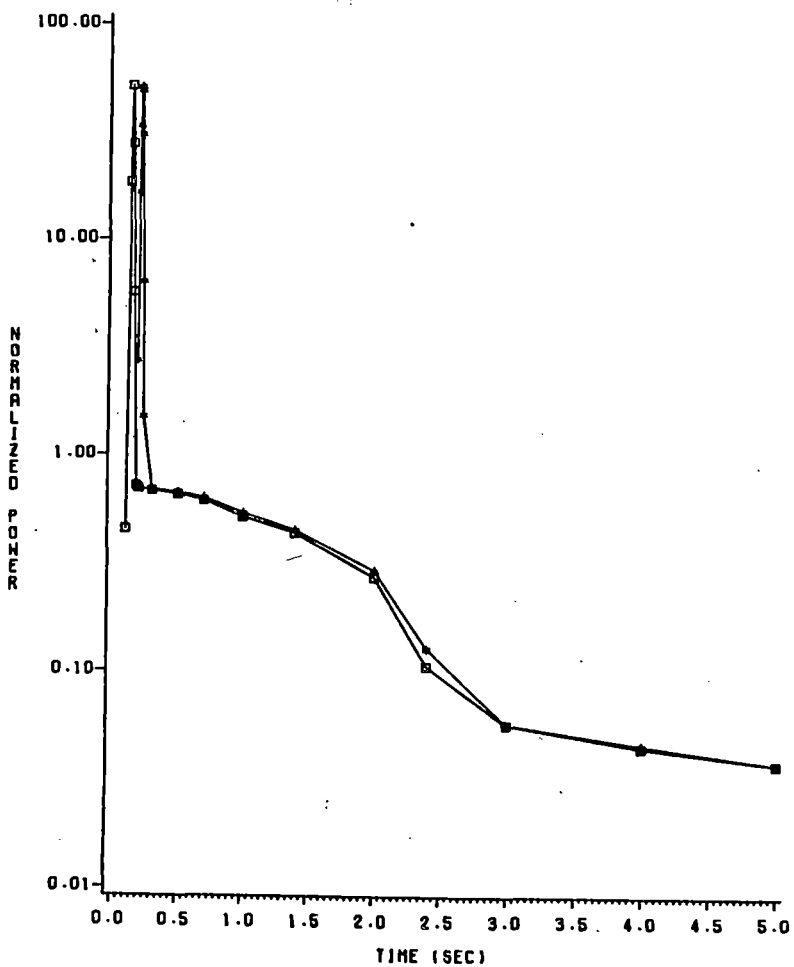
STAR: NOMINAL CASE
SQUARE: SENSITIVITY CASE (LESS 10%)

FIGURE 3-10B

FIGURE 3-11

SENSITIVITY STUDY - INITIAL ZERO POWER LEVEL

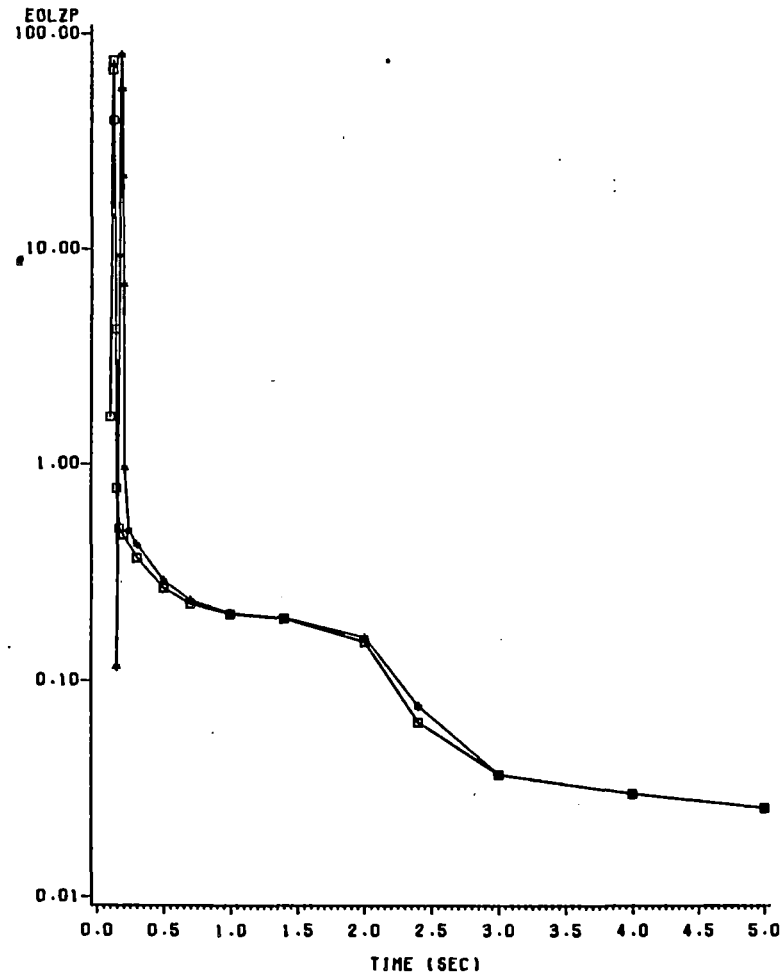
BEGINNING OF LIFE, HOT ZERO POWER



STAR: NOMINAL CASE (1.0E-9)
SQUARE: SENSITIVITY CASE (0.001)

FIGURE 3-11A

END OF LIFE, HOT ZERO POWER



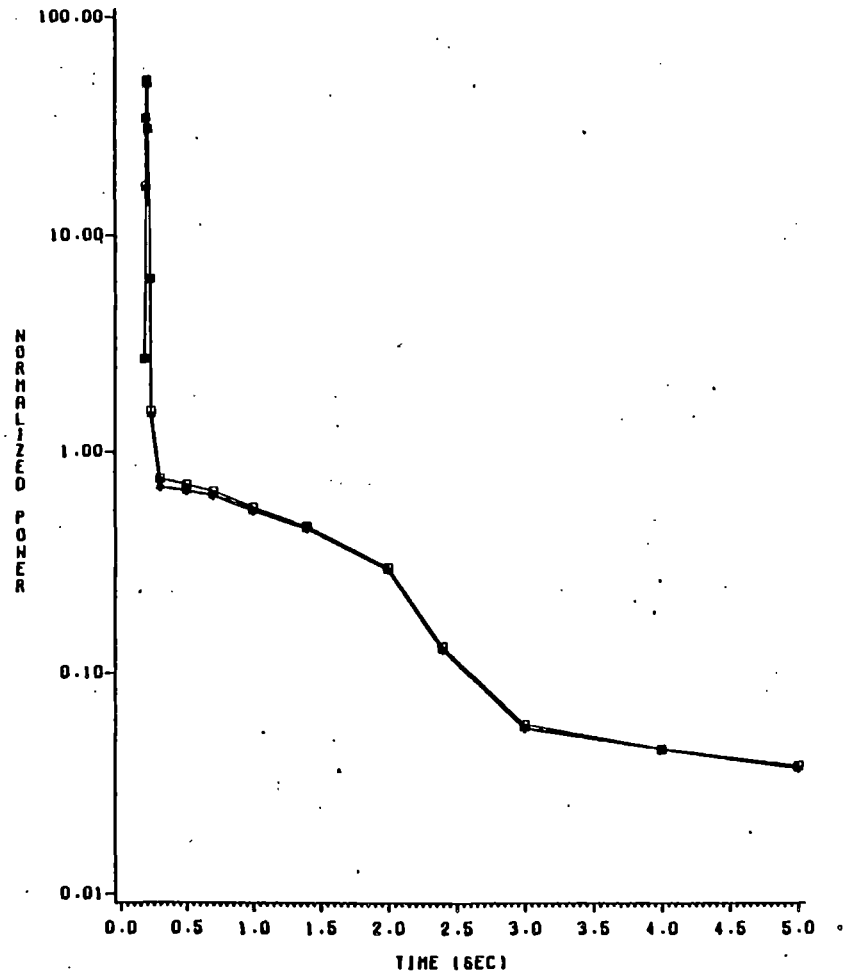
STAR: NOMINAL CASE (1.0E-9)
SQUARE: SENSITIVITY CASE (0.001)

FIGURE 3-11B

FIGURE 3-12

SENSITIVITY STUDY - BETA YIELD FRACTIONS

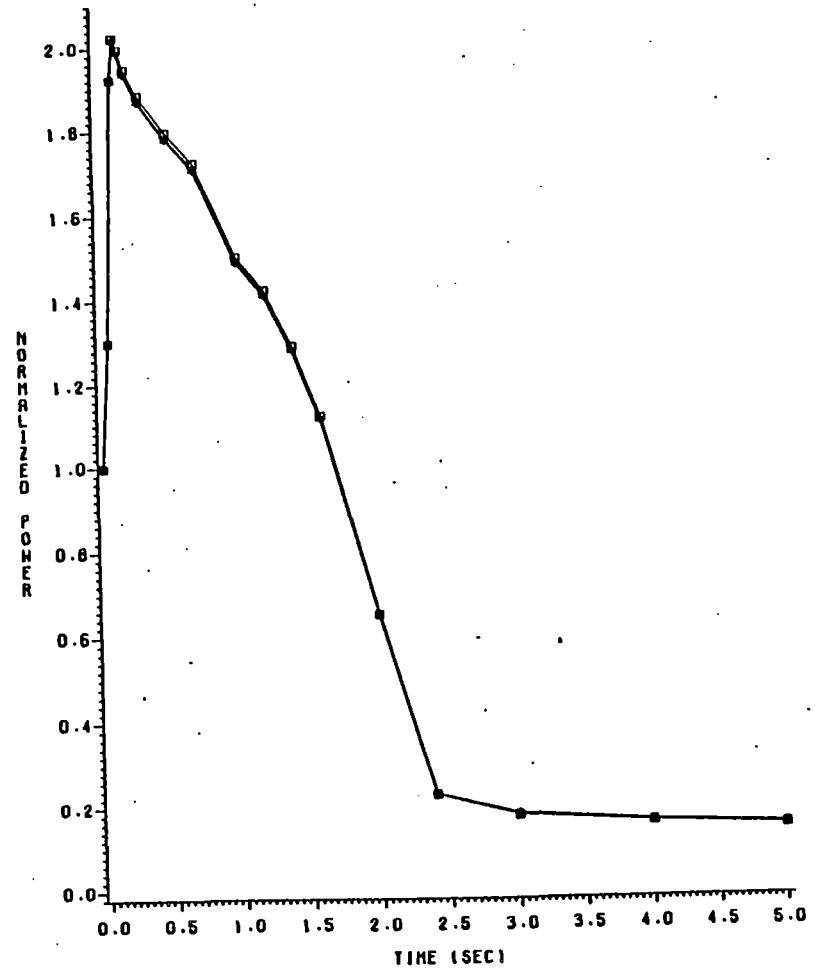
BEGINNING OF LIFE, HOT ZERO POWER



STAR: NOMINAL CASE (RETRAN CODE VALUES)
SQUARE: SENSITIVITY CASE (Surry Reload Values)

FIGURE 3-12A

BEGINNING OF LIFE, HOT FULL POWER



STAR: NOMINAL CASE (RETRAN CODE VALUES)
SQUARE: SENSITIVITY CASE (Surry Reload Values)

FIGURE 3-12B

2.2 Point Kinetics Model Thermal Hydraulic Parameters

Four parameters were evaluated in assessing the sensitivity of thermal hydraulic parameters on the RETRAN point kinetics calculation. These were:

1. gap heat transfer coefficient
2. fuel pin geometry description
3. core inlet temperature
4. system pressure

Table 3-2 presents a summary of the results in the format of Table 3-1 discussed previously.

3.2.2.1 Gap Heat Transfer Coefficient

In the RETRAN point kinetics model, the fuel-clad gap heat transfer coefficient (HGAP) is modeled by using a constant conductivity for the gap material throughout the rod ejection transient. For sensitivity study purposes, the value of HGAP was decreased by 50% from its nominal value of 1000 Btu/ft^2-hr-°F to 500 Btu/ft^2-hr-°F. The results as presented in Table 3-2 show little sensitivity.

As described in Section 2, a change in the value of HGAP is expected to result in off-setting effects on the transient. A reduction in HGAP leads to a decrease in the rate of heat transfer from the fuel to the coolant with a resultant increase in the rate of fuel heatup. This increased rise in fuel temperature in turn results in a greater negative Doppler reactivity feedback effect which leads to a more rapid reduction in core power which in turn tends to reduce the fuel temperature.

Figure 3-13 presents a comparison of the power histories for the two BOL cases.

3.2.2.2 Fuel Pin Geometry Description

The standard RETRAN Single Loop Model describes the fuel pin with two concentric fuel pin regions, a gap region and a cladding region. For the purpose of the sensitivity study, the number of concentric fuel pin regions was increased from two to six. As shown in Table 3-2, this change had little impact on the transient predictions. Figure 3-14 presents a comparison of the power histories for the two BOL cases.

3.2.2.3 Core Inlet Temperature

The initial core inlet temperature was increased 4 °F from the nominal values. (4 °F is typical of the uncertainty assumed in the initial coolant temperature for a safety analysis.) A change of this magnitude showed little effect as seen in Table 3-2.

3.2.2.4 System Pressure

Typical uncertainty assumed for system pressure in a safety analysis is 30 psia. The initial system pressure in the point kinetics analysis was reduced by this magnitude to investigate the sensitivity to system pressure. As with the core inlet temperature, no noticeable impact on the transient result was found, (Table 3-2.)

TABLE 3-2

POINT KINETICS THERMAL HYDRAULIC SENSITIVITY STUDY

I. BOL HZP Studies:

Parameter	Sensitivity	Peak Norm. Power	Energy Release (t=5 sec)
Nominal case values	Not applicable	51.1	2.23
Gap heat transfer coefficient	Reduced 50%	51.0	2.02
# of fuel pin meshes	Changed from 2 to 6	51.1	2.15
Core inlet temperature	Increased 4 °F	51.1	2.23
System pressure	Increased 30 psia	51.1	2.23

II. BOL HFP Studies:

Parameter	Sensitivity	Peak Norm. Power	Energy Release (t=5 sec)
Nominal case values	Not applicable	2.02	3.50
Gap heat transfer coefficient	Reduced 50%	2.02	3.45
# of fuel pin meshes	Changed from 2 to 6	2.02	3.45
Core inlet temperature	Increased 4 °F	2.02	3.50
System pressure	Increased 30 psia	2.02	3.50

Notes:

Peak Norm. Power = peak normalized power occurring during transient, (1.0 is equivalent to full power level.)

Energy Release (t=5 sec) = total core energy release up to a transient time of 5.0 seconds, (units of full-power-seconds.)

TABLE 3-2 (cont.)

POINT KINETICS THERMAL HYDRAULIC SENSITIVITY STUDY

I. EOL HZP Studies:

Parameter	Sensitivity	Peak Norm. Power	Energy Release (t=5 sec)
Nominal case values	Not applicable	80.2	1.87
Gap heat transfer coefficient	Reduced 50%	80.1	1.86
# of fuel pin meshes	Changed from 2 to 6	80.2	1.87
Core inlet temperature	Increased 4 °F	80.1	1.87
System pressure	Increased 30 psia	80.3	1.88

II. EOL HFP Studies:

Parameter	Sensitivity	Peak Norm., Power	Energy Release (t=5 sec)
Nominal case values	Not applicable	2.60	3.02
Gap heat transfer coefficient	Reduced 50%	2.60	3.06
# of fuel pin meshes	Changed from 2 to 6	2.60	3.05
Core inlet temperature	Increased 4 °F	2.60	3.02
System pressure	Increased 30 psia	2.60	3.02

Notes:

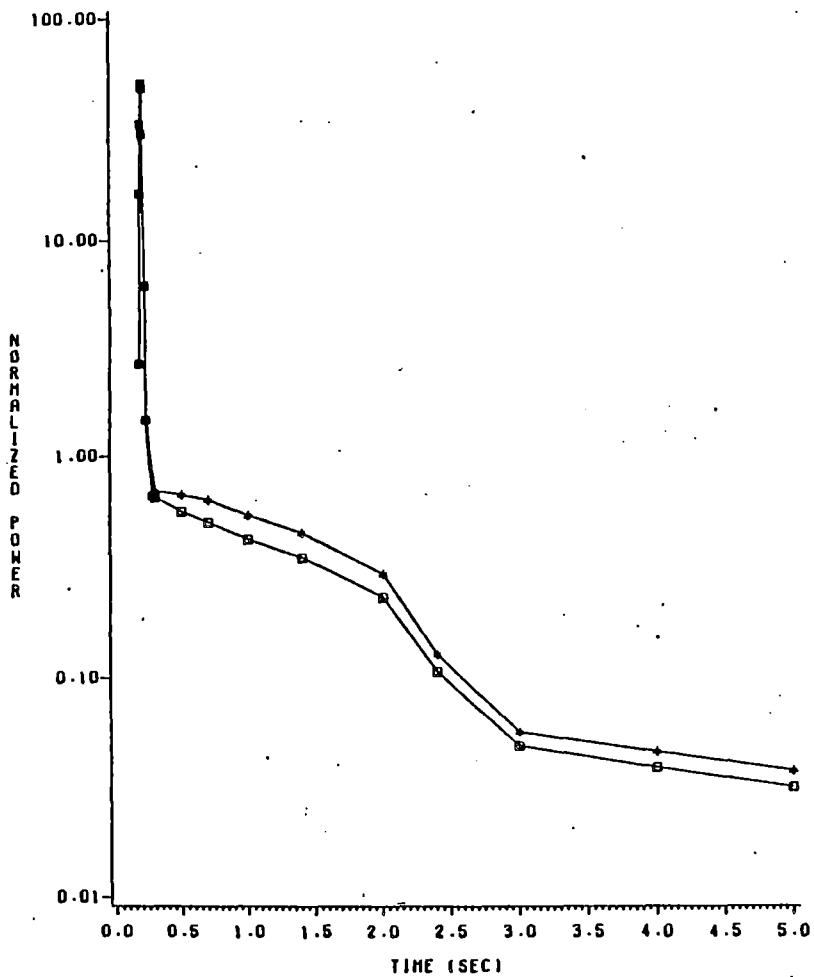
Peak Norm. Power = peak normalized power occurring during transient, (1.0 is equivalent to full power level.)

Energy Release (t=5 sec) = total core energy release up to a transient time of 5.0 seconds, (units of full-power-seconds.)

URE 3-13

SENSITIVITY STUDY - GAP HEAT TRANSFER

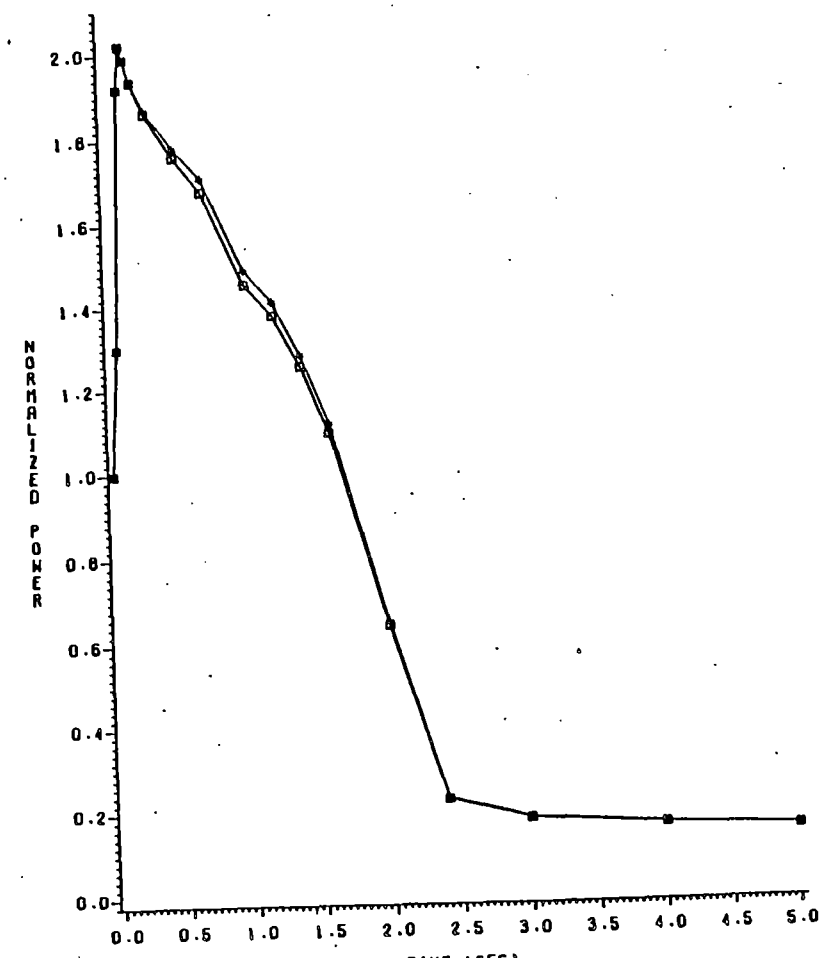
BEGINNING OF LIFE, HOT ZERO POWER



STAR: NOMINAL CASE
SQUARE: SENSITIVITY CASE (LESS 50%)

FIGURE 3-13A

BEGINNING OF LIFE, HOT FULL POWER



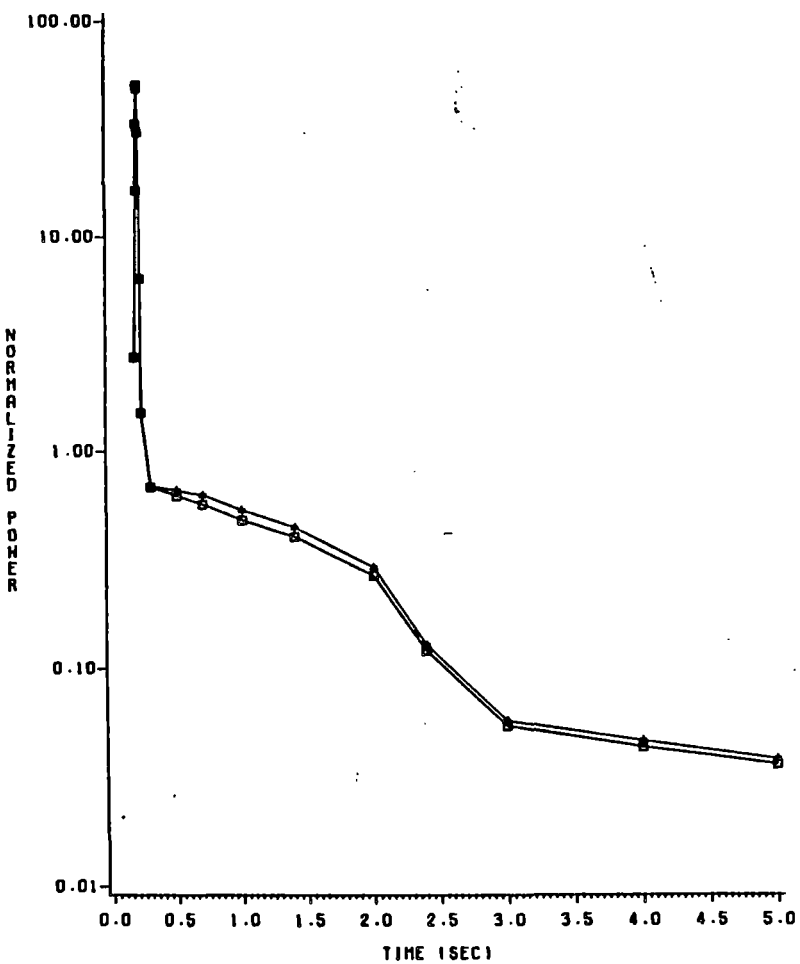
STAR: NOMINAL CASE
SQUARE: SENSITIVITY CASE (LESS 50%)

FIGURE 3-13B

FIGURE 3-14

SENSITIVITY STUDY - POINT KINETICS FUEL PIN GEOMETRY

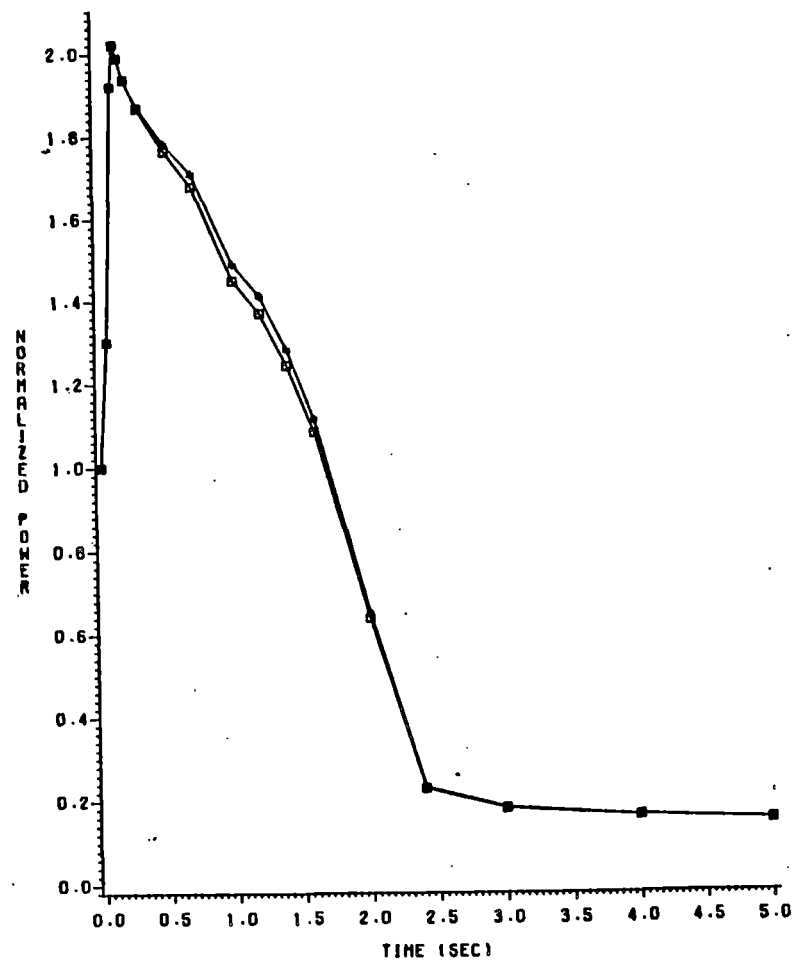
BEGINNING OF LIFE, HOT ZERO POWER



STAR: NOMINAL CASE (TWO FUEL REGIONS)
SQUARE: SENSITIVITY CASE (SIX FUEL REGIONS)

FIGURE 3-14A

BEGINNING OF LIFE, HOT FULL POWER



STAR: NOMINAL CASE (TWO FUEL REGIONS)
SQUARE: SENSITIVITY CASE (SIX FUEL REGIONS)

FIGURE 3-14B

2.3 Hot Spot Parameters

Sensitivity studies were performed with the RETRAN Hot Spot Model to evaluate the impact of various thermal hydraulic and neutronics parameters on the temperature and enthalpy history of the hot spot location. For a given nuclear unit at a given initial power level, the only differences in the RETRAN Hot Spot Model between a BOL and EOL case are the assumed temperature of fuel melt and the input hot spot power history. Therefore, the sensitivity studies were performed only at BOL conditions for both a zero power and full power case.

The sensitivities investigated were:

1. total power peaking factor
2. power history shape
3. gap heat transfer coefficient
4. fuel pin geometry description
5. inlet temperature
6. system pressure
7. mass flow rate
8. metal-water reaction option

Table 3-3 presents a summary of the sensitivity study results for the Hot Spot Model. The percentage of fuel melted, maximum fuel centerline temperature and clad temperature, and maximum fuel average enthalpy are compared for each sensitivity case with those predicted for a nominal case since these are the key parameters in assessing the potential for fuel damage and radiological release for the transient.

2.3.1 Power Peaking Factor

The core average power history predicted with the point kinetics model is weighted by a hot spot total power peaking factor (Fq) shape before input to the hot spot calculation. The design uncertainty on Fq as predicted by Vepco's steady state physics models is presently 7.5%. The power histories for the two sensitivity study cases were increased by this factor (7.5%). The results as presented in Table 3-3 show an increase in the temperatures and average enthalpy for both cases.

3.2.3.2 Power History Shape

One of the point kinetics neutronics sensitivies investigated was the time of rod ejection which is nominally set at 0.1 seconds for the rod to be completely ejected from the core. To model a similar type of sensitivity with the Hot Spot Model, the hot spot power history was displaced 0.1 seconds later in time; i.e., the time of the peak power and all later powers were increased by 0.1 seconds. As seen in Table 3-3, the sensitivity was minimal.

3.2.3.3 Gap Heat Transfer Coefficient

For a HFP case, the gap heat transfer coefficient (HGAP) is initially adjusted to produce a target initial radial temperature distribution for the fuel pin. At 0.1 seconds into the transient, this value is increased to 10000 Btu/ft^2-hr-°F and held there for the remainder of the transient to model the expected closure of the gap. Since the initial value of HGAP has no impact on the pin's initial radial temperature distribution

For a HZP case, the value of HGAP is maintained at 10000 Btu/ft^2-hr-°F for the entire transient in order to simplify the analysis. (Additional sensitivity studies showed this change to have no significant impact on the Hot Spot Model's predictions for HZP cases.)

For the sensitivity investigation, the values of HGAP used in the nominal analysis were reduced by 10%. This would be expected to lead to higher fuel temperatures since the ability of the fuel to transfer heat to the coolant has been reduced. The results presented in Table 3-3 show little significant impact from such a reduction.

3.2.3.4 Fuel Pin Geometry Description

The fuel pin geometry of the Hot Spot Model is described with ten concentric fuel regions, a single gap region, and three concentric cladding regions. To test the sensitivity of this geometry, the number of fuel regions was reduced from ten to six. The results shown in Table 3-3 show little impact on the model's predictions.

An additional note is required on the percentage of fuel melt which is shown for the HFP case in Table 3-3. For the sensitivity case this is listed as <11.1 % while for the nominal case it is given as <9 %. Although the acceptance criteria for percentage of fuel melt is <10 %, the six node sensitivity case value of <11.1 % does not necessarily imply that the acceptance criteria was violated. Due to the closeness of the comparisons of the average fuel enthalpy and fuel temperature histories between the nominal and sensitivity cases, the percentage of fuel melt for the sensitivity case is indeed <9 %, (see Figure 3-15.)

Additional confirmation is presented in Figure 3-16 which provides a comparison of the radial fuel temperature distribution through the pellet for the sensitivity and nominal cases at the time of maximum fuel melt.

The simple technique used for determining percentage of fuel melt with the RETRAN Hot Spot Model merely puts upper and lower bounds on the percentages. As shown in Table 2-6, the estimate of maximum fuel melt being <9 % for the ten fuel region nominal case is derived from the fact that the highest numbered node for which the transient fuel temperature did not exceed the assumed melting point was node 4. For the six fuel region sensitivity case, this table is invalid. The highest numbered fuel node for the six fuel region case which did not exceed the assumed melting temperature was node 3. Applying the methodology used to derive Table 2-6 to the six region case gives a percentage of fuel melt somewhere between 2.8 % and 11.1 %.

3.2.3.5 Inlet Temperature

The initial inlet temperature to the hot spot volume was increased by 5 °F for the sensitivity study. As expected the maximum temperatures for both cases showed a slight rise over the nominal values (Table 3-3), but overall the effect was insignificant.

3.2.3.6 System Pressure

The initial system pressure was decreased by 30 psia. This produced a slight increase in the maximum clad temperatures for the sensitivity

ses (Table 3-3), but again the overall effect was minimal.

3.2.3.7 Mass Flow Rate

Reducing the mass flow rate throughout the Hot Spot Model by 5% produced the most impact on the clad temperature where it would be expected to show the most noticable effect (Table 3-3). The impact on the pellet centerline temperature, as shown in the HZP case results, was insignificant.

3.2.3.8 Metal-Water Reaction

The RETRAN Hot Spot Model includes a calculation of the additional energy release due to reaction between the coolant the the Zircaloy metal of the cladding. Turning off this option reduced the temperatures and enthalpy predicted by the model as expected (Table 3-3). Again the sensitivity was more pronounced in the clad than in the fuel.

TABLE 3-3

HOT SPOT SENSITIVITY STUDY

I. HZP Studies:

Parameter	Sensitivity	% Fuel Melt	Max. Tcl (°F)	Max. Tclad (°F)	Max. Enthalpy (Btu/lb)
Nominal Case Values	Not Applicable	0	4017	2460	266
Power Peaking Factor	Increased 7.5%	0	4205	2609	285
Power History Shape	0.1 sec delay of peak	0	4018	2463	266
Gap Heat Transfer Coefficient	Reduced 10%	0	4018	2456	266
# of Fuel Pin Meshes	Changed from 10 to 6	0	4012	2471	266
Inlet Temperature	Increased 5 °F	0	4020	2462	266
System Pressure	Reduced 30 psia	0	4018	2474	266
Mass Flow Rate	Reduced 5%	0	4019	2496	267
Metal-Water Reaction	Turned off	0	4014	2372	263

Abbreviations:

% Fuel Melt = Maximum % pellet melting at hot spot

Max. Tcl = Maximum pellet centerline temperature at hot spot

Max. Tclad = Maximum cladding temperature at hot spot

TABLE 3-3 (cont.)

HOT SPOT SENSITIVITY STUDY

II. HFP Studies:

Parameter	Sensitivity	% Fuel Melt	Max. Tcl (°F)	Max. Tclad (°F)	Max. Enthalpy (Btu/lb)
Nominal Case Values	Not Applicable	<9	4903	2285	317
Power Peaking Factor	Increased 7.5%	<9	4904	2358	328
Power History Shape	0.1 sec delay of peak	<9	4903	2297	318
Gap Heat Transfer Coefficient	Reduced 10%	<9	4904	2304	322
# of Fuel Pin Meshes	Changed from 10 to 6	<11.1*	4902	2296	316
Inlet Temperature	Increased 5 °F	<9	4903	2288	317
System Pressure	Reduced 30 psia	<9	4903	2298	317
Mass Flow Rate	Reduced 5%	<9	4903	2319	318
Metal-Water Reaction	Turned off	<9	4903	2239	315

Abbreviations:

% Fuel Melt = Maximum % pellet melting at hot spot

Max. Tcl = Maximum pellet centerline temperature at hot spot

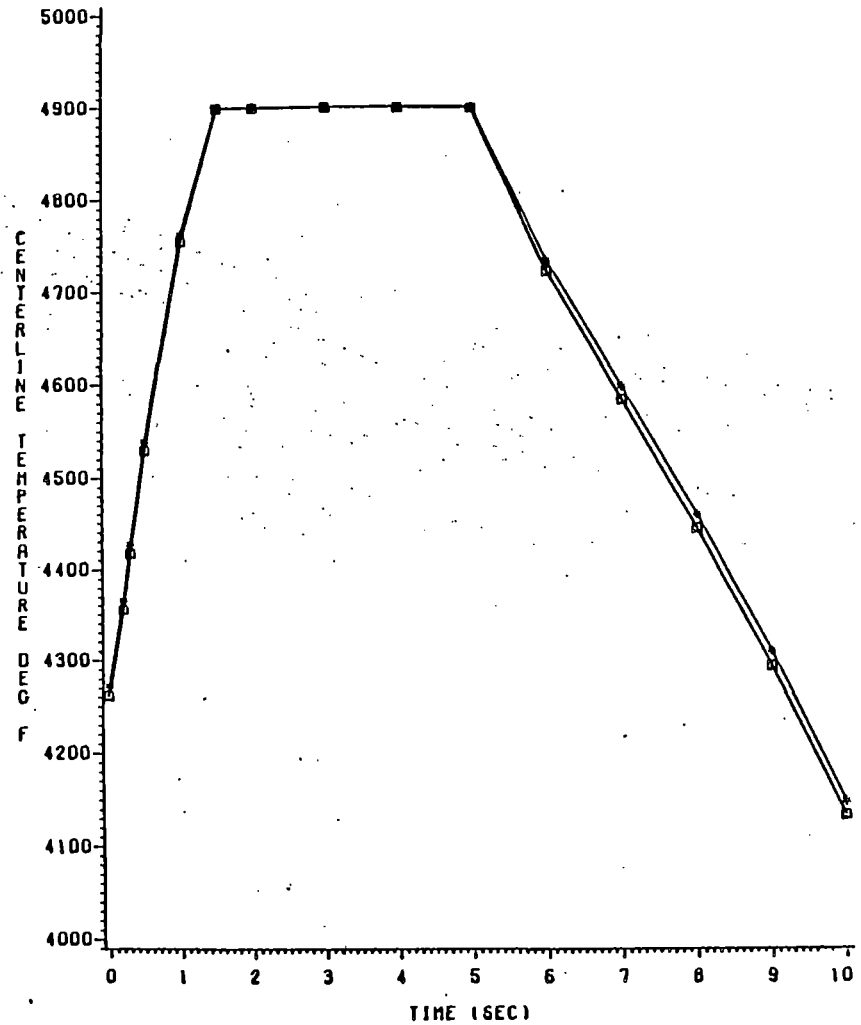
Max. Tclad = Maximum cladding temperature at hot spot

* As described in Section 3.2.3.4, the actual % of fuel melt for the sensitivity case is <9%.

FIGURE 3-15

SENSITIVITY STUDY - HOT SPOT FUEL PIN GEOMETRY

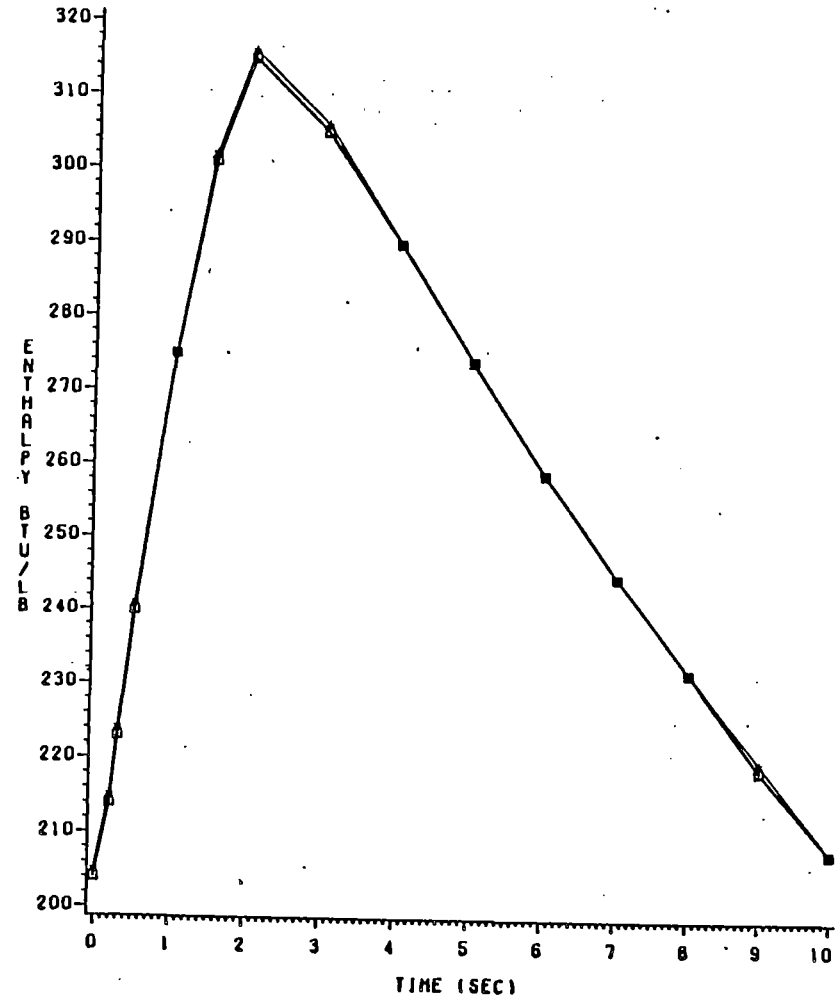
FULL POWER HOT SPOT CENTERLINE TEMPERATURE



STAR: NOMINAL TEN FUEL REGION CASE
SQUARE: SENSITIVITY SIX FUEL REGION CASE

FIGURE 3-15A

FULL POWER HOT SPOT AVERAGE ENTHALPY

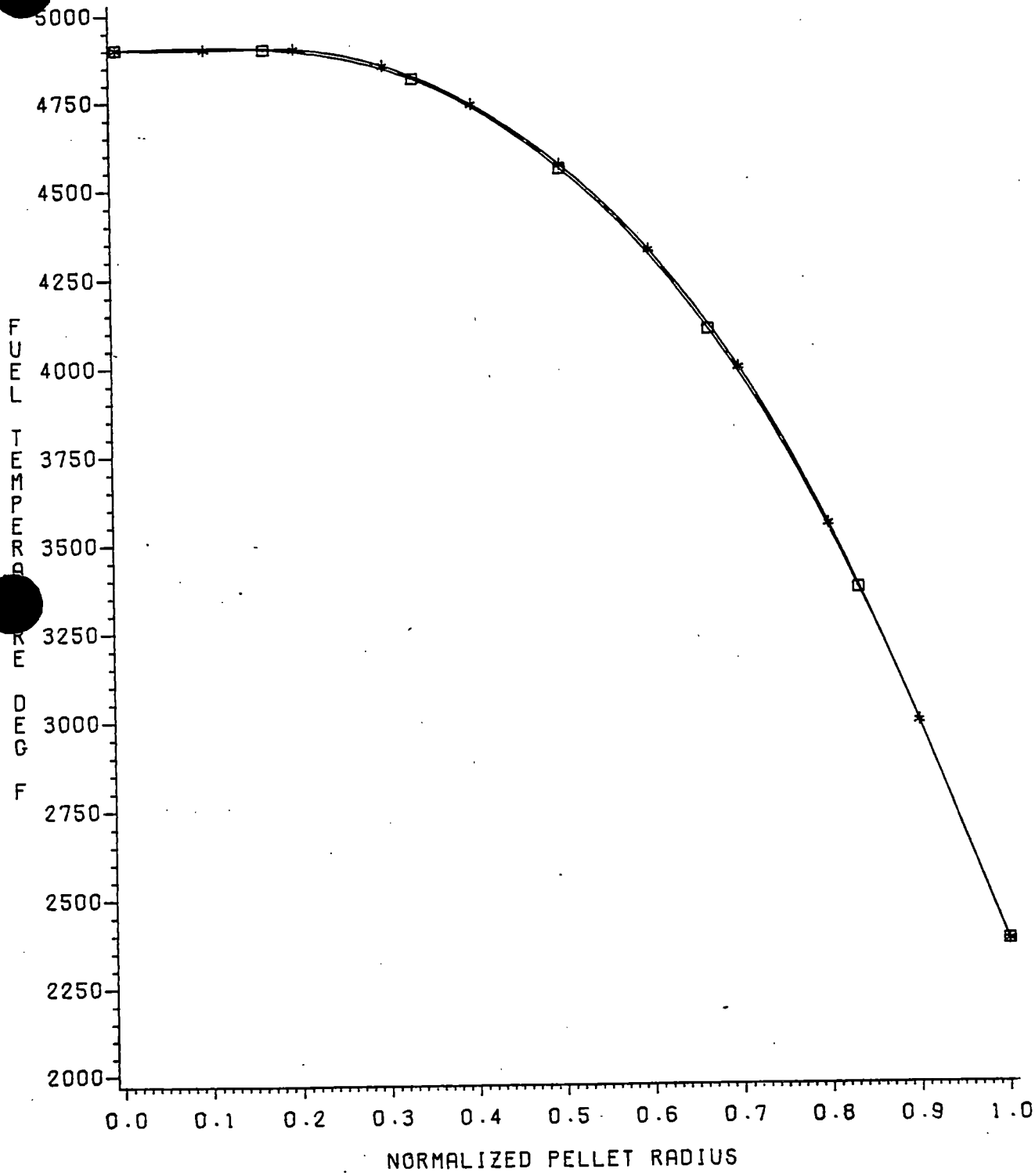


STAR: NOMINAL TEN FUEL REGION CASE
SQUARE: SENSITIVITY SIX FUEL REGION CASE

FIGURE 3-15B

FIGURE 3-16

SENSITIVITY STUDY -- HOT SPOT PELLET TEMPERATURE DISTRIBUTION



STAR = 10 NODE PELLET GEOMETRY
SQUARE = 6 NODE PELLET GEOMETRY

SECTION 4 - VERIFICATION COMPARISONS

4.1 Introduction

The purpose of this report is to document Vepco's analytical capability for performing reload core safety and licensing analysis for the rod ejection transient. As verification of this capability, appropriate results and comparisons are provided for a representative series of analyses of licensing transients.

A description of the licensing methodology and codes presently used by the vendor (Westinghouse) is provided below. These methods are currently approved for the licensing of Vepco reload cores and were implemented by Vepco to analyze six specific cases of the rod ejection transient--the four cases reported for the reload analysis of Surry 1 Cycle 5 (BOL HZP, BOL HFP, EOL HZP and EOL HFP), (Ref. 19), and two BOL cases (HZP and HFP) documented in the Surry positive moderator coefficient submittal (Ref 22).

Results from the Vepco analyses performed with the vendor's codes and techniques are compared to the vendor's results for the same analyses to demonstrate Vepco's ability to replicate vendor results. A comparison is then made between the licensing results obtained with the vendor methodology (using vendor codes) to those obtained with the Vepco methodology using the RETRAN code.

A comparison is also presented between the Vepco calculated results using the vendor and Vepco methodologies and results obtained by Vepco

using the vendor's three-dimensional space-time kinetics code. This comparison demonstrates the conservatism of the Vepco approach to the rod ejection analysis.

2 Vendor Licensing Methodology

The vendor licensing analysis of the rod ejection transient parallels that of Vepco in that the analysis may be broken into two main phases:

1. a core average nuclear power transient calculation, and
2. a transient thermal-hydraulics analysis of the core's hot spot.

Unlike Vepco, which uses a point kinetics model to derive a core average nuclear power history, the vendor uses a one-dimensional (1-D) space-time kinetics model, the TWINKLE code, (Ref. 18.) TWINKLE is a space-time neutron diffusion code which uses nuclear macroscopic cross sections generated by the vendor's steady state physics design codes to solve the neutron diffusion equations for two energy groups. The code's geometry may be specified in one, two or three dimensions with up to 2000 spatial points. Six delayed neutron groups are assumed. A detailed multiregion, transient fuel-clad-coolant heat transfer model is included for predicting pointwise Doppler and moderator feedback effects.

The vendor uses the 1-D axial geometry configuration for the rod ejection analysis (Ref. 17). Input for the model is specified to bound the conditions for a given nuclear power plant. This approach has been accepted for providing the core average power for the transient. The delayed neutron fraction, Doppler reactivity feedback, trip insertion characteristics, trip setpoints, and total trip reactivity worth are conservatively adjusted to fit the core conditions and plant characteristics for the specific analysis under consideration. The moderator feedback effects are adjusted by changing the value of the

soluble boron concentration. As with the Vepco methodology, a weighting factor is applied to the Doppler reactivity calculation to more accurately model the effects of severe flux redistribution in a three-dimensional geometry.

Despite having an explicit representation of the axial core geometry, the ejection of the rod is not modeled by perturbing the nuclear cross sections in a space-time dependent manner. Instead the code's eigenvalue is ramped over a 0.1 second time interval by a value equivalent to the assumed ejected rod worth. Upon reactor trip, (with an appropriate trip time delay), a modeling of the trip banks being inserted into the core is performed. This accounting of the axial effects is the major difference between the TWINKLE model and the RETRAN point kinetics model where in the latter the scram is based on a generic normalized trip reactivity table.

The vendor thermal-hydraulics analysis of the core hot spot is performed with a detailed fuel and clad transient heat transfer code, FACTRAN, (Ref. 9.) This code computes the transient temperature distribution in a cross section of a metal clad uranium dioxide fuel rod, and the heat flux at the surface of the rod, using as input the normalized core average power history predicted by the TWINKLE code and the local coolant conditions for the actual plant and core conditions under consideration. The fuel rod geometry is input for the particular plant being analyzed. A Zircaloy-water reaction is modeled, and all material properties are represented as functions of temperature. The fuel rod assumes an initial parabolic radial power generation.

The code uses the Dittus-Boelter or Jens-Lottes heat transfer correlations to determine the surface heat transfer characteristics before the onset of DNB. AT 0.1 seconds into the transient, the code is forced into DNB by specifying a conservative DNB heat flux, and the Bishop-Sandberg-Tong correlation is used to determine the film boiling heat transfer coefficient. The gap heat transfer coefficient is adjusted in order to provide agreement of the full power steady state temperature distribution with the distribution predicted by Westinghouse design fuel heat transfer codes. As the transient progresses, the gap heat transfer coefficient is ramped from its initial value to a very high value to model the gap closure expected to occur during the temperature transient (Ref. 17).

As with the Vepco methodology, the pre- and post-ejection design maximum total power peaking factor (Fq) is input. The value of Fq used by the code is assumed to increase from the pre-ejection value to the post-ejection value over a time interval of 0.1 seconds, and remain there for the duration of the transient.

In summary, the analytical techniques in modeling the transient of both Vepco and the vendor are similar. Table 4-1 presents an overview of the two methodologies. Table 4-2 presents the results obtained for the six comparison cases using the TWINKLE/FACTRAN codes with vendor methodology performed by both the vendor and by Vepco. The results from the vendor analysis are provided in Ref. 19 for the Surry 1 Cycle 5 reanalysis cases and Ref. 22 for the Surry +MTC cases. The close agreement between the results demonstrates Vepco's ability to perform the rod ejection

nalysis using the TWINKLE/FACTRAN codes and vendor methodology.

TABLE 4-1

COMPARISON OF VENDOR/VEPCO LICENSING METHODOLOGIES

I. Core Average Power History Calculation

Item	Vepco	Vendor
Principal code	RETRAN	TWINKLE
Physics model	Point kinetics	1-D space-time kinetics
Rod ejection	Reactivity ramp	Reactivity ramp
Doppler feedback	Reactivity vs. temp. table	Calculated from nuclear cross sections
Moderator feedback	Moderator temp. coefficient	Calculated from nuclear cross sections
Trip reactivity insertion	Reactivity vs. time table	Calculated from nuclear cross sections
Delayed neutron groups	6	6
Doppler reactivity weighting factor	Yes	Yes
Fuel heat transfer model	Yes	Yes
HGAP	Constant	Variable

TABLE 4-1 (cont.)

COMPARISON OF VENDOR/VEPCO LICENSING METHODOLOGIES

II. Hot Spot Thermal-hydraulics Calculation

Item	Vepco	Vendor
Principal code	RETRAN	FACTRAN
Number of fuel pellet regions	10	6
Fq history	Ramped over 0.1 sec	Ramped over 0.1 sec
Zircaloy-water reaction	Yes	Yes
Pre-DNB heat transfer correlation	Thom	Dittus-Boelter or Jens Lottes
Post-DNB heat transfer correlation	Bishop-Sandberg-Tong	Bishop-Sandberg-Tong
Forced DNB at 0.1 sec	Yes	Yes
Material properties	Function of temp.	Function of temp.
Gap closure modeled	Yes	Yes

TABLE 4-2

VENDOR/VEPCO ANALYSIS RESULTS USING VENDOR METHODOLOGIES

(TWINKLE-FACTRAN CODES)

The vendor calculated value is followed by the Vepco calculated value separated by a slash (/).

Parameter	Surry 1 Cycle 5 Values BOL HZP	BOL HFP	EOL HZP	EOL HFP
Fuel Pellet Melting (%)	0/0	<10/<10	0/0	<10/<10
Max. Fuel Average Temp. (°F)	3430/3432	4185/4210	3373/2949	3844/3794
Max. Clad Average Temp. (°F)	2500/2520	2488/2482	2317/2169	2151/2167
Max. Fuel Enthalpy (cal/g)	145/145	185/186	142/121	166/164

Parameter	Surry +MTC Values BOL HZP	BOL HFP
Max. Fuel Average Temp. (°F)	2883/3394	3639/3630
Max. Fuel Center Temp. (°F)	3353/3918	4958/4874
Max. Clad Average Temp. (°F)	2123/2488	2013/2079

3 Verification With Licensing Analyses

Six cases of the rod ejection transient were compared for benchmarking the Vepco methodology with that of the vendor. Assumptions for each comparison calculation were matched as closely as possible between the two methodologies.

Due to the greater flexibility of specifying reactivity effects in the Vepco RETRAN model, the steady state values of the Doppler defect (without any additional weighting factor), moderator temperature coefficient, and total trip rod worth were first calculated by Vepco using the TWINKLE code; these calculations used nuclear cross section input chosen to give conservative values for these parameters. These values were then used in the corresponding RETRAN point kinetics calculation. An exception was the reactivity feedback weighting factor used for each methodology. For the Vepco calculation, the weighting factor (PWF) was obtained from Figure 2-6 based on the value of Fq assumed for each analysis. The corresponding weighting factor used in TWINKLE was derived from the data in Ref. 19 for the Surry 1 Cycle 5 reanalyses cases and from Ref. 22 for the Surry +MTC cases. Table 4-3 presents a summary of the input conditions for the six cases.

TABLE 4-3

VERIFICATION COMPARISON CASES

Parameter	S1C5 BOL HZP	S1C5 BOL HFP	S1C5 EOL HZP	S1C5 EOL HFP	S+MTC BOL HZP	S+MTC BOL HFP
Ejected rod worth (pcm)	900	300	840	300	920	160
Delayed neutron fraction	.0055	.0055	.0044	.0044	.0059	.0059
Pre-ejection Fq	NA	2.55	NA	2.55	NA	2.55
Post-ejection Fq	15.2	5.46	15.2	5.46	15.2	4.8
Zero to full power Doppler defect (pcm)	-1088	-1088	-850	-850	-1088	-1088
RETRAN moderator temperature coefficient (pcm/°F)	7.7	5.4	-20.0	-30.8	7.7	5.4
Number of operating pumps	2	3	2	3	2	3
TWINKLE reactivity feedback weighting factor	2.725	1.3	2.32	1.3	2.725	1.2
RETRAN Power Weighting Factor	2.95	1.4	2.95	1.4	2.95	1.25

Notes: S1C5 = Surry 1 Cycle 5
S+MTC = Surry Positive Moderator Coefficient
NA = not applicable
pcm = percent mille

3.1 Core Average Power History

Results from the nuclear power transient analyses for the six cases are graphed in Figures 4-1 through 4-6. Each figure presents a normalized power history comparison from the TWINKLE 1-D and RETRAN point kinetics calculations and the corresponding normalized energy release curves. As with the sensitivity study results presented in Section 3, a value of 1.0 on the "normalized power" axis represents a value of 100% full power while a value of 1.0 on the "energy release" axis represents one-full-power second of energy. For the HZP cases, both the TWINKLE and RETRAN models assumed an initial normalized full power level of 1.0E-9.

The three HZP cases, (Figures 4-1, 4-3 and 4-5), show excellent agreement between the power histories. The minor differences occurring after the initiation of the trip can be explained by the different methods used in modeling the trip reactivity insertion. Whereas RETRAN uses a table of normalized trip worth insertion versus time which is identical for each case, TWINKLE calculates the change in core reactivity through an axial variation of nuclear cross sections with time. This is a more realistic modeling of the insertion of the control banks and takes into account the impact of the axial flux shape on the rate of change of reactivity.

Although the maximum normalized core power predicted by the two methods may differ for a specific case, the effect is not significant due to sharpness of the power peaking portion of the curve. This is verified in the plot of total energy release where both models track reasonably well. After the trip insertion begins, the normalized power curve

predicted by the TWINKLE code asymptotically approaches a lower power level relative to that predicted by RETRAN and this accounts for the eventual divergence of the energy release curves.

For the three HFP cases, (Figures 4-2, 4-4 and 4-6), the divergence between the two models is more pronounced. Typically, TWINKLE predicts a higher peak power level and maintains the power at a relatively high value longer than the RETRAN model. As with the HZP cases, a deviation in the predictions is noticable during the trip portion of the transient with TWINKLE once again settling out at a lower relative power level. The energy release curves show reasonable agreement at the most critical part of the transient between the two models with a larger divergence in the energy release predictions occurring only after the power level of the core has dropped to a relatively low value.

Comparison of the trends inherent in the three HFP cases show the effect of the delayed neutron fraction and ejected rod worth on the transient. The BOL and EOL cases for the Surry 1 Cycle 5 reanalysis, (Figures 4-2 and 4-4), both assumed the same ejected rod worth. The only significant difference between the two cases was in the delayed neutron fractions, (0.0055 for BOL versus 0.0044 for EOL), with the higher value causing the power curve to hang, (this is due more to delay in the core response than to changes in core conditions.) That this effect is due entirely to the delayed neutron fraction is especially obvious from the RETRAN analysis where it was the only parameter which changed between the two calculations.

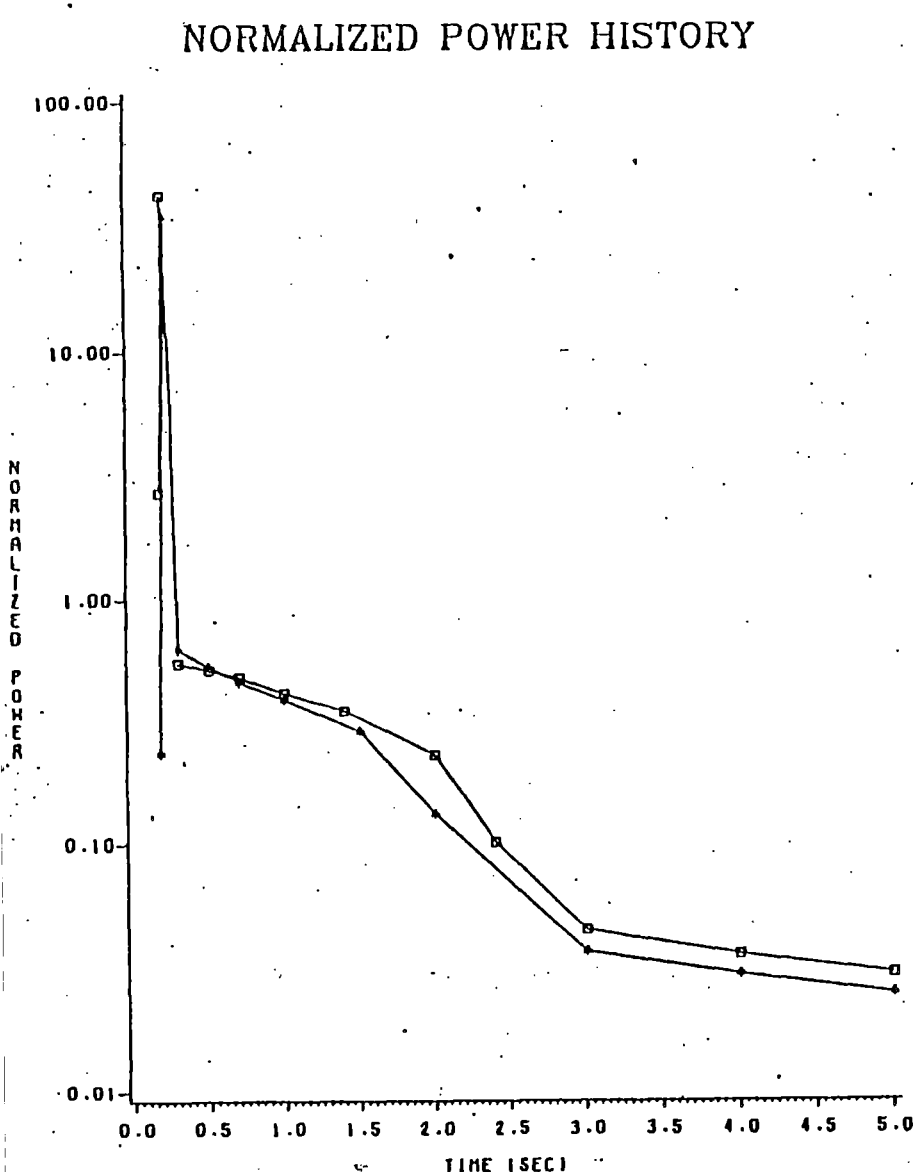
A comparison between the two BOL HFP cases, (Figures 4-2 and 4-6),

demonstrate a similar trend. Here, the difference in the values of the delayed neutron fractions is less pronounced than in the previous example, but the S+MTC case has a significantly lower ejected rod worth than the Surry 1 Cycle 5 case. This results in less reactivity insertion into the core which in turn causes less of a rise in fuel temperature and therefore a less pronounced negative Doppler reactivity feedback effect to turn the transient's power excursion around.

Considering the differences in the two models used, the power history predictions compare well. Both show the same characterics for the transients and demonstrate reasonably close agreement in actual magnitude.

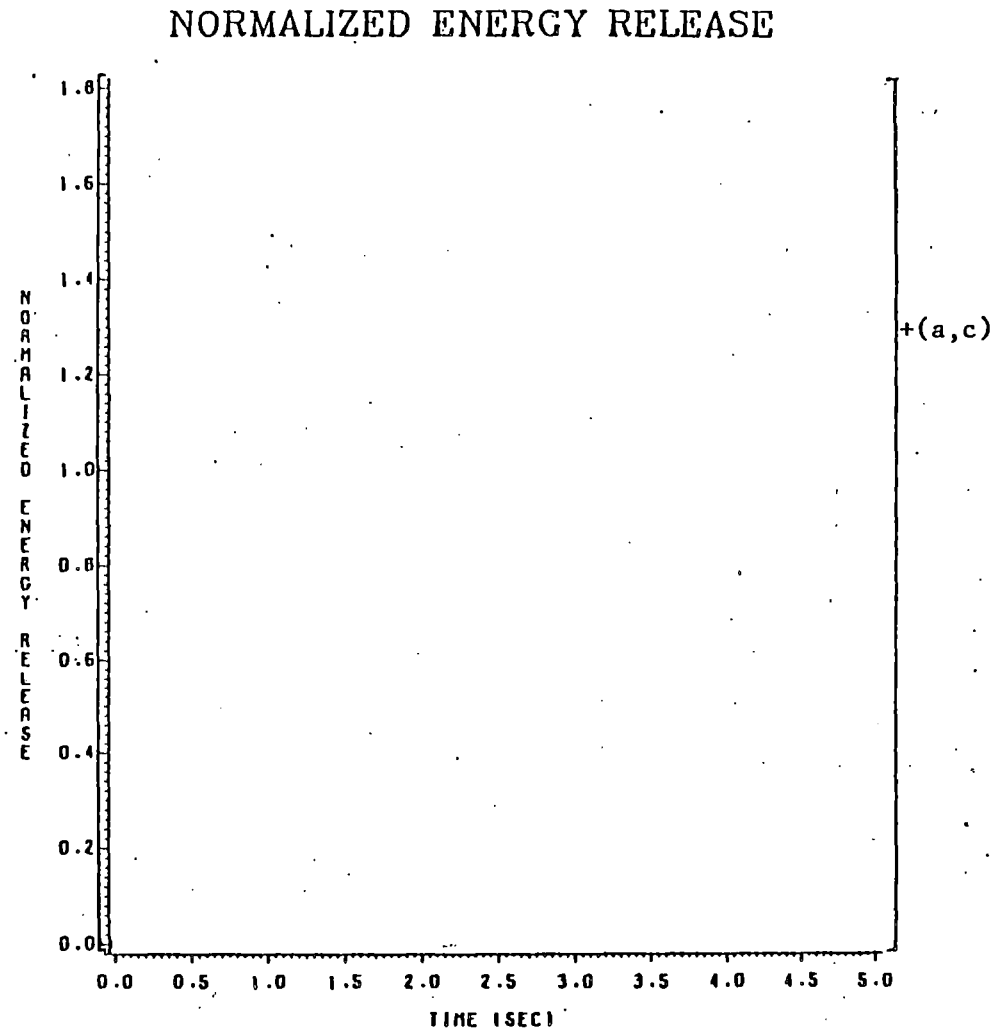
FIGURE 4-1

NUCLEAR POWER TRANSIENT - S1C5 BOL HZP



STAR: THINKLE 1-D MODEL
SQUARE: RETRAN POINT KINETICS MODEL

FIGURE 4-1A



+(a,c)

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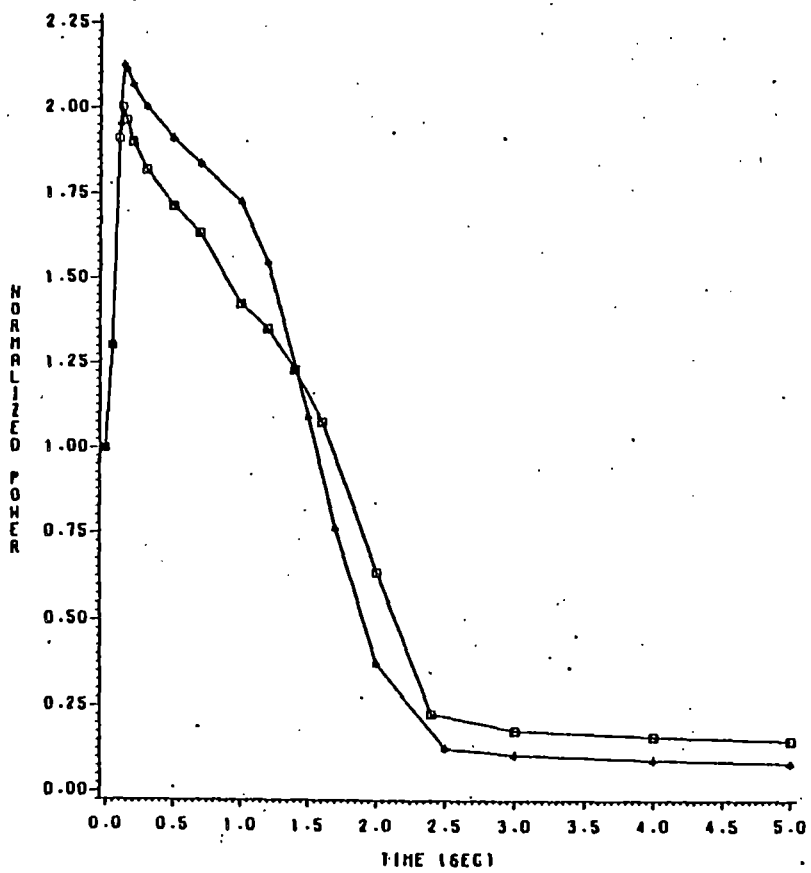
STAR: THINKLE 1-D MODEL
SQUARE: RETRAN POINT KINETICS MODEL

FIGURE 4-1B

FIGURE 4-2

NUCLEAR POWER TRANSIENT - S1C5 BOL HFP

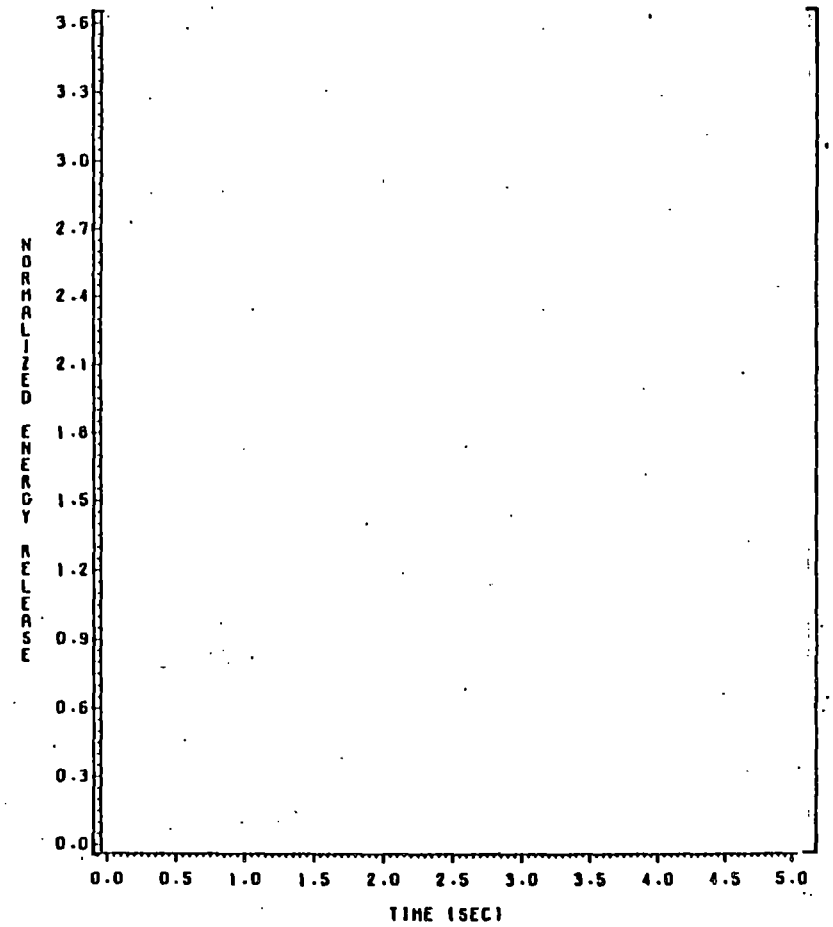
NORMALIZED POWER HISTORY



STAR: TWINKLE 1-D MODEL
SQUARE: RETRAN POINT KINETICS MODEL

FIGURE 4-2A

NORMALIZED ENERGY RELEASE



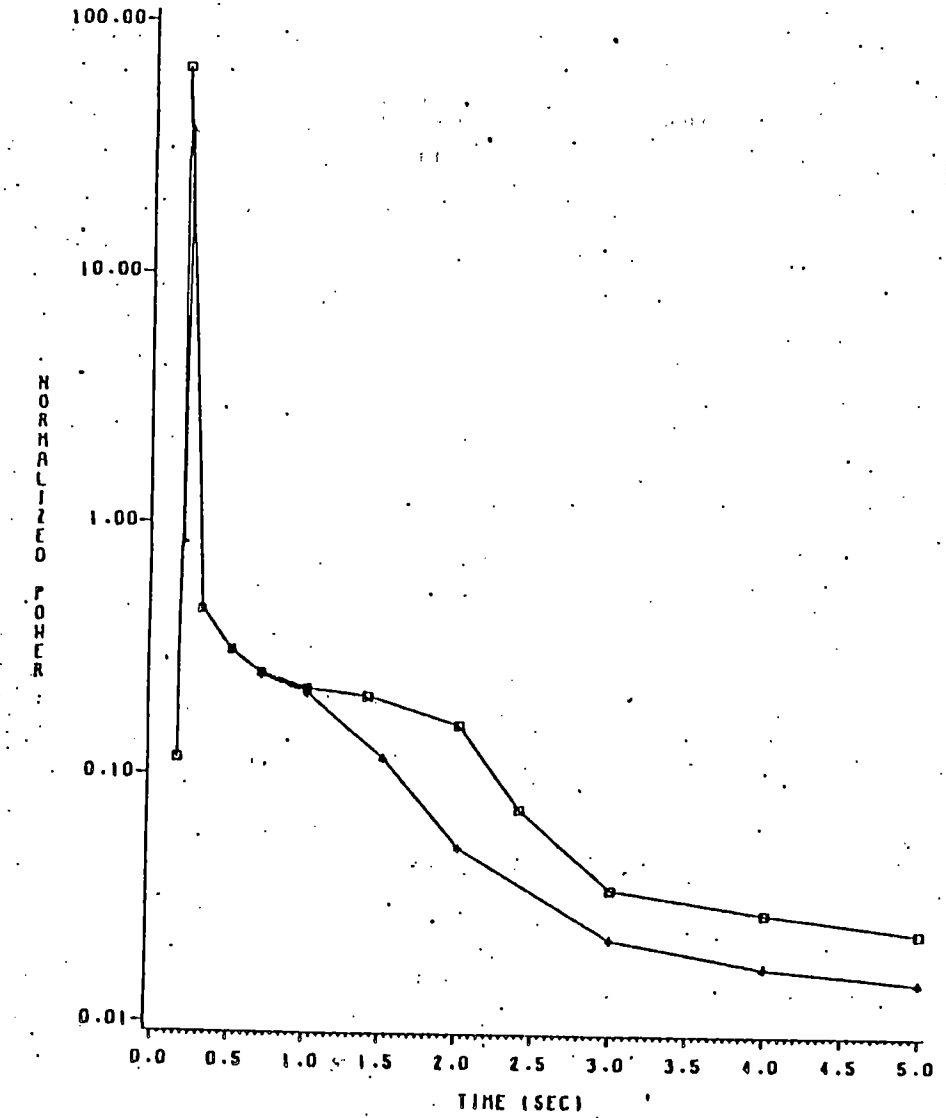
+(a,c)

WESTINGHOUSE PROPRIETARY CLASS 2

STAR: TWINKLE 1-D MODEL
SQUARE: RETRAN POINT KINETICS MODEL

FIGURE 4-2B

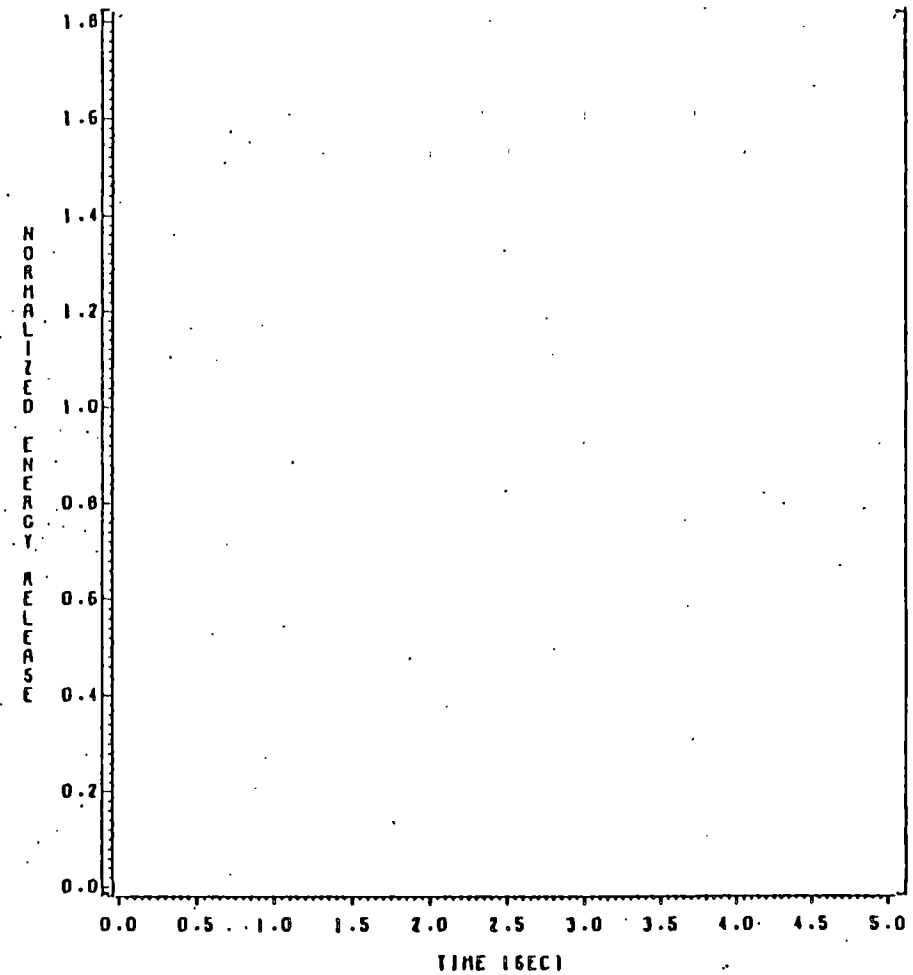
NORMALIZED POWER HISTORY



STAR: THINKLE 1-D MODEL
SQUARE: RETRAN POINT KINETICS MODEL

FIGURE 4-3A

NORMALIZED ENERGY RELEASE



+(a,c

STAR: THINKLE 1-D MODEL
SQUARE: RETRAN POINT KINETICS MODEL

FIGURE 4-3B

FIGURE 4-4

NUCLEAR POWER TRANSIENT S1C5 EOL HFP

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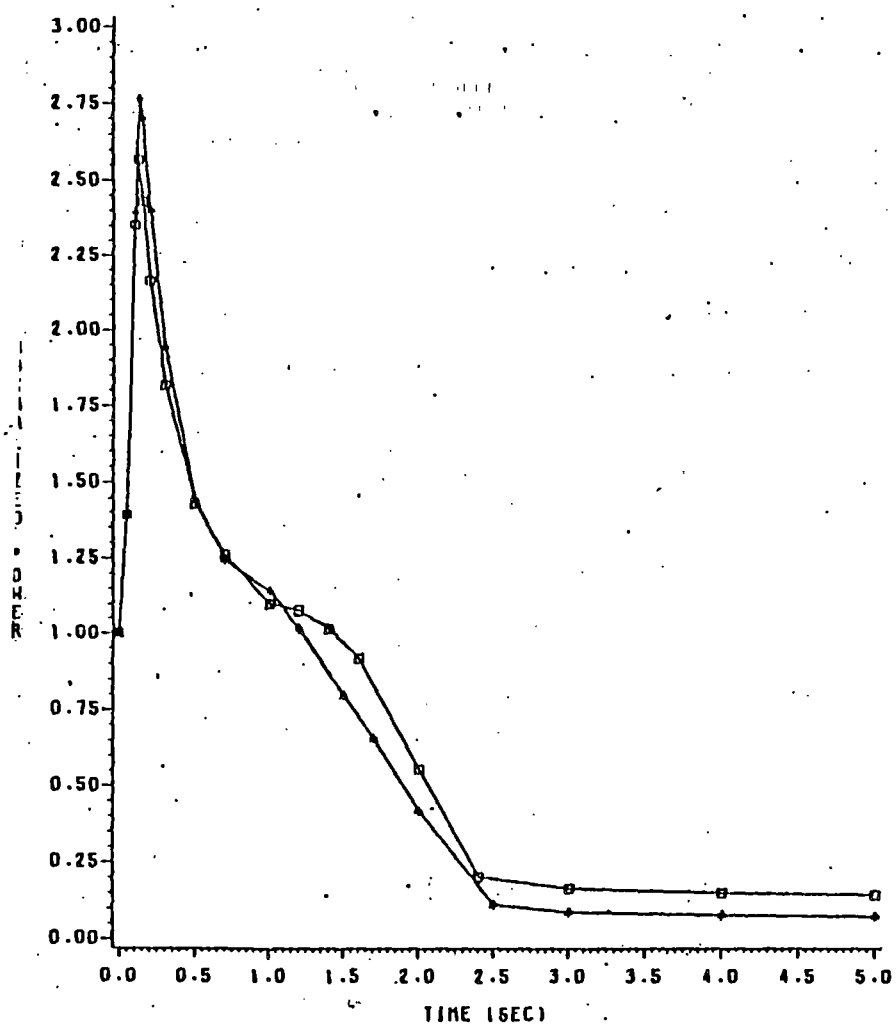
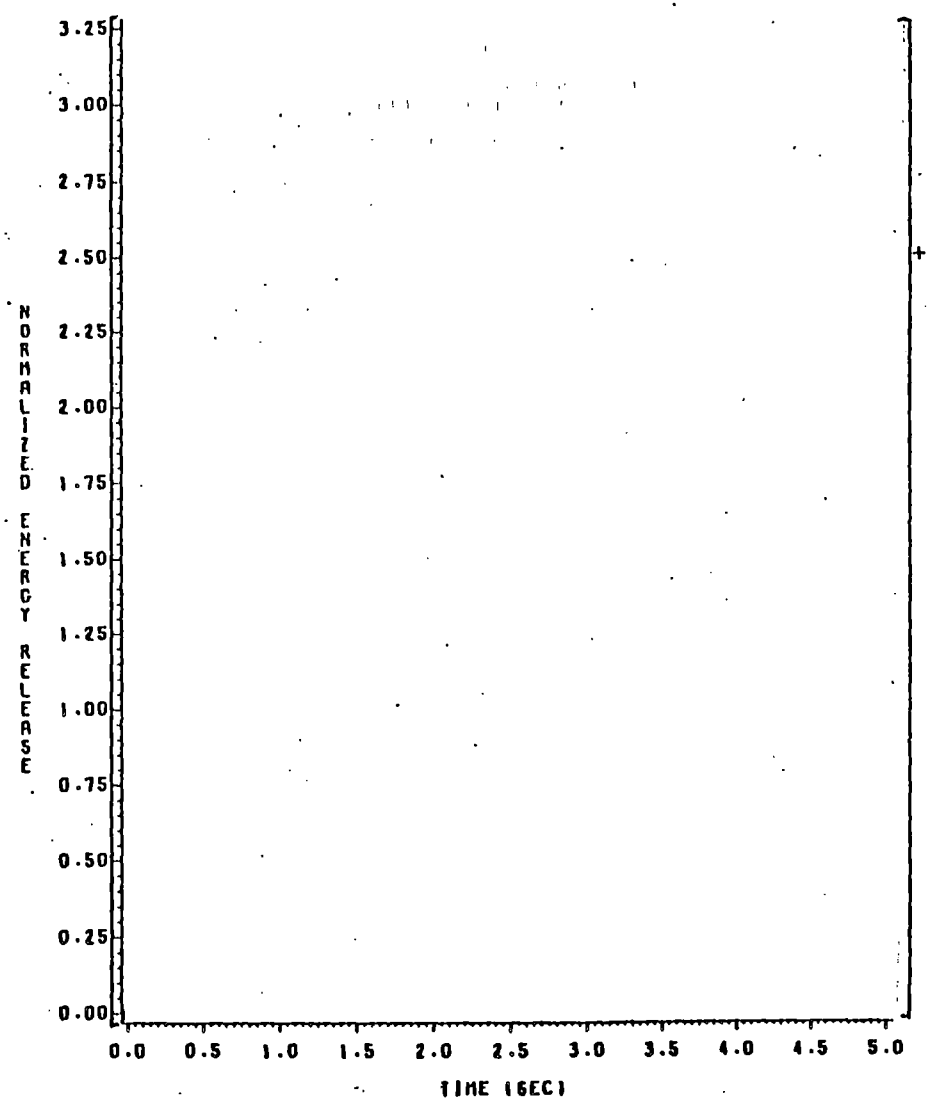
NORMALIZED POWER HISTORY



STAR: TWINKLE 1-D MODEL
SQUARE: RETRAN POINT KINETICS MODEL

FIGURE 4-4A

NORMALIZED ENERGY RELEASE



+(a,c)

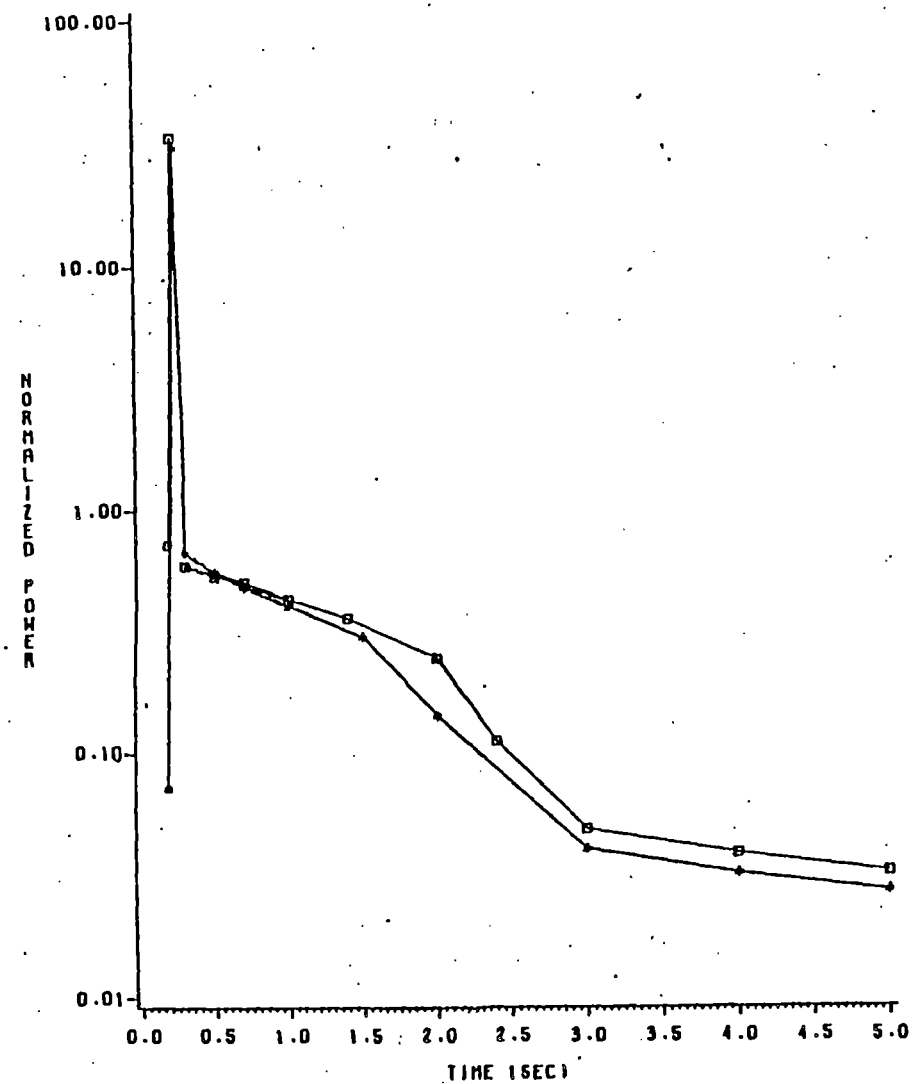
STAR: TWINKLE 1-D MODEL
SQUARE: RETRAN POINT KINETICS MODEL

FIGURE 4-4B

FIGURE 4-5

NUCLEAR POWER T[illegible]ENT - S+MTC BOL HZP

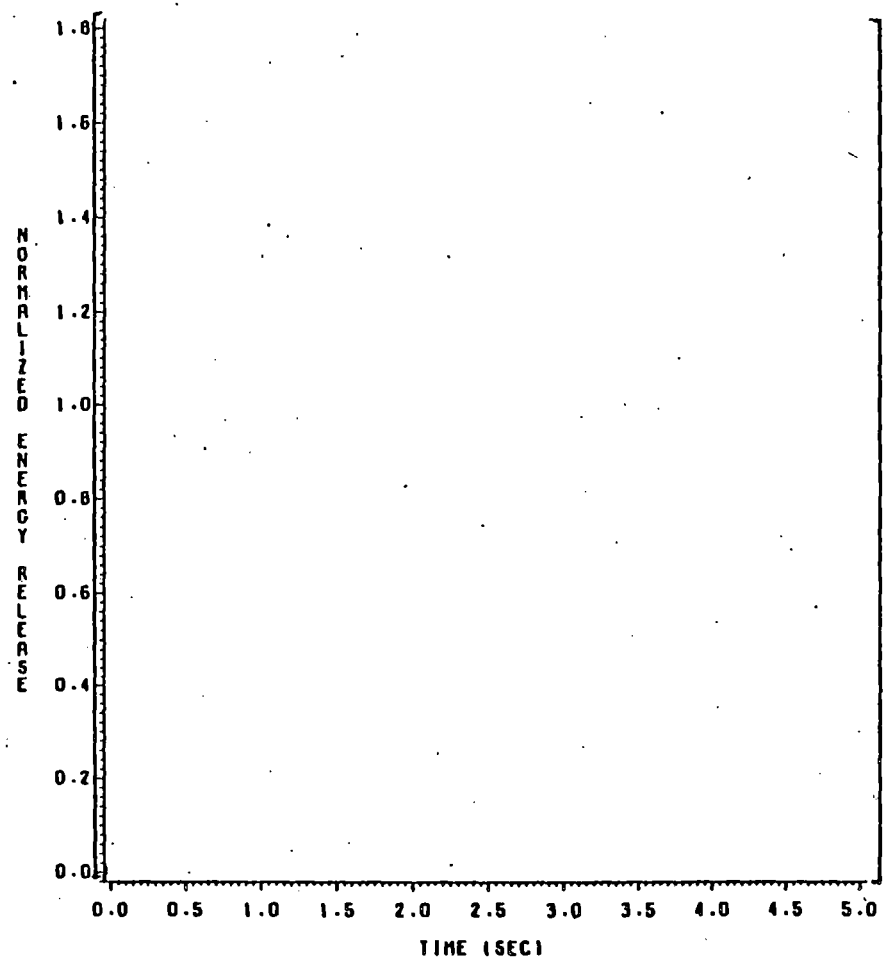
NORMALIZED POWER HISTORY



STAR: THINKLE 1-D MODEL
SQUARE: RETRAN POINT KINETICS MODEL

FIGURE 4-5A

NORMALIZED ENERGY RELEASE



+(a,c)

WESTINGHOUSE PROPRIETARY CLASS 2

STAR: THINKLE 1-D MODEL
SQUARE: RETRAN POINT KINETICS MODEL

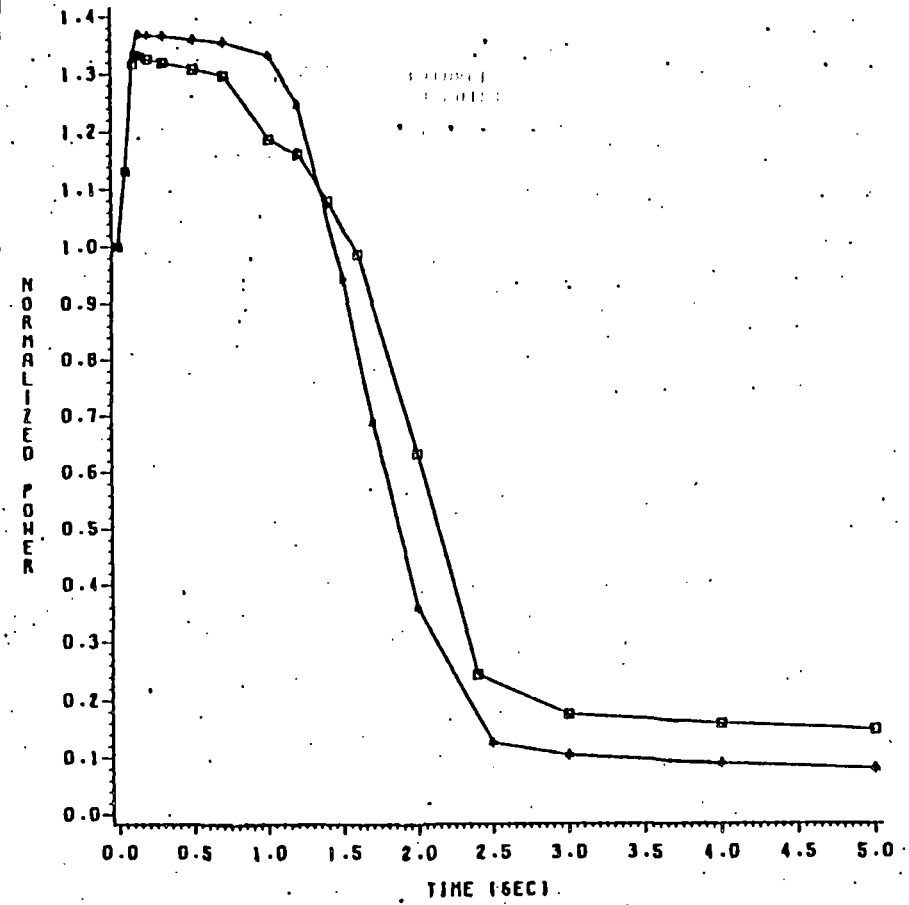
FIGURE 4-5B

FIGURE 4-6

NUCLEAR POWER TRANS[illegible] - S+MTC BOL HFP

WESTINGHOUSE PROPRIETARY CLASS 2

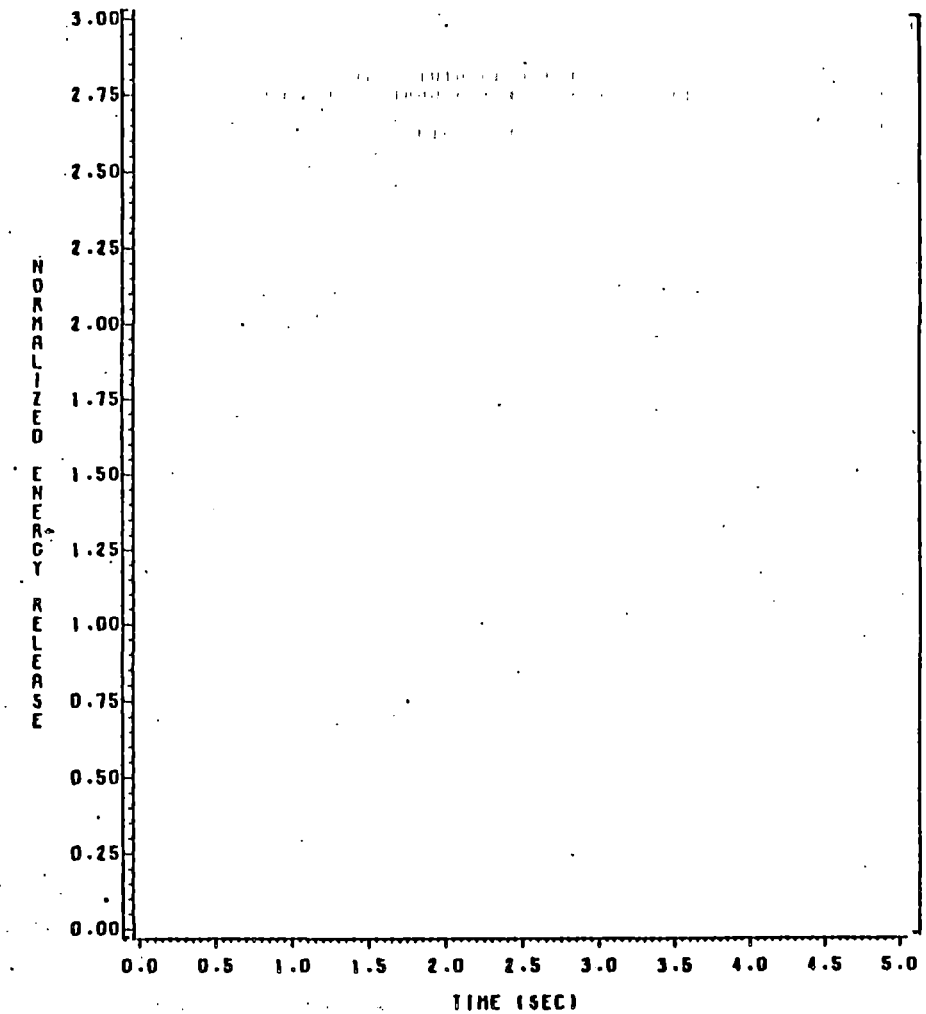
NORMALIZED POWER HISTORY



STAR: THINKLE 1-D MODEL
SQUARE: RETRAN POINT KINETICS MODEL

FIGURE 4-6A

NORMALIZED ENERGY RELEASE



+(a,c)

STAR: THINKLE 1-D MODEL
SQUARE: RETRAN POINT KINETICS MODEL

FIGURE 4-6B

3.2 Hot Spot Analysis

Results of the comparisons between the FACTRAN and RETRAN Hot Spot Model predictions for the hot spot transient are presented in Figures 4-7 through 4-12 and Table 4-4. All FACTRAN calculations were performed based on the appropriate TWINKLE 1-D model predicted core average power histories, while RETRAN Hot Spot Model calculations were performed based on the predicted RETRAN point kinetics power histories. Both hot spot models used identical assumptions as to the Fq history curve input; i.e., between a time of zero and 0.1 seconds the Fq value was linearly ramped from the pre-ejection value to the post-ejection value and held there for the remainder of the transient.

Figures 4-7 through 4-9 plot the fuel centerline transients at the hot spot location. The curves for the HZP cases demonstrate good agreement and follow the trends predicted by the TWINKLE/RETRAN core average power history predictions--i.e., RETRAN predicts a higher fuel temperature than FACTRAN later into the transient just as it predicts a higher power level than TWINKLE at that point. In none of the HZP cases did any fuel melt occur.

The corresponding HFP cases (Figures 4-7 through 4-9) demonstrate more deviation, especially in the two Surry 1 Cycle 5 cases where fuel melt occurred. For the RETRAN cases, upon reaching the assumed fuel melting temperature, (4900 °F for the BOL case and 4800 °F for the EOL case), the temperature remained constant for some time due to the high heat of fusion required to melt the fuel, (Ref. 8). Although FACTRAN assumed the same fuel melting temperatures as RETRAN, it apparently has a smaller

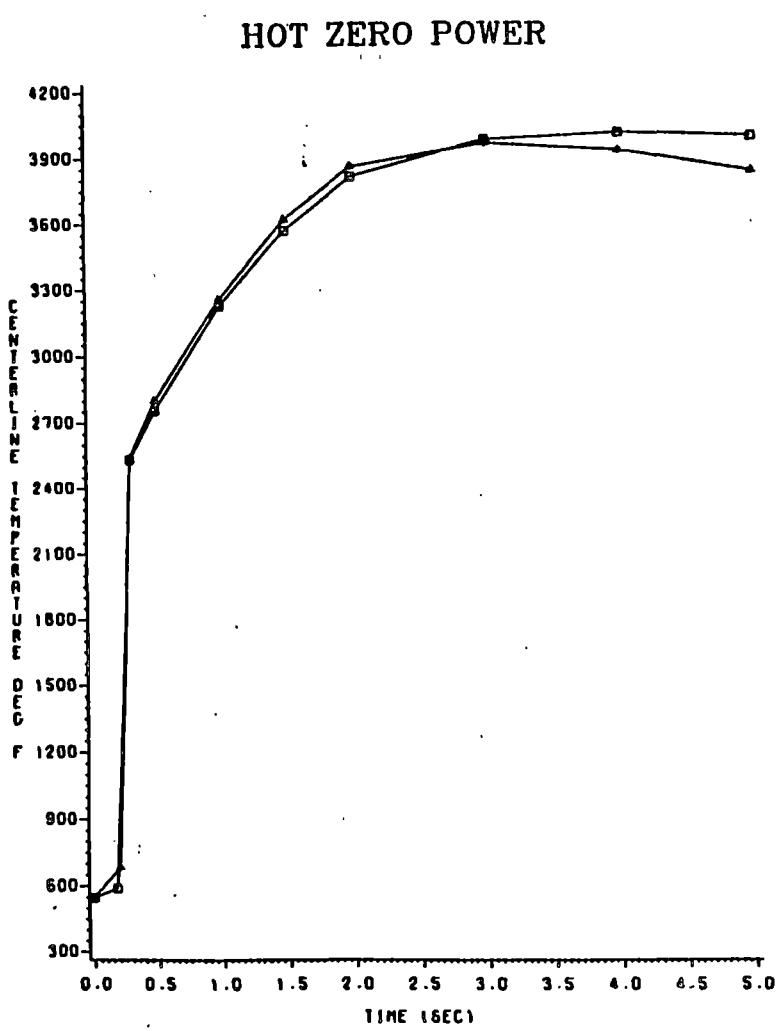
lue for the heat of fusion of uranium dioxide. Thus the FACTRAN temperature can at first level off at the melting point, and resume its rise sooner than the RETRAN Hot Spot model temperature. The main differences between the two predictions appear to be in the material properties tables used by the two codes. To check this, the TWINKLE power history used by FACTRAN was input to the RETRAN Hot Spot Model. The RETRAN predicted temperature transient characteristics were consistent with those predicted by the RETRAN Hot Spot Model using a RETRAN point kinetics power history input.

Figures 4-10 through 4-12 are comparisons of the outer clad temperature transients predicted by the vendor and Vepco models. Again agreement is better for the HZP cases, although for all cases the curves show excellent convergence late in the transient. The slight deviations between the predictions at the very start of the transient (where the FACTRAN predicted clad temperature actually decreases in value) are due to modeling differences between the two models of the gap and surface heat transfer coefficients before the onset of film boiling.

Table 4-4 is a summary of the maximum values of the hot spot temperatures, average enthalpies and percentage of fuel melt for the two models. As described above, two Surry 1 Cycle 5 HFP cases experience fuel melt. For these cases, the FACTRAN temperature rises above the assumed melting temperature while the RETRAN temperature has yet to overcome the heat of fusion input into that model's material properties tables. The FACTRAN code predicts an actual percentage of fuel melt while RETRAN predicts only an upper boundary based on Table 2-6.

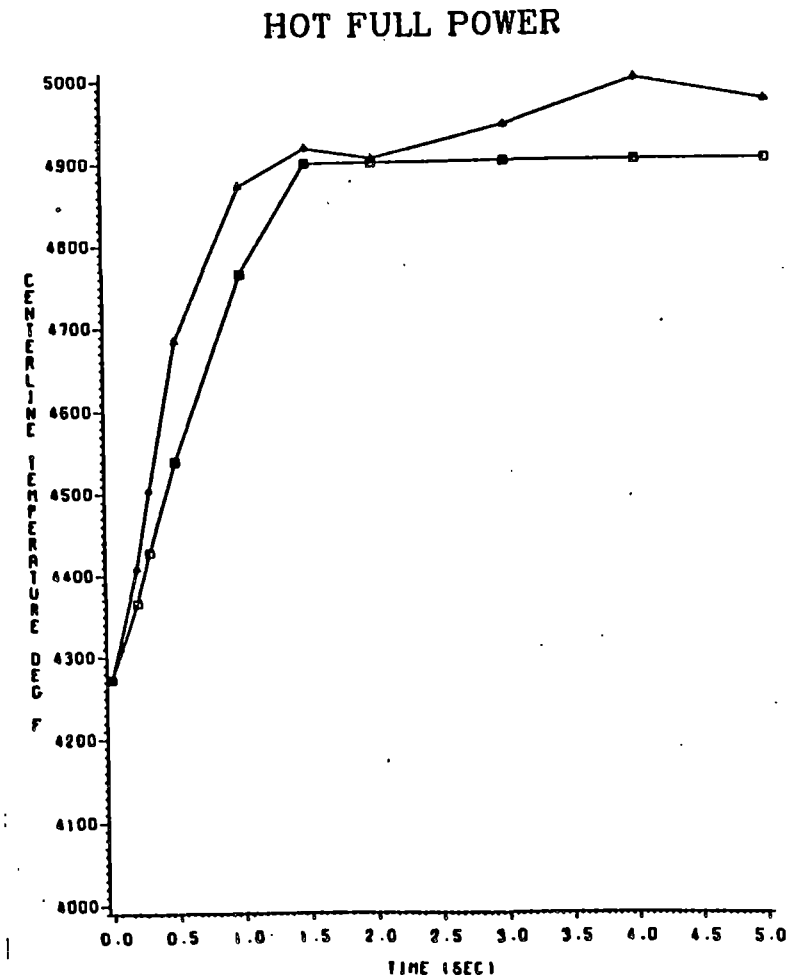
FIGURE 4-7

HOT SPOT FUEL CENTERLINE TEMPERATURE TRANSIENTS - S1C5 BOL CASES



STAR: FACTRAN CODE
SQUARE: RETRAN HOT SPOT MODEL

FIGURE 4-7A



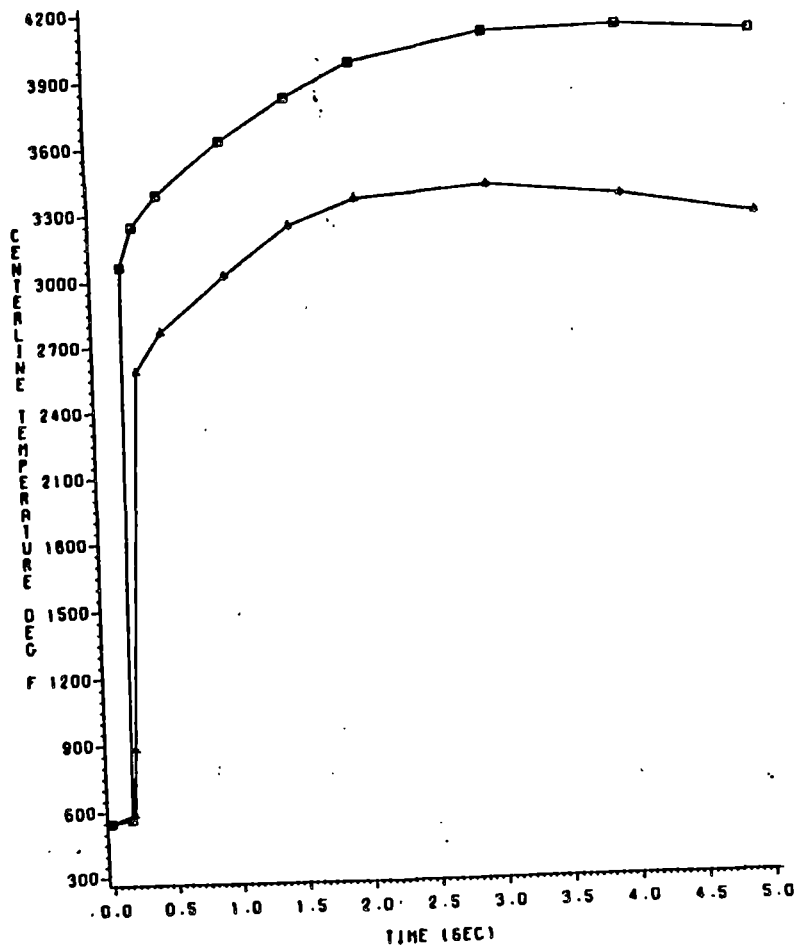
STAR. FACTRAN CODE
SQUARE. RETRAN HOT SPOT MODEL

FIGURE 4-7B

FIGURE 4-8

HOT SPOT FUEL CENTERLINE TEMPERATURE TRANSIENTS - S1C5 EOL CASES

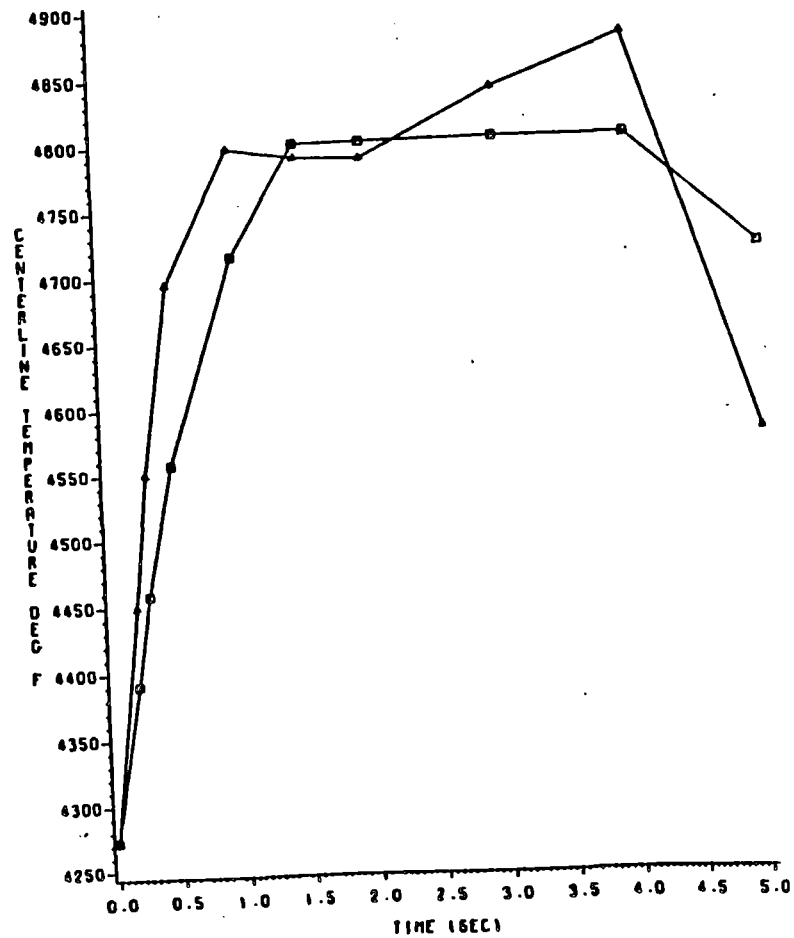
HOT ZERO POWER



STAR: FACTRAN CODE
SQUARE: RETRAN HOT SPOT MODEL

FIGURE 4-8A

HOT FULL POWER



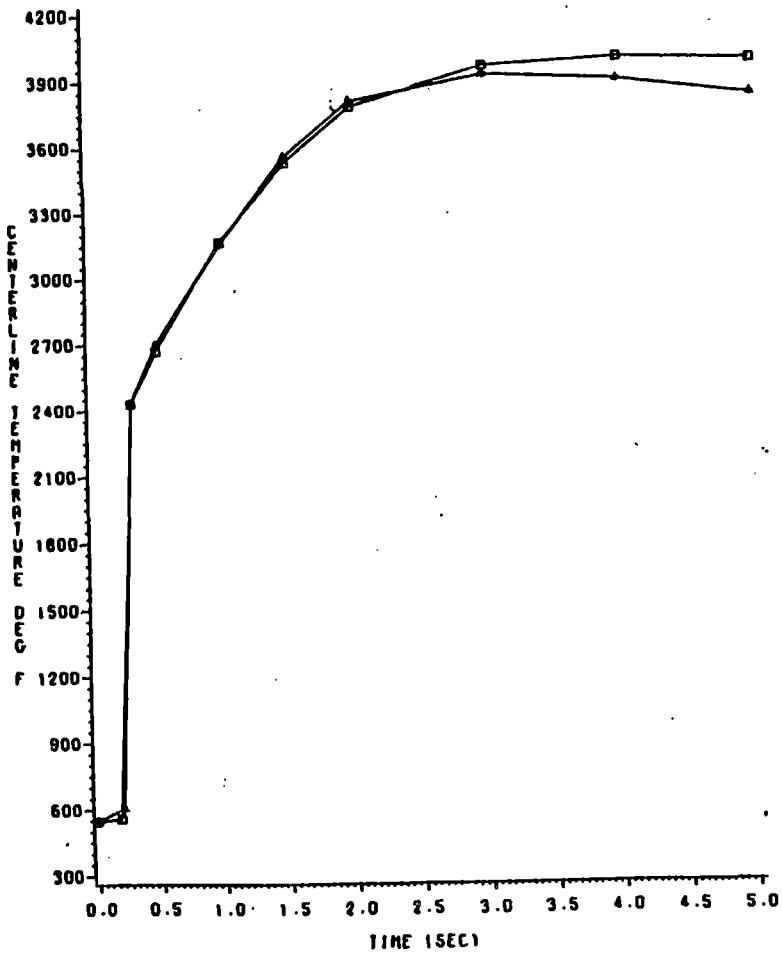
STAR. FACTRAN CODE
SQUARE. RETRAN HOT SPOT MODEL

FIGURE 4-8B

FIGURE 4-9

HOT SPOT FUEL CENTERLINE TEMPERATURE TRANSIENTS - S+MTC CASES

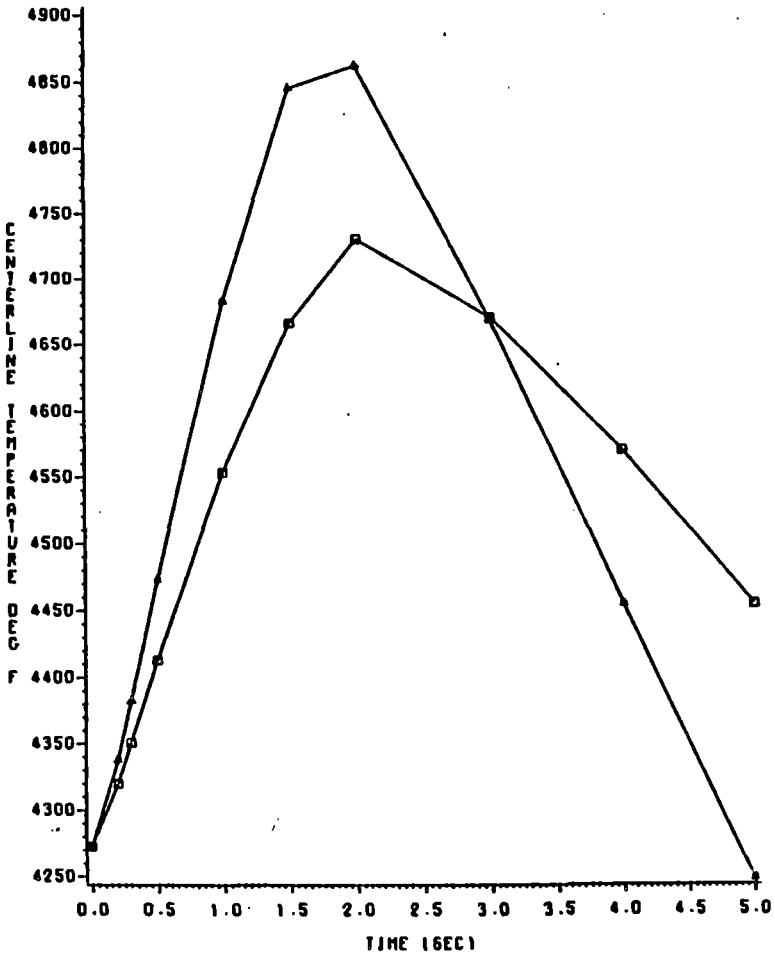
HOT ZERO POWER



STAR: FACTRAN CODE
SQUARE: RETRAN HOT SPOT

FIGURE 4-9A

HOT FULL POWER

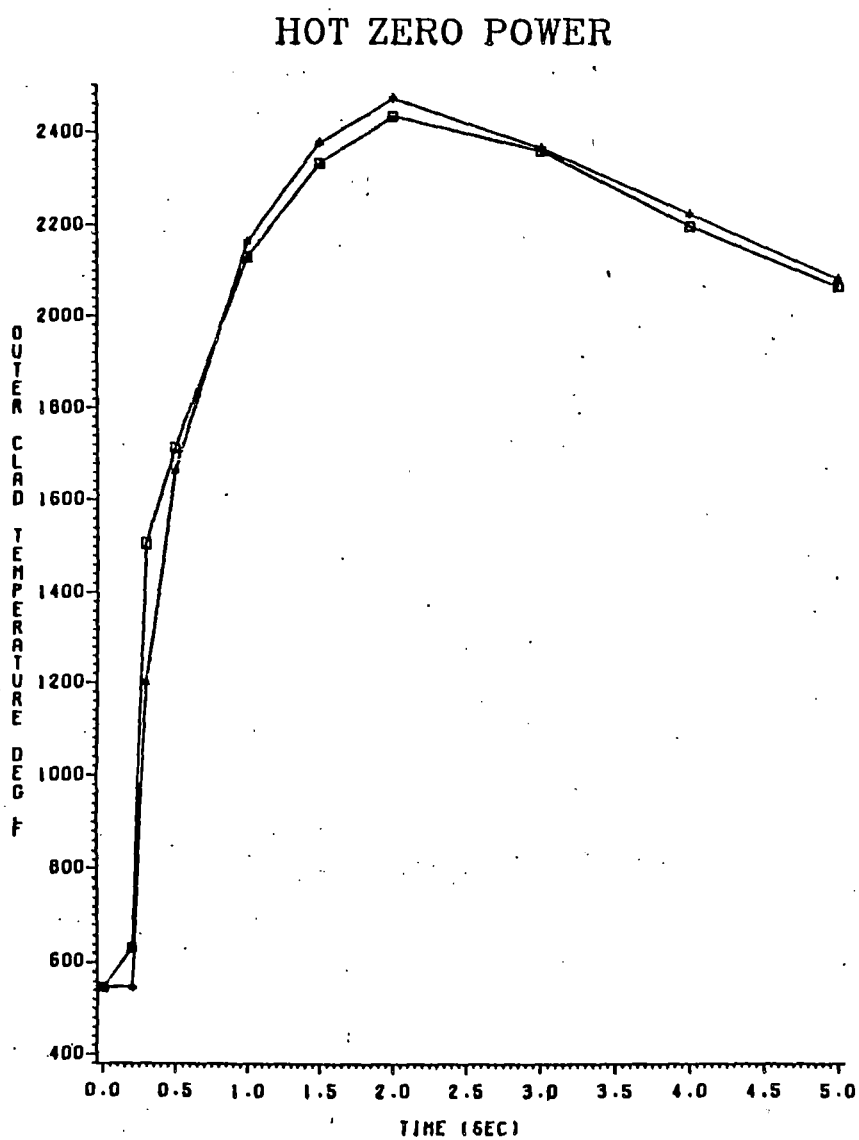


STAR: FACTRAN CODE
SQUARE: RETRAN HOT SPOT MODEL

FIGURE 4-9B

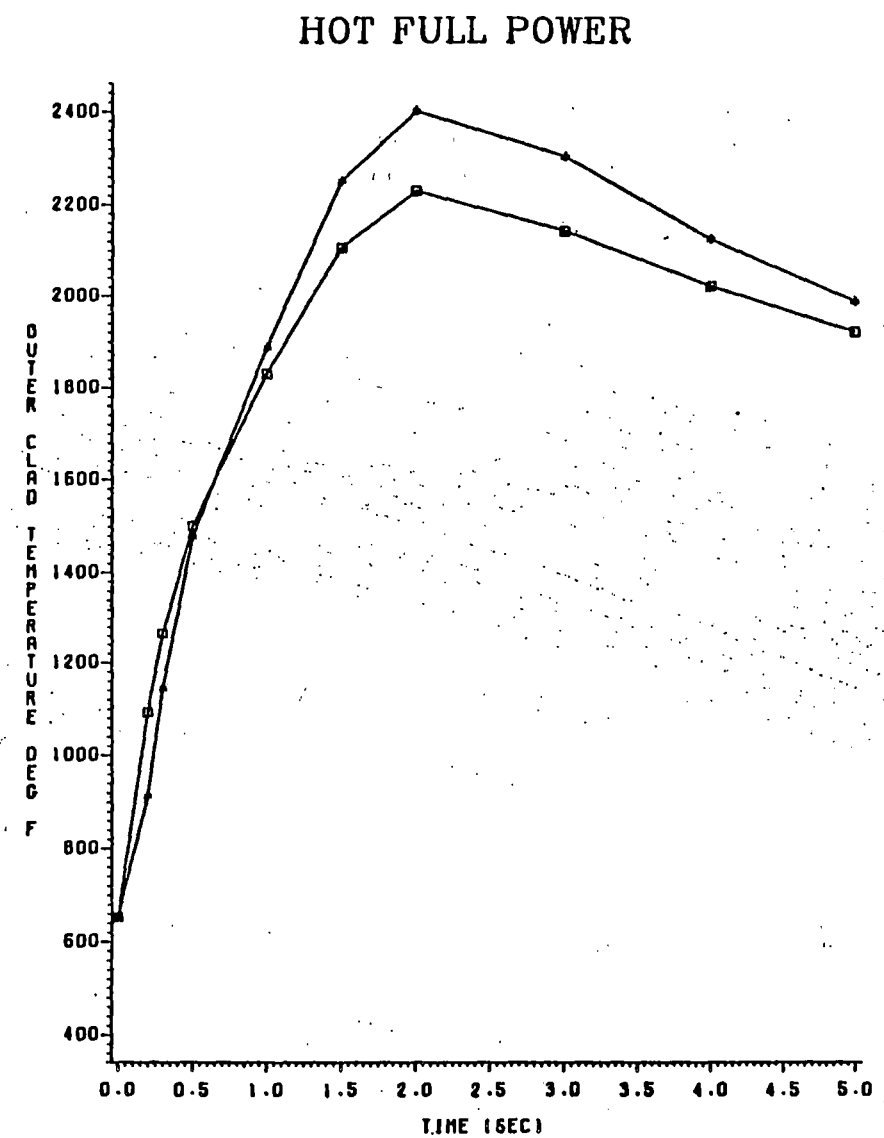
FIGURE 4-10

HOT SPOT FUEL OUTER CLAD TEMPERATURE TRANSIENTS - S1C5 BOL CASES



STAR: FACTRAN CODE
SQUARE: RETRAN HOT SPOT MODEL

FIGURE 4-10A



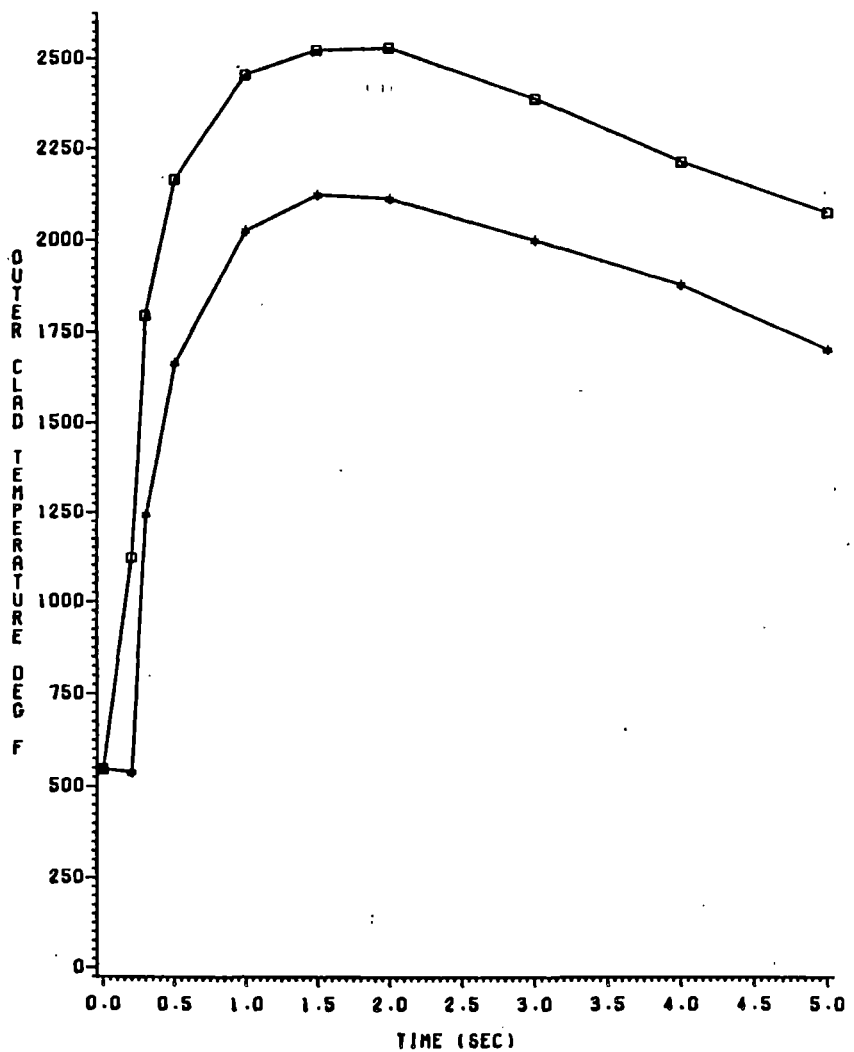
STAR: FACTRAN CODE
SQUARE: RETRAN HOT SPOT MODEL

FIGURE 4-10B

FIGURE 4-11

HOT SPOT FUEL OUTER CLAD TEMPERATURE TRANSIENTS - S1C5 EOL CASES

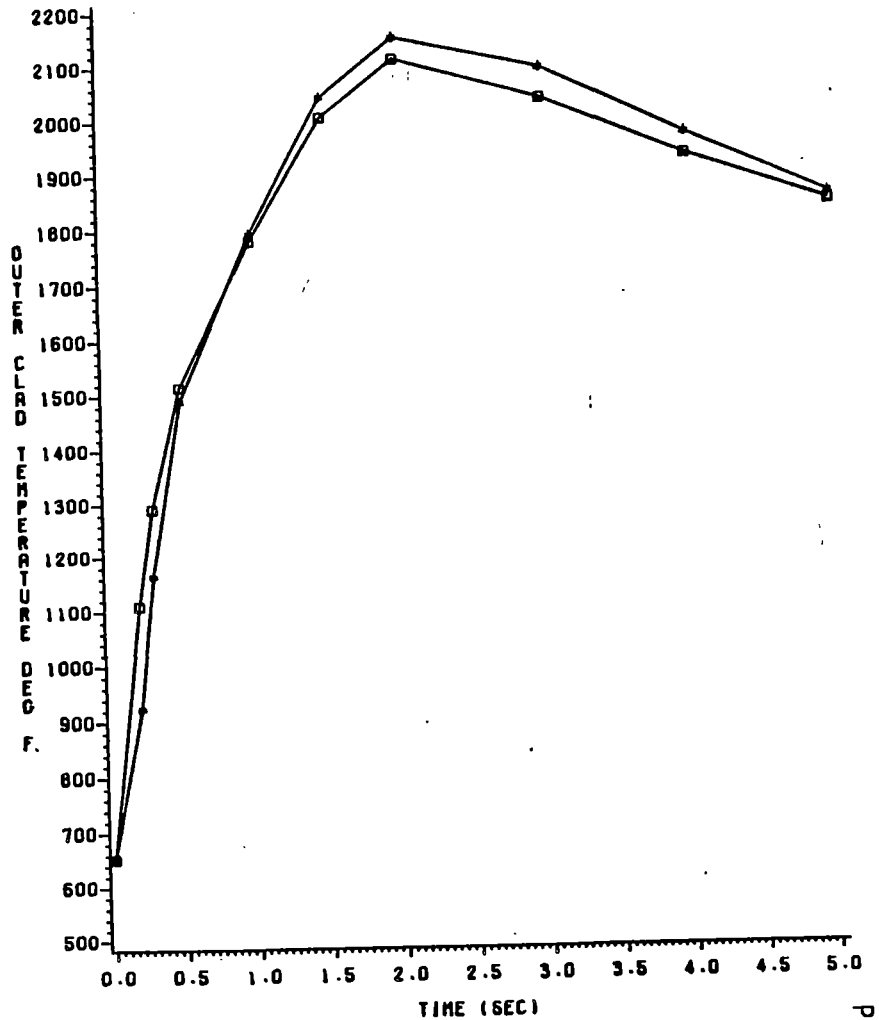
HOT ZERO POWER



STAR: FACTRAN CODE
SQUARE: RETRAN HOT SPOT MODEL

FIGURE 4-11A

HOT FULL POWER

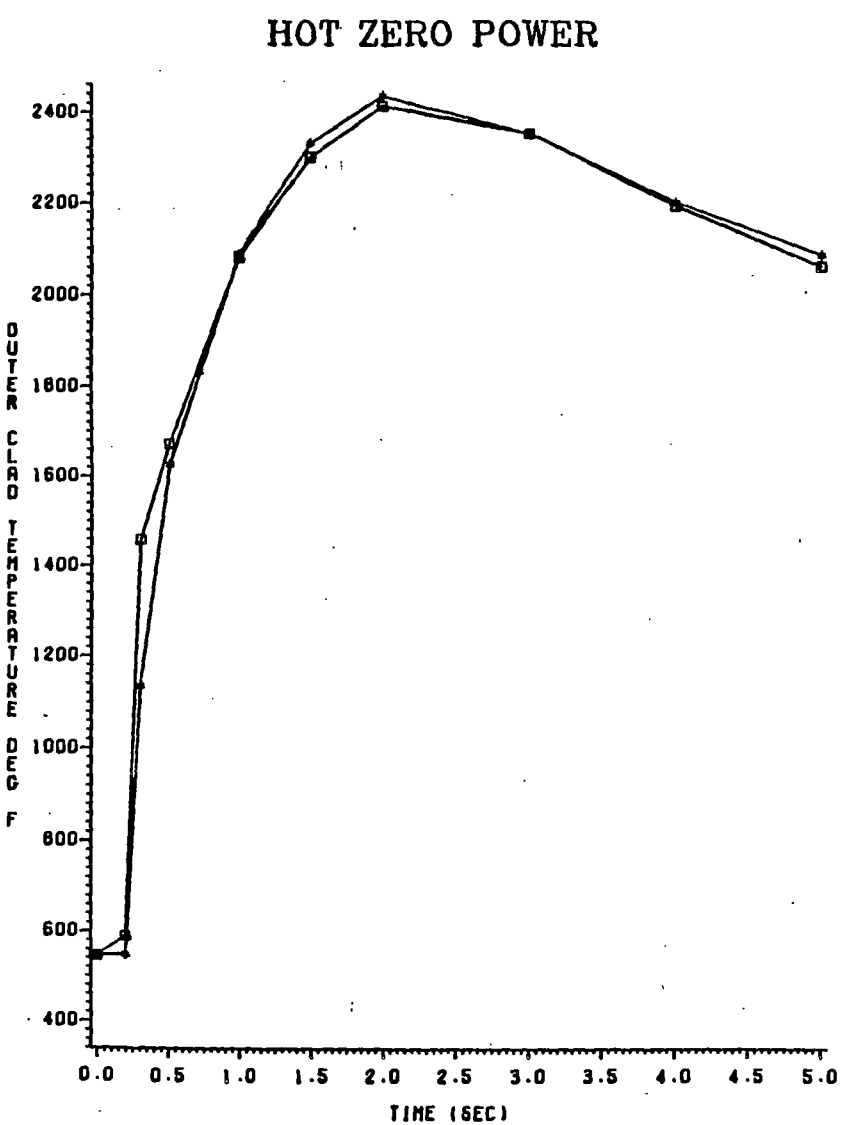


STAR: FACTRAN CODE
SQUARE: RETRAN HOT SPOT MODEL

FIGURE 4-11B

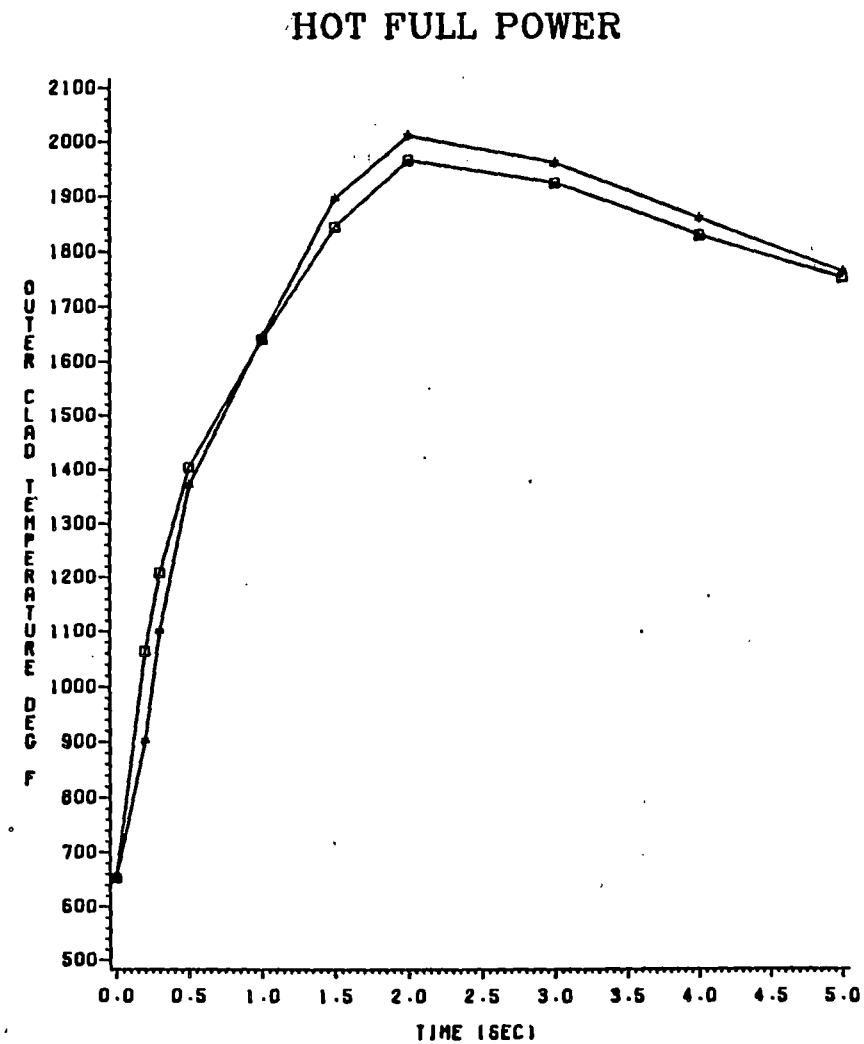
FIGURE 4-12

HOT SPOT FUEL OUTER CLAD TEMPERATURE TRANSIENTS - S+MTC CASES



STAR: FACTRAN CODE
SQUARE: RETRAN HOT SPOT MODEL

FIGURE 4-12A



STAR: FACTRAN CODE
SQUARE: RETRAN HOT SPOT MODEL

FIGURE 4-12B

TABLE 4-4

FACTRAN/RETRAN Hot Spot Model Comparisons

The FACTRAN calculated value is followed by the RETRAN calculated value separated by a slash (/).

Parameter	Surry 1 Cycle 5 Values BOL HZP	BOL HFP	EOL HZP	EOL HFP
Fuel Pellet Melting (%)	0/0	<10/<10	0/0	<10/<10
Max. Fuel Center Temp. (°F)	3954/4003	5017/4903	3390/4096	4889/4802
Max. Clad Temp. (°F)	2520/2484	2482/2292	2169/2568	2228/2172
Max. Fuel Enthalpy (Btu/lb)	261/267	334/318	219/278	301/299

Parameter	Surry +MTC Values BOL HZP	BOL HFP
Fuel Pellet Melting (%)	0/0	0/0
Max. Fuel Center Temp. (°F)	3918/3990	4874/4733
Max. Clad Temp. (°F)	2488/2471	2079/2036
Max. Fuel Enthalpy (Btu/lb)	258/266	280/275

3.3 Conclusions

Comparisons between results predicted with the Vepco and vendor methodologies for the analysis of the rod ejection event for Vepco nuclear units showed similar trends and close agreement despite differences in the two models. For the cases analyzed in this report, both methods predicted a comparable degree of fuel melting for the cases in which it occurred, but for all cases it was under the 10% limit imposed by the acceptance criteria. Likewise, neither method predicted a violation of the acceptance criteria for the peak clad temperature or peak average fuel enthalpy.

4 Comparison to Three-Dimensional Space-Time Kinetics

A three-dimensional space-time kinetics code was used to verify the acceptability and conservatism of the point kinetics approach to predict the core average power history. As noted by Yasinsky, (Ref. 20), the accuracy and acceptability of the point kinetics approach is dependent both on the method used and the particulars of the nuclear core under analysis. The use of a weighting factor to increase the magnitude of the Doppler reactivity feedback used in the point kinetics model (as is done with the Vepco methdology) is appropriate as indicated by the verification of the conservatism of the approach.

The TWINKLE code (Ref. 18) was used to analyze two rod ejection cases using three dimensional geometry:

1. BOL, HZP, ejected rod worth of 1024 pcm
2. BOL, HFP, ejected rod worth of 200 pcm

The results from these analyses were compared to the results from the RETRAN point kinetics method and the TWINKLE 1-D method for similar transient conditions in order to verify the conservatism of the latter two methods.

4.1 Three-Dimensional Model

A three-dimensional (3-D) TWINKLE model was constructed for an initial Surry unit fuel loading (Cycle 1) for a core consisting of all fresh fuel assemblies. A quarter core loading map of the cycle is presented in Figure 4-13, and gives the initial fuel enrichment of the three batches (w/o U235) and number of fresh burnable poison rodlets (# BP rods) present in each assembly. Appropriate nuclear cross section data describing the core loading was derived from vendor steady state core physics design codes, and input to the TWINKLE code following the procedures outlined in Ref. 18. A three-dimensional geometry was specified with one radial mesh point per assembly, 11 axial mesh points per assembly, one mesh point per top and bottom axial reflectors and a radial reflector region one mesh point deep at the nearest approach of a peripheral fuel assembly. A half core geometry model was used with the full core being split along the vertical axis giving a total of 1989 mesh points. The location of the ejected rod is on the vertical axis through the center of the core and produces a radially symmetric power and flux distribution about the vertical axis. Therefore a half core geometry is sufficient for modeling the transient in three dimensions. (See Figure 4-14.) The patterns used for withdrawing the control rod banks from the core from HZP to HFP conditions cause the ejected rod to be located on an axis of symmetry.

The steady state radial power distributions at HZP and HFP for an all rods withdrawn core configuration were normalized to those calculated by the vendor for the initial FSAR for the Surry units, (Ref. 21.) This

rmalization was carried out by adjusting the fast diffusion coefficient for the radial reflector region and the macroscopic absorption cross sections for the various fuel batches until reasonable agreement was obtained with the radial power distributions published in Ref. 21. The resulting power distribution comparisons for initial steady state, all rods withdrawn core conditions are presented in Figure 4-15.

In a similar fashion, the remaining critical core parameters as predicted by the steady state 3-D TWINKLE model were normalized to values typical of those used in a rod ejection analysis. For example, the zero to full power Doppler defect was normalized to a value of -1164 pcm by adjusting a multiplier to the fuel temperature component of the macroscopic fast absorption cross section. Typical values of the oderator temperature coefficient predicted by the model were obtained by adjusting the soluble boron concentration input to the code.

Control rod worths were normalized by adjusting the thermal absorption cross section used to model the presence of control rods. Using the values provided by the steady state physics code for these cross sections and ejecting the rod from the rod insertion limits specified for the initial Surry cycle resulted in ejected rod worths of such small magnitude that only minor power excursions were produced. To model transients which were more typical of those analyzed for reload cores, the insertion limits were deepened and the worth of the ejected rod increased by increasing the thermal absorption cross section of the D bank. This resulted in a HZP ejected rod worth of 1024 pcm (percent mille) for an initial core configuration of the D bank fully inserted

and the C bank inserted 43.5% into the core. For the HFP case, an ejected rod worth of 200 pcm resulted with the D bank inserted 56.5% into the core. All other control and shutdown banks were initially out of the core.

For simplicity of modeling, reactor scrams were modeled by inserting the control banks (banks D, C, B and A) less the ejected rod. The thermal absorption cross section for banks A,B and C was modified to yield conservative total trip worths typical of those used in the rod ejection analysis, (i.e., approximately 2000 pcm for the HZP case and 4000 pcm for the HFP case.)

Table 4-5 presents a summary of the steady state core physics parameters for the final 3-D TWINKLE model used for the benchmark cases.

The two transient cases were initiated by ejecting the chosen rod from the core at a constant velocity over a 0.1 seconds time interval. Reactor trip was initiated on the same trip setpoints assumed in the RETRAN point kinetics analysis and assumed a 0.5 seconds trip delay between activation of the trip and the start of rod motion. The trip rods entered the core using the same rod insertion model as is assumed in the standard TWINKLE 1-D analysis of the transient. Since the 3-D model predicts the effects of three-dimensional flux redistribution during the transient, no weighting factor was applied to the calculation of the Doppler reactivity feedback.

FIGURE 4-13

SURRY UNIT 1 CYCLE 1 CORE LOADING PLAN

(Eighth Core Geometry)

	H	G	F	E	D
08	1				
09	2 12	1			
10	1	2 12	1		
11	2 12	1	2 12	1	
12	1	2 12	1	2 12	1
13	2 12	1	2 12	3 12	3
14	1	3 12	3	3	
15	3	3			

Batch	# Fuel Assys	w/o U235
1	53	1.85
2	52	2.55
3	52	3.10

Legend:

x	==> Batch #
xx	==> # of BP Rods

FIGURE 4-14

RADIAL GEOMETRY FOR 3-D TWINKLE

	H	G	F	E	D	C	B	A
01								
02	D		A					
03		SA						
04			B		C			
05				SB				
06	C		D		B		A	
05		SB				SA		
8			C				D	
09		SB				SA		
10	C		D		B		A	
11				SB				
12			B		C			
13		SA						
14	X		A					
15								

Legend:

| | ==> Fuel Mesh Point

:...: ==> Reflector Mesh Point

Control Banks:
D
C
B
A
SB
SA
X = Location of Ejected Rod

FIGURE 4-15

STEADY STATE RADIAL POWER DISTRIBUTIONS

Beginning of Life, Hot Zero Power, All Rods Withdrawn

	H	G	F	E	D
08	1.130 1.210 -6.6				
09	1.180 1.220 -3.3	1.120 1.190 -5.9			
10	1.120 1.170 -4.3	1.160 1.180 -1.7	1.090 1.130 -3.5		
11	1.150 1.150 0.0	1.100 1.120 -1.8	1.110 1.090 1.8	1.000 1.000 0.0	
12	1.100 1.090 0.9	1.130 1.090 3.7	1.030 1.010 2.0	0.970 0.910 6.6	0.780 0.720 8.3
13	1.140 1.070 6.5	1.080 1.040 3.8	1.040 0.970 7.2	0.920 0.910 1.1	0.630 0.620 1.6
14	1.060 1.000 6.0	1.120 1.120 0.0	1.000 1.080 -7.4	0.650 0.670 -3.0	
15	0.930 0.940 -1.1	0.710 0.760 -6.6			

Legend:

x.xxx	==> FSAR Relative Power
y.yyy	==> TWINKLE 3-D Relative Power
z.z	==> % Difference

FIGURE 4-15 (cont.)

STEADY STATE RADIAL POWER DISTRIBUTIONS

Beginning of Life, Hot Full Power, All Rods Withdrawn

	H	G	F	E	D
08	1.190 1.240 -4.0				
09	1.230 1.240 -0.8	1.170 1.220 -4.1			
10	1.160 1.200 -3.3	1.200 1.200 0.0	1.130 1.160 -2.6		
11	1.180 1.170 0.9	1.120 1.150 -2.6	1.130 1.110 1.8	1.020 1.020 0.0	
12	1.110 1.100 0.9	1.130 1.100 2.7	1.040 1.030 1.0	0.970 0.930 4.3	0.790 0.750 5.3
13	1.120 1.060 5.7	1.060 1.040 1.9	1.010 0.960 5.2	0.900 0.910 -1.1	0.630 0.640 -1.6
14	1.010 0.970 4.1	1.060 1.070 -0.9	0.950 1.020 -6.9	0.630 0.660 -4.5	
15	0.870 0.890 -2.2	0.670 0.730 -8.2			

Legend:

x.xxx	==> FSAR Power
y.yyy	==> TWINKLE 3-D Power
z.z	==> % Difference

TABLE 4-5

3-D COMPARISON CASES

Parameter	BOL HZP	BOL HFP
Ejected rod worth (pcm)	1024	200
Delayed neutron fraction	.0059	.0059
Pre-ejection Fq	NA	2.55
Post-ejection Fq *	10.6	4.33
Trip rod worth (pcm)	2210	4069
Zero to full power Doppler defect (pcm)	-1164	-1164
TWINKLE 3-D soluble boron concentration (ppm)	2260	1700
RETRAN moderator temperature coefficient (pcm/°F)	2.9	1.5
Number of operating pumps	2	3
TWINKLE 1-D reactivity ** feedback weighting factor	1.74	1.2
RETRAN Power Weighting Factor	2.3	1.25

Notes: * Peak 3-D steady state nodal power weighted by a generic pin-to-box ratio and uncertainty factor

** Weighting factor for 3-D TWINKLE = 1.0

NA = not applicable

ppm = parts per million

pcm = percent mille

4.2 Comparison Results

In order to select the proper power weighting factor (PWF) for the RETRAN point kinetics analysis for comparison to the 3-D analysis, a hot spot total power peaking factor (Fq) for the condition of the rod being ejected must first be known. Since the core conditions being analyzed are not comparable to those which would typically be predicted for Surry Unit 1, Cycle 1, (i.e., the ejected rod worths have been arbitrarily increased), values of Fq used in the comparison analysis were derived from the 3-D TWINKLE model. Steady state values of peak core nodal power at initial control rod core configurations less the ejected rod were calculated with the 3-D TWINKLE model for both HZP and HFP conditions. (The HFP calculation assumed frozen thermal hydraulic feedback.) These values were increased by a 20% generic pin-to-box ratio to convert them from peak nodal powers to hot spot total power peaking factors. An additional 10% was added for uncertainty. As presented in Table 4-5 above, the assumed post-ejection total power peaking factors were therefore 10.6 for the HZP case and 4.33 for the HFP case. The PWFs for the RETRAN analysis were derived from these values of Fq based on Figure 2-6. Likewise, the reactivity feedback weighting factors used in the 1-D TWINKLE analysis were based on these values of Fq.

Typically, the higher the value of Fq assumed for the analysis, the less severe will be the power history predicted by the point kinetics calculation since a higher value of Fq implies a higher PWF which in turn causes a lower power history curve to be predicted. The derivation of the Fq values was based on steady state 3-D TWINKLE calculations

instead of the transient calculations for this reason, since the transient calculation derives some benefit from thermal hydraulic feedback effects and therefore predicts lower Fq values than the steady state calculation. This is the method used for core reload analysis where the Fq values are obtained with steady state physics codes assuming frozen thermal hydraulic feedback. The peak nodal powers predicted by the TWINKLE 3-D model with the rod ejected from the core (and before the initiation of the scram) were as follows:

	HZP	HFP
Post-ejection steady state	8.04	3.28
Transient	8.39	2.59

In essence, the use of conservatively high values of Fq will yield lower RETRAN point kinetics power history predictions and therefore lead to a closer power history comparison with the TWINKLE 3-D prediction than might otherwise be expected. From the standpoint of the hot spot calculation, the lower core average power history will tend to lower the predicted peak temperature and enthalpy predictions. But this is offset by the higher Fq values used to weight the core average power history which will in turn cause the peak temperature and enthalpy predictions to be more conservative. The latter effect is the more dominant, especially for the HFP case where the PWF has little effect on the core average power history prediction.

Figure 4-16 presents the core average power history comparisons. For both the HZP and HFP cases, the RETRAN point kinetics predictions are conservative compared to the TWINKLE 3-D prediction thus verifying the

ceptability of the point kinetics methodology in general, and the usage of the PWF in particular. The conservatism of the point kinetics approach is even more apparent in Figure 4-17 which compares the total energy release of the two models. Both figures show reasonably close agreement between the RETRAN point kinetics predictions and the TWINKLE 1-D predictions.

The hot spot power histories input to the RETRAN Hot Spot Model were calculated using the method outlined in Section 2 based on the Fq values provided in Table 4-5. A similar method was used for the FACTRAN calculations based on the TWINKLE 1-D predicted core average power histories. That is, the value of Fq as a function of time was linearly ramped form its pre-ejection value to its post-ejection value over a time interval of 0.1 seconds and maintained at the post-ejection value for the remainder of the transient. The core average power history is then multiplied by this Fq curve to derive a hot spot power history.

A different approach was used to derive the hot spot power histories for the FACTRAN calculations for the 3-D cases. For these cases, the hot spot power for a specific time was found by multiplying the 3-D core average normalized power and the 3-D peak nodal power (increased by a 20% pin-to-box ratio and a 10% uncertainty factor.) This method was used to reflect the fluctuation of Fq throughout the transient. Typically, the Fq value at any time will be less than that assumed for the point kinetics methodology. However, because of the flux redistribution which results from the movement of the scram banks into the core, it is possible for the peak normalized nodal power in the core to exceed that

assumed for the post-ejection condition. Use of a constant generic pin-to-box ratio to convert the peak normalized nodal power to a Fq value may result in a 3-D Fq prediction later in the transient which exceeds that calculated for a post-ejection condition by steady state methods. The actual impact of this on the transient is mitigated by two effects: (1) the actual core location of the hot spot in the core will be expected to shift throughout the transient, and (2) the actual power level of the core during the trip is relatively low compared to that at the transient peak. Therefore, the Fq transient shape input to the point kinetics analysis would still yield a conservative hot spot power history compared to what would actually transpire in a three-dimensional core. Figure 4-18 presents a comparison of the hot spot power histories input to the RETRAN Hot Spot Model and the FACTRAN code for the three methodologies under comparison. Since the 3-D TWINKLE derived hot spot power history is appreciably lower than that of the RETRAN point kinetics calculation, it may be reasonably predicted that the results of the RETRAN hot spot analysis based on the RETRAN hot spot power history will be conservative compared to the FACTRAN analysis results based on the 3-D TWINKLE power history.

Table 4-6 presents the results of these analyses based on the hot spot power histories presented in Figure 4-18. Plots of the fuel centerline and outer clad temperature transients for the three models are presented in Figures 4-19 and 4-20. As expected, the RETRAN point kinetics and TWINKLE 1-D based analyses are conservative compared to the 3-D analysis.

TABLE 4-6

3-D HOT SPOT MODEL COMPARISON RESULTS

I. HZP Case:

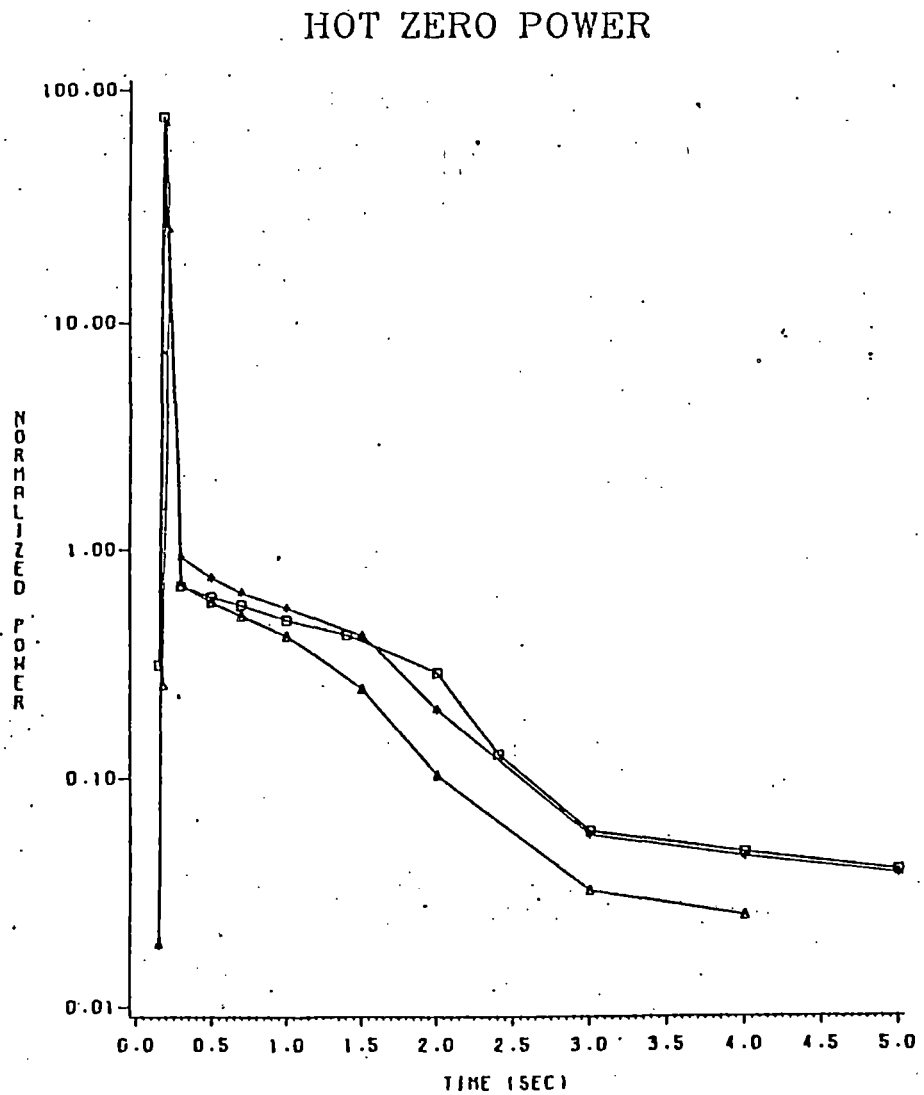
Parameter	1-D TWINKLE /FACTRAN	RETRAN	3-D TWINKLE /FACTRAN
Fuel Pellet Melting (%)	0	0	0
Max. Fuel Center Temp. (°F)	4346	3872	2659
Max. Clad Temp. (°F)	2869	2387	1703
Max. Fuel Enthalpy (Btu/lb)	296	255	161

II. HFP Case:

Parameter	1-D TWINKLE /FACTRAN	RETRAN	3-D TWINKLE /FACTRAN
Fuel Pellet Melting (%)	0	0	0
Max. Fuel Center Temp. (°F)	4819	4685	4419
Max. Clad Temp. (°F)	2037	1990	1851
Max. Fuel Enthalpy (Btu/lb)	274	268	235

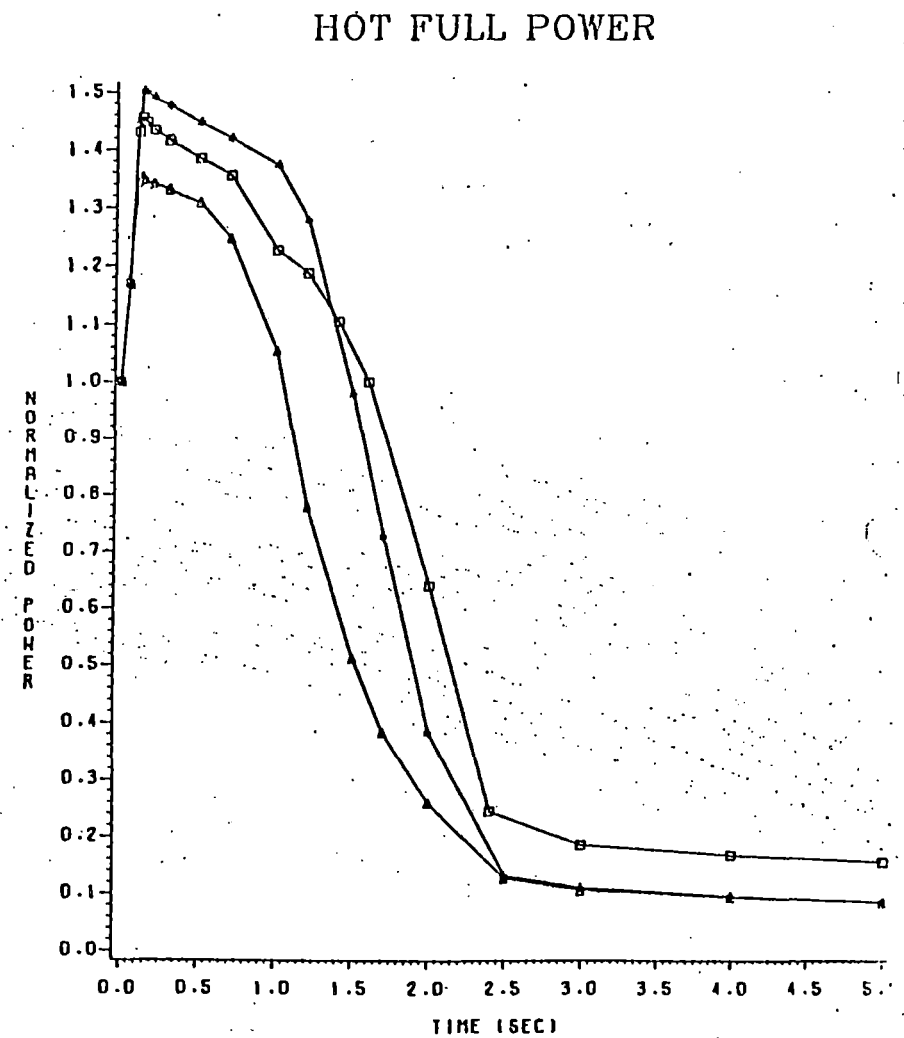
FIGURE 4-16

NUCLEAR POWER TRANSIENTS - 3-D BENCHMARKS



STAR: TWINKLE 1-D MODEL
SQUARE: RETRAN POINT KINETICS MODEL
TRIANGLE: TWINKLE 3-D MODEL

FIGURE 4-16A



STAR: TWINKLE 1-D MODEL
SQUARE: RETRAN POINT KINETICS MODEL
TRIANGLE: TWINKLE 3-D MODEL

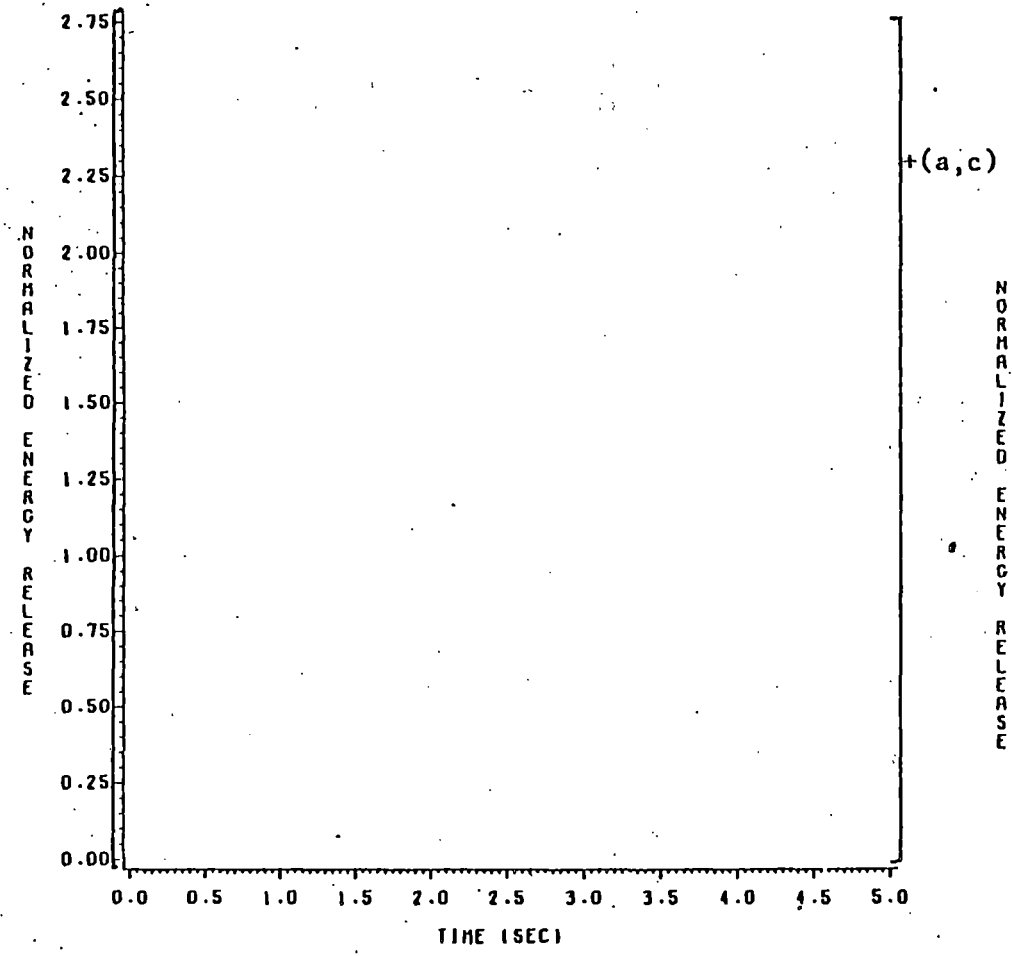
FIGURE 4-16B

FIGURE 4-17

TOTAL ENERGY RELEASE - 3-D BENCHMARKS

WESTINGHOUSE PROPRIETARY CLASS 2

HOT ZERO POWER

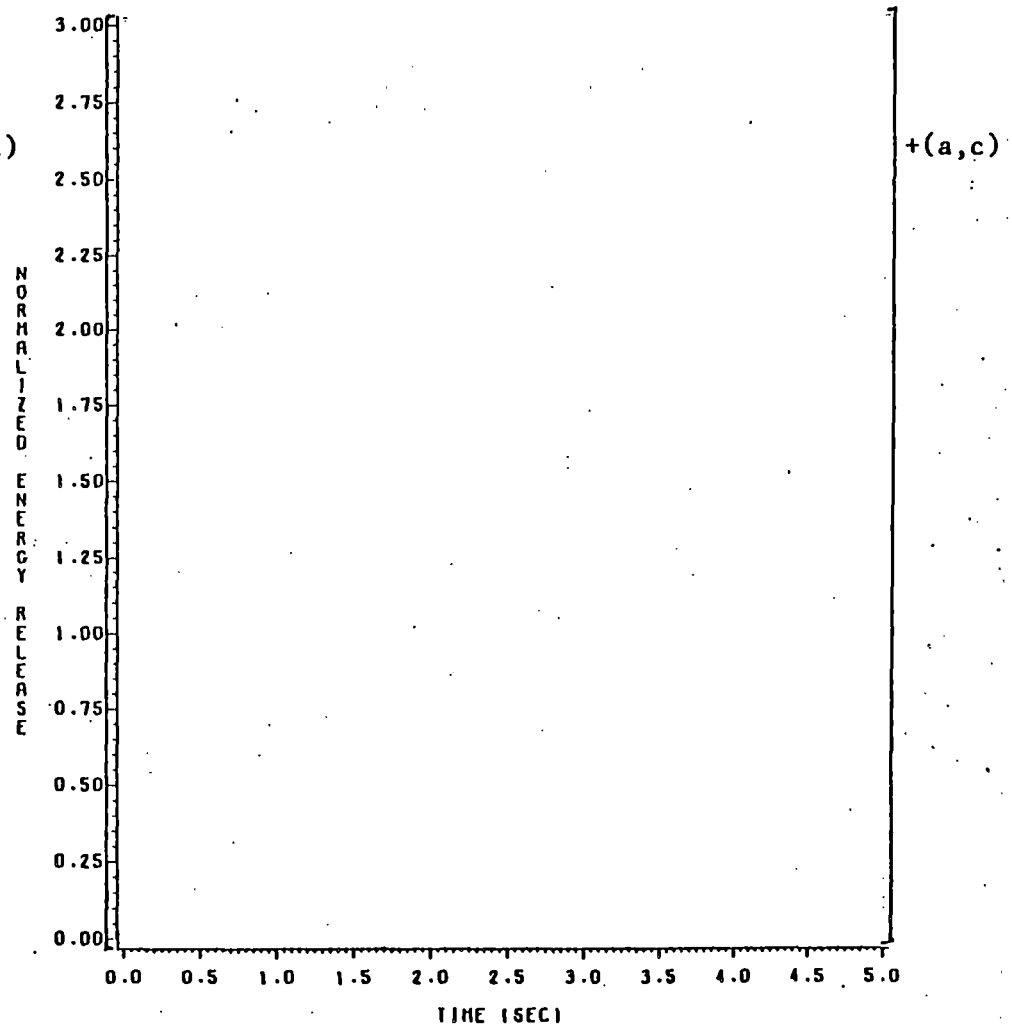


STAR: THINKLE 1-D MODEL
SQUARE: RETRAN POINT KINETICS MODEL
TRIANGLE: THINKLE 3-D MODEL

FIGURE 4-17A

WESTINGHOUSE PROPRIETARY CLASS 2

HOT FULL POWER



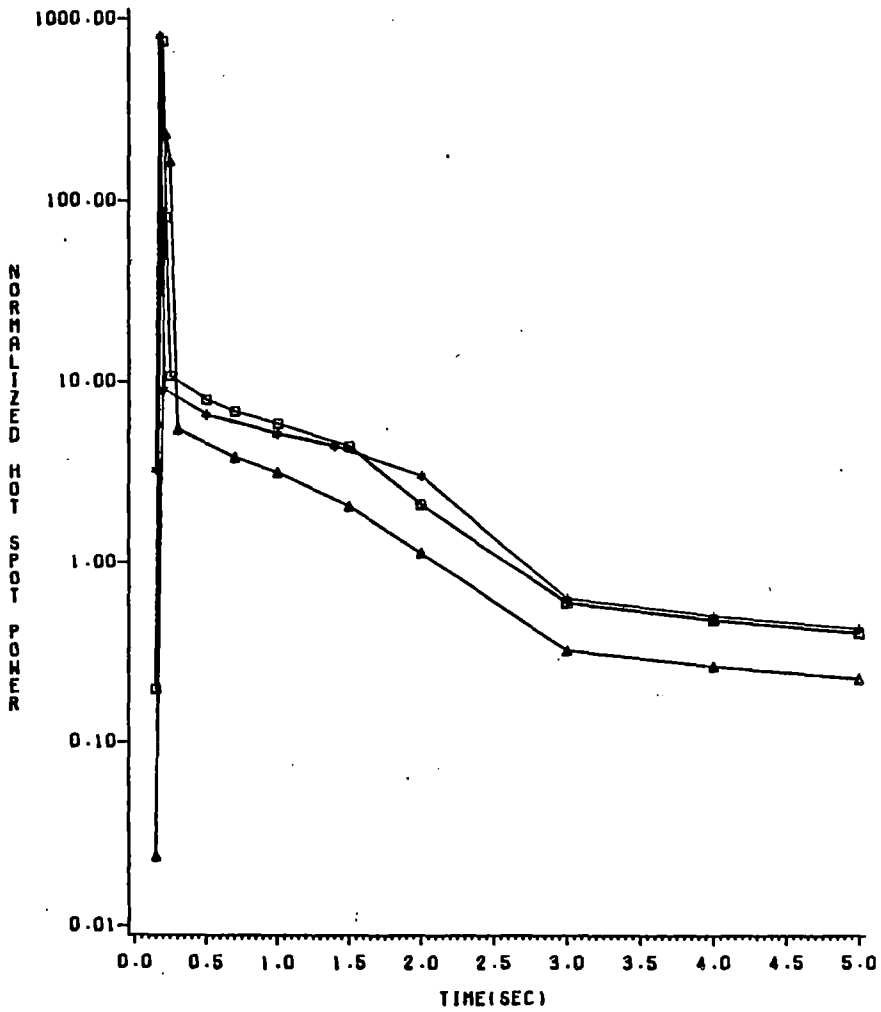
STAR: THINKLE 1-D MODEL
SQUARE: RETRAN POINT KINETICS MODEL
TRIANGLE: THINKLE 3-D MODEL

FIGURE 4-17B

FIGURE 4-18

HOT SPOT POWER [illegible]ORY - 3-D BENCHMARKS

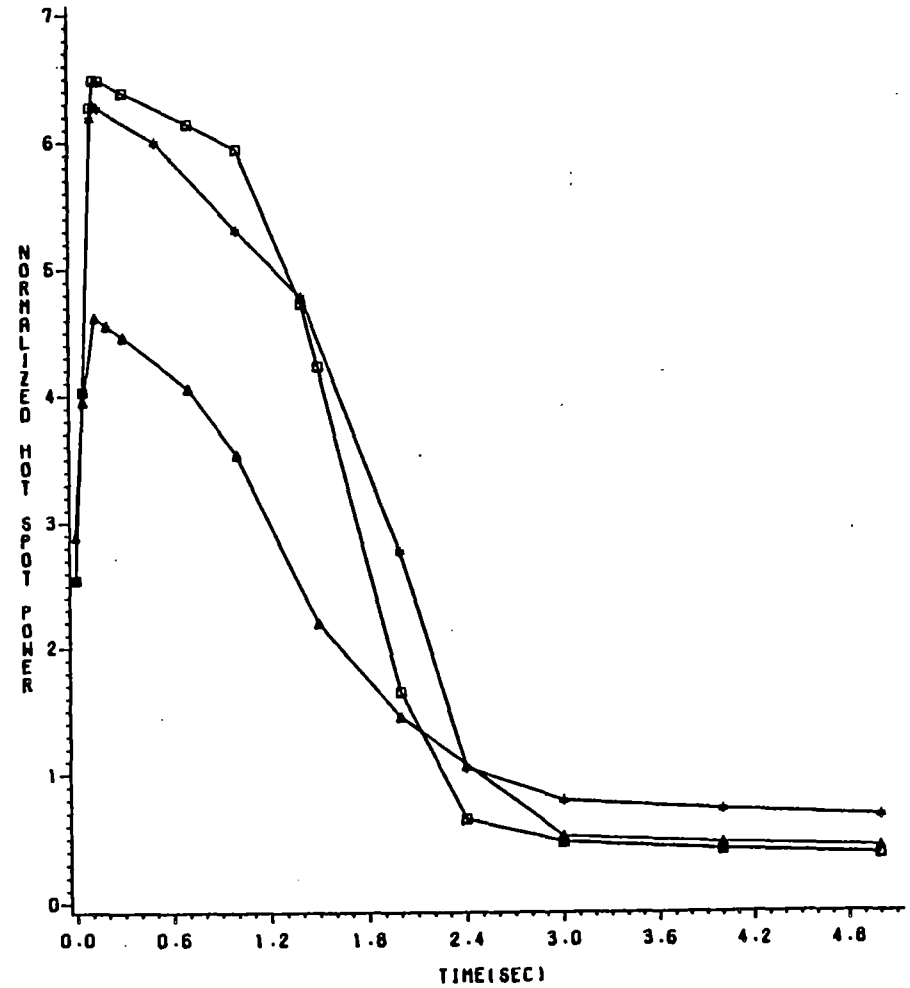
HOT ZERO POWER



STAR: RETRAN POINT KINETICS
SQUARE: TWINKLE 1-D
TRIANGLE: TWINKEL 3-D

FIGURE 4-18A

HOT FULL POWER

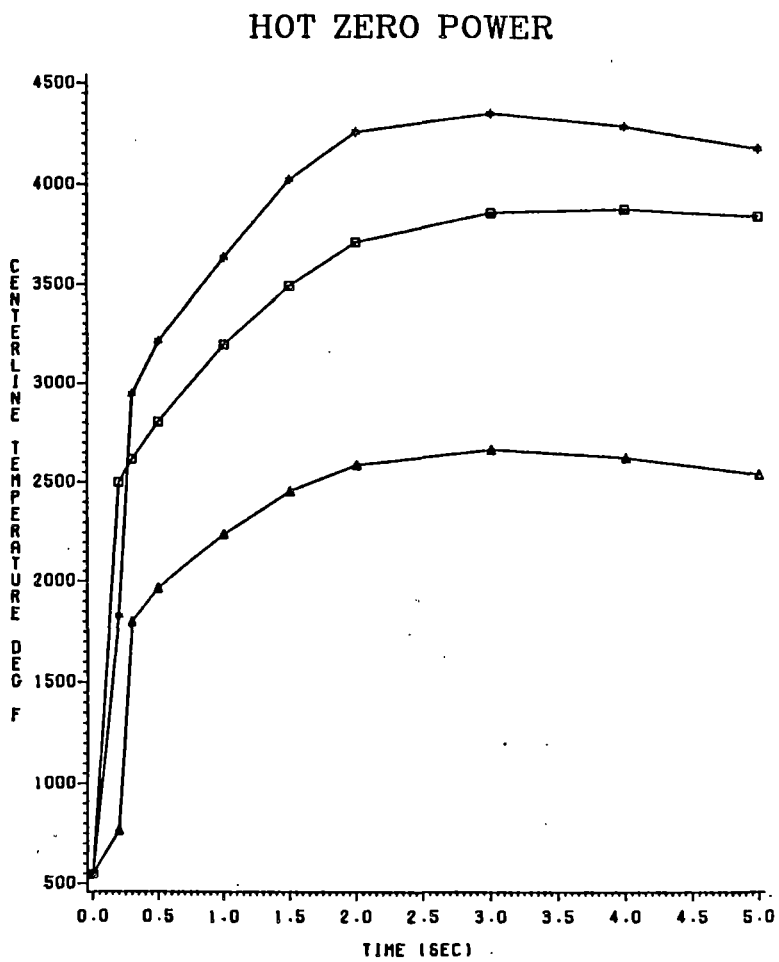


STAR: RETRAN POINT KINETICS
SQUARE: TWINKLE 1-D
TRIANGLE: TWINKEL 3-D

FIGURE 4-18B

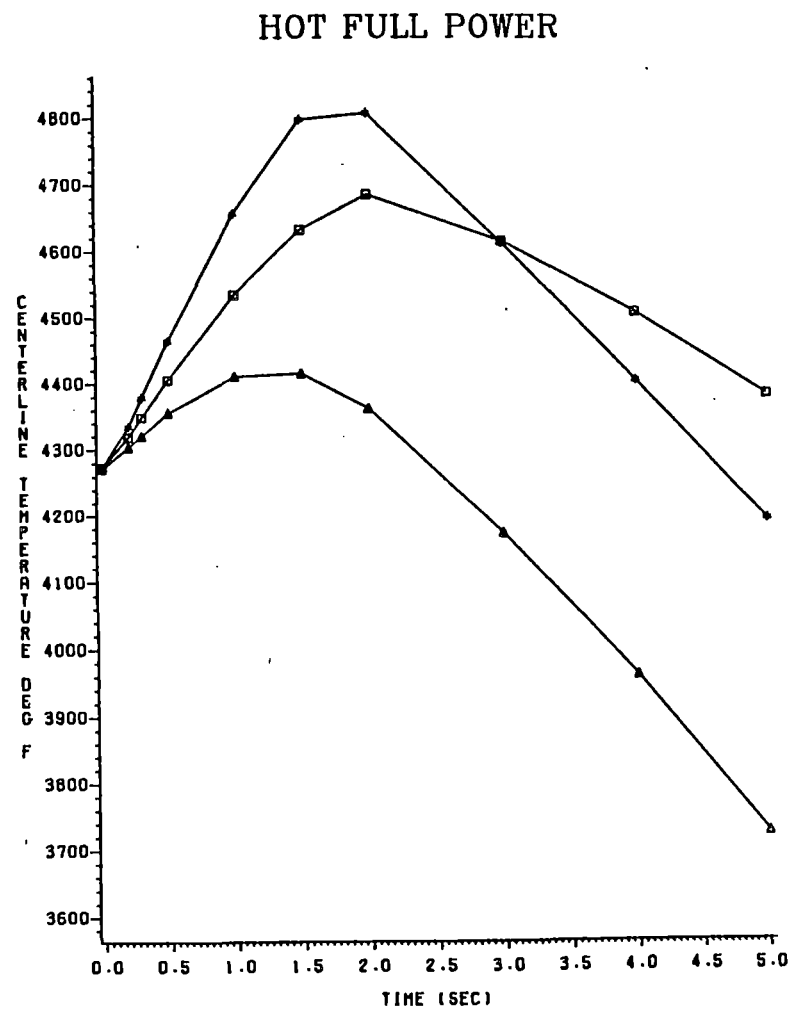
FIGURE 4-19

HOT SPOT FUEL CENTERLINE TEMPERATURE TRANSIENTS - 3-D BENCHMARKS



STAR: FACTRAN CODE (1-D POWER HISTORY)
SQUARE: RETRAN HOT SPOT (POINT KINETICS POWER HISTORY)
TRIANGLE: FACTRAN CODE (3-D POWER HISTORY)

FIGURE 4-19A

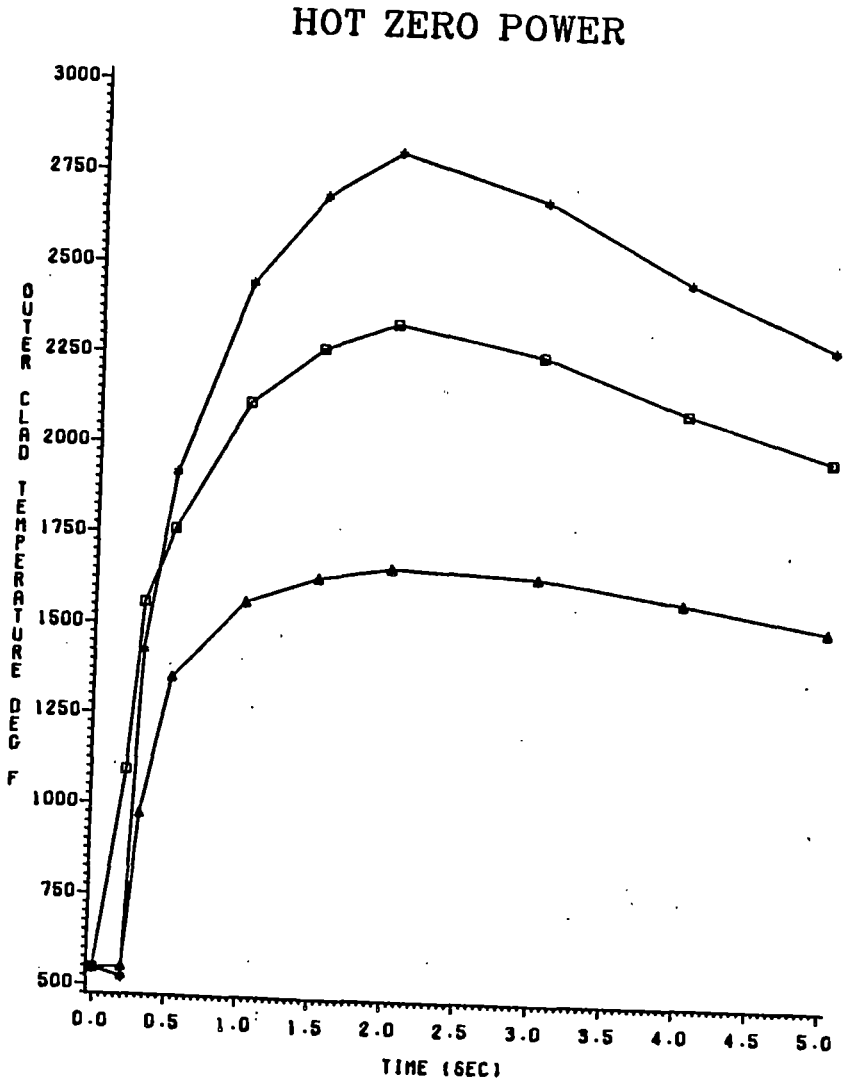


STAR: FACTRAN CODE (1-D POWER HISTORY)
SQUARE: RETRAN HOT SPOT (POINT KINETICS POWER HISTORY)
TRIANGLE: FACTRAN CODE (3-D POWER HISTORY)

FIGURE 4-19B

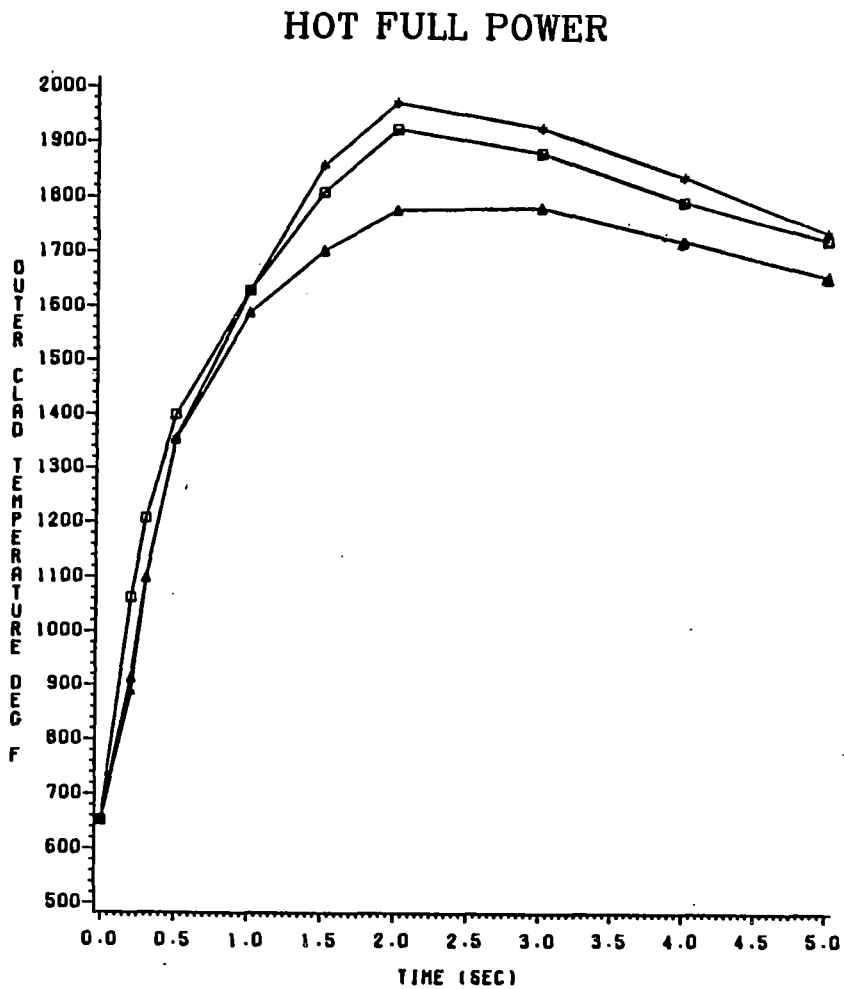
FIGURE 4-20

HOT SPOT FUEL OUTER CLAD TEMPERATURE TRANSIENTS - 3-D BENCHMARKS



STAR: FACTRAN CODE (1-D POWER HISTORY)
SQUARE: RETRAN HOT SPOT (POINT KINETICS POWER HISTORY)
TRIANGLE: FACTRAN CODE (3-D POWER HISTORY)

FIGURE 4-20A



STAR: FACTRAN CODE (1-D POWER HISTORY)
SQUARE: RETRAN HOT SPOT (POINT KINETICS POWER HISTORY)
TRIANGLE: FACTRAN CODE (3-D POWER HISTORY)

FIGURE 4-20B

SECTION 5 - SUMMARY AND CONCLUSIONS

The Virginia Electric and Power Company (Vepco) has developed a methodology using the RETRAN transient thermal hydraulics code to analyze the control rod ejection event for Vepco's Surry and North Anna Nuclear Power Stations. The analysis is performed in two steps: (1) a point kinetics calculation to determine the core average power history, and (2) a thermal hydraulics analysis, based on the core average power history from step (1), to predict the fuel and cladding temperatures and average fuel enthalpy history of the core hot spot. Results from step (2) are used to confirm that safe plant operation acceptance criteria are met. These criteria are those in USNRC Reg. Guide 1.77 (Ref. 3) and the additional in-house limits discussed in Section 1.3.

Acceptability of the methodology has been demonstrated by comparison of selected analytical results obtained with the Vepco methodology to corresponding results obtained with a NRC approved vendor methodology. The conservatism of the Vepco methodology, specifically the point kinetics calculation and its use of a Doppler reactivity feedback weighting factor with the very conservative hot spot static peaking, was demonstrated through the comparison of the Vepco results with those based on a three-dimensional space-time kinetics model and a detailed hot spot thermal hydraulic model. In Section 3, a comprehensive sensitivity analysis was performed and reported for neutronic and thermal hydraulic parameters to verify the range of applicability. The overall good agreement between the two methodologies affirms that the Vepco methodology can be used for performing the reload safety analysis

for the control rod ejection analysis for Vepco's nuclear power plants.

SECTION 6 - REFERENCES

1. Surry Power Station Units 1 and 2, "Updated Final Safety Analysis Report," Virginia Electric and Power Company, 1982.

2. North Anna Power Station Units 1 and 2, "Updated Final Safety Analysis Report," Virginia Electric and Power Company, 1982.

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