WPSRSEM-NP-A
REVISION 2
OCTOBER, 1988

KEWAUNEE NUCLEAR POWER PLANT

RELOAD SAFETY EVALUATION METHODS FOR APPLICATION TO KEWAUNEE

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WPSRSEM-NP-A
REVISION 2
October, 1988

RELOAD SAFETY EVALUATION METNODS FOR APPLICATION TC KEWAUNEE

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

April 11, 1988

Docket No. 50-305

Mr. D. C. Hintz
Vice President - Nuclear Power
Wisconsin Public Service
Corporation
P.O. Box 19002
Green Bay, Wisconsin 54037-9002

Dear Mr. Hintz:

SUBJECT: WISCONSIN PUBLIC SERVICE CORPORATION "RELOAD SAFETY EVALUATION METHODS FOR APPLICATION TO KEWAUNEE" (TAC No. 65155)

By letter dated March 27, 1987, you submitted for review a topical report entitled "Reload Safety Evaluation Methods for Application to Kewaunee". Additional information was submitted on February 12 and March 7, 1988. The report includes the necessary methods for Kewaunee reloads except for the loss-of-coolant accident (LOCA) and the fuel mishandling accident, which will be submitted by the fuel vendor on a cycle-specific basis.

The analyses employed the DYNODE-P (Version 5.4), the RETRAN-02, the VIPRE-01 and the TOODEE-2 codes. The description and the performance of these codes is part of this review. In addition, the analyses, procedures and the results of specific calculations and reload evaluations were examined. The NRC finds that the topical report is acceptable for referencing in licensing Kewaunee reloads.

The staff's safety evaluation is enclosed.

Sincerely,

Joseph G. Giitter

Joseph G. Giitter, Project Manager
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Enclosure:
As stated

cc: See next page

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATING TO TOPICAL REPORT ON
"RELOAD SAFETY EVALUATION METHODS FOR APPLICATION TO KEWAUNEE
WISCONSIN PUBLIC SERVICE CORPORATION
KEWAUNEE NUCLEAR POWER STATION
DOCKET NO. 50-305

1.0 INTRODUCTION

This report describes the calculational methods employed by the Wisconsin Public Service (WPS) Corporation for determining Kewaunee cycle-specific safety parameters, and their evaluation with respect to bounding values used in the reference safety analyses. The calculation of the following safety parameters is described in the report.

1. Moderator Temperature Coefficient of Reactivity
2. Power Reactivity Coefficient
3. Doppler Reactivity Coefficient
4. Boron Reactivity Coefficient
5. Shutdown Margin
6. Scram Reactivity Curve
7. Nuclear Heat Flux Hot Channel Factor
8. Nuclear Enthalpy Rise Hot Channel Factor
9. Effective Delayed Neutron Fraction
10. Prompt Neutron Lifetime

The evaluation section describes parameter monitoring. If for any accident the cycle-specific value of a relevant parameter (accounting for a model bias and a reliability factor) falls outside the current bounds, a reanalysis of that accident becomes necessary. For each accident, the following material is presented in the report as part of the safety evaluation methods: Accident definition, accident analysis, safety analysis results, cycle-specific calculations and reload safety evaluation.

The report addresses the following specific accidents and transients which are considered in the safety evaluation of a core reload:

1. Uncontrolled Rod Cluster Control Assembly (RCCA) Withdrawal from Subcritical
2. Uncontrolled RCCA Withdrawal at Power
3. Control Rod Misalignment
4. Control Rod Drop
5. Uncontrolled Boron Dilution
6. Startup of an Inactive Loop
7. Feedwater System Malfunction
8. Excessive Load Increase
9. Loss of External Load
10. Loss of Normal Feedwater Flow
11. Loss of Reactor Coolant Flow due to Pump Trip
12. Loss of Reactor Coolant Flow due to Locked Rotor
13. Fuel Mishandling Accident
14. Main Steam Line Break
15. Control Rod Ejection
16. Loss of Coolant
17. Power Distribution Control (PDC II) Procedures

The report describes the following codes used in the analyses:

(a) DYNODE-P (Version 5.4) is used to analyze the NSSS response. DYNODE-P provides a simulation of the core average power, fuel temperature, and coolant channel thermal-hydraulic responses.

(b) RETRAN-02 is used to analyze the NSSS transient response, both, to verify analyses performed with DYNODE-P, and independently analyze a particular transient.

(c) VIPRE-01 is used to analyze the hot channel thermal hydraulic response including the margin to critical heat flux (CHF). VIPRE-01 provides sub-channel analyses, including the analysis of the response of individual coolant channels and their associated fuel rods.

(d) TOODEE-2 is used to compute the temperature of the fuel hot spot for certain accidents. TOODEE-2 is used when VIPRE-01 hot channel analysis yields a Departure from Nucleate Boiling Ratio (DNBR) less than the value corresponding to the 95% probability limit for CHF at a 95% confidence level.

2.0 EVALUATION

The Reload Safety Evaluation Methods (RSEM) for Kewaunee are a revised version of a report first prepared in January 1979. The major revisions incorporated are:

(a) conversion from COBRA-IV to VIPRE-01 as the primary code for fuel thermal-hydraulic analysis,

(b) development of RETRAN-02 as an additional system analysis code, and performance of selected analyses using RETRAN-02, and

(c) generation of transient analysis results with DYNODE-P (Version 5.4), i.e, the current version of the DYNODE-P code.

2.1 Acceptability of DYNODE-P, RETRAN-02, TOODEE2

DYNODE-P and TOODEE2 have been approved (Ref. 1) for the reload safety evaluation of the Prairie Island Units 1 and 2, plants similar to Kewaunee. RETRAN-02 is a derivative of RELAP, and both codes have been extensively used to provide best estimate as well as conservative analyses of the transients under consideration. The staff utilized RETRAN-02 to qualify DYNODE-P for the reload safety evaluation of the Prairie Island Units (Ref. 1).

2.1.1 Acceptability of VIPRE-01

The VIPRE-01 code has been found acceptable by the staff
with the following conditions (Ref. 2):

(1) The application is limited to heat transfer modes up to critical heat flux

(2) An analysis is made to ensure that the minimum DNBR of the CHF correlation used in VIPRE-01 can predict its data base of DNB occurrence with at least a 95% probability at a 95% confidence level

(3) Documentation is submitted by each user to provide justification for the modeling assumptions, choice of particular two-phase flow models, correlations and input values of plant specific-data

(4) If a profile fit subcooled boiling model which was developed based on steady state data is used in boiling transients, care should be taken in time step size used for transient analysis to avoid a Courant number less than 1.

(5) Each user should abide by the quality assurance program established by EPRI for the VIPRE-01 code.

We have found that the Kewaunee RSEM report meets conditions (1) and (3). WPS has agreed to abide by the quality assurance program established by EPRI for VIPRE-01, and has determined that for all transient calculations where the profile fit Levy subcooled boiling correlation is used, Courant number is greater than 1 (Ref. 3). Lastly, in order to meet condition (2), WPS has analyzed test bundle measured data on critical heat flux using VIPRE-01 and the W-3 correlation (Ref. 4). A discussion of the results of this analysis is presented below.

2.1.2 WPS Analysis of DNB Test Results VIPRE-01

WPS selected four test bundles typical of current Advanced Nuclear Fuel (ANF) and Westinghouse 14 x 14 fuel designs which envelope many aspects of the current Kewaunee fuel designs. A statistical assessment was made of the VIPRE-01 DNBR calculations for the test bundles. A total of 246 calculations of critical heat flux were made, and the results were presented as tables of predicted critical heat flux, measured critical heat flux, and the ratio of predicted to measured DNBR. Of the 246 data points, 29 were rejected because they lay outside the limits of applicability of the W-3 correlation. The remaining 217 points were analyzed by two different methods. In the first the distribution of data was tested for normality using the W-statistic for small data sets and the D-prime test for larger data sets. Having determined that the data show acceptable normal behavior, an analysis of variance was performed on an equivalent sample size of 5.9 and 7.4 degrees of freedom. Based on a population mean of 0.7548, a true variance of .023481, and a one-sided tolerance factor of 3.203, the ratio of predicted/measured DNBR at a 95/95 limit was determined to be 1.246. In the second method, a distribution free analysis was performed to determine the ratio of predicted to measured DNBR at a 95/95 limit to be 1.027. The limit determined by the distribution free analysis is, therefore, bounded by that determined on the basis of normal distribution of data, and both limits are bounded by the 1.3 safety limit assumed by WPS in safety analyses.

2.2 General Physics Methods

We have reviewed the definitions of and the brief calculational procedures for the safety parameters indicated in Section 2.0 of this Safety Evaluation Report (SER), and found them consistent among themselves, and with the currently approved methods (Ref. 5). We have also examined the current model biases and reliability factors associated with these safety parameters (Ref. 3) and found them to be acceptable. We, therefore, find that the general physics methods, as described in Section 2 of the Topical Report, are acceptable for use in reload safety evaluations.

2.3 Description of Accident

We have reviewed the brief descriptions of the sixteen accidents listed in Section 2.0 of this SER and their possible consequences. We determined that the major features of the accidents and their possible consequences have been satisfactorily accounted for. We, therefore, find the descriptions of the accidents acceptable.

2.4 Accident Analysis

We have reviewed the section on Accident Analysis for each of the 16 accidents listed in Section 2.0 of this SER. Except for the cases indicated below, the accident analysis is carried out utilizing DYNODE-P and/or RETRAN-02 in conjunction with VIPRE-01 subchannel analysis. The exceptions are:

(a) The limiting $F_{\Delta H}$ for the most limiting dropped rod configuration for the Control Rod Drop Accident is determined using steady-state analysis; and the thermal margin at steady state is determined using subchannel analysis with VIPRE-01. This is acceptable since automatic rod control is administratively limited by constraints on power (less than 90%) and control rod bite (greater than 215 steps), and analyses of transients for which a trip does not occur (with concomitant power overshoots) are not necessary (Ref. 3).

(b) WPS does not analyze the Fuel Mishandling Accident or the Loss of Coolant Accident. The Loss of Coolant Accident analysis is contracted by the fuel vendor. In case a reanalysis of the Fuel Handling Accident is necessary, it will be performed by the fuel vendor.

For the accidents analyzed we have determined that:

(1) The input assumptions are consistent with the USAR and the codes used in the accident analyses.

(2) The acceptance criteria are consistent with the USAR design bases.

(3) The reactor state points (power level, control bank positions, exposures, etc.) have been chosen to ensure conservative results.

Based on this determination, we find the sections on Accident Analysis acceptable.

2.5 WPS Safety Analysis Results

We have reviewed the sections of the topical report in which WPS safety analysis results obtained with DYNODE-P, RETRAN-02, VIPRE-01 and TWODEE2 have been presented and compared with USAR results. All the accidents listed in Section 2.0 of this SER with the exception of the Fuel Handling Accident and the Loss of Coolant Accident, have been analyzed. Altogether 121 plots of parameters such as fuel rod heat flux, average moderator temperature, fuel and clad temperatures, core power level, pressurizer pressure, and minimum DNBR have been presented as functions of time, making up more than 3,000 data points. We find that the WPS results presented are either consistent with or conservative with respect to USAR results, and are indicative of WPS's ability to analyze these accidents.

2.6 Cycle-Specific Physics Calculations

We have reviewed the sections of the topical report which describe the cycle-specific physics calculations for the accident analyses, and find that:

(a) The cycle-specific physics calculations of the key safety parameters have been performed consistently with the general physics methods described in Section 2 of the topical report.

(b) The calculations have been performed at limiting core conditions to ensure conservatism.

2.7. Reload Safety Evaluation

We have reviewed the sections of the topical report that describe the reload safety evaluation with respect to the accidents under discussion by comparing the cycle-specific values of safety parameters with the bounding values. In our evaluation we examined the following aspects of the key safety parameters:

(a) completeness of the set of parameters for the accident in question,
(b) augmentation of the parameters with model biases and reliability factors to ensure conservatism, and
(c) the limiting directions of the safety parameters.

We have reviewed in detail the reload safety evaluations, and have noted the following:

(1) The prompt neutron lifetime, lp*, had been originally omitted from the list of key safety parameters used for the evaluation of Uncontrolled RCCA Withdrawal from Subcritical. In response to our request for additional information, WPS has agreed to include this parameter in the list of key safety parameters (Ref. 3). With this change, the reload safety evaluation of this accident is acceptable.

(2) For the Loss of Coolant Flow (Pump Trip and Locked Rotor) Accidents, the initial maximum fuel temperature is an important safety parameter, and must be included in the reload safety evaluation. Additionally, the bounding value of $F_{\Delta H}$ for these accidents has been indicated as the PDCII technical specification limit. Since Loss of Reactor Coolant Flow is a Condition III (Infrequent Occurrence) event, PDCII analysis does not extend to this accident, and the PDCII $F_{\Delta H}$ limit does not apply. The bounding value of $F_{\Delta H}$ is, therefore, the $F_{\Delta H}$ limit assumed in the USAR analysis.

(3) The acceptance conditions for the Doppler and moderator coefficients in the case of the Control Rod Ejection Accident as specified in the topical report are in error. They should read:

$$alpha_D*(1-RF_D) \leq alpha_D \quad \text{(most negative bounding value)}$$

$$alpha_M+PF_M+B_M \leq alpha_M \quad \text{(most negative bounding value).}$$

(4) For the Loss of Coolant Accident only three safety parameters, F_Q, $F_{\Delta H}$, and scram reactivity, are included in the reload safety evaluation. Other parameters such as fuel rod temperature, fuel rod internal pressure, decay heat, densification spike factor, and axial rod shrinkage also impact the consequences of this accident. In response to our request for additional information, WPS indicated that these parameters are bounded by the assumptions of the Loss of Coolant Accident analysis, which is performed by the fuel vendor.

3.0. CONCLUSION

We have reviewed WPS's topical report on Reload Safety Evaluation Methods for Application to Kewaunee. Our evaluation of the topical report addressed the applicability of the computer codes used, the general physics methods employed, the accident analysis procedures and results, cycle-specific calculations and reload safety evaluation. We find the topical report acceptable for referencing in licensing documents with the observations indicated in Section 2.7 above

4.0 REFERENCES

1. Safety Evaluation by the Office of Nuclear Reactor Regulation of the Reactor Physics and Reload Safety Evaluation Methods Technical Reports NSPNAD-8101P and -8102P for the Northern States Power Company Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2, dated February 17, 1983.

2. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to the VIPRE-01 Code and WRB-1 Correlation for Facility Operating License Nos. DPR-42 and -60 Northern States Power Company Prairie Island Units 1 and 2 Docket Nos. 50-282 and 50-306, dated May 30, 1983.

3. Letter from D.C. Hintz (WPS) to Document Control Desk, U.S. Nuclear Regulatory Commission, dated February 12, 1988.

4. Letter from D.C. Hintz (WPS) to Document Control Desk, U.S. Nuclear Regulatory Commission, dated March 16, 1988 (with attachments).

5. Letter from A. Schwencer (USNRC) to E.R. Mathews (WPS), dated October 22, 1979.

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ABSTRACT

This document is an updated Topical Report describing the Wisconsin Pubic Service Corporation (WPS) reload safety evaluation and transient analysis methods for application to Kewaunee.

The report addresses the methods for the calculation of cycle specific physics parameters and their comparison to the bounding values used in the safety analyses.

In addition, comparisons of WPS safety analysis results to the Kewaunee Updated Safety Analysis Report (4) are presented to verify WPS safety analysis models and methods.

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1.0 INTRODUCTION

This report addresses the methods for the calculation of Kewaunee cycle specific physics parameters and their comparison to and establishment of the bounding values used in Kewaunee plant transient, reload, and safety analyses. This document is an update of previous submittals which were reviewed by the NRC (3).

A brief description of the general physics calculational procedures is reviewed in Section 2. The specific detailed calculations are controlled by written procedures in accordance with the WPSC Operational Quality Assurance Program (OQAP). General methods are described for each of the key physics parameters of interest in reload safety evaluations.

Cycle specific physics calculations and their comparisons to the safety analyses are described for each accident in Section 3. The specific applications of the reliability factors described in Reference 1 are also presented in this section.

A general description is given in Section 3 of each of the accidents analyzed in the Kewaunee Updated Safety Analysis Report (USAR) (Ref. 4) that is sensitive to core physics parameters and is therefore of concern for a reload evaluation. For each accident, a discussion of the general input assumptions, consequences, and sensitivities to various physics characteristics is provided.

Calculation of core physics parameters for the purpose of performing reload safety evaluations requires an intimate knowledge of the safety analyses to which cycle specific comparisons are to be made. Specifically, one must understand the manner in which the bounding physics parameters have been used in each of the analyses and the conservatisms inherent in the values chosen. In order to acquire such an understanding, Wisconsin Public Service (WPS) has developed models for performing various safety analyses for Kewaunee and has acquired over ten years of experience in performing independent reload core design and safety analysis.

Section 3 discusses the Kewaunee USAR transients as follows:

- A definition of the transient is given describing the physical phenomena involved.

- A description of the analyses methodology to be applied, and the assumptions used in the analysis are given.

- The results of a representative analysis are presented, discussed, and compared to the USAR and/or other independent results as appropriate.

- The sensitive reload parameters used as input to the transient analysis are described and their conservative direction determined, for determination as to whether or not an accident analysis must be reanalyzed to accommodate the behavior of a specific fuel reload.

- Lastly, any limiting safety system setpoint specified in the Kewaunee Technical Specifications (5), which assures a limit to an input parameter, is identified and compared to the conservative specific reload value.

An updated list of bounding safety analyses applicable to Kewaunee is compiled for each Reload Safety Evaluation Report submitted to the NRC (Ref. 6). The specific bounding values for each analysis are provided in the cycle specific Reload Safety Evaluation Report utilizing the most up-to-date analysis methodology.

The computer models applied to Kewaunee were developed in accordance with documented guidelines which accompany each of the computer codes.

The development of the computer models described in this report was controlled by procedures in accordance with the WPS OQAP. The control of these models is periodically audited by WPSC Quality Assurance as well as the NRC (7). A brief description of the models is provided as follows.

Appendix A gives an overview of the computer code package that is used to simulate the transients and accidents listed in this report.

Appendix B gives a description of the DYNODE-P computer code which is used to simulate the transient response of the Nuclear Steam Supply System. DYNODE-P has been reviewed by the NRC and found to be acceptable for safety analysis and licensing applications (8).

Appendix C gives a description of the VIPRE-01 computer code which is used to simulate the thermal hydraulic response of the reactor core and hot coolant subchannel. WPSC participated in the UGRA group which provided the computer analysis on which the NRC based its SER (9). WPSC is therefore experienced and knowledgeable in the application of VIPRE-01 to Kewaunee. A discussion of the WPS thermal margin methodology and various VIPRE sensitivity studies is included in this section.

Appendix D gives a description of the TOODEE 2 computer code which is used to simulate the thermal response of the hot fuel rod and associated coolant channel under transient conditions. A discussion of the WPS fuel thermal response methodology is also included in this appendix.

Appendix E gives a description of the RETRAN-02 computer code which is used to simulate the transient response of the Nuclear Steam Supply System. RETRAN-02 underwent a generic review by the NRC resulting in the issuance of a SER (11).

Appendix F describes the development of WPS best estimate models and presents the results of best estimate analyses using RETRAN and DYNODE compared to Kewaunee plant and simulator data for selected transients.

Appendix G contains additional information requested by the Nuclear Regulatory Commission.

2.0 GENERAL PHYSICS METHODS

In this section the general physics calculational methods are described for application to reload safety evaluations for Kewaunee.

Cycle specific calculations, the application of reliability factors and comparisons to the safety analyses are discussed in Section 3 for each accident considered.

2.1 MODERATOR TEMPERATURE REACTIVITY COEFFICIENT, α_M

Definition: α_M is the change in core reactivity associated with a 1°F change in average moderator temperature at constant average fuel temperature.

Calculations of α_M are performed in three dimensions with the nodal model [1]. The average moderator temperature is varied while the independent core parameters such as core power level, control rod position and RCS boron concentration are held constant. Dependent core parameters such as power distribution and moderator temperature distribution are permitted to vary as dictated by the changes in core neutronics and thermal-hydraulics. The average fuel temperature is held constant and no changes in nodal xenon inventory are permitted.

2.2 POWER REACTIVITY COEFFICIENT, α_P

Definition: α_P is the change in core reactivity associated with a 1% (of full power) change in core average power level.

Calculations of α_P are performed in three dimensions with the nodal model[1]. Core power is varied while all other independent parameters such as rod position and RCS boron concentration are held constant. Dependent core parameters such as power distribution, average fuel and moderator temperatures and moderator temperature distribution are permitted to vary as dictated by the changes in core neutronics and thermal-hydraulics. No changes in nodal xenon inventory are permitted.

2.3 DOPPLER REACTIVITY COEFFICIENT, α_D

Definition: α_D is the change in core reactivity associated with a 1°F change in average fuel temperature at constant average RCS moderator temperature.

α_D is computed as the difference between the power coefficient, α_P, and the moderator coefficient, α_M, as shown below.

$$\alpha_D = \alpha_P * \frac{\partial P}{\partial T_f} - \alpha_M * \frac{\partial TM}{\partial P} * \frac{\partial P}{\partial T_f}$$

2.4 BORON REACTIVITY COEFFICIENT, α_B

Definition: α_B is the change in reactivity associated with a 1PPM change in core average soluble boron concentration.

Calculations of α_B are performed in three dimensions with the nodal model (1). The core average boron concentration is varied while the independent core parameters such as core power level and control rod position are held constant. Dependent core parameters such as power distribution and moderator temperature distribution are permitted to vary as dictated by the changes in core neutronics and thermal hydraulics. No changes in xenon inventory are permitted.

2.5 SHUTDOWN MARGIN, SDM

Definition: SDM is the amount of reactivity by which the core would be subcritical following a reactor trip, assuming the most reactive control rod is stuck out of the core and no changes in xenon or RCS boron concentration.

Calculations of SDM are performed in three dimensions with the nodal model (1). The general calculational sequence is given below.

Case #1 - At power condition with rods at the power dependent insertion limits.

Case #2 - Hot Zero power condition with all rods in except the stuck rod. No changes in xenon or boron are assumed.

Case #3 - Hot Zero power conditions with rods at the positions of case #1.

The dependent core parameters such as power distribution and temperature distribution are permitted to vary as dictated by the changes in core neutronics and thermal-hydraulics. All spatial effects and rod insertion allowances are explicitly accounted for in each calculation. The SDM is computed as the change in core reactivity between case 1 and case 2. This value is conservatively adjusted using case #3 and model reliability factors, RF_i, and biases (1). These uncertainty factors are applied to the inserted rod worth, the moderator defect, and the Doppler defect.

2.6 SCRAM REACTIVITY CURVE, $\Delta\rho_{scram}(t)$

Definition: $\Delta\rho_{scram}(t)$ is the rod worth inserted into the core as a function of time after rod release. The most reactive rod is assumed to remain fully withdrawn.

The independent core parameters such as power level, RCS boron concentration and xenon inventory are held constant during the insertion. Neutron flux, a dependent parameter, is assumed to redistribute instantaneously during rod insertion. However, the effects of moderator and doppler feedback on the scram reactivity shape are not included. The total scram reactivity insertion is conservatively normalized to the minimum shutdown margin.

The reactivity dependence on rod position calculated above is converted into a time dependent function using empirical data relating rod position to time after rod release. The empirical data is normalized such that the total time to full rod insertion is equal to or greater than the limits defined by the Technical Specifications (5).

2.7 NUCLEAR HEAT FLUX HOT CHANNEL FACTOR, FQ

Definition: The maximum local fuel rod linear power density divided by the core average fuel rod linear power density.

Calculations of F_Q are based on three dimensional power distributions obtained with the nodal model [1] coupled with local peak pin to assembly power ratios obtained from the quarter core PDQ model [1]. Statistical factors defined in Reference 1 are applied to increase the F_Q to a conservative value.

2.8 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR, $F_{\Delta H}$

Definition: The maximum integral linear power along a fuel rod rod divided by the core average fuel rod integral power.

Calculations of $F_{\Delta H}$ are based on three dimensional power distributions obtained with the nodal model [1] coupled with the local peak pin to assembly power ratios obtained from the quarter core PDQ model [1]. Statistical factors defined in Reference 1 are applied to increase $F_{\Delta H}$ to a conservative value.

2.9 EFFECTIVE DELAYED NEUTRON FRACTION, β_{eff}

Values for β_i are determined by weighting the delayed neutron fractions from each fissile isotope by the fission sharing of that isotope as determined from PDQ. The importance factor I, applied as .97, conservatively accounts for the effects of reduced fast fissioning, increased resonance escape, and decreased fast leakage by the delayed neutrons. β_{eff} is the product of β and I, where

$$\beta = \sum_{i=1}^{6} \beta i.$$

2.10 PROMPT NEUTRON LIFETIME, ℓ^*

ℓ^* is calculated as a function of core exposure from two dimensional PDQ calculations. This is accomplished by using two group flux weighted PDQ parameters to compute the slowing down time and the thermal diffusion time based on the 1/v nature of the boron cross section.

2.11 FUEL TEMPERATURE, T_f

T_f is calculated as a function of linear heat generation rate. Conditions of maximum fuel densification, low oxide conductivity and low gap conductance are assumed in the analysis.

3.0 SAFETY EVALUATION METHODS

This section addresses the evaluation of the cycle specific physics parameters with respect to the bounding values used in the safety analyses. Specific methods are described for each accident or transient by which the determination is made as to whether or not any reanalysis is required. For each accident or transient the following material is described:

a. Definition of Accident - a brief description of the causes and consequences.

b. Accident Analysis - a brief description of the typical methods employed and discussion of the sensitive physics parameters. Included is a list of the acceptance criteria.

c. WPS Safety Analysis Results - a brief summary of the WPS calculational experience and results of the comparisons of WPS models to the Kewaunee Updated Safety Analysis Report (4).

d. Cycle Specific Physics Calculations - a description of the specific physics calculations performed each cycle for the purposes of a reload safety safety evaluation.

e. Reload Safety Evaluation - a description of the comparisons of the cycle specific physics characteristics and the bounding values used in the safety analysis. Specific applications of the model reliability factors and biases which are determined as described in Reference (1) are also addressed. Biases and reliability factors are to be applied in the following manner:

- Moderator Temperature Coefficient α_M
  Apply in a conservative direction as follows:
  α_M = α_M (MODEL) + $B_M \pm RF_M$
  B_M = Moderator temperature coefficient bias (pcm/°F)
  RF_M = Moderator temperature coefficient reliability factor (pcm/°F)

- Doppler Coefficient α_D
  Apply in a conservative direction as follows:
  α_D = α_D (MODEL)*(1$\pm RF_D$)
  RF_D = Doppler coefficient reliability factor

- Boron Reactivity Coefficient α_B
  Apply in a conservative direction as follows:
  α_B = α_B (MODEL)*(1$\pm RF_B$)
  RF_B = Boron coefficient reliability factor

- Nuclear Heat Flux Hot Channel Factor (F_Q)
  Apply in a conservative direction as follows:
  F_Q = (F_Q(MODEL))*(1+RF_{FQ})*(1+T)
  RF_{FQ} = nuclear heat flux hot channel factor reliability
  T = Technical Specification Tilt Limit

- Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$)
  Apply in a conservative direction as follows:
  $F_{\Delta H}$ = $(F_{\Delta H}(MODEL))*(1+RF_{F\Delta H})*(1+T)$
  $RF_{F\Delta H}$ = nuclear enthalpy rise hot channel factor reliability
  T = Technical Specification Tilt Limit

- Effective Delayed Neutron Fraction: β_{eff}
  Apply in a conservative direction as follows:
  $\beta_{eff} = \beta_{eff}(MODEL)*(1 \pm RF_{\beta})$
  RF_{β} = β_{eff} relative reliability factor

- Prompt Neutron Lifetime (ℓ^*)
  Apply in a conservative direction as follows:
  $\ell^* = \ell^*(MODEL)*(1 \pm RF_{\ell^*})$
  RF_{ℓ^*} = relative prompt neutron lifetime reliability factor

- Scram Reactivity $\Delta\rho$ scram(t)
  Apply in a conservative direction as follows:
  $\Delta\rho$ scram(t) = $\Delta\rho$ scram(t)(MODEL) x $(1 - RF_R)$
  RF_R = rod worth relative reliability factor

- Rod Worth ($\Delta\rho_R$)
  Apply in a conservative direction as follows:
  $(\Delta\rho_R) = \Delta\rho_R(MODEL)*(1 \pm RF_R)$
  RF_R = rod worth relative reliability factor

- Fuel Temperature (T_f)
  Apply in a conservative direction as follows:
  $T_f = T_f(MODEL) * (1 \pm RFT_f)$
  RFT_f = fuel temperature relative reliability factor

The specific numerical values assigned as the bounding values for each accident for purposes of performing the Kewaunee reload safety evaluations are presented in the cycle specific Reload Safety Evaluation Reports (6).

If an accident or transient requires reanalysis because any one of the cycle specific physics parameters exceeds the current bounding value, the reanalysis will be performed utilizing the transient analysis methodology as described herein for that specific event and which has been qualified by the presented results. If the parameter exceeded involves a Technical Specification Limit the reanalysis will be submitted to the NRC in support of the appropriate Technical Specification amendment.

3.1 UNCONTROLLED RCCA WITHDRAWAL FROM A SUB-CRITICAL CONDITION

3.1.1 DESCRIPTION OF THE ACCIDENT

An uncontrolled addition of reactivity due to an uncontrolled withdrawal of a Rod Cluster Control Assembly (RCCA) results in a power excursion. The nuclear power response is characterized by a very fast rise terminated by the reactivity effect of the negative fuel temperature coefficient. After the initial power burst, the reactor power is reduced by this inherent feedback and the accident is terminated by a reactor trip on high nuclear power. Due to the small amount of energy released to the core coolant, pressure and temperature excursions are minimal during this accident.

3.1.2 SUMMARY OF ACCIDENT ANALYSIS

The uncontrolled RCCA withdrawal from a sub-critical condition is analyzed using a dynamic simulation incorporating point neutron kinetics, including delayed neutrons and decay heat; fuel, clad, and gap heat conduction; and channel coolant thermal-hydraulics. The reactivity effects due to moderator and fuel temperature effects, as well as that due to control rod insertion after trip, are included.

The core is assumed initially to be at hot zero power, HZP. Power is supplied to the RCCA drive mechanisms such that no more than two banks may be withdrawn simultaneously. The maximum reactivity insertion due to the rods is therefore conservatively assumed as that due to two banks of maximum worth moving simultaneously at maximum speed through the region of highest differential worth.

The magnitude of the power peak reached during the transient is strongly dependent upon the Doppler reactivity coefficient for a given rate of reactivity insertion. A value conservatively small in absolute magnitude, which generally occurs at Beginning of Cycle (BOC), is assumed for the accident analysis. The magnitude of the power spike is relatively insensitive to the value of moderator temperature reactivity coefficient chosen. The least negative value, occurring at BOC, maximizes the calculated consequences of the accident. For conservatism, however, a positive value is used in the analysis.

In calculating reactivity due to control rod insertion by reactor trip, the most adverse combination of instrument and setpoint errors and time delays is assumed. The power range - low range trip setpoint is assumed to be 10% (of full power) above its nominal value. The most reactive rod is assumed to stick in the fully withdrawn position when the trip signal is actuated.

As long as the reactivity insertion remains small compared to β_{eff}, the total delayed neutron yield, the shortest reactor period during the transient will remain large compared to ℓ^*, the prompt neutron lifetime. In this case, the transient core power response is relatively insensitive to the value of ℓ^* and is determined predominantly by the yields and decay constants of the delayed neutron precursors. The transient power response is sensitive to ℓ^* in cases where ultra-conservative reactivity insertion rates, in which prompt criticality is achieved, are assumed. The postulated initial core pressure and temperature are conservatively taken as the minimum and maximum, respectively, consistent with the assumed rod and power configurations.

The results of the analysis are compared to the following acceptance criteria:

a. The maximum power density in the fuel must be less than that at which centerline melting or other modes of fuel failure occur.

b. The minimum departure from nucleate boiling ratio (DNBR) calculated using the W-3 correlation must be greater than 1.30.

3.1.3 WPS SAFETY ANALYSIS RESULTS

WPS has analyzed the uncontrolled RCCA withdrawal from a subcritical condition transient using input assumptions consistent with the Kewaunee USAR (4).

The models described in Appendix A were used to analyze the case of a rapid, $8.2 * 10^{-4}$ $\Delta k/sec$, RCCA withdrawal from subcritical conditions. The results of these calculations are compared to USAR Figures 14.1-2, 14.1-4, and 14.1-5 in Figures 3.1-1 through 3.1-5 of this report.

The WPS model predicts higher fuel, clad, and coolant temperatures than those of the USAR, however, the nuclear power and core average heat flux results compare well to the USAR, thereby demonstrating consistency in the doppler and moderator reactivity coefficients used.

A sensitivity study showing the effect of initial power level on peak heat flux was performed and the results are compared to Figure 14.1-2 of Ref. 4 in Figure 3.1-1. The results indicate that the maximum peak heat flux occurs following a rod withdrawal from the minimum initial power level. WPS models predict slightly higher peak heat fluxes.

3.1.4 CYCLE SPECIFIC PHYSICS CALCULATIONS

a. DOPPLER TEMPERATURE COEFFICIENT, α_D

Calculations of α_D are performed in accordance with the general procedures described in Section 2. Cycle specific calculations for this accident are performed as a function of power level over a range of 0 - 50 percent power at BOC and EOC.

b. MODERATOR TEMPERATURE COEFFICIENT, α_M

Calculations of α_M are performed in accordance with the general procedures described in Section 2. Cycle specific calculations for this accident are performed at BOC for the HZP, unrodded, no xenon core condition. This produces the least negative moderator coefficient due to both the unrodded and high critical boron concentration core core conditions.

c. MAXIMUM REACTIVITY INSERTION RATE, $\Delta\rho/\Delta t$

In order to compare with the reactivity insertion rate assumed by the safety analysis for uncontrolled rod withdrawal transients, the assumption is made that two banks of highest worth will be withdrawn simultaneously at maximum speed. This value requires two components. First, the maximum withdrawal speed is required in inches per second. A maximum value for Kewaunee is 0.76 in/sec. The second component is the maximum differential reactivity insertion per inch for the two maximum worth rod banks moving in 100% overlap. This has been obtained by first calculating the two banks which have the maximum worths and then moving these two banks simultaneously at HZP conditions in an area of highest differential worth. These calculations are performed at both BOC and EOC. Finally the reactivity insertion rate is divided by the minimum βeff determined according to (e) of this subsection to yield the maximum reactivity insertion rate in dollars.

d. SCRAM REACTIVITY CURVE, $\Delta\rho_{SCRAM}(t)$

Calculations of the scram reactivity are performed in accordance with the general procedures described in Section 2. Cycle specific calculations for this accident are performed at BOC and EOC for the zero power condition. A conservatively slow scram curve is generated by making the following assumptions:

1. The integral of the scram curve is based on an initial rod position at or below the zero power insertion limits specified in the Technical Specifications (5).

2. The shape of the scram curve is based on an initial rod position of full out. This provides the longest possible delay to significant negative reactivity insertion.

3. The xenon distribution is that which causes the minimum shutdown margin.

4. Instantaneous redistribution of flux is assumed to occur during the rod insertion.

e. EFFECTIVE DELAYED NEUTRON FRACTION, β_{eff}

Calculations of β_{eff} are performed in accordance with the general procedures described in Section 2. Cycle specific calculations are performed at BOC and EOC.

f. PROMPT NEUTRON LIFETIME, ℓ^*

The value of ℓ^* is calculated in accordance with the general procedures given in Section 2. Cycle specific calculations are performed at BOC and EOC.

3.1.5 RELOAD SAFETY EVALUATIONS

Each of the physics parameters calculated above is adjusted to include the model reliability factors, RF_i (1) and biases. These adjusted values are the cycle specific parameters which are then compared to the bounding values assumed in the safety analysis.

The cycle specific parameters are acceptable if the following inequalities are met:

CYCLE SPECIFIC PARAMETERS		SAFETY ANALYSIS PARAMETERS
a. $\alpha_M + RF_M + B_M$	$\leq$	α_M (least negative bounding value)
b. α_D * (1-RFD)	$\leq$	α_D (least negative bounding value)
c. $\beta eff * (1-RF_\beta)$	$\geq$	βeff (minimum)
d. $\left\lvert \dfrac{\Delta\rho/\Delta t * (1+RF_{RODS})}{\beta eff * (1 - RF_\beta)} \right\rvert$	$\leq$	$\left\lvert \dfrac{\Delta\rho/\Delta t \text{ (bounding)}}{\beta eff \text{ (maximum)}} \right\rvert$
e. $\lvert \Delta\rho_{SCRAM}(t) * (1-RF_{RODS}) \rvert$	$\geq$	$\lvert \Delta\rho_{SCRAM}(t) \rvert$ (bounding)
f. $\ell^* * (1-RF\ell^*)$	$\geq$	ℓ^* (minimum)

The integral of the bounding value of the scram curve, $\Delta\rho_{SCRAM}(t)$, is taken as that rod worth required to produce the shutdown margin assumed in the safety analysis for the most limiting cycle specific core conditions discussed in 3.1.4d above.

FIGURE 3.1-1

KEWAUNEE NUCLEAR POWER PLANT

UNCONTROLLED WITHDRAWAL FROM SUBCRITICAL

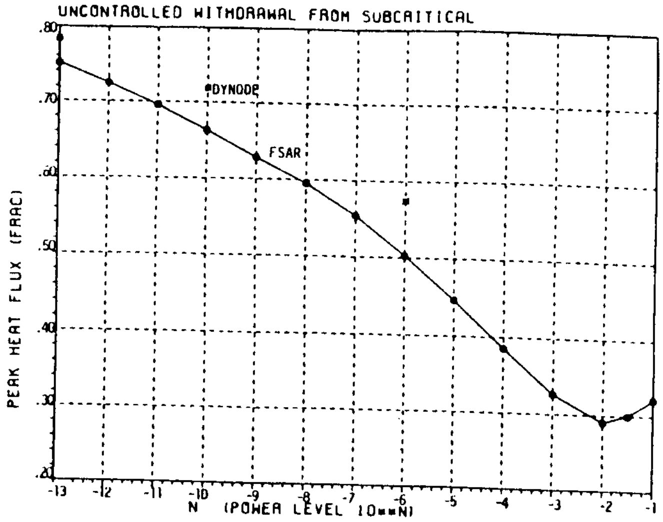


FIGURE 3.1-2

KEWAUNEE NUCLEAR POWER PLANT

UNCONTROLLED ROD WITHDRAWAL SUBCRITICAL

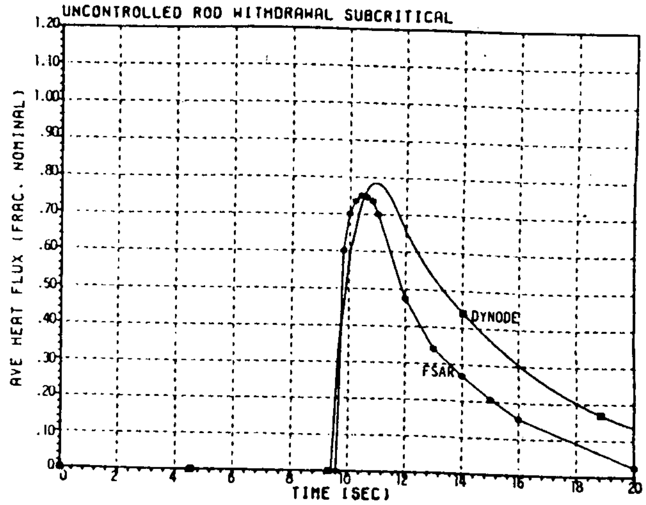


FIGURE 3.1-3

KEWAUNEE NUCLEAR POWER PLANT

UNCONTROLLED ROD WITHDRAWAL SUBCRITICAL

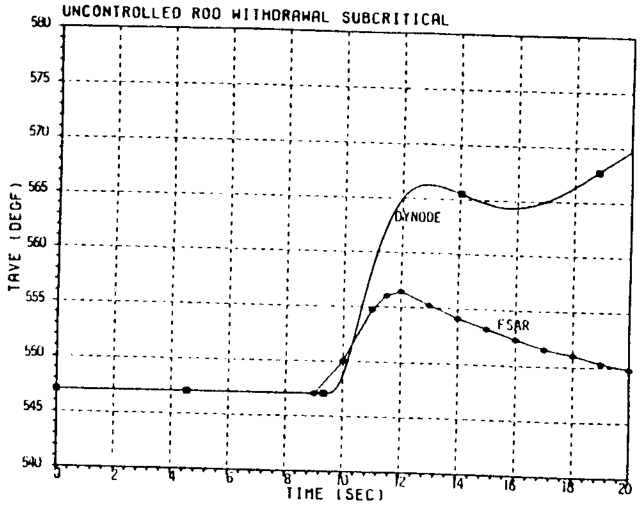


FIGURE 3.1-4

KEWAUNEE NUCLEAR POWER PLANT

UNCONTROLLED ROD WITHDRAWAL SUBCRITICAL

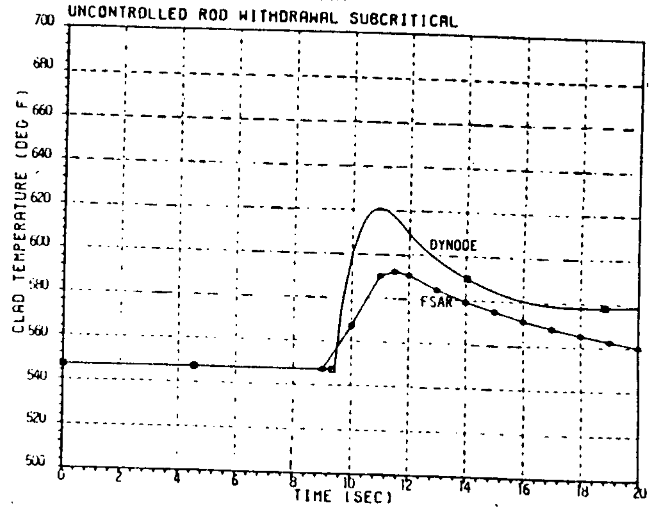
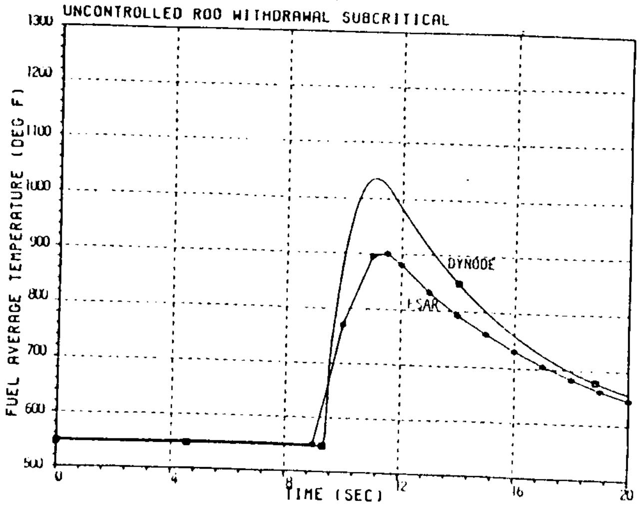


FIGURE 3.1-5

KEWAUNEE NUCLEAR POWER PLANT

UNCONTROLLED ROD WITHDRAWAL SUBCRITICAL



3.2 UNCONTROLLED RCCA WITHDRAWAL AT POWER

3.2.1 DEFINITION OF ACCIDENT

An uncontrolled RCCA withdrawal at power results in a gradual increase in core power followed by an increase in core heat flux. The resulting mismatch between core power and steam generator heat load results in an increase in reactor coolant temperature and pressure. The reactor core would eventually suffer departure from nucleate boiling if the power excursion were not checked by the reactor protection system. Depending on the initial power level and rate of reactivity insertion the following trips serve to prevent fuel damage or over-pressurization of the coolant system: nuclear power, core coolant ΔT, high pressurizer level, and high pressurizer pressure. For the more rapid rates of reactivity insertion, the maximum power reached during the transient will exceed the power at the time the trip setpoint is exceeded by an amount proportional to the insertion rate and the time delay associated with trip circuitry and rod motion.

3.2.2 SUMMARY OF ACCIDENT ANALYSIS

The uncontrolled RCCA withdrawal at a power condition is analyzed using a dynamic simulation incorporating point neutron kinetics; reactivity effects of moderator fuel and control rods; and decay heat. A simulation of the reactor vessel, steam generator tube and shell sides, pressurizer, and connecting piping is required to evaluate the coolant pressure and core inlet temperature response and their effect on core thermal margins. The reactor trip system, main steam and feedwater systems, and pressurizer control systems are also included in the model. This model calculates the response of the average core channel thermal-hydraulic conditions and heat generation and is coupled to a detailed model of the hot channel. This latter model calculates the departure from nucleate boiling ratio (DNBR) as a function of time during the accident.

In order to maximize the peak power during the transient the fuel and moderator temperature coefficients used in the analysis are the least negative values likely to be encountered. Although during normal operation the moderator coefficient will not be positive at any time in core life, a value of zero or slightly greater, may be conservatively assumed for the purposes of the analysis. The least negative fuel and moderator temperature coefficients are normally encountered at BOC.

The reactivity insertion due to reactor trip is calculated by considering the most adverse combination of instrument and setpoint errors and time delays. The rate of reactivity insertion corresponding to the trip of the RCC assemblies is calculated assuming that the most reactive assembly is stuck in the fully withdrawn position.

Since the reactivity insertion rate determines which protective system function will initiate termination of the accident, a range of insertion rates must be considered. Relatively rapid insertion rates result in reactor trip due to high nuclear power. The maximum rate is bounded by that calculated assuming that the two highest worth banks, both in their region of highest incremental worth, are withdrawn at their maximum speed. Relatively slow rates of reactivity insertion result in a slower transient which is terminated by an overtemperature ΔT trip signal, or in some cases, a high pressurizer level signal. The minimum rate which need be considered in the analysis is determined by reducing the reactivity insertion rates until the analysis shows no further change in DNBR.

The acceptance criteria for this accident are that the maximum pressures in the reactor coolant and main steam systems do not exceed 110% of design values and that cladding integrity be maintained by limiting the minimum DNB ratio greater than 1.30.

3.2.3 WPS SAFETY ANALYSIS RESULTS

WPS has analyzed a spectrum of control rod withdrawal transients using input consistent with the USAR (4).

The models described in Appendix A were used to analyze the following four control rod withdrawal transients:

* Fast Rate from Full Power (FRFP)
* Slow Rate from Full Power (SRFP)
* Fast Rate from Intermediate Power (FRIP)
* Slow Rate from Intermediate Power (SRIP)

In addition to DYNODE-P, the NSSS response to the FRFP and SRFP uncontrolled rod withdrawal transient was predicted by RETRAN-02 for comparison. The transient forcing function inputs to VIPRE required for MDNBR analysis were derived from the OYNODE-P results in all cases.

The results of these calculations are compared to the corresponding cases (same initial power and reactivity insertion rate) reported in Section 14.1-2 of the USAR.

The transient response of the NSSS and hot channel for the FRFP case are compared with the results of Figures 14.1-6 and 14.1-7 of the USAR in Figures 3.2-1 and 3.2-4 of this report. The reactor trip is actuated on high neutron power for this case. NSSS parameter trends predicted by the WPS models (both DYNODE and RETRAN) are in good agreement with the USAR results. MDNBR is predicted slightly higher than the USAR results due to a slightly earlier reactor trip and therefore slightly less severe power response.

A similar comparison for the SRFP case is shown in Figures 3.2-5 to 3.2-8, where the results of the WPS models (both DYNODE and RETRAN) are compared to Figures 14.1-8 and 14.1-9 of the KNPP USAR. The WPS results show a slower power increase than the USAR results and reactor trip occurs about three seconds later. Because the power and Tave responses are less severe in the WPS model, the MDNBR is slightly higher than the USAR value. Reactor trip is caused by the overtemperature - ΔT trip function in both the WPS and USAR analyses.

Figure 3.2-9 shows the comparison of the WPS model minimum DNBR results with those of Figure 14.1-10 of the USAR. It should be noted that the USAR results are given over a wide range of reactivity insertion rates (2×10^{-6} to 8×10^{-4} Δk/second) while the WPS model was used only to analyze the two cases indicated in Figure 3.2-9. The two cases selected represent typical uncontrolled rod withdrawal transients which are terminated by the two important Reactor Protection System trip functions for this type of transient; the high nuclear power and overtemperature ΔT trips. The minimum DNBR calculated in the Chemical and Volume Control System Malfunction analysis (described in Section 3.5) is also included on Figure 3.2-9 for comparison.

The responses of the NSSS and the hot channel for the fast and slow rate from Intermediate Power cases (FRIP and SRIP) are shown in Figures 3.2-10 to 3.2-13 and 3.2-14 to 3.2-17, respectively, of this report. The USAR reports only the minimum DNBR for these cases, and the results of the WPS single channel VIPRE model are compared to those of Figure 14.1-11 in Figure 3.2-18 of this report.

The consequences of uncontrolled RCCA withdrawal accidents were computed using the methodology described in Appendix A. Insertion rates of 8.2×10^{-4} and 3×10^{-5} Δk/second, at full power; and 8.2×10^{-4} and 1×10^{-4} Δk/second, at 60% of full power were considered. Sensitivity studies were performed for the slower rates of reactivity insertion to determine the effect of parameters in the overtemperature ΔT trip setpoint formulation on time of reactor trip and minimum DNBR.

The WPS hot channel DNBR analyses were computed using a single channel model, a multi-channel 1/8 assembly model, and a 1/8 core lumped subchannel model. Based on the comparison of results for the full power rod withdrawal transients, the single channel model provides the most conservative MDNBR results and the best comparison to the USAR results.

3.2.4 CYCLE SPECIFIC PHYSICS CALCULATIONS

a. DOPPLER TEMPERATURE COEFFICIENT, α_D

Calculations of α_D are performed in accordance with the general procedures described in Section 2. Cycle specific calculations for this accident are performed as a function of power level over the range 0 to 100% power at BOC and EOC.

b. MODERATOR TEMPERATURE COEFFICIENT, α_M

Calculations of α_M are performed in accordance with the general procedures described in Section 2. Cycle specific calculations for this accident are performed at BOC to determine the moderator coefficients over the operating range of 0 - 100% power under various conditions of xenon inventory. The model Bias, Bm, is applied to the moderator temperature coefficient as shown in section 3.2.5.

c. MAXIMUM REACTIVITY INSERTION RATE, $\Delta\rho/\Delta t$

Calculations similar to those described in Section 3.1.4(c) are performed at the full power, equilibrium xenon conditions. The reactivity insertion rate is divided by the minimum cycle βeff determined according to (f) of this subsection to yield the maximum reactivity insertion rate in dollars.

d. SCRAM REACTIVITY CURVE, $\Delta\rho_{SCRAM}(t)$

Calculations of the scram reactivity curve are performed in accordance with the general procedures described in Section 2. Cycle specific calculations for this accident are performed at BOC and EOC for the full and zero power conditions. A conservatively slow scram curve is generated by making the following assumptions:

1. The integral of the scram curve is based on an initial rod position at or below the power dependent insertion limits.

2. The shape of the scram curve is based on an initial rod position of full out. This provides the longest possible delay to significant reactivity insertion.

3. The xenon distribution is that which causes the minimum shutdown margin.

4. Instantaneous redistribution of flux is assumed to occur during the rod insertion.

e. NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR, $F_{\Delta H}$

Calculations of $F_{\Delta H}$ are performed in accordance with the general procedures described in Section 2. Maximum core $F_{\Delta H}$s are verified to remain within Technical Specification limits for allowable combinations of axial offset, power level, and control rod insertion. For Kewaunee, the continuous surveillance of the power distribution is accomplished with the ex-core detectors using a Power Distribution Control scheme, PDC-II (12). The cycle specific physics calculations performed for the verification of the PDC-II scheme with respect to the $F_{\Delta H}$ limits are described in Section 3.17.

f. EFFECTIVE DELAYED NEUTRON FRACTION, β_{eff}

Calculations of βeff are performed in accordance with the general procedures described in Section 2. Cycle specific calculations are performed at BOC and EOC.

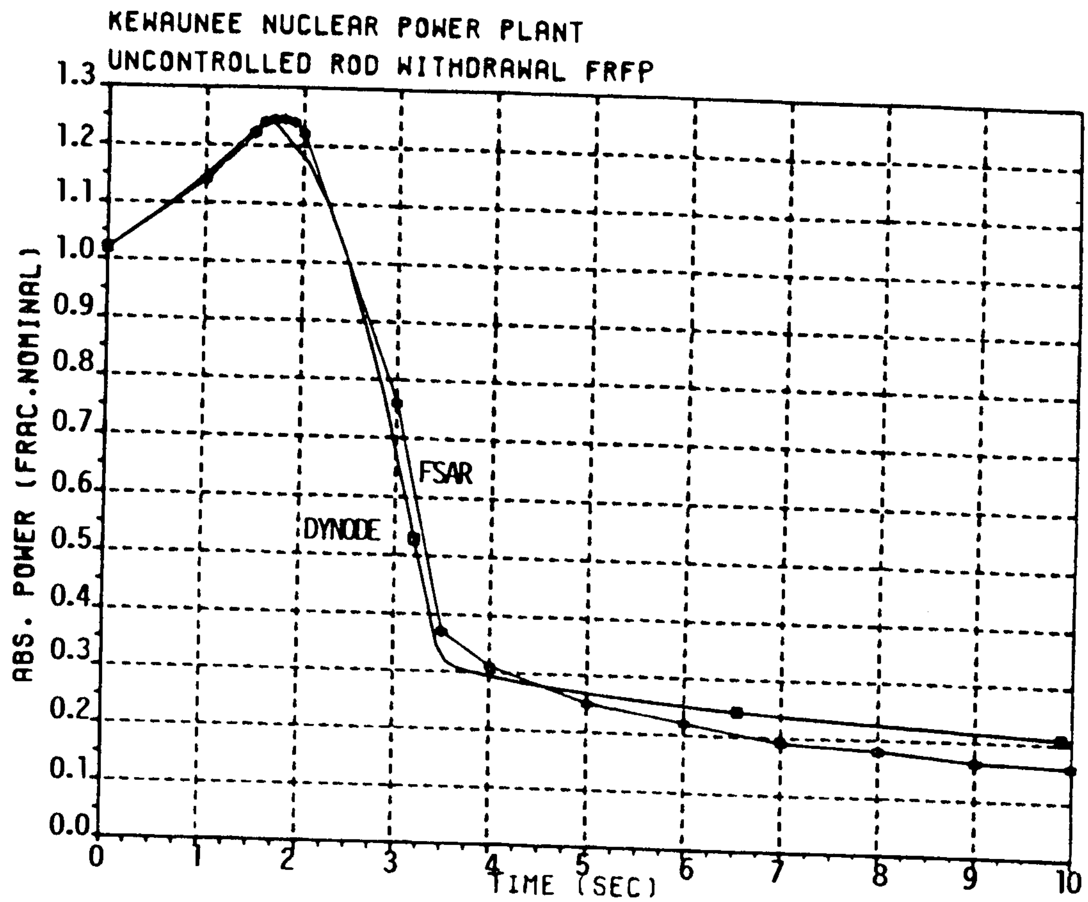
3.2.5 RELOAD SAFETY EVALUATIONS

Each of the physics parameters calculated above is adjusted to include the model reliability factors, RF_i(1). These adjusted values are the cycle specific parameters which are then compared to the bounding values assumed in the safety analysis. The cycle specific parameters are acceptable if the following inequalities are met:

CYCLE SPECIFIC PARAMETERS		SAFETY ANALYSIS PARAMETERS
a. $\alpha_D*(1 - RF_D)$	$\leq$	α_D (least negative bounding value)
b. $\alpha_M+RF_M+B_M$	$\leq$	α_M (least negative bounding value)
c. $\left\lvert \frac{\Delta\rho/\Delta t*(1+RF_{RODS})}{\beta eff * (1-RF_\beta)} \right\rvert$	$\leq$	$\left\lvert \frac{\Delta\rho/\Delta t \text{ (bounding)}}{\beta eff \text{ (maximum)}} \right\rvert$
d. $\lvert \Delta\rho_{SCRAM}(t)*(1-RF_{RODS}) \rvert$	$\geq$	$\lvert \Delta\rho_{SCRAM}(t) \text{ (bounding)} \rvert$
e. $F_{\Delta H}*(1+RFF_{\Delta H})*(1+T)$	$\leq$	Technical Specifications (Refer to Section 3.17)
f. $\beta_{eff} * (1-RF_\beta)$	$\geq$	β_{eff} (minimum)

The integral of the bounding value of the scram curve, $\Delta\rho_{SCRAM}(t)$, is taken as that rod worth required to produce the shutdown margin assumed in the safety analysis for the most limiting cycle specific core conditions discussed in 3.2.4d above.

FIGURE 3.2-1


KEWAUNEE NUCLEAR POWER PLANT
UNCONTROLLED ROD WITHDRAWAL FRFP
ABS. POWER (FRAC.NOMINAL)
0.0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1.0
1.1
1.2
1.3
FSAR
DYNODE
TIME (SEC)
0
1
2
3
4
5
6
7
8
9
10

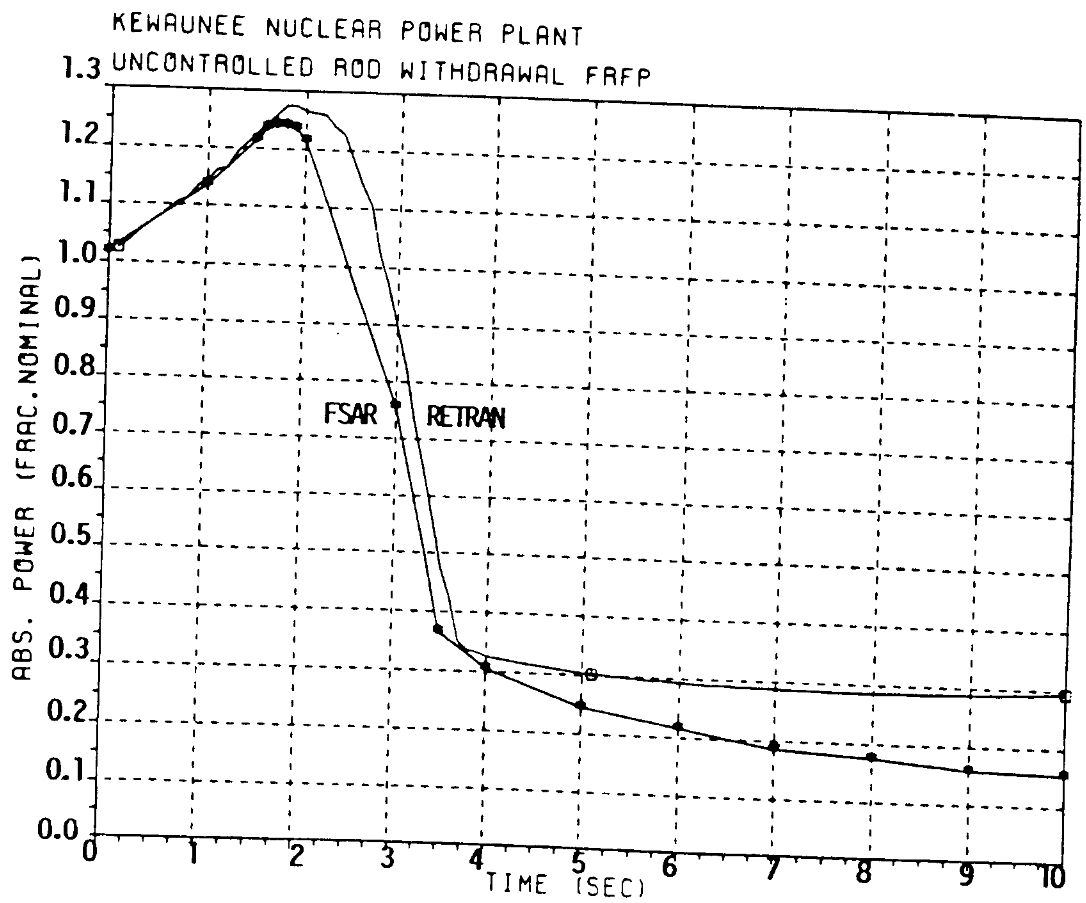
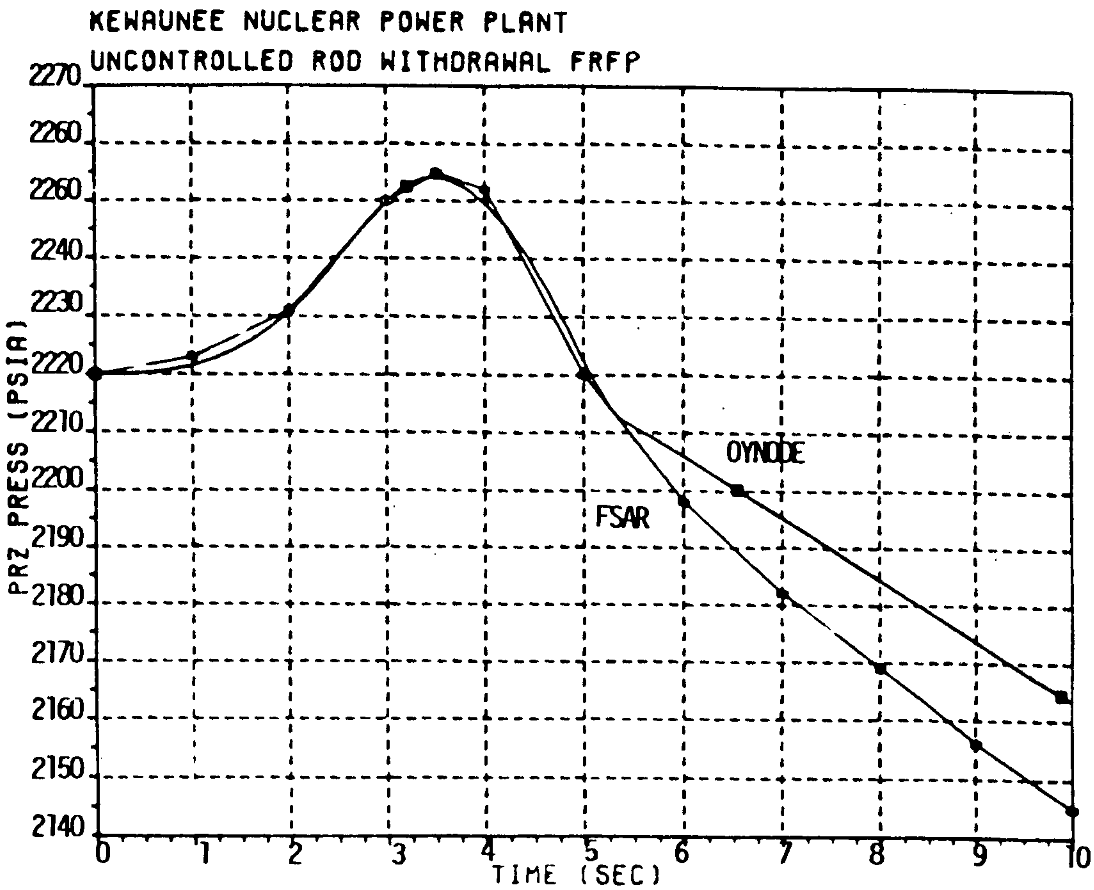
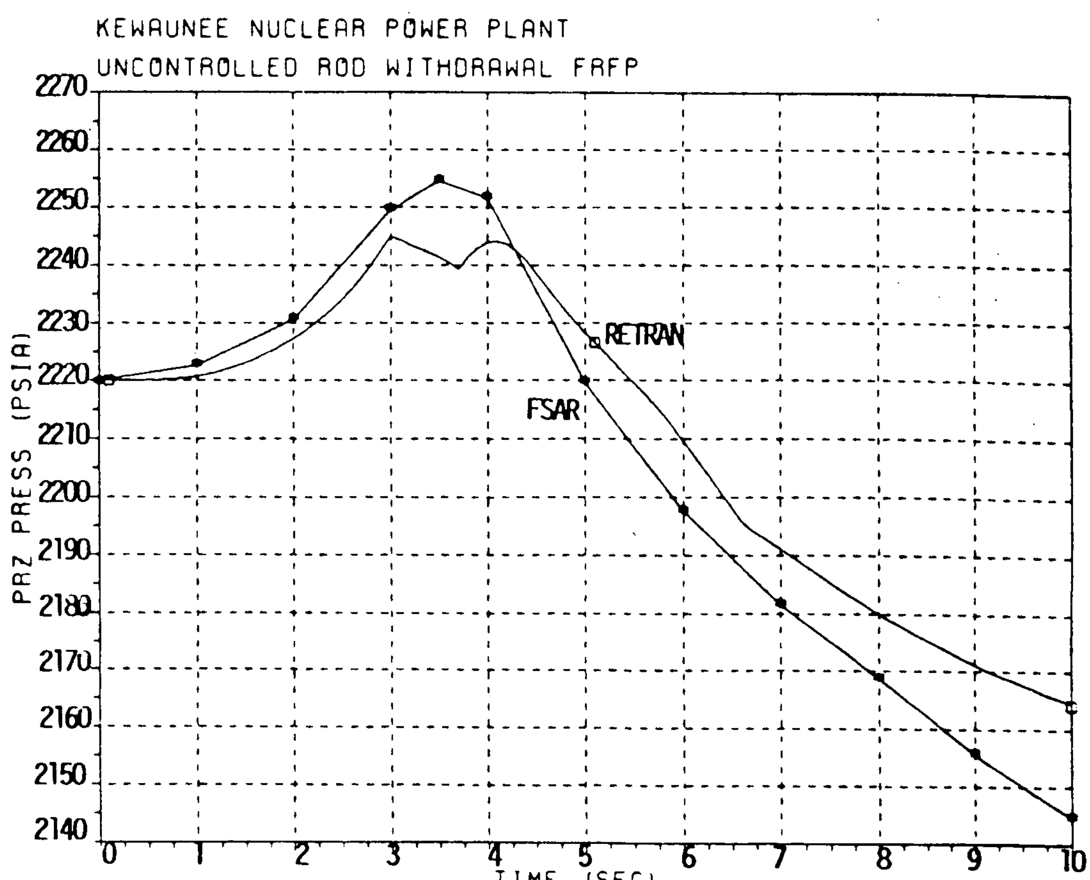

KEWAUNEE NUCLEAR POWER PLANT
UNCONTROLLED ROD WITHDRAWAL FRFP
ABS. POWER (FRAC.NOMINAL)
0.0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1.0
1.1
1.2
1.3
FSAR
RETRAN
TIME (SEC)
0
1
2
3
4
5
6
7
8
9
10

FIGURE 3.2-2

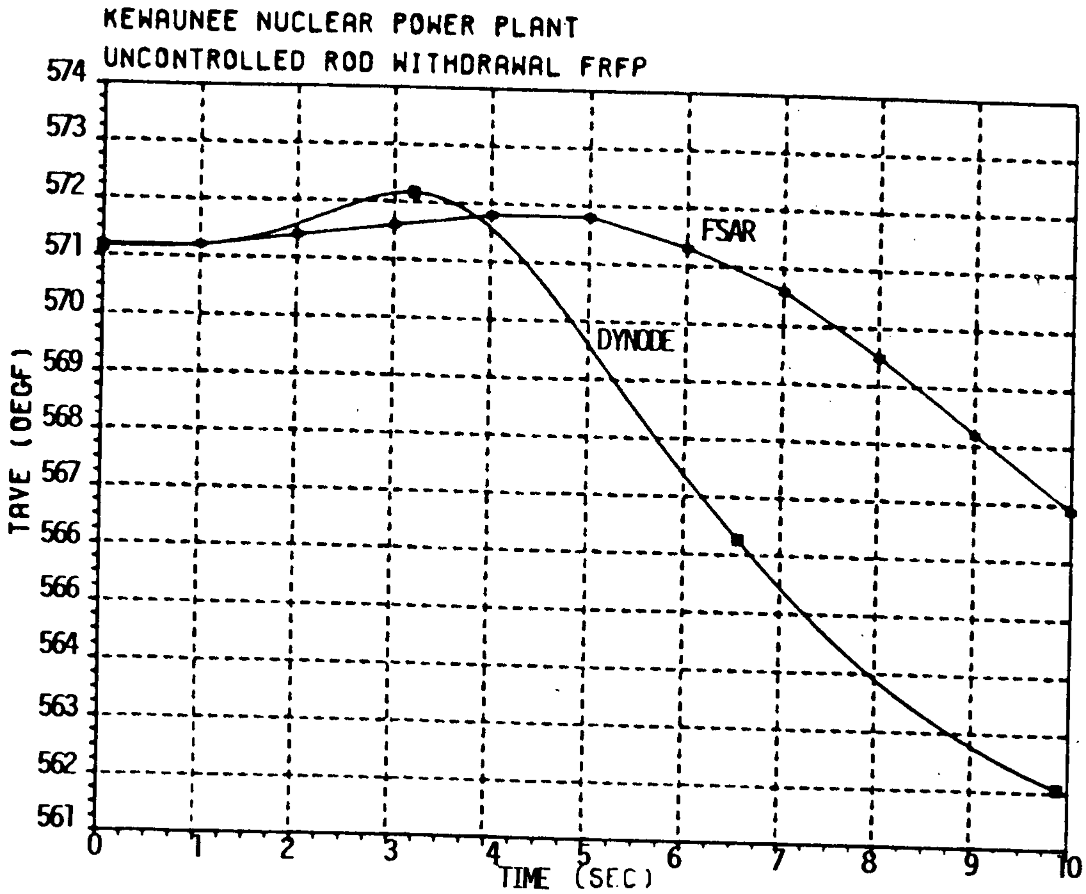


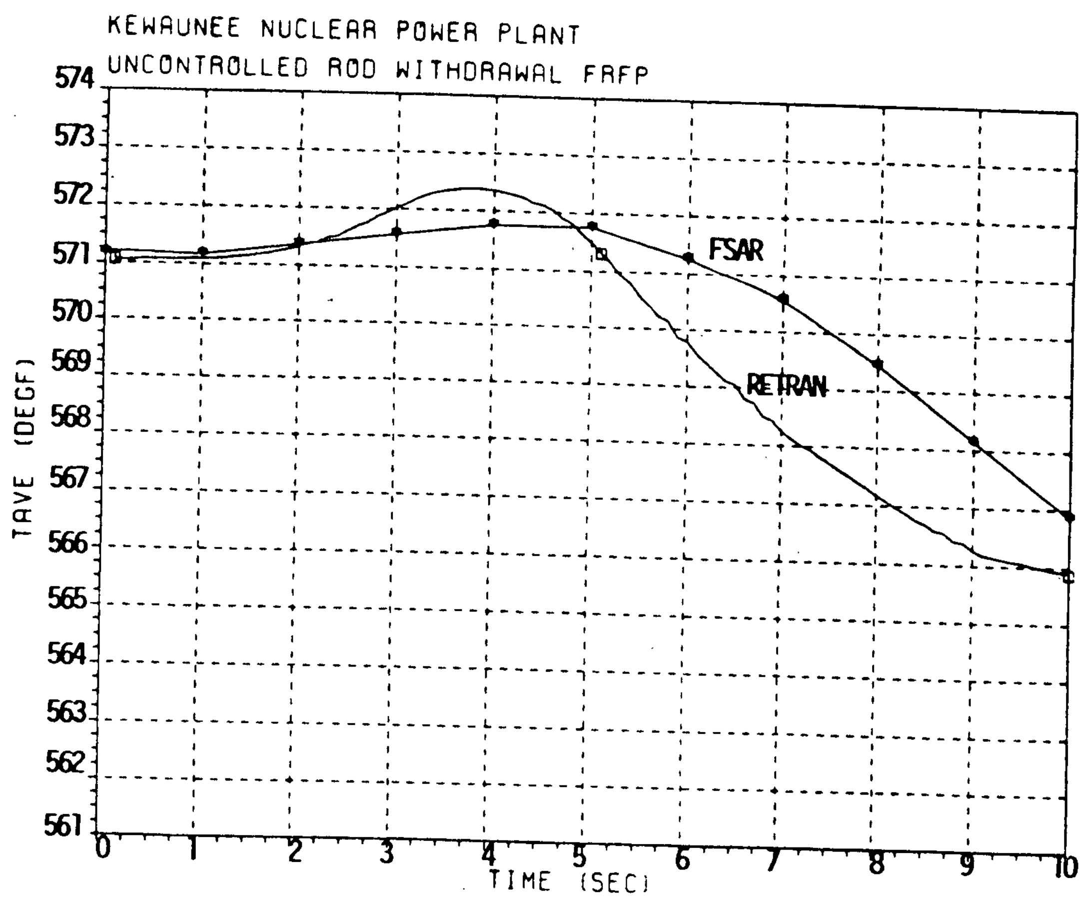
KEWAUNEE NUCLEAR POWER PLANT
UNCONTROLLED ROD WITHDRAWAL FRFP
PRZ PRESS (PSIA)
TIME (SEC)
OYNODE
FSAR



KEWAUNEE NUCLEAR POWER PLANT
UNCONTROLLED ROD WITHDRAWAL FRFP
PRZ PRESS (PSIA)
TIME (SEC)
RETRAN
FSAR

FIGURE 3.2-3


KEWAUNEE NUCLEAR POWER PLANT
UNCONTROLLED ROD WITHDRAWAL FRFP
574
573
572
571
570
569
568
567
566
566
564
563
562
561
TAVE (DEGF)
FSAR
DYNODE
0
1
2
3
4
5
6
7
8
9
10
TIME (SEC)


KEWAUNEE NUCLEAR POWER PLANT
UNCONTROLLED ROD WITHDRAWAL FRFP
574
573
572
571
570
569
568
567
566
565
564
563
562
561
TAVE (DEGF)
FSAR
RETRAN
0
1
2
3
4
5
6
7
8
9
10
TIME (SEC)

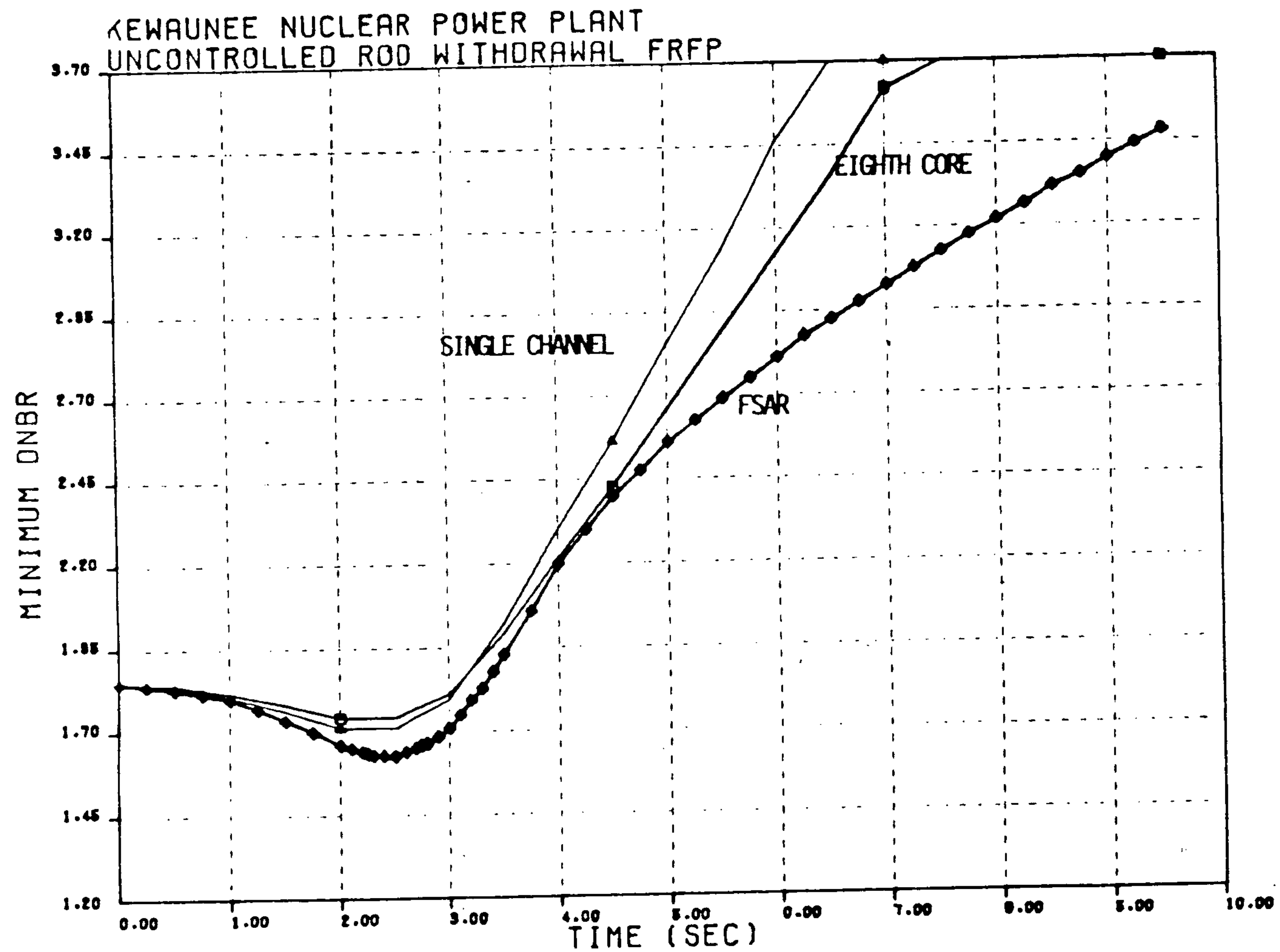
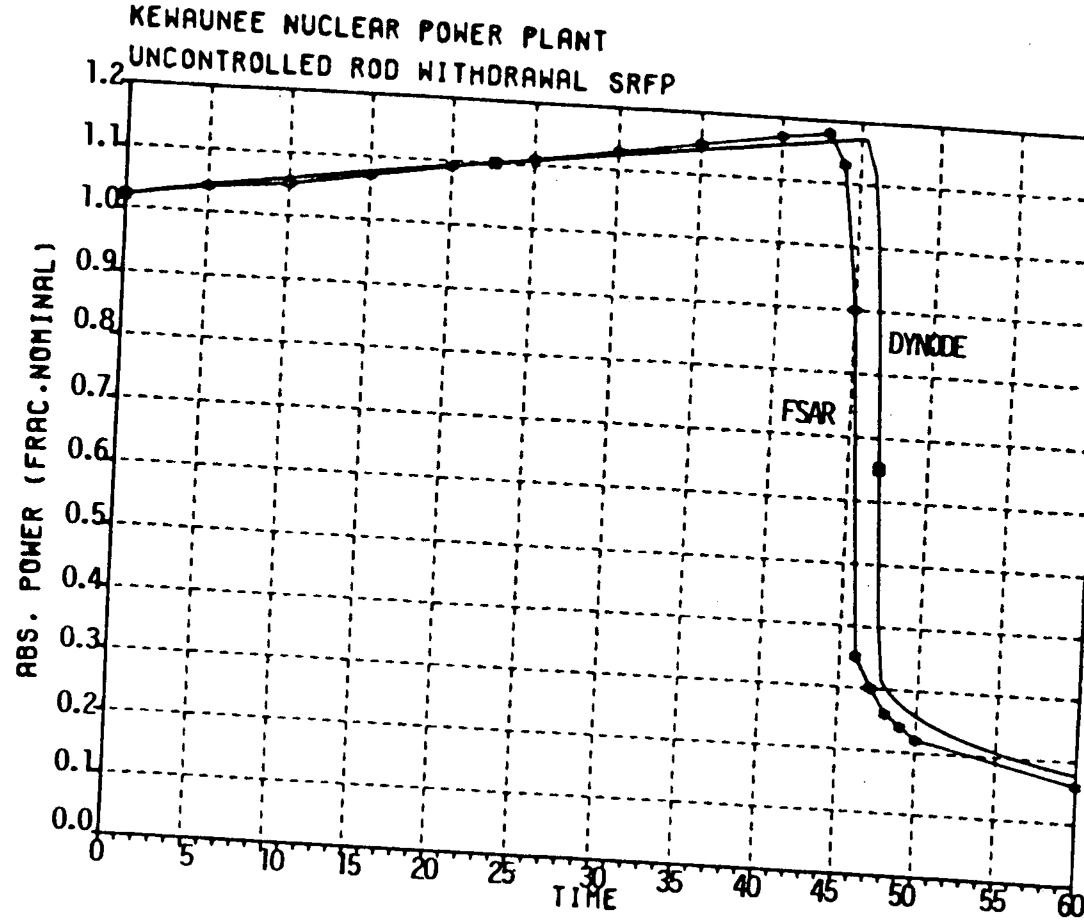

KEWAUNEE NUCLEAR POWER PLANT
UNCONTROLLED ROD WITHDRAWAL FRFP
MINIMUM DNBR
TIME (SEC)
SINGLE CHANNEL
EIGHTH CORE
FSAR

FIGURE 3.2-4

FIGURE 3.2-5


KEWAUNEE NUCLEAR POWER PLANT
UNCONTROLLED ROD WITHDRAWAL SRFP
ABS. POWER (FRAC.NOMINAL)
1.2
1.1
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0.0
DYNODE
FSAR
TIME
0
5
10
15
20
25
30
35
40
45
50
55
60

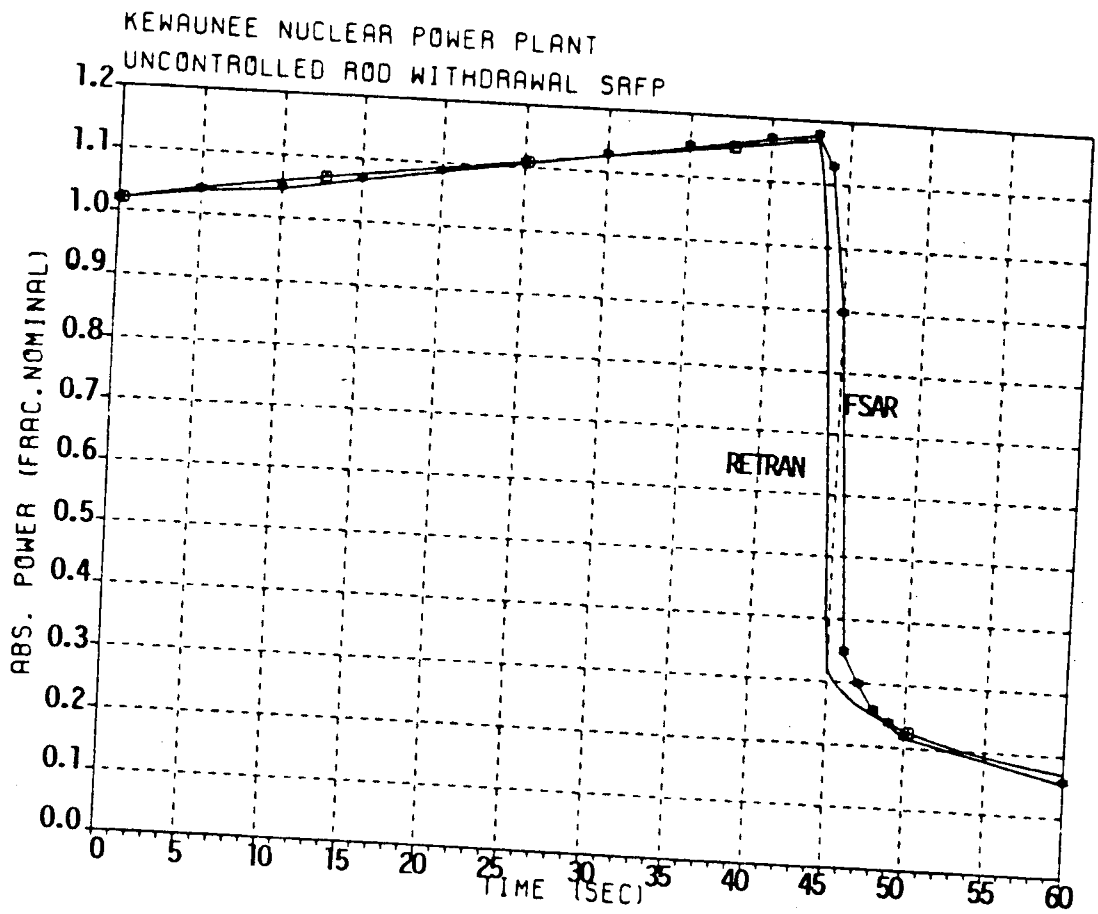
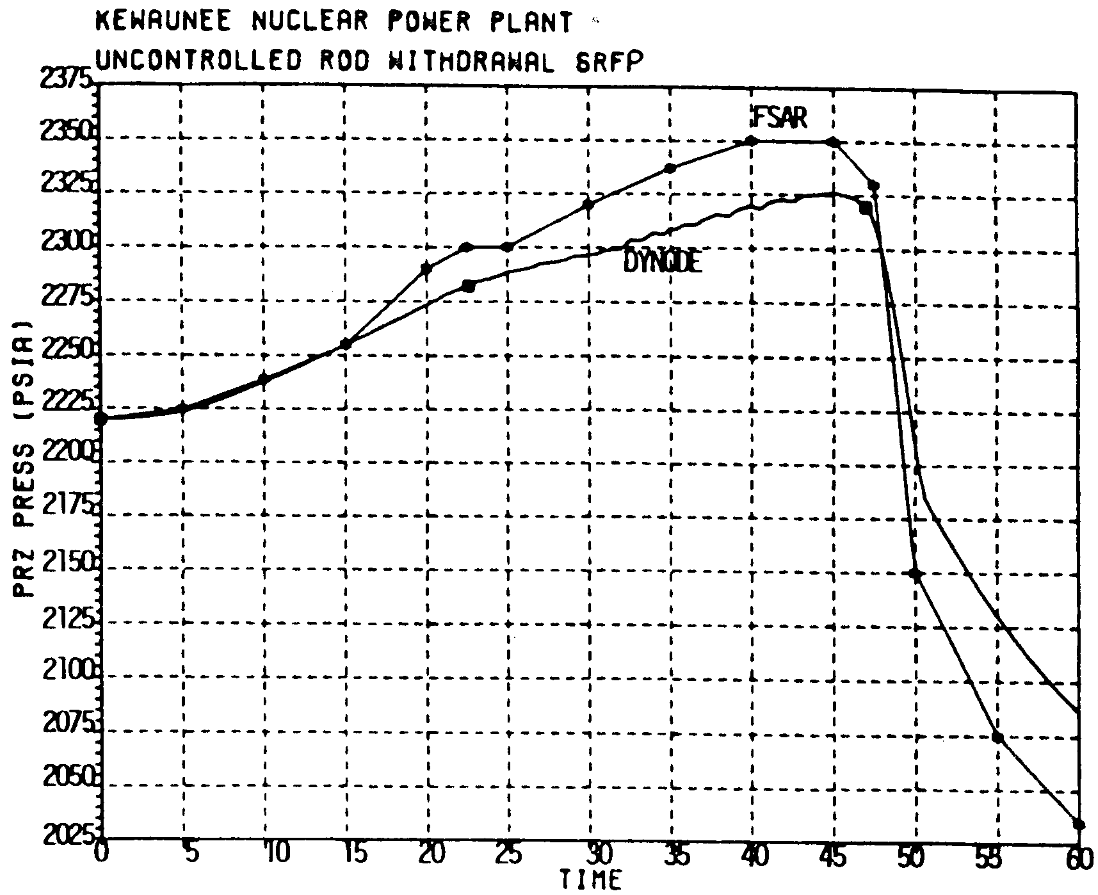
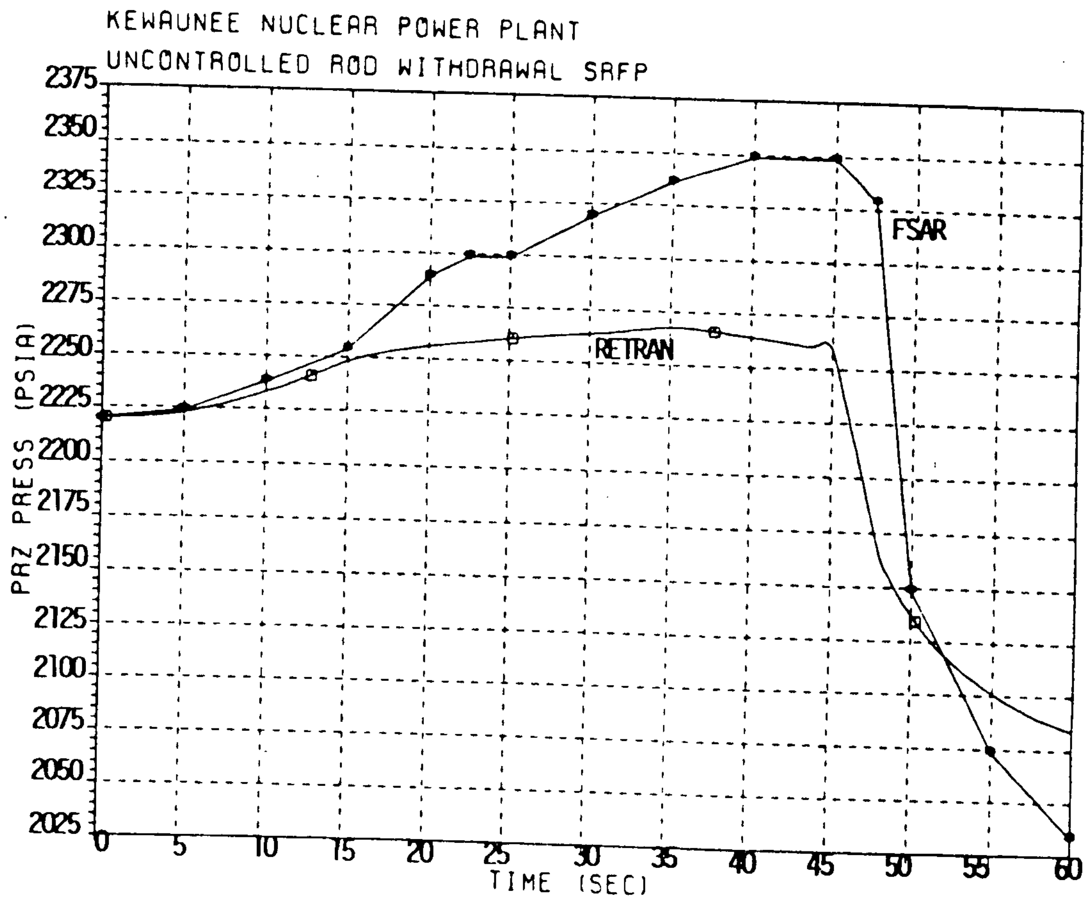

KEWAUNEE NUCLEAR POWER PLANT
UNCONTROLLED ROD WITHDRAWAL SRFP
ABS. POWER (FRAC.NOMINAL)
1.2
1.1
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0.0
FSAR
RETRAN
TIME (SEC)
0
5
10
15
20
25
30
35
40
45
50
55
60

FIGURE 3.2-6

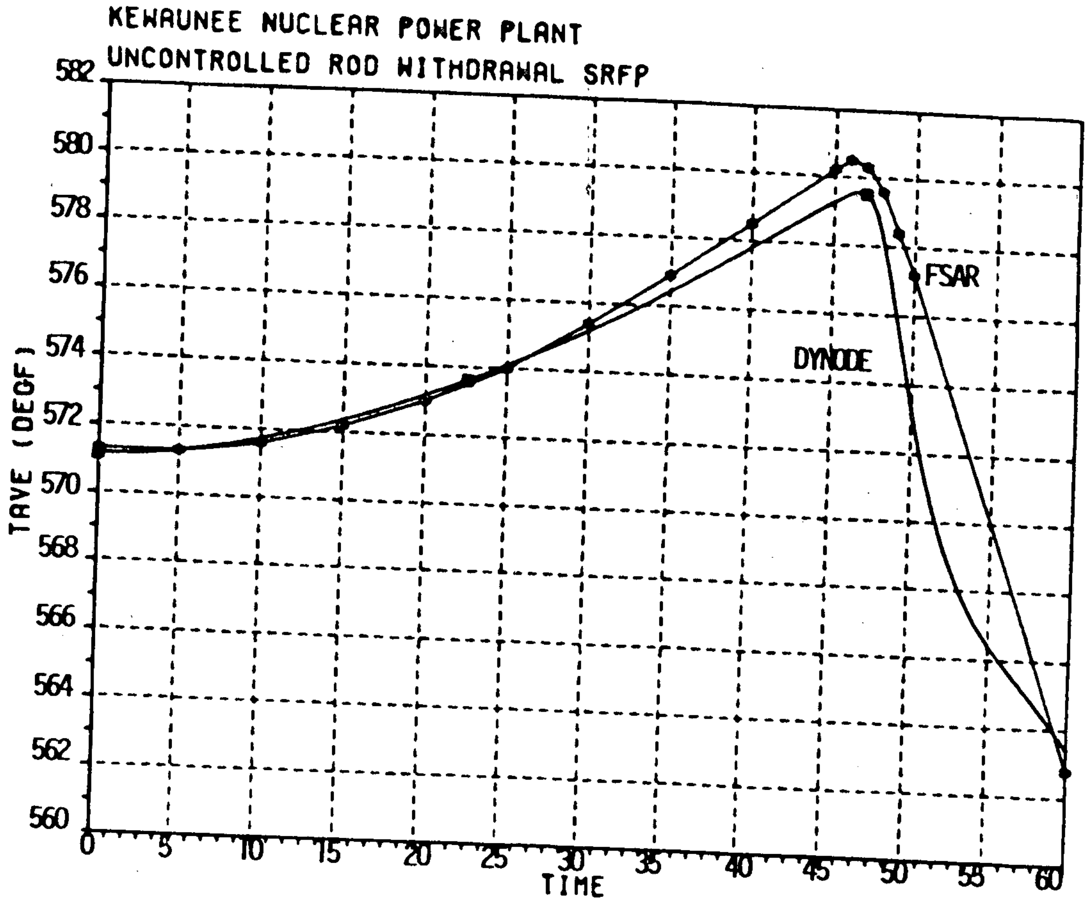


KEWAUNEE NUCLEAR POWER PLANT
UNCONTROLLED ROD WITHDRAWAL SRFP
FSAR
DYNODE
PRZ PRESS (PSIA)
TIME
2375
2350
2325
2300
2275
2250
2225
2200
2175
2150
2125
2100
2075
2050
2025
0
5
10
15
20
25
30
35
40
45
50
55
60

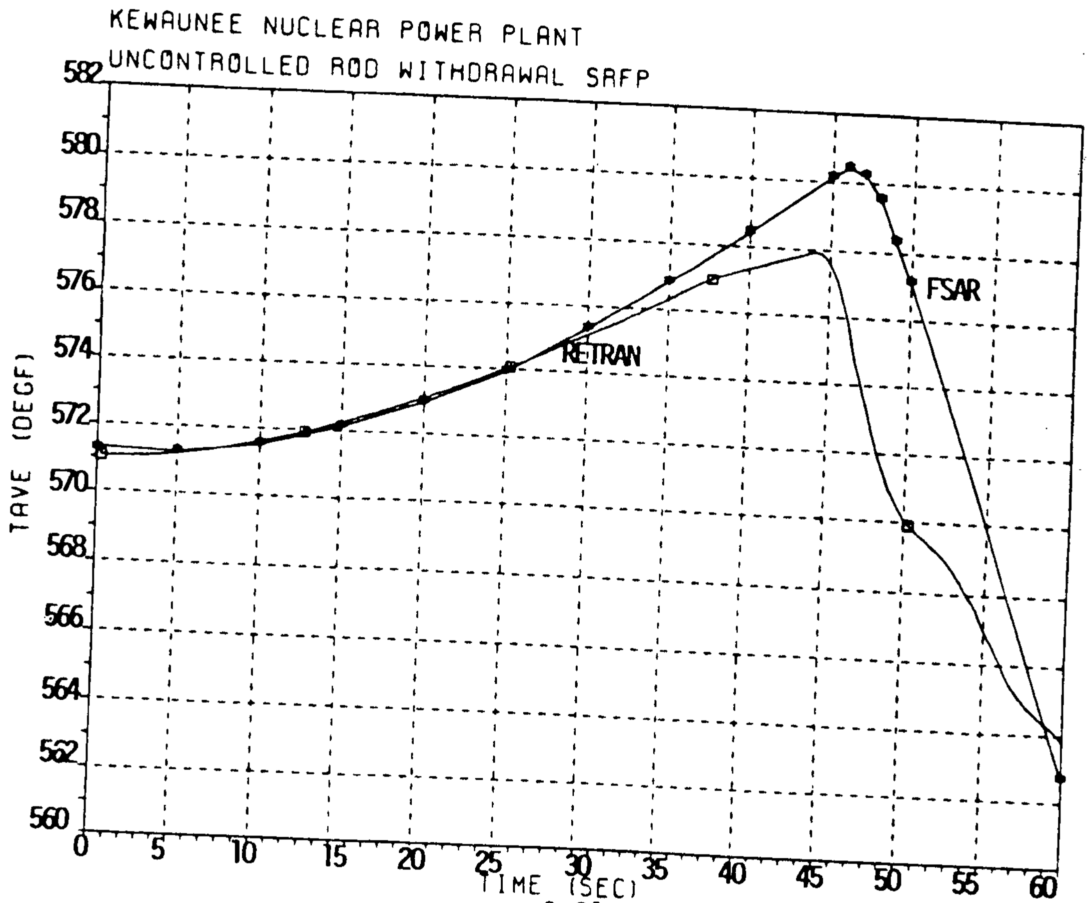


KEWAUNEE NUCLEAR POWER PLANT
UNCONTROLLED ROD WITHDRAWAL SRFP
FSAR
RETRAN
PRZ PRESS (PSIA)
TIME (SEC)
2375
2350
2325
2300
2275
2250
2225
2200
2175
2150
2125
2100
2075
2050
2025
0
5
10
15
20
25
30
35
40
45
50
55
60

FIGURE 3.2-7



KEWAUNEE NUCLEAR POWER PLANT
UNCONTROLLED ROD WITHDRAWAL SRFP
582
580
578
576
574
572
570
568
566
564
562
560
TAVE (DEGF)
0
5
10
15
20
25
30
35
40
45
50
55
60
TIME
FSAR
DYNODE



KEWAUNEE NUCLEAR POWER PLANT
UNCONTROLLED ROD WITHDRAWAL SRFP
582
580
578
576
574
572
570
568
566
564
562
560
TAVE (DEGF)
0
5
10
15
20
25
30
35
40
45
50
55
60
TIME (SEC)
FSAR
RETRAN

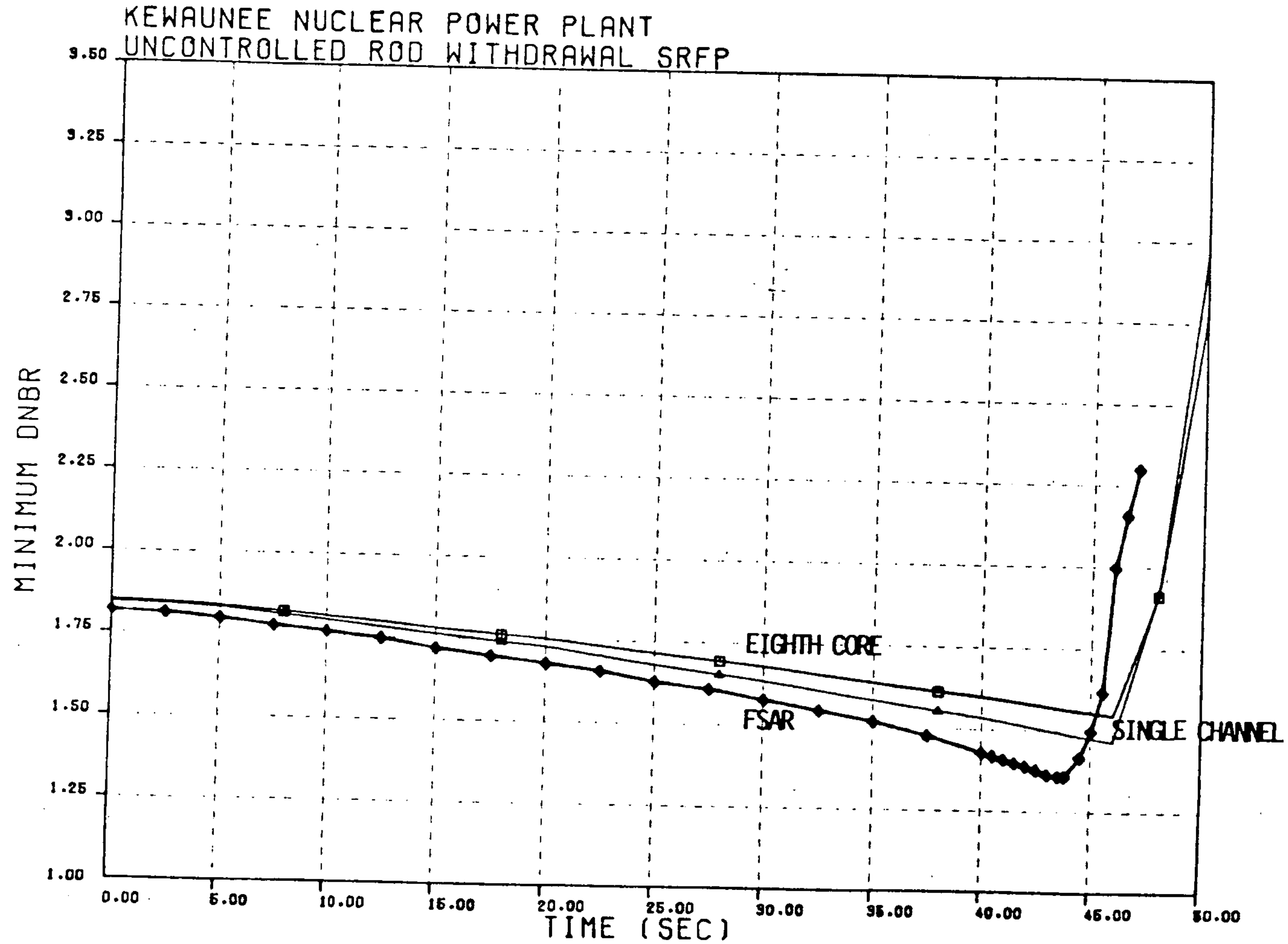

KEWAUNEE NUCLEAR POWER PLANT
UNCONTROLLED ROD WITHDRAWAL SRFP
MINIMUM DNBR
TIME (SEC)
EIGHTH CORE
FSAR
SINGLE CHANNEL
3.50
3.25
3.00
2.75
2.50
2.25
2.00
1.75
1.50
1.25
1.00
0.00
5.00
10.00
15.00
20.00
25.00
30.00
35.00
40.00
45.00
50.00

FIGURE 3.2-8

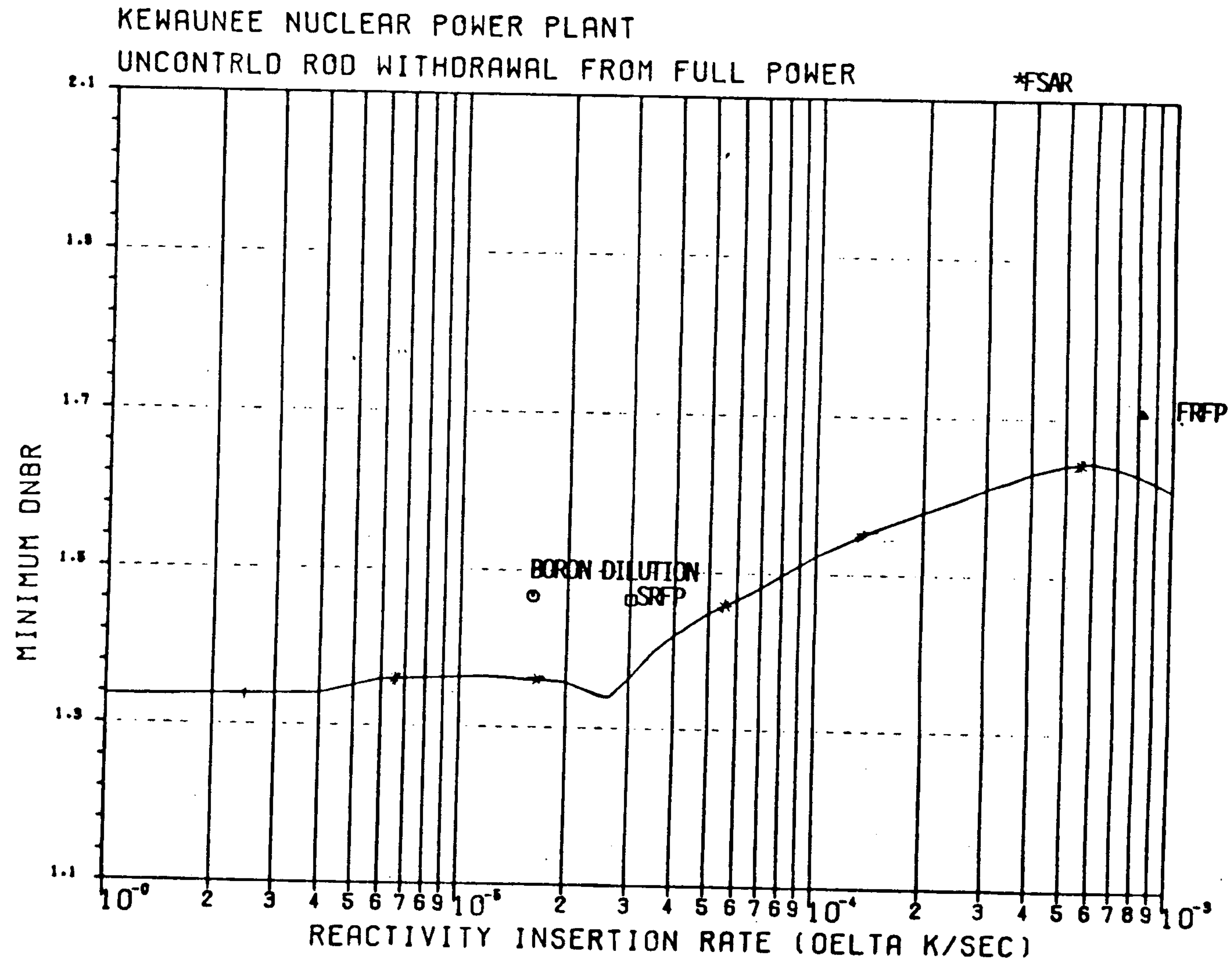

KEWAUNEE NUCLEAR POWER PLANT
UNCONTRLD ROD WITHDRAWAL FROM FULL POWER
*FSAR
FRFP
BORON DILUTION
SRFP
MINIMUM DNBR
2.1
1.7
1.1
REACTIVITY INSERTION RATE (DELTA K/SEC)

FIGURE 3.2-9

FIGURE 3.2-10

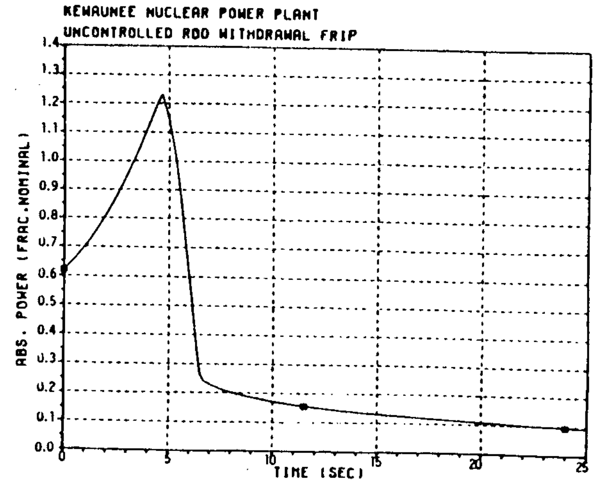


FIGURE 3.2-11

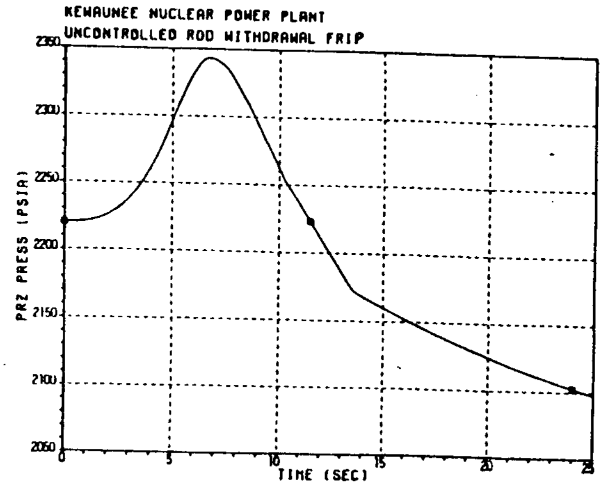


FIGURE 3.2-12

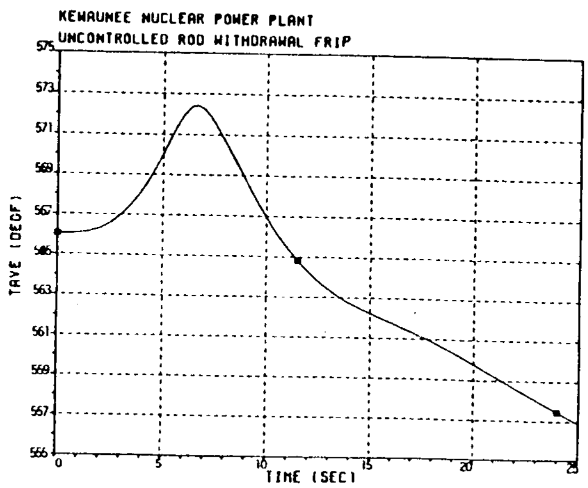


FIGURE 3.2-13

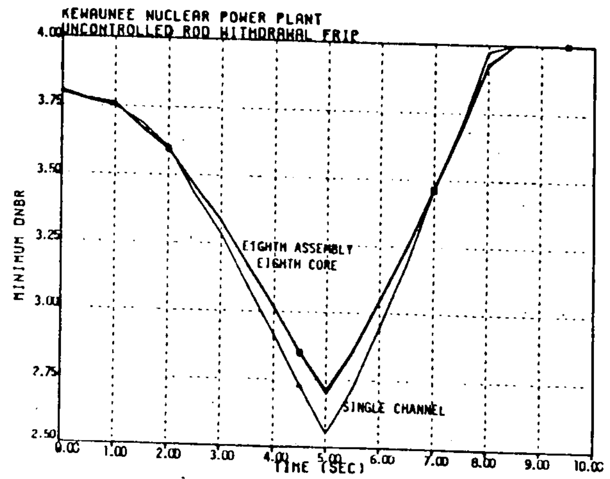


FIGURE 3.2-14

KEWAUNEE NUCLEAR POWER PLANT
UNCONTROLLED ROD WITHDRAWAL SRIP

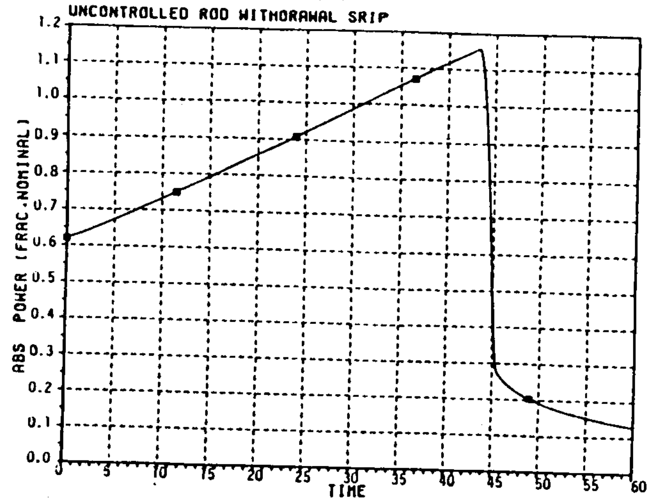


FIGURE 3.2-15

KEWAUNEE NUCLEAR POWER PLANT
UNCONTROLLED ROD WITHDRAWAL SRIP

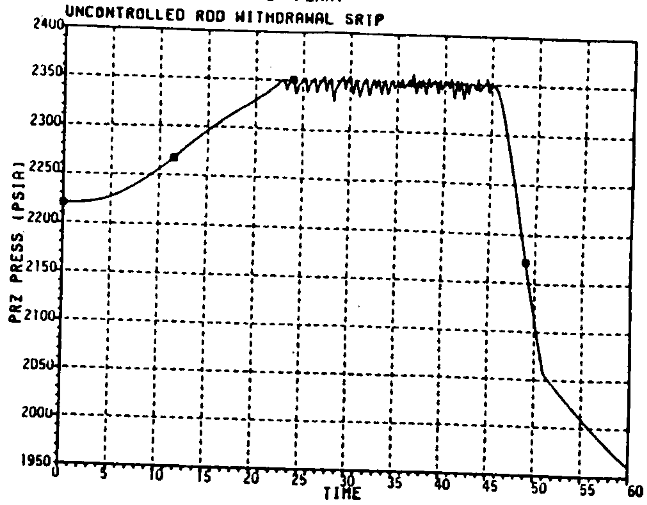


FIGURE 3.2-16

KEWAUNEE NUCLEAR POWER PLANT
UNCONTROLLED ROD WITHDRAWAL SRIP

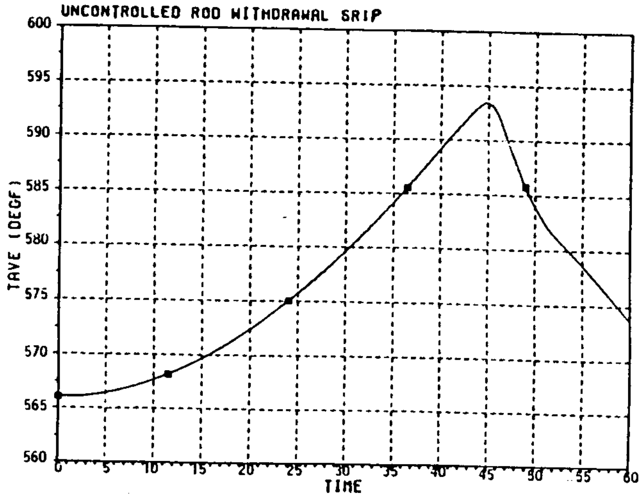
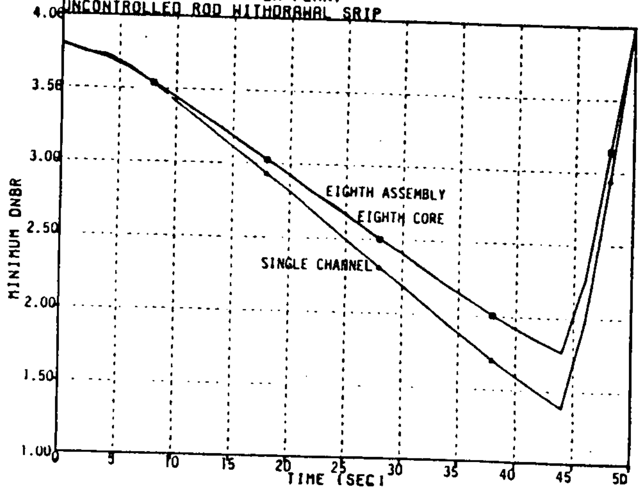


FIGURE 3.2-17

KEWAUNEE NUCLEAR POWER PLANT
UNCONTROLLED ROD WITHDRAWAL SRIP



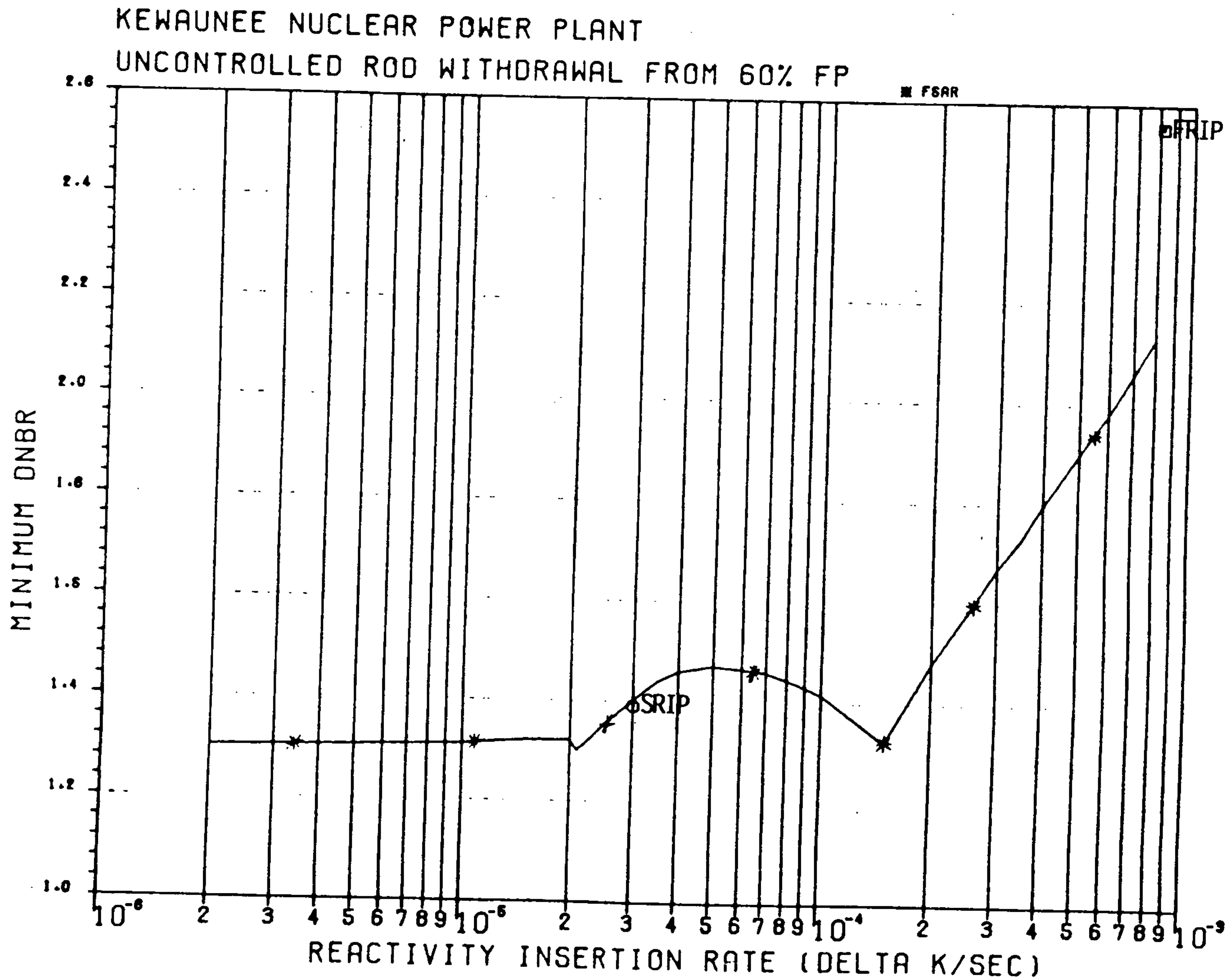

KEWAUNEE NUCLEAR POWER PLANT
UNCONTROLLED ROD WITHDRAWAL FROM 60% FP
FSAR
FRIP
SRIP
MINIMUM DNBR
2.6
2.4
2.2
2.0
1.8
1.6
1.4
1.2
1.0
10^{-6} 2 3 4 5 6 7 8 9 10^{-5} 2 3 4 5 6 7 8 9 10^{-4} 2 3 4 5 6 7 8 9 10^{-3}
REACTIVITY INSERTION RATE (DELTA K/SEC)

FIGURE 3.2-18

3.3 CONTROL ROD MISALIGNMENT

3.3.1 DEFINITION OF ACCIDENT

In the analysis of this accident, one or more rod cluster control assemblies is assumed to be statically misplaced from the normal or allowed position. This situation might occur if a rod were left behind when inserting or withdrawing banks, or if a single rod were to be withdrawn. Full power operation under these conditions could lead to a reduction in DNBR and is subject to limitations specified in the plant Technical Specifications.

3.3.2 ACCIDENT ANALYSIS

In the analysis of misaligned control rods, $F_{\Delta H}$ will be determined for the most limiting configuration. In general, the worst case is that with Bank D fully inserted except for a single withdrawn assembly, since Bank D is the only bank which may be inserted at full power. In practice, multiple independent alarms would alert the operator well before the postulated conditions are approached.

The limiting value of $F_{\Delta H}$ is input to a steady state thermal-hydraulic sub-channel calculation to determine the departure from nucleate boiling ratio (DNBR). This calculation assumes the most adverse combination of steady state errors applied to core neutron flux level, coolant pressure, and coolant temperature at the core inlet.

The acceptance criterion for this accident is that fuel rod failure is not permitted and this is insured by calculating the MDNBR using the W-3 correlation, and demonstrating that it is not less than 1.3.

3.3.3 WPS SAFETY ANALYSIS RESULTS

WPS has analyzed the control rod misalignment accident using input consistent with the Kewaunee USAR (4). Using the methods described in Appendix C, the control rod misalignment incident was analyzed using a hot channel factor ($F_{\Delta H}$) of 1.92. The MDNBR obtained was in agreement with the USAR result of "greater than 1.9".

3.3.4 CYCLE SPECIFIC PHYSICS CALCULATIONS

The nuclear enthalpy rise hot factor channel ($F_{\Delta H}$) is calculated for this accident consistent with the procedure described in Section 2. The maximum $F_{\Delta H}$ for a control rod misalignment at full power is calculated with Bank D fully inserted and one rod cluster of Bank D fully withdrawn. This is more conservative than the worst case that can occur since the control rod insertion limits restrict Bank D insertion to approximately midcore at full power. The rod misalignment calculations are performed for both BOC and EOC.

3.3.5 RELOAD SAFETY EVALUATIONS

The $F_{\Delta H}$ calculated above is conservatively adjusted to account for the model reliability factor, $RFF_{\Delta H}$. Additionally, a further conservatism is applied to account for the maximum initial quadrant tilt condition (T) allowed by the Technical Specifications. The resulting $F_{\Delta H}$ is then compared to the value used in the safety analysis as follows:

CYCLE SPECIFIC PARAMETER		SAFETY ANALYSIS PARAMETER
$F_{\Delta H}*(1+RFF_{\Delta H}) * (1+T)$	$\leq$	$F_{\Delta H}$ (Rod Misalignment)

3.4 CONTROL ROD DROP

3.4.1 DEFINITION OF ACCIDENT

In the analysis of this accident, a full-length RCCA is assumed to be released by the gripper coils and to fall into a fully inserted position in the core. The reactor is assumed to be operating in the manual mode of control.

A dropped rod cluster control assembly (RCCA) typically results in a reactor trip signal from the power range negative neutron flux rate circuitry. The core power distribution is not adversely affected during the short interval prior to reactor trip. The drop of a single RCCA assembly may or may not result in a reactor trip. If the plant is brought to full power with an assembly fully inserted, a reduction in core thermal margins may result because of a possible increased hot channel peaking factor.

3.4.2 ACCIDENT ANALYSIS

In the analysis of dropped RCCAs, $F_{\Delta H}$ will be determined for all possible dropped rod configurations and the most limiting configuration will be used in an analysis to determine the DNBR that would result if the core were returned to full power.

The limiting value of $F_{\Delta H}$ is input into a steady state thermal-hydraulic subchannel calculation which computes the DNBR using the W-3 correlation. The calculation is performed assuming full power with the most adverse combination of steady state errors applied to core neutron flux level, coolant pressure, and coolant temperature at the core inlet.

The acceptance criteria for the accident is that fuel rod failure is not permitted and this is insured by calculating MDNBR using the W-3 correlation and demonstrating that it is not less than 1.3.

3.4.3 WPS SAFETY ANALYSIS RESULTS

WPS has analyzed the control rod drop accident using input consistent with the Kewaunee USAR (4). Using the methods described in Appendix A, the control rod drop incident was analyzed using a hot channel factor ($F_{\Delta H}$) of 1.92. The MDNBR obtained was in agreement with the USAR result of "greater than 1.9".

3.4.4 CYCLE SPECIFIC PHYSICS CALCULATIONS

The nuclear enthalpy rise hot channel factor ($F_{\Delta H}$) is calculated for all possible dropped rods, consistent with the procedure described in Section 2. Each rod is dropped at full power, equilibrium xenon conditions and the rod yielding the largest $F_{\Delta H}$ is determined. This rod is then dropped into the core assuming various initial xenon and flux distributions to determine the maximum $F_{\Delta H}$ under dropped rod conditions. Additionally, peak $F_{\Delta H}$ occurring during the xenon transient following the dropped rod is calculated and compared to the initial dropped rod $F_{\Delta H}$. These calculations are performed at both BOC and EOC.

3.4.5 RELOAD SAFETY EVALUATION

The nuclear enthalpy rise factor $F_{\Delta H}$ calculated according to subsection 3.4.4 of this report is conservatively adjusted to account for calculational uncertainties.[1] It is further increased to account for the Technical Specification allowance for quadrant tilt (T).

Cycle Specific Parameter		Cycle Analysis Parameter
$F_{\Delta H} * (1+RFF_{\Delta H}) * (1+T)$	$\leq$	$F_{\Delta H}$ (Control Rod Drop)

3.5 UNCONTROLLED BORON DILUTION

3.5.1 DEFINITION OF ACCIDENT

The accident considered here is the malfunction of the chemical and volume control system in such a manner as to deliver unborated water at the maximum possible flowrate to the reactor coolant system under full power conditions. Dilution during refueling or startup is assumed to be recognized and terminated by operator intervention before loss of shutdown margin. With the reactor in automatic control, the power and temperature increase from boron dilution at power results in the insertion of the RCC assemblies and a decrease in shutdown margin. Rod insertion limit alarms would alert the operator to isolate the source of unborated water and initiate boration prior to the time that shutdown margin is lost. With the reactor in manual control, the power and temperature rise due to boron dilution would eventually result in an overtemperature ΔT reactor trip if the operator did not intervene. After such a trip, the operator would be expected to isolate the unborated water source and initiate boration procedures.

3.5.2 ACCIDENT ANALYSIS

The system transient response to an uncontrolled boron dilution is simulated using a detailed model of the plant which includes the core, reactor vessel, steam generators, pressurizer, and connecting piping. The model also includes a simulation of the charging and letdown systems, rod control system, pressurizer control system, and the reactor protection system. Reactivity effects due to fuel and moderator feedback coolant boron concentration, and control rod motion before and after trip are included in the analysis. This model provides the transient response of average core power, reactor coolant pressure, and coolant temperature at the core inlet which are applied as forcing functions to a thermal-hydraulic simulation of the hot channel. The hot channel model uses the W-3 correlation to calculate the departure from nucleate boiling ratio in the hot channel.

The reactivity due to boron dilution is calculated by assuming the maximum possible charging flow and minimum reactor coolant volume and taking into account the effect of increasing boron worth as dilution continues. The core burnup and corresponding boron concentration are selected to yield the most limiting combination of moderator temperature coefficient, Doppler temperature coefficient and spatial power distribution. This is normally the BOC condition. The minimum shutdown allowed by the technical specifications is conservatively assumed to exist prior to the initiation of the transient. The maximum time delay is assumed to exist between the time the trip setpoint is reached and the rods begin to move into the core. The most reactive rod is assumed to remain in its fully withdrawn position after receipt of the trip signal.

The acceptance criteria for this accident are that pressures in the reactor coolant system and main steam system do not exceed 110% of the respective design pressures, and that fuel clad integrity is maintained by limiting the MDNBR to greater than 1.3.

3.5.3 WPS SAFETY ANALYSIS RESULTS

WPS has analyzed a chemical and volume control system malfunction resulting in a decrease in the boron concentration of the reactor coolant. The analysis was performed using the models described in Appendix A with input consistent with the USAR (4). The results are compared to those presented in Section 14.1-4 of the USAR. Sensitivity studies indicate that the most critical parameters in the analysis of the boron dilution accident are the moderator temperature coefficient, the boron worth coefficient, and the parameters used in the overtemperature ΔT trip set point algorithm.

The NSSS and hot channel transient response calculated by the WPS model are shown in Figures 3.5-1 to 3.5-4. No corresponding transient results are given in the USAR. However, reactor trip on overtemperature ΔT was stated to occur at 78 seconds. The trip time calculated using the WPS model was 84 seconds, also on overtemperature ΔT.

For the charging flow rate used in both analyses (180 gallons/minute), the USAR quotes a reactivity insertion rate of 1.6×10^{-5} Δk/sec. The WPS model calculated an insertion rate of 1.4×10^{-5} Δk/sec at this charging flow. The slower reactivity insertion rate is the cause of the later reactor trip time.

From Figure 14.1-10 of the USAR, the minimum DNBR corresponding to this rate of reactivity insertion is 1.37. The WPS model predicts a MDNBR of 1.47.

3.5.4 CYCLE SPECIFIC PHYSICS CALCULATIONS

a. DOPPLER TEMPERATURE COEFFICIENT, α_D

Calculations of α_D are performed in accordance with the procedure described in Section 2. Cycle specific calculations are made as a function of power at BOC and EOC.

b. MODERATOR TEMPERATURE COEFFICIENT, α_M

Calculations of α_M are performed using the methods described in Section 2. Cycle specific calculations are made at unrodded, full power, and zero power conditions. The model bias, B_M is included in the calculations.

c. BORON REACTIVITY INSERTION RATE, $\Delta\rho B/\Delta t$

Calculations of α_B, the boron reactivity coefficient, are performed using the methods described in Section 2. Cycle specific calculations for these accidents are threefold: full power, all rods out; zero power, all rods in, less one stuck rod; and zero power, all rods in. These are performed at both BOC and EOC. The most negative reactivity coefficient is multiplied by the maximum rate to yield the boron reactivity insertion rate. The reactivity insertion rate is divided by the minimum cycle βeff determined according to (f) of this subsection to yield the maximum reactivity insertion rate in dollars.

d. SHUTDOWN MARGIN, SDM

For refueling and startup modes (cold), the shutdown margin is calculated directly with all rods in rather than with one stuck rod, consistent with the assumptions made in the safety analysis.

e. NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR, $F_{\Delta H}$

Calculations of $F_{\Delta H}$ are performed in accordance with the general procedures described in Section 2. Maximum core $F_{\Delta H}$s are verified to remain within Technical Specification limits for allowable combinations of axial offset, power level, and control rod insertion. For Kewaunee, the continuous surveillance of the power distribution is accomplished with the ex-core detectors using a Power Distribution Control scheme, PDC-II (12). The cycle specific physics calculations performed for the verification of the PDC-II scheme with respect to the $F_{\Delta H}$ limits are described in Section 3.17.

f. EFFECTIVE DELAYED NEUTRON FRACTION, β_{eff}

Calculations of β_{eff} are performed in accordance with the general procedures described in Section 2. Cycle specific calculations are performed at BOC and EOC.

3.5.5 RELOAD SAFETY EVALUATION

All the cycle specific parameters discussed above are adjusted to include model reliability factors RF_i[1] and these results are then compared to the bounding values assumed in the safety analysis. The cycle specific parameters are acceptable if the following inequalities are met:

CYCLE SPECIFIC PARAMETERS		SAFETY ANALYSIS PARAMETERS
a. Refueling and Cold Startup Conditions		
SDM (ARI)	$\geq$	SDM (bounding)

b. At Power Conditions

SDM	$\geq$	SDM (bounding)
$\left\lvert \frac{\Delta\rho B/\Delta t\ (1+RFB)}{\beta_{eff}\ (1-RF\beta)} \right\rvert$	$\leq$	$\left\lvert \frac{\Delta\rho B/\Delta t\ (bounding)}{\beta_{eff}\ (maximum)} \right\rvert$
$\alpha_M+RF_M+B_M$	$\leq$	α_M (least negative bounding value)
$\alpha_D*(1-RF_D)$	$\leq$	α_D (least negative bounding value)
$F\Delta H*(1+RF_{F\Delta H})*(1+T)$	$\leq$	Technical Specifications (Refer to Section 3.17)
β_{eff} * $(1-RF_\beta)$	$\geq$	β_{eff} (minimum)

FIGURE 3.5-1

KEWAUNEE NUCLEAR POWER PLANT
BORON DILUTION

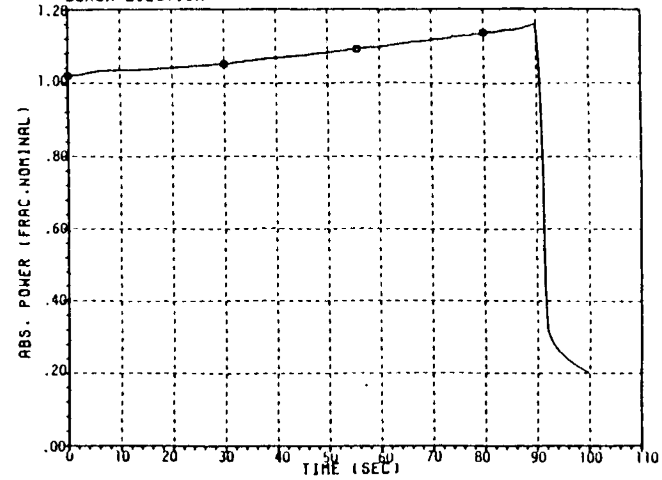


FIGURE 3.5-2

KEWAUNEE NUCLEAR POWER PLANT
BORON DILUTION

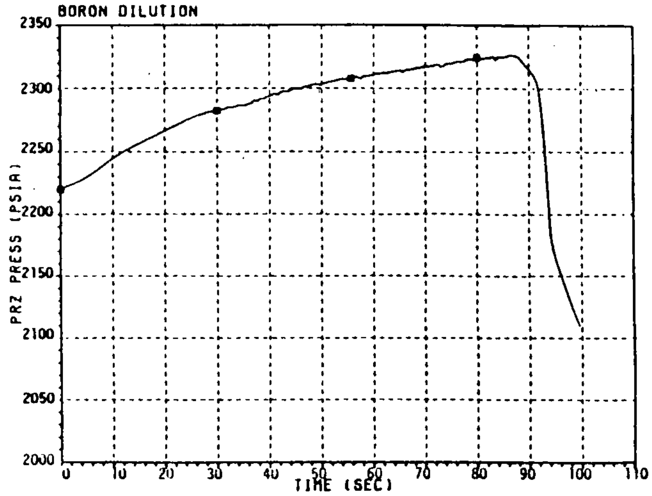


FIGURE 3.5-3

KEWAUNEE NUCLEAR POWER PLANT
BORON DILUTION

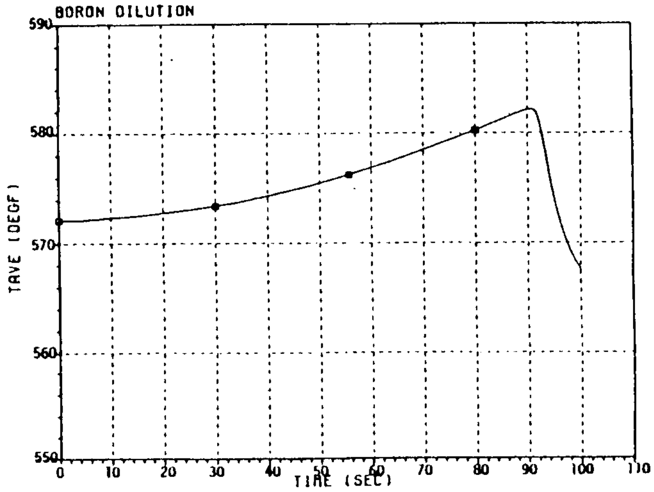
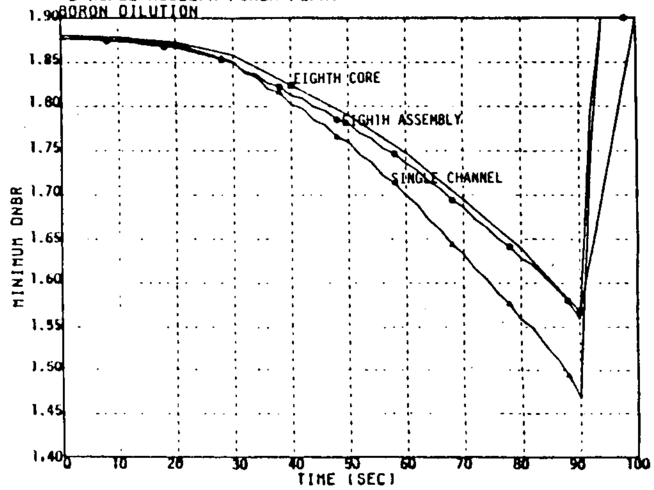


FIGURE 3.5-4

KEWAUNEE NUCLEAR POWER PLANT
BORON DILUTION



3.6 STARTUP OF AN INACTIVE COOLANT LOOP

3.6.1 DEFINITION OF ACCIDENT

Since there are no isolation valves or check valves in the Kewaunee reactor coolant system, operation of the plant with an inactive loop causes reverse flow through that loop. If there is a thermal load on the steam generator in the inactive loop, the hot leg coolant in that loop will be at a lower temperature than the core inlet temperature. The startup of the pump in the idle loop results in a core flow increase and the injection of cold water into the core, followed by a rapid reactivity and power increase. The resulting increase in fuel temperature limits the power rise due to Doppler feedback. Above 10% rated power, however, the reactor protection system prevents operation with an inactive loop, and consequently the temperature differential in an inactive loop would be small enough to minimize the accident consequences. Furthermore, the Kewaunee Technical Specifications do not permit operation with a reactor coolant pump out of service except during low power physics testing.

3.6.2 ACCIDENT ANALYSIS

The system transient response to an inactive loop startup is simulated using a detailed model which includes the core, reactor vessel, steam generators, main steam and reactor coolant piping, and the plant control and protection systems. This model calculates the time-dependent behavior of the average core power, coolant pressure, and core inlet flow and temperature which are supplied as forcing functions to a model of the hot channel for calculation of DNBR.

The accident is analyzed using the most negative moderator temperature coefficient and the least negative Doppler coefficient calculated to occur during the cycle. No credit is taken for reactivity reduction caused by reactor trip.

The reactor is initially assumed to be operating at 12% of rated power with reverse flow through the inactive loop. This includes a 2% uncertainty for calibration error above the 10% power setpoint in the protection system for single loop operation. The assumption of this high initial power level is conservative since it maximizes the temperature difference between the hot leg and cold leg in the inactive loop. The most adverse combination of initial coolant pressure and core inlet temperature is chosen to minimize the margin to core DNB limits.

The acceptance criteria for this accident are that the maximum pressures in the reactor coolant and main steam systems do not exceed 110% of design values and that cladding integrity be maintained limiting the minimum DNB ratio greater than 1.30.

3.6.3 WPS SAFETY ANALYSIS RESULTS

WPS has analyzed the inactive loop startup accident using the models and methods described in Appendix A. The results obtained are compared to the results presented in Section 14.1-5 of the USAR (4). Sensitivity studies have confirmed that the value of the moderator temperature coefficient exerts a controlling influence on the calculated accident consequences. Increasing the absolute magnitude of the negative moderator coefficient by 30% increases the maximum neutron power by 11% referenced to the peak.

The USAR (4) states that the flow was linearly ramped to the nominal value in 10 seconds. The flow response of the WPS model exhibits similar behavior. However, a four second delay due to pump and fluid inertia is observed before the fluid achieves significant acceleration. Nominal flow and pump speed in the startup loop is reached by about 14 seconds in the WPS model.

Figures 3.6-1 to 3.6-5 provide a comparison of NSSS transient to Figures 14.1-14 and 14.1-15 of the USAR. The results of the WPS model compare well with those of the USAR.

3.6.4 CYCLE SPECIFIC PHYSICS CALCULATIONS

a. DOPPLER TEMPERATURE COEFFICIENT, α_D

Calculations of α_D are performed in accordance with the general procedures described in Section 2. Specific evaluation of α_D for this accident is made assuming 12% power for both BOC and EOC.

b. MODERATOR TEMPERATURE COEFFICIENT, α_M

Calculations of α_M are performed in accordance with the general procedures described in Section 2. Specific calculations for this accident are performed for hot zero power, rodded, no xenon conditions at both BOC and EOC. The model bias, B_M, is included in the calculations.

c. NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR, $F_{\Delta H}$

Calculations of $F_{\Delta H}$ are performed in accordance with the general procedures described in Section 2. Maximum core $F_{\Delta H}$s are verified to remain within Technical Specification limits for allowable combinations of axial offset, power level, and control rod insertion. For Kewaunee, the continuous surveillance of the power distribution is accomplished with the ex-core detectors using a Power Distribution Control scheme, PDC-II (12). The cycle specific physics calculations performed for the verification of the PDC-II scheme with respect to the $F_{\Delta H}$ limits are described in Section 3.17.

3.6.5 RELOAD SAFETY EVALUATIONS

Each of the physics parameters calculated above is conservatively adjusted to include the model reliability factors RF_i[1]. These adjusted values are the cycle specific parameters which are then compared to the bounding values assumed in the safety analysis. The cycle specific parameters are acceptable if the following inequalities are met:

CYCLE SPECIFIC PARAMETERS		SAFETY ANALYSIS PARAMETERS
a. $\alpha_M - RF_M + B_M$	$\geq$	α_M (most negative bounding value)
b. $\alpha_D*(1-RF_D)$	$\leq$	α_D (least negative bounding value)
c. $F\Delta H*(1+RFF_{\Delta H})*(1+T)$	$\leq$	Technical Specifications (Refer to Section 3.17)

FIGURE 3.6-1

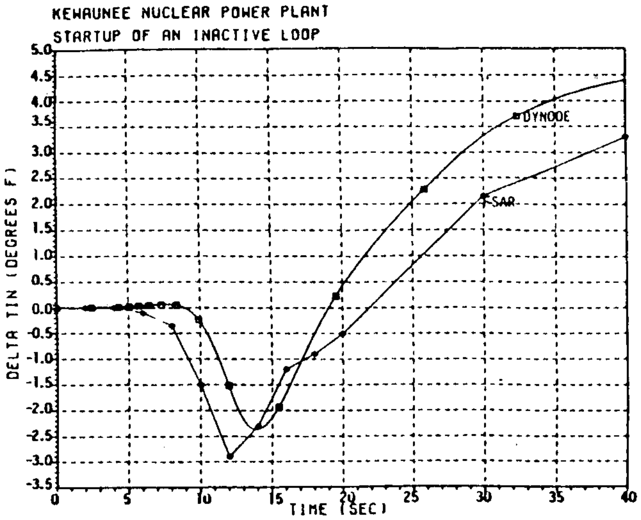


FIGURE 3.6-2

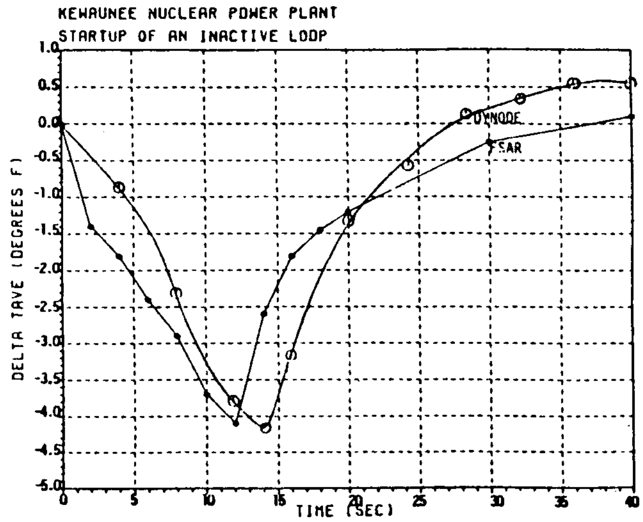


FIGURE 3.6-3

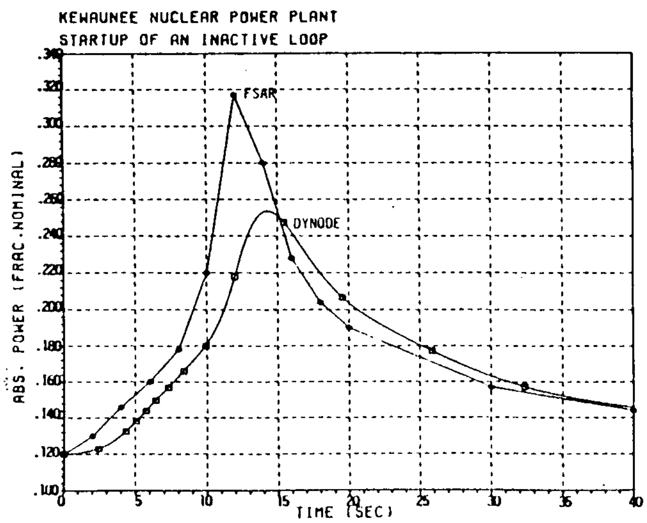


FIGURE 3.6-4

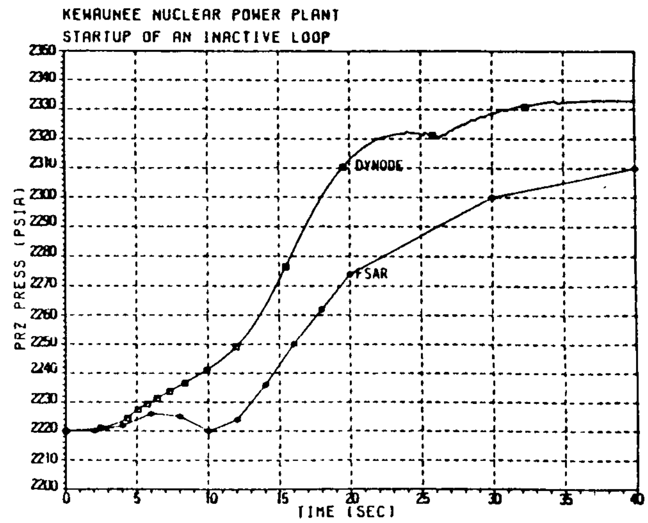
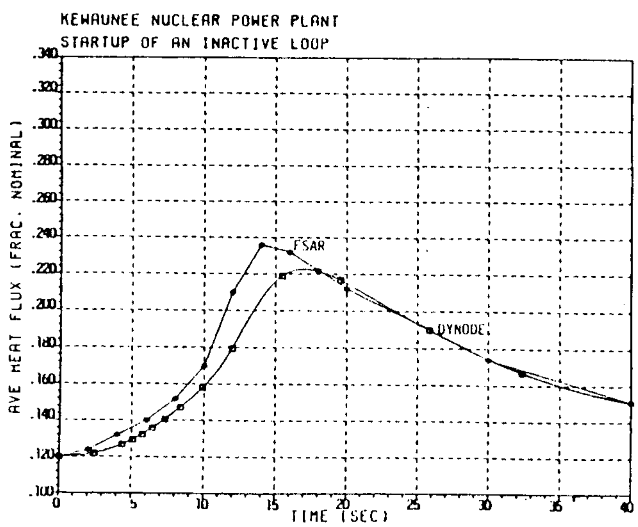


FIGURE 3.6-5



3.7 FEEDWATER SYSTEM MALFUNCTION

3.7.1 DEFINITION OF ACCIDENT

Two classes of accidents are to be considered under this classification: Those that result in a decrease in feedwater temperature and those that result in an increase in feedwater flow. Either condition will result in an increased heat transfer rate in the steam generators, causing a decrease in the reactor coolant temperature and an increased core power level due to negative reactivity coefficients and/or control system action. For the case of the decrease in feedwater temperature, the worst accident which may be postulated involves opening the bypass valve which diverts flow around the feedwater heaters. For the case of an increase in feedwater flow rate, the worst accident which may be postulated involves the full opening of a feedwater control valve. For this case, sustained high feedwater flow rate would ultimately result in a reactor trip due to high steam generator water level.

3.7.2 ACCIDENT ANALYSIS

The feedwater system malfunction transient is analyzed using a dynamic simulation which includes core kinetics and heat transfer, reactor vessel and coolant piping, steam generators, pressurizer, and control systems. Pertinent variables obtained from the NSSS simulation are then applied as forcing functions to a separate thermal-hydraulic model of the hot channel which calculates DNBR.

Two cases are analyzed. The first case is for a reactor without automatic control and with a zero moderator temperature coefficient. This represents the situation where the reactor has the least inherent transient response capability. In this case, the core power slowly increases due to Doppler and moderator reactivity effects until the core power level again matches the load demand and a new steady state is achieved. The reactor does not trip. The coolant temperature decreases which has the effect of increasing the margin to DNB. This increase in DNBR is larger than the decrease caused by the higher heat flux and the net effect is that MDNBR increases during the transient.

The second case analyzed assumes that the reactor automatic control system responds to the decreasing coolant temperature and matches reactor power to load demand. A conservatively large (in absolute value) negative moderator temperature coefficient is assumed to exist. The value chosen is more negative than that calculated to actually occur at EOC. This case results in a somewhat higher final core power level than the uncontrolled case without moderator feedback; this in turn results in a net decrease in DNBR but the decreased coolant temperature again maintains a significant margin above the 1.3 limit.

The core neutronic characteristics which exert a significant influence on the calculated results of this transient are the Doppler and moderator reactivity coefficients. The most negative moderator temperature coefficient calculated to occur during the cycle is used in the analysis to maximize the power increase. For such slow rates of reactivity addition as are encountered, the transient response is insensitive to the value of ℓ^*, the prompt neutron lifetime. Trip reactivity insertion characteristics are not relevant, since the reactor does not trip.

The acceptance criteria for the feedwater system malfunction transient are that cladding integrity be maintained by limiting the minimum DNBR to be greater than 1.3 and that maximum pressure in the reactor coolant and main steam system not exceed 110% of the design pressure.

3.7.3 WPS SAFETY ANALYSIS RESULTS

Not all classes of the feedwater system malfunction transient are analyzed here. These types of malfunctions are represented by the analysis of a decrease in feedwater temperature transient using the models described in Appendix A. This calculation has been performed using input consistent with the Kewaunee USAR (4). Specifically, the transient analyzed was the opening of the feedwater heater by-pass valve.

The model used corresponds to BOC conditions without control. The feedwater enthalpy transient forcing function was derived from the USAR results.

The response of the NSSS is compared to Figures 14.1-16 and 14.1-17 of the USAR in Figures 3.7-1 to 3.7-4. The WPS models predict the same trends throughout the transient as the USAR results. Mass and energy balances have been performed which substantiate the validity of the WPS model. Hot channel MDNBR analyses were not compared since the MDNBR increases during this transient.

3.7.4 CYCLE SPECIFIC PHYSICS CALCULATIONS

a. DOPPLER TEMPERATURE COEFFICIENT, α_D

Calculations of α_D are performed in accordance with the general procedures described in Section 2. Cycle specific calculations for this accident are performed as a function of power level over the full operating range from 0 - 100% power at BOC and EOC.

b. MODERATOR TEMPERATURE COEFFICIENT, α_M

Calculations of α_M are performed in accordance with the general procedures described in Section 2. Cycle specific calculations are performed at BOC and EOC to determine the least negative α_M at full power conditions and the most negative α_M under all operating conditions. The model bias is included in these calculations.

c. NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR, $F_{\Delta H}$

Calculations of $F_{\Delta H}$ are performed in accordance with the general procedures described in Section 2. Maximum core $F_{\Delta H}$s are verified to remain within Technical Specification limits for allowable combinations of axial offset, power level, and control rod insertion. For Kewaunee, the continuous surveillance of the power distribution is accomplished with the ex-core detectors using a Power Distribution Control scheme, PDC-II (12). The cycle specific physics calculations performed for the verification of the PDC-II scheme with respect to the $F_{\Delta H}$ limits are described in Section 3.17.

3.7.5 RELOAD SAFETY EVALUATIONS

Each of the physics parameters calculated above are adjusted to include the model reliability factors RF_i and biases (1). These adjusted values are the cycle specific parameters which are then compared to the bounding values assumed in the safety analysis. The cycle specific parameters are acceptable with regard to feed water malfunction transients if the following inequalities are met:

CYCLE SPECIFIC PARAMETERS		SAFETY ANALYSIS PARAMETERS
BOC a. $\alpha_D*(1-RF_D)$	$\leq$	α_D (least negative bounding value) if $\alpha M < 0$
$\alpha_D * (1+RF_D)$	$\geq$	α_D (most negative bounding value) if $\alpha M \geq 0$
b. $\alpha_M + RF_M + B_M$	$\leq$	α_M (least negative bounding value)
EOC a. $\alpha_M - RF_M + B_M$	$\geq$	α_M (most negative bounding value)
b. $\alpha_D * (1-RF_D)$	$\leq$	α_D (least negative bounding value)
c. $F_{\Delta H}*(1+RFF_{\Delta H})*(1+T)$	$\leq$	Technical Specifications (Refer to Section 3.17)

FIGURE 3.7-1

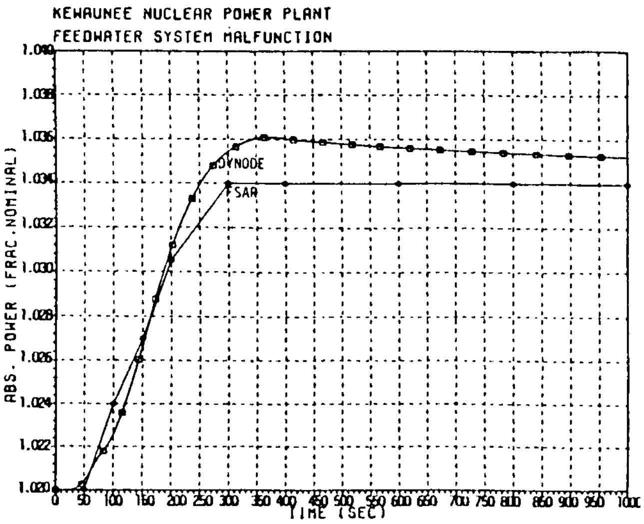


FIGURE 3.7-2

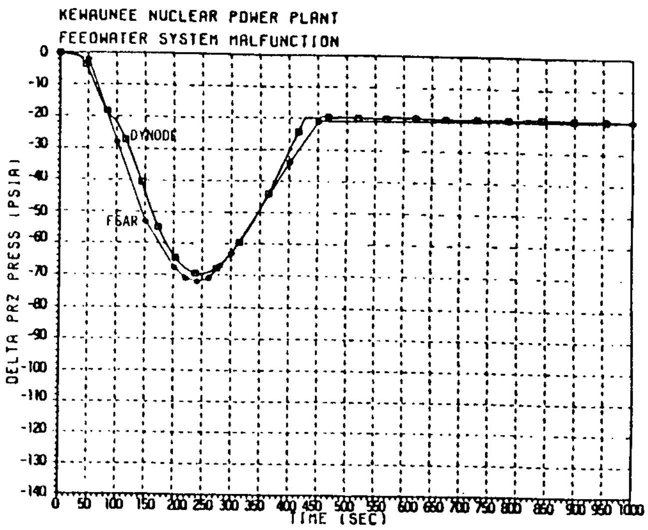


FIGURE 3.7-3

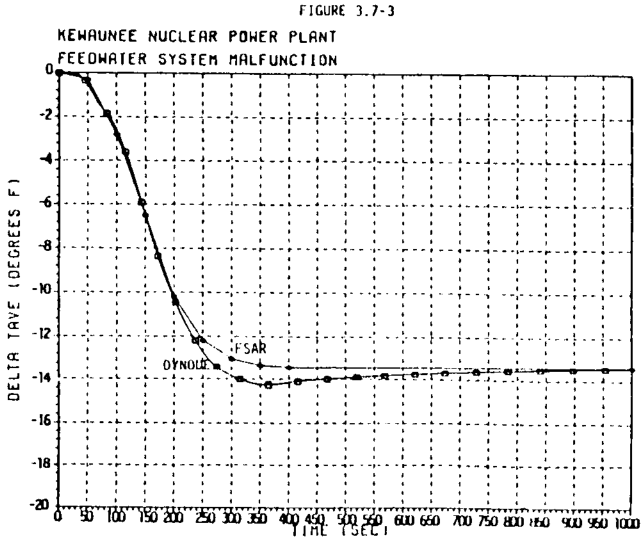
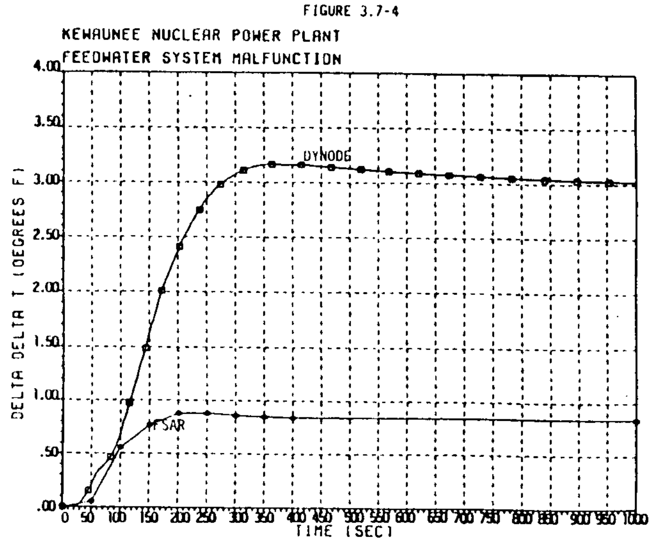


FIGURE 3.7-4



3.8 EXCESSIVE LOAD INCREASE

3.8.1 DEFINITION OF ACCIDENT

An excessive load increase accident is defined as a rapid increase in steam generator steam flow that causes a power mismatch between core heat generation and secondary side load demand. The ensuing decrease in reactor coolant temperature results in a core power increase due to fuel and moderator feedback and/or control system action. Only steam flow increases within the capability of the turbine control valves are considered here; larger flow increases are considered in connection with main steam line break accidents (Section 3.14).

3.8.2 ACCIDENT ANALYSIS

The excessive load increase transient is analyzed using a dynamic simulation which includes the reactor core, reactor vessel, steam generators, pressurizer and connecting piping. The main steam and feedwater systems and control and protection systems are also modeled. The departure from nucleate boiling ratio is computed using a separate model of the hot channel thermal-hydraulic behavior and the W-3 correlation. This model is coupled to the NSSS simulation which supplies core power and coolant temperature and pressure as a function of time.

The transient is initiated by imposing a rapid increase in steam flow to 120% of rated full power flow. Initial pressurizer pressure, reactor coolant temperature, and core power are assumed at their extreme steady-state values to minimize the calculated margin to DNB. Typically, four cases are analyzed: moderator reactivity coefficient at minimum and maximum values; with and without automatic reactor control.

For the cases without control, the case with the least negative moderator coefficient shows a large coolant temperature decrease relative to the power increase and the net effect is to increase the DNBR. The case with the more negative moderator coefficient shows a larger increase in power and a decrease in DNBR. The cases with reactor control show similar behavior but the control system acts to maintain average coolant temperature by increasing reactor power, so the DNBR decreases in both cases. However, all cases presented in the Kewaunee USAR (4) exhibit a large margin to the 1.3 DNBR limit.

Reactor trip does not occur during any of the transients considered, consequently scram reactivity insertion characteristics are not factors in the evaluation of this accident. Moderator and Doppler reactivity coefficients are the most significant kinetics parameters. The most negative Doppler and moderator temperature coefficients are assumed to provide the most conservative evaluation since they maximize the core power increase. The acceptance criteria for this accident are that the fuel cladding integrity be maintained by limiting the minimum DNBR to be greater than 1.3 and reactor coolant and main steam system maximum pressures not be greater than 110% of the design pressures.

3.8.3 WPS SAFETY ANALYSES RESULTS

The Excessive Load Increase transient is analyzed for the BOC-No Control and EOC-Auto Control conditions using the models described in Appendix A. These two cases represent the extremes for this event. The calculations have been performed using input consistent with the Kewaunee USAR (4).

The response of the NSSS and the hot channel are compared to Figures 14.1-19 through 14.1-24 of the USAR in Figures 3.8-1 to 3.8-10 of this report.

The WPS model predicts a more severe temperature and pressure response for the BOC-No Control case causing MDNBRs to be slightly less than the USAR results. Both models show MDNBR increasing with time.

The parameter trends in the EOC-Auto Control case show good agreement. The WPS model, however, predicts a more conservative MDNBR response due to a slightly more severe reactor power transient illustrated in Figure 3.8-7.

DNBR analyses were computed using a single channel model and multi-channel 1/8 core and 1/8 assembly models. The single channel model is shown to be the most conservative when MDNBR is decreasing.

3.8.4 CYCLE SPECIFIC PHYSICS CALCULATIONS

a. DOPPLER TEMPERATURE COEFFICIENT, α_D

Calculations of α_D are performed in accordance with the general procedures described in Section 2. Cycle specific calculations for this accident are performed at the full power equilibrium xenon conditions at BOC and EOC.

b. MODERATOR TEMPERATURE COEFFICIENT, α_M

Calculations of α_M are performed in accordance the with general procedures described in Section 2. Cycle specific calculations are performed at BOC and EOC to determine the maximum and minimum values of the moderator coefficient at full power equilibrium xenon conditions.

c. NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR, $F_{\Delta H}$

Calculations of $F_{\Delta H}$ are performed in accordance with the general procedures described in Section 2. Maximum core $F_{\Delta H}$s are verified to remain within Technical Specification limits for allowable combinations of axial offset, power level, and control rod insertion. For Kewaunee, the continuous surveillance of the power distribution is accomplished with the ex-core detectors using a Power Distribution Control scheme, PDC-II (12). The cycle specific physics calculations performed for the verification of the PDC-II scheme with respect to the $F_{\Delta H}$ limits are described in Section 3.17.

3.8.5 RELOAD SAFETY EVALUATIONS

Each of the physics parameters calculated above are adjusted to include the model reliability factors RF_i and biases (1). These adjusted values are the cycle specific parameters which are then compared to the bounding bounding values assumed in the safety analysis. The cycle specific parameters are acceptable with regard to excessive load increase transients if the following inequalities are met:

	CYCLE SPECIFIC PARAMETERS		SAFETY ANALYSIS PARAMETERS
BOC	a. $\alpha_D*(1-RF_D)$	$\leq$	α_D (least negative bounding value) if $\alpha M < 0$
	$\alpha_D * (1+RF_D)$	$\geq$	α_D (most negative bounding value) if $\alpha M \geq 0$
	b. $\alpha_M + RF_M + B_M$	$\leq$	α_M (least negative bounding value)
EOC	a. $\alpha_M - RF_M + B_M$	$\geq$	α_M (most negative bounding value)
	b. $\alpha_D * (1-RF_D)$	$\leq$	α_D (least negative bounding value)
	c. $F_{\Delta H}*(1+RFF_{\Delta H})*(1+T)$	$\leq$	Technical Specifications (Refer to Section 3.17)

FIGURE 3.8-1

KEWAUNEE NUCLEAR POWER PLANT
EXCESSIVE LOAD INCREASE BOL NO CONTROL

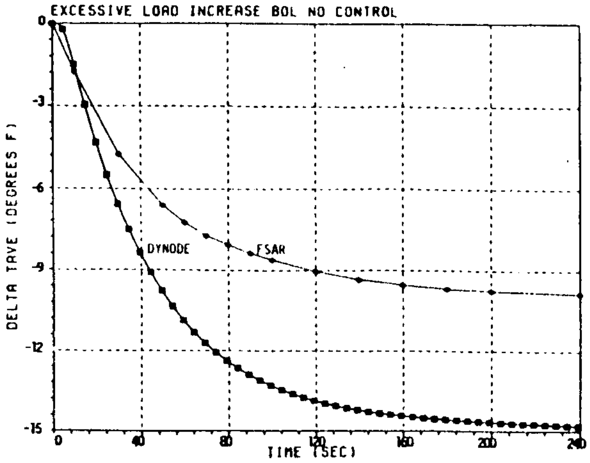


FIGURE 3.8-2

KEWAUNEE NUCLEAR POWER PLANT
EXCESSIVE LOAD INCREASE BOL NO CONTROL

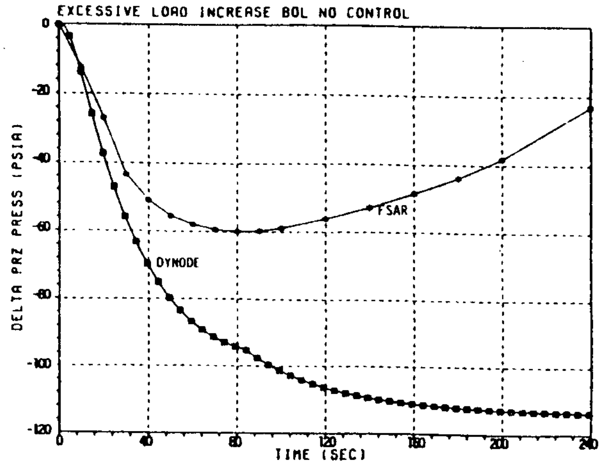


FIGURE 3.8-3

KEWAUNEE NUCLEAR POWER PLANT
EXCESSIVE LOAD INCREASE BOL NO CONTROL

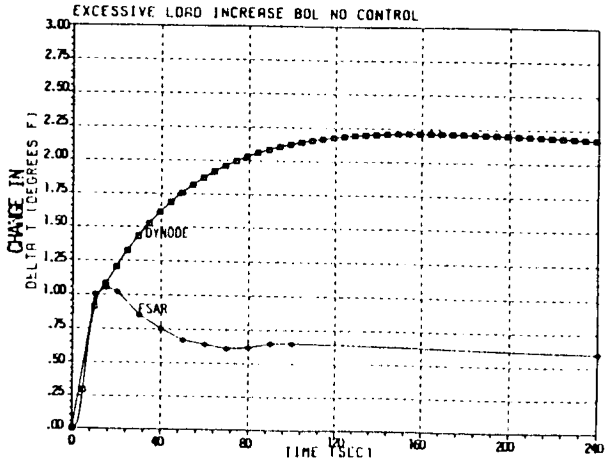


FIGURE 3.8-4

KEWAUNEE NUCLEAR POWER PLANT
EXCESSIVE LOAD INCREASE BOL NO CONTROL

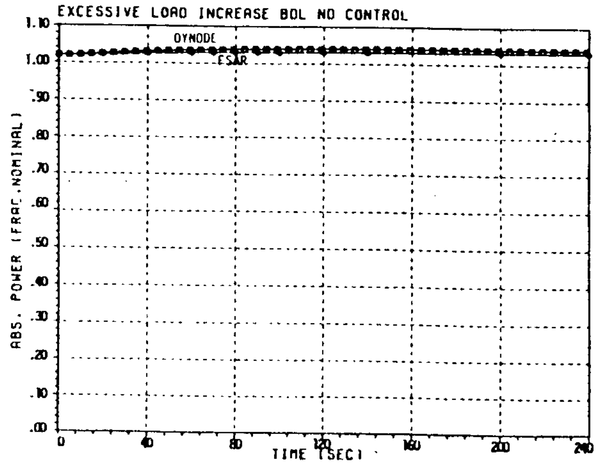


FIGURE 3.8-5

KEWAUNEE NUCLEAR POWER PLANT
EXCESSIVE LOAD INCREASE BOL NO CONTROL

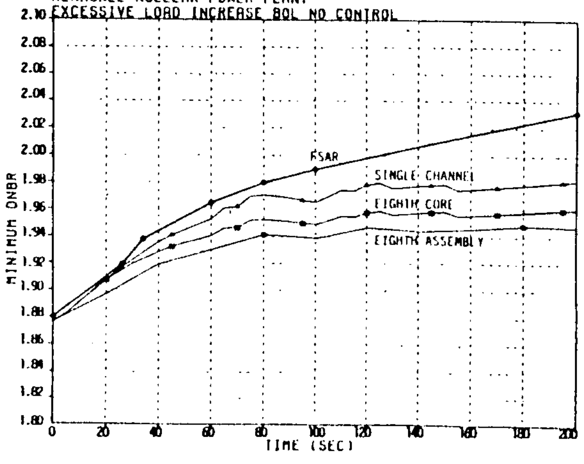


FIGURE 3.8-6

KEWAUNEE NUCLEAR POWER PLANT

EXCESSIVE LOAD INCREASE EOL AUTO CONTROL

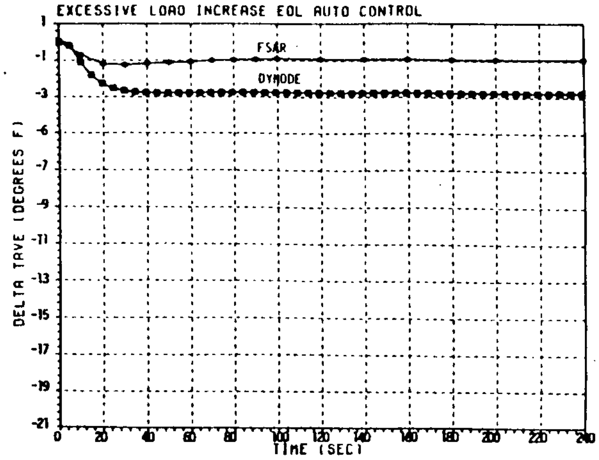


FIGURE 3.8-7

KEWAUNEE NUCLEAR POWER PLANT

EXCESSIVE LOAD INCREASE EOL AUTO CONTROL

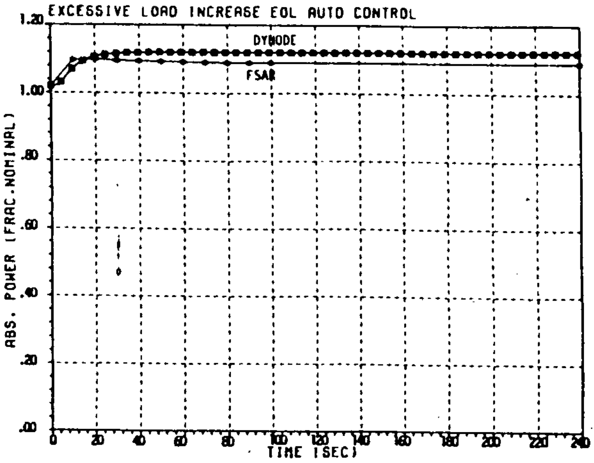


FIGURE 3.8-8

KEWAUNEE NUCLEAR POWER PLANT

EXCESSIVE LOAD INCREASE EOL AUTO CONTROL

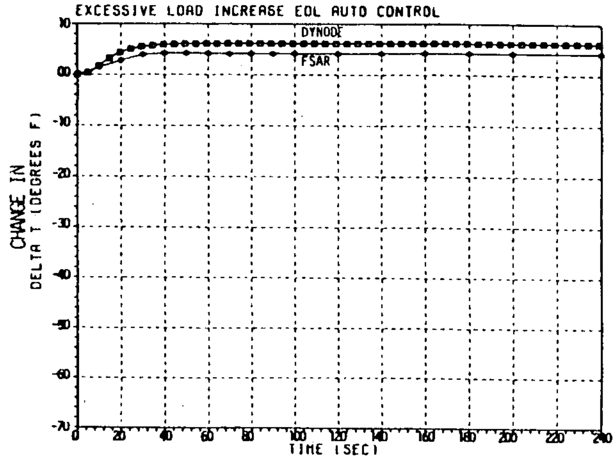


FIGURE 3.8-9

KEWAUNEE NUCLEAR POWER PLANT

EXCESSIVE LOAD INCREASE EOL AUTO CONTROL

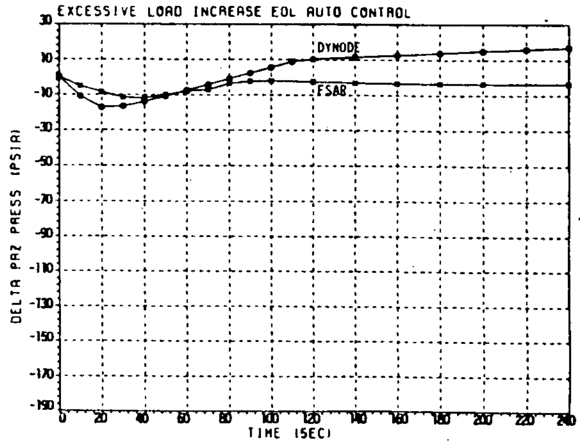
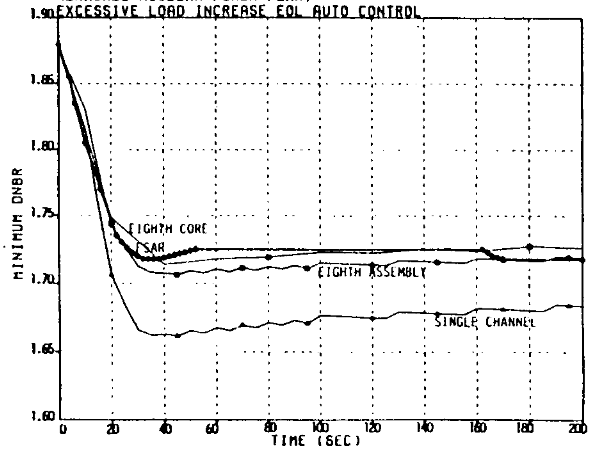


FIGURE 3.8-10

KEWAUNEE NUCLEAR POWER PLANT

EXCESSIVE LOAD INCREASE EOL AUTO CONTROL



3.9 LOSS OF EXTERNAL LOAD

3.9.1 DEFINITION OF ACCIDENT

The most likely source of a complete loss of load is a turbine-generator trip. Above approximately 10% power, a turbine trip generates a direct reactor trip which is signaled from either of two diverse inputs: release of autostop oil or stop valve closure. If credit is taken for the steam bypass system and pressurizer control system, there is no significant increase in reactor coolant temperature or pressure. To provide a conservative assessment of the accident however, no credit is taken for direct reactor trip, steam bypass actuation or pressurizer pressure control. Under these assumptions both secondary and primary pressures increase rapidly and a reactor trip is generated by the high pressurizer pressure signal.

This accident is primarily of concern from the standpoint of demonstrating the adequacy of overpressurization protection, since the hot channel MDNBR increases (or decreases only slightly) during the accident.

3.9.2 ACCIDENT ANALYSIS

The loss of external load accident is analyzed using a detailed model of the nuclear steam supply system and associated control and protection systems. Core kinetics heat transfer, reactor coolant and steam generator secondary side temperatures and pressures, steam and feedwater flowrates, and pressurizer liquid level are some of the variables computed by the model. No credit is taken for direct reactor trip caused by turbine trip, the steam bypass system or the pressurizer control system. The secondary side pressure rises to the safety valve setpoint and is limited to that pressure by steam relief through the safety valves. Scram on high pressurizer pressure mitigates the consequences of this accident and prevents water relief through the pressurizer relief and safety valves.

The worst case with respect to overpressurization assumes no control rod motion prior to reactor trip and no credit for pressurizer relief or spray valves. In this case, the magnitude of the moderator reactivity coefficient has only a very slight effect on the magnitude of the maximum reactor coolant pressure; and likewise very little effect on DNBR response.

The peak pressure is likewise insensitive to the magnitude of the Doppler reactivity coefficient, however, the least negative values of both moderator and Doppler coefficients are assumed in the analysis.

The acceptance criteria for this accident are that the maximum main steam and reactor coolant system pressures not exceed 110% of their design pressures and DNBR calculations must demonstrate that the MDNBR is not less than 1.3 at any time during the transient.

3.9.3 WPS SAFETY ANALYSIS RESULTS

WPS has analyzed the loss of load accident using input consistent with the FSAR.

The models described in Appendix A were used to analyze the loss of external load transient corresponding to BOC and EOC conditions with and without reactor control. For comparison, the NSSS response of the BOC-No Control case was also predicted with RETRAN-02. MONBR analyses were not performed since for all cases the MDNBR increased or decreased only slightly with time.

These transients are simulated by closing the turbine stop valves rapidly. The anticipated reactor trip on stop valve closure is disabled and reactor trip occurs on high pressurizer pressure. The results of these calculations are compared to USAR Figures 14.1-38 through 14.1-45 in Figures 3.9-1 through 3.9-20 of this report. In general the results of the WPS model show good agreement with those reported in the USAR.

3.9.4 CYCLE SPECIFIC PHYSICS CALCULATIONS

a. DOPPLER TEMPERATURE COEFFICIENT, α_D

Calculations of α_D are performed in accordance with the general procedures described in Section 2. Cycle specific calculations for this accident are performed at the full power equilibrium xenon condition at BOC and EOC.

b. MODERATOR TEMPERATURE COEFFICIENT, α_M

Calculations of α_M are performed in accordance with the general procedures described in Section 2. Cycle specific calculations are performed at BOC and EOC to determine the least negative value of the moderator coefficient at the full power condition. The model bias, B_M, is applied as shown in section 3.9.5.

c. SCRAM REACTIVITY CURVE, $\Delta\rho_{SCRAM}(t)$

Calculations of the scram reactivity curve are performed in accordance with the general procedures described in Section 2. Cycle specific calculations for this accident are performed at BOC and

EOC for the full power condition. A conservatively slow scram curve is generated by making the following assumptions:

1. The integral of the scram curve is based on an initial rod position at or below the full power insertion limits.

2. The shape of the scram curve is based on an initial rod position of full out. This provides the longest possible delay to significant reactivity insertion.

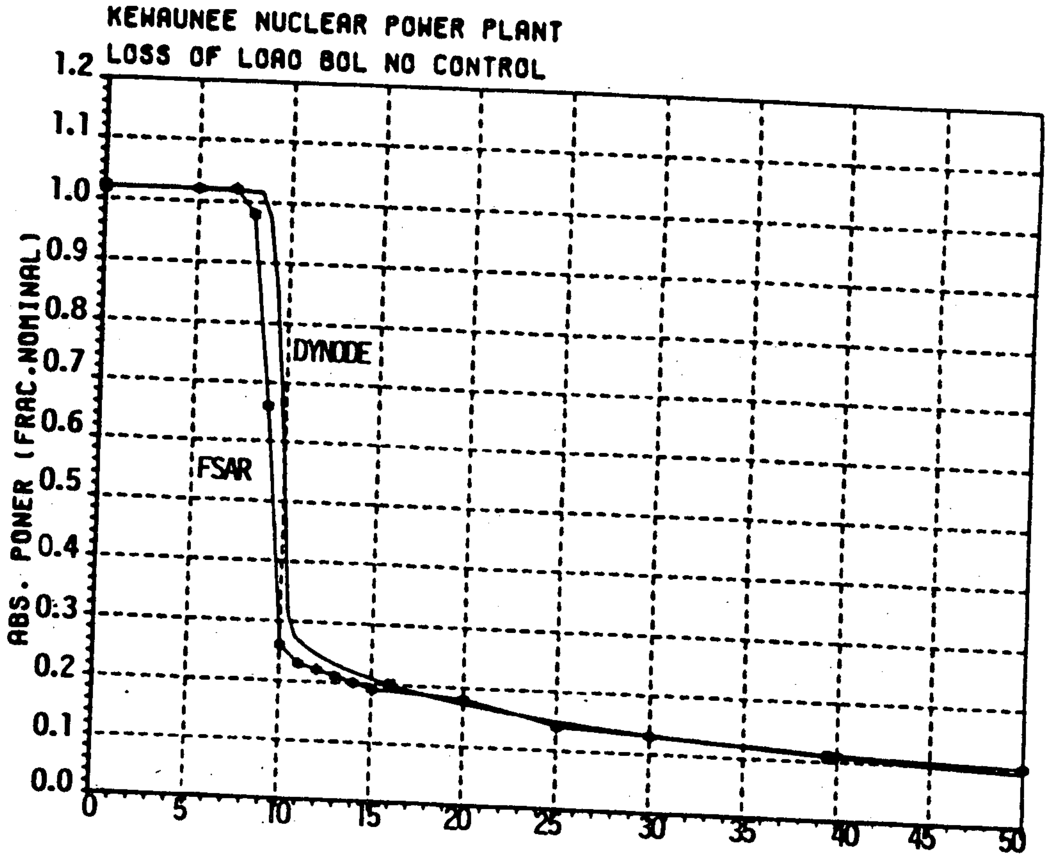
3. The xenon distribution is that which causes the minimum shutdown margin.

4. Instantaneous redistribution of flux is assumed to occur during the rod insertion.

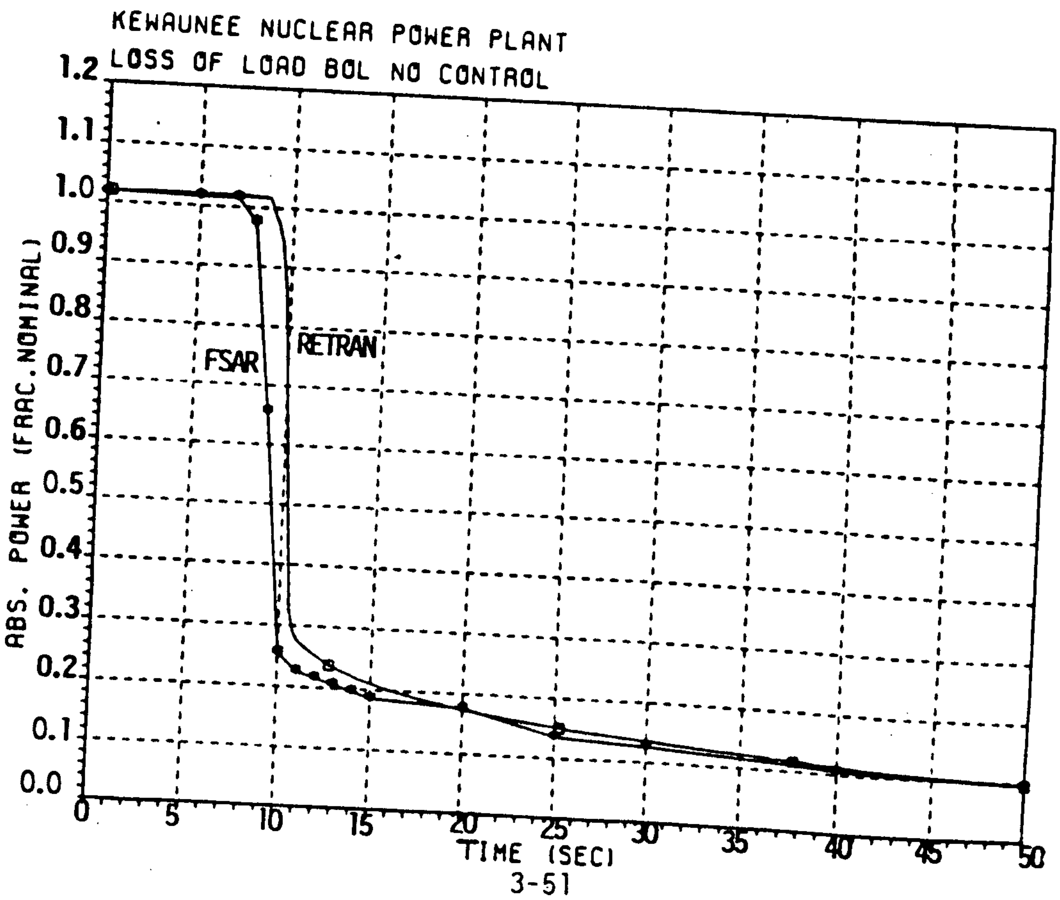
d. <u>NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR</u>, $F_{\Delta H}$

Calculations of $F_{\Delta H}$ are performed in accordance with the general procedures described in Section 2. Maximum core $F_{\Delta H}$s are verified to remain within Technical Specification limits for allowable combinations of axial offset, power level, and control rod insertion. For Kewaunee, the continuous surveillance of the power distribution is accomplished with the ex-core detectors using a Power Distribution Control scheme, PDC-II (12). The cycle specific physics calculations performed for the verification of the PDC-II scheme with respect to the $F_{\Delta H}$ limits are described in Section 3.17.

FIGURE 3.9-1

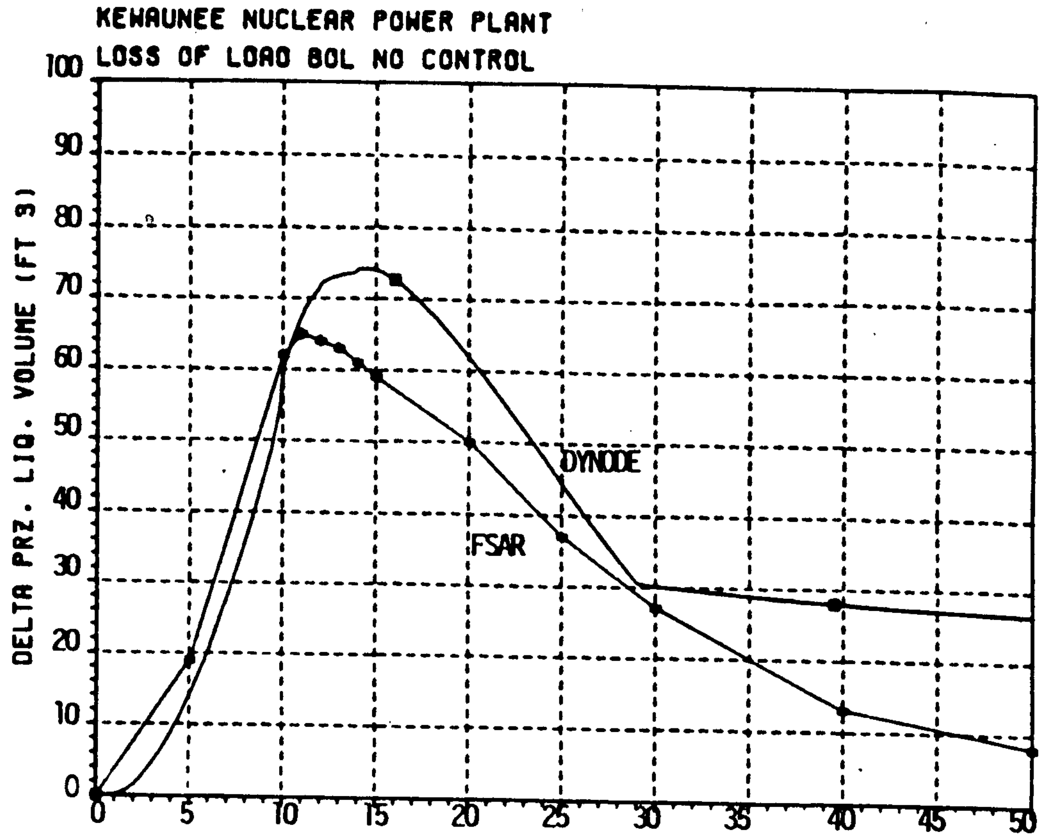


KEWAUNEE NUCLEAR POWER PLANT
LOSS OF LOAD BOL NO CONTROL
ABS. POWER (FRAC.NOMINAL)
1.2
1.1
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0.0
DYNODE
FSAR
0
5
10
15
20
25
30
35
40
45
50



KEWAUNEE NUCLEAR POWER PLANT
LOSS OF LOAD BOL NO CONTROL
ABS. POWER (FRAC.NOMINAL)
1.2
1.1
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0.0
FSAR
RETRAN
0
5
10
15
20
25
30
35
40
45
50
TIME (SEC)

FIGURE 3.9-2


KEWAUNEE NUCLEAR POWER PLANT
LOSS OF LOAD BOL NO CONTROL
DELTA PRZ. LIQ. VOLUME (FT 3)
100
90
80
70
60
50
40
30
20
10
0
0
5
10
15
20
25
30
35
40
45
50
DYNODE
FSAR

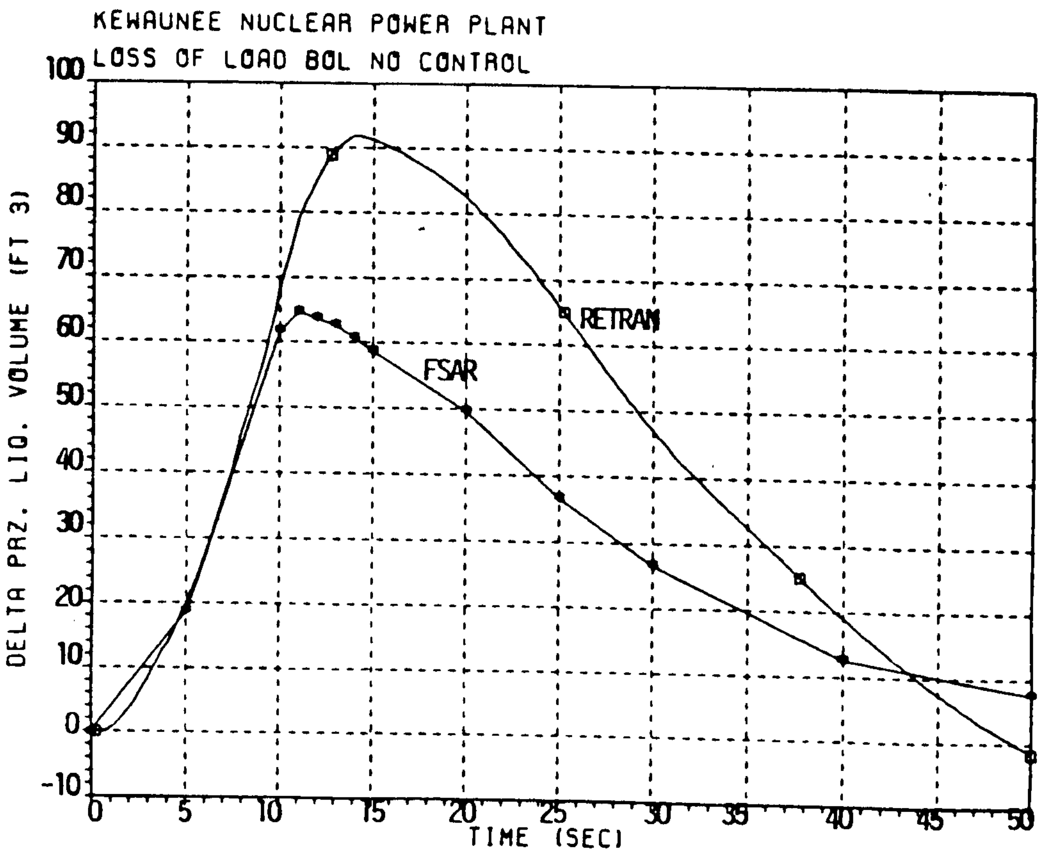
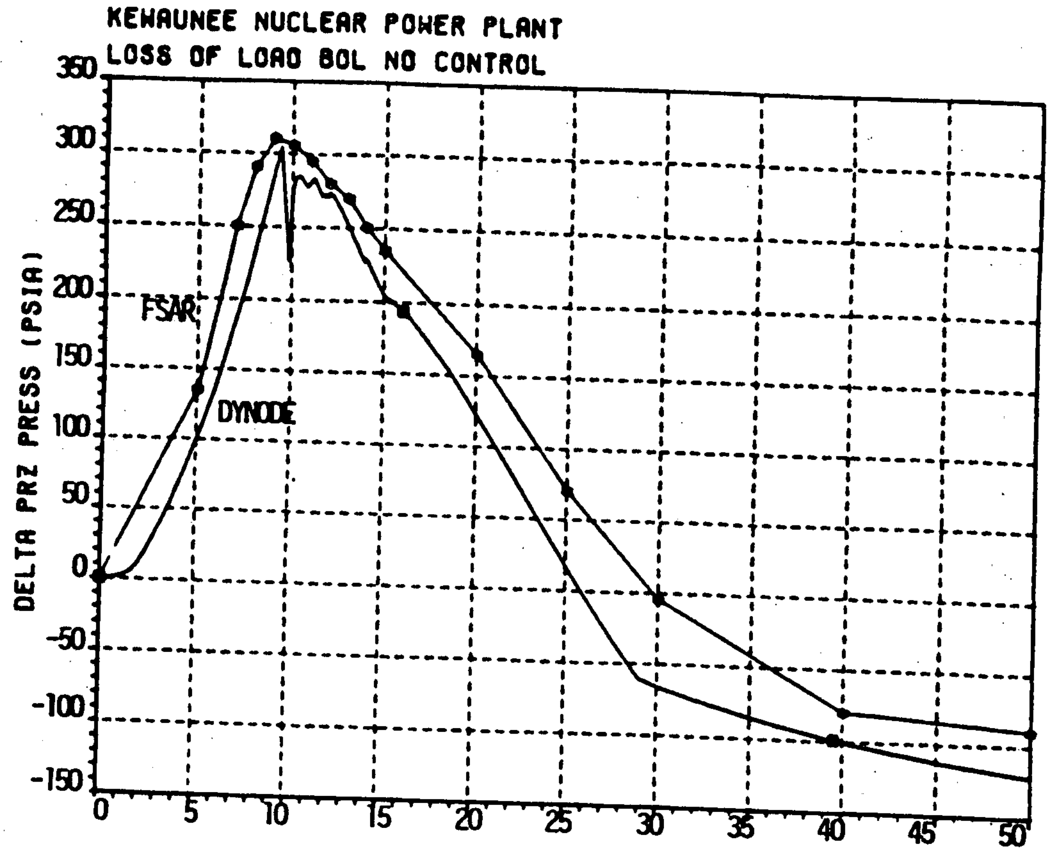

KEWAUNEE NUCLEAR POWER PLANT
LOSS OF LOAD BOL NO CONTROL
DELTA PRZ. LIQ. VOLUME (FT 3)
100
90
80
70
60
50
40
30
20
10
0
-10
0
5
10
15
20
25
30
35
40
45
50
TIME (SEC)
RETRAN
FSAR

FIGURE 3.9-3


KEWAUNEE NUCLEAR POWER PLANT
LOSS OF LOAD BOL NO CONTROL
DELTA PRZ PRESS (PSIA)
FSAR
DYNODE
350
300
250
200
150
100
50
0
-50
-100
-150
0
5
10
15
20
25
30
35
40
45
50

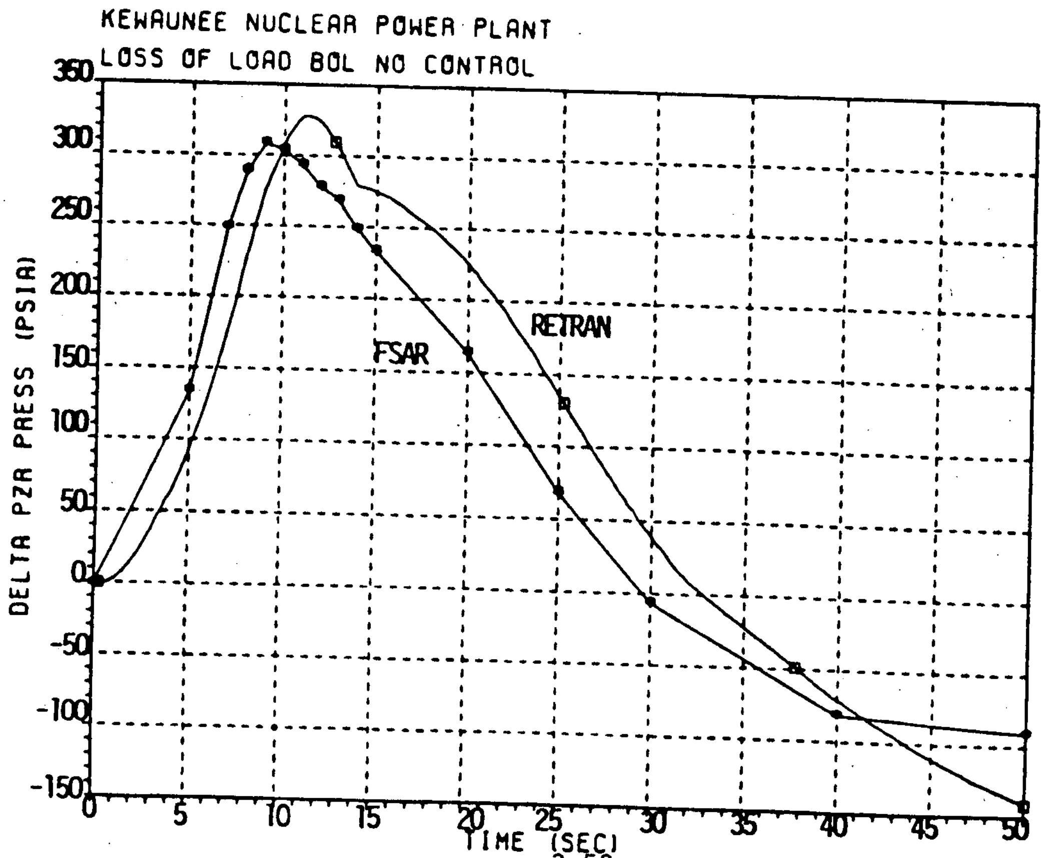
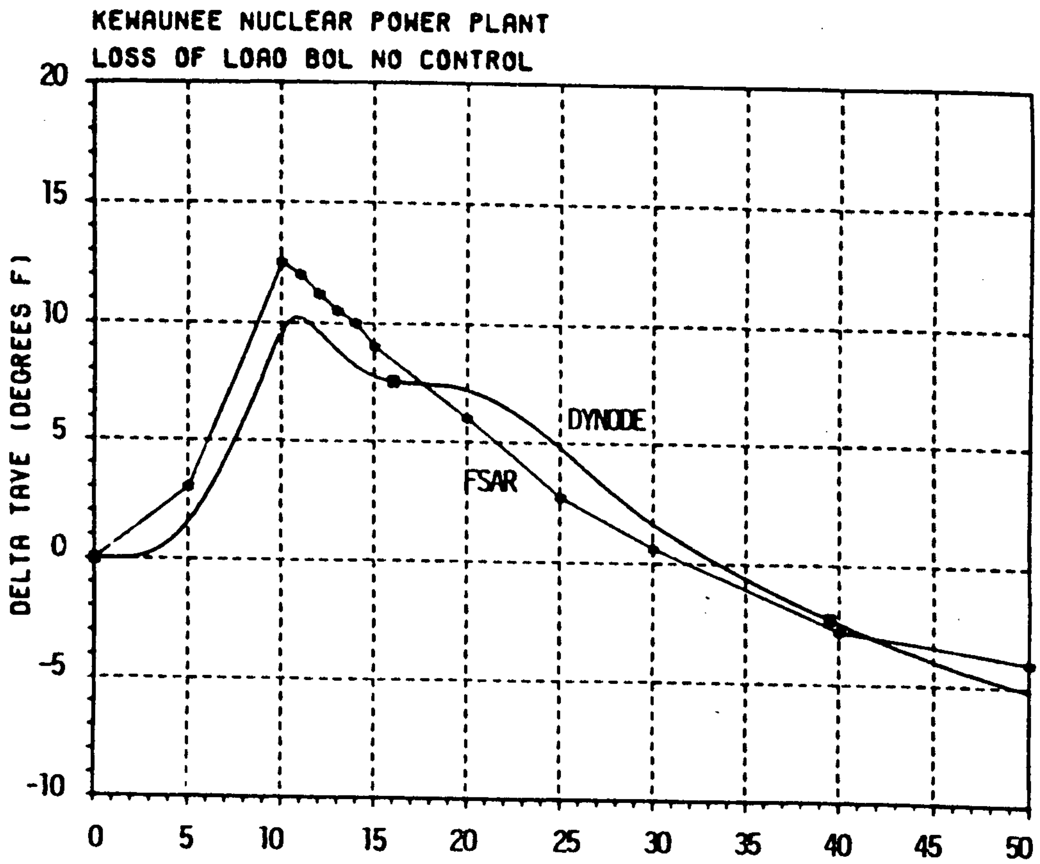

KEWAUNEE NUCLEAR POWER PLANT
LOSS OF LOAD BOL NO CONTROL
DELTA PZR PRESS (PSIA)
RETRAN
FSAR
TIME (SEC)
350
300
250
200
150
100
50
0
-50
-100
-150
0
5
10
15
20
25
30
35
40
45
50

FIGURE 3.9-4


KEWAUNEE NUCLEAR POWER PLANT
LOSS OF LOAD BOL NO CONTROL
DELTA TAVE (DEGREES F)
20
15
10
5
0
-5
-10
0
5
10
15
20
25
30
35
40
45
50
DYNODE
FSAR

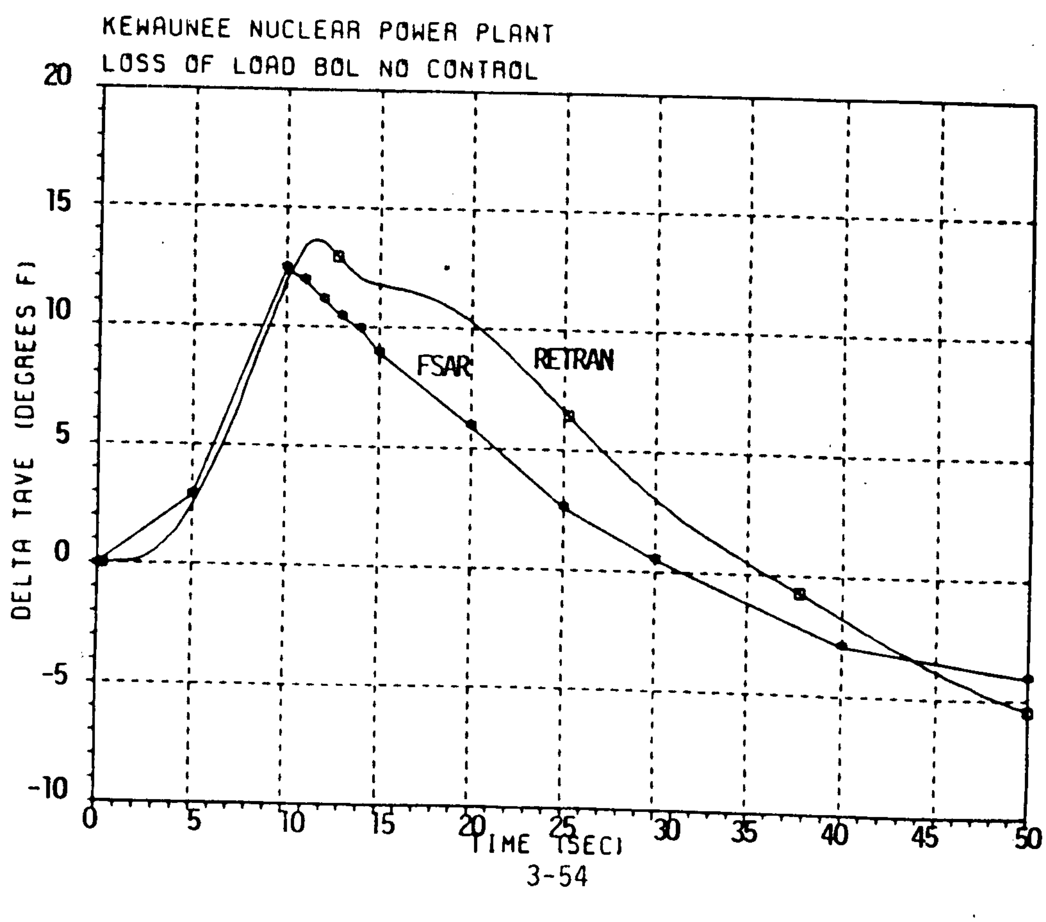
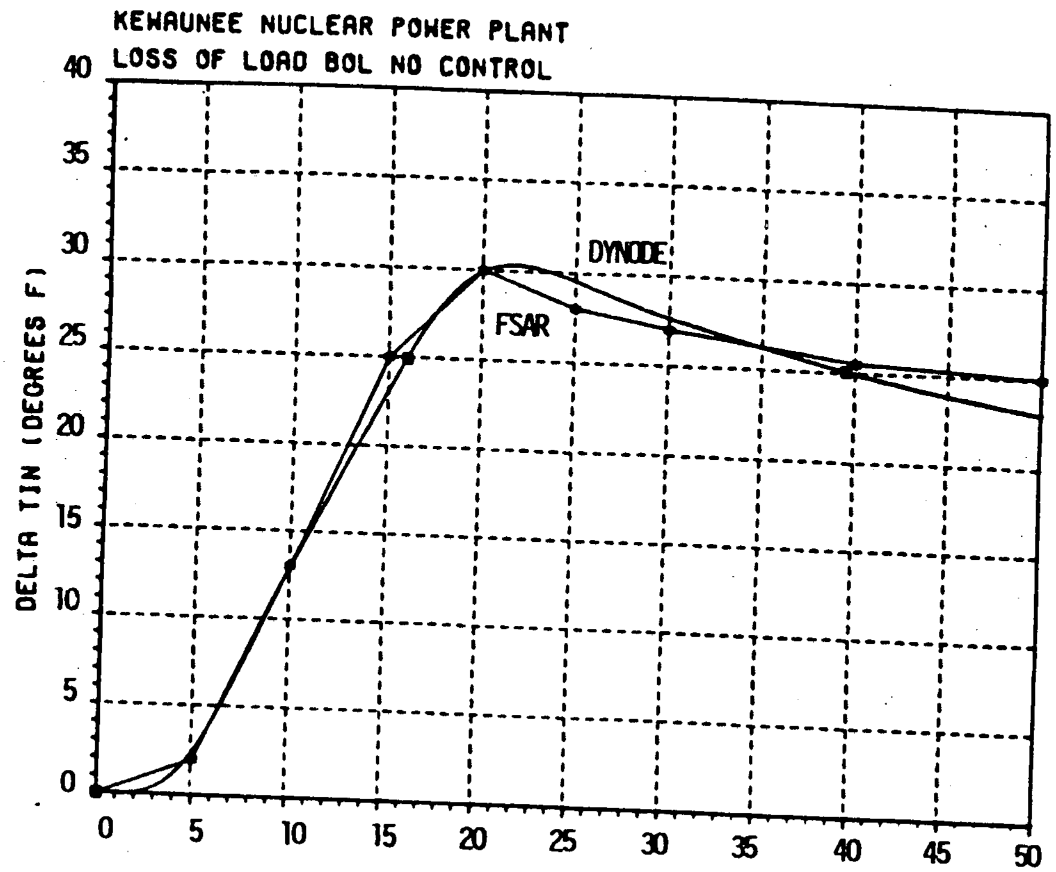

KEWAUNEE NUCLEAR POWER PLANT
LOSS OF LOAD BOL NO CONTROL
DELTA TAVE (DEGREES F)
20
15
10
5
0
-5
-10
0
5
10
15
20
25
30
35
40
45
50
TIME (SEC)
FSAR
RETRAN

FIGURE 3.9-5


KEWAUNEE NUCLEAR POWER PLANT
LOSS OF LOAD BOL NO CONTROL
DELTA TIN (DEGREES F)
DYNODE
FSAR
0
5
10
15
20
25
30
35
40
45
50

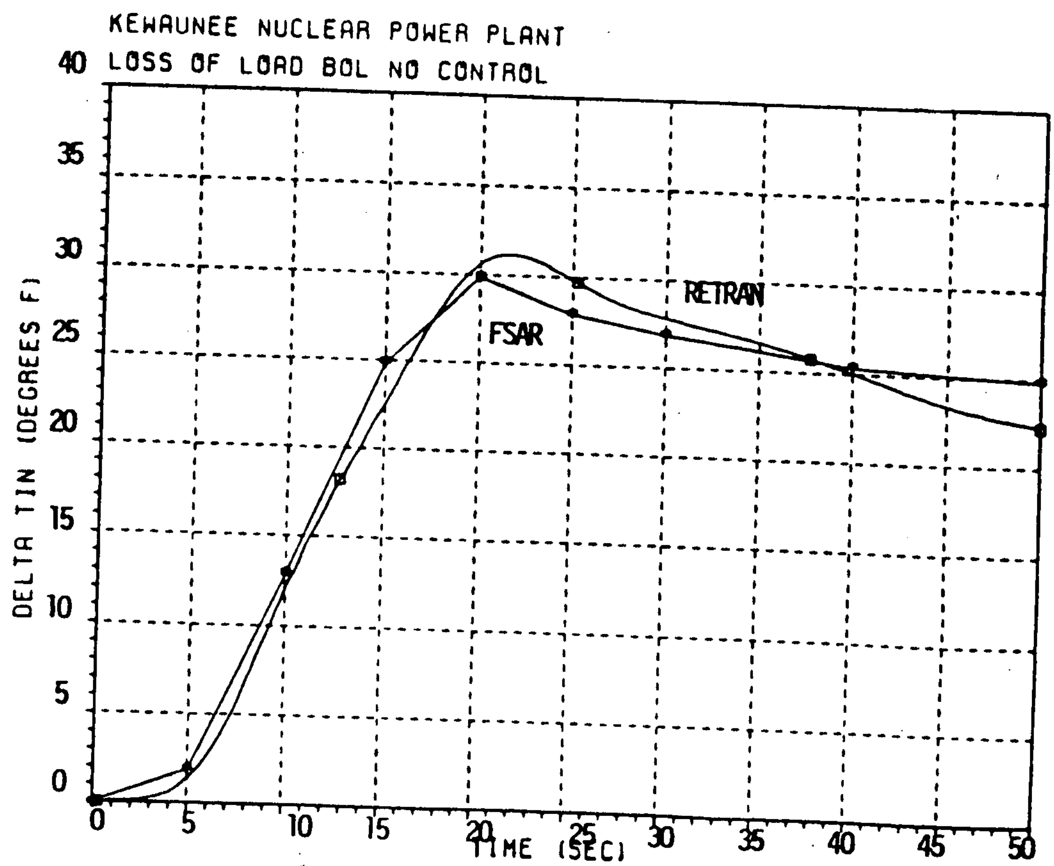

KEWAUNEE NUCLEAR POWER PLANT
LOSS OF LOAD BOL NO CONTROL
DELTA TIN (DEGREES F)
RETRAN
FSAR
TIME (SEC)
0
5
10
15
20
25
30
35
40
45
50

FIGURE 3.9-6

KEWAUNEE NUCLEAR POWER PLANT

LOSS OF LOAD EOL NO CONTROL

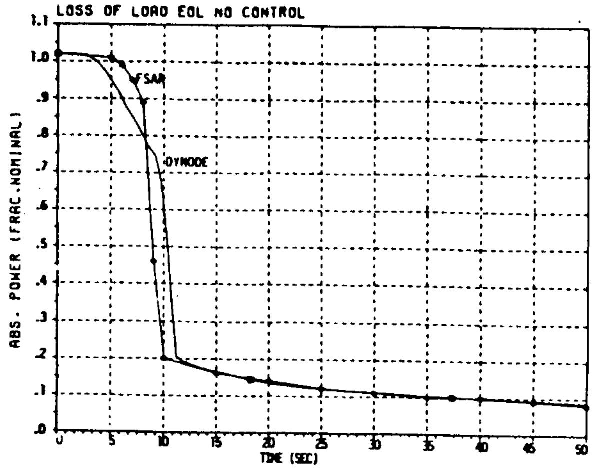


FIGURE 3.9-7

KEWAUNEE NUCLEAR POWER PLANT

LOSS OF LOAD EOL NO CONTROL

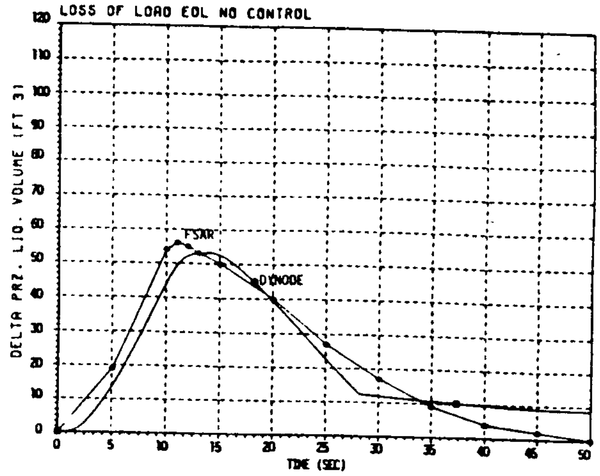


FIGURE 3.9-8

KEWAUNEE NUCLEAR POWER PLANT

LOSS OF LOAD EOL NO CONTROL

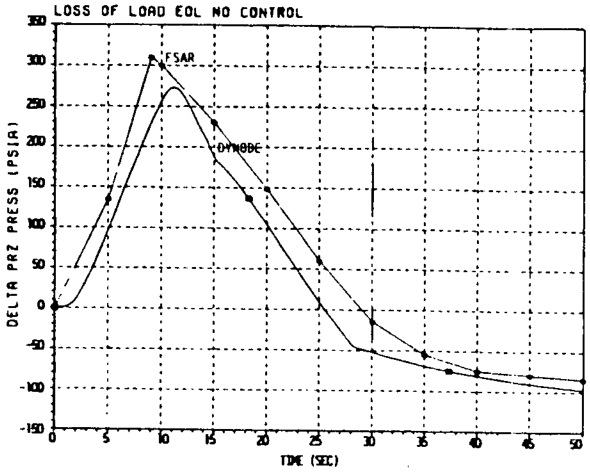


FIGURE 3.9-9

KEWAUNEE NUCLEAR POWER PLANT

LOSS OF LOAD EOL NO CONTROL

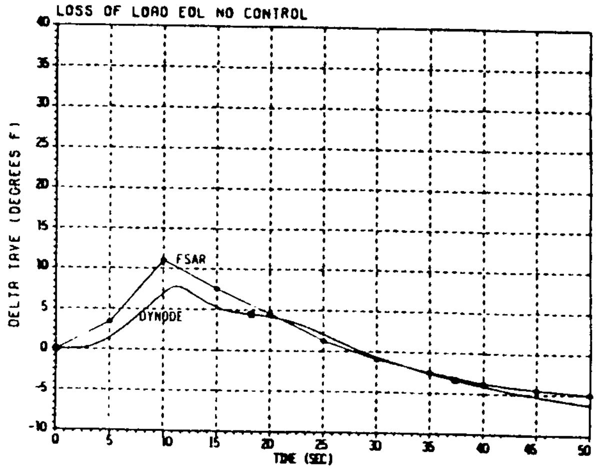
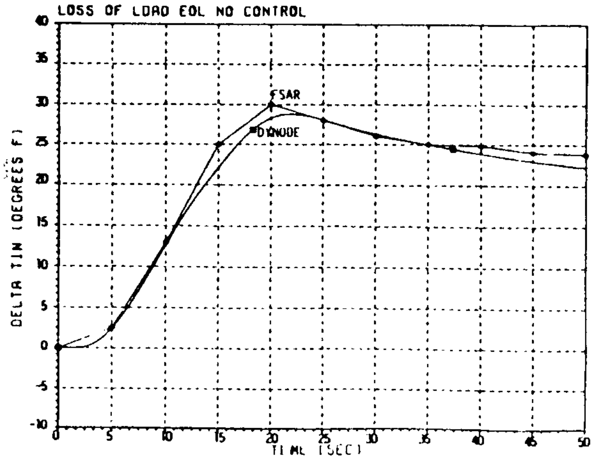


FIGURE 3.9-10

KEWAUNEE NUCLEAR POWER PLANT

LOSS OF LOAD EOL NO CONTROL



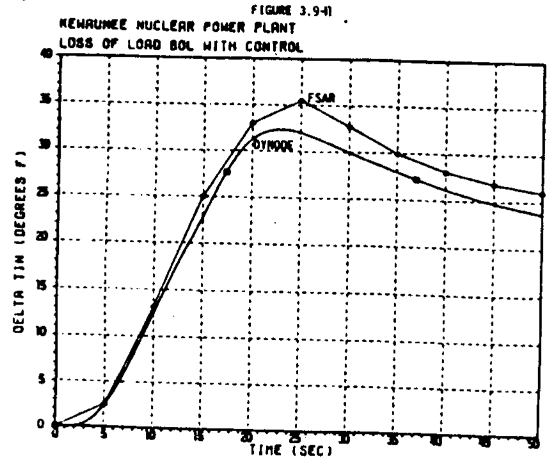


FIGURE 3.9-11
KEWAUNEE NUCLEAR POWER PLANT
LOSS OF LOAD BOL WITH CONTROL
FSAR
DYNODE
DELTA TIN (DEGREES F)
TIME (SEC)

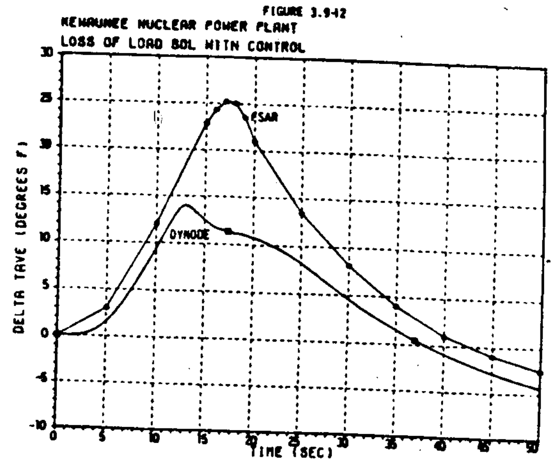


FIGURE 3.9-12
KEWAUNEE NUCLEAR POWER PLANT
LOSS OF LOAD BOL WITH CONTROL
FSAR
DYNODE
DELTA TAVE (DEGREES F)
TIME (SEC)

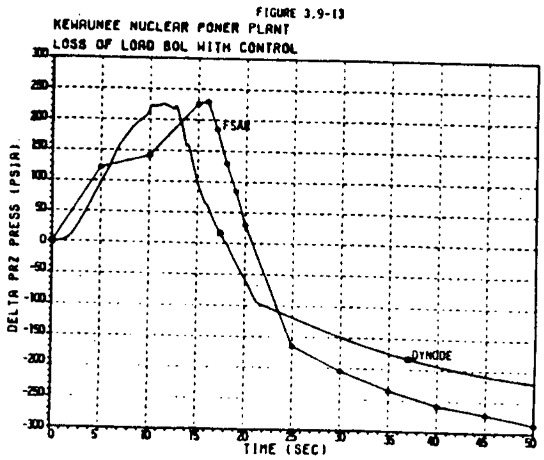


FIGURE 3.9-13
KEWAUNEE NUCLEAR POWER PLANT
LOSS OF LOAD BOL WITH CONTROL
FSAR
DYNODE
DELTA PRZ PRESS (PSIA)
TIME (SEC)

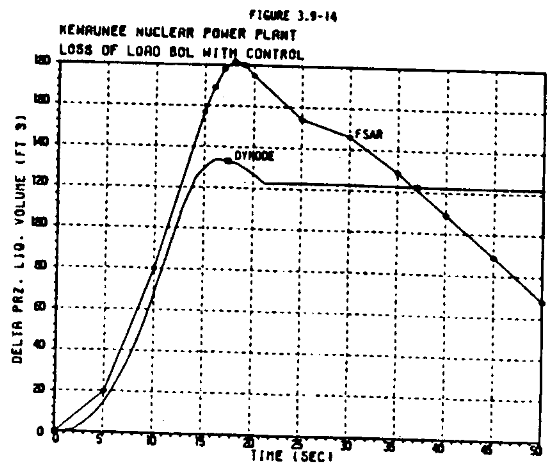


FIGURE 3.9-14
KEWAUNEE NUCLEAR POWER PLANT
LOSS OF LOAD BOL WITH CONTROL
FSAR
DYNODE
DELTA PRZ. LIQ. VOLUME (FT 3)
TIME (SEC)

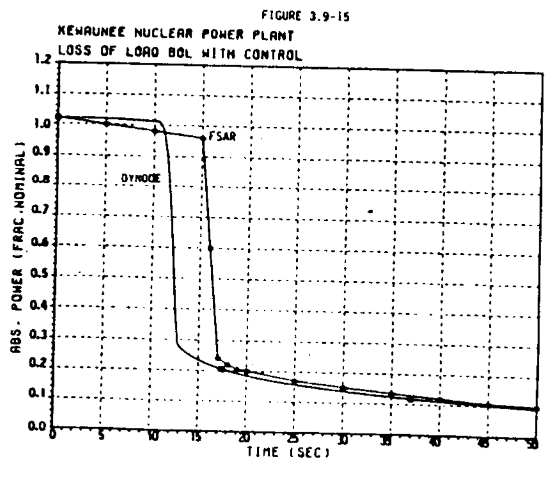


FIGURE 3.9-15
KEWAUNEE NUCLEAR POWER PLANT
LOSS OF LOAD BOL WITH CONTROL
FSAR
DYNODE
ABS. POWER (FRAC.NOMINAL)
TIME (SEC)

FIGURE 3.9-16
KEWAUNEE NUCLEAR POWER PLANT
LOSS OF LOAD EOL WITH CONTROL

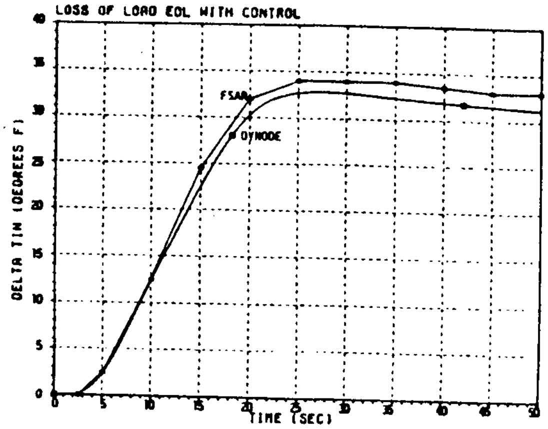


FIGURE 3.9-17
KEWAUNEE NUCLEAR POWER PLANT
LOSS OF LOAD EOL WITH CONTROL

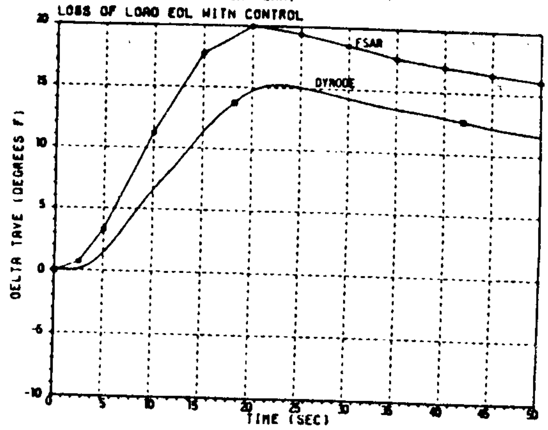


FIGURE 3.9-18
KEWAUNEE NUCLEAR POWER PLANT
LOSS OF LOAD EOL WITH CONTROL

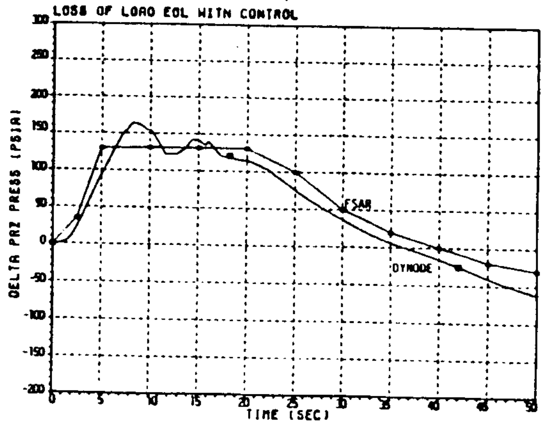


FIGURE 3.9-19
KEWAUNEE NUCLEAR POWER PLANT
LOSS OF LOAD EOL WITH CONTROL

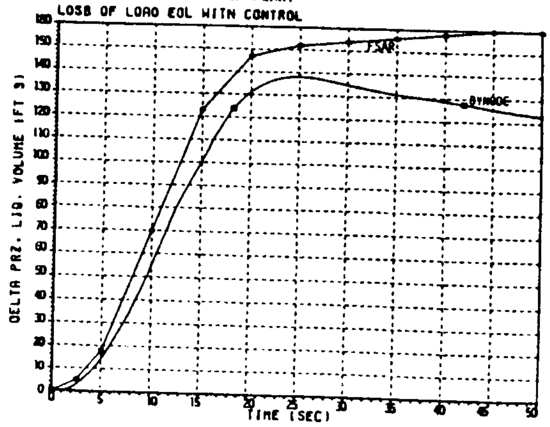
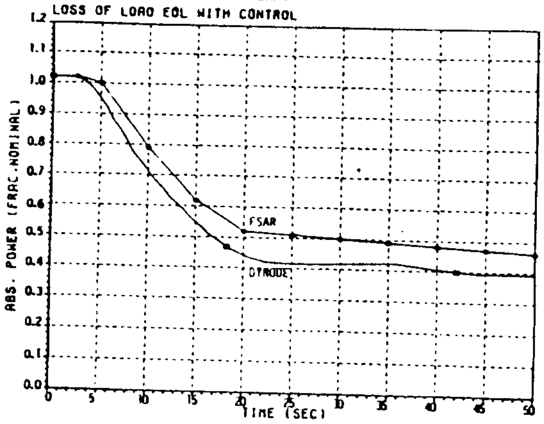


FIGURE 3.9-20
KEWAUNEE NUCLEAR POWER PLANT
LOSS OF LOAD EOL WITH CONTROL



3.9.5 RELOAD SAFETY EVALUATIONS

Each of the physics parameters calculated above was adjusted to include the model reliability factors RF_i and biases (Reference 1). These adjusted values are the cycle specific parameters which are then compared to the bounding values assumed in the safety analysis. The cycle specific parameters are acceptable with regard to loss of load transients if the following inequalities are met:

		CYCLE SPECIFIC PARAMETERS		SAFETY ANALYSIS PARAMETERS
BOC	a.	$\alpha_D * (1+RF_D)$	$\geq$	α_D (most negative bounding value) if $\alpha M < 0$
		$\alpha_D * (1-RF_D)$	$\leq$	α_D (least negative bounding value) if $\alpha M \geq 0$
	b.	$\alpha_M + RF_M + B_M$	$\leq$	α_M (least negative bounding value)
EOC	a.	$\alpha_M - RF_M + B_M$	$\geq$	α_M (most negative bounding value)
	b.	$\alpha_D * (1+RF_D)$	$\geq$	α_D (most negative bounding value)
	c.	$\lvert \Delta\rho_{SCRAM}(t)*(1-RF_{RODS}) \rvert$	$\geq$	$\lvert \Delta\rho_{SCRAM}(t) \rvert$ (bounding)
	d.	$F_{\Delta H}*(1+RFF_{\Delta H})*(1+T)$	$\leq$	Technical Specifications (Refer to Section 3.17)

The integral of the bounding value of the scram curve, $\Delta\rho_{SCRAM}(t)$, is taken as that rod worth required to produce the shutdown margin assumed in the safety analysis for the most limiting cycle specific core conditions discussed in 3.9.4.c above.

3.10 LOSS OF NORMAL FEEDWATER FLOW

3.10.1 DEFINITION OF ACCIDENT

This accident is defined as a complete loss of normal feedwater. Realistically, the plant's auxiliary feedwater pumps would be actuated and would supply sufficient feedwater to both steam generators to dissipate residual and decay heat after reactor trip. To provide a margin of conservatism however, only one of the three auxiliary feedwater pumps is assumed to deliver feedwater to one of the two steam generators. Under this assumption, the steam generator not receiving auxiliary feedwater suffers a degradation of heat transfer capability and the reactor coolant system temperature and pressure increase as a result of decay heat following reactor trip. Traditionally, an additional conservatism has been applied to the analysis of the loss of feedwater accident by assuming that the reactor coolant pumps are tripped and coast down to natural circulation conditions, further degrading the heat transfer capability of both steam generators. When analyzed in this manner, the accident corresponds to a loss of non-emergency A.C. power.

3.10.2 ACCIDENT ANALYSIS

The loss of normal feedwater accident is analyzed using a dynamic simulation model which includes the reactor and reactor coolant system and the secondary plant systems. The model includes a simulation of the natural circulation flow existing in the reactor coolant system subsequent to the assumed coastdown of the reactor coolant pumps. The model also includes the heat source due to the decay of fission products since the reactor trips on a low steam generator level signal early in the transient, and this decay heat constitutes the main energy source thereafter.

The results of the analysis of the loss of normal feedwater accident are not sensitive to the values of the core neutronics parameters. The reactor is tripped very early in the transient by the decreasing steam generator levels. Since this occurs well before steam generator heat transfer capability has been reduced, the margin to DNB is not reduced significantly prior to reactor trip. The maximum reactor coolant temperature occurs approximately 2000 seconds after accident initiation and is not significantly affected by the core neutron power transient, since decay of fission products is the major energy source over most of this time interval. The decay heat is conservatively calculated by assuming that the fission products are initially in equilibrium at the existing core power level.

The acceptance criteria for this accident are that pressure in the reactor coolant and main steam systems not exceed 110% of design pressure and that the minimum DNBR occurring during the accident be not less than 1.3 when calculated using the W-3 correlation.

3.10.3 WPS SAFETY ANALYSIS RESULTS

The loss of normal feedwater accident has been analyzed using WPS models as described in Appendix A using input data consistent with the Kewaunee USAR.

The results are compared to the corresponding results reported in Section 14.1-10 the USAR. The accident is assumed to occur as a result of isolating both steam generators from their normal supply of feedwater. Only one steam generator receives flow from one auxiliary feedpump; the other SG dries out due to steam release through the safety valves. A trip of both reactor coolant pumps, postulated to occur simultaneously, results in a further degradation of heat transfer capability.

The results obtained from the WPS models are compared to Figures 14.1-46(a)-(c) of the USAR in Figures 3.10-1 through 3.10-4.

The fact that the unfed SG is predicted to dry out at approximately the same time in both analyses indicates that comparable initial shell side mass inventories and transient safety valve flow rates were used in both the WPS model and the USAR analyses.

A volume balance based on the thermal expansion of the RCS fluid indicates that the pressurizer volume surge is consistent with the RCS temperature calculated by the WPS model. A mass, energy, and volume balance on the shell side of the SG receiving auxiliary feedwater indicates that the level response is correct. In general, the results of the WPS model and those reported in the USAR show the same trends.

3.10.4 CYCLE SPECIFIC PHYSICS CALCULATIONS

The loss of normal feedwater transient is not sensitive to core physics parameters since the reactor is assumed to trip in the initial stages (approximately 2 seconds) of the transient. This trip occurs due to a lo-lo steam generator level signal well before the heat transfer capability of the steam generator is reduced. The transient is then driven by the decay heat from the tripped reactor Also, the loss of flow transient analyzed in Section 3.11 is considered a more severe transient of this type.

Therefore no comparisons will be made for reload safety evaluations.

FIGURE 3.10-1

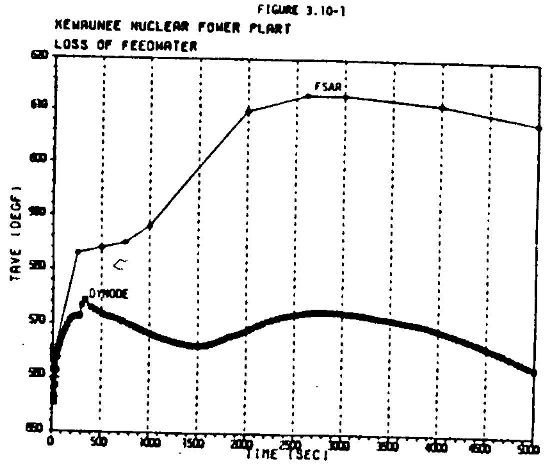


FIGURE 3.10-2

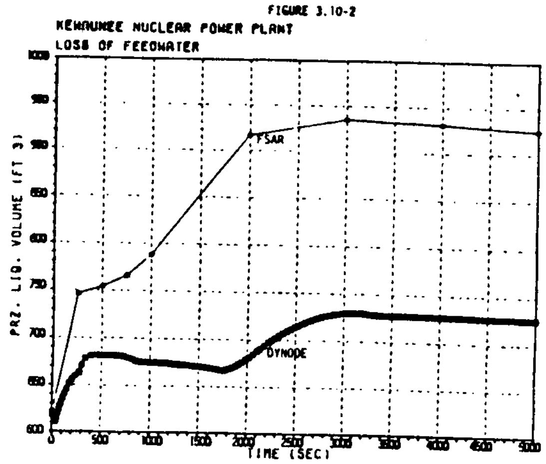


FIGURE 3.10-3

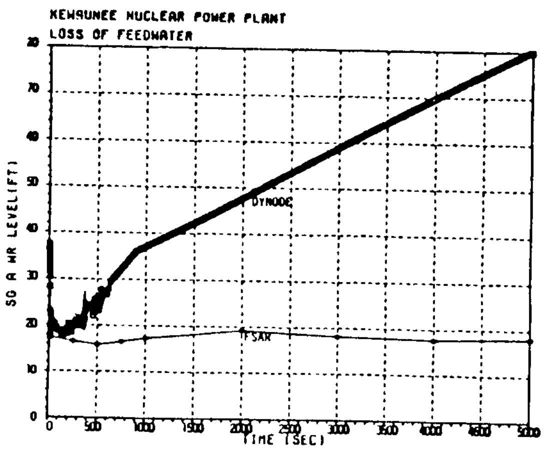
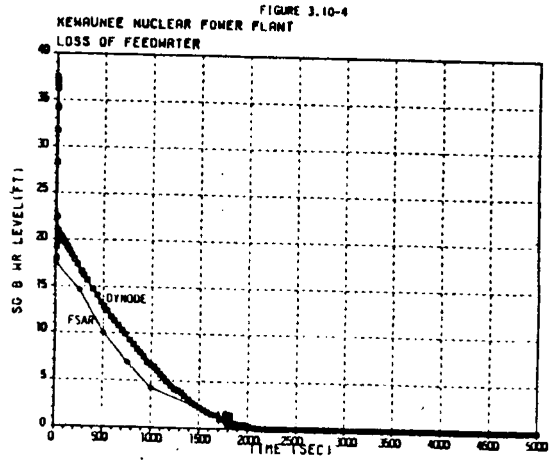


FIGURE 3.10-4



3.11 LOSS OF REACTOR COOLANT FLOW - PUMP TRIP

3.11.1 DEFINITION OF ACCIDENT

The accident considered here is the simultaneous loss of electrical power to both of the reactor coolant pumps. As a result of loss of driving head supplied by the pumps, the coolant flow rate decreases, but is retarded by the rotational inertia of the reactor coolant pump flywheel and by the hydraulic inertia of the fluid itself. The reactor is tripped by any one of several diverse and redundant signals which monitor coolant pump and coolant flow conditions. This trip results in a power reduction before the thermal-hydraulic conditions in the core approach those which could result in damage to the fuel. Loss of power to one of the pumps with both pumps initially operating is also considered, but the consequences are less severe than for the two pump trip. Seizure of the reactor coolant pump shaft is considered in section 3.12.

3.11.2 ACCIDENT ANALYSIS

The loss of forced reactor coolant flow accident is analyzed using a detailed model of the reactor coolant system thermal-hydraulics. The conservation of momentum and continuity equations for the coolant, coupled to a representation of the pump hydraulics and speed coastdown, are solved to compute the system flowrate as a function of time. Reactor core neutron kinetics and heat transfer equations are coupled to the flow coastdown equations in order to compute heat flux and coolant temperatures in the reactor. A simulation of the steam generators and pressurizer is also included in the model. A separate model analyzes the transient response of the core hot channel, using conditions supplied by the NSSS model as input, and computes the departure from nucleate boiling ratio (DNBR).

The initial conditions for the accident analysis assume the most adverse combination of power, core inlet temperature, and pressurizer pressure including allowances for steady state error so that the initial margin to DNB is the minimum expected during steady state operation.

The power transient is analyzed using the least negative value of moderator reactivity coefficient calculated to occur during the cycle. For the sake of conservatism, a value of zero is assumed in the analysis even though the moderator coefficient is expected to remain negative.

for all normal operating conditions. The most negative value of Doppler reactivity coefficient calculated to occur during the cycle is used in the analysis since this value has been shown to result in the maximum hot spot heat flux at the time of minimum DNBR. The reactivity reduction due to control rod insertion after trip is calculated by assuming the most adverse delay time expected to occur between loss of power to the pump and the initiation of rod motion. Upon reactor trip, it is assumed that the most reactive RCC assembly is stuck in its fully withdrawn position, resulting in a minimum insertion of negative reactivity. The trip reactivity insertion dominates the power response and is the most important neutronics input parameter.

The acceptance criteria for the loss of reactor coolant flow accident are that the minimum DNBR be not less than 1.3 and that the maximum reactor coolant and main steam system pressures not exceed 110% of their design values.

3.11.3 WPS SAFETY ANALYSIS RESULTS

WPS has analyzed the loss of reactor coolant flow accident using input consistent with the USAR (4).

The models described in Appendix A were used to analyze the following transient cases:

- Loss of power to one reactor coolant pump with two pumps initially running (1/2 pump trip)

- Loss of power to two reactor coolant pumps with two pumps initially running (2/2 pump trip)

The results of these two cases are compared to the corresponding results of USAR Section 14.1-8.

In addition to DYNODE-P, the NSSS response to a 2/2 pump trip transient was predicted by RETRAN-02 for comparison. The transient forcing function inputs to VIPRE required for MDNBR analysis were derived from the DYNODE-P results in all cases.

Additional comparisons using the RETRAN and DYNODE best estimate models were made to simulate two loss of flow startup tests (1/2 and 2/2 reactor coolant pump trip tests) conducted at the Kewaunee Nuclear Power Plant. The results of these comparisons are included in Appendix F of this report.

Figures 3.11-1, 3.11-2, 3.11-3, and 3.11-4 compare the results of the WPSC analyses for the 1/2 pump trip to the corresponding results from USAR Figures 14.1-29, 14.1-30, and 14.1-31. As shown, the WPS model predicts a slightly earlier decrease in nuclear power heat flux due to a slightly earlier reactor trip on low reactor coolant flow. These results cause the WPS predicted MDNBR to be slightly higher.

Comparisons of WPS results for the 2/2 pump trip case to USAR Figures 14.1-26, 14.1-27, and 14.1-28 are shown in Figures 3.11-5 through 3.11-8 of this report. The nuclear power and heat flux predicted by WPS models decrease at a slightly slower rate following reactor trip on low reactor coolant flow. This results in the WPS VIPRE model predicting a MDNBR during the transient of approximately the same magnitude as the USAR but shifted slightly to a later time.

DNBR analyses were performed with single channel and multichannel models. The single channel model calculates the most conservative MDNBR response and also demonstrates the best agreement to the USAR results.

3.11.4 CYCLE SPECIFIC PHYSIC CALCULATIONS

a. DOPPLER TEMPERATURE COEFFICIENT, α_D

Calculations of αD are performed in accordance with the general procedures described in Section 2. Cycle specific calculations for this accident are performed at BOC and EOC to determine the most negative value at full power conditions.

b. MODERATOR TEMPERATURE COEFFICIENT, α_M

Calculations of αM are performed in accordance with the general procedures described in Section 2. Cycle specific calculations are performed at BOC to determine the least negative value of the moderator coefficient at the full power condition. The model bias, B_M, is included in the calculations.

c. SCRAM REACTIVITY CURVE, $\Delta\rho_{SCRAM}(t)$

Calculations of the scram reactivity curve are performed in accordance with the general procedures described in Section 2. Cycle specific calculations for this accident are performed at BOC and EOC for the full power condition.

A conservatively slow scram curve is generated by making the following assumptions:

1. The integral of the scram curve is based on an initial rod position at or below the full power insertion limits.

2. The shape of the scram curve is based on an initial rod position of full out. This provides the longest possible delay to significant reactivity insertion.

3. The xenon distribution is that which causes the minimum shutdown margin.

4. Instantaneous redistribution of flux is assumed to occur during the rod insertion.

d. NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR, $F_{\Delta H}$

Calculations of $F_{\Delta H}$ are performed in accordance with the general procedures described in Section 2. Maximum core $F_{\Delta H}$s are verified to remain within Technical Specification limits for allowable combinations of axial offset, power level, and control rod insertion. The safety analysis value for FΔH bounds the Technical Specification limit.

e. FUEL TEMPERATURE, T_f

Fuel temperature is calculated at conditions to maximize fuel temperature as a function of linear heat generation rate. The maximum cycle specific linear heat generation rate is used to derive the maximum cycle specific fuel temperature.

3.11.5 RELOAD SAFETY EVALUATIONS

Each of the physics parameters calculated above are adjusted to include the model reliability factors RF_i(1) and biases. These adjusted values are the cycle specific parameters which are then compared to the bounding values assumed in the safety analysis. The cycle specific parameters are acceptable with regard to loss of reactor coolant flow pump trip transients if the following inequalities are met:

CYCLE SPECIFIC PARAMETERS		SAFETY ANALYSIS PARAMETERS
a. $\alpha_D*(1+RF_D)$	$\geq$	α_D (most negative bounding value) if $\alpha M \leq 0$
b. $\alpha_M+RF_M+B_M$	$\leq$	α_M (least negative bounding value)
c. $\lvert \Delta\rho_{SCRAM}(t)*(1-RF_{RODS})\rvert$	$\geq$	$\lvert \Delta\rho_{SCRAM}(t)\rvert$ (bounding)
d. $F_{\Delta H}*(1+RFF_{\Delta H})*(1+T)$	$\leq$	FΔH (bounding value Loss of Flow)
e. $T_f * (1+RFT_f)$	$\leq$	T_f (bounding value Loss of Flow)

The integral of the bounding value of the scram curve, $\Delta\rho_{SCRAM}(t)$, is taken as that rod worth required to produce the shutdown margin assumed in the safety analysis for the most limiting cycle specific core conditions discussed in 3.11.4.c above.

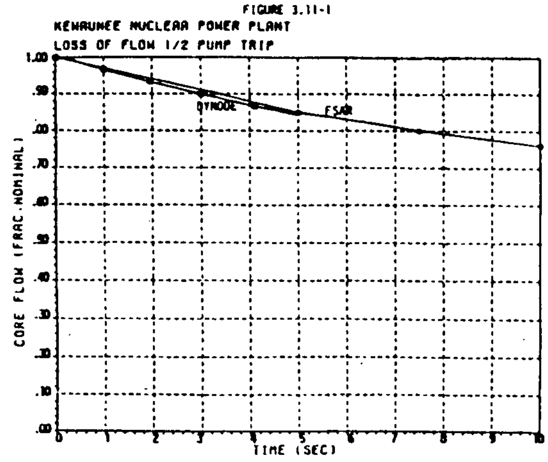


FIGURE 3.11-1
KEWAUNEE NUCLEAR POWER PLANT
LOSS OF FLOW 1/2 PUMP TRIP
DYNODE
FSAR
CORE FLOW (FRAC.NOMINAL)
TIME (SEC)

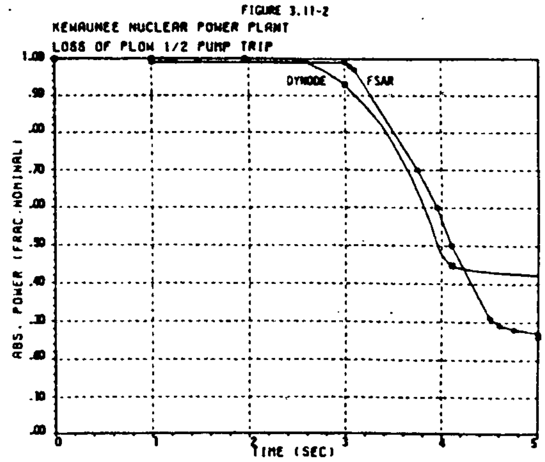


FIGURE 3.11-2
KEWAUNEE NUCLEAR POWER PLANT
LOSS OF FLOW 1/2 PUMP TRIP
DYNODE
FSAR
ABS. POWER (FRAC.NOMINAL)
TIME (SEC)

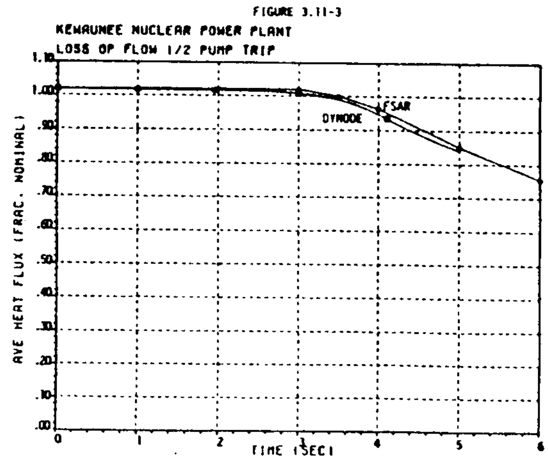


FIGURE 3.11-3
KEWAUNEE NUCLEAR POWER PLANT
LOSS OF FLOW 1/2 PUMP TRIP
FSAR
DYNODE
AVE HEAT FLUX (FRAC. NOMINAL)
TIME (SEC)

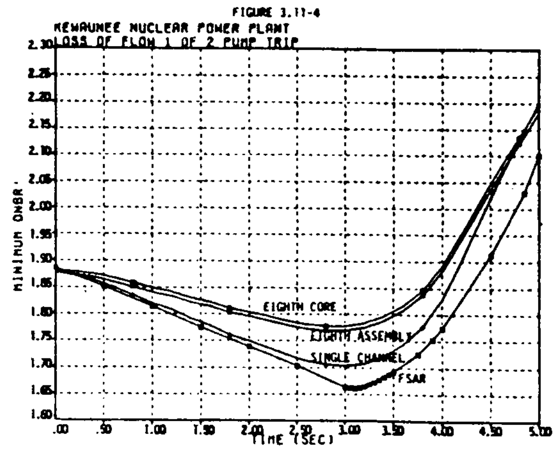
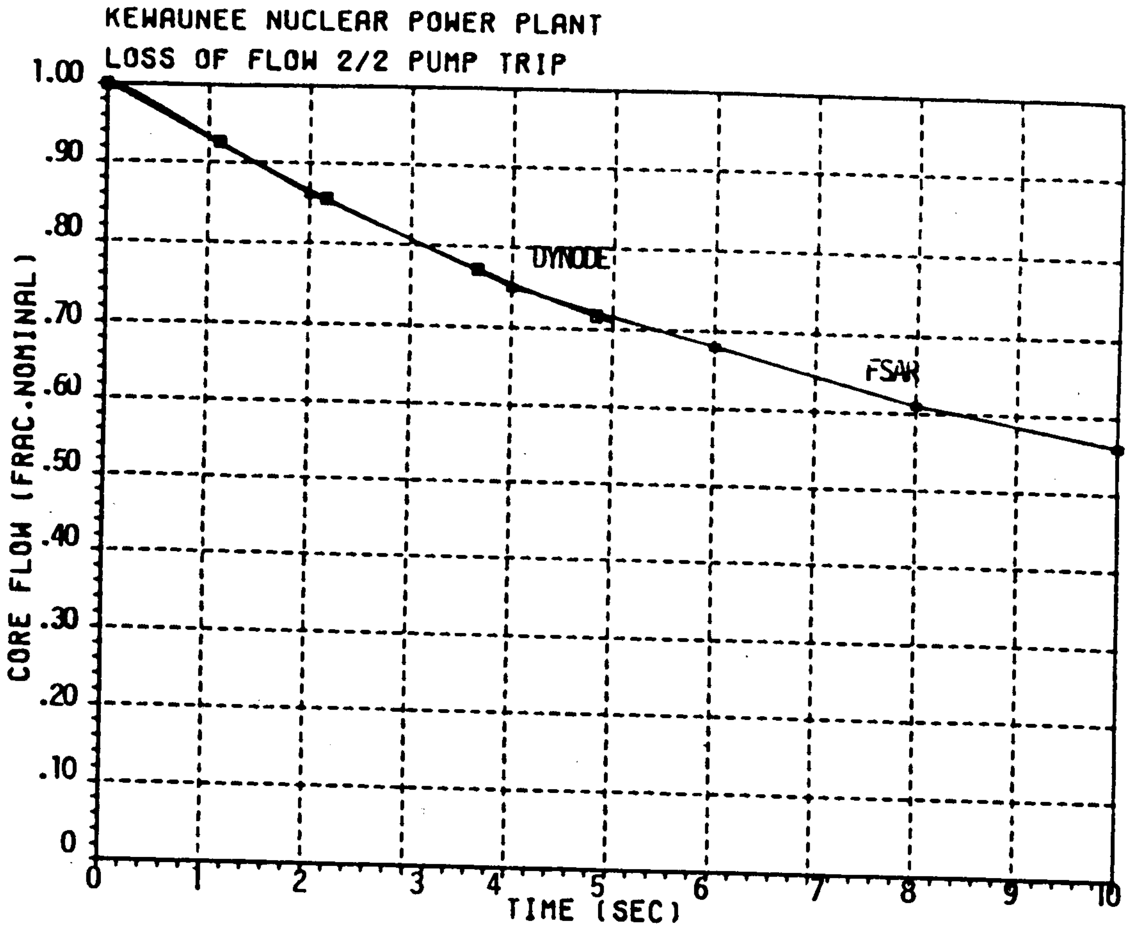


FIGURE 3.11-4
KEWAUNEE NUCLEAR POWER PLANT
LOSS OF FLOW 1 OF 2 PUMP TRIP
EIGHTH CORE
EIGHTH ASSEMBLY
SINGLE CHANNEL
FSAR
MINIMUM DNBR
TIME (SEC)

FIGURE 3.11-5


KEWAUNEE NUCLEAR POWER PLANT
LOSS OF FLOW 2/2 PUMP TRIP
CORE FLOW (FRAC.NOMINAL)
1.00
.90
.80
.70
.60
.50
.40
.30
.20
.10
0
DYNODE
FSAR
0
1
2
3
4
5
6
7
8
9
10
TIME (SEC)

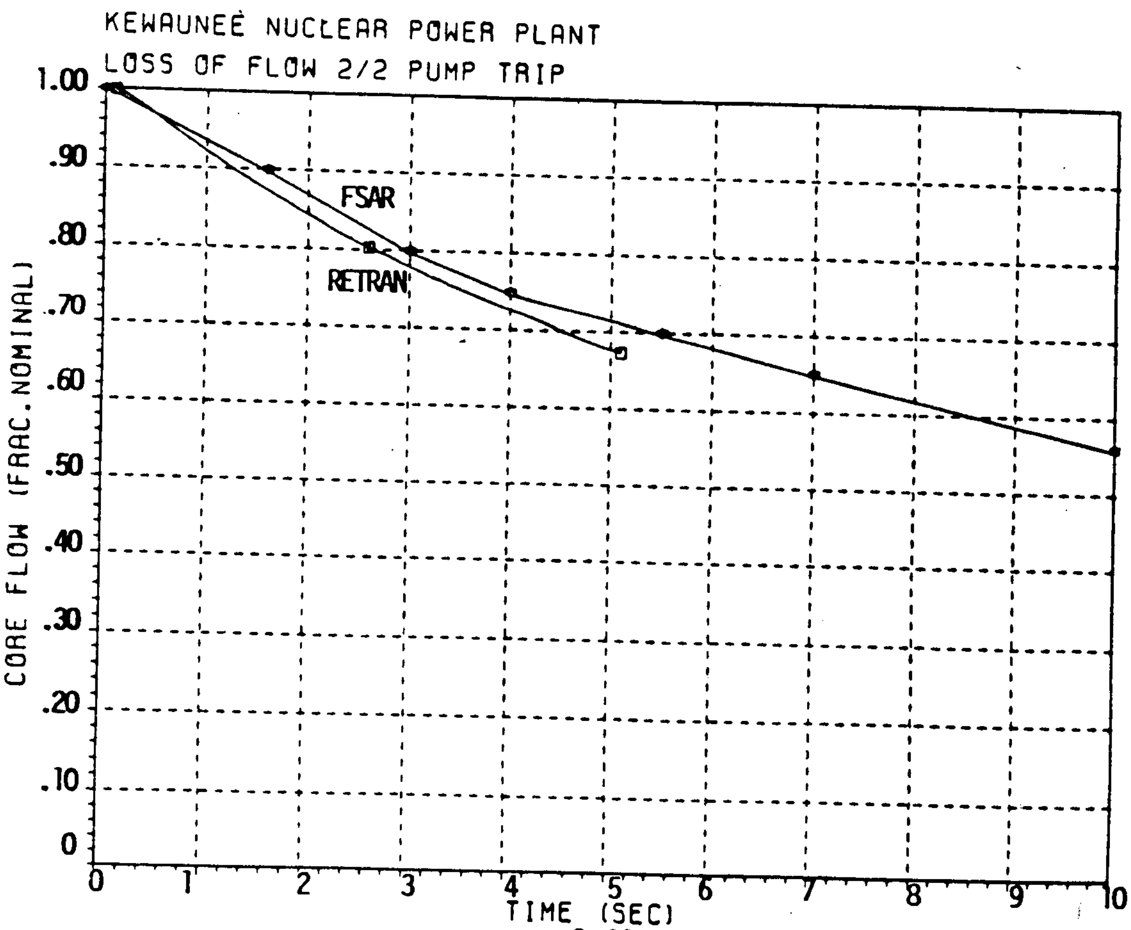
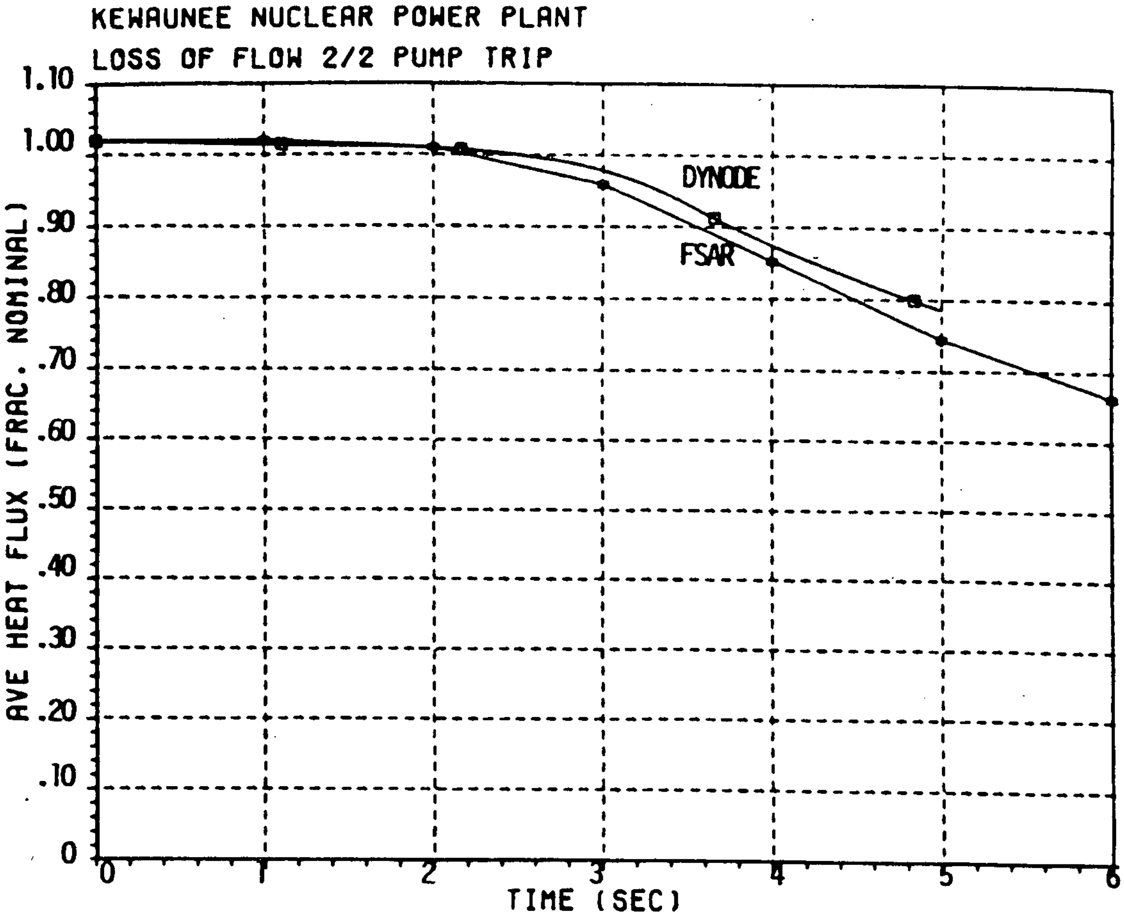
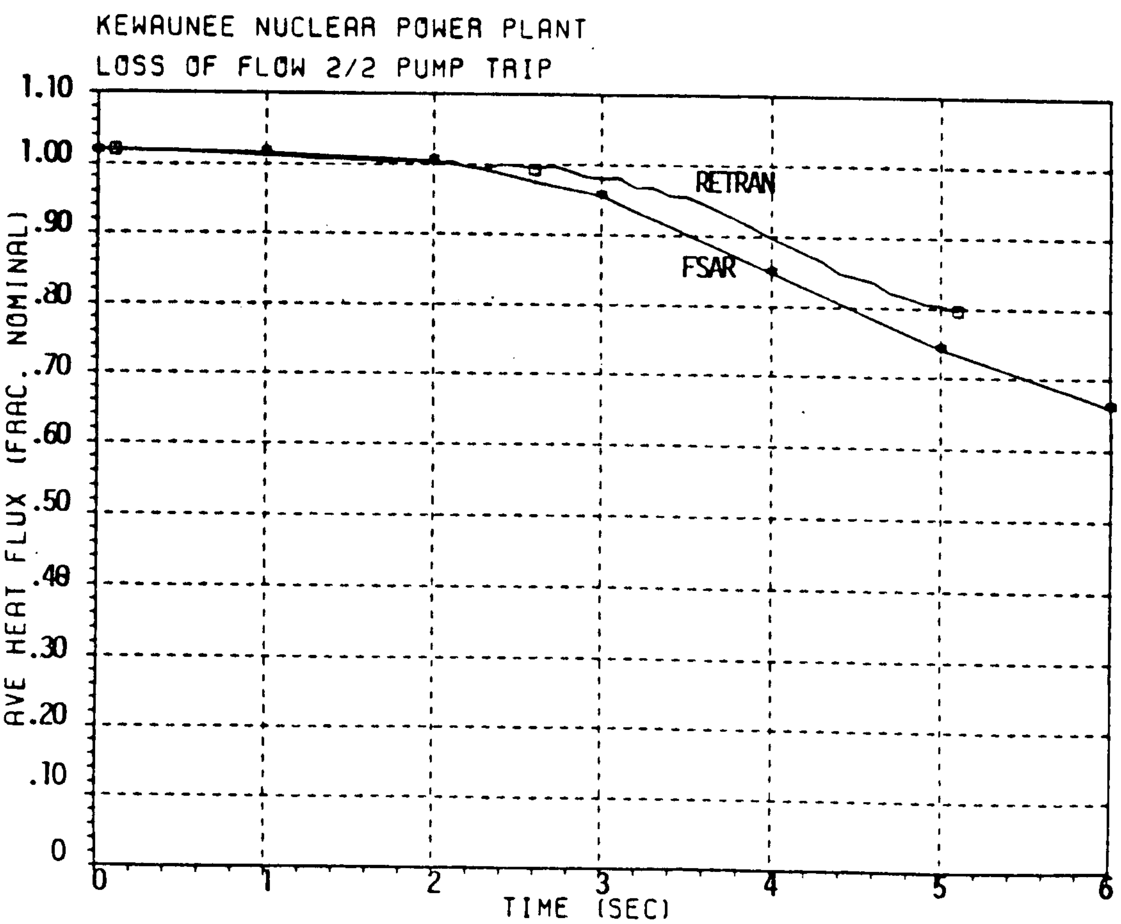

KEWAUNEE NUCLEAR POWER PLANT
LOSS OF FLOW 2/2 PUMP TRIP
CORE FLOW (FRAC.NOMINAL)
1.00
.90
.80
.70
.60
.50
.40
.30
.20
.10
0
FSAR
RETRAN
0
1
2
3
4
5
6
7
8
9
10
TIME (SEC)

FIGURE 3.11-6

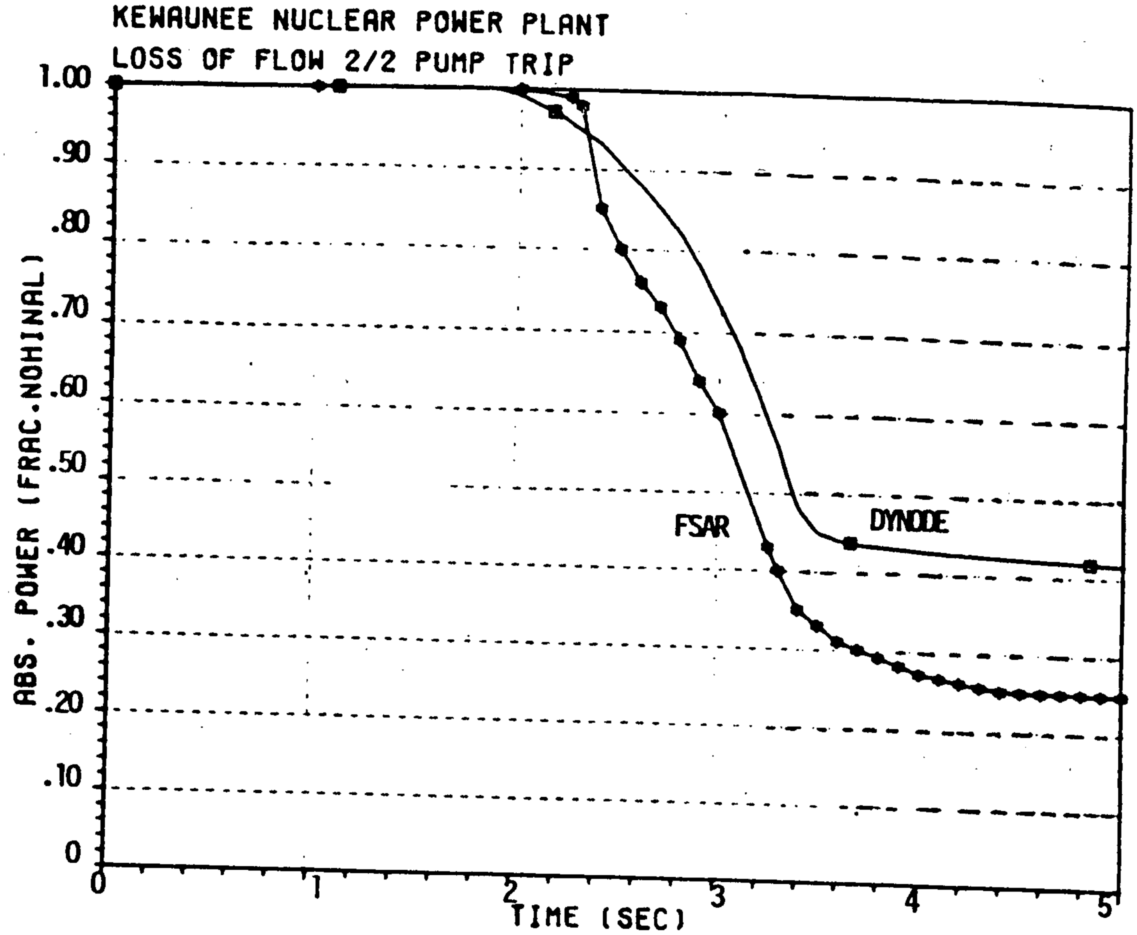


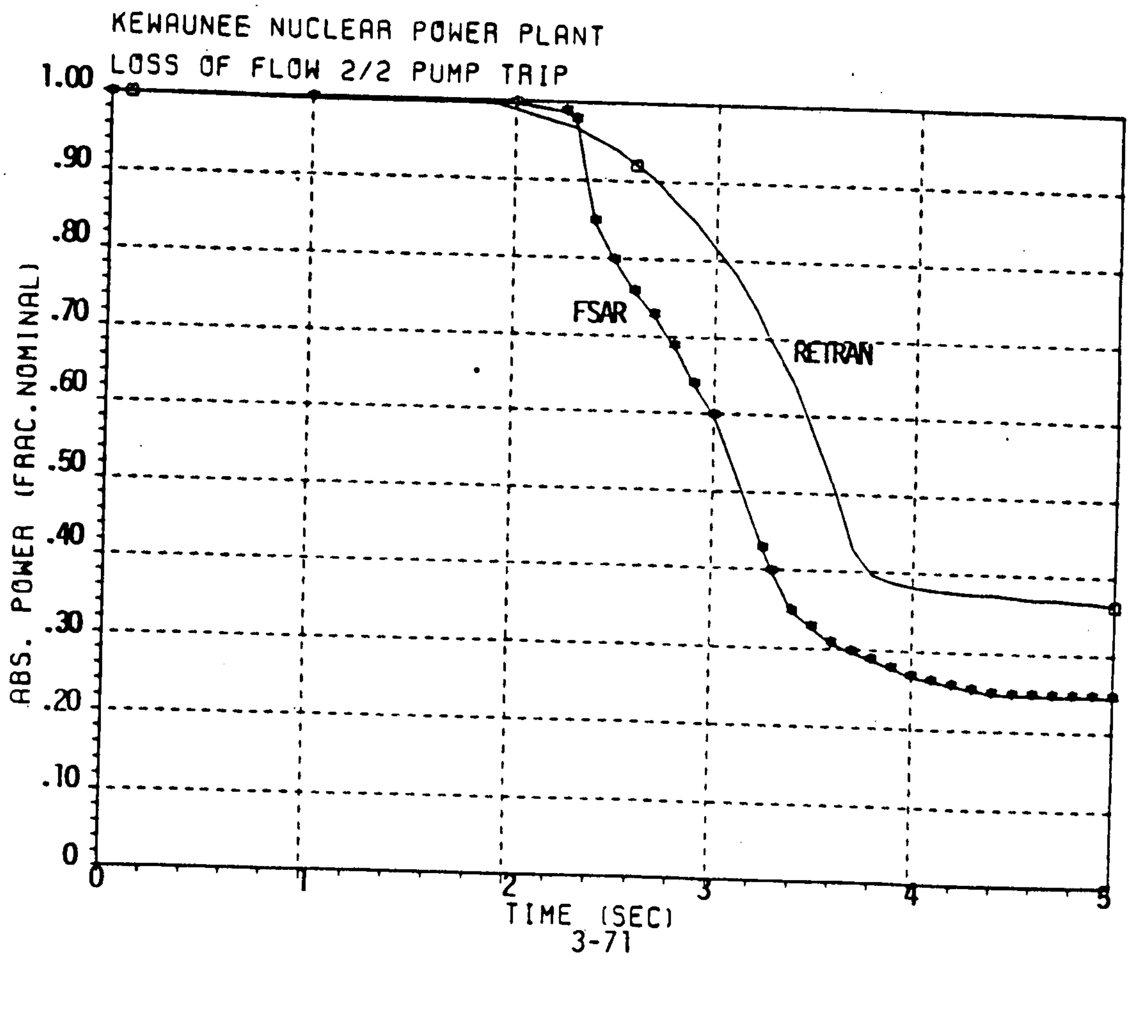
KEWAUNEE NUCLEAR POWER PLANT
LOSS OF FLOW 2/2 PUMP TRIP
1.10
1.00
.90
.80
.70
.60
.50
.40
.30
.20
.10
0
AVE HEAT FLUX (FRAC. NOMINAL)
DYNODE
FSAR
0
1
2
3
4
5
6
TIME (SEC)



KEWAUNEE NUCLEAR POWER PLANT
LOSS OF FLOW 2/2 PUMP TRIP
1.10
1.00
.90
.80
.70
.60
.50
.48
.30
.20
.10
0
AVE HEAT FLUX (FRAC. NOMINAL)
RETRAN
FSAR
0
1
2
3
4
5
6
TIME (SEC)

FIGURE 3.11-7


KEWAUNEE NUCLEAR POWER PLANT
LOSS OF FLOW 2/2 PUMP TRIP
ABS. POWER (FRAC.NOMINAL)
TIME (SEC)
FSAR
DYNODE
1.00
.90
.80
.70
.60
.50
.40
.30
.20
.10
0
1
2
3
4
5


KEWAUNEE NUCLEAR POWER PLANT
LOSS OF FLOW 2/2 PUMP TRIP
ABS. POWER (FRAC.NOMINAL)
TIME (SEC)
FSAR
RETRAN
1.00
.90
.80
.70
.60
.50
.40
.30
.20
.10
0
1
2
3
4
5

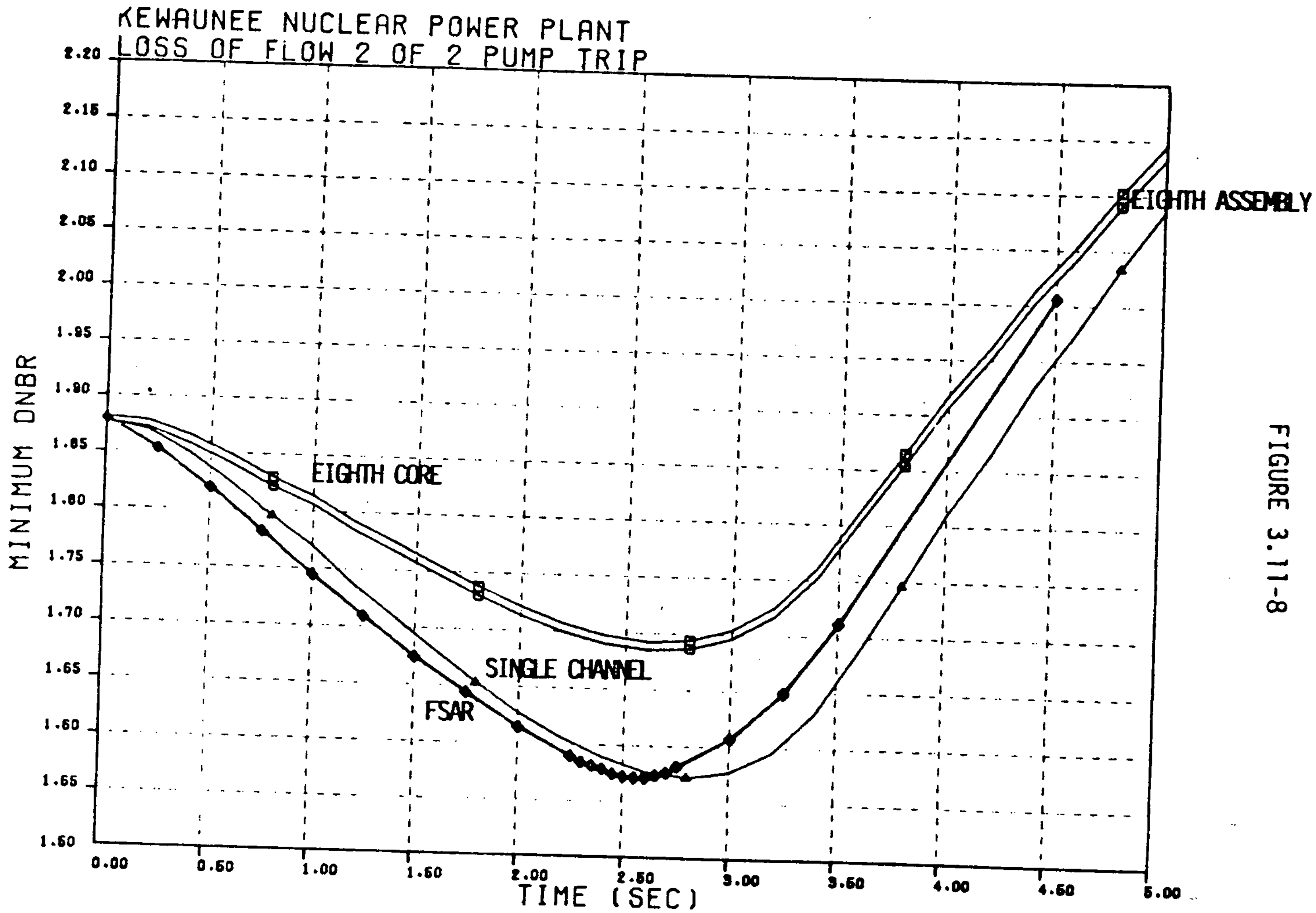

KEWAUNEE NUCLEAR POWER PLANT
LOSS OF FLOW 2 OF 2 PUMP TRIP
MINIMUM DNBR
TIME (SEC)
EIGHTH ASSEMBLY
EIGHTH CORE
SINGLE CHANNEL
FSAR
0.00
0.50
1.00
1.50
2.00
2.50
3.00
3.50
4.00
4.50
5.00

FIGURE 3.11-8

3.12 LOSS OF REACTOR COOLANT FLOW - LOCKED ROTOR

3.12.1 DEFINITION OF ACCIDENT

The accident postulated is the instantaneous seizure of the rotor of a single reactor coolant pump. Flow through the affected loop is rapidly reduced, leading to a reactor trip initiation due to low flow. The sudden decrease in core flow while the reactor is at power results in a degradation core heat transfer and departure from nucleate boiling in some of the fuel rods. The sudden degradation in steam generator heat transfer associated with the coolant flow transient causes an increase in reactor coolant temperature and a pressurizer insurge. The pressurizer safety valves are actuated and maintain the reactor coolant system pressure within acceptable limits. This accident is classified as a condition IV limiting fault.

3.12.2 ACCIDENT ANALYSIS

The analysis of the locked reactor coolant pump rotor is performed using a detailed model of the reactor coolant system thermal-hydraulics. The conservation of momentum and continuity equations for the coolant, coupled to a representation of the pump hydraulic characteristics, are solved to compute the system flow rates as a function of time. Reactor core neutron kinetics and transient heat transfer equations are coupled to the flow equations in order to compute the core heat flux and coolant temperatures in the reactor. A simulation of the pressurizer and steam generators is also included in the model. Separate models compute the thermal-hydraulic response of the coolant hot channel and fuel hot spot using conditions supplied by the NSSS model as input. These models compute heat flux, fuel and clad temperatures, and MDNBRs for a conservative evaluation of the extent of fuel damage which could occur during a locked rotor accident.

The initial conditions for the accident analysis assume the most adverse combination of power, core inlet temperature, and pressurizer pressure including allowances for steady state errors so that the initial margin to DNB is the minimum expected during steady state operation. For purposes of evaluating the reactor coolant system pressure transient, the initial pressure is assumed as the maximum expected during normal operation including allowances for instrumentation error and controller tolerances.

The power transient is analyzed using the least negative value of moderator reactivity coefficient calculated to occur during the cycle. For the sake of conservatism, a value of zero is assumed in the analysis even though only negative values are expected at normal operating conditions. The most negative Doppler reactivity coefficient is used in the analysis since this results in maximum hot spot heat flux at the time of minimum DNBR. Trip reactivity insertion characteristics are calculated by assuming the maximum time delay between a low flow signal and control rod motion. It is further assumed that the most reactive RCC assembly is stuck in a fully withdrawn position.

The acceptance criteria for the locked rotor analysis are as follows:

1. The maximum reactor coolant and main steam system pressures must not exceed 110% of the design values.

2. The number of fuel rods calculated to experience a DNBR of less than 1.3 should not exceed the number of fuel rods required to fail in order to yield doses due to released activity which will exceed the limits of 10CFR20.

3. The maximum clad temperature calculated to occur at the core hot spot must not exceed 2750 °F.

3.12.3 WPS SAFETY ANALYSIS RESULTS

WPS has analyzed the locked rotor accident for the Kewaunee Plant, using input consistent with the USAR.

The models described in Appendix A were used to analyze the locked rotor accident assuming the condition of a locked rotor in one coolant loop with two pumps initially running. The results of this analysis are compared to the corresponding results of Section 14.1-8 of the USAR.

In addition to DYNODE-P, the NSSS response to a locked rotor incident was predicted by RETRAN-02 for comparison. The transient forcing function inputs to VIPRE and TOODEE required for MDNBR and fuel temperature analyses, respectively, were derived from the DYNODE-P results.

In Figures 3.12-1 and 3.12-2, the transient response of core flow rate and RCS pressure are compared to Figures 14.1-32 and 14.1-34 of the USAR, respectively, for the locked rotor case. The WPS model predicts that the pressurizer safety valves are able to maintain the RCS pressure at about 2500 psia, while the USAR results indicate a rise in pressure beyond the safety valve set point, reaching a maximum of 2737 psia.

In Figures 3.12-3 and 3.12-4, the transient DNBRs during the locked rotor accident are compared to Figure 14.1-35 of the USAR for peak rod powers ($F_{\Delta H}$) of 1.58 and 1.420 respectively. The WPS MDNBR predictions are conservative with respect to the USAR results. Additional sensitivity analyses of the minimum DNBR to the initial $F_{\Delta H}$ have shown that the minimum DNBR is 1.3 when the initial $F_{\Delta H}$ is equal to 1.33. DNBR analyses with single channel and multi-channel models demonstrate that the most conservative MDNBR response is calculated by the single channel model.

Figure 3.12-5 compares the transient clad temperature response at the hot spot for the locked rotor accident to the corresponding results of Figure 14.1-37 of the USAR. The WPS model predicts a maximum of 1659 °F which compares to the USAR analysis maximum of 1680 °F.

WPS has performed a study of the NSSS response which demonstrates some sensitivity of the RCS peak pressure to assumptions related to the steam generator heat transfer characteristics. A difference in peak RCS pressure was found to be 70 psia between assuming normal heat transfer correlations from the tube side to the shell side and assuming (conservatively) that the heat transfer degrades with the RCS flow decrease according to $h \alpha w^{0.8}$. In all cases, the safety valve capacity was found to be sufficient to maintain the RCS pressure near the safety valve setpoint.

A sensitivity study was also performed for the hot spot cladding temperature response which demonstrated that the peak cladding temperature is sensitive to the fuel rod surface heat transfer coefficient. Typically, a 25 Btu/hr-ft °F change in the heat transfer coefficient produced a 50 °F change in peak cladding temperature.

3.12.4 CYCLE SPECIFIC PHYSICS CALCULATIONS

a. DOPPLER TEMPERATURE COEFFICIENT, α_D

Calculations of α_D are performed in accordance with the general procedures described in Section 2. Cycle specific calculations for this accident are performed at BOC and EOC to determine the most negative value at full power conditions.

b. MODERATOR TEMPERATURE COEFFICIENT, α_M

Calculations of α_M are performed in accordance with the general procedures described in Section 2. Cycle specific calculations are performed at BOC to determine the least negative value at the full power condition. The model bias, B_M, is included in the calculations.

c. <u>SCRAM REACTIVITY CURVE</u>, $\Delta\rho_{SCRAM}(t)$

Calculations of the scram reactivity curve are performed in accordance with the general procedures described in Section 2. Cycle specific calculations for this accident are performed at BOC and EOC for the full power condition. A conservatively slow scram curve is generated by making the following assumptions:

1. The integral of the scram curve is based on an initial rod position at or below the full power insertion limits.

2. The shape of the scram curve is based on an initial rod position of full out. This provides the longest possible delay to significant reactivity insertion.

3. The xenon distribution is that which causes the minimum shutdown margin.

4. Instantaneous redistribution of flux is assumed to occur during the rod insertion.

d. <u>EFFECTIVE DELAYED NEUTRON FRACTION</u>, β_{eff}

Calculations of β_{eff} are performed in accordance with the general procedures described in Section 2. Cycle specific calculations are performed at BOC and EOC at full power conditions.

e. <u>FUEL PIN CENSUS</u>, $F_{\Delta H}$

Calculation of the number of fuel rods (pin census) versus $F_{\Delta H}$ is performed in accordance with the general procedures described in Section 2. The calculations determine the number of fuel pins which exceed the limiting value of $F_{\Delta H}$ and are expected to experience DNB ratios less than 1.3. Cycle specific pin census curves are determined at the full power conditions for both BOC and EOC with the control rods at or above the power dependent insertion limits.

f. <u>NUCLEAR HEAT FLUX HOT CHANNEL FACTOR</u>, F_Q

The maximum core F_Qs are assumed to remain within the current limits as defined in the Technical Specifications for allowable combinations of axial offset and power level. The safety analysis value for FQ bounds the current Technical Specification limit.

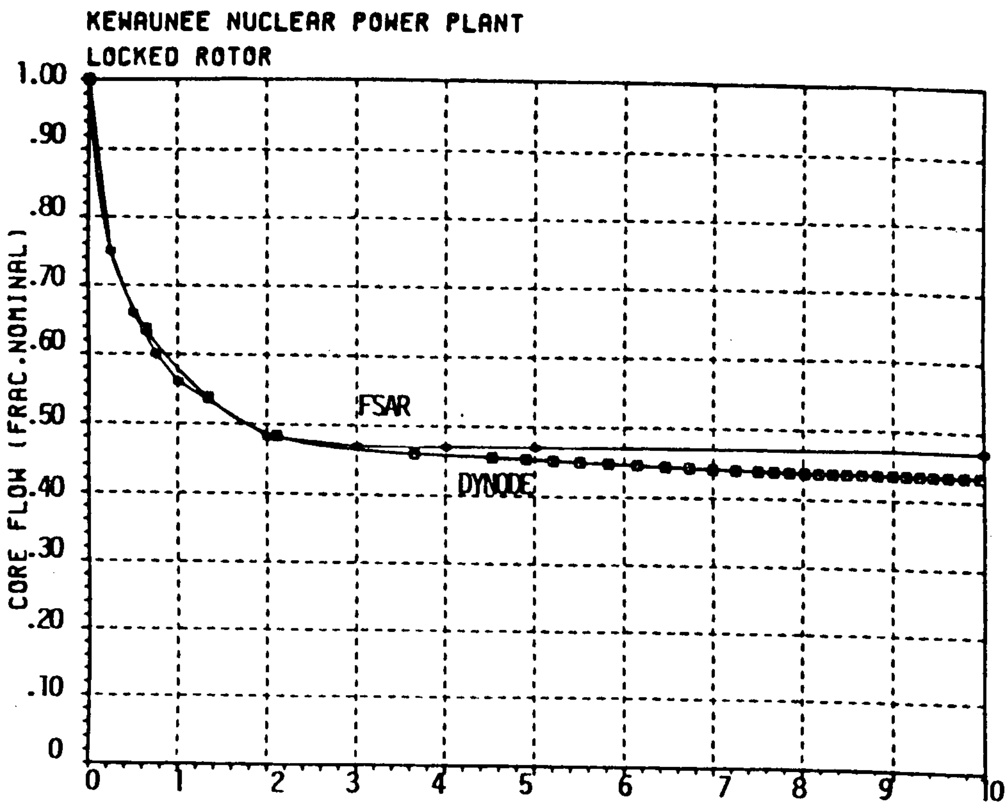
g. FUEL TEMPERATURE, T_f

Fuel temperature is calculated at conditions to maximize fuel temperature as a function of linear heat generation rate. The maximum cycle specific linear heat generation rate is used to derive the maximum cycle specific fuel temperature.

3.12.5 RELOAD SAFETY EVALUATION

Each of the physics parameters calculated is adjusted to include the model reliability factors, RF_i (1). These adjusted values are then compared to the bounding values assumed in the safety analysis. The cycle specific parameters are acceptable with regard to the locked rotor accident if the following inequalities are met:

CYCLE SPECIFIC PARAMETERS		SAFETY ANALYSIS PARAMETERS
a. $\alpha D*(1+RF_D)$	$\geq$	α_D (most negative bounding value if $\alpha M \leq D$)
b. $\alpha M+RF_M+B_M$	$\leq$	α_M (least negative bounding value)
c. $\lvert \Delta\rho_{SCRAM}(t)*(1-RF_{RODS}) \rvert$	$\geq$	$\lvert \Delta\rho_{SCRAM}(t) \rvert$ (bounding)
d. $\beta_{eff}*(1-RF_{\beta})$	$\geq$	β_{eff} (bounding value)
e. No. of fuel pins above $F_{\Delta H}(DNBR=1.3)$	$\leq$	40%
f. $FQ * (1+RFF_Q) * (1+T)$	$\leq$	FQ (bounding value Locked Rotor)
g. $T_f * (1+RFT_f)$	$\leq$	T_f (bounding value Locked Rotor)


KEWAUNEE NUCLEAR POWER PLANT
LOCKED ROTOR
CORE FLOW (FRAC.NOMINAL)
1.00
.90
.80
.70
.60
.50
.40
.30
.20
.10
0
0
1
2
3
4
5
6
7
8
9
10
FSAR
DYNODE

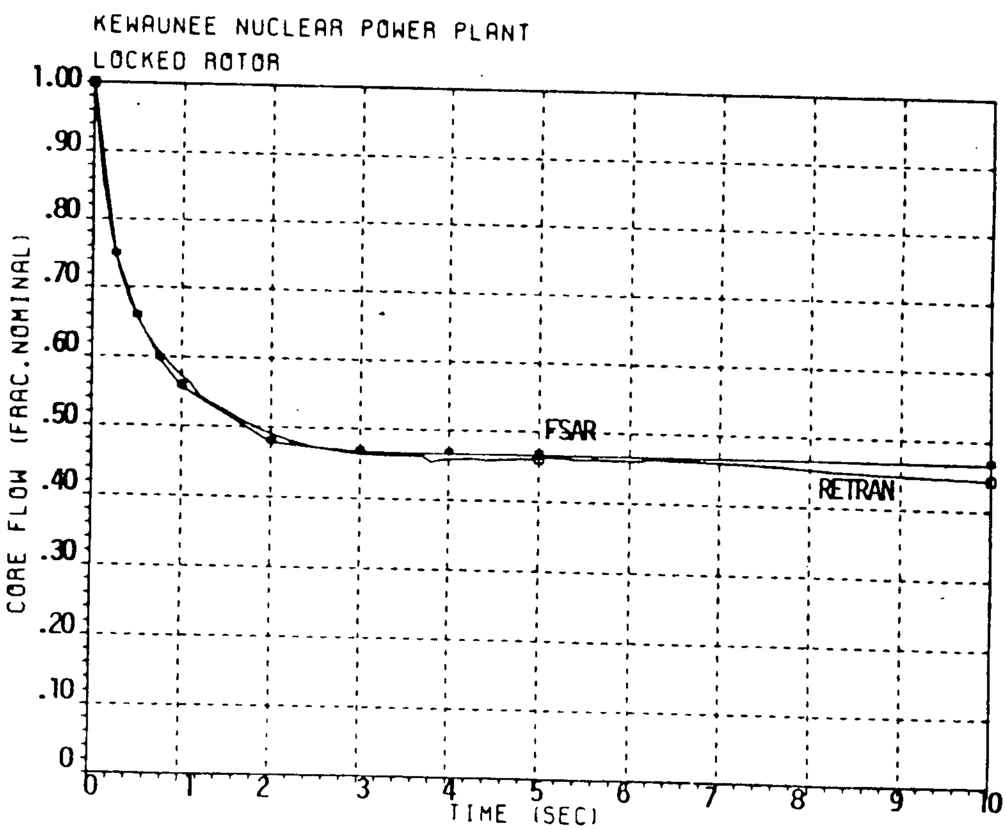
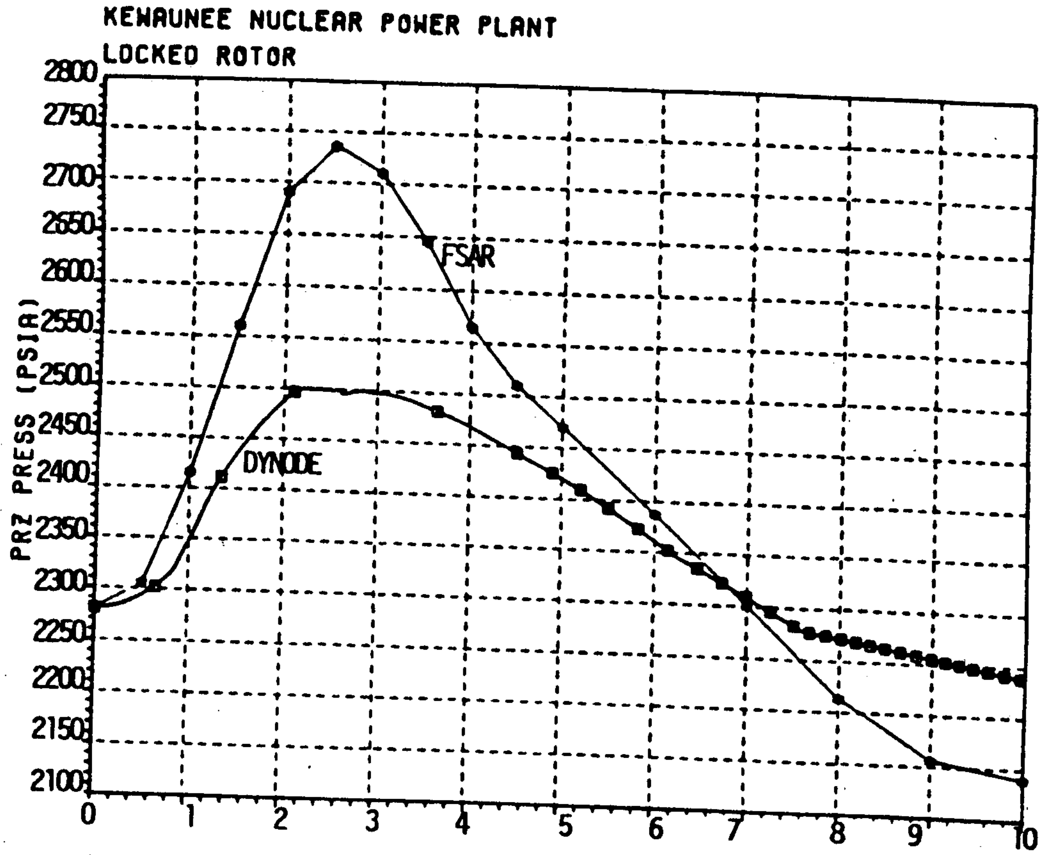
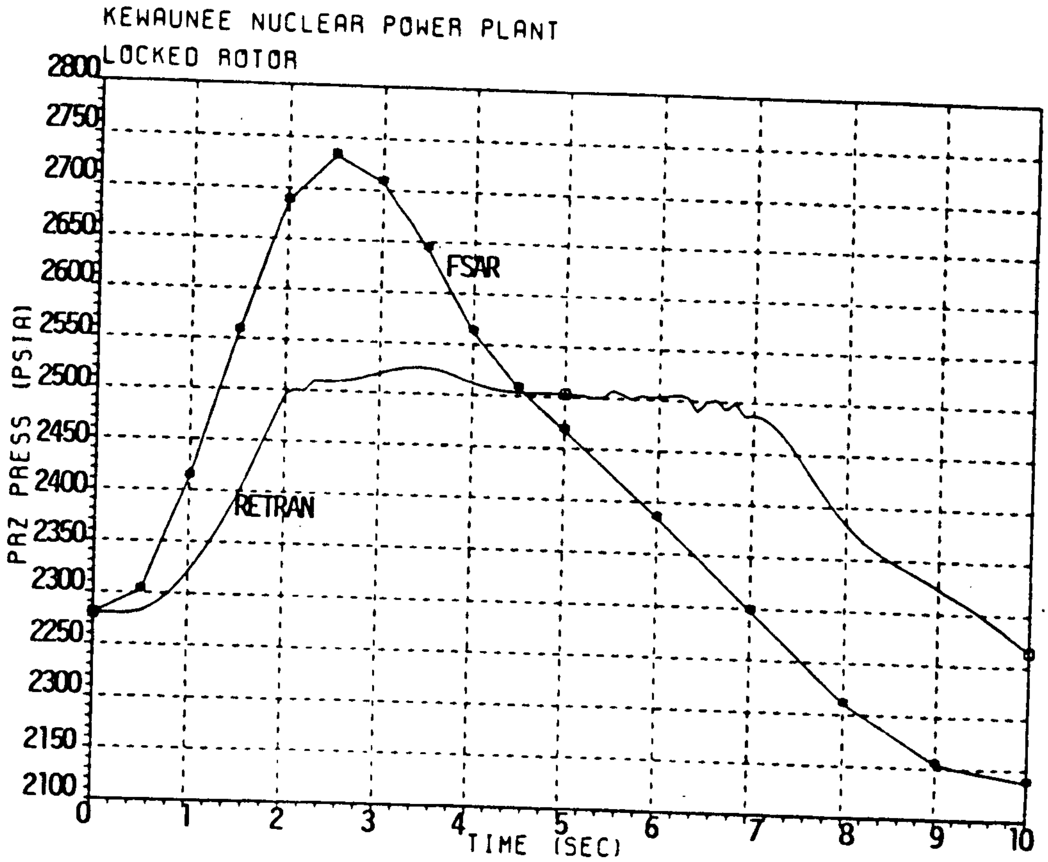

KEWAUNEE NUCLEAR POWER PLANT
LOCKED ROTOR
CORE FLOW (FRAC.NOMINAL)
1.00
.90
.80
.70
.60
.50
.40
.30
.20
.10
0
0
1
2
3
4
5
6
7
8
9
10
TIME (SEC)
FSAR
RETRAN

FIGURE 3.12-2


KEWAUNEE NUCLEAR POWER PLANT
LOCKED ROTOR
PRZ PRESS (PSIA)
2800
2750
2700
2650
2600
2550
2500
2450
2400
2350
2300
2250
2200
2150
2100
0
1
2
3
4
5
6
7
8
9
10
FSAR
DYNODE


KEWAUNEE NUCLEAR POWER PLANT
LOCKED ROTOR
PRZ PRESS (PSIA)
2800
2750
2700
2650
2600
2550
2500
2450
2400
2350
2300
2250
2300
2150
2100
0
1
2
3
4
5
6
7
8
9
10
TIME (SEC)
FSAR
RETRAN

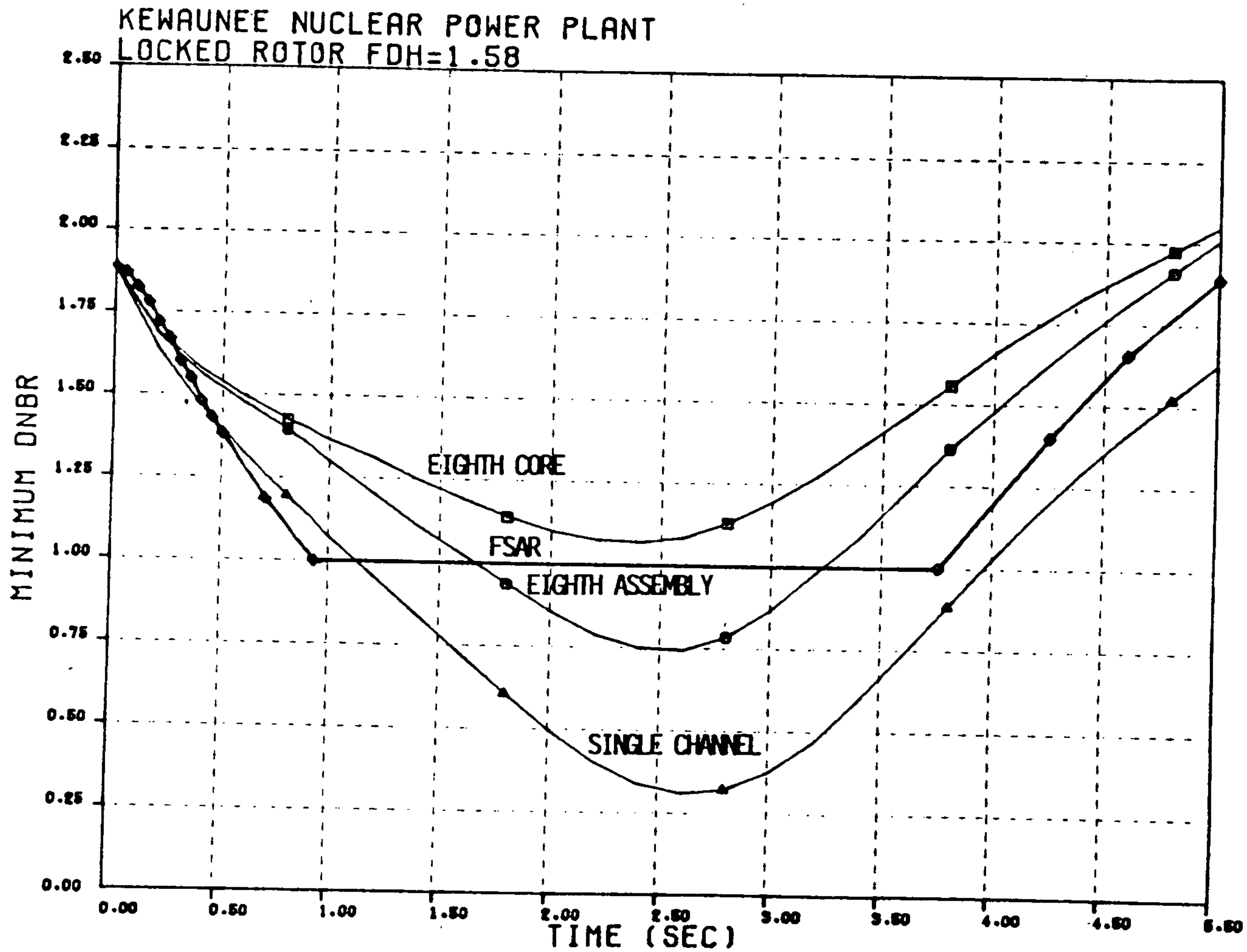

KEWAUNEE NUCLEAR POWER PLANT
LOCKED ROTOR FDH=1.58
MINIMUM DNBR
TIME (SEC)
2.50
2.25
2.00
1.75
1.50
1.25
1.00
0.75
0.50
0.25
0.00
0.00
0.50
1.00
1.50
2.00
2.50
3.00
3.50
4.00
4.50
5.50
EIGHTH CORE
FSAR
EIGHTH ASSEMBLY
SINGLE CHANNEL

FIGURE 3.12-3

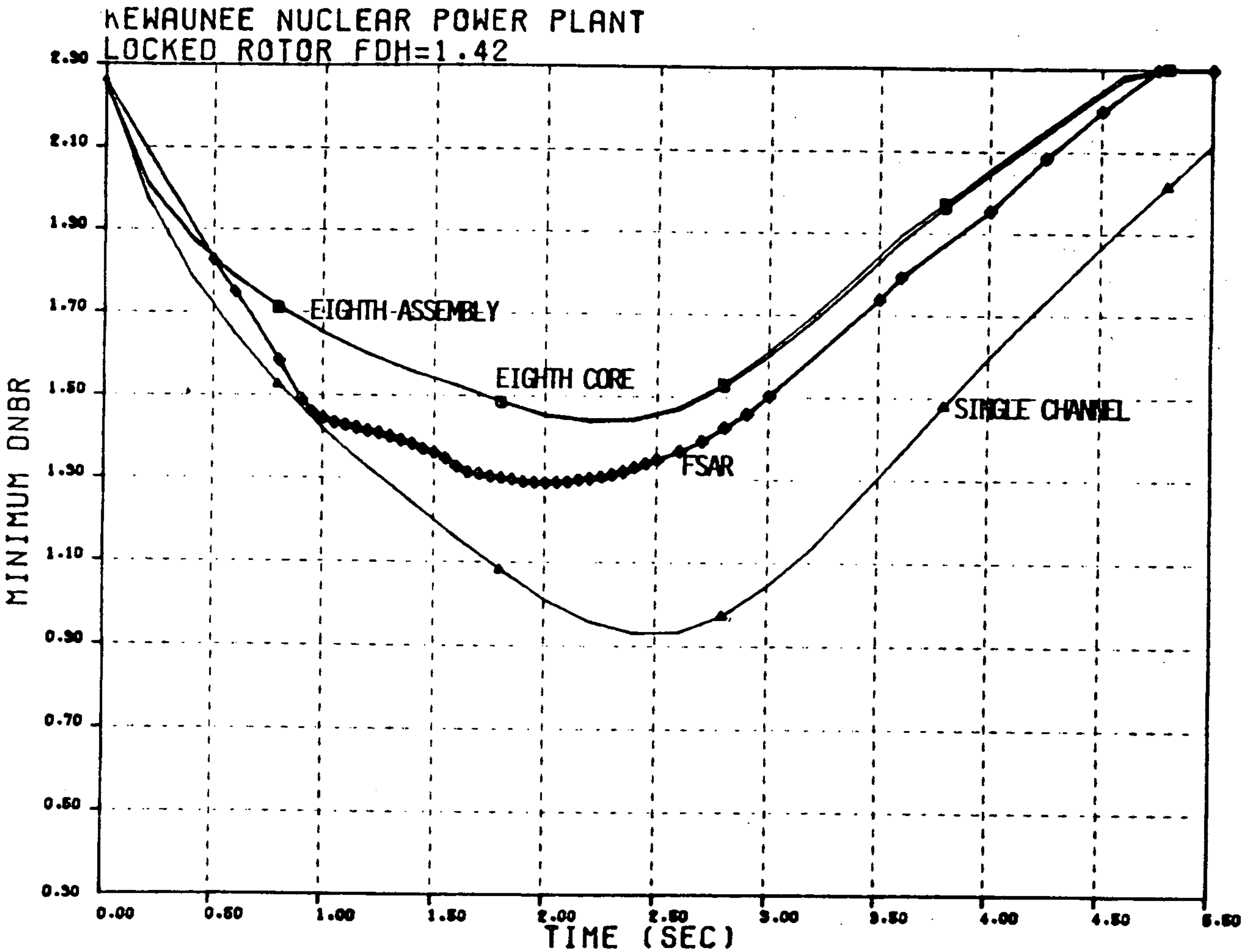

KEWAUNEE NUCLEAR POWER PLANT
LOCKED ROTOR FDH=1.42
MINIMUM DNBR
TIME (SEC)
EIGHTH-ASSEMBLY
EIGHTH CORE
FSAR
SINGLE CHANNEL
2.30
2.10
1.90
1.70
1.50
1.30
1.10
0.90
0.70
0.50
0.30
0.00
0.50
1.00
1.50
2.00
2.50
3.00
3.50
4.00
4.50
5.50

FIGURE 3.12-4

LOCKED ROTOR

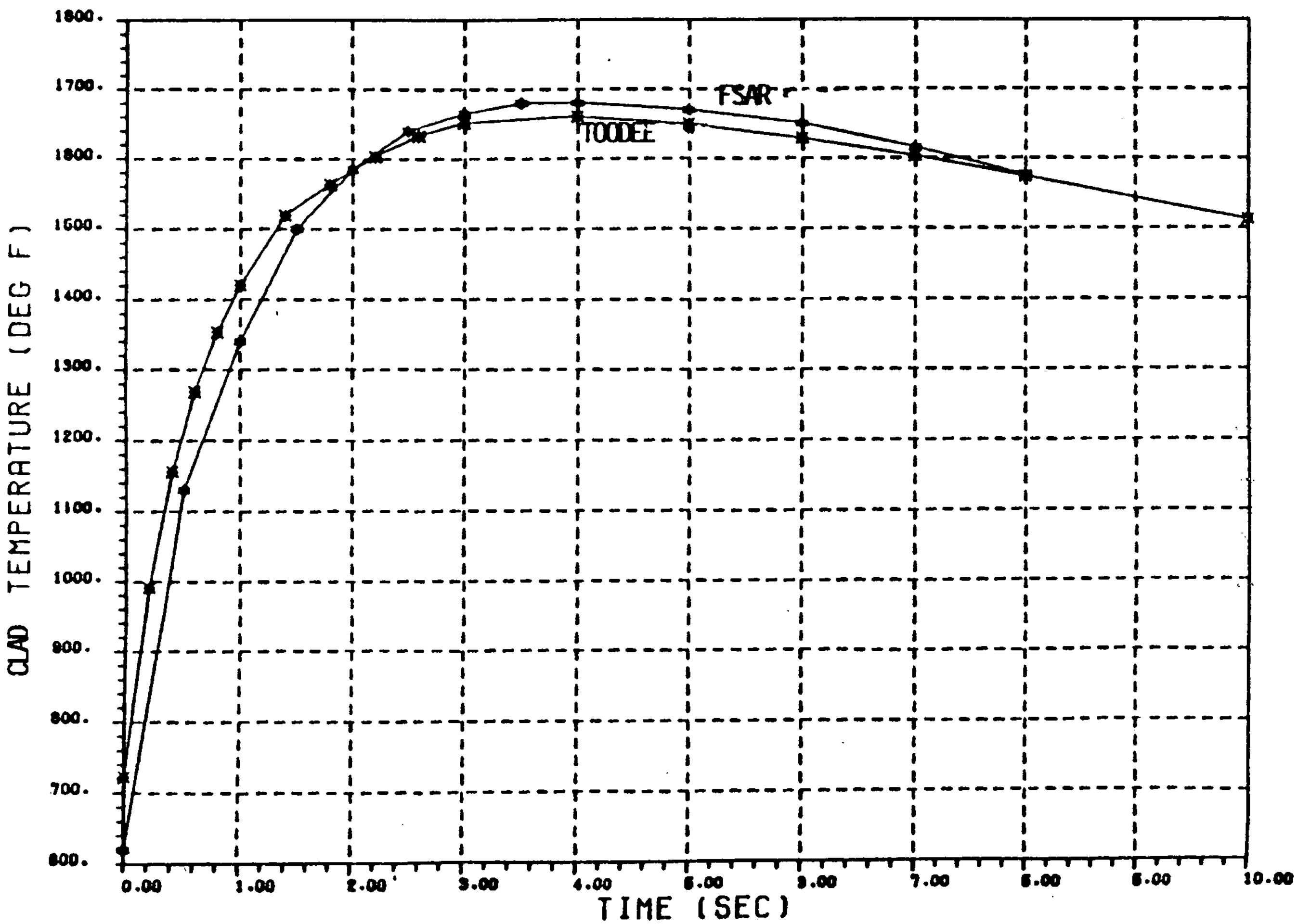


FIGURE 3.12-5

3.13 FUEL HANDLING ACCIDENT

3.13.1 DEFINITION OF ACCIDENT

The accident considered is the sudden release of the gaseous fission products held in the plenum between the pellets and cladding of one fuel assembly. The activity associated with this accident would be released either inside the Containment Building or the Auxiliary Building. A high radiation level alarm in the Containment Building would close the purging supply and exhaust ducts. A high radiation level on the Auxiliary Building vent monitor would automatically activate the special ventilation system with subsequent absolute and charcoal filtration. In calculating the offsite exposure from the accident, however, it is assumed that the activity is discharged to the atmosphere at ground level from the Auxiliary Building since this maximizes the offsite doses.

3.13.2 ACCIDENT ANALYSIS

The gap activity is calculated based on fission gas buildup in the fuel and subsequent diffusion to the fuel rod gap at rates dependent upon the operating temperature. The calculation assumes that the assembly with the maximum gap activity is the one which is damaged. Only that fraction of fission gases which has diffused into the gap and plenum regions of the fuel pin would be available for immediate release. This fraction is calculated based on a conservative evaluation of the temperature and power distribution in the highest powered assembly in the last six weeks prior to shutdown. This activity is further reduced by decay during the 100 hours elapsing after shutdown before removal of the vessel head.

The activity present in the fuel rod gaps consists predominately of halogens and noble gases. Decontamination factors are applied to account for halogen depletion by the pool water; all the noble gas inventory is assumed to escape from the pool water surface. Dispersal of the activity escaping the Auxiliary Building is calculated using the Gaussian plume dispersion formula, taking credit for building wake dilution. Using conservative radiological formulae, the activity concentrations at the site exclusion boundary are converted to integrated whole body and thyroid doses. These doses are then compared to the acceptance criteria set forth in 10CFR100.

3.13.3 WPS SAFETY ANALYSIS RESULTS

WPS has not analyzed this accident.

3.13.4 CYCLE SPECIFIC PHYSICS CALCULATIONS

The hot channel factor, F_Q is calculated for equilibrium hot full power conditions with the rods at or below the full power insertion limits. This value is determined for core exposures ranging from 1.5 MWD/MTU before EOC to EOC. Calculations of F_Q are performed in accordance with the procedure described in Section 2.

3.13.5 RELOAD SAFETY EVALUATIONS

The F_Q calculated above is conservatively adjusted to include the reliability factor, RFF_Q(1). This value is then compared to the value assumed in the accident analysis. The comparison is acceptable if the following inequality is satisfied:

CYCLE SPECIFIC PARAMETERS		SAFETY ANALYSIS PARAMETERS
$F_Q*(1+RFF_Q)*(1+T)$	$\leq$	F_Q (maximum bounding for this accident)

3.14 MAIN STEAM LINE BREAK

3.14.1 ACCIDENT DESCRIPTION

The accident considered here is the complete severance of a pipe inside containment at the exit of the steam generator with the plant initially at no load conditions and both reactor coolant pumps running. The resulting uncontrolled steam release causes a rapid reduction in reactor coolant temperature and pressure as the secondary side is depressurized. If the most reactive RCC assembly is assumed stuck in its fully withdrawn position, there is a possibility that the core will become critical and return to power due to the negative moderator coefficient. A return to power is potentially a problem mainly because of the high hot channel factors which exist with a stuck RCC assembly. The core is ultimately restored to a subcritical condition by boric acid injection via the Emergency Core Cooling System. The zero power case is considered because the stored energy of the system is at a minimum and steam generator secondary inventory is at a maximum under these conditions, thus increasing the severity of the transient. Similarly, the case with both reactor coolant pumps running is analyzed because this assumption maximizes the cooldown rate of the reactor coolant system.

3.14.2 ACCIDENT ANALYSIS

The analysis of the steam line break accident is performed using a detailed, multi-loop model of the core, reactor coolant system and pressurizer, steam generators, and main steam system. The steam flow through the severed steam line is calculated using a critical flow model. Conservation equations for the steam generator shell side mass and energy inventory are solved to predict the temperatures and pressures existing throughout the transient. Heat transfer from the reactor coolant system to the steam generators is calculated based on instantaneous fluid conditions and empirical correlations. The analytical model includes a representation of the reactor vessel upper head volume in order to predict the transient response of the reactor coolant pressure subsequent to draining the pressurizer. A simulation of the safety injection system and boron injection allows calculation of the core coolant boron concentration and its influence on core neutron kinetics. The representation of core moderator density reactivity effects must include allowances for the large change in density which the coolant undergoes as the system temperature falls. A detailed thermal-hydraulic model of the hot channel is coupled to the system simulation and provides a calculation of the departure from nucleate boiling ratio during the transient.

The core neutronics parameters input to the model are evaluated at the core conditions which yield the most limiting values of moderator and Doppler reactivity coefficients, spatial power distribution, and shutdown margin. This is normally the EOC condition, since the moderator temperature coefficient is most negative and the shutdown margin is minimum. Trip reactivity insertion characteristics need not be input to the analysis, since the reactor is assumed to be initially shutdown with minimum shutdown margin. The moderator reactivity coefficient is also calculated assuming the most reactive rod is stuck in its fully withdrawn position, and includes the local reactivity feedback from the high neutron flux in the vicinity of the stuck rod.

An important parameter which is input to the model is the boron concentration in the Boric Acid Storage Tank. The value used in the WPS model corresponds to the minimum value permitted by the Technical Specifications.

The acceptance criteria for the main steam line break accident are that reactor coolant and main steam system pressures do not exceed 110% of design pressure and that the minimum DNBR be not less than 1.3.

3.14.3 WPS SAFETY ANALYSIS RESULTS

WPS has analyzed the main steam line break using input consistent with the Kewaunee USAR (4).

a. A break at the exit of the steam generator with safety injection and offsite power assumed available.

b. A break downstream of the flow measuring nozzle with safety injection and offsite power assumed available.

c. A break equivalent to 247 lbm/sec at 1100 psia with safety injection and offsite power assumed available.

The results of case a are compared to those reported in USAR Figure 14.2-6 in Figures 3.14-1 to 3.14-5 of this report. The results of case b are compared to those reported in USAR Figure 14.2-5 in Figures 3.14-6 to 3.14-10 of this report. The results of case c are compared to those reported in USAR Figure 14.2-9 in Figures 3.14.11 to 3.14-13 of this report.

The WPS model results, in general, show good agreement with those reported in the USAR.

The hot channel model described in Appendix C was used to analyze the hot channel DNBR at the five steady state conditions listed in Table 3.14-1. The minimum DNBR calculated using the WPS model was 1.39, compared to Section 14.2-5 of the USAR which states that the DNBR was greater than 1.37 for all cases.

3.14.4 CYCLE SPECIFIC PHYSICS CALCULATIONS

a. MODERATOR TEMPERATURE COEFFICIENT, α_M

Calculations of α_M are performed in accordance with the general procedures described in Section 2. Specific calculations are performed at both BOC and EOC with hot full power boron concentrations in order to obtain a maximum negative coefficient. Additionally, calculations of α_M are made as a function of core average temperature with all rods in, except for the most reactive RCCA, at 1000 psia. Using this functional value of $\alpha_M(T)$, k_{eff} is calculated versus temperature assuming an initial 2% shutdown condition at 547 °F.

b. SHUTDOWN MARGIN, SDM

The shutdown margin is calculated consistent with the description given in Section 2, and is calculated for both BOC and EOC, HZP and HFP conditions.

c. NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR, $F_{\Delta H}$

The maximum $F_{\Delta H}$ is calculated consistent with the description given in Section 2 and is calculated for reactor conditions expected during the cooldown.

3.14.5 RELOAD SAFETY EVALUATION

Each parameter calculated above is conservatively adjusted to include the model reliability factors, RF_i, and biases (1). These results are then compared to the bounding values assumed in the safety analysis. For k_{eff} versus temperature during cooldown, the reliability factors are applied to the calculation of the moderator temperature coefficient in the determination of k_{eff}.

Uncertainties for the rod worth, moderator temperature defect, and Doppler temperature defect are applied to the shutdown margin (SDM) as discussed in Section 2.4. The cycle specific parameters are acceptable if the following inequalities are met:

CYCLE SPECIFIC PARAMETERS		SAFETY ANALYSIS PARAMETERS
$k_{eff}(T)$	$\leq$	$k_{eff}(T)$ (bounding)
SDM	$\geq$	SDM (bounding)
$F_{\Delta H}*(1+RFF_{\Delta H})*(1+T)$	$\leq$	$F_{\Delta H}$ (bounding value steam line break)
$\alpha_D * (1-RF_D)$	$\leq$	α_D (least negative bounding value)
$\alpha_B * (1-RF_B)$	$\leq$	α_B (least negative bounding value)

TABLE 3.14-1

HOT CHANNEL ANALYSES FOR STEAM LINE BREAK ACCIDENT

Case	Inlet Temperature	RCS Pressure	Core Average* Heat Flux	$F\Delta H$	WPS Model MDNBR	USAR MDNBR
1	456.8 °F	962.8 psia	6.10%	8.8	7.644	>1.37
2	425.6 °F	863.3 psia	21.76%	7.2	2.237	>1.37
3	411.2 °F	800.0 psia**	35.54%	6.25	1.390	>1.37
4	392.4 °F	670.4 psia	20.41%	7.65	2.388	>1.37
5	328.4 °F	505.3 psia	3.14%	10.35	>10.000	>1.37

*% of 1650 MWt

**Pressure taken from Figure 14.2-6 of the USAR.

FIGURE 3.14-1
KEWAUNEE NUCLEAR POWER PLANT
STEAM LINE BREAK

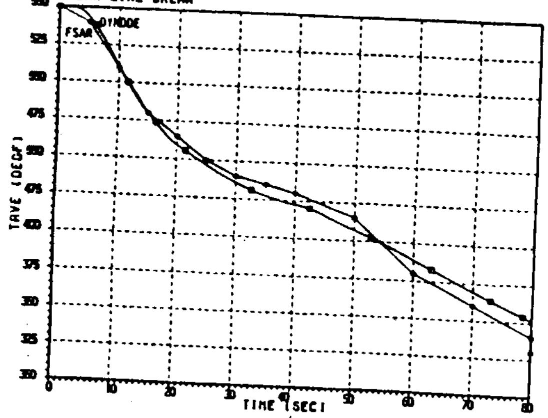


FIGURE 3.14-2
KEWAUNEE NUCLEAR POWER PLANT
STEAM LINE BREAK

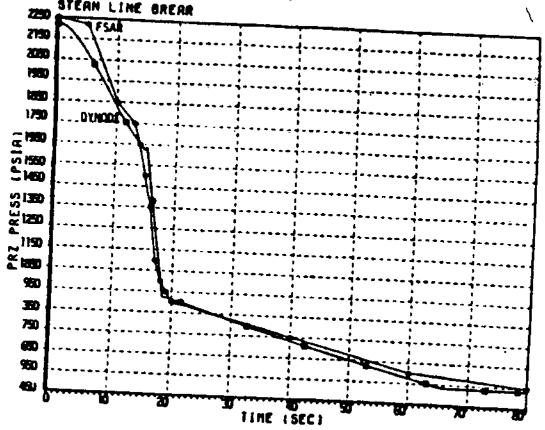


FIGURE 3.14-3
KEWAUNEE NUCLEAR POWER PLANT
STEAM LINE BREAK

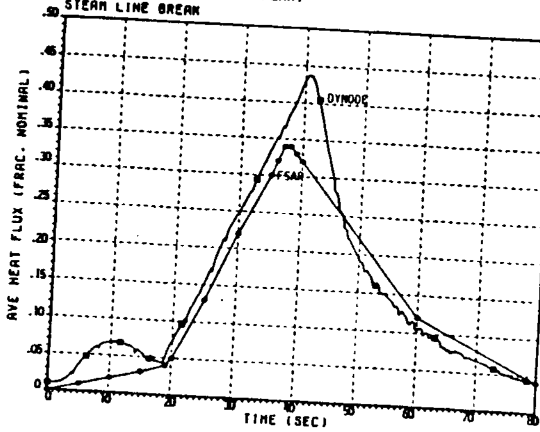


FIGURE 3.14-4
KEWAUNEE NUCLEAR POWER PLANT
STEAM LINE BREAK

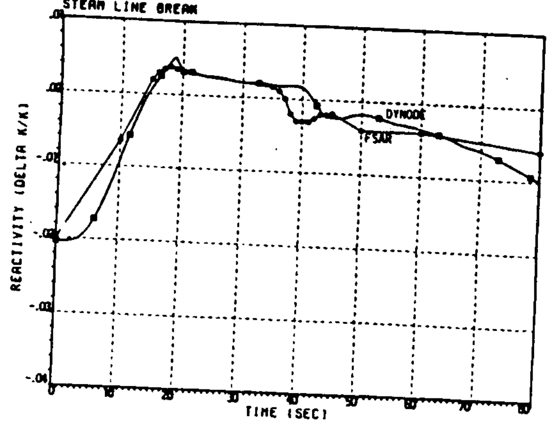


FIGURE 3.14-5
KEWAUNEE NUCLEAR POWER PLANT
STEAM LINE BREAK

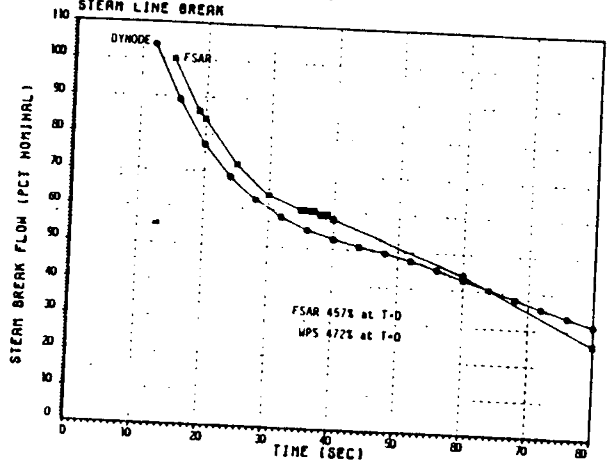


FIGURE 3.14-6

KEWAUNEE NUCLEAR POWER PLANT
STEAM LINE BREAK DOWNSTREAM FLOW RESTRCT

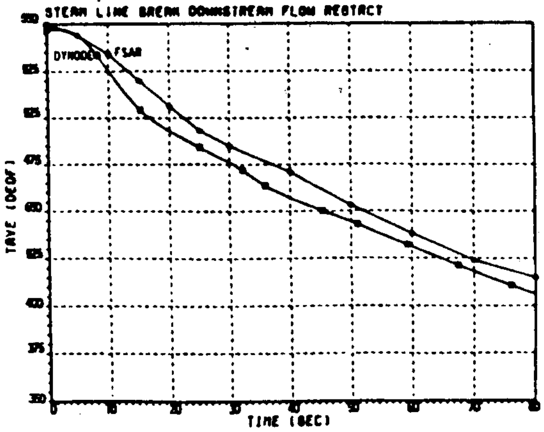


FIGURE 3.14-7

KEWAUNEE NUCLEAR POWER PLANT
STEAM LINE BREAK DOWNSTREAM FLOW RESTRCT

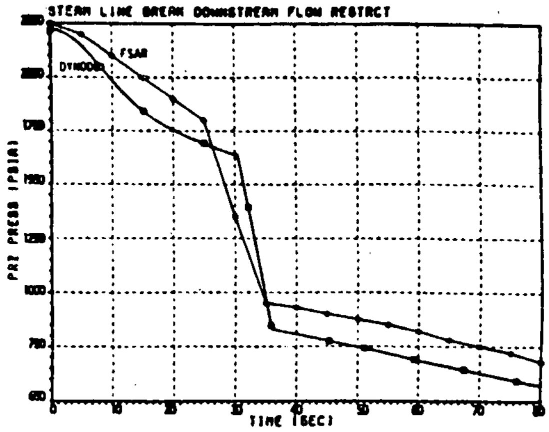


FIGURE 3.14-8

KEWAUNEE NUCLEAR POWER PLANT
STEAM LINE BREAK DOWNSTREAM FLOW RESTRCT

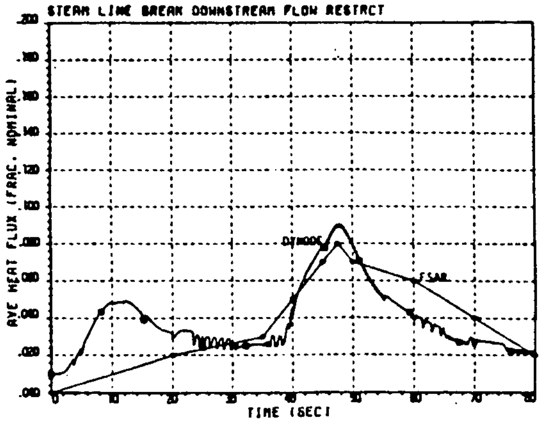


FIGURE 3.14-9

KEWAUNEE NUCLEAR POWER PLANT
STEAM LINE BREAK DOWNSTREAM FLOW RESTRCT

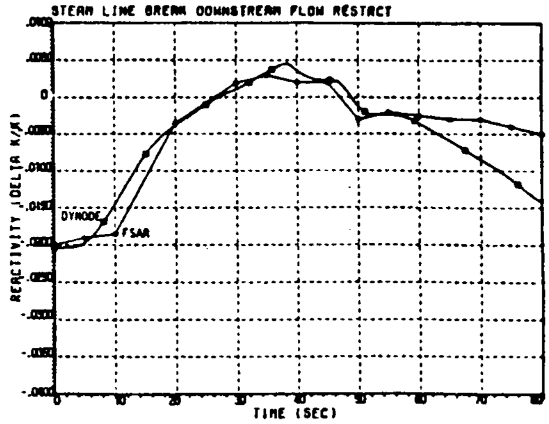


FIGURE 3.14-10

KEWAUNEE NUCLEAR POWER PLANT
STEAM LINE BREAK DOWNSTREAM FLOW RESTRCT

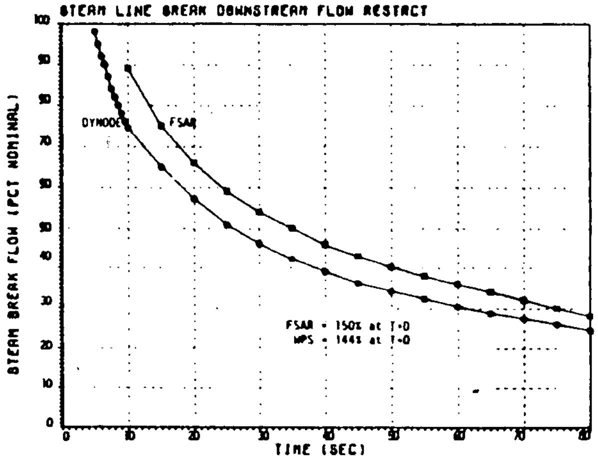


FIGURE 3.14-11

KEWAUNEE NUCLEAR POWER PLANT
STEAM LINE BREAK SPURIOUS OPENING OF SV

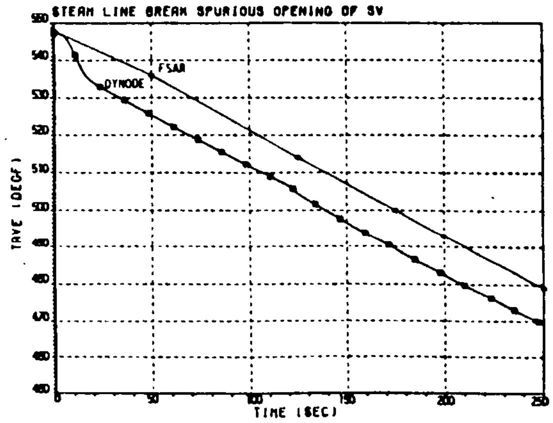


FIGURE 3.14-12

KEWAUNEE NUCLEAR POWER PLANT
STEAM LINE BREAK SPURIOUS OPENING OF SV

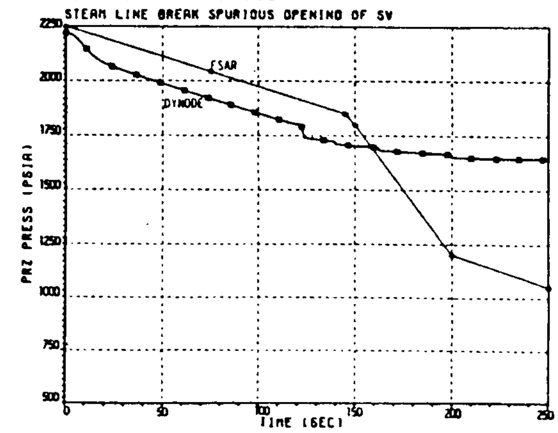
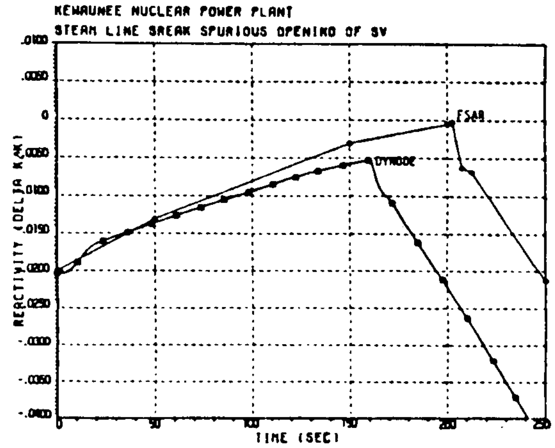


FIGURE 3.14-13

KEWAUNEE NUCLEAR POWER PLANT
STEAM LINE BREAK SPURIOUS OPENING OF SV



3.15 CONTROL ROD EJECTION

3.15.1 DEFINITION OF ACCIDENT

This accident is postulated to result from the unlikely failure of a control rod pressure housing followed by ejection of an RCC assembly by the reactor coolant system pressure. If a rod inserted in a high worth region of the core were to be ejected, the rapid reactivity insertion and unfavorable power distribution which would result might cause localized fuel rod damage.

3.15.2 ACCIDENT ANALYSIS

The analysis of the control rod ejection accident requires a model of the neutron kinetics coupled to models of the fuel and clad transient conduction and the thermal-hydraulics of the coolant channel. In practice, model sophistication has varied from point kinetics to three-dimensional spatial kinetics. When three-dimensional calculations are not employed, the reactivity feedbacks must be corrected using weighting factors to account for the spatial dimensions not included in the model. The thermal-hydraulic model used includes a multi-nodal radial model of fuel, gap, and clad conduction; and a multi-nodal axial model of the coolant channel. Since the calculations result in a maximum fuel enthalpies less than those corresponding to catastrophic fuel failures, the system pressure surge is calculated on the basis of conventional heat transfer from the fuel. The pressure surge model includes prompt heat generation in the coolant (so called "direct moderator heating"), fluid transport in the system, heat transfer in the steam generators, and the action of relief and safety valves. No credit is taken for pressure reduction caused by the assumed failure of the control rod pressure housing.

The maximum ejected rod worth is calculated with all control banks at their maximum permissible insertion for the power level of interest.

The moderator reactivity effect is included in the model by correlating reactivity with moderator density, thereby including effects of coolant temperature, pressure, and voiding. The Doppler reactivity effect is typically correlated as a function of either fuel temperature or power. The highest boron concentration corresponding to the initial reactor state is assumed in the calculation of moderator feedback. The largest temperature rise during the transient, and hence the largest reactivity effects, occurs in channels where the power is higher than average. This means that the reactivity feedback is larger than that predicted by a single average channel analysis. As a result, when a three-dimensional space-time kinetics calculation is not performed, weighting factors are applied as multipliers to the average channel Doppler feedback reactivity to account for spatial reactivity feedback effects. For the WPS model a one-dimensional kinetics model is used in which the axial dimension effects such as power distribution, scram insertion rate, and temperature distribution are accounted for.

The results of the accident analysis are relatively insensitive to β_{eff}, the effective delayed neutron fraction, except in those cases in which the ejected rod worth approaches or exceeds β_{eff}. In these cases, the minimum value of β_{eff} calculated for the assumed initial reactor state is used in the accident analysis.

The results are also relatively insensitive to ℓ^*, the prompt neutron lifetime, in the range of values normally encountered in commercial pressurized water reactors. Minimum values of ℓ^* are used in the accident analysis.

Control rod reactivity insertion during trip is obtained by combining a differential rod worth curve with a rod velocity curve, based on maximum design values for scram insertion times. The reactor trip delay time is calculated by combining the maximum time delays involved in the instrumental and actuation circuitry.

The acceptance criteria for the control rod ejection accident are as follows:

* The average hot spot fuel enthalpy must be less than 280 calories/gram.

* The maximum reactor coolant system pressure must be less than the pressure that will cause stresses to exceed the emergency condition stress limit; assumed to be 150% of design pressure.

* The maximum clad temperature calculated to occur at the core hot spot must not exceed 2750°F.

3.15.3 WPS SAFETY ANALYSIS RESULTS

WPS has analyzed the ejected rod accidents and compared the results to a Westinghouse Topical Report on rod ejection accidents analysis (13).

The models described in Appendix A were used to analyze the control rod ejection accident at the following four initial conditions:

* Zero Power Beginning of Life (ZPBOL)
* Full Power Beginning of Life (FPBOL)
* Zero Power End of Life (ZPEOL)
* Full Power End of Life (FPEOL)

The results of these calculations are compared to those reported in Chapter 4 of Reference 13 for equivalent cases (same initial power, core burnup, ejected rod worth, and transient peaking factor). Reference 13 provides documentation of generic results for Westinghouse Pressurized Water Reactors; consequently, those results are applicable to the Kewaunee Plant.

The core nuclear power response, energy release, and hot spot fuel temperatures are compared to the results of Figures 4.1 and 4.3 of Reference 13 in Figures 3.15-1 to 3.15-3 of this report for the ZPBOL case.

Similar comparisons to Figures 4.2 and 4.4 of Reference 13 for the FPBOL cases are presented in Figure 3.15-4 to 3.15-6 of this report.

Figures 3.15-7 to 3.15-10 of this report show the comparisons with the results of Figures 4.1 and 4.2 of Reference 13 for the core nuclear power and energy release for the ZPEOL and FPEOL cases, respectively.

Results of the comparison for the maximum fuel rod temperatures and enthalpies at the hot spot are given in Table 3.15.-1.

An energy balance was performed at the hot spot for the ZPEOL case out to the time at which maximum fuel temperature occurred. This energy balance verified that the WPS hot spot model results are consistent with the energy release.

A sensitivity study was performed which showed that the core average energy release is sensitive to the Doppler reactivity. Typically a decrease in Doppler reactivity of 30% produces a 25% increase in the core average energy release for zero power conditions.

A sensitivity study was also performed for the peak cladding temperature as a function of the fuel rod surface heat transfer coefficient. For zero power conditions, a change of 100 Btu/hr-ft^2-°F in the heat transfer coefficient produces about a 400°F change in peak cladding temperature.

3.15.4 CYCLE SPECIFIC PHYSICS CALCULATIONS

a. DOPPLER TEMPERATURE COEFFICIENT, α_D

The values of α_D are calculated in accordance with the general procedures described in Section 2. Full and zero power core conditions in rodded, unrodded, and ejected rod configurations are considered at BOC and EOC in order to determine the least negative value of α_D.

b. MODERATOR TEMPERATURE COEFFICIENT, α_M

Calculations of α_M are performed in accordance with the general procedures described in Section 2. Cycle specific values are computed at full and zero power, BOC and EOC core conditions to determine the least negative moderator temperature coefficients.

c. EFFECTIVE DELAYED NEUTRON FRACTION, β_{eff}

The value of β_{eff} is calculated in accordance with the general procedures given in Section 2. Cycle specific calculations are performed at BOC and EOC for both the full power and zero power conditions.

d. MAXIMUM EJECTED ROD WORTH, $\Delta\rho_{EJECT}$

Calculations of the ejected rod worth are performed with the nodal model in three dimensions. No credit is taken for either moderator or Doppler reactivity feedback mechanisms. All calculations are performed with the control rods at or below their power dependent insertion limits (PDIL). Cycle specific calculations are performed at BOC and EOC for both the full power and zero power conditions. The search for the highest worth ejected rod includes all rods initially inserted to the PDIL. The maximum worth of the ejected rod includes consideration of transient xenon conditions such as maximum positive or negative axial offsets.

e. SCRAM REACTIVITY CURVE, $\Delta\rho_{SCRAM}(t)$

Calculations of the scram reactivity curve are performed in accordance with the general procedures described in Section 2. Cycle specific calculations for this accident are performed at BOC and EOC for the full and zero power conditions. A conservatively slow scram curve is generated by making the following assumptions:

1. The integral of the scram curve is based on an initial rod position at or below the power insertion limits.

2. The shape of the scram curve is based on an initial rod position of full out. This provides the longest possible delay to significant reactivity insertion.

3. The xenon distribution is that which causes the minimum shutdown margin.

4. Instantaneous redistribution of flux is assumed to occur during the rod insertion.

f. HEAT FLUX HOT CHANNEL FACTOR, F_Q

F_Q is calculated for each of the cases investigated as described above for the determination of the maximum $\Delta\rho_{EJECT}$. The maximum value of F_Q does not necessarily correspond to the maximum value of $\Delta\rho_{EJECT}$. As described above, no calculations of F_Q for the ejected rod takes credit for the moderator or Doppler feedback mechanisms.

g. PROMPT NEUTRON LIFETIME, ℓ^*

The value of ℓ^* is calculated in accordance with the general procedures given in Section 2. Cycle specific calculations are performed at BOC and EOC.

3.15.5 RELOAD SAFETY EVALUATION

Each of the physics parameters calculated for this accident are adjusted to include the model reliability factors, RF_i, and biases (1). These adjusted values are the cycle specific parameters to be compared to the bounding values used in the safety analysis.

The cycle specific parameters are acceptable with regard to the ejected rod accident if the following inequalities are met:

CYCLE SPECIFIC PARAMETERS		SAFETY ANALYSIS PARAMETERS
a. $\alpha_D*(1-RF_D)$	$\leq$	α_D (least negative bounding value)
b. $\alpha_M+RF_M+B_M$	$\leq$	α_M (least negative bounding value)
c. $\beta_{eff}*(1-RF\beta)$	$\geq$	β_{eff} (minimum)
d. $\lvert\Delta\rho_{EJECT}*(1+RF_R)\rvert$	$\leq$	$\lvert\Delta\rho_{EJECT}\rvert$ (bounding)
e. $\lvert\Delta\rho_{SCRAM}(t)*(1-RF_R)\rvert$	$\geq$	$\lvert\Delta\rho_{SCRAM}(t)\rvert$ (bounding)
f. $F_Q*(1+RFF_Q)*(1+T)$	$\leq$	F_Q (bounding value rod ejection accident)
g. $\ell^* *(1-RF\ell^*)$	$\geq$	ℓ^*(minimum)

TABLE 3.15-1

COMPARISON OF ROD EJECTION

MAXIMUM FUEL ROD ENTHALPIES AND TEMPERATURES

CASE	MAXIMUM FUEL AVERAGE TEMP. (°F)		MAXIMUM FUEL AVERAGE ENTHALPY (Btu/lb)		MAXIMUM CLAD AVERAGE TEMP. (F°)	
	REFERENCE 13	WPS MODEL	REFERENCE 13	WPS MODEL	REFERENCE 13	WPS MODEL
ZPBOL	3529	3250	270.1	242.5	2614	2512
FPBOL	4218	4114	334.7	332.5	2605	2621
ZPEOL	3622	3116	278.7	228.6	2711	1908
FPEOL	4063	4008	319.8	318.6	2487	2487

FIGURE 3.15-1

KEWAUNEE NUCLEAR POWER PLANT

CONTROL ROD EJECTION HZP.BOL

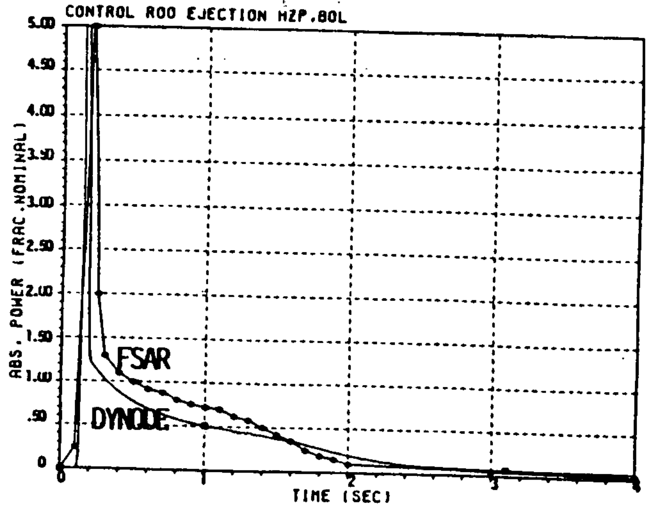


FIGURE 3.15-2

KEWAUNEE NUCLEAR POWER PLANT

CONTROL ROD EJECTION HZP.BOL

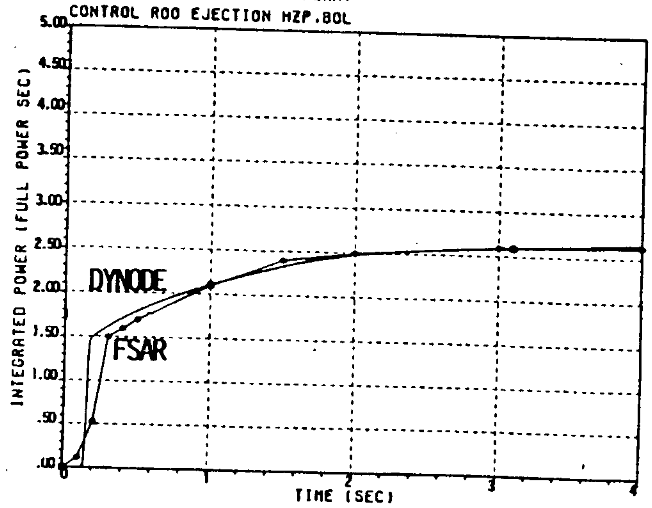


FIGURE 3.15-3

KEWAUNEE NUCLEAR POWER PLANT

CONTROL ROD EJECTION HZP BOL

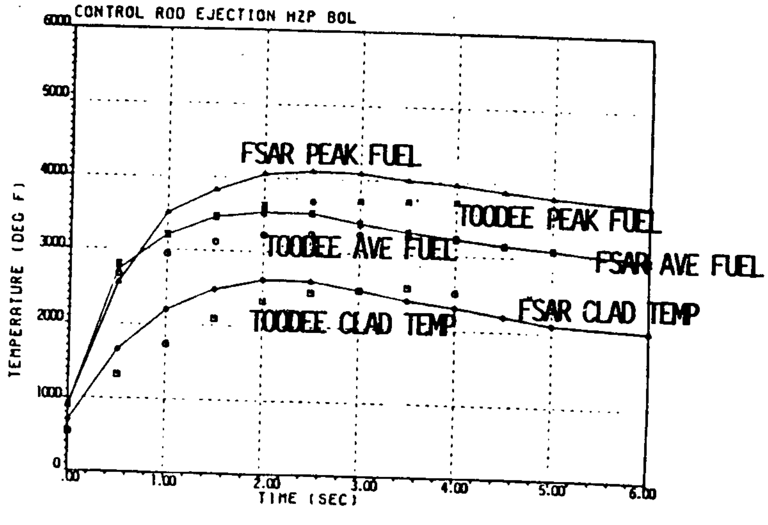


FIGURE 3.15-4

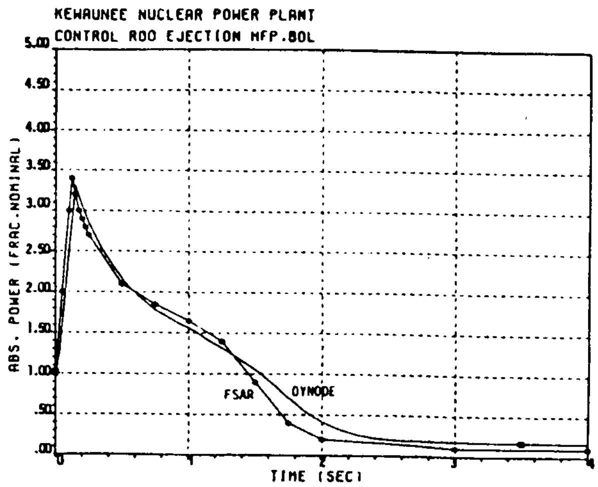


FIGURE 3.15-5

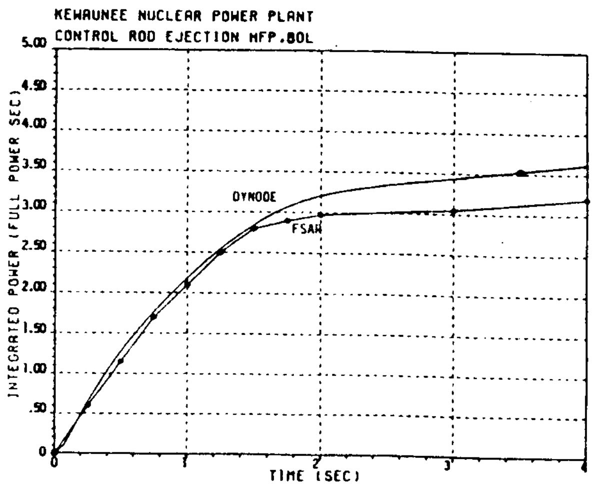


FIGURE 3.15-6

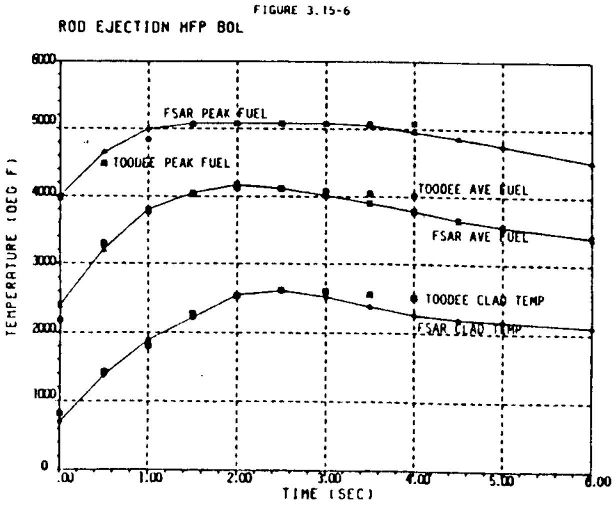


FIGURE 3.15-7

KEWAUNEE NUCLEAR POWER PLANT

CONTROL ROD EJECTION HZP.EOL

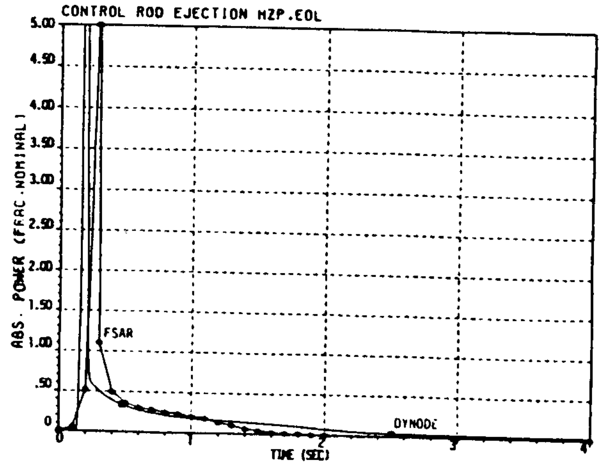


FIGURE 3.15-8

KEWAUNEE NUCLEAR POWER PLANT

CONTROL ROD EJECTION HZP.EOL

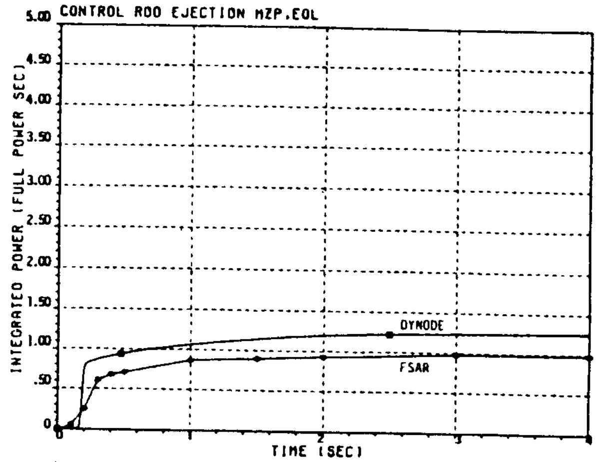


FIGURE 3.15-9

KEWAUNEE NUCLEAR POWER PLANT
CONTROL ROD EJECTION HFP.EOL

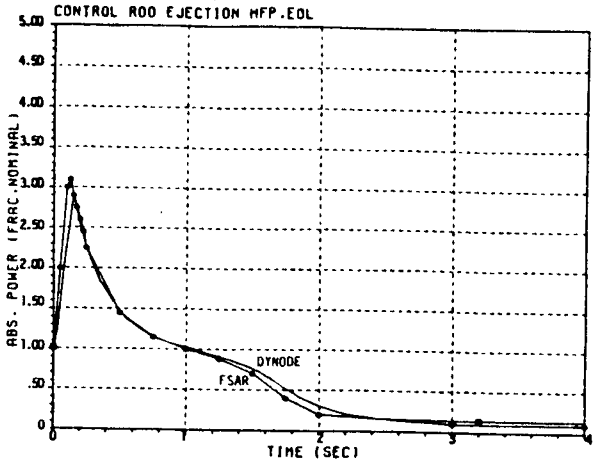
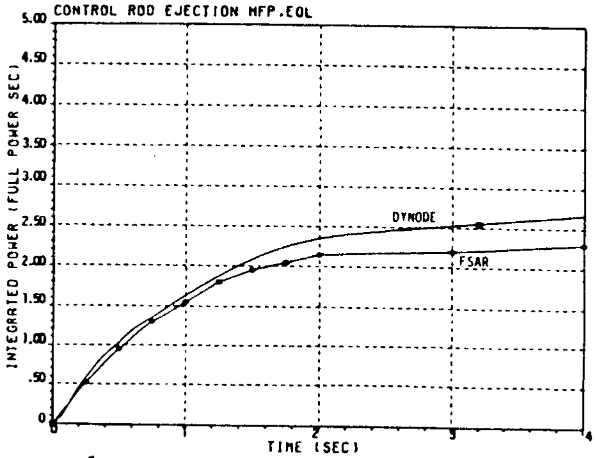


FIGURE 3.15-10

KEWAUNEE NUCLEAR POWER PLANT
CONTROL ROD EJECTION HFP.EOL



3.16 LOSS OF COOLANT ACCIDENT

3.16.1 DEFINITION OF ACCIDENT

The loss of coolant accident is defined as the rupture of the reactor coolant system piping or any line connected to the system, up to and including a double-ended guillotine rupture of the largest pipe. Ruptures of small flow area would cause coolant expulsion at a rate which would allow replacement at the same rate via the charging pumps and an orderly shutdown would be possible. A larger rupture would result in a net loss of reactor coolant inventory and a decreasing pressurizer water level and pressure. A Reactor trip occurs and safety injection is actuated resulting in the injection of borated water into the reactor coolant system, isolation of the normal feedwater, and initiation of the auxiliary feedwater supply. When the reactor coolant system depressurizes to 700 psia, the nitrogen bubble in the accumulator tanks expands, forcing additional water into the reactor coolant system. For large breaks, void formation in the core coolant during the initial blowdown phase results in almost immediate power reduction down to decay heat levels.

3.16.2 ACCIDENT ANALYSIS

An analysis of the loss of coolant accident is performed to demonstrate the effectiveness of the emergency core cooling system (ECCS) to meet the criteria of 10CFR50.46 and in preventing radioactive releases which would violate the criteria of 10CFR100. This analysis is usually performed in four steps:

1. A system blowdown analysis is performed to obtain the time-dependent behavior of core power, system pressure, flowrates, and other relevant variables. The digital model employed in this calculation is a detailed representation of the primary and secondary systems, including the hot fuel assemblies and the remainder of the core; the reactor vessel downcomer, upper plenum, upper head, and lower plenum regions; the steam generators, pressurizer, and associated piping; and the safety injection systems. The model uses a lumped "node and flow-path" approach to compute the space and time variations of the thermal-hydraulic conditions of the primary and secondary systems. Some of the phenomena which must be considered in the blowdown analysis are coolant flows between regions; heat transfer between primary and secondary fluids, and

between system metal surfaces and fluids contacting them; the hydraulic interactions of system components such as reactor coolant pumps; fuel rod swelling and rupture; and the behavior of emergency core coolant as it is injected into a system undergoing rapid decompression.

2. An analysis of the core hot channel is conducted using a detailed thermal-hydraulic model supplied with time-varying boundary conditions from the blowdown analysis. These calculations must consider cross flow between regions and any flow blockage calculated to occur as a result of clad swelling or rupture. The calculated flow must be smoothed to eliminate calculated oscillations with a period of less than 0.1 seconds. This model is used during the period extending from the beginning of blowdown to the end of ECCS bypass.

3. A reflood model continues the system blowdown analysis through the period of ECCS reflood of the reactor core. Due to the complexity of the phenomena occurring, empirical correlations of experimental data are used to define such variables as carryover fraction, heat transfer coefficients, natural convection in the secondary side of the steam generators, and slip flow in the ruptured loop cold leg nozzle.

4. A thermal calculation of the temperature transient in the hot fuel rod during refill and reflood is accomplished using a fourth model. As in the reflood model, empirical correlations of measured data are employed to represent complex phenomena such as flow blockage due to clad swelling and rupture. Metal - water chemical reaction and radiation from the fuel rod surface are included in the hot rod model.

Detailed requirements for ECCS evaluation models are are described in 10CFR Part 50, Appendix K.

Certain data related to core neutronics are required as input to the ECCS evaluation model described above. These items consist of the data required to calculate the continuing fission energy generation prior to shutdown by voiding and boron injection, the data necessary to calculate fission product decay heat subsequent to reactor shutdown, and data relating to the initial spatial power distribution.

The fission power history prior to reactor shutdown is calculated from the reactor kinetics equations, with terms included to account for fuel temperature and moderator density feedback, control rod insertion, and injection of borated water. For the larger breaks, reactor shutdown usually occurs due to coolant void formation, while for smaller breaks, scram reactivity insertion is required. A conservative calculation is assured by assuming the minimum plausible values for the various components in the reactivity balance. The moderator feedback is calculated using the boron concentration which corresponds to the core status when the Technical Specification requirement relating to non-positive moderator temperature coefficient is just met. Moderator reactivity is input to the transient calculation as a function of core coolant density. The Doppler reactivity feedback is usually much smaller than that resulting from coolant voiding.

Reactor trip may be actuated by one of several signals; the particular trip setpoint first reached and the time of trip are dependent on break size, particularly for small breaks. For large breaks, trip occurs due to high containment pressure or safety injection actuation; while for smaller breaks, pressurizer low pressure actuates the trip.

The trip reactivity insertion is calculated assuming the most reactive rod to remain in its fully withdrawn position and using a rod drop time corresponding to the Technical Specification limit. Large break accidents do not exhibit significant sensitivity to trip reactivity.

Fission product and actinide decay energy sources are calculated in accordance with the requirements of Appendix K of 10 CFR Part 50. Infinite operating time is assumed prior to accident initiation.

The spatial power distribution used in the ECCS evaluation analysis is chosen as the most limiting from the several calculated to occur over the lifetime of the core. Axial power shapes with maxima near the core mid-plane generally result in the most severe accident consequences. This is because the upper portions of the core are cooled to a greater extent during the flow reversal which occurs early in blowdown, and the lower portion of the core is cooled quickly by the initial stages of reflood. The initial hot spot peaking factor, F_Q, plays an important role in determining the severity of the worst cladding temperature response in the core. Because of the rapid degradation in heat transfer following the break, the temperature profile within the fuel rods tends to addition, larger values of F_Q will result in less effective heat transfer during the reflood period at the hot spot. Thus, a larger value of F_Q will produce a more severe cladding temperature response.

3.16.3 WPS SAFETY ANALYSIS RESULTS

WPS has not analyzed this accident. The current docketed analysis is reviewed for each reload to determine its applicability to the current core design.

3.16.4 CYCLE SPECIFIC PHYSICS CALCULATIONS

a. SCRAM REACTIVITY CURVE, $\Delta\rho_{SCRAM}(t)$

Calculations of the scram reactivity curve are performed in accordance with the general procedures described in Section 2. Cycle specific calculations for this accident are performed at BOC and EOC for the full power condition. A conservatively slow scram curve is generated by making the following assumptions:

1. The integral of the scram curve is based on an initial rod position at or below the full power insertion limits.

2. The shape of the scram curve is based on an initial rod position of full out. This provides the longest possible delay to significant reactivity insertion.

3. The xenon distribution is that which causes the minimum shutdown margin.

4. Instantaneous redistribution of flux is assumed to occur during the rod insertion.

b. NUCLEAR HEAT FLUX HOT CHANNEL FACTOR, F_Q

The maximum core F_Qs are assumed to remain within the current limits as defined in the Technical Specifications for allowable combinations of axial offset and power level. For Kewaunee, the continuous surveillance of the power distribution is accomplished with the ex-core detectors using the Power Distribution Control (PDC-II) scheme (12). The cycle specific physics calculations performed for the verification of the PDC-II scheme with respect to the F_Q limits are described in Section 3.17.

c. NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR, $F_{\Delta H}$

Calculations of $F_{\Delta H}$ are performed in accordance with the general procedures described in Section 2. Maximum core $F_{\Delta H}$s are verified to remain within Technical Specification limits for allowable combinations of axial offset, power level, and control rod insertion. For Kewaunee, the continuous surveillance of the power distribution is accomplished with the ex-core detectors using a Power Distribution Control scheme, PDC-II (12). The cycle specific physics calculations performed for the verification of the PDC-II scheme with respect to the $F_{\Delta H}$ limits are described in Section 3.17.

3.16.5 RELOAD SAFETY EVALUATION

The calculated values of F_Q and $F_{\Delta H}$ are increased to include the model reliability factors, and core tilt penalties. These adjusted cycle specific values are compared to the current peaking factor limits defined in the Technical Specifications. The details of this comparison are described in Section 3.17. The scram reactivity curve is conservatively adjusted by the rod worth reliability factor.

CYCLE SPECIFIC PARAMETERS		SAFETY ANALYSIS PARAMETERS
a. $\lvert \Delta\rho_{SCRAM}(t)*(1-RF_{RODS})\rvert$	$\geq$	$\lvert \Delta\rho_{SCRAM}(t)\rvert$ (bounding)
b. $F_Q*(1+RFF_Q)*(1+T)$	$\leq$	Tech Spec Limit (Refer to Section 3.17)
c. $F_{\Delta H}*(1+RFF_{\Delta H})*(1+T)$	$\leq$	Tech Spec Limit (Refer to Section 3.17)

The integral of the bounding value of the scram curve, $\Delta\rho_{SCRAM}(t)$, is taken as that rod worth required to produce the shutdown margin assumed in the safety analysis for the most limiting cycle specific core conditions discussed in 3.16.3.b above.

3.17 POWER DISTRIBUTION CONTROL VERIFICATION

Calculations are performed at exposures ranging from beginning to end of cycle to verify the applicability of the power distribution control (PDC-II) scheme as defined in the Technical Specifications. Specifically, the core peaking factors, $F_{\Delta H}$ and FQ(Z) are calculated at full power equilibrium core conditions and multiplied by conservative factors to verify that they remain within the limits as defined in the Technical Specification.

3.17.1 PEAKING FACTOR CALCULATIONS

Calculations of F_Q and $F_{\Delta H}$ are performed in accordance with the general procedures described in Sections 2.7 and 2.8. Cycle specific calculations are performed at full power equilibrium core conditions at exposures ranging from BOC to EOC. Statistical uncertainty factors derived from measured to predicted power distribution comparison analyses are conservatively applied to the calculated peaking factors[1].

Variations in the axial power distribution cause variations in FQ(Z) distribution, while the associated control rod motion causes variations in the $F_{\Delta H}$ distribution. To account for potential axial power distribution variations allowed by the Power Distribution Control (PDC-II) procedures, conservative factors called the V(Z) function are applied to the calculated full power equilibrium $F_Q(Z)$. The V(Z) function is determined by investigating the changes in $F_Q(Z)$ during core axial power perturbations, most of which are induced with combinations of power level and rod insertions changes (12).

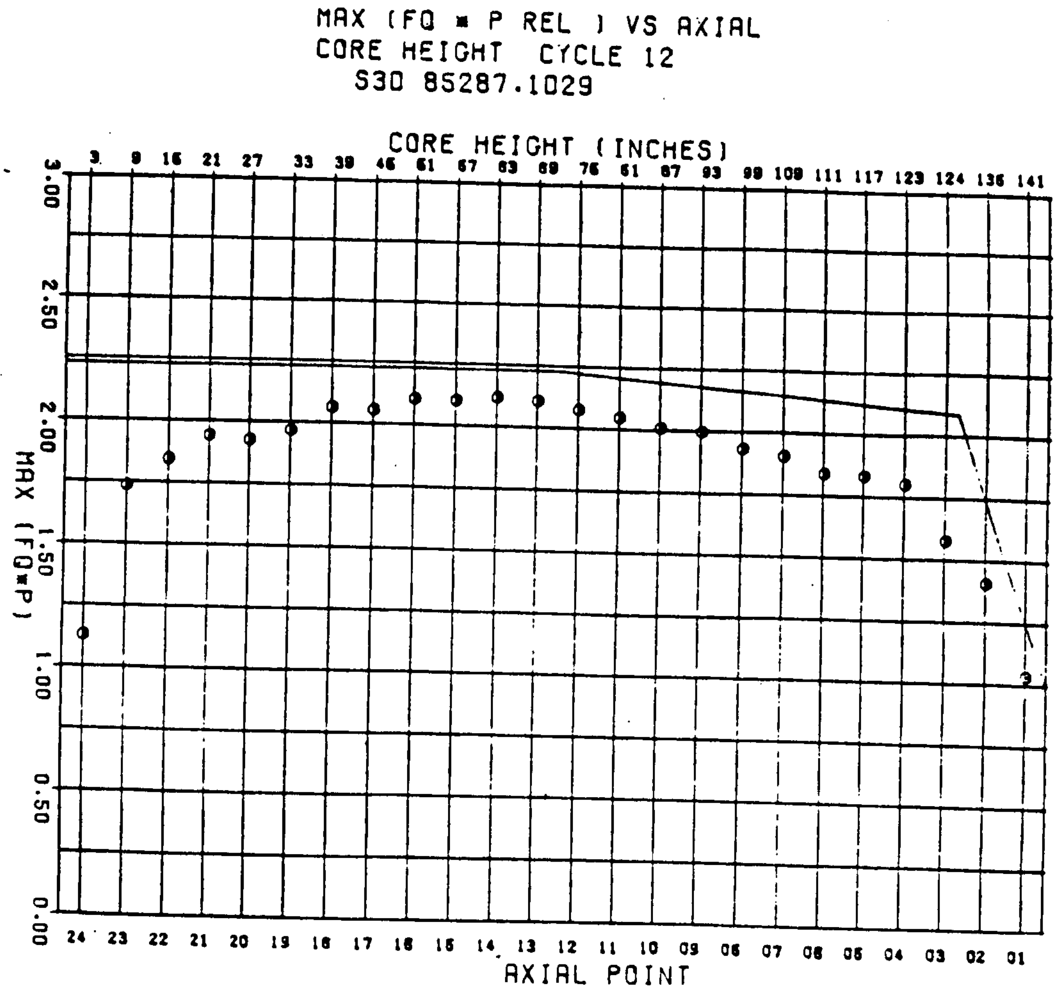
The maximum $F_{\Delta H}$ is chosen from a range of power distributions resulting from core maneuvers allowed by PDC-II in combination with control rod insertions allowed by Technical Specification Rod Insertion limits.

3.17.2 RELOAD SAFETY EVALUATION

The calculated peaking factors, $F_Q(Z)$ and $F_{\Delta H}$ are increased by statistical uncertainty factors and conservative reliability factors[1]. $F_Q(Z)$ is further increased by the V(Z) function (12). Calculated cycle specific $F_Q(Z)$ and $F_{\Delta H}$ are compared to the current Technical Specification limits. A typical comparison plot of FQ(Z) for Kewaunee is shown in Figure 3.17.1.

CYCLE SPECIFIC PARAMETERS		SAFETY ANALYSIS PARAMETERS
a. $F_Q(Z) * (1+RFF_Q) * V(Z) * (1+T)$	$\leq$	FQ (Technical Specification Limits)
b. $F_{\Delta H} * (1 + RFF_{\Delta H}) * (1+T)$	$\leq$	FΔH (Technical Specification Limits)

FIGURE 3.17-1


MAX (FQ * P REL) VS AXIAL
CORE HEIGHT CYCLE 12
S30 85287.1029
CORE HEIGHT (INCHES)
3 9 15 21 27 33 39 45 51 57 63 69 75 81 87 93 99 105 111 117 123 129 135 141
3.00
2.50
2.00
1.50
1.00
0.50
0.00
MAX (FQ*P)
24 23 22 21 20 19 18 17 16 15 14 13 12 11 10 09 08 07 06 05 04 03 02 01
AXIAL POINT

4.0 REFERENCES

1. Letter from E. W. James, WPS to A. Schwencer, NRC, transmitting Wisconsin Public Service Corporation, Kewaunee Nuclear Power Plant, topical report entitled, "Qualification of Reactor Physics Methods for Application to Kewaunee", October, 1978.

2. Letter from E. W. James, WPS to A. Schwencer, NRC, transmitting Wisconsin Public Service Corporation, Kewaunee Nuclear Power Plant, topical report entitled, "Reload Safety Evaluation Methods for Application to Kewaunee", January, 1979.

3. Letter from A. Schwencer, NRC, to E. R. Mathews, WPS, transmitting Safety Evaluation Report by the Office of Nuclear Reactor Regulation: "Qualification of Reactor Physics Methods for Application to Kewaunee," October 22, 1979.

4. Wisconsin Public Service Corporation, Kewaunee Nuclear Power Plant, Updated Safety Analysis Report, July 1, 1986.

5. Wisconsin Public Service Corporation, Technical Specifications for the Kewaunee Nuclear Power Plant, Docket Number 50-305.

6. "Reload Safety Evaluation," for Kewaunee Nuclear Power Plant, all previous reload cycles.

7. Letter from Gaston Fiorelli (NRC) to E. W. James (WPS) transmitting inspection report 30-305/78-15 dated September 6, 1978.

8. Northern States Power Company, Prairie Island Nuclear Power Plant, "Reload Safety Evaluation Methods for Application to PI Units, NSP NAD8102-A, Revision 4, dated June, 1986.

9. C. W. Stewart, et. al. "VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores", BNWL Research Project 1584-1, Revision 1, November, 1983.

10. Letter from Charles E. Rossi (NRC) to John A. Blaisell (UGRA), "Safety Evaluation Report on EPRI NP-2511-CCM VIPRE-01", May 1, 1986.

11. Letter from Cecil O. Thomas (NRC) to Dr. Thomas W. Schnatz (UGRA), "Safety Evaluation Report: RETRAN - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," September 2, 1984.

12. R. J. Burnside and J. S. Holm, "Exxon Nuclear Power Distribution Control For Pressurized Water Reactors, Phase II" XN-NF-77-57, Exxon Nuclear Company, Inc., January 1978.

13. D. H. Risher, Jr., "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Models," WCAP-7588, Revision 1-A, January 1975.

APPENDIX A

Computer Program Overview

This section describes the WPS computer programs that were used to simulate the response of the nuclear steam supply system (NSSS) and predict the thermal-hydraulic response of the hot coolant channel and hot spot in the core for the transients and accidents listed in Section 3.0.

The DYNODE-P (Version 5.4) program is used to analyze the transient response of the Nuclear Steam Supply System (NSSS). This program is described in detail in Reference B1 of Appendix B of this report. DYNODE-P provides a simulation of the core average power, the core average fuel temperature, and the core average coolant channel thermal-hydraulic responses.

The VIPRE-01 program is used to analyze the transient response of the hot channel in the core. VIPRE-01 provides a simulation of the thermal-hydraulic response of the coolant channels and associated fuel rods within the core.

The TOODEE-2 program is used to compute the transient temperature response of the hot fuel spot for certain accidents. TOODEE-2 provides a simulation of the hot fuel rod and associated coolant channel. This program is used only if the VIPRE-01 hot channel analysis yields a departure from nucleate boiling ratio (DNBR) which is less than the value corresponding to the 95% probability limit at 95% confidence level.

The sequence of calculations and interfaces of these programs are as follows: DYNODE-P is run to obtain the core average heat flux and the RCS thermal-hydraulic responses. The transient core average heat flux, core inlet coolant temperature and RCS pressure responses along with the appropriate core spatial power distribution and hot channel flow rate are then input into VIPRE-01 to obtain the hot channel transient DNBR. Similar information is also input into TOODEE-2 to analyze the thermal response of the hot fuel spot, in those cases requiring this analysis.

RETRAN-02 will also be used to analyze the NSSS system transient response. RETRAN will be used to verify analyses by DYNODE or to independently analyze a transient concern.

Best estimate versions of the safety analysis models were developed. A description of the best estimate models and comparisons of best estimate model calculations to Kewaunee plant and simulator data are provided in Appendix F.

APPENDIX B

NSSS SIMULATION, DYNODE-P

NSSS Simulation

The response of the NSSS of a PWR under transient and accident conditions is analyzed with the DYNODE-P program (B1). This program includes a simulation of the components of a PWR NSSS which significantly influence the response of the system to transient conditions. DYNODE-P analyses have been reviewed by the NRC as part of the Reload Safety Methods Topical Report submitted by Northern States Power (B2).

The major features of the DYNODE-P (version 5.4) program are:

- Point and one dimensional kinetics model for core power transients with major feedback mechanisms and decay heat represented. Initial subcritical core conditions can be modeled.
- Power forced mode option for hot channel analyses.
- Multinode radial fuel rod and multinode axial coolant channel representations in the core.
- Conservation of mass, energy, volume, and boron concentration for the reactor coolant system (RCS). Conservation of momentum is optional.
- Detailed non-equilibrium pressurizer model including spray and heater systems and safety and relief valves.
- Explicit representation of the shell side of the steam generators including conservation of mass, energy, and volume.
- Representation of heat transfer with structural metal components of the NSSS.
- Explicit representation of the main steam system with isolation, check, dump, bypass, and turbine valves including conservation of mass, energy, momentum, and volume.
- Representation of the reactor protection and high pressure safety injection systems.
- Representation of the major control systems.
- Provisions for simulating a variety of transients and accidents including a break in the main steam system and asymmetric loop transients.

The base input parameters relating to the initial conditions are:

- Core geometry and initial thermal-hydraulic characteristics.
- Initial RCS pressure and pressurizer level, core inlet enthalpy, RCS flow distribution, and RCS boron concentration.
- Initial core power level and distribution.
- RCS, steam generator, and main steam system volume distributions and hydraulic characteristics.
- Initial steam generator pressures and levels and heat transfer data.

The input parameters required to obtain the transient responses are:

- Core kinetics characteristics including control rod motion.
- Reactor coolant system inertias, pressure loss coefficients, and pump hydraulic and torque characteristics.
- Control system characteristics.
- Main and auxiliary feedwater characteristics.
- Valve characteristics.
- Safety systems characteristics.
- Transient power demand.

The major output consists of the following list of parameters which are edited at select time points during the transient:

- Core variables

  Average power - Fuel rod temperatures and heat flux - Coolant enthalpies, temperature, and mass - Kinetics variables including keff.

- RCS variables

  Mass, energy, and boron distribution of the coolant - Loop flow rates - Pressurizer pressure and level - Safety system variables - Pressure control system variables - Reactor coolant pump speeds, torques, and developed heads

- Steam generator variables

  Pressure and levels - Masses - Heat loads - Feedwater and steam flows

- Main steam system variables

  Pressure and mass distributions - Steam flows

APPENDIX B

References

B1. R. C. Kern, L. W. Cress, D. L. Harrison, "Dynode P, Version 5.4: A Nuclear Steam Supply System Transient Simulator for Pressurized Water Reactors - User Manual", UAI 83-49, November 1983.

B2. Northern States Power Company, Prairie Island Nuclear Power Plant, "Reload Safety Evaluation Methods for Application to PI Units, NSP NAD-8102A", Revision 4, dated June, 1986.

APPENDIX C

THERMAL-HYDRAULIC ANALYSIS, VIPRE-01

A thermal-hydraulic hot channel analysis is performed for transients which are DNB limiting. The methodology in the original submittal of this topical report used COBRA-IV for thermal hydraulic analysis of the hot fuel channel. VIPRE-01 has since replaced COBRA-IV as the primary WPS fuel thermal-hydraulic analysis computer code. Comparisons of VIPRE-01 to COBRA-IV have been presented (C2, C3, C4) and have shown similar results between the two codes.

VIPRE-01, a thermal hydraulic computer code developed by Battelle Northwest under the sponsorship of EPRI, computes the flow and enthalpy distribution in the fuel assembly subchannel for steady state or transient conditions. VIPRE-01 has undergone generic review by the NRC (C2) at the request of the Utility Group for Regulatory Applications (UGRA) and has been found acceptable for use in licensing applications. Wisconsin Public Service is a UGRA member and has contributed to various UGRA submittals. Utility specific submittals containing VIPRE-01 analyses have also been reviewed by the NRC (C3, C4).

The coolant regions analyzed by VIPRE-01 are divided into computational cells in which the conservation equations for mass, energy, and momentum for the fluid are solved. The independent variables; enthalpy, pressure, void fraction and velocity are averaged for each cell considering heat and momentum sources and sinks due to fixed solids such as fuel rods and grid spacers.

Heat transfer regimes from subcooled to super-heated forced convection including departure from nucleate boiling (DNB), and turbulent and diversion cross flows are considered in the subchannel analysis.

The basic input parameters are:

- Fuel rod and channel geometries
- Fluid thermal-hydraulic parameters
- Heat flux or power distribution
- Turbulent mixing parameters
- Transient forcing functions:

  Core inlet temperature, Core inlet flow, System average pressure, Core power or heat flux.

The major time-dependent output parameters are:

- Subchannel DNBR
- Subchannel flow distribution
- Subchannel fluid properties
- Fuel rod temperature distribution

The WPS safety analysis methodology establishes thermal margin on the basis of Kewaunee transient system analyses performed with DYNODE-P and fuel thermal-hydraulic analyses performed with VIPRE-01. VIPRE-01 evaluates the MDNBR response in the hot channel using DYNODE-P results for core heat flux, inlet temperature, pressure, and flow as input forcing functions.

VIPRE-01 predictions of MDNBR are benchmarked against the Kewaunee USAR (C5) to evaluate the performance of the WPS thermal margin methodology (Refer to comparisons in section 3 of this report). An effort was made to construct a model for benchmarking which is comparable to that used in the original analysis A single hot channel model of the Westinghouse fuel design which includes a constant gap coefficient was developed for the USAR benchmarking effort.

Additional 1/8 assembly and 1/8 core models were developed to more accurately account for assembly subchannel and core-wide flow distribution effects. These models provide an overall analysis of the thermal-hydraulic behavior of the core. The hot quarter assembly in the 1/8 core model is modeled by individual subchannels each consisting of an individual or a limited number of fuel rods. The remainder of the core in the 1/8 core model is modeled on an assembly-by-assembly basis. Each channel is divided axially into increments of equal lengths. Resistance to crossflow and coolant mixing between adjacent channels is considered. Flow redistribution due to localized hydraulic resistances (e.g. spacer grids) is also predicted. The effects of local variations in power, fuel rod, and fuel pellet fabrication and fuel rod spacing are also considered.

Power distributions in the expanded VIPRE-01 models are predicted by the WPS core design analysis. The VIPRE-01 thermal hydraulic analysis may be performed on power distributions which represent the specific reload core designs or which are increased to a pre-determined $F_{\Delta H}$ as was done in the USAR benchmark effort.

Once the highest powered rod has been identified in the core design analysis, the hot quarter assembly which contains this hottest rod is represented by single subchannels. The hot subchannel is identified as the one having the lowest MDNBR with that fuel rod power distribution.

In valid subchannel analysis, sufficient detail of the regions surrounding the hot channel must be considered. If a case is specified where the hot channel occurs on the edge of the hot quarter assembly, the hot channel would be adjacent to a quarter assembly lumped channel. The basic philosophy upon which subchannel analyses are based is thus not being satisfied. Therefore a new geometry must be described such that the hot channel is interior to a region of equivalently sized subchannels. This new geometry is modeled by representing the adjacent quarter assembly on a detailed subchannel basis, similar to the hot quarter assembly, rather than a lumped channel basis.

In the following sections sensitivity studies on key input parameters to VIPRE-01 are described.

Radial Power Distribution

Individual rod power generation and assembly lumped power generation are represented in the core-wide mode. Radial power distributions are obtained from the core design analysis with the hot fuel rod power increased to a predetermined $F_{\Delta H}$. The radial power distribution within the hot assembly is conservatively assumed to be the worst assembly distribution as determined from the core design analysis. This worst case distribution is then conservatively overlayed on the VIPRE-01 hot assembly location.

In the following table, the mass flux in the VIPRE-01 single channel model is adjusted so the DNBR equals that of the USAR (C5). The mass flux in the eighth assembly and eighth core are determined by keeping the flow in the hot channel equal to the single channel model flow and adjusting the core average flow accordingly.

Model	Relative MDNBR	Axial Flow Factor	Exit Enthalpy
Westinghouse	1.000	--	--
Single Channel	1.000	--	696.9
Eighth Assembly	0.995	0.986	671.7
Eighth Core	1.008	0.961	668.2

Axial flow factor, as defined here, is the ratio of the mass flux in the hot channel to the average mass flux in the core. Although cross-channel mixing causes exit enthalpy to decrease in the larger models, the overall effect on DNBR in the steady state case is small.

Two radial assembly power distributions were analyzed in the eighth core geometry model to determine the worst case, a conservatively flat and a typical locally peaked distribution. As shown the conservatively flat distribution was the most limiting since the flatter power distribution reduces the coolant mixing effects in the hot assembly.

Power Distribution	Relative MDNBR	Axial Flow Factor	Exit Enthalpy
Flat	1.000	0.948	693.3
Typical	1.049	0.960	672.2

Although the flat distribution is more conservative, the typical distribution with a locally peaked hot channel is still conservative due to the artificially raised $F_{\Delta H}$ discussed earlier. This radial distribution will be used in all future licensing calculations.

Axial Power Distribution

A constant axial flux distribution is superimposed on the radial distribution in order to yield the heat generated in each individual rod (or lumped rod) at any elevation. A cosine distribution with a conservatively high peak, 1.77, is used. A sensitivity study was performed for three axial power distributions, a middle peaked, a down-skewed and an up-skewed cosine shape (see figure C1). It should be noted that the skewed cases are well out of the operating band of the power distribution control scheme used at Kewaunee.

Power Distribution	Relative MDNBR	Point of MDNBR	Power in Upper Half of Core	Axial Flow Factor	Exit Enthalpy
COSINE	1.000	86 in	0.5	.960	672.2
UP-SKEW	0.953	110 in	0.68	.959	673.5
DOWN-SKEW	1.040	66 in	0.32	.961	671.1

The results of this study show that the axial flux shape with the most positive axial offset is the most limiting. This is expected since the location of MDNBR is in the upper portion of the core.

Based on these results, a limiting up-skewed cosine power distribution will be used for VIPRE-01 licensing calculations. A conservatively high peak, FZ, will be imposed on this power distribution and will be calculated using the current Technical Specification peaking factor limits according to:

$$FZ = F_Q^N \; / \; F_{\Delta H}^N$$

The power distribution used in VIPRE licensing calculations will be verified each cycle to ensure that it bounds all power distributions permitted by the Kewaunee power distribution control scheme and the Technical Specification.

Inlet Flow Distribution

In order to determine the effect of flow distribution, a sensitivity study was performed with three inlet flow distributions. The three distributions looked at were: a flow distribution to produce constant pressure drop in the first node; a 20 percent reduction in flow to the hot quarter assembly; and a 20 percent reduction in flow to the four channels bordering the hot rod.

Flow Distribution	Relative MDNBR	Axial Flow Factor	Recovery Point
Uniform	1.000	0.960	--
Reduced in HC	1.004	0.965	4.0"
Reduced in HQA	0.998	0.961	12.0"

Recovery point, as defined here, is the height at which flow is within 0.2% of the uniform value.

The reduction in flow had very little effect on the MONBR because the hot channel flow rate recovered very quickly. At about 20 inches from the bottom of the core the flow rate was almost identical in the three cases. This shows that the effect of cross flow is indeed substantial and that the inlet flow distribution is not important. Additional analyses (Ref. C3) have shown that MDNBR results are also insensitive to inlet pressure distributions.

Assembly Crossflow

The transverse momentum equation provides for the evaluation of cross flow between adjacent channels.

KIJ = KIJDUM*SL

where:

SL = Transverse Momentum Factor = s/l

s = gap spacing

l = centroid distance

KIJ = Crossflow Resistance Coefficient

KIJDUM = A constant determined by the user

SL is fixed at .25 by the geometry. A reasonable value for KIJDUM is 0.5. Sensitivity studies were performed for $0.01 \leq$ KIJDUM ≤ 100. The most conservative assumptions are those that increase the crossflow effect. As KIJ decreases, crossflow increases. The results, however, are relatively insensitive to KIJ, as shown below:

KIJDUM	Relative MDNBR	Axial Flow Factor	Exit Enthalpy
0.01	1.000	0.960	672.2
0.1	1.000	0.960	672.2
0.5	1.000	0.960	672.2
10.0	1.0D0	0.960	672.2
100.0	1.004	0.965	671.8

Turbulent Mixing

Turbulent Mixing is determined by:

$$W' = S\bar{G}$$

where:

W' = Turbulent crossflow
S = Gap width
$\bar{G}$ = Average flow in adjacent channels
= Turbulent mixing coefficient

β can either be a constant or a function of Reynolds number. The sensitivity of MDNBR to the value of β is shown below:

Turbulent Mixing Factor	Relative MDNBR	Axial Flow Factor	Exit Enthalpy
0.0062 Re -0.1	0.977	0.957	681.0
0.062 Re -0.1	0.998	0.960	672.8
0.62 Re -0.1	0.998	0.966	660.5
0.0	0.973	0.955	683.9
0.01	0.991	0.957	675.1
0.019	1.000	0.960	672.2
0.04	1.017	0.966	669.0

Although the effect is small, increasing β tends to increase MDNBR because increased turbulent mixing tends to even out the flow in the channels, increasing the axial flow factor in the hot channel. Reference C3 recommends β be set to 0.019 for all subchannel gaps because it is the smallest value that is physically reasonable. A larger value would be less conservative.

Another parameter in VIPRE's turbulent mixing model is the turbulent momentum factor, FTM, which determines the efficiency with which turbulent cross flow mixes momentum. It can be seen below that this parameter has very little effect on MDNBR and the Reference C1 recommended value for FTM, 0.8, will be used.

FTM	Relative MDNBR	Axial Flow Factor	Exit Enthalpy
0.0	0.999	0.964	672.2
0.8	1.000	0.960	672.3
1.0	1.004	0.965	672.2

Axial Increments

The size of the axial nodes was studied. Three cases were run in which the core was divided into 36, 24, and 12 axial increments. The three cases produced consistent results demonstrating a relative insensitivity of MDNBR to axial node height.

Axial Increments	Relative MDNBR	Axial Flow Factor	Exit Enthalpy
36	1.000	0.960	672.2
24	1.001	0.952	672.3
12	1.005	0.968	670.3

Steady State Results

In order to assess the steady-state capabilities of the W3 correlation in VIPRE-01 the safety limits curves from the Kewaunee Technical Specifications (C6) were reproduced at several selected points using VIPRE-01. These safety limits curves define the region of acceptable operation with respect to average temperature, power, and pressure. On the boundary of this region MDNBR=1.3.

A steady state single channel model was used. Inputs were derived wherever possible from Reference C7 which describes the analysis methods used to generate the safety limit curves. The following table shows the results.

VIPRE-01 Reanalysis of Safety Limit Curves

Pressure	Tave	Power	MDBNR VIPRE	MDNBR West	& Diff.
1700	582 °F	102%	1.290	1.39	-0.8
17D0	545 °F	120%	1.319	1.30	1.5
2000	604 °F	97%	1.331	1.30	2.4
2000	601 °F	100%	1.277	1.30	-1.8
20D0	567 °F	120%	1.301	1.30	0.1
2200	6D6 °F	102%	1.285	1.30	-1.2
2200	580 °F	120%	1.290	1.30	-0.8
2400	608 °F	106%	1.333	1.30	2.5
2400	590 °F	120%	1.317	1.30	1.3

VIPRE MDNBR Values - Mean = 1.305, Standard Deviation = 0.020.

The WPS results generally agree within 2 percent of the Westinghouse results indicating that the W-3 correlation used in the VIPRE-01 steady state model can adequately calculate thermal safety limits.

Final Values

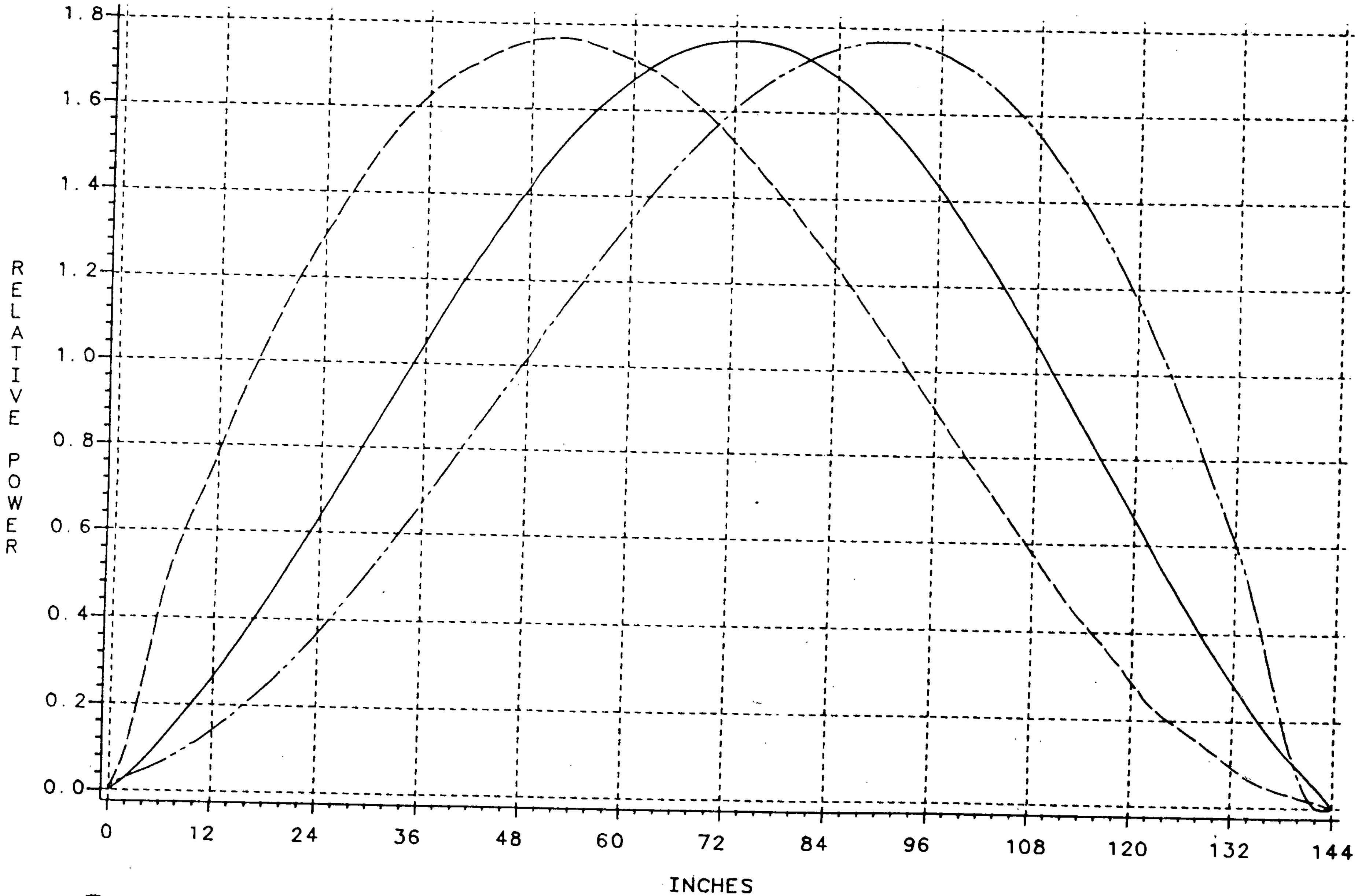
Table C1 is a list of important parameters and correlations which will be used for the Kewaunee VIPRE-01 licensing model. These parameters and correlations were selected based on the assumptions in the original Kewaunee safety analysis models, recommendations from EPRI and other utility topical reports, and WPS' sensitivity studies and benchmarks described in this report.

TABLE C1

Final WPS VIPRE Model

Geometry	
Model	1/8 core
Number of Channels	81
Number of Rods	78
Number of Axial Nodes	38
Channel Length	152 in.
Active Fuel Length	144 in.
Nuclear Parameters	
F_Z^N	1.77
$F_{\Delta H}^N$	1.58
F_{QT}	2.8
Axial Flux Profile	Cosine
Operating Conditions	
System Pressure	2220 psia
Core Inlet Temperature	539.5 °F
Core Average Mass Flux	2.314 Mlbm/hr ft^2
Core Average Heat Flux	0.1948 MBtu/hr ft^2
Flow Correlations	
Single Phase Friction	$F = 0.184\ Re^{-02}$
Two-Phase Friction Multiplier	Homogeneous
Subcooled Void	Levy
Bulk Void	Homogeneous
Heat Transfer Correlations	
Single Phase Forced Connection	h = 0.023 K/De Re 0.8 Pr 0.4
Nucleate Boiling	None
Film Boiling	None
Mixing Model	
Turbulent Mixing Factor	0.019
Turbulent Momentum Factor	0.8
Transverse Momentum Factor	0.25
Crossflow Resistance Factor	0.5 Length/Pitch
Critical Heat Flux	
CHF Correlation	W-3

FSAR VIPRE BASE CASE
1/8 CORE MODEL
AXIAL POWER PROFILE VS CHANNEL LENGTH



LEGEND: KEY —— COSINE — — DOWN-SKEW -·-·- UP-SKEW

FIGURE C1

APPENDIX C

References

C1. C. W. Stewart, et. al. "VIPRE-01: A Thermal Hydraulic Analysis Code for Reactor Cores, BNWL Research Project 1584-1, Rev. 1, November 1983.

C2. Letter from Charles Rossi (NRC) to John Blaisdell (UGRA) dated May 1, 1986, "Safety Evaluation Report on EPRI NP-2511-CCM VIPRE-01".

C3. Northeast Utilities, "NUSCO Thermal Hydraulic Model Qualification Volume II (VIPRE)" August 1, 1984.

C4. Northern States Power, Prairie Island Nuclear Plant, "Reload Safety Evaluation Methods for Application to PI Units, NSP NAD-8102-A4", Revision 4, dated June, 1986.

C5. Wisconsin Public Service Corporation, Kewaunee Nuclear Power Plant, Updated Safety Analysis Report, July 1, 1986.

C6. Wisconsin Public Service Corporation, Technical Specifications for the Kewaunee Nuclear Power Plant, Docket Number 50-305.

C7. WCAP 8092 "Fuel Densification - Kewaunee Nuclear Power Plant," March, 1973.

APPENDIX D

THERMAL-HYDRAULIC ANALYSIS, TOODEE-2

TOODEE-2 (D1) computes the thermal response of a fuel rod and associated coolant channel under transient conditions.

TOODEE-2 solves the conservation of energy equation in the fuel rod and the conservation of mass and energy in the coolant channel over the entire length of the core. The fuel-cladding gap model is the same as in the GAPCON programs. Material properties are computed based on local conditions. Cladding deformation is taken into account.

$Zr-H_2O$ reaction is also considered as part of the total heat source. Heat transfer regimes from subcooled to superheated forced convection are considered. The major input parameters are:

- Fuel rod and coolant channel geometries and properties.
- Initial power level and distribution
- Initial temperature distribution
- Time-dependent forcing functions
- Average power - Inlet flow and temperature - Saturation temperature

The major time-dependent output parameters are:

- Temperature distribution in fuel rod - Fuel rod surface and gap conditions - Energy in the fuel

APPENDIX D

References

D1. G. N. Lauben, "TOODEE-2: A Two-Dimensional Time Dependent Fuel Element Thermal Analysis Program," NUREG-75/057, May 1975.

APPENDIX E

NSSS Simulation, RETRAN-02

RETRAN-02 is a versatile and reliable thermal-hydraulic code used to analyze reactor system transients (E1). A Safety Evaluation Report has been issued by the USNRC in which it was determined that RETRAN-02 MOD003 is an acceptable program for use in licensing applications (E2). Plant-specific input data for Kewaunee have been developed. Following are the major input parameters.

- Geometry
  - Volumes
  - Junctions
  - Heat Conductors
- Core Data
  - Kinetics
  - Moderator and Fuel Temperature Coefficients
  - Control Reactivity
  - SCRAM Reactivity
- Control
  - Trips
  - Programmed Control Blocks
  - Valves
  - Fill Tables

The major output consists of the following parameters at predetermined time points during the transient.

- Volume parameters - Pressures, temperatures, liquid levels, and qualities at all nodes in the system.
- Junction parameters - Flows, pressure drops, enthalpies, and qualities at all connections between nodes.
- Core parameters - Fission power, decay heat power, Keff, control reactivity, fuel temperature reactivity, moderator temperature reactivity.
- Heat conductor parameters - Heat flux, temperature surface temperature and mass flux.

APPENDIX E

References

E1. J. H. McFadden, et. al. "RETRAN-02: A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems", Computer Code Manual, November 1984.

E2. Letter from Cecil O. Thomas (NRC) to Dr. Thomas W. Schnatz (UGRA), "Safety Evaluation Report: RETRAN-A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," September 2, 1984.

APPENDIX F

Best Estimate Models

The USAR analyses are a set of conservative calculations in that they are based on pessimistic assumptions of plant conditions. Disabling control systems, disabling or limiting reactor protection or engineered safety functions, using conservative initial plant operating conditions, maximizing delay times for mitigating actions, and using conservative reactivity feedback effects with limiting core power distributions are some of the assumptions applied in the accident analyses of the USAR. These analyses define a bounding analysis for licensing purposes. In the USAR benchmark comparisons presented in Section 3 of this report the WPS models incorporated these same conservatisms allowing the comparison of one licensing analysis to another.

More realistic evaluations of the plant response to transients are predicted by best estimate versions of the WPS safety analysis computer codes. In an effort to validate these best estimate models, actual plant measurements are compared to when available, e.g. Ref. F2. However due to the lack of actual Kewaunee plant transient data a second source of reference data, the Kewaunee plant simulator, is used.

Kewaunee best estimate safety analysis models for DYNODE-P, RETRAN-D2, and VIPRE-01 have been developed. Comparisons of predictions from these best-estimate models to Kewaunee plant and simulator data are presented and discussed in References F1 and F3 and are summarized in this appendix. These comparisons support the qualification of the WPS safety analysis models for best estimate evaluations of plant responses and demonstrate the capabilities of the computer codes to analyze events which have not been included in the analyses of the USAR. It is recognized that appropriate conservatisms should be applied to best estimate predictions to account for model uncertainties, although these uncertainties have not been quantified by the comparisons of this report.

NSSS predicted responses and plant measured results are compared in the Figures indicated for the following tests:

Plant Comparison End of Cycle 11 Test	50% Load Reduction at 5%/min.	Fig. F1
Plant Comparison Startup Test --	Pump Trip Loss of Flow	Fig. F2
Plant Comparison Startup Test --	50% Load Reduction at 200%/min.	Fig. F3

A set of six transients representing various classes of accidents were run on the Kewaunee simulator including: reactor trip, failed open pressurizer PORV, main steam line break, steam generator tube rupture, loss of AC power, and a 50% load reduction. Parameter predictions for each of these transients were compared to the simulator data and have been presented in previous reports (F1, F3). Selected comparison plots are included here for:

Simulator Comparison	Reactor Trip	Fig. F4
Simulator Comparison	SG Tube Rupture	Fig. F5
Simulator Comparison	PORV Fails Open	Fig. F6

A best estimate version of VIPRE-01 was created to evaluate thermal-hydraulic parameters for reload core designs, to compare to Kewaunee measured data, and to predict core thermal-hydraulic behavior. The important differences between the best estimate and USAR VIPRE models are mainly in the assumptions relating to the heat transfer and hydraulic correlations, the initial power distribution, and the plant initial operating conditions. The MDNBR calculated by the best estimate 1/8 core model at steady state full power conditions is 4.0 which compares to a MDNBR of 1.88 calculated for the full power steady state USAR VIPRE model.

A thermocouple map of the core exit temperature distribution was taken during flux map 1109. VIPRE power distribution inputs were derived from the measured power distribution. VIPRE predicted core exit temperatures were compared to the measured thermocouple temperatures. The results, presented in Figure F7, indicate that VIPRE predicts the measured temperatures in most assemblies to within ± 3°F.

FIGURE 1
EOC11 50% LOAD REDUCTION TEST

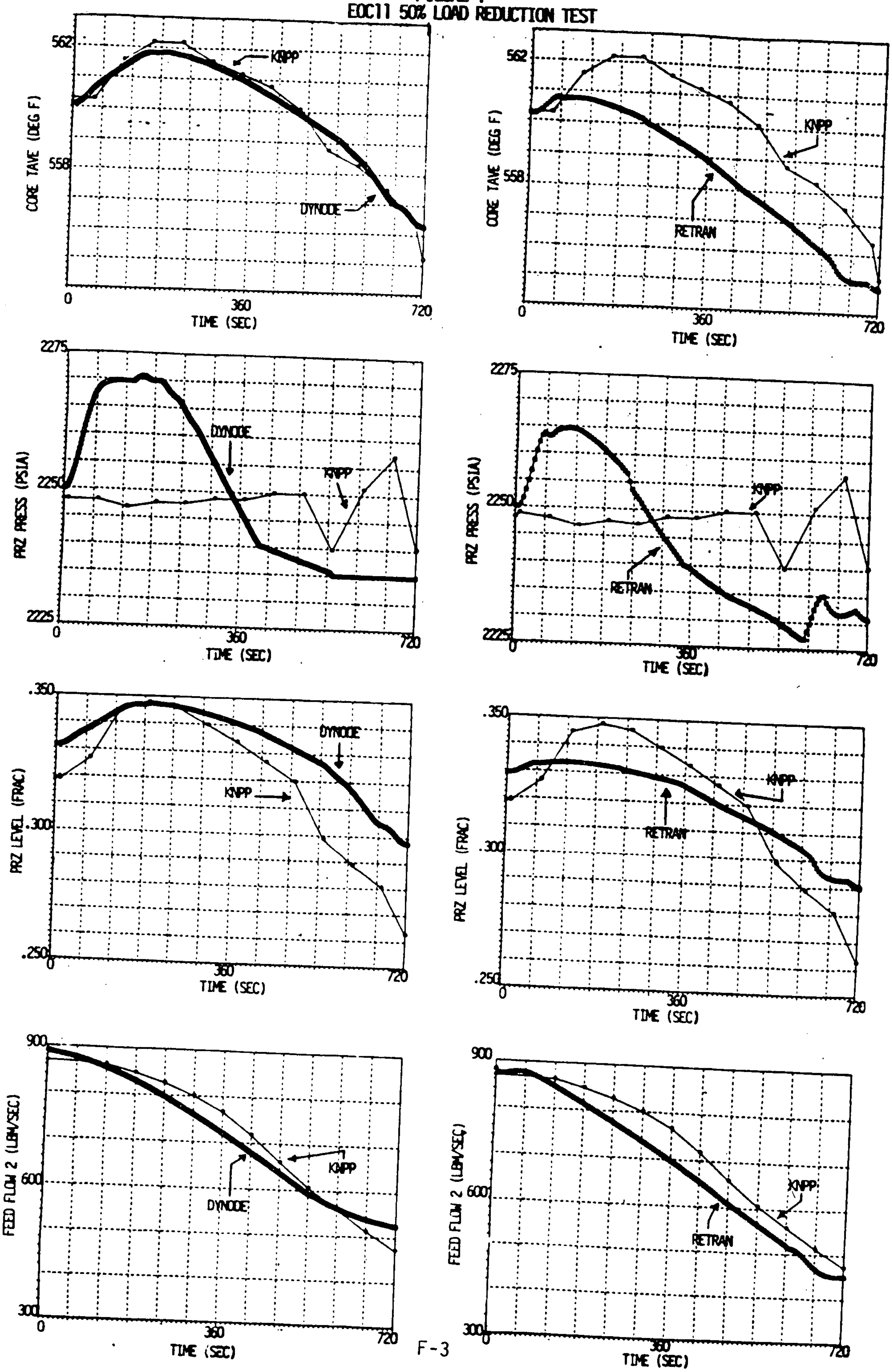


FIGURE F1 (Cont.)
EOC11 50% Load Reduction Test

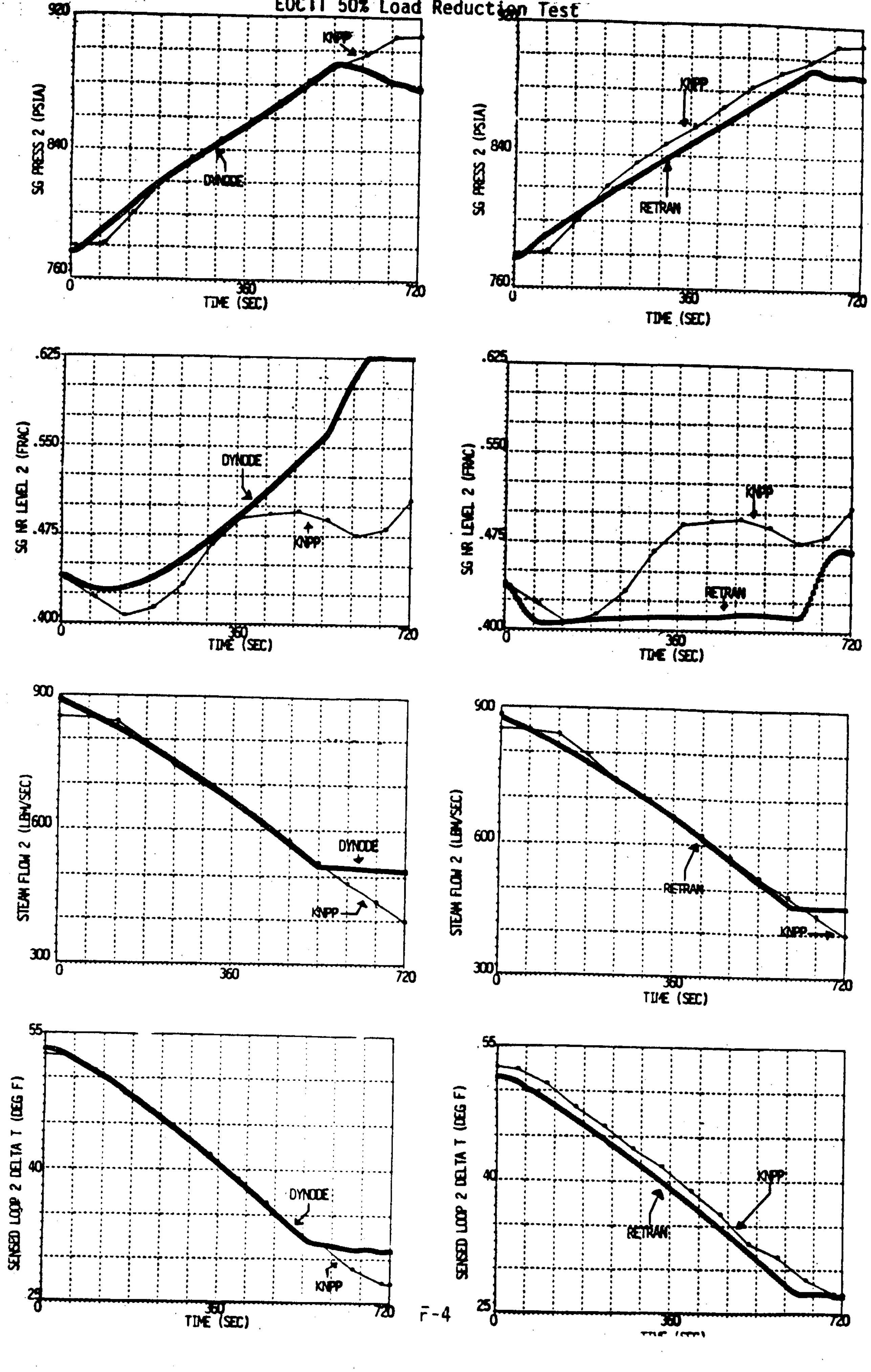
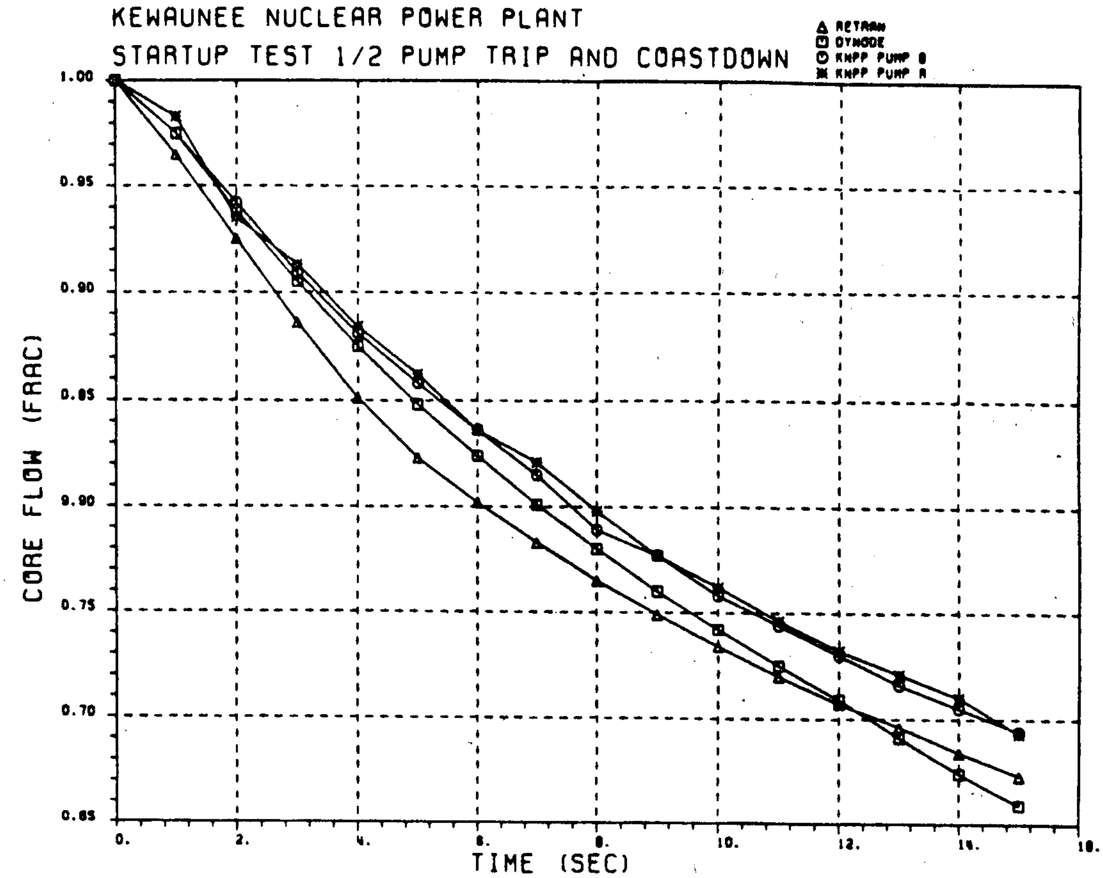


FIGURE F2
Startup Test Pump Trip



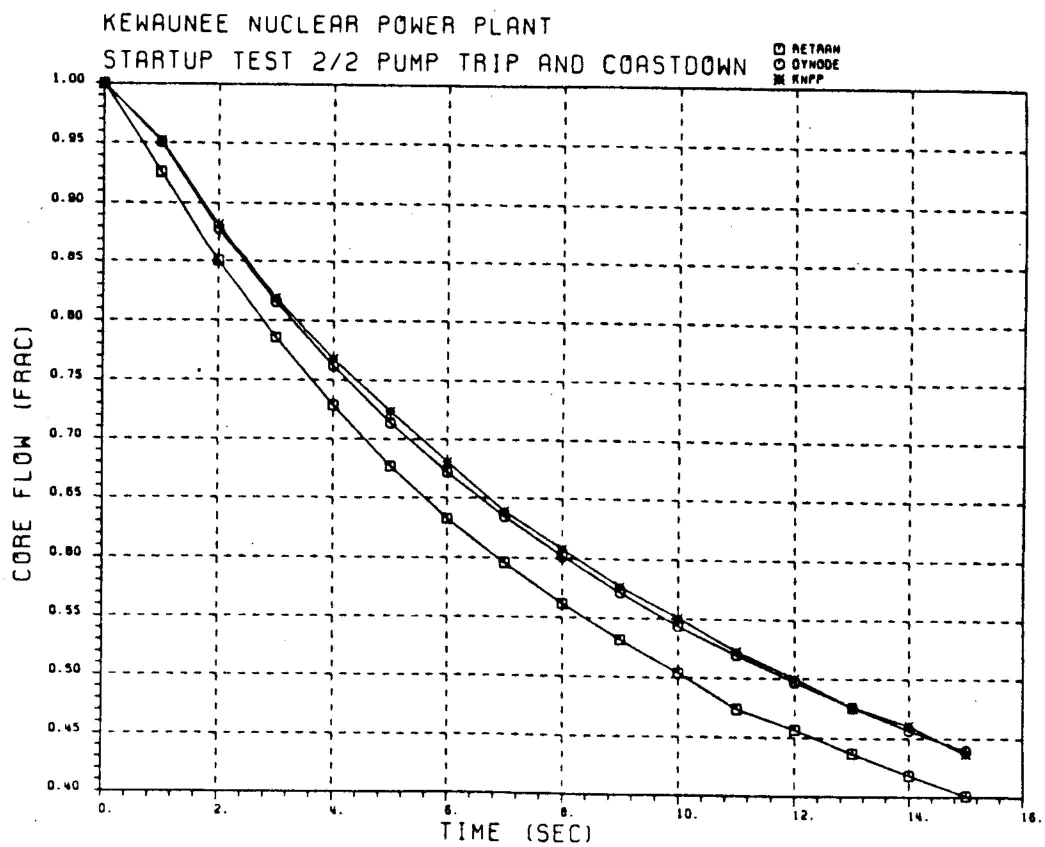


FIGURE F3

STARTUP TEST

50% LOAD REDUCTION

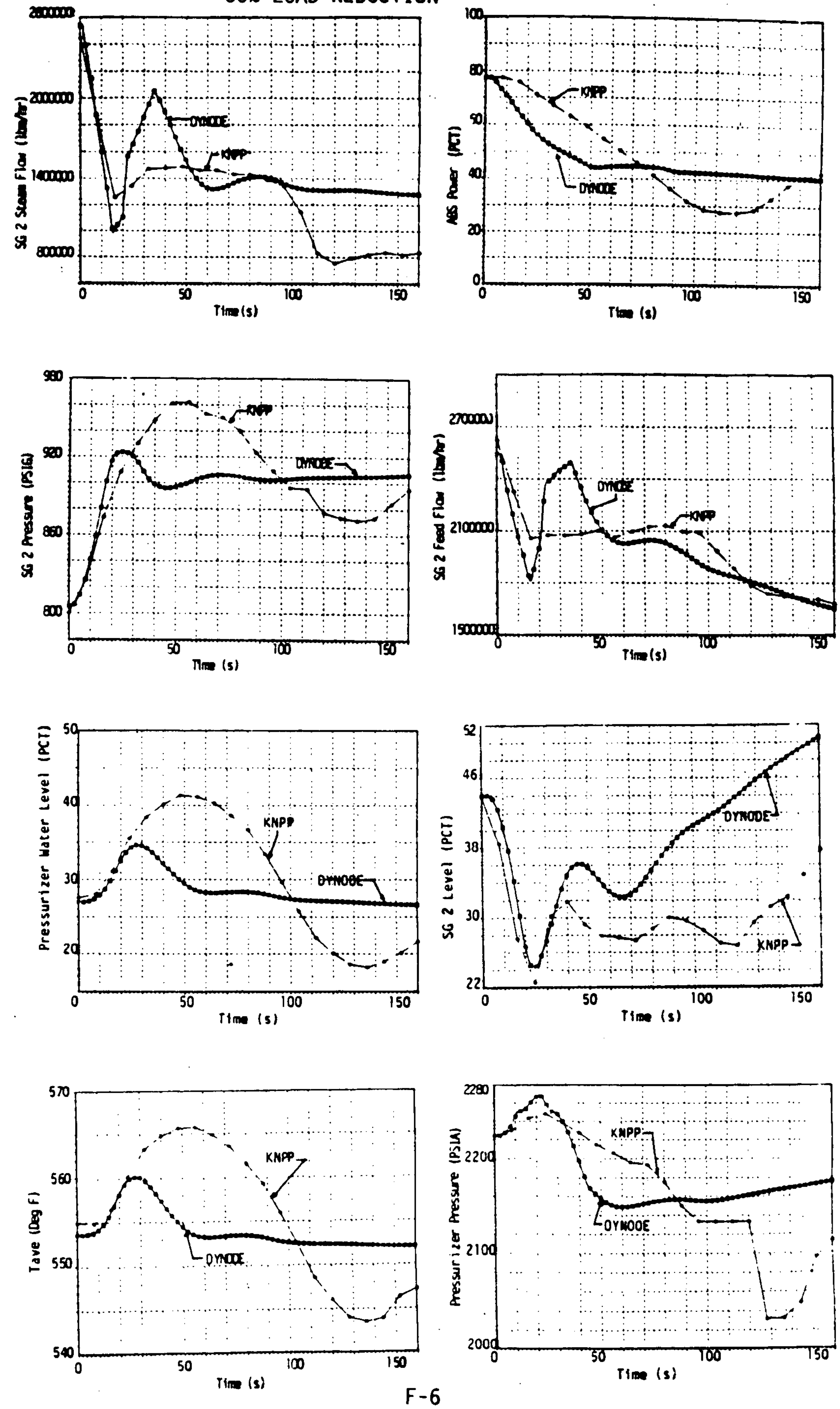


FIGURE F4
SIMULATOR COMPARISON - REACTOR TRIP

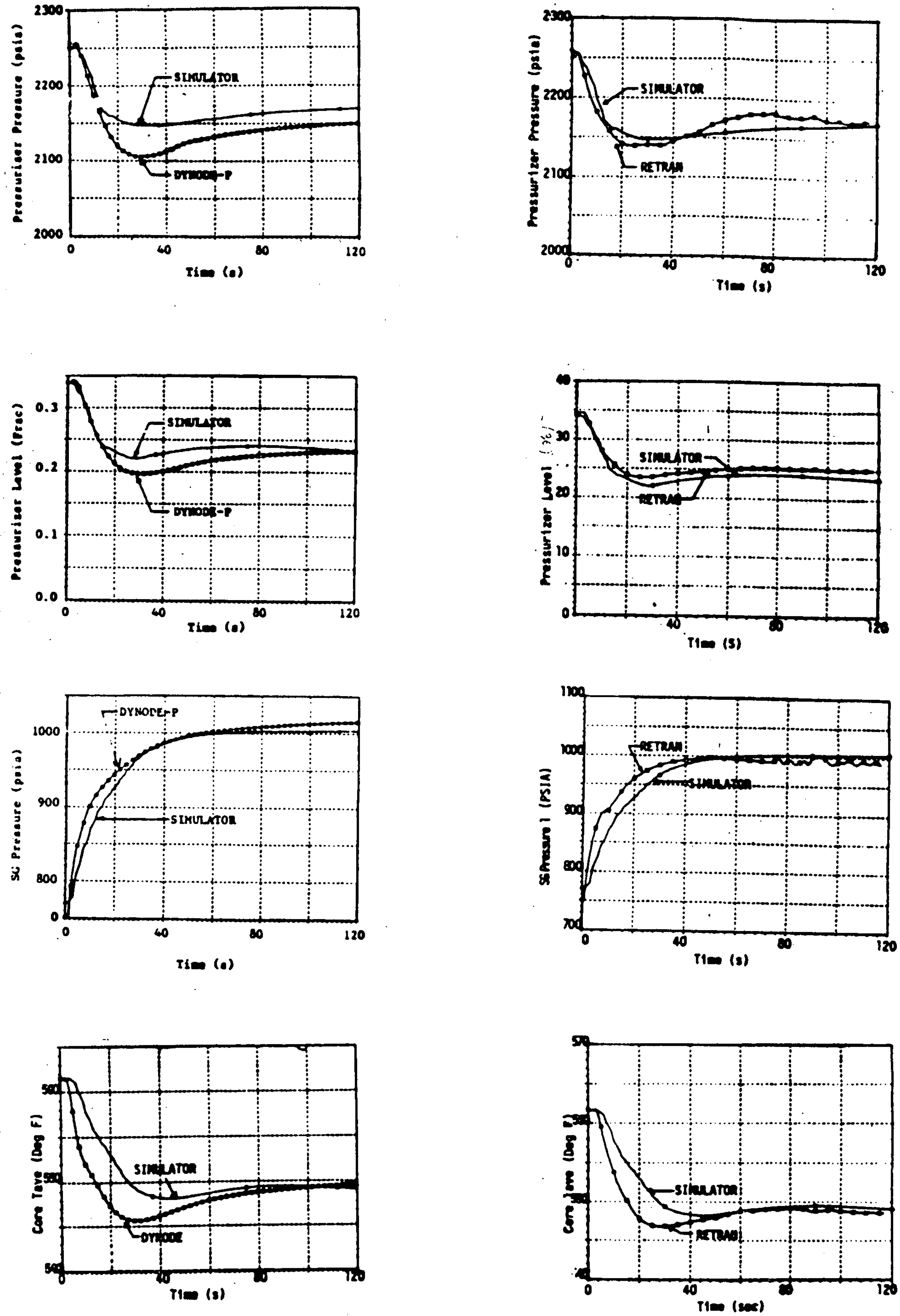


FIGURE F5
SIMULATOR COMPARISON
SG TUBE RUPTURE

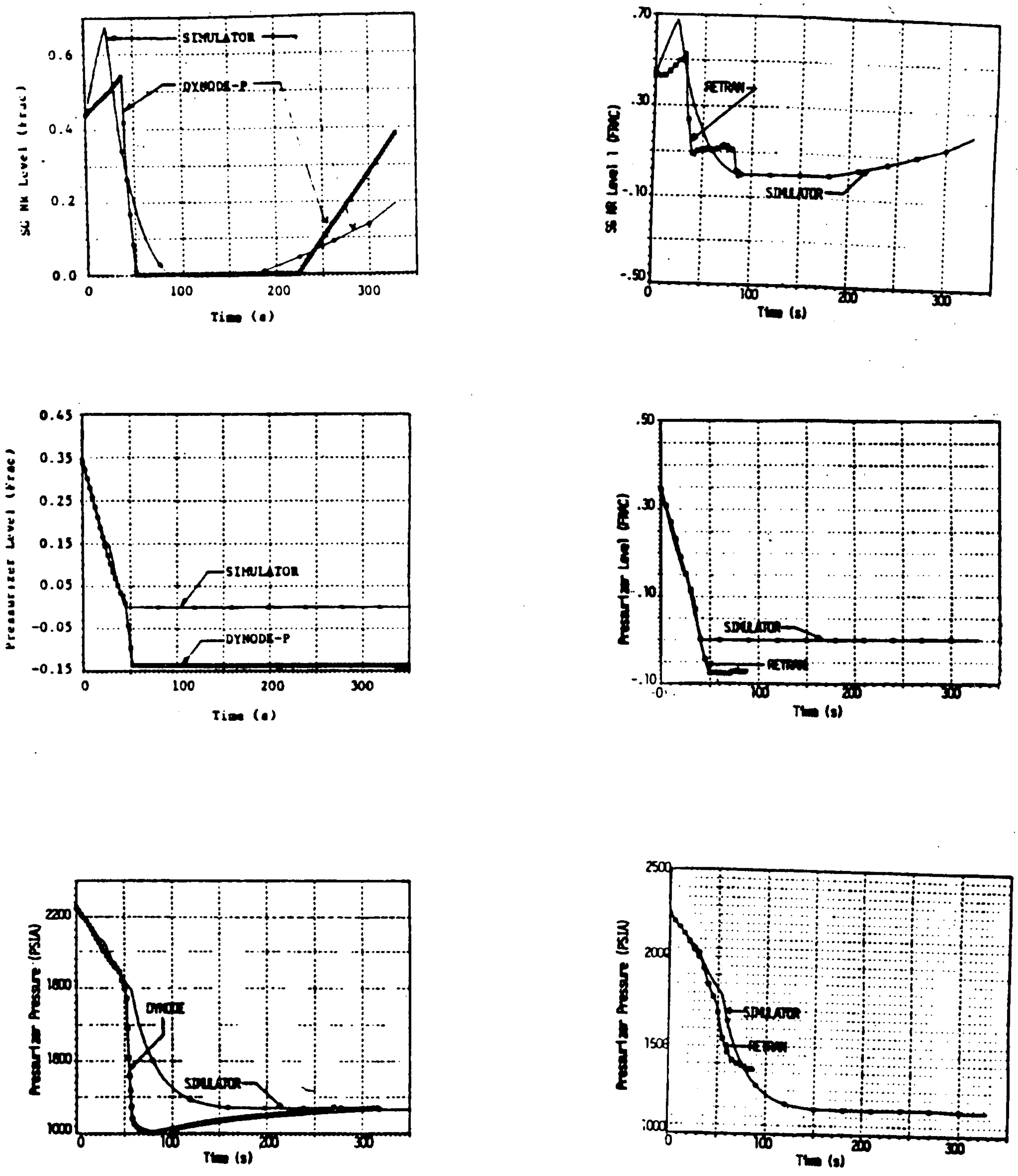
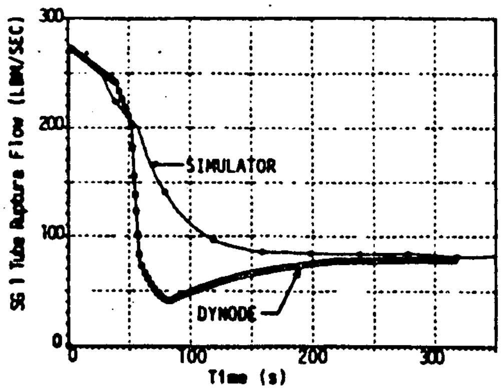
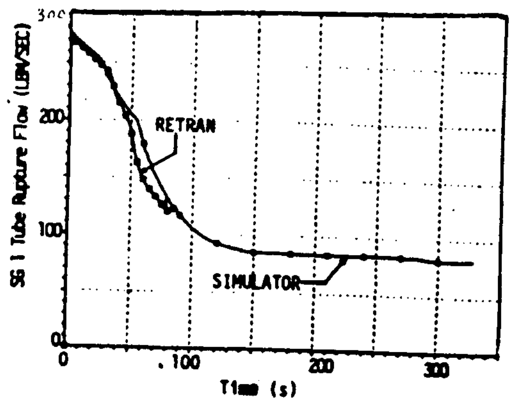
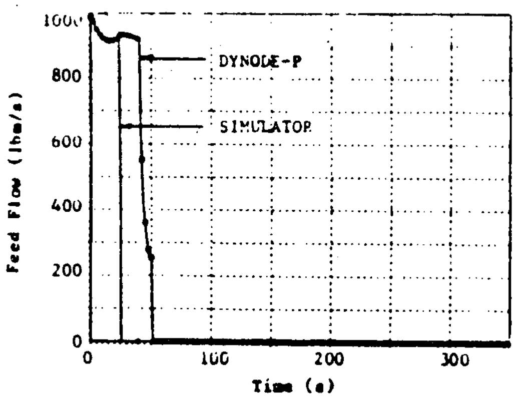
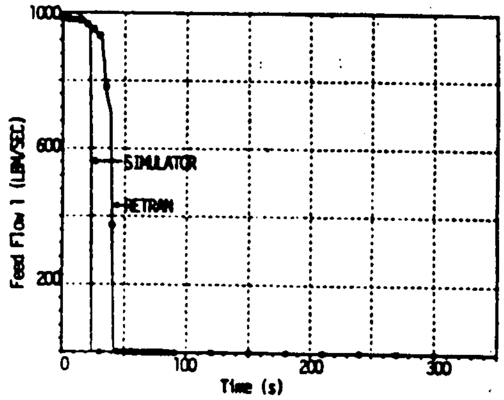


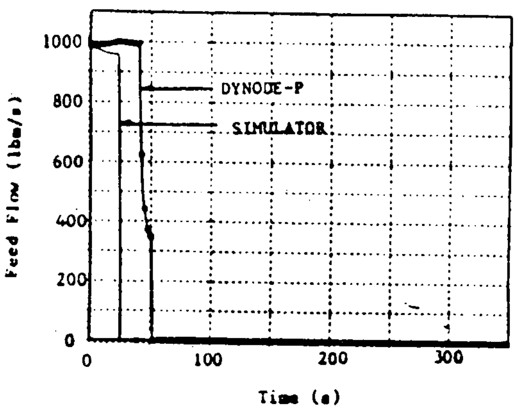
FIGURE F5 (cont.)
SIMULATOR COMPARISON
SG TUBE RUPTURE











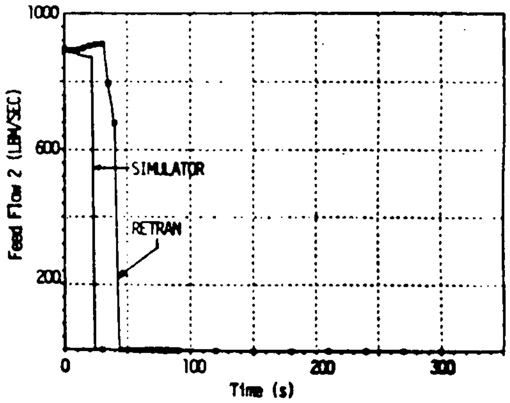


FIGURE F6

SIMULATOR COMPARISON

FAILED PORV

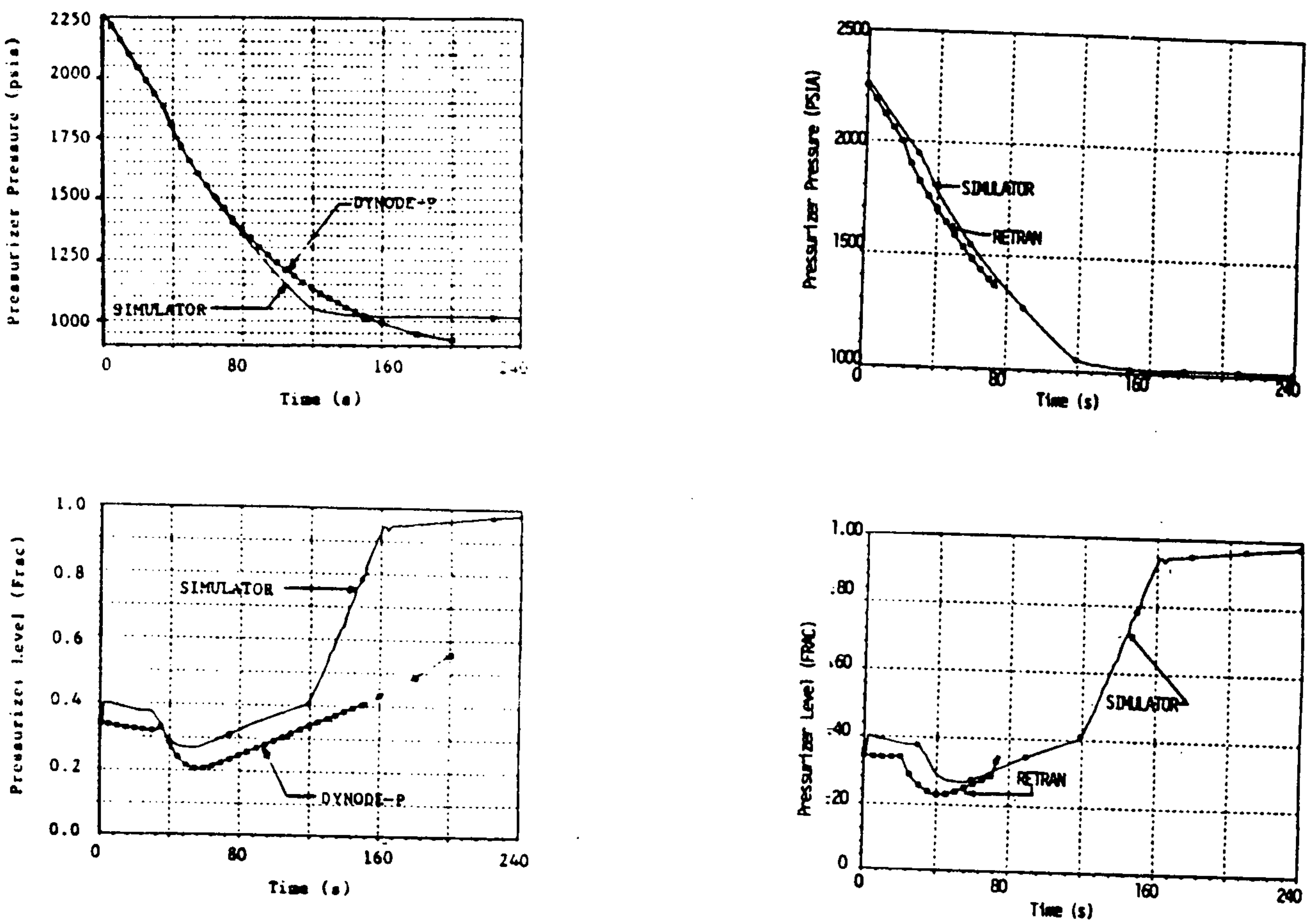


FIGURE F7

CYCLE 11 BOC 100% POWER FLUX MAP 1109

THERMOCOUPLE DATA

1.066 604.6 602.9 1.7				
1.284 -- 611.3 --	1.241 608.6 610.1 -1.5			
1.285 612.3 612.2 0.1	1.292 609.5 611.3 -1.8	1.145 -- 605.4 --		
1.267 610.0 609.8 0.2	1.160 601.4 606.5 -5.1	1.103 598.8 601.8 -3.0	1.012 -- 597.0 --	
1.054 -- 599.1 --	1.102 601.3 601.4 -0.1	1.160 -- 602.9 --	1.095 -- 596.2 --	0.493 569.1 569.5 -0.4
0.759 583.6 583.9 -0.3	1.092 -- 596.0 --	0.945 588.7 591.4 -2.7	0.468 568.8 567.4 1.4	
0.576 567.0 568.7 -1.7	0.332 -- 559.0 --			

POWER
T/C TEMP (DEG F)
VIPRE TEMP (DEG F)
DIFF (DEG F)

APPENDIX F

References

F1. R. C. Kern, et. al, "Validation of Nuclear Power Plant Simulators" proceedings of the International Conference on Power Plant Simulators", Curenavaca, Mexico, November 19-21, 1984, p. 307.

F2. J. Chao, et. al, "Using RETRAN-02 and Oynode-P to analyze Steam Generator Tube Breaks", NSAC-47, May, 1982.

F3. Kim Hammer, et. al, "An Analysis of Selected Operational Transients for the Kewaunee Pressurized Water Reactor", University of Wisconsin, Reactor Safety Research Report, May, 1986.

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APPENDIX G

Additional Information Requested by NRC

February 12, 1988

10 CFR 50.4

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Gentlemen:

Docket 50-305
Operating License DPR-43
Kewaunee Nuclear Power Plant
TAC No's. 65155 and 66562
Core Reload Safety Evaluation Methods

Please find attached the additional information requested with regard to the "Reload Safety Evaluation Methods for Application to Kewaunee, Revision 1," February 1987 (FIN A-3834). Should you require any additional information, please contact my staff.

Sincerely,

Mark L. Marchi for
D. C. Hintz
Vice President - Nuclear Power

PEM/jms

Attach.

cc - Mr. Robert Nelson, US NRC
US NRC, Region III

Attachment to

Letter from D. C. Hintz

To

Document Control Desk

Dated February 12, 1988

"Reload Safety Evaluation Methods for Application to Kewaunee, Revision 1," February 1987 (FIN A-3834)

Request for Additional Information

1. Describe how the present Reload Safety Evaluation Methods (RSEM) differ from the January 1979 version, and discuss the reasons for preparing a revised version of RSEM.

Response

The present RSEM are very similar to those submitted in the 1979 report. Key sensitive physics parameters and their required inequalities are for the most part unchanged. A significant revision to the safety analysis methods is the conversion from COBRA-IV to VIPRE-01 as the primary code for fuel thermal hydraulic analysis. The revised topical report discusses the sensitivity studies performed during the development of the VIPRE-01 model inputs.

Since 1979 WPS has developed RETRAN-02 as an additional system analysis code and has acquired experience in the development and application of best estimate transient analyses. Portions of the revised report discuss these subjects and present selected results.

Finally, the DYNODE-P model, the primary system analysis code, has undergone several upgrades since the 1979 report. The current code version used to generate the transient results in the revised report is DYNODE-P Version 5.4. The DYNODE-P Version 5.4 manual was provided to further describe this upgrade (1).

The upgraded mothods presented in the revised version of RSEM provide WPS a more complete and technically accurate evaluation of the performance and safety of the reload core. The major reason for the submittal of the revised report is to present transient analysis results using the current WPS methods and codes. WPS intends to use these methods and codes to support reload designs and Technical Specification revisions as required.

2. Since the Updated Safety Analysis Report (USAR) and the Kewaunee RSEM omploy distinct methodologies, Kewaunee RSEM safety analyses of reload cores that use the USAR as a reference analysis must ensure the following:

   (a) The differences in biases and reliability factors in the two methods are accounted for.

   (b) The definitions of safety analyses parameters used in the Kewaunee RSEM are consistent with those used in the USAR.

   How does Wisconsin Public Service Corporation (WPS) ensure (a) and (b) above?

   Response

   The Kewaunee USAR transient analyses are bounding analyses in which core and system parameters are assumed to be at conservatively limiting values. Additional biases and uncertainties, if any, are indeterminate. However, the results presented in the USAR represent the worst case results which were found acceptable to the NRC at the time of licensing.

   The RSEM require best estimate core safety parameter analyses assuming core configurations which are consistent with the evaluated transients. Biases and reliability factors are conservatively applied to the RSEM analysis results to account for model uncertainties. These uncertainty factors are determined by statistically comparing model predictions to measurements. The RSEM bias and reliability factors ensure conservative results with respect to plant responses. This is independent of the USAR analysis results. Therefore, by comparison of reload safety analysis results to bounding results which have been accepted as our licensing basis, we ensure that the actual plant performance under Chapter 14 assomptions will also remain acceptable.

   The definition of safety analysis parameters in the USAR are not presented in much detail. However, many of the basic parameters are elementary and are not subject to interpretation or inconsistency (i.e., $F_{\Delta H}$). The definition and application of the parameters presented in Section 2 of the Revised RSEM Report have been verified through comparison of mere detailed descriptions of the USAR analyses during the first four cycles of operation. These reloads were supported by Westinghouse, and reload safety analysis and core reload management reports were provided. These four reload safety analyses were duplicated by WPS and verified as consistent. In addition, prior to taking over responsibility for this activity, WPS contracted Westinghouse to provide formal instruction to WPS personnel in Westinghouse Reactor Safety Analysis methods in January, 1977. The results of this training were used to ensure consistency between reload evaluation methods.

3. Has WPS analyzed all the accidents discussed in the USAR, using the methods described in the Kewaunee RSEM Report, and determined that the design bases are met for the reference core?

Response

WPS has analyzed the majority of accidents discussed in the USAR including all RSE design basis (limiting) events with the exception of LOCA. In some cases, if a specific accident was determined to be bounded by another accident of the same class, the accident was not analyzed. By analysis of a limiting transient of each class as a minimum, WPS demonstrated and documented in the RSEM the ability to adequately understand and model the pertinent phenomena. The following list of USAR events identifies the accidents which have been analyzed by WPS. For all the accidents analyzed, the design bases were shown to be adequately met.

For a reload core, the design bases are verified provided the inequalities required by the RSEM are satisfied for each transient event. If the RSEM conditions are not satisfied, the transient is re-analyzed using a set of parameters which provides for a conservative calculation for the specific reload, and the results are compared to the evaluation criteria. Thus, should an accident which has not been specifically analyzed be required to be re-analyzed for a reload, WPS will analyze the accident at that time.

USAR Chapter 14 Accidents

USAR Section		RSEM Report Section	WPS Analyzed
14.1.1	Uncontrolled RCCA Withdrawal From a Subcritical Condition	3.1	yes
14.1.2	Uncontrolled RCCA Withdrawal at Power		
	Fast Rate 100% Power	3.2	yes
	Slow Rate 100% Power	3.2	yes
	Fast Rate 60% Power	3.2	yes
	Slow Rate 60% Power	3.2	yes
14.1.3	RCC Assembly Misalignment		
	G-7 dropped	3.4	yes
	J-10 dropped	3.4	yes
	K-7 dropped	3.4	yes
	H-8 dropped	3.4	yes
	L-8 dropped	3.4	yes
	Bank D fully inserted w/one RCCA fully withdrawn	3.3	yes
14.1.4	Chemical and Volume Control System Malfunction		
	Dilution during refueling	3.5	
	Dilution during startup	3.5	
	Dilution at power - automatic control	3.5	
	Dilution at power - manual control	3.5	yes
14.1.5	Startup of an Inactive Reactor Coolant Loop	3.6	yes
14.1.6	Excessive Heat Removal Due to FW System Malfunctions		
	BOL no control	3.7	yes
	EOL control	3.7	
14.1.7	Excessive Load Increase Incident		
	BOL no control	3.8	yes
	EOL no control	3.8	
	BOL control	3.8	
	EOL control	3.8	yes
14.1.8	Loss of Reactor Coolant Flow		
	Two-pump trip	3.11	yes
	One-pump trip	3.11	yes
	Locked rotor	3.12	yes

USAR Section		RSEM Report Section	WPS Analyzed
14.1.9	Loss of External Electrical Load		
	BOL no control	3.9	yes
	EOL no control	3.9	yes
	BOL control	3.9	yes
	EOL control	3.9	yes
14.1.10	Loss of Normal Feedwater	3.10	yes
14.1.11	Loss of AC Power to the Plant	3.10	yes
14.2.1	Fuel Handling Accidents		
	Oropped assembly or RCCA	3.13	
	Assembly stuck inside the reactor vessel	3.13	
	Assembly stuck in the penetration valve	3.13	
	Assembly stuck in the transfer tube or carriage	3.13	
14.2.2	Accidental Release - Recycle or Waste Liquid	N/A	
14.2.3	Accidental Release - Waste Gas		
	Gas decay tank rupture	N/A	
	Volume control tank rupture	N/A	
14.2.4	Steam Generator Tube Rupture	N/A	
14.2.5	Rupture of a Steam Pipe		
	Downstream of flow restrictor	3.14	yes
	Upstream of flow restrictor	3.14	yes
	Downstream of flow restrictor - loss of power	3.14	
	Upstream of flow restrictor - loss of power	3.14	
	Spurious opening of a safety valve	3.14	yes
14.2.6	RCC Assembly Ejection		
	BOL full power	3.15	yes
	EOL full power	3.15	yes
	BOL zero power	3.15	yes
	EOL zero power	3.15	yes
14.2.7	Turbine Missile Damage to Spent Fuel Pool	N/A	
14.3.1	Loss of Reactor Coolant From Small Ruptured Pipes or From Cracks in Larger Pipes		
	Four-inch	3.16	
	Six-inch	3.16	
	Three-inch	3.16	
14.3.2	Major Reactor Coolant Pipe Ruptures	3.16	

4. For each accident discussed in the Kewaunee RSEM Report, how does WPS determine that the set of safety analysis parameters discussed in the section on Reload Safety Evaluations is complete? In other words, how does WPS determine that variations in the value of a parameter not included in the set does not affect the consequences of the accident?

Response

The determination of the set of safety analysis parameters for each transient and accident considered in the Reload Safety Evaluation process has been made on the basis of two factors--namely, prior experience and training (see response to Question 2) of the engineers who established these sets, and computational results obtained from sensitivity studies performed specifically for the Kewaunee Plant. These determinations were made from a complete set of physics parameters and are based on the importance of each relative to affecting the results for the parameters which relate to the specific acceptance criteria as defined for each event in the RSEM topical report.

In addition, WPS has compared the RSEM parameter lists with the Westinghouse Reload Safety Evaluation Methodology (Reference 3) and found them to be consistent. Additional information in this regard is presented in the response to Question 2.

5. Table 1 contains a partial list of accidents discussed in the Kewaunee RSEM Report. Also listed for each accident are safety analysis parameters that are not included in the set of safety analysis parameters discussed in the reload safety evaluation of the accident, even though these parameters are expected to have a non-negligible effect on the consequence of the accident. Justify the omission in each case.

Table 1: List of Accidents and Omitted Safety Analysis Parameters

Accident	Omitted Parameter(s)
Uncontrolled RCCA Withdrawal from Subcritical Condition	ℓ^*
Control Rod Misalignment	α_D, α_M, integral and differential RCCA worths
Loss of Reactor Coolant Flow	β, initial fuel temperature
Loss of Reactor Coolant Flow-Locked Rotor	initial fuel temperature
Main Steam Line Break	F_Q
Control Rod Ejection	Doppler weighting factor
Loss of Coolant Accident	fuel rod temperature, fuel rod internal pressure, decay heat, densification spike factor, axial rod shrinkage

<u>Response</u>

A review of Table 1 in the Request for Additional Information indicates the following:

(a) Uncontrolled RCCA Withdrawal from Subcritical Conditions

This transient has a low probability for realistic positive reactivity insertion rates which would result in a prompt critical condition prior to reactor trip. In this case, as described in Section 3.1.2 of the Revised RSEM Report, the transient core power response is relatively insensitive to ℓ^* and is determined predominantly by the yields and decay constants of the delayed neutron precursors.

However, since this event is normelly analyzed with ultra-conservative reactivity insertion rates in which prompt criticality is achieved, ℓ^* is an important parameter, and as such, should be included in the RSE comparisons. Section 3.1.5 of the report will be revised to include this parameter.

(b) Control Rod Misalignment

This event is analyzed with steady-state methods since the analysis relates to either static rod misplacements or events which occur over time periods which are long compared to dynamic effects. Events relating to rapid rod motion are considered in the analyses involving control rod withdrawal, ejection, and drop. Thus, the parameters given in Table 1 do not affect the results.

(c) Loss of Reactor Coolant Flow

This event is analyzed at full power conditions with a conservatively small moderator temperature coefficient. The result is that only small power level changes occur prior to scram. Thus, β will have a very small impact on the power and heat flux responses since the minimum DNBR occurs shortly (within 1 second) after scram.

WPS recognizes the importance of fuel temperature in maximizing the heat flux response in the transient analysis. However, since fuel temperature is governed by power distribution limits it is not considered directly reload dependent. Also, the initial fuel temperature of the hot spot does not impact this analysis since the acceptance criteria preclude the occurrence of DNB during the event.

(d) Loss of Reactor Coolant Flow-Locked Rotor

Some percentage of fuel rods experience DNB in this event when it is analyzed with the conservative RSEM assumptions and, thus, initial fuel temperature will impact the transient hot spot analysis results. The hot spot fuel temperature transient analysis assumes a conservatively high initial value which bounds all reload cores.

Since fuel temperature is not directly reload dependent (see response to Question 5(c)), it is excluded from the RSEM comparisons.

(e) Main Steam Line Break

Currently, the acceptance criterion relating to fuel damage which WPS has adopted for this accident is that no fuel rods will experience DNB. Thus, F_Q, which would primarily affect the post-DNB fuel temperature response, is not of consequence for the WPS analysis.

(f) Control Rod Ejection

The Doppler Weighting Factor is a major parameter for use in the transient analyses to predict the correct plant response when 3-D models are not employed. However, for the reload evaluation, the unweighted value is computed for the core and compared to the unweighted value used in the bounding transient analyses. Refer to the response to Question 15 for additional information in this regard.

(g) Loss of Coolant Accident

The LOCA analysis is currently contracted by the reload fuel vendor. As a result, the fuel vendor places restrictions on plant operation (e.g., Technical Specification power distribution limits) which ensure that the fuel is maintained within the assomptions of the LOCA analysis. A reload core is acceptable provided the fuel adequately meets these constraints at all allowable operating conditions in the cycle (see response to Question 16).

The parameters identified in Table 1 for this accident do have a significant impact on the analysis results. However, these parameters relate to the thermal hydraulic and mechanical design aspects of the fuel (with the exception of decay heat) and not the physics aspects. The WPS RSEM cover only the latter parameters. Although not explicitly reviewed for the reload, these parameters are adequately bounded by the LOCA analysis assumptions, provided the applicable operating restrictions are adhered to.

6. Provide estimates for the uncertainties in the following calculated results for accidents analyzed with DYNODE-P:

(a) maximum vessel pressure
(b) minimum DNBR
(c) maximum fuel temperature

Response

Rigorous analyses of the uncertainties in the maximum reactor vessel pressure, minimum ONBR, and maximum fuel temperature which are obtained for accidents analyzed with DYNODE-P could be provided on the basis of extensive sensitivity studies. However, reasonable estimates that are based on the conservative inputs used, can be justified in light of the following considerations:

(a) The methodology which has been developed has been appropriately qualified by comparisons with other approved licensing results (USAR) and plant data.

(b) All the analyses which are performed with OYNODE-P are based on bounding calculations in which each important parameter is set to a limiting value which includes the uncertainty. As an example, the Locked Rotor system analysis assumes an initially high pressurizer pressure of 2280 psia, while the corresponding hot channel analysis assumes an initially low value of 2220 psia.

Since VIPRE-01 inputs are taken directly from DYNODE-P, conservatisms are included in the calculation of DNBR. In addition, the DNBR limit is conservatively set to 1.3 to ensure with 95% probability at a 95% confidence level that DNB is avoided.

This deterministic method is conservative as opposed to a less conservative method in which the uncertainties are combined in a statistical manner. Therefore, no additional penalties need be applied to the results of the analyses, which are based on DYNODE-P, to account for these uncertainties.

(c) Reasonable estimates of uncertainties are:

- Maximum Vessel Pressure ±2%, determined from plant and USAR transient comparisons.
- Minimum DNBR ±10%, determined by combining uncertainties of DNBR-related parameters at MDNBR conditions.
- Maximum fuel temperature ±5%, determined by comparison to detailed fuel rod codes such as COMETHE and ESCORE.

7. What are the margins of instrumentation error in coolant temperature, pressure and flow rate, and reactor power that are allowed in the DYNODE-P analyses?

Response

The DYNODE-P input assumptions are consistent with the USAR (i.e., input values are set to conservatively determined bounding values). DYNODE-P allows for a margin of instrument error equal to: ±4°F coolant temperature, ±2% reactor power, and ±30 psi primary pressure. Reactor coolant flow rate is assumed to be approximately 8% lower than plant measured flow rates. VIPRE, since it derives its system operating input directly from DYNODE, assumes the same input values.

8. How is Case #3 discussed in Section 2.5 (Shutdown Margin) utilized in conservatively adjusting the shutdown margin?

Response

Case #3 is used in conjunction with Case #2 to derive the worth of the control rods at hot zero power core conditions assuming all rods move from the full power insertion limit to the fully inserted position. Ten percent of this rod worth is then conservatively applied to the calculated shutdown margin to account for the rod worth uncertainty. At startup, the predicted rod worths must be within 10 percent of measured worth to verify the 10 percent uncertainty.

9. How is axial peaking within a node accounted for in the determination of F_Q (Section 2.7)?

Response

Axial peaking in a node is accounted for by applying statistical factors to the nodal calculated power. These statistical factors are derived from comparisons of measured to predicted reaction rates using the previous three operating fuel cycles. A description of these statistical factors is provided below. A more detailed discussion can be found in Reference 2.

Each flux map consists of 61 measured axial data points in 36 instrumented radial core locations. These 61 values are grouped into approximately 3 points per ½-node (a node is one foot in length). Reaction rates are computed by the nodal model and compared to the measured values from flux maps. Two sets of 24 factors are derived from these comparisons. The first set, D1, is used to split the nodal calculations axially into half-nodes and is derived as the average ratio of the calculated reaction rate in the half node at level L (L=1 to 24) to the average measured reaction rate in the half node. The second set, D2, is used to calculate the peak reaction rate within the half-node and is derived as the average ratio of the peak measured reaction rate to the average measured reaction rate within the half-node at level L. Thus, axial peaking is accounted for in the determination of FQ by applying D_1 and D_2 to the nodal calculated power.

10. How was the importance factor (I) conservatively determined to be 0.97 (Section 2.9)?

Response

The adjoint flux (importance flux) solution to the diffusion equation model of KNPP reactor cores was used to estimate the spatial importance of the delayed neutron yields. A 10% uncertainty is applied directly to the importance factor. This uncertainty is combined with other uncertainties to account for variations in spectrum, isotopes, etc. to yield a total uncertainty of 3% on delayed neutron fraction. (Refer to the response to question 11.) Additional detail can be found in Section 3.8 of Reference 2.

The calculated delayed neutron kinetics constants, with a 0.97 importance factor have yielded acceptable results during startup testing of the previous reload cores. These values are used for reactimeter input and the results are compared to independent reactivity measurements such as boron concentration.

11. Supply the bias and reliability factor for each safety analysis parameter discussed in Section 3.D.

Response

The bias and reliability factors used in the Cycle 13 reload safety evaluation are shown below:

Parameter	Reliability Factor	Bias
$F^{\Delta}H$	3.6%	0
Rod Worth	10.0%	0
Moderator Temperature Coefficient	4.68 PCM/°F	1.1 PCM/°F
Doppler Coefficient	10.0%	0
Boron Worth	5.0%	0
Delayed Neutron Parameters	3.0%	0

FQN Reliability Factors

Core Level	RF (%)
1 (Bottom)	19.94
2	8.46
3	5.B2
4	5.46
5	5.74
6	5.09
7	4.92
8	5.34
9	5.06
10	5.45
11	4.93
12	5.00
13	4.55
14	4.60
15	4.53
16	4.60
17	4.69
18	4.69
19	5.42
20	5.65
21	B.17
22	7.81
23	15.09
24 (Top)	15.57

12. Justify augmenting a local peaking factor such as F_Q or $F_{\Delta H}$ with a global parameter such as T, the technical specification tilt limit (Section 3.0).

Response

The statistically determined local peaking augmentation factors will have inherently included any small core tilts (typically on the order of 0.5%) that may have been present during the measurements of the reaction rates (see response to question 9). However, since the excore detector quadrant power tilt monitor is set to alarm at 2%, slightly higher local peaking factors could be present between monthly core surveillance without a corresponding alarm.

Increasing the peaking factor linearly by 2% is adequate given that there would not be any local perturbation such as a control rod misalignment or a large uncertainty in the excore instrumentation setpoint caused by drift.

In the first case, the rod position deviation monitors alarm if control rods are not within a few steps of their demand signal. Additionally, individual rod position indicators are checked at least once each shift as required by technical specifications.

In the second case, at least once a quarter surveillance is performed on the excore Nuclear Instrument System (NIS) detectors. According to the surveillance procedure, the excore detectors are calibrated to the movable incore detector (MID) flux measurements. This calibration normally eliminates any excore detector tilt caused by detector drift. WPS operating experience has shown that the excore detector tilt will typically increase from D% to, at most, 0.3% between calibrations.

Thus, it is highly unlikely that a control rod misalignment or a core tilt of any significance could go undetected. A multiplier of 2% on local peaking factor, in addition to the statistical uncertainty factors, provides a conservative margin on peaking factor predictions to cover operation with small credible tilts.

13. Over what ranges is the xenon distribution varied to determine the xenon distribution that causes the minimum shutdown margin (Section 3.1.4 (d)?

Response

There are a number of xenon distributions assumed in the search for the reload minimum shutdown margin. In the axial direction, xenon distributions are varied from a very negative axial offset (more flux in the bottom of the core) to a very positive axial offset (more flux in the top of the core). These axial distributions are applied to both full and zero power conditions. Different xenon concentrations are also examined in the shutdown margin evaluation. No xenon, equilibrium xenon, and transient xenon including peak xenon after shutdown and transient xenon after control rod movemant are examples of the distributions analyzed.

WPS reload core evaluation experience has demonstrated that typically the minimum shutdown margin will occur at the end of cycle at full power in a positive axial offset core condition. This configuration has yielded the minimum shutdown margin for previous operating cycles.

14. Why is the Control Rod Drop accident not analyzed assuming automatic as well as manual mode of control (Section 3.4.1)? Describe in detail the analysis performed to determine (a) the least rod worth that will trip the negative flux rate scram system, and (b) the consequences of the transients for which a trip does not occur. Specify how key safety parameters are chosen in both parts of the analysis to ensure conservatism. How are the rod drop concerns contained in the letter dated November 28, 1979, from the Nuclear Regulatory Commission to affected utilities addressed?

Response

In November, 1979 Westinghouse notified NRC and affected licensees of a concern that FSAR analyses of the control rod drop events may not represent the most limiting DNB ratio, since a reactor trip on negative flux rate could not be assured in all cases. A meeting was held with NRC and an interim solution was agreed upon (4). WPS committed to the interim solution and has procedures in effect which implement it (5).

In summary, since the automatic rod control is administratively limited by constraints on power (less than 90%) and control rod bite (greater than 215 steps), the concomitant power overshoot is mitigated. The consequences of a rod drop event under these restrictions are therefore as currently described in Section 3.4.1 (i.e., no reactor trip and no power overshoot assumed). Thus, the current analyses in manual control is sufficient to bound the operating restrictions in effect.

WPSC has participated in a Westinghouse Owners Group study to establish a generic methodology which could be used to analyze reload cores for the dropped rod event without a direct reactor trip. This methodology is currently under NRC review (6). Upon NRC approval of this methodology, our intent is to incorporate it into the cycle-specific reload analyses procedures and thereby eliminate the need for the current restrictions on automatic rod control. Since this analysis does not take credit for the negative rate trip, development of calculational procedures to analyze sensitivity of the rate trip to dropped rod worth would not be germane at this time.

15. What was the control rod ejection velocity assumed in the analysis of the Control Rod Ejection accident? Justify the use of Doppler weighting factors of 1.3 and 1.6 at zero power and full power, respectively (Section 3.15.2)

Response

The control rod ejection velocity is assumed to be 120 ft/sec (i.e., the time to eject a fully inserted control rod is 0.1 sec).

The values of the Doppler weight factors of 1.3 and 1.6 which are given in Section 3.15.2 were the values which were obtained for the model during the qualification phase (i.e., comparison to USAR analyses). When a three-dimensional space-time kinetics calculation is not performed, these weighting factors are applied to the core average doppler reactivity to account for spatial feedback effects.

For reload-specific evaluations, the unweighted Doppler reactivity coefficient is computed for the reload core and compared to the unweighted value used as input to the rod ejection transient analysis.

In the case that it is necessary for WPS to perform reload specific transient analyses for this accident, the Doppler reactivity feedback parameter which will be input to DYNODE-P will be conservatively calculated using the 3-D nodal code and will thereby account implicitly for the weight factor.

16. Describe the review process that the current docketed analysis of the loss of coolant accident undergoes to determine its applicability to a given reload cycle (Section 3.16.3).

Response

The fuel vendor responsible for the LOCA analysis reviews the plant performance and operational characteristics and the Technical Specifications for the proposed reload. Provided there are no changes which affect the LOCA results or which allow the plant or fuel to be outside the LOCA assomptions, the current docketed LOCA analysis is applicable to the reload cycle.

The reload designer must verify that the reload-dependent parameters are adequately bounded by the LOCA analysis assumptions. Since the reload design is constrained by the design inputs such as fuel design, Technical Specifications, and safety analyses (including LOCA analysis), there are few parameters under control of the reload designer which can impact the LOCA analysis. (Refer to the response to Question 5g).

As described in Section 3.16, the parameters determined by the reload which are sensitive to LOCA analyses are scram worth, $F_{\Delta H}$, and FQ. The bounding scram curve is input into the LOCA analyses. This is the same bounding scram curve used in the non-LOCA transients. The review of scram reactivity in regard to LOCA is, therefore, done as described in Sections 3.16.4 and 3.16.5.

Peaking factors are analyzed at various conditions ranging from beginning of cycle to end of cycle. The maximum values are chosen from those core conditions allowed by Technical Specifications (T.S. Section 3.10). These constraints are power distribution control strategy, control rod insertion limits, and maximum peaking factor limits.

The FQ values, including the reliability factors, at each core elevation are chosen as the maximum under equilibrium conditions from the cases described above. A conservative function, V(z), is applied at each elevation to account for non-equilibrium axial power variations. This function was determined by investigating the changes in FQ during core axial power perturbations induced by the cembination of power and control rod maneuvers (further detail can be found in Reference 12 to the RSE methods topical). The resultant axial distribution of FQ is compared to the limiting distribution used in the LOCA analyses. This limiting LOCA input is depicted as the solid line in figure 3.17.1 of the RSEM topical.

The current docketed loss of coolant analysis is applicable to the reload, provided these above reload sensitive parameters are bounded by LOCA analyses assumptions.

17. Has the fuel misloading accident been analyzed for the Kewaunee Nuclear Power Plant?

Response

WPS has not analyzed the fuel misloading accident because adequate multiple controls are in place to ensure proper loading of the reactor core. These controls include:

(a) WPS reviews the core shuffle procedure prepared by Westinghouse and a WPS representative monitors the fuel movement to assure compliance with the procedure.

(b) After the reload, a video-taped map is made and reviewed by Westinghouse, WPS Operations, WPS Reactor Engineering, and WPS QC.

(c) A spent fuel pool piece count is performed after the reload to verify the spent fuel pool inventory. Additionally, once per year the spent fuel pool is video-tape mapped to verify fuel assembly and insert locations.

(d) Throughout fuel movement, a WPS Senior Reactor Operator is present on the manipulator crane.

In the unlikely event of a power distribution or reactivity anomaly, the records generated by the above controls would be reviewed and any discrepancy would be analyzed specifically.

18. What quality assurance program does WPS intend to use to ensure consistency in the application of VIPRE-01?

Response

WPS currently uses VIPRE-01 MOD-01, which is the version reviewed and approved by NRC. To ensure consistency in application of VIPRE-01 for safety analyses, WPS intends to abide by the quality control procedures established by EPRI and which the Utility Group for Regulatory Approval committed to for the VIPRE-01 code (see Section 2.6, Reference 10 of WPS RSEM Report).

Internal responsibilities for control of the computer codes are under the auspices of the WPS Operational Quality Assurance Program which requires compliance with ANSI/ASME N45.2.11 and 10CFR50, Appendix B. Implementation is required to be by written procedure.

VIPRE-01 will be controlled by Fuel Management Procedures (specifically, FMP series 5.3) which govern control and documentation of computer codes. These procedures are audited annually by WPS Quality Assurance and have been reviewed by NRC as well (see Reference 7, RSEM Report). Future modifications to VIPRE-01 will be performed under the FMP controls and in accordance with EPRI procedures as upgraded versions become available.

19. If a profile fit subcooled boiling model (such as LEVY and EPRI models) which was developed based on steady state data, is used in boiling transients, care should be taken in the time step size used for transient analysis to avoid the Courant number less than 1.0.

Response

The WPS VIPRE-01 model uses the profile fit LEVY subcooled void correlation. Courant number (Nc) is ensured greater than 1.0 for all transient events by selecting time step sizes which are greater than the bounding minimum time step. This bounding minimum time step is derived using the minimum velocity transient event, the locked rotor accident.

A review of the locked rotor analysis results indicates that the minimum velocity in the core during the transient, including uncertainties, is approximately 6.3 ft/sec. Since Nc must be greater than 1.D, the limiting time step size is computed as follows:

$$Nc = \frac{V\Delta t}{\Delta x}$$

Where: Δx = axial node size
V = velocity
Δt = time step

If $Nc > 1.0$, then $\Delta t > \frac{\Delta x}{V}$

for all events Δx = D.333 ft.
for the locked rotor event minimum V = 6.3 ft/sec
therefore, $Nc > 1.0$ provided $\Delta t > .053$ sec

A review of the USAR event analyses shows that in all cases time steps are greater than .053 sec (typically 0.2 sec. is the minimum time step used). Thus, Nc is assured greater than 1.0 for the minimum velocity event and has even greater margin for those events with larger coolant velocities.

References

1. DYMODE-P, Version 5.4 User Manual, UAI 83-49, November 1, 1983.

2. Letter from E. W. James (WPS) to A. Schwencer (NRC) transmitting WPS, Kewaunee topical report entitled, "Qualification of Reactor Physics Methods for Application to Kewaunee," October, 1978.

3. WCAP-9272, "Westinghouse Reload Safety Evaluation Methodology," dated March, 1978.

4. NRC Summary of November 19, 1979, meeting with Westinghouse and Licensees regarding Dropped Rod Protection.

5. Letter from E. R. Mathews (WPS) to D. G. Eisenhut (NRC), dated November 21, 1979.

6. Letter from R. A Newton (WOG) to J. Lyons (NRC) transmitting WCAP-11394.

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NRC-88-33

EASYLINK 62891993

March 7, 1988

U. S. Nuclear Regulatory Commission
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Washington, D.C. 20555

Gentlemen:

Docket 50-305
Operating License DPR-43
Kewaunee Nuclear Power Plant
TAC #65155
Additional Information on Core Reload Safety Evaluation Methods

References: 1. Letter from D. C. Hintz (WPSC) to NRC Document Control Desk dated March 27, 1987
2. Letter from D. C. Hintz (WPSC) to NRC Document Control Desk dated February 12, 1988

Reference 1 submitted for Nuclear Regulatory Commission (NRC) review Revision 1 of the topical report entitled "Reload Safety Evaluation Methods for Application to Kewaunee." Subsequently, the NRC staff posed nineteen questions which Wisconsin Public Service Corporation (WPSC) answered in reference 2. The purpose of this letter is to respond to six additional questions from the NRC staff.. The six questions, along with WPSC's responses, are attached to this letter.

The only remaining WPSC response concerning the topical report is scheduled to be submitted on March 15, 1988. This submittal will provide the basis for using the W-3 correlation and safety limit with the VIPRE-01 computer code as described in the topical report. The March 15 date was agreed upon in telephone conversations with the NRC staff.

Sincerely,

D. C. Hintz
Vice President - Nuclear Power

KAH/jms
Attach.
cc - Mr. Robert Nelson, US NRC
US NRC, Region III

Attachment

To

Letter from D. C. Hintz (WPSC) to NRC Document Control Desk

Dated

March 7, 1988

Additional Information on Core Reload Safety Evaluation Methods

"Reload Safety Evaluation Methods for Application to Kewaunee, Revision 1," February 1987 (FIN A-3834)

Request for Additional Information (February 16, 1988)

1. Briefly describe the decay heat model in DYNODE-P. Is the 1-D kinetics model in DYNODE-P used for the analysis of any accidents? If it is used, describe how the input for the 1-0 model differs from that for the point kinetics model.

Response

The decay heat model in DYNODE-P is similar in concept to the delayed neutron model. By defining a "concentration" for each decay heat group and interpreting γ_i as yield fraction the decay heat precursors can be represented by:

$$\frac{dqi(t)}{dt} = \gamma_i\, n(t) - \lambda_i\, q_i(t)$$

for $i = 1, \ldots\ldots, 11$

Where:

q_i = concentration of the i^{th} decay heat group
γ_i = yield fraction of the i^{th} group
$n(t)$ = normalized reactor power
λ_i = decay constant of the i^{th} group

The power density in the core at time t is given by:

$$Z(t) = (1 - \sum_{i=1}^{11} \gamma_i)n(t) + \sum_{i=1}^{11} \lambda_i qi(t)$$

Where:

$Z(t)$ = power density time t

The DYNODE-P decay heat source is represented by a polynomial fit of eleven exponentials. The polynomial fit constants correspond to the ANSI 5.1 (1971) standard data.

The 1-D kinetics model in DYNODE-P is used for the analysis of the Control Rod Ejection Accident. The basis for generating the 1-0 kinetics parameters is the same as for the point kinetics model. The only difference is that the 1-D model requires the specification of the spatially dependent neutronic parameters, while the point kinetics model requires core average values.

The methodology for generating the 1-D kinetics parameters from the 3-D model is described in Reference 1 (a copy of which is enclosed). This methodology assures consistency between the 3-D and 1-D models for reactivity effects and describes the calculations for the initial k_∞ and M^2 distributions.

Kinetics feedback parameters relating to moderator density and fuel temperature are calculated in a conservative manner and account for the spatial weight factors (Ref. 2) and model uncertainties associated with the 3-D model. The reactivity addition due to the ejection of the control rod is represented as a linear time dependent core reactivity change based on a conservative velocity (Ref. 2). Scram reactivity is represented explicitly by moving the control rods into the core assuming the highest worth rod is stuck. Both the ejected rod and scram worths conservatively account for 3-D model uncertainties. The remaining neutronic parameters are handled in the same manner as for the point kinetics model.

Thus, the underlying principle of utilizing a conservative and bounding deterministic method of analysis is the same when either the point or 1-D kinetics model is used.

Reference 1: R. C. Kern, "Methods and Guidelines for Obtaining One-Dimensional Nodal Constants for DYNODE-P from Three-Dimensional Nodal Calculations," NAI 82-46, Rev. 0, August 9, 1982. (Attached)

Reference 2: Response to Question 15, NRC Request for Additional Information, WPSC letter dated February 15, 1988.

2. Are any transients analyzed using user-specified reactor coolant systems flow rates (IPUMP=0 option) in DYNODE-P?

Response

The following events are analyzed with user-specified reactor coolant system flow rates (IPUMP=0) since there is no significant change in flow during the event:

Uncontrolled Rod Withdrawal from Subcritical
Uncontrolled Rod withdrawal at Power
Chemical and Volume Control System malfunction
Excessive Heat Removal due to FW System malfunction
Loss of External Electrical Load
RCC Assembly Ejection

The benchmark analyses have shown that for these events the user-specified flow assumption is adequate.

For those events in which significant flow changes do occur, such as the Loss of Flow, Loss of Feedwater (with loss of power), the Idle Loop Startup, and the Steam Line Break, the dynamic pump model is used.

3. Describe typical coolant channel and fuel rod nodalizations in the core used in DYNODE-P calculations.

Response

The fuel rod representation in DYNODE-P at the average axial power location consists of a discrete radial nodalization within the oxide and cladding regions. The oxide region is divided into five equal volume nodes, while the cladding region is represented by two nodes. Axially, the coolant channel is divided into twelve nodes--each one foot in length.

4. List the accidents for which TODEE2 analyses were actually used.

Response

WPS uses TOODEE-2 for the Rod Ejection and Locked Rotor Accidents since DNB occurs in these events when analyzed with the conservative licensing basis assumptions. TOODEE-2 is used to evaluate the hot spot fuel rod thermal response.

5. Provide typical values for the following input parameters used in TODEE2 calculations:

   a) effective roughness of fuel and cladding
   b) estimated hot gas pressure
   c) mole fractions of helium and xenon in gap
   d) oxide thickness at a fuel clad node.

   Response

   TOODEE-2 has the following input values:

.059 mils	effective roughness
784 psia	estimated hot gas pressure
1.0	mole fraction of helium in gap
0.0	mole fraction of xenon in gap
0.0	oxide thickness at a fuel clad node

   Our experience has shown that these inputs yield conservative licensing calculations.

6. The nuclear parameters presented in Table C1 (Final WPS VIPRE Model) are the power peaking factor limits in the current technical specifications, and are, therefore, the limiting values of these parameters expected under normal operating conditions. Justify the assumption that they are also limiting for all accident conditions that may be analyzed with VIPRE-01.

   Response

   The nuclear peaking factors presented in Table C1 refer to full power initial conditions since DNBR-limiting accidents are typically initiated from full power. These values were used in the USAR benchmark analyses and are slightly larger than the current Technical Specification limits. In future licensing calculations, WPS intends to use the Technical Specification nuclear peaking factor limits or values which conservatively bound those limits.

   Under normal operating conditions, assuming the plant is in compliance with the Technical Specifications, the Tech. Spec. nuclear peaking limits will conservatively bound the majority of VIPRE-01 analyzed accidents. In accidents where the Table C1 nuclear parameters are not limiting (e.g., Steam Line Break, RCC Misalignment, RCC Drop and Rod Ejection), the VIPRE-01 peaking factors are conservatively increased to bound the conditions expected during the event, thus ensuring a conservative DNBR evaluation.

   Changing core conditions in the transient are conservatively accounted for by applying core average responses from DYNODE-P as boundary conditions to the VIPRE-01 hot channel analyses.

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NRC-88-36

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March 16, 1988

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Gentlemen:

Docket 50-305
Operating License DPR-43
Kewaunee Nuclear Power Plant
TAC #65155
Additional Information on Core Reload Safety Evaluation Methods

References:
1. Letter from D. C. Hintz (WPSC) to NRC Document Control Desk dated March 27, 1987
2. Letter from D. C. Hintz (WPSC) to NRC Document Control Desk dated February 12, 1988
3. Letter from D. C. Hintz (WPSC) to NRC Document Control Desk dated March 7, 1988

Revision 1 of the topical report entitled "Reload Safety Evaluation Methods for Application to Kewaunee" was submitted for Nuclear Regulatory Commission review on March 27, 1987 (reference 1). Subsequently, the NRC staff requested additional information which was supplied by Wisconsin Public Service Corporation in references 2 and 3. The purpose of this letter is to answer the remaining NRC question concerning the topical report (reference 1).

Attachment 1 to this letter provides justification for the use of the VIPRE-01 computer code with the W-3 correlation and the 1.3 minimum departure from nucleate boiling ratio (MDNBR) safety limit. This ensures that VIPRE-01 will give appropriately conservative MDNBR results when applied as described in the topical report.

Attachment 2 to this letter is an Advanced Nuclear Fuels report which is referenced by attachment 1. Advanced Nuclear Fuels considers information contained in attachment 2 to be proprietary. In accordance with the Commission's Regulation 10 CFR 2.790(b), the enclosed Affidavit (attachment 3) executed by Mr. H. E. Williamson of Advanced Nuclear Fuels provides the necessary information to support the withholding of the information in attachment 2 from public disclosure.

Accordingly, it is respectfully requested that the information which is proprietary to Advanced Nuclear Fuels Corporation be withheld from public disclosure in accordance with 10 CFR 2.790.

Correspondence with respect to the proprietary aspects of the Advanced Nuclear Fuels Affidavit should be addressed to R. A. Copeland, Manager, Reload Licensing, Advanced Nuclear Fuels Corporation, 2101 Horn Rapids Road, P.O. Box 130, Richland, Washington 99352-0130.

Sincerely,

Mark L. Marchi for

D. C. Hintz
Vice President - Nuclear Power

KAH/jms

Attach.

cc - Mr. Robert Nelson, US NRC
US NRC, Region III

Attachment 1

To

Letter from D. C. Hintz (WPSC) to NRC Document Control Desk

Dated

March 16, 1988

Additional Information on Core Reload Safety Evaluation Methods

TAC #65155

NRC Request for Additional Information

Question

Justify the use of VIPRE-01 with the W-3 CHF correlation and the 1.3 MDNBR safety limit by showing that given the correlation data base, VIPRE-01 gives the same or a conservative safety limit.

Response

WPS performed thermal hydraulic calculations using the VIPRE-01 computer code and compared critical heat flux (CHF) using the W-3 correlation to test bundle measured data. The CHF test bundle data and test results are documented in References 1 and 2. WPS selected four test bundles typical of current ANF and Westinghouse 14x14 fuel designs for comparison: ROSAL-B, ENC-3, ENC-4, and ENC-5. These bundles adequately envelope many aspects of the current Kewaunee fuel designs. A summary description of each of the test bundles is presented in Table 1.

Advanced Nuclear Fuels (ANF) was contracted to perform a statistical assessment of the WPS VIPRE-01 DNBR (P/M) calculations and provided a report to WPS on the analysis results. The ANF report is included as Attachment 2 to this response. Portions of this report are summarized in the following paragraphs.

Table A of Attachment 2 shows the VIPRE-01 DNBR (P/M) results for each run of each test series. Figures 2.1 through 2.5 of Attachment 2 show DNBR (P/M) trends versus the operating parameters--pressure, inlet mass velocity, heat flux, inlet enthalpy, and inlet temperature. The figures also indicate the range of operational conditions analyzed for the data base.

An inspection of the VIPRE-01 results indicates that some calculations of DNBR (P/M) are significantly different from the remaining data. By statistical examination and by applying the limits of the W-3 correlation, the test runs shown in Table 2.1 of Attachment 2 are determined to be outliers and are excluded from consideration in the statistical analysis.

The remaining runs (217 points total) are analyzed by two methods to assure an appropriate 95/95 limit is determined. The first method utilizes the analysis of variance approach. The results of this method are shown in Table 2.2 of Attachment 2. For the analysis of variance, an equivalent sample size of 5.9 with 7.4 degrees of freedom is determined. Based on these values a k factor, equal to 3.203 for a 95/95 DNBR (P/M) limit of 1.25, is calculated. Details of the variance analysis are presented in Appendix B of Attachment 2.

A second method of analysis uses the order statistic approach. This method considers the limit to be based upon distribution free techniques. In this case, with 217 data points, a table for distribution free limits provided the rank to use as the 95/95 limit. This is the 5th from the largest value of DNBR (P/M) and is 1.027. The distribution free analysis is thus bounded by the analyses based upon an assumption of normality.

The VIPRE-01 results can be examined for distributional characteristics. Assessment of normality is performed using the W-statistic for small data sets and the D-prime test for larger data sets. Table 2.3 of Attachment 2 presents the results of these tests. Although ENC-3 is slightly peaked, the general conclusion is that the data shows reasonable normal behavior and that a normal distribution for the data as a whole is an acceptable model.

Finally, the data can be viewed graphically. Figure 3.1 of Attachment 2 shows the predicted versus measured critical heat flux along with the W-3 95/95 limit of 1.3 and a line where predicted and measured critical heat flux are equal. Also, a histogram of the data with a superimposed normal distribution, which has a mean of 0.755 and a standard deviation of 0.153, is displayed in Figure 3.2 of Attachment 2.

Based on the statistical assessment of VIPRE-01 (W-3) CHF results, a DNBR (P/M) limit of 1.25 adequately bounds the 95/95 limit for the data base analyzed. A fuel rod predicted by VIPRE-01 to have a DNBR of 1.25 is thus assured with a 95% confidence that there is a 95% probability of avoiding DNB. Therefore, the DNBR safety limit of 1.3, which will be used in the WPS thermal margin methodology, conservatively bounds the 95/95 limit for the analyzed data base.

TABLE 1

COMPILATION OF TEST SECTION GEOMETRY PARAMETERS

	SPONSOR GEOMETRY TYPE	TOTAL NO. OF PTS	NO. OF HTD RODS	ROD PITCH (IN)	ROD DIAM (IN)	UNHEATED ROD DIAM (INCHES)	TEST SECTION LENGTH (INCHES)	NO. OF GRIDS	GRID SPACING (INCHES)	RADIAL PEAKING FACTOR	AXIAL POWER DISTRIB
ROSAL	WH-PWR	33	16	.555	.422	0.000	96.0	7.	26.0	1.047	NON-UNIFORM
ENC-3	ENC-PWR	73	21	.556	.421	0.536	72.0	5.	15.7	1.094	UNIFORM
ENC-4	ENC-PWR	80	21	.556	.421	0.536	72.0	5.	15.7	1.094	UNIFORM
ENC-5	ENC-PWR	60	22	.565	.424	0.544	66.0	4.	26.2	1.083	UNIFORM

References

1. Parametric Study of CHF Data, EPRI NP-2609, September, 1982.
2. DNB Test Data Report, XN-NF-81-80, January, 1982.