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Westinghouse Control Rod Ejection Accident Analysis Methodology Using Multi-Dimensional Kinetics

Westinghouse

Westinghouse Non-Proprietary Class 3

WCAP-15807-NP-A

WESTINGHOUSE CONTROL ROD EJECTION ACCIDENT ANALYSIS METHODOLOGY USING MULTI-DIMENSIONAL KINETICS

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Section A

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 1, 2003

Mr. Hank A. Sepp, Jr.
Manager, Regulatory & Licensing
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

SUBJECT: ACCEPTANCE FOR REFERENCING OF TOPICAL REPORT WCAP-15806-P AND WCAP-15807-NP, "WESTINGHOUSE CONTROL ROD EJECTION ACCIDENT ANALYSIS METHODOLOGY USING MULTI-DIMENSIONAL KINETICS" (TAC NO. MB4521)

Dear Mr. Sepp:

By letter dated February 25, 2002, Westinghouse Electric Company (Westinghouse) submitted licensing Topical Reports (TRs) WCAP-15806-P and WCAP-15807-NP, "Westinghouse Control Rod Ejection Accident Analysis Methodology Using Multi-Dimensional Kinetics," to the NRC for review and approval. The objective of this TR is to present for licensing approval, an improved methodology for use in the Final Safety Analysis Report (FSAR) rod ejection accident analysis for pressurized water reactors. This application methodology is based on a three-dimensional core representation, using the NRC-approved core neutron kinetics code SPNOVA and the NRC-approved core thermal-hydraulic code VIPRE-01.

The staff has completed its review of the subject TR. The TR is acceptable for referencing in licensing applications to the extent specified and under the limitations delineated in the report and in the associated NRC safety evaluation (SE), which is enclosed. The enclosed SE defines the basis for acceptance of the TR. The staff has noted both the proposals for new fuel enthalpy criteria and the Westinghouse stated interim criteria, but will not endorse any new criteria at this time. The staff review of this TR covers only the justification of the improved rod ejection application methodology to adequately and conservatively represent the rod ejection event.

Pursuant to 10 CFR 2.790, we have determined that the enclosed SE does not contain proprietary information. However, we will delay placing the SE in the public document room for ten working days from the date of this letter to provide you with the opportunity to comment on the proprietary aspects only. If you believe that any information in the enclosure is proprietary, please identify such information line by line and define the basis pursuant to the criteria of 10 CFR 2.790.

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

In accordance with the guidance provided on the NRC website, we request that Westinghouse publish an accepted version within three months of receipt of this letter. The accepted version shall incorporate (1) this letter and the enclosed SE between the title page and the abstract, (2) all requests for additional information from the staff and all associated responses, and (3) a "-A" (designating "accepted") following the report identification symbol.

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

Sincerely,

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 700

Enclosure: Safety Evaluation

cc w/encl:
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT WCAP-15806-P AND WCAP-15807-NP,

"WESTINGHOUSE CONTROL ROD EJECTION ACCIDENT ANALYSIS METHODOLOGY

USING MULTI-DIMENSIONAL KINETICS"

WESTINGHOUSE ELECTRIC COMPANY

PROJECT NO. 700

1.0 INTRODUCTION

By letter dated February 25, 2002 (Reference 1), Westinghouse Electric Company (Westinghouse), submitted Topical Reports (TRs) WCAP-15806-P and WCAP-15807-NP, "Westinghouse Control Rod Ejection Accident Analysis Methodology Using Multi-Dimensional Kinetics," to the NRC for staff review and approval. The purpose of the TR is to present for licensing approval an improved application methodology for the Final Safety Analysis Report (FSAR) rod ejection accident analysis, based upon a realistic three-dimensional core representation using the NRC-approved core neutron kinetics code SPNOVA (References 2 and 3) and the NRC-approved core thermal-hydraulic code VIPRE-01 (VIPRE) (References 4 and 5). This improved methodology can be used to replace the currently approved methodologies (References 6 and 7), which are primarily based on conservative one-dimensional (1-D) axial core neutron kinetics methods. The proposed methodology is for application to pressurized water reactors (PWRs) and the TR describes the three-dimensional (3-D) methodology for the analyses of the rod ejection accident. The two referenced codes, SPNOVA and VIPRE-01, are coupled to pass the necessary data for the nuclear, fluid and fuel temperature calculations.

The phenomena that are of importance in determining the consequences of a rod ejection accident, particularly in high burnup fuel cores, have been identified in the NRC-sponsored Phenomenon Identification and Ranking Tables (PIRT) exercise for this accident (Reference 8). The improved 3-D methodology presented in the Westinghouse TR is stated to be consistent with the identified phenomena.

As discussed in Section 2.2 of this safety evaluation (SE), ongoing reactivity insertion accident (RIA) experiments at the CABRI research facility (Reference 9) and other tests of prompt-critical events with irradiated fuel rods have indicated that the current NRC peak fuel enthalpy criterion of 280 calories per gram (cal/g) may not be conservative for high fuel burnups (> 50 GWd/tU). As part of the industry response to the recent experimental results, revised acceptance criteria have been developed by the Electric Power Research Institute (EPRI) through Working Group 2 of the EPRI Robust Fuel Program. This approach is described in an EPRI TR (Reference 10). The proposed limiting criteria for the allowable fuel enthalpy vary as

a function of either fuel burnup or clad oxidation and have been submitted by industry through the Nuclear Energy Institute (NEI) requesting NRC review and endorsement (Reference 11). The proposed revised criteria would apply to both hot zero power (HZP) and hot full power (HFP) RIAs in both PWR and boiling water reactors (BWRs), and would affect the fuel failure threshold limit as well as the core coolability limit.

In the interim, until any new limits are approved, Westinghouse will apply additional conservatism to the 3-D methodology for the peak fuel enthalpy limit for the HZP rod ejection transient. This interim conservative adjustment limits the peak fuel enthalpy increase to 100 cal/g, which is less than the anticipated future criteria. Since the current Westinghouse criteria use an absolute limit of 200 cal/g for the peak radially averaged fuel enthalpy, and the HZP nominal fuel enthalpy is about 17.5 cal/g, this is equivalent to reducing the current Westinghouse design limit by 82.5 cal/g (41.25 percent). For the full power or non-prompt-critical cases they will continue to use the current licensed peak fuel enthalpy criteria until new criteria are approved.

The staff has noted both the proposals for new fuel enthalpy criteria and the Westinghouse stated interim criteria, but will not endorse any new criteria at this time. The staff review of this TR covers only the justification of the improved rod ejection application methodology to adequately and conservatively represent the rod ejection event.

2.0 <u>REGULATORY EVALUATION</u>

2.1 Current Regulatory Requirements

The staff has reviewed the Westinghouse TR in accordance with the applicable regulations and guidelines contained in the following documents:

- 10 CFR Part, 50 Appendix A, General Design Criteria (GDC) for Nuclear Power Plants;
- 10 CFR Part 100, Reactor Site Criteria;
- U. S. AEC, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," Regulatory Guide (RG) 1.77, May 1974;
- U. S. NRC, Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-0800, July 1981.

Section 4.2 of the SRP specifies two licensing criteria for reactivity insertion accident events. A fuel coolability limit was established to restrict the energy deposition in the fuel rod to preclude fuel melt, fragmentation and dispersal. This meets the requirements of GDC 28, "Reactivity Limits," as cited in RG 1.77, that the coolability limit for violent expulsion of fuel should be 280 cal/gm of UO_2. A fuel rod failure threshold was established to limit fission product release during postulated accidents to meet the specific requirements of 10 CFR 100.11 and 10 CFR Part 50, Appendix A, GDC 19, "Control Room." Cladding failure is assumed to occur when the heat flux is greater than or equal to the departure from nucleate boiling (DNB) ratio for zero power, low power and full power RIA events in PWRs.

The rod ejection accident is a design basis reactivity insertion event for PWRs. An analysis of the radiological consequences of the event is typically presented in Chapter 15 of the plant updated FSAR. Acceptable analysis methods and criteria for the event are described in

RG 1.77, and the NRC review process for the FSAR analysis is described in Section 15.4.8 of the SRP.

The allowable dose consequences for the event are as given in RG 1.77. The number of fuel failures for the dose evaluation is based on the number of fuel rods reaching DNB, as discussed in Section 15.4.8 of the SRP.

2.2 Recent Developments

As discussed previously, ongoing RIA experiments at the CABRI Research Facility and other tests of rapid reactivity insertion (prompt-critical) events with irradiated fuel rods have indicated that the current NRC peak fuel enthalpy criterion of 280 cal/g may not be conservative for high burnups (> 50 GWd/tU).

On October 4, 1993, the Office of Nuclear Reactor Regulation (NRR) sent a user-need memorandum to the Office of Nuclear Regulatory Research (RES) involving high-burnup fuel issues. The memorandum requested work in three areas: (1) fuel performance model changes, (2) fuel performance code updates, and (3) evaluation of the fuel failure thresholds. Subsequently, an "Agency Program Plan for High-Burnup Fuel" was prepared and sent to the Commission on July 6, 1998. An attachment to the Agency Program Plan describes the high burnup fuel research schedules and resources related to several issues, as well as assessments of their safety significance. This plan was discussed with the Advisory Committee for Reactor Safeguards (ACRS) in 1998.

One of the activities listed in these documents relates to fuel damage thresholds and criteria for various events. For RIAs, GDC 28 states that:

> "The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effect of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core."

Section 4.2 of the SRP describes specific acceptance criteria for RIA analyses to demonstrate that GDC 28 is satisfied. The SRP describes a coolable geometry as:

> "...[retaining a] rod-bundle geometry with adequate coolant channels to permit removal of decay heat."

For a severe RIA, such as rod ejection in a PWR or a rod drop in a BWR, the SRP provides a specific acceptance criterion to prevent the violent expulsion of fuel and ensure core coolability:

> "...To meet the guidelines of Regulatory Guide 1.77 as it relates to preventing wide-spread fragmentation and dispersal of the fuel and avoiding the generation of pressure pulses in the primary system of a PWR, a radially averaged enthalpy limit of 280 cal/g should be observed. This 280 cal/g limit should also be used in BWRs."

Although the SRP limit is set at 280 cal/g for coolability, the reactor vendors have adopted their own lower design values to ensure coolability (approximately 225 cal/g) to assure conservatism.

The SRP also contains another lower limit of 170 cal/g that is used as a threshold for cladding failure to determine that the radiological releases from an RIA remain well below 10 CFR Part 100 limits.

Work on RIA testing is continuing through experiments at the CABRI facility in France, and at the Nuclear Safety Research Reactor (NSRR) in Japan. As a result of a fuel failure during a test at CABRI in 1993 (REP Na-1), and one in 1994 (HBO-1) at NSRR, the NRC recognized that high burnup fuel cladding might fail during an RIA at a lower enthalpy than the 280 cal/g limit currently specified in RG 1.77. However, generic analyses performed by all of the reactor vendors have indicated that the fuel enthalpy during RIAs will be much lower than the RG 1.77 limit, based on 3-D neutronic calculations. For high burnup fuel which no longer contains significant reactivity, the peak fuel enthalpy calculated using the 3-D models is expected to be lower than the value of 100 cal/g that was discussed and recommended by the RES in Research Information Letter No. 174 dated March 3, 1997, as a potential replacement for the RG 1.77 value.

The March 3, 1997 Research Information Letter and the attachment to the Agency Program Plan describe both the fuel cladding failure threshold and the core coolability criteria limits, and suggest that new interim criteria be established to reflect the experience from the CABRI and NSRR experiments. The suggested interim criteria include a cladding failure limit of 100 cal/g. In addition, the coolability limit would be left at 280 cal/g for a fuel burnup of less than 30 GWD/t, while higher burnup fuel would be subject to a "no cladding failure" limit. The plan also noted that the 280 cal/g value might be lowered to about 230 cal/g, but explained that this was not a high-burnup issue.

The staff has noted both the proposals for new fuel enthalpy criteria and the Westinghouse stated interim criteria, but will not endorse any new criteria without appropriate review and approval. The staff's review of this TR covers only the justification of the improved rod ejection application methodology to adequately and conservatively represent the rod ejection event.

Current reactor design practices result in reactor core designs and operation where the calculated peak enthalpy during RIAs remains well below the 280 cal/g (and the 230 cal/g) limit. These calculations were performed using conservative, one-dimensional neutronic methods. More realistic calculations of these scenarios by both the reactor vendors and NRC contractors show that the most reactive fuel (usually new fuel) will not suffer a fuel enthalpy rise during an RIA above the proposed 100 cal/g interim cladding failure limit. Although some of the fuel failures that occurred in CABRI and NSRR have released fuel material outside the cladding, none have shown the sort of expulsion of molten fuel that threaten the integrity of the reactor coolant pressure boundary. The currently identified future tests at these facilities would not address fuel fragmentation and dispersal effects at high temperatures. Therefore, they would not provide information relevant to the 280 cal/g (or the 230 cal/g) coolability limit. Instead, these experiments would only provide further experimental confirmation for the proposed 100 cal/g fuel failure threshold limit.

Current results of analyses to calculate off-site releases during RIAs do not challenge licensing limits, and the staff does not believe that changing the cladding failure threshold limit from 170 cal/g to 100 cal/g would significantly affect these results. If the analyses are done with realistic methods, no cladding failure is expected using either limit.

3.0 TECHNICAL EVALUATION

3.1 Accident Description and Limits

The rod ejection accident is described as the mechanical failure of a control rod mechanism pressure housing such that the reactor coolant system pressure would cause the ejection of a partially- or fully-inserted control rod and drive shaft to its fully withdrawn position. If the reactor is at or near critical, the consequences of this mechanical failure are a rapid reactivity insertion and core power increase together with an adverse core power distribution, possibly leading to localized fuel rod damage. The power increase is arrested primarily by the negative reactivity due to the Doppler feedback resulting from fuel heatup, and the transient is terminated by a reactor trip which is initiated shortly after the beginning of the transient.

Due to the extremely low probability of a rod cluster control assembly ejection, this accident is classified as a Condition IV (limiting fault) event as defined by the American Nuclear Society Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants (ANSI N18.2-1973) (Reference 12).

For a typical PWR, the position of all control rods is continuously indicated in the control room, and an alarm will occur if one rod deviates from its bank demand position by more than 5 percent of the indicated rod position. There are low and low-low insertion limit monitors with visual and audio signals. Operating instructions require boration at the low limit alarm, and emergency boration at the low-low limit alarm. These alarm functions ensure that the initial conditions prior to a control rod ejection accident will not be worse than the cases analyzed. Should a rod ejection event occur while the reactor is at or near critical, the transient will typically be terminated by one or more of the automatic features of the reactor protection system. The protection system features are typically described in the FSAR for each plant.

The physical limits of this accident are that any consequential damage to either the core or the reactor coolant system must not prevent long-term core cooling and that any off-site dose consequences must be within the guidelines of 10 CFR Part 100. More specific criteria are applied to ensure that there is no significant fuel dispersal in the coolant, gross fuel lattice distortion, or severe shock waves. Acceptable limiting criteria for this accident have been defined in RG 1.77. However, Westinghouse typically has applied the following conservative criteria:

1. The average fuel pellet enthalpy at the hot spot shall be below 200 cal/g (360 BTU/lbm) for irradiated or unirradiated fuel. (Note: RG 1.77 allows a higher limit of 280 cal/g.)

2. The peak reactor coolant pressure shall be less than that which would cause stresses to exceed the faulted condition stress limits. (Note: Westinghouse

plants meet the more stringent emergency condition stress limits as specified by RG 1.77.)

3. Fuel melting will be limited to less than the innermost 10 percent of the fuel pellet at the hot spot, even if the average fuel pellet enthalpy at the hot spot is below the limits of Criterion 1 above.

Some plants use enthalpy as alternate criteria for fuel failure (200 cal/g for clad failure and 250 cal/g for incipient centerline melting, respectively). These criteria have been used in plant FSARs which have been reviewed and accepted by the NRC.

Note that the EPRI TR (Reference 10) proposes a fuel rod failure threshold of 170 cal/g for unirradiated fuel, decreasing to approximately 125 cal/g at a rod average burnup of 62 GWd/MTU. The proposed limit for core coolability (centerline melt) ranges from 230 cal/g for unirradiated fuel to approximately 200 cal/g at a rod average burnup of 62 GWd/MTU.

The staff has noted the proposals for new fuel enthalpy criteria and the Westinghouse current conservative criteria, but will not endorse any new criteria at this time. The staff's review of this TR covers only the justification of the improved rod ejection application methodology to adequately and conservatively represent the rod ejection event.

3.2 Current Methodology

The current Westinghouse licensing basis analysis methodology for the FSAR rod ejection accident is based upon the application of conservative 1-D axial core neutron kinetics methods, and is described in the NRC-approved TR WCAP-7588, Rev. 1A (Reference 6) and in the NRC-approved TR CENPD-190 (Reference 7).

These currently approved application methods use a 1-D nuclear design model to calculate the ejected rod worth and associated peaking factor, and the Doppler and moderator feedback. In order to bound future reload cycles for an individual plant, the vendor may perform, at the request of the licensee, a more conservative bounding analysis utilizing key parameters which are not expected to be exceeded. These parameter values are then utilized with appropriate uncertainty allowances in a bounding transient analysis calculation of the core and fuel behavior. The key parameters are checked every cycle to ensure that the analysis remains bounding.

The current static analysis of the ejected rod is performed in 3-D using an adiabatic feedback model which maintains the Doppler and moderator feedback at the initial (pre-ejection) condition. This will generate a peaking factor which is larger than would be calculated with a transient calculation that includes more realistic feedback. The various control rod bank locations are evaluated to determine the location of the worst ejected control rod assemblies in the core at various fuel burnup levels.

The current effective delayed neutron fraction for the entire core is obtained by weighting the delayed neutron fraction for different fissionable isotopes by the fraction of fissions in each isotope and the power sharing in the core.

In the current static calculation, key safety parameters of a single rod ejection are evaluated at the beginning and end of each reload fuel cycle using the three-dimensional nodal code ANC (References 13, 14, 15, and 16). The evaluation considers both full power and zero power initial conditions with the control banks at their respective insertion limits. Spatial peaking factors, control bank worths and ejected rod worths are derived from multi-dimensional neutronic calculations. In the static calculation of the rod ejection event, the peaking factors are calculated with an adiabatic assumption. That is, the nuclear feedback, both Doppler and moderator, is established during the initial condition calculation and this feedback is not allowed to vary when the rod is ejected.

The current fuel temperature (Doppler) coefficient is defined as the change in reactivity per degree change in effective fuel temperature. It is primarily a measure of the Doppler broadening of U238 and Pu240 resonance absorption peaks. The fuel temperature coefficient is calculated by performing two-group multi-dimensional neutronic calculations. Moderator temperature is held constant and power level is varied. The spatial variation of fuel temperature is taken into account by calculating the effective fuel temperature as a function of local power density throughout the core.

The current transient calculation of the rod ejection is performed in two stages, first an average core channel calculation and then a hot rod calculation. The average core calculation is performed using the approved TWINKLE (Reference 17) code in the 1-D (axial) mode to determine the core average power generation with time including the various core feedback effects, i.e., Doppler temperature and moderator temperature reactivities. A Doppler weighting factor is applied to the Doppler feedback to compensate for the missing dimensions. Enthalpy and temperature transients at the hot spot are then determined by multiplying the average core energy generation by the hot channel factor and performing a fuel rod transient heat transfer calculation using the approved FACTRAN (Reference 18) code. The power distribution calculated without feedback is conservatively assumed to persist throughout the transient. The DNB heat flux is not calculated; instead DNB is conservatively assumed to occur near the start of the transient. Cases at the beginning and end-of-cycle (EOC), at full and hot zero power initial conditions, are analyzed. Input values for the ejected rod worth, peaking factors, delayed neutron fraction and feedback coefficients are determined using the nuclear design methods as discussed in the preceding subsection. Appropriate uncertainty allowances are applied to the parameters used in the calculation. A more detailed discussion of the current method of analysis can be found in typical plant FSARs, and in the Westinghouse rod ejection TR (Reference 6).

3.3 New Methodology

3.3.1 Nuclear Model

The most significant difference between the current NRC-approved analysis method and the new improved analysis method for this accident is the change from a 1-D to a 3-D core neutron kinetics and feedback model. This eliminates the need to apply Doppler weighting factors to the core kinetics calculation to simulate the increased Doppler feedback due to the skewed power distribution following the ejection of the rod. It also eliminates the conservative assumption of a constant no-feedback value of the ejected rod peaking factor in the hot rod calculation. The computer codes used with the 3-D revised analysis method have already been

reviewed and approved by the NRC. The nuclear model is based on the NRC-approved Westinghouse SPNOVA code (References 2 and 3).

In the new methodology, a detailed 3-D transient nuclear model is used for the analysis. The use of a transient model gives a more accurate analysis of the actual transient. The major changes to the model are the input of the kinetics parameters. These include the delayed neutron fractions, the delayed neutron decay constants, the neutron velocities for each energy group, and the transient driver functions which initiate the transient and the reactor trip. These parameters enable the ongoing calculation of the peaking factors as the transient unfolds, instead of assuming a constant peaking factor value throughout the transient. The overall delayed neutron fraction can be further adjusted for conservatism by applying a fixed multiplier to the individual node-by-node values.

Westinghouse proposes to use conservative preconditions for both HFP and HZP analyses that include time-of-life effects, rod insertion effects, potential xenon distributions and allowable control rod positions.

Typical cycle-specific key parameters that Westinghouse analyzes include:

- Ejected rod worth
- Ejected rod peaking factors
- Delayed neutron fractions
- Doppler temperature coefficient
- Moderator temperature coefficient

Uncertainty allowances are applied in determining these key parameters to provide conservative limiting analysis values.

3.3.2 Thermal-Hydraulic Feedback Model

The new core thermal-hydraulic and fuel rod models, as well as the hot rod model, are based on the NRC-approved Westinghouse version of the VIPRE-01 code and associated methods (References 4 and 5).

The thermal-hydraulic model used in the reactor core kinetics calculation includes the time dependent effects of reactor coolant flow, heat transfer from the clad to the coolant, and direct heat generation in the coolant. The calculation is performed with the VIPRE code using a mesh structure consistent with the nuclear analysis mesh structure. Since the rod ejection event is a very rapid transient, most of the thermal energy is retained in the fuel rod. Thus, the coolant temperature increase is relatively small. Some of the fission energy is deposited directly into the coolant through the slowing down of the fission neutrons and the absorption of gamma rays accompanying the fission process. This can become significant for transients that result in very high peak nuclear power increases, and is taken into account in the calculation.

Since the principal feedback mechanism is due to the Doppler feedback, thermal hydraulic modeling assumptions are made for the node average which conservatively maximizes heat transfer from the clad to the coolant. This includes the assumption of full reactor coolant flow

(no reactor coolant pumps (RCPs) out-of-service) in the HZP rod ejection case, and no initiation of post-DNB heat transfer.

For the HZP case, the major factor turning the transient around is the Doppler feedback, which is directly related to the rapid increase in fuel pellet temperature. Under prompt-critical conditions, the fuel rod transient is nearly adiabatic; that is, the details of the heat transfer from the fuel to the clad and from the clad to the coolant is not of high importance. For the much slower HFP case, both the Doppler and moderator feedback are important, particularly at the EOC with the most negative moderator temperature coefficient. Therefore, for the HFP case, the fuel rod internal heat transfer and the heat transfer to the clad and coolant are important and are accounted for in the model.

The average fuel rod model for the feedback calculation is performed with the VIPRE code using a multizone fuel pellet representation for the fuel rod in each neutronic/thermal-hydraulic core node. The fuel pellet-to-clad gap heat transfer is calculated using the dynamic gap conductance model in VIPRE that accounts for changes in the fuel dimensions and fill gas pressure with temperature. Design values of pellet radial power distributions, based on an assembly-average burnup, are input for each fuel assembly. The resonance effective temperature is generated at each spatial node from the radially varying temperatures using design values of the T_{eff} weighting function.

To conservatively bound the transient and cover the uncertainties in the actual T_{eff} calculation, an input multiplier on the Doppler feedback cross section adjustment is applied. This allows a uniform uncertainty allowance to be applied on the Doppler feedback adjustments. Similarly, the core parameters are adjusted to make the moderator temperature more positive to conservatively represent the moderator density feedback effect.

3.3.3 Peak Enthalpy Calculations

The hot fuel rod thermal calculation is performed independent of the node average fuel temperature feedback calculation, with additional conservatism applied to the modeling and initial conditions in order to maximize the increase in fuel temperature and enthalpy. The key limit for the accident is the calculated radially-averaged peak fuel enthalpy (RAPFE), or the maximum change in fuel enthalpy. The hot fuel rod model uses the same fuel pellet and clad mesh description as for the average rod.

The hot fuel rod model is based on the NRC-approved model described in the Westinghouse VIPRE modeling TR (Reference 4), and is similar to the model used in the approved FACTRAN code (References 6 and 18). It represents the hottest fuel rod from any assembly in the core. The pellet-to-clad gap heat transfer is calculated using the dynamic gap model in VIPRE, which is comparable to the conservative, NRC-approved FACTRAN transient gap model. In both cases, the model has been calibrated against the design fuel rod temperatures generated by an acceptable fuel rod performance code such as the approved PAD program (Reference 19), using the method described above for the average rod model. Consistent with current plant licensing applications, the heat transfer to the coolant is calculated using the Dittus-Boelter correlation for single phase forced convection, the Thom correlation for nucleate boiling, and the Bishop-Sandberg-Tong correlation (Reference 20) for transition and film boiling beyond DNB. In order to maximize the temperature and enthalpy increase within the fuel pellets, the

hot spot of the fuel rod is in post-DNB film boiling during the transient. The Baker-Just correlation (Reference 21) is used to account for heat generation in the cladding material due to the zirconium-water reaction.

A benefit of this new method compared to the current licensed analysis methodology based on 1-D kinetics is that more realistic time-dependent core average power, rod peak power, and axial power distributions are taken directly from the 3-D kinetics results for the hot rod transient calculation instead of using a more conservative constant peak power value. The calculation can be performed for the hot rod in the hottest assembly (the one with the peak fuel enthalpy), or for different fuel assemblies in the core at various levels of burnup.

3.4 Neutronic Parameters and Core Initial Conditions

Key neutronic parameters, such as Doppler and moderator feedback, are determined in order to maximize the rod ejection effect. For example, the total Doppler feedback coefficient is at a maximum at the beginning of a cycle, and at a minimum at the end of a cycle. The moderator temperature coefficient becomes more negative with decreasing boron concentrations, and with increasing temperatures.

The effect of the ejected rod worth is dependent on the arrangement of fuel assemblies within the core, the control rod pattern, the axial power distribution due to burnup and xenon effects, and the allowed control rod insertion limits. If the control rods are partially inserted, the ejected rod worth increases for power distributions which skew the power to the top of the core. The core power distribution is naturally skewed slightly to the bottom of the core at full power due to the temperature feedback. Thus a burnup skew builds into the core with cycle depletion, and is at a maximum at the end of the cycle. For this reason, the EOC condition provides the most limiting axial power distributions. The ejected rod peaking factor will also increase as the ejected rod worth increases. Thus if the ejected rod worth is conservatively increased, this also conservatively increases the ejected rod peaking factor without applying a separate additional conservatism on the peaking factors.

There are two other key core operating parameters, besides the time of life and depletion model, that have a significant effect on the ejected rod worth and peaking factors, and that can be adjusted as part of the initial conditions for the analysis. These are the xenon distribution and the control rod bank positions.

The axial xenon distribution can have a significant impact on the ejected rod worth and the ejected rod peaking factor for partially-inserted rod banks. Xenon distributions that force the power distribution more to the top of the core are more limiting since they increase the axial peaking factor and increase the worth of the rod that is being ejected.

At HFP, there is a nominal operating range in which the reactor is expected to operate. This band of operation is typically defined by axial offset limits. Those limits can be either a band around the equilibrium value, or absolute limits. Since HFP operation is the expected norm, a limiting axial xenon distribution is used in the precondition for the rod ejection evaluation. This is a highly unlikely situation since it would result in the operator having no room to control the reactor, but it provides a conservative bound.

At HZP, there are no limitations on the axial power distribution. HFP equilibrium xenon, and zero xenon conditions result in very similar ejected rod worths. Top-skewed xenon distributions decrease the ejected rod worth. Mildly bottom-skewed xenon distributions slightly increase the ejected rod worth. Therefore, an artificially skewed-to-the-bottom xenon distribution is chosen which increases the ejected rod worth beyond the zero xenon case to conservatively account for adverse power distributions.

The control bank insertion also has a significant role in the ejected rod worth. The ejected rod worth generally increases with increased control bank insertion for the same axial power shape. Thus, deeper insertions increase the ejected rod worth. Technical specification limits on control rod insertion, and the control rod insertion limit alarms, ensure that it is highly unlikely that the control rods will be inserted beyond the specified limits. Thus, the assumption that the control banks are at their insertion limit is a conservative initial condition for the rod ejection accident. In order to perform a more bounding analysis where a higher bounding ejected rod worth is desired, a deeper insertion can be utilized, and/or the control rod cross sections can be adjusted.

3.5 Reload Safety Evaluation (RSE)

The currently approved Westinghouse RSE methodology (Reference 22) uses a bounding analysis approach which is characterized by utilizing key parameters determined from a static analysis to determine whether a detailed transient case should be analyzed for the current cycle. The key parameters for the rod ejection transient that vary from cycle to cycle, assuming no change in plant operating characteristics or fuel type, are:

- Ejected rod worth
- Ejected rod peaking factors
- Delayed neutron fractions
- Doppler temperature coefficient
- Moderator temperature coefficient

The reference bounding safety evaluation calculation may be performed with more conservative values for these key parameters through the use of more conservative allowances. If a reload safety evaluation, using the cycle-specific static values, is less limiting than the reference bounding analysis of record, then a cycle-specific transient analysis does not need to be performed.

If the plant operating characteristics (power, temperature, pressure, flow, design peaking factors, etc.) or the fuel type or characteristics (clad diameter and thickness, pellet diameter, grid) should change, this is identified in the RSE methodology, and more variables are evaluated to determine if the analysis must be repeated.

These parameters are all associated with the reactor core, and the use of the 3-D methodology requires the generation of a detailed 3-D core model. Thus, these parameters are implicitly handled by the nuclear model.

The rod ejection transient is a very rapid transient, and as such there is no recirculation loop impact on the course of the transient. Therefore, variations in the primary and secondary system characteristics have no impact on the course of the transient.

The control rod ejection time is chosen to be fast enough to be of no consequence to the actual transient, so this time is insensitive to the control rod cluster geometry and rodlet absorber composition. The modeling of the ex-core detectors and the determination of the time of reactor trip are dependent on the type of plant, ex-core detector geometry and the protection system setpoints and allowances. The position of the tripped rods versus time is also dependent on the control rod cluster geometry and rodlet composition. The trip behavior thus is plant specific, and is modeled as such; and, as noted in the Westinghouse TR, the control rod trip has only a secondary impact on the limiting parameters calculated for the transient.

The currently approved RSE methodology in WCAP-9272-A continues to be applicable to analyses with the approved 3-D kinetics/thermal-hydraulics methodology (SPNOVA/VIPRE) and its application is consistent with both the current RG 1.77 criteria and any new proposed criteria. Therefore, the use of this methodology is acceptable to the staff.

The basic 3-D rod ejection methodology, as defined in the Westinghouse submittal, is generally applicable to all Westinghouse and Combustion Engineering nuclear steam supply systems (NSSS) pressurized water reactors. Application of this methodology to non-Westinghouse NSSS plants would, however, require additional justification, including submittal of sample calculations, uncertainty analyses, and applicable benchmarking results.

3.6 Sample Application Analyses

The plant selected for the sample application calculations is a Westinghouse 3-loop core with a 17x17 fuel assembly and an 8-cluster lead control bank (Bank D). This core design has typically been one of the most limiting for the HZP rod ejection transient.

3.6.1 Hot Zero Power

Analyses were performed to determine the worst ejected rod cluster and the appropriate starting conditions for the transients. Additional calculations were performed for this study to demonstrate the sensitivity of various parameters.

The HZP analyses were performed at the EOC conditions. Several sample conservative analyses were conducted:

- Statistical reload analysis: A typical single reload cycle analysis with key parameter conservative allowances added to the results statistically. (Labeled Base Case)

- Deterministic reload analysis: A typical single reload cycle analysis with conservative allowances included for each key parameter separately. (Labeled All Allowances)

- Bounding analysis: A typical multi-cycle bounding analysis which increases the key parameters prior to including the key parameter conservative allowances to create an analysis that is expected to bound most future reload cycles. (Labeled Bounding Case)

The key parameters used in the evaluations and the results are provided in Table 3.3 of the Westinghouse submittal. The key parameter sensitivities are presented in Table 3.4 of the submittal. Results of the analysis showed that the application of all the uncertainties in the key parameters together increases the core average peak power considerably, and that the peak fuel delta enthalpy also increases compared to the conservative base case without those uncertainties. The individual perturbations, when combined as the square root of the sum of the squares, provide the same impact as the transient calculation which included all of them together. This also confirms that the individual effects were independent.

The bounding analysis is a severe case with the adiabatic ejected rod worth being slightly less than $2.00 in reactivity. The Doppler multiplier has been adjusted lower, and the feedback effect is further conservatively bounded. This bounding reference calculation produces results which are far more limiting than is typically assumed. The results are summarized in Figures 3.7 and 3.8 and in tabulated form in the submittal.

The staff finds the analyses and results presented in WCAP-15806-P to be acceptable, since the overall modeling is improved and all key parameters are represented conservatively.

3.6.2 Hot Full Power

Both beginning-of-cycle (BOC) and EOC cases were analyzed at HFP. Results of the analysis show that the ejected rod reactivity worths are almost the same, but with different ejected rod core locations and with different parameters associated with the time in life. In both cases, the key parameter conservative uncertainties were included: the Doppler feedback and delayed neutron fractions were reduced by a conservative multiplier, the moderator temperature coefficient was made more positive by increasing the soluble boron concentration, and with a xenon distribution giving a limiting positive axial offset with the control rods deeply inserted.

Both HZP and HFP cases showed the characteristic rapid increase in power until the Doppler feedback balances the reactivity insertion, followed by a decrease to the new equilibrium power. The control rod trip then initiates the shutdown. The HFP transient results indicated a different profile than the HZP transient for the following reasons:

- The HFP ejected rod worth is much less than the delayed neutron fraction, and the transient is not a prompt event.
- The reactor is already operating at power so, in general, the fuel temperatures are already significant and pellet clad contact has occurred. Thus, the heat transfer is very good between the pellet and the coolant.
- Also, since the reactor is at power, there is no delay time, as seen in the HZP cases, for the flux level to increase into the significant range.

The summary of the parameters and results are provided in Tables 3.5 and 3.6 of the Westinghouse submittal. Profiles of the core average power, peak fuel enthalpy and minimum DNB ratio are shown in Figures 3.9 through 3.11 of the same submittal.

The staff finds the analyses and results presented in WCAP-15806-P to be acceptable, since the overall modeling is improved and all key parameters are represented conservatively.

3.7 Staff Audit

To resolve questions raised during the staff's review of the Westinghouse TR, the staff conducted a technical audit at the Westinghouse Nuclear Center in Monroeville, Pennsylvania on March 17-18, 2003. Selected areas identified in the staff's review were examined in detail during the audit and Westinghouse provided a formal response (Reference 23) to the staff's questions. Topics covered in the audit included:

- SE restrictions on previously approved TRs
- Uncertainty analyses
- Benchmarking
- Key parameters
- Cross section modification during rod ejection
- Input controls
- Code coupling limitations

The audit was accomplished by a combination of interviews with the Westinghouse 3-D Rod Ejection Team members, reviews of the sample case calculation notes, and reviews of the individual code Users Manuals. Setup and execution of a special test case requested by the staff were also performed during the audit. This test case examined the effect of moderator temperature feedback by driving the moderator temperature coefficient to zero.

The staff also examined the derivation and solution in SPNOVA of the 3-D neutron kinetics equations and the effect of the pulse width variation. The coupling of the SPNOVA and VIPRE codes was reviewed, including neutronic and thermal-hydraulic channel mapping, file manipulation, time step synchronization, and limitations on representation of feedback mechanisms. The statistical approach used was clarified, and justification was provided for the use of the square root of the sum of the squares combination of independent variables.

The staff requests for clarification and additional information were satisfied during this audit, and it was confirmed that the methodology, analyses and results presented in the Westinghouse TR are consistent with RG 1.77 and the applicable SRP sections and that all key parameters can be modeled conservatively. The staff finds the analyses and results presented in WCAP-15806-P and WCAP-15807-NP to be acceptable, since the overall modeling is improved and all key parameters are represented conservatively.

4.0 CONCLUSION

The staff reviewed the application methodology, analyses, and results presented in WCAP-15806-P and WCAP-15807-NP and determined that the analyses and results are performed in accordance with the guidance and limitations provided in RG 1.77 and the applicable sections of the SRP. In additicn, the new 3-D time dependent application methodology is a considerable improvement over the current methodology, leading to an more realistic calculation of the rod ejection event. The staff accepts this methodology and concludes that it is acceptable for referencing in licensing applications. The staff finds the analyses and results presented in

WCAP-15806-P to be acceptable, since the overall modeling is improved and all key parameters are represented conservatively.

The staff also noted both the proposals for new fuel enthalpy acceptance criteria and the Westinghouse stated interim criteria, but will not endorse any new criteria at this time. The staff approval of this TR covers only the justification for the use of the improved methodology to adequately and conservatively represent the rod ejection event.

The staff has concluded that although the current RG 1.77 limits may not be conservative for cladding failure at higher fuel burnups, the more realistic analyses performed by the NSSS vendors, which have been confirmed by NRC-sponsored calculations, provide reasonable assurance that the effects of postulated RIAs in operating plants with fuel burnups up to the currently approved 62 GWD/MTU will neither (1) result in damage to the reactor coolant pressure boundary nor, (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core as specified in current regulatory requirements (GDC 28).

On the basis of the above review and justification, the staff concludes that the proposed change to the Westinghouse control rod ejection accident analysis methodology is acceptable.

The basic 3-D rod ejection methodology, as defined in the Westinghouse submittal, is generally applicable to all Westinghouse and Combustion Engineering NSSS pressurized water reactors. Application of this methodology to non-Westinghouse NSSS plants would, however, require additional justification, including submittal of sample calculations, uncertainty analyses, and applicable benchmarking results.

5.0 <u>REFERENCES</u>

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Principal Contributor: A. Attard

Date: July 1, 2003

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Section B

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Westinghouse
Electric Company

Box 355
Pittsburgh Pennsylvania 15230-0355

February 25, 2002
LTR-NRC-02-9

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

Attention: J. S. Wermiel, Chief
Reactor Systems Branch
Division of Systems Safety and Analysis

Subject: Submittal of WCAP-15806-P/WCAP-15807-NP, "Westinghouse Control Rod Ejection Accident Analysis Methodology Using Multi-Dimensional Kinetics," for NRC Review and Approval (Proprietary / Non-proprietary)

Dear Mr. Wermiel:

Enclosed are copies of the Proprietary and Non-Proprietary versions of the Westinghouse document "Westinghouse Control Rod Ejection Accident Analysis Methodology Using Multi-Dimensional Kinetics", WCAP-15806-P (Proprietary) and WCAP-15807-NP (Non-Proprietary). This document presents the Westinghouse-developed three-dimensional methodology for analyzing the rod ejection accident for pressurized water reactors. The computer codes used in this methodology have been previously reviewed and approved by the NRC for this application. The document is being submitted for NRC review and approval for the licensing application to all pressurized water reactors and reload designs. Westinghouse expects this methodology to be implemented for a reload core design later this year, and may be implemented for other licensees in the future.

Also enclosed are:

1. One (1) copy of the Application for Withholding, AW-02-1513 with Proprietary Information Notice and Copyright Notice.

2. One (1) copy of Affidavit, AW-02-1513.

This submittal contains Westinghouse proprietary information of trade secrets, commercial or financial information which we consider privileged or confidential pursuant to 10 CFR 9.17(a)(4). Therefore, it is requested that the Westinghouse proprietary information attached hereto be handled on a confidential basis and be withheld from public disclosure.

LTR-NRC-02-9

This material is for your internal use only and may be used solely for the purpose for which it is submitted. It should not be otherwise used, disclosed, duplicated, or disseminated, in whole or in part, to any other person or organization outside the Office of Nuclear Reactor Regulation without the expressed prior written approval of Westinghouse.

Correspondence with respect to any Application for Withholding should reference AW-02-1513 and should be addressed to H. A. Sepp, Manager of Regulatory and Licensing Engineering, Westinghouse Electric Company, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

Henry A. Sepp, Manager
Regulatory and Licensing Engineering

Copy to:
R. Caruso, NRR
D. Holland, NRR
U. Shoop, NRR
S. L. Wu, NRR

Westinghouse
Electric Company

Box 355
Pittsburgh Pennsylvania 15230-0355

February 25, 2002
AW-02-1513

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

Attention: J. S. Wermiel, Chief,
Reactor Systems Branch
Division of Systems Safety and Analysis

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: Submittal of WCAP-15806-P/WCAP-15807-NP, "Westinghouse Control Rod Ejection Accident Analysis Methodology Using Multi-Dimensional Kinetics," for NRC Review and Approval (Proprietary / Non-proprietary)

Reference: Letter from H. A. Sepp to J. S. Wermiel, LTR-NRC-02-9 dated February 25, 2002

Dear Mr. Wermiel:

The application for withholding is submitted by Westinghouse Electric Company LLC, a Delaware limited liability company ("Westinghouse"), pursuant to the provisions of paragraph (b)(1) of Section 2.790 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary material for which withholding is being requested is identified in the proprietary version of the subject report. In conformance with 10 CFR Section 2.790, Affidavit AW-02-1513 accompanies this application for withholding, setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

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

Correspondence with respect to this application for withholding or the accompanying affidavit should reference AW-02- 1513 and should be addressed to the undersigned.

Very truly yours,

Henry A. Sepp, Manager
Regulatory and Licensing Engineering

AW-02-1513

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

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

Henry A. Sepp

Henry A. Sepp, Manager
Regulatory and Licensing Engineering

Sworn to and subscribed
before me this 25th day
of February, 2002.

Lorraine M. Piplica

Notary Public

LORRAINE M. PIPLICA
COMMONWEALTH OF PENNSYLVANIA
NOTARY

Notarial Seal
Lorraine M. Piplica, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Dec. 14, 2003
Member, Pennsylvania Association of Notaries

(1) I am Manager, Regulatory and Licensing Engineering, in the Nuclear Services, of the Westinghouse Electric Company LLC, a Delaware limited liability company ("Westinghouse") and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Electric Company.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked "Westinghouse Control Rod Ejection Accident Analysis Methodology Using Multi-Dimensional Kinetics", WCAP-15806-P (Proprietary)/WCAP-15807-NP (Non-Proprietary), February 25, 2002, for submittal to the Commission, being transmitted by Westinghouse Electric Company (W) letter (LTR-NRC-02-9) and Application for Withholding Proprietary Information from Public Disclosure, Henry A. Sepp, Westinghouse, Manager Regulatory and Licensing Engineering to the attention of J. S. Wermiel, Chief, Reactor Systems Branch, Division of Systems Safety and Analysis. The proprietary information as submitted by Westinghouse Electric Company is that associated with the Westinghouse-developed three-dimensional methodology for analyzing the rod ejection accident for pressurized water reactors. The computer codes used in this methodology have been previously reviewed and approved by the NRC for this application. The document is being submitted for NRC review and approval for the licensing application to all pressurized water reactors and reload designs. Westinghouse expects this methodology to be implemented for a reload core design later this year, and may be implemented for other licensees in the future.

This information is part of that which will enable Westinghouse to:

(a) Obtain NRC licensed approval of the 3-D methodology.

(b) Promote convergence between Westinghouse business units.

Further this information has substantial commercial value as follows:

(a) Westinghouse intends to sell this methodology usage to licensee.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended for developing the enclosed improved core thermal performance methodology.

Further the deponent sayeth not.

Proprietary Information Notice

Transmitted herewith are proprietary and non-proprietary versions of documents furnished to the NRC. In order to conform to the requirements of 10 CFR 2.790 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.790(b)(1).

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Preface

This report presents the Westinghouse Electric Company developed three-dimensional methodology for the analysis of the rod ejection accident for pressurized water reactors. The report is structured into four major sections, a list of references and four appendices. A brief overview of the content of each of these sections follows:

1.0	Introduction	A brief discussion of the accident and associated limits. It also discusses the current methodology used by Westinghouse and its limitations.
2.0	Multi-dimensional Analysis Methods	This section describes the proposed Westinghouse methodology using 3-D kinetics.
3.0	Sample Application of 3-D Methodology	Sample calculations were performed to demonstrate the methods. The calculational results are representative and are not intended for the licensing of any specific reactor unit.
4.0	Summary and Conclusions	A concise overview of the applicability of the methodology is presented in this section.
5.0	References	A list of references is provided which documents the pertinent reports and papers which are referenced throughout this report.
Appendix A	Overview of Computer Codes	Although both computer codes being used in this methodology are currently approved for use by the NRC, this appendix provides some background on the codes being used and the data interchange between the codes.
Appendix B	Qualification of Transient Analysis Method	An OECD rod ejection benchmark problem was analyzed using the computer codes described in Appendix A. The Westinghouse results are compared to the reference results in this appendix.
Appendix C	Rod Ejection Sensitivity Studies	This appendix provides background information on the sensitivities of the key factors which impact the rod ejection transient.
Appendix D	Comparison of 3-D and 1-D Analysis Method	The rod ejection transient, as modeled using current licensed 1-D methodology, is compared to the transient as modeled in 3-D. The key contributors to the differences are noted. A hot rod calculation is performed which demonstrates the consistency of these two methods.

1.0 Introduction

1.1 Objective

The rod ejection accident is a design basis reactivity insertion event for pressurized water reactors (PWRs). An analysis of the consequences of the event is typically presented in Chapter 15 of a plant Final Safety Analysis Report (FSAR). Acceptable analysis methods and criteria for the event are described in the Nuclear Regulatory Commission (NRC) Regulatory Guide 1.77 (Ref. 1), and the NRC review process for the FSAR analysis is described in the NRC Standard Review Plan 15.4.8 (Ref. 2). The current Westinghouse analysis methodology of the FSAR rod ejection accident, based upon the conservative one-dimensional (1-D) (axial) core neutron kinetics method, is described in the NRC-approved Topical Report WCAP-7588 Rev.1A (Ref. 3) and in the NRC-approved Topical Report CENPD-190 (Ref. 4).

As the discharge burnup of PWR fuel assemblies increases, some acceptance criteria of the rod ejection accident may need to be revised to account for the changes in fuel and clad behavior with irradiation. The current 1-D method may be too conservative to demonstrate that the revised acceptance criteria are met during the rod ejection accident for the high burnup fuel. The purpose of this report is to present for licensing approval a revised methodology for the FSAR rod ejection accident analysis, based on a three-dimensional core neutron kinetics method using the NRC-approved neutron kinetic code SPNOVA (Ref. 5 & 6) and the NRC-approved core thermal-hydraulic code VIPRE-01 (VIPRE) (Ref. 7 & 8). This report demonstrates that with the revised methodology and the current PWR protection system, there are significant margins in the safety analysis to the current and the postulated acceptance criteria in compliance with the General Design Criterion 28, "Reactivity Limits," of Appendix A to 10 CFR Part 50.

The phenomena that are of importance in determining the consequences of a rod ejection accident, particularly in high burnup fuel cores, have been identified in the NRC's PIRT (Phenomenon Identification and Ranking Tables) for this accident (Ref. 9). The 3-D methodology presented in this report is consistent with the identified phenomena.

As part of an industry coordinated effort, an Electric Power Research Institute (EPRI) working group is formulating guidelines for the analysis of reactivity insertion transients, particularly the hot zero power prompt-critical event. Westinghouse has participated in this working group, and the general guidelines as defined by the working group are expected to be consistent with the methodology described in this report.

1.2 Accident Description

This accident is defined as the mechanical failure of a control rod mechanism pressure housing such that the reactor coolant system pressure would cause the ejection of a partially or fully inserted control rod and drive shaft to its fully withdrawn position. If the reactor is at or near critical, the consequences of this mechanical failure are a rapid reactivity insertion and core power increase together with an adverse core power distribution, possibly leading to localized fuel rod damage. The power increase is arrested

primarily by the negative reactivity due to the Doppler feedback resulting from the fuel heatup, and the transient is terminated by a reactor trip which is initiated shortly after the beginning of the transient.

Due to the extremely low probability of a rod cluster control assembly ejection, this accident is classified as a Condition IV (limiting fault) event as defined by the American Nuclear Society *Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants* (ANSI N18.2 – 1973)(Ref. 10).

1.3 Reactor Protection

The position of all control rods is continuously indicated in the control room and an alarm will occur if one rod deviates from its bank demand position by more than 5 percent of span. There are low and low-low insertion limit monitors with visual and audio signals. Operating instructions require boration at the low limit alarm, and emergency boration at the low-low limit alarm. These alarm functions ensure that the accident will not be worse than the cases analyzed. Should a rod ejection event occur while the reactor is at or near critical, the transient will typically be terminated by one or more of the following automatic features of the Reactor Protection System. The protection system features are described in the FSAR. Typical features for Westinghouse-designed plants are summarized below; the protection system for other PWRs has a similar functionality:

1. Power Range High Neutron Flux Reactor Trip (low setting) – actuated when two out of four power range channels indicate a power level above a preset nominal setpoint (typically 25% of full power). This trip may be manually bypassed when two out of four power range channels indicate a power level above the permissive P-10 setpoint (approximately 10% of full power), and is automatically reinstated when three of four channels indicate a power level below 10% power.

2. Power Range High Neutron Flux Reactor Trip (high setting) - actuated when two out of four power range channels indicate a power level above a preset nominal setpoint (typically 109% of full power). This trip function is always active.

3. High Positive Nuclear Flux Rate Reactor Trip – actuated when the positive rate of change of neutron flux on two out of the four power range channels indicate a rate above the typical preset nominal setpoint (typically 5% in 2 seconds). This trip function is always active.

A reactor trip may also occur on low pressurizer pressure as a result of the Reactor Coolant System (RCS) depressurization caused by the failure in the control rod pressure housing. The continued depressurization would eventually cause actuation of the Emergency Core Cooling System (ECCS). While the feedback effects within the core are sufficient to turn the transient around, the negative reactivity provided by the reactor trip more than offsets the reactivity insertion due to the rod ejection and therefore maintains the reactor in a shutdown condition.

1.4 Accident Limits

The real physical limits of this accident are that any consequential damage to either the core or the reactor coolant system must not prevent long-term core cooling and that any off-site dose consequences must be within the guidelines of 10 CFR Part 100. More specific criteria are applied to ensure that there is no significant fuel dispersal in the coolant, gross fuel lattice distortion, or severe shock waves. Acceptable limiting criteria for this accident have been defined in NRC Regulatory Guide 1.77. However, Westinghouse typically has applied the following conservative criteria:

1. The average fuel pellet enthalpy at the hot spot shall be below 200 cal/g (360 Btu/lbm) for irradiated or unirradiated fuel. (Note: Regulatory Guide 1.77 allows a higher limit of 280 cal/g.)

2. The peak reactor coolant pressure shall be less than that which would cause stresses to exceed the Faulted Condition stress limits. (Note: Westinghouse plants actually meet the more stringent Emergency Condition stress limits as specified by Reg. Guide 1.77.)

3. Fuel melting will be limited to less than the innermost 10 percent of the fuel pellet at the hot spot, even if the average fuel pellet enthalpy at the hot spot is below the limits of Criterion 1.

Some plants have used enthalpy as alternate criteria for fuel failure (200 cal/g and 250 cal/g for clad failure and incipient centerline melt, respectively). These criteria have been used in numerous plant FSARs which have been reviewed and accepted by the NRC. While Westinghouse historically has chosen to utilize the above conservative criteria, the NRC criteria represent the necessary and sufficient safety limits for this accident.

The allowable dose consequences for the event are as given in Regulatory Guide 1.77. The number of fuel failures for the dose evaluation is based on the number of fuel rods reaching Departure from Nucleate Boiling (DNB), as discussed in the NRC's Standard Review Plan SRP 15.4.8.

Recent tests of rapid (prompt-critical) reactivity insertion events with highly irradiated fuel have indicated that the current NRC peak fuel enthalpy criterion may not be conservative. Therefore, various new criteria are being proposed by the Industry through EPRI for the allowable fuel enthalpy increase as a function of fuel burnup or clad oxidation. The revised criteria are expected to apply only to the zero or very low power prompt-critical case, and may affect the fuel failure limit as well as the coolability limit. The revised criteria are not currently available, but will be adopted by Westinghouse when they become finalized. In the interim, until new limits are defined by the Industry-EPRI program and approved by the NRC, a conservatism will be applied in the Westinghouse three-dimensional (3-D) methodology to the peak fuel enthalpy limit for the Hot Zero Power (HZP) rod ejection transient. This interim conservative adjustment is to limit the peak fuel enthalpy increase to 100 cal/g, which is less than the anticipated future criteria. This is equivalent to reducing the current Westinghouse limit of 200 cal/g peak radially averaged fuel enthalpy by 41.25% (82.5 cal/g). The full power or non-prompt-critical cases will continue to use the current licensed peak fuel enthalpy criteria.

1.5 Current Analysis Methods

The current Westinghouse licensing-basis analysis method for the rod ejection event are described in References 3 and 4. These methods use a 3-D static nuclear design model to calculate the ejected rod worth and associated peaking factor, and the Doppler and moderator feedback. In order to bound future reload cycles for an individual plant, even more conservative bounding analysis parameters may be chosen which are not expected to be exceeded. These values are then utilized with appropriate uncertainty allowances in a very conservative transient analysis calculation of the core and fuel behavior. The parameters are checked every cycle to ensure the analysis remains bounding. The current analysis methods are described in more detail below. The methods described are those that have been applied to Westinghouse constructed plants; although differing in detail, the methods applied to CE constructed plants (Ref. 4) are similar.

1.5.1 Static Calculations

The key safety parameters of a single rod ejection are evaluated at the beginning and end of each reload fuel cycle using the three-dimensional nodal code ANC (Ref. 11, 12, 13, 14). The evaluation considers both full power and zero power initial conditions with the control banks at their respective insertion limits. Spatial peaking factors, control bank worths and ejected rod worths are derived from multi-dimensional neutronic calculations. For the rod ejection event, the peaking factors are calculated with an "adiabatic" assumption. That is, the nuclear feedback, both Doppler and moderator, is established during the initial condition calculations and this feedback is not allowed to vary when the rod is ejected.

The fuel temperature (Doppler) coefficient is defined as the change in reactivity per degree change in effective fuel temperature. It is primarily a measure of the Doppler broadening of U^{238} and Pu^{240} resonance absorption peaks. The fuel temperature coefficient is calculated by performing two-group multi-dimensional neutronic calculations. Moderator temperature is held constant and power level is varied. The spatial variation of fuel temperature is taken into account by calculating the effective fuel temperature as a function of local power density throughout the core.

The effective delayed neutron fraction for the entire core is obtained by weighting the delayed neutron fraction for different fissionable isotopes by the fraction of fissions in each isotope and the power sharing in the core.

1.5.2 Transient Calculations

The calculation of the rod ejection transient is performed in two stages, first an average core channel calculation and then a hot rod calculation. The average core calculation is performed using the TWINKLE (Ref. 15) code in the one-dimensional (axial) mode to determine the core average power generation with time including the various core feedback effects, i.e., Doppler and moderator reactivities. A Doppler weighting factor is applied to the Doppler feedback to compensate for the missing dimensions. Enthalpy and temperature transients in the hot spot are then determined by multiplying the average core energy generation by the hot channel factor and performing a fuel rod transient heat transfer calculation

using the FACTRAN (Ref. 16) code. The power distribution calculated without feedback is conservatively assumed to persist throughout the transient. The DNB heat flux is not calculated; instead DNB is conservatively assumed to occur near the start of the transient. Cases at beginning and end of cycle, full and zero power initial conditions, are analyzed. Input values for the ejected rod worth, peaking factors, delayed neutron fraction and feedback coefficients are determined using the nuclear design methods as discussed in the preceding subsection. Appropriate uncertainty allowances are applied to the parameters used in the calculation.

A more detailed discussion of the current method of analysis can be found in typical plant FSARs, and in the Westinghouse rod ejection topical report (Ref. 3).b

1.5.3 Reload Safety Evaluation

The Westinghouse Reload Safety Evaluation (RSE) methodology (Ref. 17) or its equivalent is performed to confirm the validity of a plant's current licensing basis safety analysis for each fuel cycle reload. The current licensing basis safety analysis for a plant is defined as the latest applicable analysis as presented in the FSAR, or in other licensing documentation, that supports the plant's Technical Specifications. The Westinghouse safety analysis methodology is intended to be valid and bounding for all plant fuel cycles, provided that there are no major changes to the fuel design or to the plant. Thus, the safety analysis reload related input assumptions are selected to bound the expected values for standard reload designs. This bounding analysis concept is the key to the Westinghouse reload safety analysis methodology and minimizes cycle specific safety analyses. The determination that all of the reload related safety analysis parameters for a given event are bounded ensures that the reference safety analysis remains valid for the cycle in question. On the other hand, when a reload parameter is not bounded, further evaluation is necessary in order to demonstrate that all of the applicable safety criteria continue to be met for the reload. The new 10 CFR 50.59 guidance criteria will be followed in making the determination on whether the re-evaluation/reanalysis requires prior NRC review and approval.

1.6 Need for Multi-Dimensional Methods

As discussed in Section 1.1, higher fuel burnups with increased fuel duty has required the industry to reevaluate the limits for the HZP rod ejection event. These new limits are likely to be significantly reduced, and will require the use of improved methods to clearly demonstrate the margin that exists. Advanced analytical computer codes have already been licensed by Westinghouse that will permit a more refined and appropriate analysis for these events. With these new analytical capabilities, it is no longer necessary to apply the overly conservative and sometimes inconsistent historical methodology assumptions; however, this will require the development and licensing of new methods for the application of these codes.

2.0 Multi-Dimensional Analysis Methods

The current 1-D Westinghouse methodology for the rod ejection accident is based on a number of conservative assumptions, which sometimes results in inconsistent conditions being used for parameters in the same evaluation. This results in an very conservative analysis. The revised analysis method using 3-D space-time neutronics allow the elimination of many of these inconsistencies, and results in a more realistic, although still conservative, analysis. This is not a best-estimate method, since many conservative assumptions are maintained. In addition to the use of conservative analysis assumptions, conservative uncertainty allowances are applied to key parameters. Two methods of applying the uncertainty allowances are considered. The conventional method is to apply them in a deterministic manner, i.e. simultaneously in the worst (most limiting) direction in the same calculation. In an alternative "statistical" method, their impacts may be independently determined and the overall net impact determined using a square root of the sum of the squares of the individual impacts.

Fundamental in the Westinghouse methodology is the continued use of the reload safety evaluation process. Through this process, the impact of the reload cycle can be determined from static nuclear calculations, and the transient calculations are re-performed only if the evaluated results are outside of the space defined by previously utilized key parameters. Key parameters in analysis are defined in Reference 17.

2.1 Static Nuclear Design Methods

2.1.1 Nuclear Design Depletion Model

The static nuclear methods are based on the design model utilized for the other reload safety evaluations. Typically, the End Of Cycle (EOC), HZP case is the most limiting, but other cases are also evaluated. Calculations are performed at Beginning Of Cycle (BOC) and EOC, and at Hot Full Power (HFP) and HZP.

One key modeling variable that significantly impacts the ejected rod calculations is the impact of depletion with the lead bank inserted. The depletion of the core with the lead bank inserted reduces the power and hence the burnup at the top of the core, especially in the assemblies containing the lead bank. This can significantly increase the ejected rod worth at EOC. However, this impact takes time to build up, and is based on an operation strategy of deep rod insertion for the reactor. The cycle depletion model must incorporate the impact of rodded depletion, consistent with the operations of the reactor over the cycle. Since the typical operation is HFP with the lead control bank inserted slightly to have sufficient impact on the core with a small change in position (referred to as the "bite" position), it is sufficient to use a model which has the lead bank inserted at or below the bite position for the cycle. It is necessary to deplete with deeper insertion of the lead bank only if a significant amount of load follow operation is anticipated.

Another potential history factor is the impact at BOC of the previous cycle length. Since the safety analysis calculations are generally performed prior to the shutdown of the previous cycle, the BOC evaluations need to encompass the impact of the potential variability of the previous cycle length. [

]^{a,c}.

2.1.2 Calculation of Key Transient Parameters

a. Doppler Feedback

The fuel temperature (Doppler) coefficient is defined as the change in reactivity per degree change in effective fuel temperature. It is primarily a measure of the Doppler broadening of U^{238} and Pu^{240} resonance absorption peaks. The fuel temperature coefficient is calculated by performing two-group multi-dimensional neutronics calculations. Moderator temperature is held constant and power level is varied. The spatial variation of fuel temperature is taken into account by calculating the effective fuel temperature as a function of local power density throughout the core. At a given power level, the fuel temperatures are greatest for fresh fuel and decrease as the clad creeps down on the fuel rod during burnup. Thus the total Doppler feedback coefficient is a maximum at beginning of cycle, and a minimum at end of cycle.

b. Moderator Feedback

The moderator temperature coefficient is defined as the change in reactivity per degree change in the average moderator temperature. The primary factors that affect the value are the change in moderation with the change in the water density and the change in the absorption due to the change in the soluble boron atom density with the change in the water density. The isothermal temperature coefficient is calculated by performing two-group multi-dimensional neutronics calculations. The core power level is held constant and the inlet temperature is varied. The moderator temperature coefficient is then determined by subtracting the Doppler temperature coefficient from the isothermal temperature coefficient. The moderator temperature coefficient becomes more negative with decreasing boron concentrations, and with increasing temperatures.

c. Delayed Neutron Fraction

The effective delayed neutron fraction for the entire core is obtained by weighting the delayed neutron fraction for different fissionable isotopes by the fraction of fissions in each isotope and the power sharing in the core. The delayed neutron fraction is lower for plutonium isotopes than uranium isotopes, so as the fuel depletes the delayed neutron fraction decreases.

d. Ejected Rod Worth and Peaking Factor

The ejected rod worth is dependent on the arrangement of fuel assemblies within the core, the control rod pattern, the axial power distribution due to burnup and xenon effects, and the allowed insertion limits. If

the control rods are partially inserted, the ejected rod worth increases for power distributions which skew the power to the top of the core. The core power distribution is naturally skewed slightly to the bottom of the core at full power due to the feedback. Thus a burnup skew builds in the core with cycle depletion, being maximum at end of cycle. For this reason, the end of cycle provides the most limiting axial power distributions. The ejected rod peaking factor will also increase as the ejected rod worth increases. Thus if the ejected rod worth is conservatively pessimized (increased), this also conservatively pessimizes (increases) the ejected rod peaking factor even without applying a separate additional pessimism on the peaking factors. The method of taking into account the initial xenon distribution and control rod insertion is discussed in more detail in Section 2.1.3, Reactor Core Initial Conditions.

The static analysis of the ejected rod is performed in 3-D using an "adiabatic" feedback model which maintains the Doppler and moderator feedback at the initial (pre-ejection) condition. This will generate a peaking factor which is larger than would be calculated with a transient calculation that includes some feedback. The various control bank locations are evaluated to determine the location of the worst ejected rod.

2.1.3 Reactor Core Initial Conditions

There are two key core operation parameters aside from the time of life and depletion model that have a significant effect on the ejected rod worth and peaking factor, and can be adjusted as part of the initial conditions for the analysis. These are the xenon distribution and the control rod bank positions.

The axial xenon distribution can have a significant impact on the ejected rod worth and the ejected rod peaking factor for partially-inserted rod banks. Xenon distributions that force the power distribution more to the top of the core are more limiting since they increase the axial peaking factor and increase the worth of the rod that is being ejected.

At HFP, there is a nominal operating range in which the reactor is expected to operate. This band of operation is typically defined by axial offset limits (Ref. 18) (axial shape index for CE-designed plants). Those limits can be a band around the equilibrium value, or absolute limits. Since HFP operation is the expected norm, a limiting axial xenon distribution is used in the precondition for the rod ejection evaluation. [

] a, c. This is a highly unlikely situation since it would result in the operator having no room to control the reactor, but it is a conservative bound.

At HZP, there are no limitations on the axial power distribution. Hot full power equilibrium xenon, and no xenon, result in very similar ejected rod worths. Top-skewed xenon distributions decrease the ejected rod worth. Mild bottom-skewed xenon distributions slightly increase the ejected rod worth. Therefore, an artificially skewed-to-the-bottom xenon distribution is chosen which increases the ejected rod worth beyond the no-xenon case [ ] a, c to conservatively account for adverse power distributions. An alternate approach may be utilized in which partially inserted banks are assumed to be fully inserted. This results in an overly conservative value for the ejected rod worth, but avoids the

necessity of having to establish the skewed axial xenon distributions necessary to properly calculate the worth of partially inserted banks.

The control bank insertion also has a significant role in the ejected rod worth. The ejected rod worth generally increases with increased control bank insertion for the same axial power shape. Thus, deeper insertions increase the ejected rod worth. Technical specification limits on control rod insertion, and the control rod insertion limit alarms, ensure that it is highly unlikely that the control rods will be inserted beyond the specified limits. Thus, the assumption of the control banks at their insertion limit is used as a conservative initial condition for the rod ejection accident. In order to perform a more bounding analysis where a high bounding ejected rod worth is desired, a deeper insertion can be utilized, and/or the control rod cross sections can be adjusted.

2.2 Transient Analysis Methods

The most significant difference between the current NRC-approved analysis method and the revised analysis method for this accident is the change from a 1-D to a 3-D core neutron kinetics and feedback model. This eliminates the need to apply Doppler weighting factors to the core kinetics calculation to simulate the effect of the increased Doppler feedback due to the skewed power distribution following the ejection of the rod. It also eliminates the very conservative assumption of a constant no-feedback value of the ejected rod peaking factor in the hot rod calculation. The computer codes used with the 3-D revised analysis method have already been reviewed and approved by the NRC. The nuclear model is based on the NRC-approved Westinghouse SPNOVA code (Ref. 5 & 6). The core thermal-hydraulic and fuel rod models, as well as the hot rod model, are based on the NRC approved Westinghouse version of the VIPRE-01 code and methods (Ref. 7 & 8). An overview of these computer codes is provided in Appendix A.

2.2.1 Nuclear Model

A detailed 3-D transient nuclear model, consistent with the static nuclear model, is used for the analysis. The use of a transient model which is consistent in spatial detail and feedback characteristics as the static design model gives a more accurate analysis of the actual transient. The fuel and control rod cross sections are the same as used for the static nuclear analyses. The major changes to the model which reflect the transition to a transient calculation are the input of the kinetics parameters. These are the delayed neutron fractions, the delayed neutron decay constants, the neutron velocities for each energy group, and the transient driver functions which initiate the transient and the reactor trip. [

] ^{a,c}. The overall delayed neutron fraction can be adjusted for conservatism by applying a fixed multiplier to the individual node-by-node values.

Transient driver functions are used to define the transient. These include control rod cluster movement to simulate the rod ejection, and a driver that models the control rod bank position vs. time curve for the reactor trip. Other drivers are available to allow variations in core inlet conditions (pressure, flow and

temperature) with time, but are not used for a very rapid rod ejection transient. An excore detector response capability is available which is based on the same methodology as used for the Dynamic Rod Worth Measurement (DRWM) (Ref. 19) calculations. [

]^{a,c}. The excore normalization factors are determined at HFP prior to the transient simulation. These detector signals can then be used to determine the time of the reactor trip signal.

2.2.2 Thermal-Hydraulic Model for Feedback Calculations

The thermal-hydraulic model used in the reactor core kinetics calculation includes the time dependent effects of reactor coolant flow, heat transfer from the clad to the coolant, and direct heat generation in the coolant. The calculation is performed with the VIPRE code using a mesh structure consistent with the nuclear analysis mesh structure. As the rod ejection event is a very rapid transient, the most of the thermal energy is retained in the fuel rod. Thus the coolant temperature increase is relatively small. Some of the fission energy is deposited directly into the coolant through the slowing down of the fission neutrons and the absorption of gamma rays accompanying the fission process. This can become significant for transients that result in very high peak nuclear power increases. This is taken into account in the calculation.

Since the principle feedback mechanism is due to the Doppler feedback, thermal hydraulic modeling assumptions are made for the node average which maximize heat transfer from the clad to the coolant. This includes the assumption of full reactor coolant flow (no Reactor Coolant Pumps (RCPs) out-of-service) in the hot zero power rod ejection case, and no initiation of DNB heat transfer. [

]^{a,c}.

For the HZP case, the major factor turning the transient around is the Doppler feedback, which is directly related to the rapid increase in fuel pellet temperature. Under prompt-critical conditions, the fuel rod transient is nearly adiabatic; that is, the details of the heat transfer from the fuel to the clad and the clad to the coolant is not of high importance. For the much slower HFP case, both the Doppler and moderator feedback are important, particularly at end of cycle with a very negative moderator temperature coefficient. Therefore, for the HFP case, the fuel rod internal heat transfer and the heat transfer to the clad and coolant are important.

The average fuel rod model for the feedback calculation is performed with the VIPRE code using a multi-zone fuel pellet representation for the fuel rod in each neutronics/thermal-hydraulic core node. The fuel rod model typically uses [ ]^{a,c} mesh points in the fuel and two points in the clad. The fuel pellet-to-clad gap heat transfer is calculated using the dynamic gap conductance model in VIPRE that accounts for changes in the fuel dimensions and fill gas pressure with temperature. Design values of pellet radial power distributions, based on assembly-average burnup, are input for each fuel assembly. The resonance effective temperature is generated at each spatial node from the radially varying temperatures using design values of the T_{eff} weighting function. For consistency with the static nuclear design model, the VIPRE average fuel rod model is calibrated against the nominal design static fuel rod

model temperatures over the power range of interest [

]^{a,c}. This calibration is performed for the typical fuel compositions in the core, and as a function of fuel depletion.

To conservatively pessimize the transient and cover the uncertainties in the actual T_{eff} calculation, an input multiplier on the Doppler feedback cross section adjustment is applied. This allows a uniform uncertainty allowance to be applied on the Doppler feedback adjustments. Similarly, the core parameters are adjusted to make the moderator temperature more positive to conservatively represent the moderator density feedback effect.

2.2.3 Hot Fuel Rod Model for the Peak Fuel Enthalpy Calculation

The hot fuel rod thermal calculation is performed independent of the node average fuel temperature feedback calculations, with additional conservatism applied to the modeling and initial conditions in order to maximize the increase in fuel temperature and enthalpy. The key limit for the accident is the calculated radially-averaged peak fuel enthalpy (RAPFE), or the maximum change in fuel enthalpy, depending on the criterion. The hot rod model uses the same fuel pellet and clad mesh description as for the average rod.

The hot fuel rod model is based on the NRC-approved model described in the Westinghouse VIPRE modeling topical report (Ref. 7), and is similar to the model used in the FACTRAN code (Ref. 3 & 16). It represents the hottest fuel rod from an assembly in the core. The pellet-to-clad gap heat transfer is calculated using the dynamic gap model in VIPRE, which is comparable to the very conservative NRC-approved FACTRAN transient gap model. In either case, the model is calibrated against the design fuel rod temperatures as generated by a fuel performance code such as the PAD program (Ref. 20) using the method described above for the average rod model. As for current plant licensing applications, the heat transfer to the coolant is calculated using the Dittus-Boelter correlation for single phase forced convection, the Thom correlation for nucleate boiling, and the Bishop-Sandberg-Tong correlation (Ref. 21) for transition and film boiling beyond Departure from Nucleate Boiling (DNB). In order to maximize the temperature and enthalpy increase within the fuel pellets, the hot spot of the fuel rod is in post-DNB film boiling during the transient. The Baker-Just correlation (Ref. 22) is used to account for heat generation in the cladding material due to the zirconium-water reaction.

A benefit in this method compared to the current licensed analysis methodology based on a 1-D kinetics is that time-dependent core average power, rod peak power, and axial power distributions are taken directly from the 3-D kinetics results for the hot rod transient calculation instead of using a very conservative constant peak value. The calculation can be performed for the hot rod in the hottest assembly (the one with the peak fuel enthalpy), or for different fuel assemblies in the core at various levels of burnup.

2.2.4 Initial Conditions and Accident Assumptions

The initial conditions are the same as those used in the static analysis of the rod ejection, with the addition of the conservatisms described in Section 2.1.2 and the accident-specific analysis assumptions described below.

a. Initial Power Level

The accident is analyzed with the reactor at either HZP or HFP initial conditions. The initial power level for the HZP case is assumed to be [ ] a, c times nominal full power, or lower. No uncertainty is applied since the results are insensitive to the exact value of power once the level is below the range of sensible heat generation. For the HFP case, the core neutronics calculation is performed at 100% of nominal power. The uncertainty allowance for calorimetric errors is applied to the hot rod calculation.

b. Initial RCS Flow Rate

As discussed in Section 2.2.2, thermal-hydraulic model assumptions are made in the VIPRE feedback calculation model which maximize heat transfer from the clad to the coolant. This includes full reactor coolant flow (no RCPs out-of-service, even if allowed by the plant Technical Specifications).

For the hot rod fuel rod model, one or more RCPs are conservatively assumed out-of-service for the HZP case if allowed by the plant Technical Specifications. Otherwise, full RCS thermal design flow is assumed. The thermal design flow accounts for uncertainties in the RCS hydraulic resistances and flow measurements.

c. Initial Core Inlet Temperature and Pressure

For the core neutron kinetics calculation, the core inlet temperature and outlet pressure are assumed constant at their nominal values consistent with the nuclear design model. Typical uncertainty allowances, at the initial conditions, in inlet temperature and pressure (as defined in the FSAR) are applied to the hot rod calculation.

d. Ejected Rod Simulation

The control rod is assumed to be ejected in 0.1 second. This corresponds to the time for a fully-inserted rod to be ejected from the core when the full RCS differential pressure is acting on the rod. The time of ejection is not a critical parameter. A more rapid ejection has no impact on the HZP peak fuel enthalpy results and a very small effect on the full power case results.

e. Reactor Trip

The reactor trip is simulated by dropping in the partially or fully withdrawn rod banks using conservative control rod cluster acceleration and terminal velocity which are consistent with the plant Technical

Specifications. Additional conservatism in the trip is added by assuming that the most reactive control rod does not trip; this is a rod adjacent to the ejected rod. [ ]^{a,c}.

f. Reactor Trip Point and Trip Time Delay

The reactor trip is assumed to occur when the appropriate number of ex-core detector signals reach the trip setpoint plus the conservative error allowance. Reactor trip setpoints, error allowances, and trip time delays are given in the individual plant Technical Specifications. An increased uncertainty in the trip setpoints typically has only a small effect on the transient since this will introduce only a small additional delay in the reactor trip. The typical trip time delay is 0.5 seconds from reaching the trip point until the start of rod motion. In the analysis, it is conservatively assumed that the highest-response power range detector fails to actuate, thus requiring two other detectors to indicate sufficient power to trip the reactor.

2.2.5 Application of Conservative Allowances

Conservative allowances on the key analysis parameters may be applied in the calculation using either a "deterministic" or a "statistical" approach. In the "deterministic" method, the uncertainties in the key parameters are applied in the conservative direction simultaneously in the calculation. This leads to a very conservative result, since the key parameters are not all expected to be at their limiting value at the same time. A more reasonable analysis approach is to perform a "base case" calculation without the uncertainty allowances, and then apply the uncertainty allowances to the calculation one at a time to generate the explicit impacts on the analysis limit of interest (e.g. the resulting increase in the peak fuel enthalpy). The net impact is then determined by combining the individual effects statistically using the square root of the sum of the squares method and applying the result to the peak fuel enthalpy calculated in the base case. The key parameters are well known and discussed by Diamond (Ref. 23).

The conservative allowances and their method of application which will be applied to the key analysis parameters are shown below:

- The Doppler feedback will be reduced by [ ]^{a,c} by applying a multiplier to the change in the fast absorption cross section for the given change in the calculated fuel effective temperature.

- The moderator coefficient will be increased by at least [ ]^{a,c} by increasing the core soluble boron concentration. Alternatively, a multiplier can be made to the feedback adjustment term to provide a similar adjustment.

- The delayed neutron fraction will be reduced by [ ]^{a,c} by applying a uniform multiplier to the node-by-node values.

- The ejected rod worth is increased from the nominal condition by [ ]^{a,c}. This may be accomplished through the application of a skewed xenon distribution, deeper control bank insertion

or an adjustment to the control rod cross sections. This results in an increase of the static ejected rod peaking factors [ ]^{a,c}.

Other conservative allowances that will be applied to other parameters are:

- [ ]^{a,c}.

- The trip function uncertainties are the same as have been applied in the current analysis method. These include a conservative control rod cluster acceleration and terminal velocity which are consistent with the plant Technical Specifications, a reactor trip setpoint including Technical Specification uncertainties, a reactor trip signal based on assuming a failure of the best (largest response) detector, and a half second trip delay time.

- The hot rod DNBR (Departure from Nucleate Boiling Ratio) calculation will use the same uncertainty allowances as for the current licensing applications (Ref. 24). Typically the uncertainties include an increase in the reactor power, an increase in reactor coolant temperature at hot full power, a decrease in RCS pressure, an increase in the radial peaking factor and an engineering factor on the peak location. Plant-specific values will be applied on an individual plant basis.

- The hot rod radially averaged fuel enthalpy calculation will apply the standard uncertainties, including:

  - Discrete grid bias term (dependent on the grid type)
  - Local peaking factor uncertainty
  - Local engineering peaking factor penalty
  - Core calorimetric uncertainty for hot full power calculations

  Note that the peaking factor terms are included in a square root of the sum of the squares combination. [ ]^{a,c}.

- Recognizing that the new HZP rod ejection fuel enthalpy limit for high burnup fuel has not been established as of Original Version: February 2002, an analysis limit of 117.5 cal/g (82.5 cal/g lower than the current Westinghouse limit) will be used as an interim limit for the HZP cases.

Using the above assumptions, the transient is evaluated starting from a highly unlikely initial condition. This ensures a conservative evaluation of the transient consequences.

2.2.6 Summary of Transient Analysis Method

The transient analysis uses a number of conservatisms to ensure that the evaluated transient is bounding for the core. The conservatisms are implemented with the selection of the initial conditions, assumptions on the behavior of the plant systems, and conservative allowances used in the evaluation of the transient. The following table summarizes the elements and how they are applied with the recommended methodology, consistent with Regulatory Guide 1.77. The conservatisms applied to the hot rod calculations are similar to those used for current plant safety analyses.

Table 2.1 3-D Methodology Elements

Elements of Reg. Guide 1.77*	3-D Methodology
A. Initial Core Conditions Zero Power (BOC & EOC) Low Power (BOC & EOC) Full Power (BOC & EOC)	The full power transient must be evaluated at BOC and EOC to determine the most limiting conditions. The HFP hot rod evaluation includes the uncertainty on calorimetric power. The EOC HZP case is the most limiting of the HZP and low power transients. The BOC ejected rod worth can be compared to the EOC value to demonstrate that the EOC transient will be the most limiting. The initial core conditions need to address the potential operational history of the core. This includes consideration of the previous cycle length variation on the BOC cases and operation with control rods inserted for the EOC cases.
B. Initial Loss of Primary System Integrity	The RCS overpressure will be evaluated in the same manner as with the 1-D methodology.
C. Ejected Rod Worth	
a) maximum inserted position based on power level	a) The control rod positions will be consistent with the insertion limits.
b) additional fully inserted or partially inserted misaligned or inoperable rods, if allowed	b) The rod positioning at the insertion limit is already below that of normally expected positions, thus no additional misalignment term is added.
c) increase worth to account for calculational uncertainties	c) The uncertainty due to cross-sections is included in the uncertainty in the beta-effective. This is consistent with how the control rod worths are measured.
d) increase worth to account for xenon transients	d) The largest variation in the control rod worths is due to power distribution variations caused by transient xenon distributions, quadrant power tilts, or other similar factors. The HZP ejected rod worth is determined assuming nominal conditions (no xenon) and then increased by [ ]^{a,c} to account for the impact of potential adverse power distributions. The HFP initial condition assumes the control rods at their insertion limit consistent with a xenon distribution which results in the axial offset being at its most positive allowed value.

* Elements of Reg. Guide 1.77 as consistent with the EPRI working group 3-D methodology guidelines.

Table 2.1 3-D Methodology Elements (cont.)

Elements of Reg. Guide 1.77*	3-D Methodology
D. Reactivity Insertion Rate a) based on differential worth curve and rod position vs. time curve b) rate of ejection based on maximum ? P and weight and cross-sectional area of the control rod and drive shaft	The 3-D calculation inherently includes the rod worth as a function of position. The rod ejection time is taken to be 0.1 seconds which is consistent with the physical parameters. Variations in the speed has minimal impact on the net outcome of the transient.
E. Effective Delayed Neutron Fraction and Prompt Neutron Lifetime	The prompt neutron lifetime is replaced by the neutron velocity when using a 3-D model. Although the core transient will change with variations in the neutron velocity, the net impact on the fuel enthalpy is small. Therefore, design values are used.
a) use available data and average based on fission fractions	a) Use of the 3-D model allows for the delayed neutron fractions to vary with core position.
b) use minimum calculated value for the given reactor state	b) Values consistent with the time in life will be used. In addition, the delayed neutron fraction will be further reduced by [ ]^{a,c}.
c) consider both the power excursion and the power reduction when selecting a conservative value	c) Use of the 3-D model allows for the delayed neutron fractions to vary with core position, thus changes in the core effective value due to flux distribution changes are automatically taken into account.
F. Initial Pressure, Flow and Temperature	The nominal core values are used for the 3-D transient. Conservative values are chosen for the hot rod analysis, with uncertainties applied in the limiting direction.
G. Fuel thermal properties a) fuel-clad gap heat transfer coefficient b) fuel thermal conductivity c) direct moderator heating	Nominal values are used in the 3-D transient. The hot rod evaluation uses conservative values consistent with high fuel temperatures for the maximum fuel enthalpy calculation.
H. UO_2 Specific Heat	Use nominal values in the transients.

* Elements of Reg. Guide 1.77 as consistent with the EPRI working group 3-D methodology guidelines.

Table 2.1 3-D Methodology Elements (cont.)

Elements of Reg. Guide 1.77*	3-D Methodology
I. Moderator Reactivity Coefficient to include effects of voids, pressure, temperature and boron	The feedback is modified to make the moderator temperature coefficient more positive by [ ]^{a,c}. This can be done by either adjusting the feedback corrections, or by increasing the boron concentration in the 3-D model.
J. Doppler Coefficient to include corrections for pin shadowing and should compare conservatively to data. Uncertainty in fuel temperature to be included.	The Doppler feedback correction is adjusted and is reduced by [ ]^{a,c}.
K. Control Rod Reactivity Insertion on reactor trip to include initial position, differential worth curve, etc.	The 3-D model accounts for the variation of trip reactivity with position. The trip rod insertion time is taken to be the maximum allowed by the Technical Specifications. The trip reactivity is further pessimized by assuming a control rod cluster adjacent to the ejected rod cluster is stuck and both rods are not trippable.
L. Reactor Trip Delay Time	The conservative trip delay time, typically 0.5 seconds, is assumed. In addition, the impact of the asymmetric power distribution is accounted for in the 3-D calculation with the effective excore signal determined for each detector. It is conservatively assumed that one detector is out of service, so that three out of the four detectors must indicate a trip.
M. Computer Code a) coupled thermal/ hydraulic/ nuclear model b) all reactivity feedback mechanisms c) at least 6 delayed neutron groups d) axial and radial nodes e) coolant flow modeled f) trip on flux or pressure	These are addressed by the use of a 3-D transient code.

* Elements of Reg. Guide 1.77 as consistent with the EPRI working group 3-D methodology guidelines.

Table 2.1 3-D Methodology Elements (cont.)

Elements of Reg. Guide 1.77*	3-D Methodology
N. Analytical Models and Computer Codes	
a) Documented and justified	a) The SPNOVA and VIPRE codes have previously been approved for use by the NRC, and their usage for rod ejection is discussed in Appendix A.
b) Conservatism evaluated by comparison with experiment or more sophisticated codes	b) The uncertainties in codes were addressed in the previously approved topical reports. A comparison with an HZP rod ejection benchmark problem is presented in Appendix B.
c) Changes in flux shapes should be investigated.	c) Appendix C discusses the sensitivities of the calculated results to many parameters, including flux shapes.
d) Conservatism of the flux shapes used for reactivity input and feedback, peak energy deposition, total energy and gross heat transfer to the coolant should be evaluated.	d) The 3-D codes take into account the actual flux distributions to obtain the effective feedback. Different fuel rod models are used for the feedback parameters and for the hot rod evaluation, allowing for a more pessimistic hot rod calculation.
e) Sensitivity studies on Doppler, power distribution, fuel heat transfer parameters and other relevant parameters should be included.	e) Appendix C discusses the sensitivities of the calculated results to many parameters, including Doppler and moderator impacts.
O. Pressure Surge	This is calculated based on the net volume increase due to the heating of the coolant, and is consistent with the current licensed 1-D analysis method.
P. Pin Census	The use of a 3-D code allows for the calculation of a pin census directly.

* **Elements of Reg. Guide 1.77 as consistent with the EPRI working group 3-D methodology guidelines.**

Table 2.1 3-D Methodology Elements (cont.)

Additional Items*	3-D Methodology
AA. Initial Power Distribution/ Cross Sections	The initial power distribution is consistent with the allowed technical specifications and core operating limits. Sustained operation with control rods inserted is addressed by the methodology.
BB. Pin-to-Node Factor	The pin power reconstruction method is used to generate the peak rod power.
CC. Reload Checks	The conventional approach will be used to confirm that the analysis of record remains valid for each reload core. If the confirmation is unsuccessful, then the analysis must be revised, or the core redesigned.
DD. Pellet-gap Model	The nominal pellet gap model will be used for the node average temperature calculation. A conservative model will be used for the hot rod calculation.
EE. Onset of DNB	The onset of DNB will be conservatively calculated to force the hot spot into post-DNB film boiling during the transient.
FF. Calculation of ? cal/g	A detailed hot rod model with conservative heat transfer properties will be used to determine the fuel enthalpy increase.
GG. Fuel pellet radial power profile	The design radial power profile will be used for the node average fuel temperature calculation. Also, the fuel effective resonance temperature will be calculated using the design radial weighting factors. The hot rod radial power distribution will use a conservative profile which increases the average fuel temperature, and hence maximizes the fuel enthalpy.

* **Additional Items as consistent with the EPRI working group 3-D methodology guidelines.**

2.3 RCS Overpressure Evaluation Method

NRC Regulatory Guide 1.77 states that the peak reactor coolant system (RCS) pressure during the rod ejection accident should not exceed the Emergency Condition stress limits as defined in Section III of the ASME Boiler and Pressure Vessel Code. This is more stringent than the Faulted Condition stress limits which are normally expected to be applicable for a Condition IV event such as rod ejection. An evaluation of the RCS components for the ATWS event has shown that a pressure of 3200 psig (3215 psia) can be sustained without resulting in a stress level exceeding the Service Level C (Emergency Condition) stress limits (Ref. 25). Since the SPNOVA/VIPRE code does not contain a reactor coolant loop model, a calculated value of the peak pressure reached during the accident is not available. However, the results can be used to determine the volumetric surge out of the core vs. time. The calculation is performed using conventional heat transfer from the fuel taking into account the prompt heat generation in the coolant. Since the core inlet conditions are constant over the time of interest, the volumetric surge can be computed based on the change in the core fluid density and total mass with time. The volumetric surge can then be compared to the design pressure-relief surge rate for the plant, and if the calculated peak surge rate does not exceed the design value, then it can be concluded that the peak RCS pressurizer pressure does not exceed 110% of the RCS design pressure, which is well below the Emergency Condition stress limit. If desired, the RCS pressure transient can be calculated by using the VIPRE-predicted core heat flux vs. time in an RCS loop analysis computer code. In these calculations, no credit is taken for the pressure reduction due to the failure of the control rod pressure housing.

An existing RCS overpressure evaluation of record will continue to be used if the core power transient from the 3-D evaluation is bounded by the transient from the reference case used in the existing overpressure evaluation.

2.4 Rods in DNB Evaluation Method

The fission product inventory in the fuel rods calculated to experience DNB condition is an input to the radiological evaluation (see Section 2.5 below). The radiological criteria used in the evaluation of control rod ejection accidents are given in Appendix B of Regulatory Guide 1.77.

The method for evaluating the amount of fuel rods in DNB remains unchanged from that used for current plant analyses. A fuel rod is conservatively assumed to be in DNB if its minimum DNBR is less than the DNBR limit on a 95/95 basis during the rod ejection transient. The DNBR calculations are performed with different rod power factors using the VIPRE code, a DNB correlation applicable to the fuel geometry, and the design methods described in References 7 and 24. The amount of fuel rods in DNB is determined by summarizing the fuel rods with power factors greater or equal to that of the fuel rod having DNBR less than the limit.

2.5 Radiological Consequences

The radiological consequences are evaluated based on the number of fuel cladding failures predicted for the accident. The cladding may fail as a result of reaching DNB, or by exceeding some accident-specific

failure limit. As required by Regulatory Guide 1.77, fuel rods reaching DNB are assumed to fail, regardless of the time in DNB or peak clad temperature reached. This is considered to be very conservative for a short-duration event such as the rod ejection accident. The number of rods in DNB is a concern only for the full-power cases, where the margin to DNB is smaller. For low or zero power cases, where the transient is very rapid, fuel rods are more likely to fail as a result of brittle fracture of the cladding, particularly at end of cycle with high fuel burnups. Thus for the HZP case, failure criteria are set, based on a fuel enthalpy criterion. The number of failures for the HZP case is assessed by determining the number of fuel rods exceeding the failure limit for these cases. The radiological evaluation is also sensitive to the amount of fuel melting that occurs. Based on experience using the current 1-D analysis methodology, fuel melting is only predicted in the HFP cases as a result of reaching DNB. With the revised 3-D analysis methodology, once the number of fuel failures and extent of fuel melting (if any) is determined, the radiological evaluation is performed following the guidelines of Regulatory Guide 1.77 (or Regulatory Guide 1.183 for plants which have implemented the alternate source term).

The percentage of the core which is assumed to have fuel failure for the radiological evaluation will not be changed. It will be demonstrated that the 3-D analysis results in failure rates beneath this limit, thus the radiological evaluation of record would remain applicable and would not be redone.

2.6 Reload Safety Evaluation

The Westinghouse RSE methodology uses a bounding analysis approach which is characterized by key parameters determined from a static analysis to ascertain if a detailed transient case should be analyzed for the current cycle (Ref. 17). The key parameters for the rod ejection transient that are variable from cycle to cycle, assuming no change in plant operating characteristics or fuel type, are:

- Ejected rod worth
- Ejected rod peaking factor
- Delayed neutron fraction
- Doppler temperature coefficient
- Moderator temperature coefficient

[

]^{a,c}.

The reference bounding safety evaluation calculation may be performed with more conservative values for these key parameters through the use of more conservative allowances. If an evaluation using the cycle specific static values is less limiting than the reference bounding analysis of record, then a cycle specific transient analysis does not need to be performed.

If the plant operating characteristics (power, temperature, pressure, flow, design peaking factors, etc.) or the fuel type or characteristics (clad diameter and thickness, pellet diameter, grid) should change, this is identified in the RSE methodology, and more variables are evaluated to determine if the analysis must be repeated.

2.7 Applicability to Various Reactor Types

The basic 3-D rod ejection methodology, as defined in this report, is applicable to all pressurized water reactors. The transient is a very rapid transient, and as such there is no loop impact on the course of the transient. Therefore, variations in the primary and secondary system have no impact on the course of the transient. The key parameters of interest for the transient are:

- Ejected rod worth
- Ejected rod peaking factor
- Delayed neutron fraction
- Doppler temperature coefficient
- Moderator temperature coefficient

These parameters are all associated with the reactor core, and the use of the 3-D methodology requires the generation of a detailed 3-D core model. Thus, these parameters are implicitly handled by the nuclear model.

The control rod ejection time is chosen to be fast enough to be of no consequence to the actual transient, so this is insensitive to the control rod cluster geometry and rodlet composition. The modeling of the excore detectors and the determination of the time of trip are dependent on the type of plant, excore detector geometry and the protection system setpoints and allowances. The position of the tripped rods versus time is also dependent on the control rod cluster geometry and rodlet composition. The trip behavior thus is plant specific, and modeled as such. And as noted in this report, the control rod trip has only a secondary impact on the limiting parameters calculated for the transient.

The fuel rod model will also be plant dependent, but are defined by the fuel rod geometric and material properties, and the fuel temperature steady-state models are normalized to the design code values.

Thus, although there will be small differences in the models used for different PWRs, these changes are clearly identified primarily by the geometric differences. The 3-D methodology thus defined is therefore independent of the PWR plant type.

3.0 Sample Application of 3-D Methodology

3.1 Core Description

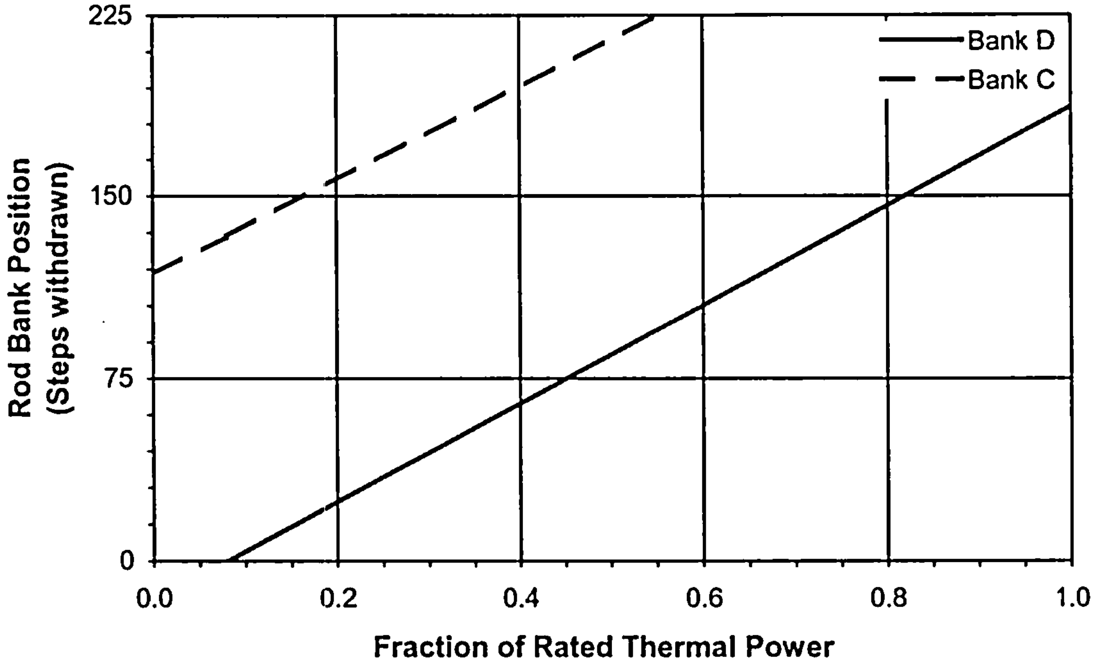
The selected core for the sample application is a Westinghouse 3-loop core with a 17x17 assembly and an 8-cluster lead control bank (Bank D). This core has typically been one of the most limiting for the HZP rod ejection transient. The core geometry and control cluster locations (control banks A, B, C, D and shutdown banks SA, SB) are shown in Figure 3.1. The shaded fuel assembly cluster at H-14 (or one of its symmetric counterparts) indicates the typical position of the worst ejected rod at end of cycle. The shaded fuel assembly cluster at F-10 (or one of its symmetric counterparts) indicates the typical position of the worst ejected rod at beginning of cycle.

The control bank insertion limits are presented in Figure 3.2.

Figure 3.1 Control and Shutdown Rod Locations

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1															
2						A		D		A					
3							SA		SA						
4				C		B				B		C			
5					SB						SB				
6		A		B		D		C		D		B		A	
7			SA				SB		SB				SA		
8		D				C				C				D	
9			SA				SB		SB				SA		
10		A		B		D		C		D		B		A	
11					SB						SB				
12				C		B				B		C			
13							SA		SA						
14						A		D		A					
15															

Figure 3.2 Control Rod Insertion Limits as a Function of Power



3.2 Static Analysis

The static analysis is performed to determine the worst ejected rod cluster and the appropriate preconditions for the transients. Additional static calculations were performed for this study to demonstrate the sensitivity of various parameters.

3.2.1 Cycle Depletion

The All Rods Out (ARO) cycle depletion results are summarized in Figures 3.3 and 3.4. For the static analysis, the more typical operational strategy of bite depletion was chosen. Sensitivity studies reflecting different depletion models are summarized in Appendix C.

3.2.2 Hot Zero Power Static Analysis

The HZP static sensitivities (Appendix C) clearly demonstrate that the EOC cases are much more severe than the BOC HZP ejected rod cases. To provide a range of typical analyses, three cases were analyzed:

- A more realistic conservative base case
- A case which increases the ejected rod worth by [ ] [a,c] compared to the first case
- A bounding case

These HZP static results are presented in Table 3.1.

Table 3.1 HZP Static Results

a, c

[ ]

Note that with the [ ] a, c increase in the ejected rod worth, there is an associated [ ] a, c increase in the adiabatic calculated static peaking factor, F_q. The bounding case has a [ ] a, c increase in the ejected rod worth and an associated [ ] a, c increase in the peaking factor.

3.2.3 Hot Full Power Static Results

The HFP static calculations were performed at BOC and EOC with a skewed xenon distribution. To provide a more limiting HFP transient, the control rods were inserted much deeper than the HFP rod insertion limit. They were inserted to 140 steps which is more representative of a 5-cluster D-bank pattern for three loop cores. The eight-cluster pattern used in this core has both the axis and diagonal clusters of the lead bank which were evaluated to determine the worst ejected rod location. With the very conservative assumption of the deep insertion, these cases are more representative of a bounding analysis.

To summarize the conditions of the representative cases:

- BOC (150 MWd/MTU) and EOC (21,000 MWd/MTU)
- Lead control bank (Bank D) at 140 steps withdrawn, all other control rods withdrawn out of the core
- Bite depletion for the EOC case
- Equilibrium xenon distribution, then skewed to give a core average axial offset of [ ] a, c with the lead control bank deeply inserted

The results are given in Table 3.2 and show that the BOC and EOC ejected rod worth reactivities are identical for this cycle, but are in different locations.

Figure 3.3 Boron Concentration versus Cycle Burnup

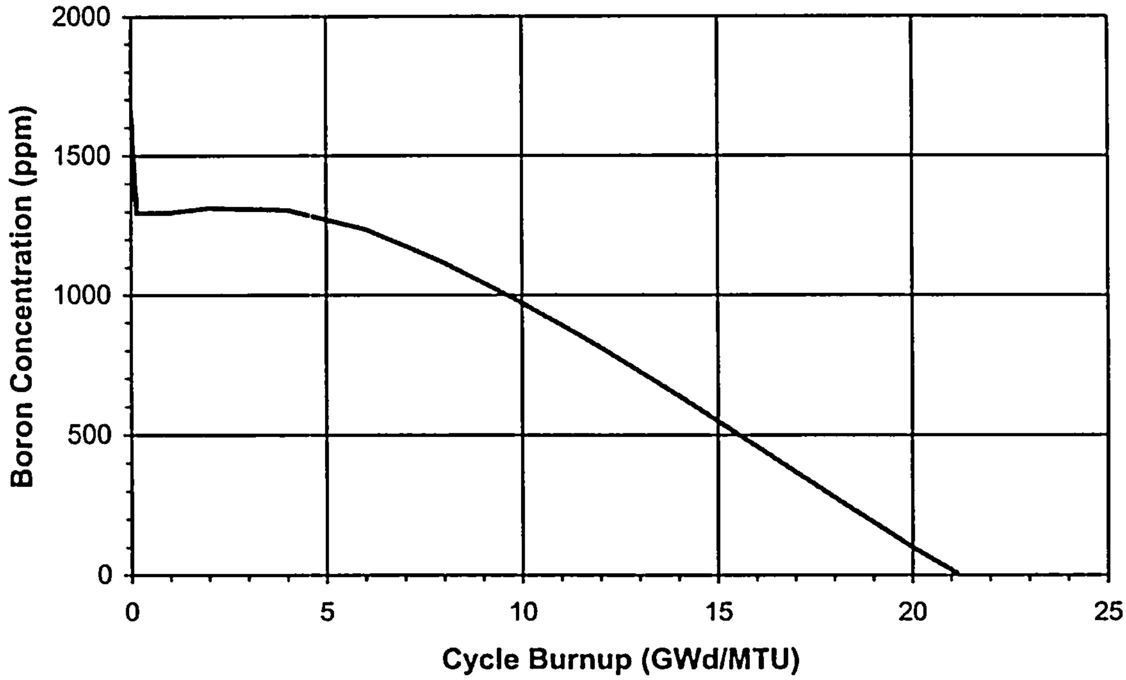


Figure 3.4 Core Peaking Factors versus Cycle Burnup

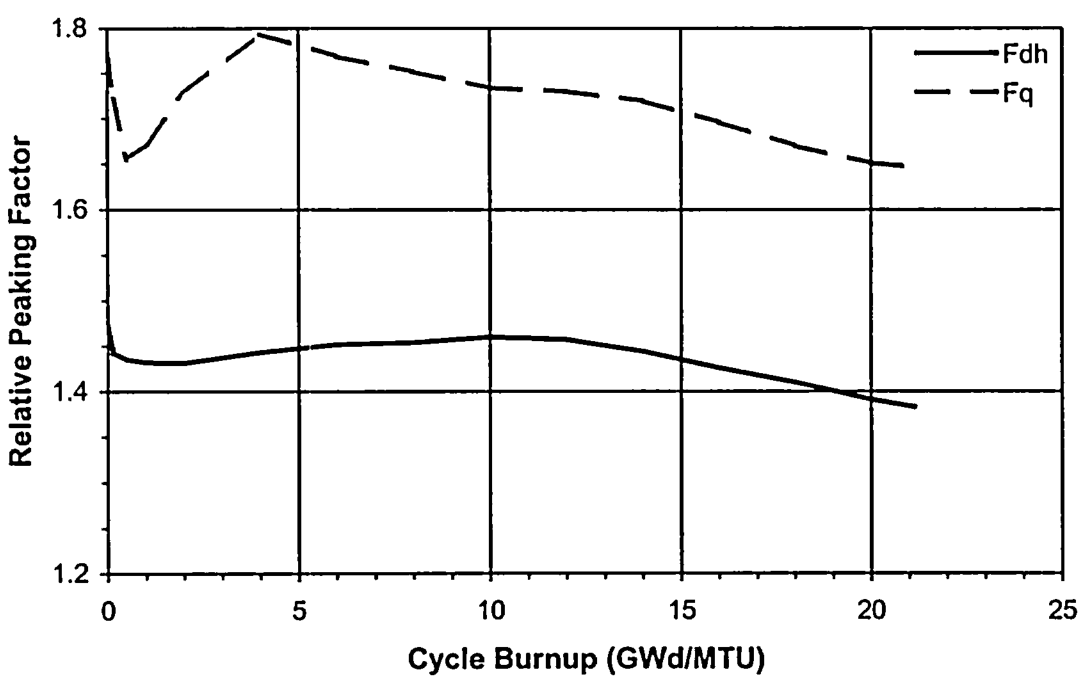


Table 3.2 HFP Static Results

a, c

[ ]

3.3 Transient Analysis

3.3.1 Hot Zero Power

The hot zero power analyses were performed at EOC. However, multiple sample conservative analyses are shown representing the following:

- Statistical reload analysis: A typical single reload cycle analysis with key parameter conservative allowances added to the results statistically. (Labeled Base Case)

- Deterministic reload analysis: A typical single reload cycle analysis with conservative allowances included for each key parameter separately. (Labeled All Allowances)

- Bounding analysis: A typical multi-cycle bounding analysis which increases the key parameters prior to including the key parameter conservative allowances to create an analysis that is expected to bound most future reload cycles. (Labeled Bounding Case)

The key parameters used in the evaluations and the results are provided in Table 3.3. The key parameter sensitivities are presented in Table 3.4. It can be seen that the application of all the uncertainties in the key parameters together increases the core average peak power by [ ] a, c and the peak fuel delta enthalpy increases by [ ] a, c compared to the conservative base case without those uncertainties. The individual perturbations, when summed, provide the same impact as the transient which included all of them together. The time transients are shown in Figures 3.5 and 3.6.

The bounding analysis is a severe case with the adiabatic ejected rod worth being slightly under $2.00 in reactivity. The Doppler multiplier has been adjusted even lower, thus the feedback effect is further pessimized. This bounding reference calculation produces results which are far more limiting. The results are summarized in Figures 3.7 and 3.8. The core average power profile is shown in Figure 3.7. The peak fuel enthalpy is presented in Figure 3.8.

a, c

Table 3.3 Results of EOC, HZP Ejected Rod Analyses with Bounding Values

Table 3.4 Results for Key Value Sensitivity Analysis

a, c

[ ]

Figure 3.5 Core Average Power Sensitivity

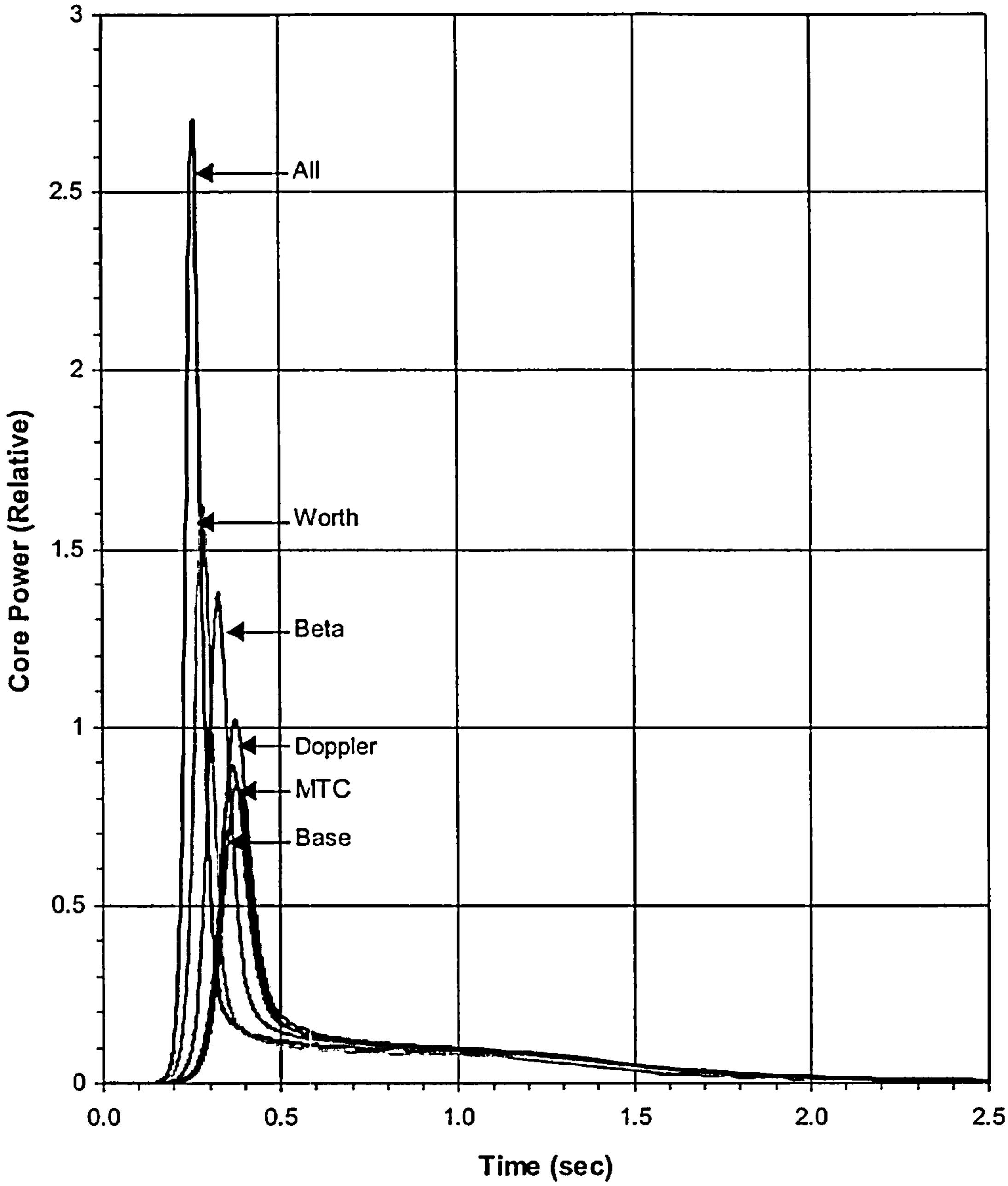


Figure 3.6 Peak Fuel Enthalpy Increase Sensitivity

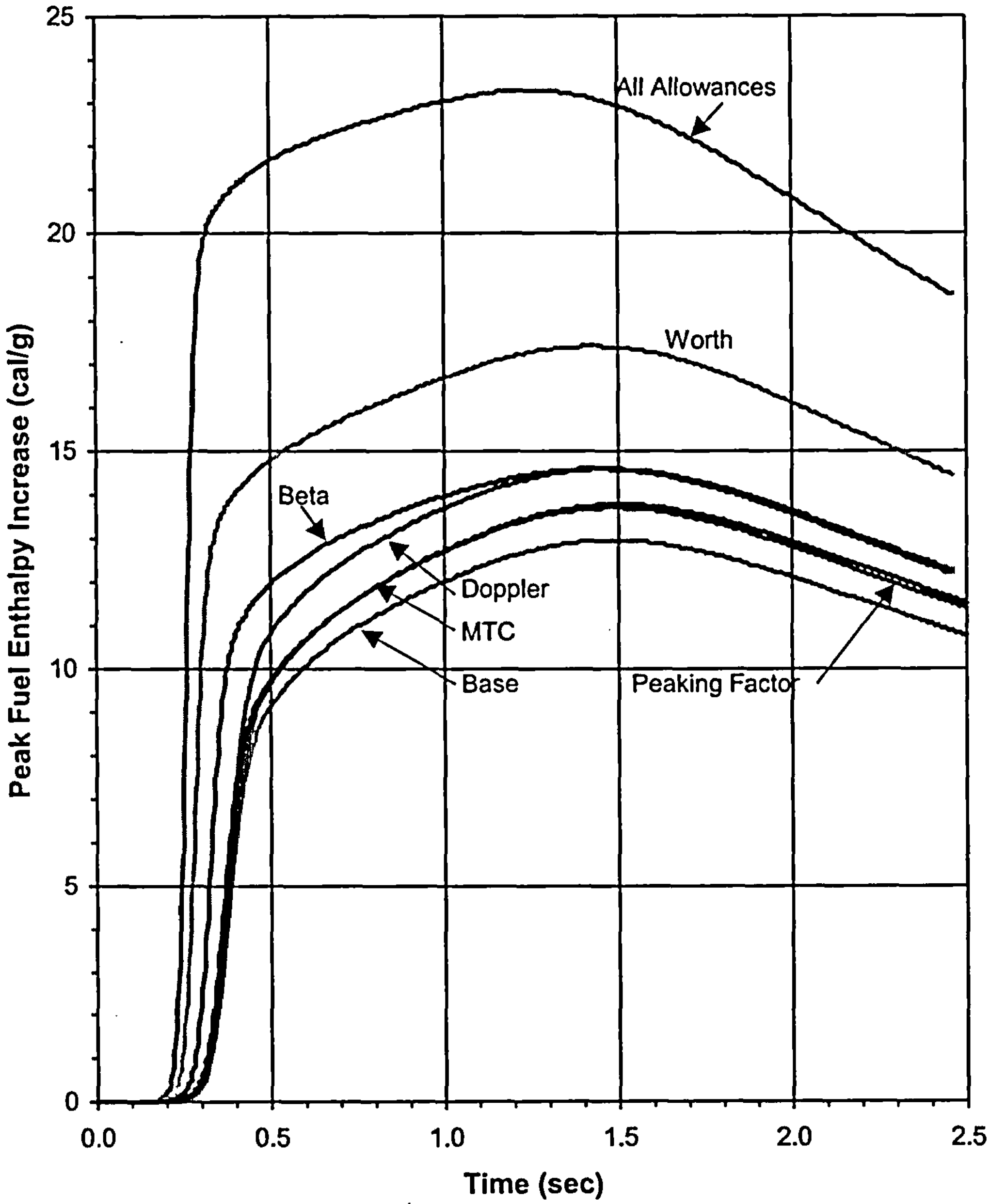


Figure 3.7 Core Average Power vs. Time for Bounding Analysis

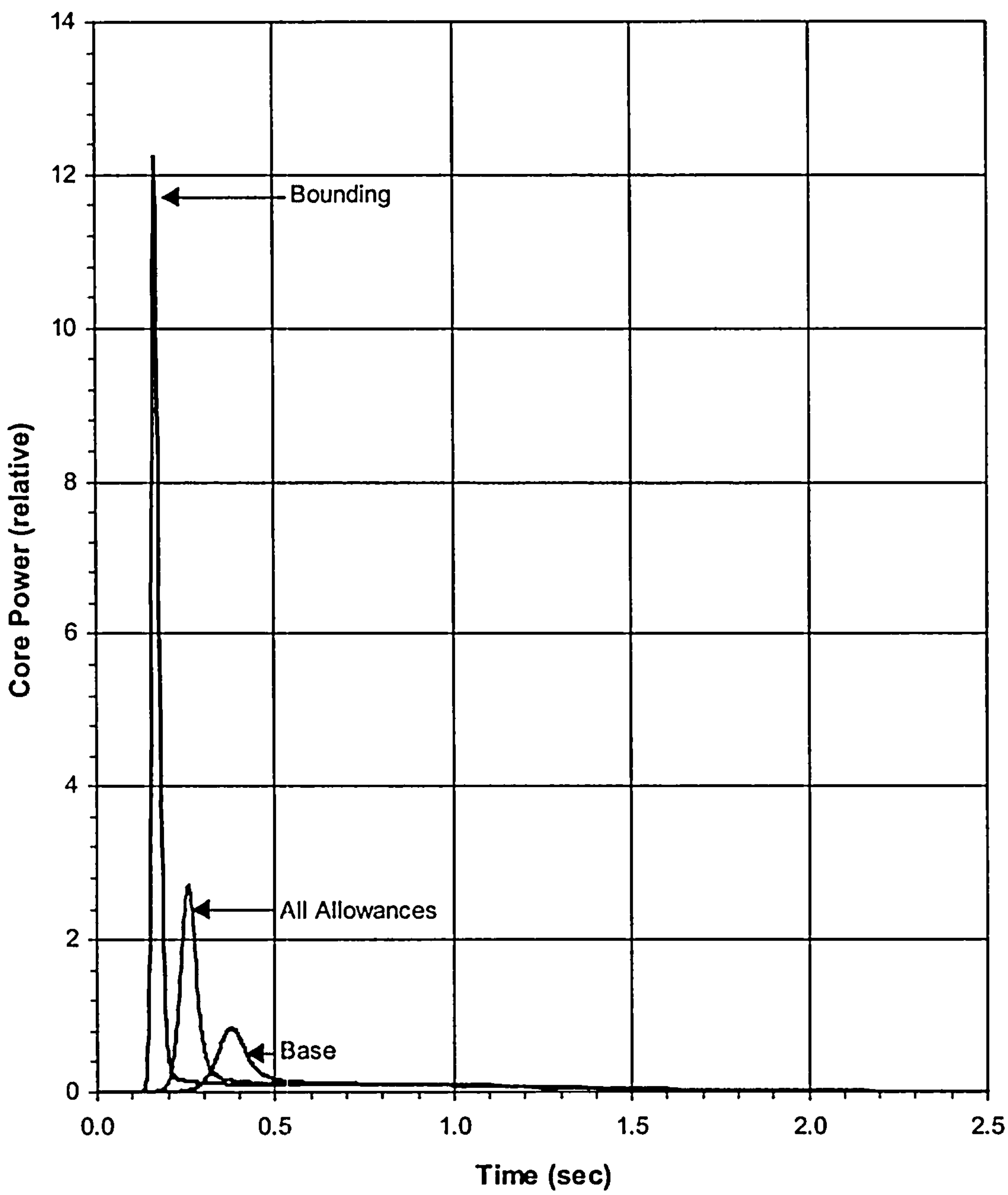
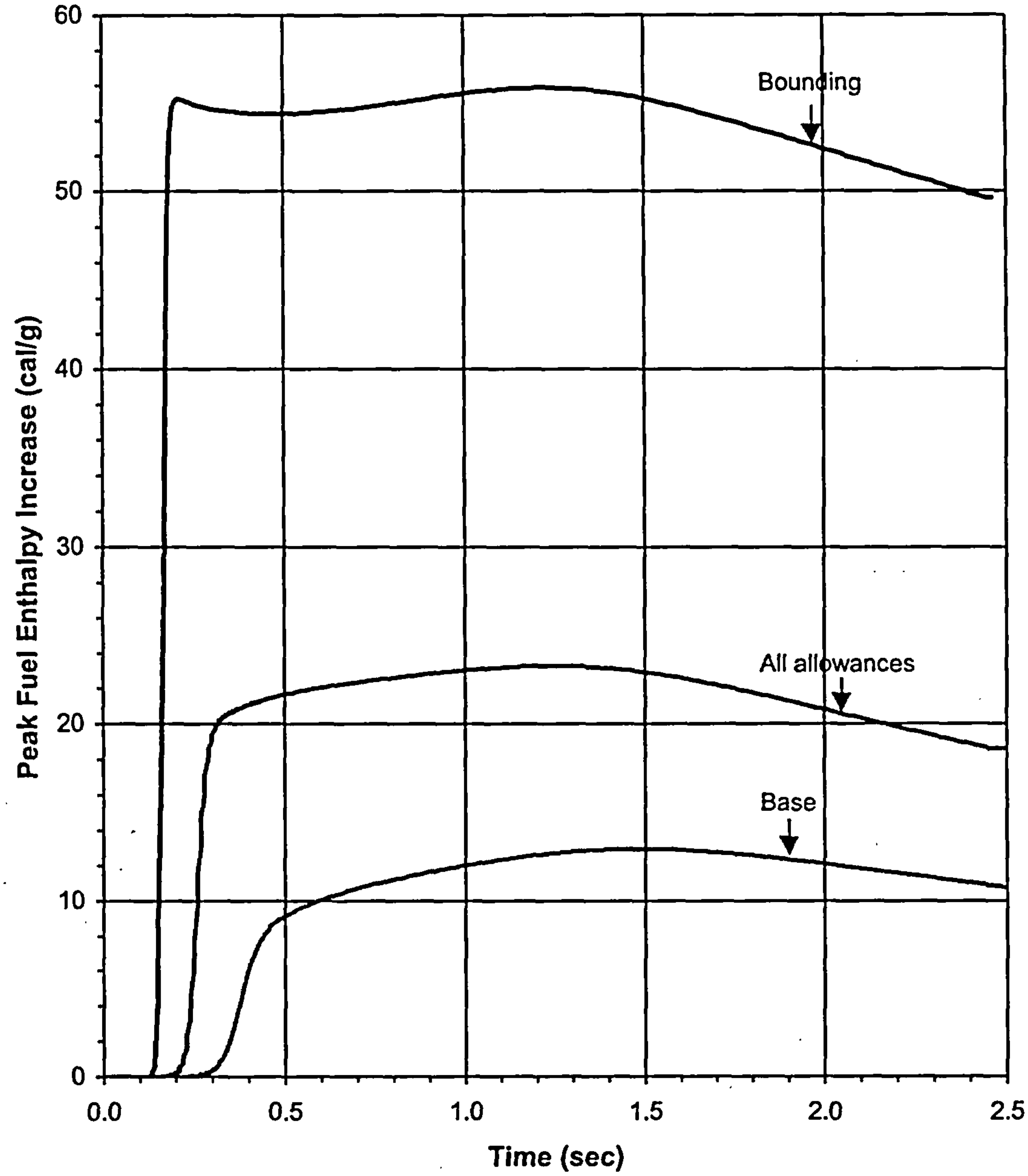


Figure 3.8 Peak Fuel Enthalpy Increase vs. Time for Bounding Analysis



3.3.2 Hot Full Power

The hot full power analysis has two cases that were analyzed – a BOC case and an EOC case. They represent nearly the same ejected rod reactivity worth, but with different ejected rod core locations and different parameters associated with the time in life. In both cases, the key parameter conservative allowances were included: the Doppler feedback and delayed neutron fraction were reduced by a conservative multiplier; the moderator temperature coefficient was made more positive by increasing the soluble boron concentration; and a xenon distribution giving a limiting positive axial offset with the control rods deeply inserted.

The cases show the characteristic rapid increase in power until the Doppler feedback balances the reactivity insertion, followed by a decrease to the new equilibrium power. The control rod trip then initiates the shutdown. The HFP transient has a different profile than the HZP transient for the following reasons:

- The ejected rod worth is much less than the delayed neutron fraction, thus the transient is not a prompt event
- The reactor is already operating at power, so in general, the fuel temperatures are already significant and pellet clad contact has occurred. Thus the heat transfer is very good between the pellet and the coolant.
- Also, since the reactor is at power, there is no delay time, as seen in the HZP cases, for the flux level to increase into the significant range.

The summary of the parameters and results is provided in Tables 3.5 and 3.6. Profiles of the core average power, peak fuel enthalpy and minimum DNBR are shown in Figures 3.9 through 3.11. The inside ejected rod has lower peaking factors and a larger delayed neutron fraction. Thus the initial power increase is smaller, but the asymptotic power level prior to the trip is higher. In neither case does the peak rod experience DNB and therefore there is no rod failure.

The BOC case has the initial peak core power and the ejected rod peak core power in the same neighborhood. The EOC case shows a shift from one region of the core to another. Thus the BOC case starts with a higher fuel enthalpy and finishes with a higher enthalpy. The DNB analysis is less dependent on the initial power distribution, and here the BOC and EOC results are very similar.

a, c

Table 3.5 HFP Rod Ejection Transient Parameters

[ ]

Table 3.6 HFP Rod Ejection Results

a, c

[ ]

Figure 3.9 HFP Ejected Rod – Core Average Power vs. Time

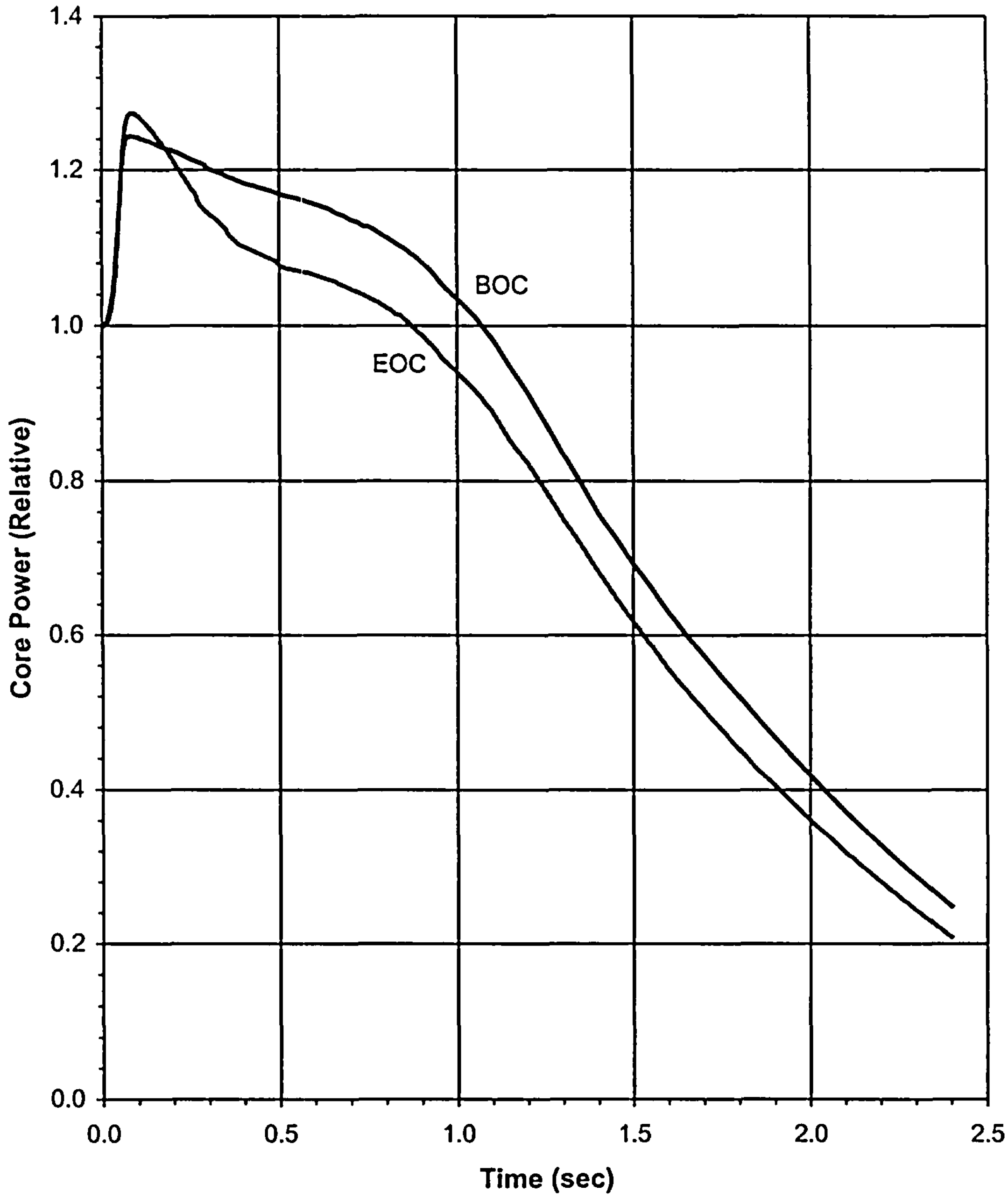


Figure 3.10 HFP Ejected Rod – Peak Fuel Enthalpy vs. Time

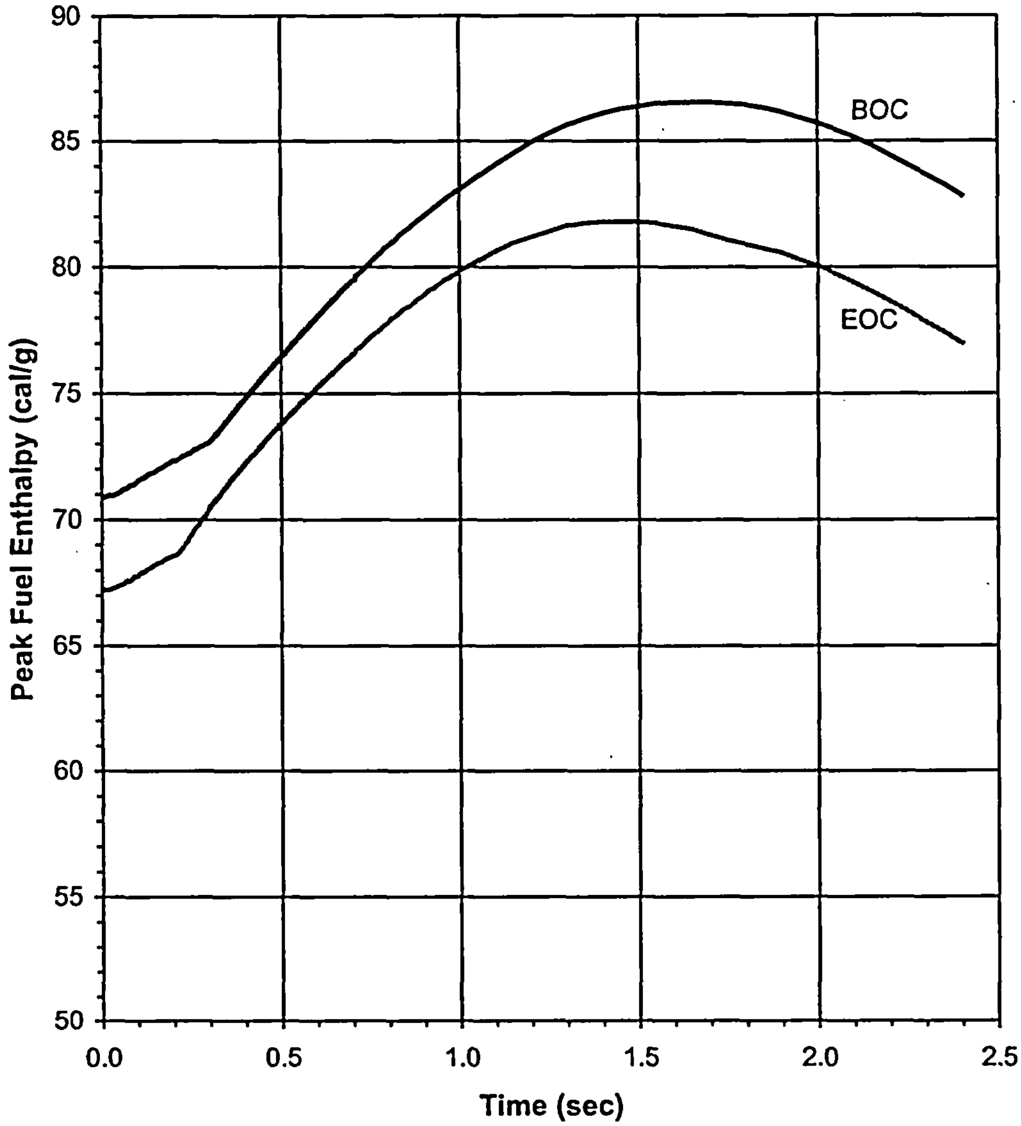
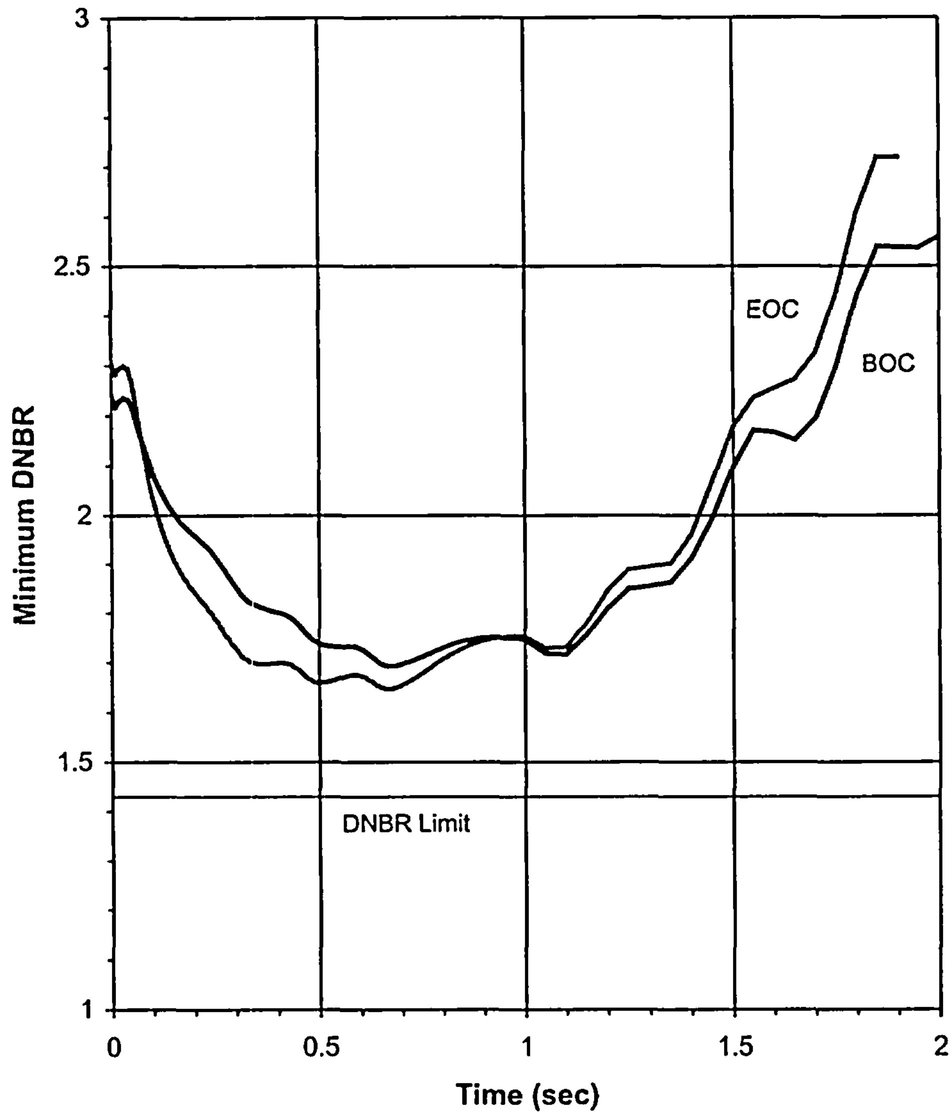


Figure 3.11 HFP Ejected Rod – Minimum DNBR vs. Time



4.0 Summary and Conclusions

This report describes the revised methodology for the analysis of the rod ejection transient in pressurized water cores using 3-D neutron kinetics in a more realistic and consistent, but still conservative, manner. The methodology utilizes the NRC-approved codes SPNOVA (Ref. 5 & 6) and VIPRE-01 (Ref. 7 & 8) that are coupled to pass the necessary data for the nuclear, fluid and fuel temperature calculations. Conservative preconditions are chosen for both the hot full power analysis and the hot zero power analysis that include time of life, rod shadowing effects, potential xenon distributions and allowable control rod positions. The cycle-dependent key parameters which have been defined are:

- Ejected rod worth
- Ejected rod peaking factor
- Delayed neutron fraction
- Doppler temperature coefficient
- Moderator temperature coefficient

Uncertainty allowances, as discussed in the report, are applied on the key parameters together, or statistically, to provide the limiting analysis values. The fuel temperature model for the core transient is based on the nuclear design reference steady state fuel temperatures. The conservative allowance for the feedback effect is included in a reduction of the feedback effect for both the Doppler temperature and the moderator temperature.

The hot rod analysis is performed using the core transient power as the forcing function with the actual 3-D peaking factors, with uncertainty allowances, as a function of time. The thermal model(s) for the hot rod are generated to provide a conservative analysis for the parameter of interest (maximum fuel enthalpy or minimum DNBR). The hot zero power maximum fuel enthalpy limits are currently being evaluated for revision by an Industry-EPRI program but are expected to be similar to the current Westinghouse limit. To cover this uncertainty when the current limits are used, a conservatism is being placed on the peak fuel enthalpy limit, reducing it to 117.5 cal/g, when used with the 3-D methodology for the HZP rod ejection. Results of the sample calculations show that using the revised methodology, the proposed interim acceptance limits were met.

The Westinghouse reload safety evaluation methodology (Ref. 17) is performed to confirm the validity of the existing safety analysis. The existing safety analysis is defined as the reference safety analysis and is intended to be valid for all cycles of the plant. Thus, safety analysis input parameter values are selected to bound the values expected in subsequent cycles. The rod ejection transient continues to be covered by the Westinghouse reload safety evaluation methodology using the methodology described in this report.

5.0 References

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Appendix A Overview of Computer Codes

A.1 Introduction

The effective 3-D analysis of the rod ejection event requires nuclear, thermal-hydraulic and fuel temperature calculations to be performed in a coupled manner for the entire core in both a steady-state environment and a very rapid transient mode. The Westinghouse methodology is utilizing previously approved computer programs: the SPNOVA computer program for the neutron kinetics, and the VIPRE-01 computer program for the thermal hydraulics and fuel temperature calculation.

A.2 SPNOVA

A.2.1 Nodal Solution

The Westinghouse standard core design methodology uses a 3-D nodal expansion method for the static analysis of the cores. This methodology is licensed and has been incorporated into the NRC-approved SPNOVA computer program (Ref. 5 & 6). The static neutronics solution in SPNOVA is also consistent with the NRC-approved ANC computer program (Ref. 11, 12, 13, 14).

A.2.2 Neutron Kinetics

The SPNOVA program includes a neutron kinetics capability. The time-dependent solution is based on the Stiffness Confinement Method which is designed to efficiently and accurately solve the time dependent equations. This method modifies the static cross sections and utilizes the same flux solution module as the static calculations. Thus, improvements to the static solution capabilities were directly utilized for the transient solution.

The applicable limitations in the Safety Evaluation Report (SER) for the use of SPNOVA for this analysis and the Westinghouse compliance are:

<u>WCAP-12983 SER Limitation</u>

The kinetics benchmarking demonstrates that SPNOVA provides an accurate method for determining both the core-wide and local power and flux response during core reactivity transients. However, in the transient application of SPNOVA the event-specific uncertainties associated with the SPNOVA methods and selected options have not been determined. In licensing applications of SPNOVA, these uncertainties are required to ensure an acceptable margin to the fuel safety limits and must be provided in event-specific submittals.

Compliance for Rod Ejection Analysis

The intent of this document is to provide the kinetics methodology for this transient including the event-specific uncertainty allowances to be used.

A.3 VIPRE-01

VIPRE-01 is a subchannel code developed from several versions of the COBRA code by the Battelle Northwest National Laboratories under the sponsorship of Electric Power Research Institute (EPRI). The subchannel analysis concept used in VIPRE is the same as in COBRA-IIIC. Conservation equations of mass, axial and lateral momentum, and energy are solved for the fluid enthalpy, axial flow rate, lateral flow and momentum pressure drop. A detailed description of the VIPRE code can be found in Reference 8.

The VIPRE heat transfer model solves the conduction equation for the temperature distribution within fuel rods and provides the heat source term for the fluid energy equation. The full boiling curve can be incorporated into the heat transfer model, from single phase convection through nucleate boiling to the Critical Heat Flux (CHF), and transition boiling to the film boiling regime.

The Westinghouse version of VIPRE-01 (Ref. 7) contains additional features as compared to the original VIPRE-01, including Westinghouse DNB correlations and heat transfer correlations consistent with the FACTRAN code (Ref. 16). For the hot fuel rod transient calculations, the following FACTRAN features have been incorporated into VIPRE-01: a) the Bishop-Sandberg-Tong heat transfer correlation for film boiling (Ref. 21), b) Baker-Just model for calculating heat generation in the cladding due to zirconium-water reaction (Ref. 22), and c) fuel enthalpy and melting predictions. However, the code additions do not alter the fundamental VIPRE-01 computational methods and functional capabilities. The modified version of VIPRE-01 is maintained in accordance with Westinghouse Quality Assurance (QA) procedures for software control.

The NRC SER on WCAP-14565 concludes that the Westinghouse VIPRE application is acceptable and that VIPRE can be used to replace THINC-IV and FACTRAN codes in the reload methodology with four conditions. The SER conditions on WCAP-14565 and Westinghouse compliance for the rod ejection analysis are provided below.

WCAP-14565 SER Condition 1

Selection of the appropriate DNB correlation, DNBR limit, engineering hot channel factors for coolant enthalpy rise and other fuel-dependent parameters for a specific plant application should be justified with each submittal.

Compliance for Rod Ejection Analysis

DNBR calculations for radiological consequence evaluation are performed with the NRC-approved VIPRE modeling assumptions described in Reference 7. Selection of a DNB correlation, DNBR limit and hot channel factors will be justified on a plant specific basis depending on fuel type. For fuel enthalpy evaluations, as described in Section 2 of this report, the hot fuel rod transient calculation is consistent with that for the post-CHF locked rotor analysis in Reference 7 and with the FACTRAN model described in WCAP-7588 (Ref. 3).

WCAP-14565 SER Condition 2

VIPRE boundary conditions from other computer codes, including core inlet coolant flow and enthalpy, core average power, power shape and nuclear peaking factors, should be justified as conservative for each use of VIPRE.

Compliance for Rod Ejection Analysis

The current design assumptions about core inlet flow rates, inlet temperature, and system pressure (Ref. 3) remain unchanged for the hot fuel rod transient calculation using the VIPRE-01 code. Time-dependent core average power, axial power shape, and nuclear peaking factors from SPNOVA/VIPRE incorporate many conservative assumptions as discussed in Section 2 of this report.

WCAP-14565 SER Condition 3

Any new correlation other than WRB-1, WRB-2 and WRB-2M will require additional justification.

Compliance for Rod Ejection Analysis

Only NRC-approved DNB correlations will be used for the rod ejection DNBR calculations.

WCAP-14565 SER Condition 4

Because VIPRE does not model the time-dependent physical changes that may occur within the fuel rods at elevated temperatures, appropriate justification should be submitted with each usage of VIPRE in the post-CHF region to ensure that conservative results are obtained.

Compliance for Rod Ejection Analysis

The VIPRE hot rod modeling retains the same conservatism as the current design method using FACTRAN for the rod ejection event. Specifically, the following conservative assumptions are made in the VIPRE calculation, in order to maximize increase in fuel enthalpy:

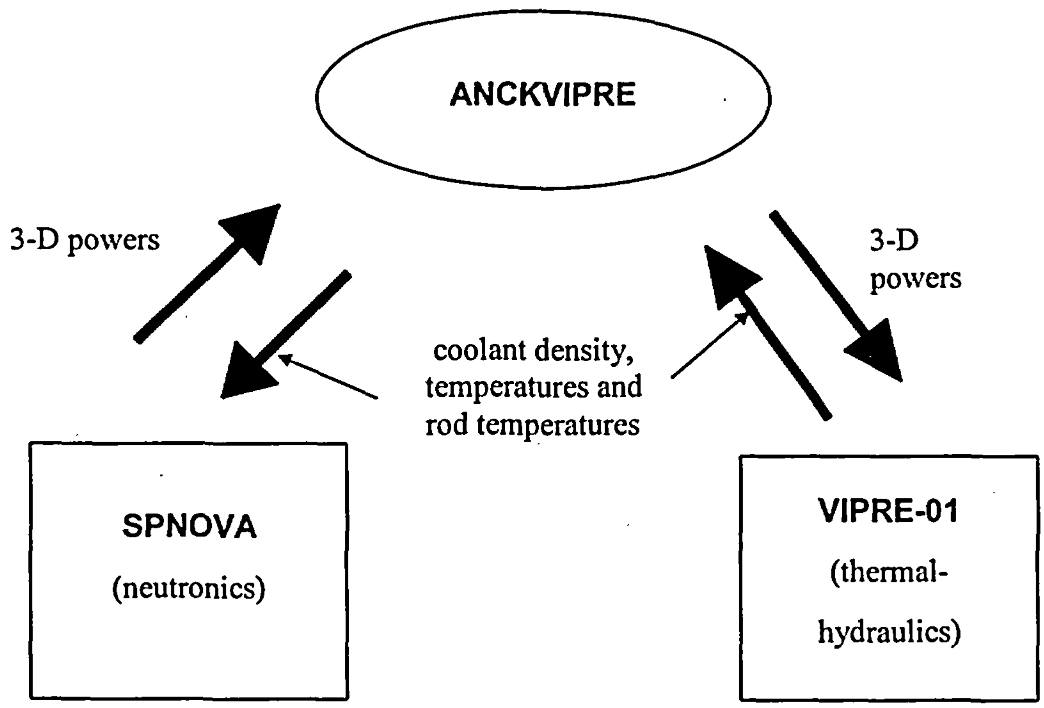
- Hot channel factors applied to rod power,
- Uncertainties in plant operating mode and parameter measurement applied to the limiting direction,
- The hot spot of the fuel rod forced into DNB and film boiling heat transfer between clad and coolant during the transient.

A.4 Computer Program Coupling

The effective 3-D analysis of the rod ejection event requires nuclear, thermal-hydraulic and fuel temperature calculations to be performed in a coupled manner for the entire core in both a steady-state environment and a very rapid transient mode. The methodology uses currently approved programs with a distributed architecture. The architecture uses a standard protocol for communication between running programs on the same or different computers to transfer data. Currently the programs utilize the Parallel Virtual Machine (Ref. 26) software for the data transfer, but this interface could be replaced with another product with no change in computational results. Thus, the actual mechanism used for the data transfer is not an inherent part of the methodology.

The only modification needed by the programs was the ability to transfer selected data into and out of the executing program. To further simplify, the data communication between the major programs is not direct; an intermediate auxiliary program (ANCKVIPRE) is utilized to coordinate the data transfer between the main programs. A schematic of the data flow is presented in Figure A.1. In addition to the data transfer, the auxiliary program also saves the hot rod information for later processing. This information is used to generate the driver functions for the hot rod analysis.

Figure A.1 Computer Program Coupling Schematic Diagram



Appendix B Qualification of Transient Analysis Method

B.1 Comparison with OECD Benchmark Problem

The Nuclear Science Committee (NSC) of the Nuclear Energy Agency (NEA)/Organization for Economic Co-Operation and Development (OECD) has released a set of computational benchmark problems for the calculation of reactivity transients in PWRs (Ref. 27). These benchmark problems verify data exchange in a coupled code system and test the neutronics coupling to fuel transient conduction methodology.

There are six PWR rod ejection benchmark problems documented in Reference 27. Of the six, Problem C1, the hot-zero-power full-core case, is most severe and challenging and therefore was chosen for the analysis with SPNOVA and VIPRE-01. The PWR core geometry is based on a Westinghouse 3-loop core, with 157 fuel assemblies. The top and bottom of the active core are covered with 30 cm thick axial reflectors. Radially the core is surrounded by one layer of reflector assemblies, which are of the same size as the fuel assemblies. The core loading pattern is a typical first core checker-board core, with three batches of fuel assemblies using burnable absorbers. The ejected rod is located near the core periphery. The reference solution of this problem was provided by Nuclear Electric using the PANTHER code (Ref. 28).

In this analysis, SPNOVA and VIPRE-01 model the core radially with one node per assembly and axially with 22 unequal meshes over the active core height and one mesh per each of the top and bottom axial reflectors. Seven equal distance regions in the fuel pin plus a gap region and a clad region are used in the fuel temperature model. As assumed in the reference solution, the Dittus-Boelter correlation was used for the clad-to-coolant heat transfer. The gap conductance was assumed at a constant value of 10 kW/m^2-°C (1762 Btu/hr-ft^2). Rod expansion and cross flow effects were not considered. A flat power distribution in the fuel rod was assumed.

Steady state SPNOVA ejected rod worth calculations were first performed using the VIPRE-01 feedback option. As shown in Table B.1, the SPNOVA rod worth predictions are in excellent agreement with the reference values. The initial power peak predicted also agrees excellently with the reference value.

Table B.1 Comparison of Rod Ejected Steady State Calculations

Solution	Core Mesh	Rod Worth (pcm)	Power Peak (frac.)
Reference	4x4x36	949.09	2.1868
SPNOVA/VIPRE	1x1x22	953.24	2.1867

A sensitivity study of time step sizes using SPNOVA and VIPRE-01 showed that for any time step less than 0.01 second, there was very little change in the solutions. Therefore, the full 5 seconds transient calculation was done with 110 time steps, using time steps of 0.005 sec up to 0.3 second, then 0.01 sec up to 0.5 second, then 0.05 sec up to 1 second, and then 0.2 sec up to 5 seconds. This SPNOVA/VIPRE-01 solution was compared to the reference solution in Table B.2. The results agree very well with the reference solution. The difference in the maximum transient core power was only −1.35%.

Table B.2 Comparison of Transient Calculations

Solution	Ejected rod worth (pcm)	Max Power in transient (frac.)	Max Power at time (sec)	Core Power at 5 sec (frac.)	Fuel Temperature at 5 sec (°F)
New Reference	949.09	4.4112	0.2712	0.1460	1245.6
SPNOVA/VIPRE	949.76	4.3516	0.2725	0.1447	1236.1

The SPNOVA/VIPRE-01 benchmark with the OECD NEACRP PWR hot zero power full core rod ejection benchmark problem shows that excellent agreement with the reference solution is obtained, even when one node per assembly and rather large time steps are used.

B.2 VIPRE Comparisons with FACTRAN

Extensive VIPRE-01 code qualifications have been performed by the code developer and users. The code qualification in Volume 4 of Reference 8 included comparisons with rod temperature measurements and heat transfer tests. The code and modeling qualification performed by Westinghouse in Reference 7 included comparisons with the FACTRAN code for hot fuel rod analysis of design-basis locked rotor transients. An additional VIPRE comparison with the FACTRAN code for a design-basis rod ejection transient initiated at the EOC HZP condition is shown in Appendix D Section D.4 using the same nuclear power vs. time transient. No adjustment of heat transfer and material properties models was performed. The results (Figure D.4) show very good agreement between the FACTRAN and VIPRE calculations.

Appendix C Rod Ejection Sensitivity Studies

C.1 Static Sensitivity Studies

C.1.1 Cycle Depletion Static Sensitivity

There were three different depletion models developed to evaluate the sensitivity of the rod positions with depletion:

- Depletion with rods fully withdrawn for the entire cycle. (This will be referred to as ARO depletion.)
- Depletion with the lead control bank inserted at a bite position. (This will be referred to as bite depletion.)
- Depletion with the lead control bank deeply inserted to account for a significant amount of at-power operation with rods deeply inserted. Because of the limitation of load following near EOC, the deep rod insertion is performed only while the boron concentration is above 200 ppm. After that point in life, the lead control bank is moved to the bite position. (This depletion will be referred to as rodded depletion.)

The sensitivity of the rodded depletion options at EOC is given in Table C.1.

Table C.1 Impact of Rodded Depletion at EOC a, c

[ ]

As can be seen, the impact of rodded depletion is very significant. Thus, the anticipated operational strategy is an important factor in the basis for the transient evaluation.

C.1.2 Control Rod Insertion Static Sensitivity

The control rods can be inserted to their insertion limits, which are a function of core power level. At HZP, the lead bank is typically fully inserted and the following control bank(s) are partially inserted as determined by their overlap with the lead bank. The worst ejected rod cluster is typically from the lead bank, and thus rod insertion has minimal impact. However, at HFP, the lead control bank is typically

inserted only a faction of the total insertion. Thus deeper insertion could be used to conservatively increase the ejected rod worth. The following table illustrates the sensitivity to lead bank insertion at EOC, HFP. To keep the axial power changes separate, all cases have the xenon adjusted to keep the pre-ejection axial offset at the same value, and to utilize a rodded depletion history.

Table C.2 Impact of Lead Bank Insertion

a, c

Table C.2 demonstrates that the insertion of the lead bank can effectively be used to increase the ejected rod worth and peaking factors for a more conservative analysis.

C.1.3 Time in Cycle Static Sensitivity Study

Static sensitivity calculations were performed to demonstrate the impact of time of life on the ejected rod static parameters. The hot full power ejected rod worth was checked at multiple times in life using a xenon distribution to maintain the same axial offset (except for the zero burnup case which had no xenon) and a deep rod insertion. An ARO depletion model was used. Also, the beginning of cycle and end of cycle HZP rod ejection cases were performed to show the impact of time of life. The results are summarized in Table C.3.

a, c

Table C.3 Impact of Time in Cycle

These results demonstrate that the end of cycle condition is far more limiting for the HZP cases. However, it points out the relative lack of insensitivity with time of life for the hot full power case condition.

C.1.4 Axial Power Distribution Static Sensitivity Study

Static sensitivity calculations were performed with various initial power distributions which were created by modifying the core xenon distribution. The xenon distributions were generated through various mechanisms, and were used to skew the radial and axial power distributions. An increase in the relative power in the upper portion of the assembly with the ejected cluster can have a significant increase in the ejected rod worth and the resultant peaking factor.

Mechanisms which were used to generate various axial xenon distributions are:

- [ ]a,c
- [ ]a,c
- [ ]a,c
- [ ]a,c

It should be noted that many of these xenon distributions are extremely unlikely, even impossible, in an operational critical core, especially at end of cycle, but are used to demonstrate the sensitivity over a broad range. The results also demonstrate that a xenon distribution can be used to create a specified conservative allowance on the ejected rod worth.

a, c

Table C.4 Impact of Axial Power Distribution at HZP

[ ]

Table C.4 illustrates that the sensitivity to the axial power distribution for reasonable to moderately severe xenon distributions is less than a [ ] [a, c] increase in the ejected rod worth. Also, the impact on the peaking factor, F_q, is more than twice the impact of the ejected rod worth.

The hot full power variability with the axial power shape was also evaluated. The deep insertion of the control bank drives the power to the bottom of the core. The xenon can then be skewed towards the bottom to drive the power back towards the top of the core. Table C.5 shows the very significant impact on the ejected rod parameters for the HFP ejected rod.

Table C.5 Impact of Axial Power Distribution at EOC, HFP

[ ] a, c

Note, that the last row (AO = -39) is using [ ] [a, c].

C.1.5 Radial Power Distribution Static Sensitivity Study at HZP

The importance of the ejected cluster position can be enhanced by increasing the relative local reactivity by decreasing the xenon concentration in the lead control bank locations. The equilibrium HFP xenon distribution can be determined with the lead control bank inserted to the HFP insertion limit. The low power in the rodded locations, when used with an equilibrium xenon condition, reduces the xenon in those locations relative to the all rods out xenon distribution. It also pushes more xenon to the bottom of the core, which increases the reactivity at the top of the core. These effects increase the reactivity in those rodded locations, thus increasing the ejected rod worth.

Table C.6 Impact of Radial Power Distribution at HZP

[ ] a, c

The results given in Table C.6 demonstrate that the sensitivity in the radial power due to the xenon distribution [ ]^{a,c}.

C.1.6 Core Radial Tilt Sensitivity Study

Many reactor cores have experienced small radial power tilts. To demonstrate the impact of quadrant power tilts, the model was redepleted using a full core model and assuming a slight temperature imbalance [ ]^{a,c} between the loops. This small temperature imbalance will create a reactivity difference in the reactor, which will in turn creates a burnup asymmetry to balance the reactivity difference. The EOC case was then used with no xenon to evaluate the impact of a quadrant power tilt on the ejected rod worth. Because of the loop inlet to core quadrant lack of alignment, two different loop asymmetries were tried. The results are summarized in Table C.7 below:

Table C.7 Impact of Radial Power Distribution Tilt at EOC, HZP

a, c

[ ]

The range of variability in the ejected rod worths was [ ]^{a,c} when compared to nominal case with no quadrant tilt. Thus, it can be seen that this impact is small, and is adequately covered by [ ]^{a,c}.

C.2 Core Transient Sensitivity Studies

C.2.1 HZP Transient Sensitivity Study

The key parameters were adjusted to demonstrate the sensitivities for the HZP rod ejection transient. The results from this sensitivity analysis were presented in Section 3.3 as part of the HZP transient analysis, and hence are not repeated here. For consistency, all of the transients assumed the same trip time. The results demonstrated the sensitivity to the key factors:

- Ejected rod worth
- Delayed neutron fraction
- Doppler temperature feedback

Of key importance is the ejected rod worth as measured in dollars of reactivity. Comparison of the key calculational results to this parameter is presented in Figure C.1. The inverse of the pulse width and the peak fuel rod enthalpy increase are both plotted versus the ejected rod worth. Both the peak enthalpy increase and the inverse of the pulse width demonstrate an approximate [ ] a, c relationship to the ejected rod (assuming the other parameters are held constant). This relationship provides an excellent approximation for the impact of increasing the ejected rod worth.

Figure C.1 Peak Fuel Enthalpy or Inverse Pulse Width vs. Ejected Rod Worth

[ ] a, c

The sensitivity studies also demonstrated a minimal impact of the moderator temperature coefficient. A review of the peak fuel enthalpy curves versus time also illustrates the impact of the trip timing. Associated with the power excursion is the rapid increase in the peak fuel enthalpy. After the prompt jump and the return to the asymptotic power level, the enthalpy rise is a very gradual increase until the trip turns around the curve (See Figures 3.6 and 3.8). This gradual increase was in the vicinity of [ ] [a, c], and the slope is basically independent of the size of the power excursion. Thus, small changes to the time of the trip and speed of the trip will have minimal impact on the peak fuel enthalpy.

C.2.2 Transient Time Step Size Sensitivity

An important aspect of the analysis of the ejected rod transient is the time step size that is used. The extremely short prompt neutron lifetime makes the neutron kinetics equations difficult to solve with large step increases in reactivity. The stiffness confinement method that is incorporated in the analysis makes larger time steps feasible. As a demonstration of the accuracy, the same transient was analyzed with different time step sizes. The time steps are variable through the transient, with the smallest time steps during the pulse. The transient pulse half width was approximately [ ] [a, c]. The time steps used are summarized in Table C.8. The results of the sensitivity analysis demonstrate the robustness of the method. The peak core average power is more sensitive than the peak fuel enthalpy, but even that impact is small. Cases C and D demonstrate the impact of a factor of four increase in the time step size between 0.0 and 0.1 seconds. During this time interval the rod is being ejected and the rapid increase in the core flux is beginning. But there is no significant feedback occurring yet, and the relatively coarse time step mesh has a very small impact on the net transient.

The peak fuel enthalpy is an integral value, and it was fairly insensitive to the time step size chosen, thus demonstrating the suitability of larger time step lengths. Only when the minimum time step was increased to a value of 10 ms was there any sensitivity. And for that case, the predicted enthalpy was high, the conservative direction.

Table C.8 Time Step Size Sensitivity

Parameter	Case A	Case B	Case C	Case D	Case E
Time Interval (sec)	Step Size (ms)	Step Size (ms)	Step Size (ms)	Step Size (ms)	Step Size (ms)
0.0 to 0.1	50.0	20.0	10.0	2.5	5.00
0.1 to 0.4	10.0	5.0	2.5	2.5	1.25
0.4 to 1.0	50.0	25.0	12.5	12.5	6.25
1.0 to 2.5	200.0	100.0	50.0	50.0	25.00

a, c

C.3 Hot Rod Transient Sensitivity Study

Sensitivity studies were performed to further justify modeling options of the hot fuel rod for the rod ejection transient under the HZP and HFP conditions. The sensitivity study of fuel enthalpy with respect to changes in channel modeling is summarized below. The initial core conditions and fuel parameters are provided in Table C.9.

VIPRE hot fuel rod transient calculations were performed using the single channel model and a multi-channel model for an eighth core described in Reference 7 (Figure C.2). A comparison of fuel enthalpy changes during the transient is shown in Figures C.3 and C.4 for the HZP and HFP case, respectively. A comparison of the peak fuel enthalpy is summarized in Table C.10 for both cases.

The comparisons show that there are no significant changes in the fuel enthalpy obtained from the single and multi-channel models. Because of the rapid increase in neutron power, and fuel rod being forced into DNB, the fuel rod heat transfer between clad and coolant is nearly adiabatic, particularly at the HZP condition. The conditions in the surrounding channels or the rest of the core have very little effect on the temperature change within the hot rod.

Table C.9 Initial Core Conditions and Parameters for Hot Rod Sensitivity Study

Parameter	Value
Fuel Rod OD	0.360 inch
Pellet OD	0.3088 inch
Clad Thickness	0.0225 inch
Heated Length	143.7 inch
Channel Flow Area	0.1442 inch^2
Core Inlet Flow (Excluding Bypass)	8.07 ft/s (HZP) 12.80 ft/s (HFP)
Core Inlet Temperature	557.0 °F (HZP) 558.2 °F (HFP)
Core Exit Pressure	2200. psia
Core Power	0.00001*2900 MWt (HZP) 1.02*2900 MWt (HFP)
Initial $F_{\Delta H}$	1.928 (HZP) 1.620 (HFP)

Table C.10 Peak Enthalpy Comparison for Single- and Multi-Channel Models

a, c

[ ]

Figure C.2 VIPRE Multi-Channel Model for 1/8th Core

a, c

[ ]

Figure C.3 HZP Fuel Enthalpy Comparison for Single- and Multi-Channel Models

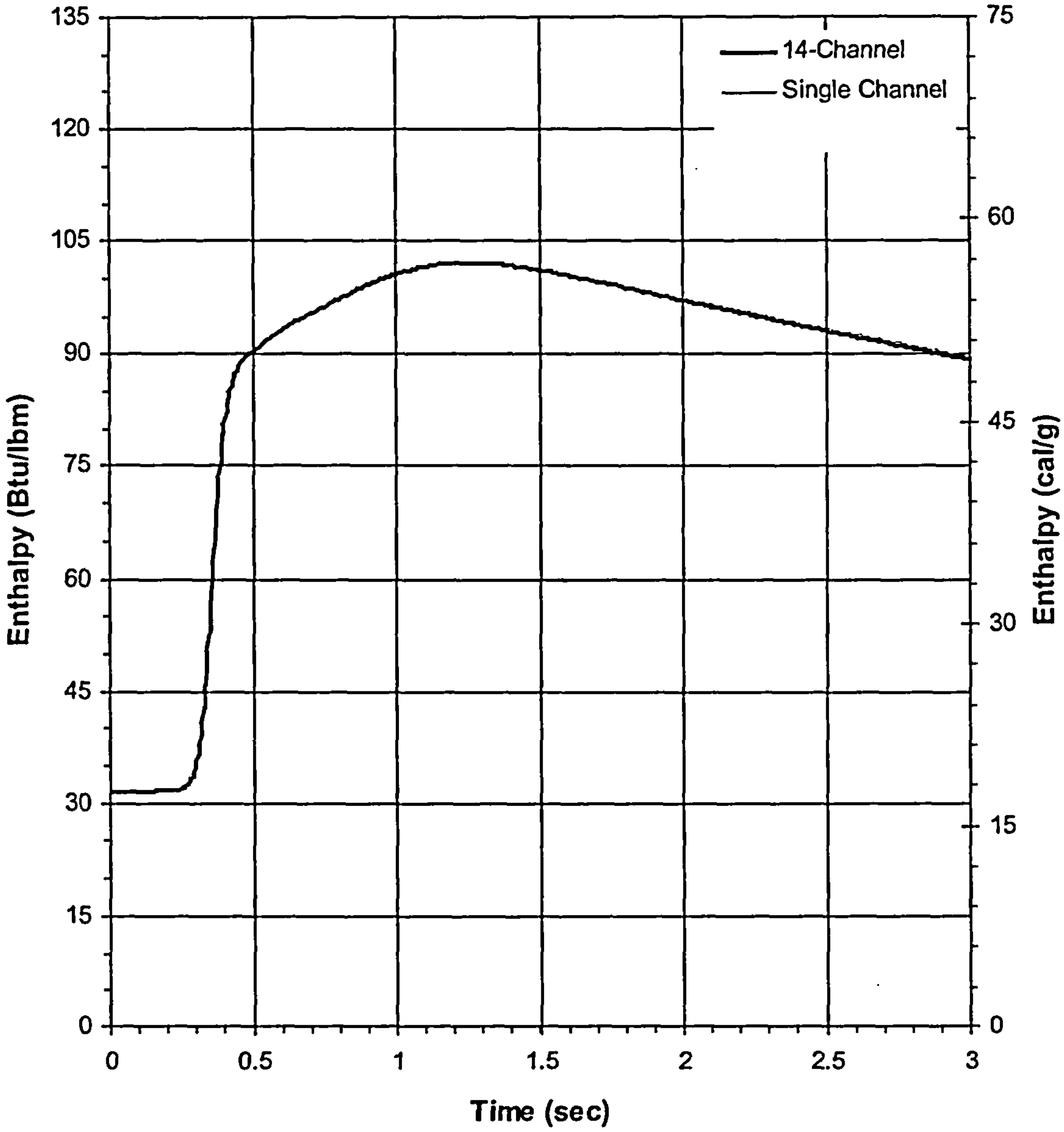
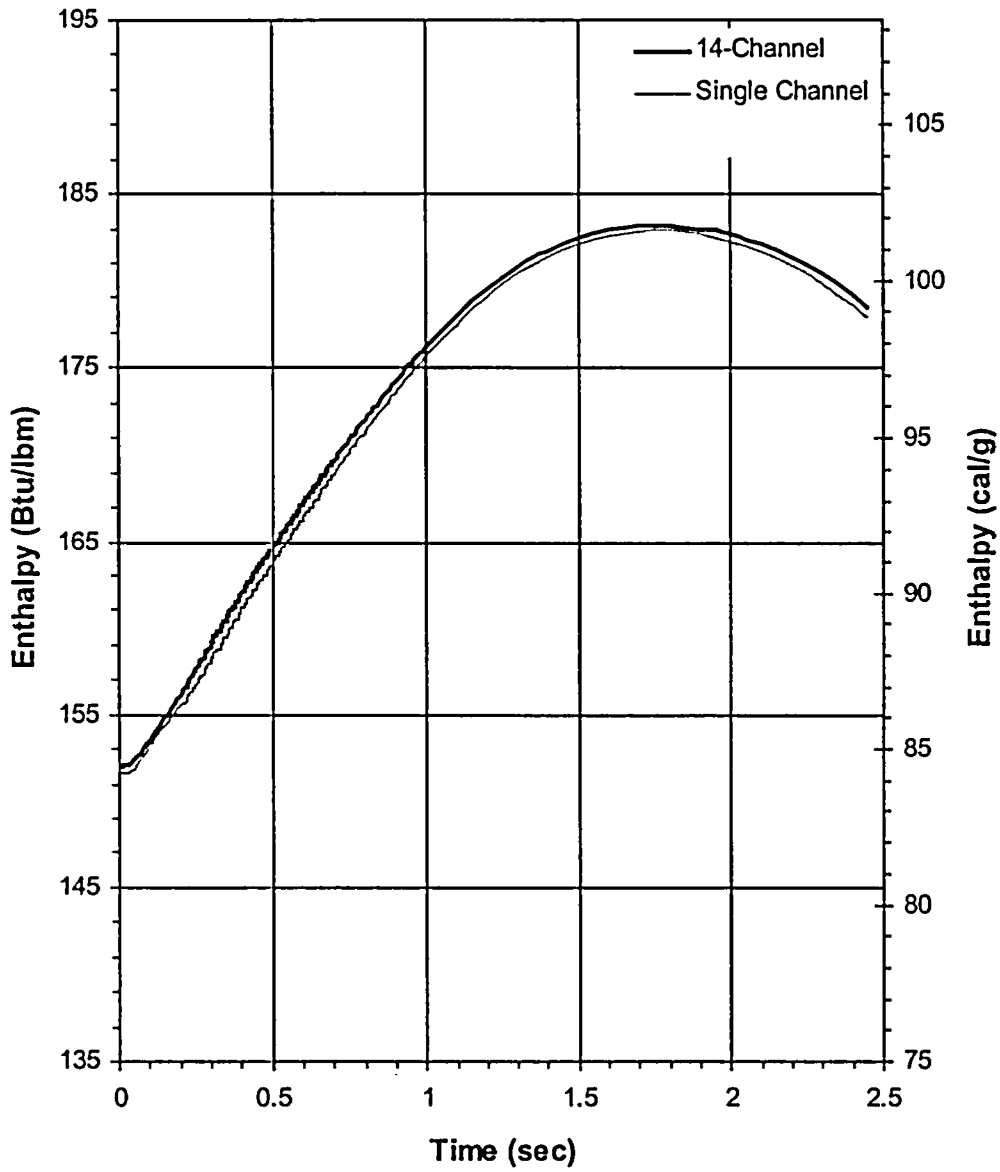


Figure C.4 HFP Fuel Enthalpy Comparison for Single and Multi-Channel Models



Appendix D Comparison of 3-D and 1-D Analysis Method

D.1 Introduction

In order to illustrate the difference between the new 3-D rod ejection analysis method and the current 1-D licensing-basis method, the 3-D EOC HZP "bounding" case presented in Section 3 was rerun using the 1-D licensing basis method and the results were compared. The 1-D method uses the TWINKLE neutron kinetics code (Ref. 15) in the 1-D axial mode to calculate the core average nuclear power vs. time, and the FACTRAN heat transfer code (Ref. 16) to calculate the hot spot fuel and clad temperatures vs. time. The parameters in the 1-D case were adjusted to yield the same ejected rod worth, peaking factor, delayed neutron fraction, and Doppler defect as the "bounding case" values shown in Table D.1. The 1-D licensing-basis method uses a conservative Doppler weighting factor in TWINKLE to partially account for the increased Doppler feedback in the ejected rod configuration. The moderator coefficient was adjusted in the 1-D case to match the 3-D calculation as close as possible. The 1-D method uses an assumed trip reactivity of 2% Δk, which is smaller than occurs by tripping in the control and shutdown banks in the 3-D method. The results are presented below.

D.2 Nuclear Power vs. Time

The nuclear power vs. time results for the 3-D and 1-D cases are shown in Figure D.1. The same fast and thermal neutron velocities were used for each case. The reactor reached a prompt-critical condition due to the rod ejection, resulting in a rapid increase in nuclear power. The nuclear power reached a peak value in the 3-D and 1-D cases of 12.23 and 18.33 times full power, both reaching their peak values at about 0.165 seconds. At the time of the peak, the feedback has increased until the core excess reactivity is essentially equal to the total delayed neutron fraction, and the core rapidly becomes sub-prompt-critical beyond this point. This results in a rapid fall-off in the nuclear power. The significantly lower nuclear power peak in the 3-D case is attributed to the stronger Doppler feedback compared to the 1-D case. The 3-D vs. 1-D pulse widths (at half maximum) are similar at about 21 vs. 23 msec. The stronger Doppler in the 3-D case also results in a lower nuclear power vs. time following the peak. The reactor trip setpoint is reached at about 0.135 seconds in both cases, with the start of rod motion 0.5 seconds later. Starting at about 1.2 seconds, the negative reactivity due to the insertion of the control and shutdown bank rods after the trip becomes significant and results in a further lowering of the nuclear power transient.

In both cases, the control rod was assumed to be ejected in 0.1 second. The difference in the shape of the nuclear power vs. time transient between 0 and just over 0.1 seconds is explained by the method used to simulate the ejection. In the 3-D case, the rod ejection was simulated by the rapid, constant acceleration of the control rod until it was fully withdrawn in about 0.1 second. With the top-peaked initial axial power distribution, this resulted in a more gradual initial increase in reactivity compared to the 1-D case in which a linear addition of reactivity in 0.1s was assumed. However, the initial power level assumption, and the rate of addition of reactivity, has no effect on the magnitude of the peak nuclear power or fuel enthalpy, provided the reactivity is added before the core reaches the power generation range.

D.3 Power Peaking Factor vs. Time

The hot spot peaking factor vs. time in the 3-D and 1-D cases is shown in Figure D.2. In the 3-D case, the hot spot peaking factor reaches its peak value just after the ejection and before the peak transient nuclear power. Since there is only a very small amount of feedback at this time, and since in a prompt-critical transient the initial delayed neutron spatial distribution is not important, the peaking factor reaches a peak value which is only slightly below the value predicted in the static 3-D ejected rod calculation. (In the sub-prompt-critical full-power case, the peak transient F_q never reaches the statically-calculated value due to the much flatter delayed neutron distribution.) Once the reactor power level reaches the power generation range, the increased Doppler feedback causes a rapid reduction in the peaking factor. Because the peak nuclear power reaches many times nominal in this transient, particularly around the ejected rod location, the direct moderator heating effect contributes to the negative reactivity feedback at the time of the peak due to the rapid (although small) increase in the local moderator temperature. Later in the transient, the moderator feedback due to heat transfer becomes more important in reducing the nuclear power transient and local power peaking. The reactor trip starts to insert a significant amount of negative reactivity at about 1.2 seconds. The peaking factor at first reduces due to the flattening of the axial power distribution, and then later, with deeper rod insertion, tends to increase as the flux returns to the top of the core. The radial peaking factor remains large due to the assumption of a stuck rod in addition to the ejected rod. However, at his point the nuclear power level is low enough that there is no significant energy addition to the core.

In a 1-D model, there is insufficient information available from the transient nuclear calculation to determine the behavior of the hot spot peaking factor vs. time. Therefore, in the current 1-D licensing basis method, the peaking factor is conservatively assumed to increase from its initial value to the statically-calculated peak value in 0.1 second, which is the time the rod is assumed to be ejected. The peaking factor is then held constant for the remainder of the transient as demonstrated in Figure D.2.

D.4 Hot Spot Fuel Enthalpy vs. Time

The hot spot fuel enthalpy vs. time results are shown in Figure D.3. The 3-D method results show an initial peak fuel enthalpy of 72.7 cal/g at about 0.21 seconds, reaching a maximum value of 73.4 (net enthalpy increase of 55.9 cal/g) at 1.23 seconds. The enthalpy then decreases throughout the rest of the transient. In the 1-D method case, the enthalpy rises to a much higher initial peak value of 133.4 cal/g at 0.20 seconds, followed by a slow enthalpy rise to a maximum 149.5 cal/g (net enthalpy increase of 131.9 cal/g) at about 1.83 seconds after the start of the transient. The higher fuel enthalpy in the 1-D case is partly due to the more severe nuclear power transient; however, the major effect is due to the very conservative use of a constant ejected rod peaking factor vs. time in the 1-D method case as discussed in Section D.3 above.

To show that the difference is due to the very conservative nuclear power and peaking factor transient used in the 1-D case, the 3-D nuclear power transient from Figure D.1 was multiplied by the 3-D peaking factor transient from Figure D.2 and used in place of the TWINKLE results as the input to FACTRAN for the hot spot fuel transient calculation. The resulting fuel enthalpy vs. time is shown in Figure D.4,

compared to the 3-D method results. The FACTRAN results are in very good agreement with the VIPRE results. This demonstrates that the differences in the fuel enthalpy behavior are due almost entirely to the differences between the 1-D and 3-D nuclear power and peaking factor transients, and not due to any differences in the heat transfer models.

D.5 Conclusions

The 3-D analysis method results for a representative bounding case at end of cycle from hot zero power was compared to the same calculation using the current 1-D licensing-basis method. The results show that the 3-D method approach results in a significant gain in peak fuel enthalpy margin. The 3-D margin gain is partly due to the reduction in the nuclear power transient as a result of the increased feedback, and also due to the use of the actual power peaking factor transient vs. time in the hot rod transient calculation vs. the overly conservative assumption of a constant ejected rod peaking factor in the 1-D method case.

Table D.1 Key Parameters for EOC HZP 3-D vs. 1-D Method Comparison

a, c

Figure D.1 3-D and 1-D Nuclear Power vs. Time

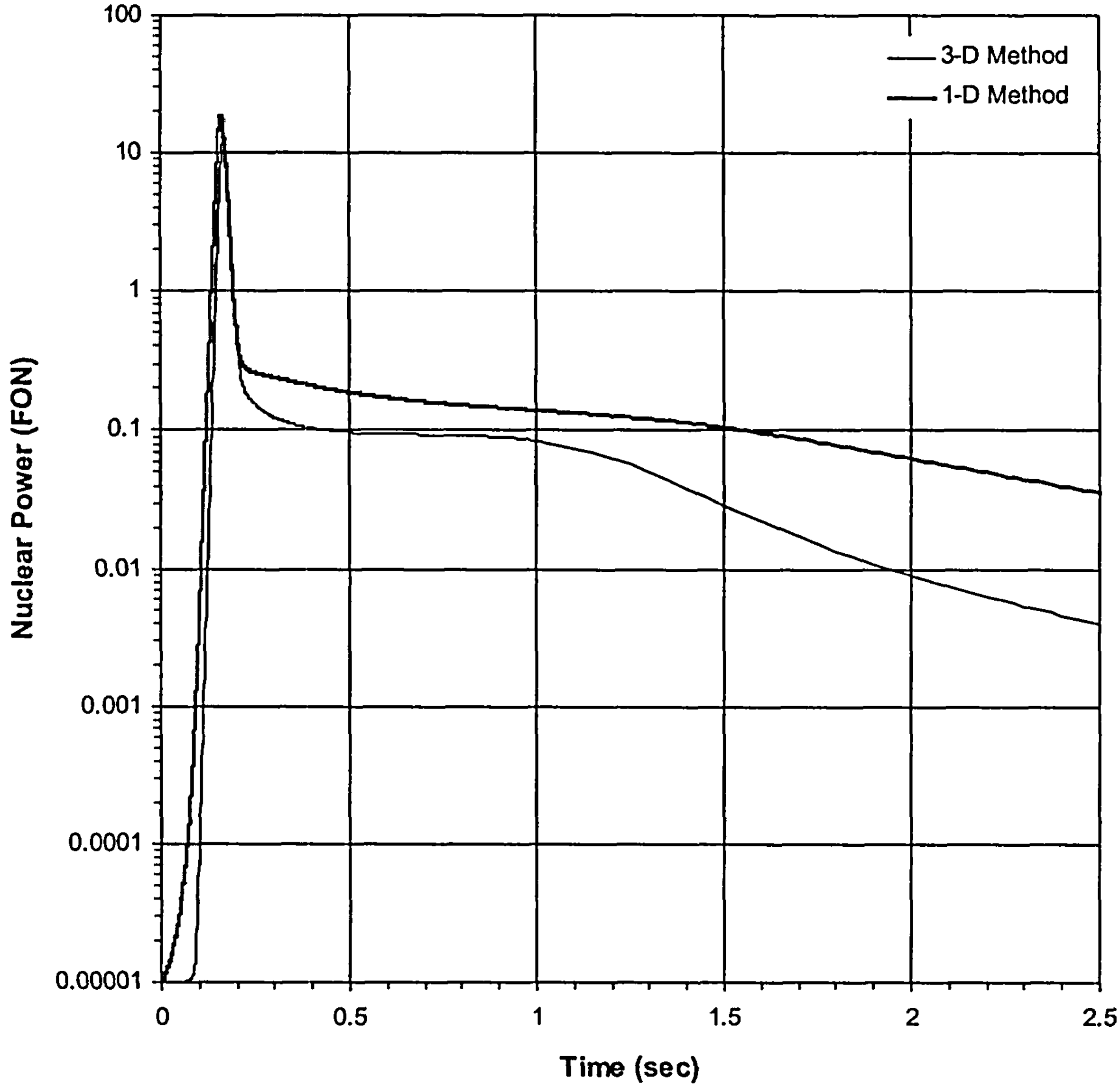


Figure D.2 3-D and 1-D F_q vs. Time

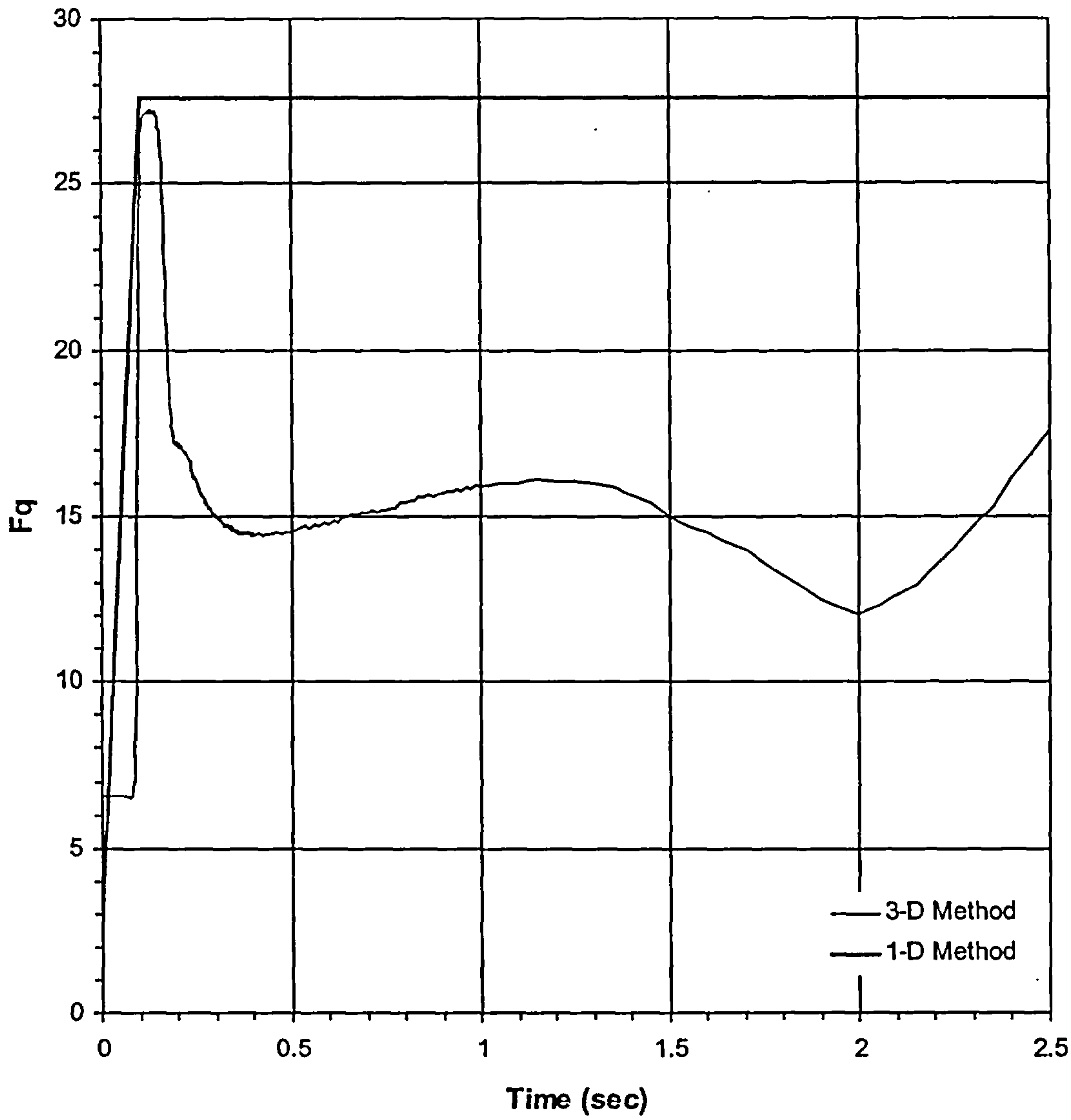


Figure D.3 3-D and 1-D Fuel Enthalpy vs. Time

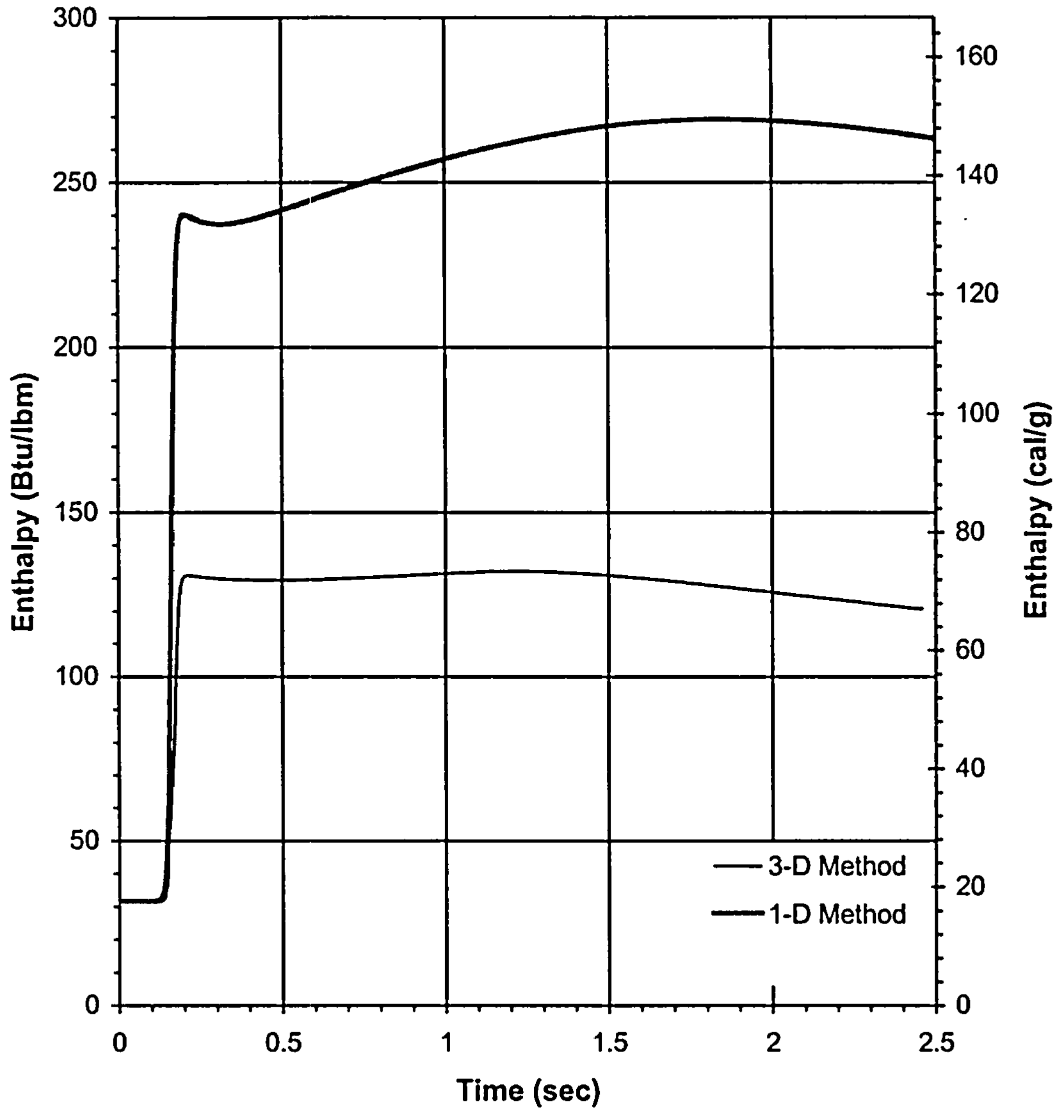
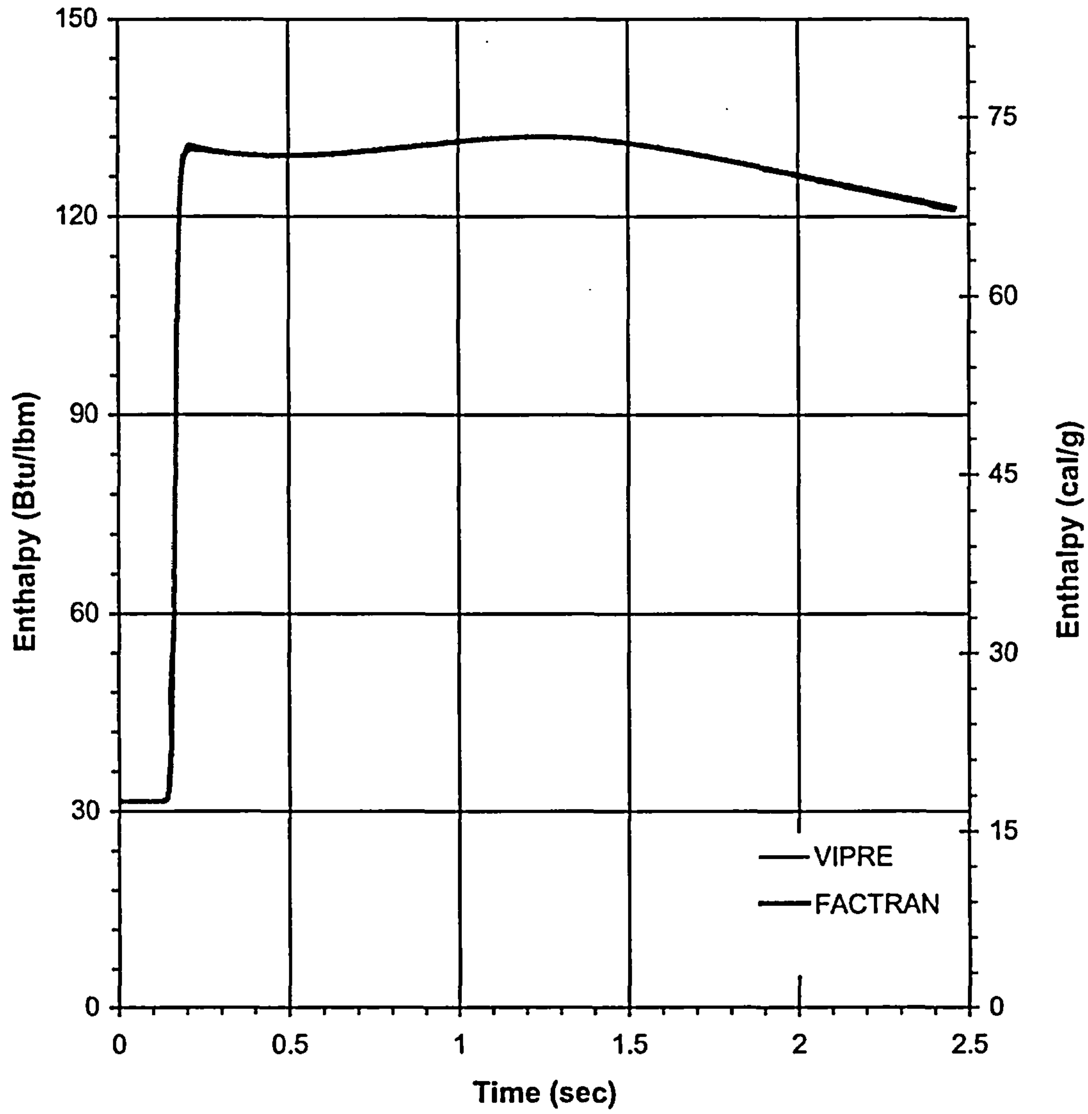


Figure D.4 FACTRAN and VIPRE Fuel Enthalpy Comparison Using 3-D Nuclear Power and Peaking Factor



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Section C

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Westinghouse

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Our ref: LTR-NRC-03-9

March 18, 2003

Enclosed is a proprietary copy of the Westinghouse response to the questions and issues raised during the NRC audit of WCAP 15806-P, "Westinghouse Control Rod Ejection Accident Analysis Methodology Using Multi-Dimensional Kinetics" (Proprietary) conducted on March 17-18, 2003 at the Westinghouse Energy Center in Monroeville, PA.

Also enclosed is:

1. One (1) copy of the Application for Withholding, AW-03-1612 (Non-Proprietary) with Proprietary Information Notice.
2. One (1) copy of Affidavit (Non-Proprietary).

This information is being submitted by Westinghouse Electric Company LLC to document the Westinghouse response to the audit questions and to obtain Nuclear Regulatory Commission ("NRC") generic approval of WCAP-15806 - "Westinghouse Control Rod Ejection Accident Analysis Methodology Using Multi-Dimensional Kinetics" (Proprietary).

This submittal contains proprietary information of Westinghouse Electric Company LLC. In conformance with the requirements of 10 CFR Section 2.790, as amended, of the Commission's regulations, we are enclosing with this submittal an Application for Withholding from Public Disclosure and an affidavit. The affidavit sets forth the basis on which the information identified as proprietary may be withheld from public disclosure by the Commission.

Correspondence with respect to this affidavit or Application for Withholding should reference AW-03-1612 and should be addressed to H. A. Sepp, Manager of Regulatory and Licensing Engineering, Westinghouse Electric Company, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

J. S. Galembush, Acting Manager
Regulatory and Licensing Engineering

Enclosures

cc: G. Shukla/NRR
R Caruso/NRR
U. Shoop/NRR
S. L. Wu/NRR

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Our ref: AW-03-1612

March 18, 2003

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: Responses to WCAP 15806-P, "Westinghouse Control Rod Ejection Accident Analysis Methodology Using Multi-Dimensional Kinetics" (Proprietary) audit questions conducted on March 17-18, 2003 at the Westinghouse Nuclear Center in Monroeville, PA.

Reference: Letter from J. S. Galembush to J. S. Wermiel, LTR-NRC-03-9, dated March 18, 2003

The Application for Withholding is submitted by Westinghouse Electric Company LLC ("Westinghouse"), pursuant to the provisions of Paragraph (b) (1) of Section 2.790 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary material for which withholding is being requested is identified in the proprietary version of the subject report. In conformance with 10 CFR Section 2.790, Affidavit AW-03-1612 accompanies this Application for Withholding, setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

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

Correspondence with respect to this Application for Withholding or the accompanying affidavit should reference AW-03-1612 and should be addressed to H. A. Sepp, Manager of Regulatory and Licensing Engineering, Westinghouse Electric Company, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

J. S. Galembush, Acting Manager
Regulatory and Licensing Engineering

Enclosures

cc: J. S. Wermiel/NRR

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AW-03-1612

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

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

J. S. Galembush, Acting Manager
Regulatory and Licensing Engineering

Sworn to and subscribed
before me this 18th day
of March, 2003

Sharon L. Fiori
Notary Public

SHARON L. FIORI
COMMONWEALTH OF PENNSYLVANIA
NOTARY PUBLIC

Notarial Seal
Sharon L. Fiori, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires January 29, 2007
Member, Pennsylvania Association Of Notaries

(1) I am Acting Manager, Regulatory and Licensing Engineering, in Nuclear Services, Westinghouse Electric Company LLC ("Westinghouse"), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Electric Company LLC.

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

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

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

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

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

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

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

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

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

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

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

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

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

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

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

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

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

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

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

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

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is enclosed as a proprietary copy of the Westinghouse response to the questions and issues raised during the NRC audit of WCAP 15806-P, "Westinghouse Control Rod Ejection Accident Analysis Methodology Using Multi-Dimensional Kinetics" (Proprietary) conducted on March 17-18, 2003 at the Westinghouse Nuclear Center in Monroeville, being transmitted by Westinghouse Electric Company (LTR-NRC-03-9) letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk, Attention Mr. J. S. Wermiel. The proprietary information as submitted by Westinghouse Electric Company LLC is that associated with Westinghouse's request for NRC approval of WCAP 15806-P - "Westinghouse Control Rod Ejection Accident Analysis Methodology Using Multi-Dimensional Kinetics" (Proprietary).

This information is part of that which will enable Westinghouse to:

(a) Obtain NRC approval of the WCAP-15806 – 3D Rod Ejection Accident Analysis Methodology.

(b) Assist our customer in obtaining enhanced nuclear design input data for fuel reload analysis.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of this information to its customers for purposes of conducting 3D rod ejection accident analysis.

(b) The information requested to be withheld reveals the distinguishing aspects of a methodology, which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar calculations and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

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AW-03-1612

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.790 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.790(b)(1).

AW-03-1612

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WCAP-15806-P, 3D Kinetics for Control rod Ejection
Additional Clarification Requested and Responses

General

1. Please provide a one-to-one comparison, in a list/Table format, between the current NRC-approved analyses methodology and the revised methodology, that are currently used (will be used) in the process of analyzing the rod ejection accident (REA). Also provide the page number in the WCAP, where the technical basis supporting the new methodology etc., can be found.

R1. In general, the proposed methodology is very similar or the same as the current licensed methodology, except for the differences resulting from the use of a three-dimensional neutron kinetics analysis computer code to calculate the core behavior vs. time. The major differences are a more accurate depiction of the ejected rod worth and power distribution behavior vs. time rather than the fixed values used in the current 1-D method, and a more accurate depiction of the Doppler feedback compared to the 1-D method. See additional information in Table 1.

2. Please provide a list of all the scenarios analyzed for the REA.

R.2 Many different cases were analyzed at BOL or EOL, HFP or HZP, including variations in input parameters and other analysis assumptions to show the sensitivity of the analysis to these variations. From these cases, the proposed Westinghouse 3-D analysis methodology was developed. The results of the cases that were run are shown in Tables 3.1 to 3.4 for the HZP scenario, and Tables 3.5 to 3.6 for the HFP scenario. Many additional cases are shown in Appendix C. For plant-specific licensing submittals, only the four cases (BOL/EOL, HFP/HZP) will be submitted.

3. The 1st paragraph on page 22, states that the volumetric surge can be computed based on the change in the core fluid density. Is this a new methodology? Please clarify.

R.3 No, this is not a new methodology. As discussed in Table 2.1, Item O on page 20, this is consistent with the current licensed 1-D analysis method (WCAP-7588 Revision 1-A).

4. Please provide additional clarification to the source of the data in Table 3.1, on page 27.

R.4 These are three representative cases. The first case represents a conservative condition with no uncertainty. It is starting from EOL, HZP, no boron, with rods at the insertion limits.

The second case represents the same case as the first one, but also including conservative allowances in the key parameters.

The third case is representative of a limiting case which we would do to bound multiple cycles. Starting with case 2, additional conservative allowances were included along with an extremely conservative xenon distribution.

5. Conditions of represented cases: Last bullet. To what depth was Bank-D inserted?

R.5 D-bank is deeply inserted for the majority of the cycle, representative of daily load follow. The specific example provided assumes D-bank inserted to 200 steps.

6. Please provide additional information/clarification to the 2nd paragraph from the bottom on page 29.

R.6 The peak power impact (%) = ((B-A)/A)*100
A = reference key parameters
B = transient including conservative allowance on all key parameters

7. Provide source of data in Tables 3.3 and 3.4

R.7 The data provided in Tables 3.3 and 3.4 are from the calculations that are documented, following internal QA procedures, in internal Westinghouse calc notes.

8. Provide confirm source of data in Figures 3.5 and 3.6.

R.8 The data provided in Tables 3.3 and 3.4 are from the calculations that are documented, following internal QA procedures, in internal Westinghouse calc notes.

9. Provide additional clarification of paragraph 4 on page 41.

R.9 This is referencing the Westinghouse standard reload design methodology as described in WCAP-9272. We generally do not perform detailed safety analyses each cycle. Instead we perform a reference analysis using conservative key parameters. This analysis is shown to meet the limits. In subsequent cycles, we demonstrate that the cycle specific key parameters are bound by those used in the reference calculations. Thus the cycle safety analysis is bound by the existing safety analysis. If the parameters for a specific analysis are not bound by the existing reference analysis, then new bounding parameters are chosen and the safety analysis redone for that event. This analysis will then serve as the new reference analysis.

10. Pg.3, 2nd sentence, please clarify.

R.10 The "low and low-low insertion limit monitors with visual and audio signals" refers to installed equipment at the plants that alerts the operators of the approach to the insertion limits.

11. Please provide references to recent tests stated in the 1st sentence of the last paragraph (p. 4).

R.11 This was meant to refer to the ongoing reactivity insertion accident (RIA) experiments at the "CABRI" test reactor in France, which indicated that cladding failure and fuel dispersal could occur at low energy deposition, significantly lower than the 280 cal/g regulatory limit of NRC Reg.Guide 1.77. One reference for these tests is:

Schmitz, F. and J. Papin, "High Burnup Effects on Fuel Behavior Under Accident Conditions: The Tests CABRI REP-Na", Journal of Nuclear Materials, Volume: 270, 1999, pp. 55-64.

12. Pg. 4, 200 cal/g--------- >100 cal/g; Explain 41.25%.

R.12 The current limit is an absolute 200cal/gm. The proposed interim limit is a delta of 100 cal/gm. Since the HZP nominal fuel enthalpy is about 17.5 cal/gm, this corresponds to an absolute limit of 117.5 cal/gm, which is a reduction of 41.25%.

Note that this limit is inserted only as an interim limit. We fully expect to utilize the industry limit when it is approved. The purpose of this report is to describe the methodology to analyze the rod ejection event, not to support or defend a limit. We were, however, requested in a meeting with the NRC on July 17, 2001 to indicate a limit that would be used with the methodology to provide a reference point for the needs of the methodology defined. The value was picked to be lower than the expected EPRI recommended value, and having a much simpler representation since it is independent of burnup. But it was thought that this value would be low enough so as to not impact the review of the methodology described in the report.

Uncertainty-Discussion

13. Page 6, 1st paragraph. Appropriate uncertainties. Please clarify.

R.13 By the phrase "appropriate uncertainty allowances" it is meant that the magnitude and direction of the conservative allowances to be applied to the analysis parameters should be specific to the parameter of interest and the accuracy or knowledge of the value of that parameter and how it is used in the analysis. The application of the conservative allowances to be applied to the Westinghouse 3-dimensional analysis methodology is discussed in Section 2.2.5 of the Topical report.

14. The 1st paragraph of Section 2.2.1, page 10, the last sentence states that the delayed fraction can be adjusted for conservatism by applying a fixed multiplier. How is this multiplier determined and applied?

R.14 The [ ]^{a,c}% conservative allowance is described in Section 2.2.5. The [ ]^{a,c}% is consistent with results from dynamic rod worth measurements on many cores. It is applied, through code input, as a uniform multiplier to the node-by-node values.

15. The 1st paragraph of Section 2.2.4 talks about conservatisms described in Section 2.1.2. Please clarify.

R.15 These conservative allowances are further elucidated in Section 2.2.5 and the response to question 16.

16. Section 2.2.5 gives a list of applicable conservative allowances. How were these values determined?

R16. The conservative values are a combination of typical values used in industry. In addition, some parameters are typically measured each cycle and the measurement variability is factored in. The MTC is typically measured and falls within ± [ ] a,c pcm/°F. (It is also not a key parameter.) The rod worth (in dollars) is measured through our dynamic rod worth measurements. These measurements show total rod worth well within ± [ ] a,c %. The delayed neutron fraction is reduced by [ ] a,c % to take this into account. The ejected rod worth is further increased by [ ] a,c %, primarily to account for variations in radial power tilts, xenon distributions and radial power distributions. The normal peaking factor uncertainties are applied.

The Doppler temperature impact is treated as a direct multiplier on the cross section feedback term. This uncertainty is typical of the industry and is consistent with that used by Diamond (see Reference 23 of the topical report).

17. Page 18, Table 2.1, 1st Box, it is stated that the variation in the speed of the ejected rod has a minimal impact on the transient. Please provide additional clarification.

R.17 The time has been chosen to be short enough such that significant nuclear heat is not added until the rod is fully withdrawn. Any more rapid withdrawal has no impact on the shape of the transient. It will only cause a slight shift in the time of the peak. This assumption is consistent with the currently licensed Westinghouse methodology that has been used for 30 years.

18. Page 18, 3rd box from the bottom. Uncertainties are alluded to. What uncertainties?

R.18 Typical uncertainties for the hot full power event are ± 2% in power, ± 5-6 °F, ± 30-60 psi in pressure, with conservative use of the Thermal Design Flow. These are plant dependent parameters and are justified on a plant-specific basis.

19. Page 41, first sentence, second paragraph states uncertainty allowances may be applied statistically. Please summarize how these are calculated and used, including source of data and statistical formulation used, with appropriate references to the sources.

R.19 The statistical formulation is simply the square root of the sum of the squares of the independent parameters. The following is the example from Table 3.4:

[ ] a,c

Additional Discussion will be needed

20. Appendix A of the WCAP, alluding to SER limitations. The staff would like some further discussion on the Cross Section adjustment process during the transient.

R.20 The kinetics formulation is included on pages 2-4 and 2-5 of WCAP-12394-A and WCAP-12983-A.

21. Appendix B: What is the difference between the "reference" in Table B.1 and the "new reference" in Table B.2?

R.21 The data provided in Tables B.1 and B.2 are based on the new reference solution, but the Table B.2 results include a slight change in the steam tables (water density).

22. Appendix C: How are the differences calculated in the tables?

R.22 The reference cases are those including the dashes. The differences are the relative proportional differences from those reference cases. ((B-A)/A)*100, see response to question 6.

Code Coupling

23. Please provide more details on data communication between SPNOVA and VIPRE-01, including data passed at the beginning of execution for each code and transfers during each time step.

R.23 For each node, SPNOVA passes normalized power distribution and relative core power data to VIPRE-01 at each time step. Similarly, for each node, VIPRE-01 passes coolant density, coolant temperature, fuel rod temperatures, and heat flux data to SPNOVA at each time step.

24. Please provide information on the numerical coupling and time step synchronization between codes.

R.24 SPNOVA and VIPRE-01 utilize identical timesteps.

25. Discuss the mapping between neutronics and thermal-hydraulics channels, and associated uncertainties and sensitivity.

R.25 The mapping between neutronics and thermal-hydraulics channels is a one-to-one mapping, so there is no additional uncertainty.

26. Discuss the cross-section generation methodology used in the coupled code suite. What are cross-section generation uncertainties. What is the format of the cross-section data file used by SPNOVA when the thermal-hydraulic feedback is provided by VIPRE-01 and what uncertainties are associated with this feedback.

R.26 The cross-section correlations are a function of burnup, water density and fuel temperature. When VIPRE-01 is used to generate the thermal hydraulic parameters, it just calculates the water density and fuel temperatures. There may be some differences between the static model in SPNOVA and the detailed thermal and hydraulic model in VIPRE-01. However, the first step in the analysis of the transient is to calculate the initial steady-state conditions with the VIPRE-01 model. The resultant k_{eff} serves as a bias term in the subsequent transient. This ensures that the initial condition is exactly critical.

The cross section uncertainties are covered in the uncertainties in the resultant parameters: peaking factors, rod worths, fuel temperature feedback and moderator temperature coefficient.

Request for Additional Sensitivity Analyses

"In Aid of Reviewing WCAP-15806-P, 3D Kinetics for Control Rod Ejection

There are several rod ejection sensitivity analyses that would be helpful to expedite staff review. It would be helpful if these were available during our visit, since the staff has not had a chance to examine the SPNOVA/VIPRE-01 codes input and output, nor exercise any test runs.

- Sensitivity to MTC over the range from the nominal (typical negative value) to the maximum allowable positive value (plant specific)

Response:

[ ] a,c

- Feedback from moderator flow (channel and sub-channel) variations during the transient

Response: A sensitivity study has been previously performed with the core flow rate reduced by [ ] a,c % in the hot zero power rod ejection transient. The results of the sensitivity study show that the peak fuel enthalpy is not very sensitive to the flow change.

- Feedback from local boiling during transient

Response: For the feedback calculations, local boiling prediction is based on the VIPRE-01 homogeneous void model without subcooled boiling. The omission of subcooled boiling is conservative for reactivity feedback, and also ensures better solution stability in the rod ejection transient calculations with very small time steps. As indicated in the SER on VIPRE-01, the VIPRE-01 subcooled void models developed from steady state data are not suitable in boiling transients with the Courant Number ($N_c = u^*\Delta t/\Delta x$) less than one.

Table 1: 3-D Methodology Elements
Comparison to Current Licensed 1-D Methodology

Elements of Reg. Guide 1.77*	3-D Methodology	1-D Methodology
B. Initial Core Conditions Zero Power (BOC & EOC) Low Power (BOC & EOC) Full Power (BOC & EOC)	The full power transient must be evaluated at BOC and EOC to determine the most limiting conditions. The HFP hot rod evaluation includes the uncertainty on calorimetric power. The EOC HZP case is the most limiting of the HZP and low power transients. The BOC ejected rod worth can be compared to the EOC value to demonstrate that the EOC transient will be the most limiting. The initial core conditions need to address the potential operational history of the core. This includes consideration of the previous cycle length variation on the BOC cases and operation with control rods inserted for the EOC cases.	Same. The full power transient is evaluated at BOC and EOC. The HFP case is most limiting for peak fuel enthalpy and centerline melting. The HFP hot rod evaluation includes the uncertainty on calorimetric power. EOL HZP case is most limiting compared to BOL HZP case. Operational history is taken into account in static 3-D calculation of input parameters to current method.
B. Initial Loss of Primary System Integrity	The RCS overpressure will be evaluated in the same manner as with the 1-D methodology	Methodology reviewed/approved by NRC in WCAP-7588 Rev.1-A.
C. Ejected Rod Worth a) maximum inserted position based on power level b) additional fully inserted or partially inserted misaligned or inoperable rods, if allowed c) increase worth to account for calculational uncertainties d) increase worth to account for xenon transients	a) The control rod positions will be consistent with the insertion limits. b) The rod positioning at the insertion limit is already below that of normally expected positions, thus no additional misalignment term is added. c) The uncertainty due to cross-sections is included in the uncertainty in the beta-effective. This is consistent with how the control rod worths are measured. d) The largest variation in the control rod worths is due to power distribution variations caused by transient xenon distributions, quadrant power tilts, or other similar factors. The HZP ejected rod worth is determined assuming nominal conditions (no xenon) and then increased by [ ]^{a,c} to account for the impact of potential adverse power distributions. The HFP initial condition assumes the control rods at their insertion limit consistent with a xenon distribution which results in the axial offset being at its most positive allowed value.	a) Same b) Same c, d) Current method increases worth by [ ]^{a,c} to account for Xenon and calculational uncertainties.

*Elements of Reg. Guide 1.77 as consistent with the EPRI working group 3-D methodology guidelines.

Table 1: 3-D Methodology Elements (cont.)
Comparison to Current Licensed 1-D Methodology

Elements of Reg. Guide 1.77*	3-D Methodology	Current (1-D) Licensed Methodology
D. Reactivity Insertion Rate a) based on differential worth curve and rod position vs. time curve b) rate of ejection based on maximum ΔP and weight and cross-sectional area of the control rod and drive shaft	a) The 3-D calculation inherently includes the rod worth as a function of position. b) The rod ejection time is taken to be 0.1 second, which is consistent with the physical parameters. Variation in the speed has minimal impact on the net outcome of the transient.	a) Linear insertion with time. This is acceptable since due to the rapid ejection time, the accident is sensitive to the total reactivity insertion, not variations in the rate. b) Same (0.1 sec.)
E. Effective Delayed Neutron Fraction and Prompt Neutron Lifetime	The prompt neutron lifetime is replaced by the neutron velocity when using a 3-D model. Although the core transient will change with variations in the neutron velocity, the net impact on the fuel enthalpy is small. Therefore, design values are used.	Same for approved 1-D method.
a) use available data and average based on fission fractions	a) Use of the 3-D model allows for the delayed neutron fractions to vary with core position.	a) Uses a single set of 3-D average design values, which take into account fission fractions.
b) use minimum calculated value for the given reactor state	b) Values consistent with the time in life will be used. In addition, the delayed neutron fraction will be further reduced by [ ] a,c.	b) Uses minimum values consistent with time in life. In addition, delayed neutron fraction is reduced by at least [ ] a,c.
c) consider both the power excursion and the power reduction when selecting a conservative value	c) Use of the 3-D model allows for the delayed neutron fractions to vary with core position, thus changes in the core effective value due to flux distribution changes are automatically taken into account.	c) Uses a single set of 3-D average design values, which take into account initial flux distribution.
F. Initial Pressure, Flow and Temperature	The nominal core values are used for the 3-D transient. Conservative values are chosen for the hot rod analysis, with uncertainties applied in the limiting direction.	Same for approved 1-D method. Typical uncertainties for HFP events are: Power: ± 2%, T_{avg} : ± 5-6°F, Pressure: ± 30-60psi, Flow: Thermal Design. These are plant dependent and are justified on a plant-specific basis.
G. Fuel thermal properties a) fuel-clad gap heat transfer coefficient b) fuel thermal conductivity c) direct moderator heating	Nominal values are used in the 3-D transient. The hot rod evaluation uses conservative values consistent with high fuel temperatures for the maximum fuel enthalpy calculation.	Same for approved 1-D method.
H. UO_2 Specific Heat	Use nominal values in the transients.	Same for approved 1-D method.

*Elements of Reg. Guide 1.77 as consistent with the EPRI working group 3-D methodology guidelines.

Table 1: 3-D Methodology Elements (cont.)
Comparison to Current Licensed 1-D Methodology

Elements of Reg. Guide 1.77*	3-D Methodology	Current (1-D) Licensed Methodology
I. Moderator Reactivity Coefficient to include effects of voids, pressure, temperature and boron	The feedback is modified to make the moderator temperature coefficient more positive by [ ]^{a,c}. This can be done by either adjusting the feedback corrections, or by increasing the boron concentration in the 3-D model.	Feedback is adjusted to make the MTC more positive by at least [ ]^{a,c} by adjusting boron concentration.
J. Doppler Coefficient to include corrections for pin shadowing and should compare conservatively to data. Uncertainty in fuel temperature to be included.	The Doppler feedback correction is adjusted and is reduced by [ ]^{a,c}.	The Doppler feedback in pre-ejection steady-state condition is reduced by at least [ ]^{a,c} compared to 3-D design values. Conservative Doppler weighting factors are used for transient calculation.
K. Control Rod Reactivity Insertion on reactor trip to include initial position, differential worth curve, etc.	The 3-D model accounts for the variation of trip reactivity with position. The trip rod insertion time is taken to be the maximum allowed by the Technical Specifications. The trip reactivity is further pessimized by assuming a control rod cluster adjacent to the ejected rod cluster is stuck and both rods are not trippable.	The 1-D model accounts for variation of trip reactivity with position. The trip rod insertion time is taken to be the maximum allowed by Tech Specs. Uses conservative total trip reactivity assuming a stuck control rod in addition to the ejected rod.
L. Reactor Trip Delay Time	The conservative trip delay time, typically 0.5 seconds, is assumed. In addition, the impact of the asymmetric power distribution is accounted for in the 3-D calculation with the effective excore signal determined for each detector. It is conservatively assumed that one detector is out of service, so that three out of the four detectors must indicate a trip.	Same trip delay time (0.5 seconds).
M Computer Code a) coupled thermal/ hydraulic/ nuclear model b) all reactivity feedback mechanisms c) at least 6 delayed neutron groups d) axial and radial nodes e) coolant flow modeled f) trip on flux or pressure	These are addressed by the use of a 3-D transient code.	Same, except no radial neutronics nodes in 1-D transient calculation. Uses radial nodes in fuel pellet calculation. In the separate hot spot calculation, radial and axial power distributions are assumed to be constant a their most limiting (post-ejection) value.

* **Elements of Reg. Guide 1.77 as consistent with the EPRI working group 3-D methodology guidelines.**

Table 1: 3-D Methodology Elements (cont.)
Comparison to Current Licensed 1-D Methodology

Elements of Reg. Guide 1.77*	3-D Methodology	Current (1-D) Licensed Methodology
N. Analytical Models and Computer Codes		
a) Documented and justified	a) The SPNOVA and VIPRE codes have previously been approved for use by the NRC, and their usage for rod ejection is discussed in Appendix A.	a) The TWINKLE and FACTRAN codes have previously been approved by the NRC for use for rod ejection.
b) Conservatism evaluated by comparison with experiment or more sophisticated codes	b) The uncertainties in codes were addressed in the previously approved topical reports. A comparison with an HZP rod ejection bench-mark problem is presented in Appendix B.	b) Uncertainties addressed in previously approved topical reports and NRC Safety Evaluation Report for 1-D methodology.
c) Changes in flux shapes should be investigated.	c) Appendix C discusses the sensitivities of the calculated results to many parameters, including flux shapes.	c) WCAP-7588 Rev.1-A discusses sensitivities of results to a number of parameters.
d) Conservatism of the flux shapes used for reactivity input and feedback, peak energy deposition, total energy and gross heat transfer to the coolant should be evaluated.	d) The 3-D code takes into account the actual flux distributions to obtain the effective feedback. Different fuel rod models are used for the feedback parameters and for the hot rod evaluation, allowing for a more pessimistic hot rod calculation.	d) Method uses input parameters based on conservative 3-D static (design) code calculations. Post-ejection flux shape assumed constant with no credit for power flattening due to feedback.
e) Sensitivity studies on Doppler, power distribution, fuel heat transfer parameters and other relevant parameters should be included.	e) Appendix C discusses the sensitivities of the calculated results to many parameters, including Doppler and moderator impacts.	e) WCAP-7588 Rev.1-A discusses sensitivities of results to a number of parameters.
O. Pressure Surge	This is calculated based on the net volume increase due to the heating of the coolant, and is consistent with the current licensed 1-D analysis method.	Methodology reviewed/approved by NRC in WCAP-7588 Rev.1-A.
P. Pin Census	The use of a 3-D code allows for the calculation of a pin census directly.	Pin census based on 3-D static post-ejection power distribution.

*Elements of Reg. Guide 1.77 as consistent with the EPRI working group 3-D methodology guidelines.

Table 1: 3-D Methodology Elements (cont.)
Comparison to Current Licensed 1-D Methodology

Additional Items*	3-D Methodology	Current (1-D) Licensed Methodology
AA. Initial Power Distribution/ Cross Sections	The initial power distribution is consistent with the allowed technical specifications and core operating limits. Sustained operation with control rods inserted is addressed by the methodology.	Uses initial power distribution peaking factors calculated by 3-D static design codes taking into account allowed Tech. Specs. and core operating limits.
BB. Pin-to-Node Factor	The pin power reconstruction method is used to generate the peak rod power.	Uses pre- and post-ejection peaking factors calculated by static design codes using pin power reconstruction method.
CC. Reload Checks	The conventional approach will be used to confirm that the analysis of record remains valid for each reload core. If the confirmation is unsuccessful, then the analysis must be revised, or the core redesigned.	Same.
DD. Pellet-gap Model	The nominal pellet gap model will be used for the node average temperature calculation. A conservative model will be used for the hot rod calculation.	Same.
EE. Onset of DNB	The onset of DNB will be conservatively calculated to force the hot spot into post-DNB film boiling during the transient for the fuel enthalpy calculation.	Onset of DNB conservatively assumed to occur at time of rod ejection or for an extremely low heat flux to force hot spot into post-DNB film boiling.
FF. Calculation of ? cal/g	A detailed hot rod model with conservative heat transfer properties will be used to determine the fuel enthalpy increase.	Uses a detailed hot spot model with conservative heat transfer properties.
GG. Fuel pellet radial power profile	The design radial power profile will be used for the node average fuel temperature calculation. Also, the fuel effective resonance temperature will be calculated using the design radial weighting factors. The hot rod radial power distribution will use a conservative profile that increases the average fuel temperature, and hence maximizes the fuel enthalpy.	Same. (Design value used for average fuel rod model in 1-D calculation, hot spot model uses a conservative pellet power profile.)

*Additional Items as consistent with the EPRI working group 3-D methodology guidelines.